Enclosure 1
Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report, Revision 2, July 2024
(non-proprietary)

Abilene Christian University Molten Salt Research Reactor Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

MASTER TABLE OF CONTENTS

CHAPTER 1 THE FACILITY 1-1

1.1 Introduction 1-1

1.2 Summary and Conclusions on Principal Safety Considerations 1-2

1.2.1 Consequences from the Operation and Use of the Facility 1-3

1.2.2 Inherent and Passive Safety Features 1-3

1.2.3 Design Features and Design Bases 1-4

1.2.4 Potential Accidents at the Facility 1-8

1.3 General Description of the Facility 1-8

1.3.1 Geographical Location 1-8

1.3.2 Principal Characteristics of the Site 1-8

1.3.3 Design Criteria, Operating Characteristics, and Safety Systems 1-9

1.3.4 Engineered Safety Features 1-10

1.3.5 Instrumentation, Control, and Electrical Systems 1-11

1.3.6 Cooling and Other Auxiliary Systems 1-11

1.3.7 Heating, Ventilation, and Air-Conditioning Systems 1-11

1.3.8 Radioactive Waste Management and Radiation Protection 1-12

1.3.9 Experimental Facilities and Capabilities 1-12

1.4 Shared Facilities and Equipment 1-12

1.5 Comparison with Similar Facilities 1-13

1.6 Summary of Operations 1-15

1.7 Compliance with the Nuclear Waste Policy Act of 1982 1-15

1.8 Facility Modifications and History 1-16

1.9 References 1-16

CHAPTER 2 SITE CHARACTERISTICS 2-1

2.1 Geography and Demography 2-1

2.1.1 Site Location and Description 2-1

2.1.2 Population Distribution 2-3

2.1.3 References 2-7

2.2 Nearby Industrial, Transportation, and Military Facilities 2-40

2.2.1 Locations and Routes 2-40

2.2.2 Air Traffic 2-43

2.2.3 Analysis of Potential Accidents at Facilities 2-47

MASTER TABLE OF CONTENTS

2.2.4 References . . . 2-48
2.3 Meteorology . . . 2-60
2.3.1 General and Local Climate . . . 2-60
2.3.2 Site Meteorology . . . 2-62
2.3.3 References . . . 2-62
2.4 Hydrology . . . 2-77
2.4.1 Drainage and Floodways . . . 2-77
2.4.2 Groundwater . . . 2-78
2.4.3 References . . . 2-78
2.5 Geology, Seismology, and Geotechnical Engineering . . . 2-89
2.5.1 Regional Geology . . . 2-89
2.5.2 Site Geology . . . 2-89
2.5.3 Seismicity . . . 2-91
2.5.4 Maximum Earthquake Potential . . . 2-92
2.5.5 Vibratory Ground Motion . . . 2-92
2.5.6 Surface Faulting . . . 2-93
2.5.7 Liquefaction Potential . . . 2-93
2.5.8 References . . . 2-94

CHAPTER 3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS . . . 3-1
3.1 Design Criteria . . . 3-1
3.1.1 Introduction . . . 3-1
3.1.2 Molten Salt Research Reactor Design Criteria . . . 3-16
3.1.3 Nuclear Regulatory Commission Guidance Documents . . . 3-31
3.1.4 Codes and Standards Used by the MSRR . . . 3-31
3.1.5 References . . . 3-33
3.2 Meteorological Damage . . . 3-38
3.2.1 Normal Wind Loads . . . 3-38
3.2.2 Tornado Loading . . . 3-40
3.2.3 Hurricane Loading . . . 3-41
3.2.4 Precipitation Loads . . . 3-41
3.3 Water Damage . . . 3-43
3.3.1 Flood Protection . . . 3-43

MASTER TABLE OF CONTENTS

3.3.2 References . . . 3-44
3.4 Seismic Damage . . . 3-45
3.4.1 Existing Multiuse Facility . . . 3-45
3.4.2 Seismic Design for Safety-Related Structures, Systems, and Components . . . 3-45
3.4.3 Seismic Instrumentation . . . 3-48
3.4.4 References . . . 3-48
3.5 Systems and Components . . . 3-60
3.5.1 General Design Basis Information . . . 3-60
3.5.2 Classification of Structures, Systems, and Components . . . 3-65
Appendix 3A City of Abilene, TX Building Permit for SERC . . . 3A-1

CHAPTER 4 MOLTEN SALT RESEARCH REACTOR DESCRIPTION . . . 4-1
4.1 Summary Description . . . 4-1
4.1.1 Reactor System . . . 4-2
4.2 Active Reactor Core . . . 4-6
4.2.1 Reactor Fuel . . . 4-6
4.2.2 Control Elements . . . 4-10
4.2.3 Neutron Moderator and Reflector . . . 4-13
4.2.4 Neutron Startup Source . . . 4-14
4.2.5 Core Support Structure . . . 4-14
4.3 Vessel . . . 4-21
4.3.1 Description . . . 4-21
4.3.2 Design Bases . . . 4-21
4.3.3 Reactor System Structural Material . . . 4-22
4.3.4 Fuel Salt Chemical Attack . . . 4-22
4.3.5 Radiation Damage to Reactor System . . . 4-23
4.3.6 Stainless Steel Oxidation and Creep . . . 4-23
4.3.7 Pressure Effects . . . 4-24
4.3.8 Thermal Design Limits . . . 4-24
4.3.9 Reactor System Integrity Monitoring . . . 4-24
4.3.10 Fuel Salt Leakage . . . 4-24
4.3.11 Description of the Reactor Access Vessel . . . 4-25

MASTER TABLE OF CONTENTS

4.4 Biological Shield 4-28
4.4.1 Description of Biological Shield 4-28
4.4.2 Design Bases 4-28
4.4.3 Functional Requirements 4-28
4.4.4 Design Methods 4-29
4.4.5 Design Approach 4-29
4.4.6 Internal Shield 4-30
4.4.7 Biological Shield Inner Layer (Sacrificial Shield) 4-30
4.4.8 Biological Shield Outer Layer (Systems Pit Floor and Walls, Top Plug) 4-30
4.4.9 Additional Shielding 4-30
4.4.10 Structural Considerations 4-31
4.4.11 Radiation Zones 4-31
4.4.12 Calculated Dose Rates and Other Relevant Shielding Parameters 4-31
4.4.13 Uncertainties in Shielding Results 4-32
4.4.14 Environmental Conditions 4-32
4.4.15 Soil Activation 4-32
4.4.16 Air Activation 4-32
4.5 Nuclear Design 4-35
4.5.1 Description of Nuclear Design 4-35
4.5.2 Normal Operating Conditions 4-37
4.5.3 Active Reactor Core Physics Parameters 4-40
4.5.4 Operating Limits 4-41
4.6 Thermal Hydraulic Design 4-51
4.6.1 Heat Removal Systems 4-51
4.6.2 Thermal-Hydraulic Methodology 4-52
4.7 Gas Management System 4-54
4.8 References 4-55

CHAPTER 5 MOLTEN SALT REACTOR COOLING SYSTEMS 5-1
5.1 Summary Description 5-1
5.2 Fuel System Boundary and Fuel Salt Heat Transport 5-2
5.2.1 Primary Cooling System 5-3
5.2.2 Secondary Cooling System 5-3

MASTER TABLE OF CONTENTS

5.2.3 Auxiliary Systems . . . 5-3
5.2.4 Instrumentation and Control Systems . . . 5-5
5.3 Cooling Systems . . . 5-5
5.3.1 Primary Cooling System . . . 5-5
5.3.2 Secondary Cooling System . . . 5-7
5.4 Fuel Salt Cleanup System . . . 5-9
5.5 Salt Makeup System . . . 5-9
5.5.1 Fuel Salt Makeup System . . . 5-10
5.5.2 Secondary Cooling Makeup System . . . 5-10
5.5.3 Auxiliary Cooling Systems . . . 5-10
5.5.4 Design Basis for the Drain Tank . . . 5-10
5.6 REFERENCES . . . 5-10

CHAPTER 6 ENGINEERED SAFETY FEATURES . . . 6-1
6.1 Summary Description . . . 6-1
6.2 Detailed Descriptions . . . 6-3
6.2.1 Confinement . . . 6-3
6.2.2 Containment . . . 6-5
6.2.3 Emergency Cooling System . . . 6-9
6.2.4 Reactor Thermal Management System . . . 6-9
6.3 Compliance with Design Criteria . . . 6-11
6.4 References . . . 6-12

CHAPTER 7 INSTRUMENTATION AND CONTROL SYSTEMS . . . 7-1
7.1 Summary Description . . . 7-1
7.1.1 Calibration of Trips, Interlocks, and Annunciation . . . 7-2
7.1.2 Distributed Control Systems . . . 7-2
7.1.3 Reactor Protection System and Engineered Safety Features Actuation System . . . 7-2
7.1.4 Distributed Control System and Control Room . . . 7-3
7.1.5 Radiation and Environmental Monitoring System . . . 7-4
7.2 Design of Instrumentation and Control Systems . . . 7-4
7.2.1 System Description . . . 7-6
7.2.2 System Performance Analysis . . . 7-6

MASTER TABLE OF CONTENTS

7.2.3 Conclusion . . . 7-6

7.3 Distributed Control System . . . 7-6

7.3.1 Design Criteria and Bases . . . 7-7

7.3.2 Control Rod Drives . . . 7-8

7.3.3 Nuclear Instrumentation . . . 7-8

7.3.4 Salt Level and Cover Gas Management System . . . 7-9

7.3.5 Auxiliary Heat Removal Cooling Air Process Monitoring and Control . . . 7-9

7.3.6 Primary Heat Removal Cooling Air Process Monitoring and Control . . . 7-9

7.3.7 Salt Pump Monitoring and Control . . . 7-9

7.3.8 System Heaters Temperature Controls and Monitors . . . 7-10

7.3.9 System Evaluation . . . 7-10

7.4 Reactor Protection System . . . 7-10

7.4.1 Design Criteria and Bases . . . 7-11

7.4.2 Reactor Trip . . . 7-15

7.4.3 RPS Trip Set Points . . . 7-17

7.4.4 System Trips . . . 7-19

7.4.5 System Evaluation . . . 7-20

7.5 Engineered Safety Features Actuation System . . . 7-20

7.5.1 Design Criteria and Bases . . . 7-21

7.5.2 ESFAS Initiation . . . 7-24

7.5.3 Initiation Limits and Signals . . . 7-26

7.5.4 System Evaluation . . . 7-27

7.6 Human-Machine Interface . . . 7-27

7.6.1 Design Criteria and Bases . . . 7-28

7.6.2 Control Room . . . 7-28

7.6.3 Control Console and Human Machine Interface . . . 7-29

7.6.4 Conclusions . . . 7-30

7.7 Radiation and Environmental Monitoring System . . . 7-30

7.7.1 Design Criteria and Bases . . . 7-30

7.7.2 Facility Sensor Stations . . . 7-32

7.7.3 Emplacement Detectors . . . 7-33

7.8 References . . . 7-33

MASTER TABLE OF CONTENTS

CHAPTER 8 ELECTRICAL POWER SYSTEMS 8-1

8.1 Normal Electrical Power System 8-1

8.2 Emergency Electrical Power Systems 8-5

8.2.1 Backup Electrical Power Systems 8-5

CHAPTER 9 AUXILIARY SYSTEMS 9-1

9.1 Heating, Ventilation, and Air Conditioning Systems 9-1

9.1.1 Design Bases 9-1

9.1.2 System Description 9-1

9.1.3 Operational Analysis and Safety Function 9-4

9.1.4 Instrumentation and Controls 9-5

9.1.5 Technical Specifications, Testing, and Inspection 9-5

9.2 Handling and Storage of Reactor Fuel 9-5

9.2.1 Design Basis 9-5

9.2.2 System Description 9-6

9.2.3 Operational Analyses and Safety Function 9-7

9.2.4 Instrumentation and Controls Requirements 9-9

9.3 Fire Protection Systems and Programs 9-9

9.3.1 Fire Protection Programs 9-9

9.3.2 Design Basis 9-9

9.3.3 System Description 9-10

9.3.4 Operational Analysis and Safety Function 9-12

9.3.5 Instrumentation and Control Requirements 9-12

9.3.6 Technical Specifications, Testing, and Inspection 9-12

9.4 Communication Systems 9-12

9.4.1 Design Basis 9-12

9.4.2 System Description 9-12

9.4.3 Operational Analyses and safety function 9-13

9.4.4 Instrumentation and Controls 9-13

9.4.5 Technical Specifications, Testing, and Inspection 9-13

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material 9-13

9.5.1 Special Nuclear Material 9-13

MASTER TABLE OF CONTENTS

9.5.2 Byproduct Material 9-14
9.6 Gas Management System 9-15
9.6.1 Design Basis 9-15
9.6.2 System Description 9-16
9.6.3 Operational Analysis and Safety Function 9-18
9.6.4 Instrumentation and Control Requirements 9-19
9.6.5 Technical Specifications, Testing, and Inspection 9-19
9.7 Cooling Systems 9-20
9.7.1 Auxiliary Heat Removal System 9-20
9.7.2 Additional Cooling Systems 9-22
9.8 Other Auxiliary Systems 9-22
9.8.1 Compressed Air, Vacuum, and Inert Gas Supply 9-22
9.8.2 Cranes 9-23
9.9 References 9-24

CHAPTER 10 EXPERIMENTAL FACILITIES AND UTILIZATION 10-1
10.1 Summary Description 10-1
10.2 Experimental Facilities 10-1
10.2.1 Salt Sampling and Measurement Experimental System 10-2
10.2.2 Gas sampling and Measurement Experimental System 10-5
10.2.3 Helium Bubble Generation and Removal Systems 10-8
10.2.4 Radiochemistry Laboratory 10-10
10.2.5 Scientific Surveillance Facilities 10-10
10.2.6 Probable Subjects of Technical Specifications 10-12
10.3 Experiment Review 10-13
10.4 References 10-13

CHAPTER 11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT 11-1
11.1 Radiation Protection 11-1
11.1.1 Radiation Sources 11-1
11.1.2 Radiation Protection Program 11-1
11.1.3 As Low As Reasonably Achievable Program 11-2
11.1.4 Radiation Monitoring and Surveying 11-2

MASTER TABLE OF CONTENTS

11.1.5 Radiation Exposure Control and Dosimetry 11-3
11.1.6 Contamination Control 11-4
11.1.7 Environmental Monitoring 11-4
11.1.8 References 11-5
11.2 Radioactive Waste Management 11-5
11.2.1 Radioactive Waste Management Program 11-5
11.2.2 Radioactive Waste Controls 11-5
11.2.3 Release of Radioactive Waste 11-6
11.2.4 Estimated Quantities of Waste Generation 11-6
11.3 Respiratory Protection Program 11-6

CHAPTER 12 CONDUCT OF OPERATIONS 12-1
12.1 Organization 12-1
12.1.1 Structure 12-1
12.1.2 Responsibility 12-1
12.1.3 Staffing 12-2
12.1.4 Selection and Training of Personnel 12-3
12.1.5 Radiation Safety 12-3
12.2 Review and Audit Activities 12-4
12.2.1 Composition and Qualification 12-4
12.2.2 Charter and Rules 12-4
12.2.3 Review/Audit Function 12-4
12.2.4 Audit Function 12-5
12.3 Procedures 12-5
12.3.1 Experiment Review and Approval 12-5
12.4 Required Actions 12-6
12.5 Reports 12-6
12.6 Records 12-6
12.7 Emergency Planning 12-6
12.8 Security Planning 12-6
12.9 Quality Assurance 12-7
12.10 Reactor Operator Training and Requalification 12-7
12.11 Startup Plan 12-7

MASTER TABLE OF CONTENTS

12.12 Material Control and Accounting Program 12-7
12.13 References 12-7
Appendix 12A ACU Research Reactor Facility Preliminary Emergency Plan . . . 12A-1

CHAPTER 13 ACCIDENT ANALYSES 13-1
13.1 Accident-Initiating Events and Scenarios 13-2
13.1.1 Maximum Hypothetical Accident 13-6
13.1.2 Reduction in Fuel Salt Inventory from a Barrier Failure 13-16
13.1.3 Increase in Fuel Salt Inventory 13-17
13.1.4 Reduction in Cooling 13-18
13.1.5 Reactivity and Power Distribution Anomalies 13-22
13.1.6 Mishandling or Malfunction of Fuel 13-37
13.1.7 Experiment Malfunction 13-37
13.1.8 External Events 13-37
13.1.9 Mishandling or Malfunction of Equipment 13-38
13.1.10 Loss of Normal Electrical Power 13-39
13.2 Accident Analysis and Determination of Consequences 13-39
13.3 Summary and Conclusions 13-42
13.4 References 13-42

CHAPTER 14 TECHNICAL SPECIFICATIONS 14-1
14.1 Probable Subjects of Technical Specifications for the Facility 14-1
14.1.1 Definitions 14-1
14.2 Safety Limits and Limiting Safety System Settings 14-2
14.2.1 Safety Limits 14-2
14.2.2 Limiting Safety System Settings 14-3
14.3 Limiting Conditions for Operations 14-3
14.3.1 Fuel Salt and Fuel System Boundary Parameters 14-4
14.3.2 Reactor Control and Safety Systems 14-5
14.3.3 Primary Cooling and Heat Dissipation Systems 14-6
14.3.4 Functional Containment 14-7
14.3.5 Ventilation Systems 14-7
14.3.6 Radiation Monitoring Systems and Effluents 14-7

MASTER TABLE OF CONTENTS

14.3.7 Experiments 14-9
14.3.8 Facility-Specific Limiting Conditions for Operations - Fuel Handling System 14-9
14.4 Surveillance Requirements 14-9
14.5 Design Features 14-10
14.6 Administrative Controls 14-10
14.7 References 14-10

CHAPTER 15 FINANCIAL QUALIFICATIONS 15-1
15.1 Financial Ability to Construct a Non-Power Reactor 15-1
15.1.1 Construction Costs and Fuel Cycle Costs 15-1
15.1.2 Sources of Funds 15-3
15.2 Financial Ability to Operate a Non-Power Reactor 15-4
15.3 Financial Ability to Decommission the Facility 15-5
15.4 Foreign Ownership, Control, or Domination (FOCD) 15-5
15.5 Nuclear Insurance and Indemnity 15-5
15.6 References 15-5
Appendix 15A 15A-1
Appendix 15B 15B-1

CHAPTER 16 OTHER LICENSE CONSIDERATIONS 16-1
16.1 Prior Use of Reactor Components 16-1
16.2 Medical Use of the Facility 16-2

CHAPTER 17 DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS 17-1
17.1 Decommissioning 17-1
17.2 Possession-Only License Amendment 17-1

CHAPTER 18 HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS 18-1

CHAPTER 19 ENVIRONMENTAL REVIEW 19-1
19.1 Introduction of the Environmental Review 19-1
19.1.1 Purpose and Need for the Proposed Action 19-1
19.1.2 Regulatory Provisions, Permits, and Required Consultations 19-2

MASTER TABLE OF CONTENTS

19.2 Proposed Action 19-3
19.2.1 Site Location and Layout 19-4
19.2.2 Non-Power Reactor 19-7
19.2.3 Water Consumption and Treatment 19-10
19.2.4 Cooling and Heating Dissipation Systems 19-10
19.2.5 Waste Systems 19-11
19.3 Description of the Affected Environment 19-14
19.3.1 Land Use 19-14
19.3.2 Visual Resources 19-20
19.3.3 Climatology, Air Quality, and Noise 19-20
19.3.4 Geologic Environment 19-25
19.3.5 Water Resources 19-27
19.3.6 Ecological Resources 19-36
19.3.7 Historic and Cultural Resources 19-37
19.3.8 Socioeconomics 19-39
19.3.9 Human Health 19-51
19.4 Impacts of Proposed Construction, Operations, and Decommissioning . . . 19-55
19.4.1 Land Use and Visual Resources 19-55
19.4.2 Visual Resources 19-56
19.4.3 Air Quality and Noise 19-57
19.4.4 Geologic Environment 19-58
19.4.5 Water Resources 19-58
19.4.6 Ecological Resources 19-59
19.4.7 Historic and Cultural Resources 19-60
19.4.8 Socioeconomics 19-60
19.4.9 Human Health 19-62
19.4.10 Waste Management 19-75
19.4.11 Transportation 19-77
19.4.12 Postulated Accidents 19-77
19.4.13 Environmental Justice 19-82
19.4.14 Cumulative Effects 19-85
19.5 Alternatives 19-99
19.5.1 No-Action Alternative 19-99

MASTER TABLE OF CONTENTS

19.5.2 Reasonable Alternatives 19-100
19.5.3 Cost-Benefit of the Alternatives 19-103
19.5.4 Comparison of the Potential Environmental Impacts 19-104
19.6 Conclusions 19-105
19.6.1 Unavoidable Adverse Environmental Impacts 19-105
19.6.2 Relationship between Short-Term Uses and Long-Term Productivity of the Environment 19-106
19.6.3 Irreversible and Irretrievable Commitments of Resources 19-107

MASTER LIST OF TABLES

Table 2.1-1 Abilene Area Populations and Projections . . . 2-10
Table 2.1-2 Number of Jobs per Industry in Abilene . . . 2-10
Table 2.2-1 Abilene Regional Airport Operations Predictions for 2022 . . . 2-50
Table 2.2-2 Annual Aircraft Operations for Dyess Air Force Base . . . 2-50
Table 2.3-1 Heaviest Daily Snowfalls in Abilene from 1950 to Present . . . 2-64
Table 2.3-2 Abilene Weather Records, 1948-2022 . . . 2-64
Table 2.3-3 Records of Most Consecutive Days of Weather Phenomena in Abilene, Texas . . . 2-65
Table 2.3-4 Maximum Wind Speeds and Reported Damage Events . . . 2-65
Table 2.3-5 Tornado Probabilities . . . 2-66
Table 2.3-6 Number of Hail Events by Size, Taylor County (1/1/1960 to 6/30/2022) . . . 2-66
Table 2.3-7 Ten Largest Recorded Hail Sizes, Taylor County (1/1/1960 to 6/30/2022) . . . 2-67
Table 2.3-8 Number of Hail Events Per Year in Taylor County, 1960-July 2022 . . . 2-67
Table 2.3-9 Taylor County Thunderstorm Wind Events (1/1/1960 to 6/30/2022) . . . 2-68
Table 2.5-1 Recommended Maximum Bearing and Skin Friction . . . 2-91
Table 2.5-2 L-Pile Information . . . 2-91
Table 2.5-3 Earthquakes of Magnitude 3 or Greater within 125 Miles (200 km) of the Molten Salt Research Reactor Site . . . 2-96
Table 3.1-1 Cross Reference to Preliminary Safety Analysis Report Sections . . . 3-34
Table 3.1-2 Cross Reference to Nuclear Regulatory Commission Guidance Documents . . . 3-36
Table 3.4-1 Safety and Seismic Classification of Structures, Systems, and Components . . . 3-49
Table 3.4-2 MSRR Horizontal and Vertical SSE . . . 3-54
Table 4.1-1 Reactor Parameters . . . 4-5
Table 4.2-1 Graphite Dimensions at 600 Degrees Celsius . . . 4-20
Table 4.2-2 Parameters Describing the Grid Plate at 600 Degrees Celsius . . . 4-20
Table 4.5-1 Reactivity Loss Stemming from the Delayed Neutron Precursors Outside the Reactor Vessel . . . 4-50
Table 4.5-2 Reactivity Coefficients Calculated in MCNP with 1 Sigma Stochastic Uncertainty . . . 4-50
Table 4.5-3 Reactivity Coefficients Calculated in SCALE . . . 4-50
Table 5.2-1 Parameter Limitations . . . 5-4
Table 5.3-1 Radiator Design Parameters . . . 5-7

MASTER LIST OF TABLES

Table 7.4-1 Deleted. . . . 7-11
Table 7.4-2 Reactor Trip Set Points 7-18
Table 7.4-3 System Trip Channels 7-19
Table 7.5-1 ESFAS Initiation. . . . 7-26
Table 10.2-1 Gamma Ray and Neutron Dose Rate at 30 cm from Sample for 1 Megawatt Operation. . . . 10-3
Table 10.2-2 Anticipated Instruments for Salt Sampling and Measurement Experimental System 10-4
Table 10.2-3 Fraction of Derived Air Concentration for Gas Sample Release in Research Bay 10-6
Table 10.2-4 Anticipated Instruments for Gas Sampling and Measurement. . . . 10-7
Table 10.2-5 Molten Salt Research Reactor Scientific Surveillance Layer Scope 10-11
Table 10.2-6 Monitoring Capabilities and Instruments 10-12
Table 12A-1 Definitions 12-2
Table 12A-2 Succession Plan for Emergency Director and Radiation Safety Officer 12-6
Table 13.1-1 Fractional Weighting of Fission Power in Each Core Volume and Temperature Reactivity Coefficients 13-4
Table 13.1-2 Maximum Hypothetical Accident Assumptions 13-7
Table 13.1-3 Initial Nuclide Inventories Assumed for the MHA. . . . 13-8
Table 13.1-4 Calculated Values of χ/Q 13-12
Table 13.1-5 Summary of Analyzed Reactivity Insertions. . . . 13-23
Table 13.2-1 Summary of Accident Scenarios Examined. . . . 13-40
Table 15.1-1 Molten Salt Research Reactor Order-of-Magnitude Cost Estimates 15-4
Table 15.1-2 Molten Salt Research Reactor Overnight Cost Estimate 15-4
Table 19.1-1 Permits and Approvals Required for Construction and Operation of the Molten Salt Research Reactor 19-2
Table 19.1-2 Permits and Approvals Required for Construction of the Science and Engineering Research Center 19-3
Table 19.3-1 National Ambient Air Quality Standards. . . . 19-23
Table 19.3-2 Demographic Profiles. . . . 19-41
Table 19.3-3 Median Family Income and Per Capita Income (2015-2019) 19-42
Table 19.3-4 People Living Below U.S. Census Poverty Thresholds 19-43
Table 19.3-5 2019 U.S. Federal Poverty Thresholds for Different Family Sizes. . . . 19-43
Table 19.3-6 Taylor County Total Housing Units and Vacancy Rates 19-43

MASTER LIST OF TABLES

Table 19.3-7 Ten Largest Employers in the City of Abilene 19-44
Table 19.3-8 School Districts in the Region of Influence and Their Tax Levies 19-44
Table 19.4-1 Chemicals Stored and Used during Operation 19-71
Table 19.4-2 Potential Occupational Hazards. 19-71
Table 19.4-3 Radiation Sources and Locations in the Facility 19-72
Table 19.4-4 Anticipated Radioactive Gaseous Effluent Production and Emissions 19-72
Table 19.4-5 Abilene, Texas Demographic Data (U.S. Census Bureau QuickFacts: Abilene, Texas) 19-84
Table 19.4-6 Abilene Poverty Statistics (LiveStories U.S. Census Bureau QuickFacts: Abilene, Texas) 19-84
Table 19.4-7 Past, Present, and Reasonably Foreseeable Projects and Other Actions Considered in the Cumulative Effects Analysis 19-96
Table 19.4-8 Cumulative Impacts on Environmental Resources, Including the Impacts of the Proposed Project 19-99
Table 19.5-1 Candidate Site Ranking for Environmental Factors 19-104
Table 19.5-2 Candidate Site Ranking for Financial Impact. 19-104
Table 19.5-3 Candidate Site Ranking for Mission Impact 19-104

MASTER LIST OF FIGURES

Figure 2.1-1 Borders and Major Cities in 200-mi (322-km) Radius from the Reactor Site . . . 2-11
Figure 2.1-2 Location of Abilene . . . 2-12
Figure 2.1-3 Molten Salt Research Reactor Location in Abilene . . . 2-13
Figure 2.1-4 Streams, Rivers, and Lakes within Several Miles of the Reactor Site . . . 2-14
Figure 2.1-5 Topography of the Abilene Area . . . 2-15
Figure 2.1-6 A 1-mile (1.5 km) Radius around the Molten Salt Research Reactor Site . . . 2-16
Figure 2.1-7 Abilene Christian University Main Campus Boundary . . . 2-17
Figure 2.1-8 Molten Salt Research Reactor Site Layout . . . 2-18
Figure 2.1-9 Science and Engineering Research Center First-floor Layout and Utilization . . . 2-19
Figure 2.1-10 Historic and Projected Population Growth . . . 2-20
Figure 2.1-11 Nearest Resident Population Distribution - 2020 . . . 2-21
Figure 2.1-12 Furthest Resident Population Distribution - 2020 . . . 2-22
Figure 2.1-13 Resident Population Bands - 2020 . . . 2-23
Figure 2.1-14 Nearest Resident Population Distribution - 2022 . . . 2-24
Figure 2.1-15 Furthest Resident Population Distribution - 2022 . . . 2-25
Figure 2.1-16 Resident Population Bands - 2022 . . . 2-26
Figure 2.1-17 Nearest Resident Population Distribution - 2027 . . . 2-27
Figure 2.1-18 Furthest Resident Population Distribution - 2027 . . . 2-28
Figure 2.1-19 Resident Population Bands - 2027 . . . 2-29
Figure 2.1-20 Nearest Resident Population Distribution - 2042 . . . 2-30
Figure 2.1-21 Furthest resident population distribution - 2042 . . . 2-31
Figure 2.1-22 Resident Population Bands - 2042 . . . 2-32
Figure 2.1-23 Historic Population Growth in Abilene and Taylor County . . . 2-33
Figure 2.1-24 Abilene Population Density . . . 2-34
Figure 2.1-25 Location of Universities in Abilene . . . 2-35
Figure 2.1-26 Abilene Hotel Locations and Occupancy . . . 2-36
Figure 2.1-27 Recreational Vehicle Parks and Campgrounds near the Reactor Site . . . 2-37
Figure 2.1-28 Locations of Hospitals in Abilene . . . 2-38
Figure 2.1-29 Concentrations of Employment across Abilene . . . 2-39
Figure 2.2-1 Industrial and Transportation Facilities Near the Site . . . 2-51

MASTER LIST OF FIGURES

Figure 2.2-2 Airfields within 10 Miles of the Site 2-52
Figure 2.2-3 Highways, Rail Lines, Dyess Air Force Base, and Abilene Regional Airport 2-53
Figure 2.2-4 Pipelines in the Abilene Area 2-54
Figure 2.2-5 Dyess Air Force Base Arrival Flight Patterns 2-55
Figure 2.2-6 Dyess Air Force Base Departure Flight Patterns 2-56
Figure 2.2-7 Dyess Air Force Base Closed Pattern Flight Patterns 2-57
Figure 2.2-8 Abilene Regional Airport Layout 2-58
Figure 2.2-9 Dyess Air Force Base Layout 2-59
Figure 2.3-2 Average Monthly Temperature Distribution for Abilene from 1970-2021 2-69
Figure 2.3-1 Rainfall Distribution for Abilene from 1948-2021 2-69
Figure 2.3-3 Maximum and Minimum Monthly Temperatures for Abilene from 1948-2021 2-70
Figure 2.3-4 Distribution of Monthly Relative Humidity in the region of Abilene between 1970 and 2021 2-70
Figure 2.3-5 Wind Speed, Direction, and Frequency Measured at Abilene Regional Airport 2-71
Figure 2.3-6 Number of Severe Weather Events within 50 miles of Abilene 2-72
Figure 2.3-7 Design-basis Hurricane Wind Speeds from Regulatory Guide 1.221 2-73
Figure 2.3-8 Hurricane Events Tracked within 50 nmi of the Reactor Site 2-74
Figure 2.3-9 Tornado Magnitude and Distance from Abilene, 1950-2010 2-75
Figure 2.3-10 Map of Tornado Pathways in Abilene from 1950-2013 2-76
Figure 2.4-1 Texas River Basins 2-80
Figure 2.4-2 Topographical Map of Brazos River Basin 2-81
Figure 2.4-3 Streams, Rivers, and Lakes within Several Miles of the Reactor Site 2-82
Figure 2.4-4 Federal Emergency Management Agency-defined Floodways 2-83
Figure 2.4-5 Topography of Abilene Area 2-84
Figure 2.4-6 Topography of Molten Salt Research Reactor Site Area 2-85
Figure 2.4-7 Major Aquifers of Texas 2-86
Figure 2.4-8 Minor Aquifers of Texas 2-87
Figure 2.4-9 Aquifers and Wells Near the Site 2-88
Figure 2.5-1 Geologic Map of Texas 2-98
Figure 2.5-2 Geologic Map of Taylor County 2-99
Figure 2.5-3 Geologic Map of Abilene Area 2-100

MASTER LIST OF FIGURES

Figure 2.5-4 Areas Prone to Collapse Sinkholes 2-101
Figure 2.5-5 Topography of the Abilene Area 2-102
Figure 2.5-6 Boring Log 1. 2-103
Figure 2.5-7 Boring Log 2. 2-104
Figure 2.5-8 Boring Log 3. 2-105
Figure 2.5-9 Boring Log 4. 2-106
Figure 2.5-10 Boring Log 5. 2-107
Figure 2.5-11 Boring Log 6. 2-108
Figure 2.5-12 Boring Log 7. 2-109
Figure 2.5-13 Boring Log 8-1 2-110
Figure 2.5-14 Boring Log 8-2 2-111
Figure 2.5-15 Boring Log 9-1 2-112
Figure 2.5-16 Boring Log 9-2 2-113
Figure 2.5-17 Boring Log Summary 2-114
Figure 2.5-18 Seismic Activity Near the Molten Salt Research Reactor Site 2-115
Figure 2.5-19 Causes and Locations of Earthquakes in Texas 2-116
Figure 2.5-20 Injection and Disposal Wells Near the Molten Salt Research Reactor Site. 2-117
Figure 2.5-21 2014 Seismic Hazard Map of Texas 2-118
Figure 2.5-22 Seismic Hazard Curves and Uniform Hazard Response Spectrum. 2-119
Figure 2.5-23 Underground Faults Near Abilene 2-120
Figure 2.5-24 Meers Fault in Oklahoma. 2-121
Figure 2.5-25 Fault Lines in Texas. 2-122
Figure 2.5-26 ASCE 7 MCE_R Hazard Comparison 2-123
Figure 3.1-1 The Science and Engineering Research Center that will House the Nuclear Energy eXperimental Testing Laboratory 3-4
Figure 3.1-2 Cross Section View of Science and Engineering Research Center. 3-5
Figure 3.1-3 Safety-related Structures, Systems, and Components in Systems Pit 3-6
Figure 3.1-4 Science and Engineering Research Center First Floor Layout and Utilization 3-7
Figure 3.1-5 Cross Section View of SERC and Research Bay (From Southeast) 3-8
Figure 3.1-6 Cross Section View of SERC and Research Bay (From Southwest). 3-9
Figure 3.1-7 Exterior View of SERC and Research Bay (From Northwest) 3-10

MASTER LIST OF FIGURES

Figure 3.1-8 Exterior View of SERC and Research Bay (From Northeast) 3-11
Figure 3.1-9 Exterior View of SERC and Research Bay (From Southeast) 3-12
Figure 3.1-10 Half Cross-section of the Systems pit, Slab, and the Fill Materials 3-13
Figure 3.1-11 SERC Pier and Tilt-up Panel Plan Showing Locations of Piers Relative to the Pit and Slab 3-14
Figure 3.1-12 SERC Pier and Tilt-up Panel Plan Notes 3-15
Figure 3.4-1 Systems Pit Cross Section (Rebar Dimensions Nominal/Typical) 3-55
Figure 3.4-2 Research Bay Dimensions (Nominal/Typical) 3-56
Figure 3.4-3 Research Bay Cross Section (Dimensions Nominal/Typical) 3-57
Figure 3.4-4 MSRR Horizontal Safe Shutdown Earthquake 3-58
Figure 3.4-5 MSRR Vertical Safe Shutdown Earthquake 3-59
Figure 3.5-1 Reactor Cell Diagram 3-65
Figure 4.1-1 Section of the Systems Pit with Major Components 4-3
Figure 4.1-2 Reactor System Components within the Reactor Enclosure 4-4
Figure 4.1-3 Reactor System Major Internals and Approximate Dimensions 4-5
Figure 4.2-1 Deleted 4-15
Figure 4.2-2 Locations of the Three Control Rod Thimbles Shown in Yellow 4-16
Figure 4.2-3 Single Hexagonal Graphite Block (Left) and Channels Formed by Lattice of Blocks (Right) 4-17
Figure 4.2-4 Hexagonal Lattice of Graphite Blocks Showing Channels at Block Corners 4-18
Figure 4.2-5 Lower Core Grid Plate 4-19
Figure 4.3-1 Reactor Access Vessel 4-27
Figure 4.4-1 Configuration and Thickness of Biological Shield Layers 4-33
Figure 4.4-2 Vertical Cuts (XZ and YZ) of Bay Building and Systems Pit 4-34
Figure 4.5-1 Variation in Pressure Coefficient of Reactivity with Respect to Void Fraction 4-46
Figure 4.5-2 Change in K_{eff} with Respect to Addition or Removal of UF_4 from Reactor 4-47
Figure 4.5-3 Loss of Reactivity Caused by Fuel Burnup 4-47
Figure 4.5-4 Axial Power Profile through the Entire Reactor Vessel 4-48
Figure 4.5-5 Radial Power Profile through Reactor Vessel Core Region 4-49
Figure 4.6-1 Simplified Reactor (Red) and Coolant (Blue) Heat Removal Systems 4-53
Figure 5.2-1 Diagram of the Molten Salt Research Reactor Primary Heat Removal 5-4

MASTER LIST OF FIGURES

Figure 5.3-1 Coolant Loop (Secondary) Cooling System Configuration 5-8
Figure 6.1-1 Physical Configuration of the Reactor Cell, Reactor Enclosure, and Reactor Thermal Management System 6-2
Figure 6.2-1 Schematic of Reactor Thermal Management System 6-11
Figure 7.2-1 Instrumentation and Controls Diagram with Subsystems 7-5
Figure 7.4-1 Reactor Trip Relay Circuits 7-15
Figure 7.4-2 Reactor Trip Relay Circuits 7-16
Figure 7.5-1 ESFAS Actuation 7-24
Figure 8.1-1 Substation and Line Location for Molten Salt Research Reactor Site 8-3
Figure 8.1-2 MSRR Electrical Configuration 8-4
Figure 9.1-1 Heating, Ventilation, and Air Conditioning Systems Schematic Diagram 9-3
Figure 9.2-1 Deleted. 9-9
Figure 9.6-1 Conceptual Design of Gas Management System 9-20
Figure 9.7-1 Auxiliary Heat Removal System Conceptual Design 9-21
Figure 9.7-2 AHRS Conceptual Flow Path in Reactor Cell 9-22
Figure 12.1-1 Abilene Christian University Research Reactor Organization 12-4
Figure 12A-1 MSRR Staffing as Shown in Section 12.1 of the PSAR. 12-3
Figure 12A-2 MSRR Site Layout 12-4
Figure 13.1-1 Dose Rate Over Time 100 m (328 feet) from the SERC 13-15
Figure 13.1-2 Dose Rate Over Time in Immediate Vicinity of Reactor Building. 13-15
Figure 13.1-3 Reactor Loop Mass Flow Rate for Reactor Pump Failure Analysis 13-19
Figure 13.1-4 Pressure from RELAP5-3D Volumes Before and After Pump Component 13-20
Figure 13.1-5 Hot Leg, Cold Leg, and Center Channel Peak Temperatures on Pump Failure 13-21
Figure 13.1-6 Power Response from Complete Loss of Pumping Power 13-21
Figure 13.1-7 Total Reactor Power with Increased Fuel Salt Density as Helium Escapes 13-25
Figure 13.1-8 Hot Leg Temperature Increase Before Return to Temperature Equilibrium 13-26
Figure 13.1-9 Excessive Reactor Cooling Increases Reactor Power. 13-27
Figure 13.1-10 Hot Leg Temperature of Reactor Loop During Increased Cooling. 13-28
Figure 13.1-11 Reactor Vessel Outlet Temperature During 1 Percent UF_4 Transient Increase 13-29

MASTER LIST OF FIGURES

Figure 13.1-12 Simplified Diagram of Adverse Reactor Vessel Geometry Accident 13-30
Figure 13.1-13 Hot Leg Temperature with Increased Cooling 13-31
Figure 13.1-14 Initial Power Spike Followed by Loss of Criticality as Flow Stops 13-32
Figure 13.1-15 Second Limiting Reactivity Insertion Temperature 13-35
Figure 13.1-16 Second Limiting Reactivity Insertion Power 13-35
Figure 13.1-17 Third Limiting Reactivity Insertion Temperature 13-36
Figure 13.1-18 Third Limiting Reactivity Insertion Power 13-36
Figure 19.2-1 Borders and Major Cities in 200-mi (322-km) Radius from Molten Salt Research Reactor Site 19-5
Figure 19.2-2 Abilene, Texas Area [5 mi (8 km) Radius] 19-6
Figure 19.2-3 Abilene Christian University Science and Engineering Research Center . . . 19-7
Figure 19.2-4 Molten Salt Research Reactor Process Flow Diagram 19-9
Figure 19.3-1 Major Land Uses for the City of Abilene 19-15
Figure 19.3-2 Aerial View of Land Use in Five-Mile (8 km) Radius of Proposed Site 19-16
Figure 19.3-3 Abilene Sensitive Development Areas 19-17
Figure 19.3-4 Planned Land Use for the City of Abilene 19-19
Figure 19.3-5 Abilene, Texas Annual Average Precipitation 19-24
Figure 19.3-6 Abilene, Texas Average Wind Speeds 19-24
Figure 19.3-7 Abilene, Texas Wind Direction 19-25
Figure 19.3-8 Abilene Area Surface Waters 19-28
Figure 19.3-9 Cedar Creek USGS Gaging Station 19-30
Figure 19.3-10 Annual Peak Streamflow 19-31
Figure 19.3-11 Potential Floodways 19-32
Figure 19.3-12 Abilene Area Emergency Services 19-47
Figure 19.3-13 Nearest Sensitive Receptors to the MSRR 19-52
Figure 19.4-1 Molten Salt Research Reactor Site Layout 19-74
Figure 19.4-2 Science and Engineering Research Center First Floor Layout 19-75
Figure 19.4-3 Abilene, Texas Demographics and Population Statistics (NeighborhoodScout) 19-85
Figure 19.5-1 Sites Considered for the MSRR 19-103

MASTER LIST OF ACRONYMS

ABI	Abilene Regional Airport
AC	alternating current
ACI	American Concrete Institute
ACU	Abilene Christian University
AEA	Atomic Energy Act of 1954, as amended
AEP	American Electric Power
AFB	Air Force Base
AHRS	auxiliary heat removal system
AHU	air handling unit
ALARA	as low as reasonably achievable
ANP	Aircraft Nuclear Propulsion
ANS	American Nuclear Society
ANSI	American National Standards Institute
AQCR	Air Quality Control Regions
Ar	argon
ARDC	advanced reactor design criteria
ARE	Aircraft Reactor Experiment
ASME	American Society of Mechanical Engineers
ASME BPVC	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
ASTM	American Society for Testing and Materials
B4C	boron carbide
Be	beryllium
CAA	Clean Air Act
CCD	Census County Division
CFR	Code of Federal Regulations
CR	control rod
DC	design criteria
DCS	distributed control system
DOE	U.S. Department of Energy
EA	environmental assessment
EFS	Enhanced Fujita Scale
EIS	environmental impact statement
EPA	Environmental Protection Agency
EPZ	emergency planning zone
ESF	engineered safety feature
ESFAS	engineered safety features actuation system
FHS	fuel handling system
GDC	general design criteria
GMS	gas management system
H	hydrogen
HMI	human machine interface
HRS	heat removal system
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
IAEA	International Atomic Energy Agency

MASTER LIST OF ACRONYMS

IEEE	Institute of Electrical and Electronics Engineers
ISA	International Society of Automation
ISG	Interim Staff Guidance
K	potassium
LCO	limiting condition for operation
LONEP	loss of NORMAL ELECTRIC power
LWR	light-water reactor
LSSS	limiting safety system settings
MCNP	Monte Carlo N-Particle Transport
MCR	main control room
MHA	maximum hypothetical accident
MSR	molten salt reactor
MSRE	molten salt reactor experiment
MSRR	molten salt research reactor
MWFRS	main wind-force resisting system
NAAQS	National Ambient Air Quality Standards
NEA	Nuclear Energy Agency
NEIMA	Nuclear Energy Innovation and Modernization Act
NEPA	National Environmental Protection Agency
NEXT	Nuclear Energy eXperimental Testing
NFPA	National Fire Protection Association
NOAA	National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission
NSR	nonsafety-related
NTU	number of transfer units
OBE	operating basis earthquake
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
PGA	peak ground acceleration
PHR	preliminary heat removal
PMF	probable maximum flood
PSAR	Preliminary Safety Analysis Report
QL	quality level
R&D	research and development
RAV	reactor access vessel
RCS	reactor control system
RG	regulatory guide
ROI	region of influence
RP	radiation protection
RPP	radiation protection program
RPS	reactor protection system
RRI	research reactor infrastructure
RTMS	reactor thermal management system

MASTER LIST OF ACRONYMS

SEI	structural engineering institute
SERC	Science and Engineering Research Center
SR	surveillance requirement
SRA	sponsored research agreement
SSC	structures, systems, and components
SSE	safe shutdown earthquake
TCEQ	Texas Commission of Environmental Quality
TS	technical specifications
UL	Underwriters Laboratory
US	United States
UPS	uninterruptible power supply
USGS	U.S. Geological Survey
VAC	volts alternating current

Chapter 1
The Facility

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 1 THE FACILITY 1-1

1.1 Introduction 1-1

1.2 Summary and Conclusions on Principal Safety Considerations 1-2

1.2.1 Consequences from the Operation and Use of the Facility 1-3

1.2.2 Inherent and Passive Safety Features 1-3

1.2.3 Design Features and Design Bases 1-4

1.2.4 Potential Accidents at the Facility 1-8

1.3 General Description of the Facility 1-8

1.3.1 Geographical Location 1-8

1.3.2 Principal Characteristics of the Site 1-8

1.3.3 Design Criteria, Operating Characteristics, and Safety Systems 1-9

1.3.4 Engineered Safety Features 1-10

1.3.5 Instrumentation, Control, and Electrical Systems 1-11

1.3.6 Cooling and Other Auxiliary Systems 1-11

1.3.7 Heating, Ventilation, and Air-Conditioning Systems 1-11

1.3.8 Radioactive Waste Management and Radiation Protection 1-12

1.3.9 Experimental Facilities and Capabilities 1-12

1.4 Shared Facilities and Equipment 1-12

1.5 Comparison with Similar Facilities 1-13

1.6 Summary of Operations 1-15

1.7 Compliance with the Nuclear Waste Policy Act of 1982 1-15

1.8 Facility Modifications and History 1-16

1.9 References 1-16

CHAPTER 1 THE FACILITY

1.1 Introduction

Abilene Christian University (ACU), the applicant, is requesting approval for a construction permit for a low-power (less than 1 MWth) Molten Salt Research Reactor (MSRR) to be located on the ACU campus in Abilene, Texas. Details of the site location are further described in Section 2.1.1. ACU is a non-profit educational institution that has been in operation since 1906 and is accredited by the Southern Association of College and Schools Commission on Colleges to award associate's, bachelor's, master's, and doctoral degrees.

The MSRR is a utilization facility as described in the regulation in Title 10 of the Code of Federal Regulations, Part 50, Section 21(c) [10 CFR 50.21(c)] useful in the conduct of research and development activities of the types specified in Section 31 of the Atomic Energy Act of 1954, as amended (AEA or the Act), and the activities meet the 10 CFR 50.2 definition of research and development. The MSRR will not be a commercial and industrial facility as specified in 10 CFR 50.21(b) or in 10 CFR 50.22. Based on these activity tests and given that the proposed MSRR is not a testing facility, ACU is seeking to obtain a license under AEA Section 104c pursuant to 10 CFR 50.21(c) as a University Research Reactor facility. Abilene Christian University is aware of the changes made to Section 104c of the AEA by the Nuclear Energy Innovation and Modernization Act (NEIMA), and the MSRR activities will be consistent with licensing under Section 104c of the AEA as amended by NEIMA.

This Preliminary Safety Analysis Report (PSAR) is submitted in accordance with the provisions of 10 CFR 50.34(a) in support of the construction permit application. This PSAR follows the appropriate content and organization of (1) NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," as augmented by the Final Interim Staff Guidance (ISG) Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for Licensing Non-Power Reactors: Format and Content for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," dated October 17, 2012, and (2) the NRC endorsed ORNL/TM-2020/1478, Appendix A, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power MSRs: Format and Content."

The purpose of the MSRR is to accelerate the development and deployment of molten salt reactor (MSR) systems through foundational research while also developing a new pipeline to a nuclear-qualified workforce. Abilene Christian University's large capital investment in the MSRR will provide a world-class MSR facility to be used by large numbers of students, staff, faculty, and outside collaborators. The intended use of the MSRR is to conduct research on molten salt systems, as well as to educate and train a new generation of operators, engineers, and scientists who will be uniquely prepared to contribute to the advancement and deployment of MSRs and applications. The research will generate dynamic, time-dependent, experimental MSR data to advance the understanding of reactor kinetics, fuel salt behavior, corrosion, and migration and behavior of fission products throughout the system, all of which can be used to validate and calibrate software for the design, licensing, and regulation of commercial MSRs.

The MSRR is a single-region, graphite-moderated core, loop-type, thermal-spectrum reactor with a fluoride-based fuel salt flowing through 316H stainless steel fuel circuit components. A cooling loop with flowing, fluoride-based salt is used to cool the fuel circuit and expel the heat to atmosphere. The MSRR is equipped with an off-gas system that can relocate fission gases from the reactor headspaces through stainless-steel circuit components for sequestration and decay. The low-power MSRR is designed to be passively safe. It relies on intrinsic properties of molten salts and engineered safety features to ensure safe and reliable operations. Inherent and passive safety features of the facility are discussed further in Section 1.2.3.

The reactor system is described as a series of interconnected subsystems that include fuel handling, reactor system, primary heat removal system, gas management system, biological shielding, and instrumentation and control. Additionally, the NRC-licensed MSRR facility includes a research bay, a control room, and a radiochemistry laboratory with a change-out room. The MSRR facility layout within the Science and Engineering Research Center (SERC) is presented in Chapter 2 and Chapter 3. The fuel handling system (FHS) resides in the fuel handling cell and is described in Section 9.2. The reactor loop resides in the reactor cell and is described in Chapter 4. The coolant loop resides in an adjacent enclosure described in Chapter 5. The gas management system is described in Chapter 4 and Chapter 9. The biological shielding is described in Section 4.4. The instrumentation and control system is described in Chapter 7. The research bay is discussed in Chapter 2 and Chapter 3. The control room is described in Chapter 7. The radiochemistry laboratory is described in Chapter 9.

The MSRR design is unique by virtue of its liquid fuel and minimal excess reactivity.

1.2 Summary and Conclusions on Principal Safety Considerations

The MSRR design is predicated on the MSR technology developed at the Oak Ridge National Laboratory (ORNL) during the 1950's and 60's (see Section 1.5), and it also takes advantage of more recent research results and technology developments. The following fundamental safety concepts were demonstrated at ORNL:

- Retention of the vast majority of radioisotopes within the salt and salt-wetted surfaces
- Highly negative reactivity coefficients
- Small online fuel additions to minimize excess reactivity

There are a number of inherent safety features built into the design and materials of the MSRR. Given the low power design of the reactor, the overall risk to people and the environment is limited by the small source term and the low fission product inventory. The MSRR is significantly different from the reactors licensed in the past by the Nuclear Regulatory Commission (NRC) and has several unique safety features not found in solid fuel systems. As an MSR, most of the inherent safety features are a result of the formulation and properties of the salts and the movement of the salts within the system (Chapter 4 and Chapter 5).

The MSRR salts are highly ionic compounds that are chemically stable, immune to radiolysis, and are compatible with the MSRR structural materials. They do not react rapidly with moisture or air. Their chemical inertness eliminates the risk of fire or

explosion due to chemical interaction. Molten salts have been used for years in industry as heat transfer media for their inertness and safety. The MSRR salts are stable to several hundred degrees above temperatures obtainable in the reactor and remain at low vapor pressure. Chapter 4 provides further details. As an inert liquid system at low pressure, safety demands are significantly reduced on the MSRR design.

Additionally, the MSRR design relies on passive decay heat removal and does not need an emergency core cooling system for decay heat removal.

Abilene Christian University does not identify a need for a research and development program as described by 10 CFR 50.34(a)(8). The MSRR is being designed and will be constructed with ample margin for safety. The MSRR has no structures, systems, or components (SSCs) that require research and development to confirm the adequacy of their design. However, as a research reactor, multiple research activities will be conducted to support the development of commercial molten salt reactors. For example, corrosion testing will more precisely define the corrosion in the MSRR environment, however ample corrosion allowance will be provided such that a safety question is not raised. Therefore, because there are no safety questions that need to be resolved for such SSCs and ACU is providing large design safety margins, ACU has not identified any need for research and development in accordance with 10 CFR 50.34(a)(8).

1.2.1 Consequences from the Operation and Use of the Facility

A key measure of safety and consequence from the operation of the facility is the magnitude of the potential source term associated with off-normal events and accidents. The source term represents the amount, timing, and nature of radioactive material released and available for release to the environment following a postulated event. The fuel salt inherently retains fission products, which together with the reactor loop boundary, reactor enclosure, and reactor cell, ensure the consequences of any accident are below regulatory limits. A maximum hypothetical accident (MHA) for the MSRR has been established as a non-mechanistic, combined failure of multiple systems resulting in a fission product release. As described in Section 13.1.1, non-physical assumptions drive radionuclide movement and bound the system response to other postulated events. This analysis is consistent with the accident analysis required for research reactors and demonstrates that, even in the presence of incredibly conservative assumptions, no member of the public who remains at the site boundary can receive more than 100 mrem of dose from the MHA.

No electrical power is required for safe shutdown or long-term decay heat removal. Loss of power causes helium gas valves to open (unpowered state), allowing the fuel salt to drain from the reactor to the drain tank, shutting down the nuclear reaction. The fuel salt in the drain tank is passively cooled by conduction, natural convection, and thermal radiation. The surrounding structures have sufficient thermal mass to absorb the decay heat indefinitely without exceeding thermal limits.

1.2.2 Inherent and Passive Safety Features

The MSRR includes a number of inherent safety features:

- Radionuclide retention in the salt and in the primary steel boundaries (Chapter 4).

- Boiling point of salts is far higher than temperatures that can be achieved by the reactor (no energetic release due to phase change) as described in Chapter 4.
- Salts are ionic and cannot be damaged in service, as described in Chapter 4.
- Salts are inert (no exothermic chemical reactions) and described in Chapter 4.
- Reactivity coefficients are highly negative (key safety) as described in Chapter 4.
- Gravity and fail-open gas valves are used to drain the fuel salt as discussed in Chapter 4 and Chapter 7.
- Drain tank is highly subcritical under all conditions, including submersion of the enclosure in water (unanticipated weather event) as discussed in Chapter 4.
- Passively cooled under loss of power (Chapter 4).
- No phase changes anywhere in the system (no energetic releases) as described in Chapter 4.
- Low operating pressures (Chapter 4 and Chapter 5).
- Stainless steel and graphite have high margins for duty as discussed in Chapter 4.
- Reactor cells and SSCs are located below grade and designed to withstand all external events. The building structural design and design consideration from natural phenomena events are described in Chapter 3.
- Radiological shielding ensures the dose rate in uncontrolled areas remains below the dose limit allowed for individual members of the public and workers and as described in Section 4.4. The radiation protection program as described in Chapter 11 will be provided in the Operating License application consistent with 10 CFR 50.34(b)(3).
- Ventilation systems are designed to support operations as described in Chapter 9.

1.2.3 Design Features and Design Bases

The design criteria for the MSRR described in Section 3.1.1 are based on the latest draft of ANSI/ANS-20.2-2021 [Reference 1.9-1]. The system-related actions throughout the PSAR describe how the design bases, including the design criteria, are satisfied.

Given the low power design of the reactor, the overall risk to people and the environment is limited by the small source term and the low fission product inventory. As noted above in Section 1.2, the reactor design relies on a layered safety approach, including low-leakage, pressure-maintaining enclosures to control the release of fission products. There are no liquid-to-gas phase transitions in the reactor system. The enclosures need only to be designed to provide margin for the heating of the gases present within the enclosures during all postulated events and are low-pressure systems.

The SSCs in the facility are all below grade and protected from external events. Details of the SSCs safety classification are provided in Section 3.5.2.

1.2.3.1 Thermal Power Level and Control of Heat Production

The MSRR is designed to operate up to 1 MWth. Excess reactivity is minimized and maintained through small uranium salt additions when necessary. The MSRR has a large negative temperature coefficient of reactivity that passively moderates heat production. Thus, the MSRR has a significant “load-following” capability where reduced heat removal through the heat exchanger leads to increased fuel salt temperature and decreased reactivity, or greater heat removal reduces fuel salt temperature and increases reactivity (see Chapter 4 and Chapter 5). The MSRR reactivity can be controlled through circulation rate adjustments in the secondary coolant salt circuit when it is being utilized for heat removal, as heat removal is not required under low-power operations. Control rods are used to compensate, maintain, and stop fission heat production during normal operations and are not relied on for safe shutdown (see Chapters 3, 4, and 7). The MSRR design provides for immediate initiation of fuel salt relocation into a non-critical configuration inside the drain tank as part of the reactor protection system (RPS), either on command or under facility loss of external electric power (see Chapters 3, 4 and 7).

1.2.3.2 Fuel Type, Enrichment, and Moderator

The MSRR is fueled with a uranium-bearing, fluoride-based salt containing minimal oxidative impurities. The baseline fuel composition is LiF-BeF_2-UF_4 in a molar ratio of approximately 67:28:5. The enrichment of U-235 will be approximately 19.75 percent. The enrichment of lithium in Li^7 will be approximately 99.99 percent or more. The mass of the fuel salt will be approximately 1600 kg (3527 lb) and is a thoroughly mixed combination of less than 500 kg (1102 lb) of UF_4 and approximately 1100 kg (2425 lb) of LiF-BeF_2. Details of the fuel are provided in Section 4.2. The fuel will be owned by the U.S. government and loaned to ACU by the Department of Energy. The moderating material is unclad graphite and described in Section 4.2.

1.2.3.3 Reactor System

All salt-facing structural metals in the MSRR fuel system are constructed of 316H stainless steel and serve as the primary fuel boundary. The entire reactor system is mechanically supported inside a steel reactor enclosure designed to isolate all potential radiological releases from the reactor loop. The reactor loop consists of the reactor vessel, reactor access vessel, reactor (fuel salt) pump, heat exchanger, drain tank and associated 2.5-in. (nominal) diameter piping. Flowing fuel salt enters the reactor vessel from the bottom, flows upward to the reactor access vessel, over to the reactor pump, down to the heat exchanger, and back to the bottom of the reactor vessel. A complete description of the reactor loop can be found in Chapter 4.

A reactor access vessel is the highest point of the reactor system and provides a gas head space with fuel salt level control and the only point of direct access to the fuel salt and off-gas while the reactor is operating. “Nano-samples” of fuel salt and gases are regularly collected from the reactor access vessel and taken to the

radiochemistry lab to monitor corrosion of the reactor system, to measure the redox potential of the fuel salt, and to collect dynamic isotopic data on the fuel salt and off gas as they evolve over time. The reactor access vessel is also used for small, fresh-fuel additions and redox potential adjustments of the fuel salt, if and when necessary. Coupons of stainless steel, as well as other materials compatible with the fuel salt, are located within the reactor access vessel to be exposed to fuel salt. These coupons are periodically withdrawn to assess the corrosion potential of the fuel salt to the reactor system boundary. All access equipment is designed to maintain reactor safety boundaries. Details can be found in Chapter 4.

1.2.3.4 Reactor Pump and Heat Exchanger

The reactor pump provides a flow rate through the loop at up to a nominal 23.9 kg/s (52.7 lb/s). The heat exchanger is the shell-and-tube type with fuel salt moving through the shell. Fuel salt leaving the heat exchanger flows back into the reactor vessel. The tube pressure is maintained above that of the shell to minimize contamination in the cooling loop in the event of a tube rupture. Details can be found in Chapters 4 and 5.

1.2.3.5 Reactor Vessel

The reactor vessel surrounds and supports the roughly cylindrical graphite core, where the vast majority of fission heat is generated in the MSRR. The core is supported by a grid plate affixed to the inside of the vessel. The reactor vessel is approximately 132 cm (52 in.) in diameter and approximately 179 cm (70 in.) tall. Details of the reactor vessel can be found in Section 4.3.

1.2.3.6 Drain Tank

The MSRR drain tank is the lowest point in the reactor system. The fuel salt drains to this location by gravity. The tank is designed to ensure a strongly subcritical fuel configuration and to passively manage the decay heat. The drain tank is electrically heated to minimize thermal shock, to maintain fuel salt readiness, and to re-melt the salt, when necessary. The drain tank system is also designed so that gas head pressure in the tank will pneumatically push the fuel salt back up into the reactor loop and maintain fuel salt column height in the reactor access vessel. Outside of normal reactor operations, the drain tank system can be configured for bulk fuel salt transfer operations as one component of the Fuel Handling System (FHS). Most of the FHS is located in the fuel handling cell (Section 9.2), located in a shielded area near the reactor system. Details of the fuel drain tank can be found in Section 9.2 and in Chapter 4.

1.2.3.7 Reactor Thermal Management System

The reactor vessel and drain tank are located inside an insulated vessel performing two safety functions. The first is to maintain the temperature of those reactor components under normal and accident conditions for long enough to allow for the system to drain. The second is to serve as a catch-pan for fuel salt that might escape the reactor loop. While this system is heated and insulated, it is

capable of removing decay heat when electric power is lost and the auxiliary heat removal system (AHRS) is not in operation. The system is described in Chapter 6, and the consequences of this accident described in Section 13.1.10.

1.2.3.8 Thermal Management

Heat generated in the MSRR is expelled to the atmosphere. Reactor power primarily is processed through a heat rejection system, which transfers heat from the fuel salt to the coolant salt (secondary) cooling system through the heat exchanger and then transfers that heat to the atmosphere through a forced air radiator. Both salt loops are heavily insulated to provide thermal stability. The operating temperature of the fuel salt and coolant salt is approximately 600 degrees Celsius (1112 degrees Fahrenheit). The primary heat removal system is located in the reactor enclosure. To minimize tritium production in the fuel salt and coolant salt, the lithium in the salts are approximately 99.99 percent enriched in Li^7. Overall thermal management during power operations is accomplished through flow rate adjustments of the salts and radiator air flow. The system is described in Chapter 5.

The melting point of the fuel salt is about 500 degrees Celsius (932 degrees Fahrenheit) and the melting point of the coolant salt is about 460 degrees Celsius (890 degrees Fahrenheit). For the salts to remain molten under low-power running conditions, fission heat can be supplemented by the reactor thermal management system (RTMS) and is described in Chapter 4 and Chapter 5.

Air circulates through the research bay to provide comfortable working conditions. Air is supplied to the enclosures and consists of a number of interrelated systems. There are large quantities of forced air, ranging from fresh to slightly radioactive. There are multiple potential sources of radioisotopes that can be introduced into the air within the facility, including the reactor cell, which is routed to exhaust while other air is routed into controlled locations before release. Air exhausts from the building are monitored for radiological release. Descriptions of these systems can be found in Chapter 5 and Chapter 9.

1.2.3.9 Helium Gas Management System

The gas contacting the MSRR salts is predominantly helium. It is used to cover and pneumatically position salts throughout the systems. Helium gas is used to pneumatically transfer salt shipments through the salt handling systems into the drain tanks of the salt loops, to transfer salt volumes from the drain tanks up into the loops, and to maintain operational salts loop levels. The design of the helium gas clean-up system, or off-gas system, for the MSRR is similar to a system that was demonstrated successfully during the molten salt reactor experiment (MSRE) program. The MSRR off-gas system is designed to provide high-fidelity performance data for the off-gas system, as well as for the individual components. The system is described in Section 9.6.

1.2.3.10 Shielding

The MSRR biological shield is a multi-layered system designed to ensure that the dose rate in uncontrolled areas remains below the dose limit allowed for individual members of the public (10 CFR 20.1301) and the dose rate in controlled areas is consistent with the occupational dose limits to individual adults (10 CFR 20.1201). From inside to outside, the multi-layered shielding system consists of an internal shield, which is located inside the reactor enclosure and surrounds the reactor vessel, a layer of sacrificial shielding inside the reactor cell to preserve the reactor cell vault components, and then the reactor cell vault itself. A complete description of the biological shielding is provided in Section 4.4.

1.2.4 Potential Accidents at the Facility

Potential events related to weather, industrial facilities, and transportation are identified in Chapter 2 and Chapter 3. All safety-related SSCs required for safe shutdown and decay heat removal are located below grade in the MSRR research bay systems pit and protected from the potential consequences of tornadoes, hurricanes, floods, explosions, and toxic gas as described in Chapter 3. Section 13.1 describes the MHA as a non-mechanistic, combined failure of multiple systems resulting in a fission product release. Non-physical and conservative assumptions drive radionuclide movement and bound the system response to other postulated events. This analysis is consistent with the accident analysis required for research reactors and demonstrates that even in the presence of incredibly conservative assumptions, no member of the public at the site boundary can receive more than 100 mrem of dose from the MHA.

1.3 General Description of the Facility

1.3.1 Geographical Location

Abilene is situated in North Central Texas approximately 142 miles (228 km) west of Fort Worth, 210 miles (340 km) northwest of San Antonio, and 146 miles (235 km) southeast of Lubbock. The main campus of ACU is a 287-acre (116-ha) property located in the northeast area of Abilene, Texas. The geographic location is described in greater detail in Chapter 2.

1.3.2 Principal Characteristics of the Site

The MSRR is sited at the southeast end of the main ACU campus in Abilene. The MSRR is housed in the existing SERC, an ACU-owned, multiuse facility as defined in 10 CFR 50.10(a)(2)(x). The SERC sits in the approximate center of 15 acres (6 ha) of previously developed land bounded by four city streets. The site also includes other ACU buildings used for research and academic purposes.

Surrounding the reactor site are ACU residence halls and a campus parking lot, which are nearest the facility, and slightly farther away there are businesses and private residences. There are no tall structures near enough to the site to affect diffusion and dispersion of airborne effluents. The elevation of the SERC is 1733 feet (528 m) above sea level.

The site is described further in Chapter 2 and the SERC in Chapters 2 and 3.

1.3.3 Design Criteria, Operating Characteristics, and Safety Systems

1.3.3.1 Design Criteria

The design criteria for the MSRR are described in Section 3.1.1 and are based on the latest draft of ANSI/ANS-20.2-2021 [Reference 1.9-1].

1.3.3.2 Operating Characteristics

The MSRR reactor is designed to achieve a maximum reactor power of 1 MWth (design rated thermal power). The reactor parameters are provided in Table 4.1-1. The MSRR will be operated for up to 5 effective full power years. ACU intends to submit a 20-year operating license application in the future.

1.3.3.3 Safety Systems

Salt loop leaks or spills in the MSRR are contained within air-cooled, sealed enclosures and cannot lead to a major accident because there are no violent reactions that can accompany their release. If a leak occurs, the molten salts cool in the enclosures designed to contain them. The freezing process is inherent and passive. Gases generated, used, and liberated are either contained or processed through the gas management system and are described in Chapter 4 and Chapter 9.

The MSRR fuel salt has a short and unobstructed path from the reactor vessel to a drain tank, allowing the fuel salt to be completely drained passively in approximately one minute. The geometry and position of the drain tank ensures a noncritical configuration under all conditions. Further details are provided in Chapter 4 and Chapter 7.

The reactor vessel is the only location where criticality can be reached and sustained in the MSRR. Unlike existing reactor types, only a fraction of the fuel participates in the sustained chain reaction at any given time, as the remainder is flowing through the rest of the reactor loop. The MSRR core is designed to have very low excess reactivity, which can be made up through occasional fuel salt additions introduced through the access vessel. An inherent safety feature of the fuel salt is that it provides a strong negative reactivity temperature coefficient, in part due to Doppler broadening but primarily from the salt coefficient of thermal expansion. Combined with the low excess reactivity, the MSRR has a high degree of passive self-regulation. Control rods capable of controlling fission rate are also included in the MSRR design for normal operations and are not required for safe shutdown. Further details are available in Chapter 4.

To ensure people and the environment are protected from radioisotope releases, the MSRR design incorporates a system of layered barriers in and around the FHS, reactor system, coolant loop (secondary) cooling system, and gas management systems. The first barrier in the systems that move or flow salt is the salt itself. The vast majority of fission products remain in the salt with some plating

out on salt-wetted surfaces. Off-gases and cover gases do not have this intrinsic feature but contain only a small fraction of the radioisotopes, which also can plate out on gas-contacted surfaces. These surfaces are the first engineered safety layer of the MSRR and define the fuel salt and gas boundaries. These primary boundaries are constructed of 316H stainless steel and are designed to be leak tight. The second engineered safety layer for the MSRR systems are steel enclosures that completely surround the primary boundaries. The enclosures are maintained at a low negative pressure, are leak tight, and are cooled by exterior air. All radionuclides potentially released from the MSRR primary boundaries are fully retained in the enclosures. The enclosures for the different systems are located in separate cells and provide another layer of protection from radiological release. The cells are connected to the air handling system, which provides cooling for the enclosures and ensures a habitable environment in accessible cells and the remainder of the MSRR facility. Further details are provided in Chapter 4 and Chapter 6.

The MSRR safety-related SSCs required for safe-shutdown and decay heat removal are located below grade (Chapters 3, 4, and 6). The MSRR also includes a biological shielding system designed to ensure the protection of people and the environment from radiation. The MSRR facility operating staff are subject to occupational radiation exposure from working in a facility that contains radioactive materials. Members of the public are potentially subject to limited exposure from radiological effluent releases during normal operations. For normal operation, such exposures are maintained below the limits of 10 CFR 20.1201 and 10 CFR 20.1301 for the operating staff and members of the public, respectively. Potential doses to the public resulting from postulated events are maintained by design to be well within the limits for research reactors. The radiation protection program, which addresses the limits in 10 CFR Part 20 as described in Chapter 11, will be provided in the Operating License application consistent with 10 CFR 50.34(b)(3).

1.3.4 Engineered Safety Features

The MSRR has three layers of engineered safety features to provide functional containment as defined in SECY-18-0096, “Functional Containment Performance Criteria for Non-Light-Water Reactors”:

- Reactor thermal management system
- Reactor enclosure
- Reactor cell

The RTMS surrounds the reactor loop and drain tank, and maintains their temperatures during normal and accident conditions to ensure the fuel salt successfully drains. It also serves as a catch pan to collect spilled fuel salt to prevent it from coming into contact with the reactor enclosure.

The reactor enclosure surrounds the reactor system, including the reactor loop, the drain tank, and the RTMS, and is a leak-tight fission product barrier. It operates under slightly negative pressure under normal and accident conditions.

The reactor cell surrounds the reactor enclosure. During normal operation, the AHRS moves air through the reactor cell to cool the reactor enclosure and reactor cell. During accident conditions, the AHRS is shut down, and louvers isolate the AHRS intake and exhaust. This provides a hold up volume for any leakage from the reactor enclosure.

The MSRR does not require or use an emergency cooling system. Heat during a loss of electrical power accident is passively removed to the surrounding structures by conduction, natural convection, and thermal radiation. These structures have sufficient thermal mass to dissipate decay heat indefinitely.

The engineered safety features systems are described in Chapter 6.

1.3.5 Instrumentation, Control, and Electrical Systems

The instrumentation and control system measures plant parameters, controls components, and provides an interface to the plant operators. It provides control signals for manual and automated functions, protective features to bring the plant to shutdown conditions, and monitoring systems for plant condition assessment and personnel protection. With it an operator can start up, operate, and shut down the reactor across all anticipated operational ranges and respond to all credible accidents. The MSRR instrumentation and control system consists of the reactor control system (RCS), the reactor protection system (RPS) and engineered safety features actuation system (ESFAS), the main control room (MCR) and the radiation monitoring system, as described in detail in Chapter 7. Electrical service is provided through the SERC facility and the MSRR electrical systems are described in Chapter 8.

1.3.6 Cooling and Other Auxiliary Systems

The Auxiliary Heat Removal System (AHRS) cools the reactor cell and ensures heat can be dissipated from the reactor enclosure under normal conditions. More detailed information on auxiliary cooling systems can be found in Chapter 5 and Chapter 9.

Detailed descriptions of the cooling and other auxiliary systems are provided in Chapter 9.

1.3.7 Heating, Ventilation, and Air-Conditioning Systems

The heating, ventilation, and air conditioning (HVAC) systems of the research bay housing the MSRR reactor and for the associated control rooms and labs provides the following functions:

- Provide humidity, pressure, and temperature control to the habitable spaces
- Ensure conditioned, clean air flows from regions of lower radiation hazard to areas of higher hazard
- Provide conditioned air for the RTMS during normal operation
- Monitor exhaust air for controlled effluent gases

- Minimize spread of contamination in the event of a radiation release

Details of the HVAC system can be found in Chapter 9.

1.3.8 Radioactive Waste Management and Radiation Protection

A radiation protection program is established to protect the radiological health and safety of workers. The program complies with the regulatory requirements of 10 CFR Parts 19, 20, and 70. This program includes the elements of an as low as reasonably achievable program, radiation monitoring and surveying, exposure control, dosimetry, contamination control, and environmental monitoring. The radiation protection program is addressed in Chapter 11 and will be provided in the Operating License application consistent with 10 CFR 50.34(b)(3).

The facility includes capabilities for the management of liquid, gaseous, and solid radioactive wastes produced by operations. The radioactive waste management systems are discussed in Chapter 11 and will be fully described in the Operating License application.

1.3.9 Experimental Facilities and Capabilities

Experimental facilities within the MSRR are described in Chapter 10 and fall under two categories: 1) instrumentation, facilities, and a laboratory for the measurement of the molten salt and off-gas and 2) scientific surveillance to characterize select aspects of the MSRR, including radiation physics, thermophysics, chemistry, material properties, radionuclide production and transport, and operation. Within these two categories are methods to measure parameters including radioactivity, redox potential, corrosion effects, and particulate formation. Molten salt measurement systems include features to measure the level (height) of molten salt within the reactor access vessel. There are helium bubble generation and removal systems to assess gaseous nuclide transport and to capture rates within the fuel salt. An associated laboratory supports the molten fuel salt and off-gas measurement capabilities.

New experiments may be added following both a management review and a review by the MSRR Review and Audit Committee. The committee reviews and approves all experimental facilities and procedures for experiments, and assesses each experiment within the requirements of 10 CFR 50.59, as described in Chapter 12.

1.4 Shared Facilities and Equipment

The MSRR facility contains a single-unit reactor that does not share any systems or equipment necessary to perform a safety function or for the safe operation of the plant with other facilities not covered by this safety analysis report. It is anticipated that some infrastructure not credited to perform a safety function or for safe operation may be shared with other nearby or onsite facilities. Examples include site utilities such as electrical, gas, and water supply systems; warehousing and storage; and site access roads.

1.5 Comparison with Similar Facilities

There has never been a molten-salt-fueled reactor licensed by the NRC or the Department of Energy; however, three critical molten salt-fueled reactors were operated at the ORNL under the auspices of the U.S. Atomic Energy Commission. These were the Aircraft Reactor Experiment (ARE), the Pratt and Whitney Aircraft Reactor-1 (PWAR-1), and the MSRE. All these machines were similar to the MSRR because they were thermal spectrum reactors in loop configuration, used fluid fuel in the form of molten fluoride salt, and operated at high temperature and low pressure. This section gives a brief history of these past developments. This section comments on the safety systems of these machines and points out similarities and differences from the MSRR design.

During the early years of applied nuclear physics, Eugene Wigner and Harold Urey, both Nobel Prize winners, argued that the U.S. should be looking at nuclear reactors in which the fuel elements are replaced by liquids. Wigner and Urey proposed that fission systems should be considered as chemical systems, rather than mechanical engineering devices, suggesting the homogeneous fluid fuel form. While aqueous solutions of uranium were first considered, Wigner suggested fluoride salts as more promising [Reference 1.9-2].

In 1946, the Nuclear Energy for Propulsion of Aircraft (NEPA) project started in the U.S., followed by the Aircraft Nuclear Propulsion (ANP) program in 1951. During this time, MSRs emerged as a primary technology for the ANP program. In March 1952, it was determined the performance and design of the circulating-fuel aircraft reactor was sufficiently encouraging that the first ARE should be constructed and tested, which took place at ORNL. The ARE loop was made from Inconel and its safety shutdown was accomplished by boron carbide absorber within stainless-steel control rods. Because the system was not drainable, enough negative reactivity was available to keep the ARE subcritical at room temperature. [Reference 1.9-3] In 1954, the ARE ran successfully for 100 hours at 860 degrees Celsius (1580 degrees Fahrenheit) [Reference 1.9-4]. The ARE circulated a NaF-ZrF_4-UF_4 fuel salt through Inconel tubes surrounded by beryllium reflectors, producing 2.5 MWth. This first MSR test showed that gaseous fission products were removed naturally through pumping action and demonstrated stable operation due to a high negative reactivity coefficient [Reference 1.9-5]. The ARE also demonstrated load-following operation without control rods.

Other MSR propulsion activities continued at ORNL, including the construction and operation of the PWAR-1, an externally heated, zero-power critical experiment designed to test the nuclear physics of an MSR with complex, non-lattice, geometry [Reference 1.9-6]. It had a central control rod. The annular absorber was made from rare earth oxides and contained within an Inconel casing. The PWAR-1 was a physics test preceding the Aircraft Reactor Test (ART) machine [Reference 1.9-7], a never-built test reactor for the aircraft nuclear engine. Both the PWAR-1 and the ART were drainable. The ANP program was canceled in 1961 when the development of reliable intercontinental ballistic missiles obviated the need for a nuclear-powered bomber.

In 1957, Raymond Bryant and Alvin Weinberg stated [Reference 1.9-8] that, “two very different schools of reactor design have emerged since the first reactors were built. One approach, exemplified by solid fuel reactors, holds that a reactor is basically a mechanical plant; the ultimate rationalization is to be sought in simplifying heat transfer machinery.

The other approach, exemplified by liquid fuel reactors, holds that a reactor is basically a chemical plant; the ultimate rationalization is to be sought in simplifying the handling and reprocessing of fuel. At the Oak Ridge National Laboratory, we have chosen to explore the second approach to reactor development."

In 1958, the Civilian Molten Salt Power Reactor program was initiated, and early MSR designs leveraged the work carried out under the NEPA and ANP military programs. In 1959, the Fluid Fuel Reactor Task Force reviewed the three fluid-fueled reactor concepts that were being developed (aqueous homogeneous reactor, liquid metal fuel reactor, and MSR) and ultimately selected the MSR as the primary fluid-fueled reactor concept, as it was recognized as having the highest probability of achieving technical feasibility. The textbook Fluid Fuel Reactors [Reference 1.9-9] provides a great overview of the technology status and design considerations at that time. From 1960 to 1964, the primary focus of MSR reactor design efforts resulted in the construction of an 8-MWth MSR that would become the centerpiece of the MSRE at ORNL and provided the first significant step towards civilian use of the technology.

The MSRE was a single-region core, loop-type reactor with a fuel salt mixture of LiF-BeF_2-ZrF4-UF_4 (65-29-5-1 mole percent) that flowed through a graphite moderator and Hastelloy-N fuel circuit components. The moderator consisted of vertical stringers of graphite formed into a cylindrical core within the reactor vessel. The fuel entered the reactor vessel and was directed downward through an outer annulus downcomer between the periphery of the graphite cylinder and the inside of the core barrel. The fuel was redirected upward through channels formed between the graphite stringers, exiting the top at approximately 663 degrees Celsius (1225 degrees Fahrenheit), out the top to a sump-type fuel pump, through a tube and shell heat exchanger, and returned to the core vessel. The MSRE used light water for cooling the containment atmosphere, containment vessel, reactor shield, drain tank, primary pump, and containment penetrations. The safety of the system was achieved by a combination of three control rods and a drain tank. An actively cooled freeze valve separated the core loop from the drain tank. This ensured that the core would drain passively if the MSRE lost external power [Reference 1.9-10], [Reference 1.9-11], [Reference 1.9-12]. The MSRE operated from June 1965 (first criticality) to December 1969 (shutdown), and eventually was terminated in 1976. Achievements included over 13,000 full power hours of operation, the first use of U-233 fuel, first use of mixed U/Pu fuel salt, online refueling, and the generation of a substantial body of knowledge and documents that form a foundation for future MSR designs [Reference 1.9-13].

The MSRR is most closely related to the MSRE in its design. Both feature a graphite moderator, a fuel salt flowing in a graphite channel lattice, a lithium fluoride and beryllium fluoride melt as a base for the fuel salt composition, and a safety system based on control rod insertion combined with the ability to drain the core. The MSRE had a freeze valve between the fuel salt loop and the drain tank, while the MSRR actively pressurizes the drain tank to keep the fuel salt in the loop.

As the development of safe, efficient, and clean MSR designs moved from the military to national labs in the past, MSR research and development is now expanding to universities in support of their deployment into the world. President Schubert of Abilene Christian University instantiated the Nuclear Energy eXperimental Testing (NEXT) Lab as

a humanitarian service and a natural fit to the ACU mission to educate students for Christian service and leadership throughout the world. Dr. Schubert summarizes this initiative by saying, “We’re looking at pushing forward to provide basic needs for humanity. I can’t think of anything that fits better with what we’re about at Abilene Christian University - being God’s hands and feet in the world.”

1.6 Summary of Operations

As noted in Section 1.1, the purpose of the MSRR is to accelerate the development and deployment of MSR systems through foundational research while also developing a new pipeline to a nuclear qualified workforce. The facility will also provide unique data and insights for the safety analysis tools and computational methodologies used for the design and licensing of future commercial reactors. The major programs to be performed in the facility will be provided in the Operating License application consistent with 10 CFR 50.34(b)(2).

The MSRR will be operated over the full range of power to research fundamental aspects of the technology. The process system designs include the necessary features to monitor and assess plant performance in support of these objectives as described elsewhere in this report. The activation product inventory and fission product inventory from the normal operation of the facility and effluent release pathways to the environment are discussed in Chapter 11, and a description of the radiation sources for the facility will be provided in the Operating License application consistent with 10 CFR 50.34(b)(3).

An analysis of postulated events and accidents from operation of the facility, including the radiological consequences of potential releases, including those from an MHA, is addressed in Chapter 13. Effluent releases to the environment are discussed in Chapter 11. Radiological consequences of unplanned releases are addressed in Chapter 13.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Abilene Christian University intends to enter into a contract with the Department of Energy for required fuel cycle services. This will be discussed further in the Operating License application, consistent with Section 302(b)(1) of the Nuclear Waste Policy Act of 1982. A letter to ACU from the Department of Energy (DOE) affirming good faith negotiations was provided to the NRC [Reference 1.9-14].

The letter from DOE refers to ACU as an owner and generator of spent nuclear fuel. However, the current plan and preference of ACU is not to own the MSRR fuel, but to obtain fuel as a loan from DOE. After the MSRR is decommissioned, the fuel and salt could be used on other projects and would be returned to DOE. All ACU discussions with DOE have been to obtain the fuel as a loan. There is the possibility that, in order to obtain fuel in a timely manner, ACU could purchase fuel from someone other than DOE. However, this path is not currently being pursued.

1.8 Facility Modifications and History

The Construction Permit application does not include the construction of a new facility. The MSRR will be installed within a portion of the existing, multiuse Science and Engineering Research Center (SERC) facility located on the ACU campus. The MSRR license boundary within the SERC, referred to as the MSRR facility, is described in Chapter 2 and Chapter 3. The SERC was designed to applicable building codes for a commercial construction classified as Business (B) occupancy and the research bay, which contains the MSRR and SSCs, is constructed to building codes applicable for industrial occupancy of type II-B (non-combustible construction). As the SERC was designed to house radiation-producing devices below grade, the research bay floor and subterranean systems pit concrete structure was designed to meet ACI 349-2103.

There are no prior operating histories of existing NRC-licensed facilities or modifications to the existing facility or site.

1.9 References

1.9-1 American National Standards Institute/American Nuclear Society, “American National Standard Nuclear Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten Salt Reactor Nuclear Power Plants,” ANSI/ANS-20.2, 2021, LaGrange Park, IL.

1.9-2 A. M. Weinberg, “The Proto-History of the Molten Salt System,” Oak Ridge National Laboratory, Oak Ridge, TN, February 28, 1997.

1.9-3 E. S. Bettis, et al, “The Aircraft Reactor Experiment-Design and Construction,” Nuclear Science and Engineering, Vol. 2.6, pp. 804-825, 1957.

1.9-4 E. S. Bettis, et al, “The Aircraft Reactor Experiment-Operation,” Nuclear Science and Engineering , Vol. 2.6, pp. 841-853, 1957.

1.9-5 W. K. Ergen, et al, “The Aircraft Reactor Experiment-Physics,” Nuclear Science and Engineering, Vol. 2.6, pp. 826-840, 1957.

1.9-6 D. Scott, et al, “A Zero Power Reflector-Moderated Reactor Experiment at Elevated Temperature,ORNL-2536,” Oak Ridge National Laboratory, Oak Ridge, TN, 1958.

1.9-7 A. P. Fraas and A. W. Savolainen, “Design Report on the Aircraft Reactor Test, ORNL-2095,” Oak Ridge National Laboratory, Oak Ridge, TN, 1958.

1.9-8 R. C. Briant and A. M. Weinberg, “Molten Fluorides as Power Reactor Fuels,” Nuclear Science and Engineering, Vol. 2, p. 797-803, 1957.

1.9-9 H. G. MacPherson, F. Maslan, and J. Lane, Fluid Fuel Reactors: Molten Salt Reactors, Aqueous Homogeneous Reactors, Fluoride Reactors, Chloride Reactors, Liquid Metal Reactors and Why Liquid Fission, Addison-Wesley Pub. Co, 1958.

1.9-10 R. C. Robertson, “MSRE Design and Operations Report Part I: Description of Reactor Design (ORNL-TM-728),” Oak Ridge National Laboratory, Oak Ridge, TN, 1965.

1.9-11 S. E. Beall, et al, “MSRE Design and Operations Report Part V: Reactor Safety Analysis Report (ORNL-TM-732),” Oak Ridge National Lab, Oak Ridge, TN, 1964.

1.9-12 M. Richardson, “Development of Freeze Valve for Use in the MSRE (ORNL-TM-128),” Oak Ridge National Laboratory, Oak Ridge, TN, 1962.

1.9-13 P. N. Haubenreich and J. R. Engel, “Experience with the Molten-Salt Reactor Experiment,” Nuclear Applications and Technology, Vol. 8.2, pp. 118-136, 1970.

1.9-14 Abilene Christian University to U.S. Nuclear Regulatory Commission, “Abilene Christian University Construction Permit Application Information Related to the Nuclear Waste Policy Act,” August 17, 2023 (USNRC ADAMS No. ML23230A392).

Chapter 2
Site Characteristics

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 2 SITE CHARACTERISTICS. 2-1

2.1 Geography and Demography 2-1

2.1.1 Site Location and Description 2-1

2.1.2 Population Distribution 2-3

2.1.3 References. 2-7

2.2 Nearby Industrial, Transportation, and Military Facilities 2-40

2.2.1 Locations and Routes 2-40

2.2.2 Air Traffic 2-43

2.2.3 Analysis of Potential Accidents at Facilities 2-47

2.2.4 References. 2-48

2.3 Meteorology 2-60

2.3.1 General and Local Climate 2-60

2.3.2 Site Meteorology 2-62

2.3.3 References. 2-62

2.4 Hydrology 2-77

2.4.1 Drainage and Floodways 2-77

2.4.2 Groundwater 2-78

2.4.3 References. 2-78

2.5 Geology, Seismology, and Geotechnical Engineering 2-89

2.5.1 Regional Geology 2-89

2.5.2 Site Geology 2-89

2.5.3 Seismicity 2-91

2.5.4 Maximum Earthquake Potential 2-92

2.5.5 Vibratory Ground Motion 2-92

2.5.6 Surface Faulting 2-93

2.5.7 Liquefaction Potential. 2-93

2.5.8 References. 2-94

LIST OF TABLES

Table 2.1-1 Abilene Area Populations and Projections 2-10
Table 2.1-2 Number of Jobs per Industry in Abilene 2-10
Table 2.2-1 Abilene Regional Airport Operations Predictions for 2022 2-50
Table 2.2-2 Annual Aircraft Operations for Dyess Air Force Base 2-50
Table 2.3-1 Heaviest Daily Snowfalls in Abilene from 1950 to Present 2-64
Table 2.3-2 Abilene Weather Records, 1948-2022 2-64
Table 2.3-3 Records of Most Consecutive Days of Weather Phenomena in Abilene, Texas 2-65
Table 2.3-4 Maximum Wind Speeds and Reported Damage Events 2-65
Table 2.3-5 Tornado Probabilities 2-66
Table 2.3-6 Number of Hail Events by Size, Taylor County (1/1/1960 to 6/30/2022) 2-66
Table 2.3-7 Ten Largest Recorded Hail Sizes, Taylor County (1/1/1960 to 6/30/2022) 2-67
Table 2.3-8 Number of Hail Events Per Year in Taylor County, 1960-July 2022 2-67
Table 2.3-9 Taylor County Thunderstorm Wind Events (1/1/1960 to 6/30/2022) 2-68
Table 2.5-1 Recommended Maximum Bearing and Skin Friction 2-91
Table 2.5-2 L-Pile Information 2-91
Table 2.5-3 Earthquakes of Magnitude 3 or Greater within 125 Miles (200 km) of the Molten Salt Research Reactor Site 2-96

LIST OF FIGURES

Figure 2.1-1 Borders and Major Cities in 200-mi (322-km) Radius from the Reactor Site . . . 2-11
Figure 2.1-2 Location of Abilene . . . 2-12
Figure 2.1-3 Molten Salt Research Reactor Location in Abilene . . . 2-13
Figure 2.1-4 Streams, Rivers, and Lakes within Several Miles of the Reactor Site . . . 2-14
Figure 2.1-5 Topography of the Abilene Area . . . 2-15
Figure 2.1-6 A 1-mile (1.5 km) Radius around the Molten Salt Research Reactor Site . . . 2-16
Figure 2.1-7 Abilene Christian University Main Campus Boundary . . . 2-17
Figure 2.1-8 Molten Salt Research Reactor Site Layout . . . 2-18
Figure 2.1-9 Science and Engineering Research Center First-floor Layout and Utilization . . . 2-19
Figure 2.1-10 Historic and Projected Population Growth . . . 2-20
Figure 2.1-11 Nearest Resident Population Distribution - 2020 . . . 2-21
Figure 2.1-12 Furthest Resident Population Distribution - 2020 . . . 2-22
Figure 2.1-13 Resident Population Bands - 2020 . . . 2-23
Figure 2.1-14 Nearest Resident Population Distribution - 2022 . . . 2-24
Figure 2.1-15 Furthest Resident Population Distribution - 2022 . . . 2-25
Figure 2.1-16 Resident Population Bands - 2022 . . . 2-26
Figure 2.1-17 Nearest Resident Population Distribution - 2027 . . . 2-27
Figure 2.1-18 Furthest Resident Population Distribution - 2027 . . . 2-28
Figure 2.1-19 Resident Population Bands - 2027 . . . 2-29
Figure 2.1-20 Nearest Resident Population Distribution - 2042 . . . 2-30
Figure 2.1-21 Furthest resident population distribution - 2042 . . . 2-31
Figure 2.1-22 Resident Population Bands - 2042 . . . 2-32
Figure 2.1-23 Historic Population Growth in Abilene and Taylor County . . . 2-33
Figure 2.1-24 Abilene Population Density . . . 2-34
Figure 2.1-25 Location of Universities in Abilene . . . 2-35
Figure 2.1-26 Abilene Hotel Locations and Occupancy . . . 2-36
Figure 2.1-27 Recreational Vehicle Parks and Campgrounds near the Reactor Site . . . 2-37
Figure 2.1-28 Locations of Hospitals in Abilene . . . 2-38
Figure 2.1-29 Concentrations of Employment across Abilene . . . 2-39
Figure 2.2-1 Industrial and Transportation Facilities Near the Site . . . 2-51

LIST OF FIGURES

Figure 2.2-2 Airfields within 10 Miles of the Site 2-52
Figure 2.2-3 Highways, Rail Lines, Dyess Air Force Base, and Abilene Regional Airport 2-53
Figure 2.2-4 Pipelines in the Abilene Area 2-54
Figure 2.2-5 Dyess Air Force Base Arrival Flight Patterns 2-55
Figure 2.2-6 Dyess Air Force Base Departure Flight Patterns 2-56
Figure 2.2-7 Dyess Air Force Base Closed Pattern Flight Patterns 2-57
Figure 2.2-8 Abilene Regional Airport Layout 2-58
Figure 2.2-9 Dyess Air Force Base Layout 2-59
Figure 2.3-2 Average Monthly Temperature Distribution for Abilene from 1970-2021 2-69
Figure 2.3-1 Rainfall Distribution for Abilene from 1948-2021 2-69
Figure 2.3-3 Maximum and Minimum Monthly Temperatures for Abilene from 1948-2021 2-70
Figure 2.3-4 Distribution of Monthly Relative Humidity in the region of Abilene between 1970 and 2021 2-70
Figure 2.3-5 Wind Speed, Direction, and Frequency Measured at Abilene Regional Airport 2-71
Figure 2.3-6 Number of Severe Weather Events within 50 miles of Abilene 2-72
Figure 2.3-7 Design-basis Hurricane Wind Speeds from Regulatory Guide 1.221 2-73
Figure 2.3-8 Hurricane Events Tracked within 50 nmi of the Reactor Site 2-74
Figure 2.3-9 Tornado Magnitude and Distance from Abilene, 1950-2010 2-75
Figure 2.3-10 Map of Tornado Pathways in Abilene from 1950-2013 2-76
Figure 2.4-1 Texas River Basins 2-80
Figure 2.4-2 Topographical Map of Brazos River Basin 2-81
Figure 2.4-3 Streams, Rivers, and Lakes within Several Miles of the Reactor Site 2-82
Figure 2.4-4 Federal Emergency Management Agency-defined Floodways 2-83
Figure 2.4-5 Topography of Abilene Area 2-84
Figure 2.4-6 Topography of Molten Salt Research Reactor Site Area 2-85
Figure 2.4-7 Major Aquifers of Texas 2-86
Figure 2.4-8 Minor Aquifers of Texas 2-87
Figure 2.4-9 Aquifers and Wells Near the Site 2-88
Figure 2.5-1 Geologic Map of Texas 2-98
Figure 2.5-2 Geologic Map of Taylor County 2-99
Figure 2.5-3 Geologic Map of Abilene Area 2-100

LIST OF FIGURES

Figure 2.5-4 Areas Prone to Collapse Sinkholes 2-101
Figure 2.5-5 Topography of the Abilene Area 2-102
Figure 2.5-6 Boring Log 1. 2-103
Figure 2.5-7 Boring Log 2. 2-104
Figure 2.5-8 Boring Log 3. 2-105
Figure 2.5-9 Boring Log 4. 2-106
Figure 2.5-10 Boring Log 5. 2-107
Figure 2.5-11 Boring Log 6. 2-108
Figure 2.5-12 Boring Log 7. 2-109
Figure 2.5-13 Boring Log 8-1 2-110
Figure 2.5-14 Boring Log 8-2 2-111
Figure 2.5-15 Boring Log 9-1 2-112
Figure 2.5-16 Boring Log 9-2 2-113
Figure 2.5-17 Boring Log Summary 2-114
Figure 2.5-18 Seismic Activity Near the Molten Salt Research Reactor Site 2-115
Figure 2.5-19 Causes and Locations of Earthquakes in Texas 2-116
Figure 2.5-20 Injection and Disposal Wells Near the Molten Salt Research Reactor Site. 2-117
Figure 2.5-21 2014 Seismic Hazard Map of Texas 2-118
Figure 2.5-22 Seismic Hazard Curves and Uniform Hazard Response Spectrum. 2-119
Figure 2.5-23 Underground Faults Near Abilene 2-120
Figure 2.5-24 Meers Fault in Oklahoma. 2-121
Figure 2.5-25 Fault Lines in Texas. 2-122
Figure 2.5-26 ASCE 7 MCE_R Hazard Comparison 2-123

CHAPTER 2 SITE CHARACTERISTICS

This chapter provides information regarding the site location and description of the Abilene Christian University (ACU) molten salt research reactor (MSRR). It also includes a discussion of the population in the vicinity of the site and the distribution of infrastructure and natural features in the area. Site characteristics are identified for use in design and analyses discussions in Chapter 3, “Design of Structures, Systems and Components,” Chapter 11, “Radiation Protection and Waste Management,” and Chapter 13, “Accident Analyses.” The site characteristics considered include:

- Geography and demography
- Nearby industrial, transportation, and military facilities
- Meteorology
- Hydrology
- Geology, seismology, and geotechnical engineering

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

The MSRR is located in the Science and Engineering Research Center (SERC), which is roughly centered on a previously developed 15-acre (6-ha) parcel of the ACU main campus in Abilene, Texas. The approximate center point of the MSRR has the following coordinates:

Latitude and Longitude

Latitude: 32° 27' 53" North

Longitude: 99° 42' 26" West

Universal Transverse Mercator Coordinates (meters [m])

Northing: 3,592,168.70 (m)

Easting: 433,537.51 (m)

Zone: 14

Abilene, located in a region of Texas loosely defined as West Central Texas and, as part of the low, rolling plains of West Texas, is flat to slightly rolling. The region is dotted with small bodies of water along with a few reservoirs. According to the Köppen climate classification, Abilene lies at the edge of a humid subtropical climate, with areas to the west being semi-arid. The average annual precipitation total is 25 inches (in.) [64 centimeters (cm)] of rain and 5 in. (13 cm) of snowfall with an annual high temperature average of 77 degrees Fahrenheit (25 degrees

Celsius) and a low temperature average of 53 degrees Fahrenheit (12 degrees Celsius). Abilene is located approximately 200 miles [322 kilometers (km)] east from the New Mexico border and 125 miles (200 km) southwest from the border with Oklahoma. Abilene is also located approximately 142 miles (340 km) west of Fort Worth, 170 miles (275 km) west of Dallas, 160 miles (260 km) east of Odessa, 190 miles (306 km) northwest of Austin, and 146 miles (235 km) southeast of Lubbock. Figure 2.1-1 shows the locations of the nearest states and the largest cities within 200 miles (322 km) of Abilene.

Figure 2.1-2 shows that most of the city of Abilene is situated in the northeast corner of Taylor County, but its northernmost portion, consisting of a 4,200 acre (1,700-ha) reservoir, Lake Fort Phantom Hill, and the French M. Robertson maximum-security state prison, extends into the southeast corner of Jones County. Abilene is the largest city in Taylor County in both size and population, and serves as the county seat.

Figure 2.1-3 shows the ACU main campus is in the northeastern region of Abilene and inside a highway loop formed by Interstate 20 to the north, Texas (TX) 322 loop to the east, and United States (U.S.) 83/84 on the west. Abilene's downtown area is roughly in the center of the city, approximately 1.5 miles (2.4 km) southwest of campus. Abilene Regional Airport is nearly four miles (6.4 km) south of the site. Dyess Air Force Base lies outside the loop on the city's southwest edge, about nine miles (14.5 km) from campus. A Union Pacific railway line lies 1.2 miles (1.9 km) south of the campus, and a short local line operated by Southern Switching Company runs perpendicular to the Union Pacific line just under one mile (1.6 km) to the west of the campus. The only rail yard in Abilene is where the two lines cross downtown nearly 1.5 miles (2.4 km) southwest of ACU.

The city of Abilene is 112.2 square miles [291 square km (km^2)] with surface water making up 5.4 square miles (14 km^2) or 4.81 percent of Abilene's area. Figure 2.1-4 shows several north-flowing creeks that branch through Abilene. Cedar Creek lies 0.6 miles (1 km) west of the site and the much smaller Rainy Creek lies 0.7 miles (1.1 km) to the east. The town's three major bodies of water, shown in Figure 2.1-4, are all man-made. Lake Fort Phantom Hill, located seven miles (11 km) north of the site, was created in 1938 by a dam constructed on Elm Creek [Reference 2.1-1]. It is used as a water source for the city as well as for recreation. Lytle Lake is just over 2 miles (3.2 km) south of campus. It is controlled by Lytle Lake Water Control and Improvement District and is not accessible to the public except as a water source for landscaping. Nearly six miles (10 km) south is Kirby Lake, which is used for both municipal and recreational purposes [Reference 2.1-2]. The region of Abilene has a rather featureless topography as illustrated in Figure 2.1-5, showing the area terrain is relatively flat and the elevation of the site is 1,733 feet (ft) (528 m) above sea level.

Near the reactor, there are no structures tall enough to affect diffusion and dispersion of airborne effluents. For the MSRR, the stack height for radioactive effluents is 50 ft (15.2 m) above ground level. The nearest tall buildings are downtown, more than 1.5 miles (2.4 km) away.

Figure 2.1-6 shows the MSRR location and the area within a one-mile (1.6 km) radius.

Figure 2.1-7 shows the MSRR is located on the southeast corner of the ACU main campus.

2.1.1.2 Boundary and Zone Area Maps

The MSRR site is an approximately 15-acre (6-ha) ACU property bounded by four city streets: East North 16th Street at the northern boundary, North Judge Ely Boulevard at the eastern boundary, East North 13th Street at the southern boundary, and Avenue F at the western boundary, as shown in Figure 2.1-8. Road access to the MSRR site is from North Judge Ely Boulevard and East North 16th Street. The site envelops the Science and Engineering Research Center (SERC), along with other ACU buildings used for research and academic purposes. Abilene Christian University property continues to the north of the site with parking lots and residence halls nearest the vicinity. To the east are business establishments, including shops and restaurants. To the south and west are private residences. The reactor is located within the SERC facility.

The SERC first-floor plan is shown in Figure 2.1-9. It includes public and non-public (controlled) areas. The public area contains a lobby, reception desk, restrooms, and a training room. The non-public area of the SERC includes a number of laboratories, a machine shop, and the MSRR operations area, or the exclusion area boundary, which is anticipated to be the area directly under the U.S. Nuclear Regulatory Commission (NRC) facility operating license. The MSRR operations area consists of the research bay, reactor control room, radiochemistry lab, and dress-out room. Based on preliminary calculations, the MSRR emergency planning zone (EPZ) is within the operations area, certainly within the site, and will be precisely delineated in the Operating License application.

2.1.2 Population Distribution

This section provides population distribution data for resident and transient populations for the area within five miles (8 km) of the center point of the site for the following years:

- Latest U.S. census data from 2020
- Beginning of the construction permit application period (2022)
- Five years after initial construction permit application period (2027)
- Approximate end of the operating license period (2042)

Estimates and projections of the resident population around the site are divided into five radial bands, represented by concentric circles. The distances from the center point of the reactor to the radial sector bands are: 0 to 0.5 miles (0 to 0.8 km), 0.5 to one mile (0.8 to 1.6 km), one to two miles (1.6 to 3.2 km), two to three miles (3.2 to 4.8 km), and three to five miles (4.8 to 8 km). These bands were then further subdivided into 16 radial directional sectors, the centers of which extend from the MSRR

coordinates towards the edge of the last band along the cardinal, intercardinal, and secondary intercardinal compass directions, each 22.5 angular degrees wide. The population data is used in dose analyses in Chapters 11 and 13.

2.1.2.1 Resident Population

Based on the 2020 U.S. census, the population of the City of Abilene was 125,182 with an average population density of 1107.5 people per square mile (427 km^2). Abilene experienced steady population growth from the 1970s to 2000. Population growth has been minimal since the turn of the century, however, gaining only about 3,200 residents by 2013. This minimal population growth is expected to continue over the next two decades with a projected population of 131,000 by the year 2030, only 12,000 more residents than today [Reference 2.1-3]. Historical and projected population growth is shown in Figure 2.1-10. The population projection from 2013 to 2020 provided by the city planners in 2014 was nearly exact. They projected an Abilene population of 125,179 for 2020, and the 2020 U.S. census counted 125,182. Using the demonstrated annual growth rate projection from the city planners of less than 1 percent, the population of Abilene is estimated to be 128,900 in 2027 and 140,300 in 2042 [Reference 2.1-3].

To establish a baseline detailed population distribution, 2020 TIGER/Line Shapefiles from the U.S. Census were tabularly joined by GEOID20 with 2020 Census Redistricting Summary Files Total Population from Table P1. The populations in the census blocks were then geographically united with the radial bands and the radial directional sectors to estimate the populations within the bands and sectors. The total population for each radial directional sector was then recalculated as the area of each new radial directional sector, divided by the original census block area, and multiplied by the census block population. This created a total population estimate that was the proportion of the area of each radial directional sector relative to the area of the census block that provided the population. The same technique was used to provide population estimates for each radial band. The results are shown in Figure 2.1-11, Figure 2.1-12, and Figure 2.1-13.

Population projections published in 2018 by The Texas Demographic Center were used for future population predictions, which were produced using the Cohort Component projection technique for each county in the state. The projections for 2020, 2022, 2027, and 2042 for the four counties (Taylor, Jones, Callahan, and Shackleford) that overlap the bands and sectors were acquired, and projected population change was calculated between each subsequent year for each county. The growth rates for each year from 2022 onward were then applied to the bands and sectors from the 2020 population to arrive at projections for the later years using the same technique described above. The results for 2022 are shown in Figure 2.1-14, Figure 2.1-15, and Figure 2.1-16. The results for 2027 are shown in Figure 2.1-17, Figure 2.1-18, and Figure 2.1-19. The results for 2042 are shown in Figure 2.1-20, Figure 2.1-21, and Figure 2.1-22. All analysis was done in two dimensional planar coordinates, projected to Texas North Central State Plane (EPSG Code 2276), using the US Survey Foot. Analysis was completed using the GDAL/OGR Geospatial Data Abstraction software, version 3.4.1, and visualized

using QGIS Version 3.22.3 “Białowieża,” revision 1628765ec7. Table 2.1-1 summarizes the 2020 area populations by radial bands surrounding the MSRR with projected populations by distance for 2022, 2027, and 2042.

In Taylor County, there were 138,034 people counted in the 2020 U.S. Census, about 90 percent of them in the Abilene Census County Division (CCD) in the northeastern corner of the county. Figure 2.1-23 shows the population of Abilene has historically represented approximately 90 percent of the total population in Taylor County and has been stable to within a couple of percentage points from 1970 to 2013 [Reference 2.1-3]. The remaining 10 percent of the county population live in the other two CCDs, Merkel and Jim Ned, which cover 730 square miles (1886 km^2) - almost 80 percent of the county land area. The Abilene population densities in the CCDs are shown in Figure 2.1-24.

Jones County is home to 20,083 people. As in Taylor County, the most populous CCD is the one that includes part of Abilene, the Hawley-Noodle CCD. It accounts for 46 percent of the county population. The small towns of Anson and Stamford account for most of the population of the CCDs of the same names; together, these CCDs make up about 34 percent of the county population. The remaining two CCDs are very sparsely populated.

The nearest permanent residences to the site are located on the west side of Avenue F and are no closer than 400 feet (120 m) from the exhaust stack for the reactor as can be seen in Figure 2.1-7. The nearest dormitory from the stack is located 460 feet (140 m) north across East North 16th Street.

2.1.2.2 Transient Population

In addition to the permanent residents around the project site, people come to Abilene temporarily for activities such as employment, education, recreation, medical care, and lodging. The transient population within five miles (8 km) of the site will be investigated when updated U.S. Census data is available and provided in the Operating License application.

The 2020 U.S. Census counted people living in transitory or temporary housing, but the data have not yet been released [Reference 2.1-4]. A significant non-permanent population within a mile of the MSRR site is college students, most of whom leave campus for the summer months. Other types of temporary accommodations are hotels, campgrounds, and hospitals.

There are 3,554 students who study at the Abilene Campus in the spring and fall semesters. This population drops to 50-100 students during the summer months [Reference 2.1-5]. Another university in Abilene, with a largely nine-month student population, is Hardin-Simmons University, which sits one mile (1.6 km) northeast of the MSRR site. Of its approximately 2,100 students, 1,600 are undergraduates and approximately 1,200 students live in college-owned or operated housing, including 87 percent of freshmen. The school’s population, similar demographically to ACU’s, can be estimated to drop to between 175 and 375 students during the summer [Reference 2.1-6]. McMurry University is home

to approximately 1,100 students, most of whom live on campus [Reference 2.1-7]. This school is 3.5 miles (5.6 km) southwest of the site. The summer student population drops to approximately 100. Figure 2.1-25 shows the locations of these universities.

A search for hotels on Google Maps shows that just outside a mile (1.6 km) from the MSRR site are 10 hotels and inns. Eight are northwest of the site north of Interstate-20 (I-20) and two are due south. Two miles (3.2 km) northwest of the site is a cluster of six hotels and motels, also north of I-20. Two more hotels are located 3 to 3.5 miles (4.8 to 5.6 km) to the southwest, on Business Route 20. The rest of Abilene's 42 hotel properties lie four or more miles (6.4 km) from the reactor site. In all, the 42 inns and hotels in Abilene have approximately 3,200 rooms as shown in Figure 2.1-26. Upper-market hotels have maintained an average occupancy of about 72 percent since 2016, excluding 2020, and are now averaging 76 percent [Reference 2.1-8].

Of the above 18 hotels, 14 are listed by the Abilene Convention and Visitors Bureau, which shows their capacity. The average number of rooms among these 14 hotels is 87 [Reference 2.1-9]. At 72-percent occupancy (adjusted for type of property) and assuming two guests per room, it can be estimated that around 2,256 visitors may be within 3.5 miles (5.6 km) of the reactor site at a given time, with variations according to season, reasons for travel, and other factors.

Abilene is home to 15 campgrounds and recreational vehicle (RV) parks, according to a search using Google maps and shown in Figure 2.1-27. The closest one is approximately one mile (1.6 km) east of the site. The next nearest is 1.5 miles (2.4 km) north by northwest. Another is sited almost four miles (6.4 km) to the south. These three campgrounds and RV parks each have approximately 100 sites. Five more are just under five miles (8 km) from the MSRR site, one west by northwest, and four to the southeast. The remaining seven campgrounds are more than five miles (8 km) from the reactor and are slightly smaller than the three above. Thus, within two miles (3.2 km) of the MSRR, as many as 200 groups and individuals may be visiting. Within five miles (8 km), as many as 550 camping and RV sites may be occupied by one or several visitors.

Within a five-mile (8-km) radius of the MSRR site, the main medical facility is Hendrick Medical Center, which has 347 beds [Reference 2.1-10]. Not included here are small walk-in clinics and outpatient surgery facilities. Hendrick Medical Center is located 1.4 miles (2.3 km) northwest of the MSRR site as illustrated in Figure 2.1-28.

2.1.2.3 Density of Workplaces and Major Industries

Workplaces are also relevant in estimating the number of people near the MSRR site at any given time. Figure 2.1-29 shows the concentration of jobs per square mile across Abilene from the most recent census data available. The total number of jobs in 2019 was 56,651, for an average of 530 jobs per square mile (205 km^2). During the day, the number of people in all of Abilene increases by 13,697 [Reference 2.1-11].

Based on those data, there are approximately 1,500 jobs within a one-mile (1.6-km) radius of the reactor, with ACU being the largest employer near the site. The major industries in Abilene are shown in Table 2.1-2. The largest industry is health care; the Hendrick Health System and the Supported Living Center are among Abilene's largest employers. Retail trade is the second largest industry, with many retail jobs distributed among the 10 malls in the city limits. Accommodations, food services, and education follow closely.

2.1.3 References

2.1-1 City of Abilene, Texas, "Our lakes," Accessed on June 18, 2020. [Online]. Available: https://abilenetx.gov/605/Our-Lakes

2.1-2 Texas Parks & Wildlife, "Kirby Lake public access facilities," Texas Parks & Wildlife. Accessed on June 25, 2020. [Online]. Available: https://tpwd.texas.gov/fishboat/fish/recreational/lakes/kirby/access.phtml

2.1-3 City of Abilene, Texas, Community Services Department, Parks and Recreation Division, "Abilene parks, recreation & senior facilities master plan," 2014. [Online]. Available: https://www.abilenetx.gov/DocumentCenter/View/1820/Introduction-PDF

2.1-4 U.S. Census Bureau, "2020 census: Conducting and motivating the count: Counting people at transitory locations," Accessed on Feb. 21, 2022. [Online]. Available: https://www.census.gov/programs-surveys/decennial-census/decade/2020/planning-management/count/transitory-locations.html

2.1-5 Abilene Christian University, "ACU achieves record enrollment for fourth year in a row," Sept. 8, 2021. Accessed on Feb. 21, 2022. [Online]. Available: https://www.acu.edu/2021/09/08/acu-achieves-record-enrollment-for-fourth-year-in-a-row/

2.1-6 "Hardin-Simmons University student life," U.S. News and World Report. Accessed on Feb. 16, 2022. [Online]. Available: https://www.usnews.com/best-colleges/hardin-simmons-university-3571/student-life

2.1-7 cMurry University public website. Accessed on March 3, 2022. Summer student population estimate based on ACU and Hardin-Simmons University population changes.

2.1-8 S. Ali and D. Weisvein, "Presale: Abilene Convention Center Hotel Development Corp," S&P Global Ratings, 2021. [Online]. Available: https://www.spglobal.com/_assets/documents/ratings/research/12080615.pdf

2.1-9 Abilene Convention and Visitors Bureau, "Accommodations." Accessed on Mar. 7, 2022. [Online]. Available: https://www.abilenevisitors.com/accommodations/category:3

2.1-10 COVID-19 hospital capacity in Taylor County and surrounding area, Reno Gazette Journal, U.S. Department of Health and Human Services. [Online]. Available: https://data.rgj.com/covid-19-hospital-capacity/texas/48/taylor-county/48441/

2.1-11 City-Data.com, “Abilene, Texas.” Accessed on March 8, 2022. [Online]. Available: http://www.city-data.com/city/Abilene-Texas.html

2.1-12 2020 Census demographic map viewer - population density, U.S. Census Bureau. [Online]. Available: https://www.census.gov/library/visualizations/2021/geo/demographicmapviewer.html

2.1-13 OSM, “OpenStreetMap,” OpenStreetMap contributors, 2021. Accessed on Feb. 13, 2022. [Online]. Available: https://www.openstreetmap.org

2.1-14 NGA, “OpenStreetMap Bright Basemap,” United States Department of Defense, National Geospatial-Intelligence Agency, 2022. Accessed on Feb. 13, 2022. [Online]. Available: https://osm.gs.mil/features/base-map

2.1-15 USGS, “National Hydrography Dataset,” United States Geological Survey, National Geospatial Program, 2022. Accessed on Feb. 13, 2022. [Online]. Available: https://apps.nationalmap.gov/downloader/

2.1-16 FAA, “Airports, Runways, Seaplane Bases, Heliports,” United States Department of Transportation, Federal Aviation Administration, 2021. Accessed on Feb. 13, 2022. Available: http://nationalmap.usgs.gov/

2.1-17 TXDOT, “Geospatial Roadway Inventory Database,” Texas Department of Transportation, 2022. Accessed on Feb. 19, 2022. [Online]. Available: https://opendata.arcgis.com/api/v3/datasets/d4f7206d27af4358acb70cb1cc819d10_0/downloads/data?format=geojson&spatialRefId=4326

2.1-18 FRA, “Main Track Centerlines,” United States Department of Transportation, Federal Railways Administrations, 2021. Accessed on Feb. 13, 2022. [Online]. Available: http://nationalmap.usgs.gov/

2.1-19 USCB, “2020 Census State Redistricting Data (Public Law 94-171) Summary File, Table P1,” United States Department of Commerce, Bureau of the Census, 2020. Accessed on Feb. 20, 2022. [Online]. Available: https://data.census.gov/cedsci/table

2.1-20 USCB, “2020 TIGER/Line Tab Block Shapefiles,” United States Department of Commerce, Bureau of the Census, 2020. Accessed on Feb. 20, 2022. [Online]. Available: https://www2.census.gov/geo/tiger/TIGER2020/TABBLOCK20/tl_2020_48_tabblock20.zip

2.1-21 TDC, “Projections of the Population of Texas and Counties in Texas by Age, Sex, and Race/Ethnicity for 2010-2050,” The Texas Demographic Center & The University of Texas at San Antonio, 2018. Accessed on Feb. 20, 2022. [Online]. Available: https://www.demographics.texas.gov/Data/TPEPP/Projections/

2.1-22 OnTheMap, United States Census Bureau. Accessed on Dec. 30, 2021. [Online]. Available: https://onthemap.ces.census.gov/

2.1-23 “Abilene downtown hotel assessment,” JLL, April 2016. [Online]. Available: https://3tkxep1rkjdb3lyf3b40suea-wpengine.netdna-ssl.com/wp-content/uploads/2018/08/Hotel-Market-Study.pdf

Table 2.1-1 Abilene Area Populations and Projections

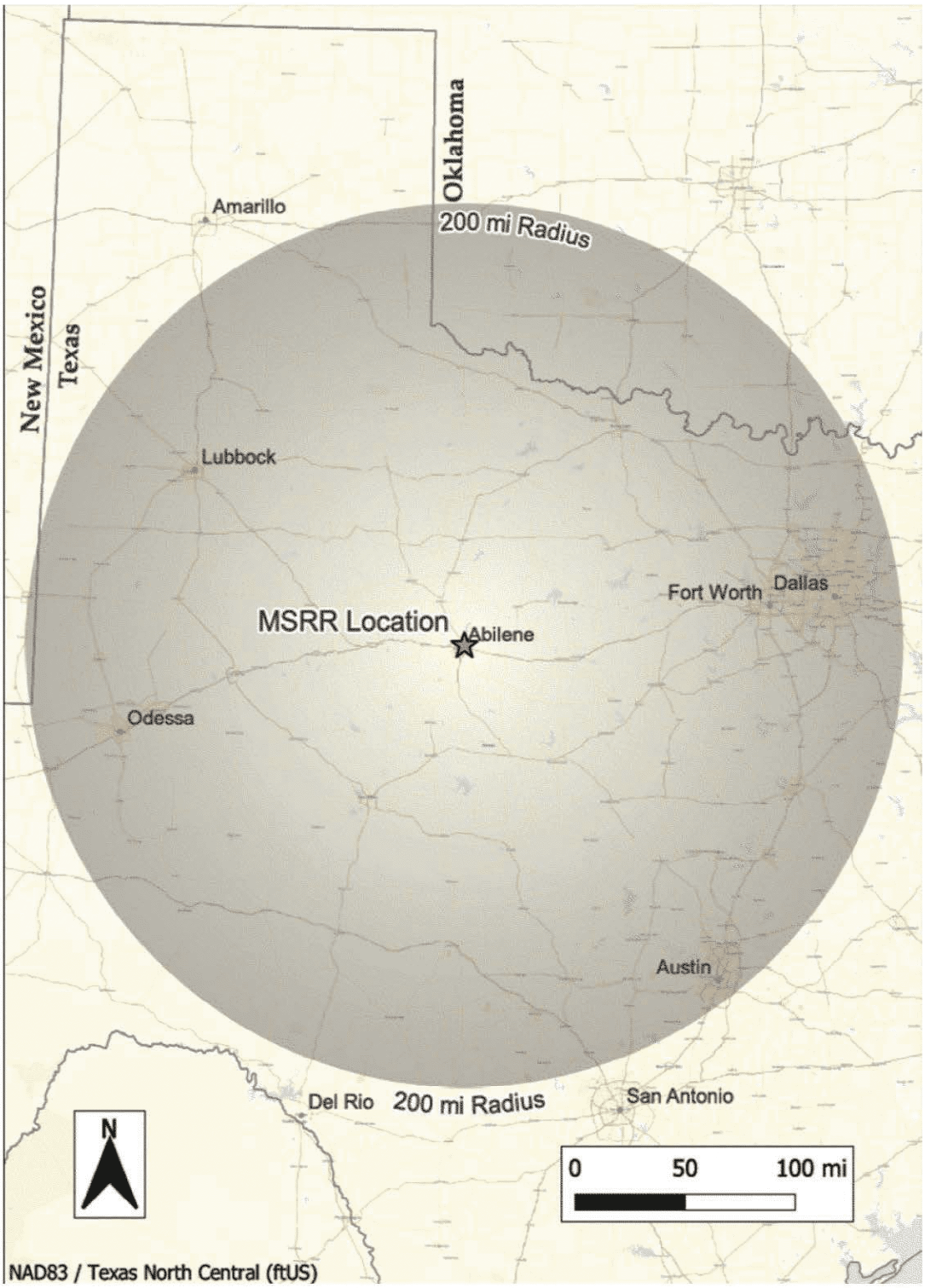
	Distance Band from Site (miles)				
Year	**0-0.5**	**0.5-1**	**1-2**	**2-3**	**3-5**
2020	3,554	5,323	8,953	18,554	42,284
2022	3,592	5,380	9,050	18,754	42,671
2027	3,676	5,506	9,263	19,195	43,513
2042	3,872	5,799	9,757	20,218	45,357

Table 2.1-2 Number of Jobs per Industry in Abilene

Industry	Number of Jobs	% of All Jobs
Agriculture, forestry, fishing, and hunting	203	0.4
Mining, quarrying, and oil and gas extraction	1355	2.7
Utilities	374	0.7
Construction	2638	5.3
Manufacturing	2547	5.1
Wholesale trade	2192	4.4
Retail trade	6205	12.4
Transportation and warehousing	1537	3.1
Information	680	1.4
Finance and insurance	2355	4.7
Real estate and rental and leasing	770	1.5
Professional, scientific, and technical services	1804	3.6
Management of companies and enterprises administration and support	317	0.6
Waste management and remediation	2252	4.5
Education services	5049	10.1
Health care and social assistance	9819	19.7
Arts, entertainment, and recreation	792	1.6
Accommodation and food services	5431	10.9
Other services (excluding public administration)	1594	3.2
Public administration	1996	4.0

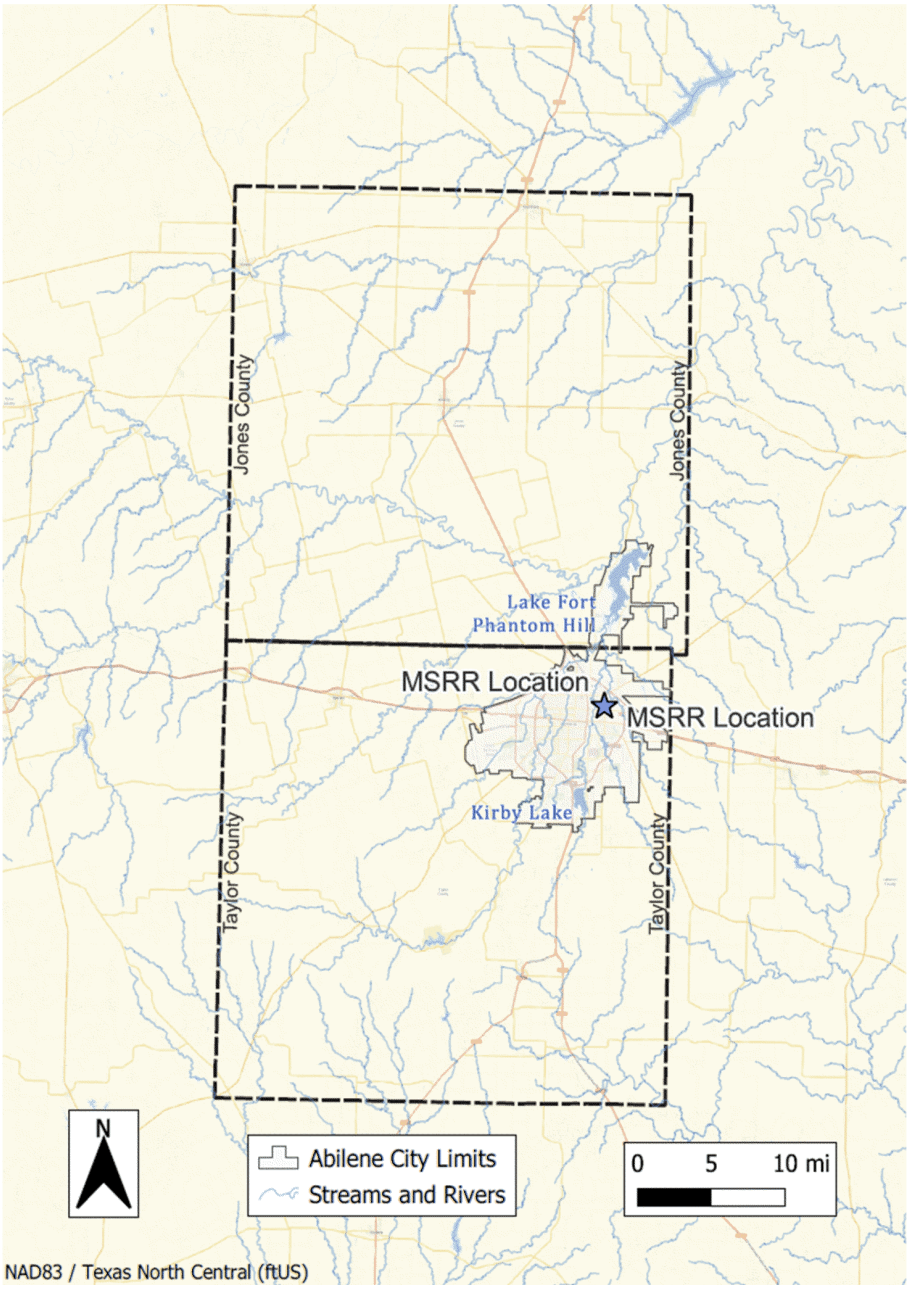
Data from CityData.com [Reference 2.1-11]

Figure 2.1-1 Borders and Major Cities in 200-mi (322-km) Radius from the Reactor Site



The purple star is the MSRR. [Reference 2.1-13], [Reference 2.1-14]

Figure 2.1-2 Location of Abilene



The City of Abilene is outlined in grey. [Reference 2.1-13], [Reference 2.1-14], [Reference 2.1-15]

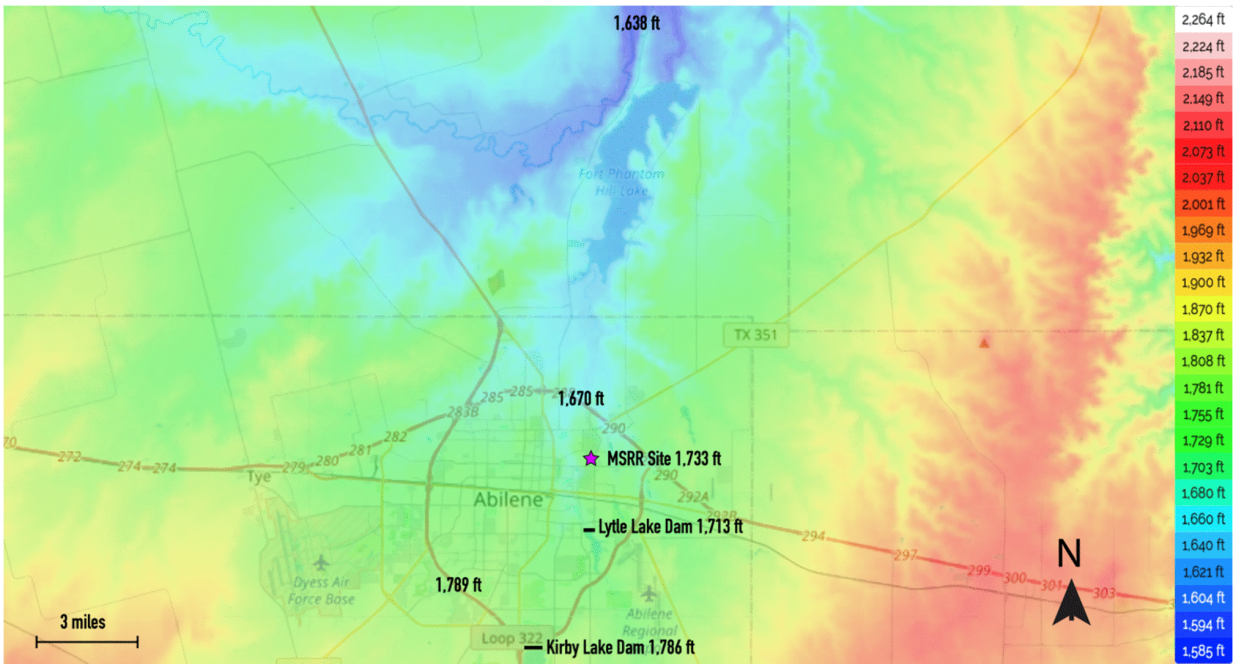
Figure 2.1-3 Molten Salt Research Reactor Location in Abilene

ACU campus is outlined and the MSRR is marked with a star. [Reference 2.1-13], [Reference 2.1-14], [Reference 2.1-15], [Reference 2.1-16]

Figure 2.1-4 Streams, Rivers, and Lakes within Several Miles of the Reactor Site

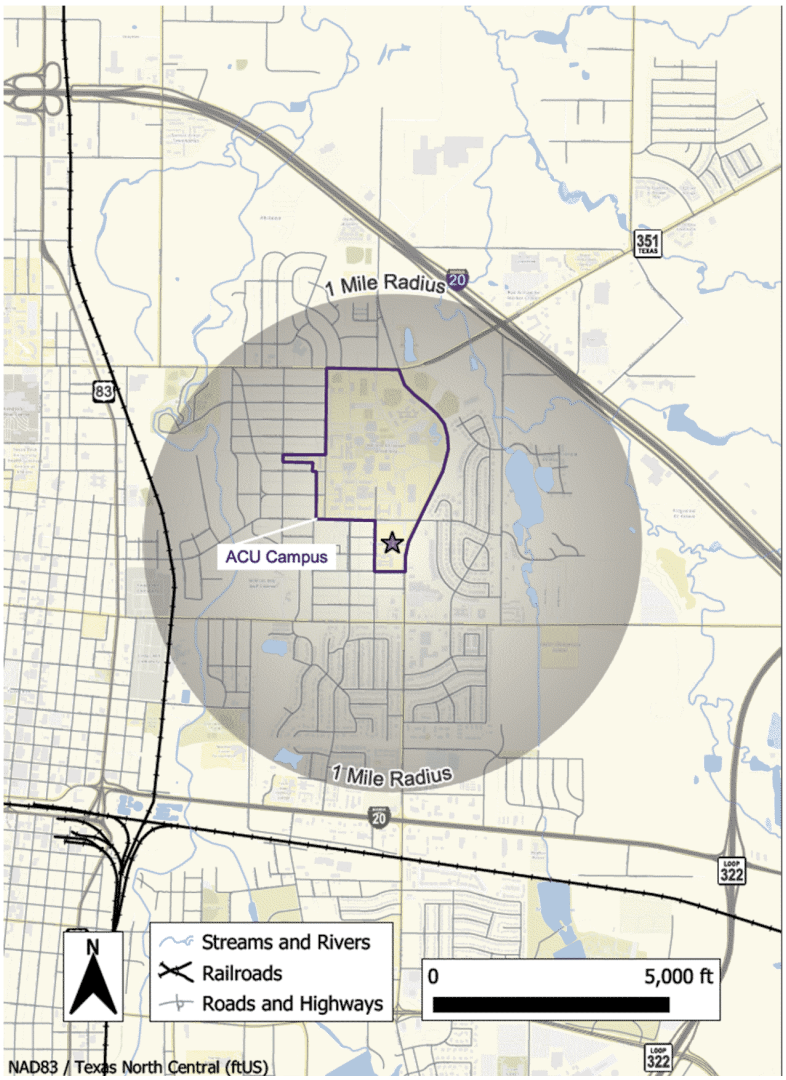
Data from OpenStreetMap [Reference 2.1-13], NGA [Reference 2.1-14], and National Hydrography Dataset [Reference 2.1-15]

Figure 2.1-5 Topography of the Abilene Area



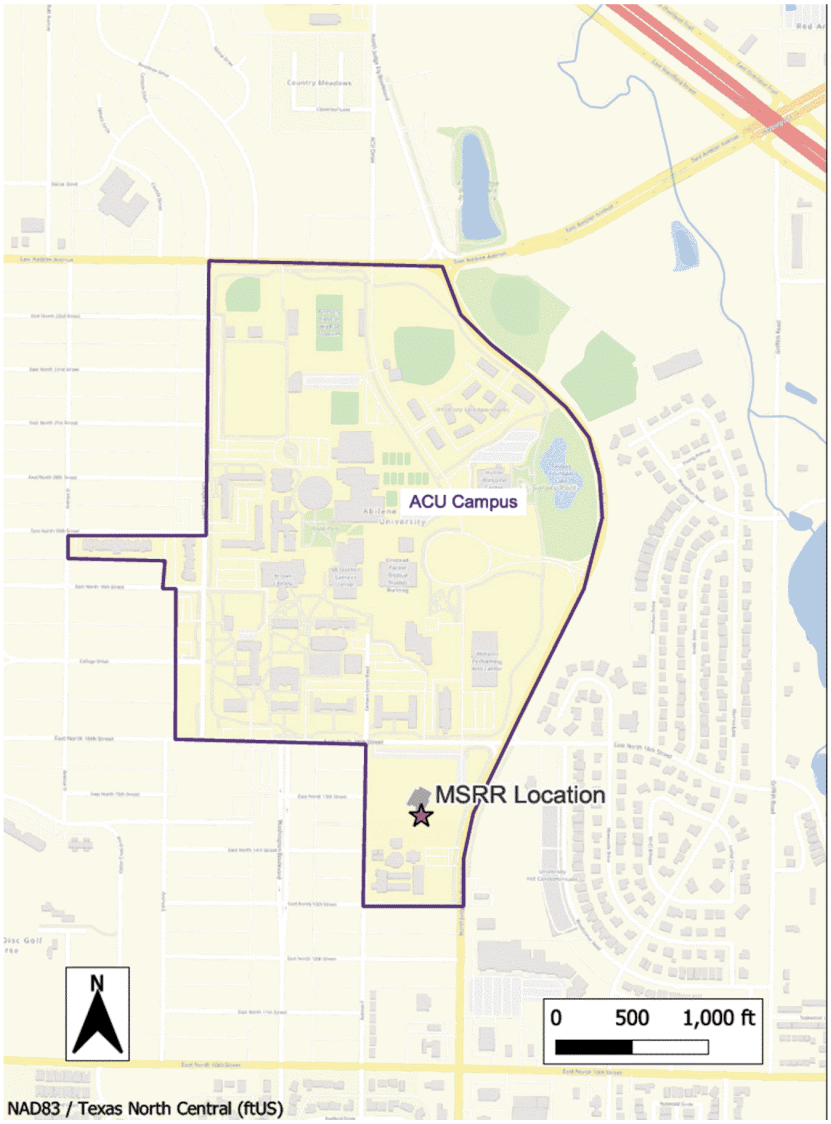
OpenStreetMap provided topography rendering. Elevation data rendered by ERIT DEM, referenced to WGS84 and EGM96.

Figure 2.1-6 A 1-mile (1.5 km) Radius around the Molten Salt Research Reactor Site



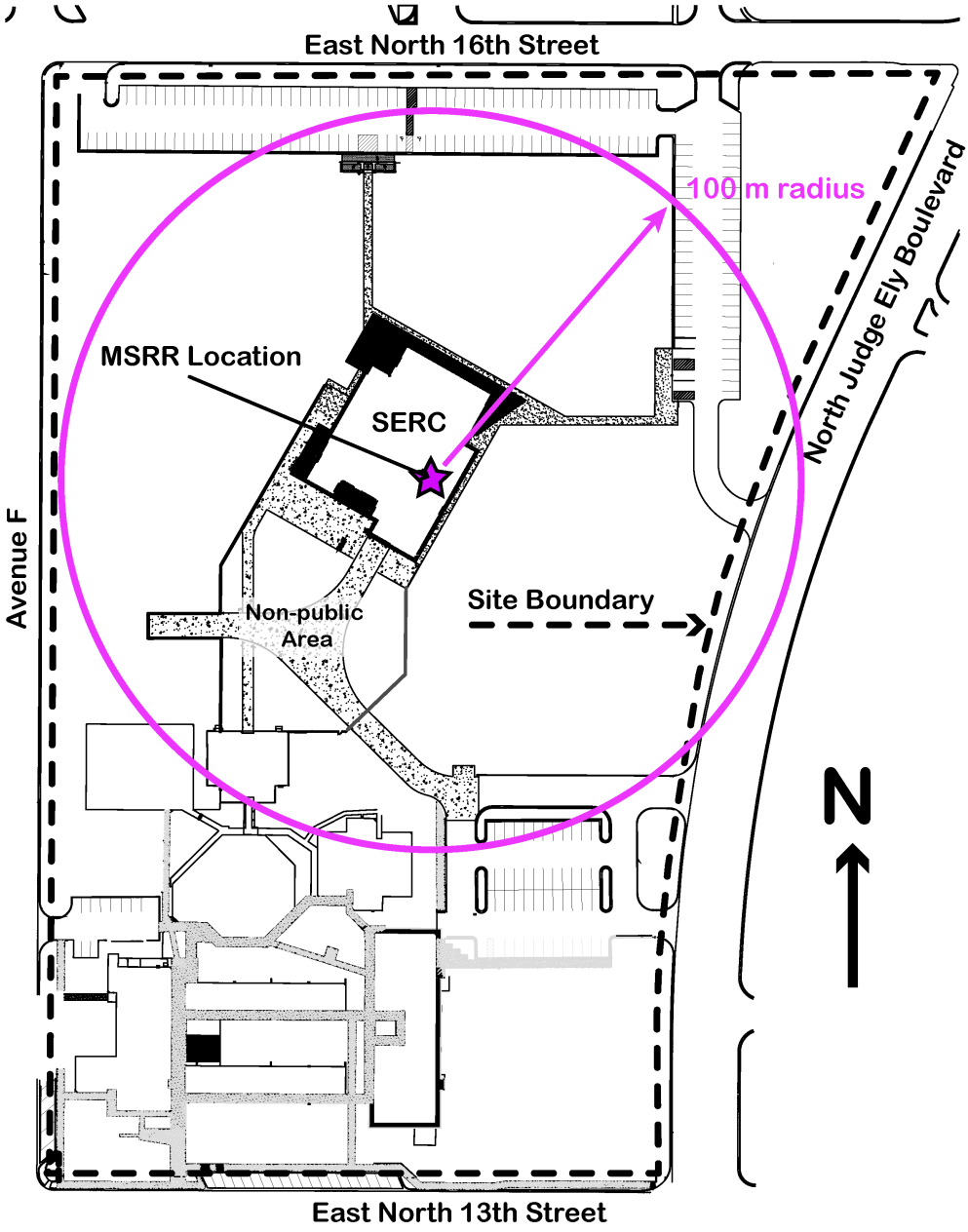
Data from OpenStreetMap [Reference 2.1-13], NGA [Reference 2.1-14] TXDOT [Reference 2.1-17], and FRA [Reference 2.1-18]

Figure 2.1-7 Abilene Christian University Main Campus Boundary



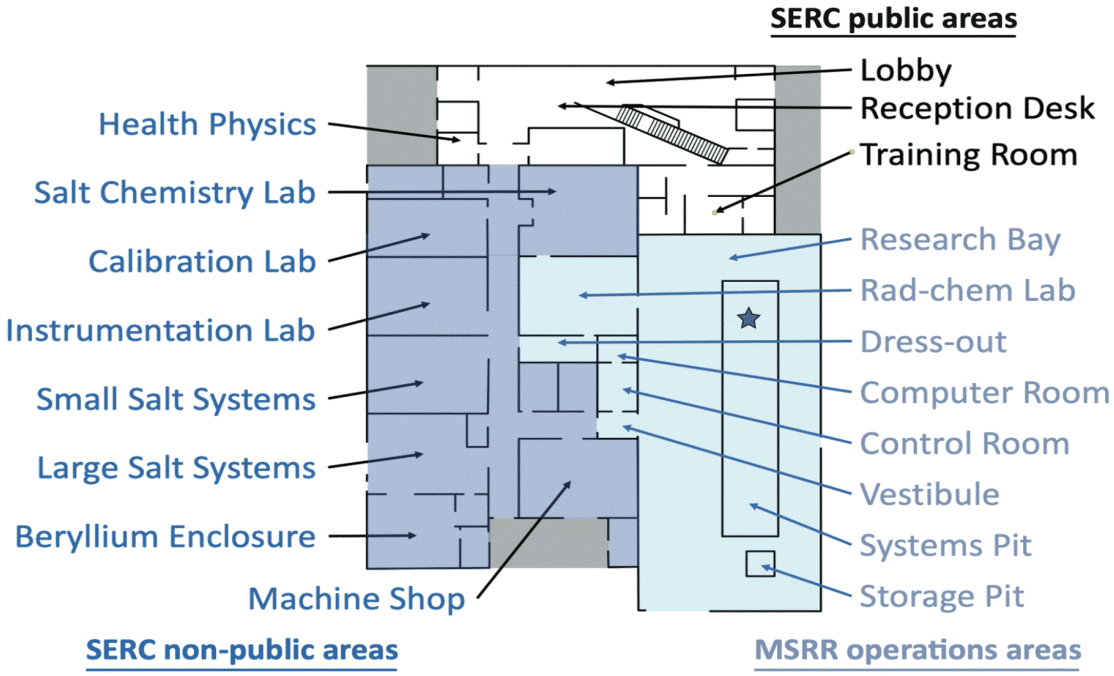
Data from OpenStreetMap [Reference 2.1-13] and NGA [Reference 2.1-14]

Figure 2.1-8 Molten Salt Research Reactor Site Layout



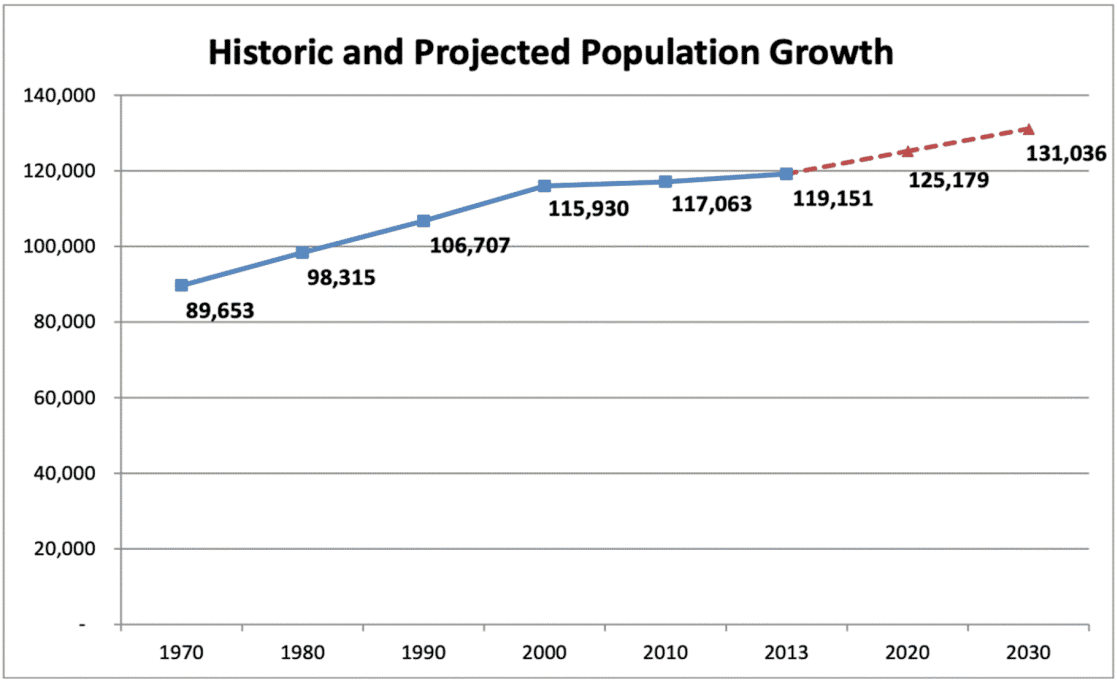
From ACU architectural archives

Figure 2.1-9 Science and Engineering Research Center First-floor Layout and Utilization



Developed using ACU architectural archives

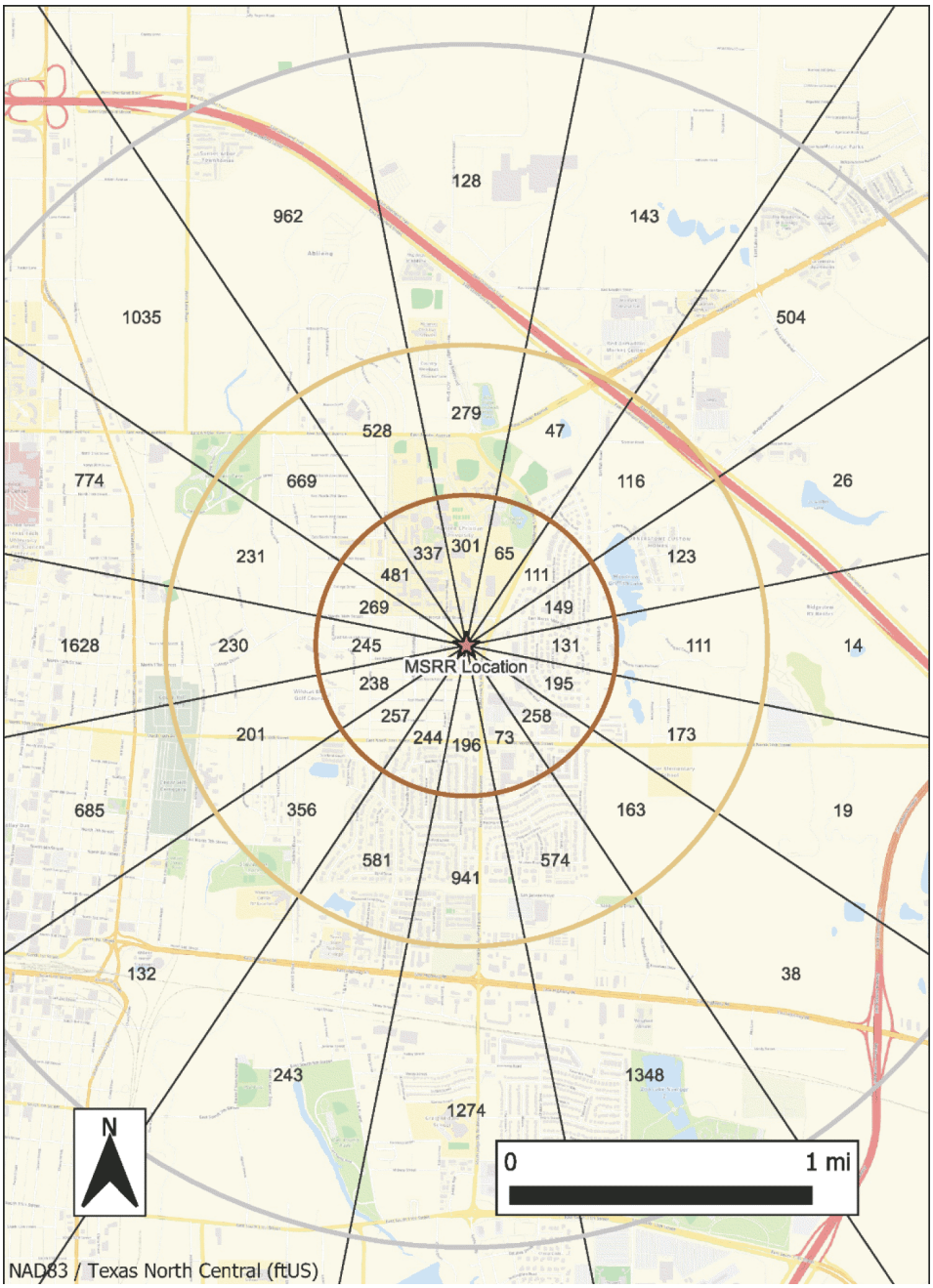
Figure 2.1-10 Historic and Projected Population Growth



1970, 1980, 1990, 2000 Source: US Decennial Census; 2013 Source: City of Abilene Nielsen 2013 Report; 2020, 2030 Source: Texas Water Development Board 2016 Regional Water Plan

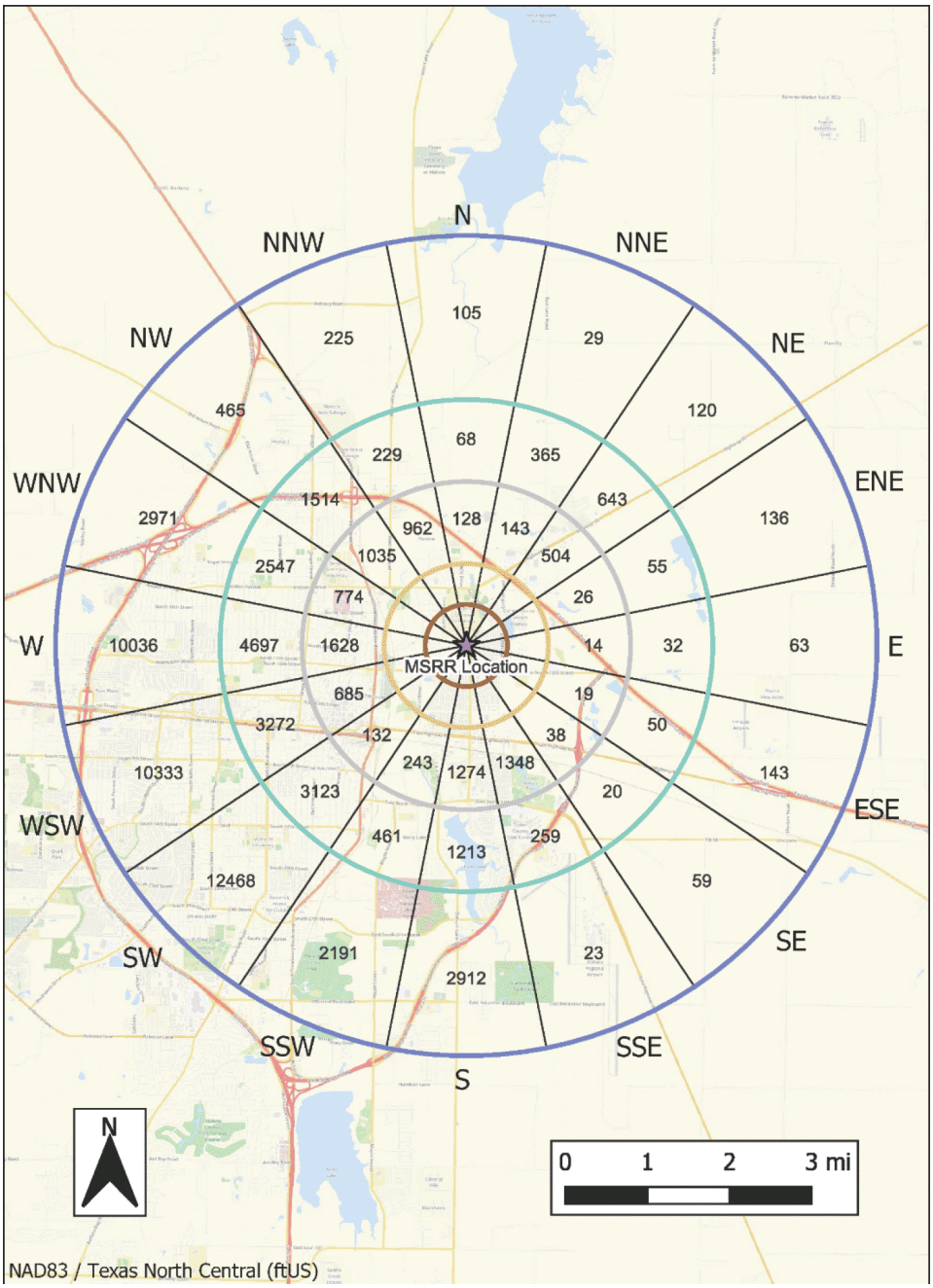
From “Abilene parks, recreation & senior facilities master plan.” [Reference 2.1-3]

Figure 2.1-11 Nearest Resident Population Distribution - 2020



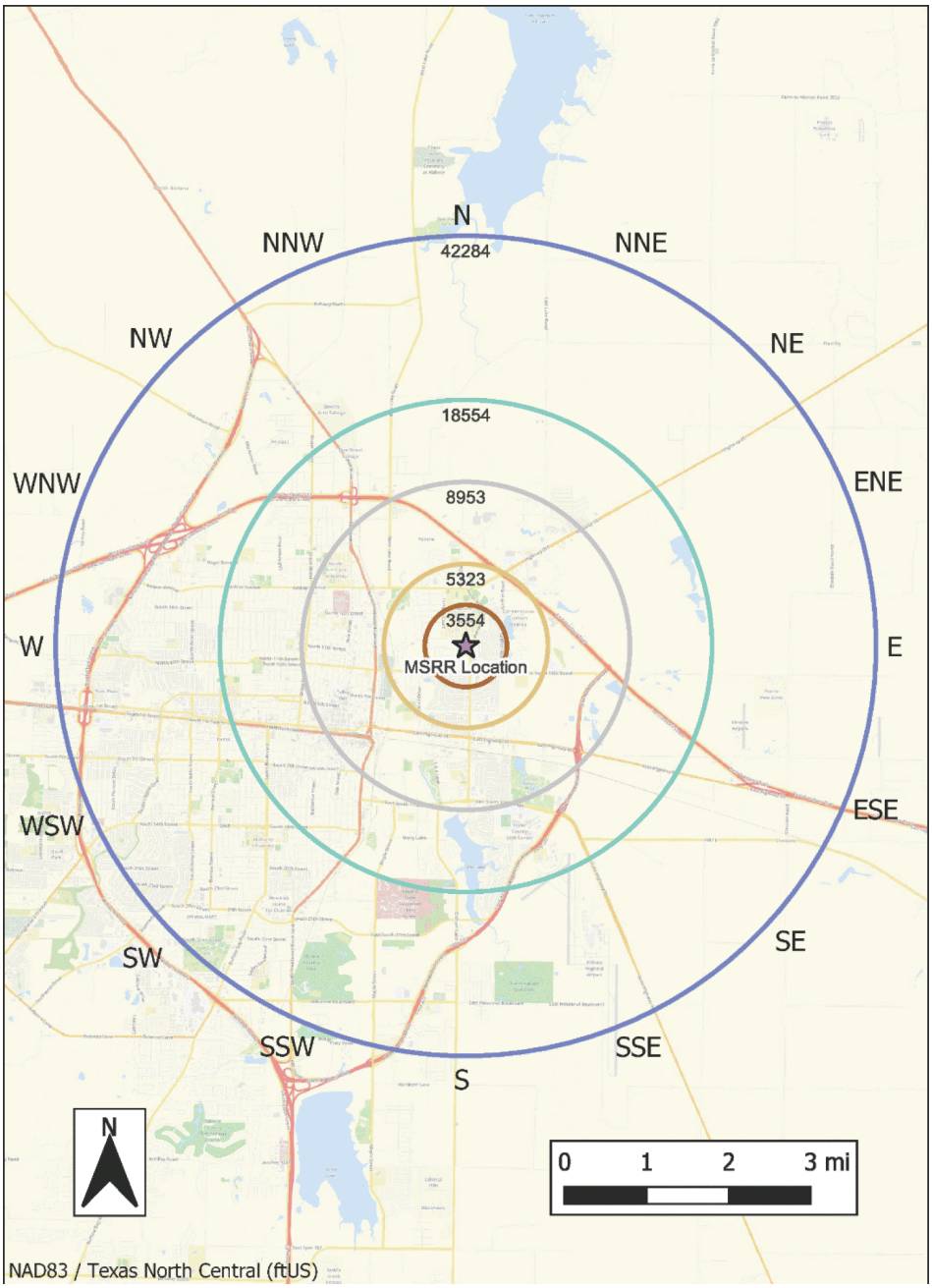
Data from USCB [Reference 2.1-19], [Reference 2.1-20]

Figure 2.1-12 Furthest Resident Population Distribution - 2020



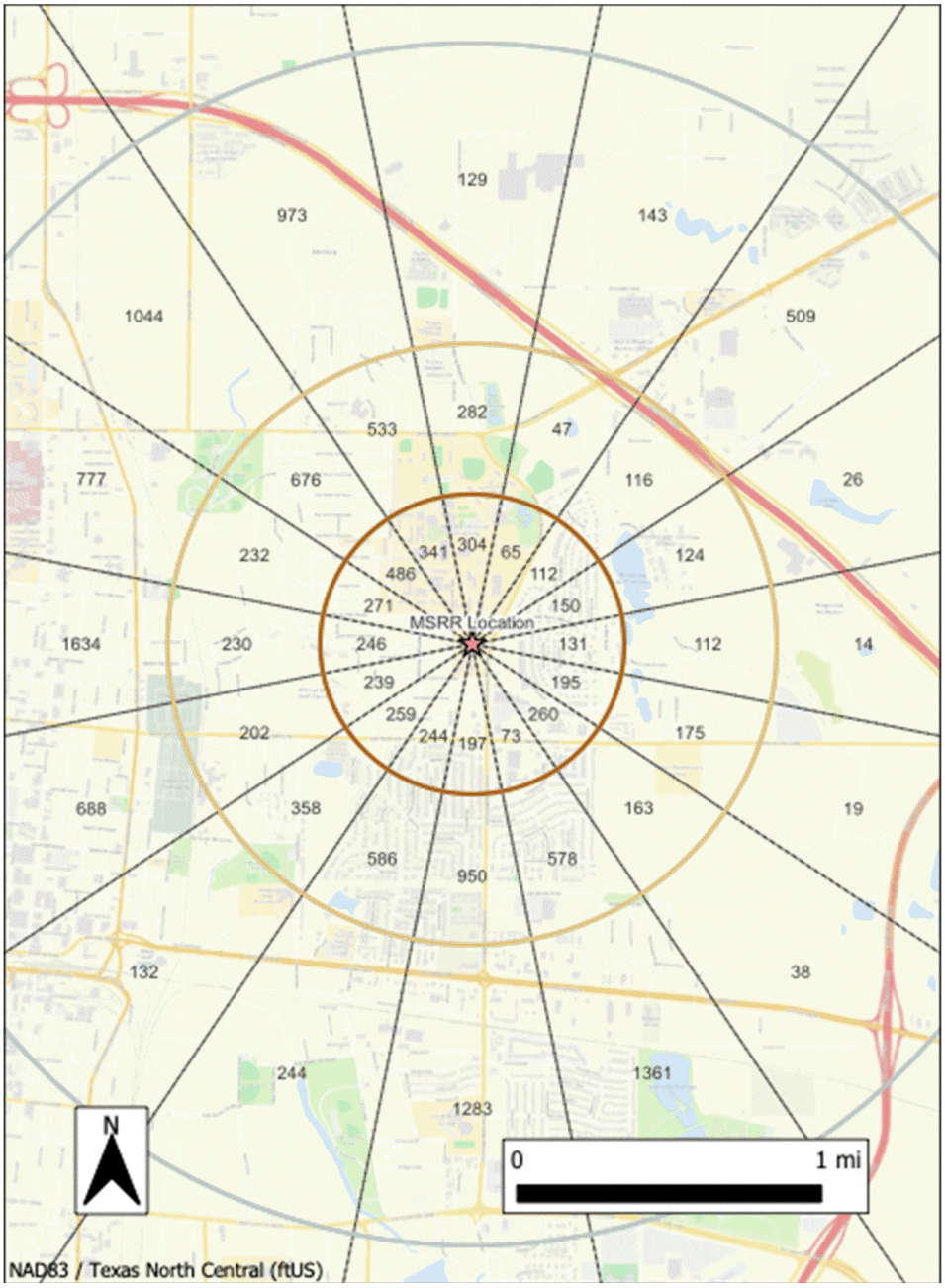
Data from USCB [Reference 2.1-19], [Reference 2.1-20]

Figure 2.1-13 Resident Population Bands - 2020



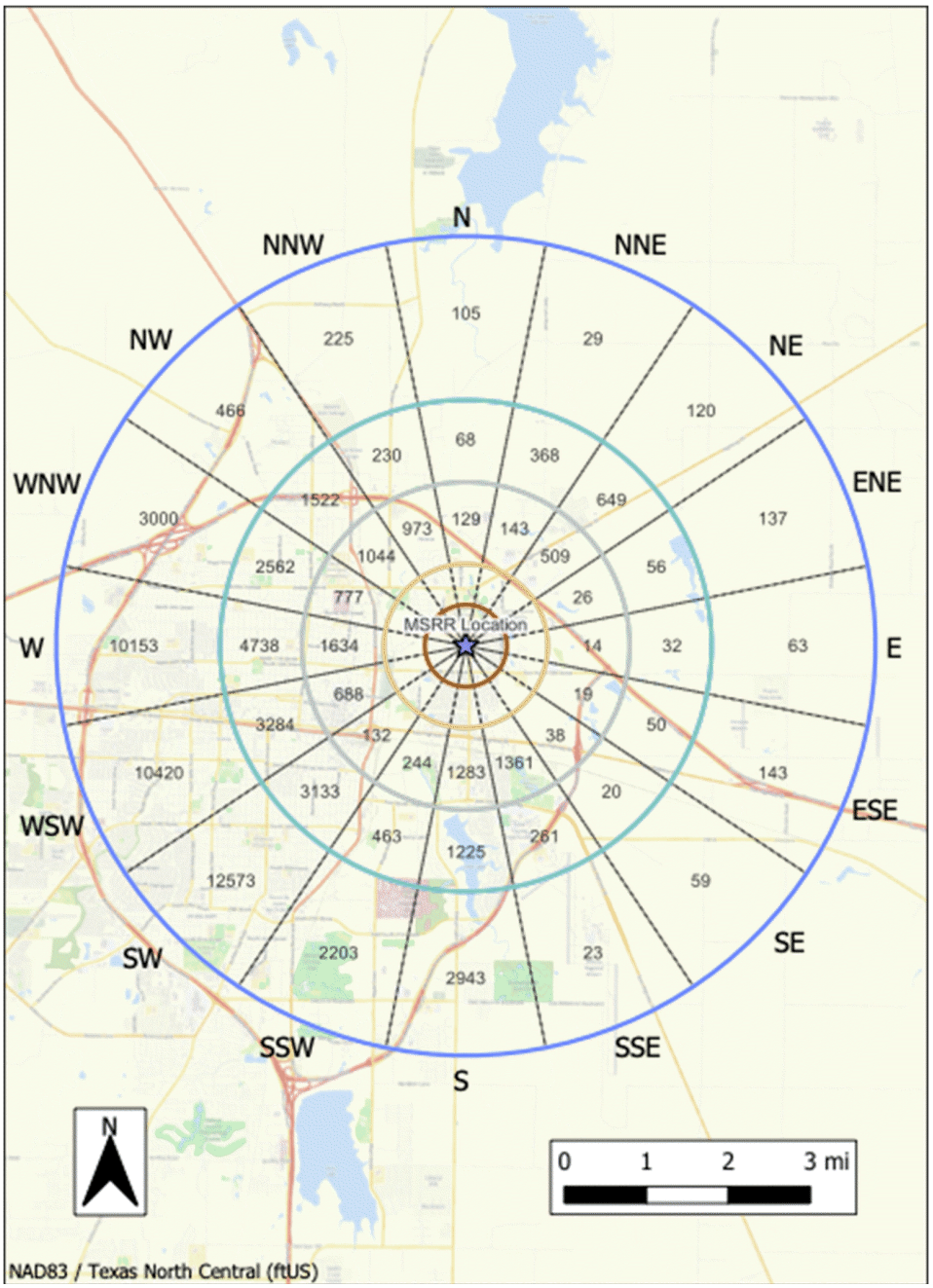
Data from USCB [Reference 2.1-19], [Reference 2.1-20]

Figure 2.1-14 Nearest Resident Population Distribution - 2022



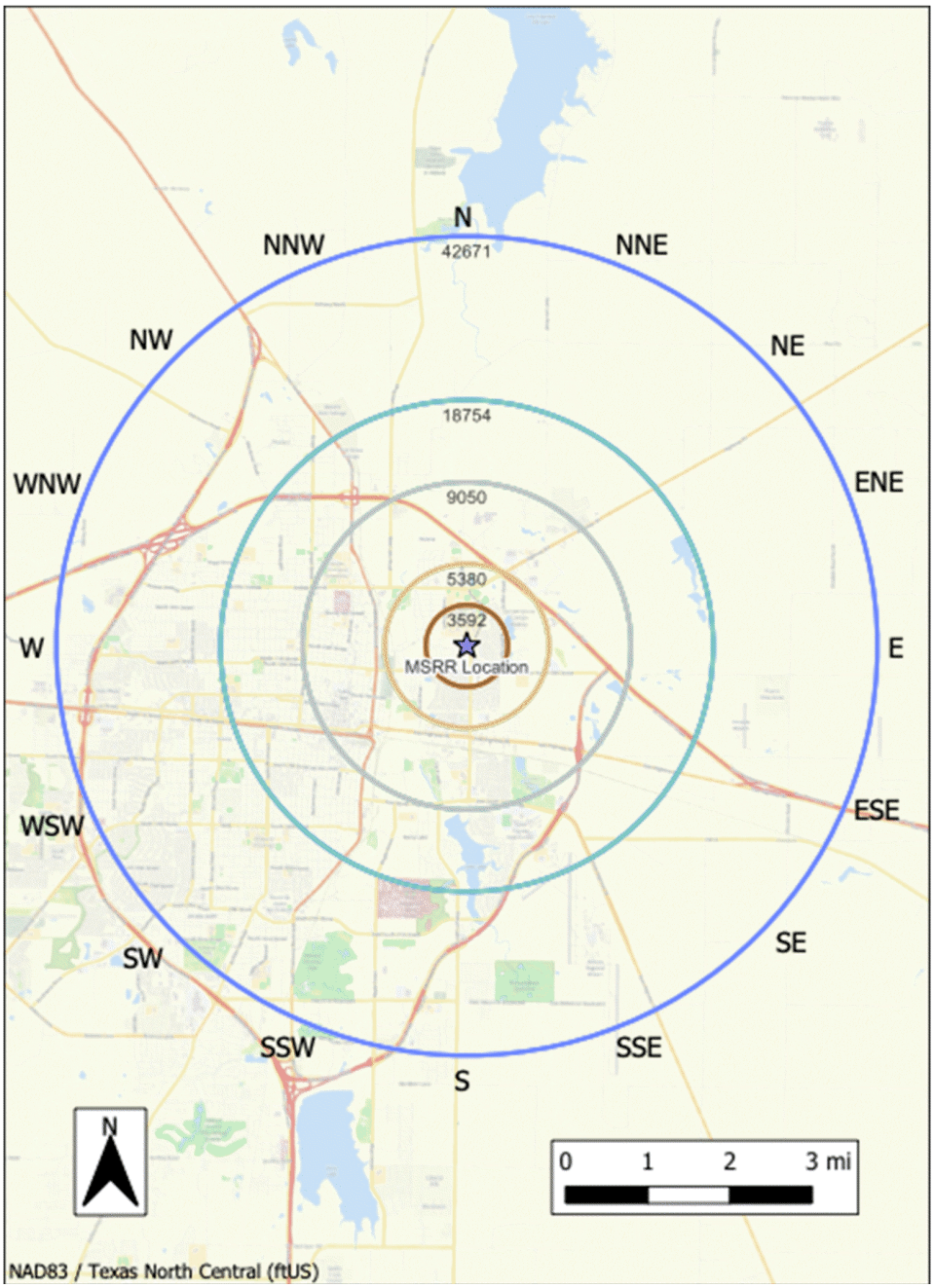
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-15 Furthest Resident Population Distribution - 2022



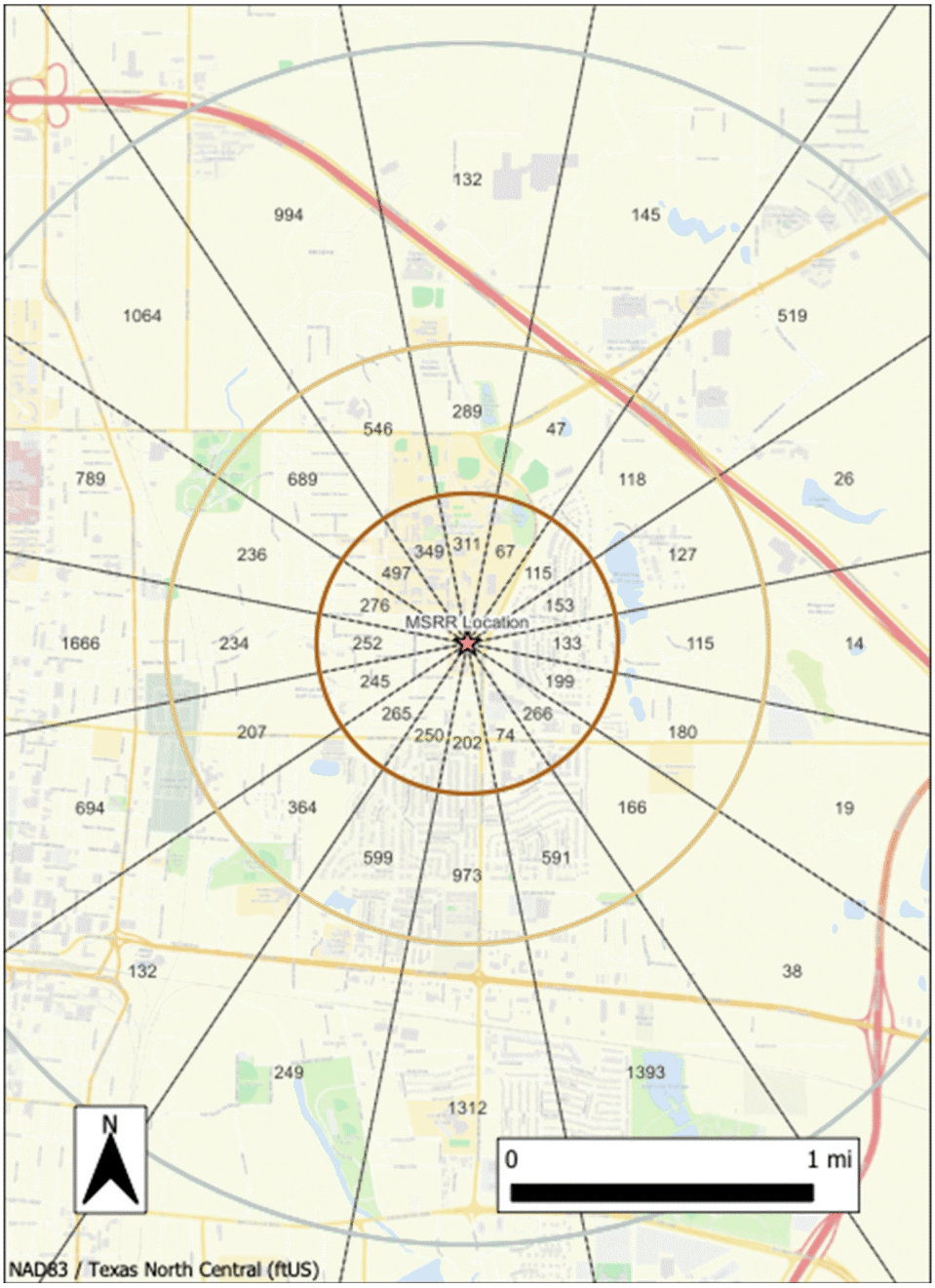
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-16 Resident Population Bands - 2022



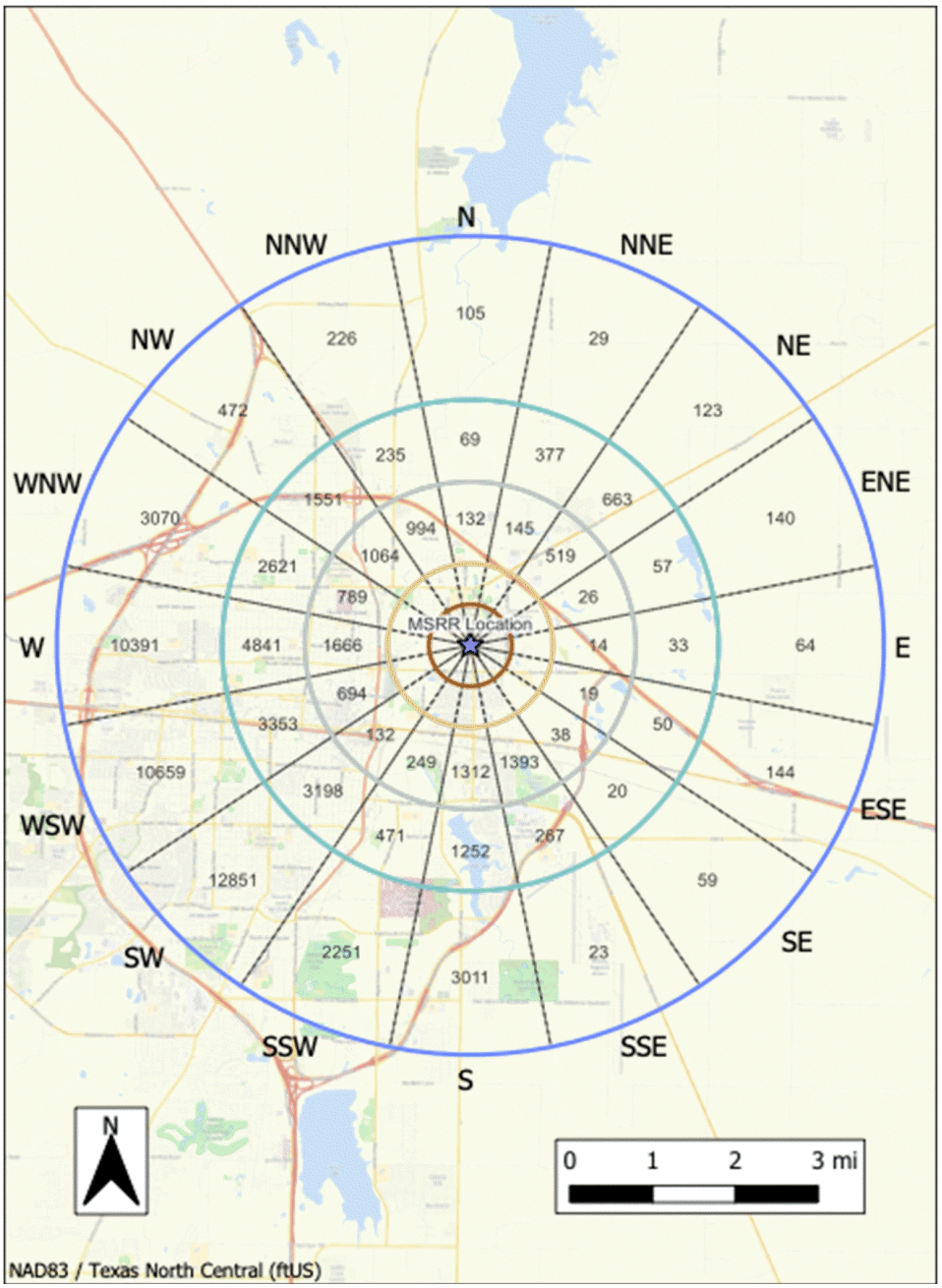
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-17 Nearest Resident Population Distribution - 2027



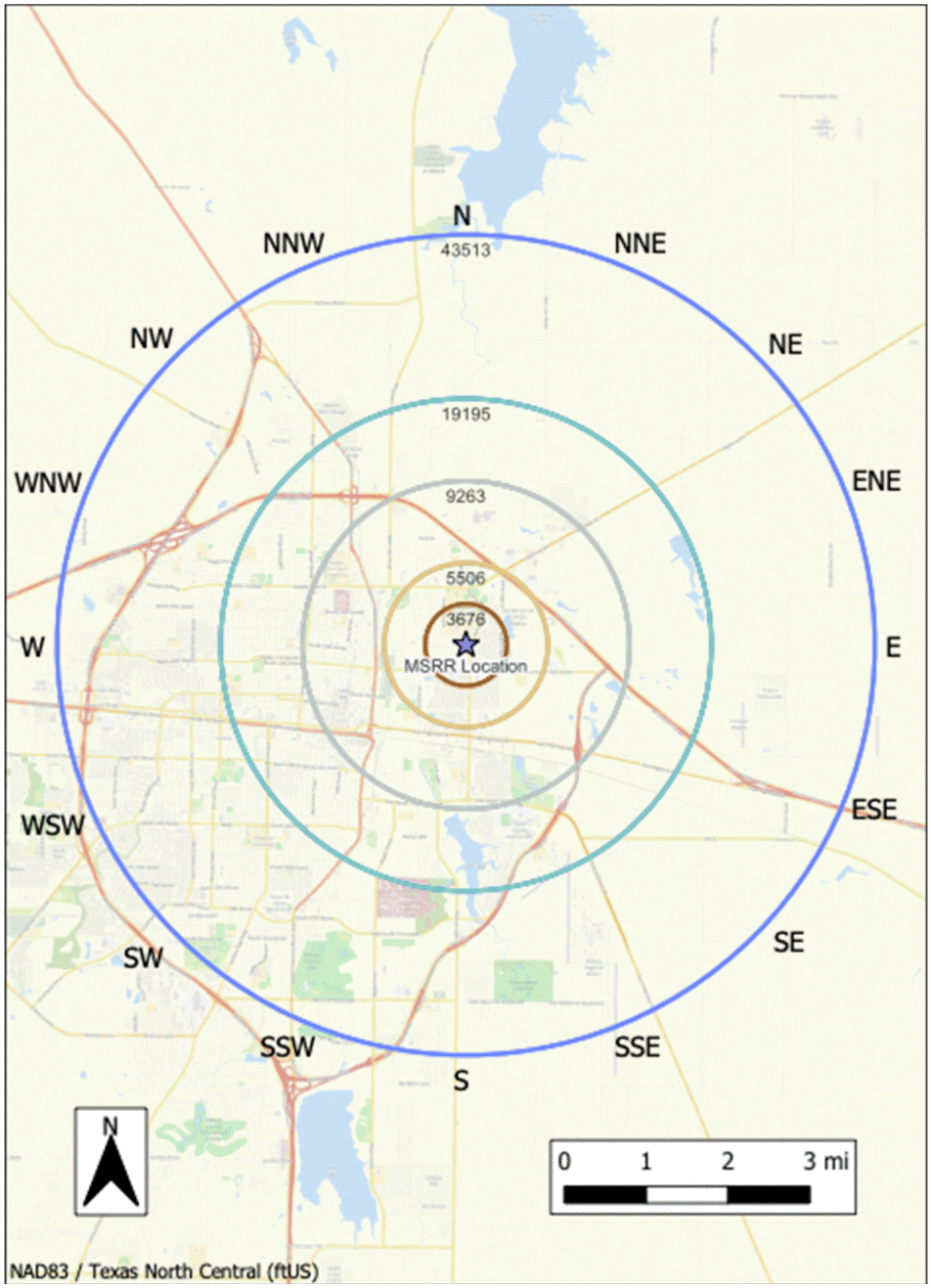
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-18 Furthest Resident Population Distribution - 2027



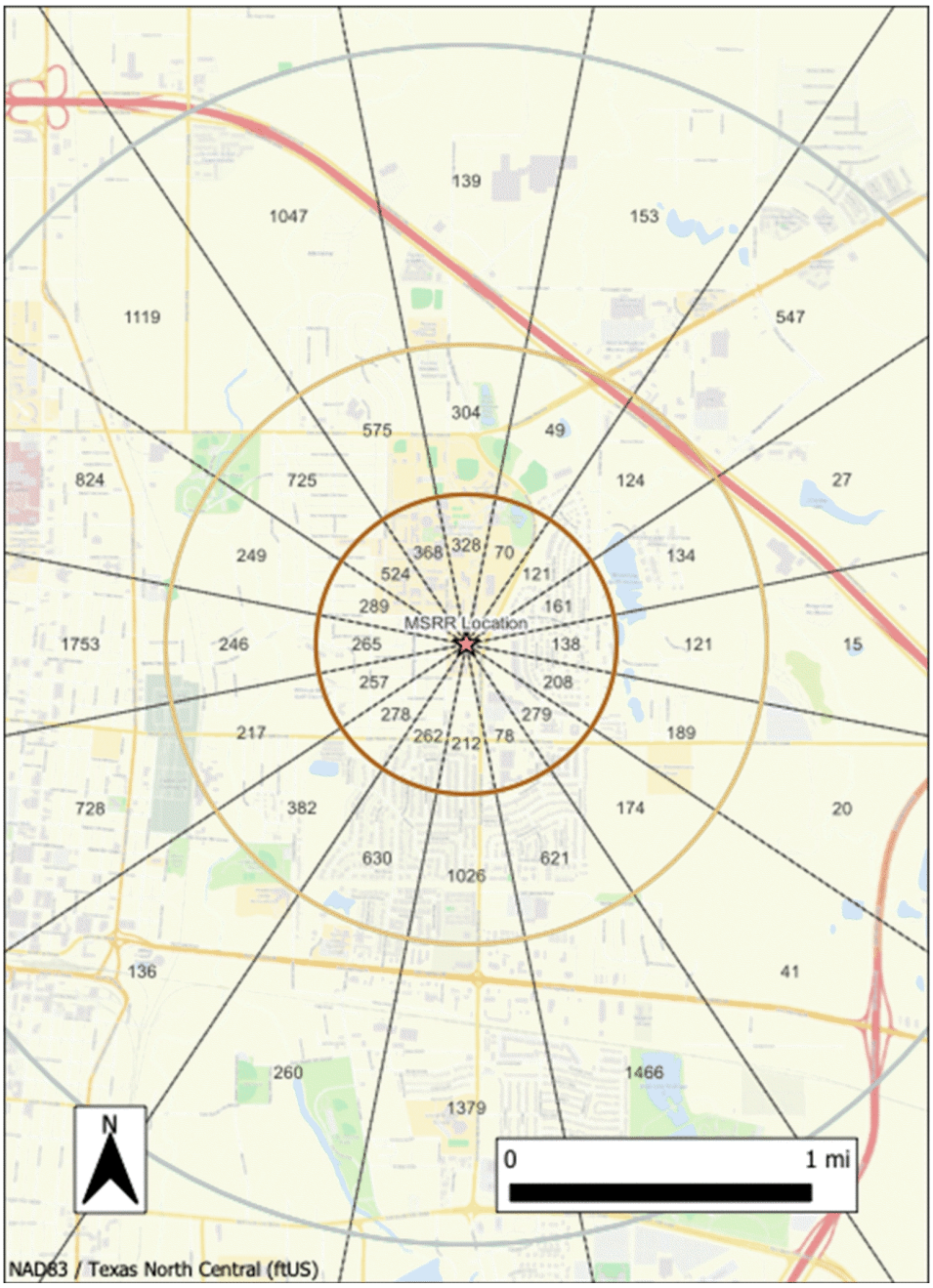
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-19 Resident Population Bands - 2027



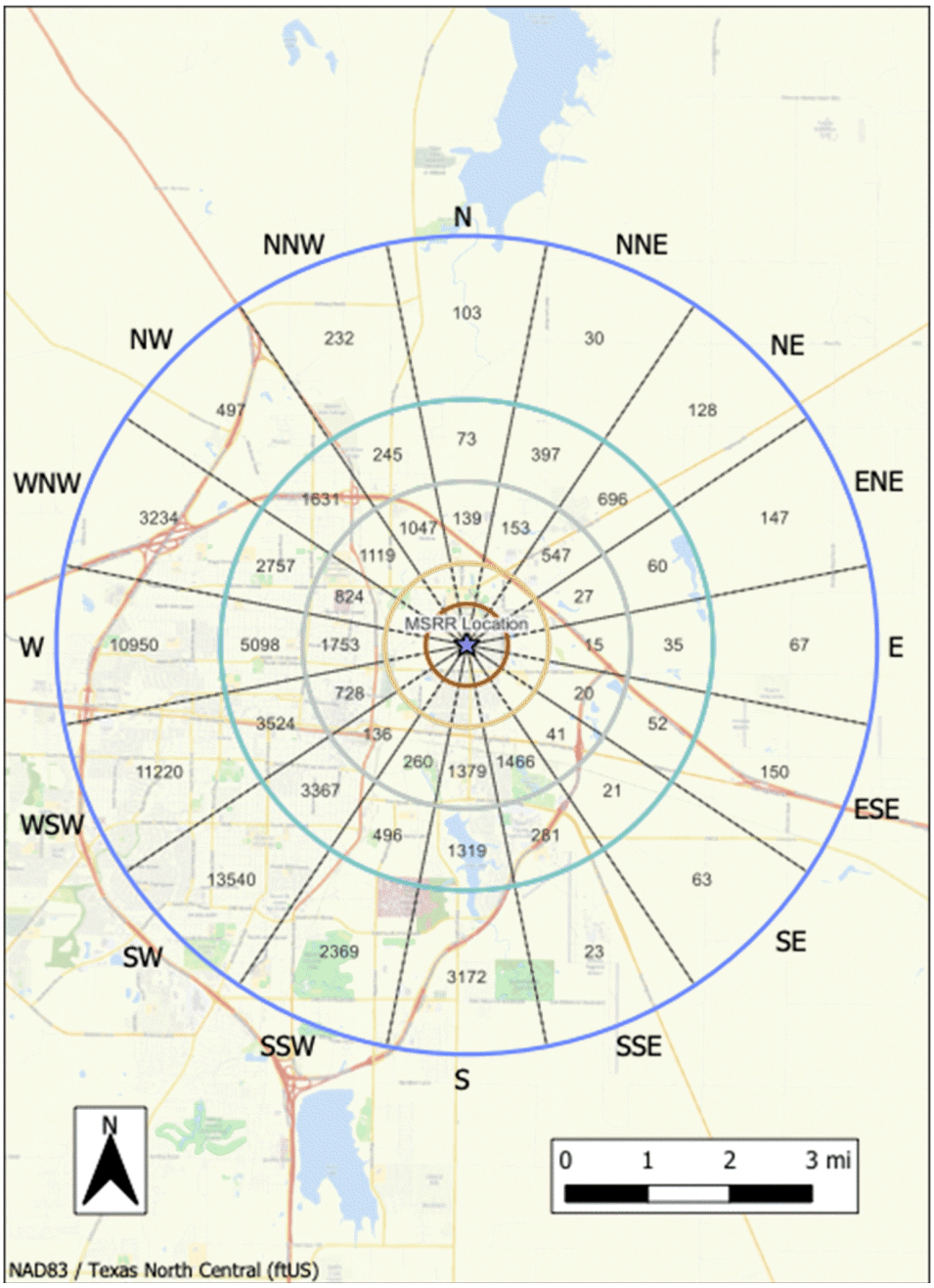
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-20 Nearest Resident Population Distribution - 2042



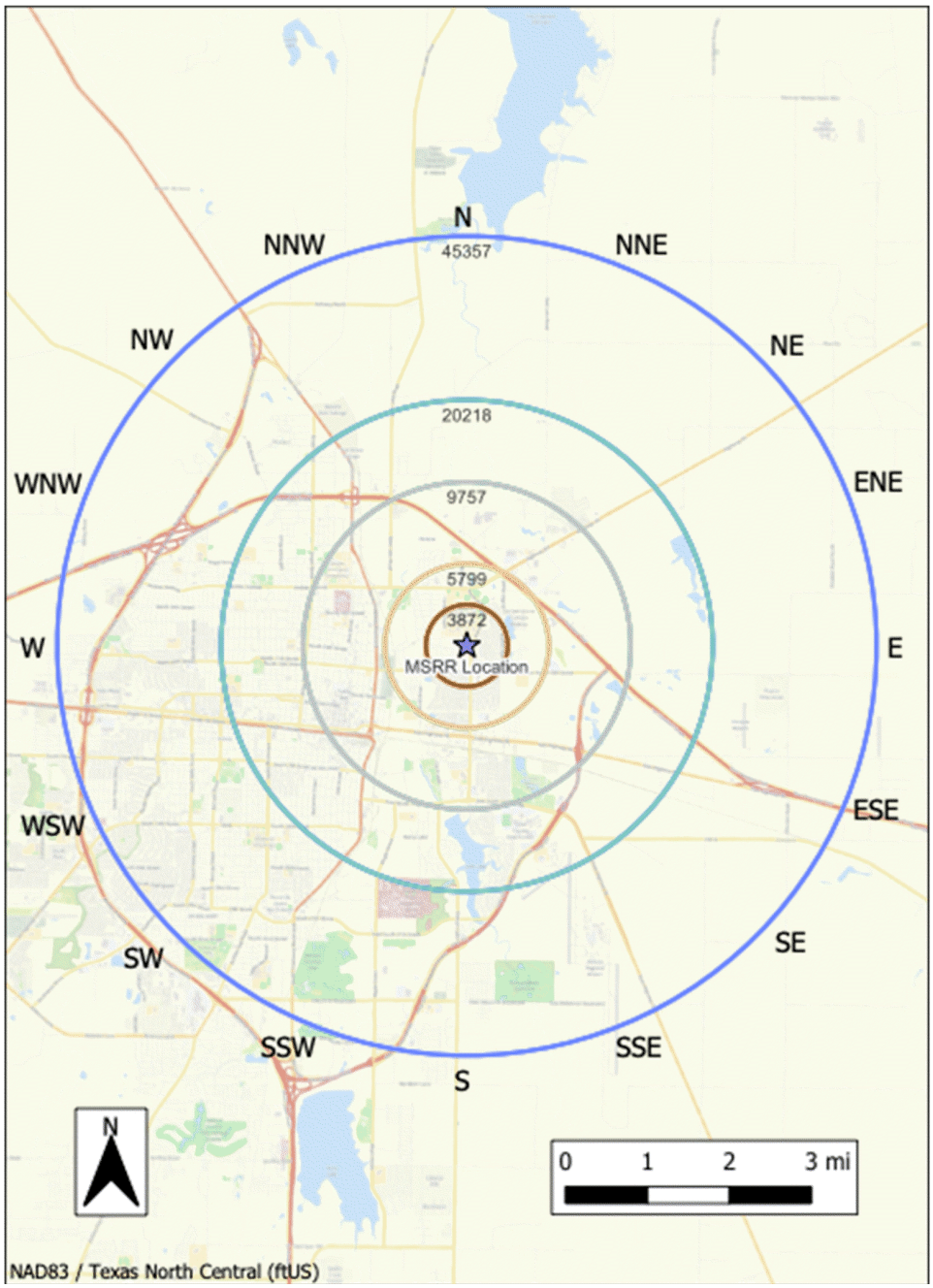
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-21 Furthest resident population distribution - 2042



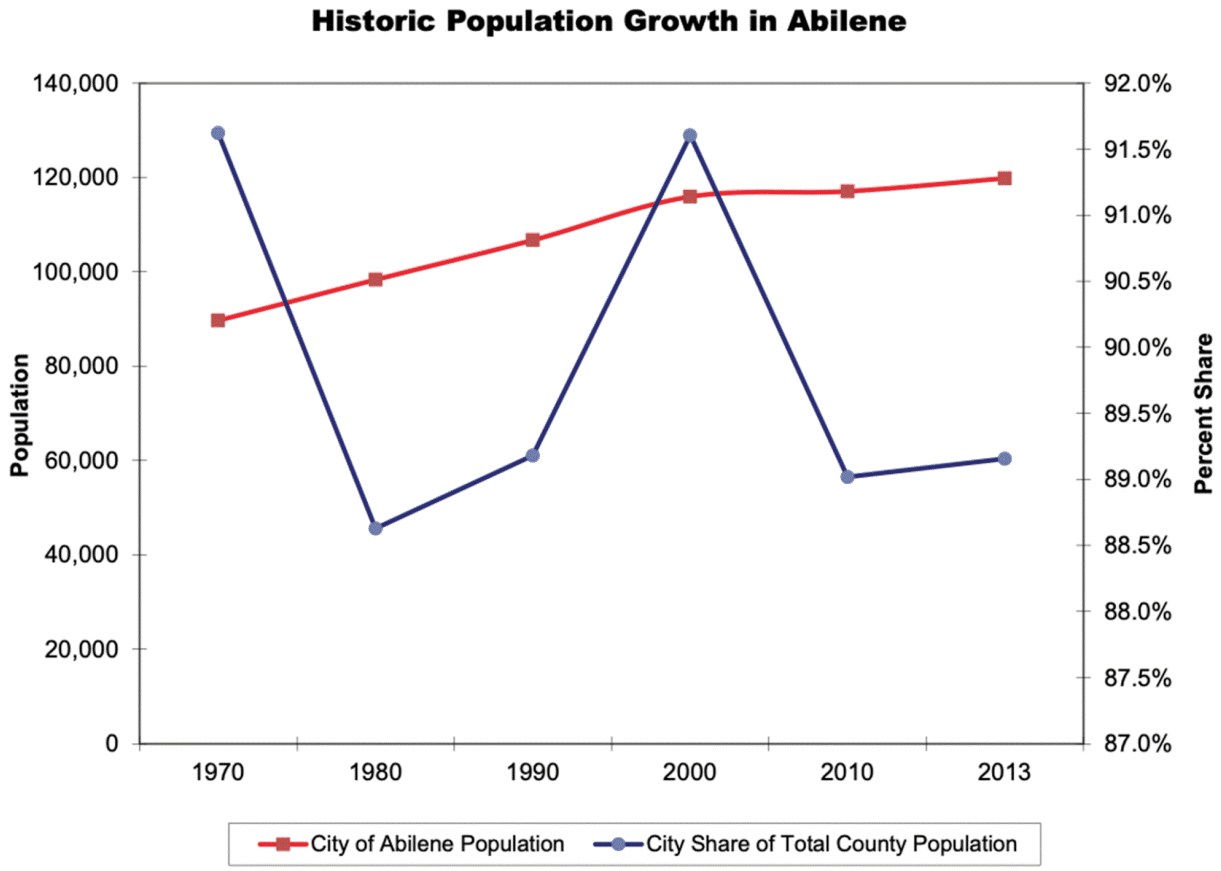
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-22 Resident Population Bands - 2042



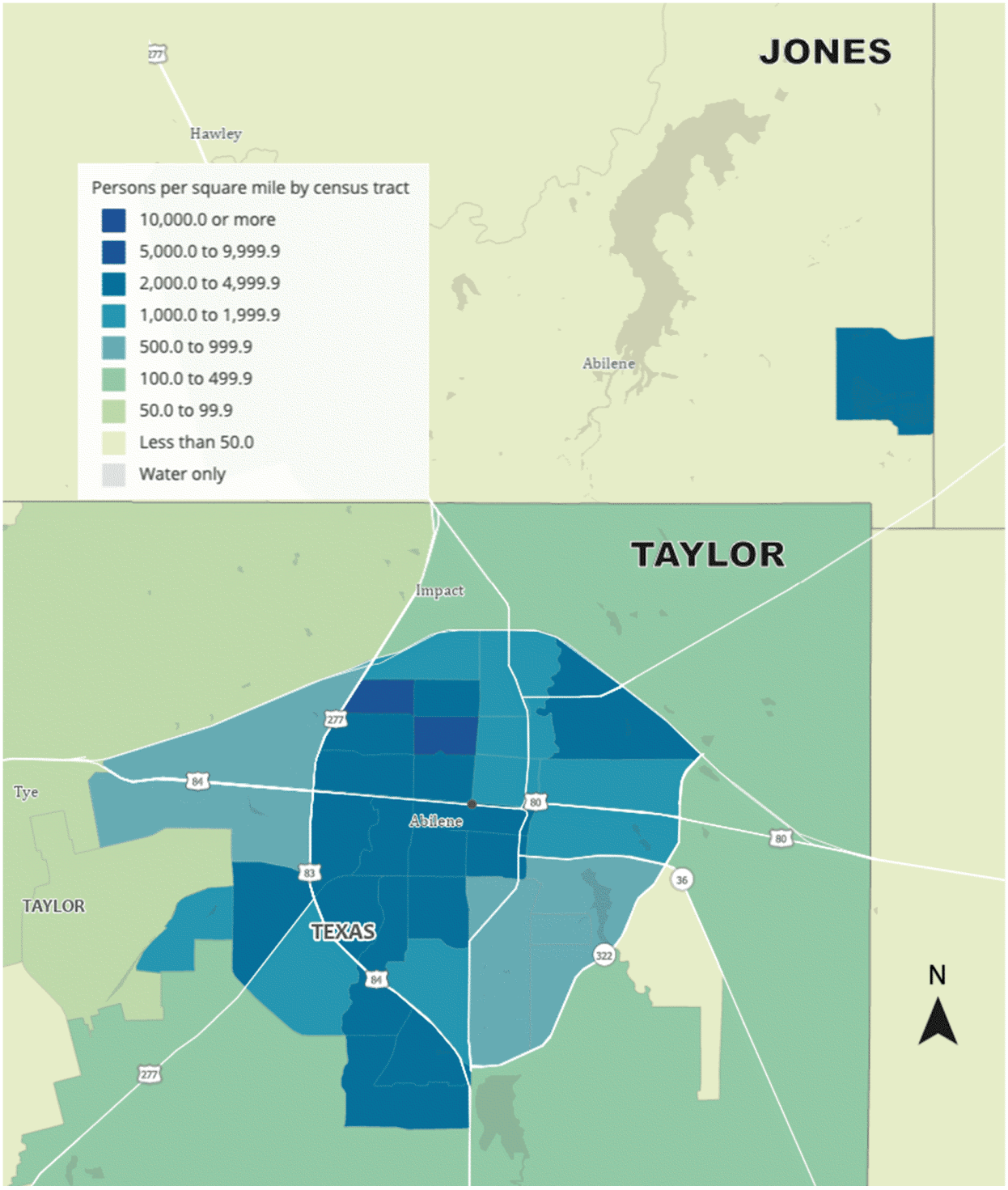
Data from USCB [Reference 2.1-19], [Reference 2.1-20], and TDC [Reference 2.1-21]

Figure 2.1-23 Historic Population Growth in Abilene and Taylor County



From “Abilene parks, recreation & senior facilities master plan” [Reference 2.1-3]

Figure 2.1-24 Abilene Population Density



From U.S. Census Bureau [Reference 2.1-12]

Figure 2.1-25 Location of Universities in Abilene

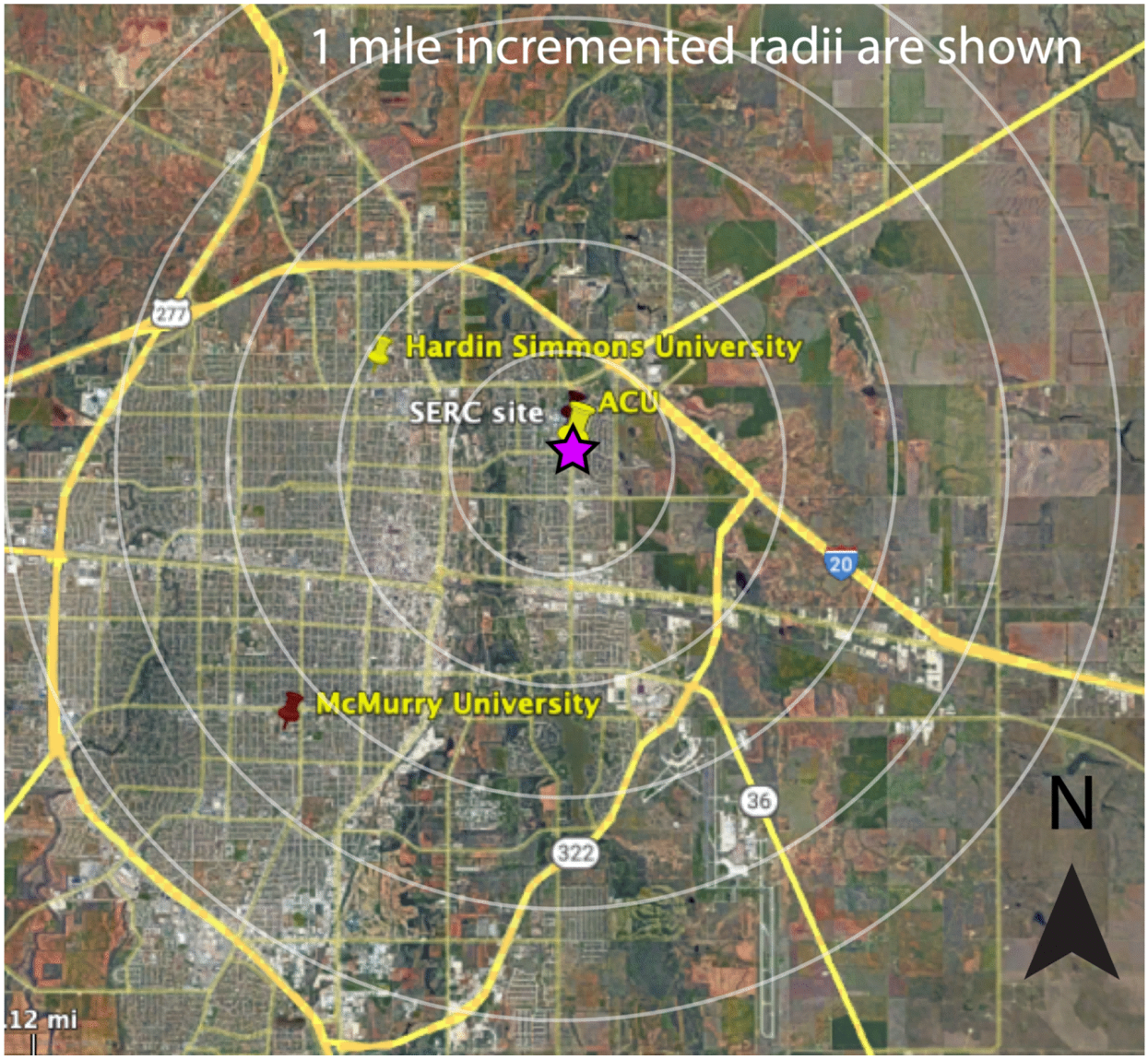
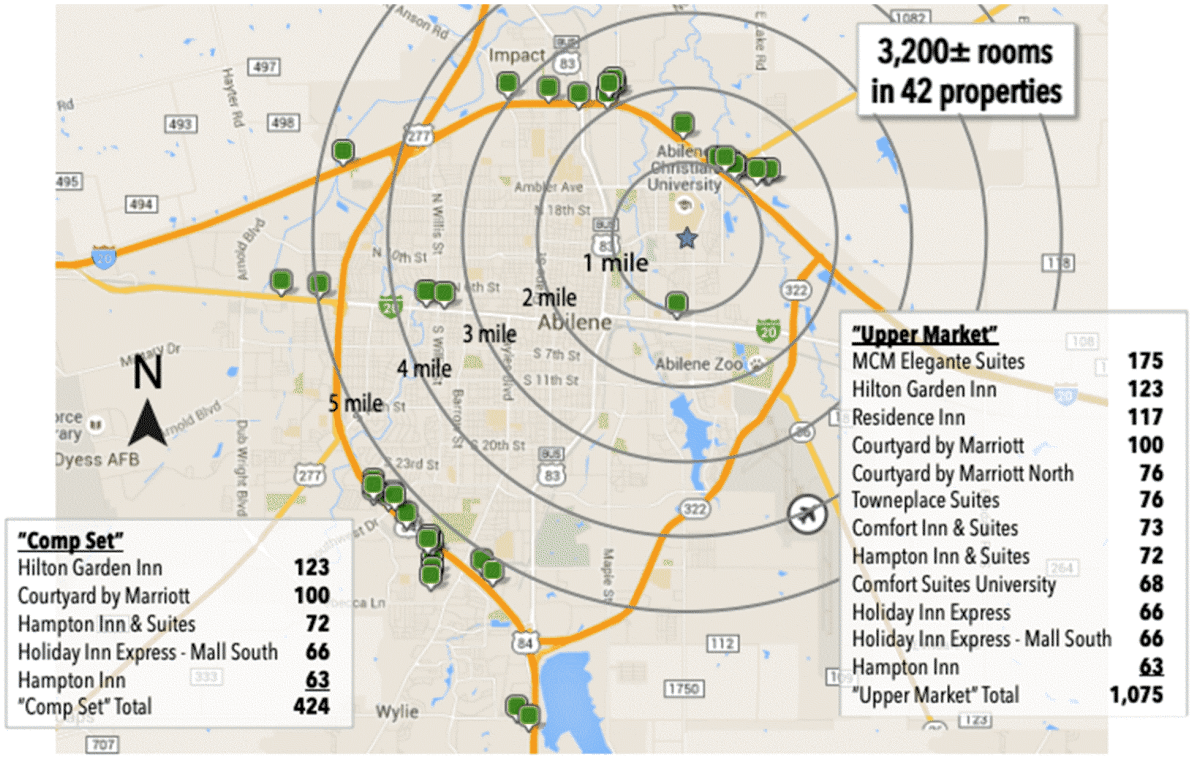


Figure 2.1-26 Abilene Hotel Locations and Occupancy

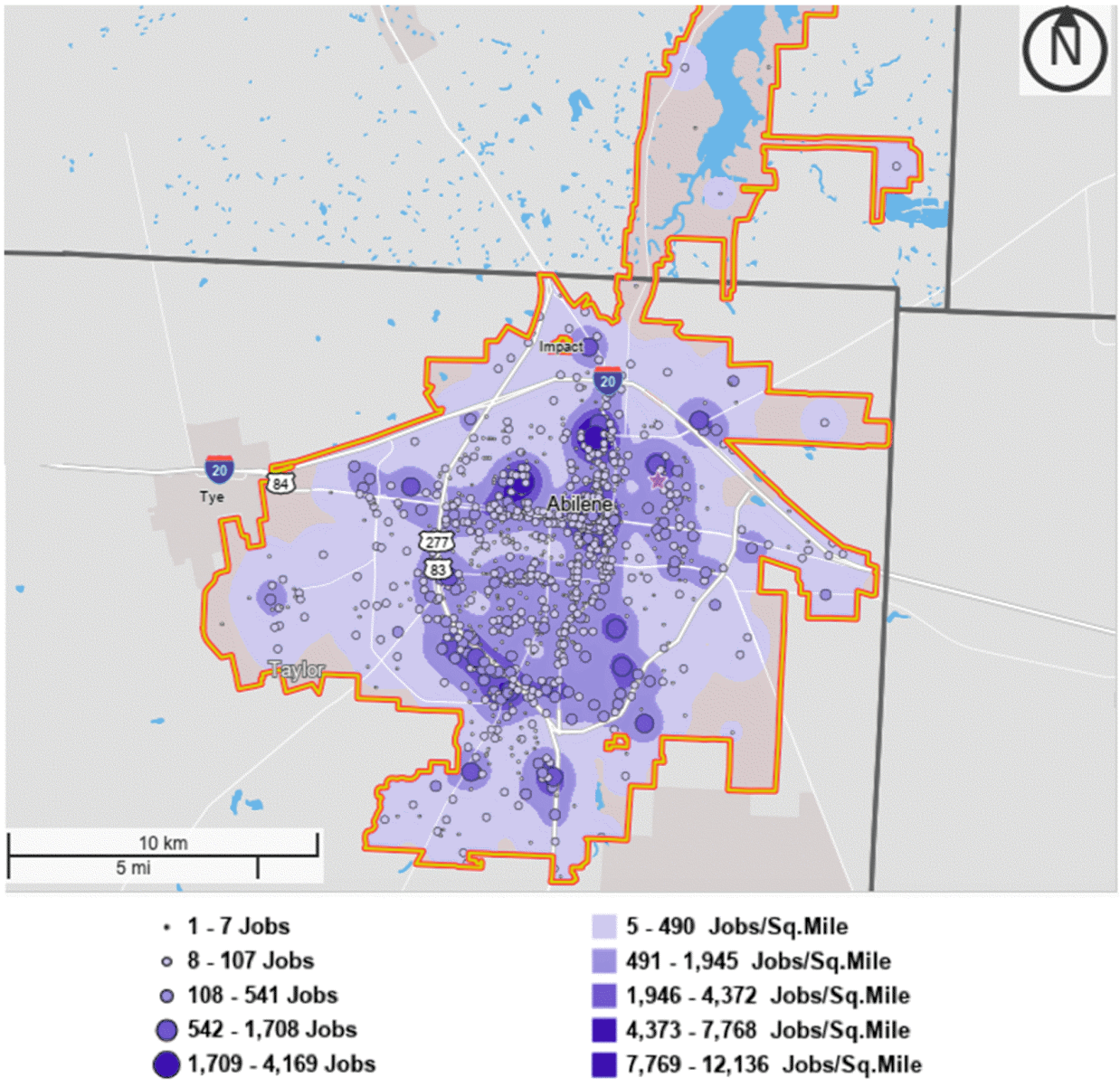


From “Abilene downtown hotel assessment” [Reference 2.1-23]

Figure 2.1-27 Recreational Vehicle Parks and Campgrounds near the Reactor Site

Figure 2.1-28 Locations of Hospitals in Abilene

Figure 2.1-29 Concentrations of Employment across Abilene



Adapted from 2020 Census demographic data viewer [Reference 2.1-22]

2.2 Nearby Industrial, Transportation, and Military Facilities

This section identifies and evaluates present and projected industrial, transportation, and military installations and operations in the area of the site. This section considers the facilities and activities within five miles (8 km) of the reactor, consistent with the guidance in NUREG-1537. Facilities and activities at greater distances were considered, and only one was analyzed in detail, because of their insignificance with respect to accident impact on the facility site.

2.2.1 Locations and Routes

An investigation of industrial, transportation, and military facilities within five miles (8 km) of the site was performed. Figure 2.2-1 shows the location of nearby facilities, including industrial and transportation facilities in the area. Figure 2.2-2 illustrates the airports, jet routes, and airway routes identified in a wider region. Figure 2.2-3 shows Abilene's highways, rail lines, Dyess Airforce Base, Abilene Regional Airport, and the downtown area. Figure 2.2-4 shows the pipelines in the Abilene area.

An evaluation of the identified transportation routes and pipelines within the five-mile (8-km) vicinity of the site identified three major highways, five major roads, one major rail line, one minor rail spur, five natural gas pipelines, and three airports for assessment:

- U.S. Interstate 20
- Business Route 20
- U.S. Highway 83/84
- TX 322 Loop
- Business U.S. Route 83
- Union Pacific Railroad
- Southern Switching Company Rail Spur
- Atmos Pipeline Texas - Natural Gas Transmission Line - XT1
- Atmos Pipeline Texas - Natural Gas Transmission Line - KC17
- Atmos Pipeline Texas - Natural Gas Transmission Line - WA6
- Atmos Pipeline Texas - Natural Gas Transmission Line - WA
- Alon USA - Non HVL product - Nitrogen, Unfilled - 2729
- Oneok NGL Pipeline, HVL, Natural Gas - 722
- Abilene Regional Airport
- Abilene Executive Airpark
- Elmdale Airpark

There are no waterways, chemical plants, refineries, hazardous chemical storage depots, missile sites, or mining/quarrying within the five-mile (8-km) vicinity of the site.

2.2.1.1 Description of Highways

United States Interstate 20 (I-20) is the main roadway through Abilene, describing an east-west arc across the northern part of the city. Texas State Highway 322 (Jake Roberts Freeway) joins I-20 in the northeast part of the city, and U.S. Route 83/84 connects to I-20 to the northwest to encompass a “loop” around Abilene of approximately six miles (10 km) from east to west and seven miles (11 km) north to south. Business Route 20 bisects the city east to west, and Business U.S. Route 83 crosses it, running north to south. Texas State Highway 36 crosses Texas State Highway 322 south of and parallel to I-20, then drops southwest. Texas State Highway 351 begins just north of the campus and heads out of town in a northeastern direction. On the opposite side of town, U.S. 277 extends southwest from U.S. Route 83/84 (see Figure 2.2-1).

Business U.S. Route 83/84 runs north-south 1.1 miles (1.8 km) west of the site. Texas State Highway 351 runs east-west near the campus and is approximately 0.7 miles (1.1 km) north of the site. These routes primarily support local traffic.

The most significant highway near the site is I-20, which runs northwest to southeast nearest the campus, and is one of the primary east-west travel routes across Texas. I-20 is the closest major highway to the MSRR. At its closest point, a two-mile stretch of I-20 is approximately one mile (1.6 km) northeast of the site. According to the Texas Department of Transportation, the annual average daily traffic count is 24,000 [Reference 2.2-1] in the stretch nearest the campus.

Business Route I-20 runs east-west through the center of Abilene just over one mile (1.6 km) south of the site. The annual average daily traffic is approximately 6,000. East of the I-20 and Business Route I-20 intersection, located nearly five miles (8 km) southeast from the site, the average daily traffic count on I-20 is 30,000 [Reference 2.2-1].

2.2.1.2 Description of Railroads

The major rail line is the Union Pacific Railroad line that parallels Business I-20, running east-west through downtown Abilene, and it is 1.2 miles (1.9 km) south of the site.

The nearest rail line to the site is a spur operated by the Southern Switching Company that runs 8.5 miles (14 km) north-south across Abilene with a closest approach to the site of 0.9 mile (1.4 km). This spur is used for shipping and receiving commodities such as grain, feed, fertilizers, oil, scrap, corn sweetener, and lumber within Abilene. The spur is infrequently operated (see Figure 2.2-3).

2.2.1.3 Description of Pipelines

Figure 2.2-4 shows the natural gas and hazardous liquid pipelines in the area surrounding Abilene. Most of the pipelines in the area are outside a five-mile (8 km) radius from the site. Those nearest the reactor are no closer than approximately four miles (6 km) and are not considered a threat to the safe

operation or shutdown of the MSRR. They include five natural gas lines, four owned by Atmos Pipeline Texas and the other by Oneok NGL Pipeline. Alon USA operates a non-high-volatile pipeline (see Figure 2.2-4).

2.2.1.4 Description of Industrial Locations

The U.S. Environmental Protection Agency Envirofacts Database was used to identify potential facilities of interest within five miles (8 km) of the MSRR. Superfund Amendments and Reauthorization Act (SARA) Title III, Tier II reports for industrial facilities within five miles (8 km) of the MSRR. All of these entities are expected to have various industrial chemicals; however, none of them has been identified as having sufficiently large quantities to be of potential concern for the MSRR at the distances as shown (see Figure 2.2-1).

Tier II Industrial Facilities

- American Electric Power West Texas Utilities (electric power distribution equipment yard) ~1.8 mi SE
- ACU Onstead Science Center (classrooms and laboratories) ~1200 ft NW
- Ben E. Keith Beer (distributor) ~1.2 mi NW
- Martin Sprocket & Gear (manufacturing plant) ~2 mi SE
- Bandag West Warehouse (tire warehouse) ~4 mi SE
- Bandag Plant (tire retread plant) ~4 mi SE
- XPO Logistics (trucking terminal) ~1.3 mi SE
- Coca Cola Southwest Beverages (soft drink bottling) ~1.2 mi NE
- Lowes North (big box hardware store) ~1 mi NE
- Coca Cola Warehouse (soft drink warehouse) ~1.5 mi N
- AEP Service Center (electric power distribution equipment yard) ~5 mi S
- Cintas (janitorial supplies, uniform cleaning) ~1.5 mi SE

2.2.1.5 Description of Military Facilities

Dyess Air Force Base (Dyess) is the only military facility in the region. Dyess is an Air Combat Command installation located on a 6,117-acre property that is 9.1 miles (14.6 km) southwest of the site. It is hosted by the 7th Bombardment Wing and includes the 317th Airlift Group. Together, they operate 27 B-1B bombers and 28 C-130J aircraft, and support a variety of transient military aircraft. As such, Dyess is identified as the only significant facility outside a five-mile (8 km) radius from the site that is evaluated for adverse risk to the facility. No plans exist to develop a military airport within 10 miles (16 km) of the site. Figure 2.2-5, Figure 2.2-6, and Figure 2.2-7 illustrate the arrival, departure, closed pattern, and holding patterns of Dyess air traffic in the Abilene area [Reference 2.2-2].

2.2.2 Air Traffic

There are two existing airports within five miles (8 km) of the MSRR site. Elmdale Airpark is located 3.5 miles (5.6 km) east-southeast of the site. The Abilene Regional Airport is located 3.4 miles (5.5 km) south of the site. A heliport owned by Hendrick Medical Center is located 1.6 miles (2.6 km) west of the site. No plans exist to develop a new commercial airport within five miles (8 km) of the site.

2.2.2.1 Elmdale Airpark

Elmdale Airpark is privately owned and operates two runways for light aircraft. It is located on the eastern edge of Abilene just north of I-20. Runway 17/35 is a 2,950 by 30-ft (900 by 9 m) asphalt surface. Runway 18/36 is a 3,280 by 50-ft (1,000 by 15 m) grass and sod surface. Both runways run north and south. There is no published information on usage, but the airpark manager stated they average about 600 operations (landings and take-offs) per year.

2.2.2.2 Abilene Regional Airport

The nearest major airport to the MSRR site is the Abilene Regional Airport, which covers 1,644 acres (665 ha) and has three asphalt runways [Reference 2.2-3].

The airport layout is shown in Figure 2.2-8. Runway 17R/35L is 7,208 ft (2,197 m) long by 150 ft (46 m) wide, oriented north-south. Runway 17L/35R is 7,198 ft (2194 km) long by 150 ft (46 m) wide, oriented north-south. Runway 4/22 is 3,681 ft (1,122 km) long by 100 ft (30 km) wide, is oriented northeast-southwest, and has been decommissioned. Based on data through 2016, it has been projected [Reference 2.2-4] that Abilene Regional Airport will have 45,982 total operations in 2022 for an average of 126 operations per day. Included in that projection is a categorical breakdown shown in Table 2.2-1 predicting that nearly half of the operations will involve general aviation aircraft, followed by large military aircraft, commercial aircraft, and then commercial air taxis.

2.2.2.3 Hendrick Medical Center Heliport

The only heliport within a five-mile (8-km) radius of the site is owned by Hendrick Medical Center. The heliport is located on the southeast corner of the hospital campus on a 120 by 100-ft (37 by 31 m) concrete pad. There is no public information regarding operations statistics. The operations manager stated they average approximately 1,000 flights per year (2,000 operations).

2.2.2.4 Dyess Air Force Base

The only military facility near Abilene is Dyess Air Force Base (Dyess). Figure 2.2-9 provides a layout of the base, which occupies 6,117 acres (2,475 ha) on the southwest outskirts of Abilene. Dyess has three parallel runways running north-northwest. Runway 16/34 is approximately 2.5 miles (4 km) long. Runway 163/343 is a landing zone approximately 0.7 mile (1.1 km) long. Runway 164/344 is also a landing zone 0.7 mile (1.1 km) long. Based on publicly available information [Reference 2.2-2], Dyess airport carries out 51,220 operations per

year. Table 2.2-2 provides a breakdown of operations by aircraft type. Approximately 80 percent of the operations are closed pattern operations, which do not leave the immediate vicinity of Dyess, leaving 10,636 operations to evaluate for potential hazard to the site.

2.2.2.5 Evaluation of Airway Hazards

The Department of Energy (DOE) provides a method for estimating the probability per year of an aircraft crashing into a facility. The methodology is outlined in DOE Standard DOE-STD-3014-2006 [Reference 2.2-5] and utilizes crash rates for non-airport operations.

The non-airport crash impact frequency evaluation is determined from using the following "four factor formula" [Reference 2.2-5]:

$$Fj = Nj \cdot Pj \cdot fj(x,y) \cdot Aj \qquad \text{Equation 2.2-1}$$

Where:

Fj = crash impact frequency

j = each type of aircraft suggested in the DOE Standard

$Nj \cdot Pj$ = expected number of in-flight crashes per year

$fj(x,y)$ = probability, given a crash, that the crash occurs in a 1-square-mile area surrounding the facility

Aj = effective area of the facility in units of square miles

Tables B-14 and B-15 of the standard provide $Nj \cdot Pj \cdot fj(x,y)$ values for general aviation aircraft, air carriers, air taxis, and small military aircraft applicable for specific DOE sites. Orthonormal distances (x,y), based on the direction of flights, between the center active runways and the site have been determined for use in identifying appropriate $fj(x,y)$ values listed in Tables B-12 through B-23. Annual aircraft crash rates, Pj, are taken from Table B-1. The number of operations, Nj, are listed in the above subsections.

The effective facility fly-in area (Aj) for the safety-related structures of the site depends on the length, width, and height of the facility, as well as the aircraft wingspan, skid distance, and impact angle as explained below:

$$Aj = Af + As \qquad \text{Equation 2.2-2}$$

where:

$$Af = (WS + R) \cdot H \cdot \cot\phi + (2 \cdot L \cdot W \cdot WS / R) + (L \cdot W) \qquad \text{Equation 2.2-3}$$

and:

$$As = (WS + R) \cdot S \qquad \text{Equation 2.2-4}$$

where:

Af = effective fly-in area

As = effective skid area

WS = aircraft wingspan

R = length of the diagonal of the facility = $(L^2 + W^2)^{0.5}$

H = facility height

$\cot\phi$ = mean of the cotangent of the aircraft impact angle

L = length of facility, facility-specific

W = width of facility, facility-specific

S = aircraft skid distance (mean value)

The maximal (conservative) dimensions of the entire SERC facility are used in the analysis and are described as box having a width (W) of 120 ft (37 m), a length (L) of 140 ft (43 m), and a height (H) of 50 ft (15 m). The remaining variables are based on generalized aircraft characteristics and provided in other tables in the standard.

2.2.2.6 Evaluation of Helicopter Hazards

Hendrick Medical Center heliport is located 1.5 miles (2.4 km) from the site. Given the helicopter crash guidance in DOE-STD-3014-2006, the distant location of the heliport from the site, and the fact that ACU will not be operating a helicopter, a helicopter impact is not considered a credible hazard to the facility.

2.2.2.7 Evaluation of Airport Hazards

2.2.2.7.1 Elmdale Airpark

Calculations of airport hazards, based on conservative statistical assumptions, found that a site crash probability from Elmdale Airpark operations is less than 9×10^{-8} per year.

2.2.2.7.2 Dyess Air Force Base

Calculations of airport hazards, based on conservative statistical assumptions, found that a site crash probability from Dyess operations is less than 1.3×10^{-8} per year.

2.2.2.7.3 Abilene Regional Airport

Initial calculations of airport hazards, based on conservative statistical assumptions, found that a site crash probability from Abilene Regional Airport operations is less than 9×10^{-6} per year. A more realistic calculation includes the actual use of the runways. Abilene Regional Airport has two active runways that are used depending on the wind direction. Runway 35R-17L is used when the wind is out of the north (approximately 20 percent of take-offs and landings), and 35L-17R is used when the wind is from the south (approximately 80 percent of take-offs and landings). The site crash probability, weighted by wind direction is 7.6×10^{-6} per year. Of the 14,830 military operations, only one-third of these are take-off and landing operations; the others are transient military aircraft communications. The site crash probability is reduced to 5.3×10^{-6} per year by removing this conservatism. The military aircraft that take-off and land at Abilene Reginal Airport are C130J from Dyess Air Force base performing touch-and-gos. Replacing the average large military aircraft wingspan of 223 ft (68 m) listed in DOE-STD-3014-2006 to the actual C130J wingspan of 133 ft (41 m), results in a site crash probability of 4.9×10^{-6} per year. This is the probability that an aircraft will strike the SERC multiuse facility in any given year with the entire building considered the target. The MSRR facility, however, is located within the research bay of the SERC, with dimensions of 120 ft by 50 ft by 50 ft (37 m by 15 m by 15 m). The probability for an airplane crash into the SERC research bay is 3.6×10^{-6} per year. The MSRR and all structures, systems, and components are positioned inside a subterranean concrete vault within the SERC research bay with an above-grade target size of approximately 3 ft by 30 ft by 18 ft (1m by 10m by 6m), which results in a probability for an airplane crash into the MSRR facility of 8.2×10^{-7} per year.

2.2.2.8 Summary of Risks from Air Traffic

Initial calculations of airport hazards, based on conservative statistical assumptions, found that the combined Elmdale Airpark and Dyess air traffic site crash probability is bounded to less than 1×10^{-7} per year and do not need be considered further. Abilene Regional Airport is the closest major airport to the site and required more realistic input to the calculations. The probability of an airplane crash into the primarily subterranean MSRR facility within the SERC has been calculated to be 8.2×10^{-7} per year.

2.2.3 Analysis of Potential Accidents at Facilities

This section provides an analysis of the possible effects on the reactor for postulated accidents or other events that could occur at the facilities listed in Section 2.2.1.

2.2.3.1 Highways and Railroads

Interstate 20 (I-20) is the closest major highway. A 2-mile (3.2 km) stretch of the interstate is approximately a mile from the MSRR. The highway is major east west corridor for cross country truck transportation as well as local truck traffic. I-20 represents the limiting hazard potential for highways.

The major rail line is the Union Pacific Railroad line that parallels Business I-20, running east/west through downtown Abilene, and is 1.2 miles (1.9 km) south of the MSRR. This is a major east-west rail corridor and represents the limiting hazard potential for railroads.

At these distances, no event on the highway or railroad is expected to produce an incident overpressure at the SERC. No significant damage would be expected. Additionally, the MSRR is located inside the reactor cell in the systems pit below the SERC (See Figure 4.1-4 and Figure 4.1-2). All systems and components necessary for safe shutdown and long-term passive decay heat removal are located within the reactor cell (nominal 4 ft.-thick steel reinforced concrete walls). The reactor cell bio shield is constructed of nominal 2 ft.-thick concrete with a nominal 4 ft.-thick concrete top plug. The reactor cell and SERC are designed such that destruction or collapse of the SERC will not impact the safety of the reactor cell and will not affect the safe shutdown capability or long-term passive decay heat removal capability.

In the case of severe damage to the SERC, loss of power will cause the reactor protection system to drain the fuel salt into the drain tank which shuts down the reactor. The fuel salt in the drain tank is passively cooled as described in Section 5.2.3.

The fuel handling system (new and used fuel salt) and gas management system are also located in the systems pit below grade and are sufficiently enclosed that damage, destruction, or collapse of the SERC due to a transportation-related explosion or fire will not damage fuel salt-containing components, impede passive cooling, or cause release of fission product gases.

Toxic gas from a highway or railroad incident would not be expected to immediately incapacitate reactor operators. In the case of toxic gas, it is expected the reactor would be manually tripped with the fuel salt draining into the drain tank where it is passively cooled. The MSRR requires no manual actions to maintain shutdown and passive cooling long term.

No credible explosion, fire, or toxic gas release due to a transportation accident will prevent safe operation, shutdown, and cooling of the MSRR.

2.2.3.2 Pipelines

The nearest pipelines to the MSRR are no closer than approximately 4 miles (6.4 km) and are not considered a threat to the safe operation or shutdown of the MSRR.

2.2.3.3 Industrial Locations

The list of industrial facilities of interest within 5 miles (8 km) of the MSRR described in Section 2.2.1.4 were evaluated. No facilities were identified that would be expected to create a more limiting hazard than those described in Section 2.2.3.1 for highways and railroads. Therefore, none of these facilities are considered a threat to safe operation or shutdown of the MSRR.

2.2.3.4 Military Facilities

The only military facility is Dyess Air Force Base. Due to the distance from the MSRR, the only hazard of interest is air traffic. This hazard is addressed in Section 2.2.2.

2.2.4 References

2.2-1 TxDOT I-20: Texas Department of Transportation, Corridor Study of I-20 in Abilene, Vision 20/20, 2018.

2.2-2 Dyess Air Force Base Air Installation Compatible Use Zone Study, January 7, 2015.

2.2-3 U.S. Dept. of Transportation and Federal Aviation Administration.

2.2-4 Abilene Regional Airport Master Plan, Chapter 3 - Aviation Activity Forecast, 2016. Available: https://abilene.airportplans.com

2.2-5 Accident Analysis for Aircraft Crash into Hazardous Facilities, DOE-STD-3014-2006 (Reaffirmed 2006), May 29, 2006.

2.2-6 OSM, “OpenStreetMap,” OpenStreetMap contributors, 2021. Accessed on Feb. 13, 2022. [Online]. Available: https://www.openstreetmap.org

2.2-7 NGA, “OpenStreetMap Bright Basemap,” United States Department of Defense, National Geospatial-Intelligence Agency, 2022. Accessed on Feb. 13, 2022. [Online]. Available: https://osm.gs.mil/features/base-map

2.2-8 TXDOT, “Geospatial Roadway Inventory Database,” Texas Department of Transportation, 2022. Accessed on Feb. 19, 2022. [Online]. Available: https://opendata.arcgis.com/api/v3/datasets/d4f7206d27af4358acb70cb1cc819d10_0/downloads/data?format=geojson&spatialRefId=4326

2.2-9 FRA, “Main Track Centerlines,” United States Department of Transportation, Federal Railways Administrations, 2021. Accessed on Feb. 13, 2022. [Online]. Available: http://nationalmap.usgs.gov/

2.2-10 NPMS Public Viewer, “National Pipeline Mapping System,” Pipeline and Hazardous Materials Safety Administration. [Online]. Available: https://pvnpms.phmsa.dot.gov/

Table 2.2-1 Abilene Regional Airport Operations Predictions for 2022

Aircraft Operations Type	Aircraft Operations (avg 126/day)[a]
Commercial airline	4,612
Air taxi	3,635
General aviation	22,905
Military	14,830[b]
Total	45,982

a. Data from Abilene Regional Airport Master Plan, Chapter 3, Table 3-24 [Reference 2.2-4]

b. Two-thirds of these operations are transient military aircraft communications.

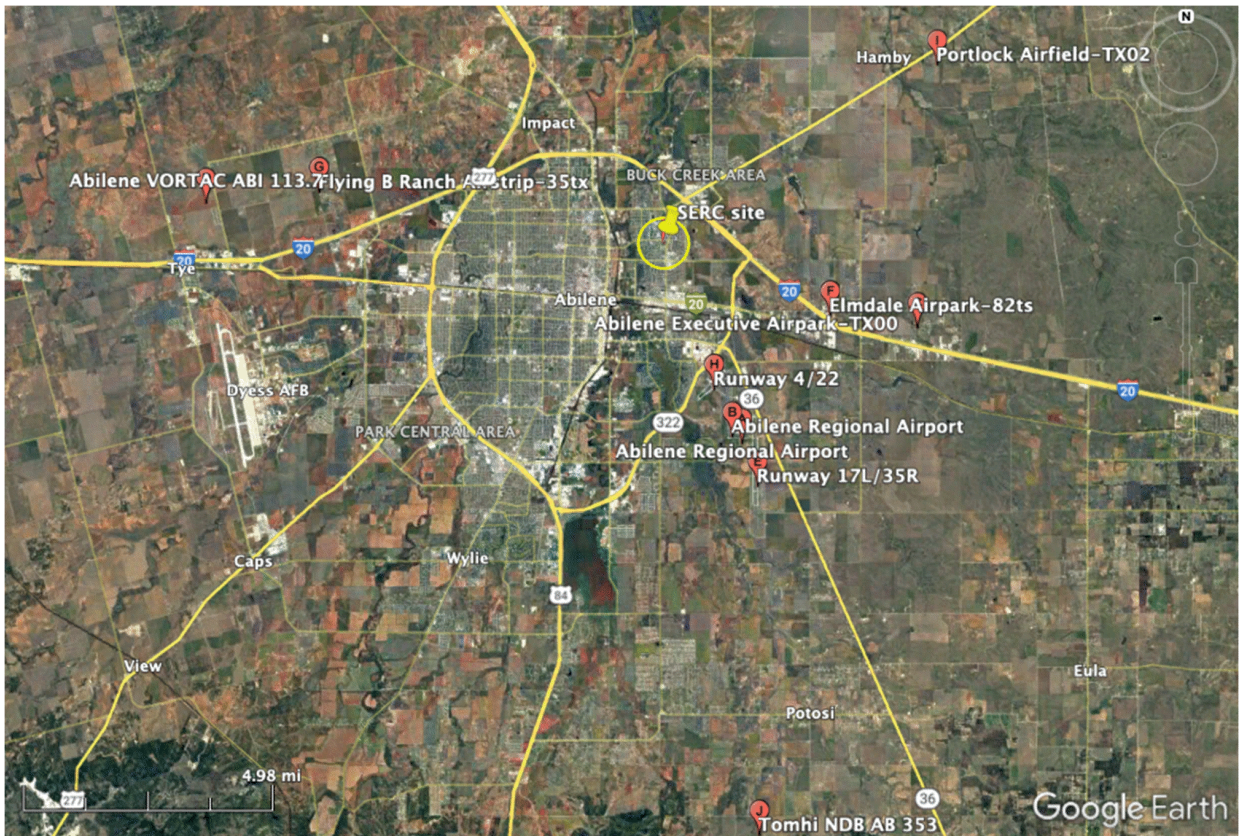
Table 2.2-2 Annual Aircraft Operations for Dyess Air Force Base

Aircraft Classification	All Operations	Closed Pattern Operations	Departure/Arrivals
Large	48,626	40,584	8,042
Small	2,570		2,570
Helicopter	24		24
Total	51,220	40,584	10,636

Data from Dyess Air Force Base study [Reference 2.2-2]

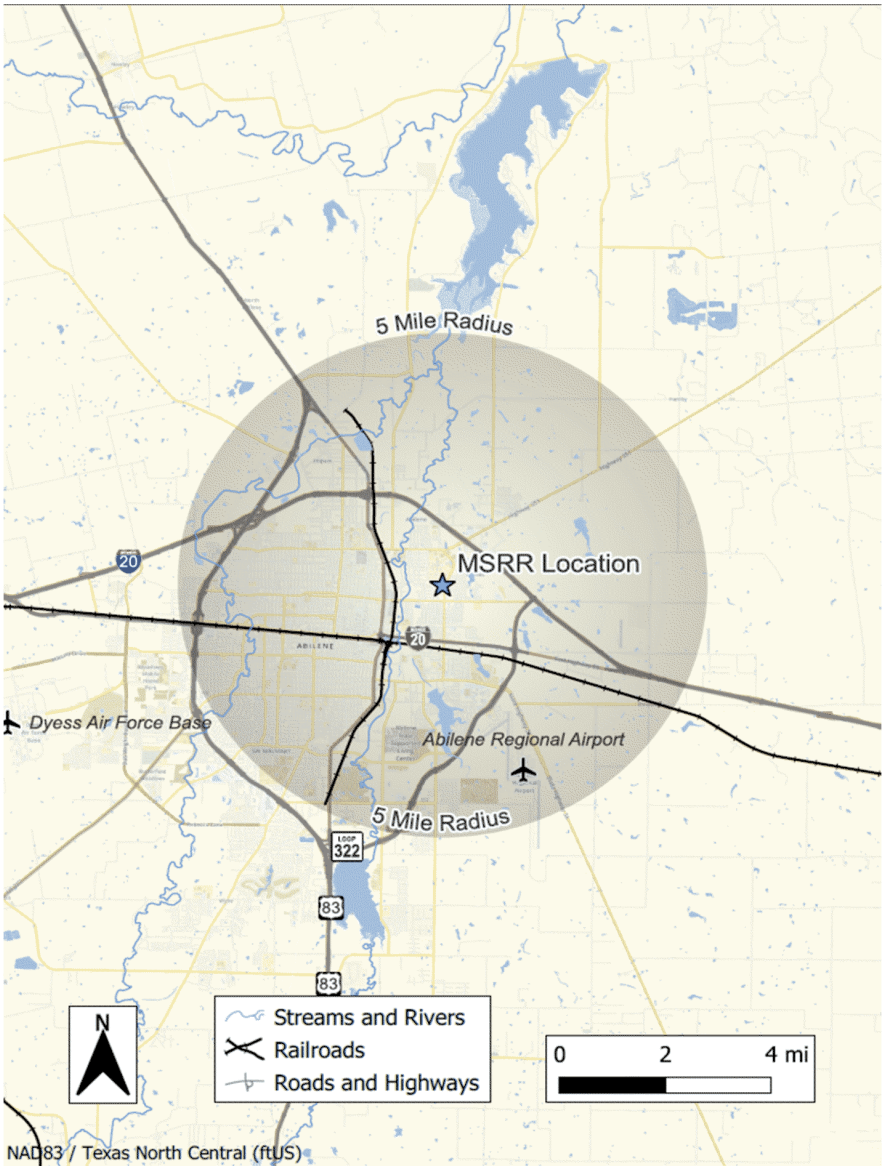
Figure 2.2-1 Industrial and Transportation Facilities Near the Site

Figure 2.2-2 Airfields within 10 Miles of the Site



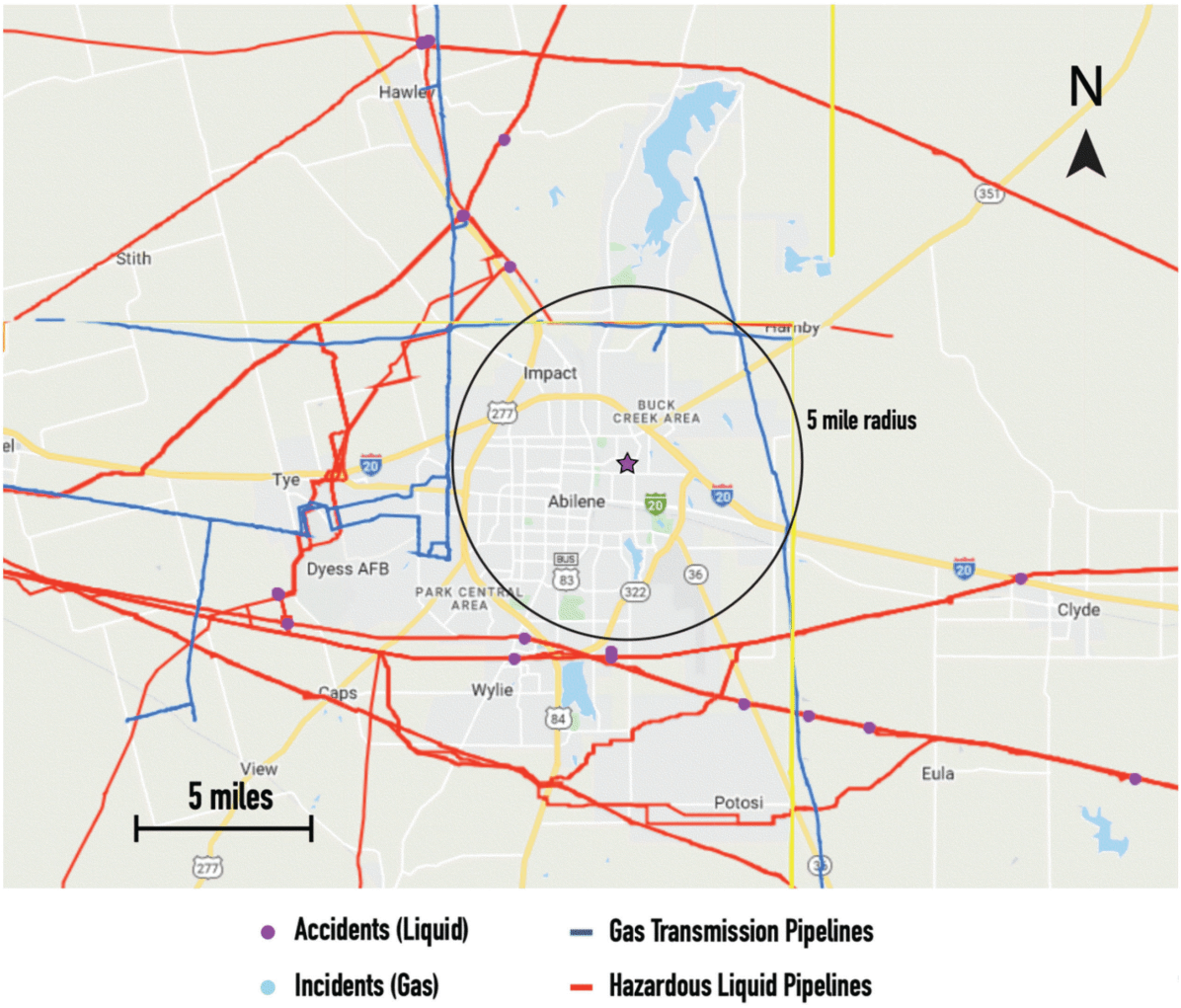
Helipads are not shown. The runways at Dyess AFB can be seen on the left side of the map below U.S. Interstate 20. (Adapted from Google Earth)

Figure 2.2-3 Highways, Rail Lines, Dyess Air Force Base, and Abilene Regional Airport



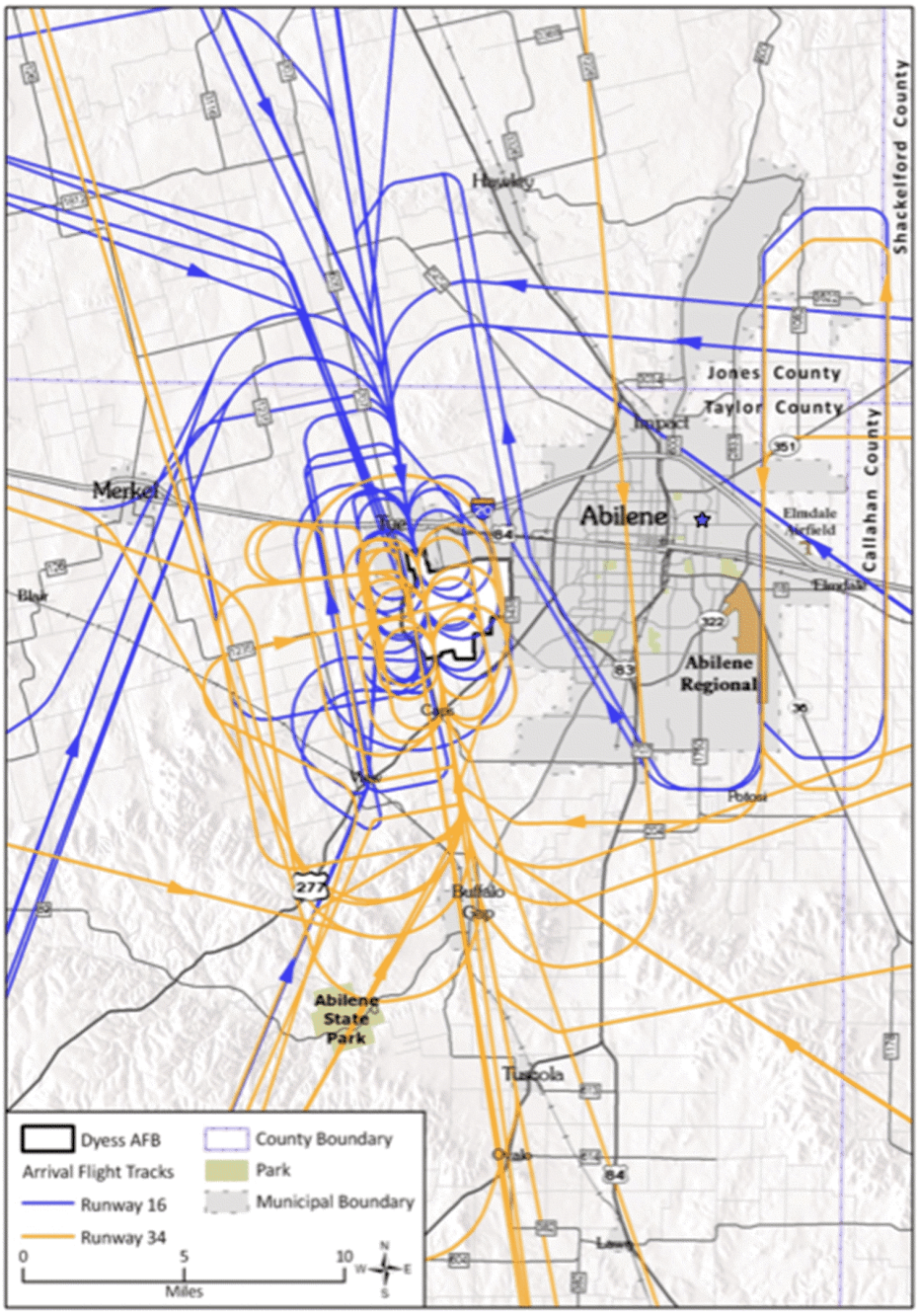
Data from OpenStreetMap [Reference 2.2-6], NGA [Reference 2.2-7], TXDOT [Reference 2.2-8], and FRA. [Reference 2.2-9]

Figure 2.2-4 Pipelines in the Abilene Area



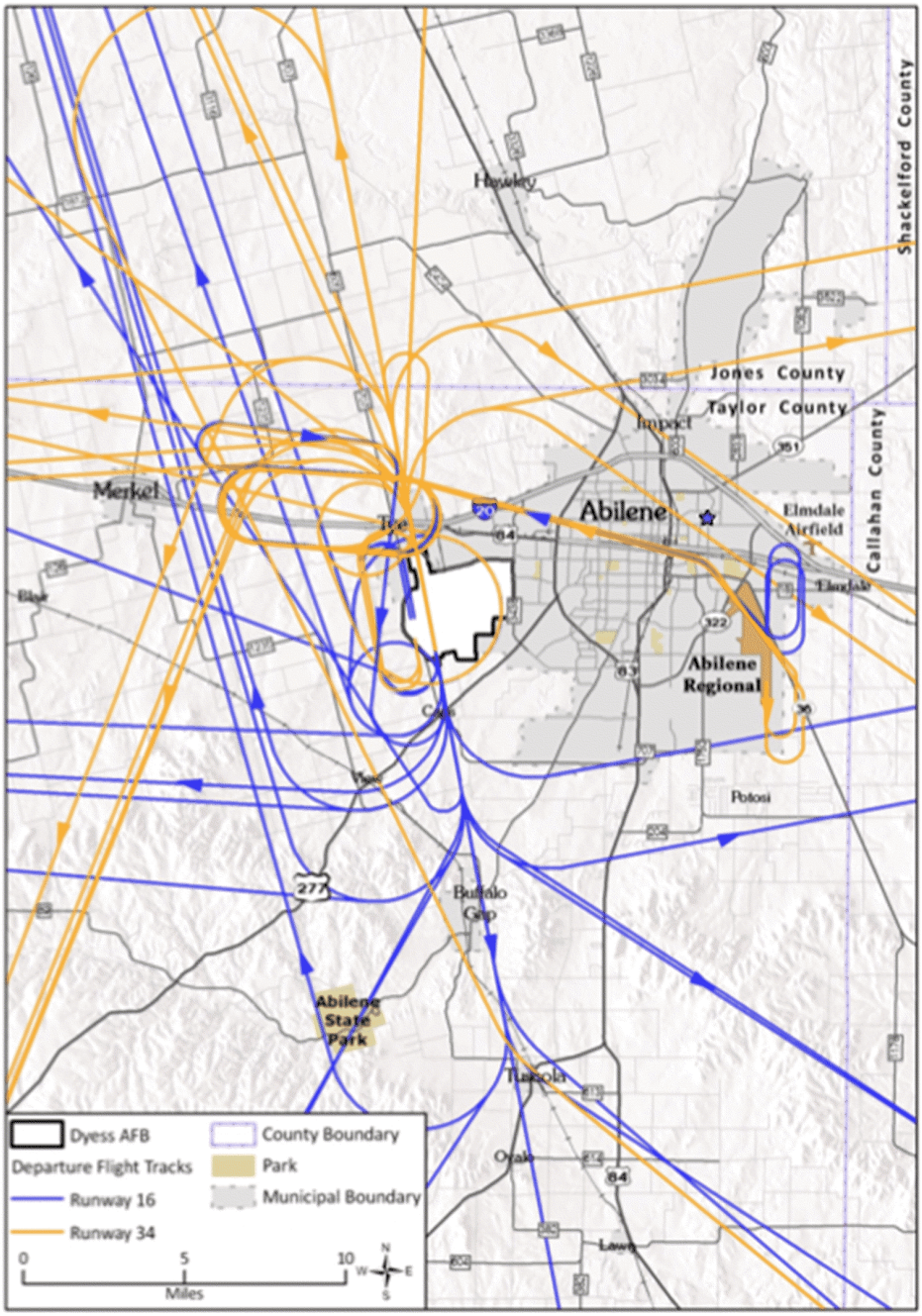
Data from National Pipeline Mapping System [Reference 2.2-10]

Figure 2.2-5 Dyess Air Force Base Arrival Flight Patterns



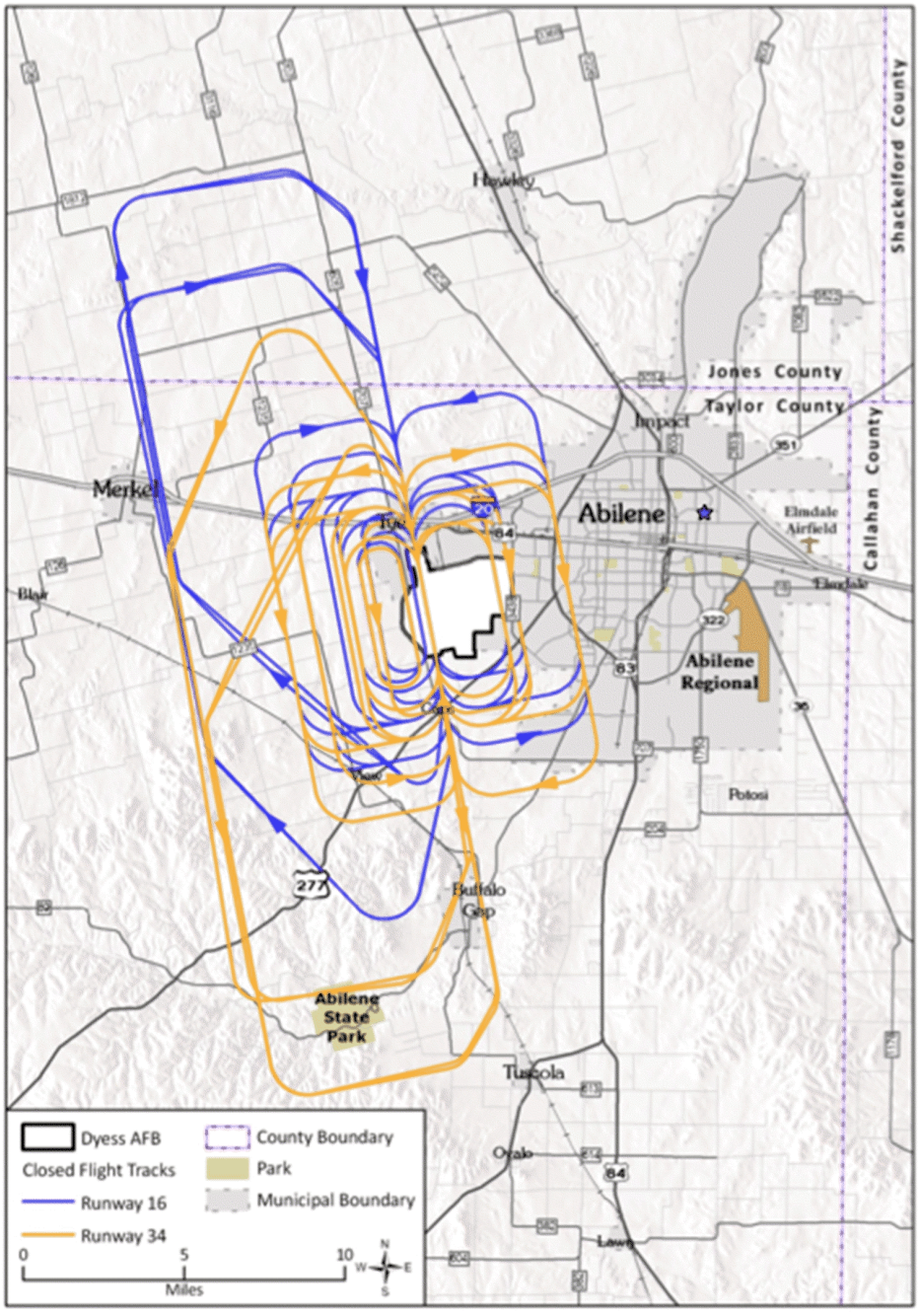
Adapted from Dyess Air Force Base [Reference 2.2-2]

Figure 2.2-6 Dyess Air Force Base Departure Flight Patterns



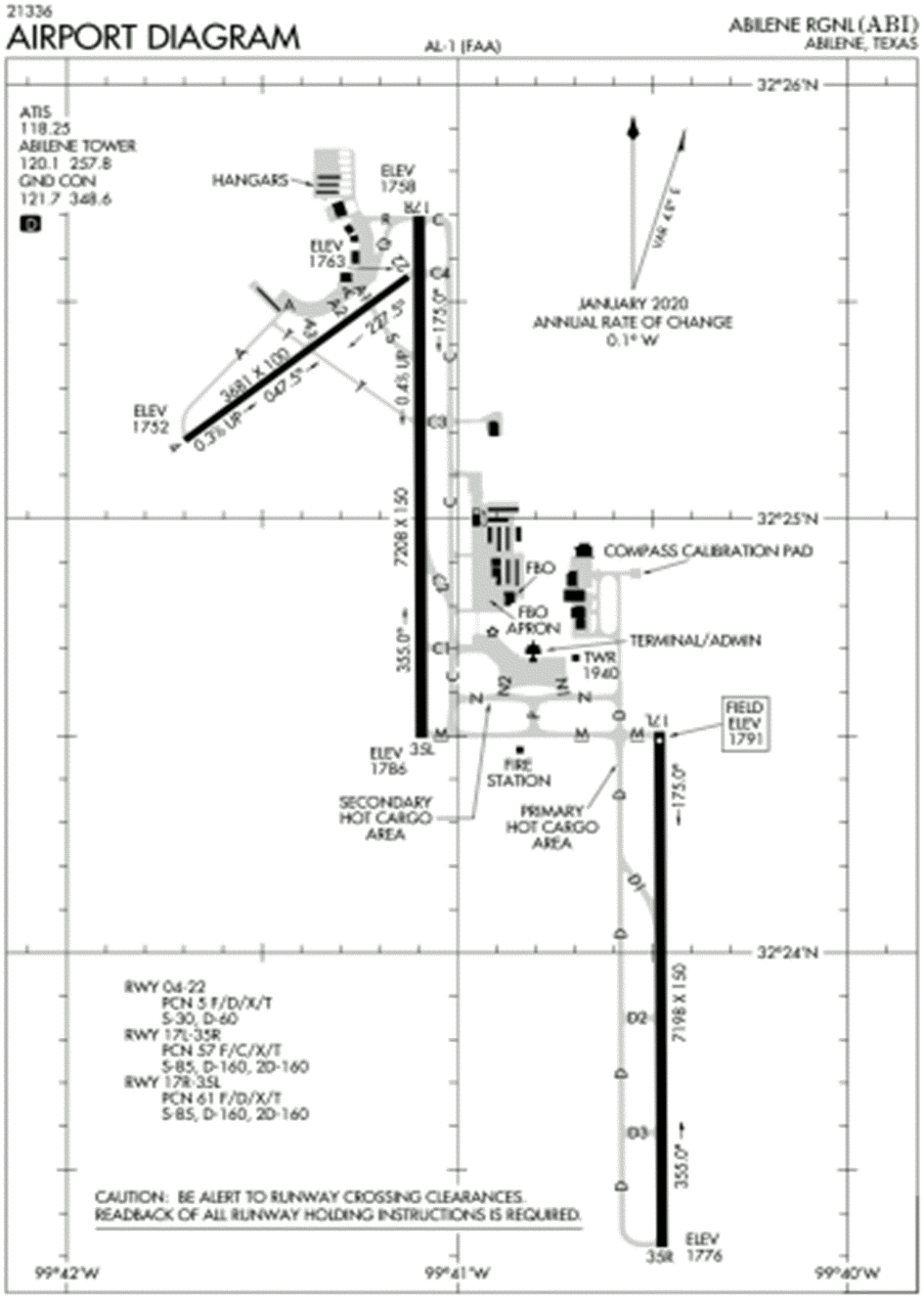
Adapted from Dyess Air Force Base [Reference 2.2-2]

Figure 2.2-7 Dyess Air Force Base Closed Pattern Flight Patterns



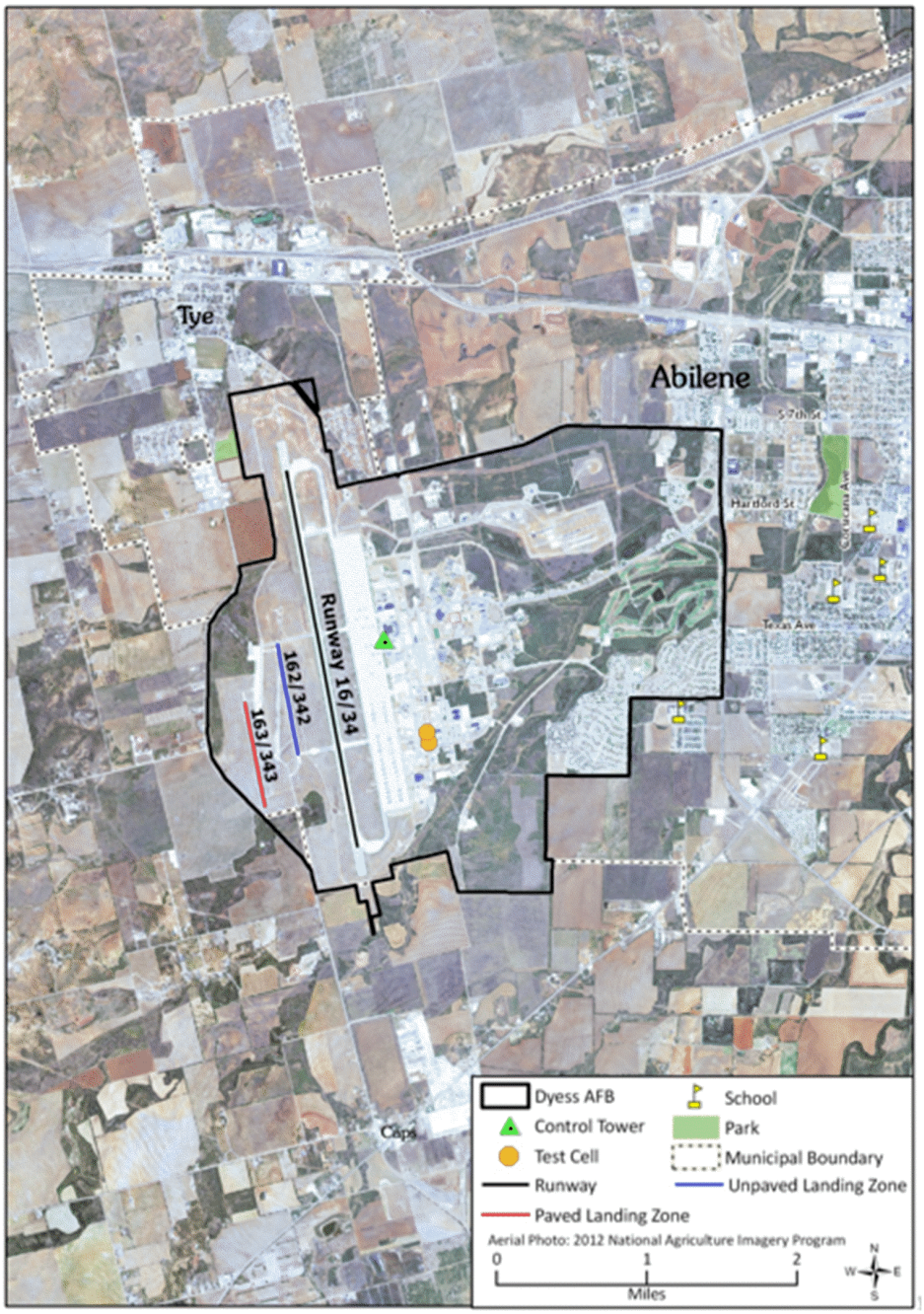
Adapted from Dyess Air Force Base [Reference 2.2-2]

Figure 2.2-8 Abilene Regional Airport Layout



Taken from Abilene Regional Airport archives

Figure 2.2-9 Dyess Air Force Base Layout



From Dyess Air Force Base [Reference 2.2-2]

2.3 Meteorology

This section describes the climate and meteorological characteristics of the site and its surrounding areas. Both average and extreme conditions are included.

2.3.1 General and Local Climate

2.3.1.1 Temperature and Precipitation

This section describes the climate of the Abilene region, beginning with characteristics of common weather phenomena and proceeding to analyses of extreme phenomena, with an emphasis on tornado history. Certain climatic features are not included because they could have no credible effect on the reactor. These are air masses, pressure systems, sleet, and relationships between synoptic-scale atmospheric processes and local meteorological conditions. A ranked list of heaviest daily snowfalls occurring in Abilene, recorded at Abilene Regional Airport as part of the National Oceanic and Atmospheric Administration's (NOAA) Global Historical Climatology Network – Daily (GHCN-Daily) data set [Reference 2.3-1] is shown in Table 2.3-1. The data captured in Table 2.3-1 is indicative of the area within a 50-mile radius of the ACU MSRR site. Table 2.3-2 and Table 2.3-3 report data derived from the NOAA database for Abilene record temperature and precipitation from 1948 to 2022.

Precipitation data is summarized in Figure 2.3-1, and extreme monthly temperatures are summarized in Figure 2.3-3. The data for both figures was recorded by Abilene Regional Airport and reported by NOAA [Reference 2.3-8]. Temperatures summarized in Figure 2.3-2 were recorded by Dyess Air Force Base and processed by the Iowa Environmental Mesonet [Reference 2.3-2]. Dyess Air Force is located within Abilene corporate limits.

2.3.1.2 Humidity

The monthly average humidity recorded at Dyess Air Force base for the region of Abilene is shown in Figure 2.3-4. The average humidity in Abilene varies throughout the year with extremes from 0 percent relative humidity to nearly 100 percent throughout the year.

2.3.1.3 Wind

Data from the NOAA National Climatic Data Center shows that Abilene's record wind gust was 78 miles (125 km) per hour recorded at Abilene Regional Airport on June 15, 1983, as part of the NOAA's GHCN-Daily data set [Reference 2.3-1]. Abilene's average wind speed measured over 51 years is 11.3 miles (18 km) per hour [Reference 2.3-2]. Table 2.3-4 provides frequency data for local damage reports due to wind speed in Abilene. Figure 2.3-5 shows a 51-year wind rose plot from the Iowa State University Mesonet system [Reference 2.3-2] using data collected at Abilene Regional Airport.

2.3.1.4 Severe Weather Phenomena

The frequency of severe weather phenomena, excluding tornadoes, can be seen in Figure 2.3-6. Hail and thunderstorms are common in Abilene. Data for hail extremes are reported in Table 2.3-6, Table 2.3-7, and Table 2.3-8. Extreme thunderstorm data are presented in Table 2.3-9. The frequency and severity of tornadoes is treated below.

2.3.1.5 Hurricane Risks

Hurricane winds are mainly a concern for coastal locations, as shown by the wind speed contours presented in Regulatory Guide (RG) 1.221, Revision 0. The RG is for power reactors and not applicable for the MSRR, but is referenced with regard to the hurricane wind speed exceedance probabilities contours. As shown in Figure 2.3-7, the MSRR site is outside an exceedance probability of 10^{-7} for nominal three-second, 130-mile (209 km) per hour gust wind speeds and is estimated to have an exceedance probability of 10^{-7} for nominal three-second, 120-mile (193 km) per hour gust speeds.

Because of the significant inland distance from the Gulf of Mexico of approximately 400 miles (644 km), tropical storm impacts are rare at the site and are mostly from storm remnants. Impacts are generally restricted to flood events from heavy rains previously addressed. According to NOAA's Office of Coastal Management's records, since 1842 there have been five tropical storms or depressions within a 50 nautical mile (57.5-mi/80-km) radius of the site as illustrated in Figure 2.3-8 [Reference 2.3-3]. Although some of these were initially classified as hurricanes, all were classified as tropical storms or depressions with maximum surface wind speed of 39 miles (63 km) per hour using a one-minute average before they reached Abilene.

2.3.1.6 Tornado Risks

Abilene sits on the southern edge of the geographical region dubbed Tornado Alley and the areas surrounding Abilene have seen several tornadoes dating back to the start of NOAA official record keeping in 1950. Figure 2.3-9 displays the magnitude and distance from the city of tornadoes between 1950 and 2010. The city of Abilene was struck twice by Fujita Scale 3 (F-3) tornadoes and once by an F-2 tornado between 1950 and 2013. Figure 2.3-10 shows their tracks through the region. Later, in 2019 and 2020, tornado outbreaks in the area spawned (Enhanced Fujita) EF 2 rated tornadoes that damaged homes [Reference 2.3-4] and caused minor damage to the prison complex northeast of the city [Reference 2.3-5].

Probabilities for tornadoes of different scale ratings are calculated considering only tornadoes rated F-2 or stronger because only they were deemed able to cause significant damage. The probability of impact is calculated based on Abilene's 1° x 1° latitudinal and longitudinal box, as defined by Pacific Northwest National Laboratory (PNNL) for the NRC [Reference 2.3-6]. The expected number of tornados per century is 360, which corresponds to the number of recorded

tornado events of any magnitude occurring within the 1° x 1° box between 1950 and 2003. Because this value is related to any tornado magnitude and not just F2 or higher, it adds a level of conservatism to the probability of impact calculations. The probabilistic area is calculated by multiplying the probabilistic lengths and widths for each tornado classification. Because there is a Weibull distribution for both the lengths and widths, the 90th, 75th, and 25th percentile values are used [Reference 2.3-7]. The likelihood values for each tornado category are based on historical data in the central U.S. from 1950 to 2003 [Reference 2.3-6] and are included in Table 2.3-5. The probability of impact per century is found by multiplying the likelihood of that tornado by the expected number of tornadoes per century by the ratio of the area of the reactor to the area of Abilene box.

$$P = \text{Tornado Likelihood} \times \frac{\text{Tornadoes per Century in Abilene Cell}}{\text{Area of Abilene Cell}} \times \text{Probabilistic Area of Tornado} \times \frac{\text{Area of Reactor site}}{\text{Area of Abilene cell}}$$

where the Area of Abilene Cell is 4,022 miles^2 (10,419 km^2) and the Area of Reactor site is 15 acres or 0.023 square mile (0.0607 km^2). The 1° latitude by 1° longitude box where Abilene is located is recorded by marking the four corners of the box on Google Maps and measuring the distance between the corners, taking the average value of the two distances for each set of corners. The average length and width of the box are multiplied together for the area of the box.

The calculated probabilities for each category of tornado are all below 1×10^{-7} per century at the 90-percent confidence level as shown in Table 2.3-5. The 25th, 75th, and 90th percentile values for the probabilistic lengths and widths of tornadoes are acquired from two images of box and whisker plots created by Harold Brooks. The 50th percentile values for the probabilistic lengths and widths are taken from Table 1 and Table 2 of the Harold Brooks document [Reference 2.3-7].

Tornado likelihood data comes from Table 2-10 of the PNNL analysis [Reference 2.3-6]. The number of segments for each tornado category are divided by the total number of segments to calculate the likelihood of each tornado size. A tornado segment includes a portion or all of a tornado, where approximately 90 percent of all tornadoes consist of one segment.

2.3.2 Site Meteorology

Details regarding the long-term dispersion modeling, modeling inputs, interpretation of modeling results, and complete description of the meteorological program for the MSRR will be provided in the Operating License application.

2.3.3 References

2.3-1 Menne, Matthew J., Imke Durre, Bryant Korzeniewski, Shelley McNeill, Kristy Thomas, Xungang Yin, Steven Anthony, Ron Ray, Russell S. Vose, Byron E.Gleason, and Tamara G. Houston (2012): Global Historical Climatology Network - Daily (GHCN-Daily), Version 3. NOAA National Climatic Data Center. doi:10.7289/V5D21VHZ [Oct. 9, 2023].

2.3-2 Iowa Environmental Mesonet, Iowa State University. [Online]. Available: https://mesonet.agron.iastate.edu/sites/locate.php

2.3-3 Historical hurricane tracks, NOAA Office for Coastal Management. [Online]. Available: https://coast.noaa.gov/hurricanes/#map=4/32/-80

2.3-4 NOAA National Weather Service, “San Angelo, TX, storm damage survey, May 18th 2019 tornado outbreak.” Accessed on Feb. 11, 2022. [Online]. Available: https://www.weather.gov/sjt/ESRIStormSurveyMap2?yyyymmdd=20190518

2.3-5 L. Gutschke, R. W. Erdrich, G. Jaklewicz, and N. Ellsworth, “Tornadoes first strike Tye, then French, Middleton prisons north of Abilene, Abilene Reporter News, March 19, 2020. [Online]. Available: https://www.reporternews.com/story/news/2020/03/19/ tornado-damages-tye-near-dyess-afb-air-force-base-abilene-texas-minor-injuries/2873370001/

2.3-6 J.V. Ramsdell, Jr. and J. P. Rinshel, “Tornado Climatology of the Contiguous United State,” NUREG/CR-4461, Rev. 2, Pacific Northwest National Laboratory, Feb. 2007. [Online]. Available: https://www.nrc.gov/docs/ML0708/ML070810400.pdf

2.3-7 H. E. Brooks, “On the relationship of tornado path length and width to intensity,” Weather and Forecasting, vol. 19, no. 2, pp. 310-319, Apr. 2004, doi: 10.1175/1520-0434(2004)019<0310:OTROTP>2.0.CO;2

2.3-8 Lawrimore, Jay H.; Ray, Ron; Applequist, Scott; Korzeniewski, Bryant; Menne, Matthew J. (2016): Global Summary of the Month (GSOM), Version 1. NOAA National Centers for Environmental Information. https://doi.org/10.7289/V5QV3JJ5. [Accessed Feb. 11, 2022].

2.3-9 Storm Events Database, NOAA, National Centers for Environmental Information, Aug. 2022. [Online]. Available: http//www.ncdc.noaa.gov/stormevents/

2.3-10 Homeland Infrastructure Foundation-Level Data. Accessed May 3, 2022. [Online]. Available at https://hifld-geoplatform.opendata.arcgis.com/datasets/historical-tornado-tracks/about

2.3-11 Jay H.; Ray, Ron; Applequist, Scott; Korzeniewski, Bryant; Menne, Matthew J. (2016): Global Summary of the Year (GSOY), Version 1. NOAA National Centers for Environmental Information. https://doi.org/10.7289/JWPF-Y430. [Oct 9, 2023].

Table 2.3-1 Heaviest Daily Snowfalls in Abilene from 1950 to Present

Rank	Snowfall (in,)	Date
1	9.8	February 14, 2021
2	9.3	April 5, 1996
3	6.6	January 21, 1983
4	6.0	January 12, 1975
4	6.0	March 20, 1970
6	5.7	March 14, 1962
7	5.5	January 10, 1973
8	5.3	January 10, 2021
9	5.0	November 28, 2001
9	5.0	January 2, 1995
11	4.5	January 1, 1996
11	4.5	February 5, 1979
13	4.4	January 13, 1982
14	4.3	December 22, 2004
15	4.1	January 1, 1983
16	4.0	February 23, 2010
16	4.0	February 20, 1964
16	4.0	December 30, 1958
16	4.0	December 28, 1954
20	3.9	February 20, 1987
20	3.9	February 3, 1956

Data from NOAA [Reference 2.3-1]

Table 2.3-2 Abilene Weather Records, 1948-2022

Record	Measurement	Date
Highest daily low	84.0°F	July 14, 2020
Lowest daily high	10.9°F	February 1, 1985
Most daily precipitation	8.3 in.	July 7, 2015
Most monthly precipitation	13.2 in.	May 1957
Most yearly precipitation	40.4 in.	2015
Earliest snow		October 27, 2020
Latest snow		April 5, 1996
Most daily snow	9.8 inches	February 14, 2021
Most monthly snow	13.5 inches	January 1973
Most yearly snow	18.6 inches	1983

Data from NOAA [Reference 2.3-1, 2.3-8, 2.3-11]

Table 2.3-3 Records of Most Consecutive Days of Weather Phenomena in Abilene, Texas

Phenomenon	# of Consecutive Days	Date
With precipitation	10	Jan 9-18, 1949
Without precipitation	73	Jun 2-Aug 13, 1970
With snow	3	Dec 26-28, 2015
High temperature ≤ 32 °F	9	Feb 10-18, 2021
High temperature ≥ 80 °F	166	May 3-Oct 15, 1956
High temperature ≥ 90 °F	78	May 27-Aug 12, 2011
High temperature ≥ 100 °F	30	Aug 3-Sep 1, 1952

Data from NOAA [Reference 2.3-1]

Table 2.3-4 Maximum Wind Speeds and Reported Damage Events

Maximum Wind Speeds Recorded (kt)	Number of Damage Reports
0-30	45
31-40	1
41-50	30
51-60	144
61-70	52
71-80	3
81-90	0
91-100	0
Unknown	25

*Unknown data points do not include a measured or estimated wind speed associated with the event that caused the reported damage.

Data from NOAA [Reference 2.3-9]

Table 2.3-5 Tornado Probabilities

Percentile	Probabilistic Area (km^2)	Probability of Site Impact per Century
EF 2 - Likelihood: 0.2114		
90	1.9	8.2E-08
75	1.6	6.9E-08
50	1.4	5.7E-08
25	1.2	4.9E-08
EF 3 - Likelihood: 0.07087		
90	9.2	1.30E-07
75	7.5	1.1E-07
50	5.9	8.5E-08
25	4.8	6.8E-08
EF 4 - Likelihood: 0.0252		
90	33	1.6E-07
75	26	1.3E-07
50	20	1.0E-07
25	15	7.6E-08
EF 5 - Likelihood: 0.00344		
50	30	2.10E-08[a]

a. Translates to 4.8E+07 years to impact

Table 2.3-6 Number of Hail Events by Size, Taylor County (1/1/1960 to 6/30/2022)

Measurement	Events
4" or larger	3
3" or larger	9
2" or larger	40
1" or larger	185

Data from NOAA [Reference 2.3-9]

Table 2.3-7 Ten Largest Recorded Hail Sizes, Taylor County (1/1/1960 to 6/30/2022)

Location	Measurement (inches)	Date
Abilene	4.5	4/24/2011
Abilene	4.5	6/12/2014
Abilene	4.25	2/22/2005
Potosi	3.5	5/14/2008
Taylor Co.	3	6/1/1962
Taylor Co.	3	5/5/1967
Taylor Co.	3	4/27/1969
Abilene	3	6/7/1996
Tuscola	3	5/8/2005
Taylor Co.	2.75	5/31/1962

Data from NOAA [Reference 2.3-9]

Table 2.3-8 Number of Hail Events Per Year in Taylor County, 1960-July 2022

Year	Events	Year	Events	Year	Events
1960	0	1981	1	2002	10
1961	1	1982	4	2003	6
1962	5	1983	3	2004	3
1963	1	1984	1	2005	4
1964	1	1985	0	2006	7
1965	2	1986	0	2007	5
1966	0	1987	5	2008	7
1967	3	1988	3	2009	3
1968	2	1989	3	2010	1
1969	4	1990	4	2011	2
1970	1	1991	5	2012	5
1971	0	1992	9	2013	6
1972	2	1993	6	2014	5
1973	3	1994	9	2015	9
1974	0	1995	11	2016	6
1975	2	1996	12	2017	4
1976	5	1997	7	2018	0
1977	2	1998	2	2019	3
1978	2	1999	11	2020	4
1979	3	2000	6	2021	3
1980	1	2001	6	2022	0

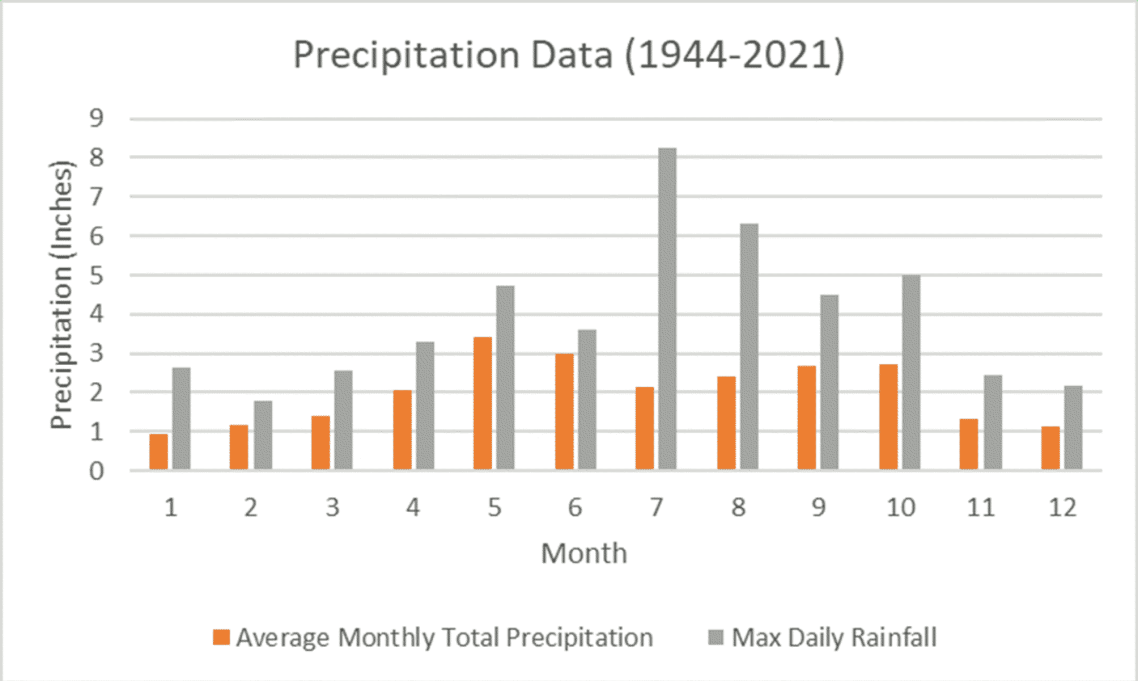
Data from NOAA [Reference 2.3-9]

Table 2.3-9 Taylor County Thunderstorm Wind Events (1/1/1960 to 6/30/2022)

Month	Thunderstorm Wind Velocity (mph)				
	70-74	75-79	80-84	85-89	90-94
January	-	-	-	-	-
February	-	-	-	-	-
March	-	1	1	-	-
April	2	-	-	-	-
May	4	6	1	-	-
June	9	3	5	-	1
July	1	1	-	-	-
August	3	-	-	-	-
September	1	-	1	-	-
October	-	1	-	-	-
November	-	-	-	-	-
December	-	-	1	-	-

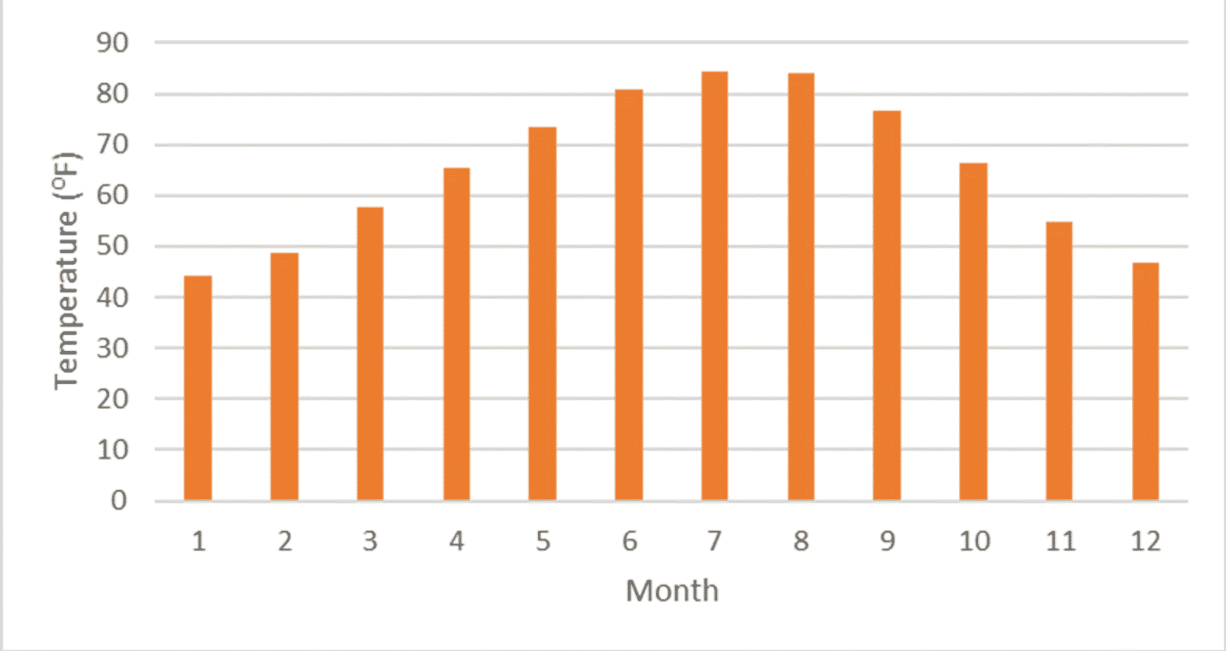
Data from NOAA [Reference 2.3-9]

Figure 2.3-1 Rainfall Distribution for Abilene from 1948-2021



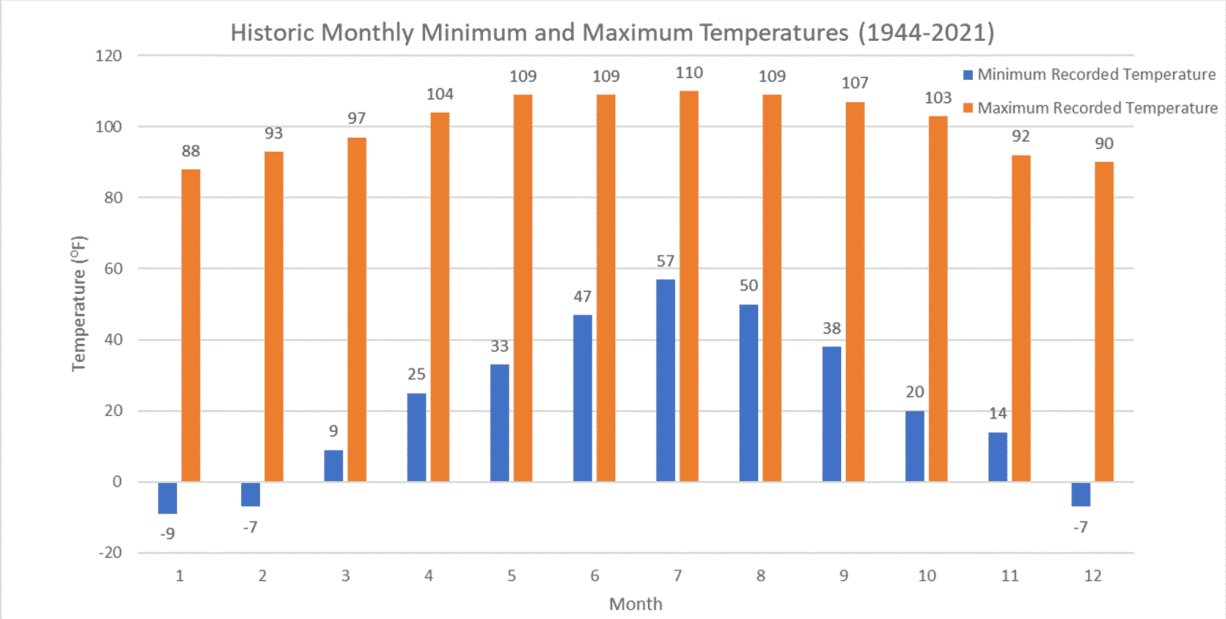
Data from NOAA [Reference 2.3-8]

Figure 2.3-2 Average Monthly Temperature Distribution for Abilene from 1970-2021



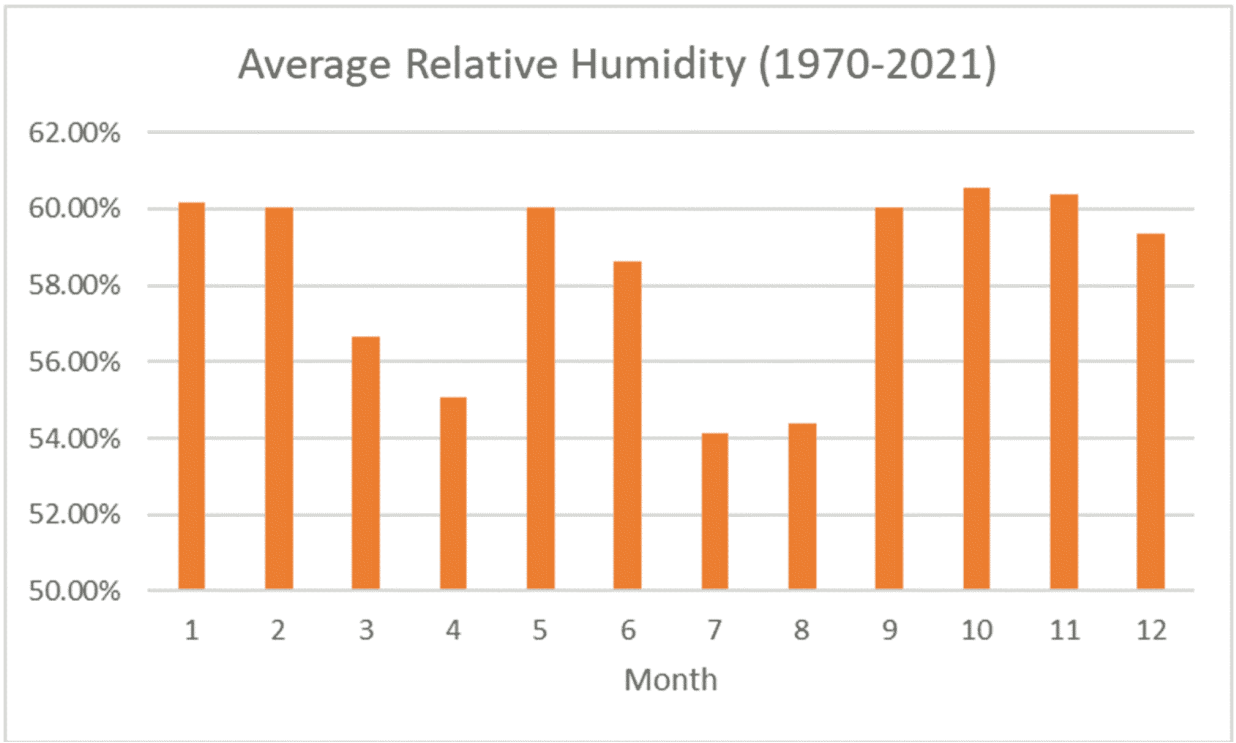
Data from Iowa Environmental Mesonet [Reference 2.3-2]

Figure 2.3-3 Maximum and Minimum Monthly Temperatures for Abilene from 1948-2021



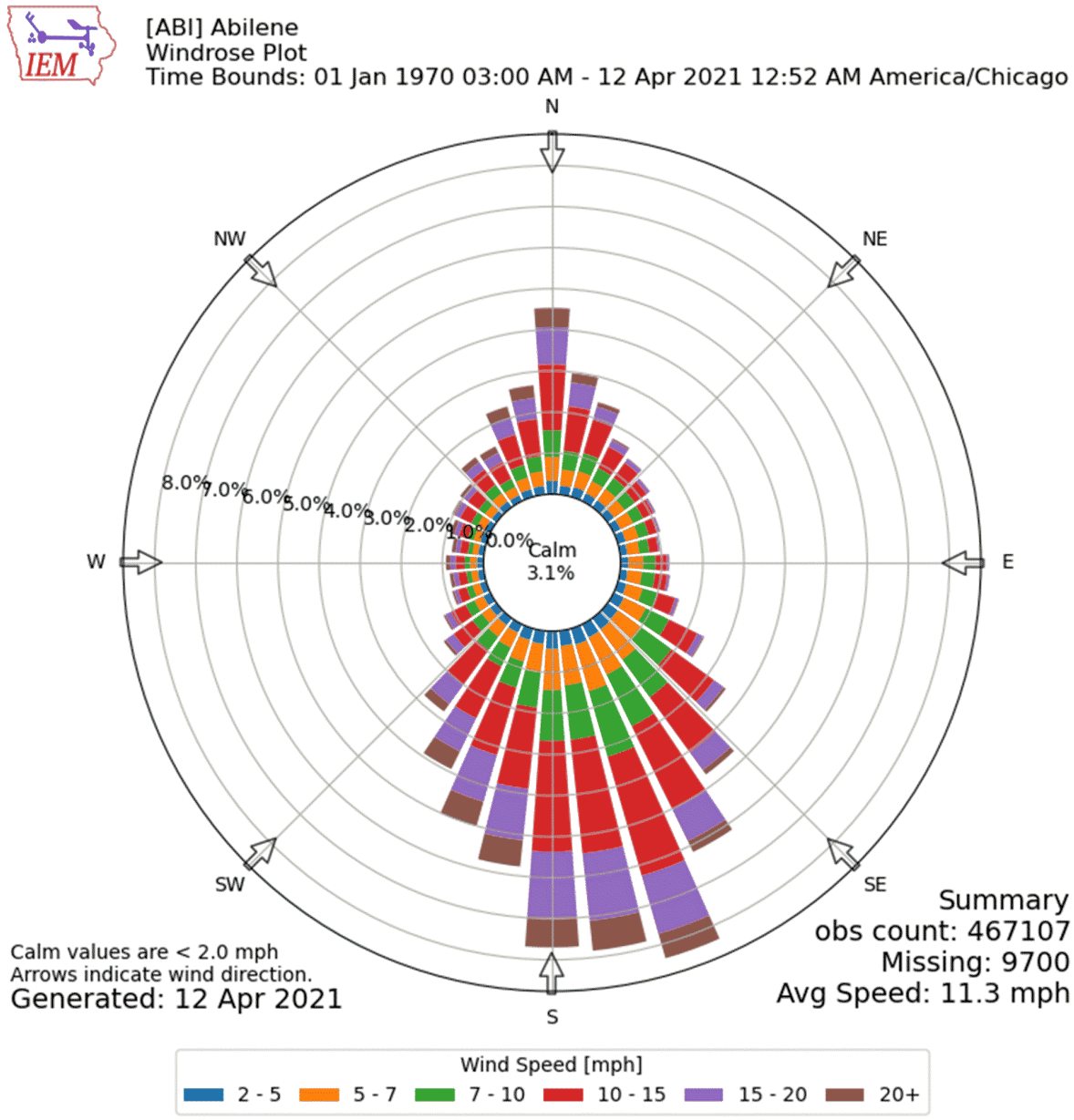
Data from NOAA [Reference 2.3-8]

Figure 2.3-4 Distribution of Monthly Relative Humidity in the region of Abilene between 1970 and 2021



Data from Iowa Environmental Mesonet [Reference 2.3-2]

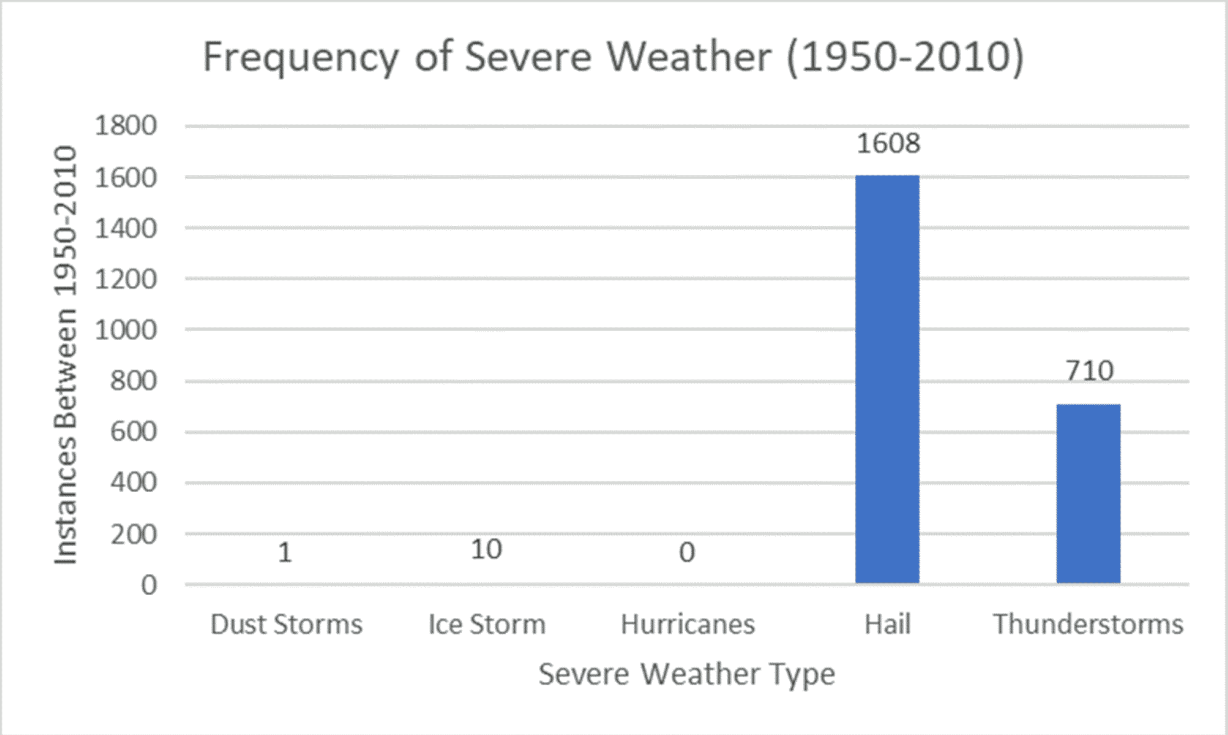
Figure 2.3-5 Wind Speed, Direction, and Frequency Measured at Abilene Regional Airport



"Obs count" in the Summary indicates the number of observations comprising the data.

From Iowa Environmental Mesonet [Reference 2.3-2]

Figure 2.3-6 Number of Severe Weather Events within 50 miles of Abilene



*"Thunderstorm Wind Events" are denoted as "Thunderstorms"

*Multiple data points in Figure 2.3-6 may be associated with the same storm or weather event depending on the location and magnitude of the damage caused.

Data from NOAA [Reference 2.3-9]

Figure 2.3-7 Design-basis Hurricane Wind Speeds from Regulatory Guide 1.221

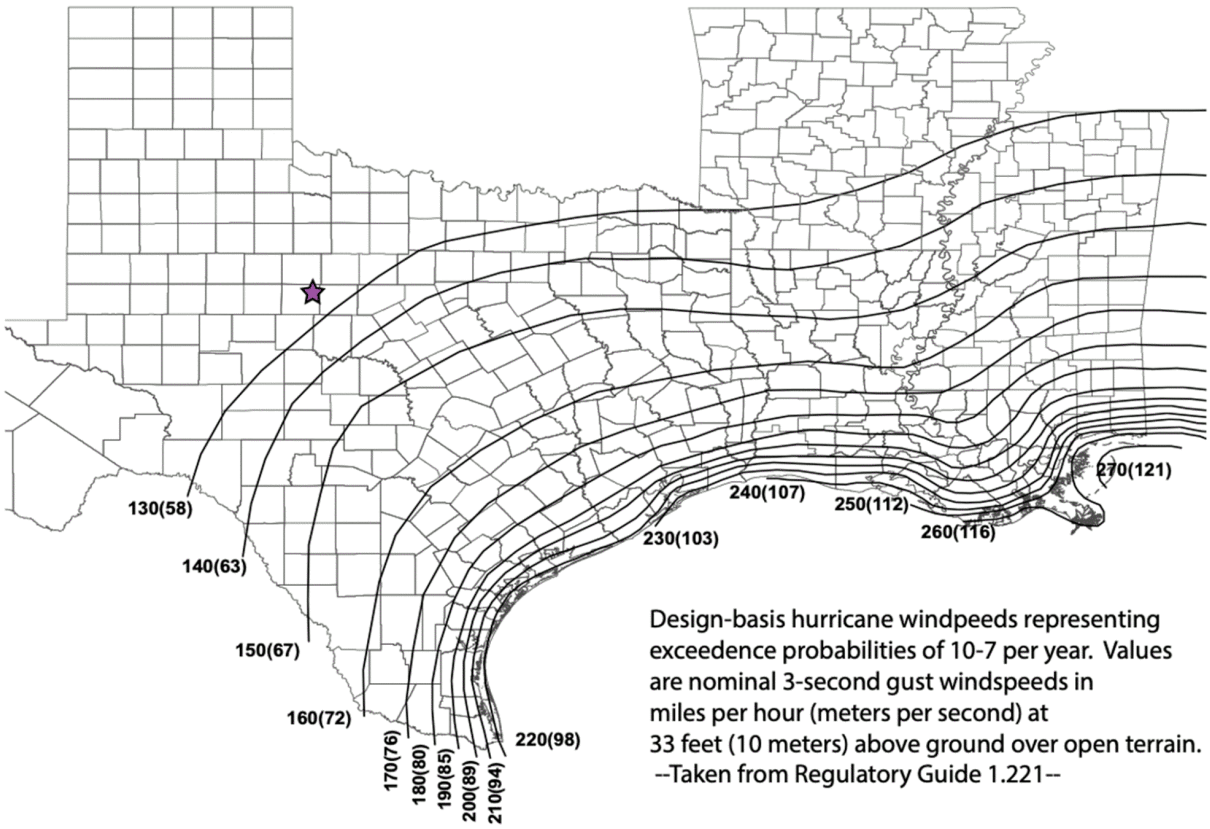
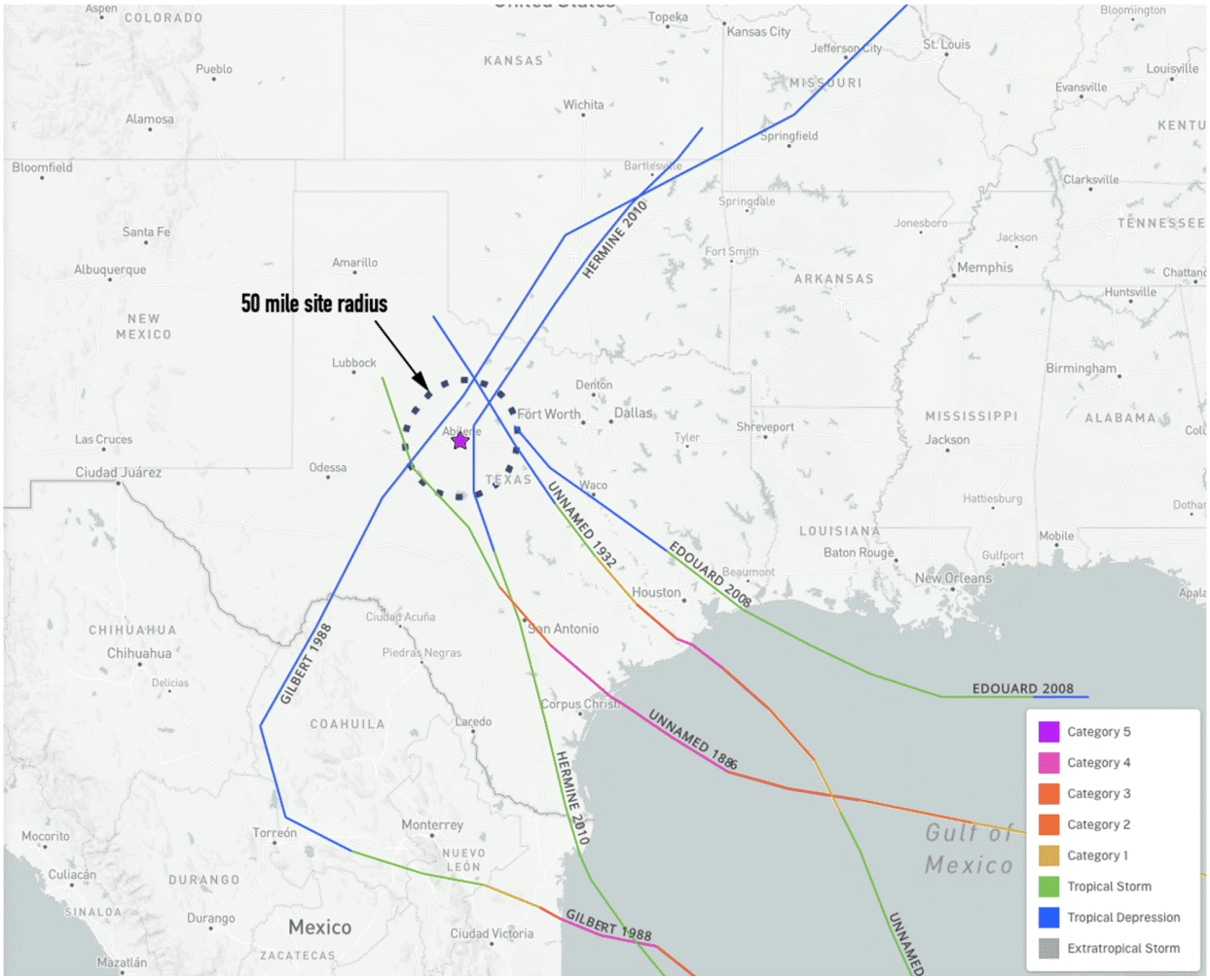
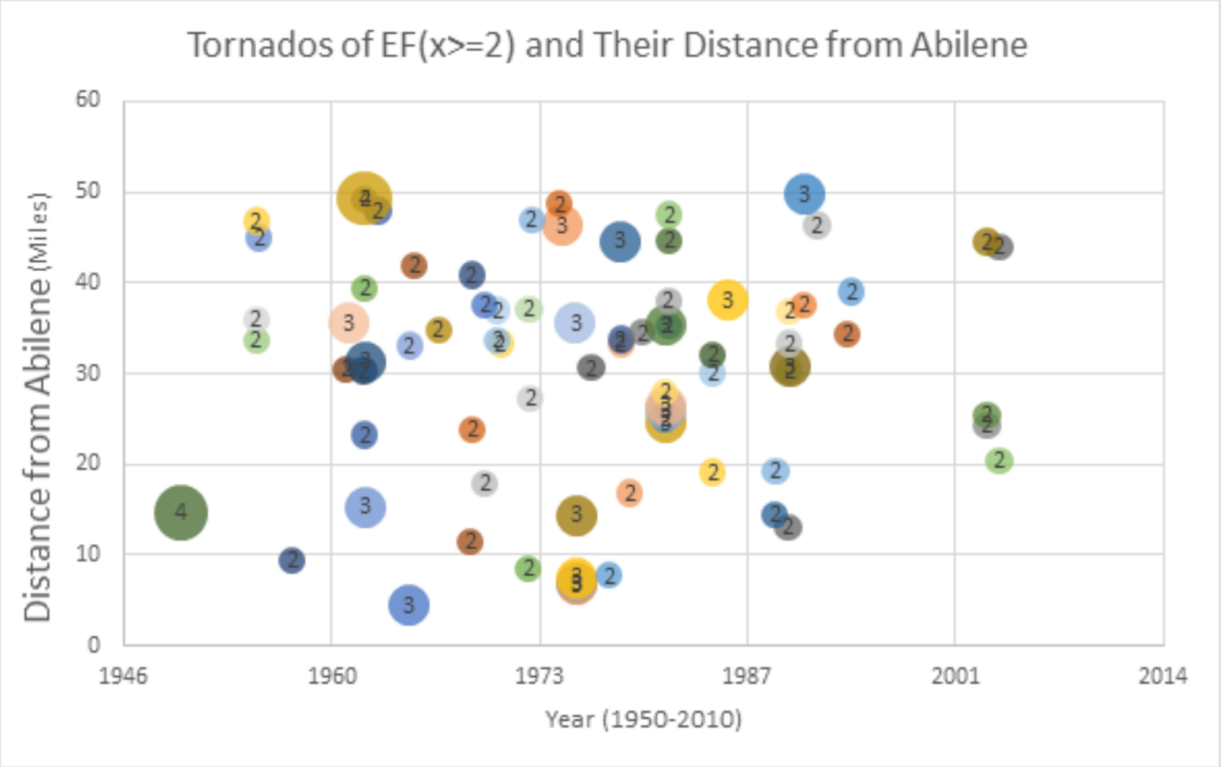


Figure 2.3-8 Hurricane Events Tracked within 50 nmi of the Reactor Site



Adapted from NOAA Office for Coastal Management [Reference 2.3-3]

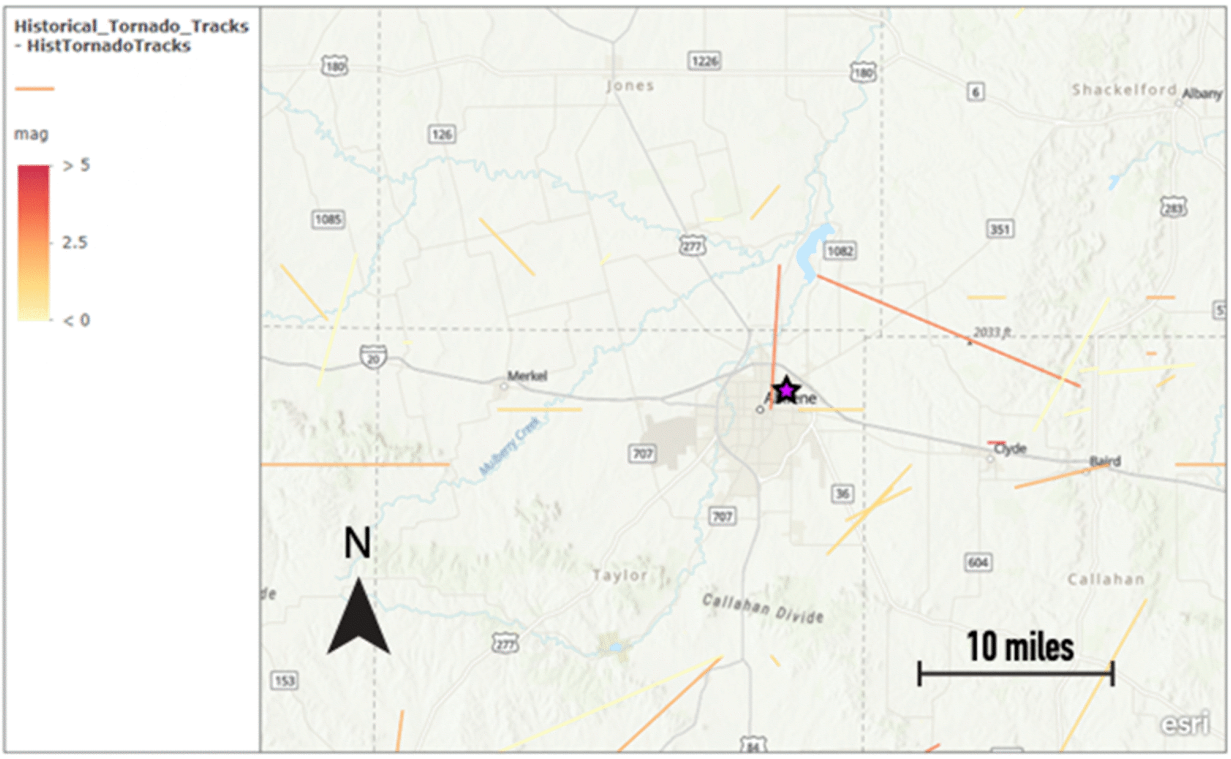
Figure 2.3-9 Tornado Magnitude and Distance from Abilene, 1950-2010



*Numbers in bubbles represent the magnitude of the recorded tornado event. Prior to February 2007 tornados were characterized with the Fujita Scale, after February 2007 tornados were characterized with the Enhanced Fujita Scale. Color variation is used to differentiate between recorded tornado events. Multiple data points in Figure 2.3-9 may be associated with the same weather event depending on the location and magnitude of the damage caused.

Data from NOAA [Reference 2.3-9]

Figure 2.3-10 Map of Tornado Pathways in Abilene from 1950-2013



Adapted from HIFLD website [Reference 2.3-10]

2.4 Hydrology

Information is provided in this section regarding groundwater and surface water features near the MSRR site to support analyses and evaluations of consequences of uncontrolled release of radioactive material from the facility.

This section discusses river basins of Texas and the Abilene region, local drainage and floodways, aquifers in the state and region, and local groundwater. Tsunami or “sea waves” reaching the MSRR site are deemed non-credible because the site location is approximately 400 miles (644 km) from the Gulf of Mexico at an elevation of greater than 1,700 ft (518 m) above sea level. The risk of flood near the reactor is deemed negligible, whether from drainage runoff, ground movement, dam failure, or creek and river blockages.

2.4.1 Drainage and Floodways

Texas river basins are shown in Figure 2.4-1. Abilene sits on the edge of the Brazos River Basin, near the Clear Fork tributary of the Brazos River, highlighted in Figure 2.4-2, and can be seen in relation to the other river basins in the state. Water from and around Abilene does not drain into this tributary because the area has no underground drainage system. Instead, water drains over land via surface channels and creeks. City-built detention ponds along creeks allow water to run off into the nearby reservoirs, Lake Kirby, Lytle Lake, and Lake Fort Phantom Hill.

The creeks nearest the MSRR are Cedar Creek to the west and the much smaller Rainy Creek to the east. Both are within 4,000 ft (1,219 m) of the site as shown in Figure 2.4-3. Neither of these creeks pose a flood risk to the MSRR site. The site is well outside FEMA-designated floodways as shown in Figure 2.4-4. The nearest 500 year floodway is to the east, more than 2,000 ft (610 m) away and 25 ft (7.6 m) below the site elevation.

Lytle Lake lies nearly two miles (3.2 km) south of the site and is the source for Lytle Creek, which feeds into Cedar Creek more than one mile (1.6 km) southwest of the site. The elevation of the Lytle Lake dam is 1,713 ft (522 m) above sea level, 20 ft (6 m) below the site, and poses no risk of flooding the site if breached. The Emergency Action Plan for Lytle Lake dam (NID ID: TX02705) prepared for the Texas Commission on Environmental Quality contains an inundation map and breach analysis for the Abilene area [Reference 2.4-1]. The potential inundation area, the area potentially affected by a dam failure shows that the MSRR site is outside the affected area.

Kirby Lake dams Cedar Creek south-southwest of the site. The Kirby Lake dam elevation is 1,786 ft (544 m) above sea level, 53 ft (16 m) higher in elevation than the site, but is located more than 5.5 miles (8.9 km) away. The Cedar Creek floodway, nearly a mile (1.6 km) from the site, is at an elevation of 1,680 ft (512 m), which is 33 ft (10 m) below the site grade, posing no hazard to the site resulting from a dam breach. Illustrations of these features are shown in Figure 2.4-5 and Figure 2.4-6. The Kirby Lake Dam (NID ID: TX02703) Breach Analysis Report contains the breach analyses and inundation maps that show the MSRR site is outside the inundation area from a

potential breach of that dam [Reference 2.4-2]. The consequences of external and internal flooding are bounded by the LONEP analysis presented in Section 13.1.10 and the criticality analysis presented in Section 4.5.

2.4.2 Groundwater

Abilene is near the Edwards-Trinity, Trinity, and Seymour Aquifers, but rests on the expansive Cross Timbers minor aquifer as illustrated in Figure 2.4-7 and Figure 2.4-8. The nearest edge of the Seymour aquifer is just over 11 miles (17.7 km) northeast of the MSRR site, underlying parts of both Taylor and Jones Counties. Five square miles (13 km^2) of outcrop from the Seymour Aquifer underlies Taylor County, but this area has zero cubic feet of average annual baseflow. However, under Jones County is 326 square miles (844 km^2) of the Seymour Aquifer's outcrop with an average of 3.3 cubic feet (0.09 cubic meter) of baseflow [Reference 2.4-3]. The nearest part of the Edwards-Trinity Aquifer is just under 10 miles (16 km) southwest of the reactor site. The Trinity Aquifer lies 8.5 miles (13.7 km) east by southeast from the site.

Groundwater is not used for Abilene's potable water supply because it is brackish and none of the aquifers discussed above is a sole-source aquifer. The city's potable water comes primarily from Fort Phantom Hill Lake, Hubbard Creek Reservoir, and O. H. Ivie Reservoir. It is treated by the Northeast Water Treatment Plant, Grimes Water Treatment Plant, and the Hargesheimer Water Treatment Plant. The wells that dot the area are used for landscaping, agriculture, and monitoring groundwater levels (Figure 2.4-9) [Reference 2.4-4]. More than 100×10^6 gallons (379,000,000 liters) of water per day are withdrawn from the Seymour Aquifer, mostly in Haskell and Knox Counties, which are just north of Jones County. The average recharge rate of this aquifer is two inches (5 cm) per year.

During the Pleistocene era, the Seymour Formation built up on eroded bedrock surface that had developed atop Permian-age poorly permeable red beds. The Seymour Formation consists of clay, silt, sand, and gravel, which coarsen downward to the primary water-producing zone [Reference 2.4-5]. The aquifer is composed of many isolated patches of alluvial deposits, which vary in area, saturated thickness, well yields, and chemical quality. The Edwards-Trinity aquifer is composed of limestone over sand and sandstone of the Cretaceous period. Trinity formations are Cretaceous, consisting of interbedded sandstone, sand, limestone, and shale [Reference 2.4-6].

The Cross Timbers Aquifer (a minor aquifer) consists of four Paleozoic-age water bearing formations including, from oldest to youngest, the Strawn, Canyon, Cisco, and Wichita groups. The aquifer is primarily composed of limestone, shale, and sandstone [Reference 2.4-7].

2.4.3 References

2.4-1 Enprotec / Hibbs & Todd, "Lytle Lake DAM Emergency Action Plan." Aug. 2012.

2.4-2 Freese and Nichols, "Emergency Action Plan." Dec. 2012 (updated June17, 2022).

2.4-3 B. Bruun, K. Jackson, P. Lake, and J. Walker, “Texas aquifers study: groundwater quantity, quality, flow, and contributions to surface water,” Texas Water Development Board, 2016. [Online]. Available: https://www.twdb.texas.gov/groundwater/docs/studies/

2.4-4 Water Data Interactive, Texas Water Development Board. Accessed on March 8, 2022. [Online]. Available: https://www3.twdb.texas.gov/apps/WaterDataInteractive/GroundwaterDataViewer/?map=gwdb

2.4-5 P. D. Ryder, “Seymour Aquifer,” Ground Water Atlas of the United States, Oklahoma, Texas, HA 730-E USGS. Accessed on Mar. 8, 2022. [Online]. Available: https://pubs.usgs.gov/ha/ha730/ch_e/E-text4.html

2.4-6 P. D. Ryder, “Edwards-Trinity Aquifer System,” Ground Water Atlas of the United States, Oklahoma, Texas, HA 730-E USGS. Accessed on Mar. 8, 2022. [Online]. Available: https://pubs.usgs.gov/ha/ha730/ch_e/E-text8.html

2.4-7 “Cross Timbers Aquifer,” Texas Water Development Board. Accessed on Mar. 9, 2022. [Online]. Available: https://www.twdb.texas.gov/groundwater/aquifer/minors/cross-timbers.asp

2.4-8 Brazos River Authority, “Brazos river authority maps.” Accessed on Feb. 11, 2022. [Online]. Available: https://www.brazos.org/About-Us/About-the-BRA/Maps

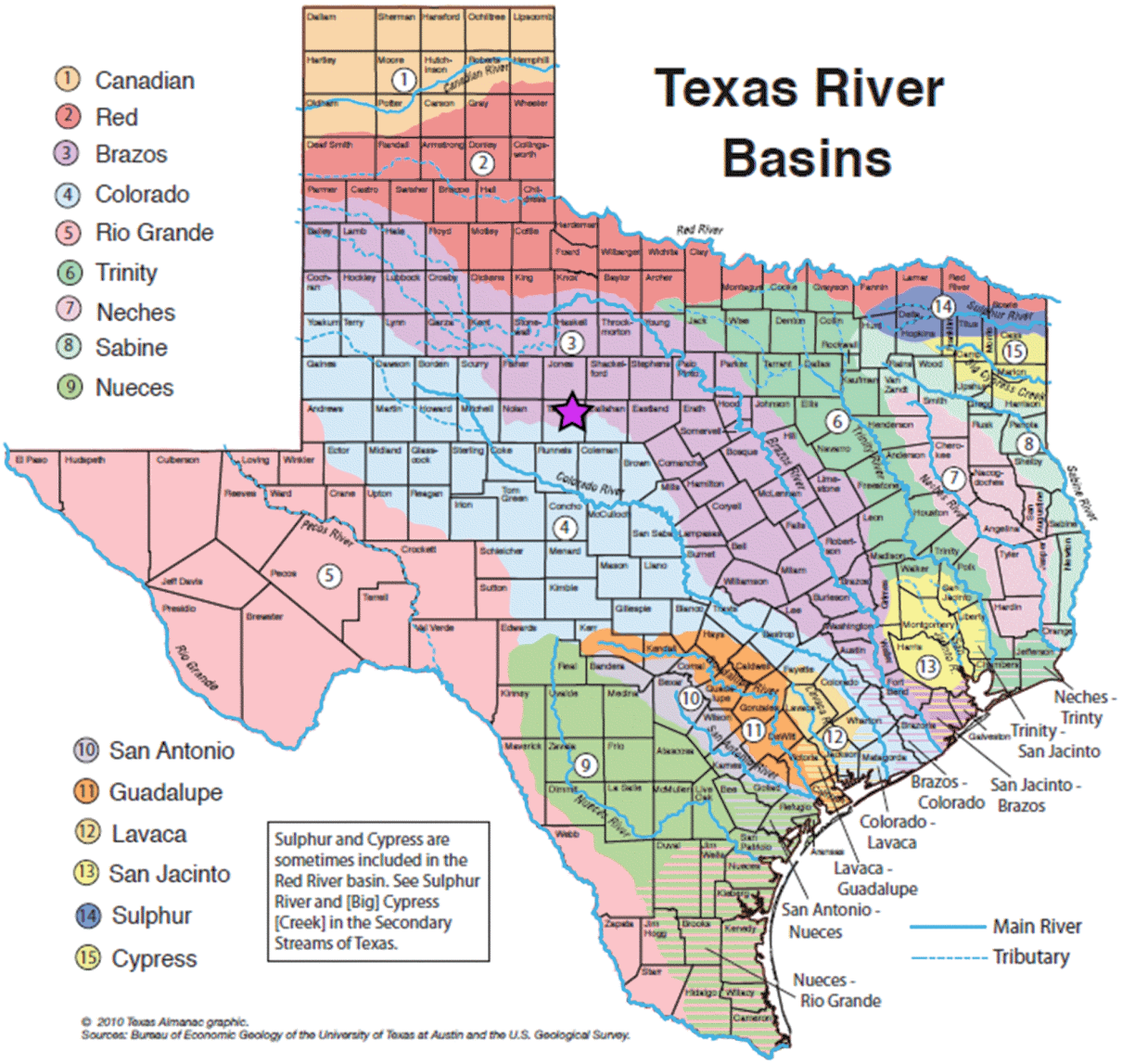
2.4-9 OSM, “OpenStreetMap,” OpenStreetMap contributors, 2021. Accessed on Feb. 13, 2022. [Online]. Available: https://www.openstreetmap.org

2.4-10 NGA, “OpenStreetMap Bright Basemap,” United States Department of Defense, National Geospatial-Intelligence Agency, 2022. Accessed on Feb. 13, 2022. [Online]. Available: https://osm.gs.mil/features/base-map

2.4-11 USGS, “National Hydrography Dataset,” United States Geological Survey, National Geospatial Program, 2022. Accessed on Feb. 13, 2022. [Online]. Available: https://apps.nationalmap.gov/downloader/

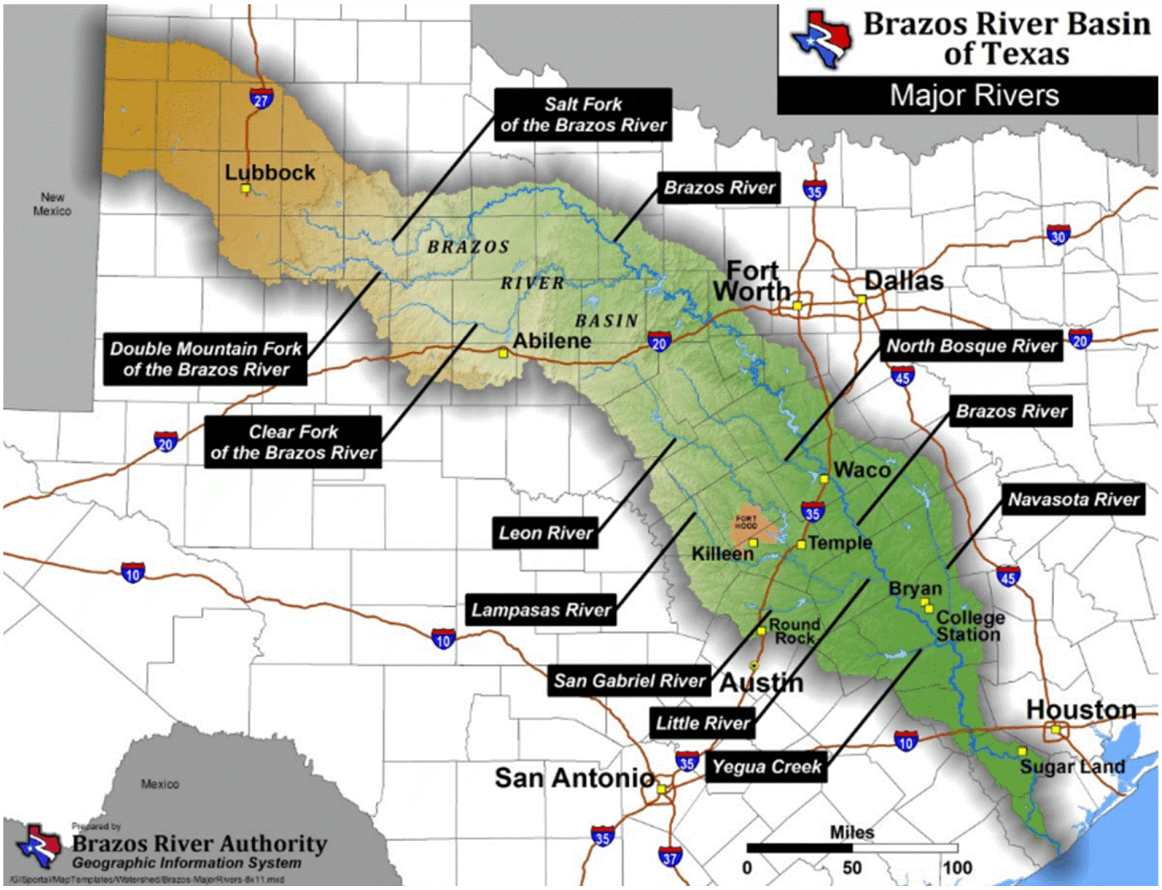
2.4-12 Flood viewer, City of Abilene Texas. [Online]. Available: https://abilene.maps.arcgis.com/apps/webappviewer/index.html?id=25cd7bb393c3417e8b272115ef810b8c

Figure 2.4-1 Texas River Basins



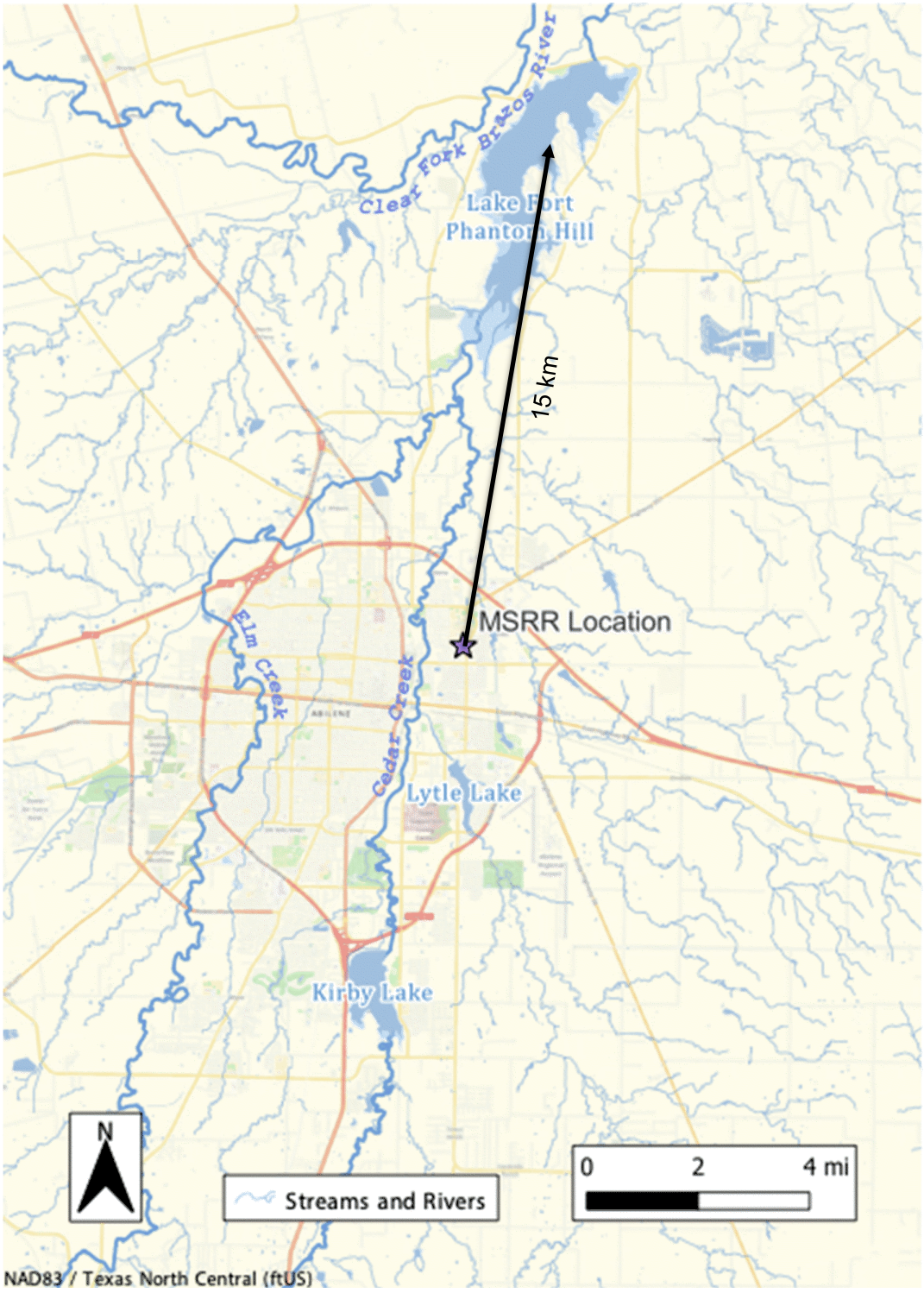
2010 Texas Almanac graphic. Sources: Bureau of Economic Geology of the University of Texas at Austin and the U.S. Geological Survey

Figure 2.4-2 Topographical Map of Brazos River Basin



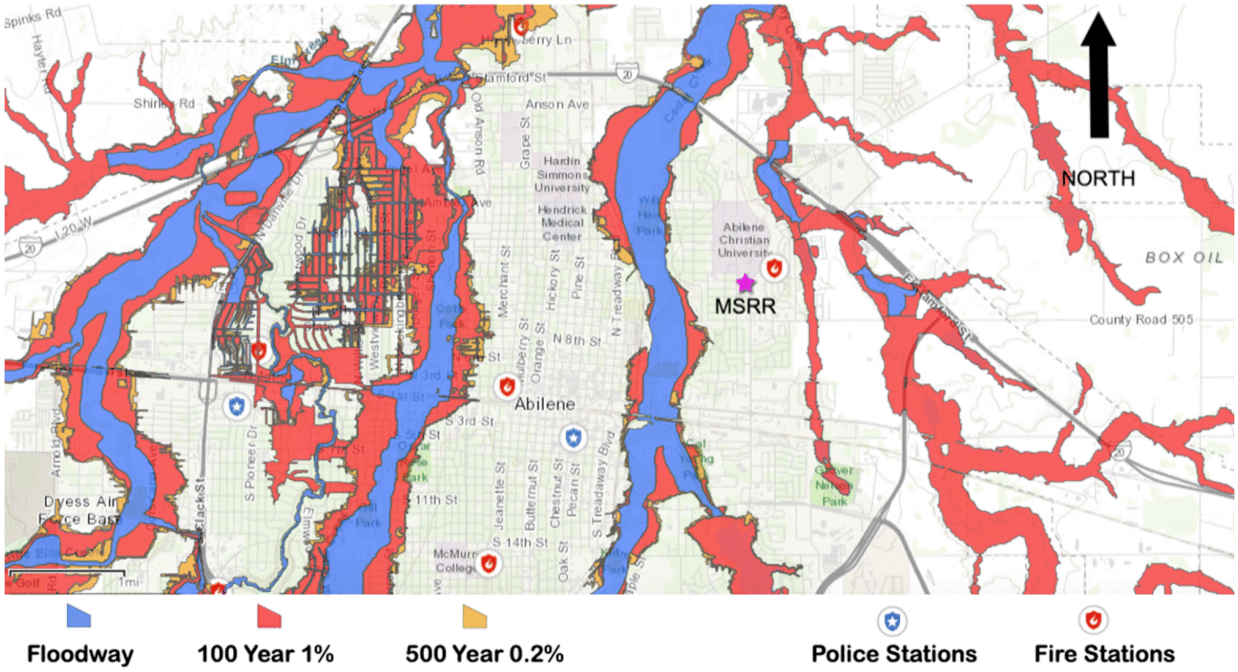
From Brazos River Authority Geographical Information System [Reference 2.4-8]

Figure 2.4-3 Streams, Rivers, and Lakes within Several Miles of the Reactor Site



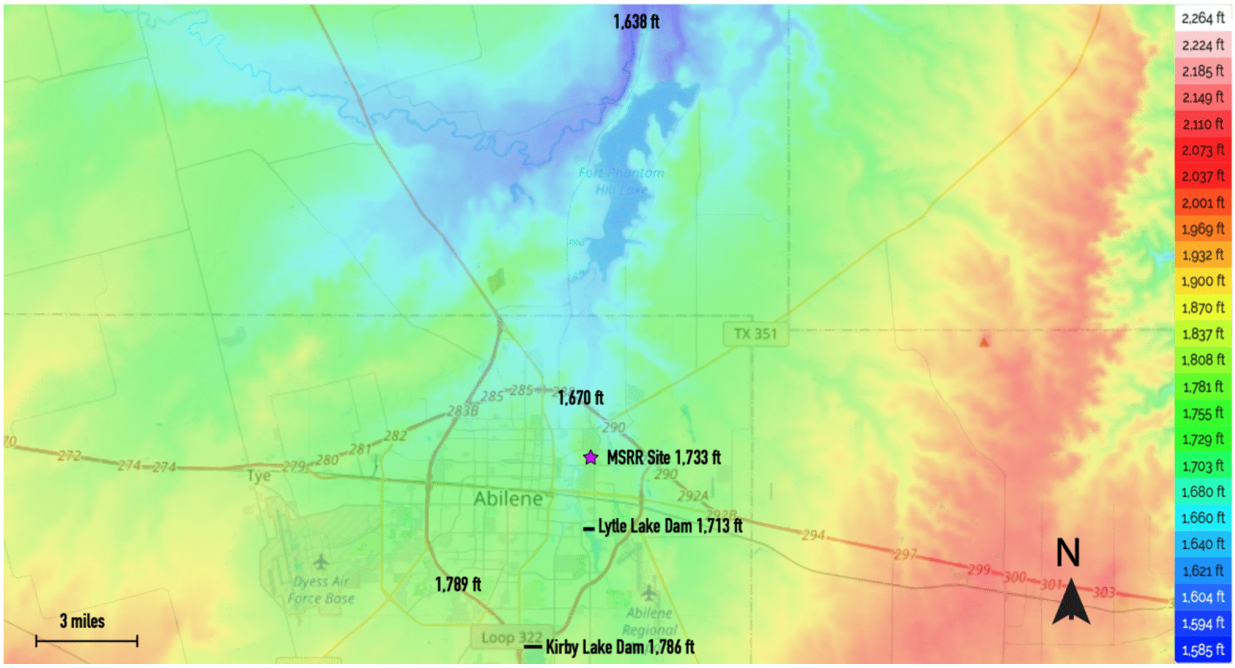
Data from OpenStreetMap [Reference 2.4-9], NGA [Reference 2.4-10], and National Hydrography Dataset [Reference 2.4-11]

Figure 2.4-4 Federal Emergency Management Agency-defined Floodways



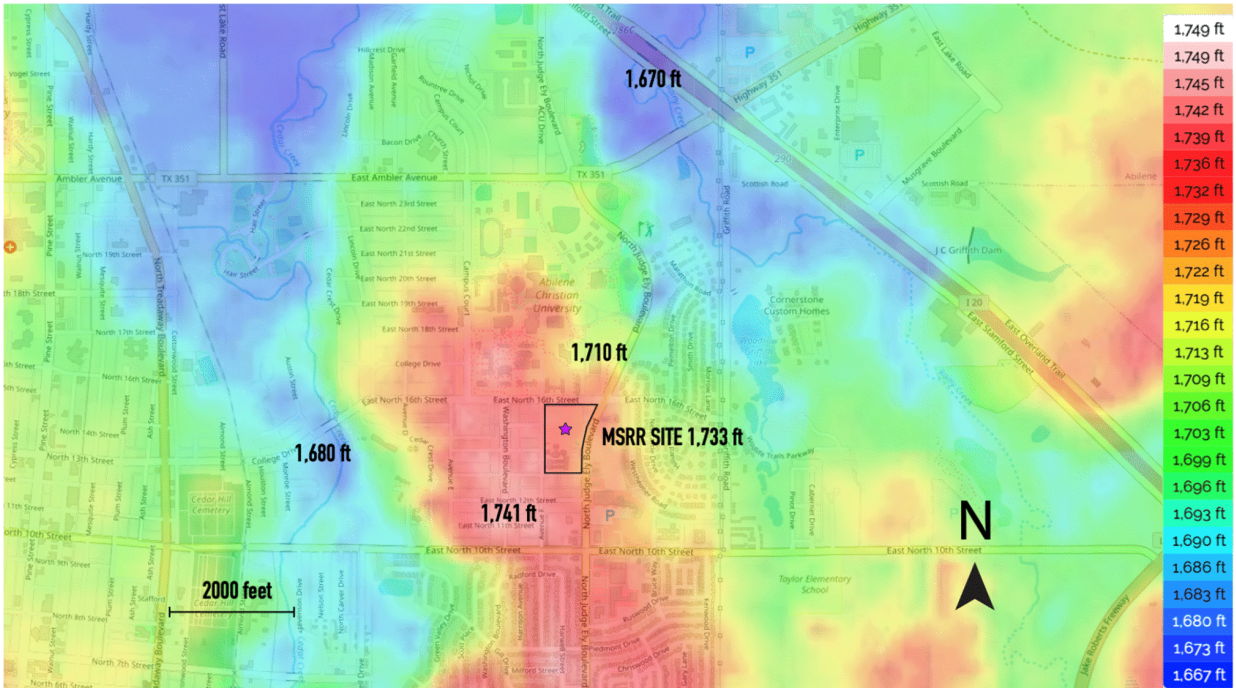
Adapted from City of Abilene Flood Viewer [Reference 2.4-12].

Figure 2.4-5 Topography of Abilene Area



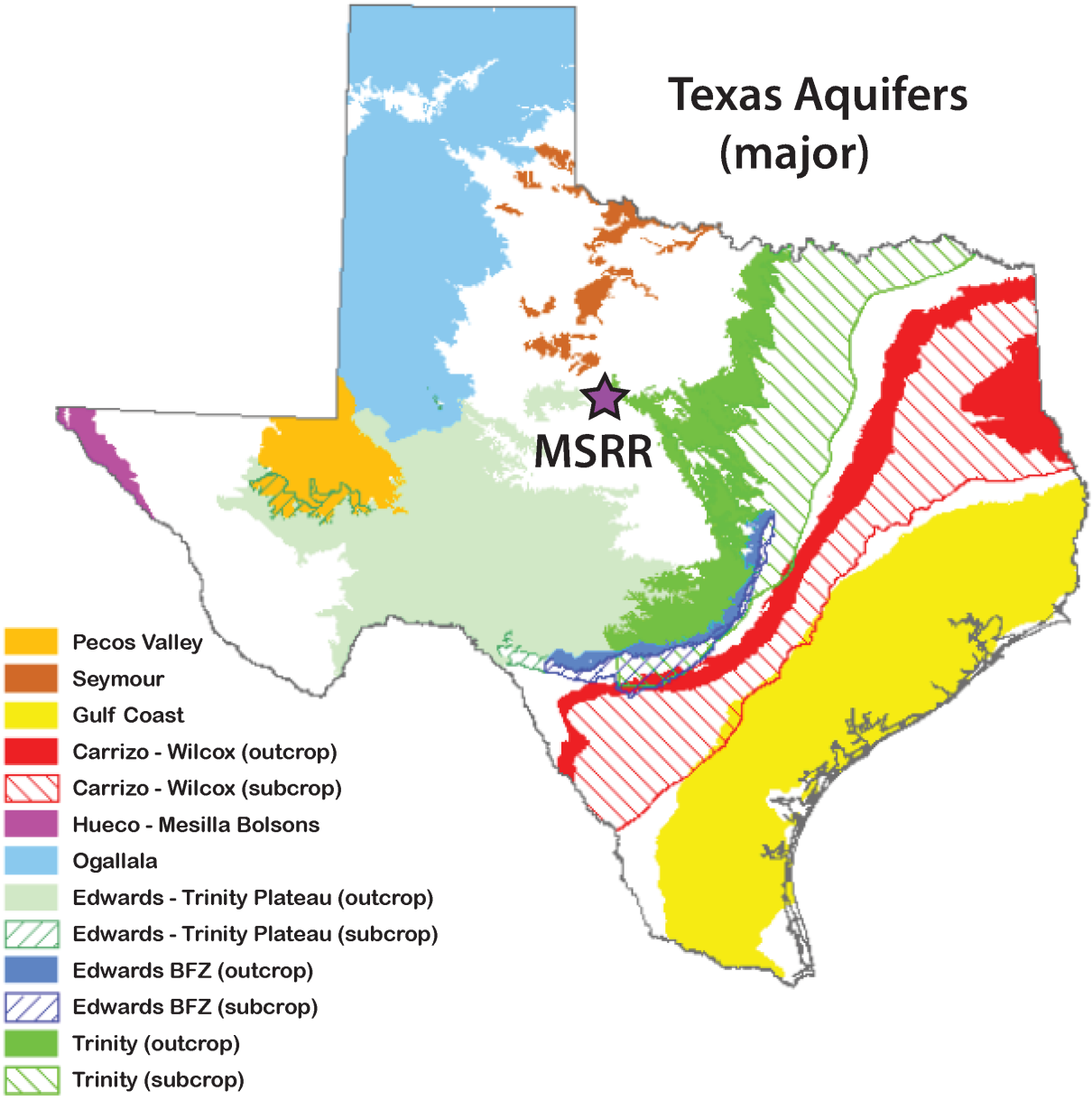
OpenStreetMap provided topography rendering. Elevation data rendered by MERIT DEM, referenced to WGS84 and EGM96.

Figure 2.4-6 Topography of Molten Salt Research Reactor Site Area



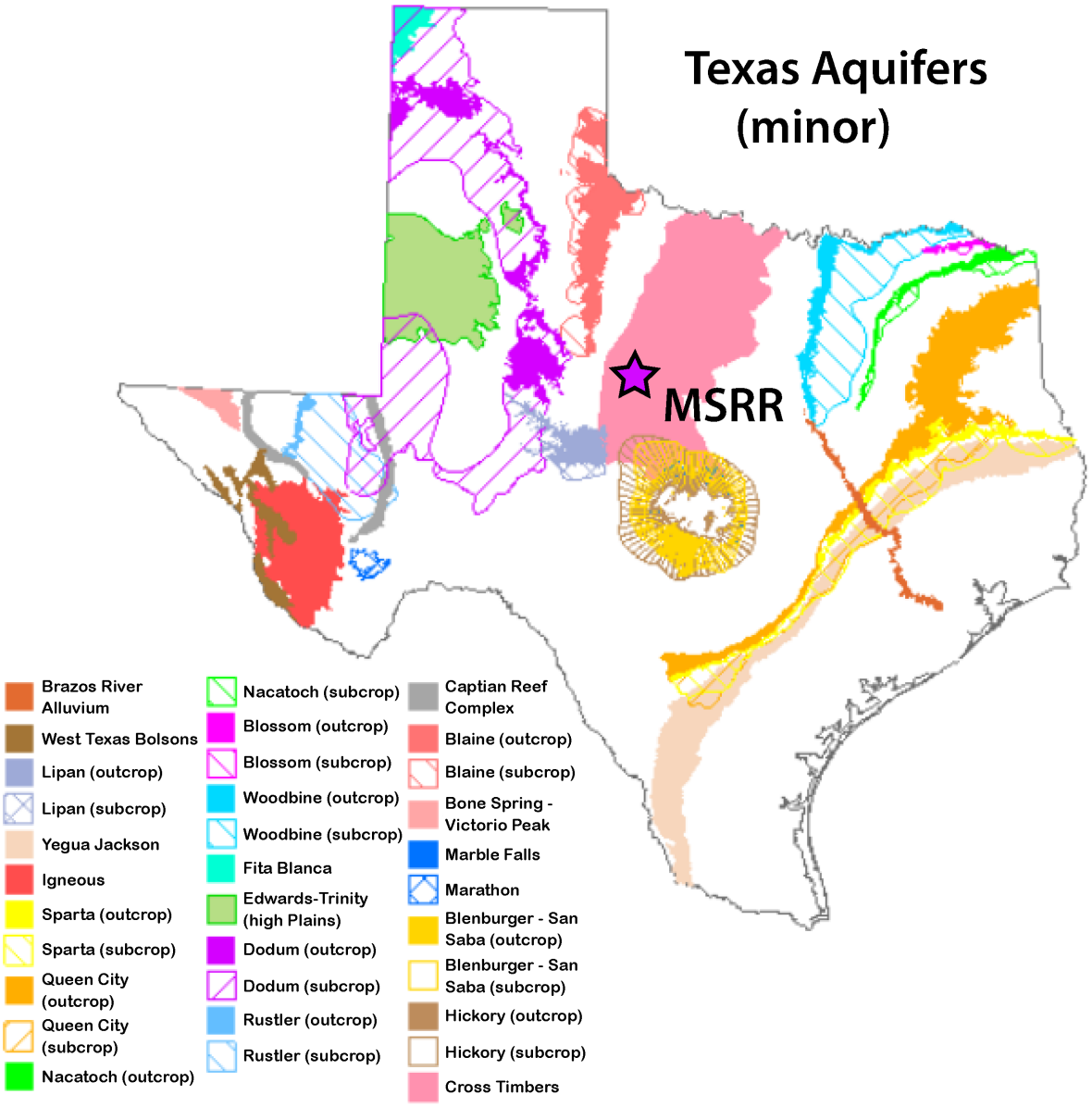
Graphical rendering provided by OpenStreet. Elevation data rendered by MERIT DEM, referenced to WGS84 and EGM96.

Figure 2.4-7 Major Aquifers of Texas



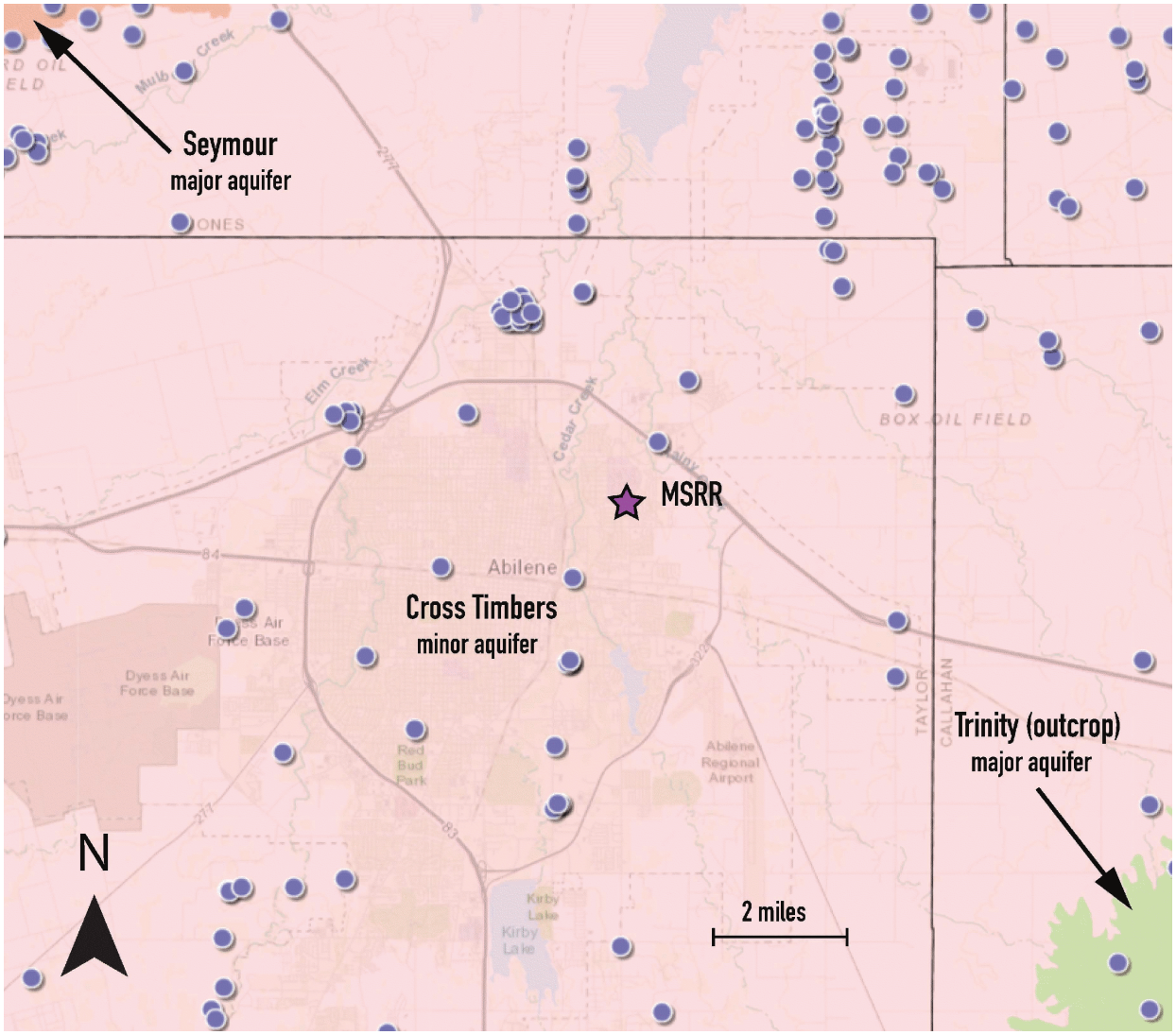
Adapted from Water Data Interactive, Texas Water Development Board [Reference 2.4-4]

Figure 2.4-8 Minor Aquifers of Texas



Adapted from Water Data Interactive, Texas Water Development Board [Reference 2.4-4].

Figure 2.4-9 Aquifers and Wells Near the Site



Adapted from Texas Water Development Board [Reference 2.4-4]

2.5 Geology, Seismology, and Geotechnical Engineering

This section describes the geologic, geophysical, seismic, and geotechnical characteristics of the MSRR site and the surrounding region. Site characteristics are developed that provide the basis for the required design input for SSCs. The seismic design basis reflects existing information from current seismic hazard publications and data from a detailed geophysical and geotechnical investigation at the site performed by Emprotec/Hibbs&Todd, Inc. (EHT). Local building codes were followed for the building that will house the reactor and for site investigation and preparation, in accordance with NUREG 1537 guidelines. EHT used assumptions and calculations consistent with local building codes and relevant standards. Local building codes are based on ASCE 7-10.

This section also includes a general regional characterization of the seismicity that can potentially impact the ground acceleration demands of the MSRR design. Although NUREG-1537 does not explicitly require specification of a radius of a region to account for seismicity, the regional description of the MSRR site investigation covers a radius close to 100 miles (161 km) around the site.

2.5.1 Regional Geology

Abilene is located on the Permian Basin near outcrops from the Cretaceous (Comanche) and Quaternary periods, as shown in Figure 2.5-1. The reactor site is in the northeastern corner of Taylor County, an area dominated by the Callahan Divide. With elevations varying from 1,898 ft (579 m) to 2,411 ft (735 m) above sea level, this short range of hills forms a barrier between the Brazos and Colorado River basins [Reference 2.5-1]. During the early Cretaceous period, shallow seas covered the area, depositing clay, sand, and bioclastic material. Today, the region is almost entirely covered in sedimentary rock, mainly limestone, sandstone, siltstone, and gypsum [Reference 2.5-2]. The only potable groundwater in Taylor County is in the far east, contained within thin sheets of limestone and shale. All other water-bearing rocks contain brine with high levels of dissolved solids [Reference 2.5-3].

Bands of the Clear Fork Group and Edwards Limestone stripe the area, formed from silt deposition [Reference 2.5-4]. Figure 2.5-2 shows the geology of Taylor County [Reference 2.5-3]. Figure 2.5-3 shows the Alluvium and Arroyo formation composition of the Abilene area [Reference 2.5-3]. Figure 2.5-4 shows that Abilene is situated outside the edge of areas defined by the USGS as prone to sinkholes [Reference 2.5-5].

2.5.2 Site Geology

The reactor site is located on a flat (approximately 1-percent grade), developed ACU property. The largest elevation changes in the Abilene area are no more than gently sloping hills with less than 100 ft (30 m) of elevation change across the entire city of Abilene as shown in Figure 2.5-5. Prior to the construction of the SERC, a geotechnical investigation was performed with sampling conducted in accordance with Shelby Type Steel Tube Sampling (ASTM D 1587M), Standard Penetration Tests and Split-Barrel Sampling (ASTM D 1586M), Texas Cone Penetration Tests (TEX-132-E, Last Reviewed 2014), and Practice for Description and Identification of Soils (Visual-Manual Procedures) (ASTM D 2488, 2017 Edition). The investigation

included nine two-inch (5 cm) borings were drilled to depths of five, 30, 40 and 60 ft (1.5, 9, 12, and 18 m) below existing ground surface elevation as reported in Figure 2.5-6 through Figure 2.5-16. At Test Boring Nos. 1, 3, and 4, groundwater was encountered between 9 and 11 ft (2.7 to 3.3 m), particularly within the alluvial soils and in contact with the Arroyo shales as shown in the log summary presented in Figure 2.5-17. The groundwater referenced in the boring log was determined to be seasonal perched groundwater trapped in the sandy clay soils above the weathered shales. The groundwater table is estimated to exist at depths greater than 60 ft (18.3 m) below current grades based on the borings. The water table likely fluctuates seasonally. These borings reveal that the soil beneath the site is composed of very stiff to hard sandy and silty clays from the surface down to 12 ft (3.7 m), below which are red-brown and tan highly-weathered to weathered shale to a depth of about 40 ft (12 m), which included sandstone seams in some areas below 25 ft (7.6 m). Below 40 ft (12 m) to the maximum bore depth of 60 ft (18.3 m) are red-brown and gray shale. The boring log summary is shown in Figure 2.5-17. There are no other significant structural units, including folds, faults, synclines, anticlines, domes, or basins, in the site area. Test borings B-8 and B-9, the 60 foot deep borings, are beneath the trench.

The geotechnical analysis by EHT and the analysis by ParkHill, the building engineering firm, both added a safety margin. ParkHill affirms that the boring data documented in the geotechnical report is verified and reliable for the areas investigated. EHT used N values correlated with Texas Department of Transportation values. The Structural design of the building ensures that the rigidity of slab combined with the rigidity of the piers is sufficient to avoid differential settlement. Soil design parameters for the lateral earth pressures were determined from the soil classification tests and from the soil properties recommended for the imported backfill at the MSRR structure. Pressures were checked with a cantilever wall assumption, at rest pressure was checked with a restrained wall condition, and no passive pressure was used to offset the forces of the retained backfill.

The straight shaft drilled piers that support the total load of the SERC are based on the existing subsurface profile, the recommended maximum net allowable end bearing and skin friction capacities given in Table 2.5-1. Additionally, Table 2.5-2 provides L-Pile information. These bearing capacities are based on published correlations for Standard Penetration Test (SPT) and Texas Cone Penetration (TCP) field test data. The recommended bearing capacities reflect a factor of safety of 3 relative to shear failure.

Table 2.5-1 Recommended Maximum Bearing and Skin Friction

Bearing Strata	Approximate Depth to Bearing Strata Below Current Grades (feet)(1)	Net Allowable Bearing Capacity (ksf)	Allowable Skin Friction (Ksf)
Shale with interbedded limestone seams	30-40'	20 ksf @ 30 feet increasing linearly to 40 ksf @ 40' bgs	3.0 ksf @ 30' increasing linearly to 5.0 ksf @ 40' bgs
Shale	Below 40'	40 ksf	5 ksf

(1) Actual embedment depths for design bearing capacity should be verified by a qualified geotechnical engineer based on subsurface conditions. Some deeper embedment depths may be required based upon subsurface variation between test boring locations.

Table 2.5-2 L-Pile Information

Depth	Soil Description	Effective Soil Unit Weight (pcf)	Avg. Undrained Shear Strength (c) (Ksf)	Soil Modulus (static) (k)(pci)	50% Axial Strain (E50) (inch/inch)
5-10'	sandy and silty clays	120	0.75	80	0.009
10-25'	Highly Weathered Shales	130	2.0	1,000	0.007
25-30'	Weathered Shales	130	5.7	1,500	0.005
30-40'	Shales with interbedded limestone seams	140	11.5	----	0.003
40-55'	Shale	135	13.5	----	0.002

2.5.3 Seismicity

The site is located on the North American Plate and is more than 620 miles (1,000 km) from any major plate boundary. Even a major earthquake on a major fault would have no effect on Abilene or the surrounding region [Reference 2.5-6]. Abilene is at no risk of earthquakes produced along major faults in North America.

Figure 2.5-18 shows the locations of earthquakes in and around the Permian basin over the last century. Since 1922, there have been 59 earthquakes of magnitude 3.0 or greater within 125 miles (200 km) of the MSRR site. Details for these earthquakes are provided in Table 2.5-3. The causes for earthquakes in Texas include small tremors induced by oil fields and wells as shown in Figure 2.5-19 [Reference 2.5-6]. Figure 2.5-20 shows the locations of injection and disposal wells in the Abilene vicinity. There is no evidence of any of these earthquakes having been felt or causing damage in Abilene.

2.5.4 Maximum Earthquake Potential

The largest earthquake in Texas since 1990 was a 4.8 magnitude in Fashing, Texas, which is more than 250 miles (400 km) from Abilene. Abilene is at low earthquake risk, with no earthquakes since 1931 [Reference 2.5-8]. The USGS database indicates that there can be a 0.17 percent threat of an earthquake within 30 miles (50 km) of Abilene, Texas, within the next 50 years.

2.5.5 Vibratory Ground Motion

Figure 2.5-21 shows the risk of earthquake vibratory ground motion across Texas via a model developed by the USGS. Abilene is an area that USGS predicts can see a peak ground acceleration (PGA) between 3 and 4 percent the force of gravity.

Seismic hazard data is taken from the ASCE Hazard Tool, an online nationally recognized website for providing relevant seismic hazards based on US Geological Hazards, governing code, site class, risk category and site location [Reference 2.5-9]. The data presented is based on the Risk-Targeted Maximum Considered Earthquake (MCE_R) adjusted for Site Class C Characteristics. The MCE_R is defined as the seismic intensity resulting in a 1% probability of collapse within a 50-year period.

The SERC was constructed to the governing edition of ASCE 7 (2010 Edition) as endorsed by the 2012 International Building Code, which was the legal code of record for Texas at the time of construction. ASCE 7-10 specifies seismic hazards from the 2008 USGS Seismic Hazard Maps. This seismic source model predicts a PGA of 0.03, PGA_M of 0.036g. The two-point mapped spectral parameters are SS 0.066g and S1 0.035g; when adjusted for site characteristics S_{MS} 0.08g and S_{M1} 0.06g. The corresponding design values, as determined by the Engineer of Record and indicated on design drawings, are S_{DS} 0.057g and S_{D1} 0.035g.

The latest edition of ASCE 7, the 2022 edition, uses the updated 2018 USGS seismic hazard maps to quantify seismic hazard. A principal change with ASCE 7-22 and the 2018 USGS seismic model is the introduction of a Multi Period Response Spectrum (MPRS) methodology in which the design spectrum is determined from 22 spectral data points (rather than 2). The 2018 update of the USGS Seismic Hazard Model includes significant changes to ground motion modeling, incorporates new findings on earthquake ground shaking, seismicity, and long-period amplification over deep sedimentary basins. The 2018 USGS model predicts a PGA_M of 0.036. In addition, primarily due to the introduction of the MPRS there is a shift toward higher frequency content in the design spectra. Beginning at approximately 3Hz and continuing to greater frequencies, the calculated ASCE 7-22 hazard is greater than the ASCE 7-10 hazard. At its peak, the ASCE 7-22 hazard exceeds the ASCE 7-10 hazard by approximately 70% at 10Hz. A comparison of the two developed hazards is shown in Figure 2.5-26.

Given the two different USGS source models and governing codes; the following conclusions are noted for the MSRR, sited at the ACU Campus:

- The predicted seismic hazard in the Abilene Texas Region is small when compared to the rest of the contiguous 48 states. The predicted seismic hazard from both models is low enough that the SERC, assigned as a Risk Category IV structure, qualifies for Seismic Design Category A (SDC-A) design methodologies. [REFERENCE ASCE 7-22 SECTION 11.6].
- The introduction of the MPRS design methodology, and the predicted higher frequency content will require dispensation when defining the SSE.

2.5.6 Surface Faulting

There is no evidence of surface faulting within 150 miles (240 km) of the site. The nearest underground faults are more than 40 miles (64 km) away near the borders of Throckmorton and Shackleford counties. They are oriented diagonally from northwest to southeast as shown in Figure 2.5-23. The nearest major fault is Meers Fault (Figure 2.5-24) located in Wichita Mountains Wildlife Refuge in Oklahoma, more than 175 miles (282 km) north-northeast of Abilene and dips to the southwest. It has a maximum estimated magnitude of 7.0 with a 1 percent probability of activity. At that distance, it is unlikely for Abilene to experience any effects [Reference 2.5-10].

Figure 2.5-25 shows the geographic locations of the fault lines in Texas. Neither Taylor nor Jones County contain any fault lines. The nearest one to Abilene is in Throckmorton County, about 75 miles (121 km) away [Reference 2.5-2].

2.5.7 Liquefaction Potential

The risk of liquefaction is very low in the Abilene area. The primary factors in soil liquefaction are soil saturation and seismic vibrations. There are no historical data in Abilene to indicate a local risk of seismic activity. Also, the borings included above show the soil is not typically saturated. The soil at the site is primarily silty clays and the most liquefiable soils are those with a higher content of sand size particles. Three factors required for liquefaction susceptibility have been reviewed for the MSRR site:

1. Loose granular soils - the soil classifications encountered at the MSRR site were found to be very stiff to hard comparative consistency sandy clay soils and weathered shale.
2. Saturation of soils by groundwater - the moisture contents of the soils tested do not indicate that the soils are saturated. Groundwater is not present in the sound shale.
3. Strong shaking - Abilene is in the lowest earthquake hazard area with up to 4%g (0.04g) peak ground acceleration.

ASCE 7-10 requirements were followed to evaluate possibility of liquefaction, including standard penetration tests performed by EHT. Additionally, following drilling for the foundation piers, there was observable water in only one hole, and that was minimal. Inspection of the holes for the piers confirmed that the walls had adequate strength to stand on their own; they did not collapse.

Liquefaction hazard evaluations generally deal with three issues: liquefaction susceptibility, initiation of liquefaction, and effects of liquefaction. Given the low liquefaction susceptibility, the initiation and effects of liquefaction were not assessed.

2.5.8 References

2.5-1 Texas State Historical Association, “Callahan Divide,” Texas State Historical Association Handbook of Texas. Accessed on Feb. 14, 2022. [Online]. Available: https://www.tshaonline.org/handbook/entries/callahan-divide

2.5-2 Texas Natural Resources Information System, Bureau of Economic Geology, U.S. Geological Survey. [Online]. Available: https://txpub.usgs.gov/txgeology/

2.5-3 Texas Department of Water Resources, “Occurrence, quantity, and quality of groundwater in Taylor County, Texas,” 1978. [Online]. Available: https://www.twdb.texas.gov/

2.5-4 Data basin, Conservation Biology Institute. [Online]. Available: https://databasin.org/maps/new/#datasets=83f6b3c68aaa4fdb8f2f22e7aeb7818f

2.5-5 U.S. Geological Survey Water Science School, “Sinkholes.” Accessed on May 11, 2022. [Online]. Available: https://www.usgs.gov/special-topics/water-science-school/science/sinkholes

2.5-6 Institute for Geophysics, Jackson School of Geosciences, “Texas earthquakes.” Accessed on Feb. 14, 2022. [Online]. Available: http://www-udc.ig.utexas.edu/external/TXEQ/

2.5-7 Railroad Commission of Texas, “Digital Map Information.” Publication Number OGA094, October 2021.

2.5-8 Institute for Geophysics, Jackson School of Geosciences, “Frequently asked questions.” Accessed on Feb. 14, 2022. [Online]. Available: http://www-udc.ig.utexas.edu/external/TXEQ/faq.html

2.5-9 U.S. Geological Survey, Unified Hazard Tool, Accessed on May 9, 2022. [Online]. Available: earthquake.usgs.gov/hazards/interactive]

2.5-10 2008 national seismic hazard maps - Source parameters, U.S. Geological Survey, Earthquake Hazards Program. [Online]. Available: https://earthquake.usgs.gov/cfusion/hazfaults_2008_search/view_fault.cfm?cfault_id=1031b

2.5-11 Texas Almanac, “Geology of Texas,” Texas State Historical Association. Accessed on Feb. 14, 2022. [Online]. Available: https://www.texasalmanac.com/articles/geology-of-texas

2.5-12 U.S. Geological Survey, “2014 Seismic Hazard Map - Texas.” Accessed on May 9, 2022. [Online]. Available: https://www.usgs.gov/media/images/2014-seismic-hazard-map-texas

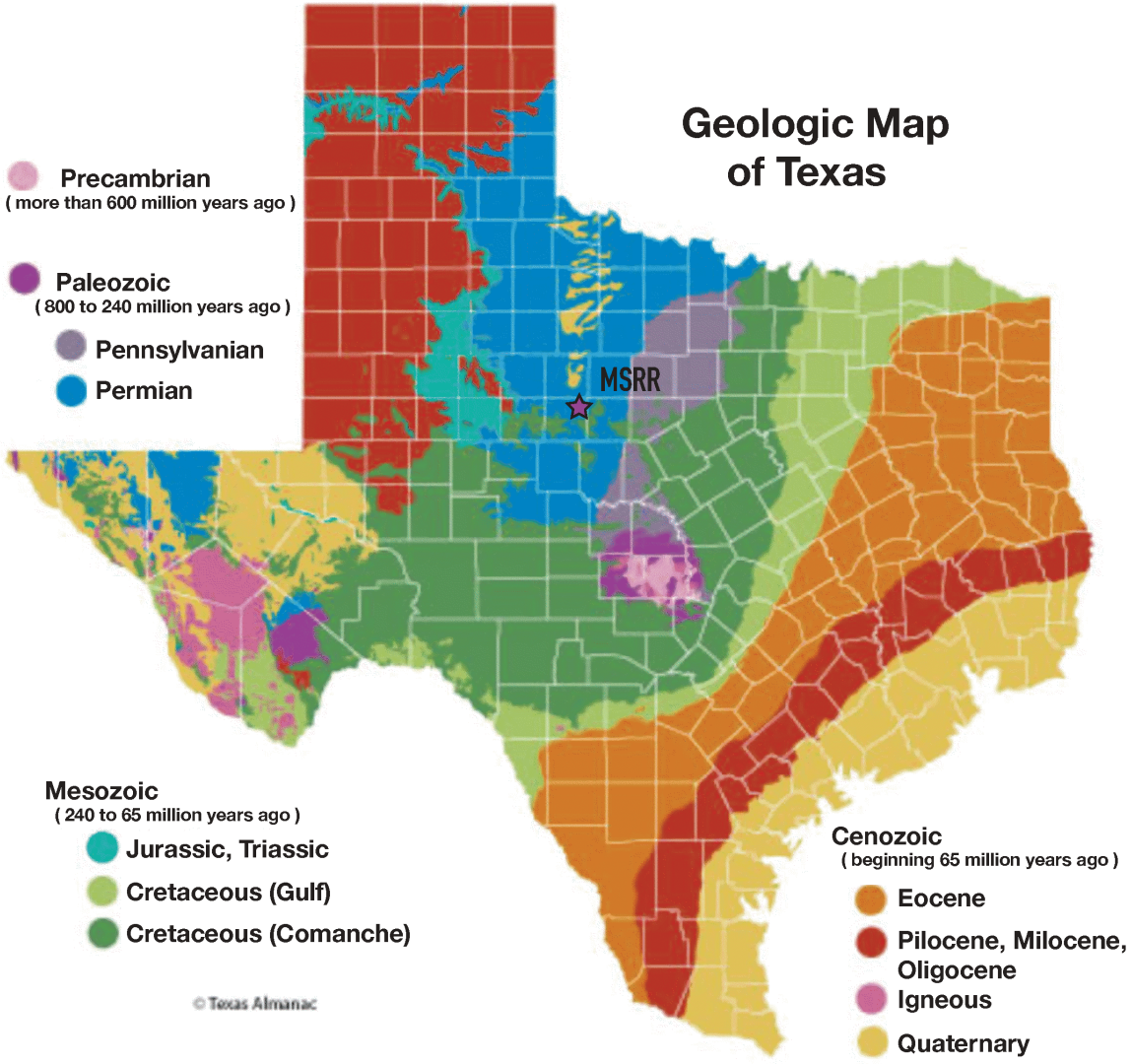
Table 2.5-3 Earthquakes of Magnitude 3 or Greater within 125 Miles (200 km) of the Molten Salt Research Reactor Site

ID	Magnitude	Distance (km)	Depth (km)	Date
tx2021zjsk	4.6	195.1km	8.4 km	December 28, 2021
usp0000v35	4.4	116.9km	10 km	June 16, 1978
usp000j7x7	4.3	107.9km	5 km	September 11, 2011
us6000d4mj	4	196.8km	5 km	December 31, 2020
us70005p48	4	119.2km	5 km	September 30, 2019
usp0000p6x	4	117.3km	5 km	June 7, 1977
us70005p8j	3.8	120.2km	5 km	October 1, 2019
usp000hwz2	3.8	115km	5 km	March 13, 2011
usb000ldzh	3.7	163.5km	5 km	December 9, 2013
usb000l7lw	3.7	161.5km	5 km	November 28, 2013
tx2021rwyl	3.6	95.12km	7.9 km	September 12, 2021
us6000d4la	3.6	196.8km	8 km	December 31, 2020
us10004d82	3.6	117.7km	2.7 km	January 10, 2016
tx2021deyk	3.5	196.5km	5.4 km	February 15, 2021
us7000cf5p	3.5	120.2km	5 km	November 15, 2020
us10004f5y	3.5	116.8km	2.1 km	January 17, 2016
usc000tc8i	3.5	116.9km	4 km	January 6, 2015
usp000jc09	3.5	120km	5 km	December 9, 2011
usp000j7yp	3.5	115.7km	7.9 km	September 12, 2011
usp0000rua	3.5	118.6km	5 km	November 28, 1977
tx2021mtdb	3.4	95.69km	6.1 km	July 1, 2021
us6000d4t2	3.4	196.2km	5 km	January 1, 2021
usp000hhr6	3.4	117km	5 km	August 8, 2010
tx2021stmj	3.3	187.7km	10.6 km	September 24, 2021
us6000d4l0	3.3	197.4km	5.8 km	December 31, 2020
us2000ixly	3.3	124.6km	1.2 km	December 26, 2018
usp000fxpp	3.3	116.3km	5 km	January 29, 2008
usp0002qkc	3.3	102.9km	5 km	January 30, 1986
usp0001r5h	3.3	121.5km	5 km	November 28, 1982
tx2022adli	3.2	195.2 km	8.2 km	January 2, 2022
usc000tazd	3.2	118.5km	5 km	December 31, 2014
usb000sc9j	3.2	114.8km	3 km	September 14, 2014
usp000jcba	3.2	113.7km	5 km	December 17, 2011
usp000j0x7	3.2	120.8km	5 km	May 2, 2011
usp00027a0	3.2	107.2 km	5 km	September 11, 1984
tx2021zmpp	3.1	95.36km	8 km	December 29, 2021
tx2021mdrc	3.1	180.2km	1.9 km	June 22, 2021
tx2021hpcf	3.1	129.1km	3.4 km	April 18, 2021
us6000d4pc	3.1	197.1km	5 km	January 1, 2021
us1000hi86	3.1	122.1km	4.9 km	October 28, 2018

Table 2.5-3 Earthquakes of Magnitude 3 or Greater within 125 Miles (200 km) of the Molten Salt Research Reactor Site (Continued)

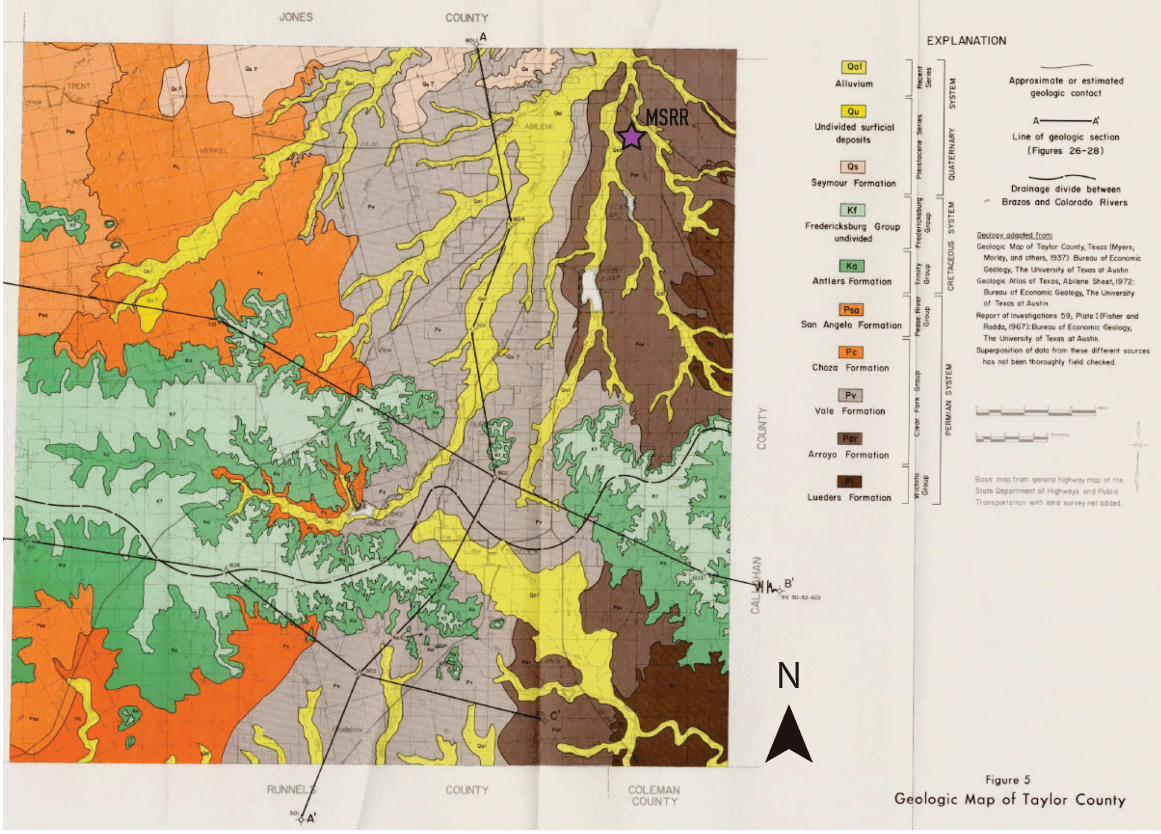
ID	Magnitude	Distance (km)	Depth (km)	Date
usb000qmf3	3.1	115.1km	4.5 km	May 14, 2014
usp000jbc2	3.1	118.8km	5 km	November 24, 2011
usp000hv6y	3.1	115.1 km	5 km	Dec 17, 2011
usp000hnku	3.1	118.1 km	5 km	October 26, 2010
usp000hmvz	3.1	121.5km	5 km	October 9, 2010
usp000h6nx	3.1	115.7 km	5 km	January 27, 2010
tx2022bwrn	3	180.2km	9.12 km	January 27, 2022
tx2021igfl	3	95.36km	6.3 km	April 28, 2021
tx2021ecyi	3	195.4km	6.4 km	February 28, 2021
us60008fga	3	122.9km	5 km	Mar 14, 2020
us2000i503	3	122.2km	5 km	October 31, 2018
usb000kv9n	3	197.5km	5 km	November 9, 2013
usa000hjmj	3	120.2 km	5.6 km	May 6, 2013
usp000hz5j	3	118.3km	5 km	April 2, 2011
usp000hysu	3	114.8km	5 km	March 28, 2011
usp000hxzc	3	114 km	5 km	March 19, 2011
usp000hx15	3	116.7km	5 km	March 14, 2011
usp000hwh0	3	120.3km	5 km	March 12, 2011
usp00027gs	3	104.5 km	5 km	September 19, 1984

Figure 2.5-1 Geologic Map of Texas



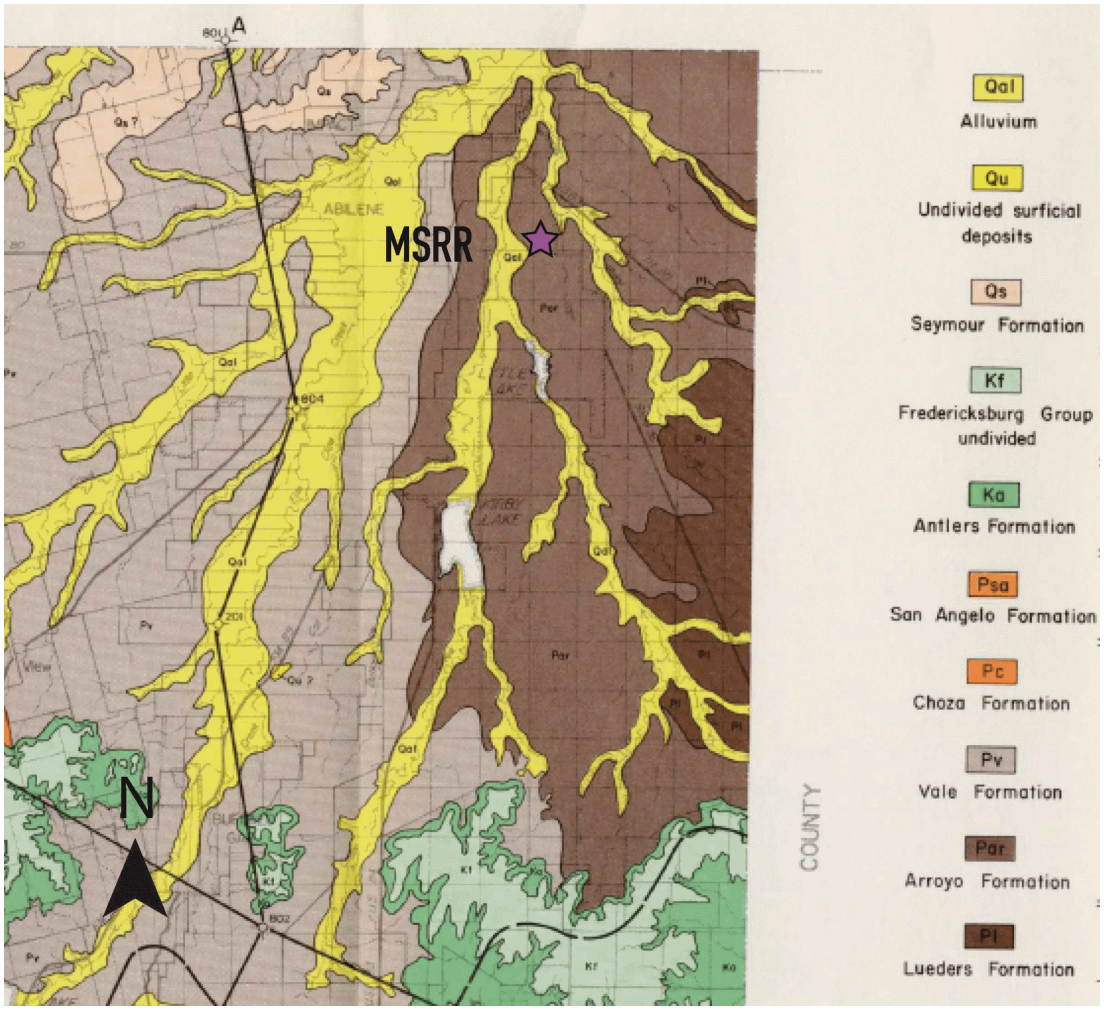
From Texas Almanac [Reference 2.5-11]

Figure 2.5-2 Geologic Map of Taylor County



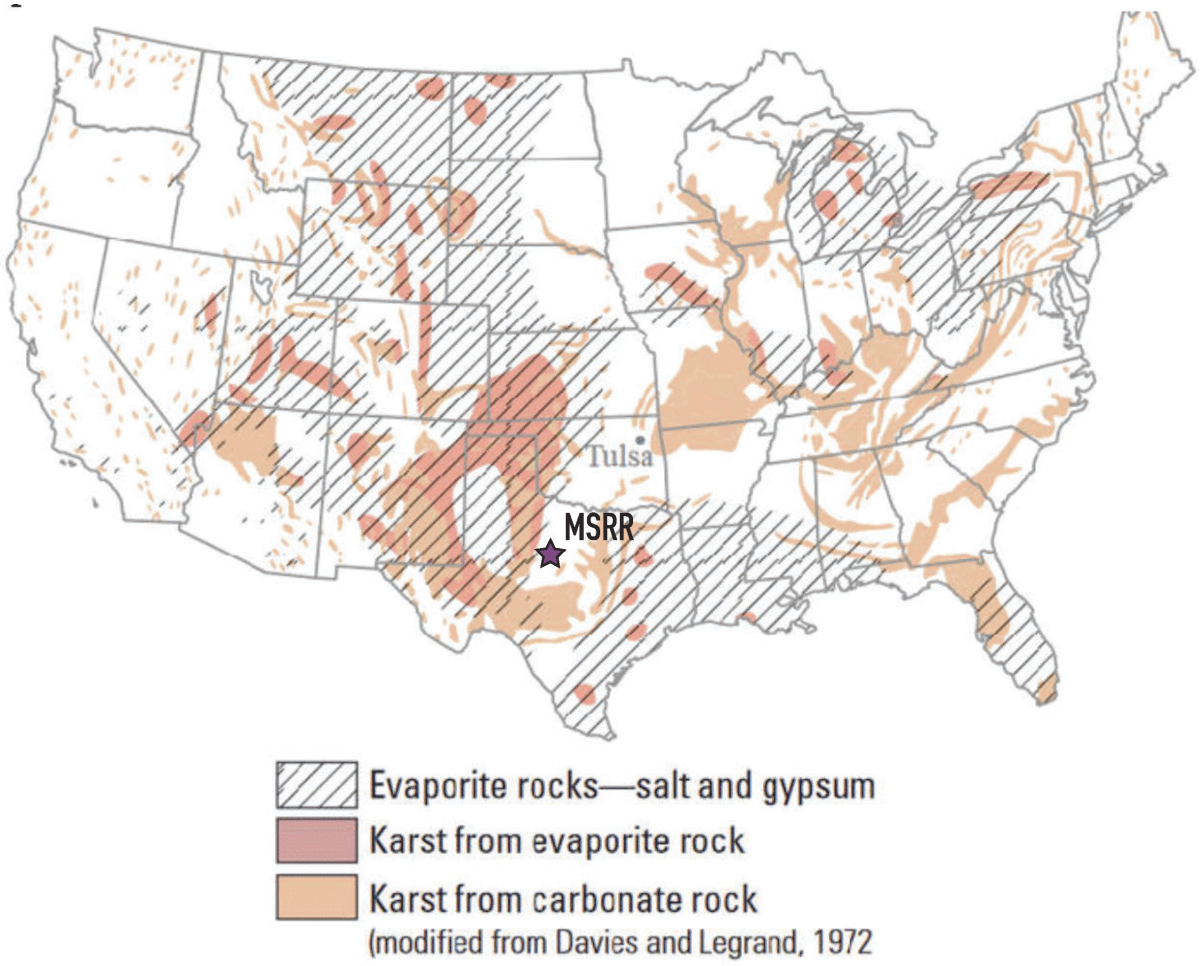
From Texas Department of Water Resources [Reference 2.5-3]

Figure 2.5-3 Geologic Map of Abilene Area



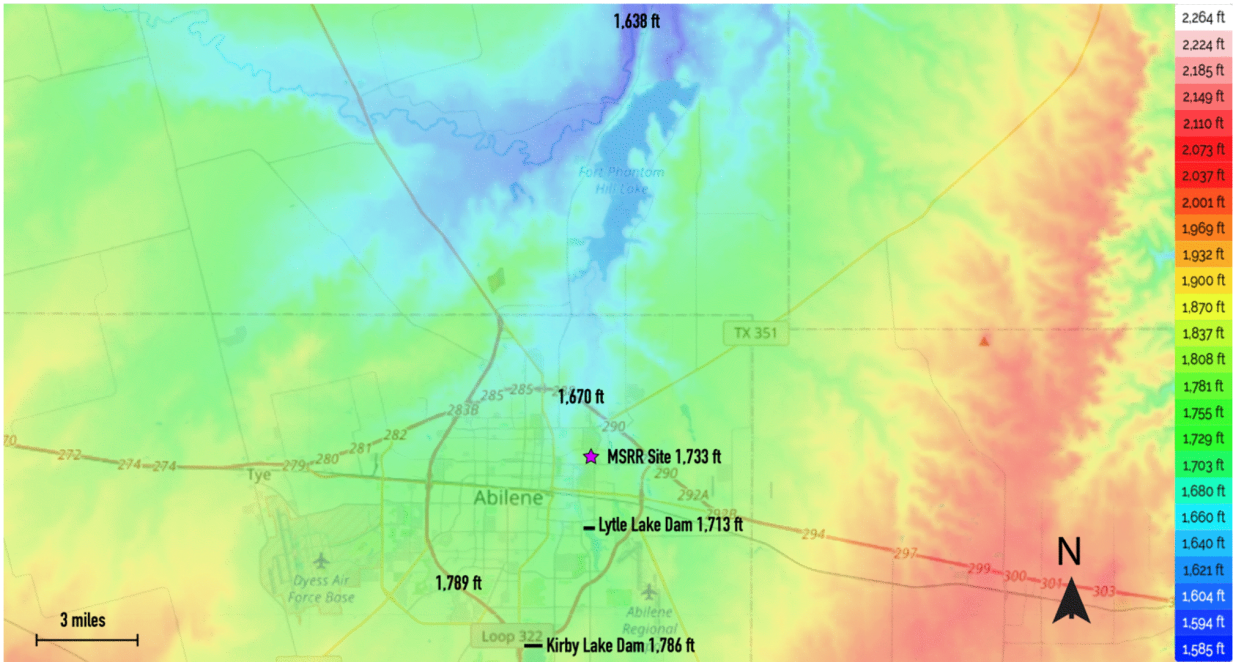
From Texas Department of Water Resources [Reference 2.5-3]

Figure 2.5-4 Areas Prone to Collapse Sinkholes



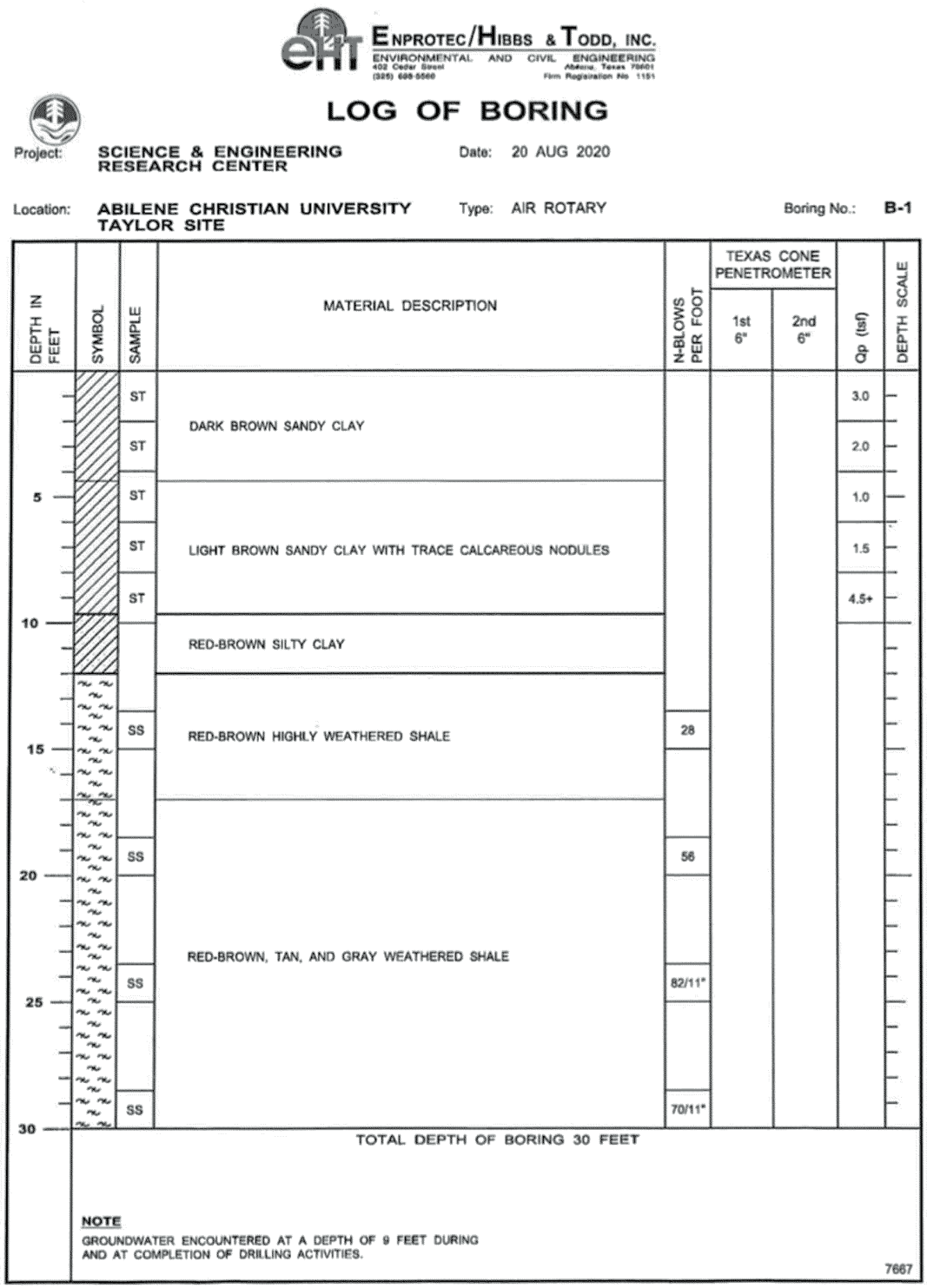
From U.S. Geological Survey [Reference 2.5-5]

Figure 2.5-5 Topography of the Abilene Area



OpenStreetMap provided topography rendering. Elevation data rendered by MERIT DEM, referenced to WGS84 and EGM96.

Figure 2.5-6 Boring Log 1



ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
402 Cedar Street Abilene, Texas 79601
(325) 698-5560 Firm Registration No. 1151

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER
Date: 20 AUG 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE
Type: AIR ROTARY
Boring No.: B-1

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
		ST	DARK BROWN SANDY CLAY				3.0	
		ST					2.0	
5		ST	LIGHT BROWN SANDY CLAY WITH TRACE CALCAREOUS NODULES				1.0	
		ST					1.5	
		ST					4.5+	
10			RED-BROWN SILTY CLAY					
		SS	RED-BROWN HIGHLY WEATHERED SHALE	28				
15								
		SS		56				
20			RED-BROWN, TAN, AND GRAY WEATHERED SHALE					
		SS		82/11"				
25								
		SS		70/11"				
30			TOTAL DEPTH OF BORING 30 FEET					

NOTE
GROUNDWATER ENCOUNTERED AT A DEPTH OF 9 FEET DURING AND AT COMPLETION OF DRILLING ACTIVITIES.

7667

Figure 2.5-7 Boring Log 2

ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
402 Cedar Street, Abilene, Texas 79601
(325) 698-5560 Firm Registration No. 1151

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER
Date: 20 AUG 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE
Type: AIR ROTARY
Boring No.: B-2

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
			INTERBEDDED LAYERS OF FINE CRUSHED LIMESTONE FOR TRACK AND BROWN SANDY CLAY					
		ST	BROWN SANDY CLAY				1.5	
5		ST					1.5	
			LIGHT BROWN SANDY CLAY WITH TRACE CALCAREOUS NODULES					
		ST	RED-BROWN SANDY CLAY WITH CALCAREOUS NODULES				4.5+	
		ST	RED-BROWN SILTY CLAY				4.5+	
10								
		SS		20				
15			RED-BROWN HIGHLY WEATHERED SHALE					
		SS		49				
20								
		SS		50/5"*				
25			RED-BROWN, TAN, AND GRAY WEATHERED SHALE -INTERBEDDED SANDSTONE SEAMS BELOW 25'					
		SS		50/6"				
30			TOTAL DEPTH OF BORING 30 FEET					

NOTE
NO GROUNDWATER WAS PRESENT DURING OR AT COMPLETION OF DRILLING ACTIVITIES.
* = WITH 6" SEAT

7667

Figure 2.5-8 Boring Log 3

ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
402 Cedar Street, Abilene, Texas, 79601
(325) 698-5560, Firm Registration No. 1151

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER
Date: 20 AUG 2020

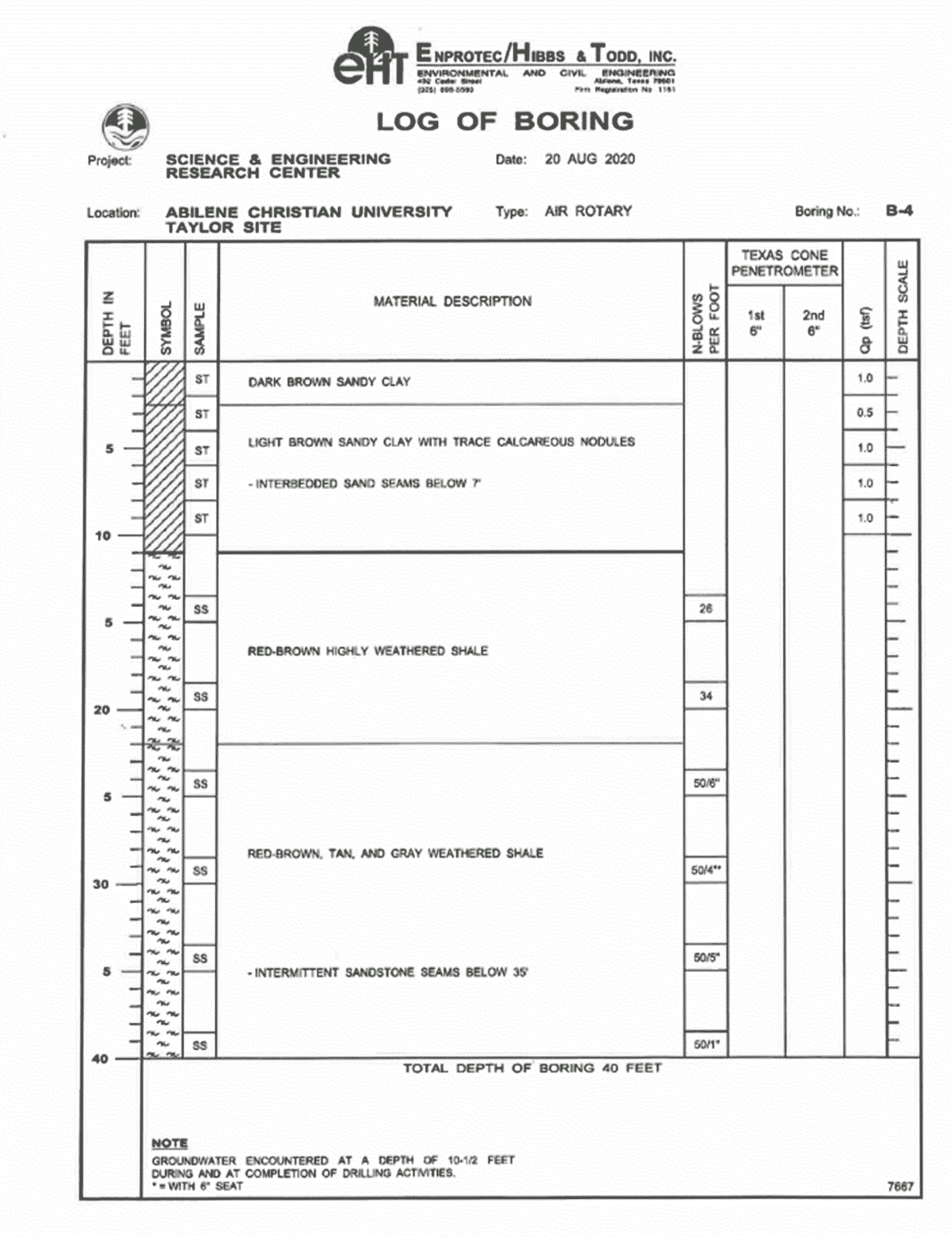
Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE
Type: AIR ROTARY
Boring No.: B-3

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
		ST	DARK BROWN SANDY CLAY				2.0	
		ST					0.5	
5		ST	LIGHT BROWN SANDY CLAY WITH TRACE CALCAREOUS NODULES				2.0	
		ST	- 4" SAND SEAM @ 6-1/2'				4.0	
		ST	- 4" SAND SEAM @ 8'				3.5	
10								
		SS	RED-BROWN HIGHLY WEATHERED SHALE	27				
15								
		SS		50/5"*				
20								
		SS	RED-BROWN, TAN, AND GRAY WEATHERED SHALE	50/5"				
25								
		SS		50/6"				
30			TOTAL DEPTH OF BORING 30 FEET					

NOTE
GROUNDWATER ENCOUNTERED AT A DEPTH OF 11 FEET DURING AND AT COMPLETION OF DRILLING ACTIVITIES.
* = WITH 6" SEAT

7667

Figure 2.5-9 Boring Log 4



ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
Abilene, Texas 79601
Firm Registration No. 1151

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER
Date: 20 AUG 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE
Type: AIR ROTARY
Boring No.: B-4

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
0		ST	DARK BROWN SANDY CLAY				1.0	
		ST					0.5	
5		ST	LIGHT BROWN SANDY CLAY WITH TRACE CALCAREOUS NODULES				1.0	
		ST	- INTERBEDDED SAND SEAMS BELOW 7'				1.0	
10		ST					1.0	
5		SS	RED-BROWN HIGHLY WEATHERED SHALE	26				
20		SS		34				
5		SS		50/6"				
30		SS	RED-BROWN, TAN, AND GRAY WEATHERED SHALE	50/4"				
5		SS	- INTERMITTENT SANDSTONE SEAMS BELOW 35'	50/5"				
40		SS		50/1"				

TOTAL DEPTH OF BORING 40 FEET

NOTE
GROUNDWATER ENCOUNTERED AT A DEPTH OF 10-1/2 FEET DURING AND AT COMPLETION OF DRILLING ACTIVITIES.
* = WITH 6" SEAT

7667

Figure 2.5-10 Boring Log 5

ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
402 Cedar Street, Abilene, Texas 79601
(325) 698-5560 Firm Registration No. 1151

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER
Date: 20 AUG 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE
Type: AIR ROTARY
Boring No.: B-5

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
		ST	BROWN SANDY CLAY				0.5	
		ST					1.0	
5		ST	LIGHT BROWN SANDY CLAY WITH TRACE CALCAREOUS NODULES				4.5+	
		ST	RED-BROWN SANDY CLAY WITH CALCAREOUS NODULES - 3" SAND SEAM				4.0	
		ST					4.5+	
10			RED-BROWN SILTY CLAY					
		SS		20				
5			RED-BROWN HIGHLY WEATHERED SHALE					
		SS		34				
20								
		SS		51				
5								
		SS		41				
30			RED-BROWN, TAN, AND GRAY WEATHERED SHALE					
		SS		50/4"				
5			- INTERMITTENT SANDSTONE SEAMS BELOW 36'					
		SS		50/0"				
40			TOTAL DEPTH OF BORING 40 FEET					

NOTE
NO GROUNDWATER WAS PRESENT DURING OR AT COMPLETION OF DRILLING ACTIVITIES.

7667

Figure 2.5-11 Boring Log 6

ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
402 Cedar Street, Abilene, Texas 79601
(325) 698-5560 Firm Registration No. 1181

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER

Date: 20 AUG 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE

Type: AIR ROTARY

Boring No.: B-6

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
		ST	DARK BROWN SANDY CLAY				4.5+	
		ST					3.0	
		ST	LIGHT BROWN SANDY CLAY WITH TRACE CALCAREOUS NODULES				3.5	
5			TOTAL DEPTH OF BORING 5 FEET					

NOTE

NO GROUNDWATER WAS PRESENT DURING OR AT COMPLETION OF DRILLING ACTIVITIES.

7667

Figure 2.5-12 Boring Log 7

ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
402 Cedar Street, Abilene, Texas 79601
(325) 698-5560 Firm Registration No. 1151

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER
Date: 20 AUG 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE
Type: AIR ROTARY
Boring No.: B-7

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
		ST	DARK BROWN SANDY CLAY				4.5+	
		ST					4.5+	
		ST	LIGHT BROWN SANDY CLAY WITH TRACE CALCAREOUS NODULES				4.5+	
5			TOTAL DEPTH OF BORING 5 FEET					

NOTE
NO GROUNDWATER WAS PRESENT DURING OR AT COMPLETION OF DRILLING ACTIVITIES.

7667

Figure 2.5-13 Boring Log 8-1

ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
402 Cedar Street Abilene, Texas 79601
(325) 698-5560 Firm Registration No. 1151

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER
Date: 3 DEC 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE
Type: AIR ROTARY
Boring No.: B-8

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
0			DARK BROWN SANDY CLAY					
5			LIGHT BROWN SANDY CLAY WITH TRACE CALCAREOUS NODULES					
10								
15			RED-BROWN HIGHLY WEATHERED SHALE (SILTY CLAY)					
20								
25			RED-BROWN, TAN, AND GRAY WEATHERED SHALE					
30			CONTINUED TO NEXT SHEET					

7667

Figure 2.5-14 Boring Log 8-2

ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER
Date: 3 DEC 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE
Type: AIR ROTARY
Boring No.: B-8 (CONT.)

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
			RED-BROWN, TAN, AND GRAY WEATHERED SHALE					
35		THD	- INTERMITTENT SANDSTONE SEAMS BETWEEN 34 AND 37 FEET		38	50=3/4"		
40		THD			50=2"	50=1"		
45		THD			50=1"	50=1/4"		
50		THD	RED-BROWN AND GRAY SHALE		50=1"	50=0"		
55		THD			50=1/2"	50=0"		
60		THD			50=1/2"	50=1"		

TOTAL DEPTH OF BORING 60 FEET

NOTE
TRACE GROUNDWATER ENCOUNTERED AT A DEPTH OF 9-1/2 FEET DURING DRILLING ACTIVITIES.

7667

Figure 2.5-15 Boring Log 9-1

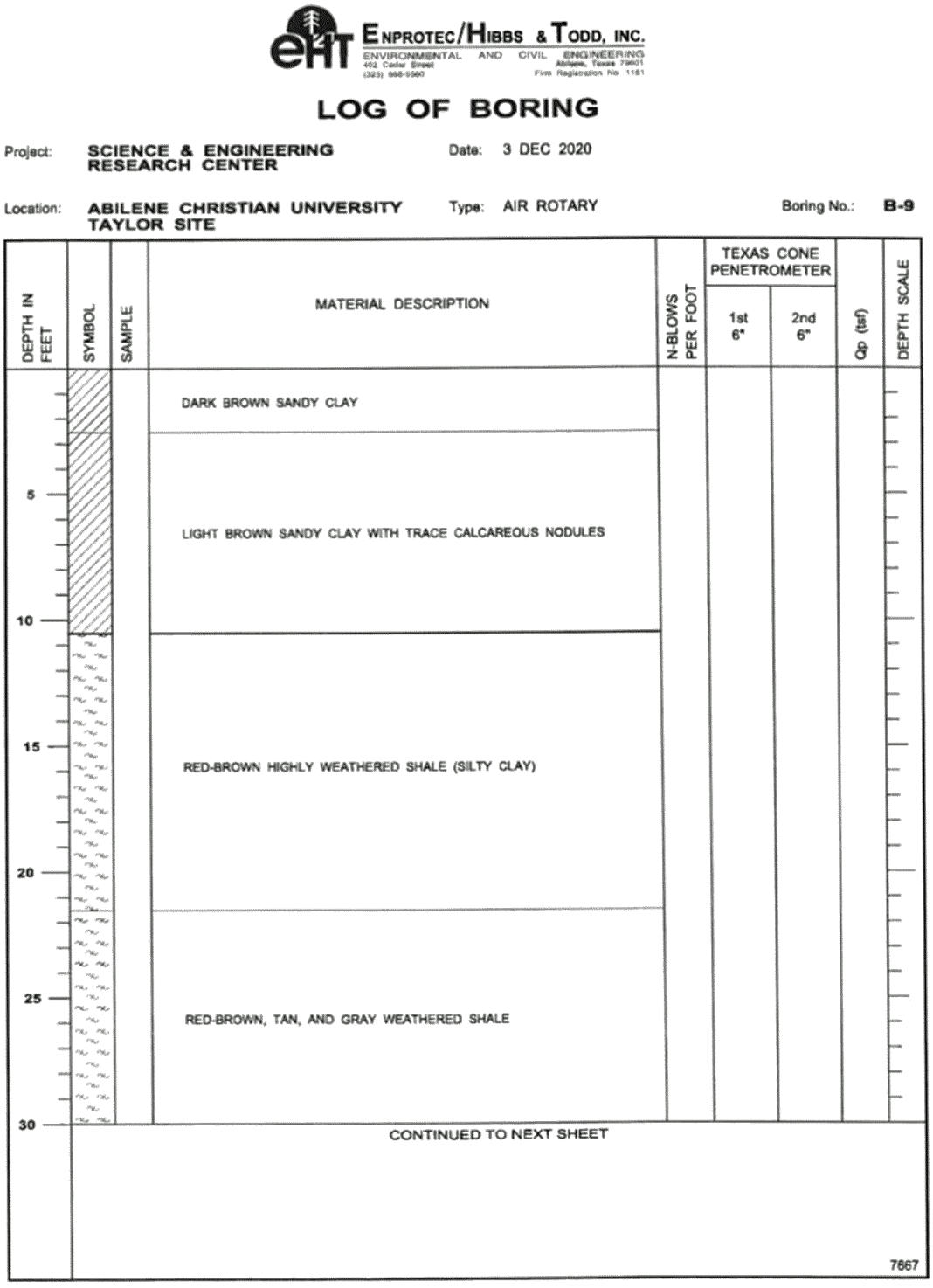


Figure 2.5-16 Boring Log 9-2

ENPROTEC/HIBBS & TODD, INC.
ENVIRONMENTAL AND CIVIL ENGINEERING
402 Cedar Street, Abilene, Texas 79601, (325) 698-5560, Firm Registration No. 1151

LOG OF BORING

Project: SCIENCE & ENGINEERING RESEARCH CENTER Date: 3 DEC 2020

Location: ABILENE CHRISTIAN UNIVERSITY TAYLOR SITE Type: AIR ROTARY Boring No.: B-9 (CONT.)

DEPTH IN FEET	SYMBOL	SAMPLE	MATERIAL DESCRIPTION	N-BLOWS PER FOOT	TEXAS CONE PENETROMETER 1st 6"	TEXAS CONE PENETROMETER 2nd 6"	Qp (tsf)	DEPTH SCALE
35		CB	RED-BROWN, TAN, AND GRAY WEATHERED SHALE - INTERMITTENT SANDSTONE SEAMS BETWEEN 31 AND 39 FEET CORE RECOVERED: 42% RQD: 18%					
40		THD			50=4"	50=2"		
45		THD			50=1/4"	50=0"		
50		THD	RED-BROWN AND GRAY SHALE		50=1"	50=1"		
55		THD			50=1/4"	50=1/4"		
60		THD			50=1-1/2"	50=1-1/2"		

TOTAL DEPTH OF BORING 60 FEET

NOTE
TRACE GROUNDWATER ENCOUNTERED AT A DEPTH OF 9 FEET DURING DRILLING ACTIVITIES.

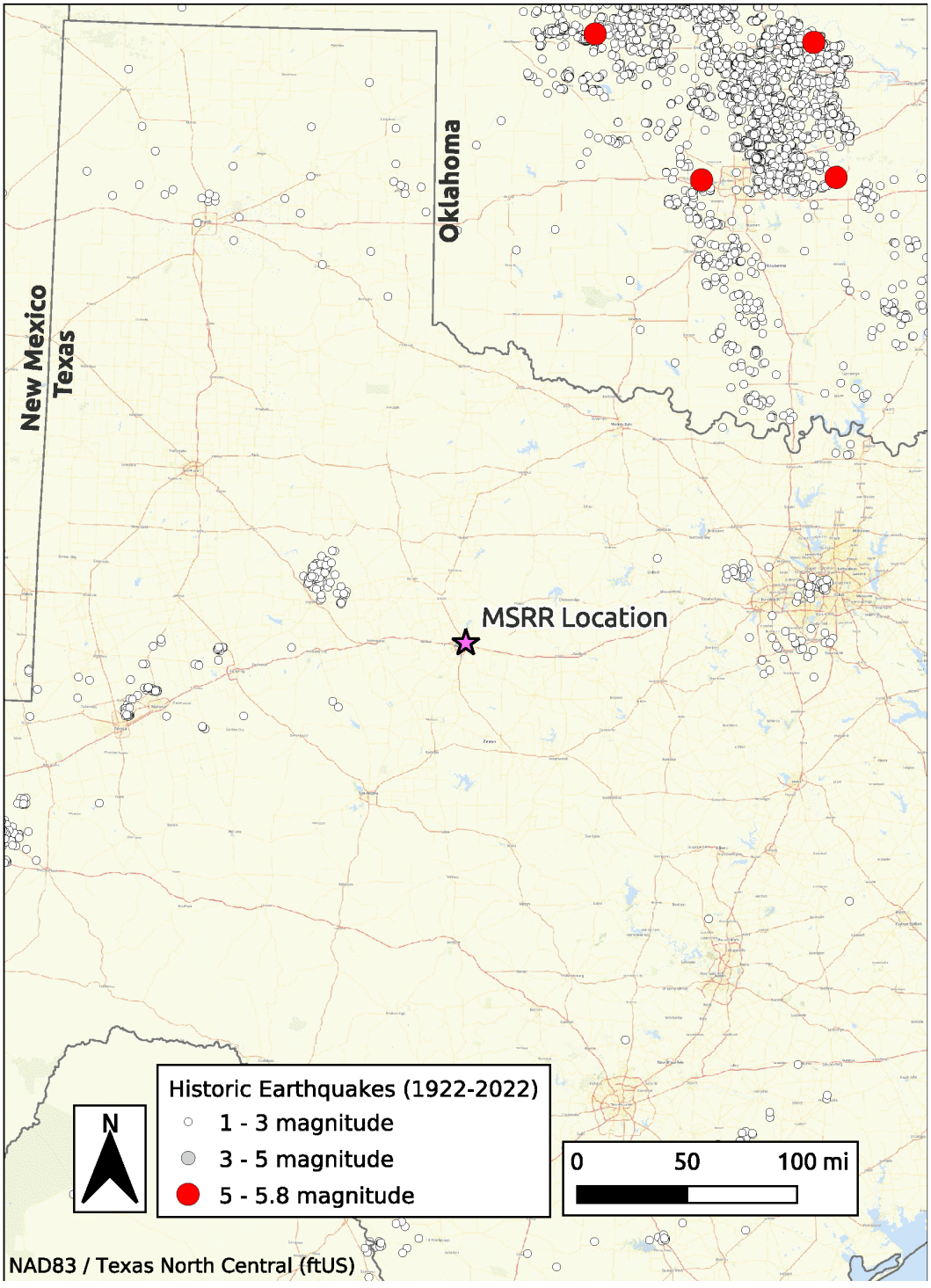
7667

Figure 2.5-17 Boring Log Summary

SCIENCE & ENGINEERING RESEARCH CENTER
ABILENE CHRISTIAN UNIVERSITY
ABILENE, TEXAS
SUMMARY OF CLASSIFICATION TESTS

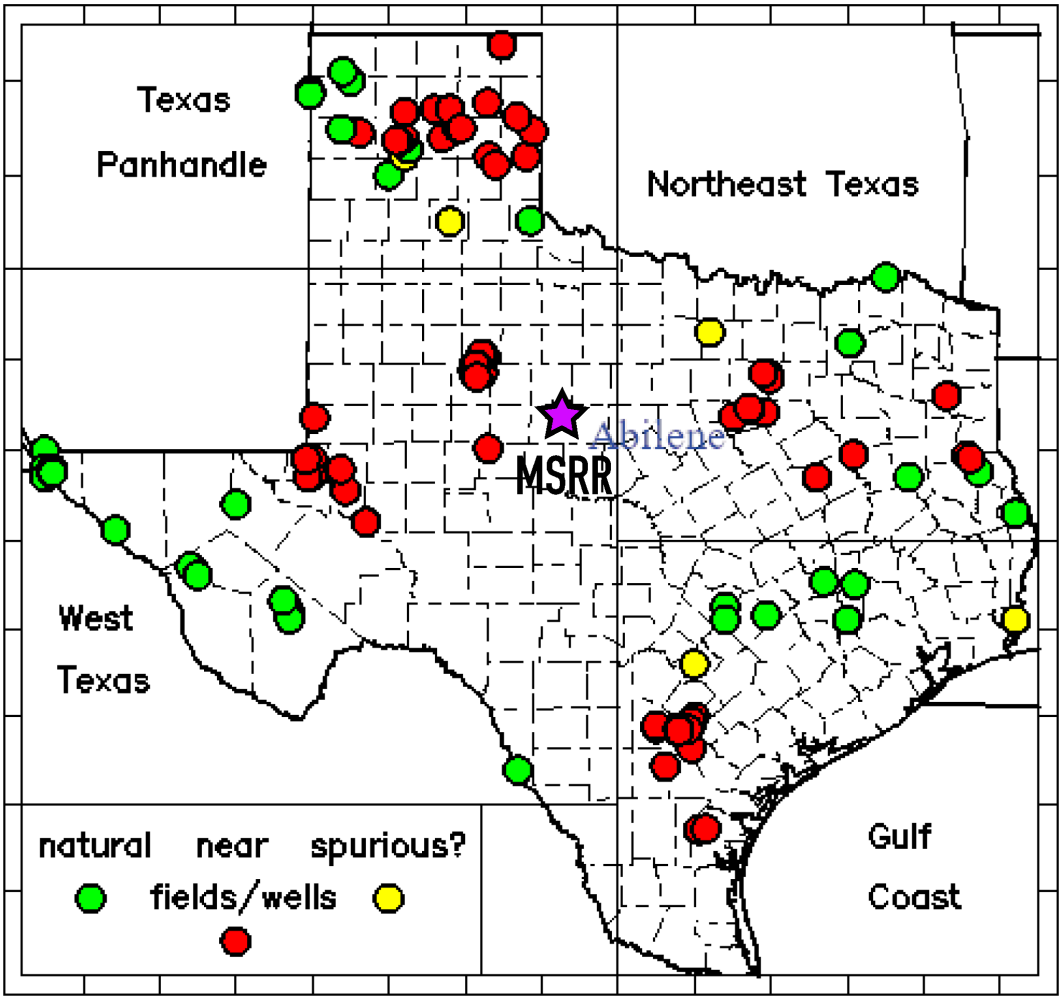
Boring No.	Depth (ft)	Liquid Limit %	Plasticity Index	% Passing #200 Mesh Sieve	Water Content %	USCS	Description
B-1	0-2'	37	21	---	17.8	CL	Dark Brown Sandy Clay
B-1	6-8'	34	19	79	15.9	CL	Light Brown Sandy Clay with trace calcareous nodules
B-2	2-4'	34	19	---	16.3	CL	Brown Sandy Clay
B-3	2-4'	32	18	63	20.0	CL	Light Brown Sandy Clay with trace calcareous nodules
B-3	8-10'	39	23	54	13.8	CL	Light Brown Sandy Clay with trace calcareous nodules
B-4	4-6'	36	21	81	16.3	CL	Light Brown Sandy Clay with trace calcareous nodules
B-4	13½-15'	42	25	87	12.6	---	Red-Brown Highly Weathered Shale
B-5	0-2'	42	25	52	20.8	CL	Brown Sandy Clay
B-5	18½-20'	47	29	99	13.6	---	Red-Brown Highly Weathered Shale
B-6	2-4'	26	13	97	16.7	CL	Dark Brown Sandy Clay
B-7	0-2'	41	25	51	10.3	CL	Dark Brown Sandy Clay

Figure 2.5-18 Seismic Activity Near the Molten Salt Research Reactor Site



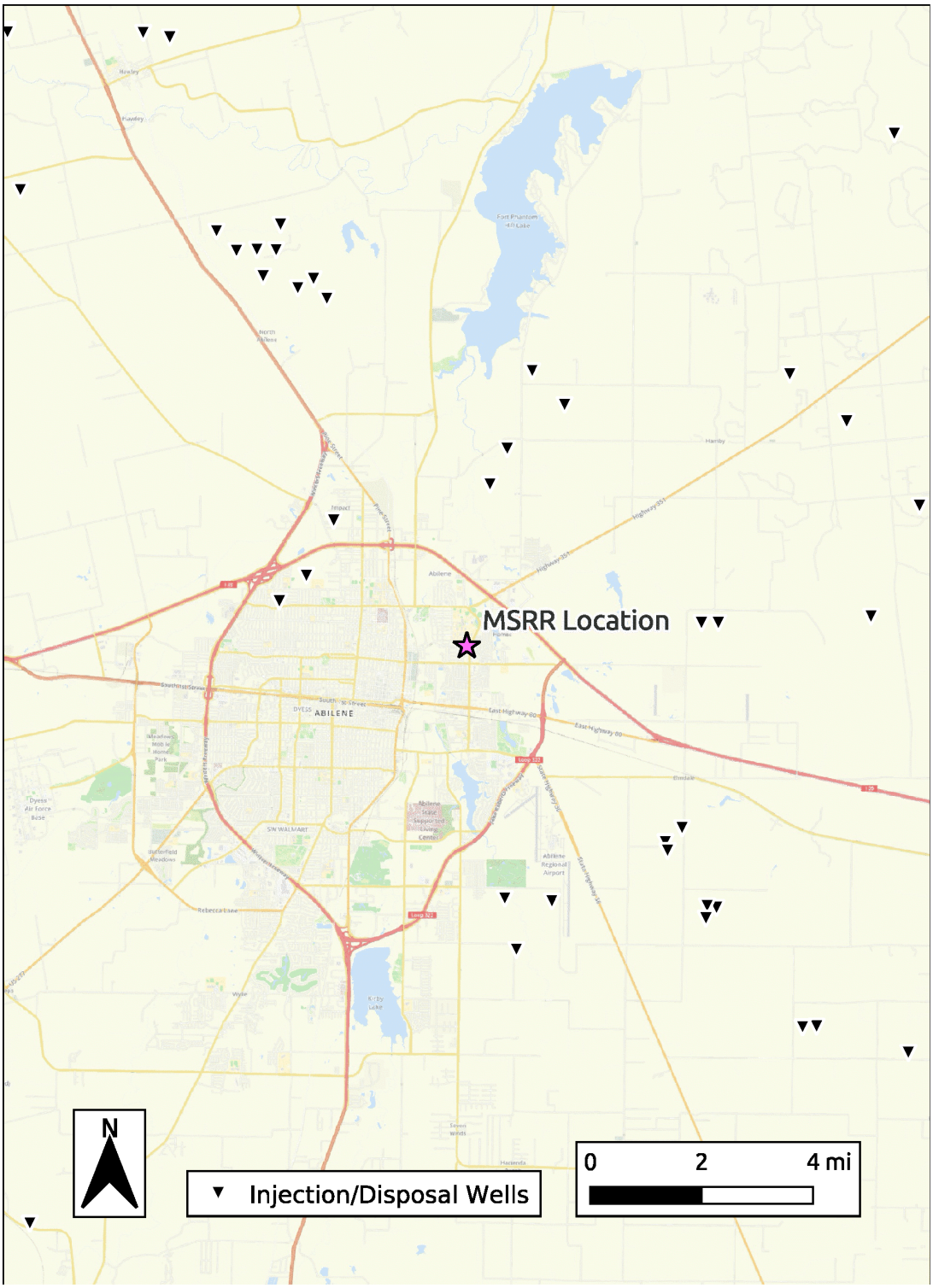
From Texas Almanac [Reference 2.5-11]

Figure 2.5-19 Causes and Locations of Earthquakes in Texas



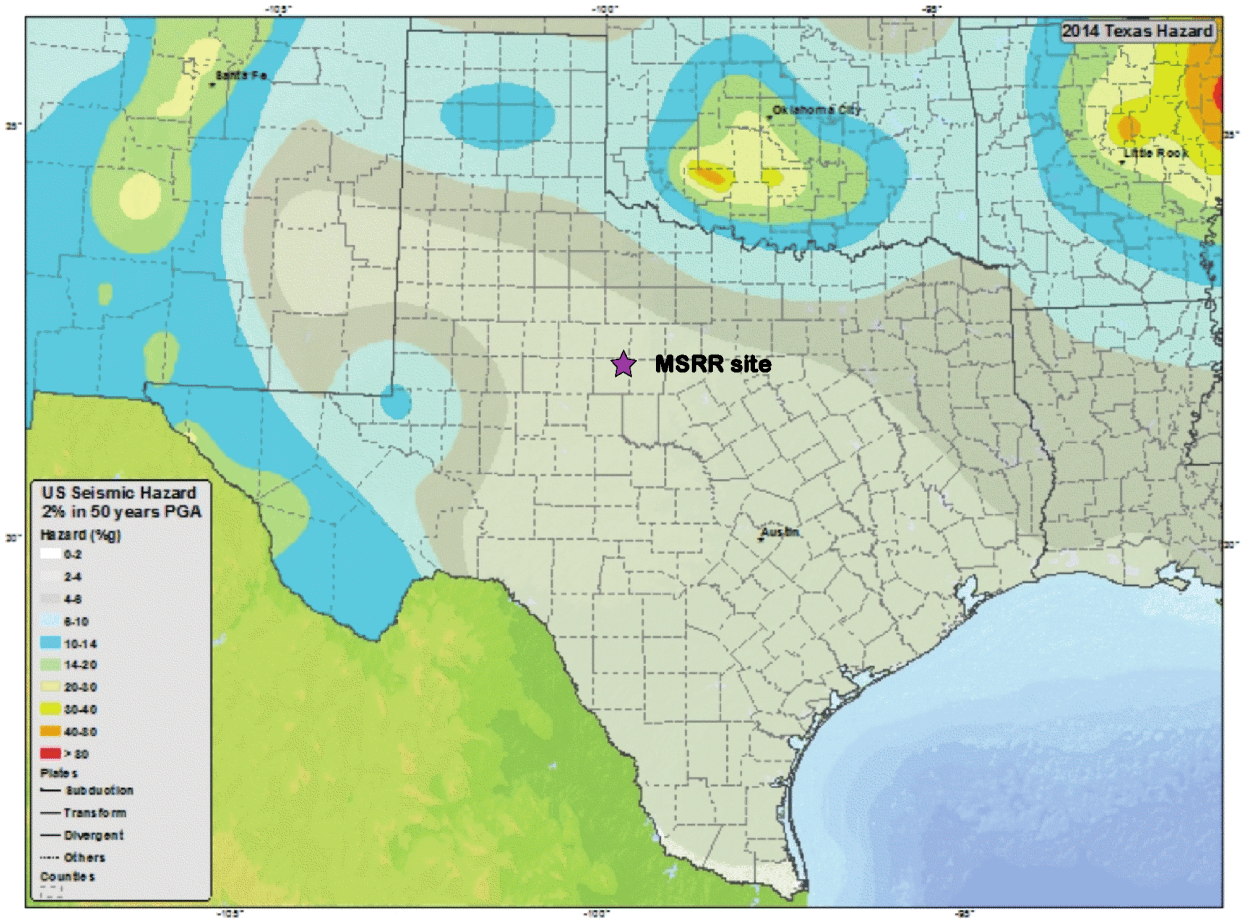
From Institute for Geophysics [Reference 2.5-6]

Figure 2.5-20 Injection and Disposal Wells Near the Molten Salt Research Reactor Site



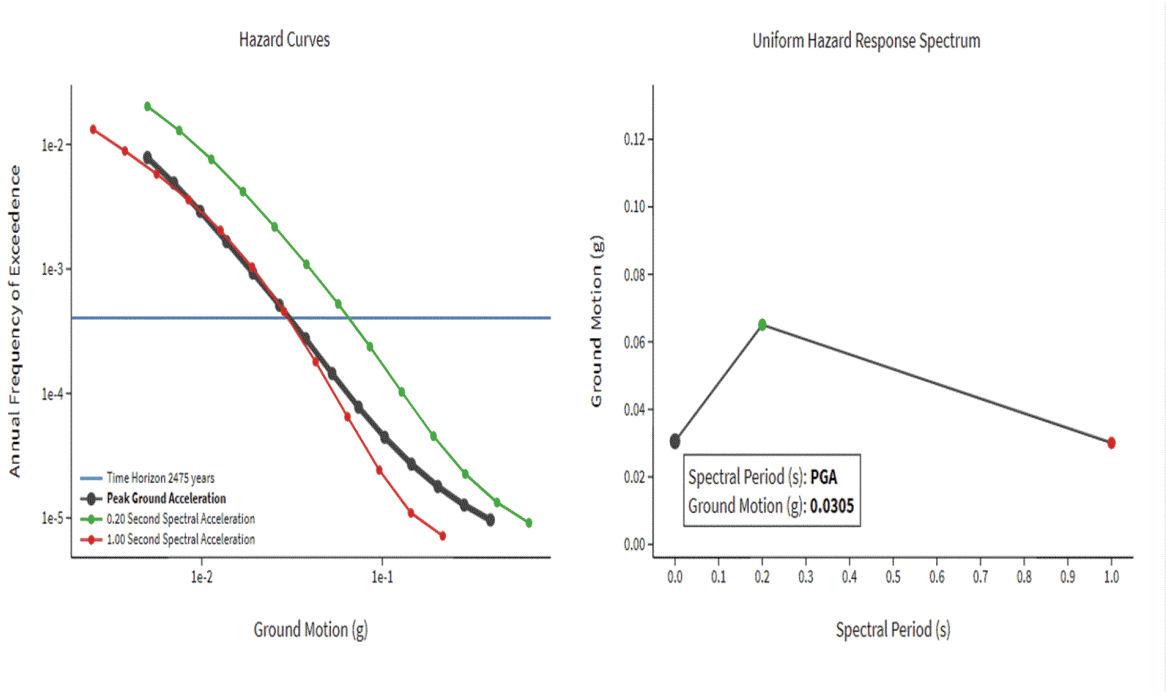
Railroad Commission of Texas [Reference 2.5-7]

Figure 2.5-21 2014 Seismic Hazard Map of Texas



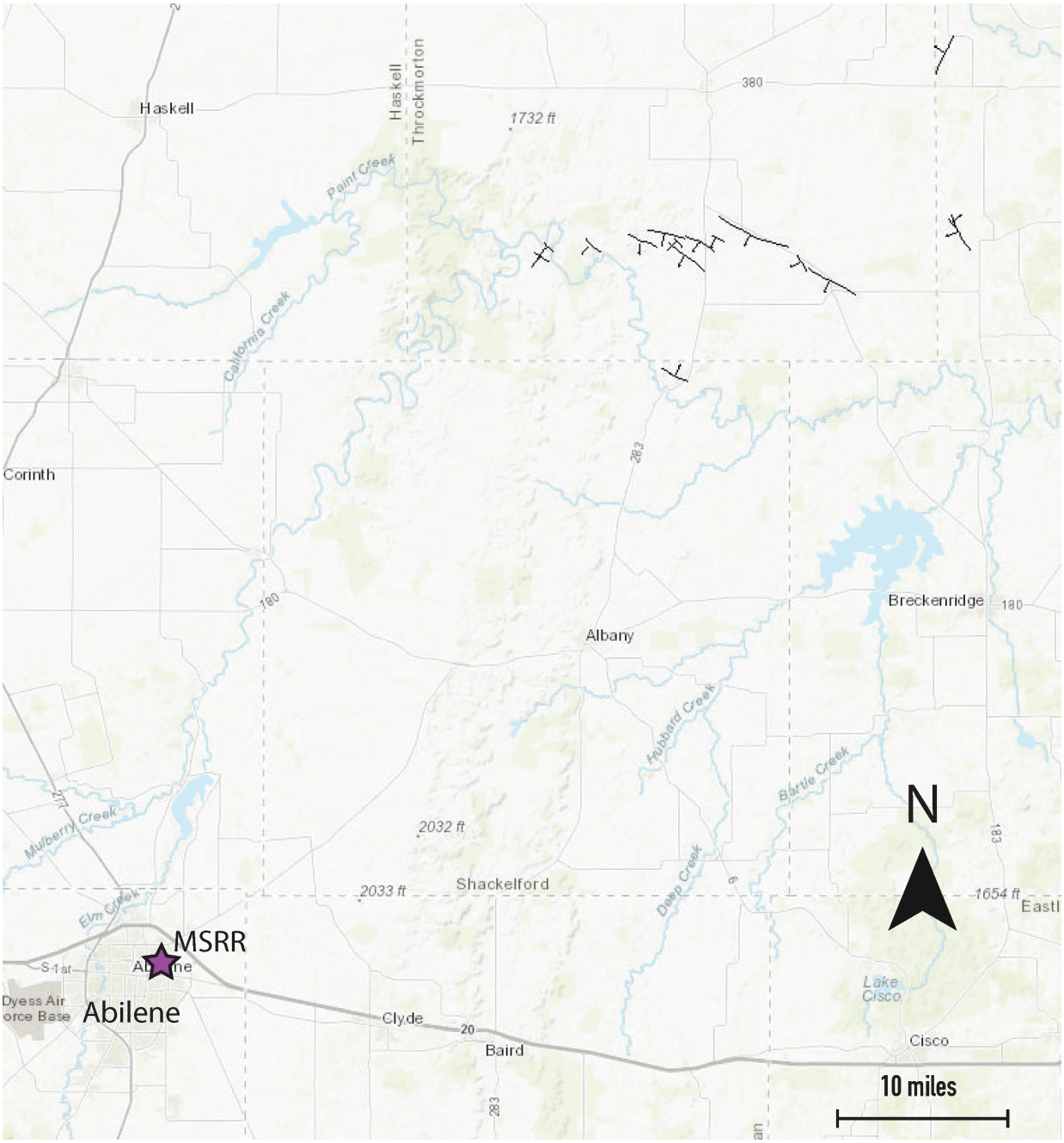
From U.S. Geological Survey [Reference 2.5-12]

Figure 2.5-22 Seismic Hazard Curves and Uniform Hazard Response Spectrum



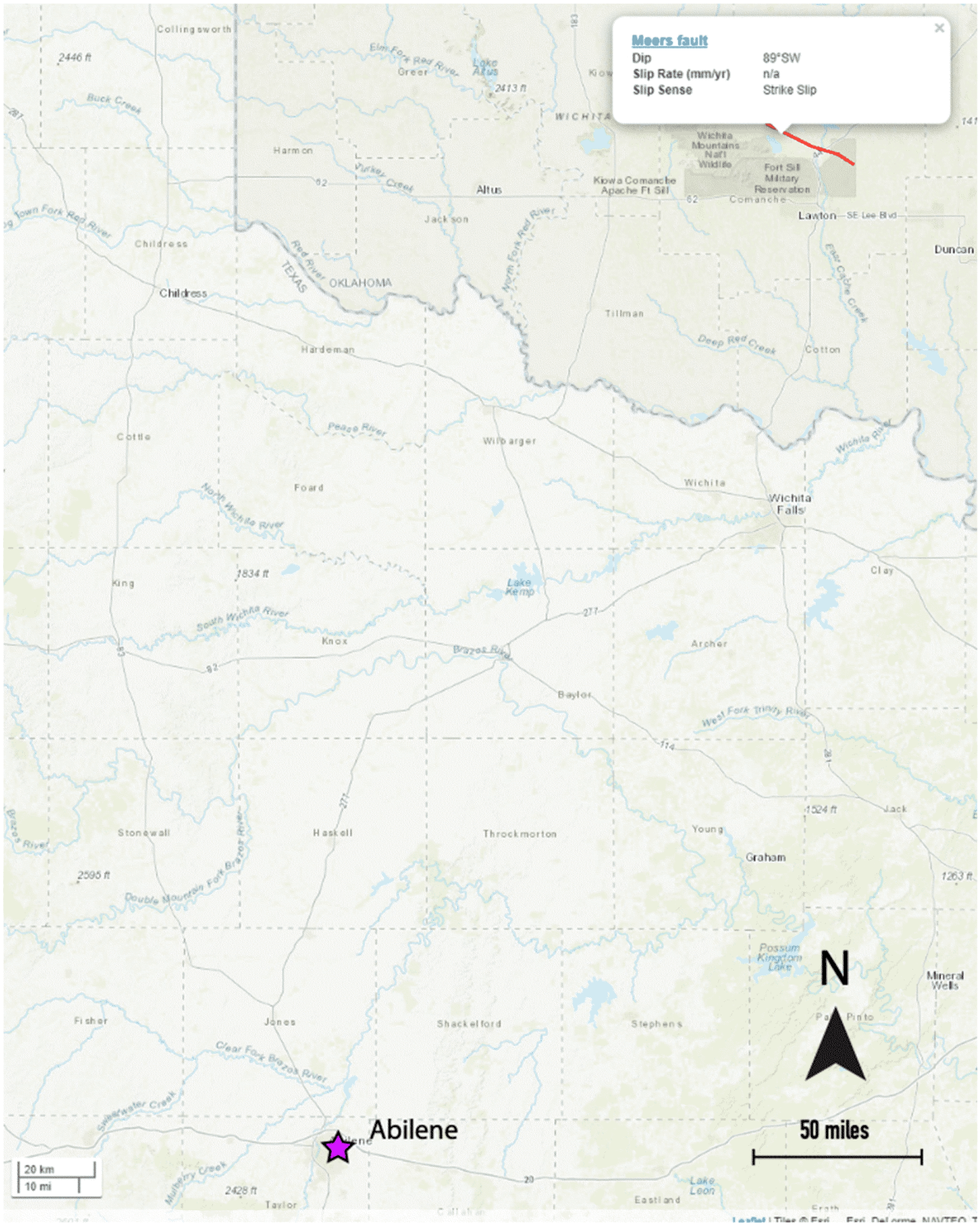
From USGS's Unified Hazard Tool [Reference 2.5-9]

Figure 2.5-23 Underground Faults Near Abilene



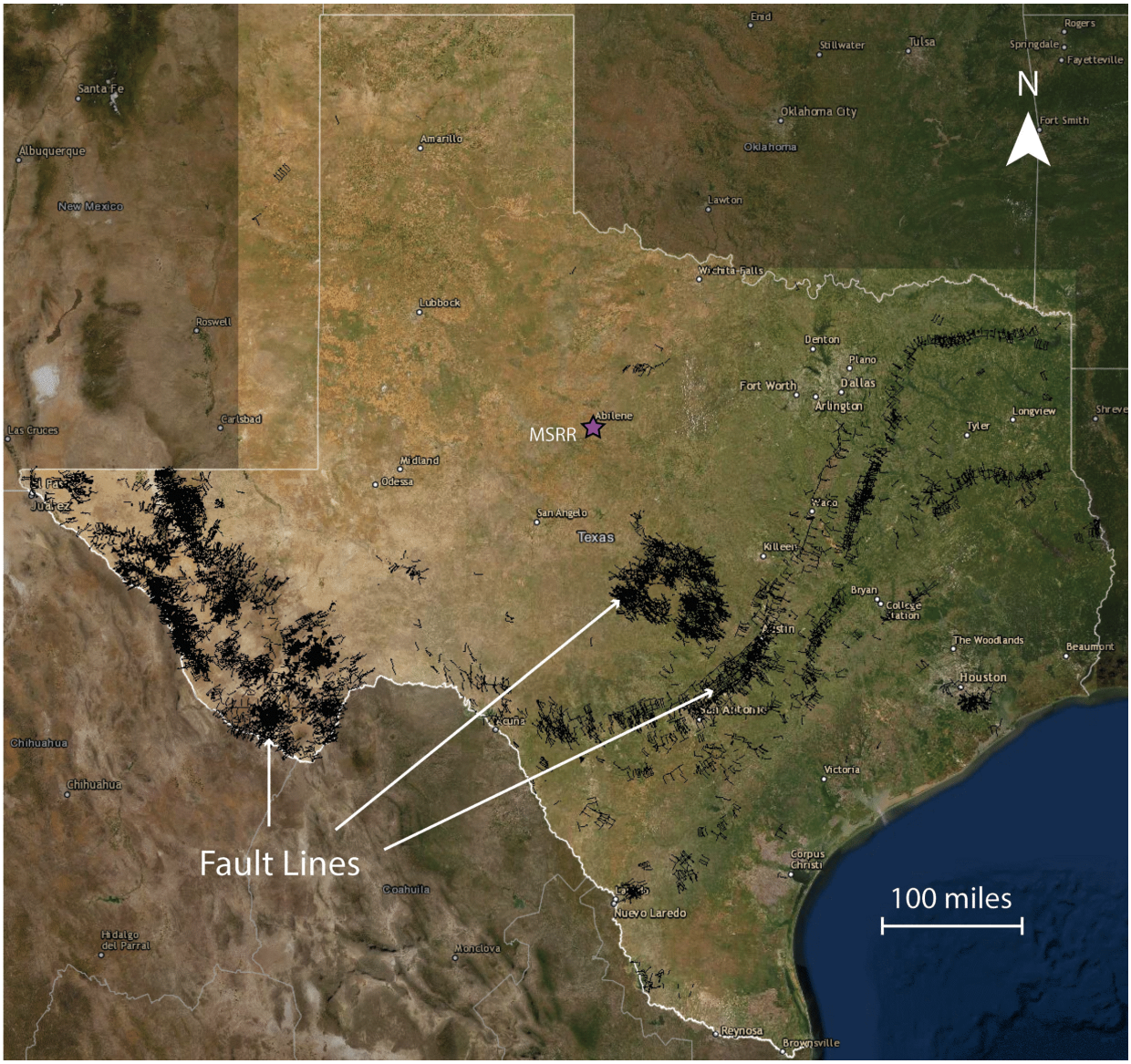
Adapted from U.S. Geological Survey [Reference 2.5-10]

Figure 2.5-24 Meers Fault in Oklahoma



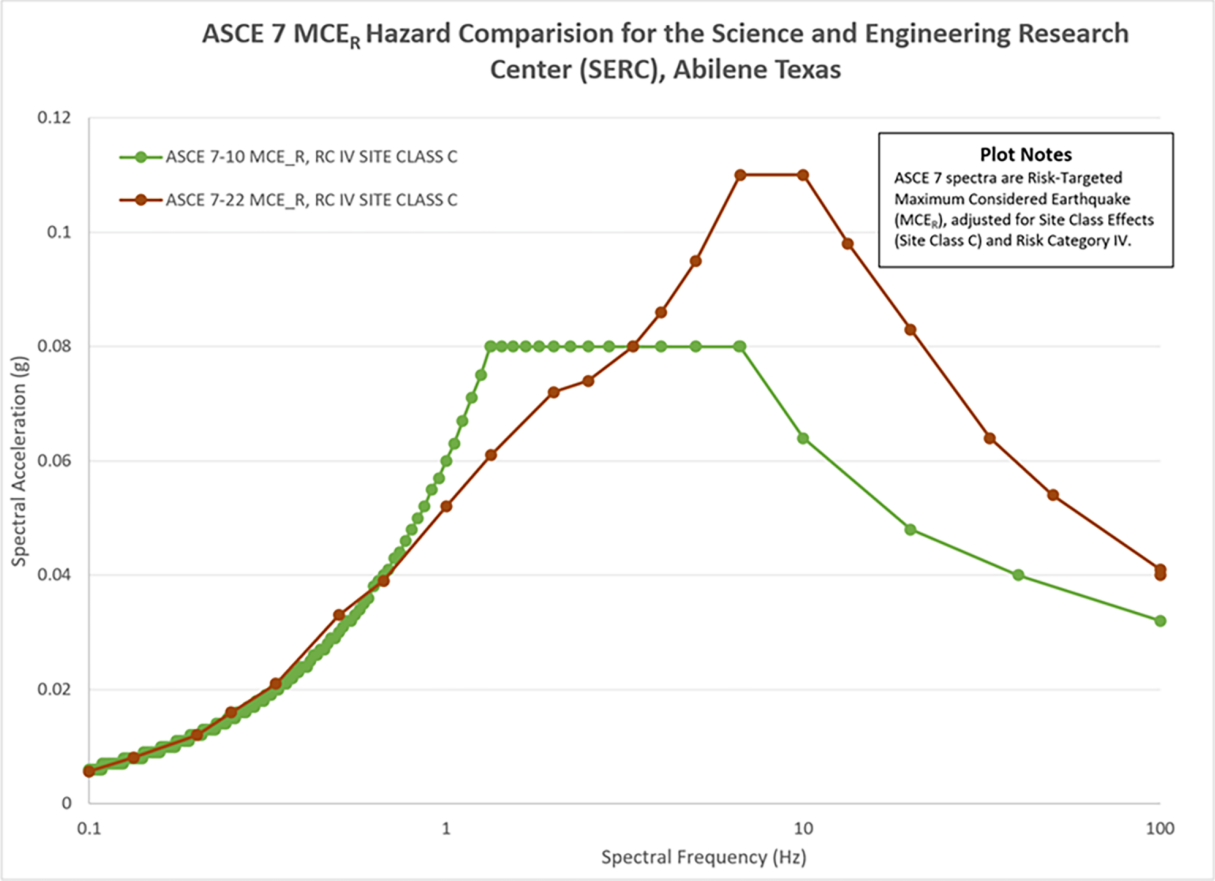
Adapted from U.S. Geological Survey [Reference 2.5-10]

Figure 2.5-25 Fault Lines in Texas



Adapted from Texas Natural Resources Information System [Reference 2.5-2]

Figure 2.5-26 ASCE 7 MCE_R Hazard Comparison



Chapter 3
Design of Structures, Systems, and Components

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS . . . 3-1

3.1 Design Criteria 3-1

3.1.1 Introduction 3-1

3.1.2 Molten Salt Research Reactor Design Criteria 3-16

3.1.3 Nuclear Regulatory Commission Guidance Documents 3-31

3.1.4 Codes and Standards Used by the MSRR 3-31

3.1.5 References. 3-33

3.2 Meteorological Damage. 3-38

3.2.1 Normal Wind Loads 3-38

3.2.2 Tornado Loading 3-40

3.2.3 Hurricane Loading 3-41

3.2.4 Precipitation Loads 3-41

3.3 Water Damage 3-43

3.3.1 Flood Protection. 3-43

3.3.2 References. 3-44

3.4 Seismic Damage. 3-45

3.4.1 Existing Multiuse Facility 3-45

3.4.2 Seismic Design for Safety-Related Structures, Systems, and Components. 3-45

3.4.3 Seismic Instrumentation. 3-48

3.4.4 References. 3-48

3.5 Systems and Components 3-60

3.5.1 General Design Basis Information 3-60

3.5.2 Classification of Structures, Systems, and Components 3-65

Appendix 3A City of Abilene, TX Building Permit for SERC 3A-1

LIST OF TABLES

Table 3.1-1 Cross Reference to Preliminary Safety Analysis Report Sections 3-34
Table 3.1-2 Cross Reference to Nuclear Regulatory Commission Guidance Documents 3-36
Table 3.4-1 Safety and Seismic Classification of Structures, Systems, and Components 3-49
Table 3.4-2 MSRR Horizontal and Vertical SSE 3-54

LIST OF FIGURES

Figure 3.1-1 The Science and Engineering Research Center that will House the Nuclear Energy eXperimental Testing Laboratory 3-4

Figure 3.1-2 Cross Section View of Science and Engineering Research Center 3-5

Figure 3.1-3 Safety-related Structures, Systems, and Components in Systems Pit 3-6

Figure 3.1-4 Science and Engineering Research Center First Floor Layout and Utilization 3-7

Figure 3.1-5 Cross Section View of SERC and Research Bay (From Southeast) 3-8

Figure 3.1-6 Cross Section View of SERC and Research Bay (From Southwest) 3-9

Figure 3.1-7 Exterior View of SERC and Research Bay (From Northwest) 3-10

Figure 3.1-8 Exterior View of SERC and Research Bay (From Northeast) 3-11

Figure 3.1-9 Exterior View of SERC and Research Bay (From Southeast) 3-12

Figure 3.1-10 Half Cross-section of the Systems pit, Slab, and the Fill Materials 3-13

Figure 3.1-11 SERC Pier and Tilt-up Panel Plan Showing Locations of Piers Relative to the Pit and Slab 3-14

Figure 3.1-12 SERC Pier and Tilt-up Panel Plan Notes 3-15

Figure 3.4-1 Systems Pit Cross Section (Rebar Dimensions Nominal/Typical) 3-55

Figure 3.4-2 Research Bay Dimensions (Nominal/Typical) 3-56

Figure 3.4-3 Research Bay Cross Section (Dimensions Nominal/Typical) 3-57

Figure 3.4-4 MSRR Horizontal Safe Shutdown Earthquake 3-58

Figure 3.4-5 MSRR Vertical Safe Shutdown Earthquake 3-59

Figure 3.5-1 Reactor Cell Diagram 3-65

CHAPTER 3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

3.1.1 Introduction

This chapter identifies and describes the architectural and engineering design criteria for the structures, systems, and components (SSCs) for the Abilene Christian University (ACU) Nuclear Energy eXperimental Testing Laboratory (NEXT Lab) Molten Salt Research Reactor (MSRR). The MSRR will be installed and operated in the NEXT Lab Science and Engineering Research Center (SERC) (see Figure 3.1-1, Figure 3.1-2, and Figure 3.1-4), an existing multiuse facility in accordance with 10 CFR 50.10(a)(2)(x). The City of Abilene building permit for the SERC is included as Appendix A to this chapter. The SERC was built to ASCE 7-10, which builds in a margin of safety. Additional safety margin was added by both the geotechnical firm and the building engineers. The ACU license boundary is the MSRR facility, which includes a radiochemistry laboratory for performing off-gas, fuel, and fuel boundary surveillance measurements, a dress out room, reactor control room, research bay, and subterranean systems pit where the MSRR will be installed and operated. The safety-related area within the MSRR facility is the systems pit. The primary safety feature of the MSRR design is the unique combination, formulation, and properties of the fuel salt, and its movement within the system. Other safety-related systems support maintaining fuel salt inventory, heat removal, and control of radionuclides within acceptable limits. All SSCs necessary for safe shutdown and long-term passive decay heat removal are located in the systems pit. Figure 3.1-3 shows a cross section of the SSCs located in the systems pit. These are all located below grade, and the SERC structure is designed such that its failure does not damage safety-related SSCs located in the systems pit. As shown in this figure, the reactor system and reactor enclosure are surrounded by thick concrete walls and a thick concrete cap. This is provided primarily for shielding, but also serves to protect the reactor system from external hazards. The fuel storage enclosure and the coolant salt and heat management enclosure are steel structures designed to retain leakage of fission product gases and to protect the safety-related SSCs from external events.

The SERC research bay is a steel support structure clad with tilt-up concrete walls constructed such that the walls fall away from the systems pit in the event of failure due to events such as tornado, large earthquake, or explosion. While there are SSCs located in the SERC that support reactor operation, they are not required for safe shutdown and long-term cooling. Examples include the control room; heating, ventilation, and air conditioning (HVAC) systems; and electrical power distribution.

The systems pit is nominally 15 ft by 80 ft and 25 ft deep (4.6 m X 24.4 m X 7.6 m) with concrete walls that are sealed to prevent ground water intrusion. Water intrusion is primarily an operational concern as flooding the pit does not affect the ability to achieve safe shutdown and long-term cooling. The subterranean design protects the reactor and SSCs necessary for safe shut down and long-term cooling from hazards associated with extreme weather and external hazards, for example explosions on Interstate 20 located about a mile north of the MSRR facility (see Section 2.1 for details).

The research bay systems pit is on a deep drilled concrete pier foundation with the drilled piers extending into the shales at a depth of approximately 55 feet below grade and are sized to have a maximum bearing pressure of 40 ksf (1915 kPa) below 40 ft. Site investigation documented that sound shales were present above and below the bearing depth of the drilled piers. The maximum net allowable bearing capacity was determined based on the information from the Texas Cone Penetrometer (TCP) Tests and information from the TxDOT Geotechnical Manual dated March 2018. Test boring data does not show concerns for seams of weakness or weathered rock once the sound shales were encountered; however, these issues were taken into account when establishing the allowable bearing capacities. Based on the TCP tests the TxDOT Manual allows for a point bearing capacity of 31 tons per square foot (tsf) or 62 kips per square foot (ksf) with a factor of safety of 2 and an allowable skin friction of 3.25 tsf or 6.5 ksf with a safety factor of 3. Supplemental drilled pier information recommends an allowable bearing capacity of 40 ksf (20 tsf) with a factor of safety of 3 and an allowable skin friction of 5 ksf (2.5 tsf) with a factor of safety of 3. Additionally, the skin friction for the top 30 feet and the bottom 5 feet of the drilled piers was neglected in the total bearing capacity calculation.

Some shallow groundwater encountered in a portion of the test borings was determined to be seasonal perched groundwater trapped in the sandy clay soils above the weathered shales. Groundwater was not a factor during excavation of the MSRR structure or during the drilling and construction of the drilled piers for the structure.

The TxDOT method takes excessive settlement into account and for a 60-inch diameter drilled pier it recommends conducting a detailed structural analysis with a maximum bearing load of 1,100 tons. The load on the MSRR piers is 1,473 kips (736.5 tons) or about 67% of the maximum allowable service load, which would not require the detailed analysis.

The floor slab is supported by the piers. Calculations for the slab support were based on the piers only and not on support from the soil. To alleviate concerns about expansive native clay soils, native soil was removed and replaced by compacted select fill near the slab. Selected fill around the safety related systems pit is compacted to not less than the following percentages of maximum dry unit weight according to ASTM D698:

- For excavations below 3 feet deep, each layer of select fill is compacted at 100 percent.
- Within 3 feet of finished surface, each layer of select fill is compacted at 95 percent.

Figure 3.1-5 through Figure 3.1-10 show cross section detail of the research bay and systems pit structure and foundation. Pier locations relative to the slab and the systems pit are shown in Figures 3.1-11 and 3.1-12.

Quality assurance (QA) plans for the SERC were implemented throughout the site evaluation, design and construction processes. The geotechnical engineering firm eHT performed site analysis, and provided quality control (QC) on those activities. Parkhill, the principle architectural engineering firm, along with McClure and Nuclear

Energy Consultants (NEC) provided QA during the design and construction phases of the SERC building. The construction contractor, Linbeck, provided primary QC for the construction phase of the building, and was supplemented by Terracon (a consulting engineering firm, specializing in environmental, facilities, geotechnical, and materials services) for field reviews during construction.

During the site investigation, eHT provided QA for the work. The QA program applied to the geotechnical report provided to Parkhill for the ACU SERC project includes a professional engineer not closely related to the geotechnical project, but with local knowledge in the areas of soil mechanics and foundations performing a peer review. A QC checklist was followed to assure that each of the required items was reviewed. The checklist was reviewed and approved by the original author of the report, a professional engineer, and a principal of eHT who is a PE.

In the design phase of the SERC, Parkhill provided the QA for the project. The structural team at Parkhill included an Engineer of Record, a Peer Engineer, and a QC Engineer. The Engineer of Record is the primary engineer responsible for the design and the production of the Construction Documents. The Peer Engineer is an experienced engineer who stays involved throughout the project and is available to the Engineer of Record for consultation and design assistance. The Quality Control Engineer is an experienced engineer who is not deeply involved in the design process who does a full QC review of the design calculations and Construction Documents. Each of the roles was performed by a Professional Engineer registered in the State of Texas. Further, ACU contracted with a third-party structural engineering firm (McClure) and Nuclear Energy Consultants, Inc. (NEC) to offer additional resources to the design team, construction team, and perform a QA review of the structural design and construction.

Parkhill and Terracon provided much of the QA during construction in collaboration with the QC from Linbeck for construction testing and inspection. A representative from Parkhill made weekly site visits and prepared field reports to document the progress of the project and address any concerns. The Linbeck QC included a global document for the project where specific areas of potential challenge were identified. A specific plan for QA of each area was implemented to minimize the likelihood of issues in that area. Inspection test plans were produced that identified tasks, the task origin, task frequency, how the task should be verified, who participated in the task, task roles, and documentation of the task once completed. Checklists were generated for each task to track them in the field and to ensure complete and quality work. There were daily QA progress inspections. Checklists were used to monitor and document progress and any non-conformance issues.

In cooperation with Linbeck, Terracon performed QA testing and observations for the SERC project as a third-party reviewer. Terracon observed work during construction and documented observations and inspection measurements including reporting any non-conformance issues for remediation and ensuring that standards were followed.

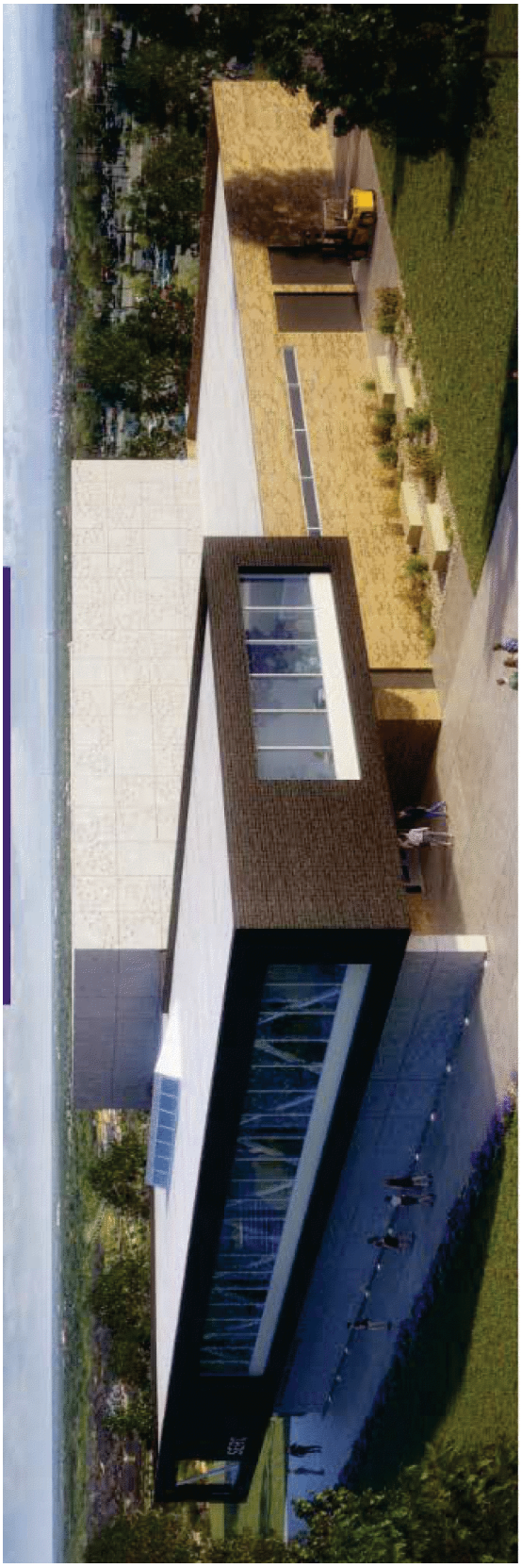


Figure 3.1-1 The Science and Engineering Research Center that will House the Nuclear Energy eXperimental Testing Laboratory

Figure 3.1-2 Cross Section View of Science and Engineering Research Center

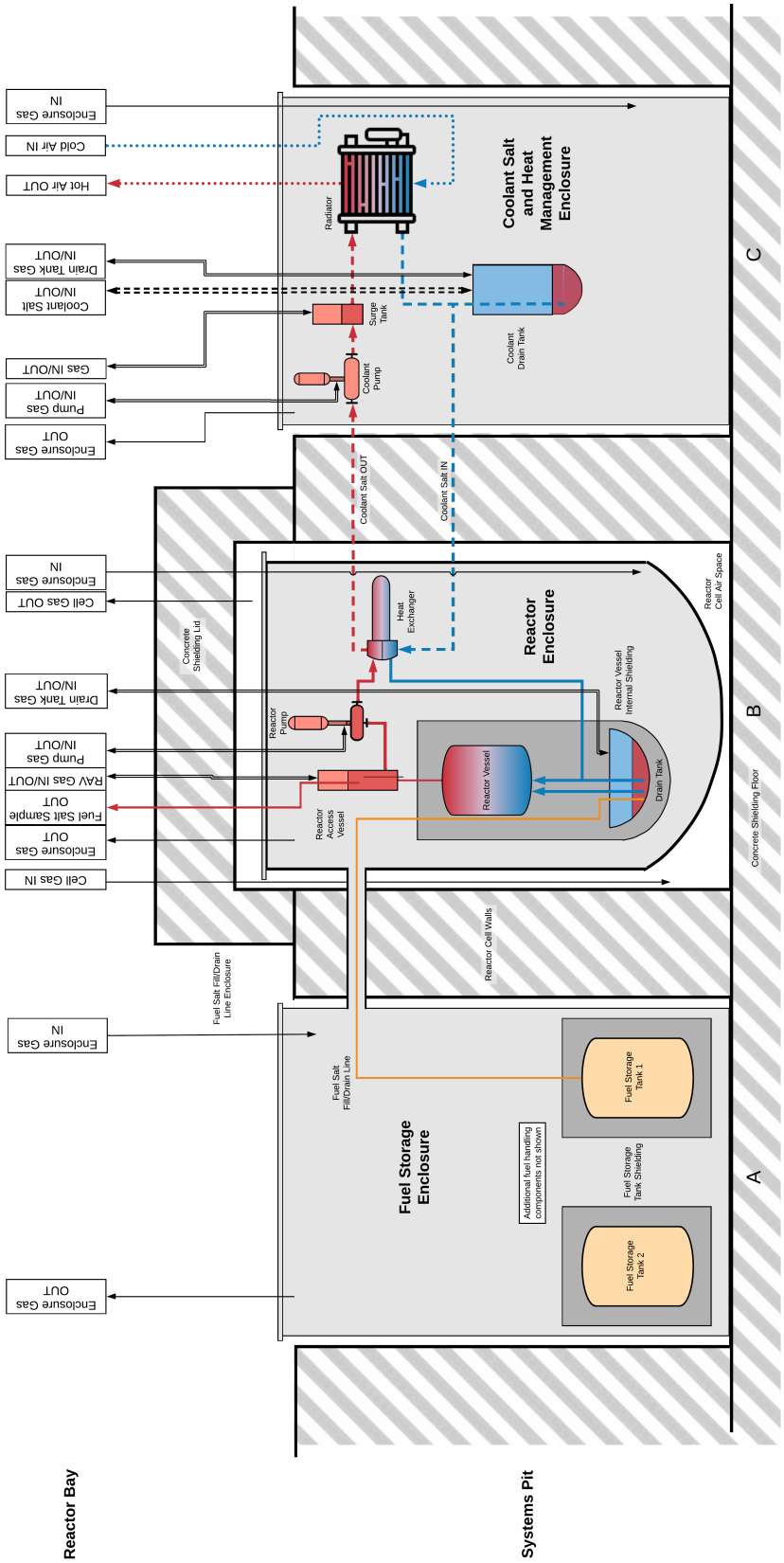
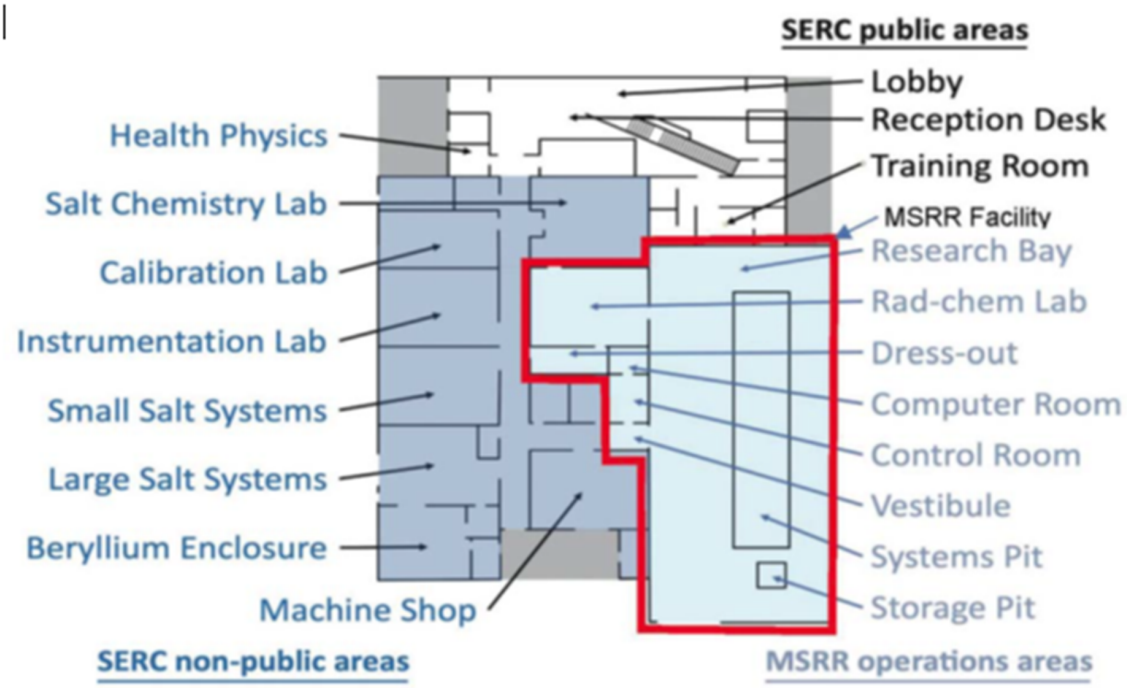


Figure 3.1-3 Safety-related Structures, Systems, and Components in Systems Pit

Figure 3.1-4 Science and Engineering Research Center First Floor Layout and Utilization



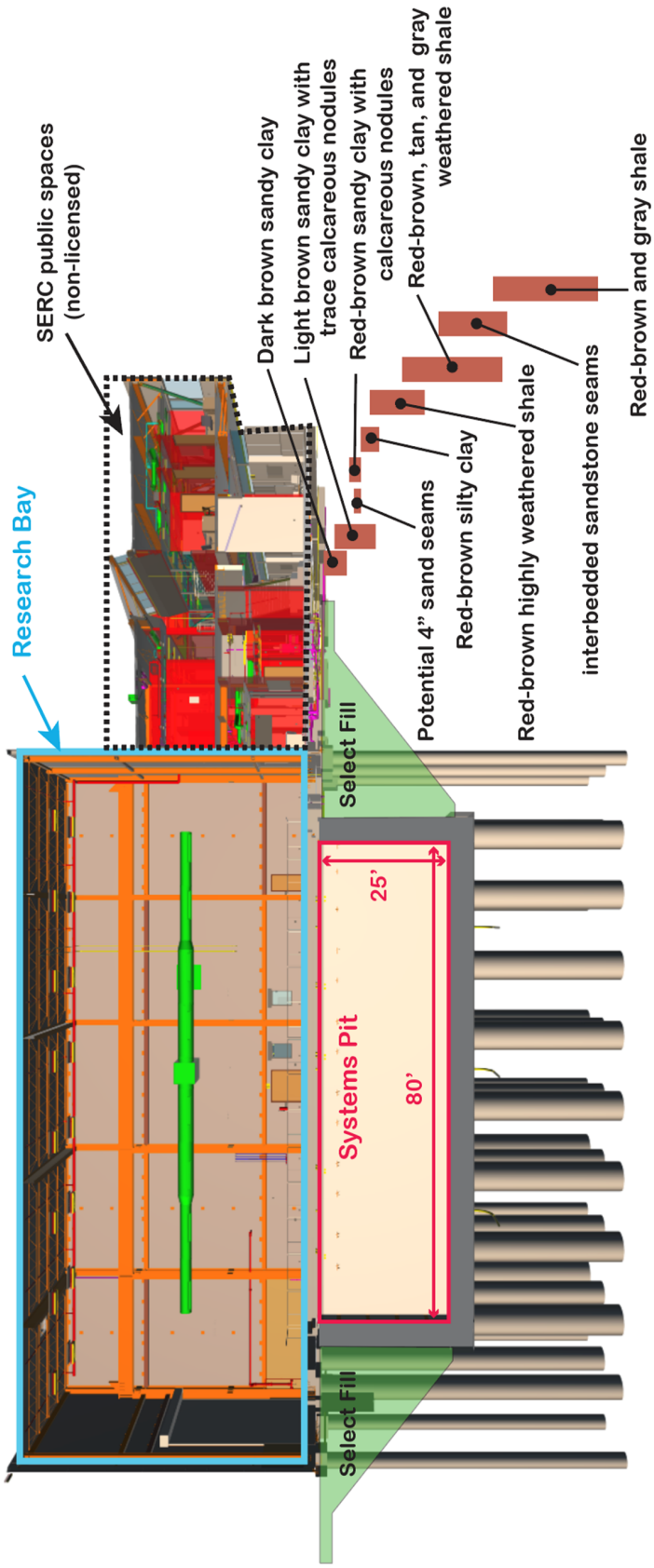


Figure 3.1-5 Cross Section View of SERC and Research Bay (From Southeast)

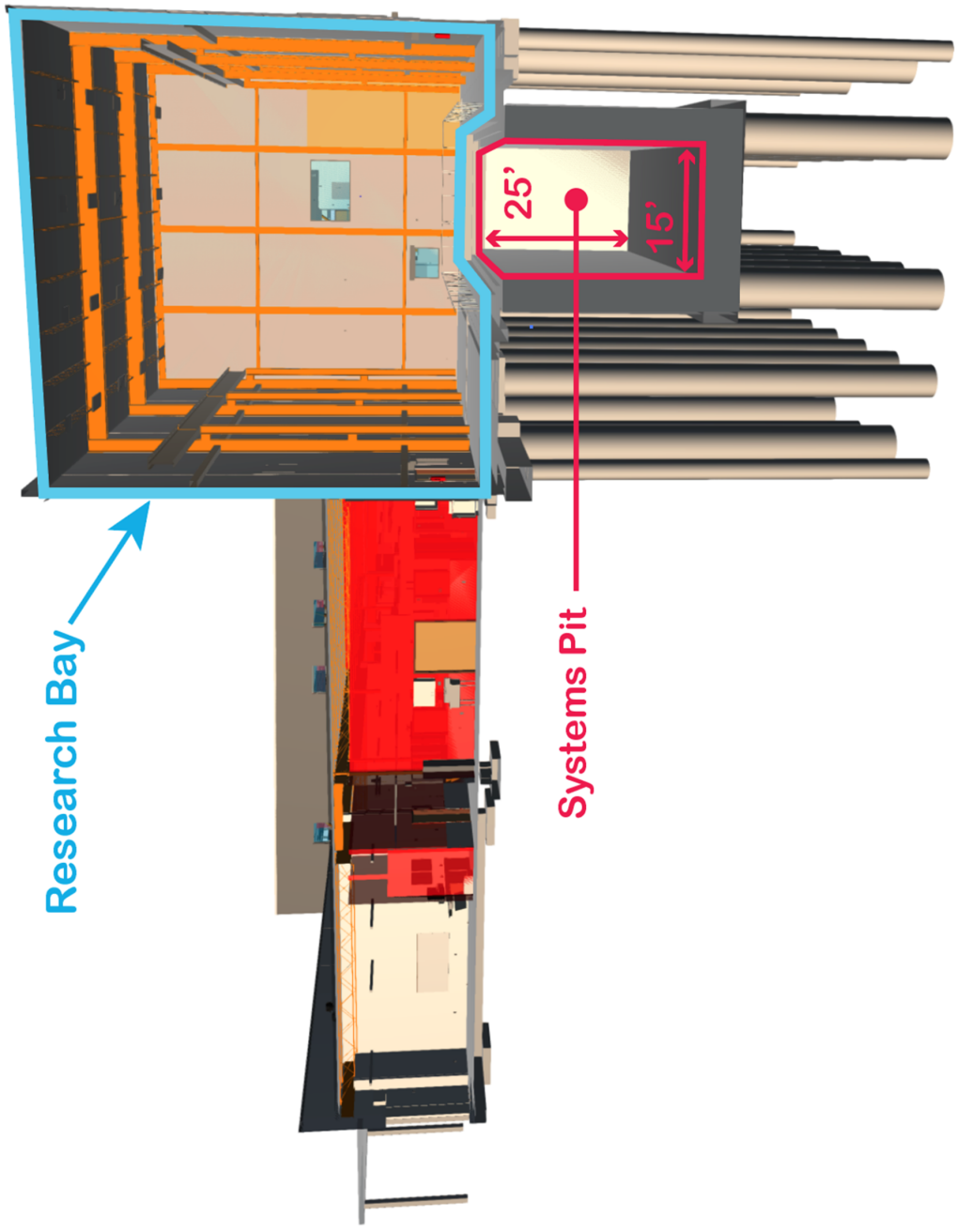


Figure 3.1-6 Cross Section View of SERC and Research Bay (From Southwest)

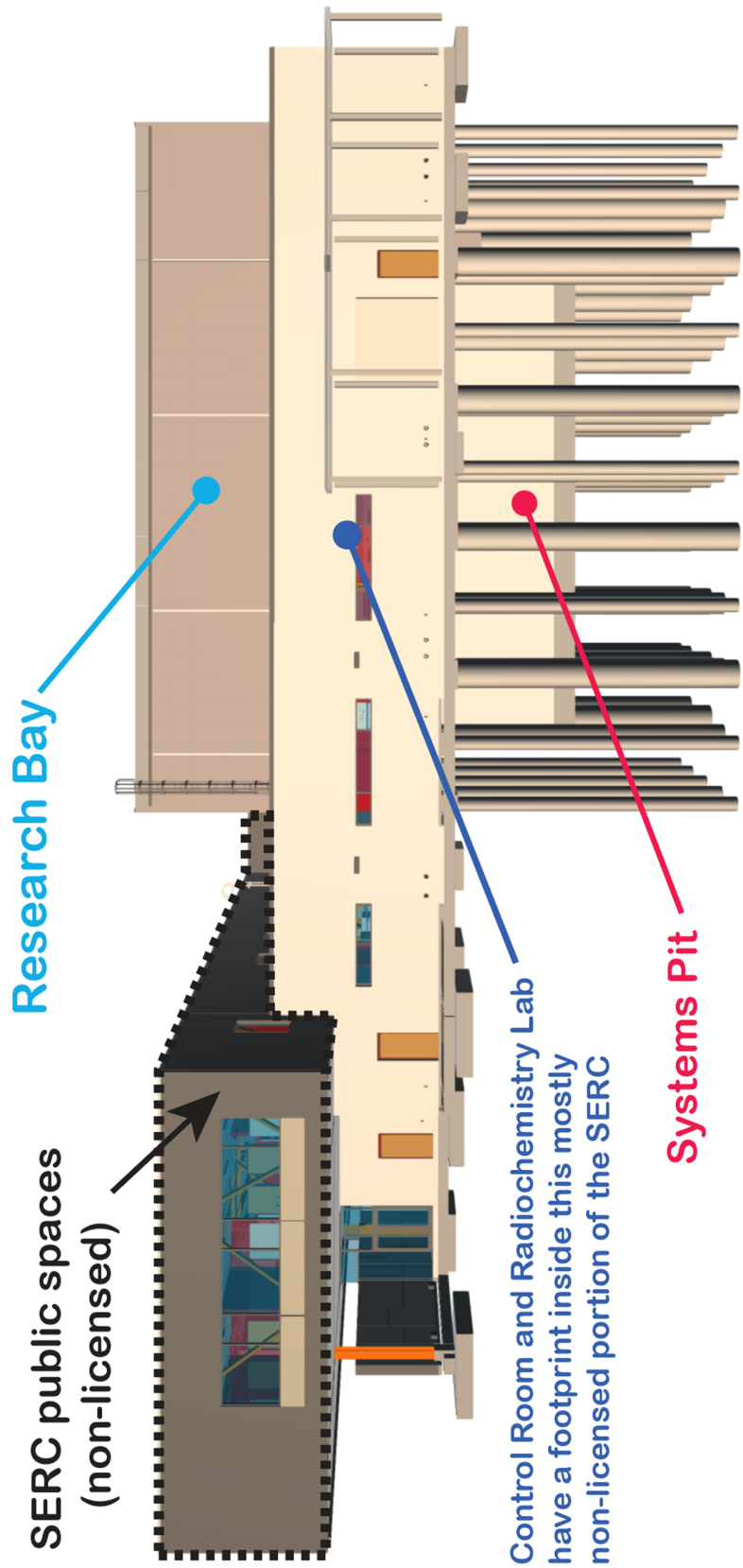


Figure 3.1-7 Exterior View of SERC and Research Bay (From Northwest)

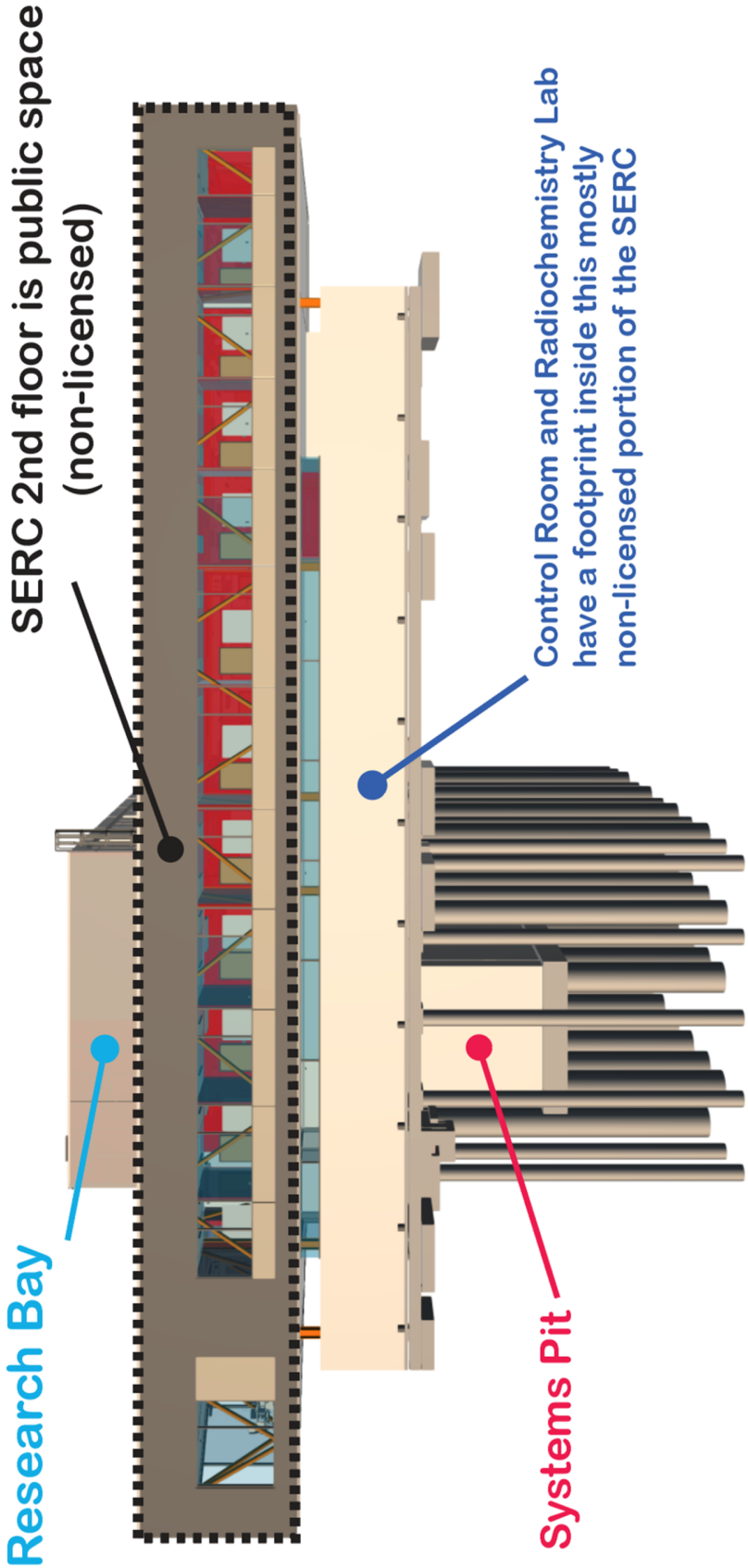


Figure 3.1-8 Exterior View of SERC and Research Bay (From Northeast)

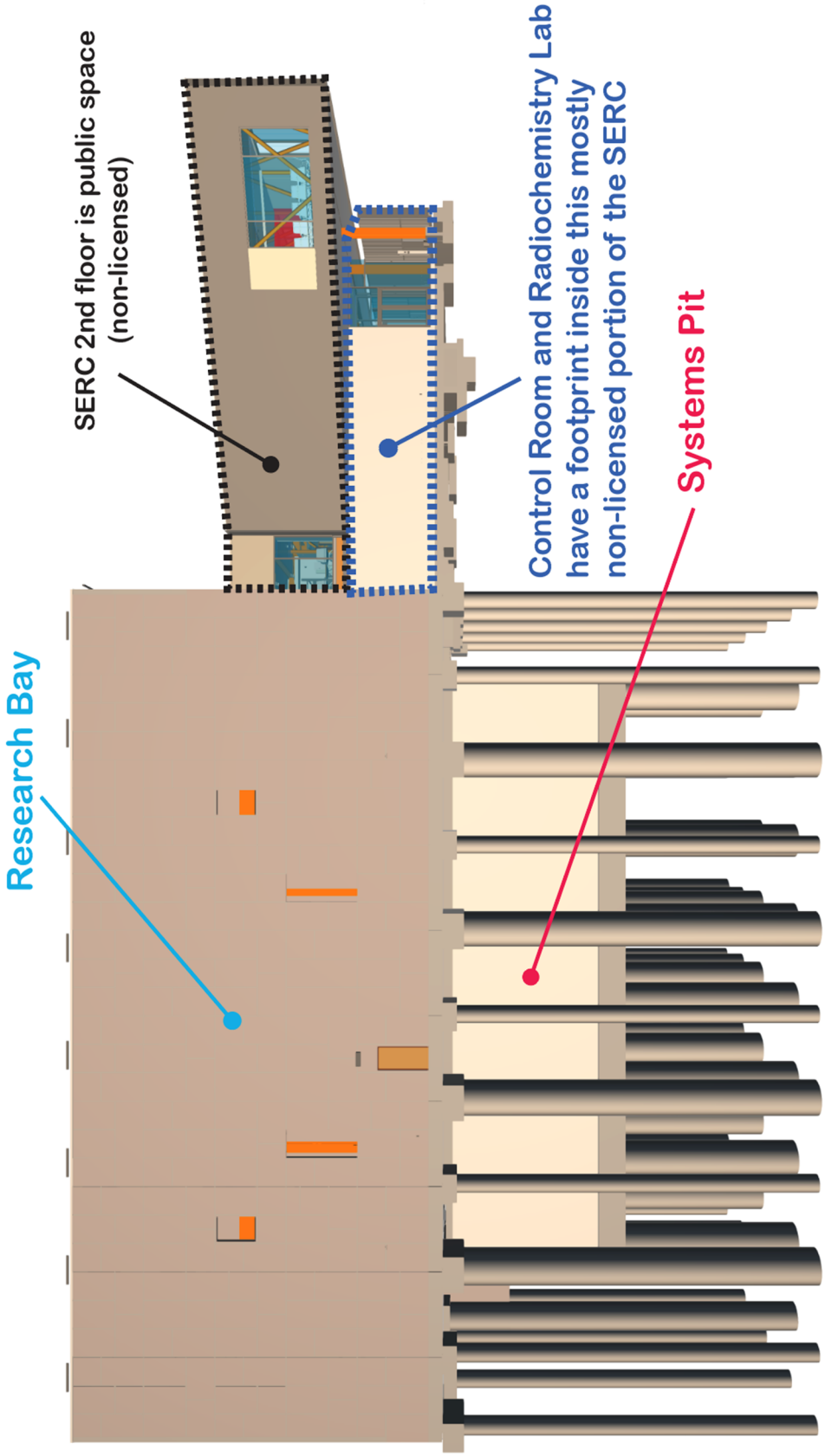
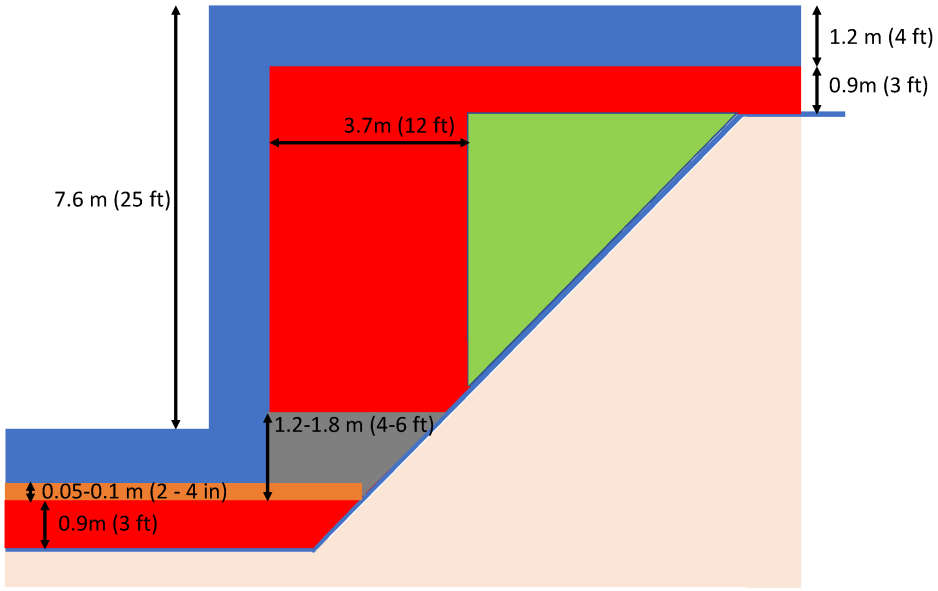


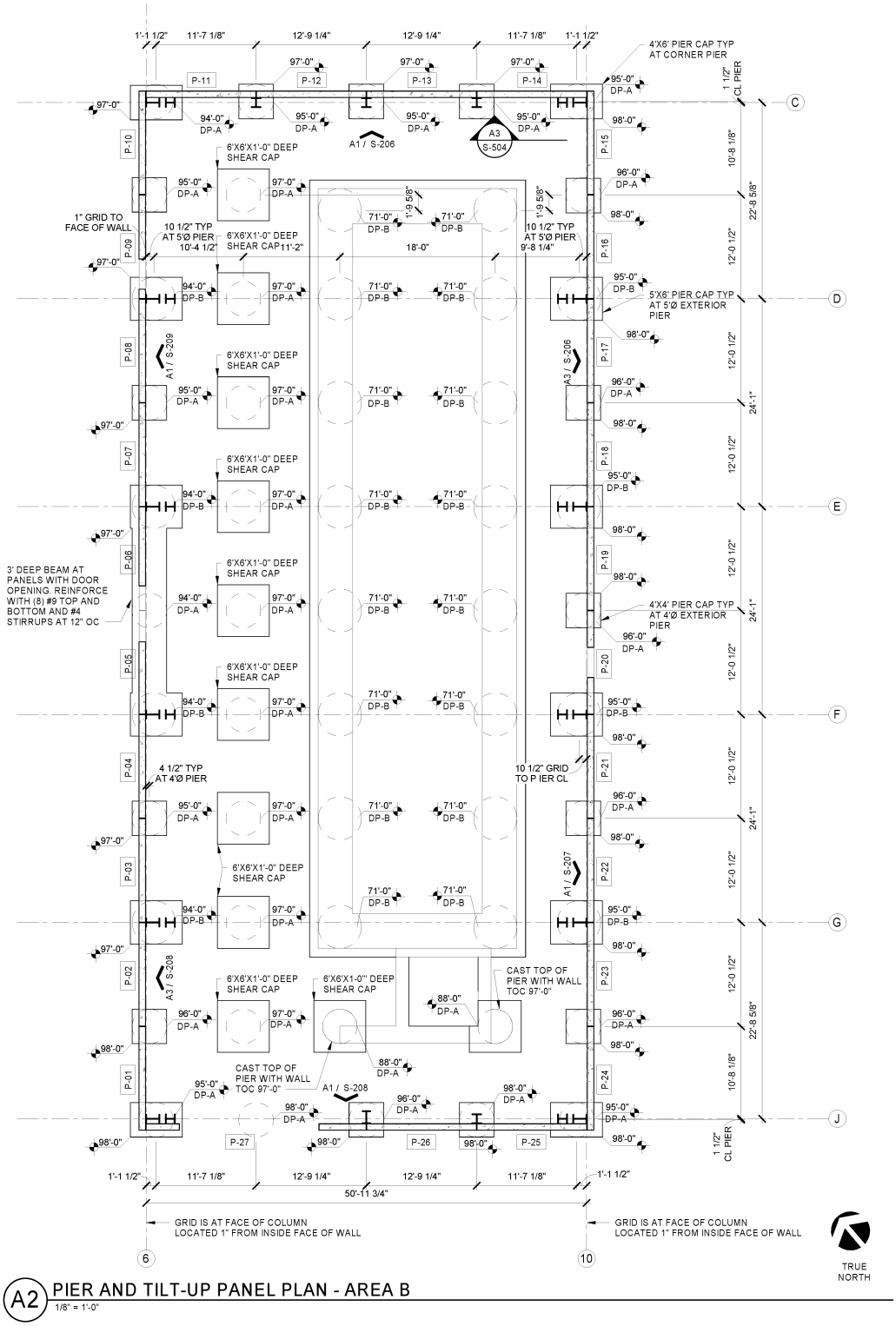
Figure 3.1-9 Exterior View of SERC and Research Bay (From Southeast)

Figure 3.1-10 Half Cross-section of the Systems pit, Slab, and the Fill Materials



The drawing is not to scale. The trench slab is approximately 1.2 m (4 ft) thick and shown in blue. A mud mat 0.05 - 0.1 m (2-4 in) thick is immediately beneath the slab and is shown in orange. A cementitious flowable fill is shown in gray and has a vertical height of 1.2 - 1.8 m (4 - 6 ft). The select fill is shown as red and extends at least 3.7 m (12 ft) out from the trench wall and is at least the top 0.9 m (3 ft) immediately below the slab. The on-site fill material is shown in green and was used further than 3.7 m (12 ft) from the trench wall and below 0.9 m (3 ft) from the bottom of the slab. The tan color in the figure represents undisturbed soil. The piers are not represented in this figure.

Figure 3.1-11 SERC Pier and Tilt-up Panel Plan Showing Locations of Piers Relative to the Pit and Slab



A2 PIER AND TILT-UP PANEL PLAN - AREA B
1/8" = 1'-0"

Figure 3.1-12 SERC Pier and Tilt-up Panel Plan Notes

PLAN NOTES

1. REFERENCE S-111B FOR OTHER FOUNDATION ELEMENTS
2. CAST PIER CAPS MONOLITHICALLY WITH SLAB
3. COORDINATE DIMENSIONS AND LOCATIONS OF ARCHITECTURAL COMPONENTS WITH ARCHITECTURAL DRAWINGS.

TILT-WALL PANEL SCHEDULE

MARK	WIDTH	LENGTH	TOP ELEVATION	BOTTOM ELEVATION	HEIGHT
P-01	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-02	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-03	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-04	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-05	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-06	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-07	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-08	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-09	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-10	9"	11'-11 3/4"	150'-0"	97'-1"	52'-11"
P-11	9"	12'-8 1/2"	150'-0"	97'-1"	52'-11"
P-12	9"	12'-8 1/2"	150'-0"	97'-1"	52'-11"
P-13	9"	12'-8 1/2"	150'-0"	97'-1"	52'-11"
P-14	9"	12'-8 1/2"	150'-0"	97'-1"	52'-11"
P-15	9"	11'-7 1/4"	150'-0"	98'-1"	51'-11"
P-16	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-17	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-18	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-19	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-20	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-21	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-22	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-23	9"	11'-11 3/4"	150'-0"	98'-1"	51'-11"
P-24	9"	11'-7 1/4"	150'-0"	98'-1"	51'-11"
P-25	9"	12'-8 1/2"	150'-0"	98'-1"	51'-11"
P-26	9"	12'-8 1/2"	150'-0"	98'-1"	51'-11"
P-27	9"	25'-5 3/4"	150'-0"	98'-1"	51'-11"

DRILLED PIER SCHEDULE

MARK	COUNT	DIAMETER	BEARING ELEVATION	VERTICAL REINFORCING	TIES	FRICTION CAPACITY (30'-40' DEPTH)	FRICTION CAPACITY (40'-50' DEPTH)	BEARING CAPACITY	TOTAL CAPACITY
DP-A	31	4'-0"	45'-0"	(24) #8	#4 AT 12" OC	502.6k	628.3k	502.6k	1633.5k
DP-B	24	5'-0"	45'-0"	(24) #10	#4 AT 12" OC	628.3k	785.3k	785.3k	2198.9k

3.1.2 Molten Salt Research Reactor Design Criteria

This section provides design criteria (DC) that establish the basic design requirements for the NEXT Lab MSRR. These design criteria were derived from the Molten Salt (Power) Reactor design criteria developed in draft American National Standards Institute/American Nuclear Society ANSI/ANS-20.2-2021 [Reference 3.1-1], which was derived from the advanced reactor design criteria (ARDC) in Regulatory Guide (RG) 1.232, Revision 0 and the General Design Criteria in Appendix A of 10 CFR 50. The bases for departures from the ARDC are noted where applicable. The MSRR-specific approach for meeting the intent for each of these criteria is described in the sections referenced in Table 3.1-1.

The MSRR-specific design criteria include:

- Series I - Overall requirements
- Series II - Protection by multiple barriers
- Series III - Protection and reactivity controls
- Series IV - Fluid systems
- Series V - Heat transport systems
- Series VI - Fuel and reactivity control
- Series VII - Salt systems and control

The DC in each series are listed in the following subsections. The DC addressed by this preliminary safety analysis report (PSAR) are listed in Table 3.1-1 and are cross referenced to the related PSAR sections.

The principle design criteria follow the regulations applicable to the NEXT Lab licensing strategy for the MSRR under 10 CFR Part 50. The NRC regulations have been evaluated for applicability to the MSRR facility for the preliminary design. The Operating License application will include a final analysis of the applicable regulations.

3.1.2.1 Series I - Overall Requirements

- Criterion 1: Quality standards and records

  The safety related SSCs shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program consistent with ANS15.8 shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate

records of the design, fabrication, erection, and testing of safety related SSCs shall be maintained by or under the control of ACU (licensee) throughout the life of the MSRR facility.

Bases: Informed by 10 CFR 50 Appendix A. ANS 15.8 defines quality assurance requirements for research reactors.

- Criterion 2: Design bases for protection against natural phenomena

  The safety related SSCs shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

  Bases: Informed by 10 CFR 50 Appendix A. Historical natural phenomena define the design basis events. This criterion does not establish a basis for the selection of beyond the design basis natural events. Tsunamis and seiches are not applicable to the MSRR site.

- Criterion 3: Fire protection

  The safety related SSCs shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire-resistant materials shall be used wherever practical throughout the MSRR facility, particularly in locations with safety related SSCs. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on safety related SSCs. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.

  Bases: Informed by RG 1.232. Changed “the unit” in second sentence to “the MSRR” to be clear where this design criteria applies.

- Criterion 4: Environmental and dynamic effects design bases

  The safety related SSCs shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. These SSCs, shall be appropriately protected against dynamic effects of events and conditions outside the MSRR facility. Dynamic effects associated with postulated pipe ruptures will be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Bases: Informed by RG 1.232. Changed “the nuclear power unit” in second sentence to “the MSRR” because the MSRR is not a nuclear power reactor. Language defining specific effects such as missiles, pipe whips, and discharging fluids has been removed because such phenomena will not be induced by the low system pressures of the MSRR. Dynamics effects will be analyzed under appropriate MSRR conditions.

- Criterion 5: Sharing of safety-related SSCs

  Not Applicable to the MSRR.

  Bases: The MSRR is a single unit. No systems are shared with another nuclear unit.

- Criterion numbers 6-9 are not used in 10 CFR 50 Appendix A.

3.1.2.2 Series II - Protection by multiple barriers

- Criterion 10: Reactor design

  The reactor system and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

  Bases: Informed by RG 1.232, Appendix C, “Modular High-Temperature Gas-Cooled Reactor (MHTGR).” Liquid salt fuel is not susceptible to thermomechanical damage. The relevant safety concept is release of radionuclides.

- Criterion 11: Reactor inherent protection

  The reactor core and associated systems that contribute to reactivity feedback shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

  Bases: Informed by RG 1.232, ARDC.

- Criterion 12: Suppression of power oscillations

  The reactor core and associated control and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.

  Bases: Informed by RG 1.232, Appendix C, MHTGR.

- Criterion 13: Instrumentation and control

  Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety, including those variables and systems that can affect the fission process and the integrity of the reactor core, the reactor fuel salt boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

  Bases: Informed by RG 1.232, Appendix C, MHTGR. “Helium pressure boundary” was changed to “fuel salt boundary” to be consistent with the MSRR.

- Criterion 14: Reactor fuel salt boundary

  The reactor fuel salt boundary shall be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, or of gross rupture.

  Basis: Informed by RG 1.232, ARDC. “Coolant” was changed to “fuel salt” to be specific to the MSRR.

- Criterion 15: Reactor fuel salt system design

  The reactor fuel salt system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor fuel salt boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

  Basis: Informed by RG 1.232, ARDC. “Coolant” was changed to “fuel salt” to be specific to the MSRR.

- Criterion 16: Containment design

  A reactor functional containment shall be provided to control the release of radioactivity to the environment and to ensure that the safety related functional containment design conditions are not exceeded for as long as postulated accident conditions require.

  Bases: Informed by RG 1.232, Appendix C, MHTGR. Removed text regarding functional containment characteristics, “consisting of multiple barriers internal and/or external to the reactor and its cooling system.” Functional containment is provided by the fuel salt itself, which retains the great majority of fission products, and three engineered safety features surrounding the reactor system. Functional containment extends to fuel salt storage and transfer outside of the reactor system. Functional containment also extends to the off-gas system which is contained within the off-gas enclosure and double walled piping connecting to the reactor enclosure.

- Criterion 17: Electric power systems

  Electric power systems shall be provided when required to permit functioning of structures, systems, and components. Electrical faults shall not impair a safety function.

  Basis: The electric power systems are not safety related. The electric power system shall not be able to impair a safety function. Specific requirements for the criterion outlined in the ARDC implicitly assume that electric power is safety related; such requirements are not applicable to the MSRR. Requirements for protection against electric faults are included.

- Criterion 18: Inspection and testing of electric power systems

  Electric power systems are not safety related. Electrical power systems will be inspected and tested to assure reasonably reliable operation to minimize system interrupts and maximize system uptime and availability.

  Basis: The electric power systems are not safety related. Testing and inspection are needed only to assure reliable operation to allow the MSRR to achieve its research mission. Specific requirements for the criterion outlined in the ARDC implicitly assume that electric power is safety related; such requirements are not applicable to the MSRR.

- Criterion 19: Control room

  A control room shall be provided from which actions can be taken to operate the MSRR facility safely under normal conditions and anticipated operational occurrences.

  Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and anticipated operational occurrences, and egress under accident conditions.

  Basis: Informed by RG 1.232, ARDC. The shutdown safety objectives were revised to be consistent with the MSRR design and performance bases. No operator action is required to trigger RPS or ESFAS. Operators are not required to be physically present within the control room during accidents. Environmental conditions such as a tornado, and internal accidents such as a fire and the release of HF, may require the control room to be evacuated. The reactor will be shut down when the control room is evacuated. Interventions such as fuel loading, inspection, and repair may require the control room. RG 1.232 ARDC are intended for power reactors and there is no requirement for research reactors to have a remote shutdown capability, so the requirement is not included in the MSRR design criterion. The remote shutdown capability is not needed in response to any accident scenario.

3.1.2.3 Series III - Protection and reactivity controls

- Criterion 20: Protection system functions

  The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to ensure that specified acceptable radionuclide release design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of safety related SSCs.

  Bases: Informed by RG 1.232, Appendix C, MHTGR. Liquid fuel cannot be readily damaged. A conservative safety concept that preserves the safety intent of the GDC is retention of radionuclides.

- Criterion 21: Protection system reliability and testability

  The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that no single failure results in loss of the protection function. The protection system shall be designed to permit periodic functional testing including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

  Bases: Informed by 10 CFR 50 Appendix A. The protection system shall be tested while the reactor is in service but not while the reactor is critical unless it is acceptable that testing of the protection system results in reactor shutdown. Removal from service of any component or channel will require shutdown.

- Criterion 22: Protection system independence

  The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

  Bases: Informed by 10 CFR 50 Appendix A

- Criterion 23: Protection system failure modes

  The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced.

  Bases: Informed by 10 CFR 50 Appendix A

- Criterion 24: Separation of protection and control systems

  The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

  Bases: Informed by 10 CFR 50 Appendix A

- Criterion 25: Protection system requirements for reactivity control malfunctions

  The protection system shall be designed to ensure that specified acceptable radionuclide release design limits are not exceeded during any anticipated operational occurrence accounting for a single malfunction of the reactivity control systems.

  Bases: Informed by RG 1.232, Appendix C, MHTGR.

- Criterion 26: Reactor protection systems

  A minimum of two reactor protection systems shall provide:

  A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences.

  A means which is independent and diverse from the other(s) shall assure that the design limits for the fission product barriers are not exceeded.

  A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the fuel salt is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.

  A means for holding the reactor in safe shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.

  Bases: Informed by RG 1.232, ARDC. The concept of cooling the fuel salt has been substituted for “cool the core” to translate the safety intent of the ARDC to the relevant properties of molten salt reactors. Statements relating to controlling the rate of reactivity change were omitted because the RPS drains the reactor to assure design limits are not exceeded.

- Criterion 27: Reactivity Control System Capability

  A non-safety related method of controlling reactivity shall be provided in addition to the reactor protection system.

  Bases: Reflects the novel nature of the MSRR. The GDC in 10 CFR 50 Appendix A contains language that is not applicable to a non-safety related reactivity control system.

- Criterion 28: Reactivity limits

  The reactor protection system shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor fuel salt boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the fuel salt.

  Bases: Informed by RG 1.232, ARDC. “Coolant” changed to “fuel salt” and “cool the core” changed to “cool the fuel salt” to be consistent with molten salt reactor technology.

- Criterion 29: Protection against anticipated operational occurrences

  The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

  Bases: Informed by 10 CFR 50 Appendix A

3.1.2.4 Series IV - Fluid systems

- Criterion 30: Quality of fuel salt boundary

  Components that are part of the fuel salt boundary shall be designed, fabricated, erected, and tested to quality standards appropriate with their function. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of fuel salt leakage.

  Bases: Informed by RG 1.232, ARDC. “Coolant” changed to “fuel salt” and “reactor coolant” changed to “fuel salt” to reflect molten salt reactor technology. ANS 15.8 defines the quality program for research reactors.

- Criterion 31: Fracture prevention of fuel salt boundary

  The fuel salt boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, anticipated operational occurrences, and postulated accident conditions the probability of rupture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and

other conditions of the boundary material(s) under operating, maintenance, testing, anticipated operational occurrences, and postulated accident conditions including uncertainties.

Bases: Informed by RG 1.232, ARDC. The specific causes for failure of a single layer were dropped in consideration of functional containment being provided through multiple layers and mechanisms. “(1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture” was replaced by “of rupture.” “Reactor coolant” changed to “fuel salt” and “reactor coolant” changed to “fuel salt” to reflect molten salt reactor technology.

- Criterion 32: Inspection of fuel salt boundary

  The fuel salt boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features, and (2) an appropriate material surveillance program for the fuel salt boundary.

  Bases: Informed by RG 1.232, ARDC. The term “coolant boundary” has been replaced by “fuel salt boundary” reflecting molten salt reactor technology. Periodic inspection, functional testing, and material surveillance will be evaluated at the design finalization.

- Criterion 33: Reactor coolant makeup

  Not applicable to the MSRR.

  Bases: A breech in the coolant loop and loss of coolant salt does not result in loss of reactor system integrity nor impair decay heat removal. Fuel salt additions to the reactor are addressed under concerns related to reactivity control, fuel salt control, and functional containment barriers.

- Criterion 34: Residual heat removal

  A passive system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the fuel salt at a rate such that specified acceptable radionuclide release design limits and the design conditions of the fuel salt boundary are not exceeded.

  Suitable redundancy in components and features shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.

  Bases: Informed by Regulatory Guide 1.232, ARDC. Changed “reactor core” to “fuel salt” and “reactor coolant” to “fuel salt” to be consistent with molten salt reactor technology. Residual heat removal is entirely passive to the foundation and atmosphere. Requirements relating to interconnections, leak detection, and isolation are subsumed into the design criteria for the functional containment (PDC 50-57).

- Criterion 35: Fuel salt cooling system

  A passive system to assure sufficient fuel salt cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the fuel salt such that effective fuel salt cooling is maintained and radionuclide release is limited during and following postulated accidents.

  Suitable redundancy in components and features shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.

  Bases: Informed by RG 1.232, ARDC. Changed “core” to “fuel salt” to reflect molten salt reactor technology. Decay heat removal is entirely passive to the foundation and atmosphere. Requirements for the passive heat removal systems relating to interconnections, leak detection, and isolation are covered by existing functional containment design criteria. (PDC 50-57).

  Note: Criteria 36, 37, and 39 are not used. Requirements for testing and inspection are subsumed under requirements for testing of functional containment heat removal and inspection of the functional containment.

- Criterion 38: Functional containment heat removal

  A passive system to remove heat from the functional containment structure(s) shall be provided as necessary to maintain the functional containment structures’ pressure and temperature within acceptable limits following any postulated accidents.

  Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure. An active auxiliary heat removal system shall be provided to maintain functional containment structure temperatures during normal operation.

  Bases: Informed by RG 1.232, ARDC. Functional containment heat removal is entirely passive to the foundation and atmosphere during shutdown.

- Criterion 40: Testing of functional containment heat removal system

  The containment heat removal system shall be designed to permit appropriate periodic functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of associated systems.

  Bases: Informed by RG 1.232, ARDC.

Criteria 41, 42, and 43 are not used. Cleanup of the reactor cell or reactor enclosure is not required in response to any accident analyzed in Chapter 13.

Criteria 44, 45, and 46 are not used because their safety requirements are subsumed under criterion 38 and 40.

- Criterion numbers 47 - 49 are not used.

3.1.2.5 Series V - Heat transport systems

- Criterion 50: Functional containment design basis

  The functional containment structure(s), including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure(s) and internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from postulated accidents. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

  Bases: Informed by RG 1.232, ARDC.

- Criterion 51: Fracture prevention of functional containment boundaries

  The boundary of the functional containment structure(s) shall be designed with sufficient margin to assure that under operating, maintenance, testing, anticipated operational occurrences, and postulated accident conditions the probability of rupture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary materials during operation, maintenance, testing, anticipated operational occurrences, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

  Bases: Informed by RG 1.232, ARDC. The specific causes for failure of a single layer were dropped in consideration of functional containment being provided through multiple layers and mechanisms. “(1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture” was replaced by “of rupture”. “Reactor coolant” changed to “fuel salt” and “reactor coolant” changed to “fuel salt” to reflect molten salt reactor technology.

- Criterion 52: Capability for functional containment leakage rate testing

  The containment structure(s) and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

  Bases: Informed by RG 1.232, ARDC.

- Criterion 53: Provisions for functional containment testing and inspection

  The functional containment structure(s) shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations and passive heat sinks, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak tightness of penetrations that have resilient seals and expansion bellows.

  Bases: Informed by RG 1.232, ARDC. Inspection of penetrations shall be visual only. Inspections will look for changes to passive heat sinks that could significantly affect heat removal, such as obstructions or surface fouling.

- Criterion 54: Piping systems penetrating functional containment

  Piping systems penetrating the functional containment structure(s) shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities. Such piping systems shall be designed with the capability to verify, by testing, the operational readiness of any isolation valves and associated apparatus periodically and to confirm that valve leakage is within acceptable limits.

  Bases: Informed by RG 1.232, ARDC.

- Criterion 55: Radionuclide interfacing lines penetrating functional containment

  Each line where a single failure could lead to a bypass of functional containment, (those lines that interface directly with fuel salt or fission products and interface with systems outside the functional containment), shall be provided with two adequately reliable containment isolation mechanisms, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, or small fuel salt or fission product sampling lines are acceptable on some other defined basis. These mechanisms shall be located to minimize the probability of failure due to environmental or external hazards.

  Bases: Informed by 10 CFR 50 Appendix A. This criterion has been made performance based, while maintaining the safety intent of the GDC. The safety intent of locating isolation valves close to containment has been preserved by including the statement “These mechanisms shall be located to minimize the probability of failure due to environmental or external hazards.” Bypass lines which interface with fuel or off-gas may be excluded from the requirements concerning two containment isolation mechanisms for several reasons: the size of the potential gaseous radionuclide source term, the consequence of in-leakage from the environment, low driving pressure differential, or other isolation mechanisms which regulate air flow through the systems pit. If relevant, such bypass lines will be delineated in the FSAR and an appropriate evaluation will be provided.

- Criterion 56: Functional containment isolation

  Each line that if left open could lead to a bypass of functional containment, such as ventilation systems that connect multiple volumes and may pass outside containment, shall be provided with two adequately reliable containment isolation mechanisms, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis. These mechanisms shall be located to minimize the probability of failure due to environmental or external hazards.

  Bases: Informed by 10 CFR 50 Appendix A. This criterion has been made performance based, while maintaining the safety intent of the GDC. The safety intent of locating isolation valves close to containment has been preserved by including the statement “These mechanisms shall be located to minimize the probability of failure due to environmental or external hazards.”

  Criterion 57: Closed system isolation valves

  Each line where a single failure in the line during off-normal conditions could lead to a bypass of functional containment, such as a cooling or instrument line that does not interface with fuel salt or fission products, shall have at least one adequately reliable containment isolation mechanism. This mechanism shall be located to minimize the probability of failure due to environmental or external hazards.

  Bases: Informed by 10 CFR 50 Appendix A. This criterion has been made performance based, while maintaining the safety intent of the GDC. The safety intent of locating isolation valves close to containment has been preserved by including the statement “These mechanisms shall be located to minimize the probability of failure due to environmental or external hazards.”

- Criterion numbers 58 and 59 are not used.

3.1.2.6 Series VI - Fuel and reactivity control

- Criterion 60: Control of releases of radioactive materials to the environment

  The reactor design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

  Bases: Informed by 10 CFR 50 Appendix A

- Criterion 61: Fuel storage and handling and radioactivity control

  The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of safety related SSCs, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a passive residual heat removal capability having reliability and testability that reflects safety related temperature control, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.

  Bases: Informed by RG 1.232, ARDC.

- Criterion 62: Prevention of criticality in fuel storage and handling

  Criticality in the fuel storage and handling systems shall be prevented by use of geometrically safe configurations.

  Bases: Informed by 10 CFR 50 Appendix A

- Criterion 63: Monitoring fuel and waste storage

  Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of temperature control capability and excessive radiation levels and (2) to initiate appropriate safety actions. Decay heat removal of fuel salt storage shall be passive.

  Bases: Informed by 10 CFR 50 Appendix A

- Criterion 64: Monitoring radioactivity releases

  Means shall be provided for monitoring the containment atmosphere, effluent discharge paths, and site environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. Radioactivity present within the reactor enclosure atmosphere during normal and accident conditions is allowed to decay before being vented to atmosphere.

  Bases: Informed by RG 1.232, ARDC.

- Criterion numbers 65-69 are not used.

3.1.2.7 Series VII - Salt systems and control

- Criterion 70: Reactor coolant (salt) purity control

  Any reactor coolant system whose moving fluids may become activated shall be designed with sufficient margin to assure that its containment function is adequately maintained.

Bases: Molten salt reactors are vulnerable to coolant activation because of their proximity to the neutron field. This criterion addresses the need to maintain containment of the activated coolant.

- Criterion 71: Fuel salt composition control

  Systems shall be provided as necessary to maintain the composition of the fuel salt within specified limits. These limits shall be based on the ability of the fuel salt to perform its safety functions.

  Bases: This criterion reflects the role of the fuel salt in the overall facility safety. Fuel salt is an essential element of providing adequate containment, heat removal, and reactivity control. Fuel salt properties are determined by its composition, which must be maintained within acceptable limits. Acceptable limits will be defined in the technical specifications.

- Criterion 72: Fuel salt temperature control systems

  Heating systems shall be provided as necessary for safety related SSCs, which contain or could be required to contain salt. These heating systems and their controls shall be appropriately designed to ensure that the temperature distribution and rate of change of temperature in systems and components containing salt are maintained within design limits assuming a single failure.

  Basis: This criterion reflects the importance of preventing freezing of salt and thermally damaging fuel salt contacting containment layers. This criterion addresses the potential for overheating of safety related components by the electrical heating systems that interact with them. This criterion addresses components which must passively provide some thermal control (such as the RTMS) for the duration of a safety related function. (The RTMS passively assures the ability of fuel salt to drain to achieve shutdown during a loss of electric power.)

- Criterion 73: Reactor trip line plugging

  If plugging of any line necessary for a reactor trip could prevent accomplishing a safety function, then the line shall be monitored with adequate reliability. Deviation from technical specifications shall trigger shutdown and appropriate corrective actions.

  Bases: This criterion reflects the distinctive characteristics of molten salt reactor protection lines.

- Criterion 74: Fuel salt system interfaces

  Where the fuel salt boundary interfaces with a structure, system, or component containing fluid that if allowed to freely interact with the fuel salt would cause the loss of a safety function, the interface location shall be designed to ensure that the fuel salt is separated from the fluid by two redundant, passive barriers.

Bases: Informed by RG 1.232, SFR-DC 78. This criterion describes the safety function of interfaces.

3.1.3 Nuclear Regulatory Commission Guidance Documents

The NRC guidance documents considered in the preliminary design of the MSRR facility are identified within this section and are listed in Table 3.1-2. Specific detail on the applicable portions of these guidance documents will be discussed in the operating license application. The sections cited in this table describe the extent of usage of these guidance documents.

3.1.4 Codes and Standards Used by the MSRR

The technical guidance of consensus codes and standards will be followed for the design, construction, and operation of the MSRR. ANS 15.8-1995 will define the quality assurance requirements for the MSRR. Guidance in codes and standards related to ASME NQA-1 or 10 CFR Part 50 Appendix B quality programs may be replaced with appropriate guidance from ANS 15.8. Other deviations from the guidance in a code or standard will be identified and justified.

Requirements pertaining to active mechanical equipment within ASME QME-1 (2017) will be satisfied subject to provisions specified in Regulatory Guide 1.100 Revision 4 (2020). Use of ASME QME-1-2017 will include application of nonmandatory Appendix QR-A for seismic qualification and nonmandatory Appendix QR-B for qualification of nonmetallic parts.

The codes and standards to be used in the design, construction, and operation of the MSRR are listed below.

ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, 2013 Edition

ANSI/ANS 15.1, The Development of Technical Specifications for Research Reactors, 2007 (R2018) Edition

ANSI/ANS 15.4, Selection and Training of Personnel for Research Reactors, 2016 (R2021) Edition

ANSI/ANS 15.7, Research Reactor Site Evaluation, 1977 (R1986) Edition

ANSI/ANS 15.8, Quality Assurance Program Requirements for Research Reactors, 1995 (R2018) Edition

ANSI/ANS 15.11, Radiation Protection at Research Reactor Facilities, 2016 (R2021) Edition

ANSI/ANS 15.16, Emergency Planning for Research Reactors, 2015 (R2020) Edition

ANSI/ANS 20.2, Nuclear Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten Salt Reactor Nuclear Power Plants, 2021 Edition

ASCE 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary, 2000

ASCE 7-10, Minimum Design Loads for Buildings and Other Structures, 2013

ASCE 43, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, 2019

ASME B16.3, Malleable Iron Threaded Fittings: Classes 150 and 300, 2021 Edition

ASME B16.22, Wrought Copper and Copper Alloy Solder-Joint Pressure Fittings, 2018 Edition

ASME B16.26, Cast Copper Alloy Fittings for Flared Copper Tubes, 2018 Edition

ASME B16.34, Valves-Flanged, Threaded, and Welding End, 2020 Edition

ASME B31.3, Process Piping, 2020 Edition

ASME BPVC Section III, Rules for Construction of Nuclear Facility Components-Division 1-Subsection NE-Class MC Components, 2017 Edition

ASME BPVC Section III, Rules for Construction of Nuclear Facility Components-Division 5-High Temperature Reactors, 2017 Edition

ASME BPVC Section VIII, Rules for Construction of Pressure Vessels Division 1, 2021 Edition

ASME BPVC Section VIII, Rules for Construction of Pressure Vessels Division 2-Alternative Rules, 2021 Edition

ASME QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Facilities, 2017 Edition

ASTM A53/A 53M, Standard Specification for Pipe, Steel, Black and Hot-Dipped, Zinc-Coated, Welded and Seamless, 2022

ASTM A234/A 234M, Standard Specification for Piping Fittings of Wrought Carbon Steel and Alloy Steel for Moderate and High Temperature Service, 2019

ASTM A269, Standard Specification for Seamless and Welded Austenitic Stainless Steel Tubing for General Service, 2013

ASTM B32, Standard Specification for Solder Metal, 2020

ASTM B88, Standard Specification for Seamless Copper Water Tube, 2020

ASTM D698, Standard Test Methods for Laboratory Compaction Characteristics of Soil Using Standard Effort, 2021

ASTM E108, Standard Test Methods for Fire Tests of Roof Coverings, 2020

AWS A5.8, Specification for Filler Metals for Brazing and Braze Welding, 2019

CAP-88, Clean Air Act Assessment Package, 1988

DOE-STD-3014, Accident Analysis for Aircraft Crash into Hazardous Facilities, 2006

EPA Federal Guidance Report 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, 1988

EPA Federal Guidance Report 12, External Exposure to Radionuclides in Air, Water, and Soil, 1993

ICRP 23, Reference Man: Anatomical, Physiological and Metabolic Characteristics: Reference Manual, 1975

IEEE 344, Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations, 2013 Edition

International Building Code (2012) as adopted and amended by the City of Abilene, Texas

International Fire Code (2009) as adopted and amended by the City of Abilene, Texas

NFPA 13, Standard for the Installation of Sprinkler Systems, 2018 Edition

NFPA 45, Standard on Fire Protection for Laboratories Using Chemicals, 2019 Edition

NFPA 70, National Electrical Code, 2014 Edition

NFPA 72, National Fire Alarm and Signaling Code, 2022 Edition

NFPA 99, Health Care Facilities Code, 2018 Edition

NFPA 801, Standard for Fire Protection for Facilities Handling Radioactive Materials, 2020 Edition

Texas Administrative Code, Title 30, Part 1, Chapter 217, Design Criteria for Domestic Wastewater Systems

UL 790, Standard Test Methods for Fire Tests of Roof Coverings, 2022 Edition

3.1.5 References

3.1-1 American National Standards Institute/American Nuclear Society, “Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten Salt Reactor Nuclear Power Plants,” draft ANSI/ANS 20.2, 2021, LaGrange Park, IL.

Table 3.1-1 Cross Reference to Preliminary Safety Analysis Report Sections

Design Criteria	PSAR Section
DC 1, Quality standards and records	3.5.1, 4.3.2, 7.4.1, 7.5.1, 7.7.1
DC 2, Design bases for protection against natural phenomena	3.5.1, 4.2.2, 4.3.2, 7.4.1, 7.5.1, 7.7.1, 9.6.1
DC 3, Fire protection	3.5.1, 7.4.1, 7.5.1, 7.7.1, 9.3.2
DC 4, Environmental and dynamic effects design bases	3.5.1, 4.2.2, 4.3.2, 7.4.1, 7.5.1, 7.7.1, 9.6.1
DC 10, Reactor design	4.2.1, 4.3.2, 4.5.1, 5.2, 6.3, 7.3.1, 7.4.1, 7.5.1, 7.71.
DC 11, Reactor inherent protection	4.5.1
DC 12, Suppression of power oscillations	4.5.1
DC 13, Instrumentation and control	7.3.1, 7.4.1, 7.6.1, 7.7.1
DC 14, Reactor fuel salt boundary	4.3.2, 5.2
DC 15, Reactor fuel salt system design	5.2, 7.3.1, 7.4.1
DC 16, Containment design	4.2.1, 6.2.1, 6.2.2
DC 17, Electric power systems	7.3.1, 7.4.1, 7.5.1, 7.6.1, 7.7.1, 8.2.1
DC 18, Inspection and testing of electric power systems	8.2.1
DC 19, Control room	7.6.1
DC 20, Protection system functions	7.4.1, 7.5.1
DC 21, Protection system reliability and testability	7.4.1, 7.5.1
DC 22, Protection system independence	7.4.1, 7.5.1
DC 23, Protection system failure modes	4.2.2, 7.4.1, 7.5.1
DC 24, Separation of protection and control systems	7.3.1, 7.4.1, 7.5.1
DC 25, Protection system requirements for reactivity control malfunctions	7.4.1
DC 26, Reactor protection systems	4.2.2, 7.4.1
DC 27, Reactivity control system capability	4.2.2, 7.3.1
DC 28, Reactivity limits	7.3.1, 7.4.1
DC 29, Protection against anticipated operational occurrences	7.4.1, 7.5.1
DC 30, Quality of fuel salt boundary	4.3.2, 5.2
DC 31, Fracture prevention of fuel salt boundary	4.3.2, 5.2
DC 32, Inspection of fuel salt boundary	4.3.2
DC 34, Residual heat removal	4.6, 6.3
DC 35, Fuel salt cooling system	4.3.2, 4.6, 6.3
DC 38, Functional containment heat removal	6.3, 9.1.1, 9.7.1
DC 40, Testing of functional containment heat removal system	6.3, 9.7.1
DC 50, Functional containment design basis	6.2.1, 6.2.2
DC 51, Fracture prevention of functional containment boundaries	6.2.2
DC 52, Capability for functional containment leakage rate testing	6.2.1, 6.2.2

Table 3.1-1 Cross Reference to Preliminary Safety Analysis Report Sections (Continued)

Design Criteria	PSAR Section
DC 53, Provisions for functional containment testing and inspection	6.2.1, 6.2.2
DC 54, Piping systems penetrating functional containment	6.2.1, 6.2.2
DC 55, Radionuclide interfacing lines penetrating functional containment	6.2.1, 6.2.2
DC 56, Functional containment isolation	6.2.1, 6.2.2
DC 57, Closed system isolation valves	6.2.1, 6.2.2
DC 60, Control of releases of radioactive materials to the environment	9.1.1, 9.7.1
DC 61, Fuel storage and handling and radioactivity control	9.2.1, 9.6.1
DC 62, Prevention of criticality in fuel storage and handling	9.2.1
DC 63, Monitoring fuel and waste storage	7.7.1, 9.2.1
DC 64, Monitoring radioactivity releases	7.7.1
DC 70, Reactor coolant (salt) purity control	9.2.1
DC 71, Fuel Salt composition control	4.2.1, 9.2.1
DC 72, Fuel salt temperature control systems	6.2.4, 7.3.1
DC 73, Reactor trip line plugging	4.2.2, 9.6.1
DC 74, Fuel salt system interfaces	4.3.2, 9.6.1

Table 3.1-2 Cross Reference to Nuclear Regulatory Commission Guidance Documents

NRC Guidance	Title	SAR Section
Regulatory Guide 1.232	Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors	3.1.1
Regulatory Guide 1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	3.2.2
Regulatory Guide 1.221	Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants	2.3.1 3.2.3
Regulatory Guide 1.29	Seismic Design Classification for Nuclear Power Plants	3.4.2
Regulatory Guide 1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	3.4.2
Regulatory Guide 1.61	Damping Values for Seismic Design of Nuclear Power Plants	3.4.2
Regulatory Guide 1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants	3.5.2
Regulatory Guide 1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	13.1.1
Regulatory Guide 1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference In Safety-Related Instrumentation and Control Systems	3.5.2
Regulatory Guide 1.87	Acceptability of ASME Section III, Division 5, High Temperature Reactors	4.3.3
Regulatory Guide 2.5	Quality Assurance Program Requirements for Research and Test Reactors	12.9
Regulatory Guide 2.6	Emergency Planning for Research and Test Reactors	12.8
Regulatory Guide 4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	11.1
Regulatory Guide 5.59	Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance	12.8
Regulatory Guide 8.2	Administrative Practices in Radiation Surveys and Monitoring	11.1
Regulatory Guide 8.4	Personnel Monitoring Device - Direct-Reading Pocket Dosimeters	11.1
Regulatory Guide 8.7	Instructions for Recording and Reporting Occupational Radiation Exposure Data	11.1
Regulatory Guide 8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	11.1
Regulatory Guide 8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	11.1
Regulatory Guide 8.13	Instruction Concerning Prenatal Radiation Exposure	11.1
Regulatory Guide 8.25	Air Sampling in the Workplace	11.1
Regulatory Guide 8.29	Instruction Concerning Risks from Occupational Radiation Exposure	11.1

Table 3.1-2 Cross Reference to Nuclear Regulatory Commission Guidance Documents

NRC Guidance	Title	SAR Section
Regulatory Guide 8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses	11.1

3.2 Meteorological Damage

This section describes the approach used to translate design basis meteorological parameters into loads used in the design of safety-related SSCs. The design basis meteorological parameters are consistent with the findings of the site characterization analysis as described in Chapter 2.

The preliminary bases for the structural design of the MSRR facility is described in this section and the final bases will be provided in the Operating License application.

The design basis meteorological parameters applicable to the design include normal wind loads, high wind loads from tornados and hurricanes, and precipitation loads. The treatment of these loads is discussed in the following subsections.

3.2.1 Normal Wind Loads

The meteorological characterization of the MSRR facility site defined the normal and high wind in Section 2.3. This section describes the approach to translating the normal winds for the site into loads on the MSRR facility.

The safety-related SSCs for the reactor are located within the MSRR facility subterranean systems pit as discussed in Section 3.5. The design of the MSRR facility provides protection for safety-related SSCs against adverse effects from winds. The loading combinations from ASCE/SEI 7-10 Chapter 2, are used for commercial and nuclear nonsafety-related SSCs. The design basis normal wind loading conditions are discussed in the following subsections.

Wind loads affect both the wind-force-resisting structural elements on the MSRR facility that provide support and the stability for the overall structure and components and cladding. The components and cladding elements of the MSRR facility envelope are not required for structural integrity and are designed such that failure does not affect the systems required for safe shutdown and long-term decay heat removal.

3.2.1.1 Applicable Design Parameters

The local building code for the SERC facility references ASCE/SEI 7-10, which defines risk categories for structures and provides design basis normal wind velocities for each risk category. Risk Category IV is the most stringent and is selected as the design basis for the MSRR facility because, in accordance with the standard, the materials in the MSRR facility are categorized as “hazardous substances.” Using Figure 26.5-1B from ASCE/SEI 7-10, the MSRR facility is designed to withstand the basic wind velocity of 120 mph for Risk Category IV structures. This bounds the expected velocities for the SERC facility site in Abilene, Texas based on the maximum observed wind in Section 2.3.1.3.

For the design of the main wind-force-resisting structural elements, the wind speed is transformed into equivalent pressure consistent with ASCE/SEI 7-10, Section 27.3. For component and cladding design, the wind speed is transformed into equivalent pressure consistent with ASCE/SEI 7-10, Section 29.3 and Section 30.3, respectively.

The mean recurrence interval of the basic wind speed for Risk Category IV buildings is 1,700 years. Wind loads determined for Risk Category IV buildings in accordance with the mean recurrence interval from ASCE/SEI 7-10, Chapters 26 to 30 are more stringent than the 100-year return period wind speed (see Section 2.3.1.3). As shown in Figure 2.5-5, the site is open terrain with scattered obstructions having heights generally less than 30 ft (10 m). Based on the site characteristics, it is consistent with ASCE/SEI 7-10 to use exposure category C" to determine inputs for the computation of applied forces on structures as discussed in Section 3.2.1.2 and Section 3.2.1.3.

3.2.1.2 Determination of Applied Forces

In accordance with Equation 27.3-1 of ASCE/SEI 7-10, the velocity pressure is

$$q_z = 0.0256 K_z K_{zt} K_d V^2 (lb/ft^2) \qquad \text{Equation 3.2-1}$$

where,

q_z = velocity pressure at height (z),

K_z = velocity pressure exposure coefficient at height (z) as determined by Table 27.3-1 of ASCE/SEI 7-10 that corresponds to the height of the safety-related structure,

K_{zt} = topographic factor equal to 1.0 as determined by Section 26.8-2 of ASCE/SEI 7-10,

K_d = wind directionality factor of 0.85 for the Main Wind-Force Resisting System (MWFRS) and components and cladding as determined by Figure 26.6-1 of ASCE/SEI 7-10, and

V = basic wind speed (3-sec gust) of 120 mph as determined by Figure 26.5-1B of ASCE/SEI 7-10 for Category IV Buildings and Other Structures.

3.2.1.3 Application of Normal Wind Load to Design of Structures

The calculated velocity pressure as determined in Section 3.2.1.2 is applied in accordance with ASCE/SEI 7-10 to confirm the MSRR facility design provides protection of safety-related SSCs against the effects of normal wind loads. See Section 3.5 for further discussion of design features that address loads from natural phenomena on the MSRR facility.

3.2.2 Tornado Loading

The meteorological characterization of the MSRR facility site defined the normal- and high-wind characteristics for the MSRR facility site in Section 2.3. This section describes the approach to translating the characteristics of design basis tornadoes for the site into loads on the safety-related portion of the MSRR facility. Tornado characteristics include high wind speed, atmospheric pressure change, and tornado-generated missile impacts. The design-basis-tornado loading conditions are discussed in the following subsections.

3.2.2.1 Applicable Design Parameters

Abilene Christian University used RG 1.76, Revision 1 to determine characteristics of the design-basis tornado to be applied to safety-related portions of the MSRR facility systems pit. Based on the MSRR facility location, the parameters for Region I with a maximum wind speed of 230 mph and an 83 mbar pressure drop are applicable in Tables 1 and 2 of the RG. Also, ASCE/SEI 7-10 was used to determine the applied forces on the safety-related portions of the MSRR facility from tornadoes. As discussed in Section 2.3.1.6 the largest tornado reported in Abilene was an EF 3 (<165mph).

3.2.2.2 Determination of Applied Forces

In accordance with Equation 27.3-1 of ASCE/SEI 7-10 the velocity pressure, or design-basis high-wind speed, is determined using Equation 3.2-1 above

where,

K_d = wind directionality factor equal to 1.0, and

V = maximum tornado wind speed as determined by RG 1.76 is 230 mph.

The design basis atmospheric pressure change, or tornado differential pressure, is 83 mbar (1.2 psi) as determined by Table 1 of RG 1.76.

Finally, the procedure used for transforming the tornado-generated missile impact into an effective or equivalent static load on the MSRR facility structure is consistent with NUREG-0800, Section 3.5.3, Subsection II. Tornado-generated missile impact effects are based on the design missile spectrum from RG 1.76.

The above-grade structure of the research bay is designed to withstand a 120-mph wind load. All safety related SSCs necessary for safe shutdown and long-term decay heat removal are located below grade in the systems pit. The systems pit is covered with precast concrete panels (see Figure 3.4-1) to protect required safety-related SSCs from tornado missiles and debris from failure of the research bay structure. Final design details will be provided in the operating license application.

3.2.3 Hurricane Loading

The meteorological characterization of the MSRR facility site defined the normal- and high-wind characteristics for the MSRR facility site as described in Section 2.3. This section describes the approach to translating the characteristics of design-basis hurricanes for the site into loads on the safety-related portion of the MSRR facility. Hurricane characteristics include high wind speed and hurricane-generated missile impacts. The design basis hurricane loading conditions are discussed in the following subsections.

3.2.3.1 Applicable Design Parameters

Regulatory Guide 1.221, Revision 0, Figure 1, was used to determine applicable design parameters for hurricane loads on the MSRR facility and is consistent with Section 2.3.1.6. It provides wind speeds for the MSRR facility location that are consistent with the definitions used in ASCE/SEI 7-10. Because ASCE/SEI 7-10 is the code of record for the SERC facility's local building code, the method from ASCE/SEI 7-10 is used to determine the applied forces from hurricanes, using the wind speeds from RG 1.221.

3.2.3.2 Determination of Applied Force

As shown in Figure 2.3-7, the MSRR site is outside an exceedance probability of 10^{-7} for nominal 3-sec, 130-mile (209 km) per hour gust wind speeds and was estimated to have an exceedance probability of 10^{-7} for nominal 3-second, 120-mile (193 km) per hour gust speeds. However, for conservatism the velocity pressure was determined using the maximum hurricane wind speed (130 mph) in Equation 3.2-1 (above) from ASCE/SEI 7-10. The procedure used for transforming the hurricane-generated missile impact into an effective or equivalent static load on the applicable portions of the structure is consistent with NUREG-0800, Section 3.5.3, Subsection II. Hurricane-generated missile impact effects are based on the design missile spectrum from RG 1.221 Table 1.

3.2.4 Precipitation Loads

The meteorological characterization of the SERC facility site defined the precipitation characteristics (see Section 2.3). This section describes the approach to translating the characteristics of design basis precipitation for the site into loads on the MSRR facility. Precipitation categories include rain, snow, and ice. The exterior shell of the MSRR facility over the research bay has a low-slope roof (1/4 in.: 12 in.) so loads due to rain accumulation are not considered a structural load in the structural design. Similarly, because of the lack of rain accumulation, ice load is anticipated to be minimal and is enveloped by the snow load. The design-basis-precipitation loading conditions are discussed in the following subsections. However, failure of the MSRR facility does not affect the ability to achieve safe shutdown or long-term passive decay heat removal because flooding of the systems pit does not affect the ability to achieve safe shutdown or to provide long term passive decay heat removal.

More detailed descriptions of the precipitation loads affecting the SERC and the MSRR facility based on the final design will be described in the Operating License application.

3.2.4.1 Applicable Design Parameters

Based on Risk Category IV characterization (see Section 3.2.1.1) and site location, Chapter 1 and Chapter 7 of ASCE/SEI 7-10 provide snow load design parameters to be applied to the MSRR facility which are consistent with maximum historical values in Section 2.3.

However, failure of the MSRR facility does not affect the ability to achieve safe shutdown or long-term passive decay heat removal because all safety-related SSCs needed for safe shutdown and long-term decay head removal are located in the systems pit, which is protected from failure of the above-grade structures.

3.2.4.2 Determination of Applied Forces

The low-slope roof snow load is calculated by Equation 3.2-2 as derived from ASCE/SEI 7-10, Sections 7.3 and 7.4.

$$p_s = 0.7 C_s C_e C_t I_s p_g \qquad \text{Equation 3.2-2}$$

where,

p_s = low-slope roof snow load (psf),

C_s = roof slope factor as determined by Sections 7.4.1 through 7.4.4 of ASCE/SEI 7-10 corresponding to the geometry of the roof,

C_e = exposure factor as determined by Table 7-2 of ASCE/SEI 7-10 equal to 1.0,

C_t = thermal factor as determined by Table 7-3 of ASCE/SEI 7-10 equal to 1.0,

I_s = importance factor as determined by Tables 1.5-1 and 1.5-2 of ASCE/SEI 7-10 equal to 1.2, and

p_g = ground snow load as set forth in Figure 7-1 of ASCE/SEI 7-10 equal to 10 psf.

Unbalanced snow loads on the ceiling of the MSRR facility are determined in accordance with Section 7.6 of ASCE/SEI 7-10 while design snow drift loads are determined in accordance with Section 7.7.

3.3 Water Damage

This section describes the approach to establishing loads on the MSRR facility from postulated internal and external flooding events.

3.3.1 Flood Protection

3.3.1.1 Conformance with Design Criterion 2 for Internal and External Flooding

Postulated internal flooding events consider the flow rates and quantities of water from sources inside the MSRR facility. In the probable maximum flood (PMF) event, the portion of the structure (subterranean systems pit) below grade could be subjected to grade-level hydrologic loads. These two flood conditions are evaluated in the following sections.

The following sections describe how the internal and external flooding design basis for the MSRR facility housing safety-related SSCs provides reasonable assurance that potential water damage does not preclude safety-related SSCs from performing their safety-related functions.

3.3.1.1.1 External Flood Design Features

The meteorological characterization from Section 2.3 provides precipitation data for Abilene, Texas. The SERC facility is a passively dry site with respect to external flooding hazards. Grading and drainage on the SERC site preclude water accumulation around the MSRR facility and precipitation loads from affecting the MSRR facility. The site is located outside and 25 ft above the Federal Emergency Management Agency 500-year flood zone and outside inundation areas affected by potential dam failures (see Section 2.4.1). There are no external flood loads on the above-ground portion of the MSRR facility. The reactor system mechanical SSCs are located in the systems pit, which is 25 ft (7.6 m) below grade. The below-grade systems pit has been designed to withstand buoyant forces associated with ground-water and grade-level flooding. The SERC facility design offsets the pressure and potential buoyant forces on the systems pit. The systems pit is a reinforced concrete structure designed to meet ACI 349-2013 [Reference 3.3-1] with the following features to prevent ground water intrusion:

- Water stops are provided in construction joints below grade level.
- Exposed external surfaces have waterproof coating.

The MSRR facility is designed such that safety-related SSCs are not affected by the postulated external flood event; therefore, DC 2 is met. Additionally, flooding the systems pit, while undesirable from an operations perspective, does not prevent safe shutdown of the reactor or long-term passive decay heat removal.

Specific grading and drainage features will be part of the final design and will be described in the application for an Operating License.

3.3.1.1.2 Internal Flood Design Features

Internal flooding in the MSRR facility housing safety-related SSCs has two potential sources: fire protection water and potable water systems located in the SERC.

The fire protection system is the primary source of concern and limiting condition for potentially flooding the systems pit. The fire protection system implements NFPA 801 [Reference 3.3-2]. The final design includes a trench around the north, east, and west sides of the systems pit to minimize fire protection water drainage into the MSRR facility systems pit that houses the reactor, FHS, and radioactive waste handling system. The MSRR safety-related SSCs are protected to prevent interaction between water and salts by enclosures, steel liners, catch pans, troughs, or similar design solutions. Safety-related SSCs that are vulnerable to water damage from internal floods are elevated above the systems pit floor as an added measure of protection. Additionally, flooding the systems pit does not prevent safe shutdown or long-term passive decay heat removal.

There is no high-energy piping in the safety-related portion of the MSRR facility systems pit; therefore, a high energy break is not considered.

The MSRR facility is designed such that safety-related SSCs are not affected by the postulated internal flood event; therefore, DC 2 is met for internal flood events.

3.3.2 References

3.3-1 American Concrete Institute, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” ACI 349, Farmington Hills, MI, 2013.

3.3-2 National Fire Protection Association, “Standard for Fire Protection for Facilities Handling Radioactive Materials,” NFPA 801, Quincy, MA, January 2020.

3.4 Seismic Damage

This section discusses the design and design bases of SSCs required to maintain function in the event of an earthquake at the MSRR facility. The MSRR facility is designed such that there is reasonable assurance an earthquake does not preclude the safe shutdown of the reactor and long-term decay heat removal.

3.4.1 Existing Multiuse Facility

The SERC is an existing multiuse facility that houses the MSRR facility. The portions of the MSRR facility containing MSRR safety-related SCCs, the systems pit, are evaluated and analyzed to confirm it is constructed to withstand a postulated earthquake and is classified as Seismic Category I. This includes analyses and review of tables, plot plans, general arrangement drawings, structural drawings, and piping and instrumentation diagrams of the existing building along with construction and vendor oversight records to the extent necessary to establish the seismic qualification of the portion of the structure housing safety-related SSCs (the systems pit). Where portions of an MSRR system are classified as Seismic Category I, the boundary limits of that portion of the SSCs designed to Seismic Category I provisions are reviewed against the design of the existing SERC facility. The review includes support systems piping and instrumentation diagrams, and for fluid systems that are partially Seismic Category I or are Seismic Category II because of location in the existing MSRR facility, the Seismic Category I portion of the system extends to the first seismic restraint beyond the isolation valves that isolate the part that is Seismic Category I. At the physical interface between seismic and non-seismic Category I piping systems, the Seismic Category I dynamic analysis is extended to either the first anchor point in the non-seismic system or to a sufficient distance into the non-seismic system so as not to degrade the validity of the Seismic Category I analysis. Those interfaces and seismic classifications are clearly identified on the final arrangement drawings of the MSRR facility.

The layout and cross sections for the research bay and systems pit are shown in Figure 3.4-1, Figure 3.4-2, and Figure 3.4-3.

3.4.2 Seismic Design for Safety-Related Structures, Systems, and Components

3.4.2.1 Seismic Design Criteria

Seismic performance criteria are described in Section 3.2(2) of ANSI/ANS 15.7 [Reference 3.4-1].

The SSCs are designed to satisfy the general seismic criteria to withstand the effects of natural phenomena (e. g., earthquakes, tornadoes, hurricanes, floods) without loss of capability to perform their safety functions. The methodology classifies SSCs into three categories: Seismic Category I, Seismic Category II, and non-seismic.

Seismic Category I applies to both functionality and integrity and is defined as SSCs that are designed and built to withstand the safe-shutdown earthquake (SSE) stresses for the MSRR site. Seismic Category II SSCs are located in the proximity of Seismic Category I SSCs, the failure of which during an SSE could result in loss of function of Seismic Category I SSCs.

- Seismic Category I applies to safety-related SSCs and to those SSCs required to support shutdown and maintain the MSRR in a safe shutdown condition.
- Seismic Category II applies to SSCs designed to prevent collapse under the SSE. Structures, systems, and components are classified as Seismic Category II to preclude structural failure during an SSE or where interaction with Seismic Category I items could degrade the functioning of a safety-related SSC to an unacceptable level or could result in an incapacitating injury to occupants of the main control room.
- Nonsafety-related (NSR) structures, systems, and components are those classified as Seismic Category II or are not classified.

Systems and components required for safe shutdown, including their foundations and supports, are designated as Seismic Category I and designed to withstand the effects of the SSE and remain functional. In addition, systems other than radioactive waste management systems, that contain, or may contain, radioactive material and whose postulated failure would result in potential offsite whole body (or equivalent) doses that are more than 0.001 Sv (0.1 rem) limit in 10 CFR Part 20, are also classified as Seismic Category I. By designing the SSCs, informed by the guidance in RG 1.29, Revision 6, to withstand the effects of an SSE, a designed-in safety margin is provided for bringing the reactor to a safe shutdown condition, while also reducing potential offsite doses from seismic events.

3.4.2.2 Design Response Spectra

The SERC was designed to ASCE 7-10 using a two-point Design Response Spectra. As noted in Section 2.5.5 Vibratory Ground Motion, the seismic hazard at the site has been refined since construction using the MPRS methodology and revised codes. The MPRS methodology includes higher frequency content. The Design Response Spectrum should envelope hazards from both USGS seismic models as described in Section 2.5.5.

The Horizontal Safe Shutdown Earthquake (SSE) for the MSRR, and its safety-related foundations, is taken as a deterministic RG 1.60 Revision 2 horizontal spectra shape anchored at the ZPA of 0.08g. The SSE is assumed to act at the top of the trench basemat. The selection of this as the SSE is conservative for the following reasons:

1. The RG 1.60 horizontal spectral shape was selected due to previous NRC endorsement, not indexed to a specific soil type, with its frequency content sufficiently broad to cover all soil types.

2. A PGA of 0.08g was arbitrarily selected to ensure the SSE spectrum bounds the 7-10 and 7-22 MCER spectra. In addition, 0.08g bounds the PGA predicted by the USGS models.
3. The SSE bounds the Design Response Spectrum (DRS) constructed per ASCE 43-19 SDC-3, when the USGS 2018 UHRS hazard data is applied as input.
4. The design spectrums developed in accordance with ASCE 7 and ASCE 43 include adjustments for site class C geotechnical conditions.

The Vertical SSE is taken as the RG 1.60 Revision 2 vertical spectra shape anchored at 0.08g.

1. The RG 1.60 vertical spectral shape was selected due to previous NRC endorsement, not indexed to a specific soil type, with its frequency content sufficiently broad to cover all soil types.
2. Selection of a vertical PGA equivalent to the horizontal PGA is conservative, given a lack of USGS data pertaining to vertical MPDS.

The SERC superstructure is non-safety related and its design basis will remain ASCE 7-10. The increased seismic hazard from ASCE 7-22 is not significant for the SERC superstructure as the building fundamental frequency was estimated to be 0.62 seconds (1.6 Hz), which is not within the region of exceedance identified in Section 2.5.5.

The MSRR SSE is quantified in Table 3.4-2 and Figures 3.4-4 and 3.4-5.

Structural damping is confirmed to follow as closely as practical the more limiting recommendations of RG 1.61, Revision 2, which range from about 3 to 7 percent. Any deviations will be justified and described as part of the Operating License application.

3.4.2.3 Soil-Structure Interaction and Dynamic Soil Pressures

The SERC was built for site class C under ASCE 7-10. As described in Chapter 2, the SERC is supported on a foundation system on stiff competent soils. The site is classified as Site Class C prescribed in ASCE/SEI 7-10, Table 20.3-1. The typical shear wave velocities for the soils present at the site are 1,200 to 2,500 ft/sec. As described in ASCE/SEI 7-10, typical practice is to define competent soil as having a shear wave velocity greater than 1,000 ft/sec. The analysis of the MSRR facility to the SSE includes the effects of soil-structure interaction. Dynamic soil pressures are determined with reference to ASCE 4 [Reference 3.4-2] Section 3.5.3.2 applied to the earth retaining walls in the MSRR facility research bay and will be provided as part of the Operating License application.

3.4.2.4 Operating Basis Earthquake

For preliminary design, the operating basis earthquake (OBE) was selected to be one-third the SSE defined previously based on RG 1.61 such that explicit design and analysis of the MSRR facility for the OBE ground-motion would not be required. An OBE is not part of the MSRR facility design.

3.4.2.5 Method of Analysis

The effect of loads other than earthquake-induced (seismic) loads is determined by static analysis methods in accordance with ASCE/SEI 7-10 and the fundamental principles of engineering. The equivalent-static and dynamic seismic analysis methods are discussed in Section 3.5.2.

3.4.3 Seismic Instrumentation

The MSRR facility is located in an area of very low seismicity and is conservatively designed for response to seismic events. In the event an earthquake is observed at the MSRR, the reactor is shut down manually and systems are evaluated based on available data.

3.4.4 References

3.4-1 American National Standards Institute/American Nuclear Society, “Research Reactor Site Evaluation,” ANSI/ANS 15.7, La Grange Park, IL, 1977 (R1986).

3.4-2 American Society of Civil Engineers, “Seismic Analysis of Safety-Related Nuclear Structures and Commentary,” ASCE 4-98, Reston, VA, January 2000.

3.4-3 American Society of Civil Engineers, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” ASCE 43, Reston, VA, January 2019.

Table 3.4-1 Safety and Seismic Classification of Structures, Systems, and Components

System Name: Components	Safety Related (SR) and Non-Safety Related (NSR) Classification and Function:	Seismic Classification
MSRR HVAC	NSR. MSRR HVAC does not play a role in accident consequence as it is not operated during an accident.	C-II
Portions of the SERC foundation	SR. Those portions of the SERC which provide structural support for SR SSCs are considered SR.	C-I
Fuel Handling System: Fuel salt purification and storage vessel as well as piping to the reactor enclosure, fuel storage enclosure	SR. Retain radionuclides under normal conditions (the salt bearing systems themselves) and accident conditions (enclosures around the salt bearing components).	C-I
Fuel Handling System: Shipping and receiving equipment	TBD. Equipment will be supplied by DOE and meet appropriate qualifications.	TBD
Fuel Salt Chemical Management System	NSR. Fuel salt samples and coupons are removed from the reactor system and analyzed to understand corrosion of the reactor system. Beryllium may be added to maintain the redox potential in an acceptable range. Penetrations of the system must be designed in accordance with functional containment DC. System functions are NSR, but these systems are integrated into the reactor system and ESFs. These systems must be designed so that they do not inhibit a SR function. Sections 4.2.1.6 and 4.3.11 of the PSAR describe the role of fuel salt sampling and coupons to monitor redox and corrosivity. This is described as a surveillance. Section 14.3.1 also describes the necessity of fuel salt chemical management.	C-I (due to interfacing with the reactor and ESFs)

Table 3.4-1 Safety and Seismic Classification of Structures, Systems, and Components (Continued)

System Name: Components	Safety Related (SR) and Non-Safety Related (NSR) Classification and Function:	Seismic Classification
Reactor System	SR. The reactor system boundary constitutes a fission product barrier under normal and accident conditions (except the MHA). Corrosion, temperature, and pressure must be controlled. The reactor system must be structurally supported by the reactor enclosure. The reactor pump seal is considered NSR as the reactor pump is enclosed in a SR leak tight pressure boundary which contains seal, motor, and bearings.	C-I
Graphite core	NSR. The graphite core cannot impair a safety function. This will be demonstrated in subsequent analysis in the FSAR.	C-II
Core support	SR. Graphite movement resulting in damage to reactor vessel shall be prevented. The core support structure shall be designed in such a way that a potential failure does not result in impairment of a safety function.	C-I
Primary Heat Removal System: Components which interface with fuel salt	SR. The HX tubes constitute fission product barriers and must retain radionuclides. Corrosion of the HX tubes from the coolant side must be controlled.	C-I
Primary Heat Removal System: Components which contain only coolant salt.	NSR. Coolant salt spills will not result in a radionuclide release exceeding 10 mrem.	C-II
Primary Heat Removal (PHR): pumps, fans, blowers,	NSR. Loss of PHR functionality does not result in the failure of a fission product barrier as demonstrated in Ch 13.	C-II
Auxiliary Heat Removal: Louvers, controllers, temperature monitors	SR. Temperature of the reactor cell triggers shutdown. The louvers controlling air flow into and out of the cell must isolate under shutdown. This is considered a part of ESFAS.	C-I
Auxiliary Heat Removal: Fans and blowers	NSR. Not required to operate during an accident but required for normal operation. Decay heat removal is entirely passive (See LONEP analysis in Ch 13).	C-II

Table 3.4-1 Safety and Seismic Classification of Structures, Systems, and Components (Continued)

System Name: Components	Safety Related (SR) and Non-Safety Related (NSR) Classification and Function:	Seismic Classification
Engineered Safety Features: Reactor Enclosure	SR. The reactor enclosure retains radionuclides under accident conditions. The reactor enclosure also provides structural support for the reactor system that must be maintained under accident conditions.	C-I
Engineered Safety Features: Reactor Cell	SR. The reactor cell limits the escape of radionuclides under accident conditions. The reactor cell also provides structural support for the reactor enclosure that must be maintained under accident conditions.	C-I
Engineered Safety Features: RTMS	SR. The RTMS contains the fuel salt in the event of an MHA. It also passively limits heat removal under normal and accident conditions.	C-I
Reactor Protection System	SR. This system shuts down the reactor.	C-I
Reactor Control System	NSR. This system does not play a role during accident scenarios.	C-II
Engineered Safety Feature Actuation Systems	SR. These systems must isolate radionuclide bearing components to retain radionuclides. Components of these systems include gas isolation valves and louvers to maintain functional containment.	C-I
Gas Management System: Reactor	SR. The gas management components in the reactor enclosure are a fission product barrier and must retain integrity under normal and accident conditions (except the MHA).	C-I
Gas Management System: FHS	SR. Used fuel will contain radionuclides and the gas piping will provide a fission product barrier which penetrates the Fuel Salt Storage enclosure.	C-I
Gas Management System: Coolant	NSR. Loss of coolant does not result in radionuclide release exceeding 10 mrem.	C-II

Table 3.4-1 Safety and Seismic Classification of Structures, Systems, and Components (Continued)

System Name: Components	Safety Related (SR) and Non-Safety Related (NSR) Classification and Function:	Seismic Classification
Gas Management System: Off-gas piping and vessels	NSR. The off-gas system will contain gaseous fission products. Functionality of the off-gas system is NSR. Functional containment of radionuclides is provided by the enclosure which surrounds the components. The off-gas enclosure retains radionuclides under accident conditions.	C-II
Gas Management System: Off-gas enclosure	SR. Contains radionuclides in the event of a rupture of the off-gas system.	C-I
Gas Management System: H2 and HF	NSR. While these components do not contain radionuclides, they pose safety hazards due to combustion (H2) and aggressive acid (HF). Vessels are in gas safety cabinets, piping will be double walled, and there will be a HF detection system in the SERC bay.	C-I
Radiation Monitoring Systems	SR. While these systems do not function during an accident, these systems trigger shutdown and ESFAS.	C-I
Backup Electrical Power	NSR. Backup power, if available, will preserve scientific equipment.	C-II
Normal Electrical Power	NSR Reactor does not depend on electric power under accident conditions.	C-II
Fire detection and suppression	NSR. Failure of fire protection systems would not prevent a safe shutdown.	C-II
Biological shield	SR. Passive, provided by design of the reactor cell and internal shield. Radiation dose of operating reactor is reduced to safe levels by the biological shield. Internal shield reduces activation of components and the dose to components within the reactor enclosure.	C-I

Table 3.4-1 Safety and Seismic Classification of Structures, Systems, and Components (Continued)

System Name: Components	Safety Related (SR) and Non-Safety Related (NSR) Classification and Function:	Seismic Classification
Scientific Surveillance	NSR. Failure of the scientific surveillance layer does not impair safety functions. Scientific surveillance systems are not required to perform a safety function. Penetrations of the scientific surveillance layer through functional containment are addressed by the SR functional containment and associated design criteria. If scientific surveillance equipment interfaces with a SR system (such as the reactor) then that interface is SR and considered a part of the SR system.	C-II
Experimental system	NSR. Failure of an experimental system does not impair safety functions. Penetrations of experimental systems through functional containment are addressed by the SR functional containment and associated design criteria. If experimental systems equipment interfaces with a SR system, then that interface is SR and considered a part of the SR system. System performance is NSR.	C-I (due to being physically integrated with reactor and ESF)

Table 3.4-2 MSRR Horizontal and Vertical SSE

MSRR Horizontal SSE at 5% Damping		
Control Point	Frequency (Hz)	Acceleration (g)
A	33	0.080
B	9	0.209
C	2.5	0.25
D	0.25	0.038

MSRR Vertical SSE at 5% Damping		
Control Point	Frequency (Hz)	Acceleration (g)
A	33	0.080
B	9	0.209
C	3.5	0.238
D	0.25	0.025

Figure 3.4-1 Systems Pit Cross Section (Rebar Dimensions Nominal/Typical)

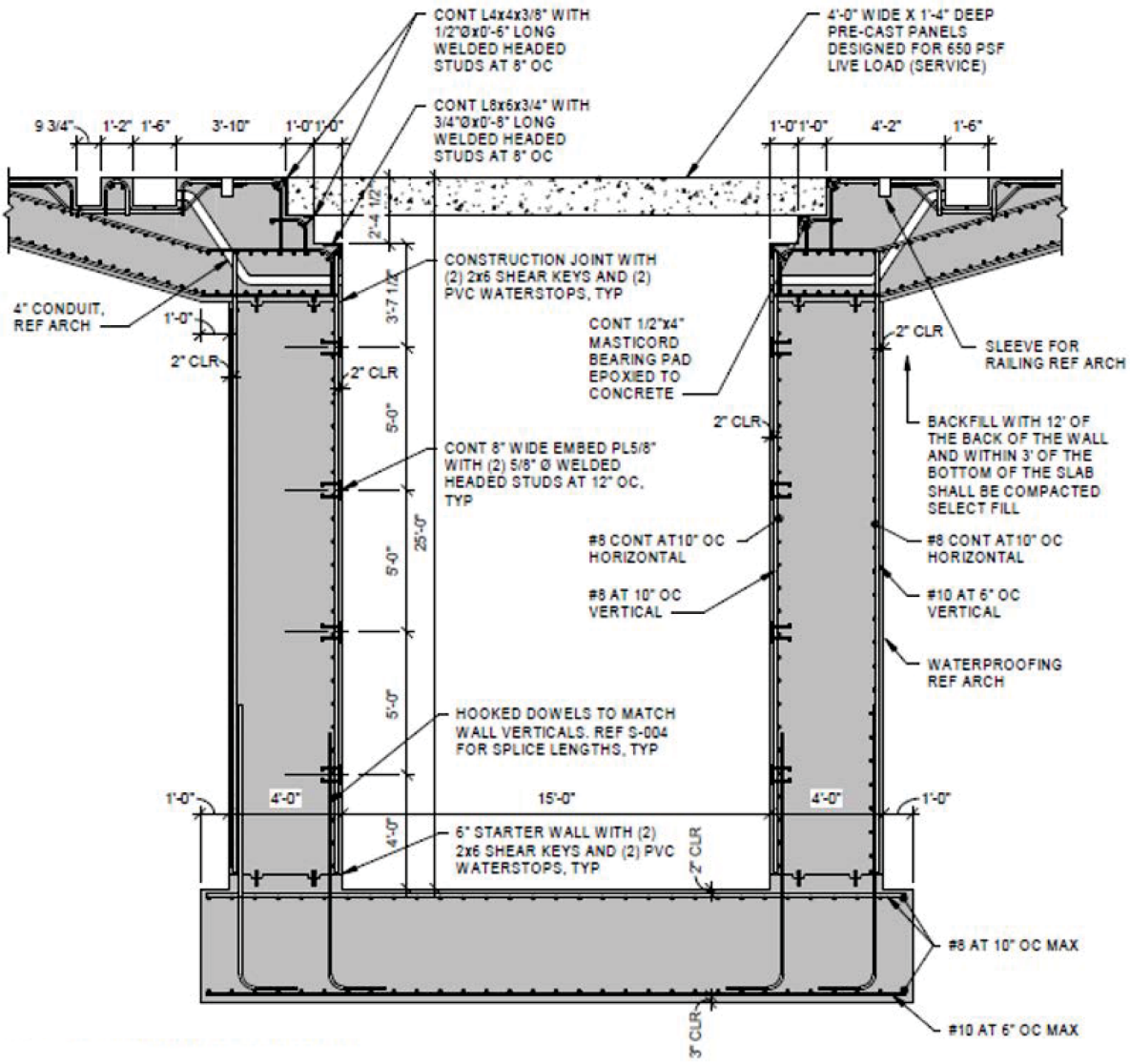


Figure 3.4-2 Research Bay Dimensions (Nominal/Typical)

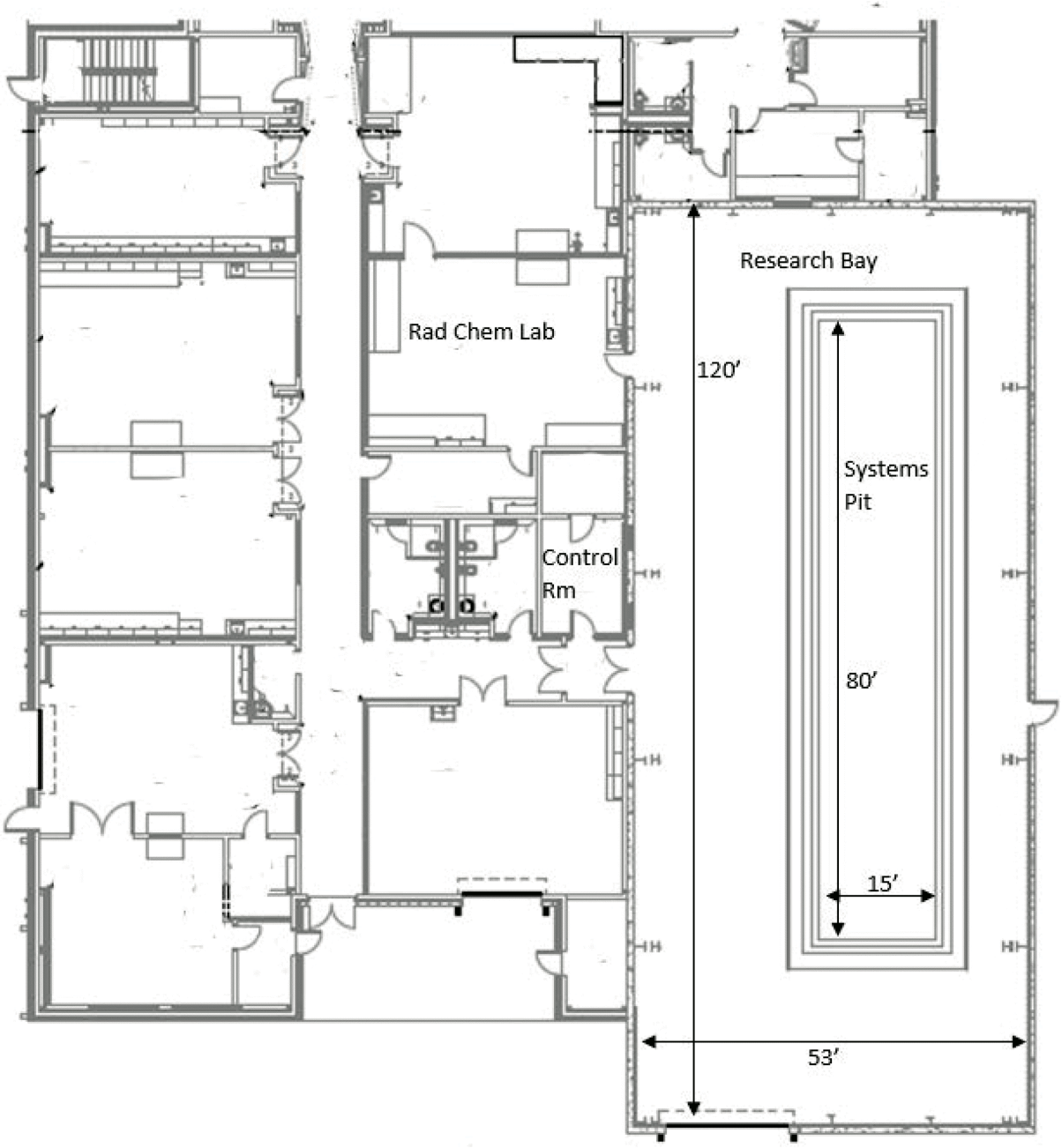


Figure 3.4-3 Research Bay Cross Section (Dimensions Nominal/Typical)

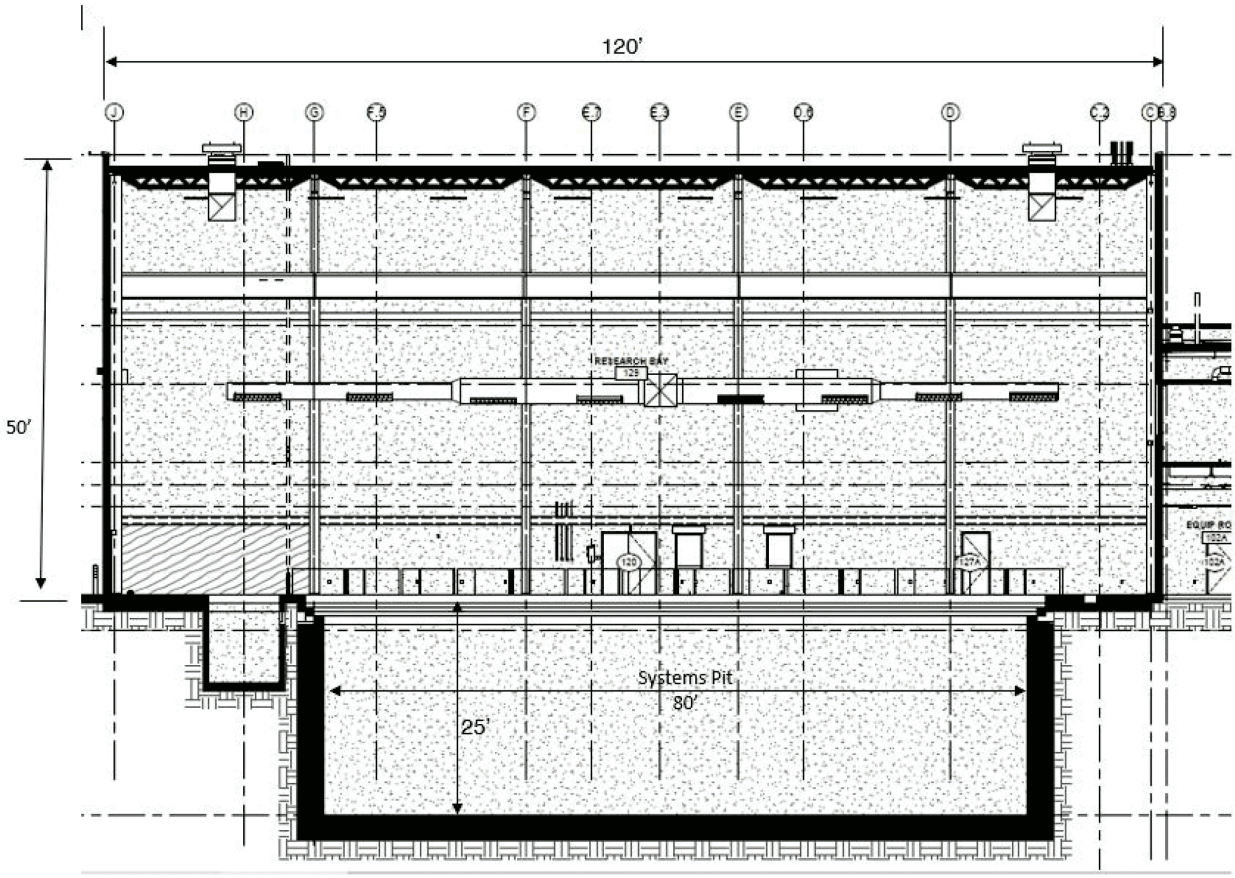


Figure 3.4-4 MSRR Horizontal Safe Shutdown Earthquake

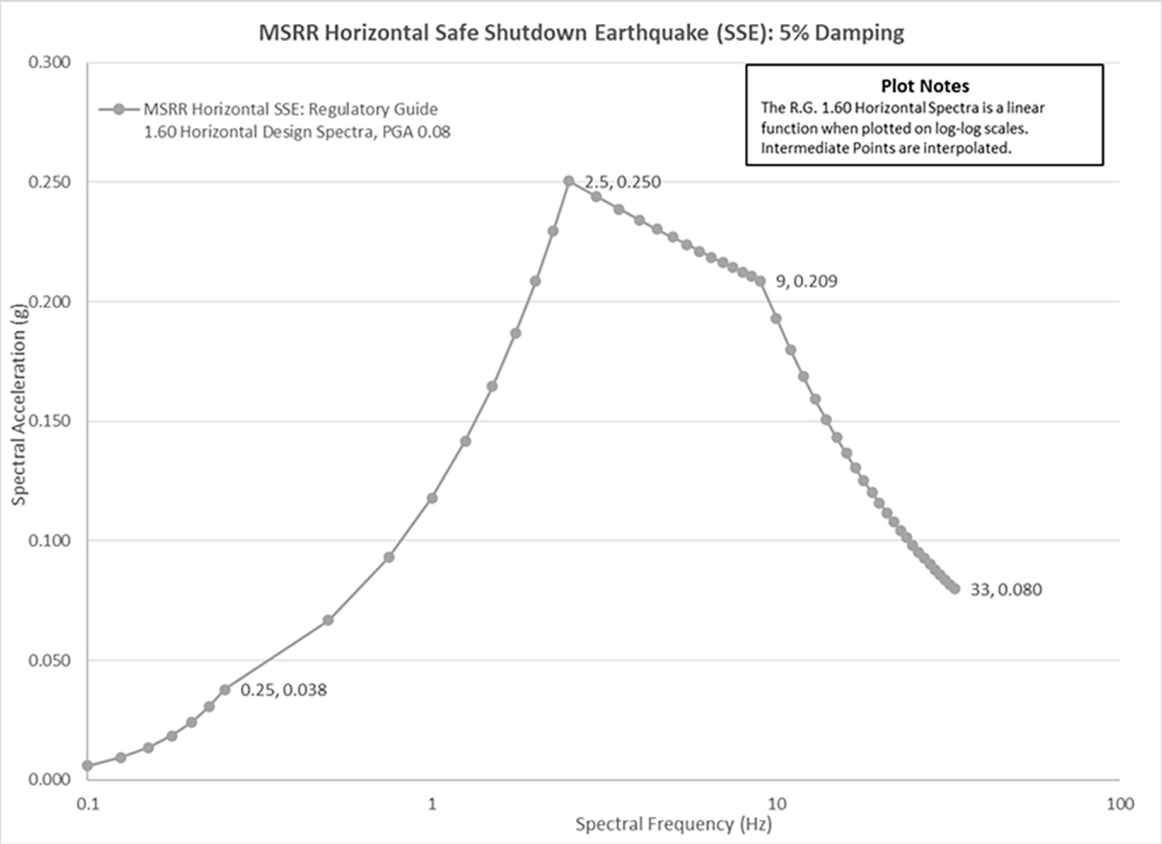
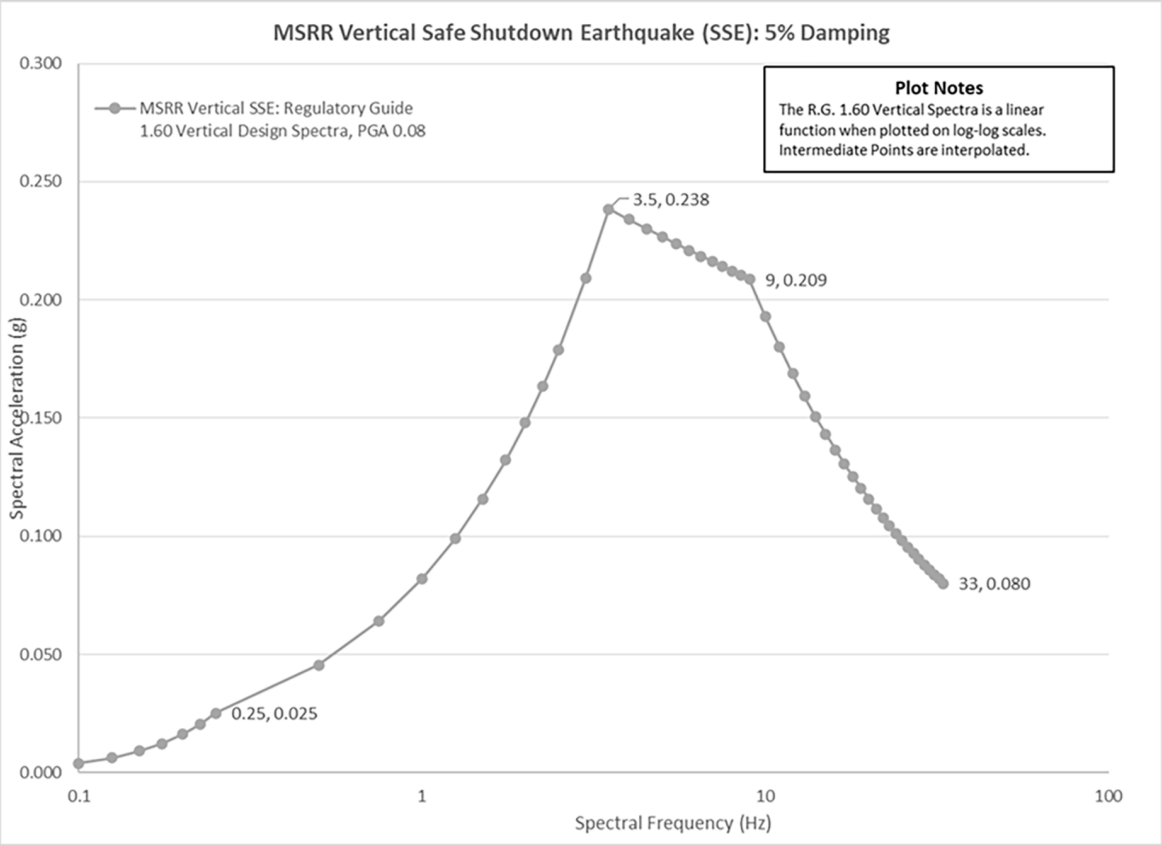


Figure 3.4-5 MSRR Vertical Safe Shutdown Earthquake



3.5 Systems and Components

This section describes the design bases for systems and components of the MSRR facility that are considered safety related because they perform safety functions during normal operations or are required to prevent or mitigate the consequences of abnormal operational transients or accidents.

3.5.1 General Design Basis Information

The SSCs relied on in the safety analysis to mitigate the consequences of postulated events serve one or more of the fundamental safety functions:

- Prevent uncontrolled release of radionuclides
- Control heat removal
- Control reactivity in the reactor core

Section 3.5.2 describes the safety classifications of SSCs based on performance of one of these fundamental safety functions. Details and design basis information related to other design criteria listed in Section 3.1.1 is discussed in the sections referenced in Table 3.1-1.

Consistent with DC 1, safety related SSCs are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they are identified and evaluated to determine their applicability, adequacy, and sufficiency, and are supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A Quality Assurance Program has been established and implemented to provide adequate assurance that these SSCs perform their safety functions satisfactorily. Appropriate records of the design, fabrication, erection, and testing of safety related SSCs are maintained by or under the control of ACU throughout the life of the MSRR facility in accordance with the approved Quality Assurance Program [Reference 12.13-6].

The MSRR will be installed and operated in an existing multiuse facility (the SERC) in accordance with 10 CFR 50.10(a)(2)(x). The SR SERC structures were designed, fabricated, erected, and tested under the quality assurance programs of Parkhill, Terracon, McClure, Nuclear Energy Consultants, and eHT (see 3.1.1). The SR SERC structures will be maintained by the ACU Quality Assurance Program.

Structures, systems, and components within a safety-related system are not necessarily safety related.

Consistent with DC 2, safety related SSCs are designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, without loss of capability to perform their safety functions. The design bases for these SSCs reflect: (1) appropriate consideration of the most severe natural phenomena historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which the historical

data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena and (3) the importance of safety functions to be performed.

Consistent with DC 3, safety related SSCs are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire-resistant materials are used wherever practical throughout the MSRR facility, particularly in locations with safety related SSCs. Fire detection and firefighting systems of appropriate capacity and capability are provided and designed to minimize adverse effects of fires on safety related SSCs. Firefighting systems are designed to ensure their rupture or inadvertent operation does not significantly impair the safety capability of SSCs.

Consistent with DC 4, safety related SSCs are designed to accommodate the effects of and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The low operating pressure of the reactor system eliminates the concern for dynamic effects, such as missiles, pipe whipping, and discharging fluids. Dynamic effects associated with the failure of a pressurized gas piping system will be accounted for in design to ensure that a breach in the gas management system does not breach the reactor system, fuel handling system, off-gas system, nor an Engineered Safety Feature.

Service levels and loads will be consistent with the requirements of ASME Section III Division 5. An example of how loads will be computed in the component design specification is shown in the table below. Loads that will be defined in component design specifications include, at a minimum, the following items:

1. Pressure Loads
2. Thermal Loads
3. Seismic Loads (SSE)
4. Piping Loads and Reactions
5. Nozzle loads
6. Bolting Loads due to Flanges, Heads, etc.
7. Fluid Loads
8. Fluid Transients
9. Support Interactions
10. Test Loads

Design / Service Level	Load Combination(s)[1]
Design Loads	$D + L + P_d + P_s$
Level A	$D + L + P_o + P_s + T_A + R_A$
Level B	$D + L + P_o + P_s + T_B + R_B$
Level C	$D + L + P_o + P_s + T_C + R_C$
Level D	$D + L + P_o + P_s + T_D + R_D + E$
Test Loads	$D + L + P_s + P_t + T_t + R_t$

Load Descriptions	
D	Dead Loads
L	Live Loads (Includes fluid and fluid transients)
P_d	Design Pressure
P_s	Static Head
P_o	Operating Pressure
P_t	Test Pressure Loads
T_A	Thermal Loads During Start-up, Normal Operation and Shutdown
T_B	Thermal Loads Due to Level B Loadings
T_C	Thermal Loads Due to Level C Loadings
T_D	Thermal Loads Due to Level D Loadings
T_t	Thermal Loads due to Testing Conditions
R_A	Design Mechanical Loads, Piping & Nozzle Reactions During Start-up, Normal Operation and Shutdown
R_B	Design Mechanical Loads, Piping & Nozzle Reactions Due to Level B Loadings
R_C	Design Mechanical Loads, Piping & Nozzle Reactions Due to Level C Loadings
R_D	Design Mechanical Loads, Piping & Nozzle Reactions Due to Level D Loadings
R_t	Design Mechanical Loads, Piping & Nozzle Reactions Due to Test Loadings
E	Design Basis Earthquake Loads

[1] Factors for individual loadings will be described in detail in the applicable Design Specifications

3.5.1.1 Prevention of Uncontrolled Release of Radionuclides

The MSRR is designed to prevent uncontrolled release of radionuclides. The fuel salt retains most of the fission and decay products. The design basis for the fuel salt is discussed in Chapter 4. The reactor system retains the fuel salt and in conjunction with the gas management system retains fission and decay product gases not retained in the fuel salt. The design bases for the reactor system are discussed in Chapter 4 and the gas management system design bases are discussed in Section 9.6.

Functional containment for the reactor system is provided by the reactor thermal management system (Section 6.2.4), the reactor cell (Section 6.2.1) and the reactor enclosure (Section 6.2.2). The fuel handling system and the coolant loop cooling system are in enclosures similar to the reactor enclosure and reactor cell to contain any leakage in those systems which will be fully described in the operating license application.

3.5.1.2 Control Heat Removal

The coolant loop described in Chapter 5 removes heat from the fuel salt through the heat exchanger and transports it to the atmosphere during reactor operation. The system is not credited for auxiliary heat removal or decay heat removal. It is located partially within the reactor enclosure and within the biological shield and partially within its separate, safety-related enclosure, the coolant salt and heat management enclosure. System components include the coolant side of the heat exchanger, the coolant pump, the radiator, the coolant loop drain tank, associated piping, and air handling equipment associated with heat removal from the radiator.

During reactor operation, the auxiliary heat removal system circulates air from the research bay through the cells in the systems pit (reactor cell, fuel handling cell, and secondary cooling cell) and out the research bay exhaust to cool the reactor enclosure (see Section 6.2.1). During accident conditions, the auxiliary heat removal system louvers isolate the reactor cell from the research bay and atmosphere.

During postulated events, the reactor vessel, internals, and drain tank passively transport heat from the fuel salt to the exterior of the reactor vessel or drain tank, to the reactor enclosure, and to the concrete bioshield and surrounding structure (described in Section 4.4). The concrete structure and surrounding earth have sufficient thermal mass to provide cooling indefinitely with a maximum concrete temperature below 100 degrees Celsius. (see Chapter 6).

3.5.1.3 Control of Reactivity in the Core

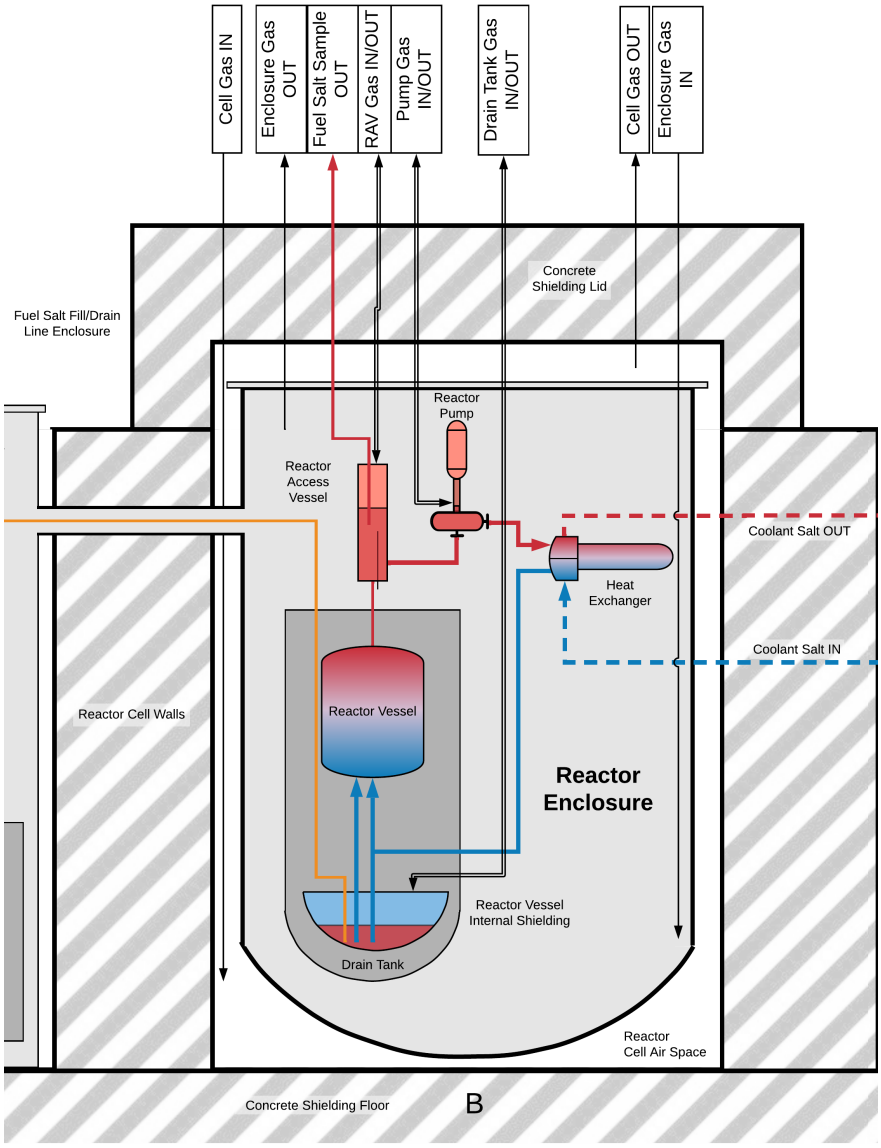
The reactivity control system is an integral system within the MSRR instrument and control (I&C) system designed to provide reactivity control for normal operations and in response to postulated events through the use of control rods, reactor pump flow, and secondary cooling. Instrumentation is provided (see Chapter 7) to monitor variables and systems over their anticipated ranges for normal operation and postulated events. This ensures adequate safety, including

those variables and systems that can affect the fission process and the integrity of the reactor core, the reactor fuel salt boundary, and functional containment. The I&C system includes appropriate controls to maintain variables and systems within prescribed operating ranges. The ability of the I&C system to perform its safety function does not rely on the performance of any auxiliary or distribution systems when normal power is available. When normal power is not available, the I&C system releases the control rods as described in Chapters 4 and 7, and the reactor protection system actuates shutting down the reactor and draining the fuel salt from the reactor system into the drain tank.

Other safety-related SSCs support the ability of the I&C system to control reactivity during postulated events:

- The reactor vessel and vessel internals (Chapter 4) maintain the geometry of the core to support control rods with their drive mechanisms and shutdown element insertion. The trip valves are located in the reactor system.
- The MSRR facility provides protection from the effects of natural phenomena on the I&C system and reactor system.

Figure 3.5-1 Reactor Cell Diagram



3.5.2 Classification of Structures, Systems, and Components

3.5.2.1 Safety Classification

Safety-related structures, systems and components means those structures, systems and components that are relied on to remain functional during and following postulated accidents identified through accident analysis (Chapter 13) to assure:

11. The integrity of the MSRR reactor system boundary;

12. The capability to shut down the MSRR and maintain it in a safe shutdown condition; or
13. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the applicable guideline exposures set forth in Chapter 13.

The terminology “anticipated operational occurrence” covers postulated events which do not threaten a fission product barrier as well as postulated events which result in minor releases of radionuclides of lesser consequence than the MHA.

3.5.2.2 Equipment Qualification

Safety-related SSCs are designed, fabricated, erected, and tested as required by the NEXT Lab Quality Assurance Program [Reference 12.13-6]. In addition, appropriate records of design, fabrication, erection, procurement, testing, and operation of SSCs are maintained throughout the life of the plant.

Safety-related systems and components are qualified under the ACU QA program. The qualification of each safety-related system or component demonstrates the ability to perform the associated safety function under environmental and dynamic service conditions in which they are required to function for the length of time the function is required.

RPS lines do not penetrate the reactor system enclosure. Safety related RPS valves are designed with diversity and redundancy and will be exercised as described in 4.2.2.1. ESFAS lines which penetrate enclosures will have valves and associated piping that will be equipped with design features to assure appropriate in-service testing in accordance with DC 54, unless containment function can be accomplished some other way. All safety related valves and associated piping will be tested before operation. Design features which could preclude the need for periodic leak testing include: double walled connection between safety related enclosures, a flow area that is so small that the leak is a fraction of the allowable enclosure leakage, or as described in the bases for DC 55. ESFAS valves that have direct connection to an enclosure atmosphere will be periodically leak tested. All ESFAS valves will be subject to in-service stroke testing. Safety related enclosures will be subject to integrated leak rate testing.

Additionally, nonsafety-related components and systems are qualified to withstand stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of the safety-related functions.

Safety-related SSCs susceptible to electro-magnetic interference (EMI) and/or radio frequency interference (RFI) are confined to the scope of the I&C system(s).

3.5.2.3 Seismic Qualification

3.5.2.3.1 Seismic Qualification by Analysis

Section 3.4 discusses SSCs and their classification for equipment qualification, and Section 3.5 discusses requirements for SSCs required to maintain their function in a postulated event. All safety-related SSCs required to mitigate a postulated event are located in the systems pit.

Safety-related systems and components are qualified to maintain their safety function during and after a postulated event depending on the function performed. The credited safety systems designed to function in a postulated event are described in Chapter 13. The specific safety function is used to define the preliminary ASCE 43 limit state used for guidance to qualify SSCs.

Seismic qualification will be accomplished through analysis, testing, or a combination of both. Acceptance criteria for the preliminary design are defined in IEEE 344-2013. Regulatory Guide 1.100 Revision 4, (Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants) states, “ASME QME-1-2017 is, in general, acceptable to the NRC staff for seismic qualification of (1) active mechanical equipment in new NPPs and (2) new or replacement active mechanical equipment in operating NPPs …” Seismic qualification of active mechanical equipment by analysis will follow the guidance in Section QR-A7100 (Qualification by Analysis).

Preliminary seismic qualification by analysis follows Section 8.2 of ASCE 43, 2019 Edition. Depending on the characteristics and complexities of the subsystem or equipment, qualification by analysis is accomplished by equivalent static or dynamic analysis methods. ASCE 43 provides certain limitations to qualification by analysis:

- Qualification of active electrical equipment by analysis is not performed.
- Qualification of active mechanical equipment by analysis may be permitted if the component is such that the functionality during an earthquake can be established and a margin of loss of functionality during an earthquake can be quantified.
- Qualification of active mechanical components by analysis shall be justified.

Seismic qualification by analysis is typically implemented for subsystems and equipment structural integrity-related capacities (e. g., anchorage, pressure boundary, pressure boundary rupture, serviceability deformations).

3.5.2.3.2 Seismic Qualification by Testing

Seismic qualification by testing includes tests and analyses to establish environmental qualification for conditions inclusive of normal and abnormal environments, and postulated seismic conditions where SSCs are installed.

Qualification by test is typically used for SSCs for which qualification by analysis is not permitted and for SSCs for which dynamic behaviors are not sufficiently understood to support qualification by analysis.

The test specimens are tested for environmental and seismic withstand capabilities in accordance with IEEE 344-2013 as informed by guidance in RG 1.100 Revision 4. Seismic qualification of active mechanical equipment by testing will follow the guidance in Section QR-A7200 (Qualification by Testing) of QME-1-2017.

3.5.2.4 Quality Classification

The quality classification for SSCs conforms to the requirements of the NEXT Lab Quality Assurance Program for the MSRR [Reference 12.13-6]. Safety-related SSCs are classified as Quality-Related, while non-safety-related SSCs are classified as Not Quality Related.

APPENDIX 3A CITY OF ABILENE, TX BUILDING PERMIT FOR SERC

City of Abilene, TX

555 Walnut St
Abilene, TX 79604

Ph: (325) 676-6273

New Building (C)

#21-002361

Project Description: A new science and engineering research facility for Abilene Christian University located on their Abilene, Texas campus, which includes public spaces (lobby, display and viewing areas), laboratories, fabrication shop, high-bay research space with bridge crane, faculty and research staff...

Issued on: 03/04/2022 at 11:55 AM by: Marissa Gonzalez

CITY OF ABILENE

ADDRESS
949 EN 16th St
Abilene, TX 79601

LEGAL
Acu Addn Blk Block A
Lot Lot 01

PERMIT HOLDER
Elizabeth Hollenbeck
Parkhill
(325) 673-8178

COLLABORATORS
- Brandon Young
  TLP/PSC
  (325) 673-8178
- Brent Bunkley
  Bunkley Electric
  (325) 677-4460
- Elizabeth Hollenbeck
  Parkhill
  (325) 673-8178
- Luke Knehans
  Linbeck Group, LLC
  (817) 889-7792
- Sean McNeill
  Batjer and Associates
  (325) 673-2556

OWNERS
-

OCCUPANCY INFORMATION

ICC - B Business - IIB
Square Footage: 27619.0000
Valuation: $15,600,000.00

INSPECTIONS 35

1. Building-Electrical Temporary Pole
2. Building- Plumbing Rough-In
3. Fire Access and Hydrants (C)
4. x1 Building Inspection
5. Electrical Ufer-Under slab rough / Ditch
6. Building Foundation (C)
7. Driveway Approach (C)
8. Fire Riser (C)
9. x2 Building Inspection
10. x3 Building Inspection
11. x4 Building Inspection
12. x5 Building Inspection
13. Building-Electrical Rough-In (C)
14. Mechanical Hood (C)
15. Building Permit - Mechanical Rough-In
16. Building Permit - Plumbing Top-Out
17. Building Permit - Plumbing Gas Rough
18. Building Framing (C)
19. Building Insulation (C)
20. Fire Sprinkler Inspection (C)
21. Electrical Above Ceiling (C)
22. Mechanical Above Ceiling (C)
23. Building- Electrical Meter Release
24. Building - Gas Meter Release
25. Dumpster Pad Pre-Pour Inspection
26. Fire Alarm Final (C)
27. STORMWATER FINAL
28. Engineering Final (C)
29. Planning & Zoning Final (C)
30. Solid Waste
31. Building-Electrical Final
32. Building Permit - Mechanical Final
33. Building Permit - Plumbing Final
34. Fire Final (C)
35. Building / Energy Final (C)

Page 1/3

MYGOV.US 21-002361, 03/04/2022 at 11:55 AM Issued by: Marissa Gonzalez

MYGOV.US City of Abilene, TX Phone: (325) 676-6273

INFORMATION FIELDS

Located within the Empowerment Zone?	NO
Plasticity Index	13-29
Declared Value	**$ 15,600,000.00**
TDLR - Project Registration #	TABS2021015539
Building - Type Of Work (Commercial)	**New Construction**
Building - CO Requirement	**Yes**
Building - Sq. Ft. (New)	27619
Building - Fire Sprinkler System	**New**
Minimum Elevation	1723.25
Located within a floodplain?	**NO**
Elevation Certificate needed?	**NO**
18" Above Gutter, Grade:	1719.85
Supplier	**City**
Water Meter Size	2"
Sanitary Sewer	**City**
GAS Service	ATMOS
Supplier	**AEP**
Structure (C) Type	**Commercial**
Business Name (if applicable)	Science and Engineering Research Center (ACU)
Square Feet - Total	**27619**
Development Review Meeting Attendance	**Accept**
Building Permit - Agree To Terms	**Yes**
Total length of sidewalk?	416
Occupant Load	**235**
Total Square Footage	**27619**
Square Feet - 1st Floor: Living	**27619**
Proposed Use	University Education
ITE Code	n/a
Sprinkler Required	**Yes**
Structure Setback: Front (curb)	30' MINIMUM

Page 2/3

MYGOV.US 21-002361, 03/04/2022 at 11:55 AM Issued by: Marissa Gonzalez

MYGOV.US City of Abilene, TX Phone: (325) 676-6273

INFORMATION FIELDS

Structure Setback: Front (property line)	10' min
Structure Setback: Rear	NA
Structure Setback: Side (N S E W)	NA
Structure Setback: Side (N S W E)	NA
Plan Review Comments	1. Follow direction of RAS as it pertains to Texas Accessibility Standards. 2. All other requirements to meet City of Abilene I-codes and NEC.

FEE	TOTAL	PAID	DUE
BLDNG - Plan Review Fee	$ 10,120.87	$ 10,120.87	$ 10,120.87
BLDNG - (C) New Building Permit Fee	$ 40,483.50	$ 40,483.50	$ 40,483.50
TOTALS	$ 50,604.37	$ 50,604.37	$ 0.00

The granting of this permit does not presume to give authority to violate or cancel the provisions of City, State, or other local laws regulating construction or the performance of construction. All provisions, laws, and ordinances governing this type of work shall be complied with, whether specified or not and shall be enforced at any and all times.

Page 3/3

MYGOV.US 21-002361, 03/04/2022 at 11:55 AM Issued by: Marissa Gonzalez

Chapter 4
Molten Salt Research Reactor Description

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 4 MOLTEN SALT RESEARCH REACTOR DESCRIPTION 4-1

4.1 Summary Description 4-1

4.1.1 Reactor System 4-2

4.2 Active Reactor Core 4-6

4.2.1 Reactor Fuel 4-6

4.2.2 Control Elements 4-10

4.2.3 Neutron Moderator and Reflector 4-13

4.2.4 Neutron Startup Source 4-14

4.2.5 Core Support Structure 4-14

4.3 Vessel 4-21

4.3.1 Description 4-21

4.3.2 Design Bases 4-21

4.3.3 Reactor System Structural Material 4-22

4.3.4 Fuel Salt Chemical Attack 4-22

4.3.5 Radiation Damage to Reactor System 4-23

4.3.6 Stainless Steel Oxidation and Creep 4-23

4.3.7 Pressure Effects 4-24

4.3.8 Thermal Design Limits 4-24

4.3.9 Reactor System Integrity Monitoring 4-24

4.3.10 Fuel Salt Leakage 4-24

4.3.11 Description of the Reactor Access Vessel 4-25

4.4 Biological Shield 4-28

4.4.1 Description of Biological Shield 4-28

4.4.2 Design Bases 4-28

4.4.3 Functional Requirements 4-28

4.4.4 Design Methods 4-29

4.4.5 Design Approach 4-29

4.4.6 Internal Shield 4-30

4.4.7 Biological Shield Inner Layer (Sacrificial Shield) 4-30

4.4.8 Biological Shield Outer Layer (Systems Pit Floor and Walls, Top Plug) 4-30

4.4.9 Additional Shielding 4-30

4.4.10 Structural Considerations 4-31

TABLE OF CONTENTS

4.4.11 Radiation Zones 4-31
4.4.12 Calculated Dose Rates and Other Relevant Shielding Parameters 4-31
4.4.13 Uncertainties in Shielding Results 4-32
4.4.14 Environmental Conditions 4-32
4.4.15 Soil Activation 4-32
4.4.16 Air Activation 4-32
4.5 Nuclear Design 4-35
4.5.1 Description of Nuclear Design 4-35
4.5.2 Normal Operating Conditions 4-37
4.5.3 Active Reactor Core Physics Parameters 4-40
4.5.4 Operating Limits 4-41
4.6 Thermal Hydraulic Design 4-51
4.6.1 Heat Removal Systems 4-51
4.6.2 Thermal-Hydraulic Methodology 4-52
4.7 Gas Management System 4-54
4.8 References 4-55

LIST OF TABLES

Table 4.1-1 Reactor Parameters 4-5
Table 4.2-1 Graphite Dimensions at 600 Degrees Celsius 4-20
Table 4.2-2 Parameters Describing the Grid Plate at 600 Degrees Celsius 4-20
Table 4.5-1 Reactivity Loss Stemming from the Delayed Neutron Precursors Outside the Reactor Vessel 4-50
Table 4.5-2 Reactivity Coefficients Calculated in MCNP with 1 Sigma Stochastic Uncertainty 4-50
Table 4.5-3 Reactivity Coefficients Calculated in SCALE 4-50

LIST OF FIGURES

Figure 4.1-1 Section of the Systems Pit with Major Components 4-3
Figure 4.1-2 Reactor System Components within the Reactor Enclosure 4-4
Figure 4.1-3 Reactor System Major Internals and Approximate Dimensions. 4-5
Figure 4.2-1 Deleted. 4-15
Figure 4.2-2 Locations of the Three Control Rod Thimbles Shown in Yellow 4-16
Figure 4.2-3 Single Hexagonal Graphite Block (Left) and Channels Formed by Lattice of Blocks (Right). 4-17
Figure 4.2-4 Hexagonal Lattice of Graphite Blocks Showing Channels at Block Corners 4-18
Figure 4.2-5 Lower Core Grid Plate 4-19
Figure 4.3-1 Reactor Access Vessel 4-27
Figure 4.4-1 Configuration and Thickness of Biological Shield Layers 4-33
Figure 4.4-2 Vertical Cuts (XZ and YZ) of Bay Building and Systems Pit 4-34
Figure 4.5-1 Variation in Pressure Coefficient of Reactivity with Respect to Void Fraction 4-46
Figure 4.5-2 Change in K_{eff} with Respect to Addition or Removal of UF_4 from Reactor 4-47
Figure 4.5-3 Loss of Reactivity Caused by Fuel Burnup 4-47
Figure 4.5-4 Axial Power Profile through the Entire Reactor Vessel 4-48
Figure 4.5-5 Radial Power Profile through Reactor Vessel Core Region. 4-49
Figure 4.6-1 Simplified Reactor (Red) and Coolant (Blue) Heat Removal Systems 4-53

CHAPTER 4 MOLTEN SALT RESEARCH REACTOR DESCRIPTION

This chapter provides principal features, operating characteristics, and parameters of the Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) to establish the design and functional characteristics of the reactor. This chapter provides the safety bases for the reactor system, reactor fuel, biological shield, nuclear design, heat removal systems, and reactor Gas Management System (GMS).

4.1 Summary Description

The MSRR facility is the NRC-licensed footprint within the Science and Engineering Research Center (SERC), a large multidisciplinary research and educational facility (see Chapter 2). The SERC contains laboratory spaces, classrooms, office spaces, and a large research bay. The MSRR facility is comprised of the research bay, research bay heating, ventilation, and air conditioning systems, research bay electrical power supply, control room, and radiochemistry lab. Figure 4.1-1 shows a cross section of the MSRR facility systems pit with major components.

Table 4.1-1 provides general reactor parameters.

The MSRR is a loop type 1 MWth molten fluoride salt reactor with uranium tetrafluoride dissolved in the salt operating at a maximum temperature of 650 degrees Celsius. The nominal fuel salt composition is (~67%)LiF-(~28%)BeF_2-(~5%)UF_4. The final composition is to be developed under the Nuclear Energy eXperimental Testing Laboratory (NEXT Lab) Quality Assurance Program with the involvement of the U. S. Department of Energy and reported in the Operating License application. The uranium is anticipated to be approximately 19.75 percent enriched in U-235 and the lithium enriched to be greater than 99.99 percent in ^{7}Li. A hexagonal lattice of graphite blocks with flow channels provides moderation for the fuel salt in the reactor vessel. Criticality is prevented by geometry and a lack of neutron moderation in other parts of the reactor loops and the Fuel Handling System (FHS).

The reactor loop consists of the reactor vessel, Reactor Access Vessel (RAV), reactor pump, and heat exchanger. Fuel salt exiting the reactor vessel flows into the RAV, circulated by the reactor pump. After the pump, fuel salt is pushed into the heat exchanger and then back to the reactor vessel through the cold leg.

The Reactor Protection System (RPS) uses a drain tank connected to the cold leg from the heat exchanger to the reactor vessel. The drain tank also has a separate, redundant, direct connection to the reactor vessel. Equalization of the cover gas pressure between the reactor access vessel and the drain tank is achieved by opening one of several valves to allow the fuel salt to passively drain out of the reactor vessel into the drain tank under the force of gravity. The drain tank is designed to hold the fuel salt in a highly subcritical configuration with passive cooling. Control Rods (CRs), located in the reactor vessel, provide reactivity control during normal operations. The CRs are not credited in any accident analysis. The drain tank also connects to the FHS and the GMS.

Thermal energy generated by fission is ultimately rejected to the atmosphere through a coolant loop (secondary) cooling system (see Section 5.2.2). The coolant salt ($67LiF\text{-}33BeF_2$) flows through a closed loop consisting of the tube side of the heat exchanger, the coolant pump, and the radiator. The coolant salt cools the fuel salt and transfers heat from the loop to the atmosphere through a forced-air radiator.

The MSRR is refueled by adding additional UF_4 directly to the fuel salt through the RAV.

The MSRR does not have traditional experimental systems; there are no beam ports or in-core experimental positions. Fuel salt and off-gas will be sampled from the RAV and coupons of various materials will be exposed to the fuel salt for monitoring and predicting corrosion of the stainless steel of the reactor system.

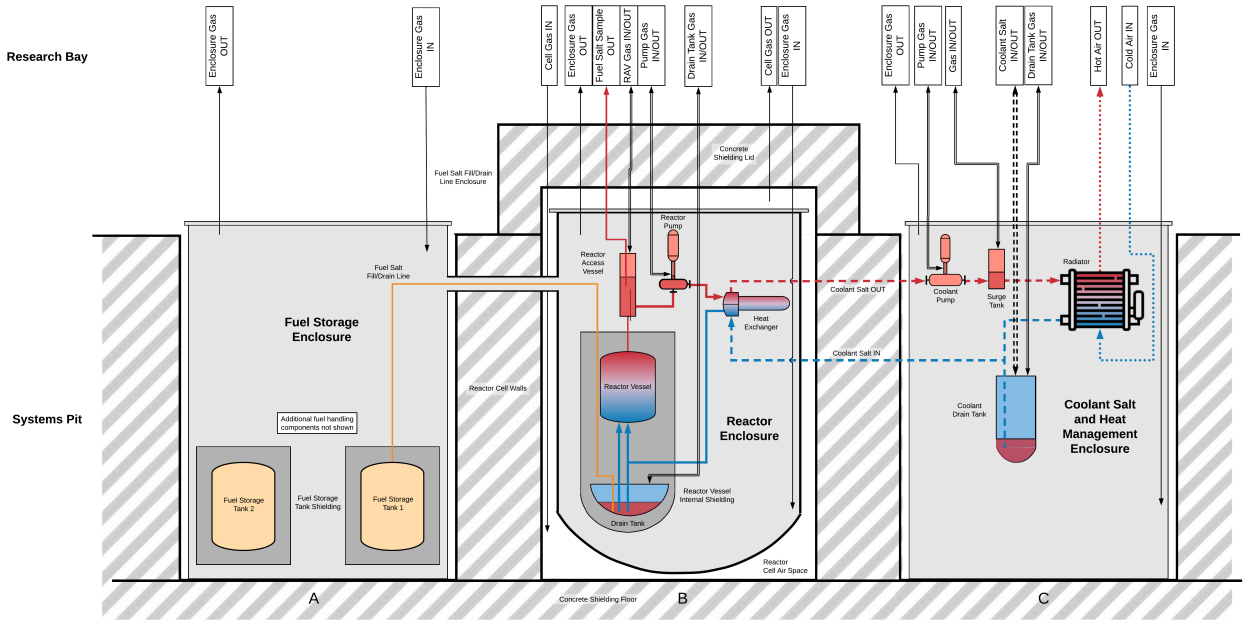
4.1.1 Reactor System

The reactor system is shown in Figure 4.1-2.

Figure 4.1-3 summarizes key dimensions of the reactor system. The reactor system is located entirely within the reactor enclosure. The reactor loop is composed of the reactor vessel, RAV, reactor pump, heat exchanger, and associated piping. The drain tank is connected to the loop through the cold leg and has a separate, redundant connection directly to the reactor vessel. Stainless steel (SS316H) which makes up the outer structure of the reactor system and fuel salts are fission product barriers. The drain tank is connected to the FHS; this connection is used to load and unload the reactor. The drain tank has two other connections at the top which are used to load and unload the reactor system. The RAV is directly above the reactor vessel. Fuel salt flows upwards through the reactor vessel, through the RAV, through the reactor pump, through the heat exchanger, and back into the bottom of the vessel. The reactor pump is a sump type with a gas shaft-seal located at the top of the system to allow easy maintenance. The reactor piping is nominal 2.5-in., [6.35 centimeter (cm)] schedule 80 stainless steel (SS316H) chosen to provide a large enough flow area to minimize pressure losses while also minimizing the amount of fuel out of the reactor vessel and providing sufficient thickness to allow high-temperature operation.

The method of moving fuel salts through the reactor affects the reactor shutdown and Reactor Protection Systems (RPS). Fuel salt is maintained in the reactor vessel by pneumatic pressure within the drain tank. Control rods located in control rod thimbles provide operational control. Control rod drives are located above the reactor shield within the reactor enclosure. Reactor shutdown is accomplished by draining the fuel salt into the drain tank. Draining of the reactor system is controlled by equalizing the pressure within the RAV and the drain tank. Pressure control is suitably redundant to ensure draining can occur any time required. The RPS includes the pressure control system of the reactor and drain tank.

Figure 4.1-1 Section of the Systems Pit with Major Components



The reactor system, FHS, and primary heat removal system are located in the systems pit.

Figure 4.1-2 Reactor System Components within the Reactor Enclosure

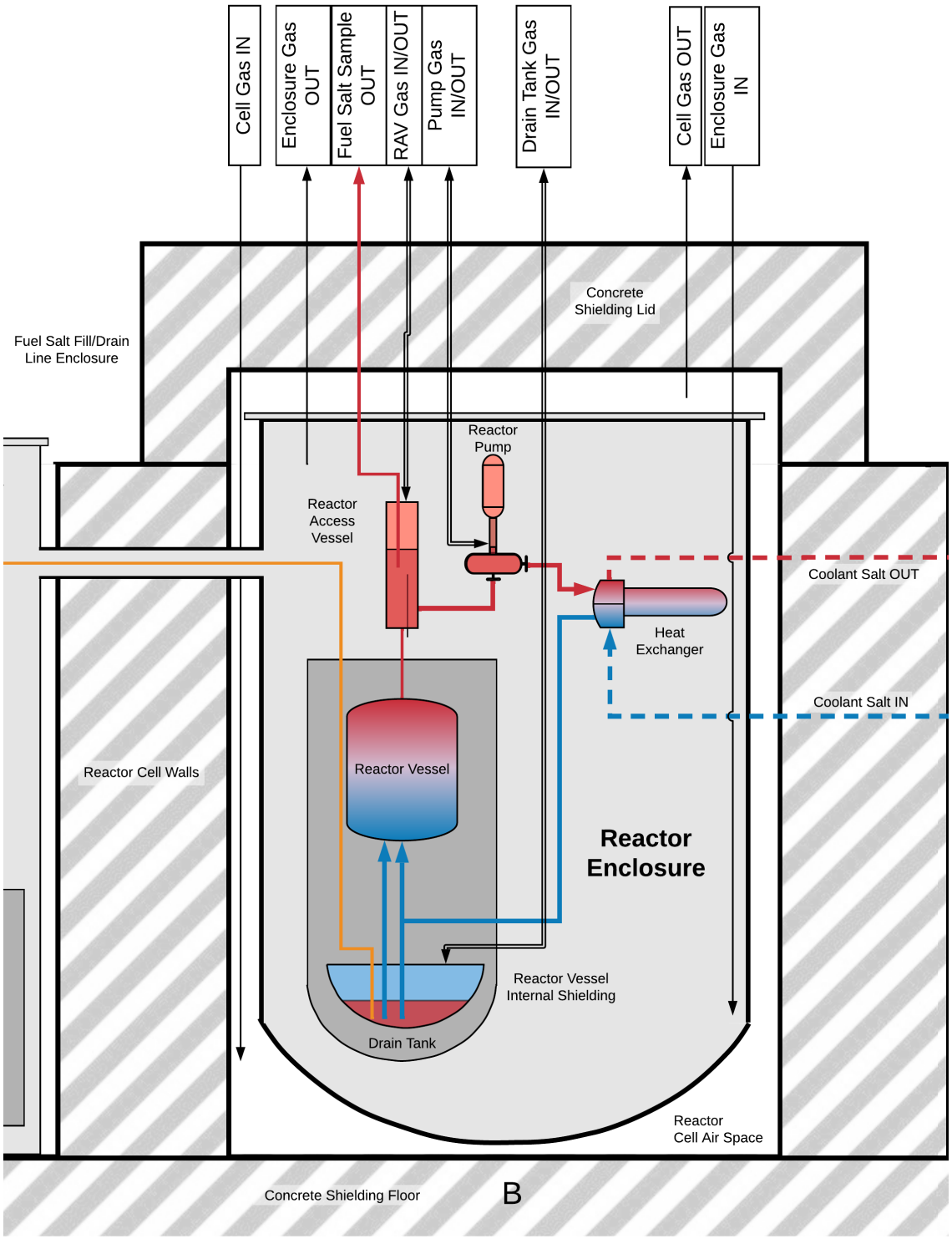


Figure 4.1-3 Reactor System Major Internals and Approximate Dimensions

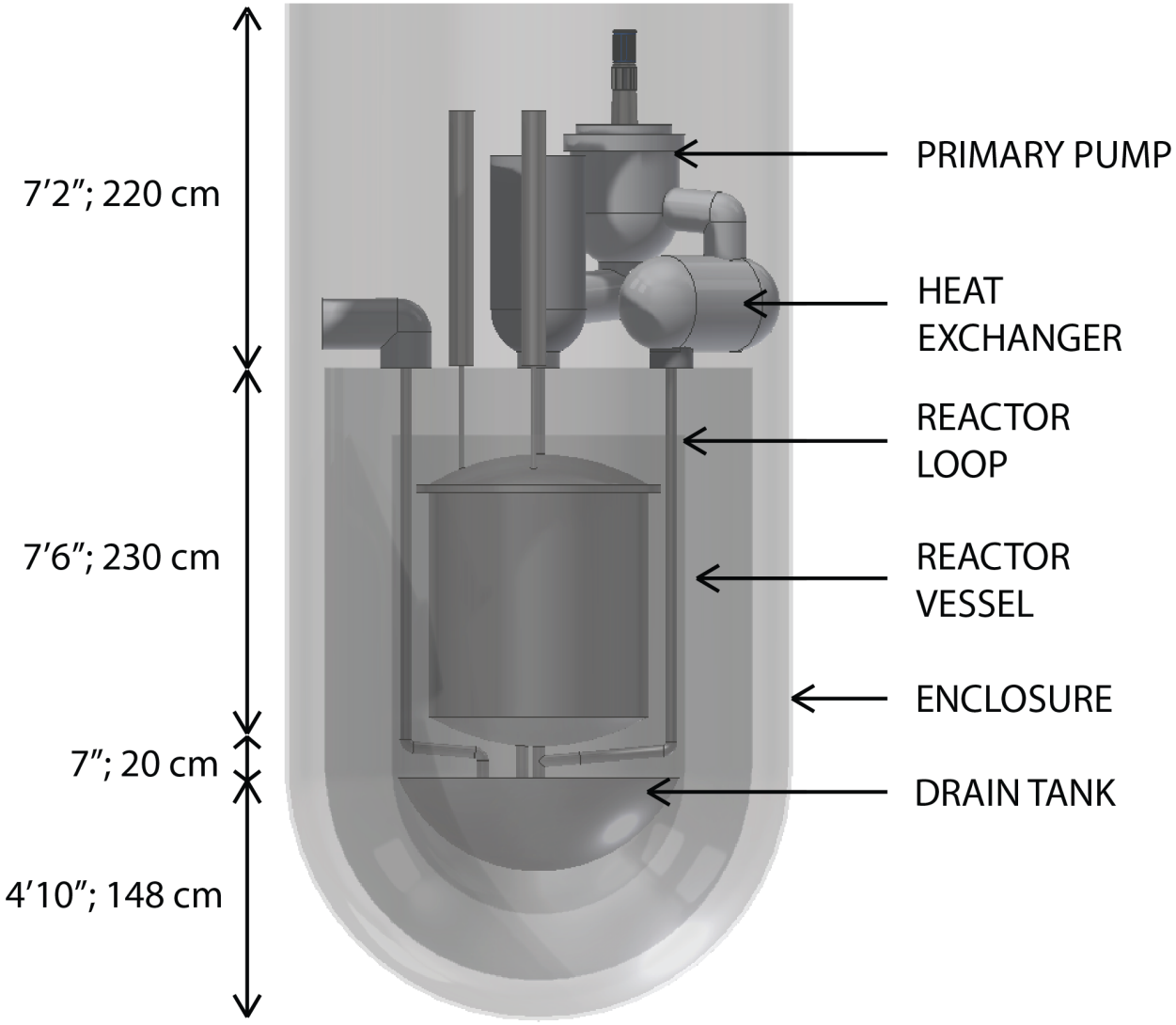


Table 4.1-1 Reactor Parameters

Parameter	Value
Thermal Power	1 MWth
Maximum reactor operating temperature	650 °C
Maximum reactor operating pressure	RAV: 2 psig, Drain tank: 30 psig
Fuel type	Uranium tetrafluoride dissolved in molten fluoride salt
Initial fuel enrichment	19.75% nominal
Moderator	Graphite
Neutron spectrum	Thermal

4.2 Active Reactor Core

This section provides the design bases and safety limits for the fuel salt, control elements, RPS, graphite moderator, startup source, graphite support structure, reactor system support structure, and reactor enclosure support structure.

4.2.1 Reactor Fuel

4.2.1.1 Description of the Fuel Salt

The nominal composition of the fuel salt at startup is 67LiF-28BeF_2-5UF_4, reported in mole percent. The lithium is enriched to 99.99 percent. The uranium is 19.75 percent enriched. The exact composition during operation varies from this nominal composition. Allowable ranges of the composition will be provided in the Operating License application. The nominal composition is assumed throughout the safety analysis unless otherwise noted.

Approximately 550 liters (L) of molten fuel salt are present within the reactor system. The reactor loop, during normal operation, will be filled with approximately 500 L of fuel salt with approximately 50 L of fuel salt in the drain tank. The reactor vessel requires 400 L to fill. The remaining 100 L is split between the RAV, reactor pump, heat exchanger, and associated piping. Allowable ranges for the quantities of fuel salt in the detailed design of the reactor system will be provided in the Operating License application. The drain tank can hold all of the fuel salt from the reactor system when not in operation.

4.2.1.2 Design Bases for the Fuel Salt

Consistent with Design Criterion (DC) 10 (see Section 3.1.2.2), the fuel salt is designed with appropriate margin to ensure specified acceptable system radionuclide release design limits are not exceeded.

Consistent with DC 16, the fuel salt retains fission products and actinides as a part of the functional containment to control release of radioactivity.

Consistent with DC 71, the composition of the fuel salt shall remain within specified limits to support safety functions.

The design basis for the MSRR is the maintenance of fuel system boundary integrity under any conditions assumed in the safety analysis, with the exception of the maximum hypothetical accident and heat exchanger tube failure accident, which assumes a non-mechanistic loss of fuel salt boundary integrity. These requirements form the design bases for the fuel salt and may be summarized with the following functional requirements.

- Thermophysical properties of the fuel salt are such that all accidents do not challenge the fuel salt boundary.
- Impurities are low enough that aggressive attack on the reactor loop is precluded.

- Redox is controlled to minimize aggressive attack on the reactor loop.
- Development of fission products does not aggressively attack the fuel salt.
- Actinides remain soluble throughout operation.
- Temperature and pressure remain within reactor system bounds.

The maximum hypothetical accident is presented in Chapter 13. The remaining accidents in Chapter 13 do not challenge the fuel salt system boundary. Sufficient information is provided about the thermophysical properties of the fuel salt to allow the safety analyses and heat removal analyses to be completed.

4.2.1.3 Location and Composition of the Fuel Salt

The fuel salt boundary is called the reactor system, comprising those components that contain fuel salt during operation. This barrier may be challenged by a number of fuel salt factors. Chemical attack by the fuel salt is driven by impurities within the fuel salt and by the redox potential of the salt. Impurity concentration and redox potential are monitored; redox potential is actively managed. Coupons of 316H are used to represent chemical attack on the reactor loop. These procedures are explained in Section 4.2.1.6. The formation of fission products coupled with the reactor design lifetime does not degrade the reactor system boundary to such a degree that barrier integrity failure becomes possible.

Fuel salt operating pressure and temperature are bound by the reactor system design parameters as presented in Section 4.3 and Section 4.5. Actinides will remain well mixed with the fuel salt during operation.

The FHS can hold, store, and process all of the fuel salt at once. A small quantity of fuel salt may be present within the FHS during reactor operation. The FHS is described in Section 9.2.

The fuel salt will contain impurities from manufacturing that cannot be removed by cleaning. Allowable bounds for these impurities will be provided in the Operating License application but it is anticipated that such impurities will not measurably impact salt properties or behavior. The fuel salt changes in composition through corrosion and through the generation of fission products and minor actinides. Estimates of corrosion products will be provided in the application for an Operating License.

At operational temperatures and pressures, the fuel salt is chemically homogeneous. Heterogeneity was not observed for molten fuel salt throughout the operational history of the Molten-Salt Research Experiment (MSRE) at Oak Ridge. Some heterogeneity is expected during freezing and thawing, but this homogenizes with temperatures above the liquidus. Analysis of the addition of UF_4 to the fuel salt is provided in Section 4.5.

For the purposes of the safety analysis, 100 percent of the xenon, krypton, tellurium, iodine, and bromine, as well as their daughters, are assumed to be present within the RAV head space. This assumption is considered to be

extremely conservative; measurements to quantify the conservatism of these assumptions will be conducted during fuel acceptance testing, startup testing, and initial low power operation [Reference 4.8-7]. These procedures are designed to confirm this behavior and demonstrate these assumptions are bounding. These procedures will be fully described in the Operating License application.

A full list of the end-of-life fuel composition will be provided in the Operating License Application.

The material composition of the fuel salt in the reactor can change because of

- fission product buildup.
- The depletion of fissile material and the addition of UF_4 to maintain criticality.
- The removal of fuel salt and fission gas through the RAV.
- The addition of corrosion products from the reactor structure.
- the addition of contaminants from the gases.
- a change in redox (or U^{4+}/U^{3+} ratio) through beryllium addition.
- a change in redox through fission itself.

These changes to fuel salt composition will be managed such that the fuel salt remains within operating bounds. Fuel salt composition will be measured directly from sampling and analysis. Specific limits may be defined for impurities (such as oxygen), chromium and iron concentration (corrosion indicators), and redox potential. Redox probes are part of the experimental system; however, chemistry control of the fuel salt will not rely on redox probes. Sampling of the gas stream will also indicate the presence of oxygen. Administrative controls governing the addition of UF_4 will limit the actinide concentration. Coupons will serve as an additional corrosion indicator. Fuel salt level will be maintained by the pressure within the RAV and drain tank and monitored indirectly through said pressure and directly through sensors. Radar sensors are the most promising choice of level sensor, with the final choice of sensor being reported in the operating license application. Reactor enclosure gas will be periodically sampled and monitored for radionuclides. The presence of fission products above a certain concentration in the enclosure will indicate the magnitude of a leak. A major leak will also result in a radical pressure change in the reactor system and gas management system and prevent continued operation. Small leaks will be monitored. ACU anticipates operating the MSRR with some defined low leak rate of contaminated helium into the reactor enclosure gas. These limits will be reported in the Operating License.

4.2.1.4 Thermophysical Properties

Evaluations of thermophysical properties will be provided in the Operating License application.

4.2.1.5 Fission Gas Formation and Reactivity

The off-gas system and bubbling system are described in more detail in Section 9.6 and Section 10.2.3.

Helium can be actively bubbled into the fuel salt through the RAV to remove xenon and krypton as part of an experimental system (see Section 10.2.3). These bubbles float upwards because of buoyancy and because the flow is oriented upwards. As these bubbles reach the free surface, most migrate to the RAV head space where they can be drawn into the off-gas system. An experimental bubble removal system, also detailed in Section 10.2.3, is positioned immediately after the RAV to remove bubbles that are entrained in the fuel salt.

Helium is present within the fuel salt; it will be the primary constituent of any gases in the reactor system. Gases are homogeneously mixed. The pressure dependency of gas is discussed in Section 4.5. Void formation collapse is discussed in Section 4.5, and the associated accident is analyzed in Chapter 13.

4.2.1.6 Stainless Steel SS316H Corrosion Monitoring and Control

Consistent with DC 71, the fuel salt is maintained in a minimally corrosive regime through control of the redox potential and impurities. Corrosion of the reactor system structural material by the fuel salt is primarily driven by the redox potential of the fuel salt and oxidizing impurities within the fuel salt. Redox potential is maintained and monitored in line with best practices from the MSRE. Oxygen within the fuel salt poses two hazards. Oxygen can enhance corrosion by oxidizing salt constituents, resulting in free fluorine and thus an increase in redox potential. Also, oxygen can react with uranium fluorides and precipitate UO_2 out of solution. The formation of large quantities of UO_2 can pose a criticality hazard and has the potential to block piping. Oxygen and other oxidizing impurities are directly monitored during operation. Based on experience with the MSRE, experience with other molten fluoride systems, and numerous scientific investigations into corrosion, corrosion is not expected to result in structural failure of the reactor system boundary. Multiple techniques are employed to monitor redox and corrosion, including coupons of SS316H placed in the fuel salt, and sampling the fuel salt and cover gas.

A detailed description of the redox potential control and monitoring system and allowable operational parameters and procedures will be provided in the Operating License application.

4.2.1.7 Temperature and Pressure Limits

Fuel salt temperature and pressure limits are defined to ensure the integrity of the reactor system under all conditions. Fuel salt temperature will be measured to ensure that the reactor operating limits and safety limit are not violated. The operating limits of the reactor system will be 550 °C and 650 °C.

The fuel salt liquidus is less than 500 °C. With further research the liquidus will be identified with greater accuracy and reported in the Operating License application. The minimum fuel salt operating limit ensures that fuel salt cannot freeze in the drain lines and impair RPS functionality. The maximum fuel salt operating limit ensures that under a maximum reactivity insertion the reactor system safety limit is not breached.

Fuel salt pressure limits are provided in Section 4.3.7.

4.2.2 Control Elements

The reactor protection system (RPS) brings the reactor to a shutdown state under all conditions. Safety analyses are based on achieving reactor shutdown by draining the fuel salt from the reactor vessel to the drain tank. Distinct from the RPS, the Reactivity Control System (RCS) controls reactivity during operation using CRs. The RCS is used to bring the reactor subcritical with fuel salt in the reactor vessel at operating conditions and is not intended to play a role in reactor shutdown.

4.2.2.1 Reactor Protection System

Consistent with DC 2, the RPS is protected against the effects of natural phenomena and can safely shutdown the reactor under those conditions.

Consistent with DC 4, the RPS is designed to accommodate the effects of and to be compatible with the expected environmental conditions during operation, maintenance, testing, and postulated accidents.

Consistent with DC 23 the RPS designed failure mode is to drain the fuel salt shutting down the reactor.

Consistent with DC 26, the RPS provides separate and diverse means for achieving reactor shutdown.

Consistent with DC 73, the gas lines associated with the RPS are monitored for potential signs of plugging which could impair a safety function.

The design basis for the RPS is to achieve reactor shutdown under the conditions and within the time required by the accident analyses to maintain fuel system boundary integrity.

The functional requirements for the RPS are presented below.

- The reactor is brought to a shutdown state by draining of the fuel salt
- Drain is affected by the equalization of pressure within the drain tank and the RAV head space.
- One minute is needed to drain the reactor vessel to a level below the graphite.
- Fuel salt must remain in liquid state during draining.

- The reactor is subcritical when the reactor core is 25 percent uncovered (less than 75 percent full).
- Upon actuation all six valves open, two valves on any single pathway are required to shutdown the reactor.
- The RPS valves and associated control equipment will be suitably diverse, reliable, and redundant.

There will be three independent paths connecting the gas space in the Reactor Access Vessel (RAV) and the gas space in the Reactor Drain Tank (RDT). Each path will contain two valves in series. The valves will be located above the liquid level so that they are not exposed to molten salt. One of these lines will be contained entirely inside the Reactor Thermal Management System (RTMS). The others will be inside the RTMS except for a short portion containing the valves that extend above the RTMS so that those valves are not exposed to high temperature. A reactor shutdown only requires one single pathway to open. It should also be noted that (although not part of the reactor shutdown system) closing the helium supply line from the medium pressure helium tank to the RDT will depressurize the RDT and result in a reactor shutdown.

While only one single pathway (with two open valves) is required to shutdown the reactor, to comply with DC 26 the MSRR will be operated with at least two pathways which have the capability to shutdown the reactor. The MSRR may operate with a single pathway non-operable. Every valve must be fail-open. In the event that the actuator fails, the valve will open. As long as the valve on a pathway fails open, the MSRR may continue to utilize that pathway with a single shutdown valve as the inoperable valve will not close. The MSRR may be operated with a single valve per pathway open to ensure that the operation of a single valve equalizes pressure and drains the reactor. In this manner, the MSRR has enhanced operability but does not compromise safety.

Specific valves are being evaluated and different valves and/or actuators may be used for each valve. The use of different valve types and/or actuators will provide redundancy and independence to satisfy DC 21. Valves and actuators will be independent of each other to satisfy DC 22. At elevated temperature, it is conceivable that a metal valve seat and plug will diffuse into each other and stick together, preventing opening. To ameliorate this, the shutdown valves operating at elevated temperature (600 C) may use a plug and seat that are sufficiently dissimilar to prevent this issue (such as a metal plug and ceramic seat). Those shutdown valves at reduced temperature (70 C) may be designed with the same metal plug and seat as this diffusion issue will not occur.

Buildup of material in the gas lines may interfere with operation of the shutdown system by clogging lines or the valves themselves. Changes in gas pressure and vessel level height will be monitored during tests of the shutdown system to observe any deviation from the technical specifications. These valves may be cycled to verify that each will close and open. If deviation does occur, the line can be cleaned or blown down during maintenance. The valves maintained at elevated temperature during operation are much less likely to clog than the relatively cold lines. Moreover, some pathways come from the stagnant vessel

headspaces while other pathways come from the existing pressure control lines. In this manner, the baseline helium flow at the entrances/exits of the shutdown lines are different, some stagnant and some flowing.

After many uses, the valve seal may no longer perfectly seal and begin to leak. Gas will leak from the drain tank to the RAV head space and helium will have to be supplied to maintain fuel salt level. This is not a safety concern, but an operational one.

Valve leakage from the stem to the reactor enclosure could result in some radionuclides escaping to the enclosure which is maintained at negative pressure. Allowable leak rates will be reported in the OL (DC 10 and DC 16).

Testing procedures for the shutdown system and a drain analysis will be provided in the Operating License application.

4.2.2.2 Reactivity Control System

The CRs are designed to bring the reactor to $k_{eff} < 0.99$ with fuel salt in the reactor vessel at a temperature of 550 degrees Celsius, and a single rod of the highest worth assumed to fail to insert into the reactor core and be fully withdrawn. Three control rods are used to achieve this design basis. The CRs are physically isolated from the fuel salt. The CRs are electrically operated; each CR assemblage is structurally attached to the top of the reactor vessel and supported by it.

Consistent with DC 27, the RCS provides a non-safety related way of controlling reactivity.

The design features of the RCS:

- Control rods are located within CR thimbles and separated from the fuel salt.
- Control rods are independently operated.
- Control rod thimbles are welded to the top of the reactor vessel.
- The atmosphere within CR thimble is helium.
- Control rods at design height can insert +350 pcm and -3600 pcm of reactivity (analysis provided in Section 4.5).
- Control rods and thimbles are constructed of SS316H, the design standards for which are discussed in Section 4.3.
- Control rod absorber material is 75 percent natural B_4C and 25 percent SS316H by volume.
- Control rod thimble material is SS316H.
- Inner and outer CR thimble radii are 1.9 cm and 1.7 cm, respectively.
- Control rod absorber radius is 1.5 cm.

- Height of the CRs when critical at 600 degrees Celsius is 50 cm above reactor centerline.
- Maximum control element withdrawal rate is 10 pcm/sec

Final design parameters, and additional considerations and analyses pertinent to the RCS will be provided in the Operating License application.

Figure 4.2-2 shows the control rod locations in the reactor core.

4.2.3 Neutron Moderator and Reflector

The MSRR design does not include a system designed to reflect leakage neutrons back into the core region.

The MSRR core design mimics the MSRE core design with a few changes. The blocks are hexagonal with one-third fuel channels machined on three of the vertices. This can be seen in Figure 4.2-3. Figure 4.2-4 shows the assembled structure.

These graphite blocks are arranged in a regular lattice to form the graphite core. The upper portion of the graphite blocks are curved to prevent fuel salt pooling on top of the graphite. The blocks are vertically restrained by the lower stainless steel grid plate as described in Section 4.2.5. The blocks and grid plate maintain graphite core geometry. The total mass of graphite is approximately 3 tons. Graphite floats in the fuel salt and must be anchored down to the grid plate. The graphite does not have a protective coating; it is in direct physical contact with the fuel salt.

The graphite moderator is not expected to react chemically with the fuel salt and its constituents. Effect of fission products or activation products on wetting or chemical reactions with graphite are being looked into by various research groups. This research will be used to define Technical Specifications for the fuel salt and will be reported in the operating license application.

The graphite core design must ensure neutron irradiation induced phenomena (such as dimensional changes, internal stresses, and hardening) do not result in structural failure of the core under anticipated operating conditions. The low power density, high fissile loading, and short operating time result in a maximum neutron fluence within the graphite less than 2×10^{20} n/cm^2 for neutron energies > 0.1MeV.

- The graphite density at room temperature must be greater than 1.75 g/cc, although greater densities are preferred, if available.
- The boron equivalent concentration must be less than 2 ppm. Boron equivalency is calculated as the effective absorption of all impurities expressed as if only natural boron was present.
- The graphite has a maximum average pore size of 1 μm in diameter
- The graphite must not have visible cracks.
- Behavior of the specific grade of graphite during neutron irradiation for the anticipated operational lifetime must be accounted for in the design.

- The graphite will be manufactured per the NEXT Lab Quality Assurance Program.
- The graphite core must be secured but tolerate limited movement or floatation. The graphite blocks must not shift away from each other as the core heats up.
- The graphite core design must ensure that the difference between the thermal expansion coefficients of the graphite and the stainless steel grid plate does not result in structural failure of the core under anticipated operating conditions and accidents.

Table 4.2-1 provides preliminary design parameters used for the safety analysis. Final design parameters will be provided in the Operating License application.

4.2.4 Neutron Startup Source

The neutron source type, strength, and location depend on the neutron detectors chosen and source placement. This information will be provided in the Operating License application.

4.2.5 Core Support Structure

Because the MSRR fuel salt is mobile throughout the reactor system, the description of the core supports explains how the entire reactor system is supported. The core support structure is divided into three parts. The first subsection describes the method by which reactor graphite is affixed to the reactor grid plate, which is attached to the reactor vessel. The second describes the method by which the entire reactor system is supported inside the reactor enclosure. The third describes the support structure for the reactor enclosure, which contains the entire reactor.

4.2.5.1 Graphite Support Structure

The tall vertical graphite blocks, seen in Figure 4.2-4, are anchored to the lower grid plate, seen in Figure 4.2-5. The graphite moderator is affixed to the lower grid plate in such a manner that the graphite core design bases are satisfied. The grid plate is welded to the ribs in the lower plenum attached to the bottom of the reactor vessel. The blocks are loosely attached to the lower core grid plate in a manner similar to that of the MSRE to account for differing thermal expansions of graphite and stainless steel.

The density of the fuel salt is greater than that of the graphite, so the graphite is buoyant in the salt. The consequent movement of graphite is accounted for in the mechanical design of both the grid plate and the graphite.

Preliminary design parameters for the grid plate are provided in Table 4.2-2. Final design parameters satisfying the design bases will be provided in the Operating License application.

4.2.5.2 Reactor System Support

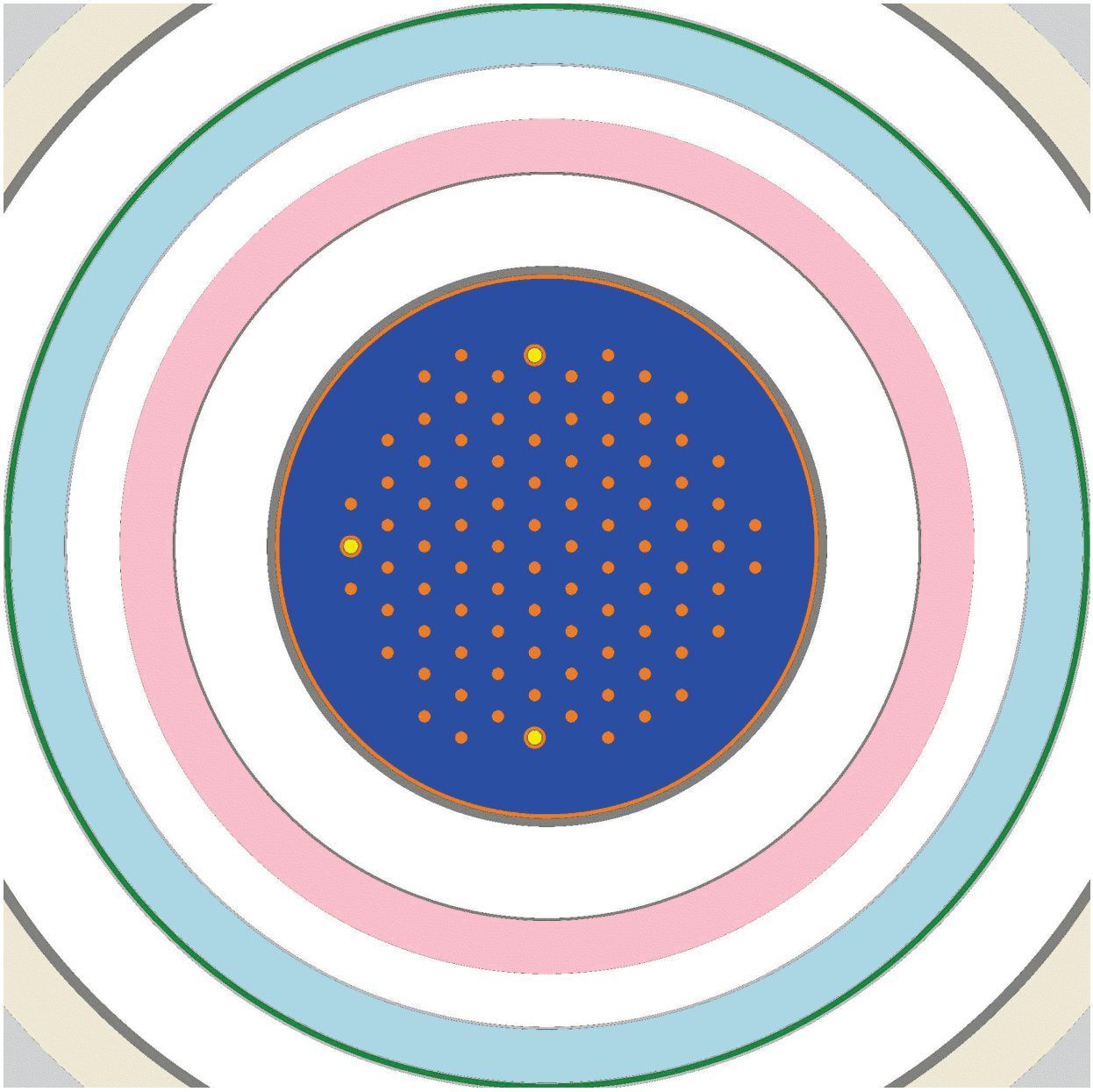
The RAV and drain tank are structurally supported by the reactor vessel. The reactor vessel and reactor pump are structurally supported by the RTMS. The primary heat exchanger is structurally supported by the reactor enclosure accounting for differential thermal expansion of the piping connecting it to the rest of the reactor system. The internal shield is structurally supported by the reactor enclosure. Thermal stresses will be minimized between the reactor vessel and reactor pump as they are attached to the same component (RTMS) which will be at relatively uniform temperatures. The final design will be presented in the Operating License application.

4.2.5.3 Reactor Enclosure Support Structure

The reactor enclosure is structurally supported from below, transmitting the load to the floor of the systems pit. The reactor enclosure support structure is rigid. Final design details will be provided in the application for an Operating License.

Figure 4.2-1 Deleted

Figure 4.2-2 Locations of the Three Control Rod Thimbles Shown in Yellow



The reactor is surrounded by a thick insulating jacket and the internal shield. The reactor enclosure vessel and cell can be seen.

Figure 4.2-3 Single Hexagonal Graphite Block (Left) and Channels Formed by Lattice of Blocks (Right)

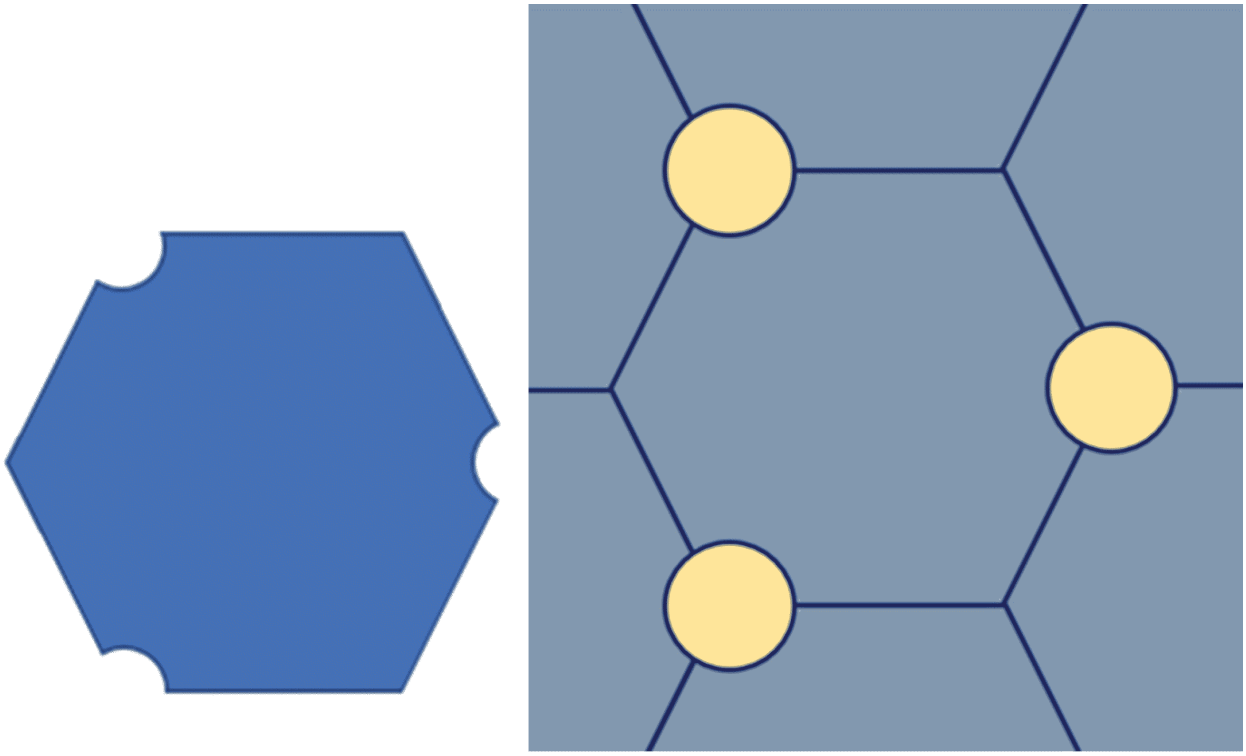


Figure 4.2-4 Hexagonal Lattice of Graphite Blocks Showing Channels at Block Corners

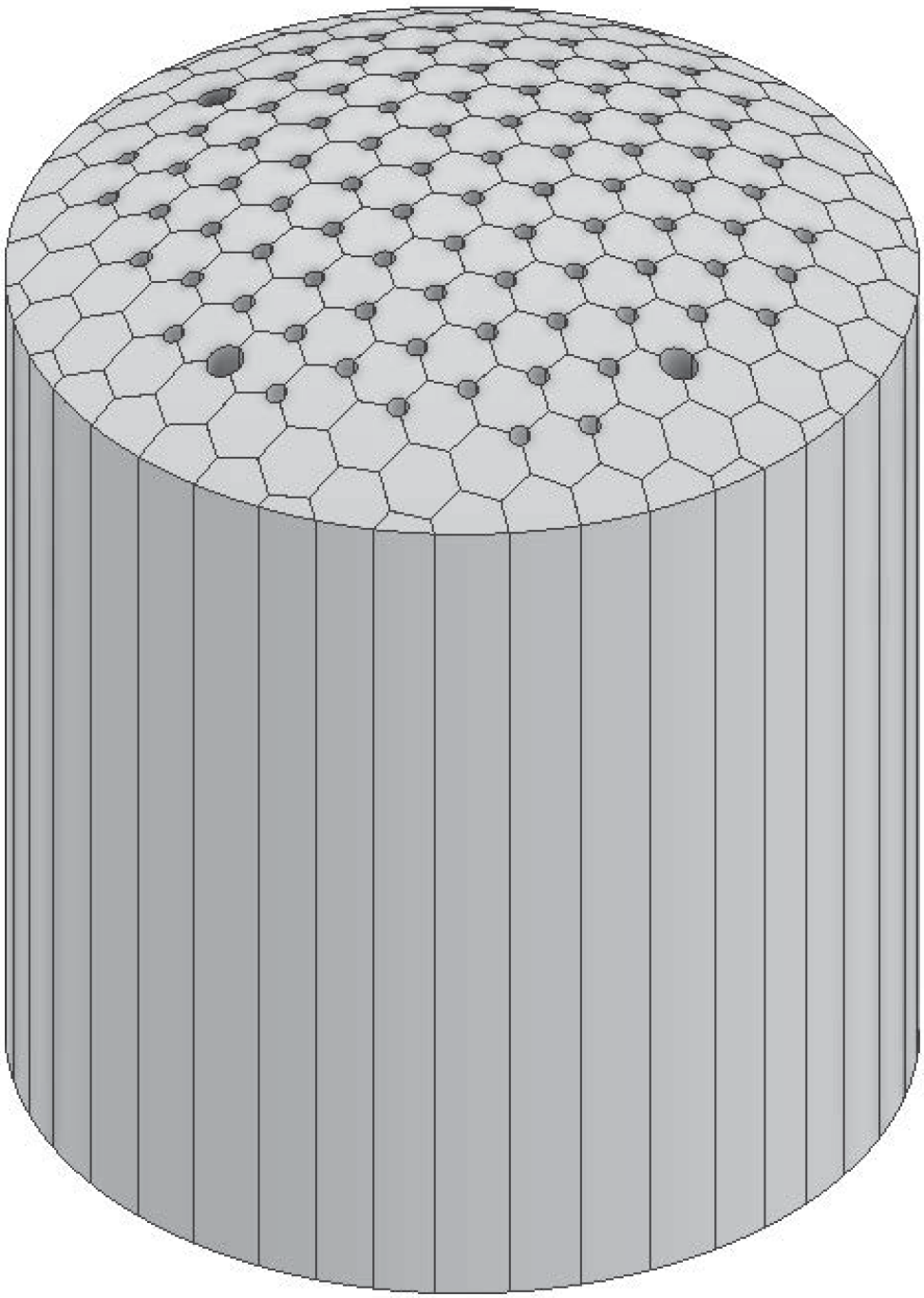


Figure 4.2-5 Lower Core Grid Plate

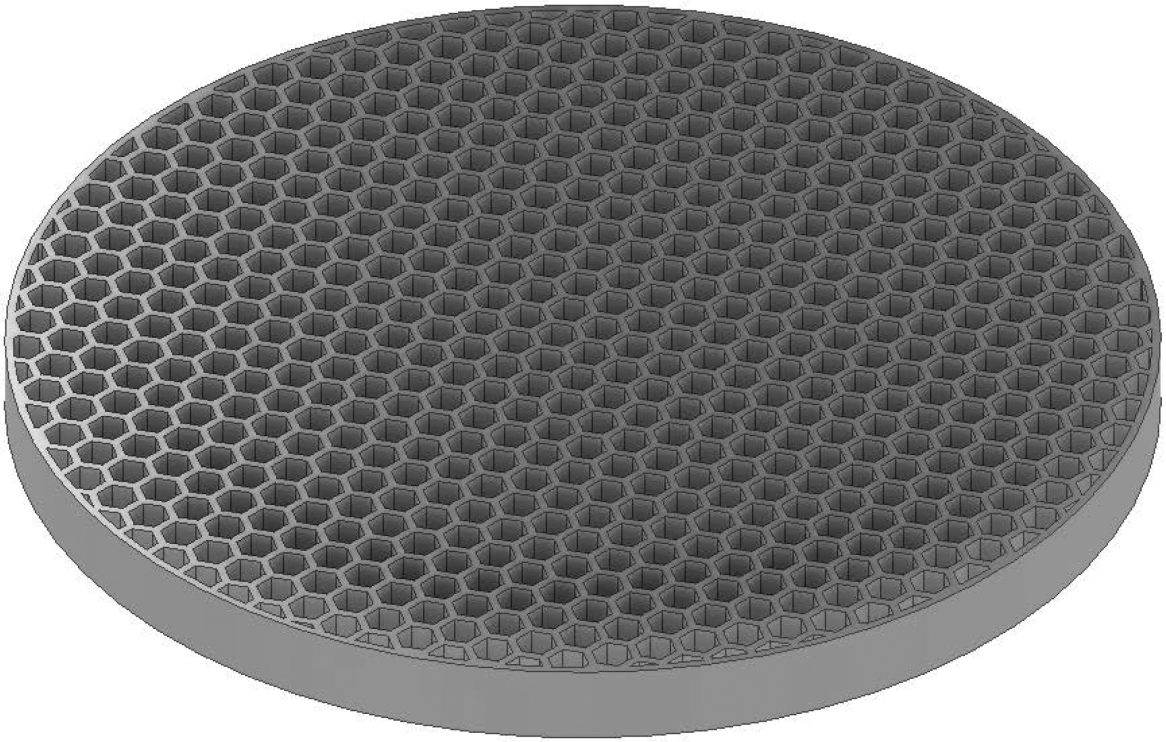


Table 4.2-1 Graphite Dimensions at 600 Degrees Celsius

Parameter	Value
Graphite lattice pitch	10.16 cm
Graphite block pitch	10.15 cm
Fuel channel diameter	3.016 cm
Graphite lattice type	Hexagonal
Outer graphite lattice radius	65.0 cm
Maximum height of graphite core	151.68 cm
Upper core radius	132 cm
Graphite volume	1758 L
Graphite density	1.8 g/cc

Table 4.2-2 Parameters Describing the Grid Plate at 600 Degrees Celsius

Parameter	Value
Grid plate material	SS 316H
Grid plate lattice type	Hexagonal
Grid plate lattice pitch	10.16 cm
Grid plate web thickness	2.032 cm
Grid plate offset relative to graphite	+x 5.866 cm
Grid plate thickness	9 cm
Thickness of outer ring	1 cm
Total diameter	130 cm
Grid plate mass	620 kg

4.3 Vessel

4.3.1 Description

The reactor system includes the reactor vessel, drain tank, heat exchanger, RAV, reactor pump, and all associated piping. This section describes design features common to all the fuel-salt-bearing components within the reactor enclosure.

The lower portion of the reactor vessel is manufactured in two pieces and welded together. The upper portion of the vessel with control rod thimbles is added once the graphite core and grid plate are placed inside. The upper lid is welded to the vessel after the graphite core and grid plate are placed inside. The bottom of the vessel has two penetrations, one that serves as a fuel salt inlet and fuel salt drain and the other as a drain only. The lid includes penetrations for the control rods and fuel salt outlet.

4.3.2 Design Bases

Consistent with DC 1, the reactor system is designed, fabricated, erected, and tested to quality standards which comply with ANS-15.8. The mechanical design will be in compliance with the codes and standards outlined in Chapter 3.

Consistent with DC 2, the reactor system is protected against the effects of natural phenomena and can safely shutdown the reactor in these conditions.

Consistent with DC 4, the reactor system is designed to accommodate the effects of and to be compatible with the expected environmental conditions during operation, maintenance, testing, and postulated accidents.

Consistent with DC 10, the reactor system is designed with appropriate margin to ensure specified acceptable system radionuclide release design limits are not exceeded.

Consistent with DC 14, the reactor system is designed to have an extremely low probability of leakage, rapidly propagating failure, or rupture.

Consistent with DC 30, the fuel salt boundaries in the reactor system are designed, fabricated, erected, and tested to the highest quality standards practical. Reactor enclosure gas will be periodically monitored. A substantial quantity of fission products detected in the enclosure will indicate the presence of a rupture. Limits and actions will be defined in Technical Specifications.

Consistent with DC 31, the fuel salt boundaries in the reactor system have sufficient margin to minimalize the probability of rupture.

Consistent with DC 32, the fuel salt boundaries in the reactor system are designed to permit periodic inspection, testing, and surveillance. The reactor system is designed to permit visual inspection of the reactor system exterior and limited ultrasonic testing of select welds. The reactor system and gas management system are designed to permit pressure testing. Surveillance requirements are met through fuel salt sampling and the use of coupons.

Consistent with DC 35, the reactor system provides a means to remove residual heat following an accident.

Consistent with DC 74, there are no fuel salt boundary interfaces with a structure, system, or component containing fluid that if allowed to freely interact with the fuel salt would cause the loss of a safety function.

Additional details on how these DC are met will be provided in the Operating License application.

Codes, standards, and design conditions applicable to the reactor system are provided in Chapter 3.

4.3.3 Reactor System Structural Material

The reactor system is constructed of stainless steel 316H or equivalent (as determined by carbon content). This material meets the applicable requirements of ASME BPVC [Reference 4.8-1]. The reactor system may utilize a redox probe which is composed of a different material. The redox probe will be electrically insulated from the reactor system structure so it will not galvanically corrode the structure.

Regulatory Guide 1.87, Revision 2 endorses ASME BPVC Section III-5 for SR components. ACU will follow the ASME Code as endorsed by Regulatory Guide 1.87 and as adjusted for quality requirements as discussed in PSAR Section 3.1.4. Any deviations from the guidance in ASME BPVC Section III-5 not related to quality assurance will be identified and justified. This part of the ASME Code covers SS 316H in terms of high-temperature strength, creep, and creep-fatigue effects up to a design life of 300,000 hours. However, the Code does not address other key requirements associated with the design of the components, such as corrosion and neutron embrittlement. ACU has a degradation management program that assesses and manages degradation mechanisms for safety related components.

Weld filler ER 316 is very similar in composition to the base metal and will be used as the weld filler material, as outlined in ASME BPVC.III.5-2017 Table HBB-I-14.1(b) "Permissible Weld Materials." Welding procedures will comply with appropriate sections of the codes and standards.

4.3.4 Fuel Salt Chemical Attack

As described in Section 4.2.1, the fuel salt chemically attacks the reactor system boundary. With active redox control this phenomenon is minimized. Sufficient thickness of the piping is provided to ensure that the expected level of corrosion does not result in possible failure of the reactor system boundary over the design life of the reactor. Anticipated corrosion rates will be provided in the Operating License application. The effects of a boundary failure and planned corrosion testing are described below.

The MHA, as described in Chapter 13, postulates the relocation of all fuel salt outside the reactor system into the RTMS. Gases present within the reactor system and reactor gas management system would come to equilibrium pressure with the reactor

enclosure, a leak tight negative pressure fission product boundary that contains the reactor system and RTMS. When the fuel salt escapes the reactor system, it is no longer in the presence of the moderator that remains in the core and is subcritical. The fuel salt will gradually cool over the course of several days due to the presence of decay heat and being inside the RTMS, which insulates the reactor system and fuel salt. Gaseous fission products are postulated by the MHA to leak from the reactor enclosure into the SERC bay and from there to the atmosphere. In such an event, the maximally exposed individual receives < 100 mrem.

This MHA provides the background against which the safety significance of degradation mechanisms is viewed. In most conceivable degradation events, the outcome would be a small leak that would be detected as fission products in the gas surrounding the reactor system. A small leak could be addressed during a subsequent maintenance and repair event. Even in the non-credible event that results in the MHA, dose limits established in 10 CFR 20 are not exceeded. As a result, it is appropriate to begin the operation of the MSRR with some uncertainty in the specific rates of degradation from one cause or another. Even if all detection mechanisms fail to warn operators before a significant leak, there is still no mechanism by which such an event results in doses to workers, the public, or the environment that exceed regulatory limits. As part of the research associated with this reactor, ACU is seeking all relevant data from the literature, consulting foremost experts in the field, and completing a rigorous testing campaign. The knowledge gained from operation of this research reactor will reduce much of the remaining uncertainty associated with molten salt fueled reactors while adequately protecting public health and safety.

4.3.5 Radiation Damage to Reactor System

According to calculations done with preliminary design inputs, the maximum damage to 316H components after 5 effective full power years for the MSRR will be bounded by 1 dpa. Considering that the MSRR is a low-pressure system with low service stresses, and that the fuel salt chemistry will be tightly controlled to minimize corrosion of SSCs; 316H components and welds in the MSRR are expected to perform their safety functions and maintain conditions specified in the design criteria. A detailed assessment of neutron dose for components in the MSRR will be conducted using final design inputs and will be provided in the operating license application.

The impact of radiation on the graphite core will be accounted for during design with the caveat that as a non safety related component, the data used for the graphite design meets a lower quality standard. The maximum dpa and fluence expected for the graphite will be 1 dpa and 1E21 n/cm^2 (E > 0.18 MeV).

4.3.6 Stainless Steel Oxidation and Creep

Stainless steel 316H has adequate mechanical properties at the operating temperature range of the reactor. These properties are well established and are characterized within the existing guidance for the use of SS 316H in the ASME BPVC

[Reference 4.8-1]. Creep and oxidation, though, pose special challenges to SS 316H. Thus, high temperature creep and oxidation are well characterized, can be accounted for by design, and do not result in structural failure of the reactor system.

4.3.7 Pressure Effects

The reactor system pressure safety limit will be defined to ensure that all components including vessels, piping, valves, and seals will retain design basis capability under all conditions. The operating range of the helium within the RAV is -2 psig to 2 psig. The design pressure of reactor components and the helium vessels V-7001 and V-7002 will meet or exceed the PRV set point associated with V-7004. The reactor system will be designed to withstand normal and accident transients that induce pressure changes. Pressure transients induced by gas control system failure will be accounted for. Pressure transients induced by reactor shutdown, where the drain tank, RAV, and low pressure helium tank reach equilibrium, will be accounted for. Pressure control for the fuel salt is described in the GMS (Section 9.6).

4.3.8 Thermal Design Limits

Thermal safety limit for the reactor system will be defined to ensure that the reactor system structural fission product barrier will not rapidly deteriorate under any condition. The reactor system safety limit will ensure that the reactor system remains with code applicability as stated in Section 4.3.3. The reactor system operating limits are 550 °C and 650 °C to ensure that sufficient margin exists to successfully drain the reactor and to ensure that the reactor system never approaches the reactor safety limit. The maximum temperature of the reactor system is likely to be within the graphite due to radiation heating. However, graphite can withstand temperatures far in excess of the fuel salt boiling point. The peak temperature of the reactor system structure is likely to be located on the upper portion of the reactor vessel. Modeling and analysis will confirm the location to ensure the reactor system remains within operating conditions.

4.3.9 Reactor System Integrity Monitoring

Corrosion of the salt-wetted steel surfaces is monitored as described in Section 4.2.1.

4.3.10 Fuel Salt Leakage

Small leaks of fuel are contained in the reactor thermal management system (RTMS), which is described in Chapter 6. The reactor system is located inside the reactor enclosure, which, in conjunction with the reactor cell, is designed to mitigate consequences of operational leakage. All structural components that could come in contact with relocated fuel salt are constructed of SS 316H. Small leaks from the GMS into the reactor enclosure will be detected by the RMS. Such leaks are bounded by the MHA.

4.3.11 Description of the Reactor Access Vessel

Fuel salt flows upwards from the reactor vessel outlet through a short section of piping into the reactor access vessel inlet. Fuel salt turns and exits the reactor access vessel from the side and travels to the reactor pump. This vessel and sections of piping are well insulated, minimizing temperature loss before the reactor heat exchanger. The lower portion of the reactor access vessel is filled with fuel salt, while the upper portion contains helium. Fuel-salt gas mixing takes place within the reactor vessel through bubbling and at the free surface. Baffles may be located within the reactor access vessel to help guide the flow.

As described in Section 4.2.1, salt chemistry will be monitored and controlled by sampling, probes, and coupons. Small quantities of fuel salt will be periodically extracted from the fuel salt through the upper portion of the reactor access vessel as described in Chapter 10. Fuel salt samples will be analyzed in the radiochemistry lab with the goal of establishing the evolution of fuel sample composition throughout the reactor's operational history. The fuel salt sample extraction system must penetrate the reactor cell, reactor enclosure, and reactor system. This system will penetrate fission product barriers. The sample extraction system will utilize a series of interlocks to ensure that the intent of DC 50-57 are implemented as described in Section 6.2.2.7. Those portions of the sample extraction system inside the reactor access vessel will be designed to the same standards as the reactor system.

Redox potential may be measured directly with probes mounted from the top of the RAV. These probes will be constructed of material compatible with the fuel salt and reactor system. The interface between the probe and the reactor system structure will be sealed to ensure radionuclide barrier integrity is maintained.

Coupons of material, including but not limited to SS316H, will be periodically introduced and extracted from the fuel salt. Coupons will be located in the bypass line from the pump discharge to the RAV; coupon equipment will be mounted from the top of the RAV and from tees along the line. The coupon system will also follow the functional containment DC 50-57 as described in Section 6.2.2.7. Material coupons will be analyzed in the radiochemistry labs to understand the fuel salt's effect on the reactor system structural material. Those portions of the sample extraction system inside the RAV and the bypass line will be designed to the same standards as the reactor system.

Slugs of UF_4 will be added to the flowing fuel salt in discrete quantities during operation through an access port on the RAV. The UF_4 will gradually dissolve into the fuel salt solution causing a minor reactivity increase as discussed in Section 4.5. The UF_4 line is doubly isolated during operation. The UF_4 slug falls by gravity into the RAV and passes through several stopping points along its travel to maintain at least two barriers separating fuel salt vapors from the SERC bay.

Beryllium may be periodically added to the fuel salt to control the redox potential of the fuel salt. This process will take place within the RAV using equipment mounted from the top and entering the free surface of the fuel salt.

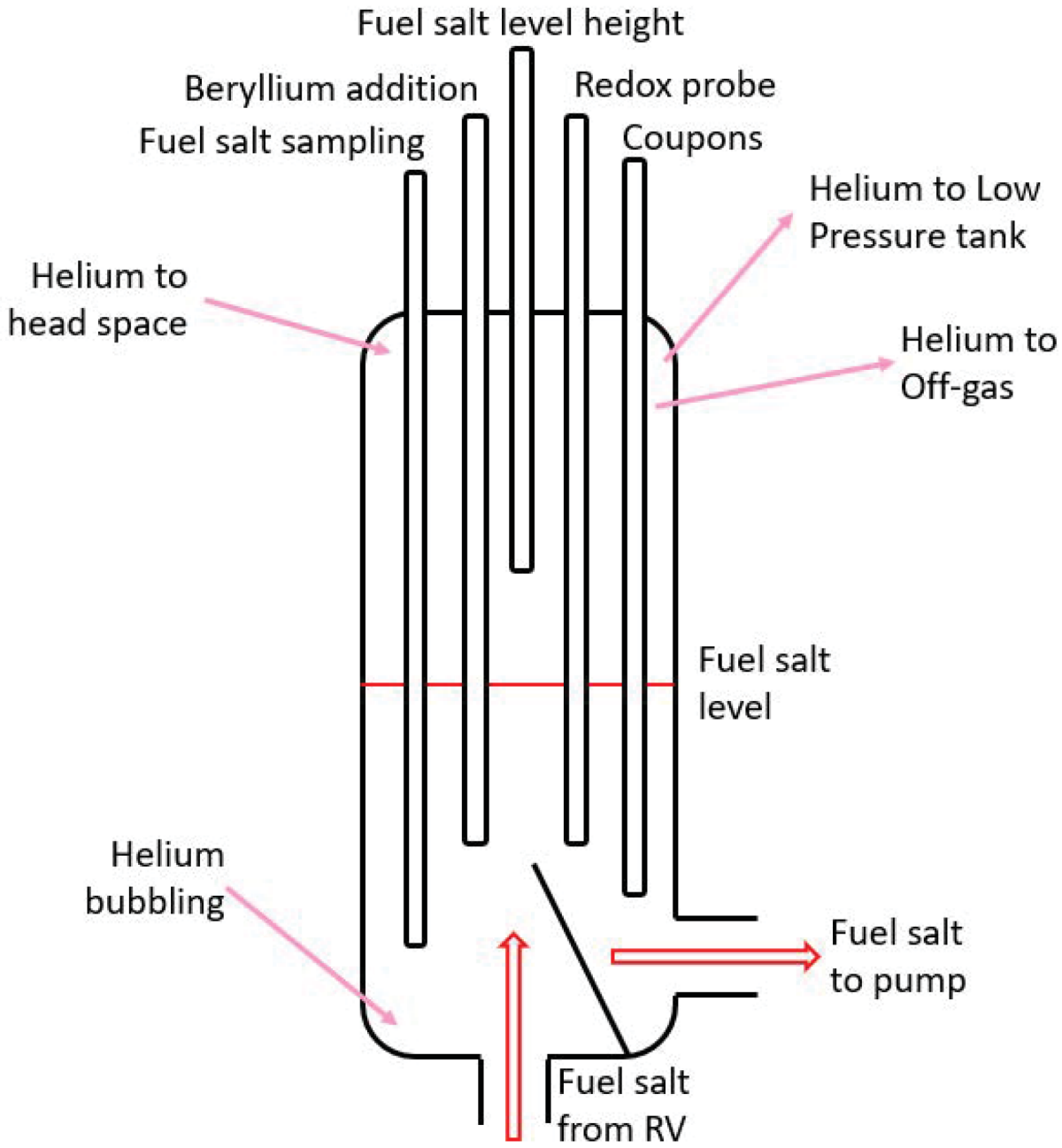
Helium pressure and flow within the RAV is foundational to the RPS, gas management system, and the off-gas system. The RAV will have helium connections to allow for helium to flow into and out of the upper portion of the vessel. Flow rate through the lines will be controlled as will the pressure of the head space. Fuel salt level sensors will be placed on the top of the RAV.

Fuel salt is not in direct contact with the top of the RAV. The RPS shall be designed to ensure, amongst other things, that the fuel salt level will not rise to such an extent that fuel salt may exit the RAV through any ports and connections at the top of the vessel.

The RAV may contain additional sensors as a part of the scientific surveillance system, described in Chapter 10.

The RAV is shown in Figure 4.3-1.

Figure 4.3-1 Reactor Access Vessel



4.4 Biological Shield

4.4.1 Description of Biological Shield

The preliminary design considerations are discussed below and the final design of the shielding will be provided in the Operating License application. According to the NRC definition, a biological shield is a mass of absorbing material placed around a reactor or radioactive source to reduce the radiation to a level safe for humans. Thus, the MSRR facility has radiation shields to protect the public, facility staff, and visitors. These shields also minimize activation of concrete in the SERC. Various shielding materials are used at the facility for specific purposes:

- Solid shielding materials (ordinary concrete, heavy concrete, high-hydrogen content material like polyethylene)
- Thermal-neutron absorbing materials (boron carbide)
- Photon-absorbing materials (steel, lead)

These materials provide necessary shielding from various radiation sources:

- Fission source terms (neutrons and photons)
- Capture photons
- Irradiated and activated circulating fuel salt source terms (neutrons and photons)
- Photon source term due to activation of components and structures
- Source term due to irradiated and activated fuel in the drain tank

4.4.2 Design Bases

The design bases for radiation shields are to ensure the dose rate in uncontrolled areas remains below the dose limit allowed for individual members of public per 10 CFR 20.1301, and the dose rate in controlled areas is consistent with the occupational dose limits to individual adults per 10 CFR 20.1201. The following target dose rate limits automatically ensure compliant annual dose, but when needed additional occupancy restrictions or operational restrictions can be implemented.

- Dose rate not exceeding 0.01 mrem/hr in uncontrolled areas results in the annual dose below 0.1 rem.
- Dose rate not exceeding 2.5 mrem/hr in controlled areas results in the annual dose below 5 rem for radiation workers as long as their annual working schedule is restricted to 2000 hours.

4.4.3 Functional Requirements

The functional requirements for the radiation shields include:

- Fission source term used in analysis corresponds to operating full power of 1 MWth.
- Source term due to activation corresponds to five years of continual operation at full power (5 effective full power years).

- Radiation source term due to activation of air inside the reactor enclosure and reactor cell, as well as activation of coolant salt. See Chapter 11.
- Established delayed neutron source term.

Additional shielding and details of the shielding design for radiation sources outside of the reactor cell will be addressed in the Operating License application.

4.4.4 Design Methods

Monte Carlo method is used for shielding analyses to allow accurate representation of the MSRR geometry and to enable accounting for penetrations. Calculational sequences available in the SCALE6.2.3/SCALE6.2.4 code system are used to generate sources and perform radiation transport shielding simulations to inform the biological shield design regarding the required composition and thickness of radiation shields and to confirm the design requirements are satisfied. SCALE is validated for shielding applications on multiple benchmarks and provides the required capability and efficiency of analysis. Criticality simulations are performed using the SCALE CSAS6 sequence and KENO-VI Monte Carlo code. Fuel depletion uses the SCALE TRITON sequence. Shielding analyses employ the SCALE MAVRIC sequence and MONACO fixed source shielding Monte Carlo code.

4.4.5 Design Approach

To achieve the design basis, radiation shields, including biological shield, are made up of multiple layers (see Figure 4.4-1). The stainless-steel reactor vessel reduces the level of radiation emanating at the outer surface.

The RTMS, which surrounds the reactor vessel, is surrounded by the internal portion of the biological shield, termed the internal shield because it is inside the reactor enclosure. The internal shield is made of a hydrogen-containing material (polyethylene) and thermal neutron absorbing materials (boron carbide). Its primary function is to efficiently degrade the energy spectrum of leakage neutrons and to capture neutrons to reduce activation of components and structures next to the reactor vessel and inside the reactor cell.

The next layer of the biological shield is located outside of and encompasses the reactor enclosure. It is termed the sacrificial shield. It is intended to reduce activation of permanent structures outside itself, but it will activate in the process. It is designed for easy removal to facilitate replacing, if needed, and ultimate decommissioning. It is made of interlocking heavy concrete blocks to reduce space requirements.

Beyond the heavy concrete, the permanent structure (systems pit structural walls), made of ordinary concrete, will further attenuate radiation.

Finally, limited areas with highest radiation levels, even if acceptable, are brought to a similar level as other bay areas by judiciously adding limited additional shielding, most likely ordinary concrete and steel or lead plates.

4.4.6 Internal Shield

The internal shield surrounds the RTMS, reducing the neutron flux in the reactor enclosure and reactor cell inside the biological shield. This reduces activation of the reactor enclosure physical structures and reduces the generation of Ar-41 in the reactor cell air. The Ar-41 generation is limited to comply with 10 CFR Part 20 as described in Chapter 11. The structural box containing the inner shield is made of Al 6061 with minimal impurities and good radiation resistance. The shield itself consists of an inner layer of high-density polyethylene, 12.7 cm thick and with a density of 0.94 g/cc, and an outer layer of packed B_4C powder, 1.27 cm thick and with an average density of 1.89 g/cc (75 percent theoretical density). Final design parameters satisfying the design bases will be provided in the Operating License application.

4.4.7 Biological Shield Inner Layer (Sacrificial Shield)

The next, inner layer of the biological shield is located outside of and fully encompasses the reactor enclosure. It is made of interlocking heavy concrete blocks with a reference composition following the PNNL-15870R1 [Reference 4.8-3] compendium specifications for M-1 heavy concrete. It is termed sacrificial shield because it will activate, and at the same time is intended to reduce activation of permanent structures (systems pit floor and walls) external to itself. It is designed to facilitate its ultimate decommissioning as well as replacement, if needed. It is made of heavy concrete to reduce space requirements.

4.4.8 Biological Shield Outer Layer (Systems Pit Floor and Walls, Top Plug)

The outer layer of the biological shield corresponds to the systems pit walls and floor and the top plug. They are made of ordinary concrete, with a reference composition following the PNNL-15870R1 [Reference 4.8-3] compendium specifications for “Concrete, Oak Ridge” with density of 2.3 g/cc. Systems pit floor and walls form a permanent structure containing rebar and are 121.92 cm (4 ft) thick. They reduce soil activation and prevent side-streaming of scattered radiation. It reduces the dose rate in the bay building to the required level.

4.4.9 Additional Shielding

Limited size areas just above the reactor (roughly above the center of the top plug) and in the adjoining systems pit cells (opposite the reactor centerline), exhibit higher dose rates than those in the bulk of the bay building volume. Additional shielding can be added there to bring these dose rates in line with the rest of the space in accordance with the As Low As Reasonably Achievable (ALARA) principle. This can be achieved by adding a layer of ordinary concrete, but because the dose rate after a thick shield tends to be primarily due to photons, a steel or lead plate is a more effective solution.

4.4.10 Structural Considerations

To ensure adequate structural strength and integrity over their lifetime, the following requirements are imposed on the biological shield components:

- Energy fluxes are limited to less than 10^{10} MeV/cm^{2}s.
- Neutron fluence is limited to 10^{18} n/cm^2.
- Integrated dose of gamma radiation is limited to 10^{10} rads.

4.4.11 Radiation Zones

Three radiation zones are identified:

A. Reactor Cell: The reactor cell is normally inaccessible, limiting personnel radiation exposure in this high-dose area. The operating dose limits protect the concrete.

B. Covered Systems Pit: There will be limited and controlled access to the covered systems pit cells adjoining the reactor cells. The target dose rate is not to exceed 2.5 mrem/hr, but a higher rate is acceptable combined with a limited access time to ensure compliance with 10 CFR Part 20.

C. Research Bay: Those areas inside the bay but outside the systems pit have a dose limit of 2.5 mrem/hr. Further optimization of the radiation shield will be performed and presented in the Operating License application to reduce the limit in most (or all) of this volume not to exceed 0.01 mrem/hr, which would allow uncontrolled access.

Cross-sectional views of the systems pit and research bay are depicted in Figure 4.4-2.

4.4.12 Calculated Dose Rates and Other Relevant Shielding Parameters

Neutron, gamma, and total dose rates are calculated throughout the research bay and the systems pit. Dose rates are calculated for all source terms listed in design bases. The dose rate from all secondary sources combined (irradiated or activated circulating fuel salt source term, photon source term due to activation of components and structures), compared to the primary dose rate (from fission neutrons, fission photons, and capture photons) is more than 10 times smaller, (i.e., it contributes less than 10 percent to the total dose rate in zones B and C). The primary dose rate is, therefore, representative of the total dose rate in areas B and C.

The dose rate limit requirement of 2.5 mrem/hr is met everywhere in the C zone and everywhere in the B zone with the addition of a small amount of shielding.

Dose rates are calculated at the research bay building boundary (i.e., external walls) to ensure the dose to the general public at unrestricted locations is below the limit. While not all walls in their entirety actually allow uncontrolled access to the general public, the target is to treat them all as such and aim to reduce the dose rate below 0.01 mrem/hr.

Dose rate measurements are performed during the initial low-power operation at about 1 percent of full power to confirm the shielding calculation and to identify if additional shielding needs to be added to guarantee the dose rate in the whole zone C below 0.01 mrem/hr.

4.4.13 Uncertainties in Shielding Results

Engineering estimates of the overall uncertainty in the presented shielding analysis is that the results are accurate within a factor of four. This is driven primarily by the uncertainty in the as-built materials (density and composition of heavy concrete), with additional Monte Carlo statistical uncertainties. These uncertainties will be reduced, and a more detailed uncertainty quantification will be presented in the Operating License application.

4.4.14 Environmental Conditions

Both the ordinary and heavy concrete will be kept at temperatures below 65 degrees Celsius, using air cooling as described in Chapter 6. Moreover, the energy flux to the structural concrete (systems pit walls and floor) is at least an order of magnitude below the 10^{10} MeV/cm^{2}s guideline.

Maximum neutron fluence to the structural concrete is about four times below 10^{18}n/cm^2, while the gamma dose is about two orders of magnitude below 10^{10} rads. These safety factors provide a sufficient margin to account for relatively large uncertainties.

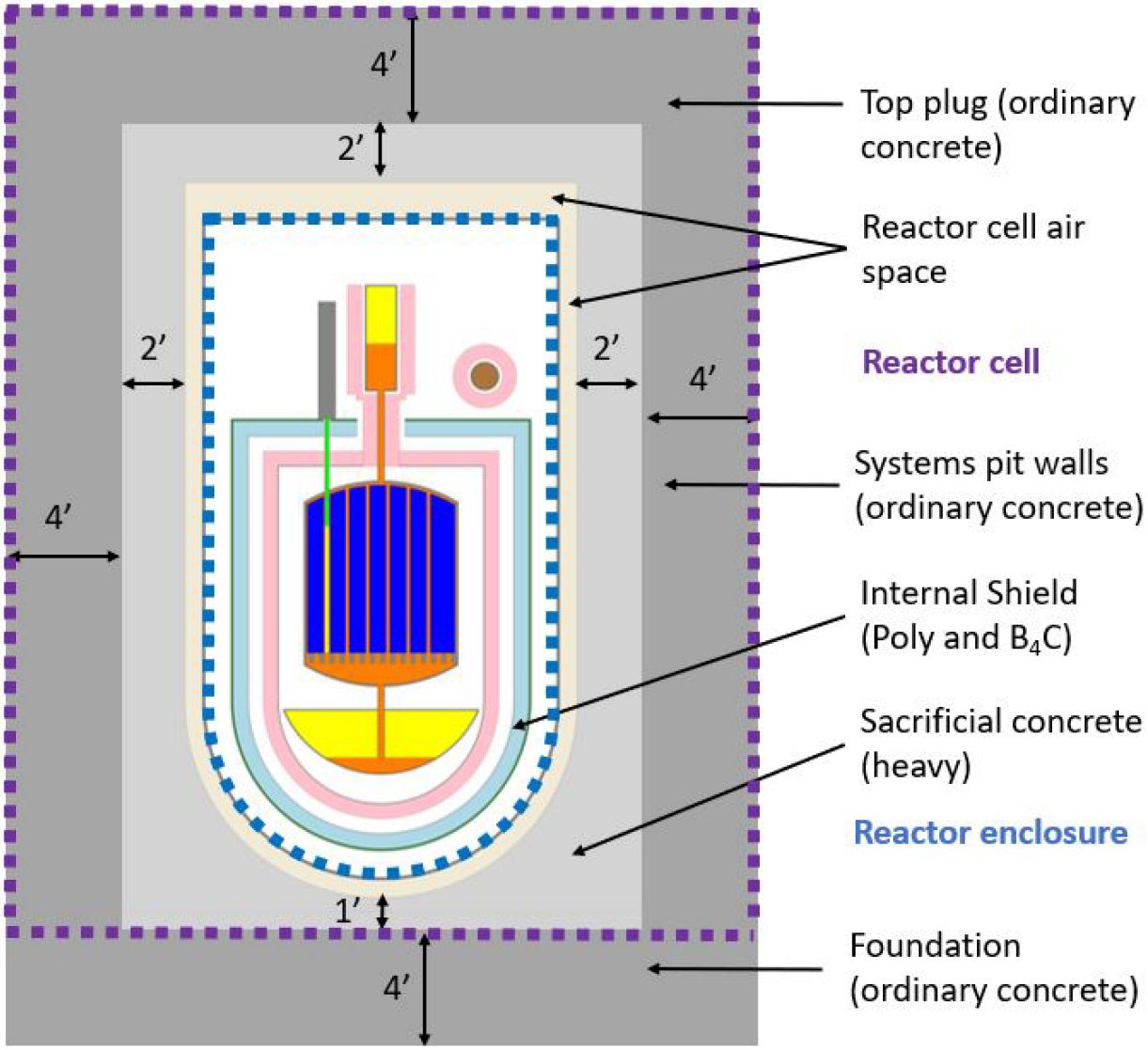
4.4.15 Soil Activation

The biological shield significantly attenuates radiation reaching the soil beneath the systems pit floor and next to the systems pit walls. For the average U. S. soil composition (PNNL Compendium [Reference 4.8-3]), average specific activity of the one-foot-thick soil layer next to the reactor cell floor and walls is less than 0.5 Bq/g (neglecting the naturally occurring ^{40}K), and, in principle, qualifies for free release. Potential activation of groundwater will be addressed in the Operating License application.

4.4.16 Air Activation

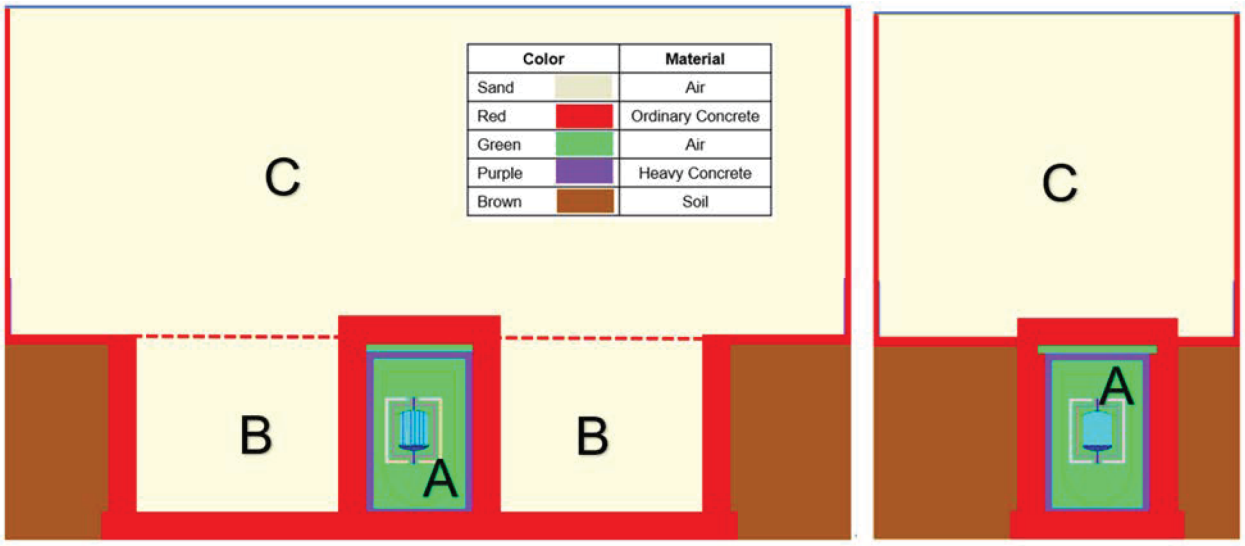
Any activation products in air are handled by the auxiliary heat removal system and the research bay heating, ventilation, and air conditioning (see Chapter 6 and Chapter 9).

Figure 4.4-1 Configuration and Thickness of Biological Shield Layers



Dimensions provided are approximate. Actual dimensions will be reported in the Operating License application.

Figure 4.4-2 Vertical Cuts (XZ and YZ) of Bay Building and Systems Pit



Depicts zones with differing dose rate limits.

4.5 Nuclear Design

4.5.1 Description of Nuclear Design

This section of the PSAR contains the preliminary nuclear design, which will be finalized in the application for an Operating License. The nuclear core is composed of the fuel salt flowing through the moderator composed of a hexagonal matrix of graphite blocks. Unmoderated fuel salt is subcritical in all other configurations and locations. The reactivity of the moderated fuel salt is affected by the following:

- System pressure and corresponding void fraction (gas bubbles)
- Fuel salt composition accounting for addition and depletion of UF_4
- Control rod position
- Flow of delayed neutrons out of and into the core
- Temperature effects
- Power profile

Operational control is provided primarily by the use of control rods. Reactor shutdown is accomplished by draining the fuel salt out of the moderator.

4.5.1.1 Design Basis

The design bases related to nuclear design are as follows:

- Consistent with DC 10, the reactor core has appropriate margin to ensure the specified acceptable system radionuclide release design limits are not exceeded.
- Consistent with DC 11, the reactor core is designed so that in the power operating range the net effect of prompt, inherent, nuclear feedback tends to compensate for rapid increase in reactivity.
- Consistent with DC 12, the reactor core ensures power oscillations that can result in conditions exceeding specified acceptable system radionuclide release design limits are not possible or can be reliably and readily detected and suppressed.

4.5.1.2 Codes

The nuclear design work is performed using three independent codes: SCALE, Serpent 2, and MCNP. The MCNP and Serpent 2 results provide an independent evaluation of SCALE for benchmarking and confirmatory analysis.

4.5.1.2.1 SCALE

SCALE, Version 6.2.4, as developed by Oak Ridge National Laboratory and released for licensing and distribution by the Radiation Safety Information Computational Center, is the primary code for neutronics analysis. This version, released in August 2016, is capable of all the required neutronics calculations.

4.5.1.2.2 Monte Carlo N-Particle Code

MCNP Version 6.1, developed by Los Alamos National Lab and released for licensing and distribution by the Radiation Safety Information Computational Center, is used for supporting neutronic and gamma transport analysis. Also, NJOY 2016 is used for select cross section processing and generation of ACE files for the MCNP analyses.

4.5.1.2.3 Serpent 2

Serpent is a multi-purpose, three-dimensional, continuous-energy Monte Carlo particle transport code developed at VTT Technical Research Centre of Finland, Ltd. The development started in 2004, and since 2009 the code has been publicly distributed by the OECD/NEA Data Bank and RSICC. Serpent started out as a simplified reactor physics code, but the capabilities of the current development version, Serpent 2, extend well beyond reactor modeling. Serpent 2 is used in this project for criticality calculations, fuel cycle studies, calculations with CFD, radiation dose rate calculations, and shielding analysis.

4.5.1.3 Code Validation Plan

A reactor physics model of the MRSE using MCNP, SCALE, and Serpent is available upon request. Code calculated k_{eff} are within one standard deviation between the three models. Reactivity coefficients are similar.

The reactor physics tools and analytical techniques are confirmed during the reactor startup, zero power physics testing, and lower power operation.

- The reactor is brought to a critical state by the slow addition of UF_4 at a given temperature and control rod height. Flowing fuel salt is required to mix the UF_4. The approach to criticality is modeled and compared to experimental results.
- Once critical, the total reactivity coefficient can be determined and compared to reactor physics modeling results by controlling the temperature of the reactor vessel through electrical heaters. Fuel salt may or may not be flowing.
- The effect of the delayed neutrons being swept out of the core is experimentally tested and compared to models.
- Radionuclide content of the reactor head space is measured at low power.

These procedures will be defined in detail and presented in the Operating License application.

4.5.2 Normal Operating Conditions

4.5.2.1 Limiting Parameters That Are of Safety Interest

The limiting parameters relevant to safety are:

- Highest power density
- Largest source term
- Highest excess reactivity (low temperature, high voids, fresh fuel)

The largest source term and greatest power density occur during maximum licensed power at the end-of-life assuming no off-gas system. Without an off-gas system in operation, gaseous fission products and their daughters build up in the reactor headspace. At maximum licensed power, the gaseous fission product concentration is maximized. After the maximum cumulative run time of 5 MW-years, the higher actinide concentration is maximized, as well as select, long-lived gaseous isotopes (mainly H-3 and Kr-85). A small quantity of helium is entrained within the fuel salt during operation; therefore, the conditions that lead to the highest power density and the greatest source term for the MSRR are

- reactor power of 1 MW_{th}.
- continuous critical operation for operating life of the reactor.
- off-gas system not in operation (gaseous fission products build up in reactor head space).

Conversely, the highest excess reactivity occurs with the reactor at minimal temperature and maximum voids. Therefore, the reactor operating conditions that would lead to the highest excess reactivity would occur at the initial criticality, with no fission products, at the lowest temperature possible, and with maximum voids in the fuel salt. These conditions are summarized as:

- Reactor vessel temperature is at minimum limit.
- Reactor pump is operating and fuel salt is flowing.
- Control rods are adjusted to attain criticality.
- Void fraction is at the maximum limit.

Power density, source term, or excess reactivity will not be directly limited during operation of the MSRR. The MSRR will be designed and licensed where those parameters are limited through control of reactor power, control rod height, and reactor temperature. Void fraction can not be directed controlled, but can be surveyed and demonstrated to be within operating limits. By maintaining reactor power, control rod height, and reactor temperature within operating limits, the power density, source term, and excess reactivity will by necessity be managed.

4.5.2.2 Impact of Gas Management System on Reactor Operation and Safety

The GMS is discussed in Section 9.6. Gaseous fission products are discussed in Section 4.2.1. The impact of this system on reactor operation and safety is discussed below.

As explained in Section 9.6, helium is used to control the pressure within the RAV, the reactor pump head space, and the drain tank head space. Pressures within these components' head space define the level height of the fuel salt in the reactor. Helium is actively bubbled into the fuel salt and removed from the fuel salt as part of the GMS. Some of this helium can become entrained in the fuel salt and flow into the reactor vessel. A fraction of fission gases, as bubbles or as diffuse atoms, come out of the fuel salt into the helium head space in the RAV and the reactor pump.

The reactor gas management system is not designed to affect void fraction during normal operation. As mentioned in Chapter 10, the bubbles may be introduced into the fuel salt through a "bubbler." Helium is supplied by the reactor gas management system. This component is experimental and not tied to an essential reactor function. Technical Specifications to monitor reactivity (which will indicate void fraction) during bubbler operation will be provided in the operating license application.

There are a couple of means by which the GMS impacts reactor operation and safety: (1) changes in the gas pressure and volume can impact the reactivity of the system and (2) in a loss of functional containment accident, the cover gas includes the radiation source terms from the gaseous fission products.

In the MSRE, the circulating void fraction is estimated to be between 0.02 percent and 0.04 percent [Reference 4.8-4]; by mechanical similitude, the MSRR is expected to have a void fraction of 0.04 percent, which corresponds to a pressure reactivity coefficient of 0.031 pcm/kPa. Figure 4.5-1 shows the behavior of the pressure coefficient relative to the void fraction.

Additional analyses have shown that the void coefficient of reactivity, at the MSRR operational temperature, 600 degrees Celsius, α_f, was -168 pcm/%f. An independent analysis of the void coefficient of reactivity provides a different value, -217 ± 27 pcm/%f. The value that is greater in magnitude is used for the safety analysis. The void coefficient of reactivity at 600 degrees Celsius should encompass the entire operating range.

4.5.2.3 Reactivity induced by fuel salt composition changes

The fuel salt contains its highest concentration of fission products, corrosion products, contaminants, and actinides at the end of life. The UF_4 concentration increases over lifetime to compensate for the depletion of U-235. Figure 4.5-2 shows the change in k_{eff} with respect the addition or removal of UF_4 from the reactor. Solid slugs of UF_4 will be dropped into the fuel salt through the RAV

where they slowly dissolve. One kg of UF_4 will induce approximately 33 pcm of reactivity. Reactor operations can easily compensate for such small additions of reactivity. Technical Specifications for the addition of UF_4 will be provided in the operating license application.

The change in reactivity induced by burnup without the addition of UF_4 is supplied in Figure 4.5-3. The MSRR is simulated at full power continuous operation for five years without adjustment of the CRs or addition of UF_4. All fission products are modeled in the fuel salt. Reactivity versus burnup with gaseous fission product removal will be evaluated and reported in the Operating License application. This curve is used to generate the fuel addition plan over the reactor lifetime.

A perturbation study demonstrating that changes in redox potential and buildup of corrosion products do not meaningfully impact reactivity will be presented in the Operating License application.

4.5.2.4 Power Oscillation Stability Analysis

As outlined in Ball and Kerlin [Reference 4.8-5], molten salt reactors are subject to oscillations. Power oscillations in the MSRR will be analyzed and results reported in the Operating License application to demonstrate that power oscillations do not impact the MSRR safety basis.

4.5.2.5 Control Rod Parameters

The three CRs operate independently of one another with a total worth of -3640 pcm of reactivity, which is sufficient to achieve $k_{eff} < 0.99$ with fuel salt in the reactor at 550 degrees Celsius.

The change in power density within the reactor induced by the movement of the CRs will be evaluated and reported in the Operating License application.

4.5.2.6 Analysis Showing the Fuel Salt Outside the Reactor Vessel Remains Subcritical at All Times

The only location within the MSRR that fuel salt can approach criticality is the reactor vessel. Without moderation, the fuel salt cannot become critical. To demonstrate, fuel salt is modeled within the drain tank and the FHS. When the entire fuel inventory is located in the drain tank and is at room temperature, k_{eff} is deeply subcritical (~0.6), considering both beginning and end of life fuel salt compositions with fission product buildup, UF_4 addition, and plutonium generation. It is possible that the entire fuel salt inventory can be located in a single tank in the FHS. Flooding is a remote concern in the MSRR, but in the event that a tank of fuel salt located in the FHS is submerged in water, k_{eff} remains deeply subcritical.

4.5.2.7 Reactor Kinetics Effects of Delayed Neutrons

The reactor kinetics effects stemming from the movement of delayed neutron precursors are described in this section. The information in this section as well as the modeling of such behavior was performed using RELAP5-3D.

The standard point kinetics model for static fuel is modified to account for the movement of delayed neutron precursors out of the reactor core. The detailed spatial distribution of the neutron precursors within the reactor vessel and how this could impact neutron importance is ignored. All delayed neutrons born within the core have the same neutron importance of 1, while all neutrons born outside the core have a neutron importance of 0.

For steady-state conditions with variable flow speeds, the reactivity loss from delayed neutrons decaying in the ex-core loop is evaluated, as shown in Table 4.5-1.

The static fuel delayed neutron fraction, Beta, has been calculated to be 672 pcm. With 100 percent flow of the delayed neutron, that fraction drops by 69 pcm; therefore, the abrupt loss of flow induced by the sudden stoppage of the reactor pump inserts 69 pcm of reactivity over the course of seconds. This transient (provided in Chapter 13) does not result in a breach of safety limits. The subjects of probable technical specifications that control important design features, limiting conditions for operation, and surveillance requirement are discussed in Chapter 14.

4.5.3 Active Reactor Core Physics Parameters

4.5.3.1 Reactor Physics Kinetics Parameters

Prompt neutron lifetime (259.3 μs) for the initial fuel salt composition is calculated using SCALE. Prompt neutron lifetime at end of life (accounting for burnup and UF_4 addition) will be reported in the Operating License application.

Reactivity coefficients account for changes in reactor vessel temperature and fuel salt pressure. Temperature coefficients are calculated for the fuel salt, graphite, stainless steel, and the entire (isothermal) reactor vessel. The fuel salt coefficient accounts for the change in density and change in cross sections. The graphite coefficient accounts for the change in cross section only. Graphite thermal expansion is simulated in Serpent 2; this phenomenon has minimal impact on reactivity and is ignored for subsequent analyses. Thermal expansion of the vessel allows more fuel salt to flow around the graphite core, increasing reactivity. The stainless steel reactivity coefficient accounts for the change in cross section and for thermal expansion. Reactivity coefficients have been calculated in MCNP. The values and 1 sigma stochastic uncertainty are reported in Table 4.5-2.

The reactivity coefficients have been calculated independently in SCALE, see Table 4.5-3.

Stainless steel 316H has a large thermal expansion coefficient relative to graphite. As the reactor vessel increases in temperature, the amount of fuel salt in the vessel increases, filling the space between the graphite and the reactor vessel wall. The vessel expansion coefficient (also called the stainless steel reactivity coefficient) is, therefore, positive. The vessel temperature coefficient is comparatively slow acting. In the event of an increase in reactor power, the fuel salt increases in temperature first, followed by the graphite, then by the vessel itself. The sum of the fuel, graphite, and vessel coefficients is strongly negative. Vessel expansion feedback is not currently considered in the safety analysis because the RTMS (see Chapter 6) acts to maintain reactor vessel temperature somewhat independently of fuel and graphite temperature. This phenomenon will be fully explored and elaborated upon in the Operating License application.

The SS316H reactivity coefficient (or the reactor vessel coefficient) has been determined to be 1.6 ± 0.3 pcm/K.

The pressure coefficient is caused by the presence of voids within the fuel salt. The void coefficient itself is reported in Section 4.5.2.2.

Delayed neutron parameters are discussed in Section 4.5.2.7.

4.5.3.2 Power Profile in Reactor Vessel

SCALE is used to calculate energy deposition within the reactor vessel fuel salt, graphite, and structure. Axial and radial power profiles are generated and are provided below. This data is used to generate power profiles for the RELAP5-3D coarse nodalization. The maximum power density within the fuel salt is 17.6 MW/m^3, while the average power density is 2.45 MW/m^3. See Figure 4.5-4 and Figure 4.5-5.

Movement of the control rods induces perturbations within the power profile. These changes do not meaningfully impact the power peaking factors. This will be demonstrated in the Operating License application.

It is not expected that changes in fuel salt composition impact the power profile throughout the reactor because the fuel salt is homogeneous throughout the loop (aside from delayed neutron precursors). A flow distribution device located in the lower plenums will be used to distribute fuel salt flow in the reactor vessel to produce a more even temperature distribution of fuel salt.

4.5.4 Operating Limits

4.5.4.1 Operational Limits on Pressure and Void fraction

This section establishes operational limits to ensure that there cannot be a reactivity insertion accident that exceeds the maximum established in Chapter 13. These operational limits include the following:

- The maximum void fraction in the fuel salt is less than 1.8 percent of the fuel salt volume. Procedures, techniques, and analysis that ensure the reactor cannot be in this state will be presented in the Operating License application.
- Maximum pressure of the fuel salt in the reactor vessel is 500 kPa. This limit is established to protect the reactor system structure, but also limits possible reactivity insertions. Pressure of the fuel salt is not directly measurable, but can be reliably determined through pressure measurement of the reactor head space and the drain tank head space.

As noted previously, in the event of accident-causing entrained bubbles in the fuel salt to completely collapse and become dissolved within the fuel salt, the fuel salt density rapidly increases, resulting in a rapid reactivity insertion. Based on MSRE operational data, the void fraction of the MSRR is expected to be less than 0.04% during operation. It is appropriate to establish a larger limit for this parameter. A void fraction of 2% was noted in the MSRE during abnormal operating modes, modes which may be induced in the MSRR to better understand its performance. Also, expanded operating bounds that can be shown to be safe should be established in the Technical Specifications to allow this research reactor to examine molten salt reactor behavior, including off-normal conditions. For these reasons, a void fraction of 1.8% was selected as the operating limit. An analysis of a bubble collapse accident is presented in Chapter 13 and bubble collapse is considered in analysis of the maximum reactivity insertion event.

Void fraction will be measured during zero power testing when the reactor is critical, which will provide a baseline during operation. Deviations in excess reactivity (as reflected in control rod height and temperature) will be noted during operation and compared to the expected change in reactivity from the depletion of uranium and build-up of fission products.

As noted previously, if there is an inadvertent increase in the cover gas pressure, there is an instantaneous reduction in the volume of any bubbles in the fuel salt, which increases the reactivity of the system. The maximum reactor pressure has the capacity to induce 356 pcm of reactivity under worst case conditions. This reactivity insertion is less than that induced by the sudden loss of voids.

As noted previously, it is hypothetically conceivable that if the reactor were to instantly lose pressure, the xenon entrained in the fuel salt can be released to the cover gas or the environment. While this is not physically possible, it is important to demonstrate that even in the worst-case scenario, the reactivity insertion due to the loss of fission gases does not exceed the limit defined in Chapter 13. The maximum worth of all xenon and krypton in the fuel salt is -120 pcm, which is significantly below the maximum reactivity insertion limit.

4.5.4.2 Control Element Limits

The control elements are not needed for the safe shutdown of the reactor. The purpose of control elements is to control reactivity during reactor startup and to make adjustments during operation. For these reasons, there are comparatively few limits associated with the control elements so long as the reactor is brought

critical during initial startup with UF_4 addition at the control rod design height. Minor deviation from the design height and design control rod worth are allowed. These limits will be reported in the Operating License application.

4.5.4.3 Definition of Excess Reactivity

Excess reactivity is affected by temperature, fission and helium gas voids, pressure, and pump operation. As these parameters will affect reactivity, they may be subjects for Technical Specifications. The most important of these is temperature. Compensation for changing conditions requires control rod movement to maintain criticality. The excess reactivity at each possible state is bounded by the following.

- For an average reactor temperature of 630 degrees Celsius, the excess reactivity during operation at 1 MWth is 70 pcm.
- For an average reactor temperature of 600 degrees Celsius, the excess reactivity during operation at 1 MWth is 420 pcm.
- For an average reactor temperature of 570 degrees Celsius, the excess reactivity during operation at 1 MWth is 780 pcm.

The average reactor temperature does not fall below 570 degrees Celsius during 1 MWth operation without violating the fuel salt low temperature limit. An average reactor temperature of 570 degrees Celsius corresponds to reactor vessel inlet temperature of 550 degrees Celsius and a reactor vessel outlet temperature of 590 degrees Celsius.

Excess reactivity is explained by considering maximum reactivity insertions during 1 MWth operation.

4.5.4.4 Maximum Reactivity Insertions

The nominal baseline operating conditions are:

- Full power, fuel salt flowing
- Average reactor vessel temperature of 600 degrees Celsius
- Critical
- No voids
- 150 kPa pressure in the reactor vessel

If the reactor increases in temperature by 30 degrees Celsius, a temperature coefficient of -12 pcm/K induces approximately 360 pcm of negative reactivity, and the control elements must be withdrawn to compensate. Since the control elements have at most +350 pcm, the average reactor temperature cannot be increased much beyond this point.

If the reactor decreases in temperature by 30 degrees Celsius, 360 pcm of positive reactivity is inserted and the control elements must be inserted to compensate. The rate of change of reactor temperature is mitigated by the large thermal mass of the fuel salt and reactor vessel. Temperature feedback is explicitly modeled with RELAP5-3D (see Chapter 13).

If the void fraction increases to 1.8 percent, approximately 400 pcm of negative reactivity is inserted and the control rods must be withdrawn to compensate. The maximum reactivity worth induced by a change in pressure is 356 pcm.

If the reactor is operating and the pump suddenly stops, 69 pcm of positive reactivity is inserted.

Control elements cannot be rapidly withdrawn or ejected by design. The maximum rate of positive reactivity insertion is 10 pcm/sec.

Maximum reactivity insertions are provided at three different reactor temperatures. These conditions describe the excess reactivities possible.

A maximum positive reactivity insertion is described by:

- Reactor is operating at full power at an average temperature of 630 degrees Celsius.
- Control rods are almost fully withdrawn to compensate for the increase in reactor temperature.
- Reactor in the aforementioned state cannot have voids, as voids will insert negative reactivity, and there is no positive worth in the CRs to compensate. Because the reactor cannot have voids, the pressure coefficient is not relevant.
- Pump suddenly stops inserting +70 pcm.
- Reactor control system malfunction is not applicable as the rods are withdrawn.

Another maximum positive reactivity insertion is described by:

- Reactor is operating at full power at an average temperature of 600 degrees Celsius.
- Pump suddenly stops inserting +70 pcm.
- Reactor has voided up to the limit; CRs are fully withdrawn to compensate.
- Reactor suddenly loses voids inserting +350 pcm because of the change in density. Void worth is limited by the allowable CR worth compensating.
- Reactor control system malfunction cannot insert positive reactivity as rods are fully withdrawn.

A further maximum reactivity insertion is described by:

- Reactor is operating at full power at an average temperature of 570 degrees Celsius.
- Control rods are inserted to compensate for lower temperature.
- Pump suddenly stops inserting +70 pcm.
- Reactor has voided up to the limit; CRs are partially withdrawn to compensate.
- Reactor suddenly loses voids inserting +400 pcm because of the change in density. Xenon migrates with the helium, inserting +120 pcm of reactivity.
- Reactor control system fails, gradually inserting positive reactivity at a rate of 10 pcm/sec. The total CR worth is the design CR worth plus the CR worth introduced from temperature compensation minus the CR worth compensating for voiding (+350 pcm + 360 pcm - 400 - 120 pcm = 190 pcm).

4.5.4.5 Shutdown Conditions

Shutdown of the reactor requires the majority of the fuel salt to be physically located in the drain tank. From there, with the drain tank head space and the reactor headspace physically connected by the open SCRAM valves, the reactor cannot become critical by any means.

Fuel salt present within the drain tank can be verified in a number of different ways.

- Pressure of the drain tank head space and the pressure of the reactor head space can be measured. If pressures are approximately equal, then the fuel salt will have relocated by gravity.
- Valves connecting the head spaces can be monitored. If at least one valve is open, then pressure has equalized, and the fuel has relocated.
- Level control sensors in the drain tank and the RAV allow for direct measurement of fuel salt location.
- The reduction in neutron source strength from draining is evident once the fuel salt drains.

4.5.4.6 Limiting Core Configuration

The maximum power density of the reactor does not change with changes in fuel salt composition nor movement of the control rods. Power density is proportional to reactor power, which is actively monitored and controlled. Flow distribution and the power distribution within the reactor vessel define the maximum temperature of the reactor system barrier, which is the relevant safety criteria for the limiting core configuration.

Given these differences between molten salt reactors versus solid-fueled reactors, it is concluded that the MSRR does not have a limiting core configuration. Power density is controlled by the systems to control reactor power. No other mechanism alters the maximum power density. Hot spots within the

upper reactor vessel (location of the maximum reactor system barrier temperature) are monitored during operation and can be accurately modeled with appropriate tools.

Figure 4.5-1 Variation in Pressure Coefficient of Reactivity with Respect to Void Fraction

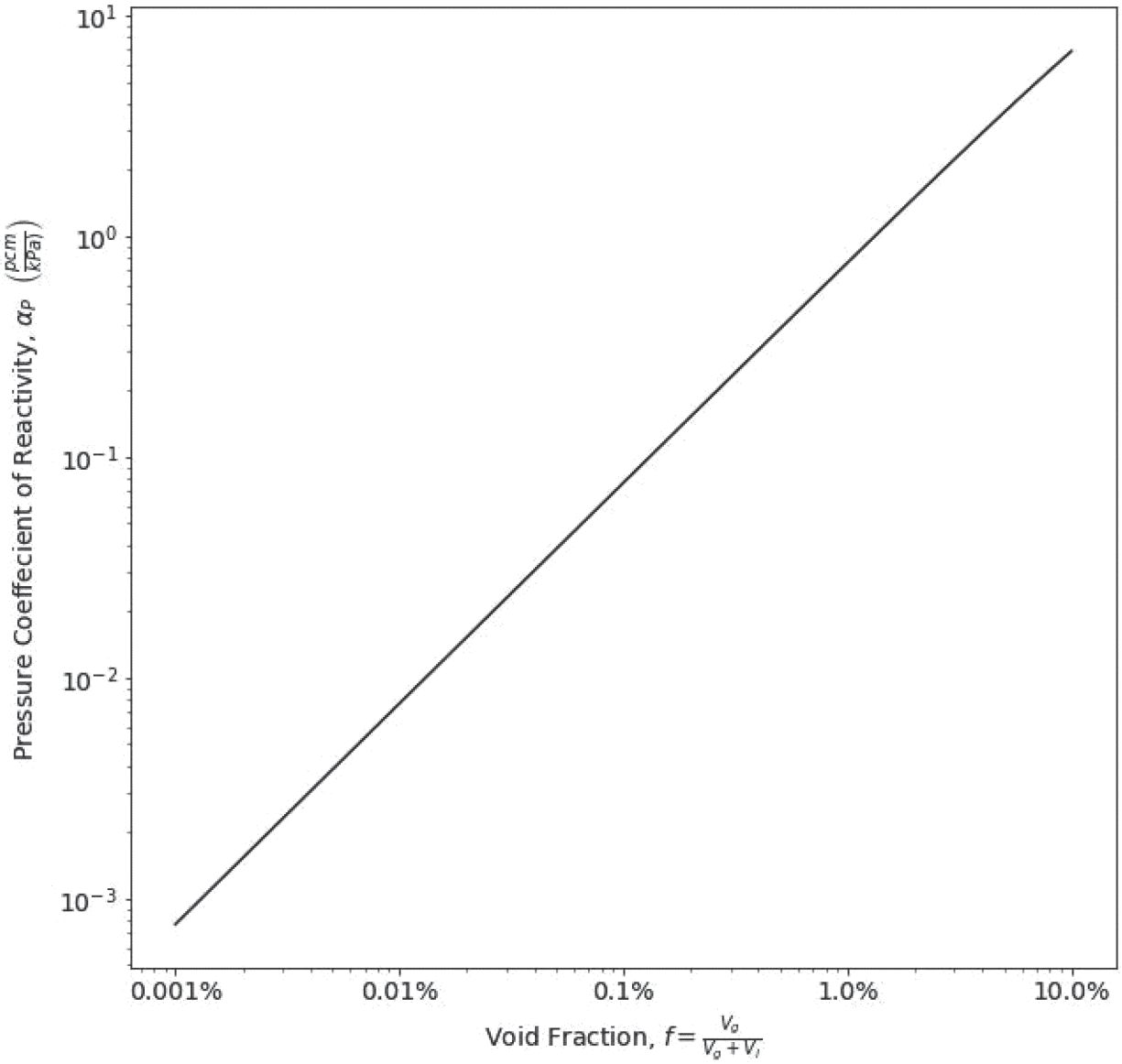


Figure 4.5-2 Change in K_{eff} with Respect to Addition or Removal of UF_4 from Reactor

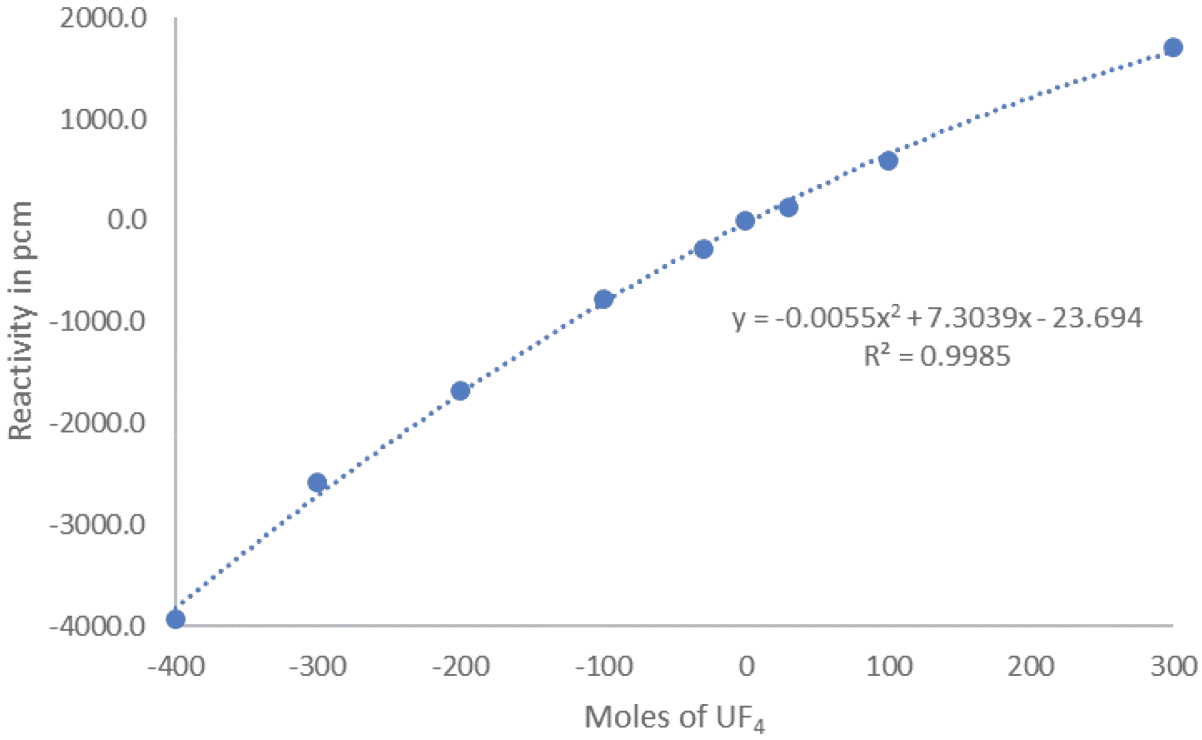


Figure 4.5-3 Loss of Reactivity Caused by Fuel Burnup

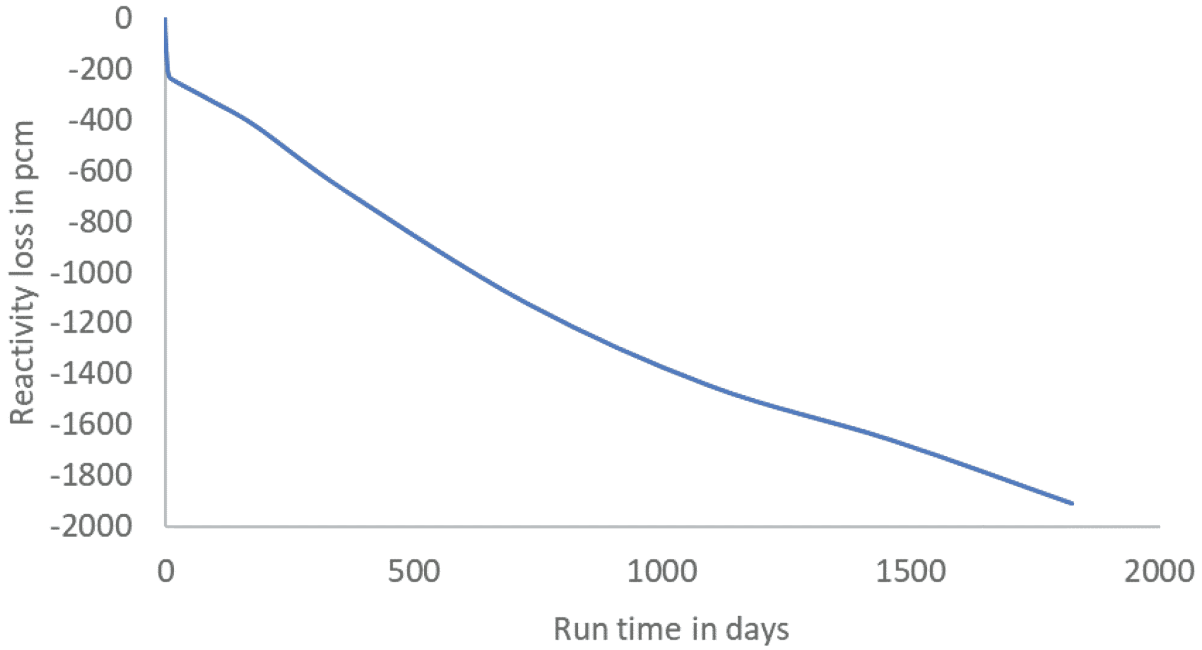


Figure 4.5-4 Axial Power Profile through the Entire Reactor Vessel

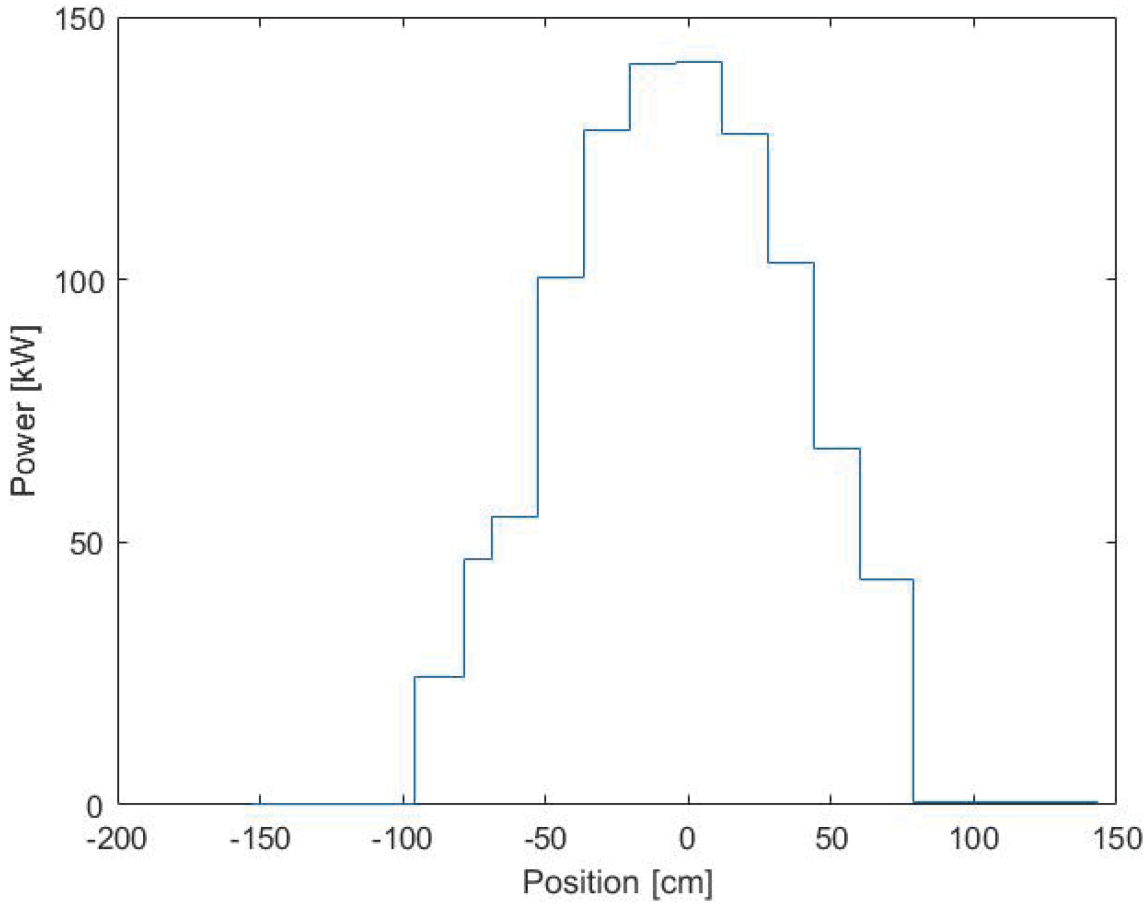


Figure 4.5-5 Radial Power Profile through Reactor Vessel Core Region

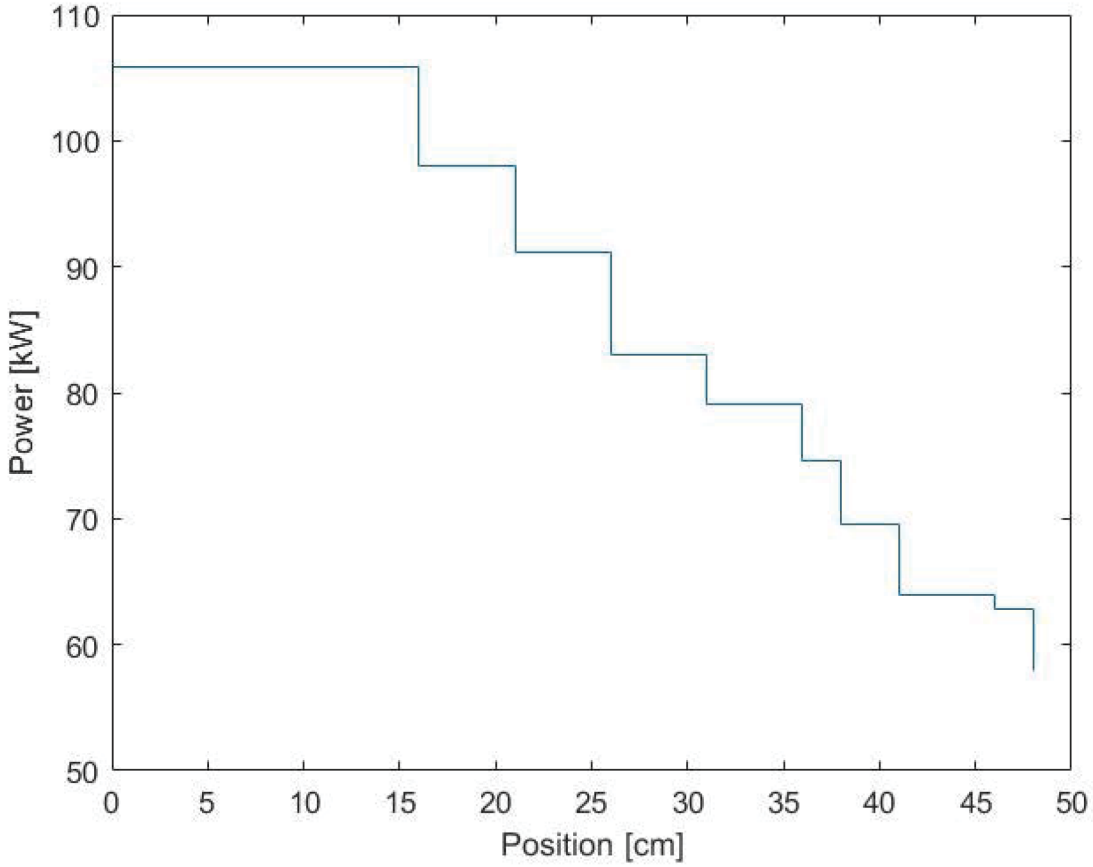


Table 4.5-1 Reactivity Loss Stemming from the Delayed Neutron Precursors Outside the Reactor Vessel

Flow (% nominal)	React Loss ($)
100	0.1020
50	0.0751
20	0.0452
10	0.0277
1	0.0032

Table 4.5-2 Reactivity Coefficients Calculated in MCNP with 1 Sigma Stochastic Uncertainty

Coefficient	Value with 1 sigma uncertainty
Fuel	-6.46 ± 0.085 pcm/K
Graphite	-5.39 ± 0.078 pcm/K
Integral	-12.045 ± 0.078 pcm/K

Table 4.5-3 Reactivity Coefficients Calculated in SCALE

Coefficient	Value
Fuel	-6.26 pcm/K
Graphite	-5.16 pcm/K
Integral	-11.94 pcm/K

4.6 Thermal Hydraulic Design

This section presents the information and analyses that show cooling capacity is sufficient to maintain salt in the reactor loop at temperatures that will not damage the reactor system and that, therefore, the integrity of the reactor system will be maintained in all foreseeable operating conditions.

Consistent with DC 34, provisions are made for heat removal during normal operations and anticipated operational occurrences such that specified acceptable radionuclide release design limits and the design conditions of the fuel salt boundary are not exceeded.

Consistent with DC 35, provisions are made to ensure sufficient fuel salt cooling during postulated accidents and to remove residual heat following postulated accidents is provided.

4.6.1 Heat Removal Systems

4.6.1.1 Thermal-Hydraulic Design

The heat removal systems in the MSRR consist of a reactor loop that transfers heat to the coolant loop through the heat exchanger. The heat exchanger is a component of the reactor system and is located within the reactor enclosure. Both are shown in Figure 4.6-1. The coolant loop then transfers heat to atmospheric air through a radiator. Most of the coolant loop, including the coolant pump, piping, and radiator, is located within the coolant loop enclosure.

In addition to this heat removal system, an auxiliary heat removal method removes heat from the reactor enclosure and the reactor cell. In the event that fuel is drained from the reactor loop to the drain tank, sufficient surface area and appropriate conditions around the drain tank can remove decay heat indefinitely.

4.6.1.2 Reactor Loop and Heat Exchange

Almost all of the reactor power is deposited within the reactor vessel, either directly within the fuel salt or within the reactor vessel structures, which are cooled by the fuel salt. Hot fuel salt is then pumped into the RAV, through the reactor pump, and into the heat exchanger. The fuel salt (outside the tubes in the shell side of the heat exchanger) exchanges heat with the coolant salt (inside the tubes) through the tube walls. The fuel salt, now cooled, flows back into the reactor vessel through the cold leg. This fluid circuit is also called the reactor or primary loop.

The drain tank is attached to the cold leg, and it is discussed in more detail in Section 5.5.4. Its RPS functions are described in Section 4.2.2. The drain tank and reactor vessel are located in the RTMS, which is described in Section 6.2.4.

4.6.1.3 Coolant Loop and Heat Exchange

The coolant loop flows through the tube side of the heat exchanger, increasing in temperature before returning through the coolant pump to the radiator. The radiator in this loop is sized to eliminate 1 MW of heat continuously to the anticipated, slightly elevated, ambient temperature.

The coolant salt is less complicated than the fuel salt because it does not contain any fissile material. The coolant loop is located within the coolant loop enclosure rather than the reactor enclosure.

The coolant loop contains a coolant drain tank where coolant is stored for use on demand. In the case of shutdown or malfunction, the salt in the coolant loop is dumped into the coolant drain tank in order to avoid its freezing in the coolant loop pipes.

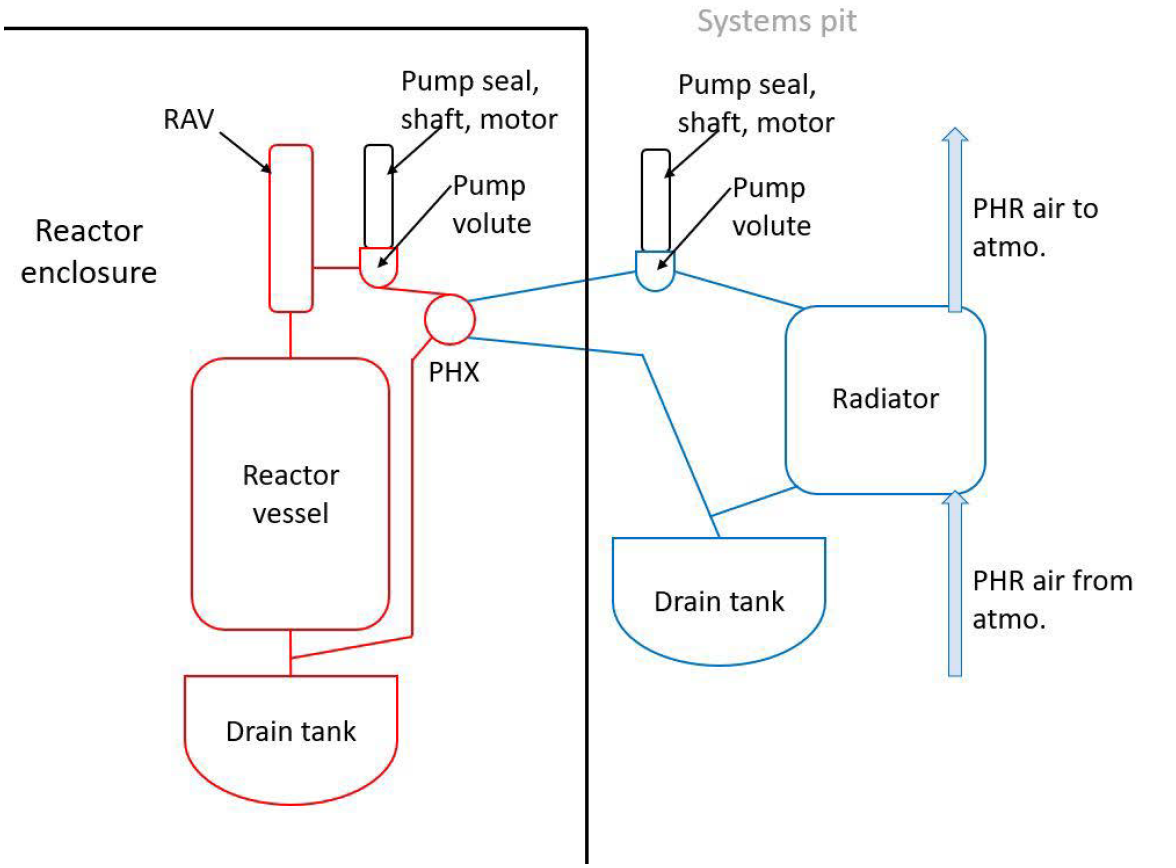
4.6.2 Thermal-Hydraulic Methodology

The main thermal-hydraulic software that is used for licensing is RELAP5-3D. A full model of the reactor loop is produced in RELAP5 and updated and modified as the design process progresses. This RELAP5 model uses point kinetics and thermal feedbacks that are informed by physics modeling in SCALE. Calculations performed outside of RELAP inform the heat exchanger, drain tank, and radiator designs. External calculations are then verified within RELAP for accuracy and consistency between methods.

The heat exchanger was sized and designed using the effectiveness-NTU method. The Gnielinski Nusselt number correlation for flow in tubes is used within the tube side of the heat exchanger. Equation 7.58 [Reference 4.8-6] is used for the shell side Nusselt correlation. The heat exchanger design is optimized to minimize volume while retaining a total heat rate of 1 MW under steady-state operating conditions. Results of the heat exchange calculations, available in Section 5.3.1, offer one potential solution that fits all design requirements though many possible solutions exist.

The Effectiveness-NTU method is adopted to design and size the radiator as well. For a designated flow rate, Gnielinski correlation is used to calculate Nusselt number on the tube side. The Grimison correlation was used to calculate the Nusselt number for the flow on the shell side. The design meets all requirements and calculations as discussed in Section 5.3.1.

Figure 4.6-1 Simplified Reactor (Red) and Coolant (Blue) Heat Removal Systems



4.7 Gas Management System

The Gas Management System provides essential safety functions to the Reactor System including cover gas control, gaseous fission product management, and Reactor Protection System functionality. The Gas Management System is described in Section 9.6.

4.8 References

4.8-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section III - "Rules for Construction of Nuclear Facility Components - Division 5 - High Temperature Reactors," New York, NY.

4.8-2 Quality Assurance Requirements for Nuclear Facility Applications, NQA-1-2019, ASME, 2019.

4.8-3 R. J. McConn, Jr., C. J. Gesh, R. T. Pagh, R. A. Rucker, and R. G. Williams, III, “Compendium of material composition data for radiation transport modeling” Radiation Portal Monitor Project, Pacific Northwest National Laboratory, PNNL 15870, Rev. 1, March 4, 2011.

4.8-4 J. R. Engel and R. C. Steffy, “Xenon behavior in the molten salt reactor experiment,” Oak Ridge National Laboratory, ORNL-TM-3464, October 1971.

4.8-5 S. J. Ball and T. W. Kerlin, “Stability analysis of the molten-salt reactor experiment,” Oak Ridge National Library ORNL-TM-1070, 1965.

4.8-6 Theodore L. Bergman, Adrienne S. Levine, Frank P. Incopera, David P. DeWitt, Fundamentals of Heat and Mass Transfer,” 7th Edition, Wiley, 2011.

4.8-7 Compere, E. L., Kirslis, S. S., Bohlmann, E. G., Blankenship, F. F., Grimes, W. R. (1975) “Fission Product Behavior in the Molten Salt Reactor Experiment” Oak Ridge National Laboratory, ORNL-TM-4865.

Chapter 5
Molten Salt Reactor Cooling Systems

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 5 MOLTEN SALT REACTOR COOLING SYSTEMS 5-1

5.1 Summary Description 5-1

5.2 Fuel System Boundary and Fuel Salt Heat Transport 5-2

5.2.1 Primary Cooling System 5-3

5.2.2 Secondary Cooling System 5-3

5.2.3 Auxiliary Systems. 5-3

5.2.4 Instrumentation and Control Systems 5-5

5.3 Cooling Systems 5-5

5.3.1 Primary Cooling System 5-5

5.3.2 Secondary Cooling System 5-7

5.4 Fuel Salt Cleanup System 5-9

5.5 Salt Makeup System 5-9

5.5.1 Fuel Salt Makeup System 5-10

5.5.2 Secondary Cooling Makeup System 5-10

5.5.3 Auxiliary Cooling Systems 5-10

5.5.4 Design Basis for the Drain Tank 5-10

5.6 REFERENCES. 5-10

LIST OF TABLES

Table 5.2-1 Parameter Limitations 5-4
Table 5.3-1 Radiator Design Parameters 5-7

LIST OF FIGURES

Figure 5.2-1 Diagram of the Molten Salt Research Reactor Primary Heat Removal 5-4
Figure 5.3-1 Coolant Loop (Secondary) Cooling System Configuration 5-8

CHAPTER 5 MOLTEN SALT REACTOR COOLING SYSTEMS

5.1 Summary Description

The molten salt research reactor (MSRR) cooling systems have the following key requirements:

- Ensure that the fuel salt temperature remains below a safety limit defined to protect stainless steel structure integrity
- Ensure that the system can remove 1 MWth of heat on a continuous basis from the reactor loop

The MSRR primary heat removal system consists of the reactor loop, which contains fuel salt and transfers heat to a coolant loop where it is ultimately dissipated to the air through a radiator. Cooling for decay heat removal is accomplished by passive cooling in the drain tank by natural circulation and thermal radiation (Section 6.2.3).

Fluid in the reactor loop travels between the reactor vessel, reactor access vessel (RAV), reactor pump, and the shell side of the heat exchanger. Fluid in the coolant loop travels between the tube side of the heat exchanger and the air-cooled radiator. A diagram of heat removal pathways during normal reactor operation at 1 MWth is shown in Figure 4.1-1.

The reactor loop is physically located inside the reactor enclosure while the coolant loop penetrates the reactor enclosure. For reactor shut down and long-term cooling, the fuel salt in the reactor system is drained to the drain tank below the reactor by equalizing the gas pressure between the drain tank head space and the RAV head space. No isolation or check valves are needed or included in the fuel salt bearing sections of the reactor system.

The coolant loop is contained mainly within the secondary enclosure but extends into the reactor enclosure and into the heat exchanger. It consists of the tube side of the heat exchanger, coolant loop pump, radiator, and connective piping. A drain tank in the coolant loop underneath the main loop stores coolant after shutdown. No isolation valves or check valves are needed or included in the coolant system.

Fuel salt within the reactor loop and salt in the coolant loop flows under a forced-convection/pumped-flow regime. In the event of pump failure, the positioning of the components allows for natural convection in both loops.

Heat exchange from the reactor system to the coolant loop occurs through tube walls within the shell-and-tube heat exchanger, and temperatures remain steady because of strong thermal reactor physics feedback.

The coolant loop relies on ultimate heat transfer to the environment through a radiator. A tube bundle containing coolant is cooled with forced outside air. Small amounts of radionuclides are produced in the coolant loop by the neutrons from the delayed neutron precursors in the reactor loop.

Both the heat exchanger and the coolant loop radiator are sized to provide a continuous heat removal of 1 MW. Air used to cool the radiator on the coolant loop is pulled from outside the facility, flows across the radiator and back to the outside environment.

The key components in the coolant loop are the tube side of the heat exchanger, the radiator, a coolant pump, and the air-exchange system. A diagram showing each heat exchange system is shown in Figure 5.2-1. Positive pressure is maintained in the coolant loop relative to the reactor loop during steady state operation.

The reactor drain tank housed in the Reactor Thermal Management System (RTMS) with the reactor vessel, has adequate capacity to store the entire system fuel salt content, and provides adequate cooling to passively remove fission product decay heat indefinitely. Details of the RTMS, drain tank design and cooling performance are in Section 6.2.4.

The fuel salt is maintained at operational temperatures by nuclear heating during full-power operation; however, electrical resistance heaters are used to bring the reactor system and heat removal system to temperature during startup and low-power operations. An auxiliary system of resistance heaters around the reactor components and heat-removal components allows the system to be preheated at a controlled rate. This minimizes thermal stresses and prevents salt from inadvertently freezing within the reactor or coolant loops. If heaters are lost, fuel salt is drained much faster than the time necessary for the salt to freeze in the system. Chapter 13 provides an analysis of the loss of off-site power accident.

This chapter describes the design bases and parameters of the reactor-to-coolant loop heat exchanger, heat exchange in the radiator, and ultimate heat sink cycling within the coolant loop containment. Auxiliary cooling in the reactor loop drain tank is also described.

5.2 Fuel System Boundary and Fuel Salt Heat Transport

Consistent with Design Criterion (DC) 10, reactor design, the fuel system boundary and fuel salt heat removal systems are designed with appropriate margin to help ensure specified acceptable system radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Consistent with DC 14, reactor fuel salt boundary, the fuel salt boundary temperature is maintained to have an extremely low probability of gross rupture.

Consistent with DC 15, reactor fuel salt system design, the fuel salt cooling systems are designed with sufficient margin to help ensure the design conditions of the reactor fuel salt boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Consistent with DC 30, quality of fuel salt boundary, the fuel salt boundary is designed, fabricated, erected, and tested to appropriate quality standards.

Consistent with DC 31, fracture prevention of fuel salt boundary, the fuel salt cooling systems have sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, the probability of rupture is minimized. The design reflects consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary materials under operating, maintenance, testing, and postulated accident conditions, including uncertainties.

The primary design criteria for the cooling systems are:

- Continuous removal of 1 MW thermal with system operating pressure between 0 and 0.5 MPa
- Maintain the fuel salt thermal limit of 816 degrees Celsius assumed in Chapter 13 analyses. (A safety limit will be defined using the final design and will be specified in the operating license application.)

5.2.1 Primary Cooling System

The primary cooling system of the MSRR is referred to as the reactor loop cooling system. The reactor loop allows fuel to flow continuously between the reactor vessel and the heat exchanger. Fuel remains in the reactor loop during normal operation, although it can be moved to the reactor drain tank to allow indefinite passive decay heat removal in the event of shutdown or malfunction. The cooling system is sized for the licensing basis of 1 MWth and is capable of continuously removing 1 MW of fission heat through the heat exchanger at appropriate coolant flow rates.

5.2.2 Secondary Cooling System

The secondary cooling system of the MSRR is referred to as the coolant loop cooling system. The coolant loop cooling system is capable of matching the continuous 1 MW of heat removal from the reactor loop at steady state. As heat is transferred to the coolant loop through the heat exchanger, it is transferred through the radiator at the same rate as steady-state power production. Air in the coolant loop enclosure region is continuously cycled from outside to maintain air temperature low enough to support the required continuous heat removal of less than or equal to 1 MW, depending on operating conditions.

5.2.3 Auxiliary Systems

A system of heating elements heats the salt prior to criticality. The reactor vessel, drain tank, and loop piping are housed in the RTMS (see Section 6.2.4). The reactor and coolant loops are preheated before operation, allowing the system temperature to be raised to operating temperature before criticality is achieved at startup. The heaters and insulation also prevent salt freezing in the pipes that would restrict circulation in each loop. Temperatures of reactor components are monitored to ensure salt does not freeze within the reactor system outside the drain tank.

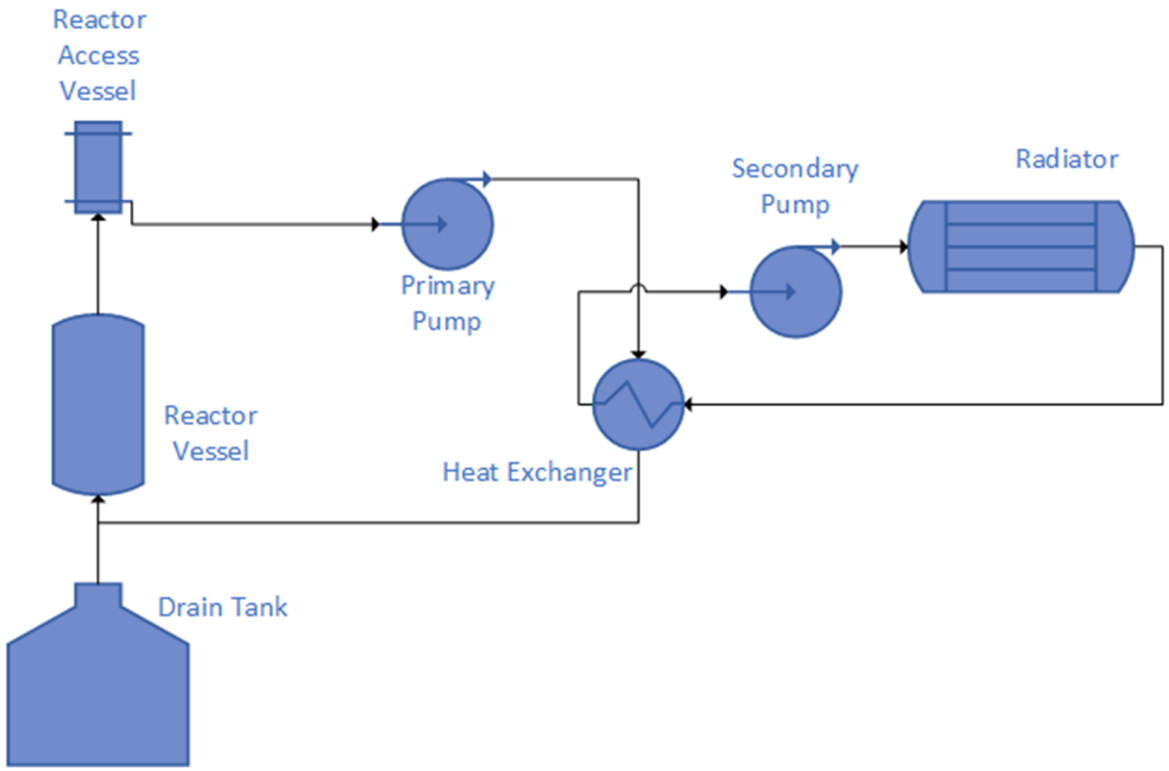
The drain tank provides decay heat removal. It is designed with sufficient surface area that it can passively remove decay heat indefinitely without exceeding the safety limit temperature for the stainless steel material.

Reactor systems have been examined to determine the parameter limitations summarized in Table 5.2-1.

Table 5.2-1 Parameter Limitations

Parameter	Minimum Value	Maximum Value	Basis
Alloy	Stainless steel 316H	N/A	Temperature and corrosion resistance
Loop pipe diameter	5.898 cm (nominal)	5.898 cm (nominal)	Velocity and pressure loss
Reactor loop flow rate	0 kg/s	~25 kg/s	Delayed neutron reactivity, heat transfer rate, pressure drop across pump, temperature change from hot leg to cold leg
Coolant loop flow rate	0 kg/s	~75 kg/s	Heat transfer and pressure loss
Fuel salt temperature	~550°C	~650°C	Fuel freezing point and stainless steel operating temperature limits

Figure 5.2-1 Diagram of the Molten Salt Research Reactor Primary Heat Removal



5.2.4 Instrumentation and Control Systems

Several control systems are used to ensure design parameters remain within the required range for safe operation. Fuel temperature is monitored by thermocouples placed at the reactor vessel outlet for temperature measurements because it is the location of the highest stainless steel reactor barrier temperature at steady-state flow. Pressure is monitored continuously through active monitoring systems in the reactor head space, drain tank, and reactor pump head space. The reactor protection system equalizes pressure between the drain tank and RAV to passively shut down the reactor by draining the fuel salt from the system. Both the reactor loop and coolant loop require variable flow rates to control power output. A combination of resistance heaters, variable pump speeds in the reactor and coolant loops, and blower flow rate across the radiator are used to control heat removal.

During normal operating conditions, salt is prevented from returning to the drain tank by a pressure differential. Pressure in the drain tank is kept higher than in the loop, preventing salt from entering the drain tank. If the pressure gradient ceases to be actively maintained, the fuel salt will drain into the drain tank passively without interaction from the operator or software. A reactor trip can also be performed automatically based on setpoints for various measurements of reactor operation (e.g., temperature and power). Alternatively, during shutdown the operator will trip the reactor by allowing the pressures in the system to equalize and allow fuel to drain into the drain tank for shutdown.

5.3 Cooling Systems

The cooling systems include the reactor loop cooling system and the coolant loop cooling system. These cooling systems interface through the heat exchanger. The coolant loop interfaces directly with the atmosphere through the ultimate heat sink at the radiator.

5.3.1 Primary Cooling System

The heat transfer in the reactor loop begins in the reactor vessel region, traveling through the graphite channels that make up the active core region. The RAV (salt expansion and sampling tank) is situated directly above the core to accommodate expansion or contraction within the loop. Piping from the core passes through the pump on the way to the heat exchanger, which is configured horizontally above the core, optimizing natural-circulation flow in the event of reactor pump failure. After leaving the heat exchanger, fuel is piped down below the core and back up through the graphite core channels.

The primary cooling system is capable of removing 1 MW of continuous thermal power at an operating pressure of no more than 0.5 MPa gage.

Heat transfer calculations (Section 4.6, Section 5.2.3, and Section 5.3.2) account for a limiting constraint on flow rate of less than 25 kg/s in the fuel loop. This is a conservative constraint to ensure reactivity insertion from delayed neutrons is minimized in the event of pump failure and subsequent stagnation of fuel within the core. The reactor operates at a lower flow rate within the reactor loop of 23.9 kg/s, which achieves the desired heat transfer rate at full power.

The fuel salt used in the reactor loop is 67LiF-28BeF2-5UF4 with the ratios given in mole percent. Details on salt chemistry are addressed in Section 4.2.1.

Heat transfer calculations show a shell-and-tube heat exchanger can be designed and sized to remove 1 MW of continuous heat production during operation. Heat exchange ultimately determines the rate at which power can be produced in the MSRR because the neutron physics result in the core matching the heat rate of the heat exchanger at equilibrium power production during steady-state operation. Accurate sizing of the heat exchanger ensures steady-state power remains within the intended range.

An effectiveness and number of transfer units (NTU) method calculation has been used for designing the heat exchanger. The primary cooling system in the MSRR limits fuel-side flow rate to 25 kg/s through pump sizing and administrative controls if necessary. This minimizes positive reactivity insertion due to delayed neutrons remaining in the core because increasing or decreasing mass flow rate results in negative or positive reactivity insertion to the core, respectively. Calculations also constrained Reynolds numbers on both the shell and the tube side to above 3000 to remain in the correct range for the Gnielinski correlation [Reference 5.6-1] to calculate the Nusselt number on the tube side. This Nusselt number correlation is appropriate for turbulent internal tube flow and commonly used in the nuclear industry. Importantly, it is also available for use in RELAP5-3D, which allows direct comparison between manual calculations and RELAP5 calculations.

Models show relatively low pressures throughout the loop with a maximum pressure of 299 kPa absolute pressure at the pump outlet. Minimum pressure at pump inlet is 106 kPa bar absolute pressure. Pressure loss across the shell (fuel) side of the heat exchanger is constrained in design calculations at 80 kPa. Pressure loss across the tube (coolant) side is also constrained to 80 kPa. The coolant loop is designed and intended for positive pressure at steady state compared to the reactor loop to prevent egress of fuel from the reactor loop into the more lightly shielded coolant loop in the event of a leak in the heat exchanger. Core inlet is at 203 kPa absolute pressure while outlet is at 138.5 kPa absolute pressure. A stagnant reactor loop ranges in pressure from 101 kPa at the highest point to 203 kPa at the lowest point in the loop based on hydrostatic pressure.

Thermal hydraulic control systems in the reactor loop control pump flow and passively drain the entire loop in the event of a reactor trip. Controls for variable pump speed or throttle valve controls may exist in the reactor loop and coolant loop to tune the reactor to the desired power rating. Under normal operating conditions, the temperature of the fuel is maintained through large negative temperature feedback, and thermal-hydraulic control systems are not necessary to maintain fuel within operating temperatures. In the event of a temperature excursion, actuation of the reactor protection system shuts down the reactor by equalizing the pressure between the drain tank and the reactor access vessel, causing the fuel salt in the reactor loop to drain to the drain tank where passive cooling of decay heat can occur indefinitely through natural convection and radiant heat transfer.

5.3.2 Secondary Cooling System

The cooling system in the coolant loop provides heat removal from the reactor loop during normal operation and shut down by transporting heated coolant salt from the heat exchanger to the radiator.

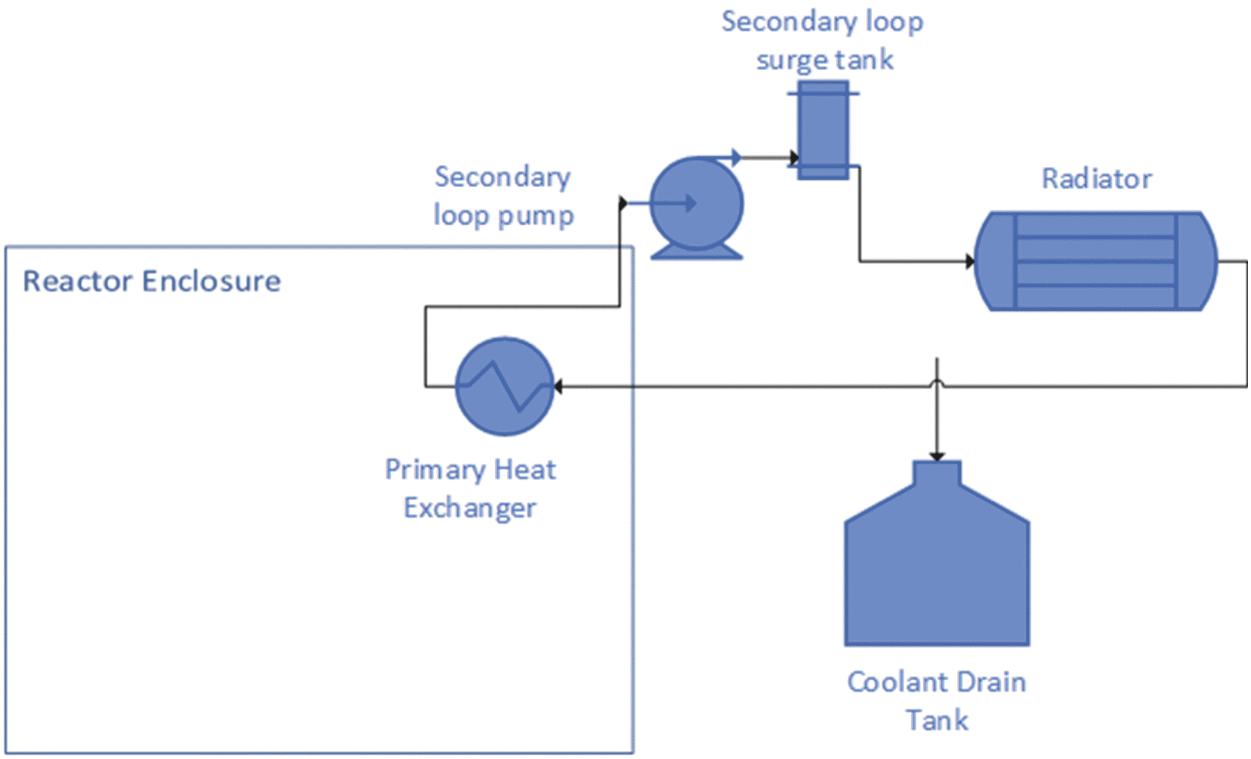
The system is placed outside the reactor enclosure and uses a mixture of enriched lithium fluoride (LiF) and beryllium fluoride (BeF_2) for its excellent thermal hydraulic properties and operating temperature range that matches the fuel salt. The 2:1 mixture forms a eutectic mixture with a melting point of 454 degrees Celsius [Reference 5.6-2], and a boiling point of 1430 degrees Celsius [Reference 5.6-3]. Coolant salt is less complicated than the fuel salt because it does not contain fissile material. With the exception of the tube side of the heat exchanger, the coolant loop is placed in a separate enclosure from the reactor loop.

The coolant loop flows hot salt from the tube side of the heat exchanger, through piping to the coolant loop pump, through the surge tank, and then to the radiator. The radiator transfers heat from the hot coolant to the air blown through it. The initial radiator design parameters shown in Table 5.3-1 are subject to change as the design progresses. The expansion tank is used to avoid sudden rises in molten salt pressure during temperature changes or pump activation. Afterward, coolant leaves the radiator and travels back to the heat exchanger.

Table 5.3-1 Radiator Design Parameters

Parameter	Nominal Value
Maximum heat rate	1 MW
Estimated tube inlet temperature (salt)	781K / 508 °C
Estimated tube outlet temperature (salt)	773K / 500 °C
Approximate shell-inlet temperature (air)	310K / 37 °C
Approximate shell-outlet temperature (air)	394K / 121 °C
Tube side-flow rate at full power	53.4 kg/s
Shell side-flow rate at full power (air)	11.7 kg/s

Figure 5.3-1 Coolant Loop (Secondary) Cooling System Configuration



The coolant loop cooling system is designed to remove at least the same amount of thermal power (1MW) as the heat exchanger. Coolant loop system pressure is maintained above reactor loop pressure to prevent any leakage from the reactor loop. A continuous flow rate of 53.4 kg/s is maintained within the coolant loop at full power of 1 MW. This flow rate can be reduced at lower power operation, though a constant flow rate can be used for the entire range of power outputs, allowing the temperature drop across the radiator to adjust rather than the flow rate within the coolant loop.

An effectiveness-NTU method has also been adopted to calculate the rate of heat transfer in the radiator. For the designated coolant flow rate (53.4 kg/s), which demonstrates a high Reynolds number, the Gnielinski correlation was used to calculate Nusselt number on the tube side. On the other hand, the Grimison correlation [Reference 5.6-1] was used to calculate the Nusselt number for flow on the shell side, with a flow rate of 11.7 kg/s.

In case of a shutdown or malfunction, the salt in the coolant loop automatically dumps into the coolant drain tank to avoid molten salt freezing in the pipes of the coolant loop.

Decay heat is not present in the coolant loop because no fuel is present. The coolant loop drain tank is also used to store coolant for use in the coolant loop on demand. Resistance heating is used to heat the coolant loop at startup and to maintain piping and component temperatures in the operating range when necessary, particularly at low power operation.

While the coolant loop does not contain fissile material, it may have radionuclides that must be contained. Tritium generated in the reactor may migrate into the coolant loop through the heat exchanger tube walls. Neutrons streaming from the internal shield and delayed neutrons generated in the heat exchanger could also activate the coolant salt. If tube leaks were to occur, then small quantities of fuel salt may relocate to the coolant loop. Tritium bears the greatest radiological hazard and is addressed in Chapter 11. The coolant salt head space can be monitored for tritium and general radiation monitoring in the coolant loop enclosure as described in Chapter 7.

5.4 Fuel Salt Cleanup System

Functional requirements for the fuel salt chemistry control system are described in Section 4.2.1 and the fuel handling system is described in Section 9.2. The fuel salt redox potential is monitored, and active metals such as beryllium are added to the fuel salt to control redox potential. Non-gaseous fission products are not removed from the fuel salt during operation. Any additional salt cleanup occurs in the fuel handling system.

Gaseous fission products are not removed from the salt except by the gas management system. As described in Section 9.6, it will remove and filter gaseous fission products on an ongoing basis to ensure all fission products are contained safely. The system primarily makes use of adsorbent materials to ensure that xenon, krypton, and iodine compounds are contained as they exit the reactor via the sparge gas and cover gas over the RAV.

Uranium tetrafluoride is added to the base fuel salt when the reactor is subcritical during reactor startup and to compensate for the depletion of fissile material during operation using high-assay low-enriched uranium. The reactor pump must be in operation when UF_4 is added to ensure appropriate mixing. Reactivity impacts of fuel salt additions are addressed in Section 4.5.2.3, and the fuel salt addition system will be described in the Operating License application.

5.5 Salt Makeup System

The salt makeup systems are essential components that allow reactor systems to adjust smoothly to changes in salt volume that occur during operation. Particularly at startup and shutdown or during changes to power output, temperature can vary, changing the density and volume of salt. Removal of small amounts of salt for data collection or addition of experimental materials to the RAV also can result in small changes in volume. These small variations in volume are accommodated by the RAV in the fuel loop and by a surge tank in the coolant loop. Drain tanks also exist in each loop to allow addition or removal of salt from circulation.

5.5.1 Fuel Salt Makeup System

The reactor loop has a small amount of excess volume in the RAV that allows volume changes to accommodate temperature and pressure fluctuations.

Multiple techniques are used to monitor RAV level directly, while pressure differential between the RAV and drain tank can be used to verify this measurement. Instrumentation and control systems used to monitor and adjust the level in the RAV are detailed in Section 7.3.4.

Excess salt can also be stored in the drain tank to allow adjustments to salt level as needed. The drain tank is in a heated zone, and fuel from the drain tank is added at the appropriate temperature to avoid reactivity insertions from a “cold slug” of new fuel flowing through the reactor vessel.

5.5.2 Secondary Cooling Makeup System

The secondary cooling makeup system of the MSRR is referred to as the coolant salt makeup system. Similar to the RAV used for makeup and overflow in the fuel loop, the coolant loop contains a surge tank to allow for small changes in coolant volume due to temperature or other factors. The description of instrumentation and controls in Section 7.3.4 can be similarly applied to the sensors that will be used in the surge tank.

5.5.3 Auxiliary Cooling Systems

Radiation heating in materials outside the reactor system will be analyzed and reported in the Operating License application. Preliminary analyses suggest that radiation heating does not lead to exceeding of safety limits.

5.5.4 Design Basis for the Drain Tank

The drain tank absorbs and passively dissipates decay heat produced after shutdown indefinitely. Drain tank performance is included in the description of the RTMS in Section 6.2.4.

5.6 REFERENCES

5.6-1 Bergman, T.L., and Lavine, A.S., “Fundamentals of Heat and Mass Transfer, 8th edition,” John Wiley and Sons, Inc., Hoboken, NJ, 2017.

5.6-2 Thoma, R.E. (ed), “Phase Diagrams of Nuclear Reactor Materials,” ORNL 2548, Oak Ridge National Laboratory, Oak Ridge, TN, 1959.

5.6-3 Baral, Khagendra, et. al., Temperature-Dependent Properties of Molten Li_2BeF_4 Salt Using Ab Initio Molecular Dynamics,” ACS Omega 2021, 6, 30, 19822-19835, July 21, 2021, https://doi.org/10.1021/acsomega.1c02528.

Chapter 6
Engineered Safety Features

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 6 ENGINEERED SAFETY FEATURES. 6-1

6.1 Summary Description 6-1

6.2 Detailed Descriptions 6-3

6.2.1 Confinement. 6-3

6.2.2 Containment. 6-5

6.2.3 Emergency Cooling System. 6-9

6.2.4 Reactor Thermal Management System 6-9

6.3 Compliance with Design Criteria 6-11

6.4 References 6-12

LIST OF FIGURES

Figure 6.1-1 Physical Configuration of the Reactor Cell, Reactor Enclosure, and Reactor Thermal Management System 6-2

Figure 6.2-1 Schematic of Reactor Thermal Management System 6-11

CHAPTER 6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

The Nuclear Regulatory Commission (NRC) describes a methodology for functional containment in SECY-18-0096 [Reference 6.4-1]. It acknowledges that non-light-water reactor technologies differ from light-water reactors in operating conditions, coolants, and fuel forms, allowing a different approach to fulfill the safety function of limiting the physical transport of radioactive material to the environment. The SECY defines functional containment as “a barrier, or a set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment.” The first fission product barrier is the fuel salt itself, which will contain the majority of fission products under normal and accident conditions as described in Chapters 4 and 13. The second barrier is the reactor system and gas management system which contain those fission products considered gaseous and are described in Chapters 4 and 9. The third, fourth, and fifth barriers are the Reactor Thermal Management System, reactor enclosure, and reactor cell. The final three fission product barriers are engineered safety features (ESFs) and are described in this chapter. The fission product barriers act in concert with atmospheric dispersion considerations to provide functional containment.

While the Molten Salt Research Reactor (MSRR) does not have a confinement and containment found in the traditional licensing paradigm, the reactor cell and reactor enclosure perform comparable safety functions. The MSRR also features a novel ESF, which is the reactor thermal management system (RTMS).

As shown in Figure 6.1-1, the MSRR has three ESF systems surrounding reactor components in nested fashion: 1) innermost is the RTMS, 2) the reactor enclosure that surrounds the RTMS, and 3) the reactor cell that encompasses the reactor enclosure. The ESFs are located inside the systems pit which is in the research bay.

The RTMS surrounds five reactor system components, the reactor vessel, the drain tank, the reactor pump, the RAV, and the fuel salt bearing portions of the heat exchanger. It maintains fuel salt temperature as needed and retains any fuel salt that leaks from the reactor system.

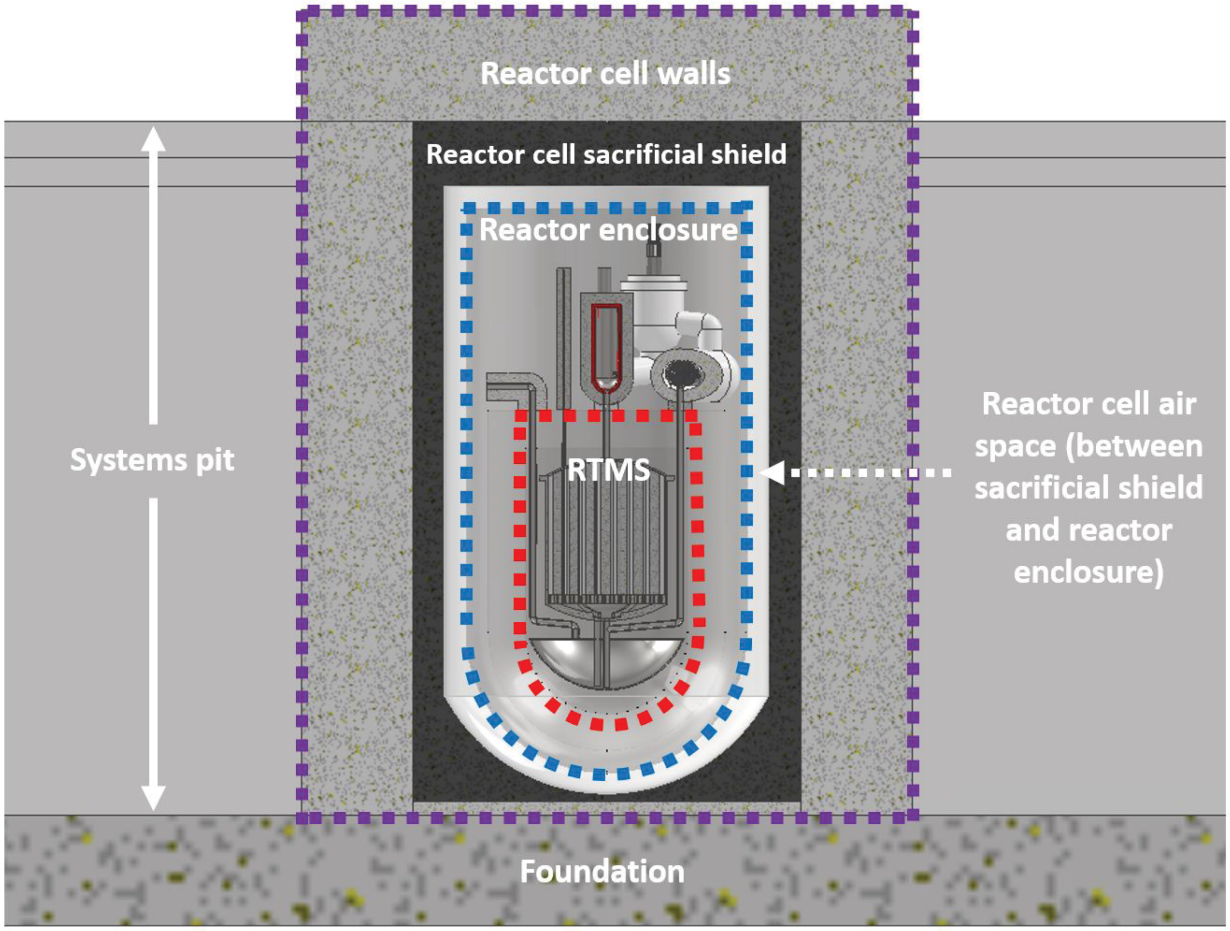
The reactor enclosure outside the RTMS contains the entire reactor system, including the heat exchanger, reactor pump, reactor access vessel, and associated piping. The reactor enclosure is a sealed, leak-tight, pressure-controlled boundary that serves as a fission product barrier. The reactor enclosure has two safety functions: to maintain negative pressure during normal and accident conditions and to provide structural support for the reactor system.

The reactor cell is a non-sealed, controlled airspace surrounding the reactor enclosure. Air is circulated through the reactor cell by the auxiliary heat removal system (AHRS) venting to the research bay exhaust and ultimately to the atmosphere to provide cooling of the reactor enclosure. The AHRS maintains reactor cell temperatures by removing

large air volumes, which creates a slightly negative pressure during operation. The reactor cell is constructed of concrete and steel, its walls serve as the biological shield, and it has three safety functions that are entirely passive under accident conditions:

- Serve as a large thermal mass to absorb excess heat in a Loss of Normal Electrical Power (LONEP) accident (Chapter 13)
- Slow the release of gases which might leak from the reactor enclosure
- Provide essential biological shielding as described in Section 4.4.

Figure 6.1-1 Physical Configuration of the Reactor Cell, Reactor Enclosure, and Reactor Thermal Management System



6.2 Detailed Descriptions

6.2.1 Confinement

6.2.1.1 Reactor Cell and Auxiliary Heat Removal System Safety Functions

The MSRR does not have a reactor confinement per se, but the reactor cell and the AHRS perform similar functions. Together they

- maintain the reactor system and reactor enclosure within defined temperature limits under normal and accident conditions.
- control the flow of radioactive air generated within the cell during normal and accident conditions.
- reduce radiological effects during complete loss of active cooling as described in the LONEP accident analysis.

The reactor cell materials and dimensions are described in Section 4.4, and the AHRS is further described in Section 9.7.1.

6.2.1.2 Reactor Cell in the Accident Analysis

The reactor cell completely surrounds the reactor enclosure. The reactor enclosure dissipates heat from the reactor system into the reactor cell by both convection and thermal radiation. During normal operations, the AHRS provides forced air flow, and in loss-of-power conditions, natural circulation of air in the reactor cell and dissipates the heat to the reactor cell structure (Section 6.2.3).

As described in Chapter 13, during the maximum hypothetical accident (MHA) in which gaseous fission product inventory is released to the reactor enclosure and from the reactor enclosure to the reactor cell, the reactor cell serves as a hold up volume for air. With cell louvers closed, the design leak rate of the reactor cell is 1 percent per day. The design leak rate of the reactor cell will be confirmed during pre-operational and acceptance testing and will encompass a variety of conditions including operation or non-operation of the HVAC. The HVAC does not provide air to the cell but may influence the pressure of the SERC bay which communicates with the cell through small gaps. This effect is bounded by the design leak rate. The slow temperature rise of the reactor cell during LONEP produces a correspondingly minor change in pressure and leakage.

6.2.1.3 Physical Description of the Reactor Cell

The reactor cell shown in Figure 6.1-1 is constructed of concrete and steel. Its walls serve as biological shielding and provide additional structural support for the reactor enclosure. The walls are designed to reduce gamma and neutron dose to a level low enough to allow operation. Section 4.4 provides details. The reactor cell and reactor enclosure mechanical supports prevent rupture of the reactor enclosure during a building collapse for any reason. The reactor cell does not maintain a substantial pressure boundary during postulated accidents due to small gaps between the blocks and around penetrations. The design leak rate is

attained due to the slow pressure rise driven by the slow and minor temperature rise during a LONEP event. The reactor cell has maintenance hatches, and the reactor cell lid is movable in pieces for access to the reactor enclosure. The reactor cell contains penetrations for necessary piping and ducting. The reactor cell is not isolatable from the research bay as there are small gaps within the cell walls and lid. The systems pit is the ultimate structural support for the reactor enclosure and reactor cell.

6.2.1.4 Safety Functions of the Auxiliary Heat Removal System

The AHRS provides safety functionality by drawing 4000 cfm of cell air from the bay and discharging it through the bay exhaust to atmosphere during operation. This cools the reactor cell and ensures cell air does not mix with research bay air. The reactor cannot operate unless the AHRS is in operation. Cell air enters and exits the cell through louvers, which may be closed when the reactor shuts down.

A complete LONEP terminates AHRS cooling, and cell louvers automatically close to isolate the cell. If AHRS cooling is lost, the reactor cell will gradually heat up. The reactor trips and resistance heaters are de-energized by the LONEP. Both actions minimize reactor cell heat up without the AHRS in operation (See Section 6.2.3).

6.2.1.5 Radionuclide Content of Reactor Cell Air

Radioactive nuclides N-16, Ar-41, Ar-37, and C-14 are generated through neutron activation in the reactor cell. This air could contain activated dust, an evaluation of which will be provided in the Operating License application.

6.2.1.6 Airflow Paths and Equipment

A detailed description of the airflow paths and equipment will be provided in the Operating License application. The air handling equipment in the reactor cell is required to be operating to support reactor operation. This equipment is not needed, however, for accident scenarios in which passive functions of the reactor cell described above mitigate certain scenarios. For more details, see the LONEP analysis in Chapter 13.

6.2.1.7 Analyses that Require or Impact the Reactor Cell

- Biological shielding, Section 4.4
- MHA, Chapter 13
- LONEP, Chapter 13
- Activated reactor cell air is considered an effluent, as discussed in Section 11.1.5.

6.2.1.8 Reactor Cell Compliance with Design Criteria

DC 16, the reactor cell is a component of functional containment of gaseous radionuclides.

DC 50, the reactor cell shall satisfy a design leak rate under pressure and temperatures consistent with a LONEP.

DC 52 and DC 53, the reactor cell shall have its leak rate tested.

DC 54, 55, 56, and 57, the reactor cell relies upon isolation of penetrations to meet the design leak rate so that consequences are enveloped by the MHA. Isolation and monitoring of penetrations will be implemented by suitably redundant valves, physical breaks, or component barriers in other portions of the facility.

6.2.2 Containment

6.2.2.1 Safety Functions of the Reactor Enclosure

The reactor enclosure functions as and achieves containment. It is a leak-tight, pressure-controlled, fission product barrier, and almost all radionuclides present within the MSRR during operation are contained inside the reactor enclosure. The reactor enclosure maintains negative pressure during operation and accident conditions. The reactor cell and the reactor enclosure work in concert to provide functional containment, thus mitigating radiological consequences during an MHA. During operation, the reactor enclosure pressure of less than 84 kPa is actively monitored and maintained. The reactor enclosure is cooled by the AHRS, which directs air through the reactor cell, cooling the exterior surface of the reactor enclosure. The reactor enclosure has a design leak rate of 0.01 percent per day with +2 psig overpressure or an absolute pressure of 115 kPa for the enclosure atmosphere. The design leak rate will be confirmed during pre-operational acceptance testing. The reactor enclosure will be periodically inspected and functionally tested.

The reactor enclosure atmosphere is inert with nitrogen gas and has minimal oxygen concentration during operation. The radionuclide content of the reactor enclosure atmosphere is monitored. If a substantial quantity of fission product is detected in the reactor enclosure, the reactor will shut down. Allowable limits will be reported in the Operating License application.

The reactor enclosure is connected to a nitrogen gas supply and a vacuum pump to maintain the enclosure atmosphere at the required composition and pressure. The vacuum pump rejects the enclosure atmosphere to the research bay exhaust. Isolation valves, normally closed during operation, are used on these nitrogen gas connections and are considered a part of ESFAS. A pressure relief valve with a setting below the design pressure of the enclosure is located between the nitrogen supply vessel and the first isolation valve. The isolation valves and pressure relief valve shall satisfy DC 4 and DC 56. The number of nitrogen bottles attached to the nitrogen supply shall be limited by design features such that the equilibration of the attached bottles with the reactor enclosure shall not overpressure the reactor enclosure in the event the pressure relief valve fails. These bottles will be secured so they do not impose a risk to SR components.

6.2.2.2 Physical Description of the Reactor Enclosure

The reactor enclosure consists of a vertical cylindrical section with a hemispherical lower head, and an upper lid with a large diameter flange bolted to the vertical vessel. The enclosure lid is removable for assembly maintenance and decommissioning. There are multiple salt-wetted penetrations into the reactor enclosure for both fuel salt and coolant salt. The fuel salt penetrations go to the fuel salt receiving vessel in the fuel handling system. The coolant salt penetrations go to the heat exchanger. The reactor enclosure lid has additional hatches for equipment and personnel that are sealed during operation but can be opened for maintenance. Multiple gas penetrations have isolation seals that collectively satisfy the enclosure design leak rate. Structural supports of the reactor system are attached to the reactor enclosure. The enclosure is supported by a dedicated support system, which distributes the load to the reactor systems pit floor. Section 4.2.5 describes the reactor enclosure structural support systems.

6.2.2.3 Reactor Enclosure in the Accident Analysis

Design features of the reactor enclosure are effective in case of an MHA. It is assumed that as a result of an MHA, fuel salt relocates to the RTMS, the reactor head space gases redistribute to the reactor enclosure atmosphere, and the AHRS ceases to function and the louvers close (see Section 6.2.4 and Chapter 13). The sudden release of the reactor head space increases reactor enclosure pressure, but the pressure remains negative. Loss of the AHRS reduces cooling capacity, causing the reactor cell and reactor enclosure to slowly heat up. The reactor system and fuel salt are initially at operating temperature and gradually lose stored thermal energy by conduction, natural convection, and thermal radiation to the reactor enclosure. Decay heat from the fuel salt and gaseous radionuclides contributes to temperature rise. In response, the reactor enclosure pressure gradually increases, but pressure remains below atmospheric. This process is modeled in detail and presented in the MHA analysis (Chapter 13), which assumes a non-mechanistic 0.01 percent per day leakage. The role the reactor enclosure plays in the LONEP progression is similar to its role in the MHA except for spilling fuel salt.

6.2.2.4 Design Bases for the Reactor Enclosure

- Design leak rate is 0.01 percent per day at +2 psig.
- Below atmospheric pressure during operation
- Below atmospheric pressure during MHA and LONEP
- Provide structural support for the reactor system
- Maintain structural integrity given a building collapse
- All gas penetrations are sealable with leak-tight valves.
- Salt-wetted penetrations are isolatable.
- All penetrations are sealed around the penetration to minimize leakage between the reactor enclosure itself and the penetration line.

- Multiple equipment and personnel hatches provided through the reactor enclosure upper lid
- The reactor enclosure atmosphere is nitrogen with minimal oxygen concentration.

6.2.2.5 Physical Description Needed for Analysis

- Internal diameter of 10 ft (3 m)
- Lower hemispherical head
- 1 in, (2.54 cm) thick vessel and lid
- Vertical portion of the reactor enclosure is 15 ft (4.6 m) tall.
- Supported from the rim or lip of the vessel
- During operation, the reactor enclosure pressure is approximately 84 kPa.
- The cool portion (see RTMS Section 6.2.4) is 70 degrees Celsius.

6.2.2.6 Analyses which Require or Impact the Reactor Enclosure

- Structural design of reactor system, Section 4.3
- Biological shielding, Section 4.4
- MHA, Chapter 13
- LONEP, Chapter 13

6.2.2.7 Compliance with Design Criteria (See Section 3.1.2)

Consistent with Design Criterion (DC) 4, environmental and dynamic effects design bases, isolation valves and the pressure relief valve on the nitrogen line outside the reactor enclosure shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The reactor enclosure is protected from dynamic effects.

The fuel storage enclosure is connected to a nitrogen gas supply and a vacuum pump to maintain the enclosure atmosphere at the required composition and pressure. The vacuum pump rejects the enclosure atmosphere to the research bay exhaust. Isolation valves, normally closed during operation, are used on these nitrogen gas connections. A pressure relief valve with a setting below the design pressure of the enclosure is located between the nitrogen supply vessel and the first isolation valve. The number of nitrogen bottles attached to the nitrogen supply shall be limited by design features such that the equilibration of the attached bottles with the enclosure shall not overpressure the enclosure in the event the pressure relief valve fails. These bottles will be secured so they do not impose a missile risk to safety related components.

The off-gas enclosure is connected to a nitrogen gas supply and a vacuum pump to maintain the enclosure atmosphere at the required composition and pressure. The vacuum pump rejects the enclosure atmosphere to the research bay exhaust.

Isolation valves, normally closed during operation, are used on these nitrogen gas connections. A pressure relief valve with a setting below the design pressure of the enclosure is located between the nitrogen supply vessel and the first isolation valve. The number of nitrogen bottles attached to the nitrogen supply shall be limited by design features such that the equilibration of the attached bottles with the enclosure shall not overpressure the enclosure in the event the pressure relief valve fails. These bottles will be secured so they do not impose a missile risk to safety related components.

Consistent with Design Criterion (DC) 16, containment design, the MSRR fuel salt, reactor enclosure, and reactor cell serve as functional containment to control the release of radioactivity to the environment for as long as postulated accident conditions require.

Consistent with DC 50, functional containment design basis, the MSRR reactor enclosure structure, including access openings and penetrations, are designed to accommodate with margin the calculated pressure and temperature conditions resulting from postulated accidents without exceeding the design leakage rate. The MSRR reactor enclosure maintains negative pressure during operation and accident conditions.

Consistent with DC 51, fracture prevention of functional containment boundaries, the boundary of the MSRR reactor enclosure structure is designed with sufficient margin to ensure under operating, maintenance, testing, and postulated accident conditions the probability of rupture is minimized. The design reflects consideration of service temperatures and other conditions of the containment boundary materials during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws. The MSRR reactor enclosure maintains negative pressure during operation and accident conditions.

Consistent with DC 52, capability for functional containment leakage rate testing, the MSRR reactor enclosure structure is maintained at a negative pressure. Leakage more than design specifications will be detectable.

Consistent with DC 53, provisions for functional containment testing and inspection, the MSRR reactor enclosure structure will allow for periodic visual inspection of important areas. Degradation of the enclosure will be detectable by monitoring leakage.

Consistent with DC 54, piping systems penetrating functional containment, the MSRR reactor enclosure relies upon isolation of penetrations to meet the functional containment design leak rate so that consequences are enveloped by the MHA. Isolation and monitoring of penetrations will be implemented by suitably redundant valves, physical breaks, or component barriers in other portions of the facility.

Consistent with DC 55, radionuclide interfacing lines penetrating functional containment, the MSRR reactor enclosure relies upon isolation of penetrations to meet the functional containment design leak rate so that consequences are enveloped by the MHA. Isolation and monitoring of radionuclide bearing penetrations will be implemented by suitably redundant valves, physical breaks, or component barriers in other portions of the facility. Two isolation valves will be located along the fuel salt transfer line from the reactor drain tank to the fuel salt purification and storage vessel in the Fuel Handling System. Two isolation valves will be located along the off-gas line from the reactor enclosure to the off-gas enclosure.

Consistent with DC 56, functional containment isolation, the MSRR reactor enclosure relies upon isolation of penetrations to meet the functional containment design leak rate so that consequences are enveloped by the MHA. Isolation and monitoring of ventilation line and equipment hatch penetrations will be implemented by suitably redundant valves, physical breaks, or component barriers in other portions of the facility.

Consistent with DC 57, closed system isolation, the MSRR reactor enclosure relies upon isolation of penetrations to meet the functional containment design leak rate so that consequences are enveloped by the MHA. Isolation and monitoring of additional penetrations such as instrumentation lines will be implemented by suitably redundant valves, physical breaks, or component barriers in other portions of the facility.

6.2.3 Emergency Cooling System

The MSRR does not utilize an emergency cooling system; the MSRR is designed such that decay heat is passively removed under adverse conditions. If the primary and auxiliary heat removal systems are lost, the reactor will trip, and the fuel salt will drain to the drain tank. Sufficient thermal mass exists within the reactor system, reactor enclosure, reactor cell, and research bay to ensure completely passive cooling mechanisms (conduction, natural convection, and thermal radiation) safely remove decay heat without violating a thermal limit. The Loss of Normal Electrical Power accident is described in Section 13.1.10.

6.2.4 Reactor Thermal Management System

The reactor vessel, drain tank, reactor pump, RAV, and fuel salt bearing portions of the heat exchanger are located within a heated and well-insulated container called the RTMS that has two general functions. The first is to ensure the fuel salt remains liquid under normal and adverse conditions long enough to allow it to drain from the reactor vessel. The second is to serve as a catch pan or container for fuel salt that escapes the reactor system. The RTMS is highlighted in Figure 6.2-1.

The RTMS maintains proper uniform temperatures within the reactor vessel and drain tank with electrical resistance heaters. These heaters are solely used to attain reactor operating limits before operation. These heaters are not used during full power operation. It is composed of an inner stainless steel container that surrounds the reactor system (reactor vessel, drain tank, RAV, reactor pump, and fuel salt bearing

portions of the heat exchanger) with an outer layer of insulation. The stainless steel is capable of withstanding, without failure, the direct contact of approximately 1.5 tons of hot fuel salt falling on it and collecting at the bottom of the RTMS. The utility of the RTMS is demonstrated in the Chapter 13 MHA analysis. In the event heater power is lost, the RTMS and the components inside will slowly cool. This cooldown period is long enough for the fuel salt to drain. The RTMS is not sealed, so nitrogen can move freely through small gaps between the RTMS and the reactor enclosure.

6.2.4.1 Design Criteria

Consistent with DC 72, fuel salt temperature control systems, the RTMS provides necessary heating for the reactor vessel and drain tank. The RTMS ensures the fuel salt remains liquid in the reactor system as necessary to support reactor protection system function.

6.2.4.2 Design Bases of the Reactor Thermal Management System

- Maintain the reactor vessel and drain tank at operating temperature under normal and adverse conditions
- Maintain the reactor vessel and drain tank above the freezing point of the fuel salt when all power is lost for long enough to ensure the fuel salt drains
- Does not maintain a pressure differential
- The RTMS shall be capable of withstanding without failure relocation of the entire fuel salt inventory from the reactor system to the RTMS.

A mechanical analysis demonstrating the capacity of the RTMS to withstand hot fuel salt interacting with it will be supplied in the Operating License application.

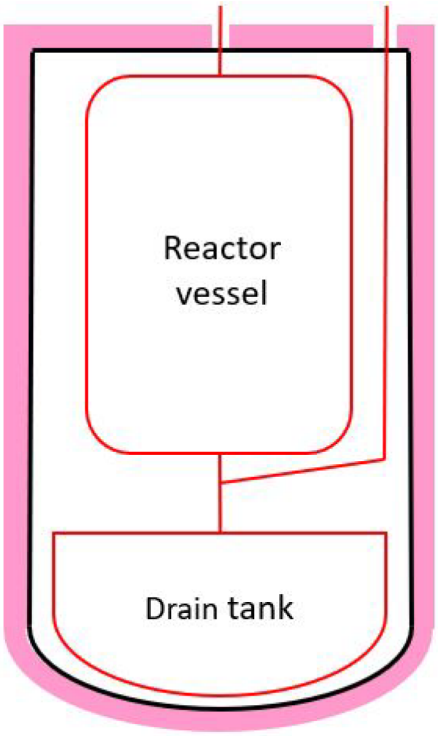
6.2.4.3 Design Parameters Needed for Analysis

- The RTMS is comprised of an inner stainless steel 316H structure and an outer layer of insulation.
- The inner RTMS has a vertical cylindrical section and a lower hemispherical body. The vertical section is approximately 79.8 in. (202.6 cm) tall, and the inner radius is approximately 34.3 in. (87.2 cm).
- Nitrogen within the RTMS is at roughly 600 degrees Celsius. It is open to reactor enclosure atmosphere.
- The RTMS is independently supported from the reactor loop.
- The internal height of the RTMS is approximately 9.5 ft. (2.9 m) and the external height of the RTMS is approximately 10.4 ft (3.1 m)

6.2.4.4 Analyses That Require or Impact the Reactor Thermal Management System

- MHA, Chapter 13
- LONEP, Chapter 13

Figure 6.2-1 Schematic of Reactor Thermal Management System



Note: Pink represents insulation. Red represents fuel salt-bearing components.

6.3 Compliance with Design Criteria

Consistent with DC 10, reactor design, the MSRR is designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Consistent with DC 34, residual heat removal, emergency cooling is provided passively to remove residual heat when required. The system safety functions to transfer fission product decay heat and other residual heat from the fuel salt at a rate such that specified acceptable radionuclide release design limits and the design conditions of the fuel salt boundary are not exceeded. During normal operations heat is removed by the AHRS (see Section 9.7.1) and the primary and secondary cooling systems (see Section 5.2)

Consistent with DC 35, fuel salt cooling system, emergency cooling is provided passively to remove residual heat to assure sufficient fuel salt cooling during postulated accidents and to remove residual heat following postulated accidents. The system passively transfers heat from the fuel salt such to the surrounding structures such that effective fuel salt cooling is maintained, and radionuclide release is limited during and following postulated accidents.

Consistent with DC 38, functional containment heat removal, in the MSRR, emergency cooling is passive and removes heat from the reactor enclosure by conduction, natural convection, and thermal radiation during loss-of-power events. Sufficient thermal mass exists in the systems and structures to dissipate decay heat indefinitely. During normal operations the AHRS (Section 9.7.1) provides reactor enclosure cooling.

Consistent with DC 40, testing of functional containment heat removal system, in the MSRR emergency cooling is passive and integral with the fuel salt boundary and reactor enclosure and provides passive cooling of the reactor enclosure. Routine monitoring of these SSCs is sufficient to ensure the integrity and capability of the emergency salt cooling system. During normal operations the AHRS provides for reactor enclosure cooling.

6.4 References

6.4-1 Nuclear Regulatory Commission, “Functional containment performance criteria for non-light-water-reactors,” SECY-18-0096, September 28, 2018.

Chapter 7
Instrumentation and Control Systems

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 7 INSTRUMENTATION AND CONTROL SYSTEMS 7-1

7.1 Summary Description 7-1

7.1.1 Calibration of Trips, Interlocks, and Annunciation 7-2

7.1.2 Distributed Control Systems 7-2

7.1.3 Reactor Protection System and Engineered Safety Features Actuation System 7-2

7.1.4 Distributed Control System and Control Room 7-3

7.1.5 Radiation and Environmental Monitoring System 7-4

7.2 Design of Instrumentation and Control Systems 7-4

7.2.1 System Description 7-6

7.2.2 System Performance Analysis 7-6

7.2.3 Conclusion 7-6

7.3 Distributed Control System. 7-6

7.3.1 Design Criteria and Bases 7-7

7.3.2 Control Rod Drives. 7-8

7.3.3 Nuclear Instrumentation. 7-8

7.3.4 Salt Level and Cover Gas Management System 7-9

7.3.5 Auxiliary Heat Removal Cooling Air Process Monitoring and Control 7-9

7.3.6 Primary Heat Removal Cooling Air Process Monitoring and Control. 7-9

7.3.7 Salt Pump Monitoring and Control 7-9

7.3.8 System Heaters Temperature Controls and Monitors 7-10

7.3.9 System Evaluation 7-10

7.4 Reactor Protection System 7-10

7.4.1 Design Criteria and Bases 7-11

7.4.2 Reactor Trip 7-15

7.4.3 RPS Trip Set Points 7-17

7.4.4 System Trips 7-19

7.4.5 System Evaluation 7-20

7.5 Engineered Safety Features Actuation System 7-20

7.5.1 Design Criteria and Bases 7-21

7.5.2 ESFAS Initiation. 7-24

7.5.3 Initiation Limits and Signals 7-26

7.5.4 System Evaluation 7-27

TABLE OF CONTENTS

7.6 Human-Machine Interface 7-27
7.6.1 Design Criteria and Bases 7-28
7.6.2 Control Room 7-28
7.6.3 Control Console and Human Machine Interface 7-29
7.6.4 Conclusions 7-30
7.7 Radiation and Environmental Monitoring System 7-30
7.7.1 Design Criteria and Bases 7-30
7.7.2 Facility Sensor Stations 7-32
7.7.3 Emplacement Detectors 7-33
7.8 References 7-33

LIST OF TABLES

Table 7.4-1 Deleted. 7-11
Table 7.4-2 Reactor Trip Set Points 7-18
Table 7.4-3 System Trip Channels 7-19
Table 7.5-1 ESFAS Initiation. 7-26

LIST OF FIGURES

Figure 7.2-1 Instrumentation and Controls Diagram with Subsystems 7-5
Figure 7.4-1 Reactor Trip Relay Circuits 7-15
Figure 7.4-2 Reactor Trip Relay Circuits 7-16
Figure 7.5-1 ESFAS Actuation 7-24

CHAPTER 7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The Instrumentation and Control (I&C) system monitors plant parameters, controls, components and provides an interface to the plant operators. It provides control signals for manual and automated functions, protective features to bring the plant to shutdown conditions, and monitoring systems for plant condition assessment and personnel protection. Using the I&C system, an operator can start up, operate, and shut down the reactor across all anticipated operational ranges and respond to all credible accidents. The information in this chapter is based on the conceptual design and will be updated with detailed design information in the application for an Operating License.

The I&C subsystems are categorized as either NonSafety-Related (NSR) or Safety Related (SR). Signal and information flow is one way from safety related to NSR and is electrically isolated by components within the SR system.

High reliability in safety systems is achieved through component selection, redundancy, and independence. Potential failures of the I&C system are eliminated by design, with emphasis on redundancy as is befitting a research reactor. The I&C system features a minimum of double redundancy on SR systems with steps taken to isolate and separate redundant I&C paths.

Signal pathways that communicate adversarial conditions are designed to comply with the building firefighting and accident analysis, and are certified to function as required in all anticipated accident and abnormal conditions. The control room employs modern tools to present an “at a glance” understanding of the reactor, with flexible and diverse display options aiding in reactor operation.

The passively safe design of the Molten Salt Research Reactor (MSRR) prompts a safety approach centered on protecting the fission product barriers and maintaining the safe envelope in which the MSRR operates. The fission product barriers are protected by automatic systems with alarms indicating the approach to operational limits. The integrity of the fuel-salt Reactor Protection System (RPS) is protected at all times and the I&C system restricts operations to when a drain is possible. The I&C system ensures the safe operation of the reactor by protecting the fission product barriers and controlling the reactor system so that draining is always available when required.

The MSRR I&C system consists of the Distributed Control System (DCS), the Reactor Protection System (RPS), the Engineered Safety Features Actuation System (ESFAS), the Human-Machine Interface (HMI), and the Radiation and Environmental Monitoring System (REMS).

Shutdown during and after a seismic event is achieved by independent components.

7.1.1 Calibration of Trips, Interlocks, and Annunciation

The DCS and RPS include trips, interlocks, and annunciations to monitor the operation of the process control and protection systems. For the RPS, this includes actuation of shutdown mode within a set of defined parameters after the onset of a postulated event.

Instrument uncertainty is considered in the determination of activation and actuation setpoints for trips, interlocks, and alarms. Setpoints are based on the following design principles:

- Maintaining operational control limits considering such variables as the time to reach limits, process conditions, and time delays.
- Maintaining operating assumptions and limits for analyzed events
- Mechanical and equipment limits
- Early detection of parameters that pose a risk to operation or of initiating a system trip.
- Annunciation of conditions that require system control response or potential operator intervention to maintain parameters within the normal operating envelope

Operational considerations such as drift, linearity, hysteresis, and operational margins are considered in the development of specific instrument loop setpoints. Consideration is also given to fixed instrument errors and environmental effects in the selection of instrument setpoints.

7.1.2 Distributed Control Systems

The DCS functions include control of plant equipment for startup, operation, and normal shutdown of the reactor, monitoring plant status, and data collection.

Data is transferred unidirectionally from SR components to NSR components. NSR displays in the control room are used to display data from both SR and NSR systems. The DCS uses information from the SR components for control functions for NSR components. The system is designed such that no failure of the DCS can prevent the RPS and ESFAS from functioning.

7.1.3 Reactor Protection System and Engineered Safety Features Actuation System

The RPS and ESFAS perform continuous monitoring of SR components by comparing measured values and alarming if predetermined limits are exceeded.

The RPS and ESFAS are implemented in a single subgroup. Their components are classified as safety related and are designed and maintained based on U.S. Nuclear Regulatory Commission (NRC) and industry guidance for SR systems, as described in this document.

The RPS consists of a series of minimum double-redundant sensors monitoring key system-limiting variables, automatic protective circuits, and manual reactor trip switches, along with system self-fault detecting fail safes. It is grouped with the ESFAS. Because of the passively safe design of the MSRR, protective systems focus on preserving fission product barriers and maintaining the safe operational envelope.

The RPS monitors core power with at least two safety channels across the full subcritical and power operation range and above, accounting for hypothetical accidents.

The system limiting temperatures are monitored by minimum double-redundant thermal sensors at key points throughout the reactor thermal management system (RTMS). The high temperature limit is monitored along the system's hottest points to prevent degradation of the material properties of the reactor system barrier. The low temperature limit is monitored throughout the RTMS to prevent approaching fuel freezing conditions.

The fuel salt level in the reactor loop is monitored to ensure that fuel is at the correct level for normal operation before start up and during operation. Pressure sensors provide an indication of system breaks or leaks and provide a signal for protective action in the event of significant pressure change indicative of an accident.

If the RPS detects a signal that, accounting for well-characterized system error, places the system outside its acceptable operational envelope, or one of the system fault or reactor trip switches is triggered, the RPS begins a protective action draining fuel salt. These protective actions are initiated by fail-safe systems that default to a safe-shutdown state in the event of total loss of power. Well-characterized gas control valves are used for this role. Once a protective action is initiated, it progresses to completion and only is available for reset once the initiating cause has been cleared.

7.1.4 Distributed Control System and Control Room

The DCS is used to monitor plant process measurements from sensors, to provide control signals to components, to provide indication to the operator of plant condition, and to alarm conditions requiring operator attention.

The control room monitors display both SR and NSR data. The DCS makes use of information from SR systems passed through one way communications protocols.

The control console provides sensor information and the controls necessary to operate the plant and to mitigate analyzed events. The control console is located in the control room and is the principle human machine interface (HMI). Multiple control console displays capable of displaying the same information are available for reliability and human factors.

The DCS monitors the health of facility systems and components. It receives self-diagnostic alarms from facility components. Trouble or anomalous comparison results are alarmed and displayed on control console monitors.

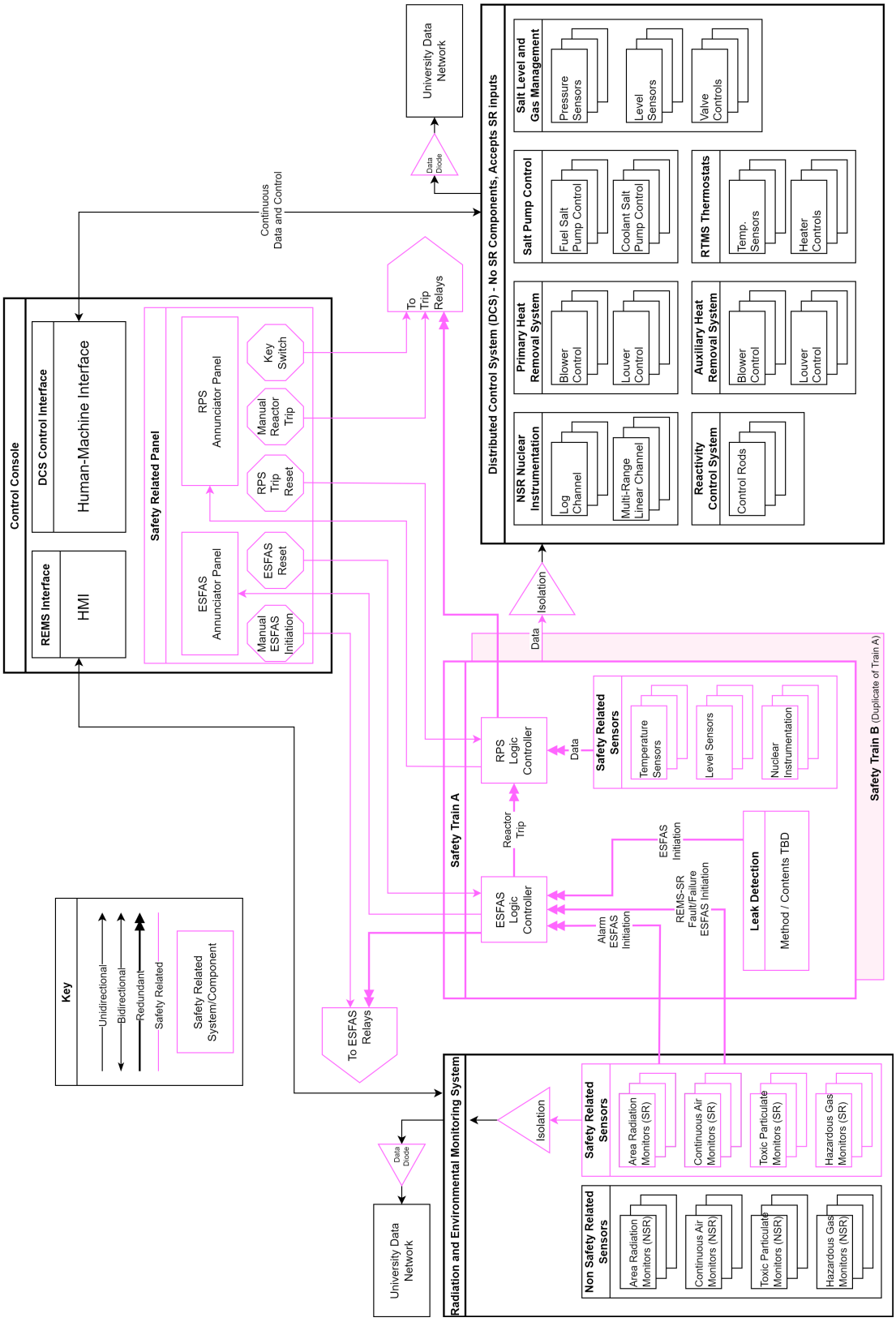
7.1.5 Radiation and Environmental Monitoring System

The REMS provides data to the ESFAS and control room and can trigger control room alarms. Instruments include both area radiation monitors and continuous air monitors placed throughout the various air volumes of the facility. During normal operation, areas containing dangerous radioactive environments are inaccessible; so the safety of personnel and the public can be ensured by monitoring for leaks and ensuring the accessible areas stay in safe limits.

7.2 Design of Instrumentation and Control Systems

The I&C system is designed to meet the requirements established by the design criteria and associated design bases and guidelines.

Figure 7.2-1 Instrumentation and Controls Diagram with Subsystems



7.2.1 System Description

The I&C system is separated into subsystems as follows and described in the sections noted:

- Distributed control systems (Section 7.3)
- Reactor protection system (Section 7.4) and the ESFAS (Section 7.5)
- Human Machine Interface (Section 7.6)
- Radiation and environmental monitoring system (Section 7.7)

7.2.2 System Performance Analysis

The RPS and ESFAS automatically trigger protective actions if set limits are reached. The limits and set points will be discussed in the Operating License application.

7.2.3 Conclusion

The I&C system meets the design criteria listed in NUREG-1537, in ORNL 1478, and in the design bases established herein from those criteria. The ability to passively cool eliminates loss-of-cooling accidents, while the robust construction and passively-safe nuclide retention controls any radioactive release to the public.

7.3 Distributed Control System

The DCS consists of the reactor controls. Its functions include controlling plant equipment for startup, operation, and normal shutdown of the reactor, monitoring plant status, and data collection.

Data is transferred unidirectionally from SR components to NSR components. The DCS displays are used to display data from both SR and NSR systems. The system is designed such that no failure of the DCS can impact the safe operation of the facility or prevent the RPS or ESFAS from functioning.

The DCS is composed of several subsystems in logical groups that monitor and control aspects of plant function. These include:

- Three control rod drives
- Log and linear power nuclear instrumentation channels
- Salt level and cover gas management system
  - Cover gas control valves
  - Cover gas supply and pressure sensors
  - Salt level sensors
- Primary and auxiliary heat removal cooling air process monitors and controls
  - Air blower power controllers
  - Air flow sensors

  - Louver position sensors and controls
  - Cooling air state monitoring sensors
- Salt pump monitoring and control systems for reactor loop and coolant salt loops
  - Salt pump power controllers
  - Pump power gauges
  - Gas seal monitors
  - Pump motor thermocouples
- System heaters and temperature monitors and controls
  - Electrical heater power controllers
  - Electrical heater power gauges
  - System temperature sensors

System interlocks for startup and operation will be discussed in the Operating License application.

7.3.1 Design Criteria and Bases

Consistent with DC 10, the DCS is designed with appropriate margin to ensure that radionuclide release design limits are not exceeded.

Therefore, the DCS shall feature interlocks to ensure that both automatic and operator action can maintain control of the system during normal operation and transients. The DCS shall feature fault and cross-checking ability sufficient for the detection of equipment failure or abnormal behavior.

Consistent with DC 13, the DCS is designed to provide both sensors and controls for systems and the facility across the full range of anticipated normal operational and transient conditions.

Therefore, the DCS shall feature sensors capable of monitoring variables across the full range of potential values for anticipated normal operational and transient conditions, and controls either designed or set to maintain variables inside the prescribed operating range.

Consistent with DC 15, the DCS is designed with sufficient margin to ensure the design conditions of the reactor fuel salt boundary are not exceeded during normal operation and transient conditions.

Therefore, the DCS shall not permit inputs or system behaviors that would exceed the design conditions of the reactor fuel salt boundary.

Consistent with DC 17, the DCS is provided with electrical power.

Therefore, the facility electrical supply shall meet the need of the DCS.

Consistent with DC 24, the DCS is separated from the protection system.

Therefore, the DCS shall be separated through unidirectional communication protocol from the protective systems to prevent any interference with protective system initiation or function.

Consistent with DC 28, the control rod drive system is designed with limits on the amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor fuel salt boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the fuel salt.

Consistent with DC 27, the DCS is designed to control reactivity through control rods. The control rods are not needed to safely shut down the reactor and are not safety related. The control rods provide control of reactivity and core power.

Consistent with DC 72, the DCS provides controls for the heaters for SR SSCs, that contain or could be required to contain salt, that ensure appropriate temperature distribution and rate of change.

Therefore, the DCS shall feature thermostats for heaters for SR SSCs that do not permit temperature change or distribution outside design limits allowing for a single failure.

Regulatory guidelines of note include local building and fire codes, NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” [Reference 7.8-1] and ORNL/TM-2020/1478 [Reference 7.8-2].

7.3.2 Control Rod Drives

Three control rod drives provide control of core power and reactivity insertions. The control rod design employs existing, proven control rod designs and drives. The position and state of the control rods is continually available to operators as part of the DCS and HMI.

7.3.3 Nuclear Instrumentation

Instrument selection for reactor control uses the best available commercial sensors. Reliability is achieved using well-proven technology with the ability to self-calibrate and fault detect in conjunction with the DCS monitoring system health. The accuracy of the sensor channel is calculated and accounted for in the selection of control setpoints. Control system uncertainty is accounted for in the determination of setpoints and will be discussed in the Operating License application.

Nuclear instrumentation consists of a linear power channel and a log power channel. Together they provide power information over the full power range of the system, from subcritical multiplication through credible accident scenarios. The log channel primarily is used for start-up and low-power operation. The linear channel is used for full-power operation.

This data is available continually on the DCS for operators and presented alongside the other neutron detectors, which are described in greater depth in Section 7.6.3.

7.3.4 Salt Level and Cover Gas Management System

This subgroup contains the cover gas control valves, cover gas supply, and the salt level detectors.

The cover gas control valves, in concert with the cover gas pressure supply and reactor pump, provide control of the salt level in the reactor access vessel. Salt is forced upward into the reactor system by delivering pressurized cover gas to the drain tank. The level of the fuel salt in the reactor access vessel can be set by adjusting the gas pressure. A separate system exists for the coolant loop to perform the same function. Monitoring systems and programs ensure valve operability over the lifetime of the plant.

The instrumentation and control of the cover gas compressors are included in this subgroup.

The salt level in the system is monitored at the top and bottom of both salt loops. In the reactor loop, this corresponds to the reactor access vessel and the drain tank; in the coolant loop, the surge and drain tank. Several sensor alternatives exist for this role, the final choices for which will be included in the Operating License application.

7.3.5 Auxiliary Heat Removal Cooling Air Process Monitoring and Control

This subgroup includes the cooling air blower power controls, louver controls, and corresponding sensors for the auxiliary heat removal system that cools the reactor cell. Sensors monitoring the variables of the state of the cooling air in the system inform on the heat load being removed by the system. Air blowers with corresponding controls and louvers allow the flow rate to be adjusted based on environmental conditions to ensure the cooling requirements of the system are being met. Additionally, in the event of a protective action, the flow louvers can be closed to achieve the system designed leak rate for the reactor cell [Section 6.2.1.2].

7.3.6 Primary Heat Removal Cooling Air Process Monitoring and Control

This subgroup includes the cooling air blower power controls, louver controls, and corresponding sensors for the primary heat removal system. Sensors monitoring the variables of state of the cooling air in the system inform on the heat load being removed by the system. Air blowers with corresponding controls and louvers allow the flow rate to be adjusted based on environmental conditions to ensure the cooling requirements of the system are being met.

7.3.7 Salt Pump Monitoring and Control

This subgroup consists of the pump controls and monitoring equipment for both the reactor loop and coolant loop. Salt flow rate influences core reactivity; thus, the reactor pump provides one of the controls of core power and reactivity insertions. Power draw from the pump is monitored, as changes in power draw can be indicative

of changing salt flow conditions. The pump motor temperature is monitored as part of the health of system monitoring and as a design limitation. This will be discussed in more detail in the Operating License application.

7.3.8 System Heaters Temperature Controls and Monitors

The system heater controls consist of various power relays for the electric resistive heaters for both the reactor loop and coolant loop and thermocouples to monitor system temperature. Thermocouples are located throughout the salt loops on the exteriors of the salt piping, heat exchangers, drain tanks, reactor access vessel, and the RTMS. They function to bring the system, in a controlled fashion, to operational temperatures prior to salt being raised into the reactor system and coolant loop and then maintain the temperature profile as needed during operation. Also included are sensors tracking the power draw of the various heaters. This serves to track the health of the system.

7.3.9 System Evaluation

The DCS provides control of the facility and reactor across normal operational modes. It does not perform a safety function except that correct operation of the facility can preclude RPS and ESFAS actuations. Details on applicable design codes will be provided in the Operating License application.

7.4 Reactor Protection System

The RPS provides protection for reactor operations by initiating signals to mitigate the consequences of postulated events and to ensure safe shutdown. The RPS components are classified as safety related and are designed and maintained based on U.S. Nuclear Regulatory Commission and industry guidance for SR systems as described in this document. The RPS performs sense and command functions to meet analyzed limits based on the following:

- Protect the integrity of piping and components that contain the fuel salt or the cover gas exposed to the fuel salt
- Maintain the assumptions of limiting plant operating parameters assumed in the Safety Analysis
- Actuate equipment when safety setpoints are reached to prevent a Chapter 13 event

The RPS consists of a series of minimum double-redundant sensors monitoring key system limiting variables; automatic protective circuits, and reactor trip switches. It is grouped with the ESFAS. Because of the passively safe design of the MSRR, protective systems focus on preserving fission product barriers and maintaining the safe operational envelope.

If the RPS detects a signal that, accounting for well-characterized system error, would place the system outside its acceptable operational envelope, or one of the system fault or reactor trip switches is triggered, the reactor protection system begins a protective action, draining fuel salt. These protective actions are initiated by fail-safe systems that default to a safe-shutdown state in the event of total loss of power or loss of outside

system control. Well-characterized gas control valves are utilized for this role. Once a protective action is initiated, it will progress to completion and only be available for reset once the initiating cause has been cleared.

SR systems automatically initiate and control protective actions. Operators are not required to take any action to achieve shutdown conditions for Chapter 13 events. Protective features and reactor trip also can be actuated manually.

The potential for common cause failures is minimized to the extent practical using physical separation and independence for the following:

- Redundant components performing the same SR function
- Automatic and manual methods of achieving a protective action
- Primary and diverse methods available to achieve reactor shutdown conditions

Safety-related I&C component reliability is established by type testing, previous operating experience, analysis, or combination of the three methods. SR components are compatible with the full range of service, environmental, and electromagnetic conditions they may experience. SR functions are testable on demand.

Table 7.4-1 Deleted

7.4.1 Design Criteria and Bases

Consistent with DC 1, the RPS must be designed, fabricated, erected, and tested in keeping with the requirements for SR SSC's, retaining functionality throughout and following a postulated accident.

Therefore, the RPS shall be designed, fabricated, erected, and tested with the highest degree of scrutiny for MSRR systems. Redundancy, diversity, and failsafe components shall be employed across all levels of its design. Common cause failures will be examined, and potential for them minimized to the highest degree practicable. RPS components will be designed for the environment in which they operate accounting for credible accident conditions. The RPS shall employ multiple diverse shutdown mechanisms to ensure the system can function to completion upon initiation. The RPS shall be monitored and tested on a schedule determined by the ACU QAPD to ensure it remains capable of performing its safety function.

Consistent with DC 2, the RPS must be designed to function in the case of a bounding natural phenomenon in combination with all credible accident conditions.

Therefore, the RPS shall be designed, fabricated, erected, and tested with the bounding natural phenomena taken into account, allowing for the antagonistic combination of credible accident conditions. Components will be selected for their compatibility with natural phenomena induced conditions.

Consistent with DC 3, the RPS must be designed and located to minimize the probability and effect of fires and explosions.

Therefore, the RPS shall be designed, and fabricated utilizing, practical, noncombustible and fire-resistant materials. RPS components shall be located away from, or protected from potential fire or explosive hazards. Analysis shall show the consequences of credible fires or explosions will not prevent the protective function of the RPS.

Consistent with DC 4, the RPS must be designed, and utilize components that are compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

Therefore, the RPS shall be designed, and utilize components that are compatible with, all credible environments that could be encountered such that changes in environment from nominal operational conditions do not prevent the RPS from performing its protective function.

Consistent with DC 10, the RPS provides margins sufficient to ensure a release of radionuclides in excess of design limits is prevented.

Therefore, the RPS shall feature sufficient redundancy in sensors and equipment to ensure the successful competition of the protective actions of the system to prevent radionuclide release in excess of design limits. Initiation set points will take into account well characterized system error to ensure protective actions initiate in a timely enough manner to protect release design limits.

Consistent with DC 13, the RPS must provide both sensors and controls for systems and the facility across the full range of anticipated normal operational and transient conditions.

Therefore, the DCS shall feature sensors capable of monitoring variables across the full range of potential values for anticipated normal operational and transient conditions, and controls either designed or set to maintain variables inside prescribed operating range.

Consistent with DC 15, the RPS provides sufficient margin to ensure the design conditions of the reactor fuel salt boundary are not exceeded.

Therefore, the RPS initiation set points must take into account system error and system response times to ensure that the design conditions are not exceeded.

Consistent with DC 17, the RPS is provided with electric power.

Therefore, the RPS electrical supply shall be built to applicable standards to ensure its normal and reliable function.

Consistent with DC 20, the RPS must be capable of detecting when the MSRR and facility exceeds specified acceptable radionuclide release design limits and initiate protective actions.

Therefore, the RPS shall monitor safety parameters with at a minimum redundant sensors and automatically initiate protective action upon the detection of a signal that, accounting for well characterized system error, would violate safety limits.

Consistent with DC 21, the RPS must be reliable and testable. It must feature redundancy and independence in its design to aid in reliability. At a minimum no single failure can lead to a loss of protective function. Its functionality must be capable of independently testing individual channels.

Therefore, the RPS shall be present on two separate but identical safety trains. In this way a total failure of one train leaves the second retaining functionality. The trains are separated as practical to prevent common cause failures. Reliability will be achieved through the use of proven off the shelf commercial products, and the oversight of the ACU QAPD. The reading of every RPS safety channel shall independently be available through the one-way communication protocol with the DCS. This will facilitate the periodic functional testing of the RPS safety channel sensors. Manual initiators of the RPS will allow functional testing of protective functions.

Consistent with DC 22, the RPS must feature sufficient external and internal independence to prevent the effects from natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels from leading to a loss of protective function. Diversity in component design and principle of operation must be employed.

Therefore, the RPS shall feature components that are diverse, both in manufacture and principle of operation, to prevent common cause failure. Safety channels will be to the extent practicable functionally independent of each other to prevent knock on failures from progressing to other channels. The SR systems utilizes one way communication protocols to isolate comms from the NSR systems into the SR systems. Additionally, the SR systems exist on two separate safety trains which will be isolated to the extent practicable such that even in the total failure of one train the second can continue to operate.

Consistent with DC 23, the RPS must be designed to fail into a safe state in the event of a loss of power or adversarial environmental conditions.

Therefore, the RPS shall be designed such that it is forced to its safe state by default and requires electrical power to be brought to the operational state. In the event of a loss of power the RPS will, through reliable mechanical means, be brought to its safe state.

Consistent with DC 24, the RPS must be designed such that no failure of the control system or shared components between the control and protective systems can compromise the protective systems functionality.

Therefore, the RPS shall feature electrical and communication independence from the DCS, operating as an isolated system with communication between the systems using one way communication protocols, with data passed from SR to NSR systems.

Consistent with DC 25, the RPS must ensure radionuclide release limits are not exceeded during any operational occurrence accounting for a single malfunction of the reactivity control systems.

Therefore, the RPS shall automatically protect safety limits to prevent damage to radionuclide barriers. The RPS can accomplish this even accounting for a single failure of the reactivity control systems as the RPS possesses the ability to insert more negative reactivity than positive reactivity from the reactivity control system, with sufficient rapidity to prevent damage according to chapter 13 analysis.

Consistent with DC 26, the RPS must possess independent and diverse means of bringing the MSRR to safe shutdown during normal operation and following postulated accidents. These methods must be capable of inserting sufficient negative reactivity rapidly enough to prevent exceeding design limits for fission product barriers.

Therefore, the RPS shall incorporate multiple trip valves on separate drain lines passing both through and outside the RTMS. These lines will provide the independent and diverse means of inserting negative reactivity via draining the reactor. Draining the reactor inserts sufficient negative reactivity to maintain the system in safe shutdown following all postulated accidents, and allows the fuel salt to safely cool.

Consistent with DC 28, the reactor protection system is designed to trip on the rate of power increase or a high power level initiated by postulated reactivity accidents to ensure that the effects can neither (1) result in damage to the reactor fuel salt boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to significantly impair the capability to cool the fuel salt.

Consistent with DC 29, the RPS must be designed to assure an extremely high probability of bringing the MSRR to its safe shutdown configuration.

Therefore, the RPS shall be built with reliability as a central feature of its design. It shall feature redundancy and diversity in every aspect of its function. Components will be selected to satisfy the ACU QAPD requirements for SR components. The entire system shall be of a fail-safe design.

Regulatory guidelines of note include local building and fire codes, NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” [Reference 7.8-1] and ORNL/TM-2020/1478 [Reference 7.8-2].

7.4.2 Reactor Trip

Figure 7.4-1 Reactor Trip Relay Circuits

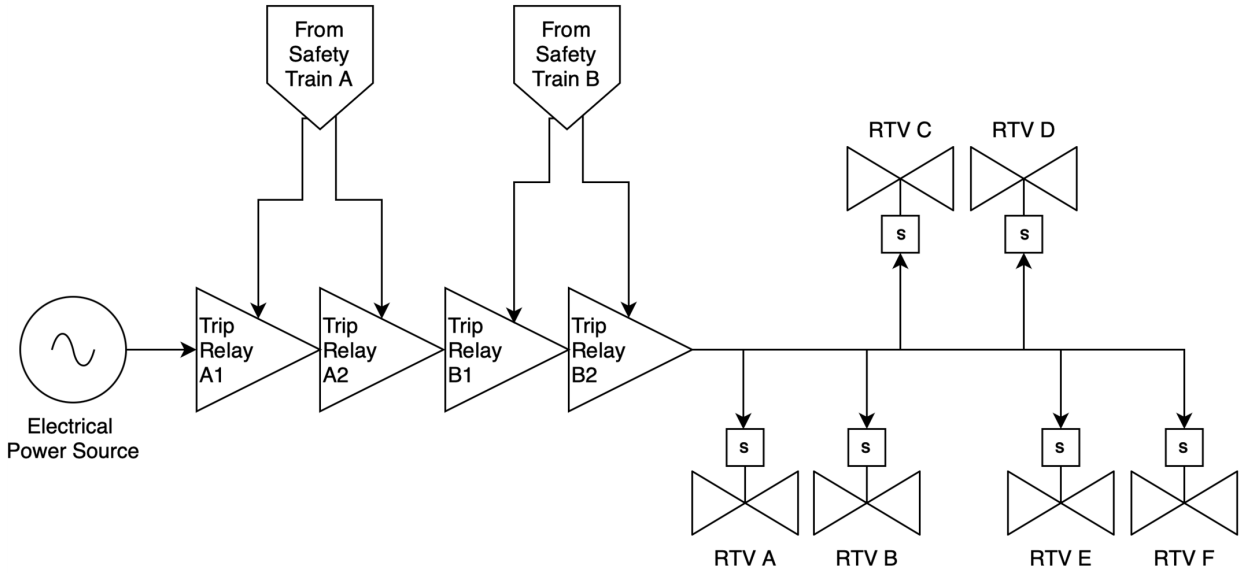


Figure 7.4-2 Reactor Trip Relay Circuits

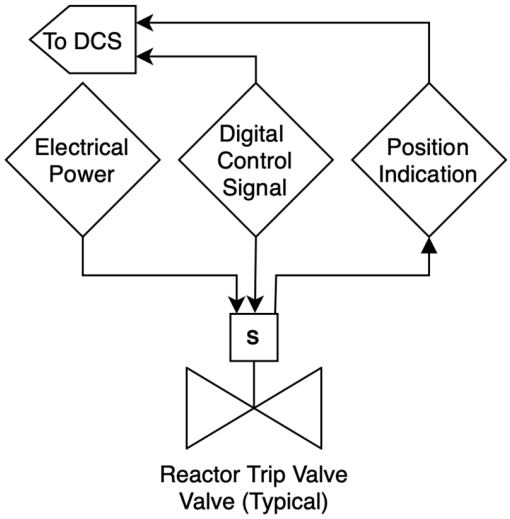


Figure 7.4-1 is a diagram of the Reactor Trip relay circuit. It displays how signals from the RPS actuate the trip relays to interrupt the electrical power to the Reactor Trip Valve solenoids. Figure 7.4-2 depicts the arrangement of inputs and outputs to the Reactor Trip Valve solenoids including the electrical power, as depicted in Figure 7.4-1.

The MSRR accomplishes a reactor trip through the exclusive use of gas valves. The control rods of the MSRR reactivity control system will only be utilized for controlling the power level of the reactor during normal operation as part of the DCS, playing no part in a trip. Figure 7.4-1 depicts the arrangement of the reactor trip relays with the reactor trip valves. Once a protective action is initiated, the action will progress to completion and only be available for reset once the protective action has been completed and the initiating cause cleared.

A reactor trip begins when the RPS logic controller receives a signal that, accounting for well-characterized system error, would place the system outside its acceptable operational envelope, or one of the system fault signals or a reactor trip switch is triggered. If either of the sensors for a trip channel on a safety train report a value outside the trip set points, then the RPS logic controller on that train will send signals for a reactor trip to that controller's trip relays.

The RPS logic controllers are field programmable gate array type controllers, such that a single controller features multiple parallel processes on one device. In this manner even if an entire process fails the RPS controller still has a redundant process to receive signals from sensors and generate trip signals to reactor trip relays.

Trip signals are passed along redundant cabling pathways each terminating at a different trip relay. Each controller sends signals to both of the trip relays to which it is connected for every initiation. The trip relays are arranged in series along the electrical power supply to the trip valve solenoids. Therefore, a single relay activation is sufficient to cut off electrical power to the trip valve solenoids.

The reactor trip valves are designed such that without electrical power they move to their safe, i.e. drain, state through the action of integrated mechanical components. The valves will be pneumatic type valves, relying on solenoid valves to switch between the pressure source and sink. When power is interrupted to the solenoids either deliberately through a relay activation, or as the result of a Loss of Normal Electrical Power (LONEP), the solenoids will move by design to the pressure sink, relieving the pneumatic pressure on the trip valves and causing them to open. Once the valves are open the salt will drain under all credible circumstances.

The trip valves consist of six valves on three gas lines. The number of valves and lines are chosen for reliability and diversity. The reactor may be operated with three of these valves locked open such that the opening of any one of the three closed valves is sufficient to drain the reactor. This provides equipment backups in the event of a failure of one of the valves in its open or safe state and preserve the ability to operate the reactor. Upon the loss of power, the valves will open connecting the gas volumes between the RAV, the fuel salt pump head space, and the fuel salt drain tank. This will equilibrate the gas pressure resulting in a fuel salt drain from the reactor loop into the drain tank. Three drain lines connect the reactor access vessel to the drain tank for reliability and diversity.

Because the reactor vessel, and the drain tank are all contained inside the thermally insulated boundary of the RTMS, fuel salt will remain molten for the period required to drain. The system is sub critical when the reactor core is 25 percent uncovered. In less than one minute the fuel salt level will be below the graphite. When the reactor is shutdown (4.2.2.1), the fuel salt inventory will relocate to the drain tank. As described in Chapter 4, there are no credible conditions for the drain tank to become critical.

7.4.3 RPS Trip Set Points

Trip channels and triggering values are listed in Table 7.4-2 and then discussed in greater detail in the following section, covering both the trigger points, safety limits, and how they will be monitored.

Table 7.4-2 Reactor Trip Set Points

Trip Channel	Triggering Value	Reference
Reactor Power	> ~1.0 MWth	14.2.2
Reactor Loop High Temperature	> 650 °C	4.3.8, 14.2.2
Reactor Loop Low Temperature	< ~550 °C	4.3.8, 14.2.2
Fuel Salt High Level	Below the non-salt wettable piping level	4.3.11, 14.3.1
Fuel Salt Low Level	Above the top of the salt outflow pipes	4.3.11, 14.3.1
Reactor Loop High Gas Pressure	> 0.5 MPa	4.3.7, 14.2.2
Reactor Loop Low Gas Pressure	< 0.1 MPa	4.3.7, 14.2.2

7.4.3.1 Reactor Power

Reactor thermal power is monitored by at least four calibrated detectors between two trains. The detectors are positioned according to analysis such that they can monitor core behavior across the full power range from subcritical multiplication through all credible accident scenarios. Current sensor selection is for the detectors to be uncompensated ion chambers.

The reactor power level trip setpoint will be set in the OL application when greater information regarding measurement accuracy and system performance will be available. It will be chosen to protect the licensed power level and is expected to be nominally 1 MWth.

7.4.3.2 Reactor Loop Temperature

The system has both high and low temperature limits. The high temperature safety limit is informed by the limiting temperature of the steel used to construct the salt system. The operating limit of 650 °C is set to prevent reaching the safety limit temperature during analyzed events. The low temperature safety limit is set by the salt freezing point, for which the precise values are unknown. It is expected to be < 500 °C and so operational limits are nominally set at 550 °C. The final limits will be set in the OL application when greater details on the properties of the fuel salt are available.

The system temperature is monitored by sets of three thermocouples placed in thermowells. The trains monitor shared sets of thermocouples and employs two out of three logic for every set. Thermowells are placed monitoring the limiting heating of the system, which is expected along the vessel head or outflow pipe, and along the system cold leg return at the heat exchanger exit and the salt drain line, to monitor and ensure the ability to drain is not compromised.

As described in the chapter 13 analysis most analyzed events do not require RPS actuation to maintain temperature limits. Section 13.1.5.10 details the maximum reactivity insertion event which was the analyzed event that caused a temperature limit to be reached.

7.4.3.3 Fuel Salt Levels

Salt level in the fuel salt system will be a limiting condition for operation. The GMS is not designed to withstand salt intrusion, and as such the salt level must be kept below the level of the GMS openings in the reactor loop. Additionally, during power operations involving the primary heat removal system, salt level in the reactor loop must be sufficiently high to establish a circulation loop, to prevent salt stagnation or restricted ability to cool the salt. Correspondingly, there is a minimum salt level requirement above the level of the outflow pipe from the fuel salt loop's RAV, and reactor pump bowl. The salt levels in the RAV, and the reactor pump bowl, are monitored to ensure the salt levels remain within the proscribed operational limits for the current operation state of the MSRR. The salt level monitoring will be accomplished by four salt level sensors each on two separate trains.

The chapter 13 analyzed event, Increase in Fuel Salt Inventory, described in Section 13.1.3 would activate the RPS through a salt level violation. 13.1.6 Mishandling or Malfunction of Fuel would result in a low-level trip.

7.4.3.4 Reactor Loop Gas Pressures

The pressure in the reactor loop is a safety limit. It is set by the mechanical properties of the system vessels and piping, and takes into account the behavior of radionuclides during potential accidents. The safety limit for the reactor loop will likely be between 0.1 and 0.5 MPa. Pressure limits may also be set by the pump seal requirements. The gas pressure is monitored at the gas head spaces, through attached gas piping. The pressures are monitored using redundant pressure sensors on two trains.

Pressure limits are monitored, but in the vast majority of cases it does not fluctuate during analyzed chapter 13 events. Barrier failures would trigger pressure limits.

7.4.4 System Trips

Table 7.4-3 System Trip Channels

System Trip	Trigger	Reference
ESFAS Initiation	Initiation	7.4.4.1
Trip Switch	User initiation	7.6.2
Power Safety Channels High Voltage	Manufacturer Specified Operational Range Limits	7.4.4.3
Fire Detection	Fire Detection	9.3.2
Key Switch	Initiation	7.4.4.5

Table 7.4-3 shows system trip channels alongside their triggers and references to PSAR sections explaining why those triggers were selected.

7.4.4.1 ESFAS Initiation

The initiation of the ESFAS always triggers the RPS. The AHRS cooling air serves both to protect the mechanical properties of the materials that comprise the reactor cell, and to maintain the initial conditions of any accident analysis. The accident analyses show that, starting from the temperature profile maintained by the auxiliary cooling system, the system retains sufficient thermal inventory to absorb and dissipate heat from an accident. Because the activation of the ESFAS restricts the flow of cooling air, any additional power operation after ESFAS initiation will place the system outside the initial conditions of the accident analyses. Therefore, to maintain the conditions upon which the accident analyses were performed the RPS must always initiate and bring the reactor to a shutdown state following ESFAS initiation.

7.4.4.2 Trip Switch

A trip switch signal directly triggers the reactor trip relays. It is expected that there will be multiple reactor trip switches in addition to the one located in the main control room.

7.4.4.3 Loss of Power Safety Channels High Voltage

Power safety channels high voltage outside of the acceptable range for operation of the detection system will trigger a trip.

7.4.4.4 Fire Detection

The detection of a fire in the facility triggers a trip.

7.4.4.5 Key Switch

Without the key inserted the trip valves cannot be closed. If the key is removed while the system is operating a trip occurs.

7.4.5 System Evaluation

The RPS provides protection for reactor operations by initiating signals to mitigate the consequences of postulated events and to ensure safe shutdown. The design is preliminary and the final design will be provided in the Operating License application.

7.5 Engineered Safety Features Actuation System

The ESFAS consists of the SR cabling and connections that deliver sensor signals to the ESFAS logic controllers on the safety trains, which are connected to the SR sensors of the Radiation and Environmental Monitoring System (REMS) and leak detection apparatus, and also includes the ESFAS initiation relays and attached SR equipment such as louvers and gas valves that can be initiated to bring the facility to its passively safe, lowest-leakage, cold, secure state. In that state the facility will remain subcritical and secure until the ESFAS initiation is cleared and reset.

The ESFAS initiation occurs automatically with the detection of a failure in one of the fission product barriers or the detection of elevated radionuclide concentrations outside of fission product barriers. The detailed approach to detection of a failure in the fission product barriers is still under development, but may include the detection of an unexpected loss of cover gas pressure, fuel salt level, or an unexpectedly elevated radiation field. Detection of elevated radionuclide concentrations outside barriers is accomplished by SR sensors from the REMS and passed in a robust and reliable manner to ESFAS. Activation of the ESFAS occurs separately from the RPS system, but an ESFAS initiation will trigger the RPS to trip. A manual ESFAS initiator is included in the controls present in the control room SR panel.

The ESFAS is a distributed system with initiating devices executing the ESFAS logic present on both safety trains. Signals from the sensor pairs pass through the ESFAS controllers on their way to the DCS and the control room. If any of the signals violate set limits a ESFAS signal will be sent to the ESFAS initiation relays bringing the system to its designed accident configuration. As with all safety systems in the MSRR this will be accomplished by a minimum of two separate signal pathways traveling along physically divided paths. Cabling suitable for the MSRR will be utilized and will be physically protected to meet the MSRR firefighting and accident plans.

The safety trains encompass both the RPS and ESFAS. An ESFAS initiation triggers the closing of the auxiliary cooling system louvers and correspondingly necessitates the shutdown of the AHRS. ESFAS initiation triggers the RPS automatically, but a reactor trip does not necessarily trigger an ESFAS initiation. ESFAS initiation only occurs upon detection of a radioactive release, since during an anticipated operational occurrence that does not require restriction of the AHRS to prevent release of radionuclides the additional cooling capacity is desirable. The ESFAS is triggered by detection of a leak or of radionuclide levels outside set points. Any initiation of the ESFAS triggers the RPS as the MSRR relies on the auxiliary cooling system to maintain the limiting conditions for operations related to the reactor cell.

7.5.1 Design Criteria and Bases

Consistent with DC 1, the ESFAS must be designed, fabricated, erected, and tested in keeping with the requirements for SR SSC’s, retaining functionality throughout and following a postulated accident.

Therefore, the ESFAS shall be designed, fabricated, erected, and tested with the highest degree of scrutiny for MSRR systems. Redundancy, diversity, and failsafe components shall be employed across all levels of its design. Common cause failures will be examined, and potential for them minimized to the highest degree practicable. ESFAS components will be designed for the environment in which they operate accounting for credible accident conditions. The ESFAS shall employ multiple diverse protection mechanisms to ensure the system can function to completion upon initiation. The ESFAS shall be monitored and tested on a schedule determined by the ACU QAPD to ensure it remains capable of performing its safety function.

Consistent with DC 2, the ESFAS must be designed to function in the case of a bounding natural phenomenon in combination with all credible accident conditions.

Therefore, the ESFAS shall be designed, fabricated, erected, and tested with the bounding natural phenomena taken into account, allowing for the antagonistic combination of credible accident conditions. Components will be selected for their compatibility with natural phenomena induced conditions.

Consistent with DC 3, the ESFAS must be designed and located to minimize the probability and effect of fires and explosions.

Therefore, the ESFAS shall be designed, and fabricated utilizing, practical, noncombustible and fire-resistant materials. ESFAS components shall be located away from, or protected from potential fire or explosive hazards. Analysis shall show the consequences of credible fires or explosions will not prevent the protective function of the ESFAS.

Consistent with DC 4, the ESFAS must be designed, and utilize components that are compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

Therefore, the ESFAS shall be designed, and utilize components that are compatible with, all credible environments that could be encountered such that changes in environment from nominal operational conditions do not prevent the ESFAS from performing its protective function.

Consistent with DC 10, the ESFAS must be designed with margins sufficient to ensure a release of radionuclides in excess of design limits is prevented.

Therefore, the ESFAS shall feature sufficient redundancy in sensors and equipment to ensure the successful competition of the protective actions of the system to prevent radionuclide release in excess of design limits. Initiation set points will take into account well characterized system error to ensure protective actions initiate in a timely manner to protect release design limits.

Consistent with DC 17, the ESFAS is provided with electrical power.

Therefore, the ESFAS electrical supply shall be built to applicable standards to ensure its normal and reliable function.

Consistent with DC 20, the ESFAS must be capable of detecting when a release in excess of specified acceptable radionuclide release design limits has occurred and initiate protective actions.

Therefore, the ESFAS shall monitor for release and automatically initiate protective action upon the detection of a signal that, accounting for well characterized system error, reaches safety setpoint(s). Should sensors or monitoring systems fail the ESFAS will initiate a protective action until such time as sensors can be repaired, or additional sensors brought online.

Consistent with DC 21, the ESFAS must be reliable and testable. It must feature redundancy and independence in its design to aid in reliability. At a minimum no single failure can lead to a loss of protective function. Its functionality must be capable of independently testing individual channels.

Therefore, the ESFAS shall be present on two separate but identical safety trains. In this way a failure of one train leaves the second retaining functionality. The trains are separated as practical to prevent common cause failures. Reliability will be achieved through the use of proven off the shelf commercial products, and the ACU QAPD. The reading of every ESFAS safety channel shall independently be available through the one-way communication protocol with the REMS and or the DCS. This will facilitate the periodic functional testing of the ESFAS safety channel sensors. Manual initiators of the ESFAS will allow functional testing of protective functions.

Consistent with DC 22, the ESFAS must feature sufficient external and internal independence to prevent the effects from natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels from leading to a loss of protective function. Diversity in component design and principle of operation must be employed.

Therefore, the ESFAS shall feature components that are diverse, both in manufacture and principle of operation, to prevent common cause failure. Safety channels will be to the extent practicable functionally independent to prevent knock on failures from progressing to other channels. The SR systems utilizes one way communication protocols to isolate comms from the NSR systems into the SR systems. Additionally, the SR systems exist on two separate safety trains which will be isolated to the extent practicable such that a failure of one train the second can continue to operate.

Consistent with DC 23, the ESFAS must be designed to fail into a safe state in the event of a loss of power or adversarial environmental conditions.

Therefore, the ESFAS shall be designed such that it is forced to its safe state by default and requires electrical power to be brought to the operational state. In the event of a loss of power the ESFAS will through reliable mechanical means be brought to its safe state.

Consistent with DC 24, the ESFAS must be designed such that no failure of the control system or shared components between the control and protective systems can compromise the protective systems functionality.

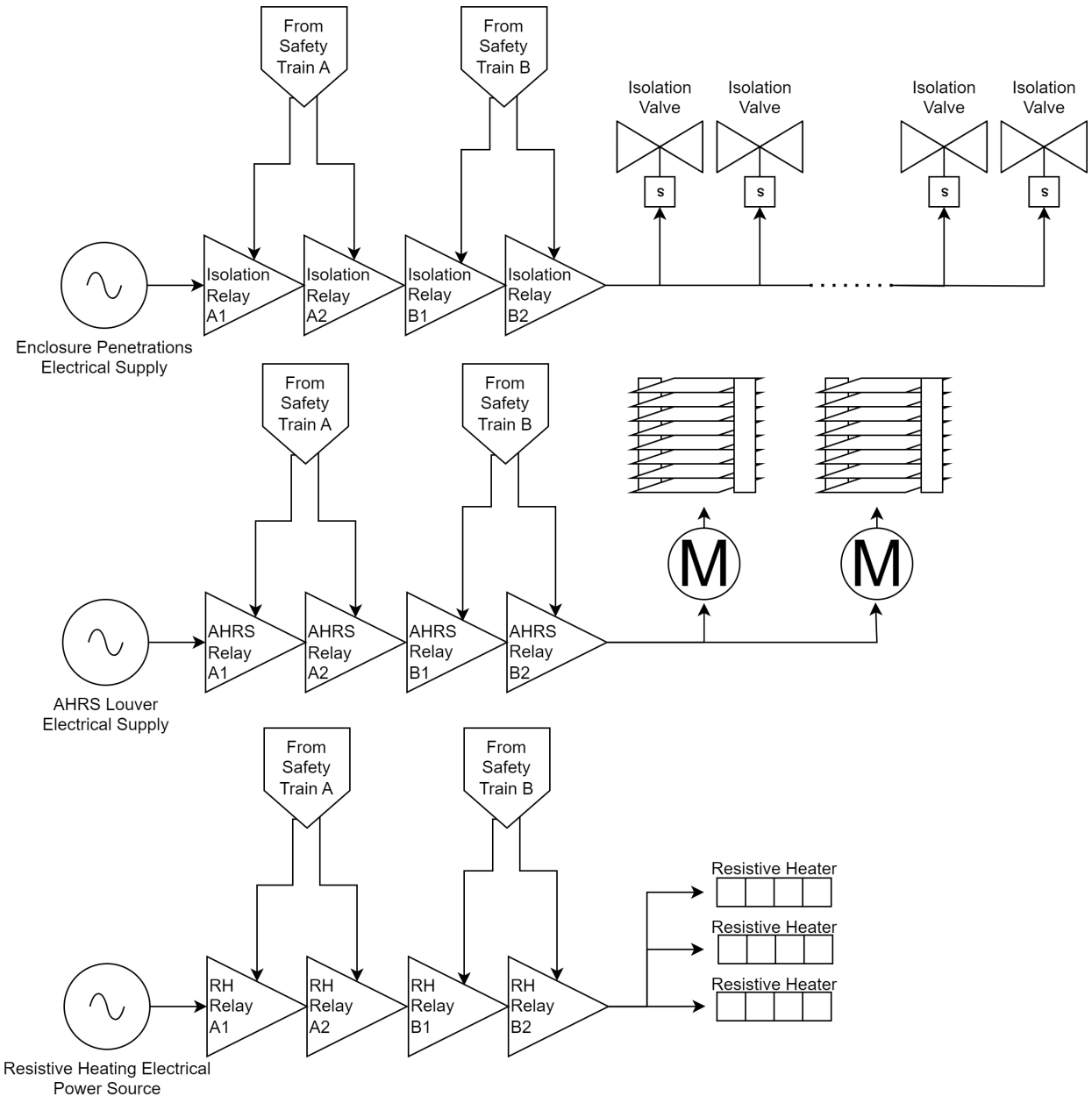
Therefore, the ESFAS shall feature communication independence from the DCS and NS REMS, operating as an isolated system with communication between the systems accomplished utilizing one-way communication protocols, with data passed from SR to NSR systems.

Consistent with DC 29, the ESFAS must be designed to assure an extremely high probability of bringing the MSRR to its lowest leakage configuration.

Therefore, the ESFAS shall be built with reliability as a central feature of its design. It shall feature redundancy and diversity in every aspect of its function. Components will be selected to satisfy the ACU QAPD requirements for SR components. The entire system shall be of a fail-safe design.

7.5.2 ESFAS Initiation

Figure 7.5-1 ESFAS Actuation



The three diagrams in Figure 7.5-1 depict the constituent actions of an ESFAS actuation. All three feature the use of relays arranged in series on the electrical supply of a system. Connections to the safety trains show how signals coming from the ESFAS controllers present on the safety trains can activate these relays to terminate electrical power to the relevant systems. From the top to bottom the diagrams are of the reactor enclosure isolation valves, the AHRS louvers, and the resistive heating in the reactor enclosure.

The ESFAS brings the MSRR into its passively safe designed leak rate configuration. It does this by sealing the air passages to and from the reactor cell air space, by sealing valves on all gas penetrations through the reactor enclosure, and disconnecting power to the electrical resistive heaters in the reactor enclosure. This is accomplished using relays that terminate electrical power to the failsafe ESFAS equipment causing it to move to its safe state.

The reactor cell air passages contain louvers capable of achieving the designed system leak rate. Following the receipt of an ESFAS signal the louvers will be closed by fail safe drives. The drives will be such that the loss of power brings the louvers to the sealed position.

The enclosure penetrating lines all feature at least a pair of valves, each rated to independently isolate the line. The valves fail safe such that in the event of a loss of power the valves seal the system to its designed accident configuration.

The electric resistive heaters present in the reactor enclosure receive power through relays that can be initiated by the ESFAS logic controllers. The ESFAS heater power disconnect relays are arranged in series such that the initiation of any of them will terminate electrical power to the heaters.

7.5.2.1 Reactor Enclosure Isolation System

Many lines penetrate the reactor enclosure, including, but not limited to the lines used to add helium to the medium and low-pressure helium tanks and take cover gas out of the enclosure, lines carrying the pneumatic gas used to actuate the reactor trip valves, the beryllium and UF4 addition lines, and any sampling lines.

The isolation system closes all necessary penetrations to achieve the low leakage condition. It consists of at a minimum one pair of fail-safe valves on each line that will seal closed upon initiation. The valves are arranged in series on each line such that the initiation of either valve successfully isolates the line, and by extension the enclosure. Each ESFAS logic controller on both safety trains actuates both isolation valves. The valves are such that without applied power the valves fail to the closed state through a reliable mechanical design. ESFAS initiation therefore actuates the valves by utilizing relays in series on the power supply to the valves to terminate electrical power to the valves and place them in their safe state.

7.5.2.2 Reactor Cell Air Louvers

During normal operation, the auxiliary heat removal system circulates 4000 cubic feet per minute of cooling air through the reactor cell to maintain an acceptable temperature in the reactor cell and by extension the reactor enclosure. Upon initiation, the ESFAS closes the auxiliary heat removal system intake and exhaust louvers automatically to control leakage from the cell.

The reactor cell air intake and exhaust contain louvers capable of achieving the designed system leak rate. Following the receipt of an ESFAS signal the louvers are closed by fail safe drives. The drives are such that the loss of power brings the louvers to the sealed position. Each ESFAS logic controller on both safety trains actuates both louvers. ESFAS initiation actuates the louvers by utilizing relays in series on the power supply to the louvers to terminate electrical power to the louvers and place them in their safe state.

7.5.2.3 Heater Power Disconnect

Power to the resistive heaters in the reactor enclosure is passed through relays that can be actuated by the ESFAS system. The relays are arranged in series such that the actuation of any of the relays terminates power to the resistive heaters. Each ESFAS logic controller features a redundant relay and actuates both relays upon initiation.

7.5.3 Initiation Limits and Signals

ESFAS initiating limits will be set in the operating license application. While the variables that will be monitored and the systems employed to monitor them are known, the exact limits will not become available until more progress has been made on the detailed engineering and design of the MSRR. Table 7.5-1 contains the ESFAS initiation channels that the ESFAS logic controller will monitor using the SR sensors incorporated into the ESFAS.

Table 7.5-1 ESFAS Initiation

System Initiation Channel	Triggering Value
REMS Limits	TBD-Preserve facility release limits
REMS-SR Fault/Failure	System fault detection
Leak Detection	TBD-Pressure/Salt level change/Radiation or gas

7.5.3.1 REMS Limits

The SR sensors in the REMS are utilized to monitor the reactor enclosure, reactor cell, and research bay for the presence of radionuclides and dangerous gas and particulate concentrations outside acceptable limits. When concentration of radionuclides or dangerous gas and particulates exceed set limits the ESFAS logic controllers will generate an ESFAS initiation signal and actuate the ESF protective equipment.

The triggering valves of those limits will be determined as part of the detailed engineering and design of the MSRR and will be set such that, accounting for well characterized system error and performance, release limits are not exceeded.

7.5.3.2 REMS SR Fault/Failure

The SR REMS sensors and equipment have self-fault detecting circuits and monitoring equipment. If a fault is detected in SR REMS equipment ESFAS will initiate until such a time as SR equipment can be brought back online.

7.5.3.3 Leak Detection

A direct detection of a leak or failure of one of the fission product barriers will initiate the ESFAS. The detailed approach to leak detection is still under development. One signal that is expected to contribute to leak detection is cover gas pressure. Sudden sharp transients in cover gas pressure indicative of a pressure barrier failure may be used as part of leak detection. Another may be salt level detection, depending on the resolution of level sensors differing levels of leakage may be detectable through the detection of unexpected fuel salt level drops. Finally, REMS sensors may be utilized to detect unusual changes in the radiation fields of the MSRR or vessel gas chemistry make up that could be indicative of a leak.

7.5.4 System Evaluation

Based on inputs from the REMS and leak detection sensors, the ESFAS isolates the reactor cell by closing the air louvers in the auxiliary heat removal system, closes the reactor enclosure penetration isolation valves, and disconnects power to the reactor enclosure resistive heaters. By doing this the ESFAS brings the reactor cell and enclosure into their passively safe low leakage state.

7.6 Human-Machine Interface

The HMI consists of control room displays, input devices, and the auditory announcement system which are primarily located on the control console, creating a digital interface with the DCS and REMS to control the reactor and facility. Utilizing integrated facility communications, it will allow for normal operation and routine activity to be coordinated from the control room. It provides controls for DCS components, REMS components, ESF systems that are utilized in routine operation, and the reactor trip valves. It can display all information recorded by the I&C system through any of the subsystems including SR data passed through one way communications from the RPS and ESFAS. The console also includes a key switch without which the reactor trip valves cannot be closed, locking the reactor in a safe state and preventing unauthorized operation of the facility.

A record of incoming sensor readings, operator inputs, and the current facility status is preserved. The DCS and HMI are designed by a certified, commercial entity to meet the needs of the MSRR and applicable regulatory requirements. Regulatory standards and best practices are observed in the creation of the DCS and HMI and are used as guidelines for creating a safe control apparatus with appropriate access restrictions and information security to ensure only authorized users gain access to the system.

7.6.1 Design Criteria and Bases

Consistent with DC 13, the HMI must provide both sensors and controls for systems and the facility across the full range of anticipated normal operational and transient conditions.

Therefore, the HMI shall feature sensors capable of monitoring variables across the full range of potential values for anticipated normal operational and transient conditions, and controls either designed or set to maintain variables inside prescribed operating ranges.

Consistent with DC 17, the HMI electrical supply must be designed to provide reliable power.

Therefore, the electrical supply shall be built to applicable standards to ensure its normal and reliable function.

Consistent with DC 19, the control room must provide a suitable workspace to control the facility during normal operation and abnormal operational occurences, and be easily evacuated during accident conditions.

Therefore, the control room shall meet all applicable building codes and regulations to create a functional and habitable workspace for control of the facility, that can be quickly and safely evacuated during postulated accidents.

Regulatory guidelines of note include local building and fire codes, NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” [Reference 7.8-1] and ORNL/TM-2020/1478 [Reference 7.8-2].

7.6.2 Control Room

The reactor control room contains equipment related to normal operation of the plant. These include operator and supervisor workstation terminals, which provide alarms, annunciations, personnel and equipment interlocks, and process information. A full accounting of interlocks, equipment, and any potential safety bypasses will be included in the Operating License.

The control room console contains a manual trip switch that allows operators to initiate a reactor trip.

The control room will meet all relevant codes and promote safe evacuation during credible accidents. It is not intended that the control room have additional shielding, air handling to limit radiation exposure under postulated conditions, or that it be reinforced to ensure its survival under postulated conditions. Instead under postulated event conditions it will be evacuated.

7.6.3 Control Console and Human Machine Interface

Operators have access to all essential values and relevant secondary information and are able to control the reactor and facility systems through the control console located in the control room. A dedicated and clearly visible alarm panel is present with auditory and visual alarm features. Access to the control room is controlled to a level sufficient to ensure licensed operators are in control of the control room, the reactor, and the facility.

The control room console is designed to allow operators to manipulate plant parameters to control the reactor within an acceptable envelope during normal operating conditions, including planned transients. The control room consoles are designed as follows.

- Control room displays implement the guidance from NUREG-1537, Section 7.6, with respect to ease of operator use. The plant controls are grouped and located in the control room so that operators can easily reach and manipulate the controls. Displays of the results of an operator's actions are readily observable.
- The screen element organization and appearance of the consoles are designed to allow operators to perform actions to operate the reactor under normal operating conditions and to monitor it under postulated event conditions.

The control console provides access to the SR inventory of the control room through a SR panel. This is expected to include a minimal inventory of push buttons to allow for acknowledgment and reset of alarms and for triggering of a reactor trip and ESFAS initiation functions.

The control console provides for the calibration and testing of connected equipment from MSRR systems. The console also allows for inspection, calibration and testing of integrated console equipment.

Operators have access to all information that travels through the I&C system and are able to utilize secondary displays to display it. The arrangement of displays and grouping of information and controls is informed by human factors engineering principles and best practices.

The control console features a reactor key, without which the reactor trip valves are locked in their open, i.e. safe, configuration.

The control console may feature a backup power supply, however this would be included only for operator convenience and not as a safety feature, as the control room is not credited in any response to analyzed events.

7.6.4 Conclusions

The preliminary design for the control room, control console, and the human-machine interface satisfies its DC and design bases. The final design will be provided as part of the Operating License.

7.7 Radiation and Environmental Monitoring System

The REMS provides sensor readings of the radiation and environment conditions in the facility at the physical sensor locations, to the ESFAS controllers, the DCS, and the control room, as appropriate. It accomplishes this utilizing gas, particulate, and radiation sensors to characterize the environment within the facility. The sensor channels provide alarms suitable for their display location, including audible, visual, or both. The REMS has both SR and NSR sensors as part of its sensing apparatus and delivers SR signals to the ESFAS. These sensors are deployed in two configurations.

A selection of sensors is placed in sensor stations serviced by multiple sensing lines, allowing them to characterize the presence of chemical and radiological species in the atmosphere of the connected gas volumes. The sensing lines provide a constant flow of gas from connected gas volumes to allow continual characterization of the environment in the connected volumes. The sensor stations consist of a number of differing sensors including continuous air monitors (CAMs), area radiation monitors (ARMs), hazardous gas detectors, and particulate sensors. This allows a small inventory of sensors to characterize the environment in a multitude of connected gas volumes. These sensor stations provide indication of releases into the connected volumes and are used as part of leak detection.

The sensor stations are supplemented with individually placed sensors, positioned throughout the facility at locations of interest where sensing lines would be insufficient. These sensors include area monitors monitoring location specific radiation fields. Additionally, they may include supplemental chemical or radiation sensors in areas where sensing lines do not provide a thorough or rapid response.

7.7.1 Design Criteria and Bases

Consistent with DC 1, the SR components of the REMS must be designed, fabricated, erected, and tested in keeping with the requirements for SR SSCs, and commensurate with the importance of the safety function it performs.

Therefore, the SR components of the REMS shall be designed, fabricated, erected, and tested in keeping with the requirements determined by the ACU QAPD. REMS components will be designed for the environment in which they operate accounting for credible accident conditions. The REMS shall be monitored and tested on a schedule determined by the ACU QAPD.

Consistent with DC 2, the SR components of the REMS must be designed to function in the case of a bounding natural phenomenon in combination with all credible accident conditions.

Therefore, the SR components of the REMS shall be designed, fabricated, erected, and tested with the bounding natural phenomena taken into account, allowing for the antagonistic combination of credible accident conditions. Components will be selected for their compatibility with natural phenomena induced conditions commensurate with their importance to safety.

Consistent with DC 3, the SR components of the REMS must be designed and located to minimize the probability and effect of fires and explosions. Fire detection and fighting systems of sufficient capacity and capability must be present to minimize the adverse effects of fires on the REMS.

Therefore, the SR components of the REMS shall be constructed of noncombustible and fire-resistant materials wherever practical. The REMS shall be located away from explosive hazards and provided sufficient firefighting capacity consistent with its importance to safety.

Consistent with DC 4, the SR components of the REMS must be designed, and utilize components that are compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents commensurate with the systems importance to safety.

Therefore, the SR components of the REMS shall be designed, and utilize components that are compatible with, the environments associated with normal operation maintenance, testing, and postulated accidents to a degree commensurate with its importance to safety.

Consistent with DC 10, the REMS must be designed with margins sufficient to ensure a release of radionuclides in excess of design limits is prevented.

Therefore, the REMS shall feature initiation set points that take into account well characterized system error to ensure protective actions initiate in a timely manner to protect release design limits.

Consistent with DC 13, the REMS must provide both sensors for systems and the facility across the full range of anticipated normal operational and transient conditions.

Therefore, the REMS shall feature sensors capable of monitoring variables across the full range of potential values for anticipated normal operational and transient conditions.

Consistent with DC 17, the REMS is provided with electrical power.

Therefore, the REMS electrical supply shall be built to applicable standards to ensure its normal and reliable function.

Consistent with DC 63, the REMS must monitor fuel storage and waste storage areas for excessive radiation levels.

Therefore, the REMS shall include radiation and environmental sensors capable of monitoring the fuel storage and waste storage areas.

Consistent with DC 64, the REMS must monitor containment atmosphere, discharge pathways, and facility environment for radioactivity.

Therefore, the REMS shall include radiation and environmental sensors capable of monitoring the containment atmosphere, discharge pathways, and facility environment.

Regulatory guidelines of note include local building and fire codes, NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” [Reference 7.8-1] and ORNL/TM-2020/1478 [Reference 7.8-2].

7.7.2 Facility Sensor Stations

Sensor stations are built inside a sealed volume (or employ a semi sealed volume for stations respirating normal atmosphere) connected through sensing lines to the volumes of interest. Inside the sealed volume a collection of sensors is present including an ARM, a CAM, and hazardous gas and particulate monitors. A gamma spectrometer may also be included. The sensing lines will feature gas valves to isolate the volumes being monitored, in series with flow meters and pumps or pressure supplies to ensure adequate flow over sensors.

The REMS layout calls for at least two sampling locations servicing the differing gas volumes in the facility. Capabilities are provided for the detection of fission produced noble gasses, radioiodine, and particulates at each sampling location. The selected instruments and their configuration provide for detection over the full range of normal and abnormal conditions. There will be both SR and NSR sensor stations. The signals from SR sensor stations are passed to each safety train at each sampling location. The logic is any detection of radiation, radioisotopes, or chemical species above limits will trigger a protective action. A failure of SR REMS sensors that compromises the system's ability to protect limits also triggers a protective action.

One SR sensor station services the chemically inert air volumes in the facility, delivering inert atmosphere to the sensors through sensing lines. These volumes include the reactor enclosure. This sensor station may include other enclosures that utilize inert atmospheres and is kept isolated from the non-inert sensor stations to prevent oxygen and moisture contamination of the inert atmosphere volumes. The other SR sensing locations service the non-inert potential leak pathways of the facility. This includes the reactor cell, fuel storage enclosure, the coolant salt and heat management enclosure, and the research bay.

There may be a final NSR sampling location that will service the regular atmosphere volumes of the facility that are not potential leak pathways for the reactor and attached secondary, fuel, and gas systems. This includes the radiochemistry lab, the salt chemistry lab, and the control room. While this sensing location is not SR it will be utilized to inform when a building evacuation is necessary. The potential exists for this sensing station to be replaced with individual detectors, with the final design selection made based on the detailed engineering and design.

Appropriate emergency plans for detection of specific radiation, gas, and particulate species will be developed and described in the operating license application.

7.7.3 Emplacement Detectors

In addition to sensor stations fed via sensing lines, sensors will also be placed throughout the facility to monitor variables of interest that the sensor stations are insufficient to characterize. These will include area radiation monitors, monitoring radiation fields at specific locations, and supplemental radiation or chemical detectors for areas where sensing lines do not provide a rapid or detailed response. A provisional list of locations for these additional emplaced detectors is presented below. This list is not exhaustive. Detectors may be added or removed as detailed design continues.

- A gamma spectrometer, or gross count detector will be emplaced on the return line for the cooling salt from the primary heat exchanger. This will serve to detect intrusion of primary salt into the secondary salt by monitoring for the elevated signal primary salt would introduce relative to the normal detection of activated secondary salt.
- Area monitors in addition to the sensor station may be employed in the research bay to ensure the radiation field in the large main area of the facility is properly characterized. This will depend on final shielding design and analysis.
- Detectors may be emplaced or embedded to monitor dose to materials in the facility such as the sacrificial concrete of the reactor cell.
- Monitors may be placed at penetrations from the various enclosures/cells to monitor for leaks.
- Monitoring fuel salt storage and radioactive waste systems/areas for excessive radiation levels.
- Detectors may be placed near public viewing areas to ensure the dose to the public is well characterized.

7.8 References

7.8-1 United States Nuclear Regulatory Commission, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”, Part 1 “Format and Content” NUREG-1537, Washington, D.C., March 1996.

7.8-2 Oak Ridge National Laboratory, “Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application,” ORNL/TM-2020/1478, Oak Ridge, TN, July 2020.

Chapter 8
Electrical Power Systems

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 8 ELECTRICAL POWER SYSTEMS 8-1

8.1 Normal Electrical Power System 8-1

8.2 Emergency Electrical Power Systems 8-5

8.2.1 Backup Electrical Power Systems 8-5

LIST OF FIGURES

Figure 8.1-1 Substation and Line Location for Molten Salt Research Reactor Site 8-3
Figure 8.1-2 MSRR Electrical Configuration 8-4

CHAPTER 8 ELECTRICAL POWER SYSTEMS

The Abilene Christian University (ACU) molten salt research reactor (MSRR) will be installed and operated in the ACU Science and Engineering Research Center (SERC), which is an existing multipurpose research facility as described by Title 10 Code of Federal Regulations, Part 50 Section 10(a)(2)(x). Electric power is supplied to the SERC by American Electric Power (AEP) Texas. The electrical power systems include the non-Class 1E normal power system (Section 8.1) and the backup power system (Section 8.2). Because of the passive design of the MSRR, safety-related structures, systems, and components do not require emergency electrical power to perform safety-related functions following a design basis event. Therefore, alternating current (AC) power from off-site or backup power sources is not required to mitigate a design basis event. As such, no technical specifications related to the overall electrical system are required to assure and maintain safe shutdown.

8.1 Normal Electrical Power System

The electrical power provided by AEP is adequate and reliable for all MSRR operations. The substation is located at the intersection of San Jacinto and Goliad Drive in Abilene, Texas (Figure 8.1-1) and has adequate capacity to provide power for the proposed facility loads. During the past three years, AEP has experienced three outages that affected the substation, two (momentary) weather related and one fire in a distribution support structure. Figure 8.1-2 provides a schematic diagram showing the basic MSRR distribution systems and circuits.

Electrical power provided by AEP will supply all required services for the MSRR during startup, normal operation, shutdown, off-normal events, and decommissioning. The operating license application will propose technical specifications containing limiting conditions for operation which will require operability of systems needing electric power for defined reactor modes, such as operation. Normal electric power will be supplied to:

- Heaters for salt systems
- HVAC
- Control rod drives
- Instrumentation and control systems
- Gas Management System, including the reactor trip valves
- Air blowers
- Pumps (fuel and coolant)
- Engineered safety features
- Scientific Surveillance System
- Experimental systems
- Security systems
- Communication systems
- Other systems and loads as needed

The normal power system does not perform any safety related functions, is not credited for the mitigation of postulated events, and is not credited with performing safe shutdown functions. The design of the MSRR is fail safe with respect to electrical power, assuring safe reactor shutdown in case of a loss of electrical power. The MSRR is passively cooled and does not require electrical power to meet safety functions. In case of a loss of power, the fuel salt automatically drains by gravity into the reactor drain tank before the salt cools. Assurance of safe shutdown in the event of loss of electrical power and prevention of uncontrolled release of radioactive material is addressed in Chapter 13.

The following will be addressed in the Operating License application:

- Ranges and specifications for required electrical power
- Special processing of the electrical service
- Design and performance specifications of principal components (standard and non-standard)
- Special routing or isolation of wiring or circuits for operations and experimental facilities
- Deviations from national or local electrical standards
- Spectrum of reactor operations and associated electrical power requirements

The SERC including the MSRR facility electrical distribution systems are designed to the NEC (NFPA 70-2014). No safety related SSC requires electric power to perform its function. Thus, the electric power system is not safety related and electric isolation of safety related and non-safety related systems is not required. The electric power system will be designed with normal electrical protective equipment. A malfunction of the normal power system or backup power system will not prevent a reactor trip nor impair other safety functions.

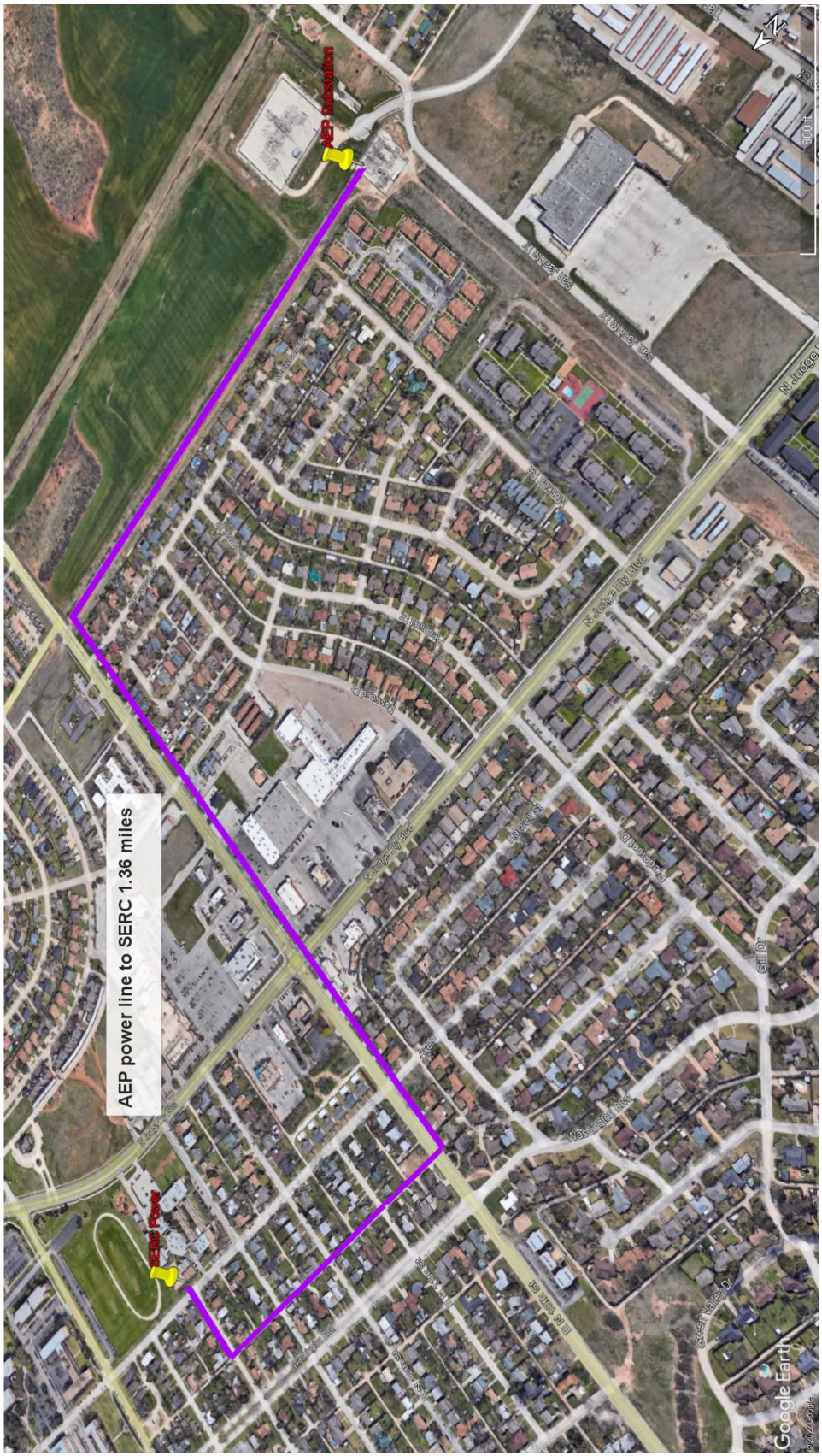


Figure 8.1-1 Substation and Line Location for Molten Salt Research Reactor Site

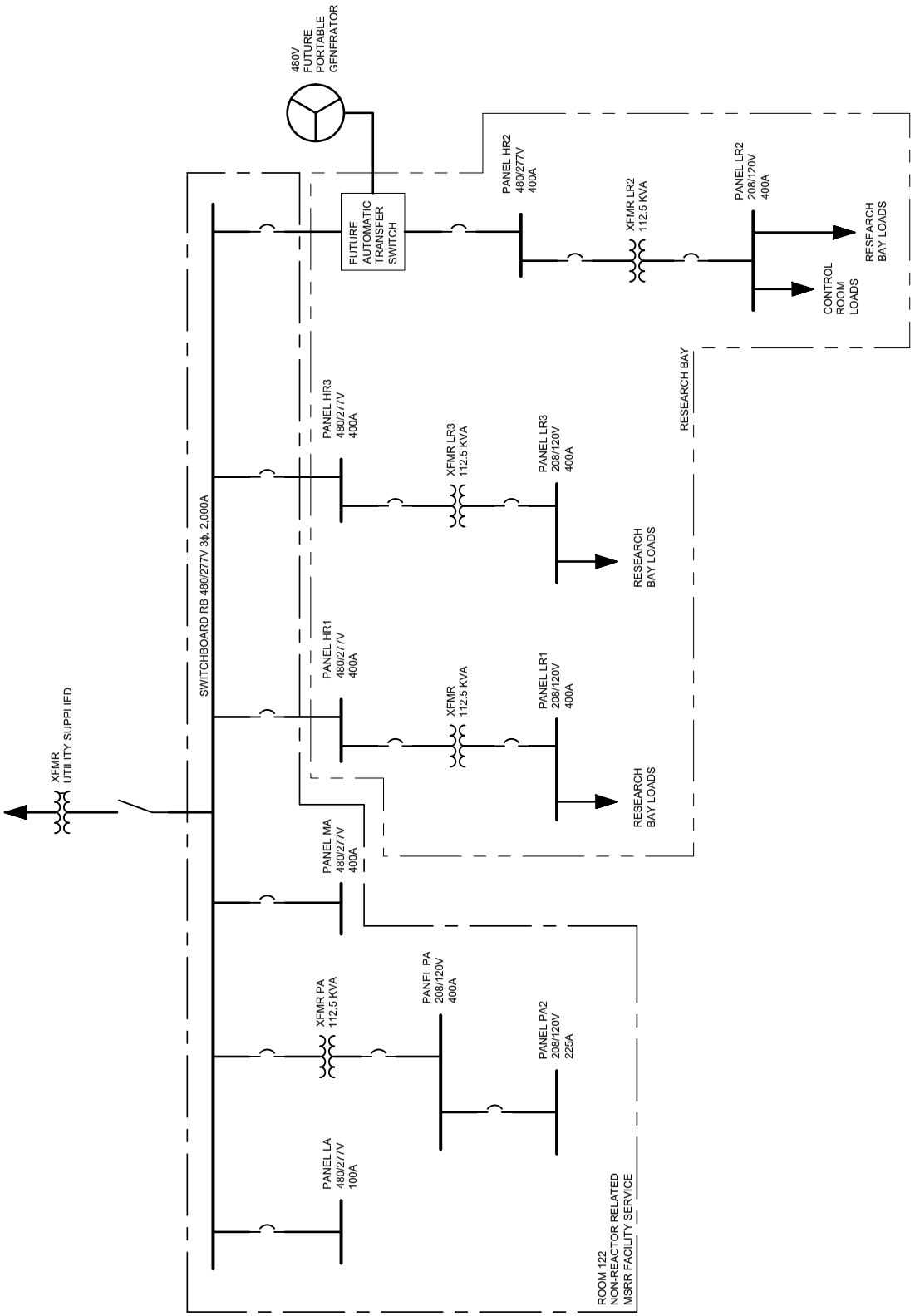


Figure 8.1-2 MSRR Electrical Configuration

8.2 Emergency Electrical Power Systems

Electrical power is not required to initiate a reactor shutdown, and passive cooling is adequate to prevent either fuel or primary containment failure (Chapter 13); therefore, no technical specifications are required for the electrical distribution system. Emergency lighting, the fire alarm system, and the security system will have uninterruptible power supplies with ratings to be provided in the Operating License application.

8.2.1 Backup Electrical Power Systems

The purpose of the backup power system is to provide AC electrical power to selected facility loads when the normal AC power supply is not available. The system is comprised of uninterruptible power supplies (UPS). Emergency lighting, the fire alarm system, and the security system will have uninterruptible power supplies (UPS) with ratings to be provided in the Operating License application. Other loads that are sensitive to power interruptions such as instrumentation and control will also have UPS. As a facility convenience, in the future the SERC will be equipped with a plug-in connection for use with a portable 480 VAC generator to provide power for non-operational activities. There are no plans to maintain a generator on site.

Selected loads are supplied with continuous AC electrical power via UPS, including instrumentation and control functions. Each UPS provides a highly-reliable power supply during normal operations and is automatically configured to provide backup power during a loss of normal electrical power event. The UPS are sized to provide sufficient power to those selected loads to maintain functionality for at least 24 hours after loss of power to the facility for operational convenience. A complete description of the UPS systems will be provided in the Operating License application.

No DC systems are included in the MSRR facility. The UPS systems do not perform any safety-related functions and are not credited for mitigation of postulated events. The systems are used for monitoring functions and are not credited with maintaining or performing safe shutdown functions.

Considering Design Criterion 17, Electric power systems, electric power systems are provided to permit functioning of structures, systems, and components. Safe shutdown and long-term decay heat removal are passive, and no electric power is required. The Chapter 13 analyses show that with complete loss of electrical power the design limits for fission product barriers are not exceeded. Thus, the bases for Design Criterion 17 are met with no electrical power required for safe shutdown, decay heat removal or accident mitigation.

Consistent with Design Criterion 18, Inspection and testing of electric power systems, electric power systems are designed to permit appropriate periodic inspection and testing. However, because loss of power shuts down the reactor and long-term decay heat removal is passive, inspection and testing of safety related components is not needed. Routine monitoring of operability and inspection of components will assure reliable MSRR operation and availability of electric power. Systems required by limiting conditions for operation to have electric power will have surveillance requirements in technical specification to confirm provision of power for required modes.

Chapter 9
Auxiliary Systems

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 9 AUXILIARY SYSTEMS 9-1

9.1 Heating, Ventilation, and Air Conditioning Systems 9-1

9.1.1 Design Bases 9-1

9.1.2 System Description 9-1

9.1.3 Operational Analysis and Safety Function 9-4

9.1.4 Instrumentation and Controls 9-5

9.1.5 Technical Specifications, Testing, and Inspection 9-5

9.2 Handling and Storage of Reactor Fuel 9-5

9.2.1 Design Basis 9-5

9.2.2 System Description 9-6

9.2.3 Operational Analyses and Safety Function 9-7

9.2.4 Instrumentation and Controls Requirements 9-9

9.3 Fire Protection Systems and Programs 9-9

9.3.1 Fire Protection Programs 9-9

9.3.2 Design Basis 9-9

9.3.3 System Description 9-10

9.3.4 Operational Analysis and Safety Function 9-12

9.3.5 Instrumentation and Control Requirements 9-12

9.3.6 Technical Specifications, Testing, and Inspection 9-12

9.4 Communication Systems 9-12

9.4.1 Design Basis 9-12

9.4.2 System Description 9-12

9.4.3 Operational Analyses and safety function 9-13

9.4.4 Instrumentation and Controls 9-13

9.4.5 Technical Specifications, Testing, and Inspection 9-13

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material 9-13

9.5.1 Special Nuclear Material 9-13

9.5.2 Byproduct Material 9-14

9.6 Gas Management System 9-15

9.6.1 Design Basis 9-15

9.6.2 System Description 9-16

9.6.3 Operational Analysis and Safety Function 9-18

TABLE OF CONTENTS

9.6.4 Instrumentation and Control Requirements 9-19
9.6.5 Technical Specifications, Testing, and Inspection 9-19
9.7 Cooling Systems 9-20
9.7.1 Auxiliary Heat Removal System. 9-20
9.7.2 Additional Cooling Systems 9-22
9.8 Other Auxiliary Systems 9-22
9.8.1 Compressed Air, Vacuum, and Inert Gas Supply 9-22
9.8.2 Cranes 9-23
9.9 References 9-24

LIST OF FIGURES

Figure 9.1-1 Heating, Ventilation, and Air Conditioning Systems Schematic Diagram 9-3
Figure 9.2-1 Deleted. 9-9
Figure 9.6-1 Conceptual Design of Gas Management System 9-20
Figure 9.7-1 Auxiliary Heat Removal System Conceptual Design 9-21
Figure 9.7-2 AHRS Conceptual Flow Path in Reactor Cell 9-22

CHAPTER 9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning Systems

9.1.1 Design Bases

The Molten Salt Research Reactor (MSRR) facility heating, ventilation, and air conditioning (HVAC) systems transfer heat from the combined heat load of the structures, systems, and components (SSCs) to the atmosphere under normal operating conditions, and maintain relative humidity and air temperature conditions in the facility within limits for personnel health and habitability. Under accident conditions the reactor cell is isolated, and fuel salt cooling is passive as described in Section 6.3. The MSRR facility HVAC system is not relied upon in any accident analyses in Chapter 13.

Consistent with DC 38, functional containment heat removal, an active auxiliary heat removal system will function to maintain facility conditions consistent with the initial conditions used in accident analyses. The MSRR HVAC systems provide conditioned air to support the auxiliary heat removal system. The HVAC systems and active auxiliary heat removal are not required for response to an accident and the systems are not safety related.

Consistent with DC 60, control of releases of radioactive materials to the environment, the MSRR facility HVAC is designed to

- ensure the research bay is maintained at a pressure below the public spaces, radiochemistry lab, dress-out room, and control room during normal operation.
- ensure the radiochemistry lab, dress-out room, and control room are at a lower pressure relative to the atmosphere and other public parts of the MSRR facility during normal operation.
- reduce the dose to restricted areas to ensure ALARA principles by exhausting activated air (see Chapter 4 and Chapter 11) during normal operation.
- monitor air exhaust for planned releases of effluent radioactivity.
- isolate on detection of high levels of radioactivity to support good ALARA principles. Isolation of the HVAC is not required to meet Chapter 13 accident analyses.

Design basis values will be specified in the Operating License application.

9.1.2 System Description

The HVAC systems for the radiation-controlled area housing the MSRR and the associated control rooms and labs provides the following functions:

- Humidity, pressure, and temperature control for the habitable spaces
- Conditioned clean air flowing from regions of lower radiation hazard to areas of higher hazard

- Conditioned air for the auxiliary heat removal system (AHRS) during normal operation
- Exhaust air for controlled effluent gases
- Contamination control in the event of a radiation release

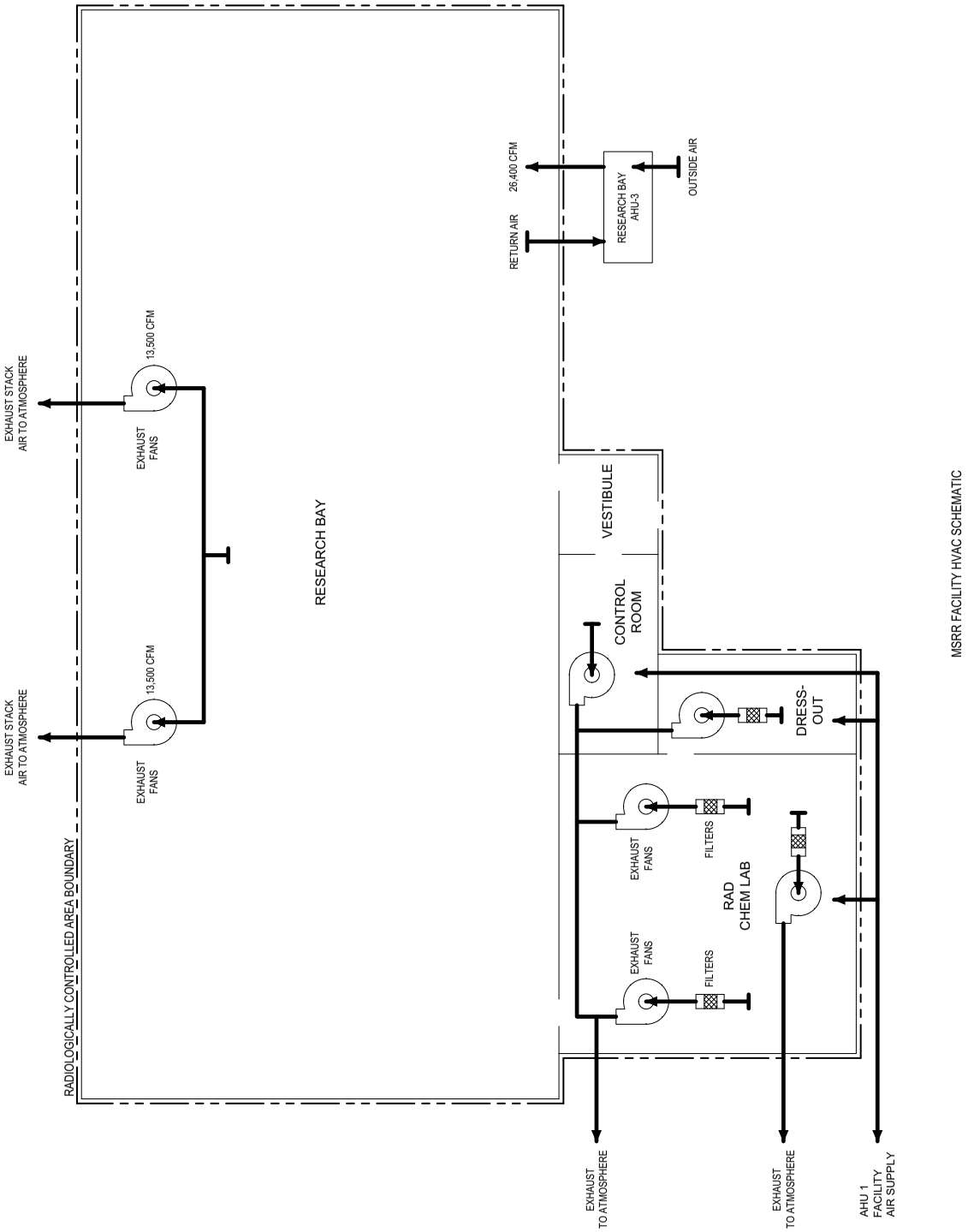
Two HVAC systems service the radiation-controlled area, a dedicated system for the research bay and a separate system for the control room and labs. The systems use air handling units (AHUs) to move air across air-to-water heat exchangers in which coils are supplied with hot or cold water. The systems share two on-site boilers to supply hot water with capacity and temperature setpoints determined by the facility HVAC requirements, and two on-site water chillers that have temperature setpoints determined by the facility cooling requirements. Local air handling controls provide an air pressure differential so pressure in the MSRR facility is lower than the atmosphere, and pressure in the research bay is lower than the rest of the building. These pressure settings are controlled independently in the control room.

There are three operational modes of the research bay HVAC system:

- Recirculation mode - when the reactor is shut down, air handling optimizes energy savings by returning 80 percent of inside air to the AHU to be mixed with 20 percent outside air.
- Once-through mode - when the reactor is in operation, the system uses 100 percent outside air supply and exhausts 27,000 cfm to generate an air exchange of 10 air changes per hour (298,000 ft^3).
- Isolation mode - when an airborne radioactive release is detected, the system shuts down air exchange and maintains an isolated condition.

In Figure 9.1-1, the research bay HVAC system layout is bounded by dotted lines (radiation-controlled area boundary). Dimensions, and air flows are considered typical or nominal.

Figure 9.1-1 Heating, Ventilation, and Air Conditioning Systems Schematic Diagram



The dedicated research bay AHU contains both hot- and cold-water coils in a single-duct system. The unit is built such that when the reactor is shutdown, 80 percent of the air supplied to the bay may be recycled through the system, but when the reactor is in operation, up to 100 percent of the air comes from outside, is monitored, and then is exhausted through a stack system. This helps maintain radiation levels ALARA by continuously diluting and exhausting gases that may have been activated.

The AHU for the control room/lab HVAC system has a single supply, but the return air and exhaust is different than for the research bay HVAC system. Air that enters the radiochemistry lab and dress-out room is filtered and exhausted through plume fans; none is recirculated. The supply and exhaust air to these rooms is isolated with control valves that can be shut by the operator in case of a radioactive release. The speed of the exhaust fans is designed to maintain pressure in these rooms lower than the outside, but higher than the research bay. This ensures clean air flows from outside into public spaces that are not in the radiation-controlled area, then into the radiochemistry lab, the dress-out room, and the control room, and finally, into the research bay. This will minimize the spread of any release of radioactive material in the facility.

During normal operation, both the control room/lab and research bay HVAC systems draw air from either a combination of return air fans and air from outside the building or exclusively from outside air. The air is prefiltered with standard HVAC filters, runs through temperature and humidity control units, then through a final standard HVAC filter and into the room.

The AHUs for the radiochemistry lab, dress-out room, and control room share the same supply, but the exhaust from the radiochemistry lab and dress-out room are not recirculated while the control room air is routed to the radiochemistry lab exhaust fan.

The HVAC design will be finalized and described in the Operating License application.

9.1.3 Operational Analysis and Safety Function

During normal operation, the HVAC systems are designed to control the exposure of operational personnel and the public to radioactive materials. The release criterion is based on 10 CFR Part 20. The HVAC systems are not credited for mitigation of accident events, nor are they required for safe shutdown functions. The HVAC does not play a role in the isolation of the reactor cell. Isolation of the reactor cell, which is part of the ESF and necessary for postulated accidents, is a function of the AHRS and described in section 9.7 and 6.2. Releases during the MHA are assumed to immediately enter the environment. This assumption bounds HVAC performance or lack thereof.

Systems, structures, and components required for safe shutdown and long-term decay heat removal in the systems pit are protected from damage from HVAC system failure during seismic activity by physical separation, barriers, or seismically mounting components.

The HVAC systems are not necessary to control radionuclide release during accidents. Operator action is not required to mitigate a postulated accident as described in chapter 13. If the control room may become uninhabitable in the near future, operators will trip the reactor and egress to a safer location. Operators may be required to leave the control room in the event of a fire, HF release, tornado, or a large radionuclide release. For this reason, the HVAC function to the control room is not considered safety related.

The radiation monitoring system provides monitoring of discharge gases to the atmosphere during normal operations.

The HVAC systems may contain radiological contaminants and are designed to minimize contamination as required in 10 CFR Part 20 and described in Chapter 11.

The HVAC system does not supply air to the fuel salt storage enclosure.

The HVAC system maintains a pressure gradient between the research bay, control room, and laboratories to ensure that any radiological release flows out the monitored exhaust stack.

9.1.4 Instrumentation and Controls

To minimize spread of contamination during potential accident conditions, the instrumentation and control system activates dampers to isolate the research bay, radiochemistry lab, control room, and dress-out room from the rest of the building when high radioactivity levels in the research bay are detected. The instrument and control functions are described in Chapter 7 and will be provided in more detail in the Operating License application.

9.1.5 Technical Specifications, Testing, and Inspection

The HVAC system is tested periodically and will be detailed in the Operating License application. Any proposed technical specifications (TS) will be included in the Operating License application.

9.2 Handling and Storage of Reactor Fuel

The fuel handling system (FHS) is designed to ensure fuel is enclosed in a manner such that radionuclides are functionally contained during handling and manipulation. Fuel salt is maintained in geometries and in proximity to materials that prevent criticality in all conditions during fuel storage and movement in the facility. Additionally, the FHS is designed to ensure appropriate radiation shielding when removing irradiated fuel.

9.2.1 Design Basis

Consistent with DC 61, fuel storage and handling and radioactivity control, the FHS is designed to assure adequate safety under normal and postulated accident conditions. Structural and mechanical integrity is assured as part of element (3) of DC 61, which requires the containment of radioactivity. It has the capability for appropriate periodic inspection and testing of safety related components, with suitable shielding for

radiation protection with passive residual heat removal. Containment is provided by the fuel salt, the safety related vessels in the system, and the safety related fuel storage enclosure.

Consistent with DC 62, prevention of criticality in fuel storage and handling, criticality in the fuel storage and handling system is prevented by geometry.

Consistent with DC 63, monitoring fuel and waste storage, the FHS temperature is controlled by passive cooling. Radiation is monitored to detect signs of leakage from the FHS.

Consistent with DC 70, fuel salt purity control, the FHS is designed to support the required chemistry control described in Section 4.2.

Consistent with DC 71, fuel salt composition control, the FHS is designed to maintain the composition of the fuel salt within specified limits.

9.2.2 System Description

The FHS for the MSRR is safety related and equipped to enable manipulation and transfer of fuel salt from reception into the facility through the reactor system and ultimate disposal. The FHS is equipped to enable final salt purification, fuel chemistry modifications, and other salt manipulations over the course of MSRR operation. In addition, the FHS is capable of handling used salt. Proprietary design conditions and equipment arrangement are transmitted by separate letter (ML2409A332, dated March 28, 2024). The FHS is made up of the fuel salt purification and storage tank, the fuel storage enclosure, the RAV, the drain tank, and the transport vessels.

The fuel salt purification vessel will be constructed from UNS N02201 (Ni-201) low-carbon nickel. The vessel will be designed in accordance with the 2021 Edition of the ASME Code Section VIII, Division 1. At the temperatures required for fuel salt purification (roughly 625 °C), pure nickel is resistant to corrosive attack by hydrogen fluoride and hydrofluoric acid. The fuel salt purification vessel will experience load cycles and time at temperature that are a small fraction of the allowable values in Section VIII, Division 1. The associated piping system, including its supports, will be designed to B31.3-2020, which is supplemented by Section III, Division 5, 2017 Edition at high temperatures.

Facilities and equipment involved with the FHS are equipped to handle beryllium. This requires purpose-built containment, ventilation, handling procedures, and personnel training with regular beryllium health screening.

The FHS is compatible with the shipping containers designed to receive fuel salt from the supplier. Received salt is inserted into the FHS. The fuel salt will be sparged with HF, H2, and He gases to remove impurities. The effluent gases are routed out of the FHS to a HF neutralization system (denoted "HF absorbent canister") and then released to atmosphere through the exhaust. Proprietary design conditions and equipment arrangement are transmitted by separate letter (ML2409A332, dated March 28, 2024). Once the salt is confirmed to be the desired composition and purity, it is transferred into the reactor system. All transfers are completed using pressurized

inert gas, and the salt travels through heated transfer lines between vessels. Salt freeze plugs and pressure controls are being considered to minimize the use of mechanical valves.

Decay heat removal is entirely passive. Fuel salt stored in the FHS will be suitably decayed to ensure that passive cooling is sufficient. The required decay time in the reactor drain tank before transfer will be governed by power history.

The FHS will be designed and constructed to accommodate inspection of welds and other select areas. The FHS design will be finalized and described in the Operating License application.

9.2.3 Operational Analyses and Safety Function

There are five key components of the FHS as described below.

Salt Reception/Insertion

A station on the facility service floor above the reactor in the research bay is equipped to receive the fuel salt shipping container and to insert the fuel into the fuel salt purification and storage tank. This may be a heated station for liquid transfer or an ambient station for powder transfer. Methods of salt inspection and verification will be reported in the Operating License application.

Fuel Storage Enclosure

Fuel salt purification and storage operations occur inside the fuel storage enclosure, a safety related, leak tight, pressure and fission product boundary. Heated transfer lines that are enclosed by a safety related pressure boundary provide salt transfer pathways between the fuel storage enclosure and the drain tank within the reactor system. Design of the enclosure takes into account pressure and temperature of fuel purification and storage during normal and accident conditions and will be specified completely in the Operating License application.

The design basis accident for the fuel storage enclosure is the rupture of the fuel purification and storage tank, resulting in the molten fuel salt and gas spilling into the enclosure. During purification operations, nitrogen gas is used to inert the Fuel Storage Enclosure, which is maintained at negative pressure. The fuel storage enclosure has a design leak rate. Leak requirements will be less stringent than those for the reactor enclosure due to the lesser radionuclide inventory attributed to decay time and the separation of gaseous fission products into the helium during operation. Design pressures, temperatures, and leak rates for the fuel storage enclosure will be determined to ensure that the radiological consequence of the aforementioned accident is bound by the MHA. Proprietary design conditions for this enclosure are transmitted by separate letter (ML2409A332, dated March 28, 2024).

All gas lines are isolated in the event of a fuel salt leak into the fuel storage enclosure. The fuel storage enclosure has a pressure relief valve in case the HF, H2 and He lines fail to isolate which vents to a water trap to collect fission products and HF.

Reactor Drain Tank

The reactor drain tank is inside the RTMS (Section 6.2.4), and salt in this tank is highly subcritical at all times. This tank serves as the primary nexus for fuel salt entry into, holding in, and exit from the reactor system. Fuel salt transfers between the reactor drain tank and reactor loop as needed to enable operation and shut down.

Reactor Access Vessel

The reactor access vessel (RAV) is a component within the reactor loop through which fuel salt passes. The RAV enables minor online fuel additions and micro-sampling of the fuel salt and off-gas as part of reactor vessel surveillance measurements.

Fuel Salt Purification and Storage Tank

Fuel salt purification occurs in the fuel salt purification and storage tank. HF, H2, and He gases are bubbled in the molten fuel salt to remove contaminants. This procedure will be performed during pre-operational testing and may be conducted during operation to correct chemistry imbalances in the salt. The system is designed to hold the entire fuel salt inventory. Fuel will be stored in this vessel during decommissioning and during maintenance periods. The vessel is constructed of a nickel alloy that is resistant to HF exposure. Transfer of fuel salt to the drain tank occurs through heated transfer lines. A fuel salt wetted valve is implemented on this transfer line. Fuel salt is transferred to the transport vessels as a liquid and then frozen for removal from the facility.

Without an effective moderator, k_{eff} in the fuel salt remains deeply subcritical under all conditions during fuel storage and movement in the facility. The configuration with the highest k_{eff} outside the core considers the fuel salt is frozen in a tank, the UF4 has separated from the FLiBe and collected at the bottom, and the tank is submerged in water. In this case k_{eff} is approximately 0.6.

The fuel salt purification and storage tank is constructed of Ni-201. Proprietary service conditions and ASME code assignments are transmitted by separate letter (ML2409A332, dated March 28, 2024).

The MSRR design will not have Ni-201 to SS316H welds. The fuel salt purification vessel will be constructed entirely from Ni-201 including all nozzles and welded support structures. Vessel supports will be welded to the Ni-201 vessels but will be bolted to the SS316H enclosure with an electrically insulating material between the two metals.

All weld procedures will be qualified in accordance with ASME Code Section IX. The Ni-201 vessels will be constructed using a TIG welding process using SFA 5.9 ERNi-1 weld rod per Section II, Part C.

Ni-201 piping or valves will be connected to SS316H by flanged connections that have isolating flange kits to electrically separate the two metals. Details of the flange and isolating material will be provided in the Operating License application.

Technical specifications guiding the FHS will be provided in the OL. These specifications will address the pressures and temperatures of lines and vessels comprising the FHS, rate of temperature change when heating up, number of thermal cycles, number of sparging cycles, total time at elevated temperatures, pressure and leak rate of the FHS enclosure, radiation monitoring, pressures and mass flow rates of the gases flowing through the vessels, maintenance procedures, and inspection/testing procedures. Any technical specifications needed for the system will be provided in the Operating License application.

Figure 9.2-1 Deleted

9.2.4 Instrumentation and Controls Requirements

Instrumentation and controls requirements will be defined in Chapter 7 in the Operating License application.

The FHS is designed for appropriate testing and inspection. Details will be provided in the Operating License application. Any necessary technical specifications will be provided in the Operating License application.

9.3 Fire Protection Systems and Programs

9.3.1 Fire Protection Programs

A complete description of the fire protection program, how it integrates with the emergency plan, the effects SSCs may have on safety or licensed materials protection, and the fire hazard analysis will be provided in the Operating License application.

9.3.2 Design Basis

The objectives of the fire protection systems are to ensure safety-related systems are able to perform as expected and to prevent injury, loss of life, and minimize property damage in the case of fire. A set of fire protection systems is employed to ensure fire

detection, distribution of alerts to response organizations and local personnel, and fire suppression. Basic design features of the reactor assembly, shielding, and instrumentation and controls provide sufficient passive protection to ensure operation of safety-related systems during design-basis events.

Consistent with DC 3, fire protection (Section 3.1.2.1), the MSRR facility fire protection system is of appropriate capacity and capability to minimize the adverse effects of fires on safety-related SSCs. The fire protection system is designed to ensure rupture or inadvertent operation does not significantly impair the safety capability of SSCs. The fire protection design basis functions are to

- provide detection and suppression of fires,
- meet 2009 Abilene, Texas, fire code requirements.
- generate alarm signals indicating presence and location of fire.
- execute commands appropriate for the particular location of the fire.
- provide fire detection in the facility and initiate fire-rated damper closures.
- provide constant flow of water to an area experiencing a fire for a minimum of 30 minutes or 60 minutes based on the hazard classification of the area.

9.3.3 System Description

Automatic fire protection systems include a wet-pipe sprinkler system with heat-responsive elements installed throughout the building that provide coverage for fire suppression to the building, and an automatic fire alarm and communication system to alert appropriate response organizations. The research bay is separated from the remainder of the MSRR facility and Science and Engineering Research Center by a two-hour fire barrier.

The sprinkler system is installed throughout the building and complies with the following regulatory codes in addition to the applicable insurance authorities underwriting requirements:

- International Building Code (2012) as adopted and amended by the City of Abilene, Texas [Reference 9.9-1]
- International Fire Code (2009) as adopted and amended by the City of Abilene, Texas [Reference 9.9-2]
- NFPA 13 (2018) [Reference 9.9-3]
- NFPA 45 (2019) [Reference 9.9-4]
- NFPA 72 (2022) [Reference 9.9-5]

The building is split into three hazard classifications according to NFPA 13.

- Light Hazard area: These areas have automatic sprinklers that supply a minimum of 0.1 gpm over 1500 sq ft and include corridors, vestibules, antechambers, restrooms, offices, conference rooms, and control rooms.

- Ordinary-Hazard, Group 1 Occupancy area: These areas have automatic sprinklers that supply a minimum of 0.15 gpm over 1500 sq ft. These areas include laboratories, equipment rooms, electrical rooms, and mechanical rooms.
- Ordinary-Hazard Group 2 area: The research bay is in this category and has a sprinkler system capable of providing 0.2 gpm over 1500 sq ft. The sprinkler system has automated alarm switches used to notify the fire department when a flow of water in the sprinkler system is detected.

An automatic local fire alarm system that complies with NFPA 72 is installed and includes connections to report alarm conditions to off-site response organizations. Local sensors provide automatic alarm coverage throughout the building. These sensors include photoelectric smoke detectors that measure smoke density and communicate to the control panel, and 57 degree Celsius thermal detectors to initiate alarm conditions. Audible and visible alarm signal devices installed in accordance with NFPA 72 throughout the building are automatically activated by the alarm system in case of a sensor activation. Additionally, normally-open fire doors are automatically closed by the alarm system in case of a fire.

Water from the sprinkler system that falls in the systems pit can pool there and require cleanup. There is no water drainage from the pit, so the runoff is contained by the concrete walls and floor and can be sampled prior to removing. Flooding in the systems pit does not impact the ability to achieve safe shutdown nor does it impede long-term decay heat removal.

Work in the radiochemistry lab falls under the ACU chemical hygiene plan, which requires all chemicals, including waste chemicals, to be stored in sealed containers. Any hazardous, volatile chemicals are used in the hood or the glove box. All radioactive chemicals are stored in a radiation cabinet when not in use. Radioactive sample containers are opened in sealed glove boxes, reducing the likelihood of sprinklers activating while a sample is being handled. The likelihood of a sprinkler being activated during the very brief time that a radioactive chemical is outside a glove box or a radiation cabinet is less than 0.001 percent of the time (86.4 seconds per day). Even in that case, the sample container is sealed until it is inside the glove box, ensuring that runoff does not carry radioactive materials into the environment.

Manual protection systems and plans include manual alarm activation, fire extinguishers, hose connections, evacuation procedures, and administrative controls. Equipment to support these manual systems and plans include manual fire alarm pull boxes and dry chemical portable fire extinguishers throughout the building, as well as the emergency communications system. A hose connection for the fire department is supplied outside the research bay to ensure complete coverage of the building.

Administrative controls are put into place to ensure flammable materials are kept at a distance from hot surfaces or areas that might have ignition sources. These controls include demarcation of hot surfaces for both health safety and fire control purposes.

Passive fire protection includes architectural features of the building, such as precast concrete walls, metal beams and doors, and other materials such as fire-rated building materials. The exterior building materials are precast concrete panels, brick,

and metal paneling. The roof consists of a metal deck with insulation and coverboard and a thermoplastic polyolefin membrane that meets ASTM E108 - 20a (2020) [Reference 9.9-6] or UL 790 (2022) [Reference 9.9-7] Class A fire rating standards.

9.3.4 Operational Analysis and Safety Function

Abilene Christian University maintains active fire protection systems and performs regular testing and inspections to ensure timely and appropriate responses.

Water-based fire suppression systems are used throughout the MSRR facility. In case of activation, a drainage system has been built around the systems pit where the reactor is located that will carry off most of the water. If a catastrophic event occurred in which the systems pit filled with water, fuel salt is stored such that there is no potential for a criticality event, there is no impact on the ability to achieve safe shutdown, and decay heat removal is not impeded.

The fuel handling system and the reactor system are located in water-tight, stainless steel enclosure vessels. Both during normal operation and during a loss of power, the maximum temperature of the enclosure vessels is less than 100 degrees Celsius, thus eliminating concerns with steam generation and rapid pressurization.

9.3.5 Instrumentation and Control Requirements

A fire alarm panel transmits alarm status information to the ACU Police and Abilene Fire Departments. Additional details on fire protection instrumentation and controls will be provided as part of the Operating License application.

9.3.6 Technical Specifications, Testing, and Inspection

The fire protection system is designed to allow for appropriate testing and inspections. Details will be provided in the Operating License application. Any proposed TS will be provided in the Operating License application.

9.4 Communication Systems

9.4.1 Design Basis

Communication systems provide support during normal and emergency operations. They are not credited for mitigation of design-basis events, and have no safe shutdown functions.

9.4.2 System Description

The telephone system is installed and maintained by ACU. At a minimum, hard line connected phones will be located in the control room, health physics office, and the research bay. The system is connected by standard telecommunications equipment to the commercial telephone network. An installed public address system allows reactor operators to communicate with the other rooms in the SERC. A cell phone in the control room allows for communication between individuals in the facility and the control room. The control room cell phone remains within the control room at all times

and is intended to be used in case the control room hard line phone is unavailable. Operators may contact personal cell phones of workers. A digital radio is kept in the control room for emergency communication with first responders in case other outside communications are unavailable.

9.4.3 Operational Analyses and safety function

Communication systems provide support during normal and emergency operations. They are not credited for mitigation of design-basis events, and have no safe shutdown functions.

9.4.4 Instrumentation and Controls

No specific instrumentation and controls are provided as part of the communication system.

9.4.5 Technical Specifications, Testing, and Inspection

Telephones and public address are in normal use. No specific testing and inspection are necessary. Radio systems are periodically tested. No TS on communications systems are anticipated.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

Minimization of contamination by special nuclear material and uncontrolled release prevention are accomplished by administrative controls that will be described in the Operating License application.

9.5.1 Special Nuclear Material

Special nuclear material is received and used in the form of dry fluoride salt delivered to the facility. Fuel is not expected to be manufactured on site. The UF_4 salt will contain high-assay, low-enriched uranium (less than 20 percent enrichment) and may be mixed with LiF-BeF_2. The exact form of this material will be specified in the Operating License application.

Special nuclear material is handled in the FHS, the reactor cell, the gas management system (GMS), the fuel salt sample system, and the radiochemistry lab. Special nuclear material is used and maintained in accordance with the facility license and regulations. Uncontrolled releases are prevented by the following features:

- Fuel salt and reactor system as discussed in Chapter 4
- Fuel handling system as discussed in Section 9.2
- Confinement, containment, and auxiliary heat removal system as discussed in Chapter 6
- Experimental facilities as discussed in Chapter 10
- Heating, ventilation, and air conditioning operations as discussed in Section 9.1

The possibility of coming into contact with special nuclear material is limited to the salt receiving station and the radiochemistry laboratory. Activity of fresh salt is very low, and administrative controls are put into place to minimize contact with special nuclear material in both locations. Fuel samples are shielded and controlled to ensure ALARA dose levels and compliance with facility license and regulations.

Besides the fuel salt and byproduct material, small quantities of SNM may be present within the facility. ACU may utilize a startup source such as PuBe, AmBe, or Cf-252. ACU may utilize fission chambers with small quantities of U-235 for neutron detection. Natural uranium, not acquired through the DOE, may be present within the radiochemistry lab to serve as a surrogate for the fuel in select experiments.

9.5.2 Byproduct Material

Byproduct materials are generated during reactor operation as a result of fission reactions and are mixed in the fuel salt. Byproduct material is present in the reactor system (including the reactor vessel, RAV, drain tank, fuel side of the heat exchanger, reactor pump, and piping), portions of the GMS connected to the reactor system, fuel handling system (fuel salt purification and storage tank and connecting pipes), fuel salt sample system, and Radiochemistry Laboratory (described in Section 9.5.2.1). The material is accessed in the fuel salt sample system and in the radiochemistry lab for purposes of monitoring and maintenance of corrosion control and research purposes.

Some byproduct material can take the form of gases that are collected for storage, monitoring, or controlled release, not to exceed limits set forth by regulation. The GMS described in Chapter 4 and Section 9.6 monitors and holds up gaseous byproduct materials until release at concentrations below limits set by 10 CFR Part 20. The administrative controls and physical shielding that maintain radiation doses in compliance with 10 CFR Part 20 and ALARA principles will be fully described in the Operating License application.

9.5.2.1 Radiochemistry Laboratory

Preliminary Safety Analysis Report (PSAR) Chapter 10, “Experimental Facilities and Utilization,” describes the experimental facilities and the movement of small fuel salt samples from the reactor. It is also planned to irradiate small material samples within the reactor access vessel and transport these samples to the Radiochemistry Laboratory for analysis. These samples will be analyzed or packaged and shipped to off-site collaborators. The location of the Radiochemistry Laboratory in relation to the research bay is shown in Figure 2.1-9 (Chapter 2).

The Radiochemistry Laboratory will be part of the reactor facility license. The reactor license allows the use of byproducts and special nuclear material initially licensed and produced by operation of the reactor. Experiments conducted in the Radiochemistry Laboratory may involve the processing of irradiated materials containing special nuclear material. The Radiochemistry Laboratory will not be a

"production facility" as defined in 10 CFR 50.2 because activities will meet the (3)(i) exception in the definition for laboratory scale facilities designed or used for experimental or analytical purposes.

The Radiochemistry Laboratory will primarily consist of analytical equipment to conduct experiments and measurements on samples removed from the reactor, such as tensile testers, scanning electron microscopes (SEM), focused ion beam (FIB) systems, inductively coupled plasma (ICP) systems, Raman spectroscopy systems, and laser-induced breakdown spectroscopy (LIBS). The Radiochemistry Laboratory will also consist of typical university laboratory equipment such as benches, sinks and cabinets. A dress-out room will be used for donning and doffing of appropriate personal protective clothing.

Activities in the Radiochemistry Laboratory will be under the reactor facility radiation protection program discussed in Chapter 11, "Radiation Protection Program and Waste Management," of this PSAR, including requirements for radioactive waste. Experiments shall be conducted within the Limiting Conditions for Operation and Surveillance requirements of the technical specifications. Experiments will be reviewed and approved in accordance with the technical specification requirements for management and MSRR Review and Audit Committee review and/or approval of experiments.

Additional information on the experimental program, transfer of samples from the reactor to the Radiochemistry Laboratory, use of the Radiochemistry Laboratory, and the design details of the Radiochemistry Laboratory will be discussed in the Operating License application. The Radiochemistry Laboratory and fume hood within the Radiochemistry Laboratory will have ventilation that is part of the heating, ventilation and air conditioning (HVAC) system discussed earlier in this PSAR chapter. The Radiochemistry Laboratory will contain appropriate radiation protection monitors and equipment for the research program to be carried out. Detailed procedures for use of the Radiochemistry Laboratory will be part of the safety analysis accompanying requests for experiment approval.

9.6 Gas Management System

The preliminary design of the GMS is described in this section and will be fully described in the Operating License application.

9.6.1 Design Basis

The GMS has both safety related and non-safety related components. The non-safety related components are in the off-gas subsystem and are described in the following section. The GMS fulfills the following safety functions.

- Retain radionuclides such that a postulated loss of barrier integrity does not result in doses to the public greater than the maximum hypothetical accident
- Dissipate decay heat from retained radionuclides in the event of a complete loss of off-site power
- Support reactor protection system functions

The GMS is designed to

- remove decay heat generated in the GMS to ensure the fuel salt boundary will not be damaged and reactor system boundary integrity will remain intact.
- ensure the control and detection of leaks so an uncontrolled release of radioactive material does not occur.
- monitor, store, and purify inert gases (helium).
- move fuel salt pneumatically throughout the facility.
- support the reactor protection system and provide for reactor shutdown.

Consistent with DC 2, design basis protection against natural phenomenon, the GMS is protected against the effects of natural phenomena and can safely shutdown the reactor in these conditions.

Consistent with DC 4, environmental and dynamic effects design basis, the GMS is designed to accommodate the effects of and to be compatible with the expected environmental conditions during operation, maintenance, testing, and postulated accidents.

Consistent with DC 61, fuel storage and handling and radioactivity control, the GMS is designed to permit appropriate periodic inspection and functional testing of safety related components to ensure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of system components, and (3) the operability of the systems during normal operating conditions.

Consistent with DC 73, reactor trip line plugging, the GMS includes provisions to monitor gas lines whose plugging could prevent the GMS safety-related function of pressure equalization, thus leading to reactor shutdown. Technical specifications for monitoring will be established.

Consistent with DC 74, fuel salt system interfaces, there are no fuel salt boundary interfaces with an SSC containing fluid that if allowed to freely interact with the fuel salt would cause the loss of a safety function.

9.6.2 System Description

The GMS handles those gases which directly interface with salts across multiple subsystems. Helium is the primary gas, although nitrogen, hydrogen, and anhydrous hydrofluoric acid will also be used. Gases will contain trace quantities of radioactive species. The GMS touches every salt-bearing component throughout the reactor. Gas pressure and flow rate through each component are essential to salt control. Pressure differentials are used to pneumatically transfer and relocate the salts to fill (prime) the systems and to move the salts between handling stations (e.g., sampling stations or fuel storage). The GMS has a robust, reliable, and fail-safe system for the measurement and control of pressure. The GMS is designed to withstand surges in volumes and pressures. Pressure relief functions will actuate if a pressure surge is detected.

The GMS is tied to the reactor protection system as described in Section 4.2.2 and Chapter 7. Equalization of GMS pressure between the RAV and the reactor drain tank causes the fuel salt to drain out of the reactor system into the drain tank, shutting down the reactor.

Reactor system functions are dependent on the GMS. That portion of the GMS located inside the reactor enclosure is denoted as the reactor GMS. The Reactor GMS interfaces with the reactor drain tank, the RAV, and the reactor pump. A medium-pressure and a low-pressure helium storage vessel are each located inside the reactor enclosure. The medium-pressure tank is supplied by the high-pressure helium tank. The low-pressure helium tank is maintained at low pressure by a compressor or by venting to the low-pressure off-gas system, which is a subsystem of the GMS. A helium line with mass flow rate control and pressure control connects each component to the low-pressure tank while a helium line with pressure control connects each vessel to the medium-pressure tank. A compressor is used to move helium between the low-pressure and medium-pressure tanks. Helium is moved from the medium-pressure helium tank through components to the low-pressure tank. A portion of helium passing through the RAV head space can be routed to the off-gas system. A conceptual design of the GMS is provided in Figure 9.6-1.

Because of the intimate contact between the gas and the radionuclide-bearing salts, the inert gases contain gaseous radionuclides. Storage and cleaning of the contaminated gas to remove non-helium components occur within the off-gas system, a subsystem of the GMS. The functions performed by the off-gas subsystem are not safety related. The off-gas subsystem will not contain molten salt and service requirements for these components are mild. The off-gas system will be fully enclosed within the Off-gas Cleanup Enclosure, which is safety related. Effluent releases from the off-gas system are in compliance with 10 CFR Part 20 Appendix B. Radionuclides separated from the gas and deposited in or on solid media generate decay heat, which is appropriately managed. Various components of the GMS are located within the reactor enclosure and the reactor cell, depending on the hazard level of the individual component. The GMS piping penetrating the engineered safety features are isolatable to ensure functional containment requirements.

The off-gas system cleans the radioactive gases from the reactor GMS using a holdup container and two charcoal beds, along with necessary piping and valves. Radioactive gases are released to the atmosphere through the bay exhaust; cleaned helium will be reused. Two air-cooled charcoal beds hold up the fission gases until they either decay away or decay into solids, which remain in the charcoal. The charcoal beds that absorb non-helium gas constituents in the off-gas system are arranged in parallel. The beds are designed to be used alternately so that one bed can be regenerated while the other is in use. The beds consist of sections of varying diameter of pipe stuffed with charcoal, connected in series. The dimensions of these piping systems will be reported in the Operating License application. Nitrogen is used to flush the bed. This nitrogen and any remaining fission products exit out the stack. A compressor (scroll pump) is used to bring both contaminated helium exiting the holdup volumes and clean helium exiting the scrubber beds back to the reactor. The pump has a second layer of containment because there are fission products in the

contaminated helium circulating through. There is a sampling port in the headspace of the reactor after the holdup volume and after the scrubber beds. Off-gas content is monitored by samples taken at these locations.

The GMS is constructed of stainless steel 316H and is designed to the same standards as the reactor system, as described in Section 4.3.

Details of GMS design and operations will be provided in the Operating License application.

9.6.3 Operational Analysis and Safety Function

Radionuclide content of the helium within the reactor system, in the coolant loop, and in the FHS are dissimilar. Cross contamination of helium is minimized by design and operations. The reactor GMS is located inside the reactor enclosure and the reactor cell, as described in Chapter 6, ensuring all gaseous fission products are doubly contained. Actinides are not gaseous; therefore, there is no possibility of inadvertent criticality in the GMS. Radionuclide content of the GMS is monitored. Release of fission gases does not exceed 10 CFR Part 20, as described in Chapter 11. The GMS is scrubbed of hazardous materials, such as fluorine or hydrogen fluoride, before release. Concentrations of hazardous chemicals and fission products are monitored to detect buildup, clogging, and leaks.

The off-gas subsystem uses a hold up volume and two large scrubber beds parallel to each other. One bed always is in use while the other one is regenerated with a flow of inert gas. The beds are switched between regeneration and use according to timelines to be provided in the Operating License application. Regeneration timelines and flow rates will be set to maintain effluent concentrations below those allowed in 10 CFR Part 20, Appendix B, and reported in the Operating License application. The total quantity of charcoal will be reported in the Operating License application. A holdup volume provides a time for the short-lived radionuclides to decay before entering the scrubber beds. The quantity of and dimensions of this holdup volume will be provided in the Operating License application. The holdup volume has a maximum flow rate to ensure only iodine, xenon, and krypton enter the scrubber beds. This flow rate will be provided in the Operating License application.

The off-gas system will be contained within the off-gas enclosure(s) which satisfy DC 4 and 61. The off-gas line connecting the reactor enclosure to the off-gas enclosure will be double walled. Contaminated helium from the reactor is stored and sequestered within the off-gas system for a time before release or reuse. A decay time of contaminated helium will be established in Technical Specifications before it is sent to the off-gas system. The design basis accident for the off-gas system is the rupture of one of the off-gas tanks within the off-gas enclosure. Penetrations of the off-gas enclosure are protected by isolation valves. These valves isolate the off-gas system in the event the system deviates from operating limits such as stack activity or off-gas enclosure pressure. Design pressures, temperatures, and leak rates for the enclosure will be selected to ensure rupture of the off-gas system does not result in radiological consequence exceeding the MHA.

Gases in the GMS will be passed through a scrubber which will remove the hydrogen and the hydrogen fluoride from the gas stream. The scrubber may utilize a liquid and a solid to effectively capture the wide variety of chemical species released by the sparging process in the fuel salt handling system. After scrubbing, this gas stream may be released to the atmosphere and subject to the appropriate limits.

9.6.4 Instrumentation and Control Requirements

There are sampling ports in the headspace of the reactor, after the holdup volume, and after the scrubber beds.

Pressure and flow rate are measured and controlled where necessary to ensure performance of the safety functions. Temperature of the system boundary and the scrubber beds are provided to ensure performance of the safety functions.

Activity of the holdup volume and scrubber beds is monitored. If the activity exceeds operating limits, appropriate actions are undertaken. This can include isolating the off-gas system or reducing reactor power. Effluent releases are monitored.

There is no indication from fundamental chemistry or the MSRE experience that actinides, such as uranium and plutonium, will volatilize in the fuel salt and migrate to the reactor gas space in substantial quantities. Any fuel salt droplets will collect on the piping and drip back into the RAV. To confirm this and other behaviors, a small quantity of reactor head space may be sampled and analyzed.

9.6.5 Technical Specifications, Testing, and Inspection

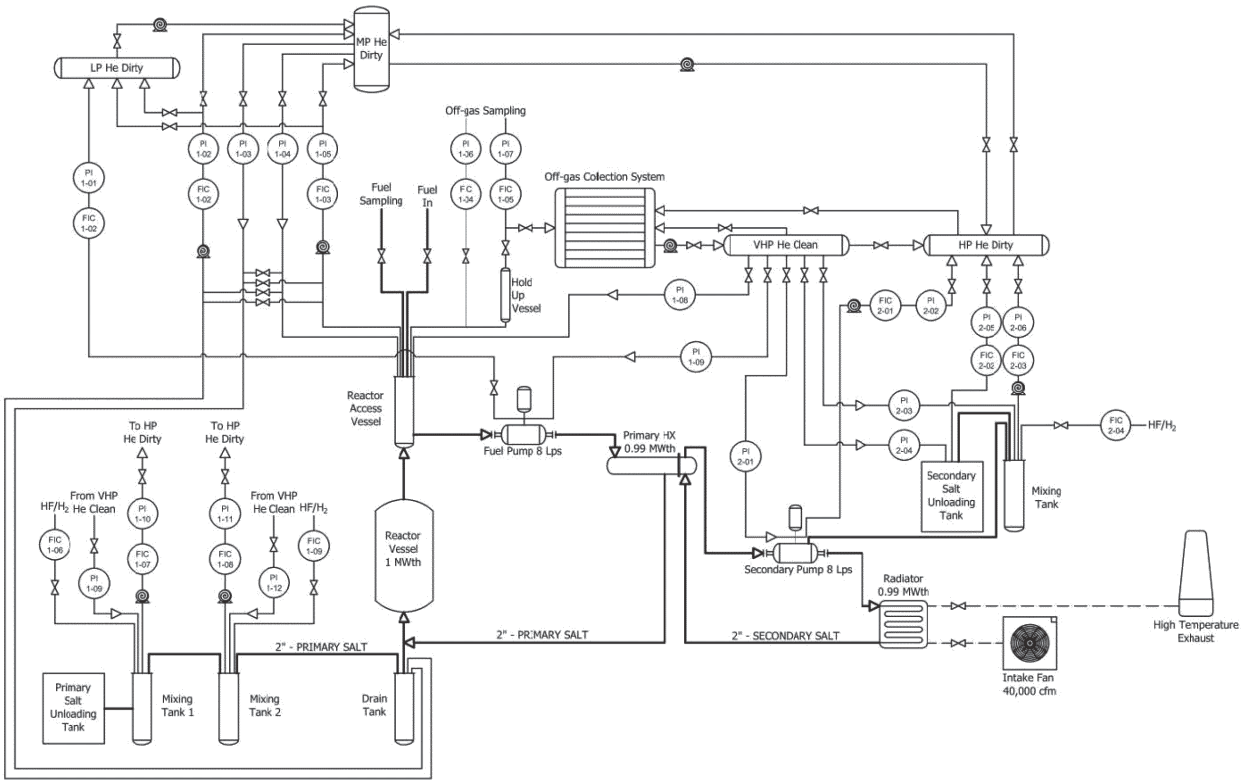
Flow rates for the off-gas loops and holdup volumes are set to maintain effluent concentrations below those allowed in 10 CFR Part 20, Appendix B, and will be reported in the Operating License application. The total quantity of charcoal will be reported in the Operating License application.

A preliminary list of parameters considered for monitoring is shown below. Values will be provided in the Operating License application.

- Gas pressure (high and low) limits
- Gas flow (high and low) limits
- Maximum allowable gas leakage rate out of the GMS
- Maximum allowable gas leakage rate into the GMS
- Maximum temperature of charcoal filter beds
- Maximum allowable activity concentration of fission products in the exhaust stack

A surveillance program is established to ensure barrier integrity, both of the GMS physical structure (first layer of piping) and of any enclosures in which gas management is located.

Figure 9.6-1 Conceptual Design of Gas Management System



9.7 Cooling Systems

9.7.1 Auxiliary Heat Removal System

9.7.1.1 Design Basis

Consistent with DC 38, functional containment heat removal, the AHRS is designed to transfer heat from the reactor cell to the atmosphere under normal operating conditions. This function is provided to maintain functional containment structure temperatures consistent with the initial conditions assumed in accident analyses.

Consistent with DC 40, testing of functional containment heat removal system, the isolation function of the AHRS is designed to permit appropriate periodic functional testing to ensure structural and leak-tight integrity of the louvers and associated ductwork, and actuation of isolation.

Consistent with DC 60, control of releases of radioactive materials to the environment, the AHRS is designed to isolate the reactor cell on loss of power and on detection of high levels of radioactivity to limit the release of radioactive materials.

9.7.1.2 System Description

As described in Section 6.2, the reactor cell and reactor enclosure temperatures are maintained by the AHRS, which routes 4000 cfm of air from the research bay into the reactor cell. This air flows upwards around the reactor enclosure and is then rejected back into the bay such that it does not mix with the rest of the bay air but goes out the bay exhaust. This system is required during reactor operation but not when the reactor is shut down. In the event of loss of power or high radiation levels, louvers in the AHRS isolate the reactor cell from the research bay and the research bay exhaust. The portions of the system necessary for detection of high radiation and isolation (louvers) are safety related. Details will be provided in the Operating License application.

9.7.1.3 Operational Analysis and Safety Function

The safety function of the AHRS is to isolate the reactor cell on detection of high levels of radiation or on loss of power to limit the release of radioisotopes. This function is described in Section 6.2. During operation, the system removes heat from the reactor cell to maintain the temperatures of the reactor enclosure and bioshield. The system is not required for safe shutdown or for decay heat removal. System isolation is the only function included in the safety analyses.

9.7.1.4 Instrumentation and Controls Requirements

Radiation and temperature of the air exiting the reactor cell are monitored. Appropriate limits will be provided in the Operating License application.

9.7.1.5 Technical Specifications, Testing and Inspection

The TS require the AHRS to operate during reactor operations. Limits on radiation and temperature will be described in the Operating License application.

Figure 9.7-1 Auxiliary Heat Removal System Conceptual Design

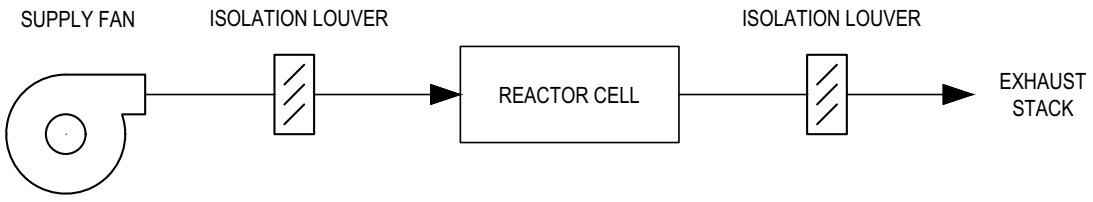
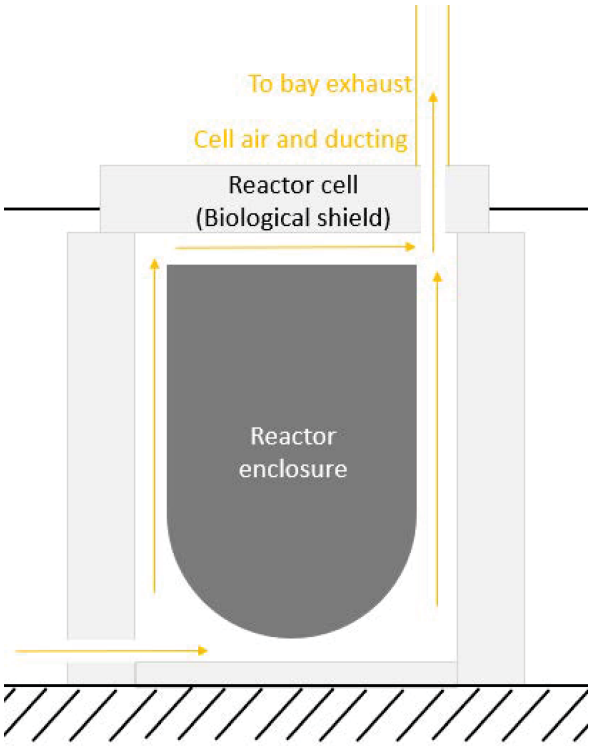


Figure 9.7-2 AHRS Conceptual Flow Path in Reactor Cell



9.7.2 Additional Cooling Systems

Additional cooling systems for the FHS, the chemical processing system, and the GMS will be identified and reported in the Operating License application.

9.8 Other Auxiliary Systems

9.8.1 Compressed Air, Vacuum, and Inert Gas Supply

The SERC building is outfitted with a compressed air system that will supply compressed air at up to 145 psi to lab spaces and the research bay. A one-inch line along the west wall of the research bay feeds that area. Two ¾-inch lines feed the radiochemistry lab; one along the west wall and another in the vent hood along the north wall. Steel piping for the system will conform to ASTM A 53/A 53M schedule 40 black. Fittings for steel pipe will conform to ASME B16.3, malleable iron, or ASTM A 234/A 234M forged steel welding type. Copper tubing used will conform to ASTM B 88, Type K drawn, fittings to ASME B 16.8 cast copper alloy or ASME B 16.22, wrought copper, and bronze. Copper joints will be brazed according to AWS A5.8 BCuP silver/phosphorous/copper alloy with a melting range of 1,190 to 1,480 degrees F. The system is equipped with 3/8 inch quick connect fittings at the terminus. Class 150 Teflon-seated ball valves are included to allow for easier use of the system.

The SERC building is also outfitted with a vacuum system that can supply vacuum up to 28.4 inches of mercury with access in the same places as the compressed air. Vacuum piping consists of copper tubing that conforms to ASTM B88, Type L, fittings that conform to ASME B16.22, wrought copper, and bronze or MSS SP 73 wrought and cast copper. Joints follow ASTM B32, alloy grade Sb5 tin-antimony or alloy grade Sn95 tin-silver, solder. Again, Teflon-seated ball valves are provided in the system for ease of use.

The SERC building is furnished with an inert gas supply system that is fed from inert gas bottles stored on a pad exterior to the building. The gases run from the gas supply on the exterior pad to the west wall of the research bay, the radiochemistry lab, and other labs in the building. The tubing for the gas supply will conform to ASTM A-269 fully annealed high-quality 316L seamless stainless steel hydraulic tubing with a hardness of R680 or less. All tubing will be free of scratches and suitable for bending and flaring. Tubing shall be designed for 75,000 psi for metal -20 to 100 degrees F and shall be designed for allowable working pressure loads from S-valves as specified by ANSI B-31.3. Tubing will be cleaned and bagged for service. A bending tool will be used where practical, and welded fittings for other direction changes. Joints will be welded. Valves will be supplied with the system and will be cleaned and purged before service. All tubing and fittings will be prepared in accordance with NFPA 99.

9.8.2 Cranes

The SERC building reactor bay includes a crane for moving heavy loads within the bay and the pit. The crane is an overhead bridge crane with a 40-ton capacity on the main hook and a 4-ton capacity on the auxiliary hook. The crane will be operated, maintained, and inspected using standards in ASME B30, NML-1, and HRT-1.

Consistent with DC 4, environmental and dynamic effects design basis, the crane is designed to accommodate the effects of and to be compatible with the expected environmental conditions during operation, maintenance, testing, and postulated accidents.

When not in use, the Crane will be maintained in a position where, should a disaster occur, it would not fall onto any components of the MSRR that would compromise safety or cause an uncontrolled release of radiation.

Administrative controls and crane use procedures will be developed for the Operating License application. The administrative controls and procedures will be informed by NUREG-0612 and the appropriate ASME standards. The usage of the crane requires a radio control pendant that is kept in a location in the control room secured by a lock. Only those people who have been trained to operate the crane will have access to open the lock and retrieve the radio control pendant. All operators for the crane will be trained in crane operation, capabilities of the lifting equipment, and inspection techniques to ensure reliable operation of handling equipment.

Loads paths for heavy loads will be predetermined and will receive a safety review and approval by appropriate management. The load paths will be determined in a way that will preclude the chance that a malfunction could prevent reactor shutdown,

cause an unanalyzed reactor accident or uncontrolled release of radioactive materials. Load path plans will take into account kinetic energy generated in an accidental drop and how that could affect SSC's along the path, and the position of SSC's relative to the path. Special lifting devices and other rigging will satisfy guidelines of appropriate standards such as ASME B30.9 and will consider static and dynamic loads. Operators will be trained in proper rigging techniques and the use of any special devices.

In addition to these administrative precautions, the MSRR will only be operated with concrete shielding panels totaling four feet thick covering safety related enclosures in the systems pit. These concrete panels will be designed to serve as a barrier, protecting the operating MSRR from heavy load drops.

9.9 References

9.9-1 International Code Council, "International Building Code," Washington, DC, 2012 (as adopted and amended by the City of Abilene, Texas).

9.9-2 International Code Council, "International Fire Code," Washington, DC, 2009 (as adopted and amended by the City of Abilene, Texas Fire Prevention Division).

9.9-3 National Fire Protection Association, "Standard for the Installation of Sprinkler Systems," NFPA 13, Quincy, MA, 2018.

9.9-4 National Fire Protection Association, "Standard on Fire Protection for Laboratories Using Chemicals," NFPA 45, Quincy, MA, 2019.

9.9-5 National Fire Protection Association, "National Fire Alarm and Signaling Code," NFPA 72, Quincy, MA, 2022.

9.9-6 American Society for Testing and Materials, "Standard Test Methods for Fire Tests of Roof Coverings," ASTM E108-20a, West Conshohocken, PA, April 2020.

9.9-7 Underwriters Laboratories, "Standard Test Methods for Fire Tests of Roof Coverings," UL 790, Ed. 9, Northbrook, IL, February 2022.

Chapter 10
Experimental Facilities and Utilization

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 10 EXPERIMENTAL FACILITIES AND UTILIZATION 10-1

10.1 Summary Description 10-1

10.2 Experimental Facilities 10-1

10.2.1 Salt Sampling and Measurement Experimental System 10-2

10.2.2 Gas sampling and Measurement Experimental System 10-5

10.2.3 Helium Bubble Generation and Removal Systems 10-8

10.2.4 Radiochemistry Laboratory 10-10

10.2.5 Scientific Surveillance Facilities 10-10

10.2.6 Probable Subjects of Technical Specifications 10-12

10.3 Experiment Review 10-13

10.4 References 10-13

LIST OF TABLES

Table 10.2-1 Gamma Ray and Neutron Dose Rate at 30 cm from Sample for 1 Megawatt Operation 10-3

Table 10.2-2 Anticipated Instruments for Salt Sampling and Measurement Experimental System 10-4

Table 10.2-3 Fraction of Derived Air Concentration for Gas Sample Release in Research Bay 10-6

Table 10.2-4 Anticipated Instruments for Gas Sampling and Measurement. 10-7

Table 10.2-5 Molten Salt Research Reactor Scientific Surveillance Layer Scope 10-11

Table 10.2-6 Monitoring Capabilities and Instruments 10-12

CHAPTER 10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

The intended purpose of the Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) is to accelerate the development and deployment of molten salt reactor systems through foundational research while developing a new pipeline to a nuclear qualified workforce. ACU's large capital investment in the MSRR provides a world-class molten salt research facility to be used by large numbers of students, staff, faculty, and outside collaborators. A preliminary description and evaluation of the systems are given in this chapter that will be finalized in the application for an Operating License.

Experimental facilities within the MSRR fall in two categories: 1) instrumentation, facilities, and a radiochemistry laboratory for the measurement of the molten salt and off-gas and 2) scientific surveillance to characterize select aspects of the MSRR including radiation physics, thermophysics, chemistry, material properties, radionuclide production and transport, and operation. In these categories there are methods to measure parameters including radioactivity, redox potential, corrosion effects, and particulate formation. Molten salt measurement systems additionally include features to measure the level (height) of molten salt within the reactor access vessel.

New experiments may be added following both a management review and a review by the MSRR Review and Audit Committee (committee). The committee reviews and approves all experimental facilities and procedures for experiments, and assesses each experiment within the regulations in Title 10 of the Code of Federal Regulations, Part 50, Section 59 (10 CFR 50.59).

Experimenters work with licensed reactor operators for experiment planning, and facilities access and utilization. Radiation monitors are placed around experimental areas that interact with extracted salts and gas. Air monitors will be installed to scan for aerosols, particulates, and droplets.

Security of the facilities is maintained by experimenters and reactor operators to ensure no personnel enter experimental systems areas without prior approval and radiation safety training. Alarms alert reactor operators of opening of experimental areas.

10.2 Experimental Facilities

Facilities and equipment for the measurement of the molten fuel salt are included in the salt sampling and measurement experimental system. Facilities and equipment for the measurement of the off-gas are included in the gas sampling and experimental system. There are helium bubble generation and removal systems to assess gaseous nuclide transport and capture rates within the fuel salt. An associated laboratory under the 10 CFR Part 50 MSRR facility license supports the molten fuel salt and off-gas measurement capabilities.

10.2.1 Salt Sampling and Measurement Experimental System

The objectives for the salt sampling and measurement experimental systems are to enhance the understanding of the evolving composition of the molten salt held within the reactor system, to study the salt's oxidation-reduction potential, and perform materials testing. Sampling will include removing salt along with integrated fuel. Removal of samples will be done only with the knowledge and consent of a reactor operator or senior reactor operator present at the controls pursuant to 10 CFR Part 55. The reactor access vessel resides above the reactor vessel and before the reactor pump. This is the region from which salt extractions and measurements are made. The salt sampling and measurement region consists of several ports penetrating the shielding and enclosure. One port is included for salt sampling directly from the salt surface within the reactor access vessel through a stainless steel tube from the top of the shielding through the port and into the salt.

Another port is included for measuring the height of salt within the reactor access vessel. Measurement of salt height provides information about the current volume of salt in the system and verifies changes due to the addition of fuel or the emptying of the system. These sensors are described in Chapter 4 and Chapter 7. Data from these sensors may be used for experimental purposes.

Another port is included to add an oxidation-reduction potential probe.

Ports and systems for coupon testing of reactor materials will be incorporated on a bypass line from the reactor pump outlet to the RAV. This line is a part of the reactor system and satisfies the same design criteria.

10.2.1.1 Design Bases

The aspects of the molten salt experimental systems which pertain to maintaining fission product boundaries are considered safety related. The function of the salt sampling system is not safety related. The salt sampling system removes a sample of fuel salt to measure the salt chemistry. The salt sampling system interfaces with the reactor system and the ESF to perform this function, therefore it will obey the reactor system and ESF design criteria.

The molten salt experimental systems are oriented around the region above the reactor enclosure top flange and passing through the biological shielding. Component lines (such as the sampling system) are stainless steel 316H tubing and are sealed against the reactor enclosure. The reactor system may utilize a redox probe which is composed of a different material. The redox probe will be electrically insulated from the reactor system structure so it will not galvanically corrode the structure. Manipulations of equipment are done by remote-controlled tooling and robotic arms where appropriate to minimize radiation worker dose consequence.

Molten salt sampling is done through a double-walled stainless steel tube through biological shielding and into the reactor access vessel. Remotely controlled isolation valves are used to maintain appropriate functional containment as described in Chapter 6 and Chapter 3. Molten salt is pulled into an inert gas environment before being packaged and sent to labs for analysis.

Consistent with DC 55 in section 3.1.2.5, radionuclide interfacing lines penetrating functional containment, the salt sampling and measurement experimental system includes containment isolation provisions.

Test coupons are submerged in molten salt for quantifying corrosion characteristics during operation and are retrievable through an opening in the shielding. Test coupon material is compatible with the reactor system and fuel salt.

10.2.1.2 Reactivity

The reactivity induced by sample extraction is anticipated to be less than 30 pcm.

10.2.1.3 Radiological Assessment

Preliminary dose calculations are performed to provide an estimate of exposure from the samples. This was performed for both gamma and delayed neutron dose with the MSRR operating at 1 MW_{th}. Calculations were performed using a SCALE 6.2.2 TRITON-6 depletion to ORIGEN decay sequence [Reference 10.4-3]. A three-dimensional model was provided in the 3D depletion module of SCALE to generate problem-specific cross-section libraries to use instead of the standard pre-generated light-water reactor ones in SCALE. ORIGEN was utilized to calculate radionuclide inventory from a 30-day irradiation period followed by incremental decay periods of one day, one week, and one month after removal of the sample. The 30-day irradiation period allows for adequate build-in of radionuclides that denominate the gamma-ray and neutron dose-rate calculations.

ORIGEN results are compiled as continuous energy bins in units of intensity. Dose rates were then calculated as a function of decay time. The results are presented in Table 10.2-1 in mrem/hr at a range of 30 cm for 1-gram sample. Results approximately scale linearly as a factor of power level and sample mass.

Table 10.2-1 Gamma Ray and Neutron Dose Rate at 30 cm from Sample for 1 Megawatt Operation

Time after removal (days)	Dose rate at 30 cm (mrem/hr/g)
0	761
1	115
7	34.7
30	6.82

The dose rate from delayed neutrons is calculated in a similar manner. Because of the delayed neutron precursor half-lives, neutron dose does not contribute to overall dose after 10 minutes of decay.

10.2.1.4 Instrumentation

Gross gamma and neutron counters are positioned above and around the RAV ports. Air monitors are positioned to measure the surrounding airspace for aerosols and droplets.

A level sensor is inserted through one port and into the molten salt reactor access vessel to monitor the height of the molten salt. A REDOX probe is inserted through a separate port to monitor salt corrosion effects.

Gamma detectors located entirely within the reactor enclosure may be used to obtain fission product spectrum data from various points around the reactor system. RAV port penetrations for these detectors are not required. Table 10.2-2 summarizes the anticipated experimental instruments that may be used in the experimental facility. The FSAR will include a final instrumentation list.

Table 10.2-2 Anticipated Instruments for Salt Sampling and Measurement Experimental System

Instrument	Measurement	Location
Gross gamma counter	Gamma field	Above/around RAV ports
Gross neutron counter	Neutron field	Above/around RAV ports
Air monitor	Aerosols, droplets	Above RAV ports
Level sensor	Molten salt height in tank	RAV port penetration
REDOX probe	Oxidation-reduction potential	RAV port penetration
Radiation detector	Gamma ray spectrum	Inside reactor enclosure

10.2.1.5 Physical Restraints and Shielding

The experimental systems have physical restraints and shielding. Physical restrictions include concrete and fenced barriers. Tamper alarms are set to notify reactor operators if physical restraints have been opened. Shielding placed around the experimental facility reduces radiation levels in the research bay area. Experimental systems shielding limits dose rate from experiments in the research bay area.

10.2.1.6 Operating Characteristics

Molten salt sampling involves extracting small quantities of salt from the sampling tube leading to the reactor access vessel. The stainless steel tube with its bottom submerged into the molten salt is a permanent fixture of the experimental system. A vacuum is pulled on the sampling line, pulling salt up the line to sample a small quantity. The salt sample is placed into a shielded transport container outside the reactor to be sent to the radiochemistry lab for analysis. Clogging of the tube with

frozen molten salt is prevented by pressurizing the tube with cover gas to dislodge any blockage, or by means of heating. Sampling tube lifetime is on the same order as the rest of the reactor, making replacement unnecessary. Because the tube material is identical to the reactor system physical structure, material longevity is consistent with the entire MSRR system.

Sampling volumes are small inducing only minimal reactivity effects. Unexpected release of extracted salt is kept small by minimizing sampling size. Radiation releases are tracked by monitoring gross gammas and neutrons around the sample transport container by reactor operators in the control room.

10.2.1.7 Safety Assessment

A safety assessment of the radiological and chemical hazards associated with the experimental system will be provided in the Operating License application. Samples of radioactive material are appropriately encapsulated and shielded to ensure compliance with 10 CFR Part 20 and applicable chemical safety guidelines.

10.2.2 Gas sampling and Measurement Experimental System

10.2.2.1 Description

The second set of experimental systems involves measurements of the off-gas emitted by reactor salts as described in Section 9.6. The objectives for this experiment are to characterize the constituent elements and molecular composition within the off-gas stream. The off-gas experimental system region consists of several lines that extend from the main off-gas reprocessing line and are sent to detection zones for analysis. Gas is sent to a sample collection cell that stores small amounts of off-gas to be removed from the system for composition analysis. Removal of samples is done only with the knowledge and consent of a reactor operator or senior reactor operator present at the controls pursuant to 10 CFR Part 55. Additional closed-loop gas lines feed in-situ gas analysis systems.

The gas sampling system is in compliance with Design Criterion 55 (see Section 3.1.2.5). An off-gas sample present within the RAV is withdrawn to a sample collection vessel located outside the reactor enclosure. At least three isolation valves separate the RAV from the collection vessel along the line: two isolation valves in the enclosure and one outside the enclosure. When open, the collection vessel obtains a quantity of off-gas. When the valves close, this volume may be transferred to the radiochemistry laboratory for processing. The collection vessel is surrounded by a pressure retaining container.

10.2.2.2 Design and Specifications

The design of the off-gas system experimental region is incorporated into the reactor off-gas system to best assist in characterizing the contents of the gas stream. Piping is stainless steel 316H within the RTMS and 316L outside the

RTMS, and is of sufficient diameter and wall thickness to tolerate corrosion by molten salt particulates and fission products. Connections between components are welded fittings.

Valves and detector components are controlled remotely through pneumatic controls or electronic signals.

10.2.2.3 Reactivity

The reactivity induced by the off-gas experimental facility is small as anticipated reactivity changes due to gas sampling and measurement are less than 30 pcm.

10.2.2.4 Radiological Assessment

Research bay activity concentrations are normalized to derived air concentration (DAC) values as provided in 10 CFR Part 20 Appendix B. These calculations are performed to obtain values for airborne radionuclides in the research bay using a TRITON-ORIGEN sequence. Individual isotope activities in units of mCi were divided by the room volume of 5100 m^3/mL. These are then divided by the DAC values in 10 CFR Part 20, Appendix B, to obtain normalized values. The sum of these individual nuclide values is used to obtain the total fractional DAC value upon removal. Table 10.2-3 provides decay for one day, one week, and one month, for 1-mg of salt irradiated for 30 days at 1 MW. Results can be scaled linearly to other masses as an approximation.

Table 10.2-3 Fraction of Derived Air Concentration for Gas Sample Release in Research Bay

Isotope	T = 0	T = 1 day	T = 1 week	T = 1 month
Kr-83m	2.65E-07	1.27E-09	1.71E-17	1.42E-17
Kr-85	7.34E-08	7.41E-08	7.40E-08	7.37E-08
Kr-85m	3.22E-04	7.96E-06	1.68E-15	0.00E+00
Kr-87	2.53E-03	5.32E-09	0.00E+00	0.00E+00
Kr-88	8.77E-03	2.51E-05	1.37E-20	0.00E+00
Xe-131m	2.39E-07	2.45E-07	2.49E-07	1.17E-07
Xe-133	3.10E-04	3.01E-04	1.51E-04	7.31E-06
Xe-133m	9.53E-06	8.72E-06	1.71E-06	1.20E-09
Xe-135	3.23E-03	1.18E-03	3.28E-08	2.17E-26
Xe-135m	6.51E-04	4.73E-05	1.19E-11	0.00E+00
Xe-138	7.78E-03	0.00E+00	0.00E+00	0.00E+00
I-126	6.63E-11	6.29E-11	4.56E-11	1.33E-11
I-128	1.05E-10	4.73E-28	0.00E+00	0.00E+00
I-129	2.15E-09	2.18E-09	2.21E-09	2.31E-09
I-130	3.46E-06	9.04E-07	2.81E-10	1.01E-23
I-131	6.55E-01	6.10E-01	3.70E-01	5.09E-02
I-132	7.12E-03	5.89E-03	1.61E-03	1.11E-05

Table 10.2-3 Fraction of Derived Air Concentration for Gas Sample Release in Research Bay (Continued)

Isotope	T = 0	T = 1 day	T = 1 week	T = 1 month
I-132m	1.12E-05	6.95E-11	0.00E+00	0.00E+00
I-133	3.31E-01	1.54E-01	1.27E-03	1.30E-11
I-134	1.94E-03	4.79E-11	0.00E+00	0.00E+00
I-135	4.45E-02	3.54E-03	8.94E-10	0.00E+00
Br-80	9.29E-12	1.73E-13	2.71E-23	0.00E+00
Br-80m	7.97E-11	1.85E-12	2.89E-22	0.00E+00
Br-82	1.35E-07	8.42E-08	4.98E-09	9.71E-14
Br-83	8.83E-05	9.99E-08	8.67E-26	0.00E+00
Br-84	2.44E-04	6.09E-18	0.00E+00	0.00E+00
H-3	8.58E-07	8.58E-07	8.57E-07	8.54E-07
Total	1.06E+00	7.74E-01	3.73E-01	5.09E-02

10.2.2.5 Instrumentation

Gross gamma and neutron counters are positioned above and around the ports. Air monitors are positioned to measure the surrounding airspace for aerosols and gas.

Detectors using optical measurement methods (e.g., Raman, LIBS) are used for monitoring the off-gas for elemental and molecular compositions.

A photon detector positioned over an off-gas line is included for characterization of the off-gas radioactivity. Table 10.2-4 summarizes the instruments anticipated for possible utilization in this facility.

Table 10.2-4 Anticipated Instruments for Gas Sampling and Measurement

Instrument	Measurement	Location
Gross gamma counter	Gamma field	Above/around manifold
Gross neutron counter	Neutron field	Above/around manifold
Air monitor	Aerosols, droplets	Above manifold
Thermocouples	Temperature	In-situ gas line
Pressure transducer	Pressure	In-situ gas line
Flow meter	Mass flow rate	In-situ gas line
Optical detector(s)	Elements, molecules	In-situ gas line
Radiation detector(s)	Gamma ray spectrum	In-situ gas line

10.2.2.6 Physical Restraints and Shielding

The experimental systems have physical restraints and shielding. Physical restrictions include concrete and fenced barriers. Tamper alarms notify operators if physical restraints have been opened. Shielding around the experimental facility reduces radiation levels in the research bay area.

10.2.2.7 Operating Characteristics

Off-gas extraction is done by first pulling a vacuum on the sample line attached to the off-gas manifold. A valve separates the main off-gas stream from this sample line. After purging the sample line and filling with cover gas, the valve to the off-gas stream is opened to allow a small amount of off-gas to enter the line. The valve to the off-gas stream is then closed. The sample line sends a small quantity of the off-gas to a remote laboratory with analysis equipment by pneumatics. Any off-gas leakage is be detected by the surrounding radiation monitors to indicate the issue to reactor personnel.

All other detector systems operate using windows to the off-gas stream without directly interacting with the gas itself. Operation of these detectors is done remotely using computers to activate and deactivate lasers and detectors, and to collect data. Detection does not interfere with the normal functioning of the off-gas system, and is included around other features of the off-gas system such as scrubbers and charcoal filters.

10.2.2.8 Safety Assessment

A safety assessment of the radiological and chemical hazards associated with the experimental system will be provided in the Operating License application. Samples of radioactive material are appropriately encapsulated and shielded to ensure compliance with 10 CFR Part 20 and applicable chemical safety guidelines.

10.2.3 Helium Bubble Generation and Removal Systems

10.2.3.1 Description

The objective of the helium bubble generation and removal systems are to conduct experiments in conjunction with the off-gas system to better understand gaseous nuclide transport and capture rates from the salt. The systems use helium bubbles as a catalyst to develop a better understanding of the gas dynamics of bubbles in the system, and determine the reactivity impacts with small injections of bubbles in the system. The bubble and removal system is capable of running in three modes. The first has no bubbles injected into the system. The second allows for varying measured generation of bubbles. The third removes gas from the salt and passes it to the off-gas system.

The bubble generation system is located just before or in the reactor access vessel and requires tubing to supply helium gas to the system. A system of remote-actuated gas valves that are part of the gas management system is used to turn the bubbling system on and off. The specific design of the bubble generator will be provided in the Operating License application.

The bubble removal system is located after the reactor access vessel, and requires tubing connection to the off-gas system. The specific design of the bubble removal system will be provided in the Operating License application.

10.2.3.2 Design and Specifications

The bubble generation and removal systems have stainless steel 316 facing the fuel salt. The 316H stainless steel is of appropriate diameter to supply or remove sufficient helium to or from the system and resist clogging. The tubing wall thickness is great enough to resist corrosion by the fuel salt and fission products.

Valves and controls for the system are remotely actuated either pneumatically or electrically.

The design of the bubble generation and removal system and specific limits on the amount of gas that can be added to the system will be provided in the Operating License application.

10.2.3.3 Reactivity

The removal of gas from the fuel salt is anticipated to have reactivity effects dominated by the removal of Xe-135. Experiments will be conducted to assess the maximum reactivity effects from gas removal and the reactor kinetic responses from this removal.

10.2.3.4 Radiological Assessment

Radiological assessment is similar to what is reported in Section 10.2.2.4.

10.2.3.5 Instrumentation

Gas-flow sensors in the helium gas supply line and gas removal lines, reactor access vessel off-gas line, and pump off-gas line monitor flow of gas into and out of the system.

Pressure transducers on each input and exhaust line monitor gas pressures.

10.2.3.6 Physical Restraints and Shielding

The experimental systems have physical restraints and shielding. Physical restrictions include concrete and fenced barriers. Tamper alarms notify reactor operators if physical restraints have been opened. Shielding is placed around the experimental facility reduces radiation levels in the research bay area.

10.2.3.7 Operating Characteristics

Specific operating characteristics of the helium bubble generation and removal system will be included in the Operating License application.

10.2.3.8 Safety Assessment

A safety assessment of the radiological and chemical hazards associated with the experimental system will be provided in the Operating License application.

10.2.4 Radiochemistry Laboratory

The radiochemistry laboratory is used for chemical and radiological analysis of materials sampled in and around the MSRR. It is equipped with a glove box designed to accommodate radioactive samples. Radiological surveillance equipment includes an alpha/beta/gamma hand and foot monitor, an alpha/beta particle monitor, a wall-mounted area radiation monitor, and an automatic low-background alpha/beta counting system for monitoring surface swipes from throughout the facility. A HPGE gamma-ray detector provides the ability to precisely measure the gamma spectrum for salt or other materials from the reactor facility. An array of chemical analysis instrumentation delivers the capability to provide in-depth atomic, isotopic, and molecular qualitative and quantitative analysis of materials from the reactor facility. Fuel salt, coolant salt, headspace gasses, coupons, and containment materials can be analyzed in this lab. A flame atomic absorption spectrometer provides elemental analysis with detection limits in the parts per million range. An inductively coupled plasma-mass spectrometer extends that capability to isotopes with detection limits into the parts per trillion range. A gas chromatograph-mass spectrometer is capable of gas-phase, liquid, and volatile solid-phase sample analysis, including molecular species, which is particularly useful for monitoring headspace gas purity and composition. Equipment similar to a LECO O836 elemental analyzer will enable researchers to monitor oxygen content in salt samples. Raman spectrometers provide the ability to monitor molecular species in reactor off-gas and in molten and quenched salt samples. An ultraviolet-visible spectrometer provides another method of monitoring components in the salt such as transition metal impurities. An electrochemical system provides the ability to monitor the electrochemical potential of the molten salt, allowing researchers to maintain that potential in a desirable range. It also provides an additional method to identify and quantify certain components within the molten salt (described in Chapter 9).

10.2.5 Scientific Surveillance Facilities

The separate group of experimental facilities of the MSRR is the scientific surveillance facilities. These facilities are designed to characterize aspects of the MSRR: radiation physics, thermophysics and mechanics, chemistry, materials, radionuclide production and transport accountancy within the system, and operation.

10.2.5.1 Description

The scientific surveillance facilities are designed to gather information and data to support future development and licensing of molten salt reactors as well as education of professionals with the focus on molten salt reactors. The scientific surveillance facilities form a layer of instrumentation, computer hardware and software, and supporting design features, called the scientific surveillance layer (SSL), which is capable of capturing the MSRR behavior during its operation.

The sensors and design features comprising the scientific surveillance layer include dedicated information and data streams, data streams from additional experiments, as well as data streams from the reactor instrumentation and controls. The scientific surveillance layer does not perform a safety related

function. Where the scientific surveillance layer penetrates or interfaces with a safety related system, it shall satisfy the design criteria and requirements of that system.

10.2.5.2 Design Bases

The SSL data acquisition system is designed to collect data and support the wide range of needs for molten salt reactors:

- Reactor physics delayed neutron precursors drift characterization, reactivity effects, validation of neutronic and coupled tools
- Salt characteristics evolution during operation (thermophysical properties, radiation stability, operational effects)
- Radionuclide production, mass transport and accountancy (fission products, activation products, solubility, diffusion, source term including volatility of instrumentation and controls)
- Operation effects on materials including corrosion [radiolysis, fission product impurities (like Te), structural materials performance]
- Operational behavior and component performance characterization.

The MSRR is intended to support development and licensing efforts for future molten salt reactors. Table 10.2-5 summarizes phenomena of interest that are within the SSL scope.

Table 10.2-5 Molten Salt Research Reactor Scientific Surveillance Layer Scope

Relevant Phenomena	Impacting Factors
Salt chemistry control and phase evolution	Irradiation and flow
Evolution of thermophysical properties during operation	Irradiation, Flow and Operational range
Fission product source term and transport	Irradiation, Flow and Operational range
Corrosion effects during operation	Irradiation and Flow
Volatilization of salt species and gas formation	Irradiation and Flow
Heat transfer characteristics	Flow
MSRR component performance demonstration	Full operational range

10.2.5.3 Reactivity

The SSL is not expected to affect reactor reactivity.

10.2.5.4 Radiological Assessment

The SSL is not expected to affect the overall radiological assessment of the MSRR.

10.2.5.5 Instrumentation

Monitoring capabilities are supported by the MSRR design and appropriate sensors. General categories of instruments are provided in Table 10.2-6. The SSL instruments cover a range of phenomena time scales and conditions including volatile gas effects on salt properties and operational characteristics. Both integral (bulk) and differential effects are of interest.

Table 10.2-6 Monitoring Capabilities and Instruments

Instrumentation	Location	Method
Radiation monitors	Includes instrumentation of both fuel and coolant side.	Fission chambers, gamma detectors, etc.
Non-radiation monitors	Throughout the system at the determined locations of interest	Thermometry, flow, pressure, vibration sensors, strain sensors, high-fidelity acoustic transducers
Fission product generation, migration, collection, and speciation	Throughout the system at the determined locations of interest	Optical spectroscopy
Corrosion product generation	Throughout the system at the determined locations of interest	Salt sample analysis

10.2.5.6 Physical Restraints, Shields, or Beam Catchers

The SSL does not have any additional physical restraints, shields, or beam catchers.

10.2.5.7 Operating Characteristics

Operation of the SSL is independent of reactor operations and is not required. It is anticipated the SSL will be capable of operation under all conditions to collect data necessary for understanding the reactor system.

10.2.5.8 Safety Assessment

No safety concerns are anticipated by operation of the SSL

10.2.6 Probable Subjects of Technical Specifications

Probable topics for technical specifications for experimental systems focus on maximum size of samples removed, security requirements for experimental systems, and maximum permissible values measured on some of the experimental system instruments.

10.3 Experiment Review

ACU's Radiation Safety Office is assigned responsibility for implementing the radiation protection program at the MSRR facility using the guidelines of ANSI/ANS-15.11 [Reference 10.4-2]. The Radiation Safety Office reports to the Vice President of Research. Management review of experiments is conducted prior to review by the MSRR Review and Audit Committee and includes representation from the Radiation Safety Office. The committee reviews and approves all experimental facilities, procedures for experiments, and assesses each experiment within the regulations of 10 CFR 50.59. Review includes the description and purpose of the experiments, experimental facilities, experimental procedures, and a safety assessment of the experiments (described in Chapter 12). Radioactivity, material hazards, and chemical safety are included within the analysis. Experiments shall be maintained within the bounds of technical specifications.

10.4 References

10.4-1 American National Standards Institute/American Nuclear Society, “The Development of Technical Specifications for Research Reactors,” ANSI/ANS 15.1, 2007 (R2018), LaGrange Park, IL.

10.4-2 American National Standards Institute/American Nuclear Society, “Radiation Protection at Research Reactor Facilities,” ANSI/ANS-15.11 1993 (R2021), LaGrange Park, IL.

10.4-3 W. A. Wieselquist, R. A. Lefebvre, and M. A. Jessee, Eds., SCALE Code System, ORNL/TM-2005/39, Version 6.2.4, Oak Ridge National Laboratory, Oak Ridge, TN (2020).

Chapter 11
Radiation Protection Program and Waste Management

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT 11-1

11.1 Radiation Protection 11-1

11.1.1 Radiation Sources 11-1

11.1.2 Radiation Protection Program 11-1

11.1.3 As Low As Reasonably Achievable Program 11-2

11.1.4 Radiation Monitoring and Surveying 11-2

11.1.5 Radiation Exposure Control and Dosimetry 11-3

11.1.6 Contamination Control 11-4

11.1.7 Environmental Monitoring 11-4

11.1.8 References 11-5

11.2 Radioactive Waste Management 11-5

11.2.1 Radioactive Waste Management Program 11-5

11.2.2 Radioactive Waste Controls 11-5

11.2.3 Release of Radioactive Waste 11-6

11.2.4 Estimated Quantities of Waste Generation 11-6

11.3 Respiratory Protection Program 11-6

CHAPTER 11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

The following subsections identify sources of radiation and describe program elements for radiation protection. The program descriptions are at an appropriately high level and identify use of existing approved NRC regulatory guides (RGs) for demonstrating conformance to 10 CFR Part 20. The identification of expected radiation sources along with commitments to existing approved guidance provide reasonable assurance, consistent with 10 CFR 50.40(a) and 10 CFR 50.34(b)(3), that the regulations in 10 CFR Part 20 are met, and the health and safety of the public will not be endangered. Additional details of these programs will be in the Operating License application.

11.1.1 Radiation Sources

This section identifies various radiation sources monitored and controlled as part of the radiation protection and radioactive waste management programs. It provides tabulations of anticipated radiation sources as a direct result of reactor operations. The sources of radiation result from fission in the fuel (fission products and decay products) and neutron activation products (including tritium) generated as a result of exposure to neutrons.

Tritium screening analysis has shown the overwhelming source of tritium production in the molten salt research reactor (MSRR) facility is from the irradiation of FLiBe salts, as both fuel and cooling salts contain beryllium and lithium. The lithium content in the MSRR salts has an enrichment of 7-lithium to greater than 99.99 percent to minimize tritium production from 6-lithium capture. The MSRR salts produce approximately 1.5 curies (Ci) of tritium per megawatt day. Because the maximum licensable power for the MSRR is limited to 1 MWth, and assuming constant operation at that power level (bounding argument), the maximum tritium production in the facility is 1.5 Ci per day. Public effects of releasing all of this tritium, along with other expected effluents, results in doses of less than 1 mrem and are discussed in Section 19.4.9.2.2.2.

There are no current estimates for the activation of corrosion products in the primary salt loop. Estimated concentrations of Cr and Fe in the primary salt loop are below 100 ppm, and even smaller in the secondary salt loop. The radiological hazard posed by these activation products is dwarfed by the prompt fission and fission products' radiological hazard. Details about radiation sources, including activity and radiation fields, will be provided in the Operating License application.

11.1.2 Radiation Protection Program

A radiation protection program (RPP) will be created for the MSRR facility as required by 10 CFR 20.1101. It is designed to meet the requirements of 10 CFR Part 20 and 10 CFR Part 19, and developed, documented, and implemented commensurate with the scope and extent of activities at a research reactor facility consistent with ANSI/ ANS15.11-2016 (R2021) [Reference 11.1.8-1], and with consideration of RG 8.2,

Administrative Practices in Radiation Surveys and Monitoring, Revision 1; RG 8.13, Instruction Concerning Prenatal Radiation Exposure, Revision 3; and RG 8.29, Instruction Concerning Risks from Occupational Radiation Exposure, Revision1.

Periodic reviews of the programs content and implementation are per 10 CFR 20.1101(c). These reviews identify areas of improvement to help keep occupational and public doses as low as reasonably achievable (ALARA), as required by 10 CFR 20.1101(b).

The initial organizational structure of facility operations is shown in Section 12.1. The radiation protection training program is designed and implemented in accordance with the requirements of 10 CFR 19.12, and recordkeeping is performed in accordance with 10 CFR Part 20, Subpart L.

Additional details of the RPP for the facility, including organization and staffing levels, authorities and responsibilities, position qualifications, personnel training requirements, and document control and record keeping procedures, will be provided in the Operating License application.

11.1.3 As Low As Reasonably Achievable Program

Management of the MSRR is committed to keeping both occupational and public radiation exposure ALARA. The ALARA policy issued by the President of the University makes it clear that all personnel are responsible for ensuring work is performed in accordance with ALARA principles. In achieving this objective, the MSRR staff keeps radiation exposure to workers and members of the public below the limits in 10 CFR 20.1201 and 10 CFR 20.1301, as well as below the airborne effluent dose limits in 10 CFR 20.1101. The ALARA program is designed with guidance from RG 8.10, Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable, Revision 2, and ANSI/ANS15.11-2016 (R2021) [1].

Additional details about the ALARA program will be provided in the Operating License application, pursuant to 10 CFR 50.34(b)(3) and 10 CFR 50.34(b)(6).

11.1.4 Radiation Monitoring and Surveying

Radiation levels within the MSRR facility are monitored to ensure they are maintained within safe bounds for both workers and the public, and to ensure continued compliance with regulatory and technical specifications requirements. Monitoring activities are performed with instruments of an appropriate range of response and sensitivity to the types of radiation relevant to the monitored area. Monitoring requirements will be established as a part of internal procedures and technical specifications, and will be primarily based on the guidance provided in the following:

- ANSI/ANS 15.11-2016 (R2021)
- RG 8.2, Administrative Practices in Radiation Surveys and Monitoring, Revision 1
- RG 8.4, Personnel Monitoring Device - Direct-Reading Pocket Dosimeters, Revision 1

- RG 8.7, Instructions for Recording and Reporting Occupational Radiation Exposure Data, Revision 4
- RG 8.9, Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program, Revision 1
- RG 8.25, Air Sampling in the Workplace, Revision 1
- RG 8.34, Monitoring Criteria and Methods to Calculate Occupational Radiation Doses, Draft Revision 1

The requirements for radiation monitoring and surveying are outlined in 10 CFR Part 20, Subpart F. The purpose of radiation monitoring and surveying is to quantify the radiation field intensity and the quantity of radioactive materials, and to detect other radiological hazards and releases of radioactive materials from facility operations and equipment. Surveys and monitoring focus on sections of the facility where external exposure or internal uptake of radionuclides can cause occupational doses in excess of limits allowed in 10 CFR Part 20. Measurements of uptake of radiological material are calculated through airborne radioactive material monitoring. Bioassays can be used when a substantial (greater than 10 percent) annual limit on intake uptake of radionuclides is suspected consistent with RG 8.9. Survey and monitoring results help to ensure exposures to radiation do not exceed dose limits set in 10 CFR Part 20.

Written procedures will be developed to establish compliance with the requirements with 10 CFR Part 20, Subpart F. Additional details about radiation monitoring and surveying programs will be provided in the Operating License application.

11.1.5 Radiation Exposure Control and Dosimetry

This subsection addresses controls for exposure and access to radioactive materials. Additional details of radiation exposure control, shielding, salt handling equipment, and expected annual radiation exposures will be provided in the Operating License application.

Radiologically-controlled areas are established to protect workers and the public against undue radiation exposures or radiological intake. Although not anticipated to be present during normal operations, access to high and very high radiation areas are controlled as prescribed by 10 CFR Part 20, Subpart G. Procedures governing access controls, signage, and labeling are developed for the MSRR pursuant to 10 CFR Part 20, Subparts G and J.

The effluents from the MSRR facility are monitored for radioactivity during normal operations and postulated events. The structures, systems, and components are designed to prevent uncontrolled leakage of liquid or gaseous effluents into working areas or the environment. Releases from normal operations are discussed in Section 19.4.9.2.2.2. Radioactivity releases under postulated accident conditions are evaluated in Chapter 13.

The MSRR facility heating, ventilation, and air conditioning system (Section 9.1) includes gaseous effluent monitoring and filtration, after which gaseous effluents are generally released to the atmosphere. Other potential gaseous effluent release points include the auxiliary heat removal system exhaust (Section 9.7.1) and the MSRR gas management system (Section 9.6). Further analysis of effluents will be included in the Operating License application.

11.1.6 Contamination Control

Per 10 CFR 20.1406, MSRR facility management is committed to ensuring contamination is minimized to the extent practicable. Bases for procedures and structure, system, and component requirements relating to working within, surveying, and monitoring contamination and possibly-contaminated areas are provided by ANSI/ANS 15.11-2016 and RG 4.21, Revision 0, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning:”

- Minimizing leakage and spills of contaminants to prevent the spreading of contamination
- Capabilities to detect leakage
- Minimizing the capability of leakages to spread contamination
- Periodic reviews of operations, procedures, and practices

A description of the design features and procedures for the control of radioactive contamination for the facility, will be provided in the Operating License application.

11.1.7 Environmental Monitoring

Environmental monitoring is required at the MSRR facility in accordance with 10 CFR 20.1302 and the technical specifications for the facility operating license. The RPP provides a program for detection and evaluation of public and occupational radiation exposure on the technical basis of ANSI/ANS 15.11-2016. Surveys of the environment are included in the RPP. Calibration of monitors and other instruments are conducted in accordance with the recommendations and are fully described in health physics procedures. The MSRR Review and Audit committee has the responsibility to review radiation monitoring procedures and audit results, and to make recommendations for continued improvement.

Baseline condition evaluations have been made by taking soil samples and gamma surveys of the area around the proposed site. The results of the analysis of these samples and where they are to be stored during the lifetime of the reactor facility will be discussed in the Operating License application. The MSRR facility staff are committed to keeping effects on the environment ALARA.

Releases to the environment are detected either through monitoring contamination, monitoring effluents, environmental surveys, or monitoring airborne radioactivity within the research bay. These monitoring procedures protect the environment and public from facility radioactive releases and allow understanding the radiological impact of the facility.

The RPP is implemented coincident with the start of plant operational activities and the environmental monitoring program will be described in the Operating License application.

11.1.8 References

11.1.8-1 American National Standards Institute/American Nuclear Society, Radiation Protection at Research Reactors, ANSI/ANS 15.11-2016, LaGrange Park, IL.

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

A description of the radioactive waste management program for the facility, including organization and staffing levels, authorities and responsibilities, position qualifications, personnel training requirements, and document control and record keeping procedures, will be provided in the Operating License application. The preliminary organizational structure for the facility is described in Section 12.1.

As part of the ALARA program, the radioactive waste management program is periodically reviewed to look for opportunities to improve the capabilities of the program to keep these waste volumes and activities ALARA. This review includes periodic assessment of the waste streams to look for ways to minimize waste generation and find methods to reduce waste volumes. Additional details will be provided in the Operating License application.

11.2.2 Radioactive Waste Controls

Radioactive waste is defined as an item or substance that contains radioactivity above background levels and is no longer of use to the mission of the MSRR facility. Equipment, consumables, and components are categorized as radioactive waste by the MSRR staff and segregated at the point of creation from other items that are not considered radioactive waste.

During normal operations, liquid radioactive wastes are packaged and disposed of using a licensed and qualified low-level radioactive waste disposal vendor. Solid radioactive waste at the MSRR facility is primarily generated by reactor operation, either as a byproduct of experiments, such as material coupons, or from maintenance, such as reactor structural components and tools. Additional radioactive waste is produced by laboratory activities, such as contaminated gloves or pipette tips. Solid radioactive waste is packaged to be stored temporarily onsite in a designated cell in the research bay. Appropriate disposal is organized with the licensing status of the material, its chemical form, and its radioactivity (or lack thereof) defined at the time of disposal. Solid radioactive wastes also include absorbing media such as off-gas charcoal and air filters. The frequency and assaying procedures for these regularly created wastes will be addressed in waste disposal procedures in the Operating License application.

Liquid radioactive waste can be generated from utilization of the decontamination showers and eye wash. If used to decontaminate personnel, the liquid is assumed to be radioactive and treated accordingly. Additionally, liquid radioactive waste generated from radiochemical operations can be a mixed chemical and radioactive hazard. Mixed wastes will be stored in a manner appropriate for the chemical and radiological hazards they pose, and solidified if chemical form permits.

Radioactive gasses produced as fission or decay products are anticipated to be released during normal operations as effluent. Details about effluent releases are in Section 11.1.5.

11.2.3 Release of Radioactive Waste

Solid and liquid radioactive wastes are assayed prior to disposal or release to help ensure the waste is released appropriately with regard to local, state, and federal regulations. Gaseous radioactivity is expected to almost entirely be emitted as effluent.

A more thorough description of radioactive wastes and effluents, release points, and monitoring systems will be provided with the Operating License application.

11.2.4 Estimated Quantities of Waste Generation

The primary operational waste is the 100mg salt samples taken for monitoring the primary fuel salt chemistry. For each 100 mg salt sample, it is estimated that an additional 25 mL of liquid waste is generated. These wastes, based on a well-mixed model of the reactor, contain ~25% U by weight but are well below GTCC limits when stabilized for disposal, even for accumulations of massively conservative quantities of samples. Long-term storage of wastes takes place in the waste pit in the research bay. Based on a proposed once-a-month sampling run of 5 samples, it is estimated that 4.5 liters of liquid waste is generated per year from the salt sampling. The specific periodicity of salt sampling is to be determined in the Operating License application.

Wastes from academic and research endeavors involving licensed radioactive material tend to be irregular in form and in nuclide composition. It is estimated from radiochemical experiment experience from the University of Texas at Austin's medical isotopes research that a radiochemical research program will generate about 16 gallons of solid waste per year in the form of gloves, coats, absorbent pads, pipette tips, and other lab wastes along with an estimated 4.8 liters of liquid waste per year. Considerations for waste generation will be made as part of experimental design procedures.

11.3 Respiratory Protection Program

The use of respirators or other respiratory protection for the prevention or restriction of intake of radionuclides is not expected at the MSRR facility. Therefore, the development of a respiratory protection program is not required. However, airborne radioactivity and inhalation doses shall still be kept ALARA by use of engineering and administrative

controls. Engineered protection is afforded by the reactor enclosure and the reactor bay HVAC system, while administrative controls will be provided via training to exit the bay during radiation monitor alarms.

Chapter 12
Conduct of Operations

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 12 CONDUCT OF OPERATIONS 12-1

12.1 Organization 12-1

12.1.1 Structure 12-1

12.1.2 Responsibility 12-1

12.1.3 Staffing 12-2

12.1.4 Selection and Training of Personnel 12-3

12.1.5 Radiation Safety 12-3

12.2 Review and Audit Activities 12-4

12.2.1 Composition and Qualification 12-4

12.2.2 Charter and Rules 12-4

12.2.3 Review/Audit Function 12-4

12.2.4 Audit Function 12-5

12.3 Procedures 12-5

12.3.1 Experiment Review and Approval 12-5

12.4 Required Actions 12-6

12.5 Reports 12-6

12.6 Records 12-6

12.7 Emergency Planning 12-6

12.8 Security Planning 12-6

12.9 Quality Assurance 12-7

12.10 Reactor Operator Training and Requalification 12-7

12.11 Startup Plan 12-7

12.12 Material Control and Accounting Program 12-7

12.13 References 12-7

Appendix 12A ACU Research Reactor Facility Preliminary Emergency Plan . . . 12A-1

LIST OF TABLES

Table 12A-1 Definitions 12-2
Table 12A-2 Succession Plan for Emergency Director and Radiation Safety Officer 12-6

LIST OF FIGURES

Figure 12.1-1 Abilene Christian University Research Reactor Organization 12-4
Figure 12A-1 MSRR Staffing as Shown in Section 12.1 of the PSAR. 12-3
Figure 12A-2 MSRR Site Layout 12-4

CHAPTER 12 CONDUCT OF OPERATIONS

This chapter describes conduct of operations for the Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) facility. It includes administrative aspects of the facility, the emergency plan, the security plan, the Quality Assurance Program, the reactor operator requalification plan, and the startup plan. The administrative aspects of facility operations are organization, review and audit activities, organization aspects of radiation safety, facility procedures, required actions in case of license or technical specification violations, or certain events or observations, reporting requirements, and recordkeeping.

12.1 Organization

This section describes the organizational structure, functional responsibilities, and levels of authority for facility operation. The organizational aspects of the radiation protection (RP) program, staffing, and selection and training of personnel are also discussed.

12.1.1 Structure

The organizational structure for facility operations is shown in Figure 12.1-1.

12.1.2 Responsibility

All functional positions in the organizational structure have responsibility for safe operation of the reactor facility, protection of the health and safety of the public and workers at the facility, and protection of the environment. Responsibilities for the key functional positions in the organizational structure are described in the following subsections, consistent with Figure 12.1-1. The responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

12.1.2.1 ACU Vice President of Research

The ACU Vice President of Research is the Level 1 organizational head with legal responsibility for holding the Construction Permit and the facility Operating License, is responsible for the overall management of the site, and has the authority to commit university resources to protect workers at the facility, public health and safety, and the environment. The ACU Vice President of Research provides direction to the MSRR Facility Director regarding ACU business and strategic objectives. The MSRR Review and Audit Committee is appointed by and reports to the ACU Vice President of Research. Abilene Christian University's Radiation Safety Office for the MSRR report to the ACU Vice President of Research.

12.1.2.2 MSRR Facility Director

The MSRR Facility Director is Level 2 management with responsibility for compliance with the Construction Permit, Operating License, and overall management and leadership of the reactor facility. The MSRR Facility Director provides direction to the senior reactor operators regarding plant business, and

facility strategic testing and performance objectives. The MSRR Facility Director is advised by the MSRR Review and Audit Committee and ACU's Radiation Safety Office.

12.1.2.3 Senior Reactor Operators

Senior reactor operators are Level 3 individuals licensed by the U.S. Nuclear Regulatory Commission (NRC) to direct the activities of reactor operators and are responsible for conforming to applicable rules, regulations, license requirements, and procedures for operation of the MSRR facility, per 10 CFR 50.54. Senior reactor operators are responsible for safe and efficient operation of the MSRR facility, and for maintaining their qualification status. The senior reactor operator provides oversight and direction of the operating staff and is advised by the Radiation Safety Office. Senior reactor operators are responsible for meeting the requirements of the security plan and the emergency plan.

12.1.2.4 Operating Staff

Operating staff consists of NRC-licensed reactor operators, reactor operations trainees, and other non-licensed personnel who assist in the operations of the MSRR. Operating staff are responsible for conforming to applicable rules, regulations, license requirements, and operating procedures for the MSRR facility. Operating staff are responsible for safe and efficient operation of the MSRR facility. Reactor operators are responsible for maintaining NRC-licensed Reactor Operator status.

12.1.2.5 Radiation Safety

ACU's Radiation Safety Office supports the ACU Vice President of Research, the MSRR Facility Director, and the senior reactor operators, as needed, and has a direct reporting line to the ACU Vice President of Research. The Radiation Safety Office is responsible for establishing and implementing the RP program and the as low as is reasonably achievable (ALARA) program, for monitoring worker doses, and for calibrating the health physics instrumentation. The Radiation Safety Office staff has the authority to terminate unsafe activities.

12.1.2.6 MSRR Review and Audit Committee

A method for independent review and audit of safety aspects of MSRR facility operations is established to advise management, consistent with 10 CFR 50.34(b)(6)(ii). These two functions will be performed by the MSRR Review and Audit Committee.

12.1.3 Staffing

Sufficient personnel and materials resources will be provided to safely conduct facility operations. Minimum staffing levels when the reactor is not secured, required contact lists, and events requiring the presence of a senior reactor operator at the facility will

meet the requirements of the regulations and are consistent with ANSI/ANS 15.1-2007 (R2018) [Reference 12.13-1]. Details will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(i).

12.1.4 Selection and Training of Personnel

An indoctrination and training program will be implemented and maintained for personnel performing or managing facility operation activities. The standard ANSI/ANS 15.4-2016 (R2021) [Reference 12.13-2] is used in the selection and training of personnel as applicable and will be described in the application for an Operating License. Records of personnel training and qualification are maintained.

A description of the training program and the required minimum qualifications for facility staff will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(i).

The licensed reactor operator and senior rector operator training program, including the requalification training program, is addressed in Section 12.10.

12.1.4.1 Qualifications

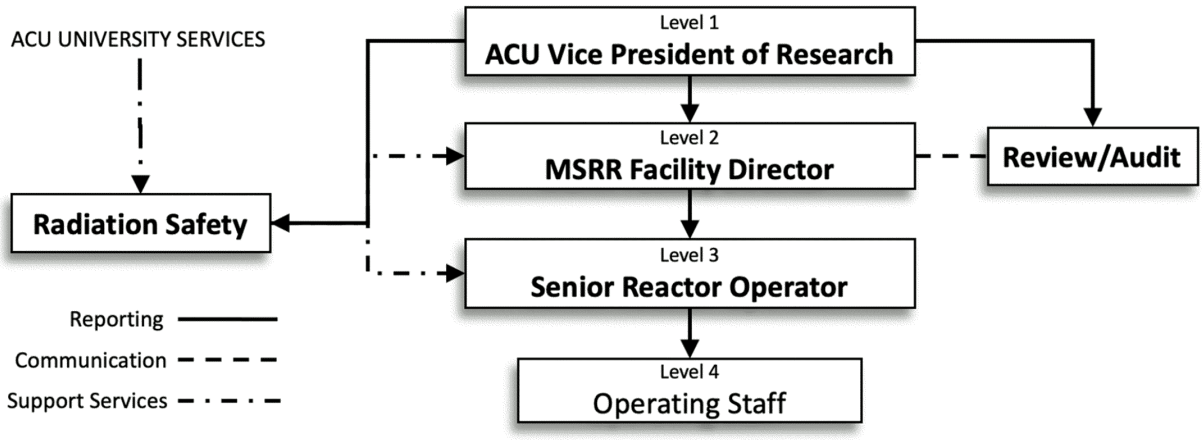
Personnel associated with the MSRR facility will have a combination of academic training, experience, and skills commensurate with the responsibility to provide reasonable assurance that decisions and actions during all normal and abnormal conditions are such that the facility is operated in a safe manner. Details will be provided in the application for an Operating License.

12.1.5 Radiation Safety

Sufficient resources in terms of staffing and equipment are provided by ACU's Radiation Safety Office to implement an effective RP program, consistent with guidance provided in ANSI/ANS 15.11-2016 [Reference 12.13-3]. Further details related to the authority of the RP program staff with respect to facility operations will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(i).

The RP program is described in Section 11.1.2.

Figure 12.1-1 Abilene Christian University Research Reactor Organization



12.2 Review and Audit Activities

The ACU Vice President of Research establishes the MSRR Review and Audit Committee and ensures the appropriate technical expertise is available for review and audit activities. Committee activities are summarized and reported to the ACU Vice President of Research and to the MSRR Facility Director. The details of review and audit activities, who holds the approval authority, how it communicates and interacts with facility and University management will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(ii).

12.2.1 Composition and Qualification

The composition and qualifications of the MSRR Review and Audit Committee will be consistent with ANSI/ANS 15.1-2007 (R2018) and details provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(ii).

12.2.2 Charter and Rules

The MSRR Review and Audit Committee charter and rules will include provisions for meeting frequency, quorums, use of subgroups, and minutes, consistent with ANSI/ANS 15.1-2007 (R2018), and will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(ii).

12.2.3 Review/Audit Function

The items that must be reviewed by the MSRR Review and Audit Committee and the rules for distribution of reports or minutes will be consistent with ANSI/ANS 15.1-2007 (R2018) and provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(ii).

12.2.4 Audit Function

The items that must be audited by the MSRR Review and Audit Committee and rules for distribution of audit findings will be consistent with ANSI/ANS 15.1-2007 (R2018) and provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(ii).

12.3 Procedures

Operating procedures provide appropriate direction to ensure the facility is operated within the design basis and technical specifications limits. Activities affecting safety are performed in accordance with approved implementing procedures. The level of detail in a procedure is dependent on the complexity of the task and considers the experience, education, and training of the users and the consequences of errors. Expectations for the use of procedures are documented and communicated to facility personnel.

Technical specifications require procedures for the following topics, consistent with Section 6.4 of ANSI/ANS 15.1-2007 (R2018). Written procedures will be prepared, reviewed, approved, and followed for:

- Startup, operation, and shutdown of the reactor
- Fuel loading, unloading, and movement within the reactor
- Maintenance of major system components that may have an effect on reactor safety
- Surveillance checks, calibrations, and inspections required by technical specifications or that may have an effect on reactor safety
- Personnel radiation protection, consistent with applicable regulations or guidelines. The procedures shall include management commitment and programs to maintain exposure and releases as low as reasonably achievable in accordance with the guidelines of ANSI/ANS 15.11-2016 [Reference 12.13-3]
- Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity
- Implementation of required plans (e.g., emergency, security)
- Use, receipt, and transfer of by-product material

A description of the facility procedures, including the review, approval, and change process, will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(vi).

12.3.1 Experiment Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures. The following provisions shall be stated:

1. all new experiments or class of experiments shall be reviewed by the MSRR Review and Audit committee and approved in writing by Level 2 or designated alternates prior to initiation;

2. Substantive changes to previously approved experiments shall be made only after review by the MSRR Review and Audit committee and approved in writing by Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

12.4 Required Actions

Technical specifications will specify actions be taken when certain events occur. These are actions to be taken in case of a safety limit violation, release of radioactivity from the site above allowed limits, and other events listed in ANSI/ANS-15.1-2007 (R2018). Details of required actions will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(vi).

12.5 Reports

Technical specifications will specify the reporting requirements for the facility. These include annual routine operating reports, special reports for violations of safety limits, release of radioactivity from the site above allowed limits, and other events listed in ANSI/ANS-15.1-2007 (R2018). In addition, reports are made concerning permanent changes in organization involving level 1 or 2 personnel and significant changes in the transient or accident analysis as described in the safety analysis report. Details of reports will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(vi).

12.6 Records

The technical specifications specify records required to be maintained, where and how they are maintained, and their length of retention. The records are grouped by retention period: 1) a period of at least five years (or for the life of the component involved if less than five years), and 2) the lifetime of the reactor facility. There are also record retention requirements for training records of NRC-licensed reactor operators and senior reactor operators. Technical specifications are described in Chapter 14 and will be provided with the Operating License application, consistent with 10 CFR 50.34(b)(6)(vi).

12.7 Emergency Planning

In accordance with 10 CFR 50.34(a)(10), the specific information required in a preliminary safety analysis report by 10 CFR Part 50, Appendix E.II is provided in Appendix 12A of this chapter. The emergency plan will be updated with the Operating License application, consistent with the requirements in 10 CFR 50.34(b)(6)(v) applicable to a research reactor. The emergency plan will meet 10 CFR 50.54(q) as applicable to a research reactor with consideration of guidance provided in ANSI/ANS 15.16-2015 (R2020) [Reference 12.13-4]; Regulatory Guide (RG) 2.6, Revision 2; and NUREG-0849.

12.8 Security Planning

A description of the security plan for the facility will be provided with the Operating License application, consistent with 10 CFR 50.34(c) and will consider the guidance provided in RG 5.59, Revision 1. Security under the operating license meets 10 CFR 50.54(p) as applicable for a research reactor facility with Category II material.

Development of a security plan requires an approved safeguards information protection plan. The ACU protection plan description has been submitted to the NRC for approval and, once approved, will be implemented to protect material designated as safeguards information- modified handling used in the development of the ACU Molten Salt Research Reactor Security Plan.

12.9 Quality Assurance

The Quality Assurance Program Description for the design and construction of the MSRR is based on ANSI/ANS 15.8-1995 (R2018) [Reference 12.13-5] and considers the guidance from RG 2.5, Revision 1. The ACU Quality Assurance Program Description was submitted as a topical report and approved by the NRC [Reference 12.13-6].

12.10 Reactor Operator Training and Requalification

The NRC-licensed reactor operator and senior reactor operator training and requalification plan is developed and implemented in accordance with 10 CFR Part 55 as it pertains to research reactor or test reactor facilities. The operator training and requalification plan will be provided with the Operating License application, consistent with the requirements in 10 CFR 50.34(b)(8) and 10 CFR 50.54.

12.11 Startup Plan

The startup plan will be provided with the Operating License application, consistent with the requirements in 10 CFR 50.34(b)(6)(iii).

12.12 Material Control and Accounting Program

The Material Control and Accounting program will be provided with the Operating License application, consistent with the requirements in 10 CFR Part 74. We anticipate using guidance provided in NUREG-2159, Revision 1 and will engage the NRC for specific guidance for preparing information in the application for an Operating License.

12.13 References

12.13-1 American National Standards Institute/American Nuclear Society, The Development of Technical Specifications for Research Reactors, ANSI/ANS 15.1, 2007 (R2018), New York, NY.

12.13-2 American National Standards Institute/American Nuclear Society, American National Standard for the Selection and Training of Personnel for Research Reactors, ANSI/ANS 15.4-2016 (R2021), New York, NY.

12.13-3 American National Standards Institute/American Nuclear Society, Radiation Protection at Research Reactor Facilities, ANS/ANSI 15.11, 2016, New York, NY.

12.13-4 American National Standards Institute/American Nuclear Society, Emergency Planning for Research Reactors, ANSI/ANS 15.16, 2015 (R2020), New York, NY.

12.13-5 American National Standards Institute/American Nuclear Society, Quality Assurance Program Requirements for Research Reactors, ANSI/ANS 15.8, 1995 (R2018), New York, NY.

12.13-6 Abilene Christian University Quality Assurance Program Description Topical Report-Accepted Version, ML22293B802, October 19, 2022.

APPENDIX 12A ACU RESEARCH REACTOR FACILITY PRELIMINARY EMERGENCY PLAN

This preliminary Emergency Plan provides a discussion of the plans for coping with emergencies related to the Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) facility, as is required by 10 CFR 50.34(a). As per Title 10 of Code of Federal Regulations, Part 50 (10 CFR Part 50), APPENDIX E to 10 CFR PART 50, "*Emergency Planning and Preparedness for Production and Utilization Facilities*," these plans are described generally here in support of a Construction Permit application. The Emergency Plan to be submitted with the Operating License application shall contain sufficient information to ensure the compatibility of proposed emergency plans for the onsite area, which includes an Emergency Planning Zone (EPZ), with facility design features, site layout, and site location with respect to access routes and land use. The following items shall be discussed in the Emergency Plan:

A. Onsite and offsite organizations for coping with emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations.

B. Contacts and arrangements made and documented with local, State, and Federal governmental agencies with responsibility for coping with emergencies, including identification of the principal agencies.

C. Protective measures to be taken within the site boundary and within each EPZ to protect health and safety in the event of an accident; procedures by which these measures are to be carried out (e.g., in the case of an evacuation, who authorizes the evacuation, how the public is to be notified and instructed, how the evacuation is to be carried out); and the expected response of offsite agencies in the event of an emergency.

D. Features of the facility to be provided for onsite emergency first aid and decontamination and for emergency transportation of onsite individuals to offsite treatment facilities.

E. Provisions to be made for emergency treatment at offsite facilities of individuals injured as a result of licensed activities.

F. Provisions for a training program for employees of the licensee, including those who are assigned specific authority and responsibility in the event of an emergency, and for other persons who are not employees of the licensee but whose assistance may be needed in the event of a radiological emergency.

G. A preliminary analysis that projects the time and means to be employed in the notification of State and local governments, NRC, and the public in the event of an emergency.

This preliminary Emergency Plan closely follows the guidance provided in the RG 2.6 endorsed ANSI/ANS 15.16-2015, "*Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities*," [Reference 12A-1] and informed by NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors" [Reference 12A-2].

12A.1 Introduction

Appendix E to 10 CFR Part 50, "*Emergency Planning and Preparedness for Production and Utilization Facilities*," establishes requirements for emergency plans to attain an acceptable state of emergency preparedness and to provide reasonable assurance that protective measures can and will be taken to protect the health and

safety of workers and the public. This appendix provides the emergency planning related information required in Preliminary Safety Analysis Reports by 10 CFR 50.34(a)(10) and 10 CFR 50 Appendix E.II (Appendix E.II).

12A.1.1 Reactor Facility Description

The reactor facility is located on a site on the southeast corner of the main Abilene Christian University (ACU) campus. The Molten Salt Research Reactor (MSRR) facility, located within the site, will occupy a fraction of the ACU Science and Engineering Research Center (SERC). The SERC is multi-purpose construction that supports the broad range of research being carried out by the Nuclear Energy eXperimental Testing (NEXT) Laboratory. A general description of the reactor facility is provided in Chapter 1 of the ACU Preliminary Safety Analysis Report.

12A.1.2 Definitions

Table 12A-1 Definitions

Term	Definition
Shall, Should, and May	The word “shall” is used to denote a requirement; the word "should" to denote a recommendation; and the word “may” to denote permission, neither a requirement nor a recommendation.
ACU	Abilene Christian University
Committed Dose Equivalents (CDE)	As defined in Title 10, Section 20.1003, of the *Code of Federal Regulations* (10 CFR 20.1003), the CDE (HT,50) is the dose to some specific organ or tissue of reference (T) that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.
Committed Effective Dose Equivalent (CEDE)	As defined in Title 10, Section 20.1003, of the *Code of Federal Regulations* (10 CFR 20.1003), the CEDE (HE,50) is the sum of the products of the committed dose equivalents for each of the body organs or tissues that are irradiated multiplied by the weighting factors (WT) applicable to each of those organs or tissues (HE,50 = ΣWTHT.50).
Deep Dose Equivalent (DDE)	The external whole-body exposure dose equivalent at a tissue depth of 1 cm (1000 mg/cm2).
Emergency	Any situation which activates the MSRR Emergency Plan.
Emergency Action Level (EAL)	A parameter or criteria used as a basis for emergency classification.
MSRR	The ACU Molten Salt Research Reactor
NEXT Lab	The ACU Nuclear Energy eXperimental Testing Lab
Off-campus	The geographical area that is beyond the main ACU campus boundaries.

Table 12A-1 Definitions (Continued)

Term	Definition
Off-site	The geographical area that is beyond the site boundary and includes on-campus and off-campus areas.
On-campus	The geographical area that is within the main ACU campus boundaries.
On-site	The on-campus geographical area that is within the site boundary.
Recovery actions	The actions taken after the emergency has been terminated to restore the facility to its pre-emergency condition.
Science and Engineering Research Center (SERC)	A two-story building on the campus of ACU that houses the MSRR reactor, NEXT Lab staff and students, their related offices, meeting rooms, and laboratories.
Site boundary	The MSRR site is bounded by the roadways E North 16th St., E North 13th St., Avenue F, and N Judge Ely Blvd.

12A.2 Organization and Responsibilities

12A.2.1 Normal Organizational Structure

The minimum staff required to conduct immediate emergency operations shall be available or maintained by ACU. Staffing is described in Section 12.1 of the Preliminary Safety Analysis Report and shown in Figure 12A-1. Facility administrative procedures shall provide the details of the normal MSRR facility organization, including reporting relationships. The EPZ is within the site boundary and no off-site emergency plan actions are required (see Section 2.1).

Figure 12A-1 MSRR Staffing as Shown in Section 12.1 of the PSAR

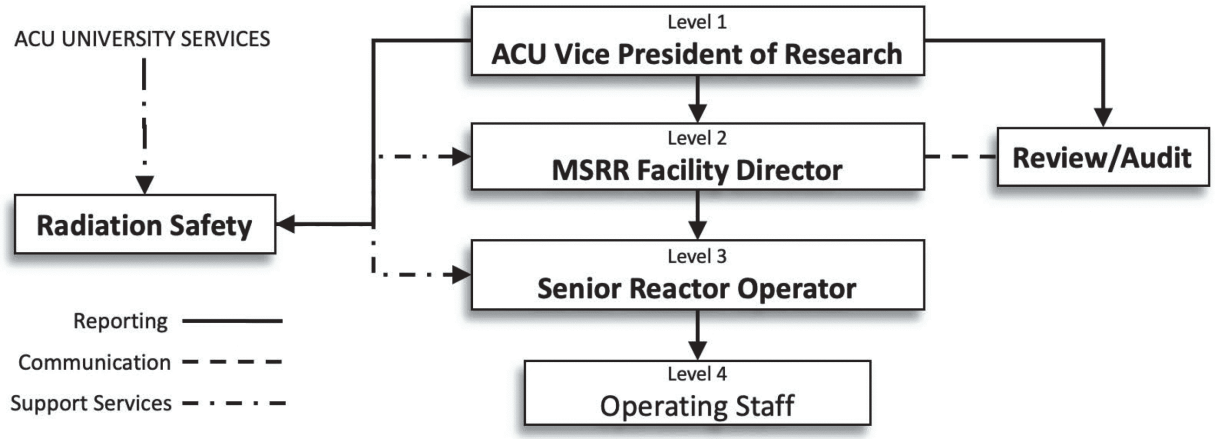
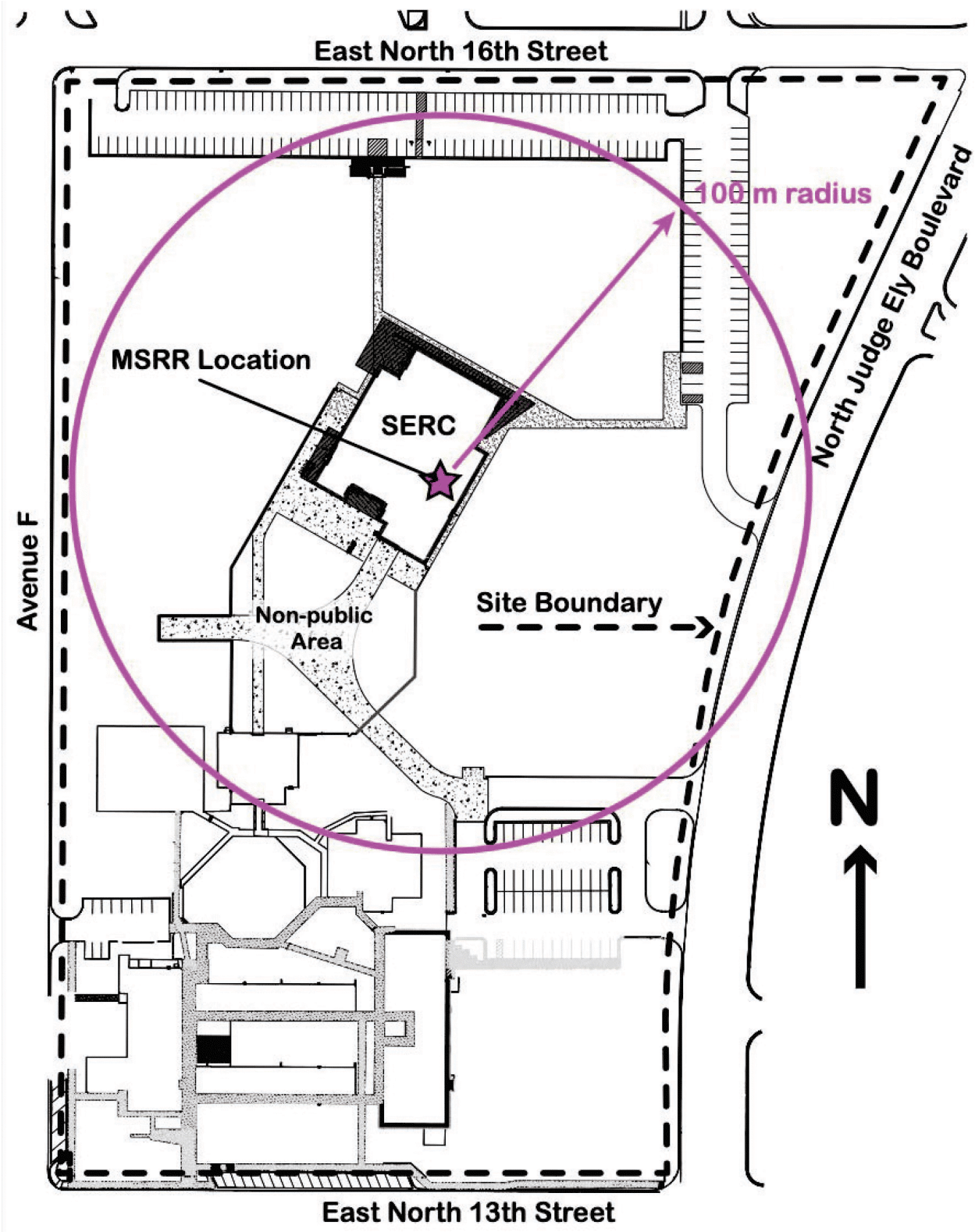


Figure 12A-2 MSRR Site Layout



12A.2.2 Emergency Organization Structure

The regulations in 10 CFR Part 50, Appendix E.II.A requires information regarding onsite and offsite organizations for coping with emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations.

Since the actual personnel to staff the emergency organization will change over time, actual persons are not identified in the plan. The senior emergency plan response qualified individual on-shift at the onset of an emergency, or if no qualified person is at the facility, the senior person on the emergency plan contact listing that can first respond to the facility is responsible for assessing and declaring an emergency, and assuming command and control responsibilities following an emergency declaration. Upon declaration of an emergency, designated members of the normal staff fulfill corresponding roles in responding to the emergency. For example, health physics personnel undertake radiation protection activities; security personnel undertake security activities; engineering personnel focus on facility assessment and technical support for operations; and operations personnel focus on facility operations.

Additional personnel may be designated by ACU management as emergency responders providing special expertise deemed beneficial, but not mandatory, to the planned response. The individuals assigned as emergency response personnel are designated by ACU management based on the technical requirements and training requirements of the position. The primary responsibilities of key emergency response personnel are outlined below. The additional roles and responsibilities for emergency response personnel will be provided in the application for an Operating License.

12A.2.2.1 Emergency Director (ED)

In the event of an emergency, the senior emergency plan response qualified individual on-shift at the onset of an emergency, or if no qualified person is at the facility, the senior person on the emergency plan contact listing that can first respond to the facility will be the Emergency Director (ED).

This is a university research reactor, and the facility is not intended to be staffed by qualified staff around the clock every day of the year. However, qualified staff will be on-site at all times that the reactor is operating or not secure. When the reactor is not operating and secure, qualified staff will be identified to be on-call and be able to be on-site in a timely manner (e.g., within 30 minutes) at any time of the day or any time of the year. In the event that there is an on-site emergency, qualified staff will be notified via phone or text (most likely by emergency personnel) and will respond in a timely manner. Local police and fire department personnel will undergo facility specific training and will participate in routine exercises at the facility. Additional details will be provided in the updated emergency plan within the application for an Operating License.

For emergency response, some individuals will be “on-call”. “On-Call” is defined as being immediately reachable by either phone or text and capable of being on-site within 30 minutes. Individuals “on-call” must be mentally and physically capable of emergency response duties.

Table 12A-2 Succession Plan for Emergency Director and Radiation Safety Officer

	Emergency Director	Radiation Safety Officer
Primary	MSRR Facility Director	Senior Health Physicist
Backup	Senior Reactor Operator	Health Physicist
Secondary Backup	Reactor Operator	Trained Radiation Worker

The responsibilities of the ED are as follows.

- Declare and classify the emergency
- Direct emergency operation and ensure proper implementation of the emergency response plan
- Ensure that any necessary NRC notifications are made in accordance with the applicable requirements
- Authorize emergency workers to incur radiation exposures in excess of normal occupational limits, with the concurrence of the Radiation Safety Officer (RSO), if available. This function cannot be delegated.
- Terminate the emergency and initiate recovery operations
- Assess conditions in the facility after termination of the emergency to determine the proper course of further recovery actions
- Authorize an evacuation of all or part of the site.
- Authorize reentry into the facility (or portion thereof) that required evacuation during the emergency
- Establish and coordinate recovery/re-entry efforts
- Evaluate the causes of the emergency and recommend corrective actions before returning the facility to a normal operating status
- Coordinate emergency response actions with the off-site emergency support services
- Request augmented support as appropriate.

Communication of emergency information to the media is the primary responsibility of the Emergency Director with the assistance of ACU communications staff. The line of succession follows the Emergency Director line of succession.

12A.2.2.2 Radiation Safety Officer

In the event of an emergency, the senior health physics person on-shift or on-call will be responsible for the radiological health physics aspects of the emergency.

The responsibilities for the Radiation Safety Officer are to:

- Evaluate public and personnel doses received during the incident
- Assess subsequent potential doses and recommend protective actions, as appropriate
- Assist the ED and help determine the course of further action

12A.2.2.3 ACU Police Department

ACU Campus Police is responsible for providing and supporting security measures during an emergency, such as crowd and traffic control, and for responding to security incidents at the MSRR facility. If a security incident occurs, ACU Campus Police shall respond and notify the ED if the incident is valid. A transition of the ED to a broader emergency response team may then occur in accordance with the Emergency Plan. The ACU Campus Police perform assessments and response actions to security incidents with necessary assistance by the ED in accordance with the MSRR Security Plan. ACU Campus Police is also the 911 Emergency Communications Center and contacts local off-campus emergency management agencies.

12A.2.3 Off-Campus Support

The regulations in 10 CFR Part 50, Appendix E.II.B requires information regarding contacts and arrangements made and documented with agencies with responsibility for coping with emergencies, including identification of the principal agencies. This section describes the authorities, responsibilities, and support functions of federal, state, county, and local governmental agencies in an emergency situation. The information presented here pertains to any class of emergency.

The arrangements with the Hendrick's Medical Center, Metrocare American Medical Response (AMR), City of Abilene Fire Department will be obtained, documented, and included in the application for an Operating License to ensure a clear understanding of the emergency support responsibilities of each organization. ACU has conducted early discussions with these entities to inform them of the plans to locate the MSRR on the ACU campus and have obtained preliminary documentation from them to indicate their awareness of their potential responsibilities in the Emergency Plan to be submitted as part of the Operating License application. Copies of those documents are included as Attachments to this document.

12A.2.3.1 Hendrick Medical Center

Hendricks Medical Center is the largest hospital in Abilene and is located within 1.4 miles (2.3 km) from the facility. Hendrick Medical Center will be the provider of services and facilities to individuals at the MSRR facility who have been injured, exposed, or contaminated for whom the hospital's services and facilities are necessary or appropriate.

Hendrick Medical Center has standard operating procedures for dealing with radiological emergencies, including contaminated patients. ACU is working with Hendrick Medical Center to develop appropriate response plans for potential incidents at the MSRR facility and a letter documenting the engagement can be found in the Attachments to this document.

12A.2.3.2 Metrocare American Medical Response (AMR)

Metrocare/AMR provides ambulatory response services for the city of Abilene and will provide transport of patients from the MSRR site to off-site medical services. ACU is working with AMR to develop appropriate response plans for potential incidents at the MSRR facility and a letter documenting the engagement can be found in the Attachments to this document.

12A.2.3.3 City of Abilene Fire Department

The City of Abilene Fire Department (AFD) will provide assistance during emergencies involving actual or potential fire, explosions, or injuries. Moreover, response incidents involving hazardous materials will be developed to incorporate emergency response and hazardous/materials/rescue units as may be appropriate. ACU is working with AFD to develop appropriate response plans for potential responses at the MSRR facility and a letter documenting the engagement can be found in the Attachments to this document.

12A.2.3.4 Nuclear Regulatory Commission

Notification procedures (e.g., telephone, electronic messaging, written reports, etc.) will be implemented as required. The response provided by the NRC is described in NUREG-0728, “NRC Incident Response Plan.” The NRC is the Coordinating Agency/Lead Federal Agency for incidents that occur at fixed facilities or activities licensed by the NRC.

12A.2.4 Staffing

ACU shall provide the capability for 24-hour notification to on-campus and off-campus organizations including a primary backup means to accomplish the required communications. On-campus organizations shall be able to call in staff as needed during off-hours (when the reactor is secure and not operating) and for emergency support. Off-campus emergency support organizations maintain their staffing as needed for emergency support. The staffing plan will be provided in the application for an Operating License.

12A.3 Emergency Classification System

This preliminary emergency plan describes several classes of emergency situations covering the spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the emergency organization. To provide for improved communications between ACU and federal, state, and local agencies and organizations, the most severe accidents are standardized in four classes of

emergency conditions that group the accidents according to the severity of off-site radiological consequences. This preliminary emergency plan includes all four standard classes that may be appropriate for dealing with accident consequences for the MSRR facility. Each class of emergency is associated with particular Emergency Action Levels (EALs) and with particular immediate actions to provide appropriate graded response. In order of increasing severity, the four standard emergency classes are described in qualitative terms in the following four subsections.

12A.3.1 Notification of Unusual Events (NOUE)

Man-made events or natural phenomena that are recognized as creating a significant hazard potential that was previously nonexistent have occurred. These events may not result in damage to the reactor but may warrant an immediate shutdown of the reactor or interruption of routine functions. No releases of radioactive material requiring off-site responses are expected. Situations that may lead to this classification include:

A. threats to or breaches of security such as bomb threats or civil disturbances directed towards the facility

B. natural phenomena such as tornadoes in the immediate vicinity of the reactor or earthquakes felt in the facility

C. facility emergencies such as prolonged fires or high off-gas activity

One or more elements of the emergency organization are likely to be activated or notified to increase the state of readiness as warranted by the circumstances.

12A.3.1.1 Alert

Events leading to an alert would be of such radiological significance as to require notification of the emergency organization and its response as appropriate for the specific emergency situation. Under this class, it is unlikely that off-site response would be necessary. Substantial modification of reactor operating status is a highly probable corrective action. Protective evacuations or isolation of certain areas within the operations boundary or within the site boundary may be necessary. Situations that may lead to this classification include:

A. severe failure of primary radionuclide boundary, where that inventory is not as well contained

B. significant releases of radioactive materials as a result of sample extraction failures

12A.3.2 Site Area Emergency

No credible accidents attributable to the MSRR or its operations are postulated that can cause emergency conditions at or beyond the site boundary for this classification. However, the Emergency Director retains the right to declare this class if necessary. Monitoring at the reactor site boundary is conducted to assess

the need for off-site protective actions. Protective measures on-site may be necessary. An armed attack directed towards or occurring at the reactor facility may result in a Site Area Emergency.

12A.3.3 General Emergency

No credible accidents attributable to the MSRR or its operation are postulated that can cause radiological emergency conditions at or beyond the site boundary for this classification. The EPZ shall be within the site boundary and provided in the application for an Operating License. However, the Emergency Director retains the right to declare this class if necessary.

Loss of physical control of the reactor facility may result in a General Emergency.

12A.3.4 Emergency Action Levels (EAL)

Emergency Action Levels (EALs) may be based on airborne Effluent Concentration (EC) fractions at the stack exhaust point and other on-site parameters for which dose rates and radiological effluent releases at the site boundary can be projected. The radiation dose levels specified below are considered adequate for the credible accidents associated with the operation of research reactors and the specified action levels provide reasonable assurance that protective measures associated with the action levels can and will be taken.

The following subsections detail the EALs for the four defined classes of emergencies. In situations where an EAL is not applicable, the Emergency Director retains the right to declare an emergency if warranted by other conditions. EALs specified in the following subsections follow guidance given in ANSI/ANS 15.16-2015 [Reference 12A-1] and NUREG-0849 [Reference 12A-2].

12A.3.5 Emergency Action Levels for Notification of Unusual Events

One or more elements of the emergency organization are likely to be activated or notified to increase the state of readiness as warranted by the circumstances. Although the situation may not have caused damage to the reactor, it may warrant an immediate shutdown of the reactor or interruption of nonessential routine functions.

Situations that may lead to this class include:

- Actual or projected airborne radioactive effluent at the site boundary or beyond for an exposure of 24 hours or less exceeding a 15 mrem DDE or the following CEDE limits:

For noble gases	50 EC x 24 h = 1200 EC·h ≈ 15 mrem CEDE	Equation 12A-1
For radionuclides other than noble gases	100 EC x 24 h = 2400 EC·h ≈ 15 mrem CEDE	Equation 12A-2

- Fire within the reactor facility not extinguished within 15 minutes.

- Credible security threat affecting the reactor facility.
- Receipt of bomb threat affecting the reactor facility.
- Reports or observations of severe natural phenomena affecting the reactor facility.

12A.3.6 Action Levels for an Alert

Situations that lead to this class of protective action include:

- Actual or projected radioactive airborne effluent concentration at the site boundary or beyond for an exposure of 24 hours or less exceeding a 75 mrem DDE or the following CEDE limits:

For noble gases	250 EC x 24 h = 6000 EC·h ≈ 75 mrem CEDE	Equation 12A-3
For radionuclides other than noble gases	500 EC x 24 h = 12,000 EC·h ≈ 75 mrem CEDE	Equation 12A-4

- Radiation levels at site boundary of 0.2 mSv/hour (20 mrem/h) DDE for 1 hour or 1.0 mSv (100 mrem) CDE to the thyroid.
- A security breach of the reactor facility may result in an alert.

12A.3.7 Action Levels for a Site Area Emergency

Situations that lead to this class of protective action include:

- Actual or projected radioactive airborne effluent concentration at the site boundary or beyond for an exposure of 24 hours or less exceeding a 375 mrem DDE or the following CEDE limits:

For noble gases	1250 EC x 24 h = 30,000 EC·h ≈ 375 mrem CEDE	Equation 12A-5
For radionuclides other than noble gases	2500 EC x 24 h = 60,000 EC·h ≈ 375 mrem CEDE	Equation 12A-6

- Actual or projected radiation levels within or at the site boundary of 1.0 mSv/hour (100 mrem/h) DDE for 1 hour or 5.0 mSv (500 mrem) CDE to the thyroid.
- An armed attack directed towards or occurring at the reactor facility may result in a Site Area Emergency.

12A.3.8 Action Levels for a General Emergency

The MSRR Preliminary Safety Analysis Report indicates that no credible accidents attributable to the reactor or its operation are postulated to cause an emergency condition at or beyond the site boundary or beyond in excess of:

- Sustained actual or projected radiation levels of 500 mrem/h DDE
- Actual or projected dose in the plume exposure pathway of 1,000 mrem TEDE or 5,000 mrem CDE to the thyroid

Loss of physical control of the reactor facility may result in a General Emergency and may be declared.

12A.4 Emergency Planning Zones

As part of emergency planning, ACU shall identify any radiological emergencies that may result in exposures exceeding 10 mSv deep dose (1 rem whole body) or 50 mSv (5 rem) CDE to the thyroid and identify an appropriate Emergency Planning Zone (EPZ). The MSRR facility shall be designed such that no credible accidents attributable to the reactor or its operation are postulated to require the EPZ to be extended off-site. The EPZ will be defined in the application for an Operating License.

12A.4.1 Determining an EPZ

The EPZ size depends on the distance at which the protective actions are calculated to be warranted and will be provided in the application for an Operating License.

12A.5 Emergency Response

Emergency response measures shall be identified for each emergency in the ACU Operating License application. These response measures shall be related to the emergency class and action levels that specify what measures shall be implemented.

12A.5.1 Activation of Emergency Organization

The method for activating the emergency organization shall be described in the application for an Operating License. The plan shall specify the location(s) of current notification lists, specific actions to notify and mobilize the emergency organization, and the applicable off-site support organizations for each emergency class.

12A.5.2 Assessment Actions

The methods, systems, and equipment for gathering and processing information and data on which to base decisions to escalate or deescalate emergency response actions shall be described in the ACU Operating License application.

12A.5.3 Corrective Actions

The corrective actions for taking control of the emergency situation, protecting or providing aid to affected personnel, and mitigating the consequences of the emergency shall be described in the Operating License application.

12A.5.4 Protective Actions

The emergency plan submitted with an Operating License application shall describe protective actions appropriate for the emergency class and described. It is anticipated that the emergency plan may include the following:

1. conditions for either partial or complete on-site evacuation, evacuation routes, and primary and alternate assembly areas;
2. methods to assure personnel accountability and the segregation of potentially contaminated personnel;
3. protective measures and exposure guidelines for emergency personnel;
4. provisions for isolation and access control of facility areas to minimize exposures to radiation and the spread of radioactive contamination; and
5. the methods for monitoring radiation dose rates and contamination levels, both on-site and off-site, including provisions for transmitting collected information and data to the element of the emergency organization responsible for accident assessment.

12A.6 Emergency Facilities and Equipment

Descriptions of the emergency facilities, types of equipment, and their locations will be provided in the Operating License application.

12A.6.1 Emergency Support Center

A facility or defined area within a facility shall be designated in the Operating License application to serve as an emergency support center from which emergency control directions shall be given. The support center shall be located to oversee operations effectively and shall be separated from actual activities to function efficiently.

12A.6.2 Assessment Facilities

Monitoring systems and laboratory facilities that are to be used to determine the need to initiate emergency measures as well as those to be used for continuing assessment shall be identified in the Operating License application. These monitoring systems may consist of equipment such as radiological monitors, personnel monitoring, sampling equipment, earthquake sensors, fire and combustion product detectors, and process monitors that provide pertinent facility system or status information.

12A.6.3 First Aid and Medical Facilities

The measures that will be used to provide necessary assistance to persons injured or exposed to radiation will be identified in the Operating License application. The capabilities for decontamination, administering first aid, and transporting injured personnel along with the arrangements for medical treatment shall be described. The following items shall be included:

1. capabilities for decontaminating personnel for their own protection and to prevent or minimize further spread of contamination;
2. first aid training and capabilities of the emergency organization;

3. arrangements for transporting injured personnel who also may be contaminated to medical treatment facilities;
4. arrangements for local hospital and medical services; and
5. assurance that hospital and medical services can provide the required services, and those persons providing them are available, prepared, and qualified to handle radiological emergencies; written agreements with respect to arrangements made for hospital and medical services shall be included.

12A.6.4 Communication Equipment

The systems of emergency communications that will be available to communicate instructions and information both on-site and off-site throughout the course of the emergency shall be identified in the Operating License application. ACU shall establish reliable primary and backup means of communication (e.g., public telephone and radio) that are compatible with local off-site support groups.

12A.6.5 Contingency Planning

Contingency plans in the case an emergency renders any of the above facilities or equipment unusable shall be established and described in the Operating License application. This does not mean fully redundant dedicated backup facilities or equipment, but rather established agreements with alternate facilities from which the needed functions can be performed in the event of an emergency.

12A.7 Recovery

As part of the ACU Operating License application, this element of the emergency plan shall describe the criteria for restoring the reactor facility to a safe status including reentry into the reactor facility or portions of the facility that may have been evacuated because of the accident. The operations to recover from most severe accidents will be complex and depend on the actual conditions at the facility, therefore it is not practicable to plan detailed recovery actions for all conceivable situations.

12A.8 Maintaining Emergency Preparedness

The elements necessary for maintaining an acceptable state of emergency preparedness shall be described in the Operating License application. A description shall be provided of how the effectiveness of the emergency plan will be maintained, including training, review, and update of the emergency plan and associated implementing procedures along with maintenance and inventory of equipment and supplies that would be used in emergencies. Frequent coordination with emergency support organizations shall also be maintained to ensure the necessary training and the efficient use of their capabilities.

12A.8.1 Training and Drills

The following shall be identified or described, as applicable, to demonstrate emergency preparedness in the Operating License application:

1. programs to train and periodically retrain on-site personnel for participation in the emergency plan and to provide specified training to on-site and off-site personnel who have specific emergency assignments;
2. on-site emergency drills to be conducted as action drills with each required emergency measure being executed as realistically as is reasonably possible, including the use of appropriate emergency equipment; and
3. provision for critiques of all drills, including timely evaluation of observer comments, correction of identified deficiencies, and revision of implementing procedures, as needed.

12A.8.2 Plan Review and Update

Reviewing, revising, and updating of the emergency plan shall be described in the Operating License application. This includes specifying the methods to ensure that changes and revisions are reviewed, approved, and distributed to appropriate elements of the emergency organization.

12A.8.3 Equipment Maintenance

Provisions to ensure operational readiness of emergency equipment and supplies, including required maintenance and calibrations, testing, and periodic inventory shall be described in the Operating License application.

12A.9 Citations

12A-1 *Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities*, ANSI/ANS 15.16, 2015.

12A-2 “Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors,” NUREG?0849, Sec. 4.0, U.S. Nuclear Regulatory Commission (Oct. 1983).

ATTACHMENTS:

METROCARE

May 23, 2022

Rusty Kruzelock, Ph.D.
Vice President for Research
Abilene Christian University
1600 Campus Court
Abilene, TX 79601

Dear Dr. Kruzelock,

The Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) site is within the area served by the Metrocare American Medical Response (AMR) unit and will provide transport of patients from the MSRR site to off-site medical services.

This written letter serves to recognize the cooperative effort between ACU and AMR, and acknowledges that ACU is working with AMR to develop appropriate response plans to an emergency incident at the ACU MSRR site.

Patients will be transported to medical facilities as requested by patients or as directed. Emergency response plans for patient care and transport shall be done while following Metrocare Protocol and Metrocare Standard Operating Procedures to include foreseeable incidents.

Sincerely,

Jarrod Greene NRP-LP, CES
Metrocare/AMR

ABILENE FIRE DEPARTMENT – OFFICE OF THE FIRE CHIEF

May 26, 2022

Rusty Kruzelock, Ph.D.
Vice President for Research
Abilene Christian University
1600 Campus Court
Abilene, TX 79601

Dear Dr. Kruzelock,

This written letter serves to recognize the cooperative effort between Abilene Christian University (ACU) and the City of Abilene Fire Department (AFD). This letter acknowledges that ACU is working with the AFD to develop appropriate response plans to an emergency alert at the ACU Molten Salt Research Reactor (MSRR) facility.

Future AFD response will be contingent upon the type of emergency and the information provided by the caller to the AFD. ACU is working to provide AFD responders with knowledge and understanding of MSRR facility operations to assist in this effort.

Moreover, response to incidents involving hazardous materials will be developed to incorporate emergency response, including nearby fire apparatus and hazardous/materials/rescue units as may be appropriate. Further, it is my understanding that ACU plans to provide an MSRR staff member(s) who will also respond to emergencies to help provide assistance. The representative(s) from ACU is expected to be knowledgeable of the facilities, the hazards, and have the ability to assist in coordinating necessary emergency response activities.

Sincerely,

Cande Flores
Fire Chief
Abilene Fire Department

DocuSign Envelope ID: A22F1C11-11B9-4D24-BD32-68EE4B1FF230

HENDRICK HEALTH

28 June 2022

Rusty Kruzelock, Ph.D.
Vice President for Research
Abilene Christian University
1600 Campus Court
Abilene, TX 79601

Dear Dr. Kruzelock,

The Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) site is within the area served by Hendrick Medical Center.

This written letter serves to recognize the cooperative effort between ACU and Hendrick Medical Center and acknowledges that ACU is working with Hendrick Medical Center to develop appropriate response plans to an emergency incident at the ACU MSRR facility.

In accordance with its capacity, licensure, and then-current capabilities, Hendrick Medical Center will be the provider of services and facilities to individuals at the MSRR facility who have been injured, exposed, or contaminated for whom the hospital's services and facilities are necessary or appropriate.

Sincerely,

DocuSigned by:
C21AE5BE42274FD...
Jeremy Walker
Vice President & CFO,

DS
Reviewed and Approved by
Lynette Tatum, Esq.
Associate Corporate Counsel
Hendrick Health

300 Pine Street • Abilene, Texas 79601-2432 • 325-670-2000 • hendrickhealth.org

Chapter 13
Accident Analyses

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 13 ACCIDENT ANALYSES 13-1

13.1 Accident-Initiating Events and Scenarios 13-2

13.1.1 Maximum Hypothetical Accident 13-6

13.1.2 Reduction in Fuel Salt Inventory from a Barrier Failure 13-16

13.1.3 Increase in Fuel Salt Inventory 13-17

13.1.4 Reduction in Cooling 13-18

13.1.5 Reactivity and Power Distribution Anomalies 13-22

13.1.6 Mishandling or Malfunction of Fuel 13-37

13.1.7 Experiment Malfunction 13-37

13.1.8 External Events 13-37

13.1.9 Mishandling or Malfunction of Equipment 13-38

13.1.10 Loss of Normal Electrical Power 13-39

13.2 Accident Analysis and Determination of Consequences 13-39

13.3 Summary and Conclusions 13-42

13.4 References 13-42

LIST OF TABLES

Table 13.1-1 Fractional Weighting of Fission Power in Each Core Volume and Temperature Reactivity Coefficients 13-4
Table 13.1-2 Maximum Hypothetical Accident Assumptions 13-7
Table 13.1-3 Initial Nuclide Inventories Assumed for the MHA 13-8
Table 13.1-4 Calculated Values of χ/Q 13-12
Table 13.1-5 Summary of Analyzed Reactivity Insertions 13-23
Table 13.2-1 Summary of Accident Scenarios Examined 13-40

LIST OF FIGURES

Figure 13.1-1 Dose Rate Over Time 100 m (328 feet) from the SERC 13-15
Figure 13.1-2 Dose Rate Over Time in Immediate Vicinity of Reactor Building 13-15
Figure 13.1-3 Reactor Loop Mass Flow Rate for Reactor Pump Failure Analysis 13-19
Figure 13.1-4 Pressure from RELAP5-3D Volumes Before and After Pump Component 13-20
Figure 13.1-5 Hot Leg, Cold Leg, and Center Channel Peak Temperatures on Pump Failure 13-21
Figure 13.1-6 Power Response from Complete Loss of Pumping Power 13-21
Figure 13.1-7 Total Reactor Power with Increased Fuel Salt Density as Helium Escapes 13-25
Figure 13.1-8 Hot Leg Temperature Increase Before Return to Temperature Equilibrium 13-26
Figure 13.1-9 Excessive Reactor Cooling Increases Reactor Power 13-27
Figure 13.1-10 Hot Leg Temperature of Reactor Loop During Increased Cooling 13-28
Figure 13.1-11 Reactor Vessel Outlet Temperature During 1 Percent UF_4 Transient Increase 13-29
Figure 13.1-12 Simplified Diagram of Adverse Reactor Vessel Geometry Accident 13-30
Figure 13.1-13 Hot Leg Temperature with Increased Cooling 13-31
Figure 13.1-14 Initial Power Spike Followed by Loss of Criticality as Flow Stops 13-32
Figure 13.1-15 Second Limiting Reactivity Insertion Temperature 13-35
Figure 13.1-16 Second Limiting Reactivity Insertion Power 13-35
Figure 13.1-17 Third Limiting Reactivity Insertion Temperature 13-36
Figure 13.1-18 Third Limiting Reactivity Insertion Power 13-36

CHAPTER 13 ACCIDENT ANALYSES

This chapter provides information and analyses related to the health and safety of the public, facility staff, and the environment. The analyses consider the potential consequences of a diverse array of adverse events and accidents, as well as the capability of the facility to accommodate such disturbances. The health and safety of the workers will be demonstrated in the Operating License application. A chemical hazards analysis will be completed based on the detailed engineering design and provided in the application for an Operating License. Analyzed events include scenarios proposed in the NRC-endorsed guidance document ORNL/TM-2020/1478 [Reference 13.4-1], the NUREG-1537 guidance document, as well as additional accident scenarios identified as needed to produce a thorough and comprehensive set of events. Data has been produced for all events identified to determine, either through bounding cases or scenario-specific modeling, that safety of the public, as measured by reactor system or radiological dose consequences of possible failure of reactor integrity, is maintained both in normal operational conditions and credible accident scenarios. The Molten Salt Research Reactor (MSRR) is passively safe, requiring no operator intervention in the event of facility loss of power, even if this occurs during accident conditions. Such a scenario is bounded by a maximum hypothetical accident (MHA).

The reactor design relies on a functional containment approach, as described in Section 6.2, that mitigates the consequences of a radionuclide barrier failure, and helps ensure compliance with the regulations in Title 10 of the Code of Federal Regulations, Part 20, Section 1301 (10 CFR 20.1301). The accident analysis presented in this chapter bounds potential accident source terms by evaluating the dose consequences of an MHA. The MHA, described in Section 13.1.1, is a hypothetical scenario conservatively defined to bound the potential dose consequences of other fission product release events postulated for the reactor design basis. Containment of radiological material is accomplished through several tiers of barriers throughout the system. The fuel salt is considered to be the first barrier for the majority of radionuclides, as they remain chemically or physically bound within the salt or salt-wetted surfaces. A subset of fission products is gaseous under reactor conditions, is not always entrained in the fuel, and can exit the reactor system if boundary integrity is compromised. The second barrier to gaseous fission product release is comprised of the reactor system and the gas management system (GMS) boundaries. During operation, gaseous fission products are contained within the reactor access vessel (RAV) and the GMS. The reactor enclosure provides a third boundary. The reactor cell, which surrounds the reactor enclosure, acts as a final barrier to the environment.

None of the transient scenarios examined result in failure of any radionuclide barriers, so in addition to the initiating events analyzed through RELAP5-3D [Reference 13.4-2], the MHA assumes non-mechanistic compromise of the primary reactor loop boundary integrity to ensure that failure of the single most important radionuclide barrier, the reactor loop stainless steel boundary, still results in acceptable and safe conditions for the public and facility staff.

An official release version of RELAP5-3D is used for all calculations. A point kinetics model is chosen within RELAP5-3D to represent power production and reactivity feedback. This model is informed primarily by SCALE calculations [Reference 13.4-3], with Monte Carlo N-Particle (MCNP) modeling used for select reactivity insertions [Reference 13.4-4]. Appropriate adjustments to account for flowing fuel and its effects on delayed neutron precursors in RELAP5-3D point kinetics are being made to support the Operating License application. These changes to RELAP5-3D are not included in the data presented here and will undergo further

validation prior to inclusion. In the current version of RELAP5-3D, insertions are added manually to account for delayed neutron effects to model the same behavior. Differences in accuracy are expected to be minimal.

For the purpose of accident scenario limits, an operating temperature limit of 816 degrees Celsius (1500.8 °F) and a pressure limit of 0.5 MegaPascal (MPa) (73 pounds per square inch) were set. The model remains within the set operating limits during all analyzed scenarios.

13.1 Accident-Initiating Events and Scenarios

Relevant accident-initiating events are addressed in the following sections and fall into the following categories:

- MHA (reactor system rupture)
- non-MHA salt spills
- Cooling anomalies
- Fuel handling anomalies
- Positive reactivity insertions
- Surveillance system malfunction
- External events
- Loss of normal electrical power

When analyzing the results of these events, a special focus has been placed on outputs that have a direct impact on reactor safe operation. These are primarily temperature, pressure, power excursions, and a radiologically hazardous material release. Because no transient event analyzed directly causes materials failure, or otherwise causes radionuclide release, a massive salt spill in the reactor enclosure of undetermined cause, followed by an immediate loss of electrical power, is assumed as a bounding case for the MHA. Salt released from the fuel salt storage tanks was also considered as a similar bounding accident but with the same or lower dose exposure (due to additional decay time relative to a spill directly from the fuel loop) and is therefore bounded by the chosen MHA.

The RELAP5-3D calculations that support the transient analysis have been performed using temperature reactivity feedback coefficients produced in a geometrically complete SCALE model of the reactor vessel and entered into the RELAP5-3D point kinetics model. Reactor physics details are in Section 4.5. An MCNP model of the reactor vessel region was used for specific reactivity accident scenarios.

The reactor loop, described in Chapter 5 and shown in Figure 4.1-2, was modeled along with a partial coolant loop that interacts with the heat exchanger of the reactor loop. In a loop-style molten salt reactor, the flowing fuel salt carries delayed neutron precursors out of the reactor. Therefore, reductions in fuel salt flow rate will introduce positive reactivity because of the increase in delayed neutrons remaining in the reactor. A delayed neutron reactivity addition of 0.25 $ was used for loss-of-flow accidents to reflect the effect accurately. The assumption of a 0.25 $ reduction in reactivity due to flow (relative to stagnant salt conditions where 1.0 $ of reactivity is defined as 672 pcm) is conservative at

the nominal fuel loop flow rate of 25 kg/s [55 pounds per second (lb/s)]. A variable delayed neutron fraction is beyond the scope of RELAP5-3D currently, so insertions were added manually to produce the necessary changes in reactivity as pump flow changes. Conservative inputs were used as well to ensure a margin of safety. The most significant conservatism used is the assumption that all delayed neutron precursors decay immediately upon leaving the reactor and are not present upon returning through the reactor loop. A more thorough inclusion of the effects of delayed neutron precursor flow will be provided in the Operating License application.

An MCNP model of the reactor has been used to produce the reactivity insertions that correspond with specific transient events. These reactor kinetic parameters, as described in Table 13.1-5, are implemented as inputs for the point kinetics model in RELAP5-3D. Transient behavior is simulated and informed by these inputs in RELAP5-3D, and key metrics such as temperature, pressure, or fission rate are observed to determine if they are adversely affected to produce unsafe operating conditions.

In the standard operation of a molten salt reactor there are two gas entrainment phenomena that can result in sudden reactivity insertions. During operation the reactor produces various gaseous fission products, notably Xenon-135. When the solubility limit of the salt is reached, bubbles of gaseous fission products, primarily Xenon and Krypton, form bubbles which may either remain entrained in the flowing salt or exit through the headspace of the reactor. To mitigate the problem of sporadic release of fission products from the fuel salt, helium is sparged through the salt to continuously remove the noble gas fission products as they are formed. This sparging process in turn creates a second phenomenon that can affect the reactivity of the core. The entrained sparge gas, which contains both helium and fission product gases, has the effect of lowering the overall density of the circulating fuel.

In the event that fuel circulation ends suddenly, the entrained gases would rise out of the core and cease circulating, creating an increase in density. This density increase results in additional fuel being present in the core and therefore an increase in reactivity. This increase in reactivity must be counteracted by negative reactivity from either the inherent physics of the reactor or operational procedures such as control rod insertion to maintain safe temperatures within the reactor core. In current MSRR RELAP analysis, the temperature reactivity coefficient is the only means utilized for this purpose.

The point reactor kinetics (PRK) model is used in RELAP5-3D to model the feedback behavior of the MSRR in both steady state and transient simulations. The PRK model is informed by reactor physics parameters from SCALE, SERPENT, and MCNP. The temperature reactivity feedback includes separate values for the graphite and fuel. Graphite reactivity feedback is applied to each volume associated with a graphite channel and volumes both inside and outside of graphite channels, including the annulus of fuel outside the core and the upper and lower plenums. A pseudo 2D approach is used to model the core, with six defined rings of channels and an outer annulus of fuel that flows around the graphite core moderator representing the radial gradation of channel behavior. In addition to these volumes, the lower plenum, lower grid plate, and upper plenum are all treated as single volumes with small amounts of fission power associated with each.

The table below shows the proportional weighting of both fission power and reactivity feedback. The weighting of the fission power is identical to the reactivity weighting shown for the fuel volumes in the upper portion of the table, while the weighting for the graphite reactivity is reapportioned from the fission power weighting and normalized to sum to unity. Radial and axial dependance of the fuel temperature reactivity coefficient is therefore accounted for within RELAP by apportioning the reactivity coefficient based on fission power distribution. The fuel temperature reactivity coefficient can be treated as constant in this case because it is primarily a function of the derivative of density with respect to temperature, and density changes are linear in the temperature range of interest.

Table 13.1-1 Fractional Weighting of Fission Power in Each Core Volume and Temperature Reactivity Coefficients

Fuel Volumes	**Center Ring (or 1D value)**	**Ring 2**	**Ring 3**	**Ring 4**	**Ring 5**	**Ring 6**	**Outer Fuel Annulus**	**Fuel Temp Reactivity Coefficient**
Lower plenum	0.00872	n/a	n/a	n/a	n/a	n/a	n/a	-6.69 pcm/K
Grid plate	0.00944	n/a	n/a	n/a	n/a	n/a	n/a	-6.69 pcm/K
Axial Core 1	0.000538	0.001564	0.002361	0.002362	0.002617	0.000848	0.0000506	-6.69 pcm/K
Axial Core 2	0.00328	0.009535	0.014394	0.014401	0.015952	0.005172	0.000309	-6.69 pcm/K
Axial Core 3	0.005567	0.016181	0.024426	0.024438	0.027069	0.008777	0.000524	-6.69 pcm/K
Axial Core 4	0.007079	0.020578	0.031063	0.31079	0.034424	0.011161	0.000666	-6.69 pcm/K
Axial Core 5	0.007608	0.022115	0.033383	0.33421	0.036995	0.011995	0.000715	-6.69 pcm/K
Axial Core 6	0.007608	0.022115	0.033383	0.33421	0.036995	0.011161	0.000715	-6.69 pcm/K
Axial Core 7	0.007079	0.020578	0.031063	0.31079	0.034424	0.0011161	0.000666	-6.69 pcm/K
Axial Core 8	0.005567	0.016181	0.024426	0.024438	0.027069	0.008777	0.000524	-6.69 pcm/K
Axial Core 9	0.00328	0.009535	0.014394	0.014401	0.015952	0.005172	0.000309	-6.69 pcm/K
Axial Core 10	0.000538	0.001564	0.002361	0.002362	0.002617	0.000848	0.0000506	-6.69 pcm/K
Upper plenum	0.05714	n/a	n/a	n/a	n/a	n/a	n/a	-6.69 pcm/K
Graphite Associated Volumes	**Center Ring (or 1D value)**	**Ring 2**	**Ring 3**	**Ring 4**	**Ring 5**	**Ring 6**	**Outer Fuel Annulus**	**Graphite Temp Reactivity Coefficient**
Axial Core 1	0.000584	0.001699	0.002564	0.002565	0.002842	0.000921	n/a	-5.58 pcm/K
Axial Core 2	0.003563	0.010355	0.015632	0.01564	0.017324	0.005617	n/a	-5.58 pcm/K
Axial Core 3	0.006046	0.017573	0.026528	0.026541	0.029399	0.009532	n/a	-5.58 pcm/K
Axial Core 4	0.007688	0.022349	0.033736	0.033753	0.037387	0.012121	n/a	-5.58 pcm/K
Axial Core 5	0.008263	0.024018	0.036256	0.036297	0.040178	0.013027	n/a	-5.58 pcm/K
Axial Core 6	0.008263	0.024018	0.036256	0.036297	0.040178	0.013027	n/a	-5.58 pcm/K
Axial Core 7	0.007688	0.02234	0.033736	0.033753	0.037387	0.012121	n/a	-5.58 pcm/K
Axial Core 8	0.006046	0.017573	0.026528	0.026541	0.029399	0.009532	n/a	-5.58 pcm/K
Axial Core 9	0.003563	0.010355	0.015632	0.01564	0.017324	0.005617	n/a	-5.58 pcm/K
Axial Core 10	0.000584	0.001699	0.00256	0.00256	0.002842	0.000921	n/a	-5.58 pcm/K

In the case of any manual reactivity insertion, such as is performed for the void collapse scenario, an input table of values is given to RELAP of both the reactivity insertions in dollars and the period of time over which the insertion is performed. Between discrete time values, RELAP linearly interpolates the insertion, increasing the insertion linearly

from the previous time step's value to the next time step's value over the defined period. The void coefficient of reactivity described in PSAR Section 4.5.2.2 was calculated using SCALE and Serpent through modifying the fuel salt density. The bounding void coefficient coupled with the hypothetical maximum void coefficient were used to calculate the maximum reactivity insertion from voids as reported in PSAR Section 4.5.4.4. The value used in RELAP is $0.6084 and is assumed to occur over a period of seven seconds as bubbles rise through the core and fail to replenish due to a lack of continued salt circulation and/or gas entrainment. RELAP adjusts the overall fission power according to the PRK equation below, where ρ = (k-1)/k and n represents the overall fission power.

$$\frac{dn}{dt} = \frac{\rho-\beta}{\Lambda}n + \sum_i \lambda_i c_i \qquad \text{Equation 13.1-1}$$

The strongly negative temperature reactivity feedback coefficient allows the reactor to maintain safe temperatures during the analyzed reactivity transients in RELAP. In each positive reactivity insertion, we expect and are able to observe a similar trend based on the modeled physics. The initial reactivity insertion causes a significant spike in core power and temperature, which then results in a subcritical reactor core due to the strong temperature reactivity feedback. This allows the temperature of the reactor to decrease and equilibrate to a new steady state temperature, generally slightly higher than the original steady state temperature based on the size of the insertion. For example, an insertion of 240 pcm in the reactor core should be expected to result in a 19.6 °C increase in steady state temperature. In each case this is what is observed and verified with hand calculations (with small deviations due to weighting factor nuance).

Similarly, based on modeled physics, we expect steady state fission power to be driven primarily by the rate of heat transfer at the heat exchanger. This behavior can be observed and verified in the excessive cooling transient shown in Section 13.1.5.2. In each case we are able to verify quantitatively that the RELAP model behaves as we should expect based on the underlying neutron physics and heat transfer equations.

The model uses several simplifying assumptions in the heat transfer of the transient simulations. 1) An adiabatic boundary condition between the reactor loop and the reactor enclosure. This can be expected to eliminate the oscillatory patterns in temperature and power that may be ordinarily expected on the tail end of a transient reactivity excursion. 2) Conjugate heat transfer between core channels is not modeled. The primary effect of this assumption is slightly larger temperature differences between core channels than would be expected in experimental data, making it a conservative assumption. 3) Core channels are represented as homogeneous “rings” rather than individual channels and therefore some minor differences between channels that are at similar radial distances from the center of the reactor are compressed into an average of the behavior of constituent channels. Where necessary, these assumptions will be refined for the operating license application of the MSRR.

13.1.1 Maximum Hypothetical Accident

The MHA is the loss of reactor system boundary structural integrity with coincidental loss of electric power and relies on the reactor enclosure and reactor cell to control the release of fission products to the environment. Figure 4.1-3 shows the reactor vessel and related reactor loop components located inside the reactor enclosure. The MHA is modeled as a massive relocation of all salt contents from the reactor system into the reactor thermal management system (RTMS). Fission products do not escape the reactor system during normal operation; therefore, breaching the reactor enclosure or cell without the reactor system will not result in migration of fission products outside the system. This accident is non-mechanistic and has no credible mechanism. It is shown in the rest of the accident analysis that reactor system structural integrity is not lost for any conceivable event. The escape of fission products past all established MSRR barriers and their release to the unrestricted environment is the most hazardous radiological accident conceivable at the MSRR facility. However, the MSRR is designed and operated so that a fission product release is not credible. The MHA bounds all credible fission product release accidents and can be used to illustrate the analysis of events and consequences during the accidental release of radioactive material. Further conservative conditions and assumptions are used to demonstrate the public would be protected during this non-credible event. Engineered safety features described in Chapter 6 provide the necessary degree of passive safety to mitigate off-site radiological consequences.

The MHA begins after a year of continuous operations at 1 MWth (maximum licensed power). During the year leading into the MHA, the off-gas system has not been operated, so all potentially gaseous radionuclides are concentrated within the reactor system. The MHA starts with a leakage of fuel salt to the RTMS that surrounds the reactor system, which retains the fuel salt. The relocation of the fuel salt is then conservatively followed immediately by a highly unlikely loss of facility power. Decay heat is removed passively to surrounding structures. Fission product release is controlled by natural forces and the leak tightness of the reactor enclosure and reactor cell to mitigate radiological consequences. Normally, the reactor enclosure would be at a pressure of 12.2 psia or 84 kPa. These and further assumptions are summarized in Table 13.1-2.

Decay heat and the latent heat of the reactor system will heat the reactor enclosure atmosphere because of the loss of active reactor cell cooling. Lack of cooling and the relocation of higher-pressure reactor system gases into the reactor enclosure cause the reactor enclosure internal pressure to rise by approximately 2 psi or 14 kPa. This pressure rise is insufficient to pressurize the reactor enclosure space, but in the MHA, the negative relative pressure is not considered, so the enclosure is modeled as if it were in equilibrium with the atmosphere (further conservatism). The MHA will proceed with the assumption that the noble gases begin to leak from the enclosure to the reactor cell at a rate of 0.01 percent per day, which remains constant for the duration of the MHA. Iodine, and other particulates are assumed to leak at 10% of the rate of the noble gases. The elevated temperatures do not challenge the system structural integrity. To further exacerbate the scenario, the gas inside the reactor enclosure is assumed to contain 100 percent of all possible gaseous radionuclides; none are entrained or re-entrained in the salt, nor do they plate out on any surfaces before or after the MHA. Having received radioactive gas from the enclosure, the reactor cell

leaks at a rate of 1 percent per day into the systems pit, and from the pit into the research bay. In this MHA, no credit is taken for hold-up in the research bay nor system pit, so all gases leaving the reactor cell are modeled as having leaked directly into the atmosphere at ground level.

The nuclides considered in the MHA are based on the empirical chemical experience from the MSRE and the accident analysis performed by the Atomic Energy Commission (AEC) in preparation for the MSRE's runs. From operational experience, the AEC categorized radionuclides into three broad categories: Salt-seekers, Noble Metals, and Noble Gases (ORNL-4865). Salt seekers, such as Cesium and Rubidium, dissolve readily in the salt and are stable within it, and thus do not migrate away from the salt itself. Noble Metals, such as Molybdenum and Ruthenium, do not readily dissolve and form much of the deposits observed on the piping as well as the entrained particulates that flow with the salt (ORNL-TM-3884). Noble gases, such as Xenon and Krypton, readily diffuse out of the salt and migrate into the off-gas system or are entrained as bubbles and foam. Of these three categories, only the noble gases are expected to escape from the reactor system. All noble gases are assumed to escape during the MHA.

Tellurium and Iodine were considered exceptions and granted their own section in ORNL-4865 “Fission Product Behavior in the MSRE.” Tellurium is unique among the noble metals in that it has a vapor pressure of 0.0017 MPa at the 650C operating temperatures of the MSRE, and some uncertainty was present about where the Tellurium was migrating (ORNL-4865). This in turn influences its daughter product Iodine, which is nominally stable in the salt as an iodide ion as evidenced by the tendency of Iodine isotopes with shorter-lived tellurium precursors to behave as salt-seekers. (ORNL-3151) However, due to the tellurium transport phenomena influencing the iodine's distribution, the ORNL report does not classify iodine as a salt-seeker. This analysis conservatively estimates that all of the tellurium and iodine is lost in the event of an accident. Additionally, Bromine was also assumed to be lost due to its similarity to iodine and importance in Cesium production.

Prior to startup of the MSRE, the AEC analyzed a “Maximum Credible Accident,” a water intrusion into a salt spill resulting in radionuclide dispersal. That analysis assumed that only 10% of the iodine and solid particulates would likely escape, along with 100% of the noble gases. Although the MSRR MHA does not include water intrusion causing dispersal, this analysis has opted for 100% escape of any nuclide considered to conservatively envelope accident consequences in all cases.

Table 13.1-2 Maximum Hypothetical Accident Assumptions

Assumption	Quantity	Comments
Pre-MHA operations	1 MW-year with no removal or plating of fission gases	Equilibrium conditions for fission gas inventory
MHA	Breach of reactor system boundary and loss of facility power	Full inventory of fuel and fission gases relocated to reactor thermal management system inside reactor enclosure

Table 13.1-2 Maximum Hypothetical Accident Assumptions (Continued)

Assumption	Quantity	Comments
Source term	100% of Te, I, Xe, Br, Kr, and H-3 are considered free gases (no salt retention or plate out on surfaces before or after the MHA)	Only losses are from radioactive decay
Reactor enclosure leak rate	0.01% per day (noble gas radionuclides) 0.001% per day (all other nuclides)	For all gases
Reactor cell leak rate	1% per day	For all gases
Research bay holdup	None	Not considered when evaluating public dose
Atmospheric conditions	Constant 1 m/s, stability class F winds in a single direction	Includes building wake (conservative assumption)

Table 13.1-3 Initial Nuclide Inventories Assumed for the MHA

Nuclide	Initial Quantity Released to Enclosure (Bq)
H-3	3.92E+13
Te-131	1.59E+15
Te-131m	2.62E+14
I-131	1.77E+15
Xe-131m	1.93E+13
Te-132	2.64E+15
I-132	2.66E+15
I-132m	7.44E+12
Te-133	2.06E+15
Te-133m	2.30E+15
I-133	4.10E+15
I-133m	2.91E+14
Xe-133	3.98E+15
Xe-133m	1.18E+14
Te-134	4.24E+15
I-134	4.78E+15
I-134m	2.30E+14
Te-135	2.03E+15
I-135	3.86E+15
Xe-135	3.83E+15
Xe-135m	7.34E+14
Cs-135	0
Te-137	2.50E+14
I-137	1.87E+15
Xe-137	3.75E+15
Cs-137	0

Table 13.1-3 Initial Nuclide Inventories Assumed for the MHA (Continued)

Nuclide	Initial Quantity Released to Enclosure (Bq)
Br-85	7.83E+14
Kr-85	4.70E+13
Kr-85m	7.86E+14
Br-88	1.05E+15
Kr-88	2.13E+15
Rb-88	0
Br-89	6.57E+14
Kr-89	2.71E+15
Rb-89	0
Sr-89	0
Br-90	3.36E+14
Kr-90	2.81E+15
Rb-90	0
Sr-90	0
Y-90	0
Br-91	1.35E+14
Kr-91	2.01E+15
Rb-91	0
Sr-91	0
Y-91	0
Y-91m	0

The MHA presented is unrealistic in that fission gases are sequestered neither by the fuel salt nor from plating on internal reactor surfaces, either prior to or after the MHA. Also, holdup of tritium in the graphite is not considered, so that all tritium generated is treated as being released. The MHA takes no credit for uncertainties associated with those processes, which represents the most conservative approach to calculating a maximum hypothetical dose to the public. Further, it is assumed the gaseous fission products include 100 percent of each of the nuclides (1) that credibly exist in the gas phase, or that could potentially produce gaseous compounds within the reactor environment, and (2) that could contribute non-negligibly to the overall dose. These nuclides are tellurium (which, while not gaseous, has an appreciable vapor pressure), iodine, xenon, bromine, hydrogen, and krypton in the system.

The MHA event sequence has been analyzed in RELAP5-3D to determine the extent of pressure and temperature changes in the reactor enclosure atmosphere as well as temperatures of structural materials. At no time during the MHA sequence does temperature have a compounding effect on the accident. Modeling the dose consequence of the MHA is broken into three separate tasks: transport from enclosure to environment atmosphere, atmospheric dispersion, and the calculation of resultant dose.

Transport modeling of gaseous or particulate radionuclides was achieved by solving the transient differential equations governing the inventory of radionuclides using the MATLAB symbolic toolbox [Reference 13.4-5]. For some general nuclide k, the differential equation governing the inventory is:

$$\frac{dN_k}{dt} = -N_k(L_k + \lambda_k) + \sum_{i=1}^{i}(Br_i \times N_i \times \lambda_i) + (N_{source}L_{source}) \qquad \text{Equation 13.1-2}$$

where,

N_k = the number of atoms of the isotope of interest in the current compartment,

L_k = the leakage rate from the current compartment to the next (sec^{-1}),

λ_k = the decay constant for the nuclide of interest (sec^{-1}),

Br_i = the branching ratio to the nuclide of interest from parent isotope i,

N_i = the number of atoms of parent isotope i in the current compartment,

λ_i = the decay constant for the parent nuclide being analyzed (sec^{-1}),

N_{source} = the number of atoms in the upstream compartment,

L_{source} = the leak rate of the upstream compartment (sec^{-1}),

and these are bounded by the initial conditions:

100 percent of fuel salt Te, I, Xe and Br, Kr liberate at T_0 in the enclosure, and

$N_0 = 0$ at T_0 for all other compartments or isotopes.

For the purposes of this MHA, Te-I-Xe-Cs mass chains 131, 132, 133, 134, 135, and 137 are included. These mass chains make up 75 percent of the Te-I-Xe-Cs activity for mass chains 121 to 147 upon accident initiation, and 99 percent of the Te-I-Xe-Cs activity after 24 hours. Likewise, Br-Kr-Rb-Sr mass chains 85, 87, 88, 89, 90, and 91 are included. These make up 81 percent of the initial activity of the Br-Kr-Rb-Sr of mass chains 79 to 100 but make up 99 percent of that decay chain's activity within the first 24 hours.

These equations are solved for the enclosure, cell, and the systems pit. From the systems pit inventory, we then calculate the leak rate in Ci/sec for some isotope i by Equation 13.1-3 below:

$$\dot{Q}_i(t) = (N_i^{pit}(t) \times \lambda_i) L_{pit}$$ Equation 13.1-3

where,

$\dot{Q}_i(t)$ = the activity leak rate at time t (in Ci/sec),

$N_i^{pit}(t)$ = the number of atoms of isotope i in the systems pit,

λ_i = the decay constant of isotope i in sec^{-1}, and

L_{pit} = the systems pit leak rate, in sec^{-1}.

The systems pit leak rate is defined to be so large such that there is no meaningful holdup and radionuclides emitted from the reactor cell immediately escape to the atmosphere. Dilution factors were calculated for two distances from the reactor: the first dilution factor was calculated at the property boundary 100 m (328 feet) away using the methodology proposed in Regulatory Guide (RG) 1.145, Revision 1. This methodology calls for a comparison of three different equations, shown below:

$$\frac{\chi^d}{\dot{Q}} = \frac{1}{\overline{U}_{10}\left(\pi\sigma_y\sigma_z + \frac{A}{z}\right)}$$ Equation 13.1-4

$$\frac{\chi}{\dot{Q}} = \frac{1}{\overline{U}_{10}(3\pi\sigma_y\sigma_z)}$$ Equation 13.1-5

$$\frac{\chi}{\dot{Q}} = \frac{1}{\overline{U}_{10}(\pi\Sigma_y\sigma_z)}$$ Equation 13.1-6

where,

$\frac{\chi^d}{\dot{Q}}$ = the relative concentration, in sec/m^3, at distance d [100 m (328 feet)],

π = 3.14159,

$\overline{U}_{10}$ = the wind speed at 10 meters (33 feet) above plant grade in m/sec,

σ_y = the lateral plume spread, in m, as a function of atmospheric stability and distance,

σ_z = the vertical plume spread, in m, as a function of atmospheric stability and distance,

Σ_y = lateral plume spread with meander and building wake effects, in m, as a function of atmospheric stability, windspeed, and distance. For distances under 800 m (2625 feet), where M is determined from Figure 3 of the regulatory guide to be equal to 4, and

A = the smallest vertical-plane cross-sectional area of the reactor building and is equal to 140 m^2 (1604 square feet) for the Science and Engineering Research Center (SERC).

Per the methodology in RG 1.145, Equation 13.1-4 and Equation 13.1-5 are first compared, and the higher of those two values is then compared against Equation 13.1-6. Whichever is the lower value is then taken to be the relative concentration. Table 13.1-4, below, shows the relative concentrations as calculated for stability class F winds.

Table 13.1-4 Calculated Values of χ/Q

Equation Number	(χ/Q (sec/m^3)
Equation 13.1-4	0.10111 (1/98.9)
Equation 13.1-5	0.01153 (1/86.7)
Equation 13.1-6	0.00865 (1/115.6)

Thus, for class F stability at a wind speed of 1 m/s, we calculate a relative concentration of 1/115.6. This, multiplied by the release rate in Ci/second, yields the concentration of the nuclide at 100 meters (328 feet) from the release point (site boundary).

The methodology presented in RG 1.145 is only valid at and beyond 100-meter (328-feet) distances from the release point. Thus, dilution factors in the immediate vicinity of the building are calculated by using the methodology presented in Lamarsh and Baratta [Reference 13.4-6] for evaluating concentrations at a vent from a leaking building. The equation is as follows:

$$\frac{\chi^d}{\dot{Q}} = \frac{1}{D_B} \qquad \text{Equation 13.1-7}$$

where,

D_B= cAv,

$\frac{\chi^d}{\dot{Q}}$ = the relative concentration at distance d (in the building vicinity)(sec/m^3),

c = a constant depending on building geometry, usually conservatively taken at 0.5 [Reference 13.4-6],

A = the cross-sectional area of the building [140 m^2 (1604 square feet)], and

v = the average wind speed over the duration of the release (m/sec).

Thus, within the direct building wake, we calculate a dilution factor of 0.01428, or 1/70 for individuals located directly within the wake of the building.

Dose modeling was then performed utilizing these concentrations, using a combination of EPA Federal Guidance Reports 11 [Reference 13.4-7] and 12 [Reference 13.4-8] dose conversion factors, along with ICRP 23 [Reference 13.4-9] reference man values for inhalation rates [Reference 13.4-10, Reference 13.4-11, and Reference 13.4-12]. Activity concentrations for each nuclide at each distance from the SERC were calculated using Equation 13.1-8, below:

$$X_i^d = \dot{Q}_l(t) \times \frac{X^d}{\dot{Q}} \qquad \text{Equation 13.1-8}$$

where,

X_i^d = the concentration (Ci/m^3) of nuclide i, at distance d (m),

$\dot{Q}_l(t)$ = the release rate (Ci/sec) of nuclide i at time t (sec), and

$\frac{X^d}{\dot{Q}}$ = the relative concentration at distance d (sec/m^3).

Dose rates can then be found for both submersion and inhalation of the air in the plume centerline, using the following equations:

For internal exposure, committed effective dose rates are calculated by:

$$\dot{H}_i^d(t) = X_i^d(t) \times F_i^{inhalation} \times V \qquad \text{Equation 13.1-9}$$

where,

$\dot{H}_i^d(t)$ = the dose rate due to inhalation of nuclide i at distance d, using the concentration of the plume centerline (mrem/hr),

$X_i^d(t)$ = the concentration of nuclide i, at distance d at the plume centerline (Bq/m^3),

$F_i^{inhalation}$ = the inhalation dose conversion factor from FGR 11, converted to (mrem)/(Bq), and

V = is the total inhalatory volume of an individual per hour, averaged between 8 hours of rest, 8 hours of work, and 8 hours of non-working activity, from ICRP 23 (0.958 m^3).

For external exposures, external dose rates were calculated by

$$\dot{H}_i^d(t) = X_i^d(t) \times F_i^{submersion} \times V \qquad \text{Equation 13.1-10}$$

where,

$\dot{H}_i^d(t)$ = the dose rate due to inhalation of nuclide i at distance d, using the concentration of the plume centerline (mrem/hr),

$X_i^d(t)$ = the concentration of nuclide i, at distance d at the plume centerline (Bq/m^3),

$F_i^{submersion}$ = the inhalation dose conversion factor from FGR 11, converted to (mrem)/(Bq), and

These dose rates were added together to form the Total Effective Dose Equivalent (TEDE) dose rate, and the results are shown in Figure 13.1-1 and Figure 13.1-2.

Figure 13.1-1 Dose Rate Over Time 100 m (328 feet) from the SERC

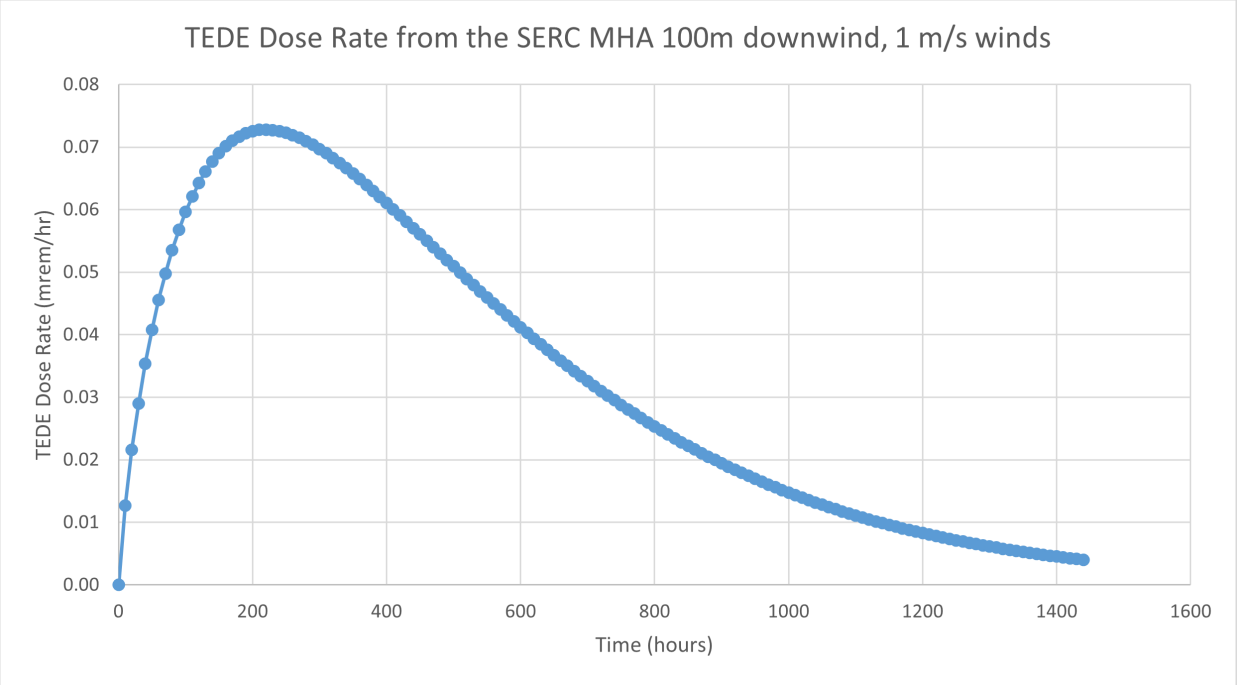
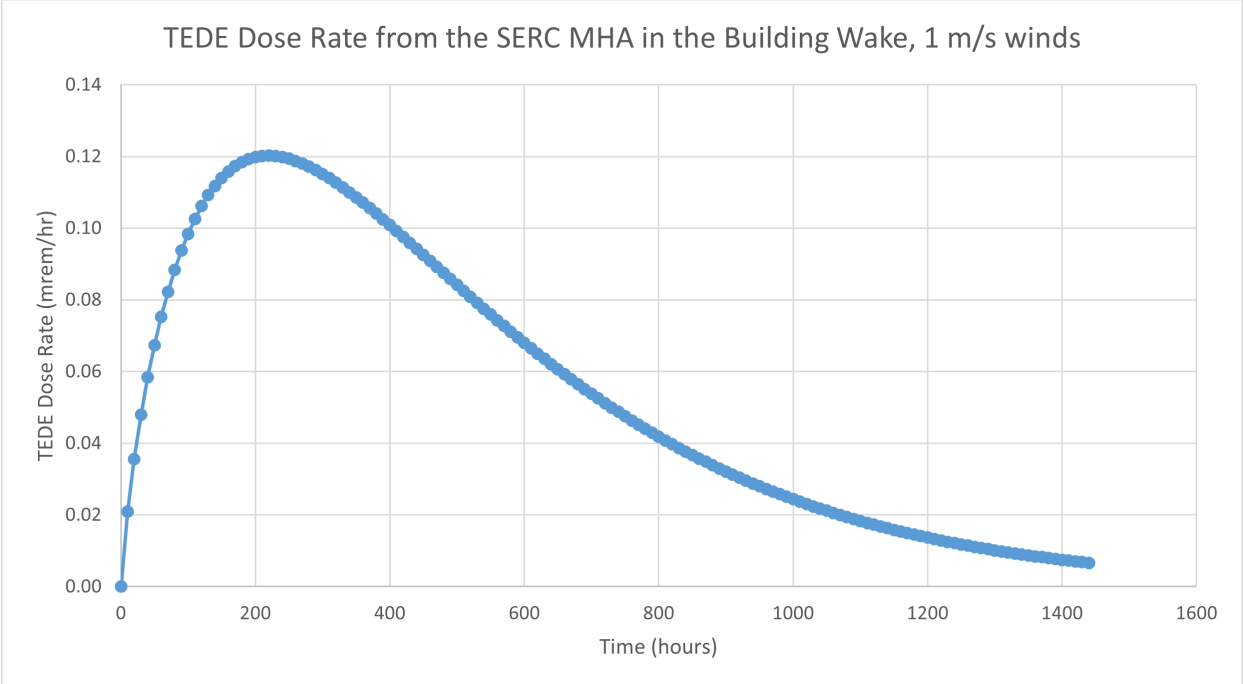


Figure 13.1-2 Dose Rate Over Time in Immediate Vicinity of Reactor Building



Integrating the line of best fit over 60 days (1440 hrs) for the dose rates in Figure 13.1-1 yields a dose of 48.66 mrem for an individual who spends 60 days directly 100 m (238 feet) downwind of the release point. Repeating over the same

duration in the immediate wake of the building, using the curve in Figure 13.1-2 yields a dose of 80.36 mrem through a 60-day duration. This indicates that even in the presence of incredibly conservative assumptions, no member of the public who remains on site can receive more than 100 mrem of dose from the MHA.

13.1.2 Reduction in Fuel Salt Inventory from a Barrier Failure

In each salt-containing system, there are several barriers between radionuclides within the salt and the outside environment. The first barrier is the salt itself. The second barrier is the physical structure enclosing the salt. The third barrier is the enclosure surrounding the system. The fourth barrier is the cell in which the enclosure is located.

The salt chemically binds to the majority of the fission products, maintaining them in solution. Other fission products, primarily represented by noble metals and refractory metals, fall in a second category that is retained within the salt as a solid but does not chemically bind to the salt. A third category of fission products is gaseous, and except for a small concentration that can be soluble within the salt, can eventually leave the salt. This group is primarily represented by isotopes of xenon and krypton, potentially iodide and bromide compounds as well. A more detailed description of fuel chemistry can be found in Section 4.2.1.

Within the reactor systems, three barrier failures that would result in a loss of fuel inventory are possible:

- Fuel salt loss from the piping and components of the reactor system, including the drain tank
- Leakage of fuel salt through the heat exchanger into the coolant loop
- Loss of salt from the fuel storage tanks in the fuel handling system (FHS)

This section shows how radionuclides would be contained in each of these possible barrier failures. All possible leaks are bounded by MHA, described in Section 13.1.1.

In the first scenario, fuel salt egress from the reactor system would result in the presence of fission products in the reactor enclosure as assumed in the MHA.

The second scenario is a tube break in the heat exchanger. Because the fuel and coolant salts are chemically compatible with each other, a tube break or leak in the heat exchanger results in intermixing of some of the fuel and coolant salts. The coolant loop is kept at a higher pressure during operation, through operator management of flow rates and pump pressures of each loop, than the reactor loop to minimize the amount of fission product infiltration into the coolant loop. Any radiation increase detected in the coolant loop cooling system or the coolant salt and heat management enclosure also triggers an immediate shutdown and draining of the loops. Given that the leak rate of the coolant salt and heat management enclosure is designed to the same standard as the reactor enclosure, the radiological consequences of this event are bounded by the MHA. Radiation surveillance would drain both the fuel and coolant loops, isolating all salt in the respective drain tanks,

should an egress of fuel into the coolant loop occur. More detailed information on the radiation monitoring system and the fuel handling system can be found in Sections 11.1.4 and 9.2.

The coolant salt also contains small quantities of radionuclides generated by nuclear interactions and activation. Tritium generated in the fuel salt can diffuse through the heat exchanger and accumulate in the coolant salt. In the event of a breach in the coolant salt piping, the coolant salt and gases present enter the coolant salt and heat management enclosure. Consequently, conditions within this enclosure are monitored to detect such an event. Further, the enclosure is designed to mitigate release of radionuclides to the research bay. The consequence of this accident is bounded by the MHA.

In the last scenario, fuel from the fuel storage tanks is able to exit the tanks and enter the fuel storage enclosure. This is a nearly identical scenario to the MHA but does not involve critical salt and, therefore, whether recently irradiated or unirradiated fuel is involved in such an accident, the concentration of radionuclides is the same or less than calculated for the MHA. The FHS is contained within an enclosure to mitigate the release of radionuclides.

During operation, systems will be in place to prevent flow either from the fuel storage tank into the reactor and coolant loops or from the loops back to the storage tanks. This prevents any fuel loading accidents from occurring due to operator error.

13.1.3 Increase in Fuel Salt Inventory

Excess fuel salt can be present in the drain tank, presenting the possibility of excess fuel salt in the fuel loop.

An increase in fuel salt volume in the fuel loop has no negative effect on either reactor safety or performance. If additional fuel salt inadvertently is moved from the drain tank to the reactor loop, the level in the RAV rises, allowing adequate space for the excess volume to be absorbed within the reactor loop. There is no effect on criticality, thermal hydraulics, or other safety-related systems. Salt level is measured by multiple redundant sensors, as described in Section 7.3.4. Redundant monitoring ensures that excess fuel salt cannot push its way into the gas line above the RAV. Absent significant temperature or pressure shifts, changes to the level in the RAV indicate unintentional addition of excess salt volume. Regardless of other parameters, exceeding the high-level limit results in a high salt level reactor trip. Because of the positioning of the tanks, a loss of power fully mitigates this accident scenario.

Another possible inventory increase accident involves an increase in the concentration of UF_4 as described in Section 4.5. The small quantities of UF_4, kept separately for the purpose of mixing into the non-critical FLiBe-UF_4 mixture on startup, can enter the reactor loop at a higher concentration than anticipated. In this event, positive reactivity is inserted and equilibrium temperatures rise. Control rods are then relied upon to insert negative reactivity to return to the desired steady-state operating temperature. While accidental fuel insertion is unlikely and requires direct action by the operator, and either through operator action or natural temperature

feedbacks, power restabilizes at a temperature that is safe for the materials in the reactor. Additionally, any increase in reactor power above 1 MWth actuates the Reactor Protection System (RPS), shuts down the reactor, and drains the fuel salt to the drain tank. This increase in reactivity is covered in more depth in Section 13.1.5.3.

13.1.4 Reduction in Cooling

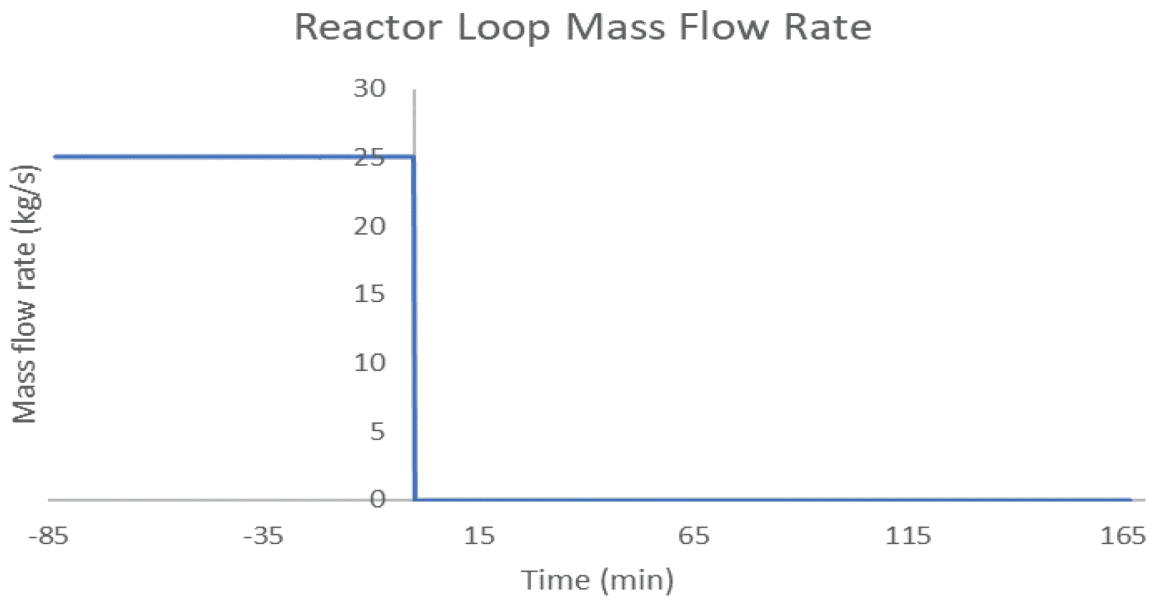
A complete loss of electrical power is considered in Section 13.1.10. Loss of auxiliary heat removal (reactor cell cooling, see Section 9.7.1) is bounded by the complete loss of electrical power.

Reductions in cooling rate can occur for a variety of reasons: electrical power loss, pump malfunction, pipe rupture, full or partial blockage of flow in either the fuel or coolant loops, accidental drain of the coolant salt, or loss of secondary cooling system heat sink. A full drain of the fuel salt from the reactor loop is performed in response to these scenarios, allowing safe, indefinite decay heat removal from the drain tank.

A full loss of reactor loop pumping power results in a slow, continued circulation of fuel salt within the reactor loop that theoretically continues to dissipate decay heat indefinitely; however, this phenomenon is not relied on as a safety system for this analysis and is ignored for conservatism. The transient analyzed here is a complete loss of flow such as might occur in a full obstruction in the reactor loop. The analysis assumes fuel salt remains in the reactor loop, and is not drained to the drain tank. Hence, this scenario represents a bounding case for all kinds of loss of flow events.

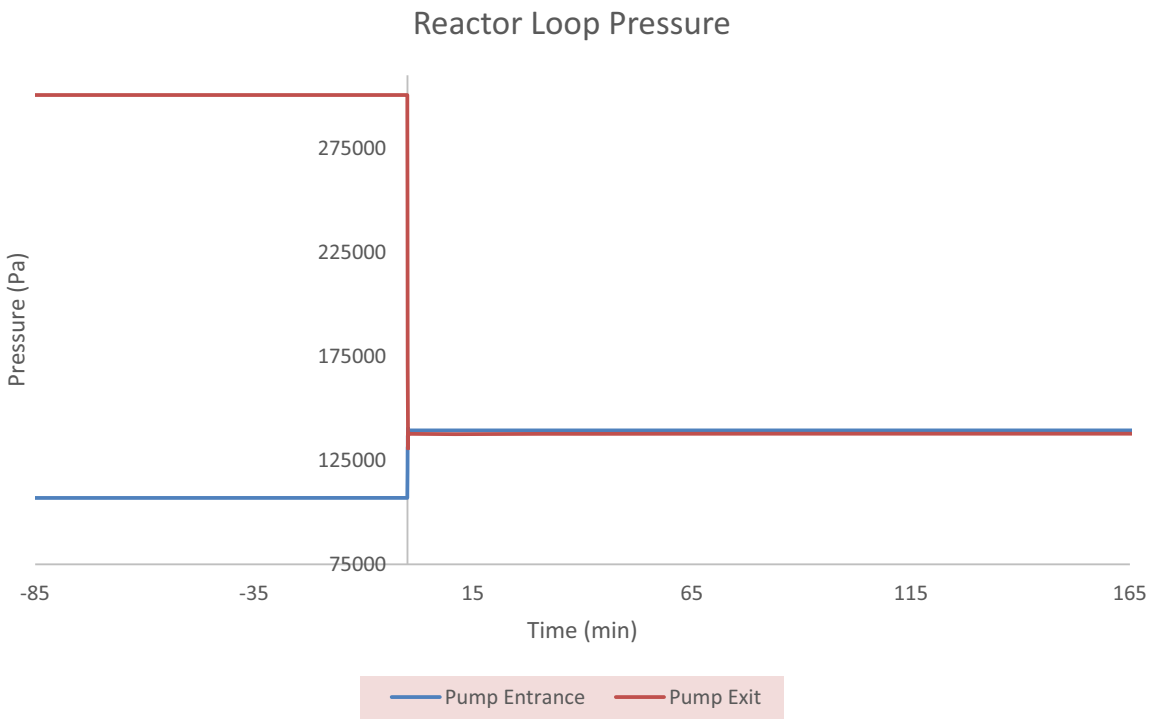
A transient analysis performed in RELAP5-3D shows the effects of a full reactor pump failure, which is shown on the graph below as beginning at time=0. Prior to time=0, the reactor loop flow rate is 25 kg/s (55 lb/s) at 1 MWth of power. At time=0, a pump failure occurs and the fluid flow comes to a stop over 10 seconds (see Figure 13.1-3).

Figure 13.1-3 Reactor Loop Mass Flow Rate for Reactor Pump Failure Analysis



Pressure is calculated to remain under the maximum desired pressure of 0.5 MPa (73 pounds per square inch), as described in Section 4.3, and no significant effects are seen on pressure from power excursions or reactor pump trip. During operation, the minimum and maximum reactor loop pressures are found immediately before and after the reactor pump. No pressure excursions are seen to occur either during reactor pump failure or after reactor pump failure, resultant from temperature excursions. Analyses of the pressures within each volume reveal that no volume pressures deviate during either standard operation or the transient scenario. The reactor loop reverts to pressures associated with relevant hydrostatic forces after reactor pump failure, consistent with elevation in the loop in the absence of reactor pump pressure (see Figure 13.1-4).

Figure 13.1-4 Pressure from RELAP5-3D Volumes Before and After Pump Component



The operating temperature limit is 650 °C (1202 °F). Fuel salt remains within this limit for the duration of this transient, and an automated shutdown via the RPS is not required (see Section 4.2) because of the robustness of the reactivity feedback of the reactor. The maximum temperature seen is in the center channels of the graphite, peaking at 617 degrees Celsius. This temperature remains far under the temperature limits for both the salt and graphite, which are the relevant material limits in this region. The highest temperature achieved within a steel-contacting region is 601 degrees Celsius, which is within the operating range of stainless steel 316 (see Section 4.3). See Figure 13.1-5.

Loss of fuel flow immediately decreases fission power in the MSRR. Strong negative temperature feedback of the reactor causes reactor power to drop from just under 1 MWth to only a few tens of kWth over the course of a few minutes. The remaining power, along with the heat capacity of the salt and insulation, then allows the salt to be maintained in a molten state (see Figure 13.1-6).

Figure 13.1-5 Hot Leg, Cold Leg, and Center Channel Peak Temperatures on Pump Failure

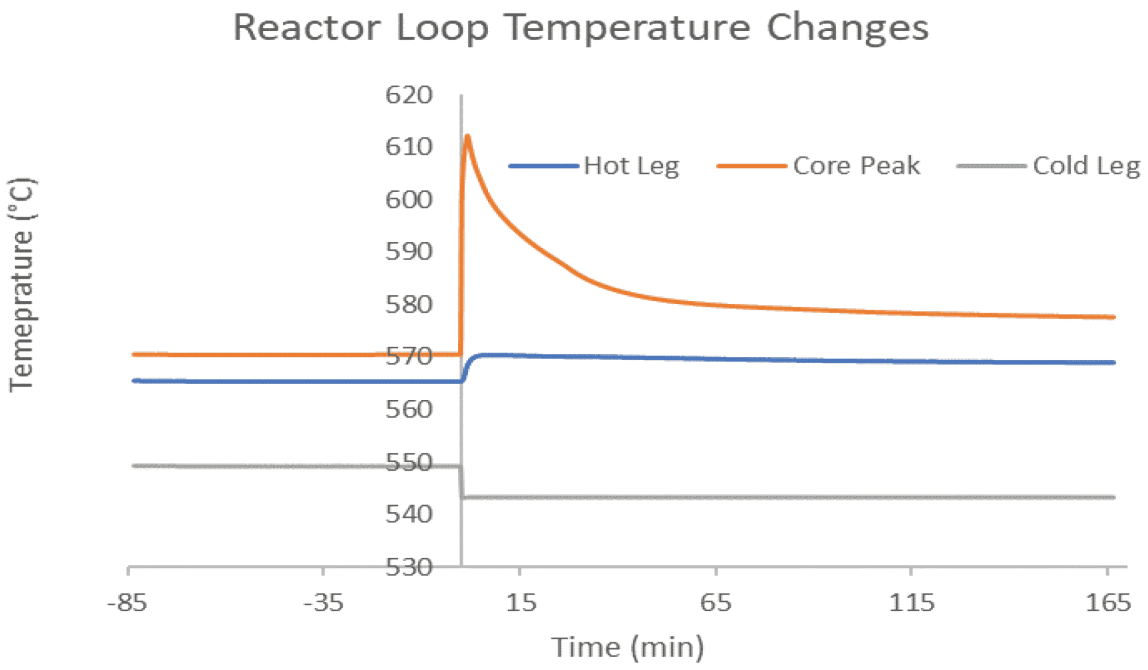
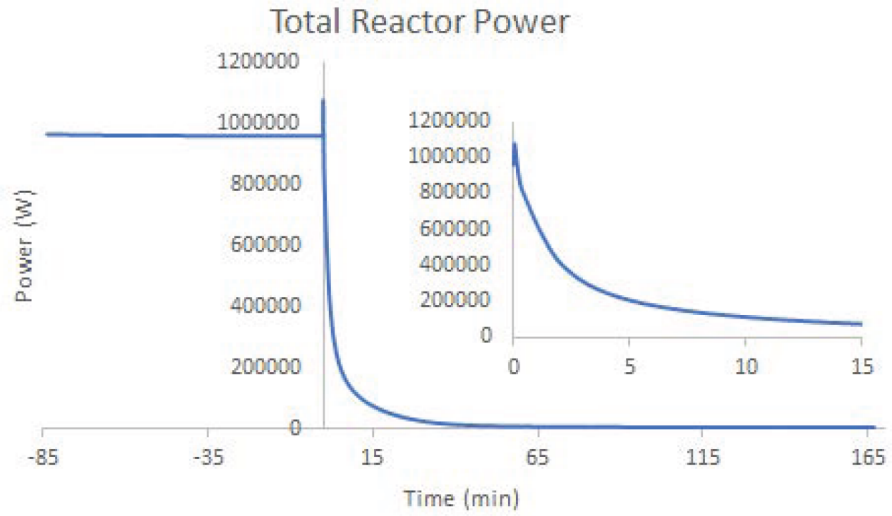


Figure 13.1-6 Power Response from Complete Loss of Pumping Power



A reduction in cooling is shown to have no substantial bearing on the fundamental safety of the reactor. Even without control rod activation, there are no safety-critical negative consequences; no unsafe pressure, temperature, or large power excursions occur. The strong negative reactivity temperature feedback inherent in the reactor prevents such consequences. This feedback gives the reactor a strong anchor to the initial temperature based on reactor physics phenomena and quickly returns the

reactor vessel temperature to a temperature that is similar to the initial reactor temperature distribution. The resulting increase in temperature is small, about 7 degrees Celsius.

13.1.5 Reactivity and Power Distribution Anomalies

A number of scenarios can result in reactivity increases in the MSRR core. Each of these scenarios is analyzed through RELAP5-3D analysis to ensure the transient case of each does not exceed safety limits. The nominal operating power of 1 MWth is used for these transients.

Potential sources of reactivity insertions are the presence of voids or entrained gases, the presence of xenon in the fuel salt, reactivity control system malfunction, reactor pump trip, addition of excess uranium, temperature changes, and pressure changes. These phenomena and their potential impact on reactivity are described in Section 4.5 and accounted for in the analyzed scenarios.

The relevant limiting conditions for these scenarios are the pressure and temperature safety limits associated with the reactor system. Reactor power is an indirect concern only to the degree that it may result in rapid temperature increases in the fuel or structural steel of the reactor. Despite pressure being a primary concern, it was found not to fluctuate in the vast majority of cases and, therefore, is included only when found to have any changes during the transient. Otherwise, pressure curves can be considered to be identical to the steady-state case.

Reactor power is included in some transient cases, particularly those with the most intense reactivity insertions (large or short-time period insertions). Changes in power are summarized in Table 13.1-5.

Table 13.1-5 Summary of Analyzed Reactivity Insertions

Section	Insertion Event	Reactivity Insertion (pcm)	Time of Insertion (s)	Maximum Temperature Change	Equilibrium Temperature Change (°C)	Peak % Change in Power
13.1.5.7	Spurious control rod movement	130	13s		11	Bounded
13.1.5.1	Pressure transient and bubble collapse	396	7	78.9	25.2	638.7
13.1.5.2	Excessive cooling	0	0	3.8	2.4	60.0
13.1.5.3	Excess fuel salt injection	79.4	60	15.3	5.2	71.8
13.1.5.4	Interfacing systems failure	Not applicable - only viable scenario is covered in 13.1.5.3				
13.1.5.5	Adverse reactor vessel geometry changes	806	10 (transitory)	0.1	0.1	34.37
13.1.5.6, 13.1.4	Loss of forced circulation	162.5	7	39.5	3.629	25.77
13.1.5.8	Misaligned control elements	Not relevant to safe operation see Section 4.5				
13.1.5.9	Accidental reactivity insertion from surveillance interactions and experiments	Not applicable				
13.1.5.10	Void collapse and reactor pump trip	420	19	78	N/A	215
13.1.5.10	Void collapse, reactor coolant system malfunction, and reactor pump trip	780	7	161	N/A	880

Temperatures took 60-100 seconds to peak in the transients examined, meaning control rods and fuel drain both can be used to bring the reactor below critical before reaching a significant fraction of peak power. Over the short term, starting from an operating temperature of 600 degrees Celsius, the operating temperature of stainless steel will be exceeded by any insertion that exceeds 500 pcm. The reactor system will be designed to accommodate brief thermal excursions as represented by the limiting reactivity insertions.

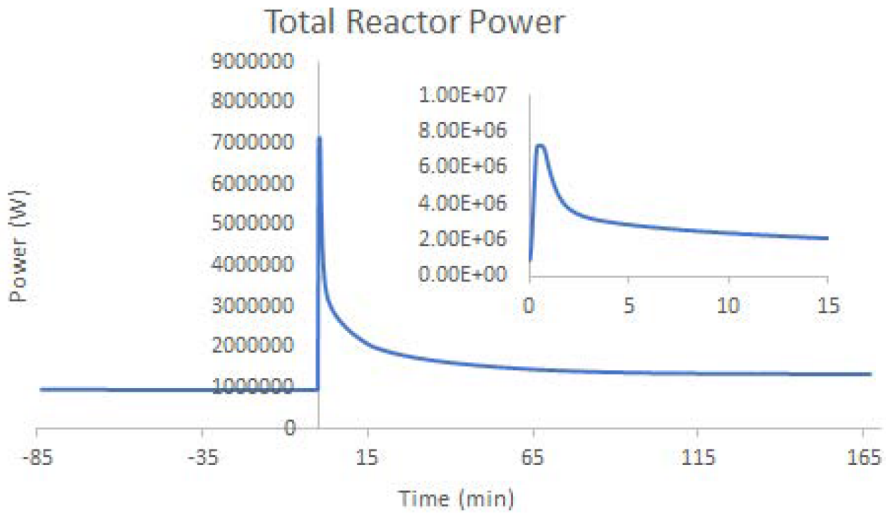
Each respective transient is shown as occurring at time=0 in the figures. No credit is taken for operator actions, RPS functions, or Reactivity Control System (RCS) functions in order to analyze the natural response of the system.

13.1.5.1 Pressurization of Fuel Fluid or Excessive Reactor Vessel Voiding and Subsequent Bubble Collapse

An instantaneous removal of all entrained helium in the fuel salt has been shown to result in a 396 pcm positive reactivity insertion for helium entrainment of 1.8 percent by volume (Section 4.5). This scenario can occur as the reactor pump is shut down and entrained helium bubbles rise to the surface through the RAV. The temperature change is expected to be less than 50 degrees Celsius and has no adverse effect on the safe operation of the reactor. While not included in this analysis, during normal operations, this temperature change is easily ameliorated with control rods.

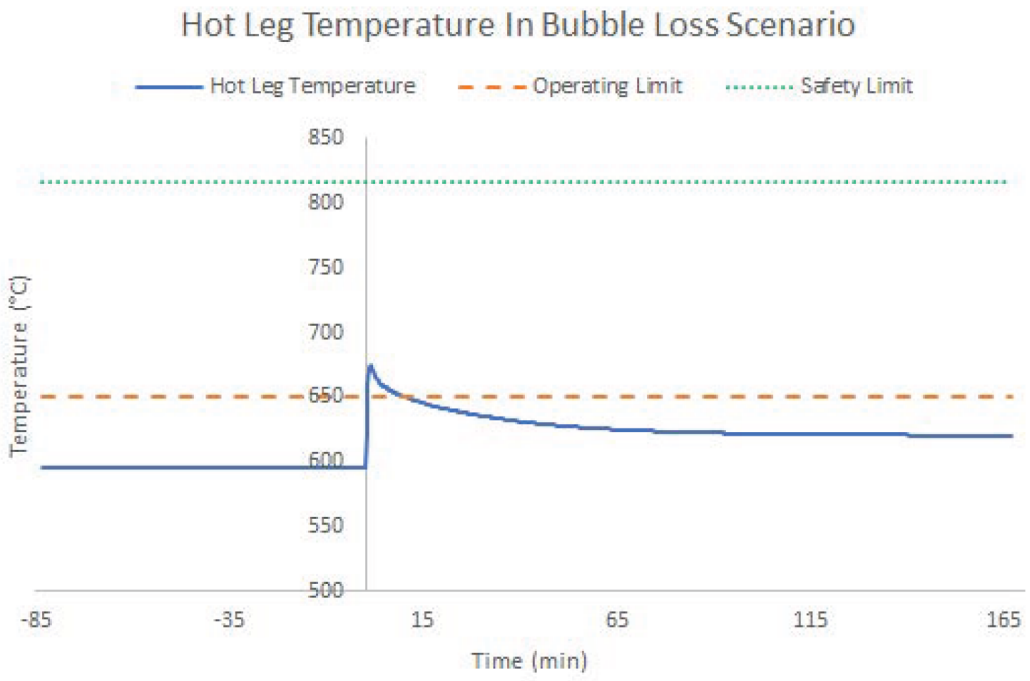
Additionally, the 396 pcm reactivity insertion from gas loss occurs over a relatively long period of time. Based on a force balance used to calculate the rate of bubble rise within the reactor, larger bubbles of 1 centimeter (0.4 inch) or more in radius may leave the reactor vessel in three to five seconds while smaller bubbles in the range of a millimeter (0.04 inch) or smaller radius may take as much as 10 to 100 seconds to leave the reactor. Meanwhile, new helium bubbles are introduced to the reactor vessel from the rest of the loop as it flows through and arrives at the reactor vessel. The total loop travel time is in the range of 60 seconds at the maximum reactor loop fluid flow rate. Time spent in the reactor vessel constitutes about 58 out of those 60 seconds. An insertion time frame of seven seconds was used, over which the insertion addition is approximated as linear.

Figure 13.1-7 Total Reactor Power with Increased Fuel Salt Density as Helium Escapes



Power increases due to helium leaving the reactor are significant, showing a short period where power is more than six times the intended license power (see Figure 13.1-7). This has no direct impact on the safe operation of the reactor. Reactor protection system actuation is unnecessary in this case because the temperature stabilizes at a safe level based on temperature reactivity feedback alone. During normal reactor operation, however, RPS actuation occurs as soon as the reactor power set-point is exceeded.

Figure 13.1-8 Hot Leg Temperature Increase Before Return to Temperature Equilibrium



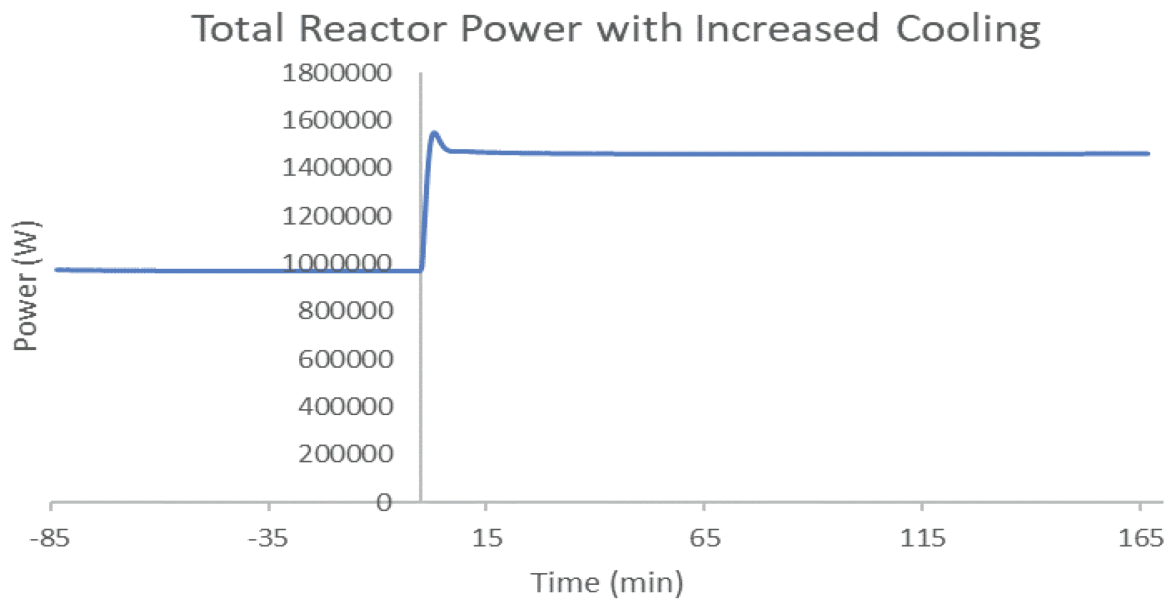
Overall, there does not appear to be any significant danger of loss of the primary fission product barrier that can be achieved through an unplanned voiding of helium from the reactor vessel. Temperatures outside the operational range of stainless steel 316H or FLiBe fuel salt are never achieved (see Figure 13.1-8).

13.1.5.2 Excessive Cooling

The RCS is responsible for maintaining the coolant loop within the appropriate range using both the heaters and air exchange rate in the coolant salt and coolant loop enclosure. In the event of a reduction in coolant loop temperatures due to overcooling at the radiator, power and hot leg temperatures in the reactor loop are expected to rise to compensate.

A RELAP5-3D simulation was performed to determine the transient response in a limiting case where the coolant salt at the heat exchanger inlet is within a few degrees Celsius of the freezing point. In this model, the reference case is allowed to approach steady state for the first 5000 seconds, at which point the temperature of the coolant salt inlet is dropped from 500 degrees Celsius to 462 degrees Celsius, just above the freezing point of FLiBe at 459 degrees Celsius. For purposes of this analysis, no RPS or RCS actuation is assumed in order to evaluate the natural response of the system to overcooling.

Figure 13.1-9 Excessive Reactor Cooling Increases Reactor Power

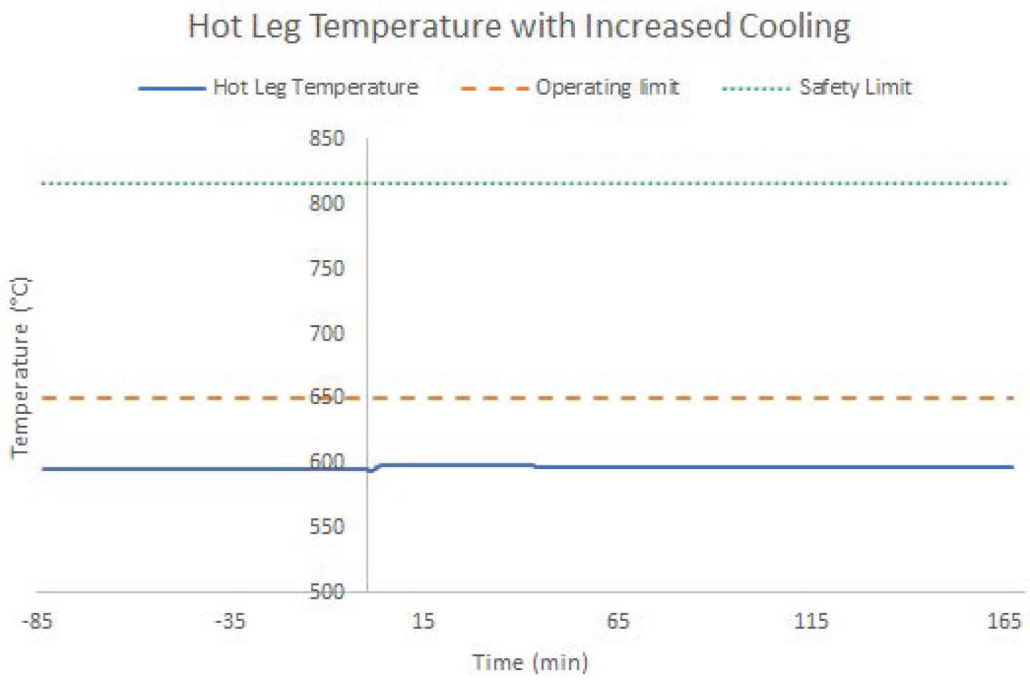


The result of this increase in cooling is a reactivity insertion and corresponding increase in power (see Figure 13.1-9). Although a cooling rate increase causes the reactor to exceed its licensed power rate as shown here, no direct effect on reactor safety is seen either in pressure excursion or temperatures that exceed the operational range of stainless steel 316H.

Only a very modest temperature rise in the hot leg is seen per Figure 13.1-10, as predicted, which compensates for the colder cold leg temperature to maintain a consistent average temperature within the graphite core. Given that a colder temperature cannot be achieved within the coolant loop without freezing the coolant loop, this transient can reasonably represent the bounding case for excessive cooling of the reactor loop. Colder temperatures within the coolant loop would represent a complete loss of flow, which can be represented by the reduction in cooling transients of Section 13.1.4. Given that the reduction-in-cooling transient scenarios also result in safe steady-state temperatures with or without operator intervention, it can reasonably be concluded that the MSRR can safely operate regardless of any changes in coolant temperature.

While not assumed here, in operation, the RPS actuates as soon as power exceeds the set-point and terminates the event.

Figure 13.1-10 Hot Leg Temperature of Reactor Loop During Increased Cooling

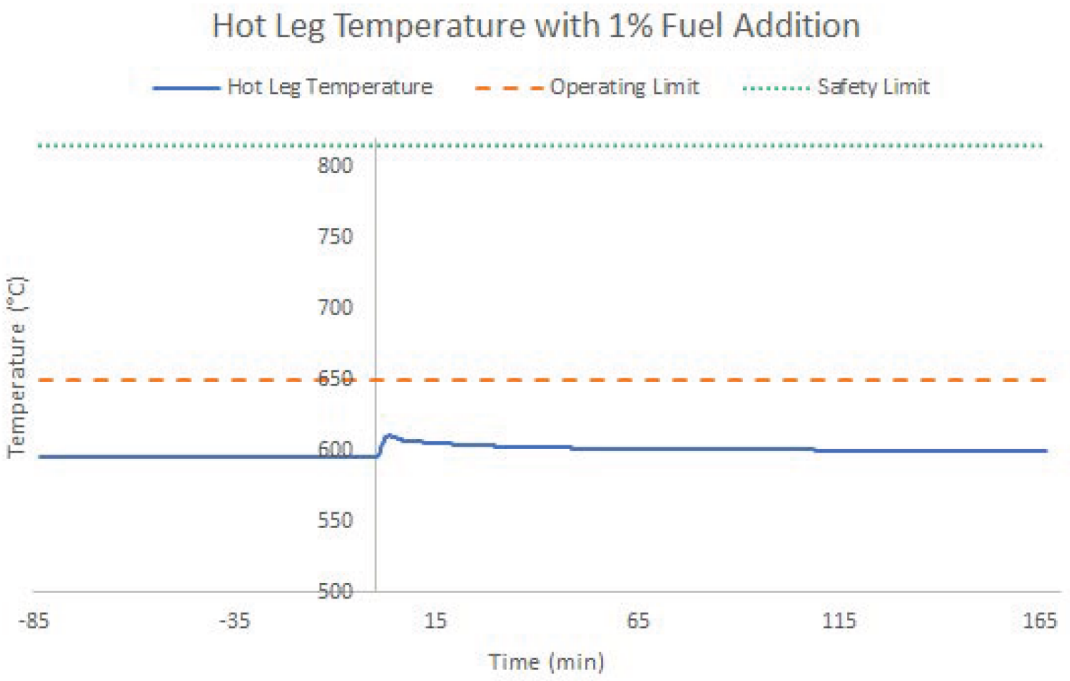


13.1.5.3 Fuel or Fuel Salt Injection

No amount of excess fuel salt injected into the reactor loop affects the fuel level in the reactor vessel once it is filled. Because the RAV acts as an expansion tank and remains subcritical at any fuel level because of its geometry, criticality accidents cannot occur from overfilling.

If the fuel salt UF_4 concentration were to change, as discussed in Section 4.5, there is a realistic mechanism to increase power and temperature. To model this case, MCNP was used to simulate a 1 percent relative increase in the overall UF_4 concentration of the fuel. This can be considered to be a bounding case of UF_4 concentration increase because this amount of fuel constitutes 10 MW-yrs of fuel and it is safe to say that no more fuel than this would be kept within the facility. Figure 13.1-11 shows the temperature at the reactor vessel outlet representative of the hot leg fluid temperature during a 1-percent increase in UF_4 transient with no RPS or RCS activation.

Figure 13.1-11 Reactor Vessel Outlet Temperature During 1 Percent UF_4 Transient Increase



No RCS or RPS actuations are assumed in this analysis to evaluate the system's natural response. Reactor power increases by 71.8 percent to 1.72 MWth. Equilibrium temperature stabilizes at a 5.2 degrees Celsius increase. Temperatures remain well within the system's structural limits.

In operation, the event is terminated by RPS actuation before power increases above the set-point.

13.1.5.4 Interfacing Systems Failure

The systems interfacing with the reactor system that can produce positive reactivity changes are limited to fuel storage tanks in the FHS. This transient scenario is addressed in Section 13.1.5.3. No other credible accident scenarios have been found with interfacing system failures.

No other non-fuel liquids can interface with the reactor during operation; therefore, this accident can be considered inapplicable.

13.1.5.5 Adverse Active Reactor Vessel Geometry Changes

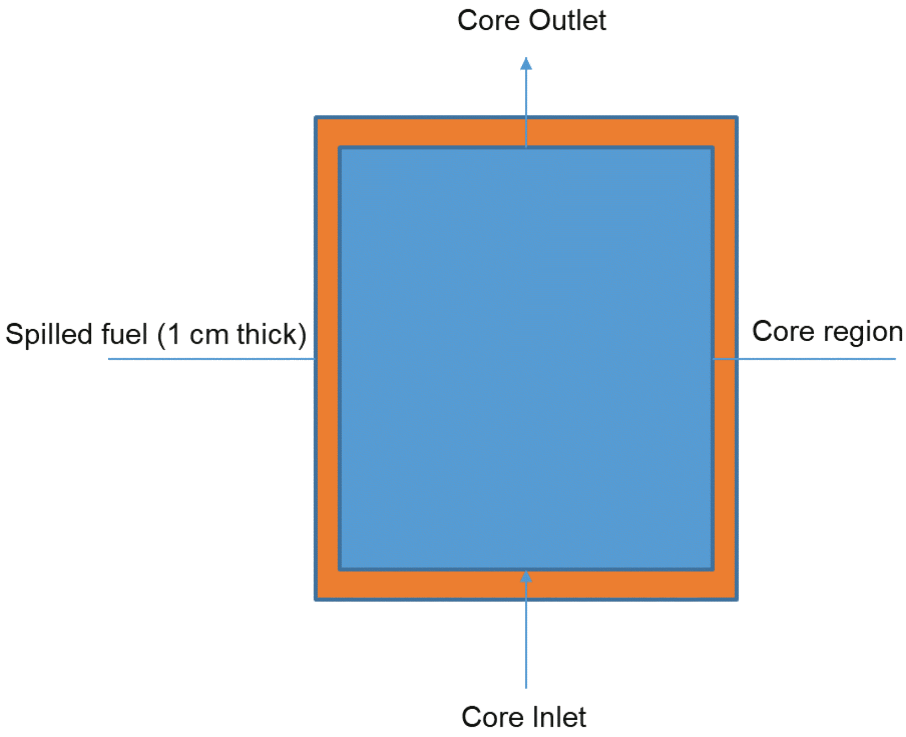
Graphite movement is precluded by the design of the graphite moderator and reactor vessel support structures, as presented in Section 4.2.5. Structural failure of a single block, which may result in the block floating upwards and being

replaced by fuel salt, is unlikely to result in a large reactivity insertion. An analysis demonstrating this will be based on the detailed design of the graphite blocks and presented in the Operating License application.

The reactor geometry change of primary concern is the addition of fuel around the outside of the reactor vessel structure. In the event of a large leak above the reactor vessel, it is possible for fuel to leak down and around the reactor vessel. The worst-case reactivity insertion requires that all the salt positioned above the reactor vessel in the loop runs down symmetrically around the reactor vessel, creating a concentric cylinder of fuel, approximately 1 cm (0.4 inch) in thickness, flowing around the reactor vessel. This effectively increases reactor vessel diameter until the salt flows past and settles below the reactor vessel and, therefore, has a strong propensity to briefly increase reactivity.

While the formation of an even, concentric cylinder of fuel around the outside of the reactor vessel is non-physical, it provides a larger, albeit temporary, positive reactivity insertion relative to the rest of the accidents analyzed. See Figure 13.1-12.

Figure 13.1-12 Simplified Diagram of Adverse Reactor Vessel Geometry Accident



This scenario was calculated in MCNP to determine the maximum possible insertion from fuel leakage onto the outside of the reactor vessel. The maximum theoretical increase in reactivity from this event has been determined to be 806 pcm, under the conservative assumption that the reactor loop remains full. In RELAP5-3D, this is modeled as occurring over the course of 10 seconds. Of all

the transients analyzed, this is the highest reactivity insertion. Because it is complete after only a few seconds and results in a loss of pump prime and loop flow, the increase in power and temperature is minimal. The primary result is a loss of power shortly after the transient begins. In this case, passive temperature feedbacks are allowed to bring the temperature back down to a safe operating level to determine behavior without RPS or RCS intervention to evaluate the inherent safety of the reactor design. This accident progression results in fuel salt in the reactor thermal management system and reactor gases in the reactor enclosure, which is bound by the MHA.

Figure 13.1-13 shows the increase in temperature is less than 1 degree Celsius while Figure 13.1-14 shows the power increases to 1.34 MWth. The temperature increase is well within the structural limits of the system. In operation, the event is terminated by the RPS as soon as power increases above the set-point.

Figure 13.1-13 Hot Leg Temperature with Increased Cooling

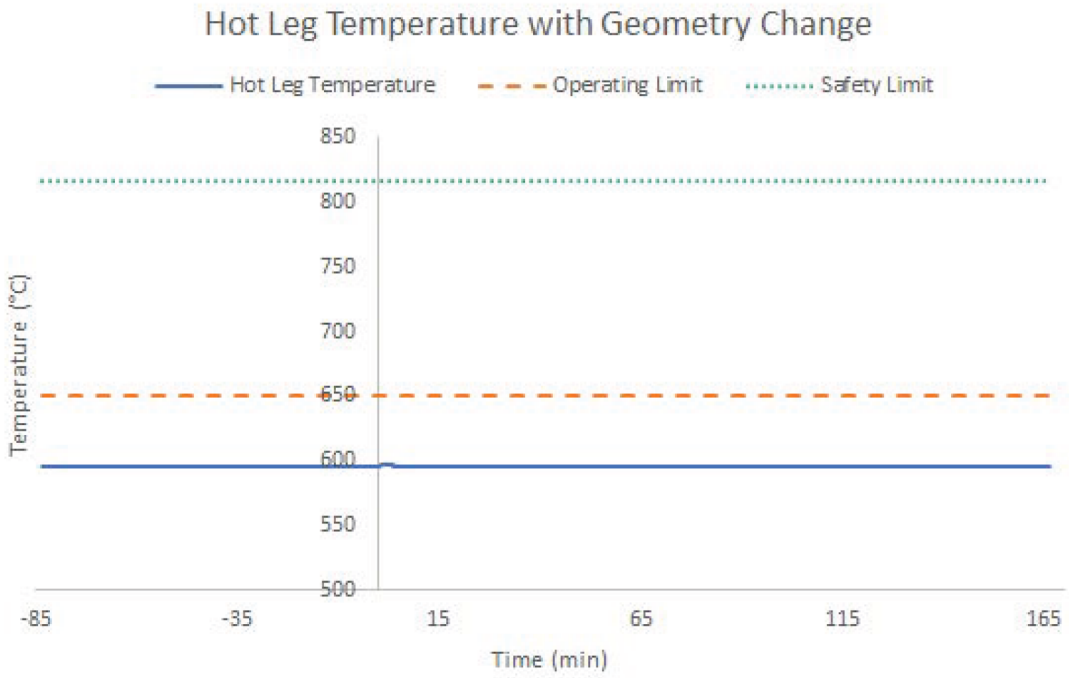
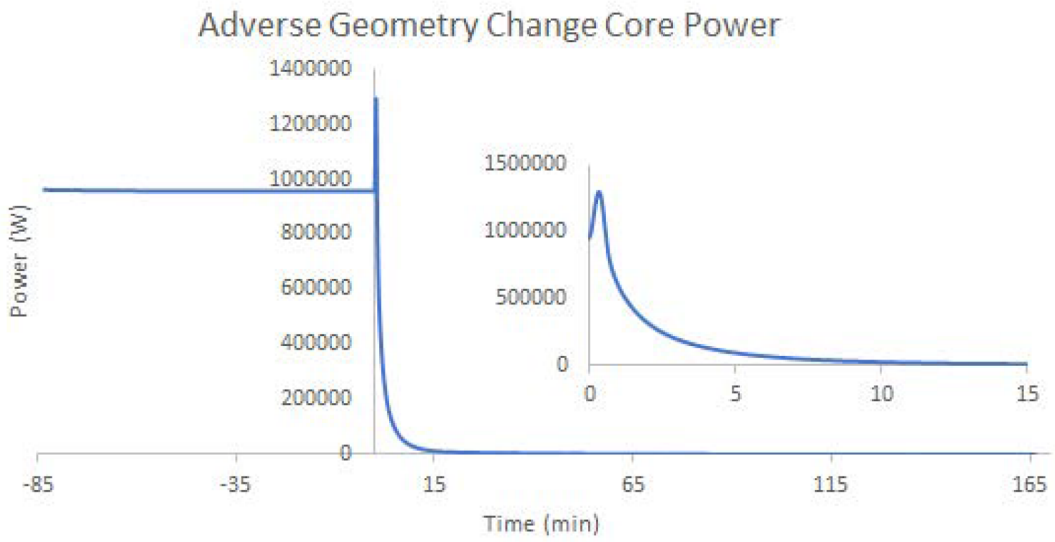


Figure 13.1-14 Initial Power Spike Followed by Loss of Criticality as Flow Stops



13.1.5.6 Loss of Forced Circulation Including Loss of Electrical Power

A loss of forced circulation is modeled in the reduction-in-cooling scenario presented in Section 13.1.4. This scenario shows the most extreme reduction in cooling, with a complete loss-of-flow accident. Analysis finds that after a small and brief spike in temperature staying well within the materials limits of the reactor, the fuel temperature within the reactor vessel homogenizes over time, and the steady-state high temperature within the graphite channels at the center channel outlet reduces because of a flatter temperature profile at extremely low power rates. Loss of electric power is analyzed in Section 13.1.10. The loss of forced circulation with loss of electrical power accident is bounded by the reactivity insertions analyzed in Section 13.1.5.10, which also include the reactivity insertion due to helium entrainment loss during a pump failure scenario, as well as accidental rod removal in two of the three cases examined.

13.1.5.7 Spurious Control Element Actuation

Removal of the largest worth control rod from the normal operational position results in a reactivity insertion of 130 pcm. With a total temperature feedback of -12.045 pcm per degree Celsius, this results in an equilibrium increase in temperature of 10.8 degrees Celsius. Relative to other accidents analyzed, this control rod removal causes a small reactivity insertion that is considered to be bounded by other accidents analyzed in this section.

13.1.5.8 Misaligned Control Elements

While changes to power distribution can occur depending on the relative height of each control rod within the reactor vessel, the alignment of the control rods is not relevant to safe operation of the MSRR, as stated in Section 4.5.

13.1.5.9 Accidental Reactivity Insertion from Surveillance Interactions and Experiments

Surveillance and experiment system malfunction is not foreseen to have an effect on reactivity or power. The RAV is outside the graphite core region, so any experiments performed would have no effect on reactivity. Objects added to the RAV are contained to prevent them from entering the other reactor system components. Relocation of materials added to the surveillance system do not insert reactivity. The safety compliance and approval process of all reactor surveilance work can be found in Chapter 10 of this document.

13.1.5.10 Maximum Reactivity Insertions

A subset of the reactivity insertions may occur simultaneously or in relatively rapid succession as discussed in Section 4.5, necessitating that these events be analyzed together. The operating limits presented in Section 4.5 are derived to accommodate a limiting reactivity insertion based on the cumulative effect of several simultaneous system failures. The limiting reactivity insertions are briefly summarized here.

The first limiting reactivity insertion is a loss-of-flow accident at the maximum operating temperature described in Section 4.5.

The second limiting reactivity insertion is a combined void collapse and coolant pump trip. This has a reactivity insertion of 350 pcm and a loss-of-flow accident (70 pcm) at normal operating conditions. The accident scenario is described in Section 4.5, as follows:

- Reactor is operating at full power at an average temperature of 600 degrees Celsius.
- Pump suddenly stops inserting +70 pcm.
- Reactor has voided up to the limit; control rods fully withdrawn to compensate.
- Reactor suddenly loses voids inserting +350 pcm because of the change in density. Void worth is limited by the allowable control rod worth compensating.
- An RCS malfunction cannot insert positive reactivity because rods are fully withdrawn.

The third and most limiting reactivity insertion is combined void collapse, RCS malfunction, and coolant pump trip. This has a reactivity insertion of 520 pcm over seven seconds, followed by 190 pcm over 19 seconds, and a loss-of-flow accident (70 pcm). The scenario as described in Section 4.5, as follows:

- Reactor is operating at full power at an average temperature of 570 degrees Celsius.
- Control rods are inserted to compensate for lower temperature.
- Coolant pump suddenly stops inserting +70 pcm.
- Reactor has voided up to the limit, control rods are partially withdrawn to compensate.
- Reactor suddenly loses voids inserting +400 pcm because of the change in density. Xenon migrates with the helium, inserting +120 pcm of reactivity.
- The RCS fails, gradually inserting positive reactivity at a rate of 10 pcm/s. The total control rod worth is the design control rod worth plus the control rod worth introduced from temperature compensation minus the control rod worth compensating for voiding (+350 pcm + 360 pcm - 400 - 120 pcm = 190 pcm).

The second and third limiting reactivity insertions are analyzed assuming RPS actuation.

Figure 13.1-15 shows temperature versus time in the first 120 seconds for the second limiting reactivity insertion. The orange dashed line is the operating temperature limit when the reactor begins to drain. Figure 13.1-16 shows power versus time in the first 120 seconds. Draining of the fuel salt is initiated several seconds after the maximum temperature reaches 650 °C. The fuel loop has drained sufficiently to ensure the reactor is subcritical 60 seconds after initiation. The temperature of the reactor system does not exceed the safety limit.

Figure 13.1-17 shows temperature versus time in the first 120 seconds for the third limiting reactivity insertion. The orange dashed line is the operating temperature limit when the reactor begins to drain. Figure 13.1-18 shows power versus time in the first 120 seconds. The temperature of the reactor system does not exceed the safety limit.

The reactor system operating temperature limit is briefly exceeded during these scenarios for a short time. The reactor system will be designed to accommodate brief thermal excursions such as these.

Figure 13.1-15 Second Limiting Reactivity Insertion Temperature

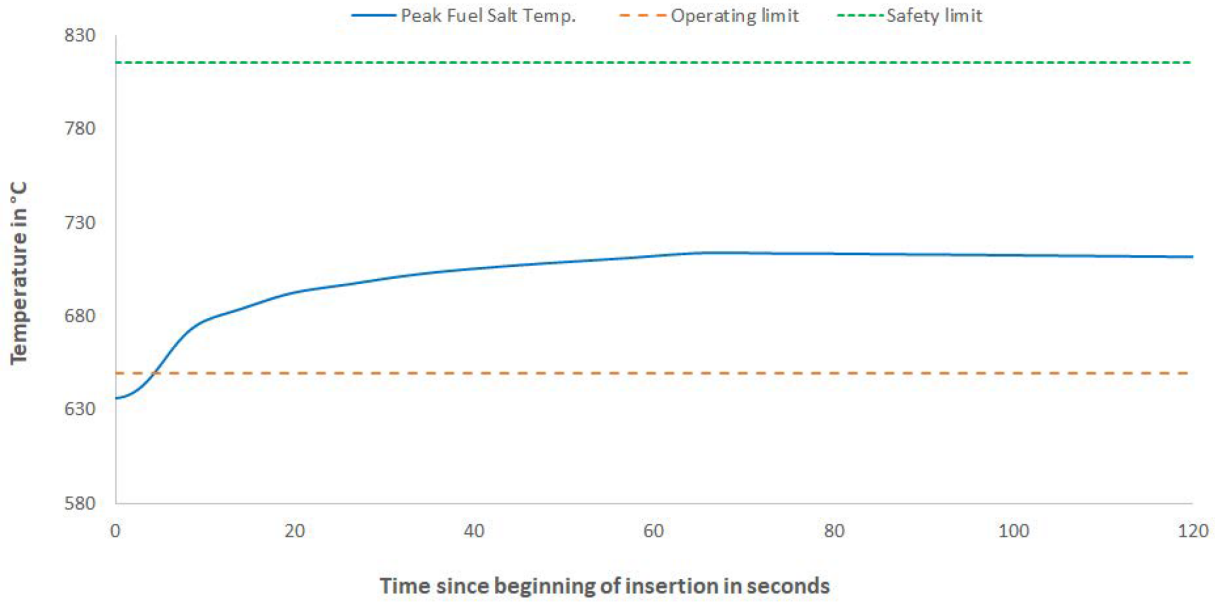


Figure 13.1-16 Second Limiting Reactivity Insertion Power

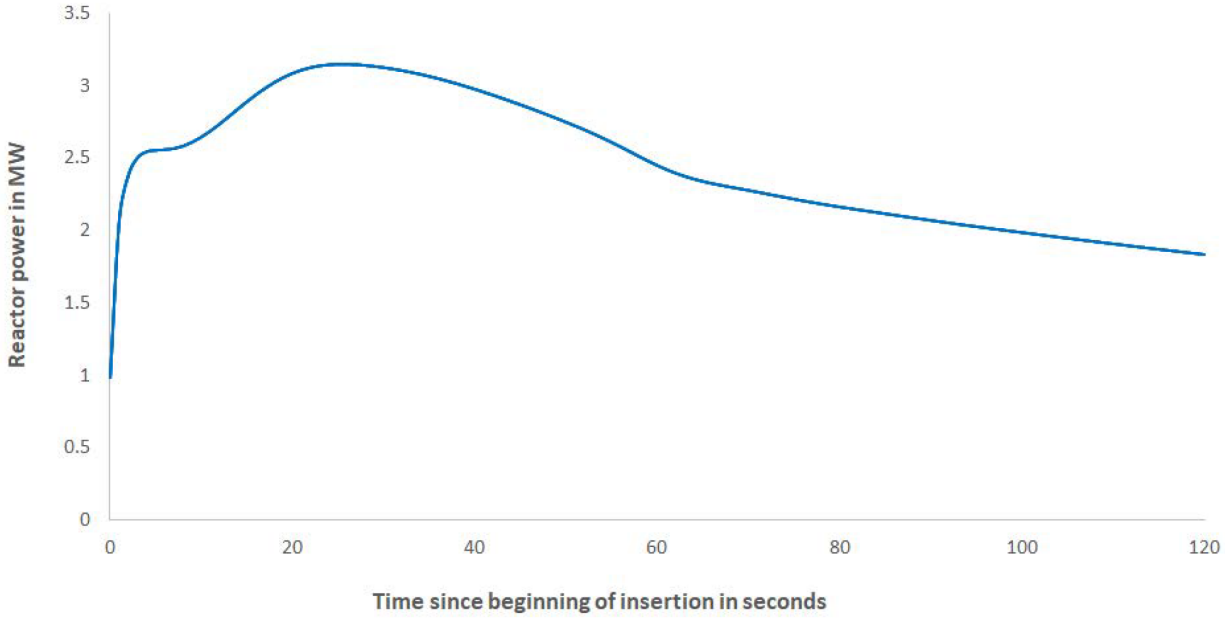


Figure 13.1-17 Third Limiting Reactivity Insertion Temperature

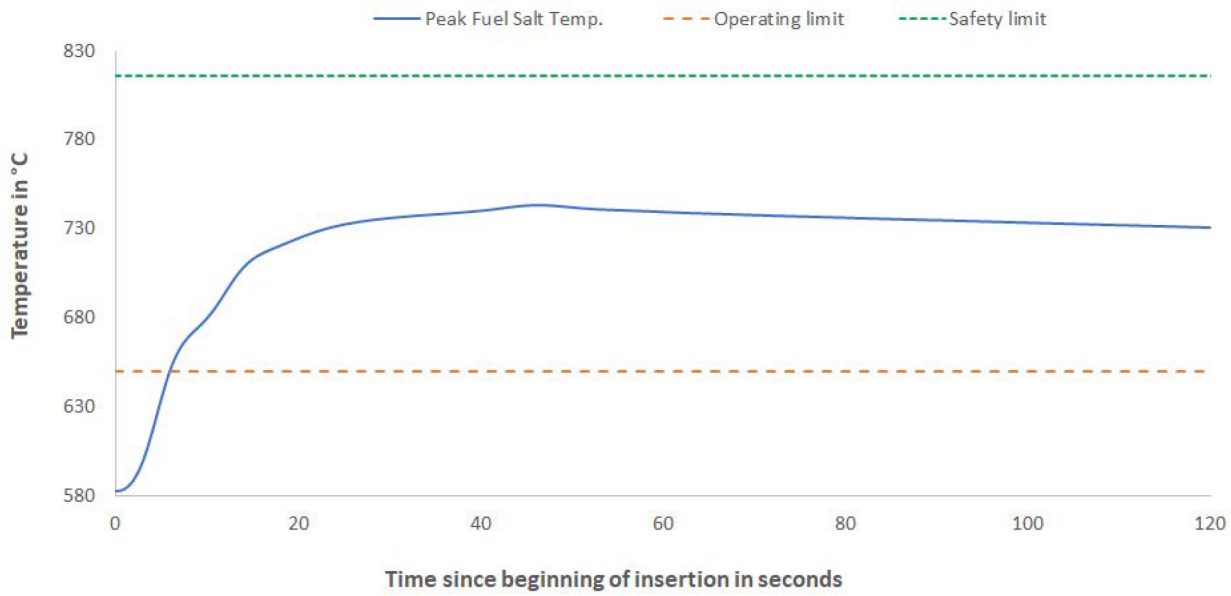
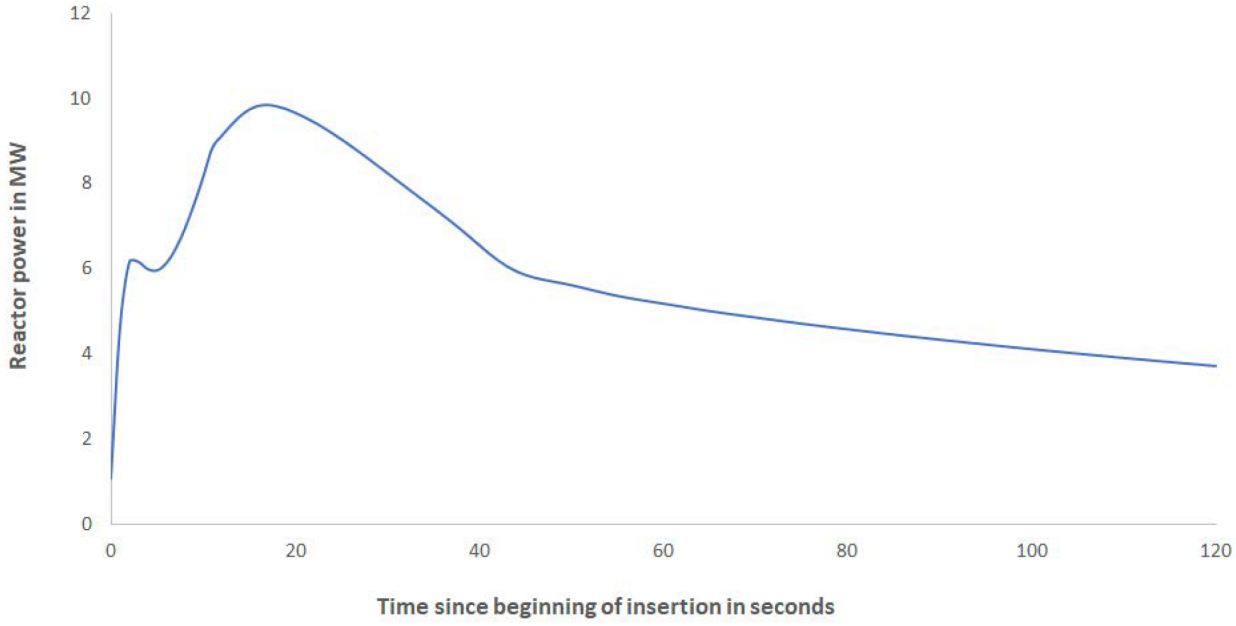


Figure 13.1-18 Third Limiting Reactivity Insertion Power



13.1.6 Mishandling or Malfunction of Fuel

The primary accident that can result from fuel mishandling is the addition of excess fuel, which is described in Section 13.1.5.3. Even in the case of a large amount of fuel injected (larger even than the amount kept onsite), the reactor maintains safe temperatures and operating conditions. That scenario is outside of the realm of feasibility and the results in Section 13.1.5.3 show a conservative bounding case produces no safety concerns in relevant metrics such as temperature and pressure.

Another possible accident from fuel mishandling is an inappropriate salt level (level below the pump), which results in loss of salt flow. This situation results in convective flow in the reactor vessel and reduces reactor power to a low level of passive heat removal through the walls of the reactor vessel until the reactor fuel salt is drained into the drain tank.

More in-depth description of the FHS, the fuel handling process, and safety-related concerns are provided in Section 9.2.

13.1.7 Experiment Malfunction

There are five types of experiments and surveillance in the MSRR.

- Coupons are used as a proxy to examine corrosion over time and to monitor the overall health of the structural material as the reactor continues to operate.
- Samples of salt are taken from the RAV. A sample canister is filled in the RAV, then sealed and removed for surveillance work with salt chemistry.
- Cover gas samples also are taken to determine the amounts and kinds of fission products found in the head space above the RAV, giving additional insight into the behavior and transport of fission products.
- Beryllium can be added to the fuel salt to maintain redox conditions.
- Helium gas may be bubbled into the experimental tank and a bubble removal system employed following the tank.

In the event a capsule, coupon, or redox material escapes from its holder, it is prevented from leaving its apparatus by design.

Samples of fuel salt are precluded from physically damaging the reactor system structure. Samples of fuel salt are not cooled; their capsule design is sufficient to retain radionuclides by removal of heat to the ambient environment, which is much cooler than the reactor system.

13.1.8 External Events

13.1.8.1 Earthquakes

Significant earthquakes are not known to occur within the vicinity of the chosen site. A search of historical earthquake data finds that the largest earthquake that has occurred near Abilene was a magnitude 4.6 earthquake at a distance of

121 mi (195.1 km). This magnitude is insufficient to cause damage to modern buildings, so earthquakes are not considered a credible source of transient disturbance. More in-depth discussion of seismic hazards is provided in Section 2.5.3 and Section 3.4.

13.1.8.2 Tornadoes

Tornado hazards are described in Section 2.3 and an analysis of the ability of the MSRR to withstand extreme winds is provided in Section 3.2.

13.1.8.3 Floods

The site has not historically experienced flooding, is outside the 500-year flood plan, and is outside the inundation area for dam failures, so flooding is not foreseeably a credible external event (see Section 2.4). Flooding of the reactor pit with water will not pose a criticality concern, as stated in Section 4.5. Any damage to electrical equipment caused by flooding is bounded by the loss of electric power accident in Section 13.1.10.

13.1.8.4 Volcanism

The chosen site is geologically stable, and volcanic activity is not a credible threat to the reactor facility given its geological location, as described in Section 2.5.

13.1.8.5 Missile Impact

Credible missile impact can result from objects carried by high winds. The reactor is located below grade surrounded by thick concrete shielding and has a thick concrete lid. It is designed to withstand a complete collapse of the above-grade SERC structure. All safety functions required for safe shutdown and cooling are located inside the concrete shielding below grade, as described in Chapter 3.

13.1.9 Mishandling or Malfunction of Equipment

The function of the off-gas system is described in detail in Section 9.6. Mishandling of equipment in the off-gas system will result in off-gas enclosure isolation, preventing a large release.

An overhead crane malfunction can result in a heavy object dropping onto the external concrete shield of the reactor. This concrete barrier is able to withstand the weight and impact of any object that would foreseeably be moved with the overhead crane. Prior to the concrete barrier being in place, neither fuel salt nor radioactive material are present as a safety concern, making reactor system damage the only consequence in the case of a crane malfunction.

Finally, any sample taken from the reactor has the possibility of exposing fission products to the environment if handled improperly. To minimize the risk of exposure to workers and the environment, samples are taken in small amounts in appropriately

sealed containers for transport from the reactor to the radiochemistry lab. Where necessary, work on samples is done in glove boxes with appropriate levels of radiation shielding.

13.1.10 Loss of Normal Electrical Power

The MSRR does not require an emergency core cooling system, an active decay heat removal system, or any kind of backup or emergency electrical power supply. If off-site electrical power is lost, all electrically operated systems, including the auxiliary heat removal system, will stop in the MSRR.

In case of a loss of normal electric power (LONEP), the following key events will occur more or less simultaneously:

- Reactor trip valves deenergize, equilibrating pressure in the RAV and drain tank, thus shutting down the reactor and draining the fuel salt to the drain tank.
- Reactor pump and coolant pumps stop.
- The reactor thermal management system deenergizes; sufficient heat remains in the fuel salt to ensure it drains.
- Auxiliary heat removal system deenergizes; reactor cell louvers close.
- Reactor enclosure isolates.

At this point in the incident progression, decay heat is removed passively. Detailed modeling in RELAP5-3D using preliminary design parameters demonstrated this incident does not challenge any radionuclide barriers. Heat transfer is primarily by natural convection within enclosures and by thermal radiation between solid bodies. The maximum temperature of the drain tank is less than 660 degrees Celsius; the maximum temperature of the reactor enclosure is less 100 degrees Celsius; and the maximum temperature of the biological shielding is less than 60 degrees Celsius. This situation is maintained for the duration of the incident. The temperature of the drain tank peaks within the first few days, then steadily cools. The temperature of the enclosure and cell peaks after several weeks. The pressure of the reactor enclosure is negative initially but will gradually become neutral with time. There is no driving pressure differential to the outside. No safety-related temperatures or pressures are breached.

13.2 Accident Analysis and Determination of Consequences

Table 13.2-1 summarizes the accidents analyzed in this document. All accidents, including the MHA potentially resulting in a loss of reactor functionality, were found to maintain safe conditions for the public, staff, and environment. These events present no technical need for direct action from an operator to maintain safe conditions, and most events do not require RPS actuation to maintain temperature limits. Any event that would result in exceeding license power or administratively set limits to temperature or pressure, however, would result in an RPS actuation and draining of fuel to the drain tank.

Table 13.2-1 Summary of Accident Scenarios Examined

Accident Scenario	Initiating Event	Safety Consequences	Safety Measures Needed to Avoid Unsafe Conditions	Safety Metric(s)
Maximum hypothetical accident (Section 13.1.1)	Loss of offsite power and release of irradiated fuel salt into the reactor enclosure	Dose to the public of <100 mrem	Full RPS actuation is performed passively on power loss	Reactor enclosure pressure remains below atmospheric pressure; public dose remains below 100 mrem
Reduction in cooling (Section 13.1.4)	Reactor loop flow loss, coolant loop flow loss, ultimate heat sink loss	Reactor shutdown due to equipment malfunction, bounded by 13.1.5.10	None	Maximum reactor system temperature < 816 °C
Increase in fuel salt inventory or concentration (Section 13.1.3)	Fuel storage leak, operator error	Reactor drains to drain tank for fuel rebalancing, up to 3.8°C increase in temperature	None	Maximum reactor system temperature < 816 °C
Bubble collapse/ void fraction decrease (Section 13.1.5.1)	Pressure decrease, pressure increase, sparge gas flow stoppage	Use of control rods to return to desired power, 78.9°C increase in temperature	None	Maximum reactor system temperature < 816 °C
Spurious control rod activation (Section 13.1.5.7)	Reactivity insertion	Small, safe increase in temperature and power, bounded by 13.1.5.10	None	Brief increase in reactor power
Misaligned control rods (Section 13.1.5.8)	Operator error or control rod malfunction	Operational power limited below design and license power	None	Not an accident scenario in this design
Surveillance malfunction (Section 13.1.5.9, Section 13.1.7)	Dislodged sample in the RAV	Object retained within the RAV	None	Bounded by loss of forced circulation
Excessive cooling (Section 13.1.5.2	Overcooling of radiator and coolant salt	Power increase and modest increase to hot leg temperature, temperature increase of 3.8°C	None	Brief increase in reactor power

Table 13.2-1 Summary of Accident Scenarios Examined (Continued)

Accident Scenario	Initiating Event	Safety Consequences	Safety Measures Needed to Avoid Unsafe Conditions	Safety Metric(s)
Loss of normal electrical power (Section 13.1.10)	Power outage	Reactor scram, loss of auxiliary heat removal, temperature increase of 39.5°C	None	Temperature of radionuclide barriers does not breach safety limit
Adverse reactor vessel geometry changes (Section 13.1.5.5)	Fuel salt relocation outside the reactor vessel inside the reactor thermal management system	Release of radionuclides to the reactor enclosure	Bounded by MHA	See MHA for consequences
Flooding of reactor cell (Section 13.1.8)	Ingress of water from adverse weather or rupture of building water lines	Loss of electrical systems	Full reactor shutdown and removal of water to avoid corrosion	Temperature of radionuclide barriers does not breach safety limit
Maximum reactivity insertion (Section 13.1.5.10)	Loss of voids, control element malfunction, loss of reactor pump	Reactor system temperature briefly exceeds operating limit and RPS is activated to prevent further temperature increase	RPS actuation	Temperature of radionuclide barriers breach operating limit but not safety limit
Reduction in fuel salt inventory (Section 13.1.2)	Leakage of fuel salt	Bounded by MHA		
Mishandling or malfunction of fuel (Section 13.1.6)	Bounded by fuel or salt injection (Section 13.1.5.3)			
External events (Section 13.1.8)	Various	None	RPS actuation, reactor cell isolation, passive cooling	Temperature maintained by passive cooling
Mishandling or malfunction of equipment (Section 13.1.9)	Mishandling and release of fission product gases	Fission products retained in enclosures	FHS, reactor, and coolant loop enclosure integrity	Radiological dose

Table 13.2-1 Summary of Accident Scenarios Examined (Continued)

Accident Scenario	Initiating Event	Safety Consequences	Safety Measures Needed to Avoid Unsafe Conditions	Safety Metric(s)
Fuel or fuel salt injection (Section 13.1.5.3)	Addition of UF_4 to the fuel salt	Increase power and temperature	None	Power and temperature
Interfacing systems failure (Section 13.1.5.4)	Bounded by fuel salt injection (Section 13.1.5.3)			
Loss of forced circulation (Section 13.1.5.6)	Bounded by loss of electric power (Section 13.1.10)			

13.3 Summary and Conclusions

None of the credible transient scenarios analyzed result in unsafe conditions for the reactor as designed. Even without insertion of control rods, short term transients do not result in unsafe temperatures or pressures within the reactor. Further, taking no credit for the drain tank or control rods in the safety case, the reactor continues to maintain safe temperatures and pressures. This shows that the design has sufficient inherent safety to withstand every anticipated event that is within the design basis of the reactor.

In the maximum exposure scenario analyzed in the MHA, several barriers fail, resulting in the release of radioactive fission products. The maximally exposed individual does not exceed annual worker dose limits and does not suffer negative health repercussions as a result. Additionally, the maximally exposed individual at the site boundary does not receive more than 100 mrem of exposure.

13.4 References

13.4-1 Oak Ridge National Laboratory Report, "Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application."

13.4-2 Idaho National Laboratory, RELAP5-3D, Idaho Falls, ID, 2015.

13.4-3 W. A. Wieselquist, R. A. Lefebvre, and M. A. Jessee, Eds., “SCALE Code System,” ORNL/TM-2005/39, Version 6.2.4, Oak Ridge, TN, 2020.

13.4-4 Los Alamos National Laboratory, “Monte Carlo N-Particle Transport Code”, Los Alamos, NM, 2016.

13.4-5 The MathWorks, “Symbolic Math Toolbox,” Natick, MA, 2019, retrieved from https://www.mathworks.com/help/symbolic/.

13.4-6 J. R. Lamarsh and A. J. Baratta, “Introduction to Nuclear Engineering 3rd ed.” Ch. 11, Pg 650. Upper Saddle River, New Jersey, Prentice Hall, 2001, accessed May 03, 2022.

13.4-7 U.S. Environmental Protection Agency, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” Federal Guidance Report No. 11, EPE-520/1-88-020, Washington, DC, September 1988.

13.4-8 U.S. Environmental Protection Agency, “External Exposure to Radionuclides in Air, Water, and Soil,” Federal Guidance Report No. 12, EPA-402-R-93-081, Washington, DC, September 1993.

13.4-9 International Commission on Radiological Protection, “Reference Man: Anatomical, Physiological and Metabolic Characteristics: Reference Manual,” ICRP Publication 23, Pergamon Press, Oxford, UK, 1975.

13.4-10 FGR 11: U.S. Environmental Protection Agency, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion” Federal Guidance Report 11, EPA-520/1-88-020, Sept 1988. Accessed Sept 20, 2021.

13.4-11 FGR 12: U.S. Environmental Protection Agency, “External Exposure to Radionuclides in Air, Water, and Soil” Federal Guidance Report 12, EPA-402-R-93-081, Sept 1993. Accessed Sept 20, 2021.

13.4-12 ICRP 23: International Commission on Radiological Protection “Report of the Task Group on Reference Man” ICRP Report 23, Oct 1974. Pergamon Press. Accessed Sept 24, 2021.

Chapter 14
Technical Specifications

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 14 TECHNICAL SPECIFICATIONS 14-1

14.1 Probable Subjects of Technical Specifications for the Facility 14-1

14.1.1 Definitions 14-1

14.2 Safety Limits and Limiting Safety System Settings 14-2

14.2.1 Safety Limits 14-2

14.2.2 Limiting Safety System Settings 14-3

14.3 Limiting Conditions for Operations 14-3

14.3.1 Fuel Salt and Fuel System Boundary Parameters 14-4

14.3.2 Reactor Control and Safety Systems 14-5

14.3.3 Primary Cooling and Heat Dissipation Systems 14-6

14.3.4 Functional Containment 14-7

14.3.5 Ventilation Systems 14-7

14.3.6 Radiation Monitoring Systems and Effluents 14-7

14.3.7 Experiments 14-9

14.3.8 Facility-Specific Limiting Conditions for Operations - Fuel Handling System 14-9

14.4 Surveillance Requirements 14-9

14.5 Design Features 14-10

14.6 Administrative Controls 14-10

14.7 References 14-10

CHAPTER 14 TECHNICAL SPECIFICATIONS

14.1 Probable Subjects of Technical Specifications for the Facility

The purpose of this section is to address the requirements of the regulations in 10 CFR 50.36 by providing the Technical Specifications (TS) for the Molten Salt Research Reactor (MSRR) located at Abilene Christian University (ACU). The TS bases other than for administrative controls are included in the TS as required by 10 CFR 50.36(a)(1), but are not part of the TS.

In accordance with 10 CFR 50.34(a)(5), the variables and conditions subject to TS control for the MSRR facility are provided in this chapter. The TS and parameter limits will be submitted with the Operating License application, consistent with 10 CFR 50.34(b)(6)(vi), and will address the requirements in 10 CFR 50.36.

The structure and content of the TS follow the guidance provided in ORNL/TM-1478 (Reference 14.7-1), including Appendix 14.1, which is based on the guidance provided in ANSI/ANS-15.1 (Reference 14.7-2) and include

- definitions.
- safety limits and limiting safety system settings.
- limiting conditions for operation.
- surveillance requirements.
- design features.
- administrative controls.

14.1.1 Definitions

The U.S. Nuclear Regulatory Commission (NRC) and the non-power-reactor community have agreed on most of the definitions given in this section of ANS 15.1. Those applicable to a particular facility should be included verbatim. Facility-specific definitions may be added to clarify terms referred to in the TS. Examples of modifications and additional definitions presented below are based on ORNL/TM-1478/2020 to help clarify the meaning of terms used in ANS 15.1 as applied to the MSRR. A full set of definitions will be included in the Operating License application.

- Core configuration. The active reactor core configuration includes the fuel salt, neutron moderator, control rods, and neutron startup source.
- Experiment. Any operation, hardware, or target that is designed to investigate reactor characteristics or that is intended for irradiation within the reactor structure, or in an irradiation facility. The reactor access vessel (RAV) is not considered an experiment.
- License. The NRC's written authorization for ACU and its reactor operators and senior reactor operators to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

- Reactor secured. The MSRR is secured when the following conditions exist:
  - The reactor fuel salt inventory is in the drain tank and
    - the console key switch is in the off position and removed from the lock.
    - no work is in progress involving fuel salt within the fuel system boundary, the active reactor core, or vessel structure.
    - no experiments are being moved or serviced in the RAV.
- Reference core condition. The reference core condition is the reactivity condition of the active reactor core at a reference set of parameters (e.g., fuel salt temperature, pressure, void fraction, and control element position.)
- Shutdown. The MSRR is shutdown when fuel salt is relocated to the drain tank to ensure subcriticality under all conditions. The gas head spaces of the RAV and drain tank are linked together (same pressure).
- Control rods. Control rods are employed to adjust reactivity during operation. Control rods act through neutron absorption, are solid, and can be actively positioned. Control rods are not needed for reactor shutdown.

The following definitions are added:

- Functional containment. A barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, anticipated operational occurrences, and accident conditions.
- Secured shutdown. Secured shutdown is achieved when the reactor meets the requirements of the definition of "reactor secured" and the facility administrative requirements for leaving the facility with no licensed reactor operators present.

14.2 Safety Limits and Limiting Safety System Settings

14.2.1 Safety Limits

All reactor licensees are required by 10 CFR 50.36(c) to specify safety limits in the TS. Safety limits are placed on important process variables identified in the facility as necessary to reasonably protect the integrity of the primary reactor system barrier against the uncontrolled release of radioactivity.

For the MSRR, the radioactivity of concern is generally the fission products in the fuel salt. For the MSRR, this primary reactor system barrier is the surfaces that could come in contact with fuel salt or gaseous fission products

There are four safety limits for the MSRR: maximum reactor system temperature, minimum reactor system temperature, maximum reactor system pressure, and minimum reactor system pressure. Reactor system temperature must remain below the safety limit to ensure that safety related steel components remain within ASME Code allowable values. Reactor system pressure must remain below design limits to

ensure integrity of valves and seals. Reactor system pressure must remain within high and low operating pressure limits to restrict void fraction in the fuel salt and limit possible reactivity insertions through void collapse.

Values for the safety limits will be submitted in the Operating License application.

14.2.1.1 Important Process Variables

The important process variables related to the safety limits are reactor system temperature and pressure. Controlling reactor system temperature is important to keep the materials in reactor system barriers within operating limits to help prevent failure. Controlling reactor system pressure is important because high pressure can exceed reactor system material failure limits and lead to failure of the reactor system barrier. The reactor system temperature will be monitored through combined readings from multiple locations in the fuel salt. The pressure will be monitored through measurement of pressure in multiple gas spaces within the reactor system.

14.2.2 Limiting Safety System Settings

Limiting Safety System Settings (LSSS) for nuclear reactors are settings for automatic protective devices related to important process variables having a significant safety function. The MSRR has four LSSS that result in the protective trip of the reactor; maximum temperature of the reactor system, minimum temperature of the reactor system, maximum pressure of the reactor system, and minimum pressure of the reactor system. The safety limits are protected by monitoring and tripping the reactor if the set points of these parameters are exceeded.

The calculated setpoints for these protective actions, providing the minimum acceptable safety margin considering process uncertainty, overall measurement uncertainty, and the transient phenomena of the process instrumentation will be presented in the Operating License application. Because the LSSSs are analytical limits, the protective channels may be set to actuate at more conservative values. The more conservative values may be established as Limiting Conditions for Operation (LCOs).

14.3 Limiting Conditions for Operations

The LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the MSRR facility. Possible topics for LCOs are derived from the safety analyses, which provide the bases for the LCOs. The LCOs are implemented administratively or by control and monitoring circuitry to ensure the MSRR is not damaged, the MSRR is capable of performing its intended function, and that no one suffers undue radiological exposures because of MSRR operations. Final LCO topics and LCO setpoints will be presented in the Operating License application.

The following proposed LCO parameters will be monitored to assure the topics identified below remain within acceptable bounds during operation. The parameters may address multiple topics. If there is not a parameter identified for a particular topic, then no LCO is anticipated for that topic.

- Control rod insertion depth
- Control rod insertion speed
- Fuel salt temperature
- DCS neutron detectors for monitoring power level
- O_2 and Cr concentration in fuel salt
- UF_3 / UF_4 ratio in fuel salt
- Fuel salt level in the RAV
- Reactor system cover gas pressure
- Reactor enclosure pressure
- radioactivity in the reactor enclosure
- radioactivity in the AHRS cooling air
- radioactivity in the coolant salt system
- Gas Management System Hydrogen concentration
- Gas Management System activity
- radioactivity in the SERC bay
- radioactivity in the exhaust stack
- O_2 and Cr concentration in the coolant salt

Systems that need to be operable for the reactor operation:

- Operability of DCS
- Operability of RPS
- Operability of ESFAS
- Operability of REMS
- Operability of RCS
- Operability of electrical power systems
- Operability of Reactor System
- Operability of ESFs
- Operability of primary heat removal system
- Operability of AHRS

14.3.1 Fuel Salt and Fuel System Boundary Parameters

Excess Reactivity

To control excess reactivity in the reactor, fuel salt temperature, pressure, void fraction, and control rod reactivity worth are maintained within design basis.

Fuel Composition Changes

Fuel salt chemistry and rate of chemical attack on the reactor system is monitored by fuel salt sampling, off-gas sampling, and redox potential probes. The composition attributes to be monitored, and the method and timing of monitoring will be submitted with the Operating License application.

Reactivity Coefficients

Because reactivity coefficients cannot vary unacceptably with reactor operation, the MSRR does not require TS for reactivity coefficients

Fuel Salt Level Limits

Fuel salt level in the RAV is monitored to confirm the level remains within design basis bounds. Fuel salt levels outside the acceptable limits immediately triggers the protective action of shutting down the reactor by draining the fuel salt into the reactor drain tank.

Detection of Leakage or Loss of Fuel Salt

In addition to monitoring the fuel salt level in the RAV, the reactor enclosure and primary cooling system are monitored for indication of fuel salt leakage.

Detection of Fission Product Activity

The pressure within the reactor enclosure and radiation content of cooling air removed from engineering safety features is monitored.

Hydrogen Concentration (Gas Management System) Limits

Composition of the gases in the gas management system is monitored for constituents that indicate the system is outside the design basis.

14.3.2 Reactor Control and Safety Systems

Operable Control Rods

The MSRR has three control rods used to control reactivity during reactor operation. They are not used to scram the reactor. Draining the fuel salt into the reactor drain tank accomplishes reactor shut down. The reactor drain tank and control rods are operated within the design basis as specified by the TS.

Reactivity Insertion Rates

The maximum rates of adding positive reactivity are determined for the control rods. The TS ensure the reactor control system remains within design basis.

Reactor Protection System

The preliminary RPS is described in Chapter 7; the final design will be submitted with the Operating License application

Instrumentation and Control Requirements for Operation

Preliminary details of the instrumentation and control requirements for operation are given in Chapter 7. The MSRR has redundant and accurate power level monitors that cover the range from subcritical source multiplication to above the full power level. The TS ensure the instrumentation and control system remains within the design basis.

Interlocks

The need for required interlocks that inhibit or prevent control system actions will be determined and submitted with the Operating License application.

14.3.3 Primary Cooling and Heat Dissipation Systems

Cooling systems are discussed in Chapter 5 and Chapter 9. Heat from the fuel salt is transferred through a heat exchanger to the coolant salt and then to the environment through a salt to air radiator. On reactor shutdown, decay heat is passively removed from the fuel salt. The MSRR may be operated without a working primary heat removal system with a restriction on reactor power.

The auxiliary heat removal system allows the reactor to operate at any licensed power.

Isolation Valves

Any isolation valves that are safety related will have TS to ensure they remain within the design basis.

Coolant Level Limits

Upper and lower coolant salt levels will be determined as part of the final design submitted with the Operating License application. If these levels are determined to be safety-related, appropriate TS limits will be proposed.

Detection of Leakage or Loss of Coolant Salt

The coolant salt in the cooling system is monitored for indication of coolant salt leakage. The final design will determine when the system will be operable and appropriate actions when it is not operating, and TS will be submitted with the Operating License application.

Detection of Fission Product Activity in the Cooling System

Radiation monitors detect the presence of fission products outside the reactor enclosure. The final design will determine when these systems are required to be operable and appropriate actions in the event they are not operable. Appropriate TS will be submitted with the Operating License application. The proposed TS will provide for prompt detection of fission products escaping from the reactor enclosure or into the secondary cooling system.

Cooling System and Heat Dissipation System Radioactivity Limits

The coolant salt in the cooling system remains within the allowable radioactivity limit with special emphasis on tritium as determined by the design basis. These limits will be determined, and a TS limit on radioactivity and sampling, if necessary, will be submitted with the Operating License application.

Coolant Salt Chemistry Requirements

The control of corrosion within the coolant salt cooling system is important to protecting the primary fission product barrier through the heat exchanger. Coolant salt chemistry and rate of chemical attack are monitored. The composition attributes to be monitored and the method and timing of monitoring will be submitted as proposed TS with the Operating License application.

14.3.4 Functional Containment

The MSRR is housed in a system that has a series of barriers, which are engineered safety features, comprising a functional containment. Details of the design of the functional containment are given in Chapter 6 and Chapter 9. The design basis of the engineered safety features is maintained when required to ensure operability of the functional containment. The reactor enclosure is maintained at a negative pressure and the reactor cell temperature is maintained below operational limits during operation. The reactor cell is maintained isolable under all operating and accident conditions. Proposed TS on system operation and required design features will be submitted as part of the Operating License application.

14.3.5 Ventilation Systems

The heating ventilation and air conditioning system for the MSRR facility is discussed in Chapter 9. Ventilation and exhaust flow rates and the systems are designed to achieve the controlled release of effluents. Proposed TS on system operation and required design features will be submitted as part of the Operating License application.

14.3.6 Radiation Monitoring Systems and Effluents

Monitoring systems and effluents are discussed in Chapter 11 with instrumentation and control of the radiation monitoring system discussed in Chapter 7.

14.3.6.1 Monitoring Systems

The MSRR facility is designed with appropriate area radiation monitors, continuous air monitors, and effluent monitors. The required radiation monitors, the function each performs (e.g., reactor shutdown or containment isolation), the approximate location of each, the type of radiation detected, and the alarm or automatic action setting, will be submitted with the Operating License application along with proposed TS. Setpoints and calibrations are listed in terms of radiation exposure rates and concentrations rather than as count rates that can change with calibration. For specified monitors that become inoperable, the proposed TS state that reactor operations may continue for a limited time to be determined and justified only if the monitor is replaced by a substitute or portable monitor. The replacement monitor performs essentially the same function until the original monitor is repaired or replaced.

Air Monitors (Gas and Particulate)

Monitors are specified in the final design, as needed, for radioactive gas and those radioactive particulates that might be airborne in the research bay. These monitors are capable of alerting facility personnel to the presence of radioactivity.

Fission Product Monitors

The need for fission product monitors and related TS will be determined in the final design and submitted as part of the Operating License application.

Area Monitors

There are area monitors and related TS to be determined by the final design, and will be included in the Operating License application. The type of radiation detected, such as gamma rays or neutrons, is specified. How this system is used to meet the regulatory requirements of 10 CFR 70.24 will be determined. These area monitors give information on the potential exposure rates from reactor-related radiation. Alarm and automatic action setpoints are specified to help ensure personnel exposures and potential doses remain well below limits of 10 CFR 20 and are consistent with the facility as low as reasonably achievable program.

Environmental Monitors

The need for and the details of a TS-required environmental monitoring program will be discussed in the Operating License application.

14.3.6.2 Effluents

Effluents released from the MSRR facility are monitored by the radiation monitoring system to remain within the design basis to ensure compliance with 10 CFR 20 while complying with the facility as low as reasonably achievable program. Proposed TS for effluents will be submitted with the Operating License application.

14.3.7 Experiments

Experimental facilities and the experimental program for the MSRR are discussed in Chapter 10.

The experimental program anticipates the removal of very small fuel salt and gas samples from the reactor. Also, material coupons testing the response to the fuel salt environment may be removed from the bypass line between the pump discharge and the RAV. The impact on reactivity from these sample removals is very small if not undetectable. Technical specification reactivity limits, if needed on experiments, will be submitted with the Operating License application.

It is anticipated that material coupons testing the response to the fuel salt environment are the only experimental materials used with the reactor. Experiments containing fissile materials, potentially corrosive materials, liquid and gas samples, and explosive material are not performed.

It is anticipated that small samples of fuel salt and gases will be removed from the reactor for analysis in the Radiochemistry Laboratory. The design details and basis of the Radiochemistry Laboratory are being developed and potential TS, if needed, that address the failure and malfunction of an experiment will be proposed and submitted with the Operating License application.

14.3.8 Facility-Specific Limiting Conditions for Operations - Fuel Handling System

The MSRR has a fuel handling system used for initial fuel salt loading and for fuel salt unloading at the end of reactor operations before decommissioning. The system has the capability to store fuel salt. Small fuel additions are made using the RAV. Any TS requirements associated with the fuel handling system will be submitted with the Operating License application.

14.4 Surveillance Requirements

Most of the LCOs established in Section 3 of the TS are accompanied by a surveillance requirement in Section 4. The surveillance-related TS clearly identify the parameter or function to be measured or tested as well as the method, the frequency, and the acceptable deviation or error.

If a surveillance is not required for safety while the MSRR is shut down, it may be deferred, but it must be performed before reactor startup. These surveillance requirements are clearly identified. Scheduled surveillances that cannot be performed while the reactor is operating may be deferred until the next planned reactor shutdown. Surveillances that may be deferred and the reasons for deferment are clearly stated in the TSs, and will be justified in the Operating License application and noted in the basis of the specification.

In general, any time a reactor system or component is modified or repaired, the appropriate surveillance for that system is performed as part of the operability check of the system or component. This is done regardless of when the surveillance was last performed or when it is next due.

Surveillance requirements will be submitted as part of the Operating License application.

14.5 Design Features

The TS contain design features. To ensure that the NRC-issued operating license remains valid, TS-required design features cannot be changed without prior NRC review and approval. The aspects of the facility design that will become TS-required design features will be submitted with the Operating License application. Potential TS-required design features are the type and enrichment of the fuel salt, active reactor core and fuel salt configurations, engineered safety features, site features, cooling systems, and fuel salt storage.

14.6 Administrative Controls

Administrative control remains within the safety basis as described in Chapter 12. The TS for administrative controls follows the recommendations of ANS 15.1 (Reference 14.7-2) as appropriate for the MSRR design. The Degradation Management Program will define the design basis of some safety related SSCs. Administrative controls may be established for the Program to assure that the design basis is maintained for the life of the facility. Final administrative TSs will be submitted with the Operating License application.

14.7 References

14.7-1 Oak Ridge National Laboratory, "Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application," ORNL/TM-2020/1478, Oak Ridge, TN.

14.7-2 American National Standards Institute/American Nuclear Society, "The Development of Technical Specifications for Research Reactors," ANSI/ANS-15.1-2007 (R2018).

Chapter 15
Financial Qualifications

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 15 FINANCIAL QUALIFICATIONS 15-1

15.1 Financial Ability to Construct a Non-Power Reactor 15-1

15.1.1 Construction Costs and Fuel Cycle Costs 15-1

15.1.2 Sources of Funds 15-3

15.2 Financial Ability to Operate a Non-Power Reactor 15-4

15.3 Financial Ability to Decommission the Facility 15-5

15.4 Foreign Ownership, Control, or Domination (FOCD) 15-5

15.5 Nuclear Insurance and Indemnity 15-5

15.6 References 15-5

Appendix 15A 15A-1

Appendix 15B 15B-1

LIST OF TABLES

Table 15.1-1 Molten Salt Research Reactor Order-of-Magnitude Cost Estimates 15-4
Table 15.1-2 Molten Salt Research Reactor Overnight Cost Estimate 15-4

CHAPTER 15 FINANCIAL QUALIFICATIONS

This chapter provides financial information establishing that Abilene Christian University (ACU) is financially qualified to own, construct, operate, and decommission the molten salt research reactor (MSRR) facility. The ACU financial information is provided in accordance with 10 CFR 50.33(d)(3), 10 CFR 50.33(f), and with the implementing regulations regarding the Price-Anderson Act contained in 10 CFR Part 140. This information is consistent with guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," and with the Final Interim Staff Guidance Augmenting NUREG-1537, Part 1.

15.1 Financial Ability to Construct a Non-Power Reactor

Abilene Christian University is a non-profit educational institution that has been in continual operation since its founding in 1906 and is accredited by the Southern Association of College and Schools Commission on Colleges to award associate's, bachelor's, master's, and doctoral degrees.

Abilene Christian University also earned Doctoral/Professional University status, as determined by the Carnegie Classification of Institutions of Higher Education in its latest update, released in December 2021. The principal location for ACU business is:

Abilene Christian University
1600 Campus Court
Abilene, TX 79601

Abilene Christian University is an established secondary educational institution pursuing development of world-class, on-campus nuclear research and training opportunities for its faculty, staff, and students in advanced nuclear technologies. The MSRR that ACU seeks to license is foundational in this pursuit.

Abilene Christian University is not acting as an agent or representative of another entity in the filing of this Construction Permit application.

Pursuant to 10 CFR 50.33(f)(1), ACU has provided estimates associated with the total construction of the facility and related fuel costs, as well as funding sources.

15.1.1 Construction Costs and Fuel Cycle Costs

The MSRR will be housed in the Science and Engineering Research Center (SERC), located on the main campus. The SERC is a multiuse facility the NRC staff determined could meet the 10 CFR 50.10(a)(2)(x) exclusion from the definition of "construction" (ML20365A024). The SERC facility funding ($23M) has been committed and considered covered costs, including the estimated $11M for the portion of the SERC to house and operate the MSRR. As such, construction costs for the SERC or the portion for the MSRR will not be included in the preliminary cost estimate. Expenses for anticipated modifications to the SERC and support systems required to facilitate installation and operation of the MSRR, however, are included in this preliminary cost estimate.

Rough order-of-magnitude (ratio) estimates for MSRR overnight costs can be obtained using the cost-capacity equation (Equation 15.1-1) from Peters and Timmerhaus [1]. Cost (C_i), capacity (P_i), and exponents (α) for scaling cost ratios over a wide variety of non-nuclear plants have been developed over the years and range from 0.3 to 1.02, with an overwhelming fraction of α values in the 0.6 to 0.7 range.)

$$C_x = C_y\left(\frac{Px}{Py}\right)^{\alpha} \times inflation \qquad \text{Equation 15.1-1}$$

At best, the accuracy of such an order-of-magnitude estimate is probably limited to -30 to +50 percent for simple system comparisons and worse for large, complicated ones. The accuracy is dependent on the similarity of systems being compared and the historical data to support the exponent used. Accuracy further suffers as the ratio becomes greater. Further, comparisons with other GEN-IV reactors are dependent on construction cost estimates because none have actually been built. The GEN-IV systems, however, are very similar to the MSRR in that they have containment structures far smaller and less expensive than those used in current light-water reactor facilities. Table 15.1-1 contains order-of-magnitude cost estimate ranges for the MSRR based on the University of Texas at Austin (UT) TRIGA Mark II reactor, the Kairos Hermes reactor, the Versatile Test Reactor (VTR), and the Very High Temperature Reactor (VHTR) central cost estimates, using cost capacity exponent endpoints of 0.6 and 0.7 to bracket the projections and adjusting for inflation.

Most of the existing non-power reactors for cost comparisons are TRIGA systems that vary in power, configuration, and capability. The UT TRIGA MARK II reactor provides an interesting initial consideration in a cost discussion for the MSRR. The UT reactor was installed in a research facility constructed by UT that housed the new reactor similarly to the ACU plan to house the MSRR in the multi-purpose SERC facility. The UT reactor is approximately the same physical size and power level of the MSRR, although significant design differences exist.

The Kairos Hermes reactor is a fluoride salt-cooled, high-temperature, low-power test reactor. There are a number of similarities between the MSRR and Hermes, including a high operating temperature, a low-pressure fluoride salt loop-type cooling system, graphite and salt moderation, a novel fuel form, and power level. Primary differences include facility construction, fuel fabrication, and power coupling system costs.

The VTR is a sodium fast test reactor that includes facilities for extensive testing capabilities and research infrastructure. The similarities to the MSRR include a high operating temperature, a low-pressure liquid loop-type cooling system, and the lack of power coupling systems. Primary differences include power level and the facility construction and fuel fabrication costs.

The VHTR is a gas-cooled reactor and is less similar to the MSRR. Similarities include high temperature operations, graphite moderation, and a novel fuel form. Primary differences include power level and the cogeneration coupling, fuel fabrication, and large facility construction costs.

Given the similarities and differences of these systems to the MSRR, these comparisons suggest that the rough order-of-magnitude overnight cost estimate for the MSRR is probably less than $100M.

A detailed overnight cost estimate prepared by the Engineering and Construction Manager for the MSRR project is summarized in Table 15.1-2. The MSRR construction project is separated into five budget categories: licensing activities, design engineering, materials and construction, SERC modifications, and research and development (R&D). The costs for all activities and components in each category are estimated with contingencies ranging from 10 to 100 percent, to arrive at a central overnight cost estimate of $65M with a $30M contingency based on statistical considerations of the sub-component cost estimates and estimated uncertainties.

Given the novelty of the MSRR, ACU is engaged with the Department of Energy Office of Nuclear Energy (DOE-NE) to develop appropriate fuel cycle services for the MSRR, as discussed below. Abilene Christian University will obtain fuel cycle cost estimates from analysis of MSRR operations, consider recent costs for other operating research reactors, and provide the results in the Operating License application.

15.1.2 Sources of Funds

Abilene Christian University has entered into a pair of funding agreements with Natura Resources, LLC, that support the development and construction of the MSRR. The first of these agreements is titled “Sponsored Research and Collaboration Agreement” (SRA), and it covers the costs of conceptual design, R&D, and licensing work required to submit and support during NRC review of the ACU construction permit application. The statement of work under the SRA is funded at the $21.5M level for the activities at ACU. Similar SRAs are in place with university collaborators, including Texas A&M, UT, and Georgia Tech, with the statements of work under those SRAs totaling $9M of activities.

The second of these agreements is titled “Project Management Agreement,” and covers the costs of the MSRR final design, fabrication, construction, Operating License development and submission to NRC, and support of the NRC Operating License review efforts.

ACU has entered into a contract with Teledyne-Brown Engineering to develop the necessary front end engineering design (FEED) work necessary to refine the cost and schedule of the MSRR construction project.

Abilene Christian University informed the DOE-NE early in the project about ACU’s intention to build and operate an MSRR and to develop reasonable assurances that the DOE-NE Research Reactor Infrastructure (RRI) program would provide fuel cycle services for the MSRR free of charge, as it currently does for other university research reactors. In November 2019, ACU received a letter of support from DOE-NE (Appendix 15A) stating the ACU molten salt research reactor will be added to the RRI program listing, and requests for fuel cycle services consistent with appropriations and availability of services will be considered. In January 2022, the DOE assigned a

federal project lead to engage ACU and develop a plan for DOE support of the MSRR fuel cycle and associated licensing aspects. In response to a DOE-NE request at that time, ACU submitted a formal request for MSRR fuel services support (Appendix 15B), which is included in the Construction Permit application. Both ACU and the DOE currently are developing a fuel qualification plan to support the Operating License application.

Table 15.1-1 Molten Salt Research Reactor Order-of-Magnitude Cost Estimates

Reactor System	~Overnight Cost ($M)	Year of Cost Estimate	Inflation Factor	~ Operating Thermal Power Level megawatt (MWth)	2022 1 MW overnight cost estimate
UT TRIGA[1]	25	1986	2.54	1	$63
Kairos Hermes[2]	629	2021	1.03	35	$54-77M
VTR[3]	3600	2019	1.09	350	$72-128M
VHTR[4]	4300	2010	1.24	600	$61-115M

1. Data from conversation with UT TRIGA owner
2. Data from Dec 16, 2020, DOE Advanced Reactor Demonstration Program press release
3. Data from INL presentation [2]
4. Data from 600 MWth NHSS configuration from INL/EXT-11-23282

Table 15.1-2 Molten Salt Research Reactor Overnight Cost Estimate

Budget Category	Overnight Cost Estimate ($M)	Cost Estimate with Contingency
Licensing activities	10	13
Design engineering	30	43
Materials and construction	20	31
SERC modifications	2	3
R&D	3	5
Total	65	95

15.2 Financial Ability to Operate a Non-Power Reactor

Abilene Christian University has reasonable assurance of obtaining the necessary funds to cover facility operation costs for the period of the Operating License. The specific information to address ACU's financial ability to cover estimated operational costs is out of scope in a Construction Permit application. Operational cost estimates and financial resources will be provided to the NRC in the Operating License application, per 10 CFR 50.33(f)(2).

15.3 Financial Ability to Decommission the Facility

Abilene Christian University has reasonable assurance that funds will be available to decommission the MSRR facility in accordance with 10 CFR 5.33(k)(1). This information will be submitted to the NRC in accordance with 10 CFR 50.75(d)(1) as part of the Operating License application.

15.4 Foreign Ownership, Control, or Domination (FOCD)

Abilene Christian University is the applicant for the Construction Permit and subsequent Operating License for the MSRR. Abilene Christian University is an established, non-profit educational institution with a principal place of business in Abilene, Texas, and is not acting as an agent or representative of another person in filing the application. The university is led by a president who responsible to the board of trustees. The president and all members of the board are US citizens; therefore, ownership and control are not dominated by foreign entities or individuals. The Board of Trustees is currently composed of 30 members, all of whom are U.S. citizens.

The principal officers of ACU include the President, Provost, Senior Vice President of Operations, Vice President of Research, and Vice President and General Counsel. All of these officers are U.S. citizens.

15.5 Nuclear Insurance and Indemnity

Abilene Christian University is a non-profit educational institution applicant under 10 CFR 140.71, and as such, is not required to provide nuclear liability insurance. The NRC will indemnify ACU for any claims arising from a nuclear incident under the Price-Anderson Act, Section 170 of the Atomic Energy Act, as amended, and in accordance with the provisions of its indemnity agreement pursuant to 10 CFR 140.95, Appendix E, above $250,000 up to $500 million. Because ACU is not requesting to possess special nuclear material as part of its Construction Permit application, ACU will request the indemnity agreement as part of the Operating License application. Per 10 CFR 50.54(w), ACU is not required to purchase property insurance. If such arrangements are made, however, the details will also be provided in the Operating License application.

15.6 References

15.6-1 M. S. Peters and K. O. Timmerhaus, Plant Design and Economics for Chemical Engineers, 3rd ed., New York: McGraw-Hill, 1980.

15.6-2 K. Pasamehmetoglu, "Versatile Test Reactor," Idaho National Laboratory, 2020.

APPENDIX 15A

NOV 25 2019

Department of Energy
Washington, DC 20585
November 15, 2019

Dr. Philip J. Schubert, President
Abilene Christian University
206 Hardin Administration Building
ACU Box 29100
Abilene, Texas 79699-9100

Dear Dr. Schubert:

The United States Department of Energy (DOE) Office of Nuclear Energy (NE) appreciates the ongoing dialogue with Abilene Christian University (ACU) regarding ACU's plan to build a Molten Salt Research and Test Reactor (MSRTR) and ACU's interest in joining DOE-NE's Research Reactor Infrastructure (RRI) Program.

The Department is committed to supporting U.S. universities and the continued advancement of nuclear technology, through various means including opportunities provided through its Nuclear Energy University Program, Integrated University Program, and the RRI Program.

The DOE-NE is supportive of ACU's intent to establish this MSRTR capability at ACU. Once ACU has a Nuclear Regulatory Commission (NRC) licensed research reactor, they will be added to the RRI program listing of universities with licensed research reactors. At that time, DOE-NE will consider ACU requests for services (e.g. fuel, fuel delivery, fuel disposition) administered through the RRI program. DOE-NE is committed to assessing these requests as well as those of other universities that have NRC licensed research reactors and fulfilling such requests consistent with appropriations and availability of services. Prior to and after achieving an NRC license, ACU is encouraged to continue working with the DOE-NE Molten Salt Reactor (MSR) program office for information and advice.

ACU is encouraged to submit applications to various competitive DOE-NE funding opportunity announcements that ACU believes can assist in their research and development interests and goals.

NOV 25 2019

We look forward to supporting ACU by providing necessary information and guidance as ACU continues to work towards the development of a licensed MSRTR and advancing nuclear technology.

Sincerely,

Alice Caponiti
Acting Deputy Assistant Secretary
for Reactor Fleet and
Advanced Reactor Deployment
Office of Nuclear Energy

APPENDIX 15B

Finding global solutions to the world's most critical needs

Abilene Christian University
Nuclear Energy eXperimental Testing Lab
ACU Box 28208, Abilene, Texas 79699-8208
acunextlab.org

NEXT

Mr. Aaron Gravelle
University Competitive Research Team
Office of Nuclear Energy
Department of Energy
1000 Independence Ave. SW
Washington, DC 20585

Dear Mr. Gravelle,

Abilene Christian University (ACU) is seeking a license from the Nuclear Regulatory Commission (NRC) under section 104c of the Atomic Energy Act (AEA) and pursuant to Part 50 in Chapter 10 of the Code of Federal Regulations (10 CFR Part 50) to design, construct, operate, and decommission a fluoride-based, liquid-fueled Molten Salt Research Reactor (MSRR) on the ACU campus to enhance research and education opportunities in molten salt technology. The MSRR is a first-of-a-kind reactor to be licensed by the NRC, and ACU's goal is to minimize the time to licensure through early NRC engagement and strategic reactor design choices.

ACU has been engaged in pre-licensing activities with the NRC since 2019 and plans to submit the MSRR Construction Permit (CP) application this year. Early Operating License (OL) application development activities are also underway at ACU with the goal of initial criticality by the end of 2025. The MSRR design is predicated on the reactor and experiences gained during the Molten Salt Reactor Experiment (MSRE) and shares many of the design features of the MSRE reactor as part of the strategic approach to minimize time to licensure.

In order for ACU to be successful in constructing, operating, and decommissioning the MSRR, ACU will require support from the DOE, as is the case with all the other US-based university research reactors. However, the MSRR differs substantially from existing university reactors, not only in its design and function, but also because the fuel form is novel and will require an approved fuel qualification effort (in collaboration with ACU and approved by the NRC) that is consistent with MSRR project licensing requirements and timelines.

ACU is requesting assistance from the DOE for the MSRR. This request includes, but may not be limited to:

MSRR Licensure Support

- A letter stating that DOE will provide fuel for the MSRR and support the fuel cycle (CP application requirement)
- Support ACU as it develops and executes a Fuel Qualification (FQ) plan
  - Initial FQ plan to a level of detail required in 10 CFR 50.34(a)8 to support the ACU CP application
  - Detailed FQ plan including R&D execution schedule
  - Execute R&D under an appropriate QA program in collaboration with ACU
- Collaborate with ACU to document a fuel safety case able to be approved by the NRC in support of the ACU OL application

NEXT

MSRR Fluoride Salts Requirements

- Bulk commodity estimates and specifications
  - No more than 500 kg of HALEU with enrichment near 19.75% with minimal oxidative impurities
  - No more than 2200 kg of Li^7F-BeF_2 salt (~66-34 mole %) with lithium enrichment > 99.99% Li^7 and minimal oxidative impurities
- Salt forms and quantities
  - Unfueled Salt
    - No more than 1100 kg of Li^7F-BeF_2 salt
  - Fueled Salts
    - No more than 1100 kg of Li^7F-BeF_2-based salt combined and thoroughly mixed with 90–95% of the required critical mass of HALEU
    - Approximately 50 kg of UF_4 (HALEU) in a form that can be titrated into the reactor to bring the system to criticality and maintain long-term operations.

The final design of the MSRR (described in the OL application) will evolve from the preliminary design (described in the CP application) during the detailed engineering phase of the project, which will begin this year. The MSRR design evolution will certainly sharpen the fuel commodity requirements, but more importantly, will facilitate potential MSRR design modifications as a result of the fuel specification, which directly impact the costs and timelines of the fuel fabrication effort and the MSRR project. It will be important for DOE and ACU to work together closely in order to optimize and manage costs and schedules.

Sincerely,

Dr. Rusty Towell

Director of NEXT Lab

cc: Andrew Griffith, Acting Assistant Secretary for Nuclear Energy
Alice Caponiti, Deputy Assistant Secretary for Reactor Fleet and Advanced Reactor Deployment
Suibel Schuppner, Director, Office of Nuclear Energy Technologies

Chapter 16
Other License Considerations

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 16 OTHER LICENSE CONSIDERATIONS 16-1

16.1 Prior Use of Reactor Components 16-1

16.2 Medical Use of the Facility 16-2

CHAPTER 16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

The Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) will be constructed within the ACU Science and Engineering Research Center (SERC), which is an ACU-owned multi-use facility as described in 10 CFR 50.10(a)(2)(x) and designed as a location to house a variety of potential radiation producing systems for education, research and training. The MSRR will be the first of such systems and will be integrated into a portion of the pre-existing SERC facility as described in Chapter 2. The SERC facility will provide the MSRR facility with a habitable environment for workers and operators, power for operations, emergency fire sprinklers, and a 40-ton crane, which will be used to lower the MSRR reactor components into position in the subterranean systems pit (see Chapter 2) for assembly. The MSRR reactor components (see Chapter 4) will be purpose built at an off-site facility and will not contain any prior-use components in their construction. The weight of the MSRR reactor system will be supported by the systems pit walls and floor (subterranean), and as such, are classified as SSCs in the MSRR facility (see Chapter 3). All other components designed and built to facilitate operations within the SERC, such as the ductwork for the air exchange with the cooling system (see Chapter 5), will not contain any prior-use components in their construction and installation.

ACU does not plan on utilizing safety related SSCs before their incorporation into the MSRR.

ACU has not yet determined a provider for the UF_4 fuel. The preferred source is high assay low enriched uranium from the Experimental Breeder Reactor II (EBR II) via the U.S. Department of Energy. The EBR II used metallic uranium of high enrichment. To be used by the MSRR, the metallic uranium would need to be down blended and recomposed as a salt to make the UF_4. In this sense, the MSRR fuel is “new,” even though it is not derived from enriched uranium that came directly from ore. However, because the uranium was used in the EBR II, it may contain different amounts of U-233 or U-239 than uranium enriched from ore.

ACU has also not determined a provider for the LiF-BeF (Flibe) carrier salt. The preferred source is coolant salt from the Molten Salt Reactor Experiment (MSRE) via the U.S. Department of Energy. The MSRE salt will likely contain some tritium and metal impurities. Purity specifications for the MSRR salt are being developed. The MSRR salt will be purified as part of the preoperational activities before fuel is introduced into the system. Thus, the prior use of the carrier salt will not have an adverse effect on the operation or safety of the MSRR.

If other sources for fuel or Flibe are needed to supply the MSRR, the same purity specifications and preoperational processing will apply. The other sources may or may not have prior use. The operating license application will identify the sources of fuel and Flibe and will describe any prior use.

16.2 Medical Use of the Facility

The MSRR facility does not contain equipment or facilities for medical therapy and will not be used for medical therapy. ACU is not applying for a Class 104a license as describe in 10 CFR 50.21(a).

Chapter 17
Decommissioning and Possession-Only License Amendments

Abilene Christian University Molten Salt Research Reactor Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 17 DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS 17-1

17.1 Decommissioning 17-1

17.2 Possession-Only License Amendment 17-1

CHAPTER 17 DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS

17.1 Decommissioning

A decommissioning report for the facility will be provided with the Operating License application consistent with 10 CFR 50.33(k) and address the content requirements in 10 CFR 50.75(d)(2). Section 15.3 will describe the financial assurances for the availability of funding to support decommissioning.

17.2 Possession-Only License Amendment

This section relates to a possession-only amendment license and is not applicable to the construction and operation phases of the facility.

Chapter 18
Highly Enriched to Low-Enriched Uranium Conversions

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 18 HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS 18-1

CHAPTER 18 HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS

The Abilene Christian University molten salt research reactor fuel is a salt mixture of LiF, BeF_2 and UF_4, using high-assay, low-enriched uranium. The Molten Salt Research Reactor does not have a unique purpose as defined in 10 CFR 50.2 and will meet the requirement of 10 CFR 50.64(b) by not being fueled by highly enriched uranium. The facility may possess small quantities of highly enriched uranium in the form of fission chambers and instrument calibration sources.

Chapter 19
Environmental Review

Abilene Christian University Molten Salt Research Reactor
Preliminary Safety Analysis Report

Revision 2
July 2024

© 2024 Abilene Christian University NEXT Lab

TABLE OF CONTENTS

CHAPTER 19 ENVIRONMENTAL REVIEW 19-1

19.1 Introduction of the Environmental Review 19-1

19.1.1 Purpose and Need for the Proposed Action 19-1

19.1.2 Regulatory Provisions, Permits, and Required Consultations 19-2

19.2 Proposed Action 19-3

19.2.1 Site Location and Layout 19-4

19.2.2 Non-Power Reactor 19-7

19.2.3 Water Consumption and Treatment 19-10

19.2.4 Cooling and Heating Dissipation Systems 19-10

19.2.5 Waste Systems 19-11

19.3 Description of the Affected Environment 19-14

19.3.1 Land Use 19-14

19.3.2 Visual Resources 19-20

19.3.3 Climatology, Air Quality, and Noise 19-20

19.3.4 Geologic Environment 19-25

19.3.5 Water Resources 19-27

19.3.6 Ecological Resources 19-36

19.3.7 Historic and Cultural Resources 19-37

19.3.8 Socioeconomics 19-39

19.3.9 Human Health 19-51

19.4 Impacts of Proposed Construction, Operations, and Decommissioning . . . 19-55

19.4.1 Land Use and Visual Resources 19-55

19.4.2 Visual Resources 19-56

19.4.3 Air Quality and Noise 19-57

19.4.4 Geologic Environment 19-58

19.4.5 Water Resources 19-58

19.4.6 Ecological Resources 19-59

19.4.7 Historic and Cultural Resources 19-60

19.4.8 Socioeconomics 19-60

19.4.9 Human Health 19-62

19.4.10 Waste Management 19-75

19.4.11 Transportation 19-77

TABLE OF CONTENTS

19.4.12 Postulated Accidents 19-77
19.4.13 Environmental Justice 19-82
19.4.14 Cumulative Effects 19-85
19.5 Alternatives 19-99
19.5.1 No-Action Alternative 19-99
19.5.2 Reasonable Alternatives 19-100
19.5.3 Cost-Benefit of the Alternatives 19-103
19.5.4 Comparison of the Potential Environmental Impacts 19-104
19.6 Conclusions 19-105
19.6.1 Unavoidable Adverse Environmental Impacts 19-105
19.6.2 Relationship between Short-Term Uses and Long-Term Productivity of the Environment 19-106
19.6.3 Irreversible and Irretrievable Commitments of Resources 19-107

LIST OF TABLES

Table 19.1-1 Permits and Approvals Required for Construction and Operation of the Molten Salt Research Reactor 19-2
Table 19.1-2 Permits and Approvals Required for Construction of the Science and Engineering Research Center 19-3
Table 19.3-1 National Ambient Air Quality Standards 19-23
Table 19.3-2 Demographic Profiles 19-41
Table 19.3-3 Median Family Income and Per Capita Income (2015-2019) 19-42
Table 19.3-4 People Living Below U.S. Census Poverty Thresholds 19-43
Table 19.3-5 2019 U.S. Federal Poverty Thresholds for Different Family Sizes 19-43
Table 19.3-6 Taylor County Total Housing Units and Vacancy Rates 19-43
Table 19.3-7 Ten Largest Employers in the City of Abilene 19-44
Table 19.3-8 School Districts in the Region of Influence and Their Tax Levies 19-44
Table 19.4-1 Chemicals Stored and Used during Operation 19-71
Table 19.4-2 Potential Occupational Hazards 19-71
Table 19.4-3 Radiation Sources and Locations in the Facility 19-72
Table 19.4-4 Anticipated Radioactive Gaseous Effluent Production and Emissions 19-72
Table 19.4-5 Abilene, Texas Demographic Data (U.S. Census Bureau QuickFacts: Abilene, Texas) 19-84
Table 19.4-6 Abilene Poverty Statistics (LiveStories U.S. Census Bureau QuickFacts: Abilene, Texas) 19-84
Table 19.4-7 Past, Present, and Reasonably Foreseeable Projects and Other Actions Considered in the Cumulative Effects Analysis 19-96
Table 19.4-8 Cumulative Impacts on Environmental Resources, Including the Impacts of the Proposed Project 19-99
Table 19.5-1 Candidate Site Ranking for Environmental Factors 19-104
Table 19.5-2 Candidate Site Ranking for Financial Impact 19-104
Table 19.5-3 Candidate Site Ranking for Mission Impact 19-104

LIST OF FIGURES

Figure 19.2-1 Borders and Major Cities in 200-mi (322-km) Radius from Molten Salt Research Reactor Site . . . 19-5
Figure 19.2-2 Abilene, Texas Area [5 mi (8 km) Radius] . . . 19-6
Figure 19.2-3 Abilene Christian University Science and Engineering Research Center . . . 19-7
Figure 19.2-4 Molten Salt Research Reactor Process Flow Diagram . . . 19-9
Figure 19.3-1 Major Land Uses for the City of Abilene . . . 19-15
Figure 19.3-2 Aerial View of Land Use in Five-Mile (8 km) Radius of Proposed Site . . . 19-16
Figure 19.3-3 Abilene Sensitive Development Areas . . . 19-17
Figure 19.3-4 Planned Land Use for the City of Abilene . . . 19-19
Figure 19.3-5 Abilene, Texas Annual Average Precipitation . . . 19-24
Figure 19.3-6 Abilene, Texas Average Wind Speeds . . . 19-24
Figure 19.3-7 Abilene, Texas Wind Direction . . . 19-25
Figure 19.3-8 Abilene Area Surface Waters . . . 19-28
Figure 19.3-9 Cedar Creek USGS Gaging Station . . . 19-30
Figure 19.3-10 Annual Peak Streamflow . . . 19-31
Figure 19.3-11 Potential Floodways . . . 19-32
Figure 19.3-12 Abilene Area Emergency Services . . . 19-47
Figure 19.3-13 Nearest Sensitive Receptors to the MSRR . . . 19-52
Figure 19.4-1 Molten Salt Research Reactor Site Layout . . . 19-74
Figure 19.4-2 Science and Engineering Research Center First Floor Layout . . . 19-75
Figure 19.4-3 Abilene, Texas Demographics and Population Statistics (NeighborhoodScout) . . . 19-85
Figure 19.5-1 Sites Considered for the MSRR . . . 19-103

CHAPTER 19 ENVIRONMENTAL REVIEW

19.1 Introduction of the Environmental Review

In accordance with the provisions of the regulations in Title 10 of the Code of Federal Regulations (10 CFR) Part 50 and supporting guidance, Abilene Christian University (ACU) is providing this Environmental Review in support of an application to construct and install a non-power research reactor (Molten Salt Research Reactor - MSRR) facility within the ACU Science and Engineering Research Center (SERC) on the campus of ACU in Abilene, Texas. The ACU MSRR is a utilization facility as described in 10 CFR 50.21(c) that is useful in the conduct of research and development activities of the types specified in Section 31 of the Atomic Energy Act of 1954, as amended (AEA), and the activities meet the 10 CFR 50.2 definition of research and development. The MSRR is not a commercial and industrial facility as specified in 10 CFR 50.21(b) or in 10 CFR 50.22.

This Environmental Review is provided with the Construction Permit application as required by 10 CFR 50.30(f). The Environmental Review provides information to the U.S. Nuclear Regulatory Commission (NRC) to facilitate preparation of environmental documentation in accordance with the provisions of 10 CFR Part 51. This chapter provides an assessment of the environmental effects of construction, operation, and decommissioning of the MSRR on the site and surrounding areas.

This Environmental Review follows and is organized consistent with the NRC guidance provided in Final Interim Staff Guidance (ISG) Augmenting NUREG-1537, Part 1, Chapter 19, and supports the regulatory review that is performed by the NRC under 10 CFR Part 51. Although this ISG is specific to medical isotope facilities, it includes guidance for non-power reactors and reflects more recent NRC staff guidance for environmental reports and is useful for other non-power reactor facilities.

This Environmental Review describes the project, potential alternatives, and the methods and sources used in the environmental impact analysis. This Environmental Review also describes the existing environment at the site and vicinity, and summarizes the environmental impacts of construction, operation, and decommissioning.

19.1.1 Purpose and Need for the Proposed Action

The proposed Federal action is the issuance of a Construction Permit under the provisions of 10 CFR Part 50, which allows the construction of a non-power research reactor facility to accelerate the development and deployment of molten salt reactor systems through foundational research while also developing a new pipeline to a nuclear qualified workforce. ACU's capital investment in the MSRR provides a world-class, molten salt research facility to be utilized by large numbers of students, staff, faculty, and outside collaborators. The intended use of the MSRR is to conduct research on molten salt systems, as well as to educate and train a new generation of engineers and scientists who are uniquely prepared to contribute to the advancement and deployment of molten salt reactors and applications. The research generates experimental molten salt reactor data to advance the understanding of the generation and migration of gases and vapors in a fluid-fueled fluoride reactor and the behavior

of delayed neutron precursors during normal and off-normal operating conditions, all of which can be used in the validation and calibration of software for the design, licensing, and regulation of commercial molten salt reactors.

19.1.2 Regulatory Provisions, Permits, and Required Consultations

This review is for an application for the off-site construction and installation of a non-power, research reactor facility within a pre-existing multi-use structure - the SERC. Therefore, many of the environmental considerations and associated permits for on-site construction activities are not applicable. Table 19.1-1 lists the permits required for installation and operation of the MSRR along with the current status of each. Permits required for decommissioning the facility will be provided in the environmental report submitted with the decommissioning plan at the end of facility operation. Although development of the SERC is not part of this proposed action, the permits and approvals for construction of the SERC are provided in Table 19.1-2.

Table 19.1-1 Permits and Approvals Required for Construction and Operation of the Molten Salt Research Reactor

Agency	Regulatory Authority	Permit or Approval	Activity Covered	Status
Nuclear Regulatory Commission	AEA 10 CFR 50.50	Construction Permit	Construction of the facility	Addressed in the Construction Permit application
Nuclear Regulatory Commission	10 CFR 50.57	Operating License	Operation of the facility	To be addressed in the Operating License application
Nuclear Regulatory Commission	10 CFR 40	Source Material License	Receipt, possession, use, and transfer of radioactive source material	To be addressed in the Operating License application
Nuclear Regulatory Commission	10 CFR 30	By-Product Material License	Receipt, possession, use and transfer of radioactive by-product material	To be addressed in the Operating License application
Nuclear Regulatory Commission	10 CFR 70	Special Nuclear Material License	Receipt, possession, use, and transfer of special nuclear material	To be addressed in the Operating License application
Nuclear Regulatory Commission	National Environmental Policy Act (NEPA) 10 CFR 51	Environmental Assessment or Environmental Impact Statement in accordance with NEPA	Approval for construction and operation of a radiation facility	Addressed in this Environmental Report

Table 19.1-1 Permits and Approvals Required for Construction and Operation of the Molten Salt Research Reactor (Continued)

Agency	Regulatory Authority	Permit or Approval	Activity Covered	Status
Environmental Protection Agency	Resource Conservation and Recovery Act 40 CFR 261 and 40 CFR 262	Acknowledgment of Notification of Hazardous Waste Activity	Generation of hazardous waste	Notification not yet submitted
Environmental Protection Agency	Clean Water Act 40 CFR 112, Appendix F	Spill Prevention, Control and Countermeasure (SPCC) Plans for Construction and Operation	Storage of oil during construction and operation	SPCC Plans not yet prepared
Department of Transportation	Hazardous Material Transportation Act	Certificate of Registration	Transportation of hazardous materials	Registration application not yet submitted
Texas Department of State Health Services	Texas Radiation Control Act	Shipping Registration	Shipment of low-level radioactive waste	Application not yet submitted
Department of Energy	Nuclear Waste Policy Act of 1982 Research Reactor Infrastructure Program	Contract	Fuel cycle services including disposal of high-level radioactive waste streams and spent nuclear fuel	Contract not yet awarded

Table 19.1-2 Permits and Approvals Required for Construction of the Science and Engineering Research Center

Agency	Permit or Approval	Activity Covered	Status
City of Abilene, Texas	New Building Permit	Construction and inspection of the SERC	Permit #21-002361 issued March 4, 2022
Texas Department of Licensing and Regulation	Registration	Architectural barriers (accessibility) review and inspection	Registration complete May 12, 2021 Project #: TABS2021015539

19.2 Proposed Action

The proposed federal action is issuance of a Construction Permit for the MSRR to be installed in the SERC. Connected activities to the issuance of a Construction Permit discussed in this Environmental Review are operation and decommissioning of the MSRR. This non-power, research reactor is to provide educational, research, and training opportunities for faculty, staff, and students. The SERC is an existing facility that houses the MSRR.

The applicant for this Construction Permit and owner of the MSRR is ACU, a private, non-profit, educational institution in operation since 1906 and not created for the purpose of designing, constructing, and operating the MSRR described herein. Additional information about the ACU organization and key personnel is provided in Section 12.1.

19.2.1 Site Location and Layout

The proposed MSRR is part of the SERC on the ACU campus in Abilene, Texas, at a latitude of 32° 27’ 53” N, and a longitude of 99° 42’ 26” W. The Universal Transverse Mercator coordinates are in Zone TXNC-4202 at Northing 6,854,150.457 (usft) and Easting 1,596,177.717 (usft).

The SERC site lies on the southeast corner of the ACU campus in the City of Abilene, which is located 150 miles (mi) [241 kilometers (km)] west of Fort Worth, in West Central Texas as shown in Figure 19.2-1. Part of the low rolling plains of west Texas, the Abilene area is flat to slightly rolling. The region is dotted with small bodies of water along with a few large reservoirs and natural lakes. Most of the city lies in Taylor County, but its northernmost portion, which consists mostly of the 4200-acre [1700-hectare (ha)] reservoir, Lake Fort Phantom Hill, extends into Jones County (Figure 19.2-2). The site location is approximately 8 mi (12.9 km) south of Lake Fort Phantom Hill.

The SERC site encompasses approximately 15.17 acres (6.14 ha) of land. The northern border of the site is East North 16th Street and the eastern border is North Judge Ely Boulevard. The building in which the MSRR is located covers approximately 25,000 square feet (2323 square meters). The SERC location is shown in Figure 19.2-3.

A road connecting the SERC site has been constructed extending west from the main entrance along North Judge Ely Boulevard and connecting to the northern parking lot. The nearest major intersection to the MSRR is approximately 470 feet (ft) [143 meters (m)] north of the facility main entrance, and it is the intersection of East North 16th Street and North Judge Ely Boulevard. The main transportation ways in the vicinity include Interstate 20, which passes from the northwest to the southeast roughly 1 mi (1.6 km) north of ACU. Texas State 351 merges with Interstate 20 to the north of ACU. Approximately 3.8 mi (6.11 km) to the south-southeast of the SERC is Abilene Regional Airport.

The facility uses municipal water from the City of Abilene. There are water lines, sanitary water lines, and fire water lines connecting to the SERC. The installation of wastewater mains conforms to Texas Administrative Code, Title 30, Part 1, Chapter 217, Design Criteria for Domestic Wastewater Systems. This installation also conforms to Chapter 290, Public Drinking Water. This water supply is used during construction, operations, and decommissioning.

Figure 19.2-1 Borders and Major Cities in 200-mi (322-km) Radius from Molten Salt Research Reactor Site

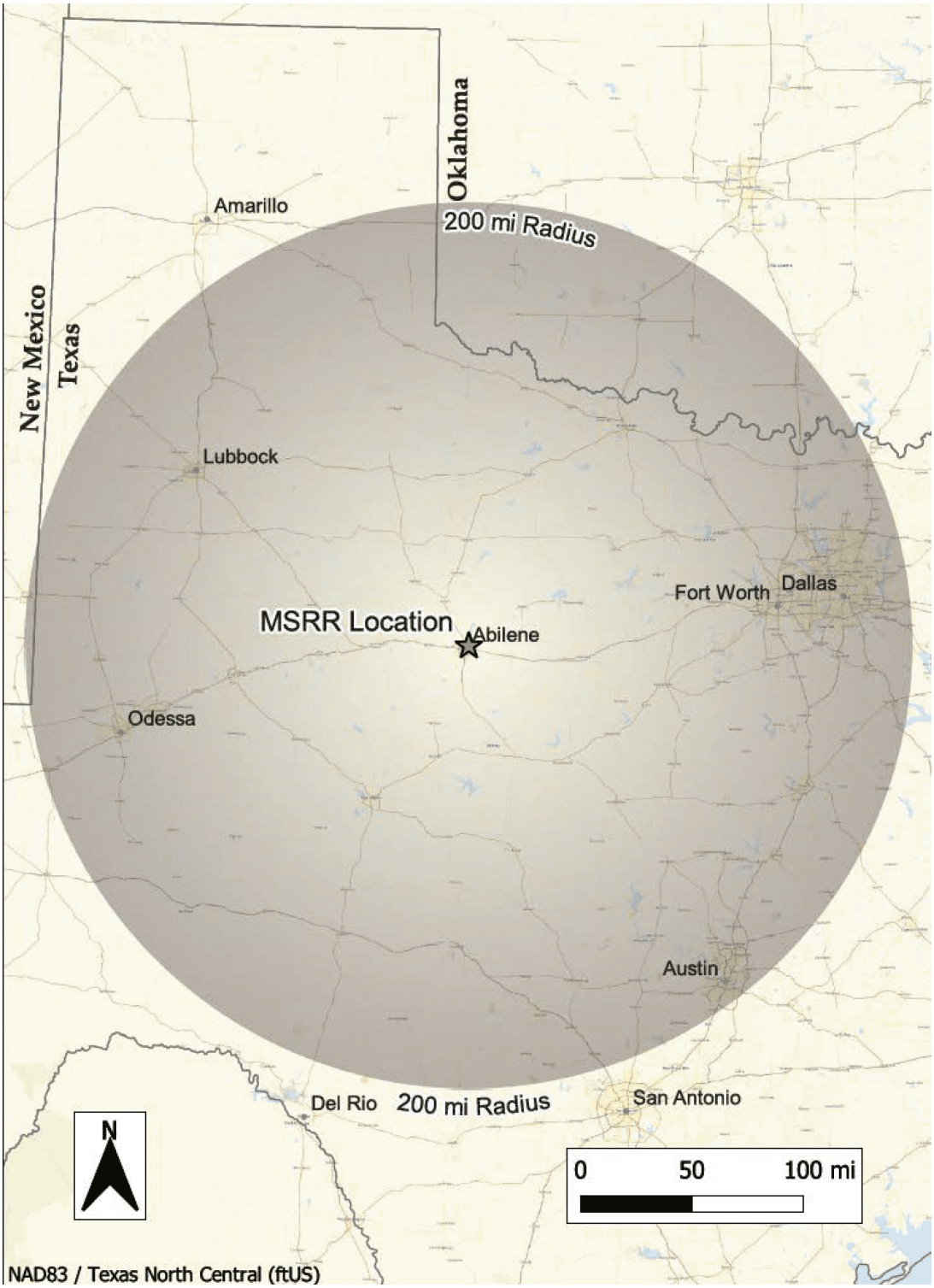


Figure 19.2-2 Abilene, Texas Area [5 mi (8 km) Radius]

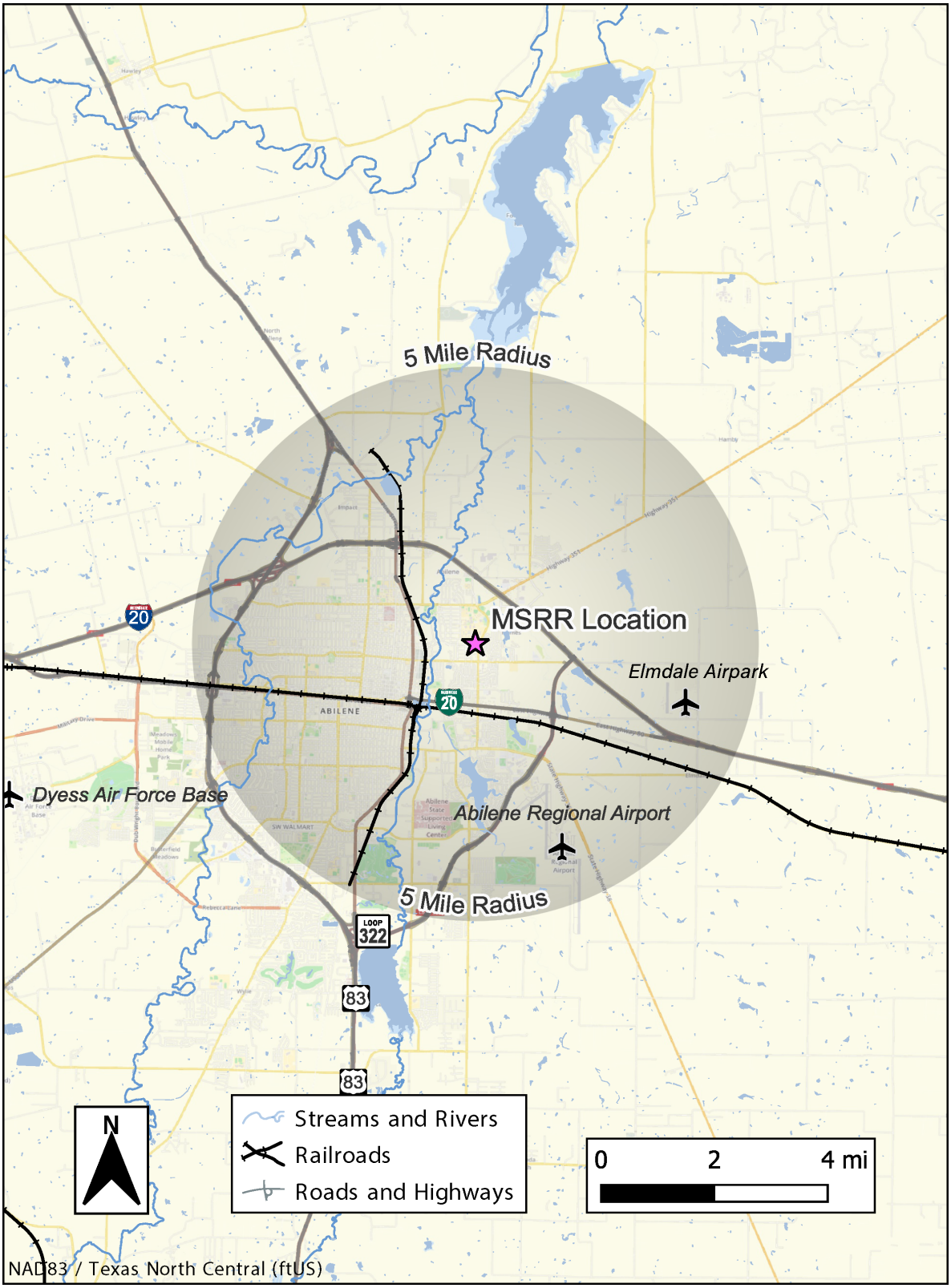
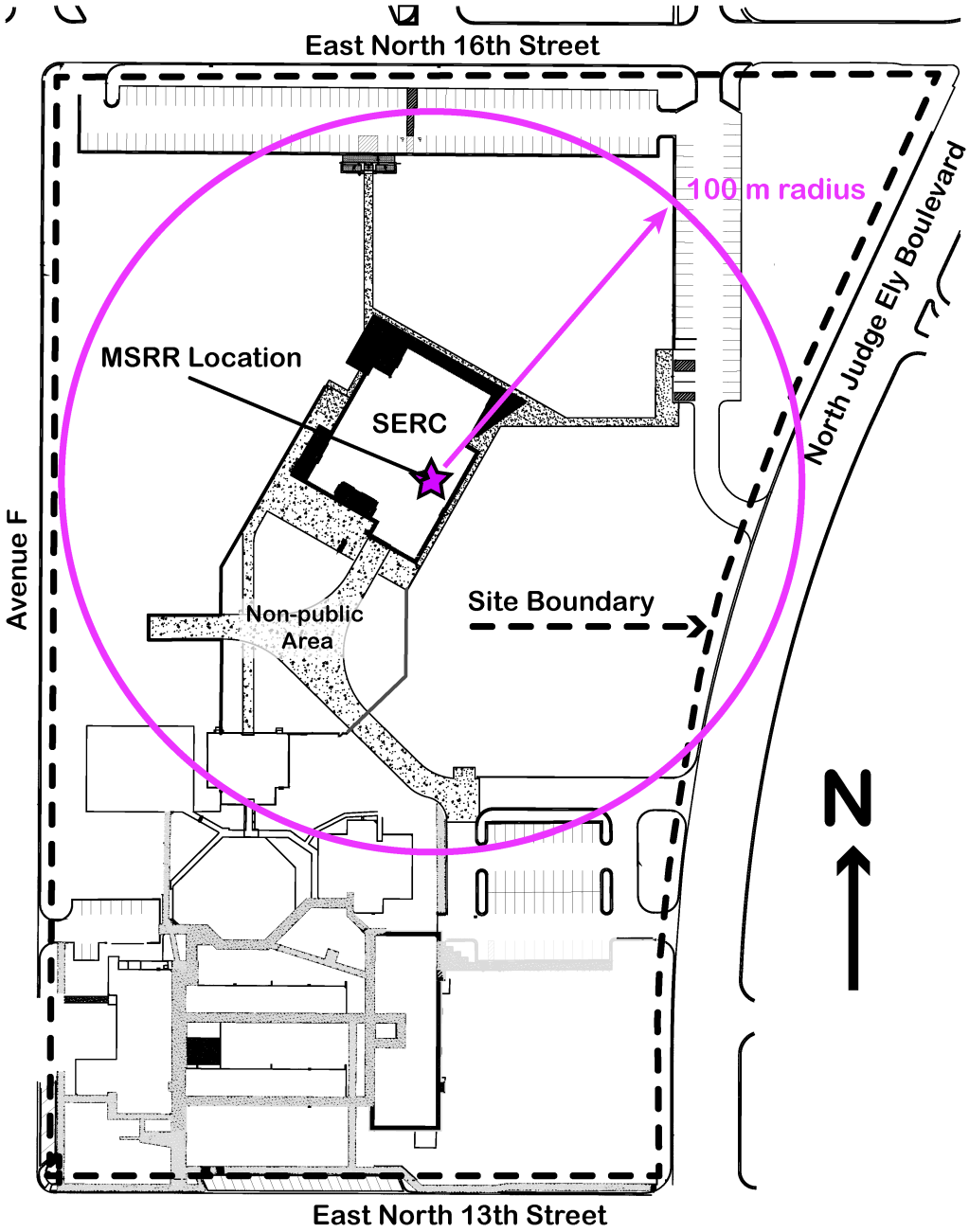


Figure 19.2-3 Abilene Christian University Science and Engineering Research Center



19.2.2 Non-Power Reactor

The facility contains one research reactor. The MSRR will be a 1 megawatt thermal power (MWth) molten fluoride salt reactor with uranium tetrafluoride dissolved in the salt. It is physically located within the SERC research bay. The nominal fuel salt composition is (67.2%)LiF-(27.8%)BeF_2-(5%)UF_4. The final composition is developed by ACU under the Nuclear Energy eXperimental Testing (NEXT) Lab Quality Assurance program with the involvement of the Department of Energy (DOE) and will

be reported in the Operating License application. The uranium is anticipated to be approximately 19.75 percent enriched in U-235 and the lithium enriched to be greater than 99.99 percent ^{7}Li. The MSRR is a loop-type reactor. Fuel salt actively fissions within a vessel that contains a graphite moderator. The fuel salt then flows into a reactor access vessel (RAV), circulated by the reactor pump. After the reactor pump, fuel salt is pushed into the heat exchanger and then back to the reactor vessel through the cold leg. The cold leg from the heat exchanger to the reactor vessel connects to the drain tank. The drain tank connects to the fuel handling system and the gas handling system. Figure 19.2-4 shows the process flow diagram for the reactor and heat dissipation.

The reactor vessel is the only location within the MSRR where fuel salt actively fissions and can achieve criticality. All components that contain salt or fission gases are constructed of stainless steel 316H.

The intended use of the MSRR is to conduct research on molten salt systems, as well as to educate and train a new generation of engineers and scientists who are prepared to contribute to the advancement and deployment of molten salt reactors and applications. It is a non-power research reactor, in which heat generated in the core is not used for production of electric power. Thermal energy generated by fission is ultimately rejected to the atmosphere through a coolant salt loop. The coolant salt ($67LiF\text{-}33BeF_2$) flows through an enclosed loop consisting of the shell side of the heat exchanger, the coolant pump, and the radiator. The coolant salt cools the fuel salt and transfers heat from the loop to the atmosphere through a radiator. Control rods, located in the reactor vessel, are used to provide a level of reactivity control. The reactor is brought to a safe shutdown state through passive draining of fuel salt into a highly subcritical configuration within the reactor drain tank below the reactor vessel.

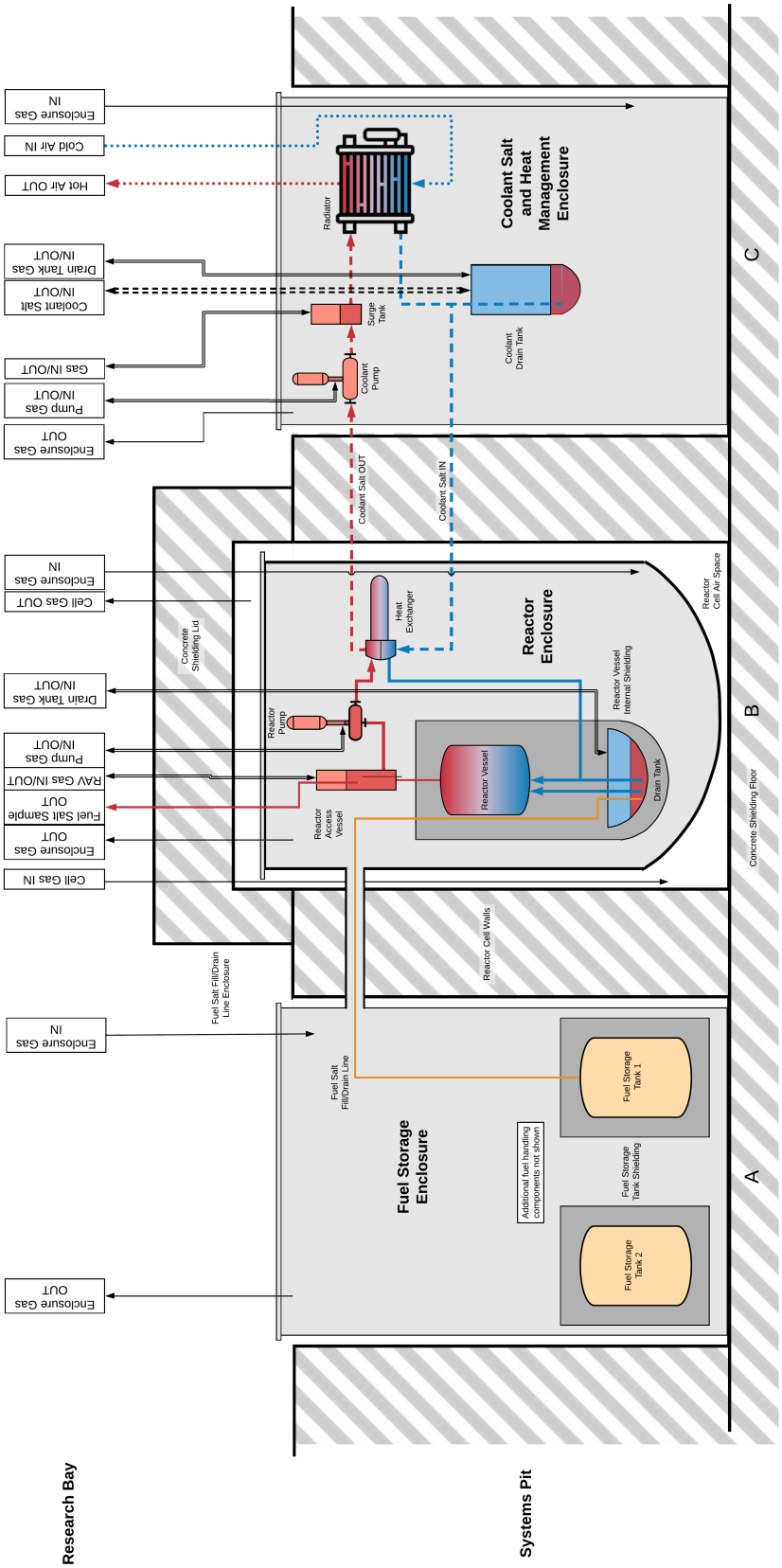


Figure 19.2-4 Molten Salt Research Reactor Process Flow Diagram

19.2.3 Water Consumption and Treatment

No water beyond that consumed or used for sanitary purposes by workers is expected to be used during construction and installation of the MSRR. During operations, water is used for drinking and sanitary systems and for facility heating and cooling, fire suppression, and industrial purposes. A water-based fire-protection system is used throughout the SERC facility. The fire suppression system provides between 0.1 and 0.2 gallons per minute (0.4 to 0.8 liters per minute) over 1500 square feet (487 square meters), depending on the area classification according to NFPA 13.

There are no water treatment systems at the facility as there is ample potable water available from the municipality.

All wastewater generated outside the radiologically controlled area is discharged directly to the City of Abilene wastewater collection system, which conveys it to the Abilene Hamby Water Reclamation Facility. All industrial or wastewater generated in the radiologically controlled area either is evaporated or solidified and disposed of in accordance with the SERC waste management plan and applicable laws and regulations.

19.2.4 Cooling and Heating Dissipation Systems

The MSRR primary heat removal system consists of the reactor loop, which contains fuel salt and transfers heat to a coolant loop, where it is ultimately dissipated to the air through a radiator. Fluid (fuel salt) in the reactor loop travels between the reactor vessel, the RAV, the reactor pump, and the shell side of the heat exchanger. Fluid (salt) in the coolant loop travels between the tube side of the heat exchanger and the air-cooled radiator. A diagram of heat removal pathways during normal reactor operation at 1 MWth is shown in Figure 19.2-4.

The reactor loop is physically located inside the reactor enclosure while the coolant loop penetrates the reactor enclosure. The reactor drain tank below the reactor system allows fuel salt to drain at shutdown.

The coolant loop is contained mainly within the secondary enclosure but extends into the heat exchanger. It consists of the tube side of the heat exchanger, coolant loop pump, radiator, and connective piping. A drain tank in the coolant loop underneath the main loop stores coolant after shutdown.

Fuel salt within the reactor loop and salt in the coolant loop flows under a forced-convection/pumped-flow regime. In the event of pump failure, the positioning of the components allows for some natural convection in both loops.

Heat exchange in the reactor system occurs through tube walls within the shell-and-tube heat exchanger and temperatures remain steady because of strong thermal neutronic feedback.

Similarly, the coolant loop relies on heat transfer through a radiator. A tube bundle containing coolant salt is exposed to air that is actively exchanged with the outside environment.

Both the heat exchanger and the coolant loop radiator are sized to provide a continuous heat removal of 1 MW. Air to cool the radiator is pulled from outside, flowing across the radiator, and back to the outside environment.

The key components in the coolant loop are the heat exchanger, radiator, and secondary air-exchange system. Positive pressure is maintained in the coolant loop relative to the reactor loop during steady-state operation. While the thickness of the heat exchanger tube walls normally separates the two loops, a negative pressure gradient is maintained in the reactor loop relative to the coolant loop to ensure that in case of leakage between the two, fuel salt is contained in the reactor loop.

Cooling systems also control building ambient air temperature in association with heating, ventilation, and air conditioning (HVAC) needs. Two HVAC systems serve the MSRR facility, a dedicated system for the research bay, and a separate system for the control room and Radiochemistry Laboratory. The systems use air handling units to move air across air-to-water heat exchangers in which coils are supplied with hot or cold water. The systems share two on-site boilers to supply hot water with capacity and temperature setpoints determined by the facility HVAC requirements, and two on-site water chillers with temperature setpoints determined by facility cooling requirements. Local air handling controls provide an air pressure differential so pressure in the MSRR facility is lower than the atmosphere, and pressure in the research bay is lower than the rest of the building.

19.2.5 Waste Systems

Construction, operations, and decommissioning result in the accumulation of radioactive and nonradioactive wastes. The MSRR anticipates some long-term storage of radioactive and nonradioactive materials. The MSRR treats and temporarily stores the solid radioactive and nonradioactive waste generated as a part of operation processes within the facility until it can ship the waste off site for disposal. The information below details the generation, storage, waste management activities, waste minimization, pollution prevention measures, and transportation of radioactive and nonradioactive waste.

19.2.5.1 Radioactive Waste

Operation of the MSRR generates liquid, solid, and/or gaseous radioactive waste during the following activities:

- Reactor operations
- Experiment byproducts (e.g., material coupons)
- Laboratory activities (e.g., gloves, pipette tips)
- Maintenance activities
- Decontamination and decommissioning activities

Gaseous radioactive waste is not anticipated to be produced in significant quantities in the MSRR facility. Radioactive gases are anticipated to be released as effluent as part of normal operations. Released gases will be monitored and controlled to ensure that total dose to the public does not exceed applicable regulations.

Liquid radioactive waste can be generated from use of the decontamination showers and eye wash. If used to decontaminate personnel, the liquid is assumed to be radioactive and treated procedurally. Additionally, liquid radioactive waste generated from radiochemical operations can be a mixed chemical and radioactive hazard.

Solid radioactive waste at the MSRR facility is primarily generated by operation of the reactor, either as a byproduct of experiments, such as material coupons, or from maintenance, such as reactor structural components and tools. Additional radioactive waste is produced by laboratory activities, such as contaminated gloves or pipette tips. Solid radioactive waste is packaged to be stored temporarily onsite until appropriate disposal can be organized for the licensing status of the material, its chemical form, and its radioactivity (or lack thereof) at the time of disposal. Solid radioactive wastes also include sorbing media such as off-gas charcoal, ion exchange resins, and air filters. It is anticipated that solid waste during decommissioning will be packaged and sent for disposal similar to solid wastes generated during operation.

19.2.5.2 Nonradioactive Wastes

The MSRR generates nonradioactive waste as part of construction, installation, routine operations, maintenance, cleaning, and decommissioning activities.

Nonradioactive Liquid Waste

Nonradioactive liquid waste is generated during construction. For example, lubricating oil, hydraulic oil, and grease can be necessary to assemble various pieces of equipment and systems. During operations, nonradioactive liquid waste includes hazardous waste, such as certain chemicals. A list of chemicals used during facility operation is provided in Section 19.4.9. Some chemicals are in liquid form, and are controlled and confined in containers, tanks, and pipes.

The MSRR releases small amounts of nonradioactive chemicals into the city sewer system as a result of routine facility maintenance activities and routine laboratory analytical procedures using chemicals. The MSRR has administrative controls in place to ensure its nonradioactive effluents meet requirements pertaining to the types, quantity, and concentrations specified as acceptable for processing by the City of Abilene wastewater treatment facility. Additionally, sanitary wastewater from the SERC facility is sent to the City of Abilene wastewater treatment facility for treatment and disposal. The MSRR does not intend to treat or permanently store hazardous wastes on site. The MSRR disposes of hazardous wastes generated at the facility at a licensed hazardous waste disposal site. Nonradioactive liquid waste during decommissioning will be treated similar to waste during operation.

Nonradioactive Solid Waste

During construction, operations, and decommissioning, MSRR expects to generate the following nonradioactive wastes:

- Wood from crates
- Packaging from receiving activities
- Used personal protective equipment
- Broken mechanical parts
- Metal shavings
- Piping
- Wires
- Batteries
- Air filters
- Expired lights and fixtures
- Paper
- Hoses
- Empty plastic containers
- Expired ink cartridges

The MSRR temporarily collects and stores non-radioactive solid wastes on site and then transports them off site to either a landfill, a storage facility, or a recycling facility. For example, scrap metal, batteries, mercury-containing equipment and bulbs, and used oil are collected and stored temporarily and then recycled or recovered at an offsite permitted recycling or recovery facility, as appropriate.

19.2.5.3 Waste Minimization and Pollution Prevention Program

The MSRR radioactive and nonradioactive waste management program is based on a pollution-prevention and waste-minimization framework that augments ACU's existing program. The program includes:

- Waste minimization and recycling
- Employee training and education on general environmental activities and hazards associated with the facility, operations, and the pollution prevention program; and waste minimization requirements, goals, and accomplishments
- Employee training and education on specific environmental requirements and issues
- Designation of employees responsible for pollution prevention and waste minimization
- Recognition of employees for efforts to improve environmental conditions

- Requirements for employees to consider pollution prevention and waste minimization in day-to-day activities and engineering

19.3 Description of the Affected Environment

19.3.1 Land Use

This section describes the characteristics of the land use and the region. A description of the visual resources of the site is provided in Section 19.3.2.

19.3.1.1 Site

The MSRR is contained within the ACU SERC, a multipurpose facility designed to house the expanding NEXT Lab activities. The SERC site includes 15.17 acres (6.14 ha) of land located in the southeast corner of ACU main campus (see Figure 19.2-3) and is approximately 1.5 mi (2.41 km) from Abilene City Hall.

19.3.1.2 Vicinity

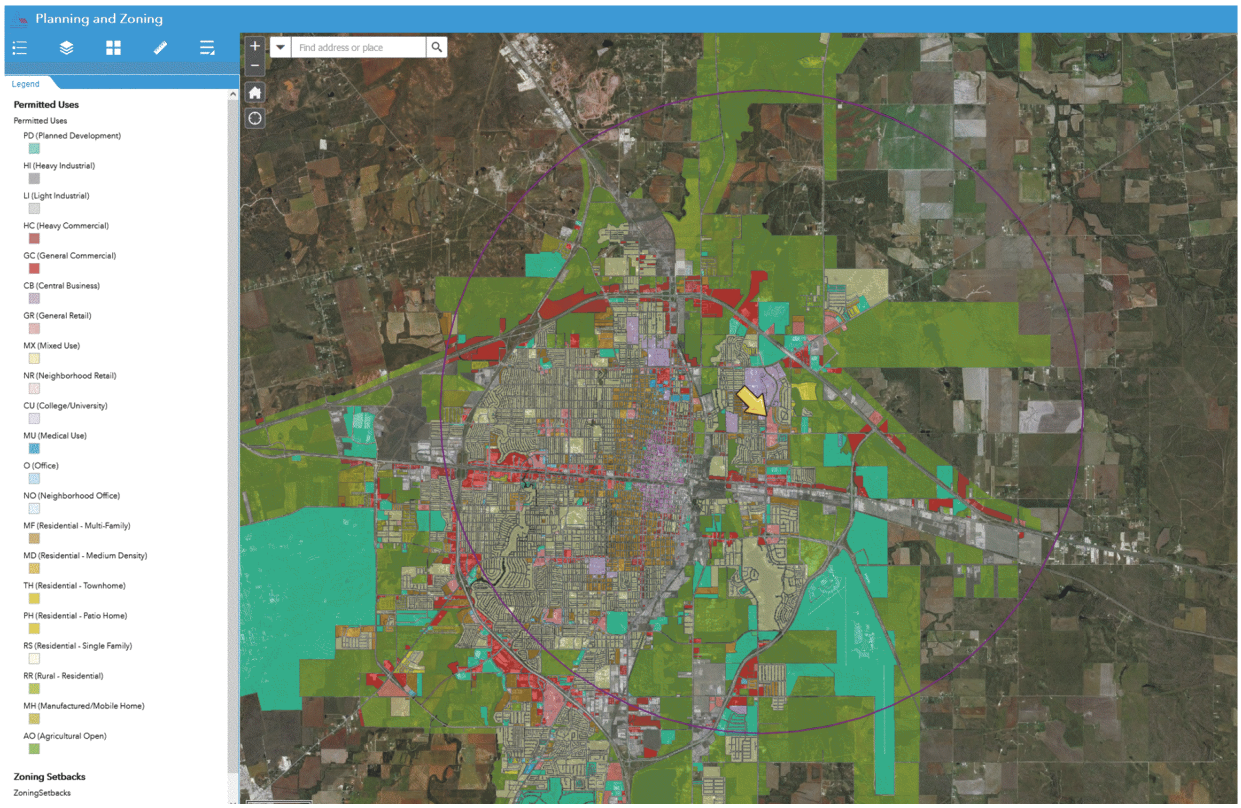
Figure 19.3-1 lists the major land uses within a 5-mi (8-km) radius. To the south and west of the site are mainly low-density, single-family neighborhoods with the occasional restaurant or business spread throughout. Directly north is ACU, which has a total student population of approximately 3,500. North of ACU are a few neighborhoods that quickly transition into open fields the farther north one travels until reaching Interstate-20. The east and southeast are dominated by open fields with some being used for agriculture. Nearly the entirety of the city of Abilene can be found within 5 mi (8 km) of the proposed site, which includes a population of 123,420 as of the 2019 census.

19.3.1.3 Major Land Uses

Land use in the Abilene, Texas region is shown in Figure 19.3-1.

The main transportation routes in the vicinity include Interstate 20, which passes from northwest to southeast roughly 1 mi (1.6 km) north of ACU. Texas State 351 merges with Interstate 20 to the north of ACU. Abilene Regional Airport is located 3.84 mi (6.17 km) to the south-southeast of the proposed site, which accommodated 49,155 flights in the period from 2016-2017.

Figure 19.3-1 Major Land Uses for the City of Abilene



19.3.1.4 Special Land Uses

Special land uses and sensitive development areas within 5 mi (8 km) of the facility are shown in Figure 19.3-2 and Figure 19.3-3. The site within a 5-mi (8 km) radius has low density, single-family housing, which is marked as number 1 in Figure 19.3-2; open fields with some agricultural plots; (number 2), Abilene Regional Airport (number 3), and Dyess Air Force Base (number 4).

“Sensitive development areas,” as determined by the City of Abilene, are shown in Figure 19.3-3. The site for the MSRR does not fall into either region.

Figure 19.3-2 Aerial View of Land Use in Five-Mile (8 km) Radius of Proposed Site

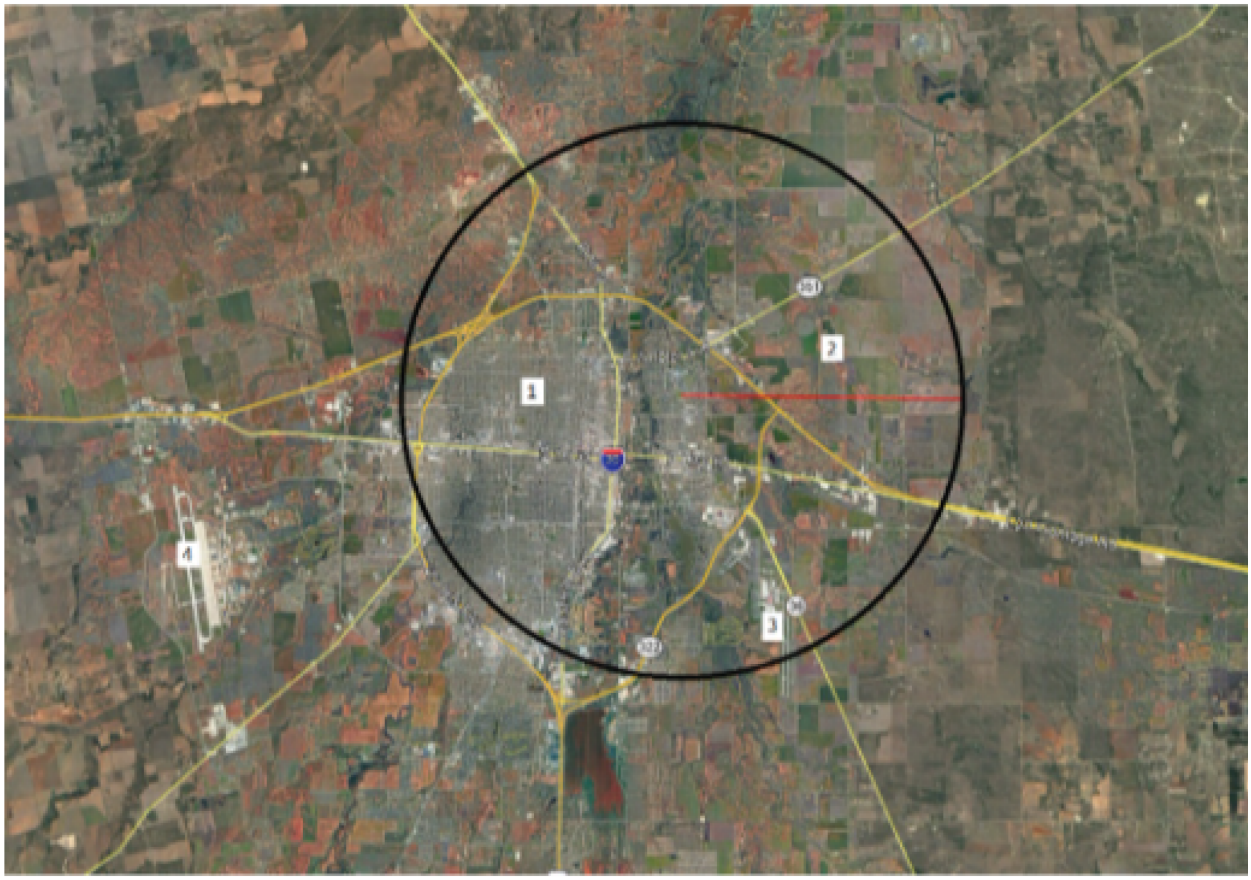
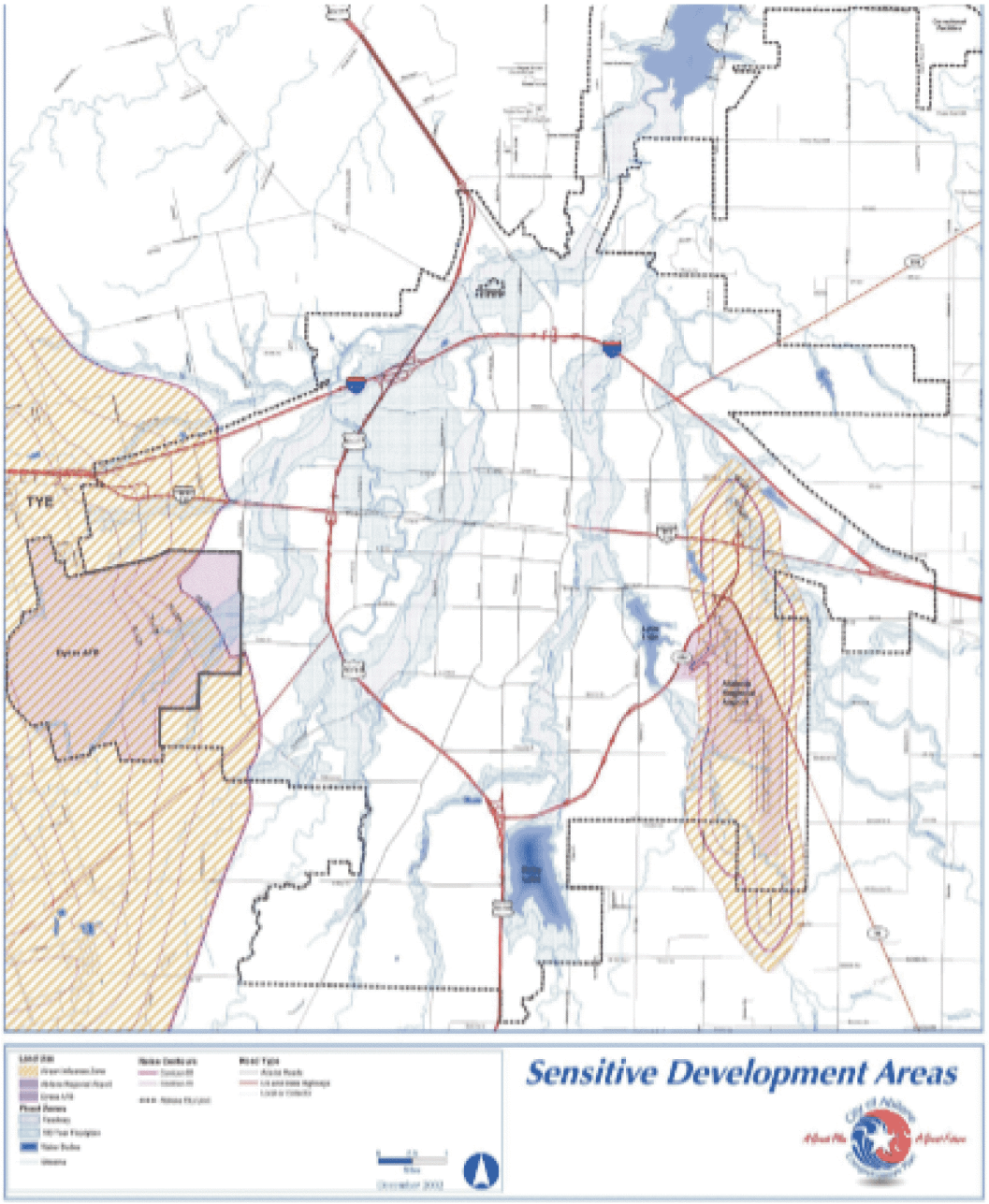


Figure 19.3-3 Abilene Sensitive Development Areas



19.3.1.5 Coastal Impacts

The Coastal Zone Management Act of 1972, as amended (16 U.S.C. 1451 et seq.), was enacted to address increasing pressures of over development upon the nation's coastal resources. Given that no federally-designated coastal zone areas are within 5 mi (8 km) of the SERC site, and that the proposed action does not affect any land or water use or natural resource within the coastal zone, the applicant's certification and the State's concurrence are not applicable.

19.3.1.6 Agricultural Resources and Facilities

The Farmland Protection Policy Act of 1981 (7 U.S.C. 4201 et seq.) and its implementing regulations requires agencies to make evaluations as part of the process under the National Environmental Policy Act of 1969, as amended, to reduce the conversion of farmland to nonagricultural uses by federal projects and programs. Construction of the MSRR does not convert any land with soils that may qualify as prime farmland or farmland of statewide significance to industrial use, cool season grasses, or native prairie. Furthermore, the site is currently zoned for commercial use. Because the SERC site has been committed to urban development and zoned for commercial use, the site does not have farmland soils subject to the Farmland Protection Policy Act.

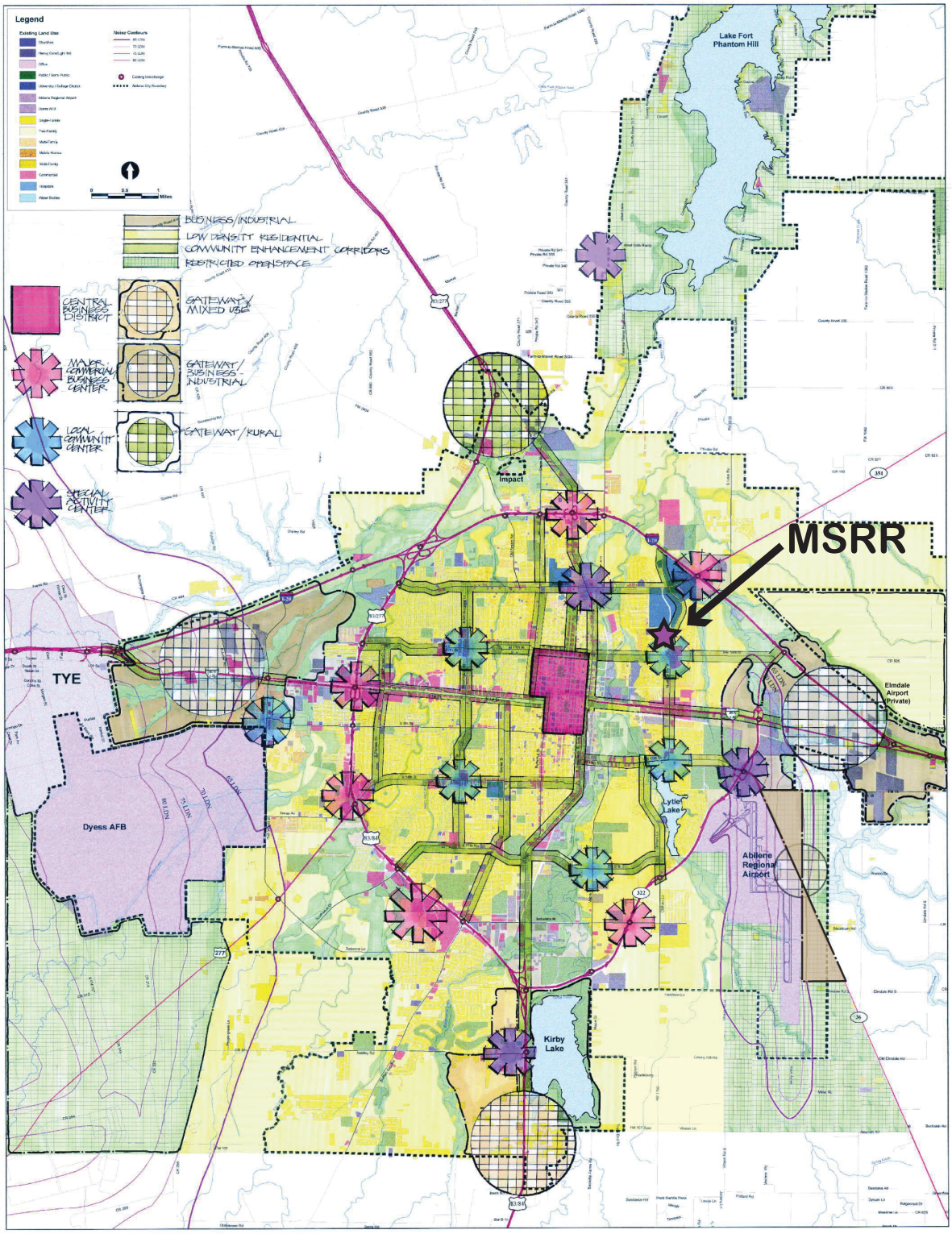
19.3.1.7 Land Use and Plans

As shown in Figure 19.3-4, the future of the surrounding area appears to be unchanged as it remains low-density residential. Also, land-use impacts are confined to the 15.17 acre (6.14 ha) SERC site. Therefore, areas with a special land use or mineral resources are not affected by the construction of the MSRR.

19.3.1.8 References

19.3.1-1 City of Abilene Texas, Planning and Zoning Services, https://abilene.maps.arcgis.com/apps/webappviewer/index.html?id=fc76608ae8394a5cb2b4d8a245262275.

Figure 19.3-4 Planned Land Use for the City of Abilene



19.3.2 Visual Resources

From the north, the affected views are those of the members of ACU, which has a student body and full-time faculty of 5,558. The current closest on-campus buildings are Wessell Hall, which is currently under construction, and Sikes Hall, which is set for renovation. The west and south of the site are designated as single-family residential. The views of a total of 14 houses are directly altered by the SERC building. Addition of the MSRR within the SERC has no visual impact. The site is bordered on the east by North Judge Ely Blvd, so travelers' views are obscured.

The tallest structure present at the SERC is the main building which has a height of 50 ft (15 m). The majority of the main building and the tallest point at the SERC are a light gray color. The exhaust stack is present at the top of the main building, but during operation emissions from the exhaust stack are not visible.

The site has low scenic quality because of a lack of notable features, uniform landform, low vegetation diversity, an absence of water, muted colors, and a commonality within the physiographic province. The site has a low sensitivity rating because it is in an area with low scenic values resulting from a moderate amount of use by viewers, low public interest in changes to the visual quality of the proposed site, a low sensitivity to changes in visual quality by the type of users in the area, and a lack of special natural and wilderness areas. Areas surrounding the SERC vegetate open areas with cool-season grasses or native prairie grasses. Vegetation partially mitigates impacts to visual resources.

19.3.3 Climatology, Air Quality, and Noise

19.3.3.1 Regional Climatology

The proposed site is in Texas low rolling plains (#2) climate division. It is characterized by hot and humid summers, with drier, cool-to-cold winters. Rain falls year-round with more in the summer. There are many small reservoirs in the area, but none large enough to have an impact on climate or provide any moderating effect.

The nearest National Climatic Data Center station is the Abilene Station, 4.9 mi (7.88 km) from the proposed site. Data from the Abilene Station indicate the annual average precipitation is about 15.79 inches (in.) [40.11 centimeters (cm)] of which 3.34 in (8.48 cm) (Reference 19.3.3-1) or about 21 percent of the total falls in summer. The maximum average high temperature ranges between 57 degrees Fahrenheit (14 degrees Celsius) in winter and 95 degrees Fahrenheit (35 degrees Celsius) in the summer, but annually the temperature ranges between 35 to 95 degrees Fahrenheit (2 to 35 degrees Celsius) (Reference 19.3.3-1). See Figure 19.3-5.

Annual average wind speeds are shown in Figure 19.3-6. The average monthly wind speed is 11 miles per hour (mph) [17.7 kilometers per hour (kph)] for windier seasons and drops to 10.2 mph (16.4 kph) for calmer seasons. The windier part of

the year lasts five months, from January 30 to June 28, with average wind speeds of more than 11 mph (17.7 kph). The windiest day of the year is April 2, with an average hourly speed of 12.8 mph (20.6 kph) (Reference 19.3.3-1).

The calmer time of the year lasts for seven months, from June 28 to January 30. The calmest day of the year is August 23, with an average hourly wind speed of 9.1 mph (14.6 kph) (Reference 19.3.3-1).

The predominant average hourly wind direction in Abilene is from the south throughout the year (Figure 19.3-7).

19.3.3.2 Regional Air Quality

In accordance with the Federal Clean Air Act of 1970, as amended (CAA), the Environmental Protection Agency established National Primary and Secondary Ambient Air Quality Standards for six pollutants (often referred to as criteria pollutants) to protect the environment and public health. These pollutants include, ozone, carbon monoxide, nitrogen dioxide, sulfur dioxide, lead, and particulate matter. Particulate matter includes particulates less than 10 microns and less than 2.5 microns, which are particles with equivalent aerodynamic diameters less than or equal to 10 and 2.5 microns, respectively.

Other air pollutants of concern include greenhouse gases, such as carbon dioxide and methane, and hazardous air pollutants as identified in Section 112 of the CAA. The Texas Commission on Environmental Quality (TCEQ) regulates the release of air contaminants in the state of Texas through the Texas Clean Air Act. State rules for regulating hazardous air pollutants are found in the Texas Administrative Code, Title 30, Part 1, Chapter 113, Subchapter B.

National Ambient Air Quality Standards (NAAQS) (Reference 19.3.3-2) limit the concentrations of the six criteria pollutants established to protect human health and welfare. Table 19.3-1 shows the current NAAQS. Areas in which pollutant concentrations exceed these standards are designated nonattainment areas. Attainment areas are areas in which recent monitoring data demonstrate that concentrations are lower than the NAAQS. If monitoring has been insufficient to determine whether an area meets the standards, the area is designated as an unclassifiable area (Reference 19.3.3-3).

The CAA requires development of regulatory plans for nonattainment areas to reduce pollution levels until the area meets the NAAQS within the specified time frame. State agencies typically complete these plans, which are called State Implementation Plans. After air quality has improved in an area to the point that monitoring data meet the NAAQS, the area is designated as a maintenance area.

Air quality designations are generally made at the county level, but designations may also be made for smaller, localized areas for the purpose of planning and maintaining ambient air quality with respect to the NAAQS. The EPA has created Air Quality Control Regions (AQCRs), which are intrastate or interstate areas that

share a common airshed. The site is in the Abilene-Wichita Falls Intrastate ACQR. This region and Taylor County are classified as attainment/unclassifiable for all criteria pollutants.

The nearest nonattainment area is the Dallas-Fort Worth area. The closest Dallas-Fort Worth area county designated as nonattainment is Parker County, approximately 94 mi (151 km) northeast (Reference 19.3.3-3).

19.3.3.3 Noise

Noise is unwanted or unwelcome sound usually caused by human activity that is added to the natural acoustic setting of a locale. It is further defined as sound that disrupts normal activities and diminishes the quality of the environment. The setting of this proposed activity is the ACU campus, an urban/suburban area within the City of Abilene, Texas. Consequently, the natural noise environment of the area is what can be expected in an urban/suburban area (e.g., people, recreation, automobiles, trucks, maintenance activities, and commercial and military aircraft).

The MSRR is installed and operated within the SERC. Consequently, there is little addition to the existing natural noise environment. There is a slight addition because of the additional vehicle use from installation, operation and decommissioning workers driving to work.

19.3.3.4 References

19.3.3-1 Weatherspark. Climate and Average Weather Year Round in Abilene, June 16, 2022, https://weatherspark.com/y/6226/Average-Weather-in-Abilene-Texas-United-States-Year-Round.

19.3.3-2 U.S. Environmental Protection Agency. “NAAQS Table.” Accessed July 9, 2022. https//www.epa.gov/criteria-air-pollutants/naaqs-table.

19.3.3-3 U.S. Environmental Protection Agency. “Green Book.” Accessed July 12, 2022. www3.epa.gov/Green Book/National Area and County-Level Multi-Pollutant Information.

Table 19.3-1 National Ambient Air Quality Standards

Pollutant		Primary/ Secondary	Averaging Time	Level	Form
Carbon monoxide (CO)		Primary	8 hr	9 ppm	Not to be exceeded more than once per year
			1 hr	35 ppm	
Lead (Pb)		Primary and secondary	Rolling 3-month average	0.15 µg/m^3 [a]	Not to be exceeded
Nitrogen dioxide (NO_2)		Primary	1 hr	100 ppb	98th percentile of 1-hour daily maximum concentrations, averaged over 3 years
		Primary and secondary	1 year	53 ppb[b]	Annual Mean
Ozone (O_3)		Primary and secondary	8 hr	0.070 ppm[c]	Annual fourth-highest daily maximum 8-hour concentration, averaged over 3 years
Particle pollution (PM)	$PM_{2.5}$	Primary	1 year	12.0 µg/m^3	Annual mean, averaged over 3 years
		Secondary	1 year	15.0 µg/m^3	Annual mean, averaged over 3 years
		Primary and secondary	24 hr	35 µg/m^3	98th percentile, averaged over 3 years
	PM_{10}	Primary and secondary	24 hr	150 µg/m^3	Not to be exceeded more than once per year on average over 3 years
Sulfur dioxide (SO_2)		Primary	1 hr	75 ppb[d]	99th percentile of 1-hour daily maximum concentrations, averaged over 3 years
		Secondary	3 hr	0.5 ppm	Not to be exceeded more than once per year

a. In areas designated nonattainment for the Pb standards prior to the promulgation of the current (2008) standards, and for which implementation plans to attain or maintain the current standards have not been submitted and approved, the previous standards (1.5 µg/m^3 as a calendar quarter average) also remain in effect.

b. The level of the annual NO_2 standard is 0.053 ppm. It is shown here in terms of ppb for the purposes of clearer comparison to the one-hour standard level.

c. Final rule signed October 1, 2015, and effective December 28, 2015. The previous (2008) O_3 standards are not revoked and remain in effect for designated areas. Additionally, some areas may have certain continuing implementation obligations under the prior revoked one-hour (1979) and 8-hr (1997) O_3 standards.

d. The previous SO_2 standards (0.14 ppm 24-hr and 0.03 ppm annual) additionally remain in effect in certain areas: (1) any area for which it is not yet one year since the effective date of designation under the current (2010) standards, and (2) any area for which an implementation plan providing for attainment of the current (2010) standard has not been submitted and approved and which is designated nonattainment under the previous SO_2 standards or is not meeting the requirements of a State Implementation Plan (SIP) call under the previous SO_2 standards [40 CFR 50.4(3)]. A SIP call is an EPA action requiring a state to resubmit all or part of its SIP to demonstrate attainment of the required NAAQS.

Figure 19.3-5 Abilene, Texas Annual Average Precipitation

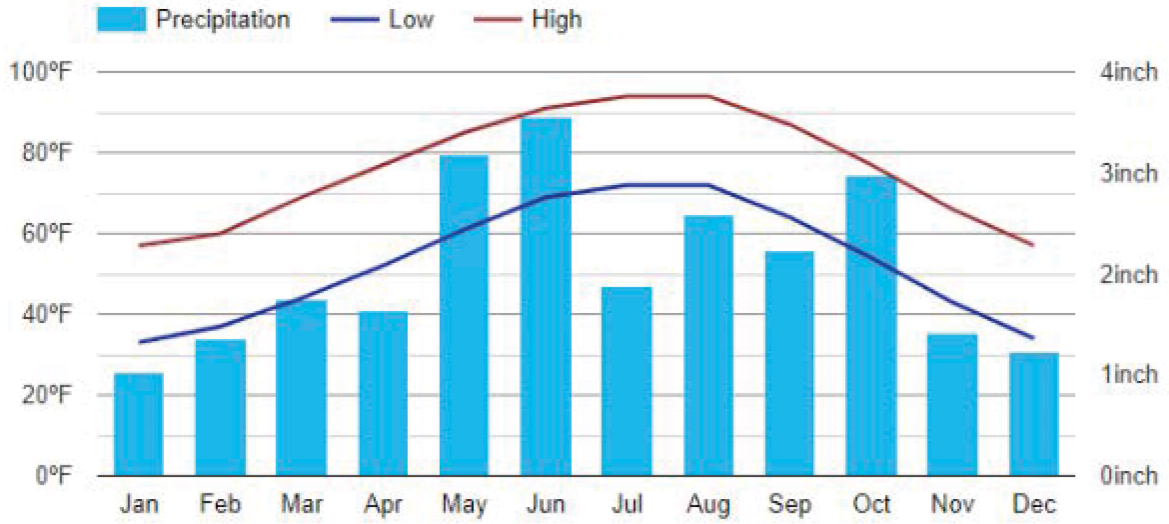
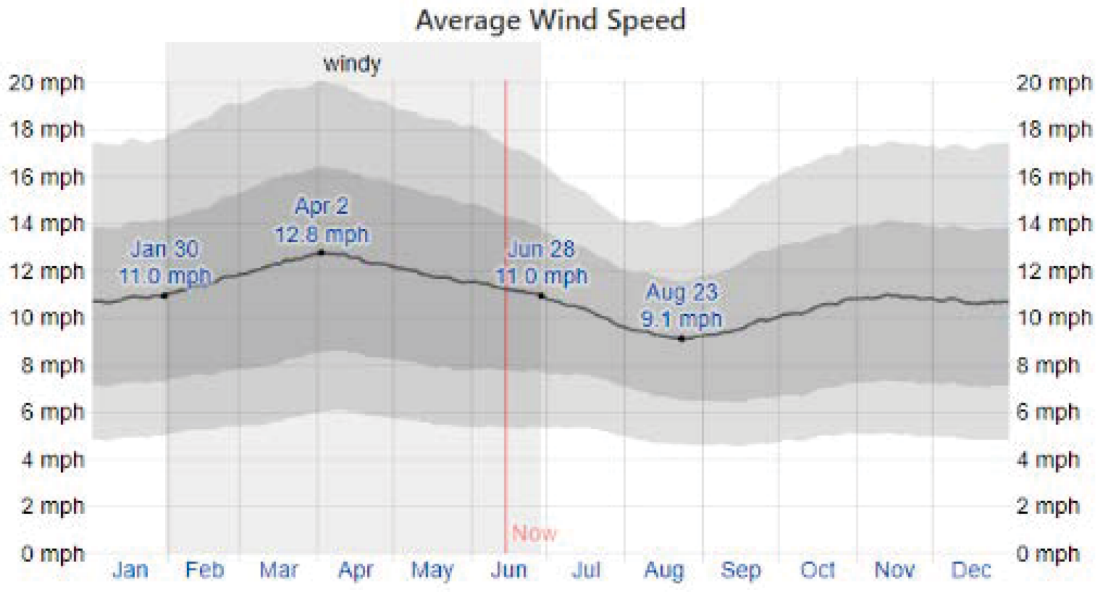
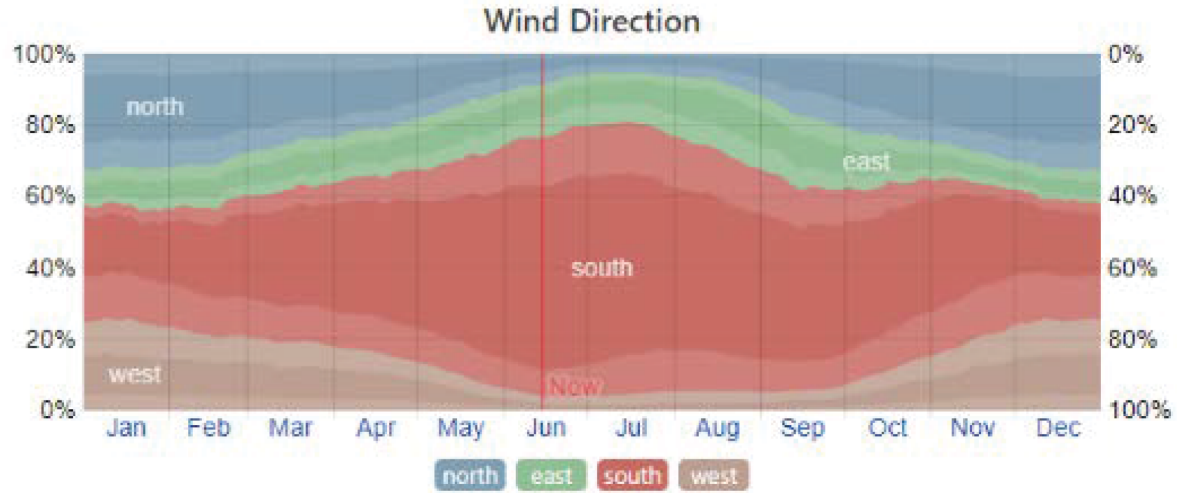


Figure 19.3-6 Abilene, Texas Average Wind Speeds



The average hourly wind speeds (dark gray line), with 25th to 75th and 10th to 90th percentile bands (Reference 19.3.3-1)

Figure 19.3-7 Abilene, Texas Wind Direction



The percentage of hours in which the mean wind direction is from each cardinal wind directions, excluding hours in which the mean wind speed is less than 1.0 mph (1.6 kph). The lightly tinted areas at the boundaries are the percentage of hours spent in the implied intermediate directions (northeast, southeast, southwest, and northwest) (Reference 19.3.3-1).

19.3.4 Geologic Environment

19.3.4.1 Site Geology

Tectonic activity has little or no effect on the proposed MSRR location. The site is located on the North American Plate and is more than 621 mi (1000 km) from any major plate boundary. A major earthquake on a major fault does not have any effect on Abilene or the surrounding region.

The current topography is flat but slopes gently toward the east. The eastern edge of the proposed site is roughly 3 ft (0.9 m) lower than the western edge, which creates a very slight slope over the course of the approximately 600 ft (183 m) plot of land.

The overall region of Abilene, Texas, has a download slope from the southern part with the lowest portion of the city being the area surrounding Lake Fort Phantom Hill and the Clear Fork of the Brazos River. Elevations across the entire site range from about 1729 ft (527 m) above mean sea level at the southeastern corner to 1735 ft (529 m) on the western edge. The composition of the ground onsite, based on the boring survey investigation conducted during the SERC design and construction, is primarily clay and sandy clay, which extends for roughly 2 ft (0.6 m) at all boring locations.

The nearest faults are more than 40 mi (64 km) away near the borders of Throckmorton and Shackleford counties. They are oriented diagonally from northwest to southeast. The nearest major fault is Meers Fault, located in Wichita Mountains Wildlife refuge in Oklahoma more than 175 mi (282 km) north-northwest of Abilene, and it dips to the southwest. It has a maximum estimated magnitude of 7.0 with a 1 percent probability of activity. From this distance, it is unlikely for Abilene to experience any effects.

19.3.4.2 Soils

Based on the boring survey conducted during the SERC design and construction, the top layers of soil at all boring locations are composed of brown, sandy clay extending downward 5 to 10 ft (1.5 to 3.0 m) with calcareous soils becoming more prevalent as depth increases. After about 10-12 ft (3.0-3.7 m), the material changes to red-brown weathered shale, which continues down for the remainder of the boring depth. Water was encountered in three of the seven bore holes drilled between 9 and 11 ft (2.7 and 3.4 m). This was determined to be seasonal perched groundwater trapped in the sandy clay soils above the weathered shales. Based on soil moisture content, the groundwater table is considered to exist below 60 ft (18.3 m) across the site.

19.3.4.3 Prime Farmland

The proposed facility is to be installed in the SERC on the ACU campus within the City of Abilene, an urban/suburban setting; therefore, no prime farmlands or farmland of statewide importance is present onsite.

19.3.4.4 Seismology

The site is located on the North American Plate. The nearest area with any historic seismological activity is more 62 mi (100 km) to the northwest near the town of Snyder, Texas. This area has had multiple earthquakes since recording started in 1973, with magnitudes ranging from 4.0 to 4.9, but an earthquake of this magnitude occurring in Snyder would not be felt nor cause any damage to buildings in Abilene.

The closest earthquake to Abilene occurred 42 years ago near the city of Jayton, Texas, approximately 46 mi (75 km) to the northwest. It registered a magnitude of 4.4 and occurred at a depth of nearly 6.2 mi (10 km).

The largest earthquake in Texas since 1990 was a 4.8 magnitude in Fashing, Texas, more than 250 mi (400 km) from Abilene. No damage was done to the city of Abilene.

In the same time period, a 4.4 magnitude earthquake occurred in the nearer city of Snyder, Texas. Again, no damage was reported in the city of Abilene.

Abilene has a low earthquake risk, with no quakes reported since 1931. The USGS database indicates there is a 0.17 percent likelihood of a primary earthquake within 31 mi (50 km) of Abilene, Texas, within the subsequent 50 years.

Using the earthquake return period, it is estimated that the peak ground acceleration for the proposed site is 0.0305 g. The calculated acceleration is consistent with the evaluates published in the national seismic maps.

19.3.4.5 Other Hazards

Located in west central Texas, the site is inland and not subject to threats from tsunamis.

The site is distant from active volcanism and not subject to threats from volcanic action.

19.3.5 Water Resources

19.3.5.1 Surface Water

No natural surface water features originate, end, or run through the proposed site, and there is no direct impact from construction on drainages. The closest bodies of water to the proposed site are two creeks running parallel to each other and are both approximately 1500 ft (457 m) from, and on either side of, the site (Figure 19.3-8). These two creeks, Cedar Creek (west of site) and Rainy Creek (east of site), both originate from Lake Kirby and Lytle Lake respectively, and flow from south to north into Lake Fort Phantom Hill north of ACU and outside Taylor County. Lake Fort Phantom Hill is about 5 mi (8.04 km) from ACU and the proposed site. No surface water is to be diverted, redirected, or dammed to support installation of the MSRR.

Figure 19.3-8 Abilene Area Surface Waters

The flow of Cedar Creek can be monitored using the nearest USGS gaging station, which is in the northeastern portion of the City of Abilene. The site number is 08083480, and the site name is Cedar Creek at IH 20, Abilene, Texas. The gaging station is a stream with a drainage area of 136 square miles (352 square kilometers) (Figure 19.3-9). The mean annual discharge at Cedar Creek is approximately 11.83 cubic feet per second (0.33 cubic meters per second), per data from 2002-2020. The annual peak streamflow for the years 2002-2020 can be found in Figure 19.3-10.

Figure 19.3-9 Cedar Creek USGS Gaging Station

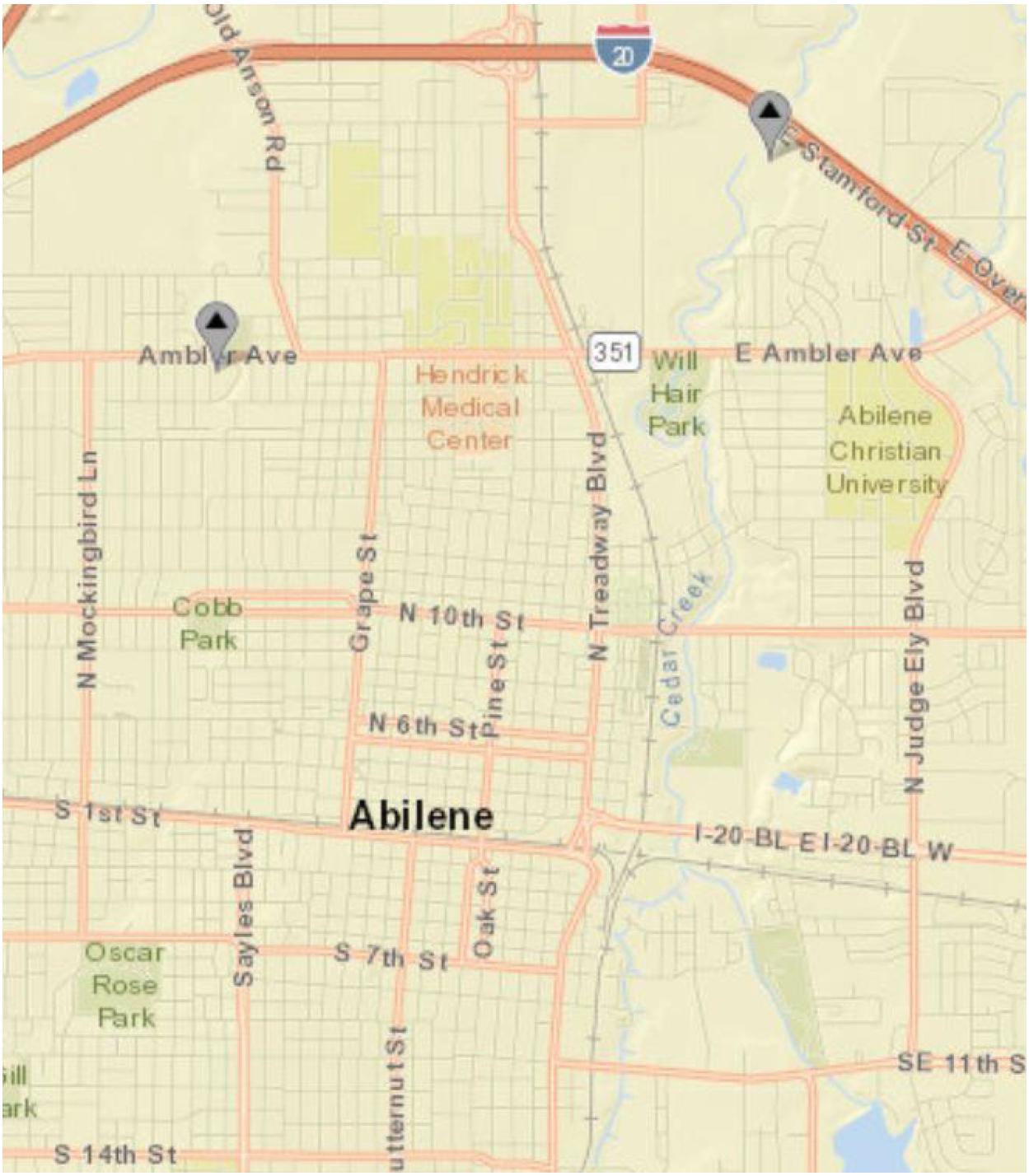
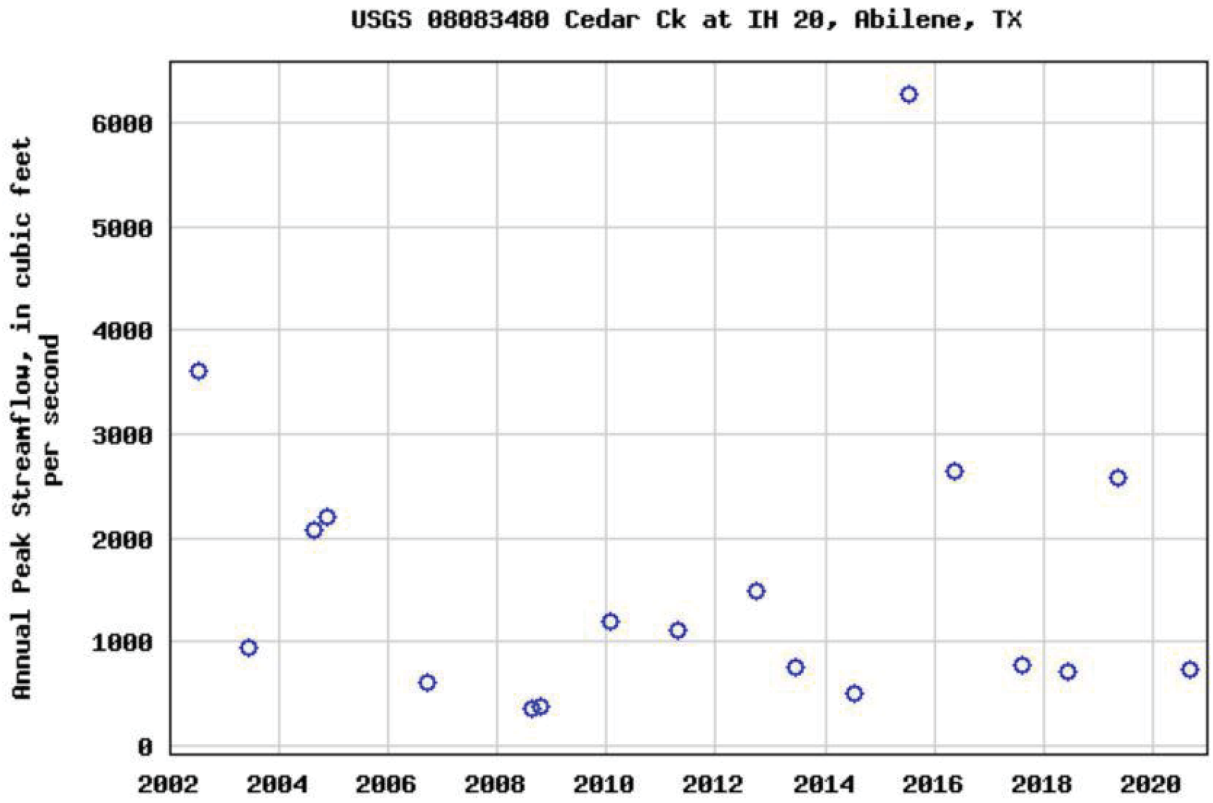


Figure 19.3-10 Annual Peak Streamflow



Peak flows tend to occur during late spring and early summer in Taylor County, particularly in May and July. There is little data to indicate a risk of flooding. The proposed site is in an area of minimal flood hazard, although there are multiple floodways northeast and directly west of the site (Figure 19.3-11). There is nothing to indicate the SERC site is prone to high water tables.

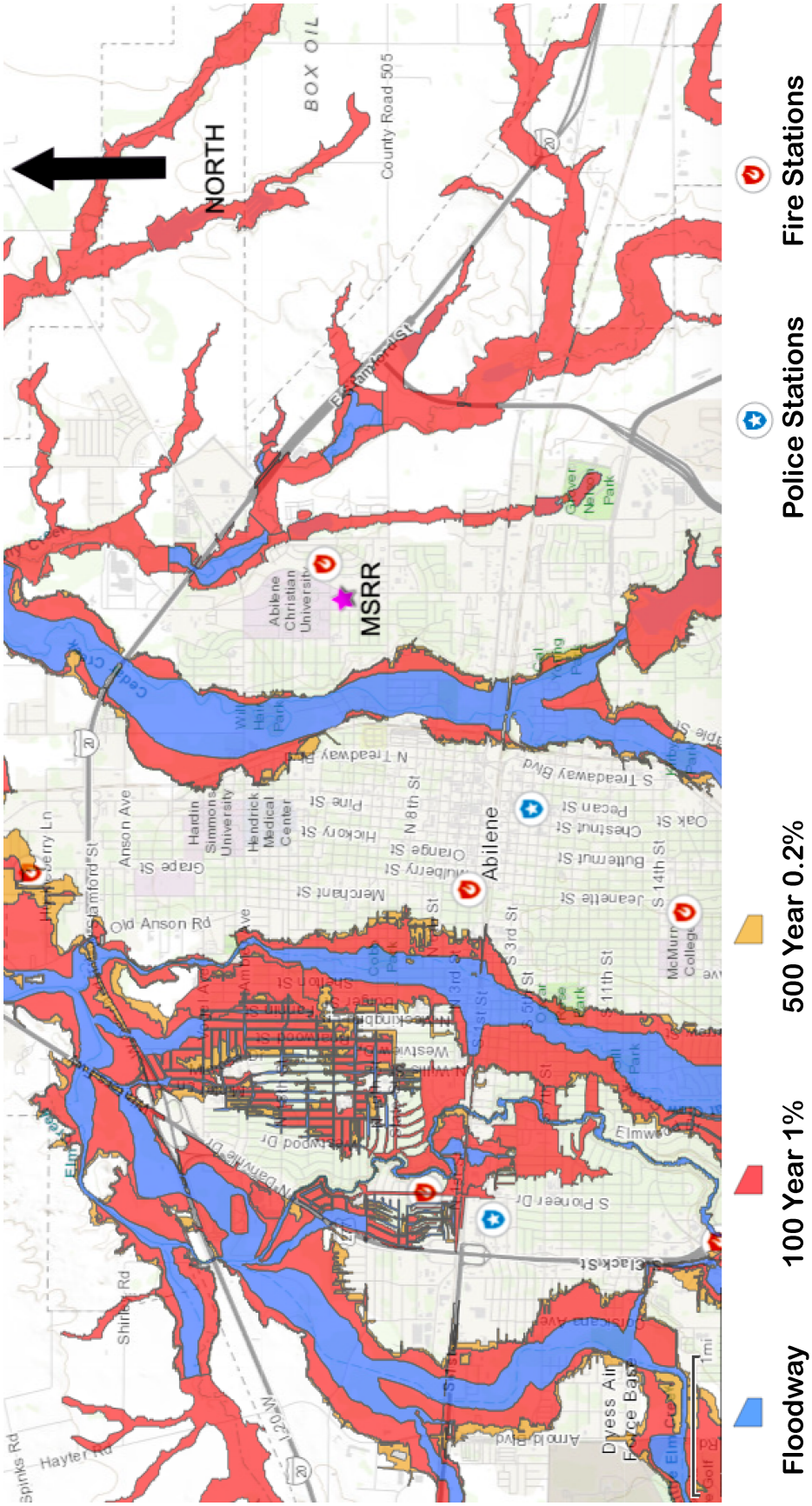


Figure 19.3-11 Potential Floodways

Kirby lake is located just east of highway 83-84 and south of Texas Highway Loop 322 approximately 6 miles from the SERC site. It is fed by Cedar Creek, covers 740 acre (299 ha), and has a maximum depth of 16 ft. Fort Phantom Hill is a reservoir between Farm Roads 600 and 2833. It covers a surface area of 4246 acres (1759 ha) and has a storage capacity of 74,310 acre-feet (92 million cubic meters). Fort Phantom Hill is owned by the City of Abilene and is used for recreational and municipal purposes.

The northern two-thirds of Taylor County is drained by the Brazos River drainage system. The southern third is drained by various tributaries of the Colorado River drainage system, most notably Jim Ned Creek.

Many small dams can be found in the area:

- Cameron Dam southwest of ACU
- Diamond Lake Dam southwest of ACU
- Elmer Griffith Lake Dam east of ACU
- Fairways Reservoir Development Number 2 Dam farther southwest of ACU
- JC Griffith Dam east of ACU (just past Highway 20)
- Lake Abilene Dam southwest of ACU
- Lake Bulger Dam south of Lake Abilene
- Lake Kirby Dam south of ACU
- Lytle Lake Dam south of ACU
- RJ Griffith Dam east of ACU
- Woodrow Griffith Lake Dam east of ACU.

19.3.5.2 Groundwater

There are three major aquifers that surround Abilene but are not near the proposed site. These include Seymour to the north, Edwards-Trinity Plateau to the southwest, and Trinity (outcrop) to the east. There are no minor aquifers in Taylor County.

Groundwater beneath Taylor and Jones Counties comes from the Seymour aquifer (primarily in Jones County with a slight extension into Taylor County) and the Edwards-Trinity aquifer system (only in Taylor County) (Reference 19.3.5-1). Neither of these systems are sole-source aquifers (Reference 19.3.5-2).

The Seymour aquifer was formed from erosion of the Seymour Formation during the Pleistocene age. The Seymour Formation consists of clay, silt, sand, and gravel, which coarsen downward to the primary water-producing zone. This sediment overlays poorly permeable bedrock of the Permian-age Clear Fork Group, which primarily consists of claystone (Reference 19.3.5-3). This aquifer is composed of many isolated patches of alluvial deposits, which vary in area, saturated thickness, well yields, and chemical quality. Taylor County contains 5 square miles (13 square kilometers) of outcrop from this aquifer and has zero cubic feet of average annual base flow. Jones County contains 326 square miles (844 square kilometers) of outcrop from this aquifer and has 3.3 cubic feet (0.09 cubic meters) of average annual baseflow (Reference 19.3.5-4).

More than 100 million gallons (379 million liters) of water per day are withdrawn from the Seymour Aquifer, mostly in Haskell and Knox Counties, which are just north of Jones County. The average recharge rate of this aquifer is two in. (5 cm) per year.

Both the Edwards-Trinity aquifer and the Trinity aquifer underlie part of Taylor County, Edwards-Trinity on the west side and Trinity on the east. The Edwards-Trinity aquifer is composed of limestone over sand and sandstone. In Taylor County, these rocks are from the Permian age.

At the proposed site, seven 2-in. (5-cm) borings were drilled from 5 ft to 40 ft (1.5 m to 12 m) below existing ground surface elevation. At Test Boring Nos. 1, 3, and 4, groundwater was encountered between 9 ft and 11 ft (2.7 m and 3.4 m), particularly within the alluvial soils and in contact with the shales. This was determined to be seasonal perched groundwater trapped in the sandy clay soils above the weathered shales. The groundwater table was estimated to exist at depths greater than 60 ft (18.3 m) below current grades. The water table likely fluctuates seasonally. These borings also found the soil beneath the proposed site is composed of very stiff-to-hard sandy and silty clays from the surface to 11-12 ft (3.4-3.7 m), below which were highly weathered-to-weathered shales to a depth of at least 40 ft (15 m). There were intermittent sandstone seams in some areas below a depth of 25 ft (7.6 m).

19.3.5.2.1 Groundwater Quality and Use

Almost all groundwater in Taylor County is considered fresh; most of the groundwater in Jones County is slightly saline with some being moderately saline, an indication of variation in chemical quality in the Seymour aquifer.

The part of the Seymour Aquifer in the northern part of Taylor County, which is the closest area to the proposed site, is considered slightly saline with a dissolved solids concentration of 1000-3000 milligrams per liter. The part of the Trinity Aquifer closest to the proposed site is considered fresh with a dissolved solids concentration of 0-1000 milligrams per liter. The part of the Edwards-Trinity Aquifer closest to the proposed site is also considered fresh.

The City of Abilene does not currently use any groundwater in its potable water supply. The city water supply primarily comes from Fort Phantom Hill Lake, Hubbard Creek Reservoir, and O.H. Ivie Reservoir. It is treated by the Northeast Treatment Plant, Grimes Treatment Plant, and Hargesheimer Water Treatment Plant.

High levels of nitrate are present in the groundwater in Taylor County, likely due to oxidation of soil organic nitrogen during initial cultivation followed by leaching of fertilizers on cultivated land. The Seymour Aquifer also contains excessive amounts of chloride and sulfate. The water in the Trinity Aquifer is generally fresh but very hard, and also contains high concentrations of chloride and sulfate.

19.3.5.3 Water Consumption

The only water consumption of note is water consumed through basic utilities such as sprinkler systems, toilets, sinks, and, potentially, chemical showers. Water in this region is plentiful relative to the present population. This facility has little to no noticeable impact on local water supplies.

City water is needed during decommissioning for purposes of controlling dust and possibly site-restoration purposes. With other considerations that will be stated, impacts on groundwater resources from the facility decommissioning are negligible. Contaminated materials or fuels are removed and treated. The water is sourced from Abilene reservoirs and city water facilities during decommissioning. Portable sanitary facilities most likely are used and removed for treatment as well.

19.3.5.4 References

19.3.5-1 Ryder, Paul D. Groundwater Atlas of the United States - Hydrologic Investigations Atlas 730-E. U.S. Geological Survey, Reston, VA, 1996. PDF ebook. https://pubs.usgs.gov/ha/730e/report.pdf

19.3.5-2 ArcGIS Web Application, Sole Source Aquifers, accessed June 16, 2022. https://epa.maps.arcgis.com/apps/webappviewer/index.html?id=9ebb047ba3ec41ada1877155fe31356b.

19.3.5-3 Clear Fork Group (TXPCF;0). Interactive Maps and Downloadable Data for Regional and Global Geology, Geochemistry, Geophysics, and Mineral Resources; Products of the USGS Mineral Resources Program, accessed June 16, 2022. https://mrdata.usgs.gov/geology/state/sgmc-unit.php?unit=TXPcf%3B0.

19.3.5-4 Texas Water Development Board, Texas Aquifers Study, December 31,2016 https://www.twdb.texas.gov/groundwater/docs/studies/TexasAquifersStudy_2016.pdf#page=143

19.3.6 Ecological Resources

19.3.6.1 Ecoregion

The proposed site is located within the rolling plains ecoregion of Texas which primarily consists of moraines or areas with accumulated soil and rocks formed during the Paleozoic and Mesozoic Eras. Approximately one-third of the rolling plains ecological region of North Central Texas is used for intensive agriculture with a variety of different crops. The rest of the region is dominated by cattle ranching. Natural vegetation has been converted to agricultural crops or pastures on about 90 percent of the area. There is a low density of small intermittent streams and few associated rivers, all with low volume of water flowing at low velocity. The region has seen significant change from human activity as livestock grazing practices that have introduced many invasive species such as perennial forbs, legumes, and woody species (Reference 19.3.6-1).

19.3.6.2 Site and Near-Site Areas

Before the recent SERC building was erected, the site consisted of roughly 15.17 acres (6.14 ha) of well-manicured, grassy fields used for ACU sports activities. Approximately 10-20 trees border the fields. Vegetation on this plot is highly limited as it is constantly being used for outdoor activities and is likely only common grass and weeds. Because of the urban location of the site, as well as the frequent use by local people, it is unlikely to support any permanent residence of animals such as prairie dogs, badgers, or squirrels. Immediately to the east of the site is a more densely populated area of trees (23) that covers about 0.49 acres (0.2 ha).

19.3.6.3 Habitats in the Vicinity of the Proposed Site

The immediate vicinity of the site is urban and does not allow natural development of the ecosystem. Approximately 0.5 mi (0.8 km) to the east, the location becomes much more open and allows for wildlife to live. This area consists of mostly grassy plains with patches of trees and many small ponds scattered throughout. To the west of the site is Cedar Creek, which is a low-volume, shallow river which runs through Abilene.

The rolling plains ecoregion is largely defined by gently rolling hills that are broken up by rivers and streams draining from west to east. Bottomlands frequently surround larger streams and rivers and contain American elm, button willow, pecan, and cottonwood trees. Some other dominant woody species are redberry juniper, yucca, mesquite, lotebush, hackberry, bumelia, prickly pear, skunk bush sumac, ephedra, plum, western soapberry, little leaf sumac, shin oak, tasajillo, agarito, catclaw acacia, lime prickly ash, and sand sage.

Watersheds of the Colorado, Brazos, Red, and Trinity Rivers bisect this region of North Central Texas. Riparian zones along these streams and their tributaries contain important wildlife habitat for the region and support good populations of

white-tailed deer and Rio Grande turkeys. Bobwhites, scaled quail, mourning doves, collared peccary, and a variety of songbirds, small mammals, waterfowl, shorebirds, reptiles, and amphibians are found in this region (Reference 19.3.6-2).

19.3.6.4 Protected Species and Habitats

Federally listed endangered species found in the rolling plains include (Reference 19.3.6-3):

- Gray wolf (canus lupus) - This species historically occurred throughout the high plains of Texas and is now thought to be extirpated.
- Black-footed ferret (mustela nigripes) - This species is thought to be extinct in the high plains of Texas; however, it did occur in the black-tailed prairie dog/ burrowing owl/prairie rattlesnake complex that exists today in prairie dog towns on shortgrass prairie sites.
- American peregrine falcon (Falco peregrinus anatum) - This is a neotropical (summer) migratory species infrequently observed gliding over prairies and croplands.

19.3.6.5 References

19.3.6-1 “Rolling Plains.” TPWD, Texas Parks and Wildlife, accessed June 16, 2022. https://tpwd.texas.gov/landwater/land/habitats/cross_timbers/ecoregions/rolling_plains.phtml.

19.3.6-2 “Rolling Plains.” TPWD, Texas Parks and Wildlife. Accessed June 16, 2022. https://tpwd.texas.gov/landwater/land/habitats/cross_timbers/ecoregions/rolling_plains.phtml.

19.3.6-3 “High Rolling Plains Wildlife Management Activities Appendix L to T.” Texas Parks and Wildlife Department, accessed June 16, 2022. https://tpwd.texas.gov/publications/pwdpubs/media/pwd_bk_w7000_793_high_rolling_plains_AppendixLtoT.pdf.

19.3.7 Historic and Cultural Resources

This section discusses the cultural background and the known historic and cultural resources at the proposed site in the city of Abilene, Texas, and in the surrounding area, including Taylor County and the State of Texas. This discussion is based on information from the Texas State Historical Association and the database of Texas historic sites. Although the MSRR will be installed in an existing building and no alteration of the character of historic properties will occur, the area of potential effects is defined as the site boundary identified in Figure 19.2-3.

19.3.7.1 Cultural Background

Human occupation in north and central Texas is generally characterized according to the following chronological sequence.

19.3.7.1.1 Paleoindian Period (13,400-9,800 B.P.)

The earliest peoples in Texas lived in small, mobile groups that hunted megafauna, such as mammoths and bison, as well as foraged for grasses and tubers. This period is characterized by Clovis technology (knapped tools used for basic cutting, woodworking, and digging) followed by Folsom technology (tools scraped on both sides; unnotched, fluted, lanceolate points). Other projectile points have been found as well, including the unnotched Barber and Golondrina points, as well as the stemmed Wilson point.

19.3.7.1.2 Archaic Period (9,800-1,250 B.P.)

During the Archaic Period, smaller dart points used with atlatls became more common. Food patterns were diversified with a variety of game and plants found in the diet, many of which were cooked in earth ovens or on large rock mounds. Roots and tubers were especially common in the diet. There was also more variety in stone tools, with manos and metates, gouges, drills, and burins found at various sites. Points in the Archaic period include Agostura, Hoxie, Uvalde, Bandy, Andice-Bell-Calf Creek, Taylor, Nolan, Bulverde, Ensor, and Fairland. Cooking became more refined in the late Archaic period with rock ovens and hearths becoming more common. Small cemeteries containing individuals killed during conflict have been found as well.

19.3.7.1.3 Late Prehistoric Period (1,250-375 B.P., ca. A.D. 750-1600)

The introduction of the bow and arrow came about during the late Prehistoric Period, with arrow points such as the Scallorn, Granbury, and Edwards points found. The diet became more dependent on freshwater mussels. Drills, awls, and perforators were also found during this period. At the end of the Late Prehistoric period, plain bone-tempered pottery came into use, and tools became even more complex, including choppers, hammer stones, spatulas, and gravers. Bones of animals were used for tools and for ornaments such as beads.

19.3.7.1.4 Historic Period (375 B.P.-present, ca. A.D. 1600 to present)

Texas was explored by both Spanish and French expeditions in the 1500s and 1600s. The first Spanish mission in Texas was established in 1690, and Spanish colonization continued until Mexico won its independence in 1821. Anglo-American settlers moved in and established farms until eventually Texas won its independence from Mexico, then became a state. Farming continues to be important in Texas to this day, although the oil and gas, entertainment, business, and tech industries have grown as well.

19.3.7.2 Historic and Archeological Resources

Taylor County, Texas, has 54 individual places and five districts on the National Register of Historic Places. All are buildings, most built in the early to mid-1900s with a few built in the late 1800s. Most are designated as historic places for design or architectural reasons, but a few are indicated as historic for significant persons or historical events as well.

There are no historic properties within the boundary of the site. The closest historical place, approximately a quarter mile (0.4 km) from the proposed site, is the ACU Administration Building (Hardin Administration Building). It was built as part of the university's expansion after World War II in the classical revival style. It is considered historically important because of historic events and its design and architecture.

The closest archaeological resource is Fort Phantom, approximately 13 mi (20.92 km) north of the site. The fort was built in the 1850s and is not currently being surveyed or dug. This project is unlikely to affect the fort.

19.3.7.3 References

19.3.7-1 Texas Historical Commission. (2020). Texas Historic Sites Atlas [Map]. Texas.gov. https://atlas.thc.texas.gov/Map

19.3.7-2 Timothy K. Perttula, Thomas R. Hester, Stephen L. Black, Carolyn E. Boyd, Michael B. Collins, Myles R. Miller, J. Michael Quigg, Wilson W. Crook III, Bryon Schroeder, Ellen Sue Turner, Drew Sitters, Nancy Velchoff, Richard A. Weinstein, and Thomas J. Williams, "Prehistory," Handbook of Texas Online, https://www.tshaonline.org/handbook/entries/prehistory. Published by the Texas State Historical Association.

19.3.8 Socioeconomics

This section describes socioeconomic factors that have the potential to be directly or indirectly affected from construction, operations, and decommissioning of the proposed MSRR facility. The MSRR facility and the communities that support it can be described as a dynamic socioeconomic system. The communities provide the people, goods, and services required to construct, install and operate the proposed MSRR facility. The MSRR facility operations, in turn, provide wages and benefits for people and dollar expenditures for goods and services. The measure of a community's ability to support the construction, operations, and decommissioning of the proposed MSRR facility depends on its ability to respond to changing environmental, social, economic, and demographic conditions.

The socioeconomic region of influence is defined by the area in which MSRR operations employees and their families likely reside, spend their income, and use their benefits-all of which affect the economic condition of the region. For the purposes of analysis and because of the small size of the MSRR operations workforce (10 to 20 people), this area includes all of Taylor County and the City of Abilene.

19.3.8.1 Population Growth Rates and Projections

Taylor County has four cities, four towns, one census-designated place, and four unincorporated communities. On July 1, 2019, the estimated total population was 138,034 (Reference 19.3.8-1). The two most populated cities in the region are Abilene and Merkel, with a population of 123,420 people (Reference 19.3.8-2) and 2614 people (Reference 19.3.8-3), respectively. Merkel is directly west of Abilene off Interstate-20. As of 2019, the population of the rest of the county was approximately 12,000 people. There was a 5 percent change in population between April 1, 2010, to July 1, 2019, and there has been an average increase of approximately 0.45 percent per year over the past 20 years. The largest change in population was a 13.4 percent increase between 1970 and 1980 (Reference 19.3.8-4).

19.3.8.2 Race and Ethnicity

Table 19.3-2 presents the demographic profiles for the City of Abilene and Taylor County. The total minority population in Abilene is 53,688, which is 43.5 percent of the total population. The largest minority groups in the City of Abilene are Hispanics or Latinos (of any race) at 26.6 percent, followed by Black or African American at 10.6 percent. The total minority population in Taylor County is 55,076 which is 39.9 percent of the total population. The largest minority groups are Hispanic and Latinos at 25 percent and Black or African American at 8.4 percent.

Table 19.3-2 Demographic Profiles

	Taylor County	City of Abilene
Total Population	138,034	123,420
Race (percent of total population, Not Hispanic or Latino)		
White alone	85.2	72.2
Black or African American Alone	8.4	10.6
American Indian and Alaska Native alone	1	0.9
Asian alone	2.3	2.3
Native Hawaiian and Other Pacific Islander alone	0.2	0
Two or More Races	3	3.1
Ethnicity		
Hispanic or Latino	25	26.6
White alone, not Hispanic or Latino	62.7	58.3
Minority Population (including Hispanic or Latino ethnicity)		
Total minority population	55,076	53,688
Percent minority	39.9	43.5

19.3.8.3 Transient Population

Colleges and recreational opportunities attract daily and seasonal visitors who create a demand for temporary housing and services. In 2021, approximately 8791 students attended colleges and universities within 20 mi. (32 km) of the SERC facility (Reference 19.3.8-5). According to the 2000 census, there were 204 seasonal housing units in Taylor County.

Migrant farm workers are individuals whose employment requires travel to harvest agricultural crops. These workers may or may not have a permanent residence. Some migrant workers follow the harvesting of crops, particularly fruit, throughout rural areas of the country. Others may be permanent residents living near Abilene and Merkel and traveling from farm-to-farm harvesting crops.

Migrant workers may be members of minority or low-income populations. Because migrant workers travel and can spend a significant amount of time in an area without being actual residents, they may be unavailable for counting by census takers. If uncounted, these minority and low-income workers are “underrepresented” in the decennial census population counts.

In the 2017 Census of Agriculture, farm operators were asked whether they hired migrant workers - defined as a farm worker whose employment required travel - to do work that prevented the migrant workers from returning to their permanent place of residence the same day. The census is conducted every five years and results in a comprehensive compilation of agricultural production data for every county in the nation.

Information about migrant and temporary farm labor (working less than 150 days) was collected in the 2017 Census of Agriculture. According to that census, approximately 352 farm workers were hired to work for less than 150 days on 182 farms in Taylor County, Texas. Two farms in Taylor County reported hiring migrant workers in the 2017 census (Reference 19.3.8-6).

19.3.8.4 Labor Force, Employment, and Unemployment

This section depicts labor force, employment, and unemployment data for the city of Abilene, where 57.2 percent of the population is age 16 or older and can work. As of March 2021, the civilian labor force was approximately 79,000 people with an unemployment rate of 5.8 percent. The average over the previous six months (from October 2020 to March 2021) was 78,687 with an unemployment rate of 5.43 percent. Between 2020 and 2021, unemployment increased in the City of Abilene and in Taylor County. One of the reasons that were attributed to this was the COVID-19 pandemic, which led to the closure of many workplaces in the country. A Bureau of Labor Statistics report of employment by industry shows Dyess Air Force base is the largest employer. Outside of that, many people are employed in health, education, and manufacturing.

19.3.8.5 Income and Wages

Table 19.3-3 compares the median family and per capita income figures for the City of Abilene, Taylor County, and the State of Texas (Reference 19.3.8-7). According to the U.S. Census Bureau 2015-2019, the City of Abilene had a median family income and per capita income lower than both Taylor County and the State of Texas. Overall, both Taylor County and the City of Abilene had lower median family and per capita income than that of the State of Texas.

Table 19.3-3 Median Family Income and Per Capita Income (2015-2019)

Median Family Income	**2015-2019**
City of Abilene	$50,659
Taylor County	$53,143
State of Texas	$61,874
Per Capita Income	**2015-2019**
City of Abilene	$24,529
Taylor County	$27,162
State of Texas	$31,277

19.3.8.6 Poverty Rates

Table 19.3-4 compares the percentage of individuals living below the federal poverty threshold in the City of Abilene, Taylor County, and the State of Texas. Table 19.3-5 shows the poverty levels determined by the U.S. Census Bureau in 2019 (Reference 19.3.8-8).

Table 19.3-4 People Living Below U.S. Census Poverty Thresholds

All People	2015-2019
City of Abilene	16.5%
Taylor County	14%
State of Texas	13.6%

Table 19.3-5 2019 U.S. Federal Poverty Thresholds for Different Family Sizes

Size of family unit	Related children under 18 years								
	None	One	Two	Three	Four	Five	Six	Seven	Eight or more
One person (unrelated individual):									
Under age 65	13,465								
Aged 65 and older	12,413								
Two people:									
Householder under age 65	17,331	17,839							
Householder aged 65 and older	15,644	17,771							
Three people	20,244	20,832	20,852						
Four people	26,695	27,131	26,246	26,338					
Five people	32,193	32,661	31,661	30,887	30,414				
Six people	37,027	37,174	36,408	35,674	34,582	33,935			
Seven people	42,605	42,871	41,954	41,314	40,124	38,734	37,210		
Eight people	47,650	48,071	47,205	46,447	45,371	44,006	42,585	42,224	
Nine people or more	57,319	57,597	56,831	56,188	55,132	53,679	52,366	52,040	50,035

Source: U.S. Census Bureau.

19.3.8.7 Housing

Table 19.3-6 shows the amount of housing units and the vacancy rates for Taylor County (Reference 19.3.8-9).

Table 19.3-6 Taylor County Total Housing Units and Vacancy Rates

Housing Units	Amount
Total Number of Housing Units	57,952
Vacant Housing Units	7,650
Vacancy Rates	**Percentage**
Homeowner Vacancy Rates	1.9%
Rental Vacancy Rates	6.4%

19.3.8.8 Local Employers

Table 19.3-7 ranks the 10 largest employers in the City of Abilene (Reference 19.3.8-10).

Table 19.3-7 Ten Largest Employers in the City of Abilene

Company	Product/Service	Employees
Dyess AFB	Air Force Base	8400
Hendrick Health System	Hospital	3200
Abilene Christian Univ.	Private university	1900
State supported living center	Mental health facility	1225
TX Dept of Criminal Justice	Prisons	1190
Blue Cross Blue Shield	Call center	1090
Abilene Regional Medical Center	Hospital	830
AbiMar Foods	Food manufacturing	680
First Financial Bank	Banking and financial	540
Rentech Boiler Systems	Metal fabrication	400

19.3.8.9 Taxes

Counties, municipalities, and boards of education may impose sales taxes in addition to the state sales tax. Local and state entities in the region of influence impose sales and property taxes. These include the school district, the City of Abilene, Taylor County, and the State of Texas. Tax rates can vary by jurisdiction. The retail sales tax in the City of Abilene is 8.25 percent. There is no personal income tax in the State of Texas. The sales tax in the State of Texas is 6.25 percent, but certain jurisdictions can raise it to a maximum of 8.25 percent (Reference 19.3.8-11). The combined corporate tax rate in the State of Texas is 21 percent. Table 19.3-8 compares the school district tax levies in the region of influence as of 2019 (Reference 19.3.8-12).

Table 19.3-8 School Districts in the Region of Influence and Their Tax Levies

School District	Tax Levies per $100 valuation
Abilene ISD	$1.3214
Jim Ned ISD	$1.1195
Merkel ISD	$1.1855
Trent ISD	$1.28835
Wylie ISD	$1.197

19.3.8.10 Education

In the 2018-2019 school year, the Abilene Independent School District (ISD) had approximately 16,500 students in pre-kindergarten (K) through grade 12 (Reference 19.3.8-13). The Abilene ISD has 17 pre-K to grade 5 elementary schools, six grade 6 to 9 middle schools, and six grade 9 to 12 high schools (two traditional, one STEM academy, one medical magnet program, and one alternative school) (Reference 19.3.8-14). The ATEMS High School, Woodson Center for Excellence, Craig Middle School, Abilene Christian School, Thomas Elementary School, Taylor Elementary, and Holland Medical School are all within 2.5 mi (4 km) of the site.

The second largest school district in Taylor County is Wylie ISD, which is also in the City of Abilene. Wylie ISD has approximately 4773 students (Reference 19.3.8-15) who attend seven different schools. There are four pre-K to grade 5 elementary schools, two grade 6 to 9 middle schools, and one grade 9 to 12 high schools (Reference 19.3.8-16).

Taylor County has nine public school districts serving K through grade 12. The total enrollment for all Taylor County school districts is approximately 26,335. These school districts, with their current total enrollment data, include:

- Abilene ISD (16,500)
- Blackwell CISD (163) (Reference 19.3.8-17)
- Clyde CISD (1,453) (Reference 19.3.8-18)
- Eula ISD (426) (Reference 19.3.8-19)
- Jim Ned CISD (1,362) (Reference 19.3.8-20)
- Merkel ISD (1,106) (Reference 19.3.8-21)
- Trent ISD (157) (Reference 19.3.8-22)
- Winters ISD (568) (Reference 19.3.8-23)
- Wylie ISD (4,773)

Five private schools are in Taylor County, all of which are in the City of Abilene for K through grade 12 (Reference 19.3.8-24), seven post-secondary schools are located in Taylor County (Reference 19.3.8-25). Texas State Technical College and Cisco College are two-year technical and community colleges located in the City of Abilene. ACU, Hardin-Simmons University, and McMurry University all offer four-year degrees and are located in the City of Abilene.

19.3.8.11 Tourism, Activity Centers, and Recreation for the City of Abilene and Taylor County

19.3.8.11.1 Tourism and Activity Centers (Shopping, Business, Agricultural, and Sporting Events)

The City of Abilene has several tourist attractions and activity centers, including Abilene Zoo, West Texas Fair and Rodeo, National Center for Children's Illustrated Literature, Paramount Theatre, and Frontier Texas. Also, there are several museums and educational centers, including The Center for Contemporary Arts, Dyess Memorial Museum, and Taylor County History Center (Reference 19.3.8-26).

19.3.8.11.2 Public Recreational Facilities

The City of Abilene has approximately 730 acres of parks (Reference 19.3.8-27). Most of the outdoor activities are in Abilene State Park [529.8 acres (214 ha)] (Reference 19.3.8-28), including hiking, biking, swimming, fishing, and camping. Other parks in Taylor County have walking trails.

19.3.8.12 Public Services

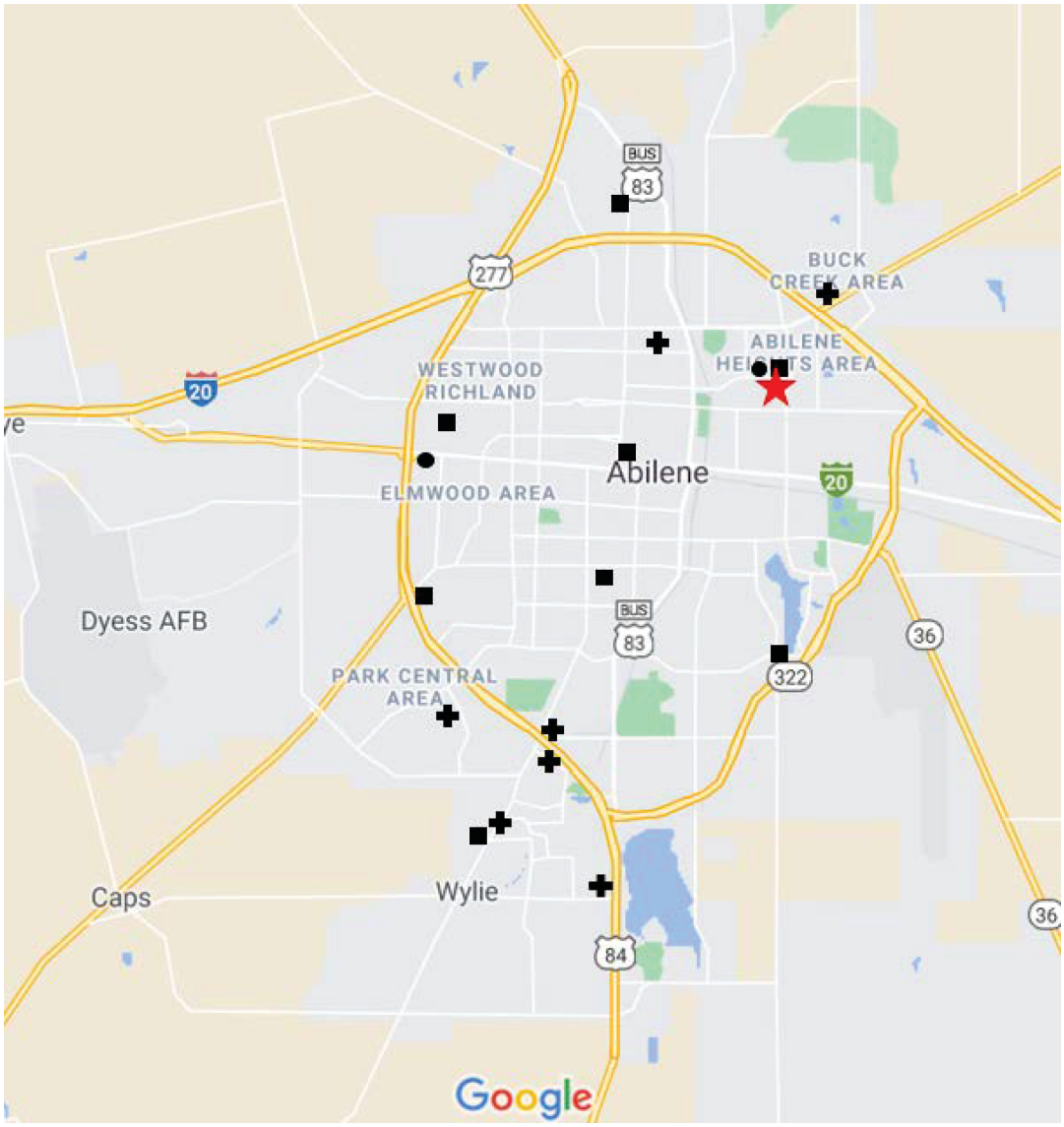
19.3.8.12.1 Medical

The City of Abilene has several medical care facilities including two hospitals. Hendrick Medical Center South has 231 beds (Reference 19.3.8-29), and Hendrick Medical Center has 350 beds (Reference 19.3.8-30). There are four homeless shelters and approximately 12 assisted living care facilities and nursing homes in Abilene (Reference 19.3.8-31).

19.3.8.12.2 Emergency Services

Fire/rescue and emergency medical services are located throughout Taylor County, primarily in the two largest cities, Abilene and Merkel. The location of emergency services in Abilene are shown in Figure 19.3-12.

Figure 19.3-12 Abilene Area Emergency Services



Squares are fire stations, circles are police departments (Abilene PD to the west and ACU PD near the star), crosses are emergency rooms, and the star is the proposed MSRR site.

Water Treatment

Taylor County supplies water to community residents by various water systems and well types - municipal, other than municipal, transient noncommunity, nontransient noncommunity, and private sources of supply water. Water is brought from Fort Phantom Hill Reservoir, Hubbard Creek Reservoir, and O. H. Ivie Reservoir, then treated and purified in Abilene (Reference 19.3.8-32). Some of the city water supply also comes from reclaimed wastewater (Reference 19.3.8-33).

There are 13 water utilities in Taylor County. The cities of Abilene, Buffalo Gap, Lawn, Merkel, Trent, and Tye provide municipal services. Other areas of Taylor County are mostly served by water supply corporations; however, there is one Water Control and Improvement District run jointly by Tuscola and Taylor County (Reference 19.3.8-34). Because the City of Abilene is the only water utility supplier in Abilene, the water utility system likely serves almost all if not all the population of Abilene, which is approximately 124,000. The City of Abilene does not use groundwater sources for its municipal water supply. The wastewater in Taylor County is reclaimed and treated in the areas with access to municipal services (Reference 19.3.8-35).

19.3.8.13 References

19.3.8-1 Bureau, US Census. “U.S. Census Bureau Quick Facts: Taylor County, Texas.” United States Census Bureau, accessed June 16, 2022. https://www.census.gov/quickfacts/taylorcountytexas.

19.3.8-2 Bureau, US Census. “U.S. Census Bureau Quick Facts: Abilene City, Texas.” United States Census Bureau, accessed June 16, 2022. https://www.census.gov/quickfacts/abilenecitytexas.

19.3.8-3 Data USA Merkel, TX Census Place. “Merkel, TX Census Place,” Data USA, Accessed June 16, 2022. https://datausa.io/profile/geo/merkel-tx.

19.3.8-4 “Texas Almanac: City Population History from 1850-2000.” Texas Historical Association, accessed June 16, 2022. https://texasalmanac.com/sites/default/files/images/CityPopHist%20web.pdf

19.3.8-5 “Hardin-Simmons University - Profile, Rankings and Data | US News Best ...” Hardin-Simmons University, U.S. News and World Report, 2022. Accessed June 16, 2022. https://www.usnews.com/best-colleges/hardin-simmons-university-3571.

19.3.8-6 “Table 7. Hired Farm Labor - Workers and Payroll: 2017.” 2017 Census of Agriculture Full Report Volume 1, Chapter 2, United States Department of Agriculture, accessed June 16, 2022.

19.3.8-7 Bureau, US Census. “Quick Facts: Texas.” U.S. Census Bureau, United States Census Bureau, 2021, accessed June 16, 2022. https://www.census.gov/quickfacts/fact/table/TX/PST045219.

19.3.8-8 Bureau, US Census. “Poverty Thresholds.” Census.gov, United States Census Bureau, May 9, 2022. https://www.census.gov/data/tables/time-series/demo/income-poverty/historical-poverty-thresholds.html.

19.3.8-9 Bureau, US Census. “American Community Survey (ACS).” Census.gov, United States Census Bureau, April 18, 2022. https://www.census.gov/programs-surveys/acs.

19.3.8-10 “Why Abilene.” Why Abilene, Development Corporation of Abilene, January 30, 2022. https://developabilene.com/community-profile#cp-workforce.

19.3.8-11 Hegar, Glenn. “Sales and Use Tax.” Sales and Use Tax, Texas Comptroller of Public Accounts, 2022, accessed June 16, 2022. https://comptroller.texas.gov/taxes/sales/#:~:text=Texas%20imposes%20a%206.25%20percent,as%20well%20as%20taxable%20services.

19.3.8-12 “School District Property Tax Rates in Taylor County.” School District Property Tax Rates in Texas by County, Texas Association of Counties, 2022, accessed June 16, 2022. https://txcip.org/tac/census/schoolhist.php?FIPS=48441.

19.3.8-13 “Abilene ISD at a Glance.” District Home Abilene Independent School District, 2022, accessed June 16, 2022. https://www.abileneisd.org/our-district/district-information/#:~:text=Approximately%2016%2C500%20students%20are%20enrolled%20in%20Abilene%20ISD.

19.3.8-14 “Explore Abilene Independent School District.” Abilene Independent School District, Niche, March 15, 2021. https://www.niche.com/k12/d/abilene-independent-school-district-tx/.

19.3.8-15 Murphy, Ryan, et al. “Wylie ISD.” Texas Public Schools Wylie ISD, The Texas Tribune, June 10, 2022. https://schools.texastribune.org/districts/wylie-isd-taylor/.

19.3.8-16 Explore Wylie Independent School District (Abilene).” Niche, May 3, 2021. https://www.niche.com/k12/d/wylie-independent-school-district-abilene-tx/.

19.3.8-17 Murphy, Ryan, et al. “Blackwell CISD.” Texas Public Schools, The Texas Tribune, June 10, 2022. https://schools.texastribune.org/districts/blackwell-cisd/.

19.3.8-18 Murphy, Ryan, et al. “Clyde CISD.” Texas Public Schools, The Texas Tribune, June 10, 2022. https://schools.texastribune.org/districts/clyde-cisd/.

19.3.8-19 Murphy, Ryan, et al. “Eula ISD.” Texas Public Schools, The Texas Tribune, June 10, 2022. https://schools.texastribune.org/districts/eula-isd/.

19.3.8-20 Murphy, Ryan, et al. “Jim Ned CISD.” Texas Public Schools, The Texas Tribune, June 10, 2022. https://schools.texastribune.org/districts/jim-ned-cisd/.

19.3.8-21 Murphy, Ryan, et al. “Merkel ISD.” Texas Public Schools, The Texas Tribune, June 10, 2022. https://schools.texastribune.org/districts/merkel-isd/.

19.3.8-22 Murphy, Ryan, et al. “Trent ISD.” Texas Public Schools, The Texas Tribune, June 10, 2022. https://schools.texastribune.org/districts/trent-isd/.

19.3.8-23 Murphy, Ryan, et al. “Winters ISD.” Texas Public Schools, The Texas Tribune, June 10, 2022. https://schools.texastribune.org/districts/winters-isd/.

19.3.8-24 “Best Taylor County Private Schools.” Private School Review, Private School Review, 2022, accessed June 16, 2022.

19.3.8-25 “Colleges near Taylor County, Texas.” Univstats, JNIVSTATS, 2022, accessed June 16, 2022. https://www.univstats.com/colleges/texas/taylor-county/.

19.3.8-26 “Experience Culture, Cuisine, & Cowboys.” Official Tourism Website of Abilene, Texas, Abiline Convention and Visitors Bureau, 2022, accessed June 16, 2022. https://www.abilenevisitors.com/things-to-do/.

19.3.8-27 “My Abilene Parks & Recreation.” About the Parks Department Abilene, TX, City Of Abilene, Texas, 2022, accessed June 16, 2022. https://abilenetx.gov/347/About-the-Parks-Department.

19.3.8-28 “Abilene State Park.” Abilene State Park History - Texas Parks & Wildlife Department, Texas Parks and Wildlife, August 4, 2016. https://tpwd.texas.gov/state-parks/abilene/park_history#:~:text=The%20park%2C%20with%20529.4%20acres,Guide%20to%20Abilene%20State%20Park.

19.3.8-29 “Hendrick Medical Center South.” Hendrick Health, Hendrick Health, 2022, accessed June 16, 2022. https://www.hendrickhealth.org/locations/hendrick-medical-center-south/.

19.3.8-30 “Hendrick Medical Center.” PracticeLink, Pittsburgh Critical Care Associates, Inc., 2022, accessed June 16, 2022. https://www.practicelink.com/facility/Hendrick-Medical-Center/Intensivist-Jobs/. “Hendrick Medical Center.” Hendrick Medical Center (450229) - Free Profile, American Hospital Directory, 2022, accessed June 16, 2022/

19.3.8-31 “Assisted Living Abilene Texas.” Google Maps, Google, 2022, accessed June 16, 2022. https://www.google.com/maps/search/assisted+living+abilene+tx/@32.4281257,-99.7616945,12z.

19.3.8-32 “Our Lakes.” Our Lakes | Abilene, TX, City of Abilene Texas, 2022, accessed June 16, 2022. https://abilenetx.gov/605/Our-Lakes.

19.3.8-33 “Wastewater Treatment.” City of Abilene Texas, City of Abilene Texas, 2022, accessed June 16, 2022. https://abilenetx.gov/455/Wastewater-Treatment.

19.3.8-34 “Find as Water Utility.” Public Utility Commission of Texas Water Utility Information, Public Utility Commission of Texas, 2022, accessed June 16, 2022. https://www.puc.texas.gov/WaterSearch/Search/Find?UtilityName=&RepPartyName=&CCNRegnum=&UtilityTypeId=W&OwnershipTypeId=&CountyId=221&ActivityStatusId=.

19.3.8-35 “Wastewater Treatment.” Wastewater Treatment, City of Abilene Texas, 2022, accessed June 16, 2022. https://abilenetx.gov/455/Wastewater-Treatment.

19.3.9 Human Health

The AEA gives the NRC authority to issue and enforce standards that provide an adequate level of protection for public health and safety and that protect the environment. The NRC staff evaluates the latest radiation protection recommendations from national and international scientific bodies as a basis for its radiation protection standards. The facilities that the NRC licenses to possess radioactive material must adhere to these radiation protection standards to protect workers and the public against potential health risks from exposure to radioactive material used, generated, and released from the licensed facility. The NRC staff periodically inspects a licensed facility to ensure the facility operates within the NRC requirements.

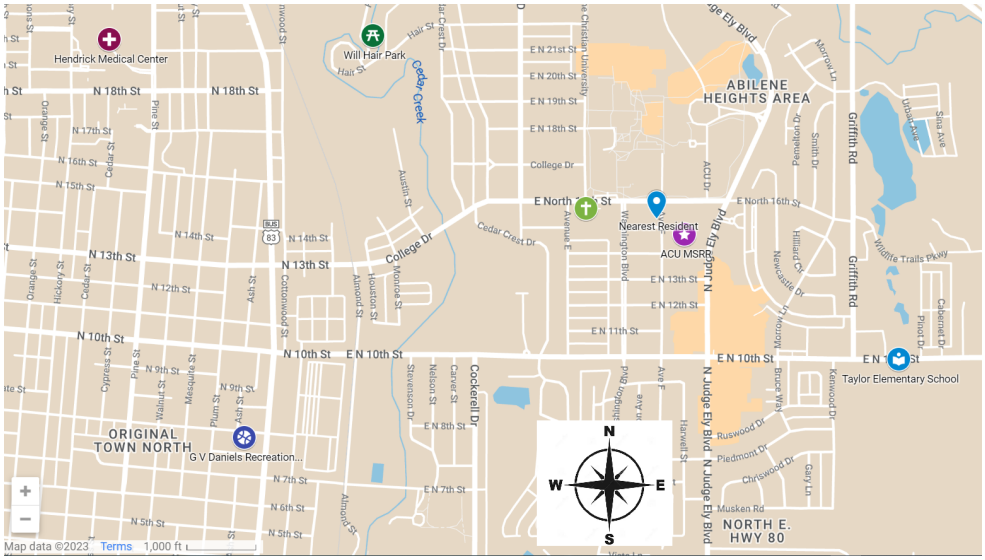
19.3.9.1 Maps of Potentially Sensitive Surrounding Facilities

Nearest sensitive receptors (shown on Figure 19.3-13) include:

- Residents
- Parks
- Educational facilities
- Medical facilities
- Religious facilities
- Community centers

The site boundary distances are illustrated in Figure 19.4-1. The nearest full-time resident is approximately 410 ft (125 m) from the MSRR. The nearest school is Taylor Elementary School 0.67 mi (1 km) away and the nearest hospital is Hendrick Medical Center about 1.6 miles (2.5 km) from the facility. Figure 2.4-3 shows the nearest drinking water intake, which is on Lake Fort Phantom, 15 km NNE of the facility.

Figure 19.3-13 Nearest Sensitive Receptors to the MSRR



19.3.9.2 Background Radiation Exposure

There is no background radiation from natural or man-made sources in the vicinity of the site that results in abnormal radiation hazards to the public. Average background radiation to the public is 620 mrem/yr [6.2 millisieverts (mSv)] in the United States, with approximately half from medical procedures and man-made sources and half from natural sources (Reference 19.3.9-1).

19.3.9.3 Natural Sources

Natural sources of radiation can be divided into three categories: cosmic, internal, and terrestrial. Cosmic radiation comes from extremely energetic particles from the sun and stars that enter Earth's atmosphere. Internal contamination occurs when people swallow or inhale radioactive materials, or when radioactive materials enter the body through an open wound or are absorbed through the skin. Terrestrial radiation is due to the decay of radioactive materials in the earth. This source represents the majority of the background radiation for an average member of the public (Reference 19.3.9-2).

19.3.9.4 Man-Made Sources

A portion of background radiation comes from human activities. Man-made sources of radiation to the public include medical sources, consumer products, and nuclear reactor facilities as discussed below. Trace amounts of radioactive elements have dispersed in the environment from nuclear weapons tests and accidents like the one at the Chernobyl nuclear power plant in Ukraine. Normally operating nuclear reactors emit small amounts of radioactive elements (Reference 19.3.9-3).

An accident at a nuclear reactor could result in offsite radiological exposure. The risk of this happening at nuclear power plants is very small because of the diverse and redundant barriers and safety systems in place at nuclear power plants, the training and skills of the reactor operators, testing and maintenance activities, and regulatory requirements and oversight. Even in the unlikely event of such an accident, impacts are local and do not contribute significantly to background radiation. The nuclear fuel cycle also creates various waste products; these waste products are also subject to strict regulation such that the contribution to background radiation is very minor. (Reference 19.3.9-2)

19.3.9.5 Radioactive and Nonradioactive Hazardous Material Stored Onsite

No radioactive materials or hazardous materials are currently stored on the site. As such, there are no liquid, gaseous, or solid waste management systems of effluent control systems at the facility. The MSRR is installed in the SERC, a 28,000 square foot (2323 square meter), multilevel structure containing spaces designed to support research in chemistry, physics, and a variety of engineering disciplines. It is expected that eventually there will be other radiation sources in the SERC in addition to the MSRR.

ACU possesses radiation sources in the form of X-ray machines, which are not associated with the MSRR. These sources are licensed by the Texas Department of State Health Services (Reference 19.3.9-4).

19.3.9.6 Public Radiation Exposure

The proposed MSRR, if approved by the NRC, would be licensed to possess, use, generate, and release radioactive effluents under controlled conditions into the environment during normal operation. The MSRR uses radioactive waste management systems to remove radioactivity to maintain the doses to workers and members of the public within the NRC dose limits and to be as low as is reasonably achievable.

The radioactive materials potentially released under controlled conditions include fission gases, such as Xe-138, Xe-135, Xe-135m, Kr-87, and Kr-88, and activated air products, such as Ar-41, H-3, and C-14. N-16 is the most common potential effluent but decays before reaching public populations.

Gaseous radioactive effluents are monitored to ensure compliance with the NRC requirements. The SERC does not plan to release any liquid radioactive effluents.

The radiation dose to the maximally exposed individual is discussed in Section 19.4.9.2.

19.3.9.7 Occupational Radiation Exposure

The SERC workers who conduct activities involving radioactively contaminated systems or who work in radiation areas can be exposed to radiation. ACU radiation protection staff monitors its radiation workers in accordance with NRC requirements. Most of the occupational radiation dose results from external

exposure rather than from internal exposure resulting from inhaled or ingested radioactive materials. Facility workers are exposed to radiation throughout the labs because of exposures from

- radiation emitted from radionuclides in the salts.
- activated components, potentially with deposited fission products, during reactor maintenance and coupon monitoring programs.
- activated air from reactor spaces.

19.3.9.8 Chemical Hazards

Chemicals can enter the body through the skin, by inhalation, or by ingestion. Chemical exposure produces different effects on the body, depending on the chemical and the amount of exposure. Chemicals can cause cancer, affect reproductive capability, disrupt the endocrine system, and have other health effects. Acute effects from chemical exposure occur immediately. Chronic or delayed effects result in symptoms, such as skin rashes, headaches, breathing difficulties, and nausea.

At the SERC facility, chemical effects can result from the routine use of chemicals and hazardous materials during research processes. A list of chemicals used at the MSRR is provided in Table 19.4-1.

All chemical waste is bottled and disposed using a chemical waste disposal contractor per ACU's Chemical Hygiene Plan.

19.3.9.9 Other Hazards

The SERC facility is a research facility with many of the typical occupational hazards found at other laboratory research facilities. Workplace hazards can be grouped into physical hazards (e.g., hazards from slips, trips, and falls from a height, and those from transportation, temperature, humidity, and electricity); physical agents (e.g., noise, vibration, and ionizing radiation); chemicals; and psychosocial issues (e.g., work-related stress).

19.3.9.10 References

19.3.9-1 U.S. Environmental Protection Agency, Radiation Sources and Doses, Website: https://www.epa.gov/radiation/radiation-sources-and-doses.

19.3.9-2 U.S. Nuclear Regulatory Commission, Sources of Radiation, Website: https://www.nrc.gov/about-nrc/radiation/around-us/sources.html,

19.3.9-3 U.S. Nuclear Regulatory Commission, Sources of Radiation, Website: https://www.nrc.gov/about-nrc/radiation/around-us/sources.html.

19.3.9-4 Texas Department of State Health Services, License Numbers R00161 and R00161-00, Website: https://www.dshs.texas.gov/.

19.4 Impacts of Proposed Construction, Operations, and Decommissioning

This chapter provides an analysis of the impacts of construction, operation, and decommissioning of the facility. Overall impact rankings are given to each environmental resource evaluated. Unless otherwise defined, criteria followed the guidance given in NRC Impact Rankings in 10 CFR Part 51, Subpart A, Appendix B, Table B-1, Footnote 3, as follows:

- SMALL (S) - Environmental effects are not detectable or are so minor that they neither destabilize nor noticeably alter any important aspect of the resource.
- MODERATE (M) - Environmental effects are sufficient to alter noticeability, but not to destabilize, important aspects of the resource.
- LARGE (L) - Environmental effects are clearly noticeable and are sufficient to destabilize important aspects of the resource.

19.4.1 Land Use and Visual Resources

19.4.1.1 Land Use - Construction

Installation of the MSRR does not require construction of new structures. Components of the MSRR will be constructed off-site in existing facilities that perform fabrication work of systems similar to the MSRR. The MSRR is installed in the existing SERC. As the SERC is already constructed and not the subject of this environmental review, the following summary description of the SERC is for information.

The SERC site currently includes approximately 15.17 acres (6.14 ha) of developed land. Construction of the SERC facility converted developed area into an industrial area that includes the MSRR building, an employee parking lot, and facility access roads. The portion of the site that remains undisturbed either is covered by the former building of Taylor Elementary or by cool-season grasses.

Construction of the SERC facility did not convert any land with soils that may qualify as prime farmland or farmland of statewide significance to industrial use, cool-season grasses, or native prairie. Furthermore, the site is currently zoned for commercial use (Reference 19.4.2-1). Because the SERC site has been committed to urban development and zoned for commercial use, the proposed site does not have farmland soils subject to the Farmland Protection Policy Act (7 CFR 658.2).

Land-use impacts are confined to the proposed 15.17-acre (6.14-ha) site; therefore, areas with a special land use or mineral resources are not affected by construction and installation of the MSRR within the SERC facility.

The Coastal Zone Management Act of 1972, as amended (16 U.S.C. 1451 et seq.), was enacted to address the increasing pressures of overdevelopment upon the nation’s coastal resources. Given that no federally designated coastal zone

areas are within 5 mi (8 km) of the SERC site, and that the proposed action does not affect any land or water use or natural resource within the coastal zone, the applicant's certification and the state's concurrence are not required.

Based on the proposed project not converting permanently any farmland to other land uses, the lack of qualifying important farmland soils within affected areas, the location of the proposed facility within an area zoned for commercial use, and the lack of special land use or mineral resources on site, land use impacts from construction are SMALL.

19.4.1.2 Land Use - Operations

Operations of the MSRR do not require any new land or require land use changes beyond that required for construction (installation in the SERC); therefore, land use impacts during operations are SMALL.

19.4.1.3 Land Use - Decommissioning

Decommissioning activities are similar to construction and installation activities because they involve equipment to dismantle and remove the MSRR from within the SERC. Land requirements to perform these activities are the same or less than those required during construction and installation. After decommissioning activities are complete, the SERC continues to be used for research activities. Given that land requirements are similar to those described during construction and that, after decommissioning is complete, land is either industrial or open space, land use impacts during decommissioning are SMALL.

19.4.2 Visual Resources

19.4.2.1 Visual Resources - Construction

The visual setting of the SERC is described in Section 19.3.2. The site currently has the former building for Taylor Elementary School on the southern portion and the SERC building northern portion. The SERC facility building is approximately 50 ft high, 173 ft long, and 150 ft wide (15 x 53 x 46 m).

The activities associated with the construction and installation of the MSRR require equipment, alter onsite conditions, and partially obstruct views of the existing landscape. The proposed site has low scenic quality because of a lack of notable features, uniform landform, low vegetation diversity, an absence of water, muted colors, and a commonality within the physiographic province. The proposed site has a low sensitivity rating because it is in an area with low scenic values resulting from a low amount of use by viewers, low public interest in changes to the visual quality of the proposed site, a low sensitivity to changes in visual quality by the type of users in the area, and a lack of special natural and wilderness areas. Further, when construction and installation activities are complete, ACU may vegetate open areas with cool-season grasses or native prairie grasses. Vegetation can partially mitigate impacts to visual resources. Based on the low scenic quality and the views within the vicinity, construction-related aesthetic impacts are SMALL.

19.4.2.2 Visual Resources - Operation

The appearance of the SERC facility does not change during operation of the MSRR; therefore, visual impacts during operations are SMALL.

19.4.2.3 Visual Resources - Decommissioning

Decommissioning activities are similar to construction and installation activities because they involve heavy equipment to dismantle the MSRR and to remove equipment. After MSRR decommissioning activities, the SERC building remains in use; therefore, visual impacts during decommissioning are SMALL.

19.4.2.4 References

19.4.2-1 “Property Search 56979 ABILENE CHRISTIAN UNIV for Year 2022.” Taylor CAD - Property Details, Taylor CAD, 2022, accessed June 16, 2022. https://propaccess.taylor-cad.org/clientdb/Property.aspx?cid=1&PROP_ID=56979

19.4.3 Air Quality and Noise

19.4.3.1 Air Quality Impacts

Modest air and noise emissions occur during construction, operations, and decommissioning of the MSRR. There is interior “remodeling” to accommodate the MSRR within the SERC, that are primarily, if not entirely, indoor activities. Air and noise emissions are limited to delivery trucks.

During construction and installation, both air quality and noise levels can be affected near the SERC facility. Air pollutants include fugitive dust from vehicles and exhaust gases from worker vehicles as they commute to and from the SERC facility. Noise is emitted from increased traffic as workers commute to and from the SERC facility. As the number of additional delivery trucks to ACU associated with installation and operation of the MSRR is a small fraction of the number of routine, daily delivery vehicles, air quality impacts are SMALL.

19.4.3.2 Noise Impacts

Noise emissions during operation occur because of increased traffic volumes from additional workers associated with the MSRR. The number of MSRR workers is small compared the number of workers and associated traffic from faculty and staff whose offices are located in the SERC. Noise from operating equipment is contained inside the building and inaudible outside of the SERC facility.

Given that noise emissions from operating equipment are not expected to be audible beyond the building, and that additional noise emissions caused by worker vehicles are minor, it can be concluded that offsite noise impacts during construction and operation are SMALL.

19.4.4 Geologic Environment

This section addresses the direct and indirect effects of the construction, operations, and decommissioning of the proposed MSRR on the geologic environment when added to the aggregate effects of other past, present, and reasonably foreseeable future actions. The cumulative impacts on the geologic environment primarily relate to land disturbance, the potential for soil erosion and loss, and the projected consumption of geologic resources. The description of the affected geologic environment is provided in Section 19.3.4. The geographic area of analysis for evaluation of cumulative impacts on soil resources includes the 5-mi (8 km) vicinity surrounding the proposed site. For geologic resources, the extent of the geologic area of analysis has been expanded to all of Taylor County to encompass potential commercial sources of rock and mineral resources to support construction activities at the proposed site and vicinity. Because the aspects of land disturbance and conversion are addressed in Section 19.4.1, the cumulative impact analysis focused on soil loss, including the loss of prime farmland soils and other important farmland soils, and the consumption of geologic resources. These loses are minimal because the MSRR is installed in the existing SERC.

The incremental impacts from construction, operations, and decommissioning of the proposed MSRR on the geologic environment, including geologic and soil resources, are SMALL.

19.4.5 Water Resources

19.4.5.1 Surface Water

As described in Section 19.3.5, the closest bodies of water to the proposed site are two creeks that run parallel to each other and are both approximately 1500 ft (457 m) from, and on either side of, the SERC site. There is no direct impact to these water bodies, however, as all construction and operational activities are contained within the SERC. No surface water is diverted, redirected, or dammed to support installation of the MSRR. No intake or discharge facilities are constructed or renovated as part of this project.

The SERC facility uses municipal water from the City of Abilene. There are water lines, sanitary water lines, and water fire lines connecting to the SERC; therefore, both direct and indirect impacts to surface water during construction and operation are SMALL.

19.4.5.2 Groundwater

Groundwater in the area is described in Section 19.3.5.2. The MSRR facility uses municipal water from the City of Abilene and sanitary wastes are discharged to the sanitary sewer system. Groundwater withdrawal or returns are not required during construction, operation, or decommissioning of the facility. Consequently, direct and indirect impacts to groundwater during construction, operations, and decommissioning are SMALL.

19.4.5.3 Water Use

The only water consumption of note is water consumed through basic utilities such as sprinkler systems, toilets, sinks, cleaning and potentially chemical showers. Water in this region is plentiful relative to the present population. This facility has little to no noticeable impact on local water supplies. Because there is excess capacity within the City of Abilene public works, potential indirect effects of the demand from the facility are SMALL.

19.4.5.4 Monitoring

Because of the absence of direct impacts to surface waters and to groundwater, the low potential for indirect impacts, and the use of management measures and controls to prevent releases to groundwater, no non-radiological groundwater monitoring activities are planned for the facility.

19.4.6 Ecological Resources

This section addresses the impacts of construction, operation, and decommissioning of the facility on the ecological resources within the vicinity of the site. The impacts discussed below are based on the characterization and description of terrestrial and aquatic ecosystems from Section 19.3.6.

19.4.6.1 Places and Entities of Special Interest

As described in Section 19.3.6, there are no places or entities of special interest on the site. The site is part of the ACU campus. Impacts from construction, operations, and decommissioning on places and entities of special interest are SMALL because such resources are not present on the site or near the location of the proposed facility where construction, operations or decommissioning occurs. Specific mitigation measures and management controls are not needed.

19.4.6.2 Aquatic Communities and Wetlands

As described in Section 19.3.5, the closest bodies of water to the proposed site are two creeks that run parallel to each other and are approximately 1500 ft (457 m) from, and on either side of, the SERC site. There is no direct impact to these water bodies, however, as all construction and operational activities are contained within the SERC. The facility does not withdraw or discharge water for cooling or other processes from surface water bodies or groundwater. Process water and potable water for the facility are obtained from the City of Abilene. Construction and decommissioning does not affect waterbodies near the site. Thus, direct and indirect impacts to aquatic communities and wetlands from construction and operations are SMALL.

19.4.6.3 Terrestrial Communities

The proposed MSRR is part of the SERC on the ACU campus in Abilene, Texas (see Section 19.2.1). The setting is urban/suburban. The terrestrial communities on the site and adjacent areas are described in Section 19.3.7.2. As construction

involves installation of the MSRR in the existing SERC, construction, operation and decommissioning activities have little to no effect on mobile wildlife species or birds. Therefore, impacts to wildlife and terrestrial communities from construction and operations are SMALL.

19.4.6.4 Protected Species

Federally Listed Endangered Species found in the rolling plains area are delineated in Section 19.3.6.4. No designated critical habitat is identified onsite or in the vicinity of the site. No Federal or state-listed threatened, endangered or special concern plant species have been observed on or in the immediate vicinity of the site. Potential impacts from construction, operations, and decommissioning on protected species potentially occurring on the site or in near offsite areas are SMALL. Specific mitigation measures and management controls are not needed.

19.4.7 Historic and Cultural Resources

As is described in Section 19.3.7.2, no onsite historic properties are associated with the site. The nearest listed National Register of Historic Places property, approximately a quarter mile (0.4 km) from the proposed site, is the ACU Administration Building (Hardin Administration Building). No direct impacts are expected to occur to the Hardin Administration Building in association with either construction, operational, or decommissioning activities of the site. As construction involves installation of the MSRR within an existing structure, no visual or other indirect impacts occur. Therefore, potential impacts to historic and archaeological resources are SMALL.

19.4.8 Socioeconomics

This section describes potential impacts to the socioeconomic environment, including transportation system impacts associated with the construction and operation of the facility. The assessment of potential socioeconomics impacts addresses potential changes in the regional population, economy, housing availability, and public services. The evaluation of transportation system impacts addresses routes and expected modes involved with transporting materials, workers, and equipment to the site.

19.4.8.1 Socioeconomics Impact

This section evaluates impacts to the population, housing, public services (e.g., water supply), public education, and tax-revenues in the region of influence (ROI) that result from constructing, operating, and decommissioning the facility. The ROI is identified as Taylor County, Texas. Potential impacts of constructing the facility are attributable to the size of the construction workforce, the expenditures needed to support the construction program, and any tax payments made to political jurisdictions. Because direct impacts are those that occur onsite, the only direct impacts are associated with the presence of the workforce at the site. All other socioeconomic impacts are considered to be indirect, as they occur offsite.

Construction activities associated with the MSRR are essential installation of reactor and support equipment within the existing SERC systems pit. Typical heavy construction activities associated with structural building are not expected. Consequently, the construction/installation labor force is much smaller than typical: dozens, up to 100, workers. During operation, NEXT Lab anticipates only 10 to 20 additional employees associated with operation of the MSRR. The analysis presented in this subsection is based on these parameters for the projected workforces for construction, operation, and decommissioning.

19.4.8.1.1 Population Impacts

On July 1, 2019, the estimated total population of Taylor County was 138,034 and the population of Abilene was 123,420 people (Section 19.3.8.1). Abilene Christian University has 1900 employees and a transient undergraduate student enrollment of approximately 3500. The number of workers anticipated for construction and operations is very small compared to the ROI population. Thus, the estimated ROI labor force in the construction trades is demonstrated to be abundant relative to construction workforce requirements, which greatly reduces the potential for large numbers of trade workers to relocate in the ROI. It is possible that some workforce commutes or temporarily relocates to the site from non-ROI counties, but these numbers are not significant or do not cause a perceptible increase in the ROI population. Therefore, the impact of construction, operation, and decommissioning of the facility on population are SMALL.

19.4.8.1.2 Housing Impacts

Section 19.3.8.7 delineates the amount of housing units and vacancy rates for Taylor County, which had 57,952 total housing units with 7650 units vacant. The rental vacancy rate is higher than the homeowner vacancy rate in Taylor County at 6.4 percent. The impact of construction, operation, and decommissioning workforces on housing are SMALL.

19.4.8.1.3 Public Services Impact

During construction, operation, and decommissioning the City of Abilene public works supplies water to the site, including potable water uses, fire protection uses, and nominal construction uses. As the city water public works serves almost all of the population of City of Abilene, which is approximately 124,000 (Section 19.3.8.7), the impact of the relatively small number of construction and operations workers on public water supply is SMALL. Similarly, impacts to wastewater treatment facilities are small.

19.4.8.1.4 Public Education Impacts

Schools and student populations are discussed in Section 19.3.8.10. Population increase due to construction workforce, operational workforce, and decommissioning workforce requirements is not expected. The estimated ROI labor force in the construction trades is demonstrated to be abundant relative to construction and operations workforce requirements, which greatly reduces

the potential for large numbers of trade workers to relocate in the ROI. It is possible that some workforce may commute or temporarily relocate to the site from non-ROI counties, but these numbers are not significant or cause a perceptible increase in the area's population or result in any perceptible change in school enrollment. Therefore, the level of impact to the local public education system is SMALL.

19.4.8.1.5 Tax Impacts

Tax revenues associated with the construction, operation, and decommissioning of the facility include payroll taxes on wages and salaries of the construction and operations work forces, and sales and use taxes on purchases made by ACU and the construction and operations workforces. Increased employment and the associated tax collections are a benefit to the state, county, and municipal-level jurisdictions as well as school districts. The overall tax revenues, however, are relatively small in comparison to the established tax base of Taylor County. Therefore, total tax revenues from the MSRR project result in SMALL positive impacts at the community level.

19.4.8.2 Transportation Impacts

The area around the site is served by a transportation network of federal and state highways; one commercial passenger airport (Abilene Regional Airport); and one air force base. Goods and services to support the facility reach the site using existing roadways. The operation, and decommissioning of the facility does not alter existing transportation routes for conveying materials or personnel to the site. Therefore, the impacts to transportation routes are SMALL.

The construction and operation of the facility does not alter existing traffic patterns to and from the site because of the limited increase in number of workers; therefore, the impacts to traffic patterns are SMALL.

19.4.8.3 Public Recreational Facilities

Traffic, visual impairments, and other negative impacts to surrounding recreational facilities are minimal because the MSRR is located within the SERC structure, which mitigates sounds and limits construction, operation, and decommissioning impacts from these recreational facilities locations. During operation, noise from the facility is limited by walls and other physical barriers; therefore, operational impacts due to noise are SMALL.

19.4.9 Human Health

The following sections discuss the potential non-radiological and radiological health impacts to the public and to occupational workers from construction, operation, and decommission of the facility. Federal regulations for generating, managing, handling, storing, treating, protecting, and disposing of wastes during construction, operation, and decommissioning are issued and overseen by the NRC and the EPA. These regulations include compliance with provisions of the Clean Air Act, the Clean Water Act, the AEA, and the Resource Conservation and Recovery Act.

19.4.9.1 Non-radiological Impacts

Non-radiological hazards pertaining to the facility include hazards from emissions, discharges, and waste from normal facility processes and from accidental releases. Nonradioactive wastes from construction, operation, and decommissioning activities of the facility include solid waste and liquid waste, and air emissions. Wastes, discharges, and emissions are managed in accordance with applicable Federal, state, and local laws and regulations.

During construction, nonradioactive chemicals, which include fuels, oils, solvents, and other materials necessary for MSRR installation, are expected to be present onsite. During operation, non-radioactive chemicals associated with the MSRR are stored onsite in small quantities and include lubricating oil for rotating equipment and cleaning materials and consumables used for cleaning and maintenance. The estimated inventory of other major chemicals used during operations of the MSRR is provided in Table 19.4-1. Other NEXT Lab and ACU research with various chemicals occurs that is related to the reactor but is not included in this table. During decommissioning, nonradioactive chemicals expected on site is similar to the construction materials.

19.4.9.1.1 Nonradioactive Liquid Wastes

The primary source of liquid wastes are sanitary wastes discharged to the City of Abilene wastewater collection system that conveys to the Abilene Hamby Water Reclamation Facility. Insignificant volumes of non-radioactive liquid chemical wastes are generated during construction, operation, or decommissioning. Some lab-scale chemical use occurs outside the research bay area.

There is waste associated with purification of reactor coolant salts. This purification results in an aqueous solution of sodium hydroxide, universal indicator, and sodium fluoride resulting from the neutralization of hydrofluoric acid. Additionally, there is liquid waste from laboratory chemical analyses. Chemical waste is bottled and disposed using a chemical waste disposal contractor per the ACU Chemical Hygiene Plan.

All industrial or wastewater generated in the radiologically controlled area is either evaporated or solidified and disposed of in accordance with the SERC waste management plan and applicable laws and regulations. Therefore, the direct and indirect impacts from liquid effluents during construction, operation, and decommissioning are SMALL.

19.4.9.1.2 Non-radioactive Gaseous Waste

The facility generates gaseous effluents resulting from process operations and the ventilation of operating areas. Cooling air exhausted from the coolant loop radiator is vented through the side of the research bay. Effluent from the reactor off-gas system is mixed with room air and ventilated to the roof. This

effluent is monitored prior to release. Fume hoods, instrument exhaust, and beryllium handling lab exhaust are ventilated to the roof via high-efficiency particulate air and carbon filtration.

Non-radioactive gaseous waste produced as a result of MSRR operation is similar to that produced in other science buildings on the ACU campus and does not require a separate TCEQ permit or registration.

As the estimated offsite emissions are insignificant and meet regulatory requirements from all phases of the project, direct and indirect human health impacts beyond the site boundary during construction, operations, and decommissioning are SMALL.

19.4.9.1.3 Non-radioactive Solid Waste

The following is a representative list of nonradioactive solid wastes that are generated by the project during construction, operation, and decommissioning:

- Construction and demolition debris
- Spent personal protective equipment
- Maintenance items such as heating, ventilation, and air conditioning filters, pipes, electronics, etc.
- Batteries (alkaline, lithium)
- Expired light bulbs and fixtures
- Office supplies
- Expired ink cartridges
- Cleaning supplies
- Empty plastic containers
- Food wastes

Solid waste management and control measures for the facility include waste reduction, recycling, and waste minimization practices. The impact of solid waste is SMALL.

19.4.9.1.4 Physical Occupational Hazards

Physical occupational hazards exist during all phases of the project, particularly during the construction and decommissioning phases. Because occupational hazards occur onsite and during construction, operation, and decommissioning of the facility, they are considered direct impacts. No indirect impacts (offsite) are identified. Table 19.4-2 lists the general types of occupational physical hazards (physical, electrical, and chemical) that can be present at the facility during the phases of the project.

The following hazardous chemicals are expected to be stored at the facility:

- Lithium fluoride
- Sodium fluoride
- Potassium fluoride
- Beryllium fluoride
- Hydrogen fluoride
- Hydrogen gas
- Uranium tetrafluoride (or similar as reactor fuel)
- Sodium hydroxide
- Beryllium
- Nitric acid

No hazardous chemicals in quantities above the threshold quantities of 29 CFR 1910.119 Appendix A are expected to be stored at the facility.

Occupational physical hazards are reduced or eliminated through implementation of safety practices, training, and physical control measures. Operations adheres to the regulations and standards established by the Occupational Safety and Health Administration regulations; therefore, the impacts from occupational hazards are SMALL.

19.4.9.2 Radiological Impacts

This section describes the public and the occupational health impacts from radioactive materials at the facility. During the construction phase, radioactive material is on site for construction-related activities such as radiography. These radioactive materials are present as sealed sources covered by contractor radioactive materials licenses. The impacts from the use of these radioactive materials on both occupational health and public health are SMALL when the devices containing the radioactive materials are operated according to standard operating procedures and their respective license conditions. The radiological impacts addressed in the following subsections result from reactor-related source during the operation and decommission phases of the facility.

19.4.9.2.1 Layout and Location of Radioactive Material

Figure 19.4-1 and Figure 19.4-2 illustrate the physical layout of the site, indicating site features, the site boundary, structures, and designated areas. Radioactive materials are within the research bay and laboratory areas with the high-radiation materials limited to the systems pit and storage pit within the research bay. Table 19.4-3 provides an overview of the significant radioactive materials present within the MSRR facility. Access to the research bay is strictly controlled, and personnel entering these buildings are participants in the occupational dose monitoring program. In addition, a variety of relatively

small sources are present consequent to the reactor's experimental and educational purposes. These include calibration sources, the off-gas system, material coupons, and salt chemistry experiments

19.4.9.2.2 Gaseous Sources of Radiation

19.4.9.2.2.1 Sources

No airborne sources of radioactivity will be produced during construction. Airborne radioactive sources during operation within the MSRR consist of radioactive gases produced during the operation of the reactor from both fission and activation of nearby air, primary / coolant salts, and structural and shielding components. Under normal operation conditions, these sources are held within the reactor loop and the off-gas system or are ejected from the building as controlled effluents with release not to exceed regulatory limits. When the off-gas system is not in use, the gaseous effluents from the reactor consist primarily of the fission daughter noble gases produced within the core. These gases may escape the reactor loop through valves, joints, and flanges, seep into the reactor enclosure, and then escape into the reactor cell where they are swept away into the stack by the cover gas system.

An additional consideration for radiological purposes is tritium from fission and activation of salt and air. Tritium is capable of seeping out of the reactor system, diffusing through piping, valves, and other structural materials. Tritium from fission and primary salt activation is produced at a rate of 1.75E-5 Curies (Ci)/sec [6.48E-4 gigabecquerel (GBq)/sec] in a 1 MWth, continuous operation scenario. Cell activation produces tritium at a rate of 9.8E-12 Ci/sec (3.29E-10 GBq/sec) and activation of the secondary salt produces 7.9E-9 Ci/sec (2.92E-7 GBq/sec).

Table 19.4-4 delineates the anticipated gaseous radioactive production and emissions.

19.4.9.2.2.2 Dose to Maximally-Exposed Individual

The radiation source release rates described in table in Table 19.4-4 were calculated using the methodology described in the following paragraphs.

The escape pathway for nuclides follows one of two paths: either a nuclide is considered to be pulled into the stack the moment it is generated (for tritium and air activation products) or the nuclides leak through the reactor into the enclosure, and from the enclosure into the cell where they are swept away into the stack during normal operations. The former case is modeled as:

$$Release\ Rate\left(\frac{Ci}{year}\right) = S\left(\frac{Ci}{sec}\right) * \left(60\frac{sec}{min} * 60\frac{min}{hr} * 24\frac{hrs}{yr} * 365.24\frac{days}{yr}\right)$$

For fission gases that must diffuse through the reactor, the release was modeled as a two-compartment system, flowing from the reactor to the enclosure to the cell using the following equations:

$$N_{Reactor} = \frac{S + \left[N_{Reactor}^{Parent} \lambda^{parent} * Br_{parent \rightarrow daughter} \right]}{\lambda^{daughter} + (L_{reactor} + L_{Pump\ Seal})}$$

and

$$N_{Enclosure} = \frac{(N_{Reactor}(L_{reactor} + L_{Pump\ Seal})) + \left[N_{Enclosure}^{Parent} \lambda^{parent} * Br_{parent \rightarrow daughter} \right]}{\lambda^{daughter} + (L_{Enclosure} + L_{Pressure})}$$

Where N is the number of atoms in the specified subscript, S is the production rate in atoms per second, Br is the branching ratio of the parent isotope to the daughter isotope,Λ is the decay constant, and L is the leak rate of the specified section of the reactor. $L_{pump\ seal}$ represents losses through the reactor pump's gasket (0.2462%/day); $L_{reactor}$ represents escape through the piping and welds of the reactor (1%/day); $L_{enclosure}$ is, as defined in the MHA, the leak tightness of the steel enclosure vessel (0.01%/day), and $L_{pressure}$ is the quantity of gas that must be pumped out of the enclosure during normal operations to maintain the negative differential pressure in the enclosure (0.0625%/day). For isotopes that do not have a parent considered, $N^{Parent} = 0$.

The nuclides are removed as they enter the cell, as during normal operations the auxiliary heat removal system sweeps the cell atmosphere up to the stack, where the airflow is mixed into the 13,500 m^3/minute being ejected from each of the two 15 ft^2 vents atop the reactor bay. The inventory in these compartments is then turned into a release rate in Ci/year by the following equation:

$$Release\ Rate_{total}\left(\frac{Ci}{yr}\right) = N_{Enclosure} * \lambda\, s^{-1} * L_{Enclosure} s^{-1} * 37,000,000 \frac{Bq}{Ci} * 31556736 \frac{seconds}{year}$$

This result is then divided by 2 for the pair of stacks. Because CAP88 considers all sources to be co-located and both stacks have identical release parameters, this does not change the resultant dose.

These release rates are modeled for release assuming continuous reactor operation via the EPA CAP-88 program (Reference 19.4.9-1). The Clean Air Act Assessment Package - 1988 (CAP-88) computer model is a set of computer programs, databases and associated utility programs for

estimating dose and risk from radionuclide emissions to air. The CAP-88 is a regulatory compliance tool under the National Emissions Standard for Hazardous Air Pollutants.

Data and assumptions used in the CAP-88 analysis include:

- Meteorological wind rose data from the Abilene Regional Airport (one mile from the SERC)
- Release from the two vent stacks atop the research bay [50 ft (15.2 m) tall]
- Vent stack flow rate is 13,500 cubic feet per minute (383 cubic meters per minute) each through a 15 ft^2 opening
- Momentum based plume with a 4.57 m/s exit velocity
- Local food sourcing to bound potential intake doses
- 100-year buildup time

The analysis shows that the maximally-exposed individual from this simulated continuous release is located 656 ft (200 m) north and receives a dose of 0.493 mrem/year (7.26E-3 mSv) after 100 years of buildup, primarily driven by ingestion of tritium. For comparison, the average background dose in the United States from natural sources is approximately 311 mrem/year (3.11 mSv/year). This dose does not include the direct radiation dose from the operating reactor. That dose is expected to be small and will be evaluated for operating license application.

Because the doses to members of the public from normal operations are calculated to be well within the regulatory limits for protection of the maximally exposed individual, the radiological impacts to members of the public from normal operations at the facility are SMALL.

19.4.9.2.3 Liquid Sources of Radiation

No liquid sources of radiation are expected during construction. The MSRR is not expected to produce or handle liquid radioactive sources during operations aside from the molten salts. Small quantities, anticipated to be approximately a few liters over the facility lifetime, of liquid radioactive waste are generated as a byproduct of radiochemistry lab operations. Additionally, usage of the emergency shower or other decontamination processes creates liquid wastes, although these are anticipated to be minute in volume and activity.

While the salt is melted and mobile, salt-borne sources are considered as liquid sources. The fuel salt contains the highest quantity and activity of radionuclides. However, few of these are expected to be able to exit the fuel salt during normal operations. Notable exceptions to this are the noble gases, which are likely to escape the salt and instead enter the sealed reactor head space.

Liquid radioactive wastes are expected to be produced in minute quantities, and no effluent pathway is expected to result in a dose to a member of the public. Therefore, the public dose impacts from liquid effluents are SMALL.

19.4.9.2.4 Solid Radiation Sources

No solid radiation sources are expected as a result of construction. Solid radioactive sources include frozen salts, reactor components and tools that have either been activated or contaminated by the fuel salt, operational startup and calibration sources, and experimental samples such as coupons, as well as solid waste generated from lab operations such as used gloves or contamination covers for surfaces.

The most significant solid radioactive source in terms of activity is the fuel salt when it is solidified in the fuel storage tanks. Solid fuel salt contains numerous fission product species that do not plate out or freely evolve from the salt, and thus when fuel salt is in storage it is anticipated to carry the same material balance as it does in the liquid phase.

Solid radioactive waste at the MSRR facility is primarily generated by operation of the reactor, either as a byproduct of experiments, such as material coupons, or from maintenance, such as reactor structural components and tools. Additional radioactive waste can be produced by laboratory activities, such as contaminated gloves or pipette tips. Solid radioactive wastes may also include sorbing media such as off-gas charcoal, ion exchange resins, and air filters. Solid radioactive waste is packaged to be stored temporarily onsite in a designated cell in the research bay until appropriate disposal.

Solid radioactive wastes are released from the facility following appropriate assay and waste disposal procedures. In the case of materials for which the radioactivity has decayed to a level at or below background, the waste can be disposed of as non-radioactive waste. Alternatively, wastes that take up a substantial volume or are not expected to decay to background in a reasonable time frame can be disposed of by shipment to licensed low-level radioactive waste disposal facilities in containers approved by the Department of Transportation. As Texas is a member of the Texas Low-Level Waste Compact, low-level radioactive waste can be disposed of at the Waste Control Specialists, LLC, facility near Andrews, Texas.

The impact from solid radiation sources is SMALL.

19.4.9.2.5 Dose to Maximally-Exposed Worker

Occupational radiation exposures to workers from all sources at the facility do not result in a dose greater than the occupational dose limits provided in 10 CFR Part 20, Subpart C; therefore, the dose impact to workers from direct exposure sources are SMALL.

19.4.9.3 Radiological Monitoring

Radiological monitoring includes effluent monitoring and environmental monitoring. Environmental monitoring is required per 10 CFR Part 20, Subpart F and as specified in the MSRR Radiological Effluent Technical Specifications. Effluent monitoring consists of continuous measurements of some effluent streams; periodic measurement of radioactive particles trapped on filters, and measurement of samples from effluents released in batches. Surveys of the environment are included in the facility Radiation Protection Program. Baseline condition evaluations are made by taking soil samples and gamma surveys of the area around the site.

Releases to the environment are detected either through the monitoring of contamination, the monitoring of effluents, environmental surveys, or the monitoring of airborne radioactivity within the research bay. These monitoring procedures help protect the environment and public from radioactive releases from the facility and allow understanding of the radiological impact of the facility.

Impacts to public health from implementing radiological monitoring described above are SMALL. The information gained from monitoring helps to control radiological impacts and ensures they remain SMALL.

19.4.9.4 References

19.4.9-1 CAP-88 (Clean Air Assessment Package - 1988) (Version 4.1) [Computer Program]. (1988). U.S. Environmental Protection Agency. http://www.sc.doe.gov/sc-80/sc-83/cap88pc.shtml.

Table 19.4-1 Chemicals Stored and Used during Operation

Chemical	Quantity
Lithium-7 fluoride	567 kg
Lithium fluoride	686 kg
Beryllium fluoride	938 kg
Uranium (IV) fluoride	509 kg
Anhydrous hydrogen fluoride	15 kg
Hydrogen gas	0.5 kg
Sodium hydroxide	1 kg
Universal Indicator	1 gm
Nitric acid, 68%	3.5 kg
Standard solution containing less than 0.1 g each of 20 non-radioactive metals in a 2% aqueous nitric acid solution	1 kg
Helium	1.4 kg
Argon	13 kg

Table 19.4-2 Potential Occupational Hazards

Construction and Decommissioning		
Physical	**Electrical**	**Chemical**
Construction equipment	Power connects/disconnects	Oils and fuels
Working from heights	General wiring	Decontamination fluids
Heavy lifts	Power tools	Cleaners
Slips and falls		Paints/solvents
Hot work		Natural gas
Operations		
Physical	**Electrical**	**Chemical**
Ergonomics	General electrical	Lithium fluoride salts
Slips and falls	Wiring	Beryllium fluoride salts
Lifting	Electronics	Cleaners
Loading and unloading		Oils and fuels
Cranes and hoists		
Elevated work surfaces		
Stairs		

Table 19.4-3 Radiation Sources and Locations in the Facility

Description	Contents
Reactor vessel, fuel handling storage, fuel handling system, internal components	Fuel, dissolved fission products, graphite stringers, startup source, circulating fission products, activated salt, gaseous fission products, activated structural components, particulate/deposited solids, tritium
Reactor piping	Fuel, circulating fission products, particulate/ deposited solids, activated salt, activated structural components, tritium
Off-gas charcoal bed and lines	Tritium, gaseous fission products, particulate solids
Coolant salt loop	Tritium, activated salt
Radiochemistry laboratory and fume hood	Liquid/chemical radioactive wastes
Systems pit atmosphere	Activated atmospheric gases, tritium
Research bay	Calibration and check sources, escaped noble gases and tritium, experimental samples
Research bay storage pit	Radioactive wastes

Table 19.4-4 Anticipated Radioactive Gaseous Effluent Production and Emissions

Nuclide	Production Rate (Ci/Sec) at 1 MW	Steady-state Emission Rate (Ci/Sec)	Total Yearly Emissions per Stack (Ci/Year)
H-3	1.75E-05	1.75E-05*	2.79E+02
C-14	6.78E-12	6.78E-12*	1.07E-04
N-16	1.77E-06	1.77E-06*	2.79E+01
Ar-37	1.67E-11	1.67E-11*	2.63E-04
Ar-41	2.86E-07	2.86E-07*	4.51E+00
Kr-83m	4.72E-01	5.97E-13	9.42E-06
Kr-85m	4.81E-01	3.64E-12	5.74E-05
Kr-85	6.27E-06	5.76E-11	9.08E-04
Kr-87	3.27E+00	2.00E-12	3.15E-05
Kr-88	1.99E+00	6.06E-12	9.56E-05
Xe-131m	1.91E-04	8.86E-12	1.40E-04
Xe-133m	2.00E-03	7.14E-12	1.13E-04
Xe-133	8.60E-02	1.00E-09	1.58E-02
Xe-135m	5.04E+00	1.50E-12	2.37E-05
Xe-135	1.12E+00	9.15E-11	1.44E-03
Xe-138	4.30E+01	8.98E-13	1.42E-05
I-130	5.11E-05	2.92E-15	4.61E-08
I-131	2.40E-02	2.92E-10	4.60E-03
I-132	3.03E+00	6.02E-12	9.50E-05
I-132m	1.10E-02	7.99E-15	1.26E-07

Table 19.4-4 Anticipated Radioactive Gaseous Effluent Production and Emissions

Nuclide	Production Rate (Ci/Sec) at 1 MW	Steady-state Emission Rate (Ci/Sec)	Total Yearly Emissions per Stack (Ci/Year)
I-133	5.17E-01	8.33E-11	1.31E-03
I-134	1.44E+01	4.17E-12	6.57E-05
I-135	1.54E+00	2.50E-11	3.95E-04

* Emission rates with a * indicate nuclides that the nuclide is assumed to immediately escape into the environment.

Figure 19.4-1 Molten Salt Research Reactor Site Layout

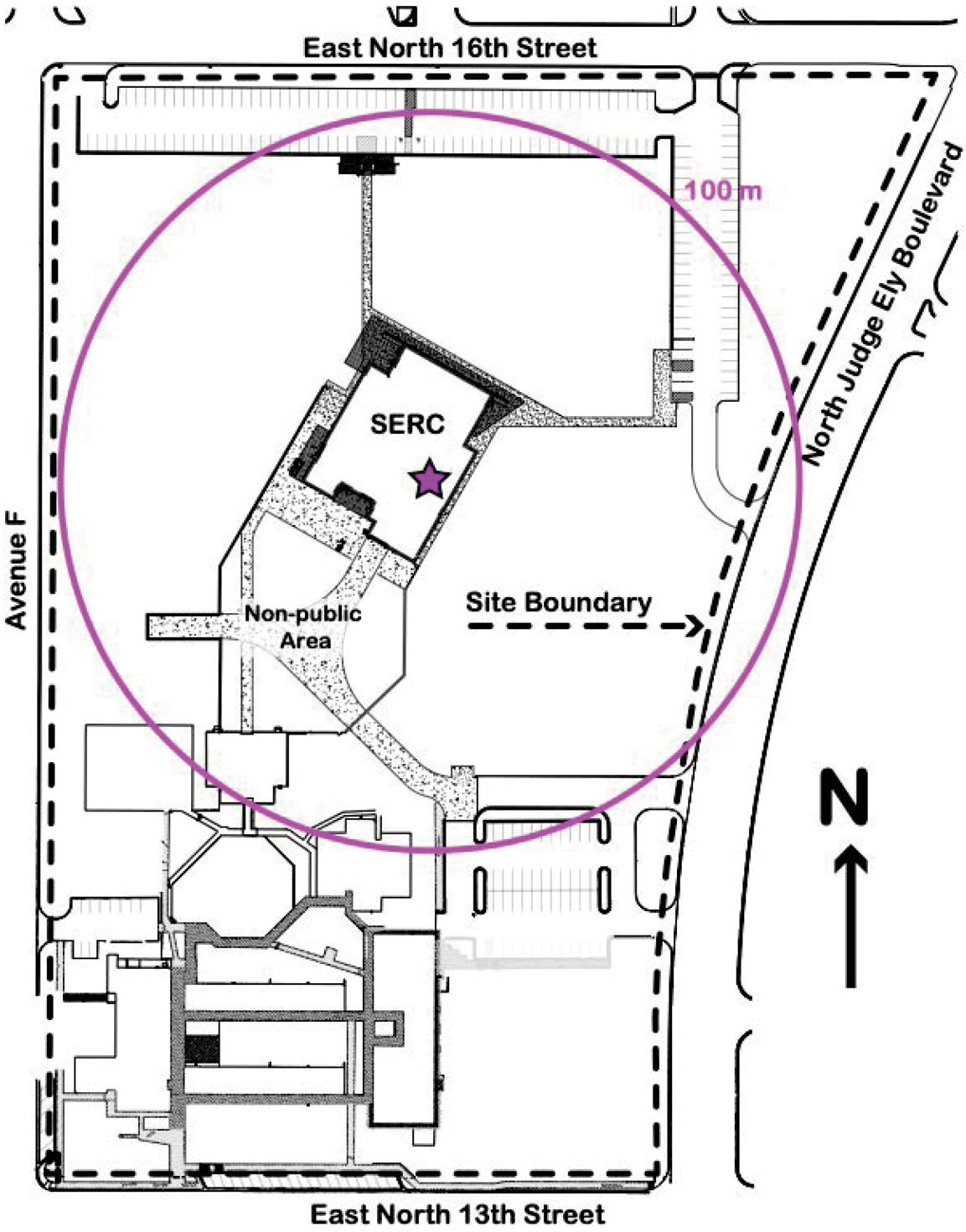
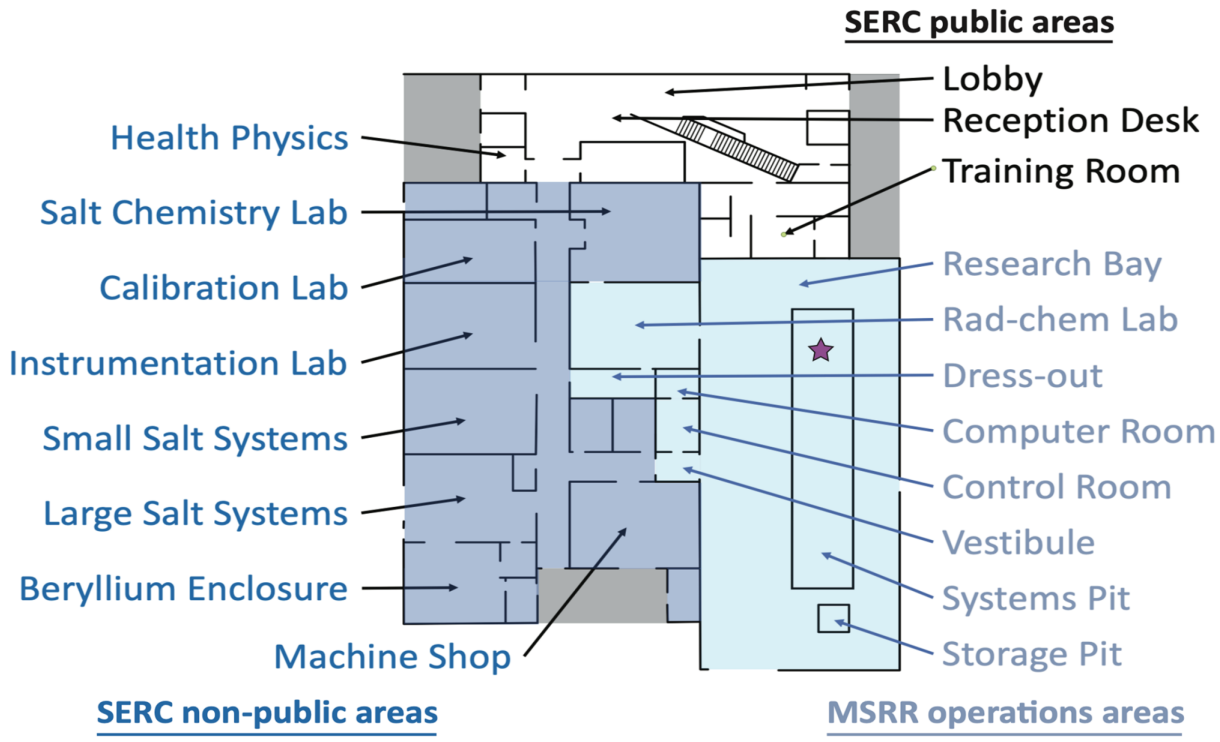


Figure 19.4-2 Science and Engineering Research Center First Floor Layout



19.4.10 Waste Management

19.4.10.1 Construction

During the construction phase, the majority of waste generated is construction and demolition (C&D) waste. Local solid waste haulers are contracted to dispose of C&D waste in permitted local landfills. Such waste includes material produced directly or incidentally by C&D. Examples include scrap lumber, bricks, sandblast grit, glass, wiring, non-asbestos insulation, scrap metal, concrete with reinforcing steel, nails, wood, electrical wiring, rebar, concrete, rubble, and similar construction and demolition wastes. ACU or the construction contractor secures the necessary contracts for proper disposal of C&D wastes.

Only small amounts of hazardous waste (e.g., acids and degreasers) are generated during construction. No radioactive waste is generated during construction. As the facility is a small industrial facility, the impacts of waste management from construction activities are SMALL.

19.4.10.2 Operation

Facility operations generate municipal solid waste commonly known as “trash,” consisting of food waste, plastic film, paper waste, and food product packaging waste. General office and industrial supplies waste is generated at the facility.

Solid wastes generated in conjunction with operation of the facility are managed in accordance with applicable state and federal environmental regulations and disposed in approved and licensed disposal facilities. Solid wastes (e.g., office waste, recyclables) are collected and stored temporarily onsite and are disposed of or recycled locally. The waste is transported from the site by a local sanitary waste entity without being treated or packaged. These activities are typical for a general commercial facility within the Abilene area.

There is expected to be no significant sources of hazardous waste during facility operations; however, hazardous wastes are managed in accordance with a written waste management plan that conforms to state and Federal regulations regarding the storage and disposal of hazardous waste.

Radioactive waste generated by the operation of the facility include, but are not limited to

- reactor operations.
- experiment byproducts (e.g., material coupons).
- laboratory activities (e.g., gloves, pipette tips).
- maintenance activities.
- decontamination activities.

Systems designed to support the safe and efficient management of radioactive waste streams are described in Section 19.2.5. These waste systems are operated in accordance with procedures such that the final waste form is acceptable for transport in Department of Transportation or NRC-certified shipping containers. There is no on-site disposal of radioactive wastes during operations.

Used fuel is not removed from reactor coolant salts and, therefore, is not stored separately. The entire reactor coolant inventory, including fuel, gets returned to the DOE at the end of the performance period.

Reprocessing is currently unlikely in the United States, and an open fuel cycle is anticipated. Management of used nuclear fuel is addressed in 10 CFR 51 23 and the associated NUREG-2157, “Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel” (Reference 19.4.10-1). NUREG-2157 concluded that the impact for at-reactor storage would be small for short-term, long-term, and indefinite storage. NUREG-2157 did not address non-LWRs. To provide additional guidance for non-LWR license applications, the DOE prepared “Non-LRW Reactor Fuel Environmental Data” (PNNL-29367) (Reference 19.4.10-2). The environmental impacts for continued storage of LWR fuel described in NUREG-2157 are considered to bound any impacts of the MSRR fuel storage until return to DOE.

Based on the quantities of waste, systems designed to manage radioactive waste streams, and waste management, impacts from all types of waste generated during operations, including impacts on the capacity of waste management facilities, are SMALL. Impacts from waste transportation are discussed in Section 19.2.5.

19.4.10.3 References

19.4.10-1 U.S. Nuclear Regulatory Commission. “Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel,” NUREG-2157, Vol. 1, September 2014.

19.4.10-2 Pacific Northwest National Laboratory. “Non-LRW Reactor Fuel Environmental Data,” PNNL-29367, September 2020.

19.4.11 Transportation

This section addresses the direct and indirect contributory effects from the construction, operations, and decommissioning of the proposed MSRR when added to the effects from other past, present, and reasonably foreseeable future actions on transportation infrastructure. The geographic area of analysis for evaluation of cumulative impacts on transportation is primarily the same as that used in Section 19.2.1 and Section 19.3.1 and includes the 15.17 acres (6.14 ha) within the site boundary and the 5 mi (8 km) region surrounding the proposed MSRR. However, the roads for routes that could be used for shipments or disposal of wastes also are considered. Transportation infrastructure includes roadways, rail lines, airports, and traffic control devices. As discussed in Section 19.4.8, the traffic impacts are SMALL during construction and decommissioning and SMALL during operations.

Other, non-related construction projects in the area can produce an increase in vehicle traffic on roads within the 5 mi (8 km) radius of the proposed MSRR. Most existing roads are sufficient to handle the project transportation activities, and alternative routes can be used to minimize transportation impacts. In some cases, however, a nominal increase in traffic can occur, especially if MSRR construction workers and vehicles use the same roads during peak traffic periods.

19.4.12 Postulated Accidents

This section describes the postulated events that are within the design basis of the facility and a maximum hypothetical accident (MHA) that bounds the radiological consequences of the postulated events that release fission products. No radiological accidents exist during construction. It is anticipated that accidents during decommissioning would be bounded by the MHA.

19.4.12.1 Event Categories

The events are grouped according to type and characteristics of the events. The event categories are:

- MHA (fuel salt spill into reactor thermal management system)

- non-MHA salt spills
- Cooling anomalies
- Fuel handling anomalies
- Positive reactivity insertions
- Surveillance system malfunction
- External events
- Loss of normal electrical power

Acceptance criteria for these figures of merit represent design limits that ensure the MHA is bounding. The MHA dose consequences are evaluated in Section 13.2.

19.4.12.1.1 Maximum Hypothetical Accident

The MHA for the MSRR is a combined failure of multiple systems resulting in a fission product release. After one full-power year of continuous operations at 1 MWth with no gas removal, the accident is initiated with a simultaneous failure of the reactor system and loss of normal electric power, resulting in a fuel salt and gas spill into the RTMS and reactor enclosure that cannot be mitigated by routing the gases to the off-gas system. In this scenario, fission gases and particulate matter begin within the reactor enclosure (assuming immediate liberation from the salt), followed by a slow leakage into the reactor cell. As described in Section 13.1.1, non-physical assumptions drive radionuclide movement and bound the system response to other postulated events. The MHA is a bounding event with conservative radionuclide transport assumptions that challenge the important radioactive retention features of the functional containment. The dose of a maximally exposed person staying at the site boundary full time for 60 days would be less than 81 mrem as shown in Chapter 13. Thus the impact of the MHA would be SMALL.

19.4.12.1.2 Non-Maximum Hypothetical Accident Salt Spills

In each salt containing system, there are several barriers between any radionuclides within the salt and the outside environment. The first is the salt itself. The second barrier is the piping and components where the salt is intended to remain throughout operation. This includes the fuel handling system, reactor system, and coolant system. A third barrier is the enclosure in which each salt system is housed. The fourth barrier is the reactor cell.

Within the reactor systems several possible barrier failures can occur that might result in a loss of fuel or coolant salt inventory:

- Salt loss from the piping and components of the reactor system, including the drain tank, is one possible failure scenario.
- Egress of coolant salt from the coolant loop.

- The third and final salt inventory failure scenario is loss of salt from the storage tanks in the fuel handling system.
- Salt from the reactor loop may enter the coolant loop through a tube break in the heat exchanger
- Salt from the fuel handling system may exit the storage tanks and enter the salt storage enclosure. This accident involves either unirradiated salt or post-critical irradiated fuel salt.
- Heat exchanger tube breaks and fuel storage tank failure are bounded by the MHA.

As described in Section 13.1.2, all of these scenarios are bounded by the MHA.

19.4.12.1.3 Cooling Anomalies

Reductions in cooling rate can occur for a variety of reasons, including electrical power loss, pump malfunction, pipe rupture, full or partial blockage of flow in either the fuel or coolant loops, accidental drain of the coolant salt, or loss of ultimate heat sink. A full drain of the fuel salt from the reactor loop is performed in response to these scenarios, allowing for safe indefinite decay heat removal from the drain tank.

Fission power is immediately affected by a loss of fuel flow. This is due to the strong negative temperature feedback of the MSRR. The power quickly drops from just under 1 MWth to only a few tens of kilowatts over the course of a few minutes following the loss of fuel flow. The remaining power, along with the heat capacity of the salt and insulation, then allows for the fuel salt to be maintained in a molten state as it flows into the drain tank after a loss of power.

As described in Section 13.1.4, a reduction in cooling accident is shown to have no substantial bearing on the fundamental safety of the reactor. Even with no control element activation, there are no safety-critical, negative consequences: no unsafe pressure, temperature, or power excursions occur. This is primarily due to the strong negative reactivity temperature feedback inherent in the reactor. This feedback gives the reactor a strong anchor to the initial temperature based on neutronics and quickly returns the reactor vessel temperature to a temperature that is similar to the initial reactor temperature distribution, though there is a small increase in temperature.

As described in Section 13.1.10, a loss of normal electric power does not result in the breach of any fission product barrier and radiological consequence. Fuel salt rapidly relocates to the drain tank while the reactor cell is isolated from the environment. Auxiliary Heat Removal is terminated. Sufficient thermal mass exists within the reactor system, RTMS, reactor enclosure, reactor cell, and research bay to ensure that no thermal limits are breached. Heat transfer is through entirely passive means such as conduction, natural convection, and thermal radiation. The system may be maintained in this state indefinitely.

19.4.12.1.4 Fuel Handing Anomalies

No amount of excess fuel salt injected into the system affects the fuel level in the reactor vessel once filled. Because the RAV also acts as an expansion tank and remains subcritical at any fuel level due to its geometry, criticality accidents cannot occur due to overfilling. If the fuel salt concentration were to change, however, there is a realistic mechanism to increase power and temperature. Section 13.1.5.3 presents the analysis to simulate a 1 percent relative increase in the overall UF_4 concentration of the fuel. The results demonstrate the lack of need for intervention, even from control rods, to maintain a safe temperature within the reactor. While it certainly is desirable to use control rods to maintain the desired power output and return to the standard operating temperature, even a complete failure of all three rods to lower farther into the graphite results in a safe, steady-state operation for an indefinite period of time.

19.4.12.1.5 Positive Reactivity Insertions

A number of scenarios can result in reactivity increases in the MSRR graphite core. As described in Section 13.1.5, these scenarios are analyzed via RELAP5 analysis to ensure that the transient case of each does not exceed safety limits.

Pressure and temperature limits are the most direct concern in each case, with power being an indirect concern only to the degree that it may result in rapid temperature increases in the fuel or structural steel of the reactor. Despite pressure being a primary concern, it is found not to fluctuate in the vast majority of cases and, therefore, is included only when found to have any changes during the transient.

Table 13.1-3 summarizes temperature and power changes for a number of reactivity insertion events.

19.4.12.1.6 Surveillance System Malfunction

There are currently plans for three types of surveillance in the MSRR:

- Coupons are used as a proxy to examine corrosion over time and to monitor the overall health of the structural material as the reactor continues to operate.
- Samples of salt are taken from the RAV. A sample canister is filled in the RAV and then sealed and removed for surveillance work with salt chemistry.
- Cover gas samples are taken to determine the amounts and kinds of fission products that are found in the head space above the RAV, giving additional insight into the behavior and transport of fission products.

Surveillance can contribute to accidental malfunction in several ways. A flow-loss accident can occur if a metal coupon or other surveillance apparatus blocks flow. This accident scenario is discussed in Section 13.1.7 where it is shown that the reactor is able to safely bring down power to compensate for lack of flow.

In another scenario, a sealed sample capsule could rupture and cause damage to the RAV. There is no pressure differential or temperature increase to the sealed capsule that drives this event, and this event is bounded by the analysis presented in Section 13.1.5.5.

Finally, any removal of material from the reactor for study can result in fission product release and radiological exposure. Minimizing sample sizes, constant monitoring, correct use of personal protective equipment, and use of shielded glove boxes for chemistry work all are essential to maintaining safe conditions when working with samples.

19.4.12.1.7 External Events

19.4.12.1.7.1 Earthquakes

Significant earthquakes are not a natural phenomenon known to occur in the vicinity of the chosen site. A search of historical earthquake data finds the largest earthquake that has occurred near Abilene was a magnitude 4.6 earthquake at a distance of 121 mi (195.1 km). This magnitude is insufficient to cause damage to modern buildings, so earthquakes are not considered a credible source of transient disturbance.

19.4.12.1.7.2 Tornadoes

An analysis of the ability of the facility to withstand extreme winds can be found in Section 3.2. Regardless of the building's level of resistance to winds, the concrete and steel containment structures are set into the systems pit and far exceed the weight and density at which heavy winds are a concern for the reactor itself. Although damage to the reactor building is likely if it is in the direct path of a strong tornado, a reactor shutdown is standard protocol in the event of severe weather warnings, and no safety critical components are foreseeably affected by this occurrence. Even with a complete loss of the facility building, the reactor remains in a safely shutdown state without operator intervention.

19.4.12.1.7.3 Floods

The siting location does not historically experience flooding, so flooding is not foreseeably a credible external event. Nonetheless, analyses performed show that filling the reactor containment area with water does not pose a criticality concern. The loss of electrical component functionality is bounded by the loss of normal electric power accident in Section 13.1.10.

19.4.12.1.7.4 Volcanism

The chosen site is geologically stable and volcanic activity is not a credible threat to the reactor facility.

19.4.12.1.7.5 Missile Impact

Credible missile impact can result from objects transported by high winds or from tools or other equipment falling onto the reactor from above. While detailed structural analysis has not been performed, the heavy shielding concrete is considered more than sufficient to withstand any credible impact that might occur.

19.4.12.1.8 Loss of Normal Electrical Power

The MSRR does not have backup or emergency electrical power required to safely bring the reactor to a shutdown state from operating or emergency conditions. If the off-site electrical power is lost, all electrically operated systems stop in the MSRR. As demonstrated in Section 13.1.10, a complete loss of normal electric power causes the fuel salt to drain, the reactor cell to be confined, and the auxiliary heat removal system to cease. These actions do not damage any radionuclide barriers. Fuel salt can remain in the drain tank where it remains sufficiently cool indefinitely through passive heat dissipation without environmental impact.

19.4.13 Environmental Justice

This section addresses the potential health and environmental effects from the construction, operations, and decommissioning of the proposed SERC facility on low-income populations and minorities of Abilene and the area surrounding the proposed site. The addressed topics are in accordance with the National Environmental Policy Act. These health effects are measured by risks and rates that could concern fatal or nonfatal adverse impacts on human health. This includes cancer and other deficiencies that go under bodily impairment, infirmity, illness, or death. A high number of adverse health effects occur when the risk or rate of exposure to any kind of environmental hazard is significant. This affects the minority population and the low-income communities. This refers to the significant environmental impacts that are harmful to the ecological, cultural, human health, economic, and social parts of the environment. All these possible impacts to low-income and minority groups are considered.

With a 2010 census recording a population of 117,420, approximately 28 percent of the city of Abilene identified themselves as a minority group. Below is a chart labeling what percentages of people identified as a certain race. As shown in Table 19.4-5, white/Caucasian held the majority group at 72.2 percent.

Figure 19.4-3 shows a demographic plot of the highest income and the lowest income groups in Abilene. Red means there is an income decrease currently, and blue represents an income raise. The selected area is where ACU is located; it is neither red nor blue. Since it is not in an area of income decrease, it most likely is not negatively affecting the poor in Abilene.

Based on 2010-2019 data, an average of 16.5 percent of people in Abilene, Texas, identified as persons in poverty and living below the federal poverty threshold. Persons in poverty were at 16.5 percent (Table 19.4-6). In 2018, the federal poverty income threshold was $25,465 for a family of four with two children and $17,308 for a single parent of one child. In the past 12 months, 2500 hundred women and 1500 men between the ages of 18 to 24 live in poverty in Abilene, Texas.

The environmental justice impact analysis evaluates the potential for negative effects on minorities and low-income communities. These effects can occur during construction, operations, and decommissioning of the MSRR facility. These potential impacts have been described in other sections of this Environmental Review.

Human health and environmental life are two general subjects that need to be considered; however, these fit into radiological human health impacts, non-radiological human health impacts, noise impacts, and traffic impacts.

Environmental impacts during MSRR construction include extra noise, dust, traffic, employment, and possibly housing impacts. All construction, noise and dust are definite impacts although they are short-term and only onsite. People living on access roads to the site are temporarily affected by traffic as well. These effects are not expected to be adverse because they are temporary impacts. Additionally, the construction activities are modest (i.e., installation of MSRR components within the existing SERC). With some construction sites, increased demand for housing rises. This is not likely to be the case because most construction workers available to this job already live in the Taylor County area, which means the cost of housing does not rise. Overall, construction is not likely to damage minorities or the low-income population.

Potential impacts to minority and low-income populations during MSRR facility operations consist of mostly radiological human health impacts which are SMALL as noted previously. Other than that, noise and traffic impacts are SMALL because the increase in staff to operate the MSRR is small.

Similar to the impacts coming from construction, potential impacts to minority and low-income groups consist of noise, dust, traffic, employment, and possible housing impacts. These, similar to construction, are short-term. Activities associated with the construction, operations and decommissioning of the proposed project are SMALL or non-adverse to the general population.

Where the proposed facility is located, buildings and roads already surround it, so it is not in a place that endangers the wildlife. This means that the typical lifestyles of people in the region stay intact because there is little to no impact on area wildlife. There are no foreseen disproportionate effects on wildlife. Minority and low-income populations that reside in accessible areas might be affected temporarily by noise,

dust, and an increase in traffic during construction, operation, and decommissioning. These impacts are temporary, however, and have a SMALL impact due to their short-lived nature. Facility operations do not adversely affect these groups of people because the level of potential radiological doses to the public from the MSRR are well below the limits placed by the federal, state, and local governments. Minorities and low-income communities do not experience adverse effects from the proposed MSRR facility.

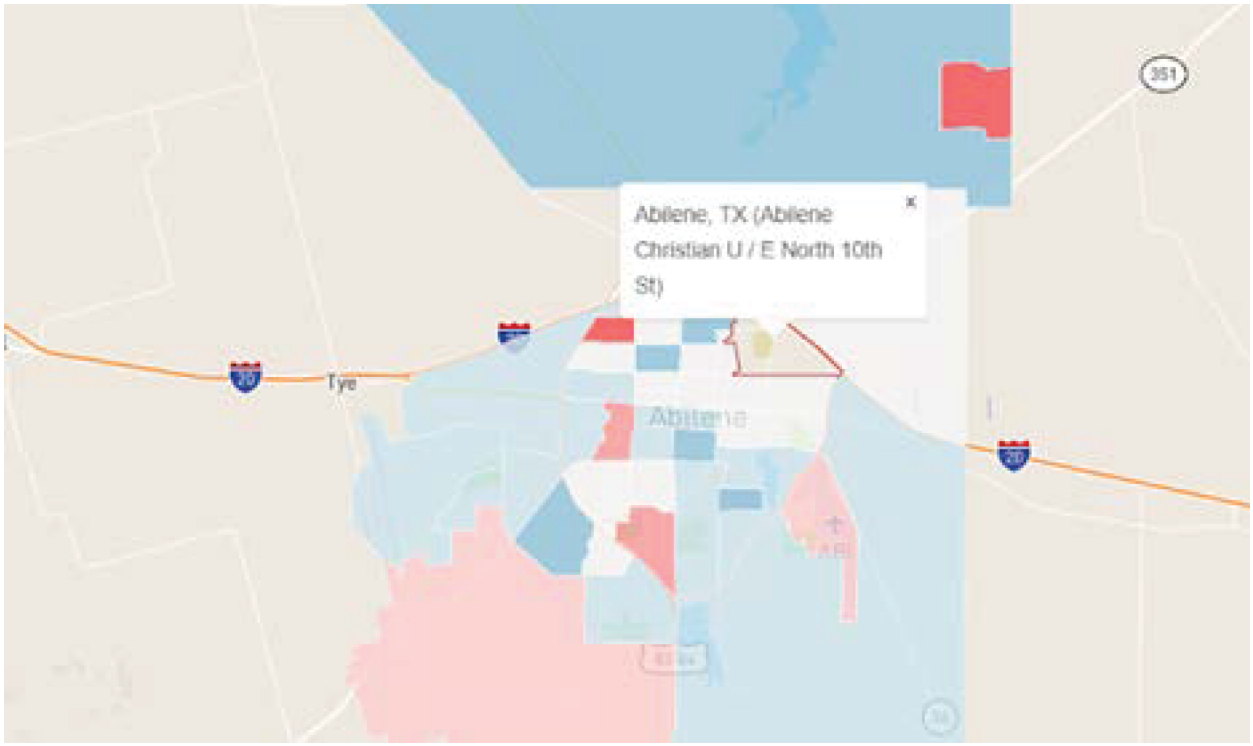
Table 19.4-5 Abilene, Texas Demographic Data (U.S. Census Bureau QuickFacts: Abilene, Texas)

Race and Hispanic Origin	Percent
White alone	72.2%
Black or African American alone	10.6%
American Indian and Alaska Native alone	0.9%
Asian alone	2.3%
Native Hawaiian and Other Pacific Islander alone	0.0%
Two or More Races	3.1%
Hispanic or Latino	26.6%
White alone, not Hispanic or Latino	58.3%

Table 19.4-6 Abilene Poverty Statistics (LiveStories U.S. Census Bureau QuickFacts: Abilene, Texas)

Income and Poverty	
Median household income (in 2019 dollars), 2015-2019	$50,659
Per capita income in past 12 months (in 2019 dollars), 2015-2019	$24,529
Persons in poverty, percent	16.5%

Figure 19.4-3 Abilene, Texas Demographics and Population Statistics (NeighborhoodScout)



19.4.14 Cumulative Effects

This section contains a summary of potential cumulative environmental impacts associated with construction, operation, and decommissioning activities for the MSRR site in combination with other past, present, and reasonably foreseeable actions or projects in the area. The term cumulative impact, which was previously defined in the regulations of the Council on Environmental Quality implementing NEPA, has been repealed. However, in the revised 2022 NEPA regulations, the definition of effects or impacts (40 CFR 1508.1(g)) includes evaluation of cumulative effects.

A summary of past, present, and reasonably foreseeable projects that could have a cumulative effect within the geographic area of interest are listed in Table 19.4-7. The cumulative impact assessment for each geographic area of interest that may be affected by the project is presented below. The resources assessed include land use and visual resources; air quality and noise; geologic environment; water resources; ecological resources; historic and cultural resources; socioeconomics; human health; waste management; transportation; and environmental justice. According to the Council on Environmental Quality's "Considering Cumulative Effects Under NEPA," the establishment of an appropriate geographic area of analysis depends on the project impact zone or resource or system that is affected by the project (Reference 19.4.14-1). The analysis of cumulative environmental impacts is resource specific. The geographic area of analysis of the past, present, and reasonably

foreseeable actions considered are based on the environmental effects that can occur to each of the affected resources under consideration and is described below for each resource area.

19.4.14.1 Land Use and Visual Resources

The description of the affected environment in Section 19.3 serves as a baseline for the land use and visual resources cumulative impact assessment. The geographic area of analysis for evaluation of cumulative effects on land use and visual resources is the same as that used in Section 19.4.1 and includes 15.17 acres (6.14 ha) within the site boundary and the 5-mi (8 km) region surrounding the site. As discussed in Section 19.4.1.1, construction and operation and decommissioning impacts from the facility on land use are SMALL. Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be assessed to determine cumulative effects on land use and visual resources. Relevant actions that are considered in this cumulative effects analysis are limited to proposed or in-progress developments that could alter current land use status and the visual character of the area.

19.4.14.1.1 Land Use Resources

The proposed site is located in the existing SERC that encompasses 15.17 acres (6.14 ha) of land located on the southwest corner of the ACU campus in Abilene, Texas. The site is located in Taylor County, immediately south of the university. As discussed in Section 19.4.1, impacts to land use as a result of the construction, operation, and decommissioning of the MSRR are SMALL.

The “region” of the site is defined as the area within a 5-mi (8 km) radius of the existing SERC site (Figure 19.3-3). Major land use within the region is described in Section 19.3.1.3 and depicted in Figure 19.3-2. The dominant land use in the region is residential and commercial businesses (as almost the entirety of the city of Abilene is located within the 5-mi (8 km) radius) and open fields with some being used for agriculture to the east and southeast.

As noted in Table 19.4-7, endeavors that can contribute to cumulative impacts include proposed and in-progress construction and road maintenance projects. Land use impacts are confined to the 15.17-acre (6.14-ha) SERC site. The projects that can cause cumulative impacts to land use include proposed and in-progress commercial facilities development projects in the downtown Abilene area. The noted projects are in areas previously impacted. Also included is the Cedar Creek Waterway project that would develop a greenbelt region along the creek and could have minor beneficial land use changes. Therefore, cumulative impacts to land use are SMALL.

19.4.14.1.2 Visual Resources

The projects and activities described in Table 19.4-7 result in minimal changes to the existing view because construction, operations, and decommissioning occurs primarily, if not entirely, inside existing facilities. Changes to the view shed within the vicinity of the MSRR project area occur temporarily during construction because of the presence of heavy equipment if construction/ demolition/renovation activities listed in Table 19.4-7 in the vicinity of the project occur at the same time as the MSRR construction or decommissioning activities. These impacts are temporary, however, and cease when construction or decommissioning is complete. As described in Section 19.4.2.1, the proposed MSRR site is considered of low scenic quality and has a low sensitivity rating.

Overall, because of the MSRR construction and decommissioning activities occurring within or adjacent to existing facilities and the proposed MSRR site being of low scenic quality, cumulative impacts to visual resources are SMALL.

19.4.14.2 Air Quality and Noise

19.4.14.2.1 Air Quality

The description of the affected environment in Section 19.4.3.1 serves as a baseline for the air quality cumulative impact assessment. Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be considered contributors to cumulative impacts to air quality.

As described in Section 19.4.3.1, air emission impacts from construction and operations and decommissioning are SMALL as emissions are controlled at the source where practicable and maintained within established regulatory limits designed to minimize impacts. Air emission impacts as a result of concurrent construction activities are expected at both the proposed facility and some of the projects listed in Table 19.4-7. These proposed and in-progress projects can overlap with the construction schedule for the MSRR. The majority of the facilities listed in Table 19.4-7 can contribute to cumulative impacts to regional and local air quality during operations. However, mitigation measures to ensure compliance within applicable regulatory limits established by the National Primary and Secondary Ambient Air Quality Standards (40 CFR Part 50) and National Emission Standards for Hazardous Air Pollutants (40 CFR Part 61) are implemented to minimize impacts to local ambient air quality and the nuisance impacts to the public in proximity to the project. Impacts to air quality from construction activities are expected to be minor, localized, and short-term; therefore, overlapping construction schedules are not expected to contribute significantly to cumulative effects.

Given their proximity to the site, the proposed and in-progress construction projects and operational facilities can contribute to cumulative impacts to air quality. The multiple projects and facilities are governed, as required, through

the TCEQ. Implementation of TCEQ regulation and processes, as appropriate, ensure that new projects and existing operational facilities do not result in regional air quality degradation. Therefore, the cumulative impacts of construction and operational related criteria pollutants to air quality are SMALL.

19.4.14.2.2 Noise

The description of the affected environment in Section 19.3.3.3 serves as a baseline for the noise cumulative impact assessment. The geographic area of analysis for evaluation of cumulative effects from noise emissions includes the ACU campus. Noise impacts resulting from construction, operation, and decommissioning of the facility are discussed in Section 19.3.3.3 and Section 19.4.3.2 and are SMALL. The MSRR is installed and operated within the existing SERC. There is little addition to the existing natural noise environment during operation. During the construction periods for the site and the potential or in-progress facilities listed in Table 19.4-7, additional impacts to noise are expected in the immediate area around each site if construction happens at the same time. Noise levels from construction equipment are expected to attenuate rapidly with distance. Table 19.4-71 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be assessed to determine cumulative effects on noise.

Therefore, incremental contribution of cumulative impacts to noise in the region are SMALL.

19.4.14.3 Geologic Environment

The description of the affected environment in Section 19.3.4 serves as a baseline for the geologic environment cumulative impact assessment. The geographic area of analysis for evaluation of cumulative effects on geologic resources is the same as that used in Section 19.4.4 and includes the 5-mi (8 km) vicinity surrounding the site. As discussed in Section 19.4.4, construction, operation, and decommissioning impacts from the site on the geologic environment are SMALL.

Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be assessed to determine cumulative effects on the geologic environment. Of these projects, none are considered likely to contribute to cumulative impacts to geological resources.

The MSRR is installed in the existing SERC on the ACU campus. No prime farmlands or farmland of site-wide importance are present onsite. As discussed in Section 19.4.4, soil loss, including the loss of prime farmland soils and other important farmland soils and the consumption of geologic resources is minimal. Therefore, cumulative impacts to geological resources are SMALL.

19.4.14.4 Water Resources

The description of the affected environment in Section 19.3.5 serves as a baseline for the water resources cumulative impact assessment. The geographic area of analysis for evaluation of cumulative effects on water resources is the same as that used in Section 19.4.5 and includes the 15.7 acres (6.14 ha) within the site boundary. Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be assessed to determine cumulative effects on water resources. Construction, operation, and decommissioning impacts to water resources are SMALL.

19.4.14.4.1 Hydrology

There are no surface water resources located on the site; therefore, there are no direct impacts as a result of alteration of streams or water bodies. The nearest water bodies are the nearby Cedar Creek (west of site) and Rainy Creek (east of site), both of which flow into Lake Fort Phantom Hill north of the university and outside Taylor County. Lake Fort Phantom Hill is located approximately 5 mi (8-km) north of ACU. No surface water is diverted, redirected, or dammed to support installation of the MSRR. Construction, operation, and decommissioning activities are contained within the SERC. No intake or discharge facilities are constructed or renovated as part of this project. It is anticipated that at the potential new construction sites and maintenance activities, including greenbelt development listed in Table 19.4-7, best management practices, are used in accordance with TCEQ and federal rules to prevent sediment runoff and subsequent siltation in receiving streams during construction. Because of the minimal impacts to hydrology presented by each potential project in the vicinity of the site, cumulative hydrologic impacts are SMALL.

19.4.14.4.2 Surface and Groundwater Resources

As discussed in Section 19.3.5, Section 19.4.5.1, and Section 19.4.5.2, impacts to surface water and groundwater use during construction, operation, and decommissioning are SMALL. Table 19.4-7 shows the proposed and in-progress projects and facilities that can contribute to cumulative impacts to water use.

The SERC facility in which the MSRR is located uses municipal water from the City of Abilene. The City of Abilene does not currently use groundwater in its potable water supply. The only water use of note is water consumed through basic utilities such as fire protection systems, toilets, sinks, and potentially chemical showers. The City of Abilene operates three water treatment plants with a maximum capacity of 46 million gallons per day.

Wastewater from the SERC discharges to the sanitary collection system, then to the Buck Creek Pump Station (with a 24 million gallons per day capacity) and then to the Hamby Water Reclamation Facility that discharges to Kirby Lake and Lake Fort Phantom Hill. Industrial users are required to adhere to an industrial pre-treatment program to protect wastewater quality.

It is currently unknown what amount of water is used and how much wastewater is generated by the operating and proposed or in-progress projects that share these same water sources and discharge facilities; however, because of the relatively small impacts to water and wastewater systems during construction, operation, and decommissioning of the MSRR, and the present operating capacities of the water systems, cumulative impacts from water use are SMALL, and the incremental contribution to cumulative impacts from the installation and operation of the MSRR are SMALL.

Groundwater withdrawals or returns are not required during construction, operation, or decommissioning of the facility. Because of the lack of groundwater use, the use of municipal water systems, a pre-treatment program for industrial users, and NPDES/TPDES regulations, cumulative impacts to water quality are SMALL.

19.4.14.5 Ecological Resources

The description of the affected environment in Section 19.3.6 serves as a baseline for the ecological resources cumulative impact assessment. The geographic area of analysis for evaluation of cumulative effects on ecological resources is the same as that used in Section 19.4.6 and includes the 15.17 acres (6.14 ha) within the site boundary and an approximate half-mile (0.8 km) region surrounding the site. As discussed in Section 19.4.4, construction, operation, and decommissioning activities are contained within the SERC. Impacts from construction on terrestrial and aquatic ecosystems, including protected species, are SMALL.

Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be assessed to determine cumulative effects on ecological resources. Projects identified for consideration in this cumulative effects analysis are on already developed or disturbed land. The development of a greenbelt along Cedar Creek can provide minor habitat improvement for aquatic and terrestrial species as it is developed; therefore, cumulative impacts to terrestrial and aquatic ecological resources are SMALL.

19.4.14.6 Historical and Cultural Resources

The description of the affected environment in Section 19.3.7.2 serves as a baseline for the historical and cultural resources cumulative impact assessment. The geographic area considered in this analysis of cumulative effects on historical and cultural resources is the same as that used in Section 19.4.7 and includes the 15.17 acres (6.14 ha) within the site boundary. The nearest listed National Register of Historic Places property is located on the ACU campus approximately a quarter mile (0.4 km) from the proposed MSRR site; however, no onsite historic properties are associated with the proposed site. As discussed in Section 19.4.7, impacts from construction, operation, and decommissioning of the facility are SMALL.

Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be assessed to determine cumulative effects on historical and cultural resources. No historic properties are impacted by the site; therefore, no additional cumulative impacts to historic and cultural resources occur. Consequently, potential cumulative impacts of the site are SMALL.

19.4.14.7 Socioeconomic Environment

The description of the affected environment in Section 19.3.8 and Section 19.4.8 serves as a baseline for the socioeconomic cumulative impact assessment. The geographic area of analysis for evaluation of cumulative effects on socioeconomic resources is the same as that used in Section 19.4.8 and includes the 15.17 acres (6.14 ha) within the site boundary and Taylor County. As discussed in Section 19.4.8.1, impacts from construction, operation, and decommissioning of the facility have a SMALL impact on socioeconomic conditions. Impacts to transportation associated with the construction, operation, and decommissioning of the site are discussed in Section 19.4.8.2 and are SMALL.

Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that could contribute to determine cumulative socioeconomic effects. Cumulative impacts from actions identified in Table 19.4-7 on aspects of socioeconomics, including water/wastewater systems, population growth, local tax base, the labor force, transportation, public education, and recreational facilities in conjunction with the incremental contribution to cumulative impacts are SMALL.

19.4.14.8 Human Health

The geographic area of analysis for evaluation of cumulative effects on human health is the same as that used in Section 19.4.9 and includes the 15.17 acres (6.14 ha) within the site boundary and the 5-mi (8-km) region surrounding the site. As discussed in Section 19.4.9, impacts from construction, operation, and decommissioning of the facility have a SMALL impact on human health.

Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be assessed to determine cumulative effects on human health.

19.4.14.8.1 Non-radiological Impacts

Construction, operation, and decommissioning of any facility includes potential hazards to workers typical of any construction site. Normal construction safety practices are employed to promote worker safety and to reduce the likelihood of worker injury during construction. Therefore, because controls are in place to limit injuries and illnesses and occupational impacts rarely reach beyond the construction site, cumulative occupational hazards from construction are SMALL.

Potential non-radiological public health hazards pertaining to the construction and operation of facilities in the 5-mi (8-km) region surrounding the site are associated with routine emissions and discharges as well as accidental spills/ releases. To minimize potential exposure to the public, control systems are in place to limit emissions in accordance with Federal, state, and local requirements. These controls include conveyance of wastewater to appropriate, approved, wastewater treatment facilities; discharges to Waters of the United States in accordance with NPDES/TPDES permits; implementation of Spill Prevention Control and Countermeasure Plans; and air emission controls. Cumulative environmental impacts to water and air resources to human health are SMALL.

19.4.14.8.2 Radiological Impacts

As described in Section 19.4.9, the radiological impacts from construction, operation, and decommissioning of the MSRR are SMALL. There are no nuclear reactor facilities that contribute to radioactive exposure. As described in Section 19.3.9.2, there is no background radiation for natural or man-made sources in the vicinity of the site that result in abnormal radiation hazards to the public. For analysis of cumulative impacts, the geographic area of interest considered was 5 mi (8-km) beyond the site boundary. Hendrick Health operates two hospitals in Abilene, Texas, that provide imaging services to patients. Hendrick Health is Abilene's largest medical imaging provider. Services also include radiation therapy and nuclear medicine. Table 19.4-7 summarizes past, present, and future projects and actions that contribute to cumulative effects.

Based on the anticipated negligible radiation exposure to members of the public outside these facilities and the distance of these facilities from the proposed MSRR site, cumulative radiological impacts to members of the public during operation are SMALL.

19.4.14.9 Waste Management

The geographic region of interest for the evaluation of cumulative impacts from the disposal of nonradioactive and radioactive waste is the area within a 5-mi (8-km) radius of the SERC facility. There are no nuclear power plants that contribute to radioactive exposure. There are two operating medical facilities: Hendrick Medical Center and Hendrick Medical Center South.

Because of its relatively small size and operating staff, the contribution of the MSRR project on the local non-radioactive waste management resources and disposal capacity are SMALL, and the percentage contributed when considering other current and proposed projects also are SMALL. Non-radioactive liquid waste generated during construction, operation, and decommissioning, including hazardous waste, is disposed at a licensed facility. Incidental amounts of non-radioactive chemicals can enter the city wastewater collection system as a result of routine facility maintenance or laboratory activities. Administrative controls are implemented to ensure effluents meet applicable pre-treatment standards. There are no expected significant sources of hazardous waste during

operations. The MSRR radioactive- and nonradioactive-waste management program is based on a waste minimization and pollution prevention framework as described in Section 19.2.5.3. It is anticipated that liquid and solid waste from the projects and facilities, such as the medical facilities, listed in Table 19.4-7 use the same waste disposal facilities as the MSRR. The MSRR construction, operation, and decommissioning have a negligible cumulative impact on non-radioactive waste management and disposal resources.

For radioactive waste generated during operation and decommissioning, disposal is available at existing facilities located outside the local region. The MSRR anticipates some long-term and temporary storage of solid radioactive waste until appropriate disposal arrangements are implemented based on licensing status of the material. Fuel salt will be returned to DOE during decommissioning. Given the volumes of low-level radioactive waste (LLRW) received at radioactive waste disposal facilities from industries such as the nuclear power industry (93 operating commercial reactors in 2021), the medical industry, and research and development, the operation and future decommissioning of a single non-power reactor does not contribute significantly to LLRW management and disposal resources. Likewise, each of the existing facilities that generate LLRW within 5 mi (8-km) of the site have only a small effect on the nationwide LLRW management and disposal infrastructure. Therefore, the cumulative impact of the proposed project on waste management resources is SMALL.

19.4.14.10 Transportation

Section 19.4.11 addresses the direct and indirect contributory effects from the construction, operations, and decommissioning of the proposed MSRR when added to the effects from other past, present, and reasonably foreseeable future actions on transportation infrastructure. The geographic area of analysis for evaluation of cumulative impacts on transportation is primarily the same as that used in Section 19.2.1 and Section 19.3.1 and includes the 15.17 acres (6.14 ha) within the site boundary and the 5-mi (8-km) region surrounding the proposed MSRR. The roads and routes that are used for shipments or disposal of wastes are also considered. Transportation infrastructure includes roadways, rail lines, airports, and traffic control devices. As discussed in Section 19.4.8, the traffic impacts are SMALL during construction and decommissioning and SMALL during operations.

Other, non-related construction projects in the area produce an increase in vehicle traffic on roads within the 5-mi (8-km) radius of the proposed MSRR. Most existing roads are sufficient to handle the project transportation activities, and alternative routes can be used to minimize transportation impacts. In some cases, however, a nominal increase in traffic can occur, especially if MSRR construction workers and vehicles use the same roads during peak traffic periods. However, cumulative impacts to transportation are temporary and SMALL.

19.4.14.11 Environmental Justice

The geographic area of analysis for evaluation of cumulative effects on environmental justice includes the City of Abilene and the area surrounding the proposed site. As described in Section 19.4.13, no disproportionate impacts on low-income or minority populations from construction, operation or decommissioning of the proposed MSRR or other actions are expected.

Table 19.4-7 identifies recent past, present, and reasonably foreseeable future actions within the geographic extent of analysis that can be assessed to determine cumulative effects on environmental justice. No present or on-going actions are identified that significantly impact or are impacted by this proposed project. Thus, the cumulative impacts on environmental justice are SMALL.

19.4.14.12 Conclusion

Table 19.4-8 summarizes the cumulative impacts in all resource areas. In conclusion, there are no significant cumulative adverse environmental impacts from the construction, operation, and decommissioning of the site when considered together with other past, present, and reasonably foreseeable future projects in the area.

19.4.14.13 References

19.4.14-1 Council on Environmental Quality. Environmental Justice: Guidance Under the National Environmental Policy Act. December 10, 1997.

19.4.14-2 SHINE Medical Technologies. “Preliminary Safety Analysis Report.” Section 19.4, Impacts of Proposed Construction, Operations, and Decommissioning. October 13. 2015.

19.4.14-3 U.S. Environmental Protection Agency. “Green Book National Area and County-Level Multi-Pollutant Information.” www3.epa.gov/Green Book/National Area and County-Level Multi-Pollutant Information, accessed July 12, 2022.

19.4.14-4 “Your Abilene Investments.” www.abilenetx.gov/Communications/Your Abilene Investments/Projects, accessed July 14, 2022.

19.4.14-5 City of Abilene Texas Water Utilities. www.abilenetx.gov/Departments/ Water Utilities/Water Treatment, accessed July 18, 2022.

19.4.14-6 City of Abilene Texas Water Utilities. www.abilenetx.gov/Departments/ Water Utilities/Wastewater Treatment, accessed July 18, 2022.

19.4.14-7 Texas Department of Transportation. “Inside TxDOT.” www.txdot.gov/ Texas Department of Transportation/Inside TxDOT/Projects, accessed July 19,2022.

19.4.14-8 Texas Commission on Environmental Quality. Site-level summary of emissions data for 2014-2020. www.tceq.texas.gov/Air Quality/Point Source Emissions Inventory/2014-2020statesum, accessed July 18, 2020.

19.4.14-9 Hendrick Medical Center Radiology & Diagnostic Imaging. www.hendrickhealth.org/Services, accessed July 19, 2022.

19.4.14-10 Hendrick Medical Center. www.hendrickhealth.org/locations/Hendrick Medical Center, accessed July 19, 2022

19.4.14-11 Hendrick Medical Center South. www.hendrickhealth.org/locations/ Hendrick Medical Center South, accessed July 19, 2022.

19.4.14-12 Higher Ground: The Campaign for Abilene Christian University. www.higherground.acu.edu, accessed July 27, 2022.

Table 19.4-7 Past, Present, and Reasonably Foreseeable Projects and Other Actions Considered in the Cumulative Effects Analysis

Project Name	Summary of Project	Location (from MSRR)	Status	Potentially Affected Resource(s)	Retained for Cumulative Effects Analysis	Basis
Abilene Heritage Square	Rehabilitation of former middle school and high school to house a new city library, event center, restaurant and more	Approximately 2 miles southwest	In final design phase; construction start date not specified	Transportation, water resources, air quality	Y	Potential ongoing construction
Cedar Creek Waterway	Development of a greenbelt from Kirby Lake to Lake Ft. Phantom Hill into a series of park spaces by hiking, biking and jogging trails	Varying distances up to 6 miles southwest and 6 miles north; the closest point is 1500 feet west	Bond package approved; land and easement acquisition in progress; completion date not specified	Land use, water resources, air quality, human health, ecological resources	Y	Close proximity and potential ongoing construction activities
Abilene Downtown Convention Center Hotel	Construction of a full-service convention center hotel downtown	Approximately 2 miles southwest	Projected completion is Spring 2023	Land use, water resources	Y	Potential ongoing construction
TXDOT projects	Road maintenance and projects including, but not limited to, overpass installation, Buffalo Gap Road study, and I-20 corridor study	Various	In planning or progress	Transportation, air quality, noise, socioeconomics	Y	Close proximity and potential overlapping construction
City of Abilene	Road maintenance projects including, but not limited to, street re-surfacing and re-sealing	Various	In planning or progress	Transportation, air quality, noise, socioeconomics	Y	Close proximity and potential overlapping construction

Table 19.4-7 Past, Present, and Reasonably Foreseeable Projects and Other Actions Considered in the Cumulative Effects Analysis (Continued)

Project Name	Summary of Project	Location (from MSRR)	Status	Potentially Affected Resource(s)	Retained for Cumulative Effects Analysis	Basis
Abilene Landfill TX LP	Waste/Refuse management	Approximately 5 miles northwest in Jones County	Operational	Air quality, water resources	Y	Operational
Bisdgestone Bandag, LLC	Tires and inner tubes	Approximately 6 miles southeast	Operational	Air quality, water resources	Y	Operational
Tige Boats, Inc.	Boat building and repairing	Approximately 4 miles southeast	Operational	Air quality, water resources	Y	Operational
Petrosmith Equipment	Fabricated plate work (boiler shops)	Approximately 13 miles southwest	Operational	Regional air quality	N	Operational
Broadwind Heavy Fabrications, Inc.	Fabricated structural metal	Approximately 10 miles southwest	Operational	Regional air quality	N	Operational
Hendrick Medical Center	Existing hospital providing medical services imaging, radiation therapy, and nuclear medicine	Approximately 2 miles southwest	Operational	Human health, water resources, waste management	Y	Operational
Hendrick Medical Center South	Existing hospital providing medical services imaging, radiation therapy, and nuclear medicine	Approximately 10 miles southwest	Operational	Human health, water resources, waste management	N	Operational
Abilene Regional Airport	Public airport	Approximately 4 miles southeast	Operational	Land use, water resources, regional air quality, noise	Y	Operational

Table 19.4-7 Past, Present, and Reasonably Foreseeable Projects and Other Actions Considered in the Cumulative Effects Analysis (Continued)

Project Name	Summary of Project	Location (from MSRR)	Status	Potentially Affected Resource(s)	Retained for Cumulative Effects Analysis	Basis
Dyess Air Force Base	United States Air Force base	Approximately 12 miles southwest	Operational	Land use, water resources, regional air quality, noise	N	Operational
Higher Ground	Abilene Christian University campus infrastructure improvement plan including various building demolition, renovation and construction	Within 1 mile at various locations on the ACU campus	In planning or progress	Transportation, air quality, noise, water resources	Y	Close proximity and potential overlapping construction

Table 19.4-8 Cumulative Impacts on Environmental Resources, Including the Impacts of the Proposed Project

RESOURCE CATEGORY	CUMULATIVE IMPACT LEVEL
Land use and visual resources	
Land use	Small
Visual resources	Small
Air quality and noise	
Air quality	Small
Noise	Small
Geologic environment	Small
Water resources	
Hydrology	Small
Water use	Small
Water quality	Small
Ecological resources	
Terrestrial ecosystems	Small
Aquatic ecosystems	Small
Historic and cultural resources	Small
Socioeconomics	Small
Human health	
Nonradiological health	Small
Radiological health	Small
Waste management	Small
Transportation	Small
Environmental justice	Small

19.5 Alternatives

19.5.1 No-Action Alternative

The proposed federal action is issuance of a Construction Permit for a non-power, MSRR facility to accelerate the development and deployment of molten salt reactor systems through foundational research while also developing a new pipeline to the nuclear workforce. The intended use of the MSRR is to conduct research on molten salt systems, as well as to educate and train a new generation of engineers and scientists who are prepared to contribute to the advancement and deployment of molten salt reactors and applications. The research will generate experimental molten salt reactor data to advance the understanding of the generation and migration of gases and vapors in a fluid-fueled fluoride reactor and the behavior of delayed neutron precursors during normal and off-normal operating conditions, all of which can be used in the validation and calibration of software for the design, licensing, and regulation of commercial molten salt reactors.

Under the No-Action Alternative, the NRC would not issue the Construction Permit, and there would be no subsequent construction, operation, or decommissioning. Consistent with the guidance in the Final Interim Staff Guidance (ISG) Augmenting NUREG-1537, Chapter 19, the environmental consequences of the No-Action Alternative are assumed to be the status quo.

If the MSRR is not constructed, operated, and decommissioned the adverse environmental consequences discussed in Section 19.4 are avoided; however, as discussed in Section 19.4, the adverse impacts of construction, operation, and decommissioning of the research reactor are concluded to be SMALL. Therefore, the benefit of avoiding those impacts is not significant. Construction and operation of the MSRR provides socioeconomic benefits as described in Section 19.4.8, including increases in tax revenues to local jurisdictions, which are not realized if the reactor is not constructed and operated.

As described in Section 19.1.1, construction, operation, and decommissioning of the MSRR provides a means to conduct research on molten salt systems, as well as to educate and train a new generation of engineers and scientists who are prepared to contribute to the advancement and deployment of molten salt reactors and applications. These programmatic benefits would not be realized under the No-Action Alternative. The programmatic benefits support deployment of advanced nuclear technologies that result in less reliance on carbon fuel-based forms of energy production.

19.5.2 Reasonable Alternatives

This section describes how the site is developed and potential alternatives to the proposed project, based on the guidance in Section 19.5 of the Final ISG Augmenting NUREG-1537, Part 1 and Section 19.5 of the Final ISG Augmenting NUREG-1537, Part 2. Based on the guidance in these ISG documents, this section:

- describes the process used to develop, identify, and evaluate reasonable alternatives,
- describes reasonable alternatives considered.
- identifies the alternatives that were eliminated from further evaluation.
- considers whether alternatives may avoid or reduce adverse effects.

According to the two ISG documents, reasonable alternatives may include, but are not limited to, alternative sites, alternative siting within a proposed site, modification of existing facilities, alternative technologies, and/or alternative transportation methods. The proposed project involves the demonstration and testing of new technology. Therefore, alternative technologies are not considered, and modification of a different existing facility to house the proposed project is not feasible. With respect to alternative transportation methods, the proposed location of the site on a specific site within the SERC limits transportation options to vehicle transport using the existing road network. Alternative routes, types of vehicles, carpooling of workers, or other transportation-related features may be considered as mitigation measures, but do not merit full analysis as alternatives. Similarly, the proposed project has been specifically sited within the SERC on the ACU campus to minimize potential impacts.

Because none of these potential types of alternatives are feasible or reduce or avoid adverse effects, this subsection is limited to identification and analysis of potential alternative locations. This approach meets the guidance of the ISG documents, which state that, if new construction is proposed, then the alternatives should include at least one alternative location. Similarly, the level of analysis of the alternative sites in this subsection is commensurate with the context, degree, and intensity of the potential impacts, as indicated in the ISG documents.

19.5.2.1 Alternate Sites

ACU is submitting a Construction Permit application for the MSRR at the SERC on the campus of ACU in Abilene, Texas. Part of the process for developing the application is selection of a site that provides the geographic setting for the facility, as well as identification of potential alternative sites that may be considered if they meet ACU objectives and could result in a reduction in environmental impacts. This section provides a description of the bases, assumptions, and processes applied to the identification of the candidate sites for detailed analysis, and the selection of the proposed site and potential alternative sites.

The process used by ACU for identifying and evaluating alternate research reactor sites included defining a region of interest, identifying potential sites within candidate sites, evaluating and scoring candidate sites. There were four categorical factors important to ACU in the site evaluation process:

<u>Environmental Impact</u>

1. Site preparation
2. Facility construction
3. Reactor construction
4. Reactor operations
5. Reactor decommissioning
6. Facility decontamination and remediation

<u>Financial Impact</u>

1. Site preparation
2. Facility construction
3. Reactor construction
4. Reactor operations
5. Reactor decommissioning
6. Facility decontamination and remediation

<u>Mission Impact</u>

Timeline to criticality

1. Site preparation

2. Facility construction
3. Reactor construction

Proximity

1. Education
2. Security
3. Emergency response

The following elements define the region of interest: candidate sites are in the North Abilene area; potential sites that are owned by ACU; candidate sites that have sufficient space around the proposed facility and had existing utilities.

Three candidate sites for the MSRR are identified and subsequently evaluated using the above factors. The three candidate sites are:

- Rhoden Farm Site
- Sherrod Site
- Science and Engineering Research Center

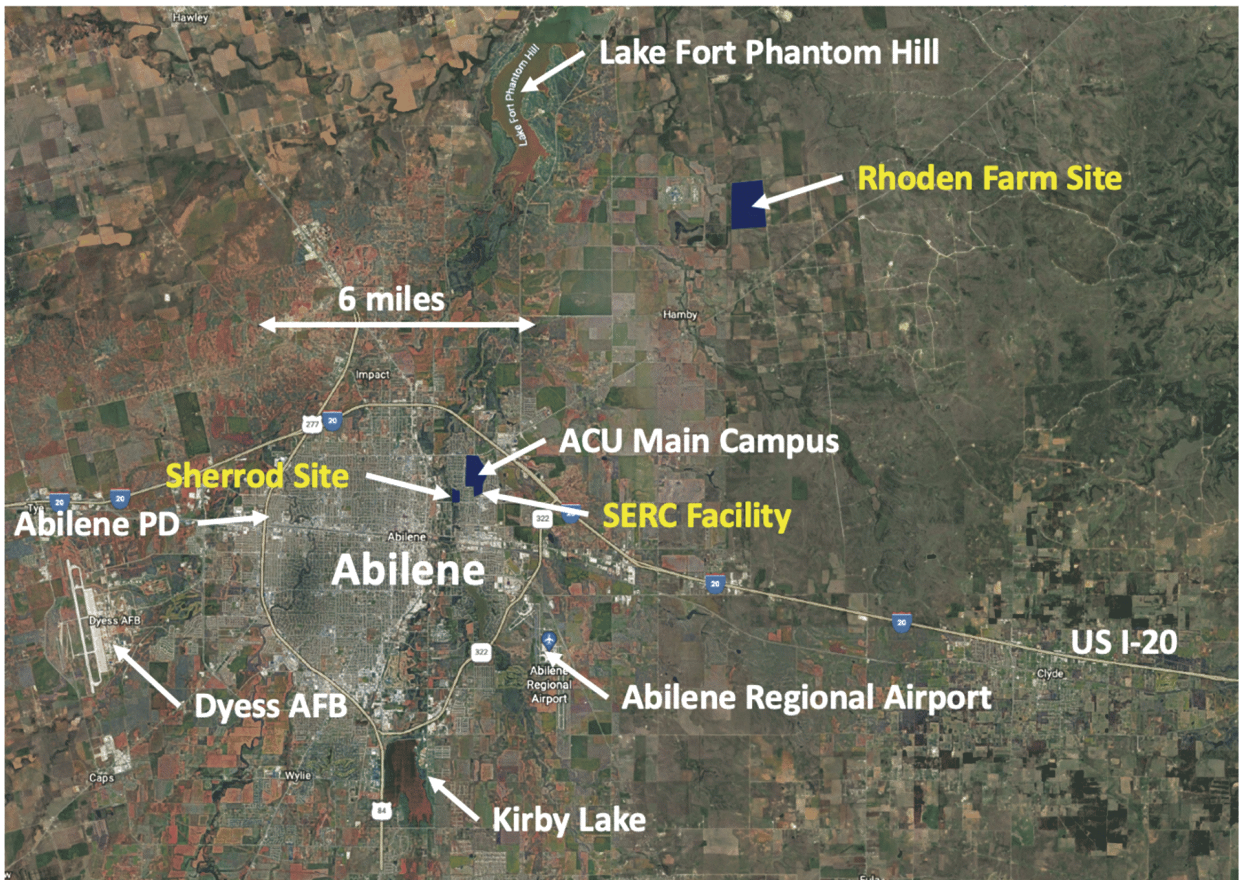
The Rhoden Farm site is an ACU facility located approximately 10 mi (16 km) north of the main ACU campus. It is an active research farm of approximately 300 acres (121 ha). Rhoden Farm is used by the ACU Agriculture and Environmental Sciences Department for sustainable agriculture and environmental systems.

The Sherrod site is an area less than one mile (1.6 km) southwest of the main ACU campus. It is presently a disc golf park and contains student housing units that are no longer in use (“Sherrod Apartments”). The housing units need to be removed if the site is used for the MSRR.

The SERC is located on the ACU main campus close to the Engineering, Chemistry, and Physics Departments. It is an existing facility designed to house a variety of radiation producing devices.

The candidate sites are shown in Figure 19.5-1.

Figure 19.5-1 Sites Considered for the MSRR



19.5.3 Cost-Benefit of the Alternatives

The objectives of the siting process are to identify a site that meets ACU business objectives for the project, satisfy applicable NRC site suitability requirements, and are compliant with the NRC implementation guidance for the NEPA analysis requirements for research reactors with respect to the consideration of alternative sites. This section describes how the potential sites for the MSRR are evaluated and the bases for selecting the SERC site.

The three candidate MSRR sites identified above are evaluated using the criteria defined in Section 19.5.2.1. The objective of the qualitative evaluation is to determine the preferred site for the MSRR based on the three categorical factors important to ACU. Rank scoring (1st, 2nd, and 3rd) is applied to the considerations in each of the three categories. Results of the site ranking process are shown in Table 19.5-1, Table 19.5-2, and Table 19.5-3.

Table 19.5-1 Candidate Site Ranking for Environmental Factors

Environmental Impact	Rhoden Farm Site	Sherrod Site	SERC
1) Site preparation	3rd	2nd	1st
2) Facility construction	3rd	2nd	1st
3) Reactor construction	1st	1st	1st
4) Reactor operations	1st	1st	1st
5) Reactor decommissioning	1st	1st	1st
6) Facility decon and remediation	2nd	2nd	1st

Table 19.5-2 Candidate Site Ranking for Financial Impact

Financial Impact	Rhoden Farm Site	Sherrod Site	SERC
1) Site preparation	2nd	3rd	1st
2) Facility construction	3rd	2nd	1st
3) Reactor construction	2nd	1st	1st
4) Reactor operations	2nd	1st	1st
5) Reactor decommissioning	2nd	1st	1st
6) Facility decon and remediation	3rd	2nd	1st

Table 19.5-3 Candidate Site Ranking for Mission Impact

Mission Impact	Rhoden Farm Site	Sherrod Site	SERC
A) Timeline to Criticality			
1) Site preparation	2nd	2nd	1st
2) Facility construction	2nd	2nd	1st
3) Reactor construction	2nd	1st	1st
B) Proximity			
1) Education	3rd	2nd	1st
2) Security	3rd	2nd	1st
3) Emergency Response	2nd	1st	1st

19.5.4 Comparison of the Potential Environmental Impacts

The environmental impact for all three candidate sites is approximately the same with regard to reactor construction, reactor operations, and decommissioning activities because of similar installation or “drop-in” of the reactor. The primary difference between the sites from an environmental perspective is that construction of a new facility needs to be included in the project and the associated licensing review at the Rhoden farm and Sherrod sites.

From a financial impact perspective (Table 19.5-2), there are additional costs for reactor construction, reactor operations, and decommissioning at the Rhoden Farms site because of the more remote location. The financial impact rankings are primarily driven by costs associated with construction of a new purpose-built facility at the Rhoden Farm or Sherrod sites.

The proposed SERC facility is ranked first from a mission impact perspective. Construction of a new purpose-built facility as part of the Construction Permit application results in a substantial delay and negatively impacts the goal to minimize the time to criticality. Further, proximity to the central ACU campus supports deeper and broader integration with students and departments. Also, security posture and emergency response times increase with distance from campus and first responders.

19.6 Conclusions

19.6.1 Unavoidable Adverse Environmental Impacts

This section describes principal unavoidable adverse environmental impacts for which mitigation measures either do not exist or cannot entirely eliminate the impact.

19.6.1.1 Unavoidable Adverse Environmental Impacts of Construction

Because construction of the MSRR involves installation of components and equipment within an existing structure, the environmental impacts of construction are minimal. All of the impacts assessed in Section 19.4 are SMALL because they either are not detectable or are minor compared to the availability of the affected resources. Land use is not affected as no new structures are required. No construction related activities result in disproportionately high and adverse environmental or health effects on minority or low-income populations.

Unavoidable adverse impacts from construction of the facility include:

- There are temporarily impacted land areas used for employee parking and equipment staging. Permanent university employee parking likely suffices, but if additional short-term construction employee parking is used, the area needs to be restored with either native plants or landscaping.
- Unavoidable water use impacts are minimal because of the abundant water available in the City of Abilene municipal system.
- Impacts to social services and traffic patterns due to a potential influx of population are minor.
- There are potential minor impacts to the capacity of waste disposal facilities managing construction waste.
- Activities associated with the use of construction and staging equipment and construction workforce traveling to and from the project site can result in varying, minor amounts of dust, air emissions, noise, and vibration.

19.6.1.2 Unavoidable Adverse Environmental Impacts of Operations

Impacts due to the operation of the facility are discussed in detail in Section 19.4. Operations-related impacts are SMALL because they either are not detectable or are minor compared to the availability or status of the affected resource. Operation of the MSRR has no negative effect on land use, ecological resources, or historic and cultural resources. No operations or decommissioning-related activities result in disproportionately high and adverse environmental or health effects on minority or low-income populations.

Unavoidable adverse impacts from operation of the facility include:

- Air quality and noise potentially can be impacted by air emissions from the facility and increased vehicle traffic from the MSRR workforce.
- There is minor impact to water supply and sanitary treatment systems. The City of Abilene has ample capacity to provide the balance of water required for the facility. The City of Abilene public works has ample capacity to treat the volume of wastewater required for operations.
- There are small impacts to social services because of the small increase in the ACU operations staff. There is a small, positive, socioeconomic impact from increased tax revenue and local expenditures by the operations staff and from visiting faculty members using the MSRR.
- There are potential impacts to the general public and operations workforce from radiation sources and airborne radioactive effluents. Additionally, there are potential impacts because of exposure to physical, electrical, and chemical hazards at the facility.
- There are potential impacts to the capacity of waste disposal facilities managing low-level radioactive waste.

19.6.2 Relationship between Short-Term Uses and Long-Term Productivity of the Environment

This Environmental Review focuses on the analyses and resulting conclusions associated with the environmental impacts from activities during the new plant construction, operation, and decommissioning at the site. These activities are considered short-term uses for purposes of this section. In this section, the long-term is considered to be initiated with the conclusion of proposed facility decommissioning at the site. This section includes an evaluation of the extent that the short-term uses preclude any options for future long-term use of the site.

Section 19.6.1.1 summarizes the potential unavoidable adverse environmental impacts of construction. Some small adverse environmental impacts can remain after all practical measures to avoid or mitigate them are taken; however, none of these impacts represent long-term effects that preclude any options for future use of the site and the SERC.

Similarly, Section 19.6.1.2 summarizes the unavoidable adverse environmental impacts of facility operations. Some small adverse environmental impacts can remain after all practical measures to avoid or mitigate them are taken; however, none of these impacts represent long-term effects that preclude any options for future use of the SERC.

During operation, exposures to and management of radioactive and hazardous materials and waste are unavoidable; however, adherence to exposure and dose limits established by regulatory agencies and implementation of a robust facility health and safety program limits impacts to workers and the public. Exposures to radioactive materials during transportation is also unavoidable; however, the dose impacts to the exposed populations are only a small fraction of background dose.

Operation of the facility has a comparable impact on all populations in the region around the site. No disproportionate impacts are expected to either minority or low-income populations as such populations are not identified within the region around the site. Therefore, there are no long-term effects to environmental justice that preclude any options for future use of the site.

In summary, the impacts resulting from the facility construction, operation, and decommissioning result in both adverse and beneficial short-term impacts. The principal short-term adverse impacts are SMALL residual impacts to local traffic, noise, and air quality. There are no long-term impacts to the environment. The principal short-term benefits are the creation of additional jobs, additional tax revenues, and knowledge gains from the research conducted at the MSRR. The principal long-term benefits are the knowledge and skills acquired by a new generation of scientists and engineers from work at the MSRR, advancements in understanding molten salt reactor technology, and potential benefits from increased tax revenues after facility decommissioning. The short-term impacts and benefits and long-term benefits do not affect long-term productive use of the site and the SERC.

19.6.3 Irreversible and Irretrievable Commitments of Resources

This section describes the anticipated irreversible and irretrievable commitments of environmental resources used in the construction, operation, and decommissioning of the facility. The term “irreversible commitments of resources” describes environmental resources that potentially are changed by the new facility construction or operation and that cannot be restored at some later time to the resource’s state prior to construction or operation. Irretrievable commitments of resources are materials used for the proposed facility in such a way that they cannot, by practical means, be recycled or restored for other uses.

Irreversible environmental resource commitments resulting from the new facility include

- degradation of air and water resources.
- land disposal of wastes, including hazardous and low-level radioactive waste.

The proposed facility requires water from the City of Abilene for construction, potable water, fire protection, and facility heating and cooling. The City of Abilene provides water supply for both public drinking and fire protection through treatment of surface water. Because there is sufficient capacity within the Abilene water supply system, there are no indirect effects associated with the demand from the facility. There are no direct impacts to water quality or hydrology from the facility; therefore, there are no irreversible impacts.

Construction and decommissioning activities create dust and other emissions, such as vehicle exhaust. During operations, emissions are a product of vehicle exhaust, ventilation system exhaust, and fuel combustion, resulting in very low levels of gaseous pollutants and particulates released from the facility into the air. Emissions during operations are in compliance with applicable federal and state regulations, minimizing their impact on public health and the environment. No irreversible impacts to air quality are anticipated.

Irretrievable commitments of resources during new plant construction are similar to that of any small-scale industrial facility construction project. Materials consumed during construction and operation are described in Section 19.2.5.2. These materials are irretrievable unless they are recycled at decommissioning. Use of construction materials in the quantities associated with the facility have a SMALL impact with respect to the commitment of such resources.

During operations, the main resources that are irretrievably committed are the nuclear fuel and the fluoride salts used as reactor coolant. The spent nuclear fuel cannot be recycled, and the coolant salts are disposed as low-level radioactive waste. Materials used in the construction of the reactor, spent fuel canisters, and other waste containers and metals and concrete activated as result of reactor operations also are irretrievable and disposed as radioactive waste.

While a given quantity of material consumed during facility construction, operation, and decommissioning at the site is irretrievable, the impact on material availability is SMALL as the quantity of required materials, mostly stainless steel, is relatively small. Furthermore, the reasonably stable supply of minerals required for MSRR materials suggests that these minerals continue to be available for the foreseeable future in response to demand.