Number	Title	Rev	Document Owner
IAG-31	Authorization Agreement for the Reactor Technology Complex (RTC) Advanced Test Reactor (ATR) Facility	5	Site Operations Director
LST-100	Safety Basis for Advanced Test Reactor Facility	20	ATR Operations Manager
TSR-186	Technical Safety Requirements for the Advanced Test Reactor	14	ATR Operations Manager

Document ID:IAG-31
Revision ID:5
Effective Date:02/01/05

Authorization Agreement

U.S. Department of Energy
Idaho Operations Office
and Battelle Energy Alliance, LLC
Authorization Agreement
for the Reactor Technology Complex (RTC)
Advanced Test Reactor (ATR) Facility

Battelle Energy Alliance

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PREAMBLE

This Authorization Agreement documents the agreement between the Department of Energy Idaho Operations Office (NE-ID) and the Idaho National Laboratory (INL) contractor, Battelle Energy Alliance, LLC (BEA), on the overall basis for the safe operation of the Reactor Technology Complex Advanced Test Reactor (ATR) facility. DOE has reviewed and approved BEA's proposed Safety Basis as defined in LST-100 for the defined scope of work. This authorization agreement contains key terms and conditions (controls and commitments) under which BEA is authorized to perform work. Any changes to these terms and conditions require DOE approval. In documenting the overall basis for the safe conduct of work, this authorization agreement highlights the scope of the authorization, the safety basis, and the key terms and conditions (controls and commitments). This agreement highlights existing requirements rather than introducing new requirements. However, the agreement does distill and succinctly state the overall basis for the safe conduct of work in support of effective DOE and BEA integrated safety management of the facility.

AGREEMENT

1. The DOE has determined, based on the completion of the following activities, that implementation of the authorization agreement is sufficient to adequately protect the public, the workers, and the environment, and DOE is confident BEA is capable of meeting the commitments identified in the authorization agreement:

   A. Through the periodic assessment of the contractor's performance and through the review and approval of the documents that are identified by this authorization agreement, DOE ensures that the contractor, BEA, has identified the basis for safe operation of the INL.

   B. Through review and approval, the ATR Upgraded Final Safety Analysis Report (UFSAR) and Technical Safety Requirements (TSR) adequately documents the analysis of facility hazards, specifies the controls necessary to prevent and mitigate the hazards, and defines a safety management program, which affords an acceptable level of safety to the public, the workers, and the environment. A complete list of applicable safety documents is identified in LST-100.

   C. Through the use of the Environmental Checklist/National Environmental Policy Act (NEPA) process as described in MCP-3480, "Environmental Instructions for Facilities, Processes, Materials, and Equipment," the ATR facility evaluates the effects to the environment resulting from changes in its operations.

D. Through the performance of restart assessments in accordance with DOE Order 425.1A and revisions thereto as identified in contract DE-AC07-05-ID14517, there is assurance the facility is being operated without endangering the health and safety of the public, the workers, or the environment.

E. Based on items B, C, and D above, and on its institutional and support programs and processes, BEA has adequate technical and managerial capability to safely operate the ATR facility.

2. The authorization agreement applies to the ATR facility, located within the Reactor Technology Complex of the INL located 50 miles west of Idaho Falls, Idaho. The facilities, major functions, subsystems, and support facilities are described in the safety documents identified in LST-100.

3. The scope of activities authorized in conjunction with the operation of ATR is defined by: (a) the Work Authorization and Change Control process specified in Contract Number DE-AC07-05-ID14517, and (b) the safety documents identified in LST-100. The following activities involving the processing, handling, and storage of radioactive material are authorized:

A. Operation of the ATR, a Category A, Department of Energy nuclear reactor with a design thermal power of 250 MW, a nominal operation pressure of 360 psig at the top of the core, and a nominal reactor outlet temperature of 167°F

B. Irradiation of experiments as defined in the facility Technical Safety Requirements

C. Experiment change-out operations

D. Reactor refueling operations and reactor fuel and experiment storage operations

E. Handling and storage of radioactive waste materials generated from the above activities

F. All additional work which may be required in order for BEA to meet its contractual obligations under Contract Number DE-AC07-05-ID14517 at the ATR facility.

4. Operation of the ATR is subject to the following requirements and conditions:

A. Safety Basis, Environmental Permits, and DOE-approved Operational Readiness Reviews/Readiness Assessments, as applicable, were used to develop this agreement. This agreement forms the basis for operation of the ATR. Authorized deviations shall be documented by BEA and approved by DOE through Exemption Requests, Compliance Schedule Agreements, Implementation Plans, modifications to the contract, or other means as appropriate.

B. BEA shall operate the ATR in accordance with the operational controls specified in the Safety Basis as defined by LST-100.

C. BEA shall maintain the Safety Basis, defined in LST-100. As Safety Basis documents are revised, added, or deleted as approved by DOE, the list shall be updated. Changes to LST-100 do not require separate DOE approval under the following conditions:

- Change in the revision date or number of a listed document if DOE has formally approved the new document revision
- Addition of a new Safety Basis document [e.g., Safety Analysis Report/Technical Safety Requirement document, Unreviewed Safety Question (USQ), hazard classification] that has been formally approved by DOE
- Change in the revision date/number of a listed document if the new document revision has received a USQ screening/evaluation and determined not to involve a USQ
- Deletion of a listed document if cancellation of the document has been formally approved by DOE
- Editorial changes (e.g., typographical errors)
- All other changes to the Safety Basis List documents require the approval of both DOE and BEA management at the level of the Authorization Agreement signatories.

D. BEA shall implement and maintain in effect an ATR Facility Emergency Preparedness and Planning Program as described in BEA Manual 16, *Emergency Preparedness.*

E. BEA shall comply with the requirements specified in the active revisions of the following approved permits and compliance agreements:

- Idaho Department of Environmental Quality, "State of Idaho Permit to Construct for TRA Evaporation Pond," Permit Number 023-00001, September 9, 2002
- James Johnston, Department of Environmental Quality, letter to Mr. Stacey Madson, "INEEL Test Reactor Area (TRA) Cold Waste Pond and Water Reactor Research Test Facility (WRRTF) Wastewater Disposal Ponds," January 19, 2001.

F. Through the use of a facility Configuration Management Process controlled by PRD-115, "Configuration Management," BEA maintains the ATR hardware, procedures, and organizational integrity.

G. Through the use of An Integrated Safety Management System detailed by PDD-1004, "INEEL Integrated Safety Management System," BEA integrates the environmental, safety, and health issues into the totality of the work for the ATR facility.

5. This Authorization Agreement is subject to the conditions listed herein. Nonconformance to the components of this agreement will be reported using currently applicable reporting systems such as the Issues Communications and Resolution Environmental (ICARE) System and the Occurrence Reporting and Processing System. DOE and BEA agree to the conditions and limitations contained in the Authorization Agreement. The conditions and limitations within this Authorization Agreement are effective upon the date identified in the first page header of this document and shall expire upon expiration or termination of Contract Number DE-AC07-05-ID14517.

6. This Authorization Agreement is not intended to change or add to the Management and Operations (M&O) contract, but is intended to implement applicable requirements of the M&O contract. If there is an inconsistency between the M&O contract and this Authorization Agreement, the M&O contract terms will prevail.

Battelle Energy Alliance, LLC:

ART CLARK Deputy Laboratory Director for Operations (Printed)	1/7/05 Deputy Laboratory Director for Operations (Signature)

Department of Energy Idaho Operations Office:

Elizabeth D Sellers Manager of Idaho Operations (Printed)	1/20/05 Manager of Idaho Operations (Signature)

Department of Energy Idaho Headquarters Office:

OWEN W. LOWE Associate Director Nuclear Facilities Management, NE-40 (Printed)	1/31/05 Associate Director Nuclear Facilities Management, NE-40 (Signature)

Safety Basis List for

ADVANCED TEST REACTOR FACILITY

NOTE: *Many of the institutional safety programs are administered by functional area organizations that are outside of the organizational structure directly responsible for operation of the ATR. See SAR-153, Section ES.1.1.3 for commitments regarding these programs.*

1. SAR-153, "Upgraded Final Safety Analysis Report for the Advanced Test Reactor," Current Issue.

2. TSR-186, "Technical Safety Requirements for the Advanced Test Reactor," Current Issue.

3. DOE-NE Approval letter to John E. Dwight, Director, ATR Programs, March 30, 2005, Transmittal of NE-40 Approval of the Advanced Test Reactor (ATR) Upgraded Final Safety Analysis Report Addendums, Engineering Design Files (EDF) 5488 and 5614, and Associated Technical Safety Requirements (TSR) Revisions. (See LO-RTC/SMC-05-025.)

4. Nuclear Criticality Safety Supplement – none required.

5. Test Authorizations – none required.

6. Hazard Classification and Categorization – see SAR-153.

7. Standard Practice (SP)-10.3.1.31 DOE concurrence, D. M. Lucoff to R. V. Furstenau, Nuclear Facility Restart Authorization and Power Reduction Recovery, DML-65-01, dated May 29, 2001.

8. Readiness Assessment approval T. F. Burns Jr., to J. C. Okeson, Approval to Restart the Advanced Test Reactor (Initial Criticality) Following the Core Internals Changeout – OPE-TRA-94-067, dated July 9, 1994.

9. Other Documents and process associated with the safety basis of DOE-ID activities – E. C. Anderson Letter to E. J. Ziemianski, ECA-10-94, Modifications to ATR Canal Operations, dated March 8, 1994.

10. Unreviewed Safety Questions – TRA-USQ-2004-176, "Reactor Shutdown System Reactor Vessel Level Limiting Control Setting."

TRA-USQ-SE-2003-145[1,2],"ATR Firewater Supply System Modeling Issues."

TRA-USQ-2004-214[2], Revision 1, "ATR Seismic Safety Basis Deterioration."

TRA-USQ-2004-385, Revision 1, "ATR Surge Tank Level Instrument Limiting Control Settings."

TRA-USQ-2004-396[2], "ATR Seismic Primary Coolant System Break Size and TRA Support Building Vulnerabilities."

TRA-USQ-2004-413[2], "ATR Seismic Primary Coolant System Break Size Contribution from Letdown Valves."

RTC-USQ-2005-173, "Impact of Potential PCS Leakage on Fuel Element Thermal-Hydraulic Conditions Prior to Reactor Vessel Venting."

RTC-USQ-2005-197, "M-11 Emergency Coolant Pump Flow Measurement and Uncertainty."

RTC-USQ-2005-248, "GAP-005-05 – Radiation Monitoring and Seal System Setpoint Derivation."

RTC-USQ-2005-336, "Derivation of Emergency Coolant Pump Performance Parameters Modeled in the ATR Accident Analyses."

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[1] An interim resolution supporting the continued operation with this USQ was approved by DOE. See: O. W. Lowe memorandum to E. D. Sellers, November 21, 2003, Approval of the Safety Evaluation Report (SER) for Resolution of Unreviewed Safety Questions Related to August 21, 2003 Shutdown of the Advanced Test Reactor (ATR) and Approval for ATR Restart.

[2] An interim resolution supporting the continued operation with these USQs were approved by DOE. See: J. A. Malmo letter to J. E. Dwight, March 30, 2005, Transmittal of NE-40 Approval of the Advanced Test Reactor (ATR) Upgraded Final Safety Analysis Report Addendums, Engineering Design Files (EDF) 5488 and 5614, and associated Technical Safety Requirements (TSR) revisions (LO-RTC/SMC-05-025).

Document ID:TSR-186
Revision ID:14
Effective Date:04/07/05

OFFICIAL USE ONLY

Technical Safety Requirements

Technical Safety Requirements for the Advanced Test Reactor

INEEL
Idaho National Engineering and Environmental Laboratory

Form 412.14
10/9/2003
Rev. 05

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REVISION LOG

Rev.	Date	Affected Sections	Revision Description
NOTE:	*Each section of this document has an individual revision number. The revision number identified in the first column is the revision number for the Front Matter (TSR-186) section of this document.*		
0	03/23/1998	All	See DAR 80588.
1	05/21/98	TSR-186, TSR-186-3/4, TSR-186-A-3/4.8	See DAR 62529. See DOE-HQ approval Letter OPE-TRA-98-035. Interim Resolution of ATR Confinement Leak Rate.
2A	04/17/99	TSR-186, TSR-186-1, TSR-186-3/4, TSR-186-5, TSR-186-A-3/4.2, TSR-186-A-3/4.3, TSR-186-A-3/4.6	See DAR 63972. See DOE-HQ approval Letter OPE-TRA-99-028. LOCA Interim.
2B	09/24/99	TSR-186, TSR-186-1, TSR-186-3/4, TSR-186-5, TSR-186-A-3/4.2, TSR-186-A-3/4.3, TSR-186-A-3/4.4, TSR-186-A-3/4.6	See DAR 64433. See DOE-HQ approval Letter OPE-TRA-99-072. LOCA MOD.
3	02/28/2000	TSR-186, TSR-186-1, TSR-186-3/4, TSR-186-5, TSR-186-A-3/4.5	See DAR 63218. See DOE-HQ approval Letter OPE-TRA-99-041. 1998 Annual Update.
4	N/A	N/A	Revision 4 was not used see Letter VMB-13-03.
5	03/28/01	TSR-186, TSR-186-A-3/4.8	See DAR 66766. See DOE-HQ approval Letter TPO-TRA-01-017. ATR Confinement Leak Rate USQ Resolution.
6	12/17/01	TSR-186, TSR-186-3/4, TSR-186-A-3/4.2	See DAR 66769. See DOE-HQ approval Letter TPO-TRA-01-114. 2000 Annual Update.
7	09/26/02	TSR-186, TSR-186-3/4, TSR-186-A-3/4.5	See DAR 89147. See DOE-HQ approval Letter TPO-TRA-02-061. Resolution of USQ Evaluation TRA-USQ-SE-2001-063.
8	10/31/02	TSR-186, TSR-186-5	See DAR 88852. See DOE-HQ approval Letter TPO-TRA-02-084. 2001 Annual Update.

Rev.	Date	Affected Sections	Revision Description
9	02/06/03	TSR-186, TSR-186-1, TSR-186-5	See DAR 95459 and 96613. See DOE-HQ approval Letters TPO-TRA-03-01 and TPO-TRA-03-011. Resolution of USQ Evaluation TRA-USQ-SE-2002-150 and change to training AC.
10	11/23/03	TSR-186, TSR-186-3/4, TSR-186-A-3/4.2, TSR-186-3/4.3	See DAR 105152. See DOE-NE approval Letter TPO-TRA-03-113. Resolution of USQ Evaluation SE-2003-126, SE-2003-145, SE-2003-146, SE-2003-155, SE-2003-156, and SE-2003-171.
11	08/10/04	TSR-186, TSR-186-1, TSR-186-2, TSR-186-3/4, TSR-186-5, TSR-186-A-2.0, TSR-186-A-3/4.1, TSR-186-A-3/4.2, TSR-186-A-3/4.3, TSR-186-A-3/4.6, TSR-186-A-3/4.8	See DAR 98666. See DOE-NE approval Letter TPO-TRA-04-051. 2002 Annual Update.
12	08/10/04	TSR-186, TSR-186-1, TSR-186-3/4, TSR-186-5, TSR-186-A-2.0, TSR-186-A-3/4.0, TSR-186-A-3/4.2, TSR-186-A-3/4.3, TSR-186-A-3/4.4, TSR-186-A-3/4.5, TSR-186-A-3/4.6, TSR-186-A-3/4.7, TSR-186-A-3/4.8, TSR-186-A-3/4.9	See DAR 102367. See DOE-NE approval Letter TPO-TRA-04-052. 2003 Annual Update.
13	01/22/05	TSR-186, TSR-186-3/4, TSR-186-A-3/4.2	See DAR 119365. See DOE-NE Approval Letter TPO-TRA-05-010. Emergency Firewater Injection System Vessel Pressure Subsystem.

Rev.	Date	Affected Sections	Revision Description
14	04/07/05	TSR-186, TSR-186-1, TSR-186-2, TSR-186-3/4, TSR-186-5, TSR-186-A-2.0, TSR-186-A-3/4.1, TSR-186-A-3/4.2, TSR-186-A-3/4.3, TSR-186-A-3/4.5	See DAR 120036. See DOE-NE Approval Letter LO-RTC/SMC-05-025. Seismic Vulnerability and DBR Program Identified USQ's SAR-153 and TSR-186 Changes.

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LIST OF ACRONYMS, ABBREVIATIONS, AND SYMBOLS

AC	Administrative Control
AHTL	ATR high-temperature loop
ALARA	As-low-as-reasonably-achievable
ATP	Approved Transportation Plan
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
AWIT	aluminum water ignition temperature
BH	Bottom Head
C	Center
CAM	Constant Air Monitor
CLLC	Contact to Logic Level Converter
CFR	Code of Federal Regulations
CHF	CRITICAL HEAT FLUX
CLOFA	Complete Loss of Flow Accident
CSAP	Core Safety Assurance Package
DBA	Design Basis Accident
DBR	Design Basis Report (ATR)
DOE	Department of Energy
EO	Experiment Operator
ECP	Emergency Coolant Pump
EPA	Environmental Protection Agency
ESA	Experiment Safety Analysis
ESAP	ESAP Experiment Safety Assurance Package
ESF	ENGINEERED SAFETY FEATURE
EFIS	EMERGENCY FIREWATER INJECTION SYSTEM
FSAR	Final Safety Analysis Report
ICPP	Idaho Chemical Processing Plant
INEL	Idaho National Engineering Laboratory
IPT	Inpile tube
K_{eff}	effective multiplication factor
kW	kilowatt

LIST OF ACRONYMS, ABBREVIATIONS, AND SYMBOLS (continued).

Section		Page
LCO(s)	LIMITING CONDITION(S) for OPERATION	
LCRM	log count rate meter	
LCS(s)	LIMITING CONTROL SETTING(S)	
LDW	Low Pressure Demineralized Water	
LIPT	Large Inpile Tube	
LOCA	Loss-of-coolant accident	
OPLIMS	OPERATING LIMITS	
OSCC	Outer Shim Control Cylinder	
MUCH	Maximum Utilization Capacity Holder	
MW	Megawatts	
NA	Not Applicable	
NE	Northeast	
N_F	Reactor operating steady power level	
N_L	Reactor power level that corresponds to 1% N_F during power and low power operation or the maximum allowed power (0.5 MW) during depressurized operation.	
NOAA	National Oceanic and Atmospheric Administration	
NW	Northwest	
Δp	Differential pressure	
PCP	Primary Coolant Pumps	
PCS	Primary Coolant System	
pH	Measurement of acidity	
PPS	Plant Protection System	
RAO	Reactor Auxiliary Operator	
RCR	Reactor Control Room	
RCT	Radiological Controls Technician	
RDAS	Reactor Data Acquisition System	
RMSS	Radiation Monitoring and Seal System	
RO	Reactor Operator	
RSS	Reactor Shutdown System	
SATS	Surveillance and Test System	
SCS	Secondary Coolant System	
SDD	System Design Description	
SE	Southeast	
SEO	Senior Experiment Operator	
SIPT	Standard Inpile Tube	
SL(s)	SAFETY LIMIT(S)	
SNM	Special nuclear material	

LIST OF ACRONYMS, ABBREVIATIONS, AND SYMBOLS *(continued)*.

Section		Page
SR	Surveillance Requirement	
SRAO	Senior Reactor Auxiliary Operator	
SRO	Senior Reactor Operator	
SS	Shift Supervisor	
SSC	Structures, systems, and components	
σ	Standard deviation for normal distribution	
SW	Southwest	
TBD	To be determined	
TRA	Test Reactor Area	
TSR	Technical Safety Requirements	
UT	Ultrasonic Testing	
UV	Upper Vessel	
VVS	Vessel Vent Subsystem	
WCC	Warning Communications Center	

Section 1

Use and Applications

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1.0 USE AND APPLICATIONS

1.1 DEFINITIONS

NOTE

The defined terms of this section appear in capitalized type and are applicable throughout the Technical Safety Requirements and the Bases.

TERM	DEFINITION	
ACCURACY	The maximum difference between the nominal protective channel SETPOINT and the actual value, of the measured variable, at which channel trip occurs, not including time lags due to channel response time.	
ACTION	That part of the Technical Safety Requirements that prescribes Required Actions to be taken under designated Conditions within specified COMPLETION TIMES.	
ACTUATION LOGIC CIRCUIT	Components and interconnections controlled by the PPS, from the output of the 2 of 3 logic module(s) through the components listed for the following ACTUATION SYSTEMS:	
	RSS	Rod Clutch Controller Relays (the K-1 relay associated with each module, A1 through A8) for a total of four ACTUATION LOGIC CIRCUITS.
	EMERGENCY FIREWATER INJECTION	Circuit and relay modules for Vessel Level and Pressure, located in the relay logic panel in each PPS room with four circuits for each UV and BH subsystem. The circuit includes four Vessel Pressure bypass switches.
	RMSS	Radiation Monitoring and Seal System (RMSS) relays R1AX2, R1AX1, R1A1, R1AX2B, R1AX1B, R1A1B, R1A2B, R1 (A through C), R1A (A through C) and BDR1 through BDR4 configured into two 2 out of 2 logic circuits for argon delay line valves and for the primary and secondary ventilation dampers.
	Primary Pump Shutoff	ESF 1 and 4 relays, 500-1X, 500 ESF-1, 62-1, 500-4X, 500 ESF-4, and 62-4 consisting of two circuits.

1.1 DEFINITIONS (continued)

Pressurizing and Gland Seal Pump Shutoff	ESF 1 and 4 relays 502-1X, 502 ESF-1, 502-11X, 502 ESF-1B, 502-44X, 502ESF-4, 502-4X, and 502 ESF-4B consisting of two circuits.
LOCA PCP Shutoff	There are 4 circuits; one for each PCP, M-6 PCP: Bypass Relays K1-IN-1, K1-IN-4, K1-OUT-1, K1-OUT-4; Input Relays K2-IN-1, K2-IN-4, K2-OUT-1, and K2-OUT-4; Time Delay Relays TMR-IN-1, TMR-IN-4, TMR-OUT-1, TMR-OUT-4; and relays KM6-IN1-1, KM6-IN1-2, KM6-IN4-1, KM6-IN4-2, KM6-OUT1-1, KM6-OUT1-2, KM6-OUT4-1, and KM6-OUT4-2. M-7 PCP: Bypass Relays K1-IN-1, K1-IN-4, K1-OUT-1, K1-OUT-4; Input Relays K2-IN-1, K2-IN-4, K2-OUT-1, and K2-OUT-4; Time Delay Relays TMR-IN-1, TMR-IN-4, TMR-OUT-1, TMR-OUT-4; and relays KM7-IN1-1, KM7-IN1-2, KM7-IN4-1, KM7-IN4-2, KM7-OUT1-1, KM7-OUT1-2, KM7-OUT4-1, and KM7-OUT4-2. M-8 PCP: Bypass Relays K1-IN-1, K1-IN-4, K1-OUT-1, K1-OUT-4; Input Relays K2-IN-1, K2-IN-4, K2-OUT-1, and K2-OUT-4; Time Delay Relays TMR-IN-1, TMR-IN-4, TMR-OUT-1, TMR-OUT-4, and relays KM8-IN1-1, KM8-IN1-2, KM8-IN4-1, KM8-IN4-2, KM8-OUT1-1, KM8-OUT1-2, KM8-OUT4-1, and KM8-OUT4-2. M-9 PCP: Bypass Relays K1-IN-1, K1-IN-4, K1-OUT-1, K1-OUT-4, Input Relays K2-IN-1, K2-IN-4, K2-OUT-1, and K2-OUT-4, Time Delay Relays TMR-IN-1, TMR-IN-4, TMR-OUT-1, TMR-OUT-4, and relays KM9-IN1-1, KM9-IN1-2, KM9-IN4-1, KM9-IN4-2, KM9-OUT1-1, KM9-OUT1-2, KM9-OUT4-1, and KM9-OUT4-2.

ACTUATION SYSTEM	Components and interconnections of a system, from the sensor(s) to the ACTUATOR.
ACTUATOR	A device that acts directly on the plant to provide protective action when released, energized, or deenergized by a trip. ACTUATORS include, but are not limited to, SAFETY RODS, Pressurizing and Gland Seal Pumps, PCPs, CONFINEMENT fans and dampers, and EFIS level control valves.
ACTUATOR CONTROLLER	A device that directly provides a protective action signal to the ACTUATOR.

1.0 USE AND APPLICATIONS

1.1 DEFINITIONS (continued)

APPROVED FUEL STORAGE	A fixed single or multi-item location or container outside the reactor fuel annulus or irradiation facilities within the reflector tank whose k_{eff}, cooling requirements, and related administrative controls for containing fissile material have been analyzed in approved safety documentation. Carriers for transporting fissile material forms within the INEEL that operate in accordance with an Approved Transportation Plan (ATP) that meets the k_{eff}, cooling, movement and placement requirements, and related administrative controls, qualify as APPROVED FUEL STORAGE provided applicable requirements defined in the ATP to meet these specifications are carried out at the ATR FACILITY. Containers for shipping fissile material forms that are authorized shipping packages qualify as APPROVED FUEL STORAGE.
ATR QUADRANT	A portion of the ATR CORE consisting of the ten FUEL ELEMENTS cooled by one of the four quadrant flow systems and whose power is measured by one of the quadrant water power channels. There are four quadrants designated as NW, NE, SW and SE.
AUTOMATIC	Structures, systems or components configured to perform their intended functions independent of human influence.
BOUNDARY FAULT LEAKAGE	As applied to the PCS is leakage through a fault in a PCS component body, pipe wall, or vessel wall within the RADIOGRAPHIC LIMIT (does not include leakage past seals, gaskets, or valve packing).
CALIBRATION	A local test performed on instrumentation to ensure its output responds within acceptable tolerance to known values of the input parameters. The acceptable tolerance is based on ensuring that the instrument remains operational between successive tests by accounting for uncertainties in the test and instrumentation drift.
CASCADING	The positive reactivity feedback due to voiding in flux trap EXPERIMENT facilities during a reactor power excursion. The reactivity insertion from the flux trap EXPERIMENT facility that initiated the excursion is not included in the CASCADING.
CHANNEL / INSTRUMENT CHANNEL	All components and interconnections, from sensor to COMPARATOR/CLLC inclusive, necessary to monitor a plant variable or condition and to initiate a trip when the variable or the condition deviates beyond a set limit. A CHANNEL terminates at the output of the COMPARATOR/CLLC (bistable).
COMPARATOR	A module having a logical output which is a function of the SETPOINT and the value of the monitored analog variable input. The logical output has two, and only two, operating discrete stable output states defined as "reset" (normal) and "tripped".
COMPLETION TIME	The maximum permitted time interval between the discovery of exceeding a SAFETY LIMIT or OPERATING LIMIT and the satisfactory completion of the associated Required Action.

1.1 DEFINITIONS (continued)

CONFINEMENT	The ATR substructure and superstructure intended to partially contain fission products in case of a severe accident involving the PRIMARY COOLANT SYSTEM and/or installed EXPERIMENTS, with all penetrations designed to minimize leakage. The substructure includes all first and second basement areas except stairwell Number 3. The superstructure includes the reactor area, reactor control room, instrument repair room, reactor data acquisition computer room, freight elevator, passenger elevator, and stairwells Number 1, 2 and 4.
CONTRACTOR	The company to which DOE has awarded the primary contract for operation of the ATR facility.
CONTROL RODS	Particular rods that are used for making adjustments in reactivity or power distribution. These are further classified as outer shim control cylinders, SAFETY RODS, neck shim rods, and regulating rods. Certain of the neck shim rods may be selected as regulating rods.
CORE	The general region in the Reactor Vessel, within the Core Reflector Tank, containing FUEL ELEMENTS, CONTROL RODS, EXPERIMENT irradiation facilities and reflector.
CORE CHANGES	The manual movement or manipulation of any component within the Core Reflector Tank with two or more FUEL ELEMENTS in the reactor CORE. The manual movement of components (e.g., SAFETY RODS) using their normal drive mechanisms is not considered a CORE CHANGE however; movement of these components with their normal drive mechanisms disconnected is considered a CORE CHANGE.
CRITICAL HEAT FLUX	A heat transfer crisis condition in which a heat flux is reached or exceeded and a transition from nucleate boiling to film boiling occurs leading to a rapid surface temperature rise.
DEPRESSURIZED OPERATION	Defined in this text in Table 1.3-1, Reactor Operating Conditions.
DEPRESSURIZED SHUTDOWN	Defined in this text in Table 1.3-1, Reactor Operating Conditions.
EMERGENCY FIREWATER INJECTION SYSTEM (EFIS)	An ESF that provides an emergency coolant source to maintain water level over the CORE, consisting of two subsystems: 1) upper vessel injection (UV) and 2) bottom head (BH) injection with ACTUATION SYSTEMS to control each, and supplied via the critical firewater path by TRA Firewater System through valve GT-10-63 for the upper vessel and through valve GT-10-35 for the bottom head.
ENGINEERED SAFETY FEATURE (ESF)	An essential protective system or subsystem that functions to mitigate the consequences of an incident by providing protection other than scram. ESF systems are the Primary Pump Shutoff, Pressurizing and Gland Seal Pump Shutoff, LOCA PCP Shutoff, EFIS, Vessel Vent, RMSS, PCS Relief Valves and Vessel Level Alarm.

1.1 DEFINITIONS (continued)

EXPERIMENT	An inpile test assembly, pneumatic irradiation capsule, or other hardware device inserted into the reactor to utilize the reactor neutrons or radiation generated therein. The material content of such hardware device including structural components, encapsulation or confining boundaries, and contained fluids or solids are part of the EXPERIMENT.
FISSION BREAK	A significant release of fission products to the PCS due to the breach of a fissile material container.
FLOW INSTABILITY	A heat transfer crisis condition in thin, heated cooling channels with down flowing coolant due to hydraulic instability. The normal downflow becomes hydraulically unstable due to increased flow resistance caused by significant boiling within the channel, resulting in channel voiding, drastically reduced heat transfer, and high surface temperatures. In ATR FUEL ELEMENTS, FLOW INSTABILITY is predicted to occur when the bulk coolant temperature rise of the hot track equals or exceeds the temperature rise required to produce saturation temperature at the outlet.
FUEL ELEMENT	The assembly of fuel plates, side plates and end boxes designed to be installed in the ATR fuel annulus.
FUNCTIONAL TEST	Test performed on a system or piece of equipment to determine if it is capable of performing its intended function.
HOLDDOWN REACTIVITY MARGIN	The amount the reactor remains subcritical for reactivity changes during shutdown evolutions.
IMMEDIATELY	Term used as a COMPLETION TIME to indicate the Required Action is to be pursued without delay and in a controlled manner.
INSERVICE INSPECTION	Inspection that comprises the methods and actions for assuring the structural and pressure-retaining integrity of plant components.
IRRADIATED	As applied to FUEL ELEMENTS, any FUEL ELEMENT which has cooled less than or equal to 1200 days after operation in the CORE with reactor power greater than 20 kW unless the FUEL ELEMENT has been shown to have a decay heat generation < 30 W.
LEAKAGE	As applied to the PCS is leakage from identified sources and from sources that have not been specifically located.
LIMITING CONDITIONS OPERATION (LCO)	The lowest acceptable conditions to ensure functional capability or performance level of SAFETY-RELATED structures, systems, and components and their support systems required for normal safe operation of the facility.
LIMITING CONTROL SETTING (LCS)	Setting on the RSS that prevents process variables from exceeding SAFETY LIMITS during normal operation (Condition 1) and during anticipated transients (Condition 2).
LOBE	One of five portions of the CORE consisting of eight FUEL ELEMENTS surrounding one inner flux trap. Identified as Northwest (NW), Northeast (NE), Center (C), Southwest (SW), or Southeast (SE).

1.1 DEFINITIONS (continued)

LOOP FACILITY	An irradiation facility consisting of independent piping, pumps, pressurizers, and in-pile piping installed to use one or more ATR flux trap baffle positions.
LOW POWER OPERATION	Defined in this text in Table 1.3-1, Reactor Operating Conditions.
MANUAL	As applied to the EFIS, means one manually operated valve in applicable flow path(s), in the closed position.
N_F	The reactor operating steady power level (up to and including 250 MW). The power level N_F represents may change, not only for each reactor operating cycle but also during an operating cycle. During a SCHEDULED STARTUP, N_F is the projected/anticipated full operating power level. Actual EXPERIMENT requirements or operational considerations may necessitate small deviations from the projected value. The neutron level and wide range neutron SUBSYSTEMS are adjusted such that upon reaching N_F the indication is equal to or greater than 100%.
N_L	For POWER OPERATION and LOW POWER OPERATION, a power level which corresponds to 1% N_F. For DEPRESSURIZED OPERATION, the maximum allowed reactor power level (0.5 MW).
OPERABLE/ OPERABILITY	When a system, subsystem, component, or device is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, components, electrical power, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, component, or device to perform its function(s) are also capable of performing their related support function(s). General principles of operability are as follows: 1. A system is considered OPERABLE as long as there exists assurance that it is capable of performing its specified safety function(s). 2. A system can perform its specified safety function(s) only when all of its necessary support systems are capable of performing their related support functions. 3. Assuring the capability to perform a safety function is an ongoing and continuous process. 4. When all systems designed to perform a certain safety function are not capable of performing that safety function, a loss of function condition exists. Facility operation with such a condition may not continue (ACTION statements are provided to identify and respond to a loss of function). 5. When a system is determined to be incapable of performing its intended safety function(s), the declaration of inoperability shall be immediate. 6. Any exception to an immediate determination of inoperability must be justified.
OPERATING LIMITS (OPLIMS)	A term used to represent LIMITING CONTROL SETTINGS and LIMITING CONDITIONS for OPERATION.

1.1 DEFINITIONS (continued)

PLANT PROTECTIVE SYSTEM (PPS)	Those active devices, with the associated equipment that initiates their action, whose function in conjunction with passive structures of the plant is necessary to prevent unacceptable release or spread of radioactive materials by preventing plant variables or conditions from reaching their respective SAFETY LIMITS, or mitigating the consequences of exceeding SAFETY LIMITS. The PPS also prevents unacceptable plant damage in accordance with the limits specified by the CONTRACTOR. The PPS includes the Reactor Shutdown Systems (RSS) which cause the release of the SAFETY RODS and the ENGINEERED SAFETY FEATURES (ESF). The RSS instrument SUBSYSTEMS are listed in LCO 3.1.1 Table 3.1.1-1. The PPS ESF systems are the Primary Pump Shutoff, Pressurizing and Gland Seal Pump Shutoff, LOCA PCP Shutoff, Emergency Firewater Injection, Radiation Monitoring and Seal, Vessel Vent Valves, PCS Relief Valves and Vessel Level Alarm.
POWER OPERATION	Defined in this text in Table 1.3-1, Reactor Operating Conditions.
POWER RECOVERY	The attempted restoration of reactor power after a power reduction prior to shutdown by xenon buildup. During a POWER RECOVERY all surveillances shall be maintained for the reactor operating condition (Table 1.3-1) at the time of the power reduction; i.e., for surveillances, mode Applicability is not exited. A POWER RECOVERY is distinguished from, and does not include XENON RESTART.
PRESSURIZED SHUTDOWN	Defined in this text in Table 1.3-1, Reactor Operating Conditions.
PRESSURIZED STANDBY	Defined in this text in Table 1.3-1, Reactor Operating Conditions.
PRIMARY COOLANT SYSTEM (PCS)	The system, including all connecting piping and associated valves, and pumps that circulates demineralized water at elevated pressure to remove reactor generated heat. It consists of the following major components: Reactor Vessel, Primary Coolant Pumps, Emergency Primary Coolant Pumps, Primary Heat Exchangers (tube side), Primary Surge Tank, Primary Pressurizing Pumps and Primary Degassing Tank.
RADIOGRAPHIC LIMIT	The interface between the PCS piping larger than 3 inches in diameter and certain other piping branches. The limits are defined by Drawing 414577.
REACTOR DEFUELED	Defined in this text in Table 1.3-1, Reactor Operating Conditions.
REACTOR OPERATION	Any DEPRESSURIZED OPERATION, LOW POWER OPERATION, or POWER OPERATION as defined in Table 1.3-1, Reactor Operating Conditions.
REACTOR OUTAGE	A REACTOR SHUTDOWN for the performance of evolutions such as maintenance, refueling, modification, or EXPERIMENTS change.
REACTOR SHUTDOWN	Whenever the reactor is not in the REACTOR OPERATION condition.
REACTOR STARTUP	Withdrawal of SAFETY RODS with the intent of making the reactor critical.

1.0 USE AND APPLICATIONS

1.1 DEFINITIONS (continued)

SAFETY LIMIT (SL)	Limit on a process variable associated with those physical barriers that are necessary for the intended facility function and that are found to be required to guard against the uncontrolled release of radioactivity.
SAFETY RELATED	Those structures, systems, and components relied upon during or following design-basis events to ensure (a) the integrity of the reactor coolant pressure boundary, (b) the capability to shut down the reactor and maintain it in a safe REACTOR SHUTDOWN condition, and (c) the capability to prevent or mitigate the consequences of accidents that result in potential offsite exposures comparable to the 10 CFR 100 guidelines.
SAFETY RODS	Those cylindrical assemblies with hafnium absorber plates which are released from their drive mechanisms and drop into the CORE upon receipt of a trip signal from the Reactor Shutdown System (RSS). A safety rod assembly with the hafnium absorber plates removed is not a safety rod.
SCHEDULED STARTUP	Withdrawal of SAFETY RODS for REACTOR OPERATION following a REACTOR OUTAGE.
SETPOINT	The value at which a protective subsystem is set to initiate an alarm or required protective action.
SOURCE POWER	The power generated in the vicinity of a flux trap. For inner flux traps (i.e., those inside of the fuel annulus), the SOURCE POWER is the local LOBE power, whereas for outer flux traps, the SOURCE POWER is the average of the three adjacent LOBE powers.
STANDBY	The condition in which the subject equipment is not operating but has been functionally tested to verify it is OPERABLE and capable of being placed in-service, either AUTOMATICALLY or manually, within a specified time frame.
SUBSYSTEM	A group of PPS CHANNELS which monitor the same variable.
SUBSYSTEM LOGIC	A PPS subassembly which receives trip signals from SUBSYSTEM INSTRUMENT CHANNELS and provides a trip signal as an output when a sufficient number of INSTRUMENT CHANNEL trips occur in a prescribed combination.
TOTAL RESPONSE TIME/RESPONSE TIME	The interval between the time when a measured variable reaches the SETPOINT at the SUBSYSTEM sensor and the time at which the SAFETY RODS release, the pump starts to coast down, or the valve is open. It is the sum of the appropriate CHANNEL RESPONSE TIME, logic RESPONSE TIME, ACTUATOR CONTROLLER RESPONSE TIME and ACTUATOR RESPONSE TIME. The RESPONSE TIME may be measured by means of any series of sequential, overlapping, or total steps so that the entire RESPONSE TIME can be determined.
XENON RESTART	Withdrawal of the SAFETY RODS with the intent of making the reactor critical, following loss of startup reactivity due to xenon poisoning, after xenon poison decays sufficiently to allow adequate startup reactivity.

1.2 ORGANIZATION

The SAFETY and OPERATING LIMITS and their associated ACTIONS and surveillances are specified in Section 2 - SAFETY LIMITS, and Section 3/4 - LIMITING CONTROL SETTINGS and LIMITING CONDITIONS for OPERATION. Section 2 is organized by the specific SAFETY LIMIT (SL). Each SL has an applicability statement and an ACTION table. Section 3/4 is organized into general application and specific LIMITING CONTROL SETTINGS (LCS) or LIMITING CONDITIONS for OPERATION (LCO). Section 3/4.0 establishes general requirements applicable to all LCSs and LCOs. Sections 3/4.1 through 3/4.9 provide specific LCS and LCO requirements. Each LCS and LCO has an applicability statement(s), ACTION(s), and surveillance requirements. Many of the LCSs and LCOs specify limitations using tables to describe parameters of components, such as valve alignment, instrumentation, or CORE power.

ACTION tables are set up in three columns that list the Conditions, Required Actions for the particular conditions, and the COMPLETION TIMES for the Required Action. The surveillance tables list the surveillance requirements and their accompanying surveillance intervals.

Administrative Controls are specified in Section 5.

1.2.1 SAFETY AND OPERATING LIMITS

SAFETY LIMITS and OPERATING LIMITS maintain the facility operation within the safety envelope established by SAR-153 (limit the consequences of high-probability events to those of operational incidents).

1.2.1.1 SAFETY LIMITS

SAFETY LIMITS (SLs) are limits on process variables associated with those physical barriers that are necessary for the intended facility function and that are found to be required to guard against the uncontrolled release of radioactivity.

The SAFETY LIMITS for the CORE are selected from the thermal-hydraulic parameters and the LOBE power. The total CORE power is not a significant parameter for operation compared to the LOBE power; therefore, the SAFETY LIMIT is on LOBE power. For the CORE, the SLs are the values of each variable for which the margin to CRITICAL HEAT FLUX (which includes flow instabilities that could lead to rapid temperature increase) is at least 3σ (standard deviations of the normal distribution). SLs for the piping are selected from the design values considered for the piping. For the primary coolant piping, the SLs are values that ensure the probability of failure for the pressure boundary of the PRIMARY COOLANT SYSTEM is acceptably low.

The parameters that have direct impact on the safety of the reactor through protection of the barriers to fission product release are (a) LOBE steady state power, (b) CORE differential pressure (high), (c) PRIMARY COOLANT SYSTEM (PCS) pressure, (d) CORE flow rate (low CORE differential pressure), and (e) PCS fluid temperature.

1.2 ORGANIZATION (continued)

Each SAFETY LIMIT describes as precisely as possible the parameters being limited in measurable units (pressure, temperature, etc.). Applicability statements have been provided to describe the mode(s) in which the SL shall be complied with. ACTION statements describe the ACTION(s) to be taken when an SL is not met. Exceeding an SL is a TSR violation and a reactor scram shall be IMMEDIATELY initiated and any other applicable specific ACTION(s) completed. Follow up actions as described in Section 5 Administrative Controls, include a technical evaluation to determine the extent of damage and the OPERABILITY of the system(s) or component(s) and place restrictions on approval for the restart of the affected system(s) or component(s).

1.2.1.2 General Application

Section 3/4.0 establishes the general requirements applicable to all LIMITING CONTROL SETTINGS and LIMITING CONDITIONS for OPERATION, including surveillance requirements, and apply at all times, unless otherwise stated. This section provides guidance in a narrative format and does not contain ACTION and surveillance tables. In general, the direction provided in this section is in itself a description of use and application for LCSs, LCOs and SRs.

1.2.1.3 LIMITING CONTROL SETTINGS

LIMITING CONTROL SETTINGS (LCSs) are settings on safety systems that control process variables to prevent exceeding SAFETY LIMITS during normal operation and during Condition 2 events. The LCSs provide REACTOR SHUTDOWN through the Reactor Shutdown System (RSS) as required in the accident analysis. The RSS is designed to limit high-probability events to operational incidents (no damage to the FUEL ELEMENT cladding).

The LCS(s) describe the RSS SETPOINT values, ACCURACY, and RESPONSE TIMES for parameters in measurable units (pressure, temperature, etc.). Applicability statements have been provided to describe the mode(s) in which the LCS shall be complied with. ACTION statements describe the ACTIONS to be taken when an LCS is not met.

If an AUTOMATIC safety system does not function as required, appropriate action shall be taken to compensate for the failure of the AUTOMATIC safety systems. This action may take the form of a REACTOR SHUTDOWN or ENGINEERED SAFETY FEATURE initiation. Failure to comply with LCS requirements or ACTION(s) as described in Section 5.4 Administrative Controls, constitutes a TSR violation and follow up actions may be required as specified in AC 5.4.4.2.

1.2.1.4 LIMITING CONDITIONS FOR OPERATION

LIMITING CONDITIONS for OPERATION (LCOs) are the lowest functional capability or performance level of SAFETY-RELATED structures, systems, and components (SSC) and their support systems required for normal safe operation of the facility.

1.2 ORGANIZATION (continued)

The LCO(s) describe SETPOINT values, ACCURACY, and RESPONSE TIMES for parameters in measurable units (pressure, temperature, etc.). Applicability statements have been provided to describe the mode(s) in which the LCO shall be complied with. ACTION statements describe the ACTIONS to be taken when an LCO is not met. Failure to comply with LCO requirements or ACTION(s) as described in Section 5.4 Administrative Controls, constitutes a TSR violation and follow up actions may be required as specified in AC 5.4.4.2.

1.2.1.5 Surveillance Requirements

Surveillances are required to ensure that structures, systems, and components (SSC) required for the safe operation of the reactor are OPERABLE and to ensure that required ranges of parameters are maintained. Surveillance requirements are requirements relating to tests, test intervals, CALIBRATIONS, or inspections to ensure OPERABILITY of SAFETY-RELATED SSC and to ensure that the facility operation will be within the SAFETY LIMITS and that the LIMITING CONTROL SETTINGS and the LIMITING CONDITIONS for OPERATION will be met.

Since surveillance requirements must be met for all SSC to be considered OPERABLE, failure to perform a surveillance within the required time interval or failure of a surveillance test may result in the SSC being declared inoperable. Section 3/4.0 provides specific guidance for failing to meet or perform surveillance requirements and places limitations on mode changes when all surveillance requirements for the SSC required for the new condition are not met. Failure to comply with surveillance requirements as described in Section 5.4 Administrative Controls, constitutes a TSR violation and follow up actions may be required as specified in AC 5.4.4.3.

1.2.2 Administrative Controls

Administrative Controls are the requirements for organization and management, procedures, record keeping, reviews and audits, and programs necessary to ensure safe operation of the ATR in accordance with the other sections of the TSR. Among other requirements, Administrative Controls contain the requirements for reporting deviations from TSR, for staffing facility positions important to safe operation of the ATR, and the physical and administrative controls of the nuclear criticality safety program.

Administrative Controls (ACs) are in effect during all modes of applicability unless otherwise stated within a particular AC. Failure to comply with an AC is considered a TSR violation and follow up actions may be required as specified in AC 5.4.4.4. There are three types of Administrative Controls; specific requirements (e.g., staffing, fuel handling and storage requirements), requirements for establishing, implementing, and maintaining institutional safety programs (e.g., fire protection, maintenance, configuration management), and requirements for certain elements of institutional safety programs (e.g., identification of configuration items, procedures for normal operation). Programs and programmatic elements encompass a large number of details. Noncompliance with a specific part or detail of a required TSR institutional safety program or programmatic element (e.g., failure to identify or control an individual

1.2 ORGANIZATION (continued)

configuration item) does not constitute failure to comply with the Administrative Control unless it significantly increases the risk of facility operation. Noncompliance with a specific requirement in the TSR Administrative Controls section (e.g., a fuel handling or storage requirement) does constitute a failure to comply with the Administrative Control. The specific categorization of each Administrative Control is shown in the following table.

Specific Requirements	Institutional Safety Programs and Programmatic Elements
AC 5.3.2.1, AC 5.4.3, AC 5.4.4.1 a, AC 5.4.4.2 a, AC 5.4.4.3 a, AC 5.7.7.2.	AC 5.2.1, AC 5.2.2, AC 5.3.1, AC 5.3.2.2, AC 5.3.3.1, AC 5.3.3.2, AC 5.4.1, AC 5.4.2, AC 5.4.4.1 b, c&d, AC 5.4.4.2 b&c, AC 5.4.4.3 b&c, AC 5.4.4.4, AC 5.4.5, AC 5.5.1, AC 5.6.1, AC 5.6.2, AC 5.7.1, AC 5.7.2, AC 5.7.3, AC 5.7.4, AC 5.7.5, AC 5.7.6, AC 5.7.7.1, AC 5.7.8, AC 5.7.9, AC 5.7.10, AC 5.8.1, AC 5.8.2, AC 5.8.3, AC 5.8.4, AC 5.8.5, AC 5.8.6, AC 5.8.7, AC 5.9.1.

1.0 USE AND APPLICATIONS

1.3 MODE APPLICABILITY

The reactor operating conditions for which each Technical Safety Requirement applies is stated with each requirement under the heading of Applicability. Table 1.3-1 provides a list of reactor operating conditions or modes established for the ATR. The table distinguishes between different modes of operation based on reactor power level, reactor vessel inlet pressure, AUTOMATIC or MANUAL operation of EFIS and fuel loading. The modes described have been chosen in an effort to provide definite lines of demarcation between modes, yet hold the number of modes to a minimum.

It should be recognized that certain facility conditions exist or evolutions are performed in which the modes listed in Table 1.3-1 are not linked directly to the condition. In these circumstances a modifying description has been added to the modes to further distinguish the required conditions. The most common example of a mode modifier is the term "with IRRADIATED FUEL ELEMENTS". This modifier when used with a reactor operating condition such as DEPRESSURIZED SHUTDOWN further describes and establishes a submode of DEPRESSURIZED SHUTDOWN when the Reactor Vessel contains IRRADIATED FUEL ELEMENTS. Following is a list of examples showing different modifiers or unique conditions used throughout the TSR:

Modifiers

- For 30 minutes following ...
- ...(2 PCP) or (3 PCP)
- Prior to initial use in ...

Special Conditions

- All times
- During cask movement with IRRADIATED FUEL ELEMENTS in the Reactor Vessel
- With IRRADIATED FUEL ELEMENTS in the canal
- PCP operation
- During CORE CHANGES
- Following POWER OPERATION
- When loop conditions are > 200°F and 750 psig (1,500 psig AHTL)
- until:

  1. SR 4.5.1.2, FUEL ELEMENT cooling time, is met

  OR

  2. ≤ 1 operating PCP.

1.0 USE AND APPLICATIONS

1.3 MODE APPLICABILITY (continued)

Table 1.3-1 Reactor Operating Conditions

REACTOR OPERATING CONDITION	REACTOR POWER LEVEL (MW)	ADDITIONAL IDENTIFYING CONDITIONS	METHODS OF CHANGING REACTOR OPERATING CONDITIONS
POWER OPERATION	> 3	PCS at operating pressure EFIS in AUTOMATIC	Enter from LOW POWER OPERATION by increasing reactor power level.
LOW POWER OPERATION	≤ 3	PCS at operating pressure EFIS in AUTOMATIC	Enter from PRESSURIZED STANDBY by withdrawing SAFETY RODS with the intent of making the reactor critical. Enter from POWER OPERATION by reducing the reactor power level.
PRESSURIZED STANDBY	0	PCS above atmospheric pressure EFIS in AUTOMATIC	Enter from POWER or LOW POWER OPERATION as a result of reactor scram or shutdown procedure. Enter from PRESSURIZED SHUTDOWN or REACTOR DEFUELED by establishing the PCS boundary with EFIS in AUTOMATIC.
PRESSURIZED SHUTDOWN	0	PCS above atmospheric pressure EFIS in MANUAL	Enter from PRESSURIZED STANDBY by placing EFIS in MANUAL. Enter from DEPRESSURIZED SHUTDOWN or REACTOR DEFUELED by establishing the PCS boundary with EFIS in MANUAL.
DEPRESSURIZED OPERATION	≤ 0.5	PCS at atmospheric pressure EFIS in AUTOMATIC or MANUAL	Enter from DEPRESSURIZED SHUTDOWN by withdrawing SAFETY RODS with the intent of making the reactor critical.
DEPRESSURIZED SHUTDOWN	0	PCS at atmospheric pressure EFIS in AUTOMATIC or MANUAL	Enter from DEPRESSURIZED OPERATION as a result of reactor scram or shutdown procedure. Enter from PRESSURIZED SHUTDOWN or PRESSURIZED STANDBY by opening a Vessel Vent valve or equivalent (above CORE elevation). Enter from REACTOR DEFUELED by loading a FUEL ELEMENT into the reactor vessel.
REACTOR DEFUELED	0	PCS at atmospheric pressure EFIS not required	Enter from DEPRESSURIZED SHUTDOWN by removing all FUEL ELEMENTS from the Reactor Vessel. (Fueled EXPERIMENTS may remain in the CORE.) Enter from PRESSURIZED STANDBY or PRESSURIZED SHUTDOWN by opening a Vessel Vent valve or equivalent (above CORE elevation).

1.0 USE AND APPLICATIONS

1.4 LOGICAL CONNECTORS

1.4.1 Discussion

Logical connectors are used in the TSR to discriminate between, and yet connect, discrete Conditions, Required Actions, COMPLETION TIMES, Surveillances, and Frequencies. The only logical connectors that appear in the TSR are **AND** and **OR**. The physical arrangement (e.g., indentation) of these connectors constitutes logical conventions with specific meanings.

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to the Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and successive indentations of the logical connectors.

When logical connectors are used to state a Condition, COMPLETION TIME, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of Condition, COMPLETION TIME, Surveillance, or Frequency. Nesting of logical connectors is used in Required Actions as needed to convey the requirements.

1.4.2 Examples

The following examples illustrate the use of logical connectors.

Example 1.4.2-1

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify ... **AND** A.2 Restore ...	

In this example the logical connector **AND** is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.4 LOGICAL CONNECTORS (continued)

Example 1.4.2-2

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore ... **OR** A.2.1 Trip ... **AND** A.2.2.1 Verify ... **OR** A.2.2.2 Reduce... **OR** A.3 Scram the reactor.	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector **OR** and the left justified placement. Any one of these three ACTIONS may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector **AND**. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector **OR** indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.5 COMPLETION TIMES

1.5.1 Discussion

LIMITING CONDITIONS for OPERATION specify minimum requirements for ensuring safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and COMPLETION TIME(S).

COMPLETION TIME is the time interval allowed to complete an ACTION. It is referenced to the time a situation (e.g., inoperable equipment or process variable not within limits) is discovered that requires entering an ACTIONS Condition, provided the facility is in the operating condition as stated in the applicability statement. The Required Action(s) must be completed before the specified COMPLETION TIME expires. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the system or facility is no longer within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions should be performed within their associated COMPLETION TIME. When in multiple Conditions, each COMPLETION TIME shall be tracked separately starting from the time of discovery of the situation that required entry into the Condition.

Once a condition has been entered, subsequent failures or process variables discovered to be not within limits, will result in a separate entry into the Condition. The Required Actions of the Condition apply to each additional failure, with the COMPLETION TIMES based on the entry time for each failure.

IMMEDIATE COMPLETION TIME	When IMMEDIATELY is used as a COMPLETION TIME, the Required Action is to be pursued without delay and in a controlled manner. A COMPLETION TIME of IMMEDIATELY is used throughout the TSR when the intent is to take ACTION as soon as practical in a controlled manner. Typically, most ACTIONS can be completed within a few minutes; however conditions / evolutions may exist that require longer time periods to complete the ACTION. In these circumstances it is not desirable to provide a finite time period for completion since the intent is to initiate ACTION once the condition has been detected; however performance of the ACTIONS may require a undetermined time period for safe, effective completion.

The following examples illustrate the use of COMPLETION TIMES with different types of conditions and changing conditions.

1.5 COMPLETION TIMES (continued)

1.5.2 Examples

Example 1.5.2-1

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify ...	2 Hours
	AND	
	A.2 Reduce ...	4 Hours

Condition A has two Required Actions. Each Required Action has its own separate COMPLETION TIME. Each COMPLETION TIME is referenced to the time Condition A is entered.

The Required Actions of Condition A are to verify ... within 2 hours **AND** to reduce ... within 4 hours. A total of 2 hours is allowed for verifying ... and a total of 4 hours (not 6 hours) for reducing ...

Example 1.5.2-2

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore ...	30 Minutes
	OR	
	A.2 Reduce ...	

Condition A provides a choice of either A.1 **OR** A.2 as Required Actions. The COMPLETION TIME of 30 minutes applies to either of the choices. When only one COMPLETION TIME is stated for a condition, the time applies to all Required Actions.

Example 1.5.2-3

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Declare system inoperable.	IMMEDIATELY
	AND	
	A.2 Verify pressure greater than ...	Every 8 Hours

When the LCO has not been met the Required Action A.2 requires repetitive verification (identified by the term "Every") on an interval of 8 hours. The initial completion of A.2 must be within 8 hours of entering Condition A. A 25% extension is allowed for completion of repetitive ACTIONS (as allowed for in SR 4.0.2), thus each succeeding verification may be performed within 1.25 times the specified 8 hour interval or 10 hours.

1.5 COMPLETION TIMES (continued)

Example 1.5.2-4

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pump inoperable.	A.1 Restore pump OPERABILITY.	30 Minutes
B. Required Action and associated COMPLETION TIME of Condition A not met.	B.1 Verify ... **AND** B.2 Reduce ...	4 Hours 8 Hours

When the pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 30 minutes, Condition B is entered and the COMPLETION TIME clocks for ACTIONS B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

Example 1.5.2-5

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. LEAKAGE greater than 10 gpm.	C.1 Reduce LEAKAGE to 10 gpm or less.	Prior to entering LOW POWER OPERATION

In this example an LCO limit on LEAKAGE for a system prior to entering a mode change into LOW POWER OPERATION has been exceeded. The Required Action for reducing the leakage to less than 10 gpm does not have a finite COMPLETION TIME but must be completed prior to entering the LOW POWER OPERATION mode. Where a COMPLETION TIME is based upon changing modes of operation, continued operation in the present mode is considered acceptable for an indefinite period of time unless otherwise specified.

1.0 USE AND APPLICATIONS

1.6 FREQUENCY NOTATIONS

1.6.1 Discussion

Each Surveillance Requirement (SR) has a specified frequency which must be met in order to meet the associated LCO. Frequency notation may at times be used to specify surveillance intervals. The frequency notations are listed in Table 1.6.1-1.

Table 1.6.1-1 Frequency Notations

NOTATION	FREQUENCY
Each Shift	At least once per 12 hours
Daily	At least once per 24 hours
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Semi-Annually	At least once per 184 days
Annually/Yearly	At least once per 366 days
Startup	Prior to SCHEDULED STARTUP **OR** XENON RESTART **NOTE** **Does not include POWER RECOVERY**

1.6.2 Examples

The following examples illustrate the various ways that frequencies are specified.

Example 1.6.2-1

SURVEILLANCE REQUIREMENTS:

SR #	SURVEILLANCE REQUIREMENT	FREQUENCY
	Verify CALIBRATION	Annually

Example 1.6.2-1 contains the type of SR most often encountered in the TSR. The frequency specifies an interval of (Annually) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as Annually, an extension of the time interval to 1.25 times the interval specified is allowed by SR 4.0.2 for operational flexibility. The measurement of this interval continues at all times even when the SR is not required to be met per SR 4.0.1 (such as when the equipment is inoperable or when the unit is outside the applicability of the LCO).

If the interval specified (plus the 25% extension) is exceeded while the system or facility is in a mode where the surveillance is required, then SR 4.0.3 provides guidance to complete the SR within 24 hours or the specified frequency, whichever is less.

1.0 USE AND APPLICATIONS

1.6 FREQUENCY NOTATIONS (continued)

If the interval specified (plus the 25% extension) is exceeded while the system or facility is not in a mode where the surveillance is required, the Surveillance must be performed within the frequency requirements prior to entry into the mode or specified condition. Failure to do so would result in a violation of SR 4.0.4.

Example 1.6.2-2

SURVEILLANCE REQUIREMENTS:

SR #	SURVEILLANCE REQUIREMENT	FREQUENCY
	Verify connection valves (critical firewater path) are open.	Startup **AND** Monthly

Example 1.6.2-2 has two frequencies. The first one is a one time performance frequency and the second one is of a type shown in Example 1.6.2-1. The logical connector **AND** indicates that both Frequency requirements must be met. The use of "startup" indicates a single performance as defined in Table 1.6.1-1. This type of frequency does not qualify for the extension allowed by SR 4.0.2.

Table 1.6.1-1 defines "startup" as "Prior to SCHEDULED STARTUP **OR** XENON RESTART". The surveillance must be performed prior to each startup. The use of "monthly" indicates that the surveillance requirement must be repeated each month (extension applies) during applicability. There are no time restrictions associated with "startup" that limit the performance of the SR within a finite time prior to startup; however the SR must be performed after any activity that may invalidate the SR. If the SR was performed and subsequently invalidated for any reason, the SR must be performed again to comply with SR 4.0.4.

Example 1.6.2-3

SURVEILLANCE REQUIREMENTS:

SR#	SURVEILLANCE REQUIREMENT	FREQUENCY
	Verify charger supplying E-11 (A through D)	Prior to entering Applicability

Example 1.6.2-3 contains the type of SR commonly used in the TSR. Prior to entering any single mode listed in an LCOs Applicability, the requirement of the LCO surveillance must be met. Since the same surveillance applies to each mode individually, it can be considered to apply to all listed modes as a group. Thus, when a Surveillance has a frequency of prior to entering Applicability, the Surveillance must only be completed prior to entry into any one of the modes listed, but not prior to subsequent entry into other listed mode(s). As long as the system remains in one of the applicable modes the Surveillance Requirement continues to met. If the system is subsequently configured such that none of the listed modes are Applicable, then the surveillance must again be performed once prior to entering any of the Applicable modes.

1.6 FREQUENCY NOTATIONS (continued)

There are no time restrictions associated with "prior to Applicability" that limit the performance of the SR within a finite time. However, if the SR was performed and subsequently invalidated for any reason, the SR must be performed again to comply with SR 4.0.4. This type of frequency does not qualify for the extension allowed by SR 4.0.2.

Consider as an initial condition, the reactor operating at 100 MWs and the Applicability for an LCO is as follows:

APPLICABILITY:

- POWER OPERATION
- For 30 minutes following POWER OPERATION
- LOW POWER OPERATION
- PRESSURIZED STANDBY
- PRESSURIZED SHUTDOWN
- DEPRESSURIZED OPERATION
- DEPRESSURIZED SHUTDOWN with IRRADIATED FUEL ELEMENTS in the Reactor Vessel
- ≥ 2 PCP Operation.

This concept is illustrated in the following example. If the reactor is shutdown for a SCHEDULED OUTAGE, all Surveillances must be maintained current and mode Applicability will exist until all IRRADIATED FUEL ELEMENTS have been discharged from the Reactor Vessel. Mode Applicability could again be entered by several different evolutions e.g., reloading IRRADIATED FUEL ELEMENTS, pressurizing the PCS with EFIS in MANUAL (PRESSURIZED SHUTDOWN) or EFIS in AUTOMATIC (PRESSURIZED STANDBY), or by withdrawing SAFETY RODS for DEPRESSURIZED OPERATION. For this example, the reactor will be returned to 100 MW operation. Mode Applicability will be entered by loading an IRRADIATED FUEL ELEMENT in the Reactor Vessel. The surveillance is required prior to entering Applicability (i.e., loading the first IRRADIATED FUEL ELEMENT) and is maintained (is not to be performed again) through all subsequent Applicable mode changes. The surveillance is required only once since mode Applicability is maintained from the time the first IRRADIATED FUEL ELEMENT was loaded and continues during 100 MW POWER OPERATION.

Section 2

Safety Limits

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2.0 SAFETY LIMITS

2.1 CORE THERMAL PERFORMANCE PARAMETERS

2.1.1 STEADY-STATE POWER

SL: LOBE steady-state power shall not exceed (POWER OPERATION):

88.5 MW for operation with two PCP.
OR
96.0 MW for operation with three PCP.

CORE steady-state power shall not exceed (DEPRESSURIZED OPERATION):

7.3 MW

APPLICABILITY:

- POWER OPERATION
- DEPRESSURIZED OPERATION

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LOBE or CORE Power above limit.	A.1 Scram the reactor.	IMMEDIATELY

2.0 SAFETY LIMITS

2.1 CORE THERMAL PERFORMANCE PARAMETERS

2.1.2 REACTOR VESSEL DIFFERENTIAL PRESSURE (LOW)

SL: Reactor Vessel Differential Pressure shall not be less than:

30.6 psid for operation with two PCPs.

OR

39.3 psid for operation with three PCPs.

APPLICABILITY:

- POWER OPERATION
- LOW POWER OPERATION

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Differential Pressure below the limit with SAFETY RODS not released.	A.1 Scram the reactor.	IMMEDIATELY

2.0 SAFETY LIMITS

2.1 CORE THERMAL PERFORMANCE PARAMETERS

2.1.3 REACTOR INLET PRESSURE

SL: Reactor Inlet Pressure (measured in the 36-inch inlet piping to the vessel) shall not be less than:

146.7 psig for operation with two PCP.

OR

170.6 psig for operation with three PCP.

APPLICABILITY:

- POWER OPERATION
- LOW POWER OPERATION

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Inlet Pressure below the limit with SAFETY RODS not released.	A.1 Scram the reactor.	IMMEDIATELY

2.0 SAFETY LIMITS

2.1 CORE THERMAL PERFORMANCE PARAMETERS

2.1.4 REACTOR COOLANT FLOW

SL: Reactor Coolant Flow shall not be less than 302 gpm.

APPLICABILITY:

- DEPRESSURIZED OPERATION

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Coolant Flow below the limit with SAFETY RODS not released.	A.1 Scram the reactor.	IMMEDIATELY

2.0 SAFETY LIMITS

2.2 PRIMARY COOLANT SYSTEM

2.2.1 PRIMARY COOLANT SYSTEM PRESSURE

SL: The PCS pressure shall not exceed:

a. 634 psig, pump discharge to the first block valve.
b. 533 psig, first block valve to butterfly valve.
c. 468 psig, butterfly valve to Reactor Vessel outlet.
d. 429 psig, Reactor Vessel outlet to pump suction.

APPLICABILITY:

- All times

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressure limit(s) exceeded.	A.1 Verify reactor scram. **AND**	IMMEDIATELY
	NOTE **Pressurizing and Gland Seal Pump operation is allowed to maintain CORE inventory or to support evolutions required by a radiological assessment.**	
	A.2 Cease operation of Pressurizing and Gland Seal Pumps. **AND**	IMMEDIATELY
	NOTE **If a FISSION BREAK is suspected, perform a radiological assessment prior to reducing PCS pressure to atmospheric.**	
	A.3 Be in DEPRESSURIZED SHUTDOWN.	2 Hours or per radiological assessment

2.0 SAFETY LIMITS

2.2 PRIMARY COOLANT SYSTEM

2.2.2 **PRIMARY COOLANT SYSTEM TEMPERATURE**

SL: The PCS Temperature shall not exceed 240°F.

APPLICABILITY:

- All times

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PCS temperature limit exceeded.	A.1 Verify reactor scram. **AND** **NOTE** **If a FISSION BREAK is suspected, perform a radiological assessment prior to reducing PCS pressure to atmospheric.** **NOTE** **If a FISSION BREAK is suspected, PCS feed and bleed is only allowed if approved by a radiological assessment.** A.2 Establish a method to reduce PCS temperature to ≤ 240°F.	IMMEDIATELY IMMEDIATELY *or per* radiological assessment

2.0 SAFETY LIMITS

2.2 PRIMARY COOLANT SYSTEM

2.2.3 REACTOR VESSEL DIFFERENTIAL PRESSURE (HIGH)

SL: The Reactor Vessel Differential Pressure shall not exceed 150 psid.

APPLICABILITY:

- All times

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. High Vessel Differential Pressure limit exceeded.	A.1 Verify reactor scram.	IMMEDIATELY
	AND	
	A.2 Reduce the PCS flow until differential pressure ≤ 150 psid.	30 Minutes

2.0 SAFETY LIMITS

2.2 PRIMARY COOLANT SYSTEM

2.2.3 **REACTOR VESSEL DIFFERENTIAL PRESSURE (HIGH)**

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Chapters 3/4

Operating Limits and Surveillance Requirements

General Application

Redacted in Entirety

FOIOA Exemption 3 UCNI, Exemption 7f

3/4 OPERATING LIMITS AND SURVEILLANCE REQUIREMENTS

3/4.0 GENERAL APPLICATION

FOIA Exemption 3 UCNI, Exemption 7f (Chap 3/4)

FOIA Exemption 3 UCNI, Exemption 7f (Chap 3/4)

3.0 LIMITING CONDITIONS FOR OPERATION

LCO 3.0.1	LCOs shall be met during the modes and other specified conditions identified in the Applicability statement, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. If the LCO is restored or is no longer applicable before the specified COMPLETION TIME(S) expires, completion of the ACTION is not required, unless otherwise stated.
LCO 3.0.3	When an LCO is not met and the associated ACTIONS are not met or an ACTION is not provided, the facility shall be placed in a mode or other specified condition in which the LCO is not applicable. Activities shall be initiated to place the affected system(s) or facility in a mode or other specified condition in which the LCO is not applicable within 1 hour. When the appropriate ACTIONS have been determined for placing the system or facility in a mode in which the LCO is not applicable, the COMPLETION TIME of the ACTIONS shall be IMMEDIATELY. Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the ACTIONS required by LCO 3.0.3 are not required.
LCO 3.0.4	A mode or other specified condition in the Applicability shall not be entered, if doing so results in an LCO not being met, except when the associated ACTIONS to be entered permit continued operation for an unlimited period of time. LCO 3.0.4 shall not prevent changes in modes or other specified conditions in the Applicability that are required to comply with ACTIONS. If while operating within an ACTION a power reduction occurs, re-entry into modes or other specified Conditions in the applicability is allowed during POWER RECOVERY, when the ACTIONS to be entered permit operation for a limited time. The COMPLETION TIME remains as the time from which the Condition was initially entered. Exceptions to LCO 3.0.4 are stated in the individual LCOs. When an individual LCO states that LCO 3.0.4 does not apply, then it allows entry into modes or other specified conditions in the Applicability when the associated ACTIONS to be entered permit operation in the mode or other specified condition for only a limited time.
LCO 3.0.5	Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control.

3/4 OPERATING LIMITS AND SURVEILLANCE REQUIREMENTS

FOIA Exemption 3 UCNI, Exemption 7f (Chap 3/4)

3/4.0 GENERAL APPLICATION

FOIA Exemption 3 UCNI, Exemption 7f (Chap 3/4)

4.0 SURVEILLANCE REQUIREMENTS

SR 4.0.1 SRs shall be met during the modes or other specified conditions in the Applicability for individual LCOs unless otherwise stated in the SR. Failure to meet a surveillance required for LCO compliance (whether such failure is experienced during the performance of the surveillance or between performances of the surveillance) shall constitute failure to meet the LCO. Failure to perform a surveillance within the specified frequency shall constitute failure to meet the LCO, except as provided in SR 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 4.0.2 The specified frequency for each SR is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance or as measured from the time a specified condition of the frequency is met.

For frequencies that are required to be performed only once or that are linked to a specific event, the above interval extension does not apply.

If a COMPLETION TIME for an ACTION statement requires repetitive performance on a time interval, the above frequency extension applies to each performance after the initial performance.

SR 4.0.3 If it is discovered that a surveillance required for LCO compliance was not performed within its specified frequency, compliance with the requirement to declare the LCO not met may be delayed from the time of discovery up to 24 hours or up to the limit of the specified frequency, whichever is less. This delay period is permitted to allow performance of the surveillance.

If the surveillance is not performed within the delay period, the LCO shall IMMEDIATELY be declared not met, and the applicable ACTIONS shall be entered. The COMPLETION TIMES of the ACTIONS begin IMMEDIATELY on expiration of the delay period. When the surveillance is performed within the delay period and the surveillance is not met, the LCO shall IMMEDIATELY be declared not met, and the applicable ACTIONS shall be entered. The COMPLETION TIMES of the ACTIONS begin IMMEDIATELY on failure of the surveillance.

SR 4.0.4 Entry into a mode or other specified condition in the Applicability of an LCO shall not be made unless the LCO's surveillances have been met within their stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to Modes or other specified conditions in the Applicability that are required to comply with ACTIONS.

Section 5

Administrative Controls

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5.0 ADMINISTRATIVE CONTROLS

5.1 PURPOSE

Administrative Controls are the requirements for organization and management, procedures, record keeping, reviews and audits, and programs necessary to ensure safe operation of the ATR such that it meets the Technical Safety Requirements.

Administrative Controls (ACs) apply to all activities at the ATR during all REACTOR OPERATING CONDITIONS unless otherwise stated with the particular AC.

5.2 MANAGEMENT RESPONSIBILITIES

AC 5.2.1 The ATR Operations Manager shall be responsible for the following:

a. Overall facility operation and shall establish a policy for the succession to his authority during his absence.

b. Ensuring that the ATR is operated and surveillances are conducted in accordance with the ATR TSR.

c. Establishing controls and policies that ensure operations are conducted following approved procedures.

d. Ensuring that each on-duty shift shall be composed of at least the minimum shift crew composition required in AC 5.3.2.

e. Ensure approved administrative procedures establish overtime and stress guidance for general working conditions that include the maximum number of consecutive working hours and maximum number of consecutive days.

AC 5.2.2 The Shift Supervisor (SS) shall be responsible for the local command function of the facility. During any absence of the SS a certified Senior Reactor Operator (SRO) shall be designated to assume the local command function.

5.0 ADMINISTRATIVE CONTROLS

5.3 ORGANIZATION AND STAFFING

AC 5.3.1 Organization chart updates showing the operations line of authority shall be available to DOE-ID. Organization charts shall show the lines of authority, responsibility, and communication directly affecting ATR. The organization charts shall include operation staff, training, radiation control, and quality assurance support as well as technical and craft support.

AC 5.3.2 Minimum Operations Shift Complement and Actions

AC 5.3.2.1 The minimum allowable ATR staff for various plant conditions shall be as follows:

a. During REACTOR OPERATION.

1. A certified RO or SRO shall be at the controls.

2. A certified SRO shall be in the Reactor Control Room.

3. Minimum Reactor Control Room staffing shall be two.

4. An Experiment Operator (EO) shall be in the vicinity of the loop control consoles to act on alarms.

5. A qualified operator capable of responding to areas outside of building TRA-670 to take accident mitigating action in addition to the staffing requirements for RO, SRO, and EO noted above.

b. During REACTOR SHUTDOWN with two or more fuel elements in the CORE.

NOTE

Not applicable if physical restraints are in place to prevent changes in reactivity.

1. An RO shall be at the controls.

c. During CORE CHANGES through the reactor vessel refueling ports.

1. An SRO certified for fissile material handling shall be at the reactor top with no other duties except supervision of CORE CHANGES.

d. During CORE CHANGES other than through the reactor vessel refueling ports.

1. An SRO certified for fissile material handling shall directly supervise personnel performing the changes.

5.3 ORGANIZATION AND STAFFING (continued)

e. Radiological Control Technician (RCT) staffing.

1. An RCT shall be available at TRA at all times. The RCT may respond to requests for emergency assistance in areas outside TRA, provided that no work requiring constant radiological surveillance is in progress or is begun during the RCT absence.

f. During REACTOR SHUTDOWN with IRRADIATED FUEL ELEMENTS in the reactor vessel.

1. A qualified operator capable of responding to areas outside of building TRA-670 to take accident mitigating action.

AC 5.3.2.2 Approved administrative procedures shall define (a) the specific areas where the RO and SRO must remain to be considered "at the reactor controls" and "in the reactor control room," (b) the emergency conditions under which the RCT may be called away from TRA, and (c) the specific areas where the EO must remain to be considered "in the vicinity."

AC 5.3.3 Operating Support

AC 5.3.3.1 A list of facility support personnel by name, title, and work and home telephone number shall be available to operating staff. The list shall be current (updated quarterly) and shall include as a minimum Operations management, a radiation engineer, and key mechanical, electrical, reactor safety, and maintenance specialists.

AC 5.3.3.2 The minimum supporting staff required during any operational shift shall be available. Minimum supporting staff is those personnel required to administer and effectively carry out the responsibilities of the programs listed under AC 5.7 during the shift.

5.4 TECHNICAL SAFETY REQUIREMENTS

AC 5.4.1 General

The TSR Shall:

a. Be prepared, independently reviewed, and approved in accordance with DOE Order 5480.22.

b. Define the controls to ensure the ATR remains within the safe operating envelope defined in SAR-153. The TSRs formally document the requirements in the following sections:

   1. Use and Application.
   2. SAFETY LIMITS.
   3/4. OPERATING LIMITS, which include LIMITING CONTROL SETTINGS. LIMITING CONDITIONS FOR OPERATION, Surveillance Requirements.
   5. Administrative controls.

c. Be complied with except for reasonable action taken in an emergency when this action is immediately needed to protect the health and safety of workers and the public, and when ACTION consistent with the TSR is not immediately apparent.

d. Be procedurally controlled to require that changes are:

   1. Prepared with a submittal package, including a description of the revision, justification for the change, and supporting analysis.
   2. Reviewed and approved by the contractor.
   3. Approved by the DOE prior to incorporation of the TSR change.

NOTE

Changes to the TSR Bases do not require DOE approval if they meet the conditions of Section 5.4.5.

AC 5.4.2 Compliance

The Contractor is responsible for ensuring the requirements of the ATR TSR are met. Compliance shall be demonstrated by:

a. Operating within the SLs.

b. Operating within the LCSs, LCOs, and the associated SRs during their applicability.

c. Operating within the ACTIONS of LCSs and LCOs when required.

d. Performing all SRs as required.

e. Establishing, implementing, and maintaining the required ACs.

5.4 TECHNICAL SAFETY REQUIREMENTS (continued)

AC 5.4.3 Violation of TSR

Violations of the TSR occur as a result of the following:

a. Exceeding an SL.

b. Failure to perform an ACTION within the required COMPLETION TIME, including LCO 3.0.3, following:

1. Exceeding an LCS.
2. Failure to meet an LCO.
3. Failure to successfully meet an SR.

c. Failure to perform a surveillance within the required frequency.

d. Failure to comply with an AC (as specified in Section 1.2.2).

AC 5.4.4 Response to TSR Violations

AC 5.4.4.1 Response to Safety Limit Violations.

If an SL is violated proceed as follows:

a. Place the facility in a safe condition in accordance with SL ACTIONS.

b. Notify DOE of the violation in accordance with DOE O 232.1.

c. Prepare an Occurrence Report in accordance with DOE O 232.1.

d. Perform and document a technical evaluation of the SL violation to determine if any damage has occurred, evaluate the capacity of the facility to restart, and obtain DOE approval prior to restart.

AC 5.4.4.2 Response to an LCS or LCO Violation.

If a TSR violation occurs as a result of the failure to perform an ACTION within the required COMPLETION TIME, including LCO 3.0.3, following the failure to meet an LCO or exceeding an LCS, proceed as follows:

a. Place the facility in a safe condition by entering LCO 3.0.3.

b. Notify DOE of the violation in accordance with DOE O 232.1.

c. Prepare an Occurrence Report in accordance with DOE O 232.1.

5.4 TECHNICAL SAFETY REQUIREMENTS (continued)

AC 5.4.4.3 Response to an SR Violation.

If an SR has not been successfully performed within the required frequency, proceed as follows:

a. Perform the SR as provided in SR 4.0.3.

   1. If the SR is successfully met, Exit SR 4.0.3 and continue operation in a compliant condition.

   2. If the SR is not successfully met, enter the ACTIONS of the applicable LCO.

b. Notify DOE of the violation in accordance with DOE O 232.1.

c. Prepare an Occurrence Report in accordance with DOE O 232.1.

AC 5.4.4.4 Response for failure to comply with an AC.

If an AC is has not been complied with, proceed as follows:

a. Notify DOE of the violation in accordance with DOE O 232.1.

b. Prepare an Occurrence Report in accordance with DOE O 232.1.

c. Prepare a recovery plan describing the steps leading to compliance with the AC.

d. Perform and document a technical evaluation of the AC violation to determine if any damage has occurred.

AC 5.4.5 TSR Bases Control

Changes may be made to the TSR Bases without prior DOE approval provided the changes do not involve any of the following:

a. A change to the TSR.

b. A change to SAR-153 that involves an unreviewed safety question as defined by 10 CFR 830.

c. A change to the way that operability or the TSR could be met, applied, or interpreted.

Proposed changes that meet the criteria of (a), (b), or (c) above shall be reviewed and approved by the DOE prior to implementation. Changes to the Basis that may be implemented without prior DOE approval shall be provided to the DOE for information at least annually.

5.0 ADMINISTRATIVE CONTROLS

5.5 PROCEDURES

AC 5.5.1 Procedures shall be established, implemented, maintained, and controlled that govern the following activities:

a. Administrative aspects of operation of the ATR.

b. Normal operation of the ATR.

c. Emergency operation of the ATR.

d. Surveillance requirements specified in Section 3/4 of the TSR.

e. Surveillance activities to support and maintain operation including:

   i. Reactor Startup (initial critical evaluation).
   ii. Safety instrument performance.
   iii. INSERVICE INSPECTION plan completion.

f. Procedure creation, review, approval, revision, and temporary changes.

5.6 RECORD KEEPING

AC 5.6.1 Record creation and retention practices shall be in accordance with configuration management of documents and plant modification and the quality assurance programs.

AC 5.6.2 The following documents shall be created and retained as records:

a. Records and logs of facility operation.

b. Records and logs of principal maintenance activities, inspections, repairs, and replacements of principal equipment items related to nuclear safety.

c. All reportable events and occurrences.

d. Records of surveillance activities, inspections, and calibrations required by the TSR.

e. Records of changes made to procedures supporting the TSR.

f. Records and drawing changes reflecting facility design modifications made to systems and equipment described in SAR-153.

g. Records of radiation exposure for all individuals entering radiologically controlled areas.

h. Records of gaseous and liquid radioactive material released to the environment.

i. Records of radioactive material releases.

j. Records of principal facility tests and experiments.

k. Records of unreviewed safety question (USQ) determinations performed for changes made to procedures or equipment, or USQ determinations for tests and experiments.

l. Records of fuel inventory, loading and storage.

5.7 PROGRAMS

AC 5.7.1 Configuration Management Control

A program shall be established, implemented, and maintained for the control of ATR configuration including both documents and facility structures, systems, and components, including software. The program shall have the following elements:

a. Establish policy and responsibilities for configuration management.

b. Identify and document configuration items. Configuration items shall include documents that apply to design, construction, and operation of the ATR. The configuration items should also include documents that support the safety basis such as, SAR-153, TSR, operating, maintenance, and training manuals, program implementation plans, procedures, and manuals.

c. Establish verification, validation, and acceptance criteria for configuration items.

d. Establish review and approval of configuration items.

e. Establish control of configuration changes and review and approval of changes.

AC 5.7.2 Emergency Preparedness

A program shall be established, implemented, and maintained for the control of facility emergencies and the response thereof to protect personnel, property, and the environment. The program shall have the following elements:

a. Establish a policy and responsibilities for emergency preparedness.

b. Establish specific procedures and actions to be taken to minimize injuries to personnel, damage to property, and impact on the environment.

c. Base planning and preparedness on and commensurate with the major hazards and consequences.

d. Consider operational emergencies including incidents involving hazardous material, fires, explosions, natural phenomena, and terrorist threats or acts.

e. Establish training, drills, and exercises for emergency preparedness with an annual independent audit.

5.7 PROGRAMS (continued)

AC 5.7.3 Fire Protection

A program shall be established, implemented, and maintained for the control of fire protection. The program shall have the following elements:

a. Establish a policy and responsibilities for the control of fire protection.

b. Minimize the threats to public health or welfare resulting from fire, hazards to site personnel from a fire, delays to important DOE programs as a result of a fire, and minimize the extent of property damage to a manageable level.

c. Minimize the potential for the occurrence of a fire and hazards associated with a fire.

d. Establish a defense-in-depth fire-protection philosophy.

e. Establish responsibility, accountability, and authority for fire protection compliance.

AC 5.7.4 Hazardous Materials Protection

A program shall be established, implemented, and maintained for the control of hazardous materials for protection of site personnel and the public health and welfare. The program shall have the following elements:

a. Establish the policy and the responsibilities for hazardous material protection.

b. Incorporate the as-low-as-reasonably-achievable (ALARA) philosophy for personnel exposures. Eliminate or control all potential health hazards to the ALARA level while allowing efficient conduct of operations.

c. Incorporate an industrial hygiene program and policy to provide a work place free from recognized hazards and protect personnel against recognized hazards. Control the level of hazardous materials to which personnel may, at times, be exposed.

d. Incorporate bioassay and medical monitoring.

AC 5.7.5 Inservice Inspection

A program shall be established, implemented, and maintained for the control of inservice inspection. The program shall have the following elements:

a. Establish a policy and responsibilities for the control of inservice inspection.

b. Identify and document structures, systems, and components for which periodic, systematic, and comprehensive inspection or testing will be implemented.

AC 5.7.6 Maintenance

A program shall be established, implemented, and maintained for the control of maintenance of the ATR and supporting facilities.

5.7 PROGRAMS (continued)

AC 5.7.7 Nuclear Criticality Safety

AC 5.7.7.1 A program shall be established, implemented, and maintained for the control of nuclear criticality safety. The program shall have the following elements:

a. Establish a policy and the responsibilities for the control of nuclear criticality safety.

b. Control nuclear criticality safety parameters and their bases, identification, and verification.

c. Establish access controls and physical security protection for fissile-material.

d. Establish identification, accountability, control, management, and storage of fissile-material.

e. Establish physical protection and control for onsite handling and transport of fissile-material.

AC 5.7.7.2 Fuel storage and handling shall meet the following requirements:

NOTE

Fissile material equivalents of U-235 are: 1 gram of U-235 = 1/2 gram of Pu = 1/4 gram of U-233.

NOTE

The fissile material mass of flux monitors or similar measuring devices shall be included as part of the fissile material mass of the fissile material forms in which they are installed.

a. Allowable fissile material forms in the ATR FACILITY shall be limited to:

1. ATR FUEL ELEMENTS containing ≤ 1100 g of U-235.
2. Fueled LOOP FACILITY EXPERIMENTS containing equivalent of ≤ 1500 g of U-235 and with a fuel diameter ≤ 4 in.
3. Miscellaneous fissile material specimen containing equivalent of ≤ 365 g U-235 (e.g., capsule EXPERIMENTS, flux monitors and sources).

5.7 PROGRAMS (continued)

b. Fissile material shall be stored in APPROVED FUEL STORAGE that is subject to the following limits:

1. k_{eff} shall not exceed 0.95 for the service conditions.
2. Cooling shall be adequate to remove decay heat without reaching saturation temperature in the coolant.
3. Storage shall be stable and not susceptible to tipping from credible natural phenomena or work activities.
4. Relocation of storage units shall be completed only when fissile materials have been removed from the unit. (Carriers for transporting fissile material forms and shipping containers for unirradiated fissile material forms that are APPROVED FUEL STORAGE are exempt from this limit.)
5. Storage shall be located away from areas where heavy loads are routinely handled (e.g., crane assisted activities) or specific limitations shall be established to preclude physical contact between heavy loads and materials in storage.

APPLICABILITY

Applies at all times except as specified for fissile material forms outside of APPROVED FUEL STORAGE (AC 5.7.7.2(d)). Miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) are excluded from and/or do not need to show compliance with these requirements.

c. APPROVED FUEL STORAGE for FUEL ELEMENTS in the Reactor Vessel but not in their CORE positions shall be limited to two FUEL ELEMENTS in any two locations of the in-vessel fuel storage racks.

APPLICABILITY

Storing FUEL ELEMENTS in the Reactor Vessel outside of the fuel annulus is allowed only during DEPRESSURIZED SHUTDOWN.

5.7 PROGRAMS (continued)

d. Fissile material forms outside of APPROVED FUEL STORAGE shall be limited to (limits apply to each independently):

1. Canal

i. No more than four FUEL ELEMENTS, including canal transfer tube.

ii. No more than one fueled EXPERIMENT. Miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) are excluded from this requirement.

iii. No more than 365 g U-235 equivalent in miscellaneous specimen.

iv. No more than one type (FUEL ELEMENT(S), fueled LOOP FACILITY EXPERIMENT or miscellaneous fissile material specimen) of fissile material shall be out of approved storage at any time. Miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) are excluded from this requirement.

2. Vessel

i. No more than two FUEL ELEMENTS outside the fixed locations (APPROVED FUEL STORAGE) or the CORE.

ii. No more than one fueled EXPERIMENT outside the CORE. Miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) are excluded from this requirement.

iii. No more than 365 g U-235 equivalent in miscellaneous specimen outside the CORE.

iv. No more than one type (FUEL ELEMENT(S), fueled LOOP FACILITY EXPERIMENT or miscellaneous fissile material specimen) of fissile material shall be out of approved storage at any time. Miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) are excluded from this requirement.

5.7 PROGRAMS (continued)

3. Other

i. No more than four unirradiated FUEL ELEMENTS outside the canal or the reactor vessel.

ii. No more than one fueled EXPERIMENT outside the canal or the reactor vessel. Miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) are excluded from this requirement.

iii. No more than 365 g U-235 equivalent in miscellaneous specimen outside the canal or the reactor vessel.

iv. No more than one type (FUEL ELEMENT(S), fueled LOOP FACILITY EXPERIMENT or miscellaneous fissile material specimen) of fissile material shall be out of approved storage at any time. Miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) are excluded from this requirement.

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e. In water, a minimum distance of one foot shall be maintained between any two of the individual items of the fissile material forms outside APPROVED FUEL STORAGE, except for special circumstances during loading or unloading of FUEL ELEMENTS from the fuel annulus. When tolerances or other interferences do not allow loading or unloading a single FUEL ELEMENT from the fuel annulus, a pair may be inserted or removed provided the SRO in charge of handling has completed a specific evaluation that establishes limits to preclude interaction with any other fissile material out of APPROVED STORAGE.

Miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) are excluded from minimum distance requirements.

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f. All activities requiring movement of fissile materials to be out of APPROVED FUEL STORAGE shall be completed with at least two staff members trained in handling of fissile material. In addition, the Shift Supervisor or his designated alternate shall be present to direct fuel handling when more than two FUEL ELEMENTS are outside approved storage in the canal including canal transfer tube.

5.7 PROGRAMS (continued)

Activities requiring movement of miscellaneous fissile material specimens containing in aggregate the equivalent of ≤ 15 g of U-235 (e.g., EXPERIMENTS, flux monitors and sources) shall be completed with at least one staff member trained in handling of fissile material.

g. Inspect the structural integrity of the EXPERIMENT storage racks and hooks at least every 2 years* or remove the racks and/or hooks from service.

h. Inspect the structural integrity of all normally stationary canal fuel storage racks at least once every eight years* or remove the racks from service.

i. The CORE is not subject to the requirements of TSR-186, Section 5.7.7.2.

AC 5.7.8 Quality Assurance

A program shall be established, implemented, and maintained for the control of quality assurance. The program shall have the following elements:

a. Establish a policy and the responsibilities for the control of quality assurance.

b. Establish the requirements and responsibilities for design, procurement, fabrication, installation, testing, and operation of structures, systems, and components.

c. Establish the requirements and responsibilities for review and approval, control, and storage of quality documents.

d. Establish the requirements and responsibilities for control of work processes.

AC 5.7.9 Radiation Protection

A program shall be established, implemented, and maintained for the control of radiation protection. The program shall be based on 10 CFR 835 and shall have the following elements:

a. A policy and the responsibilities for the control of radiation protection.

b. Incorporate as-low-as-reasonably-achievable (ALARA) philosophy to personnel exposure to ionizing radiation.

* A 25% extension may be applied to this inspection interval consistent with SR 4.0.2.

5.7 PROGRAMS (continued)

c. Radiation exposure limits and controls including the use of personal protective clothing and equipment.

d. Radiological monitoring and contamination control.

e. Requirements for record keeping and training.

AC 5.7.10 Unreviewed Safety Question

A program shall be established, implemented, and maintained for the control of unreviewed safety questions, based on the requirements in 10 CFR 830 Subpart B.

5.8 REVIEWS AND AUDITS

AC 5.8.1 A review and audit system shall be established to ensure that the reactor is operated in accordance with approved procedures, that day-to-day activities are conducted in a safe manner, and that activities and programs affecting nuclear safety are conducted in a manner consistent with the safety basis and the TSR.

AC 5.8.2 CONTRACTOR management shall specify the functions, organization, responsibilities, qualifications, and reporting requirements for each functional element that contributes to the required reviews and audits.

AC 5.8.3 A CONTRACTOR designated, independent review committee shall review all matters with nuclear safety implications. The membership, responsibilities, and procedures of the review committee shall be formally documented and approved by CONTRACTOR management.

AC 5.8.4 Facility staff reviews shall ensure that the day-to-day activities are conducted in a safe manner. The facility staff reviews shall include:

a. Unreviewed safety question determinations

b. Proposed tests and experiments

c. Procedures

d. Facility changes and modifications

e. TSR changes

f. Facility operations

g. Facility maintenance and testing including evaluation of system performance for indication of degradation.

h. DOE and industry issues of safety significance

i. Assessment of the effectiveness of the reviews conducted by facility staff

j. Any other safety significant issues or items.

AC 5.8.5 Reviews of facility activities and programs affecting nuclear safety shall be conducted by personnel independent of facility staff. These reviews shall include:

a. Unreviewed safety question determinations

b. Proposed changes to the TSR

c. Violations of codes, Orders, and procedures that have safety and health significance

5.8 REVIEWS AND AUDITS (continued)

d. Occurrence Reports

e. Staff performance

f. Unanticipated deficiencies of structures, systems, or components that could affect nuclear safety

g. Significant, unplanned radiological or toxic material releases

h. Significant operational abnormalities

i. Assessment of the effectiveness of the reviews conducted by facility staff.

AC 5.8.6 INTENTIONALLY LEFT BLANK

AC 5.8.7 Audits shall be conducted by a safety organization independent of facility staff that includes the following:

a. Conformance with the TSR

b. Training and qualification of facility staff

c. Program implementation

d. Deficiency corrective action

e. Quality program adherence

f. Other activities of safety significance.

5.9 STAFF QUALIFICATION and TRAINING

AC 5.9.1 A program shall be established, implemented, and maintained for qualification and training of Reactor Operators (ROs), Radiological Control Technicians (RCTs), Senior Reactor Operators (SROs), Experiment Operators (EOs), Senior Reactor Auxiliary Operators (SRAOs), and Shift Supervisors (SSs).

FOIA Exemption 3 UCNI, Exemption 7f (TSR-186 Appendix A)

ADVANCED TEST REACTOR

UPGRADED FINAL SAFETY ANALYSIS REPORT

EXECUTIVE SUMMARY

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ACRONYMS/ABBREVIATIONS

ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ATR	Advanced Test Reactor
CAMs	continuous air monitors
CFR	Code of Federal Regulations
DOE	U.S. Department of Energy
DBR	ATR PPS and Technical Specifications Design Basis Report
EDE	effective dose equivalent
EFIS	emergency firewater injection system
ES	executive summary
ESF	engineered safety feature
HFE	human factors engineering
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LMITCO	Lockheed Martin Idaho Technologies Company, Inc.
LOCA	loss of coolant accident
NRC	U.S. Nuclear Regulatory Commission
PCS	primary coolant system
PGA	peak ground acceleration
PPS	plant protection system
PRA	probabilistic risk assessment
RMSS	radiation monitoring and seal system
RSS	Reactor Shutdown System
SAR	Safety Analysis Report
SSC	structures, systems, and components
SSE	safe shutdown earthquake
TEDE	total effective dose equivalent
TRA	Test Reactor Area
TSR	Technical Safety Requirements
UFSAR	Upgraded Final Safety Analysis Report
USQ	unreviewed safety question
VVS	Vessel Vent System

ES-1. Safety Analysis Report Summary

The Advanced Test Reactor (ATR) is a Department of Energy (DOE)-owned test reactor whose principal function is to provide a high neutron flux for testing reactor fuels and other materials. The reactor has also provided other irradiation services such as radioisotope production. The reactor has a design thermal power of 250 MW with a nominal operating pressure of 360 psig at the top of the core and a nominal maximum reactor outlet temperature of 170°F. The ATR and its support facilities are located at the Test Reactor Area (TRA) of the Idaho National Engineering and Environmental Laboratory (INEEL).

This Upgraded Final Safety Analysis Report (UFSAR) for the ATR was prepared following the requirements of DOE Order 5480.23, Nuclear Safety Analysis (DOE 1994), and using U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (NRC 1978). The current requirements for safety analyses for DOE reactors are contained in 10 CFR 830 Subpart B, Safety Basis Requirements (CFR 2002a). 10 CFR 830 specifies following the NRC Regulatory Guide 1.70 as the safe harbor standard, thus this UFSAR follows the standard for Documented Safety Analysis specified in the current regulations and contains some additional material called for by DOE Order 5480.23.

ES-1.1 Safety Analysis Report (SAR) Commitments

DOE Order 5480.23 (DOE 1994) requires contractors be held responsible for adhering to assumptions and commitments set forth in the safety analysis. The interim guidance attached to DOE Order 5480.23 states that contractors are required to adhere to commitments made in the SAR and to conduct operations in such a way that the assumptions made in the SAR are valid. The implementation Guide For Use in Developing Documented Safety Analyses to Meet Subpart B 10 CFR 830 (DOE 2001) states that development of a documented safety analysis is the process whereby facility hazards are identified, controls to prevent or mitigate potential accidents involving those hazards are proposed, and commitments are made for design, construction, operation, and disposition so as to ensure adequate safety at DOE Nuclear Facilities. This section describes the Contractor's position on SAR commitments in the ATR UFSAR.

The Contractor (Baranick 1995) has defined a commitment as any concept in a SAR and its references that (a) establishes guidelines for operations, reliability, availability, and maintainability of any system or subsystem that is required for operation of a facility within the risk accepted by DOE; (b) limits the risk accepted by DOE; or (c) establishes a parametric value that if changed in any arbitrary manner could increase the risk relative to the safety or protection of worker, the public, or the environment.

There are three types of commitments in the UFSAR; design commitments, operational commitments, and commitments for institutional safety programs.

ES-1.1.1 Design Commitments

The design commitments include the assumed physical configuration of the important-to-safety and safety-related structures, systems, and components (SSC); and the facility-specific general design criteria; and the stated requirements for modifications for important-to-safety and safety-related SSC. There are design commitments in Chapters 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 15, 18, and 20. While design commitments are not specifically identified in the UFSAR, design commitments are preserved through the facility configuration control and the facility modification processes. Design commitments are identified as requirements in the facility modification process or the modification supports a change in the design commitment.

Inspection discrepancies related to design commitments can be evaluated and resolved via the backward looking Unreviewed Safety Question (USQ) process. The application of the USQ process to screen issues related to the plant configuration is supported by the DOE Young memorandum (Young 1992). The Young memo defines a condition where the facility physical configuration and the configuration assumed in the safety analysis do not agree as a discrepant as-found state. There are several discussions of discrepant as-found states in the Young memo. If the discrepancy is between the facility and the Technical Safety Requirements (TSRs), DOE Order 5480.21 (DOE 1991b) and 10 CFR 830 (2002a) has clear requirements for reporting and performing a USQ determination. If the discrepancy is between the facility and the safety analysis the memo provides guidance. The first action is to determine what (the physical configuration or the safety analysis) is correct. If the facility configuration is correct and the Contractor determines there is no safety significance, the USQ process is used to determine if the Contractor has the authority to approve the configuration. This is the backward looking USQ determination. The backward looking USQ resolves the discrepancy and establishes justification for the configuration.

The Young memo further elaborates that the backward looking USQ process was needed by nuclear plant operating licensees and the U.S. Nuclear Regulatory Commission (NRC) when the commercial nuclear industry embarked on the design basis reconstruction programs. Those programs revealed information that questioned many aspects of the current configuration, resulting in the documentation of thousands of apparent discrepancies. The discovery of such discrepancies could have resulted in extensive justifications for continued operation or forced plant shutdowns and expensive restart efforts and performance improvement plans. Both the licensees and the NRC needed to screen these discrepancies for safety significance to avoid such burdens. The backward looking USQ process proved to be a successful tool in that effort.

ES-1.1.2 Operational Commitments

Operational commitments are requirements on important-to-safety and safety-related SSCs operability; system and parameter surveillance activities; and specified parametric values that are under the control of the operating staff. Examples of operational commitments are primary coolant system (PCS) chemistry limits, fuel element qualifications, reflector stress conditions, and non-TSR instrumentation requirements. There are operational commitments in Chapters 3, 4, 7, 8, 9, 10, 13, 15, 16, and 20. Operational commitments are identified as SAR commitments in the procedures that implement the requirement.

Inspection issues related to operational commitments will be evaluated. The above definition will be used to establish if the issue identifies an operational commitment. Once identified, commitments will be verified to be implemented and identified in procedures. The inspection issue would not constitute violation of the SAR commitment unless the issue identifies a commitment that is not implemented.

ES-1.1.3 Commitments For Institutional Safety Programs

The UFSAR includes comprehensive descriptions of institutional safety programs. Examples of institutional safety programs are fire protection, hazardous material protection, radiation protection, quality assurance, and conduct of operations. There are institutional safety program commitments in Chapters 1, 9, 10, 11, 12, 13, 14, 15, 17, 18, 19, 20, and 21. Programmatic commitments encompass a large number of details that are more appropriately covered in specific program documents external to the UFSAR. The Contractor develops these programs in Contractor requirements documents and procedures that are equal to the UFSAR in the Contractor document hierarchy. Many of the requirements for institutional safety programs are contractual requirements. Additionally, many of the programs are administered by Contractor organizations that are outside of the organizational structure in the UFSAR that is directly responsible for operation of the ATR. As a result, the SAR material related to the institutional safety programs is descriptive only. The SAR commitment is to maintain an effective implementation of the programs described, not the details of the program or its governing documents. Inspection discrepancies in a program would not constitute violation of the SAR commitment unless the discrepancies were so gross as to render premises of the summary invalid.

ES-1.2 Hazards Analysis and Classification

The hazards classification for the ATR facility, as discussed in this section, consists of identifying the potential hazards and evaluating the consequences associated with the hazards. This process of hazards classification follows the guidelines established by the U.S. Department of Energy (DOE 1997).

ES-1.2.1 Hazard Analysis

The hazards analysis performed for the ATR included identifying the maximum potential hazardous material inventory available at the facility. This included identifying the three classes of hazardous material present: radioactive material, explosive material, and hazardous chemicals.

Four potential sources of radionuclides were identified for the ATR. The sources include the reactor core, the radioactive materials (irradiated fuel and other hardware) stored in the canal, isotope production targets, and experiments containing fueled and non-fueled components. The ATR is a Category A reactor with an operating power level of up to 250 MW_t and, as such, has a radioactive material inventory with the potential for significant offsite consequences.

No explosives have been separately stored at the ATR. The only explosive or pyrophoric hazards identified for the ATR facility are associated with the severe accident scenarios in which hydrogen gas is released. The evaluation of these hazards is covered by the evaluation of the radiological hazards.

The inventories of potentially hazardous chemicals have been identified, and potential accident scenarios have been evaluated. The results of the evaluation indicate that significant quantities of sulfuric acid are stored in ATR buildings. Small amounts of other hazardous chemicals are stored throughout the ATR. Personnel exposure, handling, and storage of these materials conform to the Contractor's *Safety and Health* manual. None of the postulated chemicals released in an accident would require additional guidance beyond those already addressed by existing institutional safety programs.

ES-1.2.2 Hazard Classification

The classification of operational hazards at the ATR is shown in the Table ES-1, which identifies the hazards considered and the resulting hazard type or classification. Many of the hazards evaluated were classified as standard industrial hazards. The radiological hazard is such that there could be significant offsite consequences in the event of an unmitigated release of radioactive material from the facility.

The ATR is a Category A reactor with an operating power level of up to 250 MW_t with potential for significant offsite radiological consequences. The ATR is classified as a Hazard Category 1 nuclear facility in accordance with Department of Energy standards for hazard classifications of nuclear facilities (DOE 1997).

ES-1.3 Design Basis Accidents

The design basis accidents that provide the greatest challenge to the engineered safety features (ESFs) are listed in Table ES-2. For the design basis accidents, the ESFs prevent or mitigate damage to the reactor core and PCS.

ES-1.4 Major Accident Sequences

The design basis events considered in safety analyses of the ATR are similar to those considered for the commercial reactor industry. To the extent possible, the accident sequences are organized in a format consistent with a commercial safety analysis report. However, some differences exist to accommodate the unique features of the ATR. For the ATR UFSAR, the event categories and major accident sequences are listed in Table ES-3. The event categories correspond to the Sections of Chapter 15 (Accident Analyses).

The first seven event categories correspond to those identified in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70. The limiting events in each of these seven categories have been identified and are included in the major accident sequences listed in Table ES-3. The frequency of occurrence of each of the event sequences has been classified into categories as specified in the standard RDT C16-1T (AEC 1969). The categories include Condition 2 (anticipated), Condition 3 (unlikely), and Condition 4 (extremely unlikely). These correspond to the 1973 ANSI/ANS Standard 18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" (ANS 1973), which includes Condition 2 - Incidents of Moderate Frequency, Condition 3 - Infrequent Incidents, and Condition 4 - Limiting Faults.

The development of the accident analyses in the UFSAR built on the analyses in the *ATR Plant Protection System and Technical Specifications Design Basis Report* (DBR) (EG&G Idaho 1994). Where appropriate, the limiting events in the DBR from each of the event classification categories (e.g., decrease in reactor coolant flow, reactivity and power distribution anomalies, decrease in reactor coolant inventory, etc.) were reanalyzed. In addition, events were included if the analysis assumptions needed to be addressed for applicability. Event classification categories 8 through 15 include events that are contained in the DBR that arise due to the unique mission and design of the ATR. For the most part, these events do not readily fit into the categories established for commercial power reactors. The limiting events in categories 8 through 11 have been identified and are included in the major accident sequences listed in Table ES-3. Event Category 8 contains the evaluations made of fuel storage and cask handling events. Category 9 contains the events identified for depressurized operation. The ATR is operated in this mode to run physics tests after core internal changeouts and core configuration changes. This mode of operation occurs infrequently and is limited to low power. Category 10 contains evaluations of fuel failures that can be postulated to occur without the occurrence of a reactor or PCS accident. Category 11 contains events initiated by failures of supporting systems. Category 12 contains a discussion of severe accidents analyzed for the ATR PRA. These are beyond design basis events and include the anticipated transient without scram (ATWS) and the hypothetical large break LOCA. The hypothetical large break LOCA is analyzed to determine the radiological releases from the ATR confinement. Categories 13 and 14 contain the PCS high pressure and flow events. The limiting events are included in the major accident sequences listed in Table ES-3. Category 15 discusses the cracking expected in a reflector that is nearing the end of its useful life. Evaluations performed for Category 15 establish core power limits and fuel loading limits for operation with an aged reflector. Category 16 discusses external events. Analyses of these events are based primarily on the evaluations in the PRA.

ES-1.5 Safety Features and Mitigative Measures

The following safety features and mitigative measures are available at the ATR to mitigate postulated accident scenarios:

LOCA Primary Coolant Pump (PCP) Shutoff System—The LOCA Primary Coolant Pump Shutoff System is an ESF function provided by the inputs from the Reactor Shutdown System (RSS) Reactor Inlet Pressure and Outlet Pressure Subsystems. The purpose of the LOCA Primary Coolant Pump Shutoff System is to shutoff all operating primary coolant pumps in the event of a loss-of-coolant-accident (LOCA). Shutoff of the primary coolant pumps needs to occur prior to air being drawn from the surge tank into the Primary Cooling System (PCS). Air in the PCS could cause a degradation of emergency pump flow.

Radiation Monitoring and Seal System—The RMSS prevents the uncontrolled release of airborne radioactive contamination from the ATR confinement. The subsystem is composed of stack exhaust radiation monitors, 2 of 3 trip logic, two actuation channels, and quick-closing dampers with their associated solenoids and actuators. The system actuates the building isolation system when detectors in the stack breech sense high radiation.

Emergency Firewater Injection System—The Emergency Firewater Injection System (EFIS) provides emergency reactor core cooling and flooding of the reactor vessel in the event of a major break in the Primary Coolant System piping or other events resulting in loss of primary coolant. This ESF is designed to activate at a predetermined low pressure, or when the water level lowers to a predetermined height above the top of the core. The EFIS comprises three systems: the Upper Vessel EFIS Piping System, the Bottom Head EFIS Piping System, and the Actuation System.

Vessel Vent System—The Vessel Vent System (VVS) provides the capability to depressurize the ATR vessel. It is needed to mitigate high-pressure events such as the long-term complete loss-of-flow accident (CLOFA). In the CLOFA, core decay heat produces steam and causes a pressure increase in the vessel exceeding the supply pressure of the EFIS. Venting the vessel under such conditions is required to allow the EFIS coolant to flow into the ATR vessel and keep the core covered. The VVS is manually actuated by operator action and cannot be automatically actuated.

Primary Pump Shutoff System—The Primary Pump Shutoff System for ATR is part of the Plant Protection System (PPS), using the same transducers and comparators as RSS Differential Pressure. The purpose of the Primary Pump Shutoff System is to shut down one or more primary coolant pumps before a high differential pressure across the reactor core can approach damage thresholds for the core components.

Pressurizing and Gland Seal Pump Shutoff System—The Pressurizing and Gland Seal Pump Shutoff System for ATR is part of the PPS, using the same transducers and comparators as RSS Inlet Pressure. The purpose of the Pressurizing and Gland Seal Pump Shutoff System is to protect the PCS from exceeding the system design pressure. Upon receiving an overpressure trip signal, the Pressurizing and Gland Seal Pump Shutoff System stops the primary pressurizing pumps and the gland seal pumps; a reactor scram also occurs on high pressure.

Primary Coolant System Overpressure Relief System—The Overpressure Relief System consists of two spring-loaded safety relief valves mounted on the reactor vessel 8-in. lower drain line.

Vessel Level Alarm System—The Vessel Level Alarm System actuation is an ESF function provided by the inputs from the Reactor Shutdown System Reactor Vessel Water Level Subsystem or by the reactor vessel water level channels of the EFIS. The purpose of the Vessel Level Alarm System actuation is to alert the operators of low vessel liquid level for potential actuation of the EFIS. The vessel level alarm is used to alert against low-probability events that result in a loss of PCS inventory during depressurized and outage operations when irradiated fuel elements are in the reactor vessel.

Emergency Coolant Pumps—There are two emergency coolant pumps, either of which provides the necessary forced flow decay heat removal capability to ensure adequate fuel cooling when the primary coolant pumps are not functioning. Each emergency coolant pump (M-10 and-11 in Figures 5.1-1, -10), has a capacity of 4,700 gpm at 22 ft Total Dynamic Head (TDH). Emergency coolant pump M-10 is ac motor-driven from the 480 V diesel-commercial bus. The second pump, M-11, is dc motor-driven from the utility dc power control center.

Canal Firewater Emergency Makeup System—The canal firewater emergency makeup system prevents irradiated fuel elements from being uncovered following a canal draining event. The system provides automatically actuated raw water makeup to the irradiated fuel storage section of the canal.

ES-2. Safety Analysis Report Applicability

The applicability of this safety analysis report is to evaluate the safety of the ATR facility for normal, abnormal, and accident conditions. The principal function of the ATR facility is to provide a high neutron flux for testing reactor fuels and other materials. The reactor has a design thermal power of 250 MW with a nominal operating pressure of 360 psig at the top of the core and a nominal maximum reactor outlet temperature of 170°F. The reactor and PCS, along with the control room and much of the auxiliary and experimental support equipment, are located in TRA Building 670. Buildings and structures in other parts of TRA provide additional support including commercial electric power, raw and demineralized water, and fire protection. A summary of the specific facilities and their locations is provided. The following buildings* and structures that are necessary for safe operation of ATR and define the boundaries of the ATR Facility were considered in this UFSAR:

- Buildings
  - TRA-670, ATR Reactor Building (excluding ATR Critical Facility [ATRC])
  - TRA-671, ATR Cooling Tower Pumphouse
  - TRA-609, Compressor Building, Switchgear and Compressor Room
  - TRA-674, Diesel Building
  - TRA-608, Demineralizer Building
  - TRA-619, Raw Water Pumphouse
  - TRA-633, Firewater Pumphouse
  - TRA-634, ATR Storage Facility (during experiment storage in casks)
  - TRA-688, Firewater Pumphouse.
- Structures
  - TRA-770, ATR Vent Stack
  - TRA-771, ATR Cooling Towers
  - TRA-774, Transformer Yard
  - TRA-776, ATR 1,500-gal Diesel Fuel Tank
  - TRA-763, Compressed Air Storage Tanks
  - TRA-718, Overhead Raw Water Storage Tank

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* Even though building TRA-689 is tied to TRA-670 by pipe, building TRA-689 is not included in this list, since the building is not in use at this time. The piping between TRA-689 and TRA-670 is isolated by both MOV-32-1 and MOV-1-909. Since the piping connections to TRA-689 are downstream of either a <3-in. orifice or valve, any break is bounded by the analysis in Chapter 15.6.

- TRA-719, Ground-Level Storage Tanks
- TRA-708A, Demineralized Water Storage Tank
- TRA-754, Demineralized Water Storage Tank
- TRA-781, Firewater Storage Tank.

The analysis results are documented in Chapter 15 (Accident Analyses) of this report, which includes references to the ATR PRA (Eide, Khericha, and Thatcher 1991; Thatcher et al. 1994) and to specific engineering design files.

ES-3. Risk Assessment

ES-3.1 Normal Operations

The risks associated with normal plant operations have been assessed from three specific areas: radiation protection, human factors engineering, and hazardous material protection. A summary of the results of this assessment is presented in the following subsections.

ES-3.1.1 Radiation Protection

The personnel radiological risk due to normal operation is minimized by the features and programs associated with the facility such as shielding, ventilation, and the radiological controls program. Each of these facility features or programs associated with radiation protection is discussed below.

ES-3.1.1.1 Shielding—For normal operation, the shielding has been designed to limit the radiation dose rate for operating personnel to 0.25 mrem/hr in areas that require full-time occupancy. For areas with higher, or potentially higher dose rates, access time is restricted to limit doses to levels that are commensurate with the work to be accomplished that are within "as low as reasonably achievable" (ALARA) limits that have been established. Administrative controls and an aggressive ALARA program have succeeded in maintaining personnel doses in an acceptable downward trend. Whenever work is to be performed on equipment that exceeds the guideline dose rates, special localized shielding and appropriate administrative controls are provided to minimize worker exposure. Thus, the operating personnel risk from radiation exposure is limited consistent with design limits considered in the facility design.

ES-3.1.1.2 Ventilation—The ATR ventilation systems are designed to provide airflow from areas of low contamination to areas of potentially high contamination. The air supply and exhaust fan controls are designed to operate in a manner that would prevent the air flow from reversing due to interrupted operation of ventilation equipment. Flow reversal could cause a spread of radioactive contamination to areas normally occupied by workers. In addition, the dampers for the ventilation system are used to minimize leakage from the reactor confinement during reactor accident scenarios. Thus, the ventilation system minimizes the spread of contamination and reduces the potential personnel radiological risk and consequences during normal and off normal operation of the ATR.

ES-3.1.1.3 Radiological Control Program—The personnel radiation exposure controls program embodies the ALARA philosophy, and meets the requirements of 10 CFR 835 (CFR 1998). This program, implemented through Contractor procedures, is aggressively executed to keep radiation exposures to the minimum practical, consistent with job and program needs and associated costs.

External dosimetry provides for monitoring personnel radiation exposures during normal operation and accidents. This includes the use of personal dosimeters and external radiation monitors. The internal dosimetry requires personnel with jobs that have the potential for internal exposure to participate in the bioassay program to determine the accumulation doses of internal exposure.

Other instrumentation is provided to monitor the area airborne activity; Continuous Air Monitors (CAMs) determine if the airborne radioactive concentration is within prescribed limits, and monitoring the stack effluent determines the amount of radioactivity released to the environment.

The above-described facility features and programs minimize radiation exposure risk and consequences to operating personnel.

ES-3.1.2 Human Factors Engineering

Human Factors Engineering (HFE) has been an explicit design and operational consideration at ATR dating back to the beginning of the 1980s. As described in Chapter 18 (Human Factors Engineering), current instrumentation, workstation layouts, communications, and other performance aids used by ATR personnel during normal operations and maintenance, consider person-machine requirements established in DOE-referenced HFE standards and guidance documents. HFE criteria established in these standards and guidelines are also considered for the design of the general working environment involving such factors as heat, light, noise, and protective clothing as they may impact performance, especially under abnormal operating conditions. Finally, it is recognized in Chapter 18 that person-machine interactions can impact the likelihood of an operator committing an error and, in turn, impacting his own safety and that of the public. Therefore, Chapter 18 uses the risk assessment reported in Chapter 15 (Accident Analyses), along with operating experience, as the technical bases for assessing the degree to which HFE may be a safety issue at ATR. The initiator portion of the ATR risk assessment, reported in Chapter 15, focuses on human tasks during normal operations involving reactor, process, and experimental loop control, with potentially the most significant consequences. In each case, HFE standards referenced by DOE are used to determine the quality of the person-machine interface. The results of these determinations are incorporated into the risk assessment to estimate human error likelihood. The ATR risk assessment results, reported in Chapter 15, indicate that the impact of HFE on human actions is not a significant contributor to core damage under the current ATR operating configuration. Further, the ATR risk assessment results show that while HFE may be a factor in the operation of the ATR, HFE has been adequately addressed for the mitigation of design-basis accidents requiring human intervention. Nevertheless, in cases where human error, due to HFE, is believed to be a possible contributor to initiation of an abnormal or accident condition, HFE deviations from the DOE-referenced standards are being used as a technical basis for person-machine upgrades.

ES-3.1.3 Hazardous Material Protection

The Contractor maintains a hazardous material protection program that complies with Occupational Safety and Health Administration (OSHA) requirements, DOE orders, and Contractor's procedures and standard practices. Hazardous materials historically have been evaluated assuming a de minimus level below which little harm is associated with exposure. The concept of As Low As Reasonably Achievable for these materials is applied qualitatively by maintaining individual exposures below industrial workplace permissible exposure limits.

A monitoring program is maintained to determine employee health hazards through bioassay and medical monitoring. The environment in which the employee must work is monitored by surveys and by permanent and portable monitoring instrumentation.

ES-3.2 Accident Analysis

ES-3.2.1 Design Basis Accidents

Not all the events analyzed in the DBR (EG&G Idaho 1994) were reanalyzed for the UFSAR. The DBR was used to identify the limiting event(s) in each accident category (e.g., loss of heat sink, loss of flow, reactivity/power anomalies, etc.) and, with results from the Reactivity Insertion Accident (RIA) and PRA programs, to ascertain the proper classification for event probability. Reanalyses and DBR analyses provide a bounding safety basis for plant operation. These analyses provide the bases for ATR safety systems and Technical Safety Requirements such that the required margins to protection criteria can be maintained, thus preventing severe fuel damage and the release of significant fission products from ATR. Fission products are contained by the three traditional defense-in-depth barriers: fuel cladding, the primary coolant boundary, and the confinement.

A more conservative reactor protection limit for unlikely events was used in the UFSAR compared to what had been used in the DBR. The reactor protection limit in the DBR was 2σ to cladding liquids, which may not preclude loss of coolable core geometry. During their visit to the ATR facility, the National Research Council (report published by the National Academy Press, Washington, DC. 1988: "Safety Issues at DOE Test and Research Reactors") suggested that ATR damage criteria be reviewed to ensure that a coolable core geometry be maintained. This suggestion was incorporated as the revised damage limit for unlikely events; i.e., to maintain the fuel plate in a coolable geometry. To ensure cooling of the entire plate, the limit for unlikely events involving fuel elements in the reactor core was modified to preclude fuel plate buckling which could result in coolant channel closure.

The results from the PRA were used to review the probability of occurrence of events analyzed in the DBR and the spectrum of accidents for the UFSAR. Following the review and development of additional bases, the classification of several events was changed in the UFSAR. The rapid closure of the butterfly flow control valve and the complete loss-of-flow events were changed from unlikely events to extremely unlikely events. Only a few previously unanalyzed design basis events involving the reactor core were identified, and they were determined to be enveloped by existing analyses. One event, the complete loss-of-heat sink (CLOHS) was determined, based on the PRA analysis, to need reanalysis for long-term, post-scram consequences.

Numerous events involving handling casks and other heavy loads have been identified. The analyses of these events are discussed in Chapter 15 (Accident Analyses).

Evaluation of Condition 2, Condition 3, and Condition 4 events in Chapter 15 showed no events where direct fuel damage occurred concurrent with a breach of the PCS. The offsite and onsite radiological consequences were reanalyzed only for the hypothetical beyond design basis loss-of-coolant accident resulting in 100% core damage. Radiological consequences from this event bound all design basis events.

ES-3.2.2 Probabilistic Risk Assessment (PRA)

A comprehensive full scope PRA was performed for the ATR for severe fuel damage event sequences (Level 1 PRA) and the risk to public and INEEL populations (Level 3 PRA) (Eide, Khericha, and Thatcher 1991; Thatcher et al. 1994). The ATR PRA is a detailed assessment of accident sequences which could result in significant ATR fuel damage and release of fission products to the confinement and to the environment.

The PRA was developed in an analysis effort separate from the UFSAR. The PRA for a reactor takes the safety analysis one step further than the traditional UFSAR analysis. The UFSAR analysis evaluates maximum design conditions to define the safe operating envelope of the reactor and to specify PPS requirements. The PRA analysis starts with the plant in a nominal analyzed operating condition and then looks at the consequences of internal and external initiating events, considering single and multiple equipment failures and operator errors. The PRA evaluates the consequences of major fuel damage events that are not considered in the UFSAR analysis either because the probability of the initiating event is too low or because multiple system failures or operator errors are involved.

While the PRA and the UFSAR analyses each considered severe accidents that resulted in complete core meltdown, the consequences are calculated and reported differently. The PRA consequence analyses were calculated using distributions of site-specific weather patterns applicable to the INEEL. The weather distribution accounted for variation in wind direction, wind velocities, and dispersion conditions up to at least the 95th percentile. The consequences were reported in terms of either early or latent fatalities. Latent fatalities were cancer related. In the UFSAR, consequences were determined using meteorological conditions described in Section 15.12.10, and were reported in terms of radiation exposure.

The ATR PRA includes event sequences during power operation, shutdown operation, heavy load drop events, and storage canal draining events for system failures, flooding and fires, and external hazards. The ATR PRA considered the contribution of human error and performed detailed human reliability analyses for risk-significant human errors. Significant improvements to emergency operating procedures and abnormal operating procedures, as well as a definition of restrictions on certain shutdown and heavy load handling operations, resulted from PRA evaluations. The ATR PRA does not include risk assessments for radionuclide sources other than the ATR fuel. It also does not include risk assessment for minor fuel damage events such as fuel channel blockage or damage during fuel handling.

The results of the ATR PRA are summarized in Table ES-4. The results assume the completion of risk-reduction or safety upgrades, including operating procedures and practices, which were identified and committed to during the performance of the PRA. These upgrades have been completed. A detailed summary of the ATR PRA and its results, conclusions, and insights is in the Updated ATR PRA summary report (Atkinson et al. 1994).

The probability for a severe fuel damage event and fission product release is dominated by the probability for seismic events (earthquakes) of an intensity greater than the Safe Shutdown Earthquake (SSE). The ATR SSE is defined as an earthquake with a peak ground acceleration (PGA) of 0.24g, which would have a mean frequency of about 1×10^{-4}/yr (8×10^{-5}/yr for the INEEL seismic hazard curves used for the ATR PRA). The top seismic fuel damage sequence group, with a mean frequency of 1.3×10^{-5}/yr, would correspond to earthquakes with a PGA > 0.5g. The individual ATR internal system failure fuel damage event sequences have a mean frequency $< 1 \times 10^{-6}$/yr and the sum of the fuel damage frequencies for all internal sequences is 7.7×10^{-6}/yr.

The ATR PRA risk results (risk = event sequence consequences multiplied by the event sequence frequency) are based on scoping accident analyses and on severe accident progression and confinement failure event tree analyses using conservative severe accident phenomenological event probabilities for ATR fuel damage accident sequence groups. The results of these analyses defined fission product release or source term groups, which were then analyzed for doses and health consequences for the general TRA population, the population at the nearest co-located INEEL facility, the Idaho Nuclear Technology and Engineering Center (INTEC), and for the offsite public out to 60 miles from the ATR. The consequence analyses for the public included ingestion doses from consumption of agricultural products and water but assumed no actions to evacuate any of the public. However, land and milk interdiction was considered. The TRA and INTEC analyses included evacuation but assumed no sheltering. The population out to 60 miles from ATR includes the major population centers near the INEEL, Idaho Falls and Pocatello. Risk analyses were also performed out to 30 miles from the ATR to estimate the risk to the public within 10 miles of the INEEL for comparison to the DOE risk-based safety goals (DOE 1991a).

The ATR PRA estimated a near-zero, insignificant risk for a prompt or early fatality for each population group. This analysis does not include the risk to operators within the ATR, but only to the co-located population group at TRA outside of the ATR. The DOE risk-based safety goal (DOE 1991a) for a latent cancer fatality due to radiation exposure of individuals within 10 miles of the DOE site boundary is defined as 0.1% of the normal population risk for a cancer fatality. This goal has been defined as about 2×10^{-6}/yr per individual based on United States cancer death statistics (Kim and Bradley 1993). The ATR risk results for the public within 10 miles of the INEEL are a factor of 100 below the DOE goal, which is a factor of 1,000 below the normal total cancer risk. Even at TRA, the risk is a factor of 25 below the DOE goal. Other uncertainties are implicitly included by conservative accident analysis modeling and assumptions.

ES-3.2.3 Accident Analysis Conclusions

All the design basis events analyzed met the damage criteria established for their event classification. No Condition 2 events resulted in predicted rupture of the fuel element cladding as a consequence of the initiating fault. The bounding event used for radiological consequences was the hypothetical large-break LOCA, which is classified as a severe accident. This accident results in offsite doses within 10 CFR 100 limits at the low population zone of 185 rem thyroid and 13.2 rem total effective dose equivalent (TEDE)[a]. For workers evacuating from TRA, the potential doses are 75.7 rem thyroid and 5.3 rem TEDE.

The ATR PRA identified the total probabilistic risk to both onsite and offsite public population groups to be significantly below the DOE risk-based safety guidelines by a factor of 100 for the public risk and a factor of 25 for the TRA population for all severe fuel damage accident sequences.

A summary of Chapter 15 events that resulted in substantial fission product release from the fuel is shown in Table ES-5. The table presents the predicted consequences of the event.

[a] The terms EDE, TEDE, or whole body dose are used interchangeably in the UFSAR.

Table ES-1. Hazard classification for the Advanced Test Reactor.

TRA Buildings	670	671	609	674	608	619	633	634	688	689	770	771	774	776	763	718	719	708A	754	781
Operational Hazard																				
Nuclear Hazard	X																			
Radiological	X	X						X		X	X	X								
Corrosive	X	X	X	X	X		X		X											
Explosive or pyrophoric	X		X	X		X	X		X											
Thermal and Thermal Radiation	X			X	X	X	X		X					X						
Flammable gas/materials	X		X	X	X	X	X	X	X					X						
Toxic/pathogenic	X	X	X					X												
Electrical	X	X	X	X	X	X	X	X	X	X		X	X				X	X	X	
Ionizing and nonionizing electromagnetic radiation	X							X												
Asphyxiation or drowning	X	X	X	X	X	X	X	X		X		X	X	X	X	X	X	X	X	X
Mass/gravity/height	X										X	X				X				X
Pressure-volume	X	X	X	X	X	X				X	X				X					
Kinetic-rotational and kinetic-linear	X	X	X	X	X	X	X	X	X			X								
High noise levels	X	X	X	X			X		X											
Lifting heavy objects	X		X			X		X												
Chemical Storage	X	X	X		X		X		X											
Slip, Trips, and Falls	X	X	X	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X
Batteries Lead/Acid	X		X	X		X	X		X											
Carcinogenics	X		X		X	X	X	X		X							X	X	X	

a. Information can be obtained from EDF TRA-1554 (Dinneen 2001) and in Chapter 20, Hazard Analysis and Classification.

Table ES-2. Engineered safety feature design basis accidents.

Accident Sequence	Engineered Safety System
3-in. diameter opening in the PCS due to opening a drain valve, relief valve, or vent valve	EFIS and LOCA Primary Coolant Pump Shutoff System
Experiment loop piping failure with experiment fuel melt	RMSS
Long-term complete loss of flow or complete loss of heat sink	VVS and EFIS
Opening of flow control Butterfly valve to full open	Primary Pump Shutoff System
Loss of PCS pressure control (Loss of instrument air)	Pressurizing and Gland Seal Pump Shutoff System
Loss of pressure control of PCS and failure of the Pressurizing Pumps and Gland Seal Pumps Shutoff System	Primary Coolant Overpressure Relief System
Loss of PCS inventory during depressurized and outage operations when irradiated fuel elements are in the reactor vessel	Vessel Level Alarm System

Table ES-3. Accident sequences.

	Event Categories	Major Accident Sequences	Frequency[a]
1.	Increase in Heat Removal by the Secondary (Cold Water Addition)	The major accident sequences in this event category are bounded by the limiting reactivity addition events described in Section 15.4 of Chapter 15. The major accident sequences are:	
		a. Secondary Coolant System Flow Control Valve Fail to the Open Position	Condition 2
		b. Excessive Heat Removal from the Secondary Coolant System	Condition 2
		c. Starting a Standby Primary Coolant Pump	Condition 3
2.	Decrease in Heat Removal by the Secondary	Complete Loss of Heat Sink	Condition 4
3.	Decrease in Reactor Primary Coolant Flow Rate	Loss of Commercial Power to the Site	Condition 2
		Slow Closing Primary Pump Discharge Check Valve	Condition 3
		Long Term Complete Loss of Flow	Condition 4
		Complete Closure of the Flow Control Butterfly Valve	Condition 4
4.	Reactivity and Power Distribution Anomalies	Withdrawal of all Outer Shim and Neck Shim from 10^{-5} N_F	Condition 2
		0.50$ Step Insertion[b]	Condition 2, 3, and 4
		Inpile Tube Decompression with Subsequent Voiding due to failure of the 1/2 in. piping	Condition 2
		Inpile Tube Decompression with Subsequent Voiding due to a failure of piping > 1/2 in. in diameter	Condition 4
5.	Increase in Reactor Coolant Inventory[c]	A Loss of Instrument Air closes back pressure control value PCV-1-1 and opens pressurizing flow control value FCV-1-8	Condition 2
		Loss of Instrument Air closes PCV-1-1 and opens FCV-1-8 with concurrent failure of the Inlet Pressure Subsystem	Condition 4
6.	Decrease in Reactor Coolant Inventory	Heat Exchangers Tube Rupture	Condition 1
		A 3-in. diameter opening in the PCS due to opening a drain valve, relief valve, or vent valve	Condition 2
		Rupture of PCS Outside Radiographic Limits	Condition 3
		Opening Valve outside Radiographic limits or a Radiographic Boundary (shutdown)	Condition 3
		A seismic induced LOCA	Condition 4
		Pressure Tube Penetration or Bottom Head Closure Plug Failure (shutdown)	Condition 4
7.	Radioactive Release from a Subsystem or Component	1/2-in. Experiment Loop LOCA	Condition 2
		Greater than 1/2 in. Experiment Loop LOCA	Condition 4

Table ES-3. (continued).

	Event Categories	Major Accident Sequences	Frequency[a]
8.	Fuel Storage and Cask Handling Events	Dropping a heavy cask from an elevation less than 1 ft above the canal floor, limited PCS leakage with the drop chute cover removed or other small or limited failure of the storage canal.	Condition 3
		Inadvertent Opening or failure of a Canal Drain Valve or drain piping with the Drain Cover also having been Removed	Condition 4
		A large failure of the storage canal, canal bellows or a rapid draining of the PCS with the drop chute cover removed.	Condition 4
		Dropping an Irradiated Fuel Element during Fuel Handling either in the Reactor Vessel or in the working canal with no significant damage to the Fuel Element	Condition 2
		Minor damage to one fuel element in the canal with a minor fission product release	Condition 3
		Significant melting of one fuel element (or partial melting in more than one element) due to crushing or other accidents	Condition 4
		Cask and Heavy Load Drops within handling limits[d]	Condition 3
		Cask and Heavy Load Drops outside handling limits[d]	Condition 4
9.	Depressurized Operation - Low Flow Events	Momentary Loss of Flow due to an emergency pump failure or inadvertently turned off	Condition 3
		Extended Loss of Flow Due to Extended Failure of Both Emergency Coolant Pumps	Condition 4
	Depressurized Operation - Loss of Coolant Accidents	Opening Valve Outside Radiographic Limits or at Radiographic Boundary	Condition 3
		Withdrawal of All Outer Shims from 10^{-5} N_F	Condition 2
	Depressurized Operation - Reactivity Accidents	Rapid Regulating Rod Withdrawal	Condition 3
		Loop Heaters Accidentally On	Condition 3
		Slow Power Increase or Lobe Power Balance Shift	Condition 3
		Perched Fuel Drop from Within Measurement Accuracy	Condition 3
		Cold Water Addition due to inadvertent startup of a PCP or inadvertent actuation of EFIS	Condition 3
		Cold Experiment Loop Draining Due to an Open Normally Inaccessible Valve	Condition 4
		Step Reactivity Insertions	Condition 4

Table ES-3. (continued).

	Event Categories	Major Accident Sequences	Frequency[a]
10.	Fuel Failures	Partial Fuel Channel Blockage from Small Debris in the PCS	Condition 2
		Unexpected High Peaking in the Fuel due to an Analysis or loading error	Condition 2
		Small Fuel Element Defects or Minor Fuel Handling Damage to fuel	Condition 2
		Fuel Channel Blockage from a Sight Glass or Other Inventoried Items Left in the Reactor System after a Shutdown or Due to Failure of Small Structural Elements in the ATR Vessel above the Core or Due to Failure of the Inlet Screen.	Condition 3
		Structural Fuel Element Defects Affecting a few Fuel Elements in the Core Due to Design or Fabrication Errors or Handling Damage	Condition 3
		Fuel Plate Buckling Due to Excessive Buildup of Oxide on Fuel Plate Surface	Condition 3
		Gas Leakage into the Fuel Element Coolant Channels due to an Insulating Jacket Failure during Normal Pressurized Operation	Condition 3
		Fuel Channel Blockage or Fuel Damage Due to the Failure of Large Structural Elements above or within the core	Condition 4
		Structural Fuel element Defects Affecting many Fueled Elements in the Core Due to Design or Fabrication Errors or Severe Handling Damage	Condition 4
11.	Failure of Supporting Systems	Various	Various
12.	Severe Accidents	Direct Damage LOCA	Beyond Design Basis
13.	PCS High Pressure Anomalies	A Primary Pump Discharge Valve or Check Valve closes abruptly	Condition 3
		Flow Control Valve (Butterfly) Failure	Condition 4
14.	PCS High Flow Anomalies	Opening of Flow Control Butterfly Valve to Full Open or the starting of a third PCP with two already in operation	Condition 2
15.	Reflector Aging	Reflector Cracking	Condition 2

Table ES-3. (continued).

	Event Categories	Major Accident Sequences	Frequency[a]
16.	External Events	Internal Fire, Internal Flooding Accidental Release of Chemicals from Onsite storage, Industrial Facility Accident, Seismic, External Flooding, Volcanic, Transportation, Aircraft, Extreme Wind, and Tornado.	[e]

a. Frequency listed by event category.
Condition 1 - Normal Operations
Condition 2 – Anticipated Faults
Condition 3 – Unlikely Faults
Condition 4 – Extremely Unlikely Faults

b. While the 0.50$ step does not have a definitive initiator, its; consequences are used to envelope numerous Condition 3 events. The consequences of a 0.50$ step will also meet Condition 2 damage criteria and is used to bound Condition 2 events. The consequences of a 0.50$ step and bound consequences of some Condition 4 faults.

c. There are two identified Condition 3 Events for this category, but they are not carried on this table because each is bounded by the Condition 2 event listed.

d. Event discussed for several areas in ATR-670.

e. Total Fuel damage frequency for all events is 3.3 E^{-5}/yr. Individual events leading to fuel damage are beyond design basis.

Table ES-4. ATR PRA results summary.

Significant ATR fuel damage accidents (Level 1 PRA)	Frequency[a]
Total mean fuel damage frequency	3.3×10^{-5}/yr
95% uncertainty bound	9.6×10^{-5}/yr
Internal events (system failures)	
During power operation	2.4×10^{-6}/yr
Shutdown operations, heavy load drops, canal draining	5.3×10^{-6}/yr
External events (earthquakes, severe winds, etc.)	2.5×10^{-5}/yr
Top accident sequence group	
Seismic events resulting in long-term loss of ac power and firewater injection failure	1.3×10^{-5}/yr
Top internal events (system failures) accident sequences	
Power operation	9.8×10^{-7}/yr
Small LOCA with firewater injection failure	
Shutdown, heavy load handling, canal draining	4.1×10^{-7}/yr
Cask drop over reactor above lift height limit	
Public and onsite risk (Level 3 PRA)	**Frequency[b]**
Risk of an early/prompt fatality (per exposed individual)	$< 1 \times 10^{-10}$/yr
Total mean risk of a latent cancer fatality per exposed individual	
Out to 30 miles from ATR (approximately 10 miles from INEEL)	1.5×10^{-8}/yr
TRA (with evacuation)	8.0×10^{-8}/yr
INTEC (with evacuation)	2.1×10^{-9}/yr

a. Frequency listed is the total frequency for all accidents in an event category.

b. Frequency listed is the total from all events.

Table ES-5. Summary table of worse case events resulting in a radioactive consequence.

Event	Worker Dose TEDE[1]/Thyroid (rem)	EVAC Dose TEDE/Thyroid (rem)	Dose at EA TEDE/Thyroid (rem)	Dose at LPZ TEDE/Thyroid (rem)	Event Classification	Acceptance Criteria TEDE/Thyroid (rem)
Hypothetical Large Break LOCA (Section 15.12)	NC/NC	5.28/75.7	7.53/82.7 (2 hour exposure)	13.2/185	Beyond design basis	25/300 @ LPZ
Flow Blockage (Section 15.10.3)	NC/NC	0.26/3.78	0.38/4.13 (2 hour exposure)	0.66/9.25	Condition 4	25/300 @ Evacuating workers, EA, and LPZ
Canal Fuel Damage Event (Section 15.8.7)	NC/NC	1.64/23.5	2.3/25.6 (2 hour exposure)	4.1/57.4	Condition 4	25/300 @ Evacuating workers, EA, and LPZ
Loop Piping Failure - < 1/2 in. (Section 15.7)	1.95/NC	0.00741/0.188	0.192/4.64 (Full plume passage)	NC/NC	Condition 2	5/NC to workers, 0.5/year for off-site personnel
Heat Exchanger Tube Failure (Section 15.6.3)	<0.5/NC	NC/NC	<0.0025/NC	NC/NC	Condition 1	5/NC to workers, < 0.010/NC @ nearest site boundary

EA Exclusion area
EVAC Evacuation
LPZ low population zone
NC Not calculated
TEDE Total Effective Dose Equivalent
(1) Doses in the UFSAR are reported either as EDE, TEDE, or whole body dose. These terms are used interchangeably.

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CHAPTER 1

FACILITY OVERVIEW

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1.3-1. Matrix of DOE Order 5480.23 requirements correlated with sections in the Upgraded Final Safety Analysis Report 1-13

ACRONYMS/ABBREVIATIONS

ATR	Advanced Test Reactor
CLOFA	complete loss of flow accident
DOE	U.S. Department of Energy
EDF	engineering design file
EFIS	emergency firewater injection system
INEEL	Idaho National Engineering and Environmental Laboratory
NRC	U.S. Nuclear Regulatory Commission
PCS	primary coolant system
PPS	plant protective system
RP	Reactor Programs
RMS	radiation monitoring system
TRA	Test Reactor Area
UFSAR	Upgraded Final Safety Analysis Report

1. Facility Overview

1.1 Introduction

This section briefly describes the Advanced Test Reactor (ATR), the facilities and buildings which are considered in the Upgraded Final Safety Analysis Report (UFSAR), and the specific safety or mitigative features designed to minimize the consequences associated with the postulated accident scenarios.

1.1.1 General Description of Facility

Construction began in November 1961 on the ATR, the world's largest test reactor, with an operating thermal power level of up to 250 MW. The first nuclear startup of ATR, at zero power, was achieved July 2, 1967, and full power was achieved August 16, 1969. The first operation with inpile experiments began December 25, 1969.

The ATR is designed for use in testing advanced nuclear fuel systems and materials. It provides a high neutron flux (up to 1×10^{15} n/cm^2/sec thermal) environment for flux traps that may contain inpile tubes for high-pressure loops or other flux trap irradiation facilities. The cylindrical symmetry of experimental loops, their diameter (averaging about 3 in.), and the large number of samples to be irradiated at one time strongly influenced the design of the reactor.

The ATR is a light-water moderated and cooled reactor. It provides high neutron flux levels in the test spaces without excessive power densities in the fuel elements. The ATR is unique in that the fuel is not in a compact core, but provides a close coupling of the flux traps and the fuel by means of a serpentine arrangement. When viewed from above, the fuel configuration looks like a four leaf clover with the fuel elements winding in and around a three-by-three array of flux traps. The flux traps within the lobes that correspond to the leaves of the clover are almost entirely surrounded by fuel as is the irradiation position in the center of the core. The flux traps between the lobes have fuel on half the circumference. The flux traps provide an intense source of neutrons. The ATR contains a control configuration that allows the lobe powers to be controlled independently (within design limitations). This flexibility not only allows the tailoring of the fluxes to meet the need of each experiment, but it also provides for the most efficient use of the reactor power.

In addition to the nine flux traps, space is provided in the neck shim housing and reflector for experiments of various sizes. Several design features incorporated in the ATR to improve experiment performance are unique in test reactor technology. These features are: (a) the use of flux trapping to provide high thermal neutron populations in the nine irradiation locations, (b) incorporation of special shim designs to minimize perturbations to the axial flux symmetry throughout a fuel cycle, and (c) regional power measurement and control to shift power (within limits) in the core lobes, thereby optimizing neutron flux for particular experiments.

1.1.2 Physical Boundaries of Facilities Considered in the UFSAR

The reactor and Primary Coolant System, along with the control room and much of the auxiliary and experimental support equipment, are located in Test Reactor Area (TRA) Building 670. Buildings and structures in other parts of TRA provide additional support including commercial electric power, raw and demineralized water, and fire protection. The following buildings[a] and structures that are necessary for safe operation of the ATR and define the boundaries of the ATR Facility were considered in the UFSAR:

- Buildings
  - TRA-670, ATR Reactor Building (excluding ATR Critical Facility)
  - TRA-671, ATR Cooling Tower Pumphouse
  - TRA-609, Compressor Building, Switchgear and Compressor Room
  - TRA-674, Diesel Building
  - TRA-608, Demineralizer Building
  - TRA-619, Raw Water Pumphouse
  - TRA-633, Firewater Pumphouse
  - TRA-634, ATR Storage Facility (during experiment storage in casks)
  - TRA-688, Firewater Pumphouse
- Structures
  - TRA-708A, Demineralized Water Storage Tank
  - TRA-770, ATR Vent Stack
  - TRA-771, ATR Cooling Towers
  - TRA-774, Transformer Yard
  - TRA-776, ATR 1,500-gal Diesel Fuel Tank
  - TRA-763, Compressed Air Storage Tanks
  - TRA-718, Overhead Raw Water Storage Tank
  - TRA-719, Ground-Level Storage Tanks
  - TRA-754, Demineralized Water Storage Tank
  - TRA-781, Firewater Storage Tank

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[a] Even though building TRA-689 is tied to TRA-670 by pipe, building TRA-689 is not included in this list, since the building is not in use at this time. The piping between TRA-689 and TRA-670 is isolated by both MOV-32-1 and MOV-1-909. Since the piping connections to TRA-689 are downstream of either a <3-in. orifice or valve, any break is bounded by the analysis in Chapter 15.6.

Other facilities in the Test Reactor Area may provide support to the ATR on an as-needed or indirect basis. These facilities are not included in the UFSAR. Such facilities include TRA-607, Carpenter Shop; TRA-653, Maintenance Building (Machine Shop, Welding Shop); TRA-625, Maintenance Support building (Mechanical Shop, Instrument Shop); TRA-662, Storage and Receiving Building; TRA-679, Training and ATR Simulator building; and various office and training facilities. The UFSAR also does not include TRA facilities under the TRA Landlord's jurisdiction where operation of these facilities does not directly support operation of the ATR. These facilities include TRA-605, TRA waste water treatment; TRA-624, Sewage Treatment Plant; TRA-715, the evaporation pond; and TRA-702, Cold Waste Pond. A plot plan of the Test Reactor Area is shown in Figure 1.1-1.

The ATR Critical Facility contains a nuclear duplicate of the ATR core. This facility provides data as needed for ATR fuel and experiment loadings to ensure safe and efficient operation of the ATR. This facility has a safety analysis report and is therefore excluded from this analysis. A number of nonreactor facilities—including TRA-632, Hot Cells; and TRA-621, Nuclear Materials Inspection and Storage Building—also have their own safety analysis reports; these facilities are excluded from this analysis.

The analysis of steady state normal operation of the reactor is given in Chapter 4 (Reactor). The analysis of off-normal events is documented in Chapter 15 (Accident Analyses) of this report, which includes references to the ATR Probabilistic Risk Assessment (Eide, Khericha, and Thatcher 1991; Thatcher et al. 1994) and to specific engineering design files. The off-normal events presented in Chapter 15 range from operational transients to severe accidents.

1.1.3 Safety Features of the ATR

The safety features of systems available at the ATR to mitigate postulated accident scenarios include the following:

- Plant Protective System—The plant protective system (PPS) constitutes those active devices, with the associated equipment that initiates their action, whose function in conjunction with passive structures of the plant is necessary to prevent unacceptable release or spread of radioactive materials by preventing plant variables or conditions from reaching their respective safety limits, or mitigating the consequences of exceeding safety limits. The PPS also prevents unacceptable plant damage in accordance with the limits specified by the contractor. The PPS includes reactor shutdown systems and the engineered safety features (ESFs).

- Engineered Safety Feature—An ESF is an essential protective system or subsystem that functions to mitigate the consequences of an incident by providing protection other than scram.

- LOCA Primary Coolant Pump (PCP) Shutoff System—The LOCA Primary Coolant Pump Shutoff System is an ESF function provided by the inputs from the Reactor Shutdown System Reactor Inlet Pressure and Outlet Pressure Subsystems. The purpose of the LOCA Primary Coolant Pump Shutoff System is to shutoff all operating primary coolant pumps in the event of a loss-of-coolant-accident (LOCA). Shutoff of the primary coolant pumps needs to occur prior to air being drawn from the surge tank into the primary cooling system (PCS). Air in the PCS could cause a degradation of emergency pump flow.

- Radiation Monitoring and Seal System—The Radiation Monitoring and Seal System prevents the uncontrolled release of airborne radioactive contamination from the ATR confinement. The system is composed of a stack exhaust monitoring subsystem, two actuation channels, two Radiation Monitoring System (RMS-1) relays, and quick-closing dampers with their associated solenoids and actuators. The system actuates the building isolation system when high radiation is detected in the stack exhaust.

- Emergency Firewater Injection System—The Emergency Firewater Injection System (EFIS) provides emergency reactor core cooling and flooding of the reactor vessel in the event of a major break in the PCS piping or other events resulting in loss of primary coolant. This engineered safety feature is designed to activate at a predetermined low pressure or when the water level lowers to a predetermined height above the top of the core. The ATR EFIS comprises three systems: (a) the Upper Vessel EFIS System, (b) the Bottom Head EFIS System, and (c) the EFIS Actuation System. The configurations of the piping systems are similar, but the Upper Vessel EFIS is routed to distribute emergency firewater coolant through the reactor vessel lower drain piping, while the Bottom Head EFIS is routed to distribute emergency firewater coolant through the reactor vessel bottom head.

  The Upper Vessel EFIS connection to the PCS piping is in the nozzle trench. The bottom head EFIS routes firewater directly into the reactor bottom head. Both systems are equipped with failsafe valving (i.e., valves fail open), independent level and pressure channels, and two-out-of-three actuation logic. Three level channels and three pressure channels transmit water level and vessel pressure signals to both the upper vessel and bottom head actuation logic. The EFIS instrumentation is designed to actuate the EFIS in time to prevent the core from uncovering, and to not actuate the EFIS when firewater injection into the PCS is not needed.

- The Vessel Vent System—The Vessel Vent System functions to depressurize the ATR vessel. It is used to mitigate high pressure events such as the long-term complete loss-of-flow accident (CLOFA). In the CLOFA, core decay heat maintains pressure in the vessel in excess of the supply pressure of the EFIS. Venting the vessel under such conditions is required to allow the EFIS coolant to flow into the ATR vessel and keep the core covered. The Vessel Vent System is designed to be manually actuated. The system is actuated by opening two parallel power-actuated vent valves located in the reactor vessel upper drain line.

  An alternate procedure for preventing a pressure buildup in the ATR vessel is to establish a purge through the reactor vessel and out the reactor vessel upper drain piping. However, if the reactor vessel upper drain valve is inaccessible or cannot be opened due to mechanical problems, then the Vessel Vent System allows purging to prevent pressure buildup.

- Primary Pump Shutoff System—The Primary Pump Shutoff System for ATR is part of the PPS and uses the same transducers and comparators as the Reactor Shutdown System. The purpose of the Primary Pumps Shutoff System is to stop one or more primary coolant pumps before a differential pressure across the reactor core causes damage to the fuel elements or core structures.

  The differential pressure across the reactor core stresses the core structures and fuel elements. If these structures fail or strain excessively, core geometry may not be maintained, and the fuel elements may be damaged. The Primary Pumps Shutoff System will stop at least one primary pump during three-pump operation to prevent overstressing the ATR core components or fuel elements from high flow. A reactor scram is also generated by the Reactor Shutdown System.

- Pressurizing and Gland Seal Pump Shutoff System—The Pressurizing Pumps and Gland Seal Pumps Shutoff System for ATR is part of the PPS, using the same transducers and comparators as the Reactor Shutdown System. The purpose of the Pressurizing Pump and Gland Seal Pump Shutoff System is to protect the PCS from exceeding the system design pressure. Upon receiving an overpressure trip signal, the Pressurizing Pumps and Gland Seal Pumps Shutoff System stops the primary pressurizing pumps and gland seal pumps; a reactor scram is also initiated by the Reactor Shutdown System.

- Primary Coolant System Overpressure Relief System—The Overpressure Relief System consists of two spring-loaded safety relief valves mounted on the reactor vessel 8-in. lower drain line. This safety feature provides overpressure protection for the primary coolant boundary. The safety relief valves ensure that PCS piping does not exceed 120% of design pressures in the event the Pressurizing Pump and Gland Seal Shutoff System fails to trip the pressurizing pumps.

FOIA Exemption 3 UCNI, Exemption 7f (Page 1-10 -- Plot Plan of RTC)

1.2 Facility Management

1.2.1 Prime Contractors

This section discusses the prime contractors associated with the construction and operation of the ATR facility.

1.2.1.1 Design, Construction, and Architect Engineer Contractors. The ATR design contractor was Ebasco Services, Incorporated, with the Babcock and Wilcox Company as the nuclear subcontractor. These firms also provided inspection services for the fabrication of components and for construction. The prime contractor for construction was the Fluor Corporation.

1.2.1.2 Current Contractors. The initial operating contractor was Phillips Petroleum Company. Subsequent operating contractors included Idaho Nuclear Company, Aerojet Nuclear Company, EG&G Idaho, Inc., Lockheed Martin Idaho Technologies Company, and Bechtel B&W Idaho.

As discussed in Chapter 13 (Management, Organization, and Institutional Safety Provisions), there are established policies and procedures to ensure the safe operation of the ATR. Reactor Programs (RP) was established by the Contractor with the specific responsibility to safely operate the ATR. The RP organizational structure provides clear lines of authority to coordinate and communicate the discovery, management, and resolution of safety issues. Independent internal organizations have also been established that are responsible for providing surveillance over nuclear safety activities. The RP organization provides the ATR with the following functional responsibilities: reactor operations staffing and management, facility training, engineering (plant, nuclear, and experiments), and maintenance.

1.3 Document Structure

The initial plan for this UFSAR was prepared and submitted in August 1989. The 1989 plan was based on Department of Energy (DOE) orders and applicable Nuclear Regulatory Commission (NRC) guidance existing at that time, particularly the requirements for the safety analysis reports for DOE nuclear reactors in DOE Orders 5480.6 (DOE 1986) and 5481.1B (DOE 1987). DOE Order 5480.6 required new safety analysis reports to follow NRC guidelines for standard format and content. Based on this requirement, the ATR UFSAR plan is based on the NRC guidelines for format and content (NRC 1978, 1987).

Evolving DOE guidance at that time required both probabilistic risk assessment-based and traditional deterministic safety analyses in its safety programs. This guidance was contained in DOE Secretary of Energy Notice SEN-35-91 and DOE Order 5480.23 (DOE 1991, 1994). As part of the program to upgrade the safety envelope for ATR, the UFSAR upgrade was coordinated with other safety upgrade activities, which involved considerable interaction. Specifically, the probabilistic risk assessment and the Strategic Evaluation Program (Buescher 1993) results were used to minimize the number of new quantitative analyses and to focus on improving the technical safety requirements (either to relax them when a restriction did not significantly increase safety, or to tighten them as necessary to provide additional safety). At the same time, the program recognized the expanded scope of DOE-required safety analyses to include severe accidents, risk assessments, human factors engineering, and risk management programs.

The DOE Order for safety analysis reports, DOE 5480.23 (DOE 1994), included increased concern for worker safety, safety concerns about factors other than hardware (human factors, training, management and organization, institutional, communication, procedures, etc.), and concerns about hazardous and toxic chemical hazards. DOE Order 5480.23 canceled the paragraphs of DOE 5480.6 and 5481.1B that had defined the requirements for the safety analysis reports for DOE nuclear reactors, in particular Paragraph 8c of 5480.6. Paragraph 8c required that SARs follow the NRC guidelines on standard format and content. The 5480.23 Order established different requirements for minimum SAR content. The guidance in the Order also acknowledged existing upgrade programs and was concerned that SARs include the required information and not that they conform to an arbitrary pattern. This safety analysis report takes advantage of much of the work that was already completed prior to the publication of DOE 5480.23. It followed much of the original outline developed using NRC guidelines and included the additional content required by the DOE 5480.23.

Paragraph 8.b.(3) of DOE Order 5480.23, "Nuclear Safety Analysis Reports" (SAR), specified required topics to be included in the SAR (executive summary; applicable statutes, rules, regulations and Departmental Orders; site characteristics; etc.). Attachment 1 to the Order, "Interim Guidance for DOE Order 5480.23, Safety Analysis Report (SAR)," Paragraph 4.f.(3)(d), provided guidance on the intended content for each of the required topics of 8.b.(3) of the Order. A detailed comparison was made between the requirements of DOE Order 5480.23 and the guidance of Regulatory Guide 1.70 (NRC 1978). This resulted in identification of the need to develop new chapters or sections within existing chapters for the UFSAR, because some of the requirements found in DOE Order 5480.23 had not been covered by Regulatory Guide 1.70. Table 1.3-1 correlates the guidance paragraphs (related to the requirements of paragraph 8.b.(3)) of Attachment 1 of the Order to each UFSAR section or chapter which addresses or encompasses the identified guidance paragraph for each requirement.

The current requirements for safety analyses for DOE reactors are contained in 10 CFR 830 Subpart B, Safety Basis Requirements (CFR 2002). 10 CFR 830 specifies following the NRC Regulatory Guide 1.70 as the safe harbor standard, thus this UFSAR follows the standard for Documented Safety Analyses specified in the current regulations.

Because ATR is an operating facility, operating experience and existing information have been used as appropriate in the SAR upgrade program rather than performing system calculations. Much of the content developed in the UFSAR has been descriptive rather than analytical, consistent with DOE guidance.

All chapters in the UFSAR contain reference lists. The guidelines followed for references are that all references should be either published material or documents available from an Idaho National Engineering and Environmental Laboratory (INEEL) document center. New engineering evaluations and analyses are documented in controlled documents, usually Engineering Design Files (EDFs). Some older documents, which were used as references, were not controlled or available from an INEEL document center. Many of these older documents have been archived in the TRA Nuclear Engineering library. Other of the older documents have been incorporated into EDFs to make them controlled documents. In the case of the latter, the author and date of the original documents, however, are superseded by the author currently preparing the EDF and by date that the EDF is approved and filed at a controlled document center.

Table 1.3-1. Matrix of DOE Order 5480.23 requirements correlated with sections in the Upgraded Final Safety Analysis Report.

DOE 5480.23 Requirement	UFSAR Section
1. **Executive Summary**	Executive Summary and Chapter 1
1 a	Executive Summary
1 b	Section 1.1 and 1.2
1 c	Section 1.3
1 d	Section 1.3
2. **Applicable Statutes, Rules, and Departmental Orders**	Chapter 19
3. **Site Characteristics**	Chapter 2
3 a	Section 2.1
3 b i, ii, iii, v	Section 2.1
3 b iv	Sections 2.1 and 2.2
3 c	Chapter 2
3 d	Sections 2.1, 2.3, 2.4, and 2.5
3 e	Section 2.2
4. **Facility Description and Operation, Including Design of Principal Structures, Components, Systems, Engineered Safety Features, and Processes**	
4 a	Chapters 3 to 10[a]
4 a i	Chapters 3 and 6
4 a ii	Chapters 3 to 11[a]
4 a iii	Chapters 3 to 10[a]
4 a iv	Sections 9.2 and 9.3
4 a v	Chapters 7 and 9
4 b	Chapters 3 to 10[a]
4 b i, ii, iii, iv,	Chapter 6
4 c	Chapters 6 and 14
5. **Hazard Analysis and Classification of the Facility**	Chapter 20
6. **Principal Health and Safety Criteria**	Chapters 3 and 19
6 a, d, e	Section 3.1
6 b, c	Section 3.2 and 3.11
7. **Radioactive and Hazardous Material Waste Management**	Chapter 11
7 a	Sections 11.1, 11.3, 11.4, and 11.5
7 b	Sections 11.1 and 11.2
7 b i	Section 11.1
7 b ii	Sections 11.2
7 b iii	Sections 11.3, 11.4, 11.5, and 11.6
8. **Inadvertent Criticality Protection**	Sections 4.3, 9.1 and 13.5
9. **Radiation Protection**	Chapter 12
9 a through m	Section 12.7

a. DOE 5480.23 Requirement 4, Facility description and Operation, is organized based on information requirements in Parts 4a and 4b. The UFSAR is organized based on the major functional systems of the facility in Chapters 3-10. Thus, the information required by the Order in each of these various parts are included as appropriate in each of these Chapters.

Table 1.3-1. (continued).

DOE 5480.23 Requirement	UFSAR Section
10. **Hazardous Material Protection**	Chapter 21
10 a	Section 21.1
10 b	Section 21.3
10 c, d	Section 21.4
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CHAPTER 2

SITE CHARACTERISTICS

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ACRONYMS/ABBREVIATIONS

ANL-W	Argonne National Laboratory-West
ARA	Auxiliary Reactor Area
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical Facility
CFA	Central Facilities Area
cfs	Cubic Feet per Second
CH	Contact-Handled
CTB	Centennial Tectonic Belt
DBE	design basis earthquake
DOE	U.S. Department of Energy
DOE-ID	U.S. Department of Energy, Idaho Operations Office
EBR-I	Experimental Breeder Reactor I
EG&G Idaho	EG&G Idaho, Inc.
ESRP	Eastern Snake River Plain
FAST	Fluorinel Dissolution Process and Fuel Storage Facility
FSA	Fuel Storage Area
FSAR	Final Safety Analysis Report
HEPA	High Efficiency Particulate Air Filter
HLLW	High-Level Liquid Waste
HLW	High-Level Waste
ICPP	Idaho Chemical Processing Plant
IDLH	Immediately Dangerous to Life and Health
ILTSF	Intermediate Level Transuranic Storage Facility
INEL	Idaho National Engineering Laboratory
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center (formerly Idaho Chemical Processing Plant ICPP)
IPC	Idaho Power Company
ISB	Intermountain Seismic Belt
Ka	thousand years ago
LLW	Low-Level Waste
LOFT	Loss-of-Fluid Test
LPZ	low population zone
Ma	million years ago
M_L	Low Magnitude
M_s	Surface-Wave Magnitude
msl	mean sea level
MST	Mountain Standard Time

NCDC	National Climatic Data Center
NMIS	Nuclear Materials Inspection and Storage
NOAA	National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission
NRF	Naval Reactors Facility
PBF	Power Burst Facility
PMF	probable maximum flood
PMP	probable maximum precipitation
RH	Remote-Handled
RP	Reactor Programs
RTU	Remote Terminal Unit
RWMC	Radioactive Waste Management Complex
SDA	Subsurface Disposal Area
SNM	special nuclear material
SPERT	Special Power Excursion Reactor Test
SPOII	Security Police Officer Level II
SSE	safe shutdown earthquake
TAN	Test Area North
TLV-STEL	Threshold Limit Value - Short Term Exposure Limit
TRA	Test Reactor Area
TRAHC	Test Reactor Area Hot Cell
TREAT	Transient Reactor Test Facility
TRU	Transuranic
TSA	Transuranic Storage Area
UP&L	Utah Power and Light
USGS	U.S. Geological Survey
UTM	Universal Transverse Mercator
VMF	Vehicle Monitoring Facility
VRZ	volcanic rift zone
V_s	Shear-Wave Velocity
WCC	Woodward-Clyde Consultants
WINCO	Westinghouse Idaho Nuclear Company
WIPP	Waste Isolation Pilot Plant

2. SITE CHARACTERISTICS

This chapter, using available data and analyses, quantifies characteristics of the surrounding environment that influence the design, procedures, and safety of the Advanced Test Reactor (ATR) Facility operations as required by U.S. Department of Energy (DOE) Order 5480.23, "Nuclear Safety Analysis Reports" (DOE 1994). Only data relevant to safety analysis activities are included.

This chapter

- Provides site characteristics investigation and results summaries that meet the applicable recommendations of U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70 (NRC 1978).
- Provides characterization information necessary to support the requirements for (a) identifying and quantifying environmental threats that may contribute significantly to the risks posed by the facility, (b) facility hazard classification, (c) accident analyses, and (d) selecting principal safety criteria. Site characterization information includes site and vicinity information in the fields of geology, seismology, hydrology, meteorology, population distribution (present and projected), land use, site activities, and site controls.
- Describes natural phenomena affecting and affected by the facility. The level of detail provided is commensurate with the magnitude of the hazards associated with the facility or operation. An overall site plan is referenced when appropriate and applicable.
- Defines facility exclusion area(s) and the site boundary with respect to public exclusion areas, access control areas, property lines, and distances from the site boundary to potential effluent release points.

2.1 Geography and Demography

This section (a) provides a precise description of the location of the Test Reactor Area (TRA) site and general descriptions of local natural features and other Idaho National Engineering and Environmental Laboratory (INEEL) facilities and capabilities; (b) establishes the exclusion area authority and control; and (c) provides population distribution and land use information for the INEEL and vicinity.

2.1.1 Site Location and Description

2.1.1.1 Specification of Location. The ATR site is located in the south-central portion of the INEEL, a U.S. Government-owned facility managed by DOE and established for building, testing, and operating developmental nuclear reactors, research, and related facilities. Maps of the INEEL and vicinity showing locations of surface features are provided in Section 2.1.1.2.

The INEEL occupies approximately 890 mi^2 (2,305 km^2) of the northwestern portion of the Eastern Snake River Plain (ESRP) in southeastern Idaho. The INEEL is nearly 39 mi (63 km) long from north to south (extreme latitudes are 43° 26' and 44° 01' N) and approximately 36 mi (58 km) wide in its broadest southern portion (extreme longitudes are 112° 28' and 113° 9' W). The INEEL includes portions of five Idaho counties (Bingham, Bonneville, Butte, Clark, and Jefferson) and lies within Townships 2 to 8 N and Ranges 28 to 34 E, Boise baseline and meridian.

The surface of the INEEL is relatively flat-lying, semiarid, sagebrush desert, with predominant relief manifested either as volcanic buttes rising above the desert floor or as unevenly surfaced basalt flows, flow vents, and fissures. The Lemhi, Lost River, and Bitterroot mountain ranges border the Site on the north and west. The average elevation of the INEEL is 5,000 ft (1,526 m) above sea level. East, Middle, and Big Southern Buttes are in the southern portion of the INEEL and have elevations of 6,410 ft (1,952 m), 6,572 ft (2,003 m), and 7,575 ft (2,309 m) above sea level, respectively. Typical elevations on the INEEL (aside from the buttes) range from 5,200 ft (1,585 m) in the northeast to 4,750 ft (1,450 m) in the southwest. A broad topographic ridge extends northward through the INEEL. This ridge effectively separates the drainage of mountain ranges northwest of the INEEL from the Snake River.

The Big Lost River, Little Lost River, and Birch Creek flow from the valleys between the northwestern mountain ranges to the Lost River Sinks in the northwest portion of the INEEL, which acts as a closed drainage basin. Any water trapped in these sinks (also called playas, i.e., intermittent lakes) serves to recharge the Snake River Plain aquifer. The Big Lost River enters the southwestern corner of the INEEL, flows north, and percolates into the Snake River Plain aquifer at the Big Lost River Sinks near Howe, Idaho. During peak flows of the Big Lost River, water is diverted to spreading areas approximately 5 mi (8 km) east of the location where the river enters the INEEL Site. The intermittently flowing waters of the Big Lost River have created a floodplain along the lower areas on the west side of the INEEL, extending north as far as the sinks. In the past, the Little Lost River and Birch Creek entered the INEEL along the west and northern borders, respectively. However, all surface waters are now diverted for agricultural irrigation before reaching the Site and only occasionally flow onto the INEEL.

The ATR is located in Building TRA-670 at TRA. This location is E 341520; N 4827507 in universal transverse mercator (UTM) coordinates. The main floor of the reactor building is at an elevation of 4,926.26 ft (1,501.5 m) mean sea level (msl). ATR is in Butte County, Idaho.

2.1.1.2 Site Area Maps. Figure 2.1-1 shows the configuration of the INEEL and major on-Site facilities. Figure 2.1-2 shows the location of the INEEL and the Lost River, Lemhi, and Bitterroot Ranges, and Figure 2.1-3 shows the surface water features at and near the INEEL. Figure 2.1-4 is a detailed map of the TRA, showing the location of ATR and other principal plant structures.

The INEEL has many facilities and technical capabilities. These include existing reactor facilities, high-level waste (HLW) and low-level waste (LLW) processing and storage, hot cells, analytical laboratories, machine shops, and administrative facilities. The following sections briefly describe INEEL general facilities (Section 2.1.1.2.1), reactor facilities (Section 2.1.1.2.2), transportation infrastructure (Section 2.1.1.2.3), and electric power (Section 2.1.1.2.4).

2.1.1.2.1 General Site Facilities—The major INEEL facility areas are shown on Figure 2.1-1 and are listed as follows:

- Test Area North (TAN), which has large shops and hot cells, conducts special materials experiments. It is the location of the former Loss-of-Fluid Test (LOFT) facilities.
- Naval Reactors Facility (NRF), which includes ship propulsion reactors and training facilities.
- Test Reactor Area (TRA), which conducts irradiations, materials testing, and isotope production.
- Idaho Nuclear Technology and Engineering Center (INTEC), which stores defense and test reactor fuels and waste.

- Central Facilities Area (CFA), which provides services including transportation, large industrial shops, health services, and radiation monitoring.

- Argonne National Laboratory-West (ANL-W), which conducts liquid metal breeder reactor development and reactor research.

- Radioactive Waste Management Complex (RWMC), which provides LLW disposal and stores retrievable transuranic (TRU) waste.

- Power Burst Facility (PBF)/Special Power Excursion Reactor Tests (SPERTs), which conducted light water reactor fuel transient behavior studies. There are no reactors currently operating in this area. The SPERT facilities, currently store and process waste (see RWMC facilities).

In addition to the major facilities at the INEEL, other office buildings, a computer center, and two major laboratories are located in Idaho Falls, Idaho. A significant portion of INEEL-related technical, analytical, and administrative support capability is concentrated in Idaho Falls. Detailed descriptions of the INEEL facilities are provided in *Idaho National Engineering Laboratory Site Development Plan* (DOE-ID 1994).

The following sections briefly describe the INEEL facilities and capabilities that could impact or be impacted by ATR operations if accidents or other off-normal conditions occur. These include fuel storage, HLW and LLW processing and storage, reactor test facilities, transportation facilities, and INEEL electrical power systems.

2.1.1.2.1.1 Idaho Nuclear Technology and Engineering Center—The INTEC stores various types of radioactive waste and spent reactor fuel. It also processes certain types of radioactive waste. The capabilities of the INTEC include:

- Storing highly enriched (93% U-235) and other types of reactor fuel

- Storing high-level liquid waste (HLLW) in an acid form in stainless steel tanks housed in underground concrete vaults

- Converting HLLW to a solid by a production scale waste calcining facility.

2.1.1.2.1.2 Radioactive Waste Management Complex—The RWMC disposes of INEEL-generated solid, low-level, and beta/gamma contaminated wastes and also stores retrievable Contact-Handled (CH) and Remote-Handled (RH) TRU waste. The RWMC consists of two main areas: the 88-acre (33.6 ha) Subsurface Disposal Area (SDA) and the 55-acre (22.3 ha) Transuranic Storage Area (TSA).

RWMC operations presently consist of the following: (a) subsurface disposal of LLW at SDA, (b) CH TRU waste storage at TSA, and (c) RH TRU waste storage in watertight vaults at the Intermediate Level Transuranic Storage Facility (ILTSF).

The solid waste inventories and forecasts from waste generators indicate that the RWMC will be able to continue accepting waste until 2093 (EG&G Idaho 1984).

2.1.1.2.2 Reactor Facilities—The major currently operable test reactors at the INEEL are the Advanced Test Reactor (ATR) and Transient Reactor Test Facility (TREAT). These reactors provide test space at prescribed radiation conditions to simulate steady-state and accident environments.

TREAT operates as a low-power, steady-state, or transient reactor. This is a graphite-moderated, air-cooled reactor designed to produce short, intense power bursts for studying fuel behavior under transient power conditions. TREAT provides significant support to fast reactor technology and materials behavior programs.

2.1.1.2.3 Transportation—The INEEL and its associated facilities are served by a transportation network that includes public highways, local air service, onsite roads, and a railroad. This transportation system adequately supports operations of ATR.

Within the INEEL boundaries, there are approximately 87 mi (140 km) of paved roads restricted to use by employees and visitors on official business. Numerous miles of paved service roads provide access to areas including CFA, TAN, and others.

An additional 90 mi (144.4 km) of paved highways passing through the Site are used by the public. U.S. Highways 20 and 26 cross the southern portion of the Site, and Idaho State Highways 22, 28, and 33 cross the northeastern part of the Site.

Figure 2.1-1 shows the network of highways, roads, and railways on the INEEL and its vicinity.

Over 100 mi (160.9 km) of unpaved INEEL roads and trails provide access for emergency, security, and service vehicles throughout the Site.

In addition to U.S. and state highways, two major interstate highways traverse the southeast Idaho region. Interstate 15 provides north-south connections, while Interstate 86/84 provides an east-west connection from Boise and points west of Pocatello, Idaho.

Idaho Falls receives railroad freight service from Butte, Montana, to the north, and Pocatello, Idaho, and Salt Lake City, Utah, to the south. From these points, interconnections are made throughout the U.S. From Blackfoot, Idaho, the north-south railroad track carries rail freight service to the INEEL over the Mackay Branch line that connects to a Site spur south of CFA. Railroad cars are picked up at the Scoville Siding along the spur. Fourteen miles (22.5 km) of the Mackay Branch line traverses the southern part of the Site.

A Government-owned railroad track passes north from the Mackay Branch through CFA and past the east side of INTEC to NRF. A spur line runs westerly to connect this tract into and through the south end of the INTEC storage facility (INTEC Building CCP-603). INEEL-owned, -operated, and -maintained equipment is used to move freight along the Scoville Spur.

Idaho Falls air service includes a major airline, commuter airlines, and charter service.

2.1.1.2.4 INEEL Electrical Power Systems—Commercial electrical power is delivered to the operating areas at the INEEL by an extensive power transmission system (Figure 2.1-5). The system is a loop configuration with 56.5 mi (90.9 km) of a 138-kV transmission line, several miles of secondary voltage feeder lines, and seven major substations with a cumulative transformer capacity of 122.6 MVA (DOE-ID 1994). Power management personnel operate the transmission loop as a service utility providing 24-hour staffed dispatch control, a high-voltage line crew, technical and engineering assistance, and general electrical maintenance.

Electric power is supplied to the INEEL from the Idaho Power Company (IPC), a privately owned utility. Commercial electric power is transmitted to the INEEL through the Antelope Substation (see Figure 2.1-5), owned by Utah Power and Light (UP&L) Company, via two major substations on the Pacific Northwest Power Grid: the IPC Brady Substation and the UP&L Goshen Substation. If necessary, power may also be delivered to the Antelope Substation from Montana Power Company's Anaconda Substation.

At the Antelope Substation, the 230-kV bus connecting the power lines from the Brady and Anaconda Substations feeds a single transformer where 230 kV is reduced to 161 kV. The 161-kV bus is interconnected with the UP&L 161-kV Goshen-Antelope transmission line and feeds two DOE-owned 55-MVA transformers where the 161 kV is further stepped down to 138 kV. The secondary of each of these transformers is connected to a separate 138-kV transmission line, thereby providing two completely redundant (duplicate) power paths to the INEEL Substation (Scoville Substation), located at CFA.

Power is then fed over a 138-kV loop from the Scoville Substation to six other substations, including one at TRA. The loop feeds power in both directions to ensure continuous power supply. At ATR, there are three 4,160 Vac buses, two supplied from commercial power and one supplied from normally operated diesel generation independent of the commercial power. See Chapter 8 (Electric Power) for details of the ATR power supply and distribution system.

2.1.1.3 Boundaries for Establishing Effluent Release Limits. The TRA is enclosed by a double fence, which defines the restricted area. The DOE has the authority to control all activities within the boundaries of the INEEL; therefore, the exclusion area is defined as the INEEL boundary. The nearest INEEL boundary is 6.7 mi (10.8 km) to the northwest of ATR. In addition to access control onto the INEEL, access to the TRA is controlled as specified in a Reactor Programs (RP) controlled document. These access control practices are the basis for the Guard Post Orders, which define the procedures followed by the TRA security police officers (SPOII). In addition, only ATR authorized personnel have unlimited access to the ATR Building (TRA-670). All others must enter as visitors under escort.

2.1.1.4 Special Features. Archeological surveys in the 100-m zone around the perimeter of the TRA have indicated no significant cultural resources.

No known material resources of commercial value, such as minerals or lumber, exist at ATR or in the vicinity of the INEEL.

A critical habitat for threatened or endangered species has not been identified at the INEEL. The bald eagle and the peregrine falcon are the only threatened or endangered species observed at the INEEL.

2.1.1.5 Conclusion. The INEEL site and ATR location and features are described above in sufficient detail to provide the necessary framework for subsequent discussion and analyses. Transportation routes are sufficiently distant that use is not likely to interfere with ATR operation.

2.1.2 Exclusion Area Authority and Control

Except for the areas where State and Federal highways cross the INEEL, described in Section 2.1.1.3, access to the remainder of the INEEL is restricted to employees and visitors on official business.

2.1.2.1 Authority. All land within the INEEL is either withdrawn from the public domain or is DOE-acquired lands under the authority of DOE. Issues do not exist concerning land ownership.

DOE Order 473.1, "Physical Protection Program," describes the DOE policy for the physical protection of special nuclear material (SNM) and vital equipment including nuclear reactors (DOE 2002c). The provisions of this order apply to all Departmental Elements and contractors performing work for DOE as provided by law and/or contract, and as implemented by the appropriate contracting officer.

2.1.2.2 Control of Activities Unrelated to Plant Activities. The contractor's *Safeguards and Security Manual* describes the requirements and procedures to be followed for any cleared visitors, uncleared visitors, or foreign visitors to the TRA and ATR.

2.1.2.3 Arrangements for Traffic Control. The TRA restricted area is not traversed by a highway, railway, or waterway.

Vehicle access procedures into TRA are described in a RP controlled document. Only TRA business-related and construction vehicles are allowed into TRA. All vehicles that enter TRA pass through the Vehicle Monitoring Facility (VMF) before entry.

Emergency access and exit from TRA and the ATR are discussed in the *INEEL Emergency Plan/RCRA Contingency Plan and TRA Facility Specific Addendum.*

2.1.2.4 Abandonment or Relocation of Roads. No public roads traversing the exclusion area had to be abandoned or relocated.

2.1.2.5 Conclusions. The Contractor has complete personnel access control within TRA, and DOE-ID has control at the INEEL in general, including areas that are not part of a specific facility.

2.1.3 Population Distribution

The following sections contain information about population distribution within a 50-mi (80-km) radius of ATR. The 1990 census data were used to compile information on residential population distributions. Figure 2.1-6 shows the counties covered by the 50-mi (80-km) radius.

2.1.3.1 Population Within 10 Miles. In Figure 2.1-7, concentric circles have been drawn at 1, 2, 3, 4, 5, and 10 mi (1.6, 3.2, 4.8, 6.4, 8.0, and 16 km) from ATR. A residential population does not exist on the INEEL, and only a small portion of the 10-mi (16-km) radius surrounding the ATR falls outside the boundary of the INEEL. The northwest sector at 0 to 10 mi (0 to 16 km) (Figure 2.1-8) indicates a population of 1, although the population resides outside the INEEL boundary.

2.1.3.2 Population Between 10 and 50 Miles (16 and 80 Kilometers). The population for each sector (formed by radial lines dividing the 10-mi [16-km] interval concentric rings into 22.5° segments centered on the 16 compass points) is shown in Figure 2.1-8 and tabulated in Table 2.1-1. Population projections for the years 2000 and 2010 are shown in Figures 2.1-9 and 2.1-10, and are tabulated in Tables 2.1-2 through 2.1-3. The total 1990 population within the 50-mi (80-km) radius is 104,646.

2.1.3.3 Cyclic and Transient Population. Employees at the INEEL are a cyclic population. They are concentrated during assigned work shifts at several separate facilities, as shown in Figure 2.1-1. Two recreational areas, Craters of the Moon National Monument and Experimental Breeder Reactor No. 1 (EBR-I), a National Historic Landmark, are locations for transient populations. Craters of the Moon is located west and west-southwest of ATR, in the sectors 20 to 30 mi (32 to 48 km) and 30 to 40 mi (48 to 64 km) from ATR (Figure 2.1-8). Visitation is concentrated in these sectors.

During FY 1991, a total of 207,000 people visited Craters of the Moon, which has a peak season in June, July, and August.

EBR-I is generally open to the public from late May to early September. It is located southwest of the ATR approximately 6 mi (10 km). An average of approximately 10,000 people, as of 1994, visit EBR-I each year.

2.1.3.4 Low Population Zone. The low population zone (LPZ) has been defined as the area between the INEEL boundary and three-fourths the distance to the nearest population center with a population greater than 25,000, which is Idaho Falls. The distance from the TRA to the boundary of the LPZ is 34.5 mi (55.5 km). Public highways cross through the LPZ along the southern and northern portions of the INEEL (see Figure 2.1-1). The State of Idaho Transportation Department reported in 1988 that a daily average of 1,840 vehicles traveled U.S. 20 along the southern portion of the INEEL (Hardinger 1990). This number includes bus and vehicle commuters to the INEEL. State Highway 33 carried a daily average of 530 vehicles (which includes commuters) from the eastern boundary of the INEEL to Test Area North, and carried a daily average of 260 vehicles from TAN to the western boundary of the INEEL. State Highway 28 carried a daily average of 430 vehicles through the northern portion of the INEEL, and State Highway 22 carried a daily average of 150 vehicles.

2.1.3.5 Population Center. The nearest population center of more than 25,000 is Idaho Falls, which had a population of 43,929 in the 1990 census. It is located 46 mi (74 km) east of ATR (Figure 2.1-6).

2.1.3.6 Population Density. The cumulative population in 1990 within a 30-mi (48-km) radius of the ATR was 3,430. Therefore, population density within this area would be 1.2 people per mi^2 (0.47 people per km^2); excluding the area of the INEEL, the density would be 1.75 people per mi^2 (0.68 people per km^2). The cumulative population in 1990 within a 50-mi (80-km) radius is 104,646, which results in a population density of 13.3 people per mi^2 (5.2 people per km^2); excluding the INEEL, the density would be 15 people per mi^2 (5.9 people per km^2). For purposes of projecting future potential impacts as a result of population growth in the INEEL area, population projections have been made. Population density within the 50-mi (80-km) radius for the year 2000 would be 15.6 people per mi^2 (6.1 people per km^2) and 17.6 people per mi^2 (6.9 people per km^2) if the area of the INEEL is excluded. For 2010, the population density within the 50-mi (80-km) radius would be 17.3 people per mi^2 (6.7 people per km^2), and 19.5 people per mi^2 (7.6 people per km^2) if the area of the INEEL is excluded.

2.1.3.7 Conclusions. All of the population vs distance values required for safety analysis are presented above. A low population zone (LPZ) boundary is established such that the nearest population center is one and one-third times the distance to the LPZ.

2.1.4 Land Use

Approximately 300,000 acres (121,407 hectares) of the INEEL Site are open to controlled grazing by cattle or sheep. Grazing on the INEEL is limited to the shaded areas shown in Figure 2.1-11. The nearest grazing lands are 3.3 mi (5.3 km) from the ATR. Grazing is prohibited within 2 mi (3.2 km) of any nuclear facility, and livestock populations are controlled. Dairy cows are not allowed. Other uses of the land are extremely limited because of the climate, lava flows, and general desert soil characteristics. The only land at the INEEL suitable for farming is near the terminations of the Big and Little Lost Rivers, near the town of Howe. Arable land with a moderate irrigation limitation is present on both sides of the Big Lost River and in the remains of an ancient lake bed between Mud Lake and Howe. The remainder of the INEEL, approximately 65% of the surface area, has a low subsurface water-holding capacity, is rocky or covered with basalt, or is classified as having moderate-to-severe limitations for agricultural irrigation.

Land use in the region surrounding INEEL consists of about 45% agriculture, 45% open land, and 10% urban area. Agricultural use includes cropland, mostly wheat and potatoes, and grazing land for cattle and sheep.

Because public access to the INEEL is controlled, activities other then grazing within 5 mi (8 km) of the ATR are limited to DOE and DOE-support contractor activities conducted in the facilities described in Section 2.1.1.2.1.

Table 2.1-1. 1990 population distribution within a 50-mi (80-km) radius of the Advanced Test Reactor.

	Miles				
Sector	0-10	10-20	20-30	30-40	40-50
N	0	137	46	17	5
NNE	0	9	3	73	21
NE	0	0	170	778	184
ENE	0	0	64	621	1,205
E	0	0	1	376	54,853
ESE	0	0	2	264	12,658
SE	0	25	258	8,259	13,480
SSE	0	9	44	1,414	3,292
S	0	3	13	164	2,375
SSW	0	0	0	0	20
SW	0	1	3	0	0
WSW	0	3	7	8	142
W	0	330	46	10	17
WNW	0	1,194	767	26	2
NW	1	1	245	810	127
NNW	0	24	24	6	9
Total	1	1,736	1,693	12,826	88,390

Total Population 0-50 mi: 104,646.

Table 2.1-2. Projected population distribution within a 50-mi (80-km) radius of the Advanced Test Reactor for the year 2000.[a]

	Miles				
Sector	0-10	10-20	20-30	30-40	40-50
N	0	171	56	21	7
NNE	0	11	3	91	28
NE	0	0	215	969	225
ENE	0	0	79	770	1,484
E	0	0	2	444	63,982
ESE	0	0	3	318	14,931
SE	0	27	313	9,835	15,959
SSE	0	11	53	1,684	3,420
S	0	4	17	192	2,766
SSW	0	1	0	0	4
SW	0	1	4	1	0
WSW	0	4	9	13	215
W	0	401	57	15	27
WNW	1	1,434	927	34	3
NW	2	2	298	969	154
NNW	0	30	31	7	
Total	3	2,097	2,067	15,363	103,216

Total Population 0-50 mi: 122,747.

a. Source for growth projections: Idaho Power Company (IPC), 1991, *County Economic Forecasts-Demand Planning Project Report*, January.

Table 2.1-3. Projected population distribution within a 50-mi (80-km) radius of the Advanced Test Reactor for the year 2010.[a]

	Miles				
Sector	0-10	10-20	20-30	30-40	40-50
N	0	186	62	23	7
NNE	0	11	4	103	32
NE	0	0	240	1,087	252
ENE	0	0	88	861	1,661
E	0	0	2	492	70,328
ESE	0	0	3	350	16,459
SE	0	30	348	10,831	17,591
SSE	0	12	59	1,859	4,336
S	0	5	19	216	3,091
SSW	0	1	0	0	30
SW	0	1	4	1	0
WSW	0	4	9	17	290
W	0	442	63	20	36
WNW	1	1,568	1,019	37	3
NW	2	3	329	1,075	172
NNW	1	32	33	8	13
Total	4	2,295	2,282	16,980	114,301

Total Population 0-50 mi: 135,862.

a. Source for growth projections: Idaho Power Company (IPC), 1991, *County Economic Forecasts-Demand Planning Project Report*, January.

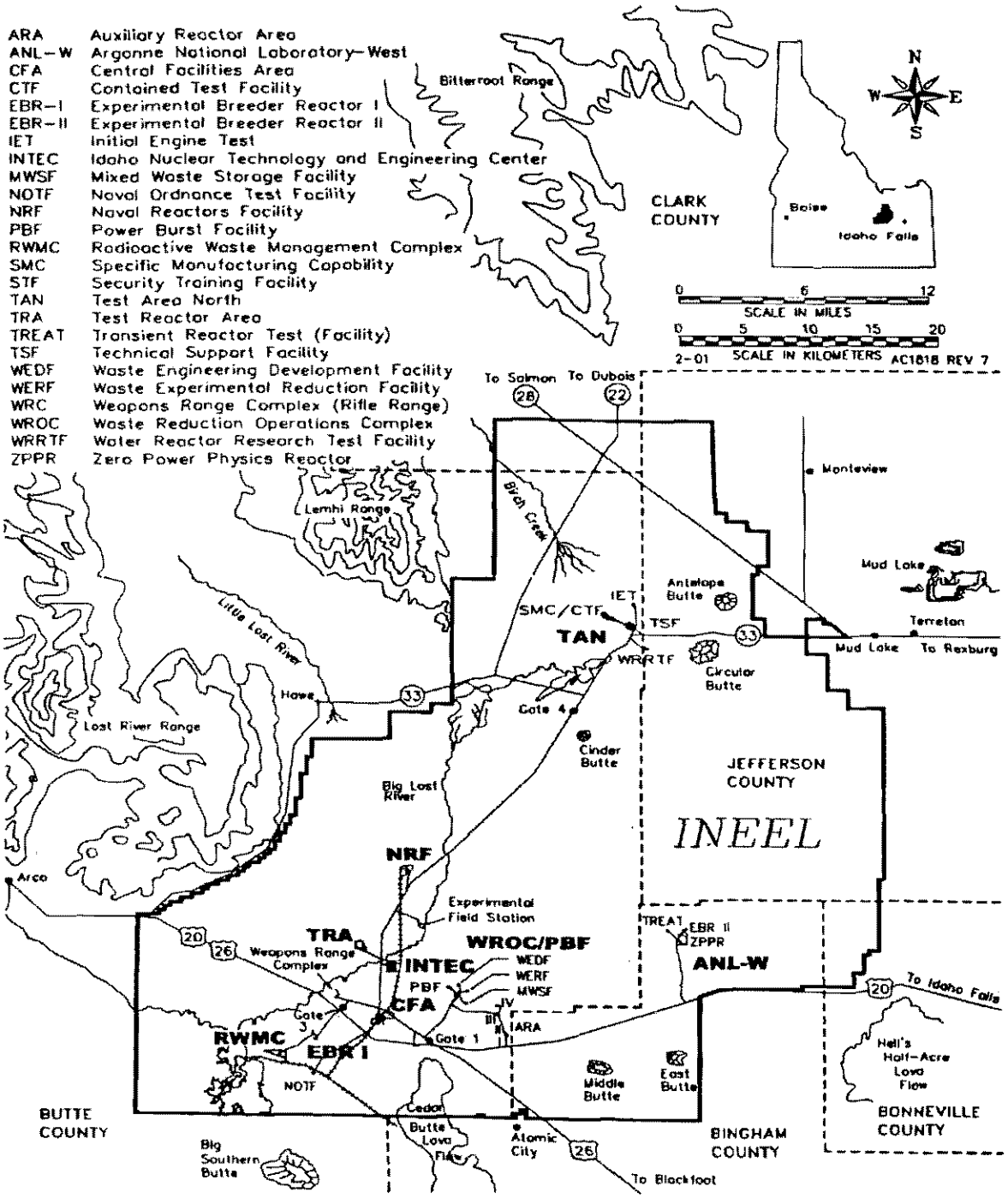

Figure 2.1-1. Major Facilities on the INEEL. **(For Information Only)**

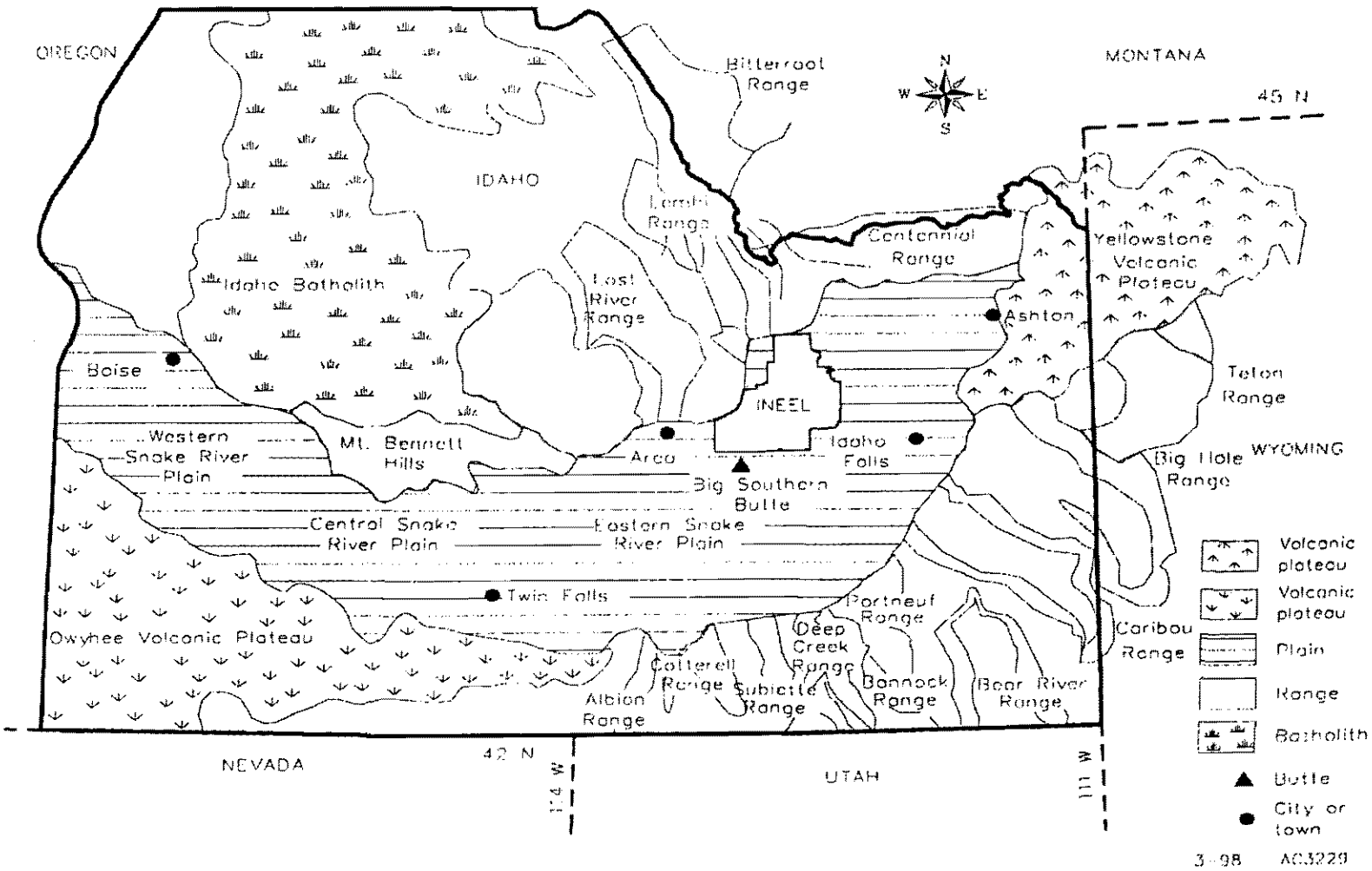

Figure 2.1-2. Geological Setting of the INEEL (modified from EG&G Idaho 1984). **(For Information Only)**

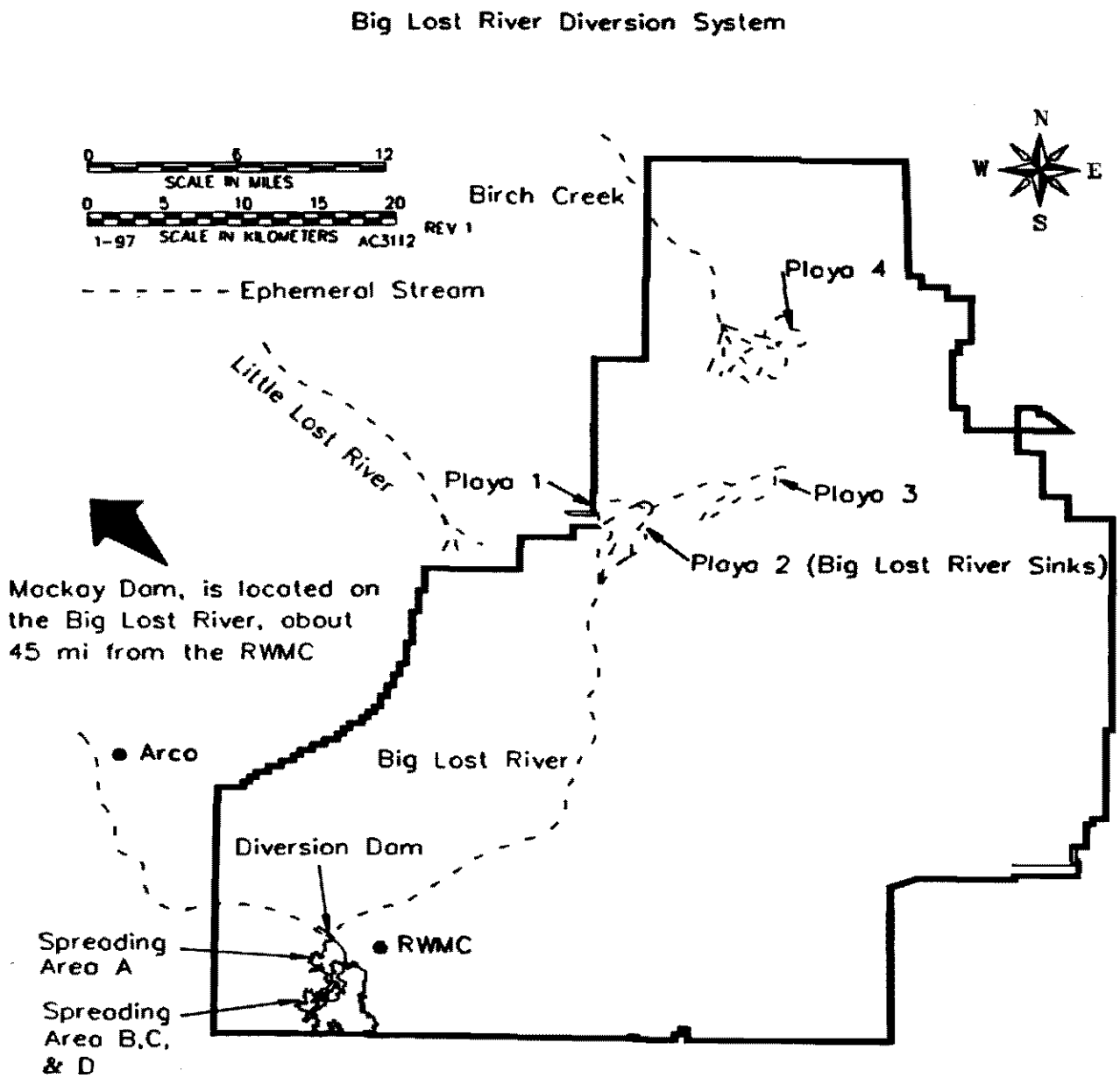

Figure 2.1-3. Surface Water Features at or near the INEEL. **(For Information Only)**

FOIA Exemption 3 UCNI, Exemption 7f (Page 2-23 -- Plot Plan of RTC)

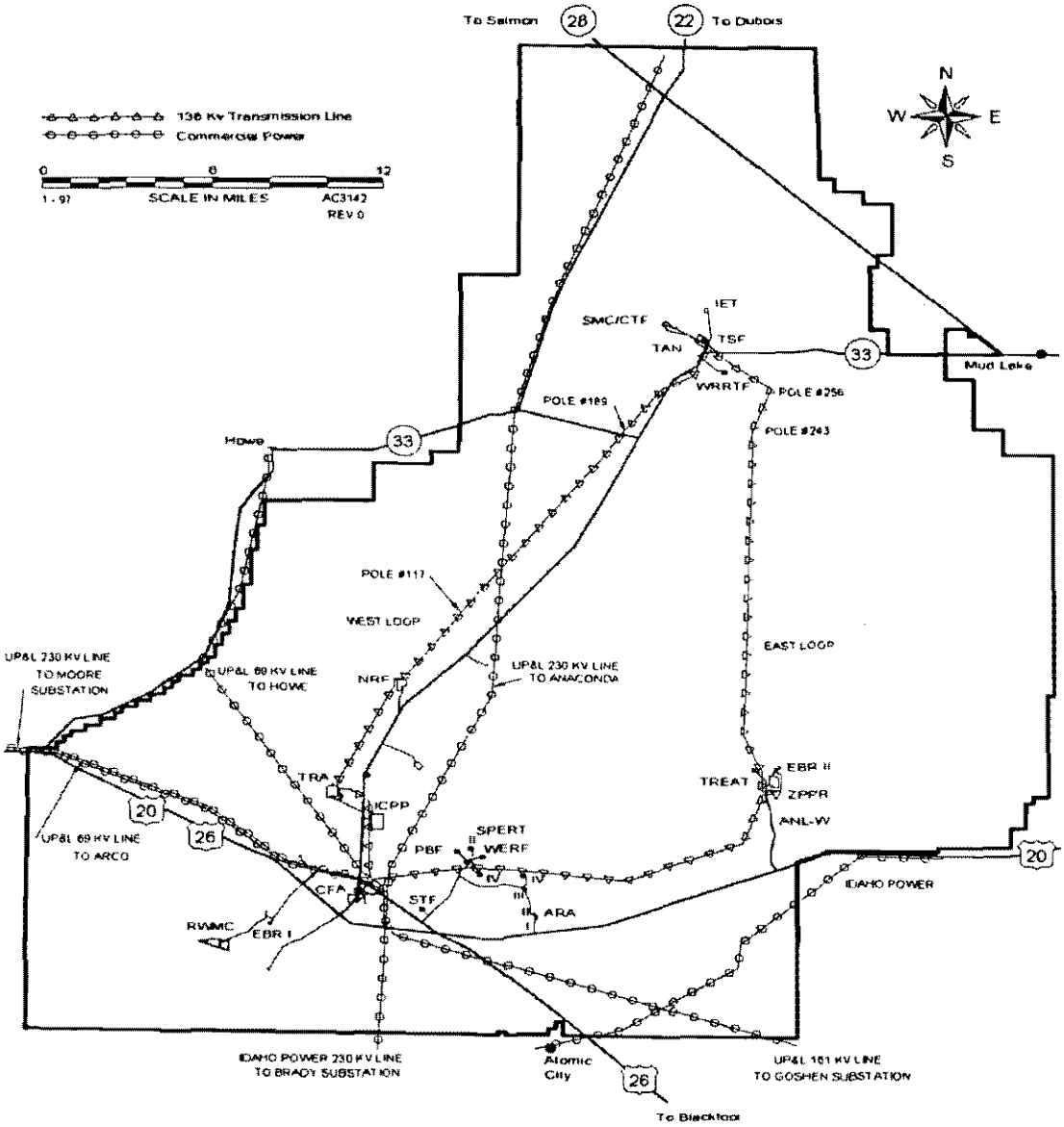

Figure 2.1-5. Power Lines within the INEEL. **(For Information Only)**

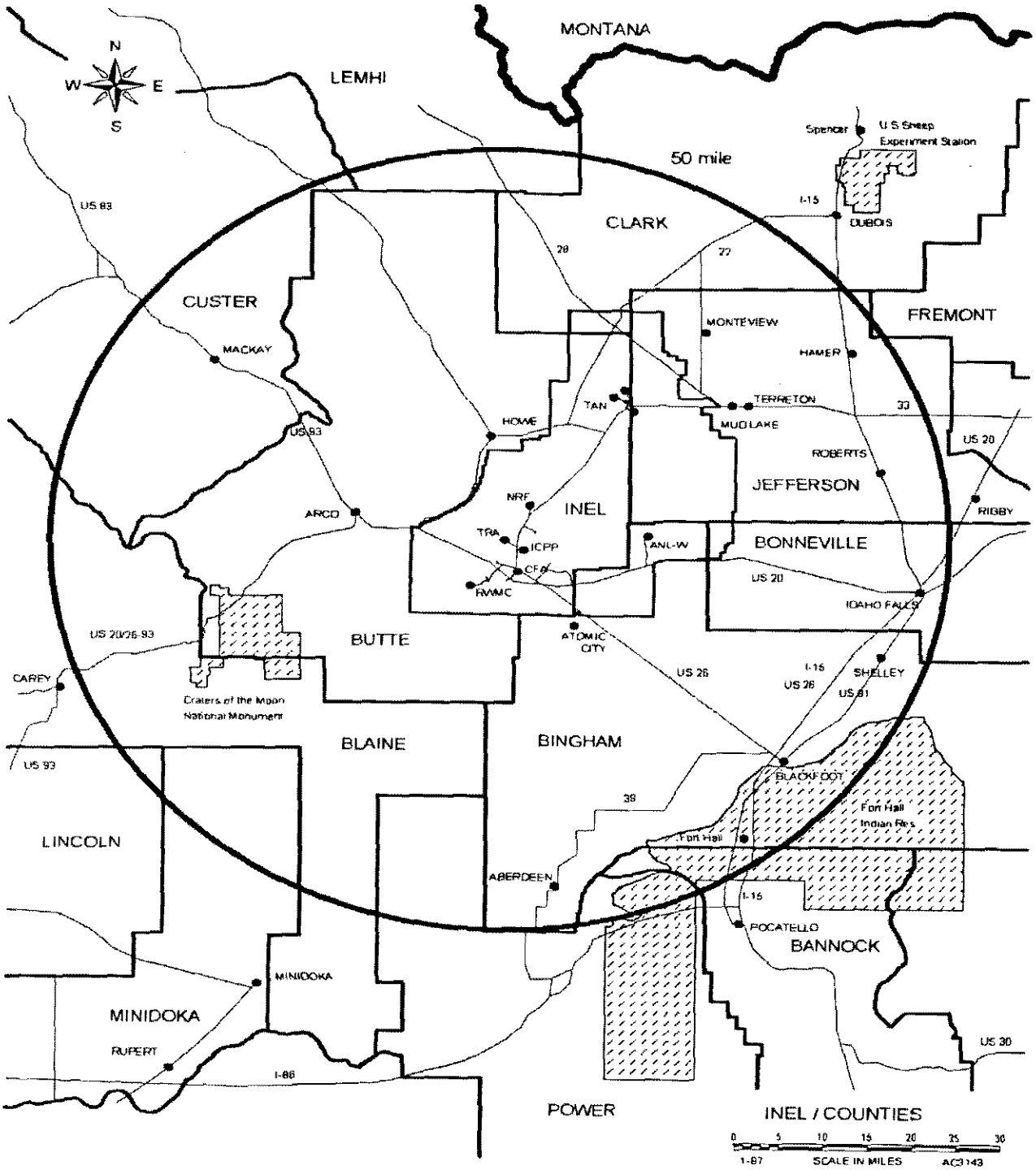

Figure 2.1-6. Counties Covered within a 50-mi (80-km) Radius of the Advanced Test Reactor. **(For Information Only)**

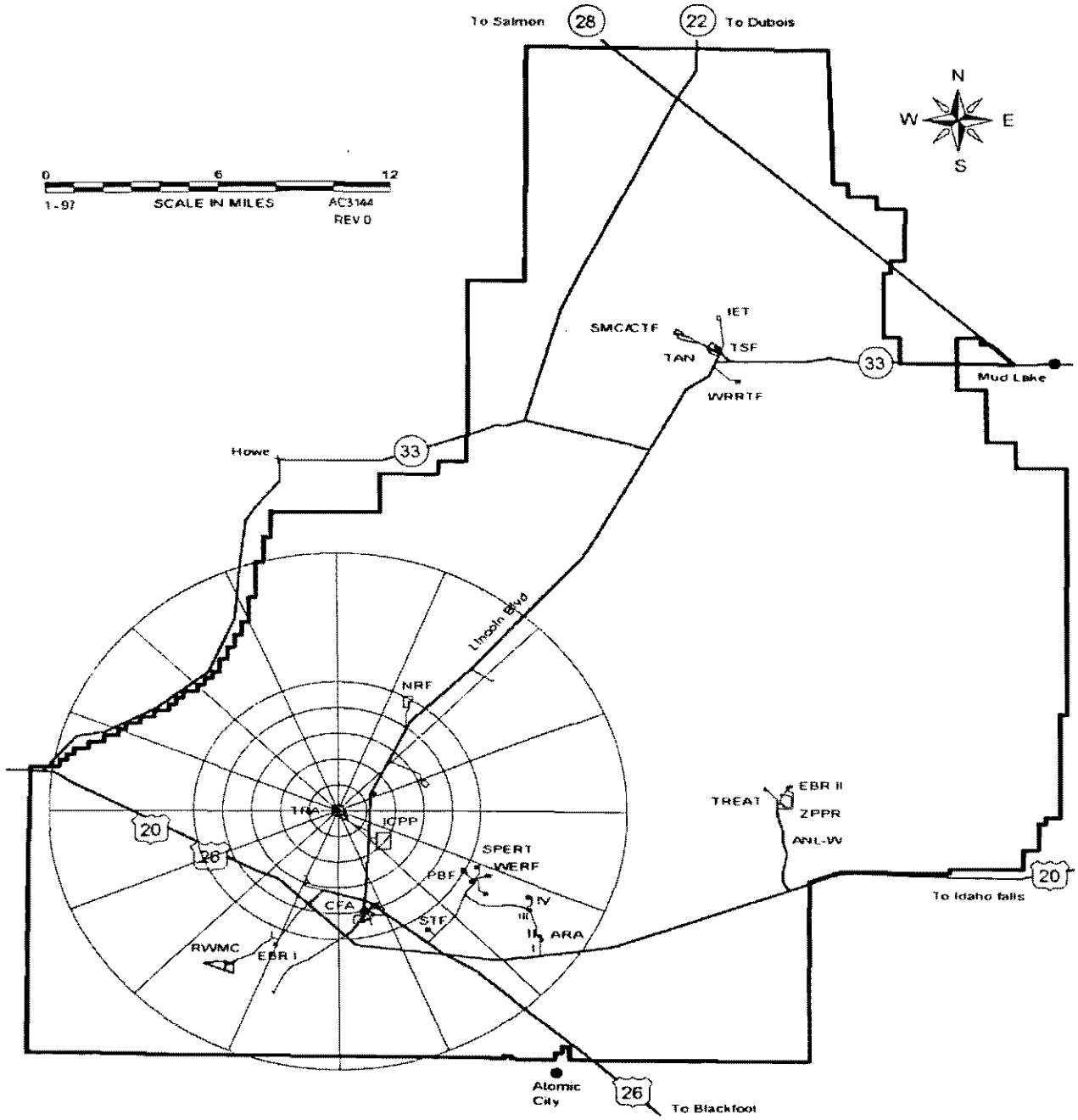

Figure 2.1-7. Sectors within a 10-mi (16-km) Radius of the Advanced Test Reactor.
(For Information Only)

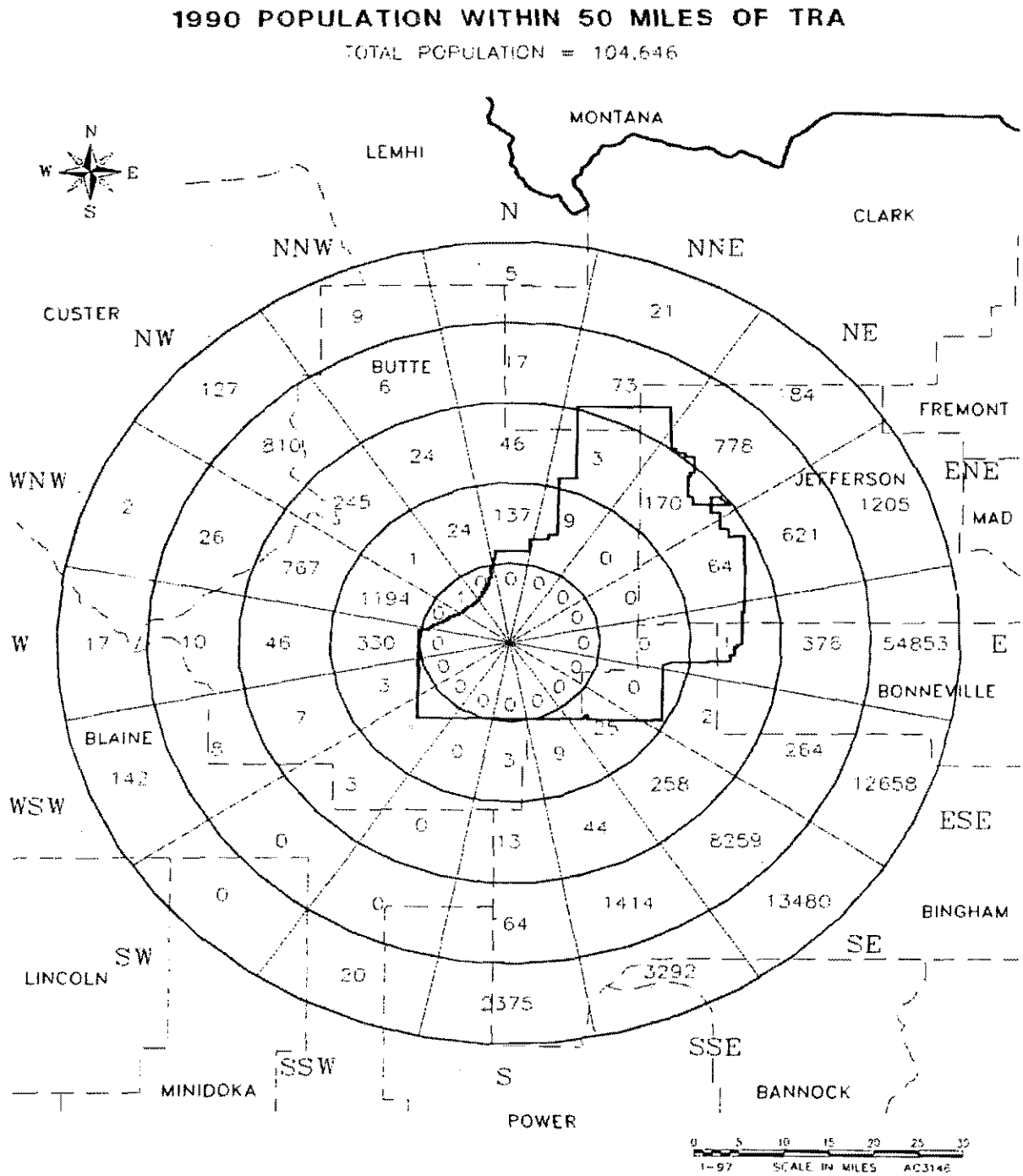

Figure 2.1-8. 1990 Population in Each Sector within a 50-mi (80-km) Radius of the Advanced Test Reactor. **(For Information Only)**

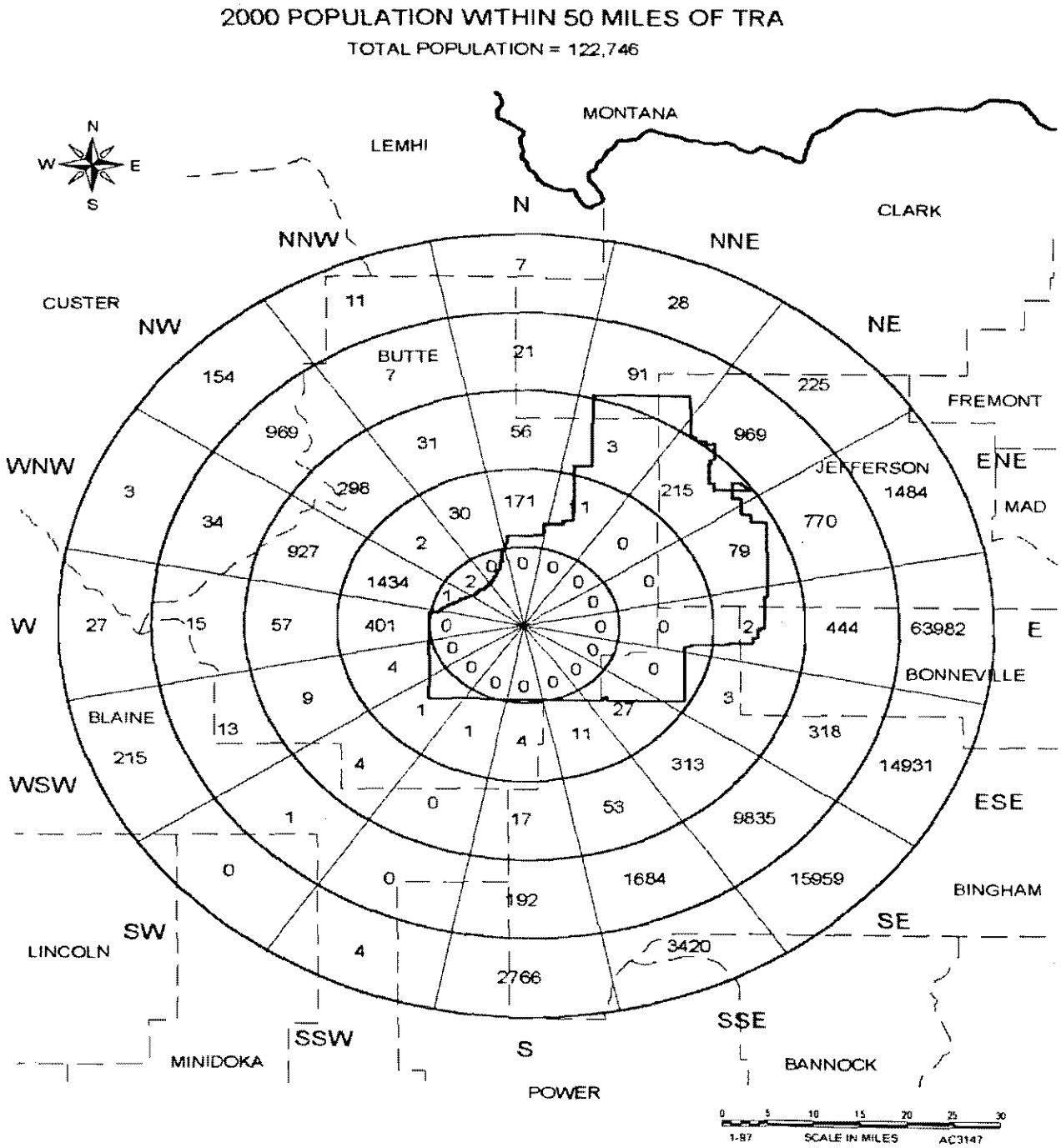

Figure 2.1-9. 2000 Population in Each Sector within a 50-mi (80-km) Radius of the Advanced Test Reactor. **(For Information Only)**

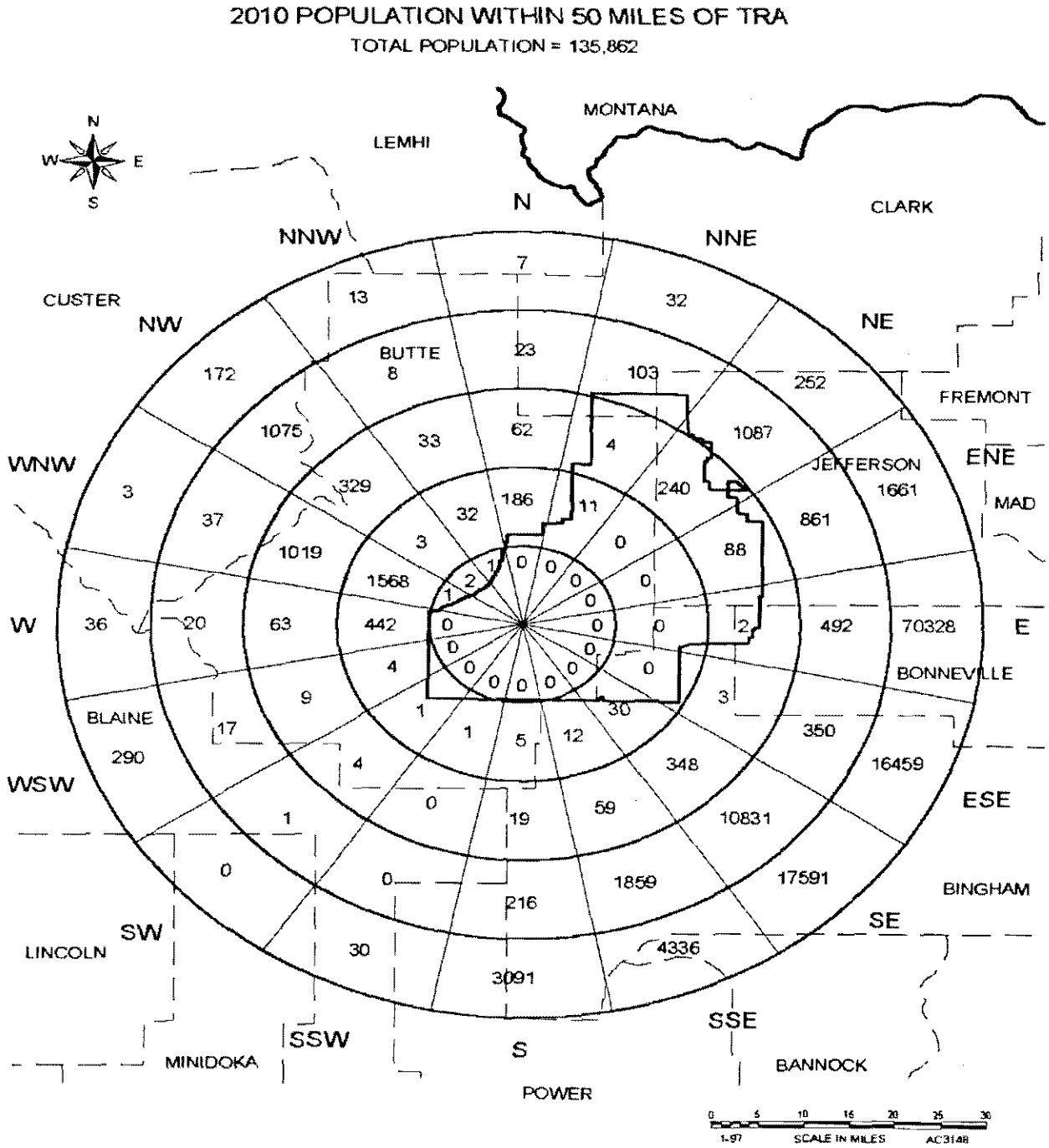

Figure 2.1-10. 2010 Population in Each Sector within a 50-mi (80-km) Radius of the Advanced Test Reactor. **(For Information Only)**

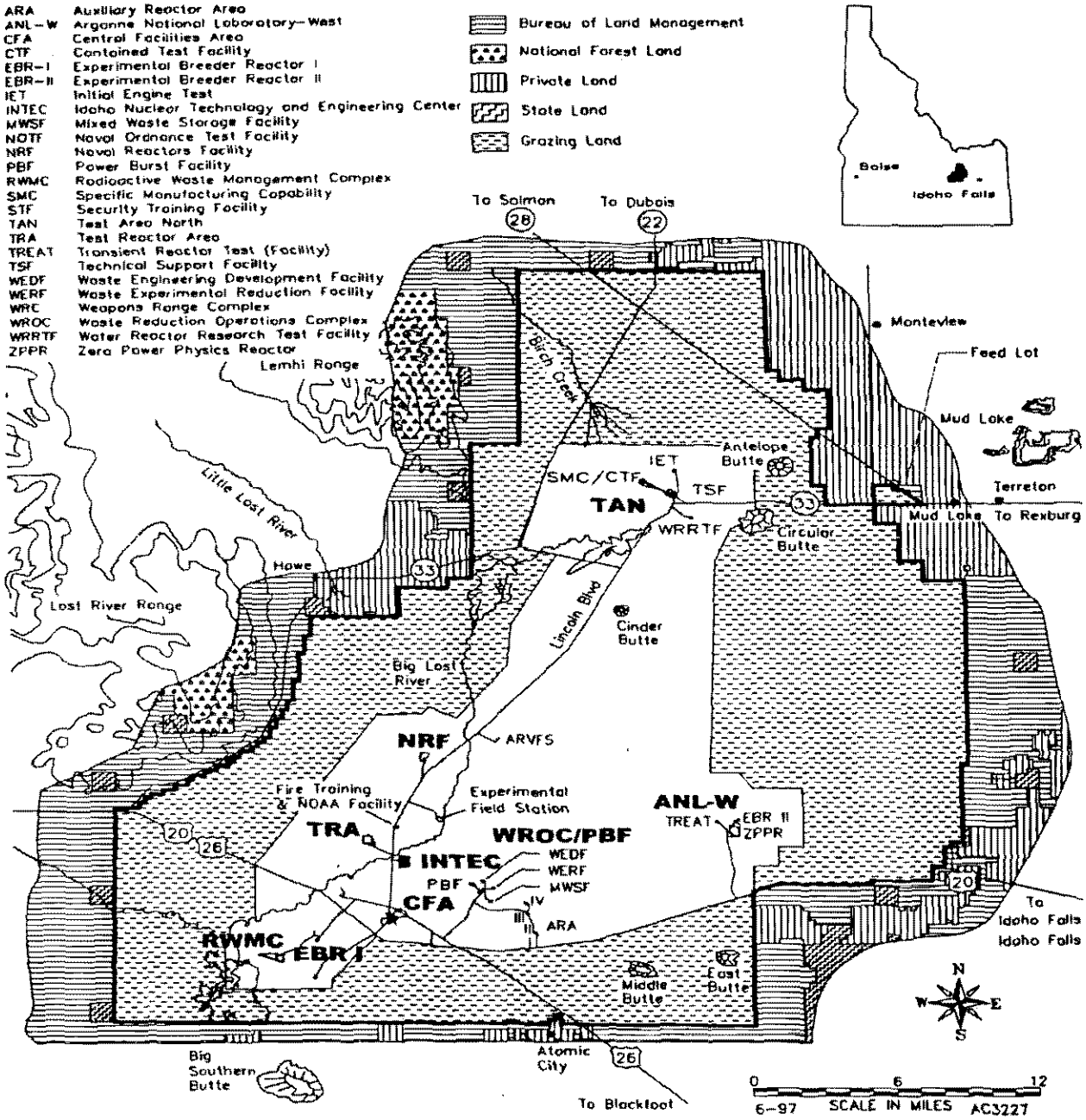

Figure 2.1-11. Geographical Features and Land Use at the INEEL and Vicinity. **(For Information Only)**

2.2 Nearby Industrial, Transportation, and Military Facilities

This section provides a description of nearby industrial, transportation, and military facilities that could potentially be at risk as a result of activities at TRA or that could impact TRA activities as a result of normal operations or accidents. The accidents at nearby facilities that could impact TRA operations are also described.

2.2.1 Location and Routes

The INEEL is a site where nuclear reactors and support facilities have been built and tested to demonstrate the applications of reactor technology, to conduct safety research, and to support defense programs. The existing facilities include a number of research and experimental reactors, waste management facilities, and a chemical processing plant (INTEC) that is a primary facility for processing radioactive waste and for storing radioactive waste and reactor fuel.

There are no commercial industries or activities, commercial or public transportation, or military installations or activities (except NRF) in the vicinity that could be impacted by, or cause impact to the ATR at the INEEL.

2.2.2 Descriptions

2.2.2.1 Description of Facilities. The INEEL facilities within 5 mi (8 km) of the ATR are described in Section 2.1.1.2.1. Detailed descriptions of these facilities are contained in the *Idaho National Engineering Laboratory Site Development Plan* (DOE-ID 1994) and are not repeated in this section.

2.2.2.2 INEEL Facility Evacuation Times. Estimated evacuation times for INEEL facilities within a 5 mi radius of the ATR are as follows: INTEC (at 1.8 mi) can be evacuated in 35 minutes following notification from the INEEL Warning Communications Center, NRF (at 4.9 mi) in 45 minutes, CFA (at 4.1 mi) in 10 minutes, and PBF (at 4.9 mi) in 6 minutes (Knudson and Matthews 1989).

An additional delay time would precede these evacuation times due to reporting and decision making processes. This delay time is estimated to be 20 minutes (Knudson and Matthews 1989). The TRA can be evacuated within 18 minutes following an evacuation notification (Knudson and Matthews 1989). Several facilities within TRA, such as the Advanced Test Reactor Critical Facility (ATRC), the TRA Hot Cell (TRAHC) and Nuclear Materials Inspection and Storage (NMIS) facilities, could experience accidents which might affect personnel at the ATR. Personnel within the TRA, however, are protected by a common evacuation plan. A serious TRA facility accident, if expected to affect the safety of personnel in other TRA facilities, would result in a common evacuation of the TRA.

2.2.2.3 Pipelines. There are no pipelines other than those within the boundaries of the specified nearby facilities or within 5 mi (8 km) of the ATR, and none traverse the INEEL.

2.2.2.4 Waterways. The Big Lost River has recently been classified as a navigable waterway; however, no effluents are discharged to that waterway by any INEEL operations (see Section 2.4.1.2).

2.2.2.5 Airports. Airports do not exist within 10 mi (16 km) of ATR. The nearest commercial airport is in Idaho Falls, which is 47 mi (75.6 km) east. There are infrequent security helicopter flights near TRA, and a helicopter landing pad is designated immediately south of the southern TRA perimeter fence.

2.2.2.6 Projections of Industrial Growth. The following projections are from the Idaho Power Company (IPC 1992):

- The 1990 to 2010 projected annual rate of Idaho population growth of 1.53% per year will produce a projected population gain of nearly 356,000 people over the two decades.
- Total employment in Idaho is expected to increase at an annual rate of 1.93% per year over the 1990 to 2010 period.

2.2.2.7 Conclusions. The locations and activities of all industrial, military, and transportation facilities that potentially could affect, or be affected by, ATR safety were considered. The data are adequate for identification of potential accidents to be evaluated in the next section.

2.2.3 Evaluation of Potential Accidents

The following subsections evaluate potential chemical and radiological accidents at adjacent INEEL facilities that may impact operations at ATR. The facilities considered are those located within approximately a 5-mi (8-km) radius of ATR as recommended for commercial reactors (NRC 1978). Of the facilities considered, only INTEC, presumably NRF, ATRC, NMIS, and TRAHC Facility present a potentially significant hazard at the ATR. Accident consequences and safety analysis for NRF are not included in the ATR UFSAR because they are classified information.

2.2.3.1 Potential Chemical Accident Impacts to ATR from Adjacent Facilities. Of the facility areas within a 5-mi (8-km) radius of the TRA, only INTEC has the potential for sufficient chemical material release to significantly impact ATR operations.

This section provides an assessment of potential impacts at ATR resulting from accidental chemical spill at INTEC based on Rood (1991). Spills postulated to occur at the Lincoln Boulevard turnoff to INTEC were also evaluated. Peak and time-weighted-average concentrations were calculated for receptors at the ATR facility and the TRA guard station at a height 1 m above ground level. Calculated concentrations were then compared to the 15-minute averaged threshold limit value—short term exposure limit (TLV-STEL) and the 30-minute averaged immediately dangerous to life and health (IDLH) limit. Several different methodologies were used to estimate source strength and dispersion.

Hydrofluoric acid, anhydrous ammonia, and chlorine were identified as chemicals that may have a potential impact at the ATR facility. These chemicals required evaluations using 50% meteorological conditions. Subsequent to the analysis by Rood, reprocessing of spent fuel at INTEC has been eliminated, thus eliminating the use of bulk quantities of hydrofluoric acid. Removal of bulk quantities of hydrofluoric acid from INTEC was completed in 1994. Therefore, hydrofluoric acid spills have been deleted from consideration. Using 50% meteorological conditions, chlorine did not exceed any of the standards. Nitric acid and ammonium hydroxide required reevaluation using an evaporative model, and the results showed minimal impact at the ATR facility and TRA guard station. The remaining chemicals were eliminated from consideration based on conservative screening calculations and, thus, were not considered to pose a potential threat to operation of the ATR facility. Table 2.2-1 shows the results of the final analysis for each chemical. The 15-minute average concentration was normalized to the TLV-STEL value, and the 30-minute average concentration was normalized to the IDLH value. These concentrations represent the air concentration at 1-m height. They do not represent the air concentration inside the ATR control room facility. The air concentration inside the control room will depend on the air exchange rate, where the air intake is located, and on any filtration equipment for the incoming air stream.

Fifteen-minute time-weighted-averaged concentrations of anhydrous ammonia exceeded TLV-STEL values for the cases considered. The IDLH value for hydrofluoric acid was also exceeded, indicating that even a limited spill presents significant health hazards to personnel in the immediate vicinity and perhaps at the ATR facility. Calculated concentrations of ammonium hydroxide, hexone, nitric acid, propane, gasoline, chlorine, and liquid nitrogen were all below the TLV-STEL value.

Subsequent to the analysis summarized above (Rood 1991), INTEC activities have changed. The anhydrous ammonia inventory and use is greatly decreased. A spill may still be possible, but the likelihood is considered low. In the event of a spill outside the INTEC fenced boundary, the CFA Emergency Action Center would be in control. The risk to TRA personnel is acceptably low.

2.2.3.2 *Potential Radiological Accident Impacts to ATR from Adjacent Facilities.*

Of the facility areas within a 5-mi (8-km) radius of the TRA location, only five have the potential for significant radioactive material releases: ATRC, NMIS, TRAHC, INTEC, and NRF.

The *Safety Analysis Report for the ATRC (TRA-670) (SAR-192)* (INEEL 2003b) considers several radiological events for ATRC, which include inadvertent canal draining and out-of-core criticality accident.

1. The inadvertent draining accident resulted in calculated radiation fields on the ATRC main floor on the order of 7 R/hr immediately after the accident. Recovery action could be implemented soon after a worst-case canal draining accident. Also the radiation levels are low enough that ATRC operators could evacuate without accumulating a significant dose.

2. The calculated dose for the hypothetical severe fuel damage accident is 100 rem in the ATRC bay. The receptor was assumed to evacuate the ATRC room within 7.5 minutes. This accident is beyond the design basis for ATRC.

The receptor location analyzed was within the ATRC, which is located within the ATR building. Doses and dose rates to ATR Facility personnel would be less than the values reported above due to additional shielding and distance. Complete details of the ATRC radiological accidents are available in the *Safety Analysis Report for the ATRC (TRA-670)* (SAR-192) (INEEL 2003b).

The *TRA Safety Analysis Report (SAR) Chapters for Nuclear Materials Inspection and Storage (NMIS) TRA-621 (SAR-154)* (INEEL 2003a) postulates two accidents in the facility hazard analysis that could result in radiological exposure to the co-located worker and one accident that could result in hazardous material exposure to the co-located worker. The postulated radiological accidents are a nuclear criticality and a facility fire; the postulated hazardous material release accident is a facility fire.

1. A nuclear criticality accident is postulated to involve nine ATR fuel elements with a prior irradiation that results in a dose rate of 200 mrem/hr on contact. The calculated dose for a co-located worker at 200 m is 1.1 rem total effective dose equivalent (TEDE) assuming an unfiltered release through the building ventilation system.

2. A fire in the facility is postulated to damage 132 ATR fuel elements with a prior irradiation that results in a dose rate of 200 mrem/hr on contact. The calculated dose for a co-located worker at 200 m is 16 rem (TEDE) assuming an unfiltered release through the building ventilation system.

3. A fire in the facility is also postulated to release Cd for a fire in the storage vault or Pb for a fire in the non-vault areas. The exposure to the co-located worker at 100 m is calculated to be 2.45 mg/m^3 for Cd, which exceeds the Emergency Response Planning Guidelines (American Industrial Hygiene Association) level 1 (ERPG-1), and 11.9 mg/m^3 for Pb, which exceeds ERPG-2. The calculations do not consider the effect of the building structure on the rate or amount of material released.

The receptor location for the criticality and the radiological consequences of a fire is 200 m, the receptor location for the hazardous material release due to a facility fire is 100 m. A facility worker immediately outside the ATR facility is over 300 m away from the NMIS. The exposure at 100 and 200 m envelope the exposure to ATR facility workers. Details of the radiological and hazardous material release accident analysis are available in the NMIS SAR (INEEL 2003a).

The *Documented Safety Analysis for the TRA Hot Cell Facility (TRA-632)* (SAR-204) postulated two for further quantitative analysis (INEEL 2002). Descriptions of the cases follow:

Case 1: Unfiltered release of radioactive material from a representative ATR test train because of a fire. The release is due to the failure of the building, walls, windows, and doors from the heat of the fire.

Case 2: Direct dose from a cobalt source because of a shielding failure. The source becomes unshielded because of a shielding window leak or some other unspecified failure.

The receptor location cases included:

- 100 m from the building (facility boundary)
- General public (about 10.6 km)

The ATR is located between these two receptor locations, and the dose at ATR will be bounded by the dose at 100 m.

The results of the TRAHC analyses indicate for Case 1, an unmitigated TEDE of 18.1 rem at 100 m, using 4.5 m/s (10.07 mph) wind speed. Analysis for Case 2 indicates a dose of 77 rem in a 24 hr period.

The radiological impact for ATR facility personnel from credible radiological accidents would be much less than the unmitigated 18.1 rem (for 4.5 m/s wind speed) because controls have been implemented to reduce the heat and duration of a postulated fire and to ensure that various radiation shields remain functional (e.g. windows).

INTEC has numerous facilities, which contain radioactive material. Accidents involving several of these facilities could have releases that could result in a radiological impact at TRA. The accident with the worse case radiological consequences at TRA would be the Fuel Storage Area (FSA). Numerous events were analyzed for the FSA and documented in SAR-113 (2003). The postulated accidents selected for quantitative analysis are:

1. HEPA filter failure
2. Total fuel cladding failure
3. Spent resin release
4. Inadvertent nuclear criticality
5. FSA aboveground structure failure
6. FSA pool drain.

The pool drain event produced the most significant consequences. The event was analyzed with a concurrent loss of all cladding with a concurrent criticality. The distance between INTEC and TRA is 2.4 km. The event was analyzed using Class F meteorology with a wind speed of 2 m/sec. The radiological impact at TRA was 27 rem. Due to the high value of the unmitigated dose the structure at INTEC was classified as safety class.

Accident consequences and safety analysis for NRF are not included in the ATR UFSAR because they are classified information.

2.2.3.3 Conclusions. The conservatively calculated dose at ATR from the worst-case adjacent facility accident is significantly less than the calculated doses from ATR accidents as described in Chapter 15. The risk of adjacent facility accidents is acceptable.

Table 2.2-1. Summary of the air dispersion calculations for the 1 m level at the ATR facility for an INTEC chemical spill.

	Air Concentration at ATR		
Chemical	Peak (mg/m^3)	15 min normal[a]	30 min normal[b]
Ammonium hydroxide	2.15E+01	7.96E-01	6.14E-02
Hexone	1.88E+03	9.70E-01	1.18E-02
Liquid nitrogen	1.30E+04	—	6.90E-02
Nitric acid	3.30E+00	3.30E-01	1.27E-02
Propane	3.10E+03	2.66E-01	6.66E-03
Gasoline	3.42E+03	5.88E-01	1.76E-01
Anhydrous ammonia (1,640 kg)	8.37E+02	4.79E+01	1.65E-01
Anhydrous ammonia (65,500 kg, 50% meteorological conditions)	3.16E+02	5.16E+00	1.77E-01
Chlorine (50% meteorological conditions)	1.32E+00	1.74E-02	2.90E-04

a. The 15-min average concentration normalized to the TLV-STEL value (TLV-STEL: threshold limit value - short-term exposure limit).

b. The 30-min average concentration normalized to the IDLH value (IDLH: immediately dangerous to life and health).

2.3 Meteorology

2.3.1 Regional Climatology

The climate of the INEEL, located in the Eastern Snake River Plain (ESRP), is affected by its altitude above sea level, its latitude, and its intermountain setting. The regional aspects of this climate are described in the following subsections.

2.3.1.1 General Climate. Air masses crossing the ESRP have first traversed a mountain barrier and precipitated a large percentage of inherent moisture. Therefore, annual rainfall at the INEEL is light, and the region is classified as arid to semiarid. The relatively dry air and infrequent low clouds permit intense solar heating of the surface during the day and rapid radiational cooling at night. These factors combine to give a large diurnal range of temperature near the ground.

The moderating influence of the Pacific Ocean produces a climate that is usually warmer in the winter and cooler in summer than is found at locations with similar latitudes in the more continental regions of the U.S. to the east of the Continental Divide. The Centennial and Beaverhead Mountain Ranges are an effective barrier to movement of most of the intensely cold winter air masses passing to the south out of Canada toward the ESRP. Occasionally, however, cold air spills over the mountains and is trapped in the ESRP. The INEEL then experiences below-normal temperatures for periods lasting 7 to 10 days.

The mountains bordering the ESRP also channel the prevailing west winds into a southwesterly flow because of the northeast-southwest orientation of the ESRP between the bordering mountain ranges. The second most frequent wind direction is from the northeast.

A summary of recent climatological data from 14 National Climatic Data Center (NCDC) stations on and surrounding the INEEL (Clawson, Start, and Ricks 1989) is given in Table 2.3-1. The recent data supplement, but do not replace, an extensive early climatological report (Yanskey, Markee, and Richter 1966). The data have been compiled for a common time period (January 1981 through July 1985) to allow a climatological comparison of these sites. The data include average annual statistics for air temperature and precipitation. Normal annual air temperatures and precipitation levels, which are an average of the 30-year period of 1951-1980, are also provided when they are available. Data listed under the heading Upper Snake River Plains Division are an average of all NCDC stations in the ESRP and represent the average regional climate.

2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases. The minimum meteorological conditions for design and operating bases of the ATR are provided in the INEEL Architectural Engineering Standards (DOE-ID 2002). These criteria are based on regional and local conditions specifically applicable to the INEEL.

2.3.1.3 Conclusions. The cited references provide all of the general climate data needed for safety analyses.

2.3.2 Local Meteorology

INEEL meteorology is described in detail in Clawson, Start, and Ricks (1989) and is summarized in the following subsections, including normal and extreme values of selected meteorological parameters and an analysis of the potential influence of the ATR on local meteorology.

2.3.2.1 Normal and Extreme Values of Meteorological Parameters. Clawson, Start, and Ricks (1989) provides normal and extreme values for several climatic topics. Brief summaries follow.

Large ranges in daily surface air temperatures can be expected at any location on the INEEL. The average daily air temperature over a 39-year period of record at CFA ranges from a low of 10°F (-12.2°C) on January 2 to a high of 70°F (21.1°C) on several days in late July.

Detailed studies of the air temperature structure from the surface to 250 ft (76.2 m) have been conducted for several years using air temperatures measured on tall towers at various locations on the INEEL. The GRD3 station, approximately 1 mi (1.6 km) northeast of TRA, is representative of TRA. For GRD3, from September 1980 through May 1983, the data indicate an average annual inversion rate of 45.9% (percent of time), and a lapse rate of 54.1%.

Precipitation has been recorded since 1950 at CFA. Although the total annual precipitation is light, precipitation can be expected in any month of the year, and there are several months when no precipitation was recorded. A pronounced precipitation peak occurs in May and June. The average precipitation for each of these months at CFA is 1.2 in. (3.0 cm). Precipitation observed in every other month is approximately one-half or less of this amount. The average annual total precipitation is 8.71 in. (22.1 cm) at CFA.

Atmospheric pressure was recorded nearly continuously at CFA from February 1950 to August 1964. A mercurial barometer located at a height of 4,937.57 ft (1,504.97 m) msl was used to measure pressure during this time. The data indicate an average station pressure of 25.06 in. (63.65 cm) of mercury (Hg). Monthly averages range from 24.98 in. (63.45 cm) Hg in April to 25.13 in. (63.83 cm) Hg in December. The extreme lowest and highest atmospheric pressures recorded were 24.26 in. (61.62 cm) Hg and 25.69 in. (65.25 cm) Hg.

Moisture content of the air has been continuously monitored primarily at CFA since 1950. Originally, atmospheric moisture was recorded as hourly wet bulb temperatures. After discontinuing the hourly observations, atmospheric moisture was recorded as relative humidity using a hydrothermograph. Atmospheric humidity has more recently been recorded as dew point temperature with a chilled mirror. Measurements using each of these variables are detailed in Clawson, Start, and Ricks (1989).

The wind pattern over the INEEL can be complex. The orientation of the bordering mountain ranges, as well as the general orientation of the ESRP, play an important part in determining the wind regime. The INEEL is in the belt of prevailing westerly winds that are normally channeled within the ESRP, producing west-southwest or southwest winds. See Clawson, Start, and Ricks (1989) for wind roses for GRD3 at the 32.8-ft (20-m) and the 200-ft (61-m) levels.

The state of the ground was observed daily at CFA from 1950 through 1963. The percent of observations in each category indicates that the ground is usually snow-covered in December, January, and February. The ground is usually dry from mid-April through mid-November. March and the first half of November are transition months when the surface is either frozen or covered less than 50% with snow. For a significant portion of January and February, the ground is covered with blowing or drifting snow. The annual summary indicates that the ground is dry more than 50% of the time, and that it is either moist, frozen, or completely covered with ice, snow, or melting snow about 25% of the time.

The sun rises as early as 4:51 and sets as late as 20:17 hours Mountain Standard Time (MST). It also rises as late as 8:06 and sets as early as 16:56 hours MST. The longest day of the year is 15 hours, 26 minutes, and the shortest day of the year is 8 hours, 56 minutes. Regional solar radiation data were obtained during 1953-1975 from more than 200 observation stations. The average daily total global radiation on a horizontal surface in July is expected to be 681 ly/day[a], and drops to 131 ly/day[a] in December. The average total daily global radiation value for the entire year is 406 ly/day.[a] The average daily opaque sky cover for the INEEL, based on regional observations, is five-tenths of sky.

Several other types of meteorological phenomena occur at the INEEL. Dew formation and dissipation coincide with the formation and dissipation of the nocturnal temperature inversion. Thunderstorms have been observed during every month of the year, although seldom in November through February. Thunderstorms over the INEEL are usually much less severe than those in the nearby mountains because of high cloud-base altitudes. Hence, the precipitation from many thunderstorms evaporates before reaching the ground. Thunderstorms at the INEEL may be accompanied by micro bursts (i.e., strong, localized, gusty winds). Lightning strikes at the INEEL are monitored by the U.S. Bureau of Land Management; however, historical data for the INEEL have not been compiled. Although the number of lightning strikes on the INEEL does not appear to be high, the lack of natural targets and the poor conductivity of the dry desert soil and underlying lava rock cause manufactured structures to be susceptible to lightning strikes. Small hail, usually smaller than 1/4-in. (0.6-cm) in diameter, occasionally accompanies thunderstorms. A study of airborne dust showed dust concentrations ranged from a low of 14.1 $\mu g/m^3$ over total snow cover to a high of 772 $\mu g/m^3$ during the summer.

The National Oceanic and Atmospheric Administration (NOAA) records indicate that five funnel clouds and no tornadoes have been sighted within the INEEL boundaries since 1950. The calculated return period for a tornado at the INEEL with wind speeds exceeding 120 mph is 1×10^6 years (Coats and Murray 1985).

The monthly average wind speed, highest hourly average wind speed, and corresponding wind directions at CFA are provided in Clawson, Start, and Ricks (1989). The highest hourly average wind speed for CFA is 67 mph. The highest average wind speeds occur in April, and they are from the west-southwest and southwest. The maximum instantaneous gust recorded at CFA was 78 mph.

The topography of the INEEL influences atmospheric transport. Tower measurements at GRD3 have shown a 180-degree direction shear between levels 20 ft (6.1 m) and 50 ft (15.2 m) above the ground. This strong shear has been observed many times during the early morning hours; it is attributed to the slope of the terrain in this area being opposite the general slope of the terrain over the ESRP. Solar energy incident upon local topographic features and thermal energy radiated from the earth provide an energy source of secondary importance to the INEEL wind field. The combined influences of global synoptic-scale forcing, topographic channeling, and radiative heat energy exchanges near the INEEL result in several characteristic trajectory patterns over the INEEL.

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[a]. The unit of measurement is the langley (1 g-cal/cm^2).

Several fundamental atmospheric diffusion experiments have been conducted at the INEEL (Clawson, Start, and Ricks 1989; Yanskey, Markee, and Richter 1966). These include experiments that studied the characteristics of (a) effluent material released from a 148-ft (45-m) elevated source, (b) effluent material released at ground level, (c) effluent material released into an ultrastable, low wind speed atmosphere, (d) effluent material at extended regional and continental scales, and (e) effluent material in recirculating and stagnating unsteady atmospheric conditions.

Annual average concentration isopleths for normalized INTEC emissions using meteorological data from 1974 through 1983 generally exhibit a northeast-southwest distribution of the concentration isopleths. Areas of higher concentration are generally confined within the INEEL boundaries. A secondary concentration lobe is exhibited in the area between Blackfoot and Aberdeen. Similarities are also known to exist in the concentration patterns for effluent releases from sites located in the vicinity of the INTEC, including the ATR.

2.3.2.1.1 Natural Phenomena Hazard Studies—In 1985, the ATR building reactor room and control room (including adjacent office structure) were evaluated for extreme wind/tornado hazards. Findings indicated that the masonry block walls, liner plates, and secondary bracing have low stress levels for all wind speeds considered. Minimal to moderate damage of the ATR facility will be experienced for probabilities of $p = 1 \times 10^{-6}$ per year wind speeds (115 mph). Moderate to severe damage will be sustained at $p = 1 \times 10^{-7}$ per year wind speeds (126 mph straight wind, 167 mph tornado). The main wind force resisting frame members were (a) capable of resisting straight wind speeds of up to 154 mph ($p < 1 \times 10^{-7}$ per year) and (b) capable of resisting 176 mph tornado wind speed ($p < 1 \times 10^{-7}$ per year).

The study concluded that the ATR reactor building reactor room and control room, including adjacent office structure, have capabilities of resisting wind speeds greater than established hazard values (Arendts 1989).

2.3.2.2 Potential Influence of the Plant and its Facilities on Local Meteorology. Any significant influence on local meteorology from ATR structures or operations would be expected to be associated with the cooling towers. Although technical studies were not available for analyses, it is expected that potential impacts would include plume visibility, fogging, icing, and drift deposition. ATR is located a sufficient distance from public lands and highways to preclude any significant impacts from fogging, icing, or drift deposition. Interactions between cooling tower plumes and other industrial phases should be negligible because of the distances between facilities and because the direction of ATR to plume generators such as INTEC is perpendicular to prevailing wind paths.

2.3.2.3 Conclusions. The ATR has negligible influence on local meteorology. The cited references provide all of the local meteorology data needed for safety analyses.

2.3.3 Onsite Meteorological Measurements Program

The onsite meteorological measurements program at TRA is conducted by NOAA. The measurement station is a "Type 3 Station," using a 49-ft (15-m) tower, and the basic complement of instruments is as follows:

- High-volume air sampler control/status
- Wind speed at 49 ft (15 m)

- Wind direction at 49 ft (15 m)
- Nuclear radiation
- Temperature at 6.5 ft (2 m)
- Temperature at 49 ft (15 m).

The remote terminal unit (RTU) gathers data from individual transducers, converts the outputs to engineering units, and provides the capability to transmit the output data, on request, to a data acquisition system. The RTU is capable of storing output records for later transmission. During communications or data acquisition system failures, the RTU is capable of storing five days of data. The TRA RTU outputs a record every six minutes.

2.3.3.1 Conclusions. The meteorological measurements equipment at TRA records data in a manner suitable for updating existing long-term information.

2.3.4 Diffusion Estimates

The first series of reports using meteorological data acquired on-site was published during the time period from 1958 to 1960 (IDO-12003, IDO-12004, and IDO-12015). A major addition to that original body of climatological literature was the first edition of the "Climatography of the National Reactor Testing Station," IDO-12048 (Yanskey, Markee, and Richter 1966). One section of IDO-12048 describes the practical techniques of atmospheric diffusion and deposition calculations. The coefficients for computation were derived from field trials with various types of meteorological tracers. Other sections deal with extreme values and atmospheric transport.

The second edition of the INEEL climatography, DOE/ID-12118 (Clawson, Start, and Ricks 1989) adds extensively to the first edition but is not a replacement. The second edition includes updates on dispersion, transport, and extreme values.

2.3.4.1 Conclusions. Both the short-term and long-term meteorological data and calculational techniques at the INEEL meet the recommendations of the NRC Regulatory Guide 1.70 (NRC 1978).

Table 2.3-1. Compilation of National Climatic Data Center temperature and precipitation records for observation stations on and surrounding the INEEL from the years 1981-1985.[a]

National Climate Data Center ID	Elevation (ft msl)	Absolute maximum temperature (°F)	Absolute minimum temperature (°F)	Annual average (normal) temperature (°F)	Annual average (normal) cooling degree days (DGD/y)	Annual average (normal) heating degree days (DGD/y)	Absolute maximum daily precipitation (in.)	Annual average snowfall (in.)	Annual average (normal) precipitation (in.)
Aberdeen Exp. Station	4405	98	-38[b]	42.9[b](44.8)	247[b](269)	7975[b](7633)	0.76[b]	N/A	10.23[b](8.81)
Arco 3 SW	5328	97	-45[c]	40.9[e]	217[c](204)	7286[b](8613)	1.64[d]	N/A	14.35[c]
Blackfoot 2 SSW	4487	97	-24[c]	43.6[e]	409[e]	6144[b]	1.36[e]	N/A	10.20[d](10.70)
Craters of the Moon	5897	95	-37	42.4[d]	355[d]	7751	2.00[d]	121.9[b]	17.71[b]
Dubois Exp. Station	5450	98	-25	42.4(42.7)	335(301)	8250(8424)	1.25	50.9	16.05(11.74)
Hamer 4 NW	4791	98	-46[b]	38.7[e](42.5)	116[d](262)	8352[b](8441)	0.81[d]	31.8	11.49[b](8.46)
Howe	4820	97[b]	-38[b]	43.0[d]	244[d]	7890	2.02[d]	19.8	10.16[d](8.85)
Idaho Falls 2 ESE	4765	98[b]	-34[c]	45.0[d]	393[c]	7480[b]	1.13[d]	39.1[d]	16.39[c]
Idaho Falls FAA AP	4730	100	-38	42.8(43.8)	309(288)	8094(7995)	0.97	43.5	12.89(9.77)
Idaho Falls 46 W	4938	99	-47	40.7(42.0)	272(253)	8840(8626)	1.51	28.3	10.31(8.62)
Minidoka 10 WNW	4290	105[d]	-30[d]	40.3[e]	343[d]	7551[d]	1.95[e]	N/A	10.73[d]
Rexburg Ricks College	4920	97[b]	-31[c]	43.7[c]	310[c]	7948(7443)	1.40[d]	75.1	17.39[c]
Richfield	4306	100	-28	44.4(45.4)	369(332)	7546(8398)	1.36	39.5	14.51(11.09)
St. Anthony 1 WNW	4950	95	-39	41.6[c](42.2)	104[c](146)	8297[b]	1.14[c]	N/A	14.50[b]
Upper Snake River Plain	N/A	N/A	N/A	43.1(43.9)	N/A(N/A)	N/A(N/A)	N/A	N/A	14.62(11.21)

a. Taken from Clawson, Start, and Ricks 1989. N/A indicates that the data are not available for a given station. Normals are average data from the 30-year period of 1951 to 1980.

b. Data are missing for all or part of 1 year.

c. Data are missing for all or part of 2 years.

d. Data are missing for all of part of 3 years.

e. Data are missing for all or part of 4 years.

2.4 Hydrologic Engineering

This section provides descriptions of the surface and groundwater hydrology of the INEEL and surrounding region. Surface water discussions include general flood history and flood design considerations, the probable maximum flood (PMF) and potential dam failures, and other local surface water considerations that could affect designs. Groundwater discussions include general descriptions of the local and regional groundwater environment, including water use and other groundwater considerations that could affect designs.

2.4.1 Hydrologic Description

2.4.1.1 Site and Facilities. The area around TRA is relatively flat with elevations ranging from approximately 4,930 ft (1,503 m) mean sea level (msl) in the southwest corner to approximately 4,918 ft (1,499 m) msl in the northeast corner. The Big Lost River channel is 4,480 ft (1,365 m) from the southeast corner of the TRA security fence and 4,120 ft (1,256 m) from the southeast corner of the warm waste pond. Elevation of the river bed in the TRA area is 4,919 ft (1,499.3 m) msl.

ATR is located at the TRA Facility area. The main floor of the reactor building is at an elevation of 4,926.26 ft (1,501.5 m) msl. Elevations of support facilities such as the diesel oil pump house 4,921.0 ft. (1,499.9 m) msl and the raw water/fire water pump house 4,920.5 ft (1,499.8 m) are lower.

2.4.1.2 Hydrosphere. Most of the INEEL is located in the Pioneer Basin into which three tributaries drain: the Big Lost River, Little Lost River, and Birch Creek (Figure 2.1-3). These tributaries receive water from mountain watersheds located to the north and northwest of the INEEL. Stream flows are often depleted before reaching the INEEL by irrigation diversions and infiltration losses along the river. When water does flow onto the INEEL, it either evaporates or infiltrates into the ground because the Pioneer Basin is a closed topographic depression.

In addition to the infrequent flow of the three tributaries, INEEL surface water hydrology includes local runoff from precipitation and snowmelt. The INEEL has very little surface water except in years of heavy runoff from snowmelt or rainfall. Surface water is not used at the INEEL. The Big Lost River is the major surface water feature on the INEEL and is the only tributary of concern at TRA. At its nearest point, the Big Lost River channel is 4,480 ft (1,365.5 m) from the southeast corner of the TRA security fence with an elevation of 4,919 ft (1,499.3 m) msl. Big Lost River waters are impounded and regulated by the Mackay Dam located approximately 4 mi northwest of Mackay, Idaho. Mackay Dam is an earthen structure, 80 ft (24.4 m) high and approximately 1,430 ft (435 m) long at dam crest, which was completed in 1917 with a storage capacity of 44,500 acre-ft (54,890,000 m^3) and a surface area of 1,241 acres (502 ha) (Van Haaften, Koslow, and Naretto 1984). It is classified as a high hazard dam by the State of Idaho, based on the Army Corps of Engineers guidelines for safety inspection. An ungated overflow spillway with a weir length of 75 ft (23 m) at elevation 6,066.5 ft (1,849.1 m) msl is located near the west abutment of the dam. The spillway is rated for a discharge of 3,250 cfs (92.0 m^3/s) with 4 ft (1.2 m) of freeboard on the dam. The total discharge capacity of Mackay Dam is less than 10,000 cfs (283.2 m^3/s).

Mackay Dam was built before current seismic or hydrologic design criteria were in effect. Although the structure's ability to withstand further seismic activity is unknown, the performance of the structure following the Borah Peak earthquake demonstrated the stability of the embankment during moderate earthquake ground motion (estimated to be 0.22 to 0.3 g peak ground acceleration at the dam site). Some minor turbidity was observed for approximately 1 hour after the earthquake. A walkdown of the dam after the earthquake showed no damage to the internal structure.

A number of irrigation diversions exist between the Mackay Dam and the Snake River Plain. Upon leaving the dam, waters of the Big Lost River flow southeastward through the Big Lost River Basin past Arco and onto the ESRP. When flow in the Big Lost River reaches the INEEL, it is either diverted at the INEEL diversion dam (Figure 2.1-3) or flows northward across the INEEL in a shallow, gravel-filled channel to its terminus at the Lost River Sinks, where its flow is lost to evaporation and infiltration.

The INEEL diversion dam, constructed in 1958 to divert high runoff flows from the INEEL facilities, consists of a diversion dam, diversion channel, two 6-ft (1.8-m) diameter gated culverts, three dikes, four spreading areas, and two interconnecting channels. The diversion channel is capable of carrying 7,200 cfs (203.9 m^3/s) from the Big Lost River into the spreading areas. Two low swales located southwest of the main channel will carry an additional 2,100 cfs (59.5 m^3/s) for a combined maximum diversion channel of 9,300 cfs (263 m^3/s) (Bennett 1986). The total capacity of the spreading areas is 18,200 ac-ft (22,450,000 m^3) at elevation 5,040 ft (1,536 m) msl and 58,000 ac-ft (71,542,000 m^3) at elevation 5,050 ft (1,539 m) msl (McKinney 1985; Koslow and Van Haaften 1986). Table 2.4-1 lists Mackay Dam (Reservoir) and INEEL diversion area characteristics (Koslow and Van Haaften 1986).

The distance of the TRA facility to the Big Lost River channel and the high infiltration rates of the surface alluvium and basalt suggest it is unlikely that flood waters from the Big Lost River will reach the TRA facility (Doornbos et al. 1991).

The Little Lost River drains slopes of the Lemhi and Lost River ranges. Flow in the Little Lost River is diverted for irrigation north of Howe, Idaho, and does not reach the INEEL.

Springs below Gilmore Summit in the Beaverhead Mountains and drainage from the surrounding basin are the source for Birch Creek. Flowing in a southwesterly direction between the Lemhi and Bitterroot ranges, the water in the creek is diverted north of the INEEL for irrigation and hydropower during the summer months. In the winter months when not used for irrigation, the water is returned to a manufactured channel on the INEEL 4 mi (6.4 km) north of TAN where it infiltrates into channel gravels and recharges the aquifer.

Three waste disposal ponds are at the TRA facility (Figure 2.1-4). These include the sewage treatment lagoon, evaporation pond, and cold waste pond. The sewage treatment lagoons consist of two cells: the northern cell is approximately 185 ft long × 30 ft (56.4 m × 9.1 m) wide, and the southern cell is approximately 185 ft long × 45 ft (56.4 m × 13.7 m) wide. Elevation at the bottom of the cells is approximately 4,910 ft (1,497 m) msl. The TRA evaporation pond consists of two cells approximately 10 ft deep, surrounded by wire mesh fence which encloses an area of approximately 5 acres. The cold waste pond consists of two cells, each with dimensions of 150 × 400 ft (45.7 m × 121.9 m), and it is surrounded by a mesh wire fence. The cells in the cold waste pond are 6 ft (1.8 m) deep. Elevation at the bottom of the north cell is approximately 4,918 ft (1,499 m) msl, and the elevation of the south cell is approximately 4,917 ft (1,498.7 m) msl (Doornbos et al. 1991).

The Snake River Aquifer is located approximately 460 ft (140 m) below land surface at TRA. Groundwater from the aquifer supplies water for drinking, cooling, and other needs at TRA and the ATR. Two perched water bodies are located in the subsurface at approximately 50 ft (15.2 m) and 150 to 200 ft (45.7 to 61 m) below land surface. Both perched water bodies are attributed to infiltration from the disposal ponds located at the surface.

2.4.1.3 Conclusions. The ATR structures and land area elevations are provided above. The regional surface water features and flood control structures are described in detail. The data are adequate for the flooding evaluation in the following section.

2.4.2 Floods

2.4.2.1 Flood History. Flooding at the INEEL has occurred with the simultaneous occurrence of melting snow and spring rains. Snowmelt runoff on the INEEL occurs in January, February, and March, while runoff from the surrounding mountains generally occurs in May or June. The onset of a prolonged freeze usually takes place in late November, lasting three months or more and ending in late February or early March. If the ground is frozen when snowmelt occurs, flooding can be extensive because the infiltration capacity of the soil is greatly reduced. Localized flooding has not and is not expected to result in flooding of the ATR reactor building basements (see Chapter 3, Design of Structures, Components, Equipment, and Systems).

In July 1983, abnormally cold temperatures for December combined with large flows in the Big Lost River resulted in ice jams that impeded flow and threatened flooding upstream of TRA. An emergency dike-building effort prevented overtopping and failure of the dike. During the following summer, the height of the dikes was increased to prevent this threat in the future (McKinney 1985). TRA and, consequently, the ATR were well out of the affected area.

2.4.2.2 Flood Design Considerations. The Big Lost River is the only major natural surface water feature near TRA. The river channel is located 4,480 ft (1,365 m) from the southeast corner of the TRA security fence.

To protect downstream facilities on the INEEL from flooding of the Big Lost River, the INEEL flood diversion system was constructed in 1958. This system uses a small dam to divert river flows into spreading areas and is located in the southwestern portion of the INEEL. Since that time, the system has been upgraded, and the capacity of the diversion channels leading to the INEEL spreading areas has been significantly increased. Water not diverted at the INEEL diversion dam flows northward across the INEEL in the shallow, gravel-filled channel of the Big Lost River to its terminus in the Big Lost River Sinks.

Localized flooding that results from sudden thaw after a heavy snowfall or thunderstorm is not expected to result in flooding of the ATR reactor building basements because the reactor building main floor is located at a higher elevation than its surroundings. Similarly, other safety components in supporting facilities are also located at the ground level on pedestals at least 3 in. (7.6 cm) above the ground and are not expected to be flooded.

Table 2.4-2 presents the simulated probable maximum flood (PMF), ATR elevation, peak flood elevation, and wave arrival time upon approach to TRA. Given an overtopping failure of the dam concurrent with the PMF, the flood water is not expected to reach the TRA site (Koslow and Van Haaften 1986).

2.4.2.3 Effects of Local Intense Precipitation. The effects of precipitation in the form of rain or snowfall, and the subsequent melting of the snow, have been evaluated for flooding potential. Snowfall is a major contributor to the total yearly precipitation for the INEEL. The annual average INEEL snowfall is 27.6 in. (70.1 cm), with a maximum recorded yearly snowfall of 59.7 in. (152 cm) (Clawson, Start, and Ricks 1989). December receives the most snow with an average of 6.4 in. (16.3 cm) and a maximum monthly snowfall of 18.1 in. (46.0 cm) (Clawson, Start, and Ricks 1989). The water content of the melted snow contributes approximately one-quarter to one-third of the annual average 8.71 in. (22.1 cm) of precipitation received at the INEEL. Even though annual snowfall is significant and contributes greatly to the total precipitation at the INEEL, the dryness of the air and cold winter temperatures reduce the moisture content of the snow.

In a 1986 study, Koslow and Van Haaften found that the 25-year combined rain and snowmelt was 2.74 in. (6.96 cm)/day. The cumulative design rainfall is 0.78 in. (1.98 cm)/day, resulting in 0.52 in. (1.32 cm)/day of direct runoff. The maximum snowmelt rate for the INEEL was estimated to be 2.22 in. (5.64 cm)/day. The extent of the snow cover varied considerably with the topographic features of the basin. Snow depth was assumed to be at least 5 in. (12.7 cm) and to cover 50% of the total drainage basin considered for each facility. The peak flow resulting from this combined rain and snowmelt was 63 cfs (1.78 m^3/s) for TRA. This does not pose a significant threat to the TRA facility.

In the past, ice jams in the Big Lost River have not resulted in flooding of the TRA facilities. Nearly all flow of the Big Lost River (if any) is diverted during winter months to avoid ice jams in the main river channel (EG&G Idaho 1984). TRA and the ATR are not located in areas where ice jams can cause flooding.

2.4.2.4 Conclusions. No postulated local flooding results in ATR damage. A bounding case flood scenario is evaluated in the following section.

2.4.3 Probable Maximum Flood on Streams and Rivers

The following four possible Mackay Dam failure scenarios could result in flood waters being routed downstream onto the INEEL (see Tables 2.4-2 and 2.4-3):

1. Hydraulic (piping) failure of the dam, with the 100-year recurrence interval flood (failure mode triangle and peak reservoir level of 6,066.5 ft [1,849 m])

2. Hydraulic (piping) failure of the dam, with the 500-year recurrence interval flood (failure mode trapezoid and peak reservoir level of 6,066.5 ft [1,849 m])

3. Overtopping failure caused by the PMF (peak reservoir level of 6,077 ft [1,852 m])

4. Seismic failure of the dam, coincident with the 25-year recurrence interval flood (failure mode trapezoid and peak reservoir level of 6,066.5 ft [1,849 m]).

In all four cases analyzed by Koslow and Van Haaften (1986), the peak water surface elevation of the reservoir was assumed to be above or up to the spillway crest elevation. They also assumed that the INEEL diversion dam would be overtopped by flooding waters released from the Mackay Dam. This overtopping is expected to cause failure of the diversion dam and, thus, contributes to downstream flooding on the INEEL. None of the four scenarios considered indicated that flood waters were predicted to reach the ATR facility.

2.4.3.1 Probable Maximum Precipitation. Probable maximum precipitation (PMP) has been modeled for basins above Mackay Dam. The three PMF scenarios include a general storm PMF, a thunderstorm PMF, and PMF from rainfall on snowpack. The general storm PMF resulted from the 48-hour general storm PMP in June over the entire basin, preceded 3 days in time by an antecedent storm with a magnitude of 40% of the 48-hour PMP. The peak flow entering Mackay Reservoir for this scenario was 82,000 cfs (2,322 m^3/s) occurring 154 hours after the beginning of the storm. The thunderstorm PMF resulted from a 1-hour thunderstorm in August over the 50-mi^2 (50-km^2) drainage area adjacent to Mackay Reservoir. The peak flow entering Mackay Reservoir during the thunderstorm PMF was 7,010 cfs (198.4 m^3/s) occurring 5.5 hours after the beginning of the storm.

The PMF resulting from rainfall on snowpack occurs in June from a 48-hour general storm PMP on a 100-year snowpack. The peak flow entering Mackay Reservoir from this scenario was 139,000 cfs (3,936 m^3/s) occurring 35 hours after the beginning of the storm. This scenario yields the highest peak inflow to Mackay Reservoir. It also peaks in less time than the general storm PMF. However, this scenario is overly conservative because of the uncertainty involved in modeling snowpack in large basins in semiarid regions such as the Big Lost River above Mackay Reservoir. For design purposes, the general storm PMF represents a more reasonable estimate of upper-limit flooding in this drainage.

The flow from the general storm PMF scenario is of such great magnitude compared to the Mackay Dam spillway and outlet works capacities (3,240 cfs [91.7 m^3/s] and 2,900 cfs [82.1 m^3/s], respectively), that it will overtop and cause Mackay Dam to fail. A Mackay Dam failure will result in routing flood waters downstream onto the INEEL, which will result in localized flooding along the Big Lost River. However, none of the flood waters are predicted to reach ATR.

2.4.3.2 Precipitation Losses. The Big Lost River is perennial upstream from a point a few miles southeast of Arco. Downstream, flow in the river is lost by infiltration through the channel bottom. Infiltration losses through the permeable channel bottom are largest when the channel is initially wetted and when the hydraulic head is at a maximum. Heavy precipitation runoff collects in the river channel and infiltrates into the ground.

A maximum loss of 28 ft^3/s (0.79 m^3/s) per mile was measured in the area of the Big Lost River Sinks (Bennett 1990). Stream flow losses to evapotranspiration were minor compared to infiltration losses (Bennett 1990).

2.4.3.3 Runoff and Stream Course Models. Four different Mackay Dam failure scenarios were simulated with the NWS DAMBRK model (Koslow and Van Haaften 1986). The results of peak flow, peak water surface elevation flood, wave arrival time, and maximum water velocity were presented for eight cross sections along the Big Lost River. The results indicate that in the event of a Mackay Dam failure, flooding would occur throughout the entire study reach; no flooding was expected to reach the TRA facilities.

2.4.3.4 Probable Maximum Flood Flow. Three PMF scenarios were evaluated: a general storm PMF, a thunderstorm PMF, and a PMF from rainfall on snowpack. The general storm PMF resulted from the 48-hour general storm PMP in June over the entire basin, preceded three days in time by an antecedent storm with a magnitude of 40% of the 48-hour PMP. The peak flow entering Mackay Reservoir for this scenario was 82,000 cfs (2,322 m^3/s), occurring 154 hours after the beginning of the storm.

The thunderstorm PMF resulted from a 1-hour thunderstorm in August over the 50-mi^2 (129-km^2) drainage area adjacent to Mackay Reservoir. The peak flow entering Mackay Reservoir during the thunderstorm PMF was 7,010 cfs (199 m^3/s), occurring 5.5 hours after the beginning of the storm.

The PMF resulting from rainfall on snowpack occurs in June from a 48-hour general storm PMF on a 100-year snowpack. The peak flow entering Mackay Reservoir from this scenario was 139,200 cfs (3,942 m^3/s) occurring 35 hours after the beginning of the storm. This scenario yields the highest peak inflow to Mackay Reservoir. It also peaks sooner than the general storm PMF. This scenario resulted in failure of the Mackay Dam and subsequent flooding of localized areas along the Big Lost River on the INEEL.

2.4.3.5 Water Level Determinations. Not applicable.

2.4.3.6 Conclusions. The worst-case flood is an overtopping failure of Mackay dam caused by the probable maximum flood (PMF) in the drainage above the dam. The predicted consequences are no damage to the ATR.

2.4.4 Potential Dam Failures, Seismically Induced

Seismic failure of the Mackay dam is discussed in Chapter 3, Section 7, Seismic Design. In the case of seismic failure of the dam coincident with the 25-, 100-, or 500-year recurrence interval flood, the flood water is not expected to have any significant impact on the TRA site except the loss of commercial power (Doornbos et al. 1991).

2.4.4.1 Dam Failure Permutations. Not applicable.

2.4.4.2 Unsteady Flow Analysis of Potential Dam Failures. Not applicable.

2.4.4.3 Water Level at Plant Site. Not applicable.

2.4.4.4 Conclusions. Seismic dam failure is bounded by the previously discussed PMF dam failure.

2.4.5 Probable Maximum Surge and Seiche Flooding

Seiche flooding is not possible; therefore, no further discussion is included.

2.4.6 Probable Maximum Tsunami Flooding

Tsunami flooding is not possible; therefore, no further discussion is included.

2.4.7 Ice Effects

In July 1983, abnormally cold temperatures, combined with large flows in the Big Lost River, resulted in ice jams that impeded flow and threatened flooding on the INEEL. An emergency dike-building effort prevented overtopping and failure of the dike. During the following summer, dike heights were increased to prevent a flooding threat in the future (McKinney 1985). In winter months when water flows in the Big Lost River, nearly all of the water is diverted to the spreading areas so that INEEL structures are not threatened. Also, the diversion gate is managed to minimize ice jam flooding. In conclusion, TRA and the ATR facility are therefore well out of the ice jam affected area.

2.4.8 Cooling Water Canals and Reservoirs

Not applicable.

2.4.9 Channel Diversions

Not applicable.

2.4.10 Flooding Protection Requirements

A system of dikes protects several INEEL facilities from flooding, but TRA is not in danger of being flooded. Even in the worst-case scenario, it is unlikely that significant flooding would occur at TRA (Doornbos et al. 1991).

2.4.11 Low Water Considerations

All water used at the INEEL is obtained from the aquifer; therefore, no discussions of low water considerations are included. See Section 2.4.13.1.3 for a discussion of water levels in the aquifer.

2.4.12 Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluent in Surface Waters

Natural surface waters would not receive liquid effluent discharges from either normal operations or accidents.

2.4.13 Groundwater

Groundwater at the INEEL occurs in the vadose zone and the aquifer. Descriptions of groundwater are provided in the following sections.

2.4.13.1 Description and Onsite Use

2.4.13.1.1 Vadose Zone—The vadose zone is that portion of the subsurface that extends from the land surface to the water table. An extensive vadose zone exists at the INEEL ranging in thickness from 200 ft (61 m) in the north to greater than 900 ft (274 m) in the south. At the TRA site, the vadose zone is approximately 460 ft (140 m) thick. It consists of surface sediments, relatively thin horizontal basalt flows, and occasional interflow sediments. Surface alluvial sediments include clays, silts, sands, and gravels deposited by the Big Lost River. Surficial deposits at TRA range from 38 to 50 ft (11.6 to 15.2 m) thick.

Thick sequences of interfingering basalt flows comprise most of the vadose zone beneath TRA. These flows are characterized by large void spaces resulting from cinders, lava tubes, fractured flow tops, and the hummocky basalt-flow surfaces. Basalt flows are commonly fractured. Some fractures are filled with sedimentary material transported from the surface by water and gravity while others are open.

Some basalt flows were exposed at the surface long enough to collect deposits of interflow sediments. Subsequently, they were buried by other basalt flows and are now known as sedimentary interbeds. The most common interflow materials are wind-blown sand and loess.

2.4.13.1.2 Perched Water—Perched water occurs when water migrates vertically and, to a lesser extent, laterally from the source until an impeding layer is encountered (at the INEEL, a layer of dense basalt or fine sedimentary materials where the hydraulic conductivity is sufficiently low that vertical movement of water is restricted). After a perched water mound develops, lateral movement of the water can accelerate. Figure 2.4-1 is a generalized diagram of perched water at TRA. Perched water bodies exist at two depths beneath TRA: one at approximately 50 ft (15.2 m) and another between 150 and 200 ft (45.7 and 61 m). Both perched water bodies have been monitored since the 1960s. The shallow perched water body is believed to be leakage from the retention basin and/or associated pipelines. Water levels in boreholes completed in the shallow perched water body suggest that its extent has not changed significantly over the past 25 years. Surface ponds are also thought to be the source for the deep perched water at TRA. The general shape of the deep perched water body also has not changed appreciably over time although water levels have fluctuated in response to volume of water discharged to the surface ponds (Doombos et al. 1991). Transmissivity for the deep perched water body has been measured in several studies including those described by Ackerman (1991) and Bishop, Wylie, and Mattick (1992). Transmissivity estimates have ranged from 3.5 to 4.3 × 10^5 ft^2/day (2.8 to 4 × 10^4 m^2/day).

2.4.13.1.3 Snake River Plain Aquifer

2.4.13.1.3.1 Regional Hydrogeology—The INEEL is located on the western edge of the ESRP, which overlies the Snake River Plain aquifer, a continuous body of groundwater underlying nearly all of the ESRP. Aquifer boundaries correspond to surface physical features (i.e., mountains on the west and north and the Snake River on the east). Occurrence and movement of groundwater in the aquifer is dependent on the geologic framework, which determines aquifer transmissivity and storage, and the recharge and discharge within that framework. The aquifer is composed of numerous relatively thin basalt flows extending to depths in excess of 3,500 ft (1,067 m) below land surface. Over time, some of these flows have been exposed at the surface long enough to collect sediment and develop soil horizons. These sedimentary horizons, known as sedimentary interbeds, are sandwiched between basalt flows at various depths. Many of these sedimentary interbeds act as aquicludes. Regionally, most water moves horizontally through basalt interflow zones, which are the broken and rubble zones between lava flows. Locally, water moves vertically along joints and the interfingering edges of interflow zones and sedimentary interbeds act to restrict the vertical movement of groundwater (Garabedian 1986).

The Snake River Plain aquifer groundwater flow is primarily in a southwest direction; however, locally, the flow direction can be affected by recharge from rivers, surface water spreading areas, and heterogeneities in the aquifer.

Transmissivity of Quaternary basalt, as determined from aquifer tests, ranges from 1,000 to 100,000 ft^2/day (100 to 10,000 m^2/day) and, in places, exceeds 1 million ft^2/day (100,000 m^2/day) (Whitehead 1992). Yields of wells drilled in the Snake River basalts are among the largest in the nation. Irrigation wells open to less than 100 ft (30.5 m) of the aquifer yield as much as 7,000 gal/min (26,500 L/min) with slight drawdown; yields of 2,000 to 3,000 gal/min (7,500 to 11,400 L/min) are common.

Storage coefficients determined from aquifer tests vary widely. Values range from 10^{-5} to 10^{-1}, indicating that aquifer conditions range from confined to unconfined (Whitehead 1992).

Water storage in the Snake River Plain aquifer has been estimated at 2×10^9 acre-ft (2.5×10^{12} m^3), which is about the same volume of water stored in Lake Erie, or enough water to cover the entire state of Idaho to a depth of 4 ft (1.2 m). Approximately 5×10^8 acre-ft (6.2×10^{11} m^3) of the water stored in the aquifer could potentially be pumped from it (Robertson, Schoen, and Barraclaugh 1974).

Recharge to the ESRP aquifer is from seepage of irrigation water, stream flow and canal leakage, tributary valley underflow, and direct precipitation. Aquifer discharge is largely spring flow to the Snake River and water pumped for irrigation. Major springs are near the American Falls Reservoir and along the Snake River from Milner Dam to King Hill. About two-thirds of the groundwater discharged from the aquifer is through the series of springs between Milner and King Hill. Included in that reach are 11 of the 65 springs in the United States that discharge an average of more than 100 ft^3 (2.8 m^3)/s (Meinzer 1927).

Regional comparisons of water levels indicate that water levels have been relatively stable in the central part of the ESRP for the last 50 years (Garabedian 1986). However, on large tracts of land in the eastern plain, water levels rose an average of 60 to 70 ft (18 to 21 m) (Mundorff, Crosthwaite, and Kilburn 1964), and groundwater discharge increased soon after the initiation of surface-water irrigation (about 1910). By 1928, most surface water for irrigation was appropriated; since that time, the total amount of water diverted for irrigation has been relatively stable.

2.4.13.1.3.2 INEEL Hydrogeology Aquifer Geometry and Physical Characteristics—Depths to the water table from the INEEL land surface range from about 200 ft (61 m) in the northern part of the site to more than 900 ft (274 m) in the southern part (Pittman, Jensen, and Fischer 1988). At TRA, depth to water is 460 ft (140 m). Ackerman (1991) analyzed aquifer-test data of 183 single-well tests at 94 wells in the Snake River Plain aquifer to estimate values of transmissivity. Estimates of transmissivity for individual wells ranged from 1.1 to 7.6×10^5 ft^2/day (0.1 to 7.1×10^4 m^2/day). Lowest values are generally at the northern end of the site and the highest are near TRA.

Multiple-well aquifer tests are necessary to estimate aquifer storage, but few multiple-well aquifer tests have been conducted at the INEEL. The long-term average coefficient of storage, based on limited data, is estimated to be in the order of 5% (0.05) (Walton 1958), which is indicative of unconfined conditions. A long-term pumping test that would stress the aquifer sufficiently to measure appreciable gravity drainage has not been conducted at the INEEL.

2.4.13.1.3.3 Groundwater Flow—The elevation of the water table for the Snake River Plain aquifer and the general direction of groundwater movement in the vicinity of the INEEL are depicted in Figure 2.4-2. Groundwater flow is to the south-southwest, although, locally, the direction of groundwater flow is affected by recharge from rivers, surface water spreading areas, and heterogeneities of the aquifer.

Vertical-head gradients are usually less than 0.01 over the first 200 ft (61 m) and less than 0.02 over the first 550 ft (168 m) of saturated thickness. Across the INEEL, the horizontal gradient of the water table ranges from 1 to 15 ft/mi (0.2 to 2.8 m/km) [average is about 4 ft/mi (.76 m/km)] (Ackerman 1991). Horizontal groundwater flow velocity ranges from 5 to 25 ft/day (1.5 to 7.6 m/day); however, most of the flow ranges from 5 to 10 ft/day (1.5 to 3 m/day) (Robertson, Schoen, and Barraclaugh 1974). Data from Mundorff, Crosthwaite, and Kilburn (1964) indicate that about 2,000 ft^3/sec (56.6 m^3/s) flows beneath the INEEL at its widest point.

2.4.13.1.3.4 Recharge and Discharge—Recharge to the Snake River Plain aquifer near the INEEL originates from precipitation in the mountains to the north and west. Most of the inflow occurs as underflow from alluvial-filled valleys along the edges of the ESRP. The Big Lost River, Little Lost River, and Birch Creek terminate at sinks on or near the INEEL and recharge the aquifer. Recharge occurs through the surface of the plain from flow in the channel of the Big Lost River and its diversion areas. Additionally, recharge may occur from melting of localized snowpacks during years in which snowfall accumulates on the ESRP.

Recharge to the aquifer in the vicinity of the INEEL is closely linked to the amount of precipitation, particularly snowfall, for a given year. Historically, recharge from rivers has occurred during wet cycles that last from a few to several years. The intervening years produce little recharge.

2.4.13.1.3.5 Onsite Use—All water used at TRA is groundwater from the SRP aquifer tapped by three deep wells. Water supply and distribution at the TRA is a TRA landlord function. See Chapter 9 (Auxiliary Systems) for details on water supply and distribution for the ATR. Generally TRA uses approximately 50,000,000 gal/mo of water.

2.4.13.1.3.6 Piezometric Levels—Figure 2.4-3 is a contour map of the piezometric surface of the aquifer in the vicinity of the TRA. As discussed in the hydrology section, the flow is to the south-southwest. Vertical-head gradients are usually less than 0.01 over the first 200 ft and less than 0.02 over the first 550 ft of saturated thickness. Horizontal gradient of the water table in the aquifer underneath the INEEL is about 4 ft/mi, and flow velocity ranges from 5 to 25 ft/day.

The TRA facility is located over a very permeable section of the aquifer. Ackerman (1991) analyzed an aquifer test performed on the Materials Test Reactor test well, which produced a transmissivity of 2.0×10^5 ft^2/day (1.86×10^4 m^2/day). Because the aquifer is extremely permeable in this area, a large discharge has little effect on the hydraulic head, indicating that it would be difficult to reverse flow directions by pumping. The amounts of water currently removed from the aquifer have little or no effect on gradients or piezometric levels.

Natural aquifer recharge sources do not exist within the TRA facility. Recharge occurs from the intermittent flow of the Big Lost River.

2.4.13.2 Sources

2.4.13.2.1 Water Use on the Eastern Snake River Plain—Most groundwater pumped from the ESRP aquifer is used to irrigate crops. ESRP groundwater users have been categorized according to primary use category (some rights are granted for multiple uses), number of permits, and amount diverted (see Table 2.4-3).

Of the groundwater withdrawals in 1980 [5.4 million acre-ft (6.7×10^9 m^3)], 97.1% of total withdrawals and 99.8% of total consumed were expended on agriculture (Frenzel 1987).

Major centers for groundwater irrigation on the ESRP are clustered in the areas of Mud Lake, Osgood, American Falls, Blackfoot, Cassia, and eastern Twin Falls Counties. Smaller areas irrigated by groundwater are scattered throughout the ESRP.

Because groundwater supplies 100% of the drinking water consumed within the ESRP (Gaia Northwest 1988) and an alternative drinking water source or combination of sources are not available, the U.S. Environmental Protection Agency (EPA) designated the ESRP aquifer a sole source aquifer in 1991 (EPA 1990).

EPA defines a sole source aquifer as one that supplies 50% of the drinking water consumed in the area overlying the aquifer. Current guidelines stipulate that designated sole source aquifers have no alternative source or combination of sources that could physically, legally, and economically supply all those who obtain their drinking water from the aquifer. Because the ESRP aquifer has been designated as a sole source aquifer, no Federal financial assistance may be committed for any project if EPA determines that the project may contaminate the aquifer through the recharge zone, which would create a significant hazard to public health.

The INEEL holds a Federal Reserved Water Right that permits a pumping capacity of 80 cfs (2.26 m^3/s) and a maximum consumptive use of 35,000 acre ft/yr (4.3×10^7 m^3/yr). The water is used for drinking, process water, and noncontact cooling water. Groundwater use at the INEEL is a relatively small component of the total pumpage from the ESRP aquifer. Groundwater withdrawals at INEEL average about 6,000 acre-ft/yr (7.4×10^6 m^3/yr) from 1979-1981 (DOE 1995). Consumptive use was 37%, or 2,220 acre-ft (2.7×10^6 m^3) per year. The remaining 63% has been discharged to the aquifer through disposal wells (discontinued in 1984) and infiltration and evaporation ponds (still in use) (Pittman, Jensen, and Fisher 1988).

Total water use for the upper Snake River drainage basin and the Snake River Plain aquifer was 36,514.27 acre-ft (4.50×10^7 m^3) for 1985, which was over 50% of the water used in Idaho and about 7% of agricultural withdrawals in the nation.

2.4.13.3 Accident Effects. No significant impacts to groundwater are expected from any of the accidents analyzed (see Chapter 15, Accident Analyses).

2.4.13.4 Monitoring or Safeguard Requirements. The U.S. Geological Survey (USGS) conducts a Site-wide groundwater monitoring program at the INEEL to determine the effects of INEEL operations on groundwater quality and quantity. The basis for this program goes back to 1949 when the U.S. Atomic Energy Commission asked the USGS to investigate and describe the water resources of the INEEL and adjacent areas. Information was collected that depicted hydrogeologic conditions before reactor operations. Since that time, the USGS has maintained a groundwater monitoring system and conducted investigations to determine changes in water quality and quantity resulting from activities at the various facilities at the INEEL. Periodic summaries have been published, including those by Nace (1959), Barraclough, Teasdale, and Jensen (1967), Barraclough et al. (1967), Robertson, Schoen, and Barrcalaugh (1974), Barraclough, Robertson, and Janzer (1976), Barraclough and Jensen (1976), Barraclough, Lewis, and Jensen (1981), Lewis and Jensen (1984), Pittman, Jensen, and Fisher (1988), and Orr and Cecil (1991). Well logs, water levels, and water chemistry data are available in files at the INEEL office of the USGS.

The locations of water-level observation wells within the INEEL water-level monitoring network and the frequency of water-level measurements are provided in Figure 2.4-4. The USGS routinely monitors water levels in 160 onsite wells and boreholes (120 in the aquifer and 36 in perched water zones).

Approximately 140 wells and boreholes (99 in the aquifer and 38 in perched water zones) are monitored for water quality on or near the INEEL. The USGS conducts a comprehensive aquifer-well sampling program to identify contaminants and observe contaminant migration within the Snake River Plain aquifer. Sampling of perched water wells and surface water sites is performed to provide data on the quality of water recharging the aquifer. The locations and sampling frequency for wells within the INEEL water-quality monitoring network as reported by Orr and Cecil (1991) for the 1986-1988 monitoring period are provided in Figure 2.4-5.

2.4.13.5 Design Bases for Subsurface Hydrostatic Loading. Most of this section is not applicable to ATR at TRA because a dewatering system is not employed. The three wells that supply coolant and serve as firewater supply have not been designed for protection against seismically induced pressure waves. However, no damage from the 1983 Borah Peak earthquake (M_s 7.3) was observed (see Section 2.5.2).

2.4.13.6 Conclusions. The reported studies of regional and local groundwater are adequate to support any foreseeable safety analysis of ATR operations. No groundwater characteristics adversely affect any ATR structures.

Table 2.4-1. Mackay Dam and INEEL Diversion Dam reservoir characteristics.[a]

	Mackay Dam	INEEL Diversion Dam
Dam crest elevation (ft msl)	6,076.0	5,064.7
Dam crest length (ft)	1,430.0	500.0
Dam height (ft)	79.0	22.0
Spillway crest elevation (ft msl)	6,066.5	
Spillway	Ungated over-flow crest, 75 ft long	none
Gate centerline elevation (ft msl)	6,036.0 (upper) 6,007.8 (lower)	5,045.6
Dam base elevation (ft msl)	5,997.0	5,043.6
Spillway maximum capacity (cfs)	6,588.0	—
Gate maximum capacity (cfs)	2,960.0	1,121.0
Reservoir capacity, ac-ft[b] at Mackay Dam elevation (ft msl)	55,091 at 6,076.0 44,500 at 6,066.5 8,750 at 6,030.0 500 at 6,010.0	—

a. After Koslow and Van Haaften 1986.

b. It has been estimated that Mackay Reservoir has lost 22% of its mid- and late-season irrigation capacity because of sedimentation of the reservoir (Butte Soil Conservation District 1982).

Table 2.4-2. Simulated possible flooding scenarios resulting from Mackay Dam failure.[a]

Flood source	ATR/streambed elevation (MSL) USGS datum	Peak flood elevation (MSL) USGS datum	Time of wave arrival (hr)
Seismic failure of Mackay Dam	4,925.5/4,919	4,923	15.3
Piping failure 100-year flood	4,925.5/4,919	4,923	16.7
Piping failure 500-year flood	4,925.5/4,919	4,923	15.3
PMF-induced overtopping failure	4,925.5/4,919	4,924	13.1

a. Koslow and Van Haaften 1986.

Table 2.4-3. Groundwater appropriations issued by the Idaho Department of Water Resources on the Eastern Snake River Plain.[a]

Allocation type	Number of rights Issued	cfs diverted	Acreage watered
Irrigation	4,287	30,197.675	1,075,694.85
Irrigation storage	1	5,000.00	
Diversion to storage	1	100.00	
Municipal	34	99.07	
Fish propagation	8	91.38	
Commercial	150	58.36	1.22
Industrial	42	34.34	
Wildlife	5	14.90	
Domestic	121	11.29	58.00
Power	1	7.00	
Stock Water storage	3	2.04	
Heating	2	1.81	
Fire protection	3	1.60	
Recreation	2	0.28	
Cooling	1	0.05	
Wildlife storage	1	0.02	

a. Information obtained from the files of the Idaho Department of Water Resources.

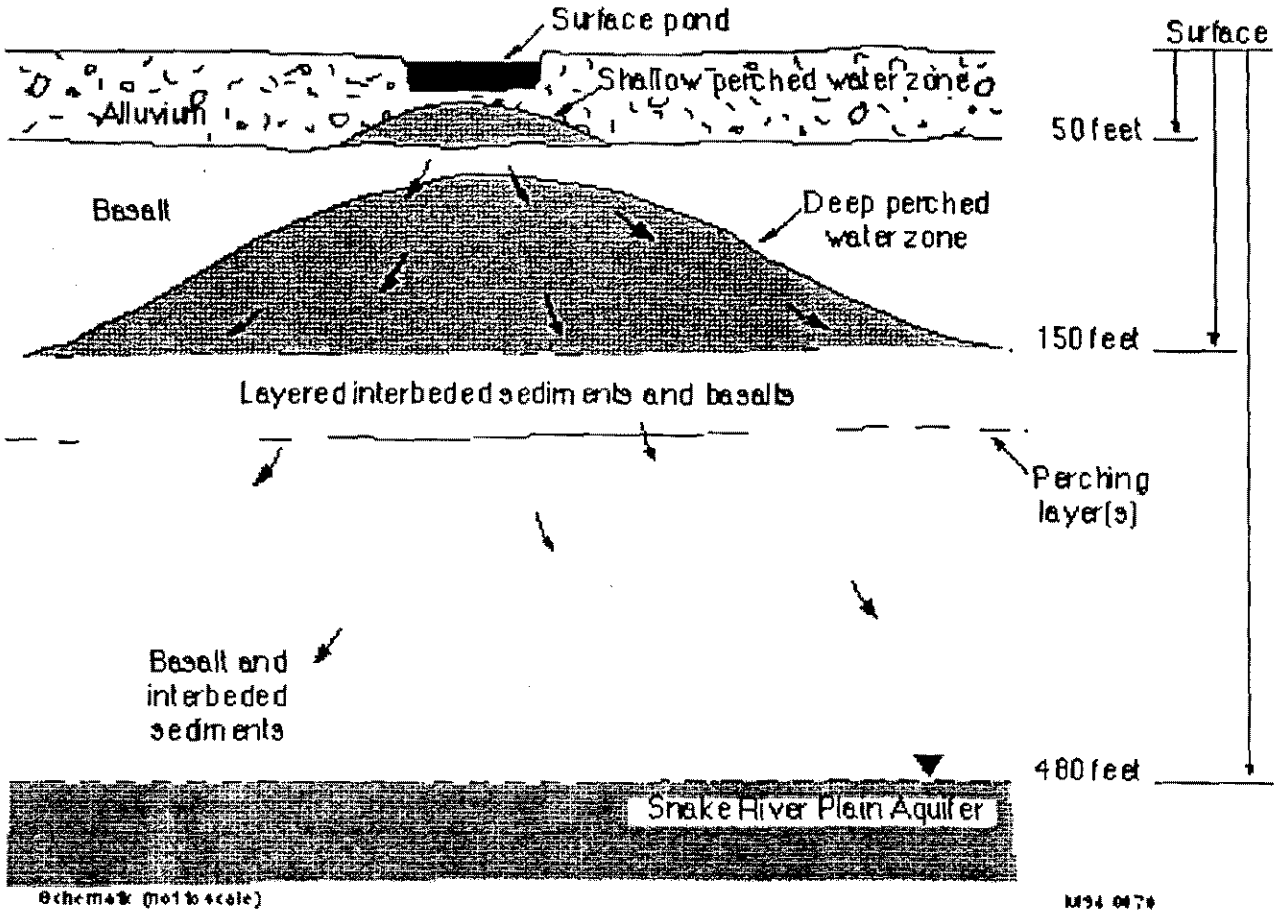

Figure 2.4-1. Generalized Diagram of Perched Water at TRA. **(For Information Only)**

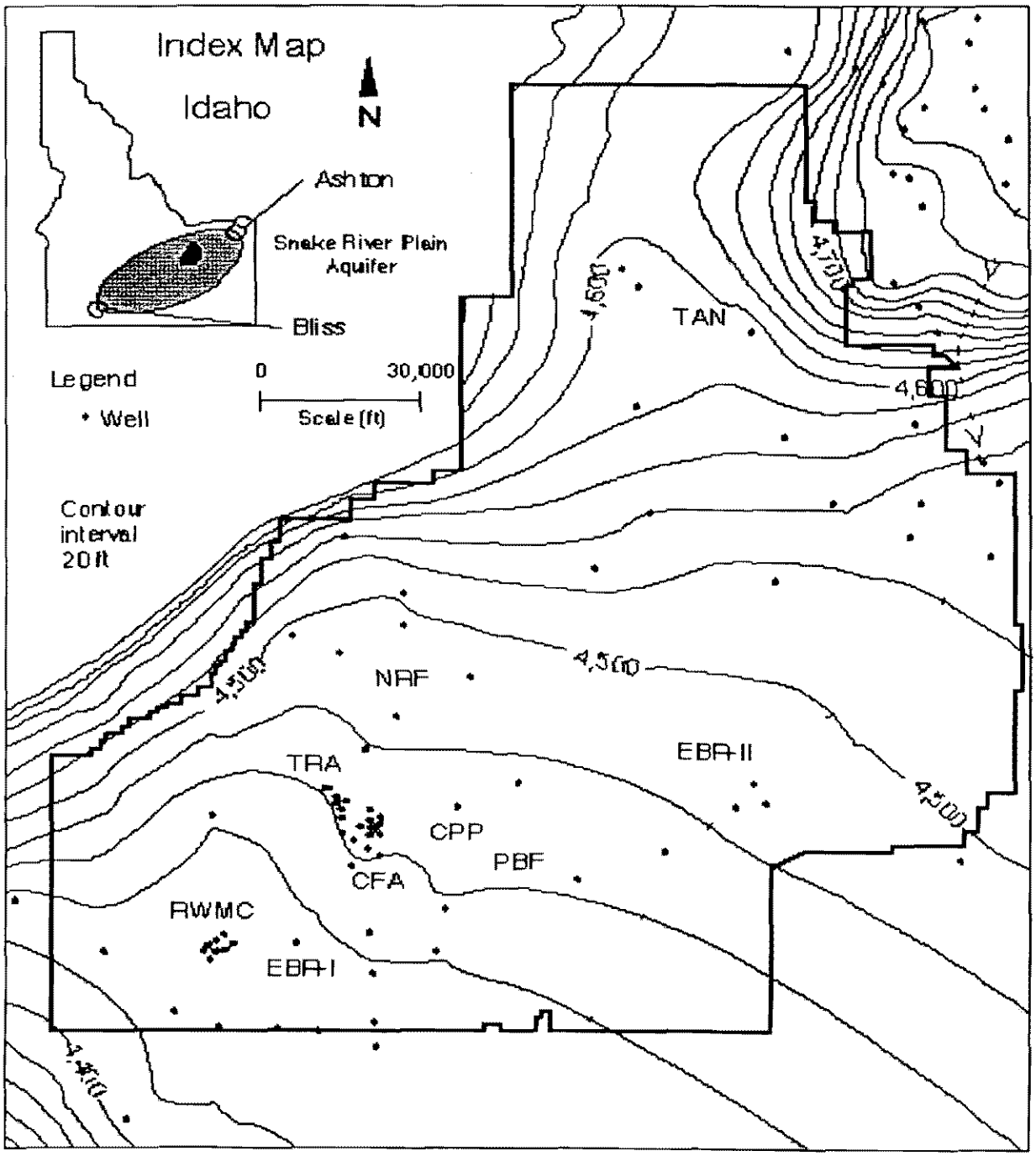

Figure 2.4-2. Elevations of the Water Table for the Snake River Plain Aquifer (after Stormberg and Zimmerle, 1992). **(For Information Only)**

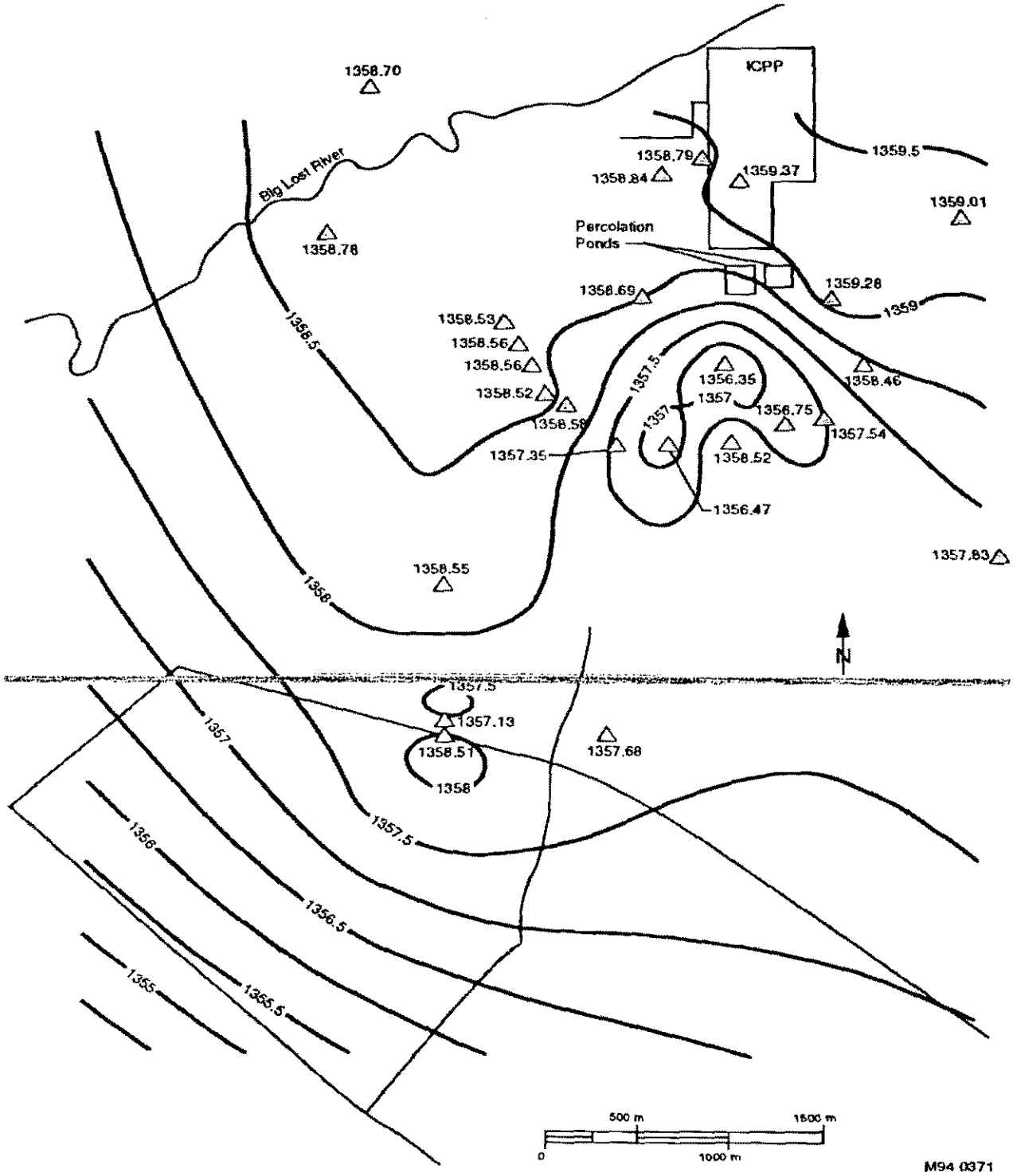

Figure 2.4-3. Contour map of Piezometric surface of the aquifer near TRA (in meters, average mean sea level) (after Keck et al, 1994). **(For Information Only)**

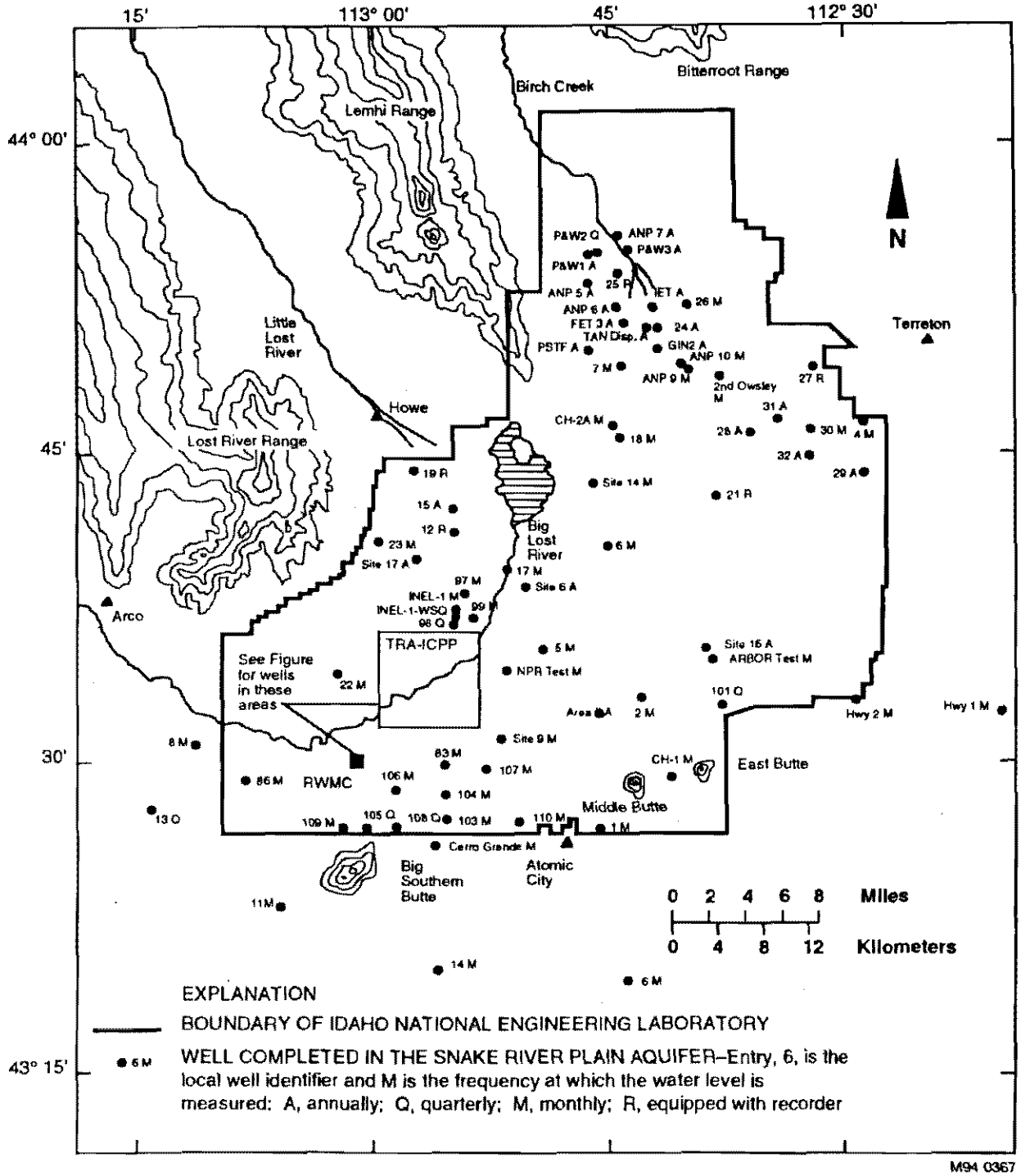

Figure 2.4-4. Locations of Water-level Observation Wells within INEEL Water-quality Monitoring Network. (after Orr and Cecil, 1991). **(For Information Only)**

FOIA Exemption 3 UCNI, Exemption 7f (Page 2-60 – Well map with coordinates)

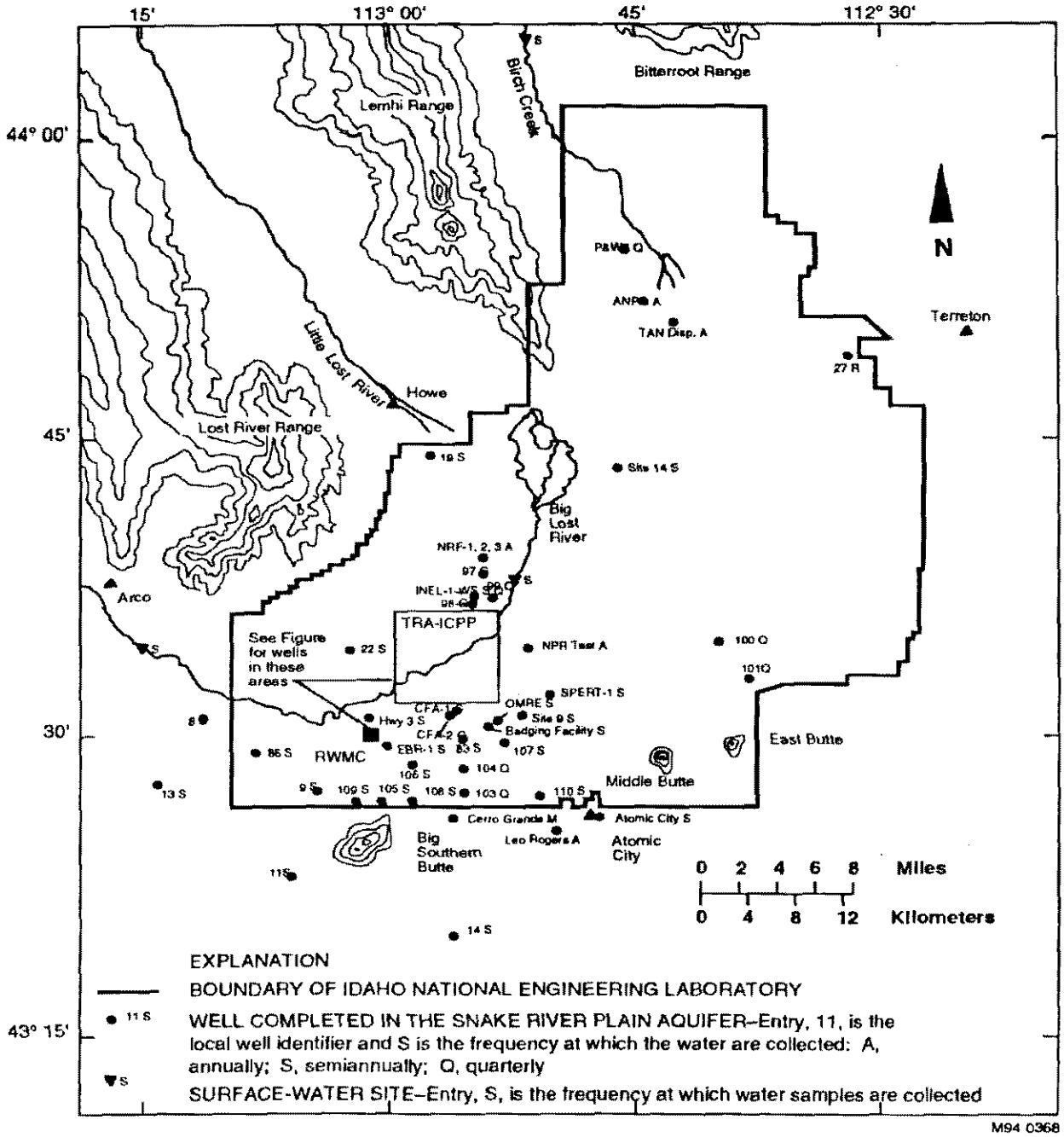

Figure 2.4-5. Locations and Sampling Frequency for Wells within the INEEL Water-quality Monitoring Network (after Orr and Cecil, 1991). **(For Information Only)**

FOIA Exemption 3 UCNI, Exemption 7f (Page 2-62 – Well map with coordinates)

2.5 Geology, Seismology, and Geotechnical Engineering

This section summarizes the basic geologic and seismic characteristics of the region and site; the geologic history; the lithologic, stratigraphic, and structural geologic conditions; the tectonic structures; and the geologic features and conditions that could affect facility structures. A more detailed discussion of the geology and seismology of the region is presented in Appendix A.

2.5.1 Basic Geologic and Seismic Information

The Snake River Plain (SRP) is a crescent-shaped, late Cenozoic, topographic and structural basin that crosses southern Idaho (Malde 1991). Early volcanism of the SRP was explosive and voluminous, with an age progression from the 15-million-year-old volcanic centers of the Owyhee Plateau in southwest Idaho, to the younger Quaternary silicic calderas and active hydrothermal system of the Yellowstone Plateau. Silicic calderas are inferred to have been active on the ESRP between about 4 to 8 million years ago, and produced the voluminous ash-flow tufts of the Heise volcanic field (Morgan, Doherty, and Leeman 1984; Hackett and Morgan 1988). These old silicic vents of the INEEL area have subsequently been covered in places by > 1 km of younger basaltic lavas and sediments.

The INEEL is located near the northwestern margin of the ESRP and is influenced by two distinct geologic provinces. The topographically subdued ESRP province is a volcanic plain having a relatively low rate of seismicity in the midst of the seismically-active, high-relief mountains of the surrounding Basin and Range province.

Borehole investigations (Doherty 1979a, b; Doherty, McBroome, and Kuntz 1979; Anderson and Lewis 1989; Anderson 1991) together with surficial-geologic mapping (Scott 1982; Kuntz et al. 1990) and paleomagnetic data (Champion, Lanphere, and Kuntz 1988) have led to a general understanding of the ages and distribution of volcanic and sedimentary deposits in the INEEL area. At least five Quaternary basalt lava-flow groups are known, ranging in age from 5,200 years for the Hell's Half Acre lava field to > 730,000 years for lavas of the northern INEEL Site. Basaltic vents on the ESRP are typically craters, small pyroclastic cones and open fissures that collectively define volcanic-rift zones. The best known and most recently active rift zone on the SRP is the Great Rift (Kuntz et al. 1988) with its eight Holocene eruptive periods and volcanism as recent as 2,100 years ago. Several older and more diffuse volcanic-rift zones cross the INEEL, and volcanic vents within the INEEL boundary are also older than 200,000 years. The Cerro Grande Flow (~ 13 Ka) flows onto INEEL from a vent to the south of INEEL.

The mountainous terrain to the north and northwest of the INEEL is part of the northern Basin and Range tectonic province. The ranges are composed of Paleozoic sedimentary rocks that have been uplifted and tilted eastward by recurrent movement of major faults situated along the western flanks of the ranges. The region is seismically active, as evidenced by the 1983 M-7.3 Borah Peak earthquake.

Paleoseismic investigations cited in Woodward-Clyde Consultants (WCC) (WCC 1992b) indicate that the Lost River, Lemhi, and Beaverhead faults north of the INEEL have evidence for Quaternary displacements and associated seismicity, but that the southernmost segments of those faults (nearest the INEEL) exhibit no evidence of Holocene seismicity.

2.5.2 Seismology

The ESRP is surrounded by the seismically active Intermountain seismic belt (ISB) and the Centennial Tectonic belt (CTB). The ISB is a zone of concentrated seismicity that is more than 1,287 km long, 97 km wide, and extends from northwestern Montana through eastern Idaho, central Utah, and southern Arizona and Nevada (Smith and Sbar 1974; Arabasz and Smith 1981). The CTB, formerly called the Idaho Seismic Zone, also a seismically active zone, extends from the ISB near the Yellowstone Plateau westward into central Idaho (Stickney and Bartholomew, 1987).

Earthquake data (Richter magnitude $M_L \geq 2.5$) for the period 1884 to 1989 have been compiled from the INEEL, USGS, Montana Bureau of Mines and Geology, United States Bureau of Reclamation, University of Utah Seismograph Stations, and Engdahl and Rinehart, 1988 by WCC (1992a). The distribution of epicenters indicates that the ESRP has a low frequency of earthquakes relative to the ISB and CTB. Ground motions produced by earthquakes that occur within the Basin and Range province will be transmitted onto the ESRP. The INEEL is classified as Seismic Zone 2B of the Uniform Building Code.

The INEEL has maintained a seismic network for monitoring earthquake activity within and near the ESRP since July 1971 (King, Doyle, and Jackson 1987). Currently, the seismic network consists of 26 seismic stations and 22 strong-motion accelerographs. Earthquake data compiled by the INEEL seismic network for a 24-year period from 1971-1995, indicate that 19 microearthquakes (Coda magnitude $M_c < 1.5$) have been located within or near the boundary of the ESRP (Jackson et al. 1993). Local seismic monitoring by the INEEL and the historical earthquake record indicate that the ESRP has a low rate of seismicity relative to the surrounding seismically active regions.

The largest earthquake within the region occurred on August 17, 1959 at Hebgen Lake, Montana (within the ISB) and was located 161 km northeast of the INEEL Site. The event had a surface-wave magnitude (M_s) 7.5 (Doser 1985) and was felt at the INEEL, but caused no damage at the INEEL. Another large earthquake M_s 7.3 occurred on October 28, 1983 within the CTB (Doser and Smith 1985). The earthquake resulted from movement along the Lost River fault; a normal fault with relative vertical movement downward to the west (Crone et al. 1987). The epicenter for this event was located in the Thousands Springs valley near the western flank of Borah Peak, approximately 90-110 km from INEEL facilities (Richins et al. 1987). Although earthquake ground motions were felt at the INEEL Site, no significant damage occurred to any facility at the INEEL (Gorman and Guenzler 1983). Peak ground accelerations ranging from 0.022 to 0.078 g were recorded at basement and freefield sites at several INEEL facility areas (Jackson 1994).

2.5.2.1 Maximum Earthquake Potential. Patterns of seismicity and locations of mapped faults have been used to assess potential sources of future earthquakes for estimating ground shaking at INEEL. The sources and maximum magnitudes of earthquakes that could produce the maximum levels of ground motions at ATR include: (a) a magnitude 7.0 earthquake at the southern end of the Lemhi fault, (b) a magnitude 7.0 earthquake at the southern end of the Lost River fault, (c) a magnitude 5.5 earthquake associated with dike-injection in either the Arco or Lava Ridge-Hell's Half Acre Volcanic Rift Zones (VRZs) and the axial volcanic zone, and (d) a "random" magnitude 5.5 earthquake occurring within the ESRP (WCC 1992a). Ground motion contributions from other sources such as the postulated ESRP boundary fault, northern Basin and Range Province, Yellowstone Plateau, and Idaho Batholith are significantly smaller because of their distant locations or lower maximum magnitudes.

2.5.2.2 Seismic Wave Transmission Characteristics. Woodward-Clyde Consultants (WCC 1990) and Woodward-Clyde Federal Services (WCFS 1996) indicate that near-surface geology [1.2 to 3.1 mi (2 to 5 km) depth] has a significant influence on earthquake ground motions at the Site, in addition to amplification resulting from soil overlying bedrock. The INEEL resides upon the ESRP, which is covered with basalt lava flows and sediments. Boreholes located throughout the INEEL site indicate that the basalt is interbedded with sedimentary layers; in some areas, the percentage of interbeds is 50%.

This unique stratigraphy has the effect of attenuating or decreasing the level of earthquake ground motions because seismic waves travel through a sequence of alternating high (basalt) and low (sediments) velocity zones that tend to scatter the seismic energy. Also, seismic energy is intrinsically dampened by the sedimentary interbeds. The net effect of the interbedded basalt is to reduce the level of earthquake ground motions when compared to a homogeneous basalt (no interbeds) (WCC 1990, WCFS 1996).

Figure 2A.2-13 in Appendix A to Chapter 2 shows the shear-wave velocity (V_s) profile determined to estimate earthquake ground motions at ATR (WCFS 1996). The velocity model was derived from well and borehole logs located near TRA. Because the velocity model has large contrasts (basalt versus sediment), the velocity profiles were smoothed to taper the large effects of scattering, which resulted in low-amplitude spectra. Regional earthquakes were digitally recorded near two boreholes at TRA. These data were used to estimate the near-surface attenuation, κ, and to determine the amount of smoothing in the velocity profiles.

2.5.2.3 Safe Shutdown Earthquake. The design basis earthquake/safe shutdown earthquake (DBE/SSE) for ATR is 0.24 g (Harris 1989; DOE-ID 1995). The NRC Regulatory Guide 1.60 horizontal spectra are appropriately scaled using the SSE value (NRC 1973). The vertical input spectra are two-thirds of the horizontal input spectra (DOE-ID 1995). The DBE/SSE value is supported by the results of the Woodward-Clyde Consultants' (1990) deterministic evaluation.

The DOE-ID AE Standards (DOE-ID 2002) have been revised to incorporate new seismic design requirements of DOE Standards DOE-STD-1020 (DOE 2002a) and DOE-STD-1021 (DOE 2002b). The TRA site specific Performance Categories PC-1 and PC-2 seismic design criteria use the International Building Code (IBC 2000) spectral accelerations. The TRA site specific PC-3 and PC-4 horizontal and vertical seismic design criteria for rock Design Basis Earthquake/Safe Shutdown Earthquake (DBE/SSE) is available in the DOE-ID Standard (DOE-ID 2002). The TRA site specific PC-3 and PC-4 horizontal and vertical seismic design criteria for soil DBE is under development, and will be incorporated into the AE Standards when completed.

2.5.2.4 Operating Basis Earthquake. None defined.

2.5.2.5 Conclusions. The potential maximum earthquakes based on regional geology and seismic wave transmission characteristics of the area near the INEEL have been studied in detail. The results support the design basis earthquake in 2.5.2.3 above.

2.5.3 Surface Faulting

Surface faulting, defined as the rupture of the earth's surface from tectonic or magmatic activity, is of concern in some areas of INEEL, but not at the ATR site.

2.5.3.1 Evidence of Site Fault Offset. No evidence of fault offset is seen in surficial sediments within 5 miles of ATR. Several northeast-trending lineaments of alluvial and eolian origin are apparent on aerial photographs of the area. In some cases, linear patterns of revegetation have developed on the margins of ancient range-fire scars and eolian redistribution of surficial materials following range fires has produced linear dune-forms along fire-scar margins (Morin-Jansen 1987).

A dense array of drill holes in the TRA area (shallow geotechnical holes, deeper groundwater monitoring wells, production wells, and injection wells) and several 40- to 50-ft deep excavations to bedrock have revealed no evidence of surface ruptures or displacements in the near-surface basalt lava flows. Geologic correlations among wells and drill holes based on lithologic, geophysical, and geochemical characteristics, and based on radiometric ages of lava flows in many of these holes show no evidence of faulting within about 300 ft of the surface. However, paleomagnetic inclinations and K/Ar ages of lava flows in coreholes (Champion, Lanphere, and Kuntz 1988; Lanphere, Champion, and Kuntz 1993) have been interpreted to suggest that a graben or downwarp exists at depths of greater than approximately 300 ft (Smith, Hersley, and Jackson 1996). In contrast, geochemical correlation and correlation based on gamma logs of wells and coreholes suggest that lavas and interbeds are essentially flat beneath the area and that no faulting is necessary to explain the positions of stratigraphic units (Anderson 1991; Anderson et al. 1996). A summary of the arguments for and against these opposing interpretations is given by Smith, Hersley, and Jackson (1996). Even if a graben or fault or fold exist in the area at depths below 300 ft, it is not likely to be active in today's stress field and its age of > 300,000 years shows that it would probably not be considered "capable" by the NRC definition (Smith, Hersley, and Jackson 1996).

2.5.3.2 Earthquakes Associated With Capable Faults. Capable faults have not been identified within 5 miles of ATR, and significant earthquakes have not been recorded or reported in the area. Several microearthquakes have been recorded in the INEEL area since 1972, but they were not felt and they do not define or correlate with known faults (see Appendix 2A.2.1, Seismicity).

2.5.3.3 Investigation of Capable Faults. Capable faults do not exist within 5 mi (8 km) of the ATR facility. The 5 mi radius for investigation is recommended by NRC Regulatory Guide 1.70 (NRC 1978). The nearest capable faults are those in the vicinity of Howe and Arco, and volcanic rift zone structures in the Arco VRZ and fissures north at NRF.

2.5.3.4 Correlation of Epicenters With Capable Faults. No capable faults have been found within 5 miles of ATR.

2.5.3.5 Description of Capable Faults. Capable faults do not exist within 5 mi (8 km) of the ATR facility (see Section 2.5.3.1).

2.5.3.6 Zone Requiring Detailed Faulting Studies. There is no zone requiring detailed faulting studies within the vicinity of the ATR.

2.5.3.7 Results of Faulting Investigations. Detailed faulting investigations are not necessary within the 5 mi (8 km) radius, and none have been done.

2.5.4 Stability of Subsurface Materials and Foundations

At the ATR site, the surficial sediments vary from 30 to 50 ft (9 to 15 m) and consist mostly of gravel, gravelly sands, and sands. In some locations, a thin [0 to 7 ft (0 to 2 m) thick] layer of clay and silt underlie the gravelly alluvium, forming a discontinuous low-permeability layer just above the basalt bedrock (Hull 1989). That low-permeability layer caused infiltration water from the formerly used TRA warm waste pond to create a local perched water body that extends a short distance inside the southeast corner of the TRA fence (Hull 1989). These perched water bodies have the potential to cause soil instability because the water saturates the sediment. However, the majority of the surficial sediments are gravelly, and the area beneath ATR does not have a perched water body. In conclusion, there is low potential for liquefaction or other soil instabilities in the ATR area.

2.5.4.1 Geologic Features. Specific details concerning geologic features are located in Appendix A.

2.5.4.2 Properties of Subsurface Materials. Specific details concerning properties of subsurface materials are located in Appendix A.

2.5.4.3 Exploration. Specific details concerning exploration are located in Appendix A.

2.5.4.4 Geophysical Surveys. Specific details concerning geophysical surveys are located in Appendix A.

2.5.4.5 Liquefaction Potential. Specific details concerning liquefaction potential are located in Appendix A.

2.5.4.6 Earthquake Design Bases. See Section 2.5.2.3, Safe Shutdown Earthquake, for derivation of the SSE. Because liquefaction is not a problem for the ATR facility, it is justified to use the derived SSE.

2.5.4.7 Static Stability. The foundations of the reactor building rest on the underlying basalt rock.

2.5.4.8 Design Criteria. See Section 2.5.2.3, Safe Shutdown Earthquake.

2.5.4.9 Techniques to Improve Subsurface Conditions. No techniques have been used to improve subsurface conditions at the ATR.

2.5.4.10 Subsurface Instrumentation. No subsurface instrumentation exists at the ATR.

2.5.4.11 Construction Notes. The ATR facility was constructed in the late 1960s, and construction notes are not available.

2.5.5 Stability of Slopes

Slopes in the ATR area are very small, a few feet per mile at most, and do not pose a threat for instability or landslides.

2.5.6 Embankments and Dams

There are no embankments or dams for the flood control or containment of cooling water at the ATR.

2.5.7 Volcanism

2.5.7.1 Introduction. The regional-tectonic framework of ESRP volcanism was previously introduced in Section 2.5.1. Basaltic and rhyolitic volcanism has affected the ESRP for about the past 10 Ma and has continued into geologically recent time. Historical eruptions have not occurred on the ESRP, but lava flows issued as recently as 2,100 years ago from the Great Rift, about 15.5 mi (25 km) southwest of the INEEL. Other Holocene basaltic lava fields near the southern INEEL boundary are nearly as young, and range from about 5,000 and 13,000 years in age (Kuntz et al. 1986). Many basaltic and three rhyolitic vents located within the present INEEL boundary erupted between about 0.2 and 1.2 Ma (Kuntz et al. 1990). For these reasons, an assessment of volcanic hazards is warranted at ATR based on the record of past volcanism in the region.

Table 2.5-1 lists hazards associated with ESRP basaltic volcanism, based on interpretation of the ESRP eruption products, and by analogy with historical observations of rift-zone volcanism in Hawaii and Iceland. The most significant hazard is inundation or burning of facilities by lava flows. Such flows vary greatly in volume and may cover a few to 155 or more mi^2 (400 km^2) (Kuntz, Covington, and Schorr 1992). On gentle terrain such as the ESRP, lava flows would generally move down-slope at a few meters per minute. Large lava flows on the ESRP seldom exceed 18.7 mi (30 km) in length, and most are 7.5 mi (< 12 km) long. Borehole investigations and outcrop studies indicate that most ESRP basaltic lava flows are 22 ft (< 10 m) thick, and taper to several-meter thickness at flow edges. Therefore, they are unlikely to surmount major topographic or constructed obstacles. The general topography and vent locations of the INEEL area suggest that future lava flows will most likely erupt from vents along the axial volcanic zone, or at the intersections of that zone with the volcanic-rift zones from which they could flow toward the central INEEL and ATR.

In summary, from Appendix A, the most probable areas of future basaltic volcanism and ground deformation, are the axial volcanic zone and the Arco volcanic-rift zone. Within this context, the Volcanism Working Group (1990) estimated the conditional probability of basaltic volcanism affecting a south-central INEEL site, including the ATR, as being $< 10^{-5}$/yr. Using an event tree analysis (Hackett and Khericha, 1993) show that for a 3×10^{-5} per year probability of eruption at a random location in the Arco volcanic rift zone or the Axial volcanic zone, the frequency of lava inundation at ATR ranges from 6.0×10^{-8} to 2.4×10^{-8} per year.

Table 2.5-1. Hazards associated with basaltic volcanism on the Eastern Snake River Plain. Entries are listed from highest to lowest relative hazard.

Phenomenon	Relative frequency	Size or area of influence	Comments
Lava flow	Common	3 km^2 to 400 km^2 area of coverage, based on sizes of ESRP lava fields of the past 15 Ka.	Significant hazard; typical basaltic phenomenon; lava from fissures or shield volcanoes may inundate large areas downslope of vents.
Ground deformation: fissuring, faulting and uplift	Common; associated with virtually all shallow magma intrusion, and eruption	Fissuring could affect areas of 2 × 20 km; minor tilting and broad uplift in areas to 5 × 20 km.	Significant hazard; due to shallow dike intrusion; "dry" intrusion may occur without lava flows.
Volcanic earthquakes	Common; associated with magma intrusion before and during eruptions	Maximum M = 5.5 and most events M < 4; ground vibration may affect facilities within 25 km.	Moderate hazard; swarms of shallow earthquakes (< 4 km focal depth) occur as dikes propagate underground.
Gas release	Common; associated with fissuring and lava effusion	May affect few km^2.	Low hazard; local plume of corrosive vapor, downwind from eruptive vent or fissure.
Tephra fall	Rare	Restricted to near-vent areas; may affect several km^2 downwind.	Low hazard; basaltic eruptions are inherently nonexplosive and may form small tephra cones but little fine ash to be carried downwind.
Base surge (ground-hugging blast of steam and tephra)	Rare	Effects limited to radius of several km from vent; < 10 km^2 area.	Low hazard; steam explosions due to interaction between ascending magma and shallow groundwater; water table too deep under most of INEEL.
Tephra flow (ground-hugging flow of hot, pyroclastic material)	Extremely rare	Near vent; may affect area < 1 km^2.	Very low hazard; as per tephra fall but affecting even smaller areas.

2.6 Environmental Analysis

The INEEL is within a cool desert ecosystem characterized by shrub-steppe communities. The INEEL contains relatively large areas of grazed and ungrazed land that contain several types of plant communities (e.g., sagebrush, grassland, juniper, and wetland). The INEEL is dominated by big sagebrush (*Artemisia tridentata*) and, to a much lesser extent, green rabbitbrush (*Chrysothamnus viscidiflorus*), other small shrubs, and various species of bunchgrass. In general, the vegetation and habitats around the TRA are typical of those developed on lava flows over much of the central portion of the INEEL.

The INEEL supports numerous animal species including 1 amphibian, 9 reptile, 184 bird, and 37 mammal species. The INEEL supports several species of game animals, but DOE allows hunting on the Site only near the northern boundary. Important game animals present include waterfowl, sage grouse, pronghorn antelope, elk, and mule deer. The INEEL Site supports a relatively large portion of Idaho's pronghorn population during winter. The INEEL provides winter and nesting habitat for sage grouse and is an important wintering area for several species of raptors.

No critical habitat for threatened or endangered species has been identified at the INEEL. The bald eagle and the peregrine falcon are the only threatened or endangered species that have been observed at the INEEL.

Flora and fauna of the INEEL and around TRA are typical of those generally found throughout sagebrush habitats of southeastern Idaho.

2.7 References

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CHAPTER 2 APPENDIX A

GEOLOGY AND SEISMOLOGY OF THE INEEL AND REGION

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ACRONYMS/ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ATR	Advanced Test Reactor
AVZ	Axial Volcanic Zone
BLWN	band-limited-white-noise
CTB	Centennial Tectonic Belt
DBE	design basis earthquake
ESRP	Eastern Snake River Plain
ETR	EngineeringTest Reactor
H-EB	Howe-East Butte
INEL	Idaho National Engineering Laboratory
INEEL	Idaho National Engineering and Environmental Laboratory
ISB	Intermountain Seismic Belt
Ka	thousand years ago
LR-HHA	Lava Ridge—Hell's Half-Acre
Ma	million years ago
MBMG	Montana Bureau of Mines and Geology
MMI	Modified Mercalli Intensity
MRE	most recent event
NRC	U.S. Nuclear Regulatory Commission
NRF	Naval Reactors Facility
NPR	New Production Reactor
OBE	operating basis earthquake
PGA	peak ground acceleration
RWMC	Radioactive Waste Management Complex
SMA	strong-motion accelerograph
SRP	Snake River Plain
SSE	safe shutdown earthquake
TRA	Test Reactor Area
USBR	United States Bureau of Reclamation
USGS	U.S. Geological Survey

UTC Universal Time Coordinated (Greenwich Mean Time)
UUSS University of Utah Seismograph Stations

VRZ volcanic rift zone

WCC Woodward-Clyde Consultants
WCFS Woodward-Clyde Federal Services

Appendix 2A

Geology and Seismology of the INEEL and Region

This section presents the fundamental geologic and seismic characteristics of the Idaho National Engineering and Environmental Laboratory (INEEL) and the surrounding region. Included are the geologic history; the lithologic, stratigraphic, and structural geologic conditions; the tectonic structures; and the geologic features and conditions that could affect facility structures.

2A.1 Fundamental Geologic And Seismic Information

The subsections that follow describe the regional and site-specific geology and discuss their geologic history.

2A.1.1 Regional Geology

2A.1.1.1 Physiographic Provinces. The INEEL is located on the Eastern Snake River Plain (ESRP). The ESRP is the eastern part of the Snake River Plain physiographic province (Figure 2A.1-1), a broad, low-relief basin floored with basaltic lava flows and terrestrial sediments. The Snake River Plain physiographic province is approximately 49 to 60 mi (80 to 100 km) wide and over 348 mi (560 km) long, extending in a broad arc from the Idaho-Oregon border on the west to the Yellowstone Plateau on the east. It transects and sharply contrasts with the mountainous country of the Northern Basin and Range Province and the Idaho Batholith.

Surface elevations on the Snake River Plain decrease continually and gradually from about 6,600 ft (2,000 m) near Yellowstone to about 2,100 ft (650 m) near the Idaho-Oregon border. Summits of mountains surrounding the Plain range up to 12,100 ft (3,700 m) in elevation, producing a maximum elevation contrast of about 7,550 ft (2,300 m).

The northern Basin and Range Province, which bounds the ESRP on the north, is composed of north- to northwest-trending mountain ranges [with peaks up to 12,100 ft (3,700 m) high] separated by intervening basins 4,600 to 5,250 ft (1,400 to 1,600 m) in elevation filled with terrestrial sediments and volcanic rocks. Individual mountain ranges in the vicinity of the Snake River Plain are up to 125 mi (200 km) long and 19 mi (30 km) wide. They are sharply separated from the intervening basins by late Tertiary to Quaternary normal faults (Section 2A.2.2.1). The basins are 3 to 12.5 mi (5 to 20 km) wide and grade onto the ESRP.

The Yellowstone Plateau, at the northeastern end of the ESRP, is a high volcanic plateau underlain by Pleistocene (less that 2 million years old) rhyolitic volcanic rocks (Figure 2A.1-1). Its elevation [about 6,890 to 8,530 mi (2,100 to 2,600 m)] is significantly higher than that of the ESRP, but not as high as the mountain summits of the northern Basin and Range Province. The Plateau is characterized by extremely high heat flow from the surface; extremely high temperatures at shallow depths; abundant hot-spring, fumarolic, and geyser activity; and landforms controlled by thick rhyolitic lava flows. These characteristics reflect the recent volcanic activity in the area [2 million years ago (Ma) to several tens of thousands of years ago (Ka)].

The Idaho Batholith, which adjoins the northern margin of the central Snake River Plain, is a large area of irregular mountainous terrain with peaks ranging in elevation from 7,870 to 12,100 ft (2,400 to 3,700 m). Streams dissecting the area usually have dendritic drainage patterns that reflect the homogeneity of the underlying granitic rocks that comprise the batholith.

The four physiographic provinces described here (the ESRP, the northern Basin and Range Province, the Yellowstone Plateau, and the Idaho Batholith) also correspond to tectonic or seismotectonic provinces. Each province has a different seismogenic potential that is determined by its intrinsic tectonic processes. The nature and seismogenic potential of these tectonic processes is discussed in Sections 2A.2.2 and 2A.2.4.7 through 2A.2.4.9.

2A.1.1.2 Geologic History

2A.1.1.2.1 Paleozoic, Mesozoic, and Early Cenozoic History—The mountains northwest of the ESRP near the INEEL are composed of thick sequences of sedimentary strata that date from the late Precambrian through Pennsylvanian ages (Table 2A.1-1). The Precambrian through lower Ordovician rocks are mostly clastic (shales and quartzites); the upper Ordovician through Pennsylvanian rocks are mostly carbonates (dolomites and limestones). They occur within westward-dipping thrust sheets that formed during east-directed Mesozoic compressional tectonism (Link et al. 1988; Skipp and Hait 1977).

During the Paleozoic and Mesozoic Eras, continental-shelf carbonates (limestones and dolomites) were deposited along the western margin of the North American continent in a north-trending belt that included southeastern Idaho (Oldow et al. 1989). Thrust faulting that accompanied deposition of these sediments produced three orogenies: in Paleozoic time (Antler orogeny), at the Paleozoic/Mesozoic boundary (Sonoma orogeny); and in Mesozoic/Cenozoic time (Sevier/Laramide orogenies). This thrust faulting produced the Idaho/Wyoming thrust belt (the Overthrust Belt) that extends through eastern Idaho (Figure 2A.1-2). In early Cenozoic time, eastward-directed thrust faults and belts of deformation may have continued uninterrupted through southeast Idaho.

Large volumes of granitic rock emplaced by igneous intrusion into the upper crust during Mesozoic and early Cenozoic thrusting produced the Idaho Batholith in central Idaho (Figure 2A.1-2). Subduction of the Pacific plate beneath the North American plate caused large-scale melting of lithospheric rocks all through the western Cordillera. In addition to the Idaho Batholith, the Sierra Nevada Batholith and other large granitic intrusive bodies were formed during this time.

In the early Cenozoic, northwest-southeast-directed extension produced the northeast-trending Trans-Challis fault zone and the associated Custer and Panther Creek grabens (Figure 2A.1-3). Accompanying volcanism caused caldera subsidence along the trend of the grabens. Volcanic rocks of the Challis volcanic field, which covers much of south-central Idaho adjacent to the northwestern margin of the ESRP, were erupted from sources along the Trans-Challis zone and elsewhere in south-central Idaho.

2A.1.1.2.1.1 Late Cenozoic and Quaternary History of the Yellowstone-ESRP Volcano-Tectonic Province

2A.1.1.2.1.1.1 The Yellowstone Hotspot—The processes that caused development of the ESRP began about 17 Ma, when a rising plume of anomalous hot rocks in the earth's mantle (the Yellowstone hotspot) first impinged on the base of the lithosphere. Because the mantle plume is rooted deep in the earth (probably at the mantle-core boundary), it has remained stationary while plate tectonic processes have shifted the lithosphere and crust (North American plate). Seventeen million years ago, the North American plate was positioned so that the area now located in north-central Nevada was directly above the hotspot. As plate tectonic activity has moved the plate southwestward at about 1.38 in./yr (3.5 cm/year), the hotspot has left its distinctive effects strung out in a band extending from its present position at Yellowstone National Park to north-central Nevada.

The effects of the hotspot on the lithosphere and crust have been profound. Two types of large-scale melting have occurred. The first type was melting of the hot mantle material in the rising plume itself, which generated basaltic melts (magmas) that migrated to mid-crustal levels [about 12 mi (20 km) depth]. This melting resulted from a decrease in pressure on high-temperature mantle material as it moved from great depth. The second type was melting of mid-crustal rocks, which produced granitic melts that migrated upward to near-surface reservoirs and caused widespread explosive and effusive rhyolitic volcanism typical of that at Yellowstone National Park. This melting resulted from heating of mid-crustal rocks by the much hotter basaltic magmas that rose from the mantle plume.

The effects of the hotspot that can be observed at the surface of the earth today include widespread, large-volume sheets of rhyolitic volcanic rocks emplaced by explosive processes; large-volume rhyolitic lava flows; and calderas from which the rhyolitic volcanic rocks erupted. The calderas are typically 19 to 43 mi (30 to 70 km) across. They resulted when the roofs of shallow magma chambers collapsed as voluminous explosive eruptions occurred. As the roofs fell into the evacuated magma chambers, the resulting depressions were filled with thick sequences of rhyolitic volcanic rocks. Other effects of the hotspot are elevated topography in the area directly over the hotspot (from buoyant effects of the hotspot), and the basin (depressed topography) of the ESRP caused by subsidence as plate motion moved volcanic highlands southwestward from the hotspot.

A string of such calderas and volcanic fields has formed in the wake of the Yellowstone hotspot as the North American plate has migrated to the southwest across it (Figure 2A.1-4). Surface subsidence resulting from crustal cooling along this string of volcanic fields formed the ESRP.

Another way to view the progression of hotspot-related rhyolitic volcanism is in a plot of age of volcanic rocks versus distance from Yellowstone along the ESRP (Figure 2A.1-5). This diagram shows that the beginning of rhyolitic volcanism becomes younger toward Yellowstone and that basaltic volcanism has covered most of the rhyolitic volcanic fields in the ESRP.

2A.1.1.2.1.2 Modifications to Crustal Structure Resulting from Hotspot Processes—In addition to causing the large-scale melting and volcanism described in previous sections, melting processes associated with the hotspot modified the crust beneath the ESRP significantly. Crystallization of large volumes of basaltic magma in the mid-crust produced a roughly 6-mi (10-km) -thick lens of anomalous dense rock that transmits seismic waves faster than the material above or below (Figure 2A.1-6). The weight that this material added to the crust, along with the contraction from cooling after passing over the hotspot, has caused the ESRP to subside in elevation by about 1.2 mi (2 km) during the past 4 million years.

2A.1.1.2.1.3 Basalt Volcanism and Sedimentation in the Subsiding ESRP Basin—The subsidence of the ESRP has produced an elongated northeast-trending basin in which basalt lava flows and deposits of sedimentary material have accumulated to a total thickness of 0.6 to 1.2 mi (1 to 2 km). The basalt lava flows were generated by residual heat in the upper mantle beneath the ESRP, and rose to the surface to erupt into the subsiding basin.

Sedimentary materials have formed interbeds between lava flows. The sediments are composed of (a) fine-grained silts that were deposited by wind action; (b) silts, sand, and gravels deposited by streams such as the Big Lost River; and (c) clays, silts, and sands deposited in lakes such as Mud Lake and its much larger ice-age predecessor, Lake Terreton.

The accumulation of these two types of rocks in the ESRP has resulted in the sequence of interlayered basalt lava flows and sedimentary interbeds because volcanism is a sporadic process. During the long periods of quiescence between volcanic periods, sediments accumulated to thicknesses of < 3 ft to > 200 ft (< 1 m to > 60 m). In contrast, several lava flows commonly accumulated to thicknesses reaching several tens of meters during short periods of volcanic activity.

2A.1.1.2.2 Basin and Range Tectonic Activity—The Basin and Range Province of the western United States (Figure 2A.1-1) is a region of extending crust, high elevations, high heat flow, and extensive Cenozoic volcanism (Eaton 1982). The north to north-northwest trends of normal faults and mountain ranges in the Basin and Range Province, as well as various types of in situ stress determinations (Zoback and Zoback 1989), show that the area is subjected to tension directed east-west to northeast-southwest. In the northern Basin and Range Province, which is transected by the ESRP, the extension produces north-trending normal faults and mountain ranges on the southern side of the ESRP and northwest-trending faults on the northern side. The mountain ranges are caused by a process called block faulting. As extension stretches the area, the brittle upper crust (upper 6 to 10 mi [10 to 16 km]) can respond only by breaking into blocks that rotate slightly along the faults between to produce long, narrow mountain ranges with intervening basins (valleys).

The rugged topography and high elevations characteristic of these mountain ranges terminate at the margins of the ESRP and give way to the relatively flat and low-lying topography characteristic of the Plain. The activity on the normal faults that bound the ranges must terminate at the plain margins, or the mountain ranges would continue across the Plain.

2A.1.2 Site Geology

2A.1.2.1 Topographic and Physiographic Description

2A.1.2.1.1 INEEL Area—The topographic relief of the ESRP is subdued with respect to the surrounding Basin and Range Province. Total relief of the floor of the ESRP in the INEEL area is about 656 ft (200 m), ranging from 4,790 ft (1,460 m) at Big Lost River Sinks to about 5,410 ft (1,650 m) on the northeast-trending axial ridge of the Plain (Figure 2A.1-7). Four prominent buttes occur along the axial ridge of the ESRP, and they stand noticeably higher than the Plain. Big Southern Butte (7,572 ft [2308 m]), Cedar Butte (5,827 ft [1776 m]), Middle Butte (6,391 ft [1948 m]), and East Butte (6,571 ft [2003 m]) stand 390 to 2,130 ft (120 to 650 m) above the axial ridge.

The axial ridge, known as the Axial Volcanic Zone (AVZ) (Hackett and Smith 1992), constrains the Snake River to the southeastern edge of the ESRP and causes rivers from the mountains to the north of the ESRP to drain into closed basins (sinks). The most prominent example is the Big Lost River, which flows onto the ESRP near Arco, turns northeastward in the southwestern part of INEEL, and flows north to the Big Lost River Sinks in the northern part of INEEL. The Little Lost River and Birch Creek also empty into sinks (playas) in the northern part of INEEL.

Although the ESRP's large-scale topography is rather subdued, its detailed topography is very rough and uneven because of the character of the numerous basalt lava flows that make up the surface. The topography is characterized by lobate forms, numerous steep-walled closed depressions and mounds, and anastomosing fissures. Erosional processes have not established classic drainage patterns. Streams tend to be intermittent, wandering, and blind as they follow lava-flow contacts and lava channels, commonly ending in closed depressions.

In many areas, the lava flow topography is softened by deposition of windblown silt into fissures and depressions. In some areas, the silt deposition has been so great that the topography is dominated by dune forms and rolling terrain with little or no basalt at the surface. Development of intermittent lakes and ponds in many closed depressions in the lava-flow surface has resulted in deposition of fine silts and clays, producing small, flat-floored playas (Nace et al. 1975).

2A.1.2.1.2 Advanced Test Reactor Site—The Advanced Test Reactor (ATR) site is located in a flat-lying area near the Big Lost River in the southern part of the INEEL. It is underlain by about 49 ft (15 m) of Big Lost River alluvial silts, sands, and gravels, which lie on an alternating sequence of basalt lava flows and interbedded sediments extending to a depth of about 1,970 to 2,300 ft (600 to 700 m). Landforms in the vicinity of ATR consist of braided channels (some abandoned) of the Big Lost River to the east and southeast of the site, and irregular flow lobes of basalt lavas to the west of the site.

2A.1.2.2 Stratigraphy and Areal Geology

2A.1.2.2.1 INEEL Area Stratigraphy—Table 2A.1-2 summarizes the thickness, age, distribution, characteristics, and origin of stratigraphic units on and near INEEL. During the past 4 million years, the ESRP (including the INEEL area) has experienced volcanic activity, mostly in the form of mild outpourings of basaltic lava flows. Vents for the basaltic volcanism are concentrated in northwest-trending volcanic rift zones (VRZs) and along the Axial Volcanic Zone (Hackett and Smith 1992; Kuntz, Covington, and Schorr 1992) (Figure 2A.1-8). Sediments deposited by wind action, streams, and lakes have also accumulated in the ESRP, concurrently with the basaltic lava flows. Lithologic logs of four INEEL deep holes [> 2,000 ft (> 610 m) deep] (Figure 2A.1-9) (Doherty 1979a, 1979b; Doherty, McBroome, and Kuntz 1979; Hackett and Smith 1992) and hundreds of more shallow drill holes (Nace et al. 1975) show that an interlayered sequence of basalt lava flows and poorly consolidated sedimentary interbeds, known as the Snake River Group (Malde 1991), occurs to depths of 0.6 to 1.2 mi (1 to 2 km) beneath the INEEL. This sequence is underlain by a large, but unknown, thickness of Late Tertiary rhyolitic volcanic rocks.

Sedimentary interbeds within the Snake River Group have diverse origins. The interbeds include silts deposited by wind action; silts, sands, and gravels deposited by streams such as the Big Lost River; and clays, silts, and sands deposited in playas and lakes such as Mud Lake and its much larger Pleistocene predecessor, Lake Terreton. All of these sedimentary processes continue to operate today, producing surficial deposits of alluvial, eolian, and lacustrine/playa origin. The interlayering of unconsolidated and poorly consolidated sediments within the basalts has several implications for facilities at INEEL.

- The interbedded sediments are composed mostly of fine-grained materials (silts and clays), which have very low permeability and high absorption capabilities (Nace et al. 1975). Therefore, they retard the downward migration of water and contaminants to the water table.

- The low permeability of the sedimentary interbeds commonly causes localized perched water zones beneath some INEEL infiltration ponds (Hull 1989) and beneath natural infiltration/recharge zones such as the Big Lost River channel and sinks at flood stage (Bennett 1990).

- The sediments can be confining or semiconfining layers in the aquifer, thereby affecting the way that water (and contaminants) move vertically and horizontally.

- The alternating high and low seismic velocities associated with basalts and poorly consolidated sedimentary interbeds, respectively, causes greater than normal attenuation of earthquake strong ground motions (WCC 1990, 1992a).

- The unconsolidated sands and clays intercalated within the hard, brittle basalts contribute to very difficult drilling and downhole geophysical logging conditions, increasing the expense and time necessary for developing exploratory drill holes and monitoring wells at the INEEL.

2A.1.2.2.2 INEEL Area Areal Geology—Surface rocks on and near the INEEL are mostly Quaternary basalt lava flows, the upper part of the Snake River Group, ranging in age from < 15,000 to > 730,000 years (Figure 2A.1-10) (Kuntz et al. 1990). A wide band of Quaternary mainstream alluvium extends along the course of the Big Lost River from the southwestern corner of the INEEL to the Lost River Sinks area in north-central INEEL. Lacustrine (lake) deposits of clays and sands deposited in Ice Age Lake Terreton occur in the northern part of the INEEL. Beach sands deposited at the high stand of Lake Terreton were reworked by winds in late Pleistocene and Holocene time to form large dune fields (eolian deposits) in the northeastern part of the INEEL (Nace et al. 1975; Scott 1982). Several Quaternary rhyolite domes occur along the Axial Volcanic Zone near the south and southeast borders. Paleozoic limestones, late-Tertiary rhyolitic volcanic rocks, and large alluvial fans occur in limited areas along the northwest margin of INEEL.

2A.1.2.2.2.1 Vertical and Horizontal Facies of Basalt Lavas—An idealized section showing distribution of vertical and horizontal facies variation in ESRP basalt lava flows is shown in Figure 2A.1-11. From bottom to top, basalt lava flows typically are composed of a basal rubble zone, a lower vesicular zone, a massive columnar jointed zone, an upper vesicular and fissured zone, and a cap of platy jointed crust.

The near-vent facies of lava flows is typified by thin, vesicular, platy flows (shelly pahoehoe). Also, pyroclastic ash and breccia layers are commonly interleaved within the thin flow layers. With distance from the vent, the shelly pahoehoe grades rapidly into the layered facies structure, described above, which typifies the medial and distal portions of the lava flow (Figure 2A.1-11). Deflation pits, in which solidified crust has subsided over areas where lava has drained away, are common throughout the flow but more numerous near the terminus.

2A.1.2.2.2.2 Sediment Facies—Sediments of diverse origins occur covering and interbedded with basalts of the ESRP. Surface lava flows throughout INEEL and surrounding regions are covered by varying thicknesses of windblown silt (loess). Alluvial sands and gravels are common along the Big Lost River channel through the INEEL, and lacustrine clays deposited in Pleistocene Lake Terreton are common in the northern and northeastern part of the INEEL. Because the sedimentary depositional processes operating in the geologic past are similar to those operating today, the same types of sediments make up the interbeds in the subsurface.

2A.1.2.2.3 ATR Site Stratigraphy—At the ATR site, the surficial sediments (Big Lost River alluvium) vary from 30 to 50 ft (9 to 15 m) thick, and consist mostly of gravel, gravelly sands, and sands. In some locations, a thin [0 to 6.6 ft (0 to 2 m) thick] layer of clay and silt underlies the gravelly alluvium, forming a discontinuous low-permeability layer just above the basalt bedrock (Hull 1989). That low-permeability layer caused infiltration water from the warm-waste pond no longer used at TRA to create a local perched water body that extends a short distance inside the southeast corner of the TRA fence (Hull 1989). These perched water bodies have the potential to cause soil instability because the water saturates the sediment. However, because the majority of the surficial sediments are gravelly and the area beneath ATR does not have a perched water body, there is low potential for liquefaction or other soil instabilities in the ATR area.

Sedimentary interbeds within the Snake River Group beneath the ATR site are composed mostly of silts, clayey silts, and sandy silts. Cross sections showing the positions and thickness of interbeds are presented by Anderson (1991). Additional, more detailed cross sections of the area beneath the warm waste pond are given by Hull (1989). These sections show that a thick interbed occurs at a depth of about 150 to 200 ft (45 to 60 m) below the surface. Hull (1989) presents evidence that this zone is really a fractured basalt with silty infillings in the fractures. Several more interbeds are shown to occur between 200 and 600 ft (60 and 180 m), and they presumably occur throughout the entire thickness of the basalt section [about 2,300 ft (700 m) in this area] because they are present in Deep Exploration Well INEL-1 (Figure 2A.1-9), which is located about 2.5 mi (4 km) north of ATR.

2A.1.2.2.4 ATR Site Areal Geology—The basalts at the surface near the ATR site, just to the west and north of the site (Figure 2A.1-10) and those lying beneath the surficial sediment layer, are 400,000 to 730,000 years old and flowed from vents located about 10 mi (16 km) west and 1 mi (1.6 km) north of the site (Kuntz et al. 1990). Basalt flows beneath those at the surface are older and range in age to as much as ~4.3 million years at the base of the basalt sequence (Hackett and Smith 1992). These basalts have accumulated in a basin (the ESRP) that has continuously subsided at a rate of about 1 mm/yr since passage of the Yellowstone hotspot about 4.3 Ma.

In contrast to vent locations for surface basalts, the source vents for basalts in the subsurface are poorly known. It is clear that some of the subsurface basalts (Anderson 1991) were erupted from the volcanic vent at AEC Butte about 0.9 mi (1.5 km) north of TRA. Others came from vents in the Arco VRZ, and possibly the Lava Ridge–Hell's Half-Acre (LR-HHA) VRZ (see Sections 2A.2.2 and 2A.6).

Basalts in the TRA area and throughout the ESRP are olivine tholeiites. They are mostly porphyritic and contain up to 20% by volume phenocrysts of olivine and plagioclase. The groundmass is composed of olivine, plagioclase, clinopyroxene, magnetite, ilmenite, and minor amounts of apatite, glass, rutile, and oxidation products. An average of 78 chemical analyses (Leeman 1982) and ion exchange capacity on a fresh sample of basalt from the subsurface at INEEL (Nace et al. 1975) are presented in Table 2A.1-3.

2A.1.2.3 Structural Geology. No evidence of fault offset is seen in surficial sediments within 5 miles of ATR. Several northeast-trending lineaments of alluvial and eolian origin are apparent on aerial photographs of the area. In some cases, linear patterns of revegetation have developed on the margins of ancient range-fire scars and eolian redistribution of surficial materials following range fires has produced linear dune-forms along fire-scar margins (Morin-Jansen 1987).

A dense array of drill holes in the TRA area (shallow geotechnical holes, deeper groundwater monitoring wells, production wells, and injection wells) and several 40- to 50-ft deep excavations to bedrock have revealed no evidence of surface ruptures or displacements in the near-surface basalt lava flows. Geologic correlations among wells and drill holes based on lithologic, geophysical, and geochemical characteristics, and based on radiometric ages of lava flows of many of these holes show no evidence of faulting within about 300 ft of the surface. However, paleomagnetic inclinations and K/Ar ages of lave flows in coreholes (Champion, Lanphere, and Kuntz 1988; Lanphere, Champion, and Kuntz 1993) have been interpreted to suggest that a graben or downwarp exists at depths of greater than approximately 300 ft (Smith, Hersley, and Jackson 1996). In contrast, geochemical correlation and correlation based on gamma logs of wells and coreholes suggest that lavas and interbeds are essentially flat beneath the area and that no faulting is necessary to explain the positions of stratigraphic units (Anderson 1991; Anderson et al. 1996). A summary of the arguments for and against these opposing interpretations is given by Smith, Hersley, and Jackson (1996). Even if a graben or fault or fold exist in the area at depths below 300 ft, it is not likely to be active in today's stress field and its age of > 300,000 years shows that it would probably not be considered "capable" by the NRC definition (Smith, Jackson, and Hackett 1996).

Individual basalt lava flows have well-developed fissure sets that formed during emplacement of the lava. These fissures result from bending of the solidified lava crust as still-molten lava flows away, leaving deflated areas (Figure 2A.1-11). In addition, post-solidification cooling joints developed in the lava flows, usually producing columnar joints with polygonal patterns. These emplacement- and cooling-related fissures and joints are ubiquitous in ESRP lava flows; they are not through-going tectonic structures, and they should not be viewed as indications of folding or faulting. They are separate and distinct from fissuring related to dike injection in volcanic rift zones (see Section 2A.2.2), which is a seismogenic process and has significance for safety of INEEL facilities.

2A.1.2.4 Geologic History Related to Regional Geologic History. The following is a summary of the geologic history at the ATR site and its relationship to regional geologic history:

- Eruption of voluminous, explosive silicic volcanic rocks during passage of the Yellowstone hotspot beneath the area at 6.5 to 4.3 Ma.
- Subsidence of the area as the hotspot passed with coeval eruption of basaltic lavas and accumulation of clastic sediments in the ESRP basin.
- Accumulation of about 2,300 ft (700 m) of interbedded basalts and sediments, the Snake River Group, from 4.3 Ma to present.
- Establishment of the Big Lost River's course through the central part of INEEL, probably within the last 0.5 to 1.0 million years. Upstream of the town of Arco the river's course is controlled by the positions of Basin and Range block fault mountain ranges; downstream the river's course is controlled by the positions of volcanic zones and the local slope of the surface of the ESRP.
- The last volcanism at ATR occurred > 400,000 years ago. Since that time, Big Lost River alluvium has accumulated to a depth of 30 to 50 ft (9 to 15 m).

2A.1.2.5 Engineering Geologic Conditions. The engineering geologic conditions of the site are presented in Section 2A.4. Letters near T-axes denote stress-indicator types: F, focal mechanism; G, fault-slip data; B, borehole breakouts; V, alignment of rift-zone features. Quality ranking (a through d) is from Zoback and Zoback (1989). Borehole-televiewer log of well INEL-1 showed no borehole breakouts, implying low deviatoric stresses within the ESRP (Moos, Barton, and Smith 1990). Faults are from Witkind (1975), Bond (1978), Anderson and Miller (1979), Stickney and Bartholomew (1983), and Hecker (1991). Volcanic-Fissure Sets are from Smith, Hackett, and Rogers (1989) and Kuntz, Covington, and Schorr (1992).

Table 2A.1-1. Summary of geologic units in the INEEL area.

Age	Unit, thickness, distribution	Physical character	Origin	Reference
	Eolian deposits, < 30m; covers much of INEEL, sand dunes in NE part and loess blankets elsewhere	Fine sand and silt, dune forms and blankets	Sand from Lake Terreton beaches; loess from glacial and alluvial fan deposits	Scott 1982; Nace et al. 1975
	Playa deposits, < 10 m, in sinks areas of northern INEEL and scattered small playas across INEEL	Poorly sorted clayey, sandy silt, some with high carbonate content	Deposited in closed depressions during wet periods	Scott 1982; Nace et al. 1975; Kuntz et al. 1990
Holocene to Late Pleistocene	Alluvium and alluvial fans, up to 30m, along streams and at base of mountains	Gravels with varying content of sand, silt, clay	Mainstream deposits along rivers, debris flows and intermittent stream deposits in fans	Scott 1982; Nace et al. 1975; Kuntz et al. 1990
	Holocene lava fields, up to 30m, in Great Rift and along Axial Volcanic Zone	Olivine tholeiite basalt to rhyodacite and ferroliatite	Erupted from VRZs and AVZ	Kuntz et al. 1986, 1990
	Lake Terreton deposits, < 50m, northeastern INEEL and Mud Lake area	Sandy and clayey light colored silt with lenses of pure sand, silt, and clay	Deposited in beaches, bars, and floor of Pleistocene Lake Terreton	Nace et al. 1975; Scott 1982
Pleistocene	Rhyolite Domes, along axis of ESRP	Porphyritic to aphric rhyolite flows and minor pyroclastics	Dome building eruptions at Big Southern, East, Middle, Cedar, and unnamed	Armstrong, Leeman, and Malde 1975; Hackett and Smith 1992; Hayden 1992
Late Tertiary to Quaternary	Snake River Group basalts and interbedded sediments ~1km, ESRP	Porphyritic to aphyric olivine tholeiites, fractured and vesicular, flows ~20 ft thick, interbedded with alluvial, eolian, and lacustrine sediments	Erupted from numerous vents and fissures in VRZs and AVZ of ESRP	Armstrong, Leeman, and Malde 1975; Malde 1991; Hackett and Smith 1992
Late Tertiary	Heise Volcanics, Several thousand meters, buried beneath ~1km of ESRP basalt and outcrops along margins of ESRP	Ash flow tuffs and rhyolitic lavas	Erupted from calderas buried beneath the basalts of the ESRP	Morgan 1988; Hackett and Morgan 1988; Hackett and Smith 1992
Late Precambrian to Paleozoic	Carbonate and clastic sedimentary rocks, > 5,000 m, mountains north and south ESRP	Quartzites overlain by middle to upper Paleozoic carbonates	Marine deposition followed by Mesozoic thrusting	Link et al. 1988; Skipp and Hait 1977.

Table 2A.1-2. Stratigraphic column for bedrock units in the Lost River and Lemhi Ranges (showing geologic ages, formation names, and stratigraphic thicknesses).

Period	Formation	Thickness (M)
Pennsylvanian	Snaky Canyon Formation	> 1,032
Mississippian	Arco Hills Formation	124
	Surrett Canyon Formation	260-335
	South Creek Formation	87-140
	Scott Peak Formation	620-635
	Middle Canyon Formation	180-240
	McGowan Creek Formation	60-150
Devonian	Three Forks Formation	50-80
	Jefferson Formation	0-> 135
	Unnamed carbonates	> 460
	Beartooth Butte Formation	0-135
Silurian	Laketown Dolomite	20-100
	Roberts Mountains Formation	780
	//////////////////////	//////////////
Ordovician	Fish Haven Dolomite	150-300
	Kinnikinic Quartzite	100-240
	Summerhouse Formation	30-360
Cambrian	//////////////////////	//////////////
	Formation of Tyler Peak	> 32
Precambrian	Wilbert Formation	190
	//////////////////////	//////////////
	Lemhi Group	> 50

//// Hiatus caused by nondeposition, erosion, or faulting

Table 2A.1-3. Chemical analyses and ion exchange capacities of Eastern Snake River Plain basalts.

Chemical Analysis			Ion Exchange Capacity[a]	
SiO_2	47.04		4 mm	0.98
TiO_2	2.70		2 mm	0.89
Al_2O_3	15.11		1 mm	0.73
FeO	13.51		0.5 mm	0.61
MnO	0.20		0.25 mm	0.86
MgO	7.65		0.125 mm	1.32
CaO	10.06		0.062	2.80
Na_2O	2.54		< 230 mesh	2.30
K_2O	0.61			
P_2O_5	0.58			

a. Meq per 100 g of rock

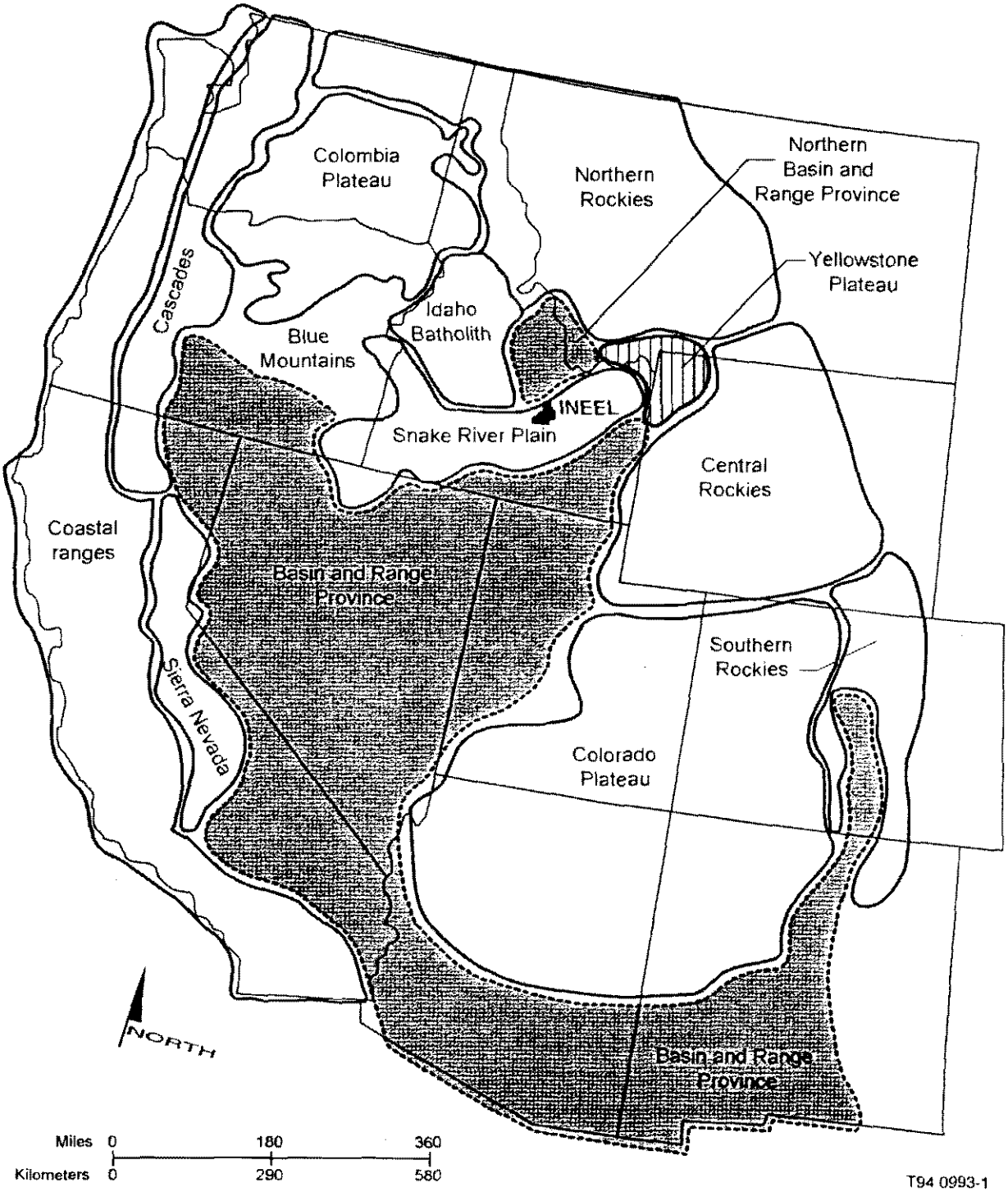


Figure 2A.1-1. Map of major physiographic and tectonic provinces of the western U.S. **(For Information Only)**

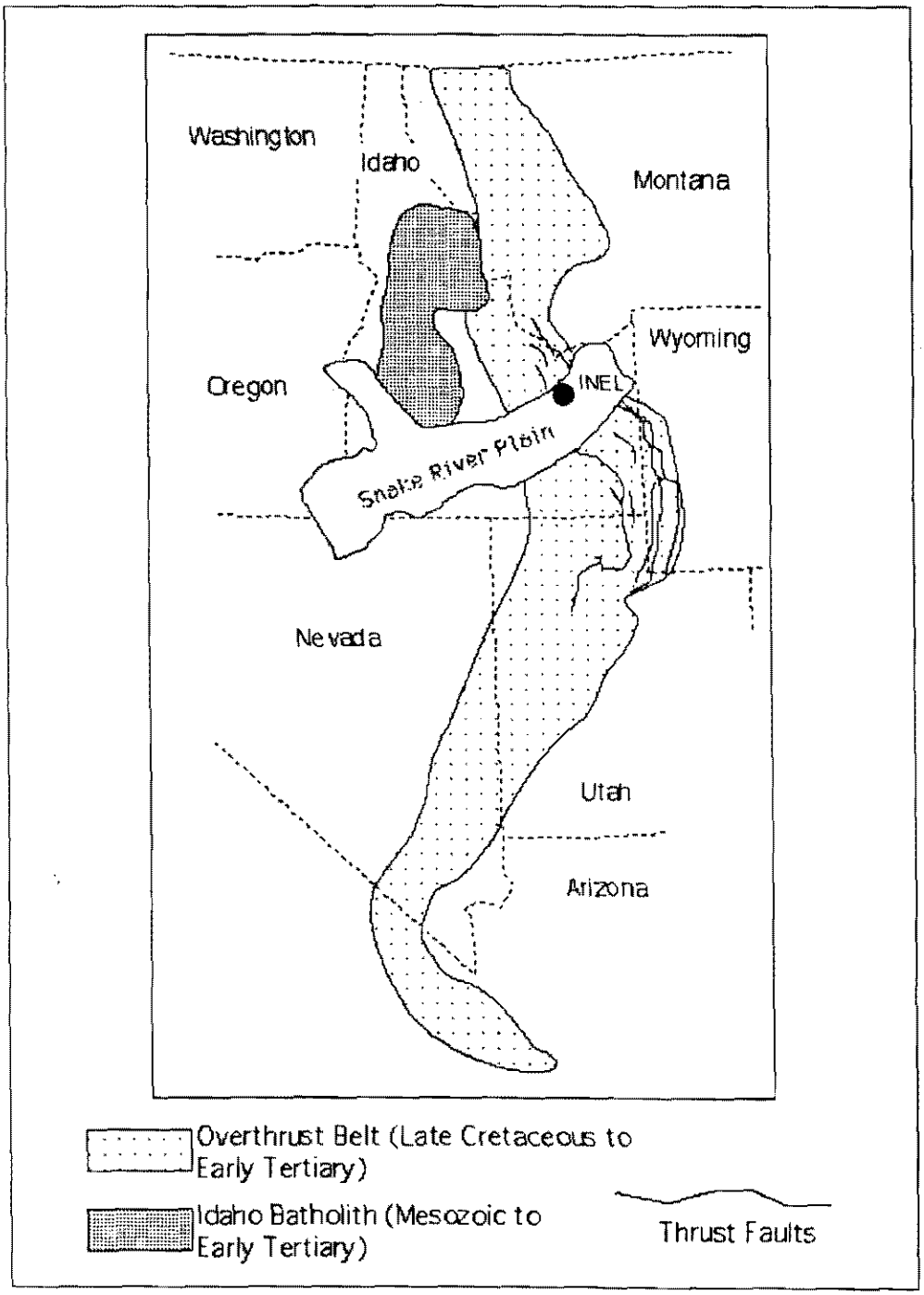


Figure 2A.1-2. Generalized map of the Overthrust Belt and Idaho Batholith. **(For Information Only)**

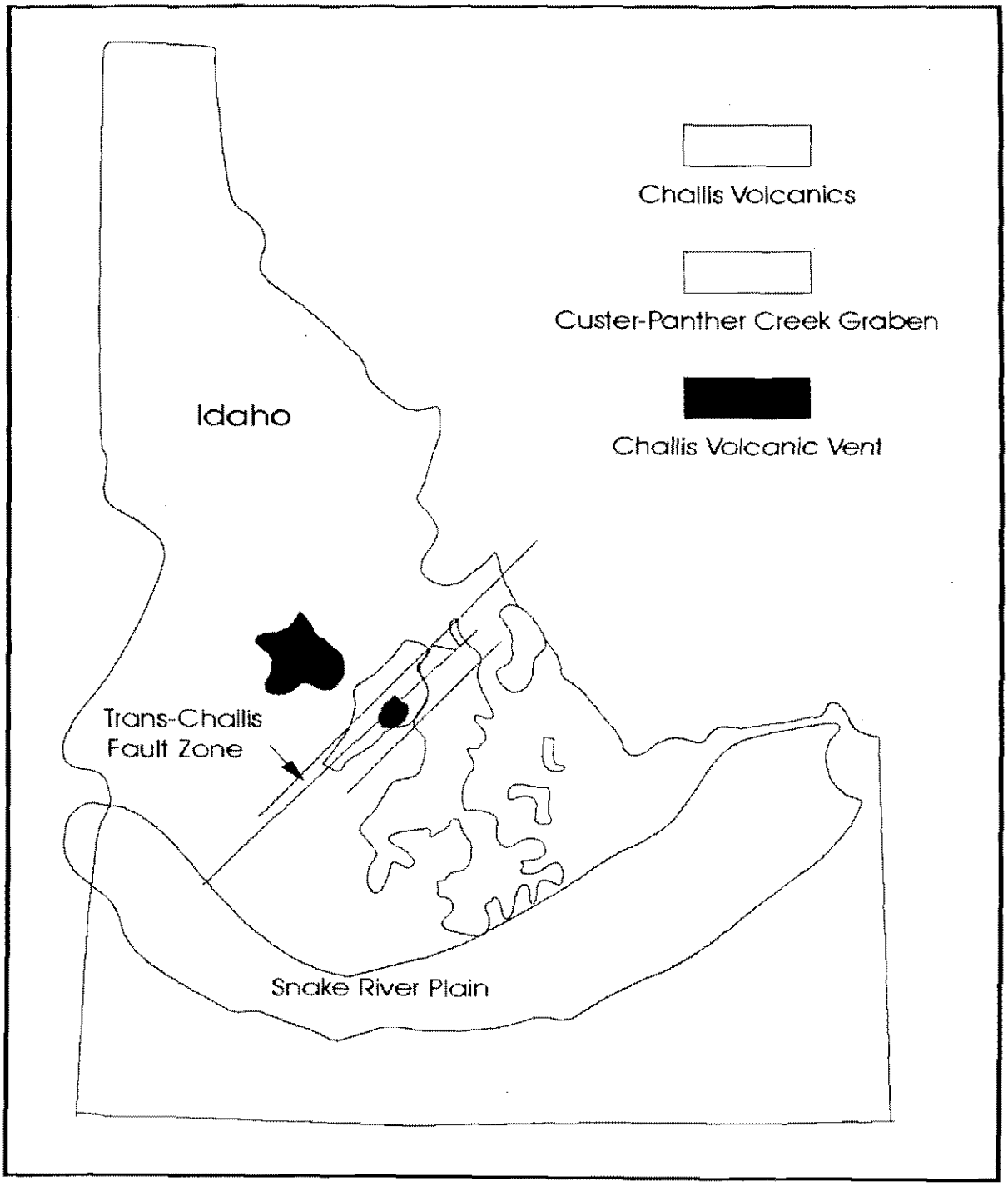


Figure 2A.1-3. Map showing the location of the Eocene Challis Volcanic Field and related tectonic elements. **(For Information Only)**

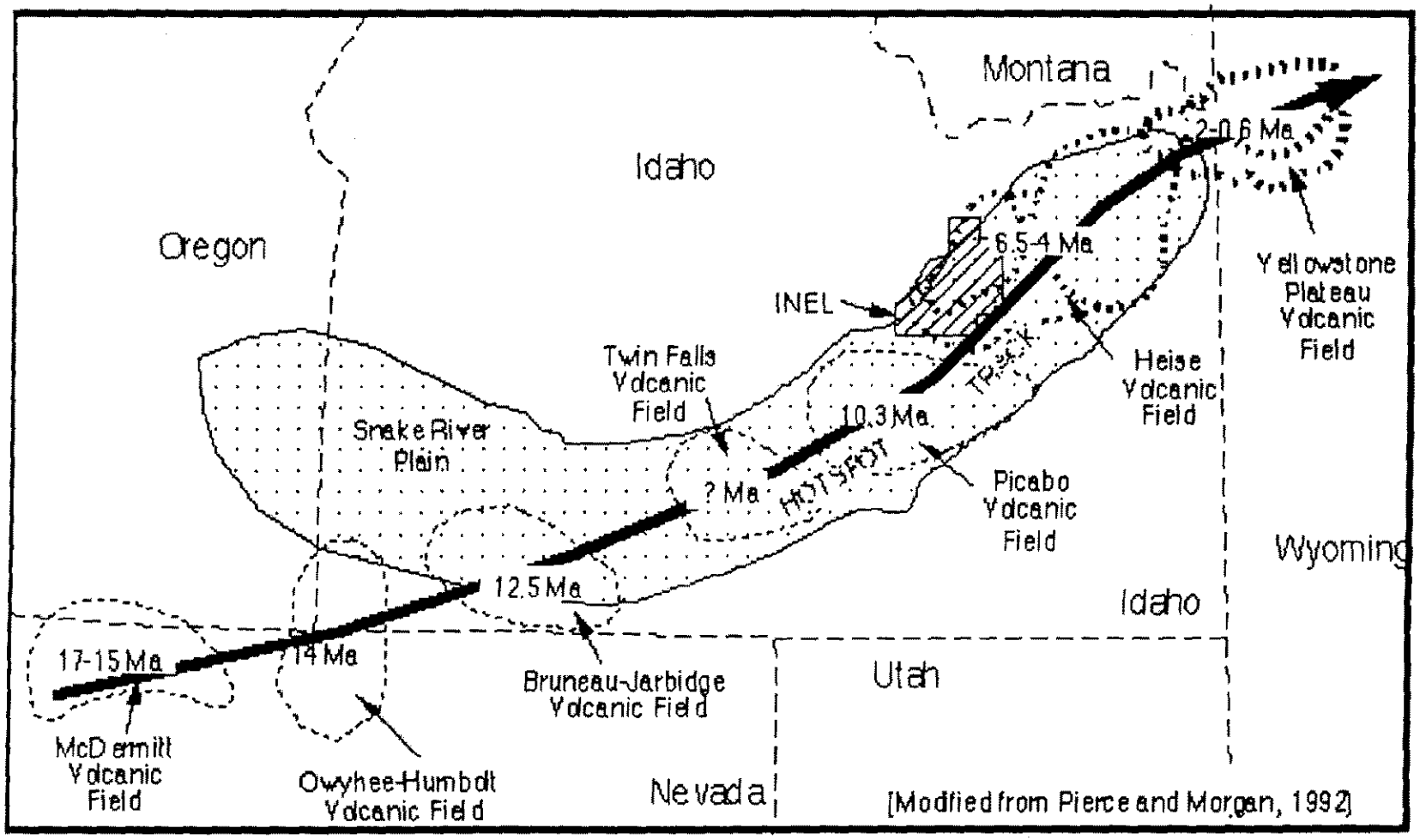


Figure 2A.1-4. Map of Southern Idaho and adjacent states showing the track of the Yellowstone Hotspot and names and locations of major silicic volcanic centers of the Snake River Plain. **(For Information Only)**

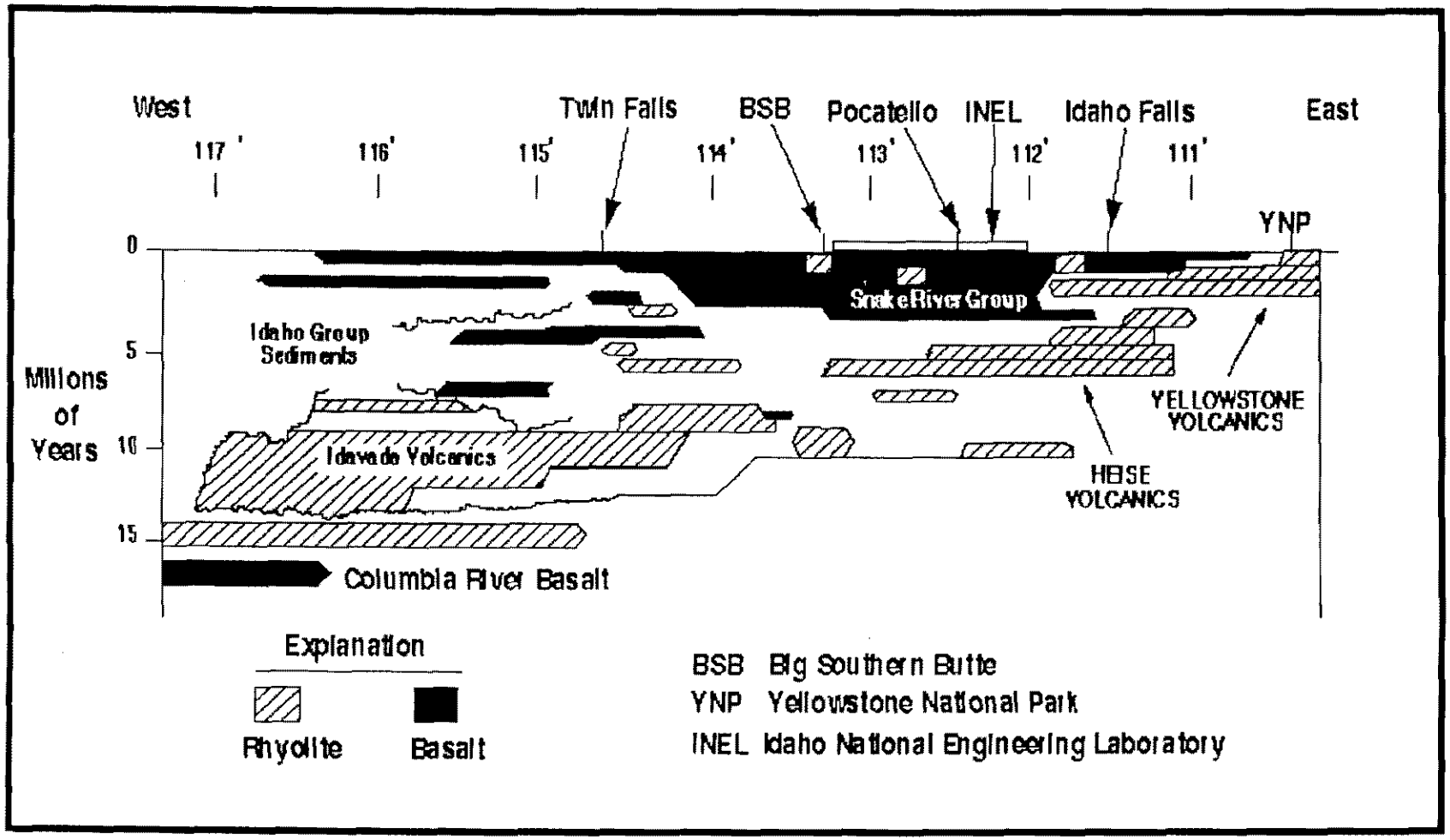
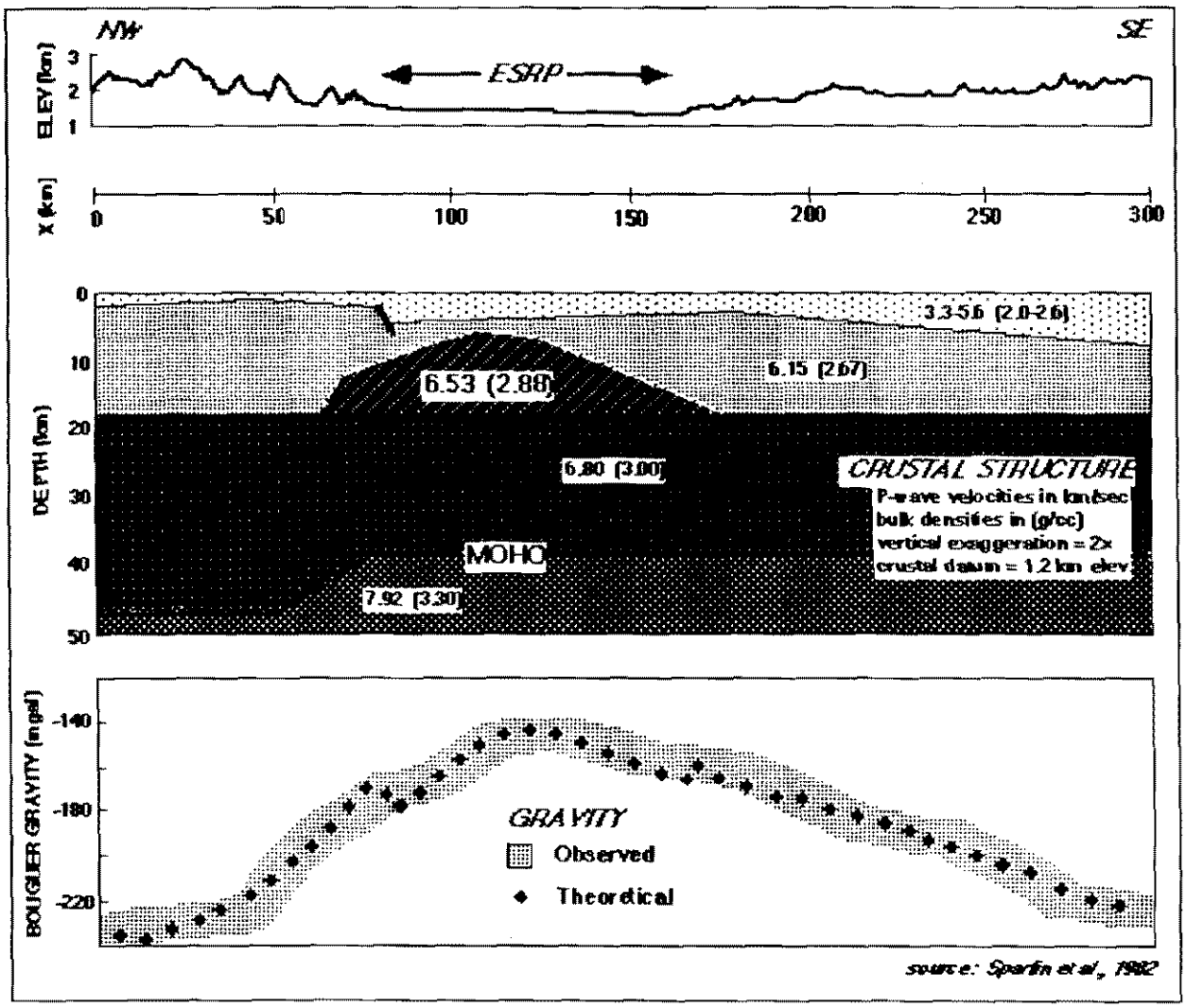


Figure 2A.1-5. Summary plot of age-distance relationships among volcanic and sedimentary rock units of the Snake River Plain and Yellowstone (adapted from Armstrong, Leeman, and Malde 1975; Hackett and Morgan 1988). **(For Information Only)**



EASTERN SNAKE RIVER PLAIN REGIONAL TOPOGRAPHY, GRAVITY AND CRUSTAL STRUCTURE

Figure 2A.1-6. Schematic cross sections of the Eastern Snake River Plain near the INEEL showing regional topography, bouguer gravity, and inferred crustal structure based on gravity and seismic-refraction data (adapted from Sparlin, Braile, and Smith 1982). **(For Information Only)**

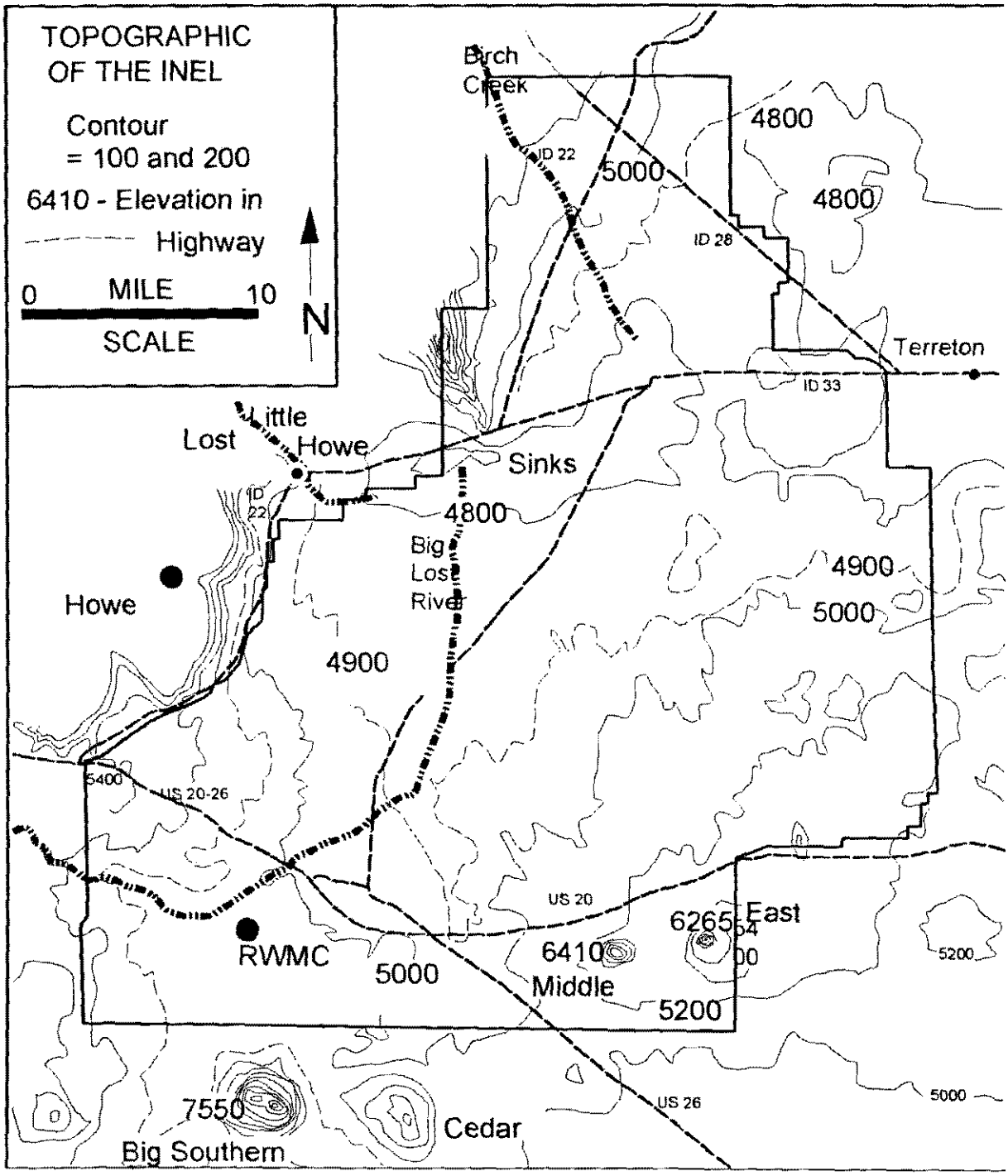
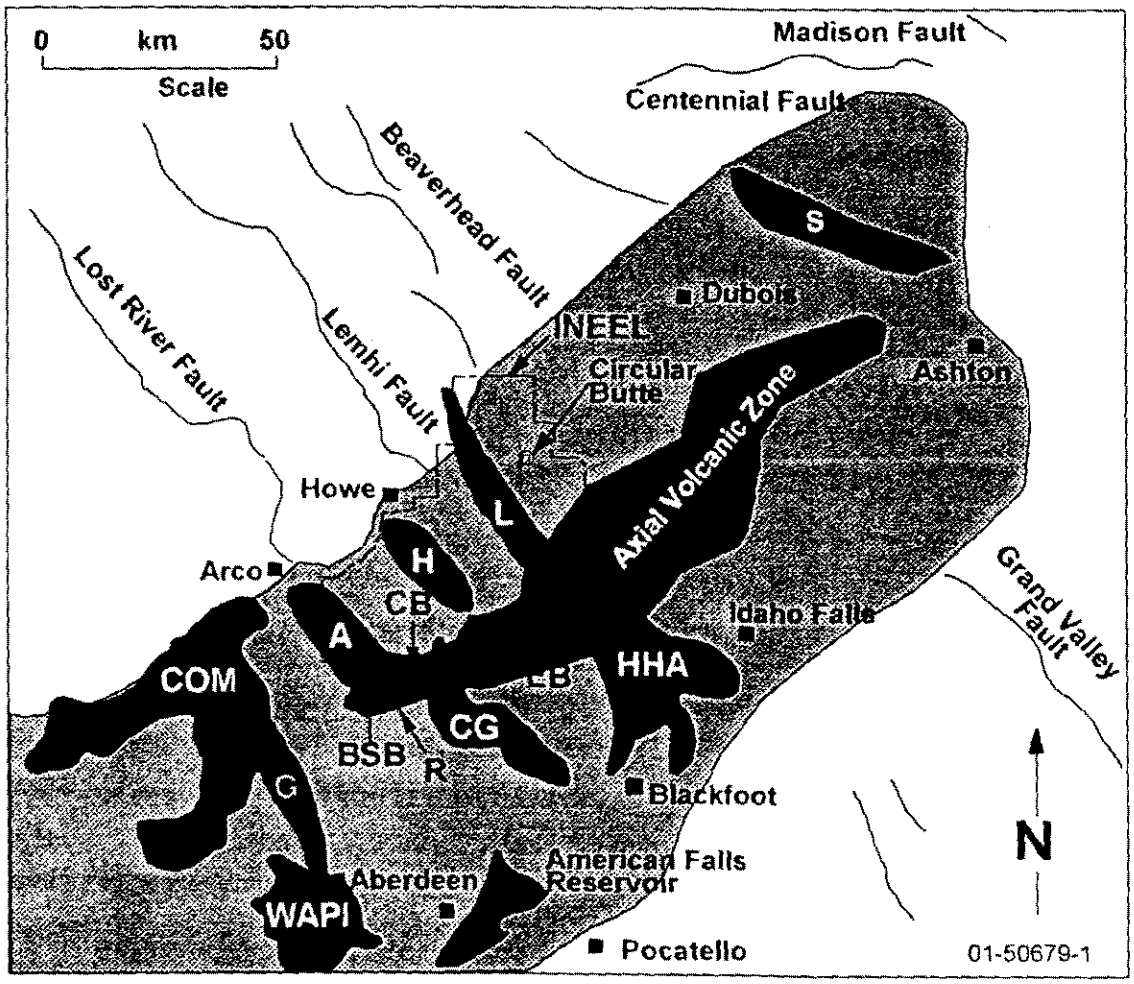


Figure 2A.1-7. Topographic map and major geographic features of the INEEL area.
(For Information Only)



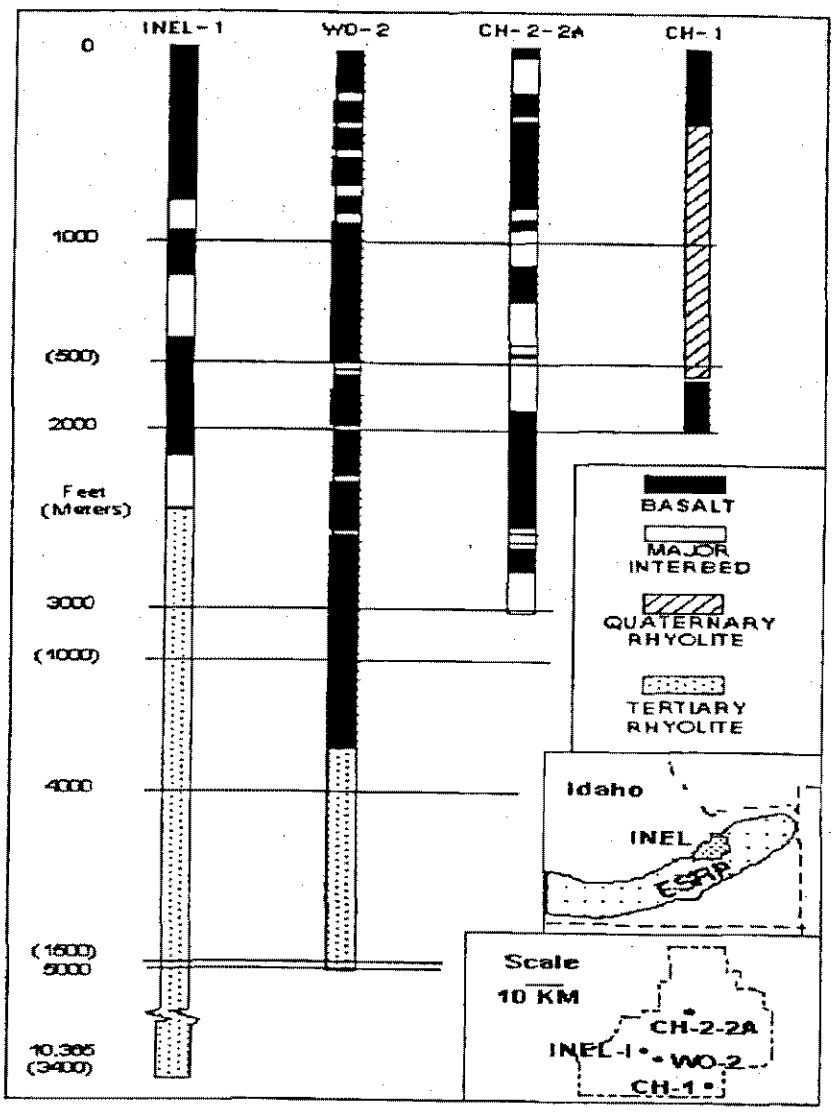
Holocene Lava Fields
COM=Craters of the Moon
CG=Cerro Grande
HHA=Hells Half Acre
R=North & South Robbers

Volcanic Rift Zones
S=Spencer-High Point
L=Lava Ridge-Hells Half Acre
H=Howe-East Butte
A=Arco
G=Great Rift

ESRP
BSB=Big Southern Butte
MB=Middle Butte
EB=East Butte

Basin and Range Province

Figure 2A.1-8. Map the Eastern Snake River Plain showing the locations of Holocene lava field, volcanic rift zones, and axial volcanic zone (adapted from Kuntz, Covington, and Schorr 1992). **(For Information Only)**



Simplified lithologic logs of deep drillholes on the INEL. (INEL-1 from Doherty et al., 1979; CH-1 from Doherty, 1979a; CH-2-2A from Doherty, 1979b; and WO-2 from Hackett and Smith, 1992)

Figure 2A.1-9. Generalized lithologies of four deep drillholes on the INEEL (insets show rock types, INEEL location, and drillhole locations on the INEEL). **(For Information Only)**

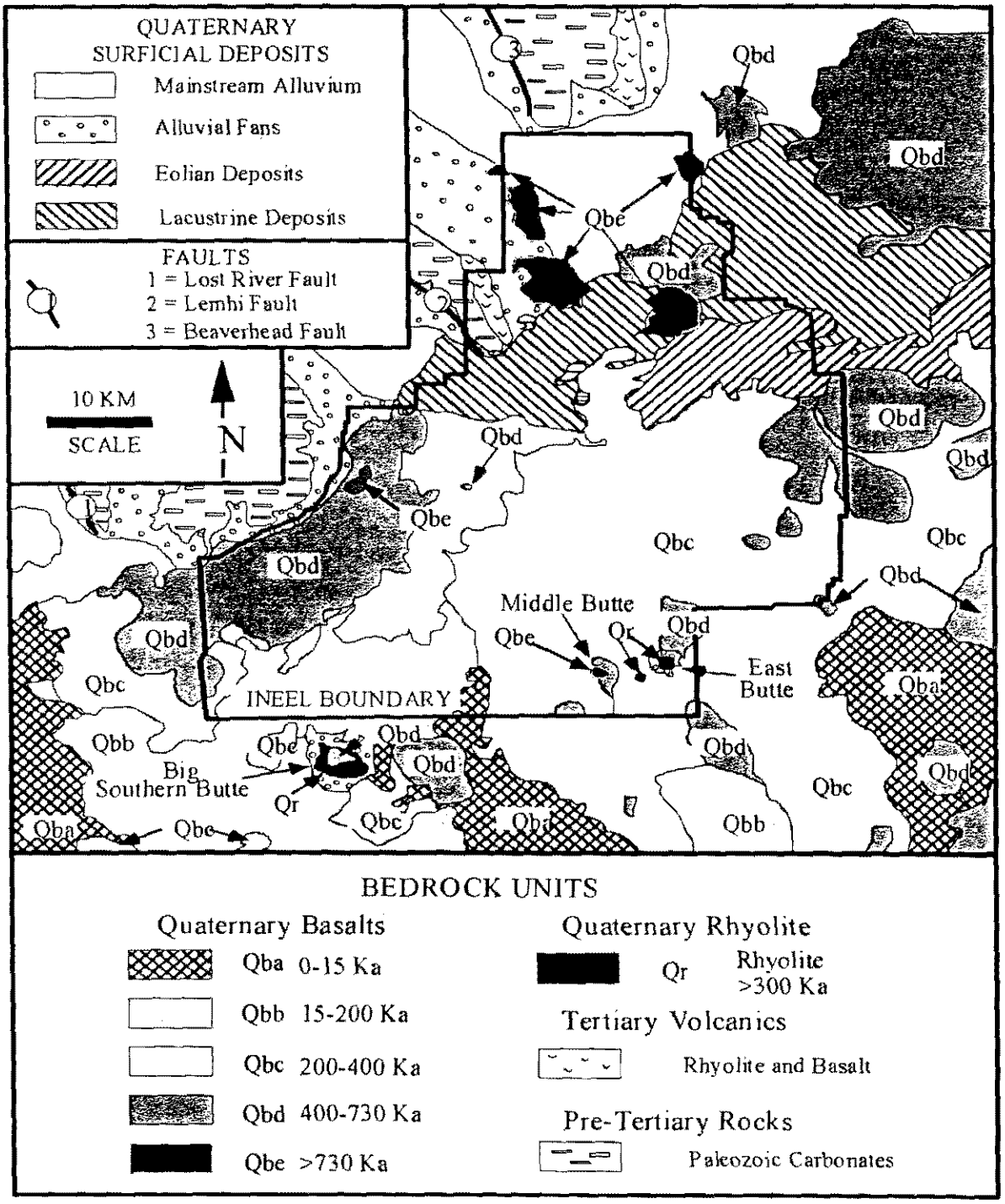


Figure 2A.1-10. Geologic map of the INEEL area (adapted from Kuntz et al. 1990; Hackett and Smith 1992). **(For Information Only)**

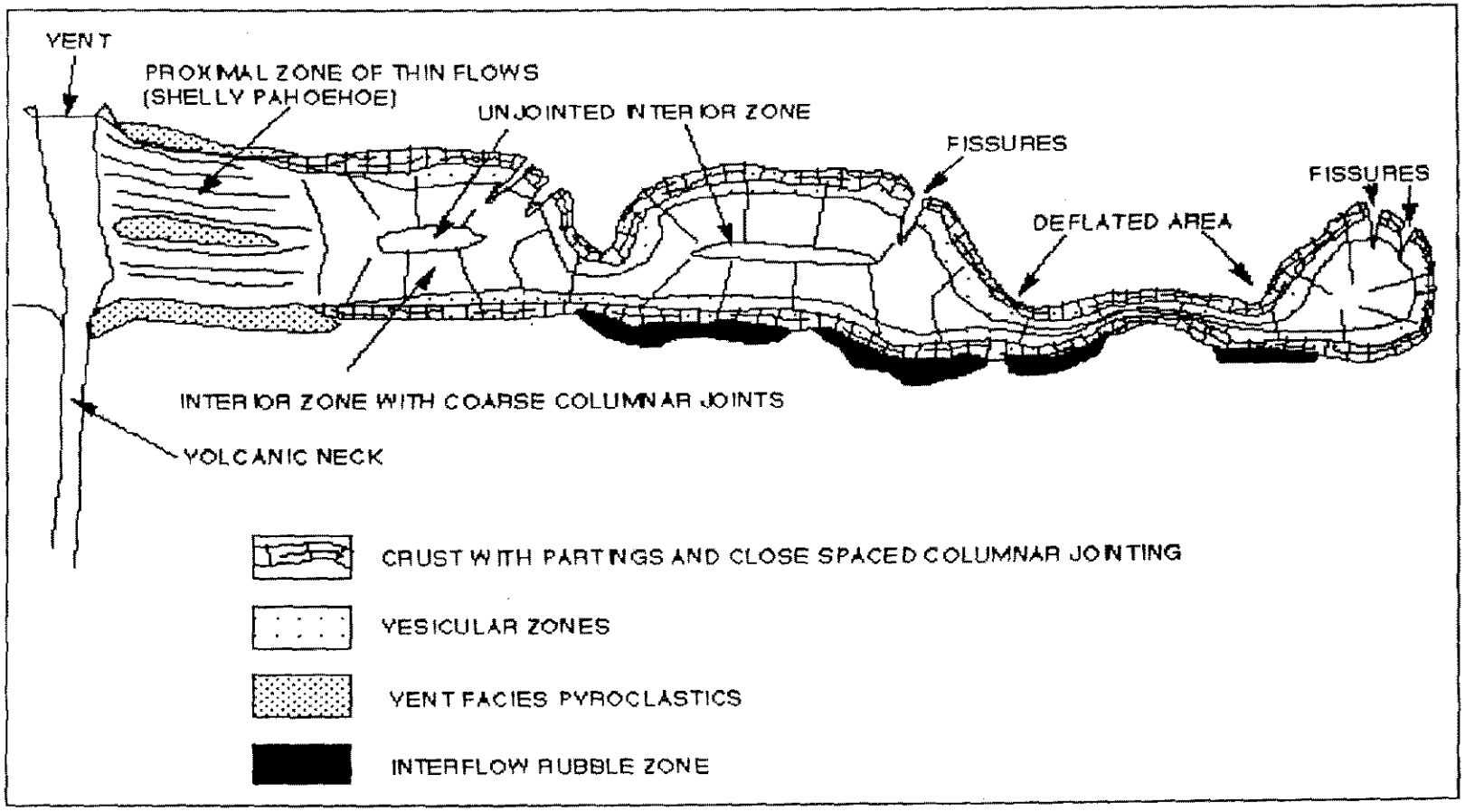


Figure 2A.1-11. Generalized cross section of a basalt lava flow on the Eastern Snake River Plain showing typical lithologic features, structures, and facies variations with distance from vent. **(For Information Only)**

2A.2 Vibratory Ground Motion

The following subsections provide descriptions and discussions of the seismicity of the region, the geologic structures and tectonic activity, earthquake activity in relation to geologic structures or tectonic provinces, maximum earthquake potential regionally and locally, characteristics of seismic-wave transmissions, safe-shutdown earthquake, and operating-basis earthquake.

2A.2.1 Seismicity

This subsection provides information on the seismic nature of the region, historical records of earthquakes, and the seismic monitoring network at the INEEL and regionally.

2A.2.1.1 Regional Setting. The ESRP is defined as the eastern portion of the SRP extending from the Yellowstone Plateau to the Great Rift (Figure 2A.1-8). The relatively aseismic ESRP is surrounded by the seismically active Intermountain Seismic and Centennial Tectonic belts. The Intermountain Seismic belt (ISB) is a zone of concentrated seismicity that extends from northwestern Montana through eastern Idaho and central Utah into southern Nevada (Sbar and Barazangi 1970; Smith and Sbar 1970, 1974; Arabasz and Smith 1981; Smith and Arabasz 1991). It is divided into three parts, which are referred to as northern (Montana), central (Idaho), and southern (Nevada and Utah) ISB (Smith and Arabasz 1991). North of the ESRP, seismicity extends from Hebgen Lake, Montana, westward into central Idaho, and has been characterized as an independent zone of earthquake activity that is referred to as the Centennial Tectonic Belt (CTB) (Stickney and Bartholomew 1987). Smith and Arabasz (1991) consider the CTB (formerly called the Idaho Seismic Zone) as a part of the central ISB that wraps around the ESRP. In the following discussions, this zone of seismicity will be referred to as the CTB to distinguish it from the north-trending zone of seismicity within the central ISB.

2A.2.1.2 Earthquake Data. Earthquakes of Richter magnitudes > 2.0 for the time period 1884-1989 were compiled by Woodward-Clyde Consultants (1992a), from the following sources:

- Idaho National Engineering Laboratory (INEL), 1986-1989
- United States Geological Survey (USGS), 1986-1989
- Montana Bureau of Mines and Geology (MBMG), 1986-1989
- United States Bureau of Reclamation (USBR), 1986-1989
- University of Utah Seismograph Stations (UUSS), 1986-1989
- Engdahl and Rinehart (1988, 1991), 1884-1985.

Jackson (1993) lists the earthquake location parameters (date, location, depth, magnitude, data source, etc.) for earthquakes with magnitude > 3.0 for a 200-mi (320-km) radius around the TRA site.

For the central ISB, the earthquake record extends back to November 10, 1884, the date of the first documented earthquake [Richter magnitude (M_L) 6.3], which occurred near Paris, Idaho. Before the 1960s, seismographic coverage of the ESRP and surrounding Basin and Range was relatively poor, with only earthquakes larger than magnitude 5.0 recorded by seismographs worldwide. Detection of earthquakes before this time was based on motion and damage reports made by local residents.

The regional seismographic network, beginning in the mid-1960s, in Utah and eastern Idaho, including the INEL in the mid-1970s, is operated by UUSS. The Yellowstone Park network was operated from 1970 to 1981 by the USGS and from late-1983 to present by UUSS (WCC 1992a). Seismic stations were installed near Teton Dam, Idaho, (currently operated by Ricks College) beginning in 1980; in southwestern Montana (MBMG) starting in 1981; and in western Wyoming near Jackson Lake (USBR) during 1986. With additional seismic stations, smaller-magnitude earthquakes could be detected. Based on the number of seismic stations operating over specific time intervals, periods of completeness can be established for various magnitudes. The periods of completeness are the time periods over which independent earthquakes (excluding aftershocks) can be considered to be completely detected (WCC 1992a). Table 2A.2-1 shows the periods of completeness for various magnitudes given in Jackson (1993) (based on Youngs et al. 1987; Geomatrix Consultants 1989; Engdahl and Rinehart 1988; WCC 1992a). The completeness periods indicate that for historic times, the data base for larger-magnitude earthquakes is more complete than for smaller-magnitude events.

2A.2.1.3 Moderate-to-Large Earthquakes. Moderate-to-large earthquakes of magnitude ≥ 5.5 have occurred within 200-mi (320-km) of ATR. For these events, Table 2A.2-2 lists the largest magnitude computed, moment magnitude if computed, and Modified Mercalli intensities that are based on damage at the epicenter. Because earthquakes of magnitudes $M \geq 2.5$ occur at distances greater than 31 mi (50 km) from ATR, only events of $M \geq 5.5$ are listed in Table 2A.2-2. Figure 2A.2-1 shows their locations relative to Basin and Range normal faults and the ESRP.

The largest earthquake, with surface-wave magnitude (M_s) 7.5, occurred within the ISB on August 17, 1959, at Hebgen Lake, Montana (Figure 2A.2-1) (Doser 1985a). It was located 120 mi (190 km) northeast of ATR. Although the earthquake was felt at the INEEL, it caused no damage at ATR or other INEEL facilities (Gorman and Guenzler 1983).

The M_s 7.3, Borah Peak, Idaho, earthquake occurred on October 28, 1983, in the CTB near the INEL (Doser and Smith 1985). The earthquake resulted from normal faulting along the Lost River fault (Crone et al. 1987). The epicenter for this event was located in the Thousand Springs valley near the western flank of Borah Peak (Richins et al. 1987). Substantial damage occurred to masonry structures in the local communities of Mackay and Challis, Idaho, near the epicentral area (Stover 1985).

ATR was located in Modified Mercalli Intensity Zone VI during the earthquake (Stover 1985). Inspections following the earthquake revealed no apparent structural or component damage that would compromise structural integrity at ATR. The reactor automatically scrammed without incident when the ATR Plant Protective System's trip was triggered by earthquake ground motions that exceeded 0.01 g, which was the level at which the trip was set (Gorman and Guenzler 1983).

Currently, the INEEL operates 22 strong-motion accelerographs (SMAs). They are located at various levels (e.g., basement, first floor, and roof tops) within critical facilities and at free-field sites (not within buildings). There are two instruments located at TRA: one in ATR and one in the Engineering Test Reactor (ETR). Instruments within facilities record the response of the building to the earthquake ground shaking; at free-field sites, instruments record the level of earthquake ground motions at the earth's surface. At the time of the Borah Peak earthquake, the INEL had 15 SMAs in operation including the two at TRA. Peak horizontal accelerations recorded at INEL ranged from 0.022 to 0.078 g for basement and free-field sites (Jackson 1985). Table 2A.2-3 shows the corrected peak accelerations, velocities, and displacements measured by the SMAs at ATR and ETR, which were 53 mi (86 km) from the Borah Peak epicenter (Jackson et al. 1991). Jackson et al. contains copies of the corrected acceleration, velocity, and displacement time-histories and response and Fourier spectra for the vertical and two horizontal components for the SMAs at ATR and ETR. Peak horizontal accelerations (0.022-0.030 g) recorded at TRA were well below the current safe shutdown earthquake (SSE) design accelerations of 0.24 g for ATR.

2A.2.2 Geologic Structures and Tectonic Activity

2A.2.2.1 Tectonic Provinces. The tectonic provinces of greatest concern for seismic and volcanic hazards at INEEL are the ESRP and the northern Basin and Range Province (Figure 2A.1-1). Other provinces sufficiently close to INEEL to warrant possible consideration, especially for probabilistic seismic-hazards assessments, are the Yellowstone Plateau and the Idaho Batholith.

The ESRP is distinguished from the surrounding provinces by subdued topography, lower elevations, absence of Basin and Range faults and mountain ranges (Figure 2A.1-1), and historic aseismicity (Anders et al. 1989). In addition, it is associated with a regional gravity high (Bankey et al. 1985), positive aeromagnetism that result from basalt lava flows at the surface (Zeitz, Gilbert, and Kirby 1978), and high seismic velocity (Sparlin, Braile, and Smith 1982), which is associated with zones of dense mafic rocks in the mid-crust beneath the Plain. This zone of mafic material in the mid-crust is believed to represent the zone of accumulation and solidification of mafic magmas that were generated by the Yellowstone hotspot as it passed beneath the ESRP.

The northern Basin and Range Province is distinguished by north to northwest trending block-fault mountain ranges that formed in response to east to northeast extension. Normal faults bounding these ranges have accumulated 0.6 to 1.2 mi (1 to 2 km) of vertical displacement during Late Tertiary and Quaternary time (Pierce and Morgan 1992). Seismicity and Holocene paleoseismicity along these faults are concentrated in a parabolic zone that passes through the Yellowstone Plateau and flanks both sides of the ESRP (Figure 2A.2-2) (Anders et al. 1989).

The Yellowstone Plateau is distinguished by Quaternary and continuing crustal response to rapid uplift over the Yellowstone hotspot close to areas (northeastern ESRP) that are rapidly subsiding. This combination results in development of large faults with high slip rates (Pierce and Morgan 1992) and with trends inconsistent with the direction of regional extension (e.g., the Centennial, Teton, and Hebgen/Red Canyon faults) (Figure 2A.2-9) (Doser 1985a; Eddington, Smith, and Renggli 1987; Peyton, Smith, and Pechmann 1991). In addition, the Yellowstone Plateau has much greater levels of seismicity than either the ESRP or the northeastern Basin and Range Province (Smith and Arabasz 1991), a situation resulting from interaction of regional extension with rapid local vertical crustal movements. Occurrence of voluminous Quaternary explosive silicic volcanism (Christiansen 1984) and significant delays in teleseismic P-waves beneath the caldera area suggest extremely high temperatures and presence of magma in the crust and upper mantle (Iyer et al. 1981; Evans 1982; Smith and Braile 1994).

The Idaho Batholith is distinguished by high, rugged topography, sparsity of Basin and Range faults, and absence of late Tertiary and Quaternary volcanism (Figure 2A.1-1). Seismicity is much less intense than that associated with the Basin and Range (Smith and Arabasz 1991), with maximum magnitude of about 5. Nevertheless, the seismicity in the Idaho Batholith is considered in seismic hazards assessment for INEEL. The batholith appears to have been relatively unaffected by regional extension, perhaps because the granitic rocks are stronger or more coherent than rocks in the Basin and Range Province to the east and southwest.

2A.2.2.2 Faults. Faults of several ages and origins occur in the INEEL region. Some of them are old and inactive, presenting no earthquake threat; others are capable of generating earthquakes that could affect INEEL facilities. A detailed correlation of faults with earthquakes is presented in Section 2A.2.4.

Mesozoic thrust faults occur in the mountain ranges bordering the ESRP (Figure 2A.1-2) (Skipp and Hait 1977; Link et al. 1988). They formed during a period of east-directed thrusting related to the Sevier orogeny. They are gently westward-dipping structures that separate major Paleozoic thrust sheets. These faults are mostly inactive at the present time because the compressional forces that created them no longer exist. However, it is possible that parts of some of the thrust faults have been reactivated by Basin and Range normal faults in Late Tertiary to recent times (Smith and Bruhn 1984).

Eocene to Oligocene normal faults trend northward across the Lost River, Lemhi, and Beaverhead ranges north of the ESRP (Janecke 1992). Although these faults have several kilometers of accumulated displacement, their orientation is not optimum for reactivation in the present stress field. Therefore, they are not active today and pose no threat for earthquake hazards.

Basin and Range normal faults (Figure 2A.2-2) of Miocene to Recent age bound the present northwest-trending mountain ranges north and south of the ESRP (Scott, Pierce, and Hait 1985). These faults have accumulated 0.6 to 1.8 mi (1 to 3 km) of displacement in the past 4 to 7 Ma and are still active today as evidenced by fault scarps cutting latest Quaternary and Holocene alluvial fan deposits and by the occurrence of the 1983 Borah Peak earthquake. Table 2A.2-4 summarizes the important characteristics of most Basin and Range normal faults around the ESRP. The closest of these faults to INEEL facilities (the Lost River, Lemhi, and Beaverhead faults [Figure 2A.2-2]) each bound the southwest side of a mountain range, producing typical Basin and Range half graben. Large normal faults extend from the northern margin of the ESRP northward to the Salmon River. Based on seismic and paleoseismic investigations, they are capable of generating earthquakes of magnitude 7 (Doser and Smith 1985; WCC 1992b). Because of their size, activity, and proximity to many INEEL facilities, they control much of the INEEL seismic hazard.

2A.2.2.2.1 Lemhi Fault—Detailed paleoseismic and structural investigations have been performed on the southern Lemhi fault (WCC 1992b; Bruhn, Wu, and Lee 1992), with the following results:

- Segmentation of the southern Lemhi fault is redefined based on timing of paleoseismic events and on detailed mapping of the structure of the fault in bedrock and surficial deposits (Figure 2A.2-3).
- The most recent earthquake events on the various segments range from 15,000 to 25,000 years ago (Figure 2A.2-4).
- There is evidence for temporal clustering of earthquake events (i.e., clusters of several events over a few thousand years separated by long intervals [tens of thousands of years] of quiescence).
- Maximum magnitude of earthquakes in the southern part of the fault is about M 7 (WCC 1992a).
- Bedrock structural features of the southern part of the fault suggest that Quaternary displacement dies out at the south end of the Lemhi Range and that significant seismogenic fault movements do not extend onto the ESRP (Figure 2A.2-6).

Summary of earthquake events deduced from 4 trenches logged in 1990 and 1991. The data suggest that proposed segment boundaries (solid arrowheads) do not terminate earthquake ruptures. Instead, single events (E2, etc) may extend from Coyote Springs south to the end of the range (open arrowheads). Resulting rupture lengths of 34-35 km and corresponding Mw=6.9 establish the MCE for the southern Lemhi fault.

2A.2.2.2.2 Lost River Fault—The Lost River Fault is slightly farther away from most INEEL facilities than the Lemhi Fault, but is similar with respect to seismic hazards. Detailed paleoseismic and structural investigations were completed on the Arco and Pass Creek segments of the Lost River Fault (closest to the INEEL boundary) (Woodward-Clyde Federal Services (WCFS) 1996) (Figure 2A.2-5). Results indicate:

- Activity on both segments was younger than previously believed. The ages of the two most recent events on the Arco segment are between 21±4 ka and 20±4 ka (±2σ). The ages of the three most recent events on the Pass Creek segment are between 18±3 ka and 17±4 ka. Because of the overlap in age estimates (within 2σ), the two most recent events on both segments may have been contemporaneous.
- Ages of individual earthquake events indicate temporal clustering (i.e., clusters of several events over a few thousand years separated by long intervals [tens of thousands of years] of quiescence). Recurrence intervals varied from around 1,000 years or less to 40,000 years or more on both segments.

- Paleomagnitude estimates based on vertical displacements yield a range of moment magnitudes (MW) 6.6 to 7.3 for the Arco segment and 6.7 to 7.5 for the Pass Creek segment, depending on whether displacements are assumed to be average or maximum values. Length-based magnitude for the Pass Creek segment is MW 6.7, and for the Arco segment ranges from MW 6.6 - 6.8.

- The Arco segment of the Lost River Fault may extend south of the Lost River Range onto the ESRP and into the northwestern portion of the Arco Volcanic Rift Zone.

2A.2.2.2.3 Beaverhead Fault—Although considerably farther from INEEL than the Lemhi and Lost River faults, earthquakes from this fault will contribute significantly to the probabilistic hazard assessment.

2A.2.2.2.4 Grand Valley—Star Valley Fault—Although this fault is located at great distance from the INEEL area, it will also contribute to the probabilistic hazard. Field investigations by Anders et al. (1989), Piety et al. (1986), and McCalpin, Piety, and Anders (1990) have shown that the northern part of this fault system was very active from about 4 to 2 Ma, but since then it has been inactive. The southern end of the fault, in the Alpine and Star Valley area, however, has experienced late Pleistocene and Holocene earthquake activity.

The northwest boundary of the ESRP has been considered as a possible seismic source (EG&G Idaho 1984). While it is true that a northeast-trending fault or faults may exist at depth (Pankratz and Ackerman 1982), the only evidence for active faulting consists of the presence of a small northeast-trending topographic scarp on an alluvial fan on the southeast side of the Arco Hills (Scott 1982). That scarp was trenched in 1989 by the Idaho Geological Survey under contract to EG&G Idaho. The resulting trench logging showed that faulting does not occur there and that the scarp results from some surficial processes such as eolian modification of a fire scar (Breckenridge and Othberg 1991). Other information about faulting parallel to the ESRP margins is given in Section 2A.2.4.4.

Nontectonic lineaments on and near the INEEL can be observed from the air, on aerial photographs, and on satellite images. One of the most pronounced of these lineaments, the Principal Lineament, has been studied extensively and was shown to be caused by eolian modifications to a large fire scar (Morin-Jansen 1987). This process produces many lineaments and perhaps even small topographic scarps on the ESRP. Other lineaments are caused by unmodified fire scars, linear stream drainages, alignments of vegetative or soil contrast with unknown causes, fluvial (stream, river) deposits, paleoflood deposits, and eolian deposits (dunes) (Golder 1992; Breckenridge and Othberg 1991). A discussion of lineaments near the ATR site is presented in Section 2A.3.

Late Tertiary caldera boundary faults are postulated to exist in the silicic volcanic rocks beneath the Snake River Group. There are several bases for this postulation:

- Calderas like those that exist on the Yellowstone Plateau today must have been associated with the late-Tertiary silicic volcanic fields occurring along the margins of the ESRP.
- In some areas (southern ends of the Lemhi and Beaverhead Ranges near INEEL and northern ends of the Caribou and Snake River Ranges near Rexburg) structures interpreted to be caldera boundary structures have been recognized (Morgan 1988).
- The great thicknesses of silicic volcanic rocks observed in INEEL deep exploration holes, INEL-1 and WO-2 (Figure 2A.1-9), indicate that they were emplaced into an intra-caldera setting.

The exact sizes, shapes, and locations of the buried calderas are uncertain, but interpretations have been made (Figure 2A.2-2) on the basis of geophysical anomalies, positions of volcanic fields, flow-direction indicators in ash flow sheets, and paleomagnetic data (Morgan 1988). Several general observations are possible. A caldera could be large enough to span the entire width of the ESRP. Caldera shape, and thus the configuration of associated caldera boundary faults, is generally circular to oval. Because calderas tend to overlap (Figure 2A.2-2), it is likely that most of the ESRP boundary is characterized by caldera boundary faults buried beneath the edges of the Snake River Group. Caldera boundary faults can explain, in a manner consistent with data and concepts, Pankratz and Ackerman's (1982) interpreted buried fault along the northwest margin of the ESRP.

Several lines of evidence, summarized in Section 2A.6, show that the calderas are no longer active because the causative heat source has moved to a new position beneath Yellowstone. The possibility of reactivation of the faults resulting from contemporary tectonism should be considered but does not seem to be a cause for concern for two reasons: (a) because the faults have a circular to oval configuration, they are not likely to have long sections oriented properly for movement in contemporary stress fields, and (b) late-Pleistocene and Holocene faulting observed on the ESRP could not be related to reactivation of these faults (Golder 1992).

2A.2.2.3 Volcanic Rift Zones and Axial Volcanic Zone. Volcanic vents on the ESRP are concentrated in northwest-trending and northeast-trending linear belts (Figure 2A.1-8). The northwest-trending belts have associated ground deformation features and are referred to as volcanic rift zones (VRZs). The ground deformation features are fissures, faults, grabens, and monoclines that form from dilational stresses above the tops of basalt dikes as magma moves from depth to the surface. Three well-defined VRZs occur in the INEEL region of the ESRP: the Great Rift VRZ (which extends southeastward from Craters of the Moon National Monument); the Arco VRZ (which extends southeast from Arco across the southwestern corner of the INEL); and the Lava Ridge-Hell's Half Acre (LR-HHA) VRZ (which extends from the south end of the Lemhi Range to the Hell's Half-Acre lava field) (Figure 2A.1-8). An older, and poorly defined, VRZ (the Howe - East Butte VRZ) may extend from the vicinity of Howe to the SE corner of INEL. Since it is defined by only two fissures and four vents (as compared to scores of fissures and vents for the other VRZs) its mere existence is in question. This VRZ is not sufficiently defined to be included in Figure 2A.1-8.

By analogy with active VRZs in other parts of the world (Iceland and Hawaii), it can be inferred that VRZs are sources of earthquakes during periods of volcanic activity (see Section 2A.6.2.3.1). Even though the magnitudes of VRZ earthquakes are smaller (M < 5.5) than those of the Basin and Range faults (M > 7.0) (see Section 2A.2.4), their closer proximity to INEEL facilities indicates that they can be significant contributors to both deterministic and probabilistic seismic hazards (WCC 1992a).

The VRZs and the northeast-trending Axial Volcanic Zone (AVZ) have remained as highlands throughout much of the development of the ESRP and, thus, do not receive as much sediment as the basins. This results in large-scale heterogeneities in seismic energy transmission properties and in aquifer properties.

Some volcanic vents on the ESRP are concentrated along the AVZ (Figure 2A.1-8). It is important to distinguish the AVZ from VRZs because the AVZ does not contain northeast-trending ground-deformation features that would qualify it to be a VRZ. The few ground-deformation features that do occur in the AVZ are northwest-trending fissures. This fact indicates that the volcanic vents in the AVZ are fed by northwest-trending dikes, and that even though it is not a VRZ, seismicity can be associated with volcanism there. Thus, it also plays an important role in deterministic and probabilistic seismic hazards assessment (WCC 1992a).

2A.2.3 Correlation of Earthquake Activity with Geologic Structures or Tectonic Provinces

Figure 2A.2-7 shows a compilation of the minimum-principal-stress directions for the ESRP region derived from focal mechanisms, geologic indicators, and borehole breakouts (Jackson et al. 1993; Pierce and Morgan 1992; Peyton, Smith, and Pechmann 1991; Zoback and Zoback 1989). The minimum-principal-stress directions indicate northeast-trending extension northwest of the ESRP and a more east-trending direction south of the ESRP. Although a rotation in the stress field may occur somewhere within the ESRP, the ESRP appears to be subjected to the same extensional stress field as the surrounding region (Jackson et al. 1993). Strain rates have been compiled by Eddington, Smith, and Renggli (1987) for the ESRP region (Figure 2A.2-8). Strain rates for the region around the ESRP range between 1.1×10^{-15}/s for Yellowstone Plateau to 3.8×10^{-17}/s for the ISB. The preliminary estimate for the ESRP is 1×10^{-16}/s based on the amount of extension measured within the ESRP rift zones for the Holocene, and is similar to strain rates outside the ESRP (Hackett, Smith, and Jostin 1991).

Table 2A.2-5 lists earthquakes with M ≥ 5.5 that have occurred within 200 mi (320 km) of the ATR, and that can be correlated with tectonic structures. Table 2A.2-5 includes the seismic moments, focal mechanisms, focal depths, rupture lengths, and horizontal and vertical displacements computed by various seismological methods for these earthquakes (see references in Table 2A.2-5 for more details). The following discussion of earthquakes and their relationships to geologic structures or provinces is separated into areas based on tectonic provinces.

2A.2.3.1 Eastern Snake River Plain Province

2A.2.3.1.1 Seismicity—Stover, Reagor, and Algermissen (1986) noted 14 historic ears similar to the input used in the NPR probabilistic evaluation, with minor exceptions (WCC 1992a). Geologic profiles for ATR were refined based on new borehole information and additional estimates of V_s. The BLWN ground-motion model was used along with empirical relationships. Sensitivity analyses were performed to determine seismic sources that contribute most to the hazard.

2A.2.3.1.2 Causes of Eastern Snake River Plain—Microearthquakes— Investigators have also suggested possible mechanisms for microearthquakes that occur within the ESRP. Because the ESRP is a volcanic province, magmatic processes have been considered as a possible mechanism for the low-level microearthquakes. Brott, Blackwell, and Ziagas (1981) suggested that microearthquakes may be a result of subsidence from cooling and contraction of the ESRP following the passage of the hotspot. Pelton, Vincent, and Anderson (1990) suggested association with dike-injection or mass loading of the crust by the rhyolite domes located near the axis of the ESRP. Jackson et al. (1993) observed that the microearthquakes occurring in the ESRP from 1972-1990 do not have the distinct spatial or temporal patterns observed during contemporary dike-injection events at Kilauea, Hawaii, or Krafla, Iceland (Brandsdottir and Einarsson 1979; Einarsson and Brandsdottir 1980; Klein et al. 1987) and, therefore, they are not attributed to volcanism. Although no detailed analyses of mass loading and its role in producing microearthquakes within the ESRP have been performed, Jackson et al. (1993) attribute the occurrence of microearthquakes ($M \leq 1.5$) to small-scale faulting in the shallow crust, in response to the regional extensional tectonic stress field.

2A.2.3.1.3 Volcanic Seismicity—Small dike-induced normal faults are located within the VRZs (see Section 2A.6.2.3.1). Seismic studies at active VRZs, such as in Hawaii and Iceland, and theoretical and physical models of the resulting surficial deformation features indicate that dike-injection can produce small normal faults that extend up to or slightly below the top of the dike 1.2 to 2.4 mi (2 to 4 km) (Mastin and Pollard 1988; Ryan 1987; Rubin and Pollard 1987, 1988; Rubin 1990, 1992). Because a dike-injection event has not been observed within an ESRP rift zone, active analogs are used to estimate the maximum magnitude of earthquakes that would accompany future ESRP volcanism (Table 2A.2-6). Small-magnitude earthquakes, commonly less than M_L 4.5, usually accompany basalt dike injection, although magnitude 5.5 earthquakes have been observed (Hackett, Jackson, and Smith 1996; WCC 1992a). Rubin (1992) suggests that some small normal faults form aseismically through multiple offsets as dike-injection occurs. Bjornsson et al. (1977) observed offsets of 3 to 6 ft (1 to 2 m) along normal faults during intrusion into a Krafla rift zone where the largest magnitude of the associated seismicity was M_L 3.8.

In addition to observed seismicity during active dike-injection events, the empirically based relationship of fault-area versus moment magnitude (M_w) developed by Wells and Coppersmith (1994) is used to estimate the upper bound for the maximum magnitude of seismicity associated with dike-injection in ESRP VRZs. Table 2A.2-6 shows the range of magnitudes, 3.3 M_w 5.3, derived from the relationship between fault area and moment magnitude for normal fault lengths within the Arco and LR-HHA VRZs. These values are somewhat similar to the observational values shown in Table 2A.2-7. The relationship between fault-area and maximum magnitude was used to estimate the maximum magnitude results in an upper bound for several reasons (Hackett, Jackson, and Smith 1996): (a) deformation can occur aseismically, and seismic moment release may be small compared to total moment released through inelastic deformation (Filson, Simkin, and Leu 1973; Stein et al. 1991; Rubin 1992); (b) faults move in small increments because of slow dike propagation resulting in short, discontinuous segments with variations in amounts of displacement; (c) dike-induced normal faults have shallow downdip widths resulting in small areas for rupture (Daz and Scholz 1983); (d) using magnitude—fault-area relationships assumes rupture along the entire length, but observations indicate that the faults move in small increments or even aseismically; and (e) the relationship between moment magnitude and fault area assumes a crustal value for rigidity of 4.35×10^6 psi (3×10^{11} dyne/cm^2) (which may be lower for near-surface volcanic rocks) to appropriately describe volume changes [0.725 to 2.61×10^6 psi (0.5 to 1.8×10^{11} dyne/cm^2)] (Filson, Simkin, and Leu 1973; Mori et al. 1989; Stein et al. 1991).

Recurrence intervals of the dike-induced seismicity within the ESRP VRZs are based on the volcanic rock record. For the current ATR probabilistic assessment, the maximum magnitude earthquake is assumed to occur during each dike-injection episode (Table 2A.6-3; Section 2A.6). Observational seismicity during dike-injection events at Krafla and Hawaii suggests that some episodes of dike injection are accompanied by earthquakes with maximum magnitudes of 5.5.

2A.2.3.2 Northern Basin and Range

2A.2.3.2.1 Centennial Tectonic Belt—The 1983 M_s 7.3 Borah Peak, Idaho earthquake is the largest event to occur in the CTB (Figure 2A.2-1). It originated at a depth of 10 ± 2.5 mi (16 ± 4 km), near the base of the seismogenic crust (Doser and Smith 1985). It ruptured to the northwest, producing 22 mi (36 km) of surface faulting along the Thousand Springs segment and a portion of the Warm Springs segment of the Lost River fault. It also produced a surface scarp with a maximum of 8.9 ft (2.7 m) vertical displacement (Crone et al. 1987). The Borah Peak mainshock and aftershocks define a normal fault dipping 40-50 degrees to the southwest, which is consistent with dips determined from first motions, body-wave analysis, and geodetic observations (Table 2A.2-5) (Richins et al. 1987).

The stress drop determined from seismic moment is 17 bars, and from geologic data, 12 bars. Even considering the possible sources of error in the calculations, the stress drop probably did not exceed 75 bars, suggesting that the Borah Peak earthquake was a low-stress-drop event compared to other normal faulting earthquakes in the same magnitude range (Doser and Smith 1985).

2A.2.3.2.2 Intermountain Seismic Belt—Several moderate-to-large magnitude earthquakes that can be correlated to tectonic structures have occurred within the central part of the ISB near the ESRP (Figure 2A.2-1):

- Hansel Valley. The March 12, 1934, M_L 6.6, Hansel Valley, Utah earthquake was felt over an area of 170,000 mi^2 (440,000 km^2) and reached MMI of VIII (Smith and Arabasz 1991). Shenon (1936) mapped north-trending subparallel fractures displacing salt flats and unconsolidated late Quaternary sediments in the southwestern part of Hansel Valley over an area 3.7 mi (6 km) wide and 7.5 mi (12 km) long. Up to 20 in. (50 cm) of vertical displacement and 10 in. (25 cm) horizontal offset were reported by dePolo et al. (1989). The focal mechanism from seismic waveform modeling by Doser (1989a) indicates that the mainshock occurred along a strike-slip fault with left-lateral slip on a northeast-trending structure. The event originated at focal depth of 5 to 6.2 mi (8 to 10 km) and had a subsurface rupture length of 6.8 mi (11 km) (Doser 1989a).

- Cache Valley. Reanalysis of seismograms for the August 30, 1962, Ms 5.7, Cache Valley earthquake indicates that it may be associated with the Temple Ridge fault, a less prominent feature with only 1,640 ft (500 m) of Neogene throw located east of the East Cache fault (Westaway and Smith 1989). Focal depth is estimated to be 6.2 ± 1.2 mi (10 ± 2 km), and focal mechanisms from first motions and body-wave analysis suggest a dip of 49 and 58 degrees, respectively, to the west (Table 2A.2-5). WCC (1992a) estimated Brune and root-mean-square stress drops of 25.2 ± 5.2 bars and 45.4 bars, respectively.

- Pocatello Valley. The March 28, 1975, M_b 6.1, Pocatello Valley earthquake occurred along a northeast-trending structure with a large left-lateral component of slip (Bache, Lambert, and Barker 1980). Studies of the aftershock sequence were consistent with a fault dip of 39 degrees to the northwest (Arabasz, Richins, and Langer 1979). The event originated at a focal depth of about 6 mi (10 km) (Table 2A.2-5) and has an inferred stress drop of about 50 bars for initial faulting (Bache et al. 1980).

2A.2.3.3 Yellowstone Plateau

2A.2.3.3.1 Hebgen Lake—The August 18, 1959, M_s 7.5 Hebgen Lake earthquake is the largest event to occur in the ISB region. Seismic waveform analysis by Doser (1985a) indicates that the mainshock was a double event consisting of subevent one (an M_b 6.3) followed 5 seconds later by subevent two (an M_b 7.0). 'Doser's analysis also suggests that the rupture occurred along one or more fault planes with east-west strike orientations (Table 2A.2-5), slightly discordant with the trace of surface faulting along the Hebgen and Red Canyon faults.

Maximum vertical displacements of 22 ft (6.7 m) over a surface scarp length of 14.3 mi (23 km) and 20 ft (6.1 m) over 9.6 mi (14.5 km) were observed along the Red Canyon and Hebgen faults (Myers and Hamilton 1964; Witkind 1964). A 3-ft (1-m) scarp was observed along a 1.8-mi (3-km) segment of a fault adjacent to Madison Canyon, but it was difficult to determine whether it was related to coseismic movement associated with the Hebgen Lake earthquake (Myers and Hamilton 1964).

Focal mechanisms derived from first motions and body-wave analysis for the subevents indicate fault dips ranging between 40 to 60 degrees to the southwest. Subevent 1 initiated at a focal depth of 6.2 mi (10 km) and subevent 2, 9.3 mi (15 km). The estimated stress drop for the mainshock is 115 bars (Doser 1985a).

2A.2.3.3.2 Yellowstone Caldera—The June 30, 1975, M_L 6.1, Yellowstone Park earthquake occurred near the northern rim of the Yellowstone caldera. The focal depth of this event was shallow: 3.7 mi (6 km). Aftershock studies and first motions suggest normal faulting along a northwest-trending structure dipping about 70 degrees to the northeast (Pitt, Weaver, and Spence 1979; Bache, Lambert, and Barker 1980).

2A.2.3.4 The Northern Rockies

2A.2.3.4.1 Clarkston Valley—The July 10, 1925, M 6.8, Clarkston, Montana earthquake was felt over an 309,400 mi^2 (800,000 km^2) area and reached a MMI of VIII in the epicentral area (Smith and Arabasz 1991). Although this earthquake was large, it produced no surface scarp, but some ground cracks were observed (Pardee 1926). Seismic-wave analysis indicates a focal depth of 5.6 mi (9 km), a rupture length of 15.5 mi (25 km), and oblique normal slip on a northwesterly-dipping plane trending north-south (Table 2A.2-5) (Doser 1989b).

2A.2.3.4.2 Virginia City—The November 23, 1947, M 6.3, Virginia City earthquake may be associated with rupture along a portion of the northwest-trending Madison Canyon fault based on first motions (Dewey et al. 1973). Reanalysis using seismic waveforms by Doser (1989b) suggests right-lateral slip along a fault striking east-west. Doser suggests that fault motion at depth in this part of the Hebgen Lake/Madison region occurs along structures striking nearly east-west and that the northwest strike of surface faulting may reflect the trend of preexisting weaknesses that the earthquake ruptures exploited as they propagated to the surface. The event originated at a focal depth of about 5 mi (8 km) (Doser 1989b).

Patterns of seismicity and locations of mapped faults have been used to assess potential sources of future earthquakes for estimating ground shaking at INEEL. The sources and maximum magnitudes of earthquakes that could produce the maximum levels of ground motions at ATR include (Figure 2A.2-11) (a) a magnitude 7.0 earthquake at the southern end of the Lemhi fault along the Howe and Fallert Springs segments, (b) a magnitude 7.0 earthquake at the southern end of the Lost River fault along the Arco segment, (c) a magnitude 5.5 earthquake associated with dike-injection in either the Arco or LR-HHA VRZs and the AVZ, and (d) a "random" magnitude 5.5 earthquake occurring within the ESRP (WCC 1992a). Ground-motion contributions from other sources, such as the postulated ESRP boundary fault, northern Basin and Range Province, Yellowstone Plateau, and Idaho Batholith, are significantly smaller because of their distant locations or lower maximum magnitudes.

2A.2.4 Maximum Earthquake Potential

2A.2.4.1 Lemhi Fault—Howe and Fallert Springs Segments, The Howe and Fallert Springs segments are located at the southern end of the Lemhi segment. The Howe segment is closest to INEEL (Figure 2A.2-11). ATR is located 11.8 mi (19 km) from the closest point of rupture along the Howe fault (WCC 1990). The most recent event (MRE) occurred between 15,000 and 24,000 years ago (WCC 1992b). The lengths of the Howe and Fallert Springs segments are approximately 9.3 to 12.4 and 15.5 to 18.6 mi (15 to 20 and 25 to 30 km), respectively (Turko 1988; Baltzer, Knuepfer, and Turko 1989; Haller 1988). Recent paleoseismic investigations (four trenches excavated across the faults) by WCC (1992a and b) indicate that the MRE could have ruptured portions of both the Howe and Fallert Springs segments resulting in a total length of 21.8 mi (35 km). For the MRE, maximum and average displacements are 8.2 and 4.9 ft (2.5 and 1.5 m), respectively (WCC 1992b). The maximum magnitude estimated for the MRE is 7.0 based on empirical data from Wells and Coppersmith (1994) using (a) surface-rupture length, (b) subsurface-rupture length, (c) rupture area [length × downdip extent: 19.3 × 13.1 mi (31 × 21 km)] (Figure 2A.2-12), (d) maximum displacement, and (e) average displacement (WCC 1992a). The slip rate of 0.004 in./yr (0.1 mm/yr) for both the Howe and Fallert Springs segments is lower than 0.012 in./yr (0.3 mm/yr) for the Thousand Springs segment of the Lost River fault, indicating that the Howe segment is less active (Crone et al. 1987).

2A.2.4.2 Lost River Fault—Arco Segment. The Arco segment is located at the southernmost end of the Lost River Fault and is the closest part of the fault to the INEEL (Figure 2A.2-11). The north and south boundaries of the Arco segment have been mapped at different locations by various investigators. The northern boundary was originally mapped at King Mountain (Scott 1982; Pierce 1985), but has more recently been pinpointed at Ramshorn Canyon (Crone et al. 1987; Crone and Haller 1991; Janecke 1993; Wu and Bruhn 1994). Woodward Clyde Federal Services (1996) use the Ramshorn Canyon boundary in their detailed analyses of fault behavior. The location of the southern boundary is less certain. Fault structures are mapped along the main range front to a point about 1 km south of Arco where they disappear in alluvium in the Arco basin (21 km total length, 9 km west of INEEL). Additional scarps are mapped about 2 km south of the main-range front scarps west of Butte City (Scott, Pierce, and Hait 1985). Woodward Clyde Federal Services (1996) support a location just west-southwest of Butte City, roughly 7 km west of the INEEL boundary and 25 km in total length. Wu and Bruhn (1994) suggest the boundary may extend as far as 7 km southeast of Butte City at a set of monoclinal features located about 1 km west of the INEEL boundary, thus making the segment 30 km long.

The MRE and penultimate events on the Arco segment occurred between 21±4 ka and 20±4 ka, possibly with contemporaneous rupture on the Pass Creek segment to the north. The best estimate of maximum magnitude for the Arco segment is M_W 7.0 with uncertainties of about ±1/4 magnitude, given the following factors: uncertainty of rupture lengths, uncertainty of whether displacements are more representative of average or maximum values, and the apparent discrepancy between length-based and displacement-based magnitudes. The net vertical displacement at the Arco Peak site (on the Arco segment) averages about 1.2 to 1.5 m per event. The average vertical slip rate between 58 and 20 ka was 0.12 mm/yr (WCFS 1996).

2A.2.4.3 Beaverhead Fault—Blue Dome Segment. The Blue Dome segment is located at the southernmost end of the Beaverhead fault (Figure 2A.2-10). ATR is located 28 to 31 mi (45 to 50 km) from the closest point of rupture along the Blue Dome fault. Based on scarp morphology, the MRE occurred more than 30,000 years ago (Stickney and Bartholomew 1987), and recent unpublished mapping in the area suggests that it has not been active for several hundred thousand years. The length of the segment is estimated to be about 15.5 mi (25 km) (Haller 1988). Because no detailed paleoseismic investigations have been conducted along this segment, and because faults northwest of INEEL are undergoing similar deformational processes, WCC (1992a) estimates a maximum magnitude of 7.0 for an earthquake along the Blue Dome fault. Data are not available to estimate a slip rate quantitatively, but several investigators suggest that this segment has an activity rate similar to the southern segments of the Lemhi and Lost River faults (Crone et al. 1987; Anders et al. 1989).

2A.2.4.4 Eastern Snake River Plain Boundary Faults. ATR is located about 7.5 mi (12 km) from the postulated ESRP boundary fault (Figure 2A.2-11). Deep seismic refraction profiling across the northwest boundary of the ESRP near the INEEL suggests the presence of a buried northeast-trending fault parallel to the ESRP boundary (Pankratz and Ackerman 1982). Scott (1982) suggested that a northeast-trending topographic scarp observed on an alluvial fan along the southeast side of the Arco Hills may be a result of past movement on the boundary fault. Breckenridge and Othberg (1991) excavated a trench across the scarp. Their trench logs indicated that no fault offset was present within the alluvial fan deposits. They concluded that the scarp may have developed from a buildup of eolian sediments along an old fire scar.

Other investigations have been conducted on northeast-trending faults at the southern terminations of the Lemhi Range and Beaverhead Mountains near the margins of the ESRP (Rodgers and Zentner 1988; Zentner 1989; Bruhn, Wu, and Lee 1992). Results of their studies indicate that these faults were active more than 2 Ma. Based on the following lines of evidence these faults are not considered significant seismogenic sources: (a) their northeast trend is not consistent with the direction of the active northwest-trending normal faults that are produced by regional extensional stress fields; (b) they do not displace sediments and volcanic rocks younger than 2 Ma; and (c) their lengths are small, generally less than 6.2 mi (10 km), and they have small total displacements. Furthermore, even if they were active, these faults would be capable of producing earthquakes of magnitude 5.0 based on their fault lengths and small displacements. This magnitude does not exceed the magnitudes of earthquakes that could occur in the Arco VRZ or randomly on the ESRP at equal or closer distances (WCC 1992a).

2A.2.4.5 Eastern Snake River Plain Volcanic Zones

2A.2.4.5.1 Arco Volcanic Rift Zone. The Arco VRZ extends from the southern end of the Lost River Range across the southwestern tip of the INEEL (Figure 2A.2-11). ATR is about 7.5 mi (12 km) away from the closest point on the boundary of the rift zone (Smith, Hackett, and Rogers 1989). The rift zone is about 3.7 mi (6 km) wide and 12.4 mi (20 km) long (Kuntz et al. 1990; Kuntz, Covington, and Schorr 1992). Small normal faults within the rift zone are 3.1 to 3.7 mi (5 to 6 km) in length, have cumulative vertical offsets of about 39 ft (12 m) (multiple offsets), and are postulated to extend to a depth of 1.2 mi (2 km) below the surface (Mastin and Pollard 1988; Smith, Hackett, and Rogers 1989; Kuntz et al. 1990; WCC 1992a). A set of fissures in the Box Canyon graben area is collinear with the small normal faults [3.1 mi (5 km length)] (Table 2A.2-8) bounding the graben, which results in a total length of 5 mi (8 km). Based on the compilation of earthquake data for active rift zones (Table 2A.2-6), a maximum magnitude of 5.5 is assumed possible for future dike-injection events within the rift zone. This value is conservative with respect to a magnitude of 5.2 based on the assumption that an earthquake associated with dike injection ruptures a fault area of 6.2 mi^2 (16 km^2) [length x depth: 5 × 1.2 mi (8 × 2 km)] (Figure 2A.2-12) (WCC 1992a). The most recent volcanic activity within the rift zone appears to have been about 95,000 years ago (Kuntz et al. 1986, 1990; Smith, Hackett, and Rogers 1989; Forman et al. 1993).

2A.2.4.5.2 Lava Ridge—Hell's Half-Acre Volcanic Rift Zone. The Lava Ridge—Hell's Half Acre (LR-HHA) VRZ extends from the southern end of the Lemhi range across the INEEL to the southeastern corner (Figure 2A.2-11). ATR is about 9.3 mi (15 km) away from the closest point on the boundary of the rift zone. The rift zone is 1.9 to 3.7 mi (3 to 6 km) wide and 31 mi (50 km) long. At the southern end of the rift zone, two sets of fissures, which may or may not be associated with small normal faults [0.9 mi (1.4 km) in Table 2A.2-8], are about 2.5 mi (4 km) in length (Golder 1992). Because portions of the fissures are covered by younger lava flows, the fissure sets could extend 6.8 mi (11 km) farther south. A maximum magnitude of 5.5 was assumed possible for earthquakes associated with future dike-injection events within the LR-HHA rift zone based on the compilation of earthquake data shown in Table 2A.2-6. This assumption is consistent with a magnitude of 5.5, which was estimated using fault area [9.3 × 1.9 mi = 17.5 mi^2 (15 × 3 km = 45 km^2)] and assuming rupture along the entire fissure lengths (WCC 1992a). The most recent volcanic activity within the LR-HHA rift zone occurred about 5,200 years ago (Kuntz et al. 1986, 1990).

2A.2.4.5.3 Howe—East Butte Volcanic Rift Zone. The postulated Howe—East Butte (H-EB) VRZ extends across the central portion of the INEEL from the range-front south of Howe to East Butte (Figure 2A.2-11). It is poorly expressed surficially and is mostly covered by younger lava flows and fluvial and lacustrine sediment (Kuntz 1978) (see Section 2A.6.2.3.1). ATR is located within the postulated H-EB VRZ. WCC (1992a) considered the maximum magnitude for the H-EB to be 5.5, similar to the Arco and LR-HHA VRZ. Volcanic vents in the H-EB VRZ are dated as 580,000 to 641,000 years old (Kuntz et al. 1990), and a conservative minimum age for the H-EB VRZ is estimated by its being overlain by a dated lava flow 230,000 years old.

2A.2.4.5.4 Axial Volcanic Zone. The Axial Volcanic Zone (AVZ) is located along the ESRP axis, and crosses portions of the INEEL's eastern boundary. ATR is about 9.3 mi (15 km) from the closest point of the AVZ boundary. Dike-induced features are located near the intersections of the Arco and LR-HHA VRZs with the AVZ. Thus, a maximum magnitude of 5.5 is assumed possible based on the interpretation that dike-injection mechanisms in the AVZ are similar to those in other ESRP VRZs. The most recent volcanic activity occurred 5,000 years ago (Kuntz et al. 1986).

2A.2.4.6 Eastern Snake River Plain Province. Although instrumental seismicity indicates that the ESRP is relatively aseismic, an earthquake similar in size to the 1905 Shoshone event is considered possible within the ESRP. For estimating ground motions at INEEL, an earthquake of maximum magnitude 5.5 is postulated to occur anywhere within a 15.5-mi (25-km) radius of each facility. This event is referred to as a random earthquake, and is commonly used for design of commercial nuclear reactors to assess effects from earthquakes that may occur on unknown faults (i.e., those without surface exposures).

2A.2.4.7 Northern Basin and Range Province. The northern Basin and Range source region surrounds the ESRP. Excluding known normal faults, which are capable of generating magnitude 7.0 events, an earthquake with a maximum magnitude of 6.75 is postulated to occur anywhere within this source region on an unknown subsurface structure (WCC 1992a). Doser (1985b) suggests that earthquakes of magnitude 6.0 to 6.75 could occur in the ISB without producing fault scarps for evidence of their past occurrence. An example of this phenomena is the 1975 M_L 6.0 Pocatello Valley earthquake near the Idaho-Utah border. This event occurred on a "blind" (not evident in surface geology) cross-fault, which trended transverse to the trend of nearby Basin and Range normal faults (Arabasz, Richins, and Langer 1981).

2A.2.4.8 Idaho Batholith Province. The Idaho Batholith is a seismically quiet region and its boundaries are defined by the extent of granitic rocks associated with the batholith. No extensive or well-defined Quaternary faults are mapped within the Idaho Batholith (WCC 1992a). Although seismographic coverage is poor (a detection threshold of $M \geq 3$), it appears to have a low seismic potential (Smith and Arabasz 1991). Woodward-Clyde Consultants (WCC 1992a) estimated the maximum magnitude to be M_w 5.5.

2A.2.4.9 Yellowstone Plateau Province. The Yellowstone Plateau Province includes the faults associated with the 1959 Hebgen Lake event, the largest earthquake in the ISB. Thus, the maximum magnitude is M_s 7.5 based on this event (WCC 1992a).

2A.2.5 Transmission Characteristics of Seismic Waves

2A.2.5.1 Regional Attenuation. For the ground-motion modeling studies, regional attenuation was characterized by a frequency-dependent quality factor, Q(f). Singh and Herrman (1983) determined a regional crustal coda Q_o of 450 and of 0.2 for Q(f) in the Basin and Range northwest of the ESRP. Braile et al. (1982) observed high attenuation in the 1978 ESRP seismic refraction experiment within the ESRP for the P-wave quality factor Q_p. They attributed it to low Q values in the volcanic rocks (Q_p 20 to 200) and throughout the crust (Q_p 160 to 300). Woodward-Clyde used the model parameters of Q_o and from Singh and Herrman (1983) in their 1990 deterministic analysis (WCC 1990), 1992 probabilistic analysis (WCC 1992a), and 1996 probabilistic analysis (WCFS 1996). They also suggest that the relatively short source-to-site distance of 12.4 mi (20 km) [less than 18.6 mi (30 km)] does not significantly attenuate earthquake ground motions.

2A.2.5.2 Near-Surface Geological Attenuation. WCC (1990) indicate that near-surface geology [1.2 to 3.1 (2 to 5 km) depth] has a significant influence on earthquake ground motions at a site, in addition to amplification resulting from soil overlying bedrock. The INEEL resides upon the ESRP, which is covered with basalt lava flows and sediments (see Section 2A.1.2.2). Boreholes located throughout the INEEL site indicate that the basalt is interbedded with sedimentary layers; in some areas, the percentage of interbeds is 50%.

This unique stratigraphy has the effect of attenuating or decreasing the level of earthquake ground motions because seismic waves travel through a sequence of alternating high- (basalt) and low- (sediments) velocity zones that tend to scatter the seismic energy. Also, seismic energy is intrinsically dampened by the sedimentary interbeds. The net effect of the interbedded basalt is to reduce the level of earthquake ground motions when compared to a homogeneous basalt (i.e., basalt with no interbeds) (WCC 1990). The amount of attenuation depends upon the difference between the velocities, numbers, and thickness of the basalt and sedimentary layers.

Figure 2A.2-13 shows the shear-wave velocity (V_s) profile determined to estimate earthquake ground motions at ATR (WCFS 1996). The velocity model was derived from logs of wells and borehole located near TRA. Regional earthquakes were digitally recorded near two boreholes at TRA. These data were used to estimate the near-surface attenuation, κ.

The evaluation summaries are in the following Section 2A.2.6. The Design-Basis Earthquake (DBE)/SSE for ATR has a ground acceleration of 0.24 g.

2A.2.6 Safe Shutdown Earthquake

2A.2.6.1 Ground-Motion Studies. Seismic-hazards assessments have been conducted at the INEEL since the early 1970s to evaluate potential earthquake ground motions and establish seismic design criteria. Since that time, ground-motion seismology and Federal regulations have continued to evolve, and geoscience investigations have continued at INEEL. In keeping pace with these changes, a deterministic (worst-case scenario) earthquake-hazard assessment was conducted recently for all facilities at INEEL (including ATR) using new geologic data and state-of-the-art ground motion modeling techniques (WCC 1990). Deterministic and probabilistic hazards assessments conducted for the proposed New Production Reactor (NPR) site at INEEL evaluated ground-motion contributions from the "random" ESRP earthquake (WCC 1992a). Also, an INEEL probabilistic seismic-hazard assessment is currently underway to assess the contributions from all earthquake sources on the levels of ground motions and their return periods. This study includes an evaluation for ATR.

2A.2.6.1.1 1990 INEL Deterministic Evaluation—The 1990 deterministic study estimated peak ground accelerations for ATR based on the largest earthquake (M_L 7) that could occur along the Lemhi fault (WCC 1990). This evaluation incorporated all available results from geoscience investigations pertaining to the earthquake source, subsurface stratigraphy beneath the ESRP (crustal structure), and ATR's near-surface stratigraphy. WCC (1990) developed a site-specific geologic profile beneath the facility to assess the nature of seismic-wave propagation from the Howe fault to ATR.

The geologic profile was used with the Band-Limited-White-Noise (BLWN) ground-motion model to determine site-specific accelerations for expected levels of earthquake ground motions at ATR. Sensitivity analyses performed on the results indicated that the size of the earthquake (stress drop) and near-surface geology (κ) had the most significant effects on the levels of earthquake ground motions. Peak horizontal and vertical accelerations and response spectra were estimated for the 50th and 84th percentiles based on the range of uncertainties in the geologic input.

For ATR, WCC (1990) estimated peak ground (horizontal) accelerations (PGA) of 0.117, 0.176, and 0.265 g for the 16th, 50th, and 84th percentiles, respectively. These values were compared to PGA of 0.255, and 0.203 g from the empirical relationships of Joyner and Boore (1988) and Campbell (1992), respectively (WCC 1990). Figure 2A.2-14 shows the response spectra for ATR at various return periods (WCFS 1996).

2A.2.6.1.1.1 The INEEL Probabilistic Evaluation—The INEEL site specific probabilistic seismic-hazard study (WCFS 1996) includes an assessment for ATR that provides seismic hazard curves (peak ground accelerations versus return periods) and response spectra at various return periods. The geologic input is similar to the input used in the NPR probabilistic evaluation, with minor exceptions (WCC 1992a). Geologic profiles for ATR were refined based on new borehole information and additional estimates of Vs. The BLWN ground-motion model was used along with empirical relationships. Sensitivity analyses were performed to determine seismic sources that contribute most to the hazard.

All known earthquake sources are considered in the probabilistic assessment: Basin and Range faults north and south of the ESRP, volcanic (dike-induced) seismicity of the ESRP VRZs, background seismicity of the ESRP (random earthquakes), background seismicity of the Basin and Range north and south of the plain, and seismicity of the Idaho batholith area.

2A.2.6.2 Safe Shutdown Earthquake Determination .The Design-Basis Earthquake (DBE)/SSE for ATR has a ground acceleration of 0.24 g (see Harris 1989; DOE-ID 1995). The U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.60 (NRC 1973) horizontal spectra are appropriately scaled using the SSE value. The vertical input spectra are two-thirds of the horizontal input spectra (DOE-ID 1995). The DBE/SSE value is supported by the results of the WCC (1990) deterministic evaluation.

The DOE-ID AE Standards (DOE-ID 2002) have been revised to incorporate new seismic design requirements of DOE Standards DOE-STD-1020 (DOE 2002a) and DOE-STD-1021 (DOE 2002b). The TRA site specific Performance Categories PC 1 and PC 2 seismic design criteria use the International Building Code (IBC 2000) spectral accelerations. The TRA site specific PC 3 and PC 4 horizontal and vertical seismic design criteria for rock Design Basis Earthquake/Safe Shutdown Earthquake (DBE/SSE) is available in the DOE-ID Standard (DOE-ID 2002). The TRA site specific PC 3 and PC 4 horizontal and vertical seismic design criteria for soil DBE is under development, and will be incorporated into the AE Standards when completed.

2A.2.7 Operating Basis Earthquake

None defined.

Table 2A.2-1. Time periods of earthquake data completeness.[a]

Magnitude Interval	Completeness Period
2.0–4.0	1975–1989
4.0–5.0	1963–1989
5.0–5.5	1950–1989
5.5–6.0	1925–1989
6.0–6.5	1900–1989
6.5–7.0	1875–1989
7.0 +	1850–1989

a. Modified from WCC (1992a).

Table 2A.2-2. Earthquakes with magnitudes greater than 5.5 within 200 mi of ATR.

Earthquake date and time (Hr:Mn – UTC)[a]	Magnitude[b]	Modified Mercalli intensity[c]	Geographical location[d]	Radial distance (km)[e]	References
1884 November 10 08:50	6.3 M_I	VIII	Bear Lake, Utah	225	1
1905 November 11 21:26	5.5 M_L	VII	Shoshone, Idaho	164	2
1909 October 6 02:50	6.3 M_I	VIII	Hansel Valley, Utah	216	1
1914 May 13 17:15	5.7 M_I	VII	Ogden, Utah	283	1
1925 July 10 14:45	6.8 M6.6 M_w	VI	Clarkston, Montana	201	3,4,5
1930 June 12 09:15	5.8 M_L	VI	E of Soda Springs, Idaho	190	5
1934 March 12 15:05	6.6 M_L6.6 M_w	IX	Hansel Valley, Utah	222	1,3,6
1934 March 12 18:20	6.2 M_L5.9 M_w	VII	Hansel Valley, Utah[f]	222	1,3,6
1934 April 14 21:26	5.6 M_L	VII	Hansel Valley, Utah[f]	245	1
1934 May 06 08:09	5.6 M_L	VI	Hansel Valley, Utah[f]	222	1
1944 July 12 19:30	6.1 M_b	VII	N of Stanley, Idaho	235	7
1945 February 14 03:01	6.0 M_L	VI	N of Stanley, Idaho	235	7
1947 December 17 12:38	6.3 M6.1 M_w	ND	Virginia City, Montana	225	3,4
1959 August 18[g] 06:37	7.5 M_s 6.3, 7.3 M_w	X	Hebgen Lake, Montana	187	3,8,9,10
1959 August 18 07:56	6.5 M	ND	Hebgen Lake, Montana[h]	208	3
1959 August 18 08:41	6.0 M	ND	Hebgen Lake, Montana[h]	208	3
1959 August 18 11:03	5.6 M	ND	Hebgen Lake, Montana[h]	182	3
1959 August 18 15:26	6.5 M_b6.3 M_w	ND	Hebgen Lake, Montana[h]	209	10
1959 August 19 04:04	5.9 M_s6.0 M_w	ND	Hebgen Lake, Montana[h]	209	4,10

Table 2A.2-2. (continued).

1962 August 30 13:35	5.7 M_s5.9 M_w	VII	Cache Valley, Utah	208	1,3,11
1964 October 21 07:38	5.8 M_b5.6 M_w	ND	Hebgen Lake, Montana[h]	154	3,4
1975 March 28 02:31	6.1 M_b6.2 M_w	VIII	Pocatello Valley, Utah	183	3,12
1975 June 30 18:54	6.1 M_L	VII	Yellowstone Park, Wyoming	209	3,13
1976 December 8 14:40	5.5 M_b	ND	Yellowstone Park, Wyoming	198	5
1983 October 28 14:06	7.3 M_s6.8 M_w	IX	N W of Mackay, Idaho	93	3,8,9,14
1983 October 28 19:51	5.8 M_L5.4 M_w	ND	N W of Mackay, Idaho	98	3,15
1983 October 29 23:29	5.8 M_L5.5 M_w	ND	N W of Mackay, Idaho	121	3,15
1984 August 22 09:46	5.8 M_L5.6 M_w	ND	Challis, Idaho[i]	127	3,15
1994 February 3 09:05	5.9 M_w	ND	West of Afton, Wyoming	167	16

a. UTC - Universal Time Coordinated (Greenwich Mean Time).

b. Highest magnitude value is reported in this Table. Moment magnitudes are included, if calculated. Magnitude Scales: M_I – Conversion from Intensity; M_L - Local or Richter; M - magnitude type not specified; M_w - Moment; M_b - Body-wave; M_s - Surface-wave.

c. Modified Mercalli intensity based on Wood and Neumann (1931).

d. Latitude and Longitude coordinates are listed in Appendix 2-A.

e. Radial distances based on coordinates 43o 42.0', 112o 48.0'.

f. Aftershock following the M_L = 6.6, 1934 Hansel Valley, Utah earthquake.

g. Hebgen Lake usually referred to as M_s 7.5, but is actually two events having magnitudes of M_w 6.3 and 7.3 per Doser 1985a.

h. Aftershock following the M_s = 7.5, 1959 Hebgen Lake, Montana earthquake.

i. Aftershock following the M_s = 7.3, 1983 Borah Peak, Idaho earthquake.

References: (1) Richins 1979; (2) Oaks 1992; (3) Doser and Smith 1989; (4) Doser 1989b; (5) Engdahl and Rinehart 1991; (6) Doser 1989a; (7) Stover, Reagor, and Algermissen 1986; (8) Doser 1985b; (9) Stover 1985; (10) Doser 1985a; (11) Westaway and Smith 1989; (12) Arabasz, Richins, and Langer 1979; (13) Pitt, Weaver, and Spence 1979; (14) Doser and Smith 1985; (15) Richins et al. 1987; (16) Nava, Arabasz, and Pechmann 1994.

Table 2A.2-3. Ground motions recorded during the Borah Peak earthquake at TRA.

	Acceleration (cm/sec^2)		Velocity (cm/sec)		Displacement (cm)	
Component	ATR	ETR	ATR	ETR	ATR	ETR
Longitudinal	0.022	0.030	1.12	2.15	0.16	0.21
Transverse	0.023	0.029	1.45	1.88	0.25	0.30
Vertical	0.019	0.018	0.72	1.07	0.12	0.13
Reference: Jackson et al. 1991.						

Table 2A.2-4. Basin and range faults around the Eastern Snake River Plain.[a]

Fault	Reference	Important points
Sawtooth, White Cloud Peaks area	Dewey (1987), Jackson and Zollweg (1988), Smith, Richins, and Doser (1985)	Contemporary earthquake swarms, maximum magnitude = 6.1, several mapped normal faults.
Lost River Fault	Pierce (1985, 1988), Scott, Pierce, and Hait (1985), Crone et al. (1987), Schwartz (1988), Malde (1987), Piety et al. (1986)	**Arco Segment** - MRE~30 Ka, trenching, D~2-3 m, L~10 km, SR~0.12 mm/y, RI~30-40 Ka. **Pass Creek Segment** -MRE~30-50 Ka, scarp morphology, ND, L~30 km, RI~30-50 Ka? **Mackay Segment** -MRE~4.3-6.8 Ka, trenching, ND, L~22 km, SR~0.3 mm/yr, RI~4-7 Ka. **Thousand Springs Segment** - MRE=1983, trenching/earthquake, D=2.7 m, L=36 km, SR=0.3 mm/yr, RI=6-7Ka. **Warm Spring Segment** - MRE~5.5-6.2 Ka, trenching, ND, L~15 km, SR=0.3 mm/yr, RI< 15 Ka. **North Segment** - MRE > Late Quaternary, scarp morphology, ND, L~20 km., ND, ND, low structural relief.
Lemhi Fault	WCC (1992b), Knuepfer (1989), Turko (1988), Baltzer, Knuepfer, and Turko (1989), Malde (1987), Haller (1988), Scott, Pierce, and Hait (1985), Bruhn, Wu, and Lee (1992)	**Southern Segments (Howe and Fallert Springs segments)** - MRE~15-24 Ka, trenching/scarp morphology, D~2-3 m, L~25 km, SR~0.1 mm/yr, RI=3.3 Ka (ave.) **Sawmill Gulch Segment** - MRE < 10 Ka, trenching, D=1.7 m, L=43 km, ND, ND. **Goldburg Segment** - MRE~10-15 Ka, scarp morphology, ND, L=12 km, ND, ND. **Patterson Segment** - MRE < 10 Ka?, scarp morphology, ND, L~23 km, ND, ND. **May Segment** - MRE~15-30 Ka?, scarp morphology, ND, L~23 km, ND, ND.

Table 2A.2-4. (continued).

Fault	Reference	Important points
Beaverhead Fault	Haller (1988), Stickney and Bartholomew (1987)	**Blue Dome Segment** - MRE > 30 Ka, scarp morphology, ND, L~25 km, ND, ND. **Nicholia Segment** - MRE~10-15 Ka, scarp morphology, ND, L~42 km, ND, ND. **Baldy Mountain Segment** - MRE > 30 Ka, scarp morphology, ND, L~21 km, ND, ND. **Leadore Segment** - MRE < 10 Ka, scarp morphology, ND, L~23 km, ND, ND. **Mollie Gulch Segment** - MRE~10-15 Ka?, scarp morphology, ND, L~20 km, ND, ND. **Lemhi Segment** - MRE > 30 Ka, scarp morphology, ND, L~20 km, ND, ND.
Red Rock Fault	Haller (1988), Stickney and Bartholomew (1987)	**Sheep Creek Segment** - MRE < 10 Ka, scarp morphology, ND, L~16 km, ND, ND. **Timber Butte Segment** - MRE~10-15 Ka, scarp morphology, ND, L~11 km, ND, ND.
Centennial Fault	Stickney and Bartholomew (1987), Johnson (1981)	**Western Centennial Valley Segment** - MRE < 10 Ka, scarp morphology, ND, L~23 km, ND, ND. **Red Rocks Lake Segment** - MRE > 20 Ka?, scarp morphology, ND, L~24 km, ND, ND. **Henrys Lake Segment** - MRE < 10 Ka, scarp morphology, ND, L~1 km, ND, ND.
Madison Fault	Stickney and Bartholomew (1987), Schneider (1985)	**Madison Canyon Segment** - MRE~Late Holocene (1947?, 1959?), scarp morphology, ND, L~34 km (total fault length = 117 km). Additional scarps exist but no segments have been delineated (a short segment of this fault ruptured in 1959).
Hebgen Fault and Red Canyon Fault	Stickney and Bartholomew (1987), Doser (1985a)	**Hebgen Fault** - MRE=1959, scarp morphology, D=6.7 m, L~13 km (+ 14 km on R.C.Fault), ND, ND. **Red Canyon Fault** - MRE=1959, scarp morphology, D=6.7m, L~45 km, SR~1.2-1.5 mm/yr (pre-1959), RI=4.3 Ka.
Yellowstone area	Pitt, Weaver, and Spence (1979), USGS (1972), Doser (1985a)	Numerous north-trending normal faults around Yellowstone Caldera with Quaternary movement. Contemporary seismicity, maximum magnitude = 6.1. RI = 700-750 years for M7 earthquakes based on seismic moment rates.

Table 2A.2-4. (continued).

Fault	Reference	Important points
Teton Fault	Piety et al. (1986), Smith et al. (1990), Byrd et al. (1988), Susong, Smith, and Bruhn (1987), Gilbert, Ostenaa, and Wood (1983)	**South Segment** - MRE~7Ka, trenching and scarp morphology, D=4.1 m, L~24 km, SR~1.7-2.2 mm/yr, RI~1.4-2.3 Ka. **Middle Segment** - MRE < 11-14 Ka, scarp morph., D~3m, L~20 km, SR~1.7-2.2 mm/yr, RI~1.4-2.3Ka. **North Segment** - MRE < 11-14 Ka, scarp morph., D~3 m, L~20 km, SR~1.7-2.2 mm/yr, RI~1.4-2.3 Ka.
Grand Valley Fault	Anders et al. (1989), Piety et al. (1986)	**Grand Valley Fault** - MRE > 15-30 Ka, scarp morphology, ND, L~72 km, SR~0.02-0.04 mm/yr, ND.
Snake River Fault	Anders et al. (1989), Piety et al. (1986).	**Snake River Fault** - MRE > 15-30 Ka, scarp morphology, ND, L~50 km, SR~0.001 mm/yr, ND
Star Valley Fault	Anders et al. (1989), Piety et al. (1986), McCalpin, Piety, and Anders (1990)	**Northern Segment** - MRE < 9 Ka, scarp morphology, D~3.6-6.3, L~30 km, SR~0.8-1.2 mm/yr, RI~5-7Ka. **Southern Segment** - MRE < 9 Ka, trenching and scarp morphology, D~5.0-6.3 m, L~28 km, SR~0.6-1.1 mm/yr, RI~5-7 Ka.
Northern Wasatch Fault Segments	Schwartz (1988), Machette, Personius, and Nelson (1992), Swan, Schwartz, and Cluff (1980), Piety et al. (1986).	**Collinston Segment** - MRE > 13 Ka, scarp morphology, ND, L~25 km, ND, ND. **Brigham City Segment** - MRE=3400 yrs, trenching, ND, L~40 km, ND, ND. **Weber Segment** - MRE~500 years, 1.7-3.7 m, trenching, D~1.7-3.7m, L~50 km, SR~1.2-2.8 mm/yr, RI~1 Ka?

a. For each segment or fault, the information under Important Points is presented as follows: Most recent event (MRE) in thousands of years (Ka), type of study, displacement per last event (D), length (L), slip rate (SR), and recurrence interval (RI). ND = no data available.

Table 2A.2-5. Earthquakes within 200 mi that have occurred on tectonic structures.

Earthquake date and time (Hr:Mn - UTC)[a]	Seismic moment[b] ($\times 10^{25}$ dyne-cm)	Focal mechanism strike/dip/rake[c] (degrees)	Tectonic structure, source parameters and dimensions, and references[d]
1925 July 10 14:45	10 ± 2 B	30 80 -175 FM 250 56 - 38 BW	Associated with a fault oriented in an oblique manner north of the Clarkston Valley Fault north of Bozeman, Montana. Z=9± 5 km (LP); RL=25± 5 km (BW), 59±5 km (SF); SD=2.0± 1.0 m (v). (1,2)
1934 March 12 15:05	0.95 G 8.6 ± 2 B	7 80 - 70 FM 40 87 - 11 BW 0 73 -110 SF	Caused a fault scarp along an unnamed fault in Hansel Valley, Utah. Z= 8± 2 km (LP); RL=11± 3 km (BW), 6± 2 km (SF); BWD= -2.1± 0.1 m (h), 0.2± 0.05 m (v); SD= -0.2 (h), 2.0± 1.0 m (v); GD= 0.4± 0.1 m (v). (1,3)
1934 March 12 18:20	0.77 ± 0.3 B	25 85 - 20 BW	Aftershock to March 12, 1934 earthquake. Z= 8± 7 km (LP); RL=7± 3 km (BW); BWD= -0.5± 0.1 m (h). (1,4)
1947 December 17 12:38	1.8 ± 0.5 B	120 60 -120 FM 104 48 -170 BW	Possibly associated with the Madison Fault northwest of Hebgen Lake, Montana. Z= 8± 2 km (LP); RL=9± 2 km (BW); BWD= -0.7± 0.2 m (h). (1,2,5)
1959 August 18 06:37 (M_s 7.5)	41 G 150 L 120 S	102 60 - 90 SW 120 70 - 90 SF 132 45 - 90 GE	Caused a fault scarp along the Hebgen and Red Canyon faults near Hebgen Lake, Montana. No distinction between subevents: Z= 11± 2 km (LP); RL=24± 4 km (SF), 40± 4 km (GE); SD= 4.4 m (v); GD= 7.4± 0.4 m (v). (1,6)

Table 2A.2-5. (continued).

Earthquake date and time (Hr:Mn - UTC)[a]	Seismic moment[b] ($\times 10^{25}$ dyne-cm)	Focal mechanism strike/dip/rake[c] (degrees)	Tectonic structure, source parameters and dimensions, and references[d]
1959 August 18 06:37 (M_w 6.3)	2.8 B	102 60 - 90 FM 95 42 - 90 BW	Subevent 1: Z= 10± 2 km (LP); RL=7± 1 km (BW); BWD= 0.95 m (v). (1,6)
1959 August 18 06:37 (M_w 7.3)	92 B	100 54 - 90 FM 95 42 - 90 BW	Subevent 2: Z= 15± 3 km (LP); RL=21± 5 km (BW); BWD= 6.8 m (v). (1,6)
1959 August 18 07:56	nd	70 55 - 45 FM	Aftershock to August 18, 1959 earthquake. nd. (1)
1959 August 18 08:41	nd	70 65 - 15 FM	Aftershock to August 18, 1959 earthquake. nd. (1)
1959 August 18 11:03	nd	50 64 31 FM	Aftershock to August 18, 1959 earthquake. nd. (1)
1959 August 18 15:26	3.10 B 5.5 S	90 60 - 70 FM 83 50 - 90 BW	Aftershock to August 18, 1959 earthquake. Z= 10± 2 km (LP); RL=9± 1 km (BW). (1)
1959 August 19 04:04	1.1 ± 0.3 B 4.8 S	60 75 -155 FM 57 80 -161 BW	Aftershock to August 18, 1959 earthquake. Z= 8± 2 km (LP); RL=11± 2 km (BW). (1)
1962 August 30 13:35	0.52 ± 0.2 B	185 58 - 85 FM 201 49 -108 BW	Associated with the Temple Ridge fault, Cache Valley, Utah. Z= 12± 2 km (LP); RL=3± 1 km (BW); BWD= 0.55 ± 0.2 m (h). (1,7)
1964 October 21 07:38	1.10 ± 0.3 B	310 60 18 FM 307 56 14 BW	Aftershock to August 18, 1959 earthquake. RL=3± 1 km (BW). (1)

Table 2A.2-5. (continued).

Earthquake date and time (Hr:Mn - UTC)[a]	Seismic moment[b] ($\times 10^{25}$ dyne-cm)	Focal mechanism strike/dip/rake[c] (degrees)	Tectonic structure, source parameters and dimensions, and references[d]
1975 March 28 02:31	1.23 ± 0.6 B 2.4 L 1.2 S	225 39 - 53 FM 200 38 - 70 BW 210 60 - 90 GE	Associated with an unnamed fault in Pocatello Valley, Utah. Z= 9± 2 km (LP), 5± 2 km (SP), 12 km (GE); RL=12± 2 km (BW), 18± 2 km (GE); BWD= 0.75 ± 0.25 m (v); GD= 0.50 m (v). (1,8,9)
1975 June 30 18:54	0.75 S	302 71 -129 FM	Associated with an unnamed fault near the north-central boundary of the Yellowstone Caldera, Wyoming. Z= 6± 1 km (SP); GD= 0.12 m (v). (1,9,10)
1983 October 28 14:06	28 G 21 B 29 L	138 45 - 60 FM 155 50 - 65 BW 160 70 - 70 SF 152 49 nd GE	Caused a fault scarp along the Thousand Springs segment of the Lost River Fault in central Idaho. Z= 16± 4 km (LP), 12± 2 km (SP), 14 km (GE); RL= 21± 2 km (BW), 19± 2 km (SF), 20± 2 km (GE); BWD= -0.20 m (h), 1.30 m (v); SD= -0.30 m (h), 1.50 m (v); GD= 2.10 m (v). (1,11)
1983 October 28 19:51	0.13 B 0.24 S	287 58 -165 FM 282 48 -159 BW 286 70 -155 SF	Aftershock to October 28, 1983 earthquake. Z= 10 km (LP), 10 km (SP), 10 km (GE); RL= 6± 2 km (BW). (1)
1983 October 29 23:29	0.20 B	309 51 - 65 FM 317 45 - 90 BW	Aftershock to October 28, 1983 earthquake. Z= 19± 9 km (LP), 10 km (SP); RL= 8± 1 km (BW). (1)
1984 August 22 09:46	0.24 B	170 70 - 5 FM 348 85 -160 BW	Aftershock to October 28, 1983 earthquake. Associated with the Challis segment of the Lost River fault and possibly caused slip (M 5.0) on an antithetic fault, the Lone Pine fault, central Idaho. Z= 10 km (LP), 10 km (SP), 12± 0.7 km ($_pP$); RL= 7± 1 km (BW). (1,12)

Table 2A.2-5. (continued).

Earthquake date and time (Hr:Mn - UTC)[a]	Seismic moment[b] ($\times 10^{25}$ dyne-cm)	Focal mechanism strike/dip/rake[c] (degrees)	Tectonic structure, source parameters and dimensions, and references[d]
1994 February 3 09:05	nd	nd	Occurred 18 km to the east of the west-dipping Star Valley fault. Preliminary results suggest it was associated with unknown fault. The mainshock was located at the northern end of the aftershock zone suggesting rupture propagated to the south. These aftershocks exceeded magnitude 5.0. Z= nd. (13)

a. UTC – Universal Time Coordinated (Greenwich Mean Time).

b. Seismic moments based on: G - geology; B - body-wave analysis; L - Geodetic observations; S surface-wave analysis from Doser and Smith (1989).

c. Focal mechanisms based on: FM - first motions; BW - body-wave analysis; SF - surface-wave analysis; SF - surface faulting; GE - geodetic observations from Doser and Smith (1989).

d. Earthquake source parameters and dimensions: Z - focal depth from long-period waves (LP), short period waves (SP), geodetic observations (GE) or body wave depth phases ($_pP$); RL – rupture length from body waves (BW), surface faulting (SF), and geodetic observations (GE); BWD - body-wave displacement, SD - surface displacement, and GD - geodetic displacement for horizontal (h) and vertical (v) orientations. Numbers in parentheses corresponds to references listed below. For obtaining original references see Doser and Smith (1989).

References: (1) Doser and Smith (1989); (2) Doser (1989b); (3) Shenon (1936); (4) Doser (1989a); (5) Dewey et al. (1973); (6) Doser (1985a); (7) Westaway and Smith (1989); (8) Cook and Nye (1979); (9) Bache, Lambert, and Barker (1980); (10) Pitt, Weaver, and Spence (1979); (11) Crone et al. (1987); (12) Jackson (1994a); (13) Nava, Arabasz, and Pechmann (1994).

Table 2A.2-6. Maximum magnitudes of earthquakes associated with dike injection in volcanic rift zones.[a]

Location[b]	Rifting event[c] (year)	Maximum magnitude[d]	References
Iceland			
Krafla	1975-76	5.0[e]	1,2
Krafla	1977	3.8[e]	3
Krafla	1978	4.1[e]	4
Hawaii, USA			
Kilauea Rift Zones			
East	1965	4.4(M_L)	5
East	1968, Aug.	3.3	6
East	1968, Oct.	3.1	6
East	1969	4.7	7
Southwest	1975	3.0	8
East	1976-77	4.0	8
East	1980, Aug.	3.0(M_c)[f]	9
East	1980, Nov.	3.1(M_c)[f]	9
Southwest	1981	3.4(M_c)[f]	9
East	1982	3.0(M_c)[f]	9
Japan			
Izu Peninsula	1989	5.5(M_{JMA})	10,11,12
Africa			
Asal, Afar	1978	5.3(M_b)	13,14
New Zealand			
Taupo Volcanic Zone[g]	1964-65	4.6	15
Taupo Volcanic Zone[g]	1983	4.3	15
California, USA			
Mono Craters[h]	1325 ± 20	> 6.5(M_s)	16
Mean and 1-sigma/n=18[i]		4.1 ± 1.0	17

a. Modified from Hackett, Jackson, and Smith (1996).

b. Worldwide dike-injection events associated with mafic magma; compositions of magma for Japan and New Zealand episodes are unknown.

c. An episode of dike-injection and associated seismicity having a beginning and end.

d. Maximum magnitude reported for the dike-injection event. Magnitudes: ML - Local or Richter; Mc - Coda; MJMA - Japan Meteorological Agency; Mb - Body-wave; Ms - Surface-wave.

e. Einarsson (1991) reports earthquakes of magnitude > 5.0 are usually associated with caldera deflation events and magnitude < 4.0 with dike injection at Krafla.

f. Coda magnitudes greater than amplitude magnitudes for these events (Nakata, Tanigawa, Klein 1982; Tanigawa, Nakata, and Klein 1981 and 1983).

g. Possibly associated with tectonic subsidence of the basin or triggered by dike injection (14).

h. Minimum estimate of the largest of five historic earthquakes based on liquefaction deposits as per (16). These earthquakes may be associated with movements along tectonic faults.

i. Mean and one standard deviation computed based on magnitudes as presented.

nd: No data obtained. References: (1) Einarsson & Bjornsson (1979); (2) Bjornsson et al. 1977; (3) Brandsdottir and Einarsson 1979; (4) Einarsson and Brandsdottir 1980; (5) Bosher and Duennebier 1985; (6) Jackson et al. 1975; (7) Swanson et al. 1976; (8) Dzurisin et al. 1980; (9) Karpin and Thurber 1987; (10) Okada and Yamamato 1991; (11) Takeo 1992; (12) Oura, Yoshida, and Judo 1992; (13) Abdallah et al. 1979; (14) Lepine and Hirn 1992; (15) Grindley and Hull 1986; (16) Sieh and Bursik 1986; (17) Hackett, Jackson, and Smith 1996.

Table 2A.2-7. Historical earthquakes possibly located within the Eastern Snake River Plain.[a]

Date	Origin time (UTC)[b]	Intensity[c]	Magnitude	Location error (km)[d]
11 November 1905	22:29[e]	MM VII	M_L 5.5[f]	± 100-200
20 February 1909	01:nd[j]	MM II	nd	± 50-100
6 December 1925	16:16	Felt	nd	± 50-100
7 August 1927	nd	Felt	nd	± 50-100
5 September 1928	05:36	Felt	M_L 5.2[g]	≥ ± 100
6 June 1932	11:00	MM II	nd	± 50-100
21 December 1932	08:00	MM II	nd	± 50-100
28 April 1934	09:30	MM IV	nd	± 100-200
28 April 1934	10:00	MM III	nd	± 100-200
29 April 1934	06:10	MM III	nd	± 100-200
18 November 1937	23:50	nd	M_L 5.4[h]	nd
1 February 1954	03:33:19.[i]	nd	nd	± 50-100
20 January 1964	10:09:39.7[i]	nd	nd	± 22-56
28 February 1969	15:30:24.4[i]	nd	nd	± 22-56

a. Modified from WCC (1992a).

b. Origin time shown as hour:minute:second for Universal Time Coordinated (UTC), as reported in Stover, Reagor, and Algermissen (1986).

c. MM = Modified Mercalli Intensity [Wood and Neumann (1931)]. Felt = Earthquake felt but no intensity assigned.

d. Estimated from Stover, Reagor, and Algermissen (1986).

e. According to Oaks (1992) Shoshone earthquake occurred at 22:29 and not 21:29 as reported by Stover, Reagaor, and Algermissen (1986).

f. Richter magnitude (M_L) calculated by Oaks (1992).

g. Magnitude obtained from Slemmons, Jones, and Gimlett (1965).

h. Information obtained from Engdahl and Rinehart (1988).

i. Instrumentally computed location.

j. nd = No data.

Table 2A.2-8. Calculated moment magnitudes from fault area for normal faults in Eastern Snake River Plain volcanic rift zones.

Rift zone	Normal fault	Length[a] (km)	Depth extent[b] (km)	Fault area[c] (km^2)	Moment magnitude[d]
Arco, ESRP[e]	Railroad Monocline	5	4	20	5.32
	Jaylin Monocline	1.3	1.3	1.69	4.25
	Box Canyon-NE	1.3	1.3	1.69	4.25
	Box Canyon-SW	3	3	9	4.97
	East-West	0.65	0.65	1.2	4.13
	Section 6	0.50	0.50	1.0	3.62
	Sect. 11	0.20	0.20	0.4	3.62
	Sect. 3	0.10	0.10	0.20	3.32
	Sect. 8	0.30	0.30	0.60	3.80
	Sect. 17	0.50	0.50	1.00	4.02
	Sect. 7	0.30	0.30	0.60	3.80
LR-HHA, ESRP[e]	Lapoint Monocline	1.4	1.4	2.8	4.47
Mean +1-sigma (N=12)					4.13 ± 0.58

a. Maximum normal fault length observed at the surface in the VRZ having vertical offset and produced by dike-injection (Jackson 1994b).

b. Maximum downdip width of the normal fault is based on depth for the level of neutral buoyancy, 4 km (Ryan, 1987). In cases where lengths are less than 4 km, the downdip width is assumed to be equivalent to its length. The intent is to estimate the maximum area possible.

c. Surface length × downdip width.

d. Calculated using: $M_w = 4.01 + 1.0 \times \mathrm{Log}_{10}(A)$; where A = fault-area and M_w = moment magnitude (Wells and Coppersmith 1994). The relationship is independent of the sense of slip.

e. Smith, Hackett, and Rogers (1989); Golder (1992); Kuntz, Covington, and Schorr (1992).

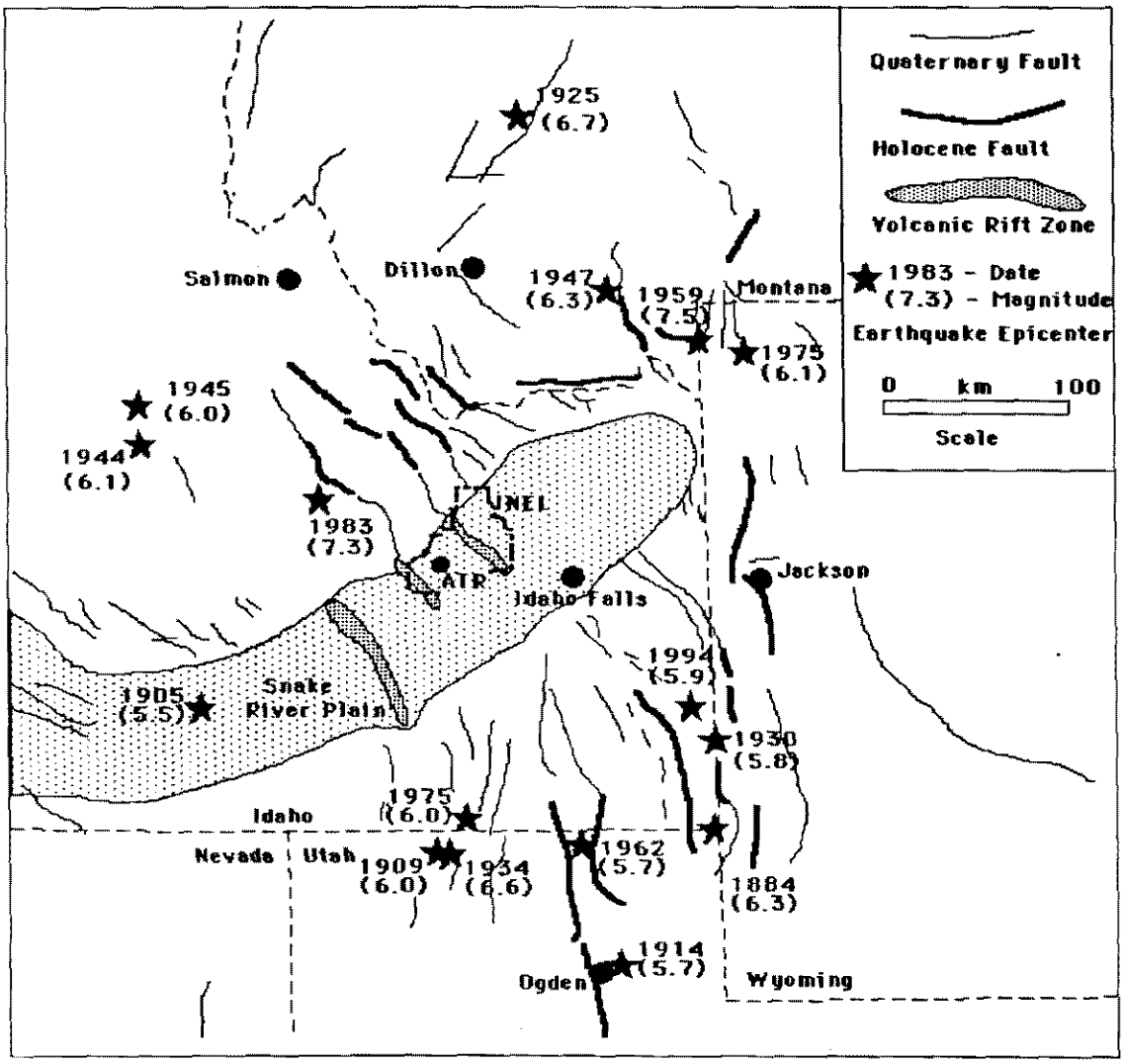


Figure 2A.2-1. Map showing the locations, dates, and magnitudes of observed earthquakes with M> 5.5 in the INEEL area (also shown are locations of Quaternary and Holocene normal faults in the basin and range province and volcanic-rift zones of the Eastern Snake River Plain). **(For Information Only)**

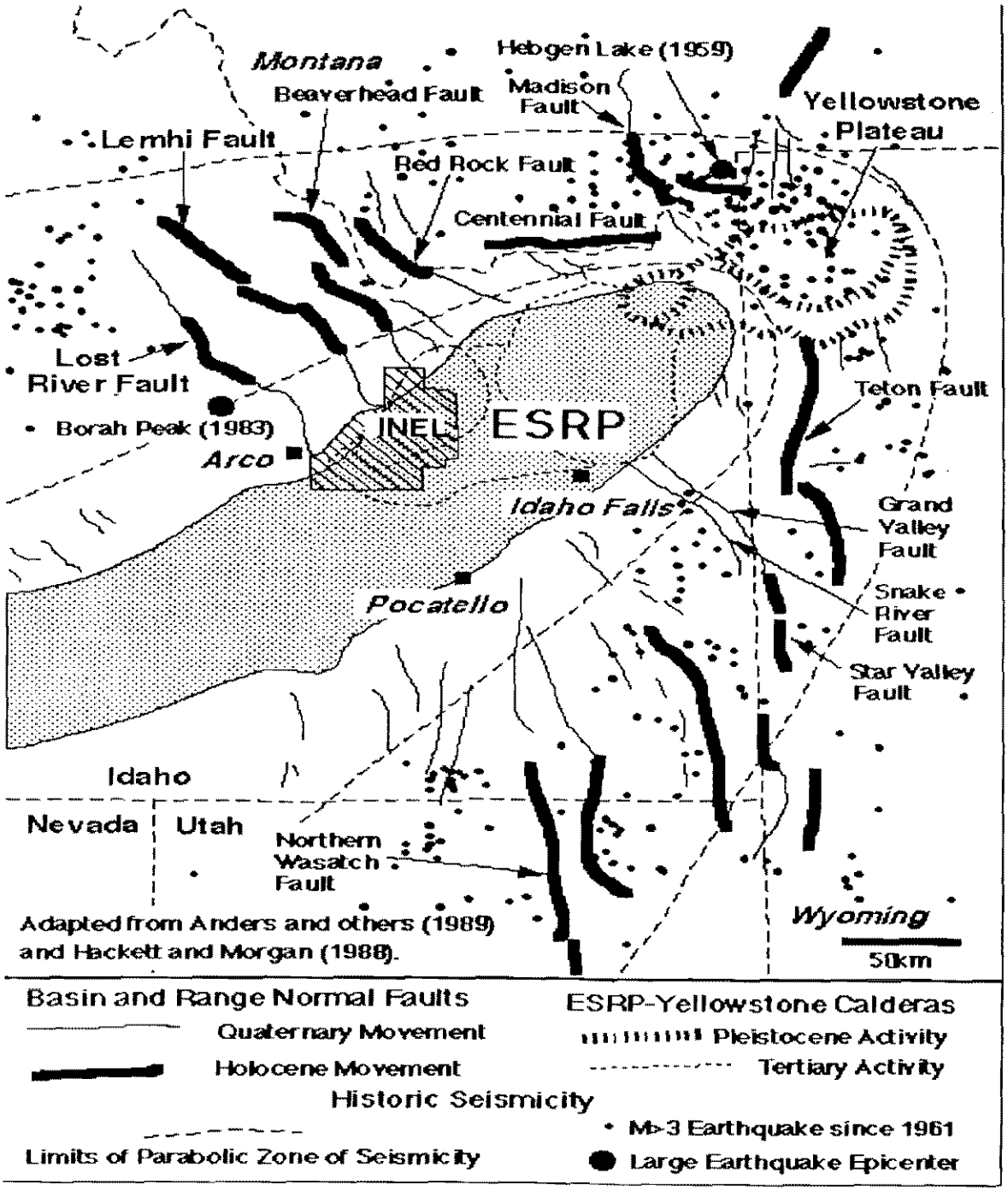


Figure 2A.2-2. Map summarizing the major physiographic, geologic, and seismotectonic elements of the INEEL region. **(For Information Only)**

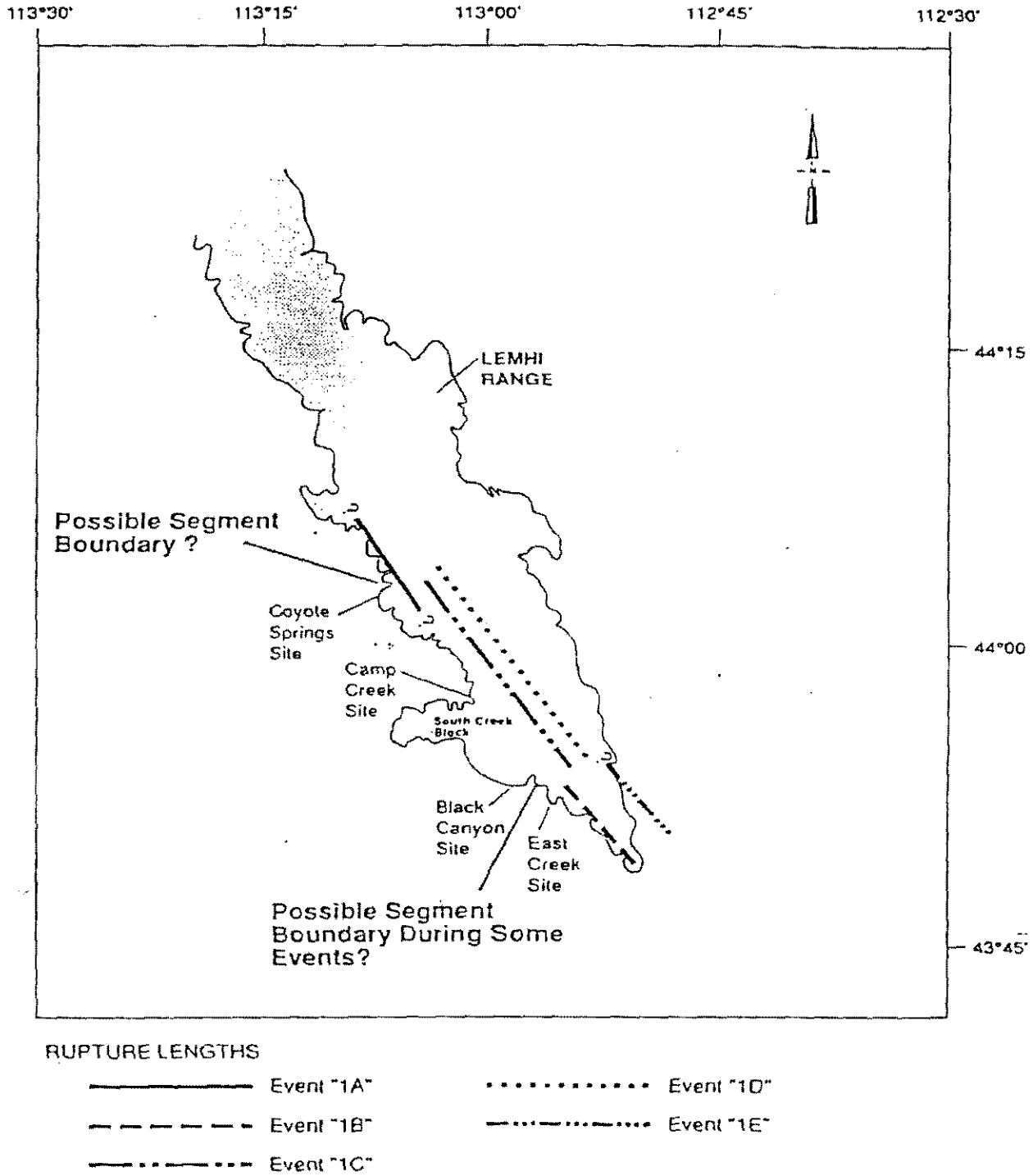


Figure 2A.2-3. Possible rupture scenarios for the Southern Lemhi Fault (WCC 1992b). **(For Information Only)**

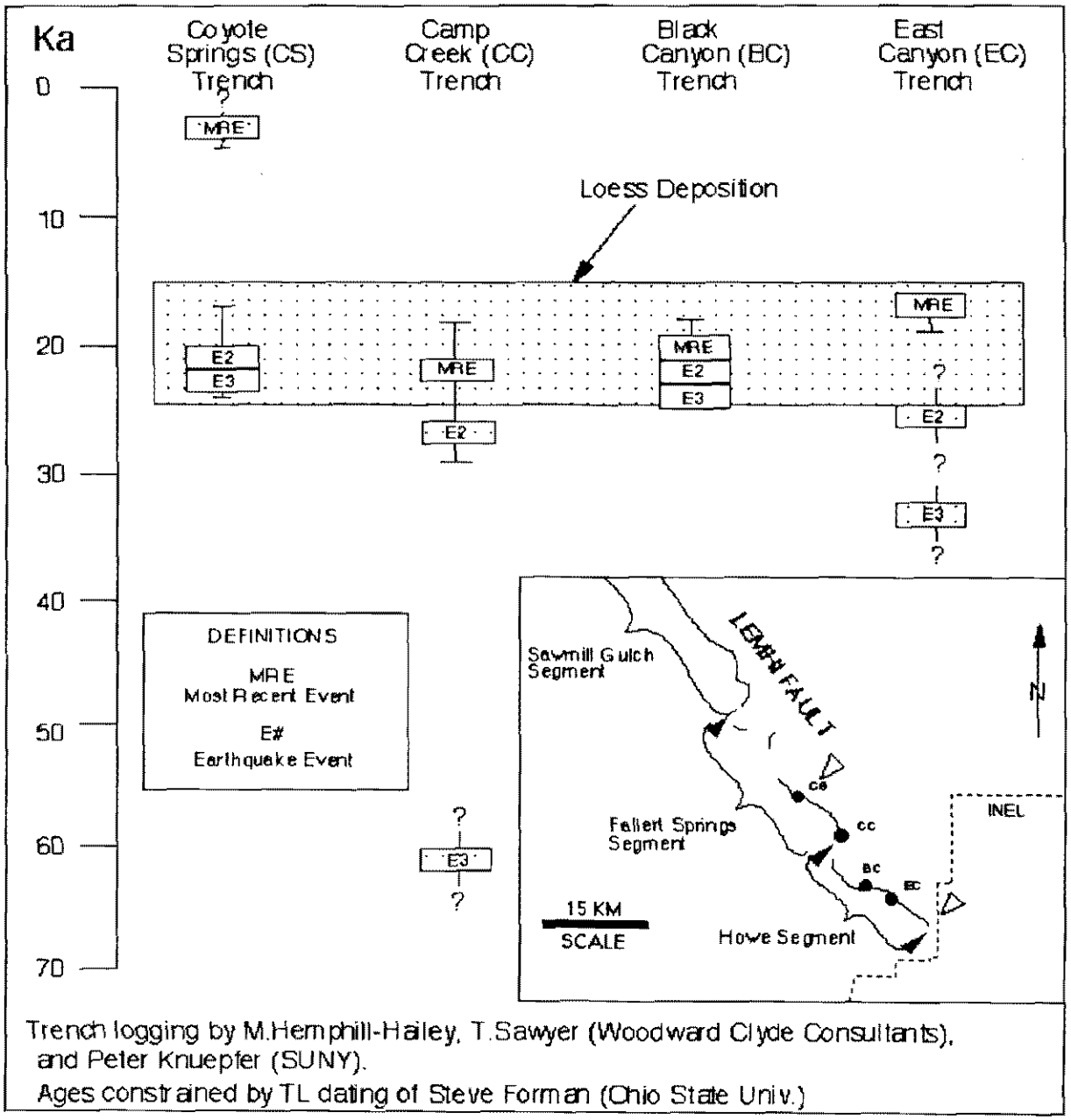


Figure 2A.2-4. Summary diagram of earthquake chronology for the Southern Lemhi Fault. **(For Information Only)**

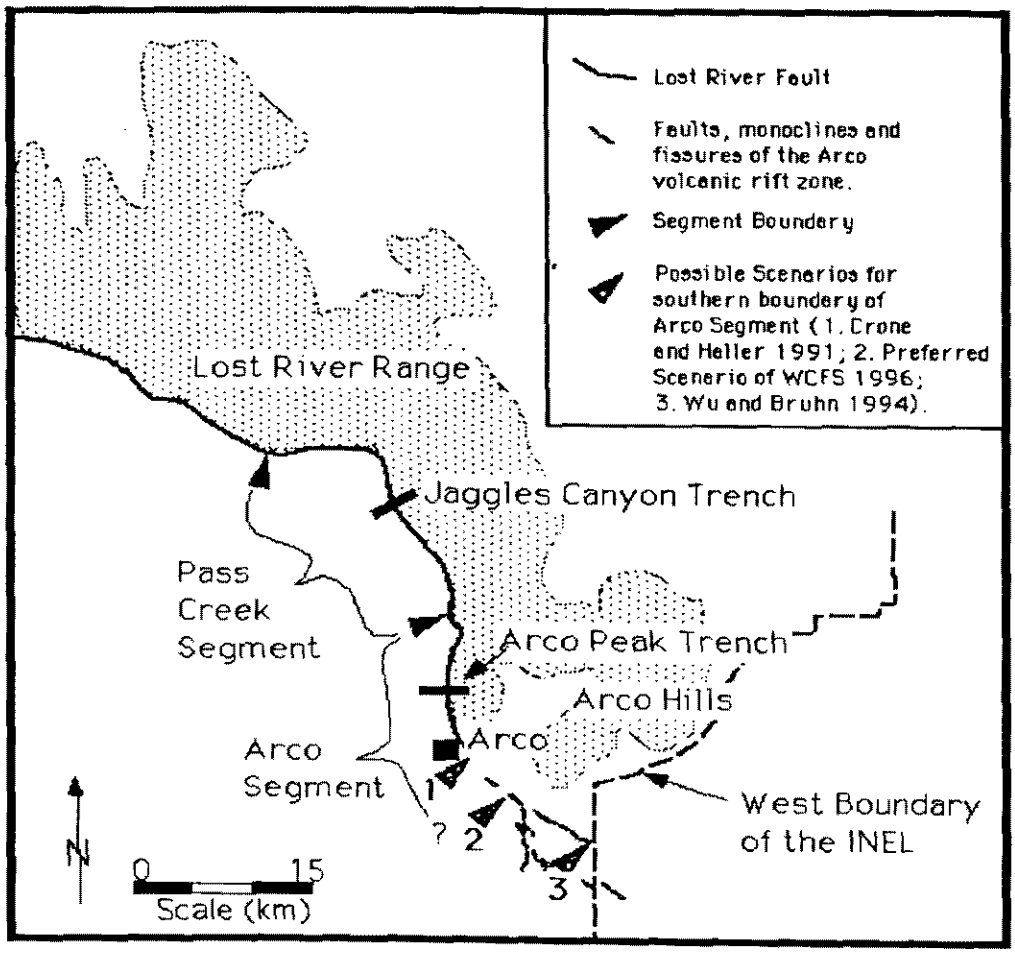


Figure 2A.2-5. Map showing segmentation of the southern Lost River Fault, and possible scenarios for the southern termination of the Arco Segment. Locations of WCFS 1996 trenches are also shown. **(For Information Only)**

STRUCTURE OF THE SOUTHERN LEMHI FAULT

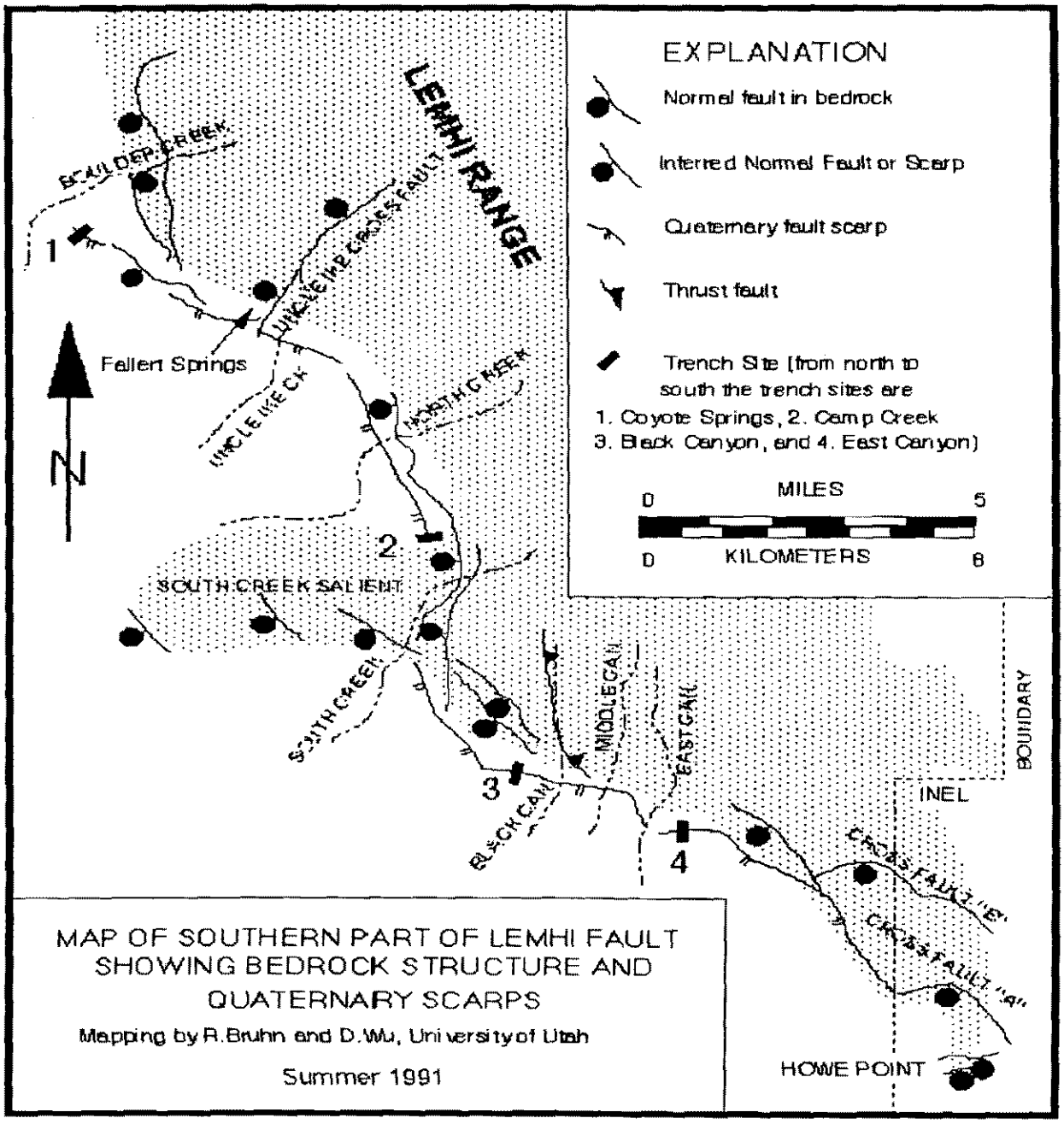


Figure 2A.2-6. Geologic map of the Southern Lemhi fault showing bedrock structure and Quaternary fault scarps (Bruhn, Wu, and Lee 1992). **(For Information Only)**

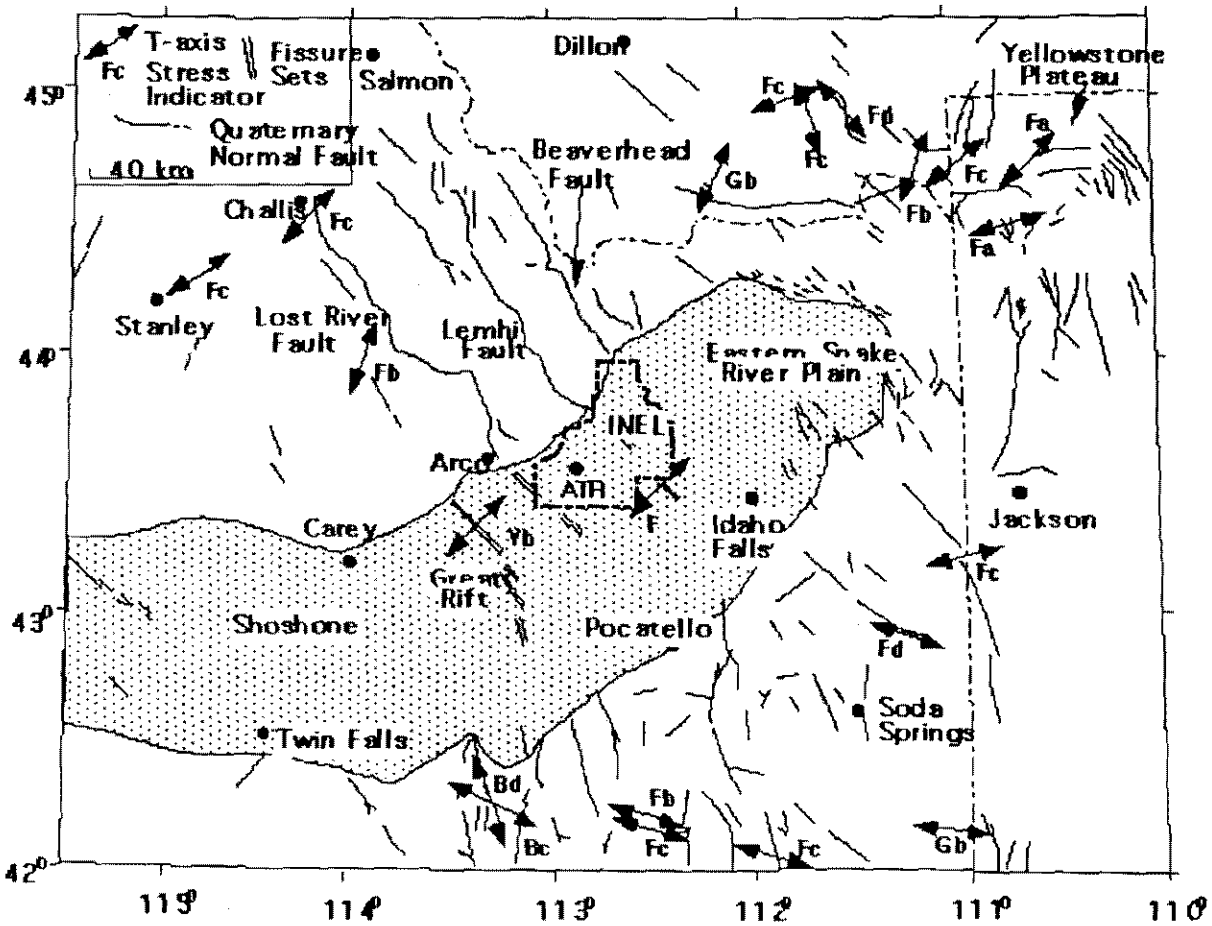


Figure 2A.2-7. Map of Southern Idaho showing the distribution of minimum-principal-stress orientations [as compiled by Zoback and Zoback (1989) and from the 1989 Eastern Snake River Plain composite microearthquake focal mechanisms of Jackson et al. (1993)]. **(For Information Only)**

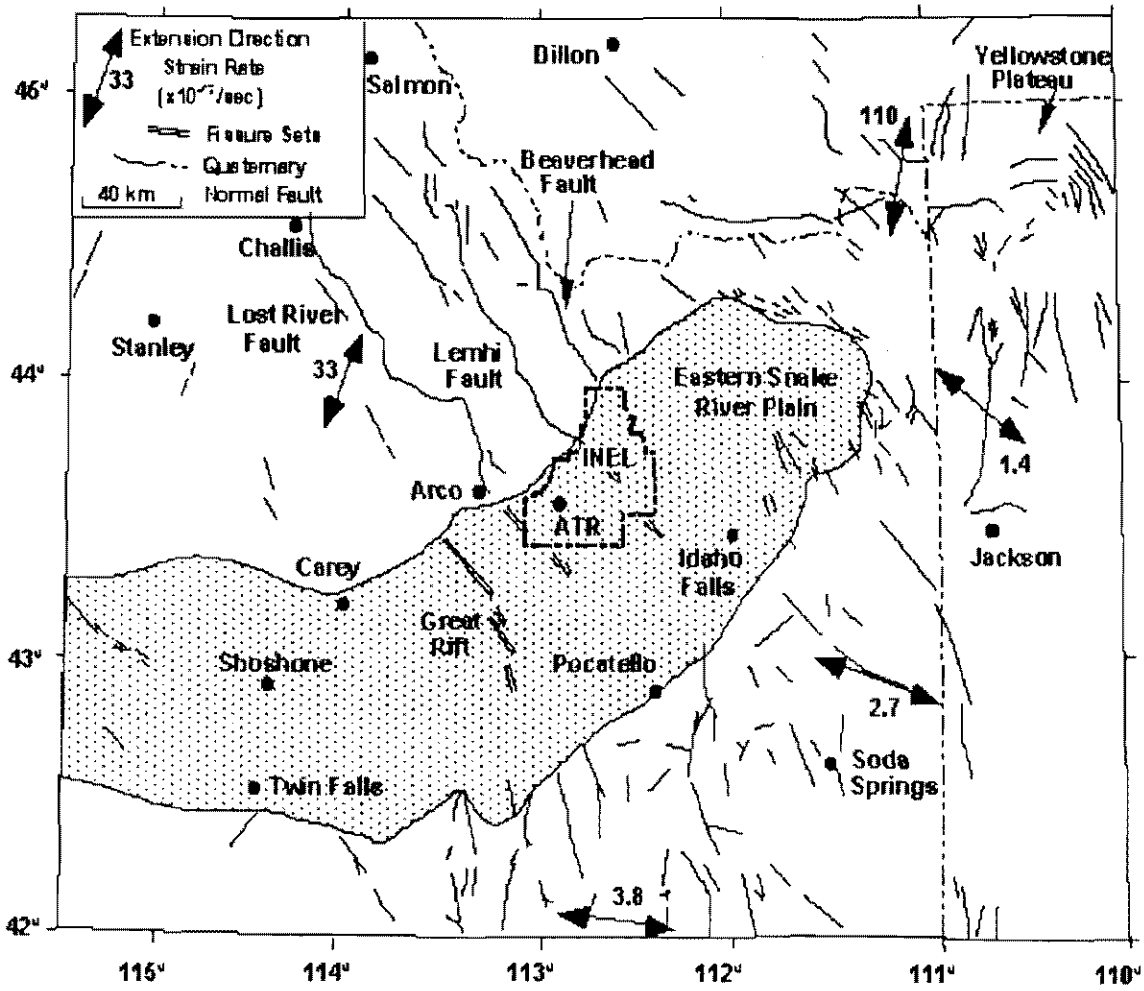


Figure 2A.2-8. Strain-rate map of Southern Idaho (adapted from Eddington, Smith, and Renggli 1987). **(For Information Only)**

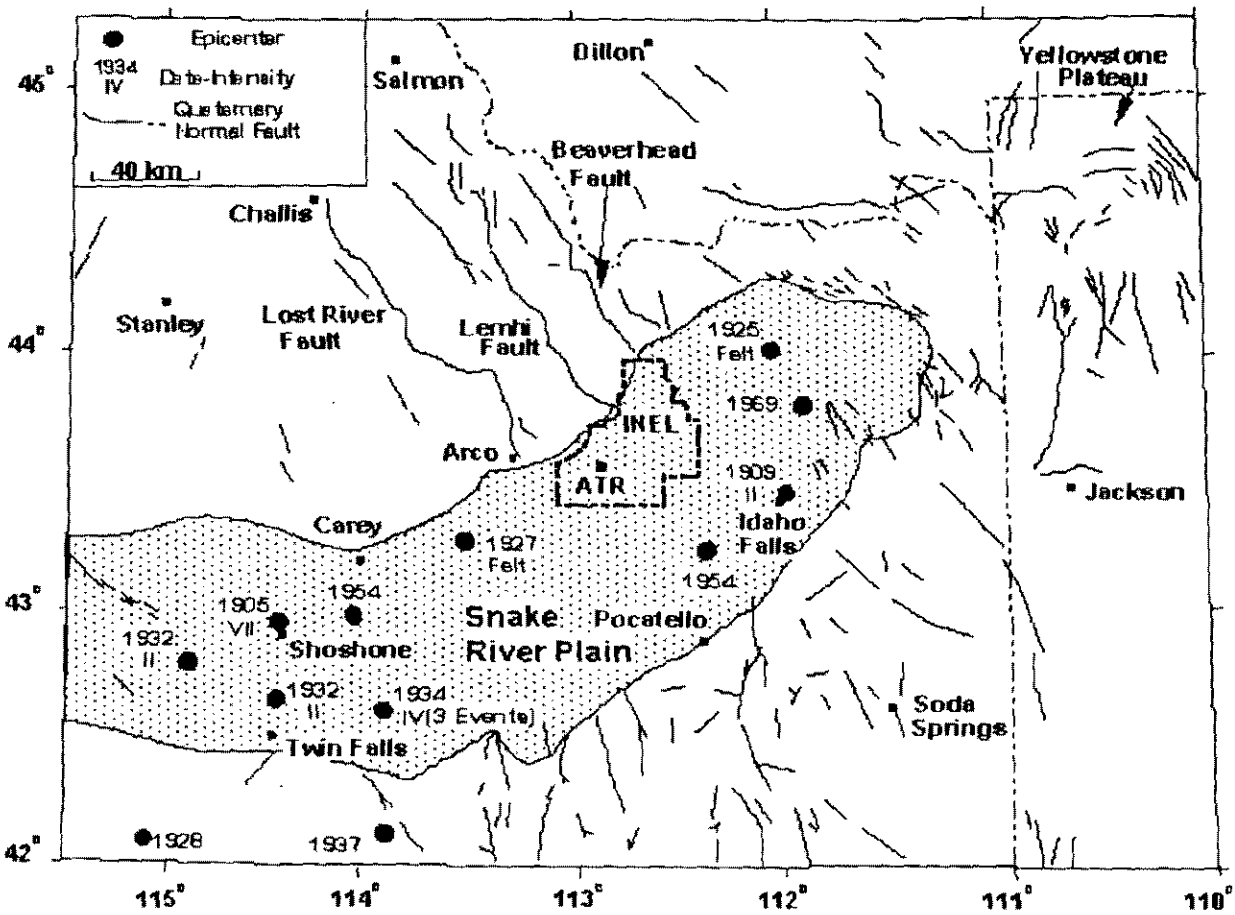


Figure 2A.2-9. Historic earthquakes on the Eastern Snake River Plain (locations from Stover, Reagor, and Algermissen 1986). **(For Information Only)**

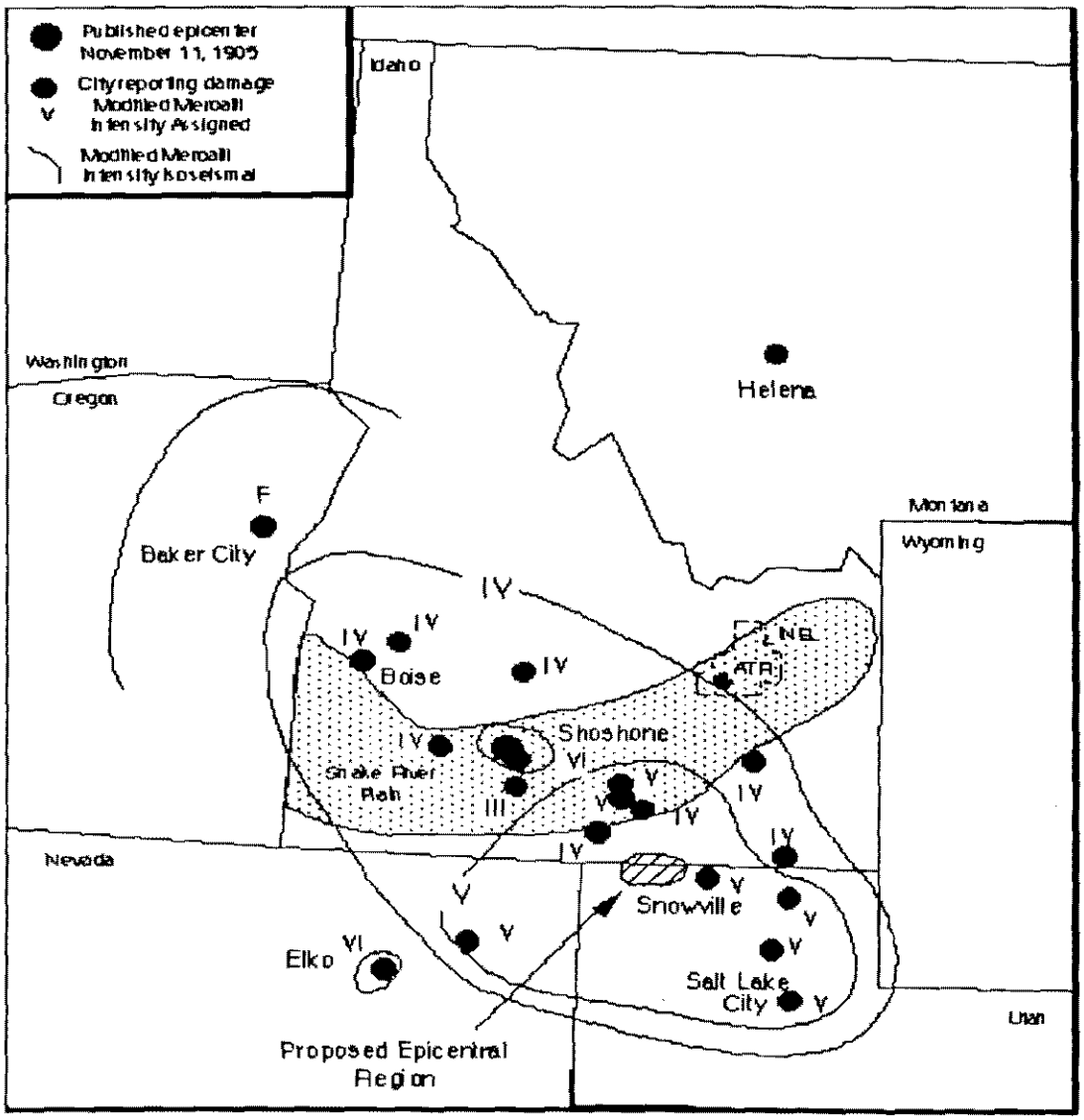


Figure 2A.2-10. Isoseismal-contour map for the 1905 Shoshone, Idaho, Earthquake (modified from Oaks 1992). **(For Information Only)**

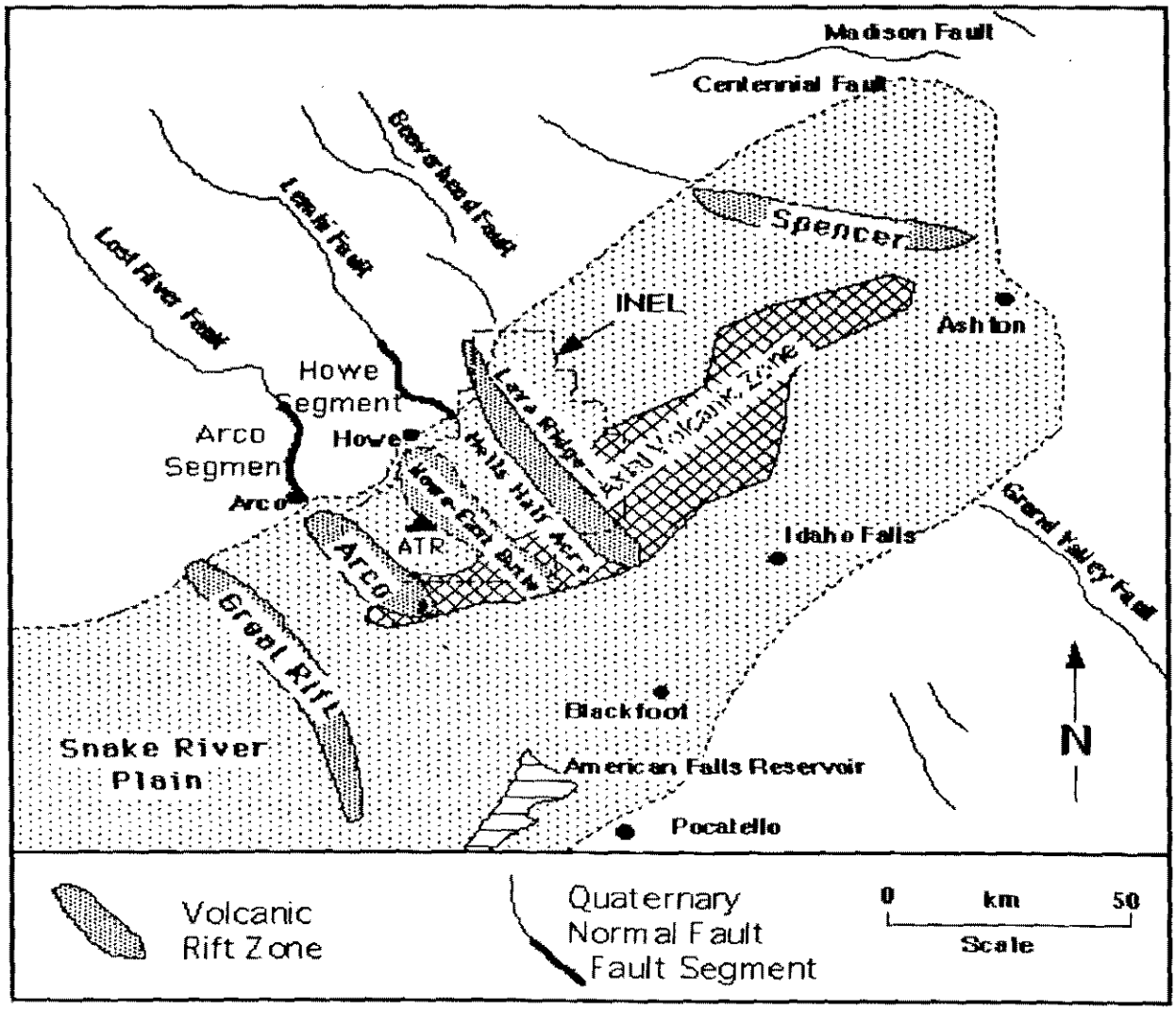


Figure 2A.2-11. Map of the INEEL area showing locations of volcanic-rift zones on the Eastern Snake River Plain and Quaternary normal faults of the surrounding Basin and Range Province (southern segments of faults near INEEL are shown as bold lines). **(For Information Only)**

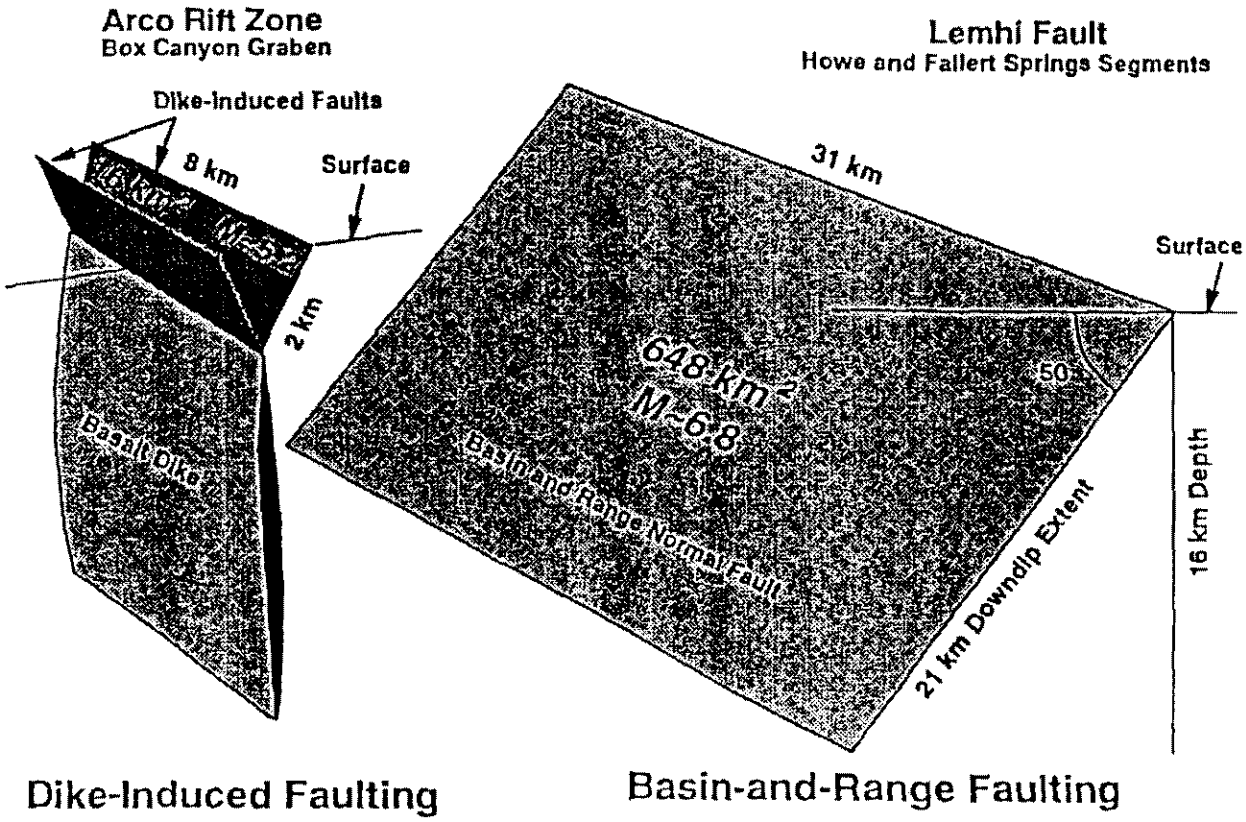


Figure 2A.2-12. Comparison of fault dimension, fault orientation, and calculated moment magnitudes for dike-induced faults on the Eastern Snake River Plain and normal faults of the surrounding Basin and Range tectonic province. **(For Information Only)**

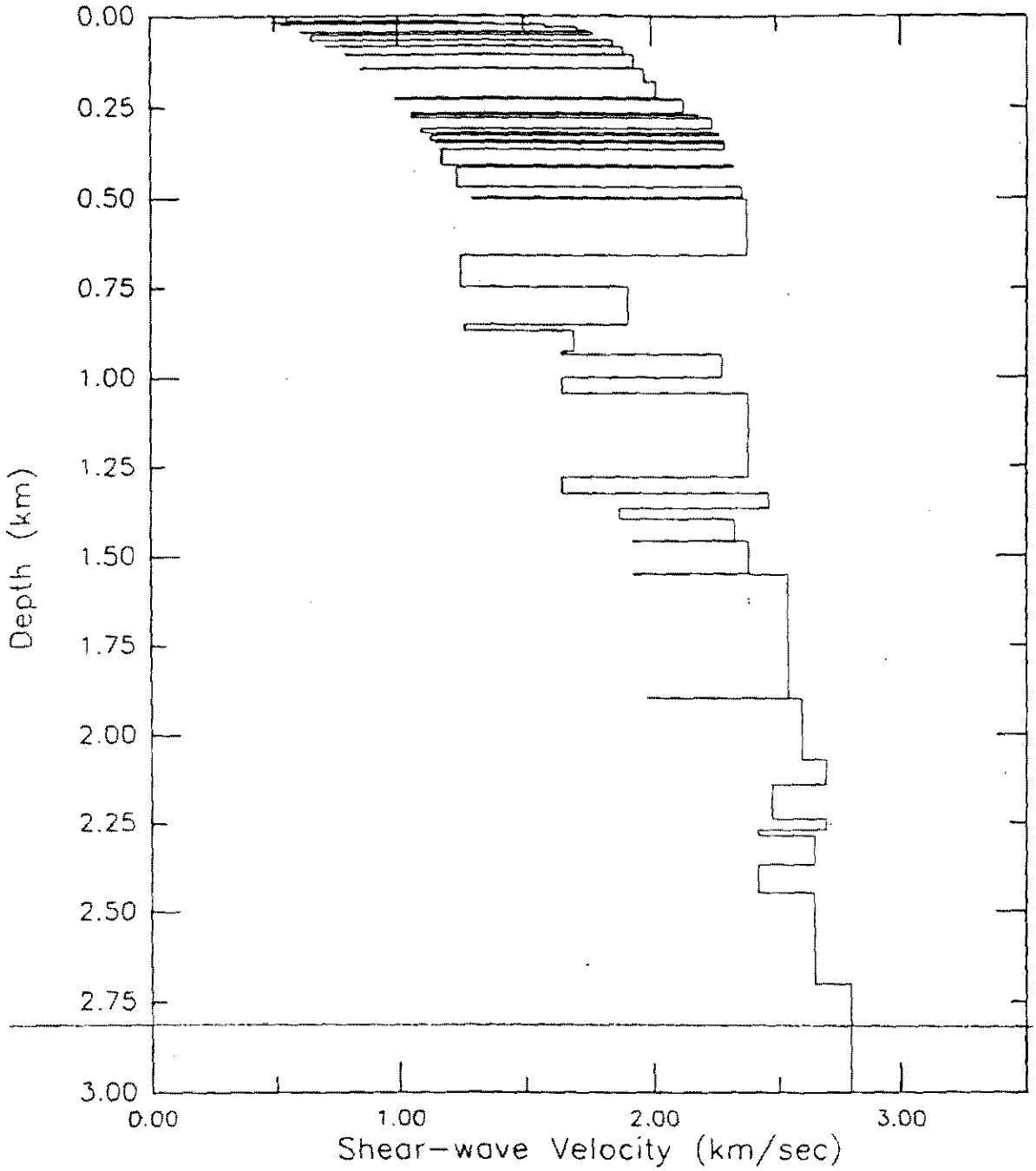


Figure 2A.2-13. S-Wave velocity model for ATR site (WCFS 1996). **(For Information Only)**

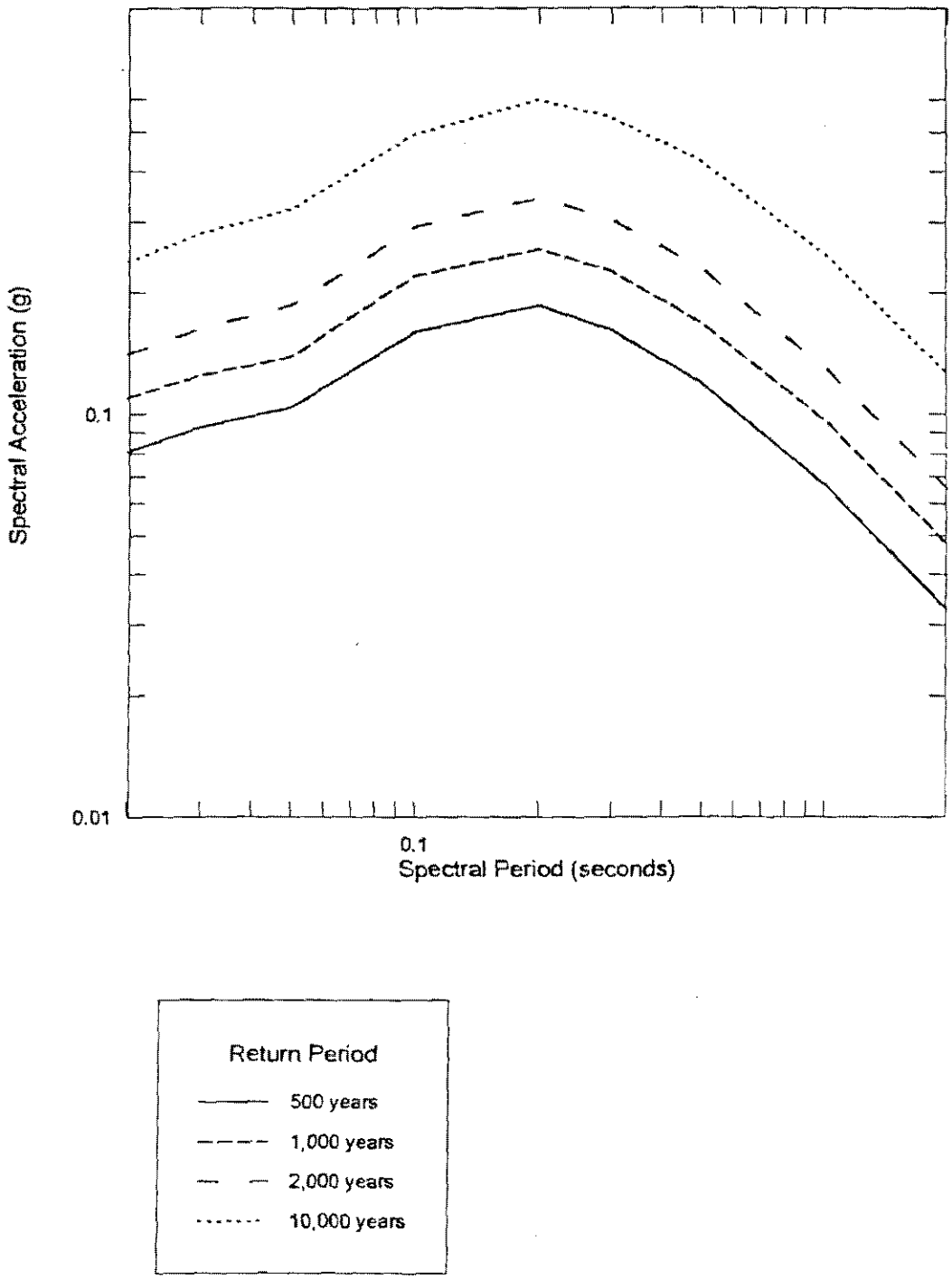


Figure 2A.2-14. ATR response spectra at 500 year, 1000 year, 2000 year, and 10,000 year return periods (WCFS 1996). **(For Information Only)**

2A.3 Surface Faulting

Surface faulting, defined as the rupture of the earth's surface from tectonic or magmatic activity, is of concern in some areas of INEEL, but not at the ATR site. The only place on the INEEL that could be affected by surface faulting related to tectonic activity is near the southern tip of the Lemhi fault (Figures 2A.2-2 and 2A.2-5). It is conceivable that surface faulting associated with an earthquake on the Howe and Fallert Springs segments could extend into the INEEL for a distance of several kilometers in the area just east of the Big Lost River Sinks. The age of most recent earthquake activity on the southern Lemhi fault is given in Section 2A.2.4.1.

Other areas in which surface faulting is of concern are in VRZs (see Section 2A.6.2.3.1). Areas in and near the Arco and the LR-HHA VRZs (Figure 2A.1-8) have the greatest potential for such dike-induced surface faulting (Figure 2A.2-12). (See Section 2A.6.2.3.1for a description of the surface deformation process.) Also, the fissures north of the Naval Reactor Facility (NRF) appear to be dike-induced fissures. The character of possible subsurface faults is discussed in Section 2A.1.2.3. The potential recurrence of such fissuring is tied closely to periods of volcanic activity in VRZs and is quantified in Section 2A.6.

2A.3.1 Geologic Conditions of the Site

See Section 2A.1.2.

2A.3.2 Evidence of Site Fault Offset

Evidence for fault offset at or near the surface does not exist in the immediate vicinity of ATR. Several lineaments are visible on aerial photographs and Landsat images. These lineaments are mostly northeast-trending alignments of contrasting density and distribution of vegetation that most likely originated from eolian modifications of old range-fire scars (see Section 2A.2.2) (Morin-Jansen 1987). The character of possible subsurface faults is discussed in Section 2A.1.2.3.

2A.3.3 Earthquakes Associated With Capable Faults

Capable faults have not been identified in the ATR area, and significant earthquakes have not been recorded or reported in the area. Several microearthquakes have been recorded in the INEEL area since 1972, but they were not felt and they do not define or correlate with known faults (see Section 2A.2.1).

2A.3.4 Investigation of Capable Faults

See Section 2A.3.6.

2A.3.5 Correlation of Epicenters With Capable Faults

The only earthquake epicenters in the ATR area are microearthquakes (see Section 2A.2.1); they are not correlated with, nor do they define, capable faults. No capable faults have been found in the ATR area.

2A.3.6 Description of Capable Faults

Capable faults do not exist within 5 mi (8 km) of the ATR facility. The nearest capable fault is the Howe segment of the Lemhi fault at 11.8 mi (17 km) from the ATR. The seismic evaluations are in Section 2A.2.6.

2A.4 Stability Of Subsurface Materials And Foundations

2A.4.1 Geologic Features

2A.4.1.1 Surface or Subsurface Subsidence. The nature of geologic materials at ATR and the processes of their formation could create several potential conditions that would contribute to subsidence at ATR. As summarized below, it is demonstrated that none of these potential conditions exist at ATR.

2A.4.1.2 Lava Tubes—Lava tubes are linear open cavities that allowed lava to flow from its source vent. Their observed dimensions in basalts of the ESRP range up to several tens of miles (kilometers) in length and 33 ft (10 m) in diameter. Lava tubes are not recognized in the lava flows at the surface west and north of ATR, and the dense pattern of drill holes in the Test Area North area has revealed none in the subsurface. Therefore, no potential for subsidence because of lava tubes exists at ATR.

2A.4.1.3 Interflow Rubble Zones. In some areas of the ESRP and the INEEL, interflow rubble zones with large void volumes have been observed in outcrops and in drill holes. However, none have been revealed in the drilling that has been done in the ATR area.

2A.4.1.3.1 Fine-Grained Sediments—Water saturation, especially when combined with ground shaking or excessive weight of structures, often causes subsidence. Because the surficial sediments at ATR are mostly dry gravels and gravelly sands, and because the reactor is founded on bedrock, fine surficial sediments do not pose a risk of subsidence. The first sedimentary interbed, at a depth of about 150 ft (45 m) is silty, and saturated by a perched water body beneath the TRA warm waste pond. However, its great depth below bedrock and Hull's (1989) interpretation that the interbed is really a silt-infilled fracture zone indicate that it does not have serious potential for subsidence.

2A.4.1.4 Previous Loading History. Rocks at the surface of the ESRP do not have a previous loading history. The slow subsidence of the ESRP basin during the past 4 million years has resulted in the continuous accumulation of the basalts and sediments of the Snake River Group. Rocks and sediments at the surface have never been subjected to lithostatic or tectonic loading.

2A.4.1.5 Rock Jointing and Weathering Patterns, Weak Materials. This discussion of rock jointing and other zones of discontinuity in the rocks beneath ATR focuses on two types of discontinuity. The first is discontinuity between lava flows, a result of the emplacement process of the lava flows. The zones between lava flows typically are characterized by a layer of rubble or breccia (Figure 2A.1-11), which is composed of blocks of basalt that broke from the advancing front of each lava flow and formed a layer of broken blocks over which the flow advanced. These interflow rubble zones range up to 3 ft (1 m) thick and commonly possess a great amount of void space between blocks. That void space can remain open after burial and contribute to groundwater flow in the aquifer, or it can become infilled with silty sediments and become a barrier to water flow. The latter seems to be the case for the perched water body layer at about 150 ft (45 m) below ATR (Hull 1989). In addition to basalt rubble zones forming at the bottom of the lava flows, it is common for fissures to develop in the upper part of lava flows during emplacement. The fissures form because sections of solidified crust (sometimes several meters thick) bend and tilt during flow of still-molten underlying lava. Such fissures can be up to 6.6 ft (2 m) wide and 10 to 16.5 ft (3 to 5 m) deep. They form complex, irregular patterns on the lava flow surface and often are crudely parallel to the edge of the flow. Sometimes they are filled by surficial sediments before burial by younger lava flows.

The second type of structural discontinuity in lava flows is related to cooling and contraction of the lava flow after solidification. This process produces columnar jointing in the interior zone of the lava flow, with columns being polygonal in cross section and perpendicular to the lava-flow surfaces (Figure 2A.1-11). The cooling process also causes development of platy joints parallel to and near the upper and lower surfaces of the lava flow. These two sets of joints cause the basalt to break into columnar blocks and irregular plates when it is weathered and eroded, or when it is broken by excavation or mining processes.

Weathering of basalts beneath ATR is minor to nonexistent because the lava flows are not exposed to surface weathering processes for sufficiently long periods of time before they are buried by younger flows and sediments. As basalts are buried to greater and greater depths, they become altered by hydrothermal waters, but that does not happen until depths of over 1,320 ft (400 m) are reached. Therefore, altered basalts are not present near the surface beneath ATR.

Fine-grained sedimentary interbeds between lava flows can cause structural weakness in some areas, but at ATR the first interbed occurs at a depth of about 150 ft (45 m), and would not affect foundation integrity. Surficial sediments, being composed of gravels and coarse sands, are not prone to structural weakness; however, because ATR is founded on the basalt bedrock, the composition of surficial sediments is irrelevant.

2A.4.1.6 Unrelieved Residual Stresses. Geologic units at and near the surface at ATR and throughout the ESRP have never been buried to greater depths; therefore, they have not acquired residual stresses from great lithostatic loads. The stresses that were generated during cooling and contraction of the basalt lavas were relieved by development of the columnar and platy fracture patterns.

2A.4.1.7 Hazardous Soils. Two areas of hazardous soils exist near ATR, but neither area has the potential to affect the foundation integrity of the plant. The first is the discontinuous layer of silt and clay beneath the surficial gravels and above the basalt bedrock (Hull 1989). Infiltration of water from the TRA warm waste pond has saturated this layer in the area directly beneath the pond, but the zone of saturation does not extend beneath the area inside the TRA fence. The second area of hazardous soil is the low-permeability layer at a depth of 150 ft (45 m) beneath the site. Even though the perched water body above this layer extends beneath the TRA facilities, it is at such great depth beneath the basalt bedrock surface that it has little significance to foundation integrity.

2A.4.2 Earthquake Design Bases

Section 2A.2.6 presents the SSE for ATR. An Operating-Basis Earthquake (OBE) has not been determined for ATR.

2A.5 Stability Of Slopes

Slopes in the ATR area are very small (a few feet per mile at most) and do not pose a threat for instability or landsliding.

2A.6 Volcanism

2A.6.1 Introduction

The regional-tectonic framework of ESRP volcanism has previously been introduced in Section 2A.1. Basaltic and rhyolitic volcanism has affected the ESRP for about the past 10 Ma, and has continued into geologically recent time. Eruptions have not occurred on the ESRP in historical times, but lava flows issued as recently as 2,100 years ago from the Great Rift, about 15 mi (25 km) southwest of the INEEL. Other Holocene basaltic lava fields near the southern INEEL boundary are nearly as young and range between about 5,000 and 13,000 years in age (Kuntz et al. 1986). Many basaltic and three rhyolitic vents located within the present INEEL boundary erupted between about 200,000 and 1.2 Ma (Kuntz et al. 1990). For these reasons, an assessment of volcanic hazards at ATR based on the record of past volcanism in the region is warranted.

The present section summarizes information on the timing, distribution, and eruptive character of volcanism that could affect ATR. Potential volcanic hazards are grouped into two categories: (a) those related to volcanic sources within the INEEL area and (b) those related to distant, non-ESRP sources. For near-field volcanism, the volcanic history of the ESRP and the INEEL area (Table 2A.1-1, Figures 2A.1-2, 2A.1-3) requires three varieties of volcanism to be evaluated:

- The formation of future silicic calderas and associated eruptions of voluminous ash and pumice, as occurred in the INEEL area between about 6.5 and 4.3 Ma, during passage of the Yellowstone mantle plume (Pierce and Morgan 1992) (Figure 2A.1-4)

- The growth of new silicic lava domes near INEEL, as occurred at Big Southern Butte (0.3 Ma), East Butte (0.6 Ma), and elsewhere along the AVZ near the southern INEEL (Kuntz et al. 1990)

- Phenomena related to Quaternary ESRP basaltic volcanism, largely involving the effusion of lava flows and magma-induced ground fissuring across the INEEL area.

Potential effects from distant volcanic sources include (a) pyroclastic flows or tephra fall from explosive-silicic eruptions of the Yellowstone Plateau, 62 to 124 mi (100 to 200 km) northeast of the INEEL and (b) tephra fall from the Cascade volcanoes and other explosive volcanic centers in the western U.S.

Up-to-date references on general aspects of volcanism and associated volcanic hazards include Wohletz and Heiken (1992), Blong (1984), and Latter (1989). Data and analyses specific to INEEL volcanic hazards have been compiled by the Volcanism Working Group (1990), Hackett and Khericha (1993), and Hackett and Smith (1994). Subsequent, related publications address ESRP regional tectonics (Rodgers, Hackett, and Ore 1990; Malde 1991; Parsons and Thompson 1991; Pierce and Morgan 1992; Hackett and Smith 1992) and ESRP volcanism (Kuntz 1992; Kuntz, Covington, and Schorr 1992; Morgan 1992). In addition, detailed geologic mapping (Kuntz et al. 1990; Golder 1992) has led to improved knowledge about locations of fissures and basaltic vents in the INEEL area.

2A.6.2 Potential Volcanic Hazards of the INEEL Area and ATR

Table 2A.6-1 summarizes the important characteristics of volcanism in the INEEL area. The nature and timing of volcanism is reconstructed from interpretation of ESRP volcanic deposits, and from the results of K-Ar dating of volcanic rocks. Also useful in understanding prehistoric INEEL volcanism are observations of historical volcanic phenomena, particularly eruptions along basaltic rift zones in Iceland and Hawaii, and the growth of silicic-lava domes at various volcanic centers along the Pacific Rim.

2A.6.2.1 Formation of ESRP Silicic Calderas and Related Volcanism. Explosive, voluminous eruptions of silicic pumice and ash, and associated caldera collapse occurred on the ESRP during passage of the Yellowstone hotspot between about 6.5 and 4.3 Ma (Malde 1991; Pierce and Morgan 1992; Figure 2A.1-4). Tephra-fall and pyroclastic-flow deposits from these eruptions were dispersed over tens of thousands of square kilometers in southern Idaho and adjoining states, and are known as the Heise Volcanics (Morgan 1992).

The risk of explosive-silicic volcanism and caldera formation in the INEEL area and at ATR is considered negligible for these reasons (Volcanism Working Group 1990):

The mantle plume (Yellowstone hotspot)—considered the apparent energy source of voluminous, caldera-forming, silicic volcanism on the ESRP—has now moved under the Yellowstone Plateau, 62 to 124 mi (100 to 200 km) northeast of the INEEL (Malde 1991; Pierce and Morgan 1992).

Thermal modeling (Blackwell 1989) and geophysical studies of the ESRP crustal structure (Sparlin, Braile, and Smith 1982) show that the silicic magma chambers inferred to have existed in the shallow crust of the ESRP during late Tertiary time are now entirely solidified and are therefore incapable of erupting.

The recurrence intervals (quiescent periods) between major caldera eruptions on the ESRP and the Yellowstone Plateau were 0.5 to 1.7 Ma long. Therefore, 2 1/2 to 8 recurrence intervals (4.3 Ma since latest such ESRP eruption) have elapsed in the INEEL area, suggesting that caldera-related silicic volcanism has ceased.

The time-transgressive pattern of ESRP-Yellowstone silicic volcanism suggests that explosive silicic volcanism expires after basaltic lava flows have filled the calderas. On the ESRP, the late-Tertiary silicic calderas are buried by up to several km of late-Tertiary-to-Quaternary basalt and sediment.

Geophysical, geothermal, and geodetic anomalies indicating the presence of large shallow magma chambers occur at such areas as Yellowstone National Park and Long Valley, California, where explosive silicic volcanism is a potential hazard. The anomalies include extremely high heat flow (Morgan et al. 1977), low seismic velocities in the upper and middle crust (Smith and Braile 1994; Smith and Christiansen 1980), abundant hot spring and geyser activity (Christiansen 1989; White et al. 1975), persistent swarms of seismic activity (Smith and Braile 1994; Rundle and Hill 1988), rapid rise and fall (meter-scale inflation and deflation within months to years) of land surface elevations (Smith and Braile 1994; Yamashita et al. 1992). Absence of these phenomena in the ESRP and near INEEL implies absence of large, shallow, silicic magma chambers and absence of the hazards associated with such magma chambers.

2A.6.2.2 Growth of Rhyolitic Domes, Intrusions, and Related Phenomena

Volcanic domes are steep-sided mounds of lava, commonly of silicic (rhyolitic) composition, from which the magma is too viscous to flow more than a few miles (kilometers) from the vent. The growth of domes is predominately an effusive process, and blocks of the surrounding terrain can be uplifted and tilted as the viscous magma approaches the surface (Spear and King 1982). Growing domes are steep-sided, rubbly, and gravitatively unstable; therefore, they are prone to slope failure. In addition, dome lavas commonly contain sufficient dissolved gas to generate small explosions. As a result, small-volume tephra-fall deposits and blocky pyroclastic flows are frequently associated with dome growth (Fink 1990; Wohletz and Heiken 1992).

Several small [< 1.7 mi^3 (< 7 km^3)] rhyolite domes were emplaced in the INEEL area during the past 1.2 Ma, along the AVZ (Table 2A.2-2 and Figure 2A.1-8): (a) Big Southern Butte (0.3 Ma) (Spear and King 1982; Fishel 1993), (b) Cedar Butte (0.4 Ma) (Hayden 1992), (c) East Butte (0.6 Ma), (d) Middle Butte (inferred as uplifted by a shallow silicic intrusion; uplifted basalt dated at 1.1 Ma); and (e) an unnamed butte (1.2 Ma). Based on emplacement of the five domes cited above within a 1 million year period (1.2 to 0.3 Ma), the estimated recurrence interval for ESRP silicic-dome effusion in the INEEL area is 200 Ka (5×10^{-6}/yr).

The Quaternary rhyolitic domes postdate the earlier caldera-related silicic volcanism by about 3 million years, and their composition is not similar to the caldera rhyolites (Leeman 1982), suggesting they are volcanologically distinct phenomena. Although tephra falls and small-volume pyroclastic flows are commonly associated with silicic-dome growth, these deposits have not been identified in the INEEL area, probably because of coverage by younger basaltic lava and sediment. Several centimeters of tephra could accumulate 6.2 mi (10 km) or more downwind of growing volcanic domes. Given the flat terrain of the ESRP, the major effects of dome effusion, intrusion and uplift, pyroclastic volcanism, and corrosive gases would likely be restricted to about 3.1 mi (5 km) from a growing volcanic dome. Any fumes and tephra associated with dome growth along the AVZ would probably be carried northeastward along the southern INEEL boundary and eventually off-Site, by prevailing southwesterly winds.

Based on the apparent 200-Ka recurrence interval (5×10^{-6}/yr) and the likely restriction of hazardous phenomena to near-vent areas, the probability of a silicic dome affecting the central and northern INEEL (including ATR) is judged to be very small (< 10^{-6}/yr). The most likely area of future silicic-dome emplacement is along the AVZ; hence, the probabilistic risk of impact on southern-INEEL facilities would be somewhat higher but still < 10^{-6}/yr.

2A.6.2.3 Basaltic Volcanism and Related Phenomena

With the exception of localized and infrequent silicic-dome volcanism, Quaternary volcanism of the INEEL area has been predominately basaltic. Potassium-argon (K-Ar) dating of lava flows (Kuntz et al. 1990) demonstrates that the ages of basaltic vents on the INEEL range from > 1 Ma on the northern INEEL, to about 0.2 Ma on the southern INEEL near the AVZ. Although their vents are not situated on the INEEL, four Holocene basalt lava fields erupted along the AVZ between about 13,000 and 5,000 years ago (Kuntz et al. 1986). In one case, the 13.4 Ka Cerro Grande lava field crossed what is now the southern INEEL boundary. Quaternary basaltic volcanism on the ESRP has largely involved mild, effusive outpourings of fluid lava flows from eruptive fissures and small, low-lying shield volcanoes (Greeley 1982; Kuntz, Covington, and Schorr 1992).

2A.6.2.3.1 Volcanic-Rift Zones—Basaltic vents are not randomly distributed across the INEEL area, but tend to concentrate in northwest-trending, linear belts (Kuntz 1992; Kuntz, Covington,and Shorr 1992; Hackett and Smith 1992, 1994) known as volcanic rift zones (VRZs). These belts are marked by basaltic vents as well as open fissures, monoclines, and small normal faults: structures that were produced during propagation of vertical dikes [0 to 2.4 mi (0 to 4 km deep)] that fed the surface eruptions (Rubin and Pollard 1988; Smith, Hackett, and Rogers 1989, Smith, Hersley, and Jackson 1996; Hackett and Smith 1992, 1994).

ESRP VRZs are inferred to be underlain by basaltic-dike swarms, based on their surface-deformation features (Smith, Hackett, and Rogers 1989, Smith, Hersley, and Jackson 1996) and their equivocal correspondence with positive aeromagnetic and gravity anomalies (Josten, Hackett, and Smith 1993). ESRP VRZs are polygenetic features (i.e., were apparently active through numerous cycles of volcanism). The Great Rift has well-developed volcanic and surface-deformation features that formed during eight cycles of Holocene volcanism (Kuntz et al. 1988) (Figure 2A.1-2). Others such as the Arco VRZ are more diffuse and diachronous, with fissures and vents dispersed across a 5-mi- (8-km)-wide belt, formed by multiple cycles of volcanism during the period 600 to 10 Ka. The LR-HHA VRZ is a strongly diachronous feature; its northern portion is occupied by lavas > 1 Ma in age, and its southern terminus is marked by the 5.2 Ka Hell's Half-Acre lava field and dike-induced fissures (Kuntz 1992; Kuntz et al. 1994). Its central region is poorly developed, and is marked by a single monocline (LaPoint 1977; Golder 1992) that was likely induced by dike intrusion. The Howe-East Butte (H-EB) VRZ (Kuntz 1978) is poorly expressed surficially, and is largely covered by fluvial and lacustrine sediment on the central INEEL. Four to five vents and two isolated fissures are associated with a positive, northwest-trending aeromagnetic anomaly (Josten, Hackett, and Smith 1993).

2A.6.2.3.2 Axial Volcanic Zone—The most voluminous and recent volcanism in the INEEL area occurred during the past 1.2 Ma along the Axial Volcanic Zone (AVZ): a broad, northeast-trending constructional-volcanic highland consisting of coalesced basaltic shield volcanoes, tephra cones, and isolated silicic domes. The AVZ forms a topographic divide along the ESRP axis (Kuntz 1992; Hacket and Smith 1992). It differs from VRZs because northwest-trending fissure swarms that typify VRZs are rare, and its overall topographic orientation is parallel to (rather than perpendicular to) the regional direction of extension. Basaltic-dike-intrusion processes along the AVZ are probably similar to those of VRZs, but increased magma supply along the ESRP axis and the predominance of large shield volcanoes has apparently covered most of the dike-induced surface deformation along the AVZ. Northwest-trending fissures and fissure swarms are present, however, at North and South Robbers flows near Big Southern Butte and at the northern end of the Hells Half Acre lava field (Figure 2A.6-1) (Kuntz 1992; Golder 1992).

2A.6.2.3.3 Volcanic Hazards at ATR—Table 2A.6-2 lists hazards associated with ESRP basaltic volcanism, based on interpretation of the ESRP eruption products, and by analogy with historical observations of rift-zone volcanism in Hawaii and Iceland. The hazards due to volcanism in the INEEL area include lava flows, ground deformation, volcanic gases, and ash fall. Seismicity related to dike intrusion is discussed in Section 2A.2.3.1.3 and 2A.2.4.5. The hazards due to base surge and ash flow are judged to be so low in the basaltic terrain of the INEEL that they are not considered further.

The most significant hazard is inundation or burning of facilities by lava flows (Volcanism Working Group 1990; Hackett and Smith 1994). Such flows vary greatly in volume and may cover a few to 155 mi^2 (400 km^2) or more (Kuntz, Covington, and Schorr 1992). On gentle terrain such as the ESRP, lava flows would generally move down-slope at a few meters per minute. Large lava flows on the ESRP seldom exceed 18.7 mi (30 km) in length, and most are < 7.5 mi (< 12 km) long. Borehole investigations and outcrop studies indicate that most ESRP basaltic lava flows are < 33 ft (< 10 m) thick, and taper to several-meter thickness at flow edges. Therefore, they are unlikely to surmount major topographic or constructed obstacles. The general topography and vent locations of the INEEL area (Figure 2A.6-2) suggest that future lavas will most likely erupt from vents along the AVZ or at the intersections of that zone with the VRZs, from which they could flow toward the central INEEL and ATR.

An INEEL-area lava flow hazard map (Figure 2A.6-3) has been developed, based on statistical analysis of the dimensions of lava flows (Table 2A.6-3) (Hackett and Smith 1994). The map shows that TRA is located at a distance of twice the mean lava flow length from volcanic vents younger than 400,000 years. At this distance, only 14% of the lava flows originating from the closest potential vent location would reach the facility. Hazard maps for ground deformation and gases/ash-fall (Figures 2A.6-4 and 2A.6-5) show that TRA is outside the areas subject to the greatest hazard from these phenomena.

2A.6.2.3.4 Volcanic Recurrence and Probabilistic Risk for ATR—Although volcanic hazards can be qualitatively identified on the basis of the geologic record, risk is more difficult to assess because risk is usually expressed quantitatively and in probabilistic terms. Volcanic hazards may exist, but there can be no risk unless life or property are threatened. Risk can be thought of as the product or interplay of (some hazardous phenomenon) multiplied by (its potential impact on life or property at a given location). Quantitative risk assessment requires not only detailed knowledge of the timing and nature of past volcanism, but also the conditional probability of its impact on human life or property. Conditional probabilities take into account not only volcanic-recurrence intervals, but also nonvolcanic parameters such as distance from vents, local-terrain configuration, prevailing-wind directions, and existing or planned mitigation actions.

Table 2A.6-3 gives estimated volcanic-recurrence intervals for INEEL volcanic zones and borehole sites, estimated by the summation of individual vents and fissures in the respective volcanic zones and dividing the sum by the total time period of volcanism within each zone. This approach gives minimum-recurrence estimates and is very conservative because it is assumed that every vent or fissure (sometimes a set of fissures, when they could be confidently grouped as cogenetic) represents a single eruptive episode. It is more likely that each eruptive episode involved eruptions from several vents and the opening of multiple fissures, based on the record of Holocene volcanism and on analysis of ESRP magma generation, rise, and storage (Kuntz 1992; Hackett and Smith 1992).

Table 2A.6-3 suggest that the shortest recurrence intervals (8,000 to 15,000 years), the most recent volcanism (Holocene lava fields), and therefore the most probable areas of future basaltic volcanism and ground deformation, are the AVZ and the Arco VRZ. Within this context, the Volcanism Working Group (1990) estimated the conditional probability of basaltic volcanism affecting a south-central INEEL site as being < 10^{-5}/yr.

A probabilistic event tree analysis was used to estimate the frequency for a basaltic volcanism event to adversely affect the ATR, i. e., for basaltic lava to flow over the ATR site before reactor fuel is removed from the site (Hackett and Khericha, 1993). The event tree analysis considers direction of lava flow, the distance of ATR from the most probable vents and the probable length of the lava flows, and the probability for successful intervention or fuel removal. A mean frequency for an adverse effect at ATR for a basaltic volcanism event near or on the INEEL was estimated to be 8.4×10^{-8}/yr for a mean event recurrence interval of 3×10^{-5}/yr.

2A.6.3 Potential Volcanic Hazards from Distant Sources

The locations and general characteristics of potentially hazardous volcanoes in the western U.S. are summarized by Bailey et al. (1983) and Wright and Pierson (1992). Potential effects upon the INEEL from eruptions of those volcanoes are addressed by the Volcanism Working Group (1990). The selective analysis given below supports the general conclusion that significant effects upon the INEEL from distant volcanic eruptions are highly improbable.

2A.6.3.1 Yellowstone Plateau. Geologic and geophysical investigations (Smith and Christiansen 1980) indicate that the mantle plume that left its 15-million-year track across southern Idaho and formed the Snake River Plain now resides beneath the Yellowstone Plateau. The plume's presence there explains the Yellowstone Plateau's crustal structure, high heat flow, geothermal features, and explosive silicic volcanism.

During the past 2.1 Ma, the Yellowstone Plateau volcanic field has produced more than 1,443 mi^3 (6,000 km^3) of silicic tephra in three cycles of explosive, caldera-related volcanism. The tephra was produced largely in the form of tephra-fall and pyroclastic-flow deposits (Christiansen 1984). Ash layers from Yellowstone have been identified in the Quaternary stratigraphic record across much of western North America (Izett and Wilcox 1982). Eruptions of this magnitude are rare in the worldwide geologic record, and the three climactic Yellowstone eruptions occurred at 2.1, 1.3, and 0.6 Ma (Christiansen 1984), for an average recurrence interval of 700,000 years.

Hazards at ATR from potential Yellowstone eruptions include blanketing by pyroclastic flows or volcanic ash. The facility lies about 100 mi (160 km) from the Yellowstone caldera rim and more than 124 mi (200 km) from the Hot Springs Basin area of northeastern Yellowstone, a likely site of future eruptions. For comparison, maximum runout distances of large-volume pyroclastic flows (ignimbrites) from Yellowstone, the ESRP, and elsewhere, traveling on relatively flat terrain, generally range from 62 to 93 mi (100 to 150 km). Hence, the likelihood of pyroclastic flows from even the largest Yellowstone eruptions reaching the INEEL is essentially nonexistent, because of the great distance and intervening topographic barriers.

Although there is no direct relationship between ashfall thickness and damage parameters (Blong 1984), the historical eruptions of Mt. St. Helens demonstrate that about 3 in. (8 cm) of ash can generally be accommodated by the infrastructure of a technologically advanced nation without serious long-term consequences (Warrick et al. 1981). Ash-fall thickness from Yellowstone could exceed 3 in. (8 cm) if there were a large [> 10 mi^3 (> 40 km^3)] eruption, and if wind conditions were to disperse the ash cloud directly over the INEEL. Such conditions are conceivable in light of past Yellowstone volcanism, but they are highly improbable because prevailing winds would not likely direct ash toward INEEL and because the recurrence intervals of such events are extremely long (0.5 to 1 Ma). Less than 2 in. (5 cm) of Yellowstone ash have been found on the ESRP at INEEL-equivalent distances (Izett and Wilcox 1982).

2A.6.3.2 Cascade Volcanoes and Other Western-U.S. Centers. The Cascade volcanoes of Northern California, Oregon, and Washington have produced many Quaternary tephra layers, some of them widely dispersed across the western U.S. (Izett 1981). These centers lie 435 to 497 mi (700 to 800 km) west of the INEEL, at distances and prevailing-wind directions that prevent all but the largest ashfall eruptions from affecting the INEEL area. The Mazama ash is a voluminous [10 mi^3 (ca. 40 km^3)] and widespread ash layer that erupted from what is now Crater Lake, Oregon, and is a product of the largest-known Cascade eruption.

In the INEEL area, the Mazama ash is 0.2 to 0.8 in. (0.5 to 2 cm) thick. Theoretical considerations and field measurements indicate that 2.4 in. (< 6 cm) of Mazama ash would have fallen on the INEEL, if the dispersal axis of the cloud were directly overhead. This result effectively eliminates all Cascade volcanoes as sources of significant [> 3 in. (> 8 cm)] ashfall at the INEEL.

A similar conclusion is reached for other western-U.S. volcanoes, such as the Long Valley caldera, which erupted about 600 Ka and produced the 144-mi^3 (600-km^3) Bishop Tuff. Long Valley is more than 500 mi (800 km) southwest of the INEEL. As in the Yellowstone analysis, significant [> 3 in. (> 8 cm)] ash fall could be expected only for improbable conditions and at extremely long recurrence intervals.

2A.6.4 Conclusions

The most significant hazards and risks to ATR are associated with basaltic volcanism and related phenomena from ESRP vents.

For volcanic areas such as the ESRP, with no historical volcanism and an incomplete chronological record of prehistoric volcanism, assessments of potential volcanic hazards and volcanic risk are based on interpretation of the long-term geologic record and on the documented effects of historical eruptions in analogous regions such as Iceland and Hawaii. Volcanic hazards to the ATR are related to future basaltic and rhyolitic eruptions along VRZs and the AVZ. The most significant volcanic hazard to the INEEL is inundation or burning of facilities by basaltic lava flows from VRZs. A significant related hazard is disruption of facilities resulting from ground deformation accompanying magma intrusion along VRZs: opening of fissures, normal faulting, and broad-region tilting and uplift within several miles (km) of vents (see Section 2A.3 and 2A.3.6). Other, less-significant basaltic hazards include volcanic-gas emission and disruption of groundwater.

Available geologic-map data and geochronometry of INEEL basalt lava flows suggest minimum (most conservative) volcanic-recurrence intervals of 10^{-4} to 10^{-5}/yr, for the AVZ and the Arco and LR-HHA VRZs. Therefore, the probabilistic risk of basalt-lava inundation or intrusion-related ground disturbance is estimated to be < 10^{-5}/yr for sites on the southern INEEL. When the direction of lava flow, the distance of ATR from the most probable vents, and the potential for successful intervention are considered in the risk assessment for basaltic volcanism for the ATR, then the probabilistic risk at ATR is < 10^{-7}/yr. Risk from these phenomena at northern-INEEL sites is still lower because volcanism there has been less frequent and less recent. The probability of significant impact from all other volcanic phenomena, such as growth of new rhyolite domes on the ESRP or tephra falls thicker than 3 in. (8 cm) from non-ESRP vents, is estimated to be < 10^{-5}/yr because of the combined effects of great distance, infrequency, low volume, and topographic or atmospheric barriers to the dispersal of tephra on the INEEL.

Table 2A.6-1. Characteristics of volcanism in the INEEL area. (See Figure 2A.2-6 for map distribution of volcanic zones and related features.)

Geological features	Caldera formation	Rift-zone volcanism	Axial-zone volcanism	Areas between volcanic zones
Magma types	Rhyolite (viscous and gas-rich)	Basalt (fluid and gas-poor)	Basalt and subordinate rhyolite	Basalt (and minor rhyolite[a])
Volcanic style and products	Highly explosive; voluminous pumice and fine ash	Mild and effusive; erupt mainly lava flows from fissures, low shield volcanoes and small tephra cones	As per rift zones, but also local rhyolite domes and intrusions (Big Southern, Middle, East Buttes) with local explosive phenomena	As per VRZs and AVZ
Stratigraphy	Filled with thick (up to several km), welded, silicic ash- flow tuffs, lava flows and sediments (Heise Volcanics)	Piles of 1- to 30-m-thick basalt lava flows and minor interbedded sediment; total lave thickness 0 to > 1 km (Snake River Group)	Basaltic lava flows and dispersed small tephra cones; isolated rhyolite domes and intrusions (Snake River Group)	Fine clastic sediment of fluvial, lacustrine and eolian origins; fewer lava flows than near VRZs (Snake River Group)
Tectonics and physical configuration	Collapse: broad, oval depressions, 10s to 100 km wide and 1-2 km deep, ringed by inward-dipping fractures	Extensional: NW-trending, linear belts of open fissures, monoclines, small normal faults and basaltic vents	External, but magma-induced fissures or faults are rare; a diffuse, NE-trending, volcanic highland along the ESRP axis	Subsidence: broad, low topographic basins between extensional and constructional volcanic highlands; seldom disturbed by magma intrusion
Geologic age in INEEL area	6.5 Ma to 4.3 Ma in INEEL area, now covered by younger basaltic lava. (2 Ma to 0.6 Ma on Yellowstone Plateau)	Surficial INEEL basalts: 1.2 Ma to 0.05 Ma; most are 0.7 Ma to 0.1 Ma. Inception of major ESRP basaltic volcanism began about 4 Ma.	Basalt: > 1 Ma (Middle Butte) to 5.2 Ka (Hell's Half-Acre). Rhyolite: > 1 Ma (near East Butte) to 300 Ka (Big Southern Butte)	As per VRZs
Quaternary eruption frequency	Zero in INEEL area; Quaternary calderas closest to INEEL occur on Yellowstone Plateau	Low; one eruption per 10,000 to 100,000 years (see Table 3.7-7)	Low: one basaltic eruption per 10,000 years (see Table 3.7-7); one rhyolitic intrusion or dome every 200,000 years or longer	Very low; by definition less frequent than along VRZs (one eruption per 100,000 years or more)

a. Rhyolite domes may exist in the subsurface.

Table 2A.6-2. Hazards associated with basaltic volcanism on the Eastern Snake River Plain. Entries are listed from highest to lowest relative hazard.

Phenomenon	Relative frequency	Size or area of influence	Comments
Lava flow	Common	3 km^2 to 400 km^2 area of coverage, based on sizes of ESRP lava fields of the past 15 Ka	Significant hazard; typical basaltic phenomenon; lava from fissures or shield volcanoes may inundate large areas downslope of vents.
Ground deformation: fissuring, faulting and uplift	Common; associated with virtually all shallow magma intrusion, and eruption	Fissuring could affect areas of 2 × 20 km; minor tilting and broad uplift in areas to 5 × 20 km.	Significant hazard; due to shallow dike intrusion; "dry" intrusion may occur without lava flows.
Volcanic earthquakes	Common; associated with magma intrusion before and during eruptions	Maximum M = 5.5 and most events M < 4; ground vibration may affect facilities within 25 km.	Moderate hazard; swarms of shallow earthquakes (< 4 km focal depth) occur as dikes propagate underground.
Gas release	Common; associated with fissuring and lava effusion	May affect few $km.^2$ near or downwind of vents	Low hazard; local plume of corrosive vapor, downwind from eruptive vent or fissure. Cooled gases may collect in topographic depressions.
Tephra fall	Rare	Restricted to near-vent areas; may affect several km^2 downwind	Low hazard; basaltic eruptions are inherently nonexplosive and may form small tephra cones but little fine ash to be carried downwind.
Base surge (ground-hugging blast of steam and tephra)	Rare	Effects limited to radius of several km from vent; < 10 km^2 area	Low hazard; steam explosions due to interaction between ascending magma and shallow groundwater; water table too deep under most of INEEL.
Tephra flow (ground-hugging flow of hot, pyroclastic material)	Extremely rare	Near vent; may affect area < 1 km^2	Very low hazard; as per tephra fall but affecting even smaller areas

Table 2A.6-3. Statistics of dimensions of lava flows less than 400,000 years old on the ESRP (from Hackett and Smith, 1994).

Statistic/Parameter	Length (km)	Area (km^2)
Maximum	24.9	400
Minimum	0.5	0.1
Range	24.4	399.9
Mean	8	47.1
Median	6.6	20.1
Standard Deviation	6.4	70.6
Number	56	55

Table 2A.6-4. Estimated volcanic-recurrence intervals (and corresponding eruption probabilities in parentheses) for volcanic zones and borehole sites in the INEEL area.

Volcanic zone or borehole	Data sources	Time interval of volcanism (years before present)	Number of vents, fissures, or flow groups	Comments	Estimated recurrence interval
Great Rift (25 km southwest of INEEL)	Kuntz et al. 1986, 1988	2,100–15,000 years (radiocarbon dating)	8 Holocene eruptive periods (each lasting a few decades or centuries, and each including multiple flows and cones); > 100 vents	No impact on INEEL; most recently and frequently active of all ESRP rift zones; thus provides minimum-recurrence for entire ESRP;	2,000 years (5×10^{-4}/yr)
AVZ (southern INEEL)	Kuntz et al. 1986, 1990	5,000–730,000 years (K-Ar dating; radiocarbon; paleomagnetic data)	65 mapped vents; 4 Holocene lava fields, 3 of them shared by volcanic rift zones	Could affect much of southern INEEL; most recently and frequently active of all volcanic zones that could affect INEEL	8,000 years (1.2×10^{-4}/yr)
Arco Volcanic Rift Zone (southwestern INEEL)	Kuntz, 1978; Smith, Hackett, and Rogers 1989; Kuntz et al. 1990	10,000–600,000 years (radiocarbon, K-Ar and TL dating; paleomagnetic data)	30–50 vents/vent groups and fissure sets; 2 Holocene lava fields	Volcanism could affect southwestern INEEL.	15,000 yrs (6.7×10^{-5}/yr)
LR-HHA VRZ (northern and eastern INEEL)	Kuntz et al. 1986, 1990	5,000–1,200,000 years (K-Ar dating; radiocarbon; paleomagnetic data)	35 mapped vents, fissures and monoclines; 1 Holocene lava field: Hell's Half-Acre	Could affect northern and eastern INEEL; extremely long eruptive history; includes oldest and youngest basalts in the INEEL area.	50,000 years (2×10^{-5}/yr)
H-EB VRZ (central INEEL)	Kuntz, 1978; Golder 1992	230,000–730,000 years (K-Ar dating; paleomagnetic data)	5–7 mapped vents and fissures	Old, poorly exposed, and sediment-covered; identified in part by subsurface geophysical anomalies.	100,000 years (1×10^{-5}/yr)
Borehole NPR Site E (south-central INEEL)	Champion, Lanphere, and Kuntz 1988	230,000–640,000 years (K-Ar dating; paleomagnetic data)	9 lava-flow groups (each group contains multiple flows, erupted over a short time)	Dates from 600-ft interval of subsurface lavas give recurrence estimate consistent with surficial geology of the area.	45,000 years (2.2×10^{-5}/yr)
Borehole RWMC 77-1 (southwestern INEEL)	Kuntz 1978; Anderson and Lewis 1989	100,000–565,000 years (K-Ar and TL dating; paleomagnetic data)	11 lava-flow groups (each group contains multiple flows, erupted over a short time)	Dates from 600-ft interval subsurface lavas give longer recurrence interval than nearby Arco and Axial zones, reflecting flow-group (sub-surface) vs. vent-counting (surface geology) approaches.	45,000 years (2.2×10^{-5}/yr)

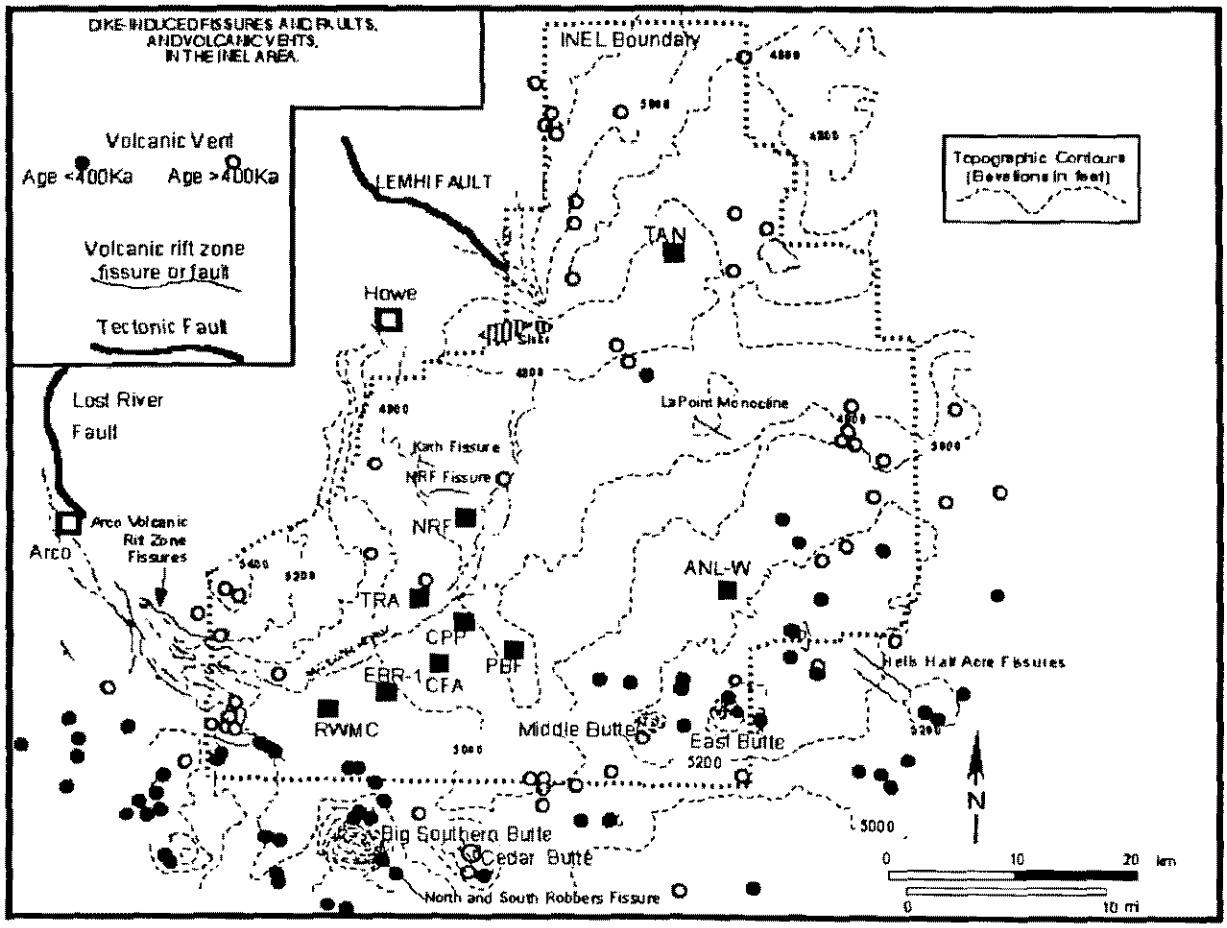


Figure 2A.6-1. Dike Induced Fissures and Faults, Volcanic Vents in the INEEL Area.
(For Information Only)

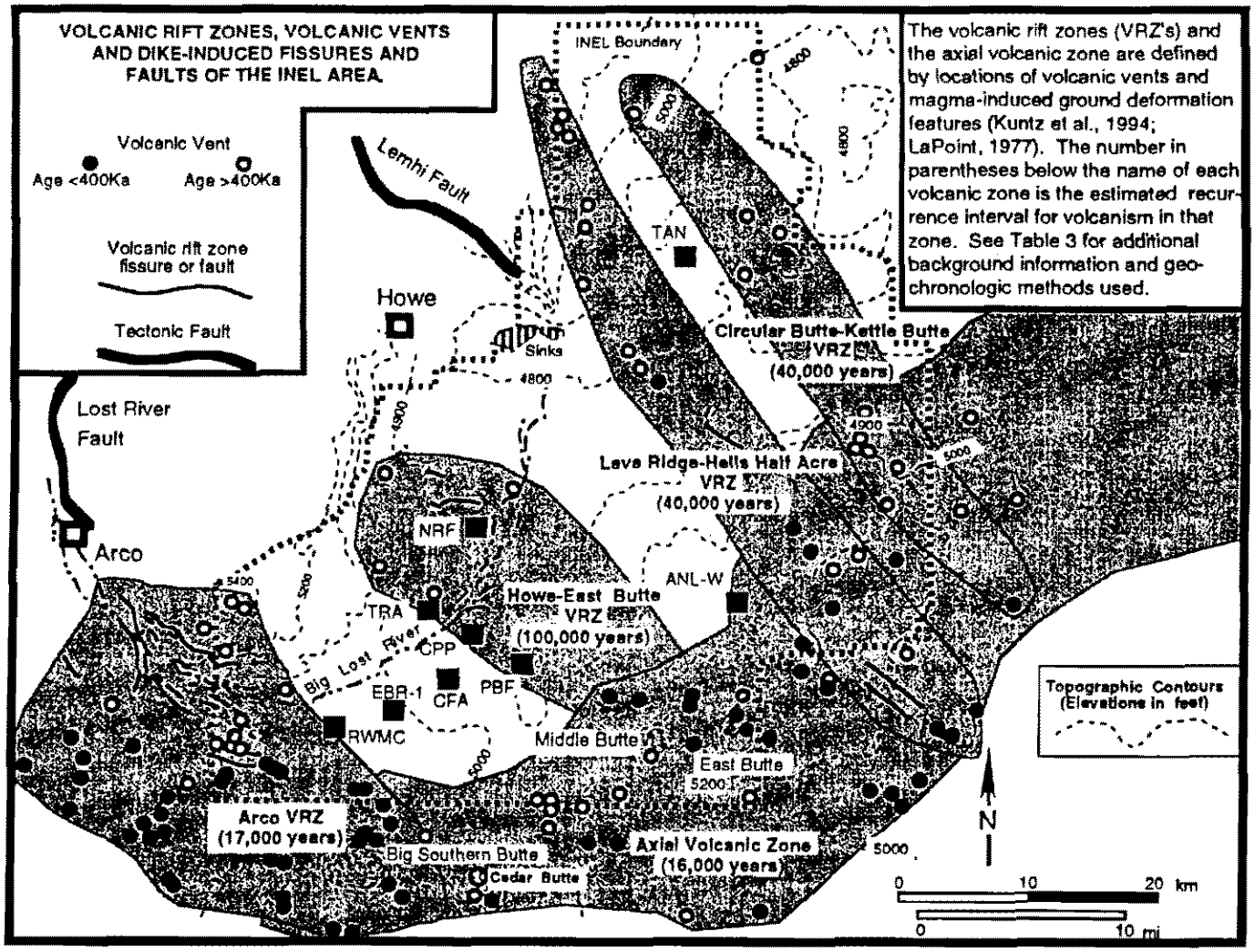


Figure 2A.6-2. Map showing volcanic vents and volcanic recurrence intervals for volcanic zones in the INEEL area. **(For Information Only)**

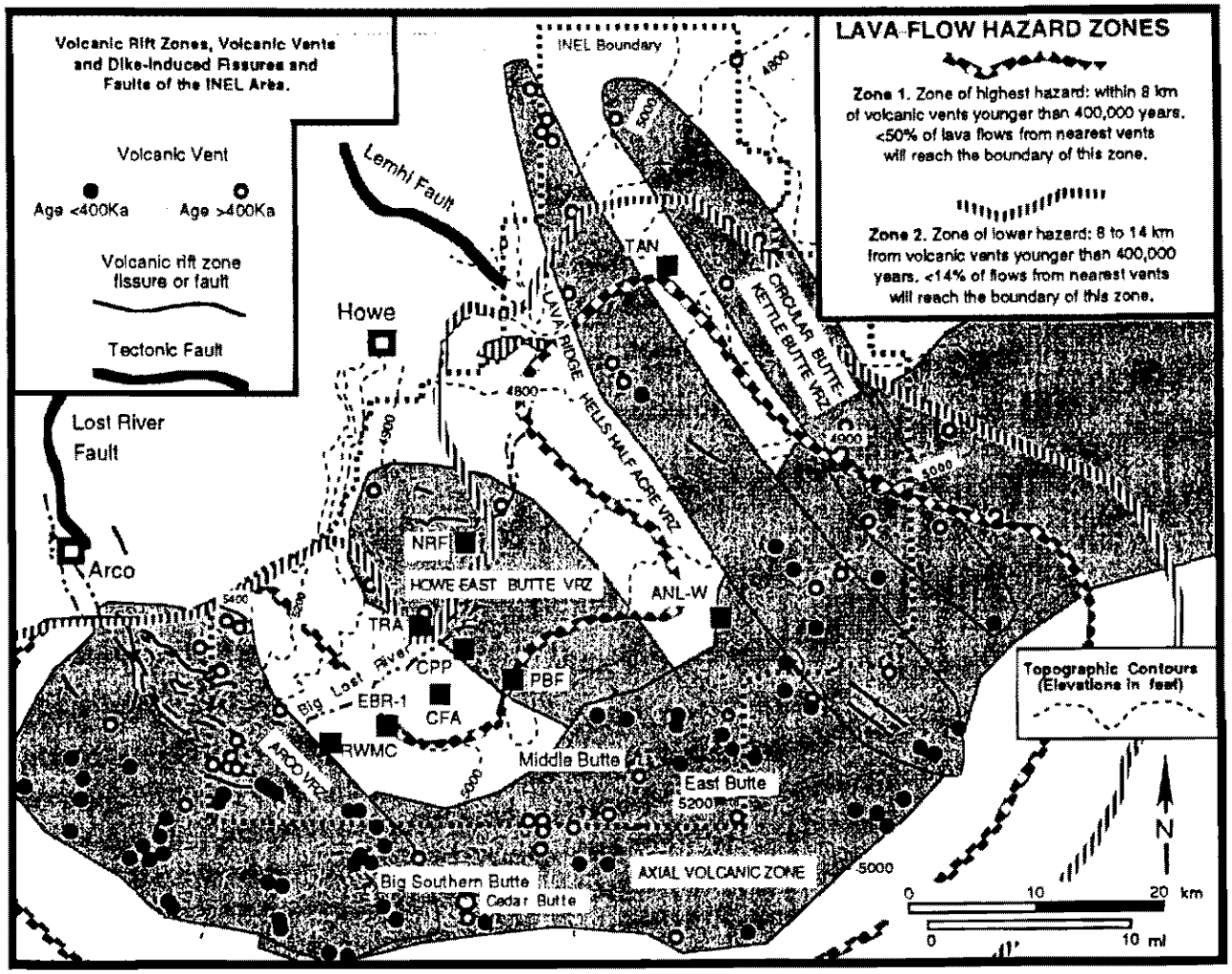


Figure 2A.6-3. Lava flow hazard zones for the INEEL area. **(For Information Only)**

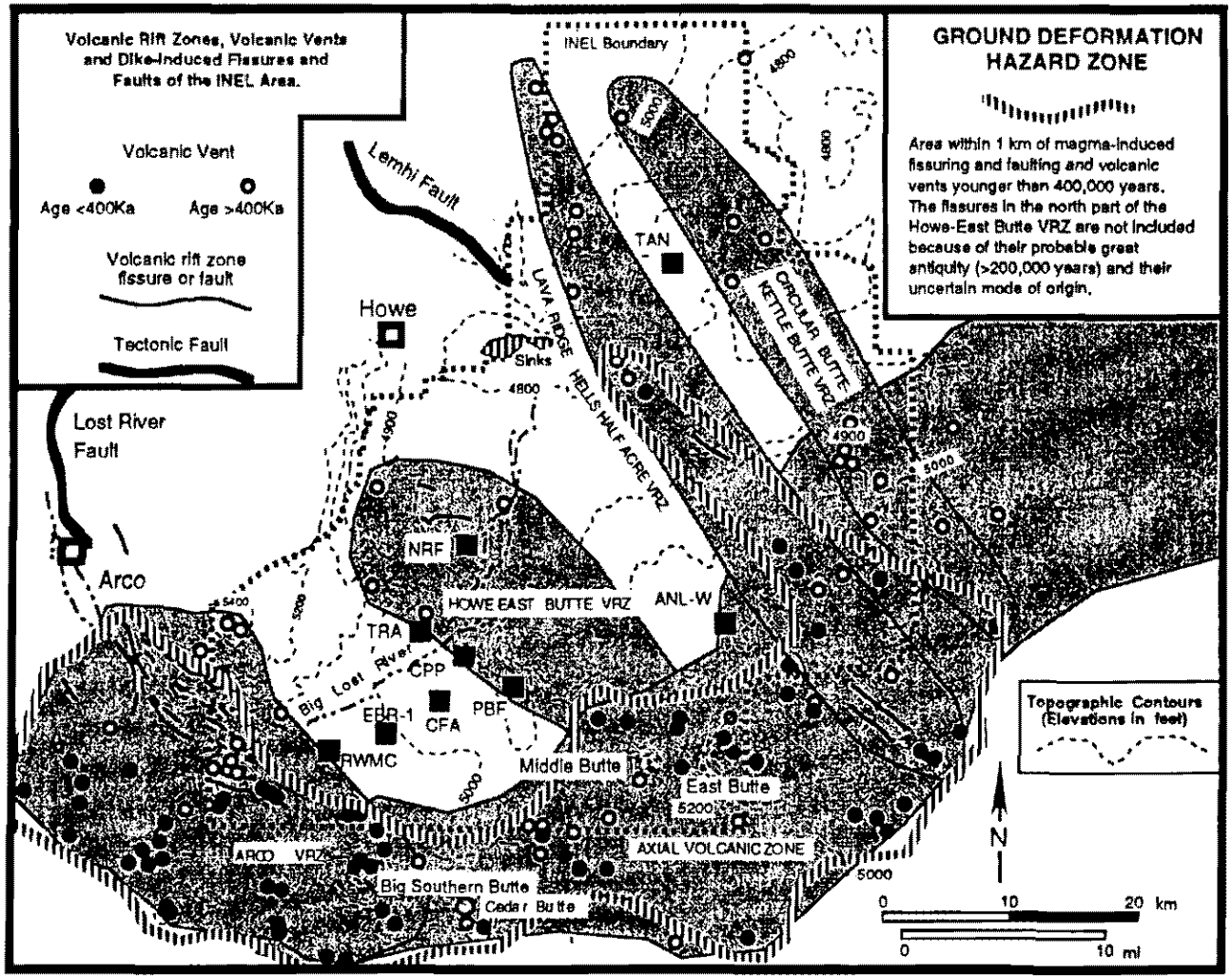


Figure 2A.6-4. Ground deformation hazard zone for INEEL. **(For Information Only)**

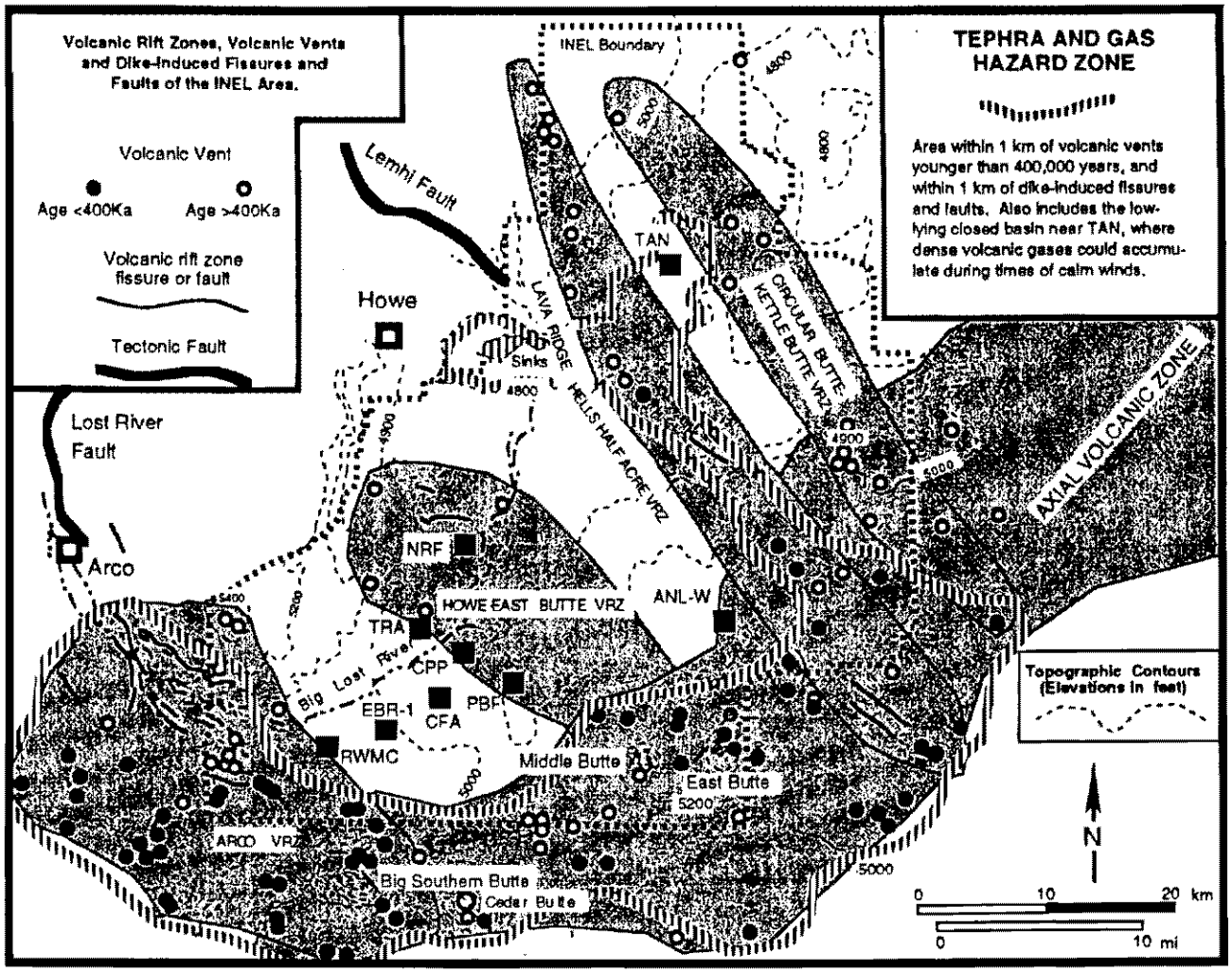


Figure 2A.6-5. Tephra (ash) and gas hazard zone for INEEL. **(For Information Only)**

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