Document ID: EHA-50
Revision ID: 0
Effective Date: 01/13/2010

Emergency Management Hazards Assessment

For

Advanced Test Reactor Complex

EMERGENCY MANAGEMENT HAZARDS ASSESSMENT FOR ADVANCED TEST REACTOR COMPLEX	Identifier: EHA-50 Revision: 0 Effective Date: 01/13/10	Page: 1 of 75

Emergency Management	Emergency Management Hazards Assessment		eCR Number: 568083

Manual: Hazards Assessment — Advanced Test Reactor (ATR) Complex

REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	01/13/2010	All	New document.

CONTENTS

FIGURES

TABLES

ACRONYMS

AEGL	Acute Exposure Guideline Level
AF	adjustment factor
ARF	airborne release fraction
ARR	airborne release rate
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical (Facility)
CAM	continuous air monitor
CAS	criticality alarm system
CDE	committed dose equivalent
DOE	Department of Energy
DOT	Department of Transportation
DR	damage ratio
EAL	emergency action level
EDF	Engineering Design File
EFIS	emergency firewater injection system
EHA	emergency management hazards assessment
EHS	emergency management hazards survey
EPIcode	Emergency Prediction Information Code
EPZ	emergency planning zone
ERG	Emergency Response Guidebook
ERPG	Emergency Response Planning Guideline
FACP	fire alarm control panel
FTVA	flux trap voiding accident
GE	general emergency

HC	Hazard Category
HEPA	high-efficiency particulate air
HPBO	high-pressure boil-off
HWST	hot waste storage tank
ICDF	Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
LOCA	loss-of-coolant accident
LPBO	low-pressure boil-off
LPF	leak path factor
MAR	material-at-risk
MCA	mass criticality storage
MSDS	material safety data sheet
MTR	Materials Test Reactor
N/A	not applicable
NFPA	National Fire Protection Association
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head
NTS	Nevada Test Site
OE	operational emergency
PA	protective action
PAC	protective action criteria
PAD	protective action distance
PAG	Protective Action Guide
PCB	polychlorinated biphenyl

PCP	primary coolant pump
PCS	primary coolant system
PE-g	plutonium equivalent grams
RAM	remote area monitor
RCRA	Resource Conservation and Recovery Act
RF	respirable fraction
RIA	reactivity insertion accident
RSAC	Radiological Safety Analysis Computer Program
SAE	site area emergency
SNF	spent nuclear fuel
SNM	special nuclear material
ST	source term
TEDE	total effective dose equivalent
TEEL	temporary emergency exposure limit
TEL	threshold for early lethality
TRU	transuranic
TTAF	Test Train Assembly Facility
UOE	unclassified operational emergency
WAC	waste acceptance criteria
WWFT	warm waste feed tank
WWTF	Warm Waste Treatment Facility

1. INTRODUCTION

DOE O 151.1C,[1] supported by DOE G 151.1-2,[2] requires a facility-specific emergency management hazards assessment (EHA) to provide the technical basis for facility emergency planning efforts. This report, in conjunction with the emergency management hazards survey (EHS) for the Advanced Test Reactor (ATR) Complex,[3] documents the EHA for facilities, operations, and activities associated with the Idaho National Laboratory (INL) ATR Complex in accordance with DOE O 151.1C and DOE G 151.1-2.

This report primarily relies on existing INL source documents as referenced herein. In most cases, the detailed information contained in the source documents was not reproduced in this EHA.

The EHS provides the initial qualitative hazards survey for all of the ATR Complex to identify specific buildings that require further assessment by the comprehensive emergency management program. This EHA addresses those additional assessments recommended in the EHS. It also considers any changes that may have occurred since the EHS was performed and broader generic issues (i.e., natural phenomena and fire). Consistent with these criteria, all identified radiological and nonradiological hazardous material stored, used, or produced from the different operations at the ATR Complex were reviewed and screened in accordance with Department of Energy (DOE) guidance. Accident scenarios were developed for the hazardous material that exceeded screening quantities to assess the potential consequences based on the specific hazard of concern. Emergency classifications, protective actions (PAs), and emergency action levels (EALs) were developed, as appropriate, based on consequence assessments documented in this report.

2. FACILITIES AND SITE DESCRIPTION

The INL Site is owned by DOE and operated as a multiprogram laboratory. DOE operates the Site through management and operating contractors. The primary mission of the Site is to provide the nation with innovations in nuclear technologies and unique scientific and engineering capabilities in nonnuclear programs that provide commercialization potential or enhance the quality of the environment.

The Site has no permanent residents and ingress and egress of Site and visiting personnel are strictly controlled. No casual visitations are permitted, except for persons driving through the Site on one of four public highways and visitors to the Experimental Breeder Reactor-I National Historic Landmark, which is only open during the summer. Security forces may interrupt traffic on Site roads or public roads that transverse the Site during emergencies and other times to support laboratory operations. Detailed information about the Site, including the potential effects of natural phenomena and external hazards on the ATR Complex, is described in the INL Safety Analysis Report.[4]

As shown in Figure 1, the Site includes a portion of five Idaho counties (i.e., Bingham, Bonneville, Butte, Clark, and Jefferson). Figure 2 shows a plot map of the ATR Complex.

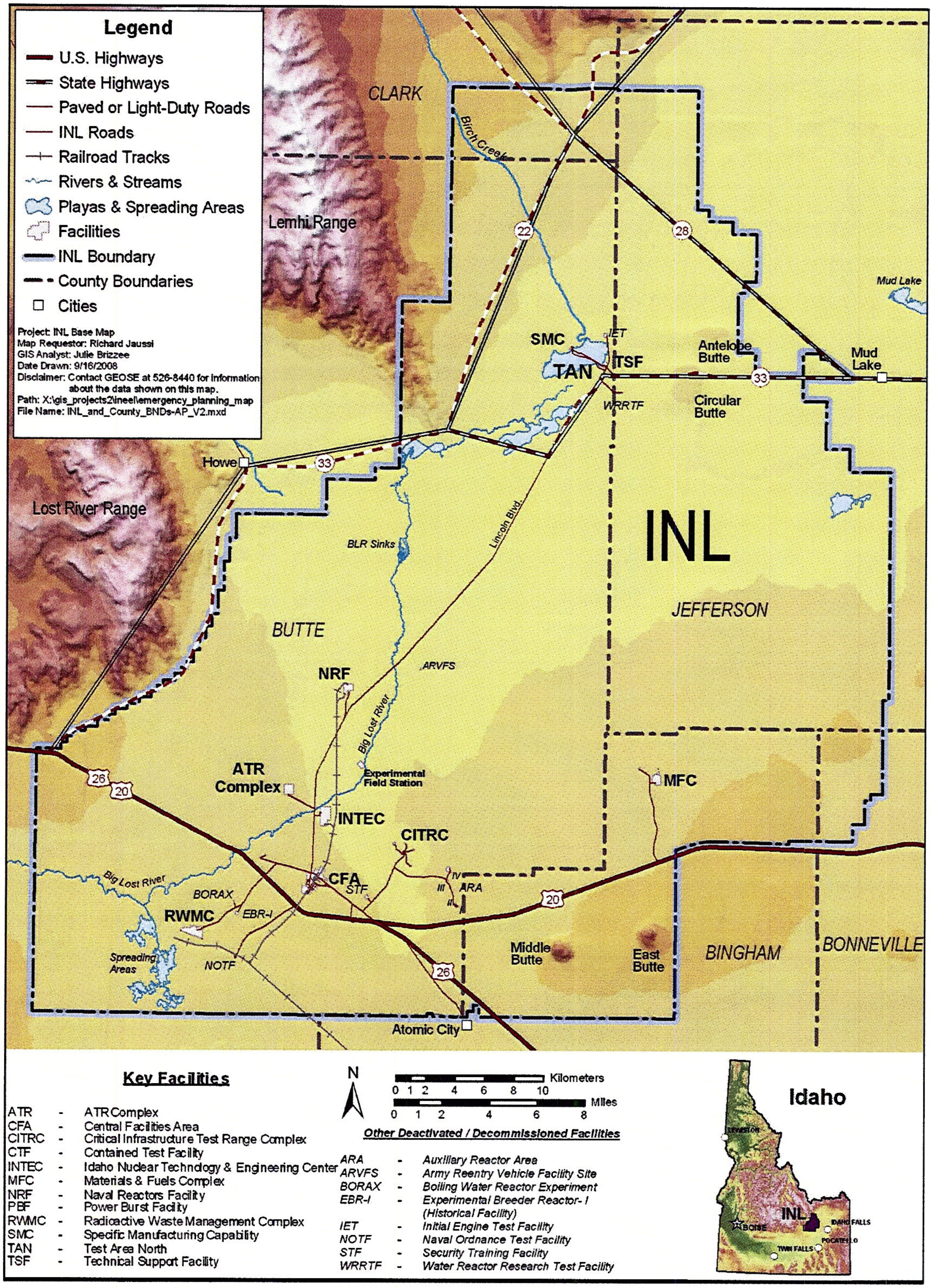


Figure 1. Location of the Idaho National Laboratory Site relative to surrounding five populated counties.

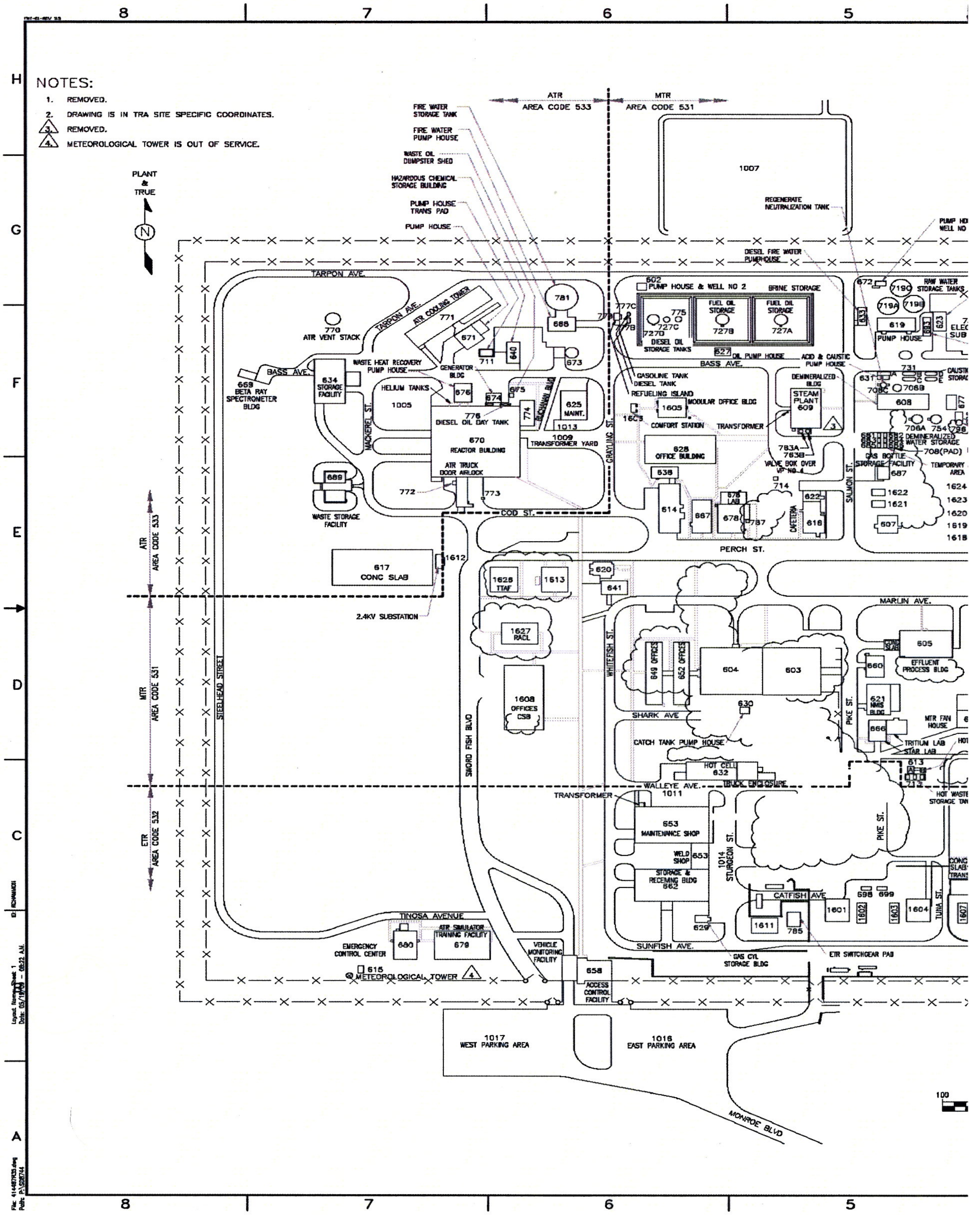
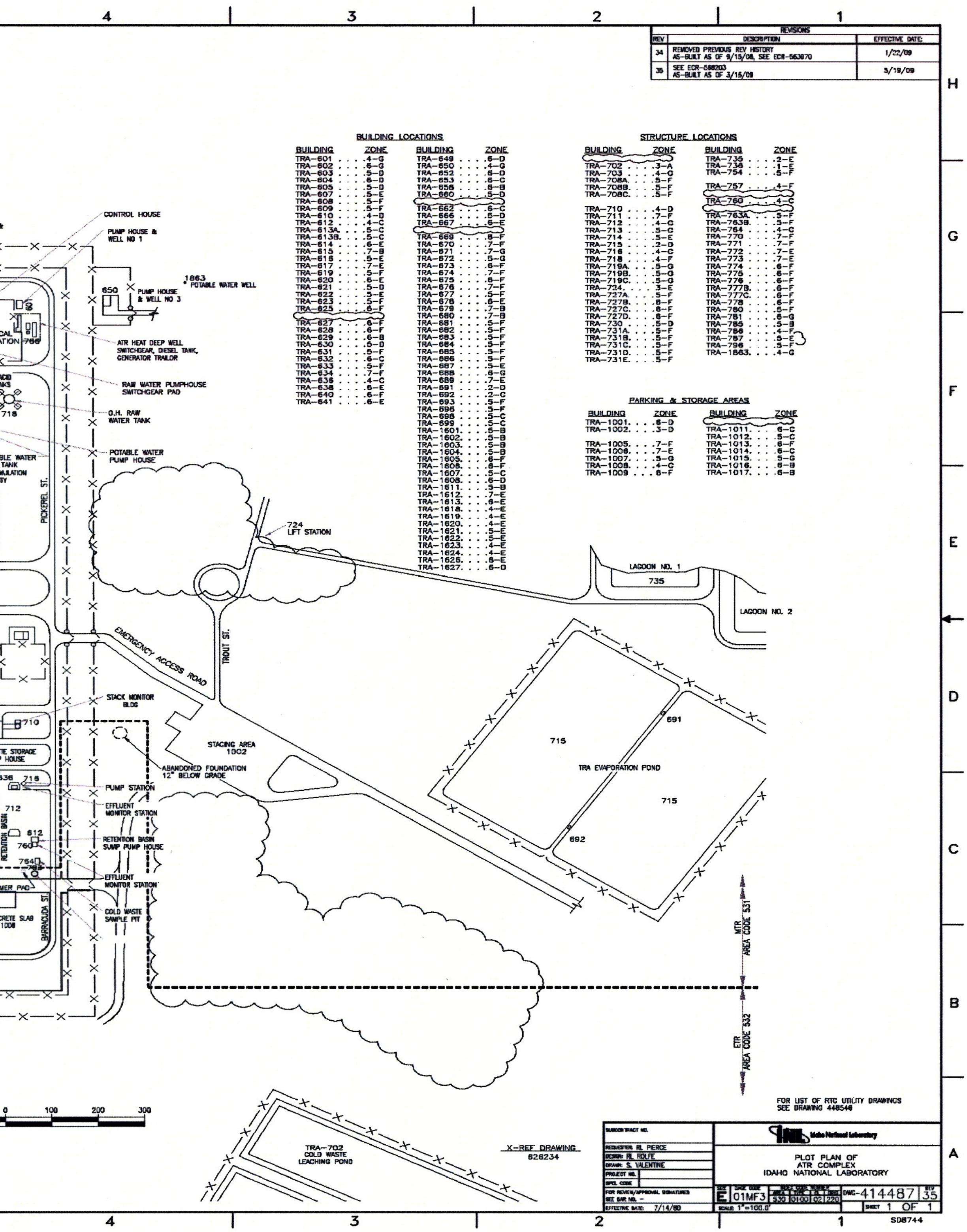


Figure 2. Map of the Advanced Test Reactor Complex with associated facilities/buildings.


REVISIONS
REV | DESCRIPTION | EFFECTIVE DATE:
34 | REMOVED PREVIOUS REV HISTORY AS-BUILT AS OF 9/15/08, SEE ECR-563870 | 1/22/09
35 | SEE ECR-588203 AS-BUILT AS OF 3/16/09 | 5/19/09
BUILDING LOCATIONS
BUILDING ZONE
TRA-601 . . . .4-G
TRA-602 . . . .6-G
TRA-603 . . . .5-D
TRA-604 . . . .6-D
TRA-605 . . . .5-D
TRA-607 . . . .5-E
TRA-608 . . . .5-F
TRA-609 . . . .5-F
TRA-610 . . . .4-D
TRA-612 . . . .4-C
TRA-613A. . . .5-C
TRA-613B. . . .5-C
TRA-614 . . . .6-E
TRA-615 . . . .7-B
TRA-616 . . . .5-E
TRA-617 . . . .7-E
TRA-619 . . . .5-F
TRA-620 . . . .6-E
TRA-621 . . . .5-D
TRA-622 . . . .5-E
TRA-623 . . . .5-F
TRA-625 . . . .6-F
TRA-627 . . . .6-F
TRA-628 . . . .6-F
TRA-629 . . . .6-B
TRA-630 . . . .5-D
TRA-631 . . . .5-F
TRA-632 . . . .6-C
TRA-633 . . . .5-F
TRA-634 . . . .7-F
TRA-636 . . . .4-C
TRA-638 . . . .6-E
TRA-640 . . . .6-F
TRA-641 . . . .6-E
TRA-649 . . . .6-D
TRA-650 . . . .4-G
TRA-652 . . . .6-D
TRA-653 . . . .6-C
TRA-658 . . . .6-B
TRA-660 . . . .5-D
TRA-662 . . . .6-C
TRA-666 . . . .5-D
TRA-667 . . . .6-E
TRA-669 . . . .8-F
TRA-670 . . . .7-F
TRA-671 . . . .7-G
TRA-672 . . . .5-G
TRA-673 . . . .6-F
TRA-674 . . . .7-F
TRA-675 . . . .6-F
TRA-676 . . . .7-F
TRA-677 . . . .5-F
TRA-678 . . . .6-E
TRA-679 . . . .7-B
TRA-680 . . . .7-B
TRA-681 . . . .5-F
TRA-682 . . . .5-F
TRA-683 . . . .5-F
TRA-684 . . . .5-F
TRA-685 . . . .5-F
TRA-686 . . . .5-F
TRA-687 . . . .5-E
TRA-688 . . . .6-G
TRA-689 . . . .7-E
TRA-691 . . . .2-D
TRA-692 . . . .2-C
TRA-693 . . . .5-F
TRA-696 . . . .3-F
TRA-698 . . . .5-C
TRA-699 . . . .5-C
TRA-1601. . . .5-B
TRA-1602. . . .5-B
TRA-1603. . . .5-B
TRA-1604. . . .5-B
TRA-1605. . . .6-F
TRA-1606. . . .6-F
TRA-1607. . . .5-C
TRA-1608. . . .6-D
TRA-1611. . . .5-B
TRA-1612. . . .7-E
TRA-1613. . . .6-E
TRA-1618. . . .4-E
TRA-1619. . . .4-E
TRA-1620. . . .4-E
TRA-1621. . . .5-E
TRA-1622. . . .5-E
TRA-1623. . . .4-E
TRA-1624. . . .4-E
TRA-1626. . . .6-E
TRA-1627. . . .6-D
STRUCTURE LOCATIONS
BUILDING ZONE
TRA-702 . . . .3-A
TRA-703 . . . .4-C
TRA-708A. . . .5-F
TRA-708B. . . .5-F
TRA-708C. . . .5-F
TRA-710 . . . .4-D
TRA-711 . . . .7-F
TRA-712 . . . .4-C
TRA-713 . . . .5-C
TRA-714 . . . .5-E
TRA-715 . . . .2-D
TRA-716 . . . .4-C
TRA-718 . . . .4-F
TRA-719A. . . .5-G
TRA-719B. . . .5-G
TRA-719C. . . .5-G
TRA-724. . . . 3-E
TRA-727A. . . .5-F
TRA-727B. . . .6-F
TRA-727C. . . .6-F
TRA-727D. . . .6-F
TRA-730 . . . .5-D
TRA-731A. . . .5-F
TRA-731B. . . .5-F
TRA-731C. . . .5-F
TRA-731D. . . .5-F
TRA-731E. . . .5-F
TRA-735 . . . .2-E
TRA-736 . . . .1-E
TRA-754 . . . .5-F
TRA-757 . . . .4-F
TRA-760 . . . .4-C
TRA-763A. . . .5-F
TRA-763B. . . .5-F
TRA-764 . . . .4-C
TRA-770 . . . .7-F
TRA-771 . . . .7-F
TRA-772 . . . .7-E
TRA-773 . . . .7-E
TRA-774 . . . .6-F
TRA-775 . . . .6-F
TRA-776 . . . .6-F
TRA-777B. . . .6-F
TRA-777C. . . .6-F
TRA-778 . . . .6-F
TRA-780 . . . .5-F
TRA-781 . . . .6-G
TRA-785 . . . .5-B
TRA-786 . . . .4-F
TRA-787 . . . .5-E
TRA-796 . . . .5-F
TRA-1863. . . .4-G
PARKING & STORAGE AREAS
BUILDING ZONE
TRA-1001. . . .6-D
TRA-1002. . . .3-D
TRA-1005. . . .7-F
TRA-1006. . . .7-E
TRA-1007. . . .5-G
TRA-1008. . . .4-C
TRA-1009 . . . 6-F
TRA-1011. . . .6-C
TRA-1012. . . .5-C
TRA-1013. . . .6-F
TRA-1014. . . .6-C
TRA-1015. . . .5-G
TRA-1016. . . .6-B
TRA-1017. . . .6-B
CONTROL HOUSE
PUMP HOUSE & WELL NO 1
650
PUMP HOUSE & WELL NO 3
1863 POTABLE WATER WELL
ATR HEAT DEEP WELL SWITCHGEAR, DIESEL TANK, GENERATOR TRAILOR
RAW WATER PUMPHOUSE SWITCHGEAR PAD
O.H. RAW WATER TANK
POTABLE WATER PUMP HOUSE
PICKEREL ST.
724 LIFT STATION
LAGOON NO. 1
735
LAGOON NO. 2
EMERGENCY ACCESS ROAD
TROUT ST.
STACK MONITOR BLDG
STAGING AREA 1002
ABANDONED FOUNDATION 12" BELOW GRADE
PUMP STATION
EFFLUENT MONITOR STATION
RETENTION BASIN SUMP PUMP HOUSE
EFFLUENT MONITOR STATION
COLD WASTE SAMPLE PIT
BARRACUDA ST.
715
691
TRA EVAPORATION POND
715
692
MTR AREA CODE 531
ETR AREA CODE 532
0 100 200 300
TRA-702 COLD WASTE LEACHING POND
X-REF DRAWING 626234
FOR LIST OF RTC UTILITY DRAWINGS SEE DRAWING 448546
SUBCONTRACT NO.
REQUESTER: RL PIERCE
DESIGN: RL ROLFE
DRAWN: S. VALENTINE
PROJECT NO.
SPCL CODE
FOR REVIEW/APPROVAL SIGNATURES SEE EAR NO. -
EFFECTIVE DATE: 7/14/80
Idaho National Laboratory
PLOT PLAN OF ATR COMPLEX IDAHO NATIONAL LABORATORY
SIZE E
CAGE CODE 01MF3
INDEX CODE NUMBER AREA 530 TYPE 0100 CL 02 ORIG 220
DWG-414487
REV 35
SCALE: 1"=100.0'
SHEET 1 OF 1
SD8744

2.1 Facility Background and Mission

In 1949, the United States Atomic Energy Commission established the Site as the National Reactor Testing Station for testing various types of nuclear reactors and associated equipment. In 1974, it was made a national laboratory by congressional action and renamed the Idaho National Engineering Laboratory. It is presently called the INL Site (or "Site"). New facilities have been built and original buildings modified to accommodate activities (e.g., testing and gathering data for reactor safety experiments; fuel studies; boiling water reactors; breeder reactors; naval reactors; waste management programs; geothermal research; computer code development, validation, and verification; and environmental research).

2.2 Processes and Operations Overview

The ATR Complex mission is to provide buildings, utilities, and support capabilities to enable government and private agencies to conduct experiments associated with the development, testing, and analysis of material used in nuclear and reactor applications. As a functioning operation, the ATR Complex is divided into the following basic areas that interact to fulfill the ATR Complex mission:

- ATR section
- Utility section
- Engineering and light laboratories
- New reactor research section
- Maintenance and security
- Research and development
- Sanitary waste and radioactive effluent operation.

3. IDENTIFICATION AND SCREENING OF HAZARDS

The EHS provides the initial qualitative hazards survey for all of the ATR Complex to identify specific buildings or structures that require further assessment by the comprehensive emergency management program. This EHA also considers any changes that may have occurred since the qualitative hazards survey was performed and broader generic issues such as natural phenomena and fire.

3.1 Buildings and Logical Groups Identified for Further Analyses

Table 1 lists all the buildings (or logical groups) identified in the EHS that are recommended for further analyses in this EHA.

Table 1. Buildings or logical groups recommended for further analyses.

Building or Structure	Potential Hazards	Basis for Recommendation
ATR Complex — All (onsite transportation)	Radiological, hazardous chemicals, radiological contamination	EHS-50,[3] Table 1
TRA-605, Effluent Processing Facility	Radiological	EHS-50, Table 1

Table 1. (continued).

Building or Structure	Potential Hazards	Basis for Recommendation
TRA-621, Nuclear Materials Inspection and Storage Facility	Radiological	EHS-50, Table 1
TRA-634, Advanced Test Reactor Storage Facility	Radiological	EHS-50, Table 1
TRA-670, Advanced Test Reactor Building	Radiological	EHS-50, Table 1
TRA-671, Cooling Tower Pumphouse	Hazardous chemical	EHS-50, Table 1
TRA-780, 90-Day Storage Area (TRA-682, -683, and -684)	Hazardous chemicals	EHS-50, Table 1

3.2 Buildings or Structures Removed from Further Consideration

No buildings identified as requiring further analysis have been screened out.

3.3 Hazardous Material Screening Results

Screening of radiological and nonradiological hazardous material stored, used, and produced at the ATR Complex is documented in EHS-50.

The radiological and nonradiological hazardous material in excess of screening thresholds is listed in Tables 2 and 3.

Table 2. Radiological hazardous material exceeding screening quantity thresholds.

Material	Location	Maximum Quantity	Details
Radiological (fission products)	TRA-605	Greater than Hazard Category (HC) 3	Appendix A
Radiological (fissile material)	TRA-621	Greater than HC-3	Appendix B
Radiological (fission products)	TRA-634	Greater than HC-3	Appendix C
Radiological (fission products)	TRA-670	Greater than HC-2	Appendix D
Radiological (spent nuclear fuel, irradiated test trains, contaminated waste)	ATR Complex (All)	Greater than HC-3	Appendix G

Table 3. Nonradiological hazardous material exceeding screening quantity thresholds.

Material	Location	Maximum Quantity	Details
Hazardous chemical (sulfuric acid)	TRA-671	Greater than 40 lb	Appendix E
Hazardous chemical (polychlorinated biphenyls, toxic metals)	90-Day Storage Area (TRA-682, -683, and -684)	Greater than 40 lb	Appendix F
Hazardous chemical (all Department of Transportation classes)	ATR Complex (All)	Greater than 40 lb solids 10 lb gases 5 gal liquids	Appendix G

4. HAZARDS CHARACTERIZATION

Radiological and nonradiological hazardous material that exceeded the screening criteria is characterized in the appropriate appendices.

5. EVENT SCENARIOS

Event scenario analyses details are provided in the appropriate appendices.

6. EVENT CONSEQUENCES

6.1 Calculation Models and Bases

6.1.1 Radiological Hazardous Material Model

Radiological hazardous material releases were modeled using the Radiological Safety Analysis Computer Program (RSAC), Version 6.2.[5] RSAC is a radiological safety analysis program that has been used extensively at the INL. It has been independently verified and validated for these types of calculations.

RSAC and computer information is shown in Table 4.

Table 4. Radiological Safety Analysis Computer Program and computer information for radiological hazardous material release modeling.

Parameter	Data
Program name	RSAC 6
Version No.	6.2
Configuration control No.	69104
Operating system	Microsoft Windows XP
Computer type	Dell OPTIPLEX 755
Central processing unit No.	413430

6.1.2 Nonradiological Hazardous Material Model

Nonradiological hazardous material releases were modeled using the Emergency Prediction Information Code Program, Version 7.0,[6] or Areal Locations of Hazardous Atmospheres Program.[7]

6.1.3 Meteorology

Radiological and nonradiological hazardous material release scenarios were evaluated for both the 95% worst-case and 50% typical meteorological conditions for the ATR Complex as described below.

Airborne releases of radiological hazardous material were modeled using the χ/Q values shown in Table 5 that were derived from data published by the National Oceanic and Atmospheric Administration for individual INL facilities.[8,9] The χ/Q values are relative measures of ground-level atmospheric diffusion compared to the initial points of release. In other words, the airborne concentration at a given distance is the product of the release source term times the χ/Q value for that distance.

Table 5. Meteorological parameters used for radiological hazardous material release modeling.

Distance (m)	95% χ/Q (s/m^3)	50% χ/Q (s/m^3)
100	2.34E−03	1.50E−05
200	1.05E−03	6.74E−06
300	6.56E−04	4.21E−06
400	4.71E−04	3.02E−06
500	3.63E−04	2.33E−06
600	2.94E−04	1.89E−06
700	2.46E−04	1.57E−06
800	2.11E−04	1.34E−06
990	4.47E−05	1.05E−06
1,170	3.62E−05	8.15E−07
1,390	2.78E−05	5.69E−07
1,640	2.15E−05	5.36E−07

Table 5. (continued).

Distance (m)	95% χ/Q (s/m^3)	50% χ/Q (s/m^3)
1,950	1.71E−05	5.08E−07
2,130	1.30E−05	4.66E−07
2,740	9.87E−06	2.88E−07
3,250	7.49E−06	2.60E−07
3,850	5.43E−06	1.80E−07
4,560	3.99E−06	1.57E−07
5,410	2.89E−06	1.22E−07
6,410	2.06E−06	9.42E−08
7,600	1.58E−06	7.16E−08
9,010	1.27E−06	5.24E−08
10,680	9.92E−07	4.28E−08
12,650	7.03E−07	3.48E−08
15,000	4.95E−07	2.94E−08
21,120	3.00E−07	2.02E−08
30,720	1.00E−07	Not determined
36,480	7.00E−08	Not determined
58,560	3.00E−08	Not determined
95,040	1.00E−08	Not determined
99,840	7.00E−09	Not determined
109,440	3.00E−09	Not determined
144,000	1.00E−09	Not determined

Airborne releases of nonradiological hazardous material were modeled using the default facility meteorological parameters shown in Table 6.

Table 6. Meteorological parameters used for nonradiological hazardous material release modeling.[a]

Meteorological Parameter	95% Worst-Case Meteorological Conditions	Typical (50%) Meteorological Conditions
Wind speed	1.04 m/s	2.46 m/s
Inversion height	Program default	Program default
Stability class	F	D
Ground roughness	Open country	Open country

a. For elevated releases only (typically fire), an additional case using the "typical meteorological conditions" wind speed with Stability Class A is considered. This condition also applies "open country" to account for the characteristics of the ATR Complex area.

6.1.4 Receptor Locations

This EHA considers, as a minimum, the potential consequences at the receptor locations shown in Table 7.

Table 7. Receptor locations of concern.

Receptor Identification	Description	Distance from Release Point
30 m	Nominal start distance alert level emergency determination	30 m
100 m	Potential alert level designation point	100 m
Facility boundary	Distance from the release point to the nearest facility boundary for an alert level emergency determination	48 m to a maximum of 200 m
Collocated facility	Distance from the release point to the nearest collocated facility boundary	2,865 m
Site boundary	Distance from the release point to the nearest Site boundary	10,855 m

6.1.5 Other Bases

None.

6.2 Analyses Results

6.2.1 Emergency Classification Criteria

This EHA assesses the potential consequences from the identified release scenarios. Based on these assessments, the analyzed scenarios are categorized as nonemergency or operational emergency (OE) events. Initiating events [i.e., criticalities, explosions, fires, seismic, and other natural phenomena (high winds, tornadoes, or flooding)] that may or may not release hazardous material are considered to be at least unclassified OEs. Classifiable OE events are identified as alerts, site area emergencies (SAEs), or general emergencies (GEs), depending on their degree of severity. Additional detail is provided in PLN-114.[10]

6.2.1.1 Radiological Release Emergency Classification Criteria. A radiological release scenario is classified as an alert if the airborne concentration for the analyzed material exceeds the PA Guide (PAG) at 30 m, but does not exceed the PAG value at the facility boundary or 200 m if the facility's boundary is greater than 200 m from the release point. If the facility boundary is not physically defined, 200 m is used as the alert/SAE boundary for classification purposes. Computer codes used to model airborne releases caution that airborne concentration predictions for distances closer than 100 m may not be reliable. The 30-m concentrations or projected dose may be approximated by using values equal to 10 times the material's concentration or projected dose value at 100 m from the release.

A radiological release scenario is classified as an SAE if the projected dose exceeds the PAG at the facility boundary or 200 m if the facility's boundary is greater than 200 m from the release point, but does not exceed the PAG at the Site boundary.

A radiological release scenario is classified as a GE if the projected dose exceeds the PAG at the Site boundary.

6.2.1.2 Nonradiological Hazardous Material Release Emergency Classification Criteria.

A nonradiological hazardous material release scenario is classified as an alert if the airborne concentration for the analyzed hazardous material exceeds the material's Acute Exposure Guideline Level (AEGL)-2, Emergency Response Planning Guideline (ERPG)-2, or temporary emergency exposure limit (TEEL)-2 value at 30 m, but does not exceed the material's AEGL-2/ERPG-2/TEEL-2 value at the facility boundary or 200 m if the facility's boundary is greater than 200 m from the release point. If the facility boundary is not physically defined, 200 m is used as the alert/SAE boundary for classification purposes. The 30-m concentrations may be approximated by using values equal to 10 times the material's concentration value at 100 m from the release.

A hazardous material release scenario is classified as an SAE if airborne concentrations for the analyzed hazardous material exceed the material's AEGL-2/ERPG-2/TEEL-2 value at the facility boundary or 200 m if the facility's boundary is greater than 200 m from the release point and the material's AEGL-2/ERPG-2/TEEL-2 value is not exceeded at the Site boundary.

A hazardous material release scenario is classified as a GE if airborne concentrations for the analyzed hazardous material exceed material's AEGL-2/ERPG-2/TEEL-2 value at the Site boundary.

6.2.2 Radiological and Nonradiological Modeling Results

Detailed information and results from the radiological and nonradiological modeling are provided in the appendices. Technical discussion covering the model development, event scenarios, and results, including bases for the EALs, are contained therein.

7. THE EMERGENCY PLANNING ZONE

The emergency planning zone (EPZ) is an area within which the EHA results indicate a need for specific and detailed planning to protect people from the consequences of hazardous material releases. The choice of the EPZ is based on an objective analysis of the hazards associated with a facility and not on arbitrary factors such as historical precedent or distance to the Site boundary. In this section, the generic results of the consequence calculations presented in this EHA are used to develop a facility EPZ in accordance with the method outlined in DOE emergency management guidance.

7.1 Determination of Emergency Planning Zone Radius

Are there any scenarios that have the potential for a classifiable OE? If so, is an EPZ required?

Analysis: Yes, 54 classifiable OEs were developed from the analyzed scenarios. Therefore, an EPZ is required.

- Do any scenarios reach the threshold for early lethality (TEL)? If so, what is the distance to early lethality under worst-case meteorological conditions and is the smallest EPZ radius considered?

  Analysis: Yes, 31 event scenarios (22 radiological material and 9 nonradiological hazardous material) exceeded the TEL or its equivalent for nonradiological hazardous material releases. The maximum TEL distance under worst-case meteorological conditions was 19,400 m. Two other scenarios had a TEL distance of 4,150 m. Therefore, the 19,400-m distance was considered for establishing the minimum EPZ radius.

- The maximum distance at which PA criteria (PAC) could be exceeded under worst-case meteorological conditions is the largest EPZ radius considered.

  Analysis: Three of the 10 GE classification scenarios exceeded the PAC at the end of an extended release period, beyond 100 km, which was the maximum distance for computer dose projections using Gaussian plume models. Nine of the 10 GE scenarios exceeded the PAC beyond 16 km by the end of the radiological material release. Seven of the 10 GE scenarios exceeded the PAC beyond 35 km by the end of the radiological material release. Six of the 10 GE scenarios exceeded the PAC beyond the Site boundary within 4 hours of the start of the radiological material release. Since all of the GE scenarios involved radiological material releases, the maximum distance considered for an EPZ radius was 16 km, which was based on the TEL distance being greater than 16 km.

- Consider relative probability of the worst-case scenario.

  Analysis: The ATR canal drain event (Scenario No. ATRPDS-7) was the worst-case scenario, which exceeded the PAC beyond 100 km. The probability of the canal drain event is much less than 1.0E−6.

- Consider nonradiological hazardous material releases favor larger EPZs and radiological releases favor smaller EPZs.

  Analysis: All of the nonradiological hazardous material release scenarios exceeded the PAC within the existing 16-km ATR Complex EPZ radius. The maximum PAC distance for the nonradiological material scenarios was 1.8 km. The TEL distance for the worst-case radiological material scenario was 19 km, which was considered when establishing the minimum EPZ size.

- Consider potentially affected populations (e.g., fishermen, campers, and farmers).

  Analysis: The populations affected by ATR Complex scenario events were primarily other collocated facilities within the Site boundaries. Some grazing lease lands on the Site and offsite agricultural lands are located immediately adjacent to the Site boundary. There are no offsite population centers and no known offsite permanent residences located within 16 km of the ATR Complex. Four population centers are located within 35 km of the ATR Complex; two of these population centers are in the prevailing downwind direction. The two population centers closest to the ATR Complex are not in the prevailing wind direction.

- Consider physical features and jurisdictional boundaries.

  Analysis: The Site boundary is the most dominant feature for establishing an EPZ boundary; however, a circular EPZ would extend a short distance beyond the Site boundary in two locations where there are no permanent residents. There is a network of "T" roads is onsite, which could be considered when describing PA areas.

Based on this analysis, the recommended EPZ for the ATR Complex is a 16-km radius centered on the ATR Complex out to the intersection with the Site boundary, at which point the Site boundary becomes the EPZ boundary

7.2 Tests of Reasonableness

Four significant considerations are presented below in the form of questions to be used as "tests of reasonableness" for the proposed EPZ size.

- Is the EPZ large enough to provide a credible basis for extending response activities outside the EPZ if conditions warrant?

  Analysis: Yes, the proposed EPZ meets this criterion.

- Is the EPZ large enough to support an effective response at and near the scene of the emergency (i.e., preclude interference from uninvolved people and activity, facilitate onsite PAs, and optimize on-scene command and control and mitigation efforts)?

  Analysis: Yes, the EPZ is large enough.

- Does the proposed EPZ conform to natural and jurisdictional boundaries where reasonable, and are other expectations and needs of the offsite agencies likely to be met by the selected EPZ?

  Analysis: The EPZ is located mostly within the Site, which makes a good jurisdictional boundary. Offsite planning already exists for the Site in general.

- What enhancement of the facility and site preparedness stature would be achieved by increasing the size of the EPZ? What resources, costs, and liabilities might a larger EPZ engender? Would a larger EPZ result in a significant increase in preparedness without large increases in cost or other detriment?

  Analysis: The 16-km EPZ radius provides complete coverage of all nonradiological hazardous material release scenarios and sets the base for any ad hoc PAs for low probability accident scenarios that may go beyond the 16-km distance. A larger radius circular EPZ incurs significant additional costs associated with evacuating nonessential personnel from areas not in the downwind direction from the affected facility. The 16-km EPZ radius impacts a Site facility that is operated by a non-DOE Idaho Operations Office entity, which could result in jurisdictional issues. Preparation of new maps and training for the onsite and offsite emergency response entities takes additional resources. An EPZ larger than a 16-km radius would not enhance the current level of emergency preparedness for the Site.

7.3 Final Emergency Planning Zone Determination

The recommended EPZ for the ATR Complex is a 16-km radius out to the intersection with the Site boundary, at which point the Site boundary becomes the EPZ boundary. The EPZ size is based on consequence assessment that determined that several radiological material event scenarios were projected to exceed the PAC beyond the existing 16-km ATR Complex EPZ radius. The TEL distance for the most severe radiological material event was 19,400 m, which is greater than the maximum 10-mi (16-km) EPZ size established by DOE and the Nuclear Regulatory Commission. Figure 3 shows the recommended EPZ for the ATR Complex.

8. EMERGENCY CLASSES, PROTECTIVE ACTIONS, AND EMERGENCY ACTION LEVELS

Table 8 presents a complete list of the EALs developed by this EHA. The appendices provide additional detail.

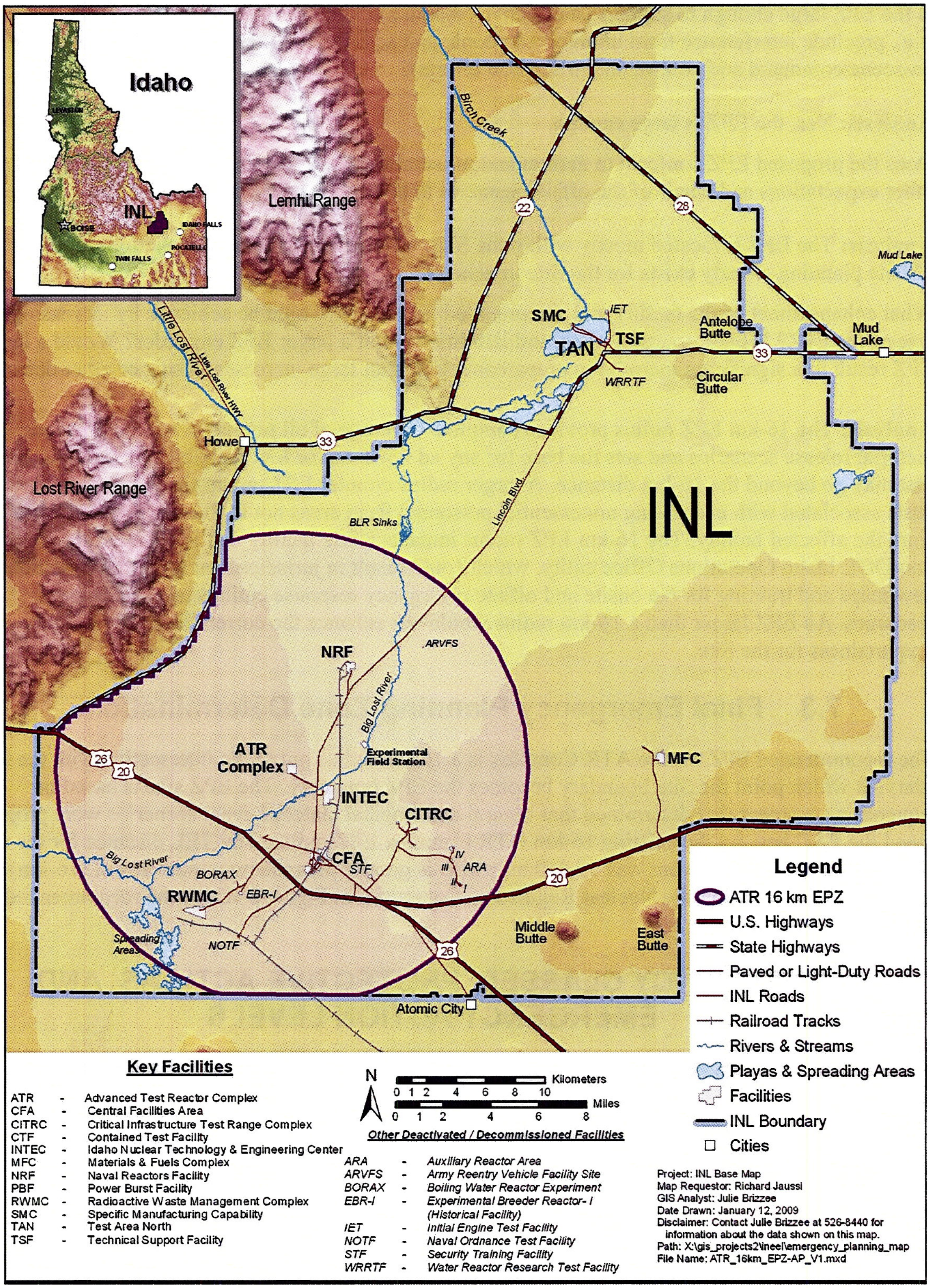


Figure 3. Recommended emergency planning zone for the Advanced Test Reactor Complex.

2.1 Facility Background and Mission

In 1949, the United States Atomic Energy Commission established the Site as the National Reactor Testing Station for testing various types of nuclear reactors and associated equipment. In 1974, it was made a national laboratory by congressional action and renamed the Idaho National Engineering Laboratory. It is presently called the INL Site (or "Site"). New facilities have been built and original buildings modified to accommodate activities (e.g., testing and gathering data for reactor safety experiments; fuel studies; boiling water reactors; breeder reactors; naval reactors; waste management programs; geothermal research; computer code development, validation, and verification; and environmental research).

2.2 Processes and Operations Overview

The ATR Complex mission is to provide buildings, utilities, and support capabilities to enable government and private agencies to conduct experiments associated with the development, testing, and analysis of material used in nuclear and reactor applications. As a functioning operation, the ATR Complex is divided into the following basic areas that interact to fulfill the ATR Complex mission:

- ATR section
- Utility section
- Engineering and light laboratories
- New reactor research section
- Maintenance and security
- Research and development
- Sanitary waste and radioactive effluent operation.

3. IDENTIFICATION AND SCREENING OF HAZARDS

The EHS provides the initial qualitative hazards survey for all of the ATR Complex to identify specific buildings or structures that require further assessment by the comprehensive emergency management program. This EHA also considers any changes that may have occurred since the qualitative hazards survey was performed and broader generic issues such as natural phenomena and fire.

3.1 Buildings and Logical Groups Identified for Further Analyses

Table 1 lists all the buildings (or logical groups) identified in the EHS that are recommended for further analyses in this EHA.

Table 1. Buildings or logical groups recommended for further analyses.

Building or Structure	Potential Hazards	Basis for Recommendation
ATR Complex — All (onsite transportation)	Radiological, hazardous chemicals, radiological contamination	EHS-50,[3] Table 1
TRA-605, Effluent Processing Facility	Radiological	EHS-50, Table 1

Table 1. (continued).

Building or Structure	Potential Hazards	Basis for Recommendation
TRA-621, Nuclear Materials Inspection and Storage Facility	Radiological	EHS-50, Table 1
TRA-634, Advanced Test Reactor Storage Facility	Radiological	EHS-50, Table 1
TRA-670, Advanced Test Reactor Building	Radiological	EHS-50, Table 1
TRA-671, Cooling Tower Pumphouse	Hazardous chemical	EHS-50, Table 1
TRA-780, 90-Day Storage Area (TRA-682, -683, and -684)	Hazardous chemicals	EHS-50, Table 1

3.2 Buildings or Structures Removed from Further Consideration

No buildings identified as requiring further analysis have been screened out.

3.3 Hazardous Material Screening Results

Screening of radiological and nonradiological hazardous material stored, used, and produced at the ATR Complex is documented in EHS-50.

The radiological and nonradiological hazardous material in excess of screening thresholds is listed in Tables 2 and 3.

Table 2. Radiological hazardous material exceeding screening quantity thresholds.

Material	Location	Maximum Quantity	Details
Radiological (fission products)	TRA-605	Greater than Hazard Category (HC) 3	Appendix A
Radiological (fissile material)	TRA-621	Greater than HC-3	Appendix B
Radiological (fission products)	TRA-634	Greater than HC-3	Appendix C
Radiological (fission products)	TRA-670	Greater than HC-2	Appendix D
Radiological (spent nuclear fuel, irradiated test trains, contaminated waste)	ATR Complex (All)	Greater than HC-3	Appendix G

Table 3. Nonradiological hazardous material exceeding screening quantity thresholds.

Material	Location	Maximum Quantity	Details
Hazardous chemical (sulfuric acid)	TRA-671	Greater than 40 lb	Appendix E
Hazardous chemical (polychlorinated biphenyls, toxic metals)	90-Day Storage Area (TRA-682, -683, and -684)	Greater than 40 lb	Appendix F
Hazardous chemical (all Department of Transportation classes)	ATR Complex (All)	Greater than 40 lb solids 10 lb gases 5 gal liquids	Appendix G

4. HAZARDS CHARACTERIZATION

Radiological and nonradiological hazardous material that exceeded the screening criteria is characterized in the appropriate appendices.

5. EVENT SCENARIOS

Event scenario analyses details are provided in the appropriate appendices.

6. EVENT CONSEQUENCES

6.1 Calculation Models and Bases

6.1.1 Radiological Hazardous Material Model

Radiological hazardous material releases were modeled using the Radiological Safety Analysis Computer Program (RSAC), Version 6.2.[5] RSAC is a radiological safety analysis program that has been used extensively at the INL. It has been independently verified and validated for these types of calculations.

RSAC and computer information is shown in Table 4.

Table 4. Radiological Safety Analysis Computer Program and computer information for radiological hazardous material release modeling.

Parameter	Data
Program name	RSAC 6
Version No.	6.2
Configuration control No.	69104
Operating system	Microsoft Windows XP
Computer type	Dell OPTIPLEX 755
Central processing unit No.	413430

6.1.2 Nonradiological Hazardous Material Model

Nonradiological hazardous material releases were modeled using the Emergency Prediction Information Code Program, Version 7.0,[6] or Areal Locations of Hazardous Atmospheres Program.[7]

6.1.3 Meteorology

Radiological and nonradiological hazardous material release scenarios were evaluated for both the 95% worst-case and 50% typical meteorological conditions for the ATR Complex as described below.

Airborne releases of radiological hazardous material were modeled using the χ/Q values shown in Table 5 that were derived from data published by the National Oceanic and Atmospheric Administration for individual INL facilities.[8,9] The χ/Q values are relative measures of ground-level atmospheric diffusion compared to the initial points of release. In other words, the airborne concentration at a given distance is the product of the release source term times the χ/Q value for that distance.

Table 5. Meteorological parameters used for radiological hazardous material release modeling.

Distance (m)	95% χ/Q (s/m^3)	50% χ/Q (s/m^3)
100	2.34E−03	1.50E−05
200	1.05E−03	6.74E−06
300	6.56E−04	4.21E−06
400	4.71E−04	3.02E−06
500	3.63E−04	2.33E−06
600	2.94E−04	1.89E−06
700	2.46E−04	1.57E−06
800	2.11E−04	1.34E−06
990	4.47E−05	1.05E−06
1,170	3.62E−05	8.15E−07
1,390	2.78E−05	5.69E−07
1,640	2.15E−05	5.36E−07

Table 5. (continued).

Distance (m)	95% χ/Q (s/m^3)	50% χ/Q (s/m^3)
1,950	1.71E−05	5.08E−07
2,130	1.30E−05	4.66E−07
2,740	9.87E−06	2.88E−07
3,250	7.49E−06	2.60E−07
3,850	5.43E−06	1.80E−07
4,560	3.99E−06	1.57E−07
5,410	2.89E−06	1.22E−07
6,410	2.06E−06	9.42E−08
7,600	1.58E−06	7.16E−08
9,010	1.27E−06	5.24E−08
10,680	9.92E−07	4.28E−08
12,650	7.03E−07	3.48E−08
15,000	4.95E−07	2.94E−08
21,120	3.00E−07	2.02E−08
30,720	1.00E−07	Not determined
36,480	7.00E−08	Not determined
58,560	3.00E−08	Not determined
95,040	1.00E−08	Not determined
99,840	7.00E−09	Not determined
109,440	3.00E−09	Not determined
144,000	1.00E−09	Not determined

Airborne releases of nonradiological hazardous material were modeled using the default facility meteorological parameters shown in Table 6.

Table 6. Meteorological parameters used for nonradiological hazardous material release modeling.[a]

Meteorological Parameter	95% Worst-Case Meteorological Conditions	Typical (50%) Meteorological Conditions
Wind speed	1.04 m/s	2.46 m/s
Inversion height	Program default	Program default
Stability class	F	D
Ground roughness	Open country	Open country

a. For elevated releases only (typically fire), an additional case using the "typical meteorological conditions" wind speed with Stability Class A is considered. This condition also applies "open country" to account for the characteristics of the ATR Complex area.

6.1.4 Receptor Locations

This EHA considers, as a minimum, the potential consequences at the receptor locations shown in Table 7.

Table 7. Receptor locations of concern.

Receptor Identification	Description	Distance from Release Point
30 m	Nominal start distance alert level emergency determination	30 m
100 m	Potential alert level designation point	100 m
Facility boundary	Distance from the release point to the nearest facility boundary for an alert level emergency determination	48 m to a maximum of 200 m
Collocated facility	Distance from the release point to the nearest collocated facility boundary	2,865 m
Site boundary	Distance from the release point to the nearest Site boundary	10,855 m

6.1.5 Other Bases

None.

6.2 Analyses Results

6.2.1 Emergency Classification Criteria

This EHA assesses the potential consequences from the identified release scenarios. Based on these assessments, the analyzed scenarios are categorized as nonemergency or operational emergency (OE) events. Initiating events [i.e., criticalities, explosions, fires, seismic, and other natural phenomena (high winds, tornadoes, or flooding)] that may or may not release hazardous material are considered to be at least unclassified OEs. Classifiable OE events are identified as alerts, site area emergencies (SAEs), or general emergencies (GEs), depending on their degree of severity. Additional detail is provided in PLN-114.[10]

6.2.1.1 Radiological Release Emergency Classification Criteria. A radiological release scenario is classified as an alert if the airborne concentration for the analyzed material exceeds the PA Guide (PAG) at 30 m, but does not exceed the PAG value at the facility boundary or 200 m if the facility's boundary is greater than 200 m from the release point. If the facility boundary is not physically defined, 200 m is used as the alert/SAE boundary for classification purposes. Computer codes used to model airborne releases caution that airborne concentration predictions for distances closer than 100 m may not be reliable. The 30-m concentrations or projected dose may be approximated by using values equal to 10 times the material's concentration or projected dose value at 100 m from the release.

A radiological release scenario is classified as an SAE if the projected dose exceeds the PAG at the facility boundary or 200 m if the facility's boundary is greater than 200 m from the release point, but does not exceed the PAG at the Site boundary.

A radiological release scenario is classified as a GE if the projected dose exceeds the PAG at the Site boundary.

6.2.1.2 Nonradiological Hazardous Material Release Emergency Classification Criteria.

A nonradiological hazardous material release scenario is classified as an alert if the airborne concentration for the analyzed hazardous material exceeds the material's Acute Exposure Guideline Level (AEGL)-2, Emergency Response Planning Guideline (ERPG)-2, or temporary emergency exposure limit (TEEL)-2 value at 30 m, but does not exceed the material's AEGL-2/ERPG-2/TEEL-2 value at the facility boundary or 200 m if the facility's boundary is greater than 200 m from the release point. If the facility boundary is not physically defined, 200 m is used as the alert/SAE boundary for classification purposes. The 30-m concentrations may be approximated by using values equal to 10 times the material's concentration value at 100 m from the release.

A hazardous material release scenario is classified as an SAE if airborne concentrations for the analyzed hazardous material exceed the material's AEGL-2/ERPG-2/TEEL-2 value at the facility boundary or 200 m if the facility's boundary is greater than 200 m from the release point and the material's AEGL-2/ERPG-2/TEEL-2 value is not exceeded at the Site boundary.

A hazardous material release scenario is classified as a GE if airborne concentrations for the analyzed hazardous material exceed material's AEGL-2/ERPG-2/TEEL-2 value at the Site boundary.

6.2.2 Radiological and Nonradiological Modeling Results

Detailed information and results from the radiological and nonradiological modeling are provided in the appendices. Technical discussion covering the model development, event scenarios, and results, including bases for the EALs, are contained therein.

7. THE EMERGENCY PLANNING ZONE

The emergency planning zone (EPZ) is an area within which the EHA results indicate a need for specific and detailed planning to protect people from the consequences of hazardous material releases. The choice of the EPZ is based on an objective analysis of the hazards associated with a facility and not on arbitrary factors such as historical precedent or distance to the Site boundary. In this section, the generic results of the consequence calculations presented in this EHA are used to develop a facility EPZ in accordance with the method outlined in DOE emergency management guidance.

7.1 Determination of Emergency Planning Zone Radius

Are there any scenarios that have the potential for a classifiable OE? If so, is an EPZ required?

Analysis: Yes, 54 classifiable OEs were developed from the analyzed scenarios. Therefore, an EPZ is required.

- Do any scenarios reach the threshold for early lethality (TEL)? If so, what is the distance to early lethality under worst-case meteorological conditions and is the smallest EPZ radius considered?

  Analysis: Yes, 31 event scenarios (22 radiological material and 9 nonradiological hazardous material) exceeded the TEL or its equivalent for nonradiological hazardous material releases. The maximum TEL distance under worst-case meteorological conditions was 19,400 m. Two other scenarios had a TEL distance of 4,150 m. Therefore, the 19,400-m distance was considered for establishing the minimum EPZ radius.

- The maximum distance at which PA criteria (PAC) could be exceeded under worst-case meteorological conditions is the largest EPZ radius considered.

  Analysis: Three of the 10 GE classification scenarios exceeded the PAC at the end of an extended release period, beyond 100 km, which was the maximum distance for computer dose projections using Gaussian plume models. Nine of the 10 GE scenarios exceeded the PAC beyond 16 km by the end of the radiological material release. Seven of the 10 GE scenarios exceeded the PAC beyond 35 km by the end of the radiological material release. Six of the 10 GE scenarios exceeded the PAC beyond the Site boundary within 4 hours of the start of the radiological material release. Since all of the GE scenarios involved radiological material releases, the maximum distance considered for an EPZ radius was 16 km, which was based on the TEL distance being greater than 16 km.

- Consider relative probability of the worst-case scenario.

  Analysis: The ATR canal drain event (Scenario No. ATRPDS-7) was the worst-case scenario, which exceeded the PAC beyond 100 km. The probability of the canal drain event is much less than 1.0E−6.

- Consider nonradiological hazardous material releases favor larger EPZs and radiological releases favor smaller EPZs.

  Analysis: All of the nonradiological hazardous material release scenarios exceeded the PAC within the existing 16-km ATR Complex EPZ radius. The maximum PAC distance for the nonradiological material scenarios was 1.8 km. The TEL distance for the worst-case radiological material scenario was 19 km, which was considered when establishing the minimum EPZ size.

- Consider potentially affected populations (e.g., fishermen, campers, and farmers).

  Analysis: The populations affected by ATR Complex scenario events were primarily other collocated facilities within the Site boundaries. Some grazing lease lands on the Site and offsite agricultural lands are located immediately adjacent to the Site boundary. There are no offsite population centers and no known offsite permanent residences located within 16 km of the ATR Complex. Four population centers are located within 35 km of the ATR Complex; two of these population centers are in the prevailing downwind direction. The two population centers closest to the ATR Complex are not in the prevailing wind direction.

- Consider physical features and jurisdictional boundaries.

  Analysis: The Site boundary is the most dominant feature for establishing an EPZ boundary; however, a circular EPZ would extend a short distance beyond the Site boundary in two locations where there are no permanent residents. There is a network of "T" roads is onsite, which could be considered when describing PA areas.

Based on this analysis, the recommended EPZ for the ATR Complex is a 16-km radius centered on the ATR Complex out to the intersection with the Site boundary, at which point the Site boundary becomes the EPZ boundary

7.2 Tests of Reasonableness

Four significant considerations are presented below in the form of questions to be used as "tests of reasonableness" for the proposed EPZ size.

- Is the EPZ large enough to provide a credible basis for extending response activities outside the EPZ if conditions warrant?

  Analysis: Yes, the proposed EPZ meets this criterion.

- Is the EPZ large enough to support an effective response at and near the scene of the emergency (i.e., preclude interference from uninvolved people and activity, facilitate onsite PAs, and optimize on-scene command and control and mitigation efforts)?

  Analysis: Yes, the EPZ is large enough.

- Does the proposed EPZ conform to natural and jurisdictional boundaries where reasonable, and are other expectations and needs of the offsite agencies likely to be met by the selected EPZ?

  Analysis: The EPZ is located mostly within the Site, which makes a good jurisdictional boundary. Offsite planning already exists for the Site in general.

- What enhancement of the facility and site preparedness stature would be achieved by increasing the size of the EPZ? What resources, costs, and liabilities might a larger EPZ engender? Would a larger EPZ result in a significant increase in preparedness without large increases in cost or other detriment?

  Analysis: The 16-km EPZ radius provides complete coverage of all nonradiological hazardous material release scenarios and sets the base for any ad hoc PAs for low probability accident scenarios that may go beyond the 16-km distance. A larger radius circular EPZ incurs significant additional costs associated with evacuating nonessential personnel from areas not in the downwind direction from the affected facility. The 16-km EPZ radius impacts a Site facility that is operated by a non-DOE Idaho Operations Office entity, which could result in jurisdictional issues. Preparation of new maps and training for the onsite and offsite emergency response entities takes additional resources. An EPZ larger than a 16-km radius would not enhance the current level of emergency preparedness for the Site.

7.3 Final Emergency Planning Zone Determination

The recommended EPZ for the ATR Complex is a 16-km radius out to the intersection with the Site boundary, at which point the Site boundary becomes the EPZ boundary. The EPZ size is based on consequence assessment that determined that several radiological material event scenarios were projected to exceed the PAC beyond the existing 16-km ATR Complex EPZ radius. The TEL distance for the most severe radiological material event was 19,400 m, which is greater than the maximum 10-mi (16-km) EPZ size established by DOE and the Nuclear Regulatory Commission. Figure 3 shows the recommended EPZ for the ATR Complex.

8. EMERGENCY CLASSES, PROTECTIVE ACTIONS, AND EMERGENCY ACTION LEVELS

Table 8 presents a complete list of the EALs developed by this EHA. The appendices provide additional detail.

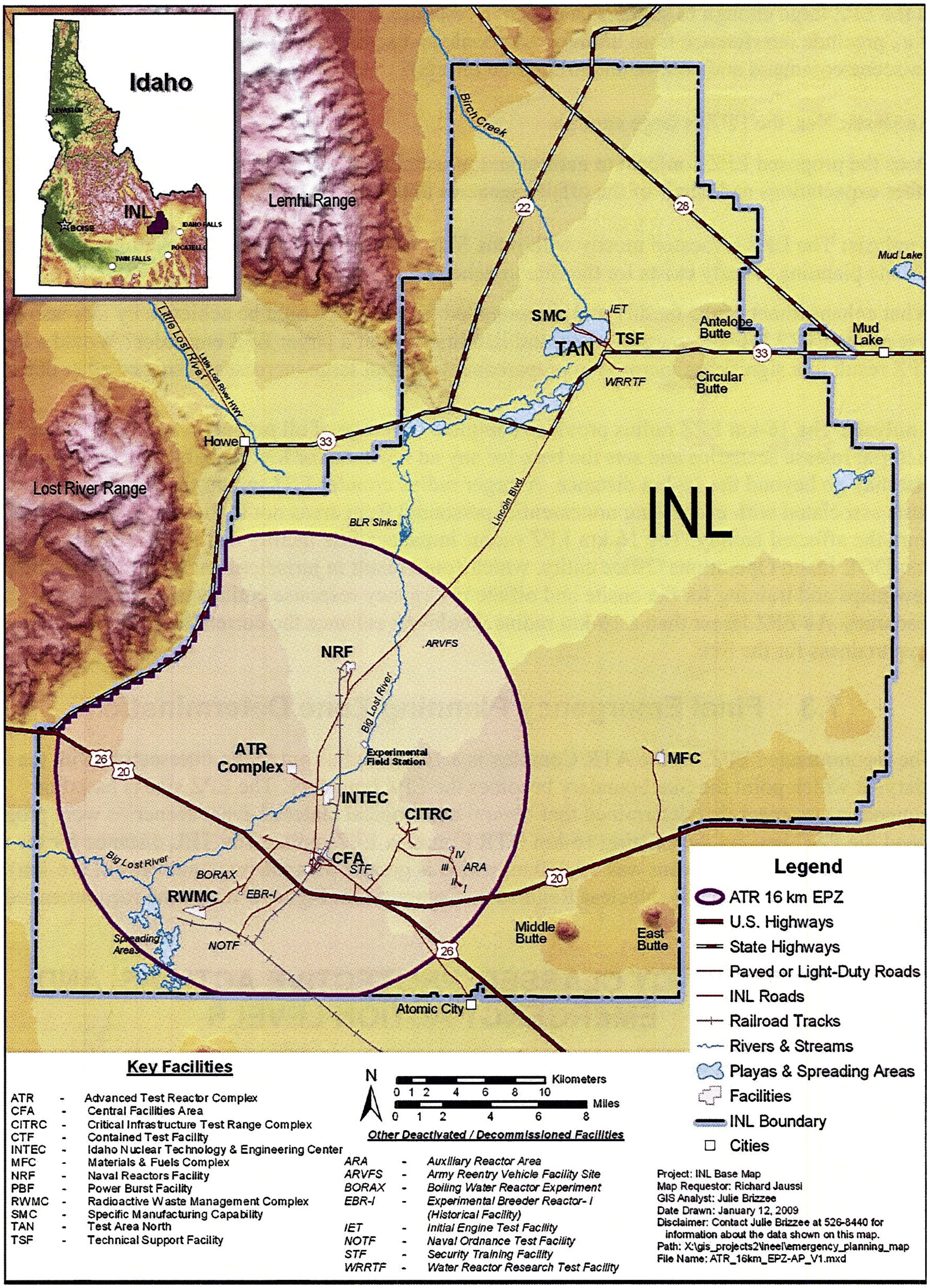


Figure 3. Recommended emergency planning zone for the Advanced Test Reactor Complex.

Table 8. Emergency classes, protective actions, and emergency action levels.

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-ALL-1.OE.1	A fire that causes or can reasonably be expected to cause significant structural damage to Idaho National Laboratory Site facilities, **AND** personnel injury or death is suspected or confirmed, **OR** requires personnel in nearby buildings to take shelter or evacuate. **NOTE:** *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Consider controlling vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.
ATR-ALL-2.OE.1	Any unplanned explosion that results in known or suspected personnel injury or damage to facilities.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Consider controlling vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.
ATR-ALL-2.OE.2	Any event that may reasonably be expected to cause a 1- to 40-gal propane tank to catastrophically fail and the propane tank is less than 100 m from the nearest facility fence.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Establish a **100-m (328-ft)** exclusion zone around the affected propane tank.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-ALL-2.OE.3	Any event that may reasonably be expected to cause a 41- to 1,000-gal propane tank to catastrophically fail.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Establish a **300-m (984-ft)** exclusion zone around the affected propane tank.	None.
ATR-ALL-2.OE.4	Any event that may reasonably be expected to cause a 1,001- to 5,000-gal propane tank to catastrophically fail.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Establish a **500-m (1,640-ft)** exclusion zone around the affected propane tank.	None.
ATR-ALL-2.OE.5	Any event that may reasonably be expected to cause a 5,001- to 13,000-gal propane tank to catastrophically fail.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Establish a **700-m (2,297-ft)** exclusion zone around the affected propane tank.	None.
ATR-ALL-2.OE.6	Any event that may reasonably be expected to cause a greater than 13,000-gal propane tank to catastrophically fail.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Establish a **900-m (2,953-ft)** exclusion zone around the affected propane tank.	None.
ATR-ALL-3.OE.1	Discovery of radioactive material contamination from past Department of Energy/National Nuclear Security Administration operations that is causing or may reasonably be expected to cause uncontrolled personnel exposures exceeding protective action criteria.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Consider controlling vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-ALL-4.OE.1	Any actual or potential release of hazardous material and a technician-level-trained hazardous material response team is required to mitigate the release, **AND** personnel injury or death is suspected or confirmed, **OR** requires personnel in nearby buildings to take shelter or evacuate.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Consider controlling vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.
ATR-ALL-4.OE.2	Discovery of hazardous material contamination from past Department of Energy/National Nuclear Security Administration operations that is causing or may reasonably be expected to cause uncontrolled personnel exposures exceeding protective action criteria.	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Consider controlling vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-ALL-5.OE.1	Any natural phenomena that may impact Idaho National Laboratory Site operations, communications, transportation, and/or health and safety of personnel such as: • Weather extremes, to include: – High or low temperatures – High winds – Tornadoes – Lightning – Floods – Snow • Earthquakes • Volcanic activity. **NOTE:** *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by the incident commander or appropriate responsible personnel. Consider controlling vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.
ATR-ALL-6.OE.1	Loss of power at any Idaho National Laboratory Site facility that compromises personnel health and safety. **NOTE:** *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Consider controlling nonessential vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-ALL-10.OE.1	Any unplanned criticality. (A criticality is considered an unclassified operational emergency, unless the criticality causes the release of hazardous material in quantities exceeding screening thresholds.)	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Establish a **100-m (328-ft)** exclusion zone around the affected facility.	None.
ATR-ALL-11.OE.1	Any event that damages or compromises structures or equipment that are intended to protect the health and safety of personnel, **AND** results in suspected or confirmed personnel injury or death or substantial degradation of health and safety, **OR** requires personnel in nearby buildings to take shelter or evacuate. **NOTE:** *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Consider controlling vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-ALL-12.OE.1	External event at any Idaho National Laboratory Site facility that involves an offsite hazardous material event not associated with Department of Energy operations that is observed to have or is predicted to have an impact on a Department of Energy site such that protective actions are required for Department of Energy workers, **OR** occurrence causes or can reasonably be expected to cause significant structural damage to Department of Energy facilities that results in suspected or confirmed personnel injury or death or substantial degradation of health and safety (e.g., airplane crash, train derailment). **NOTE:** *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*	Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel. Consider controlling vehicle access to the affected facility/area. Consider restricting nonessential personnel access to the affected facility/area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-TRN-3.A.1	Radiological waste shipment to the Nevada Test Site is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes the breach and spill of 25% to 100% of a single non-Department of Transportation Type-B radiological waste container or breach of a shipment of non-Department of Transportation Type-B radiological waste containers and a radiological release, **AS INDICATED BY** visual confirmation of the container breach, **OR** radiological control technician confirmation of an airborne radiological release by radiation survey.	Evacuate nonessential personnel to a distance of at least **105 m (345 ft)** from the accident site. Control nonessential vehicle and personnel access to the evacuated area.	None.
ATR-TRN-3.A.2	Cask lid dislodges from an Advanced Test Reactor 30-W spent nuclear fuel shipment due to a vehicle accident within the Advanced Test Reactor Complex and causes direct gamma exposure, **AS INDICATED BY** visual observation of the dislodged cask lid, **OR** radiological control technician confirmation of gamma radiation by radiation survey.	Evacuate nonessential personnel to a distance of at least **100 m (328 ft)** from the accident site. Control nonessential vehicle and personnel access to the evacuated area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-TRN-3.A.3	Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes the breach of one drum and spill of some of the transuranic material outside of the drum causing a radiological release, **AS INDICATED BY** visual confirmation of the drum breach, **AND** the container is substantially intact, **AND** the spilled material is not involved in a fire.	Evacuate nonessential personnel to a distance of at least **100 m (328 ft)** from the event. Control nonessential vehicle and personnel access to the evacuated area.	None.
ATR-TRN-3.SAE.1	Unsuppressed fire, lasting 20 minutes after fire suppression activities began, on a tractor trailer carrying a cask containing an irradiated test train causes fuel cladding damage and a radioactive material release, **AS INDICATED BY** visual observation of the fire engulfing the cask, **OR** radiological control technician confirmation of an airborne radiological release by radiation survey.	Evacuate nonessential personnel to a distance of at least **3,000 m (9,843 ft or 1.9 mi)** from the accident site. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated area.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-TRN-3.SAE.2	Unsuppressed fire, lasting 20 minutes after fire suppression activities began, on a tractor trailer carrying an Advanced Test Reactor cask containing eight Advanced Test Reactor spent nuclear fuel elements causes fuel cladding damage and a radioactive material release, **AS INDICATED BY** visual observation of the fire engulfing the cask, **OR** radiological control technician confirmation of an airborne radiological release by radiation survey.	Evacuate nonessential personnel to a distance of at least **8,075 m (26,500 ft or 5 mi)** from the accident site. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate. Control nonessential vehicle and personnel access to the evacuated area.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*
ATR-TRN-3.SAE.3	Cask lid dislodges from an irradiated test train shipment due to a vehicle accident and causes direct gamma exposure, **AS INDICATED BY** visual observation of the dislodged cask lid, **OR** radiological control technician confirmation of gamma radiation by radiation survey.	Evacuate nonessential personnel to a distance of at least **682 m (2,238 ft or 0.4 mi)** from the accident site. Control nonessential vehicle and personnel access to the evacuated area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-TRN-3.SAE.4	Cask lid dislodges from an Advanced Test Reactor 634-W spent nuclear fuel shipment due to a vehicle accident within the Advanced Test Reactor Complex and causes direct gamma exposure, **AS INDICATED BY** visual observation of the dislodged cask lid, **OR** radiological control technician confirmation of gamma radiation by radiation survey.	Evacuate nonessential personnel to a distance of at least **478 m (1,568 ft or 0.3 mi)** from the accident site. Control nonessential vehicle and personnel access to the evacuated area.	None.
ATR-TRN-3.SAE.5	Radiological waste shipment to the Nevada Test Site is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes the shipment of non-Department of Transportation Type-B radiological waste containers to catch fire causing a radiological release, **AS INDICATED BY** visual observation of the radiological waste shipment fire, **OR** radiological control technician confirmation of an airborne radiological release by radiation survey.	Evacuate nonessential personnel to a distance of at least **410 m (1,345 ft)** from the accident site. Control nonessential vehicle and personnel access to the evacuated area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-TRN-3.SAE.6	Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes a catastrophic breach of one drum and spill of most of the transuranic material outside of the drum causing a radiological release, **AS INDICATED BY** visual confirmation of the drum breach, **AND** the spilled material is not involved in a fire.	Evacuate nonessential personnel to a distance of at least **100 m (328 ft)** from the event. Shelter all nonessential personnel out to a distance of at least **200 m (656 ft)** from the event. Control nonessential vehicle and personnel access to the evacuated area.	None.
ATR-TRN-3.SAE.7	Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes an explosion/deflagration of one drum or a standard waste box causing a radiological release, **AS INDICATED BY** visual confirmation of the drum or standard waste box breach, **AND** the spilled material is involved in a fire.	Evacuate nonessential personnel to a distance of at least **100 m (328 ft)** from the event. Shelter all nonessential personnel out to a distance of at least **700 m (2,297 ft or 0.4 mi)** from the event. Consider evacuating nonessential personnel to a distance of at least **700 m (2,297 ft or 0.4 mi)** from the event. Control nonessential vehicle and personnel access to the evacuated area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-TRN-3.SAE.8	Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that involves more than one transuranic material container of any type, and up to 40 drums or 10 standard waste boxes, causing a radiological release, **AS INDICATED BY** visual confirmation of the drum or standard waste box breach, **AND** the spilled material is not involved in a fire.	Evacuate nonessential personnel to a distance of at least **200 m (656 ft)** from the event. As a precaution, shelter all nonessential personnel out to a distance of at least **1,000 m (3,280 ft)** from the event. Control nonessential vehicle and personnel access to the evacuated area.	None.
ATR-TRN-3.SAE.9	Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that involves a shipment of up to 40 drums or 10 standard waste boxes with a drum or standard waste box breach and fire causing a radiological release, **AS INDICATED BY** visual confirmation of the drum breach, **AND** the spilled material is involved in a fire.	Evacuate nonessential personnel to a distance of at least **200 m (656 ft)** from the event. Shelter all nonessential personnel out to a distance of at least **1,100 m (3,610 ft)** from the event. As soon as practicable, evacuate nonessential personnel to a distance of at least **1,100 m (3,610 ft)** from the event. Control nonessential vehicle and personnel access to the evacuated area.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-TRN-4.A.1	Any Advanced Test Reactor Complex transportation accident involving nonradiological hazardous material that results in a small spill greater than 5 gal and less than or equal to 60 gal, **AS INDICATED BY** visual observation of damage to the transport container and spillage of the contents, **AND** the incident commander recommends an evacuation/isolation distance less than or equal to **200 m (656 ft)**, **AND** protective actions are recommended for nearby buildings.	Follow instructions provided by the incident commander with respect to evacuation and isolation distances. Control nonessential vehicle and personnel access to the protective action area around the vehicle accident site.	None.
ATR-TRN-4.SAE.1	Any Advanced Test Reactor Complex transportation accident involving nonradiological hazardous material that results in a large spill equivalent to more than one 55-gal drum (greater than 60 gal) or a tanker, **AS INDICATED BY** visual observation of damage to the transport container and spillage of the contents, **AND** the incident commander recommends an evacuation/isolation distance greater than **200 m (656 ft)**.	Follow instructions provided by the incident commander with respect to evacuation and isolation distances. Control nonessential vehicle and personnel access to the protective action area around the vehicle accident site.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-605-3.A.1	Radiological release resulting from transferring contaminated (greater than 0.9 μCi/mL and less than or equal to 9 μCi/mL) primary coolant system water to TRA-605 tanks, **AS INDICATED BY** radiological control technician/chemist report of a primary coolant sample greater than 0.9 μCi/mL and less than or equal to 9 μCi/mL, **AND** loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout), **AND** radiation area monitor alarm in TRA-605, **OR** radiological control technician radiation survey in TRA-605 higher than normal background, **OR** other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.	Establish a **100-m (328-ft)** exclusion zone around TRA-605. Relocate nonessential personnel from the exclusion zone to the conference room in TRA-653. Shelter the remainder of Advanced Test Reactor Complex personnel in place. Control nonessential vehicle and personnel access to the exclusion zone.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-605-3.SAE.1	Radiological release resulting from transferring contaminated (greater than 9 µCi/mL and less than or equal to 20 µCi/mL) primary coolant system water to TRA-605 tanks, **AS INDICATED BY** radiological control technician/chemist report of a primary coolant sample greater than 9 µCi/mL and less than or equal to 20 µCi/mL, **AND** loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout), **AND** radiation area monitor alarm in TRA-605, **OR** radiological control technician radiation survey in TRA-605 higher than normal background, **OR** other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.	Establish a **220-m (722-ft or 0.14-mi)** exclusion zone around TRA-605. Relocate nonessential personnel from the exclusion zone to the conference room in TRA-652. Shelter the remainder of Advanced Test Reactor Complex personnel in place. Control nonessential vehicle and personnel access to the exclusion zone. Consider authorizing potassium iodide for essential emergency workers.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-605-3.SAE.2	Radiological release resulting from transferring contaminated (greater than 20 μCi/mL and less than or equal to 2.60E+3 μCi/mL) primary coolant system water to TRA-605 tanks, **AS INDICATED BY** radiological control technician/chemist report of a primary coolant sample greater than 20 μCi/mL and less than or equal to 2.60E+3 μCi/mL, **AND** loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout), **AND** radiation area monitor alarm in TRA-605, **OR** radiological control technician radiation survey in TRA-605 higher than normal background, **OR** other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **2,900 m (9,515 ft or 1.8 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609. Consider authorizing potassium iodide for essential emergency workers.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-605-3.SAE.3	Radiological release resulting from transferring contaminated (greater than 2.60E+3 μCi/mL and less than or equal to 1.67E+4 μCi/mL) primary coolant system water to TRA-605 tanks, **AS INDICATED BY** radiological control technician/chemist report of a primary coolant sample greater than 2.60E+3 μCi/mL and less than or equal to 1.67E+4 μCi/mL, **AND** loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout), **AND** radiation area monitor alarm in TRA-605, **OR** radiological control technician radiation survey in TRA-605 higher than normal background, **OR** other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.	Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,200 m (26,900 ft or 5.1 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Consider authorizing potassium iodide for essential emergency workers.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-605-3.SAE.4	Radiological release resulting from transferring contaminated (equal to 8.19E+3 μCi/mL) primary coolant system water to TRA-605 tanks, **AS INDICATED BY** radiological control technician/chemist report of a primary coolant sample equal to 8.19E+3 μCi/mL, **AND** Materials Test Reactor stack is operational (as indicated by the Materials Test Reactor building local readout), **AND** radiation area monitor alarm in TRA-605, **OR** radiological control technician radiation survey in TRA-605 higher than normal background, **OR** other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **2,900 m (9,515 ft or 1.8 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609. Consider authorizing potassium iodide for essential emergency workers.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-605-3.SAE.5	Radiological release resulting from transferring contaminated (greater than 8.19E+3 μCi/mL and less than or equal to 2.54E+4 μCi/mL) primary coolant system water to TRA-605 tanks, **AS INDICATED BY** radiological control technician/chemist report of a primary coolant sample greater than 8.19E+3 μCi/mL and less than or equal to 2.54E+4 μCi/mL, **AND** Materials Test Reactor stack is operational (as indicated by the Materials Test Reactor building local readout), **AND** radiation area monitor alarm in TRA-605, **OR** radiological control technician radiation survey in TRA-605 higher than normal background, **OR** other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.	Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **6,400 m (20,998 ft or 4 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Consider authorizing potassium iodide for essential emergency workers.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-605-3.GE.1	Radiological release resulting from transferring contaminated (greater than 2.54E+4 μCi/mL) primary coolant system water to TRA-605 tanks, **AS INDICATED BY** radiological control technician/chemist report of a primary coolant sample greater than 2.54E+4 μCi/mL, **AND** loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout), **AND** radiation area monitor alarm in TRA-605, **OR** radiological control technician radiation survey in TRA-605 higher than normal background, **OR** other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. If wind is from **165 to 255** degrees, consider sheltering Specific Manufacturing Capability/Test Area North personnel. If wind is from **225 to 315** degrees, consider sheltering Materials and Fuels Complex personnel. Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.	Control access on all public roadways entering the Idaho National Laboratory Site. Advise state and county representatives that offsite protective actions may be necessary depending on wind direction and speed. **NOTE:** *Sheltering or evacuation should be considered for those individuals working on property immediately adjacent to the Idaho National Laboratory Site northwest boundary **(12,000 m or 7.5 mi)**.* If wind is from **95 to 170** degrees, make protective action recommendation to Butte County to shelter or evacuate anyone working near the northwest Idaho National Laboratory Site boundary between the junction of State Highway 33 and United States Highway 20/26 and Mile Marker 12 on State Highway 33. **NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-621-3.A.1	Any unplanned criticality resulting in a radioactive material release, **AS INDICATED BY** activation of criticality alarm system RM-50-1 with a voice message from the autodialer stating "criticality alarm," **AND** security camera operator confirmation of flooding and/or fuel disruption from containers, **OR** TRA-621 personnel confirmation of conditions to cause a criticality, **OR** radiological control technician confirmation by radiation survey.	Evacuate nonessential Advanced Test Reactor Complex personnel. Control nonessential vehicle and personnel access to the Advanced Test Reactor Complex.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-621-3.SAE.1	Fire inside TRA-621 that has <u>**not**</u> been extinguished in 20 minutes after Idaho National Laboratory Fire Department fire suppression activities began, causing a release of radioactive material, **AS INDICATED BY** visual observation of the ongoing fire 20 minutes after fire suppression began, **OR** radiological control technician survey indicates an airborne radiological release.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **310 m (1,017 ft or 0.2 mi)** of the Advanced Test Reactor Complex. Control nonessential vehicle and personnel access to the evacuated area. Depending on wind direction, consider relocating the emergency control center to the alternate emergency control center at CF-609.	None.
ATR-621-10.OE.1	Any unplanned criticality, **AS INDICATED BY** activation of criticality alarm system RM-50-1 with a voice message from the autodialer stating "criticality alarm," **AND** security camera operator confirmation of flooding and/or fuel disruption from containers, **OR** TRA-621 personnel confirmation of conditions to cause criticality, **OR** radiological control technician confirmation by radiation survey.	Evacuate nonessential personnel from the Advanced Test Reactor Complex. Control nonessential vehicle and personnel access to the Advanced Test Reactor Complex.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-634-3.A.1	Radiological release from unsuppressed fire, lasting greater than 10 minutes and less than 20 minutes after fire suppression activities began, on the TRA-634 loading area that involves an irradiated test train in its storage cask, **AS INDICATED BY** visual confirmation of the fire or activation of the fire alarm/sprinkler system, **AND** radiological control technician confirmation by radiation survey outside TRA-634.	Establish a **100-m (328-ft)** exclusion zone, or other distance recommended by the incident commander, around TRA-634. Relocate nonessential personnel from the exclusion zone to the cafeteria assembly area in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone.	None.
ATR-634-3.SAE.1	Radiological release from unsuppressed fire, lasting 20 minutes after fire suppression activities began, on the TRA-634 loading area that involves an irradiated test train in its storage cask, **AS INDICATED BY** visual confirmation of the fire or activation of the fire alarm/sprinkler system, **AND** radiological control technician confirmation by radiation survey outside TRA-634.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **3,000 m (9,840 ft or 1.9 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.A.1 (Advanced Test Reactor)	Elevated primary coolant system activity that may result in a radiological release, **AS INDICATED BY** radiological control technician/chemist report of a primary coolant system sample greater than or equal to 20 μCi/mL.	Establish a **100-m (328-ft)** exclusion zone around TRA-670. Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.
ATR-670-3.A.2 (Advanced Test Reactor Critical Facility)	Radiological release from a fuel handling accident (control bridge drop onto Advanced Test Reactor Critical Facility core), **AS INDICATED BY** visual confirmation of the control bridge drop, **AND** remote area monitor alarm.	Establish a **100-m (328-ft)** exclusion zone around TRA-670. Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.A.3 (Advanced Test Reactor Critical Facility)	Any unplanned criticality resulting in a radiological release, **AS INDICATED BY** remote area monitor alarm while moving fissile material on the Advanced Test Reactor Critical Facility floor, **OR** log count-rate alarms while moving fuel into or out of the Advanced Test Reactor Critical Facility core, **AND** dose rate readings on radiological control technician-handheld instruments significantly higher than normal background.	Establish a **100-m (328-ft)** exclusion zone around TRA-670. Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.SAE.1 (Advanced Test Reactor)	Experiment loop loss-of-coolant accident with a radiological release from a less than or equal to 200-kW fueled experiment, **AS INDICATED BY** loop instrumentation indicating low-low pressurizer level and low-low pressure alarm, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm.	Evacuate nonessential personnel at least **2,190 m (7,185 ft or 1.4 mi)** in all directions from the Advanced Test Reactor Complex. Control nonessential vehicle and personnel access to the evacuated area. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609. Consider authorizing potassium iodide for essential emergency workers. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.SAE.2 (Advanced Test Reactor)	Radiological release from a fuel damage event, **AS INDICATED BY** radiological control technician/chemist report of increased primary coolant activity levels greater than 20 μCi/mL following a power variation greater than 3%, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **2,830 m (9,285 ft or 1.8 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609. Consider authorizing potassium iodide for essential emergency workers. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.SAE.3 (Advanced Test Reactor)	Radiological release from dropping highly radioactive components from a cask or problems occurring during an open air transfer of highly radioactive components that preclude immediately returning the components to the reactor vessel or canal, **AS INDICATED BY** direct observation of the event, **AND** multiple continuous air monitor and remote area monitor alarms, **OR** radiological control technician confirmation by radiation survey.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **1,000 m (3,281 ft)** of the Advanced Test Reactor Complex. Control nonessential vehicle and personnel access to the evacuated area. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.
ATR-670-3.SAE.4 (Advanced Test Reactor)	Radiological release from an unshielded dropped loop experiment accident, **AS INDICATED BY** direct observation of the event, **AND** multiple continuous air monitor and remote area monitor alarms in the reactor control room, **OR** radiological control technician confirmation by radiation survey.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **1,000 m (3,281 ft)** of the Advanced Test Reactor Complex. Control nonessential vehicle and personnel access to the evacuated area. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.SAE.5 (Advanced Test Reactor)	Experiment loop loss-of-coolant accident with a radiological release from a greater than 200-kW and less than or equal to 1-MW fueled experiment, **AS INDICATED BY** loop instrumentation indicating low-low pressurizer level and low-low pressure alarm, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **OR** radiological control technician confirmation of the release by radiation survey, **AND** release ongoing **less than or equal to 4 hours.** **RELATED INFORMATION:** This event is a precursor to emergency action level ATR-670-3.GE.3.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **4,000 m (13,125 ft or 2.5 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Consider authorizing potassium iodide for essential emergency workers. Relocate the emergency control center to the alternate emergency control center at CF-609. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.SAE.6 (Advanced Test Reactor)	Radiological release from a fuel melt caused by an experimental reactivity insertion event, **AS INDICATED BY** loop instrumentation indicating low-low pressurizer level and low-low pressure alarm, **AND** fission break monitor system alarms, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **OR** radiological control technician confirmation of the release by radiation survey, **AND** release ongoing **less than or equal to 4 hours**. **RELATED INFORMATION:** This event is a precursor to emergency action level ATR-670-3.GE.2.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **3,000 m (9,840 ft or 1.9 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Consider authorizing potassium iodide for essential emergency workers. Relocate the emergency control center to the alternate emergency control center at CF-609. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.SAE.7 (Advanced Test Reactor Critical Facility)	Multiple system failures at maximum operating power leads to an Advanced Test Reactor Critical Facility fuel melt and a radiological release, **AS INDICATED BY** major reactivity control system failure, **AND** reactor safety system failure, **AND** safety rod drive system failure to insert safety rods, **AND** remote area monitor alarm.	Evacuate nonessential personnel at least **1,710 m (5,611 ft or 1.1 mi)** in all directions from the Advanced Test Reactor Complex. Control nonessential vehicle and personnel access to the evacuated area. Consider authorizing potassium iodide for essential emergency workers. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.SAE.8 (Advanced Test Reactor)	Radiological release from an experiment fuel melt caused by a cask drop from above lift limits that damages the reactor cooling system, **AS INDICATED BY** visual observation of the cask drop, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **OR** radiological control technician confirmation of the release by radiation survey.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **1,450 m (4,757 ft or 0.9 mi)** of TRA-670. **NOTE:** *If wind persists in the same direction longer than 4 hours, evacuation distance should be extended to* ***2,650 m (8,695 ft or 1.6 mi).*** Control nonessential vehicle and personnel access to the evacuated area. Consider authorizing potassium iodide for essential emergency workers. Relocate the emergency control center to the alternate emergency control center at CF-609. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.GE.1 (Advanced Test Reactor)	Radiological release from a canal drain event, **AS INDICATED BY** canal water level alarm, **OR** direct observation of the canal water level approaching or below the 13-ft level, **AND** firewater system or other makeup water sources **not** stabilizing canal water level, **AND** multiple continuous air monitor and remote area monitor alarms, **AND** radiological control technician confirmation of the release of airborne radioactive material beta-gamma emitters outside TRA-670.	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel. If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel. Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations. (continued next page)	Control access on all public roadways entering the Idaho National Laboratory Site. If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco. If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe. If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton. If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City. **NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.GE.1 (Advanced Test Reactor) (continued)		Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	
ATR-670-3.GE.2 (Advanced Test Reactor)	Radiological release from a fuel melt caused by an experimental reactivity insertion event, **AS INDICATED BY** loop instrumentation indicating low-low pressurizer level and low-low pressure alarm, **AND** fission break monitor system alarms, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **OR** radiological control technician confirmation of the release by radiation survey, **AND** release ongoing **longer than 4 hours**.	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. If wind is from **165 to 255** degrees, consider sheltering Specific Manufacturing Capability/Test Area North personnel. If wind is from **225 to 315** degrees, consider sheltering Materials and Fuels Complex personnel. Control nonessential vehicle and personnel access to the evacuated areas. (continued next page)	Control access on all public roadways entering the Idaho National Laboratory Site. Advise state and county representatives that offsite protective actions may be necessary depending on wind direction and speed. **NOTE:** *Sheltering or evacuation should be considered for those individuals working on property immediately adjacent to the Idaho National Laboratory Site northwest boundary **(12,000 m or 7.5 mi)**.* If wind is from **95 to 170** degrees, make protective action recommendation to Butte County to shelter or evacuate anyone working near the northwest Idaho National Laboratory Site boundary between the junction of State Highway 33 and United States Highway 20/26 and Mile Marker 12 on State Highway 33. (continued next page)

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.GE.2 (Advanced Test Reactor) (continued)		Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*
ATR-670-3.GE.3 (Advanced Test Reactor)	Experiment loop loss-of-coolant accident with a radiological release from a greater than 200-kW and less than or equal to 1-MW fueled experiment, **AS INDICATED BY** loop instrumentation indicating low-low pressurizer level and low-low pressure alarm, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **OR** radiological control technician confirmation of the release by radiation survey, (continued next page)	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. If wind is from **165 to 255** degrees, consider sheltering Specific Manufacturing (continued next page)	Control access on all public roadways entering the Idaho National Laboratory Site. Advise state and county representatives that offsite protective actions may be necessary depending on wind direction and speed. **NOTE:** *Sheltering or evacuation should be considered for those individuals working on property immediately adjacent to the Idaho National Laboratory Site northwest or southern boundaries* ***(17,000 m or 10.6 mi)****.* If wind is from **95 to 170** degrees, make protective action recommendation to Butte (continued next page)

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-3.GE.3 (Advanced Test Reactor) (continued)	**AND** release ongoing **longer than 4 hours**.	Capability/Test Area North personnel. If wind is from **225 to 315** degrees, consider sheltering Materials and Fuels Complex personnel. Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	County to shelter or evacuate anyone working near the northwest Idaho National Laboratory Site boundary between the junction of State Highway 33 and United States Highway 20/26 and Mile Marker 12 on State Highway 33. If wind is from **335 to 25** degrees, make protective action recommendation to Bingham County to shelter or evacuate anyone working near the southern Idaho National Laboratory Site boundary from 4 to 15 mi west of Atomic City. **NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.A.1 (Advanced Test Reactor)	Loss-of-coolant flow with potential for core damage, **AS INDICATED BY** reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow), **AND** flow **not** restored within 30 minutes after reactor shutdown, **OR** following initial 30 minutes of flow after reactor shutdown, flow lost and **not** restored, **AND** reactor feed and bleed **not** demonstrated operable within 6 hours after reactor shutdown. **RELATED INFORMATION:** This event could be a precursor to emergency action levels ATR-670-7.SAE.1 and ATR-670-7.GE.3 or ATR-670-7.SAE.4 and ATR-670-7.GE.2.	Establish a **100-m (328-ft)** exclusion zone around TRA-670. Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.A.2 (Advanced Test Reactor)	Loss-of-coolant accident with failure of the emergency firewater injection system, **AS INDICATED BY** primary coolant leak rate greater than 300 gpm, **AS INDICATED BY** flow balance [pressurizing flow FIC-1-8B or -8A to PCV-1-1 flow (PCS F• DEGAS)], **AND** emergency firewater injection system pressure less than 58 psig, **AS INDICATED BY** PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure). **RELATED INFORMATION:** This event could be a precursor to emergency action levels ATR-670-7.SAE.2 and ATR-670-7.GE.1 or ATR-670-7.SAE.3.	Establish a **100-m (328-ft)** exclusion zone around TRA-670. Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.SAE.1 (Advanced Test Reactor)	Loss-of-coolant flow with potential for core damage and a radiological release, **AS INDICATED BY** reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow), **AND** flow **<u>not</u>** restored within 30 minutes after reactor shutdown, **OR** following initial 30 minutes of flow after reactor shutdown, flow lost and **<u>not</u>** restored, **AND** reactor feed and bleed **<u>not</u>** demonstrated operable within 6 hours after reactor shutdown, **AND** primary coolant system **<u>not</u>** vented to atmosphere, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, (continued next page)	Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **9,000 m (29,530 ft or 5.6 mi)** in the downwind direction. Consider evacuating the Critical Infrastructure Test Range Complex if it is in the downwind direction. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate. Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction. Control nonessential vehicle and personnel access to the evacuated areas. Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I. Consider authorizing potassium iodide for essential emergency workers. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. (continued next page)	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.SAE.1 (Advanced Test Reactor) (continued)	**AND** release ongoing **less than or equal to 4 hours.** **RELATED INFORMATION:** This event is a precursor to emergency action level ATR-670-7.GE.3.	Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	
ATR-670-7.SAE.2 (Advanced Test Reactor)	Loss-of-coolant accident bounds both small break and large break loss-of-coolant accidents with failure of emergency firewater injection system that could result in core damage and fuel melting with consequent radiological release, **AS INDICATED BY** primary coolant leak rate less than 300 gpm, **AS INDICATED BY** flow balance [pressurizing flow FIC-1-8B or -8A to PCV-1-1 flow (PCS F• DEGAS)], **AND** emergency firewater injection system pressure less than 58 psig, **AS INDICATED BY** PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure), (continued next page)	Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **10,850 m (35,599 ft or 6.75 mi)** in the downwind direction. Consider evacuating the Critical Infrastructure Test Range Complex if it is in the downwind direction. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate. Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction. Control nonessential vehicle and personnel access to the evacuated areas. (continued next page)	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.SAE.2 (Advanced Test Reactor) (continued)	**AND** vessel coolant level below the emergency firewater injection system injection level and decreasing, **AS INDICATED BY** LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing, **AND** multiple continuous air monitor and remote area monitor alarms and/or stock monitor alarm, **AND** release ongoing **less than or equal to 4 hours.** **RELATED INFORMATION:** This event is a precursor to emergency action level ATR-670-7.GE.1.	Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I.Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Consider authorizing potassium iodide for essential emergency workers. Verify implementation of AOP-0.1, “Operator Evacuation Procedure,” as ensured by the shift supervisor/emergency action manager, as appropriate.	

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.SAE.3 (Advanced Test Reactor)	Radiological release from an interfacing system loss-of-coolant accident, **AS INDICATED BY** primary coolant leak rate greater than 300 gpm, **AS INDICATED BY** flow balance [pressurizing flow FIC-1-8B or -8A to PCV-1-1 flow (PCS F• DEGAS)], **AND** emergency firewater injection system pressure less than 58 psig, **AS INDICATED BY** PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure), **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm.	Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **5,300 m (17,389 ft or 3.3 mi)** of the Advanced Test Reactor Complex. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate. Control nonessential vehicle and personnel access to the evacuated areas. Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609. Consider authorizing potassium iodide for essential emergency workers. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.SAE.4 (Advanced Test Reactor)	Radiological release from a low-pressure boil-off event, **AS INDICATED BY** reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow), **AND** flow <u>**not**</u> restored within 30 minutes after reactor shutdown, **AND** emergency firewater injection system pressure less than 58 psig, **AS INDICATED BY** PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure), **AND** primary coolant system vented to atmosphere, **AND** vessel coolant level below the emergency firewater injection system injection level and decreasing, **AS INDICATED BY** LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing, (continued next page)	Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **9,000 m (29,530 ft or 5.6 mi)** in the downwind direction. Consider evacuating the Critical Infrastructure Test Range Complex and Power Burst Facility if they are in the downwind direction. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate. Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction. Control nonessential vehicle and personnel access to the evacuated areas. Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I. Consider authorizing potassium iodide for essential emergency workers. (continued next page)	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.SAE.4 (Advanced Test Reactor) (continued)	**AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **AND** release ongoing **less than or equal to 4 hours**. **RELATED INFORMATION:** This event is a precursor to emergency action level ATR-670-7.GE.2.	Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	
ATR-670-7.GE.1 (Advanced Test Reactor)	Loss-of-coolant accident bounds both small break and large break loss-of-coolant accidents with failure of the emergency firewater injection system that could result in core damage and fuel melting with a consequent radiological release, **AS INDICATED BY** primary coolant leak rate greater than 300 gpm, **AS INDICATED BY** flow balance [pressurizing flow FIC-1-8B or -8A to PCV-1-1 flow (PCS F• DEGAS)], **AND** emergency firewater injection system pressure less than 58 psig, (continued next page)	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel. (continued next page)	Control access on all public roadways entering the Idaho National Laboratory Site. If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco. If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe. If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton. If wind is from **270 to 360** degrees, make protective action recommendation to (continued next page)

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.GE.1 (Advanced Test Reactor) (continued)	**AS INDICATED BY** PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure), **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **AND** vessel coolant level below the emergency firewater injection system injection level and decreasing, **AS INDICATED BY** LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing, **AND** release ongoing **longer than 4 hours**.	If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel. Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	Bingham County to shelter and prepare to evacuate Atomic City. **NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*
ATR-670-7.GE.2 (Advanced Test Reactor)	Radiological release from a low-pressure boil-off event, **AS INDICATED BY** reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow), **AND** flow <u>**not**</u> restored within 30 minutes after reactor shutdown, (continued next page)	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho (continued next page)	Control access on all public roadways entering the Idaho National Laboratory Site. If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco. If wind is from **135 to 225** degrees, make protective action recommendation to Butte (continued next page)

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.GE.2 (Advanced Test Reactor) (continued)	**AND** emergency firewater injection system pressure less than 58 psig, **AS INDICATED BY** PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure), **AND** primary coolant system vented to atmosphere, **AND** vessel coolant level below the emergency firewater injection system injection level and decreasing, **AS INDICATED BY** LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **AND** release ongoing **longer than 4 hours**.	Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel. If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel. Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	County to shelter and prepare to evacuate Howe. If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton. If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City. **NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.GE.3 (Advanced Test Reactor)	Radiological release from a high-pressure boil-off event, **AS INDICATED BY** reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow), **AND** flow **not** restored within 30 minutes after reactor shutdown, **OR** following initial 30 minutes of flow after reactor shutdown, flow lost and **not** restored, **AND** primary coolant system **not** vented to atmosphere, **AND** vessel coolant level below the emergency firewater injection system injection level and decreasing, **AS INDICATED BY** LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing, (continued next page)	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel. If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel. Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the (continued next page)	Control access on all public roadways entering the Idaho National Laboratory Site. If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco. If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe. If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton. If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City. **NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-7.GE.3 (Advanced Test Reactor) (continued)	**AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **AND** release ongoing **longer than 4 hours**.	Advanced Test Reactor Complex and other downwind locations. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	
ATR-670-8.A.1 (Advanced Test Reactor)	Reactor **not** subcritical after completion of FRP-S.1, "Uncontrolled Power Generation/ATWS" (function restoration procedure subcriticality 1). **RELATED INFORMATION:** This event could be a precursor to emergency action levels ATR-670-8.SAE.1 and ATR-670-8.GE.1 or ATR-670-8.SAE.2 and ATR-670-8.GE.2.	Establish a **100-m (328-ft)** exclusion zone around TRA-670. Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-8.SAE.1 (Advanced Test Reactor)	Radiological release from an anticipated transient without scram event, **AS INDICATED BY** • Power greater than N_L • Period — positive, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **AND** release ongoing **less than or equal to 4 hours.** **RELATED INFORMATION:** This event is a precursor to emergency action level ATR-670-8.GE.1.	Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **9,000 m (29,530 ft or 5.6 mi)** in the downwind direction. Consider evacuating the Critical Infrastructure Test Range Complex if it is in the downwind direction. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate. Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction. Control nonessential vehicle and personnel access to the evacuated areas. Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I. Consider authorizing potassium iodide for essential emergency workers. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. (continued next page)	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-8.SAE.1 (Advanced Test Reactor) (continued)		Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	
ATR-670-8.SAE.2 (Advanced Test Reactor)	Radiological event from a large reactivity insertion event, **AS INDICATED BY** control room observation of a reactor scram with indication of a power increase, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **AND** release ongoing **less than or equal to 4 hours**. **RELATED INFORMATION:** This event is a precursor to emergency action level ATR-670-8.GE.2. It is similar to emergency action level ATR-670-7.SAE.2.	Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **10,850 m (35,599 ft or 6.75 mi)** in the downwind direction. Consider evacuating the Critical Infrastructure Test Range Complex if it is in the downwind direction. Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate. Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction. Control nonessential vehicle and personnel access to the evacuated areas. Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I. (continued next page)	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-8.SAE.2 (Advanced Test Reactor) (continued)		Consider authorizing potassium iodide for essential emergency workers. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	
ATR-670-8.GE.1 (Advanced Test Reactor)	Radiological release from an anticipated transient without scram event, **AS INDICATED BY** • Power greater than NL • Period — positive, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **AND** release ongoing **longer than 4 hours**.	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel. If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel. (continued next page)	Control access on all public roadways entering the Idaho National Laboratory Site. If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco. If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe. If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton. If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City. (continued next page)

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-8.GE.1 (Advanced Test Reactor) (continued)		Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, “Operator Evacuation Procedure,” as ensured by the shift supervisor/emergency action manager, as appropriate.	**NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*
ATR-670-8.GE.2 (Advanced Test Reactor)	Radiological event from a large reactivity insertion event, **AS INDICATED BY** control room observation of a reactor scram with indication of a power increase, **AND** multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm, **AND** release ongoing **longer than 4 hours**.	Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary. Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate. (continued next page)	Control access on all public roadways entering the Idaho National Laboratory Site. If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco. If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe. If wind is from **180 to 270** degrees, make protective action recommendation to (continued next page)

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-8.GE.2 (Advanced Test Reactor) (continued)		If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel. If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel. Control nonessential vehicle and personnel access to the evacuated areas. Control access on all public roadways entering the Idaho National Laboratory Site. Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations. Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton. If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City. **NOTE:** *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-10.OE.1 (Advanced Test Reactor Critical Facility)	Any unplanned criticality, **AS INDICATED BY** remote area monitor alarm while moving fissile material on the Advanced Test Reactor Critical Facility floor, **OR** log count-rate alarms while moving fuel into or out of the Advanced Test Reactor Critical Facility core, **AND** dose rate readings on radiological control technician-handheld instruments significantly higher than normal background. **RELATED INFORMATION:** This event is a precursor to emergency action level ATR-670-3.A.3.	Establish a **100-m (328-ft)** exclusion zone around TRA-670. Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone. Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.	None.

Table 8. (continued).

Facility Emergency Action Levels	Initiating Event/Condition	Onsite Protective Action	Offsite Protective Action Recommendations
ATR-670-11.A.1 (Advanced Test Reactor)	Potential for an Advanced Test Reactor canal drain event, **AS INDICATED BY** canal water level alarm, **OR** visual observation of the canal water level decreasing faster than the emergency fire water injection system makeup rate. **RELATED INFORMATION:** This event could be a precursor to emergency action level ATR-670-3.GE.1.	Establish a **100-m (328-ft)** exclusion zone, or other distance recommended by the incident commander, around TRA-670. Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616. Control nonessential vehicle and personnel access to the exclusion zone.	None.
ATR-683-4.A.1	A waste storage drum containing hydrofluoric acid is breached by puncture, crushing, or drum degradation causing a spill of hydrofluoric acid, **AS INDICATED BY** visual observation of the drum breach, **AND** visual observation of a liquid spill on the floor of TRA-683 or adjacent area.	Establish a **109-m (358-ft)** exclusion zone around TRA-683. Relocate nonessential personnel from the exclusion zone to the conference room in TRA-652. Control nonessential vehicle and personnel access to the exclusion zone.	None.

9. MAINTENANCE AND REVIEW OF THIS HAZARDS ASSESSMENT

This EHA is maintained in accordance with PLN-114[10] and GDE-438.[11]

10. REFERENCES

1. DOE O 151.1C, "Comprehensive Emergency Management System," United States Department of Energy, November 2, 2005.
2. DOE G 151.1-2, "Technical Planning Basis," United States Department of Energy, July 11, 2007.
3. EHS-50, "Emergency Management Hazards Survey for the Advanced Test Reactor Complex," Rev. 0, October 22, 2008.
4. SAR-400, "INL Standardized Safety Analysis Report (SAR)," Chapter 1, Site Characteristics, Rev. 1, October 1, 2008.
5. Radiological Safety Analysis Computer Program (RSAC-6), Version 6.2, INEEL, 2002.
6. Emergency Prediction Information Code, Version 7.0, Homann and Associates, Inc., 2006.
7. Areal Locations of Hazardous Atmospheres, Version 5.4, Office of Response and Restoration, National Ocean Service, National Oceanic and Atmospheric Administration (NOAA), February 2006.
8. Eckman, R. M., 2003, "A Statistical Investigation of Atmospheric Dispersion at the Idaho National Engineering and Environmental Laboratory (INEEL)," NOAA Technical Memorandum OAR ARL-246, NOAA Air Resources Laboratory, Silver Springs, MD, February 2003.
9. Salmonson, B. J., to W. S. Schum, May 25, 2004, "Derived χ/Qs for 50% Meteorological Conditions," BJS-02-04.
10. PLN-114, "INL Emergency Plan/RCRA Contingency Plan."
11. GDE-438, "Developing and Maintaining Emergency Management Hazards Assessments," Rev. 4, September 19, 2008.

11. APPENDICES

Appendix A, Emergency Management Hazards Assessment for TRA-605, Effluent Processing Facility

Appendix B, Emergency Management Hazards Assessment for TRA-621, Nuclear Materials Inspection and Storage Facility

Appendix C, Emergency Management Hazards Assessment for TRA-634, Advanced Test Reactor Storage Facility

Appendix D, Emergency Management Hazards Assessment for TRA-670, Advanced Test Reactor Building

Appendix E, Emergency Management Hazards Assessment for TRA-671, Cooling Tower Pumphouse

Appendix F, Emergency Management Hazards Assessment for TRA-780, 90-Day Storage Area

Appendix G, Emergency Management Hazards Assessment for Advanced Test Reactor Complex Onsite Transportation

Appendix H, Emergency Management Hazards Assessment for Advanced Test Reactor Complex in General

APPENDIX A, EMERGENCY MANAGEMENT HAZARDS ASSESSMENT FOR TRA-605, EFFLUENT PROCESSING FACILITY	Identifier: EHA-50 Revision: 0 Effective Date: 01/13/10 Page: A-1 of A-56

Emergency Management	Emergency Management Hazards Assessment		eCR Number:	568083
Manual: Hazards Assessment — Advanced Test Reactor (ATR) Complex				

REVISION LOG

Rev.	Date	Affected Pages	Revision Description
1	01/13/2010	All	New Document

CONTENTS

FIGURE

TABLES

A-1. TRA-605, EFFLUENT PROCESSING FACILITY

A-1.1 Assessment Overview

This appendix documents the emergency management hazards assessment (EHA) for TRA-605, Effluent Processing Facility, to provide the technical basis for facility emergency planning efforts, as required by DOE O 151.1C[1] and supported by DOE G 151.1-2.[2]

A-1.2 Facility and Process Descriptions

TRA-605 is located south of Marlin Avenue, east of Pike Street, and north of Star Fish Avenue in the southeast part of the Advanced Test Reactor (ATR) Complex as shown in Figure A-1. The location of TRA-710, Materials Test Reactor (MTR) Stack, is also shown in Figure A-1.

TRA-605 is a one-story (part basement and part two-story in the east portion) building. It was built in 1952 and is a total of 22,235 ft^2. The roof is composition-on-steel beams; the walls and floors are concrete. TRA-605 is operated by Battelle Energy Alliance, LLC.

The TRA-605 mission is to collect, store, process, and dispose of radioactive-contaminated liquid waste streams generated in the ATR. TRA-605 reduces radioactivity levels in the ATR Complex warm liquid waste by removing radioactive ions by ion exchange mechanisms. Other purposes of TRA-605 include monitoring warm waste released to the evaporation pond and collecting and storing hot liquid waste before transporting the waste for treatment and disposal. The significant systems that interface with TRA-605 are the ATR warm liquid waste transfer/treatment system and ATR hot liquid waste transfer system.[3]

TRA-605 houses the Warm Waste Treatment Facility (WWTF), an effluent radiation monitoring system, a hot waste storage tank (HWST) (TRA-605), and the warm waste discharge pumps. The activities in TRA-605 include processing, sampling, monitoring, transferring, and storing radiologically-contaminated liquid.[4]

The purpose of the WWTF is to process radiologically-contaminated liquid waste. WWTF can be operated automatically while in the recirculation mode or manually while in the batch mode. WWTF is designed to process waste at a maximum flow rate of 150 gpm at 120°F and 100 psi. The nominal flow rate is 20 gpm. In the recirculation mode, a portion of the flow is diverted back to the 17,000-gal warm waste feed tank (WWFT). In the batch mode, waste flow through the WWTF is not routed back to the WWFT. In the batch mode, WWTF will shut down automatically when the WWFT reaches 10% of its capacity, and typically is manually restarted when the WWFT level reaches 50% or more of its capacity. When the WWFT reaches 90% of its capacity, waste from the WWFT will overflow into the 100,000-gal TRA-605 HWST. The WWFT receives waste from the ATR warm waste transfer pipeline, TRA-605 building sump pit, and TRA-605 WWTF recirculation system. The TRA-605 WWFT and TRA-605 HWST are vented to the MTR stack exhaust system.

APPENDIX A, EMERGENCY MANAGEMENT HAZARDS ASSESSMENT FOR TRA-605, EFFLUENT PROCESSING FACILITY

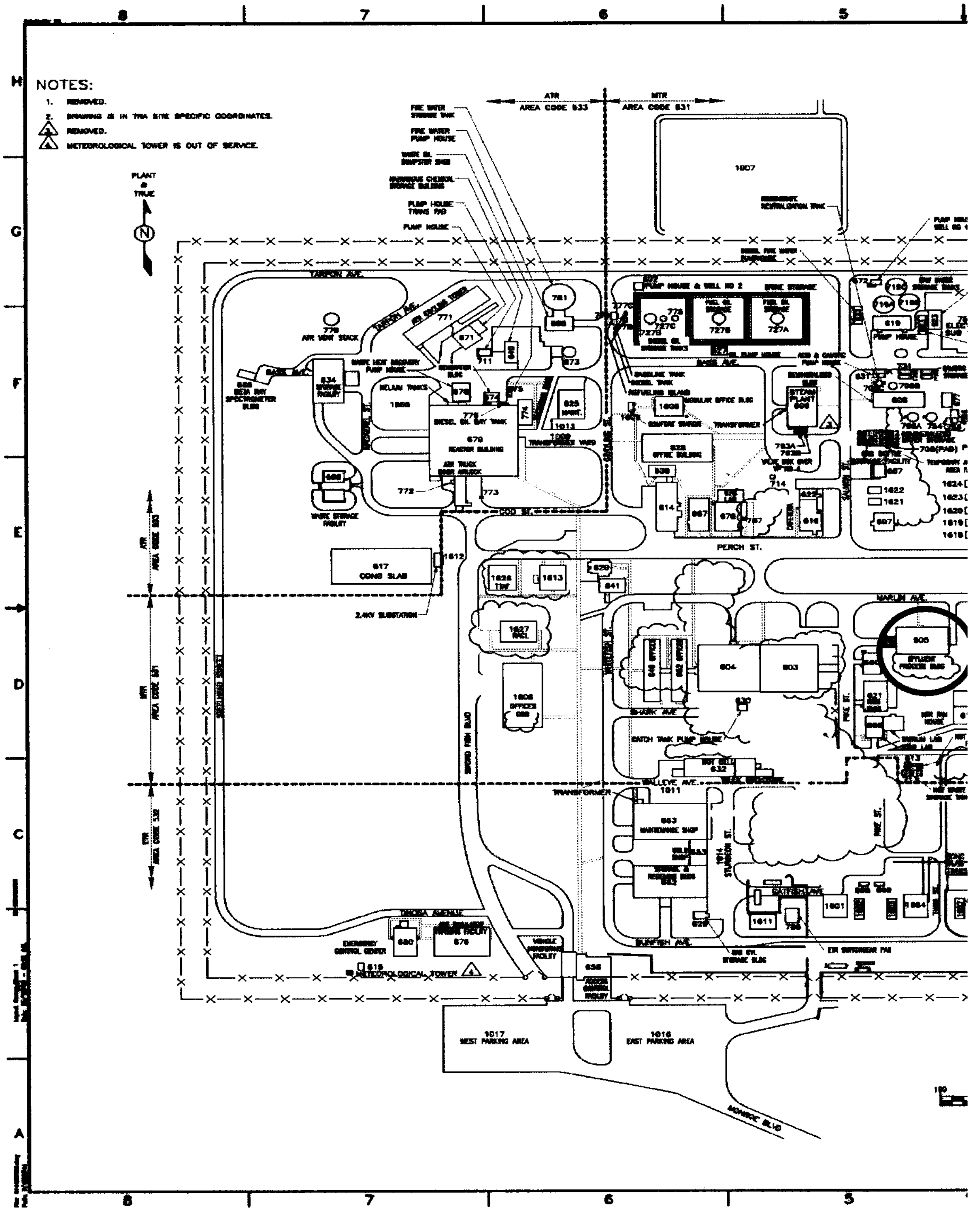
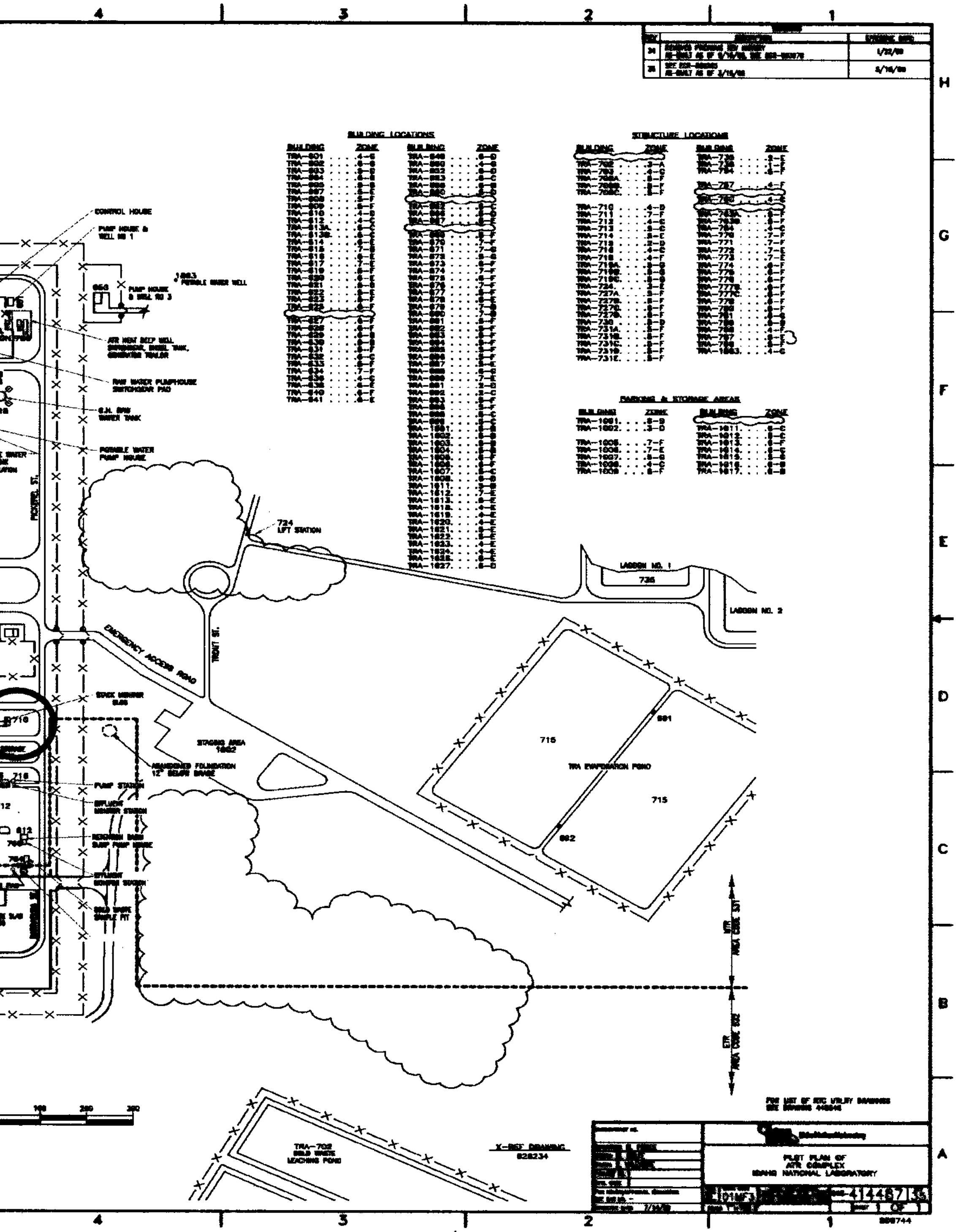


Figure A-1. Map of Advanced Test Reactor Complex showing locations of TRA-605, Effluent Processing Facility, and TF



-710, Materials Test Reactor Stack.

The TRA-605 WWTF contains the following components, which are not discussed in detail in this EHA, because the WWTF and HWST are considered the release points for radionuclide consequence assessment:

- Warm waste feed pumps/piping
- Filters/strainers
- Ion exchange resin beds
- Valve nest.

A-1.3 Identification of Hazards

No nonradiological hazardous material was identified in TRA-605. Although TRA-605 has been identified as a less than Hazard Category (HC) 3 radiological facility,[5] there is a possibility for larger quantities of radiological hazardous material in TRA-605 if the ATR primary coolant reached technical specification levels (20 µCi/mL) and had to be pumped to TRA-605 for treatment. With the abandonment of the TRA-689 radioactive waste system upgrade, any radioactive liquid waste from the ATR, including contaminated primary coolant system (PCS) liquids, will be directed to TRA-605.[4]

Table A-1 lists the radiological hazardous material that is stored, used, or produced in TRA-605 that is retained for further analyses based on the screening criteria presented in the main document.

Table A-1. Radiological material requiring further analyses.

Location	Material	Maximum Quantity (Ci)	Threshold Quantity (Ci)	Notes
TRA-605 (All)	Fresh fission products	Greater than radiological HC-3; sum of ratios is 8.97E+1	Radiological HC-3	ATR primary coolant that exceeds technical specification concentration limits may be sent to TRA-605 for processing

A-1.4 Hazardous Material Characterization and Analyses

The hazardous material listed in Table A-1 is addressed below by location.

A-1.4.1 TRA-605 (All)

A-1.4.1.1 Radiological Hazardous Material — Fresh Fission Products

The purpose of this evaluation is to identify the ATR PCS concentration, which would trigger a classifiable event if PCS liquids were transferred to TRA-605. Current ATR procedures call for pumping of PCS water to TRA-605 if the PCS concentrations exceed the technical specification concentration level of 20 µCi/mL.

The material-at-risk (MAR) is the respirable radioactive material that may become airborne while filling the TRA-605 effluent storage tank (HWST) and WWFT with contaminated primary coolant. Subsection A-1.4.1.1.1 provides an overview of the properties for the radiological hazardous material that may be found in TRA-605 when the ATR primary coolant is being processed through TRA-605.

A-1.4.1.1.1 Properties. The properties for the TRA-605 radiological material are listed in Table A-2.

Table A-2. Radiological properties for Advanced Test Reactor contaminated primary coolant transferred to TRA-605.

Radioactive mixed fission products	Fresh fission products contained in ATR primary coolant that may exceed the ATR primary coolant chemistry limit of 20 μCi/mL.
Physical form	Liquid containing particulate and gaseous radionuclides.
Particle size	Respirable.
Flammability	Not applicable (N/A).
Reactivity	N/A.
Density	N/A.
Special firefighting concerns	None.
Health concerns	Inhalation is the primary route for personnel exposure. Thyroid committed dose equivalent (CDE) is controlling for Protective Action Guide (PAG) evaluation, due to the higher concentration of radioactive iodine isotopes present in the fresh fission products and the fact that the thyroid gland concentrates iodine. Fission products are primarily beta-gamma emitters.

A-1.4.1.1.2 Conditions of Storage and Use. The primary coolant is used to cool the ATR core to keep the fuel elements from melting. The ATR core and associated piping store the coolant until conditions warrant ion exchange treatment of the coolant to remove contaminants.

Engineering Design File EDF-6873,[6] Table A-6, lists an RSAC file, pcsst, that provides the MAR based on a primary coolant concentration of 20 μCi/mL and volume of 153,363 gal. Table A-3 identifies the radionuclides that contribute 99% of dose and their activities that make up the MAR. The actual ATR primary coolant volume is 81,000 gal, so the MAR in Table A-3 must be adjusted accordingly for use in source term (ST) development. For example, to determine the MAR for a 20-μCi/mL concentration and a volume of approximately 81,000 gal, the Table A-3 values must be multiplied by the volume ratio of approximately 81,000: 153,000 or 0.53. Initial dose calculations were made using the 20-μCi/mL MAR, which projected a dose that exceeded the thyroid PAG value. The 20-μCi/mL dose projection was then used to determine concentration levels that would trigger a range of classification levels and protective actions for collocated facilities. For these concentrations, a fraction multiplier, F, was developed by dividing the thyroid PAG and 10% of the thyroid PAG at 100 m and the thyroid PAG at the ATR Complex perimeter fence, collocated facilities and the Idaho National Laboratory Site boundary distance by the computed values for the 20-μCi/mL ST multiplied by the 0.53 volume adjustment. For example, it was determined that a ground-level release from a 9-μCi/mL concentration would meet the conditions for an alert classification level (i.e., greater than 10% of the PAG at 100 m). This method identified the MAR concentrations of interest for a full spectrum of operational emergency (OE) conditions.

Table A-3. Material-at-risk for TRA-605.

Isotope	Quantity (Ci)
Rb-88	1.82E+2
Sr-89	1.65E+1
Sr-90	1.38E−1
Sr-91	3.48E+1
Sr-92	3.56E+1
Y-91	1.21E+0
Zr-95	1.26E+0
Mo-99	1.23E+1
Ru-103	4.22E+0
Ru-106	9.62E−2
Sb-127	4.97E+0
Te-129m	2.24E+0
Sb-129	2.08E+1
I-131	7.36E+1
Te-131m	1.12E+1
I-132	1.11E+2
Te-132	1.32E+2
I-133	1.69E+2
I-134	1.89E+2
Cs-134	9.26E−1
I-135	1.57E+2
Cs-137	1.19E+0
Cs-138	3.35E+2
Ba-140	3.67E+1
La-140	2.47E+0
Ce-141	1.72E+0
Pr-143	2.21E+0
Ce-144	3.03E−1
Pu-238	3.40E−5

The primary barrier is the ATR fuel cladding or experimental fuel cladding.

The ATR PCS, effluent piping from the ATR to TRA-605, WWFT, and HWST are the secondary barriers.

The TRA-605 WWFT and TRA-605 HWST are vented to the MTR stack exhaust system.

The ATR PCS water chemistry has a concentration limit of 20 μCi/mL.[7]

Event indicators are primary coolant water chemistry sample analyses. Also, instrument alarms on the ATR hot waste tank piping to TRA-605.

A-1.4.1.1.3 Barrier and Failure Mode Analyses. In addition to the specific barrier failure event, barrier and failure mode analyses must consider and document, where appropriate, available instruments and other indicators that serve as precursors to the event or help quantify the event.

The barrier and failure mode analyses presented below for TRA-605 are summarized in Table A-4.

Table A-4. Radiological failure modes and barriers for TRA-605.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA-605-R-1 (Ventilation system inoperable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 8.88E−6 damage ratio (DR)[a] for single fuel element or 2.22E−7 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 0.9 µCi/mL. Based on release fractions taken from Regulatory Guide 1.4[8] and Reactor Safety Study[6,9] (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST
TRA-605-R-2 (Ventilation system operable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 8.88E−6 DR for single fuel element or 2.22E−7 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 0.9 µCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST

Table A-4. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA-605-R-3 (Ventilation system inoperable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 8.88E−5 DR for single fuel element or 2.22E−6 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 9 μCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST
TRA-605-R-4 (Ventilation system operable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 8.88E−5 DR for single fuel element or 2.22E−6 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 9 μCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST

Table A-4. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA-605-R-5 (Ventilation system inoperable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 1.97E−4 DR for single fuel element or 4.93E−6 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 20 μCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST
TRA-605-R-6 (Ventilation system operable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 1.97E−4 DR for single fuel element or 4.93E−6 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 20 μCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST

Table A-4. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA-605-R-7 (Ventilation system inoperable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 1.17E−1 DR for single fuel element or 2.93E−3 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 2.54E+4 μCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST
TRA-605-R-8 (Ventilation system operable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 1.17E−1 DR for single fuel element or 2.93E−3 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 2.54E+4 μCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST

Table A-4. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA-605-R-9 (Ventilation system operable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 8.08E−2 DR for single fuel element or 2.02E−3 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 8.19E+3 μCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST
TRA-605-R-10 (Ventilation system inoperable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 2.56E−2 DR for single fuel element or 6.41E−4 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 2.60E+3 μCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST

Table A-4. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA-605-R-11 (Ventilation system inoperable)	Radioactive (fresh fission products)	Fuel cladding	Possible fuel clad imperfection — 1.65E−1 DR for single fuel element or 4.12E−3 DR for the ATR core	Normal ATR operations following insertion of a defective fuel element during refueling or insertion of a defective experimental fuel element or possible mechanical damage due to foreign object	Fission product release from fuel element(s) to PCS reaching a concentration of 1.67E+4 µCi/mL. Based on release fractions taken from Regulatory Guide 1.4 and Reactor Safety Study (see EDF-6873, Table 1).	Primary coolant pumped into WWFT or HWST

a. The DR is calculated based on the assumption taken from Engineering Design File TRA-ATR-1804[10] that melting one third of one fuel element would result in a primary coolant fresh fission product concentration of 33,800 µCi/mL. For example, a primary coolant concentration of 0.9 µCi/mL would be equivalent to a DR of (0.9 µCi/mL)/(3 × 33,800 µCi/mL) = 8.88E−6 for a single fuel element or DR = (0.9 µCi/mL)/(40 × 3 × 33,800 µCi/mL) = 2.22E−7 for the 40 fuel element ATR core.

A-1.4.1.1.4 Scenario Bases and Analytical Assumptions

1. Scenario TRA-605-R-1, Advanced Test Reactor Primary Coolant Sample of 0.9 μCi/mL (Ground Level)

Detailed Scenario Description	Fuel cladding breach causes an ATR primary coolant sample concentration of 0.9 μCi/mL.
Material-at-Risk	See Table A-3.
Release Characteristics	The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. It takes approximately 9 hours to pump the PCS volume from the ATR to TRA-605. The TRA-605 ventilation exhaust system is inoperable, which results in a ground-level release to the environment.
Airborne Release Fraction	2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94,[11] Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92,[12] p. A-8).
Respirable Fraction	5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).
Adjustment Factor	F = 0.0238 (81,000/153,000 × 0.9/20 = 0.0238, adjusts MAR for a concentration of 0.9 μCi/mL).
Leak Path Factor	5.0E−1 (ground-level release).
Source Term	The ST shown in Table A-5 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF. Where F = adjustment factor LPF = leak path factor ARF = airborne release fraction RF = respirable fraction.
Modeling Software and Inputs	Radiological Safety Analysis Computer Program (RSAC), Version 6.2.[13] Ground-level release. Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-5. Scenario TRA-605-R-1 source term development, primary coolant 0.9 µCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	0.0238	0.5	2.00E−4	5.00E−1	2.17E−4
Sr-89	1.65E+1	0.0238	0.5	2.00E−4	5.00E−1	1.96E−5
Sr-90	1.38E−1	0.0238	0.5	2.00E−4	5.00E−1	1.64E−7
Sr-91	3.48E+1	0.0238	0.5	2.00E−4	5.00E−1	4.14E−5
Sr-92	3.56E+1	0.0238	0.5	2.00E−4	5.00E−1	4.24E−5
Y-91	1.21E+0	0.0238	0.5	2.00E−4	5.00E−1	1.44E−6
Zr-95	1.26E+0	0.0238	0.5	2.00E−4	5.00E−1	1.50E−6
Mo-99	1.23E+1	0.0238	0.5	2.00E−4	5.00E−1	1.46E−5
Ru-103	4.22E+0	0.0238	0.5	2.00E−4	5.00E−1	5.02E−6
Ru-106	9.62E−2	0.0238	0.5	2.00E−4	5.00E−1	1.14E−7
Sb-127	4.97E+0	0.0238	0.5	2.00E−4	5.00E−1	5.91E−6
Te-129m	2.24E+0	0.0238	0.5	2.00E−4	5.00E−1	2.67E−6
Sb-129	2.08E+1	0.0238	0.5	2.00E−4	5.00E−1	2.48E−5
I-131	7.36E+1	0.0238	0.5	5.00E−1	1.00E+0	4.38E−1
Te-131m	1.12E+1	0.0238	0.5	2.00E−4	5.00E−1	1.33E−5
I-132	1.11E+2	0.0238	0.5	5.00E−1	1.00E+0	6.60E−1
Te-132	1.32E+2	0.0238	0.5	2.00E−4	5.00E−1	1.57E−4
I-133	1.69E+2	0.0238	0.5	5.00E−1	1.00E+0	1.01E+0
I-134	1.89E+2	0.0238	0.5	5.00E−1	1.00E+0	1.12E+0
Cs-134	9.26E−1	0.0238	0.5	2.00E−4	5.00E−1	1.10E−6
I-135	1.57E+2	0.0238	0.5	5.00E−1	1.00E+0	9.34E−1
Cs-137	1.19E+0	0.0238	0.5	2.00E−4	5.00E−1	1.42E−6
Cs-138	3.35E+2	0.0238	0.5	2.00E−4	5.00E−1	3.99E−4
Ba-140	3.67E+1	0.0238	0.5	2.00E−4	5.00E−1	4.37E−5
La-140	2.47E+0	0.0238	0.5	2.00E−4	5.00E−1	2.94E−6
Ce-141	1.72E+0	0.0238	0.5	2.00E−4	5.00E−1	2.05E−6

Table A-5. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Pr-143	2.21E+0	0.0238	0.5	2.00E−4	5.00E−1	2.63E−6
Ce-144	3.03E−1	0.0238	0.5	2.00E−4	5.00E−1	3.61E−7
Pu-238	3.40E−5	0.0238	0.5	2.00E−4	5.00E−1	4.05E−11

a. Listed nuclides contribute 99% of dose. Based on 20 µCi/mL of ATR primary coolant.

2. Scenario TRA-605-R-2, Advanced Test Reactor Primary Coolant Sample of 0.9 µCi/mL (Stack Release)

Detailed Scenario Description Fuel cladding breach causes an ATR primary coolant sample concentration of 0.9 µCi/mL.

Material-at-Risk See Table A-3.

Release Characteristics The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. It takes approximately 9 hours to pump the PCS volume from the ATR to TRA-605. The TRA-605 ventilation exhaust system is operable, which results in an unfiltered stack release to the environment.

Airborne Release Fraction 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor 0.0238 (81,000/153,000 × 0.9/20 = 0.0238, adjusts MAR for a concentration of 0.9 µCi/mL).

Leak Path Factor 2.5E−1 (stack release, 0.5 wall plate out × 0.5 duct work and stack plate out).

Source Term The ST shown in Table A-6 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2. Stack release (MTR stack is 250 ft[14] tall with no credit taken for jet plume rise). Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-6. Scenario TRA-605-R-2 source term development, primary coolant 0.9 µCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	0.0238	0.25	2.00E−4	5.00E−1	1.08E−4
Sr-89	1.65E+1	0.0238	0.25	2.00E−4	5.00E−1	9.82E−6
Sr-90	1.38E−1	0.0238	0.25	2.00E−4	5.00E−1	8.21E−8
Sr-91	3.48E+1	0.0238	0.25	2.00E−4	5.00E−1	2.07E−5
Sr-92	3.56E+1	0.0238	0.25	2.00E−4	5.00E−1	2.12E−5
Y-91	1.21E+0	0.0238	0.25	2.00E−4	5.00E−1	7.20E−7
Zr-95	1.26E+0	0.0238	0.25	2.00E−4	5.00E−1	7.50E−7
Mo-99	1.23E+1	0.0238	0.25	2.00E−4	5.00E−1	7.32E−6
Ru-103	4.22E+0	0.0238	0.25	2.00E−4	5.00E−1	2.51E−6
Ru-106	9.62E−2	0.0238	0.25	2.00E−4	5.00E−1	5.72E−8
Sb-127	4.97E+0	0.0238	0.25	2.00E−4	5.00E−1	2.96E−6
Te-129m	2.24E+0	0.0238	0.25	2.00E−4	5.00E−1	1.33E−6
Sb-129	2.08E+1	0.0238	0.25	2.00E−4	5.00E−1	1.24E−5
I-131	7.36E+1	0.0238	0.25	5.00E−1	1.00E+0	2.19E−1
Te-131m	1.12E+1	0.0238	0.25	2.00E−4	5.00E−1	6.66E−6
I-132	1.11E+2	0.0238	0.25	5.00E−1	1.00E+0	3.30E−1
Te-132	1.32E+2	0.0238	0.25	2.00E−4	5.00E−1	7.85E−5
I-133	1.69E+2	0.023 8	0.25	5.00E−1	1.00E+0	5.03E−1
I-134	1.89E+2	0.0238	0.25	5.00E−1	1.00E+0	5.62E−1
Cs-134	9.26E−1	0.0238	0.25	2.00E−4	5.00E−1	5.51E−7
I-135	1.57E+2	0.0238	0.25	5.00E−1	1.00E+0	4.67E−1
Cs-137	1.19E+0	0.0238	0.25	2.00E−4	5.00E−1	7.08E−7
Cs-138	3.35E+2	0.0238	0.25	2.00E−4	5.00E−1	1.99E−4
Ba-140	3.67E+1	0.0238	0.25	2.00E−4	5.00E−1	2.18E−5
La-140	2.47E+0	0.0238	0.25	2.00E−4	5.00E−1	1.47E−6
Ce-141	1.72E+0	0.0238	0.25	2.00E−4	5.00E−1	1.02E−6

Table A-6 (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Pr-143	2.21E+0	0.0238	0.25	2.00E−4	5.00E−1	1.31E−6
Ce-144	3.03E−1	0.0238	0.25	2.00E−4	5.00E−1	1.80E−7
Pu-238	3.40E−5	0.0238	0.25	2.00E−4	5.00E−1	2.02E−11

[a] Listed nuclides contribute 99% of dose. Based on 20 µCi/mL of ATR primary coolant.

3. Scenario TRA-605-R-3, Advanced Test Reactor Primary Coolant Sample of 9 µCi/mL (Ground Level)

Detailed Scenario Description	Fuel cladding breach causes an ATR primary coolant sample concentration of 9 µCi/mL.
Material-at-Risk	See Table A-3.
Release Characteristics	The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is inoperable, which results in a ground-level release to the environment.
Airborne Release Fraction	2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).
Respirable Fraction	5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).
Adjustment Factor	0.238 (adjusts MAR for a concentration of 9 µCi/mL, 81,000/153,000 × 9/20).
Leak Path Factor	5E−1 (ground-level release).
Source Term	The ST shown in Table A-7 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.
Modeling Software and Inputs	RSAC, Version 6.2. Ground-level release. Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-7. Scenario TRA-605-R-3 source term development, primary coolant 9 µCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	0.238	0.5	2.00E−4	5.00E−1	2.17E−3
Sr-89	1.65E+1	0.238	0.5	2.00E−4	5.00E−1	1.96E−4
Sr-90	1.38E−1	0.238	0.5	2.00E−4	5.00E−1	1.64E−6
Sr-91	3.48E+1	0.238	0.5	2.00E−4	5.00E−1	4.14E−4
Sr-92	3.56E+1	0.238	0.5	2.00E−4	5.00E−1	4.24E−4
Y-91	1.21E+0	0.238	0.5	2.00E−4	5.00E−1	1.44E−5
Zr-95	1.26E+0	0.238	0.5	2.00E−4	5.00E−1	1.50E−5
Mo-99	1.23E+1	0.238	0.5	2.00E−4	5.00E−1	1.46E−4
Ru-103	4.22E+0	0.238	0.5	2.00E−4	5.00E−1	5.02E−5
Ru-106	9.62E−2	0.238	0.5	2.00E−4	5.00E−1	1.14E−6
Sb-127	4.97E+0	0.238	0.5	2.00E−4	5.00E−1	5.91E−5
Te-129m	2.24E+0	0.238	0.5	2.00E−4	5.00E−1	2.67E−5
Sb-129	2.08E+1	0.238	0.5	2.00E−4	5.00E−1	2.48E−4
I-131	7.36E+1	0.238	0.5	5.00E−1	1.00E+0	4.38E+0
Te-131m	1.12E+1	0.238	0.5	2.00E−4	5.00E−1	1.33E−4
I-132	1.11E+2	0.238	0.5	5.00E−1	1.00E+0	6.60E+0
Te-132	1.32E+2	0.238	0.5	2.00E−4	5.00E−1	1.57E−3
I-133	1.69E+2	0.238	0.5	5.00E−1	1.00E+0	1.01E+1
I-134	1.89E+2	0.238	0.5	5.00E−1	1.00E+0	1.12E+1
Cs-134	9.26E−1	0.238	0.5	2.00E−4	5.00E−1	1.10E−5
I-135	1.57E+2	0.238	0.5	5.00E−1	1.00E+0	9.34E+0
Cs-137	1.19E+0	0.238	0.5	2.00E−4	5.00E−1	1.42E−5
Cs-138	3.35E+2	0.238	0.5	2.00E−4	5.00E−1	3.99E−3
Ba-140	3.67E+1	0.238	0.5	2.00E−4	5.00E−1	4.37E−4
La-140	2.47E+0	0.238	0.5	2.00E−4	5.00E−1	2.94E−5
Ce-141	1.72E+0	0.238	0.5	2.00E−4	5.00E−1	2.05E−5

Table A-7. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Pr-143	2.21E+0	0.238	0.5	2.00E−4	5.00E−1	2.63E−5
Ce-144	3.03E−1	0.238	0.5	2.00E−4	5.00E−1	3.61E−6
Pu-238	3.40E−5	0.238	0.5	2.00E−4	5.00E−1	4.05E−10

a. Listed nuclides contribute 99% of dose. Based on 20 μCi/mL of ATR primary coolant.

4. Scenario TRA-605-R-4, Advanced Test Reactor Primary Coolant Sample of 9 μCi/mL (Stack Release)

Detailed Scenario Description Fuel cladding breach causes an ATR primary coolant sample concentration of 9 μCi/mL.

Material-at-Risk See Table A-3.

Release Characteristics The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is operable, which results in an unfiltered stack release to the environment.

Airborne Release Fraction 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor 0.238 (81,000/153,000 × 9/20, adjusts MAR for a concentration of 9 μCi/mL).

Leak Path Factor 2.5E−1 (stack release, 0.5 wall plate out × 0.5 duct and stack plate out).

Source Term The ST shown in Table A-8 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2. Stack release (MTR stack is 250 ft tall with no credit taken for jet plume rise). Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-8. Scenario TRA-605-R-4 source term development, primary coolant 9 µCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	0.238	0.25	2.00E−4	5.00E−1	1.08E−3
Sr-89	1.65E+1	0.238	0.25	2.00E−4	5.00E−1	9.82E−5
Sr-90	1.38E−1	0.238	0.25	2.00E−4	5.00E−1	8.21E−7
Sr-91	3.48E+1	0.238	0.25	2.00E−4	5.00E−1	2.07E−4
Sr-92	3.56E+1	0.238	0.25	2.00E−4	5.00E−1	2.12E−4
Y-91	1.21E+0	0.238	0.25	2.00E−4	5.00E−1	7.20E−6
Zr-95	1.26E+0	0.238	0.25	2.00E−4	5.00E−1	7.50E−6
Mo-99	1.23E+1	0.238	0.25	2.00E−4	5.00E−1	7.32E−5
Ru-103	4.22E+0	0.238	0.25	2.00E−4	5.00E−1	2.51E−5
Ru-106	9.62E−2	0.238	0.25	2.00E−4	5.00E−1	5.72E−7
Sb-127	4.97E+0	0.238	0.25	2.00E−4	5.00E−1	2.96E−5
Te-129m	2.24E+0	0.238	0.25	2.00E−4	5.00E−1	1.33E−5
Sb-129	2.08E+1	0.238	0.25	2.00E−4	5.00E−1	1.24E−4
I-131	7.36E+1	0.238	0.25	5.00E−1	1.00E+0	2.19E+0
Te-131m	1.12E+1	0.238	0.25	2.00E−4	5.00E−1	6.66E−5
I-132	1.11E+2	0.238	0.25	5.00E−1	1.00E+0	3.30E+0
Te-132	1.32E+2	0.238	0.25	2.00E−4	5.00E−1	7.85E−4
I-133	1.69E+2	0.238	0.25	5.00E−1	1.00E+0	5.03E+0
I-134	1.89E+2	0.238	0.25	5.00E−1	1.00E+0	5.62E+0
Cs-134	9.26E−1	0.238	0.25	2.00E−4	5.00E−1	5.51E−6
I-135	1.57E+2	0.238	0.25	5.00E−1	1.00E+0	4.67E+0
Cs-137	1.19E+0	0.238	0.25	2.00E−4	5.00E−1	7.08E−6
Cs-138	3.35E+2	0.238	0.25	2.00E−4	5.00E−1	1.99E−3
Ba-140	3.67E+1	0.238	0.25	2.00E−4	5.00E−1	2.18E−4
La-140	2.47E+0	0.238	0.25	2.00E−4	5.00E−1	1.47E−5
Ce-141	1.72E+0	0.238	0.25	2.00E−4	5.00E−1	1.02E−5

Table A-8. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Pr-143	2.21E+0	0.238	0.25	2.00E−4	5.00E−1	1.31E−5
Ce-144	3.03E−1	0.238	0.25	2.00E−4	5.00E−1	1.80E−6
Pu-238	3.40E−5	0.238	0.25	2.00E−4	5.00E−1	2.02E−10

a. Listed nuclides contribute 99% of dose. Based on 20 μCi/mL of ATR primary coolant.

5. Scenario TRA-605-R-5, Advanced Test Reactor Primary Coolant Sample of 20 μCi/mL (Ground Level)

Detailed Scenario Description — Fuel cladding breach causes an ATR primary coolant sample concentration of 20 μCi/mL.

Material-at-Risk — See Table A-3.

Release Characteristics — The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is inoperable, which results in a ground-level release to the environment.

Airborne Release Fraction — 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction — 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor — 0.53 (81,000/153,000, adjusts MAR for a concentration of 20 μCi/mL).

Leak Path Factor — 5E−1 (ground-level release).

Source Term — The ST shown in Table A-9 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs — RSAC, Version 6.2. Ground-level release. Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-9. Scenario TRA-605-R-5 source term development, primary coolant 20 μCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	0.53	0.5	2.00E−4	5.00E−1	4.82E−3
Sr-89	1.65E+1	0.53	0.5	2.00E−4	5.00E−1	4.37E−4

Table A-9. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Sr-90	1.38E−1	0.53	0.5	2.00E−4	5.00E−1	3.66E−6
Sr-91	3.48E+1	0.53	0.5	2.00E−4	5.00E−1	9.22E−4
Sr-92	3.56E+1	0.53	0.5	2.00E−4	5.00E−1	9.43E−4
Y-91	1.21E+0	0.53	0.5	2.00E−4	5.00E−1	3.21E−5
Zr-95	1.26E+0	0.53	0.5	2.00E−4	5.00E−1	3.34E−5
Mo-99	1.23E+1	0.53	0.5	2.00E−4	5.00E−1	3.26E−4
Ru-103	4.22E+0	0.53	0.5	2.00E−4	5.00E−1	1.12E−4
Ru-106	9.62E−2	0.53	0.5	2.00E−4	5.00E−1	2.55E−6
Sb-127	4.97E+0	0.53	0.5	2.00E−4	5.00E−1	1.32E−4
Te-129m	2.24E+0	0.53	0.5	2.00E−4	5.00E−1	5.94E−5
Sb-129	2.08E+1	0.53	0.5	2.00E−4	5.00E−1	5.51E−4
I-131	7.36E+1	0.53	0.5	5.00E−1	1.00E+0	9.75E+0
Te-131m	1.12E+1	0.53	0.5	2.00E−4	5.00E−1	2.97E−4
I-132	1.11E+2	0.53	0.5	5.00E−1	1.00E+0	1.47E+1
Te-132	1.32E+2	0.53	0.5	2.00E−4	5.00E−1	3.50E−3
I-133	1.69E+2	0.53	0.5	5.00E−1	1.00E+0	2.24E+1
I-134	1.89E+2	0.53	0.5	5.00E−1	1.00E+0	2.50E+1
Cs-134	9.26E−1	0.53	0.5	2.00E−4	5.00E−1	2.45E−5
I-135	1.57E+2	0.53	0.5	5.00E−1	1.00E+0	2.08E+1
Cs-137	1.19E+0	0.53	0.5	2.00E−4	5.00E−1	3.15E−5
Cs-138	3.35E+2	0.53	0.5	2.00E−4	5.00E−1	8.88E−3
Ba-140	3.67E+1	0.53	0.5	2.00E−4	5.00E−1	9.73E−4
La-140	2.47E+0	0.53	0.5	2.00E−4	5.00E−1	6.55E−5
Ce-141	1.72E+0	0.53	0.5	2.00E−4	5.00E−1	4.56E−5
Pr-143	2.21E+0	0.53	0.5	2.00E−4	5.00E−1	5.86E−5
Ce-144	3.03E−1	0.53	0.5	2.00E−4	5.00E−1	8.03E−6
Pu-238	3.40E−5	0.53	0.5	2.00E−4	5.00E−1	9.01E−10

a. Listed nuclides contribute 99% of dose. Based on 20 μCi/mL of ATR primary coolant.

6. Scenario TRA-605-R-6, Advanced Test Reactor Primary Coolant Sample of 20 µCi/mL (Stack Release)

Detailed Scenario Description — Fuel cladding breach causes an ATR primary coolant sample concentration of 20 µCi/mL.

Material-at-Risk — See Table A-3.

Release Characteristics — The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is operable, which results in an unfiltered stack release to the environment.

Airborne Release Fraction — 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction — 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor — 0.53 (81,000/153,000, adjusts MAR for a concentration of 20 µCi/mL).

Leak Path Factor — 2.5E−1 (stack release, 0.5 wall plate out × 0.5 duct work and stack plate out).

Source Term — The ST shown in Table A-10 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs — RSAC, Version 6.2. Stack release (MTR stack is 250 ft tall with no credit taken for jet plume rise). Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-10. Scenario TRA-605-R-6 source term development, primary coolant 20 µCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	0.53	0.25	2.00E−4	5.00E−1	2.41E−3
Sr-89	1.65E+1	0.53	0.25	2.00E−4	5.00E−1	2.19E−4
Sr-90	1.38E−1	0.53	0.25	2.00E−4	5.00E−1	1.83E−6
Sr-91	3.48E+1	0.53	0.25	2.00E−4	5.00E−1	4.61E−4
Sr-92	3.56E+1	0.53	0.25	2.00E−4	5.00E−1	4.72E−4
Y-91	1.21E+0	0.53	0.25	2.00E−4	5.00E−1	1.60E−5
Zr-95	1.26E+0	0.53	0.25	2.00E−4	5.00E−1	1.67E−5

Table A-10. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Mo-99	1.23E+1	0.53	0.25	2.00E−4	5.00E−1	1.63E−4
Ru-103	4.22E+0	0.53	0.25	2.00E−4	5.00E−1	5.59E−5
Ru-106	9.62E−2	0.53	0.25	2.00E−4	5.00E−1	1.27E−6
Sb-127	4.97E+0	0.53	0.25	2.00E−4	5.00E−1	6.59E−5
Te-129m	2.24E+0	0.53	0.25	2.00E−4	5.00E−1	2.97E−5
Sb-129	2.08E+1	0.53	0.25	2.00E−4	5.00E−1	2.76E−4
I-131	7.36E+1	0.53	0.25	5.00E−1	1.00E+0	4.88E+0
Te-131m	1.12E+1	0.53	0.25	2.00E−4	5.00E−1	1.48E−4
I-132	1.11E+2	0.53	0.25	5.00E−1	1.00E+0	7.35E+0
Te-132	1.32E+2	0.53	0.25	2.00E−4	5.00E−1	1.75E−3
I-133	1.69E+2	0.53	0.25	5.00E−1	1.00E+0	1.12E+1
I-134	1.89E+2	0.53	0.25	5.00E−1	1.00E+0	1.25E+1
Cs-134	9.26E−1	0.53	0.25	2.00E−4	5.00E−1	1.23E−5
I-135	1.57E+2	0.53	0.25	5.00E−1	1.00E+0	1.04E+1
Cs-137	1.19E+0	0.53	0.25	2.00E−4	5.00E−1	1.58E−5
Cs-138	3.35E+2	0.53	0.25	2.00E−4	5.00E−1	4.44E−3
Ba-140	3.67E+1	0.53	0.25	2.00E−4	5.00E−1	4.86E−4
La-140	2.47E+0	0.53	0.25	2.00E−4	5.00E−1	3.27E−5
Ce-141	1.72E+0	0.53	0.25	2.00E−4	5.00E−1	2.28E−5
Pr-143	2.21E+0	0.53	0.25	2.00E−4	5.00E−1	2.93E−5
Ce-144	3.03E−1	0.53	0.25	2.00E−4	5.00E−1	4.01E−6
Pu-238	3.40E−5	0.53	0.25	2.00E−4	5.00E−1	4.51E−10

a. Listed nuclides contribute 99% of dose. Based on 20 µCi/mL of ATR primary coolant.

7. **Scenario TRA-605-R-7, Advanced Test Reactor Primary Coolant Sample of 2.53E+4 μCi/mL (Ground Level)**

Detailed Scenario Description — Fuel cladding breach causes an ATR primary coolant sample concentration of 2.53E+4 μCi/mL. A ground-level release at this concentration has the potential to reach PAG limits at the Site boundary.

Material-at-Risk — See Table A-3.

Release Characteristics — The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is inoperable, which results in a ground-level release to the environment.

Airborne Release Fraction — 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction — 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor — 670 (81,000/153,000 × 2,53E+4/20, adjusts MAR for a concentration of 2.53E+4 μCi/mL).

Leak Path Factor — 5E−1 (ground-level release).

Source Term — The ST shown in Table A-11 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs — RSAC, Version 6.2. Ground-level release. Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-11. Scenario TRA-605-R-7 source term development, primary coolant 2.53+4 μCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	670	0.5	2.00E−4	5.00E−1	6.10E+0
Sr-89	1.65E+1	670	0.5	2.00E−4	5.00E−1	5.53E−1
Sr-90	1.38E−1	670	0.5	2.00E−4	5.00E−1	4.62E−3
Sr-91	3.48E+1	670	0.5	2.00E−4	5.00E−1	1.17E+0
Sr-92	3.56E+1	670	0.5	2.00E−4	5.00E−1	1.19E+0
Y-91	1.21E+0	670	0.5	2.00E−4	5.00E−1	4.05E−2
Zr-95	1.26E+0	670	0.5	2.00E−4	5.00E−1	4.22E−2
Mo-99	1.23E+1	670	0.5	2.00E−4	5.00E−1	4.12E−1
Ru-103	4.22E+0	670	0.5	2.00E−4	5.00E−1	1.41E−1

Table A-11. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Ru-106	9.62E−2	670	0.5	2.00E−4	5.00E−1	3.22E−3
Sb-127	4.97E+0	670	0.5	2.00E−4	5.00E−1	1.66E−1
Te-129m	2.24E+0	670	0.5	2.00E−4	5.00E−1	7.50E−2
Sb-129	2.08E+1	670	0.5	2.00E−4	5.00E−1	6.97E−1
I-131	7.36E+1	670	0.5	5.00E−1	1.00E+0	1.23E+4
Te-131m	1.12E+1	670	0.5	2.00E−4	5.00E−1	3.75E−1
I-132	1.11E+2	670	0.5	5.00E−1	1.00E+0	1.86E+4
Te-132	1.32E+2	670	0.5	2.00E−4	5.00E−1	4.42E+0
I-133	1.69E+2	670	0.5	5.00E−1	1.00E+0	2.83E+4
I-134	1.89E+2	670	0.5	5.00E−1	1.00E+0	3.17E+4
Cs-134	9.26E−1	670	0.5	2.00E−4	5.00E−1	3.10E−2
I-135	1.57E+2	670	0.5	5.00E−1	1.00E+0	2.63E+4
Cs-137	1.19E+0	670	0.5	2.00E−4	5.00E−1	3.99E−2
Cs-138	3.35E+2	670	0.5	2.00E−4	5.00E−1	1.12E+1
Ba-140	3.67E+1	670	0.5	2.00E−4	5.00E−1	1.23E+0
La-140	2.47E+0	670	0.5	2.00E−4	5.00E−1	8.27E−2
Ce-141	1.72E+0	670	0.5	2.00E−4	5.00E−1	5.76E−2
Pr-143	2.21E+0	670	0.5	2.00E−4	5.00E−1	7.40E−2
Ce-144	3.03E−1	670	0.5	2.00E−4	5.00E−1	1.02E−2
Pu-238	3.40E−5	670	0.5	2.00E−4	5.00E−1	1.14E−6

a. Listed nuclides contribute 99% of dose. Based on 20 µCi/mL of ATR primary coolant.

8. Scenario TRA-605-R-8, Advanced Test Reactor Primary Coolant Sample of 2.53E+4 µCi/mL (Stack Release)

Detailed Scenario Description Fuel cladding breach causes an ATR primary coolant sample concentration of 2.53E+4 µCi/mL. The basis for this concentration is that a ground-level release at this concentration has the potential to reach PAG limits at the Site boundary.

Material-at-Risk See Table A-3.

Release Characteristics The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is operable, which results in an unfiltered stack release to the environment.

Airborne Release Fraction 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor 670 (81,000/153,000 × 2.53E+4/20, adjusts MAR for a concentration of 2.53E+4 μCi/mL).

Leak Path Factor 2.5E−1 (stack release, 0.5 wall plate out × 0.5 duct work and stack plate out).

Source Term The ST shown in Table A-12 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2. Stack release (MTR stack is 250 ft tall with no credit taken for jet plume rise). Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-12. Scenario TRA-605-R-8 source term development, primary coolant 2.53E+4 μCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	670	0.25	2.00E−4	5.00E−1	3.05E+0
Sr-89	1.65E+1	670	0.25	2.00E−4	5.00E−1	2.76E−1
Sr-90	1.38E−1	670	0.25	2.00E−4	5.00E−1	2.31E−3
Sr-91	3.48E+1	670	0.25	2.00E−4	5.00E−1	5.83E−1
Sr-92	3.56E+1	670	0.25	2.00E−4	5.00E−1	5.96E−1
Y-91	1.21E+0	670	0.25	2.00E−4	5.00E−1	2.03E−2
Zr-95	1.26E+0	670	0.25	2.00E−4	5.00E−1	2.11E−2
Mo-99	1.23E+1	670	0.25	2.00E−4	5.00E−1	2.06E−1
Ru-103	4.22E+0	670	0.25	2.00E−4	5.00E−1	7.07E−2
Ru-106	9.62E−2	670	0.25	2.00E−4	5.00E−1	1.61E−3
Sb-127	4.97E+0	670	0.25	2.00E−4	5.00E−1	8.32E−2
Te-129m	2.24E+0	670	0.25	2.00E−4	5.00E−1	3.75E−2
Sb-129	2.08E+1	670	0.25	2.00E−4	5.00E−1	3.48E−1
I-131	7.36E+1	670	0.25	5.00E−1	1.00E+0	6.16E+3
Te-131m	1.12E+1	670	0.25	2.00E−4	5.00E−1	1.88E−1
I-132	1.11E+2	670	0.25	5.00E−1	1.00E+0	9.30E+3
Te-132	1.32E+2	670	0.25	2.00E−4	5.00E−1	2.21E+0

Table A-12. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
I-133	1.69E+2	670	0.25	5.00E−1	1.00E+0	1.42E+4
I-134	1.89E+2	670	0.25	5.00E−1	1.00E+0	1.58E+4
Cs-134	9.26E−1	670	0.25	2.00E−4	5.00E−1	1.55E−2
I-135	1.57E+2	670	0.25	5.00E−1	1.00E+0	1.31E+4
Cs-137	1.19E+0	670	0.25	2.00E−4	5.00E−1	1.99E−2
Cs-138	3.35E+2	670	0.25	2.00E−4	5.00E−1	5.61E+0
Ba-140	3.67E+1	670	0.25	2.00E−4	5.00E−1	6.15E−1
La-140	2.47E+0	670	0.25	2.00E−4	5.00E−1	4.14E−2
Ce-141	1.72E+0	670	0.25	2.00E−4	5.00E−1	2.88E−2
Pr-143	2.21E+0	670	0.25	2.00E−4	5.00E−1	3.70E−2
Ce-144	3.03E−1	670	0.25	2.00E−4	5.00E−1	5.08E−1
Pu-238	3.40E−5	670	0.25	2.00E−4	5.00E−1	5.70E−7

a. Listed nuclides contribute 99% of dose. Based on 20 µCi/mL of ATR primary coolant.

9. Scenario TRA-605-R-9, Advanced Test Reactor Primary Coolant Sample of 8.19E+3 µCi/mL (Stack Release)

Detailed Scenario Description Fuel cladding breach causes an ATR primary coolant sample concentration of 8.19E+3 µCi/mL. This concentration potentially exceeds the thyroid PAG limit at the nearest collocated facility under Class D stability.

Material-at-Risk See Table A-3.

Release Characteristics The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is operable, which results in an unfiltered stack release to the environment.

Airborne Release Fraction 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor 217 (81,000/153,000 × 8.19E+3/20, adjusts MAR for a concentration of 8.19E+3 µCi/mL).

Leak Path Factor 2.5E−1 (0.5 wall plate out × 0.5 duct work and stack plate out).

Source Term The ST shown in Table A-13 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2. Stack release (MTR stack is 250 ft tall with no credit taken for jet plume rise). Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-13. Scenario TRA-605-R-9 source term development, primary coolant 8.19E+3 μCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	217	0.25	2.00E−4	5.00E−1	9.86E−1
Sr-89	1.65E+1	217	0.25	2.00E−4	5.00E−1	8.94E−1
Sr-90	1.38E−1	217	0.25	2.00E−4	5.00E−1	7.48E−4
Sr-91	3.48E+1	217	0.25	2.00E−4	5.00E−1	1.89E−1
Sr-92	3.56E+1	217	0.25	2.00E−4	5.00E−1	1.93E−1
Y-91	1.21E+0	217	0.25	2.00E−4	5.00E−1	6.56E−3
Zr-95	1.26E+0	217	0.25	2.00E−4	5.00E−1	6.83E−3
Mo-99	1.23E+1	217	0.25	2.00E−4	5.00E−1	6.67E−2
Ru-103	4.22E+0	217	0.25	2.00E−4	5.00E−1	2.29E−2
Ru-106	9.62E−2	217	0.25	2.00E−4	5.00E−1	5.21E−4
Sb-127	4.97E+0	217	0.25	2.00E−4	5.00E−1	2.69E−2
Te-129m	2.24E+0	217	0.25	2.00E−4	5.00E−1	1.21E−2
Sb-129	2.08E+1	217	0.25	2.00E−4	5.00E−1	1.13E−1
I-131	7.36E+1	217	0.25	5.00E−1	1.00E+0	2.00E+3
Te-131m	1.12E+1	217	0.25	2.00E−4	5.00E−1	6.07E−2
I-132	1.11E+2	217	0.25	5.00E−1	1.00E+0	3.01E+3
Te-132	1.32E+2	217	0.25	2.00E−4	5.00E−1	7.15E−1
I-133	1.69E+2	217	0.25	5.00E−1	1.00E+0	4.58E+3
I-134	1.89E+2	217	0.25	5.00E−1	1.00E+0	5.13E+3
Cs-134	9.26E−1	217	0.25	2.00E−4	5.00E−1	5.02E−3
I-135	1.57E+2	217	0.25	5.00E−1	1.00E+0	4.26E+3

Table A-13. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Cs-137	1.19E+0	217	0.25	2.00E−4	5.00E−1	6.45E−3
Cs-138	3.35E+2	217	0.25	2.00E−4	5.00E−1	1.82E+0
Ba-140	3.67E+1	217	0.25	2.00E−4	5.00E−1	1.99E−1
La-140	2.47E+0	217	0.25	2.00E−4	5.00E−1	1.34E−2
Ce-141	1.72E+0	217	0.25	2.00E−4	5.00E−1	9.32E−3
Pr-143	2.21E+0	217	0.25	2.00E−4	5.00E−1	1.20E−2
Ce-144	3.03E−1	217	0.25	2.00E−4	5.00E−1	1.64E−3
Pu-238	3.40E−5	217	0.25	2.00E−4	5.00E−1	1.84E−7

a. Listed nuclides contribute 99% of dose. Based on 20 µCi/mL of ATR primary coolant.

10. Scenario TRA-605-R-10, Advanced Test Reactor Primary Coolant Sample of 2.60E+3 µCi/mL (Ground Level)

Detailed Scenario Description Fuel cladding breach causes an ATR primary coolant sample concentration of 2.60E+3 µCi/mL. This concentration has the potential to exceed the thyroid PAG limit at the nearest collocated facility.

Material-at-Risk See Table A-3.

Release Characteristics The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is inoperable, which results in a ground-level release to the environment.

Airborne Release Fraction 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor 69.6 (81,000/153,000 × 2.63E+3/20, adjusts MAR for a concentration of 2.60E+3 µCi/mL).

Leak Path Factor 5E−1 (ground-level release).

Source Term The ST shown in Table A-14 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release. Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-14. Scenario TRA-605-R-10 source term development, primary coolant 2.60E+3 μCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	69.6	0.5	2.00E−4	5.00E−1	6.34E−1
Sr-89	1.65E+1	69.6	0.5	2.00E−4	5.00E−1	5.74E−2
Sr-90	1.38E−1	69.6	0.5	2.00E−4	5.00E−1	4.80E−4
Sr-91	3.48E+1	69.6	0.5	2.00E−4	5.00E−1	1.21E−1
Sr-92	3.56E+1	69.6	0.5	2.00E−4	5.00E−1	1.24E−1
Y-91	1.21E+0	69.6	0.5	2.00E−4	5.00E−1	4.21E−3
Zr-95	1.26E+0	69.6	0.5	2.00E−4	5.00E−1	4.39E−3
Mo-99	1.23E+1	69.6	0.5	2.00E−4	5.00E−1	4.28E−2
Ru-103	4.22E+0	69.6	0.5	2.00E−4	5.00E−1	1.47E−2
Ru-106	9.62E−2	69.6	0.5	2.00E−4	5.00E−1	3.35E−4
Sb-127	4.97E+0	69.6	0.5	2.00E−4	5.00E−1	1.73E−2
Te-129m	2.24E+0	69.6	0.5	2.00E−4	5.00E−1	7.80E−3
Sb-129	2.08E+1	69.6	0.5	2.00E−4	5.00E−1	7.24E−2
I-131	7.36E+1	69.6	0.5	5.00E−1	1.00E+0	1.28E+3
Te-131m	1.12E+1	69.6	0.5	2.00E−4	5.00E−1	3.90E−2
I-132	1.11E+2	69.6	0.5	5.00E−1	1.00E+0	1.93E+3
Te-132	1.32E+2	69.6	0.5	2.00E−4	5.00E−1	4.59E−1
I-133	1.69E+2	69.6	0.5	5.00E−1	1.00E+0	2.94E+3
I-134	1.89E+2	69.6	0.5	5.00E−1	1.00E+0	3.29E+3
Cs-134	9.26E−1	69.6	0.5	2.00E−4	5.00E−1	3.22E−3
I-135	1.57E+2	69.6	0.5	5.00E−1	1.00E+0	2.73E+3
Cs-137	1.19E+0	69.6	0.5	2.00E−4	5.00E−1	4.14E−3
Cs-138	3.35E+2	69.6	0.5	2.00E−4	5.00E−1	1.17E+0
Ba-140	3.67E+1	69.6	0.5	2.00E−4	5.00E−1	1.28E−1
La-140	2.47E+0	69.6	0.5	2.00E−4	5.00E−1	8.60E−3
Ce-141	1.72E+0	69.6	0.5	2.00E−4	5.00E−1	5.99E−3

Table A-14. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Pr-143	2.21E+0	69.6	0.5	2.00E−4	5.00E−1	7.69E−3
Ce-144	3.03E−1	69.6	0.5	2.00E−4	5.00E−1	1.05E−3
Pu-238	3.40E−5	69.6	0.5	2.00E−4	5.00E−1	1.18E−7

a. Listed nuclides contribute 99% of dose. Based on 20 μCi/mL of ATR primary coolant.

11. Scenario TRA-605-R-11, Advanced Test Reactor Primary Coolant Sample of 1.67E+4 μCi/mL (Ground Level)

Detailed Scenario Description — Fuel cladding breach causes an ATR primary coolant sample concentration of 1.67E+4 μCi/mL. This concentration has the potential to exceed the thyroid PAG limits at the Central Facilities Area, and could prohibit the use of the Central Facilities Area as an alternate emergency control center.

Material-at-Risk — See Table A-3.

Release Characteristics — The ATR primary coolant is pumped to TRA-605 for ion exchange treatment. Splashing in the receiving tanks results in some of the radioactive material going airborne. The TRA-605 ventilation exhaust system is inoperable, which results in a ground-level release to the environment.

Airborne Release Fraction — 2.0E−4 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 5.0E−1 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Respirable Fraction — 5.0E−1 (bounding for most nuclides in liquid spills, DOE-HDBK-3010-94, Section 3.2.3.1, p. 3-33) and 1.0 for iodine isotopes (DOE-STD-1027-92, p. A-8).

Adjustment Factor — 442 (81,000/153,000 × 1.67E+4/20, adjusts MAR for a concentration of 1.67E+4 μCi/mL).

Leak Path Factor — 5E−1 (ground-level release).

Source Term — The ST shown in Table A-15 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs — RSAC, Version 6.2. Ground-level release. Leakage constant (K1) is set to 3.09E−5 s^{-1} [1/(3,600 s/hr × 9 hr)] and release time for exponential decay function is set to 3.24E+4 (s) (9 hr × 3,600 s/hr).

Table A-15. Scenario TRA-605-R-11 source term development, primary coolant 1.67E+4 μCi/mL.

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Rb-88	1.82E+2	442	0.5	2.00E−4	5.00E−1	4.02E+0
Sr-89	1.65E+1	442	0.5	2.00E−4	5.00E−1	3.65E−1
Sr-90	1.38E−1	442	0.5	2.00E−4	5.00E−1	3.05E−3
Sr-91	3.48E+1	442	0.5	2.00E−4	5.00E−1	7.69E−1
Sr-92	3.56E+1	442	0.5	2.00E−4	5.00E−1	7.87E−1
Y-91	1.21E+0	442	0.5	2.00E−4	5.00E−1	2.67E−2
Zr-95	1.26E+0	442	0.5	2.00E−4	5.00E−1	2.78E−2
Mo-99	1.23E+1	442	0.5	2.00E−4	5.00E−1	2.72E−1
Ru-103	4.22E+0	442	0.5	2.00E−4	5.00E−1	9.33E−2
Ru-106	9.62E−2	442	0.5	2.00E−4	5.00E−1	2.13E−3
Sb-127	4.97E+0	442	0.5	2.00E−4	5.00E−1	1.10E−1
Te-129m	2.24E+0	442	0.5	2.00E−4	5.00E−1	4.95E−2
Sb-129	2.08E+1	442	0.5	2.00E−4	5.00E−1	4.60E−1
I-131	7.36E+1	442	0.5	5.00E−1	1.00E+0	8.13E+3
Te-131m	1.12E+1	442	0.5	2.00E−4	5.00E−1	2.48E−1
I-132	1.11E+2	442	0.5	5.00E−1	1.00E+0	1.23E+4
Te-132	1.32E+2	442	0.5	2.00E−4	5.00E−1	2.92E+0
I-133	1.69E+2	442	0.5	5.00E−1	1.00E+0	1.87E+4
I-134	1.89E+2	442	0.5	5.00E−1	1.00E+0	2.09E+4
Cs-134	9.26E−1	442	0.5	2.00E−4	5.00E−1	2.05E−2
I-135	1.57E+2	442	0.5	5.00E−1	1.00E+0	1.73E+4
Cs-137	1.19E+0	442	0.5	2.00E−4	5.00E−1	2.63E−2
Cs-138	3.35E+2	442	0.5	2.00E−4	5.00E−1	7.40E+0
Ba-140	3.67E+1	442	0.5	2.00E−4	5.00E−1	8.11E−1
La-140	2.47E+0	442	0.5	2.00E−4	5.00E−1	5.46E−2
Ce-141	1.72E+0	442	0.5	2.00E−4	5.00E−1	3.80E−2

Table A-15. (continued).

Nuclide[a]	MAR (Ci)	F	LPF	ARF	RF	ST (Ci)
Pr-143	2.21E+0	442	0.5	2.00E−4	5.00E−1	4.88E−2
Ce-144	3.03E−1	442	0.5	2.00E−4	5.00E−1	6.70E−3
Pu-238	3.40E−5	442	0.5	2.00E−4	5.00E−1	7.52E−7

a. Listed nuclides contribute 99% of dose. Based on 20 μCi/mL of ATR primary coolant.

A-1.5 Evaluation Results

A-1.5.1 Calculational Models and Bases

Calculational models and bases are addressed in the main document.

A-1.5.2 Calculation Results

In most cases, release scenarios were analyzed for airborne releases using both worst-case and average meteorological conditions based on either National Oceanic and Atmospheric Administration-provided Chi/Q values as described in the main document or in the cases of stack releases, Markee Chi/Q values that were generated by the model for Stability Classes A, D, and F. Based on the National Oceanic and Atmospheric Administration Chi/Q data shown in the main document Table A-4, the expected airborne concentrations under 50% meteorological conditions will be significantly less than the airborne concentrations under the 95% meteorological conditions for all downwind distances.

A-1.5.2.1 Radiological Hazardous Material Release Results

Radiological hazardous material release results are summarized in Table A-16 for the 95% worst-case weather TEDE, Table A-17 for the 95% worst-case weather thyroid CDE, Table A-18 for the TEDE evaluations using Stability Class D with a 50% (average) wind speed of 2.46 m/s, Table A-19 for the thyroid CDE evaluations using Stability Class D with a 50% (average) wind speed of 2.46 m/s, Table A-20 for elevated release TEDE evaluations only using Stability Class A with an average (50%) wind speed of 2.46 m/s, and Table A-21 for elevated release thyroid CDE evaluations only using Stability Class A with an average (50%) wind speed of 2.46 m/s.

Table A-16. Radiological total effective dose equivalent results for 95% worst-case meteorology.

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)				Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	Facility Boundary (100 m)	INTEC[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA-605-R-1 (Ground release)	Primary coolant at 0.9 µCi/mL	2.50E−01	2.50E−02	9.95E−05	9.82E−06	N/A	N/A	N/A
TRA-605-R-2 (Stack release)	Primary coolant at 0.9 µCi/mL	N/A	3.66E−06	8.34E−07	2.21E−07	N/A	N/A	N/A
TRA-605-R-3 (Ground release)	Primary coolant at 9 µCi/mL	2.50E+00	2.50E−01	9.95E−04	9.82E−05	100	N/A	Alert
TRA-605-R-4 (Stack release)	Primary coolant at 9 µCi/mL	N/A	3.66E−05	8.34E−06	2.21E−06	N/A	N/A	N/A
TRA-605-R-5 (Ground release)	Primary coolant at 20 µCi/mL	5.56E+00	5.56E−01	2.21E−03	2.18E−04	100	Not exceeded	Alert
TRA-605-R-6 (Stack release)	Primary coolant at 20 µCi/mL	N/A	8.15E−05	1.86E−05	4.93E−06	N/A	N/A	N/A
TRA-605-R-7 (Ground release)	Primary coolant at 2.54E+4 µCi/mL	7.02E+03	7.02E+02	2.97E+00	2.76E−01	4,700	500	Site area emergency (SAE)
TRA-605-R-8 (Stack release)	Primary coolant at 2.54E+4 µCi/mL	N/A	1.03E−01	2.35E−02	6.23E−3	100	N/A	Alert
TRA-605-R-9 (Stack release)	Primary coolant at 8.19E+3 µCi/mL	N/A	3.34E−02	7.60E−03	2.02E−03	N/A	N/A	N/A

Table A-16. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)				Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	Facility Boundary (100 m)	INTEC[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA-605-R-10 (Ground release)	Primary coolant at 2.60E+3 µCi/mL	7.30E+02	7.30E+01	2.90E−01	2.87E−02	2,000	Less than 100	SAE
TRA-605-R-11 (Ground release)	Primary coolant at 1.67E+4 µCi/mL	4.64E+03	4.64E+02	1.84E+00	1.82E−01	3,800	350	SAE

a. TEDE = total effective dose equivalent
INTEC = Idaho Nuclear Technology and Engineering Center.

Table A-17. Radiological thyroid committed dose equivalent results for 95% worst-case meteorology.

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)				Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	Facility Boundary (100 m)	INTEC Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA-605-R-1 (Ground release)	Primary coolant at 0.9 µCi/mL	5.01E+00	5.01E−01	1.97E−03	1.99E−04	100	N/A	Alert
TRA-605-R-2 (Stack release)	Primary coolant at 0.9 µCi/mL	N/A	0.00E+00	1.56E−17	6.10E−10	N/A	N/A	N/A
TRA-605-R-3 (Ground release)	Primary coolant at 9 µCi/mL	5.01E+01	5.01E+00	1.97E−02	1.99E−03	100	N/A	Alert
TRA-605-R-4 (Stack release)	Primary coolant at 9 µCi/mL	N/A	0.00E+00	1.56E−16	6.10E−09	N/A	N/A	N/A

Table A-17. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)				Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	Facility Boundary (100 m)	INTEC Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA-605-R-5 (Ground release)	Primary coolant at 20 μCi/mL	1.12E+02	1.12E+01	4.38E−02	4.43E−03	220	N/A	SAE
TRA-605-R-6 (Stack release)	Primary coolant at 20 μCi/mL	N/A	0.00E+00	3.49E−16	1.36E−08	N/A	N/A	N/A
TRA-605-R-7 (Ground release)	Primary coolant at 2.54E+4 μCi/mL	1.41E+05	1.41E+04	5.53E+01	5.59E+00	10,855	N/A	General emergency
TRA-605-R-8 (Stack release)	Primary coolant at 2.54E+4 μCi/mL	N/A	0.00E+00	4.40E−13	1.76E−05	N/A	N/A	N/A
TRA-605-R-9 (Stack release)	Primary coolant at 8.19E+3 μCi/mL	N/A	0.00E+00	1.43E−13	5.57E−06	N/A	N/A	N/A
TRA-605-R-10 (Ground release)	Primary coolant at 2.60E+3 μCi/mL	1.46E+04	1.46E+03	5.75E+00	5.81E−01	2,900	N/A	SAE
TRA-605-R-11 (Ground release)	Primary coolant at 1.67E+4 μCi/mL	9.14E+04	9.14E+03	3.65E+01	3.69E+00	8,200	N/A	SAE

Table A-18. Radiological total effective dose equivalent results for 50% (typical) meteorology.

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% (Typical) Meteorology (rem)				Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	Facility Boundary (100 m)	INTEC Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA-605-R-1 (Ground release)	Primary coolant at 0.9 µCi/mL	1.63E−03	1.63E−04	2.94E−06	4.08E−07	N/A	N/A	N/A
TRA-605-R-2 (Stack release)	Primary coolant at 0.9 µCi/mL	N/A	1.69E−06	3.13E−05	4.85E−06	N/A	N/A	N/A
TRA-605-R-3 (Ground release)	Primary coolant at 9 µCi/mL	1.63E−02	1.63E−03	2.94E−05	4.08E−06	N/A	N/A	N/A
TRA-605-R-4 (Stack release)	Primary coolant at 9 µCi/mL	N/A	1.69E−05	3.13E−04	4.85E−05	N/A	N/A	N/A
TRA-605-R-5 (Ground release)	Primary coolant at 20 µCi/mL	3.61E−02	3.61E−03	6.64E−05	9.54E−06	N/A	N/A	N/A
TRA-605-R-6 (Stack release)	Primary coolant at 20 µCi/mL	N/A	3.75E−05	6.97E−04	1.08E−04	N/A	N/A	N/A
TRA-605-R-7 (Ground release)	Primary coolant at 2.54E+4 µCi/mL	4.56E+01	4.56E+00	8.38E−02	1.20E−02	360	Not exceeded	SAE
TRA-605-R-8 (Stack release)	Primary coolant at 2.54E+4 µCi/mL	N/A	4.74E−02	8.82E−01	1.36E−1	N/A	N/A	N/A
TRA-605-R-9 (Stack release)	Primary coolant at 8.19E+3 µCi/mL	N/A	1.54E−02	2.86E−01	4.42E−02	N/A	N/A	N/A
TRA-605-R-10 (Ground release)	Primary coolant at 2.60E+3 µCi/mL	4.74E+00	4.74E−01	8.81E−03	1.25E−03	100	Not exceeded	Alert
TRA-605-R-11 (Ground release)	Primary coolant at 1.67E+4 µCi/mL	3.01E+01	3.01E+00	5.54E−02	7.96R−03	275	Not exceeded	SAE

Table A-19. Radiological thyroid committed dose equivalent results for 50% (typical) meteorology.

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% (Typical) Meteorology (rem)				Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	Facility Boundary (100 m)	INTEC Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA-605-R-1 (Ground release)	Primary coolant at 0.9 μCi/mL	3.21E−02	3.21E−03	5.73E−05	8.24E−06	N/A	N/A	N/A
TRA-605-R-2 (Stack release)	Primary coolant at 0.9 μCi/mL	N/A	2.14E−29	6.16E−04	9.53E−05	N/A	N/A	N/A
TRA-605-R-3 (Ground release)	Primary coolant at 9 μCi/mL	3.21E−01	3.21E−02	5.73E−04	8.24E−05	N/A	N/A	N/A
TRA-605-R-4 (Stack release)	Primary coolant at 9 μCi/mL	N/A	2.14E−28	6.16E−03	9.53E−04	N/A	N/A	N/A
TRA-605-R-5 (Ground release)	Primary coolant at 20 μCi/mL	7.15E−01	7.15E−02	1.28E−03	1.87E−04	N/A	N/A	N/A
TRA-605-R-6 (Stack release)	Primary coolant at 20 μCi/mL	N/A	4.77E−28	1.37E−02	2.12E−03	N/A	N/A	N/A
TRAC-605-R-7 (Ground release)	Primary coolant at 2.54E+4 μCi/mL	9.03E+02	9.03E+01	1.62E+00	2.37E−01	1,180	N/A	SAE
TRA-605-R-8 (Stack release)	Primary coolant at 2.54E+4 μCi/mL	N/A	6.03E−25	1.74E+01	2.68E+00	6,350	N/A	SAE
TRA-605-R-9 (Stack release)	Primary coolant at 8.19E+3 μCi/mL	N/A	1.96E−25	5.62E+00	8.70E−01	2,880	N/A	SAE
TRA-605-R-10 (Ground release)	Primary coolant at 2.60E+3 μCi/mL	9.39E+01	9.39E+00	1.68E−01	2.46E−02	190	N/A	SAE
TRA-605-R-11 (Ground release)	Primary coolant at 1.67E+4 μCi/mL	5.96E+02	5.96E+01	1.07E+00	1.56E−01	750	N/A	SAE

Table A-20. Radiological total effective dose equivalent results for Stability Class A meteorology.

Scenario Release Designator	Short Description	Downwind TEDE Estimates for Stability Class A Meteorology (rem)				Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	Facility Boundary (100 m)	INTEC[a] Boundary (92,865 m)	Site Boundary (10,855 m)			
TRA-605-R-1 (Ground release)	Primary coolant at 0.9 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-2 (Stack release)	Primary coolant at 0.9 µCi/mL	N/A	1.42E−06	2.68E−06	7.60E−07	N/A	N/A	N/A
TRA-605-R-3 (Ground release)	Primary coolant at 9 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-4 (Stack release)	Primary coolant at 9 µCi/mL	N/A	1.42E−05	2.68E−05	7.60E−06	N/A	N/A	N/A
TRA-605-R-5 (Ground release)	Primary coolant at 20 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-6 (Stack release)	Primary coolant at 20 µCi/mL	N/A	3.17E−05	5.96E−05	1.69E−05	N/A	N/A	N/A
TRA-605-R-7 (Ground release)	Primary coolant at 2.54E+4 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-8 (Stack release)	Primary coolant at 2.54E+4 µCi/mL	N/A	4.00E−02	7.53E−02	2.14E−2	N/A	N/A	N/A
TRA-605-R-9 (Stack release)	Primary coolant at 8.19E+3 µCi/mL	N/A	1.30E−02	2.44E−02	6.93E−03	N/A	N/A	N/A
TRA-605-R-10 (Ground release)	Primary coolant at 2.60E+3 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-11 (Ground release)	Primary coolant at 1.67E+4 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table A-21. Radiological thyroid committed dose equivalent results for Stability Class A meteorology.

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for Stability Class A Meteorology (rem)				Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	Facility Boundary (100 m)	INTEC Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA-605-R-1 (Ground release)	Primary coolant at 0.9 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-2 (Stack release)	Primary coolant at 0.9 µCi/mL	N/A	2.57E−11	5.16E−05	1.49E−05	N/A	N/A	N/A
TRA-605-R-3 (Ground release)	Primary coolant at 9 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-4 (Stack release)	Primary coolant at 9 µCi/mL	N/A	2.57E−10	5.16E−04	1.49E−04	N/A	N/A	N/A
TRA-605-R-5 (Ground release)	Primary coolant at 20 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-6 (Stack release)	Primary coolant at 20 µCi/mL	N/A	5.13E−10	1.15E−03	3.33E−04	N/A	N/A	N/A
TRA-605-R-7 (Ground release)	Primary coolant at 2.54E+4 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-8 (Stack release)	Primary coolant at 2.54E+4 µCi/mL	N/A	7.23E−07	1.45E+00	4.21E−01	1,400	N/A	SAE
TRA-605-R-9 (Stack release)	Primary coolant at 8.19E+3 µCi/mL	N/A	2.35E−07	4.71E−01	1.36E−01	N/A	N/A	N/A
TRA-605-R-10 (Ground release)	Primary coolant at 2.60E+3 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A
TRA-605-R-11 (Ground release)	Primary coolant at 1.67E+4 µCi/mL	N/A	N/A	N/A	N/A	N/A	N/A	N/A

A-1.6 Emergency Classes, Protective Actions, and Emergency Action Levels

Based on the analyses documented above, the following emergency action levels (EALs) are identified for TRA-605.

A-1.6.1 Unclassified Operational Emergency — Emergency Action Levels

Unless otherwise noted, the generic unclassified OE EALs are covered by a separate appendix to this EHA.

There are no TRA-605 facility-specific unclassified OE EALs covered in this appendix.

A-1.6.2 Alert — Emergency Action Levels

A-1.6.2.1 ATR-605-3.A.1

A-1.6.2.1.1 Event Description

Radiological release resulting from transferring contaminated (greater than 0.9 μCi/mL and less than or equal to 9 μCi/mL) primary coolant system water to TRA-605 tanks,

A-1.6.2.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of a primary coolant sample greater than 0.9 μCi/mL and less than or equal to 9 μCi/mL,

AND

loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout),

AND

radiation area monitor alarm in TRA-605,

OR

radiological control technician radiation survey in TRA-605 higher than normal background,

OR

other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.

A-1.6.2.1.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone around TRA-605.

Relocate nonessential personnel from the exclusion zone to the conference room in TRA-653.

Shelter the remainder of Advanced Test Reactor Complex personnel in place.

Control nonessential vehicle and personnel access to the exclusion zone.

A-1.6.2.1.4 Offsite Protective Action Recommendations

None.

A-1.6.2.1.5 Basis. The ATR operating specifications require primary coolant concentrations less than 20 μCi/mL. The normal primary coolant radioactivity concentration range is much less than 1.0 μCi/mL. Any primary coolant radioactivity concentration above normal will require some type of treatment. If the PCS volume is pumped to TRA-605 for storage and treatment to remove the radioactivity, there will be some release of airborne radioactivity from the TRA-605 receiving and storage tanks during the 9-hour pumping time. At these concentrations (0.9 to 9.0 μC/mL, TRA-605-R-1 and R-3), the radioactivity in the PCS volume exceeds screening thresholds, and the projected thyroid doses for a ground-level release exceed the protective action criteria (PAC) at 100 m. Since the MTR stack is not operational, the conditions for an alert are met.

A-1.6.3 Site Area Emergency — Emergency Action Levels

A-1.6.3.1 ATR-605-3.SAE.1

A-1.6.3.1.1 Event Description

Radiological release resulting from transferring contaminated (greater than 9 μCi/mL and less than or equal to 20 μCi/mL) primary coolant system water to TRA-605 tanks,

A-1.6.3.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of a primary coolant sample greater than 9 μCi/mL and less than or equal to 20 μCi/mL,

AND

loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout),

AND

radiation area monitor alarm in TRA-605,

OR

radiological control technician radiation survey in TRA-605 higher than normal background,

OR

other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.

A-1.6.3.1.3 Onsite Protective Actions

Establish a **220-m (722-ft or 0.14-mi)** exclusion zone around TRA-605.

Relocate nonessential personnel from the exclusion zone to the conference room in TRA-652.

Shelter the remainder of Advanced Test Reactor Complex personnel in place.

Control nonessential vehicle and personnel access to the exclusion zone.

Consider authorizing potassium iodide for essential emergency workers.

A-1.6.3.1.4 Offsite Protective Action Recommendations

None.

A-1.6.3.1.5 Basis. The ATR operating specifications require primary coolant concentrations less than 20 µCi/mL. The normal primary coolant radioactivity concentration range is much less than 1.0 µCi/mL. Any primary coolant radioactivity concentration above normal will require some type of treatment. If the PCS volume is pumped to TRA-605 for storage and treatment to remove the radioactivity, there will be some release of airborne radioactivity from the TRA-605 receiving and storage tanks during the 9-hour pumping time. At these concentrations (greater than 9 µCi/mL and less than or equal to 20 µCi/mL, TRA-605-R-5), the radioactivity in the PCS volume exceeds screening thresholds, and the projected thyroid doses for a ground-level release exceed the PAC beyond 200 m. Since the MTR stack is not operational, the conditions for an SAE are met.

A-1.6.3.2 ATR-605-3.SAE.2

A-1.6.3.2.1 Event Description

Radiological release resulting from transferring contaminated (greater than 20 µCi/mL and less than or equal to 2.60E+3 µCi/mL) primary coolant system water to TRA-605 tanks,

A-1.6.3.2.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of a primary coolant sample greater than 20 µCi/mL and less than or equal to 2.60E+3 µCi/mL,

AND

loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout),

AND

radiation area monitor alarm in TRA-605,

OR

radiological control technician radiation survey in TRA-605 higher than normal background,

OR

other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.

A-1.6.3.2.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **2,900 m (9,515 ft or 1.8 mi)** of the Advanced Test Reactor Complex.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate.

Control nonessential vehicle and personnel access to the evacuated areas.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.

Consider authorizing potassium iodide for essential emergency workers.

A-1.6.3.2.4 Offsite Protective Action Recommendations

None.

A-1.6.3.2.5 Basis. The ATR operating specifications require primary coolant concentrations less than 20 μCi/mL. The normal primary coolant radioactivity concentration range is much less than 1.0 μCi/mL. Any primary coolant radioactivity concentration above normal will require some type of treatment. If the PCS volume is pumped to TRA-605 for storage and treatment to remove the radioactivity, there will be some release of airborne radioactivity from the TRA-605 receiving and storage tanks during the 9-hour pumping time. At these concentrations (greater than 20 μCi/mL and less than or equal to 2.60E+3 μCi/mL, TRA-605-R-10), the radioactivity in the PCS volume exceeds screening thresholds, and the projected thyroid doses for a ground-level release exceed the PAC well beyond 200 m. Since the MTR stack is not operational, the conditions for an SAE are met.

A-1.6.3.3 ATR-605-3.SAE.3

A-1.6.3.3.1 Event Description

Radiological release resulting from transferring contaminated (greater than 2.60E+3 μCi/mL and less than or equal to 1.67E+4 μCi/mL) primary coolant system water to TRA-605 tanks,

A-1.6.3.3.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of a primary coolant sample greater than 2.60E+3 μCi/mL and less than or equal to 1.67E+4 μCi/mL,

AND

loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout),

AND

radiation area monitor alarm in TRA-605,

OR

radiological control technician radiation survey in TRA-605 higher than normal background,

OR

other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.

A-1.6.3.3.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,200 m (26,900 ft or 5.1 mi)** of the Advanced Test Reactor Complex.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate.

Control nonessential vehicle and personnel access to the evacuated areas.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Consider authorizing potassium iodide for essential emergency workers.

A-1.6.3.3.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

A-1.6.3.3.5 Basis. The ATR operating specifications require primary coolant concentrations less than 20 µCi/mL. The normal primary coolant radioactivity concentration range is much less than 1.0 µCi/mL. Any primary coolant radioactivity concentration above normal will require some type of treatment. If the PCS volume is pumped to TRA-605 for storage and treatment to remove the radioactivity, there will be some release of airborne radioactivity from the TRA-605 receiving and storage tanks during the 9-hour pumping time. At these concentrations (greater than 2.60E+3 µCi/mL and less than or equal to 1.67E+4 µCi/mL, TRA-605-R-3), the radioactivity in the PCS volume exceeds screening thresholds, and the projected thyroid doses for a ground-level release exceed the PAC well beyond 200 m. Since the MTR stack is not operational, the conditions for an SAE are met.

A-1.6.3.4 ATR-605-3.SAE.4

A-1.6.3.4.1 Event Description

Radiological release resulting from transferring contaminated (equal to 8.19E+3 µCi/mL) primary coolant system water to TRA-605 tanks,

A-1.6.3.4.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of a primary coolant sample equal to 8.19E+3 μCi/mL,

AND

Materials Test Reactor stack is operational (as indicated by the Materials Test Reactor building local readout),

AND

radiation area monitor alarm in TRA-605,

OR

radiological control technician radiation survey in TRA-605 higher than normal background,

OR

other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.

A-1.6.3.4.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **2,900 m (9,515 ft or 1.8 mi)** of the Advanced Test Reactor Complex.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate.

Control nonessential vehicle and personnel access to the evacuated areas.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.

Consider authorizing potassium iodide for essential emergency workers.

A-1.6.3.4.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

A-1.6.3.4.5 Basis. The ATR operating specifications require primary coolant concentrations less than 20 μCi/mL. The normal primary coolant radioactivity concentration range is much less than 1.0 μCi/mL. Any primary coolant radioactivity concentration above normal will require some type of treatment. If the PCS volume is pumped to TRA-605 for storage and treatment to remove the radioactivity, there will be some release of airborne radioactivity from the TRA-605 receiving and storage tanks during the 9-hour pumping time. At this concentration (equal to 8.19E+3 μCi/mL, TRA-605-R-9), the radioactivity in the PCS volume exceeds screening thresholds, and the projected thyroid doses for either a ground-level release or a stack release exceed the PAC well beyond 200 m (maximum dose at 2,865 m). Since the MTR stack is operational, the conditions for an SAE are met.

A-1.6.3.5 ATR-605-3.SAE.5

A-1.6.3.5.1 Event Description

Radiological release resulting from transferring contaminated (greater than 8.19E+3 μCi/mL and less than or equal to 2.54E+4 μCi/mL) primary coolant system water to TRA-605 tanks,

A-1.6.3.5.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of a primary coolant sample greater than 8.19E+3 μCi/mL and less than or equal to 2.54E+4 μCi/mL,

AND

Materials Test Reactor stack is operational (as indicated by the Materials Test Reactor building local readout),

AND

radiation area monitor alarm in TRA-605,

OR

radiological control technician radiation survey in TRA-605 higher than normal background,

OR

other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.

A-1.6.3.5.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **6,400 m (20,998 ft or 4 mi)** of the Advanced Test Reactor Complex.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Consider authorizing potassium iodide for essential emergency workers.

A-1.6.3.5.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

A-1.6.3.5.5 Basis. The ATR operating specifications require primary coolant concentrations less than 20 μCi/mL. The normal primary coolant radioactivity concentration range is much less than 1.0 μCi/mL. Any primary coolant radioactivity concentration above normal will require some type of treatment. If the PCS volume is pumped to TRA-605 for storage and treatment to remove the radioactivity, there will be some release of airborne radioactivity from the TRA-605 receiving and storage tanks during the 9-hour pumping time. At these concentrations (greater than 8.19E+3 μCi/mL and less than or equal to 2.54E+4 μCi/mL, TRA-605-R-8), the radioactivity in the PCS volume exceeds screening thresholds, and the projected thyroid doses for either a ground-level release or a stack release exceed the PAC well beyond 200 m (maximum dose at 2,865 m). Since the MTR stack is operational, the conditions for an SAE are met.

A-1.6.4 General Emergency — Emergency Action Levels

A-1.6.4.1 ATR-605-3.GE.1

A-1.6.4.1.1 Event Description

Radiological release resulting from transferring contaminated (greater than 2.54E+4 μCi/mL) primary coolant system water to TRA-605 tanks,

A-1.6.4.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of a primary coolant sample greater than 2.54E+4 μCi/mL,

AND

loss of flow indication in the Materials Test Reactor stack (as indicated on the Materials Test Reactor building local readout),

AND

radiation area monitor alarm in TRA-605,

OR

radiological control technician radiation survey in TRA-605 higher than normal background,

OR

other potential indicators that include radiation area monitor RM-5-49 alarms on the Advanced Test Reactor warm waste tank piping leading to TRA-605.

A-1.6.4.1.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider sheltering Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider sheltering Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

A-1.6.4.1.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

Advise state and county representatives that offsite protective actions may be necessary depending on wind direction and speed.

NOTE: *Sheltering or evacuation should be considered for those individuals working on property immediately adjacent to the Idaho National Laboratory Site northwest boundary* ***(12,000 m or 7.5 mi)****.*

If wind is from **95 to 170** degrees, make protective action recommendation to Butte County to shelter or evacuate anyone working near the northwest Idaho National Laboratory Site boundary between the junction of State Highway 33 and United States Highway 20/26 and Mile Marker 12 on State Highway 33.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

A-1.6.4.1.5 Basis. The ATR operating specifications require primary coolant concentrations less than 20 μCi/mL. The normal primary coolant radioactivity concentration range is much less than 1.0 μCi/mL. Any primary coolant radioactivity concentration above normal will require some type of treatment. If the PCS volume is pumped to TRA-605 for storage and treatment to remove the radioactivity, there will be some release of airborne radioactivity from the TRA-605 receiving and storage tanks during the 9-hour pumping time. At these concentrations (greater than 2.54E+4 μCi/mL, TRA-605-R-7), the radioactivity in the PCS volume exceeds screening thresholds, and the projected thyroid doses for a ground-level release exceed the PAC at the INL boundary. Since the MTR stack is not operational, the conditions for a general emergency are met.

A-1.6.5 Emergency Action Levels and Scenario Release Designators Cross Reference

Table A-22 shows the link between the EAL and the scenario release designator by which the EAL was generated. The technical bases for the EALs are contained in the analysis sections for the various hazardous material corresponding to the appropriate scenario release designators.

Table A-22. Emergency action levels and scenario release designators forming the technical basis for the emergency action level.

EAL No.	Scenario Release Designator
ATR-605-3.A.1	TRA-605-R-1 and -3
ATR-605-3.SAE.1	TRA-605-R-5
ATR-605-3.SAE.2	TRA-605-R-10
ATR-605-3.SAE.3	TRA-605-R-11
ATR-605-3.SAE.4	TRA-605-R-9 (Class D)
ATR-605-3.SAE.5	TRA-605-R-8 (Class D)
ATR-605-3.GE.1	TRA-605-R-7

A-1.6.6 Emergency Planning Zone

The consequences postulated from the events analyzed in the preceding sections do not indicate that a change to the current emergency planning zone is warranted.

A-2. REFERENCES

1. DOE O 151.1C, "Comprehensive Emergency Management System," United States Department of Energy, November 2, 2005.

2. DOE G 151.1-2, "Technical Planning Basis," United States Department of Energy, November 1, 2006.

3. J. C. Chapman, "Facility Hazard Categorization for the Test Reactor Area Effluent Processing Facility," INEEL/INT-2000-00251, May 2000.

4. K. C. Axelson, "Abandoning the Radioactive Waste System Upgrade Construction Project," ECF 7.4.10-1/6883, Rev. 1, May 27, 2003.
5. LWP-18005, "INL LTHC3 Radiological Facilities and Facility Managers," Rev. 2, August 16, 2007.
6. EDF-6873, "Radiological Analysis for Release From the TRA-605 Effluent Processing Facility," Rev. 0, April 6, 2006.
7. OMM-5.1.2.5.1, Rev. 11, "ATR Primary Coolant System Chemistry Limits," June 5, 2003.
8. United States Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," NRC RG 1.4, Rev. 2, Washington, D.C., June 1974.
9. United States Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risks in United States Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, Washington, D.C., October 1975.
10. Engineering Design File TRA-ATR-1804, "TRA-689 Radiological Analysis," Rev. 0, February 4, 2002.
11. DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities," December 1994.
12. DOE-STD-1027-92, "Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports," December 1992, Change Notice No. 1, September 1997.
13. Radiological Safety Analysis Computer Program (RSAC-6), Version 6.2, INEEL, 2002.
14. EDF-2024, "TRA-710 Stack Evaluation — Excavation for Warm Waste Piping (30 in.) Replacement Project," April 8, 2002.

REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	01/13/2010	All	New document.

CONTENTS

FIGURES

TABLES

B-1. TRA-621, NUCLEAR MATERIALS INSPECTION AND STORAGE FACILITY

B-1.1 Assessment Overview

This appendix documents the emergency management hazards assessment (EHA) for TRA-621, Nuclear Materials Inspection and Storage Facility, to provide the technical basis for facility emergency planning efforts, as required by DOE O 151.1C[1] and supported by DOE G 151.1-2.[2]

B-1.2 Facility and Process Descriptions

TRA-621 is within the Advanced Test Reactor (ATR) Complex protected perimeter security area south and east of TRA-603, Materials Test Reactor Building, in the central section of the ATR Complex. Figure B-1 shows the location of TRA-621 at the ATR Complex. TRA-621, which is a fuel storage, safeguards, and security facility, is used to consolidate the storage of strategic quantities of special nuclear material (SNM) at the ATR Complex and quality assurance fuel inspection and Safeguards and Security nondestructive assay activities. The activities occurring within TRA-621 include receiving, handling, storing, examining, and shipping unirradiated or slightly irradiated (i.e., reading less than 200 mR/hr on the surface of the fuel element) fissile material.[3]

TRA-621 is a vault-type structure with poured reinforced-concrete exterior walls and a precast-concrete beam roof with a poured concrete overlay. It has an underground power distribution system, fire detection systems, an electrical heating system, and a criticality alarm system (CAS). It is divided into four operational areas, including an SNM storage vault, an inspection area, an assay area, and a staging area. Figure B-2 shows the floor plan of TRA-621.

The dimensions of TRA-621 are 88 × 85 × 19.3 ft. The SNM storage vault has a sloping roof with 12.67-ft average inside clearance height. The interior walls are metal stud and sheetrock construction. The wall separating the assay and inspection areas from the staging area has 0.25-in.-lead radiation shield incorporated in the sheetrock. The staging area includes a receiving area. TRA-621 has three access points, including one personnel access door, one cargo access door in the staging area, and one emergency exit door in the inspection area. TRA-621 has no windows.

The SNM storage vault contains areas to store ATR fuel elements and miscellaneous fuels. Areas are also provided to store drums containing fissile material controlled by restricting the drum loading and quantity of drums. The storage configuration in the storage vault is within metal racks with earthquake gates in front. The fuel storage racks are constructed of fire-retardant wood with galvanized-steel fire shields under and along the sides of each fuel storage rack shelf assembly. The shelf assemblies have a fail-safe (cadmium poison) design that prevents an accidental criticality. Figure B-3 shows a single storage rack with the fire shields installed, but with only the top shelf assembly installed. As part of the facility criticality safety controls, each shelf assembly includes two cadmium sheets sandwiched between other structural material; there is one cadmium sheet across the top and one cadmium sheet across the bottom of each shelf assembly. The storage vault has an emergency firewater floor drain that drains to the ground under the vault.

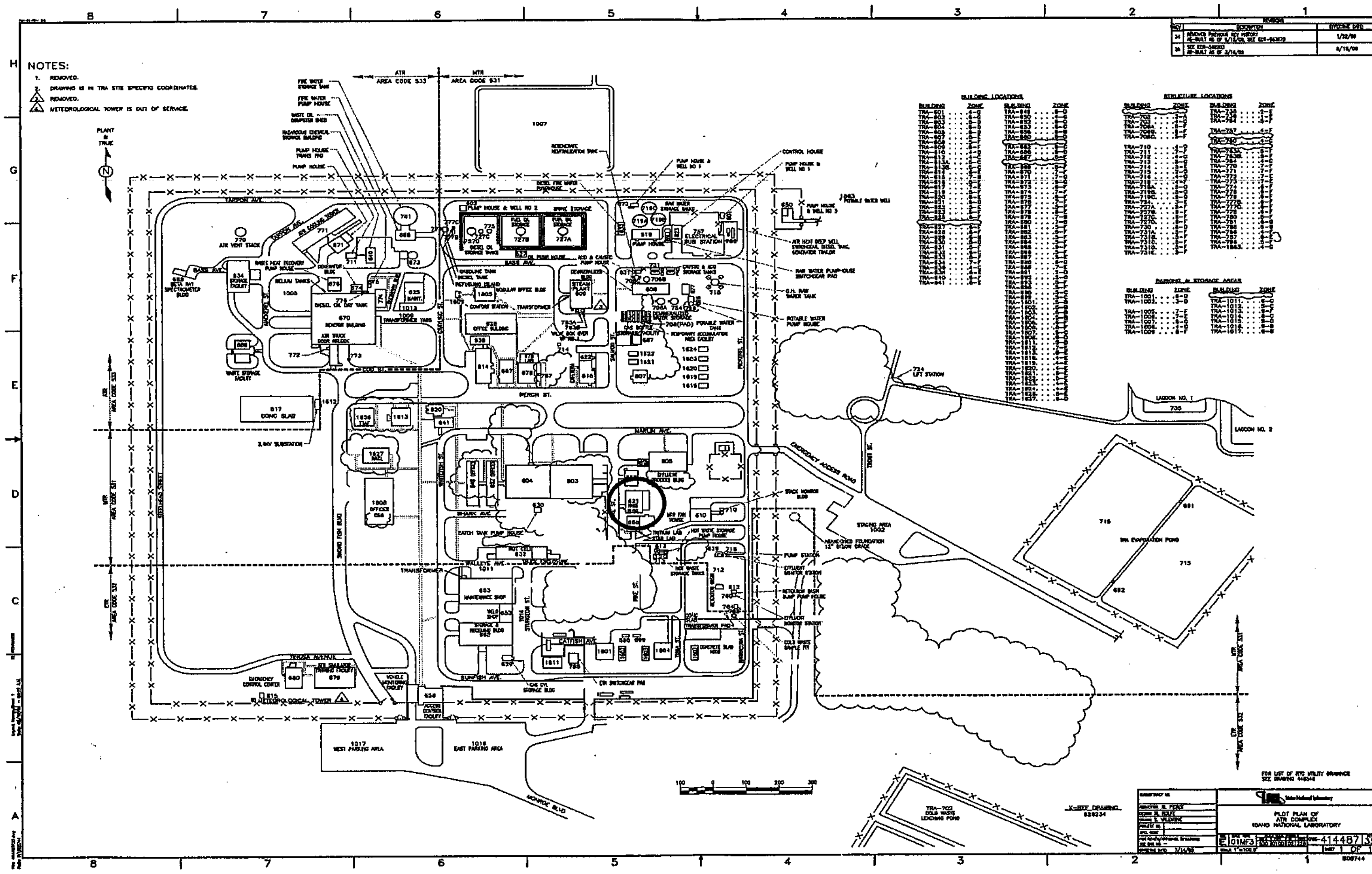


Figure B-1. Map showing location of TRA-621, Nuclear Materials Inspection and Storage Facility, at the Advanced Test Reactor Complex.

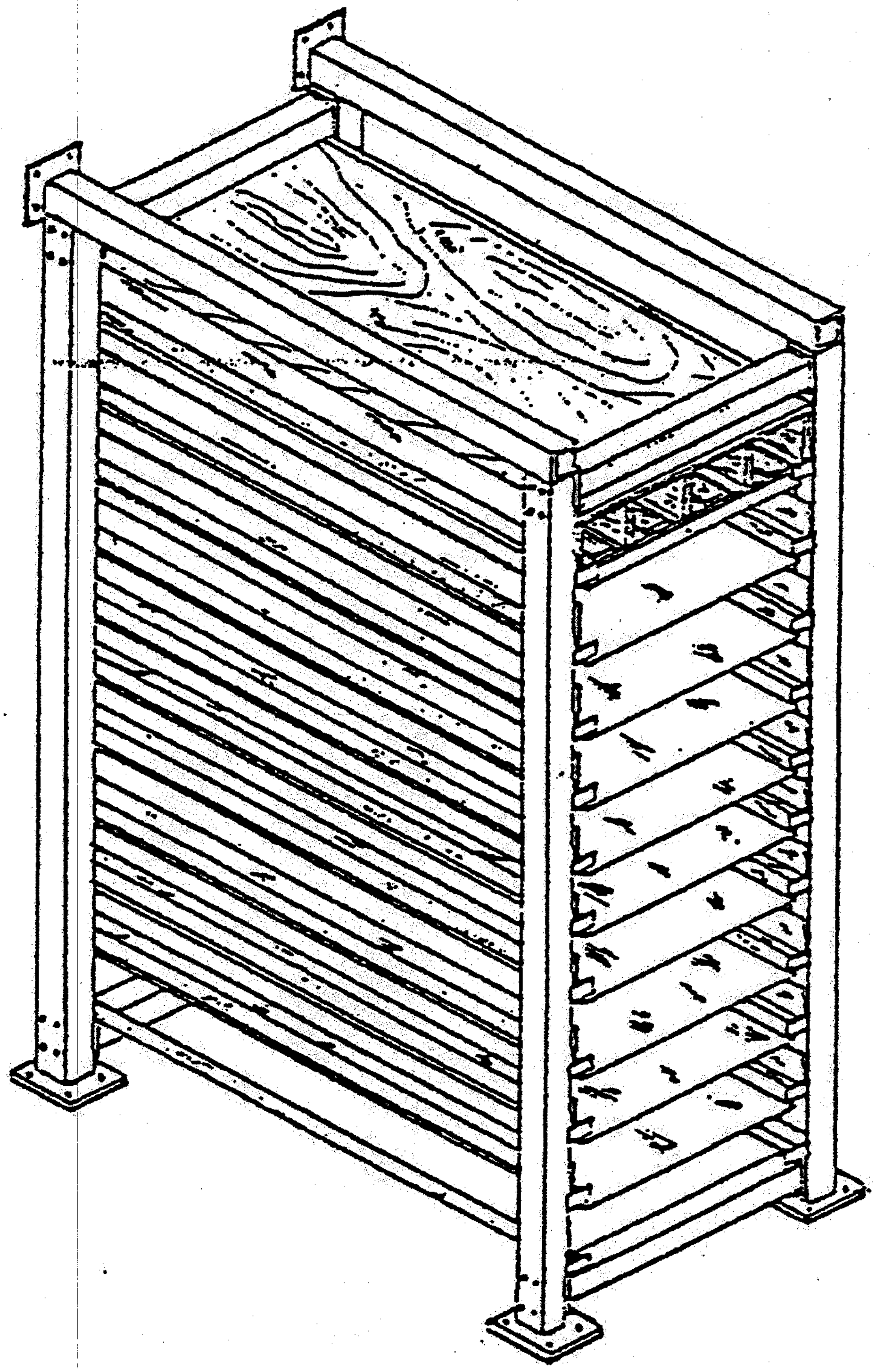


Figure B-3. Fuel storage rack with fire shields (typical).

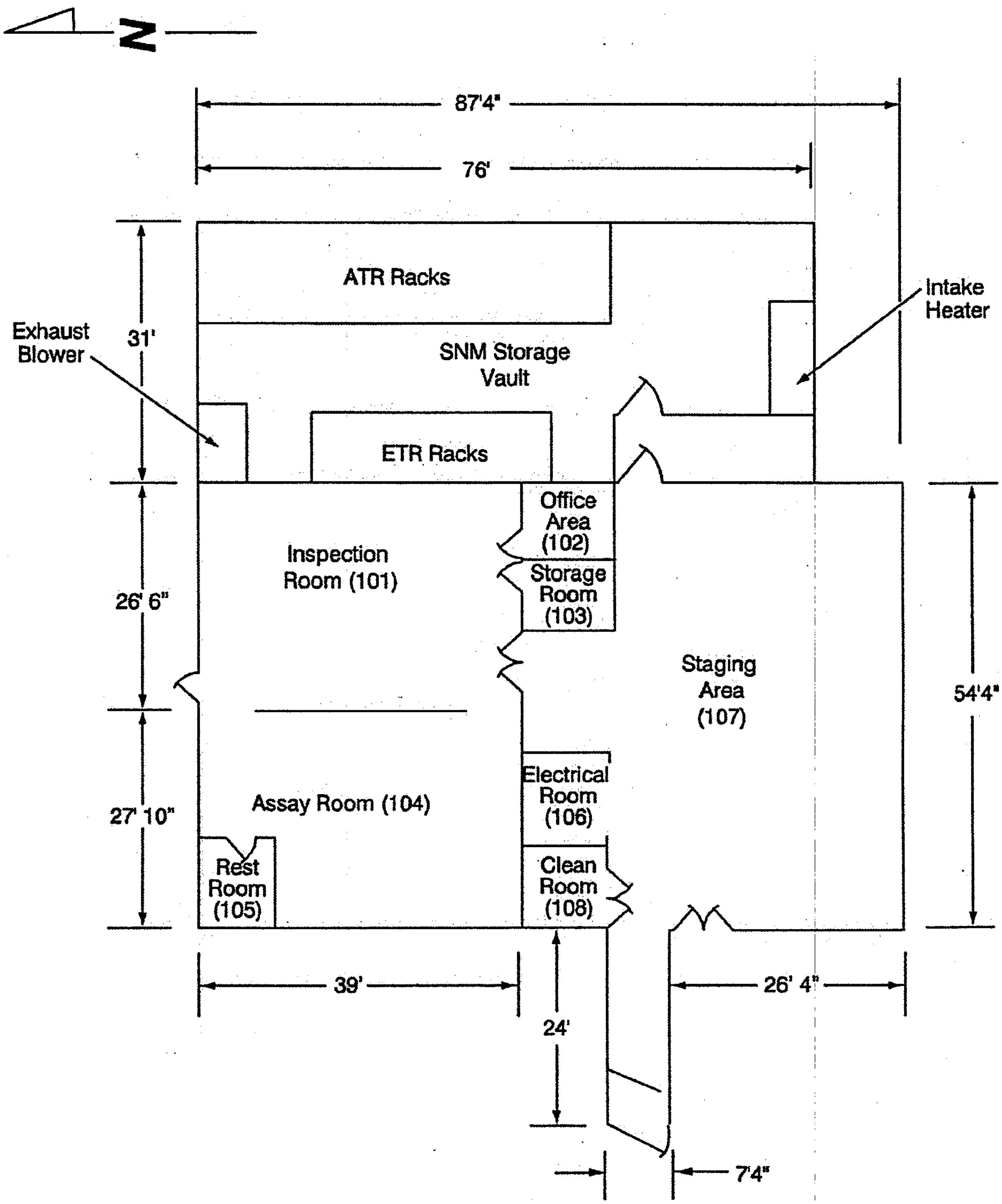


Figure B-2. Floor plan for TRA-621, Nuclear Materials Inspection and Storage Facility.

The fuel storage rack design allows the storage of ATR fuel assemblies and other miscellaneous fissile material forms. Engineering Test Reactor fuel storage racks are 4 × 11 arrays, approximately 200 cm (80 in.) high. There are 11 Engineering Test Reactor racks, creating a 44 × 11 array of storage positions. ATR fuel storage racks are also 4 × 11 arrays, approximately 200 cm (80 in.) high. There are 18 ATR racks, creating a 72 × 11 array of storage positions. Transport racks designed to hold ATR fuel elements in a critically safe configuration are provided to safely transport fuel elements between the SNM storage vault and other areas within TRA-621. The transport racks maintain a linear array and were originally designed to hold eight fuel elements, but have been modified, blocking off four positions.

TRA-621 activities include fissile material transfers that may include some repackaging, inspecting fissile material, storing fissile material, and performing safeguards-related assays of fissile material. Fissile material permitted in TRA-621 include neutron check sources used for calibrating instruments. TRA-621 is limited to 1,838 kg of nominally 93% enriched U-235, including up to 132 ATR fuel elements or equivalent that could have been slightly irradiated but are reading less than or equal to 200 mR/hr on the surface of the fuel element.

For the purposes of this EHA, TRA-621 is analyzed as a single entity.

B-1.3 Identification of Hazards

The current radiological inventory in TRA-621 is primarily comprised of (1) fissile material in the form of reactor fuel or oxide, (2) fission products contained within the fuel generated as a result of reactor operation, (3) potentially contaminated fuel elements, and (4) sealed neutron sources. The americium-beryllium neutron source (approximately 4.39 Ci) is used for testing the CAS and may be stored at TRA-621. This neutron source produces radiation fields of 500 mrem/hr on contact and 80 mrem/hr at 1 ft. Other sealed neutron sources may be brought into TRA-621, which would produce similar dose rates. Also located in TRA-621 are some instruments that have internal sources used for calibration. All these sealed sources are controlled in accordance with LRD-15001.[4]

For TRA-621, the material-at-risk (MAR) is a maximum projected inventory of 1,838 kg of U-235. In general, the U-235 is unmoderated, unirradiated, and nominally 93% enriched, including 132 irradiated ATR fuel elements or equivalent that have been decayed until radiation readings are less than 200 mR/hr on contact with the fuel element.

The cadmium is in the form of solid plates imbedded in fuel storage racks, transport racks, and 4-element ATR fresh fuel shipping containers [USA 9099/B(U)F (NRC)]. The amount of cadmium in the SNM storage vault storage racks is 132 kg (291 lb); four ATR transport racks is 10.8 kg (23.8 lb), and 12 4-element ATR fresh fuel shipping containers is 101 kg (224 lb) and, as such, is not in a readily releasable form. The total for TRA-621 was determined to be 244 kg (539 lb). The largest quantity of cadmium is in the fuel storage racks in the vault. Under normal use conditions in TRA-621, cadmium is in a monolithic solid form that does not have a vapor pressure, and as such, screens out as an airborne hazard. The Department of Energy guidance and Department of Energy Complex-wide consensus on the implementation of the guidance for the fire scenario as a release mechanism for hazardous material, is that the hazardous material released is no more toxic than the toxic combustion products created by ordinary fires. Only in the case where a documented fire hazards analysis suggests that there is an extraordinary toxic release potential, should the toxic material release be addressed in an EHA. Therefore, no further analysis is required for cadmium.

The lead is in the form of solid 1/8-in. sheets that line the walls between the staging area and the nuclear material inspection area of TRA-621 and as lead bricks. This material is not in a readily releasable form. The total amount of lead in TRA-621 was calculated to be 6.4E+3 kg (1.4E+4 lb) of which 41.5 kg (9.1E+1 lb) is lead bricks and the remaining (6.37E+3 kg or 1.4E+4 lb) is in the form of sheets.

Uranium is toxic and must be considered as a hazardous material in addition to a radiological hazard. The majority of the uranium is fuel in bulk pieces that are not considered to be in a readily releasable form. However, there is some uranium powder stored in TRA-621 and, as such, is in a readily releasable form, but the quantity, 26 lb, is less than the screening threshold and is not further considered in this appendix.

Table B-1 lists the radiological hazardous material that is stored, used, or produced in TRA-621 that is retained for further analyses based on the screening criteria presented in the main document.

Table B-1. Radiological hazardous material requiring further analyses.

Location	Material	Maximum Quantity (Ci)	Screening Threshold (Ci)	Notes
TRA-621 (All)	Fission products	Greater than Hazard Category (HC) 2	Greater than HC-3	TRA-621 is categorized as an HC-2 facility
TRA-621 (All)	Fissile material	Greater than HC-2	Greater than HC-3	TRA-621 is categorized as an HC-2 facility

B-1.4 Hazardous Material Characterization and Analyses

The hazardous material listed in Table B-1 is addressed below by location.

B-1.4.1 TRA-621 (All)

B-1.4.1.1 Radiological Hazardous Material

B-1.4.1.1.1 Properties. The properties for TRA-621 radiological hazardous material are listed in Table B-2.

Table B-2. Radiological properties for TRA-621.

Uranium and radioactive mixed fission products	Fission products from low power irradiated ATR fuel and fissile material from other odd lot fuel.
Physical form	Solid containing some gaseous fission products and some powder form. Each ATR fuel element is made up of 19 curved plates covered by an aluminum alloy cladding. The fission products are contained within the fuel matrix in the form of gases, volatile, semi-volatile, or particulate material.
Particle size	Respirable.
Flammability	Not applicable (N/A).
Reactivity	N/A.
Density	N/A.
Special firefighting concerns	None.
Health concerns	Inhalation is the primary route for personnel exposure. Thyroid committed dose equivalent is controlling for Protective Action Guide (PAG) evaluation due to the higher concentration of radioactive iodine isotopes present in the fresh fission products and the fact that the thyroid gland concentrates iodine. Fission products are primarily beta-gamma emitters. There may be industrial hygiene type concerns (not emergency preparedness concerns) related to heavy metal toxicity related to powdered uranium.

B-1.4.1.1.2 Conditions of Storage and Use. Fresh ATR fuel, low-power irradiated ATR fuel, Training Research Isotope (General Atomic)-Fuel Life Improvement Program fuel, and some odd lot fuel pieces are received, inspected, and stored in the SNM storage vault. For criticality control, the fuel elements are placed in specially designed storage racks located in the vault.

Table B-3 presents the radionuclides that are the primary contributors to dose.

Table B-3. Radionuclides that are the primary contributors to dose.

Isotope	Total Inventory (kg)	Specific Activity (Ci/g)	Curies of Isotope (Ci)
U-235	1,838	2.2E−6	4.01E+0
U-234	23.7	6.2E−3	1.52E+2
U-236	13.8	6.5E−5	8.90E−1
U-238	119.5	3.4E−7	4.01E−2

Under normal operating conditions, the primary barriers are the fuel element cladding, storage containers for fuel pieces or powdered uranium, monolith form of fuel elements, and high-efficiency particulate air (HEPA) filters in the SNM storage vault.

A secondary barrier is the TRA-621 building structure, which consists of poured concrete walls, a concrete beam roof with a concrete overlay, and no windows. The secondary barrier provides surfaces for plate-out, which can be assigned a leak path factor (LPF).

TRA-621 safety systems consist of smoke detectors in the SNM storage vault and staging area, a wet pipe sprinkler fire suppression system in all areas except the security corridor and vestibule leading to the storage vault, a Halon fire suppression system in the storage vault, manual fire extinguishers, manual fire alarms, and an exhaust damper that closes upon activation of the Halon fire suppression system in the storage vault.

Some of the administrative controls that are in place include the following:

- A maximum of three fuel handling units may be out of storage at one time, where a fuel handling unit is one fuel element, or a maximum of nineteen loose fuel plates containing up to a maximum of 1,085 g of U-235 may be out of storage
- A maximum of one fuel handling unit in one storage position
- No uranium powder may be removed from approved storage in the clean room, and individual packages of uranium powder containing greater than 4,000 g total uranium may not be removed from storage containers in the storage vault

The TRA-621 CAS consists of a central data acquisition system, an uninterruptible power supply, three detector clusters, and other miscellaneous support material. Each detector cluster has three neutron detectors associated with it to incorporate a two out of three logic circuitry. The detectors are located in the storage vault, quality assurance inspection area, and staging area.

B-1.4.1.1.3 Barrier and Failure Mode Analyses. For the purposes of barrier and failure mode analyses for radiological material, the accident scenarios are divided into two types: (1) airborne release of radiological material (fission products), and (2) criticality and associated release of radiological material. The two accident scenarios are based on those described in Chapters 3 and 6 of SAR-154.[3] The results of the barrier and failure mode analyses are detailed in the following subsection; SAR-154, TRA-NMIS-1771,[5] and Engineering Design File TRA-NMIS-1731[6] served as the principal references in developing the scenarios.

The radiological barrier and failure mode analyses for TRA-621 are summarized in Table B-4.

Table B-4. Radiological failure modes and barriers for TRA-621.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA621RR-1	Radioactive fission products and fissile material	Fuel cladding or storage container	Explosion/fire damages fuel cladding (melt or mechanical), SNM storage vault walls (mechanical or burned), and HEPA filters (blown out or burned)	Explosion or large fire in SNM storage vault	100% fuel cladding damage with release fractions taken from DOE-HDBK-3010-94,[7] Table 6-10	Degraded, as HEPA filters and SNM storage vault walls are breached; only plate-out is a mechanism for removal
TRA621CRIT-1	Prompt neutron pulse and radioactive fission products	Physical and administrative controls on fuel storage and fuel cladding	Fuel mishandling event, which positions seven or more fuel elements in a critical configuration in SNM storage vault, or fire burns fuel storage racks causing a critical configuration	Fire burns fuel storage racks, mechanical system fails while moving fuel elements, or failure to implement technical specification requirements causes criticality to occur	1E+18 fissions and 20-minute halogen in-growth period	Essentially none, as HEPA filters may be compromised and SNM storage vault walls do not provide shielding for neutrons

B-1.4.1.1.4 Scenario Bases and Analytical Assumptions

1. **Scenario TRA621RR-1, TRA-621 Fire (Lasting 20 Minutes or Longer) Causes a Radiological Release**

Detailed Scenario Description	A fire in the SNM storage vault damages cladding of stored fuel elements and causes a release of radioactivity.
Material-at-Risk	See Table B-5. The MAR represents a truncated list of radionuclides that contribute to greater than 99% of dose.
Release Characteristics	Up to 132 low power radiation (82.5 mW for 24 hours and decayed for 24 hours) ATR fuel elements and residual (unirradiated) fissile material are stored in the fuel storage rack in the SNM storage vault.[3,8] Six fuel transport racks are used for movement of the fuel elements from the staging area to the storage vault. A fire in the storage vault damages the fuel cladding and releases fission products via the fire damaged exhaust filtration system, which is located on the side of TRA-621. Some plate-out on the surfaces of the storage vault is expected to occur before release to the environment. The release is modeled as ground level.
Airborne Release Fraction	4.0E−4 for tetravalent (DOE-HDBK-3010-94, Table 6-10, p. 6-23).
Respirable Fraction	1.00E+0 (assumption based on information in DOE-HDBK-3010-94).
Damage Ratio	1.00E+0 (assumes fuel cladding damage).
Leak Path Factor	5.5E−1 (EPA 550-B-99-009,[9] Appendix D, Section D.1.2).
Source Term	The source term (ST) shown in Table B-5 was developed according to the following equation: ST = MAR × DR × LPF × ARF × RF Where DR = damage ratio ARF = airborne release fraction RF = respirable fraction.
Modeling Software and Inputs	Radiological Safety Analysis Computer (RSAC) Program, Version 6.2.[10] Ground-level release.

Table B-5. Source term for scenario release designator TRA621RR-1.

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
U-234	1.52E+2	1.00E+0	5.50E−1	4.00E−4	1.00E+0	3.33E−2
U-235	4.01E+0	1.00E+0	5.50E−1	4.00E−4	1.00E+0	8.82E−4
U-236	8.90E−1	1.00E+0	5.50E−1	4.00E−4	1.00E+0	1.96E−4
U-238	4.01E−2	1.00E+0	5.50E−1	4.00E−4	1.00E+0	8.83E−6

2. Scenario TRA621CRIT-1, TRA-621 Criticality

Detailed Scenario Description: Fuel mishandling event, which positions seven or more fuel elements in a critical configuration in the SNM storage vault, or fire burns fuel storage racks causing a critical configuration.

Material-at-Risk: Uranium and fission products contained in nine low power irradiated fuel elements (assume 1E+18 fissions during criticality).

Release Characteristics: The criticality event assumes that 10% fuel damage occurs. A minimum of seven fuel elements are needed to sustain a criticality. For an operational event, the HEPA filters remain intact; however, for a fire or external event initiator, the HEPA filters would be damaged or there could be a breach in the building structure. A ground-level release path is assumed. Plate-out may occur on the wall, floor, and ceiling surfaces.

Airborne Release Fraction: 4.0E−4 for tetravalent (DOE-HDBK-3010-94, Table 6-10, p. 6-23).

Respirable Fraction: 1.00E+0 (assumption based on information in DOE-HDBK-3010-94).

Damage Ratio: 1.00E−1 (assumes 10% fuel cladding damage).

Leak Path Factor: 5.5E−1 (EPA 550-B-99-009, Appendix D, Section D.1.2).

Source Term: Table B-6 provides the RSAC batch file used in the criticality calculations. The cloud gamma from these fresh fission products contribute up to 90% of the external dose at 100 m.

Modeling Software and Inputs: RSAC Program, Version 6.2. Ground-level release. A disc operating system batch file containing the criticality algorithm with a 20-minute halogen in-growth period following the 1E+18 fissions was used to calculate the dose due the radiological release following the criticality.

Table B-6. Radiological Safety Analysis Computer batch file for scenario release designator TRA621CRIT-1.

```
* Criticality in the NMIS-effects to downwind workers ground level
# External file for input for existing fuel
2000,0
2002,lowpower
2999
# Multiply by 9.0 CANISTERS involved in the crit
1000
1001,1,0,0
1004,0,9.
1999
# Criticality event 1.0E+18 Fissions
1000
1001,1,0,0
1003,0,3.3E10,0.001
# release 0.1% solids and 100% nobles
1200,1,1
1201,1.,0,1.E3,1.,1.
1003,0,1.E-6,1200.
1200,20,1
1201,0,1.,0,0,0
1999
# Bring solids inventory back to 100%
1000
1001,1,0,0
1004,-1,1000.,1.,1.,1000.,1000.
1999
# Fraction for 100%release
1000
1001,1,0,0
1004,0,1.
1999
# Fuel gap fractions
1000
1001,1,0,0
1004,1,1.
1101,1,5.E-1,4,3.E-2,6,6.E-4,11,2.E-1,14,6.E-4
1102,15,6.E-4,16,7.E-2,17,5.E-2,18,5.E-1,19,2.E-1
1103,20,3.E-2,21,6.E-4,24,3.E-2,25,3.E-2,26,3.E-2
1104,27,3.E-2,28,2.E-3,30,4.E-3,33,4.E-3,34,7.E-2
1105,35,5.E-2,36,5.E-1,37,2.E-1,38,3.E-2,39,6.E-4
1106,40,4.E-4,41,3.E-2,42,3.E-2,43,3.E-2,44,2.E-3
```

Table B-6. (continued).

```
1107,45,2.E-3,46,2.E-3,47,4.E-3,48,4.E-3,49,4.E-3
1108,50,4.E-3,51,4.E-3,52,7.E-2,53,5.E-2,54,5.E-1
1109,55,2.E-1,56,3.E-2,57,6.E-4,58,4.E-4,59,6.E-4
1110,60,6.E-4,61,6.E-4,62,6.E-4,63,6.E-4,64,6.E-4
1111,77,2.E-3,81,4.E-3,83,4.E-3,84,7.E-2,85,5.E-2
1112,86,5.E-1,87,2.E-1,88,3.E-2,89,6.E-4,90,4.E-4
1113,91,4.E-4,92,4.E-4,93,4.E-4,94,4.E-4,95,6.E-4
1114,98,6.E-4
1999
# Release to workers downwind
5000
# Dose summary
5001,1.04,0,400.,1.099E3,0,1
5002,.001,.01,0.,.001,.001
5101,100.,200.,300.,400.
5102,500.,600.,700.,800.
5201,1.,0
5400,2,2.68E1,7.62,0
5410,2,6,0
5999
# initiate dose summary option
3000,1
# Inhalation
7000,0,-2,1,0,2
7001,3.33E-04,0,0,0,0
7002,10,24
7999
# Ground surface exposure for 40/168
7000,4,-2,1,0,2
7001,0,0,2.38E-1,0,0
7002,10,24
7999
# Cloud gamma
9000,0,0
#Summary
3000,2
```

B-1.5 Evaluation Results

B-1.5.1 Radiological Hazardous Material Release Results

Radiological hazardous material release results for 95% worst-case and 50% typical weather conditions as described in the main document are presented in Tables B-7 and B-8.

Table B-7. Radiological release scenario calculation results for 95% worst-case meteorology.

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA621RR-1	Explosion/fire in SNM storage vault damages fuel cladding, releasing radioactive material	3.54E+1	3.54E+0	3.48E+0	1.41E−2	1.46−3	310	Not exceeded	Site area emergency
TRA621CRIT-1 (Prompt neutron)	Criticality in SNM storage vault	5.80E+1[b]	5.10E+0[b]	5.00E+0[b]	Not calculated	Not calculated	N/A	N/A	Unclassified operational emergency (UOE)
TRA621CRIT-1 (Fission product release)	Criticality in SNM storage vault	3.13E+0	3.13E−1	1.90E−1	Not calculated	Not calculated	100	Not exceeded	Alert (greater than 10% of PAG)

a. TEDE = total effective dose equivalent
INTEC/ICDF = Idaho Nuclear Technology and Engineering Center/Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility.

b. Derived from TRA-NMIS-1276.[11]

Table B-8. Radiological release scenario calculation results for 50% average meteorology.

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (122 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA621RR-1	Explosion/fire in SNM storage vault damages fuel cladding, releasing radioactive material	2.27E−1	2.27E−2	1.89E−2	4.09E−4	6.36E−5	N/A	Not exceeded	Not an emergency
TRA621CRIT-1 (Prompt neutron)	Criticality in SNM storage vault	5.80E+1[a]	5.10E+0[a]	5.00E+0[a]	Not calculated	Not calculated	N/A	N/A	UOE
TRA621CRIT-1 (Fission product release)	Criticality in SNM storage vault	5.06E+0	5.06E−1	4.21E−1	Not calculated	Not calculated	100	Not exceeded	Alert (greater than 10% of PAG)

a. Derived from TRA-NMIS-1276.

B-1.6 Emergency Classes, Protective Actions, and Emergency Action Levels

Based on the analyses documented above, the following emergency action levels (EALs) are identified for TRA-621.

B-1.6.1 Unclassified Operational Emergency — Emergency Action Levels

Unless otherwise noted, the generic UOE EALs are covered by a separate appendix to this EHA.

Facility-specific UOE EALs are covered in this appendix.

B-1.6.1.1 ATR-621-10.OE.1

B-1.6.1.1.1 Event Description

Any unplanned criticality,

B-1.6.1.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

activation of criticality alarm system RM-50-1 with a voice message from the autodialer stating "criticality alarm,"

AND

security camera operator confirmation of flooding and/or fuel disruption from containers,

OR

TRA-621 personnel confirmation of conditions to cause criticality,

OR

radiological control technician confirmation by radiation survey.

B-1.6.1.1.3 Onsite Protective Actions

Evacuate nonessential personnel from the Advanced Test Reactor Complex.

Control nonessential vehicle and personnel access to the Advanced Test Reactor Complex.

B-1.6.1.1.4 Offsite Protective Action Recommendations

None.

B-1.6.1.1.5 Basis. A criticality (TRA621CRIT-1) is considered a UOE, unless the criticality causes a radioactive material release in quantities that cause doses in excess of the protective action criteria. The protective action distance (PAD) is based on the radiological material release associated with the post-criticality 20-minute in-growth period. Normally, the PAD is 100 m for a criticality. However, due to the decommissioning and demolition activities at the ATR Complex, ATR Complex management has decided to implement an ATR-Complex-wide evacuation of nonessential personnel.

B-1.6.2 Alert — Emergency Action Levels

B-1.6.2.1 ATR-621-3.A.1

B-1.6.2.1.1 Event Description

Any unplanned criticality resulting in a radioactive material release,

B-1.6.2.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

activation of criticality alarm system RM-50-1 with a voice message from the autodialer stating "criticality alarm,"

AND

security camera operator confirmation of flooding and/or fuel disruption from containers,

OR

TRA-621 personnel confirmation of conditions to cause a criticality,

OR

radiological control technician confirmation by radiation survey.

B-1.6.2.1.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel.

Control nonessential vehicle and personnel access to the Advanced Test Reactor Complex.

B-1.6.2.1.4 Offsite Protective Action Recommendations

None.

B-1.6.2.1.5 Basis. The projected radiological dose from a 20-minute in-growth period (TRA621CRIT-1) is greater than 10% of the PAG at 100 m, which meets the criteria for an alert classification. The PAD is based on the radiological material release associated with the post-criticality 20-minute in-growth period. Normally, the PAD is 100 m for a criticality. However, due to the decommissioning and demolition activities at the ATR Complex, ATR Complex management has decided to implement an ATR-Complex-wide evacuation of nonessential personnel.

B-1.6.3 Site Area Emergency — Emergency Action Levels

B-1.6.3.1 ATR-621-3.SAE.1

B-1.6.3.1.1 Event Description

Fire inside TRA-621 that has **not** been extinguished in 20 minutes after Idaho National Laboratory Fire Department fire suppression activities began, causing a release of radioactive material,

B-1.6.3.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the ongoing fire 20 minutes after fire suppression began,

OR

radiological control technician survey indicates an airborne radiological release.

B-1.6.3.1.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **310 m (1,017 ft or 0.2 mi)** of the Advanced Test Reactor Complex.

Control nonessential vehicle and personnel access to the evacuated area.

Depending on wind direction, consider relocating the emergency control center to the alternate emergency control center at CF-609.

B-1.6.3.1.4 Offsite Protective Action Recommendations

None.

B-1.6.3.1.5 Basis. An explosion/fire damages cladding to 100% of the 132 fuel elements and SNM in the SNM storage vault fuel storage racks (TRA621RR-1). The explosion/fire breaches the HEPA filter system and storage vault walls, so the radioactive material release is direct to the environment except for plate-out on the vault surfaces.

NOTE: *The cadmium release associated with this event controls the PAD.*

B-1.6.4 General Emergency — Emergency Action Levels

None.

B-1.6.5 Emergency Action Levels and Scenario Release Designators Cross Reference

Table B-9 shows the link between the EALs and the scenario release designators used as the basis for the EAL.

Table B-9. Emergency action levels and scenario release designators.

EAL No.	Scenario Release Designator
ATR-621-10.OE.1	TRA621CRIT-1
ATR-621-3.A.1	TRA621CRIT-1
ATR-621-3.SAE.1	TRA621RR-1

B-1.6.6 Emergency Planning Zone

The maximum PAD considered in this EHA for TRA-621 was 1,800 m. That distance is less than the existing 16-km ATR Complex emergency planning zone. No change to the existing emergency planning zone size is recommended based on this EHA.

B-2. REFERENCES

1. DOE O 151.1C, "Comprehensive Emergency Management System," United States Department of Energy, November 2, 2005.
2. DOE G 151.1-2, "Technical Planning Basis," United States Department of Energy, July 11, 2007.
3. SAR-154, "Safety Analysis Report for the Nuclear Materials Inspection and Storage (NMIS) Facility TRA-621," Rev. 5, February 20, 2007.
4. LRD-15001, "Radiological Control Manual," Rev. 1, December 23, 2008.
5. TRA-NMIS-1771, "Consequence Analysis Calculations for Cadmium and Lead Releases During a NMIS Fire," Rev. 0, October 30, 2001.
6. Engineering Design File TRA-NMIS-1731, "Accident Calculations for NMIS Fire Release Scenario and Criticality Release Scenario," Durante, R. P., October 30, 2001.
7. DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities," December 1994.
8. Engineering Design File TRA-ATR-1729, "ORIGEN2 Calculations of Radionuclide Inventory of an ATR Fuel Element Irradiated at Low Power Levels," May 7, 2001.
9. EPA 550-B-99-009, "Risk Management Program Guidance for Offsite Consequence Analysis," United States Environmental Protection Agency, April 1999.
10. Radiological Safety Analysis Computer Program (RSAC-6), Version 6.2, INEEL, 2002.
11. TRA-NMIS-1276, "12 Rad Boundary Calculation for a NMIS Criticality," Rev. 0, November 10, 1997.

REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	01/13/2010	All	New document.

CONTENTS

FIGURES

TABLES

D-1. TRA-670, ADVANCED TEST REACTOR BUILDING

D-1.1 Assessment Overview

This appendix documents the emergency management hazards assessment (EHA) for TRA-670, Advanced Test Reactor (ATR) Building, to provide the technical basis for facility emergency planning efforts, as required by DOE O 151.1C[1] and supported by DOE G 151.1-2.[2]

D-1.2 Facility and Process Descriptions

ATR is the major project operating at the ATR Complex. ATR is located within TRA-670, which is in the northwest part of the ATR Complex. Figure D-1 shows the location of TRA-670. Figure D-2 shows the floor plan of the ATR confinement area.

ATR is designed to test advanced nuclear fuel systems and material for Reactor Program sponsors. It provides a high neutron flux (1E+18 thermal neutrons/cm^2/s) environment for flux traps that may contain in-pile tubes for high-pressure loops or other flux trap irradiation facilities for irradiation of sample and isotope production capsules.

ATR is a light-water moderated and cooled 250-MW$_{th}$ nuclear reactor designed to study the effects of intense irradiation on samples of reactor material, especially fuels. The unique design of ATR includes a core containing 40 fuel elements in a serpentine arrangement; beryllium reflector blocks that contain cutouts for the outer shim control cylinders and capsule irradiation facilities; the neck shim rod housing, which contains the neck shim rods, regulating rods, fixed shim rods, and irradiation facilities; safety rods; and in-pile pressure tubes, which are occupied by loop experiments. ATR design provides high thermal neutron density and the ability to adjust neutron density within the test region, relatively constant axial flux distribution, and the ability to vary the fast-to-thermal-flux ratio. ATR uses the flux trap principle and the ATR fuel geometry forms nine cylindrical flux traps, one on each side of the core and the others located in the northeast, southeast, northwest, and southwest corners and one in the center. The ATR vessel is a pressurized, solid-stainless-steel container that houses the reactor internals, core, and reflector assembly. The design parameters of the primary coolant system (PCS) are 390 psig and 240°F, with a capacity of approximately 83,000 gal of water, not including the water in the bypass demineralizer system. The normal operating parameters are an inlet pressure of approximately 365 psig at the core inlet, an inlet temperature of less than 125°F, a core differential pressure of 100 psid for a three-pump operation or equal to or greater than 77 psid for a two-pump operation, and a flow of 47,000 to 48,000 gpm for a three-pump operation or 42,000 to 43,000 gpm for a two-pump operation. Four primary coolant pumps (PCPs) are available; however, the normal pump configuration is either three PCPs running with one in standby or two PCPs running with two in standby. All four PCPs are powered from 4,160-Vac commercial power. The PCPs discharge into a common 36-in. reactor supply line. A butterfly valve in the supply line adjusts flow through the vessel. The position of the butterfly valve is set to control the core differential pressure. Flow varies depending on core and experiment loadings. To prevent complete loss of primary coolant flow, a minimum stop on the valve stem prevents complete closure. This minimum stop was designed to allow 10,000-gpm flow with a standard valve disc. Also, six holes drilled through the valve disc allow 20,000-gpm flow with one PCP operating even if the valve is in the fully closed position. Furthermore, the butterfly valve is designed to fail in the "as is" position.

ATR is housed in TRA-670, which is approximately 200 × 200 ft and extends 60 ft above and below grade. Two full basements house experiment cubicles and control panels, a subpile room, a control rod

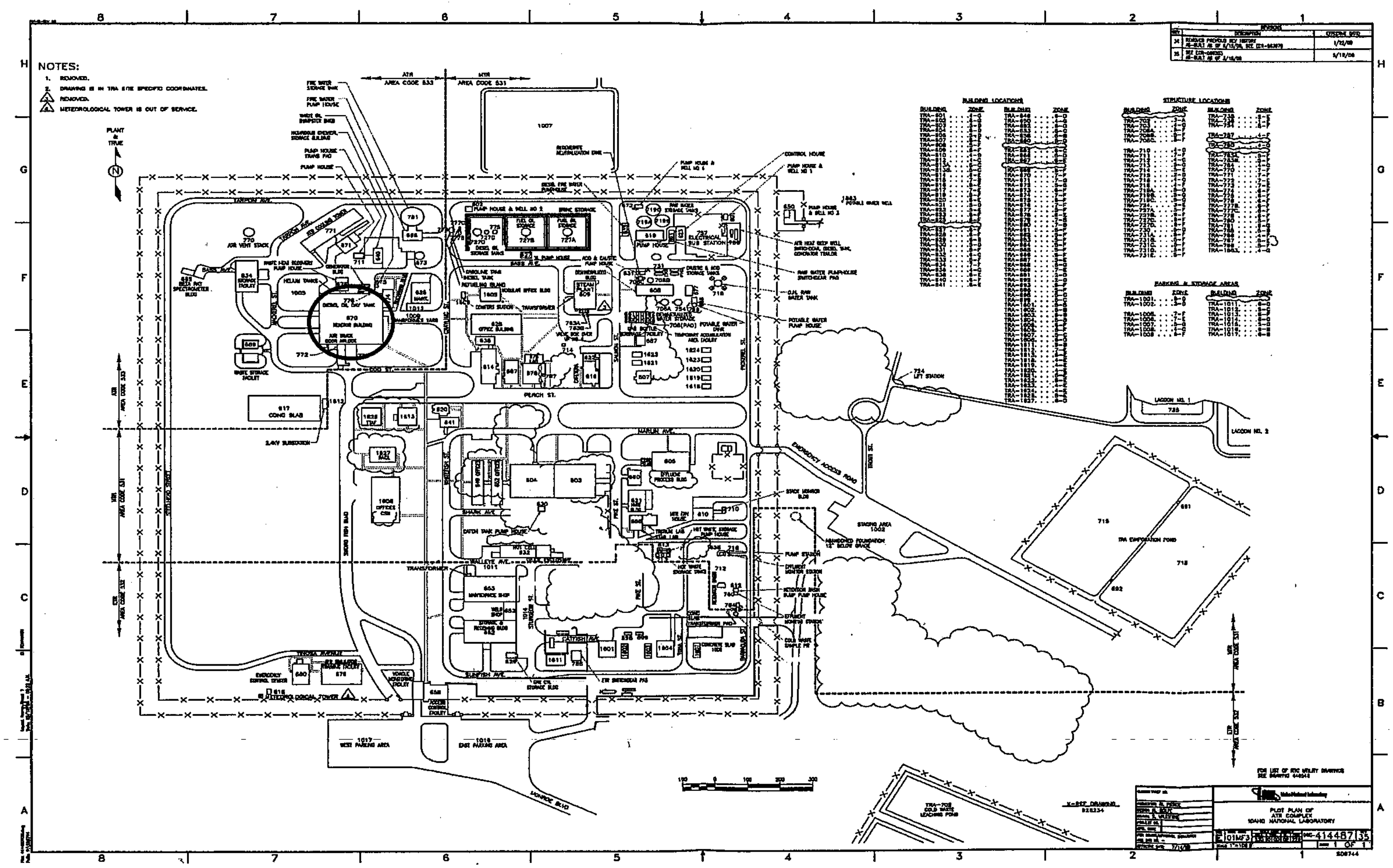


Figure D-1. Map showing location of TRA-670, Advanced Test Reactor Building, at the Advanced Test Reactor Complex.

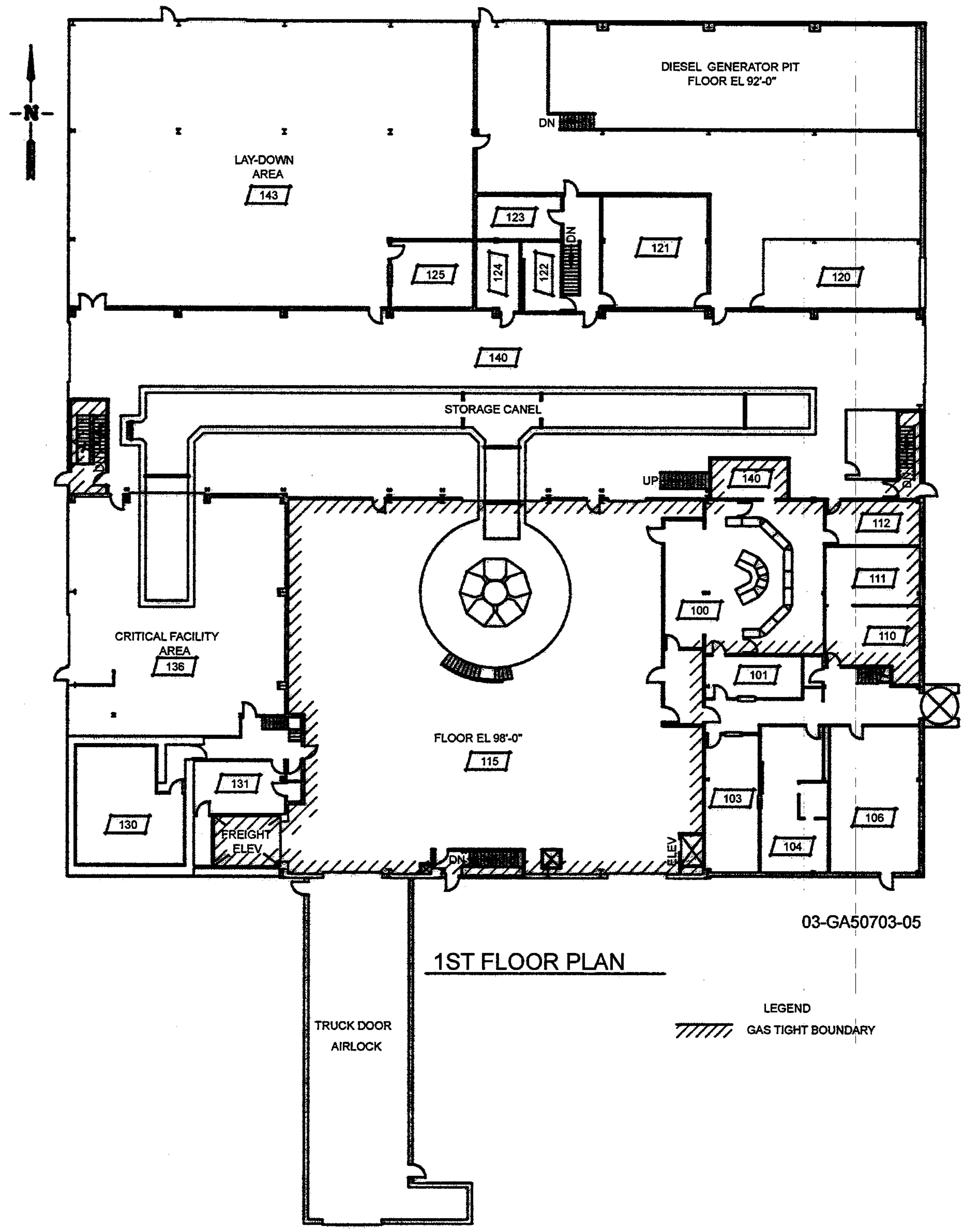


Figure D-2. Floor plan for the Advanced Test Reactor confinement area.

access room, reactor primary system equipment, and ventilation equipment. The TRA-670 substructure is constructed of heavily reinforced concrete supported by subsurface rock. The superstructure is constructed of steel frame with exterior walls primarily of insulated aluminum sandwich panels. A major portion of TRA-670 is a low leakage confinement that includes the ATR operating area, Control Room, and basements. The perimeter surfaces of the confinement are of concrete, masonry, or welded steel plate with all penetrations, including doors, hatches, and ventilation ducts, carefully sealed or gasketed. Also in TRA-670, the ATR canal, constructed of reinforced concrete, is used to temporarily store irradiated ATR fuel and experiments. The canal contains fuel storage grids; general storage grids; a gamma facility grid; one-piece grids; X, S, and T basket grids; trash liners; an experiment viewing window; a saw table; an in-pile tube storage rack; a deep-well barrier; a safety rod storage rack; and a cask loading area. The cask loading area is situated over reinforced concrete walls to support large shipping casks. A cask drop impact absorbing pad is installed in the canal cask loading area to prevent canal floor failure if a cask drops. Stored ATR fuel in the canal, subject to melting if uncovered, is protected by isolating bulkheads and redundant emergency canal makeup systems actuated manually or automatically on low canal level. The canal walls and floors are constructed of reinforced concrete. The canal shielding is designed to protect personnel in working areas adjacent to various parts of the canal. The working dose rate at the canal surface is normally less than 1.0 mR/hr and is normally less than 50 mR/hr during irradiated fuel transfers. The canal bottom is 5.5 to 7 ft of concrete and the expected dose rate below the floor is less than 1 mR/hr. The canal area is immediately north of the ATR Control Room.

The ATR Critical (ATRC) Facility is housed in a 50- × 66-ft bay in the west side of TRA-670, which is outside the ATR confinement volume. The walls of the ATRC Facility bay are made of pumice block, concrete, or aluminum. The north and east walls are pumice block, the south wall is either concrete (counting room wall) or pumice block, and the west wall is a ribbed aluminum insulation sandwich. The ATRC Facility bay floor is built of reinforced concrete and withstands a floor loading of 1,000 psf. The floor and east wall are two of the barriers of the ATR confinement volume. Figure D-3 shows the floor plan of the ATRC Facility.

The ATRC Facility is used to obtain accurate and timely data on nuclear characteristics of the ATR core such as rod worths and calibrations, excess reactivities, neutron flux distributions, gamma-heat generation rates, fuel-loading requirements, and the effects of inserting and removing experiments. ATR could accomplish most of these measurements. However, measurements using the ATRC Facility instead of ATR are easier and more accurate. Also, using ATR to perform the measurements reduces the duty cycle of the reactor.

TRA-670 is analyzed in this EHA as a segmented facility consisting of three areas: 1) ATR confinement area, 2) ATR canal area, and 3) ATRC Facility.

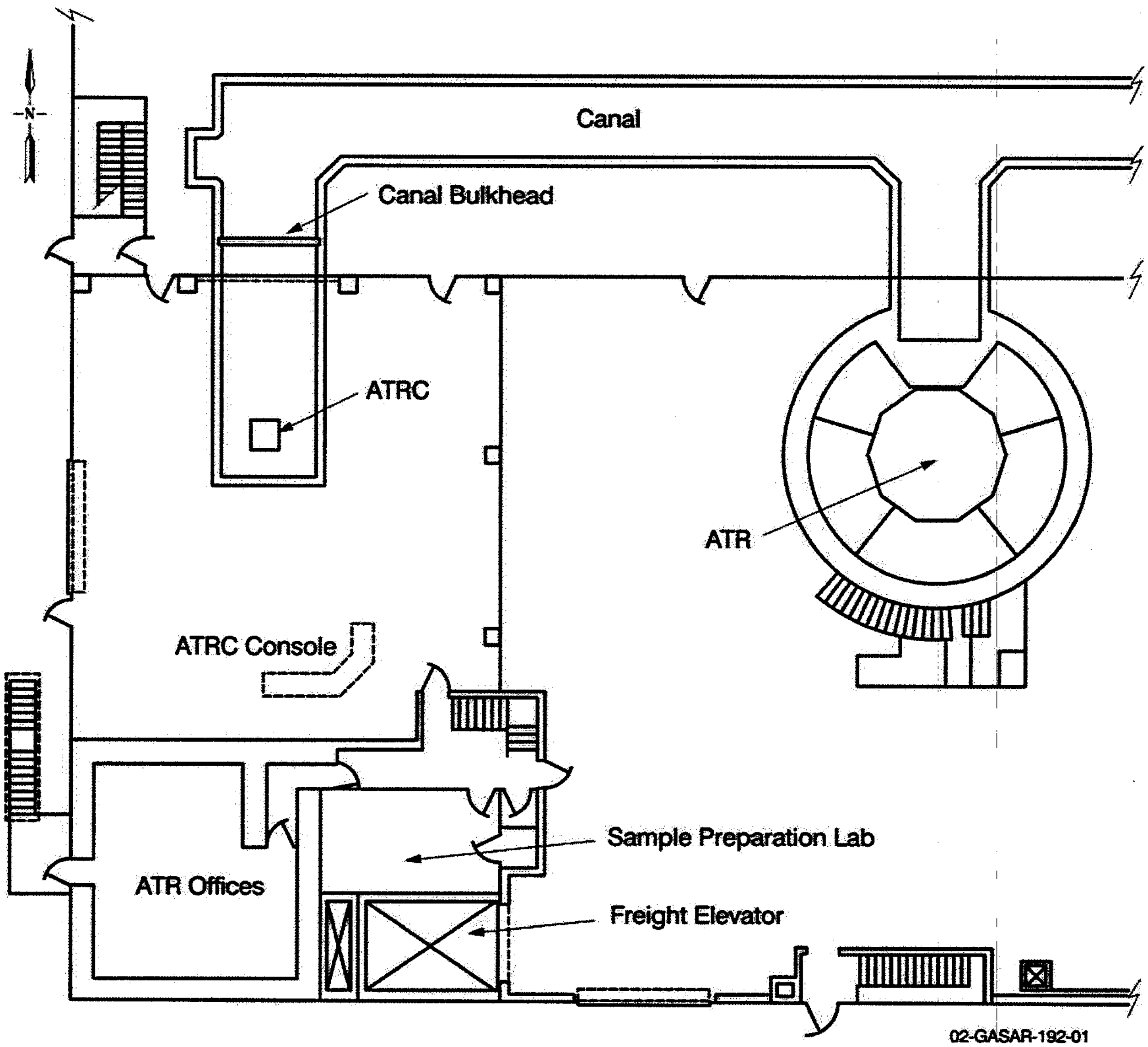


Figure D-3. Floor plan for Advanced Test Reactor Critical Facility.

D-1.3 Identification of Hazards

Table D-1 lists the radiological hazardous material that is stored, used, or produced in TRA-670 that is retained for further analyses based on the screening criteria presented in the main document.

Table D-1. Radiological hazardous material requiring further analyses.

Location	Material	Maximum Quantity (Ci)	Screening Threshold (Ci)	Notes
TRA-670 (ATR)	Fission Products, fissile material, and gamma radiation	Greater than Hazard Category (HC) 2	Greater than HC-3	ATR is categorized as a HC-1 facility. The ATRC Facility is categorized as a HC-2 facility.

D-1.4 Hazardous Material Characterization and Analyses

The hazardous material listed in Table D-1 is addressed below by location.

D-1.4.1 Advanced Test Reactor Confinement Area

D-1.4.1.1 Radiological Hazardous Material

D-1.4.1.1.1 Properties. The properties for the ATR confinement area radiological hazardous material are listed in Table D-2.

Table D-2. Radiological properties for the Advanced Test Reactor confinement area.

Physical form	Each ATR fuel element is made up of 19 curved plates covered by an aluminum alloy cladding. The fission products are contained within the fuel matrix in the form of gases, volatile, semi-volatile, or particulate materials. Electromagnetic gamma radiation from activated internal core components.
Particle size	Respirable.
Flammability	Nonflammable.
Reactivity	Not reactive.
Density	Not applicable (N/A).
Special firefighting concerns	None.
Health concerns	Potential acute and latent health effects (cancer or genetic effects) caused by inhalation or ingestion of the fission products.

D-1.4.1.1.2 Conditions of Storage and Use. The ATR core contains 40 fuel elements in a serpentine array. The fuel elements remain in the ATR core until the end of fuel life and then they are transferred to the ATR canal for storage during a cooldown period before being sent to an irradiated fuel storage facility.

The ATR core, after operating at 250 mW for 60 days, contains 3.01E+8 Ci of noble gases and volatile fission products (see SAR-153,[3] Table 20.2-1, Scram inventory). About 8.91E+7 Ci of those fission products are the primary contributors to dose. Table D-3 provides the fission product material-at-risk (MAR). Table D-4 provides a listing of the plant damage states for the severe accidents identified in SAR-153.

Table D-3. Material-at-risk for the Advanced Test Reactor confinement area.

Isotope	Quantity (Ci)
Kr-85m	2.56E+06
Kr-88	7.32E+06
Xe-133	1.36E+07
Xe-133m	4.11E+05
Cs-134	3.76E+04
Cs-137	4.85E+04
Te-127m	1.36E+04
Te-129m	1.52E+05
Te-132	7.90E+06
I-131	5.98E+06
I-132	9.06E+06
I-133	1.38E+07
I-134	1.54E+07
I-135	1.28E+07

Table D-4. Advanced Test Reactor plant damage states.

Plant Damage State	Description
1	Transients with failure of PCS recirculation, successful PCS depressurization, but failure of low-pressure coolant injection systems. Low-pressure boil-off (LPBO) occurs with eventual core uncovery and core-wide fuel damage.
2	Transients with failure of PCS recirculation and failure to depressurize the PCS. High-pressure boil-off (HPBO) occurs with eventual core uncovery and core-wide fuel damage.
3	Loss-of-coolant accidents (LOCAs) with failure of low-pressure coolant injection systems. Core uncovery and core-wide fuel damage occurs.
3M	Direct-damage large LOCA, resulting in initial fuel damage, even with successful scram and coolant injection.

Table D-4. (continued).

Plant Damage State	Description
4	Sequences involving failure to shut down ATR, termed anticipated transient without scram sequences. Potential fuel damage even with the core covered.
5	Very large reactivity insertion event potentially resulting in initial core-wide fuel damage and ATR vessel failure.
5M	Direct-damage large reactivity insertion events, resulting in initial fuel damage even with successful scram.
6	Interfacing system LOCAs with failure of low-pressure coolant injection systems.
7	Storage canal draining with failure to prevent complete fuel uncovery. The uncovery of hot or irradiated fuel stored in the canal results in fuel damage without confinement.

All of the ATR accident scenarios involve a release to the confinement area, and at that time the stack isolates and the release out of the confinement area is related to the diffusion out of the confinement area. For most accident scenarios, the confinement leakage rate 51% per day or a linear release duration of 48 hours. The large break LOCA has a confinement leakage rate of 99% per day or a release duration of 24 hours. The difference in confinement leakage rates is due to filling of the reactor basements with water during the large break LOCA event, which effectively reduces confinement volume. Where SAR-153 has specified a different release duration (i.e., 12 hours), the specified release duration has been used in that specific accident scenario.

Under normal operating conditions, the primary barrier is the fuel element cladding.

The secondary barrier is the PCS.

The following are some of the engineering controls and safety systems:[3]

- Radiation monitoring and seal system
- Emergency firewater injection system (EFIS)
- Vessel vent system
- Primary pump shutoff system
- Pressurizing and gland-seal pump shutoff system
- Primary relief or safety relief valves
- Vessel level alarm system
- LOCA PCP shutoff system
- Confinement system
- Habitability system
- Fission product removal and control system.

ATR is operated at 200 mW, which is lower than the maximum 250-mW authorized power level.

The following are some of the instruments or indicators used in the emergency action levels (EALs):

- Log count rate metering system
- Log-N period system
- Emergency firewater canal makeup system
- Primary coolant flow system.

D-1.4.1.1.3 Barrier and Failure Mode Analyses. The radiological barrier and failure mode analyses for the ATR confinement area presented below are summarized in Table D-5.

Table D-5. Radiological failure modes and barriers for the Advanced Test Reactor confinement area.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
ATRPDS-1	Radioactive fission products	Fuel cladding	Loss of primary coolant due to LPBO	Transients with failure of PCS recirculation, successful PCS depressurization, but failure of low-pressure coolant injection systems. Failure of pressurizing system with no low-pressure coolant injection due to system or operational failures. Decay heat slowly boils ATR vessel liquid. LPBO occurs with eventual core uncovery and core-wide fuel damage.	100% fuel melt (core uncovery in 18 hours and fuel melt at 19 hours)	ATR confinement slows release to 51% per day

Table D-5. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
ATRPDS-2	Radioactive fission products	Fuel cladding	Loss of primary coolant due to HPBO	Transients with failure of PCS recirculation and failure to depressurize the PCS. Failure of emergency PCPs and to depressurize PCS. Decay heat slowly boils ATR vessel liquid. HPBO occurs with eventual core uncovery and core-wide fuel damage.	100% fuel melt (core uncovery and fuel melt at 29 hours)	ATR confinement slows release to 51% per day
ATRPDS-3	Radioactive fission products	Fuel cladding	Loss of primary coolant due to small diameter break in inlet coolant piping outside radiographic boundary	Small break in inlet piping causes LOCA, and coupled with failure of low-pressure coolant injection system, core uncovery and core-wide fuel damage occurs.	100% fuel melt [core uncovery in 1.5 hours and release to (confinement) environment at 10 hours]	ATR confinement slows release to 51% per day

Table D-5. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
ATRPDS-3M	Radioactive fission products	Fuel cladding	Loss of primary coolant due to large break in coolant piping (outlet piping is more severe than inlet piping)	Direct damage large break LOCA, resulting in initial fuel damage, even with successful scram and coolant injection. Failure of PCS, resulting in core flow stagnation.	30% fuel melt containing 45% of core fission product inventory (fuel melt begins as early as 85 seconds)	ATR confinement slows release to 99% per day
ATRPDS-4	Radioactive fission products	Fuel cladding	Anticipated transient without scram (loss of ATR control, which may result in loss of coolant)	Sequences involving failure to shut down ATR, termed anticipated transient without scram sequences. Potential fuel damage even with the core covered. With pipe break, similar to Scenario ATRPDS-3 or -3M; without pipe break, similar to Scenario ATRPDS-1 or -2.	100% fuel melt (core uncovery in 18 hours and fuel melt at 19 hours)	ATR confinement slows release to 51% per day

Table D-5. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
ATRPDS-5	Radioactive fission products	Fuel cladding	Failure of multiple independent controls, causing rapid reactivity insertion, resulting in vessel movement and rupture of piping system leading to loss of coolant	A grossly perched fuel element drops into core after criticality is achieved, causing a very large and rapid reactivity insertion, resulting initial core-wide fuel damage and reactor vessel failure.	100% fuel melt [core uncovery in 1.5 hours and release to environment (confinement) at 10 hours]	ATR confinement slows release to 51% per day

Table D-5. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
ATRPDS-5M	Radioactive fission products	Fuel cladding	Fuel melt due to extremely rapid, positive ramp insertion of reactivity in ATR core from voiding in a high positive reactivity worth experiment flux trap	Sudden rupture of experiment loop pressure tube, causing sudden rupture of insulating gas envelope tube in core region of ATR core flux trap. This is a direct-damage large reactivity insertion event, resulting in initial fuel damage even with successful scram.	Immediate fuel melting, but release is limited (assumes 2% of Scenario ATRPDS-3)	PCS remains intact; ATR confinement slows gaseous release to 51% per day
ATRPDS-6	Radioactive fission products	Fuel cladding	Break in small piping outside radiographic boundary, causing loss of primary coolant, resulting in core uncovering	Failure of valve or piping in an interfacing system connected to PCS and failure of low-pressure injection system.	Some early fuel damage, but release is limited to noble gases; similar to Scenario ATRPDS-3	Liquid remains in interfacing system piping loop seals; ATR confinement slows gaseous release to 51% per day

Table D-5. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
ATREXPLR-1	Radioactive fission products	Fuel cladding	Loss of experiment loop coolant due to break in PCS piping	Failure of experimental loop cooling system.	100% (1-MW) experiment fuel melt	ATR confinement slows release to 51% per day
ATREXPLR-2	Radioactive fission products	Fuel cladding	Loss of experiment loop coolant due to break in PCS piping	Failure of experimental loop cooling system.	100% (200-kW) experiment fuel melt	ATR confinement slows release to 51% per day
ATRFCB-1	Radioactive fission products	Fuel cladding	Fuel element damage or fuel channel blockage due to failure of large structural elements in ATR vessel, above or in the core, with possible damage to PCS	Fuel channel blocked by debris, forgotten equipment, or structural failure.	100% of two fuel elements	ATR confinement slows release to 99% per day

Table D-5. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
ATRCaskDp-1	Radioactive fission products	Experimental fuel cladding	Fuel experiment damage due to cask drop onto open ATR head with fueled tests in ATR vessel	Crane failure or operator error during cask lift drops cask onto ATR vessel from above height limit established for routine lifts. Possible damage to emergency core cooling system leading to experimental fuel melt.	100% of fission products contained in 275-g U-235 fuel experiment	ATR confinement slows release to 99% per day
TRA670RR-1	Radioactive fission products	Fuel cladding	Fuel cladding imperfection	Normal ATR operations following refueling.	Normal gaseous fission product release to PCS greater than 20 μCi/mL	PCS water contains released radioactivity

D-1.4.1.1.4 Scenario Bases and Analytical Assumptions

1. **Scenario ATRPDS-1, Low-Pressure Boil-Off Due to Loss-of-Coolant Flow**

Detailed Scenario Description	Transients with failure of PCS recirculation, successful PCS depressurization, but failure of low-pressure coolant injection systems. LPBO occurs with eventual uncovering of the core and core-wide fuel damage.
Material-at-Risk	8.91E+07 Ci of fresh fission products (Table D-3).
Release Characteristics	100% fuel melt (core uncovery in 18 hours and fuel melt in 19 hours). ATR confinement slows release to 51% per day. The release is at ground level.
Airborne Release Fraction	1.00E+0 for noble gases, 1.79E−2 for iodine, 4.99E−3 for cesium, and 4.77E−3 for tellurium (SAR-153, Table 15.12-2, p. 15.12-22 of 15.12-34).
Respirable Fraction	1.0E+0.
Damage Ratio	1.0E+0.
Leak Path Factor	1.0E+0 (factored into SAR-153, Table 15.12-2).
Adjustment Factor	8.33E−2 [adjusts source term (ST) to represent the first 4 hours of a 48-hour release period]. The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)
Source Term	The ST shown in Table D-6 was developed according to the following equation: ST = (MAR × DR × LPF × ARF × RF) × AF. Where DR = damage ratio LPF = leak path factor ARF = airborne release fraction RF = respirable fraction AF = adjustment factor.
Modeling Software and Inputs	Radiological Safety Analysis Computer (RSAC) Program, Version 6.2.[4] Ground-level release.

Table D-6. Source term for scenario release designator ATRPDS-1.

Nuclide	MAR (Ci)	DR	LPF × RF	ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	8.33E−2	2.13E+5
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	8.33E−2	6.10E+5
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	8.33E−2	1.13E+6
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	8.33E−2	3.43E+4
Cs-134	3.76E+04	1.00E+0	1.00E+0	4.99E−3	8.33E−2	1.56E+1
Cs-137	4.85E+04	1.00E+0	1.00E+0	4.99E−3	8.33E−2	2.02E+1
Te-127m	1.36E+04	1.00E+0	1.00E+0	4.77E−3	8.33E−2	5.41E+0
Te-129m	1.52E+05	1.00E+0	1.00E+0	4.77E−3	8.33E−2	6.04E+1
Te-132	7.90E+06	1.00E+0	1.00E+0	4.77E−3	8.33E−2	3.14E+3
I-131	5.98E+06	1.00E+0	1.00E+0	1.79E−2	8.33E−2	8.92E+3
I-132	9.06E+06	1.00E+0	1.00E+0	1.79E−2	8.33E−2	1.35E+4
I-133	1.38E+07	1.00E+0	1.00E+0	1.79E−2	8.33E−2	2.06E+4
I-134	1.54E+07	1.00E+0	1.00E+0	1.79E−2	8.33E−2	2.30E+4
I-135	1.28E+07	1.00E+0	1.00E+0	1.79E−2	8.33E−2	1.91E+4

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E−2 = 1.0).

2. Scenario ATRPDS-2, High-Pressure Boil-Off Due to Loss-of-Coolant Flow

Detailed Scenario Description Transients with failure of PCS recirculation and failure to depressurize the PCS. HPBO occurs with eventual uncovering of the core and core-wide fuel damage.

Material-at-Risk 8.91E+07 Ci of fresh fission products (Table D-3).

Release Characteristics 100% fuel melt (core uncovery and fuel melt at 29 hours). ATR confinement slows release to 51% per day. The release is at ground level.

Airborne Release Fraction 1.00E+0 for noble gases, 3.89E−2 for iodine, 4.97E−2 for cesium, and 6.68E−2 for tellurium (SAR-153, Table 15.12-3, p. 15.12-22 of 15.12-34).

Respirable Fraction 1.0E+0.

Damage Ratio 1.0E+0.

Leak Path Factor 1.0E+0 (factored into SAR-153, Table 15.12-3).

Adjustment Factor 8.33E−2 (adjusts ST to represent the first 4 hours of a 48-hour release period). The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)

Source Term The ST shown in Table D-7 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-7. Source term for scenario release designator ATRPDS-2.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	8.33E−2	2.13E+5
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	8.33E−2	6.10E+5
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	8.33E−2	1.13E+6
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	8.33E−2	3.43E+4
Cs-134	3.76E+04	1.00E+0	1.00E+0	4.97E−2	8.33E−2	1.56E+2
Cs-137	4.85E+04	1.00E+0	1.00E+0	4.97E−2	8.33E−2	2.01E+2
Te-127m	1.36E+04	1.00E+0	1.00E+0	6.68E−2	8.33E−2	7.57E+1
Te-129m	1.52E+05	1.00E+0	1.00E+0	6.68E−2	8.33E−2	8.46E+2
Te-132	7.90E+06	1.00E+0	1.00E+0	6.68E−2	8.33E−2	4.40E+4
I-131	5.98E+06	1.00E+0	1.00E+0	3.89E−2	8.33E−2	1.94E+4
I-132	9.06E+06	1.00E+0	1.00E+0	3.89E−2	8.33E−2	2.94E+4
I-133	1.38E+07	1.00E+0	1.00E+0	3.89E−2	8.33E−2	4.47E+4
I-134	1.54E+07	1.00E+0	1.00E+0	3.89E−2	8.33E−2	4.99E+4
I-135	1.28E+07	1.00E+0	1.00E+0	3.89E−2	8.33E−2	4.15E+4

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E−2 = 1.0).

3. Scenario ATRPDS-3, Small Break Loss-of-Coolant Accident

Detailed Scenario Description LOCA with failure of low-pressure coolant injection systems. Uncovering of the core and core-wide fuel damage occur. Loss of primary coolant due to a small diameter break in the inlet coolant piping outside of the radiographic boundary.

Material-at-Risk 8.91E+07 Ci of fresh fission products (Table D-3).

Release Characteristics 100% fuel melt (core uncovery in 1.5 hours and release to environment at 10 hours). ATR confinement slows release to 51% per day. The release is at ground level.

Airborne Release Fraction 1.00E+0 for noble gases, 1.22E−1 for iodine, 2.74E−2 for cesium, and 6.97E−2 for tellurium (SAR-153, Table 15.12-4, p. 15.12-22 of 15.12-34).

Respirable Fraction 1.0E+0.

Damage Ratio 1.0E+0.

Leak Path Factor 1.0E+0 (factored into SAR-153, Table 15.12-4).

Adjustment Factor 8.33E−2 (adjusts ST to represent the first 4 hours of a 48-hour release period). The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)

Source Term The ST shown in Table D-8 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-8. Source term for scenario release designator ATRPDS-3.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	8.33E−2	2.13E+5
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	8.33E−2	6.10E+5
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	8.33E−2	1.13E+6
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	8.33E−2	3.43E+4
Cs-134	3.76E+04	1.00E+0	1.00E+0	2.74E−2	8.33E−2	8.59E+1
Cs-137	4.85E+04	1.00E+0	1.00E+0	2.74E−2	8.33E−2	1.11E+2
Te-127m	1.36E+04	1.00E+0	1.00E+0	6.97E−2	8.33E−2	7.90E+1

Table D-8. (continued).

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Te-129m	1.52E+05	1.00E+0	1.00E+0	6.97E−2	8.33E−2	8.83E+2
Te-132	7.90E+06	1.00E+0	1.00E+0	6.97E−2	8.33E−2	4.59E+4
I-131	5.98E+06	1.00E+0	1.00E+0	1.22E−1	8.33E−2	6.08E+4
I-132	9.06E+06	1.00E+0	1.00E+0	1.22E−1	8.33E−2	9.21E+4
I-133	1.38E+07	1.00E+0	1.00E+0	1.22E−1	8.33E−2	1.40E+5
I-134	1.54E+07	1.00E+0	1.00E+0	1.22E−1	8.33E−2	1.57E+5
I-135	1.28E+07	1.00E+0	1.00E+0	1.22E−1	8.33E−2	1.30E+5

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E−2 = 1.0).

4. Scenario ATRPDS-3M, Large Break Loss-of-Coolant Accident

Detailed Scenario Description — Direct-damage large LOCA, resulting in initial fuel damage, even with successful scram and coolant injection. Loss of primary coolant due to a large break in coolant piping (outlet piping is more severe than inlet piping).

Material-at-Risk — 8.91E+07 Ci of fresh fission products (Table D-3).

Release Characteristics — 30% fuel melt containing 45% of core fission product inventory (fuel melt begins as early as 85 seconds). ATR confinement slows release to 99% per day. The release is at ground level.

Airborne Release Fraction — 2.33E–1 for noble gases, 2.39E–2 for iodine, 3.92E–6 for cesium, and 1.30E–6 for tellurium (SAR-153, Table 15.12-5, p. 15.12-22 of 15.12-34).

Respirable Fraction — 1.0E+0.

Damage Ratio — 1.0E+0.

Leak Path Factor — 1.0E+0 (factored into SAR-153, Table 15.12-5).

Adjustment Factor — 1.67E−1 (adjusts ST to represent the first 4 hours of a 24-hour release period). The ST is based on the first 4 hours of a 24 hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately at this fractionation step. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 6 at 24 hours.)

Source Term The ST shown in Table D-9 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-9. Source term for scenario release designator ATRPDS-3M.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	2.33E−1	1.67E−1	9.96E+4
Kr-88	7.32E+06	1.00E+0	1.00E+0	2.33E−1	1.67E−1	2.85E+5
Xe-133	1.36E+07	1.00E+0	1.00E+0	2.33E−1	1.67E−1	5.29E+5
Xe-133m	4.11E+05	1.00E+0	1.00E+0	2.33E−1	1.67E−1	1.60E+4
Cs-134	3.76E+04	1.00E+0	1.00E+0	3.92E−6	1.67E−1	2.46E−2
Cs-137	4.85E+04	1.00E+0	1.00E+0	3.92E−6	1.67E−1	3.18E−2
Te-127m	1.36E+04	1.00E+0	1.00E+0	1.30E−6	1.67E−1	2.95E−3
Te-129m	1.52E+05	1.00E+0	1.00E+0	1.30E−6	1.67E−1	3.30E−2
Te-132	7.90E+06	1.00E+0	1.00E+0	1.30E−6	1.67E−1	1.72E+0
I-131	5.98E+06	1.00E+0	1.00E+0	2.39E−2	1.67E−1	2.39E+4
I-132	9.06E+06	1.00E+0	1.00E+0	2.39E−2	1.67E−1	3.62E+4
I-133	1.38E+07	1.00E+0	1.00E+0	2.39E−2	1.67E−1	5.51E+4
I-134	1.54E+07	1.00E+0	1.00E+0	2.39E−2	1.67E−1	6.15E+4
I-135	1.28E+07	1.00E+0	1.00E+0	2.39E−2	1.67E−1	5.11E+4

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are six 4-hour time blocks in the 24-hour release duration. At the end of 24 hours, the AF = 1.0 (6 × 1.67E−1 = 1.0).

5. Scenario ATRPDS-4, Anticipated Transient Without Scram

Detailed Scenario Description Sequences involving failure to shut down the ATR, termed anticipated transients without scram sequences. Potential fuel damage even with the core covered.

Material-at-Risk 8.91E+07 Ci of fresh fission products (Table D-3).

Release Characteristics 100% fuel melt (core uncovery in 18 hours and fuel melt in 19 hours). ATR confinement slows release to 51% per day. The release is at ground level.

Airborne Release Fraction 1.00E+0 for noble gases, 1.79E–2 for iodine, 4.99E–3 for cesium, and 4.77E–3 for tellurium (SAR-153, Table 15.12-2, p. 15.12-22 of 15.12-34).

Respirable Fraction 1.0E+0.

Damage Ratio 1.0E+0.

Leak Path Factor 1.0E+0 (factored into SAR-153, Table 15.12-2).

Adjustment Factor 8.33E–2 (adjusts ST to represent the first 4 hours of a 48-hour release period). The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)

Source Term The ST shown in Table D-10 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-10. Source term for scenario release designator ATRPDS-4.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	8.33E–2	2.13E+5
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	8.33E–2	6.10E+5
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	8.33E–2	1.13E+6
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	8.33E–2	3.43E+4
Cs-134	3.76E+04	1.00E+0	1.00E+0	4.99E–3	8.33E–2	1.56E+1
Cs-137	4.85E+04	1.00E+0	1.00E+0	4.99E–3	8.33E–2	2.02E+1
Te-127m	1.36E+04	1.00E+0	1.00E+0	4.77E–3	8.33E–2	5.41E+0
Te-129m	1.52E+05	1.00E+0	1.00E+0	4.77E–3	8.33E–2	6.04E+1
Te-132	7.90E+06	1.00E+0	1.00E+0	4.77E–3	8.33E–2	3.14E+3
I-131	5.98E+06	1.00E+0	1.00E+0	1.79E–2	8.33E–2	8.92E+3
I-132	9.06E+06	1.00E+0	1.00E+0	1.79E–2	8.33E–2	1.35E+4
I-133	1.38E+07	1.00E+0	1.00E+0	1.79E–2	8.33E–2	2.06E+4
I-134	1.54E+07	1.00E+0	1.00E+0	1.79E–2	8.33E–2	2.30E+4
I-135	1.28E+07	1.00E+0	1.00E+0	1.79E–2	8.33E–2	1.91E+4

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E–2 = 1.0).

6. Scenario ATRPDS-5, Very Large Reactivity Insertion Event

Detailed Scenario Description Very large reactivity insertion event potentially resulting in initial core-wide fuel damage and ATR vessel failure. A grossly perched fuel element drops into core after criticality is achieved, causing a very large and rapid reactivity insertion resulting in vessel movement and rupture of piping system leading to loss of coolant.

Material-at-Risk 8.91E+07 Ci of fresh fission products (Table D-3).

Release Characteristics 100% fuel melt (core uncovery in 1.5 hours and release to environment at 10 hours). ATR confinement slows release to 51% per day. The release is at ground level.

Airborne Release Fraction 1.00E+0 for noble gases, 1.22E−1 for iodine, 2.74E−2 for cesium, and 6.97E−2 for tellurium (SAR-153, Table 15.12-4, p. 15.12-22 of 15.12-34).

Respirable Fraction 1.0E+0.

Damage Ratio 1.0E+0.

Leak Path Factor 1.0E+0 (factored into SAR-153, Table 15.12-4).

Adjustment Factor 8.33E−2 (adjusts ST to represent the first 4 hours of a 48-hour release period). The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)

Source Term The ST shown in Table D-11 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-11. Source term for scenario release designator ATRPDS-5.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	8.33E−2	2.13E+5
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	8.33E−2	6.10E+5
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	8.33E−2	1.13E+6
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	8.33E−2	3.43E+4
Cs-134	3.76E+04	1.00E+0	1.00E+0	2.74E−2	8.33E−2	8.59E+1
Cs-137	4.85E+04	1.00E+0	1.00E+0	2.74E−2	8.33E−2	1.11E+2

Table D-11. (continued).

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Te-127m	1.36E+04	1.00E+0	1.00E+0	6.97E−2	8.33E−2	7.90E+1
Te-129m	1.52E+05	1.00E+0	1.00E+0	6.97E−2	8.33E−2	8.83E+2
Te-132	7.90E+06	1.00E+0	1.00E+0	6.97E−2	8.33E−2	4.59E+4
I-131	5.98E+06	1.00E+0	1.00E+0	1.22E−1	8.33E−2	6.08E+4
I-132	9.06E+06	1.00E+0	1.00E+0	1.22E−1	8.33E−2	9.21E+4
I-133	1.38E+07	1.00E+0	1.00E+0	1.22E−1	8.33E−2	1.40E+5
I-134	1.54E+07	1.00E+0	1.00E+0	1.22E−1	8.33E−2	1.57E+5
I-135	1.28E+07	1.00E+0	1.00E+0	1.22E−1	8.33E−2	1.30E+5

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E−2 = 1.0).

7. Scenario ATRPDS-5M, Direct-Damage Reactivity Insertion Event

Detailed Scenario Description Direct-damage large reactivity insertion event, resulting in initial fuel damage even with successful scram. Sudden rupture of experiment loop pressure tube, causing sudden rupture of insulating gas envelope tube in core region of ATR core flux trap causes fuel melt due to extremely rapid, positive ramp insertion of reactivity in ATR core from voiding in a high positive reactivity worth experiment flux trap.

Material-at-Risk 8.91E+07 Ci of fresh fission products (Table D-3).

Release Characteristics Immediate fuel melting, but release is limited (assumes 2% of Scenario ATRPDS-3). PCS remains intact. ATR confinement slows gaseous release to 51% per day. The release is at ground level.

Airborne Release Fraction 1.00E+0 for noble gases, 1.22E−1 for iodine, 2.74E−2 for cesium, and 6.97E−2 for tellurium (SAR-153, Table 15.12-4, p. 15.12-22 of 15.12-34).

Respirable Fraction 1.0E+0.

Damage Ratio 2.00E−2 (assumes 2% fuel cladding damage).

Leak Path Factor 1.0E+0 (factored into SAR-153, Table 15.12-4).

Adjustment Factor 8.33E−2 (adjusts ST to represent the first 4 hours of a 48-hour release period). The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)

Source Term The ST shown in Table D-12 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-12. Source term for scenario release designator ATRPDS-5M.

Nuclide	MAR (Ci)[a]	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	2.00E−2	1.00E+0	1.00E+0	8.33E−2	4.26E+3
Kr-88	7.32E+06	2.00E−2	1.00E+0	1.00E+0	8.33E−2	1.22E+4
Xe-133	1.36E+07	2.00E−2	1.00E+0	1.00E+0	8.33E−2	2.27E+4
Xe-133m	4.11E+05	2.00E−2	1.00E+0	1.00E+0	8.33E−2	6.85E+2
Cs-134	3.76E+04	2.00E−2	1.00E+0	2.74E−2	8.33E−2	1.72E+0
Cs-137	4.85E+04	2.00E−2	1.00E+0	2.74E−2	8.33E−2	2.21E+0
Te-127m	1.36E+04	2.00E−2	1.00E+0	6.97E−2	8.33E−2	1.58E+0
Te-129m	1.52E+05	2.00E−2	1.00E+0	6.97E−2	8.33E−2	1.77E+1
Te-132	7.90E+06	2.00E−2	1.00E+0	6.97E−2	8.33E−2	9.17E+2
I-131	5.98E+06	2.00E−2	1.00E+0	1.22E−1	8.33E−2	1.22E+3
I-132	9.06E+06	2.00E−2	1.00E+0	1.22E−1	8.33E−2	1.84E+3
I-133	1.38E+07	2.00E−2	1.00E+0	1.22E−1	8.33E−2	2.80E+3
I-134	1.54E+07	2.00E−2	1.00E+0	1.22E−1	8.33E−2	3.13E+3
I-135	1.28E+07	2.00E−2	1.00E+0	1.22E−1	8.33E−2	2.60E+3

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E−2 = 1.0). In this special case, Scenario ATRPDS-5M is assumed to be equivalent to 2% (0.02) of the Scenario ATRPDS-3 ST, therefore, the DR value is 2E−2.

8. Scenario ATRPDS-6, Interfacing System Loss-of-Coolant Accident

Detailed Scenario Description Interfacing system LOCA with failure of low-pressure coolant injection systems. Failure of valve or piping in an interfacing system connected to PCS and failure of low-pressure injection system results in core uncovery.

Material-at-Risk Noble gas fresh fission products (Table D-3).

Release Characteristics Some early fuel damage, but release is limited to noble gases; similar to Scenario ATRPDS-3. Liquid remains in interfacing system piping loop seals. The pipe break and release pathway are outside of the ATR confinement, so the release is assumed to occur over a 12-hour time period. The release is at ground level.

Airborne Release Fraction 1.0E+0 for noble gases (SAR-153, Table 15.12-4, p. 15.12-22 of 15.12–34; only noble gases are released in this scenario event).

Respirable Fraction 1.0E+0.

Damage Ratio 1.0E+0 (assumes fuel cladding damage).

Leak Path Factor 1.0E+0 (factored into SAR-153, Table 15.12-4).

Adjustment Factor 3.33E−1 (adjusts ST to represent the first 4 hours of a 12-hour release period). The ST is based on the first 4 hours of a 12-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 3 at 12 hours.)

Source Term The ST shown in Table D-13 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-13. Source term for scenario release designator ATRPDS-6.

Nuclide	MAR (Ci)[a]	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	3.33E−1	8.52E+5
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	3.33E−1	2.44E+6
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	3.33E−1	4.53E+6
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	3.33E−1	1.37E+5

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are three 4-hour time blocks in the 12-hour release duration. At the end of 12 hours, the AF = 1.0 (3 × 3.33E−1 = 1.0).

9. Scenario ATREXPLR-1, Experiment Loop Loss-of-Coolant Accident, 1-MW Experiment

Detailed Scenario Description Loss of experiment loop coolant due to break in piping.

Material-at-Risk Fraction of fresh fission products in Table D-3. [This scenario assumes that a 1-MW fueled experiment is involved. The MAR must be adjusted by multiplying by 4.0E–3 (1 MW/250 MW).]

Release Characteristics 100% (1-MW) experiment fuel melt. ATR confinement slows release to 51% per day. The release is at ground level.

Airborne Release Fraction 1.00E+0 for noble gases, 1.00E+0 for iodine, 1.00E+0 for cesium, and 1.00E+0 for tellurium.

Respirable Fraction 1.0E+0.

Damage Ratio 1.0E+0 (assumes fuel cladding damage).

Leak Path Factor 1.0E+0 (factored into SAR-153, Table 15.12-6).

Adjustment Factor 3.33E–4 (adjusts ST to represent the first 4 hours of a 48-hour release period and 1 MW/250 MW, 8.33E–2 × 4.00E–3 = 3.33E–4). The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)

Source Term The ST shown in Table D-14 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-14. Source term for scenario release designator ATREXPLR-1.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	3.33E–4	8.53E+2
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	3.33E–4	2.44E+3
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	3.33E–4	4.53E+3
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	3.33E–4	1.37E+2
Cs-134	3.76E+04	1.00E+0	1.00E+0	1.00E+0	3.33E–4	1.25E+1
Cs-137	4.85E+04	1.00E+0	1.00E+0	1.00E+0	3.33E–4	1.62E+1
Te-127m	1.36E+04	1.00E+0	1.00E+0	1.00E+0	3.33E–4	4.53E+0
Te-129m	1.52E+05	1.00E+0	1.00E+0	1.00E+0	3.33E–4	5.07E+1
Te-132	7.90E+06	1.00E+0	1.00E+0	1.00E+0	3.33E–4	2.63E+3

Table D-14. (continued).

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
I-131	5.98E+06	1.00E+0	1.00E+0	1.00E+0	3.33E–4	1.99E+3
I-132	9.06E+06	1.00E+0	1.00E+0	1.00E+0	3.33E–4	3.02E+3
I-133	1.38E+07	1.00E+0	1.00E+0	1.00E+0	3.33E–4	4.60E+3
I-134	1.54E+07	1.00E+0	1.00E+0	1.00E+0	3.33E–4	5.13E+3
I-135	1.28E+07	1.00E+0	1.00E+0	1.00E+0	3.33E–4	4.27E+3

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E–2 = 1.0). In this case, the ST must be further adjusted to reflect the 1-MW fueled experiment instead of the 250-MW reactor core inventory. This is accomplished by multiplying by 4.0E–3 (1 MW/250 MW), so the AF = 8.33E–2 × 4.0E–3 = 3.33E–4.

10. Scenario ATREXPLR-2, Experiment Loop Loss-of-Coolant Accident, 200-kW Experiment

Detailed Scenario Description Loss of experiment loop coolant due to break in piping.

Material-at-Risk Fraction of fresh fission products in Table D-3 [This scenario assumes that a 200-kW fueled experiment is involved. The MAR must be adjusted by multiplying by 8.0E–4 (0.2 MW/250 MW).]

Release Characteristics 100% (200-kW) experiment fuel melt. ATR confinement slows release to 51% per day. The release is at ground level.

Airborne Release Fraction 1.00E+0 for noble gases, 1.00E+0 for iodine, 1.00E+0 for cesium, and 1.00E+0 for tellurium.

Respirable Fraction 1.0E+0.

Damage Ratio 1.0E+0 (assumes fuel cladding damage).

Leak Path Factor 1.0E+0.

Adjustment Factor 6.67E–5 (adjusts ST to represent the first 4 hours of a 48-hour release period and 0.2 MW/250 MW, 8.33E–2 × 8.00E–4 = 6.67E–5). The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)

Source Term The ST shown in Table D-15 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-15. Source term for scenario release designator ATREXPLR-2.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	6.67E–5	1.71E+2
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	6.67E–5	4.88E+2
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	6.67E–5	9.07E+2
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	6.67E–5	2.74E+1
Cs-134	3.76E+04	1.00E+0	1.00E+0	1.00E+0	6.67E–5	2.51E+0
Cs-137	4.85E+04	1.00E+0	1.00E+0	1.00E+0	6.67E–5	3.23E+0
Te-127m	1.36E+04	1.00E+0	1.00E+0	1.00E+0	6.67E–5	9.07E–1
Te-129m	1.52E+05	1.00E+0	1.00E+0	1.00E+0	6.67E–5	1.01E+1
Te-132	7.90E+06	1.00E+0	1.00E+0	1.00E+0	6.67E–5	5.27E+2
I-131	5.98E+06	1.00E+0	1.00E+0	1.00E+0	6.67E–5	3.99E+2
I-132	9.06E+06	1.00E+0	1.00E+0	1.00E+0	6.67E–5	6.04E+2
I-133	1.38E+07	1.00E+0	1.00E+0	1.00E+0	6.67E–5	9.20E+2
I-134	1.54E+07	1.00E+0	1.00E+0	1.00E+0	6.67E–5	1.03E+3
I-135	1.28E+07	1.00E+0	1.00E+0	1.00E+0	6.67E–5	8.53E+2

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E–2 = 1.0). In this case, the ST must be further adjusted to reflect the 200-kW fueled experiment instead of the 250-MW reactor core inventory. This is accomplished by multiplying by 8.0E–4 (0.2 MW/250 MW), so the AF = 8.33E–2 × 8.0E–4 = 6.67E–5.

11. Scenario ATRFCB-1, Fuel Channel Blockage Event

Detailed Scenario Description — Fuel element damage or fuel channel blockage due to failure of large structural elements in ATR vessel, above or in the core, with possible damage to PCS. Fuel channel blocked by debris, forgotten equipment, or structural failure.

Material-at-Risk — This event is assumed to have an ST that is equivalent to 5% of Scenario ATRPDS-3M.

Release Characteristics — 100% of two fuel elements. ATR confinement slows release to 99% per day. The release is at ground level.

Airborne Release Fraction — 2.33E–1 for noble gases, 2.39E–2 for iodine, 3.92E–6 for cesium, and 1.30E–6 for tellurium (SAR-153, Table 15.12-5, p. 15.12-22 of 15.12-34).

Respirable Fraction — 1.0E+0.

Damage Ratio 5.00E–2 [assumes 5% (2/40 = 0.05) fuel cladding damage].

Leak Path Factor 1.0E+0 (factored into SAR-153, Table 15.12-5).

Adjustment Factor 1.67E–1 (adjusts ST to represent the first 4 hours of a 24-hour release period). The ST is based on the first 4 hours of a 24 hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately at this fractionation step. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 6 at 24 hours.)

Source Term The ST shown in Table D-16 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-16. Source term for scenario release designator ATRFCB-1.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	5.00E–2	1.00E+0	2.33E–1	1.67E–1	4.98E+3
Kr-88	7.32E+06	5.00E–2	1.00E+0	2.33E–1	1.67E–1	1.42E+4
Xe-133	1.36E+07	5.00E–2	1.00E+0	2.33E–1	1.67E–1	2.65E+4
Xe-133m	4.11E+05	5.00E–2	1.00E+0	2.33E–1	1.67E–1	8.00E+2
Cs-134	3.76E+04	5.00E–2	1.00E+0	3.92E–6	1.67E–1	1.23E–3
Cs-137	4.85E+04	5.00E–2	1.00E+0	3.92E–6	1.67E–1	1.59E–3
Te-127m	1.36E+04	5.00E–2	1.00E+0	1.30E–6	1.67E–1	1.48E–4
Te-129m	1.52E+05	5.00E–2	1.00E+0	1.30E–6	1.67E–1	1.65E–3
Te-132	7.90E+06	5.00E–2	1.00E+0	1.30E–6	1.67E–1	8.58E–2
I-131	5.98E+06	5.00E–2	1.00E+0	2.39E–2	1.67E–1	1.19E+3
I-132	9.06E+06	5.00E–2	1.00E+0	2.39E–2	1.67E–1	1.81E+3
I-133	1.38E+07	5.00E–2	1.00E+0	2.39E–2	1.67E–1	2.75E+3
I-134	1.54E+07	5.00E–2	1.00E+0	2.39E–2	1.67E–1	3.07E+3
I-135	1.28E+07	5.00E–2	1.00E+0	2.39E–2	1.67E–1	2.55E+3

[a] For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are six 4-hour time blocks in the 24-hour release duration. At the end of 24 hours, the AF = 1.0 (6 × 1.67E–1 = 1.0). In this special case, Scenario ATRFCB-1 is assumed to be equivalent to 5% (0.05) of the Scenario ATRPDS-3M ST, therefore, the DR value is 5E–2.

12. Scenario ATRCskDp-1, Advanced Test Reactor Cask Drop Onto Open Advanced Test Reactor Head

Detailed Scenario Description — Crane failure or operator error during cask lift drops cask onto ATR vessel from above height limit established for routine lifts; possible damage to emergency core cooling system leading to experimental fuel melt.

Material-at-Risk — This event is assumed to have an ST that is equivalent to 0.63% of Scenario ATRPDS-3M.

Release Characteristics — 100% of fission products contained in 275-g U-235 fuel experiment. ATR confinement slows release to 99% per day. The release is at ground level.

Airborne Release Fraction — 2.33E−1 for noble gases, 2.39E−2 for iodine, 3.92E−6 for cesium, and 1.30E−6 for tellurium (SAR-153, Table 15.12-5, p. 15.12-22 of 15.12-34).

Respirable Fraction — 1.0E+0.

Damage Ratio — 6.30E−3 [assumes 0.63% fuel cladding damage (i.e., adjusts the ST to 275 g U-235, 275 g/43,400 g U-235 in core = 0.00634 or 6.3E−3)].

Leak Path Factor — 1.0E+0 (factored into SAR-153, Table 15.12-5).

Adjustment Factor — 1.67E−1 (adjusts ST to represent the first 4 hours of a 24-hour release period). The ST is based on the first 4 hours of a 24 hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately at this fractionation step. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 6 at 24 hours.)

Source Term — The ST shown in Table D-17 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs — RSAC, Version 6.2. Ground-level release.

Table D-17. Source term for scenario release designator ATRCaskDp-1.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	6.30E−3	1.00E+0	2.33E−1	1.67E−1	6.26E+2
Kr-88	7.32E+06	6.30E−3	1.00E+0	2.33E−1	1.67E−1	1.79E+3
Xe-133	1.36E+07	6.30E−3	1.00E+0	2.33E−1	1.67E−1	3.33E+3
Xe-133m	4.11E+05	6.30E−3	1.00E+0	2.33E−1	1.67E−1	1.01E+2
Cs-134	3.76E+04	6.30E−3	1.00E+0	3.92E−6	1.67E−1	1.55E−4
Cs-137	4.85E+04	6.30E−3	1.00E+0	3.92E−6	1.67E−1	2.00E−4
Te-127m	1.36E+04	6.30E−3	1.00E+0	1.30E−6	1.67E−1	1.86E−5

Table D-17. (continued).

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Te-129m	1.52E+05	6.30E–3	1.00E+0	1.30E–6	1.67E–1	2.07E–4
Te-132	7.90E+06	6.30E–3	1.00E+0	1.30E–6	1.67E–1	1.08E–2
I-131	5.98E+06	6.30E–3	1.00E+0	2.39E–2	1.67E–1	1.50E+2
I-132	9.06E+06	6.30E–3	1.00E+0	2.39E–2	1.67E–1	2.27E+2
I-133	1.38E+07	6.30E–3	1.00E+0	2.39E–2	1.67E–1	3.46E+2
I-134	1.54E+07	6.30E–3	1.00E+0	2.39E–2	1.67E–1	3.86E+2
I-135	1.28E+07	6.30E–3	1.00E+0	2.39E–2	1.67E–1	3.21E+2

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are six 4-hour time blocks in the 24-hour release duration. At the end of 24 hours, the AF = 1.0 (6 × 1.67E–1 = 1.0). In this special case, Scenario ATRCaskDp-1 is assumed to be equivalent to 0.63% (0.0063) of the Scenario ATRPDS-3M ST (adjusts the ST to 275 g U-235, 275 g/43,400 g U-235 in core = 0.00634 or 6.3E–3), therefore, the DR value is 6.3E–3.

13. Scenario TRA670RR-1, Primary Coolant Concentration Greater Than 20µCi/mL

Detailed Scenario Description

Normal ATR operations following refueling. Cladding imperfections in new or experimental fuel cause a release of fission products to the ATR primary coolant.

Material-at-Risk

This event is assumed to have a ST that is equivalent to 20 µCi/mL in the PCS, which is equivalent to 5.60E+3 Ci (73,970 gal × 3,785 mL/gal × 20 µCi/mL = 5.60E+9 µCi or 5.60E+3 Ci).

Release Characteristics

Normal gaseous fission product release to PCS greater than 20 µCi/mL. PCS water contains released radioactivity. The release from the ATR primary coolant is to the ATR confinement during coolant cleanup activities within the ATR. The release is assumed to be at ground level.

Airborne Release Fraction

1.00E+0 for noble gases, 2.70E–1 for iodine, 1.00E–4 for cesium, and 1.00E–4 for tellurium (DOE-HDBK-3010-94,[5] Section 4.3.1.3.1, p 4-49, for noble gases and iodine, and Section 3.2.3.1, p 3-33, for cesium and tellurium).

Respirable Fraction

1.0E+0.

Damage Ratio

5.05E–6 (20 µCi/mL inventory/total inventory, 5.60E+3 Ci/1.11E+9 Ci = 5.05E–6). See SAR-153, Table 20.2-1, for total inventory at scram.

Leak Path Factor

5.50E–1 (reduction due to plate-out).

Adjustment Factor 8.33E−2 (adjusts ST to represent the first 4 hours of a 48-hour release period). The ST is based on the first 4 hours of a 48-hour release duration, therefore, release durations longer than 4 hours must be multiplied appropriately. (For example, with a 6-hour release duration, the multiplier would be 1.5, at 8 hours the multiplier would be 2, and so on up to a multiplier of 12 at 48 hours.)

Source Term The ST shown in Table D-18 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-18. Source term for scenario release designator TRA670RR-1.

Nuclide	MAR (Ci)	DR	LPF	RF × ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	5.05E−6	5.50E−1	1.00E+0	8.33E−2	5.94E−1
Kr-88	7.32E+06	5.05E−6	5.5E−1	1.00E+0	8.33E−2	1.69E+0
Xe-133	1.36E+07	5.05E−6	5.5E−1	1.00E+0	8.33E−2	3.15E+0
Xe-133m	4.11E+05	5.05E−6	5.5E−1	1.00E+0	8.33E−2	9.52E−2
Cs-134	3.76E+04	5.05E−6	5.5E−1	1.00E−4	8.33E−2	8.69E−7
Cs-137	4.85E+04	5.05E−6	5.5E−1	1.00E−4	8.33E−2	1.12E−6
Te-127m	1.36E+04	5.05E−6	5.5E−1	1.00E−4	8.33E−2	3.15E−7
Te-129m	1.52E+05	5.05E−6	5.5E−1	1.00E−4	8.33E−2	3.52E−6
Te-132	7.90E+06	5.05E−6	5.5E−1	1.00E−4	8.33E−2	1.83E−4
I-131	5.98E+06	5.05E−6	5.5E−1	2.70E−1	8.33E−2	3.73E−1
I-132	9.06E+06	5.05E−6	5.5E−1	2.70E−1	8.33E−2	5.67E−1
I-133	1.38E+07	5.05E−6	5.5E−1	2.70E−1	8.33E−2	8.64E−1
I-134	1.54E+07	5.05E−6	5.5E−1	2.70E−1	8.33E−2	9.63E−1
I-135	1.28E+07	5.05E−6	5.5E−1	2.70E−1	8.33E−2	7.98E−1

a. For each 4-hour increment in time, the ST must be multiplied by the number of 4-hour blocks of time involved to determine the amount of radioactive material released. For example, there are 12 4-hour time blocks in the 48-hour release duration. At the end of 48 hours, the AF = 1.0 (12 × 8.33E−2 = 1.0).

D-1.4.2 Advanced Test Reactor Canal Area

D-1.4.2.1 Radiological Hazardous Material

D-1.4.2.1.1 Properties. The properties for the ATR canal area radiological hazardous material are listed in Table D-19.

Table D-19. Radiological properties for the Advanced Test Reactor canal area.

Physical form	Each ATR fuel element is made up of 19 curved plates covered by an aluminum alloy cladding. The fission products are contained within the fuel matrix in the form of gases, volatile, semi-volatile, or particulate materials. Electromagnetic gamma radiation from activated internal core components.
Particle size	Respirable.
Flammability	Nonflammable.
Reactivity	Not reactive.
Density	N/A.
Special firefighting concerns	None.
Health concerns	Potential acute and latent health effects (cancer or genetic effects) caused by inhalation or ingestion of the fission products.

D-1.4.2.1.2 Conditions of Storage and Use. The ATR canal is used for storing irradiated ATR spent nuclear fuel during a cooldown period before being sent to an irradiated fuel storage facility. The canal is also used to store activated internal components removed from the ATR during reactor outages to refurbish and refuel the reactor.

The ATR core, after operating at 250 mW for 60 days, contains 3.01E+8 Ci of noble gases and volatile fission products. About 8.91E+7 Ci of those fission products are the primary contributors to dose. Multiple core quantities of spent nuclear fuel may be in storage at any given time. Table D-20 provides the fission product MAR.

Table D-20. Material-at-risk for the Advanced Test Reactor canal area.

Isotope	Quantity (Ci)
Kr-85m	2.56E+06
Kr-88	7.32E+06
Xe-133	1.36E+07
Xe-133m	4.11E+05
Cs-134	3.76E+04
Cs-137	4.85E+04
Te-127m	1.36E+04
Te-129m	1.52E+05
Te-132	7.90E+06

Table D-20. (continued).

Isotope	Quantity (Ci)
I-131	5.98E+06
I-132	9.06E+06
I-133	1.38E+07
I-134	1.54E+07
I-135	1.28E+07

Under normal operating conditions, the primary barrier is the fuel element cladding.

The secondary barrier is the canal water.

The emergency fire water supply system for canal make-up water is one of the engineering controls and safety systems.[3]

ATR is operated at 200 mW, which is lower than the maximum 250-mW authorized power level.

The following are some of the instruments or indicators used in the EALs:

- Canal water level alarm system
- Continuous air monitors (CAMs) and remote area monitors (RAMs).

D-1.4.2.1.3 Barrier and Failure Mode Analyses. For the purposes of barrier and failure mode analyses, the accident scenarios are related to the airborne release of radiological material (fission products). The accident scenario types are based on information obtained from SAR-153, Chapter 15. The results of the barrier and failure mode analyses are detailed in the following subsection; Table D-21 summarizes the results. SAR-153 served as the principal references in developing the scenarios.

Table D-21. Radiological failure modes and barriers for the Advanced Test Reactor canal area.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
ATRPDS-7	Radioactive fission products	Fuel cladding	Loss of coolant water due to breach of canal wall or floor from dropped cask, falling crane, or seismic event	Dropped cask, falling overhead crane, or seismic event	100% fuel melt	Essentially none, as canal is outside ATR confinement
ATRDLEXP-1	Direct gamma radiation	Experiment transfer shield	Uncoupling of experiment from transfer shield assembly	Failure of coupling mechanism or mishandling	100% gamma radiation	Essentially none
ATRCIC-1	Direct gamma radiation	Transfer shield	Uncoupling of core internal part or equipment from the transfer shield assembly	Failure of coupling mechanism or mishandling	100% gamma radiation	Essentially none

D-1.4.2.1.4 Scenario Bases and Analytical Assumptions

1. Scenario ATRPDS-7, Advanced Test Reactor Canal Drain

Detailed Scenario Description — Loss of coolant water due to breach of canal wall or floor due to dropped cask, dropped resin block, falling crane, or seismic event.

Material-at-Risk — See Table D-20.

Release Characteristics — ATR is assumed to have operated at 250 mW for 60 days, which provides the maximum fission product inventory within the 40 fuel elements that have been removed from the ATR core and placed in the canal for a cooldown period. Irradiated experiments and ATR core internal components contain activation products, which create very high gamma radiation exposure rates (i.e., 1.0E+6 R/hr on contact). The release is assumed to be at ground level and takes place over a 2.5-hour time period.

Airborne Release Fraction — 1.00E+0 for noble gases, 1.00E+0 for iodine, 1.00E+0 for cesium, and 1.00E+0 for tellurium (SAR-153, Table 15.12-6, p. 15.12-23 of 15.12-34).

Respirable Fraction — 1.0E+0.

Damage Ratio — 1.0E+0.

Leak Path Factor — 1.0E+0 (factored into SAR-153, Table 15.12-6).

Adjustment Factor — N/A (release duration is less than 4 hours).

Source Term — The ST shown in Table D-22 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs — RSAC, Version 6.2. Ground-level release.

Table D-22. Source term for scenario release designator ATRPDS-7.

Nuclide	MAR (Ci)	DR	LPF × RF	ARF	AF[a]	ST (Ci)
Kr-85m	2.56E+06	1.00E+0	1.00E+0	1.00E+0	1.00E+0	2.56E+06
Kr-88	7.32E+06	1.00E+0	1.00E+0	1.00E+0	1.00E+0	7.32E+06
Xe-133	1.36E+07	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.36E+07
Xe-133m	4.11E+05	1.00E+0	1.00E+0	1.00E+0	1.00E+0	4.11E+05
Cs-134	3.76E+04	1.00E+0	1.00E+0	1.00E+0	1.00E+0	3.76E+04
Cs-137	4.85E+04	1.00E+0	1.00E+0	1.00E+0	1.00E+0	4.85E+04
Te-127m	1.36E+04	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.36E+04
Te-129m	1.52E+05	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.52E+05

Table D-22. (continued).

Nuclide	MAR (Ci)	DR	LPF × RF	ARF	AF[a]	ST (Ci)
Te-132	7.90E+06	1.00E+0	1.00E+0	1.00E+0	1.00E+0	7.90E+06
I-131	5.98E+06	1.00E+0	1.00E+0	1.00E+0	1.00E+0	5.98E+06
I-132	9.06E+06	1.00E+0	1.00E+0	1.00E+0	1.00E+0	9.06E+06
I-133	1.38E+07	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.38E+07
I-134	1.54E+07	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.54E+07
I-135	1.28E+07	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.28E+07

a. No adjustment, as the release is assumed to occur over a 2.5-hour time period.

2. Scenario ATRDLEXP-1 and ATRCIC-1, High Gamma Radiation Exposure Rates

Detailed Scenario Description Operational error results in dropped loop experiment or core internal change-out irradiated materials outside of shielded container.

Material-at-Risk Electromagnetic gamma radiation.

Release Characteristics Irradiated experiments and ATR core internal components contain activation products, which create very high gamma radiation exposure rates (i.e., 1.0E+6 R/hr on contact).

Airborne Release Fraction N/A.

Respirable Fraction N/A.

Damage Ratio N/A.

Leak Path Factor N/A.

Adjustment Factor N/A.

Source Term The exposure rate ST is shown in Table D-23.

Modeling Software and Inputs Inverse square law used to calculate exposure rates at a distance.

Table D-23. Advanced Test Reactor dropped loop experiment exposure rates by distance (Scenario ATRDLEXP-1) and Advanced Test Reactor dropped core internal change-out irradiated parts or equipment (Scenario ATRCIC-1).

Distance (m)	Exposure Rate (R/hr)
30	516
68[a]	100
100	46.5
124[b]	30.2
485[c]	1.97
815[d]	7.00E−1
2,865[e]	5.66E−02

a. Threshold for early lethality equivalent.
b. Closest ATR facility boundary.
c. ATR staging area in parking lot.
d. PA distance.
e. Idaho Nuclear Technology and Engineering Center (INTEC).

D-1.4.3 Advanced Test Reactor Critical Facility

D-1.4.3.1 Radiological Hazardous Material

D-1.4.3.1.1 Properties. The properties for the ATRC Facility radiological hazardous material are listed in Table D-24.

Table D-24. Radiological properties for the Advanced Test Reactor Critical Facility.

Physical form	Each ATR fuel element is made up of 19 curved plates covered by an aluminum alloy cladding. The fission products are contained within the fuel matrix in the form of gases, volatile, semi-volatile, or particulate materials. Electromagnetic gamma radiation from activated internal core components.
Particle size	Respirable.
Flammability	Nonflammable.
Reactivity	Not reactive.
Density	N/A.
Special firefighting concerns	None.
Health concerns	Potential acute and latent health effects (cancer or genetic effects) caused by inhalation or ingestion of the fission products.

D-1.4.3.1.2 Conditions of Storage and Use. The ATRC Facility is located in the ATR canal. The ATRC Facility contains 40 low-power irradiated fuel elements. The ATR canal water provides cooling for the ATRC Facility. Non-fuel fissile material is stored in locked cabinets located on the ATRC Facility main floor. Those cabinets are considered a mass criticality storage (MCA) area. Table D-25 provides a listing of the fissile materials and their location.

Table D-25. Advanced Test Reactor Critical Facility bay nominal fissile material inventory.[6]

Fuel Category	Area/Room	Rack, Cabinet, or Core	Fissile Mass (U-235 or Equivalent) (g)
40 fuel elements – U-235	TRA-670/ATRC Facility bay	Canal storage grid or core	1,075 ± 10 g each
Fuel plate bundle	TRA-670/ATRC Facility bay	ATRC Facility storage cabinet or core when assembled	Less than or equal to 1,075 ± 10 g each
Odd lot	TRA-670/ATRC Facility bay	ATRC Facility storage cabinet	Less than or equal to 700 g each
Test train	TRA-670/ATRC Facility bay	ATRC Facility storage cabinet or core	Less than or equal to 365 g each

The primary source of radioactive inventory is the fuel material and fission products within the ATRC Facility fuel element cladding, which is assumed equivalent to reactor operation at 5 kW for 7 days, followed by zero power for 23.4 days, in a cycle that is repeated for 60 years. This power history provides a bounding inventory for all foreseeable operations. The inventory of gaseous and volatile radionuclides is 6.28E+3 Ci and 1.77E+3 Ci contribute the majority of dose. Table D-26 provides the radiological MAR.

Table D-26. Material-at-risk for the Advanced Test Reactor Critical Facility [Engineering Design File (EDF) TRA-ATR-1729[7]].

Isotope	Quantity (Ci)
Kr-85m	5.35E+01
Kr-88	1.11E+02
Xe-133	3.10E+02
Xe-133m	9.41E+00
Cs-134	8.12E−01
Cs-137	1.06E+00
Te-127m	2.68E−01
Te-129m	2.72E+00
Te-132	1.77E+02
I-131	1.16E+02
I-132	1.76E+02
I-133	2.68E+02
I-134	2.98E+02
I-135	2.50E+02

Under normal operating conditions, the primary barrier is the fuel element cladding.

The secondary barrier is the ATR canal water.

The emergency fire water supply system for canal make-up water is one of the engineering controls and safety systems.[6]

The MCA is limited to a maximum of 650 g of fissile material.

The following are some of the instruments or indicators used in the EALs:

- Canal water level alarm system
- CAMs and RAMs
- Log count rate alarm.

D-1.4.3.1.3 Barrier and Failure Mode Analyses. For the purposes of barrier and failure mode analyses, the accident scenarios are related to the airborne release of radiological material (fission products). The accident scenario types are based on information obtained from SAR-192,[6] Chapters 8 and 10. The results of the barrier and failure mode analyses are detailed in the following subsection; Table D-27 summarizes the results. SAR-192 served as the principal references in developing the scenarios.

D-1.4.3.1.4 Scenario Bases and Analytical Assumptions

1. **Scenario ATRCCD-1, Advanced Test Reactor Critical Facility Fuel Handling Accident**

Detailed Scenario Description	Control bridge drops onto the ATRC Facility reactor core causing mechanical damage to fuel cladding.
Material-at-Risk	See Table D-26.
Release Characteristics	100% fuel cladding damage occurs releasing gaseous and volatile radionuclides to a depth of one-grain thickness. The release is at ground level and it is assumed to last for 4 hours.
Airborne Release Fraction	2.36E−2 for noble gases, 2.36E−2 for iodine, 2.36E−2 for cesium, and 2.36E−2 for tellurium (based on cladding damage of one-grain thickness[8]).
Respirable Fraction	1.0E+0 (assumption based on information in SAR-192).
Damage Ratio	1.0E+0 (assumes fuel cladding damage).
Leak Path Factor	1.0E+0 (release is direct to the ATRC Facility main floor, which is outside of the confinement).
Adjustment Factor	1.0 (adjusts ST to represent the first 4 hours of a 4-hour release period).
Source Term	The ST shown in Table D-28 was developed according to the following equation: $ST = (MAR \times DR \times LPF \times ARF \times RF) \times AF$.
Modeling Software and Inputs	RSAC, Version 6.2. Ground-level release.

Table D-27. Radiological failure modes and barriers for the Advanced Test Reactor Critical Facility.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release from Primary Barrier	Other Barriers and Their Effects
ATRCCD-1	Radioactive fission products	Fuel cladding	Mechanical damage to fuel cladding	Fuel handling accident, control bridge drops onto ATRC Facility reactor core	Release fraction of 2.36E−2 derived from a one-grain thickness fuel damage depth and 100% cladding damage	Essentially none, as ATRC Facility is outside ATR confinement
ATRCFM-1	Radioactive fission products	Fuel cladding	Unknown, requires multiple system failures at maximum operating power leading to fuel melt and fuel cladding breach	A major reactivity control system failure, reactor shutdown system failure, and safety rod drive system failure to insert safety rods	100% fuel melt	Essentially none, as ATRC Facility is outside ATR confinement
ATRCCRIT	Prompt neutron pulse and radioactive fission products	Physical and administrative controls on fuel storage and fuel cladding	Fuel mishandling event, which positions seven or more fuel element in a critical configuration at bottom of ATRC Facility canal	Mechanical system failure while moving fuel elements or failure to implement technical specification requirements causes criticality to occur	1E+18 fission products and 20-minute halogen in–growth	Essentially none, as ATRC Facility is outside ATR confinement

Table D-28. Source term for scenario release designator ATRCCD-1.

Nuclide	MAR (Ci)	DR	LPF × RF	ARF	AF[a]	ST (Ci)
Kr-85m	5.35E+01	1.00E+0	1.00E+0	2.36E−2	1.00E+0	1.26E+0
Kr-88	1.11E+02	1.00E+0	1.00E+0	2.36E−2	1.00E+0	2.62E+0
Xe-133	3.10E+02	1.00E+0	1.00E+0	2.36E−2	1.00E+0	7.32E+0
Xe-133m	9.41E+00	1.00E+0	1.00E+0	2.36E−2	1.00E+0	2.22E−1
Cs-134	8.12E−01	1.00E+0	1.00E+0	2.36E−2	1.00E+0	1.92E−2
Cs-137	1.06E+00	1.00E+0	1.00E+0	2.36E−2	1.00E+0	2.50E−2
Te-127m	2.68E−01	1.00E+0	1.00E+0	2.36E−2	1.00E+0	6.32E−3
Te-129m	2.72E+00	1.00E+0	1.00E+0	2.36E−2	1.00E+0	6.42E−2
Te-132	1.77E+02	1.00E+0	1.00E+0	2.36E−2	1.00E+0	4.18E+0
I-131	1.16E+02	1.00E+0	1.00E+0	2.36E−2	1.00E+0	2.74E+0
I-132	1.76E+02	1.00E+0	1.00E+0	2.36E−2	1.00E+0	4.15E+0
I-133	2.68E+02	1.00E+0	1.00E+0	2.36E−2	1.00E+0	6.32E+0
I-134	2.98E+02	1.00E+0	1.00E+0	2.36E−2	1.00E+0	7.03E+0
I-135	2.50E+02	1.00E+0	1.00E+0	2.36E−2	1.00E+0	5.90E+0

a. No adjustment, as the release is assumed to occur over a 4-hour time period.

2. Scenario ATRCFM-1, Advanced Test Reactor Critical Facility Severe Reactor Accident

Detailed Scenario Description — Multiple system failures such as a major reactivity control system failure, reactor shutdown system failure, and safety rod drive system failure to insert safety rods cause 100% fuel melting.

Material-at-Risk — See Table D-26. The gaseous and volatile fission and activation product inventory at ATRC Facility scram following reactor operation at 5 kW for 7 days, followed by zero power for 23.4 days, in a cycle that is repeated for 60 years is 6.28E+03 Ci.

Release Characteristics — 100% fuel melt. The ATRC Facility is outside of the ATR confinement. The release is at ground level and it is assumed to last for 4 hours.

Airborne Release Fraction — 1.00E+0 for noble gases, 2.50E−1 for iodine, 5.00E−1 for cesium, and 3.00E−1 for tellurium (based on EDF TRA-ATRC-1755[9]).

Respirable Fraction — 1.0E+0 (assumption based on information SAR-192).

Damage Ratio 1.0E+0 (assumes fuel cladding damage).

Leak Path Factor 1.0E+0 (release is direct to the ATRC Facility main floor, which is outside of the confinement).

Adjustment Factor 1.0 (adjusts ST to represent the first 4 hours of a 4-hour release period).

Source Term The ST shown in Table D-29 was developed according to the following equation:
ST = (MAR × DR × LPF × ARF × RF) × AF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-29. Source term for scenario release designator ATRCFM-1.

Nuclide	MAR (Ci)	DR	LPF × RF	ARF	AF[a]	ST (Ci)
Kr-85m	5.35E+01	1.00E+0	1.00E+0	1.00E+0	1.00E+0	5.35E+1
Kr-88	1.11E+02	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.11E+2
Xe-133	3.10E+02	1.00E+0	1.00E+0	1.00E+0	1.00E+0	3.10E+2
Xe-133m	9.41E+00	1.00E+0	1.00E+0	1.00E+0	1.00E+0	9.41E+0
Cs-134	8.12E−01	1.00E+0	1.00E+0	5.00E−1	1.00E+0	4.06E−1
Cs-137	1.06E+00	1.00E+0	1.00E+0	5.00E−1	1.00E+0	5.30E−1
Te-127m	2.68E−01	1.00E+0	1.00E+0	3.00E−1	1.00E+0	8.04E−2
Te-129m	2.72E+00	1.00E+0	1.00E+0	3.00E−1	1.00E+0	8.16E−1
Te-132	1.77E+02	1.00E+0	1.00E+0	3.00E−1	1.00E+0	5.31E+1
I-131	1.16E+02	1.00E+0	1.00E+0	2.50E−1	1.00E+0	2.90E+1
I-132	1.76E+02	1.00E+0	1.00E+0	2.50E−1	1.00E+0	4.40E+1
I-133	2.68E+02	1.00E+0	1.00E+0	2.50E−1	1.00E+0	6.70E+1
I-134	2.98E+02	1.00E+0	1.00E+0	2.50E−1	1.00E+0	7.45E+1
I-135	2.50E+02	1.00E+0	1.00E+0	2.50E−1	1.00E+0	6.25E+1

a. No adjustment, as the release is assumed to occur over a 4-hour time period.

3. Scenario ATRCCRIT, Advanced Test Reactor Critical Facility Criticality

Detailed Scenario Description Fuel mishandling event, which positions seven or more fuel element in a critical configuration at bottom of the ATRC Facility canal or operator error resulting in over batching in the MCA outside of the ATRC Facility canal.

Material-at-Risk See Table D-25 (assumes 1E+18 fissions during criticality). In addition to the criticality, there will be a release of pre-existing fission products.

Release Characteristics Following an unplanned criticality, it is assumed that the fission products produced due to in-growth during the 20-minute period after the initiating criticality events are released to the atmosphere. The ATRC Facility is outside of the ATR confinement and the release is at ground level.

Airborne Release Fraction 1.00E+0 for noble gases, 2.50E−1 for iodine, 5.00E−1 for cesium, and 3.00E−1 for tellurium (based on EDF TRA-ATRC-1755).

Respirable Fraction 1.00E+0 (assumption based on information in SAR-192).

Damage Ratio 1.00E−1 (assumes 10% fuel cladding damage).

Leak Path Factor 5.5E−1 (EPA 550-B-99-009,[10] Appendix D, Section D.1.2).

Source Term The ST shown in Table D-30 was developed according to the following equation: ST = MAR × F × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table D-30. Source term for scenario release designator ATRCCRIT.

Nuclide	MAR (Ci)	DR	LPF × RF	ARF	AF[a]	ST (Ci)
Kr-85m	5.35E+01	1.00E−1	1.00E+0	1.00E+0	1.00E+0	5.35E+0
Kr-88	1.11E+02	1.00E−1	1.00E+0	1.00E+0	1.00E+0	1.11E+1
Xe-133	3.10E+02	1.00E−1	1.00E+0	1.00E+0	1.00E+0	3.10E+1
Xe-133m	9.41E+00	1.00E−1	1.00E+0	1.00E+0	1.00E+0	9.41E−1
Cs-134	8.12E−01	1.00E−1	1.00E+0	5.00E−1	1.00E+0	4.06E−2
Cs-137	1.06E+00	1.00E−1	1.00E+0	5.00E−1	1.00E+0	5.30E−2
Te-127m	2.68E−01	1.00E−1	1.00E+0	3.00E−1	1.00E+0	8.04E−3
Te-129m	2.72E+00	1.00E−1	1.00E+0	3.00E−1	1.00E+0	8.16E−2
Te-132	1.77E+02	1.00E−1	1.00E+0	3.00E−1	1.00E+0	5.31E+0
I-131	1.16E+02	1.00E−1	1.00E+0	2.50E−1	1.00E+0	2.90E+0
I-132	1.76E+02	1.00E−1	1.00E+0	2.50E−1	1.00E+0	4.40E+0
I-133	2.68E+02	1.00E−1	1.00E+0	2.50E−1	1.00E+0	6.70E+0
I-134	2.98E+02	1.00E−1	1.00E+0	2.50E−1	1.00E+0	7.45E+0
I-135	2.50E+02	1.00E−1	1.00E+0	2.50E−1	1.00E+0	6.25E+0

a. No adjustment, as the release is assumed to occur over a 4-hour time period.

D-1.5 Evaluation Results

D-1.5.1 Radiological Hazardous Material Release Results

Radiological hazardous material release results for 95% worst-case and 50% typical weather conditions as described in the main document are presented in Tables D-31 through D-34.

Table D-31. Advanced Test Reactor 4- and 48-hour and Advanced Test Reactor Critical Facility radiological material release scenario calculation results (total effective dose equivalent) for 95% worst-case meteorology.

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-1	Loss of primary coolant due to LPBO from failure of pressurizing system with no low-pressure coolant injection	5.02+3 (4 hr)	5.02E+2 (4 hr)	4.93E+2 (4 hr)	2.19E+0 (4 hr)	2.23E−1 (4 hr)	4,300 (4 hr)	400 (4 hr)	Site area emergency (SAE) (release less than or equal to 4 hours; potential for ingestion advisories)
		6.02E+4 (48 hr)	6.02E+3 (48 hr)	5.92E+3 (48 hr)	2.62E+1 (48 hr)	2.68E+0 (48 hr)	54,000 (48 hr)	1,800 (48 hr)	General emergency (GE) (release greater than 4 hours; potential for ingestion advisories)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-2	Loss of primary coolant due to HPBO from failure of emergency PCPs and failure to depressurize PCS	2.39E+4 (4 hr)	2.39E+3 (4 hr)	2.35E+3 (4 hr)	9.74E+0 (4 hr)	9.98E−1 (4 hr)	10,800[b] (4 hr)	1,200 (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		2.87E+5 (48 hr)	2.87E+4 (48 hr)	2.82E+4 (48 hr)	1.17E+2 (48 hr)	1.20E+1 (48 hr)	35,000 (48 hr)	3,100 (48 hr)	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-3	Loss of primary coolant due to small break (3 in. diameter) in coolant inlet piping and failure of low-pressure injection system	4.20E+4 (4 hr)	4.20E+3 (4 hr)	4.13E+3 (4 hr)	1.69E+1 (4 hr)	1.73E+0 (4 hr)	14,000[c] (4 hr)	1,500 (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		5.10E+5 (48 hr)	5.10E+4 (48 hr)	5.02E+4 (48 hr)	1.99E+2 (48 hr)	2.00E+1 (48 hr)	48,000 (48 hr)	4,150 (48 hr)	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-3M	Loss of primary coolant due to large break in outlet coolant piping, resulting in core flow stagnation	1.19E+4 (4 hr)	1.19E+3 (4 hr)	1.17E+3 (4 hr)	4.87E+0 (4 hr)	4.88E−1 (4 hr)	6,700 (4 hr)	810 (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		7.14E+4 (24 hr)	7.14E+3 (48 hr)	7.02E+3 (24 hr)	2.92E+1 (24 hr)	2.93E+0 (24 hr)	15,000 (24 hr)	1,810 (24 hr)	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-4	Anticipated transient without scram (loss of ATR control, which may result in loss of coolant)	5.02+3 (4 hr)	5.02E+2 (4 hr)	4.93E+2 (4 hr)	2.19E+0 (4 hr)	2.23E−1 (4 hr)	4,300 (4 hr)	400 (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		6.02E+4 (48 hr)	6.02E+3 (48 hr)	5.92E+3 (48 hr)	2.63E+1 (48 hr)	2.68E+0 (48 hr)	54,000 (48 hr)	1,800 (48 hr)	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-5	A perched fuel element drops into core after criticality is achieved, causing very large and rapid reactivity insertion that causes vessel movement and rupture of piping leading to loss of coolant	4.20E+4 (4 hr)	4.20E+3 (4 hr)	4.13E+3 (4 hr)	1.69E+1 (4 hr)	1.73E+0 (4 hr)	14,000 (4 hr)	1,500 (4 hr))	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		5.10E+5 (48 hr)	5.10E+4 (48 hr)	5.02E+4 (48 hr)	1.99E+2 (48 hr)	2.00E+1 (48 hr)	48,000 (48 hr)	4,150 (48 hr	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-5M	A sudden rupture of experiment loop pressure tube ruptures insulating gas envelope tube in core region of ATR flux trap; fuel melt due to extremely rapid, positive ramp insertion of reactivity in ATR core from voiding in a high positive reactivity worth experiment flux trap	8.41E+2 (4 hr)	8.41E+1 (4 hr)	8.27E+1 (4 hr)	3.39E−1 (4 hr)	3.46E−2 (4 hr)	2,130 (4 hr)	90 (4 hr)	SAE [based on thyroid committed dose equivalent (CDE) release less than or equal to 4 hours; potential for ingestion advisories]
		1.01E+4 (48 hr)	1.01E+3 (48 hr)	9.92E+2 (48 hr)	4.07E+0 (48 hr)	4.15E−1 (48 hr)	6,100 (48 hr)	700 (48 hr)	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem) 30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)	Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
ATRPDS-6	Interfacing system LOCA due to failure of valve or piping in an interfacing system connected to PCS and failure of low-pressure injection system	1.66E+3 (4 hr) 4.98E+3 (12 hr)	1.66E+2 (4 hr) 4.98E+2 (12 hr)	1.66E+2 (4 hr) 4.98E+2 (12 hr)	2.73E+0 (4 hr) 8.19E+0 (12 hr)	2.74E−1 (4 hr) 8.22E−1 (12 hr)	5,300 (4 hr) 9,800 (12 hr)	200 (4 hr) 575 (12 hr)	SAE
ATRPDS-7	Canal drain, loss of coolant water due to breach of canal wall or floor from dropping shield cask, falling overhead crane, or seismic event	7.09E+6 (2.5 hr)	7.09E+5 (2.5 hr)	6.97E+5 (2.5 hr)	2.85E+3 (92.5 hr)	2.86E+2 (2.5 hr)	105,000 (2.5 hr)	19,400 (2.5 hr)	GE (potential for ingestion advisories)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATREXPLR-1	Loss of experiment loop coolant due to break in PCS piping, causing fission product release from a 1-MW experiment	1.78E+3 (4 hr)	1.78E+2 (4 hr)	1.75E+2 (4 hr)	7.10E−1 (4 hr)	7.25E−2 (4 hr)	2,630 (4 hr)	200 (4 hr)	SAE (based on thyroid CDE release less than or equal to 4 hours; potential for ingestion advisories)
		1.94E+4 (48 hr)	1.94E+3 (48 hr)	1.91E+3 (48 hr)	7.63E+0 (48 hr)	7.75E−1 (48 hr)	10,000 (48 hr)	1,090 (48 hr)	GE (release greater than 4 hours; potential for ingestion advisories)
ATREXPLR-2	Loss of experiment loop coolant due to break in PCS piping, causing fission product release from a 200-kW experiment	3.56E+2 (4 hr) 4.27E+3 (48 hr)	3.56E+1 (4 hr) 4.27E+2 (48 hr)	3.50E+1 (4 hr) 4.20E+2 (48 hr)	1.423E−1 (4 hr) 1.70E+0 (48 hr)	1.45E−2 (4 hr) 1.74E−1 (48 hr)	1,575[d] (4 hr) 3,700 (48 hr)	30 (4 hr) 350 (48 hr)	SAE

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRFCB-1	Fuel element damage or fuel channel blockage due to failure of large structural elements in ATR vessel, above or in the core, with possible damage to PCS	5.92E+2 (4 hr)	5.92E+1 (4 hr)	5.82E+1 (4 hr)	2.43E−1 (4 hr)	2.43E−2 (4 hr)	1,900 (4 hr)	75 (4 hr)	SAE
		3.55E+3 (24 hr)	3.55E+2 (24 hr)	3.49E+2 (24 hr)	1.46E+0 (24 hr)	1.46E−1 (24 hr)	3,500 (24 hr)	330 (24 hr)	SAE
ATRCaskDp-1	Cask drop from above lift limits onto open ATR vessel may damage EFIS, causing experimental fuel to melt	7.45E+1 (4 hr)	7.45E+0 (4 hr)	7.33E+0 (4 hr)	3.05E−2 (4 hr)	3.06E−3 (4 hr)	570 (4 hr)	Not exceeded	SAE (4 hr)
		4.47E+2 (24 hr)	4.47E+1 (24 hr)	4.39E+1 (24 hr)	1.83E−1 (24 hr)	1.84E−2 (24 hr)	1,710 (24 hr)	Less than 100	SAE (24 hr)
TRA670RR-1 (ATR ground-level release)	Fuel cladding imperfections cause PCS concentrations greater than 20 µCi/mL	1.58E−1 (4 hr)	1.58E−2 (4 hr)	1.56E−2 (4 hr)	6.34E−5 (4 hr)	6.41E−6 (4 hr)	N/A	Not exceeded	Not an emergency
		1.90E+0 (48 hr)	1.90E−1 (48 hr)	1.87E−1 (48 hr)	7.61E−4 (48 hr)	2.24E−5 (48 hr)	100 (default distance)	Not exceeded	Alert

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRDLEXP-1	Dropped loop experiment due to uncoupling of irradiated experiment from transfer shield assembly, resulting in gamma radiation exposure	516 R/hr	46.5 R/hr	30.2 R/hr	5.66E−2 R/hr	N/A	815	68	SAE
ATRCIC-1	Dropped core internal changeout component or equipment due to uncoupling of irradiated core internal component or equipment from transfer shield assembly, resulting in gamma radiation exposure	516 R/hr	46.5 R/hr	30.2 R/hr	5.66E−2 R/hr	N/A	815	68	SAE

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRCCD-1	Fuel handling accident, control bridge drops onto ATRC Facility reactor core, damaging fuel cladding	3.54E+0	3.54E−1	3.48E−1	1.42E−3	1.42E−4	100	Not exceeded	Alert
ATRCFM-1	Severe accident, multiple system failures at maximum operating power cause fuel to melt	4.62E+2	4.62E+1	4.54E+1	1.83E−1	1.89E−3	1,710	60	SAE
ATRCCRIT (prompt neutron)	Criticality in ATRC Facility canal	5.80E+1[e]	5.10E+0[c]	5.00E+0[c]	Not calculated	Not calculated	N/A	N/A	Unclassified operational emergency (UOE)

Table D-31. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRCCRIT (Fission product release)	Criticality in ATRC Facility canal	4.18E+0	4.18E−1	4.11E−1	1.70E−3	1.71E−4	100	Not exceeded	Alert

a. TEDE = total effective dose equivalent
ICDF = Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility.
b. 7,100 m actual distance to PA Guide (PAG) 4 hours after release start.
c. 8,600 m actual distance to PAG 4 hours after release start.
d. The thyroid CDE is controlling and is used for all PA distances.
e. Derived from TRA-NMIS-1276.[11]

Table D-32. Advanced Test Reactor 4- and 48-hour and Advanced Test Reactor Critical Facility radiological material release scenario calculation results (thyroid committed dose equivalent) for 95% worst-case meteorology.

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-1	Loss of primary coolant due to LPBO from failure of pressurizing system with no low-pressure coolant injection	8.91E+4 (4 hr)	8.91E+3 (4 hr)	8.75E+3 (4 hr)	3.51E+1 (4 hr)	3.58E+0 (4 hr)	8,600 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		1.07E+6 (48 hr)	1.07E+5 (48 hr)	1.05E+5 (48 hr)	4.21E+2 (48 hr)	4.30E+1 (48 hr)	30,000 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-2	Loss of primary coolant due to HPBO from failure of emergency PCPs and failure to depressurize PCS	2.50E+5 (4 hr)	2.50E+4 (4 hr)	2.45E+4 (4 hr)	9.85E+1 (4 hr)	1.00E+1 (4 hr)	16,000[a] (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		3.00E+6 (48 hr)	3.00E+5 (48 hr)	2.94E+5 (48 hr)	1.18E+3 (48 hr)	1.20E+2 (48 hr)	50,000 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-3	Loss of primary coolant due to small break (3 in. diameter) in coolant inlet piping and failure of low-pressure injection system	6.44E+5 (4 hr)	6.44E+4 (4 hr)	6.33E+4 (4 hr)	2.54E+2 (4 hr)	2.59E+1 (4 hr)	27,000[b] (4 hr)	N/A	SAE (release equal to or less than 4 hours; potential for ingestion advisories)
		7.73E+6 (48 hr)	7.73E+5 (48 hr)	7.60E+5 (48 hr)	3.05E+3 (48 hr)	3.11E+2 (48 hr)	115,000 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-3M	Loss of primary coolant due to large break in outlet coolant piping, resulting in core flow stagnation	2.51E+5 (4 hr)	2.51E+4 (4 hr)	2.46E+4 (4 hr)	9.87E+1 (4 hr))	1.00E+1 (4 hr)	16,000 (4 hr)	N/A	SAE (release equal to or less than 4 hours; potential for ingestion advisories)
		1.51E+6 (24 hr)	1.51E+5 (24 hr)	1.48E+5 (24 hr)	5942E+2 (24 hr	6.00E+1 (24 hr)	32,500 (24 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-4	Anticipated transient without scram (loss of ATR control, which may result in loss of coolant)	8.91E+4 (4 hr)	8.91E+3 (4 hr)	8.75E+3 (4 hr)	3.51E+1 (4 hr)	3.58E+0 (4 hr)	8,600 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		1.07E+6 (48 hr)	1.07E+5 (48 hr)	1.05E+5 (48 hr)	4.21E+2 (48 hr)	4.30E+1 (48 hr)	30,000 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-5	A perched fuel element drops into core after criticality is achieved, causing very large and rapid reactivity insertion that causes vessel movement and rupture of piping leading to loss of coolant	6.44E+5 (4 hr)	6.44E+4 (4 hr)	6.33E+4 (4 hr)	2.54E+2 (4 hr)	2.59E+1 (4 hr)	27,000 (4 hr)	N/A	SAE (release equal to or less than 4 hours; potential for ingestion advisories)
		7.73E+6 (48 hr)	7.73E+5 (48 hr)	7.60E+5 (48 hr)	3.05E+3 (48 hr)	3.11E+2 (48 hr)	115,000 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-5M	A sudden rupture of experiment loop pressure tube ruptures insulating gas envelope tube in core region of ATR flux trap; fuel melt due to extremely rapid, positive ramp insertion of reactivity in ATR core from voiding in a high positive reactivity worth experiment flux trap	1.29E+4 (4 hr)	1.29E+3 (4 hr)	1.27E+3 (4 hr)	5.09E+0 (4 hr))	5.18E−1 (4 hr)	2,900 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		1.55E+5 (48 hr)	1.55E+4 (48 hr)	1.52E+4 (48 hr)	6.11E+1 (48 hr	6.22E+0 (48 hr)	12,000 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-6	Interfacing system LOCA due to failure of valve or piping in an interfacing system connected to PCS and failure of low-pressure injection system	3.09E+1 (4 hr)	3.09E+0 (4 hr)	3.04E+0 (4 hr)	1.12E−2 (4 hr)	7.06E−4 (4 hr)	N/A (4 hr)	N/A	Not an emergency (4 hr)
		9.27E+1 (12 hr)	9.27E+0 (12 hr)	9.12E+0 (12 hr)	3.36E−2 (12 hr)	2.12E−3 (12 hr)	210 (12 hr)	N/A	SAE (12 hr)
ATRPDS-7	Canal drain, loss of coolant water due to breach of canal wall or floor from dropping shield cask, falling overhead crane, or seismic event	8.60E+7 (2.5 hr)	8.60E+6 (2.5 hr)	8.46E+6 (2.5 hr)	3.38E+4 (2.5 hr)	3.41E+3 (2.5 hr)	105,000 (2.5 hr)	N/A	GE (potential for ingestion advisories)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATREXPLR-1	Loss of experiment loop coolant due to break in PCS piping, causing fission product release from a 1-MW experiment	2.28E+4 (4 hr)	2.28E+3 (4 hr)	2.24E+3 (4 hr)	8.98E+0 (4 hr))	9.16E−1 (4 hr)	3,900 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		2.74E+5 (48 hr)	2.74E+4 (48 hr)	2.69E+4 (48 hr)	1.08E+2 (48 hr	1.10E+1 (48 hr)	16,900 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)
ATREXPLR-2	Loss of experiment loop coolant due to break in PCS piping, causing fission product release from a 200-kW experiment	4.57E+3 (4 hr)	4.57E+2 (4 hr)	4.49E+2 (4 hr)	1.80E+0 (4 hr)	1.83E−1 (4 hr)	2,190 (4 hr)	N/A	SAE
		5.48E+4 (48 hr)	5.48E+3 (48 hr)	5.39E+3 (48 hr)	2.16E+1 (48 hr)	2.20E+0 (48 hr)	6,300 (48 hr)	N/A	SAE

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRFCB-1	Fuel element damage or fuel channel blockage due to failure of large structural elements in ATR vessel, above or in the core, with possible damage to PCS	1.25E+4 (4 hr)	1.25E+3 (4 hr)	1.23E+3 (4 hr)	4.92E+0 (4 hr)	5.00E−1 (4 hr)	2,830 (4 hr)	N/A	SAE
		7.50E+4 (24 hr)	7.59E+3 (24 hr)	7.38E+3 (24 hr)	2.95E+1 (24 hr)	3.00E+0 (24 hr)	7,600 (24 hr)	N/A	SAE
ATRCaskDp-1	Cask drop from above lift limits onto open ATR vessel may damage EFIS, causing experimental fuel to melt	1.57E+3 (4 hr)	1.57E+2 (4 hr)	1.55E+2 (4 hr)	6.19E−1 (4 hr)	6.28E−2 (4 hr)	1,450 (4 hr)	N/A	SAE (4 hr)
		9.42E+3 (24 hr)	9.42E+2 (24 hr)	9.30E+2 (24 hr)	3.71E+0 (24 hr)	3.77E−1 (24 hr)	2,650 (24 hr)	N/A	SAE (24 hr)

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA670RR-1 (ATR ground-level release)	Fuel cladding imperfections cause PCS concentrations greater than 20 µCi/mL	3.53E+0 (4 hr)	3.53E−1 (4 hr)	3.47E−1 (4 hr)	1.39E−3 (4 hr)	1.420E−4 (4 hr)	N/A	N/A	Not an emergency
		4.24E+1 (48 hr)	4.24E+0 (48 hr)	4.16E+0 (48 hr)	1.67E−2 (48 hr)	1.70E−3 (48 hr)	100 (48 hr)	N/A	Alert
ATRCCD-1	Fuel handling accident, control bridge drops onto ATRC Facility reactor core, damaging fuel cladding	4.09E+1	4.09E+0	4.02E+0	1.61E−2	1.62E−3	100	N/A	Alert

Table D-32. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRCFM-1	Severe accident, multiple system failures at maximum operating power cause fuel to melt	4.49E+2	4.49E+1	4.42E+1	1.77E−1	1.78E−2	680	N/A	SAE
ATRCCRIT	Criticality in ATRC Facility canal	4.51E+1	4.51E+0	4.43E+0	1.77E−2	1.79E−3	100	N/A	Alert

a. 9,000 m actual distance to PAG 4 hours after release start.
b. 10,850 m actual distance to PAG 3.7 hours after release start.

Table D-33. Advanced Test Reactor 4- and 48-hour and Advanced Test Reactor Critical Facility radiological material release scenario calculation results (total effective dose equivalent) for 50% typical meteorology.

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-1	Loss of primary coolant due to LPBO from failure of pressurizing system with no low-pressure coolant injection	3.40E+1 (4 hr)	3.40E+0 (4 hr)	2.30E+0 (4 hr)	6.85E−2 (4 hr)	1.04E−2 (4 hr)	320 (4 hr)	Not exceeded (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		4.08E+2 (48 hr)	4.08E+1 (48 hr)	2.76E+1 (48 hr)	8.22E−1 (48 hr)	1.25E−1 (48 hr)	2,600 (48 hr)	50 (48 hr)	SAE (release greater than 4 hours; potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-2	Loss of primary coolant due to HPBO from failure of emergency PCPs and failure to depressurize PCS	1.56E+2 (4 hr)	1.56E+1 (4 hr)	1.04E+1 (4 hr)	2.91E−1 (4 hr)	4.47E−2 (4 hr)	1,200 (4 hr)	50 (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		1.87E+3 (48 hr)	1.87E+2 (48 hr)	1.25E+2 (48 hr)	3.49E+0 (48 hr)	5.36E−1 (48 hr)	6,900 (48 hr)	200 (48 hr)	SAE (release greater than 4 hours; potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-3	Loss of primary coolant due to small break (3 in. diameter) in coolant inlet piping and failure of low-pressure injection system	2.73E+2 (4 hr)	2.73E+1 (4 hr)	1.83E+1 (4 hr)	5.04E−1 (4 hr)	7.73E−2 (4 hr)	1,950 (4 hr)	60 (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		3.28E+3 (48 hr)	3.28E+2 (48 hr)	2.22E+2 (48 hr)	6.05E+0 (48 hr)	9.28E−1 (48 hr)	10,300 (48 hr)	300 (48 hr)	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-3M	Loss of primary coolant due to large break in outlet coolant piping, resulting in core flow stagnation	7.80E+1 (4 hr)	7.80E+0 (4 hr)	5.24E+0 (4 hr)	1.48E−1 (4 hr)	2.24E−2 (4 hr)	615 (4 hr)	Not exceeded (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		4.68E+2 (24 hr)	4.68E+1 (48 hr)	3.14E+1 (24 hr)	8.88E−1 (24 hr)	1.35E−1 (24 hr)	2,700 (24 hr)	80 (24 hr)	SAE (release greater than 4 hours; potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-4	Anticipated transient without scram (loss of ATR control, which may result in loss of coolant)	3.40E+1 (4 hr)	3.40E+0 (4 hr)	2.30E+0 (4 hr)	6.85E−2 (4 hr)	1.04E−2 (4 hr)	320 (4 hr)	Not exceeded (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		4.08E+2 (48 hr)	4.08E+1 (48 hr)	2.76E+1 (48 hr)	8.22E−1 (48 hr)	1.25E−1 (48 hr)	2,600 (48 hr)	50 (48 hr)	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-5	A perched fuel element drops into core after criticality is achieved, causing very large and rapid reactivity insertion that causes vessel movement and rupture of piping leading to loss of coolant	2.73E+2 (4 hr)	2.73E+1 (4 hr)	1.83E+1 (4 hr)	5.04E−1 (4 hr)	7.73E−2 (4 hr)	1,950 (4 hr)	60 (4 hr)	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		3.28E+3 (48 hr)	3.28E+2 (48 hr)	2.20E+2 (48 hr)	6.05E+0 (48 hr)	9.28E−1 (48 hr)	10,300 (48 hr)	300 (48 hr)	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-5M	A sudden rupture of experiment loop pressure tube ruptures insulating gas envelope tube in core region of ATR flux trap; fuel melt due to extremely rapid, positive ramp insertion of reactivity in ATR core from voiding in a high positive reactivity worth experiment flux trap	5.46E+0 (4 hr)	5.46E−1 (4 hr)	3.65E−1 (4 hr)	1.01E−2 (4 hr)	1.55E−3 (4 hr)	100 (4 hr)	Not exceeded	SAE (based on thyroid CDE release less than or equal to 4 hours; potential for ingestion advisories)
		6.55E+1 (48 hr)	6.55E+0 (48 hr)	4.38E+0 (48 hr)	1.21E−1 (48 hr)	1.86E−2 (48 hr)	500 (48 hr)	Not exceeded	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-6	Interfacing system LOCA due to failure of valve or piping in an interfacing system connected to PCS and failure of low-pressure injection system	2.82E+1 (4 hr)	2.82E+0 (4 hr)	2.25E+0 (4 hr)	1.35E−1 (4 hr)	1.88E−2 (4 hr)	400 (4 hr)	Not exceeded (4 hr)	SAE
		8.46E+1 (12 hr)	8.46E+0 (12 hr)	6.75E+0 (12 hr)	4.05E−1 (12 hr)	5.64E−2 (12 hr)	1,470 (12 hr)	Not exceeded (12 hr)	SAE
ATRPDS-7	Canal drain, loss of coolant water due to breach of canal wall or floor from dropping shield cask, falling overhead crane, or seismic event	4.61E+4 (2.5 hr)	4.61E+3 (2.5 hr)	3.09E+3 (2.5 hr)	8.53E+1 (92.5 hr)	1.30E+1 (2.5 hr)	Greater than 25,000 (2.5 hr) (cannot extrapolate further)	2,600 (2.5 hr)	GE (potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATREXPLR-1	Loss of experiment loop coolant due to break in PCS piping, causing fission product release from a 1-MW experiment	1.15E+1 (4 hr)	1.15E+0 (4 hr)	7.66E−1 (4 hr)	2.09E−2 (4 hr)	3.22E−3 (4 hr)	130 (4 hr)	N/A (4 hr)	SAE (based on thyroid CDE release less than or equal to 4 hours; potential for ingestion advisories)
		1.38E+2 (48 hr)	1.38E+1 (48 hr)	9.19E+0 (48 hr)	2.51E−1 (48 hr)	3.86E−2 (48 hr)	1,040 (48 hr)	35 (48 hr)	SAE (release greater than 4 hours; potential for ingestion advisories)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATREXPLR-2	Loss of experiment loop coolant due to break in PCS piping, causing fission product release from a 200-kW experiment	2.30E+0 (4 hr)	2.30E−1 (4 hr)	1.54E−1 (4 hr)	4.19E−3 (4 hr)	6.46E−4 (4 hr)	100[a] (4 hr)	N/A (4 hr)	Alert (4 hr)
		2.76E+1 (48 hr)	2.76E+0 (48 hr)	1.85E+0 (48 hr)	5.03E−2 (48 hr)	7.75E−3 (48 hr)	230 (48 hr)	N/A (48 hr)	SAE (48 hr)
ATRFCB-1	Fuel element damage or fuel channel blockage due to failure of large structural elements in ATR vessel, above or in the core, with possible damage to PCS	3.89E+0 (4 hr)	3.89E−1 (4 hr)	2.62E−1 (4 hr)	7.38E−3 (4 hr)	1.12E−3 (4 hr)	55 (4 hr)	Not exceeded	Alert (4 hr)
		2.39E+1 (24 hr)	2.39E+0 (24 hr)	1.57E+0 (24 hr)	4.43E−2 (24 hr)	6.72E−3 (24 hr)	245 (24 hr)	Not exceeded	SAE (24 hr)

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRCaskDp-1	Cask drop from above lift limits onto open ATR vessel may damage EFIS, causing experimental fuel to melt	4.90E−1 (4 hr)	4.90E−2 (4 hr)	3.29E−2 (4 hr)	9.29E−4 (4 hr)	1.41E−4 (4 hr)	N/A (4 hr)	Not exceeded	Not an emergency (4 hr)
		2.94E+0 (24 hr)	2.94E−1 (24 hr)	1.97E−1 (24 hr)	5.57E−3 (24 hr)	8.46E−4 (24 hr)	100 (24 hr)	Not exceeded	Alert (24 hr)
TRA670RR-1 (ATR ground-level release)	Fuel cladding imperfections cause PCS concentrations greater than 20 µCi/mL	1.03E−3 (4 hr)	1.03E−4 (4 hr)	6.89E−5 (4 hr)	1.87E−6 (4 hr)	2.80E−7 (4 hr)	N/A	Not exceeded	Not an emergency
		1.24E−2 (48 hr)	1.24E−3 (48 hr)	8.27E−4 (48 hr)	2.24E−5 (48 hr)	3.36E−6 (48 hr)	N/A	Not exceeded	Not an emergency
ATRDLEXP-1	Dropped loop experiment due to uncoupling of irradiated experiment from transfer shield assembly, resulting in gamma radiation exposure	516 R/hr	46.5 R/hr	30.2 R/hr	5.66E−2 R/hr	N/A	815	68	SAE

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRCIC-1	Dropped core internal changeout component or equipment due to uncoupling of irradiated core internal component or equipment from transfer shield assembly, resulting in gamma radiation exposure	516 R/hr	46.5 R/hr	30.2 R/hr	5.66E−2 R/hr	N/A	815	68	SAE
ATRCCD-1	Fuel handling accident, control bridge drops onto ATRC Facility reactor core, damaging fuel cladding	2.31E−2	2.31E−3	1.54E−3	4.27E−5	6.47E−6	N/A	Not exceeded	Not an emergency

Table D-33. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRCFM-1	Severe accident, multiple system failures at maximum operating power cause fuel to melt	2.97E+0	2.97E−1	1.98E−1	5.35E−3	8.29E−4	100	Not exceeded	Alert
ATRCCRIT (Prompt neutron)	Criticality in ATRC Facility canal	5.80E+1[b]	5.10E+0[b]	5.00E+0[b]	Not calculated	Not calculated	N/A	N/A	UOE
ATRCCRIT (Fission product release)	Criticality in ATRC Facility canal	2.73E−2	2.73E−3	1.83E−3	5.14E−5	7.79E−6	N/A	Not exceeded	Not an emergency

a. The thyroid CDE is controlling and is used for all PA distances.
b. Derived from TRA-NMIS-1276.

Table D-34. Advanced Test Reactor 4- and 48-hour and Advanced Test Reactor Critical Facility radiological material release scenario calculation results (thyroid committed dose equivalent) for 50% typical meteorology.

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-1	Loss of primary coolant due to LPBO from failure of pressurizing system with no low-pressure coolant injection	5.71E+2 (4 hr)	5.71E+1 (4 hr)	3.81E+1 (4 hr)	1.02E+0 (4 hr)	1.58E−1 (4 hr)	820 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		6.85E+3 (48 hr)	6.85E+2 (48 hr)	4.57E+2 (48 hr)	1.22E+1 (48 hr)	1.90E+0 (48 hr)	5,700 (48 hr)	N/A	SAE (release greater than 4 hours; potential for ingestion advisories)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-2	Loss of primary coolant due to HPBO from failure of emergency PCPs and failure to depressurize PCS	1.60E+3 (4 hr)	1.60E+2 (4 hr)	1.07E+2 (4 hr)	2.87E+0 (4 hr)	4.43E−1 (4 hr)	2,110 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		3.19E+6 (48 hr)	3.19E+5 (48 hr)	3.13E+5 (48 hr)	1.26E+3 (48 hr)	1.28E+2 (48 hr)	11,400 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-3	Loss of primary coolant due to small break (3 in. diameter) in coolant inlet piping and failure of low-pressure injection system	4.13E+3 (4 hr)	4.13E+2 (4 hr)	2.75E+2 (4 hr)	7.41E+0 (4 hr)	1.14E+0 (4 hr)	3,850 (4 hr)	N/A	SAE (release equal to or less than 4 hours; potential for ingestion advisories)
		4.96E+4 (48 hr)	4.96E+3 (48 hr)	3.30E+3 (48 hr)	8.89E+1 (48 hr)	1.37E+1 (48 hr)	80,000 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-3M	Loss of primary coolant due to large break in outlet coolant piping, resulting in core flow stagnation	1.61E+3 (4 hr)	1.61E+2 (4 hr)	1.07E+2 (4 hr)	2.88E+0 (4 hr)	4.44E−1 (4 hr)	2,100 (4 hr)	N/A	SAE (release equal to or less than 4 hours; potential for ingestion advisories)
		9.66E+3 (24 hr)	9.66E+2 (24 hr)	6.42E+2 (24 hr)	1.73E+1 (24 hr)	2.66E+0 (24 hr)	6,900 (24 hr)	N/A	SAE (release greater than 4 hours; potential for ingestion advisories)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-4	Anticipated transient without scram (loss of ATR control, which may result in loss of coolant)	5.71E+2 (4 hr)	5.71E+1 (4 hr)	3.81E+1 (4 hr)	1.02E+0 (4 hr)	1.58E−1 (4 hr)	820 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		6.85E+3 (48 hr)	6.85E+2 (48 hr)	4.57E+2 (48 hr)	1.22E+1 (48 hr)	1.90E+0 (48 hr)	5,700 (48 hr)	N/S	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-5	A perched fuel element drops into core after criticality is achieved, causing very large and rapid reactivity insertion that causes vessel movement and rupture of piping leading to loss of coolant	4.13E+3 (4 hr)	4.13E+2 (4 hr)	2.75E+2 (4 hr)	7.41E+0 (4 hr)	1.14E+0 (4 hr)	3,850 (4 hr)	N/A	SAE (release equal to or less than 4 hours; potential for ingestion advisories)
		4.96E+4 (48 hr)	4.96E+3 (48 hr)	3.30E+3 (48 hr)	8.89E+1 (48 hr)	1.37E+1 (48 hr)	80,000 (48 hr)		GE (release greater than 4 hours; potential for ingestion advisories)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-5M	A sudden rupture of experiment loop pressure tube ruptures insulating gas envelope tube in core region of ATR flux trap; fuel melt due to extremely rapid, positive ramp insertion of reactivity in ATR core from voiding in a high positive reactivity worth experiment flux trap	8.27E+1 (4 hr)	8.27E+0 (4 hr)	5.51E+0 (4 hr)	1.48E−1 (4 hr)	2.29E−2 (4 hr)	185 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		9.92E+2 (48 hr)	9.92E+1 (48 hr)	6.61E+1 (48 hr)	1.78E+0 (48 hr)	2.75E−1 (48 hr)	1,490 (48 hr)	N/A	GE (release greater than 4 hours; potential for ingestion advisories)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRPDS-6	Interfacing system LOCA due to failure of valve or piping in an interfacing system connected to PCS and failure of low-pressure injection system	1.98E−1 (4 hr)	1.98E−2 (4 hr)	1.32E−2 (4 hr)	3.49E−4 (4 hr)	4.58E−5 (4 hr)	N/A	N/A	SAE (based on TEDE)
		5.94E−1 (12 hr)	5.94E−2 (12 hr)	3.96E−2 (12 hr)	1.05E−3 (12 hr)	1.37E−4 (12 hr)	N/A	N/A	SAE (based on TEDE)
ATRPDS-7	Canal drain, loss of coolant water due to breach of canal wall or floor from dropping shield cask, falling overhead crane, or seismic event	5.52E+5 (2.5 hr)	5.52E+4 (2.5 hr)	3.68E+4 (2.5 hr)	9.88E+2 (2.5 hr)	1.52E+2 (2.5 hr)	Greater than 25,000 (2.5 hr) (cannot extrapolate further)	N/A	GE (potential for ingestion advisories)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATREXPLR-1	Loss of experiment loop coolant due to break in PCS piping, causing fission product release from a 1-MW experiment	1.46E+2 (4 hr)	1.46E+1 (4 hr)	9.74E+0 (4 hr)	2.62E−1 (4 hr)	4.04E−2 (4 hr)	280 (4 hr)	N/A	SAE (release less than or equal to 4 hours; potential for ingestion advisories)
		1.75E+3 (48 hr)	1.75E+2 (48 hr)	1.17E+2 (48 hr)	3.14E+0 (48 hr)	4.85E−1 (48 hr)	2,230 (48 hr)	N/A	SAE (release greater than 4 hours; potential for ingestion advisories)
ATREXPLR-2	Loss of experiment loop coolant due to break in PCS piping, causing fission product release from a 200-kW experiment	2.93E+1 (4 hr)	2.93E+0 (4 hr)	1.95E+0 (4 hr)	5.25E−2 (4 hr)	8.10E−3 (4 hr)	100 (4 hr)	N/A	Alert (4 hr)
		3.52E+2 (48 hr)	3.52E+1 (48 hr)	2.34E+1 (48 hr)	6.30E−1 (48 hr)	9.72E−2 (48 hr)	530 (48 hr)	N/A	SAE (48 hr)

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
ATRFCB-1	Fuel element damage or fuel channel blockage due to failure of large structural elements in ATR vessel, above or in the core, with possible damage to PCS	8.02E+1 (4 hr)	8.02E+0 (4 hr)	5.35E+0 (4 hr)	1.44E−1 (4 hr)	2.21E−2 (4 hr)	180 (4 hr)	N/A (24 hr)	SAE (4 hr)
		4.81E+2 (24 hr)	4.81E+1 (24 hr)	3.21E+1 (24 hr)	8.64E−1 (24 hr)	1.33E−1 (24 hr)	720 (24 hr)	N/A (24 hr)	SAE (24 hr)
ATRCaskDp-1	Cask drop from above lift limits onto open ATR vessel may damage EFIS, causing experimental fuel to melt	1.01E+1 (4 hr)	1.01E+0 (4 hr)	6.734E−1 (4 hr)	1.81E−2 (4 hr)	2.78E−3 (4 hr)	100 (4 hr)	N/A	Alert
		6.06E+1 (24 hr)	6.062E+0 (24 hr)	4.04E+0 (24 hr)	1.09E−1 (24 hr)	1.67E−2 (24 hr)	120 (24 hr)	N/A	Alert

Table D-34. (continued).

Scenario Release Designator	Short Description	Downwind Thyroid CDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (124 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA670RR-1 (ATR ground-level release)	Fuel cladding imperfections cause PCS concentrations greater than 20 μCi/mL	2.26E−2 (4 hr)	2.26E−3 (4 hr)	1.53E−3 (4 hr)	4.04E−5 (4 hr)	6.16E−6 (4 hr)	N/A	N/A	Not an emergency
		2.71E−1 (48 hr)	2.71E−2 (48 hr)	1.84E−2 (48 hr)	4.85E−4 (48 hr)	7.39E−5 (48 hr)	N/A	N/A	Not an emergency
ATRCCD-1	Fuel handling accident, control bridge drops onto ATRC Facility reactor core, damaging fuel cladding	2.62E−1	2.62E−2	1.75E−2	4.70E−4	7.22E−5	N/A	N/A	Not an emergency
ATRCFM-1	Severe accident, multiple system failures at maximum operating power cause fuel to melt	2.88E+0	2.88E−1	1.92E−1	5.16E−3	7.93E−4	N/A	N/A	Not an emergency
ATRCCRIT	Criticality in ATRC Facility canal	2.89E−1	2.89E−2	2.93E−2	5.18E−4	7.96E−5	N/A	N/A	Not an emergency

D-1.6 Emergency Classes, Protective Actions, and Emergency Action Levels

Based on the analyses documented above, the following EALs are identified for TRA-670.

D-1.6.1 Unclassified Operational Emergency — Emergency Action Levels

Unless otherwise noted, the generic UOE EALs are covered by a separate appendix to this EHA.

Facility-specific UOE EALs are covered in this appendix.

D-1.6.1.1 ATR-670-10.OE.1

D-1.6.1.1.1 Event Description

Any unplanned criticality,

D-1.6.1.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

remote area monitor alarm while moving fissile material on the Advanced Test Reactor Critical Facility floor,

OR

log count-rate alarms while moving fuel into or out of the Advanced Test Reactor Critical Facility core,

AND

dose rate readings on radiological control technician-handheld instruments significantly higher than normal background.

RELATED INFORMATION:

This event is a precursor to emergency action level ATR-670-3.A.3.

D-1.6.1.1.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone around TRA-670.

Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616.

Control nonessential vehicle and personnel access to the exclusion zone.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure,"[12] as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.1.1.4 Offsite Protective Action Recommendations

None.

D-1.6.1.1.5 Basis. A criticality is considered a UOE, unless the criticality causes a radioactive material release in quantities that cause doses in excess of the PAC (Scenario ATRCCRIT). The PA distance is based on the radiological material release associated with the post-criticality 20-minute in-growth period. The PA distance is 100 m for a criticality.

D-1.6.2 Alert — Emergency Action Levels

D-1.6.2.1 ATR-670-3.A.1

D-1.6.2.1.1 Event Description

Elevated primary coolant system activity that may result in a radiological release,

D-1.6.2.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of a primary coolant system sample greater than or equal to 20 µCi/mL.

D-1.6.2.1.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone around TRA-670.

Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616.

Control nonessential vehicle and personnel access to the exclusion zone.

Verify implementation of AOP-0.1, “Operator Evacuation Procedure,” as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.2.1.4 Offsite Protective Action Recommendations

None.

D-1.6.2.1.5 Basis. ATR technical specifications require PCS concentrations less than 20 µCi/mL (Scenario TRA670RR-1).

D-1.6.2.2 ATR-670-3.A.2

D-1.6.2.2.1 Event Description

Radiological release from a fuel handling accident (control bridge drop onto Advanced Test Reactor Critical Facility core),

D-1.6.2.2.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual confirmation of the control bridge drop,

AND

remote area monitor alarm.

D-1.6.2.2.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone around TRA-670.

Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616.

Control nonessential vehicle and personnel access to the exclusion zone.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.2.2.4 Offsite Protective Action Recommendations

None.

D-1.6.2.2.5 Basis. During fuel changeout, the ATRC Facility control bridge fails and falls onto the ATRC Facility reactor core, damaging 100% of the ATRC Facility fuel (Scenario ATRCCD-1). The ATRC Facility canal bay area RAM alarms. The ATRC Facility canal is outside the ATR confinement area so the release path is via normal air exchange (four to ten air volume changes per hour).

D-1.6.2.3 ATR-670-3.A.3

D-1.6.2.3.1 Event Description

Any unplanned criticality resulting in a radiological release,

D-1.6.2.3.2 Event Recognition Factors and Related Information

AS INDICATED BY

remote area monitor alarm while moving fissile material on the Advanced Test Reactor Critical Facility floor,

OR

log count-rate alarms while moving fuel into or out of the Advanced Test Reactor Critical Facility core,

AND

dose rate readings on radiological control technician-handheld instruments significantly higher than normal background.

D-1.6.2.3.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone around TRA-670.

Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616.

Control nonessential vehicle and personnel access to the exclusion zone.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.2.3.4 Offsite Protective Action Recommendations

None.

D-1.6.2.3.5 Basis. Following an unplanned criticality (Scenario ATRCCRIT), it is assumed that the fission products produced due to in-growth during the 20-minute period after the initiating criticality event are released to the atmosphere. The ATRC Facility canal bay area RAM alarms, log count-rate alarms, and elevated count-rate alarms in the ATRC Facility canal area are observed. The ATRC Facility canal is outside the ATR confinement area so the release path is via normal air exchange (four to ten air exchanges per hour).

D-1.6.2.4 ATR-670-7.A.1

D-1.6.2.4.1 Event Description

Loss-of-coolant flow with potential for core damage,

D-1.6.2.4.2 Event Recognition Factors and Related Information

AS INDICATED BY

reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow),

AND

flow **not** restored within 30 minutes after reactor shutdown,

OR

following initial 30 minutes of flow after reactor shutdown, flow lost and **not** restored,

AND

reactor feed and bleed **not** demonstrated operable within 6 hours after reactor shutdown.

RELATED INFORMATION:

This event could be a precursor to emergency action levels ATR-670-7.SAE.1 and ATR-670-7.GE.3 or ATR-670-7.SAE.4 and ATR-670-7.GE.2.

D-1.6.2.4.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone around TRA-670.

Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616.

Control nonessential vehicle and personnel access to the exclusion zone.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.2.4.4 Offsite Protective Action Recommendations

None.

D-1.6.2.4.5 Basis. The loss of ATR coolant flow (Scenario ATRPDS-2) is a serious condition, which, if left unchecked, can result in evaporation of the primary coolant uncovering the ATR core, causing fuel melt. At this point, there are no indications of a radioactive material release. EAL ATR-670-7.A.1 could be a precursor to EALs ATR-670-7.SAE.1 and ATR-670-7.GE.3 or ATR-670-7.SAE.4 and ATR-670-7.GE.2, which are more serious in that some fuel damage has occurred and there are indications that a radioactive material release is occurring.

D-1.6.2.5 ATR-670-7.A.2

D-1.6.2.5.1 Event Description

Loss-of-coolant accident with failure of the emergency firewater injection system,

D-1.6.2.5.2 Event Recognition Factors and Related Information

AS INDICATED BY

primary coolant leak rate greater than 300 gpm,

AS INDICATED BY

flow balance [pressurizing flow FIC-1-8B or -8A to PCV-1-1 flow (PCS F• DEGAS)],

AND

emergency firewater injection system pressure less than 58 psig,

AS INDICATED BY

PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure).

RELATED INFORMATION:

This event could be a precursor to emergency action levels ATR-670-7.SAE.2 and ATR-670-7.GE.1 or ATR-670-7.SAE.3.

D-1.6.2.5.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone around TRA-670.

Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616.

Control nonessential vehicle and personnel access to the exclusion zone.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.2.5.4 Offsite Protective Action Recommendations

None.

D-1.6.2.5.5 Basis. ATR LOCA results from a break (small or large diameter) in the PCS piping (Scenarios ATRPDS-3, -3M, or -6). Depending on break size and location (inlet or outlet piping) and operability of the EFIS, the time to core uncovery and fuel melting is quite variable. If an inlet pipe break is small (3-in. diameter or less), core uncovery may or may not occur if the EFIS is operational. If core uncovery does occur, it will be in the hours timeframe after the pipe break occurs. On the other hand, if a large diameter pipe break (24-in. diameter) occurs in the outlet piping, core uncovery and fuel damage could occur within a matter of minutes. At this point, there are no indications of a radioactive material release. EAL ATR-670-7.A.2 could be a precursor to EALs ATR-670-7.SAE.2 and ATR-670-7.GE.1 or ATR-670-7.SAE.3, which are more serious events in that fuel damage and a radioactive material release have occurred.

D-1.6.2.6 ATR-670-8.A.1

D-1.6.2.6.1 Event Description

Reactor **<u>not</u>** subcritical after completion of FRP-S.1, "Uncontrolled Power Generation/ATWS"[13] (function restoration procedure subcriticality 1).

D-1.6.2.6.2 Event Recognition Factors and Related Information

RELATED INFORMATION:

This event could be a precursor to emergency action levels ATR-670-8.SAE.1 and ATR-670-8.GE.1 or ATR-670-8.SAE.2 and ATR-670-8.GE.2.

D-1.6.2.6.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone around TRA-670.

Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616.

Control nonessential vehicle and personnel access to the exclusion zone.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.2.6.4 Offsite Protective Action Recommendations

None.

D-1.6.2.6.5 Basis. ATR is not responding correctly to reactor control signals (Scenario ATRPDS-4 or -5). In essence, ATR is out of control. Some type of transient scrams the ATR, but ATR does not completely shut down. This is a very unstable reactor condition so it is necessary to take some precautionary PAs, even though there has not been a radioactive material release. This event may be a precursor to EALs ATR-670-8.SAE.1 and ATR-670-8.GE.1 or ATR-670-8.SAE.2 and ATR-670-8.GE.2.

D-1.6.2.7 ATR-670-11.A.1

D-1.6.2.7.1 Event Description

Potential for an Advanced Test Reactor canal drain event,

D-1.6.2.7.2 Event Recognition Factors and Related Information

AS INDICATED BY

canal water level alarm,

OR

visual observation of the canal water level decreasing faster than the emergency fire water injection system makeup rate.

RELATED INFORMATION:

This event could be a precursor to emergency action level ATR-670-3.GE.1.

D-1.6.2.7.3 Onsite Protective Actions

Establish a **100-m (328-ft)** exclusion zone, or other distance recommended by the incident commander, around TRA-670.

Relocate nonessential personnel from the exclusion zone to the cafeteria in TRA-616.

Control nonessential vehicle and personnel access to the exclusion zone.

D-1.6.2.7.4 Offsite Protective Action Recommendations

None.

D-1.6.2.7.5 Basis. The water level in the ATR canal is decreasing and, if a makeup source of water is not available, the fuel stored in the canal is uncovered. This causes a source to direct gamma radiation in the canal area and eventually the ATR core most recently put into storage begins to melt the fuel cladding, causing a radioactive material release. At this point, there has not been any fuel melting and no radioactive material release. This event is a precursor to EAL ATR-670-3.GE.1.

D-1.6.3 Site Area Emergency — Emergency Action Levels

D-1.6.3.1 ATR-670-3.SAE.1

D-1.6.3.1.1 Event Description

Experiment loop loss-of-coolant accident with a radiological release from a less than or equal to 200-kW fueled experiment,

D-1.6.3.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

loop instrumentation indicating low-low pressurizer level and low-low pressure alarm,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm.

D-1.6.3.1.3 Onsite Protective Actions

Evacuate nonessential personnel at least **2,190 m (7,185 ft or 1.4 mi)** in all directions from the Advanced Test Reactor Complex.

Control nonessential vehicle and personnel access to the evacuated area.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.

Consider authorizing potassium iodide for essential emergency workers.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.1.4 Offsite Protective Action Recommendations

None.

D-1.6.3.1.5 Basis. Scenario ATREXPLR-2, as described in SAR-153, Chapter 15, Subsection 15.7, is an experiment loop LOCA that assumes complete meltdown of a 200-kW test. The experiment loop cooling system is isolated from the PCS so an experiment loop LOCA does not affect the PCS.

The pressurized water experiment loop facilities described in SAR-153, Chapter 10, are systems whose failure results in the uncontrolled release of radioactivity to the environment under transient conditions. The testing program includes irradiation of fueled tests in the experiment loops. A loss of flow or LOCA in the experiment loop could result in melting of the test. Both a 1/2-in. experiment loop LOCA and an experiment loop flow coastdown are considered Condition 2 occurrences. The Condition 2 loop flow coastdown is terminated by a loop scram that should prevent fuel damage. A larger than 1/2-in. experiment loop LOCA and loss of flow without a loop scram are considered Condition 4 events. If test damage occurred, only the LOCA events allow fission product release into the loop cubicle. Any fission products released during a loss-of-flow event remain contained in the loop piping and the release to TRA-670 and the environment is enveloped by those from the LOCA.

D-1.6.3.2 ATR-670-3.SAE.2

D-1.6.3.2.1 Event Description

Radiological release from a fuel damage event,

D-1.6.3.2.2 Event Recognition Factors and Related Information

AS INDICATED BY

radiological control technician/chemist report of increased primary coolant activity levels greater than 20 μCi/mL following a power variation greater than 3%,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm.

D-1.6.3.2.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **2,830 m (9,285 ft or 1.8 mi)** of the Advanced Test Reactor Complex.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate.

Control nonessential vehicle and personnel access to the evacuated areas.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.

Consider authorizing potassium iodide for essential emergency workers.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.2.4 Offsite Protective Action Recommendations

None.

D-1.6.3.2.5 Basis. Scenario ATRFCB-1, as described in SAR-153, Chapter 15, Subsection 15.10.3, assumes that two complete fuel elements suffer extensive damage. Fuel element damage or fuel channel blockage is due to failure of large structural elements in the ATR vessel, above or in the core, with possible damage to the PCS. However, it is assumed that the PCS remains intact. It is estimated that the resulting dose is less than 5% of that calculated for Scenario ATRPDS-3M. Based on SAR-153, it is assumed that confinement leakage is 99% per day since there is emergency firewater injection. It is assumed that the ST is totally released to the environment over 24 hours.

Failure of large structural elements in the ATR vessel, above or in the core, could block the fuel element coolant channel or severely damage fuel elements. Both events could cause fuel failure without the occurrence of ATR or PCS accidents. Both events are considered Condition 4 events and envelop those events that result in large blockages.

Gross fuel element damage may result in the release of a significant amount of fission products into the PCS. However, without a PCS break, most fission products are retained in the PCS, except for gaseous fission products and those escaping due to PCS leakage. Because of the serpentine geometry of the ATR core, it is improbable that a substantial portion of the core would be involved for a major component failure. Based on engineering judgment, it is assumed that no more than two complete fuel elements would be affected. This judgment is based on the serpentine core design, which makes it difficult to block multiple elements and historical experience with material found in the vessel.

CAM and RAM alarms indicate that a radioactive material release is occurring.

D-1.6.3.3 ATR-670-3.SAE.3

D-1.6.3.3.1 Event Description

Radiological release from dropping highly radioactive components from a cask or problems occurring during an open air transfer of highly radioactive components that preclude immediately returning the components to the reactor vessel or canal,

D-1.6.3.3.2 Event Recognition Factors and Related Information

AS INDICATED BY

direct observation of the event,

AND

multiple continuous air monitor and remote area monitor alarms,

OR

radiological control technician confirmation by radiation survey.

D-1.6.3.3.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **1,000 m (3,281 ft)** of the Advanced Test Reactor Complex.

Control nonessential vehicle and personnel access to the evacuated area.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.3.4 Offsite Protective Action Recommendations

None.

D-1.6.3.3.5 Basis. Every 6 years or so, maintenance must be conducted on the ATR, which involves the removal and replacement of nonfuel core internal components and equipment. These core internal components have been subjected to an intense neutron flux during ATR operations and consequently the core components have become highly radioactive due to neutron activation. These neutron-activated core components emit gamma radiation and the gamma exposure rate may range from 1E6 to 5E6 R/hr at 1 ft (0.3048 m). If these core internal components are dropped from the transfer shield, significant exposure rates could occur across the ATR Complex (Scenario ATRCIC-1).

D-1.6.3.4 ATR-670-3.SAE.4

D-1.6.3.4.1 Event Description

Radiological release from an unshielded dropped loop experiment accident,

D-1.6.3.4.2 Event Recognition Factors and Related Information

AS INDICATED BY

direct observation of the event,

AND

multiple continuous air monitor and remote area monitor alarms in the reactor control room,

OR

radiological control technician confirmation by radiation survey.

D-1.6.3.4.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **1,000 m (3,281 ft)** of the Advanced Test Reactor Complex.

Control nonessential vehicle and personnel access to the evacuated area.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.4.4 Offsite Protective Action Recommendations

None.

D-1.6.3.4.5 Basis. The pressurized water loop facilities contain tests with a significant inventory of radioactive material. The radioactive material contained in the tests could be released by dropping a test out of the cask during test handling. Based on an evaluation for this event,[3,14] gamma radiation levels at the ATR Complex main parking lot at 500 m could be approximately 2 R/hr. Table D-23 provides exposure rate information at several distances, which is based on an initial exposure rate of 5.0E6 R/hr at 1 ft (0.3048 m) and applying the inverse square equation (Scenario ATRDLEXP-1).

D-1.6.3.5 ATR-670-3.SAE.5

D-1.6.3.5.1 Event Description

Experiment loop loss-of-coolant accident with a radiological release from a greater than 200-kW and less than or equal to 1-MW fueled experiment,

D-1.6.3.5.2 Event Recognition Factors and Related Information

AS INDICATED BY

loop instrumentation indicating low-low pressurizer level and low-low pressure alarm,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

OR

radiological control technician confirmation of the release by radiation survey,

AND

release ongoing **less than or equal to 4 hours.**

RELATED INFORMATION:

This event is a precursor to emergency action level ATR-670-3.GE.3.

D-1.6.3.5.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **4,000 m (13,125 ft or 2.5 mi)** of the Advanced Test Reactor Complex.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate.

Control nonessential vehicle and personnel access to the evacuated areas.

Consider authorizing potassium iodide for essential emergency workers.

Relocate the emergency control center to the alternate emergency control center at CF-609.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.5.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.3.5.5 Basis. Scenario ATREXPLR-1, as described in SAR-153, Chapter 15, Subsection 15.7, is an experiment loop LOCA that assumes complete meltdown of a 1-MW test (current technical specification limits the experiment size to 200 kW so an exemption is required to run a 1-MW test). The experiment loop cooling system is isolated from the PCS so an experiment loop LOCA does not affect the PCS.

The pressurized water experiment loop facilities described in SAR-153, Chapter 10, are systems whose failure results in the uncontrolled release of radioactivity to the environment under transient conditions. The testing program includes irradiation of fueled tests in the experiment loops. A loss of flow or LOCA in the experiment loop could result in melting of the test. Both a 1/2-in. experiment loop LOCA and an experiment loop flow coastdown are considered Condition 2 occurrences. The Condition 2 loop flow coastdown is terminated by a loop scram that should prevent fuel damage. A larger than 1/2-in. experiment loop LOCA and loss of flow without a loop scram are considered Condition 4 events. If test damage occurred, only the LOCA events allow fission product release into the loop cubicle. Any fission products released during a loss-of-flow event remain contained in the loop piping and the release to TRA-670 and the environment is enveloped by those from the LOCA.

D-1.6.3.6 ATR-670-3.SAE.6

D-1.6.3.6.1 Event Description

Radiological release from a fuel melt caused by an experimental reactivity insertion event,

D-1.6.3.6.2 Event Recognition Factors and Related Information

AS INDICATED BY

loop instrumentation indicating low-low pressurizer level and low-low pressure alarm,

AND

fission break monitor system alarms,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

OR

radiological control technician confirmation of the release by radiation survey,

AND

release ongoing **less than or equal to 4 hours.**

RELATED INFORMATION:

This event is a precursor to emergency action level ATR-670-3.GE.2.

D-1.6.3.6.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **3,000 m (9,840 ft or 1.9 mi)** of the Advanced Test Reactor Complex.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate.

Control nonessential vehicle and personnel access to the evacuated areas.

Consider authorizing potassium iodide for essential emergency workers.

Relocate the emergency control center to the alternate emergency control center at CF-609.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.6.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.3.6.5 Basis. The sudden rupture of an experiment loop pressure tube causes a sudden rupture of the insulating gas envelope tube in the core region of an ATR flux trap. This in turn causes an extremely rapid, positive ramp insertion of reactivity in the ATR core due to voiding in a high positive reactivity worth experiment flux trap (Scenario ATRPDS-5M). This event of a rupture in an experiment loop may result in fuel melt with vapor explosion. It is expected that the release and potential doses to receptors are significantly less than for other core damage events since the PCS remains intact and less than 2% of the core is expected to melt. It is assumed that confinement leakage is 51% per day. The ST is assumed to totally release to the environment over 2 days. For calculated radiological doses, it is assumed that this ST released is that of the first 4 hours (4 hr/48 hr).

Scenario ATRPDS-5M is a reactivity insertion accident (RIA) initiated by experiment loop ruptures that are severe enough to result in fuel melting before the ATR scram can be effective in terminating the reactivity transient. This event category is represented by a single event referred to as the flux trap voiding accident (FTVA). The FTVA is a bounding, extremely rapid, positive ramp insertion of reactivity in the ATR core due to voiding in a high positive reactivity worth experiment flux trap. The initiating event for the FTVA is a sudden rupture of an experiment loop pressure tube, which then also causes a sudden rupture of the insulating gas envelope tube in the core region of an ATR core flux trap. The double rupture results in an expulsion of high-temperature, high-pressure loop water into the relatively low-pressure (ATR vessel pressure) flux trap annulus between the gas envelope tube and the flux trap baffle, which very rapidly voids the flux trap annulus. Since ATR flux traps, including the annulus, have a positive void reactivity coefficient, ATR experiences a very rapid positive reactivity insertion and power transient. This rapid positive reactivity insertion and the resulting power transient can be large if the event occurred in a flux trap position with a high potential reactivity worth in the flux trap annulus. The large negative reactivity feedback from the core fuel terminates the transient as the fuel and core coolant heat up in response to the power rise, and a high-power scram shuts down ATR.

Although this accident results in a significantly damaged core and internals, the vessel remains intact and water full. Therefore, the water and vessel barriers to the release of fission products remain. Some early fission product release may occur from lifting of the relief valves during the pressure transient. Following the very short pressure transient, fission product release from the vessel and PCS is by primary coolant leakage. This leakage may increase if relief valves fail to reseat, or if seals are damaged by the severe pressure transient. But, the PCS and support systems remain operable and therefore, the coolant inventory can be controlled and most of the soluble fission products retained and cooled within the vessel. Nonsoluable, gaseous fission products, primarily noble gases, may initially be released through normal operation of the degassing tank and vented through the ATR stack. But, the stack breach radiation monitors detect the release and isolate the stack and confinement very shortly after the accident. Therefore, only a small fraction of the fission products available for release are released from the vessel into the confinement. Although no specific fission product release or dose analyses have been performed for this accident, the release and the potential doses to onsite and offsite receptors are significantly less than for the other core damage events for which fission product release and consequence analyses have been done (i.e., complete loss-of-coolant flow accident and LOCAs).

D-1.6.3.7 ATR-670-3.SAE.7

D-1.6.3.7.1 Event Description

Multiple system failures at maximum operating power leads to an Advanced Test Reactor Critical Facility fuel melt and a radiological release,

D-1.6.3.7.2 Event Recognition Factors and Related Information

AS INDICATED BY

major reactivity control system failure,

AND

reactor safety system failure,

AND

safety rod drive system failure to insert safety rods,

AND

remote area monitor alarm.

D-1.6.3.7.3 Onsite Protective Actions

Evacuate nonessential personnel at least **1,710 m (5,611 ft or 1.1 mi)** in all directions from the Advanced Test Reactor Complex.

Control nonessential vehicle and personnel access to the evacuated area.

Consider authorizing potassium iodide for essential emergency workers.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.7.4 Offsite Protective Action Recommendations

None.

D-1.6.3.7.5 Basis. Multiple system failures such as a major reactivity control system failure, reactor shutdown system failure, and safety rod drive system failure to insert safety rods cause 100% fuel melting (Scenario ATRCFM-1). Indicators of the system failures are observed and the ATRC Facility canal bay area RAM alarms. The ATRC Facility canal is outside the ATR confinement area so the release path is via normal air exchange (four to six air changes per hour).

D-1.6.3.8 ATR-670-3.SAE.8

D-1.6.3.8.1 Event Description

Radiological release from an experiment fuel melt caused by a cask drop from above lift limits that damages the reactor cooling system,

D-1.6.3.8.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the cask drop,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

OR

radiological control technician confirmation of the release by radiation survey.

D-1.6.3.8.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **1,450 m (4,757 ft or 0.9 mi)** of TRA-670.

NOTE: *If wind persists in the same direction longer than 4 hours, evacuation distance should be extended to **2,650 m (8,695 ft or 1.6 mi).***

Control nonessential vehicle and personnel access to the evacuated area.

Consider authorizing potassium iodide for essential emergency workers.

Relocate the emergency control center to the alternate emergency control center at CF-609.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.8.4 Offsite Protective Action Recommendations

None.

D-1.6.3.8.5 Basis. During heavy lift operations over the ATR vessel, the fissile material limits for ATR or experiments are set at 275 g of U-235 (i.e., ATR must be defueled to the point where there is no greater than 275 g of fissile material present before the heavy lift is initiated). The cask drop event (Scenario ATRCaskDp-1) assumes that the EFIS may be damaged, cooling is lost, and the 275-g U-235 experimental fuel is melted, releasing fission products to the reactor building confinement.

D-1.6.3.9 ATR-670-7.SAE.1

D-1.6.3.9.1 Event Description

Loss-of-coolant flow with potential for core damage and a radiological release,

D-1.6.3.9.2 Event Recognition Factors and Related Information

AS INDICATED BY

reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow),

AND

flow **not** restored within 30 minutes after reactor shutdown,

OR

following initial 30 minutes of flow after reactor shutdown, flow lost and **not** restored,

AND

reactor feed and bleed **not** demonstrated operable within 6 hours after reactor shutdown,

AND

primary coolant system **not** vented to atmosphere,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

release ongoing **less than or equal to 4 hours.**

RELATED INFORMATION:

This event is a precursor to emergency action level ATR-670-7.GE.3.

D-1.6.3.9.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **9,000 m (29,530 ft or 5.6 mi)** in the downwind direction.

Consider evacuating the Critical Infrastructure Test Range Complex if it is in the downwind direction.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate.

Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I.

Consider authorizing potassium iodide for essential emergency workers.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.9.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.3.9.5 Basis. Scenario ATRPDS-2 was initiated by a loss of commercial power, which also scrams ATR. The emergency PCPs were lost after 4 hours when the battery backup power was expended. There was no intentional depressurization through the ATR vessel vent valves. Natural circulation was established within the ATR vessel with liquid flowing up through the core and down through the reflector; some flow also went through the siphon breaker lines from the top of the ATR vessel to the outlet lines.

The water in the ATR vessel slowly heated up, reaching saturation in the core after 8.5 hours. As the coolant was heated, the pressure increased. The ATR vessel relief valves first lifted at 14.4 hours, then cycled to control the pressure through the rest of the transient. Continued boiling in the core, with flow through the relief valves, resulted in a decreasing liquid level in the vessel. The level reached the top of the core near 28.7 hours. Fuel melting temperatures were reached by 29 hours. With no coolant makeup available, the entire core melted with a resultant release of all the highly-volatile fission products to the PCS. This event is a precursor to EAL ATR-670-7.GE.3.

A release to the ATR confinement is occurring as indicated by CAM and RAM alarms.

NOTE: *Since RSAC calculates an instantaneous dose based on the full ST (4-hours exposure at each identified receptor location), the actual EAL distances determined for 4 hours after the release start were based on wind speed, which determines the actual duration of exposure to the airborne plume. Therefore, the 4-hour dose projections (Tables D-31 and D-32) have been adjusted accordingly to determine the PA distance.*

D-1.6.3.10 ATR-670-7.SAE.2

D-1.6.3.10.1 Event Description

Loss-of-coolant accident bounds both small break and large break loss-of-coolant accidents with failure of emergency firewater injection system that could result in core damage and fuel melting with consequent radiological release,

D-1.6.3.10.2 Event Recognition Factors and Related Information

AS INDICATED BY

primary coolant leak rate less than 300 gpm,

AS INDICATED BY

flow balance [pressurizing flow FIC-1-8B or -8A to PCV-1-1 flow (PCS F• DEGAS)],

AND

emergency firewater injection system pressure less than 58 psig,

AS INDICATED BY

PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure),

AND

vessel coolant level below the emergency firewater injection system injection level and decreasing,

AS INDICATED BY

LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing,

AND

multiple continuous air monitor and remote area monitor alarms and/or stock monitor alarm,

AND

release ongoing **less than or equal to 4 hours.**

RELATED INFORMATION:

This event is a precursor to emergency action level ATR-670-7.GE.1.

D-1.6.3.10.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **10,850 m (35,599 ft or 6.75 mi)** in the downwind direction.

Consider evacuating the Critical Infrastructure Test Range Complex if it is in the downwind direction.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate.

Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Consider authorizing potassium iodide for essential emergency workers.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.10.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.3.10.5 Basis. ATR LOCA results from a break (small or large diameter) in the PCS piping (Scenario ATRPDS-3 or -3M). Depending on break size and location (inlet or outlet piping) and operability of the EFIS, the time to core uncovery and fuel melting is quite variable. If an inlet pipe break is small (3-in. diameter or less), core uncovery may or may not occur if the EFIS is operational. If core uncovery does occur, it will be in the hours timeframe (10 hours) after the pipe break occurs (Scenario ATRPDS-3). On the other hand, if a large diameter pipe break (24-in. diameter) occurs in the outlet piping, core uncovery and fuel damage could occur within a matter of minutes (Scenario ATRPDS-3M).

Although the probable break locations for the small LOCA, which dominates this event frequency, are outside the radiographic boundary for the PCS (the radiographic boundary is discussed in SAR-153, Chapter 5), for the consequence analysis, this transient was initiated by an assumed 3-in.-diameter break in the outlet piping. The assumed break location, a partial break of one of the 18-in. outlet lines below the core elevation, was selected to try to provide a vapor-filled path between the core and the break so that fission product transport to the gas space in the confinement is maximized. The PCS depressurized rapidly with scram occurring on low inlet pressure within 2 seconds. The PCPs and emergency PCPs were tripped when their respective suction pressures were well below atmospheric pressure. This simulated the flow degradation that occurs as these pumps were operated below the minimum required net positive suction head (NPSH). Although the PCS was depressurized to atmospheric pressure, the low-pressure injection systems failed to provide coolant makeup. This led to a low-pressure core uncovering near 1.5 hours that is similar to the complete loss-of-flow accident described in Scenario ATRPDS-1.

Scenario ATRPDS-3M is associated with a "direct damage" LOCA in which the initiating event alone is sufficient to cause fuel melting. The fuel melting is due to flow stagnation in the core. With the low pressures and high heat fluxes that exist early in a large break LOCA, critical heat flux occurs, resulting in rapidly increasing fuel temperatures. With the large volumetric expansion associated with boiling at these subatmospheric pressures, and no large driving force to encourage flow through the core, liquid is not easily reintroduced to the flow channels in the core.

A 24-in. outlet break, initiated from full power of 250 MW, was analyzed in detail. The blowdown resulted in subatmospheric pressures in the vessel within 1 second because the top of the vessel was about 55 ft above the break, which is at atmospheric pressure. The PCPs and emergency PCPs were tripped on suction pressures of 2 psi to simulate flow degradation below the NPSH requirements of the pump. The low-pressure trip for emergency firewater injection was disabled to maximize the core damage; early calculations indicated that the firewater quenched the core as the heat of fusion was added to the fuel, thus preventing significant fission product release. Fuel melting began near 85 seconds and relocation of molten material to the flow distribution tank occurred between 100 and 110 seconds. The liquid level in the upper plenum dropped below the EFIS setpoint near 174 seconds and firewater injection to the vessel began at about 179 seconds. The injection flow was able to recover the liquid level in the ATR vessel. Although the PCS thermal-hydraulic calculation was stopped at 200 seconds, the transient continues with intermittent operation of the EFIS and injection occurring as the level drops below 87 ft, then stopping once the level exceeds 90 ft. This continues until the confinement fills sufficiently to a level above 87 ft that can be maintained in the ATR vessel. This event is a precursor to EAL RTC-670-7.GE.1.

A release to the ATR confinement is occurring as indicated by CAM and RAM alarms.

NOTE: *Since RSAC calculates an instantaneous dose based on the full ST (4-hours exposure at each identified receptor location), the actual EAL distances determined for 4 hours after the release start were based on wind speed, which determines the actual duration of exposure to the airborne plume. Therefore, the 4-hour dose projections (Tables D-31 and D-32) have been adjusted accordingly to determine the PA distance.*

D-1.6.3.11 ATR-670-7.SAE.3

D-1.6.3.11.1 Event Description

Radiological release from an interfacing system loss-of-coolant accident,

D-1.6.3.11.2 Event Recognition Factors and Related Information

AS INDICATED BY

primary coolant leak rate greater than 300 gpm,

AS INDICATED BY

flow balance [pressurizing flow FIC-1-8B or -8A to PCV-1-1 flow (PCS F• DEGAS)],

AND

emergency firewater injection system pressure less than 58 psig,

AS INDICATED BY

PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure),

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm.

D-1.6.3.11.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex personnel and field workers within **5,300 m (17,389 ft or 3.3 mi)** of the Advanced Test Reactor Complex.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate.

Control nonessential vehicle and personnel access to the evacuated areas.

Depending on radiological conditions, consider relocating the emergency control center to the alternate emergency control center at CF-609.

Consider authorizing potassium iodide for essential emergency workers.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.11.4 Offsite Protective Action Recommendations

None.

D-1.6.3.11.5 Basis. This event of a rupture in an interfacing system (e.g., small pipe break outside radiographic boundary and outside confinement) may result in fuel melt (Scenario ATRPDS-6). It is expected that the release and potential doses to receptors are significantly less than for other core damage events because the PCS remains intact and the release is limited to noble gases. The release is assumed to occur outside confinement. The noble gas ST is assumed to totally release to the environment over 12 hours.

Interfacing systems are isolated from the PCS by use of valves, orifices, and small diameter pipes. Therefore, the behavior of the plant as a result of the failure of an interfacing system is very similar to a small LOCA that is described in Scenario ATRPDS-3. The bulk of the core damage is due to the result of a core uncovering, which occurs at low pressure because of the pipe break. Some early damage may occur, but sufficient water around the core retains the soluble fission products in the liquid.

Even though the break is outside the confinement, the ST should be relatively small. The noble gases are released, but only a very small fraction of the other highly-volatile fission products released from the core are released to the environment through the break. Long runs of uninsulated piping between the core and the break enhance condensation of steam and deposition of released fission product aerosols, reducing the amount of fission products in vapor form. Water-filled loop seals between the core and the break also prevent the soluble fission products from reaching the break in vapor form. With the soluble fission products retained in the water, the ST is confined to the noble gases released from the fuel.

CAM and RAM alarms outside ATR confinement indicate that a radioactive material release is occurring.

D-1.6.3.12 ATR-670-7.SAE.4

D-1.6.3.12.1 Event Description

Radiological release from a low-pressure boil-off event,

D-1.6.3.12.2 Event Recognition Factors and Related Information

AS INDICATED BY

reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow),

AND

flow **not** restored within 30 minutes after reactor shutdown,

AND

emergency firewater injection system pressure less than 58 psig,

AS INDICATED BY

PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure),

AND

primary coolant system vented to atmosphere,

AND

vessel coolant level below the emergency firewater injection system injection level and decreasing,

AS INDICATED BY

LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

release ongoing **less than or equal to 4 hours.**

RELATED INFORMATION:

This event is a precursor to emergency action level ATR-670-7.GE.2.

D-1.6.3.12.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **9,000 m (29,530 ft or 5.6 mi)** in the downwind direction.

Consider evacuating the Critical Infrastructure Test Range Complex and Power Burst Facility if they are in the downwind direction.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate.

Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I.

Consider authorizing potassium iodide for essential emergency workers.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.12.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.3.12.5 Basis. The pressurizing system fails with no low-pressure coolant injection due to system or operational failures. Decay heat slowly boils ATR vessel liquid with core uncovery at 18 hours and fuel melt at 19 hours (Scenario ATRPDS-1). Based on SAR-153, it is assumed that confinement leakage is 51% per day. It is assumed that the ST is totally released to the environment over 2 days. For calculated radiological doses, it is assumed that this ST released is that of the first 4 hours (4 hr/48 hr) since the first 4 hours of an event are when time-urgent PAs are necessary.

Scenario ATRPDS-1 was initiated by a failure of the pressurizing system. ATR scram on low inlet pressure occurred near 94 seconds. The combination of leakage from the PCS (a limiting nominal leakage of 30 gpm at normal operating pressure was assumed) and the cooldown following scram depressurized the PCS. The PCPs were tripped shortly after 20 minutes, at a suction pressure of 50 psig, to protect the pumps as the NPSH limit was approached. At that time, the emergency coolant flow failed. The ATR vessel vent valves were opened 60 seconds after the PCPs were tripped, but no low-pressure coolant injection low-pressure demineralized water or emergency firewater injection occurred because of system or operational failures.

With no source of water available, the decay heat slowly boiled the liquid in the ATR vessel. The resulting core uncovering, which began at 18.2 hours, occurred near atmospheric pressure because the vent valves were open. The fuel melting temperature was reached by 18.9 hours. A complete core melt ensued, with a complete release of all the highly-volatile fission products (i.e., xenon, krypton, iodine, cesium, tellurium) in the core.

The release path for the fission products was from the ATR vessel through vent valves, through the discharge piping to a header connecting to the warm waste tank, then back up through floor drains connected to the header into the second basement. While about 13% of the tellurium was retained in the ATR vessel by chemiabsorption, nearly all of the remaining fission products were released through the vent valves. About 15% of soluble fission products (i.e., iodine, cesium, tellurium) were retained in the piping downstream of the vent valves. The dominant deposition mechanisms were turbulent deposition and deposition associated with effects of bends in the piping.

As steam was released into the confinement, it slightly pressurized the building, causing some leakage to the environment. However, as that steam condensed, the pressure decreased below ambient, and some inflow to the confinement occurred. These competing effects resulted in a confinement pressure that fluctuated very little from atmospheric pressure. Since there was no large driving force, most of the released fission products were condensed or deposited in the confinement.

D-1.6.3.13 ATR-670-8.SAE.1

D-1.6.3.13.1 Event Description

Radiological release from an anticipated transient without scram event,

D-1.6.3.13.2 Event Recognition Factors and Related Information

AS INDICATED BY

- Power greater than N_L
- Period — positive,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

release ongoing **less than or equal to 4 hours.**

RELATED INFORMATION:

This event is a precursor to emergency action level ATR-670-8.GE.1.

D-1.6.3.13.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **9,000 m (29,530 ft or 5.6 mi)** in the downwind direction.

Consider evacuating the Critical Infrastructure Test Range Complex if it is in the downwind direction.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate.

Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I.

Consider authorizing potassium iodide for essential emergency workers.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.13.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.3.13.5 Basis. ATR is not responding correctly to reactor control signals. In essence, ATR is out of control. Some type of transient scrams ATR, but ATR does not completely shut down (Scenario ATRPDS-4). This is a very unstable reactor condition. This event may or may not result in fuel melt. If the accompanying event is a LOCA, the behavior is similar to that described in Scenario ATRPDS-3 or -3M. If the transient does not involve a pipe break, but results in core uncovering, behavior is similar to that described in Scenario ATRPDS-1 or -2. For the purposes of this EHA consequence assessment, a Scenario ATRPDS-1 event was assumed.

If the accompanying event is a LOCA, the behavior is similar to that described in Scenario ATRPDS-3 or -3M. Any fission products released early in the transient, while there is still a large volume of water above and below the core, are scrubbed by and carried in that liquid, minimizing the eventual release to the environment. Even if the core damage results from a long-term core uncovering, most of the soluble fission products are deposited and retained in the confinement.

If the transient does not involve a pipe break or failure, pressure control is through the degassing tank or the ATR vessel relief or vent valves. If core damage and fission product release occur early in the transient while there is still a large amount of water surrounding the core, significant fission product scrubbing occurs. Noble gases are released either through the relief valves, vent valves, or degassing tank. This delays the release to the environment. Soluble fission products are carried by and retained in liquid, again minimizing the ST to the environment. Some of the fission products may leak out of the PCS (normal leakage from the system), but this flow should be minor compared to the release through the relief valves and degassing tank. Again, the water retains the soluble fission products. If the event results in an eventual core uncovering, the fission product behavior is similar to that described in Scenario ATRPDS-1 or -2, depending on whether the PCS is depressurized. The timing of the melt may be accelerated because there was no scram, but the phenomenology of the fission product behavior should be the same.

D-1.6.3.14 ATR-670-8.SAE.2

D-1.6.3.14.1 Event Description

Radiological event from a large reactivity insertion event,

D-1.6.3.14.2 Event Recognition Factors and Related Information

AS INDICATED BY

control room observation of a reactor scram with indication of a power increase,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

release ongoing **less than or equal to 4 hours**.

RELATED INFORMATION:

This event is a precursor to emergency action level ATR-670-8.GE.2. It is similar to emergency action level ATR-670-7.SAE.2.

D-1.6.3.14.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex and Central Facilities Area personnel and field workers within **8,000 m (26,250 ft or 5 mi)** of the Advanced Test Reactor Complex and field workers within **10,850 m (35,599 ft or 6.75 mi)** in the downwind direction.

Consider evacuating the Critical Infrastructure Test Range Complex if it is in the downwind direction.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate.

Recommend that the Advanced Mixed Waste Treatment Project and Radioactive Waste Management Complex consider evacuating if they are in the downwind direction.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on United States Highway 20/26 and close Experimental Breeder Reactor-I.

Consider authorizing potassium iodide for essential emergency workers.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.3.14.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.3.14.5 Basis. A perched fuel element drops into the core after criticality is achieved, causing a very large and rapid reactivity insertion (Scenario ATRPDS-5). This may cause ATR vessel movement and rupture of piping systems leading to loss of coolant. This event may result in fuel melt. If the accompanying event is a LOCA, the behavior is similar to that described in Scenario ATRPDS-3.

The only identified mechanism for obtaining a very large and rapid reactivity insertion in the ATR core is a perched fuel element dropping into the core from a large height above its seated position after criticality has been achieved. This type of an event requires multiple failures of independent operational controls. Analyses performed for probable perched fuel element drops determined that those events resulted in no fuel damage. Therefore, only a grossly perched fuel element, a difficult condition to achieve which should be readily recognized before ATR startup, that fails to drop into the core when subjected to high downward forces from the startup of the PCS and only drops in after core criticality is achieved (the reactivity anomaly should be detected during the approach to criticality), can provide sufficient rapid reactivity input to exceed the accident level and consequences of the bounding direct fuel damage RIA for experiment loop-initiated RIAs (Scenario ATRPDS-5M).

Following possible vessel movement and rupture of piping and other connecting systems, the vessel can be expected to empty of coolant and the core could boil off. If the emergency firewater injection paths to the vessel are broken, then no emergency coolant could be provided to prevent uncovering the core. A complete core meltdown in the absence of water could occur. Therefore, the consequences of the accident are similar to those for the LOCA without emergency coolant or described in Scenario ATRPDS-3.

D-1.6.4 General Emergency — Emergency Action Levels

D-1.6.4.1 ATR-670-3.GE.1

D-1.6.4.1.1 Event Description

Radiological release from a canal drain event,

D-1.6.4.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

canal water level alarm,

OR

direct observation of the canal water level approaching or below the 13-ft level,

AND

firewater system or other makeup water sources **not** stabilizing canal water level,

AND

multiple continuous air monitor and remote area monitor alarms,

AND

radiological control technician confirmation of the release of airborne radioactive material beta-gamma emitters outside TRA-670.

D-1.6.4.1.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.4.1.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco.

If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe.

If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton.

If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.4.1.5 Basis. Canal drain, loss of coolant water due to breach of canal wall or floor from dropping a shield cask, falling overhead crane, or seismic event. Scenario ATRPDS-7 is associated with draining the liquid from the fuel storage canal. There are two major concerns with this event: (1) the canal contains a large number of irradiated fuel assemblies so the potential ST could be very large, and (2) the canal is outside confinement so that any fission product release from the fuel has a more direct and rapid path to the environment.

The ST for this event assumed complete melting of the two most recently discharged cores, each of which operated for 60 days at 250 MW, with decay times of 12 hours and 70 days, respectively.[15] This is conservative with respect to typical core power levels and cycle length. Previously discharged fuel would have experienced an even greater cooling time and fission product decay, hence, the release is dominated by the most recently discharged core.

After fuel melting begins, it takes about 3 hours to release all of the highly-volatile fission products. The iodine is in the form of CsI. The remaining cesium is transported as CsOH, Cs_2Te, or in elemental form. The tellurium forms Cs_2TeO_4, Cs_2Te, CdTe, or remains in elemental form. These fission products exist as condensed species and are transported on aerosols or dissolved in water.[16]

D-1.6.4.2 ATR-670-3.GE.2

D-1.6.4.2.1 Event Description

Radiological release from a fuel melt caused by an experimental reactivity insertion event,

D-1.6.4.2.2 Event Recognition Factors and Related Information

AS INDICATED BY

loop instrumentation indicating low-low pressurizer level and low-low pressure alarm,

AND

fission break monitor system alarms,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

OR

radiological control technician confirmation of the release by radiation survey,

AND

release ongoing **longer than 4 hours**.

D-1.6.4.2.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider sheltering Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider sheltering Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.4.2.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

Advise state and county representatives that offsite protective actions may be necessary depending on wind direction and speed.

NOTE: *Sheltering or evacuation should be considered for those individuals working on property immediately adjacent to the Idaho National Laboratory Site northwest boundary **(12,000 m or 7.5 mi)**.*

If wind is from **95 to 170** degrees, make protective action recommendation to Butte County to shelter or evacuate anyone working near the northwest Idaho National Laboratory Site boundary between the junction of State Highway 33 and United States Highway 20/26 and Mile Marker 12 on State Highway 33.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.4.2.5 Basis. The sudden rupture of an experiment loop pressure tube causes a sudden rupture of the insulating gas envelope tube in the core region of an ATR flux trap. This in turn causes an extremely rapid, positive ramp insertion of reactivity in the ATR core due to voiding in a high positive reactivity worth experiment flux trap (Scenario ATRPDS-5M). This event of a rupture in an experiment loop may result in fuel melt with vapor explosion. It is expected that the release and potential doses to receptors are significantly less than for other core damage events since the PCS remains intact and less than 2% of the core is expected to melt. It is assumed that confinement leakage is 51% per day. It is assumed that the ST is totally released to the environment over 2 days. For calculated radiological doses, it is assumed that this ST released is that of the first 4 hours (4 hr/48 hr).

This event is an RIA initiated by experiment loop ruptures that are severe enough to result in fuel melting before the ATR scram can be effective in terminating the reactivity transient. This event category is represented by a single event referred to as the FTVA. The FTVA is a bounding, extremely rapid, positive ramp insertion of reactivity in the ATR core due to voiding in a high positive reactivity worth experiment flux trap. The initiating event for the FTVA is a sudden rupture of an experiment loop pressure tube, which then also causes a sudden rupture of the insulating gas envelope tube in the core region of an ATR core flux trap. The double rupture results in an expulsion of high-temperature, high-pressure loop water into the relatively low-pressure (ATR vessel pressure) flux trap annulus between the gas envelope tube and the flux trap baffle, which very rapidly voids the flux trap annulus. Since ATR flux traps, including the annulus, have a positive void reactivity coefficient, ATR experiences a very rapid positive reactivity insertion and power transient. This rapid positive reactivity insertion and the resulting power transient can be large if the event occurred in a flux trap position with a high potential reactivity worth in the flux trap annulus. The large negative reactivity feedback from the core fuel terminates the transient as the fuel and core coolant heat up in response to the power rise, and a high-power scram shuts down ATR.

Although this accident results in a significantly damaged core and internals, the vessel remains intact and water filled. Therefore, the water and vessel barriers to the release of fission products remain. Some early fission product release may occur from lifting of the relief valves during the pressure transient. Following the very short pressure transient, fission product release from the vessel and PCS is by primary coolant leakage. This leakage may increase if relief valves fail to reseat, or if seals are damaged by the severe pressure transient. But, the PCS and support systems remain operable and therefore, the coolant inventory can be controlled and most of the soluble fission products retained and cooled within the vessel. Nonsoluable gaseous fission products, primarily noble gases, may initially be released through normal operation of the degassing tank and vented through the ATR stack. But, the stack breach radiation monitors detect the release and isolate the stack and confinement very shortly after the accident. Therefore, only a small fraction of the fission products available for release are released from the vessel into the confinement.

D-1.6.4.3 ATR-670-3.GE.3

D-1.6.4.3.1 Event Description

Experiment loop loss-of-coolant accident with a radiological release from a greater than 200-kW and less than or equal to 1-MW fueled experiment,

D-1.6.4.3.2 Event Recognition Factors and Related Information

AS INDICATED BY

loop instrumentation indicating low-low pressurizer level and low-low pressure alarm,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

OR

radiological control technician confirmation of the release by radiation survey,

AND

release ongoing **longer than 4 hours**.

D-1.6.4.3.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider sheltering Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider sheltering Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.4.3.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

Advise state and county representatives that offsite protective actions may be necessary depending on wind direction and speed.

NOTE: *Sheltering or evacuation should be considered for those individuals working on property immediately adjacent to the Idaho National Laboratory Site northwest or southern boundaries* ***(17,000 m or 10.6 mi)****.*

If wind is from **95 to 170** degrees, make protective action recommendation to Butte County to shelter or evacuate anyone working near the northwest Idaho National Laboratory Site boundary between the junction of State Highway 33 and United States Highway 20/26 and Mile Marker 12 on State Highway 33.

If wind is from **335 to 25** degrees, make protective action recommendation to Bingham County to shelter or evacuate anyone working near the southern Idaho National Laboratory Site boundary from 4 to 15 mi west of Atomic City.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.4.3.5 Basis. Scenario ATREXPLR-1, as described in SAR-153, Chapter 15, Subsection 15.7, is an experiment loop LOCA that assumes complete meltdown of a 1-MW test (current technical specification limit the experiment size to 200 kW so an exemption is required to run a 1-MW test). The experiment loop cooling system is isolated from the PCS so an experiment loop LOCA does not affect the PCS.

The pressurized water experiment loop facilities described in SAR-153, Chapter 10, are systems whose failure results in the uncontrolled release of radioactivity to the environment under transient conditions. The testing program includes irradiation of fueled tests in the experiment loops. A loss of flow or LOCA in the experiment loop could result in melting of the test. Both a 1/2-in. experiment loop LOCA and an experiment loop flow coastdown are considered Condition 2 occurrences. The Condition 2 loop flow coastdown is terminated by a loop scram that should prevent fuel damage. A larger than 1/2-in. experiment loop LOCA and loss of flow without a loop scram are considered Condition 4 events. If test damage occurred, only the LOCA events allow fission product release into the loop cubicle. Any fission products released during a loss–of-flow event remains contained in the loop piping and the release to TRA-670 and environment is enveloped by those from the LOCA.

D-1.6.4.4 ATR-670-7.GE.1

D-1.6.4.4.1 Event Description

Loss-of-coolant accident bounds both small break and large break loss-of-coolant accidents with failure of the emergency firewater injection system that could result in core damage and fuel melting with a consequent radiological release,

D-1.6.4.4.2 Event Recognition Factors and Related Information

AS INDICATED BY

primary coolant leak rate greater than 300 gpm,

AS INDICATED BY

flow balance [pressurizing flow FIC-1-8B or -8A to PCV-1-1 flow (PCS F• DEGAS)],

AND

emergency firewater injection system pressure less than 58 psig,

AS INDICATED BY

PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure),

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

vessel coolant level below the emergency firewater injection system injection level and decreasing,

AS INDICATED BY

LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing,

AND

release ongoing **longer than 4 hours.**

D-1.6.4.4.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.4.4.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco.

If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe.

If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton.

If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.4.4.5 Basis. This EAL encompasses Scenarios ATRPDS-3 and -3M. Failure of the PCS (small diameter inlet pipe break outside radiographic boundary) and failure of low-pressure injection. This results in a low-pressure core uncovering at 1.5 hr and release to the environment at 10 hr (Scenario ATRPDS-3). Based on SAR-153, it is assumed that confinement leakage is 51% per day. It is assumed that the ST is totally released to the environment over 2 days. For calculated radiological doses, it is assumed that this ST released is that of the first 4 hours (4 hr/48 hr).

A 3-in. diameter inlet break was investigated. This initiator showed some potential for early (within 10 minutes) core damage in the higher-powered regions of the core. When this damage occurred, the upper plenum was still nearly full of liquid. This liquid retains most of the volatile fission products, while still in the PCS and after release to the confinement. The transient continued to an LPBO of the remaining liquid, which resulted in core uncovering near 30 minutes. The remaining fuel in the core is expected to melt by that time.

The layout of the PCS piping is such that there are several loop seals in the inlet and outlet piping that retain water during a LOCA. For example, any breaks beyond the pipe space, after the piping rises to the pipe tunnel, trap liquid in the piping in the pipe space. If one or more of these water-filled loop seals are located between the core and the break, this trapped water is effective in reducing the fission product release to the environment in two ways. The water plug could prevent vapor flow to the break, allowing the fission products more time to deposit in the piping. If vapor does flow through the liquid to get to the break, significant scrubbing of the soluble fission products occurs. Since the release to the environment consists of those fission products in vapor form in the confinement, any fission products retained in the PCS or carried in the water is not part of the ST. Breaks from the 36-in. outlet pipe through the 36-in. inlet pipe have scrubbed releases, which result in lower releases to the environment.

This event is characterized by failure of the PCS due to a large diameter pipe break. This resulted in core flow stagnation and fuel melting. Based on SAR-153, it is assumed that confinement leakage is 99% per day since there is emergency firewater injection. It is assumed that the ST is totally released to the environment over 24 hours. For calculated radiological doses, it is assumed that this ST released is that of the first 4 hours (4 hr/24 hr).

Scenario ATRPDS-3M is associated with a "direct damage" LOCA in which the initiating event alone is sufficient to cause fuel melting. The fuel melting is due to flow stagnation in the core. With the low pressures and high heat fluxes that exist early in a large break LOCA, critical heat flux occurs, resulting in rapidly increasing fuel temperatures. With the large volumetric expansion associated with boiling at these subatmospheric pressures, and no large driving force to encourage flow through the core, liquid is not easily reintroduced to the flow channels in the core.

A 24-in. outlet break, initiated from full power of 250 MW, was analyzed in detail. The blowdown resulted in subatmospheric pressures in the vessel within 1 second because the top of the vessel was about 55 ft above the break, which is at atmospheric pressure. The PCPs and emergency PCPs were tripped on suction pressures of 2 psia to simulate flow degradation below the NPSH requirements of the pump. The low-pressure trip for the emergency firewater injection was disabled in order to maximize the core damage; early calculations indicated that the firewater quenched the core as the heat of fusion was added to the fuel, thus preventing significant fission product release. Fuel melting began near 85 seconds, and relocation of molten material to the flow distribution tank occurred between 100 and 110 seconds. The liquid level in the upper plenum dropped below the EFIS setpoint near 174 seconds, and firewater injection to the vessel began at about 179 seconds. The injection flow was able to recover the liquid level in the ATR vessel. Although the PCS thermal-hydraulic calculation was stopped at 200 seconds, the transient continues with intermittent operation of the EFIS and injection occurring as the level drops below 87 ft, then stopping once the level exceeds 90 ft. This continues until the confinement fills sufficiently to a level above 87 ft that can be maintained in the ATR vessel.

D-1.6.4.5 ATR-670-7.GE.2

D-1.6.4.5.1 Event Description

Radiological release from a low-pressure boil-off event,

D-1.6.4.5.2 Event Recognition Factors and Related Information

AS INDICATED BY

reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow),

AND

flow **not** restored within 30 minutes after reactor shutdown,

AND

emergency firewater injection system pressure less than 58 psig,

AS INDICATED BY

PT-10-6 and PS-1-110 low-pressure alarm (firewater pressure),

AND

primary coolant system vented to atmosphere,

AND

vessel coolant level below the emergency firewater injection system injection level and decreasing,

AS INDICATED BY

LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

release ongoing **longer than 4 hours.**

D-1.6.4.5.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.4.5.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco.

If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe.

If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton.

If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.4.5.5 Basis. Scenario ATRPDS-1 is characterized by failure of the pressurizing system with no low-pressure coolant injection due to system or operational failures. Decay heat slowly boils ATR vessel liquid with core uncovery at 18 hours and fuel melt at 19 hours. Based on SAR-153, it is assumed that confinement leakage is 51% per day. It is assumed that the ST is totally released to the environment over 2 days. For calculated radiological doses, it is assumed that this ST released is that of the first 4 hours (4 hr/48 hr), since the first 4 hours of an event are when time-urgent PAs are necessary.

This event was initiated by a failure of the pressurizing system. ATR scram on low inlet pressure occurred near 94 seconds. The combination of leakage from the PCS (a limiting nominal leakage of 30 gpm at normal operating pressure was assumed) and the cooldown following scram depressurized the PCS. The PCPs were tripped shortly after 20 minutes, at a suction pressure of 50 psig, in order to protect the pumps as the NPSH limit was approached. At that time, the emergency coolant flow failed. The ATR vessel vent valves were opened 60 seconds after the PCPs were tripped, but no low-pressure coolant injection low-pressure demineralized water or emergency firewater injection occurred because of system or operational failures.

With no source of water available, the decay heat slowly boiled the liquid in the ATR vessel. The resulting core uncovering, which began at 18.2 hours, occurred near atmospheric pressure because the vent valves were open. The fuel melting temperature was reached by 18.9 hours. A complete core melt ensued, with a complete release of all the highly-volatile fission products (i.e., xenon, krypton, iodine, cesium, tellurium) in the core.

The release path for the fission products was from the ATR vessel through vent valves, through the discharge piping to a header connecting to the warm waste tank, then back up through floor drains connected to the header into the second basement. While about 13% of the tellurium was retained in the ATR vessel by chemiabsorption, nearly all of the remaining fission products were released through the vent valves. About 15% of soluble fission products (i.e., iodine, cesium, tellurium) were retained in the piping downstream of the vent valves. The dominant deposition mechanisms were turbulent deposition and deposition associated with effects of bends in the piping.

As steam was released into the confinement, it slightly pressurized the building, causing some leakage to the environment. However, as that steam condensed, the pressure decreased below ambient, and some inflow to the confinement occurred. These competing effects resulted in a confinement pressure that fluctuated very little from atmospheric pressure. Since there was no large driving force, most of the released fission products were condensed or deposited in the confinement.

D-1.6.4.6 ATR-670-7.GE.3

D-1.6.4.6.1 Event Description

Radiological release from a high-pressure boil-off event,

D-1.6.4.6.2 Event Recognition Factors and Related Information

AS INDICATED BY

reading of less than 3,200 gpm on FRS-1-5A (total primary coolant outlet flow),

AND

flow **not** restored within 30 minutes after reactor shutdown,

OR

following initial 30 minutes of flow after reactor shutdown, flow lost and **not** restored,

AND

primary coolant system **not** vented to atmosphere,

AND

vessel coolant level below the emergency firewater injection system injection level and decreasing,

AS INDICATED BY

LI-514X (plant protection system vessel level gauge) reading 0 in. (92 ft) and LI-535X (emergency firewater injection system vessel level gauge) reading less than 6 ft 6 in. and decreasing,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

release ongoing **longer than 4 hours.**

D-1.6.4.6.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.4.6.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco.

If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe.

If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton.

If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.4.6.5 Basis. Failure of the emergency PCPs and failure to depressurize the PCS. Decay heat slowly boils ATR vessel liquid with core uncovery and fuel melt at 29 hours (Scenario ATRPDS-2). Based on SAR-153, it is assumed that confinement leakage is 51% per day. It is assumed that the ST is totally released to the environment over 2 days. For calculated radiological doses, it is assumed that this ST released is that of the first 4 hours (4 hr/48 hr).

This transient was initiated by a loss of commercial power, which also scrammed ATR. The emergency PCPs were lost after 4 hours when the battery backup power was expended. There was no intentional depressurization through the ATR vessel vent valves. Natural circulation was established within the ATR vessel with liquid flowing up through the core and down through the reflector; some flow also went through the siphon breaker lines from the top of the ATR vessel to the outlet lines.

The water in the ATR vessel slowly heated up, reaching saturation in the core after 8.5 hours. As the coolant was heated, the pressure increased. The ATR vessel relief valves first lifted at 14.4 hours, then cycled to control the pressure through the rest of the transient. Continued boiling in the core, with flow through the relief valves, resulted in a decreasing liquid level in the vessel. The level reached the top of the core near 28.7 hours. Fuel melting temperatures were reached by 29 hours. With no coolant makeup available, the entire core melted with a resultant release of all the highly-volatile fission products to the PCS.

As for the LPBO described above, there was little energy release to the confinement to drive fission products to the environment. The small amounts of steam released from the PCS condensed in the confinement, resulting in the confinement pressure remaining near ambient. Nearly all of the fission products released to the environment are the result of normal air exchange between the confinement and the atmosphere. With no impetus for driving fission products from the confinement, most of the soluble fission products were deposited on the walls and floors in the confinement. Only those fission products existing as vapors at 35 hours are released to the environment.

D-1.6.4.7 ATR-670-8.GE.1

D-1.6.4.7.1 Event Description

Radiological release from an anticipated transient without scram event,

D-1.6.4.7.2 Event Recognition Factors and Related Information

AS INDICATED BY

- Power greater than N_L
- Period — positive,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

release ongoing **longer than 4 hours**.

D-1.6.4.7.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.4.7.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco.

If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe.

If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton.

If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.4.7.5 Basis. ATR is not responding correctly to reactor control signals. In essence, ATR is out of control. Some type of transient scrams ATR, but ATR does not completely shut down (Scenario ATRPDS-4). This is a very unstable reactor condition. This event may or may not result in fuel melt. If the accompanying event is a LOCA, the behavior is similar to that described in Scenario ATRPDS-3 or -3M. If the transient does not involve a pipe break, but results in core uncovering, behavior is similar to that described in Scenario ATRPDS-1 or -2. For the purposes of this EHA consequence assessment, a Scenario ATRPDS-1 event was assumed.

If the accompanying event is a LOCA, the behavior is similar to that described in Scenario ATRPDS-3 or -3M. Any fission products released early in the transient, while there is still a large volume of water above and below the core, are scrubbed by and carried in that liquid, minimizing the eventual release to the environment. Even if the core damage results from a long-term core uncovering, most of the soluble fission products are deposited and retained in the confinement.

If the transient does not involve a pipe break or failure, pressure control is through the degassing tank or the ATR vessel relief or vent valves. If core damage and fission product release occur early in the transient while there is still a large amount of water surrounding the core, significant fission product scrubbing occurs. Noble gases are released either through the relief valves, vent valves, or degassing tank. This delays the release to the environment. Soluble fission products are carried by and retained in liquid, again minimizing the ST to the environment. Some of the fission products may leak out of the PCS (normal leakage from the system), but this flow should be minor compared to the release through the relief valves and degassing tank. Again, the water retains the soluble fission products. If the event results in an eventual core uncovering, the fission product behavior is similar to that described in Scenario ATRPDS-1 or -2, depending on whether the PCS is depressurized. The timing of the melt may be accelerated because there was no scram, but the phenomenology of the fission product behavior should be the same.

D-1.6.4.8 ATR-670-8.GE.2

D-1.6.4.8.1 Event Description

Radiological event from a large reactivity insertion event,

D-1.6.4.8.2 Event Recognition Factors and Related Information

AS INDICATED BY

control room observation of a reactor scram with indication of a power increase,

AND

multiple continuous air monitor and remote area monitor alarms and/or stack monitor alarm,

AND

release ongoing **longer than 4 hours**.

D-1.6.4.8.3 Onsite Protective Actions

Evacuate nonessential Advanced Test Reactor Complex, Central Facilities Area, Critical Infrastructure Test Range Complex, and Power Burst Facility personnel and all field workers and grazing rights personnel within the Idaho National Laboratory Site boundary.

Recommend that the Advanced Mixed Waste Treatment Project; Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; Naval Reactors Facility; and Radioactive Waste Management Complex evacuate.

If wind is from **165 to 255** degrees, consider evacuating Specific Manufacturing Capability/Test Area North personnel.

If wind is from **225 to 315** degrees, consider evacuating Materials and Fuels Complex personnel.

Control nonessential vehicle and personnel access to the evacuated areas.

Control access on all public roadways entering the Idaho National Laboratory Site.

Consider authorizing potassium iodide for essential emergency workers at the Advanced Test Reactor Complex and other downwind locations.

Relocate the emergency control center to the alternate emergency control center at the Willow Creek Building in Idaho Falls.

Verify implementation of AOP-0.1, "Operator Evacuation Procedure," as ensured by the shift supervisor/emergency action manager, as appropriate.

D-1.6.4.8.4 Offsite Protective Action Recommendations

Control access on all public roadways entering the Idaho National Laboratory Site.

If wind is from **45 to 135** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Arco.

If wind is from **135 to 225** degrees, make protective action recommendation to Butte County to shelter and prepare to evacuate Howe.

If wind is from **180 to 270** degrees, make protective action recommendation to Jefferson County to shelter and prepare to evacuate Mud Lake/Terreton.

If wind is from **270 to 360** degrees, make protective action recommendation to Bingham County to shelter and prepare to evacuate Atomic City.

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

D-1.6.4.8.5 Basis. A perched fuel element drops into the core after criticality is achieved, causing a very large and rapid reactivity insertion (Scenario ATRPDS-5). This may cause ATR vessel movement and rupture of piping systems leading to loss of coolant. This event may result in fuel melt. If the accompanying event is a LOCA, the behavior is similar to that described in Scenario ATRPDS-3.

The only identified mechanism for obtaining a very large and rapid reactivity insertion in the ATR core is a perched fuel element dropping into the core from a large height above its seated position after criticality has been achieved. This type of an event requires multiple failures of independent operational controls. Analyses performed for probable perched fuel element drops determined that those events resulted in no fuel damage. Therefore, only a grossly perched fuel element, a difficult condition to achieve which should be readily recognized before ATR startup, that fails to drop into the core when subjected to high downward forces from the startup of the PCS and only drops in after core criticality is achieved (the reactivity anomaly should be detected during the approach to criticality), can provide sufficient rapid reactivity input to exceed the accident level and consequences of the bounding direct fuel damage RIA for experiment loop-initiated RIAs (Scenario ATRPDS-5M).

Following possible vessel movement and rupture of piping and other connecting systems, the vessel can be expected to empty of coolant and the core could boil off. If the emergency firewater injection paths to the vessel are broken, then no emergency coolant could be provided to prevent uncovering the core. A complete core meltdown in the absence of water could occur. Therefore, the consequences of the accident are similar to those for the LOCA without emergency coolant or described in Scenario ATRPDS-3.

D-1.6.5 Emergency Action Levels and Scenario Release Designators Cross Reference

Table D-35 shows the link between the EALs and the scenario release designators used as the basis for the EAL.

Table D-35. Emergency action levels and scenario release designators.

EAL No.	Scenario Release Designator
ATR-670-10.OE.1 (ATRC Facility)	ATRCCRIT
ATR-670-3.A.1 (ATR)	TRA670RR-1
ATR-670-3.A.2 (ATRC Facility)	ATRCCD-1
ATR-670-3.A.3 (ATRC Facility)	ATRCCRIT
ATR-670-7.A.1 (ATR)	ATRPDS-2 (no release)
ATR-670-7.A.2 (ATR)	ATRPDS-3 and -3M (no release)
ATR-670-8.A.1 (ATR)	ATRPDS-4 and -5 (no release)
ATR-670-11.A.1 (ATR)	ATRPDS-7 (no release)
ATR-670-3.SAE.1 (ATR)	ATREXPLR-2
ATR-670-3.SAE.2 (ATR)	ATRFCB-1
ATR-670-3.SAE.3 (ATR)	ATRCIC-1
ATR-670-3.SAE.4 (ATR)	ATRDLEXP-1
ATR-670-3.SAE.5 (ATR)	ATREXPLR-1
ATR-670-3.SAE.6 (ATR)	ATRPDS-5M

Table D-35. (continued).

EAL No.	Scenario Release Designator
ATR-670-3.SAE.7 (ATRC Facility)	ATRCFM-1
ATR-670-3.SAE.8 (ATR)	ATRCaskDp-1
ATR-670-7.SAE.1 (ATR)	ATRPDS-2
ATR-670-7.SAE.2 (ATR)	ATRPDS-3 and -3M
ATR-670-7.SAE.3 (ATR)	ATRPDS-6
ATR-670-7.SAE.4 (ATR)	ATRPDS-1
ATR-670-8.SAE.1 (ATR)	ATRPDS-4
ATR-670-8.SAE.2 (ATR)	ATRPDS-5
ATR-670-3.GE.1 (ATR)	ATRPDS-7
ATR-670-3.GE.2 (ATR)	ATRPDS-5M
ATR-670-3.GE.3 (ATR)	ATREXPLR-1
ATR-670-7.GE.1 (ATR)	ATRPDS-3 and -3M
ATR-670-7.GE.2 (ATR)	ATRPDS-1
ATR-670-7.GE.3 (ATR)	ATRPDS-2
ATR-670-8.GE.1 (ATR)	ATRPDS-4
ATR-670-8.GE.2 (ATR)	ATRPDS-5

D-1.6.6 Emergency Planning Zone

The maximum PA distance considered in this EHA for TRA-670 was greater than 16,000 m (16 km). Nine of the event scenarios evaluated were classified as GEs. Eight of the GE scenarios had PA distances greater than 16 km at the end of the ATR confinement release duration of 24 or 48 hours. One GE scenario did not exceed 16 km, but the PA distance was greater than the distance to the nearest INL Site boundary. The existing 16-km radial distance is appropriate for the ATR Complex emergency planning zone. Therefore, no change to the ATR Complex emergency planning zone size is recommended based on this EHA.

D-2. REFERENCES

1. DOE O 151.1C, "Comprehensive Emergency Management System," United States Department of Energy, November 2, 2005.
2. DOE G 151.1-2, "Technical Planning Basis," United States Department of Energy, July 11, 2007.
3. SAR-153, "Upgraded Final Safety Analysis Report for the Advanced Test Reactor," Rev. 12, February 5, 2008.
4. Radiological Safety Analysis Computer Program (RSAC-6), Version 6.2, INEEL, 2002.

5. DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities," December 1994.

6. SAR-192, "Safety Analysis Report for the Advanced Test Reactor Critical (ATRC) Facility," Rev. 8, March 27, 2008.

7. Engineering Design File TRA-ATR-1729, "ORIGEN2 Calculations of Radionuclide Inventory of an ATR Fuel Element Irradiated at Low Power Levels," October 30, 2001.

8. TRA-ATRC-1748, "Radiological Analysis of an ATRC Fuel Handling Accident Supporting a Upgraded Final Safety Analysis Report (UFSAR) Revision," Rev. 0, July 16, 2001.

9. Engineering Design File TRA-ATRC-1755, "Radiological Analysis of a Severe Accident Supporting an ATRC UFSAR Revision," Rev. 0, July 16, 2001.

10. EPA 550-B-99-009, "Risk Management Program Guidance for Offsite Consequence Analysis," April 1999.

11. TRA-NMIS-1276, "12 Rad Boundary Calculation for a NMIS Criticality," Rev. 0, November 10, 1997.

12. AOP-0.1, "Operator Evacuation Procedure," Rev. 6, March 23, 2005.

13. FRP-S.1, "Uncontrolled Power Generation/ATWS," Rev. 1, May 10, 2000.

14. TRA-ATR-1300, "Exposures from a Dropped Loop Experiment," Rev. 1, March 26, 1998.

15. Thatcher, T. A., et al., "Update to the Advanced Test Reactor Probabilistic Risk Assessment (Level 1, 2 and 3, Including Shutdown Operations)," Volume 1, EGG-PRP-11229, May 01, 1994.

16. Adams, J. P., and M. L. Carboneau, "Fission Product Behavior During ATR Transient," CAN2/EQC4, PG-T-92-127, EG&G Idaho, Inc., August 1992.

REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	01/13/2010	All	New document.

CONTENTS

FIGURES

TABLES

E-1. TRA-671, COOLING WATER PUMPHOUSE

E-1.1 Assessment Overview

This appendix documents the emergency management hazards assessment (EHA) for TRA-671, Cooling Water Pumphouse, to provide the technical basis for facility emergency planning efforts, as required by DOE O 151.1C[1] and supported by DOE G 151.1-2.[2]

E-1.2 Facility and Process Descriptions

TRA-671 is inside the fenced area of the Advanced Test Reactor (ATR) Complex, north of TRA-670, ATR Building. It is bordered on the south by Bass Avenue and is in the northwest quadrant of the ATR Complex. Figure E-1 shows the location of TRA-671 at the ATR Complex. Figure E-2 shows the floor plan of TRA-671.

TRA-671 is one of the facilities necessary for safe operation of ATR.[3] It contains the secondary cooling pumps to maintain proper cooling of ATR, water treatment equipment, and chemical storage tanks. The water treatment chemicals include sulfuric acid, corrosion inhibitors, and biocides.

TRA-671 is a 3,586-ft^2 single-story building. The original 2,500-ft^2 building has concrete block walls and composition roofing on steel deck and beams. In 1995, a 1,080-ft^2 addition was built on the northeast side of the building to store the water treatment chemicals. The addition was a single-story pre-engineered insulated metal building with a steel panel roof. The addition flooring is steel grating above a recessed sump, which acts as a secondary containment basin.

TRA-671 contains an 8,000-gal pressurized carbon-steel sulfuric acid storage tank, two 1,500-gal high-density poly tanks for corrosion inhibitors, two 900-gal polymer biocide/sulfuric acid storage tanks, and an underground 234-gal waste water tank.

TRA-671 consists of two large open rooms and a storage closet, which are analyzed as a single entity in this EHA.

E-1.3 Identification of Hazards

Table E-1 lists the nonradiological hazardous material that is stored, used, or produced in TRA-671 that is retained for further analyses based on the screening criteria presented in the main document. No radiological material was identified in TRA-671.

E-1.4 Hazardous Material Characterization and Analyses

The hazardous material listed in Table E-1 is addressed below by location.

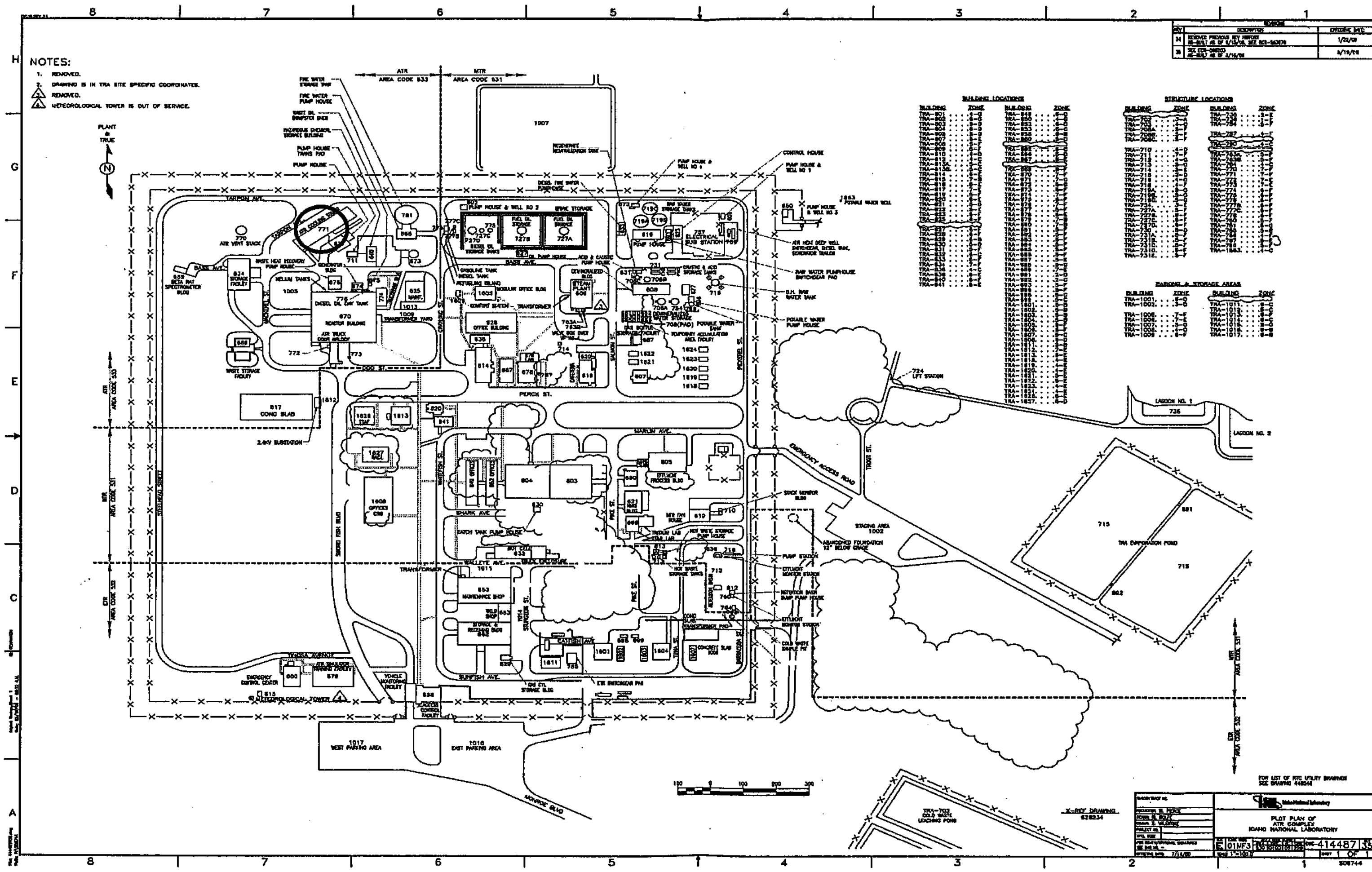


Figure E-1. Map showing location of TRA-671, Advanced Test Reactor Cooling Tower Pumphouse, at the Advanced Test Reactor Complex.

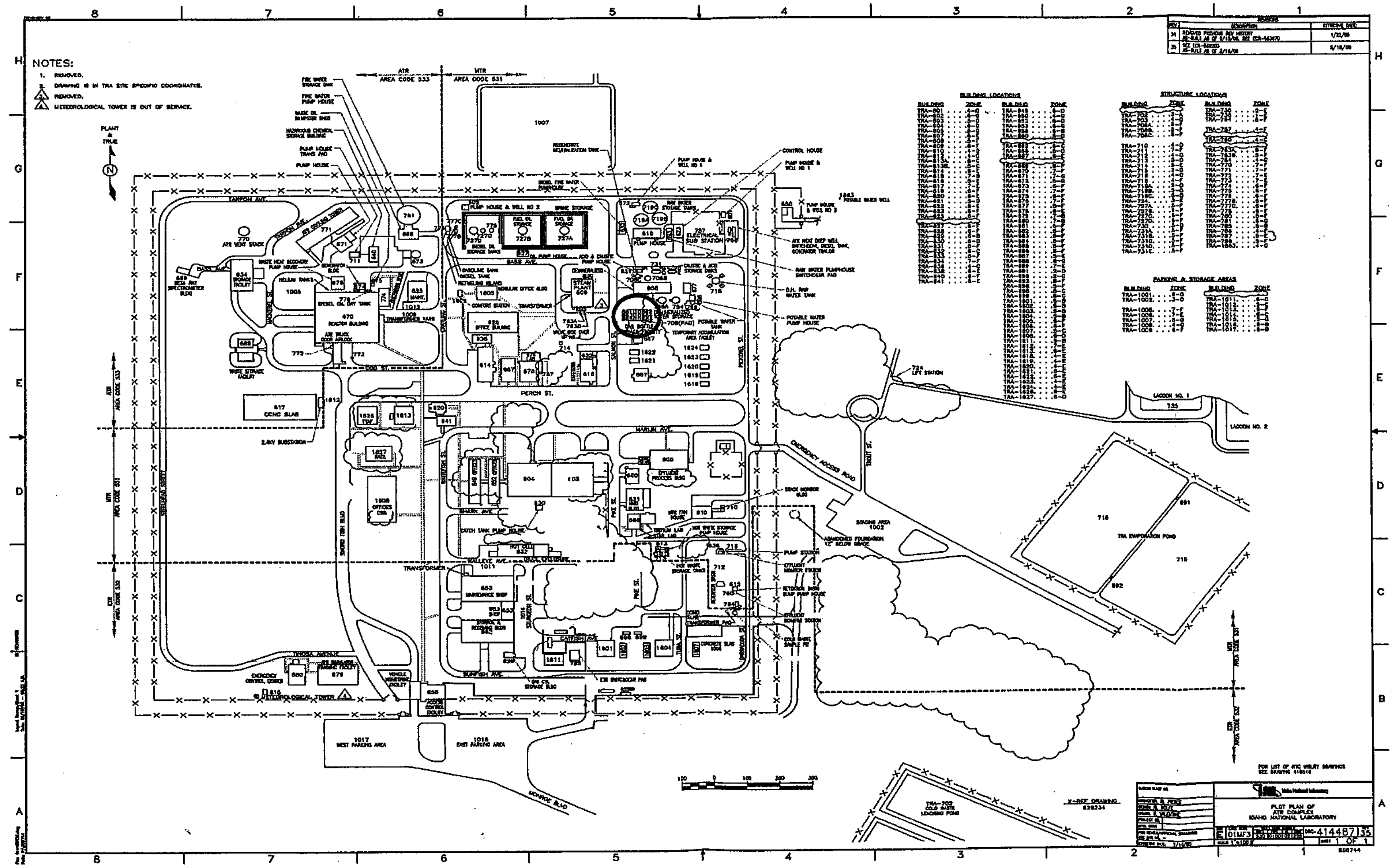


Figure F-1. Map showing location of TRA-780, 90-Day Storage Area, at the Advanced Test Reactor Complex.

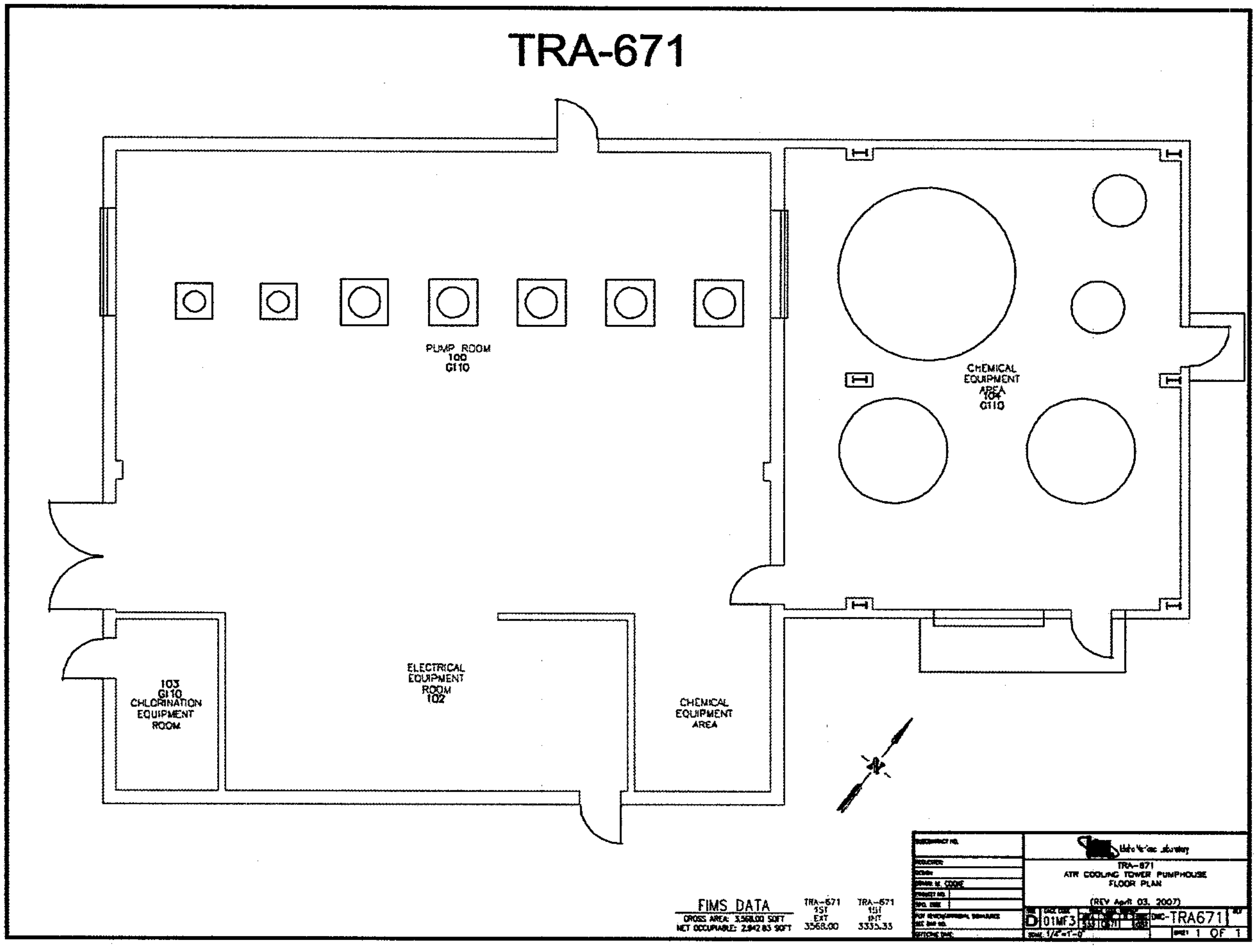


Figure E-2. Floor plan for TRA-671, Advanced Test Reactor Cooling Tower Pumphouse.

Table E-1. Nonradiological hazardous material requiring further analyses.

Location	Material	Maximum Quantity (lb)	Screening Threshold (lb)	Notes
TRA-671 (All)	Sulfuric acid	119,123	40	Only a concern in the external aircraft or vehicle crash/fire scenario, which could cause an airborne release

E-1.4.1 TRA-671 (All)

E-1.4.1.1 Nonradiological Hazardous Material — Sulfuric Acid

E-1.4.1.1.1 Properties. The properties for sulfuric acid stored in the 8,000-gal tank in the chemical equipment area of TRA-671 are listed in Table E-2.

Table E-2. Nonradiological properties for sulfuric acid stored in the 8,000-gal tank in TRA-671.

Chemical Abstract Service registry No.	7664-93-9.	
Idaho National Laboratory material safety data sheet No.	File Name 045073.	
Protective action criteria	Protective action criteria basis: (60-minute Acute Exposure Guideline Level values, Rev. 24). • PAC1 = 0.2 mg/m^3 • PAC2 = 8.7 mg/m^3 • PAC3 = 160 mg/m^3.	
Physical form	Material form	Liquid (93% sulfuric acid and 7% water).
	Molecular weight	98.08 g/mole.
	Specific gravity	1.84 at 60°F.
	Boiling point	530°F (276.7°C).
	Freezing point	−31 to 51°F (−35 to 10.6°C).
	Vapor pressure	1 mm/Hg at 294°F. 100% volatiles at 644°F.
Special firefighting concerns	Avoid any contact with acid. Wear full protective rubber clothing, gloves, and boots and self-contained breathing apparatus. Not flammable, but highly reactive, and can cause ignition by contact with combustible material. Reacts violently with water and organics. Use dry chemicals or carbon dioxide to extinguish small fires.	
Health concerns	Ingestion may cause severe injury or death. Inhalation is not a normal route of entry. Eye contact may be slight to severe with irritation, burns, or corneal necrosis (loss of sight). Skin contact may cause irritation or burns on skin. Prolonged contact may cause severe, deep burns to tissue; very corrosive effects.	

E-1.4.1.1.2 Conditions of Storage and Use. The sulfuric acid is used for water treatment pH control. The 93% sulfuric acid is stored in an 8,000-gal pressurized carbon-steel tank. The maximum inventory of sulfuric acid reported for the TRA-671 inside tank for the last 3 years is 119,123 lb. At present (December 1, 2008), there is a tank inventory of 69,662 lb.

The primary barrier is the 8,000-gal storage tank.

Secondary barriers are the building structure and recessed sump with a 765-ft^2 (71.1-m^2) surface area.

Engineered controls at TRA-671 are fire sprinklers for fire suppression and a recessed sump to contain any spill from the 8,000-gal storage tank preventing a direct release to the environment.

There are no administrative controls, however, the tank size (8,000 gal) provides a physical limitation on the maximum quantity of material that may be present.

TRA-671 is fully sprinklered and provided with an automatic/manual fire alarm system with manual pull stations.

E-1.4.1.1.3 Barrier and Failure Mode Analyses. The barrier and failure mode analyses presented below for sulfuric acid stored in the 8,000-gal tank in the chemical equipment area of TRA-671 are summarized in Table E-3.

Table E-3. Nonradiological failure modes and barriers for sulfuric acid stored in the 8,000-gal tank in the chemical equipment area of TRA-671.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA671NR-1	Sulfuric acid (Table E-1)	8,000-gal carbon-steel sulfuric acid storage tank	Tank breach or valve or piping failure releases 93% sulfuric acid to containment sump	Explosion, impact (airplane or vehicle crash into building), seismic event, or tank degradation (77% acid put in tank by mistake) causes breach of storage tank	100% due to tank or piping failure	Containment sump remains intact; airborne release is by evaporation only
TRA671NR-2	Sulfuric acid (Table E-1)	8,000-gal carbon-steel sulfuric acid storage tank	Tank breach or valve or piping failure releases 93% sulfuric acid to containment sump; sump is also breached releasing acid to environment	Explosion, impact (airplane or vehicle crash into building), seismic event, or tank degradation (77% acid put in tank by mistake) causes breach of storage tank	100% due to tank or piping failure	Essentially none; containment sump is breached, which releases the acid directly to environment; airborne release is by evaporation only
TRA671NR-3	Sulfuric acid (Table E-1)	8,000-gal carbon-steel sulfuric acid storage tank	Tank breach or valve or piping failure releases 93% sulfuric acid to containment sump; fire increases evaporation rate from sump	Impact (airplane or vehicle crash into building) and fire causes breach of storage tank	100% due to tank or piping failure and building breach	Containment sump remains intact; airborne release is by evaporation only, but evaporation rate is increased due high temperatures caused by fire

E-1.4.1.1.4 Scenario Bases and Analytical Assumptions

1. Scenario TRA671NR-1, Sulfuric Acid Spill to Containment Sump

Detailed Scenario Description	Explosion, vehicle impact (truck or aircraft), seismic event, or tank degradation causes a breach of the 8,000-gal carbon-steel sulfuric acid storage tank releasing the contents to the recessed containment sump.
Material-at-Risk	The maximum sulfuric acid quantity in storage during the last 3 years is 119,123 lb.
Release Characteristics	The release is due to evaporation from the fixed surface area (71.1 m^2) of the containment sump. This release is conservative because the containment sump is inside TRA-671 and may not be influenced by wind speed. There will be some plate-out on the building surfaces.
Airborne Release Rate	Based on the sump surface area, Emergency Prediction Information Code (EPIcode) calculated an airborne release rate (ARR) of 7.39E−6 g/s or 5.87E−5 lb/hr for Stability Class F with a wind speed of 1.04 m/s and temperature of 90°F (32.2°C). EPIcode also calculated an ARR of 4.09E−6 g/s or 3.25E−5 lb/hr for Stability Class D with a wind speed of 2.46 m/s and temperature of 70°F (21.1°C).
Leak Path Factor	5.5E−1.
Source Term	Source term (ST) = ARR × T × LPF = 5.87E−5 lb/hr × 1 hr × 5.5E−1 = 3.23E−5 lb (Class F) Where T = release duration time (1 hour) LPF = leak path factor (EPA 550-B-99-009)[4] ST = 3.25E−5 lb/hr × 1 hr × 5.5E−1 = 1.79E−5 lb (Class D).
Modeling Software and Inputs	Two calculations were required using EPIcode, Version 7.0. The liquid spill model with a sump surface area of 71.1 m^2, spill temperature of 32.2°C, spill release of 119,123 lb, and wind speed of 1.04 m/s for Stability Class F, and a spill temperature of 21.1°C and wind speed of 2.46 m/s for Stability Class D were used to determine the ARR.

The calculated ST is then entered into the term release model for consequence assessment. In order to get the correct χ/Q values, such as the same χ/Q values used in the spill model, complex geometry and an area source must be selected. The vertical dimension is set to zero meters and horizontal dimension is set to two times the radius of the spill area (71.1 m^2), which is equivalent to 9.51 m, as determined from: $\pi r^2 = 71.1\ m^2$. The release duration is 60 minutes and the sampling time is 15 minutes.

2. Scenario TRA671NR-2, Sulfuric Acid Spill to Environment

Detailed Scenario Description

Explosion, vehicle impact (truck or aircraft), or seismic event causes a breach of the 8,000-gal carbon-steel sulfuric acid storage tank and containment sump releasing the contents to the environment.

Material-at-Risk

The maximum sulfuric acid quantity in storage during the last 3 years is 119,123 lb.

Release Characteristics

The sulfuric acid spill to the environment outside of TRA-671 was allowed to default to a 1-cm depth, which resulted in a spill surface area of 2,950 m^2. Evaporation rate STs were developed for ambient temperature scenarios.

Airborne Release Rate

Based on the spill surface area, EPIcode calculated an ARR of 3.07E−4 g/s or 2.44E−3 lb/hr for Stability Class F with a wind speed of 1.04 m/s and temperature of 90°F (32.2°C). EPIcode also calculated an ARR of 1.70E−4 g/s or 1.35E−3 lb/hr for Stability Class D with a wind speed of 2.46 m/s and temperature of 70°F (21.1°C).

Leak Path Factor

1.0 (the spill area is assumed to be open to the environment).

Source Term

ST = ARR × T × LPF = 2.43E−3 lb/hr × 1 hr × 1.0 = 2.43E−3 lb (Class F)

Where

T = release duration time (1 hour)

ST = 1.35E−3 lb/hr × 1 hr × 1.0 = 1.35E−3 lb (Class D).

Modeling Software and Inputs

EPIcode, Version 7.0, using the liquid spill model with a spill area of 2,950 m^2, spill temperature of 32.2°C, spill release of 119,123 lb, and wind speed of 1.04 m/s for Stability Class F, and a spill temperature of 21.1°C and wind speed of 2.46 m/s for Stability Class D. The release duration is 60 minutes and the sampling time is 15 minutes.

3. Scenario TRA671NR-3, Sulfuric Acid Spill to Containment Sump with Fire

Detailed Scenario Description

Impact (airplane or vehicle crash into building) and fire causes breach of the 8,000-gal carbon-steel sulfuric acid storage tank and enhanced evaporation from the containment sump.

Material-at-Risk

The maximum sulfuric acid quantity in storage during the last 3 years is 119,123 lb.

Release Characteristics

The release is due to evaporation from the fixed surface area (71.1 m^2) of the containment sump. The fire increases the surface temperature to the boiling point, 530°F (276.7°C). The building walls and roof are assumed to breach as a result of the vehicle impact. Based on a fuel volume of 250 gal (full tank on delivery truck) spill into the containment sump, the fire burn time is 2.6 minutes, as calculated by EPIcode based on the fuel volume and the fire radius. The plume is elevated due to thermal buoyancy.

Airborne Release Rate

Based on the sump surface area, EPIcode calculated an ARR of 1.97E+2 g/s or 1.56E+3 lb/hr for Stability Class F with a wind speed of 1.04 m/s and spill temperature of 530°F (276.7°C). EPIcode also calculated an ARR of 6.37E+2 g/s or 5.06E+3 lb/hr for Stability Class D with a wind speed of 2.46 m/s and spill temperature of 530°F (276.7°C). For Stability Class A with a wind speed of 2.46 m/s and a spill temperature of 530°F (276.7°C), the EPIcode calculated ARR is 7.04E+2 g/s or 5.59E+3 lb/hr.

Leak Path Factor

1.0 (the building walls and roof are assumed to be breached by the vehicle impact).

Source Term

ST = ARR × T × LPF = 1.56E+3 lb/hr × 11 hr × 1 = 1.56E+3 lb (Class F).

Where

T = release duration time (1 hour)

ST = 5.06E+3 lb/hr × 1 hr × 1 = 5.06E+3 lb (Class D)

ST = 5.59E+3 lb/hr × 1 hr × 1 = 5.59E+3 lb (Class A).

Modeling Software and Inputs:

Two calculations were required using EPIcode, Version 7.0.

The liquid spill model with a sump surface area of 71.1 m^2, spill temperature of 276.7°C, spill release of 119,123 lb, and wind speed of 1.04 m/s for Stability Class F, and a spill temperature of 276.7°C and wind speed of 2.46 m/s for Stability Classes D and A were used to determine the ARR.

The calculated ST is then entered into EPIcode, Version 7.0, using the fire release model and fuel burn duration data. The fuel volume is assumed to be 250 gal, which defaults to a 5.5-m release radius; however, the physical size of the sump limits the radius to 4.8 m. The values for physical height of fire (0), heat of combustion (10,730 cal/g), and air temperature (20°C) are used. Clicking on the fire duration will bring up a default of 2.6 minutes. The release duration and sampling time are both set at 15 minutes, which conservatively releases the 1-hour evaporative ST in a 15-minute time period.

E-1.5 Evaluation Results

E-1.5.1 Nonradiological Hazardous Material Release Results

Nonradiological hazardous material release results for 95% worst-case and 50% typical weather conditions as described in the main document are presented in Tables E-4 and E-5. In addition, scenarios analyzed as elevated releases were also modeled using Stability Class A with an average (50%) wind speed of 2.46 m/s. The results of these analyses are presented in Table E-6.

Table E-4. Nonradiological release scenario calculation results for 95% worst-case meteorology.

Scenario Release Designator	Short Description	Downwind Airborne Concentrations for 95% Worst-Case Meteorology					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (50 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA671NR-1 (Sulfuric acid)	Breach of 8,000-gal carbon-steel sulfuric acid storage tank, valves, or piping with containment sump intact	1.10E−5 (mg/m^3)	1.80E−4 (mg/m^3)	1.30E−4 (mg/m^3)	1.10E−6 (mg/m^3)	2.30E−7 (mg/m^3)	Not applicable (N/A)	Not exceeded	None (this is an operational upset)
TRA671NR-2 (Sulfuric acid)	Breach of 8,000-gal carbon-steel sulfuric acid storage tank, valves, or piping with containment sump breached	1.90E−4 (mg/m^3)	4.40E−3 (mg/m^3)	2.50E−3 (mg/m^3)	7.50E−5 (mg/m^3)	1.70E−5 (mg/m^3)	N/A	Not exceeded	None (this is an operational upset)

Table E-4. (continued).

Scenario Release Designator	Short Description	Downwind Airborne Concentrations for 95% Worst-Case Meteorology					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (50 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA671NR-3 (Sulfuric acid)	Breach of 8,000-gal carbon-steel sulfuric acid storage tank, valves, or piping with containment sump intact and fire in sump area	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	1.70E−14 (mg/m^3)	5.60E−9 (mg/m^3)	N/A	Not exceeded	Unclassified operational emergency (UOE) (due to fire and damage to structure)

a. INTEC/ICDF = Idaho Nuclear Technology and Engineering Center/Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility.

Table E-5. Nonradiological release scenario calculation results for 50% typical meteorology.

Scenario Release Designator	Short Description	Downwind Airborne Concentrations for 50% Typical Meteorology					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (50 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA671NR-1 (Sulfuric acid)	Breach of 8,000-gal carbon-steel sulfuric acid storage tank, valves, or piping with containment sump intact	2.80E−5 (mg/m^3)	5.70E−6 (mg/m^3)	1.60E−5 (mg/m^3)	2.20E−8 (mg/m^3)	2.60E−9 (mg/m^3)	N/A	Not exceeded	None (this is an operational upset)
TRA671NR-2 (Sulfuric acid)	Breach of 8,000-gal carbon-steel sulfuric acid storage tank, valves, or piping with containment sump breached	5.80E−4 (mg/m^3)	1.90E−4 (mg/m^3)	4.00E−4 (mg/m^3)	1.60E−6 (mg/m^3)	2.70E−7 (mg/m^3)	N/A	Not exceeded	None (this is an operational upset)
TRA671NR-3 (Sulfuric acid)	Breach of 8,000-gal carbon-steel sulfuric acid storage tank, valves, or piping with containment sump intact and fire in sump area	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	2.60E−23 (mg/m^3)	9.70E−10 (mg/m^3)	N/A	Not exceeded	UOE (due to fire and damage to structure)

Table E-6. Nonradiological release scenario calculation results for Stability Class A and average wind speed meteorology.

Scenario Release Designator	Short Description	Downwind Airborne Concentrations for Stability Class A Meteorology					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (50 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA671NR-3 (Sulfuric acid)	Breach of 8,000-gal carbon-steel sulfuric acid storage tank, valves, or piping with containment sump intact and fire in sump area	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	2.40E−6 (mg/m^3)	7.00E−7 (mg/m^3)	N/A	Not exceeded	UOE (due to fire and damage to structure)

E-1.6 Emergency Classes, Protective Actions, and Emergency Action Levels

Based on the analyses documented above, the following emergency action levels (EALs) are identified for TRA-671.

E-1.6.1 Unclassified Operational Emergency — Emergency Action Levels

Unless otherwise noted, the generic UOE EALs are covered by a separate appendix to this EHA.

There are no TRA-671 facility-specific UOE EALs covered in this appendix.

E-1.6.2 Alert — Emergency Action Levels

None.

E-1.6.3 Site Area Emergency — Emergency Action Levels

None.

E-1.6.4 General Emergency — Emergency Action Levels

None.

E-1.6.5 Emergency Action Levels and Scenario Release Designators Cross Reference

None.

E-1.6.6 Emergency Planning Zone

No change to the existing emergency planning zone size is recommended based on this EHA.

E-2. REFERENCES

1. DOE O 151.1C, "Comprehensive Emergency Management System," United States Department of Energy, November 2, 2005.
2. DOE G 151.1-2, "Technical Planning Basis," United States Department of Energy, July 11, 2007.
3. SAR-153, "Executive Summary — Upgraded Final Safety Analysis Report for the Advanced Test Reactor," Rev. 12, December 5, 2007.
4. EPA 550-B99-009, "Risk Management Guidance for Offsite Consequence Analysis," April 1999.

REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	01/13/2010	All	New document.

CONTENTS

FIGURES

TABLES

F-1. TRA-780, 90-DAY STORAGE AREA

F-1.1 Assessment Overview

This appendix documents the emergency management hazards assessment (EHA) for TRA-682, -683, and -684, which are part of TRA-780, 90-Day Storage Area, to provide the technical basis for facility emergency planning efforts, as required by DOE O 151.1C[1] and supported by DOE G 151.1-2.[2]

F-1.2 Facility and Process Descriptions

TRA-780 is inside the fenced area of the Advanced Test Reactor (ATR) Complex on Salmon Street, south of TRA-608, Demineralizer Building. Figure F-1 shows the location of TRA-780 at the ATR Complex. TRA-780 supports ATR Complex maintenance operations, ATR operations, Radiation Measurements Laboratory operations, and other ATR Complex tenants. TRA-780 is used to temporarily store (90 days or less) hazardous and mixed waste in compliance with Resource Conservation and Recovery Act (RCRA) requirements. TRA-780 is also used to store Toxic Substance Control Act and Comprehensive Environmental Response, Compensation, and Liability Act waste.

TRA-780 is comprised of six modular, single-story steel storage buildings (TRA-681 through -686) and a 44- × 54-ft, 6-in.-thick concrete pad on which the buildings sit. Figure F-2 shows the layout of the buildings. Five of the buildings are bolted at the four corners to the concrete pad. The buildings are prefabricated and built of welded 10- and 12-gauge steel with double opening lockable doors. The interior dimensions are 6 ft × 10 ft 7 in. and the height is 7 ft 3/4 in. Each building has 60 ft^2 of storage area that can store 15 55-gal drums or six 3- × 3- by 1-ft wooden boxes for lead storage or a commensurate amount of waste in suitable containers. Drum, box, and container weights will not exceed the floor loading of 300 lb/ft^2. An 8-in.-deep secondary containment sump of 316 gal is provided for spill control. The sump is equipped with a float level detector that activates an alarm on the outside of the building. Flooring is a pultruded vinyl ester resin fiberglass floor grating. R-14 insulated walls and ceiling are 2-hour fire rated, while the door is 1-1/2-hour fire rated. Two 1-1/2-hour fire-rated fusible link damper air vents are installed. The buildings have an automatic 25-lb dry chemical fire extinguishing system activated by a pull box or interior spray nozzle fusible link. Activating a building's fire suppression system actuates an alarm on the outside of the building. TRA-686 is used to store nonflammables only and has no fire suppression system. Each building is equipped with 208/120-V electric power from TRA-608. Explosion-proof sodium lights, an electric heater, and a ventilation fan are installed in each building. The buildings are electrically grounded.

Within TRA-780, EHS-50[3] identified TRA-682, -683, and -684 as requiring further assessment. Each building is treated as single entity in this EHA.

F-1.3 Identification of Hazards

Table F-1 lists the nonradiological hazardous material that is stored, used, or produced in TRA-682, -683, and -684 that is retained for further analyses based on the screening criteria presented in the main document. No radiological material above the screening threshold was identified in TRA-780.

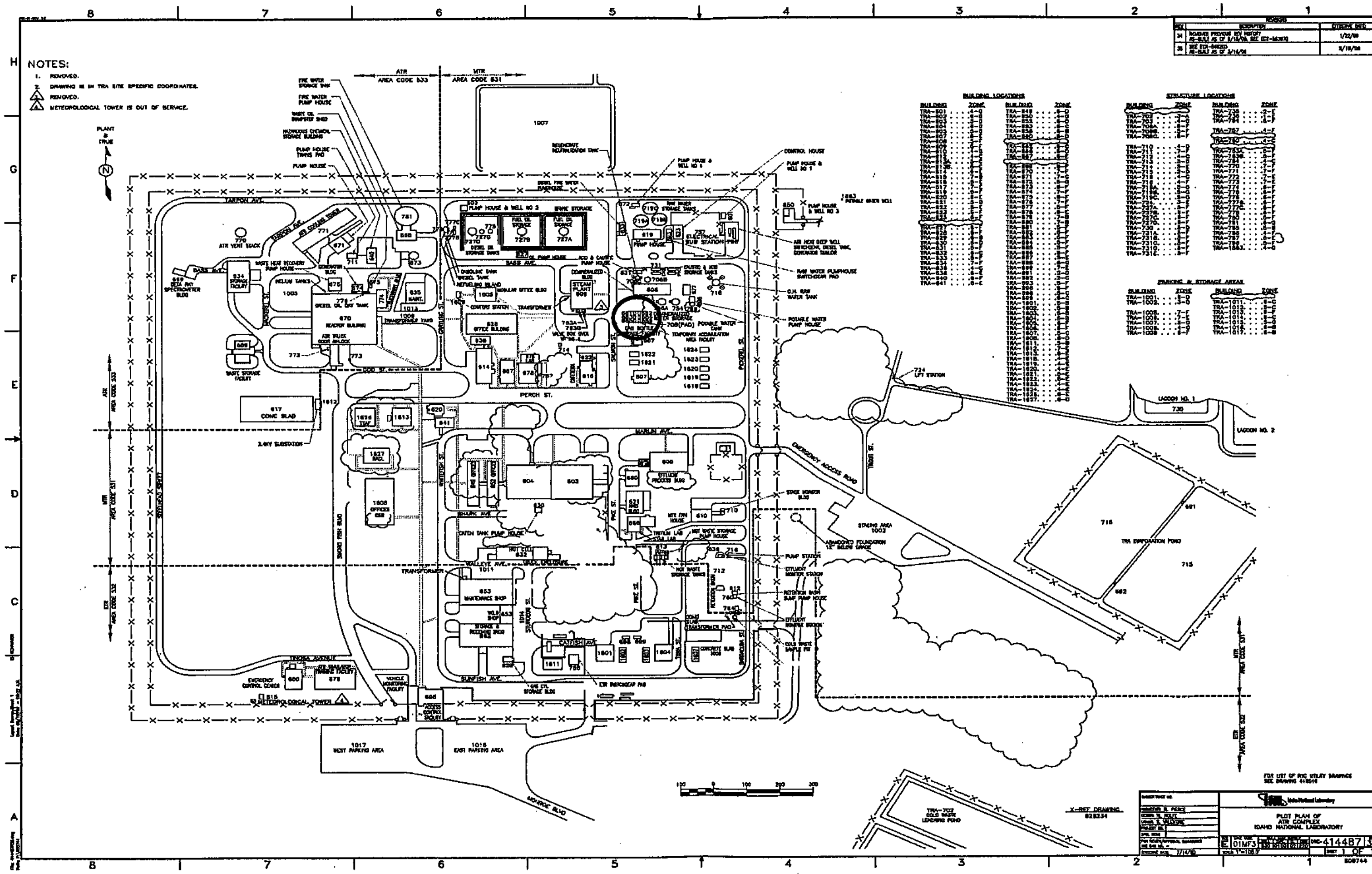


Figure F-1. Map showing location of TRA-780, 90-Day Storage Area, at the Advanced Test Reactor Complex.

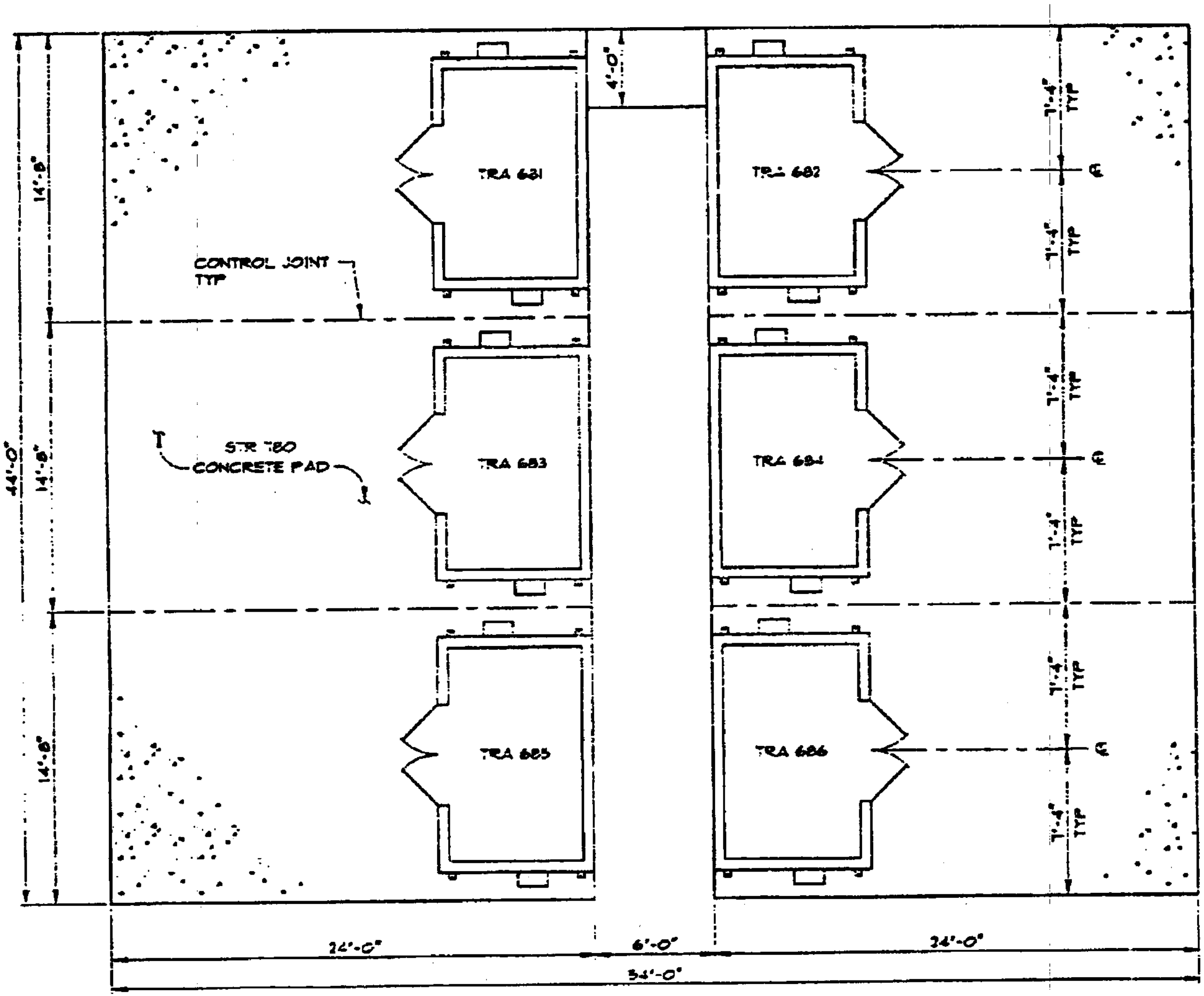


Figure F-2. Layout and construction of TRA-780, 90-Day Storage Area.

Table F-1. Nonradiological hazardous material requiring further analyses.

Location	Material	Maximum Quantity (L)	Screening Threshold (lb)	Notes
TRA-682 (All)	Polychlorinated biphenyl (PCB) contaminated waste	7,500	40	The RCRA regulatory limit is established at 7,500 L. Department of Energy (DOE) requires analysis at the regulatory authorized limit, rather than historical maximum values.
TRA-683 (All)	Silver waste from film developer, hydrofluoric acid, and other hazardous waste (unwanted, unused, or expired chemicals and process waste)	7,500	40	The RCRA regulatory limit is established at 7,500 L. DOE requires analysis at the regulatory authorized limit, rather than historical maximum values.
TRA-684 (All)	Toxicity characteristic leaching procedure/toxic metals (As, Ba, Cd, Cr, Pb, Hg, Ag, and Se)	7,500	40	The RCRA regulatory limit is established at 7,500 L. DOE requires analysis at the regulatory authorized limit, rather than historical maximum values.

F-1.4 Hazardous Material Characterization and Analyses

The hazardous material listed in Table F-1 is addressed below by location. Further examination has revealed that the predominant hazard in TRA-684 is mixed waste containing lead, which is not dispersible and screens out from further analysis. Therefore, TRA-684 is further excluded in this EHA.

F-1.4.1 TRA-682 (All)

F-1.4.1.1 Nonradiological Hazardous Material — Polychlorinated Biphenyls

F-1.4.1.1.1 Properties. The properties for PCBs stored in TRA-682 are listed in Table F-2.

F-1.4.1.1.2 Conditions of Storage and Use. Waste material (old fluorescent lighting fixtures and electrical devices containing PCB capacitors and old microscope and hydraulic oils) contaminated with PCBs is temporarily stored (up to 270 days) at TRA-682 before shipment to an approved waste disposal facility. The waste material is stored in 55-gal drums. Up to 15 55-gal drums are stored in TRA-682.

Table F-2. Nonradiological properties for polychlorinated biphenyls stored in TRA-682.

Chemical Abstract Service registry No.	53469-21-9.
Idaho National Laboratory material safety data sheet No.	None found in Dolphin Comply Plus.
Protective action criteria (PAC)	PAC basis: Temporary emergency exposure limit. • PAC1 = 3 mg/m^3 • PAC2 = 5 mg/m^3 • PAC3 = 5 mg/m^3.
Physical form	Solid or liquid.
Particle size	No data found.
Flammability	Listed as 1. May burn, but does not readily ignite.
Reactivity	Listed as zero.
Density	1.44 at 30°C.
Special firefighting concerns	None.
Health concerns	Causes acne-like skin conditions in adults and neurobehavioral and immunological changes in children. Known to cause cancer in animals. Possible human carcinogen.

Although the RCRA permit limit for the TRA-780 storage pad is 7,500 L for PCBs, the physical limit for TRA-682 is 15 55-gal drums, which is equivalent to 3,000 L. The material-at-risk (MAR) is therefore, based on the physical limit of 3,000 L for TRA-682. The upper limit of the PCB concentration in the contaminated material (light ballasts) is 500 ppm (5.0E–4). Using 3,000 L and a density of 1.44 g/mL, the PCB maximum MAR is calculated as follows:

$$\text{MAR} = 3{,}000 \text{ L} \times 1{,}000 \text{ mL/L} \times 1.44 \text{ g/mL} \times 0.0005 = 2.16\text{E+3 g} = 2.16 \text{ kg or } 4.75 \text{ lb.}$$

The primary barrier is the 55-gal storage drum.

The secondary barrier is the TRA-682 structure. The secondary barrier is limited to drum leaks, as an explosion or fire would breach the structure's walls or roof.

Engineered controls at TRA-682 consist of a dry chemical fire suppression system.

The administrative control for TRA-682 would be equivalent to the RCRA permit limit, which in this case is 2.5 times larger than the physical space limit for TRA-682. Therefore, the physical space limit is controlling.

A fusible link heat sensor activates the dry chemical fire suppression system. The activation of the dry chemical fire suppression system displays at the ATR Complex alarm panel and Fire Alarm Center located at the Central Facilities Area.

F-1.4.1.1.3 Barrier and Failure Mode Analyses. The barrier and failure mode analyses presented below for PCBs in TRA-682 are summarized in Table F-3.

Table F-3. Nonradiological failure modes and barriers for polychlorinated biphenyls in TRA-682.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA682NR-1	PCBs (Table F-1)	55-gal waste drums or smaller waste storage containers	Drum or container breach by puncture, crushing, or degradation	Operator error, natural phenomenon, or chemical reaction	Minimal release since the waste material is primarily solids; use ARF for spill of contaminated waste material	The TRA-682 structure would be a secondary barrier for a drum failure due to degradation; essentially none for operator error or natural phenomenon events
TRA682NR-2	PCBs (Table F-1)	55-gal waste drums or small waste storage containers	Drum or container breach by fire	Operator error, natural phenomenon, or incompatible chemicals stored in the same area and failure of the fire suppression system	Use fire ARF for contaminated waste material	Essentially none; the TRA-682 storage unit would be breached by fire if the fire suppression system failed

F-1.4.1.1.4 Scenario Bases and Analytical Assumptions

1. **Scenario TRA682NR-1, Waste Drum Spill**

Detailed Scenario Description	A waste storage drum is breached due to operator error, natural phenomenon, or degradation of the drum spilling part of the contents on the floor of TRA-682.
Material-at-Risk	15 55-gal drums contain 4.75 lb of PCBs.
Release Characteristics	The release is considered a ground-level release. The spill would behave similarly to radioactive contaminated waste; therefore, an airborne release fraction (ARF) and a respirable fraction from DOE-HDBK-3010-94,[4] p. 5-4, Section 5.1, would be appropriate to use for this chemical release.
Airborne Release Fraction	1.0E−3 (DOE-HDBK-3010-94, p. 5-4, Section 5.1).
Respirable Fraction	1.0E−1 (DOE-HDBK-3010-94, p. 5-4, Section 5.1).
Damage Ratio	1/15 = 7.0E−2.
Leak Path Factor	1.0 (assumes a release directly to the environment).
Source Term	4.75 lb × 1.0E−3 × 1.0E−1 × 7.0E−2 × 1.0 = 3.33E−5 lb.
Modeling Software and Inputs	Emergency Prediction Information Code (EPIcode), Version 7.0,[5] using the term release model, a 15-minute release duration, and a 15-minute sampling time.

2. **Scenario TRA682NR-2, All Waste Drums and Fire**

Detailed Scenario Description	A waste transport vehicle accidentally hits TRA-682, the impact disables the TRA-682 fire suppression system, and leaking fuel starts a fire that spreads to TRA-682 breaching the waste storage drums.
Material-at-Risk	15 55-gal drums contain 4.75 lb of PCBs.
Release Characteristics	The release is considered a thermally-buoyant release. The release would behave similarly to radioactive contaminated waste; therefore, an ARF and a respirable fraction from DOE-HDBK-3010-94, p. 5-1, Section 5.1, would be appropriate to use for this chemical release.
Airborne Release Fraction	5.0E−4 (DOE-HDBK-3010-94, p. 5-1, Section 5.1).
Respirable Fraction	1.0 (DOE-HDBK-3010-94, p. 5-1, Section 5.1).
Damage Ratio	1.0 (assumes that all drums are damaged).
Leak Path Factor	1.0 (assumes a release direct to the environment).
Source Term	4.75 lb × 5.0E−4 × 1.0 × 1.0 × 1.0 = 2.38E−3 lb.

Modeling Software and Inputs	EPIcode, Version 7.0, using the fire release model and fuel burn duration data. The fuel volume is assumed to be 100 gal, which defaults to a 3.5-m release radius, and the burn duration is set at 10 minutes. The default values for physical height of fire (0), heat of combustion (12,000 cal/g), and air temperature (32°C for Stability Class F and 21°C for Stability Classes D and A) are used. Both the release duration and sampling time are set at 15 minutes. The source term (ST) is 2.38E−3 lb.

F-1.4.2 TRA-683 (All)

F-1.4.2.1 Nonradiological Hazardous Material — Hydrofluoric Acid

TRA-683 provides temporary waste storage space for unwanted, unused, or expired chemicals or process waste. Hydrofluoric acid is one of the most toxic materials that may be in temporary storage in TRA-682; therefore, for EHA purposes, hydrofluoric acid is used as a surrogate for all of the other hazardous material that may be stored in TRA-683.

F-1.4.2.1.1 Properties. The properties for hydrofluoric acid stored in TRA-683 are listed in Table F-4.

Table F-4. Nonradiological properties for hydrofluoric acid stored in TRA-683.

Chemical Abstract Service registry No.	7664-39-3.
Idaho National Laboratory material safety data sheet No.	File No. 500656.
PAC	PAC basis: Acute Exposure Guideline Level. • PAC1 = 0.818 mg/m^3 • PAC2 = 19.6 mg/m^3 • PAC3 = 36 mg/m^3.
Physical form	Liquid (surrounded by absorbent material when in temporary accumulation area storage).
Particle size	Not applicable (N/A).
Flammability	Listed as zero.
Reactivity	Instability listed as 1.
Density	1.175 at 15.5°C.
Special firefighting concerns	Use self-contained breathing apparatus in pressure demand mode and full protective gear.
Health concerns	May be fatal if inhaled. May cause severe irritation of the upper respiratory tract with pain, burns, and inflammation. May be fatal if absorbed through the skin.

F-1.4.2.1.2 Conditions of Storage and Use. Waste material (unwanted, unused, or expired chemicals or process waste) is temporarily stored (up to 90 days) at TRA-683 before shipment to an approved waste disposal facility. The waste material is stored in 5-, 10-, or 55-gal drums. Up to 15 55-gal drums could be stored in TRA-683.

Although the RCRA permit limit for the TRA-780 storage pad is 4,000 L for corrosives, the physical limit for TRA-683 is 15 55-gal drums, which is equivalent to 3,000 L. The MAR is therefore, based on the physical limit of 3,000 L for TRA-683. Packaging and transportation require sufficient packing material to completely stabilize the individual waste containers within the storage drums. Using 3,000 L and a 10% hazardous material content, the MAR for hydrofluoric acid is calculated as follows:

$$\text{MAR} = 3{,}000\ \text{L} \times 1{,}000\ \text{mL/L} \times 1.175\ \text{g/mL} \times 0.1 = 3.53\text{E}{+}5\ \text{g} = 3.53\text{E}{+}2\ \text{kg or } 7.76\text{E}{+}2\ \text{lb}.$$

The primary barrier is the individual hazardous material container within the storage drum. Secondary barriers are the storage drum and TRA-683 structure. The secondary barrier is limited to drum leaks, as an explosion or fire would breach the structure's walls or roof. Engineered controls at TRA-683 consist of a dry chemical fire suppression system. The administrative control for TRA-682 would be equivalent to the RCRA permit limit, which in this case is 2.5 times larger than the physical space limit for TRA-682. Therefore, the physical space limit is controlling. A fusible link heat sensor activates the dry chemical fire suppression system. The activation of the dry chemical fire suppression system displays at the ATR Complex alarm panel and Fire Alarm Center located at the Central Facilities Area.

F-1.4.2.1.3 Barrier and Failure Mode Analyses. The barrier and failure mode analyses presented below for hydrofluoric acid in TRA-683 are summarized in Table F-5.

Table F-5. Nonradiological failure modes and barriers for hydrofluoric acid stored in TRA-683.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRA683NR-1	Hydrofluoric acid (Table F-1)	Individual waste chemical container	Drum or container breach by puncture, crushing, or degradation	Operator error, natural phenomenon, or chemical reaction	Minimal liquid release, dependant on the number of containers that break; ground-level release controlled by evaporation	The waste storage drum and TRA-683 structure would be secondary barriers for a drum failure due to degradation; essentially none for operator error or natural phenomenon events
TRA683NR-2	Hydrofluoric acid (Table F-1)	Individual waste chemical container	Drum or container breach by fire	Operator error, natural phenomenon, or incompatible chemicals stored in the same area and failure of the fire suppression system	Assume all individual containers are breached; thermally-buoyant plume	Essentially none; the TRA-683 storage unit would be breached by fire if the fire suppression system failed

F-1.4.2.1.4 Scenario Bases and Analytical Assumptions

1. **Scenario TRA683NR-1, Waste Drum Spill**

Detailed Scenario Description	A waste storage drum is breached due to operator error, natural phenomenon, or degradation of the drum spilling part of the contents on the floor of TRA-683.
Material-at-Risk	15 55-gal drums contain 7.76E+2 lb of hydrofluoric acid. Although the RCRA permit limit for the TRA-780 storage pad is 4,000 L for corrosives, the physical limit for TRA-683 is 15 55-gal drums, which is equivalent to 3,000 L. The MAR is therefore, based on the physical limit of 3,000 L for TRA-683. Packaging and transportation require sufficient packing material to completely stabilize the individual waste containers within the storage drums. Using 3,000 L and a 10% hazardous material content, the MAR for hydrofluoric acid is calculated as follows: MAR = 3,000 L × 1,000 mL/L × 1.175 g/mL × 0.1 = 3.53E+5 g = 3.53E+2 kg or 7.76E+2 lb.
Release Characteristics	The release is considered a ground-level release. The release will be controlled by the default spill area based on the quantity spilled and evaporation rate.
Damage Ratio	1/15 = 7.0E−2.
Leak Path Factor	1.0 (release is direct to the environment).
Source Term	7.76E+2 lb × 7.0E−2 × 1.0 = 5.43E+1 lb.
Modeling Software and Inputs	Areal Locations of Hazardous Atmospheres, Version 5.4.1,[6] using the Puddle Source option. Time and date are set to June 21, 2008, at 1400 hours with a temperature of 90°F for Stability Class F and 70°F for Stability Class D. The spill area is 3.52 m^2 as determined by a 1-cm spill depth from EPIcode, Version 7.0.

2. **Scenario TRA683NR-2, All Waste Drums and Fire**

Detailed Scenario Description	A waste transport vehicle accidentally hits TRA-683, the impact disables the TRA-683 fire suppression system, and leaking fuel starts a fire that spreads to TRA-683 breaching the waste storage drums.

Material-at-Risk	15 55-gal drums contain 7.76E+2 lb of hydrofluoric acid. Although the RCRA permit limit for the TRA-780 storage pad is 4,000 L for corrosives, the physical limit for TRA-683 is 15 55-gal drums, which is equivalent to 3,000 L. The MAR is therefore, based on the physical limit of 3,000 L for TRA-683. Packaging and transportation require sufficient packing material to completely stabilize the individual waste containers within the storage drums. Using 3,000 L and a 10% hazardous material content, the MAR for hydrofluoric acid is calculated as follows: MAR = 3,000 L × 1,000 mL/L × 1.175 g/mL × 0.1 = 3.53E+5 g = 3.53E+2 kg or 7.76E+2 lb.
Release Characteristics	The release is considered thermally buoyant.
Other Release Parameters	ST = 7.76E+2 lb.
Modeling Software and Inputs	EPIcode, Version 7.0, using the fire release model and fuel burn duration data. The fuel volume is assumed to be 100 gal, which defaults to a 3.5-m release radius and the burn duration is set at 10 minutes. The default values for physical height of fire (0), heat of combustion (12,000 cal/g), and air temperature (32°C for Stability Class F and 21°C for Stability Classes D and A) are used. Both the release duration and sampling time are set at 15 minutes. The ST is 7.76E+2 lb.

F-1.5 Evaluation Results

F-1.5.1 Nonradiological Hazardous Material Release Results

Nonradiological hazardous material release results for 95% worst-case and 50% typical weather conditions as described in the main document are presented in Tables F-6 and F-7. In addition, scenarios analyzed as elevated releases were also modeled using Stability Class A with an average (50%) wind speed of 2.46 m/s. The results of these analyses are presented in Table F-8.

F-1.6 Emergency Classes, Protective Actions, and Emergency Action Levels

Based on the analyses documented above, the following emergency action levels (EALs) are identified for TRA-683.

F-1.6.1 Unclassified Operational Emergency — Emergency Action Levels

Unless otherwise noted, the generic UOE EALs are covered by a separate appendix to this EHA.

There are no TRA-780 facility-specific UOE EALs covered in this appendix.

Table F-6. Nonradiological release scenario calculation results for 95% worst-case meteorology.

Scenario Release Designator	Short Description	Downwind Airborne Concentrations for 95% Worst-Case Meteorology					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (115 m)	INTEC/ ICDF[a] Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA682NR-1 (PCBs)	Breach of a single waste drum	1.40E−4 (mg/m^3)	1.20E−3 (mg/m^3)	1.00E−3 (mg/m^3)	5.00E−6 (mg/m^3)	2.00E−7 (mg/m^3)	N/A	Not exceeded	None (this is an operational upset)
TRA682NR-2 (PCBs)	Breach of all waste drums and fire	0.00E+0 (mg/m^3)	0.0E+0 (mg/m^3)	0.00E+0 (mg/m^3)	2.80E−8 (mg/m^3)	5.00E−7 (mg/m^3)	N/A	Not exceeded	Unclassified operational emergency (UOE) (due to fire and damage to structure)
TRA683NR-1 (Hydrofluoric acid)	Breach of a single waste drum	2.62E+2 (ppm)	2.84E+1 (ppm)	2.17E+1 (ppm)	7.07E−3 (ppm)	Not calculated (more than 1 hour downwind)	109	80	Alert
TRA683NR-2 (Hydrofluoric acid)	Breach of all waste drums and fire	0.00E+0 (ppm)	0.00E+0 (ppm)	0.00E+0 (ppm)	1.10E−2 (ppm)	1.10E−3 (ppm)	N/A	Not exceeded	UOE (due to fire and damage to structure)

[a] INTEC/ICDF = Idaho Nuclear Technology and Engineering Center/Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility.

Table F-7. Nonradiological release scenario calculation results for 50% typical meteorology.

Scenario Release Designator	Short Description	Downwind Airborne Concentrations for 50% Typical Meteorology					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (115 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA682NR-1 (PCBs)	Breach of a single waste drum	4.20E−4 (mg/m^3)	5.50E−5 (mg/m^3)	4.90E−5 (mg/m^3)	1.80E−7 (mg/m^3)	4.50E−8 (mg/m^3)	N/A	Not exceeded	None (this is an operational upset)
TRA682NR-2 (PCBs)	Breach of all waste drums and fire	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	6.00E−8 (mg/m^3)	2.80E−7 (mg/m^3)	N/A	Not exceeded	UOE (due to fire and damage to structure)
TRA683NR-1 (Hydrofluoric acid)	Breach of a single waste drum	1.60E+1 (ppm)	1.58E+0 (ppm)	1.21E+0 (ppm)	No significant concentra-tion (ppm)	Not calculated (more than 1 hour downwind)	24	17	None (this is an operational upset)
TRA683NR-2 (Hydrofluoric acid)	Breach of all waste drums and fire	0.00E+0 (ppm)	0.00E+0 (ppm)	0.00E+0 (ppm)	2.60E−2 (ppm)	1.20E−1 (ppm)	N/A	Not exceeded	UOE (due to fire and damage to structure)

Table F-8. Nonradiological release scenario calculation results for Stability Class A and average wind speed meteorology.

Scenario Release Designator	Short Description	Downwind Airborne Concentrations for Stability Class A Meteorology					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		30 m	100 m	Facility Boundary (115 m)	INTEC/ ICDF Boundary (2,865 m)	Site Boundary (10,855 m)			
TRA682NR-2 (PCBs)	Breach of all waste drums and fire	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	0.00E+0 (mg/m^3)	3.20E−7 (mg/m^3)	7.00E−9 (mg/m^3)	N/A	Not exceeded	UOE (due to fire and damage to structure)
TRA683NR-2 (Hydrofluoric acid)	Breach of all waste drums and fire	0.00E+0 (ppm)	1.00E−30 (ppm)	1.50E−30 (ppm)	1.30E−1 (ppm)	3.00E−3 (ppm)	N/A	Not exceeded	UOE (due to fire and damage to structure)

F-1.6.2 Alert — Emergency Action Levels

F-1.6.2.1 ATR-683-4.A.1

F-1.6.2.1.1 Event Description

A waste storage drum containing hydrofluoric acid is breached by puncture, crushing, or drum degradation causing a spill of hydrofluoric acid,

F-1.6.2.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the drum breach,

AND

visual observation of a liquid spill on the floor of TRA-683 or adjacent area.

F-1.6.2.1.3 Onsite Protective Actions

Establish a **109-m (358-ft)** exclusion zone around TRA-683.

Relocate nonessential personnel from the exclusion zone to the conference room in TRA-652.

Control nonessential vehicle and personnel access to the exclusion zone.

F-1.6.2.1.4 Offsite Protective Action Recommendations

None.

F-1.6.2.1.5 Basis. TRA-683 provides temporary waste storage space for unwanted, unused, or expired chemicals or process waste. Hydrofluoric acid is one of the most toxic materials that may be in temporary storage in TRA-682; therefore, for EHA purposes, hydrofluoric acid is used as a surrogate for all of the other hazardous material that may be stored in TRA-683 (Scenario TRA683NR-1).

Although the RCRA permit limit for the TRA-780 storage pad is 4,000 L for corrosives, the physical limit for TRA-683 is 15 55-gal drums, which is equivalent to 3,000 L. The MAR is therefore, based on the physical limit of 3,000 L for TRA-683. Packaging and transportation require sufficient packing material to completely stabilize the individual waste containers within the storage drums. Using 3,000 L and a 10% hazardous material content, the MAR for hydrofluoric acid is calculated as follows:

$$\text{MAR} = 3{,}000 \text{ L} \times 1{,}000 \text{ mL/L} \times 1.175 \text{ g/mL} \times 0.1 = 3.53\text{E}{+}5 \text{ g} = 3.53\text{E}{+}2 \text{ kg or } 7.76\text{E}{+}2 \text{ lb.}$$

F-1.6.3 Site Area Emergency — Emergency Action Levels

None.

F-1.6.4 General Emergency — Emergency Action Levels

None.

F-1.6.5 Emergency Action Levels and Scenario Release Designators Cross Reference

Table F-9 shows the link between the EALs and the scenario release designators used as the basis for the EAL.

Table F-9. Emergency action levels and scenario release designators.

EAL No.	Scenario Release Designator
ATR-683-4.A.1	TRA683NR-1

F-1.6.6 Emergency Planning Zone

The maximum protective action distance considered in this EHA for TRA-683 was 109 m. That distance is less than the existing 16-km ATR Complex emergency planning zone. No change to the existing emergency planning zone size is recommended based on this EHA.

F-2. REFERENCES

1. DOE O 151.1C, “Comprehensive Emergency Management System,” United States Department of Energy, November 2, 2005.
2. DOE G 151.1-2, “Technical Planning Basis,” United States Department of Energy, July 11, 2007.
3. EHS-50, “Emergency Management Hazards Survey for the Advanced Test Reactor Complex,” Rev. 0, October 22, 2008.
4. DOE-HDBK-3010-94, “Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities,” December 1994.
5. “Emergency Prediction Information Code,” Version 7.0, Homann Associates, Inc., 2006.
6. Areal Locations of Hazardous Atmospheres, Version 5.4.1, Environmental Protection Agency, February 2007.

REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	01/13/2010	All	New document.

CONTENTS

TABLES

G-1. ADVANCED TEST REACTOR COMPLEX ONSITE TRANSPORTATION

G-1.1 Assessment Overview

This appendix documents the emergency management hazards assessment (EHA) for Advanced Test Reactor (ATR) Complex onsite transportation to provide the technical basis for facility emergency planning efforts, as required by DOE O 151.1C[1] and supported by DOE G 151.1-2.[2]

G-1.2 Facility and Process Descriptions

Transportation activities can take place on roadways or loading areas anywhere within the ATR Complex fenced boundary. Onsite transportation may involve deliveries by either Idaho National Laboratory Site vehicles or commercial vendor vehicles. The quantities of material transported are in accordance with ATR Complex operational needs and may vary from single items, laboratory quantities (about 40 lb) or large volumes (hundreds to thousands of gallons). Hazardous waste, irradiated test trains, and spent nuclear fuel (SNF) shipments originate at the ATR Complex and are transported onsite to the ATR Complex main gate before exiting to their final destinations.

ATR Complex onsite transportation was analyzed by segmenting radiological and nonradiological hazardous material shipments. The radiological shipments were further segmented by type of radiological material (i.e., low-level radiological waste, irradiated test trains, or SNF). Nonradiological hazardous material shipments defer to the Emergency Response Guidebook[3] (ERG) rather than actual consequence assessment analyses for determining whether emergency action levels (EALs) are needed.

G-1.3 Identification of Hazards

Tables G-1 and G-2 list the radiological and nonradiological hazardous material that is transported within the ATR Complex that is retained for further analyses based on the screening criteria presented in the main document.

Table G-1. Radiological hazardous material requiring further analyses.

Location	Material	Maximum Quantity (Ci)	Screening Threshold (Ci)	Notes
ATR Complex (All)	Radioactive material or activated material	Greater than Hazard Category (HC) 3	HC-3	Quantities of radioactive material that exceed HC-3 threshold require further analyses

Table G-2. Nonradiological hazardous material requiring further analyses.

Location	Material	Maximum Quantity (lb)	Screening Threshold (lb)	Notes
ATR Complex (All)	All Department of Transportation (DOT) hazard classes	Variable (laboratory quantity to bulk shipments)	National Fire Protection Association Health Hazard Rating of 3, 4, or unknown: • Solids 40 lb • Liquids 5 gal • Compressed gases 10 lb	GDE-437[4] provides screening threshold values

G-1.4 Hazardous Material Characterization and Analyses

The hazardous material listed in Tables G-1 and G-2 is addressed below by the type of transportation activity. The decision has been made to include all of the DOT nonradiological hazardous material classes in this appendix, and use the ERG's protective action distances (PADs) as default values for any chemicals that are listed in the ERG. Therefore, no further analysis or source term (ST) development is required for nonradiological hazardous material.

G-1.4.1 Advanced Test Reactor Complex (All)

G-1.4.1.1 Radiological Hazardous Material — Irradiated Test Trains

G-1.4.1.1.1 Properties. The properties for irradiated test trains are listed in Table G-3.

Table G-3. Radiological properties for irradiated test trains.

Physical form	Particulate, volatile, and gaseous radionuclides, as well as gamma radiation.
Particle size	Respirable.
Flammability	Nonflammable.
Reactivity	Not reactive.
Density	Not applicable (N/A).
Special firefighting concerns	None.
Health concerns	Acute health effects caused by exposure to gamma radiation, or potential latent health effects (cancer or genetic effects) caused by inhalation or ingestion of radionuclides.

G-1.4.1.1.2 Conditions of Storage and Use. The ATR is used to irradiate various types of test trains for Department of Energy (DOE) customers. DOE customers supply experimental materials for testing. Upon completion of irradiation, the test is transferred from the ATR core to a shielded transport cask. The irradiated test train and cask may be temporarily stored in TRA-634 or immediately transported to the DOE customer. Irradiated test trains are transported in shipping casks from the ATR Complex to the DOE customer, usually the Naval Reactors Facility. The actual fueled test train composition is classified; however, a 200-kW decay power irradiated ATR fuel element serves as a surrogate for test train ST development. The material-at-risk (MAR) for a 200-kW irradiated test train is presented in Table G-4.

Table G-4. Material-at-risk for irradiated test trains [Engineering Design File (EDF) TRA-ATR-1832[5] and SAR-153, Table 20.2-1[6]].

Radionuclide	Ci
Kr-85m	1.75E+2
Kr-88	1.18E+2
Te-127m	1.10E+1
Te-129m	1.20E+2
I-131	4.56E+3
Te-132	6.93E+3
I-132	6.38E+3
I-133	6.64E+3
Xe-133	1.07E+4
Xe-133m	3.14E+2
I-134	2.12E+2
Cs-134	2.98E+1
I-135	1.91E+3
Cs-137	3.88E+1

The primary barrier is the experimental fuel cladding, if any.

The secondary barrier is the shipping cask unless there is failure of the cask seals. The shipping cask is a barrier to gamma radiation, as well as radionuclide release.

There are no engineered controls that directly apply to transportation casks.

Administrative controls limit the number of loop experiments out of approved storage to a single experiment.

Indicators are limited to visual observation or portable radiation survey instrument readings when a full irradiated test train cask is in transport.

G-1.4.1.1.3 Barrier and Failure Mode Analyses. The barrier and failure mode analyses presented below for irradiated test train transportation are summarized in Table G-5.

Table G-5. Radiological failure modes and barriers for irradiated test train transportation.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRANSRR-1	Test train radioactive fission products	Fuel cladding	Fire damages cask and fuel cladding (melt) when tractor trailer is en route to fuel examination facility	Vehicle accident involving tractor-trailer rig causes large unsuppressed fire involving transport cask containing irradiated test train experiment	100% fuel cladding damage	Essentially none; transport cask provides initial barrier, but is eventually breached due to extreme heat
TRANSRR-2	Test train radioactive fission products	Fuel cladding	Test train cask drop; cask loading equipment fails or vehicle accident occurs	Operator error, cask loading equipment failure, or vehicle accident causes cask drop to roadway	10% fuel cladding damage	Essentially none, as transport cask is assumed breached
TRANSDR-1	Gamma radiation	Transport cask	Shielding failure during transport of irradiated test train	Any cask drop dislodging cask lid; operator error, equipment failure, or vehicle accident	100% gamma radiation exposure rate available	Essentially none

G-1.4.1.1.4 Scenario Bases and Analytical Assumptions

1. **Scenario TRANSRR-1, Test Train Cask Fire**

Detailed Scenario Description Vehicle accident involving tractor-trailer rig causes a large unsuppressed fire involving the transport cask containing an irradiated test train experiment.

Material-at-Risk Fresh fission products (see Table G-6).

Release Characteristics Irradiated test train transport cask is not vented to the atmosphere. A vehicle accident resulting in a fire that engulfs the test train transport cask could cause cladding imperfections to open or cause cladding damage and damage to cask closure seals due to excessive heat resulting in a release of gaseous or volatile fission products. The release is modeled as ground level.

Airborne Release Fraction 1.0E+0 for noble gases, 2.70E−1 for iodine, 1.10E−1 for tellurium, and 2.00E−1 for cesium (DOE-HDBK-3010-94,[7] Section 4.3.1.3.1, p. 4-49).

Respirable Fraction 1.00E+0. (These radionuclides are either gaseous or volatile at fire temperatures and are within the respirable size range.)

Damage Ratio 1.00E+0 (EDF TRA-ATR-1832).

Leak Path Factor 1.00E+0 (EDF TRA-ATR-1832).

Source Term The ST shown in Table G-6 was developed according to the following equation: ST = MAR × DR × LPF × ARF × RF.

Where

DR = damage ratio

LPF = leak path factor

ARF = airborne release fraction

RF = respirable fraction.

Modeling Software and Inputs Radiological Safety Analysis Computer Program (RSAC), Version 6.2.[8] Ground-level release.

Table G-6. Source term for scenario release designator TRANSRR-1.

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Kr-85m	1.75E+2	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.75E+2
Kr-88	1.18E+2	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.18E+2
Te-127m	1.10E+1	1.00E+0	1.00E+0	1.10E−1	1.00E+0	1.20E+0
Te-129m	1.20E+2	1.00E+0	1.00E+0	1.10E−1	1.00E+0	1.32E+1
I-131	4.56E+3	1.00E+0	1.00E+0	2.70E−1	1.00E+0	1.23E+3

Table G-6. (continued).

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Te-132	6.19E+3	1.00E+0	1.00E+0	1.10E−1	1.00E+0	6.81E+2
I-132	6.38E+3	1.00E+0	1.00E+0	2.70E−1	1.00E+0	1.72E+3
I-133	6.64E+3	1.00E+0	1.00E+0	2.70E−1	1.00E+0	1.79E+3
Xe-133	1.07E+4	1.00E+0	1.00E+0	1.00E+0	1.00E+0	1.07E+4
Xe-133m	3.14E+2	1.00E+0	1.00E+0	1.00E+0	1.00E+0	3.14E+2
I-134	2.12E+2	1.00E+0	1.00E+0	2.70E−1	1.00E+0	5.72E+1
Cs-134	2.98E+1	1.00E+0	1.00E+0	2.00E−1	1.00E+0	5.96E+0
I-135	1.91E+3	1.00E+0	1.00E+0	2.70E−1	1.00E+0	5.16E+2
Cs-137	3.88E+1	1.00E+0	1.00E+0	2.00E−1	1.00E+0	7.76E+0

G-1.4.1.1.5 Scenario TRANSRR-2, Test Train Cask Drop

2. **Scenario TRANSRR-2, Test Train Cask Drop**

Detailed Scenario Description	The irradiated test train transport cask topples from the tractor trailer causing some crushing of the fueled irradiated test train and breaking the mast from the transport cask releasing gaseous and volatile radionuclides.
Material-at-Risk	Fresh fission products (see Table G-7).
Release Characteristics	There is a potential for some cladding damage due to crushing within the transport cask. If the transport cask mast breaks during the cask drop, there will be a direct release path to the environment.
Airborne Release Fraction	7.00E−2 for krypton, 7.00E−2 for xenon, 2.00E−3 for iodine, 2.00E−3 for cesium, and 2.00E−3 for tellurium (ANSI/ANS-5.10-1998,[9] Appendix A, Table A-1, p. 15).
Respirable Fraction	1.00E+0 for krypton, 1.00E+0 for xenon, 1.00E+0 for iodine, 7.00E−5 for cesium, and 7.00E−5 for tellurium. (ANSI/ANS-5.10-1998, Appendix A, Table A-1, p. 15; "Crush/Impact Spent Nuclear Fuel.")
Damage Ratio	1.00E−1 (assumes 10% fuel damage).
Leak Path Factor	1.00E+0.
Source Term	The ST shown in Table G-7 was developed according to the following equation: ST = MAR × DR × LPF × ARF × RF.
Modeling Software and Inputs	RSAC, Version 6.2.

Table G-7. Source term for scenario release designator TRANSRR-2.

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Kr-85m	1.75E+2	1.00E−1	1.00E+0	7.00E−2	1.00E+0	1.23E+0
Kr-88	1.18E+2	1.00E−1	1.00E+0	7.00E−2	1.00E+0	8.26E−1
Te-127m	1.10E+1	1.00E−1	1.00E+0	2.00E−3	7.00E−5	1.54E−7
Te-129m	1.20E+2	1.00E−1	1.00E+0	2.00E−3	7.00E−5	1.68E−6
I-131	4.56E+3	1.00E−1	1.00E+0	2.00E−3	1.00E+0	9.12E−1
Te-132	6.19E+3	1.00E−1	1.00E+0	2.00E−3	7.00E−5	8.67E−5
I-132	6.38E+3	1.00E−1	1.00E+0	2.00E−3	1.00E+0	1.28E+0
I-133	6.64E+3	1.00E−1	1.00E+0	2.00E−3	1.00E+0	1.33E+0
Xe-133	1.07E+4	1.00E−1	1.00E+0	7.00E−2	1.00E+0	7.49E+1
Xe-133m	3.14E+2	1.00E−1	1.00E+0	7.00E−2	1.00E+0	2.20E+0
I-134	2.12E+2	1.00E−1	1.00E+0	2.00E−3	1.00E+0	4.24E−2
Cs-134	2.98E+1	1.00E−1	1.00E+0	2.00E−3	7.00E−5	4.17E−7
I-135	1.91E+3	1.00E−1	1.00E+0	2.00E−3	1.00E+0	3.82E−1
Cs-137	3.88E+1	1.00E−1	1.00E+0	2.00E−3	7.00E−5	5.43E−7

3. Scenario TRANSDR-1, Test Train Cask Shielding Failure

Detailed Scenario Description Vehicle accident causes transport cask to topple over breaking the mast, which compromises the cask shielding.

Material-at-Risk Gamma radiation from fresh fission products (see Table G-8).

Release Characteristics Electromagnetic (gamma) radiation.

Airborne Release Fraction N/A.

Respirable Fraction N/A.

Damage Ratio N/A.

Leak Path Factor N/A.

Source Term The exposure rate ST shown in Table G-8 was developed according to the inverse square law equation.

Modeling Software and Inputs Manual calculations using the inverse square law equation.

$$\frac{I_1}{I_2} = \frac{d_2^2}{d_1^2}$$

Where I_1 is the exposure rate near the source, I_2 is the exposure rate at a new distance, d_2 is the new distance, and d_1 is the distance at which I_1 was measured.

Table G-8. Exposure rate source term for scenario release designator TRANSDR-1 (EDF TRA-ATR-1300[10]).

Scenario Release Designator	Distance (m)			TEL Distance (m)[a]	PAC Distance (m)[b]	Administrative Excess Exposure Authorization Distance (m)[c]
	30	100	200			
TRANSDR-1 (irradiated test train 200-kW decay power exposure rates, R/hr)	516 R/hr	46.5 R/hr	11.6 R/hr	68	682	816

a. Threshold for early lethality (TEL) = 100 R/hr for a 1-hour exposure duration.
b. Protective Action Criteria (PAC) = 1 R/hr for a l-hour exposure duration.
c. Administrative limit requiring authorization of excess dose for emergency workers = 0.7 R/hr for a 1-hour exposure duration.

G-1.4.1.2 Radiological Hazardous Material — Advanced Test Reactor Spent Nuclear Fuel

G-1.4.1.2.1 Properties. The properties for SNF are listed in Table G-9.

Table G-9. Radiological properties for spent nuclear fuel.

Physical form	Particulate, volatile, and gaseous radionuclides, as well as gamma radiation.
Particle size	Respirable.
Flammability	Nonflammable.
Reactivity	Not reactive.
Density	N/A.
Special firefighting concerns	None.
Health concerns	Acute health effects caused by exposure to gamma radiation, or potential latent health effects (cancer or genetic effects) caused by inhalation or ingestion of radionuclides.

G-1.4.1.2.2 Conditions of Storage and Use. ATR SNF is stored in the ATR canal for varying time periods before it is transported to an Idaho Nuclear Technology and Engineering Center irradiated fuel storage facility. ATR SNF is transported to the Idaho Nuclear Technology and Engineering Center for storage after sufficient cooling in the ATR canal. Cooling times may vary, with typical decay heat energy range of 30 to 634 W. The earliest time that SNF can safely be removed from the ATR canal and transported in a water-cooled ATR SNF transfer cask is 193 days, which equates to 634 W of decay heat energy. On the other end of the spectrum, the longest time that SNF is stored in the ATR canal is on the order of 1,290 days, which equates to 30 W of decay heat energy. The shorter the cooling time, the greater the amount of fission products that are left in the SNF and likelihood of fuel melting is greater under accident conditions such as fire or loss of transfer cask cooling water. Greater amounts of fission products cause higher gamma radiation exposure rates. The MAR for ATR SNF 30-W decay heat and 634-W decay heat is presented in Tables G-10 and G-11, respectively.

Table G-10. Material-at-risk for 30 W of decay heat energy spent nuclear fuel (EDF TRA-ATR-1902[11] and SAR-153, Table 20.2-1).

Radionuclide	Ci
Sr-90	2.83E+04
Ru-106	4.98E+03
Cs-134	8.02E+03
Cs-137	2.83E+04
Ce-144	3.84E+04
Pm-147	3.22E+04
Pu-238	1.51E+02
Pu-239	5.62E−01
Pu-240	5.61E−01
Pu-241	6.46E+01

Table G-11. Material-at-risk for 634 W of decay heat energy spent nuclear fuel (EDF TRA-ATR-1902 and SAR-153, Table 20.2-1).

Radionuclide	Ci
Sr-89	1.41E+05
Sr-90	3.04E+04
Y-91	2.37E+05
Zr-95	3.11E+05
Nb-95	4.38E+05
Ru-106	3.74E+04
Te-127m	2.09E+03

Table G-11. (continued).

Radionuclide	Ci
Cs-134	2.18E+04
Cs-137	3.04E+04
Ce-141	4.05E+04
Ce-144	5.40E+05
Pm-147	7.08E+04
Pu-238	1.54E+02
Pu-241	7.46E+01

The primary barrier is the fuel cladding.

The secondary barrier, the shipping cask, is degraded since the shipping cask is vented to the atmosphere. The shipping cask is a barrier to gamma radiation.

There are no engineered controls that directly apply to transportation casks.

Administrative controls require that SNF must be cooled to at least 634 W of decay heat energy before shipment.

Indicators are limited to visual observation or portable radiation survey instrument readings when a full SNF cask is in transport.

G-1.4.1.2.3 Barrier and Failure Mode Analyses. The barrier and failure mode analyses presented below for SNF transportation are summarized in Table G-12.

Table G-12. Radiological failure modes and barriers for Advanced Test Reactor spent nuclear fuel transfer cask transportation.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRANSRR-3	Eight ATR fuel elements having 30-W decay heat power (1,280 days out of reactor core)	Fuel cladding	Fire damages ATR shipping cask containing ATR SNF; transport vehicle fire	Unsuppressed fire in tractor-trailer rig transporting ATR shipping cask	100% fuel cladding damage	Essentially none, as the cask is vented; transport cask provides initial barrier, but is eventually breached due to extreme heat
TRANSRR-4	Eight ATR fuel elements having 634-W decay heat power (193 days out of reactor core)	Fuel cladding	Fire damages ATR shipping cask containing ATR SNF; transport vehicle fire	Unsuppressed fire in tractor-trailer rig transporting ATR shipping cask	100% fuel cladding damage	Essentially none, as the cask is vented; transport cask provides initial barrier, but is eventually breached due to extreme heat
TRANSRR-5	Eight ATR fuel elements having 30-W decay heat power (1,280 days out of reactor core)	Fuel cladding	Cask drop containing 30-W decay power or 1,280-day decay ATR fuel; operation incident or vehicle accident	Operator error, equipment failure, or vehicle accident causes cask drop with damage to cask and some crushing of fuel elements	10% fuel cladding damage	Essentially none, as transport cask is assumed breached

Table G-12. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRANSRR-6	Eight ATR fuel elements having 634-W decay heat power (193 days out of reactor core)	Fuel cladding	Cask drop containing 634-W decay power or 193-day decay ATR fuel; operation incident or vehicle accident	Operator error, equipment failure, or vehicle accident causes cask drop with damage to cask and some crushing of fuel elements	10% fuel cladding damage	Essentially none, as transport cask is assumed breached
TRANSDR-2	Gamma radiation	Transport cask	Shielding failure during transport of ATR fuel cask containing 30-W decay power SNF	Any cask drop dislodging cask lid; operator error, equipment failure, or vehicular accident	100% gamma radiation exposure rate available	Essentially none
TRANSDR-3	Gamma radiation	Transport cask	Shielding failure during transport of ATR fuel cask containing 634-W decay power SNF	Any cask drop dislodging cask lid; operator error, equipment failure, or vehicular accident	100% gamma radiation exposure rate available	Essentially none

1. Scenario TRANSRR-3, 30 W of Decay Heat Energy Spent Nuclear Fuel Cask Fire

Detailed Scenario Description Vehicle accident involving tractor-trailer rig causes a large unsuppressed fire involving the transport cask containing eight 30-W decay heat energy SNF elements.

Material-at-Risk Fresh fission products (see Table G-10).

Release Characteristics SNF transport cask is vented to the atmosphere. A vehicle accident resulting in a fire that engulfs the SNF transport cask could cause cladding imperfections to open or cause cladding damage resulting in a release of gaseous or volatile fission products. The release is assumed to be at ground level.

Airborne Release Fraction 3.00E−2 for strontium, 2.00E−3 for ruthenium, 4.00E−4 for cerium, 2.00E−1 for cesium, 6.00E−4 for promethium, and 4.00E−4 for plutonium (DOE-HDBK-3010-94, Table 6-10, p. 6-23).

Respirable Fraction 1.00E+0 (DOE-HDBK-3010-94, Section 6.3.1, pp. 6-17 to 6-21).

Damage Ratio 1.00E+0 (EDF TRA-ATR-1902).

Leak Path Factor 1.00E+0 (EDF TRA-ATR-1902).

Source Term The ST shown in Table G-13 was developed according to the following equation: ST = MAR × DR × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release

Table G-13. Source term for scenario release designator TRANSRR-3.

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Sr-90	2.83E+04	1.00E+0	1.00E+0	3.00E−2	1.00E+0	8.49E+2
Ru-106	4.98E+03	1.00E+0	1.00E+0	2.0E−3	1.00E+0	9.96E+0
Cs-134	8.02E+03	1.00E+0	1.00E+0	2.00E−1	1.00E+0	1.60E+3
Cs-137	2.83E+04	1.00E+0	1.00E+0	2.00E−1	1.00E+0	5.66E+3
Ce-144	3.84E+04	1.00E+0	1.00E+0	4.00E−4	1.00E+0	1.54E+1
Pm-147	3.22E+04	1.00E+0	1.00E+0	6.00E−4	1.00E+0	1.93E+1
Pu-238	1.51E+02	1.00E+0	1.00E+0	4.00E−4	1.00E+0	6.04E−2
Pu-239	5.62E−01	1.00E+0	1.00E+0	4.00E−4	1.00E+0	2.25E−4
Pu-240	5.61E−01	1.00E+0	1.00E+0	4.00E−4	1.00E+0	2.24E−4
Pu-241	6.46E+01	1.00E+0	1.00E+0	4.00E−4	1.00E+0	2.58E−2

2. Scenario TRANSRR-4, 634 W of Decay Heat Energy Spent Nuclear Fuel Cask Fire

Detailed Scenario Description Vehicle accident involving tractor-trailer rig causes a large unsuppressed fire involving the transport cask containing eight 634 W decay heat energy SNF elements.

Material-at-Risk Fresh fission products and actinides (see Table G-11).

Release Characteristics The SNF transport cask is vented to the atmosphere. A vehicle accident resulting in a fire that engulfs the SNF transport cask could cause cladding imperfections to open or cause cladding damage resulting in a release of gaseous or volatile fission products. The release is assumed to be at ground level.

Airborne Release Fraction 3.00E−2 for strontium, 6.00E−4 for yttrium, 4.00E−4 for zirconium, 2.00E−3 for ruthenium, 1.10E−1 for tellurium, 4.00E−4 for cerium, 2.00E−1 for cesium, 6.00E−4 for promethium, and 4.00E−4 for plutonium (DOE-HDBK-3010-94, Section 6-10, p. 6-23).

Respirable Fraction 1.00E+0 (DOE-HDBK-3010-94, Section 6.3.1, pp. 6-17 to 6-21).

Damage Ratio 1.00E+0 (EDF TRA-ATR-1902).

Leak Path Factor 1.00E+0 (EDF TRA-ATR-1902).

Source Term The ST shown in Table G-14 was developed according to the following equation: ST = MAR × DR × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

Table G-14. Source term for scenario release designator TRANSRR-4.

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Sr-89	1.41E+05	1.00E+0	1.00E+0	3.00E−2	1.00E+0	4.23E+3
Sr-90	3.04E+04	1.00E+0	1.00E+0	3.00E−2	1.00E+0	9.12E+2
Y-91	2.37E+05	1.00E+0	1.00E+0	6.00E−4	1.00E+0	1.42E+2
Zr-95	3.11E+05	1.00E+0	1.00E+0	4.00E−4	1.00E+0	1.24E+2
Nb-95	4.38E+05	1.00E+0	1.00E+0	3.00E−2	1.00E+0	1.31E+4
Ru-103	4.44E+04	1.00E+0	1.00E+0	2.00E−3	1.00E+0	8.88E+1
Ru-106	3.74E+04	1.00E+0	1.00E+0	2.00E−3	1.00E+0	7.48E+1
Te-127m	2.09E+03	1.00E+0	1.00E+0	1.10E−1	1.00E+0	2.30E+2
Cs-134	2.18E+04	1.00E+0	1.00E+0	2.00E−1	1.00E+0	4.36E+3
Cs-137	3.04E+04	1.00E+0	1.00E+0	2.00E−1	1.00E+0	6.08E+3

Table G-14. (continued).

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Ce-141	4.05E+04	1.00E+0	1.00E+0	4.00E−4	1.00E+0	1.62E+1
Ce-144	5.40E+05	1.00E+0	1.00E+0	4.00E−4	1.00E+0	2.16E+2
Pm-147	7.08E+04	1.00E+0	1.00E+0	6.00E−4	1.00E+0	4.25E+1
Pu-238	1.54E+02	1.00E+0	1.00E+0	4.00E−4	1.00E+0	6.16E−2
Pu-241	7.46E+01	1.00E+0	1.00E+0	4.00E−4	1.00E+0	2.98E−2

3. Scenario TRANSRR-5, 30-W of Decay Heat Energy Spent Nuclear Fuel Cask Drop

Detailed Scenario Description: The SNF transport cask, containing eight 30 W of decay heat energy SNF elements, topples from the tractor trailer causing some crushing of the fuel elements releasing gaseous and volatile radionuclides.

Material-at-Risk Fresh fission products (see Table G-10).

Release Characteristics There is a potential for some cladding damage due to crushing within the transport cask. The transport cask is vented to the atmosphere, so there is a direct release path to the environment.

Airborne Release Fraction 2.00E−3 (ANSI/ANS-5.10-1998, Appendix A, Table A-1, p. 15).

Respirable Fraction 7.00E−5 (ANSI/ANS-5.10-1998, Appendix A, Table A-1, p. 15).

Damage Ratio 1.00E−1 (assumes 10% fuel damage).

Leak Path Factor 1.00E+0 (ATR SNF transfer cask is vented to the atmosphere).

Source Term The ST shown in Table G-15 was developed according to the following equation: ST = MAR × DR × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2.

Table G-15. Source term for scenario release designator TRANSRR-5.

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Sr-90	2.83E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	3.96E−4
Ru-106	4.98E+03	1.00E−1	1.00E+0	2.00E−3	7.00E−5	6.97E−5
Cs-134	8.02E+03	1.00E−1	1.00E+0	2.00E−3	7.00E−5	1.12E−4
Cs-137	2.83E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	3.96E−4
Ce-144	3.84E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	5.38E−4

Table G-15. (continued).

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Pm-147	3.22E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	4.51E−4
Pu-238	1.51E+02	1.00E−1	1.00E+0	2.00E−3	7.00E−5	2.11E−6
Pu-239	5.62E−01	1.00E−1	1.00E+0	2.00E−3	7.00E−5	7.87E−9
Pu-240	5.61E−01	1.00E−1	1.00E+0	2.00E−3	7.00E−5	7.85E−9
Pu-241	6.46E+01	1.00E−1	1.00E+0	2.00E−3	7.00E−5	9.04E−7

4. Scenario TRANSRR-6, 634 W of Decay Heat Energy Spent Nuclear Fuel Cask Drop

Detailed Scenario Description The SNF transport cask, containing eight 634 W of decay heat energy SNF elements, topples from the tractor trailer causing some crushing of the fuel elements releasing gaseous and volatile radionuclides.

Material-at-Risk Fresh fission products (see Table G-11).

Release Characteristics There is a potential for some cladding damage due to crushing within the transport cask. The transport cask is vented to the atmosphere, so there is a direct release path to the environment.

Airborne Release Fraction 2.00E−3 (ANSI/ANS-5.10-1998, Appendix A, Table A-1, p. 15).

Respirable Fraction 7.00E−5 (ANSI/ANS-5.10-1998, Appendix A, Table A-1, p. 15).

Damage Ratio 1.00E−1 (assumes 10% fuel damage).

Leak Path Factor 1.00E+0 (ATR SNF transfer cask is vented to the atmosphere).

Source Term The ST shown in Table G-16 was developed according to the following equation: ST = MAR × DR × LPF × ARF × RF.

Modeling Software and Inputs RSAC, Version 6.2.

Table G-16. Source term for scenario release designator TRANSRR-6.

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Sr-89	1.41E+05	1.00E−1	1.00E+0	2.00E−3	7.00E−5	1.97E−3
Sr-90	3.04E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	4.26E−4
Y-91	2.37E+05	1.00E−1	1.00E+0	2.00E−3	7.00E−5	3.32E−3
Zr-95	3.11E+05	1.00E−1	1.00E+0	2.00E−3	7.00E−5	4.35E−3
Nb-95	4.38E+05	1.00E−1	1.00E+0	2.00E−3	7.00E−5	6.13E−3

Table G-16. (continued).

Nuclide	MAR (Ci)	DR	LPF	ARF	RF	ST (Ci)
Ru-103	4.44E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	6.22E−4
Ru-106	3.74E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	5.24E−4
Te-127m	2.09E+03	1.00E−1	1.00E+0	2.00E−3	7.00E−5	2.93E−5
Cs-134	2.18E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	3.05E−4
Cs-137	3.04E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	4.26E−4
Ce-141	4.05E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	5.67E−4
Ce-144	5.40E+05	1.00E−1	1.00E+0	2.00E−3	7.00E−5	7.56E−3
Pm-147	7.08E+04	1.00E−1	1.00E+0	2.00E−3	7.00E−5	9.91E−4
Pu-238	1.54E+02	1.00E−1	1.00E+0	2.00E−3	7.00E−5	2.16E−6
Pu-241	7.46E+01	1.00E−1	1.00E+0	2.00E−3	7.00E−5	1.04E−6

5. Scenario TRANSDR-2, 30 W of Decay Heat Energy Spent Nuclear Fuel Cask Shielding Failure

Detailed Scenario Description Vehicle accident causes the SNF transport cask to topple over and dislodges the cover lid.

Material-at-Risk Gamma radiation from fresh fission products (see Table G-10).

Release Characteristics Electromagnetic (gamma) radiation.

Airborne Release Fraction N/A.

Respirable Fraction N/A.

Damage Ratio N/A.

Leak Path Factor N/A.

Source Term The exposure rate ST shown in Table G-17 was developed according to the inverse square law equation.

Modeling Software and Inputs Manual calculations using the inverse square law equation.

$$\frac{I_1}{I_2} = \frac{d_2^2}{d_1^2}$$

Where I_1 is the exposure rate near the source, I_2 is the exposure rate at a new distance, d_2 is the new distance, and d_1 is the distance at which I_1 was measured.

Table G-17. Exposure rate source term for scenario release designator TRANSDR-2 (EDF-8027[12]).

Scenario Release Designator	Distance (m)			TEL Distance (m)[a]	PAC Distance (m)[b]	Administrative Excess Exposure Authorization Distance (m)[c]
	30	100	200			
TRANSDR-2 (ATR fuel cask, 30-W decay power exposure rates, R/hr)	4.9 R/hr	0.4 R/hr	0.1 R/hr	N/A	67	80

a. TEL = 100 R/hr for a 1-hour exposure duration.
b. PAC = 1 R/hr for a 1-hour exposure duration.
c. Administrative limit requiring authorization of excess dose for emergency workers = 0.7 R/hr for a 1-hour exposure duration.

6. Scenario TRANSDR-3, 634 W of Decay Heat Energy Spent Nuclear Fuel Cask Shielding Failure

Detailed Scenario Description Vehicle accident causes the SNF transport cask to topple over and dislodges the cover lid.

Material-at-Risk Gamma radiation from fresh fission products (see Table G-11).

Release Characteristics Electromagnetic (gamma) radiation.

Airborne Release Fraction N/A.

Respirable Fraction N/A.

Damage Ratio N/A.

Leak Path Factor N/A.

Source Term The exposure rate ST shown in Table G-18 was developed according to the inverse square law equation.

Modeling Software and Inputs Manual calculations using the inverse square law equation.

$$\frac{I_1}{I_2} = \frac{d_2^2}{d_1^2}$$

Where I_1 is the exposure rate near the source, I_2 is the exposure rate at a new distance, d_2 is the new distance, and d_1 is the distance at which I_1 was measured.

Table G-18. Exposure rate source term for scenario release designator TRANSDR-3 (EDF-8027).

Scenario Release Designator	Distance (m)			TEL Distance (m)[a]	PAC Distance (m)[b]	Administrative Excess Exposure Authorization Distance (m)[c]
	30	100	200			
TRANSDR-3 (ATR fuel cask, 634-W decay power exposure rates, R/hr)	252 R/hr	22.7 R/hr	5.7 R/hr	48	478	571

a. TEL = 100 R/hr for a 1-hour exposure duration.
b. PAC = 1 R/hr for a l-hour exposure duration.
c. Administrative limit requiring authorization of excess dose for emergency workers = 0.7 R/hr for a 1-hour exposure duration.

G-1.4.1.3 Radiological Hazardous Material — Nevada Test Site Shipments

G-1.4.1.3.1 Properties. The properties for Nevada Test Site (NTS) shipments are listed in Table G-19.

Table G-19. Radiological properties for Nevada Test Site shipments.

Physical form	Solid waste contaminated with transuranic (TRU) and non-TRU nuclides.
Particle size	Respirable.
Flammability	Flammable or nonflammable depending upon waste type (i.e., combustible or noncombustible).
Reactivity	Not reactive.
Density	N/A.
Special firefighting concerns	None.
Health concerns	Acute health effects caused by exposure to gamma radiation, or potential latent health effects (cancer or genetic effects) caused by inhalation or ingestion of radionuclides.

G-1.4.1.3.2 Conditions of Storage and Use. Low-level waste or mixed waste may be transported in several different package types such as cargo containers, a B-25 box, 55-gal drums, or a burrito wrap (a strong heavy fabric that may be used as a liner in a wooden or metal outer container or in some cases, as a stand-alone container). NTS waste acceptance criteria (WAC) are stated in terms of plutonium equivalent grams (PE-g). Nuclides and conversion factors for converting to PE-g are found in Appendix C of MCP-9391.[13] NTS WAC limits the quantity of radioactive material in terms of Pu-239 equivalent grams for a single waste container or full shipment of waste containers. The PE-g limit for a single container is 300 PE-g total and limit for a shipment is 2,000 PE-g. Specific activity of Pu-239 (6.22E−2 Ci/g) was used to convert the PE-g values to curie units as shown in Table G-20.

Table G-20. Material-at-risk for Nevada Test Site shipments.

Single Waste Container		Full Shipment of Waste Containers	
Radionuclide	Ci	Radionuclide	Ci
Pu-239	1.87E+1	Pu-239	1.24E+2

The primary barrier is the shipping container.

The secondary barrier is essentially none.

There are no engineered controls that directly apply to transportation containers.

Administrative controls require that shipping containers and shipments must meet the NTS WAC.

Indicators are limited to visual observation or portable radiation survey instruments readings, when a waste shipment is in transport.

G-1.4.1.3.3 Barrier and Failure Mode Analyses. The barrier and failure mode analyses presented below for NTS shipments are summarized in Table G-21.

Table G-21. Radiological failure modes and barriers for Nevada Test Site shipments.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRANSRR-7	Pu-239 equivalent (18.7 Ci)	Waste container	Radiological waste non-DOT shipping container puncture; operational incident	Forklift tine punctures waste container while loading waste shipment	10% container damage	Essentially none
TRANSRR-8	Pu-239 equivalent (18.7 Ci)	Waste container	Radiological waste non-DOT shipping container breach and partial spill; operational incident or vehicle accident	Heavy equipment damages waste container or vehicle accident damages container spilling 25% of contents	25% of container spilled	Essentially none
TRANSRR-9	Pu-239 equivalent (18.7 Ci)	Waste container	Radiological waste non-DOT shipping container total spill; operational incident	Waste container torn open while loading waste shipment due to equipment malfunction or operator error	100% of container spilled	Essentially none
TRANSRR-10	Pu-239 equivalent (124 Ci)	Waste container	Shipment of non-DOT radiological waste shipping containers damaged in vehicle accident	Tractor-trailer rollover accident damages shipping containers	25% of container spilled	Essentially none

Table G-21. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRANSRR-11	Pu-239 equivalent (124 Ci)	Waste container	Shipment of non-DOT radiological waste shipping containers damaged in vehicle accident with a fire	Vehicle accident causes fire, which engulfs radioactive waste shipment	25% of container contents released by fire	Essentially none

1. **Scenario TRANSRR-7, Small Breach of a Single Waste Container**

Detailed Scenario Description	Transport vehicle accident results in puncture of single non-DOT Type-B radiological waste shipping container.
Material-at-Risk	TRU and non-TRU radiological waste (see Table G-20; 18.7 Ci of Pu-239 equivalent).
Release Characteristics	Vehicle accident or operator error (turning too sharply and contacting an immoveable or sharp object) causes a puncture of the waste container. Waste container is assumed to contain 300 PE-g of radionuclides, which is equivalent to 18.7 Ci of Pu-239. Puncture conservatively assumes 10% damage to the container. The release is assumed to be at ground level.
Airborne Release Fraction	1.0E−3 (DOE-HDBK-3010-94, Section 5.2.3.2, paragraph 2, p. 5-20).
Respirable Fraction	1.0E−1 (DOE-HDBK-3010-94, Section 5.2.3.2, paragraph 2, p. 5-20).
Damage Ratio	1.0E−1 (scenario assumption of 10% damage).
Leak Path Factor	1.0E+0 (open to the environment).
Source Term	ST = 18.7 Ci × 1.0E−3 × 1.0E−1 × 1.0E−1 × 1.0 = 1.87E−4 Ci of Pu-239 equivalent.
Modeling Software and Inputs	RSAC, Version 6.2. Ground-level release.

2. **Scenario TRANSRR-8, Moderate Breach of a Single Waste Container**

Detailed Scenario Description	Transport vehicle accident results in puncture of single non-DOT Type-B radiological waste shipping container.
Material-at-Risk	TRU and non-TRU radiological waste (see Table G-20; 18.7 Ci of Pu-239 equivalent).
Release Characteristics	Vehicle accident or operator error (turning too sharply and contacting an immoveable or sharp object) causes a puncture of the waste container. Waste container is assumed to contain 300 PE-g of radionuclides, which is equivalent to 18.7 Ci of Pu-239. Damage to the container spills 25% of the contents. The release is assumed to be at ground level.
Airborne Release Fraction	1.0E−3 (DOE-HDBK-3010-94, Section 5.2.3.2, paragraph 2, p. 5-20).
Respirable Fraction	1.0E−1 (DOE-HDBK-3010-94, Section 5.2.3.2, paragraph 2, p. 5-20).
Damage Ratio	2.5E−1 (accident scenario assumes 25% container spill).

Leak Path Factor 1.0E+0 (open to the environment).

Source Term ST = 18.7 Ci × 1.0E−3 × 1.0E−1 × 2.5E−1 × 1.0 = 4.68E−4 Ci of Pu-239 equivalent.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

3. **Scenario TRANSRR-9, Complete Breach and Spill of a Single Waste Container**

Detailed Scenario Description Transport vehicle accident results in puncture of single non-DOT Type-B radiological waste shipping container.

Material-at-Risk TRU and non-TRU radiological waste (see Table G-20; 18.7 Ci of Pu-239 equivalent).

Release Characteristics Vehicle accident or operator error (turning too sharply and contacting an immoveable or sharp object) causes a puncture of the waste container. Waste container is assumed to contain 300 PE-g of radionuclides, which is equivalent to 18.7 Ci of Pu-239. Damage to the container spills 100% of the contents. The release is assumed to be at ground level.

Airborne Release Fraction 1.0E−3 (DOE-HDBK-3010-94, Section 5.2.3.2, paragraph 2, p. 5-20).

Respirable Fraction 1.0E−1 (DOE-HDBK-3010-94, Section 5.2.3.2, paragraph 2, p. 5-20).

Damage Ratio 1.0E+0 (scenario assumes 100% spill of container contents).

Leak Path Factor 1.0E+0 (open to the environment).

Source Term ST = 18.7 Ci × 1.0E−3 × 1.0E−1 × 1.0 × 1.0 = 1.87E−3 Ci of Pu-239 equivalent.

Modeling Software and Inputs RSAC, Version 6.2. Ground-level release.

4. **Scenario TRANSRR-10, Moderate Breach of Waste Containers**

Detailed Scenario Description Transport vehicle accident results in damage to non-DOT Type-B radiological waste shipping containers.

Material-at-Risk TRU and non-TRU radiological waste (see Table G-20 124 Ci of Pu-239 equivalent).

Release Characteristics Vehicle accident or trailer rollover causes damage to the waste containers. Waste containers are assumed to contain 2,000 PE-g of radionuclides, which is equivalent to 124 Ci of Pu-239. Damage to the containers spills 25% of the contents. The release is assumed to be at ground level.

Airborne Release Fraction 1.0E−3 (DOE-HDBK-3010-94, Section 5.2.3.2, paragraph 2, p. 5-20).

Respirable Fraction	1.0E−1 (DOE-HDBK-3010-94, Section 5.2.3.2, paragraph 2, p. 5-20).
Damage Ratio	2.5E−1 (scenario assumes spill of 25% of contents).
Leak Path Factor	1.0E+0 (open to the environment).
Source Term	ST = 124 Ci × 1.0E−3 × 1.0E−1 × 2.5E−1 × 1.0 = 3.10E−3 Ci of Pu-239 equivalent.
Modeling Software and Inputs	RSAC, Version 6.2. Ground-level release.

5. **Scenario TRANSRR-11, Moderate Breach of Waste Containers and Fire**

Detailed Scenario Description	Transport vehicle accident results in damage to non-DOT Type-B radiological waste shipping containers.
Material-at-Risk	TRU and non-TRU radiological waste (see Table G-20 124 Ci of Pu-239 equivalent.
Release Characteristics	Vehicle accident or trailer rollover causes damage to the waste containers. Waste containers are assumed to contain 2,000 PE-g of radionuclides, which is equivalent to 124 Ci of Pu-239. Damage to the containers spills 25% of the contents. The release is assumed to be at ground level.
Airborne Release Fraction	5.0E−4 (DOE-HDBK-3010-94, Section 5.2.1.1, p. 5-13, last paragraph).
Respirable Fraction	1.0E+0 (DOE-HDBK-3010-94, Section 5.2.1.1, p. 5-13, last paragraph).
Damage Ratio	2.5E−1 (scenario assumes spill of 25% of contents).
Leak Path Factor	1.0E+0 (open to the environment).
Source Term	ST = 124 Ci × 5.0E−4 × 1.0 × 2.5E−1 × 1.0 = 1.55E−2 Ci of Pu-239 equivalent.
Modeling Software and Inputs	RSAC, Version 6.2. Ground-level release.

G-1.4.1.4 Radiological — Transuranic Material Shipments

Experimental fuels containing TRU material may be transported between the Materials and Fuels Complex and the ATR Complex.

G-1.4.1.4.1 Properties. The properties for TRU material are listed in Table G-22.

Table G-22. Radiological properties for transuranic material shipments.

Physical form	Solid fuels containing TRU. TRU nuclides are listed in terms of Pu-239 fissile grams equivalent (FGE) and plutonium equivalent curies (PEC). For conversion to PEC, it is conservative to assume the FGE is composed entirely of Pu-239.[14]
Particle size	Respirable.
Flammability	Flammable or nonflammable depending upon waste type (i.e., combustible or noncombustible).
Reactivity	Not reactive.
Density	N/A.
Special firefighting concerns	None.
Health concerns	Acute health effects caused by internal exposure to alpha radiation that has potential for latent health effects (cancer or genetic effects).

G-1.4.1.4.2 Conditions of Storage and Use. TRU waste may be transported in package types such as 55-gal drums or standard waste boxes (SWBs). Unirradiated experimental fuels containing TRU material are transported in 110-gal 6M drums. Waste Isolation Pilot Plant WAC[14] limits the quantity of radioactive material in terms of Pu-239 FGE for a single container or full shipment of containers. The FGE limit for a single container is less than or equal to 200 FGE total and the limit for an SWB is less than or equal to 325 FGE. Specific activity of Pu-239 (6.22E−2 Ci/g) was used to convert the FGE values to curie units shown in Table G-23.

Table G-23. Material-at-risk for transuranic material shipping containers.

Single Waste Drum (95th Percentile)		SWB	
Radionuclide	Ci	Radionuclide	Ci
Pu-239	1.24E+1 PEC	Pu-239	2.02E+1 PEC
Single Waste Drum (Outlier)		Maximum Trailer (40 Drums)	
Radionuclide	Ci	Radionuclide	Ci
Pu-239	4.71E+1 PEC	Pu-239	4.98E+2 PEC
Normal Shipment (32 Drums)		Worst-Case SWB	
Radionuclide	Ci	Radionuclide	Ci
Pu-239	3.98E+2 PEC	Pu-239	9.36E+1 PEC
Single Drum (TRU Experimental Fuel)		Maximum Trailer (40 Drums TRU Experimental Fuel)	
Radionuclide	Ci	Radionuclide	Ci
Pu-239	1.24E+1 PEC	Pu-239	4.98E+2 PEC

The primary barrier is the shipping container.

The secondary barrier is essentially none.

There are no engineered controls that directly apply to transportation containers.

Administrative controls require that shipping containers and shipments meet the Waste Isolation Pilot Plant WAC, PLN-1851,[15] or PLN-3055.[16]

Indicators are limited to visual observation or portable radiation survey instruments readings when a shipment is in transport.

G-1.4.1.4.3 Barrier and Failure Mode Analyses. The barrier and failure mode analyses presented below for events related to transportation between the Materials and Fuels Complex and the ATR Complex are summarized in Table G-24.

G-1.4.1.4.4 Scenario Bases and Analytical Assumptions

1. **Scenario TRANSRR-12, Moderate Breach of a Single Drum**

Detailed Scenario Description	Transport vehicle accident results in breach of a single drum.
Material-at-Risk	TRU containing materials.[15,16] See Table G-23. 47.1 PEC. Outlier drum described in SAR-4,[17] Sections 3.3.1.2.2 and 3.3.2.1.2.2, “AR Project Radioactive Material Inventory.”
Release Characteristics	Vehicle accident or operator error (turning too sharply and contacting an immoveable or sharp object) causes a breach of the drum. The drum is assumed to contain 47.1 PEC. Puncture conservatively assumes 50% damage to the container. Release is assumed to be at ground level.
Airborne Release Fraction	1.0E−3 (EDF-4711,[14] Table 3, all container breach events).
Respirable Fraction	1.0E−1 (EDF-4711, Table 3, all container breach events).
Damage Ratio	5.0E−1 (assumes 50% damage).
Leak Path Factor	1.0.
Source Term	ST = 47.1 Ci × 1.0E−3 × 1.0E−1 × 5.0E−1 × 1.0 = 2.36E−3 Ci of Pu-239 equivalent.
Modeling Software and Inputs	RSAC, Version 6.2.

Table G-24. Failure modes and barriers for transuranic material transportation.

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRANSRR-12	Pu-239 equivalent (47.1 Ci)	Drum wall	Moderate drum breach (0.5 DR)	Natural phenomenon, equipment malfunction, or vehicle accident	Fraction of container contents	Essentially none
TRANSRR-13	Pu-239 equivalent (47.1 Ci)	Drum wall	Large drum breach (1.0 DR)	Natural phenomenon, equipment malfunction, or vehicle accident	Container contents	Essentially none
TRANSRR-14	Pu-239 equivalent (47.1 Ci)	Drum wall	One drum explosion/ deflagration	Natural phenomenon, equipment malfunction, or vehicle accident and fire or deflagration due to pyrophoric material or hydrogen buildup or from an external source	Container contents	Essentially none
TRANSRR-15	Pu-239 equivalent (498 Ci)	Drum wall	Tractor-trailer accident with 40 drums and no fire	Natural phenomenon, equipment malfunction, or vehicle accident	Fraction of contents	Essentially none

Table G-24. (continued).

Scenario Release Designator	Material	Primary Barrier	Failure Modes and Causes	Possible Initiating Events and Scenarios	Release From Primary Barrier	Other Barriers and Their Effects
TRANSRR-16	Pu-239 equivalent (498 Ci)	Drum wall	Tractor-trailer accident with 40 drums and fire	Natural phenomenon, equipment malfunction, or vehicle accident	Fraction of contents	Essentially none

2. **Scenario TRANSRR-13, Large Breach of a Single Drum**

Detailed Scenario Description	Transport vehicle accident results in catastrophic breach of a single drum.
Material-at-Risk	TRU containing material. See Table G-23. 47.1 PEC. Outlier drum described in SAR-4, Sections 3.3.1.2.2 and 3.3.2.1.2.2, "AR Project Radioactive Material Inventory."
Release Characteristics	Vehicle accident or operator error (turning too sharply and contacting an immoveable or sharp object) causes a breach of the drum. The drum is assumed to contain 47.1 PEC. Puncture conservatively assumes 100% damage to the container. Release is assumed to be at ground level.
Airborne Release Fraction	1.0E-3 (EDF-4711, Table 3, all container breach events).
Respirable Fraction	1.0E-1 (EDF-4711, Table 3, all container breach events).
Damage Ratio	1.0.
Leak Path Factor	1.0.
Source Term	ST = 47.1 Ci × 1.0E-3 × 1.0E−1 × 1.0 × 1.0 = 4.71E−3 Ci of Pu-239 equivalent.
Modeling Software and Inputs	RSAC, Version 6.2.

3. **Scenario TRANSRR-14, Single Drum Explosion/Deflagration**

Detailed Scenario Description	Transport vehicle accident or natural phenomenon results in the explosion/deflagration of a single drum.
Material-at-Risk	TRU containing material. See Table G-23. 47.1 PEC. Outlier drum described in SAR-4, Sections 3.3.1.2.2 and 3.3.2.1.2.2, "AR Project Radioactive Material Inventory."
Release Characteristics	Vehicle accident or operator error (turning too sharply and contacting an immoveable or sharp object) causes a breach of the drum. The drum is assumed to contain 47.1 PEC. Puncture conservatively assumes 100% damage to the container. Release is assumed to be at ground level.
Airborne Release Fraction	5.0E−4 (EDF-4711, Table 3, drum overpressure/deflagration events).
Respirable Fraction	1.0 (EDF-4711, Table 3, drum overpressure/deflagration events).
Damage Ratio	1.0.
Leak Path Factor	1.0.
Source Term	ST = 47.1 Ci × 5.0E−4 × 1.0 × 1.0 × 1.0 = 2.36E−2 Ci of Pu-239 equivalent.
Modeling Software and Inputs	RSAC, Version 6.2.

4. Scenario TRANSRR-15, Tractor-Trailer Accident With 40 Drums and No Fire

Detailed Scenario Description	Transport vehicle accident damages drum shipment.
Material-at-Risk	TRU containing materials. See Table G-23. 498 PEC. Forty 95% drums at 12.45 PEC per drum (EDF-4711, Table 1, WIPP acceptance criteria).
Release Characteristics	Vehicle accident causes damage to a shipment of drums. The drum shipment is assumed to contain 498 PEC. Conservatively assumes 10% damage to the waste containers. Release is assumed to be at ground level.
Airborne Release Fraction	1.0E−3 (EDF-4711, Table 3, all container breach events).
Respirable Fraction	1.0E−1 (EDF-4711, Table 3, all container breach events).
Damage Ratio	0.1 (EDF-4711, Table 2, all container breach events).
Leak Path Factor	1.0.
Source Term	ST = 498 Ci × 1.0E−3 × 1.0E−1 × 0.1 × 1.0 = 4.98E−3 Ci of Pu-239 equivalent.
Modeling Software and Inputs	RSAC, Version 6.2.

5. Scenario TRANSRR-16, Tractor-Trailer Accident With 40 Drums and Fire

Detailed Scenario Description	Transport vehicle accident and fire damages drum shipment.
Material-at-Risk	TRU containing material. See Table G-23. 498 PEC. Forty 95% drums at 12.45 PEC per drum (EDF-4711, Table 1, WIPP acceptance criteria).
Release Characteristics	Vehicle accident causes damage to a shipment of drums. The drum shipment is assumed to contain 498 PEC. Conservatively assumes 35% damage to the waste containers. Release is assumed to be at ground level.
Airborne Release Fraction	2.7E−4 (EDF-4711, Table 3).
Respirable Fraction	1.0 (EDF-4711, Table 3, pit/deck fires).
Damage Ratio	0.35 (EDF-4711, Table 22, conservative assumption combining all event DR values).
Leak Path Factor	1.0.
Source Term	ST = 498 Ci × 2.7E−3 × 1.0 × 0.35 × 1.0 = 4.71E−2 Ci of Pu-239 equivalent.
Modeling Software and Inputs	RSAC, Version 6.2.

G-1.5 Evaluation Results

G-1.5.1 Radiological Hazardous Material Release Results

Radiological hazardous material release results for 95% worst-case and 50% typical weather conditions as described in the main document are presented in Tables G-25 and G-26.

Table G-25. Radiological release scenario calculation results for 95% worst-case meteorology.

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-1	Explosion or fire in loading area or on Site roadway causes test train fuel cladding damage and radioactive material release	8.96E+2 (thyroid CDE[a] 1.42E+4)	8.96E+1 (thyroid CDE 1.42E+3)	4.02E+1 (thyroid CDE 6.39E+2)	3.58E−1 (thyroid CDE 5.61E+0)	3.62E−2 (thyroid CDE 5.71E−1)	3,000 (based on thyroid CDE)	Less than 100	Site area emergency (SAE) (potential for ingestion pathway advisories)
TRANSRR-2	Test train cask drop from tractor trailer while on Site roadway with visible cask damage and radioactive material release	2.41E−3	2.41E−4	1.49E−4	6.51E−6	7.32E−7	N/A	Not exceeded	None (this is an operational incident)
TRANSRR-3	Explosion or fire involving ATR SNF cask with eight (30-W) decay power fuel elements while on Site roadway	1.10E+4	1.10E+3	4.93E+2	4.37E+0	4.54E−1	6,400	800	SAE (potential for ingestion pathway advisories)

Table G-25. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-4	Explosion or fire involving ATR SNF cask with eight (634-W) decay power fuel elements while on Site roadway	1.60E+4	1.60E+3	7.16E+2	6.36E+0	6.62E−1	8,075	1,035	SAE (potential for ingestion pathway advisories)
TRANSRR-5	ATR SNF cask with eight (30-W) decay power fuel elements, cask drop while on Site roadway	1.27E−2	1.27E−3	5.69E−4	5.03E−6	5.24E−7	N/A	Not exceeded	None (this is an operational incident)
TRANSRR-6	ATR SNF cask with eight (634-W) decay power fuel elements, cask drop while on Site roadway	3.87E−2	3.87E−3	1.74E−3	1.54E−5	1.60E−6	N/A	Not exceeded	None (this is an operational incident)
TRANSRR-7	Non-DOT Type-B radiological waste shipping container puncture	6.25E−1	6.25E−2	2.81E−2	2.48E−4	2.58E−5	N/A	Not exceeded	None (this is an operational incident)

Table G-25. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-8	Non-DOT Type-B radiological waste shipping container breach with 25% spilled	1.57E+0	1.57E−1	7.02E−2	6.21E−4	6.46E−5	100 (default distance)	Not exceeded	Alert (greater than 10% of PAG[a] at 100 m)
TRANSRR-9	Non-DOT Type-B radiological waste shipping container breach with 100%37 spilled	6.25E+0	6.25E−1	2.81E−1	2.48E−3	2.58E−4	100 (default distance)	Not exceeded	Alert (greater than 10% of PAG at 100 m)
TRANSRR-10	Non-DOT Type-B radiological waste shipping container shipment rollover and breach	1.04E+1	1.04E+0	4.65E−1	4.12E−3	4.28E−4	105	Not exceeded	Alert
TRANSRR-11	Non-DOT Type-B radiological waste shipping container shipment fire	5.18E+1	5.18E+0	2.33E+0	2.06E−2	2.14E−3	410	Not exceeded	SAE

Table G-25. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-12	Moderate breach of one drum (0.5 DR)	7.89+0	7.89E−1	3.54E−1	3.13E−3	3.26E−4	100	Not exceeded	Alert
TRANSRR-13	Large breach of one drum (1.0 DR)	1.58E+1	1.58E+0	7.07E−1	6.25E−3	6.50E−4	200	Not exceeded	SAE
TRANSRR-14	One drum explosion/ deflagration	7.89E+1	7.89E+0	1.59E+0	3.13E−2	3.26E−3	700	Not exceeded	SAE
TRANSRR-15	Tractor-trailer accident with 40 drums and no fire	1.67E+1	1.67E+0	7.47E−1	6.61E−3	6.88E−4	200	Not exceeded	SAE
TRANSRR-16	Tractor-trailer accident with 40 drums and fire	7.07E+1	7.07E+0	3.17E+0	6.25E−2	6.50E−3	1,100	Not exceeded	SAE
TRANSDR-1	Test train cask with dislodged lid	516 R/hr	46.5 R/hr	11.6 R/hr	57 mR/hr	4 mR/hr	682	68	SAE
TRANSDR-2	ATR fuel cask containing 30-W decay power fuel with lid dislodged	4.9 R/hr	400 mR/hr	100 mR/hr	5.4E−1 mR/hr	3.7E−2 mR/hr	67	Not exceeded	Alert

Table G-25. (continued).

Scenario Release Designator	Short Description	Downwind TEDE[a] Estimates for 95% Worst-Case Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSDR-3	ATR fuel cask containing 634-W decay power fuel with lid dislodged	252 R/hr	22.7 R/hr	5.7 R/hr	28 mR/hr	2 mR/hr	478	48	SAE

a. TEDE = total effective dose equivalent
CDE = committed dose equivalent
PAG = Protective Action Guide.

Table G-26. Radiological release scenario calculation results for 50% typical meteorology.

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-1	Explosion or fire in loading area or on Site roadway causes test train fuel cladding damage and radioactive material release	5.82E+0 (thyroid CDE 9.13E+1)	5.82E−1 (thyroid CDE 9.13E+0)	2.63E−1 (thyroid CDE 4.10E+0)	1.05E−2 (thyroid CDE 1.63E−1	1.58E−3 (thyroid CDE 2.48E−2)	145 (based on thyroid dose)	Not exceeded	Alert

Table G-26. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-2	Test train cask drop from tractor trailer while on Site roadway with visible cask damage and radioactive material release	5.57E−7	5.57E−6	3.48E−6	2.42E−7	4.09E−8	N/A	Not exceeded	None (this is an operational incident)
TRANSRR-3	Explosion or fire involving ATR SNF cask with eight (30-W) decay power fuel elements while on Site roadway	7.04E+1	7.04E+0	3.16E+0	1.27E−1	1.98E−2	560	Not exceeded	SAE (potential for ingestion pathway advisories)
TRANSRR-4	Explosion or fire involving ATR SNF cask with eight (634-W) decay power fuel elements while on Site roadway	1.02E+2	1.20E+1	4.61E+0	1.85E−1	2.88E−2	725	Less than 100	SAE (potential for ingestion pathway advisories)

Table G-26. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-5	ATR SNF cask with eight (30-W) decay power fuel elements, cask drop while on Site roadway	8.13E−7	8.13E−6	3.65E−6	1.46E−7	2.28E−8	N/A	Not exceeded	None (this is an operational incident)
TRANSRR-6	ATR SNF cask with eight (634-W) decay power fuel elements, cask drop while on Site roadway	2.48E−6	2.48E−5	1.12E−5	4.47E−7	6.96E−8	N/A	Not exceeded	None (this is an operational incident)
TRANSRR-7	Non-DOT Type-B radiological waste shipping container puncture	4.01E−5	4.01E−4	1.80E−4	7.22E−6	1.12E−6	N/A	Not exceeded	None (this is an operational incident)
TRANSRR-8	Non-DOT Type-B radiological waste shipping container breach with 25% spilled	1.00E−4	1.00E−3	4.51E−4	1.81E−5	2.81E−6	N/A	Not exceeded	Not an emergency

Table G-26. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-9	Non-DOT Type-B radiological waste shipping container breach with 100% spilled	4.01E−4	4.01E−3	1.80E−3	7.22E−5	1.12E−5	N/A	Not exceeded	Not an emergency
TRANSRR-10	Non-DOT Type-B radiological waste shipping container shipment rollover and breach	6.65E−2	6.65E−3	2.99E−3	1.20E−4	1.86E−5	N/A	Not exceeded	Not an emergency
TRANSRR-11	Non-DOT Type-B radiological waste shipping container shipment fire	3.32E−1	3.32E−2	1.49E−2	5.98E−4	9.30E−5	N/A	Not exceeded	Not an emergency
TRANSRR-12	Moderate breach of one drum (0.5 DR)	5.06E−2	5.06E−3	3.37E−3	9.11E−5	1.42E−5	N/A	Not exceeded	Not an emergency
TRANSRR-13	Large breach of one drum (1.0 DR)	1.01E−1	1.01E−2	6.73E−3	1.82E−4	2.83E−5	N/A	Not exceeded	Not an emergency
TRANSRR-14	One drum explosion/ deflagration	5.06E−1	5.06E−2	3.37E−2	9.11E−4	1.42E−4	N/A	Not exceeded	Not an emergency

Table G-26. (continued).

Scenario Release Designator	Short Description	Downwind TEDE Estimates for 50% Typical Meteorology (rem)					Protective Action Distance (m)	Threshold for Early Lethality Distance (m)	Emergency Class
		Collocated Worker (30 m)	100 m	Facility Boundary (200 m)	Collocated Facility (2,865 m)	Site Boundary (10,855 m)			
TRANSRR-15	Tractor-trailer accident with 40 drums and no fire	1.07E−1	1.07E−2	7.12E−3	1.92E−4	6.55E−5	N/A	Not exceeded	Not an emergency
TRANSRR-16	Tractor-trailer accident with 40 drums and fire	1.01E+0	1.01E−1	6.72E−2	1.82E−3	2.83E−4	100	Not exceeded	Alert (10% of PAG at 100 m)
TRANSDR-1	Test train cask with dislodged lid	516 R/hr	46.5 R/hr	11.6 R/hr	57 mR/hr	4 mR/hr	682	68	SAE
TRANSDR-2	ATR fuel cask containing 30-W decay power fuel with lid dislodged	4.9 R/hr	400 mR/hr	100 mR/hr	5.4E−1 mR/hr	3.7 mR/hr	67	Not exceeded	Alert
TRANSDR-3	ATR fuel cask containing 634-W decay power fuel with lid dislodged	252 R/hr	22.7 R/hr	5.7 R/hr	28 mR/hr	2 mR/hr	478	48	SAE

G-1.6 Emergency Classes, Protective Actions, and Emergency Action Levels

Based on the analyses documented above, the following EALs are identified for ATR Complex transportation activities.

G-1.6.1 Unclassified Operational Emergency — Emergency Action Levels

Unless otherwise noted, the generic unclassified operational emergency (UOE) EALs are covered by a separate appendix to this EHA.

There are no onsite transportation UOE EALs covered in this appendix.

G-1.6.2 Alert — Emergency Action Levels

G-1.6.2.1 ATR-TRN-3.A.1

G-1.6.2.1.1 Event Description

Radiological waste shipment to the Nevada Test Site is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes the breach and spill of 25% to 100% of a single non-Department of Transportation Type-B radiological waste container or breach of a shipment of non-Department of Transportation Type-B radiological waste containers and a radiological release,

G-1.6.2.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual confirmation of the container breach,

OR

radiological control technician confirmation of an airborne radiological release by radiation survey.

G-1.6.2.1.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **105 m (345 ft)** from the accident site.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.2.1.4 Offsite Protective Action Recommendations

None.

G-1.6.2.1.5 Basis. Scenarios TRANSRR-8 and -9 involve the 25% and 100% spill of a single radiological waste container and scenario TRANSRR-10 involves a radiological waste shipment rollover and spill. The single non-DOT Type-B radiological waste container is assumed to contain 18.7 Ci of Pu-239 equivalent and the shipment is limited to 124 Ci of Pu-239 equivalent. Scenarios TRANSRR-8 and -9 exceed 10% of the PAC at 100 m and scenario TRANSRR-10 exceeds the PAC to a distance of 105 m. Therefore, these events meet the criteria for an alert classification.

G-1.6.2.2 ATR-TRN-3.A.2

G-1.6.2.2.1 Event Description

Cask lid dislodges from an Advanced Test Reactor 30-W spent nuclear fuel shipment due to a vehicle accident within the Advanced Test Reactor Complex and causes direct gamma exposure,

G-1.6.2.2.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the dislodged cask lid,

OR

radiological control technician confirmation of gamma radiation by radiation survey.

G-1.6.2.2.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **100 m (328 ft)** from the accident site.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.2.2.4 Offsite Protective Action Recommendations

None.

G-1.6.2.2.5 Basis. Scenario TRNSDR-2 involves a cask drop that damages the cask causing the lid to fail and exposing the fuel elements in an ATR 30-W SNF transport cask. Loss of the cask lid shielding causes a direct gamma radiation exposure pathway. An exposure rate of 1 R/hr is used to establish the PAD.

NOTE: *If a cask drop event occurs and there is loss of integrity of the cask, the direct gamma radiation exposure overwhelms the potential exposure from a radioactive material release.*

G-1.6.2.3 ATR-TRN-3.A.3

G-1.6.2.3.1 Event Description

Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes the breach of one drum and spill of some of the transuranic material outside of the drum causing a radiological release,

G-1.6.2.3.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual confirmation of the drum breach,

AND

the container is substantially intact,

AND

the spilled material is not involved in a fire.

G-1.6.2.3.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **100 m (328 ft)** from the event.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.2.3.4 Offsite Protective Action Recommendations

None.

G-1.6.2.3.5 Basis. Scenario TRANSRR-12 involves an outlier drum containing 47.1 PEC (Pu-239) assigned a DR of 0.5 (e.g., assumes that the drum lid comes off and TRU material spills out). The scenario exceeds 10% of the PAC at 100 m. Therefore, this event meets the criteria for an alert classification.

G-1.6.2.4 ATR-TRN-4.A.1

G-1.6.2.4.1 Event Description

Any Advanced Test Reactor Complex transportation accident involving nonradiological hazardous material that results in a small spill greater than 5 gal and less than or equal to 60 gal,

G-1.6.2.4.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of damage to the transport container and spillage of the contents,

AND

the incident commander recommends an evacuation/isolation distance less than or equal to **200 m (656 ft)**,

AND

protective actions are recommended for nearby buildings.

G-1.6.2.4.3 Onsite Protective Actions

Follow instructions provided by the incident commander with respect to evacuation and isolation distances.

Control nonessential vehicle and personnel access to the protective action area around the vehicle accident site.

G-1.6.2.4.4 Offsite Protective Action Recommendations

None.

G-1.6.2.4.5 ***Basis.*** The on-scene incident commander relies on the ERG to establish PADs for transportation accidents involving the spill of nonradiological hazardous material.

G-1.6.3 Site Area Emergency — Emergency Action Levels

G-1.6.3.1 ATR-TRN-3.SAE.1

G-1.6.3.1.1 Event Description

Unsuppressed fire, lasting 20 minutes after fire suppression activities began, on a tractor trailer carrying a cask containing an irradiated test train causes fuel cladding damage and a radioactive material release,

G-1.6.3.1.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the fire engulfing the cask,

OR

radiological control technician confirmation of an airborne radiological release by radiation survey.

G-1.6.3.1.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **3,000 m (9,843 ft or 1.9 mi)** from the accident site.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility and Idaho Nuclear Technology and Engineering Center evacuate.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.1.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

G-1.6.3.1.5 ***Basis.*** Scenario TRANSRR-1 involves an irradiated 200-kW test train explosion/fire. The irradiated test train is assumed to contain a maximum of 1.75E+5 Ci of mixed fission products, which are partially released due to the fire. The PAC was exceeded to a distance of 3,000 m. Therefore, this event meets the criteria for an SAE classification. A fire is considered a UOE; however, the event is classifiable if there is a radiological material release as indicated by (1) visual confirmation of the fire and (2) radiological control technician confirmation of an airborne radiological release by radiation survey near the accident scene.

G-1.6.3.2 ATR-TRN-3.SAE.2

G-1.6.3.2.1 Event Description

Unsuppressed fire, lasting 20 minutes after fire suppression activities began, on a tractor trailer carrying an Advanced Test Reactor cask containing eight Advanced Test Reactor spent nuclear fuel elements causes fuel cladding damage and a radioactive material release,

G-1.6.3.2.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the fire engulfing the cask,

OR

radiological control technician confirmation of an airborne radiological release by radiation survey.

G-1.6.3.2.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **8,075 m (26,500 ft or 5 mi)** from the accident site.

Recommend that the Idaho Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility; Idaho Nuclear Technology and Engineering Center; and Naval Reactors Facility evacuate.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.2.4 Offsite Protective Action Recommendations

NOTE: *Depending on release duration and meteorological conditions, ingestion pathway offsite protective action recommendations may be required. The Idaho National Laboratory Emergency Operations Center will determine and recommend additional protective actions and ingestion pathway advisories.*

G-1.6.3.2.5 Basis. Scenarios TRANSRR-3 and -4 involve an ATR cask explosion/fire. The ATR SNF contains aged mixed fission products, which are partially released due to the fire. The PAC was exceeded to a distance of 8,075 m. Therefore, these events meet the criteria for an SAE classification. A fire is considered a UOE; however, the event is classifiable if there is a radiological material release as indicated by (1) visual confirmation of the fire and (2) radiological control technician confirmation of an airborne radiological release by radiation survey near the accident scene.

G-1.6.3.3 ATR-TRN-3.SAE.3

G-1.6.3.3.1 Event Description

Cask lid dislodges from an irradiated test train shipment due to a vehicle accident and causes direct gamma exposure,

G-1.6.3.3.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the dislodged cask lid,

OR

radiological control technician confirmation of gamma radiation by radiation survey.

G-1.6.3.3.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **682 m (2,238 ft or 0.4 mi)** from the accident site.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.3.4 Offsite Protective Action Recommendations

None.

G-1.6.3.3.5 Basis. Scenario TRNSDR-1 involves a cask drop that damages the cask causing the lid to fail and exposing the 200-kW irradiated test train fuel element in the transport cask. Loss of the cask lid shielding causes a direct gamma radiation exposure pathway. An exposure rate of 1 R/hr is used to establish the PAD.

NOTE: *If a cask drop event occurs and there is loss of integrity of the cask, the direct gamma radiation exposure overwhelms the potential exposure from a radioactive material release.*

G-1.6.3.4 ATR-TRN-3.SAE.4

G-1.6.3.4.1 Event Description

Cask lid dislodges from an Advanced Test Reactor 634-W spent nuclear fuel shipment due to a vehicle accident within the Advanced Test Reactor Complex and causes direct gamma exposure,

G-1.6.3.4.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the dislodged cask lid,

OR

radiological control technician confirmation of gamma radiation by radiation survey.

G-1.6.3.4.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **478 m (1,568 ft or 0.3 mi)** from the accident site.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.4.4 Offsite Protective Action Recommendations

None.

G-1.6.3.4.5 Basis. Scenario TRNSDR-3 involves a cask drop that damages the cask causing the lid to fail and exposing the fuel elements in an ATR 634-W SNF transport cask. Loss of the cask lid shielding causes a direct gamma radiation exposure pathway. An exposure rate of 1 R/hr is used to establish the PAD.

NOTE: *If a cask drop event occurs and there is loss of integrity of the cask, the direct gamma radiation exposure overwhelms the potential exposure from a radioactive material release.*

G-1.6.3.5 ATR-TRN-3.SAE.5

G-1.6.3.5.1 Event Description

Radiological waste shipment to the Nevada Test Site is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes the shipment of non-Department of Transportation Type-B radiological waste containers to catch fire causing a radiological release,

G-1.6.3.5.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of the radiological waste shipment fire,

OR

radiological control technician confirmation of an airborne radiological release by radiation survey.

G-1.6.3.5.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **410 m (1,345 ft)** from the accident site.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.5.4 Offsite Protective Action Recommendations

None.

G-1.6.3.5.5 Basis. Scenario TRANSRR-11 involves a non-DOT Type-B container shipment fire. The shipment is assumed to contain a maximum of 124 Ci Pu-239 equivalent, which is partially released due to the fire. The PAC was exceeded to a distance of 410 m. Therefore, this event meets the criteria for an SAE classification.

G-1.6.3.6 ATR-TRN-3.SAE.6

G-1.6.3.6.1 Event Description

Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes a catastrophic breach of one drum and spill of most of the transuranic material outside of the drum causing a radiological release,

G-1.6.3.6.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual confirmation of the drum breach,

AND

the spilled material is not involved in a fire.

G-1.6.3.6.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **100 m (328 ft)** from the event.

Shelter all nonessential personnel out to a distance of at least **200 m (656 ft)** from the event.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.6.4 Offsite Protective Action Recommendations

None.

G-1.6.3.6.5 Basis. Scenario TRANSRR-13 involves an outlier drum containing 47.1 PEC (Pu-239) assigned a DR of 1.0 (e.g., assumes that the drum lid comes off and TRU material spills out). The scenario exceeds the PAC to a distance of 200 m. Therefore, this event meets the criteria for an SAE classification.

G-1.6.3.7 ATR-TRN-3.SAE.7

G-1.6.3.7.1 Event Description

Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that causes an explosion/deflagration of one drum or a standard waste box causing a radiological release,

G-1.6.3.7.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual confirmation of the drum or standard waste box breach,

AND

the spilled material is involved in a fire.

G-1.6.3.7.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **100 m (328 ft)** from the event.

Shelter all nonessential personnel out to a distance of at least **700 m (2,297 ft or 0.4 mi)** from the event.

Consider evacuating nonessential personnel to a distance of at least **700 m (2,297 ft or 0.4 mi)** from the event.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.7.4 Offsite Protective Action Recommendations

None.

G-1.6.3.7.5 Basis. Scenario TRANSRR-14 involves an outlier drum containing 47.1 PEC (Pu-239) and scenario TRANSRR-19 involves an SWB containing 93.6 PEC (Pu-239). Both are assigned a DR of 1.0 (e.g., assumes catastrophic failure of the drum or SWB). The scenarios exceed the PAC to a distance of 700 m. Therefore, these events meet the criteria for an SAE classification. Since the STs are nearly identical, a single EAL fits both event scenarios.

G-1.6.3.8 ATR-TRN-3.SAE.8

G-1.6.3.8.1 Event Description

Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that involves more than one transuranic material container of any type, and up to 40 drums or 10 standard waste boxes, causing a radiological release,

G-1.6.3.8.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual confirmation of the drum or standard waste box breach,

AND

the spilled material is not involved in a fire.

G-1.6.3.8.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **200 m (656 ft)** from the event.

As a precaution, shelter all nonessential personnel out to a distance of at least **1,000 m (3,280 ft)** from the event.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.8.4 Offsite Protective Action Recommendations

None.

G-1.6.3.8.5 ***Basis.*** Scenario TRANSRR-15 involves a tractor-trailer accident with a shipment of 40 drums containing 498 PEC (Pu-239) and no fire and scenario TRANSRR-17 involves an SWB containing 93.6 PEC (Pu-239) and a spill. The drum shipment is assigned a DR of 0.1 and SWB is assigned a DR of 0.5. The scenarios exceed the PAC to a distance of 200 m. Therefore, these events meet the criteria for an SAE classification. Since the STs are nearly identical, a single EAL fits both event scenarios.

G-1.6.3.9 ATR-TRN-3.SAE.9

G-1.6.3.9.1 Event Description

Shipment of transuranic material is involved in a transportation-related accident within the Advanced Test Reactor Complex that involves a shipment of up to 40 drums or 10 standard waste boxes with a drum or standard waste box breach and fire causing a radiological release,

G-1.6.3.9.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual confirmation of the drum breach,

AND

the spilled material is involved in a fire.

G-1.6.3.9.3 Onsite Protective Actions

Evacuate nonessential personnel to a distance of at least **200 m (656 ft)** from the event.

Shelter all nonessential personnel out to a distance of at least **1,100 m (3,610 ft)** from the event.

As soon as practicable, evacuate nonessential personnel to a distance of at least **1,100 m (3,610 ft)** from the event.

Control nonessential vehicle and personnel access to the evacuated area.

G-1.6.3.9.4 Offsite Protective Action Recommendations

None.

G-1.6.3.9.5 ***Basis.*** Scenario TRANSRR-16 involves a tractor-trailer accident with a shipment of 40 drums containing 498 PEC (Pu-239) and a fire. The drum shipment is assigned a DR of 0.35. The scenario exceeds the PAC to a distance of 1,100 m. Therefore, this event meets the criteria for an SAE classification.

G-1.6.3.10 ATR-TRN-4.SAE.1

G-1.6.3.10.1 Event Description

Any Advanced Test Reactor Complex transportation accident involving nonradiological hazardous material that results in a large spill equivalent to more than one 55-gal drum (greater than 60 gal) or a tanker,

G-1.6.3.10.2 Event Recognition Factors and Related Information

AS INDICATED BY

visual observation of damage to the transport container and spillage of the contents,

AND

the incident commander recommends an evacuation/isolation distance greater than **200 m (656 ft)**.

G-1.6.3.10.3 Onsite Protective Actions

Follow instructions provided by the incident commander with respect to evacuation and isolation distances.

Control nonessential vehicle and personnel access to the protective action area around the vehicle accident site.

G-1.6.3.10.4 Offsite Protective Action Recommendations

None.

G-1.6.3.10.5 Basis. The on-scene incident commander relies on the ERG to establish PADs for transportation accidents involving the spill of nonradiological hazardous material.

G-1.6.4 General Emergency — Emergency Action Levels

None.

G-1.6.5 Emergency Action Levels and Scenario Release Designators Cross Reference

Table G-27 shows the link between the EALs and the scenario release designators used as the basis for the EAL.

Table G-27. Emergency action levels and scenario release designators.

EAL No.	Scenario Release Designator
ATR-TRN-3.A.1	TRANSRR-8, -9, and -10
ATR-TRN-3.A.2	TRNSDR-2
ATR-TRN-3.A.3	TRANSRR-12
ATR-TRN-4.A.1	Small spill equivalent to one 55-gal drum (greater than 5 gal and less than or equal to 60 gal)
ATR-TRN-3.SAE.1	TRANSRR-1
ATR-TRN-3.SAE.2	TRANSRR-3 and -4
ATR-TRN-3.SAE.3	TRNSDR-1
ATR-TRN-3.SAE.4	TRNSDR-3
ATR-TRN-3.SAE.5	TRANSRR-11
ATR-TRN-3.SAE.6	TRANSRR-13
ATR-TRN-3.SAE.7	TRANSRR-14
ATR-TRN-3.SAE.8	TRANSRR-15
ATR-TRN-3.SAE.9	TRANSRR-16
ATR-TRN-4.SAE.1	Large spill equivalent to more than one 55-gal drum or a tanker (greater than 60 gal)

G-1.6.6 Emergency Planning Zone

The maximum PAD considered in this EHA for ATR Complex transportation was 8,075 m. That distance is less than the existing 16-km ATR Complex emergency planning zone. No change to the existing emergency planning zone size is recommended based on this EHA.

G-2. REFERENCES

1. DOE O 151.1C, “Comprehensive Emergency Management System,” United States Department of Energy, November 2, 2005.
2. DOE G 151.1-2, “Technical Planning Basis,” United States Department of Energy, July 11, 2007.
3. 2008 Emergency Response Guidebook, United States Department of Transportation.
4. GDE-437, “Developing and Maintaining Emergency Management Hazards Surveys,” Rev. 4, December 15, 2008.
5. Engineering Design File TRA-ATR-1832, “ATR Test Train Cask Radiological Analyses,” Rev. 0, April 23, 2002.
6. SAR-153, “Upgraded Final Safety Analysis Report for the Advanced Test Reactor,” Chapter 20, Hazard Analysis and Classification of the Facility, Table 20.2-1, Rev. 11, December 5, 2007.

7. DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities," December 1994.
8. D. R. Wenzel and B. J. Schrader, Radiological Safety Analysis Computer Program, Version 6.2, INEEL, 2002.
9. ANSI/ANS-5.10, "American National Standard for Airborne Release Fractions at Non-Reactor Nuclear Facilities," May 11, 1998.
10. Engineering Design File TRA-ATR-1300, "Exposures from a Dropped Loop Experiment," Rev. 1, March 26, 1998.
11. Engineering Design File TRA-ATR-1902, "Radiological Analysis Supporting an ATR Spent Fuel Transport Plan," April 1, 2003.
12. EDF-8027, "Structural Analysis of the ATR Spent Fuel Element Transfer Cask at INTEC," Rev. 2, August 15, 2007.
13. MCP-9391, "Waste Generator Services Certification of Waste Shipments to the Nevada Test Site," Rev. 11, September 27, 2007.
14. EDF-4711, "Radiological Dose Calculations for the Accelerated Retrieval Project Safety Analysis," Rev. 9, April 29, 2008. (**NOTE**: *There is more than one active revision to EDF-4611.*)
15. PLN-1851, "Transport Plan for the Transfer of Waste Containers Between RWMC and INTEC and RWMC and MFC," Rev. 2, May 5, 2008.
16. PLN-3055, "Transport Plan for the Transfer of Material Between MFC and the ATR," Rev. 0, TBD.
17. SAR-4, "Radioactive Waste Management Complex Safety Analysis Report," Rev. 12, July 30, 2007.

REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	01/13/2010	All	New document.

CONTENTS

H-1. GENERAL ADVANCED TEST REACTOR COMPLEX

H-1.1 Assessment Overview

This appendix documents the emergency management hazards assessment (EHA) for unclassified operational emergencies (UOEs) at the Advanced Test Reactor (ATR) Complex, as required by DOE O 151.1C.[1] Initiating events such as criticalities, explosions, fires, seismic, or other natural phenomena (high winds, tornadoes, or flooding) that may or may not release hazardous material are considered UOEs. This appendix addresses the UOE events that generally apply to the ATR Complex.

H-1.2 Facility and Process Descriptions

ATR Complex background, and mission are described elsewhere in this EHA.

H-1.3 Identification of Hazards

Radiological and nonradiological hazardous material that exceeds the threshold screening criteria is addressed elsewhere in this EHA.

H-1.4 Hazardous Material Characterization and Analyses

This subsection addresses the hazards that generally apply to all ATR Complex facilities, but are not directly tied to specific hazardous material or hazardous material inventories.

H-1.4.1 Radiological Hazardous Material

Not applicable (N/A).

H-1.4.2 Nonradiological Hazardous Material

Propane tanks of various sizes are located throughout the ATR Complex. The primary cause for concern is the potential effects of a catastrophic failure of a propane tank on nearby personnel during off-loading operations. Pertinent calculations and appropriate protective actions (PAs) are documented in EMC-1.[2]

H-1.5 Evaluation Results

H-1.5.1 Calculational Models and Bases

N/A.

H-1.5.2 Calculation Results

N/A.

H-1.6 Emergency Classes, Protective Actions, and Emergency Action Levels

H-1.6.1 Unclassified Operational Emergency — Emergency Action Levels

Based on qualitative criteria, the following UOE emergency action levels are common to most Idaho National Laboratory facilities and transportation routes.

H-1.6.1.1 ATR-ALL-1.OE.1

H-1.6.1.1.1 Event Description

A fire that causes or can reasonably be expected to cause significant structural damage to Idaho National Laboratory Site facilities,

AND

personnel injury or death is suspected or confirmed,

OR

requires personnel in nearby buildings to take shelter or evacuate.

NOTE: *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*

H-1.6.1.1.2 Event Recognition Factors and Related Information

None.

H-1.6.1.1.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Consider controlling vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.1.4 Offsite Protective Action Recommendations

None.

H-1.6.1.1.5 Basis. A fire is considered a UOE, unless the fire causes the release of hazardous material in quantities in excess of the Acute Exposure Guideline Level-2 (AEGL-2, 60 minutes), Emergency Response Planning Guideline-2 (ERPG-2), or Temporary Emergency Exposure Limit-2 (TEEL-2) concentrations.

H-1.6.1.2 ATR-ALL-2.OE.1

H-1.6.1.2.1 Event Description

Any unplanned explosion that results in known or suspected personnel injury or damage to facilities.

H-1.6.1.2.2 Event Recognition Factors and Related Information

None.

H-1.6.1.2.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Consider controlling vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.2.4 Offsite Protective Action Recommendations

None.

H-1.6.1.2.5 Basis. Any unplanned explosion that results in known or suspected personnel injury or damage to facilities is considered a UOE.

H-1.6.1.3 ATR-ALL-2.OE.2

H-1.6.1.3.1 Event Description

Any event that may reasonably be expected to cause a 1- to 40-gal propane tank to catastrophically fail and the propane tank is less than 100 m from the nearest facility fence.

H-1.6.1.3.2 Event Recognition Factors and Related Information

None.

H-1.6.1.3.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Establish a **100-m (328-ft)** exclusion zone around the affected propane tank.

H-1.6.1.3.4 Offsite Protective Action Recommendations

None.

H-1.6.1.3.5 ***Basis.*** EMC-1 overpressure calculation for a 1- to 40-gal propane tank.

H-1.6.1.4 ATR-ALL-2.OE.3

H-1.6.1.4.1 Event Description

Any event that may reasonably be expected to cause a 41- to 1,000-gal propane tank to catastrophically fail.

H-1.6.1.4.2 Event Recognition Factors and Related Information

None.

H-1.6.1.4.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Establish a **300-m (984-ft)** exclusion zone around the affected propane tank.

H-1.6.1.4.4 Offsite Protective Action Recommendations

None.

H-1.6.1.4.5 ***Basis.*** EMC-1 overpressure calculation for a 41- to 1,000-gal propane tank.

H-1.6.1.5 ATR-ALL-2.OE.4

H-1.6.1.5.1 Event Description

Any event that may reasonably be expected to cause a 1,001- to 5,000-gal propane tank to catastrophically fail.

H-1.6.1.5.2 Event Recognition Factors and Related Information

None.

H-1.6.1.5.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Establish a **500-m (1,640-ft)** exclusion zone around the affected propane tank.

H-1.6.1.5.4 Offsite Protective Action Recommendations

None.

H-1.6.1.5.5 ***Basis.*** EMC-1 overpressure calculation for a 1,001- to 5,000-gal propane tank.

H-1.6.1.6 ATR-ALL-2.OE.5

H-1.6.1.6.1 Event Description

Any event that may reasonably be expected to cause a 5,001- to 13,000-gal propane tank to catastrophically fail.

H-1.6.1.6.2 Event Recognition Factors and Related Information

None.

H-1.6.1.6.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Establish a **700-m (2,297-ft)** exclusion zone around the affected propane tank.

H-1.6.1.6.4 Offsite Protective Action Recommendations

None.

H-1.6.1.6.5 ***Basis.*** EMC-1 overpressure calculation for a 5,001- to 13,000-gal propane tank.

H-1.6.1.7 ATR-ALL-2.OE.6

H-1.6.1.7.1 Event Description

Any event that may reasonably be expected to cause a greater than 13,000-gal propane tank to catastrophically fail.

H-1.6.1.7.2 Event Recognition Factors and Related Information

None.

H-1.6.1.7.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Establish a **900-m (2,953-ft)** exclusion zone around the affected propane tank.

H-1.6.1.7.4 Offsite Protective Action Recommendations

None.

H-1.6.1.7.5 ***Basis.*** EMC-1 overpressure calculation for a greater than 13,000-gal propane tank.

H-1.6.1.8 ATR-ALL-3.OE.1

H-1.6.1.8.1 Event Description

Discovery of radioactive material contamination from past Department of Energy/National Nuclear Security Administration operations that is causing or may reasonably be expected to cause uncontrolled personnel exposures exceeding protective action criteria.

H-1.6.1.8.2 Event Recognition Factors and Related Information

None.

H-1.6.1.8.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Consider controlling vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.8.4 Offsite Protective Action Recommendations

None.

H-1.6.1.8.5 ***Basis.*** The discovery of radioactive material contamination from past Department of Energy (DOE)/National Nuclear Security Administration operations that is causing or may reasonably be expected to cause uncontrolled personnel exposures exceeding PA criteria meets the conditions for a UOE.

H-1.6.1.9 ATR-ALL-4.OE.1

H-1.6.1.9.1 Event Description

Any actual or potential release of hazardous material and a technician-level-trained hazardous material response team is required to mitigate the release,

AND

personnel injury or death is suspected or confirmed,

OR

requires personnel in nearby buildings to take shelter or evacuate.

H-1.6.1.9.2 Event Recognition Factors and Related Information

None.

H-1.6.1.9.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Consider controlling vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.9.4 Offsite Protective Action Recommendations

None.

H-1.6.1.9.5 Basis. Any actual or potential release of a hazardous material in a quantity greater than five times the reportable quantity specified for such material in 40 Code of Federal Regulations 302[3] is considered a UOE. Small spills are generally considered a nonemergency operations response.

H-1.6.1.10 ATR-ALL-4.OE.2

H-1.6.1.10.1 Event Description

Discovery of hazardous material contamination from past Department of Energy/National Nuclear Security Administration operations that is causing or may reasonably be expected to cause uncontrolled personnel exposures exceeding protective action criteria.

H-1.6.1.10.2 Event Recognition Factors and Related Information

None.

H-1.6.1.10.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Consider controlling vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.10.4 Offsite Protective Action Recommendations

None.

H-1.6.1.10.5 Basis. The discovery of hazardous material contamination from past DOE/National Nuclear Security Administration operations that is causing or may reasonably be expected to cause uncontrolled personnel exposures exceeding PA criteria meets the conditions for a UOE.

H-1.6.1.11 ATR-ALL-5.OE.1

H-1.6.1.11.1 Event Description

Any natural phenomena that may impact Idaho National Laboratory Site operations, communications, transportation, and/or health and safety of personnel such as:

- Weather extremes, to include:
  - High or low temperatures
  - High winds
  - Tornadoes
  - Lightning
  - Floods
  - Snow
- Earthquakes
- Volcanic activity.

NOTE: *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*

H-1.6.1.11.2 Event Recognition Factors and Related Information

None.

H-1.6.1.11.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by the incident commander or appropriate responsible personnel.

Consider controlling vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.11.4 Offsite Protective Action Recommendations

None.

H-1.6.1.11.5 Basis. A natural phenomena hazard such as a seismic event (earthquake) is considered a UOE, unless the event causes the release of hazardous material in quantities in excess of the AEGL-2 (60 minutes), ERPG-2, or TEEL-2 concentrations.

H-1.6.1.12 ATR-ALL-6.OE.1

H-1.6.1.12.1 Event Description

Loss of power at any Idaho National Laboratory Site facility that compromises personnel health and safety.

NOTE: *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*

H-1.6.1.12.2 Event Recognition Factors and Related Information

None.

H-1.6.1.12.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Consider controlling nonessential vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.12.4 Offsite Protective Action Recommendations

None.

H-1.6.1.12.5 Basis. Loss of offsite power to the ATR Complex for greater than 15-minutes is considered a UOE, unless the event causes the release of hazardous material in quantities in excess of the AEGL-2 (60 minutes), ERPG-2, or TEEL-2 concentrations.

H-1.6.1.13 ATR-ALL-10.OE.1

H-1.6.1.13.1 Event Description

Any unplanned criticality.

(A criticality is considered an unclassified operational emergency, unless the criticality causes the release of hazardous material in quantities exceeding screening thresholds.)

H-1.6.1.13.2 Event Recognition Factors and Related Information

None.

H-1.6.1.13.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Establish a **100-m (328-ft)** exclusion zone around the affected facility.

H-1.6.1.13.4 Offsite Protective Action Recommendations

None.

H-1.6.1.13.5 Basis. A criticality is considered a UOE, unless the criticality causes the release of hazardous material in quantities in excess of the screening thresholds.

H-1.6.1.14 ATR-ALL-11.OE.1

H-1.6.1.14.1 Event Description

Any event that damages or compromises structures or equipment that are intended to protect the health and safety of personnel,

AND

results in suspected or confirmed personnel injury or death or substantial degradation of health and safety,

OR

requires personnel in nearby buildings to take shelter or evacuate.

NOTE: *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*

H-1.6.1.14.2 Event Recognition Factors and Related Information

None.

H-1.6.1.14.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Consider controlling vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.14.4 Offsite Protective Action Recommendations

None.

H-1.6.1.14.5 Basis. Any event that causes significant damage to structures or equipment intended to protect the health and safety of personnel and results in suspected or confirmed personnel injury or death or substantial degradation of health and safety is considered a UOE.

H-1.6.1.15 ATR-ALL-12.OE.1

H-1.6.1.15.1 Event Description

External event at any Idaho National Laboratory Site facility that involves an offsite hazardous material event not associated with Department of Energy operations that is observed to have or is predicted to have an impact on a Department of Energy site such that protective actions are required for Department of Energy workers,

OR

occurrence causes or can reasonably be expected to cause significant structural damage to Department of Energy facilities that results in suspected or confirmed personnel injury or death or substantial degradation of health and safety (e.g., airplane crash, train derailment).

NOTE: *If a release of hazardous material is suspected, the appropriate section for radiological release or nonradiological hazardous material release should be referred to.*

H-1.6.1.15.2 Event Recognition Factors and Related Information

None.

H-1.6.1.15.3 Onsite Protective Actions

Follow instructions included in site- or facility-specific procedures, if applicable, and/or as directed by appropriate responsible personnel.

Consider controlling vehicle access to the affected facility/area.

Consider restricting nonessential personnel access to the affected facility/area.

H-1.6.1.15.4 Offsite Protective Action Recommendations

None.

H-1.6.1.15.5 Basis. An external event may be classified as a UOE if it involves an offsite hazardous material event not associated with DOE operations that is observed to have or is predicted to have an impact on a DOE site such that PAs are required for onsite DOE workers.

H-1.6.2 Alert — Emergency Action Levels

None.

H-1.6.3 Site Area Emergency — Emergency Action Levels

None.

H-1.6.4 General Emergency — Emergency Action Levels

None.

H-1.6.5 Emergency Action Levels and Scenario Release Designators Cross Reference

None.

H-1.6.6 Emergency Planning Zone

The consequences postulated from the events analyzed in this section do not indicate that a change to the current emergency planning zone is warranted.

H-2. REFERENCES

1. DOE O 151.1C, "Comprehensive Emergency Management System," United States Department of Energy, November 2, 2005.
2. EMC-1, "Protective Action Recommendations for Propane Tank Incidents," Rev. 0.
3. 40 CFR 302, "Designation, Reportable Quantities, and Notification."