**ENCLOSURE 2**

**Preliminary Safety Analysis Report**

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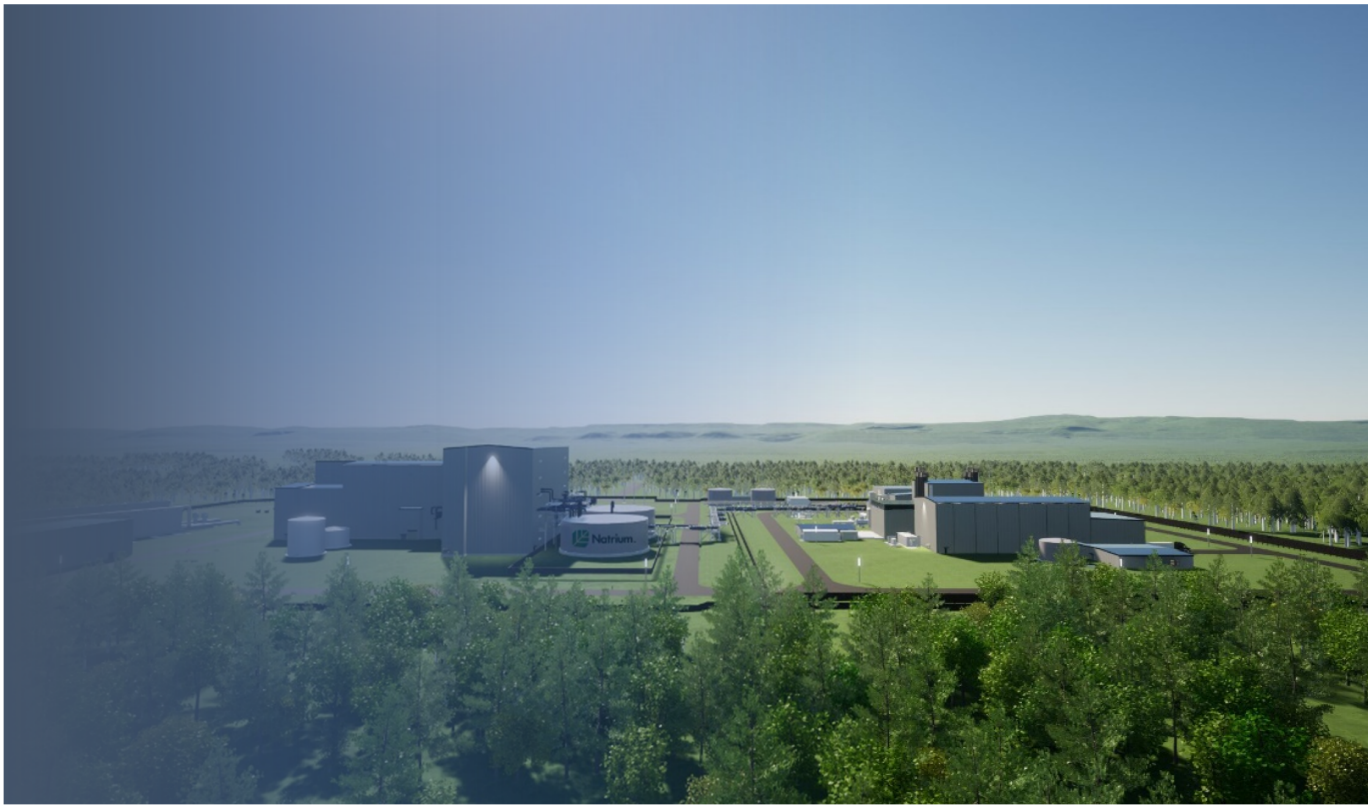

# Kemmerer Power Station Unit 1
# Preliminary Safety Analysis Report

US SFR Owner, LLC, a wholly owned subsidiary of TerraPower, LLC

NAT-9400, Preliminary Safety Analysis Report, Revision 0
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## TABLE OF CONTENTS

**Chapter 1 General Plant and Site Description . . . . . 1.1-1**

1.1 Introduction. . . . . 1.1-1

1.1.1 Reactor Supplier and Model. . . . . 1.1-1

1.1.2 Intended Use of the Reactor . . . . . 1.1-1

1.1.3 Overall Configuration . . . . . 1.1-1

1.1.4 Description of Plant Structures, Systems, and Components . . . . . 1.1-2

1.2 Site Description . . . . . 1.2-1

1.3 Plant Safety Overview . . . . . 1.3-1

1.3.1 Licensing Basis Methodology. . . . . 1.3-1

1.3.2 Fundamental Safety Functions. . . . . 1.3-1

1.3.3 Defense-in-Depth . . . . . 1.3-5

1.4 Conformance with Regulatory Criteria and Referenced Material. . . . . 1.4-1

1.4.1 Regulatory Guidance . . . . . 1.4-1

1.4.2 Topical Reports and Technical Reports . . . . . 1.4-1

1.4.3 Generic Safety Issues, Unresolved Safety Issues and TMI Action Items. . . . . 1.4-2

1.4.4 Codes and Standards. . . . . 1.4-2

**Chapter 2 Site Information . . . . . 2.1-1**

2.1 Site Characteristics and Site Parameters. . . . . 2.1-1

2.2 Geography and Demography. . . . . 2.2-1

2.2.1 Site Location and Description. . . . . 2.2-1

2.2.2 Exclusion Area Authority and Control. . . . . 2.2-1

2.2.3 Population Distribution . . . . . 2.2-2

2.3 Nearby Industrial, Transportation, and Military Facilities . . . . . 2.3-1

2.3.1 Location and Routes . . . . . 2.3-1


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TABLE OF CONTENTS

2.3.2 Descriptions . . . . . 2.3-3
2.3.3 Evaluation of Potential Hazards . . . . . 2.3-5
2.4 Regional Climatology, Local Meteorology, and Atmospheric Dispersion . . . . 2.4-1
2.4.1 Regional Climatology . . . . . 2.4-1
2.4.2 Local Meteorology and Topography . . . . . 2.4-10
2.4.3 Air Quality . . . . . 2.4-22
2.4.4 Atmospheric Dispersion . . . . . 2.4-22
2.4.5 Meteorological Monitoring . . . . . 2.4-24
2.5 Hydrologic Description . . . . . 2.5-1
2.5.1 Floods . . . . . 2.5-2
2.5.2 Flooding Protection . . . . . 2.5-36
2.5.3 Groundwater . . . . . 2.5-38
2.5.4 Low Water Considerations . . . . . 2.5-57
2.6 Geology, Seismology, and Geotechnical Engineering . . . . . 2.6-1
2.6.1 Geologic Hazards . . . . . 2.6-3
2.6.2 Vibratory Ground Motion . . . . . 2.6-22
2.6.3 Surface Deformation . . . . . 2.6-29
2.6.4 Stability of Subsurface Materials and Foundations . . . . . 2.6-35
2.6.5 Stability of Slopes . . . . . 2.6-60
2.7 Volcanic Hazards . . . . . 2.7-1
2.8 Summary of External Hazards . . . . . 2.8-1
**Chapter 3 Licensing Basis Events . . . . . 3.1-1**
3.1 Probabilistic Risk Assessment . . . . . 3.1-1
3.1.1 Overview of PRA . . . . . 3.1-1
3.1.2 Summary of PRA Results Essential to the Licensing Basis . . . . . 3.1-4


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## TABLE OF CONTENTS


3.2 Licensing Methodology for Mechanistic Source Term . . . . . 3.2-1

3.2.1 Mechanistic Source Term Summary Tables . . . . . 3.2-2

3.3 Licensing Methods for Evaluation of Licensing Basis Events . . . . . 3.3-1

3.3.1 Licensing Methodology for Design Basis Accidents . . . . . 3.3-1

3.3.2 Licensing Methodology for AOO, DBE, and BDBE . . . . . 3.3-2

3.3.3 Licensing Methodology for Major Accident . . . . . 3.3-3

3.3.4 LBE Plant Response and Analysis Overview . . . . . 3.3-3

3.4 Licensing Basis Event Selection Methodology . . . . . 3.4-1

3.5 LBE Summary . . . . . 3.5-1

3.5.1 Summary Evaluation of AOOs, DBEs, and BDBEs . . . . . 3.5-1

3.5.2 Summary Evaluation of DBAs . . . . . 3.5-1

3.6 Anticipated Operational Occurrences . . . . . 3.6-1

3.6.1 Loss of Primary Flow AOOs . . . . . 3.6-1

3.6.2 Increase or Decrease in Heat Removal AOOs . . . . . 3.6-2

3.6.3 Reactivity AOOs . . . . . 3.6-3

3.6.4 Fuel Handling AOOs . . . . . 3.6-3

3.7 Design Basis Events . . . . . 3.7-1

3.7.1 Loss of Primary Flow DBEs . . . . . 3.7-1

3.7.2 Increase or Decrease in Heat Removal DBEs . . . . . 3.7-2

3.7.3 Release from Ex-Vessel Systems DBEs . . . . . 3.7-5

3.7.4 Fuel Handling DBEs . . . . . 3.7-10

3.7.5 Local Fuel Fault DBEs . . . . . 3.7-15

3.8 Beyond Design Basis Events . . . . . 3.8-1

3.8.1 Loss of Primary Flow BDBEs . . . . . 3.8-1

3.8.2 Increase or Decrease in Heat Removal BDBEs . . . . . 3.8-5



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## TABLE OF CONTENTS


3.8.3 Reactivity BDBEs . . . . . 3.8-8

3.8.4 Release from Ex-Vessel Systems BDBEs . . . . . 3.8-11

3.8.5 Fuel Handling BDBEs . . . . . 3.8-12

3.8.6 Local Fuel Fault BDBEs . . . . . 3.8-23

3.8.7 Other BDBEs . . . . . 3.8-25

3.9 Design Basis Accidents . . . . . 3.9-1

3.9.1 Loss of Primary Flow DBAs . . . . . 3.9-1

3.9.2 Increase or Decrease in Heat Removal DBAs . . . . . 3.9-2

3.9.3 Reactivity DBAs . . . . . 3.9-4

3.9.4 Release from Ex-Vessel Systems DBAs . . . . . 3.9-4

3.9.5 Fuel Handling DBAs . . . . . 3.9-7

3.9.6 Local Fuel Fault DBAs . . . . . 3.9-10

3.10 10 CFR 50.155 Mitigation Strategies . . . . . 3.10-1

3.11 Fuel System Design . . . . . 3.11-1

3.12 Nuclear Design . . . . . 3.12-1

3.12.1 Nuclear Design Summary . . . . . 3.12-1

3.12.2 Design Basis . . . . . 3.12-3

3.12.3 Codes Used in System Evaluation . . . . . 3.12-4

3.13 Thermal and Hydraulic Design . . . . . 3.13-1

3.13.1 Description . . . . . 3.13-1

3.13.2 Design Basis . . . . . 3.13-1

3.13.3 Codes Used in System Evaluation . . . . . 3.13-2

3.13.4 Testing . . . . . 3.13-2

3.14 Criticality Safety of Fresh and Spent Fuel . . . . . 3.14-1

3.14.1 Design Basis . . . . . 3.14-1



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TABLE OF CONTENTS

3.14.2 Methodology . . . . . 3.14-1

**Chapter 4 Integrated Evaluations . . . . . 4.1-1**

4.1 Overall Plant Risk Performance Summary . . . . . 4.1-1

4.1.1 Site Boundary Dose . . . . . 4.1-2

4.1.2 Exclusion Area Boundary Early Fatality Risk . . . . . 4.1-2

4.1.3 Latent Cancer Risk . . . . . 4.1-2

4.2 Defense-in-Depth . . . . . 4.2-1

4.2.1 Plant Capability Summary . . . . . 4.2-1

4.2.2 Programmatic DID Summary . . . . . 4.2-3

4.2.3 Integrated DID Evaluation . . . . . 4.2-4

**Chapter 5 Safety Functions, Design Criteria, and SSC Safety Classification . . . . . 5.1-1**

5.1 Safety Classification of SSCs . . . . . 5.1-1

5.2 Safety-Significant PRA Safety Functions . . . . . 5.2-1

5.2.1 SR Functions . . . . . 5.2-1

5.2.2 NSRST Functions . . . . . 5.2-1

5.2.3 NST Functions . . . . . 5.2-1

5.3 Principal Design Criteria . . . . . 5.3-1

5.3.1 Overall Requirements . . . . . 5.3-1

5.3.2 Multiple Barriers . . . . . 5.3-4

5.3.3 Reactivity Control . . . . . 5.3-10

5.3.4 Fluid and Heat Transport Systems . . . . . 5.3-16

5.3.5 Fuel and Reactivity Control . . . . . 5.3-21

5.3.6 Additional PDC . . . . . 5.3-24

5.4 Safety-Related Structures, Systems, and Components . . . . . 5.4-1


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TABLE OF CONTENTS

5.5 Non-Safety-Related with Special Treatments Structures, Systems, and Components . . . . . 5.5-1

**Chapter 6 Safety-Significant SSC Criteria and Capabilities . . . . . 6.1-1**

6.1 Design Requirements for Safety-Related SSCs. . . . . 6.1-1

6.1.1 Design Basis Hazard Levels . . . . . 6.1-1

6.1.2 Summary of SRDC. . . . . 6.1-4

6.1.3 Summary of DBHL-Related Requirements for Non-Safety-Related SSCs . . . . . 6.1-5

6.2 Reliability and Capability Targets for Safety-Significant SSCs. . . . . 6.2-1

6.3 Special Treatment Requirements for Safety-Significant SSCs . . . . . 6.3-1

6.4 Design of Safety-Significant SSCs. . . . . 6.4-1

6.4.1 Seismic Design. . . . . 6.4-1

6.4.2 Design of Safety-Significant Structures . . . . . 6.4-10

6.4.3 Mechanical Systems and Components . . . . . 6.4-24

6.4.4 ASME BPVC, Section III Piping Systems, Piping Components, and Associated Supports . . . . . 6.4-33

**Chapter 7 Descriptions for Safety-Significant SSCs . . . . . 7.1-1**

7.1 Nuclear Heat Supply Systems . . . . . 7.1-1

7.1.1 Reactor Core System. . . . . 7.1-1

7.1.2 Reactor Enclosure System. . . . . 7.1-14

7.1.3 Primary Heat Transport System. . . . . 7.1-47

7.1.4 Intermediate Heat Transport System . . . . . 7.1-72

7.2 Reactor Auxiliary Systems . . . . . 7.2-1

7.2.1 Reactor Air Cooling System . . . . . 7.2-1

7.2.2 Intermediate Air Cooling System . . . . . 7.2-13

7.2.3 Sodium Cover Gas System . . . . . 7.2-22

7.2.4 Sodium Processing System . . . . . 7.2-41


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TABLE OF CONTENTS

7.2.5 Control Rod Drive System . . . . . 7.2-57
7.3 Fuel Handling Systems . . . . . 7.3-1
7.3.1 Water Pool Fuel Handling System . . . . . 7.3-1
7.3.2 Ex-Vessel Fuel Handling System . . . . . 7.3-16
7.3.3 In-Vessel Fuel Handling System . . . . . 7.3-38
7.4 Radwaste Systems . . . . . 7.4-1
7.4.1 Gaseous Radwaste Processing System . . . . . 7.4-1
7.5 Ancillary Systems . . . . . 7.5-1
7.5.1 NI HVAC System . . . . . 7.5-1
7.5.2 Nuclear Island Fire Protection System . . . . . 7.5-10
7.6 Integrated Controls Systems . . . . . 7.6-1
7.6.1 Instrumentation and Control Systems Introduction . . . . . 7.6-1
7.6.2 Nuclear Island Control System . . . . . 7.6-6
7.6.3 Reactor Protection System . . . . . 7.6-18
7.6.4 Nuclear Instrumentation System . . . . . 7.6-36
7.6.5 Reactor Instrumentation System . . . . . 7.6-48
7.6.6 Radiation Monitoring System . . . . . 7.6-60
7.6.7 Control Room and Indications . . . . . 7.6-69
7.6.8 Anticipatory Automatic Seismic Trip System . . . . . 7.6-75
7.7 Electrical Systems . . . . . 7.7-1
7.7.1 Nuclear Island Auxiliary Electrical Systems . . . . . 7.7-1
7.8 Buildings and Structures . . . . . 7.8-1
7.8.1 Reactor Building . . . . . 7.8-1
7.8.2 Fuel Handling Building . . . . . 7.8-8
7.8.3 Reactor Auxiliary Building . . . . . 7.8-18
7.8.4 Nuclear Island Control Building . . . . . 7.8-27


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## TABLE OF CONTENTS

**Chapter 8 Plant Programs . . . . . 8.0-1**

8.0 Plant Programs . . . . . 8.0-1

8.1 Quality Assurance . . . . . 8.1-1

8.1.1 Disposition of Limitations on NRC Approval of TerraPower QATR . . . . . 8.1-1

8.1.2 Responsibilities During Design and Construction . . . . . 8.1-3

8.1.3 Organization and Controls . . . . . 8.1-4

**Chapter 9 Control of Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste . . . . . 9.1-1**

9.1 Liquid and Gaseous Effluents . . . . . 9.1-1

9.1.1 Source Terms . . . . . 9.1-1

9.1.2 Offsite Releases . . . . . 9.1-1

9.1.3 RWL Summary Description . . . . . 9.1-2

9.1.4 Design Basis . . . . . 9.1-5

9.2 Contamination Control . . . . . 9.2-1

9.3 Solid Radwaste Processing . . . . . 9.3-1

9.3.1 RWS Summary Description . . . . . 9.3-1

9.3.2 Design Basis . . . . . 9.3-2

**Chapter 10 Control of Occupational Dose . . . . . 10.1-1**

10.1 Radiation Protection Program . . . . . 10.1-1

10.1.1 Radiation Surveys and Monitoring . . . . . 10.1-2

10.1.2 Access Control and Dosimetry . . . . . 10.1-2

10.1.3 High and Very High Radiation Areas . . . . . 10.1-2

10.2 ALARA Program . . . . . 10.2-1

10.3 Design Considerations . . . . . 10.3-1

10.3.1 PDC Applicable to Control Occupational Radiation Exposures . . . . . 10.3-2


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TABLE OF CONTENTS

10.3.2 Major Radiation Sources . . . . . 10.3-2
10.4 Design Provisions. . . . . 10.4-1
10.4.1 Shielding. . . . . 10.4-1
10.4.2 Reduce Accumulation of Radioactive Material. . . . . 10.4-1
10.4.3 Remote Operations . . . . . 10.4-1
10.5 Preliminary Dose Assessment for Expected Occupancy . . . . . 10.5-1
10.5.1 Chief Nuclear Officer Organization. . . . . 10.5-1
10.5.2 Operations . . . . . 10.5-1
10.5.3 Maintenance. . . . . 10.5-1
10.5.4 Training . . . . . 10.5-2
10.5.5 Engineering . . . . . 10.5-2
10.5.6 Station Sciences. . . . . 10.5-2
10.5.7 Organizational Effectiveness . . . . . 10.5-2
10.5.8 Security . . . . . 10.5-2
10.5.9 Refueling . . . . . 10.5-2
**Chapter 11 Conduct of Operations . . . . . 11.1-1**
11.1 Organization. . . . . 11.1-1
11.1.1 Executive Management Organization- TerraPower . . . . . 11.1-2
11.1.2 Natrium Project Organization- TerraPower . . . . . 11.1-3
11.1.3 Construction Organization- BPC . . . . . 11.1-4
11.1.4 Operating Organization and Technical Support . . . . . 11.1-5
11.1.5 Qualification, Training, and Conduct of Operations . . . . . 11.1-8
11.2 Human Factors Engineering. . . . . 11.2-1
11.2.1 Management of the Human Factors Engineering Program . . . . . 11.2-1
11.2.2 Operating Experience Review . . . . . 11.2-14


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## TABLE OF CONTENTS


11.2.3 Safety Function Review . . . . . 11.2-18

11.2.4 Task Analysis . . . . . 11.2-19

11.2.5 Human-System Interface Design . . . . . 11.2-21

11.2.6 Procedures, Training, and Verification and Validation . . . . . 11.2-21

11.2.7 Human Performance Monitoring . . . . . 11.2-25

11.3 Emergency Preparedness . . . . . 11.3-1

11.3.1 Bases for Emergency Preparedness Approach . . . . . 11.3-1

11.3.2 Physical Site Characteristics . . . . . 11.3-3

11.3.3 Emergency Planning Zone . . . . . 11.3-3

11.3.4 Emergency Response Organization . . . . . 11.3-4

11.3.5 Federal, State and Local Governmental Agencies . . . . . 11.3-4

11.3.6 Protective Measures . . . . . 11.3-5

11.3.7 First Aid, Decontamination, Off-site Medical Treatment, and Emergency Medical Transportation . . . . . 11.3-6

11.3.8 Training . . . . . 11.3-6

11.3.9 Emergency Response Facilities and Equipment . . . . . 11.3-6

11.3.10 Performance Monitoring . . . . . 11.3-8

11.4 Aircraft Impact Assessment . . . . . 11.4-1

11.5 Technical Specifications . . . . . 11.5-1

11.5.1 Introduction . . . . . 11.5-1

11.5.2 Operating Modes . . . . . 11.5-1



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## TABLE OF CONTENTS


11.6 Fitness-for-Duty and Security . . . . . 11.6-1

11.6.1 Fitness-for-Duty . . . . . 11.6-1

11.6.2 Security . . . . . 11.6-1

**Chapter 12 Post-Construction Inspection, Testing, and Analysis Program . . . . . 12.1-1**

12.1 Program Summary and Objectives . . . . . 12.1-1

12.2 Program Administration . . . . . 12.2-1

12.3 Construction Testing (Stage 1) . . . . . 12.3-1

12.4 PITAP Phase 1: Preoperational Test Program . . . . . 12.4-1

12.4.1 Component Testing (Stage 2) . . . . . 12.4-1

12.4.2 System Testing (Stage 3) . . . . . 12.4-1

12.5 PITAP Phase 2: Initial Startup Testing (Stage 4) . . . . . 12.5-1

12.5.1 Initial Fuel Load and Reactor Physics Tests . . . . . 12.5-1

12.5.2 Low Power Testing . . . . . 12.5-1

12.5.3 Power Ascension Testing . . . . . 12.5-2

12.6 First-of-a-Kind Testing . . . . . 12.6-1

**Chapter 13 Research and Development . . . . . 13.1-1**

13.1 Research and Development Program . . . . . 13.1-1

13.2 Structures, Systems, or Components Requiring Additional Research and Development . . . . . 13.2-1

13.2.1 Emissivity Surface Layer . . . . . 13.2-1

13.2.2 In-Vessel Fuel Handling System In-Vessel Transfer Machine Grapple Finger Operation . . . . . 13.2-1

13.2.3 Intermediate Heat Transport System Sodium – Salt Heat Exchanger Interaction . . . . . 13.2-2

13.2.4 Core Restraint System Bi-Metallic Assembly . . . . . 13.2-2



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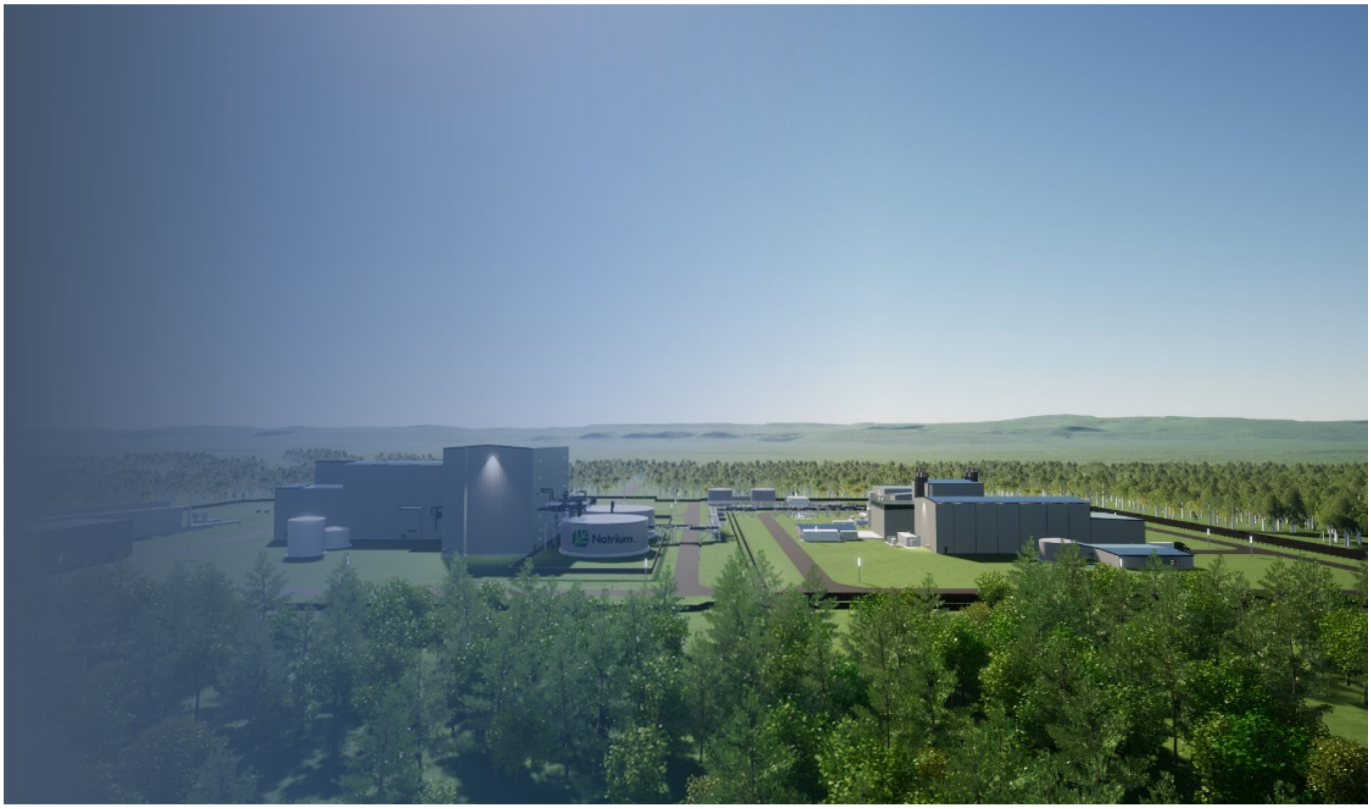

# Kemmerer Power Station Unit 1
# PSAR, Chapter 1

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## TABLE OF CONTENTS

**Chapter 1 General Plant and Site Description . . . . . 1.1-1**

1.1 Introduction. . . . . 1.1-1

1.1.1 Reactor Supplier and Model. . . . . 1.1-1

1.1.2 Intended Use of the Reactor . . . . . 1.1-1

1.1.3 Overall Configuration . . . . . 1.1-1

1.1.4 Description of Plant Structures, Systems, and Components . . . . . 1.1-2

1.2 Site Description . . . . . 1.2-1

1.3 Plant Safety Overview . . . . . 1.3-1

1.3.1 Licensing Basis Methodology. . . . . 1.3-1

1.3.2 Fundamental Safety Functions. . . . . 1.3-1

1.3.3 Defense-in-Depth . . . . . 1.3-5

1.4 Conformance with Regulatory Criteria and Referenced Material. . . . . 1.4-1

1.4.1 Regulatory Guidance . . . . . 1.4-1

1.4.2 Topical Reports and Technical Reports . . . . . 1.4-1

1.4.3 Generic Safety Issues, Unresolved Safety Issues and TMI Action Items. . . . . 1.4-2

1.4.4 Codes and Standards. . . . . 1.4-2


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## LIST OF TABLES

Table 1.4-1 Conformance with Regulatory Guides . . . . . 1.4-4

Table 1.4-2 Topical Reports . . . . . 1.4-30

Table 1.4-3 Technical Reports . . . . . 1.4-31

Table 1.4-4 Applicability of TMI-Related Requirements . . . . . 1.4-32

Table 1.4-5 Consensus Codes and Standards . . . . . 1.4-41


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## LIST OF FIGURES


Figure 1.1-1 Heat Transport Systems Overview Highlighting Operational Interface . . . . . 1.1-22

Figure 1.1-2 Plant Layout . . . . . 1.1-23

Figure 1.1-3 Heat Removal . . . . . 1.1-24

Figure 1.1-4 Safety Features . . . . . 1.1-25

Figure 1.1-5 General Arrangement RXB Section A Looking East . . . . . 1.1-26

Figure 1.1-6 General Arrangement RXB Section B Looking North . . . . . 1.1-27

Figure 1.1-7 General Arrangement FHB Section A Looking West . . . . . 1.1-28

Figure 1.1-8 General Arrangement FHB Section B Looking North . . . . . 1.1-29

Figure 1.1-9 General Arrangement FHB Section C Looking North . . . . . 1.1-30

Figure 1.1-10 General Arrangement RAB Section A Looking North . . . . . 1.1-31

Figure 1.1-11 General Arrangement RAB Section B Looking East . . . . . 1.1-32

Figure 1.1-12 General Arrangement NCB Substructure . . . . . 1.1-33

Figure 1.1-13 General Arrangement NCB Ground Floor . . . . . 1.1-34

Figure 1.1-14 General Arrangement NCB Looking West . . . . . 1.1-35

Figure 1.1-15 General Arrangement NCB Looking South . . . . . 1.1-36

Figure 1.1-16 General Arrangement FAB Ground Floor . . . . . 1.1-37

Figure 1.1-17 General Arrangement FAB Looking West . . . . . 1.1-38

Figure 1.2-1 Site Layout . . . . . 1.2-2

Figure 1.3-1 Diverse Barriers in Functional Containment . . . . . 1.3-6



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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| AC | alternating current |
| ACI | American Concrete Institute |
| ACLP | above core load pad |
| ACR | Audible Count Record |
| AEP | Annual exceedance probability |
| AFDD | accumulated freezing degree days |
| AFE | Annual Frequency of Exceedance |
| AHX | Sodium-Air Heat Exchanger |
| AIA | Aircraft Impact Assessment |
| AISC | American Institute of Steel Construction |
| ALARA | as low as reasonably achievable |
| ALOHA | areal locations of hazardous atmospheres |
| AMC | Auxiliary Monitoring and Control System |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| AOF | Allocation of Function |
| AOO | anticipated operational occurrence |
| API | American Petroleum Institute |
| ARCAP | Advanced Reactor Content of Application Project |
| ARDP | Advanced Reactor Demonstration Program |
| ARIS | assembly receipt inspection stand |
| ARM | Area Radiation Monitoring |
| ASCE | American Society of Civil Engineers |
| ASD | adjustable speed drive |
| ASE | Air Stack Structures and Equipment |
| ASME | American Society of Mechanical Engineers |
| AST | Anticipatory Automatic Seismic Trip System |
| ASTM | American Society for Testing and Materials |
| ATWS | anticipated transient without scram |
| AWOS | Automated Weather Observing System |
| AWS | American Welding Society |
| BDBE | beyond design basis event |
| bgs | below ground surface |
| BiMA | bi-metallic assembly |
| BLEVE | boiling liquid expanding vapor explosion |
| BLTC | bottom loading transfer cask |
| BPC | Bechtel Power Corporation |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| BPVC | Boiler and Pressure Vessel Code |
| BTP | Branch Technical Position |
| BTS | Battery Transfer Switch |
| BWR | boiling water reactor |
| CAID | core assembly identification |
| CAP | core assembly pot |
| CATT | core assembly transfer tube |
| CBS | Core Barrel Structures |
| CC | Climate Change |
| CCA | Collector Cylinder Assembly |
| CEUS | Central and Eastern United States |
| CFD | computational fluid dynamics |
| CFR | Code of Federal Regulations |
| CFW | Condensate and Feedwater System |
| CIP | core inlet plenum |
| CLP | cask loading pit |
| CLSM | Controlled Low Strength Material |
| CMAA | Crane Manufacturers Association of America |
| CN | runoff curve number |
| CNO | Chief Nuclear Officer |
| COLR | Core Operating Limits Report |
| COS | Climate Observing Station |
| COTS | commercial off-the-shelf |
| CP | construction permit |
| CPA | construction permit application |
| CPU | central processing unit |
| CQC | complete quadratic combination |
| CRA | control rod assembly |
| CRD | Control Rod Drive System |
| CRDM | control rod drive mechanism |
| CREAT | Climate Resilience Evaluation and Awareness Tool |
| CRS | Core Restraint System |
| CT | critical technology |
| CTC | Coolant Temperature Monitoring and Control System |
| CTE | critical technology element |
| CU | Consolidated-Undrained |
| CV | cleaning vessel |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| CVAP | Comprehensive Vibration Assessment Program |
| CVIS | core-vessel interface structure |
| CWIS | Cooling Water Intake Structure |
| D-RAP | Design Reliability Assurance Program |
| D/Q | atmospheric deposition factor(s) |
| DBA | design basis accident |
| DBE | design basis event |
| DBHL | design basis hazard level |
| DC | Direct Current |
| DCF | dose conversion factor |
| DCS | Distributed Control System |
| DG | Draft Regulatory Guide |
| DH | Downhole |
| DHS | Department of Homeland Security |
| DID | defense-in-depth |
| DIF | difficulty, importance, and frequency |
| DL | Defense Line |
| DMA | Degradation Mechanism Assessment |
| DOE | Department of Energy |
| DOE-EM | Department of Energy - Environmental Management |
| DOF | degrees of freedom |
| DOT | Department of Transportation |
| DRG | Design Review Guide |
| DRMS | Digital Radiation Monitor System |
| DRS | Design Response Spectra |
| EAB | exclusion area boundary |
| EAL | Emergency Action Level |
| EBR-II | Experimental Breeder Reactor II |
| ED | emergency director |
| EDC | EI DC power supply system |
| EEQ | electrical equipment qualification |
| EFP | Energy Island Fire Protection System |
| EHT | EI Heat Tracing System |
| EI | Energy Island |
| EIC | Energy Island Control System |
| ELV | EI AC Electrical Power Low Voltage System |
| E[M] | Expected Value of Moment Magnitude |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| EM | electromagnetic |
| EMC | electromagnetic compatibility |
| EMV | EI AC Electrical Power Medium Voltage System |
| EOP | emergency operating procedure |
| EPA | Environmental Protection Agency |
| EPZ | emergency planning zone |
| EQ | equipment qualification |
| ERF | Emergency Response Facility |
| ERFB | Emergency Response Facility Backup |
| ERFP | Emergency Response Facility Primary |
| ERM | Effluent Radiation Monitor |
| ERO | Emergency Response Organization |
| ESF | engineered safety feature |
| ESP | EI Standby AC Power System |
| ESS | Salt Heat Transport System |
| ETR | Energy Transfer Ratio |
| EUP | EI Uninterruptible AC Power Supply System |
| EVHM | Ex-Vessel Handling Machine |
| EVST | Ex-Vessel Storage Tank |
| F-C | frequency-consequence |
| FAA | Federal Aviation Administration |
| FAB | Fuel Auxiliary Building |
| FACP | Fire Alarm Control Panel |
| FE | finite element |
| FEMA | Federal Emergency Management Agency |
| FFC | Failed Fuel Canister |
| FFD | fitness-for-duty |
| FFR | failed fuel rack |
| FFV | fueling floor valves |
| FFTF | Fast Flux Test Facility |
| FGD | Flue Gas Desulfurization |
| FGR | Federal Guidance Report |
| FHA | Fire Hazards Analysis |
| FHB | Fuel Handling Building |
| FHC | Fuel Handling Supervisory Control System |
| FHCR | Fuel Handling Control Room |
| FHE | Ex-Vessel Fuel Handling System |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| FHI | In-Vessel Fuel Handling System |
| FHP | Water Pool Fuel Handling System |
| FHS | Fuel Transport and Storage System |
| FI | Fracture Index |
| FIPS | Federal Information Processing Standard |
| FIRS | Foundation Input Response Spectra |
| FIVS | fixed in-vessel shielding |
| FOAK | first-of-a-kind |
| FPC | Fuel Pool Cooling |
| FPP | Fuel Pool Purification |
| fps | feet per second |
| FRA | functional requirements analysis |
| FRMAC | Federal Radiological Monitoring and Assessment Center |
| FSAR | final safety analysis report |
| FSF | Fundamental Safety Function |
| FTA | fuel transfer adapter |
| FTL | Fuel Transfer Lift |
| FTLSP | Fuel Transfer Lift Shield Plug |
| FTP | Fuel Transfer Port |
| GAO | Government Accountability Office |
| GDC | General Design Criteria |
| GEH | General Electric Hitachi Nuclear Energy |
| GEN | Generator System |
| GFIMS | gas filtration impurities monitoring and sampling |
| GGRB | Greater Green River Basin |
| GL | generic letter |
| GMC | Ground Motion Characterization |
| GMRS | Ground Motion Response Spectra |
| GPCPD | gallons per capita per day |
| GPS | Global Positioning System |
| GSI | Geologic Strength Index |
| GUI | graphical user interface |
| GV | Guard Vessel |
| GVDS | Group-View Display System |
| HA | human action |
| HAA | Head Access Area |
| HAZMAT | hazardous materials |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| HCF | hot channel factor |
| HCl | hydrochloric acid |
| HEPA | high efficiency particulate air |
| HF | human factors |
| HFE | human factors engineering |
| HFEITS | Human Factors Engineering Issue Tracking System |
| HFEPP | Human Factors Engineering Program Plan |
| HFWUA | Hams Fork Water Users Association |
| HIC | High Integrity Container |
| HMI | human-machine interface |
| HMR | Hydrometeorological Report |
| HPM | human performance monitoring |
| HPS | The Health Physics Society |
| HRA | high radiation area |
| HRS | Heat Rejection System |
| HSI | human-system interface |
| HTFC | High Temperature Fission Chamber |
| HVAC | heating, ventilation and air conditioning |
| I&C | instrumentation and control |
| IAC | Intermediate Air Cooling System |
| IAEA | International Atomic Energy Agency |
| IBC | International Building Code |
| ICC | International Code Council |
| ICRP | International Commission on Radiological Protection |
| ICS | Integrated Control System |
| IDLH | immediately dangerous to life or health |
| IDPP | integrated decision-making process panel |
| IE | Initiating Event |
| IEC | International Electrotechnical Commission |
| IEEE | Institute of Electrical and Electronics Engineers |
| IHT | Intermediate Heat Transport System |
| IHX | Intermediate Heat Exchanger |
| INL | Idaho National Laboratory |
| INPO | Institute of Nuclear Power Operations |
| IO | input-output |
| IPCC | Intergovernmental Panel on Climate Change |
| ISA | International Society of Automation |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| ISB | Intermountain Seismic Belt |
| ISO | International Organization for Standardization |
| ISP | Intermediate Sodium Pump |
| ISRM | International Society for Rock Mechanics |
| ISRS | In-Structure Response Spectra |
| IST | in-service testing |
| ISV | Integrated System Validation |
| IVDC | In-Vessel Drive Controller |
| IVS | In-Vessel Storage |
| IVTM | In-Vessel Transfer Machine |
| LBE | licensing basis event |
| LCO | Limiting Condition for Operation |
| LCS | Local Control Station |
| LDA | Lead Demonstration Assembly |
| LFL | lower flammable limit |
| LFRS | Lateral Force Resisting System |
| LiDAR | Light Detection and Ranging |
| LIP | local intense precipitation |
| LMP | Licensing Modernization Project |
| LOOP | loss of offsite power |
| LPSD | Low-Power Shutdown |
| LRFD | Load and Resistance Factor Design |
| LPZ | low population zone |
| LSSS | Limiting Safety System Setting |
| LTA | lead test assembly |
| LTP | lead test pin |
| LWR | light water reactor |
| MANDE | monitoring and non-destructive examination |
| MANDEEP | monitoring and non-destructive examination expert panel |
| MCC | Motor Control Center |
| MCNP | Monte Carlo N-Particle Code |
| MCR | Main Control Room |
| MCT | maximum cladding temperature |
| MCV | movable closure valve module |
| MEI | maximally exposed individual |
| MIRSS | Modular Isolated Reactor Support Structure |
| mph | miles per hour |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| MPS | multi purpose sampler |
| MRE | Most-Recent Earthquake |
| MSL | Mean Sea Level |
| MSS | Manufacturers Standardization Society |
| MST | mechanistic source term |
| N/A | Not Applicable |
| NAAQS | National Ambient Air Quality Standards |
| NACP | Sodium Leak Detection Control Panel |
| NAD | North American Datum |
| NAP | North Ash Pond |
| NAPS | New Assembly Preconditioning Station |
| NAVD | North America Vertical Datum |
| NBSIR | National Bureau of Standards |
| NCB | Nuclear Island Control Building |
| NCDC | National Climatic Data Center |
| NCEI | National Centers for Environmental Information |
| NCH | Nuclear Island Cranes and Hoists |
| NDC | Nuclear Island DC Power Supply System |
| NEA | Nuclear Island Ancillary Electrical System |
| NEB | Nuclear Island Electrical Building |
| NEI | Nuclear Energy Institute |
| NEMA | National Electrical Manufactures Association |
| NES | Nuclear Island Auxiliary Electrical System |
| NFA | Nuclear Island Fire Detection and Alarm System |
| NFD | Nuclear Island Fire Water Distribution System |
| NFP | Nuclear Island Fire Protection System |
| NFPA | National Fire Protection Association |
| NFS | Nuclear Island Fire Suppression System |
| NGA | Nuclear Island Argon Gas Distribution and Storage System |
| NGI | Norwegian Geotechnical Institute |
| NGL | Nuclear Island Grounding and Lightning Protection |
| NGS | Nuclear Island Air and Gas Distribution System |
| NGVD 29 | National Geodetic Vertical Datum of 1929 |
| NHT | Nuclear Island Heat Tracing System |
| NHV | Nuclear Island Heating, Ventilation, and Air Conditioning System |
| NI | Nuclear Island |
| NIC | Nuclear Island Control System |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| NLS | Nuclear Island Lighting System |
| NLV | Nuclear Island AC Electrical Power Low Voltage System |
| NMFS | New Madrid Fault Source |
| NMV | Nuclear Island AC Electrical Power Medium Voltage System |
| NNA | Sodium Leak Detection, Collection, and Containment System |
| NOAA | National Oceanic and Atmospheric Administration |
| NRC | U.S. Nuclear Regulatory Commission |
| NRCS | Natural Resources Conservation Service |
| NSC | Nuclear Island Security System |
| NSHM23 | National Seismic Hazard Model 2023 |
| NSRST | non-safety-related with special treatment |
| NSS | Nuclear Island Salt System |
| NST | non-safety-related with no special treatment |
| NUMARC | Nuclear Management and Resources Council |
| NUP | Nuclear Island Uninterruptible AC Power Supply System |
| NWS | Nuclear Island Water System |
| NYS | Nuclear Island Yard System |
| OBE | Operating Basis Earthquake |
| OCC | Outage Control Center |
| OE | operating experience |
| OER | operating experience review |
| OL | operating license |
| OLA | operating license application |
| OQE | other quantified event |
| ORNL | Oak Ridge National Laboratory |
| OSHA | Occupational Safety and Health Administration |
| OW | Observation Well |
| P-S | Compressional (P) and Shear (S) Wave Downhole Velocities |
| PAM | Post-Accident Monitoring |
| PAV | Pool Auxiliary Vault |
| PCO | Plant Communication System |
| PCP | Process Control Program |
| PDC | Principal Design Criterion or Principal Design Criteria |
| PEM | pin extraction machine |
| PEP | plume exposure pathway |
| PFD | Probability of Failure on Demand |
| PGA | Peak Ground Acceleration |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| PGM | Plant General Manager |
| PHF | Plug Handling Fixture |
| PHM | Pool Handling Machine |
| PHT | Primary Heat Transport System |
| PI | Plasticity Index |
| PIC | Pool Immersion Cell |
| PITAP | post-construction inspection, testing, and analysis program |
| PM | Particulate Matter |
| PMC | Plant Monitoring and Control System |
| PMF | Probable Maximum Flood |
| PMP | Probable Maximum Precipitation |
| PMWP | Probable Maximum Winter Precipitation |
| POS | plant operating state |
| PRA | probabilistic risk assessment |
| PRC | Pin Removal Cell |
| PRM | Process Radiation Monitoring |
| PSAR | preliminary safety analysis report |
| PSF | probabilistic risk assessment safety function |
| PSHA | Probabilistic Seismic Hazard Analysis |
| PSP | Primary Sodium Pump |
| PSRA | Probabilistic Site Response Analysis |
| PTI | plugging temperature indicator |
| PZGW | piezometer (groundwater) |
| PZSW | piezometer (surface water) |
| QA | quality assurance |
| QAPD | Quality Assurance Program Description |
| QATR | Quality Assurance Topical Report |
| QFFD | Quaternary Fault and Fold Database |
| QHO | Quantitative Health Objective |
| R&D | research and development |
| RAB | Reactor Auxiliary Building |
| RAC | Reactor Air Cooling System |
| RCA | radiologically controlled area |
| RCC | Reactor Core System |
| RCP | Representative Concentration Pathways |
| RES | Reactor Enclosure System |
| RFDC | Required Functional Design Criteria |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| RFI | Reactor Fixed Internals |
| RG | Regulatory Guide |
| RIC | Radiation Indicator Controller |
| RIM | Reliability and Integrity Management |
| RIMEP | Reliability and Integrity Management Expert Panel |
| RIS | Reactor Instrumentation System |
| RMBS | Remote Manual Bypass Switch |
| RMC | Rod Monitoring and Control System |
| RMR | Rock Mass Rating |
| RMS | Radiation Monitoring System |
| RP | Radiation Protection |
| RPA | Rotatable Plug Assembly |
| RPD | Rotatable Plug Drive |
| RPJ | Rotatable Plug Jack |
| RPP | Radiation Protection Program |
| RPS | Reactor Protection System |
| RQD | Rock Quality Designation |
| RSB | Reactor Support Block |
| RSC | Remote Shutdown Complex |
| RSS | Reactor Support Structure |
| RTB | Reactor Trip Breaker |
| RV | Reactor Vessel |
| RVC | Reticulated Vitreous Carbon |
| RVT | Regulating Voltage Transformer |
| RWG | Gaseous Radwaste Processing System |
| RWL | Liquid Radwaste Processing System |
| RWS | Solid Radwaste Processing System |
| RXB | Reactor Building |
| SAMG | Severe Accident Management Guideline |
| SAP | South Ash Pond |
| SAR | safety analysis report |
| SARA | Superfund Amendments and Reauthorization Act |
| SARRDL | specified acceptable system radionuclide release design limit |
| SASW | Spectral Analysis of Surface Waves |
| SCADA | Supervisory, Control and Data Acquisition |
| SCG | Sodium Cover Gas System |
| SD | Standard Deviation |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| SDG | Standby Diesel Generator |
| SDOE | Secure Development and Operational Environment |
| SEI | Structural Engineering Institute |
| SFP | Spent Fuel Pool |
| SFR | sodium-cooled fast reactor |
| SGS | Steam Generation System |
| SHT | sodium holding tank |
| SHX | Sodium-Salt Heat Exchanger |
| SL | safety limit |
| SMACNA | Sheet Metal and Air Conditioning Contractors' National Association |
| SMR | small modular reactor |
| SMS | Seismic Monitoring System |
| SNM | special nuclear material |
| SOV | solenoid-operated valve |
| SPDS | Safety Parameter Display System |
| SPS | Sodium Processing System |
| SPS-E | Ex-Vessel Storage Tank Sodium Processing System |
| SPS-I | Intermediate Sodium Processing System |
| SPS-P | Primary Sodium Processing System |
| SPT | Standard Penetration Test |
| SR | safety-related |
| SRDC | Safety-Related Design Criteria |
| SRP | Standard Review Plan |
| SRSS | square root of the sum of squares |
| SSC | structure, system, or component |
| SSC | Seismic Source Characterization |
| SSE | safe shutdown earthquake |
| SSHAC | Senior Seismic Hazard Analysis Committee |
| SSI | Soil-Structure Interaction |
| SSSI | Structure-Soil-Structure Interaction |
| STP | standard temperature and pressure |
| STS | Steam Turbine System |
| SVDU | Special Video Display Unit |
| TA | Task Analysis |
| TATNF | time-at-temperature no-failure |
| TEDE | total effective dose equivalent |
| TEMA | Tubular Exchanger Manufacturers Association |


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## LIST OF ACRONYMS AND ABBREVIATIONS

| Term | Description |
|---|---|
| TFB | Turbine Facility Building |
| TFF | Sodium Test and Fill Facility |
| TI-RIPB | technology-inclusive, risk-informed, and performance-based |
| TMI | Three Mile Island |
| TMP | technology maturation plan |
| TRA | technology readiness assessment |
| TRL | technology readiness level |
| TS | technical specifications |
| TSS | Thermal Salt Storage System |
| TSTF | Technical Specification Task Force |
| TVSS | Transient Voltage Surge Suppressor |
| TWR® | Traveling Wave Reactor |
| UFC | Unified Facilities Criteria |
| UGRB | Upper Green River Basin |
| UHRS | Uniform Hazard Response Spectra |
| UIS | Upper Internal Structure |
| UMC | Utility Monitoring and Control System |
| UPS | uninterruptible power supply |
| USACE | United States Army Corps of Engineers |
| USCB | United States Census Bureau |
| USGS | United States Geological Survey |
| USL | Upper Subcritical Limit |
| USO | US SFR Owner, LLC |
| UTM | Universal Transverse Mercator |
| UU | Unconsolidated-Undrained |
| UV | undervoltage |
| V&V | verification and validation |
| VDU | Visual Display Unit |
| VHRA | Very High Radiation Area |
| $V_S$ | Shear-Wave Velocity |
| $V_{S30}$ | Time-averaged shear-wave velocity in the upper 30 meters |
| WGUEP | Working Group on Utah Earthquake Probabilities |
| WOHS | Wyoming Office of Homeland Security |
| WSEO | Wyoming State Engineer’s Office |
| WWDO | Wyoming Water Development Office |
| X/Q | atmospheric dispersion factor |
| XIS | Nuclear Instrumentation System |


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# Chapter 1 General Plant and Site Description

## 1.1 Introduction

US SFR Owner, LLC (USO), a subsidiary of TerraPower, LLC, is requesting approval for a construction permit of an 840 megawatt thermal (MWt) Natrium Reactor Plant, Kemmerer Power Station Unit 1 (Kemmerer Unit 1), to be located in Lincoln County, Wyoming approximately 3 miles (4.8 km) southwest of Kemmerer, Wyoming. Kemmerer Unit 1 is being licensed under Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities.”

This preliminary safety analysis report (PSAR) includes the information required by 10 CFR 50.34(a) to be provided as part of an application for a construction permit and demonstrates that the plant can be constructed and operated at the proposed location without undue risk to the health and safety of the public in accordance with 10 CFR 50.35(a). The layout and contents of the PSAR generally follow the guidance provided by Regulatory Guide 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” Revision 0, and Draft Regulatory Guide DG-1404, “Guidance for a Technology-Inclusive Content-Of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” Revision 1.

### 1.1.1 Reactor Supplier and Model

Kemmerer Unit 1 is a Natrium reactor supplied by TerraPower.

### 1.1.2 Intended Use of the Reactor

Kemmerer Unit 1 will be constructed and operated as part of participation in the U.S. Department of Energy's (DOE) Advanced Reactor Demonstration Program (ARDP) to demonstrate the Natrium technology and will provide electricity generation capacity in the PacifiCorp service area. The nature and inventory of contained radioactive materials is provided in Section 3.2.

### 1.1.3 Overall Configuration

The Natrium reactor plant is an 840 MWt pool-type sodium fast reactor (SFR) that contains a compact and simple safety envelope and a molten salt energy storage system which enables the plant to vary its supply of energy to the grid, up to 500 megawatts electric (MWe) net, while maintaining constant reactor power. The reactor operates near atmospheric pressure, circulating sodium through its core with pumps.

Heat is transferred from the primary loop within the Primary Heat Transport System (PHT) to an intermediate sodium piping loop within the Intermediate Heat Transport System (IHT) via intermediate heat exchangers located in the reactor pool. The IHT transports reactor heat from the intermediate heat exchangers to the Nuclear Island (NI) Salt System (NSS) via sodium-salt heat exchangers, which transport heat to the Energy Island (EI) hot salt storage tank for thermal


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energy storage. The heat stored in the hot salt tank is used to generate steam for use in commercially available steam turbine generators. Figure 1.1-1 provides an overview of these features.

The NI, which contains the reactor and its supporting systems, is designed to function as independently as possible from the EI, which contains the thermal energy storage tanks, steam generator, feedwater system, condenser, turbine, and supporting EI systems. The EI is physically connected to the NI by the NSS that transports heat between the islands and stores excess thermal energy, providing a buffer that allows steam generation and thermal energy storage operations (e.g., ramp rates) to be independent from reactor power operations as described in "Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces," (Reference 1.1-1). The EI structures, systems, and components (SSCs) are classified as non-safety-related with no special treatment (NST). Figure 1.1-2 depicts the NI, EI, and general plant layout.

The design includes reliable inherent and passive safety features including near-atmospheric operating pressures, always-on passive air cooling (Figure 1.1-3), and inherent reactivity feedback, which are summarized in Figure 1.1-4 and described in further detail in Section 1.1.4. Section 1.3 provides detail on the licensing basis and licensing basis methodology.

### 1.1.4 Description of Plant Structures, Systems, and Components

Kemmerer Unit 1 comprises an Energy Island and a Nuclear Island as described in Section 1.1.3. The major SSCs are summarized in Section 1.1.4.1 through Section 1.1.4.4.

Section 1.1.4.1 summarizes the reactor systems and components, including I&C systems needed to support reactor operation.

Section 1.1.4.2 summarizes the secondary systems and components, including salt and power cycle systems.

Section 1.1.4.3 summarizes the significant support systems and components, including fuel handing, NI support, NI fire protection, radioactive waste, and other support systems.

Section 1.1.4.4 summarizes the major structures of the NI, support buildings, and the EI facilities.

Further information on safety-related (SR) and non-safety-related with special treatment (NSRST) systems is provided in Chapter 7. The Gaseous Radwaste Processing System description is found in Chapter 7. The Solid Radwaste Processing System and Liquid Radwaste Processing System descriptions are found in Chapter 9.

#### 1.1.4.1 Reactor Systems and Components

The reactor is a metal fueled, pool-type, SFR. Safety functions are made integral to the reactor vessel, and support equipment is moved to separate structures, resulting in a simplified reactor building. The heat transfer characteristics of sodium, and the near atmospheric pressure design of the reactor, permit the use of compact and lightweight equipment. The reactor uses sodium-bonded metallic fuel with high chromium martensitic steel (HT9) cladding. The reactor


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operates at near atmospheric pressure, circulating sodium coolant through the core, with heat transfer from the primary sodium coolant (i.e., via PHT in-vessel pumps and in-vessel heat exchangers) to an intermediate sodium loop (i.e., the IHT). The intermediate sodium loop transfers heat to the NSS via Sodium-Salt Heat Exchangers (SHXs) for normal steady state operations, and to the atmosphere via the Intermediate Air Cooling System (IAC) sodium-air heat exchanger during startup and shutdown. The SSCs summarized in this section are reactor systems and components and those used for reactor control.

#### 1.1.4.1.1 Reactor Core System

The Reactor Core System (RCC) comprises various removable core assemblies (e.g., fuel, control, reflector, shield). The RCC is designed to sustain a nuclear chain reaction for fission heat generation within the thermal-mechanical constraints of the components (e.g., fuel, cladding, structures). Thermal power and reactivity are controlled by a combination of fuel management operations on a cycle-by-cycle basis and the insertion and withdrawal of control rods throughout a cycle.

The reactor core is enclosed within and supported by the core barrel structures within the Reactor Enclosure System (RES). During normal operation, the reactor core generates and transfers nuclear heat to the sodium of the PHT, which transfers heat to the sodium of the IHT. The Primary Sodium Pumps (PSPs) of the PHT supply sodium coolant flow to the core inlet plenum located at the bottom of the reactor core. Coolant enters through core assembly receptacles and nozzles from the core inlet plenum and flows upwards to remove the heat generated from the fuel pins. As the sodium coolant exits at the outlet of each core assembly, it mixes in the hot pool within the reactor vessel and flows past the upper internal structure. The sodium travels through the IHXs before it is pumped back to the core inlet. When the reactor core is shut down for refueling, the core assemblies are removed from the reactor core using the In-Vessel Transfer Machine (IVTM). Fuel assemblies are placed in the In-Vessel Storage (IVS) for decay before removing from the reactor vessel for longer-term storage.

#### 1.1.4.1.2 Reactor Enclosure System

The RES provides all structural, functional, and interface support for the PHT, RCC, and Control Rod Drive System (CRD). The RES provides one of the functional containment boundaries for the primary sodium coolant and sodium cover gas as described in Section 1.3.2.1.

The RES supports the reactor core and contains the primary sodium coolant. The RES has been designed with the following fundamental features: an integral reactor, reactor seismic isolation, top supported vessels, a passively cooled reactor head, single rotatable plug, in-vessel core assembly storage, and passive safety cooling via the Reactor Air Cooling System (RAC). Additional features, such as no penetrations through the Reactor Vessel (RV), minimize the potential for primary system leakage or rupture. Shielding features within the reactor provide radiation protection to reactor components, intermediate coolant, and the Head Access Area (HAA), while the inert gas-filled annulus between the RV and Guard Vessel (GV) protects the system from sodium-air reactions in the highly unlikely event of a RV leak. Internal RES


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structures provide primary coolant flow paths for reactor heat transfer and structural support for the PHT and RCC. The RES also provides positional control and alignment to the CRD and other reactor components critical to reactor operations.

#### 1.1.4.1.3 Reactor Air Cooling System

The RAC primary objective is to provide passive decay heat removal to the environment through natural convection of air. The RAC is inherently reliable since it is not subject to the types of failure modes normally associated with active cooling systems (such as via pumped flow through heat exchangers). It is continually in service with no operator actions nor SSC state change to initiate cooling operation. RAC heat removal performance is monitored during system operation including post-accident conditions.

Heat transfer to RAC begins with the transfer of decay heat from the core and fuel stored in the RV through the primary sodium to the RV by convection and conduction. The RV is designed to thermally radiate heat across the argon gas space to the GV. The GV transfers a portion of the heat through convection directly into the RAC air flow path. To enhance total heat transferred, heat is thermally radiated to the collector cylinder which through convection transfers heat to the RAC air flow path. This increases the capacity of RAC heat removal without increasing the surface area of the GV.

#### 1.1.4.1.4 Primary Heat Transport System

The PHT, which is contained within the RES, consists of the hot pool, cold pool, warm pool, two IHXs, and two PSPs. The principal function of the PHT is to transport heat from the reactor core to the IHT for use in power generation. The PHT safety function is to maintain primary sodium temperatures within design limits that preclude damage to the RV, the fuel, or reactor internal components. During normal operation, plant startup and shutdown, and during certain off-normal events, the two primary sodium pumps provide the necessary system head to drive primary sodium flow. The IHXs facilitate heat transfer from the primary sodium to intermediate sodium, ultimately transferring heat to the EI for electricity production. During a design basis event, the PHT safety function is to facilitate natural circulation of the primary sodium to transfer heat from the reactor core to the reactor vessel walls, where the heat is exchanged to the guard vessel and to the RAC.

The sodium volumes of the hot pool, cold pool, warm pool, along with the sodium within the PSP, IHX, primary portion of Sodium Processing System (SPS), and RCC, form the primary coolant inventory. In addition, the void area between the sodium hot pool and the reactor head is inerted using argon gas provided by the Sodium Cover Gas (SCG).

#### 1.1.4.1.5 Intermediate Heat Transport System

The IHT primary function is to transport heat away from the PHT. The IHT delivers the heat removed from the PHT to the NSS for energy storage or electrical power generation, or to the IAC to be rejected as waste heat to the atmosphere. The IHT working fluid is liquid sodium. Hot sodium is driven through the intermediate loops by the Intermediate Sodium Pumps (ISPs)


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during normal operations and is driven by natural circulation when power is not available. Heat is transferred from the PHT to the IHT as intermediate sodium passes through the PHT IHXs tubes. The boundary between the IHT and PHT occurs at the IHX nozzle to IHT pipe welds.

The IHT transfers heat to the NSS through the SHXs during normal power operation of the reactor. The boundary between the IHT and the NSS is at the SHX shell side nozzle to pipe weld. The IHT transfers heat to the IAC through the Sodium-Air Heat Exchanger (AHX) for decay heat removal and for reactor startup and shutdown operations. In this role, the IHT provides part of the normal, preferred pathway for rejection of decay heat. The AHX is part of the IAC and the boundary between the IHT and the IAC is at the AHX tube (sodium) side nozzle to pipe welds.

#### 1.1.4.1.6 Intermediate Air Cooling System

The IAC is the preferred heat removal system for the reactor during upset events. The IAC transfers heat from the IHT to the atmosphere. The IAC consists of two AHXs, each contained in a chimney structure. These chimneys are located on either side of the Reactor Auxiliary Building (RAB). Each is connected within one of the two intermediate loops.

The IAC transfers heat to the atmosphere in three system modes of operation. In active mode, there is forced circulation of the sodium in the intermediate loop provided by the ISP and forced air flow across the AHX provided by the IAC air blower. In blower mode, there is natural circulation of the sodium in the intermediate loop and forced air flow across the AHX provided by the IAC air blower. In passive mode, there is natural circulation of the sodium in the intermediate loop and natural convection flow of the air across the AHX.

#### 1.1.4.1.7 Sodium Cover Gas System

The SCG controls, monitors, and supplies inert argon gas to various systems and components throughout the Reactor Building (RXB), Reactor Auxiliary Building (RAB), and Fuel Handling Building (FHB). The SCG comprises the primary sodium cover gas supporting the PHT and the intermediate sodium cover gas supporting the IHT.

The primary SCG consists of gas distribution, monitoring, and filtering and is located in the RXB with additional pipe routing in the FHB. It receives high pressure, purified argon from the NI Air and Gas Distribution System (NGS) into the gas distribution modules in the primary supply control skid. The modules adjust gas pressure and control the flow to the various seals and penetrations connected to the RES according to the specific needs of the interface. These include the RV, RV to GV annulus, and various reactor head buffer and purge seals.

The intermediate SCG receives purified argon gas from the NGS within the RAB, adjusts pressure, and controls flow as required by the downstream processes. It then distributes the gas to the required locations, including ISP tanks, IHT drain tanks, ISP seals, ISP lube oil skids, and aerosol filter purges and sample line purges.


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#### 1.1.4.1.8 Sodium Processing System

The Sodium Processing System (SPS) monitors and removes contaminants including cesium and compounds including oxygen and hydrogen from the liquid sodium inventory. To a lesser degree, media designated for contaminant capture will also remove debris and corrosion products from the liquid sodium inventory.

The Primary Sodium Processing System (SPS-P) draws primary sodium from the RV using a submersible electromagnetic pump and delivers it to the sodium monitoring and purification equipment located below grade in the RAB. Isolation valves are located on the supply and return lines in the RXB HAA. The purification equipment consists of cold traps, which remove oxides and hydrogen impurities, and a cesium trap, which removes cesium released to the coolant from fuel pin failures. The sodium is continuously monitored for oxygen and hydrogen using a plugging temperature indicator, multi-purpose sampler and direct-measurement oxygen and hydrogen sensors. Cesium concentration is monitored using a radiation monitor. Sampling for detailed sodium chemistry is provided by the multi-purpose sampler. After analysis and purification, sodium is returned to the reactor vessel.

The Intermediate Sodium Processing System (SPS-I) draws intermediate sodium from each IHT loop and the IHT drain tanks using an electromagnetic pump, and delivers the sodium to its respective sodium monitoring and purification equipment located on grade level in the RAB. The SPS-I contains the same equipment and function as the SPS-P with the exception that the SPS-I does not contain a cesium trap or associated radiological monitoring.

The Ex-Vessel Storage Tank Sodium Processing System (SPS-E) draws sodium from the Ex-Vessel Storage Tank (EVST) using an electromagnetic pump and delivers it to sodium monitoring and purification equipment located in the vault below grade in the FHB. The SPS-E contains the same equipment and function as the SPS-I.

#### 1.1.4.1.9 Control Rod Drive System

The purpose of the CRD is to provide core reactivity control for the purpose of core power control, power shaping, and reactor shutdown. The CRD consists of control rod drive mechanisms (CRDMs) along with the associated motor controllers for the CRDM motors. The CRD driveline, linking the CRDM to the control rod assembly (CRA), is included in the CRD. The CRDMs are mounted to the reactor head on the Rotatable Plug Assembly (RPA). They provide for reactivity control for the reactor, providing insertion and withdrawal of CRAs.

The CRD consists of thirteen CRDMs that control the position of nine primary CRAs and four secondary CRAs that are used to ensure reactor shutdown when required in response to an off-normal condition. The CRD scram function is gravity-driven and is fail safe on a loss of electrical power. The CRD gravity scram is backed up by CRDM motor drive-in of the CRAs for defense-in-depth. The scram function is controlled by the RPS. The CRD also provides for CRA position indication and other system monitoring. The CRDM motor controllers are supplied power from the NI Uninterruptible AC Power Supply System (NUP) which includes battery backup.


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#### 1.1.4.1.10 Instrument and Control Systems

The instrumentation and control (I&C) systems provide manual and automatic control of plant systems during modes of operation and transient conditions. The I&C systems also protect the reactor, plant, personnel, and the public from unsafe operation through the prevention and mitigation of consequences of licensing basis events (LBEs) and design basis accidents (DBAs) and ensures safe shutdown. The systems implement the fundamental I&C system design principles of qualification, reliability, robustness, security, and defense-in-depth.

Additional I&C systems are described in more detail in Section 7.6.

#### 1.1.4.1.11 Reactor Protection System

The Reactor Protection System (RPS) is an I&C system consisting primarily of electronics and electrical equipment cabinets in the NI Control Building (NCB) substructure as well as operator interface devices in the Main Control Room (MCR) and Remote Shutdown Complex (RSC). The primary function of the RPS is to, with precision and reliability, accept input signals from plant instrumentation, apply necessary logic, and automatically generate output signals to initiate safety functions including reactor scram and engineered safety feature functions when required by plant conditions. The RPS also serves to display critical information to operators in the MCR and RSC during normal and postulated post-accident conditions.

The RPS are arranged into four redundant and independent divisions. Each division is physically separated from the other redundant RPS divisions to the degree necessary to ensure accomplishment of safety functions. Each division is physically separated or isolated from other systems, including interconnected systems and equipment in proximity, to the degree necessary to ensure accomplishment of safety functions. Each RPS division receives two parallel power feeds from the NUP, with each division’s power supply maintaining independence and separation from the others.

#### 1.1.4.1.12 Nuclear Instrumentation System

The Nuclear Instrumentation System (XIS) provides instrumentation to sense neutron flux leakage from the core during fuel movement, reactor startup, power operations, shutdown, and accident conditions. Neutron leakage is proportional to reactor flux and thus can be converted into a power indication. The XIS amplifies and conditions the instrument signals and provides corresponding input signals to the RPS and the NI Control System (NIC) for use in protective, control, and monitoring functions. The XIS comprises of four redundant divisions, each of which includes a division-specific flux detectors and signal conditioning. Each XIS division is separate and independent from the other XIS divisions.

#### 1.1.4.1.13 Radiation Monitoring System

The Radiation Monitoring System (RMS) provides radiation activity indications for systems, buildings, and release pathways throughout the plant at strategic interface points for use in the protection of plant personnel and the surrounding environment during all operating conditions. Continuous radiation monitoring, sampling, and analysis is provided for radioactive processes and for release points where radioactive effluents leave the site boundary. The RMS consists of


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three main subsystems: area radiation monitoring, effluent radiation monitoring, process radiation monitoring. The area radiation monitoring subsystem provides indication and alarms to warn personnel of a possible impending radiation health hazard. The effluent radiation monitoring subsystem monitors gaseous release points of the plant environs for indications of radiation levels above pre-determined release setpoints. The effluent radiation monitoring subsystem's primary function is to protect the health and safety of the public and to provide assurance that off-site exposure does not exceed 10 CFR 20 limits. The process radiation monitoring subsystem monitors various processes and airborne locations for indications of increasing radiation levels. The process radiation monitoring subsystem strategy is to give operators early indication of abnormal radiological conditions in a plant process that may be an indication of failures within, or external to, the monitored process (e.g., failed fuel detection). The RMS provides local and control room indication of plant radiological conditions in all areas that are required to be monitored for area radiation levels, airborne activity, and radioactive material in various gaseous and liquid process and effluent streams.

#### 1.1.4.1.14 Reactor Instrumentation System

Reactor Instrumentation System (RIS) provides measurements for parameters and sends these signals in an appropriate format to RPS, and NIC, for control, safety functions, post-accident monitoring, and informational purposes. Parameters include hot pool level, hot pool temperature, cold pool temperature, and PSP shaft speed. RIS also provides the RV-GV annulus liquid sodium leak detection parameters as an input to RPS.

#### 1.1.4.1.15 Nuclear Island Control System

The NIC is a group of I&C subsystems that perform non-safety-related with no special treatment (NST) and NSRST functions. The backbone of the NIC is implemented on a NI Distributed Control System (DCS) that provides a means for gathering information from field sensors, executing both manual and automatic logic, interfacing to field actuators, and providing a means of human interface between this equipment and operators. Additionally, the NIC includes controls that are executed outside of the DCS. These are referred to as "vendor supplied" control systems. Vendor supplied control systems include interfaces for communication with the DCS. The subsystems that are part of the NIC include the Rod Monitoring and Control System (RMC), the Coolant Temperature Monitoring and Control System (CTC), the Utility Monitoring and Control System (UMC), the Auxiliary Monitoring and Control System (AMC), the Fuel Handling Supervisory Control System (FHC), and the Plant Monitoring and Control System (PMC).

#### 1.1.4.1.16 Rod Monitoring and Control System

The Rod Monitoring and Control System (RMC) is a subsystem of the NIC that is used to control and monitor primary control rod assemblies and secondary control rod assemblies for all reactor operation modes.

During normal, startup, shutdown transition operations, anticipated operational occurrences, and design basis events, the RMC includes monitoring, indication, alarm, power runback, normal insertion and withdrawal, and rod withdrawal inhibit functions through the manipulation of the


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CRD. During shutdown and refueling, the RMC includes functions to drive the uncoupled CRDM drivelines to their fully withdrawn positions from the core and provide the necessary interlocks to the FHC so refueling operation may occur.

#### 1.1.4.1.17 Coolant Temperature Monitoring and Control System

The CTC monitors and controls the PHT, IHT, NSS, IAC, and EI Salt Heat Transport System (ESS) flow control functions. While the CTC has no active control function of the RAC, the CTC can influence the RAC performance through control of PHT.

The CTC consists of hardware networked to programmable logic controllers, software programing that implements the control functions, and a control interface implemented through the PMC. The PMC is a separate computer system from the CTC, but interfaces directly with the CTC. The PMC provides information and control displays in the MCR that the plant operators can use to monitor and modify the function of the CTC. The PMC also provides display of CTC alarm, warning, and informational messages.

#### 1.1.4.1.18 Utility Monitoring and Control System

The UMC processes information from various sources such as input-output from field devices and from digital interfaces to other systems. The UMC takes this information and makes it available to the plant data highway for use by the PMC for indication and controls as well as making it available to other NIC systems and resources.

The UMC continually monitors various parameters and variables within the NI for interfacing systems. It collects data from field instrumentation, such as temperature, pressure, and radiation levels. It is equipped to perform both manual and automatic controls allowing operators to manage and adjust auxiliary systems as necessary. Systems the UMC interfaces with include PMC, Seismic Monitoring System (SMS), NI AC Electrical Power Medium Voltage System (NMV), NI AC Electrical Power Low Voltage System (NLV), NI DC Power Supply System (NDC), NI Uninterruptible AC Power Supply System (NUP), NI Heat Tracing System (NHT).

#### 1.1.4.1.19 Auxiliary Monitoring and Control System

The AMC processes information from various sources such as input-output from field devices and from digital interfaces to other systems. The AMC takes this information and makes it available to the plant data highway for use by the PMC for indication and controls as well as making it available to other NIC systems and resources.

The AMC continually monitors various parameters and variables within the NI for interfacing systems. It collects data from field instrumentation, such as temperature, pressure, and radiation levels. It is equipped to perform both manual and automatic controls allowing operators to manage and adjust auxiliary systems as necessary. Systems the AMC interfaces with include PMC, RMS, Gaseous Radioactive Waste System (RWG), Liquid Radioactive Waste System (RWL), Solid Radioactive Waste System (RWS), SCG, and SPS.


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#### 1.1.4.1.20 Fuel Handling Supervisory Control System

The Fuel Handling Supervisory Control System (FHC) functions as the Supervisory, Control and Data Acquisition (SCADA) for the fuel handling processes. The FHC supports a supervisory control mode (remote mode) for independent fuel handling machines with onboard logic solvers. The FHC supervises the co-movement of independent machines during machine movements. The FHC supports three remote control states, remote automatic, remote step, and remote manual.

#### 1.1.4.1.21 Plant Monitoring and Control System

The monitoring and controlling of NI systems is accomplished using plant controllers, networks, and human-machine interfaces (HMIs). The PMC provides plant operators the capability to monitor plant data and control plant equipment through the HMI in all operating modes. The PMC does not have ability to perform automatic control. However, it has a user interface to perform manual control through NIC subsystems controllers. The PMC HMIs provide graphical representations of the plant's processes, systems, and equipment, provide control of various plant components and systems, generate alarms and annunciations for abnormal or hazardous conditions, store and display historical data, and provide integration of operating procedures and other reference materials.

### 1.1.4.2 Secondary Systems and Components

The secondary systems and components transform the heat generated within the reactor into electrical power. The reactor operates at near atmospheric pressure, circulating sodium coolant through its core, with heat transferred from the primary sodium coolant (via PHT in-vessel pumps and in-vessel heat exchangers) to an intermediate sodium loop (the IHT). The intermediate sodium loop transfers heat to the NSS via sodium to salt heat exchangers. The NSS transfers heat to the ESS and to the Thermal Salt Storage System (TSS) hot tank. The salt stored in the TSS hot tank is pumped through the steam generators and sent to the TSS cold tank. The steam generator equipment converts water into steam by passing the hot salt through an economizer (water preheater), evaporator, superheater, and reheater. The steam generator provides steam to the turbine to produce power.

As described in Reference 1.1-1, the NI boundary conditions have been intentionally designed so the interface with the EI does not impact the Kemmerer Unit 1 safety analysis. Similarly, steam generation and thermal energy storage operations (e.g., ramp rates) are independent from reactor power operations due to the presence of the molten salt energy storage tanks. All EI systems and functions described in the following sections are classified as NST.

#### 1.1.4.2.1 Salt Systems (NSS, ESS, TSS)

Heat is removed from the IHT SHXs in the RAB by the NSS to ESS to TSS molten salt loop during reactor operation. This salt is then stored in the TSS molten salt storage tanks to reserve thermal energy to accommodate fluctuations in steam turbine generator output. Cold salt from the TSS cold salt storage flows through the ESS to NSS cold salt piping via the ESS cold salt transfer pumps, across the EI and NI boundary, to the IHT SHXs. Hot salt from the SHXs flows through the NSS to ESS hot salt piping back across the NI and EI boundary to the TSS hot salt


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storage. The hot salt is pumped from the TSS hot salt storage through the Steam Generation System (SGS) to transfer heat from the salt to water to create steam. Cold salt is returned from the SGS to the TSS cold salt storage.

The NSS isolation valves outside of the RAB form the boundary between the NSS and ESS systems. The NSS monitors molten salt flow rate and the hot and cold salt temperatures and pressures. This provides feedback to the MCR that the NSS to ESS to TSS to SGS loop is transferring heat to the EI from the NI during normal operations. The NSS to ESS to TSS to SGS systems form the salt loop that transfers heat from the NI to the EI.

#### 1.1.4.2.2 Power Cycle Systems (SGS, CFW, STS, GEN, HRS)

Heat is removed from the TSS salt loop via the SGS. Salt flows through the SGS heat exchangers to produce superheated and reheated steam. The SGS receives high pressure feedwater from the Condensate and Feedwater System (CFW) for steam generation. The Steam Turbine System (STS) converts steam into rotational energy. The Generator System (GEN) converts the rotational energy into electricity. The CFW condenses waste steam from the STS in the condenser. Condensate is pumped through several low pressure feedwater heaters to the deaerator. The feedwater pumps take suction from the deaerator and supply high pressure feedwater to the SGS. The Heat Rejection System (HRS) removes heat from the condenser through the cooling tower.

#### 1.1.4.2.3 Main Power

The Main Power is the main power system for Kemmerer Unit 1 and consists of three subsystems:

- The 230 kilovolt (kV) power subsystem that transfers power between Kemmerer Unit 1 and the 230 kV transmission lines
- A normal main power supply subsystem that transforms and transfers the power generated by the steam-turbine generator unit to the 230 kV switchyard, and supplies plant auxiliary loads through the unit auxiliary transformer during plant normal operation
- An alternate main power supply subsystem that transforms and transfers power from the 230 kV switchyard to the plant auxiliary loads through the reserve auxiliary transformer when the normal main power supply subsystem is out of service

During normal plant operation, the normal main power supply subsystem transmits generated power from the main generator to the 230 kV switchyard through the generator circuit breaker and the main step-up transformer. Plant auxiliary power is provided from the isolated phase bus tapped connection through the unit auxiliary transformers, which, in turn, supplies the medium voltage 6.9 kV distribution system.

During plant startup and shutdown, with the generator circuit breaker open, plant auxiliary loads are back fed power via the main step-up transformer from the 230 kV switchyard to the unit auxiliary transformers. When the normal main power supply system is not available, plant auxiliary loads are supplied from the reserve auxiliary transformer, an alternate main power supply system, fed from the 230 kV switchyard.


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### 1.1.4.3 Significant Support Systems and Components

#### 1.1.4.3.1 In-Vessel Fuel Handling System

In-Vessel Fuel Handling System (FHI) is broken down into six major pieces of equipment: In-Vessel Transfer Machine (IVTM), In-Vessel Drive Controller (IVDC), Rotatable Plug Drive (RPD), Rotatable Plug Jack (RPJ), Fuel Transfer Lift (FTL), and Fuel Transfer Lift Shield Plug (FTLSP). Together, these components manipulate the positions of core assemblies during refueling operations and maintain the primary functional containment boundary during all reactor modes when the equipment is installed. The system can move core assemblies around within the core, from core location to the IVS location outside the core, and from any location within the core to the Ex-Vessel Handling Machine (EVHM). During normal refueling operation, the FHC provides interlocks and remote automatic control of the FHI equipment and systems. The RPD, RPJ, and IVTM have major interfaces with the RPA which is under the scope of the RES.

The FTL and IVTM are only installed during refueling and are removed from the reactor during power operations.

#### 1.1.4.3.2 Ex-Vessel Fuel Handling System

The Ex-Vessel Fuel Handling System (FHE) includes equipment that facilitates the receipt, conditioning, storage, installation, and removal of all core assemblies. All the FHE equipment that handles core assemblies include inert argon gas environment or sodium as a cooling medium. During a refueling outage, the ex-vessel fuel handling equipment moves non-irradiated and irradiated core assemblies into and out of short-term storage in the Ex-Vessel Storage Tank (EVST) and the reactor vessel. Following the refueling outage, irradiated core assemblies in the EVST are processed through a Pool Immersion Cell (PIC) to a spent fuel storage pool.

#### 1.1.4.3.3 Water Pool Fuel Handling System

The Water Pool Fuel Handling System (FHP) consists of the subsystems and equipment necessary to store irradiated core assemblies within the water-cooled pool. The FHP utilizes several fluid systems to maintain the Spent Fuel Pool (SFP) environment suitable to temporary storage of spent nuclear fuel. The FPC supports rejection of decay heat produced by fuel stored in the pool. The Fuel Pool Purification System (FPP) maintains pool water chemistry within specification utilizing a series of media filters and ion exchangers. Finally, a chemical processing system is employed by the PIC to strip residual sodium from core assemblies prior to entrance to the pool. This process skid retains and directs effluent separate from the rest of the SFP.

The SFP is a subgrade pool used to store spent nuclear fuel. The base structure, provided by the FHB, is concrete with an inner stainless-steel lining to prevent water migration. The SFP is a passive system with no gates, blocks, or moving parts save the Pool Handling Machine (PHM) which is used to manipulate core assemblies. There are no hydraulic penetrations or pipes extending below the minimum required water level for fuel rack shielding. There is a single permanent concrete barrier that separates the fuel storage area from the cask loading pit.


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#### 1.1.4.3.4 Main Control Room

The MCR contains equipment that is part of the NIC that allows operators to initiate or take manual control of each function necessary to control the plant and maintain safety. Information is displayed to the operators via the NIC HMI displays and Safety Parameter Display System (SPDS).

The MCR is designed to satisfy the requirements of Principal Design Criterion 19. Adequate habitability measures are provided to permit access and occupancy of the MCR during normal operations and under accident conditions.

#### 1.1.4.3.5 Nuclear Island Distributed Control System

The DCS is the primary subsystem of the NIC and provides a means for gathering information from various sources, such as field sensors, human-system Interfaces (HSIs), and data feeds from other I&C systems. The DCS also interfaces with various field actuators such that it can execute both manual and automatic logic based on the gathered information. The DCS platform provides a set of standard hardware and software that can be configured and customized as needed to meet the needs of the functions being implemented. The DCS can be expanded as needed to implement the various aspects of the NIC. This allows for the different sub-systems of the NIC to use common hardware and software which helps to simplify the design of the NIC.

#### 1.1.4.3.6 Nuclear Island Auxiliary Electrical Systems

The NI electrical power distribution is divided into several systems to distribute auxiliary electrical power to the NI of Kemmerer Unit 1. The NMV supplies power to the medium voltage loads, 6.9kV-480V transformers, and adjustable speed drives for the PSPs and ISPs. The NLV supplies power to the 480V switchgears, 480V Motor Control Center (MCC), and 480V switchboards and panelboards for NI plant auxiliary loads. The NI electrical system also includes an NHT that distributes power to heating cables on process lines and tanks to maintain specific fluid temperature to prevent freezing and crystallization of process fluids, such as molten salts, sodium, and water.

Electrical power for the plant’s critical loads such as I&C circuits, and other electrical and mechanical auxiliaries is provided by the NDC and NUP. The NMV also receives backup power from two Standby Diesel Generators (SDGs) for the plant-specific loads as investment protection.

During a loss of offsite power event, the standby diesel generators are automatically started and provide backup power to the NMV for plant-specific loads, including power to NI battery chargers, and heat tracing to avoid freezing. The standby diesel generators are part of the EI Standby AC Power System (ESP).

#### 1.1.4.3.7 Gaseous Radwaste Processing

The RWG is located in the FHB and is a once-through system that processes the gaseous waste streams from primary SCG and the SCG exhaust distribution cabinet. The system provides holdup for the decay of short-lived radioactive isotopes such as Argon-41. Waste gas is


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transported through filtration medium and activated charcoal adsorption beds, which provide enhanced decay of longer-lived isotopes like xenon and krypton prior to environmental discharges to the plant stack.

Discharge from the RWG to the plant stack is diluted by additional airflow from various Nuclear Island Heating, Ventilation, and Air Conditioning System (NHV) inputs. Exhaust flow from the NHV is combined with RWG discharge gas and is monitored to ensure dose rates are within administrative and federal limits.

#### 1.1.4.3.8 Liquid Radwaste Processing

The RWL is designed to collect, segregate, process, sample, and monitor the non-sodium and non-salt liquid radioactive waste for reuse or evaporation. The RWL equipment is primarily located in the FHB. The RWL tanks receive and store radioactive or potentially radioactive liquid waste. The RWL tanks store the waste during normal operation and during anticipated operational occurrences. Monitoring and sampling of the liquid waste is performed prior to reuse or evaporation. The RWL provides protection to plant personnel and the environment, minimizes releases and the spread of contamination, and ensures personnel exposures are as low as reasonably achievable.

#### 1.1.4.3.9 Solid Radwaste Processing

The RWS is designed to process both wet and dry solid waste from plant systems and sources. The RWS is located in the FHB, which has adequate space for onsite storage of solid waste containers and processing equipment. The RWS consists of three key design features: a dewatering skid, a compactor skid, and a radwaste storage area. Solid wastes are processed by means of decontaminating, surveying, sorting, dewatering, compaction, and classifying for storage and shipment to licensed off-site facilities. The RWS is not intended to process large radioactive waste materials, such as core assemblies, spent fuel, or large failed equipment, nor is it intended to provide long-term storage of waste.

#### 1.1.4.3.10 Nuclear Island HVAC System

The NHV is provided as required for personnel comfort, freeze protection, and equipment operability in areas identified as requiring temperature control on a per-building basis. NHV provides ventilation for areas identified as hazardous environments to mitigate hazardous concentrations in accordance with applicable standards for indoor air quality. Radiologically controlled areas (RCAs) are maintained at required relative negative pressure by NHV to control migration of airborne contaminants and prevent an unmonitored release. Additionally, NHV filters exhaust air from high efficiency particulate air (HEPA) filters, where required, to maintain radiological material releases below regulatory limits.

The NHV is locally controlled in each building with indication in the MCR. The NHV has limited remote operation for specific NHV functions such as inerting, purging, and isolation in the RXB and RAB. NHV provides a radionuclide retention function as part of the PRA defense line level four functions in the RXB.


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The NHV interfaces with the NFP to mitigate the effects of fire and smoke in NI buildings, and interfaces with the NGS to provide purging and inerting functions in the RXB and RAB areas which require an inert atmosphere during operations. Control room habitability as required per Principal Design Criterion 19 is ensured by the NHV in the NCB through monitoring for identified airborne hazards, recirculation operation, and filtering.

#### 1.1.4.3.11 Nuclear Island Ancillary Electrical System

The NI ancillary electrical system includes the NI Lighting System (NLS) (normal, egress, essential, and emergency operating lighting) and NI Grounding and Earthing and Lightning Protection (NGL). The normal lighting system provides illumination for general indoor and outdoor lighting during plant operation, maintenance, and test conditions. Electrical power for the normal lighting system is supplied from the NLV. Upon loss of normal lighting, the NLS egress lighting, essential lighting, and emergency operating lighting provides illumination in NI plant areas to support emergency plant operations, including fire events. The NLS egress, essential, and emergency operating lighting power supplies are backed up by either a self-contained battery, SDG backed low voltage AC power, or the NUP.

The NGL provides protection of plant personnel and equipment from the effect of transient over-voltages that can occur in electrical systems, voltage surges due to electrical faults or lightning strikes, and provides a proper ground reference for the instrumentation signals. The NI lightning protection system provides protection of exposed buildings, structures, stacks, substations, and associated electrical and electronic circuits from hazards due to transient over-voltage caused by lightning strikes and switching surges.

#### 1.1.4.3.12 Nuclear Island Plant Communication System

The NI plant communication system is part of the plant-wide Plant Communication System (PCO) and, provides internal and external communications during normal and emergency plant operations, as well as during a fire. The NI communication system is designed to provide an independent means of effective onsite and offsite communication during all modes of plant operation, including during abnormal and emergency operating conditions.Each subsystem is designed to provide an independent mode of communications and is not interdependent with the other subsystems. The subsystems use a converged network architecture to migrate multiple communication services into a single network for transmission. The diverse communication systems are used to assure effective communications, including provisions to accommodate usage in areas exposed to high ambient noise. The NI communication equipment is designed, fabricated, erected, constructed, and routinely tested in accordance with standard industry codes and standards. The NI communication system provides no credited safety function during a plant transient or accident and its failure or degradation does not affect any other plant system functions or have an impact upon plant safety.

The NI communication system consists of subsystems each having emergency communication capabilities. Additional details of emergency communication capabilities will be provided at the operating license stage as described in Section 11.3.

Additional details of security communication functions and cyber security controls for the NI communications system will be provided at the OL stage as described in Section 11.6.


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#### 1.1.4.3.13 Nuclear Island Cranes and Hoists

The NI Cranes and Hoists (NCH) is comprised of bridge cranes and monorail hoists in the NI (i.e., RXB, FHB, RAB). The NCH primary function is to lift and move approved loads within the NI buildings to support normal operations, maintenance, and other required activities.

The RXB and FHB bridge cranes lift and move critical heavy loads and are designed to be single failure proof in accordance with ASME NOG-1-2020, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)” (Reference 1.1-2). Elements of a heavy load program, including safe load handling paths and loads to be lifted, will be provided at the operating license stage. Monorail hoists in the FHB and RAB do not handle loads that pose a risk of adversely affecting a safe shutdown or releasing radioactivity and are not required to be single failure proof.

#### 1.1.4.3.14 Energy Island Auxiliary Electrical System

The EI auxiliary electrical system includes the EI AC Electrical Power Medium Voltage System (EMV), EI AC Electrical Power Low Voltage System (ELV), EI DC power supply system (EDC), EI Uninterruptible AC Power Supply System (EUP), and EI Heat Tracing System (EHT). Under normal operating conditions, the EI auxiliary electrical system supplies power to EI equipment required for plant startup, normal operation, and shutdown. This power is distributed through EMV, ELV, EDC, EUP, and EHT. Power for plant auxiliary loads in the EI is provided normally through the unit auxiliary transformer at the 6.9 kV level. A reserve auxiliary transformer is provided as an alternate source of power in case of failure of unit auxiliary transformer.

During a loss of offsite power event, the SDGs are automatically started and provide backup power to NMV for plant-specific loads and the EMV for plant-specific loads for investment protection. SDGs also supply power to NI and EI battery chargers, and heat tracing to avoid freezing. The SDGs are part of the ESP.

#### 1.1.4.3.15 Nuclear Island Air and Inert Gas Distribution Systems

The NI Air and Gas Distribution System (NGS) primarily includes argon gas distribution and storage systems. The argon gas distribution and storage systems distribute argon as the cover gas for sodium containing systems.

#### 1.1.4.3.16 Seismic Monitoring System

The SMS monitors the vibratory ground motion and resultant vibratory responses of representative structures so that plant personnel can take prompt action to assess the effects of an earthquake at the Kemmerer Unit 1 site. The SMS equipment is designed, built, and tested in full conformance with Regulatory Guide 1.12, “Nuclear Power Plant Instrumentation for Earthquakes,” Revision 3.

The SMS provides time-history acceleration data on seismic response in the free-field and safety-related structures. The following data is made available to plant personnel:

- Time history of acceleration


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- Derived time histories of velocity and displacement
- Derived response spectral accelerations, velocities, and displacements
- Indication of whether a specified peak acceleration has been exceeded
- Indication of whether any specified response spectral value has been exceeded
- Peak acceleration
- Standardized cumulative absolute velocity

The SMS consists of acceleration sensors that input data to a central controller cabinet for display. A hardwired connection provides an alarm input from the central controller cabinet to the UMC. The SMS contains a data recorder that is capable of receiving, processing, and storing signals from the acceleration sensors.

Acceleration sensors are responsible for monitoring seismic activity at specific locations throughout the plant. The sensors are triaxial and measure seismic motion in three orthogonal directions. Acceleration sensors are located in the NCB, RXB, FHB, and NI yard areas. The NCB is described in Section 7.8.4. The RXB is described in Section 7.8.1. The FHB is described in Section 7.8.2.

The SMS provide the data necessary to directly compare the measured motion with both site-specific ground motion response spectra and the design in-structure response spectra to evaluate the seismic response of nuclear power plant features important to safety promptly after and earthquake as required by 10 CFR 50 Appendix S. Seismic response spectra are described in Section 2.6.2.

The SMS is designed with the capabilities to perform channel checks, channel functional tests, and channel calibrations to ensure the functionality of the system.

#### 1.1.4.3.17 Nuclear Island Emergency Operating Lighting

NI emergency operating lighting provides illumination in NI plant areas during an station blackout (SBO) and consists of various components, including lighting distribution panels, lighting cables, lighting fixtures, and emergency battery pack lighting units. The NI emergency operating lighting equipment and components are designed in accordance with applicable industry standards and are not subject to any additional equipment qualification requirements. Routine tests and inspections of NI emergency operating lighting equipment are conducted to verify proper operation.

NI emergency operating lighting provides lighting to support fire suppression actions in the MCR and RSC including access and egress paths between the MCR and RSC. Emergency operating lights located in pathways between the MCR and RSC are DC self-contained, battery operated, lighting units sized to support plant recovery tasks.

NI emergency operating lighting serves no safety function and has no safety design basis. Lighting components located in the MCR, RSC, or in the proximity of safety-significant components include seismic interaction supports to mitigate adverse impacts during a seismic


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event that could reduce the functional reliability of safety-significant equipment or result in injury to MCR or RSC occupants. The NI emergency operating lighting is available during all modes of plant operation.

#### 1.1.4.3.18 Nuclear Island Fire Protection System

The primary purpose of the NI Fire Protection System (NFP) is to mitigate the effects of fire within and around the NI facilities. The NFP incorporates dedicated systems and equipment located within the NI facilities and associated yard areas.The NFP consists of the NI fire water distribution system, the NI fire suppression system, the NI fire detection and alarm system, and the sodium leak detection, collection, and containment system. The NFP conforms to Regulatory Guide 1.189, Revision 5, “Fire Protection for Nuclear Power Plants,” which complies with regulatory requirements of 10 CFR 50.48(a).

### 1.1.4.4 Major Structures

#### 1.1.4.4.1 Nuclear Island Major Structures

The Kemmerer Unit 1 plant arrangement includes five major buildings on the NI and other supporting facilities as shown on the site layout plan in Figure 1.2-1. A brief description of the primary NI structures is provided below.

#### 1.1.4.4.2 Reactor Building

The RXB is the center point of the NI and houses the reactor and associated SSCs that support reactor operation. The RXB supports the ducts and stacks for the passive RAC. The RXB is divided into two independent structures, the RXB superstructure and the RXB substructure. The above-grade RXB superstructure is a rectangular-shaped steel-framed structure with metal siding that has a symmetrical single room layout centered on the reactor. The RXB superstructure is supported by structural steel columns founded on concrete foundations that are decoupled from the below-grade reinforced concrete and steel RXB substructure which has a separate foundation.

The refueling access area floor is located at-grade in the RXB superstructure and the operating deck, or HAA, is located below grade in the RXB substructure. An overhead bridge crane, supported by the RXB superstructure, is used to support various maintenance activities that are conducted on the refueling access area floor. The RXB superstructure steel framing includes a door opening on its east wall for transport of the Ex-Vessel Handling Machine during refueling and other maintenance-related operations.

Below the refueling access area floor is the HAA, which provides maintenance access to the reactor head and its associated piping and equipment. The RES, located below the HAA, encloses the reactor and associated safety-significant systems in the underground RXB substructure that provides protection from external hazards and has subgrade waterproofing. Located on each side of the RXB are the RAC intake and discharge stacks. The RXB houses the following systems, or portions of systems:

- NI Nuclear Heat Supply Systems (RCC, RES, PHT, IHT)


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- NI Reactor Auxiliary Systems (RAC, SCG, CRD)
- NI Fuel Handling Systems (FHE, FHI)
- NI Electrical Systems (NMV, NLV, NHT, NRW, NLS and NGL)
- NI Support Systems (NCH, NHV, NFP, PCO)

The materials used for construction of the RXB, including the RXB steel superstructure and materials for the superstructure concrete foundation are described in Section 6.4.2.4. There are no special construction techniques for the RXB superstructure or substructure.

Sectional views of the RXB are provided in Figure 1.1-5 and Figure 1.1-6.

#### 1.1.4.4.3 Fuel Handling Building

The FHB is an at-grade, steel-framed superstructure with a reinforced concrete slab at-grade and a below grade reinforced concrete substructure. The FHB houses plant operations and equipment related to fuel receipt, refueling operations, storage, and the SFP, along with equipment associated with the RWL, RWG, and RWS systems. An at-grade rail system that starts inside the FHB connects the FHB to the RXB to transport fuel using the EVHM. The rail is supported by a grade slab and enclosed in the FHB-RXB corridor. An overhead bridge crane, supported by the FHB superstructure, is used to move dry fuel casks, spent fuel casks, new fuel assemblies, and other equipment located in the FHB. The primary systems located in the FHB include:

- NI Reactor Auxiliary Systems (SCG, SPS)
- NI Fuel Handling Systems (FHE, FHP)
- NI Electrical Systems (NMV, NLV, NHT, NRW, NLS, and NGL)
- NI Support Systems (NCH, NGS, NHV, NFP, PCO)
- NI Radwaste Systems (RWG, RWL, RWS)

Sectional views of the FHB are provided in Figure 1.1-7 through Figure 1.1-9.

#### 1.1.4.4.4 Reactor Auxiliary Building

The RAB is comprised of two reinforced concrete substructure levels and a steel-framed superstructure that houses critical functions and processes that support the IHT, SHXs, ISPs, and SPS. IAC AHXs are located outside the RAB to support functions inside the RAB. The RAB substructure houses the NSS and IHT drain tanks, and associated piping systems, equipment and components for these systems, along with components for reactor sodium fill and impurity monitoring on the first underground level. The second underground level contains the supplemental tank for the IHT low point drain. The RAB superstructure has a main floor at ground level supporting SHXs, two equipment platform levels that support SCG equipment and the ISPs and access platforms in the upper level, and NGS equipment, SPS sampling skids, and IHT expansion tanks in the lower level. The primary systems located in the RAB include:

- NI Nuclear Heat Supply Systems (IHT, PHT, NSS)
- NI Reactor Auxiliary Systems (IAC, SCG, SPS)


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- NI Electrical Systems (NMV, NLV, NHT, NRW, NLS, and NGL)
- NI Support Systems (NCH, NHV, NFP, NGS, PCO)

Sectional views of the RAB are provided in Figure 1.1-10 and Figure 1.1-11.

1.1.4.4.5 Nuclear Island Control Building

The NCB serves as the primary control center for the NI that houses the MCR and associated equipment to ensure safe plant operations. The NCB is designed as a steel frame one-story superstructure above grade with a reinforced concrete and steel substructure below grade that houses safety-significant systems and equipment, including the RPS vaults and battery racks. The RSC is also contained in the substructure. The NCB superstructure houses the MCR and related equipment and computer rooms, offices, and, lavatories, located within an MCR envelope that is surrounded by a reinforced enclosure that provides required protection from external hazards, radiation shielding, and fire protection. The primary systems located in the NCB include:

- NI Integrated Control Systems (NIC, UMC, PMC, RPS, XIS, RIS, RMS, SMS)
- NI Electrical Systems (NMV, NLV, NRW, NLS, and NGL)
- NI Support Systems (NHV, NFP, PCO)

Plan and sectional views of the NCB are provided in Figure 1.1-12 through Figure 1.1-15.

1.1.4.4.6 Fuel Auxiliary Building

The Fuel Auxiliary Building (FAB) is a steel-braced frame structure supported on a concrete grade slab that provides a centralized location for employee and visitor ingress to, and egress from the major NI facilities (RXB, FHB, RAB). The FAB serves as the main personnel entry point into the RCA for personnel monitoring, radiation control, decontamination, and portable hot equipment maintenance. Other areas in the FAB include non-RCA areas for personnel office space and conference rooms. The FAB also includes a small below-grade pit that contains RWL equipment to support decontamination shower and laundry functions that connect to the FHB RWL. Primary systems located in the FAB include:

- NI Electrical Systems (NGL, NLS, NLV, PCO)
- NI Support Systems (NHV, NFP, NWS)
- NI Radwaste System (RWL)

Plan and sectional views of the FAB are provided in Figure 1.1-16 and Figure 1.1-17.

1.1.4.4.7 Support Buildings and Facilities

Kemmerer Unit 1 support buildings and facilities consist of NI electrical equipment modules (E-modules) and the EI facilities. A brief description of these structures is provided below.


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#### 1.1.4.4.8 Nuclear Island E-Modules

Prefabricated E-Modules are located on the NI area to contain electrical systems and components of the electrical systems. These modules are designed to withstand applicable external weather conditions (e.g., wind, snow, rain). Appropriate NI electrical system equipment are included in the modules to support electrical system operations.

#### 1.1.4.4.9 Energy Island Facilities

The EI facilities support thermal energy storage and steam generation plant operations that are independent from reactor power operations due to the TSS as described in NATD-LIC-RPRT-0001. The independence of plant operations between the EI SSCs and the NI SSCs is a key aspect of the plant design philosophy. The NI boundary conditions have been intentionally designed so the interrelationship with the EI does not impact the NI safety analysis. Thus, the EI facilities provide appropriate housing for EI systems and components used to supply energy to the grid while maintaining isolation from any events impacting NI safety functions associated with reactivity control, core cooling, or containing the release of radioactivity. The EI contains facilities as shown on the site layout plan in Figure 1.2-1.

## References

1.1-1 TerraPower, LLC, “Regulatory Management of Natrium Nuclear Island and Energy Island Design Interfaces,” NATD-LIC-RPRT-0001-A, Revision 0, Accession No. ML24011A321.

1.1-2 ASME NOG-1, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” ASME, 2020.


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**Figure 1.1-1 Heat Transport Systems Overview Highlighting Operational Interface**

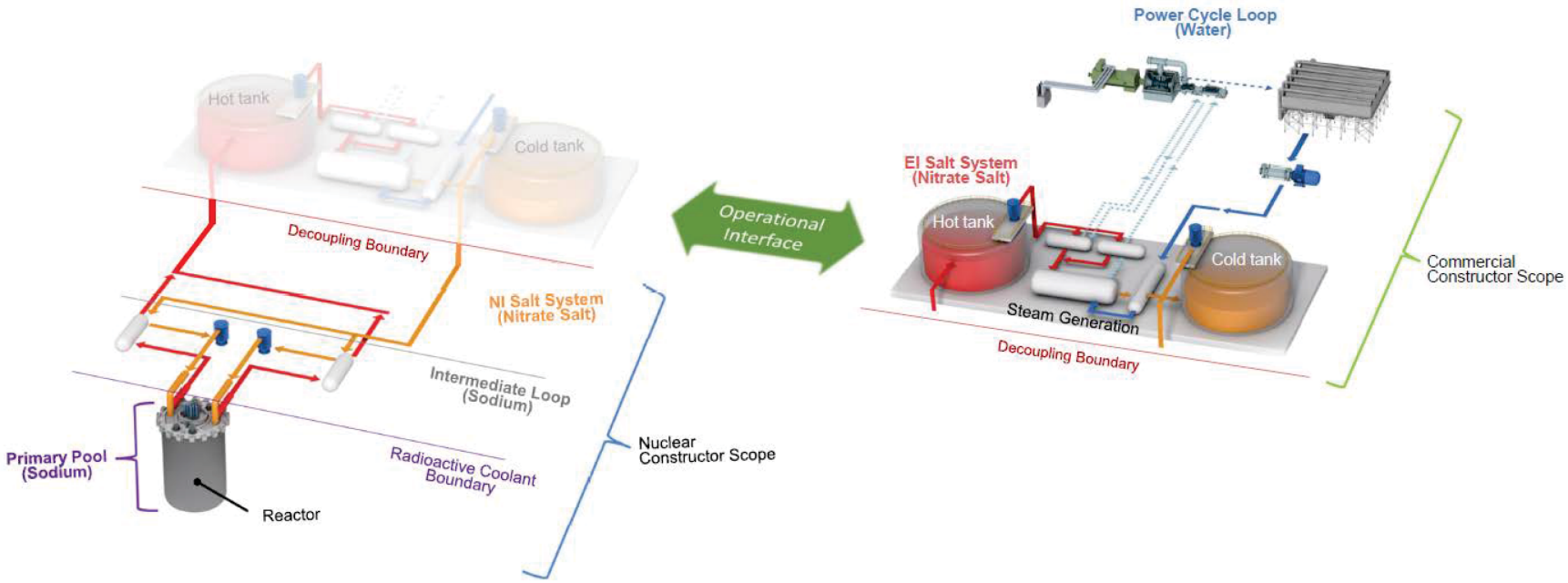



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**Figure 1.1-2 Plant Layout**

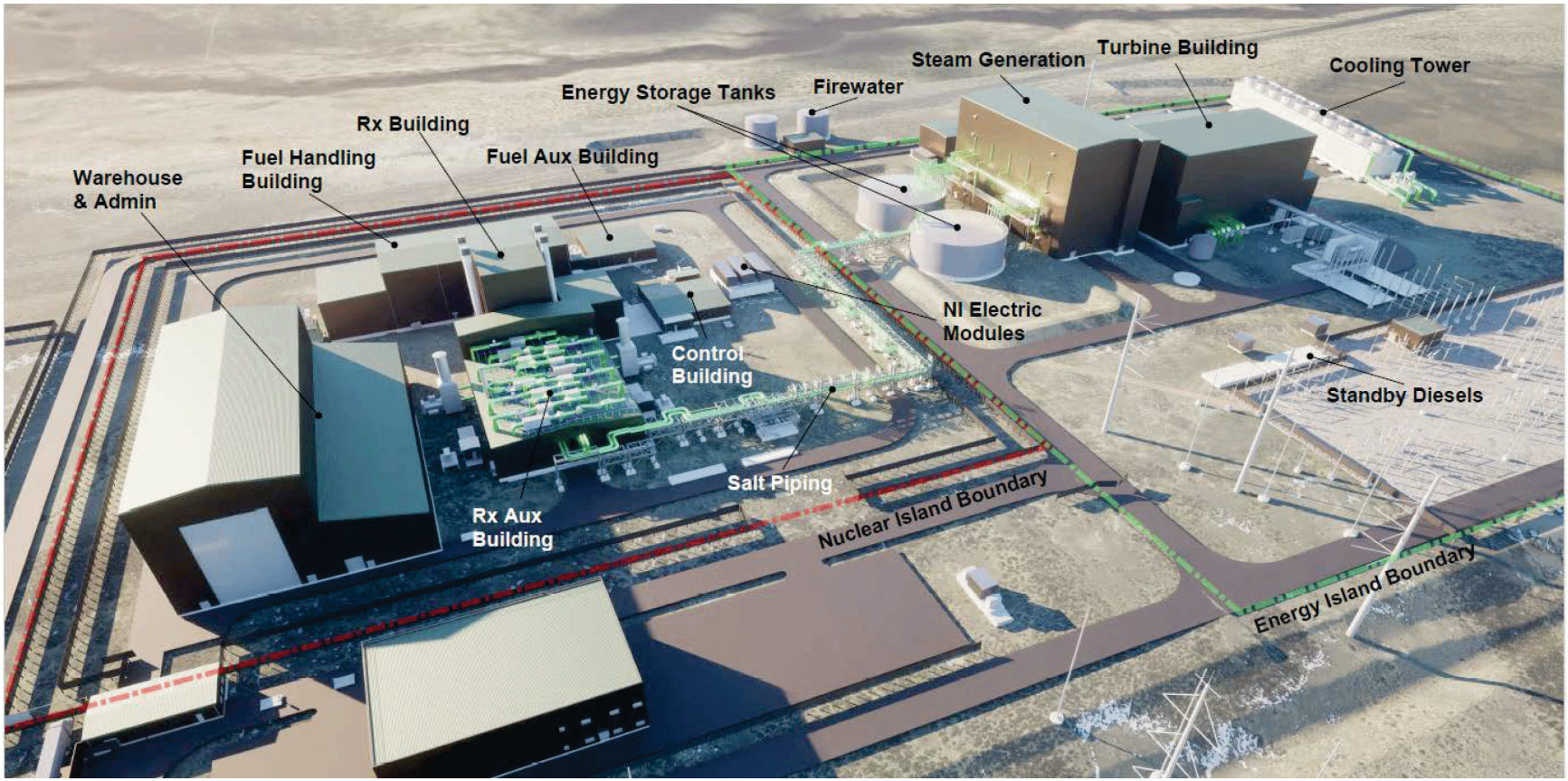



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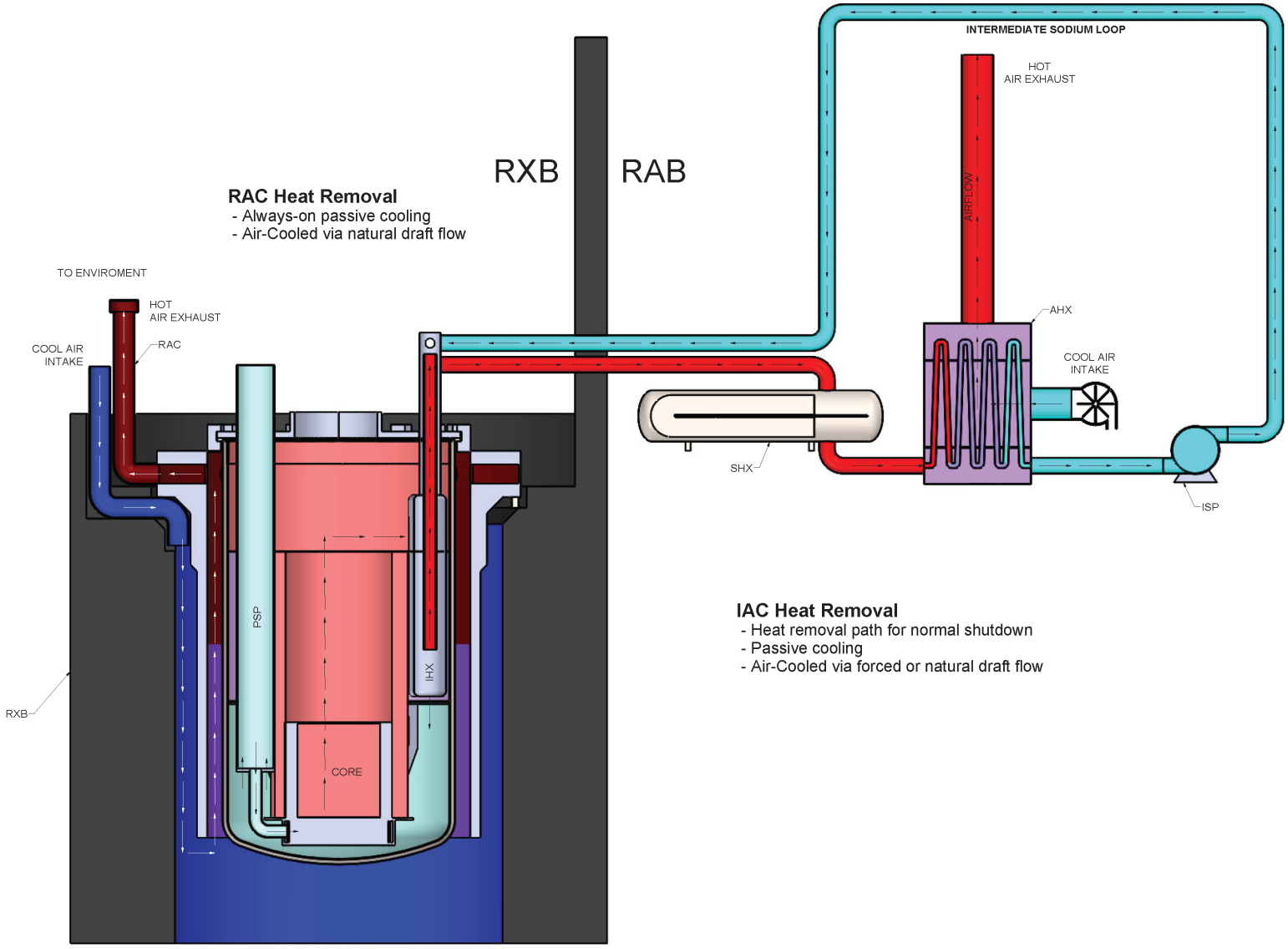
**Figure 1.1-3 Heat Removal**





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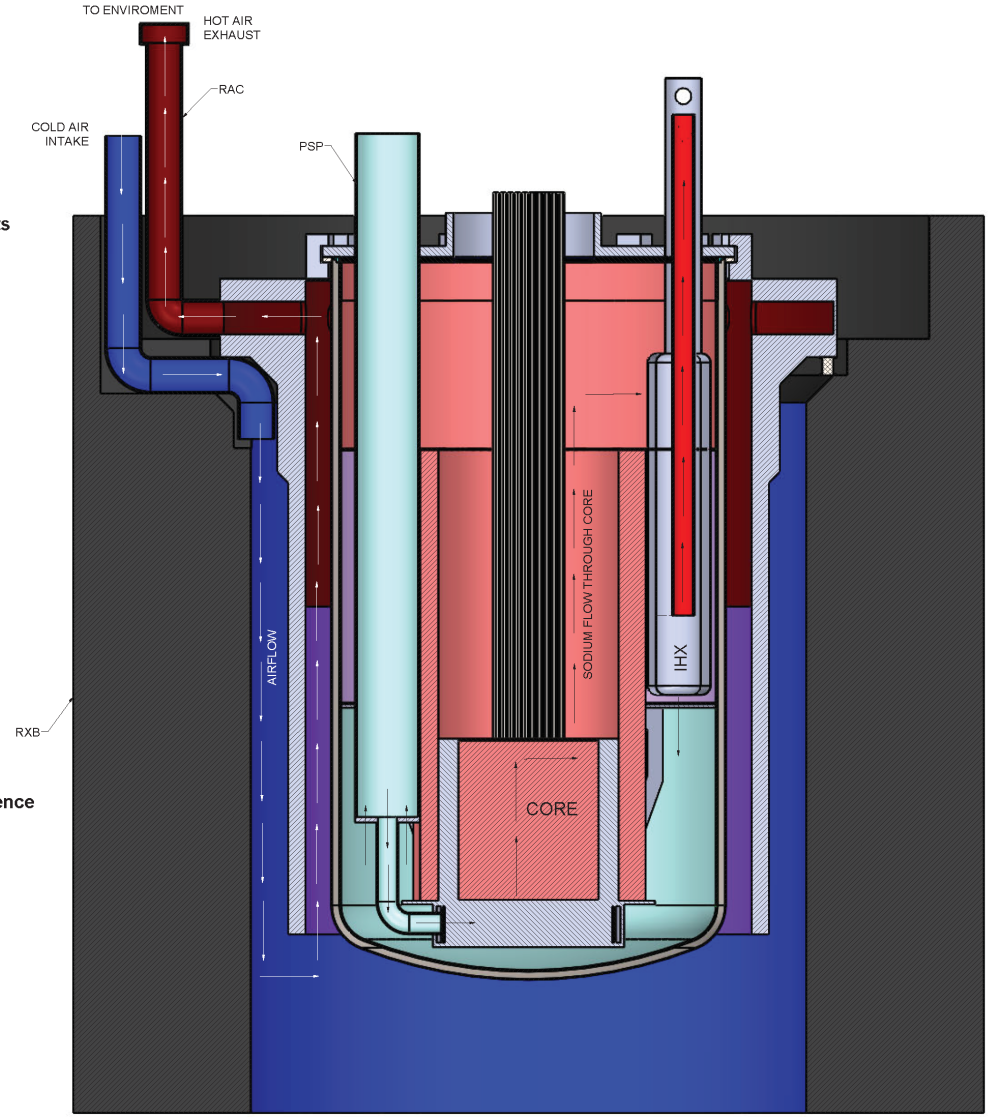
## Figure 1.1-4 Safety Features

**Simplified Response to Abnormal Events**
- Reliable reactor shutdown
- Passive emergency decay heat removal
- Near-atmospheric pressure functional containment
- No reliance on Energy Island for safety functions

**No Safety-Related Operator Actions or Safety-Related Power Systems**

**Technology Based on U.S. SFR Experience**
- SFR inherent safety characteristics demonstrated through testing in EBR-II and FFTF




**Control**
- Motor-driven control rod runback
- Gravity-driven control rod scram
- Inherently stable with increased power or temperature

**Cool**
- In-vessel primary sodium heat transport
- Intermediate air cooling natural draft flow
- Reactor air cooling natural draft flow – always on

**Contain**
- Low primary and secondary pressure
- Sodium affinity for radionuclides
- Multiple radionuclide retention boundaries


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**Figure 1.1-5 General Arrangement RXB Section A Looking East**

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**Figure 1.1-6 General Arrangement RXB Section B Looking North**

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**Figure 1.1-7 General Arrangement FHB Section A Looking West**

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**Figure 1.1-8 General Arrangement FHB Section B Looking North**

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**Figure 1.1-9 General Arrangement FHB Section C Looking North**

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**Figure 1.1-10 General Arrangement RAB Section A Looking North**

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**Figure 1.1-11 General Arrangement RAB Section B Looking East**

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**Figure 1.1-12 General Arrangement NCB Substructure**

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**Figure 1.1-13 General Arrangement NCB Ground Floor**

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**Figure 1.1-14 General Arrangement NCB Looking West**

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**Figure 1.1-15 General Arrangement NCB Looking South**

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**Figure 1.1-16 General Arrangement FAB Ground Floor**

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**Figure 1.1-17 General Arrangement FAB Looking West**

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## 1.2 Site Description

The Kemmerer Unit 1 site is located in the Elkol, Wyoming, 7.5-minute quadrangle in Zone 12T in Lincoln County (Reference 1.2-1). U.S. SFR Owner, LLC owns a 334 acre (135 hectare) property within which the approximately 290 acre (117 hectare) site is located.

Structures and facilities, parking lots, and roads related to the site are identified in Figure 1.2-1. Major plant structures are described in Section 1.1.4.4. The Reactor Building center point is located at the geographical coordinates shown in Table 2.2-1. Orientation of Kemmerer Unit 1 is such that “plant north” faces true north. Plant elevation 500 feet-0 inches equals vertical elevation 6,758 feet-0 inches NAVD88. Additional information regarding the location and exclusion, control, and site area maps are provided in Section 2.2.

Sections 2.1 through 2.8 provide information on the demographic, meteorological, hydrological, geological, and seismological characteristics of the site and the surrounding area. Table 2.1-1 identifies the site-specific external hazards and Table 2.1-2 provides a listing of site-related characteristics.

### References

1.2-1 U.S Geological Survey, Elkol Quadrangle, Wyoming-Lincoln County, 7.5-Minute Series, 2021.



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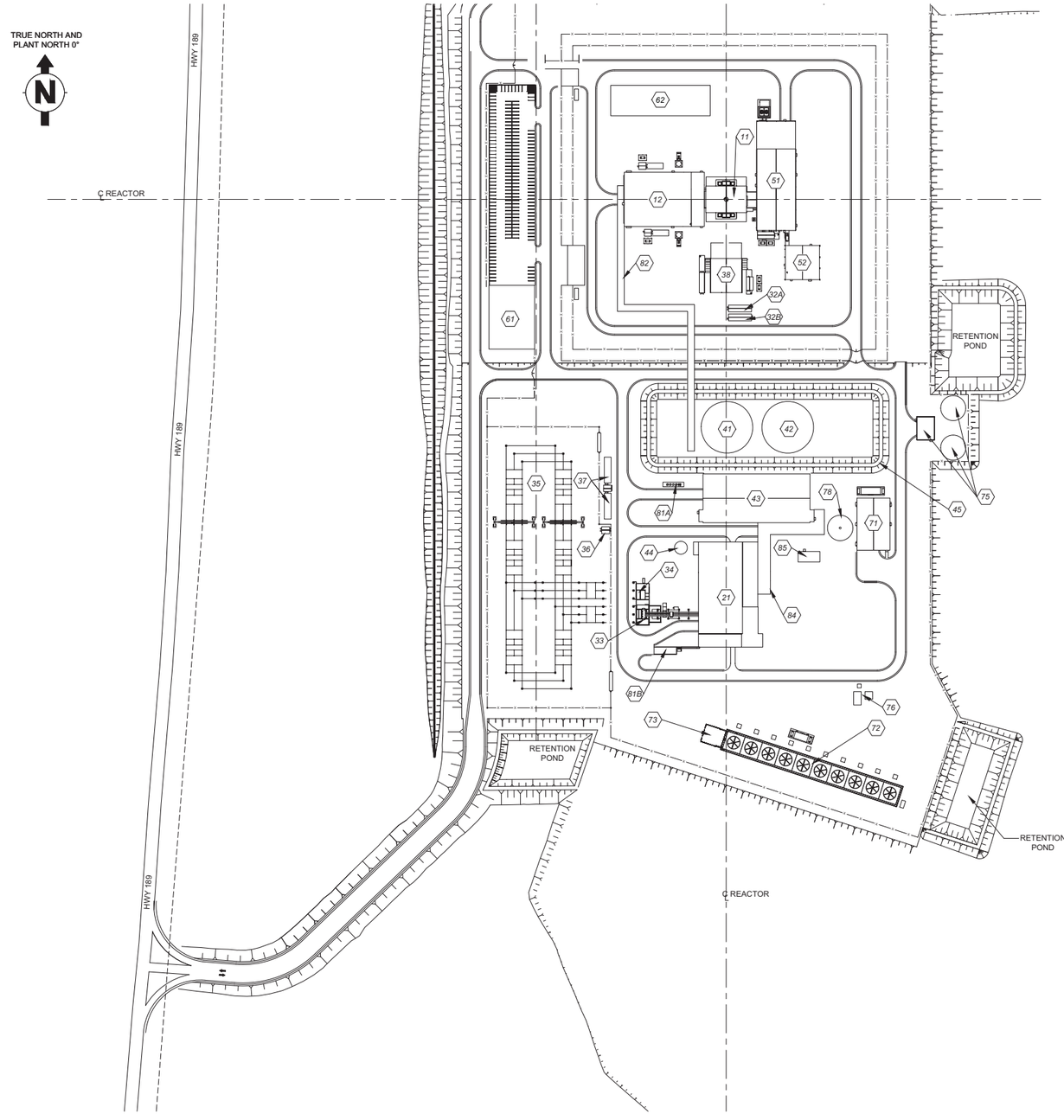
## Figure 1.2-1 Site Layout




**FACILITY NUMBER/DESCRIPTION:**

11 REACTOR BUILDING
12 REACTOR AUXILIARY BUILDING
21 TURBINE FACILITY BUILDING
32A MEDIUM VOLTAGE HOUSING
32B MEDIUM VOLTAGE HOUSING
33 MAIN AND AUXILIARY TRANSFORMERS
34 RESERVE AUXILIARY TRANSFORMERS
35 SWITCHYARD
36 SWITCHYARD CONTROL BUILDING
37 STANDBY DIESEL GENERATORS
38 NI CONTROL BUILDING
41 ENERGY STORAGE STRUCTURE SALT TANK HOT
42 ENERGY STORAGE STRUCTURE SALT TANK COLD
43 STEAM GENERATOR BUILDING
44 CONDENSATE STORAGE TANK
45 SALT CONTAINMENT BERM
51 FUEL HANDLING BUILDING
52 FUEL AUXILIARY BUILDING
61 TRAINING, ADMINISTRATION BUILDING
62 NI WAREHOUSE
71 WATER TREATMENT BUILDING
72 COOLING TOWER
73 CIRCULATING WATER PUMP AREA INTAKE FACILITY
75 FIRE TANKS AND PUMP BUILDING
76 WASTEWATER TREATMENT
78 EI DEMINERALIZED WATER TANK
81A NI ARGON AND NITROGEN BULK STORAGE
81B HYDROGEN GAS TRAILER
82 NI/EI (SALT) PIPE RACK
84 STEAM GENERATOR BUILDING/TURBINE FACILITY BUILDING (STEAM) RACK
85 AUXILIARY BOILER

**LEGEND:**

FENCE
FACILITY NUMBER

SCALE (FEET)
0 120 240 360 480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# 1.3 Plant Safety Overview

## 1.3.1 Licensing Basis Methodology

The selection of licensing basis events (LBEs); safety classification of structures, systems, or components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy were done in accordance with the methodology of Nuclear Energy Institute (NEI) Report NEI 18-04, “Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Revision 1 (Reference 1.3-1), as endorsed by Regulatory Guide 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” Revision 0. This is demonstrated in Chapters 3 through 8.

There were no deviations from the endorsed methodology or the staff positions in the regulatory guide.

An integrated decision-making process panel (IDPP), consistent with NEI 18-04, is established as a standing, multi-disciplinary panel focused on ensuring an effective, efficient, and safe design. The IDPP uses risk and performance information in the review and approval of LBEs, SSC safety classification, and DID adequacy.

## 1.3.2 Fundamental Safety Functions

Fundamental Safety Functions (FSFs) ensure the cumulative risk objectives of regulatory dose criteria and Quantitative Health Objectives are met. The cumulative risk objectives are included in the scope of the probabilistic risk assessment and are described in Section 4.1. The FSFs are retaining radionuclides, controlling heat generation, and controlling heat removal.

The Defense Line (DL) strategy provides independent and redundant mitigation strategies that ensure the FSFs are satisfied. Five DLs provide protection against unacceptable releases of radiation. The second, third, and fourth DLs include the design functions necessary to ensure performance of the FSFs. The DLs are described in Section 4.2.1.

Specified acceptable system radionuclide release design limits (SARRDLs) are established to ensure regulatory limits for offsite doses are not exceeded for any LBE. A SARRDL is a prescribed limit of radionuclide release for systems and components. SARRDLs are established so that:

- The consequences of the most limiting LBE do not exceed the siting regulatory dose limits criteria at the exclusion area boundary and low population zone.
- The 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the exclusion area boundary for normal operation and anticipated operational occurrences.

Compliance with the SARRDLs is achieved through monitoring and controlling radionuclide inventory during normal operation so that the assumed radionuclide inventory at the onset of an LBE is not exceeded. SARRDLs are established from the most-limiting event sequence inclusive of a maximum assumed radionuclide inventory (i.e., the initial radionuclide inventory at the onset


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of an LBE). SARRDL values, such as volumetric system quantities of known or bounding radionuclide concentrations, are defined for systems that contain the assumed radionuclide inventory at the onset of an LBE using system-specific methodologies defined in administrative procedures. Transient and accident analysis are used to demonstrate that SARRDLs are maintained for anticipated operational occurrences.

The exclusion area boundary and low population zone are described in Section 2.2.2. LBEs are described in Section 3.4.

The FSF of retaining radionuclides is accomplished using a functional containment strategy and is described in Section 1.3.2.1.

The FSF of controlling heat generation is performed via insertion of control rods into the reactor core and is described in Section 1.3.2.2.

The FSF of controlling heat removal is achieved through active and passive cooling systems and is described in Section 1.3.2.3.

#### 1.3.2.1 Retaining Radionuclides

The FSF of retaining radionuclides is performed using a functional containment strategy that employs diverse passive barriers. The primary goals of the functional containment strategy are to:

- Maintain at least one barrier beyond the cladding, piping, or vessel containing a radionuclide source that can reliably withstand LBE and design basis accident conditions and whose leakage is specified by design requirements or testing.
- Keep a minimum total number of required barriers between a radionuclide source and the environment to meet the established performance requirements.
- Provide a framework for necessary iterations between safety analysis, design, and probabilistic risk assessment to optimize the selection and classification of radionuclide retention barriers for LBEs and design basis accidents.

The diverse passive barriers that are used in the functional containment strategy begin at a radionuclide source and include all SSCs between that source and the environment. Figure 1.3-1 depicts an example of barriers used in the functional containment strategy.

For the fuel, the cladding provides the first safety-related (SR) radionuclide retention barrier. The FSF of controlling heat generation, described in Section 1.3.2.2, and the FSF of controlling heat removal, described in Section 1.3.2.3, are focused on maintaining cladding integrity. If a fuel assembly fails during operation, it is sealed in an SR canister prior to storage in the Spent Fuel Pool.

The Reactor Enclosure System provides the next radionuclide retention barrier following the fuel cladding. The SR portion of this boundary includes the reactor head, reactor vessel, and associated component interfaces. The guard vessel, also a component of the Reactor Enclosure System, provides an additional barrier outside of the reactor head and reactor vessel SR boundary. Concurrent leakage of both vessels is not a postulated LBE once fuel is loaded in the


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core, and there are no penetrations through either of the two vessels. The primary sodium coolant is blanketed by sodium cover gas, providing separation between the primary sodium coolant and reactor head. The Reactor Enclosure System is described in Section 7.1.2.

When irradiated fuel is not located within the reactor vessel, the fuel cladding continues to provide the first SR radionuclide retention barrier. If cladding has been breached, the fuel handling equipment and associated SSCs provide the next fission product barrier.

Functional containment barriers are also provided for systems containing radionuclides not retained within the fuel. These include systems that house and process activation products and possibly contain fission products that have leaked from failed fuel into the primary sodium coolant. Systems designed to contain radionuclides that travel beyond the primary coolant boundary include the Sodium Cover Gas System described in Section 7.2.3, Sodium Processing System described in Section 7.2.4, and Gaseous Radwaste Processing System described in Section 7.4.1.

In addition to the physical barriers, inherent features of the plant design help ensure regulatory dose criteria and Quantitative Health Objectives are met. SARRDLs are established to ensure regulatory limits for offsite doses are not exceeded as described in Section 1.3.2.

An additional inherent feature of the plant design supporting the functional containment strategy includes the high boiling point of sodium, which ensures that the reactor core remains covered by subcooled sodium at near atmospheric pressure during LBEs. The low operating pressure of the primary system and sodium cover gas ensures there is no significant driving force to transport radionuclides. The intermediate coolant pressure is greater than primary coolant pressure, therefore postulated leaks of intermediate coolant result in an ingress of sodium to the primary system rather than a release of primary coolant to the intermediate system.

The following SSCs are associated with the functional containment strategy:

- Metallic fuel matrix and fuel cladding described in Section 7.1.1
- Reactor Enclosure System, Head Access Area, and primary coolant boundary described in Section 7.1.2
- Sodium Processing System described in Section 7.2.4
- Sodium Cover Gas System described in Section 7.2.3
- Intermediate Heat Transport System (IHT) described in Section 7.1.4
- Reactor Building described in Section 7.8.1
- Reactor Auxiliary Building described in Section 7.8.3
- Fuel Handling Building described in Section 7.8.2
- Water Pool Fuel Handling System described in Section 7.3.1
- Ex-Vessel Fuel Handling System described in Section 7.3.2
- In-Vessel Fuel Handling System described in Section 7.3.3
- Nuclear Island Heating, Ventilation, and Air Conditioning System described in Section 7.5.1
- Gaseous Radwaste Processing System described in Section 7.4.1


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#### 1.3.2.2 Controlling Heat Generation

The FSF of controlling heat generation is performed by inserting control rods into the reactor core and has active, passive, and inherent means of being achieved across the DLs. Control rods are positioned in the reactor core by the Control Rod Drive Mechanism System (CRD), which uses two diverse means of inserting two control rod banks of diverse geometrical design. Control rods are positioned in the core for reactivity control during power operation, are inserted into the core as required in response to abnormal events, and maintain the reactor shutdown. The CRD is described in Section 7.2.5.

The power runback feature is an active means of achieving the FSF of controlling heat generation and prevents a scram for a variety of upset conditions during power operation. The power runback is performed by the CRD and Nuclear Island Rod Monitoring and Control System, which use control rod drive motors to insert control rods upon detection of abnormal conditions. The Nuclear Island Rod Monitoring and Control System is described in Section 7.6.2.

The scram function results in passive gravity insertion of control rods into the reactor core to achieve the FSF of controlling heat generation. The scram is performed by the CRD and Reactor Protection System. The Reactor Protection System is described in Section 7.6.3.

The CRD driveline scram follow feature provides DID to the scram function. The CRD driveline scram follow is initiated by the Reactor Protection System coincident with a scram signal and drives the control rods into the core using the control rod drive motors.

Inherent reactivity feedback of the reactor core also controls heat generation. As temperatures rise, the net negative feedback of the core reduces power. The natural feedbacks are self-regulating, inherent in the fuel design, and result in a safe and stable power level at which heat production and heat removal are in balance. The reactor core design features are described in Section 7.1.1, and the supporting analysis is provided in Chapter 3.

#### 1.3.2.3 Controlling Heat Removal

The FSF of controlling heat removal has active, passive, and inherent means of being achieved across the DLs. Two diverse residual heat removal systems, the Intermediate Air Cooling System (IAC) and Reactor Air Cooling (RAC), perform functions that satisfy the FSF of controlling heat removal. The IAC provides an active and passive means of performing the FSF of controlling heat removal. The RAC provides an inherent means of performing the FSF of controlling heat removal. The flow of sodium through the reactor core to facilitate heat removal by the IAC and RAC is performed by the Primary Heat Transport System, which is described in Section 7.1.3. This includes mechanisms for the natural circulation of sodium under certain off-normal conditions.

The IAC has two active modes of operation that perform the FSF of controlling heat removal by using the IAC blower to transfer heat from the IHT via the Intermediate Heat Exchanger. One active mode of operation removes heat using the IAC blower and forced flow through the IHT, provided by the Intermediate Sodium Pump. The second active mode of operation removes heat using the IAC blower and natural circulation flow in the IHT. The IAC also provides a passive means of performing the FSF of controlling heat removal using natural draft airflow instead of the


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IAC blower. The IAC is the primary means of heat removal during plant start-up, shutdown, and off-normal conditions. The IAC and IHT are described in Section 7.2.2 and Section 7.1.4, respectively.

The RAC provides an inherent means of performing the FSF of controlling heat removal using natural circulation of air around the exterior of the guard vessel. The RAC operates continuously, with no operator actions or moving parts required to perform its function, and removes decay heat during normal operation, anticipated operational occurrences, design basis events, and design basis accidents. Since RAC is an open system, the surrounding environment is both the ultimate coolant source and heat sink for the RAC. Meteorological data for evaluating the ultimate heat sink is provided in Section 2.4.1.4. The RAC is described in Section 7.2.1.

### 1.3.3 Defense-in-Depth

DID is a key element of the licensing basis and the demonstration of reasonable assurance of adequate protection of the health and safety of the public. The three DID elements, plant capability, programmatic capability, and integrated DID adequacy evaluation, establish the baseline DID.

The guidelines for plant capability DID adequacy outlined in NEI 18-04, Section 5.6, including the plant capability attributes provided in NEI 18-04, Table 5-2, are satisfied for each LBE as described in Sections 4.2.1.1 to 4.2.1.4.

The programmatic DID adequacy, including any additional special treatments identified, and the integrated DID adequacy evaluations will be provided at the operating license stage to establish the baseline DID.

**References**

1.3-1 Nuclear Energy Institute (NEI) 18-04, “Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Revision 1, August 2019.


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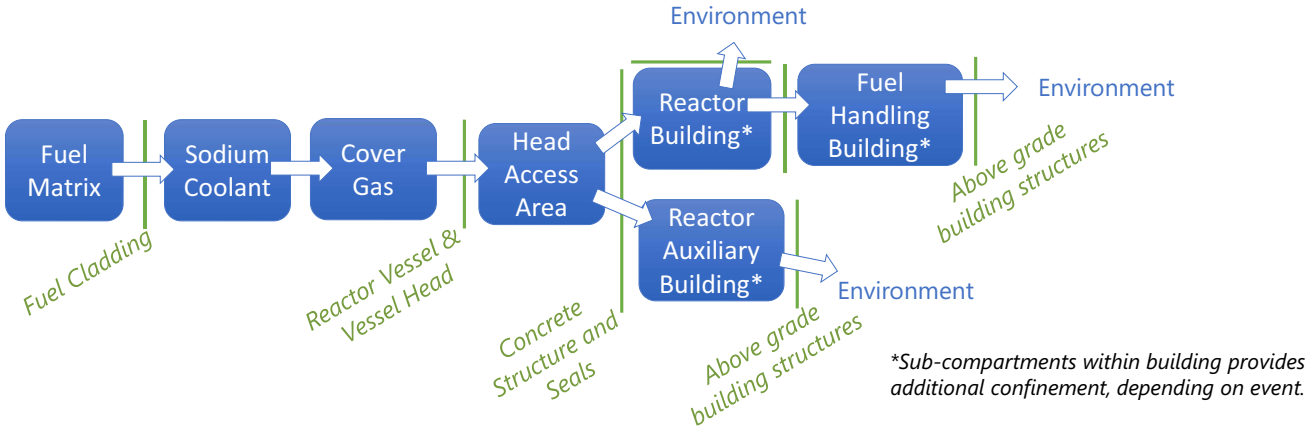
**Figure 1.3-1 Diverse Barriers in Functional Containment**



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# 1.4 Conformance with Regulatory Criteria and Referenced Material

## 1.4.1 Regulatory Guidance

All current Regulatory Guides (RGs) have been reviewed and those that are not applicable to the design have been excluded. Table 1.4-1 provides a summary of the conformance to applicable Nuclear Regulatory Commission RGs in effect 6 months before the application submittal date.

Table 1.4-1 identifies the evaluated RGs by number, title, and the revision being considered. For each RG, the table identifies the design conformance status and any notable comments regarding conformance or applicability. The conformance status for each RG is identified in the following manner:

**Full Conformance** – The regulatory guidance is relevant and applicable, and the design fully conforms to the guidance. Where options are identified in the regulatory guidance, “Conforms” indicates that the design fully conforms to the option(s) selected.

**Partial Conformance** – The guidance is partially relevant and applicable, and the design conforms to those portions of the guidance that are evaluated to be applicable. The underlying purpose or intent of the guidance is relevant to the design but cannot be appropriately applied as written, or some portion of the guidance is applicable while other portions are not applicable. The following are examples:

- A portion of the guidance is literally applicable, but the specific language refers to a different type of reactor design or a structure, system, or component (SSC) that is not part of the design.
- The regulatory guidance is applicable except for aspects that are specific to applicants for other types of licenses (operating license applicants) or to light water reactor designs.
- The intent of a regulatory guidance is applicable, but the specific language refers to one of the following:
    - a different type of reactor design
    - an SSC that is not part of the design, but for which a substantively equivalent function is served by other SSCs within the design

RGs that have been designated applicable but do not have a conformance designation are RGs that are evaluated to be not applicable to the design and construction phase of the project and will be addressed at the operating license (OL) stage.

## 1.4.2 Topical Reports and Technical Reports

The topical and technical reports identified in Table 1.4-2 and Table 1.4-3, respectively, have been submitted to the Nuclear Regulatory Commission for review. If a topical report has received a safety evaluation report, Table 1.4-2 indicates the safety evaluation report accession number. Technical reports identified in Table 1.4-3 are reviewed as part of the application. The referenced section provides additional discussion, including information being incorporated by reference.


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### 1.4.3 Generic Safety Issues, Unresolved Safety Issues and TMI Action Items

To support review of generic issues for a 10 Code of Federal Regulations (CFR) 50 application, the generic issues identified in NUREG-0933 (Reference 1.4-19) were evaluated for applicability and compliance in the design.

The Three Mile Island (TMI) Action Items identified in NUREG-0933, Appendix B, Section 1, are addressed through the codification of these items in 10 CFR 50.34(f). Table 1.4-4 provides a summary of the evaluation of the applicability of these items to the design.

**Full Compliance** – The regulation is relevant and applicable, and the design fully complies. Where options are identified in the regulation, “Full Compliance” indicates that the design fully complies to the option(s) selected.

**Partial Compliance** - The regulation is relevant and applicable, and the underlying purpose or intent of the regulation is relevant to the design but cannot be appropriately applied as written.

**Not Applicable** – The regulation is not appropriate to apply and therefore compliance is not required. The following are examples:

- The guidance is applicable only to light water reactor designs.
- The guidance is applicable to an SSC that is not part of the Natrium design.
- The guidance is only applicable to other types of applications (for example, applications under 10 CFR Part 52).

The remaining generic issues that are identified in NUREG-0933, Resolution of Generic Safety Issues, Appendix B, were also evaluated. There were no items that were identified to be applicable to the design.

### 1.4.4 Codes and Standards

The consensus codes and standards used in the design are listed in Table 1.4-5. The listing of a code or standard does not necessarily mean that it is applied in its entirety. The application of consensus codes and standards, deviations and additional system specific codes and standards are described throughout the PSAR.

## References

1.4-1 Terrapower, “QA Program Description,” TP-QA-PD-0001, Revision 14

1.4-2 U.S. Nuclear Regulatory Commission, “Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America,” NUREG/CR-7046, 2011.

1.4-3 U.S. Nuclear Regulatory Commission, JLD-ISG-2013-01, “Guidance for Assessment of Flooding Hazards Due to Dam Failure,” June 2013.



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1.4-4 U.S. Nuclear Regulatory Commission, DANU-ISG-2022-02, “Advanced Reactor Content of Application Project, Chapter 2, “Site Information” Draft Interim Staff Guidance,” May 2023.

1.4-5 American Nuclear Society, “Determining Design Basis Flooding at Power Reactor Sites,” ANSI/ANS-2.8-1992,1992

1.4-6 IEEE, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018, November 1, 2018.

1.4-7 IEEE/IEC, “International Standard - Nuclear facilities - Electrical equipment important to safety - Qualification,” IEEE/IEC 60780-323-2016, February 19, 2016.

1.4-8 IEEE, “Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” IEEE 323-2003.

1.4-9 IEEE, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497-2016, July 29, 2016

1.4-10 IEEE/IEC, “International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification,” IEEE/IEC 60980-344-2020

1.4-11 IEEE “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” IEEE 338-2012, March 23, 2012.

1.4-12 ISRM (2007). The Complete ISRM Suggested Methods for Rock Characterization, Testing and Monitoring: 1974-2006. ISRM Turkish National Group and the ISRM, 2007.

1.4-13 U.S. Environmental Protection Agency, “Ground-Water Sampling Guidelines for Superfund and RCRA Project Managers,” EPA 542-S-02-001, May 2002.

1.4-14 American National Standards Institute, “Containment Isolation Provisions for Fluid Systems,” ANSI N271-1976, 1976.

1.4-15 IEEE “IEEE Standard for System, Software, and Hardware Verification and Validation,” IEEE 1012-2016, September 29, 2017.

1.4-16 ASME, “Quality Assurance Requirements for Nuclear Facility Applications,” Requirement 11, Test Control, ASME NQA-1, 2015.

1.4-17 NEI “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” 18-04, Revision 1.

1.4-18 Terrapower, “An Analysis of Potential Volcanic Hazards at the Proposed Natrium Site near Kemmerer, Wyoming,” NAT-3226, Revision 0A

1.4-19 U.S. NRC, NUREG-0933, “Resolution of Generic Safety Issues”, (https://www.nrc.gov/sr0933/index.html), December 2011, Accessed August 10, 2023.


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 1 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.8 | Qualification and Training of Personnel for Nuclear Power Plants | 4 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.12 | Nuclear Power Plant Instrumentation for Earthquakes | 3 | Full Conformance |
| 1.13 | Spent Fuel Storage Facility Design Basis | 2 | Partial Conformance<br>General exceptions:<br>1. Although the design is non-light water reactor (LWR), the aspects of the RG are largely applicable to the SFP design.<br>2. References to “Seismic Category I” are not applicable as the design does not adopt RG 1.29 definitions.<br>3. References to “Quality Group C” are considered non-applicable as the design does not adopt the RG 1.26.<br><br>Specific regulatory guidance exceptions:<br>C.2 - Protection Against Extreme Winds<br>C.8 - Makeup Water: Normal Spent Fuel Pool makeup water system is not Seismic Category I. It is non-seismic, non-safety-related with no special treatment<br>C.9 - The design load of the Spent Fuel Pool heat sink is not at least “0.3 percent of the rated reactor thermal power.” |
| 1.20 | Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing | 4 | Partial Conformance<br>This RG is partially applicable because its scope covers LWRs and small modular reactors, however the intent and elements of a comprehensive vibration monitoring program is applicable to the design. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 2 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.21 | Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste | 3 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.22 | Periodic Testing of Protection System Actuation Functions | 0 | Full Conformance |
| 1.23 | Meteorological Monitoring Programs for Nuclear Power Plants | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.24 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure | 0 | Full Conformance |
| 1.28 | Quality Assurance Program Criteria (Design and Construction) | 5 | Full Conformance<br>RG 1.28 Revision 6 is the most recent version but Revision 5 was used in the current Nuclear Regulatory Commission approved version of the Quality Assurance (QA) Program Description Topical Report (Reference 1.4-1). |



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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 3 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.29 | Seismic Design Classification for Nuclear Power Plants | 6 | Partial Conformance<br>The general guidance and intent of RG 1.29 are applicable to the design with the exception that the overall requirements for “seismic Category I” SSCs are instead applied to safety-related (SR) SSCs. This exception is applied in the following ways:<br>C.1 - Provisions for seismic design of seismic Category I SSCs are instead applied to SR SSCs. In addition, the explicit list of SSCs in C.1.a through C.1.h are not taken to be applicable as they are specific to LWR technology; instead SR SSCs are required to be designed to withstand the effects of the SSE and remain functional.<br>C.2 - Provisions for seismic interaction prevention are not applied to items from C.1.a through C.1.h and “occupants of the control room” but are instead applied to SR SSCs.<br>C.3 - Provisions for dynamic analysis requirements at the interface of seismic Category I and non-seismic Category 1 systems are instead applicable to the interface of SR and non-safety-related with special treatment or non-safety-related with no special treatment systems.<br>C.4 - Provisions for quality requirements for seismic Category I SSCs are instead applicable to SR SSCs |
| 1.30 | Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment | 0 | Full Conformance |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 4 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.31 | Control of Ferrite Content in Stainless Steel Weld Metal | 4 | Partial Conformance<br>The design conforms with Reg Guide 1.31 with the exception of regulatory guidance C.4 that when design temperatures exceed 800F, the delta ferrite shall be limited to the range 3 ferrite number to 10 ferrite number in accordance with the ASME Section III, Division 5 Code. |
| 1.33 | Quality Assurance Program Requirements (Operation) | 3 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.34 | Control of Electroslag Weld Properties | 1 | Full Conformance |
| 1.36 | Nonmetallic Thermal Insulation for Austenitic Stainless Steel | 1 | Full Conformance<br>Alternate code or standard will be applied. |
| 1.40 | Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants | 1 | Full Conformance |
| 1.44 | Control of the Processing and Use of Stainless Steel | 1 | Full Conformance |
| 1.45 | Guidance on Monitoring and Responding to Reactor Coolant System Leakage | 1 | Partial Conformance<br>RG applies to LWRs and is not required for the Primary Coolant Boundary Leak Detection Strategy. However, select regulatory guidance in C.1.1, C.1.3, C.1.4, C.2.3, C.2.4 are utilized as reference information and guidance for design. |
| 1.47 | Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems | 1 | Full Conformance |
| 1.50 | Control of Preheat Temperature for Welding of Low-Alloy Steel | 1 | Full Conformance |



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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 5 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.53 | Application of the Single-Failure Criterion to Safety Systems | 2 | Full Conformance |
| 1.59 | Design Basis Floods for Nuclear Power Plants | 2 | Partial Conformance<br>The following exceptions are taken: More current direction from the following documents is used in conjunction with the regulatory guidance:<br>Draft Regulatory Guide DG-1290, “Proposed Revision 3 to Regulatory Guide 1.59, Design Basis Flooding for Nuclear Power Plants”<br>NUREG/CR-7046, (Reference 1.4-2).<br>JLD-ISG-2013-01, (Reference 1.4-3).<br>DANU-ISG-2022-02, (Reference 1.4-4).<br>ANSI/ANS-2.8-1992, (Reference 1.4-5). |
| 1.60 | Design Response Spectra for Seismic Design of Nuclear Power Plants | 2 | Full Conformance |
| 1.61 | Damping Values for Seismic Design of Nuclear Power Plants | 1 | Full Conformance |
| 1.68 | Initial Test Programs for Water-Cooled Nuclear Power Plants | 4 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.68.2 | Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.68.3 | Preoperational Testing of Instrument and Control Air Systems | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 6 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.69 | Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants | 1 | Full Conformance |
| 1.71 | Welder Qualification for Areas of Limited Accessibility | 1 | Full Conformance |
| 1.73 | Qualification Tests for Safety-Related Actuators in Nuclear Power Plants | 1 | Full Conformance |
| 1.75 | Criteria for Independence of Electric Safety Systems | 3 | Partial Conformance<br>The design will conform to the portion of the RG related to independence between safety related and non-safety-related circuits. In addition, the intent of some of the guidance will be used as a special treatment for applicable electric power related non-safety-related with special treatment SSCs. Some systems will use the newer version of IEEE 384-2018<br>(Reference 1.4-6) |
| 1.76 | Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants | 1 | Full Conformance |
| 1.78 | Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release | 2 | Partial Conformance<br>Regulatory guidance in C.4.1, Detection System, is used as guidance for Nuclear Island Heating, Ventilation, and Air Conditioning System instruments and control systems. |
| 1.84 | Design, Fabrication, and Materials Code Case Acceptability, ASME Section III | 39 | Full Conformance |
| 1.87 | Acceptability of ASME Section III, Division 5, “High Temperature Reactors” | 2 | Full Conformance |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 7 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.89 | Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants | 2 | Partial Conformance<br>The design uses IEEE 323-2016 (Reference 1.4-7) as endorsed by RG 1.89, with the exception of instances of qualification of equipment located within a mild radiation environment as defined by Section B of RG 1.89. Equipment located in a mild environment is qualified in accordance with IEEE-323-2003 (Reference 1.4-8) as endorsed by RG 1.209.<br>For qualification of equipment located within a harsh environment as defined in IEEE-323-2003. Equipment located in a harsh environment is qualified by testing, analysis, or a combination of testing and analysis in accordance with IEEE-323-2016 as endorsed by RG 1.89.<br>Both code years are part of the design. |
| 1.91 | Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants | 3 | Full Conformance |
| 1.92 | Combining Modal Responses and Spatial Components in Seismic Response Analysis | 3 | Full Conformance |
| 1.93 | Availability of Electric Power Sources | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 8 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.97 | Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants | 5 | Partial Conformance<br>The design of equipment used for post-accident monitoring is informed by RG 1.97 Revision 5 & IEEE 497-2016 (Reference 1.4-9). The extent of conformance to IEEE 497-2016 as endorsed by Regulatory Guide 1.97 Revision 5 will be provided at the OL stage. |
| 1.100 | Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants | 4 | Partial Conformance<br>SR Mechanical and electrical SSCs with SR Seismic Risk Significant seismic classification will conform to this standard with the exception that some systems will conform to IEEE 344-2020 (Reference 1.4-10). |
| 1.102 | Flood Protection for Nuclear Power Plants | 1 | Full Conformance |
| 1.105 | Setpoints for Safety-Related Instrumentation | 4 | Full Conformance |
| 1.109 | Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I | 1 | Full Conformance |
| 1.110 | Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors | 1 | Full Conformance |
| 1.111 | Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors | 1 | Full Conformance |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 9 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.112 | Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors | 1 | Full Conformance |
| 1.113 | Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I | 1 | Full Conformance |
| 1.114 | Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit | 3 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.115 | Protection Against Turbine Missiles | 2 | Full Conformance |
| 1.117 | Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants | 2 | Partial Conformance<br>RG 1.117 Position C.3 is not applicable to the design. The project uses NEI 18-04 (Reference 1.4-17) to confirm the SSCs that are protected from tornados, which does not follow the dose metric of RG 1.117 Position C.3. |
| 1.118 | Periodic Testing of Electric Power and Protection Systems | 3 | Partial Conformance<br>The industry standards endorsed in this RG will be applied to the extent practical. Exceptions will be identified during detailed design and qualification and will be updated at the OL stage. |
| 1.122 | Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components | 1 | Full Conformance |



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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 10 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.124 | Service Limits and Loading Combinations for Class 1 Linear-Type Supports | 3 | Full Conformance |
| 1.130 | Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Supports | 3 | Full Conformance |
| 1.132 | Geologic and Geotechnical Site Characterization Investigations for Nuclear Power Plants | 3 | Full Conformance |
| 1.134 | Medical Assessment of Licensed Operators or Applicants for Operator Licenses at Nuclear Power Plants | 4 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.138 | Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants | 3 | Partial Conformance<br>The following exceptions are taken:<br>C.3 - A more current standard ASTM C295/ISRM-2007 (Reference 1.4-12) was used.<br>C.5.2.b - Other standards (ISRM (2007), ASTM, and NGI) were used to evaluate rock mass.<br>C.5.3 - More current standards provided by the Environmental Protection Agency 542-S-02-001(Reference 1.4-13) are used.<br>Appendix A - More recent standards are used. |
| 1.140 | Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants | 3 | Full Conformance |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 11 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.141 | Containment Isolation Provisions for Fluid Systems | 1 | Partial Conformance<br>Applies to LWRs; however, design guidance supplements ANSI N271-1976<br>(Reference 1.4-14) for systems that penetrate the primary containment boundary. |
| 1.142 | Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments) | 3 | Full Conformance |
| 1.143 | Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants | 2 | Partial Conformance<br>In lieu of the deterministic classification methodology of the regulatory guidance in C.5, a project risk-specific probabilistic methodology is used as the primary classification vehicle for systems containing radioactive material.<br>For the hazards assessment described in C.6.1, a project-specific probabilistic methodology is used to determine applicable NPH and man-induced hazards.<br>C.5 and C.6.1 are considered as guidance when selecting special treatments applied to system design.<br>For the codes and standards cited by the RG, more recent versions are used for design instead of the versions cited by the RG. |
| 1.145 | Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants | 1 | Full Conformance |
| 1.147 | Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1 | 20 | Partial Conformance<br>The project may choose to implement specific code cases. In those cases, the design will conform to the RG as applicable. |



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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 12 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.149 | Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements | 4 | Full Conformance |
| 1.152 | Criteria for Programmable Digital Devices in Safety-Related Systems of Nuclear Power Plants | 4 | Full Conformance |
| 1.153 | Criteria for Safety Systems | 1 | Full Conformance |
| 1.156 | Qualification of Connection Assemblies for Production and Utilization Facilities | 2 | Full Conformance |
| 1.159 | Assuring the Availability of Funds for Decommissioning Nuclear Reactors | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.160 | Monitoring the Effectiveness of Maintenance at Nuclear Power Plants | 4 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.164 | Dedication of Commercial-Grade Items for Use in Nuclear Power Plants | 0 | Full Conformance |
| 1.166 | Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake | 1 | Full Conformance |
| 1.168 | Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants | 2 | Partial Conformance<br>IEEE 1012-2016 (Reference 1.4-15) is adopted in the design considering IEEE 1012-2016 is compatible with all life cycle models which is more representative of modern instrumentation and control system development compared to older version. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 13 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.169 | Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants | 1 | Partial Conformance<br>The Software Configuration Management Plan conforms with this RG with the exception of processes that are developed and adhere to NQA-1 (Reference 1.4-16), and project specific process and procedures. |
| 1.170 | Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants | 1 | Full Conformance |
| 1.171 | Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants | 1 | Full Conformance |
| 1.173 | Developing Software<br>Life-Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants | 1 | Full Conformance |
| 1.175 | Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.177 | Plant-Specific, Risk-Informed Decisionmaking:<br>Technical Specifications | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.178 | Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping | 2 | Full Conformance |
| 1.179 | Standard Format and Content of License Termination Plans for Nuclear Power Reactors | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |



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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 14 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.180 | Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems | 2 | Partial Conformance<br>The industry standards endorsed in this RG will be applied to the extent practical. Exceptions will be identified during detailed design and qualification and addressed during the OL stage. |
| 1.181 | Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e) | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.183 | Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors | 0 | Partial Conformance<br>The source term and radiological consequence analysis methodologies incorporate applicable portions of this RG as it relates to evaluating design basis accidents according to NEI 18-04, endorsed by RG 1.233. In general, accident specific methodologies and assumptions may be used as guidance but are not applicable when following NEI 18-04. Accident specific dose criteria do not apply when following NEI 18-04. Accident source term does not apply to non-LWR designs. |
| 1.184 | Decommissioning of Nuclear Power Reactors | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.185 | Standard Format and Content for Post-Shutdown Decommissioning Activities Report | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.186 | Guidance and Examples for Identifying 10 CFR 50.2 Design Bases | 0 | Full Conformance |
| 1.187 | Guidance for Implementation of 10 CFR 50.59, “Changes, Tests, and Experiments” | 3 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 15 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.189 | Fire Protection for Nuclear Power Plants | 5 | Full Conformance |
| 1.191 | Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.192 | Operation and Maintenance Code Case Acceptability, ASME OM Code | 4 | Full Conformance |
| 1.193 | ASME Code Cases Not Approved for Use | 7 | Full Conformance |
| 1.194 | Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants | 0 | Full Conformance |
| 1.195 | Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors | 0 | Partial Conformance<br>The source term and radiological consequence analysis methodologies incorporate portions of this RG as it relates to evaluating design basis accidents according to NEI 18-04, endorsed by RG 1.233. The only portion of RG 1.195 that is directly applicable to DBA analysis in the design are the governing dose equations presented in Section 2. Other sections such as 4 and 5.2 may be referred to for guidance in the analysis of the design basis accidents. |
| 1.196 | Control Room Habitability at Light-Water Nuclear Power Reactors | 1 | Partial Conformance<br>Applicable to LWR nuclear power plants; however, aspects of this RG 1.196 will be applied to the design as necessary to meet PDC-19. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 16 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.197 | Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors | 0 | Full Conformance |
| 1.198 | Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites | 0 | Full Conformance |
| 1.199 | Anchoring Components and Structural Supports in Concrete | 1 | Full Conformance |
| 1.202 | Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.203 | Transient and Accident Analysis Methods | 0 | Full Conformance |
| 1.204 | Guidelines for Lightning Protection for Production and Utilization Facilities | 0 | Full Conformance |
| 1.208 | A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion | 0 | Full Conformance |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 17 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.209 | Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants | 0 | Partial Conformance<br>Equipment located in a mild environment is qualified in accordance with IEEE-323-2003 as endorsed by RG 1.209. The design also uses IEEE 323-2016 as endorsed by RG 1.89, with the exception of instances of qualification of equipment located within a mild radiation environment as defined by Section B of RG 1.89. Equipment located in a harsh environment is qualified by testing, analysis, or a combination of testing and analysis in accordance with IEEE-323-2016 as endorsed by RG 1.89. Both code years are part of the design. |
| 1.211 | Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants | 0 | Full Conformance |
| 1.214 | Response Strategies for Potential Aircraft Threats (OUO-SRI) | 1 | Full Conformance |
| 1.217 | Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts | 0 | Full Conformance |
| 1.218 | Condition-Monitoring Techniques for Electric Cables Used in Nuclear Power Plants | 0 | Full Conformance |
| 1.219 | Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.226 | Flexible Mitigation Strategies for Beyond-Design-Basis Events | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 18 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.231 | Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants | 0 | Full Conformance |
| 1.232 | Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors | 0 | Full Conformance |
| 1.233 | Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors | 0 | Full Conformance |
| 1.234 | Evaluating Deviations and Reporting Defects and Noncompliance Under 10 CFR Part 21 | 0 | Full Conformance |
| 1.239 | Licensee Actions to Address Nonconservative Technical Specifications | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 1.240 | Fresh and Spent Fuel Pool Criticality Analysis | 0 | Full Conformance |
| 1.242 | Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water-Reactors, and Non-Power Production or Utilization Facilities | 0 | Full Conformance |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 19 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.243 | Safety-Related Steel Structures and Steel-Plate Composite Walls for other than Reactor Vessels and Containments | 0 | Full Conformance |
| 1.244 | Control of Heavy Loads at Nuclear Facilities | 0 | Full Conformance |
| 1.246 | Acceptability of ASME Code, Section XI, Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,” for Non-Light Water Reactors | 0 | Full Conformance |
| 1.247 | Acceptability of Probabilistic Risk Assessment Results for Non-Light Water Reactor Risk-Informed Activities | Trial | Full Conformance |
| 1.248 | Guide for Assessing, Monitoring, and Mitigating Aging Effects on Electrical Equipment Used in Production and Utilization Facilities | 0 | Full Conformance |
| 1.249 | Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors | 0 | Full Conformance |
| 1.250 | Dedication of Commercial-Grade Digital Instrumentation and Control Items for Use in Nuclear Power Plants | 0 | Full Conformance |

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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 20 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 1.253 | Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors | 0 | Full Conformance |
| 4.1 | Radiological Environmental Monitoring for Nuclear Power Plants | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 4.2 | Preparation of Environmental Reports for Nuclear Power Stations | 3 | Full Conformance |
| 4.7 | General Site Suitability Criteria for Nuclear Power Stations | 3 | Full Conformance |
| 4.11 | Terrestrial Environmental Studies for Nuclear Power Stations | 2 | Full Conformance |
| 4.13 | Environmental<br>Dosimetry - Performance Specifications, Testing, and Data Analysis | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 4.15 | Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) -- Effluent Streams and the Environment | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 4.21 | Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning | 0 | Full Conformance |

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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 21 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 4.22 | Decommissioning Planning During Operations | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 4.24 | Aquatic Environmental Studies for Nuclear Power Stations | 0 | Full Conformance |
| 4.25 | Assessment of Abnormal Radionuclide Discharges in Ground Water to the Unrestricted Area at Nuclear Power Plant Sites | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 4.26 | Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites | 0 | Partial Conformance<br>RG 4.26 Revision 1 is the most recent version but Revision 0 was used in the Volcanic Hazards Assessment Topical Report (Reference 1.4-18)<br>Based on initial site characterization activities, and recognition that the nearest Quaternary volcano was >117 kilometers from the proposed site, uncertainties in the available information appeared sufficiently constrained to successfully conduct the volcanic hazards assessment without using the Senior Seismic Hazard Analysis Committee process. |
| 5.7 | Entry/Exit Control for Protected Areas, Vital<br>Areas, and Material Access Areas | 1 | Full Conformance |
| 5.12 | General Use of Locks in the Protection and Control of: Facilities, Radioactive Materials, Classified Information, Classified Matter, and Safeguards Information | 1 | Full Conformance |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 22 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 5.20 | Training, Equipping, and Qualifying of Guards and Watchmen | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.26 | Selection of Material Balance Areas and Item Control Areas | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.29 | Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.31 | Specially Designed Vehicle with Armed Guards for Road Shipment of Special Nuclear Material | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.44 | Perimeter Intrusion Alarm Systems | 3 | Full Conformance |
| 5.51 | Independent Assessment of Nuclear Material Control and Accounting Systems | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.52 | Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (Other than Nuclear Power Plants) | 3 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.53 | Qualification, Calibration, and Error Estimation Methods for Nondestructive Assay | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.54 | Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants (SGI) | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 23 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 5.58 | Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 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 | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.60 | Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.62 | Physical Security Event Notifications, Reports, and Records | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.63 | Physical Protection for Transient Shipments | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.66 | Access Authorization Program for Nuclear Power Plants | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.69 | Guidance for the Application of Radiological Sabotage Design-Basis Threat in the Design, Development and Implementation of a Physical Security Program that Meets 10 CFR 73.55 Requirements (SGI) | 1 | Full Conformance |
| 5.73 | Fatigue Management for Nuclear Power Plant Personnel | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.74 | Managing the Safety/Security Interface | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 24 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 5.75 | Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.76 | Physical Protection Programs at Nuclear Power Reactors (SGI) | 1 | Full Conformance |
| 5.77 | Insider Mitigation Program | 1 | Full Conformance |
| 5.79 | Protection of Safeguards Information | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.80 | Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.81 | Target Set Identification and Development for Nuclear Power Reactors (OUO-SRI) | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.84 | Fitness-For-Duty Programs at New Reactor Construction Sites | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.86 | Preemption Authority, Enhanced Weapons Authority, and Firearms Background Checks | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.87 | Suspicious Activity Reports Under 10 CFR Part 73 | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 5.89 | Fitness-for-Duty Programs for Commercial Power Reactor and Category I Special Nuclear Material Licensees | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 25 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 8.2 | Administrative Practices in Radiation Surveys and Monitoring | 1 | Full Conformance |
| 8.7 | Instructions for Recording and Reporting Occupational Radiation Exposure Data | 4 | Full Conformance |
| 8.8 | Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as is Reasonably Achievable | 3 | Full Conformance |
| 8.9 | Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program | 1 | Full Conformance |
| 8.10 | Operating Philosophy for Maintaining Occupational and Public Radiation Exposures as Low as is Reasonably Achievable | 2 | Full Conformance |
| 8.13 | Instruction Concerning Prenatal Radiation Exposure | 3 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.15 | Acceptable Programs for Respiratory Protection | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.19 | Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates | 1 | Full Conformance |
| 8.20 | Applications of Bioassay for Radioiodine | 2 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.25 | Air Sampling in the Workplace | 1 | Full Conformance |


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**Table 1.4-1 Conformance with Regulatory Guides**
**(Sheet 26 of 26)**

| Guide Number | Title | Revision | Conformance Status |
|---|---|---|---|
| 8.26 | Applications of Bioassay for Fission and Activation Products | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.27 | Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.28 | Audible-Alarm Dosimeters | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.29 | Instruction Concerning Risks from Occupational Radiation Exposure | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.34 | Monitoring Criteria and Methods to Calculate Occupational Radiation Doses | 1 | Full Conformance |
| 8.35 | Planned Special Exposure | 1 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.36 | Radiation Dose to the Embryo/Fetus | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |
| 8.38 | Control of Access to High and Very High Radiation Areas of Nuclear Plants | 1 | Full Conformance |
| 8.40 | Methods for Measuring Effective Dose Equivalent from External Exposure | 0 | The guidance is not applicable to the design and construction phase of the project and will be addressed at the OL stage. |


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**Table 1.4-2 Topical Reports**

| Topical Report Number | Submittal Accession Number | Topical Report Title | Section(s) |
|---|---|---|---|
| TP-QA-PD-0001 | ML23213A199 | QA Program Description Topical Report | 8.1 |
| NAT-3056 | ML23321A036 | Plume Exposure Pathway Emergency Planning Zone Sizing Methodology | 11.3.3 |
| NAT-2806 | ML23025A409 | Fuel and Control Assembly Qualification | 7.1.1 |
| NATD-LIC-RPRT-0002 | ML23024A281 | Principal Design Criteria for the Natrium Advanced Reactor | 5.3.1 |
| NAT-3226 | ML23115A387 | An Analysis of Potential Volcanic Hazards at the Proposed Natrium Site near Kemmerer, Wyoming | 2.7 |
| NAT-2965 | ML23116A226 | Human Factors Engineering Program Plan and Methodologies | 11.2 |
| TP-LIC-RPT-0003 | ML24017A116 | Radiological Source Term Methodology | 3.1.1 |
| TP-LIC-RPT-0004 | ML23272A260 | Design Basis Accident Methodology for In-Vessel Events without Radiological Release | 3.3 |
| TP-LIC-RPT-0005 | ML23311A139 | Radiological Release Consequences Methodology | 3.1.1 |
| TP-LIC-RPT-0006 | ML23334A239 | Stability Methodology | 3.12.7 |
| TP-LIC-RPT-0008 | ML24085A822 | Partial Flow Blockage Methodology | 3.3.1 |
| NAT-4950 | ML24068A186 | Instrumentation & Control Architecture and Design Basis | 7.6 |
| NAT-8922 | ML24068A212 | Reactor Seismic Isolation System Qualification | 7.1.2 |
| TP-LIC-RPT-0007 | ML24082A262 | DBA Transient Methods for Events with Radiological Release | 3.3.1 |


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**Table 1.4-3 Technical Reports**

| Technical Report Number | Technical Report Title | Section(s) |
|---|---|---|
| TP-LIC-RPT-0011 | Core Design and Thermal Hydraulic Technical Report | 3.12,<br>3.13 |
| TP-LIC-RPT-0009 | Major Accident Methodology Report | 3.3 |
| TP-LIC-RPT-0012 | Preliminary Emergency Planning Zone Determination Analysis | 11.3.3 |



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**Table 1.4-4 Applicability of TMI-Related Requirements**
**(Sheet 1 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(1)(i) | Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8) | Full Compliance | 3.1.1 |
| 50.34(f)(1)(ii) | Perform an evaluation of the proposed auxiliary feedwater system. (II.E.1.1) | Not Applicable. Only applies to pressurized water reactors (PWRs). | Not Applicable |
| 50.34(f)(1)(iii) | Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. (II.K.2.16 and II.K.3.25) | Not Applicable. Not technically relevant to the design. | Not Applicable |
| 50.34(f)(1)(iv) | Perform an analysis of the probability of a small-break LOCA caused by a stuck-open power- operated relief valve. (PORV) (II.K.3.2) | Not Applicable. Only applies to PWRs. | Not Applicable |
| 50.34(f)(1)(v) | Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection and reactor core isolation cooling system initiation levels. (II.K.3.13) | Not Applicable. Only applies to boiling water reactors (BWRs). | Not Applicable |
| 50.34(f)(1)(vi) | Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves. (II.K.3.16) | Not Applicable. Only applies to BWRs. | Not Applicable |
| 50.34(f)(1)(vii) | Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system design modifications that would eliminate the need for manual activation. (II.K.3.18) | Not Applicable. Only applies to BWRs. | Not Applicable |


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**Table 1.4-4 Applicability of TMI-Related Requirements**
**(Sheet 2 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(1)(viii) | Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems. (II.K.3.21) | Not Applicable. Only applies to BWRs. | Not Applicable |
| 50.34(f)(1)(ix) | Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the high pressure coolant injection and reactor core isolation cooling systems. (II.K.3.24) | Not Applicable. Only applies to BWRs. | Not Applicable |
| 50.34(f)(1)(x) | Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions. (II.K.3.28) | Not Applicable. Only applies to BWRs. | Not Applicable |
| 50.34(f)(1)(xi) | Provide an evaluation of depressurization methods. (II.K.3.45) | Not Applicable. Only applies to BWRs. | Not Applicable |
| 50.34(f)(1)(xii) | Perform an evaluation of alternative hydrogen control systems. | Full Compliance | 7.2.4 |
| 50.34(f)(2)(i) | Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs. (I.A.4.2) | Full Compliance | 11.2 |
| 50.34(f)(2)(ii) | Establish a program, to begin during construction, to improve plant procedures, with the program scope to include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (I.C.9) | Full Compliance | 11.2 |


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**Table 1.4-4 Applicability of TMI-Related Requirements (Sheet 3 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(2)(iii) | Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1) | Full Compliance | 7.6.7 |
| 50.34(f)(2)(iv) | Provide a plant safety parameter display console. (I.D.2) | Full Compliance | 11.2 |
| 50.34(f)(2)(v) | Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3) | Full Compliance | 11.2 |
| 50.34(f)(2)(vi) | Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1) | Not Applicable. Venting gasses is not required to satisfy core cooling. | Not Applicable |
| 50.34(f)(2)(vii) | Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access. (II.B.2) | Full Compliance | 10.1 |


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**Table 1.4-4 Applicability of TMI-Related Requirements**
**(Sheet 4 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(2)(viii) | Provide capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials. (II.B.3) | Partial Compliance. The design will provide a capability to promptly obtain and analyze samples from the reactor coolant system that may contain accident source term materials without radiation exposures to any individual exceeding 5 rem to the whole body or 50 rem to the extremities.<br>The design will not obtain and analyze for hydrogen in the containment atmosphere, dissolved gases in the coolant or chlorides in the coolant as these are not applicable. | 7.2.4 |
| 50.34(f)(2)(ix) | Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. (II.B.8) | Not Applicable.<br>Fuel-sodium reaction does not produce hydrogen. | Not Applicable |
| 50.34(f)(2)(x) | Provide a test program and associated model development, and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves. (II.D.1) | Partial Compliance. The design will qualify reactor coolant system relief and safety valves for argon containing sodium vapors and aerosols at conditions expected during identified plant events using analysis and testing. | 7.2.3 |
| 50.34(f)(2)(xi) | Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3) | Full Compliance | 11.2 |
| 50.34(f)(2)(xii) | Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide AFW system flow indication in the control room. (II.E.1.2) | Not Applicable. Only applies to PWRs. | Not Applicable |


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**Table 1.4-4 Applicability of TMI-Related Requirements**
**(Sheet 5 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(2)(xiii) | Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (II.E.3.1) | Not Applicable. Only applies to PWRs. | Not Applicable |
| 50.34(f)(2)(xiv) | Provide containment isolation systems that (A) ensure all non-essential systems are isolated automatically; (B) ensure each non-essential penetration (except instrument lines) have two isolation barriers in series; (C) do not result in reopening of the containment isolation valves on resetting of the isolation signal; (D) utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation; and (E) include automatic closing on a high radiation signal for all systems that provide a path to the environs. (II.E.4.2) | Not Applicable. The design uses functional containment. | Not Applicable |
| 50.34(f)(2)(xv) | Capability for containment purging/venting designed to minimize the purging time consistent with as low as reasonably achievable (ALARA). (II.E.4.4) | Not Applicable. The design utilizes functional containment and does not have a single containment structure. Any leak into the head access area portion of the functional containment will be managed through normal heating, ventilation, and air conditioning system functions and a does not required a dedicated purge system. | Not Applicable |


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**Table 1.4-4 Applicability of TMI-Related Requirements**
**(Sheet 6 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(2)(xvi) | Establish design criterion for the allowable number of actuation cycles of the ECCS and reactor protection system with the expected occurrence rates of severe overcooling events. (II.E.5.1) | Not Applicable. Only applies to Babcock & Wilcox plants. | Not Applicable |
| 50.34(f)(2)(xvii) | Provide instrumentation to measure, record, and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1) | Partial Compliance. Section (B) is not applicable because there is no water used in containment and Section (C) is not applicable because hydrogen is not a byproduct of coolant-clad interaction. | 11.2 |
| 50.34(f)(2)(xviii) | Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling. (II.F.2) | Full Compliance | 7.1.3<br>11.2 |
| 50.34(f)(2)(xix) | Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3) | Full Compliance | 7.6.5<br>11.2 |
| 50.34(f)(2)(xx) | Provide power supplies for pressurizer relief valves, block valves, and level indicators. (II.G.1) | Not Applicable. Only applies to PWRs. | Not Applicable |


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**Table 1.4-4 Applicability of TMI-Related Requirements (Sheet 7 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(2)(xxi) | Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (II.K.1.22) | Not Applicable. Only applies to BWRs. | Not Applicable |
| 50.34(f)(2)(xxii) | Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (II.K.2.9) | Not Applicable. Only applies to Babcock & Wilcox plants. | Not Applicable |
| 50.34(f)(2)(xxiii) | Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (II.K.2.10) | Not Applicable. Only applies to Babcock & Wilcox plants. | Not Applicable |
| 50.34(f)(2)(xxiv) | Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (II.K.3.23) | Not Applicable. Only applies to BWRs. | Not Applicable |
| 50.34(f)(2)(xxv) | Provide an onsite Technical Support Center and onsite Operational Support Center.(III.A.1.2) | Partial Compliance. The new Emergency Preparedness rule has been approved and does not require these specific facilities. | 11.3 |


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**Table 1.4-4 Applicability of TMI-Related Requirements (Sheet 8 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(2)(xxvi) | Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials. (III.D.1.1) | Partial Compliance. The design uses functional containment and not a single containment structure. As such, barriers and boundaries classified as safety-related will be leak testable to demonstrate their performance. The design will provide for leakage control and detection in the design of safety related barriers that contain (or might contain) accident source term following an accident.<br>The program will submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from these systems. | 8 |
| 50.34(f)(2)(xxvii) | Provide for monitoring of in-plant radiation and airborne radioactivity. (III.D.3.3) | Full Compliance | 10.1 |
| 50.34(f)(2)(xxviii) | Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term release. (III.D.3.4) | Full Compliance | 10.1 |
| 50.34(f)(3)(i) | Provide administrative procedures for evaluating operating, design, and construction experience. (I.C.5) | Full Compliance | 11.2 |


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**Table 1.4-4 Applicability of TMI-Related Requirements**
**(Sheet 9 of 9)**

| Regulation | Description | Compliance Status | Section |
|---|---|---|---|
| 50.34(f)(3)(ii) | Ensure that the QA list required by Criterion II in Appendix B to 10 CFR 50 includes all SSC important to safety. (I.F.1) | Full Compliance | 11.4 |
| 50.34(f)(3)(iii) | Establish a QA Program based on the specified considerations. (I.F.2) | Full Compliance | 8.1 |
| 50.34(f)(3)(iv) | Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot-diameter opening. (II.B.8) | Not Applicable. The design uses functional containment and not a single containment structure. | Not Applicable |
| 50.34(f)(3)(v) | Preliminary Design Information - Containment Integrity. (II.B.8) | Not Applicable. The design does not utilize a containment structure and does not produce hydrogen from clad-coolant interaction. | Not Applicable |
| 50.34(f)(3)(vi) | For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations. (II.E.4.1) | Not Applicable. The design does not utilize hydrogen recombiners. | Not Applicable |
| 50.34(f)(3)(vii) | Provide a description of the management plan for design and construction activities. (II.J.3.1) | Full Compliance | 11.1 |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 1 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| ACI 318 | 2019 | Building Code Requirements for Structural Concrete Commentary on Building Code Requirements for Structural Concrete |
| ACI 349 | 2013 | Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary |
| ACI 349.1 | 2007 | Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures |
| ACI 355.2 | 2007 | Qualification of Post-Installed Mechanical Anchors in Concrete and Commentary |
| AISC SG-673 | 1986 | Specification for the Design of Cold-Formed Steel Structural Members |
| ANS 2.2 | 2016 | Earthquake Instrumentation Criteria for Nuclear Power Plants |
| ANS 2.23 | 2016 | Nuclear Power Plant Response to an Earthquake |
| ANS 57.1 | 1992 | Design Requirements for Light Water Reactor Fuel Handling Systems |
| ANSI N14.6 | 1993 | Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More |
| ANSI N271 | 1976 | Containment Isolation Provisions for Fluid Systems |
| ANSI N323D | 2002 | Installed Radiation Protection Instrumentation |
| ANSI N42.14 | 1999 | American National Standard for Calibration and Use of Germanium Spectrometers for the Measurement of Gamma-Ray Emission Rates of Radionuclides |
| ANSI/AISC 360 | 2016 | Specification for Structural Steel Buildings |
| ANSI/AISC N690 | 2018 | Specification for Safety-Related Steel Structures for Nuclear Facilities |
| ANSI/ANS 2.5 | 1984 | Determining Meteorological Information at Nuclear Power Sites |
| ANSI/ANS 3.1 | 2014 | Selection, Qualification, and Training of Personnel for Nuclear Power Plants |
| ANSI/ANS 3.2 | 2012 | Managerial, Administrative, and Quality Controls for Operational Phase of Nuclear Power Plants |
| ANSI/ANS 3.4 | 2013 | Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants |
| ANSI/ANS 3.5 | 2009 | Nuclear Power Plant Simulators for Use in Operator Training and Examination |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 2 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| ANSI/ANS 6.3.1 | 1987 | Program for Testing Radiation Shields in Light Water Reactors |
| ANSI/ANS-6.4 | 2006 | Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants |
| ANSI/ISA 67.04.01 | 2018 | Setpoints for Nuclear Safety-Related Instrumentation |
| API 527 | 2020 | Seal Tightness of Pressure Relief Valves |
| API RP 520 | 10 | Sizing, Selection, and Installation of Pressure-Relieving Devices |
| ASCE 37 | 2014 | Design Loads on Structures During Construction |
| ASCE 4 | 2017 | Seismic Analysis of Safety-Related Nuclear Structures and Commentary |
| ASCE 43 | 2019 | Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities |
| ASCE 7 | 2016 | Minimum Design Loads and Associated Criteria for Building and Other Structures with Supplement 1 |
| ASME AG-1 | 2019 | Code on Nuclear Air and Gas Treatment |
| ASME AG-1b | 2009 | Code on Nuclear Air and Gas Treatment |
| ASME B31.1 | 2022 | Power Piping |
| ASME B31.3 | 2020 | Process Piping |
| ASME B36.19M | 2004 | Stainless Steel Pipe |
| ASME BPVC Code Case N-861 | 2017 | Satisfaction of Strain Limits for Division 5 Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic Analysis |
| ASME BPVC Code Case N-862 | 2017 | Calculation of Creep-Fatigue for Division 5 Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic Analysis |
| ASME BPVC Code Case N-924 | 2022 | Design Rules and Limits for Load-Controlled Stresses for Class A Components at Elevated Temperature Service Using Elastic-Perfectly Plastic and Simplified Inelastic Analyses Section III, Division 5 |
| ASME BPVC Section II | 2019 | Materials |
| ASME BPVC Section III, Division I | 2017 | BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NE-Class MC Components |
| ASME BPVC Section IX | 2019 | Welding, Brazing, and Fusing Qualifications |
| ASME BPVC Section V | 2021 | Nondestructive Examination |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 3 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| ASME BPVC Section XI Division 1 | 2019 | Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants |
| ASME BPVC Section XI Division 2 | 2019 | Rules for Inservice Inspection of Nuclear Power Plant Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants |
| ASME BPVC, Section III, Division 5 | 2017 | ASME Boiler and Pressure Vessel Code, Section III |
| ASME-BPVC-III-A | 2017 | Section III, Appendices, Rules for Construction of Nuclear Facility Components |
| ASME BPVC-III-NCA | 2017 | Section III, Subsection NCA - General Requirements for Division 1 and Division 2 |
| ASME BPVC-VIII-1 | 2021 | Section VIII, Division 1, Rules for Construction of Pressure Vessels |
| ASME BPVC-VIII-2 | 2021 | Section VIII, Division 2, Rules for Construction of Pressure Vessels, Alternative Rules |
| ASME BTH-1 | 2017 | Design of Below-the-Hook Lifting Devices |
| ASME N509 | 2002 | Nuclear Power Plant Air-Cleaning Units and Components |
| ASME NML-1 | 2019 | Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities |
| ASME NOG-1 | 2020 | Rules for Construction of Overhead and Gantry Cranes |
| ASME NQA-1 | 2015 | Quality Assurance Program Requirements for Nuclear Facilities |
| ASME NUM-1 | 2016 | Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type) |
| ASME OM Code | 2020 | Code for Operation and Maintenance of Nuclear Power Plants |
| ASME PTC 25 | 2014 | Pressure Relief Devices |
| ASME QME-1 | 2023 | Qualification of Active Mechanical Equipment Used in Nuclear Power Plants |
| ASME STS-1 | 2021 | Steel Stacks |
| ASME/ANS RA-S-1.4 | 2021 | Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 4 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| ASTM C1055 | 2020 | Standard Guide for Heated System Surface Conditions that Produce Contact Burn Injuries |
| ASTM C535 | 16 | Standard Test Method for Resistance to Degradation of Large-Size Coarse Aggregate by Abrasion and Impact in the Los Angeles Machine |
| ASTM C692 | 2013 | Standard Test Method for Evaluating the Influence of Thermal Insulations on External Stress Corrosion Cracking Tendency of Austenitic Stainless Steel |
| ASTM C795 | 2008 | Standard Specification for Thermal Insulation for Use in Contact with Austenitic Stainless Steel |
| ASTM C871 | 18 | Standard Test Methods For Chemical Analysis Of Thermal Insulation Materials For Leachable Chloride, Fluoride, Silicate, And Sodium Ions |
| ASTM D2488 | 17e1 | Standard Practice for Description and Identification of Soils (Visual-Manual Procedure) |
| ASTM D2850 | 15 | Standard Test Method for Unconsolidated-Undrained Triaxial Compression Test on Cohesive Soils |
| ASTM D2859 | 2006 | Standard Test Method for Ignition Characteristics of Finished Textile Floor Covering Materials |
| ASTM D2974 | 20e1 | Standard Test Methods for Moisture, Ash, and Organic Matter of Peat and Other Organic Soils |
| ASTM D3286 | 1996 | Standard Test Method for Gross Calorific Value of Coal and Coke by the Isoperibol Bomb Calorimeter |
| ASTM D3359 | 2022 | Standard Test Methods for Measuring Adhesion by Tape Test |
| ASTM D3843 | 2016 | Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities |
| ASTM D3911 | 2016 | Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident Conditions |
| ASTM D3912 | 2010 | Standard Test Method for Chemical Resistance of Coatings and Linings for Use in Nuclear Power Plants |
| ASTM D4082 | 2010 | Standard Test Method for Effects of Gamma Radiation on Coatings for Use in Nuclear Power Plants |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 5 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| ASTM D4186 | 20e1 | Standard Test Method for One-Dimensional Consolidation Properties of Saturated Cohesive Soils Using Controlled-Strain Loading |
| ASTM D4227 | 5 | Standard Practice for Qualification of Coating Applicators for Application of Coatings to Concrete Surfaces |
| ASTM D4228 | 5 | Standard Practice for Qualification of Coating Applicators for Application of Coatings to Steel Surfaces |
| ASTM D4286 | 8 | Standard Practice for Determining Coating Contractor Qualifications for Nuclear Powered Electric Generating Facilities |
| ASTM D4452 | 2022 | Standard Practice for X-Ray Radiography of Soil Samples |
| ASTM D4643 | 17 | Standard Test Method for Determination of Water (Moisture) Content of Soil by Microwave Oven Heating |
| ASTM D5144 | 2008 | Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants |
| ASTM D5163 | 2016 | Standard Guide for Establishing a Program for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants |
| ASTM 6913/D6913M | 17 | Standard Test Methods for Particle-Size Distribution (Gradation) of Soils Using Sieve Analysis |
| ASTM D7167 | 2018 | Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coatings Service Level III Lining Systems in an Operating Nuclear Power Plant |
| ASTM D7230 | 06R13 | Standard Guide for Evaluating Polymeric Lining Systems for Water Immersion in Coating Service Level III Safety-Related Applications on Metal Substrates |
| ASTM E119 | 09b | Standard Test Methods for Fire Tests of Building Construction and Materials |
| ASTM E488/E488M | 2015 | Standard Test Methods for Strength of Anchors in Concrete Elements |
| ASTM E814-13A | 2017 | Fire Tests of Penetration Firestop Systems |
| ASTM E84 | 2022 | Standard Test Method for Surface Burning Characteristics of Building Materials |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 6 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| ASTM D4044/D4044M | 15 | Standard Test Method for (Field Procedure) for Instantaneous Change in Head (Slug) Tests for Determining Hydraulic Properties of Aquifers |
| ASTM D4630 | 19 | Standard Test Method for Determining Transmissivity and Storage Coefficient of Low-Permeability Rocks by In Situ Measurements Using the Constant Head Injection Test |
| AWS D1.3 | 1998 | Structural Welding Code - Sheet Steel |
| AWS D9.1 | 2000 | Sheet Metal Welding Code |
| DANU-ISG-2023-01 | Draft | Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications — Roadmap |
| FEMA-REP-10 | 1985 | Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants |
| IEC 61000-3 | 2017 | Electromagnetic Compatibility (EMC) - Part 3: Limit - All Parts |
| IEC 61000-4 | 2012 | Electromagnetic Compatibility Testing and Measurement Package - Parts 1 to 3 |
| IEC 61000-6 | 2011 | EMC Testing of Industrial Environments |
| IEEE 1008 | 1987 | Standard for Software Unit Testing |
| IEEE 1012 | 2016 | Standard for System, Software, and Hardware Verification and Validation |
| IEEE 1023 | 2020 | IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities, Institute of Electrical and Electronics Engineers |
| IEEE 1028 | 2008 | Standard for Software Reviews and Audits |
| IEEE 1050 | 2004 | Guide for Instrumentation and Control Equipment Grounding in Generating Stations |
| IEEE 1074 | 2006 | Standard for Developing Software Life Cycle Processes |
| IEEE 1187 | 2013 | Recommended Practice for Installation Design and Installation of Valve-Regulated Lead-Acid Batteries for Stationary Applications |
| IEEE 1189 | 2007 | Guide for Selection of Valve-Regulated Lead-Acid (VRLA) Batteries for Stationary Applications |
| IEEE 1202 | 2006 | Standard for Flame-Propagation Testing of Wire and Cable |
| IEEE 1375 | 1998 | Guide for the Protection of Stationary Battery Systems |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 7 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| IEEE 1491 | 2012 | Guide for Selection and Use of Battery Monitoring Equipment in Stationary Applications |
| IEEE 142 | 2007 | IEEE Recommended Practice for Grounding of Industrial and Commercial Power Systems |
| IEEE 308 | 2001 | Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations |
| IEEE 323 | 2016 | Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations |
| IEEE 336 | 2020 | IEEE Guide for Installation, Inspection, and Testing for Class 1E Power, Instrumentation, and Control Equipment at Nuclear Facilities |
| IEEE 338 | 2022 | Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems |
| IEEE 344 | 2020 | IEEE/IEC International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification |
| IEEE 352 | 2016 | IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Systems and Other Nuclear Facilities |
| IEEE 379 | 2000 | Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems |
| IEEE 382 | 2006 | IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations and Other Nuclear Facilities |
| IEEE 383 | 2003 | Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations |
| IEEE 384 | 1992 | Standard Criteria for Independence of Class 1E Equipment and Circuits |
| IEEE 420 | 2013 | Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations |
| IEEE 485 | 2010 | Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications |
| IEEE 497 | 2016 | Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 8 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| IEEE 535 | 2013 | Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations |
| IEEE 603 | 1991 | IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations |
| IEEE 610.12 | 1990 | Standard Glossary of Software Engineering Terminology |
| IEEE 634 | 2004 | Standard Cable-Penetration Fire Stop Qualification Test |
| IEEE 650 | 2006 | Standard for Qualification of Class 1E Static Battery Chargers, Inverters, and Uninterruptible Power Supply Systems for Nuclear Power Generating Stations |
| IEEE 730 | 2002 | Standard for Software Quality Assurance Plans |
| IEEE 7-4.3.2 | 2016 | IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations |
| IEEE 741 | 2007 | IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations |
| IEEE 829 | 2008 | Standard for Software and System Test Documentation |
| IEEE 946 | 2020 | Recommended Practice for the Design of DC Power Systems for Stationary Applications |
| ISA 67.02.01 | 2014 | Nuclear Safety-Related Instrument Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants |
| MSS SP-42 | 2013 | Corrosion-Resistant Gate, Globe, Angle, and Check Valves with Flanged and Butt Weld Ends |
| MSS SP-61 | 2019 | Pressure Testing of Valves |
| NFPA 11 | 2020 | Standard for Low-, Medium- and High-Expansion Foam Systems |
| NFPA 1144 | 2008 | Standard for Reducing Structure Ignition Hazards from Wildland Fire |
| NFPA 12 | 2022 | Standard on Carbon Dioxide Extinguishing Systems |
| NFPA 12A | 2004 | Standard on Halon 1301 Fire Extinguishing Systems |
| NFPA 1404 | 2018 | Standard for Fire Service Respiratory Protection Training |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 9 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| NFPA 1410 | 2020 | Standard on Training for Initial Emergency Scene Operations |
| NFPA 15 | 2022 | Standard for Water Spray Fixed Systems for Fire Protection |
| NFPA 1500 | 2021 | Standard on Fire Department Occupational Safety and Health Program |
| NFPA 16 | 2003 | Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems |
| NFPA 1620 | 2020 | Recommended Practice for Pre-Incident Planning |
| NFPA 204 | 2021 | Standard for Smoke and Heat Venting |
| NFPA 220 | 2021 | Standard on Types of Building Construction |
| NFPA 221 | 2021 | Standard for High Challenge Fire Walls and Fire Barrier Walls |
| NFPA 251 | 2006 | Standard Methods of Tests of Fire Resistance of Building Construction and Materials |
| NFPA 253 | 2018 | Standard Method of Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source |
| NFPA 259 | 2018 | Standard Test Method for Potential Heat of Building Material |
| NFPA 484 | 2022 | Standard for Combustible Metals |
| NFPA 51B | 2003 | Standard for Fire Prevention During Welding, Cutting, and Other Hot Work |
| NFPA 54 | 2021 | National Fuel Gas Code |
| NFPA 55 | 2020 | Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks |
| NFPA 58 | 2020 | Liquefied Petroleum Gas Code |
| NFPA 600 | 2020 | Standard on Industrial Fire Brigades |
| NFPA 701 | 2004 | Standard Methods of Fire Tests for Flame Propagation of Textiles and Films |
| NFPA 703 | 2021 | Standard for Fire-Retardant Treated Wood and Fire-Retardant Coatings for Building Materials |
| NFPA 750 | 2023 | Standard on Water Mist Fire Protection Systems |
| NFPA 80 | 2022 | Standard for Fire Doors and Windows |
| NFPA 80A | 2022 | Recommended Practice for Protection of Buildings from Exterior Fire Exposures |


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**Table 1.4-5 Consensus Codes and Standards**
**(Sheet 10 of 10)**

| Codes and Standards | Revision/Edition | Title |
|---|---|---|
| NUMARC 91-06 | 4D | Guidelines for Industry Actions to Assess Shutdown Management |
| NUMARC 93-01 | 4F | Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants |
| ORNL/TM-10651 | 1987 | Pressure Vessel Fluence Analysis and Neutron Dosimetry |
| PBRCTS-1 | 2000 | Technical Procedures for Resonant Column and Torsional Shear (RCTS) Testing of Soil and Rock Samples |
| SMACNA 1108 | 2008 | Accepted Industry Practice For Industrial Duct Construction |
| UFC 3-340-02 | 2008 | Structures to Resist the Effects of Accidental Explosions |
| UL 555 | 2020 | UL Standard for Safety Fire Dampers |


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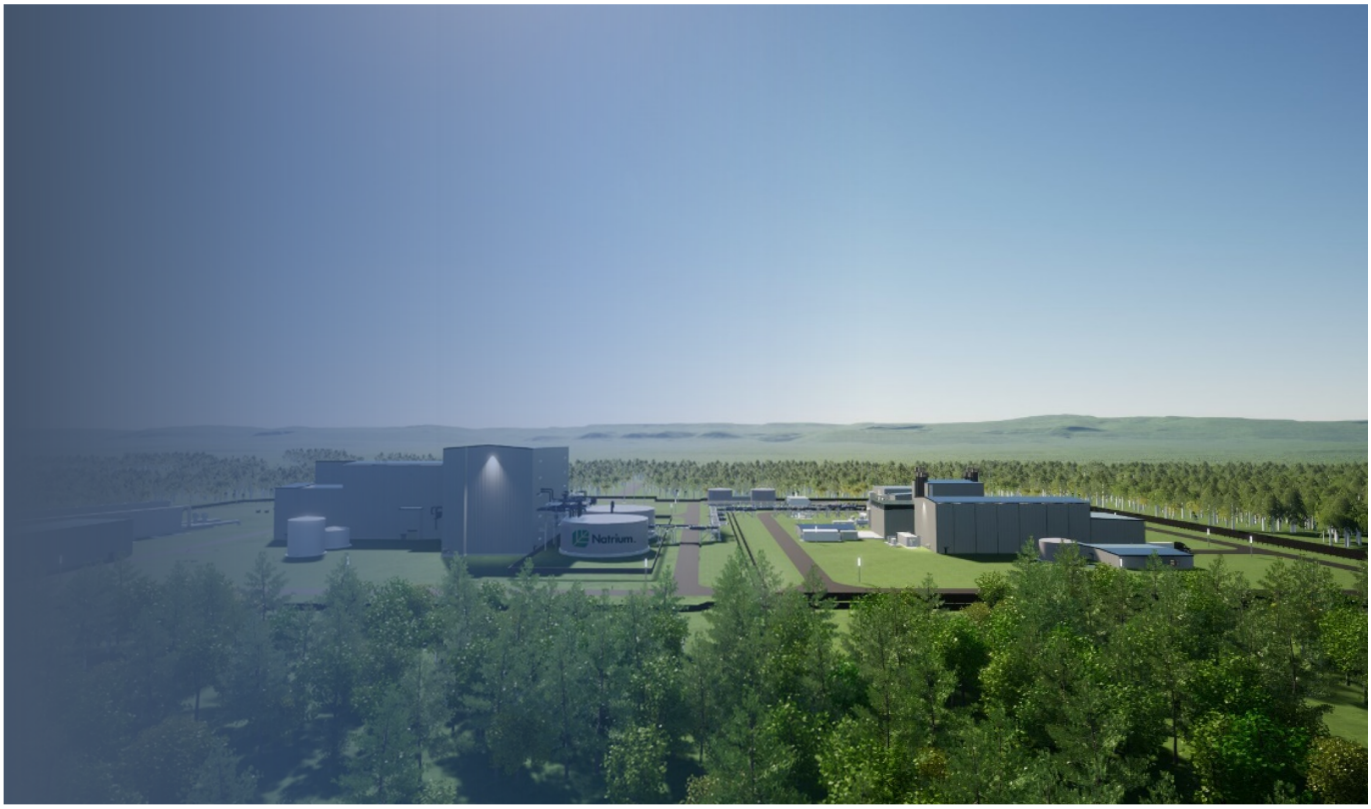

# Kemmerer Power Station Unit 1
# PSAR, Chapter 2

US SFR Owner, LLC, a wholly owned subsidiary of TerraPower, LLC

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## TABLE OF CONTENTS

**Chapter 2 Site Information . . . . . 2.1-1**

2.1 Site Characteristics and Site Parameters . . . . . 2.1-1

2.2 Geography and Demography . . . . . 2.2-1

2.2.1 Site Location and Description . . . . . 2.2-1

2.2.2 Exclusion Area Authority and Control . . . . . 2.2-1

2.2.3 Population Distribution . . . . . 2.2-2

2.3 Nearby Industrial, Transportation, and Military Facilities . . . . . 2.3-1

2.3.1 Location and Routes . . . . . 2.3-1

2.3.2 Descriptions . . . . . 2.3-3

2.3.3 Evaluation of Potential Hazards . . . . . 2.3-5

2.4 Regional Climatology, Local Meteorology, and Atmospheric Dispersion . . . . 2.4-1

2.4.1 Regional Climatology . . . . . 2.4-1

2.4.2 Local Meteorology and Topography . . . . . 2.4-10

2.4.3 Air Quality . . . . . 2.4-22

2.4.4 Atmospheric Dispersion . . . . . 2.4-22

2.4.5 Meteorological Monitoring . . . . . 2.4-24

2.5 Hydrologic Description . . . . . 2.5-1

2.5.1 Floods . . . . . 2.5-2

2.5.2 Flooding Protection . . . . . 2.5-36

2.5.3 Groundwater . . . . . 2.5-38

2.5.4 Low Water Considerations . . . . . 2.5-57

2.6 Geology, Seismology, and Geotechnical Engineering . . . . . 2.6-1

2.6.1 Geologic Hazards . . . . . 2.6-3

2.6.2 Vibratory Ground Motion . . . . . 2.6-22

2.6.3 Surface Deformation . . . . . 2.6-29



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## TABLE OF CONTENTS

2.6.4 Stability of Subsurface Materials and Foundations . . . . . 2.6-35

2.6.5 Stability of Slopes . . . . . 2.6-60

2.7 Volcanic Hazards . . . . . 2.7-1

2.8 Summary of External Hazards . . . . . 2.8-1


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## LIST OF TABLES


Table 2.1-1 Kemmerer Unit 1 Site-Specific External DBHLs . . . . . 2.1-2

Table 2.1-2 Kemmerer Unit 1 Site Characteristics . . . . . 2.1-3

Table 2.2-1 Kemmerer Unit 1 RXB Centerpoint Coordinates . . . . . 2.2-7

Table 2.2-2 Resident and Transient 10 Mi Radius Population for 2020 and Projections for 2030 through 2090 . . . . . 2.2-8

Table 2.2-3 0-to-50-Mi Population and Projections . . . . . 2.2-9

Table 2.2-4 Cumulative Population . . . . . 2.2-10

Table 2.3-1 Description of Facilities - Products and Materials . . . . . 2.3-17

Table 2.3-2 Screening of Chemicals at Nearby Facilities for Chemical Storage and Transport . . . . . 2.3-18

Table 2.3-3 Screening of Chemicals at the TFF for Chemical Storage and Transport . . . . . 2.3-20

Table 2.3-4 Transportation Routes - Highway Information Summary . . . . . 2.3-21

Table 2.3-5 (Not Used) . . . . . 2.3-22

Table 2.3-6 Transportation Routes - Pipeline Information Summary . . . . . 2.3-23

Table 2.3-7 Onsite Chemical Storage at Kemmerer Unit 1 . . . . . 2.3-24

Table 2.3-8 Meteorological Sensitivity Analysis . . . . . 2.3-25

Table 2.3-9 Offsite Facility Flammability and Explosion Hazards - Flammable Vapor Cloud and Vapor Cloud Explosion ALOHA Results . . . . . 2.3-26

Table 2.3-10 Offsite Facility Flammability and Explosion Hazards - Explosion at Source, TNT Equivalency Hand Calculation Results . . . . . 2.3-27

Table 2.3-11 Highway Transport Flammability and Explosion Hazards Results . . . . . 2.3-28

Table 2.3-12 Pipeline Flammability and Explosion Hazards Results . . . . . 2.3-29

Table 2.3-13 Onsite Storage Flammability and Explosion Hazards - Flammable Vapor Cloud and Vapor Cloud Explosion ALOHA Results . . . . . 2.3-30

Table 2.3-14 Onsite Storage Flammability and Explosion Hazards - Explosion at Source, TNT Equivalency Hand Calculation Results . . . . . 2.3-31

Table 2.3-15 Offsite Facility Toxicity Hazards Results . . . . . 2.3-32

Table 2.3-16 Highway Transport Toxicity Hazards Results . . . . . 2.3-33



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## LIST OF TABLES


Table 2.3-17 Pipeline Toxicity Hazards Results . . . . . 2.3-34

Table 2.3-18 Onsite Storage Toxicity Hazards Results . . . . . 2.3-35

Table 2.4-1 Meteorological Observation Stations and Climate Observing Stations With Respect to Kemmerer Unit 1 . . . . . 2.4-30

Table 2.4-2 Detailed Topography Within 5 Miles (8 Kilometers) . . . . . 2.4-31

Table 2.4-3 Kemmerer Unit 1 Site Terrain 0–50 Miles (0–80 Kilometers) . . . . . 2.4-32

Table 2.4-4 Summary of Normals (Temperature & Precipitation) at the Evanston Burns Field, WY Station (1991–2020) . . . . . 2.4-33

Table 2.4-5 Kemmerer Unit 1 Maximum Hourly Temperatures (2022–2023), 10 Meter Level . . . . . 2.4-34

Table 2.4-6 Kemmerer Unit 1 Minimum Hourly Temperatures (2022–2023), 10 Meter Level . . . . . 2.4-35

Table 2.4-7 Kemmerer Unit 1 Monthly Max., Min., Avg. Temperatures (°F) and Relative Humidity (2022–2023), 10 Meter Level . . . . . 2.4-36

Table 2.4-8 Maximum and Minimum Safety Temperatures, Annual Exceedance Values, and Daily Average Temperatures from Evanston 1E (Wyoming) Station . . . . . 2.4-37

Table 2.4-9 Mean Monthly and Annual Precipitation at Kemmerer Unit 1 and Surrounding Stations . . . . . 2.4-38

Table 2.4-10 Hail Events in Lincoln County . . . . . 2.4-39

Table 2.4-11 High Wind Event Frequency for Kemmerer Unit 1 . . . . . 2.4-40

Table 2.4-12 Distribution of Tornadoes and Strike Probability Within a 2 Degree Area of Kemmerer Unit 1 . . . . . 2.4-41

Table 2.4-13 Mean Monthly and Annual Snowfall (1981–2010) at Sites Around the Kemmerer Unit 1 Site . . . . . 2.4-42

Table 2.4-14 Probable Maximum Precipitation and Probable Maximum Winter Precipitation Values . . . . . 2.4-43

Table 2.4-15 Summary of Normal & Extreme Winter Precipitation Event Results and Resultant Ground Snow Loads . . . . . 2.4-44

Table 2.4-16 Monthly and Annual Percent Frequency of Precipitation Occurrence . . 2.4-45

Table 2.4-17 Monthly and Annual Average Wind Speed . . . . . 2.4-46



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## LIST OF TABLES


Table 2.4-18 Kemmerer Unit 1 Site Meteorological Persistence (2022–2023), 10 Meter Level, 1 Sector . . . . . 2.4-47

Table 2.4-19 Kemmerer Unit 1 Site Meteorological Persistence (2022–2023), 60 Meter Level, 1 Sector . . . . . 2.4-50

Table 2.4-20 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite . . . . . 2.4-53

Table 2.4-21 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite . . . . . 2.4-54

Table 2.4-22 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite . . . . . 2.4-55

Table 2.4-23 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite . . . . . 2.4-56

Table 2.4-24 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite . . . . . 2.4-57

Table 2.4-25 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite . . . . . 2.4-58

Table 2.4-26 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite . . . . . 2.4-59

Table 2.4-27 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite . . . . . 2.4-60

Table 2.4-28 Naughton Power Plant Joint Frequency Distribution 2019-2021, 50-Meter 3-Year Composite . . . . . 2.4-61

Table 2.4-29 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite . . . . . 2.4-62

Table 2.4-30 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite . . . . . 2.4-63

Table 2.4-31 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite . . . . . 2.4-64

Table 2.4-32 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite . . . . . 2.4-65

Table 2.4-33 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite . . . . . 2.4-66

Table 2.4-34 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite . . . . . 2.4-67



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## LIST OF TABLES

Table 2.4-35 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite . . . . . 2.4-68

Table 2.4-36 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-69

Table 2.4-37 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-70

Table 2.4-38 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-71

Table 2.4-39 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-72

Table 2.4-40 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-73

Table 2.4-41 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-74

Table 2.4-42 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-75

Table 2.4-43 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-76

Table 2.4-44 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual . . . . . 2.4-77

Table 2.4-45 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual . . . . . 2.4-78

Table 2.4-46 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual . . . . . 2.4-79

Table 2.4-47 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual . . . . . 2.4-80

Table 2.4-48 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual . . . . . 2.4-81

Table 2.4-49 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual . . . . . 2.4-82

Table 2.4-50 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual . . . . . 2.4-83

Table 2.4-51 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual . . . . . 2.4-84


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## LIST OF TABLES

Table 2.4-52 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-85

Table 2.4-53 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-86

Table 2.4-54 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-87

Table 2.4-55 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-88

Table 2.4-56 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-89

Table 2.4-57 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-90

Table 2.4-58 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-91

Table 2.4-59 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual . . . . . 2.4-92

Table 2.4-60 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual . . . . . 2.4-93

Table 2.4-61 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual . . . . . 2.4-94

Table 2.4-62 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual . . . . . 2.4-95

Table 2.4-63 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual . . . . . 2.4-96

Table 2.4-64 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual . . . . . 2.4-97

Table 2.4-65 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual . . . . . 2.4-98

Table 2.4-66 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual . . . . . 2.4-99

Table 2.4-67 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual . . . . . 2.4-100


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## LIST OF TABLES


Table 2.4-68 Kemmerer Unit 1 and Naughton Power Plant Stability Class Distribution for April 9, 2022–April 8, 2023, and January 1, 2019–December 31, 2021 (Naughton Power Plant Only) . . . . . 2.4-101

Table 2.4-69 Naughton Power Plant 33-Foot (10-Meter) Annual Stability Persistence Summary for Jan 2019–Dec 2021 . . . . . 2.4-102

Table 2.4-70 Naughton Power Plant 164-Foot (50-Meter) Annual Stability Persistence Summary for Jan 2019–Dec 2021 . . . . . 2.4-103

Table 2.4-71 Kemmerer Unit 1 33-Foot (10-Meter) Annual Stability Persistence Summary for April 2022–April 2023 . . . . . 2.4-104

Table 2.4-72 Kemmerer Unit 1 197-Foot (60-Meter) Annual Stability Persistence Summary for April 2022–April 2023 . . . . . 2.4-105

Table 2.4-73 Temperature Inversion Frequency and Persistence, April 2022–April 2023 . . . . . 2.4-106

Table 2.4-74 Monthly Mean Temperatures (°F) at Kemmerer Unit 1 and Surrounding Sites . . . . . 2.4-107

Table 2.4-75 Monthly Mean and Maximum Temperatures (°F) at Kemmerer Unit 1 and Surrounding Sites . . . . . 2.4-108

Table 2.4-76 Monthly, Mean, and Minimum Temperatures (°F) at Kemmerer Unit 1 and Surrounding Sites . . . . . 2.4-109

Table 2.4-77 National Ambient Air Quality Standards . . . . . 2.4-110

Table 2.4-78 ARCON 95th Percentile X/Q Values for the On-Site Control Room Evaluation for the Average Male . . . . . 2.4-112

Table 2.4-79 ARCON 99.5th Percentile X/Q Values at the EAB . . . . . 2.4-113

Table 2.4-80 XOQDOQ Distances from RXB to Nearest Receptors Per Directional Sector . . . . . 2.4-114

Table 2.4-81 XOQ and DOQ at Locations of Interest . . . . . 2.4-115

Table 2.4-82 NI 50-mile X/Q and D/Q Values, No Decay/Undepleted . . . . . 2.4-116

Table 2.4-83 NI 50-mile X/Q and D/Q Values, Decayed/Undepleted . . . . . 2.4-117

Table 2.4-84 NI 50-mile X/Q and D/Q Values, Decayed/Depleted . . . . . 2.4-118

Table 2.4-85 NI 50-mile X/Q and D/Q Values, Deposited . . . . . 2.4-119

Table 2.4-86 EI 50-mile X/Q and D/Q Values, No Decay/Undepleted . . . . . 2.4-120



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## LIST OF TABLES


Table 2.4-87 EI 50-mile X/Q and D/Q Values, Decayed/Undepleted . . . . . 2.4-121

Table 2.4-88 EI 50-mile X/Q and D/Q Values, Decayed/Depleted . . . . . 2.4-122

Table 2.4-89 EI 50-mile X/Q and D/Q Values, Deposited . . . . . 2.4-123

Table 2.4-90 Tower Instrument Specifications and Accuracies for Meteorological Monitoring Program (Preoperational and Operational) . . . . . 2.4-124

Table 2.4-91 Kemmerer Unit 1 Meteorological Percent Data Recovery . . . . . 2.4-125

Table 2.4-92 List of Wildfires in Nearby Region of Site (January 1996-December 31st, 2023) . . . . . 2.4-126

Table 2.5-1 Maximum LIP Water Levels Surrounding At-Grade NSRST Structures . . . . . 2.5-69

Table 2.5-2 Sub-Basin Drainage Areas . . . . . 2.5-70

Table 2.5-3 Composite Curve Numbers for Each Sub-Basin . . . . . 2.5-71

Table 2.5-4 Lag Times for Each Sub-Basin . . . . . 2.5-72

Table 2.5-5 Probable Maximum Precipitation Peak Discharges at the Kemmerer Unit 1 Site . . . . . 2.5-73

Table 2.5-6 Maximum Water Levels during the Design Probable Maximum Precipitation Event (including a 20 percent climate change adjustment factor) . . . . . 2.5-74

Table 2.5-7 Pond Drainage Areas . . . . . 2.5-76

Table 2.5-8 Dam Embankment Data . . . . . 2.5-77

Table 2.5-9 500-year Point Precipitation Depth . . . . . 2.5-78

Table 2.5-10 HEC-HMS 500-year Peak Discharges . . . . . 2.5-79

Table 2.5-11 Input for Breach Parameters . . . . . 2.5-80

Table 2.5-12 Selected HEC-RAS Breach Parameters . . . . . 2.5-81

Table 2.5-13 Seismic Dam Breach + 500-year Flood HEC-RAS Results . . . . . 2.5-82

Table 2.5-14 Lake Arambel Lag Time . . . . . 2.5-83

Table 2.5-15 PMF Peak Discharges . . . . . 2.5-84

Table 2.5-16 Hydrologic failure of the Lake Arambel dam during PMP HEC-RAS Results . . . . . 2.5-85



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## LIST OF TABLES


Table 2.5-17 Maximum Capacity of FGD 4, FGD 5, Lake Arambel, and Raw Water Settling Basin . . . . . 2.5-86

Table 2.5-18 Historical Ice Jams Events at Hams Fork River . . . . . 2.5-87

Table 2.5-19 Historical Ice Jams Events at Green River . . . . . 2.5-89

Table 2.5-20 Historical Ice Jams Events at Bear River . . . . . 2.5-90

Table 2.5-21 Peak AFDD and Estimated Ice Thickness for 'KEMMERER 2 N WY US' Station . . . . . 2.5-91

Table 2.5-22 Peak AFDD and Estimated Ice Thickness for 'FOSSIL BUTTE, WY US' Station . . . . . 2.5-94

Table 2.5-23 Mapped Outcrop Area Percentages of Hydrogeologic Units in the Greater Green River Basin . . . . . 2.5-95

Table 2.5-24 Municipal Groundwater Use for 2005 in the Greater Green River Basin . . . . . 2.5-96

Table 2.5-25 Greater Green River Basin Municipal Service Area Groundwater Use Projections, Low Growth . . . . . 2.5-97

Table 2.5-26 Greater Green River Basin Municipal Service Area Groundwater Use Projections, Moderate Growth . . . . . 2.5-98

Table 2.5-27 Greater Green River Basin Municipal Service Area Groundwater Use Projections, High Growth . . . . . 2.5-99

Table 2.5-28 Borehole Packer Test Results . . . . . 2.5-100

Table 2.5-29 Borehole Packer Test Summary Statistics . . . . . 2.5-103

Table 2.5-30 Hydraulic Conductivity Results for Slug Tests and Long-Term Recovery . . . . . 2.5-104

Table 2.5-31 Porosity and Specific Gravity Test Results . . . . . 2.5-107

Table 2.5-32 Water Sampling Field Parameters . . . . . 2.5-109

Table 2.5-33a Geochemical, Environmental, and Radiometric Analytical Results . . . 2.5-111

Table 2.5-33b Geochemical, Environmental, and Radiometric Analytical Results . . . 2.5-114

Table 2.5-33c Geochemical, Environmental, and Radiometric Analytical Results . . . 2.5-117

Table 2.5-33d Geochemical, Environmental, and Radiometric Analytical Results . . . 2.5-120

Table 2.5-33e Geochemical, Environmental, and Radiometric Analytical Results . . . 2.5-123



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## LIST OF TABLES


Table 2.5-34 Chloride, Sulfate, and pH Analytical Results . . . . . 2.5-126

Table 2.5-35 Construction Information for Observation Wells . . . . . 2.5-127

Table 2.5-36 Monthly Manual Groundwater Levels . . . . . 2.5-128

Table 2.5-37 Monthly Horizontal Hydraulic Gradients . . . . . 2.5-129

Table 2.5-38 Monthly Vertical Hydraulic Gradients . . . . . 2.5-138

Table 2.5-39 Hydrostatic Uplift Pressures for Plant Foundations and Other Safety-Significant SSCs . . . . . 2.5-149

Table 2.5-40 Radionuclides of Interest . . . . . 2.5-150

Table 2.5-41 Radionuclide Activity Concentrations in Groundwater . . . . . 2.5-151

Table 2.5-42 Summary of Water Demand for Naughton Units 1, 2, and 3 and Kemmerer Unit 1 . . . . . 2.5-152

Table 2.5-43 Comparison of Total Water Demand from Kemmerer Unit 1 and Naughton Power Plant with Average Monthly Discharge at Hams Fork . . . . . 2.5-153

Table 2.5-44 Comparison of Total Water Demand from Kemmerer Unit 1 and Naughton Power Plant with Average Monthly Hams Fork Discharge from USGS Daily Data and Daily Outflow at Viva Naughton Reservoir . . . . . 2.5-154

Table 2.6-1 Seismic Source Zone Characteristics . . . . . 2.6-73

Table 2.6-2 Geographic Regions Used to Help Organize the Discussion of Fault Sources . . . . . 2.6-75

Table 2.6-3 Closest Fault Source-to-Site Distances . . . . . 2.6-76

Table 2.6-4 Mean Reference-Rock Hazard Curves (AFE) by Spectral Period at the site . . . . . 2.6-78

Table 2.6-5 Mean and Fractile Hazard Curves (AFE) for 0.01 second Reference Rock Hazard at the site . . . . . 2.6-81

Table 2.6-6 Mean and Fractile Hazard Curves (AFE) for 0.1 second Reference Rock Hazard at the site . . . . . 2.6-82

Table 2.6-7 Mean and Fractile Hazard Curves (AFE) for 1 second Reference Rock Hazard at the site . . . . . 2.6-83

Table 2.6-8 Mean and Fractile Hazard Curves (AFE) for 10 second Reference Rock Hazard at the site . . . . . 2.6-84



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## LIST OF TABLES


Table 2.6-9 Mean Reference Rock UHRS (g) for 11 AFEs at the site . . . . . 2.6-85

Table 2.6-10 Mean Reference M, R, and ε Deaggregation of Reference Rock Hazard for 11 AFEs at the site (Bold Text in LF Row Indicates that Contributions from Distances less than 100 km are Ignored) . . . . . 2.6-86

Table 2.6-11 Percent Contribution of Seismic Sources to 1 Sec Mean Reference-Rock Hazard for Three AFEs at the site . . . . . 2.6-87

Table 2.6-12 Modal M and R Deaggregation of Reference-Rock Hazard for 11 AFEs at the site . . . . . 2.6-88

Table 2.6-13 Mean PSHA Control Point UHRS (g) for 11 AFEs at the site . . . . . 2.6-89

Table 2.6-14 Mean V/H Ratios and PSHA Control Point Vertical UHRS for Three AFEs at the site . . . . . 2.6-90

Table 2.6-15 Horizontal and Vertical Spectral Accelerations for GMRS Horizon and SSE . . . . . 2.6-91

Table 2.6-16 Median and Logarithmic Standard Deviation of Strain-Compatible Shear Wave Velocity and Damping Profiles Corresponding to GMRS . . . . . 2.6-92

Table 2.6-17 Mean GMRS Horizon V/H Ratios Calculated per Gülerce and Abrahamson . . . . . 2.6-94

Table 2.6-18 Summary of Subsurface Investigation Field Work . . . . . 2.6-95

Table 2.6-19 Laboratory Testing Program . . . . . 2.6-96

Table 2.6-20 Summary of Static Engineering Properties . . . . . 2.6-98

Table 2.6-21 Summary of SPT N-Values . . . . . 2.6-99

Table 2.6-22 Summary of Index Test Results . . . . . 2.6-100

Table 2.6-23 Summary of Total Unit Weight and Specific Gravity Test Results . . . . 2.6-101

Table 2.6-24 Summary of Rock Testing . . . . . 2.6-102

Table 2.6-25 Summary of Unconfined Compressive Strength Testing of Rock . . . . 2.6-103

Table 2.6-26 Hoek-Brown Parameters and Rock Mass Strength . . . . . 2.6-104

Table 2.6-27 Low Strain Shear Modulus . . . . . 2.6-105

Table 2.6-28 Summary of Exploration Depths of Borings . . . . . 2.6-106

Table 2.6-29 Summary of Borehole Depths DH Geophysical Surveys . . . . . 2.6-107



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## LIST OF TABLES

Table 2.6-30 Summary of Observation Wells . . . . . 2.6-108

Table 2.6-31 Summary of Permeability Test Locations . . . . . 2.6-109

Table 2.6-32 $V_s$ and $V_p$ Profiles of Compacted Granular Backfill . . . . . 2.6-110

Table 2.6-33 Modulus Reduction and Damping Relationships for Compacted Granular Backfill, In-Situ Soil, and Weathered Rock . . . . . 2.6-111

Table 2.6-34 Summary of Rock Mass Deformation Modulus Estimates . . . . . 2.6-113


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## LIST OF FIGURES


Figure 2.2-1 Population Grid for the 10-Mi Radius (16 Km) . . . . . 2.2-11

Figure 2.2-2 2020 Population Distribution within 10 Mi (16 Km). . . . . 2.2-12

Figure 2.2-3 2030 Population Distribution within 10 Mi (16 Km). . . . . 2.2-13

Figure 2.2-4 2040 Population Distribution within 10 Mi (16 Km). . . . . 2.2-14

Figure 2.2-5 2050 Population Distribution within 10 Mi (16 Km). . . . . 2.2-15

Figure 2.2-6 2060 Population Distribution within 10 Mi (16 Km). . . . . 2.2-16

Figure 2.2-7 2070 Population Distribution within 10 Mi (16 Km). . . . . 2.2-17

Figure 2.2-8 2080 Population Distribution within 10 Mi (16 Km). . . . . 2.2-18

Figure 2.2-9 2090 Population Distribution within 10 Mi (16 Km). . . . . 2.2-19

Figure 2.2-10 Population Grid for the 50-Mi Radius (80 Km) . . . . . 2.2-20

Figure 2.2-11 2020 Population Distribution within 50 Mi (80 Km). . . . . 2.2-21

Figure 2.2-12 2030 Population Distribution within 50 Mi (80 Km). . . . . 2.2-22

Figure 2.2-13 2040 Population Distribution within 50 Mi (80 Km). . . . . 2.2-23

Figure 2.2-14 2050 Population Distribution within 50 Mi (80 Km). . . . . 2.2-24

Figure 2.2-15 2060 Population Distribution within 50 Mi (80 Km). . . . . 2.2-25

Figure 2.2-16 2070 Population Distribution within 50 Mi (80 Km). . . . . 2.2-26

Figure 2.2-17 2080 Population Distribution within 50 Mi (80 Km). . . . . 2.2-27

Figure 2.2-18 2090 Population Distribution within 50 Mi (80 Km). . . . . 2.2-28

Figure 2.2-19 Population Density Compared to NRC Siting Criteria . . . . . 2.2-29

Figure 2.2-20 Kemmerer Unit 1 Site Plan. . . . . 2.2-30

Figure 2.2-21 Site and Property Boundaries . . . . . 2.2-31

Figure 2.2-22 Site Location Map (10 Mi, 16 Km, Radius). . . . . 2.2-32

Figure 2.2-23 Site Region Map (50 Mi, 80 Km, Radius). . . . . 2.2-33

Figure 2.3-1 Nearby Industrial Facilities and Transportation Routes within 10 Mi . . . 2.3-36

Figure 2.3-2 Pipelines within 5 Mi. . . . . 2.3-37

Figure 2.3-3 Airports and Federal Airways within 10 Mi . . . . . 2.3-38



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## LIST OF FIGURES


Figure 2.4-1 Meteorological Observation Stations and Climate Observing Stations 2.4-128

Figure 2.4-2 Naughton Power Plant Annual Precipitation Wind Rose 2019–2021, 10 Meter All . . . . . 2.4-129

Figure 2.4-3 Naughton Power Plant Annual Precipitation Wind Rose 2019–2021, 50 Meter All . . . . . 2.4-130

Figure 2.4-4 Naughton Power Plant Annual Precipitation Wind Rose 2022–2023, 10-Meter All . . . . . 2.4-131

Figure 2.4-5 Naughton Power Plant Annual Precipitation Wind Rose 2022–2023, 50-Meter All . . . . . 2.4-132

Figure 2.4-6 Kemmerer Unit 1 Annual Precipitation Wind Rose 2022–2023, 10 m . . . . . 2.4-133

Figure 2.4-7 Kemmerer Unit 1 Annual Precipitation Wind Rose 2022–2023, 60 m . . . . . 2.4-134

Figure 2.4-8 Naughton Power Plant Annual 3 Year Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-135

Figure 2.4-9 Naughton Power Plant Annual 3 Year Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-136

Figure 2.4-10 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-137

Figure 2.4-11 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-138

Figure 2.4-12 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-139

Figure 2.4-13 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-140

Figure 2.4-14 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-141

Figure 2.4-15 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-142

Figure 2.4-16 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-143

Figure 2.4-17 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-144



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## LIST OF FIGURES

Figure 2.4-18 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-145

Figure 2.4-19 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-146

Figure 2.4-20 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-147

Figure 2.4-21 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-148

Figure 2.4-22 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-149

Figure 2.4-23 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-150

Figure 2.4-24 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-151

Figure 2.4-25 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-152

Figure 2.4-26 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-153

Figure 2.4-27 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-154

Figure 2.4-28 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-155

Figure 2.4-29 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-156

Figure 2.4-30 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-157

Figure 2.4-31 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-158

Figure 2.4-32 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-159

Figure 2.4-33 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-160

Figure 2.4-34 Naughton Power Plant Annual Average Wind Rose 2022–2023, 10 m . . . . . 2.4-161


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## LIST OF FIGURES

Figure 2.4-35 Naughton Power Plant Annual Average Wind Rose 2022–2023, 50 m . . . . . 2.4-162

Figure 2.4-36 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-163

Figure 2.4-37 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-164

Figure 2.4-38 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-165

Figure 2.4-39 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 10 m . . . . . 2.4-166

Figure 2.4-40 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-167

Figure 2.4-41 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-168

Figure 2.4-42 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-169

Figure 2.4-43 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 50 m . . . . . 2.4-170

Figure 2.4-44 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 10 m . . . . . 2.4-171

Figure 2.4-45 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 10 m . . . . . 2.4-172

Figure 2.4-46 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 10 m . . . . . 2.4-173

Figure 2.4-47 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 10 m . . . . . 2.4-174

Figure 2.4-48 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 50 m . . . . . 2.4-175

Figure 2.4-49 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 50 m . . . . . 2.4-176

Figure 2.4-50 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 50 m . . . . . 2.4-177

Figure 2.4-51 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 50 m . . . . . 2.4-178


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## LIST OF FIGURES

Figure 2.4-52 Kemmerer Unit 1 Annual Wind Rose 2022–2023, 10 m . . . . . 2.4-179

Figure 2.4-53 Kemmerer Unit 1 Annual Wind Rose 2022–2023, 60 m . . . . . 2.4-180

Figure 2.4-54 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 10 m . . . . . 2.4-181

Figure 2.4-55 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 10 m . . . . . 2.4-182

Figure 2.4-56 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 10 m . . . . . 2.4-183

Figure 2.4-57 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 10 m . . . . . 2.4-184

Figure 2.4-58 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 60 m . . . . . 2.4-185

Figure 2.4-59 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 60 m . . . . . 2.4-186

Figure 2.4-60 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 60 m . . . . . 2.4-187

Figure 2.4-61 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 60 m . . . . . 2.4-188

Figure 2.4-62 UGRB Ozone Non-Attainment Area and Nearest Class 1 Federal Area . . . . . 2.4-189

Figure 2.4-63 Topographic Map of Kemmerer Unit 1 with Location of Meteorological Tower . . . . . 2.4-190

Figure 2.4-64 Topographic Map of Kemmerer Unit 1 with Location Out to 50 Miles . . . . . 2.4-191

Figure 2.5-1 General Location of the Kemmerer Unit 1 Site and Surrounding Streams . . . . . 2.5-155

Figure 2.5-2 6-Hour Local PMP Depths in 5-minute Increments (including 20 percent increase due to climate change) . . . . . 2.5-156

Figure 2.5-3 Finish Grade and Safety-Significant Structures . . . . . 2.5-157

Figure 2.5-4 Gravel (Blue colored areas) and Building Roofs, Roads and Parking Lots (Grey colored areas) . . . . . 2.5-158


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## LIST OF FIGURES


Figure 2.5-5 Estimated Maximum Water Surface Elevation (in feet NAVD 88) for LIP with Climate Change Adjustment. Red stars are the locations of the Maximum Water Surface Elevation for each Structure. . . . . 2.5-159

Figure 2.5-6 Sub-basins, Streams and Flow Paths at the Kemmerer Unit 1 Site . . . 2.5-160

Figure 2.5-7 Hyetograph of (a) the PMP and (b) the 40% PMP . . . . . 2.5-161

Figure 2.5-8 Sub-basin East Precipitation and Discharge . . . . . 2.5-162

Figure 2.5-9 Sub-basin West Precipitation and Discharge . . . . . 2.5-163

Figure 2.5-10 Sub-basin Laydown Precipitation and Discharge . . . . . 2.5-164

Figure 2.5-11 Sub-basin Confluence Precipitation and Discharge . . . . . 2.5-165

Figure 2.5-12 HEC-RAS Model Cross Sections . . . . . 2.5-166

Figure 2.5-13 Maximum Water Surface Profile Along East Lower (North Folk Little Muddy Creek) and Confluence Reaches with Climate Change (40%PMP+PMP+CC) . . . . . 2.5-167

Figure 2.5-14 Maximum Water Surface Profile along Laydown, West Lower and Confluence Reaches with Climate Change (40%PMP+PMP+CC) . . . 2.5-168

Figure 2.5-15 Fetch Length (Yellow Line) at the Kemmerer Unit 1 Site . . . . . 2.5-169

Figure 2.5-16 HEC-RAS Cross-Sections at Kemmerer Unit 1 Site. . . . . 2.5-170

Figure 2.5-17 HEC-RAS Cross-Section Locations Between Ponds and Kemmerer Unit 1 Site . . . . . 2.5-171

Figure 2.5-18 Schematic Representation of the HEC-RAS model . . . . . 2.5-172

Figure 2.5-19 Schematic Representation of the Downstream Part of the HEC-RAS Model Shown in Figure 2.5-18 . . . . . 2.5-173

Figure 2.5-20 Schematic Representation of the Upstream Part of the HEC-RAS Model Shown in Figure 2.5-18 . . . . . 2.5-174

Figure 2.5-21 Simultaneous Dam Breach of all Three Dams plus 500-Year Flood Peak Discharge. Stage and Flow Hydrograph at Section East Lower 8, Next to the Nuclear Island . . . . . 2.5-175

Figure 2.5-22 Simultaneous Dam Breach of all Three Dams Plus 500-year Flood Peak Discharge. Maximum Water Surface Profile . . . . . 2.5-176

Figure 2.5-23 Lake Arambel Watershed Lag Flow Path . . . . . 2.5-177

Figure 2.5-24 Lake Arambel PMF Runoff Hydrograph . . . . . 2.5-178



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## LIST OF FIGURES


Figure 2.5-25 North Fork Little Muddy Creek (Sub-basin East) PMF Runoff Hydrograph . . . . . 2.5-179

Figure 2.5-26 Hydrologic Failure (Overtopping) of the Lake Arambel Dam during the PMP. Stage and Flow Hydrograph at Section East Lower 8, Next to the Nuclear Island . . . . . 2.5-180

Figure 2.5-27 Maximum Water Surface Profile. Hydrologic Failure (Overtopping) of the Lake Arambel Dam during the PMP . . . . . 2.5-181

Figure 2.5-28 Dam Breach Fetch Length (yellow line) . . . . . 2.5-182

Figure 2.5-29 FGD 5 Fetch Length (yellow line) . . . . . 2.5-183

Figure 2.5-30 FGD 4 Fetch Length (yellow line) . . . . . 2.5-184

Figure 2.5-31 Kemmerer Unit 1 Site Location and NCDC Stations . . . . . 2.5-185

Figure 2.5-32 Existing Raw Water Settling Basin (Settling Pond) at Naughton Power Plant (Reference 2.5-62) . . . . . 2.5-186

Figure 2.5-33 Kemmerer Unit 1 Site and Key Features in its Vicinity . . . . . 2.5-187

Figure 2.5-34 Excerpts from Historical Maps: (a) Hall and Lesley's 1857 Map, (b) Mitchell's 1880 State, County and Township Map of Wyoming Territory, (c) Rand, McNally & Co.'s 1897 State Map of Wyoming. . . . 2.5-188

Figure 2.5-35 Excerpt from USGS Topographic Map Showing the Course of North Fork Little Muddy Creek and Tributaries in 1962 (Sheet Elkol, Wyoming, 1:24 000) . . . . . 2.5-189

Figure 2.5-36 Excerpt from USGS Topographic Map Showing the Course of North Fork Little Muddy Creek and Tributaries in 1979 (Sheet Kemmerer, Wyoming, 1:100 000) . . . . . 2.5-190

Figure 2.5-37 Excerpt from USGS Topographic Map Showing the Course of North Fork Little Muddy Creek and Tributaries in 2012 (Sheet Elkol, Wyoming, 1:24 000) . . . . . 2.5-191

Figure 2.5-38 Excerpt from USGS Topographic Map Showing the Course of North Fork Little Muddy Creek and Tributaries in 2021 (Sheet Elkol, Wyoming, 1:24 000) . . . . . 2.5-192

Figure 2.5-39 Satellite Image of the Vicinity of the Kemmerer Unit 1 Site on December 30, 1985 (source: Google Earth; Light Blue Lines Indicate Streams based on the Current Topography) . . . . . 2.5-193

Figure 2.5-40 Satellite Image of the Vicinity of the Kemmerer Unit 1 Site on August 24, 1994 (source: Google Earth; Light Blue Lines Indicate Streams based on the Current Topography) . . . . . 2.5-194



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## LIST OF FIGURES


Figure 2.5-41 Satellite Image of the Vicinity of the Kemmerer Unit 1 Site on July 10, 2006 (source: Google Earth; Light Blue Lines Indicate Streams based on the Current Topography) . . . . . 2.5-195

Figure 2.5-42 Satellite Image of the Vicinity of the Kemmerer Unit 1 Site on November 4, 2020 (source: Google Earth; Light Blue Lines Indicate Streams based on the Current Topography) . . . . . 2.5-196

Figure 2.5-43 Daily Discharge on Hams Fork River below the Viva Naughton Reservoir . . . . . 2.5-197

Figure 2.5-44 Landslide Incidence Map (Reference 2.5-79). The Circle Encloses an Area of 5-mile Radius. . . . . 2.5-198

Figure 2.5-45 Deep-Seated Landslide Susceptibility Map (Reference 2.5-76). . . . . 2.5-199

Figure 2.5-46 Surficial Materials Map Showing a Landslide Area along Hams Fork River near Kemmerer (excerpt from Reference 2.5-78). . . . . 2.5-200

Figure 2.5-47 Kemmerer Unit 1 Site Watershed. . . . . 2.5-201

Figure 2.5-48 Cross Section A-A Defined in Figure 2.5-47 . . . . . 2.5-202

Figure 2.5-49 The Blazon Gap Watershed . . . . . 2.5-203

Figure 2.5-50 The North Fork Little Muddy Creek Close to Blazon Gap . . . . . 2.5-204

Figure 2.5-51 Closeup of the East Bank Face of the North Fork Little Muddy Creek showing Interbedded Silt and Gravel Deposits . . . . . 2.5-205

Figure 2.5-52 The Stream Draining the Blazon Gap Watershed . . . . . 2.5-206

Figure 2.5-53 Schematic Map Showing the Linear Extent of Riprap Revetment along the East Side of the Nuclear Island . . . . . 2.5-207

Figure 2.5-54 Location Map for Kemmerer Unit 1 Site . . . . . 2.5-208

Figure 2.5-55 Regional Physiographic Provinces in the Kemmerer Unit 1 Site Vicinity . . . . . 2.5-209

Figure 2.5-56 Greater Green River Basin and the Kemmerer Unit 1 Site . . . . . 2.5-210

Figure 2.5-57 Geologic Features within the Greater Green River Basin . . . . . 2.5-211

Figure 2.5-58 Not Used . . . . . 2.5-212

Figure 2.5-59 Not Used . . . . . 2.5-213

Figure 2.5-60 Not Used . . . . . 2.5-214



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## LIST OF FIGURES


Figure 2.5-61 Outcrop Areas and Hydrogeologic Units by Geologic Era in Greater Green River Basin . . . . . 2.5-215

Figure 2.5-62 Potential for Recharge by Direct Infiltration of Precipitation to Mesozoic Hydrogeologic Units in the Greater Green River Basin . . . . 2.5-216

Figure 2.5-63 Hydrogeologic Nomenclature for Mesozoic Rocks in the Greater Green River Basin in Wyoming . . . . . 2.5-217

Figure 2.5-64 Mean Annual Precipitation for Lincoln County, Wyoming, 1951-80 . . . 2.5-218

Figure 2.5-65 Location of Groundwater-Dependent Towns with Relative to the Kemmerer Unit 1 Site . . . . . 2.5-219

Figure 2.5-66 Permitted Groundwater Extraction Wells within a 5-Mile Radius of the Kemmerer Unit 1 Site (DOM_GW = domestic groundwater; STK = livestock; IND_GW = industrial groundwater; MIS = miscellaneous) . . . . . 2.5-220

Figure 2.5-67 Locations of Borehole Packer Tests . . . . . 2.5-221

Figure 2.5-68 Rock Hydraulic Conductivity as a Function of Depth Below Ground Surface . . . . . 2.5-222

Figure 2.5-69 Rock Porosity as a Function of Depth Below Ground Surface . . . . . 2.5-223

Figure 2.5-70 Rock Specific Gravity as a Function of Depth Below Ground Surface . . . . . 2.5-224

Figure 2.5-71 Observation Well Locations . . . . . 2.5-225

Figure 2.5-72 Hydrographs for Observation Wells OW-106-U and M . . . . . 2.5-226

Figure 2.5-73 Hydrographs for Observation Wells OW-108-U and M . . . . . 2.5-227

Figure 2.5-74 Hydrographs for Observation Wells OW-109-U and M . . . . . 2.5-228

Figure 2.5-75 Hydrographs for Observation Wells OW-117-U, M, and L (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download) . . . . . 2.5-229

Figure 2.5-76 Hydrographs for Observation Wells OW-119-U and M . . . . . 2.5-230

Figure 2.5-77 Hydrographs for Observation Wells OW-121-U and M . . . . . 2.5-231

Figure 2.5-78 Hydrographs for Observation Wells OW-122-U, M, and L (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download) . . . . . 2.5-232

Figure 2.5-79 Hydrographs for Observation Wells OW-127-U and M . . . . . 2.5-233



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## LIST OF FIGURES


Figure 2.5-80 Hydrographs for Observation Wells OW-136-U and M . . . . . 2.5-234

Figure 2.5-81 Hydrographs for Observation Wells OW-137-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download). . . . . 2.5-235

Figure 2.5-82 Hydrographs for Observation Wells OW-140-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download). . . . . 2.5-236

Figure 2.5-83 Hydrographs for Observation Wells OW-149-U and M . . . . . 2.5-237

Figure 2.5-84 Hydrographs for Observation Wells OW-153-U, M, and L . . . . . 2.5-238

Figure 2.5-85 Hydrographs for Observation Wells OW-164-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download). . . . . 2.5-239

Figure 2.5-86 Hydrographs for Observation Wells OW-167-U, M, and L . . . . . 2.5-240

Figure 2.5-87 Hydrographs for Observation Wells OW-168-U and M . . . . . 2.5-241

Figure 2.5-88 Hydrographs for Observation Wells OW-179-U and M . . . . . 2.5-242

Figure 2.5-89 Hydrographs for Observation Wells OW-185-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download). . . . . 2.5-243

Figure 2.5-90 Hydrographs for observation wells OW-186-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download). . . . . 2.5-244

Figure 2.5-91 Hydrographs for Observation Wells OW-187-U and M . . . . . 2.5-245

Figure 2.5-92 Hydrographs for Observation Wells OW-188-U and M . . . . . 2.5-246

Figure 2.5-93 Hydrographs for Observation Wells OW-189-U and M . . . . . 2.5-247

Figure 2.5-94 Hydrographs for Observation Wells OW-190-U and M . . . . . 2.5-248

Figure 2.5-95 Hydrographs for Observation Wells OW-191-U and M . . . . . 2.5-249

Figure 2.5-96 Hydrographs for PZSW and PZGW . . . . . 2.5-250

Figure 2.5-97 Groundwater Level Contour Map and Flow Paths for August 17, 2022 (U Series Wells) . . . . . 2.5-251

Figure 2.5-98 Groundwater Level Contour Map and Flow Paths for August 17, 2022 (M Series Wells) . . . . . 2.5-252



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## LIST OF FIGURES


Figure 2.5-99 Groundwater Level Contour Map and Flow Paths for September 20-21, 2022 (U Series Wells) . . . . . 2.5-253

Figure 2.5-100 Groundwater Level Contour Map and Flow Paths for September 20-21, 2022 (M Series Wells) . . . . . 2.5-254

Figure 2.5-101 Groundwater Level Contour Map and Flow Paths for October 26-27, 2022 (U Series Wells) . . . . . 2.5-255

Figure 2.5-102 Groundwater Level Contour Map and Flow Paths for October 26-27, 2022 (M Series Wells) . . . . . 2.5-256

Figure 2.5-103 Groundwater Level Contour Map and Flow Paths for November 29-30, 2022 (U Series Wells) . . . . . 2.5-257

Figure 2.5-104 Groundwater Level Contour Map and Flow Paths for November 29-30, 2022 (M Series Wells) . . . . . 2.5-258

Figure 2.5-105 Groundwater Level Contour Map and Flow Paths for December 28-29, 2022 (U Series Wells) . . . . . 2.5-259

Figure 2.5-106 Groundwater Level Contour Map and Flow Paths for December 28-29, 2022 (M Series Wells) . . . . . 2.5-260

Figure 2.5-107 Groundwater Level Contour Map and Flow Paths for January 24-25, 2023 (U Series Wells) . . . . . 2.5-261

Figure 2.5-108 Groundwater Level Contour Map and Flow Paths for January 24-25, 2023 (M Series Wells) . . . . . 2.5-262

Figure 2.5-109 Groundwater Level Contour Map and Flow Paths for February 27-28 and March 1, 2023 (U Series Wells) . . . . . 2.5-263

Figure 2.5-110 Groundwater Level Contour Map and Flow Paths for February 27-28 and March 1, 2023 (M Series Wells) . . . . . 2.5-264

Figure 2.5-111 Groundwater Level Contour Map and Flow Paths for March 29-30, 2023 (U Series Wells) . . . . . 2.5-265

Figure 2.5-112 Groundwater Level Contour Map and Flow Paths for March 29-30, 2023 (M Series Wells) . . . . . 2.5-266

Figure 2.5-113 Groundwater Level Contour Map and Flow Paths for April 25-26, 2023 (U Series Wells) . . . . . 2.5-267

Figure 2.5-114 Groundwater Level Contour Map and Flow Paths for April 25-26, 2023 (M Series Wells) . . . . . 2.5-268

Figure 2.5-115 Average Vertical Hydraulic Gradients for the U, M, and L Series Wells . . . . . 2.5-269




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## LIST OF FIGURES


Figure 2.5-116 Kemmerer Unit 1 Site Plan . . . . . 2.5-270

Figure 2.5-117 Conceptual Model for Radionuclide Transport in Groundwater . . . . . 2.5-271

Figure 2.5-118 Pearson Type 3 Distribution Probability Density Function for Viva Naughton Reservoir Annual Low Level and Best Fit Line . . . . 2.5-272

Figure 2.6-1 Kemmerer Unit 1 Site Location Map . . . . . 2.6-114

Figure 2.6-2 Map of Regional Physiographic Provinces . . . . . 2.6-115

Figure 2.6-3 Generalized Geologic Map of the Site Region (320 km Radius) . . . . . 2.6-116

Figure 2.6-4 Generalized Map of North American Cordillera . . . . . 2.6-117

Figure 2.6-5 Generalized Geologic Map of Site Vicinity (40 km radius) . . . . . 2.6-118

Figure 2.6-6 Generalized Stratigraphic Columns of Wyoming-Idaho-Utah Thrust Belt . . . . . 2.6-119

Figure 2.6-7 Stratigraphic Column of Elkol Quadrangle . . . . . 2.6-120

Figure 2.6-8 Geologic Map of Site Area (8 km Radius) . . . . . 2.6-121

Figure 2.6-9 Geologic Map of Site Location (1 km Radius) . . . . . 2.6-122

Figure 2.6-10 Geologic Map of Wyoming Salient of the Sevier Orogenic Belt . . . . . 2.6-123

Figure 2.6-11 Cross Sections of the Sevier Orogeny in Wyoming . . . . . 2.6-124

Figure 2.6-12 Photograph of Frontier Formation . . . . . 2.6-125

Figure 2.6-13 GPS Tracks and Field Stations . . . . . 2.6-126

Figure 2.6-14 Rock Core . . . . . 2.6-127

Figure 2.6-15 Photograph of Kemmerer Unit 1 Site . . . . . 2.6-128

Figure 2.6-16 Coal Beds in Adaville Formation . . . . . 2.6-129

Figure 2.6-17 Photograph of Hams Fork Conglomerate . . . . . 2.6-130

Figure 2.6-18 Cross Section of the Elkol Quadrangle . . . . . 2.6-131

Figure 2.6-19 Looking Across Cumberland Flats . . . . . 2.6-132

Figure 2.6-20 Photograph of Qg . . . . . 2.6-133

Figure 2.6-21 SSHAC Project Region Showing 320 km radii . . . . . 2.6-134



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## LIST OF FIGURES


Figure 2.6-22 Regional Earthquake Catalog for Magnitude E[M] ≥ 2.3 . . . . . 2.6-135

Figure 2.6-23 Master Logic Tree for the SSC Model . . . . . 2.6-136

Figure 2.6-24 Map of Seismic Source Zones and Fault Sources in the Site Region . . . . . 2.6-137

Figure 2.6-25 Map of Distant Seismic Sources Beyond the Site Region showing Distances from the Kemmerer Unit 1 Site . . . . . 2.6-138

Figure 2.6-26 Seismic Source Zones Defined for the Site Region . . . . . 2.6-139

Figure 2.6-27 Master Logic Tree for Seismic Source Zones . . . . . 2.6-140

Figure 2.6-28 Contours of Seismogenic Thickness in the Site Region. . . . . 2.6-141

Figure 2.6-29 Contribution of Buoyancy in the Upper Mantle to Average Topographic Elevation . . . . . 2.6-142

Figure 2.6-30 Heat Flow in the Site Region . . . . . 2.6-143

Figure 2.6-31 Faults in the Site Region from the USGS QFFD . . . . . 2.6-144

Figure 2.6-32 Fault Sources in the Site Region from the USGS NSHM23. . . . . 2.6-145

Figure 2.6-33 Fault Sources in the Site Region Subdivided by Group or Region. . . . 2.6-146

Figure 2.6-34 Master Logic Tree for Fault Sources . . . . . 2.6-147

Figure 2.6-35 Clustered Fault Sources (Red); Other Fault Sources (Grey) . . . . . 2.6-148

Figure 2.6-36 Temporal Distribution of Surface-rupturing Earthquakes on Faults in the Wyoming Salient region, showing apparent cluster of Earthquakes in mid- to late Holocene . . . . . 2.6-149

Figure 2.6-37 Temporal Distribution of Surface-rupturing Earthquakes on Faults in the Wyoming Salient region, showing apparent cluster of Earthquakes in mid- to late Holocene . . . . . 2.6-150

Figure 2.6-38 Rock Creek Fault Mapping and Fault Source . . . . . 2.6-151

Figure 2.6-39 Master Logic Tree for the Rock Creek Fault Source . . . . . 2.6-152

Figure 2.6-40a In-Cluster Logic Tree for the Rock Creek Fault Source . . . . . 2.6-153

Figure 2.6-40b In-Cluster Logic Tree for the Rock Creek Fault Source, Detail . . . . . 2.6-154

Figure 2.6-41 Out-of-Cluster Logic Tree for the Rock Creek Fault Source. . . . . 2.6-155

Figure 2.6-42 Along-strike Distribution of Vertical Separations for the Rock Creek Fault from Interpretation of LiDAR Scarp Profiles . . . . . 2.6-156



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## LIST OF FIGURES


Figure 2.6-43 LiDAR-Based Estimates of Vertical Surface Displacement (in m) Across the Southern Rock Creek Fault (Red) from LiDAR-Based Topographic Profiles 57-61 . . . . . 2.6-157

Figure 2.6-44 LiDAR-Based Mapping of the Rock Creek Fault at Cook Canyon . . . . 2.6-158

Figure 2.6-45 Rock Creek Fault Source Slip Rates Bar Chart . . . . . 2.6-159

Figure 2.6-46 Logic Tree for the Bear River Fault Source . . . . . 2.6-160

Figure 2.6-47 Greys River Fault Mapping and Fault Source . . . . . 2.6-161

Figure 2.6-48 Logic Tree for the Greys River Fault Source . . . . . 2.6-162

Figure 2.6-49 Fault sources in the Wyoming Salient region (blue); other fault sources (grey) . . . . . 2.6-163

Figure 2.6-50 Fault sources in the Teton region (yellow); other fault sources (grey) . . . . . 2.6-164

Figure 2.6-51 Fault sources in the central Wyoming region (green); other fault sources (grey) . . . . . 2.6-165

Figure 2.6-52 Wasatch fault system logic tree for rupture models and rupture sources . . . . . 2.6-166

Figure 2.6-53 Wasatch fault system logic tree for rupture models and rupture sources for central segments . . . . . 2.6-167

Figure 2.6-54 Wasatch fault system logic tree for recurrence, slip rates, and time dependence . . . . . 2.6-168

Figure 2.6-55 Wasatch logic tree for $M_{CHAR}$ for segmented rupture models. . . . . 2.6-169

Figure 2.6-56 Map of Wasatch fault system sources . . . . . 2.6-170

Figure 2.6-57 Cascadia subduction interface source from INL SSHAC 2022 (Reference 2.6-27) . . . . . 2.6-171

Figure 2.6-58 Mean reference rock hazard curves by spectral period at the site . . . . 2.6-172

Figure 2.6-59 Contribution of source types to 0.01 second mean reference rock hazard at the site . . . . . 2.6-173

Figure 2.6-60 Contribution of source types to 0.1 second mean reference rock hazard at the site . . . . . 2.6-174

Figure 2.6-61 Contribution of source types to 1 second mean reference rock hazard at the site . . . . . 2.6-175



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## LIST OF FIGURES

Figure 2.6-62 Contribution of source types to 10 second mean reference rock hazard at the site . . . . . 2.6-176

Figure 2.6-63 Mean and fractiles of 0.01 second total reference rock hazard at the site . . . . . 2.6-177

Figure 2.6-64 Mean and fractiles of 0.1 second total reference rock hazard at the site . . . . . 2.6-178

Figure 2.6-65 Mean and fractiles of 1 second total reference rock hazard at the site . . . . . 2.6-179

Figure 2.6-66 Mean and fractiles of 10 second total reference rock hazard at the site . . . . . 2.6-180

Figure 2.6-67 Mean reference rock UHRS for eleven AFEs at the site . . . . . 2.6-181

Figure 2.6-68 Contribution of M, R Pairs to Mean 1E-4/yr 0.1 second Reference Rock Hazard at the site . . . . . 2.6-182

Figure 2.6-69 Contribution of M, R Pairs to Mean 1E-4/yr 1 second Reference Rock Hazard at the site . . . . . 2.6-183

Figure 2.6-70 Contribution of M, R Pairs to Mean 1E-5/yr 0.1 second Reference Rock Hazard at the site . . . . . 2.6-184

Figure 2.6-71 Contribution of M, R Pairs to Mean 1E-5/yr 1 second Reference Rock Hazard at the site . . . . . 2.6-185

Figure 2.6-72 Contribution of M, R Pairs to Mean 1E-6/yr 0.1 second Reference Rock Hazard at the site . . . . . 2.6-186

Figure 2.6-73 Contribution of M, R Pairs to Mean 1E-6/yr 1 second Reference Rock Hazard at the site . . . . . 2.6-187

Figure 2.6-74 Seven Median SAF Models Corresponding to Each of the Discretized Fractiles of the Distribution of Epistemic Uncertainty for Input Level 5 corresponding to an AFE of 1E-5 . . . . . 2.6-188

Figure 2.6-75 Mean Horizontal UHRS at PSHA Control Point for 11 AFEs at the site . . . . . 2.6-189

Figure 2.6-76 GA11 V/H Ratios at PSHA Control Point for an AFE of 1E-4/yr at the site . . . . . 2.6-190

Figure 2.6-77 GA11 Mean V/H Ratios at PSHA Control Point for AFEs of 1E-4, 1E-5, and 1E 6/yr at the site . . . . . 2.6-191

Figure 2.6-78 Mean Horizontal and Vertical UHRS at PSHA Control Point for AFEs of 1E-4, 1E-5, and 1E-6/yr at the site . . . . . 2.6-192



SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054
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## LIST OF FIGURES


Figure 2.6-79 Site Horizontal GMRS and UHRS at GMRS Horizon . . . . . 2.6-193

Figure 2.6-80 Damping Ratio and Shear Modulus Reduction Curves for GMRS Calculation . . . . . 2.6-194

Figure 2.6-81 Not Used . . . . . 2.6-195

Figure 2.6-82 Maximum Shear Strain Profiles Corresponding to 1E-4 and 1E-5 AFE UHRS at GMRS Horizon . . . . . 2.6-196

Figure 2.6-83 Strain-Compatible Shear Wave Velocity Profiles Corresponding to 1E-4 and 1E-5 AFE UHRS at GMRS Horizon . . . . . 2.6-197

Figure 2.6-84 Strain-Compatible Damping Ratio Profiles Corresponding to 1E-4 and 1E-5 AFE UHRS at GMRS Horizon . . . . . 2.6-198

Figure 2.6-85 Mean V/H Ratios for Vertical GMRS and SSE Calculation . . . . . 2.6-199

Figure 2.6-86 Site Horizontal and Vertical GMRS and SSE . . . . . 2.6-200

Figure 2.6-87 All VS profiles from Downhole (DH), Suspension logging (B) and SASW measurements (Array) . . . . . 2.6-201

Figure 2.6-88 Geologic Map of Site Location . . . . . 2.6-202

Figure 2.6-89 Geologic Section of Elkol Quadrangle . . . . . 2.6-203

Figure 2.6-90 Stratigraphic Column of Elkol Quadrangle . . . . . 2.6-204

Figure 2.6-91 Summary of RQD Values . . . . . 2.6-205

Figure 2.6-92 Summary of Rock Core Recovery . . . . . 2.6-206

Figure 2.6-93 Summary of Hoek-Brown Material Constant . . . . . 2.6-207

Figure 2.6-94 Summary Joint Roughness and Joint Alteration . . . . . 2.6-208

Figure 2.6-95 P-S and DH $V_s$ Profiles . . . . . 2.6-209

Figure 2.6-96 P-S and DH $V_p$ Profiles . . . . . 2.6-210

Figure 2.6-97 Short SASW Arrays . . . . . 2.6-211

Figure 2.6-98 Long, Sitewide SASW Arrays . . . . . 2.6-212

Figure 2.6-99 Combined $V_s$ Profiles . . . . . 2.6-213

Figure 2.6-100 Mean $V_s$ Profiles . . . . . 2.6-214

Figure 2.6-101 Modulus Reduction and Damping – Fresh Rock . . . . . 2.6-215



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## LIST OF FIGURES

Figure 2.6-102 Boring Location Plan . . . . . 2.6-216
Figure 2.6-103 Boring Location Plan Exploration Schedule . . . . . 2.6-217
Figure 2.6-104 East/West Geologic Cross Section through RXB . . . . . 2.6-218
Figure 2.6-105 North/South Cross Geologic Cross Section – RXB/NCB . . . . . 2.6-219
Figure 2.6-106 Post-Construction Groundwater Contour Map, Base Model . . . . . 2.6-220
Figure 2.6-107 Liquefaction Assessment – Polito . . . . . 2.6-221
Figure 2.6-108 Excavation Sections – E/W through FHB/RXB/RAB . . . . . 2.6-222
Figure 2.6-109 Excavation Sections – N/S through RXB/NCB and FHB . . . . . 2.6-223
Figure 2.6-110 Combined At-Rest Lateral Earth Pressures on Below Grade Walls . . . 2.6-224
Figure 2.6-111 Geologic Map of the Site Area . . . . . 2.6-225
Figure 2.6-112 Existing Site Topography . . . . . 2.6-226
Figure 2.6-113 NI Rough Grade Plan . . . . . 2.6-227


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# Chapter 2 Site Information

## 2.1 Site Characteristics and Site Parameters

Chapter 2 provides a description of the demographic, meteorological, hydrological, and seismological characteristics of the Kemmerer Unit 1 Site and the surrounding area, as well as current and projected population distributions, land use, and access control to surrounding areas.

Consistent with Principal Design Criteria 2, as described in Section 5.3, site-specific natural phenomena have been characterized in applicable sections of Chapter 2. Design basis hazard levels (DBHLs) are established using the site characterization information and are described in Section 6.1.1. Table 2.1-1 provides a summary of the site-specific external design basis hazards, and Table 2.1-2 provides a summary of the site characteristics. The site-specific external design basis hazards and site characteristics, which have been determined based on analyses presented in Section 2.2 through Section 2.7, are those necessary to establish the findings required by 10 CFR 50 and 10 CFR 100. Table 2.1-1 and Table 2.1-2 also contain references to the section where additional information is located.


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**Table 2.1-1 Kemmerer Unit 1 Site-Specific External DBHLs**

| Hazard | External DBHL | Section |
|---|---|---|
| Design Wind Speed | 110 mph (3-second gust) for Risk Category III structures, systems, and components (SSCs)<br>115 mph (3-second gust) for Risk Category IV SSCs<br>Wind Exposure Category C | 2.4.1.3.2<br>6.4.2.2.2 |
| Tornadoes and Tornado Missile Spectra | | |
| Maximum Wind Speed | 160 mph (72 m/s) | 2.4.1.3.3 |
| Maximum Translational Speed | 32 mph (14 m/s) | 2.4.1.3.3 |
| Maximum Rotational Speed | 128 mph (57 m/s) | 2.4.1.3.3 |
| Radius of Maximum Rotational Speed | 150 ft (45.7 m) | 2.4.1.3.3 |
| Pressure Drop | 0.6 psi (40 millibars) | 2.4.1.3.3 |
| Rate of Pressure Drop | 0.2 psi/s (13 millibars/s) | 2.4.1.3.3 |
| Tornado Missile Spectra | RG 1.76, Rev. 1, Table 2 for Region III<br><u>Tornado Horizontal Missile Velocity:</u><br>Automobile: 79 ft/s (54 mph)<br>Pipe: 79 ft/s (54 mph)<br>Sphere: 20 ft/s (13.6 mph)<br><br><u>Tornado Vertical Missile Velocity:</u><br>Automobile: 52.93 ft/s (36.1 mph)<br>Pipe: 52.93 ft/s (36.1 mph)<br>Sphere: 13.4 ft/s (9.1 mph) | 2.4.1.3.3<br>6.4.2.2.2 |
| Seismic | | |
| Ground Motion Response Spectra | Figure 2.6-79 | 2.6.2 |
| Peak ground acceleration | 0.297 g horizontal<br>0.209 g vertical | Table 2.6-15 |
| External Flooding | | |
| Design Basis Flood (including wave run-up) from external events, not including Local Intense Precipitation (Local Probable Maximum Precipitation) | 2.6 ft (6,755.4 ft NAVD 88) below plant elevation (6,758.0 ft NAVD 88) | 2.5.1.3.5 |
| Local Intense Precipitation Design Basis Flood | 0.2 ft (6,757.8 ft NAVD 88) below plant elevation (6,758.0 ft NAVD 88) | 2.5.1.1.7 |
| Extreme Winter Precipitation | 70.72 psf | 2.4.1.3.5 |


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**Table 2.1-2 Kemmerer Unit 1 Site Characteristics**
**(Sheet 1 of 3)**

| Site Characteristic | Kemmerer Unit 1 Value | Section |
|---|---|---|
| Geography and Demography | | |
| Exclusion Area Boundary | 0.25 mi (0.4 km) | 2.2.2.1 |
| Low Population Zone | 0.25 mi (0.4 km) | 2.2.2.1 |
| Population Center Distance | 63 mi (101 km) | 2.2.3.5 |
| Nearby Industrial, Transportation, and Military Facilities | | |
| Site Proximity Missiles (Except Aircraft) | Probability of occurrence is < $1x10^{-7}$ per year[1] | 2.3.3.1.4 |
| Aircraft Hazards (Accidental) | Annual aircraft crash probability is < $1x10^{-7}$ | 2.3.3.1.6 |
| Chemical Hazards | < 1 psi at SR SSCs or Probability of occurrence < $1x10^{-7}$ per year | 2.3.3.1.4 |
| Meteorology | | |
| Hurricanes and Hurricane Missiles | The site is located in Wyoming, inland from coastal areas. Due to the site location, there are no direct impacts from hurricanes and hurricane missiles at the site. | 2.4.1.3.4<br>6.4.2.2.3 |
| Ambient Design Temperature, 0.4% Annual Exceedance | 86.5° F Dry Bulb<br>55.4° F Coincident Wet Bulb | Table 2.4-8 |
| Ambient Design Temperature, 99.6% Annual Exceedance | -0.7° F Minimum Dry Bulb | Table 2.4-8 |
| Maximum and Minimum Temperatures 100-Year Return Period | 103.6° F Maximum Dry Bulb<br>-32.8° F Minimum Dry Bulb | Table 2.4-8 |
| Minimum Distance from Release Point to Exclusion Area Boundary (EAB) | 0.19 mi (300 m) | Table 2.4-80 |
| Meteorological Dispersion (X/Q) | | |
| X/Q, Short-Term Accident, EAB | | |
| 2 hr @ EAB | 1.69E-04 $s/m^3$ (North and N-NW) | Table 2.4-79 |
| 8 hr @ EAB | 1.52E-04 $s/m^3$ (North) | Table 2.4-79 |
| 24 hr @ EAB | 8.71E-05 $s/m^3$ (N-NW) | Table 2.4-79 |
| 96 hr @ EAB | 6.21E-05 $s/m^3$ (N-NW) | Table 2.4-79 |
| 720 hr @ EAB | 4.29E-05 $s/m^3$ (N-NW) | Table 2.4-79 |


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**Table 2.1-2 Kemmerer Unit 1 Site Characteristics**
**(Sheet 2 of 3)**

| Site Characteristic | Kemmerer Unit 1 Value | Section |
|---|---|---|
| Maximum Main Control Room X/Q, Accident (95th Percentile) | | |
| 0-2 hr | 3.45E-03 s/$m^3$ (Reactor Building [RXB])<br>3.97 E-03 s/$m^3$ (Reactor Auxiliary Building [RAB])<br>2.77E-03 s/$m^3$ (Fuel Handling Building [FHB]) | Table 2.4-78 |
| 2-8 hr | 3.01E-03 s/$m^3$ (RXB)<br>3.51E-03 s/$m^3$ (RAB)<br>2.28E-03 s/$m^3$ (FHB) | Table 2.4-78 |
| 8-24 hr | 7.53-04 s/$m^3$ (RXB)<br>9.02E-04 s/$m^3$ (RAB)<br>4.73E-04 s/$m^3$ (FHB) | Table 2.4-78 |
| 1-4 day | 9.41E-04 s/$m^3$ (RXB)<br>1.11E-03 s/$m^3$ (RAB)<br>6.36E-04 s/$m^3$ (FHB) | Table 2.4-78 |
| 4-30 day | 8.09E-04 s/$m^3$ (RXB)<br>9.38E-04 s/$m^3$ (RAB)<br>5.56E-04 s/$m^3$ (FHB) | Table 2.4-78 |
| Annual Average X/Q at Site Boundary, Undecayed, Undepleted | 5.128E-05 s/$m^3$ (Nuclear Island [NI])<br>1.896E-04 s/$m^3$ (Energy Island [EI]) | Table 2.4-81 |
| Hydrology | | |
| Maximum Groundwater Level (under deep foundation structures in NI) | 6,738 ft NAVD 88 | Table 2.4-39 |
| Advective Travel Time for Groundwater Flow | 2,627 days | 2.5.3.2.3.2 |
| Waste Holdup Tank Inventory of Radionuclides that Could Seep into the Groundwater and Corresponding Kd Values | Table 2.5-40 | 2.5.3.2 |
| Geology, Seismology, and Geotechnical | | |
| Static and Dynamic Bearing Capacities | RXB - 72.5 to 76 ksf*<br>FHB - 6 to 72.5 ksf*<br>NI Control Building - 19 ksf<br>* Varies based on structure foundation elevation | 2.6.4.10.1 |


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**Table 2.1-2 Kemmerer Unit 1 Site Characteristics**
**(Sheet 3 of 3)**

| Site Characteristic | Kemmerer Unit 1 Value | Section |
|---|---|---|
| Minimum Shear Wave Velocity | Figure 2.6-13 | 2.6.4.7.2 |
| Minimum Soil Angle of Internal Friction | 35 degrees | 2.6.4.2.5 |
| Settlement | < 0.5 in | 2.6.4.10.2 |
| Differential Displacement | < 0.25 in | 2.6.4.10.2 |
| Liquefaction | None | 2.6.4.8 |
| Capable Tectonic Structures or Sources | Yes | 2.6.1 |
| Volcanic Activity | The following volcanic phenomena are precluded from reaching the site:<br>• Potential for the opening of a new volcanic vent<br>• Proximal hazards<br>• Lava flows<br>• Pyroclastic density flows<br>Volcanic Tephra-Falls: < 1 x $10^{-5}$/year<br>Debris Flows: No flood condition | 2.7 |
| Turbine Missiles | | |
| Turbine Missiles | The turbine generator is located on the EI in the Turbine Facility Building approximately 500 ft south of the NI, with its rotor planned to be oriented in the north-south direction. Based on this location and orientation, the potential for damage from turbine missiles is negligible. | |

[1] A frequency analysis was not conducted for scenarios where the blast overpressure level was determined to be less than 1.0 psi per RG 1.91.



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## 2.2 Geography and Demography

### 2.2.1 Site Location and Description

#### 2.2.1.1 Specification of Location

The Kemmerer Unit 1 Site is located in the Elkol, Wyoming, 7.5-minute quadrangle in Zone 12T in Lincoln County (Reference 2.2-1). The plant and its supporting infrastructure is located on approximately 178 acres. The location of principal plant structures, including the Reactor Building (RXB) and release points on the site are identified in Figure 2.2-20. The RXB centerpoint is located at the coordinates shown in Table 2.2-1.

The site is approximately 3.8 miles (6.2 kilometers) from the PacifiCorp Naughton Power Plant in Lincoln County, Wyoming. The Naughton Power Plant is comprised of three electrical generating units with two coal units (Naughton 1 and 2 with nameplate capacities of 156 megawatts and 201 megawatts, respectively) and a natural gas-fired unit (Naughton 3 with a nameplate capacity of 247 megawatts) (Reference 2.2-3). It is approximately 3 miles (4.8 kilometers) south of the Kemmerer city municipal boundary and approximately 4.2 miles (6.8 kilometers) southwest from the town of Diamondville in Lincoln County, Wyoming (distances are measured from the RXB centerpoint to the closest municipal boundary).

The site lies south of the intersection of US 30 and US 189 and east of the intersection of the Union Pacific Railway and US 189. North Fork Little Muddy Creek runs east of the property boundary.

#### 2.2.1.2 Site Area Map

Figure 2.2-21 provides a site area map that identifies the site and property boundaries, as well as the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) (coincident with one another) and adjacent transportation routes. Figure 2.2-22 and Figure 2.2-23 depict the areas within a 10 mile (16 kilometer) and 50 mile (80 kilometer) radius of the site, respectively, and identify transportation routes, mines, industrial and military facilities, wetlands and water ways, and federal lands (e.g., Bureau of Land Management, Bureau of Reclamation, Department of Defense, National Forest Service, National Park Service).

In addition to US 189 and US 30, identified previously, there are minor roads in the immediate vicinity of the site such as County Road 325, Elkol Road, and site access and service roads for the Kemmerer Mine. The Union Pacific railway is approximately 0.15 miles (0.24 kilometers) from the site at its closest approach. The Kemmerer Municipal Airport boundary is approximately 7.1 miles (11 kilometers) north of the site.

### 2.2.2 Exclusion Area Authority and Control

#### 2.2.2.1 Authority

Figure 2.2-21 identifies the EAB and LPZ for the Kemmerer Unit 1. The EAB and LPZ are both defined by a 0.25-mile (0.4-kilometer, 400 meter) radius from the Kemmerer Unit 1 RXB centerpoint. US SFR Owner, LLC (USO) will own the Kemmerer Unit 1 property and will have the


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authority to determine all activities therein including exclusion and removal of personnel and property from the area. In certain places, the EAB extends beyond the Kemmerer Unit 1 property boundary (Figure 2.2-21). For the areas within the EAB but outside of USO ownership, USO will maintain agreements with the appropriate affected entities as part of the Kemmerer Unit 1 Emergency Plan, as indicated in Section 11.3. The agreements will ensure appropriate plans are in place that include protective actions within the areas outside of USO ownership, as warranted.

USO will acquire all mineral rights for the site. The vicinity has potentially exploitable minerals including bentonite, coal, phosphorus, sulfur, oil, and gas (Reference 2.2-7 and Reference 2.2-8). However, there are no mines or oil and gas wells at or directly adjacent to the site (Reference 2.2-9). The nearest active mineral extraction site is the Kemmerer Mine with the closest point of the Kemmerer Mine's permit boundary being approximately 2.2 miles (3.5 kilometers) west of the site.

#### 2.2.2.2 Control of Activities Unrelated to Plant Operation

As shown on Figure 2.2-21, portions of the EAB and LPZ are outside of the property boundary. The Sodium Test and Fill Facility is adjacent to the site and within the EAB and LPZ. Although owned by TerraPower, USO has control of all activities occurring within the EAB.

#### 2.2.2.3 Arrangements for Traffic Control

As shown on Figure 2.2-21, there are no Federal, State or County roads that traverse the exclusion area, and no arrangements for traffic control are required.

#### 2.2.2.4 Abandonment or Relocation of Roads

As shown in Figure 2.2-21, there are no public roads within the EAB; therefore, no public roads are required for relocation or abandonment.

### 2.2.3 Population Distribution

The population distribution, within a 50-mile (80 kilometer) radius, based on the year 2020 US Census Bureau decennial resident data was estimated (Reference 2.2-6). The population distribution is broken into 10 concentric bands (0 to 1 mile, 1 to 2 mile, 2 to 3 mile, 3 to 4 mile, 4 to 5 mile, 5 to 10 mile, 10 to 20 mile, 20 to 30 mile, 30 to 40 mile, and 40 to 50 mile) from the RXB centerpoint and within a grid of 16 directional sectors, each direction consisting of 22.5 degrees. Population is projected for 2030 (projected date of facility approval) and each decade thereafter until 2090 by calculating an annualized growth rate using the 2020 census data and state population projection data (by county). The transient population (primarily temporary housing and recreational area users) was estimated and included in the resident data for 0 to 10 miles. Population density is also calculated for comparison against 10 CFR 100 criteria.

#### 2.2.3.1 Resident Population within 10 Miles

Figure 2.2-22 shows the general location of site and other features within 10 miles (16 kilometers) of the site. Kemmerer is the largest community within 10 miles (16 kilometers) of the site. The town of Diamondville also lies within 10 miles (16 kilometers) of the site.


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The resident population distribution within 10 miles (16 kilometers) of the site is determined based on the 2020 US Census Bureau decennial resident data. The resident population in each of the 16 directional sectors and 6 concentric bands described above is calculated by overlaying the 2020 census block data on the grid shown in Figure 2.2-1 and summing all the census blocks proportional to the area of the block within the grid element. Nonresident transient populations are determined as described in Section 2.2.3.3 and added to each grid as appropriate.

Once the 2020 population (resident and transient, as appropriate) is determined for each grid element, population projections are made in 10-year increments from 2030 to 2090. Annual growth rates of the county populations are estimated using the population projections from the respective state agencies (Reference 2.2-4 and Reference 2.2-5). For each sector the calculated county annual growth rate is applied, taking into account the percentage of each sector that each county occupied. It is important to note that for any county with a negative growth rate, a growth ratio of one is used to produce conservative results. That is, using a growth ratio of one assumes no decline in population and overestimates growth, which is conservative.

The resident and transient 10-mile (16-kilometer) radius population for 2020 and projections for 2030 through 2090 are shown in Figure 2.2-2 through Figure 2.2-9 and are summarized in Table 2.2-2.

### 2.2.3.2 Resident Population Between 10 and 50 Miles

The area defined by a 50-mile (80-kilometer) radius from the RXB centerpoint (Figure 2.2-10) includes all or part of four counties in Wyoming (Lincoln, Sublette, Sweetwater, Uinta), one county in Idaho (Bear Lake), and four counties in Utah (Cache, Morgan, Rich, Summit). Estimates of the 2020 decennial census resident population between 10 and 50 miles (16 and 80 kilometers) from the site were computed using the same methodology used to develop the 10-mile (16 kilometers) radius population distribution, except that transient populations were not quantified.

The population grid up to 50 miles (80 kilometers) is shown in Figure 2.2-10. The 10-to 50-mile (16-to 80-kilometer) population for 2020 and projections for 2030 through 2090 are shown in Figure 2.2-11 through Figure 2.2-18. The 10-to-50-mile (16-to 80-kilometer) population and projections are summarized in Table 2.2-3.

### 2.2.3.3 Transient Population

The transient population segment includes individuals in the workforce, recreational areas, hotels and motels, recreational vehicle parks as well as seasonal residents and migrant populations. Major employers in the area include the Naughton Power Plant and Kemmerer Operations, LLC coal mine, Cowboy State Trucking, South Lincoln Medical Center, and South Lincoln Nursing Center. Major employers account for 263 transients within the 10-mile (16-kilometer) radius. In addition to including employees as transients, recreational facilities in the area are used by persons from locations outside the 10-mile (16-kilometer) radius. Recreational opportunities account for 1,190 transients within the 10 mile (16 kilometer) radius. The majority (1,000 transients) are visitors attending the annual Oyster Ridge Music Festival held in


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Kemmerer. Other recreational opportunities in Kemmerer that attract visitors to the area include the JC Penney Historic District National Historic Landmark, Fossil Country Frontier Museum, and Fossil Island Golf Club.

Also included in the determination of transients in the area are those people occupying temporary housing. Temporary housing includes those who reside in the area on a temporary basis. Hotels, motels, and recreational vehicle parks account for 548 transients within the 10-mile (16-kilometer) radius. Based on US Census Bureau information on seasonal housing, an estimated 64 individuals reside within the 10-mile (16-kilometer) radius on a seasonal basis.

Migrant workers are another category of transients that is considered in this analysis. The U.S. Department of Agriculture reports that there are three farms in Lincoln County employing migrant labor (Reference 2.2-2). Representatives of the University of Wyoming - Lincoln County Extension and the Wyoming Department of Workforce Services indicated that there are no crop farms or livestock ranches within the 10-mile (16-kilometers) radius. However, in a review Lincoln County parcels for the nearest residents in a 5-mile radius of the site, a parcel owned by Aggie Grazing, LLC is located 2.79 miles from the RXB centerpoint. There are also livestock trails used by local ranchers to move their livestock (mostly sheep) from winter to summer ranges and back. Those ranchers employ migrant workers as sheepherders. A maximum of about 15 sheepherders could be present at one time. Given the mobile nature of the activity, the 15 migrant worker transients were associated with each of the sectors along the trails.

The transient population was added to the resident distribution and projected for future years (Figure 2.2-2 through Figure 2.2-9).

#### 2.2.3.4 Low Population Zone

The LPZ for the site consists of the area falling within an approximately 0.25-mile (0.4-kilometer) radius of the reactor centerpoint. The LPZ is the same as the EAB. Figure 2.2-21 provides a representation of the LPZ that includes topographic features, as well as transportation routes (i.e., highways, railways, and waterways). There is no resident or transient population within the LPZ. There are no towns, recreational facilities, hospitals, schools, prisons, or beaches within the LPZ. No public roads traverse the LPZ.

Design basis accidents are evaluated in Chapter 3 to demonstrate that doses at the LPZ are within the dose limits of 10 CFR 100.21(c) and 10 CFR 50.34(a)(1)(ii).

#### 2.2.3.5 Population Center

The closest population center (population of greater than 25,000) is the city of Logan, Utah, which is approximately 63 miles (101 kilometers) west of the site. Logan had a 2020 resident population of 52,778 (Reference 2.2-6). The city of Rock Springs, Wyoming, is approximately 66 miles (106 kilometers) east of the site. Rock Springs had a 2020 population of 23,526 (Reference 2.2-6). Rock Springs may exceed 25,000 residents and meet the population center criterion during the projection period; however, it is farther away than the Logan, Utah, population center.


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The distance to the boundary of the Logan population center (approximately 63 miles, 101 kilometers) is more than 250 times the radius of the 0.25-mile (0.4-kilometer) LPZ and is greater than the minimum distance of one and one-third times the distance from the reactor to the outer boundary of the LPZ as required by 10 CFR 100.21(b).

#### 2.2.3.6 Population Density

Figure 2.2-18 is a plot of population density to radial distance from the plant. Three curves, one actual and two projected, were plotted to illustrate that the vicinity population density is well below the regulatory guidance for population density. The three curves show the cumulative population in 2020 within 20 miles (32 kilometers) of the site and projected cumulative populations in years 2030 (approximate date of facility approval) and 2090 (end of operations). On the same figure, spanning the same radial distances, the regulatory guidance population curve is plotted for a hypothetical density of 500 persons per square mile. Based on these projections, population densities, averaged over any radial distance out to 20 miles (32 kilometers), are expected to be less than 500 persons per square mile over the lifetime of the plant. The cumulative populations are summarized in Table 2.2-4.

### References


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2.2-2 U.S. Department of Agriculture, “2017 Census of Agriculture,” Wyoming State and County Data, Issued April 2019, https://www.nass.usda.gov/Publications/AgCensus/2017/Full_Report/Volume_1,_Chapter_2_County_Level. Accessed October 12, 2022. 2019.

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2.2-9 Department of Homeland Security, *Homeland Infrastructure Foundation-Level Data (HIFLD) Oil and Natural Gas Wells*, https://hifld-geoplatform.opendata.arcgis.com/datasets/geoplatform::oil-and-natural-gas-wells/explore?location=41.703166%2C-110.560188%2C14.00, accessed June 29, 2023, September 2019



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**Table 2.2-1 Kemmerer Unit 1 RXB Centerpoint Coordinates**

| | |
|---|---|
| Geographic, Decimal Degrees, NAD83 | 41.705841 degrees N (Latitude),<br>110.560547 degrees W (Longitude) |
| UTM Zone 12, Meters, NAD83 | 4,617,210.152 m N<br>12T 536,561.647 m E |
| NAD83 State Plane Wyoming West FIPS 4909, U.S. Feet | Northing (X): 767,772.00 ft<br>Easting (Y): 2,494,363.00 ft |



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**Table 2.2-2 Resident and Transient 10 Mi Radius Population for 2020 and Projections for 2030 through 2090**

| Year | 10-Mi Radius Population |
|---|---|
| 2020 | 5,757 |
| 2030 | 6,187 |
| 2040 | 6,647 |
| 2050 | 7,162 |
| 2060 | 7,687 |
| 2070 | 8,275 |
| 2080 | 8,880 |
| 2090 | 9,550 |



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**Table 2.2-3 0-to-50-Mi Population and Projections**

| Year | 10-to-50-Mi Radius Population |
|---|---|
| 2020 | 25,195 |
| 2030 | 25,559 |
| 2040 | 25,953 |
| 2050 | 26,380 |
| 2060 | 26,845 |
| 2070 | 27,347 |
| 2080 | 27,894 |
| 2090 | 28,485 |


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**Table 2.2-4 Cumulative Population**

| Radius (mi) | 2020 Census | Projected 2030 | Projected 2090 | Population equivalent to 500 persons/sq mi |
|---|---|---|---|---|
| 0 to 1 | 240 | 256 | 400 | 1,571 |
| 1 to 2 | 300 | 320 | 500 | 6,283 |
| 2 to 3 | 348 | 372 | 580 | 14,137 |
| 3 to 4 | 536 | 575 | 891 | 25,133 |
| 4 to 5 | 1,223 | 1,314 | 2,031 | 39,270 |
| 5 to 10 | 5,757 | 6,187 | 9,550 | 157,080 |
| 10 to 20 | 5,905 | 6,347 | 9,795 | 628,319 |


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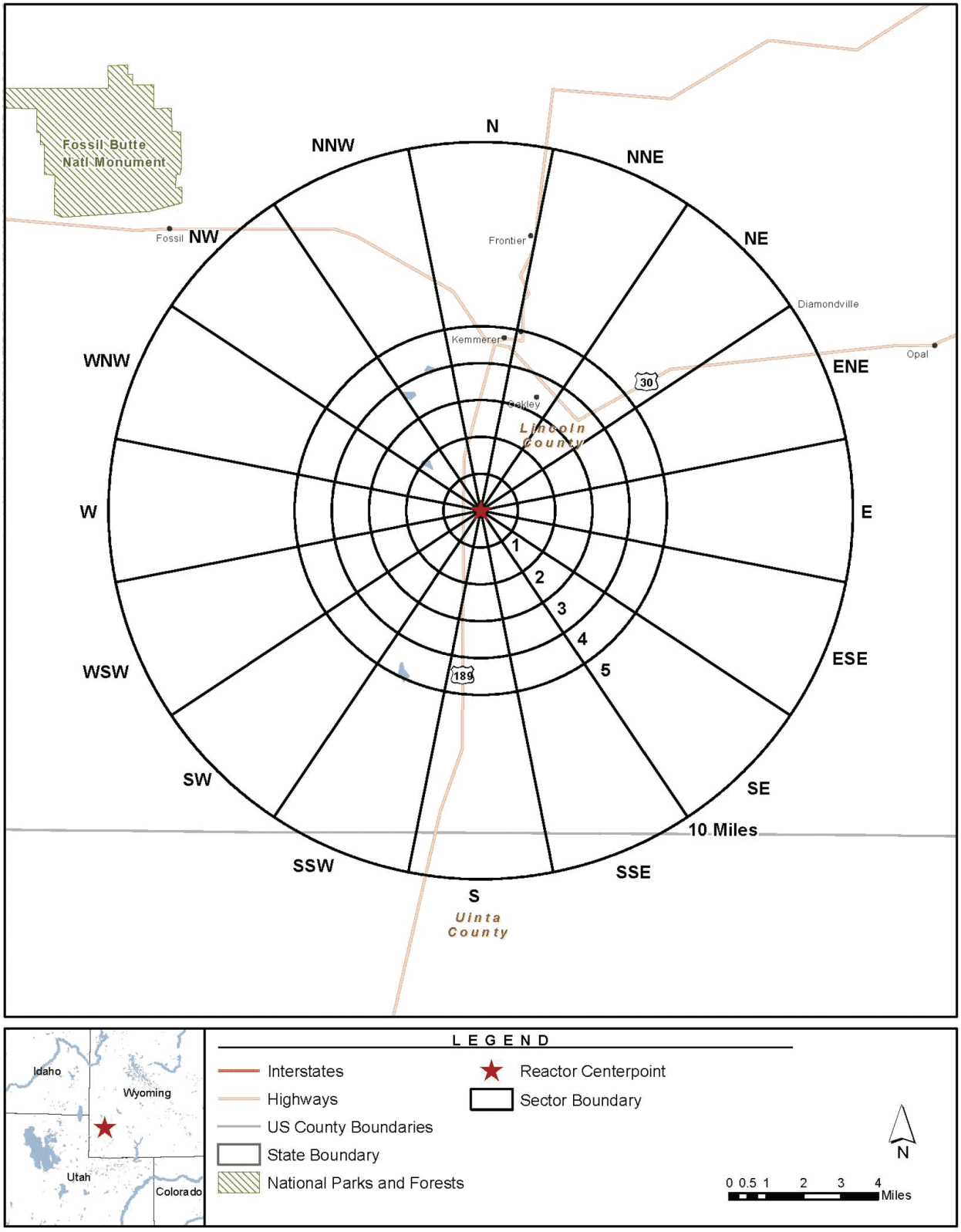
**Figure 2.2-1 Population Grid for the 10-Mi Radius (16 Km)**





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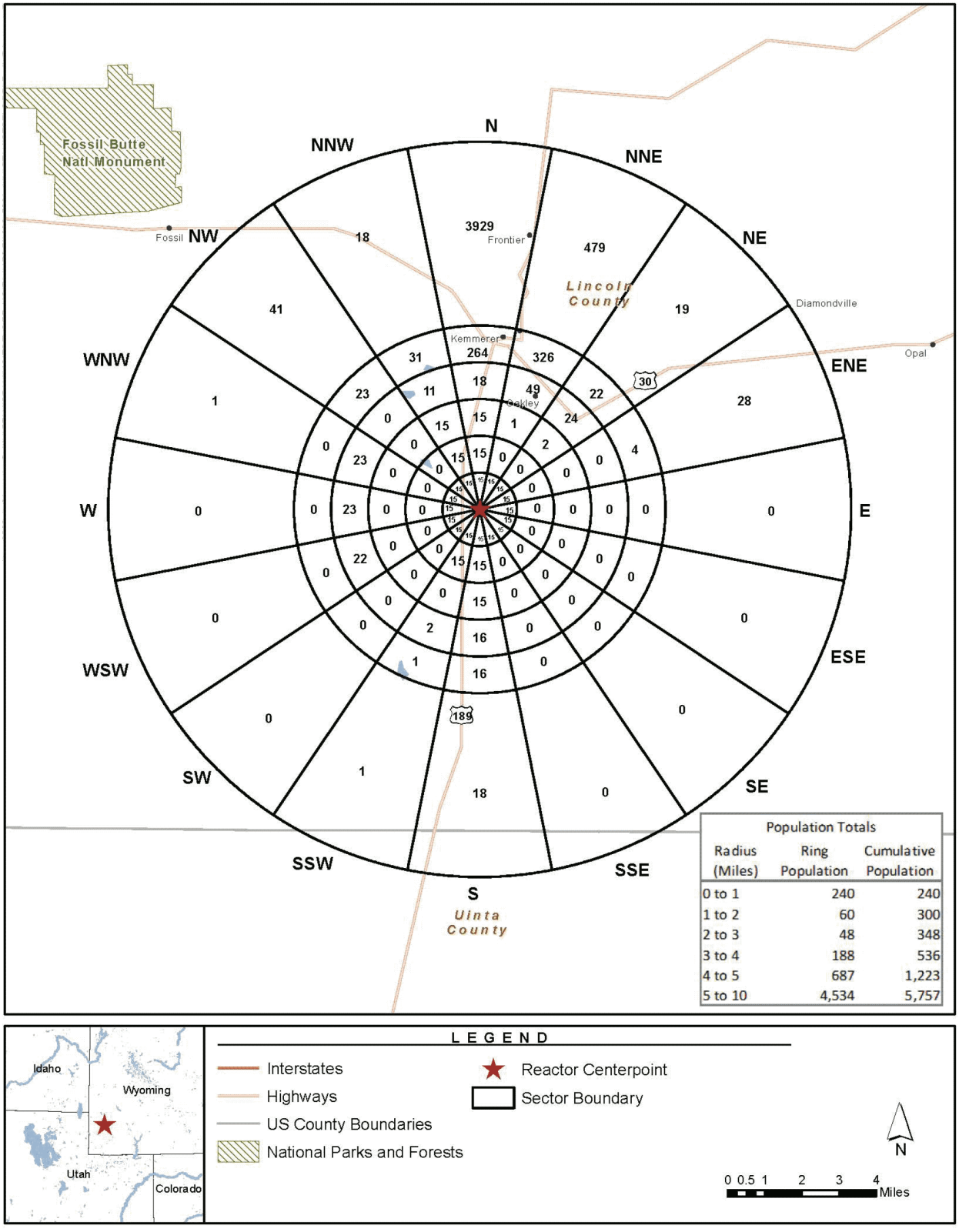
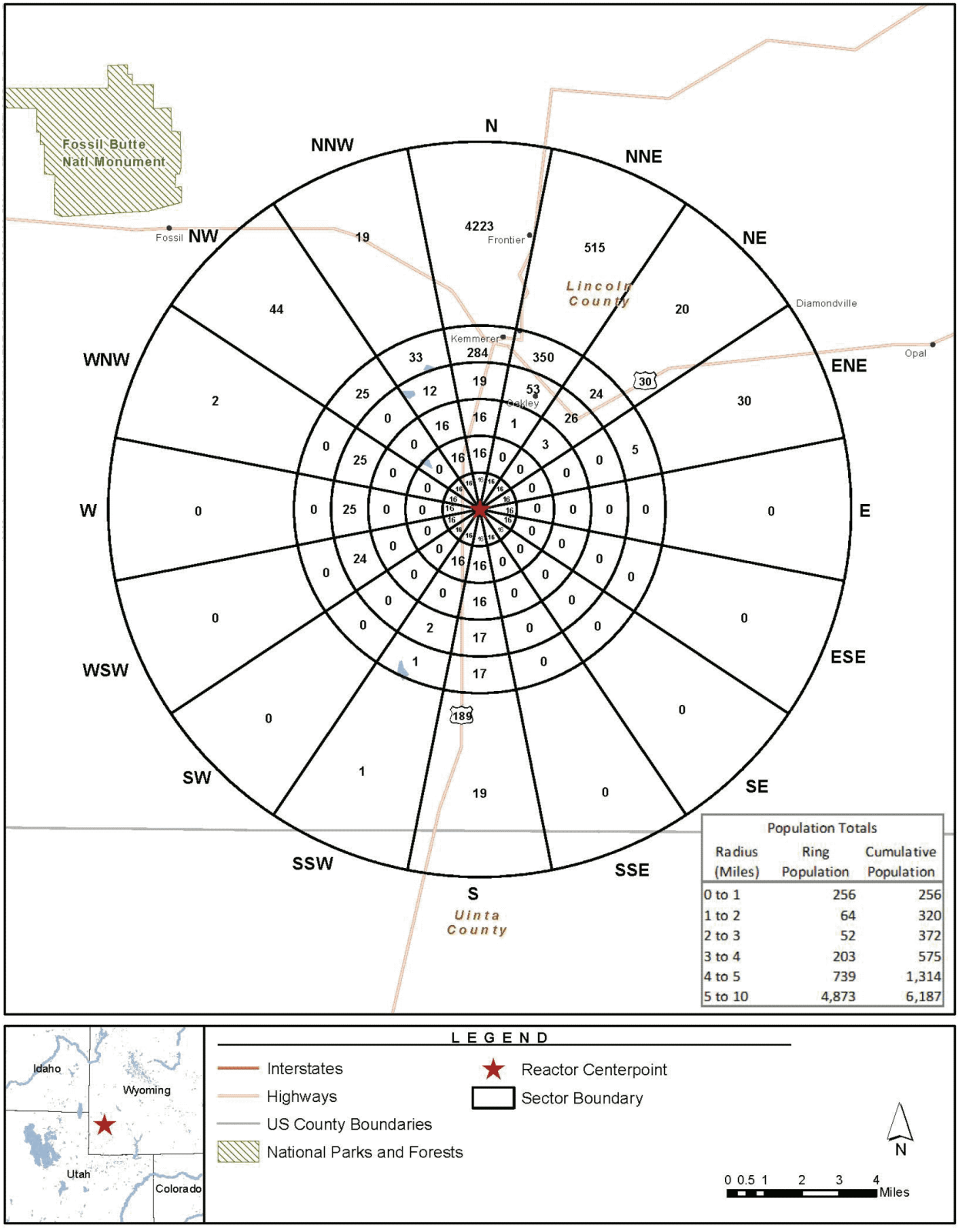
## Figure 2.2-2 2020 Population Distribution within 10 Mi (16 Km)


Fossil Butte Natl Monument
Fossil
Frontier
Kemmerer
Oakley
Diamondville
Opal
Lincoln County
Uinta County
30
189
N
NNE
NE
ENE
E
ESE
SE
SSE
S
SSW
SW
WSW
W
WNW
NW
NNW
3929
479
19
28
0
0
1
18
41
1
0
0
0
1
18
0
0
1
16
16
2
0
0
15
15
15

## Figure 2.2-3 2030 Population Distribution within 10 Mi (16 Km)





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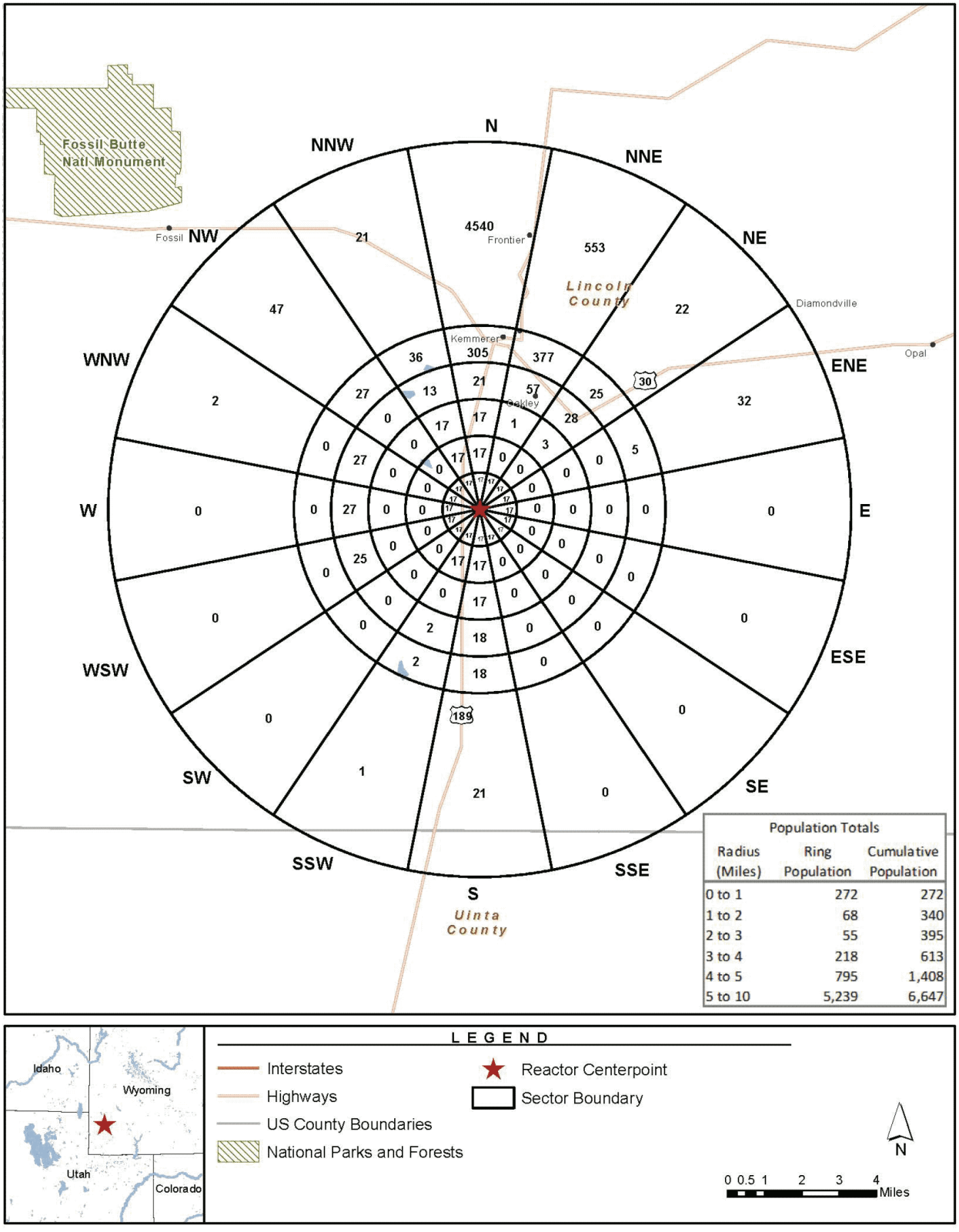
## Figure 2.2-4 2040 Population Distribution within 10 Mi (16 Km)





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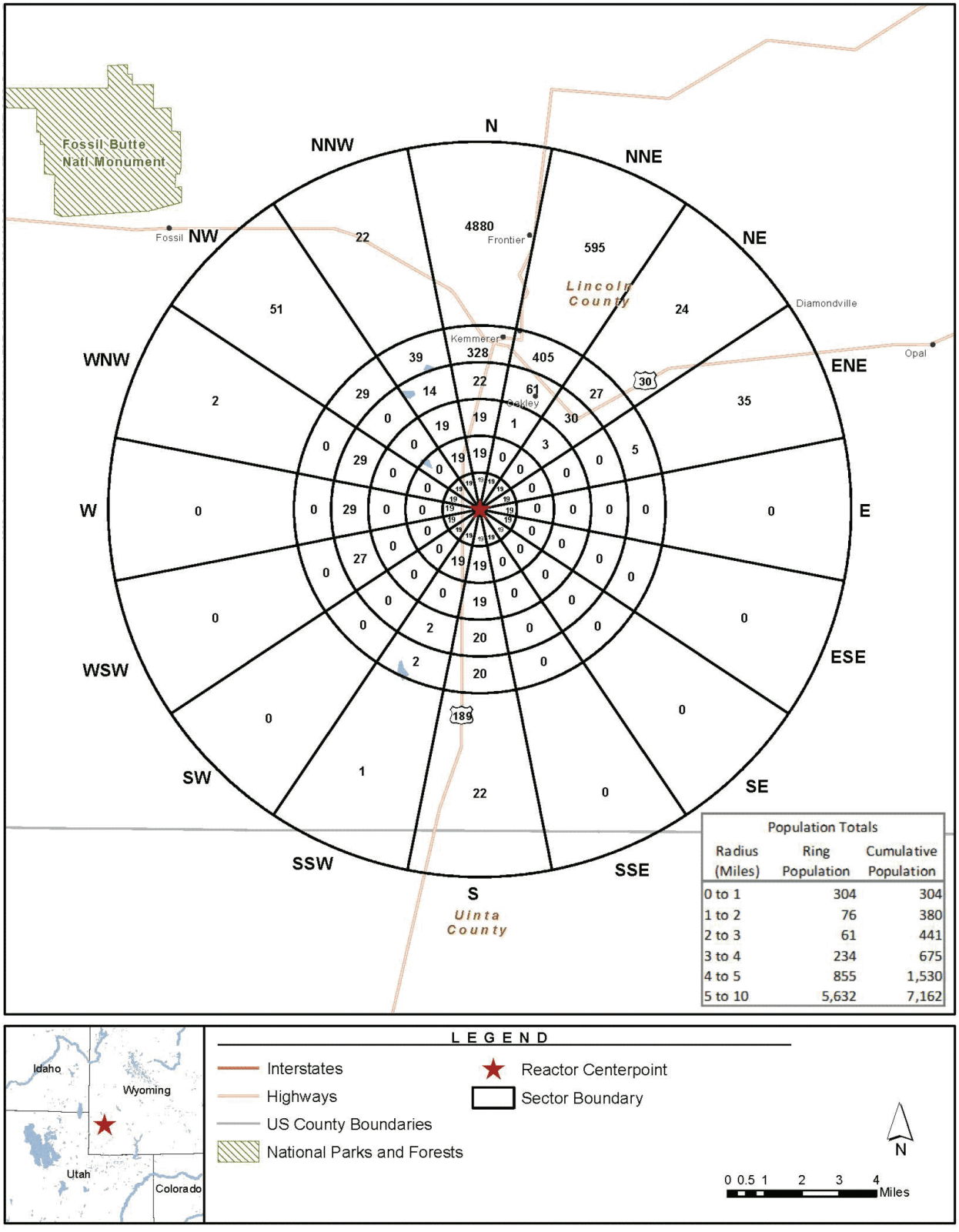
## Figure 2.2-5 2050 Population Distribution within 10 Mi (16 Km)





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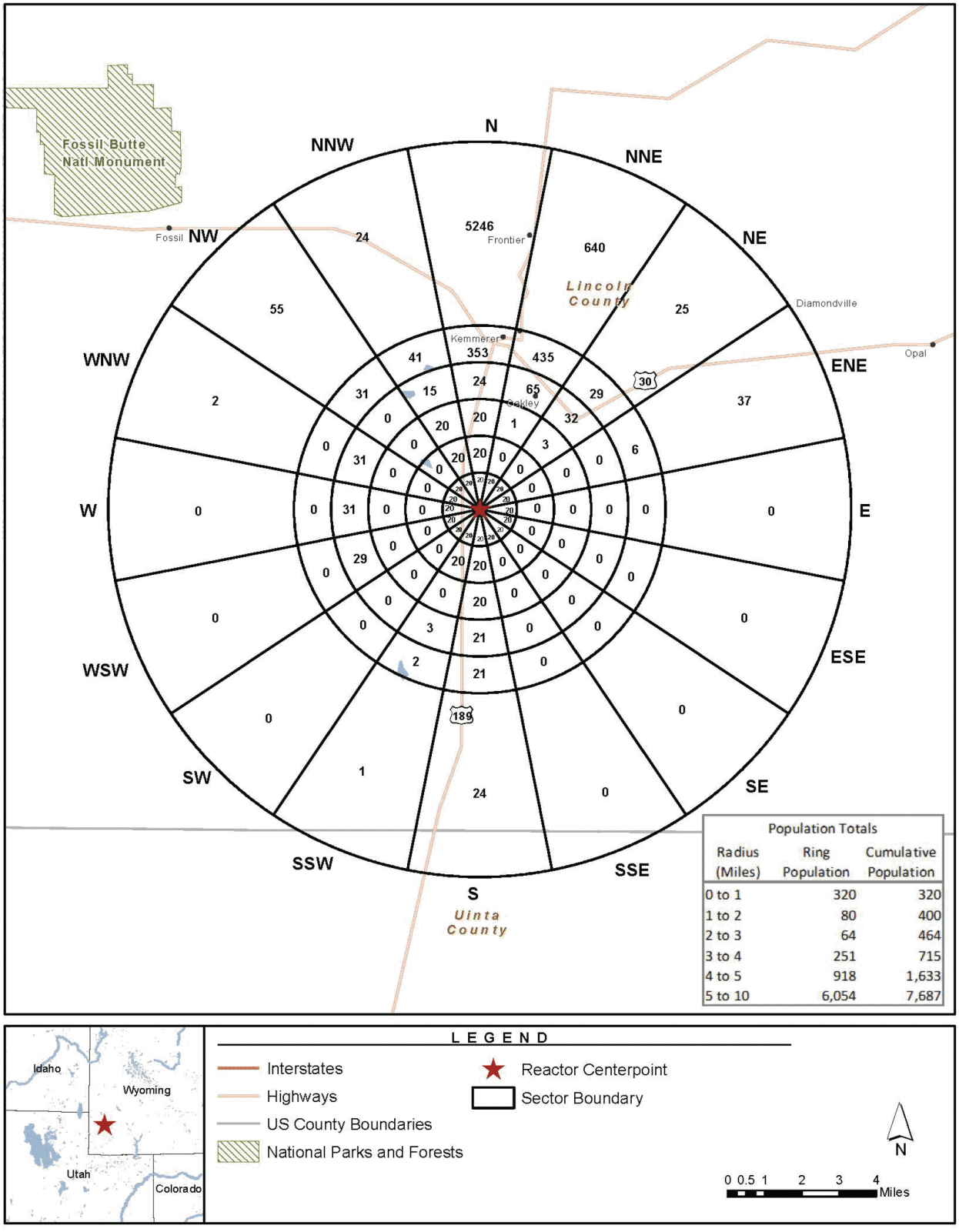
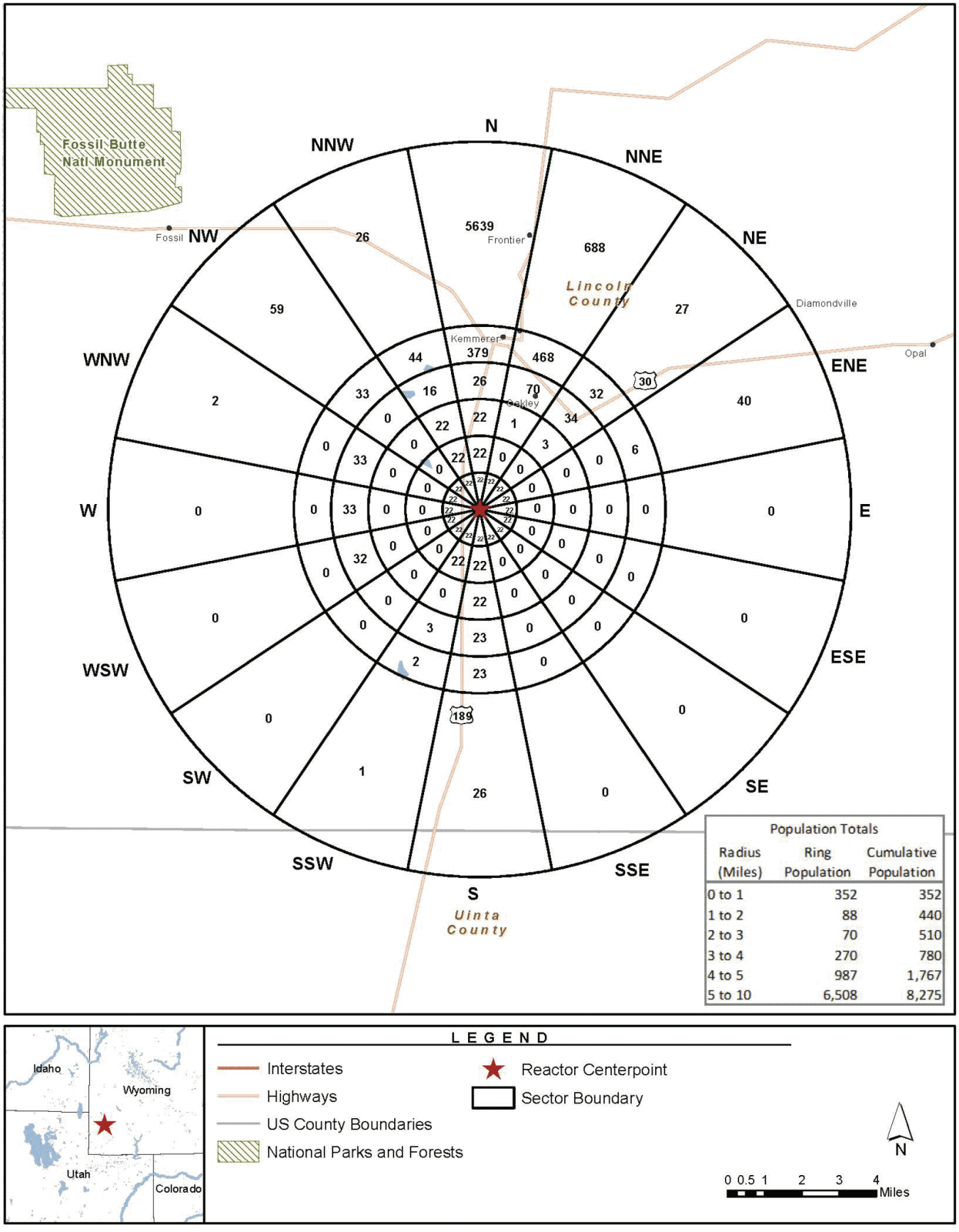
## Figure 2.2-6 2060 Population Distribution within 10 Mi (16 Km)


Fossil Butte Natl Monument
Fossil
Frontier
Kemmerer
Oakley
Diamondville
Opal
Lincoln County
Uinta County
N
NNE
NE
ENE
E
ESE
SE
SSE
S
SSW
SW
WSW
W
WNW
NW
NNW
5246
640
25
37
435
65
29
32
6
0
0
0
0
0
0
0
0
21
21
24
0
0
1
3
20
353
24
20
20
20

## Figure 2.2-7 2070 Population Distribution within 10 Mi (16 Km)


Fossil Butte Natl Monument
Fossil
Frontier
Kemmerer
Oakley
Diamondville
Opal
Lincoln County
Uinta County
N
NNE
NE
ENE
E
ESE
SE
SSE
S
SSW
SW
WSW
W
WNW
NW
NNW
5639
688
27
40
0
0
26
0
1
23
23
0
0
1
26
0
2
3
0
0
32
0
33
0
59
2
44
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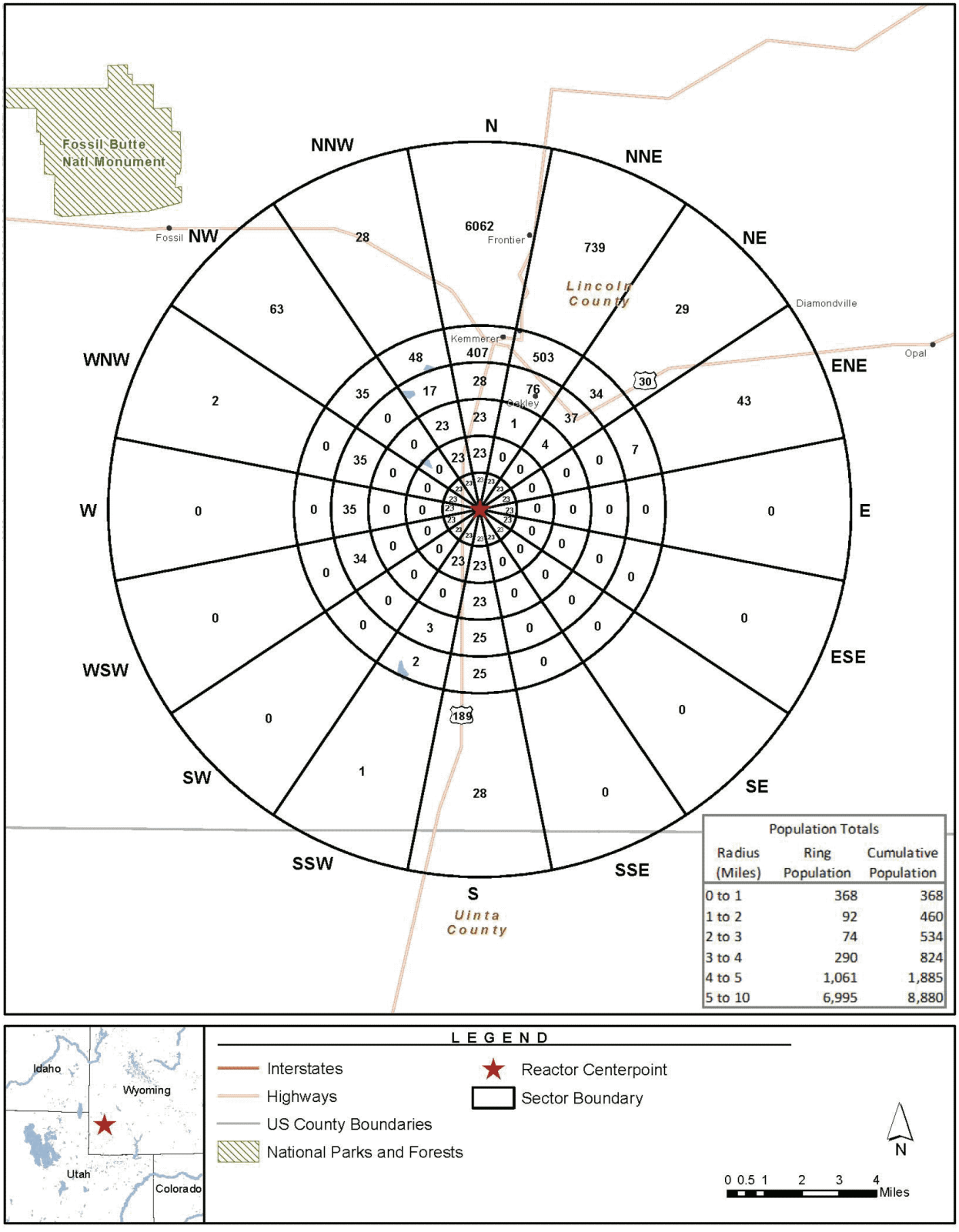
468
70
32
34
6
16
26
22
22
22

## Figure 2.2-8 2080 Population Distribution within 10 Mi (16 Km)





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## Figure 2.2-9 2090 Population Distribution within 10 Mi (16 Km)

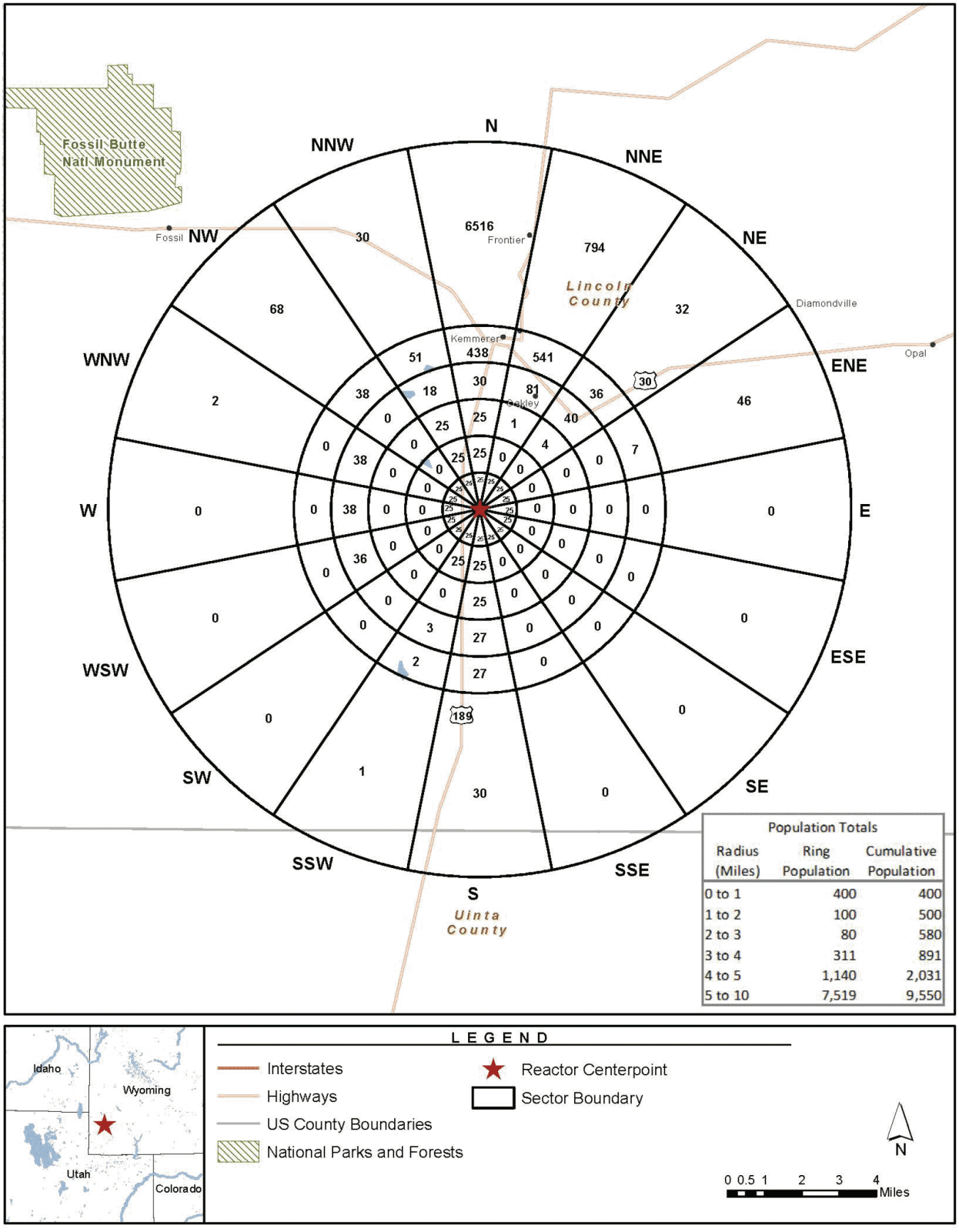



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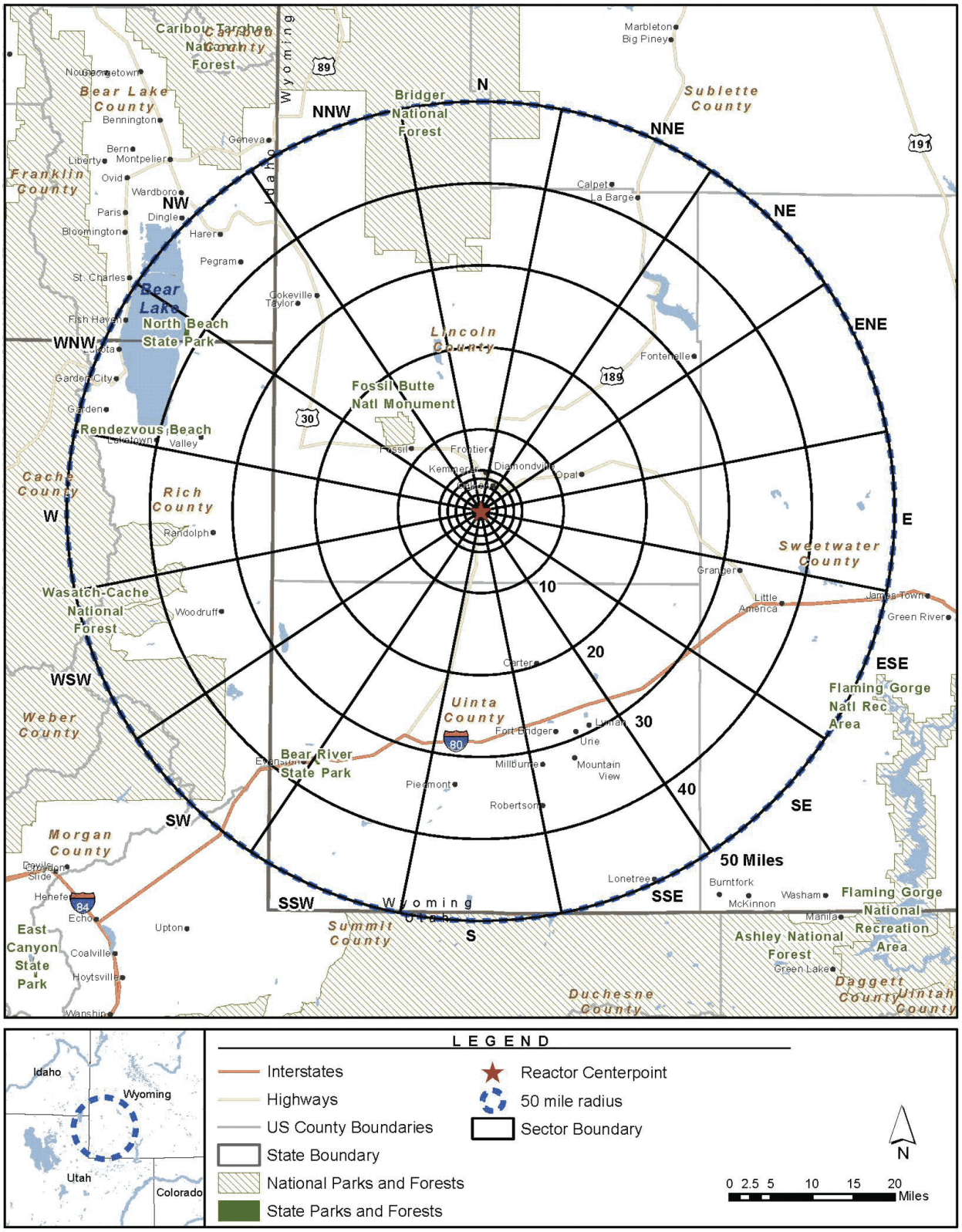
**Figure 2.2-10 Population Grid for the 50-Mi Radius (80 Km)**





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## Figure 2.2-11 2020 Population Distribution within 50 Mi (80 Km)

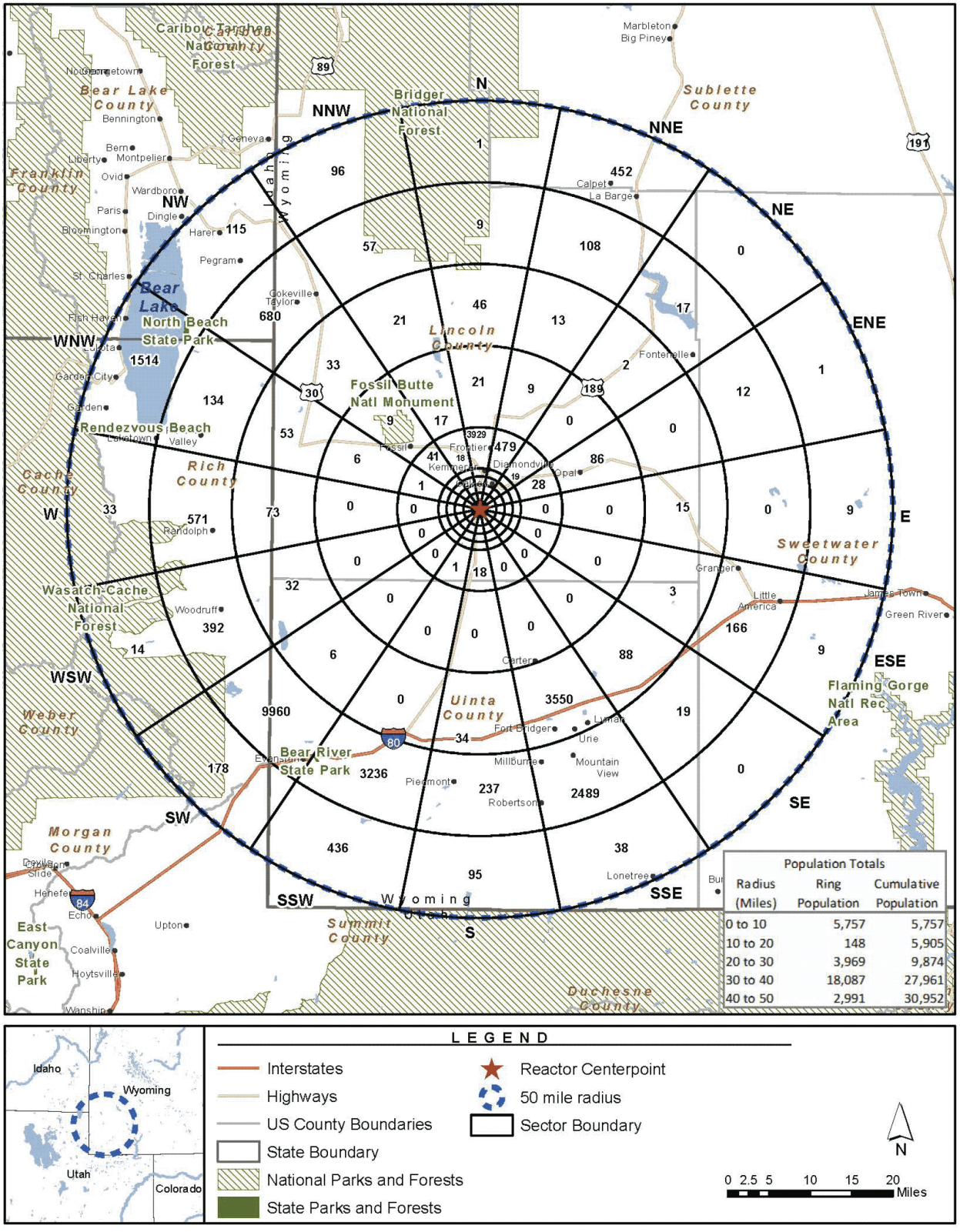



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**Figure 2.2-12 2030 Population Distribution within 50 Mi (80 Km)**

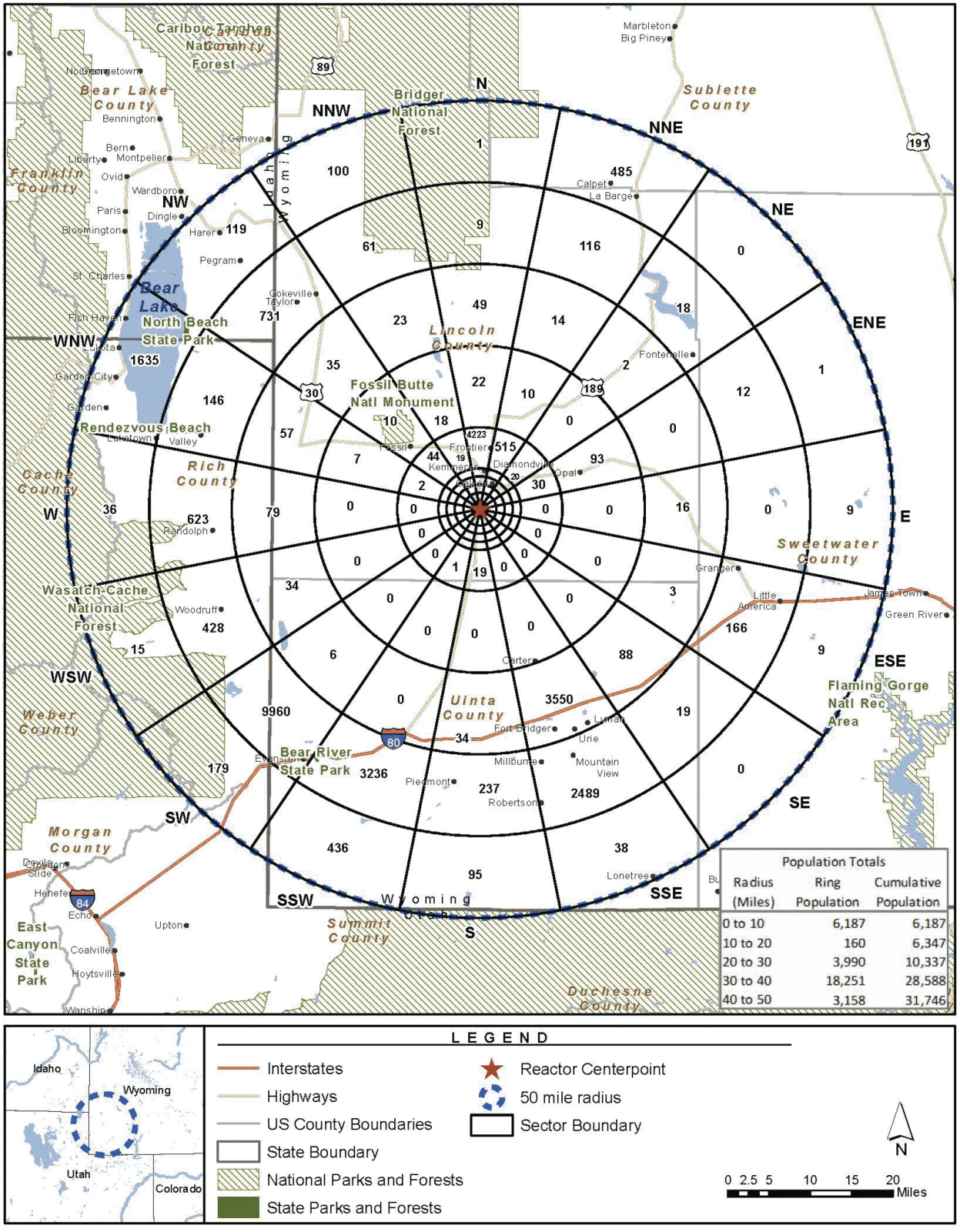



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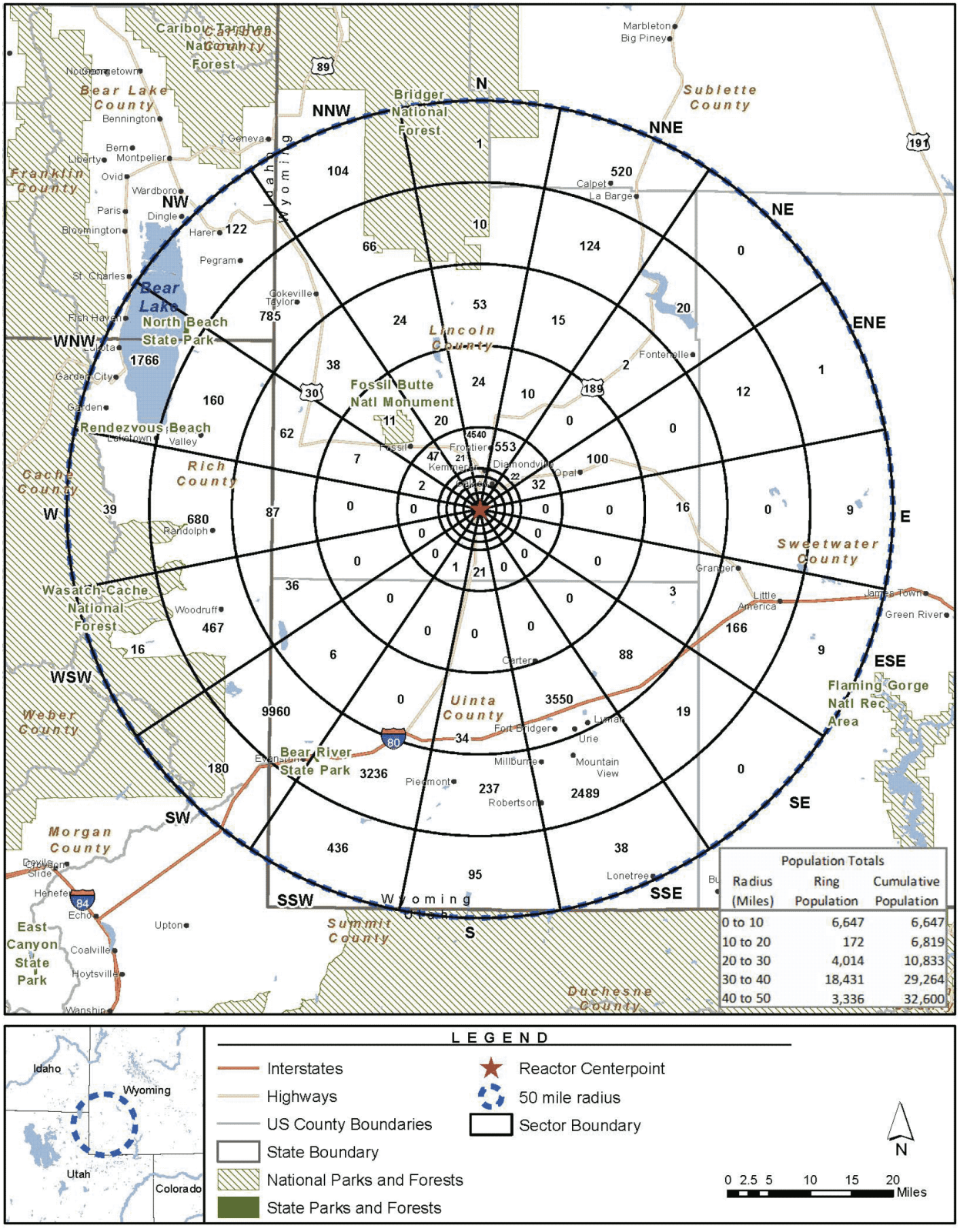
**Figure 2.2-13 2040 Population Distribution within 50 Mi (80 Km)**





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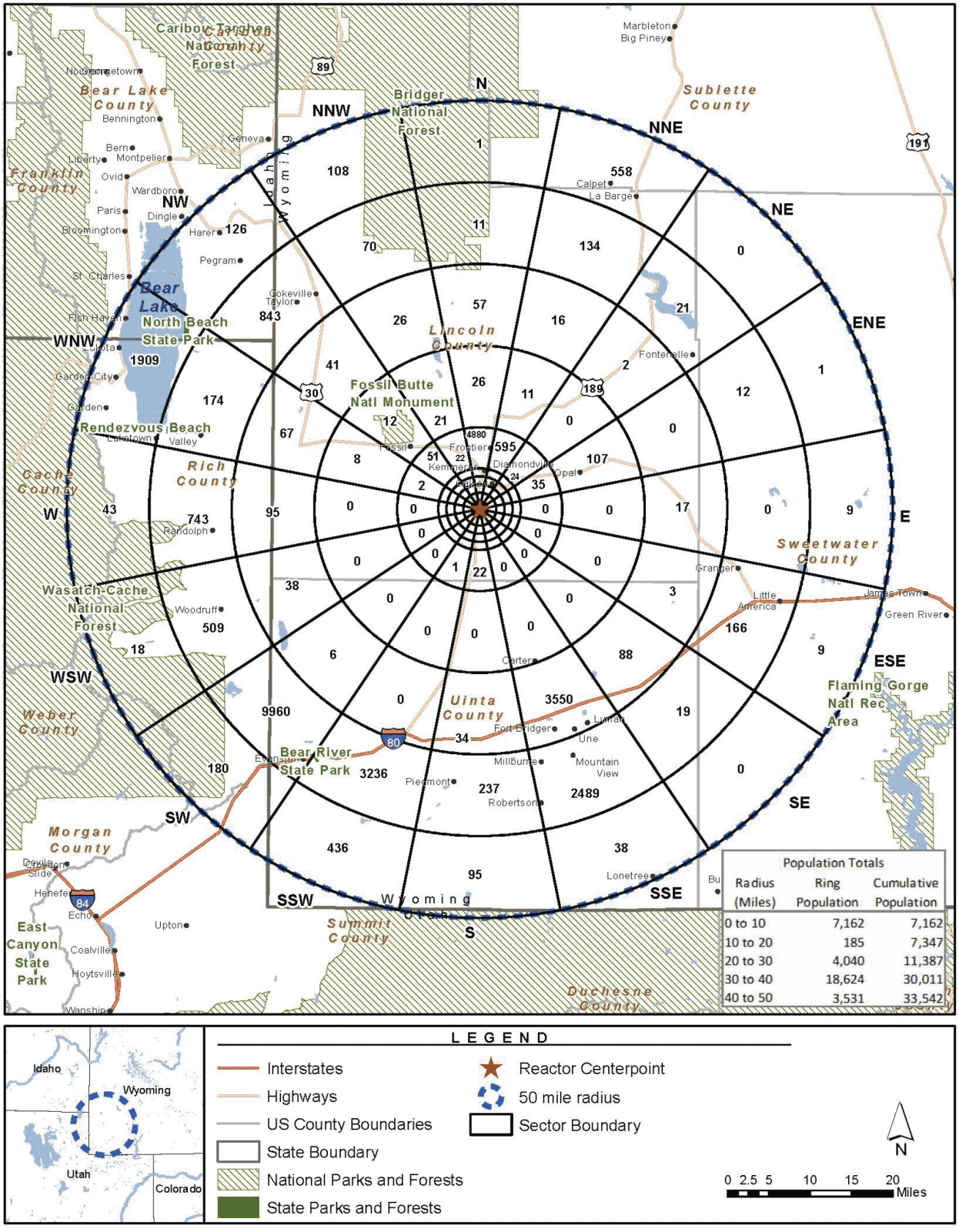
## Figure 2.2-14 2050 Population Distribution within 50 Mi (80 Km)





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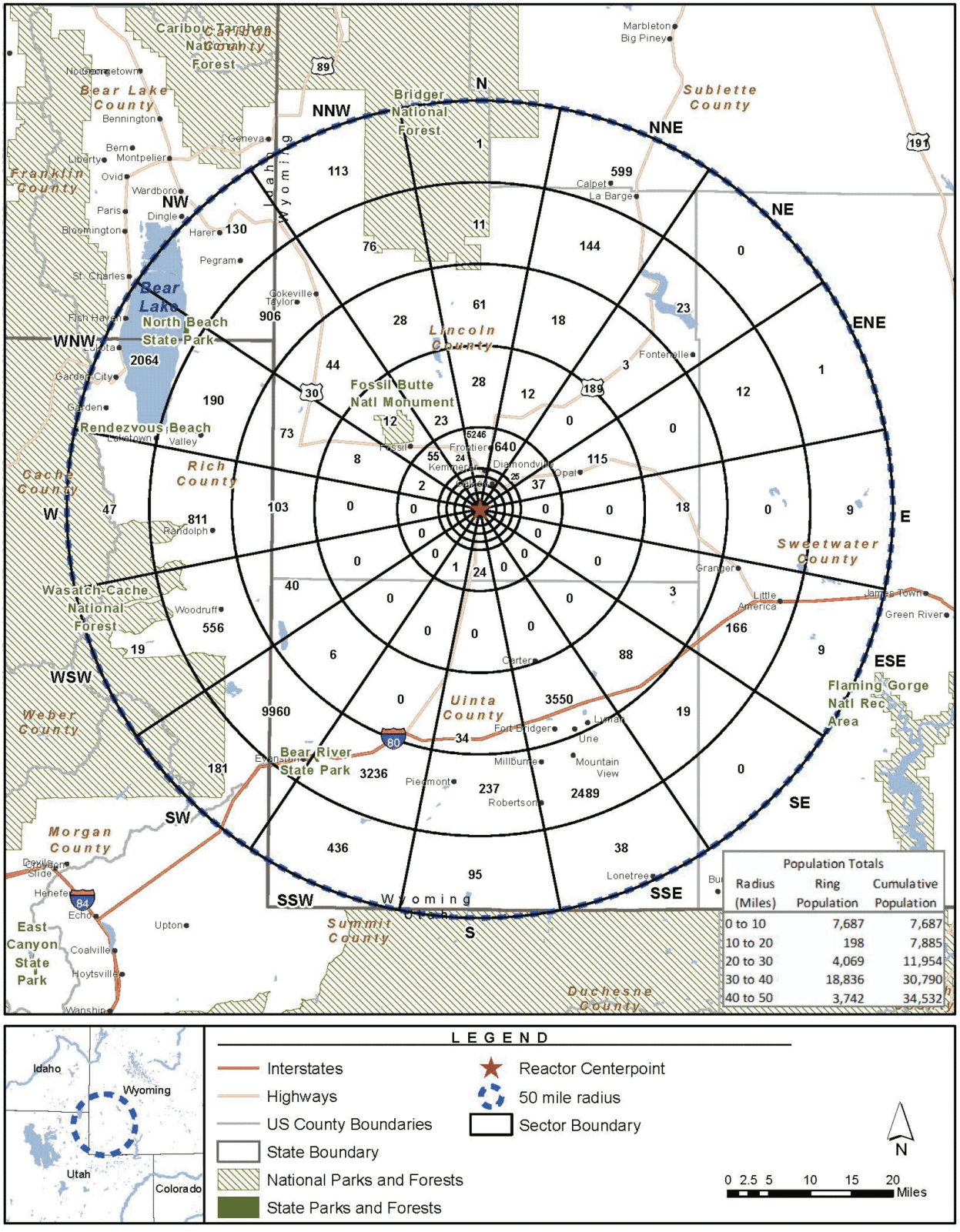
## Figure 2.2-15 2060 Population Distribution within 50 Mi (80 Km)





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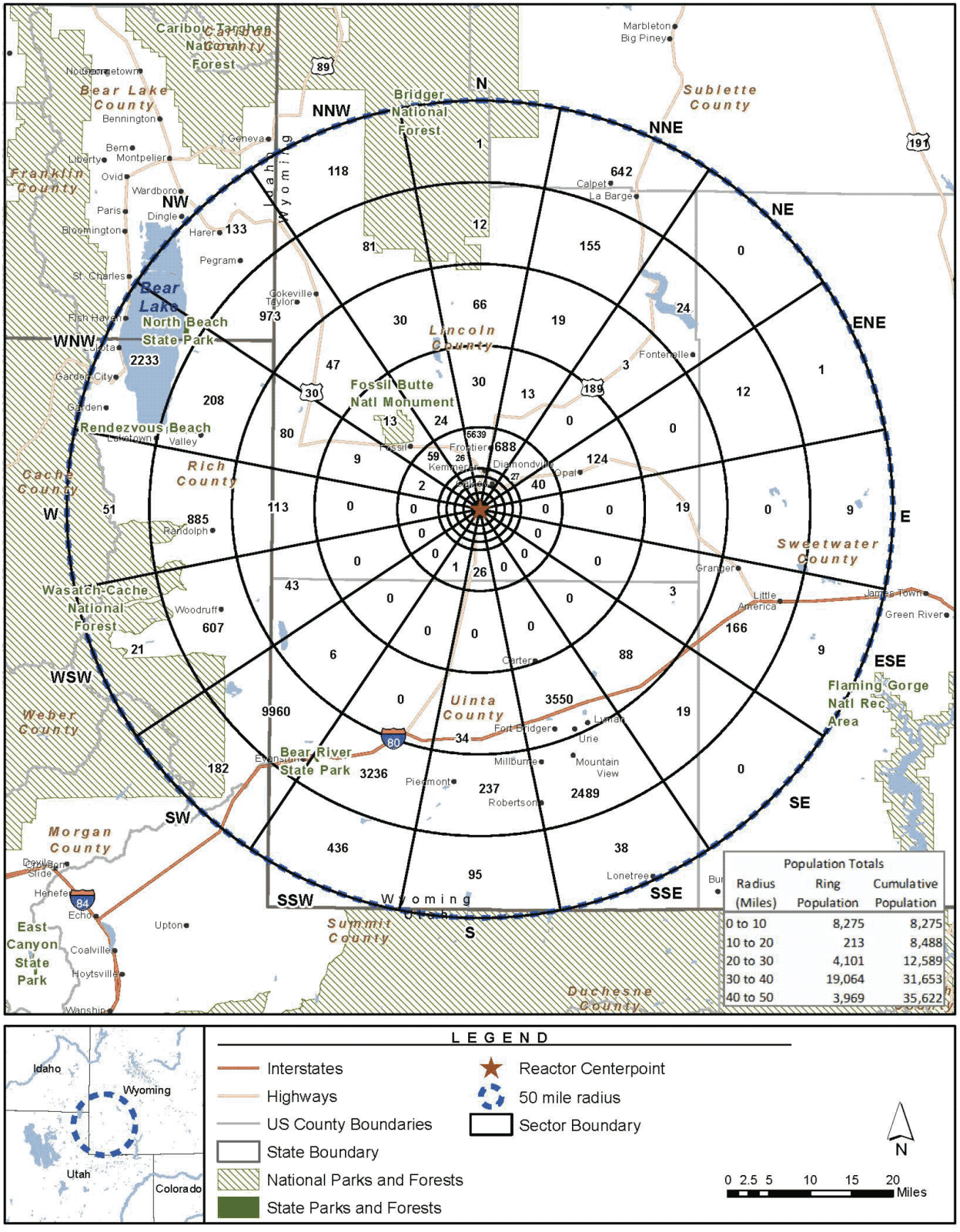
## Figure 2.2-16 2070 Population Distribution within 50 Mi (80 Km)





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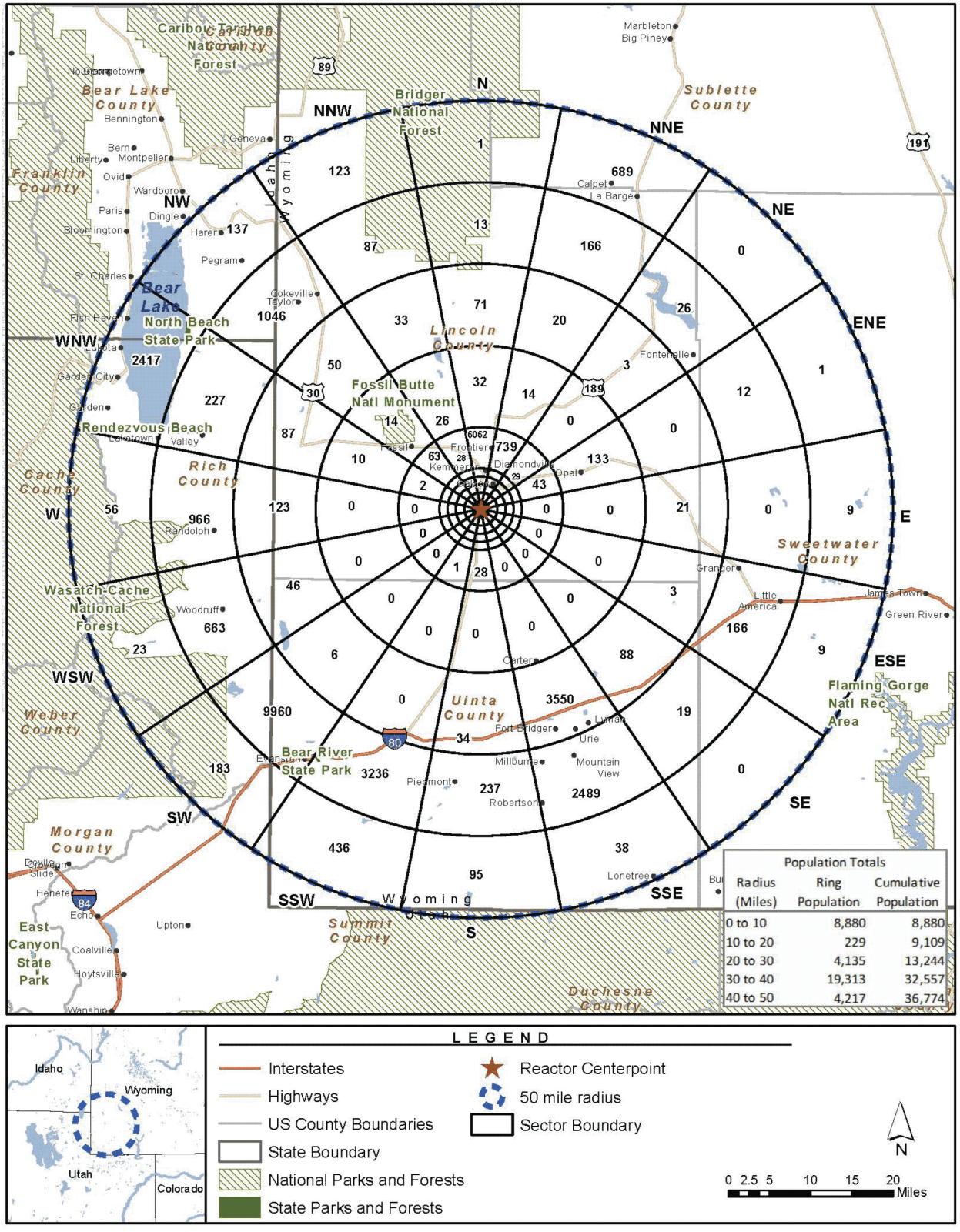
## Figure 2.2-17 2080 Population Distribution within 50 Mi (80 Km)





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## Figure 2.2-18 2090 Population Distribution within 50 Mi (80 Km)

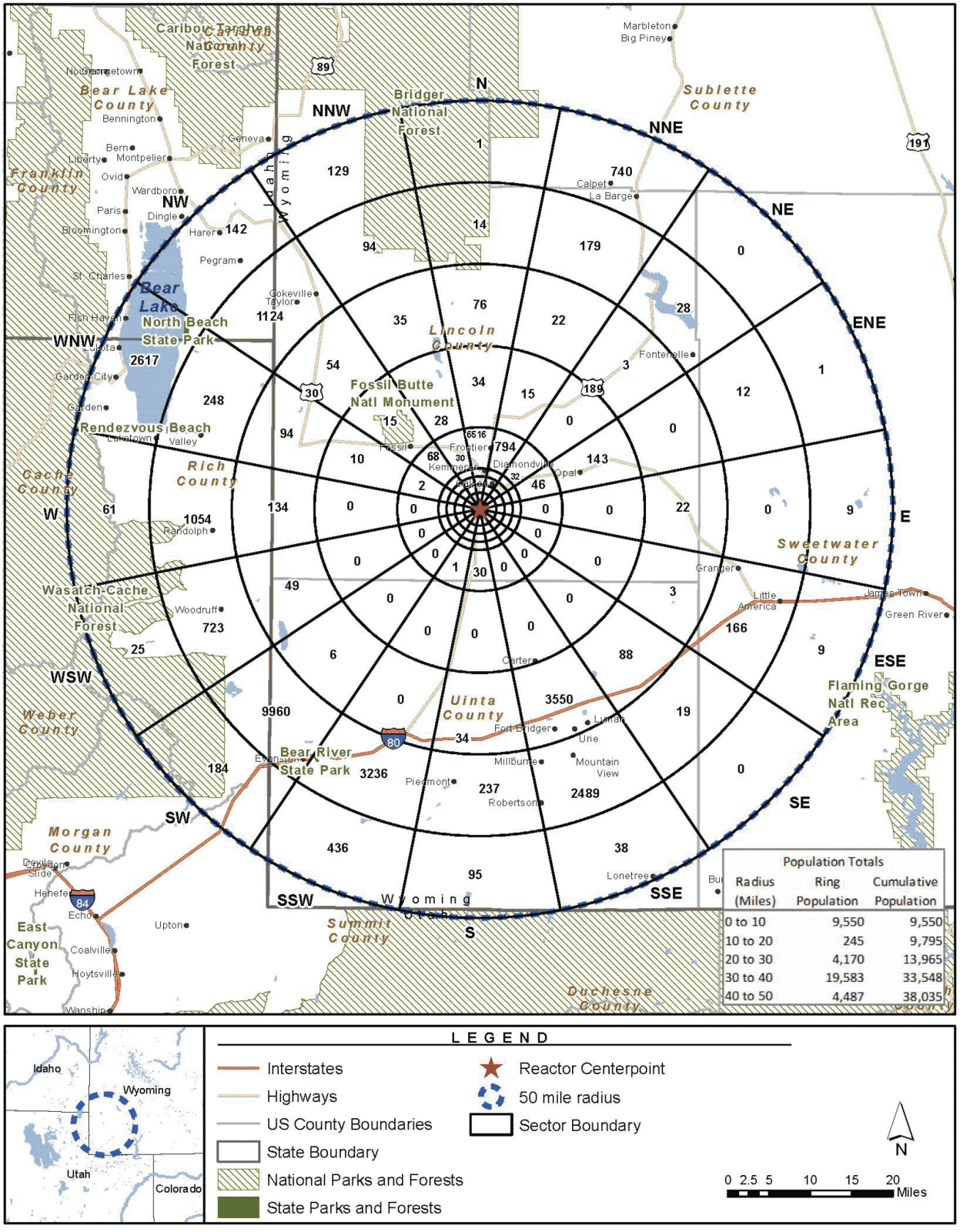



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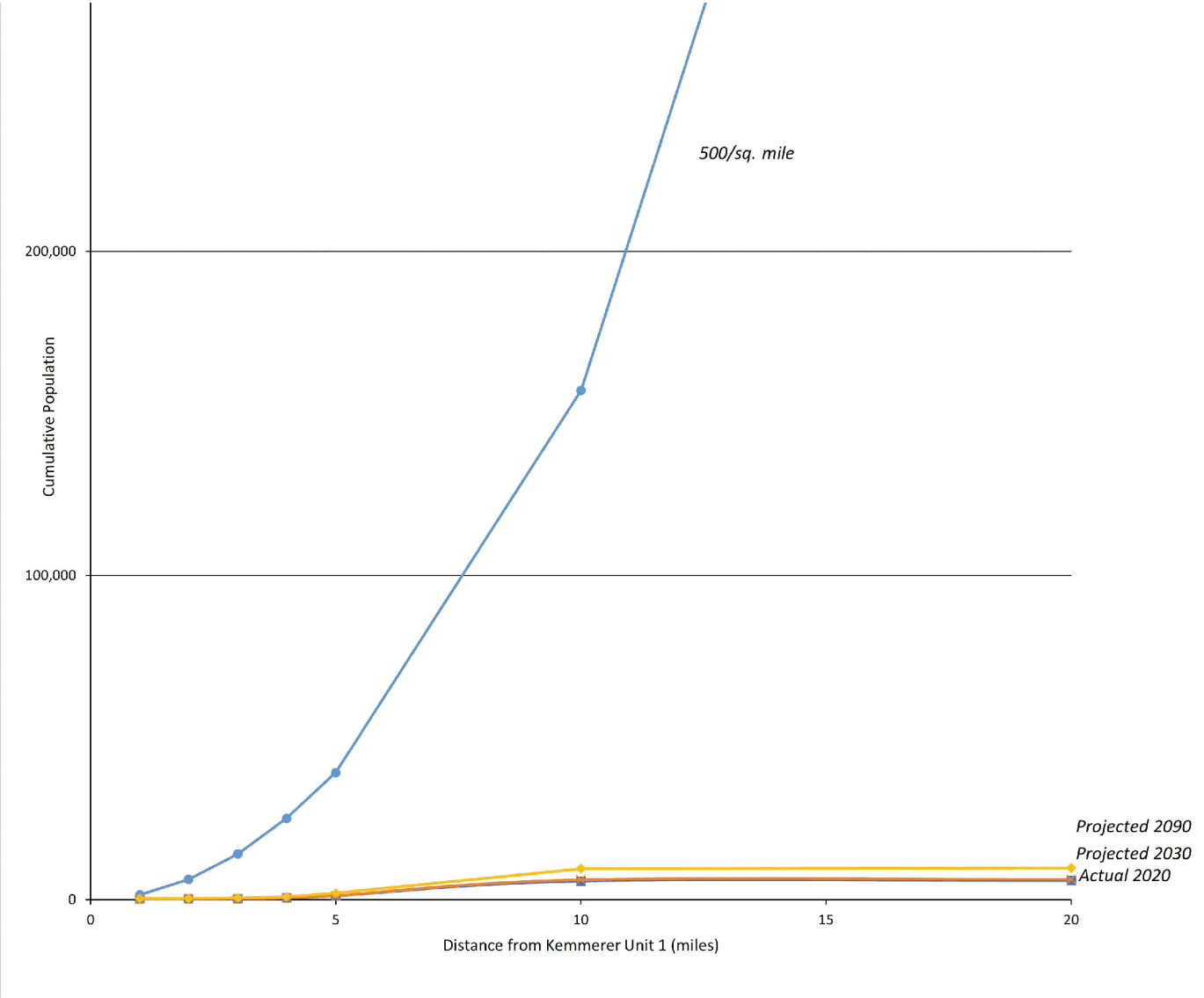
**Figure 2.2-19 Population Density Compared to NRC Siting Criteria**





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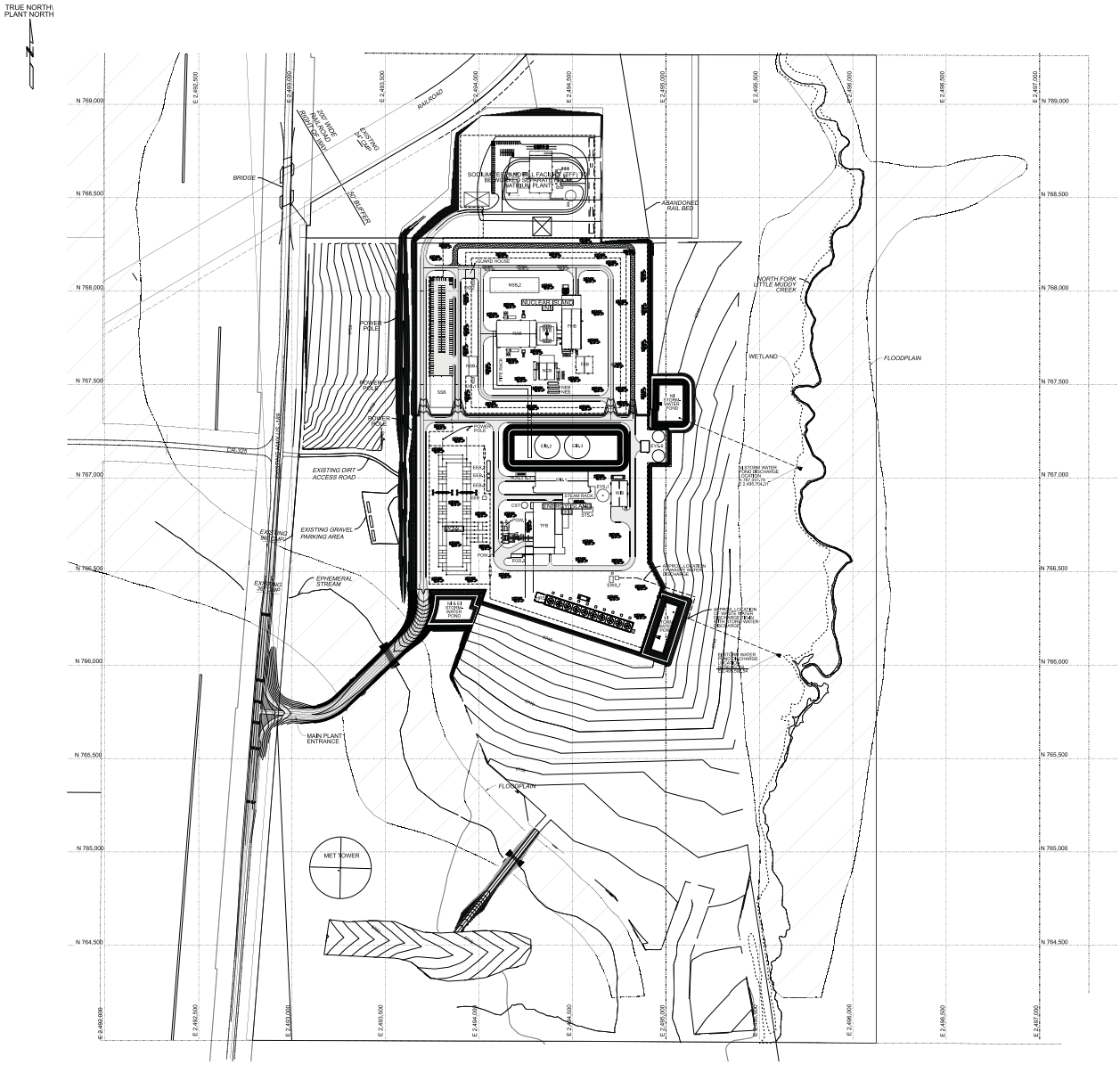
## Figure 2.2-20 Kemmerer Unit 1 Site Plan





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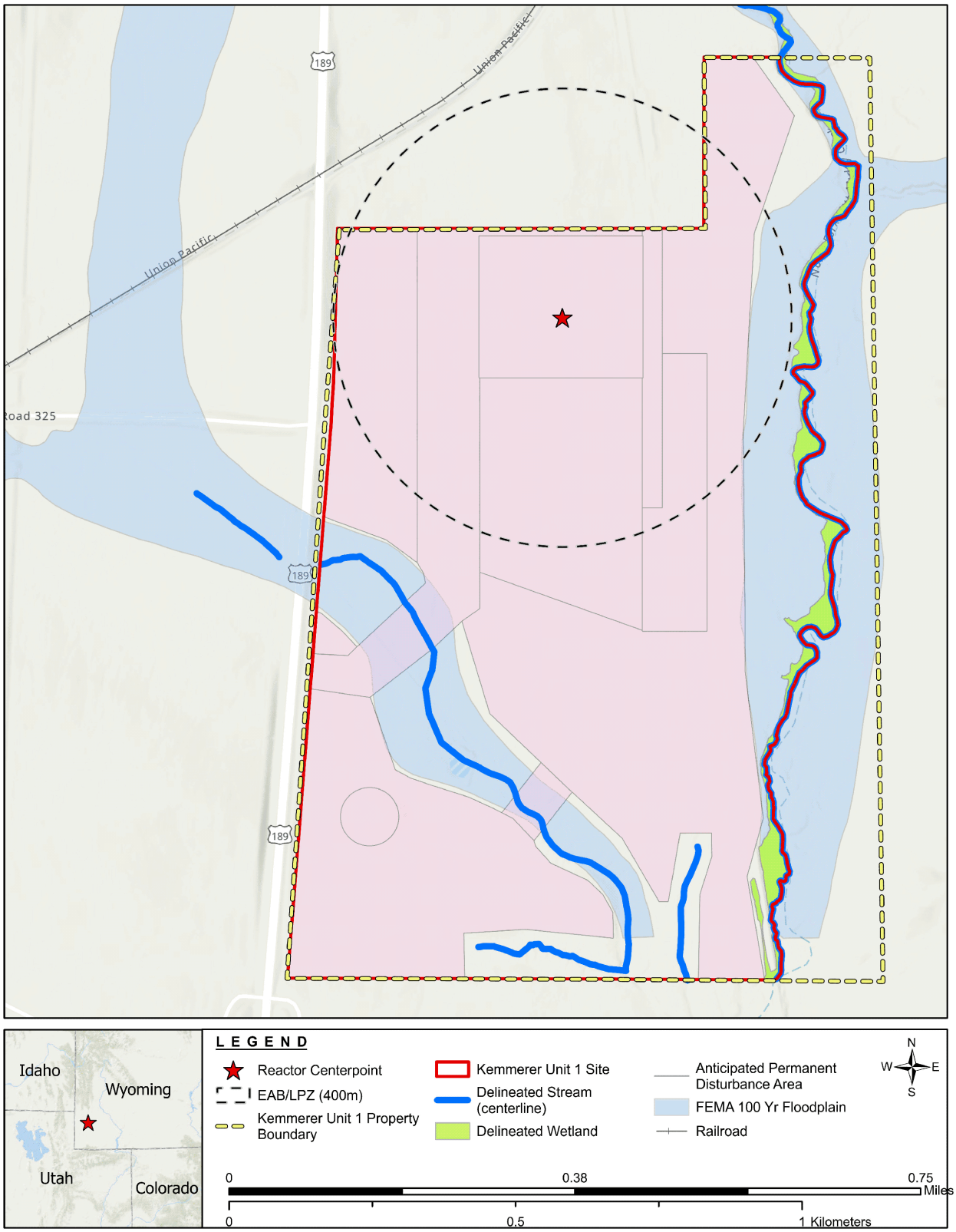
## Figure 2.2-21 Site and Property Boundaries





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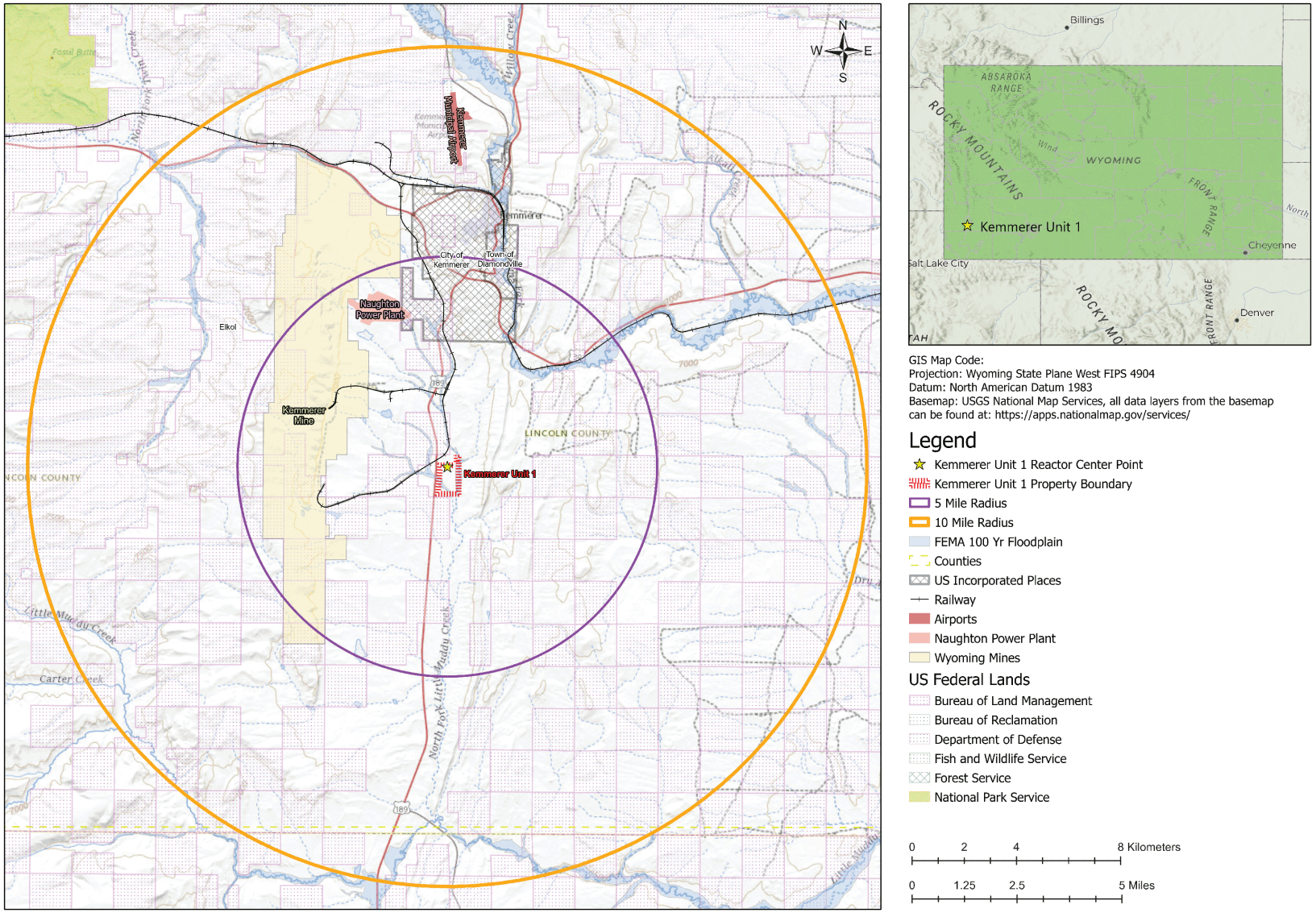
## Figure 2.2-22 Site Location Map (10 Mi, 16 Km, Radius)





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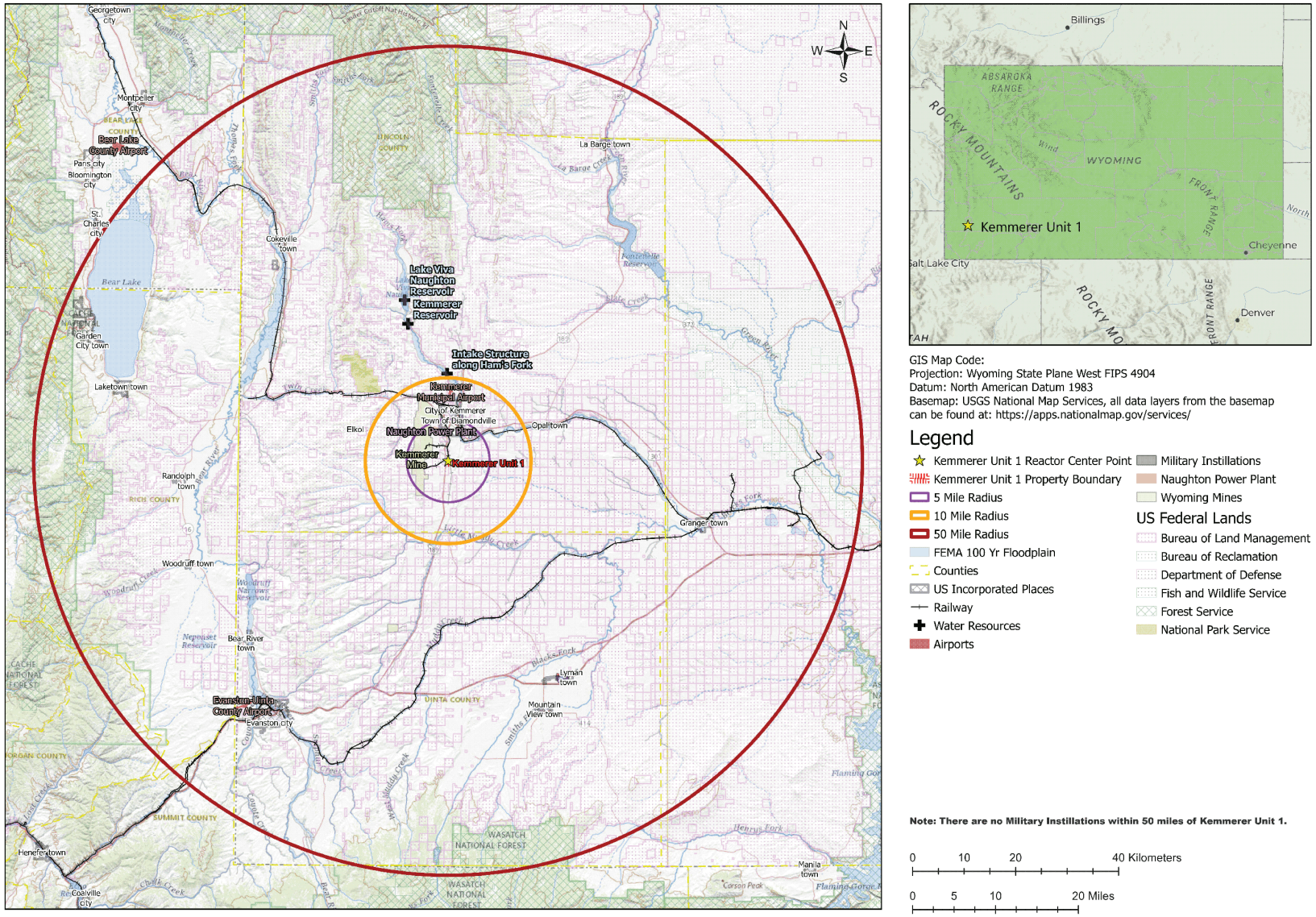
## Figure 2.2-23 Site Region Map (50 Mi, 80 Km, Radius)





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# 2.3 Nearby Industrial, Transportation, and Military Facilities

A description of potential hazards associated with nearby transportation routes, industrial and military facilities, and civilian and military airports located near the Kemmerer Unit 1 Site is provided in this section to establish whether the effects of potential hazards onsite or in the vicinity of the site should be considered as design basis hazards in accordance with 10 CFR 100.20. For each identified hazard, an assessment of the nature and extent of nearby activities was conducted. This assessment considered location, distance, frequency, and potential hazards that could result in a design basis hazard.

Evaluation of hazards associated with nearby industrial, transportation, and military facilities is conducted in accordance with the following:

- Regulatory Guide (RG) 1.78, Rev 2, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release”
- RG 1.91, Rev 3, “Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants”

## 2.3.1 Location and Routes

As discussed in Section 2.2, the site is located in Lincoln County, Wyoming and the site location and surrounding region is shown in Figure 2.2-23. Investigations identified potentially hazardous facilities and activities located within 5 miles (mi) of the site, with special attention to activities within 0.6 mi that could impose potential hazards to safety-significant structure, system, or components (SSCs). In addition, facilities and activities at distances greater than 5 mi were considered if they could affect safety-related (SR) SSCs.

### 2.3.1.1 Facilities, Activities, and Materials Located In or Transported Through the Site Vicinity

An investigation of potential external hazardous facilities and major transportation routes within 5 mi of Kemmerer Unit 1 concluded there are three industrial facilities, four highways, one railroad, and five pipelines that require further evaluation. These potential external hazards are shown in Figure 2.3-1 through Figure 2.3-3.

#### 2.3.1.1.1 Industrial Facilities Within the Site Vicinity

To identify offsite facilities that potentially store toxic, flammable, or explosive chemicals and substances, a Superfund Amendments and Reauthorization Act Title III, Tier II report (Reference 2.3-2) was obtained to provide a list of existing facilities surrounding the site within a 25 mi radius.

According to the Tier II report, the following industrial facilities are located within 5 mi of the site:

- Kemmerer Mine
- Naughton Power Plant


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An evaluation of nearby industrial facilities within 25 mi of Kemmerer Unit 1 revealed that there is one additional facility significant enough to be identified as a potentially hazardous facility:

- Kemmerer-Diamondville Joint Powers Board Water and Wastewater Plant, located 7.8 mi from Kemmerer Unit 1, was reviewed for the presence of highly toxic substances in accordance with 40 CFR 68. Although this facility is located greater than 5 mi from the site, it was identified for further analysis for the presence of chemicals stored at the wastewater treatment plant (e.g., chlorine). According to 40 CFR 68, chlorine is a highly toxic and highly volatile chemical with greater toxic endpoints and requires an offsite consequence analysis with a worst-case release scenario for risk management programs under the Clean Air Act. Chemicals stored at the Kemmerer-Diamondville Joint Powers Board Water and Wastewater Plant were verified against the Environmental Protection Agency's (EPA's) Risk Management Program Guidance, establishing they did not have toxic endpoints requiring further analysis.

#### 2.3.1.2 Transportation Routes Within the Site Vicinity

The evaluation of transport routes beyond the 5 mi vicinity of the site identified natural gas pipelines, railroads, highways, and roads. Materials transported on these transportation routes do not include highly toxic chemicals and were not considered further. Therefore, only transportation routes within a 5 mi vicinity of the site were considered for external hazard analysis.

##### 2.3.1.2.1 Highways and Roads

The site is located south of the intersection of US-30 and US-189, and east of the intersection of the Union Pacific Railroad and US-189 (Figure 2.2-22). Wyoming State highways WYO 240 and WYO 412 are within the site vicinity, as well as minor roadways in the immediate vicinity of the site, including County Road 325, Elkol Road, site access and service roads to Kemmerer Mine, Alleman Road, N Alleman Road, Naughton Road, and the site access road. US-189 is the largest and closest highway within the 5 mi vicinity of the site. Therefore, chemicals transported on US-189 bound the analysis of highways and roads located at greater distances.

##### 2.3.1.2.2 Railroads

The nearest major railroad is the Union Pacific Railroad located approximately 0.15 mi from the site (Figure 2.3-1).

##### 2.3.1.2.3 Pipelines

The following pipelines are located within 5 mi of the site as shown on Figure 2.3-2.

- Ignacio-Sumas Loop 24-inch Natural Gas Pipeline (NW)
- Ignacio-Sumas 22-inch Natural Gas Pipeline (NW)
- JL95 10-inch Natural Gas Pipeline
- FL91 3-inch Natural Gas Pipeline
- FL91 4-inch Natural Gas Pipeline


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The FL91 3-inch and FL91 4-inch natural gas pipelines at Naughton Power Plant are small connection pipelines. Due to their small sizes and short lengths, a hazard evaluation for these pipelines is not needed.

2.3.1.2.4 Airports and Federal Airways within 10 Mi of the Site

The Kemmerer Municipal Airport – Airport 1, as shown on Figure 2.3-3, is located within 10 mi of the site. There are no military airports located within 10 mi of the site and no Federal airways within 2 statute miles of the site.

2.3.2 Descriptions

2.3.2.1 Description of Facilities

2.3.2.1.1 Description of Products and Materials

A description of stored and transported hazardous chemicals and substances associated with each of the identified facilities and transportation routes is provided. These descriptions include information about the products and materials regularly manufactured, stored, used, or transported in the site vicinity.

Chemicals and Substances Stored at Nearby Industrial Facilities

The nearby industrial or offsite facilities that contain chemicals and substances that required analysis are detailed in Table 2.3-1, including identification of hazardous chemicals and the maximum stored quantity reported. The disposition of hazards associated with these chemicals is summarized in Table 2.3-2 and the analysis of these chemicals is addressed in Section 2.3.3.1.3.

2.3.2.1.2 Naughton Power Plant

Naughton Power Plant is owned by PacifiCorp and is located in Lincoln County, WY, 4 mi southwest of the City of Kemmerer and 3.1 mi northwest of the site (Figure 2.3-1). It is a three-unit power plant with total generating capacity of 700 MWe (Units 1 and 2 are coal-fired and Unit 3 was converted to natural gas in 2019).

2.3.2.1.3 Kemmerer Mine

Kemmerer Mine, owned by Kemmerer Operations LLC, is located 4 mi northwest of the site (Figure 2.3-1). Coal is processed and stored at Kemmerer Mine in concrete silos and onsite open air stockpiles. Abatement measures are used at the Kemmerer Mine to control fugitive dust from the open air stockpiles and mitigate fire hazards. These measures include water sprinkling, chemical dust suppression, and windbreak options as representative fugitive dust mitigation solutions. Another mitigation measure is the rate of coal turnover established at the Kemmerer Mine, or how frequently the stockpile is being utilized. A higher turnover value leads to lower chances of autoignition within. The coal stored at Kemmerer Mine moves quickly and is not stationary for long periods using both indoor and open-air storage systems, which distributes the stockpile quantity across multiple systems resulting in a reduction in the potential for and significance of an accident. The release of ignitable dust suspensions from some operations and


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apparatus is so infrequent that area classification is not necessary (Reference 2.3-3). According to NFPA 499 (Reference 2.3-12), where combustible dusts are processed, stored, or handled, it is usually not necessary to classify where materials are stored, transported, or where dust removal systems prevent visual dust clouds. Therefore, no further evaluation of coal stored at the Kemmerer Mine was required.

#### 2.3.2.2 Description of Pipelines

Pipelines were identified using the University of Wyoming's Wyoming Reservoir Information Tool (Reference 2.3-13). The closest and largest pipeline in diameter is the 24-inch diameter Ignacio-Sumas Loop, NW Pipeline, operated by Williams Company (Figure 2.3-2).

##### 2.3.2.2.1 24-inch and 22-inch Diameter Natural Gas Pipelines – Ignacio-Sumas Loop and Ignacio-Sumas, Northwest Pipeline

The Williams Company operates two buried natural gas pipelines within 5 mi of the site. A 24-inch diameter pipeline (Ignacio-Sumas Loop) is located approximately 2.44 mi northeast of the reactor center point, and a 22-inch diameter pipeline (Ignacio-Sumas) is located approximately 2.57 mi northeast of the reactor center point (Figure 2.3-2). The pipeline operating parameters are provided in Table 2.3-6.

##### 2.3.2.2.2 10-inch Diameter Natural Gas Pipeline – JL95

Dominion Energy Questar operates a 10-inch, buried natural gas pipeline within 5 mi of the site. The 10-inch diameter pipeline (JL95) is located approximately 2.82 mi northeast of the reactor center point (Figure 2.3-2). The pipeline operating parameters are provided in Table 2.3-6.

#### 2.3.2.3 Description of Waterways

There are no navigable waterways within 5 mi of the site vicinity.

#### 2.3.2.4 Description of Highways

The nearest highway to the Kemmerer Unit 1 site is US-189, which runs roughly north-south on the west side of the site (Figure 2.3-1). At its closest point, US-189 is roughly 0.196 mi from the nearest point to the Nuclear Island (NI). Chemicals delivered to the Naughton Power Plant and the Kemmerer Mine are transported via US-189 as summarized in Table 2.3-4.

#### 2.3.2.5 Description of Railroads

At closest approach, the nearest major railroad is operated by Union Pacific under the Rocky Mountain Division and is approximately 0.15 mi from the site (Figure 2.3-1). Solid coal is transported on the railroad from the Kemmerer Mine to the Naughton Power Plant. Coal storage and treatment methods at Kemmerer Mine were determined to provide sufficient mitigation measures against vapor cloud formation or combustion. Therefore, accident scenarios on Union Pacific Railroad were not evaluated further.


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2.3.2.6 Description of Airports and Airways

There is one municipal airport (Kemmerer Municipal Airport) that is owned by the City of Kemmerer, which is located approximately 8 mi from the site.

2.3.2.7 Projections of Industrial Growth

Sodium Test and Fill Facility

The Sodium Test and Fill Facility (TFF) is a non-nuclear testing facility located on an approximately 35 acre parcel that is owned by TerraPower directly 0.095 mi north of Kemmerer Unit 1. Potentially hazardous chemicals planned to be stored at the TFF identified for analysis are argon and nitrogen. The disposition of hazards associated with these chemicals is summarized in Table 2.3-3 and the subsequent analysis of these chemicals is addressed in Section 2.3.3.1.3.

Naughton Power Plant Conversion

Units 1 and 2 of the Naughton Power Plant will be converted from coal to natural gas. PacifiCorp plans to complete the coal-to-gas conversion by 2026. Units 1 and 2 as well as Unit 3, which was converted to natural gas in 2019, are expected to continue operating through 2036 (Reference 2.3-14). The Naughton Power Plant electric distribution and water supply systems would serve both Naughton and Kemmerer Unit 1 during the period of overlapping operations.

2.3.3 Evaluation of Potential Hazards

Based upon the information provided in Section 2.3.1 and Section 2.3.2, the potential hazards that could impact Kemmerer Unit 1 were evaluated, using the guidance in RG 1.78, RG 1.91, and RG 4.7, “General Site Suitability Criteria for Nuclear Power Stations,” Revision 3.

2.3.3.1 Determination of Potential Accidents

When determining design basis hazards or defining the principal hazard types, the following accident categories identified in RG 4.7 were considered:

i. Explosions (immediate detonation) — detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels for facilities and activities in the vicinity of the plant or onsite.

ii. Flammable vapor clouds (delayed ignition) — accidents involving the release of flammable liquids or vapors that result in the formation of unconfined vapor clouds from nearby mobile and stationary sources.

iii. Toxic chemicals — accidents involving the release of toxic chemicals (e.g., chlorine) from onsite storage facilities and nearby mobile and stationary sources. If toxic chemicals are known or projected to be present onsite or in the vicinity of a nuclear plant, or to be frequently transported in the vicinity of the plant, applicants should evaluate releases of those chemicals. For each postulated event, a range of concentrations at the site was determined for a spectrum of meteorological conditions.


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The postulated accidents within the analyzed categories were at the following locations.

Nearby Facilities:

- Naughton Power Plant
- Kemmerer Mine
- TFF

Transportation Routes:

- US-189
- Ignacio-Sumas Loop 24-inch Natural Gas Pipeline (NW)

2.3.3.1.1 Accident Hazard Categories

There are four accident categories related to explosions evaluated in this analysis:

- Flammable Vapor Clouds (delayed ignition)
- Explosions (immediate detonation)
- Jet Fire
- Toxicity analysis

Accidents involving detonations of explosives, munitions, chemicals, liquid fuels, and gaseous fuels were considered for facilities and activities within the vicinity of the site, where such materials are processed, stored, used, or transported in quantity. The effects of explosions are a concern in analyzing the structural response to blast pressures. The effects of blast pressure from explosions located at nearby facilities and transportation routes to the NI were evaluated to determine if the explosion would have an adverse effect on safety-significant SSCs located within the site which could affect safe plant operation.

2.3.3.1.2 Chemical Screening Analysis

A screening analysis was performed for chemicals and substances stored offsite and onsite, and along transportation routes. Each identified chemical that requires a hazard analysis was screened based on the physical properties of the chemical with regard to its flammability and toxicity limits, for the analyzed accident categories, [i.e., flammable and explosive vapor clouds (delayed ignition) and explosions at the source].

2.3.3.1.3 ALOHA Dispersion Model

The areal locations of hazardous atmospheres (ALOHA) dispersion model was used to model the effects from a postulated release of the identified chemicals that require hazard analysis. ALOHA models key hazards – toxicity, flammability, thermal radiation (heat), and overpressure (explosion blast force) – related to chemical releases that result in toxic gas dispersions, fires, and explosions. ALOHA employs several different models, including an air dispersion model that is intended to be used to estimate the areas, or threat flammability, or damaging overpressure that may exceed a level of concern and pose a threat to people and property. There are two


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separate dispersion models in ALOHA: Gaussian and Heavy Gas. ALOHA uses the Gaussian model to predict how gases that are about as buoyant as air will disperse in the atmosphere. The heavy gas dispersion calculations that are used in ALOHA are based on those used in the Dense Gas Dispersion Model (one of several well-known heavy gas models). ALOHA is capable of modeling a flammable vapor cloud and vapor cloud explosions, and produces a graphical “threat zone” to establish minimum safe distances from the source of the chemical hazards (Reference 2.3-15). ALOHA was also used to model the worst-case accidental boiling liquid expanding vapor explosion (BLEVE) for a complete tank failure which includes a calculated safe distance starting at the point of release.

Each postulated event was evaluated under a spectrum of meteorological conditions, to determine the worst-case meteorological condition. The spectrum of meteorological parameters chosen for the meteorological sensitivity analysis was selected based on the defined Pasquill meteorological stability classes (Table 2.3-8). Stability class has a large effect on ALOHA’s prediction of the threat zone size for dispersion scenarios. Under unstable conditions, a dispersing gas mixes rapidly with the air around it and ALOHA expects that the cloud will not extend as far downwind as it would under more stable conditions, because the vapor cloud is soon diluted. For each accident category, the ALOHA model was run over a spectrum of meteorological conditions to determine the worst-case for each postulated event. This spectrum of meteorological conditions includes the most stable meteorological class, F, allowable with the ALOHA model. The chosen wind speed inputs were based upon the defined Pasquill Stability Classes (Reference 2.3-10 and Reference 2.3-11).

Conservative assumptions used in the ALOHA model included:

- For each of the identified chemicals in the liquid state (i.e., under the atmospheric release conditions for that scenario, the physical state of the substance is expected to be a liquid), it was conservatively assumed that the entire contents of the vessel are released, instantaneously forming a 1 centimeter-thick puddle. This provided a significant surface area from which to maximize evaporation and formation of a vapor cloud.
- For each of the identified chemicals in the gaseous state, or for those chemicals that are normally gases at ambient temperatures (i.e., under the atmospheric release conditions for that scenario, the physical state of the substance is expected to be a gas), it was assumed that the quantity released from the vessel or pipeline is released over a 10-min period into the atmosphere as a continuous direct source (40 CFR 68.25).
- To model the worst-case scenario for flammable vapor cloud explosions in ALOHA, detonation was chosen as the ignition source.

Bounding scenarios were considered for the explosions and toxicity hazards analyses to capture the worst case scenario for chemicals stored in varying quantities and locations. The effects of flammable vapor clouds and vapor cloud explosions from external sources are summarized in Table 2.3-2 and are described in the following sections relative to the release source.


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### 2.3.3.1.4 Effects of Design Basis Hazards – Evaluation of Flammability and Explosion Hazards

Internal and external hazards to the nuclear plant are defined as the accidents that have a probability of occurrence on the order of magnitude of $10^{-7}$ per year or greater, and potential consequences serious enough to affect the safety of the plant to the extent that the guidelines in 10 CFR 100.20 could be exceeded. The purpose of the hazard analysis is to ensure that any evaluated chemicals or substances stored offsite and onsite, and transported via highway, rail, and pipeline would not present a design-basis hazard either through a deterministic assessment or a probabilistic analysis.

The minimum safe distances that have been calculated using the ALOHA model and the trinitrotoluene (TNT) Equivalency Method (RG 1.91) are the distances at which a nuclear power plant can operate in, unprotected, when exposed to flammable or explosive chemicals. For a given chemical, the minimum safe distance is the distance to the lower flammable limit (LFL), the distance to a peak overpressure of 1.0 psi for a vapor cloud or tank explosion, and where applicable, the distance to 5.0 $kW/m^2$ for a BLEVE tank explosion or pipeline jet fire. A comparison of the chemical storage or transport distances to the calculated minimum safe distances is provided in Table 2.3-11, Table 2.3-12, and Table 2.3-13.

Deterministic Assessment

A deterministic assessment was conducted for all stored and transported chemicals and substances to determine if an accident could occur with consequences serious enough to affect the safety of the plant. The deterministic assessment identified the minimum safe distances for external and internal chemical hazards and postulated accidents using ALOHA software and the TNT equivalency method in RG 1.91.

For the chemicals that are determined to be an unsafe distance from the site, a probabilistic analysis was performed to determine an annual probability of occurrence by following guidance in RG 1.91 and Federal Emergency Management Agency's *Handbook of Chemical Hazards Analysis Procedures* (Reference 2.3-16).

Probabilistic Assessment

A probabilistic analysis, or a determination of events with an annual probability of occurrence greater than an order of magnitude of $10^{-7}$ per year was performed for any chemicals that failed the deterministic assessment. According to RG 1.91, the distances from a nearby facility or transportation route to the SSCs that must be protected may not be great enough to allow a conclusion (based on conservative assumptions), such that the peak positive incident overpressure would be less than 1.0 psi. In such cases, an analysis of the probability of potential accidents at nearby facilities or the frequency of hazardous cargo shipment should prove that the attendant risk is sufficiently low. This assessment is based on statistical data for each identified hazard and determines if an accident with an estimated probability of occurrence is less than $10^{-7}$ with realistic or best estimate basis design inputs, or less than $10^{-6}$ with conservative estimates.


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Frequency of Occurrence Assessment

In accordance with RG 1.91, the results of the TNT Equivalency method calculation identified chemicals whose peak positive incident overpressure exceeded 1 psi for SR structures on the site. RG 1.91 requires an analysis of the frequency of hazardous cargo shipments to show that the attendant risk is sufficiently low.

Bounding Scenarios for Flammability and Explosion Hazards Analysis

For the flammability and explosion hazards analysis, a bounding area was drawn around the boundary of the NI based on the site plan (Figure 1.2-1). All measurements taken from the NI boundary to point of interest (offsite and onsite storage locations and transportation routes) were used in hazards analysis for conservatism, to account for refinements to the plant design within the NI.

The offsite chemical storage hazards analysis was conducted based on the nearest distance between the EPA Tier II reported coordinates for offsite facility location and NI for the flammability vapor cloud and explosion analyses (Reference 2.3-1). The highway transport scenarios were based on the shortest distance between US-189 and the NI for all flammability vapor cloud and explosion analyses.

Another bounding area was drawn around the SR structures within the NI for the probabilistic assessment conducted for the highway transport of propane, and the analysis of onsite chemicals.

Offsite Facility Results

A conservative analysis using the TNT equivalency methods was used to determine safe distances for the identified potentially explosive materials. The results indicate that the safe distances are less than the minimum separation distances from the site to the storage locations for any of the identified materials (Table 2.3-9).

None of the chemicals stored offsite had a facility distance that was shorter than the calculated distance to their LFL, to the distance of 1.0 psi overpressure via ALOHA or the TNT equivalency method, or to the distance of 5.0 $kW/m^2$ thermal radiation. Therefore, damaging overpressures or missiles from an explosion resulting from a complete failure of the total stored quantity for each material evaluated at each of the identified facilities with regard to the storage of materials with the potential for explosion would not adversely affect the operation of safety-significant SSCs at Kemmerer Unit 1.

Highway Transport Results

Table 2.3-4 details the hazardous materials potentially transported on US-189. The nearest approach from US-189 to the site is approximately 1,035 feet (ft).


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None of the chemicals, except for propane, had a minimum safe distance that was less than the calculated distance to their LFL, to the distance of 1 psi overpressure via ALOHA or the TNT equivalency method, or to the distance of 5.0 kW/$m^2$ thermal radiation. The results are provided in Table 2.3-11.

An analysis for the identified chemicals was conducted using the TNT mass equivalency methodologies. The worst-case scenario for the chemicals was the highway transport of 6,000 gallons (gal) of propane. A postulated scenario involving the transport and ensuing explosion of 6,000 gal of propane (highway transport deterministic assessment scenario) resulted in the largest distance to 1 psi via the TNT equivalency hand calculation 2,761 ft, which is greater than the minimum separation distance, 1,035 ft, from the site to US-189 for the identified materials (Table 2.3-11). The ALOHA Gaussian Model results for distance to LFL (2,952 ft) and distance to 1.0 psi (3,741 ft) also exceeded the minimum distance from US-189 (1,035 ft) to the NI and the SR bounding area (1,333 ft). Additionally, the calculated distance to 5.0 kW/$m^2$ via an ALOHA modeled BLEVE tank explosion (1,443 ft) exceeded the distance to the NI and SR bounding area.

None of the other offsite chemicals, except for propane, had a minimum separation distance greater than the calculated distance to their LFL, to the distance of 1 psi overpressure via ALOHA or the TNT equivalency method, or to the distance of 5.0 kW/$m^2$ thermal radiation. Therefore, an explosion from materials transported along nearby highways with the potential for explosion with the exception of propane, would not adversely affect safe plant operation of the Kemmerer Unit 1 as shown in Table 2.3-11.

Deterministic Assessment Results for Propane Transport on US-189

The worst-case scenario for all chemicals scenarios analyzed for flammability and explosion hazards is the highway transport of 6,000 gallons of propane. The calculated distance to 1.0 psi via the TNT equivalency hand calculation (2,761 ft) and ALOHA analysis (1,965 ft) was greater than the minimum safe distance from US-189 to the SR bounding area (1,035 ft). Additionally, the calculated distance to 5.0 kW/$m^2$ via an ALOHA modeled BLEVE tank explosion (1,443 ft) exceeded the distance to the SR bounding area. All other chemical results did not exceed their minimum safe distances. For these reasons, a frequency occurrence assessment was performed for highway transport of propane.

Probabilistic Assessment Results for Propane Transport on US-189

A probabilistic risk analysis of potential truck accidents at the site to determine if the transport of propane along US-189 could result in a plausible design basis hazard was conducted. For this assessment, a review of flammability and explosion hazards from US-189 to the SR bounding area was conducted to calculate the expected exposure distance to the nearest SR structures onsite (Figure 1.2-1).


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Frequency Occurrence Assessment Results

By utilizing conservative assumptions, a realistic probability was shown to be lower than the determined value. The maximum allowable number of trips that a delivery truck carrying approximately 6,000 gal of propane could make per year is approximately 19 trips per year using a $10^{-7}$ exposure rate, and 193 trips using a $10^{-6}$ exposure rate. Propane has an estimated annual shipment frequency of 13 across the stretch of the highway nearest to the site for the nearby industrial users (Kemmerer Mine and Naughton Power Plant). For the assumed shipping frequency of propane for the Kemmerer Mine and the Pacificorp Naughton Power Plant of 6,000 gallons of propane 13 times annually, there is a sufficiently low risk of a design basis accident.

Pipeline Results

The ALOHA dispersion model was used to determine the distance a vapor cloud could travel to reach the LFL boundary once a vapor cloud has formed from an accidental release of natural gas (as methane) from the pipeline. The pipeline release source module was selected in the ALOHA program to model the natural gas release. To model the pipeline release, additional assumptions in the ALOHA model included:

- The pipeline characteristics presented in Table 2.3-6 were used as inputs.
- It was conservatively assumed that the pipeline length is equivalent to 200 times the diameter of the pipe. ALOHA requires a length at least 200 times the diameter (the shorter pipeline length is conservative because as the length of the pipeline increases, the release rate of the gas decreases due to friction along the length of the pipeline).
- It was conservatively assumed that the pipeline is connected to an infinite tank source. The infinite tank source model assumes that the pipeline is connected to a very large reservoir such that gas is released from the pipeline break until the break is isolated from the reservoir.
- It was conservatively assumed that the roughness of the pipeline is smooth. The pipeline roughness was used to describe the interior surface of the pipe. Selection of a smooth pipe is a conservative assumption given that increased roughness introduces friction and turbulence to the fluid inside the pipe causing a reduction of the flow rate.

The 24-inch natural gas pipeline with a pressure of 850 psi was evaluated. Natural gas (evaluated as methane) has a molecular weight of 16.04 lb/lb-mol. Since it is lighter than air, natural gas was evaluated using ALOHA's Gaussian model. However, as methane is put under pressure, the density of the gas increases (Reference 2.3-17), causing it to disperse more like a heavy gas. At 850 psi, ALOHA determines that the Heavy Gas spreading algorithm is a more realistic way to model the gas.

The results indicate that the minimum safe distance for the Ignacio-Sumas Loop NW, 24-inch diameter pipeline is not exceeded in any hazard scenario:

- The pipeline is at a greater distance away than the calculated distance to LFL in ALOHA.
- The pipeline is at a greater distance than calculated distance to 1.0 psi via the TNT equivalency method and the calculated distance to 1.0 psi in ALOHA.


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- The pipeline is a greater distance than the calculated distance to 5 kW/m$^2$ thermal radiation from the ALOHA jet fire analysis.

The worst case scenario results for the natural gas pipeline explosion are shown in Table 2.3-12.

Since the overpressure criterion of 1.0 psi from RG 1.91 is not exceeded in the ALOHA results or in the TNT equivalency results, blast generated missile effects from a pipeline explosion are not a concern.

Onsite Chemical Storage Results

None of the onsite storage chemicals analyzed have a minimum safe distance that is shorter than the calculated distance to their LFL, to the distance of 1.0 psi overpressure via ALOHA or the TNT equivalency method, or to the distance of 5.0 kW/m$^2$ thermal radiation via an ALOHA BLEVE or an ALOHA jet fire analysis. The results of the onsite flammability and explosion hazards are provided in Table 2.3-13.

2.3.3.1.5 Effects of Design Basis Hazards – Evaluation of Toxicity Hazards

Using the ALOHA model, a minimum safe distance for Main Control Room (MCR) habitability was calculated for each chemical in a conservative spill scenario. Both transport and stored quantities of chemicals onsite and offsite were screened based on various parameters such as storage or transport distance from the MCR, chemical properties (including RG 1.78 minimum chemical weight screening), and conservative bounding scenarios. Table 2.3-16, Table 2.3-17, and Table 2.3-18 show the worst-case results for each chemical modeled in ALOHA.

Offsite Facility Results

The maximum distance to the toxicity limit was not exceeded for all offsite chemical storage scenarios.

The worst-case scenario results for a toxic vapor cloud from the release of the offsite facility storage quantity of nitrogen (liquified gas) located at the TFF exceeded the minimum distance to the toxicity limit (948 ft) and exceeded the toxic limit for the maximum outdoor concentration (88,200 ppm) at the MCR. However, the maximum indoor concentration of 2,570 ppm inside the MCR does not exceed the toxic limit for nitrogen. Offsite facility toxicity hazards results are provided in Table 2.3-15.

Highway Transport Results

The maximum distance to the toxicity limit was found to exceed the ALOHA modeled distance to the MCR for the following transport scenarios:

- Transport of Ammonium Hydroxide
- Transport of Propane


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The worst-case scenario results for a toxic vapor cloud from the release of the analyzed transport quantities of ammonium hydroxide and propane located on US-189 exceeded the minimum safe distance to the toxicity limit, and exceeded the toxicity limit for the maximum outdoor concentrations at the MCR. However, the maximum indoor concentrations inside the MCR do not exceed the toxicity limits for ammonium hydroxide or propane, and, therefore is not a potential hazard to control room habitability. Highway transport toxicity hazards results are provided in Table 2.3-16.

Pipeline Results

There are no toxicity hazards associated with a natural gas pipeline accident because natural gas (methane) does not have toxicity limits (immediately dangerous to life or health (IDLH) or a TWA) as shown in Table 2.3-17.

Onsite Chemical Storage Results

The maximum distance to the toxicity limit was found to exceed the modeled distance to the MCR for the following onsite chemical storage scenarios:

- Argon (liquified gas)
- Hydrochloric acid (concentration greater than or equal to 35 percent)
- Nitrogen (gas)
- Nitrogen (liquified gas)
- Propane
- Sodium hypochlorite (as chlorine gas)

The minimum safe distance to toxicity limit, maximum outdoor concentration, and maximum indoor concentration all exceeded the toxicity limit for sodium hypochlorite stored at the Water Treatment Building, located 620 ft from the MCR. The worst-case scenario results for sodium hypochlorite yielded a minimum distance to the toxicity limit of 870 ft, a maximum outdoor concentration of 15.5 ppm at the MCR, and a maximum indoor concentration of 10.6 ppm inside the MCR. To address the exceedance of the maximum indoor concentration inside the MCR, toxic chemical sensors are used to continually sample the MCR fresh air intake. The results of the onsite chemical storage toxicity hazards results are provided in Table 2.3-18.

2.3.3.1.6 Effects of Design Basis Hazards – Evaluation of Aircraft Hazards

Proximity Criteria – Screening Analysis

The probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR Part 100 exposure guidelines is less than an order of magnitude of $10^{-7}$ per year. The probability is considered to be less than an order of magnitude of $10^{-7}$ per year by inspection if the distances from the plant meet all of the criteria listed below:

A. The plant-to-airport distance D is between 5 and 10 statute mi, and the projected annual number of operations is less than 500 $D^2$, or the plant-to-airport distance D is greater than 10 statute mi, and the projected annual number of operations is less than 1,000 $D^2$.


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B. The plant is at least 5 statute mi from the nearest edge of military training routes, including low-level training routes, except for those associated with usage greater than 1,000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation.

C. The plant is at least 2 statute mi beyond the nearest edge of a Federal airway, holding pattern, or approach pattern.

A screening analysis following the proximity screening criteria was performed to establish whether the probability of aircraft accidents at the site is considered to be less than an order of magnitude of $10^{-7}$ per year. The screening analysis included an assessment of the projected number of operations that represented the maximum aircraft activity expected during the plant's construction period and operating life. Projected number of operations was determined using downloaded operations data from the 2022 Federal Aviation Administration (FAA) Terminal Area Forecast report (Reference 2.3-7), which includes historical data from 1990-2022 and projections up to 2050. The average rate of change from 2022-2050 was applied to projections for years 2051 to 2090. Operations data for airports not included in the FAA Terminal Area Forecast were obtained from the FAA Airport Data Information Portal (Reference 2.3-4).

As discussed in Section 2.3.1.2.4, within 10 mi of the site, there is one municipal airport (Kemmerer Municipal Airport). The FAA airport statistics and forecast operations data for this airport were reviewed and the results determined that proximity screening criteria A is met for this airport (Reference 2.3-7). Therefore, flight operations from the nearby Kemmerer Municipal Airport were not required to be considered as a potential site hazard.

There are no military training routes within 5 statute mi of Kemmerer Unit 1 according to Reference 2.3-8; therefore, criteria B is met and no further assessment regarding military training routes was needed.

In addition, no approaches or holding patterns were identified within 25 mi of the site according to the Visual Flight Rules Sectional Chart for Salt Lake City (Reference 2.3-6); thus, no further evaluations were needed.

For proximity screening criteria C, there are no Federal airways whose edges are closer than 2 statute mi to the Kemmerer Unit 1 site.

Based on the screening analysis conducted, the Kemmerer Unit 1 site meets the criteria and no further analysis is required.

#### 2.3.3.1.7 Fire Hazards

Fires that could lead to high heat fluxes or to the formation of flammable vapor clouds are considered as part of the design basis hazards assessment. As discussed in Section 2.4.1.3.6, wildfires do not pose a significant hazard in the region surrounding the Kemmerer Unit 1 site. The land coverage surrounding the NI is predominantly shrub-scrub vegetation, and areas in the proximity to NI buildings is mostly paved. Therefore, the potential for wildfires or brush fires to significantly spread from outside the site boundary across the NI to damage safety-significant SSCs is minimal. The effects of chemical explosions and resulting flammable vapor clouds from


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nearby facilities, pipelines, and transportation accidents are described in Section 2.3.3.1.4. Further evaluation of fire hazards and an analysis of the effects of fires from the TFF and Energy Island SSCs will be provided at the operating license stage.

#### 2.3.3.1.8 Conclusion of Design Basis Hazards Assessment

No hazards were identified that had a probability of occurrence on the order of magnitude of $10^{-7}$ or greater, and had potential consequences serious enough to affect safe plant operations to the extent that the dose guidelines in 10 CFR 50.34(a) could be exceeded, including highway transport of propane on US-189. However, as discussed in Section 2.3.3.1.5, a conservative analysis determined a toxicity hazard to the MCR due to onsite storage of sodium hypochlorite. As discussed in Section 7.5.1, the NI HVAC System is equipped with toxic chemical sensors to alert plant operators of a toxic hazard condition. In addition, as described in Section 7.6.7.2, in case the MCR becomes uninhabitable, plant operators can evacuate to the Remote Shutdown Complex (RSC) located in the NI Control Building substructure as shown on Figure 1.1-12. The MCR and RSC are located in close proximity to support prompt staffing of the RSC in the event of an evacuation of the MCR, and the RSC contains a bottled air injection system to support operator habitability.

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**Table 2.3-1 Description of Facilities - Products and Materials**

| Offsite Facilities | Chemicals and Substances that Require Further Analysis | Distance to Reactor Centerpoint (ft) | Distance to Reactor Centerpoint (mi) |
|---|---|---|---|
| Naughton Power Plant | Ammonium Hydroxide (15-28% as $NH_3$)<br>Unleaded Gasoline<br>Hydrochloric Acid (Conc ≥ 35%)<br>Hydrochloric Acid, Solution (Conc ≤ 35%)<br>Natural Gas (Methane)<br>Propane | 21,384 | 4.05 |
| Kemmerer Mine | Acetylene<br>Airline System Antifreeze<br>Carbon Dioxide, Argon Mix<br>Unleaded Gasoline<br>Nitrogen<br>Oxygen (store in two types of containers - type K and T)<br>Propane | 16,421 | 3.11 |


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**Table 2.3-2 Screening of Chemicals at Nearby Facilities for Chemical Storage and Transport**
**(Sheet 1 of 2)**

| Offsite Facilities and Highway Transport | | | | | | | | | Further Analysis Required? | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| **Offsite Facility Name** | **Chemical or Substance** | **Quantity** | **Physical State** | **Toxicity Limit (IDLH)** | **Vapor Pressure [1]** | **Flammable/ Explosive? Yes/No** | **Disposition** | **Notes** | **Toxicity** | **Flammability/ Explosion** | **Equivalent TNT** |
| Kemmerer Mine | Acetylene | (6) cylinders, 300 $ft^3$ each | Gas | Data unavailable | 638.14 psia @ 68°F | Yes 2.4% - 83% | Flammable and Explosive Vapor Cloud Analysis<br>Toxicity Analysis<br>Equivalent TNT Analysis<br>Transportation Analysis | [4] | Yes | Yes | Yes |
| Kemmerer Mine | Airline System Antifreeze | 156 gal | Liquid | 6,000 ppm | 1.93 psia (100 mmHg) @ 70°F | Yes 6.0% - 36.5% | Flammable and Explosive Vapor Cloud Analysis<br>Toxicity Analysis<br>Equivalent TNT Analysis<br>Transportation Analysis | | Yes | Yes | Yes |
| Naughton Power Plant | Ammonium Hydroxide (15-28% as $NH_3$) | 8,000 lbs | Liquid solution | 300 ppm | 580 mmHg @ 20°C (for 28% solution) | Yes 16% - 25% | Flammable and Explosive Vapor Cloud Analysis<br>Toxicity Analysis<br>Equivalent TNT Analysis<br>Transportation Analysis | [2] | Yes | Yes | Yes |
| Kemmerer Mine | Carbon Dioxide, Argon Mix | 6 cylinders, 300 $ft^3$ each | Liquefied Gas | 40,000 ppm ($CO_2$) | 831.04 psia @ 68°F | No | Toxicity Analysis<br>Transportation Analysis | [4] | Yes | No | Yes |
| Kemmerer Mine | Gasoline, Unleaded | 18,000 gal | Liquid | IDLH (n-heptane): 750 ppm | 5-15 psi @ 100°F | Yes 1.4% - 7.6% | Flammable and Explosive Vapor Cloud Analysis<br>Toxicity Analysis<br>Equivalent TNT Analysis<br>Transportation Analysis | [3] | Yes | Yes | Yes |
| Naughton Power Plant | Gasoline, Unleaded | 29,190 lbs | Liquid | IDLH (n-heptane): 750 ppm | Data unavailable | Yes 1.4% - 7.4% | Flammable and Explosive Vapor Cloud Analysis<br>Toxicity Analysis<br>Equivalent TNT Analysis<br>Transportation Analysis | [3] | Yes | Yes | Yes |
| Naughton Power Plant | Hydrochloric Acid (Conc ≥ 35%) | 541 lbs | Liquid | 50 ppm | 3.287 psi @ 70°F | No | Toxicity Analysis<br>Transportation Analysis | | Yes | No | Yes |
| Naughton Power Plant | Hydrochloric Acid, Solution (Conc ≤ 35%) | 2,450 lbs | Liquid | 50 ppm | 3.287 psi @ 70°F | No | Toxicity Analysis<br>Transportation Analysis | | Yes | No | Yes |
| Kemmerer Mine | Nitrogen | 6 cylinders, 300 $ft^3$ each | Liquefied gas | Asphyxiant (71,400) | 48.44 @ -300°F | No | Toxicity Analysis<br>Transportation Analysis | [4] | Yes | No | Yes |


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**Table 2.3-2 Screening of Chemicals at Nearby Facilities for Chemical Storage and Transport**
**(Sheet 2 of 2)**

| Offsite Facilities and Highway Transport | | | | | | | | | Further Analysis Required? | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| **Offsite Facility Name** | **Chemical or Substance** | **Quantity** | **Physical State** | **Toxicity Limit (IDLH)** | **Vapor Pressure [1]** | **Flammable/ Explosive? Yes/No** | **Disposition** | **Notes** | **Toxicity** | **Flammability/ Explosion** | **Equivalent TNT** |
| Kemmerer Mine | Oxygen K | 3 cylinders, 200 $ft^3$ each | Liquefied gas | Data unavailable | 36.26 psi @ -280°F | No | Toxicity Analysis<br>Transportation Analysis | [4],[5] | Yes | No | Yes |
| Kemmerer Mine | Oxygen T | 6 cylinders, 300 $ft^3$ each | Liquefied gas | Data unavailable | 36.26 psi @ -280°F | No | Toxicity Analysis<br>Transportation Analysis | [4],[5] | Yes | No | Yes |
| Kemmerer Mine | Propane | 1,000 gal | Liquefied gas | 2,100 ppm | 208 psig @ 100°F | Yes<br>2.1% - 9.5% | Flammable and Explosive Vapor Cloud Analysis<br>Toxicity Analysis<br>Equivalent TNT Analysis<br>Transportation Analysis | | Yes | Yes | Yes |
| Naughton Power Plant | Propane | 30,400 lbs | Gas (liquefied flammable gas) | 2,100 ppm | 25.4 psi @ -20°F | Yes<br>2.1% - 9.5% | Flammable and Explosive Vapor Cloud Analysis<br>Toxicity Analysis<br>Equivalent TNT Analysis<br>Transportation Analysis | | Yes | Yes | Yes |
| Naughton Power Plant | Natural Gas (Methane) | --- | Gas | Not listed | 2.640 psi @ -290°F | Yes<br>5.0% - 15% | Flammable and Explosive Vapor Cloud Analysis<br>Equivalent TNT Analysis<br>Transportation Analysis | | No | Yes | Yes |

Notes:

[1] According to RG 1.78, if a substance's vapor pressure is below 10 torr (approximately 10 mmHg); the vapor pressure is sufficiently low such that the formation of a vapor cloud is not a likely event. Therefore, an air dispersion or flammable/explosive vapor cloud hazard is not a likely exposure route.

[2] Ammonia vapor is flammable and explosive, can readily evolve from ammonia solution. The toxicity limit provided for <28% ammonium hydroxide is that of pure ammonia. Additionally, the TNT analysis is run as anhydrous ammonia for conservative values.

[3] The toxicity limit for gasoline is that of n-Heptane.

[4] According to RG 1.78, if there are several chemical containers, the evaluation normally considers only the failure of the largest container unless the containers are interconnected in such a manner that failure of a single container could cause a release from several containers.

[5] Oxygen 'K' and 'T' are designations used for each type of storage container in the Kemmerer Mine facility for specific purposes; these are both the same chemical 'oxygen' for hazard analysis evaluation.

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**Table 2.3-3 Screening of Chemicals at the TFF for Chemical Storage and Transport**

| Offsite Facility Name | Chemical or Substance | Quantity | Largest Container | Physical State | Toxicity Limit (IDLH) | Temperature [2] | | Vapor Pressure [1] [2] | | Further Analysis Required? | Notes | References [5] | Remarks |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | °C | °F | Original Units | Standardized Units (mm Hg) | | | | |
| TFF | Argon | 6,000 gal (1 cryogenic tank) | 6,000 gal (1 cryogenic tank) | Liquid | Asphyxiant (71,400) | Data unavailable | Data unavailable | Data unavailable | Data unavailable | Yes | [3] [4] | (Reference 2.3-9) | Design temperature of -190°F |
| TFF | Argon | 400,000 gal (1 tank) | 400,000 gal atmospheric tank (API 650) & -1°F to 90°F | Gas | Asphyxiant (71,400) | Data unavailable | Data unavailable | Data unavailable | Data unavailable | Yes | [3] [4] | (Reference 2.3-9) | Design temperature of -15°F (min) and 175°F (max) |
| TFF | Nitrogen | 22,000 gal (1 pressure vessel) | 22,000 gal @ atmosphere & 50°F to 200°F in a pressure vessel (ASME VIII) | Gas | Asphyxiant (71,400) | - | -300 | 48.44 psi | 2,505 | Yes | [3] [4] | (Reference 2.3-9) | Design temperatures of -15°F (min) and 180°F (max) |
| TFF | Nitrogen | 300 gal (1 pressure vessel) | 300 gal @ 150 psig & 50°F to 85°F in a pressure vessel (ASME VIII) | Gas | Asphyxiant (71,400) | - | -300 | 48.44 psi | 2,505 | Yes | [3] [4] | (Reference 2.3-9) | Design temperatures of -15°F (min) and 180°F (max) |
| TFF | Nitrogen | 6,000 gal (1 cryogenic tank) | 6,000 gal (1 cryogenic tank) | Liquified Gas | Asphyxiant (71,400) | - | -300 | 48.44 psi | 2,505 | Yes | [3] [4] | (Reference 2.3-9) | Design temperature of -265°F |

Notes:

[1] According to RG 1.78, if a substance's vapor pressure is below 10 torr (approximately 10 mmHg); the vapor pressure is sufficiently low such that the formation of a vapor cloud is not a likely event. Therefore, an air dispersion hazard is not a likely exposure route.

[2] Temperature and vapor pressure values were converted to mmHg and Fahrenheit for comparison to vapor pressure threshold in RG 1.78, as explained in note 1.

[3] Liquified gas flash boils upon release.

[4] The offsite chemical storage scenario for the TFF bounds the chemical transport scenario because chemical storage distance to the Kemmerer Unit 1 MCR is shorter than the highway transport distance from US 189 to the Kemmerer Unit 1 MCR.

[5] Chemical properties information obtained from Reference 2.3-9.

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**Table 2.3-4 Transportation Routes - Highway Information Summary**

| Nearby Transportation Routes - Highways and Roadways | Chemicals and Substances that Require Further Analysis | Transport Quantity | Distance to SR Bounding Area (ft) | Distance to SR Bounding Area (mi) |
|---|---|---|---|---|
| US-189 | Acetylene | 300 $ft^3$ cylinder | 1,035 | 0.196 |
| | Airline System Antifreeze | 156 gal | | |
| | Ammonium Hydroxide (15-28% as $NH_3$) | 2,064 lbs | | |
| | Gasoline, Unleaded | 9,000 gal | | |
| | Propane | 6,000 gal | | |


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## Table 2.3-5 (Not Used)


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**Table 2.3-6 Transportation Routes - Pipeline Information Summary**

| Pipeline ID # | Pipe Diameter | Pipe Name | Commodity | Owner | Shortest Distance Between Pipeline (within 5 mi buffer) to Reactor Centerpoint* (mi) | Operating Pipe Pressure (in psi) | Pipe Temperature |
|---|---|---|---|---|---|---|---|
| 1 | 10” | JL95 | Nat Gas | Dominion Energy Questar | 2.57 | UNK | UNK |
| 2 | 22” | Ignacio/Sumas, NW Pipeline | Nat Gas | Williams Co. | 2.82 | 850 | 55 F |
| 3 | 24” | Ignacio/Sumas Loop, NW Pipeline | Nat Gas | Williams Co. | 2.44 | 850 | 55 F |

Source: Reference 2.3-13



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**Table 2.3-7 Onsite Chemical Storage at Kemmerer Unit 1**

| Chemical | Onsite Storage Location | Distance to MCR (ft) | Toxicity Limit (ppm) |
|---|---|---|---|
| Argon (Liquified Gas) | Onsite - Bulk Plant Gases Building | 500 | Asphyxiant (71,400) |
| Carbon Dioxide (Liquified Gas) | Onsite - Turbine Storage Building | 597 | 40,000 |
| Hydrochloric Acid (≥35%) | Onsite - Cooling Tower | 1,136 | 50 |
| Hydrogen (Gas) | Onsite - Turbine Storage Building | 597 | 71,400 |
| Nitrogen (Gas) | Onsite - Bulk Plant Gases Building | 500 | 71,400 |
| Sodium Hypochlorite (as Chlorine Gas) | Onsite - Water Treatment Building | 620 | 10 |


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**Table 2.3-8 Meteorological Sensitivity Analysis**

| Stability Class | Surface Wind Speed (m/s) | Humidity | Cloud Cover | Date/Time | Temperature (°F) |
|---|---|---|---|---|---|
| A | 1.5 | 50% | 0% | June 21, 2021 / 12 pm | 92 |
| B | 1.5 | 50% | 50% | June 21, 2021 / 12 pm | 92 |
| C | 3 | 50% | 50% | June 21, 2021 / 12 pm | 92 |
| C | 5.5 | 50% | 0% | June 21, 2021 / 12 pm | 92 |
| D | 3 | 50% | 50% | June 21, 2021 / 5 am | 79 |
| D | 5.5 | 50% | 50% | June 21, 2021 / 12 pm | 92 |
| E | 1 | 50% | 50% | June 21, 2021 / 5 am | 79 |
| E | 2 | 50% | 50% | June 21, 2021 / 5 am | 79 |
| F | 1 | 50% | 0% | June 21, 2021 / 5 am | 79 |
| F | 2 | 50% | 0% | June 21, 2021 / 5 am | 79 |
| F | 3 | 50% | 0% | June 21, 2021 / 5 am | 79 |
| **BLEVE/Jetfire Meteorological Sensitivity Analysis** | | | | | |
| F | 1 | 5% | 0% | June 21, 2021 / 5 am | 79 |
| F | 1 | 25% | 0% | June 21, 2021 / 5 am | 79 |
| F | 1 | 50% | 0% | June 21, 2021 / 5 am | 79 |
| F | 1 | 75% | 0% | June 21, 2021 / 5 am | 79 |
| F | 1 | 100% | 0% | June 21, 2021 / 5 am | 79 |

Source: Table developed from Reference 2.3-10 and Reference 2.3-11


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**Table 2.3-9 Offsite Facility Flammability and Explosion Hazards - Flammable Vapor Cloud and Vapor Cloud Explosion ALOHA Results**

| Chemical | Offsite Storage Facility | Analyzed Quantity | LFL | Facility Distance to Reactor Centerpoint (ft) | Facility Distance to Reactor Centerpoint (mi) | To be Modeled? (Y/N) | Assumptions / Bounding Scenarios | Distance to LFL (ft) | Distance to 1.0 psi (ALOHA) (ft) | Distance to 1.0 psi (TNT) (ft) | Distance (ft) to Thermal radiation (5.0 $kW/m^2$) (BLEVE) | Max Thermal Radiation Downwind ($kW/m^2$) (BLEVE) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Acetylene | Kemmerer Mine | 300 ft3 | 2.4% | 16,421 | 3.11 | Y | - | 87 | 117 | 258 | N/A | N/A |
| Airline System Antifreeze | Kemmerer Mine | 156 gal | 6.0% | 16,421 | 3.11 | Y | - | 33 | 66 | 33 | N/A | N/A |
| Ammonium Hydroxide | Naughton Power Plant | 8,000 lbs | 16.0% | 21,384 | 4.05 | Y | - | 93 | 174 | 107 | N/A | N/A |
| Unleaded Gasoline | Kemmerer Mine | 18,000 gal | 1.4% | 16,421 | 3.11 | Y | - | 414 | 858 | 286 | N/A | N/A |
| Unleaded Gasoline | Naughton Power Plant | 29,190 lbs | 1.4% | 21,384 | 4.05 | Y | - | 270 | 600 | 224 | N/A | N/A |
| Propane1 | Kemmerer Mine | 1,000 gal | 2.1% | 16,421 | 3.11 | Y | - | Gaussian 363 Heavy Gas 123 | Gaussian 501 Heavy Gas 318 | 706 | 387 | 0.00207 |
| Propane1 | Naughton Power Plant | 30,400 lbs | 2.1% | 21,384 | 4.05 | Y | - | Gaussian 1,212 Heavy Gas 384 | Gaussian 1,641 Heavy Gas 921 | 1,532 | 819 | 0.00562 |

Notes:

1. Propane releases were run as a direct release using both Gaussian and Heavy Gas to show conservatism for a liquefied flammable gas; note that propane will likely act as heavy gas.


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**Table 2.3-10 Offsite Facility Flammability and Explosion Hazards - Explosion at Source, TNT Equivalency Hand Calculation Results**

| | | | | Calculated for Offsite Storage Quantities | | | | Calculated for Hwy Transport Quantities | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| **Fluid/Solid** | **Offsite Storage Facility** | **Heat of Combustion (Btu/lb)** | **Explosion Yield Factor, α** | **Flammable Mass (lb)** | **TNT Mass Equivalent (lbs)** | **Distance to 1.0 psi Over-pressure Radii, R (ft)** | **Facility Distance to Reactor Centerpoint (ft)** | **Flammable Mass (lbs)** | **TNT Mass Equivalent (lbs)** | **Distance to 1.0 psi Over-pressure Radii, R (ft)** | **Distance to Reactor Centerpoint (ft)** |
| **Atmospheric Liquids** | | | | | | | | | | | |
| Airline System Antifreeze | Kemmerer Mine | 8,419 | 100% | 0.083 | 0.367 | 33 | 16,421 | 0.083 | 0.367 | 33 | 1,035 |
| Ammonium Hydroxide | Naughton Power Plant | 7,992 | 100% | 3.14 | 13.2 | 107 | 21,384 | 3.14 | 13.2 | 107 | 1,035 |
| Gasoline, Unleaded | Naughton Power Plant | 18,720 | 100% | 12.4 | 122 | 223 | 21,384 | 23.4 | 230 | 276 | 1,035 |
| Gasoline, Unleaded | Kemmerer Mine | 18,720 | 100% | 26 | 256 | 286 | 16,421 | 23.4 | 230 | 276 | 1,035 |
| **Liquefied Gases** | | | | | | | | | | | |
| Propane | Kemmerer Mine | 19,782 | 100% | 370 | 3,848 | 705 | 16,421 | 22,176 | 230,887 | 2,761 | 1,035 |
| Propane | Naughton Power Plant | 19,782 | 100% | 3,784 | 39,397 | 1,531 | 21,384 | 22,176 | 230,887 | 2,761 | 1,035 |
| **Pressurized Gases** | | | | | | | | | | | |
| Acetylene | Kemmerer Mine | 20,747 | 100% | 17.2 | 188 | 258 | 16,421 | 17.2 | 188 | 258 | 1,035 |

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**Table 2.3-11 Highway Transport Flammability and Explosion Hazards Results**

| Flammable Vapor Cloud and Vapor Cloud Explosion ALOHA Results for Offsite Chemicals | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **Chemical** | **Chemical Transport Destination (Facility Name)** | **Analyzed Quantity** | **Highway Distance to NI (ft)** | **Distance to LFL (ft)** | **Distance to 1 psi (ALOHA) (ft)** | **Distance to 1 psi (TNT) (ft)** | **Distance (ft) to Thermal Radiation (5.0 kW/m$^2$) (BLEVE)** | **Max Thermal Radiation Downwind (5.0 kW/m$^2$) (BLEVE)** |
| Acetylene | Kemmerer Mine | 300 ft3 | 1,035 | 87 | 117 | 258 | N/A | N/A |
| Airline System Antifreeze | Kemmerer Mine | 156 gal | 1,035 | 33 | 66 | 32.2 | N/A | N/A |
| Ammonium Hydroxide | Naughton Power Plant | 8,000 lbs | 1,035 | 93 | 174 | 106 | N/A | N/A |
| Unleaded Gasoline | Kemmerer Mine | 18,000 gal | 1,035 | 393 | 816 | 276 | N/A | N/A |
| Unleaded Gasoline | Naughton Power Plant | 29,190 lbs | 1,035 | 393 | 816 | 276 | N/A | N/A |
| Propane [1] | Kemmerer Mine | 1,000 gal | 1,035 | Gaussian Mode | Gaussian | 2,761 | 1,443 | 9.76 |
| | | | | 2,952 | 3,741 | | | |
| | | | | Heavy Gas | Heavy Gas | | | |
| | | | | 807 | 1,965 | | | |
| Propane [1] | Naughton Power Plant | 30,400 lbs | 1,035 | Gaussian | Gaussian | 2,761 | 1,443 | 9.76 |
| | | | | 2,952 | 3,741 | | | |
| | | | | Heavy Gas | Heavy Gas | | | |
| | | | | 807 | 1,965 | | | |

Notes:

[1] Deterministic results for highway transport of propane.- Propane releases were run as a direct release using both Gaussian and Heavy Gas air dispersion models to show conservatism for a liquefied flammable gas; note that propane will likely act as heavy gas.


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**Table 2.3-12 Pipeline Flammability and Explosion Hazards Results**

| Pipeline | Pipe Length (ft) | Operating Pressure (psi) | Distance to Reactor Centerpoint (ft) | Distance to Reactor Centerpoint (mi) | Distance to 1 psi (TNT) (ft) | Distance to 1 psi (ALOHA) (ft) | Distance to LFL (ALOHA) (ft) | Distance to 5 kW/m$^2$ (ALOHA Jet Fire) (ft) |
|---|---|---|---|---|---|---|---|---|
| Natural Gas Pipeline Ignacio/Sumas Loop NW, 24 in. | 47,520 (actual pipe length)<br><br>400 ft pipe section (used in ALOHA analysis) | 850 | 12,883 | 2.44 | 10,194 | 4,899 | 1,257 | 1,155 |


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**Table 2.3-13 Onsite Storage Flammability and Explosion Hazards - Flammable Vapor Cloud and Vapor Cloud Explosion ALOHA Results**

| Chemical | Analyzed Quantity | LFL | To be Modeled? (Y/N) | Assumptions / Bounding Scenarios | Nearest Distance to SR Buildings (ft) | Distance to 1psi (TNT) (ft) | Distance to 1 psi (ALOHA) (ft) | | Distance to LFL (ALOHA) (ft) | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | Gaussian | Heavy Gas | Gaussian | Heavy Gas |
| Hydrogen Bottle[1] | 270 scf | 4% | Y | - | 657 | 135 | 87 | 144 | 75 | 36 |
| Hydrogen Rack[1] | 3,240 scf | 4% | Y | - | 657 | 308 | 297 | 186 | 264 | 36 |
| Coupled Hydrogen Rack[1] | 6,480 scf | 4% | Y | - | 657 | 388 | 420 | 204 | 378 | 36 |

Notes:

[1] When the source setup was completed for Hydrogen, ALOHA® reported a warning “This chemical may flash boil, which can result in a two-phase flow. Use both dispersion modules to investigate its potential behavior.” Therefore, a direct release for both Heavy Gas and Gaussian dispersion were used in the analyses.


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**Table 2.3-14 Onsite Storage Flammability and Explosion Hazards - Explosion at Source, TNT Equivalency Hand Calculation Results**

| Fluid/Solid | Heat of Combustion (Btu/lb) | Explosion Yield Factor, $\alpha$ (lbs) | Flammable Mass (lbs) | TNT Mass Equivalent (lbs) | Distance to 1 psi Over-Pressure Radii, R (ft) | Nearest Distance to SR Buildings (ft) |
|---|---|---|---|---|---|---|
| Hydrogen Bottle | 50,080 | 100% | 1.44 | 37.96 | 135 | 657 |
| Hydrogen Rack | 50,080 | 100% | 17.25 | 454.67 | 308 | 657 |
| Coupled Hydrogen Rack | 50,080 | 100% | 34.50 | 909.35 | 388 | 657 |

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**Table 2.3-15 Offsite Facility Toxicity Hazards Results**

| Chemical | Storage / Transport Location | Analyzed Quantity | Toxicity Limit / IDLH (ppm) [1] | Facility Distance to MCR as Analyzed in ALOHA (ft) [2] | Facility Distance to MCR as Analyzed in ALOHA (mi) [2] | Pvapor (mmHg) [3] | Temperature (ºF) [3] | To be Modeled? (Y/N) | Assumptions / Bounding Scenarios | Maximum Distance to Toxicity Limit (ft) | ALOHA Maximum Outdoor Concentration at MCR (ppm) | ALOHA Maximum Indoor Concentration at MCR (ppm) | Maximum Distance to Toxicity Limit Greater than Distance to MCR? | ALOHA Maximum Indoor Concentration Greater than Toxicity Limit? |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Liquefied Gases** | | | | | | | | | | | | | | |
| Argon | TFF | 6,000 gal | Asphyxiant (71,400) | 837 | 0.158 | data unavailable | data unavailable | Y | Use bounding scenario of 6,000 gal of argon stored onsite at bulk plant gases building (500ft). See onsite storage results in Table 2.3-18. | N/A | N/A | N/A | N/A | N/A |
| Nitrogen | TFF | 2,000 gal | Asphyxiant (71,400) | 837 | 0.158 | 2,505 | -300 | Y | - | 948 | 88,200 | 2,570 | Yes | No |
| **Pressurized Gases** | | | | | | | | | | | | | | |
| Argon | TFF | 400,000 gal | Asphyxiant (71,400) | 837 | 0.158 | data unavailable | data unavailable | Y | - | 144 | 8,490 | 1,480 | No | No |

Notes:
Calculated distances to the Kemmerer Unit 1 MCR are rounded down for conservatism.
[1] IDLH or similar toxicity threshold.
[2] RG 1.78 states that all hazardous chemicals within 0.3 mi of the site must be evaluated for toxicity, and Table 2 provides a minimum chemical weight screening only for chemicals stored between 0.3-5 mi from the MCR.
[3] Most vapor pressures reported are shown at standard temperature and pressure.


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**Table 2.3-16 Highway Transport Toxicity Hazards Results**

| Chemical or Substance | Storage / Transport Location | Analyzed Quantity | Toxicity Limit / IDLH (ppm) [1] | Facility Distance to MCR as Analyzed in ALOHA (ft) [2] | Facility Distance to MCR as Analyzed in ALOHA (mi) [2] | Pvapor (mmHg) [3] | Temperature (ºF) [3] | To be Modeled? (Y/N) | Assumptions / Bounding Scenarios | Maximum Distance to Toxicity Limit (ft) | ALOHA Maximum Outdoor Concentration at MCR (ppm) | ALOHA Maximum Indoor Concentration at MCR (ppm) | Maximum Distance to Toxicity Limit Greater than Distance to MCR? | ALOHA Maximum Indoor Concentration Greater than Toxicity Limit? |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Atmospheric Liquids** | | | | | | | | | | | | | | |
| Airline System Antifreeze (as Methanol) | US-189 | 156 gal | 6,000 | 1,371 | 0.259 | 1.93 | 70 | Y | Assume 100% methanol. | 99 | 102 | 53.3 | No | No |
| Ammonium Hydroxide | US-189 | 2,064 lbs | 300 | 1,350 | 0.256 | 11.9 | 80 | Y | Analysis was updated based on revised bounding scenario | 2,577 | 1,130 | 259 | Yes | No |
| Gasoline, Unleaded (as n-Heptane) | US-189 | 9,000 gal | 750 (n-heptane) | 1,371 | 0.259 | 517 | 100 | Y | Assume 100% n-heptane. | 1,122 | 491 | 258 | No | No |
| **Liquefied Gases** | | | | | | | | | | | | | | |
| Propane | US-189 | 6,000 gal | 2,100 | 1,350 | 0.256 | 1,314 | -20 | Y | Analysis was updated based on revised bounding scenario | 3,876 | 12,000 | 2,070 | Yes | No |
| Oxygen T [4] | US-189 | 300 scf | 31,600 | 1,371 | 0.259 | 1,875 | -280 | Y | - | 180 | 155 | 11.7 | No | No |
| **Pressurized Gases** | | | | | | | | | | | | | | |
| Acetylene [5] | US-189 | 300 scf | Asphyxiant (71,400) | 1,371 | 0.259 | 33,453 | 68 | Y | - | 36 | 58.7 | 10.2 | No | No |

Notes:

Calculated distances to the Kemmerer Unit 1 MCR are rounded down for conservatism.

[1] IDLH or similar toxicity threshold.

[2] RG 1.78 states that all hazardous chemicals within 0.3 mi of the site must be evaluated for toxicity, requiring minimum chemical weight screening only for chemicals stored between 0.3-5 mi from the MCR. Transport distances are 0.26 mi from the site, so they cannot be evaluated based on chemical weight.

[3] Most vapor pressures reported are shown at standard temperature and pressure.

[4] An oxygen-enriched atmosphere is defined by OHSA as an atmosphere with 23.5%+ oxygen.

[5] Acetylene cylinders are not considered interconnected, 6 cylinders 300 $ft^3$ each to be run as single container, 300 ft3 analysis.


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**Table 2.3-17 Pipeline Toxicity Hazards Results**

| Pipeline | Pipe Length (ft) | Operating Pressure (psi) | Toxicity Limit / IDLH (ppm) [1] | Pvapor (mmHg) [3] | Temperature (ºF) [3] | Assumptions/ Bounding Scenarios |
|---|---|---|---|---|---|---|
| Natural Gas Pipeline Ignacio/ Sumas Loop NW, 24in. | 47,520 | 850 | Not listed | 760 | 59 | A toxicity analysis is not required because methane does not have a reported toxicity limit (IDLH or time weighted average). |

NOTE: Calculated distances to the Kemmerer Unit 1 MCR are rounded down for conservatism.

[1] IDLH or similar toxicity threshold.

[2] RG 1.78 states that all hazardous chemicals within 0.3 mi of the site must be evaluated for toxicity, requiring minimum chemical weight screening only for chemicals stored between 0.3-5 mi from the MCR. Transport distances are 0.26 mi from the site, so they cannot be evaluated based on chemical weight.

[3] Most vapor pressures reported are shown at standard temperature and pressure.


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**Table 2.3-18 Onsite Storage Toxicity Hazards Results**

| Chemical | Storage / Transport Location | Analyzed Quantity | Toxicity Limit / IDLH (ppm) [1] | Distance to MCR (ft) [2] | Distance to MCR (mi) [2] | Pvapor (mmHg) [3] | Temperature (ºF) [3] | To be Modeled? (Y/N) | Assumptions / Bounding Scenarios | Maximum Distance to Toxicity Limit (ft) | ALOHA Maximum Outdoor Concentration at MCR (ppm) | ALOHA Maximum Indoor Concentration at MCR (ppm) | Maximum Distance to Toxicity Limit Greater than Distance to MCR? | ALOHA Maximum Indoor Concentration Greater than Toxicity Limit? |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Atmospheric Liquids** | | | | | | | | | | | | | | |
| Hydrochloric Acid (Conc ≥35%) | Onsite - Cooling Tower | 2,450 lbs | 50 | 1,136 | 0.215 | 170 | 70 | Y | Use bounding scenario of 2,450 lbs of HCl at 42% concentration stored onsite at cooling tower chemical storage building (1,050 ft). | 2,943 | 324 | 43.6 | Yes | No |
| Sodium Hypochlorite | Onsite - Water Treatment Building | 6,000 gal | 10 (chlorine) | 620 | 0.117 | 18 | 68 | Y | - | 870 | 15.6 | 10.6 | Yes | Yes |
| **Liquefied Gases** | | | | | | | | | | | | | | |
| Argon | Onsite - Bulk Plant Gases Building | 6,000 gal | Asphyxiant (71,400) | 500 | 0.095 | data unavailable | data unavailable | Y | Bounding scenario for analysis of the largest quantity of liquified argon at the closest location to the MCR. | 741 | 111,000 | 1,930 | Yes | No |
| Carbon Dioxide | Onsite - Turbine Storage Building | 5,170 scf | 40,000 | 597 | 0.113 | 44,061 | 70 | Y | - | 195 | 5,450 | 242 | No | No |
| **Pressurized Gases** | | | | | | | | | | | | | | |
| Hydrogen | Onsite - Turbine Storage Building | 3,240 scf | Asphyxiant (71,400) | 597 | 0.113 | 1,501.30 | -418 | Y | - | 120 | 3,090 | 546 | No | No |
| Nitrogen | Onsite - Bulk Plant Gases Building | 71,738 scf | Asphyxiant (71,400) | 500 | 0.094 | 2,505 | -300 | Y | Use bounding scenario of 71,738 scf of nitrogen stored onsite at Argon Bulk Storage Bldg (500 ft). | 585 | 96,600 | 17,100 | Yes | No |

Notes:

Calculated distances to the Kemmerer Unit 1 MCR are rounded down for conservatism.

[1] IDLH or similar toxicity threshold.

[2] RG 1.78 states that all hazardous chemicals within 0.3 mi of the site must be evaluated for toxicity.

[3] Most vapor pressures reported are shown at standard temperature and pressure.

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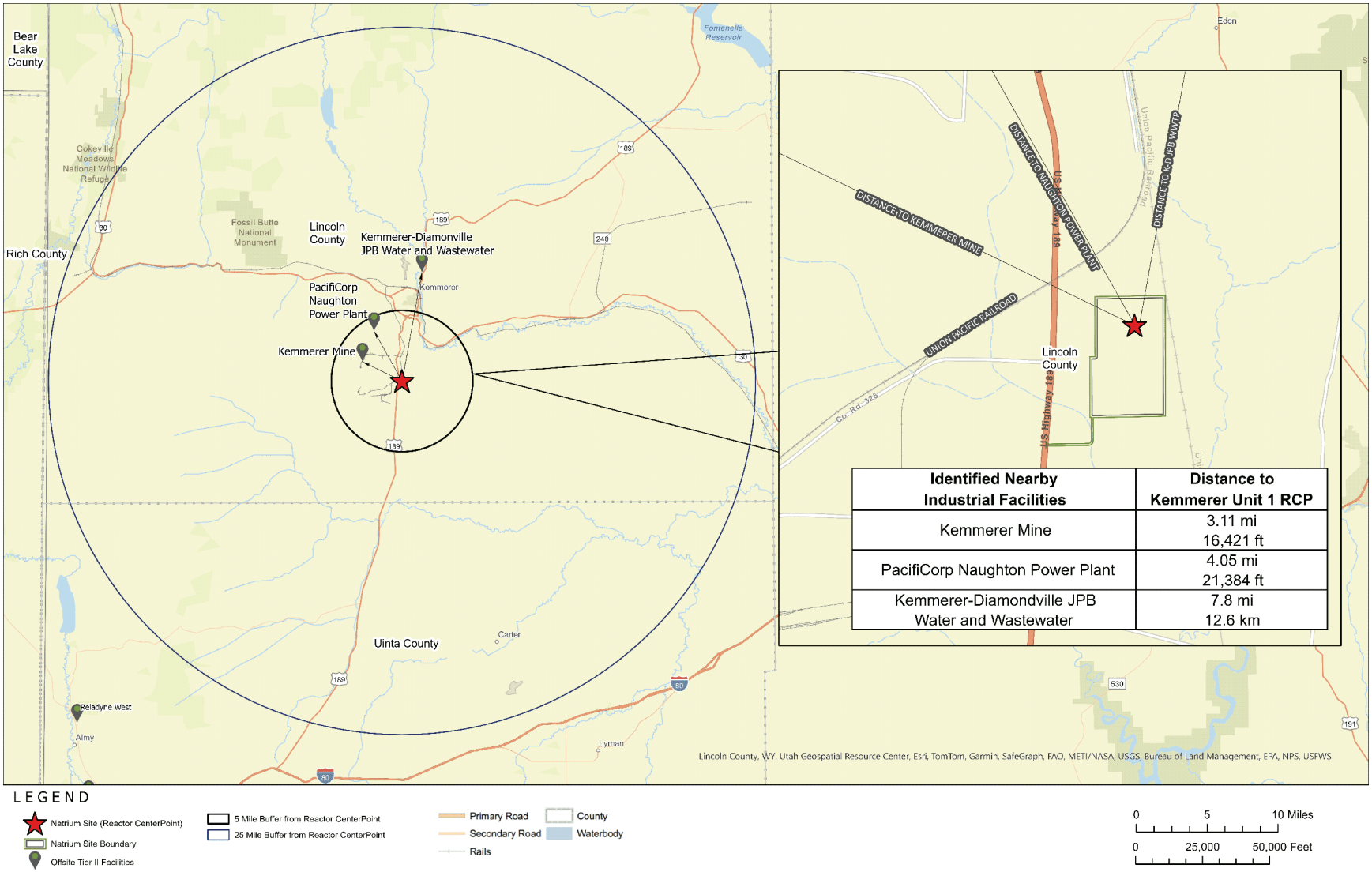
## Figure 2.3-1 Nearby Industrial Facilities and Transportation Routes within 10 Mi




| Identified Nearby Industrial Facilities | Distance to Kemmerer Unit 1 RCP |
|---|---|
| Kemmerer Mine | 3.11 mi<br>16,421 ft |
| PacifiCorp Naughton Power Plant | 4.05 mi<br>21,384 ft |
| Kemmerer-Diamondville JPB Water and Wastewater | 7.8 mi<br>12.6 km |

LEGEND

- Natrium Site (Reactor CenterPoint)
- Natrium Site Boundary
- Offsite Tier II Facilities
- 5 Mile Buffer from Reactor CenterPoint
- 25 Mile Buffer from Reactor CenterPoint
- Primary Road
- Secondary Road
- Rails
- County
- Waterbody

0 5 10 Miles

0 25,000 50,000 Feet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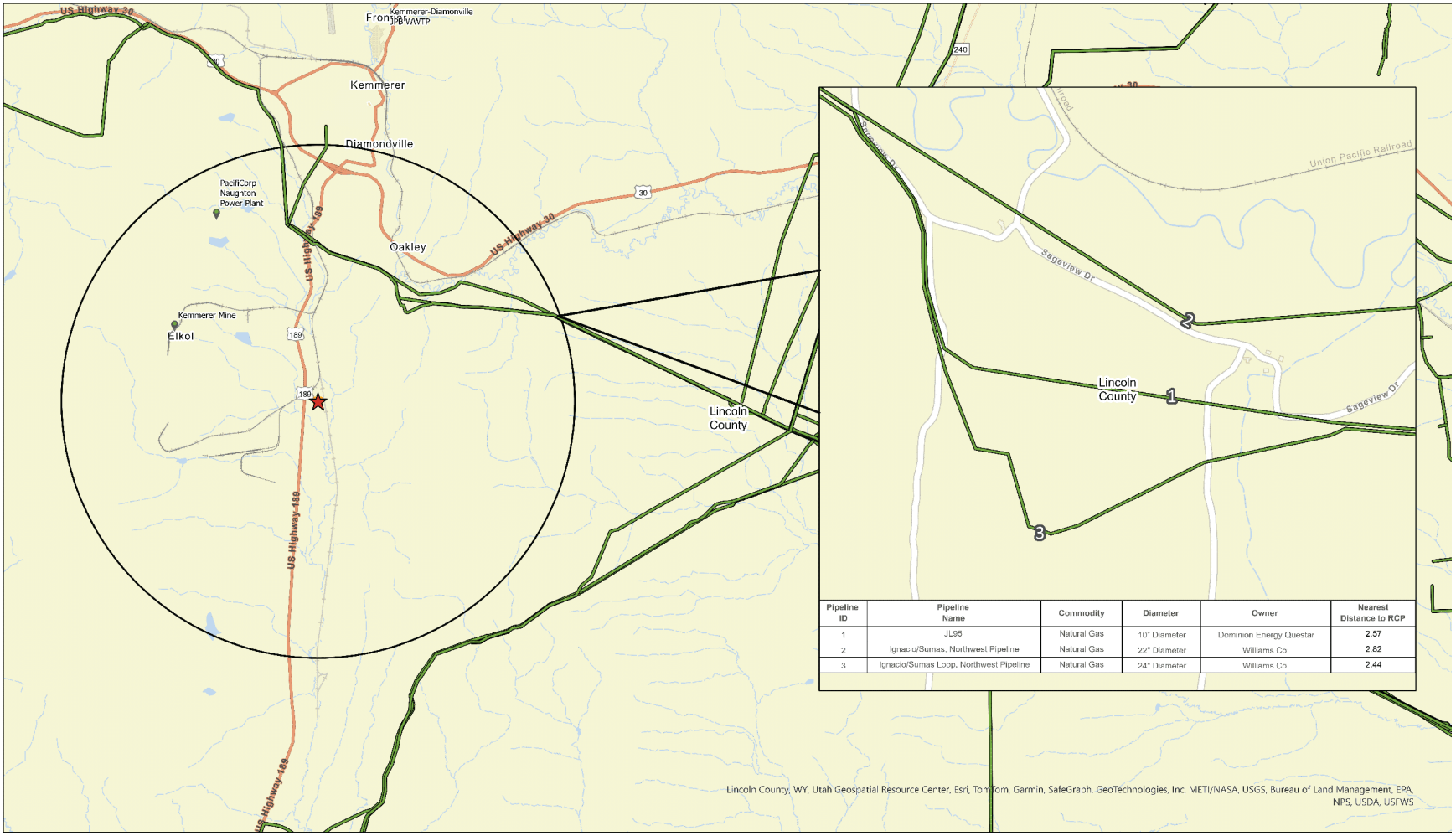
## Figure 2.3-2 Pipelines within 5 Mi




| Pipeline ID | Pipeline Name | Commodity | Diameter | Owner | Nearest Distance to RCP |
|---|---|---|---|---|---|
| 1 | JL95 | Natural Gas | 10" Diameter | Dominion Energy Questar | 2.57 |
| 2 | Ignacio/Sumas, Northwest Pipeline | Natural Gas | 22" Diameter | Williams Co. | 2.82 |
| 3 | Ignacio/Sumas Loop, Northwest Pipeline | Natural Gas | 24" Diameter | Williams Co. | 2.44 |

LEGEND

- Natrium Reactor Centerpoint
- Offsite Tier II Facilities
- 5 Mile Buffer from Reactor CenterPoint
- Wyoming Pipelines (Natural Gas)
- Safety Bounding Area
- Primary Road
- Secondary Road
- Rails
- County
- Waterbody

0 1.5 3 Miles

0 7,500 15,000 Feet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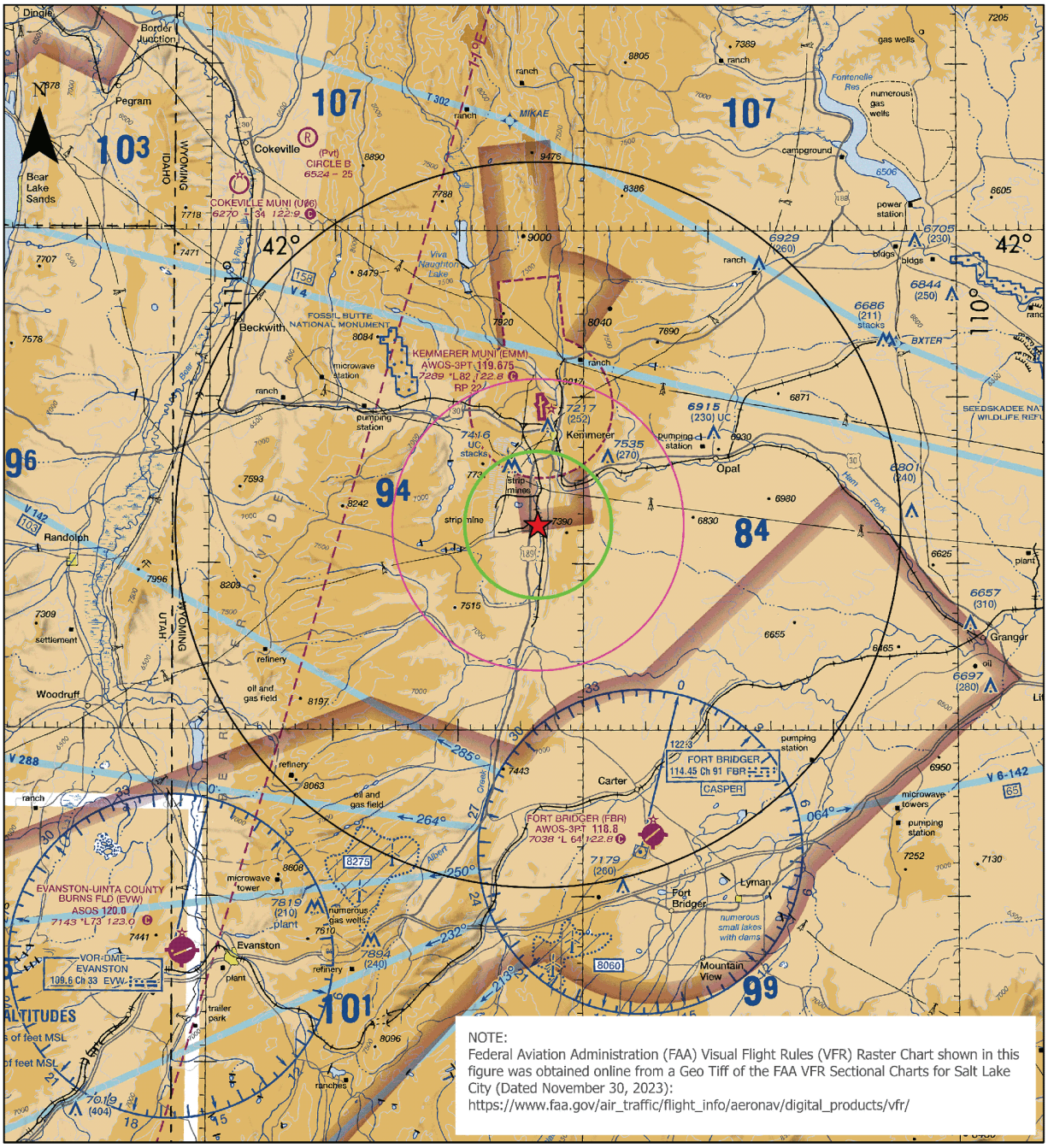
## Figure 2.3-3 Airports and Federal Airways within 10 Mi




## FAA VFR Sectional Chart - Kemmerer Unit 1 Site Map

LEGEND

- Kemmerer Unit 1 Site
- 5 Mile Buffer
- 10 Mile Buffer
- 25 Mile Buffer

0 8 16 Miles
0 40,000 80,000 Feet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# 2.4 Regional Climatology, Local Meteorology, and Atmospheric Dispersion

## 2.4.1 Regional Climatology

The regional climatology is defined as the climate within a radius of approximately 50 miles (80 kilometers) of the site. A climatological summary of normal and extreme values of relevant meteorological parameters is presented for the first-order National Weather Service (NWS) station or Automated Surface Observing System with no human augmentation. A first-order station measures a wide number of meteorological parameters, operates 24 hours a day, and is maintained by NWS trained and certified staff. Automated Surface Observing System works non-stop, updates information every minute, measures critical aviation and meteorological parameters, and serves as the primary surface weather observing network of weather stations.

### 2.4.1.1 Data Sources

Several sources of data are used to characterize local and regional climatological conditions pertinent to the Kemmerer Unit 1 Site. Global Historical Climatology Network daily and Local Climatological Data first-order stations were considered. Station selection varied with respect to the parameter evaluated (i.e., requisite data availability and coverage).

Meteorological stations were evaluated within 50 miles (80 kilometers) of the site to determine the representativeness and applicability for use in determining extreme weather values using the Federal Aviation Administration, *Surface Weather Observation Stations* (Reference 2.4-18) and results are shown in Table 2.4-1. The use of nearby offsite climatological monitoring stations ensures that the mean and extreme values measured at those locations are reasonably representative of conditions that are expected to be observed at the site. Based on this evaluation, the Evanston-Unita County Burns Airfield Station is the most representative station for the determination of dry bulb and wet bulb temperatures. The Wyoming Kemmerer 2N and Wyoming Evanston 1E COOP stations are used to perform the design-basis snow loads.

Data acquired by the NWS at the Kemmerer Municipal Airport AWOS-3PT station in Lincoln County, Wyoming and from seven other nearby locations in the network of cooperative observer stations were used to evaluate precipitation, thunderstorm activity, and lightning activity as compiled and summarized by the National Climatic Data Center. These climatological observing stations are located in Lincoln, Teton, Sublette, and Sweetwater Counties in Wyoming, Bear Lake County in Idaho, and Cache County in Utah.

Table 2.4-1 identifies the specific stations used for weather-related events and climate data and lists their approximate distance and direction from the site Reactor Building (RXB) center point. Figure 2.4-1 illustrates the station locations relative to the site used for regional weather data.

The identification of stations included considered the following:

- Proximity to the site (i.e., within the nominal 50-mile (80-kilometer) radius indicated above)
- Topographical characteristics
- Terrain elevation


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- Land use (urban versus rural)
- Requisite and adequacy of the data collection (long contiguous datasets with limited data gaps)

If an overall precipitation or temperature condition was identified for a station located beyond the nominal 50 miles (80 kilometers) and that event was representative for the site, such stations were also included, regardless of directional coverage.

Data sources used in describing the climatological characteristics of the region include:

- Wyoming Agricultural Network Water Resources Data System & State Climate Office - *Snow, Water, Climate data* (Reference 2.4-1)
- National Oceanic and Atmospheric Administration (NOAA) National Centers for Environmental Information (NCEI) Wyoming State Climate Summary 2022 (Reference 2.4-2)
- Local Climatological Data Station Details. Evanston Uinta Co Burns Field, Wyoming U.S., WBAN:04111. NOAA (Reference 2.4-3)
- Global Historical Climatology Network daily (GHCNd)

#### 2.4.1.2 General Climate

The site, located in the southwestern portion of Wyoming, falls under the Köppen climate classification of Dfb (warm-summer humid continental), which is in a region affected by both continental and mountainous influences according to Yearly & Monthly Weather (Reference 2.4-4). Since the mountain ranges lie in a general north-south direction, they are perpendicular to the prevailing westerlies. Therefore, the mountain ranges provide effective barriers, which force the air currents moving in from the Pacific Ocean to rise and drop much of their moisture along the western slopes. Wyoming is considered semi-arid east of the mountains. The prevailing westerly winds bring weather systems and moisture from the Pacific Ocean, while the Rocky Mountains to the west block some of these systems, resulting in a rain shadow effect.

Kemmerer, Wyoming experiences a semi-arid climate with distinct seasonal variations. According to *The Climate of Wyoming - Wyoming State Climate Office and Water Resources Data System* (Reference 2.4-5) and NOAA NCEI State Climate Summaries 2023 (Reference 2.4-20) for Wyoming, high-pressure systems often bring fair weather, clear skies, and calm conditions to Kemmerer. These systems are associated with descending air and typically result in dry conditions. Low-pressure systems, on the other hand, can bring more variable and dynamic weather. They are associated with rising air and often lead to cloudiness, precipitation, and sometimes thunderstorms. Frontal systems, such as cold fronts and warm fronts, influence weather patterns in Kemmerer. Cold fronts may bring cooler temperatures and precipitation, while warm fronts bring milder temperatures and the potential for rain.

Wyoming has a relatively cool climate because of its elevation. Above 6,000 feet (1829 meters), the temperature rarely exceeds 100 degrees Fahrenheit (37.8 degrees Celsius). The warmest parts of Wyoming are found in the lower portions of the Big Horn Basin, the lower elevations of the central and northeast portions of Wyoming, and along the eastern border of Wyoming. Temperatures in Wyoming have risen about 2.5 degrees Fahrenheit (1.4 degrees Celsius) since


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the beginning of the 20 century. Kemmerer experiences a wide range of temperatures throughout the year. Winters are typically cold, with temperatures often dropping below freezing. Summers have temperatures reach into the 80s degrees Fahrenheit (upper 20s to low 30s degrees Celsius) and occasionally higher.

The region has low humidity levels especially during the summer months. This low humidity contributes to dry conditions and is typical of semi-arid climates.

Precipitation in Kemmerer is low and the area is prone to drought conditions. Most precipitation occurs during the spring and early summer, with occasional thunderstorms. Winters are drier with snowfall being the primary form of precipitation. Wyoming, like the rest of the Great Plains, is susceptible to droughts, which are occasionally severe. From 1999 to 2008, large portions of the state experienced drought conditions. Wyoming then experienced several years of above average precipitation until 2012, which was Wyoming's driest year since historical records began in 1895. By October 2012, almost 90 percent of the state was in "severe" drought (the U.S. Drought Monitor's third-highest category of drought severity).

The area around the site out to 5 miles (8 kilometers) is relatively flat. The nearest ridgeline is located approximately 0.5 miles (0.8 kilometers) to the east and follows a north-south orientation. Table 2.4-2 illustrates the terrain heights in each direction out to 5 miles (8 kilometers). Figure 2.4-63 shows the local topography surrounding the site. The elevation increases approximately 3,300 feet (1,000 meters) 50 miles (80 kilometers) to the north and north-northwest of the site. *The Climate of Wyoming - Wyoming State Climate Office and Water Resources Data System* (Reference 2.4-12) was used to generate Table 2.4-3 and Figure 2.4-64, which depict the topography within a 50-mile (80-kilometer) radius of the site.

### 2.4.1.3 Severe Weather

Severe weather phenomena affecting the site and the region are considered in the design and operating bases.

#### 2.4.1.3.1 Thunderstorms, Hail, and Lightning

Lincoln County has reported 22 thunderstorm wind events over the 30-year period from 1992–2022 in Lincoln County, 6 of which were in Kemmerer. Kemmerer, Wyoming experiences thunderstorms throughout the year, with the peak season typically occurring from late spring to early fall. These storms bring heavy rainfall, strong winds, lightning, and occasional hail.

Hail events occur during the peak thunderstorm season. Lincoln County, Wyoming reported 31 hail events over the 30-year period from 1992–2022, 5 of which occurred in Kemmerer, Wyoming. Table 2.4-10 includes historical hail event data for Lincoln County.

Thunderstorms in Kemmerer produce frequent lightning strikes, particularly during the summer months. *Lightning Protection* (Reference 2.4-8) presents a methodology for estimating lightning strike frequencies, which includes consideration of the attractive area of structures. The method consists of determining the number of lightning flashes to earth per year per square kilometer and then defining an area over which the structure is expected to attract a lightning strike. Using


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the methods prescribed in Reference 2.4-8, approximately three lightning flashes per year occur in the vicinity of the site (conservatively estimated using data from Diurnal Variations of NLDN-Reported Cloud-to-Ground Lightning in the United States (Reference 2.4-13).

#### 2.4.1.3.2 Extreme Winds

No hurricane storms have been reported within 100-miles of the site based on the data from NOAA's Office for Coastal Management Historical Hurricane Tracks (Reference 2.4-9).

A total of 485 high wind events were recorded within a 100-mile radius of the Kemmerer Unit 1 Site area during the data collection period of January 1, 1996 to August 31, 2023 (27.7 years). The first and last high wind events recorded during this data collection period occurred on January 8, 1996, and April 18, 2023, respectively. Each event was assumed to be at its maximum intensity. The frequency of high wind events within 100-miles of the site, with respect to each category corresponding to the Saffir-Simpson Hurricane Categories, is listed in Table 2.4-11. The highest wind speeds in Category 3 and 4 occurred in the mountainous and forested regions in the surrounding area of the site that are not representative of the site location. The safety-significant SSCs on the site are able to withstand the wind speeds provided in Table 2.4-11.

For the structural design of safety-significant SSCs, a "basic" 110 and 115 miles per hour wind speed is used for wind loading conditions, which is the 3-second gust speed, Exposure Category C, for Risk Category III and IV buildings, respectively, and other structures as defined in ASCE/SEI 7-16 (Reference 2.4-21). Further details are provided in Section 6.4.2.2.

#### 2.4.1.3.3 Tornadoes

Wyoming experiences a low frequency of tornadoes. Tornadoes have been observed during every month of the year, though approximately 50 percent of the annual total occurred during July. Tornado intensity is measured by the EFScale. The weakest intensity (EF0) winds, 40–72 miles per hour (8–32 meters per second), occur more than half the time, producing minimal damage. Only two tornado events (EF1 on June 14, 1997 and EF0 on July 10, 2001) have been recorded in Lincoln County (La Barge and Opal) from 1992–2022 based on information from the NCEI Storm Events Database, Tornadoes (Reference 2.4-10).

The design-basis tornado characteristics applicable to safety-significant SSCs include the following parameters as identified in Regulatory Guide (RG) 1.76, Revision 1, "Design-basis Tornado and Tornado Missiles for Nuclear Power Plants":

- Maximum wind speed
- Translational speed
- Maximum rotational speed
- Radius of maximum rotational speed
- Pressure drop
- Rate of pressure drop


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Based on Figure 1 of RG 1.76, the site is located within Tornado Intensity Region III. Therefore, design-basis tornado characteristics for Tornado Intensity Region III that apply to the plant property are:

- Maximum wind speed = 160 miles per hour (72 meters per second)
- Translational speed = 32 miles per hour (14 meters per second)
- Maximum rotational speed = 128 miles per hour (57 meters per second)
- Radius of maximum rotational speed = 150 feet (45.7 meters)
- Pressure drop = 0.6 psi (40 millibars)
- Rate of pressure drop = 0.2 psi/s (13 millibars per second)

Table 2 of RG 1.76 provides the tornado missile spectrum and horizontal speeds for automobiles, schedule 40 pipe, and solid spheres. The tornado missile characteristics for the site in Region III are:

**Tornado Horizontal Missile Velocity:**

- Auto: 79 ft/s (54 mph)
- Pipe: 79 ft/s (54 mph)
- Sphere: 20 ft/s (13.6 mph)

**Tornado Vertical Missile Velocity:**

The vertical missile velocity for all missiles is 67 percent of the horizontal velocity. The vertical velocity for tornado missiles is as follows:

- Auto: 79 ft/s x 0.67 = 52.93 ft/s (36.1 mph)
- Pipe: 79 ft/s x 0.67 = 52.93 ft/s (36.1 mph)
- Sphere: 20 ft/s x 0.67 = 13.4 ft/s (9.1 mph)

Using 73.7 years of NCEI compiled storm data, 49 tornadoes ranging in intensity from EF0 to EF2 on the Enhanced Fujita-Pearson scale have been tracked within a 2-degree latitude box around the Kemmerer Unit 1 Site. No tornadoes with an EF3, EF4 or EF5 intensity were recorded. Table 2.4-12 shows the distribution of tornadoes and the calculated strike probability consistent with the methodology in NUREG/CR-4461, within the 2-degree box surrounding the site that ranged from 9.47E-08 to 2.20E-07 strikes per year ($Yr^{-1}$).

The site is not located near large bodies of surface water necessary to generate waterspouts. Of the 73.7 years of tornado data reviewed, no tornadoes were defined as waterspouts based on information in the NCEI Storm Events Database, Waterspouts (Reference 2.4-27).


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2.4.1.3.4 Tropical Cyclones and Hurricanes

The site is located far from coastal areas, therefore the likelihood of hurricanes and tropical cyclones affecting the region is extremely low. No known hurricanes or tropical cyclones were observed in the region based on the NOAA's Coastal Management Historical Hurricane Tracks (Reference 2.4-9), thus, hurricane missiles are not applicable to the site. However, increased rainfall and strong winds are possible at the site. High winds are assessed in Section 2.4.1.3.2.

2.4.1.3.5 Winter Precipitation

**Freezing Rain and Ice Storms**

There have been zero freezing rain or ice storm events recorded from the NOAA Storm Events Database (Reference 2.4-24) during the period of January 1996 through December 31st, 2023.

**Snowfall, Snow Depth, and Snow Loads**

Table 2.4-13 presents mean monthly and annual snowfall totals at sites surrounding Kemmerer Unit 1. Snowfall is common from November through May with stations at lower elevations experiencing 5 events per year, on average, exceeding 5 inches. A single storm producing 10–15 inches of snowfall occurs but is infrequent outside of the mountain areas. Due to the winds that frequently accompany snowstorms, drifts often occur that make it difficult to accurately measure snowfall. Notable heavy snow occurred in Sheridan (north-central Wyoming) from April 3$^{rd}$ through the 4$^{th}$ in 1955. Snowfall amounted to 39 inches (99 centimeters), water equivalent of 4.30 inches (10.9 centimeters) and blizzard conditions lasted more than 43 hours. These conditions sometimes last 1–2 days but rarely does a severe blizzard last over 3 days.

Total annual snowfall varies throughout Wyoming. Snowfall ranges 60–70 inches (152–178 centimeters) at lower elevations in the east, 45–55 inches (114–140 centimeters) over drier portions in the southwest, 15–20 inches (38–51 centimeters) in the Big Horn Basin lower portion, and 30–40 inches (76–102 centimeters) on the sides where the Basin elevation ranges from 5,000–6,000 feet (1,524–1,829 meters). The mountain areas experience higher annual ranges of snowfall, well over 200 inches (500 centimeters). Table 2.4-14 shows the 24-hour probable maximum precipitation values for the winter season only for Kemmerer 2N and Evanston 1E stations, as well as the 48-hour probable maximum winter precipitation (PMWP) for both stations. Both stations have the same probable maximum precipitation and PMWP values of 8.2 inches (21 centimeters) and 7.6 inches (19 centimeters), respectively, resulting in a 48-hour PMWP of 39.52 pounds per square foot.

Roof loads due to the extreme winter precipitation event are the higher roof load resulting from either the extreme frozen winter precipitation event or the extreme liquid precipitation event plus the normal winter precipitation ground snow loads. The values are converted to extreme winter precipitation event live roof loads. The antecedent snow load on non-safety-related with special treatment (NSRST) roofs is determined to be 70.72 psf based upon the Evanston, Wyoming location.


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Table 2.4-15 displays the summary of both normal and extreme winter precipitation events as well as resultant ground snow loads for the site. Data from the Evanston 1E and Kemmerer 2N meteorological stations were used to calculate the highest ground-level weight among normal winter precipitation events, which is 29.79 pounds per square foot for Kemmerer 2N and 31.2 pounds per square foot for Evanston 1E. The extreme frozen winter precipitation event (the higher ground-level weight between the 100-year return period snowfall and the historical maximum snowfall event) is 18.33 pounds per square foot for Kemmerer 2N and 31.2 pounds per square foot for Evanston 1E. Additionally, based on an extreme liquid precipitation event of 39.52 pounds per square foot, the antecedent snowpack weight is 69.31 pounds per square foot for Kemmerer 2N and 70.72 pounds per square foot for Evanston 1E. These values are considered conservative compared to ponding on the roofs.

#### 2.4.1.3.6 Droughts, Dust Storms, and Wildfires

The probability and frequency of dust storms in Kemmerer varies depending on weather patterns and specific conditions. There have been zero dust storm events reported in Lincoln County, Wyoming over the 30-year period from 1992–2022 according to information from the NCEI Storm Events Database, Dust Storm (Reference 2.4-26).

As mentioned in Section 2.4.1.2, the region tends to have low humidity levels, especially during the summer months. This low humidity contributes to dry conditions and is typical of semi-arid climates. Wyoming is the 5$^{th}$ driest state and has been affected by moderate to severe drought since 1999. Drought is an ongoing concern in Kemmerer and the region has experienced a range of drought events over the years. Notable droughts include the 1930s Dust Bowl Drought and the 1950s Drought, which led to dust storms and economic hardship for many residents in Wyoming. During the 2000-2020 time period, Kemmerer and the western United States experienced intermittent droughts which have impacted water resources and contributed to wildfires.

The NOAA storm events database (Reference 2.4-22) tracks wildfires. There were 24 wildfires in Lincoln County, Wyoming during the period of record from January of 1996 to December 31st, 2023 (Table 2.4-92). Lincoln County encompasses a larger area than a 5-mile radius from the site and this is considered a conservative number of wildfires. The return interval of fires in the sagebrush steppe environment of the site ranges from 35 to 125 years time based on Wildland Fire in Wyoming: Patterns, Influences, and Effects (Reference 2.4-23). Wildfires are not considered to be an important factor in the region surrounding the site.

During 2012, drought, along with high temperatures and high winds, resulted in one of Wyoming’s worst wildfire seasons, with more than 350,000 acres (142,000 hectares) burned. The wildfire burned more area than three times the yearly average from 2002 to 2020 of 113,000 acres (45,700 hectares). Another severe wildfire season occurred in 2020. By October 2020, about 60 percent of Wyoming was in severe drought and almost 340,000 acres (140,000 hectares) had burned.


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### 2.4.1.4 Meteorological Data for Evaluating the Ultimate Heat Sink

Monthly design dry bulb and mean coincident wet bulb temperatures were determined by using 30 years (1987 to 1990, 1992 to 1999, and 2004 to 2021) of meteorological data from Evanston, Wyoming, a nearby representative site.

The maximum two percent annual exceedance design values are 81 degrees Fahrenheit (27.2 degrees Celsius) and 54.7 degrees Fahrenheit (12.6 degrees Celsius) for the dry bulb and mean coincident wet bulb temperatures, respectively. Additionally, the minimum 100 percent annual exceedance values are -29 degrees Fahrenheit (-33.9 degrees Celsius) and 81 degrees Fahrenheit (27.2 degrees Celsius) for the dry bulb and mean coincident wet bulb temperatures, respectively.

Since there are no mechanical draft cooling towers with water storage ponds that comprise an ultimate heat sink for Kemmerer Unit 1, additional meteorological conditions to consider such as the maximum 1-hour dry bulb temperature, minimum cooling due to evaporation of water, high wet bulb temperature values, potential for water freezing, low dry bulb temperature values, and associated wind speeds are not necessary. See Section 7.2.1 for the discussion of protection of meteorological effects on the Reactor Air Cooling System.

According to the American Society of Heating, Refrigeration, and Air Conditioning Engineers Weather Data Viewer (Reference 2.4-6), the 100-year return period values of maximum and minimum dry bulb temperature are 103.6 degrees Fahrenheit (39.8 degrees Celsius) and -32.8 degrees Fahrenheit (-36.0 degrees Celsius), respectively.

### 2.4.1.5 Design-Basis Dry and Wet Bulb Temperatures

Table 2.4-7 and Table 2.4-8 present the monthly design wet bulb temperature and the mean non-coincident dry bulb temperature for the site and Evanston, Wyoming, respectively. Table 2.4-7 presents the monthly maximum, minimum, and average dry bulb and wet bulb temperatures, as well as relative humidity, obtained from the site 10-meter (33-foot) meteorological tower. For Table 2.4-8, these wet bulb temperature values correspond to 0.4 percent, 2.0 percent, 5.0 percent, and 10.0 percent annual exceedance values. These data were determined from Reference 2.4-6.

The hourly observations collected over a 30-year period (1987 to 1990, 1992 to 1999, and 2004 to 2021) at Evanston, Wyoming indicate that the 0.4 percent exceedance maximum dry bulb temperature is 92 degrees Fahrenheit (33.3 degrees Celsius). The maximum dry bulb temperature for the 100-year return period is 103.6 degrees Fahrenheit (39.8 degrees Celsius) using Reference 2.4-6. The hourly observations collected over a 30-year period (1992-2021) at Evanston, Wyoming indicate that the 100 percent exceedance minimum dry bulb temperature is -29 degrees Fahrenheit (-33.9 degrees Celsius). The minimum dry bulb temperature for a 100-year return period is estimated to be -32.8 degrees Fahrenheit (-36.0 degrees Celsius). The 0 percent exceedance maximum wet bulb temperature is 65 degrees Fahrenheit. The maximum non-coincident wet bulb temperature for a 100-year return period is estimated to be 69.5 degrees Fahrenheit (20.8 degrees Celsius).


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### 2.4.1.6 Restrictive Dispersion Conditions

Restrictive dispersion conditions considered include topography, temperature inversions, wind patterns, atmospheric stability, weather conditions, emissions sources, geographic features, and urban development, as described in *Air Stagnation Climatology for the United States (1948-1998)*, (Reference 2.4-19). Kemmerer, being situated in a valley surrounded by hills and mountains, is located at a relatively high elevation, and its proximity to geological and hydrological features could cause impacts to dispersion characteristics. Since the releases at the site are at ground level, the dispersion at the site is considered poor and unaffected by the surrounding topography as the site has very small variations in elevation, approximately 27 feet (8.2 meters).

Mixing height data were used with the air dispersion models to assess pollutant exposures and were acquired at the closest and most representative weather station that recorded these data in Riverton, Wyoming.

### 2.4.1.7 Climate Change

Most of Wyoming has warmed by one to three degrees Fahrenheit (0.6 to 1.7 degrees Celsius) during the last century, according to an Environmental Protection Agency (EPA) 2016 study, What Climate Changes Means for Wyoming (Reference 2.4-25). Heat waves are becoming more common, and snow is melting earlier in the spring. As the temperatures increase, less precipitation falls as snow and more snow melts during the winter. Since the 1950s, the snowpack in Wyoming has been inversely related to the increasing temperatures.

Warming of the atmosphere over Wyoming can also lead to a shifting of the tree line and decrease the extent of the alpine tundra ecosystems. The mountains in Wyoming contain 1,500 glaciers, and the warming will cause these glaciers to retreat or disappear altogether. With the retreat of the glaciers and decreasing snowfall, Wyoming could be in danger of extended droughts that would also mean evaporation of water resources. These extended droughts and drier soils are likely to increase the severity, frequency, and extent of wildfires.

Temperature statistics from locations around the site are presented in Table 2.4-74 through Table 2.4-76. There are NWS stations located in Western Wyoming at Kemmerer Airport and Big Piney Marbleton Airport. These tables provide monthly mean, maximum, and minimum dry bulb temperature values, respectively, for the 30-year period from 1981 to 2010, using information from 1981-2010 Normals for Wyoming Station (Reference 2.4-11). Naughton Power Plant monthly values are also presented for the period of 2019 through 2021 and for the 1-year period from April 9, 2022–April 22, 2023. There is similarity among the sites with some expected differences because of the distance between locations. The Kemmerer Unit 1 and Naughton Power Plant data for the 1-year period are similar.

The monthly mean, maximum, and minimum temperatures measured at the site follow trends at the Kemmerer and Big Piney Airports, though magnitudes in some months differ significantly from the long-term averages. For the period of April 9, 2022 to April 8, 2023, the winter months tended to be colder than the long-term averages and the summer months warmer than the long-term averages. March 2023 mean temperature at the site is approximately 10 degrees Fahrenheit (5.5 degrees Celsius) colder than long-term averages at surrounding


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locations with the monthly average maximum temperature 15–20 degrees Fahrenheit (8.3–11 degrees Celsius) colder as well. The late summer 2023 (July and August) mean temperatures are 5–8 degrees Fahrenheit (2.7–4.4 degrees Celsius) warmer than the long-term averages at surrounding airport sites but align well with the Naughton Power Plant mean temperatures. Overall, mean monthly minimum temperatures at the site are warmer than the long-term averages at surrounding locations; the annual average minimum temperature is 4 to 6 degrees Fahrenheit (2.2–3.3 degrees Celsius) warmer than the Kemmerer and Big Piney Airports.

Monthly average precipitation at the site ranged from 0.00 in (0.00 millimeters) in February to 0.72 inches (18.3 millimeters) in July. Comparing the site monthly averages to the averages at the surrounding locations indicates that April 2022–April 2023 was atypical in that there is greater than normal precipitation in late fall and winter and less than normal precipitation in late spring and summer.

Wyoming is susceptible to droughts, which are occasionally severe. From 1999 to 2008, large portions of Wyoming experienced drought conditions. Wyoming then experienced several years of above-average precipitation until 2012, which was the driest year since historical records began in 1895. By October 2012, almost 90 percent of Wyoming was in “severe” drought (the U.S. Drought Monitor’s third-highest category of drought severity). The intensity of future droughts is projected to increase, even if precipitation amounts increase. Increases in evaporation rates due to rising temperatures may increase the rate of soil moisture loss during dry spells. Future summer droughts, a natural part of Wyoming’s climate, are likely to become more intense. This in turn increases the risk of wildfires, which are projected to become more frequent and severe (Reference 2.4-2).

## 2.4.2 Local Meteorology and Topography

### 2.4.2.1 Normal and Extreme Values of Meteorological Parameters

#### 2.4.2.1.1 Data Sources

In addition to the data sources outlined in Section 2.4.1.1, measurements from the Naughton Power Plant meteorological tower and the Kemmerer Unit 1 meteorological tower were used to characterize local climatological and dispersion conditions.

As part of the preoperational meteorological monitoring program, an evaluation of the representativeness of the Naughton Power Plant meteorological data was performed to provide a basis for the use of the three-year data set from the Naughton Power Plant meteorological tower.

**Naughton Power Plant Meteorological Tower**

The Naughton Power Plant has a 164-foot (50-meter) meteorological tower that collects wind speed, wind direction, and temperature at 33 feet (10 meters) and 164 feet (50 meters). Data from the Naughton Power Plant meteorological tower, used for onsite local meteorology and dispersion analyses, includes three years of data (2019, 2020, and 2021).


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**Kemmerer Unit 1 Meteorological Data Validation Tower**

To confirm the representativeness of the Naughton Power Plant meteorological tower data, a meteorological tower was erected at the site. The site meteorological tower is 197-feet (60-meters) high and collects wind speed, wind direction, and temperature at 33 feet (10 meters) and 197 feet (60 meters). Data collection at the site meteorological tower began in April 2022.

To provide a comparison, in addition to the requisite three years of data collected at the Naughton Power Plant, one year of data collected at both the Naughton Power Plant and Kemmerer Unit 1 were analyzed (April 2022–April 2023). The data provided below includes a summary of the requisite three years of data at the Naughton Power Plant along with the comparative year of data validation.

2.4.2.1.2 Average Wind Direction and Wind Speed Conditions

**Average Wind Direction**

***Naughton Power Plant Annual and Seasonal Wind Rose Plots***

*Annual*

Figure 2.4-8 and Figure 2.4-9 present 3-year composite wind rose plots of the Naughton Power Plant data for 2019, 2020, and 2021 using the wind speed and direction at the 33 foot (10-meter) level and the 164-foot (50-meter) level, respectively.

The 2019 annual prevailing wind direction at the Naughton Power Plant at the 33-foot (10-meter) level was from the west-northwest approximately 17 percent of the time with a secondary peak from the north approximately 15 percent of the time. At the 164-foot (50-meter) level, the prevailing wind direction was also from the west-northwest approximately 19 percent of the time.

The 2020 annual prevailing wind direction at the Naughton Power Plant at the 33-foot (10-meter) level was from the west-northwest approximately 20 percent of the time with a secondary peak from the north approximately 17 percent of the time. At the 164-foot (50-meter) level, the prevailing wind direction was also from the west-northwest approximately 22 percent of the time.

The 2021 annual prevailing wind direction at the Naughton Power Plant at the 33-foot (10-meter) level was from the west-northwest approximately 17 percent of the time with a secondary peak from the north approximately 16 percent of the time. At the 164-foot (50-meter) level, the prevailing wind direction was also from the west-northwest approximately 19 percent of the time.

Figure 2.4-10 through Figure 2.4-33 present monthly composite wind rose plots for the April 2022–April 2023 period at the Naughton Power Plant at the 33-foot (10-meter) level and the 164-foot (50-meter) level, respectively.

Figure 2.4-34 and Figure 2.4-35 present the annual wind rose plots of the April 2022–April 2023 data for the Naughton Power Plant at the 33-foot (10-meter) and the 164-foot (50-meter) level, respectively.


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The April 2022–April 2023 annual prevailing wind direction at the Naughton Power Plant at the 33-foot (10-meter) level was from the west-northwest approximately 18 percent of the time with a secondary peak from the north approximately 17 percent of the time. At the 164-foot (50-meter) level, the prevailing wind direction was also from the west-northwest approximately 20 percent of the time.

*Seasonal*

Figure 2.4-36 through Figure 2.4-43 present the Naughton Power Plant seasonal 3-year composite wind rose plots for the 2019–2021 period at the 33-foot (10-meter) level and the 164-foot (50-meter) level.

Figure 2.4-44 through Figure 2.4-51 present seasonal 1-year composite wind rose plots for the April 2022–April 2023 period for Naughton Power Plant at the 33-foot (10-meter) level and the 164-foot (50-meter) level.

***Kemmerer Unit 1 Annual and Seasonal Wind Rose Plots***

*Annual*

Figure 2.4-52 and Figure 2.4-53 present annual wind rose plots of the April 2022–April 2023 meteorological data for the site at the 33-foot (10-meter) and the 197-foot (60-meter) elevations.

The April 2022–April 2023 annual prevailing wind direction at the site at the 33-foot (10-meter) level is from the north-northwest, approximately 13 percent of the time. Winds from the southwest through west sectors occur approximately 23 percent of the time. Conversely, winds from the northeast through east sectors occur approximately 7 percent of the time.

The annual prevailing wind direction at Kemmerer Unit 1 at the 197-foot (60-meter) level is from the north approximately 12 percent of the time. Winds from the southwest through west sectors occur approximately 24 percent of the time. Conversely, winds from the northeast through east sectors occur approximately 8 percent of the time.

There are no observations of calm winds at the lower level or at the upper level. At the 33-foot (10-meter) level, winds occur most infrequently from the east-southeast and at the 197-foot (60-meter) level, winds occur most infrequently from the southeast.

Annually at the site, the strongest winds (greater than 8.8 meters per second or 19.7 miles per hour) are also typically from the southwest through northwest directions while the weakest winds are typically from the north-northwest and north at 33 feet (10 meters) and from the north, north-northeast, and south at 197 feet (60 meters).

*Seasonal*

Figure 2.4-54 through Figure 2.4-61 present seasonal one-year composite wind rose plots for the April 2022–April 2023 period for the site at the 33-foot (10-meter) level and the 197-foot (60-meter) level.


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The monthly wind rose plots for the site indicate seasonal variance in wind direction but retain an overall west-northwest dominance. During the winter months (December through February), the prevailing wind direction at the 33-foot (10-meter) level was generally from the north-northwest (approximately 12 percent), though December showed a greater dominance in the west-northwest direction approximately 14 percent of the time. The same was true of December and January at the 197-foot (60-meter) level, approximately 15 percent and 13 percent of the time, respectively. February winds at the 197-foot (60-meter) level were more variable with west, north-northeast, and south directions occurring approximately 9 percent, 8 percent, and 7 percent of the time, respectively. During the spring months (March through May), April and May observed a west-northwest dominance at both levels (approximately 12 percent to 16 percent of the time), though March observed higher variability as south to southwest and north wind directions each occurred 7 percent to 11 percent of the time.

During the summer months (June through August), the prevailing wind direction at the 33-foot (10-meter) level was from the north-northwest, approximately 13 percent to 17 percent of the time, and at the 197-foot (60-meter) level was from the north, approximately 16 percent to 20 percent of the time. During the autumn months (September through November), the prevailing wind direction at the 33-foot (10-meter) level was from the north-northwest (19 percent), northwest (13 percent), and west-northwest (14 percent) for September, October, and November, respectively. At the 197-foot (60-meter) level, the prevailing wind directions varied more month-to-month and observed a greater dominance of northerly winds. September, October, and November each observed most frequent winds from the north (16 percent), northwest (12 percent), and west-northwest (16 percent), respectively, with November also showing a peak in north winds approximately 12 percent of the time, respectively.

### *Big Piney Marbleton*

*Wyoming - Annual Wind Rose (2006-2015 Average)* (Reference 2.4-7) presents a multi-year average annual wind rose plot for the Big Piney Marbleton Airport for the 10-year period from 2006 to 2015.

At the BPI Airport, the annual prevailing wind direction for the 2006 to 2015 10-year period is from the north-northwest approximately 12 percent of the time.

The Naughton Power Plant meteorological tower is in close proximity to the Kemmerer Unit 1 meteorological tower and is expected to be representative of the site. The Big Piney Airport is located approximately 80 miles (130 kilometers) to the north-northeast of the site at close to the same elevation according to Annual Summary with Comparative Data, Big Piney Marbleton Airport, Wyoming (BPI) (Reference 2.4-17). The Wyoming Range is to the west of this region and is therefore surrounded by more mountainous terrain. Large-scale winds are expected to be similar, however regional small-scale winds at the lower heights where these observations are taken affect annual wind speed and direction trends. Using the 10-year average aids in mitigating outliers due to the small-scale highly variable mountain weather.


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## Wind Speed Conditions

Table 2.4-17 presents monthly and annual average wind speeds at Naughton Power Plant, Kemmerer Unit 1, and the Big Piney Airport. The average wind speeds along with prevalent wind speed classes are presented below.

### *Naughton Power Plant Wind Speed And Prevalent Wind Speed Class*

*Wind Speed*

*Years 2019, 2020, and 2021*

Average wind speeds for the 3-year period of 2019 through 2021 are presented in Table 2.4-17 for the Naughton Power Plant. The 3-year average wind speed at Naughton Power Plant was 11.2 miles per hour (5 meters per second) at the 10-meter level and 12.5 miles per hour (5.6 meters per second) at the 164-foot (50-meter) level.

At the Naughton Power Plant for the 3-year period (2019–2021), the maximum hourly wind speed measured at the 33-foot (10-meter) level was 44.3 miles per hour (19.8 meters per second); the maximum hourly wind speed measured at the 164-foot (50-meter) level was 50.1 miles per hour (22.4 meters per second).

Annually at the Naughton Power Plant, the strongest winds (greater than 19.7 miles per hour or 8.8 meters per second) are typically from the southwest through northwest directions while the weakest winds (less than 2.1 meters per second or 4.7 miles per hour) are typically from the north and north-northeast directions.

*April 2022 through April 2023*

At the Naughton Power Plant for the 1-year period (April 2022–April 2023), the annual average wind speed was 11.1 miles per hour (5.0 meters per second) at the 10-meter level and 12.4 miles per hour (5.5 meters per second) at the 164-foot (50-meter) level.

At the Naughton Power Plant for the 1-year period (April 2022–April 2023), the maximum hourly wind speed measured at the 33-foot (10-meter) level was 37.7 miles per hour (16.8 meters per second); the maximum hourly wind speed measured at the 164-foot (50-meter) level was 43.5 miles per hour (19.4 meters per second).

*Prevalent Wind Speed Class*

*Years 2019, 2020, and 2021*

At the Naughton Power Plant for 2019, the most prevalent wind speed class for the 33-foot (10-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 28 percent of the time. The most prevalent wind speed class for the 164-foot (50-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 22 percent of the time.


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At the Naughton Power Plant for 2020, the most prevalent wind speed class for the 33-foot (10-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 29 percent of the time. The most prevalent wind speed class for the 164-foot (50-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 22 percent of the time.

At the Naughton Power Plant for 2021, the most prevalent wind speed class for the 33-foot (10-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 29 percent of the time. The most prevalent wind speed class for the 164-foot (50-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 23 percent of the time.

*April 2022 through April 2023*

At Naughton Power Plant for the April 2022–April 2023 period, the most prevalent wind speed class for the 33-foot (10-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 27 percent of the time. The most prevalent wind speed class for the 164-foot (50-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 22 percent of the time.

***Kemmerer Unit 1 Wind Speed And Prevalent Wind Speed Class***

*Wind Speed*

The annual average wind speed at the 33-foot (10-meter) level at site for the 1-year period of April 9, 2022–April 8, 2023 was 8.7 miles per hour (3.9 meters per second). At the 197-foot (60-meter) level, the annual average wind speed was 11.1 miles per hour (5.0 meters per second).

At Kemmerer Unit 1 for April 2022–April 2023, the maximum hourly wind speed measured at the 33-foot (10-meter) level was 38.2 miles per hour (17.1 meters per second); the maximum hourly wind speed measured at the 197-foot (60-meter) level was 45.1 miles per hour (20.2 meters per second).

*Prevalent Wind Speed Class*

Annually at the site meteorological data tower, the most prevalent wind speed class for the 33-foot (10-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 19 percent of the time. The most prevalent wind speed class for the 197-foot (60-meter) level is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 21 percent of the time.

On a seasonal basis at the site, the most prevalent wind speed class for the 33-foot (10-meter) level during the winter (December through February) and spring (March through May) months is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 17 percent of the time during the winter and 16 percent of the time during the spring. The 4.7 to


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6.7 miles per hour (2.1 to 3.0 meters per second) class is most prevalent during the summer (June through August) and autumn (September through November) months and occurs approximately 21 percent during the summer and 19 percent during the autumn.

At the 197-foot (60-meter) level, the most prevalent wind speed class is the 6.9 to 11.2 miles per hour (3.1 to 5.0 meters per second) class, which occurs approximately 15 percent during the winter months (December through February), 15 percent during the spring months (March through May), 25 percent during the summer months (June through August), and 24 percent during the autumn months (September through November).

**Comparative Local and Regional Wind Speed Summary**

Seasonal variations in average wind speeds at the Big Piney Airport generally agree with Naughton Power Plant and the site, though magnitude of winds is overall lower for this station with an annual average of 6.8 miles per hour (3.0 meters per second). April through June and October through December generally observed the strongest average wind speeds.

2.4.2.1.3 Wind Direction Persistence

Table 2.4-18 and Table 2.4-19 present annual and overall wind direction persistence summaries for the 33-foot (10-meter) and 197-foot (60-meter) measurement levels at the site meteorological tower. These tables were developed using one year of onsite meteorological data (April 2022–April 2023).

Approximately 42 percent of the time, wind direction persistence events last for less than four hours at the 33-foot (10-meter) level and approximately 30 percent at the 197-foot (60-meter) level. Wind direction persistence events lasting 12 hours occur 70 times per year for the lower and 132 times per year for the upper measurement levels, respectively. Wind direction persistence events lasting greater than 24 hours occur once per year for the lower and upper measurement levels.

2.4.2.1.4 Atmospheric Stability

Atmospheric stability is determined by the delta temperature method as defined in RG 1.23, “Meteorological Monitoring Programs For Nuclear Power Plants,” Revision 1. This methodology classifies atmospheric stability based on the temperature change with height (degrees Celsius per 100 meters). At the site meteorological tower, atmospheric stability is classified according to the difference between the temperature measurements at the 197-foot (60-meter) and 33-foot (10-meter) levels.

Table 2.4-20 through Table 2.4-67 present annual Joint Frequency Distributions of wind speed and direction as a function of atmospheric stability. Data are derived from the meteorological towers at the Naughton Power Plant for the years of 2019, 2020, 2021, and


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April 2022–April 2023 and at the site for the period of April 2022–April 2023. Data from both the lower and upper elevations at each tower are presented in independent annual Joint Frequency Distributions for each year of available data:

- Table 2.4-20 through Table 2.4-35 provide 3-year composite data for Naughton Power Plant for 2019–2021.
- Table 2.4-36 through Table 2.4-51 provide annual composite data for Naughton Power Plant for April 2022–April 2023.
- Table 2.4-52 through Table 2.4-67 provide annual composite data for Kemmerer Unit 1, for the period from April 2022–April 2023.

All of these tables were developed following the guidance in RG 1.23. Additional wind speed classes are included to provide greater coverage of the lower wind speeds that are most important for atmospheric dispersion.

During the one-year period from April 9, 2022–April 8, 2023, unstable conditions at the site occurred about 19 percent, neutral conditions 24 percent, and stable conditions 57 percent of the time. Table 2.4-68 shows the percent stability class (A-G) for both the site and Naughton Power Plant meteorological sites.

Table 2.4-69 through Table 2.4-72 present 3-year (2019–2021) atmospheric stability persistence summaries at the Naughton Power Plant for the 33-foot (10-meter) and 164-foot (50-meter) measurement levels and annual (April 2022–April 2023) atmospheric stability persistence summaries at the site for the 33-foot (10-meter) and 197-foot (60-meter) measurement levels.

Table 2.4-73 presents temperature inversion frequency statistics for the site for April 2022–April 2023. Stability persistence events lasting for less than 12 hours occurred 76 percent of the time and stability persistence events lasting 16 hours or less occurred 92 percent of the time. Events lasting for greater than 24 hours occurred 6 times with the longest of these inversions lasting 82 hours in early January 2023.

#### 2.4.2.1.5 Temperature and Humidity

The temperature and precipitation normals from Summary of Monthly Normals 1991–2020, the Evanston-Burns Field, Wyoming (Reference 2.4-3) first-order weather station that best represents the site are summarized in Table 2.4-4.

Normal daily mean temperatures are based on the mean daily maximum temperatures and the mean daily minimum temperatures across each month. The temperatures ranged from 13.6 to 82.0 degrees Fahrenheit (-10.2 to 27.8 degrees Celsius). The annual daily normal temperatures range from 21.7 to 67.5 degrees Fahrenheit (-5.7 to 19.7 degrees Celsius).

Table 2.4-5 and Table 2.4-6 show maximum and minimum hourly temperature data for each month of April 2022–April 2023, obtained from the site meteorological tower at the 10-meter (33-foot) level.


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Extreme weather and wind caused wiring issues at the meteorological tower in June and July 2022 and power issues in September 2022. As a result, some hours of the relative humidity, dew point, and wet bulb temperature data are missing. There were a few months when the monthly averages were based on less than 90 percent data recovery.

The monthly mean temperature at the site ranges from 12.3 degrees Fahrenheit (11.0 degrees Celsius) in February to 69.0 degrees Fahrenheit (20.6 degrees Celsius) in July. The monthly mean extreme maximum temperature at Kemmerer Unit 1 is 82.5 degrees Fahrenheit (28.1 degrees Celsius) in July and the monthly mean extreme minimum temperature was 3.6 degrees Fahrenheit (-15.8 degrees Celsius) in February. The monthly mean daily maximum temperature at the site was 48.6 degrees Fahrenheit (9.2 degrees Celsius) and the monthly mean daily minimum temperature was 26.6 degrees Fahrenheit (-3.0 degrees Celsius). The overall average temperature for the site was 37.5 degrees Fahrenheit (3.1 degrees Celsius). This agreed very well with the overall average temperature at the Naughton site which was 38.1 degrees Fahrenheit (3.4 degrees Celsius). The monthly average temperature at both sites agreed within approximately 2 degrees Fahrenheit (approximately 0.6 degrees Celsius) throughout the year. The maximum hourly temperature at the site was 92.0 degrees Fahrenheit (33.3 degrees Celsius) in September and the minimum hourly temperature was -24.9 degrees Fahrenheit (-31.6 degrees Celsius) in January. The frequency of occurrence of hourly temperature values falling below the freezing point (32 degrees Fahrenheit or 0 degrees Celsius) is approximately 46 percent.

There are NWS stations located in Western Wyoming at Kemmerer Airport and Big Piney Marbleton Airport. Monthly mean, maximum, and minimum dry bulb temperature values are from the 30-year period from 1981 to 2010 and based on Reference 2.4-11. Naughton Power Plant monthly values are also presented for the period of 2019 through 2021 and for the one-year period of April 9, 2022 through April 22, 2023. The data between the sites is similar with some expected differences because of the distance between locations. The Kemmerer Unit 1 and Naughton Power Plant data for the one-year period are similar.

The Kemmerer NWS station is approximately 10 miles (16 kilometers) to the north of the site at an elevation of 6,926 feet (2,111 meters), about 150 feet (46 meters) higher than the site. As previously mentioned, the Big Piney NWS station is approximately 80 miles (129 km) to the north-northeast of the site at an elevation of 6,820 feet (2,079 meters), a little over 50 feet (15 meters) higher than the site. Due to the high variability in elevation across the region, values of temperature, wind speed, and precipitation also vary greatly. Similar to Kemmerer Unit 1, the Kemmerer and Big Piney NWS stations are located just east of the Wyoming Range and therefore are more influenced by volatile mountain weather. Comparison of values between these stations and Kemmerer Unit 1 is expected to provide support in the overall annual trend observed at Kemmerer Unit 1 though magnitudes for the one year of data could deviate significantly from the long-term averages of the NWS stations, and even the three-year average at the Naughton Power Plant.

The monthly mean, maximum, and minimum temperatures measured at the site follow trends at the Kemmerer and Big Piney Airports, though magnitudes in some months differ significantly from the long-term averages. For the period of April 9, 2022 through April 8, 2023 the winter months tended to be colder than the long-term averages and the summer months warmer than


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the long-term averages. The March 2023 mean temperature at the site is approximately 10 degrees Fahrenheit (5.5 degrees Celsius) colder than long term averages at surrounding sites with the monthly average maximum temperature 15 degrees Fahrenheit to 20 degrees Fahrenheit (8.3 degrees Celsius to 11 degrees Celsius) colder as well. The late summer 2023 (July and August) mean temperatures are 5 degrees Fahrenheit to 8 degrees Fahrenheit (2.8 degrees Celsius to 4.4 degrees Celsius) warmer than the long-term averages at surrounding airport sites but align well with the Naughton Power Plant mean temperatures. Overall, mean monthly minimum temperatures at the site are warmer than the long-term averages at surrounding sites resulting in an annual average minimum temperature that is 4 degrees Fahrenheit to 6 degrees Fahrenheit (2.2 degrees Celsius to 3.3 degrees Celsius) warmer than the Kemmerer and Big Piney Airports.

Table 2.4-8 presents the monthly design wet bulb temperature and the mean coincident dry bulb temperature for Evanston, Wyoming, which is in the vicinity of the site. These wet bulb temperature values correspond to 0.4 percent, 2.0 percent, 5.0 percent, and 10.0 percent cumulative frequency of occurrence for the indicated month. The data were determined from Reference 2.4-6. Data for Evanston, Wyoming are from the period 1999 to 2019.

#### 2.4.2.1.6 Precipitation

Normal annual rainfall totals for nearby observing stations listed in Table 2.4-9 (Naughton, Big Piney, and Kemmerer weather monitoring stations) vary from 6.48 to 10.35 inches (16.46 to 26.29 centimeters) annually. The Local Climatological Data summary of normal rainfall totals indicates that the seasonal maximum occurs during the month of May. Normal monthly precipitation totals ranged from 0.38 to 1.75 inches (0.97 to 4.45 centimeters) for the Evanston-Burns Field first-order station.

The monthly and annual percent frequency of precipitation are presented in Table 2.4-16 for the Naughton Power Plant and Kemmerer Unit 1.

**Naughton Power Plant**

***Monthly and Annual Percent Frequency of Precipitation***

The monthly percent frequency of occurrence of precipitation at the Naughton Power Plant for the 3-year period from 2019–2021 ranged from 1.30 percent in July to 4.93 percent in May. Monthly percent frequency of occurrence of precipitation at the Naughton Power Plant for April 2022–April 2023 ranged from 0.67 percent in July to 5.56 percent in April.

***Annual Precipitation Wind Roses***

Figure 2.4-2 through Figure 2.4-5 present 3-year annual average precipitation wind roses at Naughton Power Plant from 2019–2021 and the period of April 2022–April 2023, respectively, for the 33-foot (10-meter) and 164-foot (50-meter) elevations. While the precipitation wind roses portray distributions of wind direction for only the hours in which precipitation was recorded, Figure 2.4-2 through Figure 2.4-5 show that the most frequent wind direction during precipitation was from the west to west-northwest.


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**Kemmerer Unit 1**

### *Monthly and Annual Percent Frequency of Precipitation*

Monthly percent frequency of occurrence of precipitation for April 2022–April 2023 ranged from 0 percent in February to 4.7 percent in March. Additionally, from Table 2.4-9, monthly average precipitation at the site ranged from 0.00 inches (0.00 centimeters) in February to 0.72 inches (18.3 centimeters) in July for the time period of April 9, 2022 to April 9, 2023.

### *Annual Precipitation Wind Roses*

Figure 2.4-6 and Figure 2.4-7 present annual precipitation wind roses at Kemmerer Unit 1 for the 33-foot (10-meter) and 197-foot (60-meter) elevations from April 2022–April 2023. The most frequent wind direction during precipitation was from the west to west-northwest at the 33-foot (10-meter) level. The most frequent wind direction was from a west-southwest direction at the 197-foot (60-meter) level.

### Comparative Summary of Monthly and Annual Percent Frequency and Annual Precipitation Wind Roses for the Naughton Power Plant and Kemmerer Unit 1

Although monthly values for the percent frequency of precipitation for Naughton Power Plant and Kemmerer Unit 1 vary between periods and sites, the annual values indicate that precipitation occurs only about 2 percent of the time in this region.

The most frequent wind direction during precipitation for both Naughton Power Plant and Kemmerer Unit 1 was from the west to west-northwest at the 33-foot (10-meter) level but was from the west-southwest at the 164-foot (50-meter) or 197-foot (60-meter) level. Both sites also showed a secondary peak from the north at the 33-foot (10-meter) level while the Kemmerer Unit 1 197-foot (60-meter) level showed a secondary peak from the west-northwest to north-northwest directions during the comparative period.

### Comparative Local and Regional Precipitation Summary

The monthly and annual precipitation summary from Kemmerer Unit 1 (April 2022–April 2023) is presented in Table 2.4-9, with precipitation statistics from the Naughton Power Plant (2019, 2020, 2021, 2019–2021, and April 2022–April 2023) and from NWS sites near Kemmerer Unit 1.

Comparing Kemmerer Unit 1 monthly averages to the averages at the surrounding region indicates that overall, April 2022–April 2023 was atypical in that there is greater than normal precipitation in late fall and winter and less than normal precipitation in late spring and summer. The magnitude of winter precipitation at Kemmerer Unit 1 was not as great, leading to an overall drier annual average than the surrounding sites.

Comparing Kemmerer Unit 1 and Naughton Power Plant precipitation values for April 2022–April 2023 suggests that the sites observed the same overall trend in rainfall with Kemmerer Unit 1 generally observing 0.1 inches (0.3 centimeters) to 0.3 inches (0.8 centimeters)


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less rainfall per month than Naughton Power Plant. Months which differed by more than 0.3 inches (0.8 centimeters) include May, 0.49 inches (1.2 centimeters) and July, 0.31 inches (0.79 centimeters).

Overall, the data show that precipitation is highly variable across the region, with influences from isolated showers that may lead to a large amount of precipitation recorded at one location while no precipitation at another. The three-year period at Naughton Power Plant provides a base for an expected seasonal trend near the site; spring and fall observe higher precipitation amounts than winter, while summer brings more convective showers that generally result in a higher monthly average as well, though not in some years (July 2019, August 2020, and June 2021).

Table 2.4-13 presents mean monthly and annual snowfall totals at stations near Kemmerer Unit 1. The Kemmerer Airport is the nearest station with snowfall data available. This shows that a majority of snowfall in the region occurs during the period of November through January with amounts lessening in February through April. Little to no snowfall is expected from June through September.

#### 2.4.2.1.7 Comparative Study

A study was completed which compared the meteorological data collected at the Naughton Power Plant and Kemmerer Unit 1 meteorological towers for the period from April 9, 2022–April 8, 2023. Consistent with the guidance of RG 1.23, this range was selected because it represents the first complete year of data collected at the Kemmerer Unit 1 meteorological tower. The purpose of the comparison study was to evaluate whether the Naughton Power Plant data was representative of the conditions experienced at Kemmerer Unit 1. An evaluation of the 2019, 2020, and 2021 Naughton Power Plant meteorological data showed that the Naughton Power Plant data was representative of the region, including Kemmerer Unit 1.

Each meteorological variable that was collected at both the Naughton Power Plant and Kemmerer Unit 1 meteorological tower was compared using scatter plot diagrams to calculate the correlation coefficient. This study considered a correlation coefficient of 0.7 or higher to mean the variables are representative of each other.

The study showed that the correlation coefficients were greater than 0.7 for variables examined including temperature at multiple heights, wind speed at multiple heights, delta temperature at multiple heights, wind direction at multiple heights, relative humidity, and solar radiation. The lowest correlation coefficient was 0.7006 for the 33-foot (10-meter) wind direction. This is likely the result of low wind speeds which results in wind direction variability. The highest correlation coefficient was 0.9966 for 164-foot (50-meter) and 197-foot (60-meter) temperatures obtained from the meteorological towers. Since the linear trends are the same for each of the variables examined, it is concluded that the data collected at the two meteorological towers are representative of each other. This means that the Naughton Power Plant data can be used to supplement or replace the Kemmerer Unit 1 data. This also means that the Naughton data provides a good representation for both locations and can be used to accurately represent the meteorological conditions at Kemmerer Unit 1.


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## 2.4.3 Air Quality

The U.S. EPA set National Ambient Air Quality Standards (NAAQS) (Reference 2.4-15) for six principal pollutants, which are called “criteria” pollutants. These pollutants are: CO, Pb, $NO_2$, $O_3$, PM, and $SO_2$. Units of measure for the standards are parts per million by volume, milligrams per cubic meter of air, and micrograms per cubic meter of air. Areas are either “in attainment” of the air quality standards or “in nonattainment”. Attainment means that the air quality is better than the standard. Table 2.4-77 shows the NAAQS.

### 2.4.3.1 Regional Air Quality

The portion of Lincoln County where Kemmerer Unit 1 is located is designated as in attainment with respect to the NAAQS *Air Data Air Quality Monitors* (Reference 2.4-16). The nearest nonattainment area is the UGRB Ozone Nonattainment Designation area consisting of Sublette County and portions of Lincoln and Sweetwater Counties, according to Nonattainment Map for Wyoming (Reference 2.4-14). Currently, for the 2008 Ozone NAAQS, the EPA designated the UGRB as nonattainment. However, monitored ozone in the UGRB was in attainment for the 2008 Ozone NAAQS by the attainment date of July 2015.

There are no Class 1 Federal lands within 85 miles (140 kilometers) of Kemmerer Unit 1 as of 2023. The nearest Class 1 Federal area is Grand Teton National Park, which is approximately 128 miles away. Figure 2.4-62 shows both the UGRB ozone nonattainment area and the nearest Class 1 area.

### 2.4.3.2 Projected Air Quality Conditions

The Kemmerer Unit 1 radiological systems are not sources of criteria pollutants. Supporting equipment and other non-radiological emission generating sources (e.g. diesel generators, cooling towers, storage tanks) or activities are not significant sources of criteria pollutant emissions.

Supporting equipment is operated on an intermittent basis. Therefore, these emission sources do not impact ambient air quality levels in the vicinity of the site, nor are they a significant factor in the design and operation of Kemmerer Unit 1.

## 2.4.4 Atmospheric Dispersion

### 2.4.4.1 Short-Term Diffusion Estimates

The doses for various time periods immediately following the onset of a postulated ground-level release at the Exclusion Area Boundary (EAB) and for the duration of exposure at the Low Population Zone (LPZ) are considered. The relative concentrations (X/Qs) are estimated for various time periods ranging from 2 hours to 30 days.

Site-specific meteorological data covering the 3-year period of record (2019–2021) is used to quantitatively evaluate a hypothetical accident at the site.


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#### 2.4.4.1.1 ARCON

ARCON is used to estimate X/Q values at the EAB and LPZ. RG 1.194, Revision 0, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” endorses the use of the ARCON96 computer code for calculating accident-related onsite atmospheric dispersion values that are direct inputs to habitability dose assessments.

The limiting distance from the nearest building edge is applied over all 16 wind direction sectors.

The ARCON model uses hourly meteorological data and a straight-line Gaussian model to estimate dispersion in the vicinity of wake producing structures and to calculate atmospheric dispersion factors or X/Qs at air intakes that would be exceeded no more than 0.5 percent of the time (99.5th percentile). The X/Q values are determined by averaging periods ranging from one hour to thirty days depending on the application of the X/Qs.

ARCON evaluates ground level, vent, and elevated releases. RG 1.249, Revision 0, “Use of ARCON Methodology For Calculation Of Accident-Related Offsite Atmospheric Dispersion Factors,” discusses the use of ARCON for calculation of accident-related offsite atmospheric dispersion factors. This guidance provides procedures for using ARCON code to estimate X/Qs at the EAB and the outer boundary of the LPZ out to distances of 3,937 feet (1,200 meters) from the nearest edge of a building within the Nuclear Island (NI). Due to the limited applicable distance assumed by RG 1.249, ground-level releases are assumed.

Table 2.4-78 presents the ARCON 99.5 percentile X/Q values at the height of an average male for the onsite control room evaluation. Table 2.4-79 presents the ARCON 99.5 percentile X/Q values for each of the 16 downwind sectors for the appropriate time period(s) at the EAB distance of 400 meters from 1-720 hour averaging times.

### 2.4.4.2 Long-Term Diffusion Estimates

XOQDOQ is used to calculate long-term X/Q and relative deposition (D/Q) values for specified locations and ranges within 50 miles of the site.

XOQDOQ distances from the RXB to the nearest receptors in each directional sector are provided in Table 2.4-80.

The maximum annual average X/Q value with no decay is 1.896E-04 sec/$m^3$ from the Energy Island (EI) release as modeled as coming from the cold salt storage tank and occurs at the EAB in the south sector as provided in the locations of interest in Table 2.4-81. The maximum annual average X/Q value with no decay is 5.128E-05 sec/$m^3$ from the NI release as modeled as a 100 meter circle centered on the reactor and occurs at the EAB in the south sector as provided in Table 2.4-81.

Results of the XOQDOQ modeling include 50-mile X/Q and D/Q values from the NI, displayed in Table 2.4-82 through Table 2.4-85 and 50-mile X/Q and D/Q values from the EI, displayed in Table 2.4-86 through Table 2.4-89.


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### 2.4.5 Meteorological Monitoring

Data from the meteorological monitoring stations is used to:

- Describe local and regional atmospheric transport and characteristics
- Calculate the dispersion estimates for both postulated accidental and expected routine airborne releases of effluents
- Compare with offsite sources to determine the appropriateness of climatological data used for design considerations
- Evaluate environmental risk from the radiological consequences of a spectrum of postulated accidents
- Provide a meteorological database for evaluation of the effects from plant construction and operation, including radiological and nonradiological impacts and real-time predictions of atmospheric effluent transport and diffusion
- Develop emergency response plans, including provision for real-time meteorological data and plume trajectory dispersion modeling capabilities for dose and exposure predictions.

#### 2.4.5.1 Preoperational Meteorological Measurement Program

The preoperational meteorological measurement program for Kemmerer Unit 1 is designed and maintained in accordance with the guidance provided in RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants,” Revision 1.

The 2019, 2020, and 2021 period of data taken from the Naughton Power Plant meteorological tower is determined to be the best available (validated data with least substitution), representative (tower and sensor siting in accordance with RG 1.23), and complete (with annualized composite data recovery of 90 percent), without being older than 10 years.

##### 2.4.5.1.1 Tower Location

**Naughton Power Plant Tower Location**

The meteorological tower for the Naughton Power Plant is located at 41.7576 degrees north and 110.578 degrees west, approximately 3.5 miles (5.6 kilometers) north-northwest of Kemmerer Unit 1. This tower serves as the data collection system and source of onsite meteorological data for the Naughton Power Plant. The elevation at the base of the tower is 6,902 feet (2,104 meters) above mean sea level. This elevation is about 150 feet (40 meters) higher than the elevation of the site.

The Naughton Power Plant meteorological tower is located on level, open terrain at a distance of at least 10 times the height of any nearby obstruction that exceeds one-half the height of wind measurement. The open lattice tower is located far enough away from the site and topographical features to avoid airflow modifications.


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### *Tower Location Relative to Potential Sources of Heat and Moisture*

There are no sources of moisture near the Naughton Power Plant tower that influence meteorological data monitoring.

### *Instrumentation*

The Naughton Power Plant meteorological tower instrumentation consists of wind speed, wind direction, aspirated temperature sensors, and vertical wind speed located at 164 feet (50 meters) and 33 feet (10 meters) above ground level. Additional temperature sensors are located at a 7-foot (2-meter) level along with a solar radiation sensor, a net radiation sensor, and a relative humidity sensor. A tipping bucket rain gauge is located at ground level near the tower.

## Kemmerer Unit 1 Tower Location

Figure 2.4-63 shows the location of the Kemmerer Unit 1 meteorological tower as well as the topography of the site. The 60-meter mast meteorological tower meets the guidance of RG 1.23, Revision 1, “Meteorological Monitoring Programs for Nuclear Power Plants.” Table 2.4-2 shows the terrain heights in each direction out to 5 miles (8 kilometers).

The Kemmerer Unit 1 meteorological tower is located on level, open terrain at a distance at least 10 times the height of any nearby obstruction that exceeds one-half the height of the wind measurement. The tower is located far enough away from the site and topographical features to avoid airflow modifications. The terrain height of the meteorological tower is approximately 4 feet (1 meter) lower than the NI and EI.

### *Tower Location Relative to Potential Sources of Heat and Moisture*

The ground surface surrounding the 197-foot (60-meter) Kemmerer Unit 1 tower is a grainy, light-colored material with patches of low-cut grass or weeds around the base of the tower which is typical of ground cover in the area. There are no sources of heat near the Kemmerer Unit 1 meteorological tower that would impact meteorological data monitoring.

The temperature sensors on both towers are mounted in fan-aspirated radiation shields, which are horizontal to minimize the impact of thermal radiation and precipitation.

### *Instrumentation*

The Kemmerer Unit 1 meteorological tower instrumentation consists of wind speed, wind direction, and aspirated temperature sensors located at 197 feet (60 meters) and 33 feet (10 meters) above ground level. A relative humidity sensor is located at a height of 33 feet (10 meters). A tipping bucket rain gauge is located approximately 6 feet (2 meters) north of the meteorological tower. Solar radiation instrumentation is located on a crossarm at 8 feet (2.5 meters) above ground level and extending out approximately 4 feet (1 meter) from the tower on the east side.

The specifications of the instrumentation meet or exceed the accuracy and resolution requirements of RG 1.23. The instruments are positioned on the meteorological tower in accordance with the guidance of RG 1.23.


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Table 2.4-90 provides the current meteorological instrument accuracy and resolution and compares them with regulatory guidance provided in RG 1.23.

The data logger is a micro-computer that retrieves the meteorological data and performs 15-minute and hourly averages of the various observed parameters. The data is transmitted via modem and cellular communications to a computer where it is stored and analyzed for discrepancies. Both meteorological towers use a Campbell Scientific data logger. Even though the logger series is different between the two towers, there is no difference in operation of the data loggers used at the respective towers.

### *Instrument Maintenance and Surveillance Schedules*

The Kemmerer Unit 1 meteorological instruments are inspected and serviced at a frequency that assures at least a 90 percent data recovery rate for the parameters, including the combination of wind speed, wind direction, and delta temperature, as required by RG 1.23. Percent data recovery for the Kemmerer Unit 1 and Naughton Power Plant meteorological towers are listed in Table 2.4-91.

System calibrations encompass the entire data channel for each instrument, including the data logger and a display located at the tower. The system calibrations are performed by either a series of sequential, overlapping, or total channel steps.

### *Data Reduction and Compilation*

Hourly averaged meteorological data from Kemmerer Unit 1 are reviewed and validated. This includes a check of each parameter against expected limits and like sensors on the pole. A program is used to show that the data meets or exceeds the requirements specified by RG 1.23. The data is used to generate joint frequency distributions, wind roses, wind persistence tables, and delta temperature plots. The Joint Frequency Distributions are used to determine dispersion compliance and confirm design objectives are met.

The data from the Naughton Power Plant meteorological tower hourly averaged data are reviewed and validated. Each parameter is checked against expected limits and like sensors on the tower.

### *Nearby Obstructions to Air Flow*

Downwind distances from the meteorological tower to nearby (within 0.5 miles [0.8 kilometer]) obstructions to air flow were determined using United States Geological Survey topographical maps. There are no obstructions to the meteorological instrumentation within 0.5 miles (0.8 kilometers) (Table 2.4-2).

From the information provided in Figure 2.4-63 and Table 2.4-2, and the base of the tower being at an elevation of approximately 6,754 feet (2,059 meters), there are no significant nearby obstructions to airflow.


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**Table 2.4-1 Meteorological Observation Stations and Climate Observing Stations With Respect to Kemmerer Unit 1**

| Meteorological Observation Stations | | | |
|---|---|---|---|
| **Station Name** | **Station Type** | **Distance from Kemmerer Unit 1** | **Parameters Observed** |
| KEVW: Evanston-Uinta County Airport | Automated Surface Observing System | 40 miles; SW | Temperature, Dewpoint, Pressure (altimeter), Winds, Visibility, Ceiling, Clouds |
| KBPI: Big Piney – Miley Memorial Field Airport, Sublette, Wyoming | | 62 miles; NNE | |
| KRKS: Rock Springs – Southwest Wyoming Regional Airport, Sweetwater, Wyoming | | 81 miles; ESE | |
| KLGU: Logan – Logan-Cache Airport, Cache, Utah | | 65 miles; WNW | |
| KEMM: Kemmerer, Lincoln, Wyoming | AWOS-3PT | 5 miles; NNE | |
| KFBR: Fort Bridger, Uinta, Wyoming | | 27 miles; SSE | |
| KAFO: Afton, Lincoln, Wyoming | AWOS-3P | 68 miles; NNW | |
| K1U7: Bear Lake/Paris, Bear Lake, Idaho | AWOS-3 | 51 miles; NW | |
| **Meteorological Stations used for Extreme Weather Analysis** | | | |
| **Station Name** | **Station Type** | **Distance from Kemmerer Unit 1** | **Parameters Observed** |
| Kemmerer 2N | COOP | 8 miles; NNE | Temperature and Precipitation |
| Evanston 1E | COOP | 36 miles; SW | Temperature and Precipitation |
| **Climate Observing Stations** | | | |
| **Station Name** | **Station Type** | **Distance from Kemmerer Unit 1** | **Parameters Observed** |
| Lyman 1SW | COS | 34 miles; SSE | Temperature, Vapor Pressure, Soil Temperature, Relative Humidity, Solar Radiation, Winds, Precipitation |
| Granger 6NW | COS | 31 miles; ESE | |
| Farson 5S | COS | 61 miles; ENE | |
| LaBarge 2S | COS | 39 miles; NE | |
| Big Piney 11W | COS | 55 miles; NNE | |


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**Table 2.4-2 Detailed Topography Within 5 Miles (8 Kilometers)**

| | Kemmerer Unit 1 Site Terrain 0–5 Miles | | | | |
|---|---|---|---|---|---|
| | Meters above Plant Grade (2,060 m) | | | | |
| DIRECTION | Distances (mi) | | | | |
| (From reactor) | 1 | 2 | 3 | 4 | 5 |
| North | 19 | 49 | 123 | 123 | 149 |
| North-northeast | 31 | 87 | 117 | 191 | 217 |
| Northeast | 49 | 77 | 119 | 191 | 191 |
| East-northeast | 49 | 129 | 164 | 164 | 164 |
| East | 43 | 169 | 184 | 184 | 184 |
| East-southeast | 56 | 144 | 184 | 184 | 184 |
| Southeast | 25 | 114 | 147 | 183 | 183 |
| South-southeast | 0 | 74 | 92 | 173 | 173 |
| South | 10 | 10 | 10 | 15 | 55 |
| South-southwest | 10 | 10 | 10 | 23 | 43 |
| Southwest | 4 | 13 | 45 | 74 | 178 |
| West-southwest | 7 | 25 | 156 | 238 | 245 |
| West | 13 | 25 | 185 | 188 | 263 |
| West-northwest | 7 | 42 | 165 | 293 | 293 |
| Northwest | 19 | 31 | 63 | 219 | 297 |
| North-northwest | 19 | 19 | 43 | 56 | 83 |


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**Table 2.4-3 Kemmerer Unit 1 Site Terrain 0–50 Miles (0–80 Kilometers)**

| | Distances (mi) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | 1 | 2 | 3 | 4 | 5 | 5 to 10 | 10 to 20 | 20 to 30 | 30 to 40 | 40 to 50 |
| | Meters above Plant Grade (2,060 m) | | | | | | | | | |
| North | 19 | 49 | 123 | 123 | 149 | 197 | 649 | 754 | 860 | 1,004 |
| North-northeast | 31 | 87 | 117 | 191 | 217 | 334 | 383 | 383 | 383 | 482 |
| Northeast | 49 | 77 | 119 | 191 | 191 | 195 | 232 | 232 | 232 | 232 |
| East-northeast | 49 | 129 | 164 | 164 | 164 | 165 | 165 | 165 | 165 | 165 |
| East | 43 | 169 | 184 | 184 | 184 | 184 | 184 | 184 | 184 | 184 |
| East-southeast | 56 | 144 | 184 | 184 | 184 | 184 | 184 | 184 | 184 | 184 |
| Southeast | 25 | 114 | 147 | 183 | 183 | 183 | 183 | 183 | 183 | 522 |
| South-southeast | 0 | 74 | 92 | 173 | 173 | 173 | 173 | 220 | 360 | 825 |
| South | 10 | 10 | 10 | 15 | 55 | 139 | 209 | 209 | 414 | 813 |
| South-southwest | 10 | 10 | 10 | 23 | 43 | 172 | 216 | 400 | 408 | 408 |
| Southwest | 4 | 13 | 45 | 74 | 178 | 208 | 366 | 569 | 569 | 569 |
| West-southwest | 7 | 25 | 156 | 238 | 245 | 245 | 440 | 440 | 440 | 501 |
| West | 13 | 25 | 185 | 188 | 263 | 273 | 385 | 385 | 410 | 510 |
| West-northwest | 7 | 42 | 165 | 293 | 293 | 293 | 293 | 293 | 293 | 708 |
| Northwest | 19 | 31 | 63 | 219 | 297 | 297 | 389 | 524 | 524 | 690 |
| North-northwest | 19 | 19 | 43 | 56 | 149 | 229 | 489 | 540 | 881 | 940 |


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**Table 2.4-4 Summary of Normals (Temperature & Precipitation) at the Evanston Burns Field, WY Station (1991–2020)**

| | Mean Temperature (°F) | | | Mean Precipitation (inches) |
|---|---|---|---|---|
| Month | Daily Max | Daily Min | Mean | Mean Total |
| 01 | 29.8 | 13.6 | 21.7 | 0.41 |
| 02 | 31.5 | 14.8 | 23.1 | 0.53 |
| 03 | 40.7 | 22.0 | 31.4 | 0.45 |
| 04 | 50.2 | 28.0 | 39.1 | 0.99 |
| 05 | 60.8 | 36.6 | 48.7 | 1.75 |
| 06 | 71.8 | 44.4 | 58.1 | 1.06 |
| 07 | 82.0 | 52.9 | 67.5 | 0.55 |
| 08 | 80.1 | 51.1 | 65.6 | 0.83 |
| 09 | 69.9 | 42.8 | 56.3 | 1.01 |
| 10 | 55.7 | 32.1 | 43.9 | 0.99 |
| 11 | 40.6 | 21.6 | 31.1 | 0.56 |
| 12 | 29.7 | 13.8 | 21.8 | 0.38 |
| Summary | 53.6 | 31.1 | 42.4 | 9.51 |


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**Table 2.4-5 Kemmerer Unit 1 Maximum Hourly Temperatures (2022–2023), 10 Meter Level**

| | Maximum | |
|---|---|---|
| | °C | °F |
| **January** | 1.0 | 33.8 |
| **February** | 1.1 | 34.0 |
| **March** | 3.9 | 39.0 |
| **April** | 17.1 | 62.7 |
| **May** | 24.7 | 76.5 |
| **June** | 28.9 | 84.0 |
| **July** | 32.2 | 90.0 |
| **August** | 29.8 | 85.6 |
| **September** | 33.3 | 92.0 |
| **October** | 20.2 | 68.3 |
| **November** | 12.9 | 55.3 |
| **December** | 3.8 | 38.8 |
| **Annual** | 33.3 | 92.0 |


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**Table 2.4-6 Kemmerer Unit 1 Minimum Hourly Temperatures (2022–2023), 10 Meter Level**

| | Minimum | |
|---|---|---|
| | °C | °F |
| **January** | -31.6 | -24.9 |
| **February** | -28.2 | -18.8 |
| **March** | -21.9 | -7.4 |
| **April** | -21.6 | -6.9 |
| **May** | -6.2 | 20.9 |
| **June** | -1.8 | 28.8 |
| **July** | 7.9 | 46.3 |
| **August** | 5.4 | 41.8 |
| **September** | -1.4 | 29.4 |
| **October** | -10.3 | 13.5 |
| **November** | -19.1 | -2.3 |
| **December** | -22.8 | -9.1 |
| **Annual** | -31.6 | -24.9 |


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**Table 2.4-7 Kemmerer Unit 1 Monthly Max., Min., Avg. Temperatures (°F) and Relative Humidity (2022–2023), 10 Meter Level**

| | Wet Bulb | | | Dry Bulb | | | Dew Point | | | Relative Humidity | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Min. | Max. | Avg. | Min. | Max. | Avg. | Min. | Max. | Avg. | Min. | Max. | Avg. |
| **January** | 8.6 | 20.8 | 15.3 | 7.1 | 20.5 | 14.3 | 8.6 | 20.6 | 15.4 | - | - | - |
| **February** | 1.2 | 18.1 | 9.2 | 3.6 | 21.0 | 12.0 | -0.3 | 16.3 | 7.9 | 78.2 | 99.3 | 90.6 |
| **March** | 7.9 | 22.4 | 15.8 | 8.2 | 24.0 | 16.6 | 7.6 | 21.1 | 14.9 | 81.1 | 99.5 | 92.1 |
| **April** | 19.0 | 33.1 | 27.4 | 21.2 | 41.6 | 31.1 | 14.9 | 27.1 | 21.7 | 48.3 | 93.5 | 71.5 |
| **May** | 30.5 | 42.6 | 38.6 | 32.3 | 54.9 | 45.6 | 24.1 | 33.6 | 29.6 | 34.9 | 83.1 | 58.0 |
| **June** | 42.0 | 51.0 | 48.4 | 42.0 | 71.9 | 58.6 | 29.3 | 40.3 | 35.3 | 27.6 | 70.7 | 43.1 |
| **July** | 46.4 | 57.1 | 53.0 | 52.0 | 82.5 | 69.1 | 28.3 | 45.1 | 37.8 | 14.7 | 67.6 | 36.9 |
| **August** | 47.8 | 57.5 | 53.4 | 51.8 | 78.5 | 65.5 | 33.3 | 49.5 | 43.1 | 23.5 | 82.5 | 51.7 |
| **September** | 37.3 | 50.8 | 43.3 | 41.5 | 72.8 | 57.8 | 19.8 | 38.3 | 30.2 | 15.0 | 73.0 | 47.1 |
| **October** | 32.3 | 45.3 | 39.8 | 28.0 | 52.7 | 40.7 | 24.6 | 32.4 | 27.9 | 26.3 | 67.6 | 45.8 |
| **November** | 13.3 | 26.7 | 20.1 | 14.9 | 31.5 | 23.1 | 11.9 | 22.7 | 17.5 | 63.5 | 97.2 | 81.1 |
| **December** | 8.3 | 24.7 | 17.1 | 8.7 | 26.2 | 18.0 | 7.7 | 23.3 | 15.9 | 77.0 | 98.7 | 90.8 |


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**Table 2.4-8 Maximum and Minimum Safety Temperatures, Annual Exceedance Values, and Daily Average Temperatures from Evanston 1E (Wyoming) Station**

| Design Parameters | Dry Bulb (degrees Fahrenheit) | Coincident Wet Bulb (°F) | Non-Coincident Wet Bulb (°F) |
|---|---|---|---|
| Maximum Safety Temperature | 92 | 57.3 | 64.8 |
| | 103.6 (100-Year Return) | - | 69.5 (100-Year Return) |
| Minimum Safety Temperature | -29 | - | - |
| | -32.8 (100-Year Return) | - | - |
| 0.4% Annual Exceedance Values (Maximum Normal Temperature[1]) | 86.5 | 55.4 | 58.6 |
| | 98.7 (100-Year Return) | - | - |
| 99.6% Annual Exceedance Values (Minimum Normal Temperature[1]) | -0.7 | - | - |
| | -28.4 (100-Year Return) | - | - |
| 2% Annual Exceedance Values (Maximum) | 81 | 54.7 (Mean) | 56.4 |
| 98% Annual Exceedance Values (Minimum) | 6 | - | 5 |
| 1% Annual Exceedance Values (Maximum) | 83 | 54.7 (Mean) | 58 |
| 99% Annual Exceedance Values (Minimum) | 2 | - | 1 |
| 0% Annual Exceedance Values (Maximum) | 92 | 57.3 (Mean) | 65 |
| 100% Annual Exceedance Values (Minimum) | -29 | 81 | - |
| 5% Annual Exceedance Values (Maximum) | 75 | - | 54 |
| 95% Annual Exceedance Values (Minimum) | 12 | - | 11 |
| Daily Average on Maximum Dry Bulb Temperature Day | 78 | - | - |
| Daily Average on Minimum Dry Bulb Temperature Day | -25.5 | - | - |
| Maximum High Humidity Average Wet Bulb Globe Temperature Index for 0% Exceedance Maximum Wet Bulb Temperature Day | | | 57.9 |

[1] Maximum normal and minimum normal temperature values from Reference 2.4-6, for Evanston, Wyoming.


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**Table 2.4-9 Mean Monthly and Annual Precipitation at Kemmerer Unit 1 and Surrounding Stations**

| | Kemmerer Unit 1 | | Naughton Power Plant | | | | | | | | | | Big Piney, WY | | | | Evanston-Uinta County Airport – Burns Field | | Kemmerer, WY | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 4/9/22-4/8/23 | | 2019 | | 2020 | | 1991-2020 | | 2019-2021 | | 4/9/22-4/8/23 | | 4/9/22-4/8/23 | | 1981-2010[1] | | 1991-2020 | | 1981-2010[1] | |
| | *in* | *cm* | *in* | *cm* | *in* | *cm* | *in* | *cm* | *in* | *cm* | *in* | *cm* | *in* | *cm* | *in* | *cm* | *in* | *cm* | *in* | *cm* |
| **January** | 0.50 | 1.3 | 0.22 | 0.56 | 0.16 | 0.41 | 0.27 | 0.69 | 0.22 | 0.56 | 0.76 | 1.9 | 0.79 | 2.0 | 0.31 | 0.79 | 0.41 | 1.0 | 0.65 | 1.7 |
| **February** | 0.00 | 0.0 | 0.33 | 0.84 | 0.10 | 0.25 | 0.21 | 0.53 | 0.21 | 0.53 | 0.11 | 0.28 | 0.33 | 0.84 | 0.35 | 0.89 | 0.53 | 1.3 | 0.63 | 1.6 |
| **March** | 0.60 | 1.5 | 1.06 | 2.69 | 0.76 | 1.9 | 0.36 | 0.91 | 0.73 | 1.9 | 0.90 | 2.3 | 0.34 | 0.86 | 0.44 | 1.1 | 0.45 | 1.1 | 0.80 | 2.0 |
| **April** | 0.56 | 1.4 | 1.29 | 3.28 | 0.25 | 0.64 | 0.16 | 0.41 | 0.57 | 1.4 | 0.87 | 2.2 | 0.14 | 0.36 | 0.53 | 1.3 | 0.99 | 2.5 | 0.91 | 2.3 |
| **May** | 0.44 | 1.1 | 2.85 | 7.24 | 0.28 | 0.71 | 2.01 | 5.11 | 1.71 | 4.3 | 0.93 | 2.4 | 0.96 | 2.4 | 0.84 | 2.1 | 1.75 | 4.45 | 1.13 | 2.87 |
| **June** | 0.52 | 1.3 | 0.59 | 1.5 | 1.80 | 4.57 | 0.23 | 0.58 | 0.87 | 2.2 | 0.33 | 0.84 | 0.19 | 0.48 | 0.79 | 2.0 | 1.06 | 2.69 | 0.96 | 2.4 |
| **July** | 0.58 | 1.5 | 0.06 | 0.15 | 0.29 | 0.74 | 2.83 | 7.19 | 1.06 | 2.69 | 0.27 | 0.69 | 0.14 | 0.36 | 0.71 | 1.8 | 0.55 | 1.4 | 0.69 | 1.8 |
| **August** | 0.72 | 1.8 | 0.37 | 0.94 | 0.14 | 0.36 | 1.45 | 3.68 | 0.65 | 1.7 | 0.60 | 1.5 | 1.28 | 3.25 | 0.69 | 1.8 | 0.83 | 2.1 | 1.00 | 2.54 |
| **September** | 0.36 | 0.91 | 1.57 | 3.99 | 0.32 | 0.81 | 1.27 | 3.23 | 1.05 | 2.67 | 0.59 | 1.5 | 1.60 | 4.06 | 0.78 | 2.0 | 1.01 | 2.57 | 1.01 | 2.57 |
| **October** | 0.06 | 0.15 | 0.10 | 0.25 | 0.43 | 1.1 | 1.55 | 3.94 | 0.69 | 1.8 | 0.12 | 0.30 | 0.10 | 0.25 | 0.53 | 1.3 | 0.99 | 2.5 | 0.83 | 2.1 |
| **November** | 0.22 | 0.56 | 0.61 | 1.5 | 0.35 | 0.89 | 0.41 | 1.0 | 0.46 | 1.2 | 0.34 | 0.86 | 0.45 | 1.1 | 0.21 | 0.53 | 0.56 | 1.4 | 0.92 | 2.3 |
| **December** | 0.18 | 0.46 | 0.54 | 1.4 | 0.29 | 0.74 | 0.33 | 0.84 | 0.39 | 0.99 | 0.33 | 0.83 | 0.53 | 1.35 | 0.30 | 0.76 | 0.38 | 0.97 | 0.82 | 2.1 |
| **Annual** | 4.91 | 12.5 | 9.59 | 24.4 | 5.17 | 13.1 | 11.08 | 28.14 | 8.61 | 21.9 | 6.15 | 15.6 | 6.85 | 17.4 | 6.48 | 16.5 | 9.51 | 24.2 | 10.35 | 26.29 |

[1] 1981-2010 Normals for Wyoming Station, Western and Central Wyoming Weather Forecast Office (Reference 2.4-11)



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**Table 2.4-10 Hail Events in Lincoln County**

| County | Location | Date | Diameter (in.) |
|---|---|---|---|
| Lincoln | Afton | 7/12/1995 | 1 |
| | Kemmerer | 6/21/1996 | 1 |
| | Afton | 6/5/1997 | 1 |
| | Thayne | 6/5/1997 | 1.5 |
| | Bedford | 7/19/1997 | 1 |
| | Afton | 7/30/1998 | 0.75 |
| | Alpine | 7/30/1998 | 0.75 |
| | Auburn | 8/18/2000 | 1.25 |
| | Auburn | 8/18/2000 | 0.75 |
| | Kemmerer | 9/13/2001 | 0.75 |
| | Kemmerer | 9/13/2001 | 1.75 |
| | Smoot | 6/18/2004 | 1.25 |
| | Diamondville | 6/18/2004 | 0.75 |
| | Fontenelle | 7/23/2004 | 0.75 |
| | Thayne | 6/14/2006 | 1 |
| | Star Valley Ranch | 8/9/2008 | 0.75 |
| | Thayne | 6/6/2010 | 0.88 |
| | Bedford | 8/15/2011 | 1.5 |
| | Auburn | 8/15/2011 | 1 |
| | Afton Muni Airport | 8/15/2011 | 1 |
| | Grover | 8/15/2011 | 1 |
| | Star Valley Ranch | 7/6/2013 | 1 |
| | Afton | 9/17/2013 | 0.88 |
| | Diamondville | 8/7/2014 | 0.88 |
| | Smoot | 9/18/2014 | 1.5 |
| | Star Valley Ranch | 6/15/2015 | 1.75 |
| | Afton | 6/15/2015 | 1 |
| | Kemmerer | 6/12/2016 | 1.5 |
| | Thayne | 7/4/2016 | 0.75 |
| | Star Valley Ranch | 7/10/2016 | 1 |
| | Kemmerer | 8/7/2016 | 0.88 |


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**Table 2.4-11 High Wind Event Frequency for Kemmerer Unit 1**

| Category | Number of High Wind Events | Percent of Total (%) | Event Frequency ($Yr^{-1}$) |
|---|---|---|---|
| 1 (64 to 82 kt) | 435 | 89.69 | 15.725 |
| 2 (83 to 95 kt) | 36 | 7.42 | 1.301 |
| 3 (96 to 112 kt) | 11 | 2.27 | 0.398 |
| 4 (113 to 136 kt) | 3 | 0.62 | 0.108 |
| 5 (≥ 137 kt) | 0 | 0.00 | 0.000 |
| Total | 485 | 100.00 | N/A |



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**Table 2.4-12 Distribution of Tornadoes and Strike Probability Within a 2 Degree Area of Kemmerer Unit 1**

| Classification | # Tornadoes | Length (km) | Width (km) | Area ($km^2$) | $P_p(Yr^{-1})$ |
|---|---|---|---|---|---|
| EF0 | 35 | 1.630 | 0.011 | 0.017 | 2.20E-07 |
| EF1 | 11 | 1.302 | 0.019 | 0.024 | 9.81E-08 |
| EF2 | 3 | 4.453 | 0.019 | 0.086 | 9.47E-08 |
| EF3 | 0 | 0.000 | 0.000 | 0.000 | 0 |
| EF4 | 0 | 0.000 | 0.000 | 0.000 | 0 |
| EF5 | 0 | 0.000 | 0.000 | 0.000 | 0 |

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**Table 2.4-13 Mean Monthly and Annual Snowfall (1981–2010) at Sites Around the Kemmerer Unit 1 Site**

| Site | | Jan | Feb | Mar | Apr | May | Jun | |
|---|---|---|---|---|---|---|---|---|
| Kemmerer 2N, WY | in | 10.8 | 8.9 | 8.1 | 5.0 | 2.1 | 0.1 | |
| | mm | 274.3 | 226.1 | 205.7 | 127.0 | 53.3 | 2.5 | |
| Big Piney, WY | in | NO DATA | | | | | | |
| | mm | | | | | | | |
| Naughton Coal Plant | in | NO DATA | | | | | | |
| | mm | | | | | | | |
| **Site** | | **Jul** | **Aug** | **Sep** | **Oct** | **Nov** | **Dec** | **Annual** |
| Kemmerer 2N, WY | in | 0.0 | 0.0 | 0.8 | 2.4 | 10.3 | 13.5 | 62.0 |
| | mm | 0.0 | 0.0 | 20.3 | 61.0 | 261.6 | 342.9 | 1,574.8 |
| Big Piney, WY | in | NO DATA | | | | | | |
| | mm | | | | | | | |
| Naughton Coal Plant | in | NO DATA | | | | | | |
| | mm | | | | | | | |


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**Table 2.4-14 Probable Maximum Precipitation and Probable Maximum Winter Precipitation Values**

| Station Name | Winter Season 24-hour Probable Maximum Precipitation (in) | 48-hour PMWP (in) | 48-hour PMWP (psf) L=(48-hr PMWP)*(5.2psf/in) |
|---|---|---|---|
| Kemmerer 2N | 8.2 | 7.6 | 39.52 |
| Evanston 1E | 8.2 | 7.6 | 39.52 |


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**Table 2.4-15 Summary of Normal & Extreme Winter Precipitation Event Results and Resultant Ground Snow Loads**

| Station Name | Normal Winter Precipitation Events (ground snow loads) | | | | Normal Ground Snow Load[1] (psf) | Extreme Winter Precipitation Events (ground snow loads) | | Antecedent Snowpack[3] (psf) |
|---|---|---|---|---|---|---|---|---|
| | 100-year Return Period Snowpack | Historical Maximum Snowpack (psf) | 100-year Return Period Snowfall (psf) | Historical Maximum Snowfall (psf) | | Extreme Frozen Winter Precipitation[2] (psf) | Extreme Liquid Precipitation Event (psf) | |
| Kemmerer 2N | 24.39 | 29.79 | 18.17 | 18.33 | 29.79 | 18.33 | 39.52 | 69.31 |
| Evanston 1E | 24.39 | 27.24 | 22.09 | 31.2 | 31.2 | 31.2 | 39.52 | 70.72 |

1 The normal winter precipitation event should be the highest ground-level weight (in psf) among the normal winter precipitation events.

2 The extreme frozen winter precipitation event is the higher ground-level weight (in psf) between the 100-year return period snowfall and the historical maximum snowfall event.

3 The antecedent snowpack is the normal ground snow load plus the larger of the extreme frozen winter precipitation or extreme liquid precipitation.


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**Table 2.4-16 Monthly and Annual Percent Frequency of Precipitation Occurrence**

| | Average Percent Frequency of Precipitation Occurrence (hourly) | | |
|---|---|---|---|
| | Kemmerer Unit 1 4/9/22–4/9/23 | Naughton Power Plant 4/9/22–4/9/23 | Naughton Power Plant 2019–2021 |
| **January** | 4.03% | 3.36% | 1.66% |
| **February** | 0.00% | 1.04% | 1.72% |
| **March** | 4.70% | 3.23% | 2.69% |
| **April** | 3.33% | 5.56% | 2.54% |
| **May** | 1.61% | 2.28% | 4.93% |
| **June** | 1.67% | 1.67% | 1.85% |
| **July** | 1.88% | 0.67% | 1.30% |
| **August** | 1.61% | 1.74% | 1.70% |
| **September** | 1.67% | 2.22% | 2.91% |
| **October** | 0.40% | 0.94% | 1.66% |
| **November** | 0.97% | 1.53% | 2.69% |
| **December** | 2.15% | 1.88% | 1.93% |
| **Annual** | 1.81% | 1.96% | 2.07% |


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**Table 2.4-17 Monthly and Annual Average Wind Speed**

| | Kemmerer Unit 1 | | Naughton Power Plant | | | | Big Piney, WY[1] | |
|---|---|---|---|---|---|---|---|---|
| | 4/9/22–4/8/23 | | 4/9/22–4/8/23 | | 2019–2021 | | 4/9/22–4/8/23 | |
| | *mph* | *m/s* | *mph* | *m/s* | *mph* | *m/s* | *mph* | *m/s* |
| **Jan** | 8.1 | 3.6 | 10.2 | 4.56 | 10.8 | 4.83 | 5.1 | 2.3 |
| **Feb** | 8.2 | 3.7 | 11.4 | 5.10 | 13.4 | 6.00 | 5.4 | 2.4 |
| **Mar** | 7.6 | 3.3 | 9.7 | 4.3 | 9.7 | 4.3 | 5.1 | 2.3 |
| **Apr** | 11.4 | 5.10 | 14.0 | 6.26 | 12.5 | 5.59 | 9.0 | 4.0 |
| **May** | 11.0 | 4.92 | 14.2 | 6.35 | 11.4 | 5.10 | 9.8 | 4.4 |
| **Jun** | 9.7 | 4.3 | 11.5 | 5.14 | 11.1 | 4.96 | 7.9 | 3.5 |
| **Jul** | 8.3 | 3.7 | 10.2 | 4.56 | 10.3 | 4.60 | 7.9 | 3.5 |
| **Aug** | 7.4 | 3.3 | 9.5 | 4.2 | 10.8 | 4.83 | 6.7 | 3.0 |
| **Sep** | 6.8 | 3.0 | 9.6 | 4.3 | 10.8 | 4.83 | 6.3 | 2.8 |
| **Oct** | 9.2 | 4.1 | 10.7 | 4.78 | 12.3 | 5.50 | 6.7 | 3.0 |
| **Nov** | 8.4 | 3.8 | 11.4 | 5.10 | 10.9 | 4.87 | 6.3 | 2.8 |
| **Dec** | 8.4 | 3.8 | 11.2 | 5.01 | 10.7 | 4.78 | 5.4 | 2.4 |
| **Annual** | 8.7 | 3.9 | 11.1 | 4.96 | 11.2 | 5.01 | 6.8 | 3.0 |

[1] Preliminary Monthly Climate Dat (CF6) (Reference 2.4-20)


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**Table 2.4-18 Kemmerer Unit 1 Site Meteorological Persistence (2022–2023), 10 Meter Level, 1 Sector (Sheet 1 of 3)**

| Number of Sectors Included: 1 Width in Degrees: 22.5 | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Measurement Height, m:10 Speed Sensor:1 Direction Sensor: 1 | | | | | | | | | | | | | | | | |
| Speed Greater than or Equal to: 5.00 mph | | | | | | | | | | | | | | | | |
| | Direction | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 317 | 149 | 94 | 148 | 93 | 115 | 76 | 121 | 275 | 239 | 394 | 532 | 757 | 847 | 690 | 381 |
| 2 | 121 | 44 | 23 | 64 | 26 | 52 | 13 | 20 | 80 | 51 | 183 | 245 | 357 | 392 | 348 | 122 |
| 4 | 34 | 4 | 6 | 23 | 2 | 12 | 1 | 0 | 9 | 6 | 62 | 76 | 124 | 112 | 116 | 22 |
| 8 | 22 | 0 | 0 | 4 | 0 | 2 | 0 | 0 | 0 | 0 | 10 | 6 | 28 | 17 | 11 | 0 |
| 12 | 14 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 1 | 4 | 0 | 0 |
| 18 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| Speed Greater than or Equal to: 10.00 mph | | | | | | | | | | | | | | | | |
| | Direction | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 56 | 16 | 46 | 97 | 40 | 36 | 19 | 26 | 54 | 89 | 262 | 381 | 579 | 650 | 487 | 83 |
| 2 | 29 | 3 | 17 | 48 | 14 | 19 | 3 | 5 | 13 | 22 | 137 | 186 | 282 | 309 | 262 | 26 |
| 4 | 17 | 0 | 6 | 21 | 0 | 7 | 0 | 0 | 1 | 4 | 50 | 57 | 108 | 82 | 92 | 3 |
| 8 | 9 | 0 | 0 | 4 | 0 | 1 | 0 | 0 | 0 | 0 | 9 | 5 | 26 | 8 | 9 | 0 |
| 12 | 5 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 0 | 0 | 0 | 0 |
| Speed Greater than or Equal to: 15.00 mph | | | | | | | | | | | | | | | | |
| | Direction | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 20 | 1 | 26 | 47 | 13 | 11 | 4 | 11 | 19 | 49 | 154 | 218 | 315 | 330 | 255 | 15 |
| 2 | 11 | 0 | 12 | 28 | 4 | 7 | 1 | 2 | 4 | 16 | 92 | 106 | 171 | 149 | 133 | 2 |
| 4 | 9 | 0 | 6 | 15 | 0 | 4 | 0 | 0 | 0 | 2 | 39 | 35 | 69 | 40 | 44 | 0 |
| 8 | 5 | 0 | 0 | 4 | 0 | 0 | 0 | 0 | 0 | 0 | 9 | 4 | 15 | 5 | 6 | 0 |
| 12 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 0 | 0 | 0 | 0 |


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**Table 2.4-18 Kemmerer Unit 1 Site Meteorological Persistence (2022–2023), 10 Meter Level, 1 Sector (Sheet 2 of 3)**

| Number of Sectors Included: 1 Width in Degrees: 22.5 | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Measurement Height, m:10 Speed Sensor:1 Direction Sensor: 1 | | | | | | | | | | | | | | | | |
| Speed Greater than or Equal to: 20.00 mph | | | | | | | | | | | | | | | | |
| Direction | | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 11 | 0 | 12 | 27 | 4 | 1 | 0 | 5 | 6 | 18 | 86 | 80 | 145 | 109 | 82 | 2 |
| 2 | 9 | 0 | 9 | 15 | 2 | 0 | 0 | 1 | 1 | 6 | 49 | 33 | 68 | 42 | 40 | 1 |
| 4 | 7 | 0 | 5 | 9 | 0 | 0 | 0 | 0 | 0 | 2 | 13 | 8 | 23 | 11 | 12 | 0 |
| 8 | 3 | 0 | 0 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 4 | 0 | 1 | 0 |
| Speed Greater than or Equal to: 25.00 mph | | | | | | | | | | | | | | | | |
| Direction | | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 4 | 0 | 1 | 15 | 1 | 0 | 0 | 4 | 3 | 13 | 29 | 24 | 20 | 21 | 22 | 1 |
| 2 | 3 | 0 | 0 | 12 | 0 | 0 | 0 | 1 | 1 | 5 | 12 | 9 | 9 | 10 | 10 | 0 |
| 4 | 1 | 0 | 0 | 8 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 1 | 2 | 3 | 3 | 0 |
| 8 | 0 | 0 | 0 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| Speed Greater than or Equal to: 30.00 mph | | | | | | | | | | | | | | | | |
| Direction | | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 2 | 0 | 1 | 12 | 0 | 0 | 0 | 2 | 0 | 1 | 5 | 4 | 4 | 1 | 5 | 0 |
| 2 | 1 | 0 | 0 | 10 | 0 | 0 | 0 | 1 | 0 | 0 | 1 | 2 | 2 | 0 | 1 | 0 |
| 4 | 0 | 0 | 0 | 6 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 8 | 0 | 0 | 0 | 2 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |


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**Table 2.4-18 Kemmerer Unit 1 Site Meteorological Persistence (2022–2023), 10 Meter Level, 1 Sector (Sheet 3 of 3)**

| Number of Sectors Included: 1 Width in Degrees: 22.5 | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Measurement Height, m:10 Speed Sensor:1 Direction Sensor: 1 | | | | | | | | | | | | | | | | |
| Speed Greater than or Equal to: 35.00 mph | | | | | | | | | | | | | | | | |
| Direction | | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 0 | 0 | 0 | 8 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 0 | 0 | 0 |
| 2 | 0 | 0 | 0 | 6 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 4 | 0 | 0 | 0 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |


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**Table 2.4-19 Kemmerer Unit 1 Site Meteorological Persistence (2022–2023), 60 Meter Level, 1 Sector (Sheet 1 of 3)**

| Number of Sectors Included: 1 Width in Degrees: 22.5 | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Measurement Height, m: 60 Speed Sensor: 2 Direction Sensor: 2 | | | | | | | | | | | | | | | | |
| Speed Greater than or Equal to: 5.00 mph | | | | | | | | | | | | | | | | |
| | Direction | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 745 | 407 | 112 | 173 | 109 | 133 | 72 | 99 | 283 | 255 | 399 | 590 | 825 | 980 | 838 | 288 |
| 2 | 384 | 145 | 29 | 86 | 37 | 60 | 16 | 16 | 88 | 62 | 184 | 271 | 422 | 512 | 471 | 78 |
| 4 | 147 | 42 | 6 | 35 | 3 | 16 | 1 | 0 | 8 | 8 | 68 | 86 | 158 | 160 | 182 | 9 |
| 8 | 44 | 4 | 2 | 11 | 0 | 2 | 0 | 0 | 0 | 0 | 15 | 5 | 33 | 19 | 25 | 0 |
| 12 | 23 | 0 | 0 | 2 | 0 | 0 | 0 | 0 | 0 | 0 | 3 | 0 | 6 | 2 | 4 | 0 |
| 18 | 6 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| Speed Greater than or Equal to: 10.00 mph | | | | | | | | | | | | | | | | |
| | Direction | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 165 | 65 | 50 | 123 | 56 | 58 | 21 | 33 | 74 | 110 | 301 | 495 | 728 | 842 | 644 | 109 |
| 2 | 86 | 15 | 13 | 69 | 20 | 24 | 4 | 5 | 21 | 36 | 162 | 244 | 383 | 438 | 372 | 32 |
| 4 | 35 | 1 | 6 | 31 | 3 | 8 | 0 | 0 | 0 | 8 | 64 | 77 | 146 | 132 | 149 | 3 |
| 8 | 20 | 0 | 2 | 10 | 0 | 1 | 0 | 0 | 0 | 0 | 15 | 5 | 33 | 19 | 25 | 0 |
| 12 | 12 | 0 | 0 | 2 | 0 | 0 | 0 | 0 | 0 | 0 | 3 | 0 | 6 | 1 | 4 | 0 |
| 18 | 2 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |


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**Table 2.4-19 Kemmerer Unit 1 Site Meteorological Persistence (2022–2023), 60 Meter Level, 1 Sector (Sheet 2 of 3)**

| Number of Sectors Included: 1 Width in Degrees: 22.5 | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Measurement Height, m: 60 Speed Sensor: 2 Direction Sensor: 2 | | | | | | | | | | | | | | | | |
| Speed Greater than or Equal to: 15.00 mph | | | | | | | | | | | | | | | | |
| | Direction | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 36 | 6 | 25 | 77 | 23 | 21 | 7 | 17 | 24 | 59 | 226 | 346 | 524 | 522 | 414 | 47 |
| 2 | 22 | 1 | 7 | 47 | 6 | 12 | 0 | 4 | 3 | 20 | 125 | 159 | 274 | 246 | 238 | 14 |
| 4 | 13 | 0 | 3 | 26 | 0 | 5 | 0 | 0 | 0 | 6 | 49 | 45 | 112 | 59 | 83 | 2 |
| 8 | 9 | 0 | 0 | 10 | 0 | 1 | 0 | 0 | 0 | 0 | 11 | 1 | 28 | 3 | 13 | 0 |
| 12 | 5 | 0 | 0 | 2 | 0 | 0 | 0 | 0 | 0 | 0 | 3 | 0 | 5 | 0 | 0 | 0 |
| Speed Greater than or Equal to: 20.00 mph | | | | | | | | | | | | | | | | |
| | Direction | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 14 | 1 | 12 | 43 | 8 | 8 | 3 | 10 | 8 | 35 | 147 | 194 | 331 | 253 | 200 | 14 |
| 2 | 10 | 0 | 4 | 23 | 2 | 4 | 0 | 2 | 1 | 13 | 86 | 80 | 178 | 105 | 112 | 3 |
| 4 | 8 | 0 | 2 | 9 | 0 | 2 | 0 | 0 | 0 | 3 | 34 | 17 | 73 | 24 | 42 | 0 |
| 8 | 4 | 0 | 0 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 9 | 0 | 17 | 3 | 8 | 0 |
| 12 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 1 | 0 | 0 | 0 |
| Speed Greater than or Equal to: 25.00 mph | | | | | | | | | | | | | | | | |
| | Direction | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 11 | 0 | 7 | 27 | 2 | 0 | 0 | 6 | 4 | 13 | 78 | 76 | 164 | 94 | 78 | 1 |
| 2 | 9 | 0 | 3 | 14 | 0 | 0 | 0 | 1 | 1 | 5 | 44 | 30 | 79 | 36 | 38 | 0 |
| 4 | 7 | 0 | 1 | 8 | 0 | 0 | 0 | 0 | 0 | 2 | 20 | 5 | 33 | 9 | 14 | 0 |
| 12 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 1 | 0 | 0 | 0 |


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**Table 2.4-19 Kemmerer Unit 1 Site Meteorological Persistence (2022–2023), 60 Meter Level, 1 Sector (Sheet 3 of 3)**

| Number of Sectors Included: 1 Width in Degrees: 22.5 | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Measurement Height, m: 60 Speed Sensor: 2 Direction Sensor: 2 | | | | | | | | | | | | | | | | |
| Speed Greater than or Equal to: 30.00 mph | | | | | | | | | | | | | | | | |
| Direction | | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 5 | 0 | 1 | 14 | 0 | 0 | 0 | 4 | 3 | 7 | 34 | 33 | 47 | 25 | 31 | 1 |
| 2 | 2 | 0 | 0 | 11 | 0 | 0 | 0 | 1 | 1 | 3 | 14 | 12 | 20 | 10 | 14 | 0 |
| 4 | 0 | 0 | 0 | 7 | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 2 | 5 | 4 | 4 | 0 |
| 8 | 0 | 0 | 0 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| Speed Greater than or Equal to: 35.00 mph | | | | | | | | | | | | | | | | |
| Direction | | | | | | | | | | | | | | | | |
| **Hours** | **N** | **NNE** | **NE** | **ENE** | **E** | **ESE** | **SE** | **SSE** | **S** | **WSW** | **SW** | **WSW** | **W** | **WNW** | **NW** | **NNW** |
| 1 | 3 | 0 | 0 | 11 | 0 | 0 | 0 | 2 | 0 | 2 | 13 | 8 | 9 | 3 | 5 | 0 |
| 2 | 1 | 0 | 0 | 9 | 0 | 0 | 0 | 1 | 0 | 0 | 4 | 3 | 3 | 1 | 2 | 0 |
| 4 | 0 | 0 | 0 | 6 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 8 | 0 | 0 | 0 | 2 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |


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**Table 2.4-20 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | **Direction:** | WD10 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **A** | **Delta Temperature** | | Extremely Unstable | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 0 | 1 | 2 | 1 | 1 | 0 | 5 |
| **NNE** | 0 | 2 | 1 | 0 | 0 | 0 | 3 |
| **NE** | 0 | 3 | 3 | 0 | 0 | 0 | 6 |
| **ENE** | 0 | 1 | 10 | 10 | 4 | 1 | 26 |
| **E** | 0 | 4 | 32 | 18 | 8 | 0 | 62 |
| **ESE** | 0 | 8 | 20 | 2 | 2 | 0 | 32 |
| **SE** | 0 | 12 | 10 | 5 | 1 | 0 | 28 |
| **SSE** | 1 | 18 | 17 | 4 | 0 | 5 | 45 |
| **S** | 0 | 11 | 11 | 6 | 3 | 3 | 34 |
| **SSW** | 0 | 7 | 6 | 26 | 16 | 1 | 56 |
| **SW** | 0 | 2 | 4 | 45 | 76 | 23 | 150 |
| **WSW** | 0 | 0 | 1 | 3 | 8 | 9 | 21 |
| **W** | 0 | 1 | 0 | 3 | 3 | 0 | 7 |
| **WNW** | 0 | 0 | 2 | 6 | 5 | 3 | 16 |
| **NW** | 0 | 0 | 0 | 5 | 6 | 5 | 16 |
| **NNW** | 0 | 0 | 3 | 3 | 4 | 0 | 10 |
| **Total** | 1 | 70 | 122 | 137 | 137 | 50 | 517 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 16 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 88 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 517 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-21 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | **Direction:** | WD10 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **B** | **Delta Temperature** | | Moderately Unstable | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 0 | 7 | 5 | 0 | 0 | 0 | 12 |
| **NNE** | 0 | 6 | 4 | 0 | 0 | 0 | 10 |
| **NE** | 0 | 3 | 1 | 1 | 0 | 0 | 5 |
| **ENE** | 0 | 5 | 9 | 6 | 1 | 0 | 21 |
| **E** | 0 | 16 | 22 | 11 | 0 | 0 | 49 |
| **ESE** | 0 | 17 | 23 | 1 | 1 | 0 | 42 |
| **SE** | 0 | 22 | 18 | 1 | 1 | 1 | 43 |
| **SSE** | 1 | 32 | 9 | 4 | 1 | 0 | 47 |
| **S** | 2 | 39 | 18 | 6 | 2 | 0 | 67 |
| **SSW** | 0 | 14 | 5 | 8 | 1 | 1 | 29 |
| **SW** | 0 | 8 | 1 | 14 | 14 | 1 | 38 |
| **WSW** | 0 | 4 | 1 | 7 | 26 | 10 | 48 |
| **W** | 0 | 1 | 0 | 4 | 3 | 2 | 10 |
| **WNW** | 0 | 3 | 2 | 6 | 21 | 13 | 45 |
| **NW** | 0 | 2 | 1 | 6 | 9 | 2 | 21 |
| **NNW** | 0 | 3 | 5 | 5 | 2 | 1 | 15 |
| **Total** | 3 | 182 | 124 | 80 | 82 | 31 | 502 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 16 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 88 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 502 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-22 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | **Direction:** | WD10 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **C** | **Delta Temperature** | | Slightly Unstable | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction(from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 1 | 26 | 15 | 0 | 0 | 0 | 42 |
| **NNE** | 2 | 22 | 5 | 0 | 0 | 0 | 29 |
| **NE** | 0 | 14 | 2 | 2 | 1 | 0 | 19 |
| **ENE** | 2 | 16 | 12 | 17 | 2 | 0 | 49 |
| **E** | 3 | 22 | 32 | 15 | 7 | 1 | 80 |
| **ESE** | 4 | 29 | 25 | 5 | 1 | 0 | 64 |
| **SE** | 1 | 29 | 14 | 0 | 0 | 0 | 44 |
| **SSE** | 2 | 48 | 19 | 7 | 3 | 0 | 79 |
| **S** | 5 | 49 | 23 | 4 | 2 | 1 | 84 |
| **SSW** | 2 | 27 | 14 | 6 | 0 | 1 | 50 |
| **SW** | 0 | 17 | 11 | 13 | 12 | 2 | 55 |
| **WSW** | 1 | 12 | 12 | 28 | 35 | 11 | 99 |
| **W** | 0 | 3 | 9 | 28 | 42 | 14 | 96 |
| **WNW** | 0 | 10 | 21 | 74 | 83 | 40 | 228 |
| **NW** | 0 | 4 | 21 | 38 | 20 | 9 | 92 |
| **NNW** | 0 | 8 | 7 | 8 | 2 | 0 | 25 |
| **Total** | 23 | 336 | 242 | 245 | 210 | 79 | 1,135 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 16 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 88 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 1,135 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-23 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | **Direction:** | WD10 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **D** | **Delta Temperature** | | Neutral | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 34 | 173 | 88 | 7 | 1 | 0 | 303 |
| **NNE** | 33 | 90 | 31 | 11 | 6 | 0 | 171 |
| **NE** | 18 | 48 | 55 | 47 | 5 | 1 | 174 |
| **ENE** | 20 | 41 | 57 | 63 | 40 | 18 | 239 |
| **E** | 33 | 53 | 65 | 51 | 21 | 7 | 230 |
| **ESE** | 25 | 92 | 65 | 8 | 2 | 1 | 193 |
| **SE** | 35 | 75 | 39 | 7 | 0 | 0 | 156 |
| **SSE** | 34 | 126 | 27 | 9 | 1 | 0 | 197 |
| **S** | 35 | 141 | 31 | 13 | 2 | 1 | 223 |
| **SSW** | 33 | 57 | 28 | 25 | 14 | 3 | 160 |
| **SW** | 30 | 42 | 65 | 55 | 43 | 10 | 245 |
| **WSW** | 16 | 74 | 194 | 380 | 281 | 86 | 1,031 |
| **W** | 36 | 107 | 331 | 665 | 721 | 534 | 2,394 |
| **WNW** | 33 | 129 | 729 | 1,265 | 750 | 255 | 3,161 |
| **NW** | 32 | 123 | 310 | 246 | 123 | 21 | 855 |
| **NNW** | 32 | 114 | 93 | 30 | 2 | 0 | 271 |
| **Total** | 479 | 1,485 | 2,208 | 2,882 | 2,012 | 937 | 10,003 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 16 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 88 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 10,003 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-24 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | **Direction:** | WD10 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **E** | **Delta Temperature** | | Slightly Stable | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 37 | 286 | 173 | 13 | 1 | 0 | 510 |
| **NNE** | 46 | 142 | 66 | 16 | 1 | 0 | 271 |
| **NE** | 39 | 67 | 28 | 25 | 1 | 0 | 160 |
| **ENE** | 41 | 34 | 16 | 10 | 5 | 6 | 112 |
| **E** | 56 | 39 | 18 | 20 | 3 | 0 | 136 |
| **ESE** | 44 | 47 | 14 | 3 | 0 | 0 | 108 |
| **SE** | 58 | 67 | 10 | 1 | 0 | 0 | 136 |
| **SSE** | 60 | 94 | 7 | 3 | 0 | 0 | 164 |
| **S** | 36 | 103 | 28 | 4 | 1 | 1 | 173 |
| **SSW** | 29 | 54 | 29 | 11 | 5 | 3 | 131 |
| **SW** | 14 | 51 | 60 | 37 | 18 | 3 | 183 |
| **WSW** | 12 | 46 | 151 | 253 | 81 | 23 | 566 |
| **W** | 13 | 69 | 193 | 275 | 158 | 69 | 777 |
| **WNW** | 22 | 105 | 312 | 339 | 150 | 65 | 993 |
| **NW** | 28 | 108 | 313 | 110 | 24 | 4 | 587 |
| **NNW** | 33 | 159 | 165 | 23 | 2 | 0 | 382 |
| **Total** | 568 | 1,471 | 1,583 | 1,143 | 450 | 174 | 5,389 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 16 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 88 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 5,389 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-25 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | **Direction:** | WD10 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **F** | **Delta Temperature** | | Moderately Stable | | |
| **Wind Speed (mph)** | | | | | | | |
| **Wind Direction** (from) | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 68 | 425 | 314 | 15 | 0 | 0 | 822 |
| **NNE** | 92 | 325 | 65 | 3 | 0 | 0 | 485 |
| **NE** | 83 | 105 | 11 | 1 | 0 | 0 | 200 |
| **ENE** | 59 | 53 | 3 | 2 | 0 | 0 | 117 |
| **E** | 44 | 39 | 2 | 3 | 0 | 0 | 88 |
| **ESE** | 67 | 38 | 2 | 0 | 0 | 0 | 107 |
| **SE** | 42 | 48 | 5 | 1 | 0 | 0 | 96 |
| **SSE** | 52 | 76 | 7 | 1 | 0 | 0 | 136 |
| **S** | 31 | 62 | 20 | 1 | 0 | 0 | 114 |
| **SSW** | 11 | 23 | 15 | 2 | 0 | 0 | 51 |
| **SW** | 6 | 21 | 26 | 12 | 1 | 1 | 67 |
| **WSW** | 6 | 15 | 43 | 33 | 4 | 0 | 101 |
| **W** | 8 | 31 | 65 | 28 | 4 | 1 | 137 |
| **WNW** | 14 | 50 | 112 | 22 | 3 | 0 | 201 |
| **NW** | 19 | 84 | 135 | 31 | 0 | 0 | 269 |
| **NNW** | 32 | 163 | 155 | 14 | 0 | 0 | 364 |
| **Total** | 634 | 1,558 | 980 | 169 | 12 | 2 | 3,355 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 16 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 88 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 3,355 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-26 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | **Direction:** | WD10 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **G** | **Delta Temperature** | | Extremely Stable | | |
| **Wind Speed (mph)** | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 57 | 936 | 1,666 | 74 | 0 | 0 | 2,733 |
| **NNE** | 83 | 1,069 | 556 | 2 | 0 | 0 | 1,710 |
| **NE** | 48 | 112 | 9 | 0 | 0 | 0 | 169 |
| **ENE** | 28 | 25 | 2 | 0 | 0 | 0 | 55 |
| **E** | 16 | 23 | 0 | 0 | 0 | 0 | 39 |
| **ESE** | 12 | 15 | 1 | 0 | 0 | 0 | 28 |
| **SE** | 18 | 18 | 0 | 0 | 0 | 0 | 36 |
| **SSE** | 22 | 26 | 3 | 0 | 0 | 0 | 51 |
| **S** | 9 | 22 | 2 | 0 | 0 | 0 | 33 |
| **SSW** | 8 | 6 | 4 | 0 | 0 | 0 | 18 |
| **SW** | 3 | 4 | 4 | 0 | 0 | 0 | 11 |
| **WSW** | 2 | 2 | 5 | 1 | 0 | 0 | 10 |
| **W** | 7 | 1 | 3 | 1 | 2 | 0 | 14 |
| **WNW** | 4 | 28 | 25 | 0 | 0 | 0 | 57 |
| **NW** | 13 | 38 | 25 | 7 | 0 | 0 | 83 |
| **NNW** | 37 | 65 | 135 | 15 | 0 | 0 | 252 |
| **Total** | 367 | 2,390 | 2,440 | 100 | 2 | 0 | 5,299 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 16 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 88 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 5,299 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-27 Naughton Power Plant Joint Frequency Distribution 2019–2021, 10-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | **Direction:** | WD10 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **ALL** | **Delta Temperature** | | | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 197 | 1,854 | 2,263 | 110 | 3 | 0 | 4,427 |
| **NNE** | 256 | 1,656 | 728 | 32 | 7 | 0 | 2,679 |
| **NE** | 188 | 352 | 109 | 76 | 7 | 1 | 733 |
| **ENE** | 150 | 175 | 109 | 108 | 52 | 25 | 619 |
| **E** | 152 | 196 | 171 | 118 | 39 | 8 | 684 |
| **ESE** | 152 | 246 | 150 | 19 | 6 | 1 | 574 |
| **SE** | 154 | 271 | 96 | 15 | 2 | 1 | 539 |
| **SSE** | 172 | 420 | 89 | 28 | 5 | 5 | 719 |
| **S** | 118 | 427 | 133 | 34 | 10 | 6 | 728 |
| **SSW** | 83 | 188 | 101 | 78 | 36 | 9 | 495 |
| **SW** | 53 | 145 | 171 | 176 | 164 | 40 | 749 |
| **WSW** | 37 | 153 | 407 | 705 | 435 | 139 | 1,876 |
| **W** | 64 | 213 | 601 | 1,004 | 933 | 620 | 3,435 |
| **WNW** | 73 | 325 | 1,203 | 1,712 | 1,012 | 376 | 4,701 |
| **NW** | 92 | 360 | 805 | 443 | 182 | 41 | 1,923 |
| **NNW** | 134 | 511 | 563 | 98 | 12 | 1 | 1,319 |
| **Total** | 2,075 | 7,492 | 7,699 | 4,756 | 2,905 | 1,273 | 26,200 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 16 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 88 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 26,200 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-28 Naughton Power Plant Joint Frequency Distribution 2019-2021, 50-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS50 | **Direction:** | WD50 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **A** | **Delta Temperature** | | Extremely Unstable | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 0 | 0 | 1 | 1 | 1 | 0 | 3 |
| **NNE** | 0 | 2 | 3 | 0 | 0 | 0 | 5 |
| **NE** | 0 | 1 | 4 | 0 | 0 | 0 | 5 |
| **ENE** | 0 | 0 | 4 | 5 | 1 | 3 | 13 |
| **E** | 0 | 5 | 26 | 29 | 10 | 1 | 71 |
| **ESE** | 0 | 4 | 23 | 6 | 1 | 1 | 35 |
| **SE** | 0 | 9 | 10 | 4 | 1 | 1 | 25 |
| **SSE** | 0 | 13 | 20 | 4 | 0 | 4 | 41 |
| **S** | 0 | 11 | 14 | 3 | 4 | 6 | 38 |
| **SSW** | 0 | 6 | 5 | 17 | 12 | 1 | 41 |
| **SW** | 0 | 0 | 4 | 28 | 65 | 57 | 154 |
| **WSW** | 0 | 1 | 1 | 4 | 10 | 21 | 37 |
| **W** | 0 | 0 | 0 | 2 | 2 | 0 | 4 |
| **WNW** | 0 | 1 | 1 | 6 | 7 | 3 | 18 |
| **NW** | 0 | 0 | 0 | 4 | 2 | 8 | 14 |
| **NNW** | 0 | 0 | 3 | 3 | 5 | 2 | 13 |
| **Total** | 0 | 53 | 119 | 116 | 121 | 108 | 517 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 2 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 288 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 517 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-29 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS50 | **Direction:** | WD50 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **B** | **Delta Temperature** | | Moderately Unstable | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 0 | 6 | 6 | 2 | 0 | 0 | 14 |
| **NNE** | 0 | 5 | 5 | 0 | 0 | 0 | 10 |
| **NE** | 0 | 3 | 3 | 1 | 0 | 0 | 7 |
| **ENE** | 0 | 2 | 8 | 7 | 2 | 0 | 19 |
| **E** | 0 | 12 | 16 | 12 | 1 | 0 | 41 |
| **ESE** | 1 | 11 | 24 | 6 | 0 | 1 | 43 |
| **SE** | 0 | 22 | 22 | 3 | 0 | 1 | 48 |
| **SSE** | 0 | 26 | 11 | 4 | 0 | 2 | 43 |
| **S** | 3 | 38 | 25 | 5 | 4 | 0 | 75 |
| **SSW** | 0 | 18 | 4 | 7 | 2 | 1 | 32 |
| **SW** | 0 | 5 | 1 | 6 | 14 | 7 | 33 |
| **WSW** | 0 | 2 | 1 | 4 | 24 | 21 | 52 |
| **W** | 0 | 1 | 0 | 2 | 0 | 1 | 4 |
| **WNW** | 0 | 3 | 2 | 6 | 10 | 25 | 46 |
| **NW** | 0 | 0 | 1 | 4 | 10 | 5 | 20 |
| **NNW** | 0 | 3 | 2 | 4 | 4 | 2 | 15 |
| **Total** | 4 | 157 | 131 | 73 | 71 | 66 | 502 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 2 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 288 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 502 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-30 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS50 | **Direction:** | WD50 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **C** | **Delta Temperature** | | Slightly Unstable | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 0 | 20 | 16 | 1 | 0 | 0 | 37 |
| **NNE** | 1 | 22 | 6 | 0 | 0 | 0 | 29 |
| **NE** | 1 | 18 | 0 | 2 | 1 | 0 | 22 |
| **ENE** | 1 | 12 | 10 | 12 | 6 | 0 | 41 |
| **E** | 4 | 20 | 25 | 22 | 9 | 3 | 83 |
| **ESE** | 1 | 22 | 24 | 11 | 1 | 1 | 60 |
| **SE** | 4 | 27 | 20 | 3 | 0 | 0 | 54 |
| **SSE** | 3 | 52 | 13 | 8 | 2 | 1 | 79 |
| **S** | 3 | 51 | 26 | 5 | 4 | 2 | 91 |
| **SSW** | 2 | 22 | 14 | 7 | 1 | 1 | 47 |
| **SW** | 0 | 13 | 9 | 9 | 10 | 4 | 45 |
| **WSW** | 1 | 7 | 9 | 23 | 39 | 16 | 95 |
| **W** | 0 | 2 | 6 | 21 | 31 | 22 | 82 |
| **WNW** | 0 | 8 | 17 | 53 | 75 | 53 | 206 |
| **NW** | 1 | 3 | 16 | 48 | 36 | 23 | 127 |
| **NNW** | 0 | 5 | 8 | 13 | 6 | 1 | 33 |
| **Total** | 22 | 304 | 219 | 238 | 221 | 127 | 1,131 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 2 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 288 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 1,131 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-31 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS50 | **Direction:** | WD50 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **D** | **Delta Temperature** | | Neutral | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 37 | 120 | 70 | 17 | 0 | 1 | 245 |
| **NNE** | 38 | 116 | 44 | 14 | 3 | 1 | 216 |
| **NE** | 31 | 43 | 35 | 45 | 15 | 3 | 172 |
| **ENE** | 17 | 45 | 54 | 60 | 54 | 32 | 262 |
| **E** | 29 | 46 | 60 | 48 | 27 | 27 | 237 |
| **ESE** | 23 | 76 | 68 | 33 | 6 | 4 | 210 |
| **SE** | 25 | 85 | 52 | 12 | 3 | 0 | 177 |
| **SSE** | 49 | 102 | 19 | 6 | 3 | 1 | 180 |
| **S** | 47 | 139 | 28 | 18 | 4 | 3 | 239 |
| **SSW** | 18 | 66 | 26 | 20 | 13 | 7 | 150 |
| **SW** | 26 | 50 | 36 | 40 | 34 | 33 | 219 |
| **WSW** | 8 | 46 | 119 | 276 | 252 | 106 | 807 |
| **W** | 18 | 78 | 243 | 510 | 616 | 564 | 2,029 |
| **WNW** | 18 | 100 | 514 | 1,201 | 959 | 544 | 3,336 |
| **NW** | 18 | 103 | 349 | 422 | 249 | 115 | 1,256 |
| **NNW** | 22 | 63 | 88 | 52 | 14 | 2 | 241 |
| **Total** | 424 | 1,278 | 1,805 | 2,774 | 2,252 | 1,443 | 9,976 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 2 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 288 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 9,976 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-32 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00-12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS50 | **Direction:** | WD50 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **E** | **Delta Temperature** | | Slightly Stable | | |
| **Wind Speed (mph)** | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 24 | 136 | 118 | 19 | 7 | 2 | 306 |
| **NNE** | 33 | 108 | 65 | 23 | 4 | 0 | 233 |
| **NE** | 36 | 61 | 26 | 37 | 9 | 1 | 170 |
| **ENE** | 24 | 32 | 12 | 20 | 12 | 5 | 105 |
| **E** | 21 | 28 | 20 | 23 | 12 | 8 | 112 |
| **ESE** | 26 | 34 | 26 | 9 | 2 | 0 | 97 |
| **SE** | 36 | 50 | 11 | 5 | 1 | 0 | 103 |
| **SSE** | 50 | 61 | 18 | 5 | 1 | 0 | 135 |
| **S** | 39 | 82 | 38 | 8 | 2 | 2 | 171 |
| **SSW** | 36 | 76 | 44 | 16 | 2 | 8 | 182 |
| **SW** | 21 | 42 | 40 | 36 | 29 | 16 | 184 |
| **WSW** | 12 | 54 | 67 | 174 | 105 | 35 | 447 |
| **W** | 10 | 67 | 144 | 290 | 229 | 88 | 828 |
| **WNW** | 16 | 88 | 236 | 443 | 263 | 143 | 1,189 |
| **NW** | 19 | 101 | 282 | 253 | 76 | 27 | 758 |
| **NNW** | 12 | 123 | 129 | 57 | 7 | 3 | 331 |
| **Total** | 415 | 1,143 | 1,276 | 1,418 | 761 | 338 | 5,351 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 2 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 288 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 5,351 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-33 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS50 | **Direction:** | WD50 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **F** | **Delta Temperature** | | Moderately Stable | | |
| **Wind Speed (mph)** | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 53 | 174 | 181 | 52 | 6 | 0 | 466 |
| **NNE** | 40 | 166 | 100 | 21 | 1 | 0 | 328 |
| **NE** | 35 | 52 | 20 | 5 | 2 | 0 | 114 |
| **ENE** | 34 | 19 | 4 | 5 | 3 | 0 | 65 |
| **E** | 28 | 20 | 5 | 1 | 2 | 0 | 56 |
| **ESE** | 29 | 24 | 3 | 3 | 0 | 0 | 59 |
| **SE** | 47 | 37 | 8 | 4 | 0 | 0 | 96 |
| **SSE** | 43 | 52 | 8 | 3 | 1 | 0 | 107 |
| **S** | 50 | 93 | 37 | 12 | 1 | 0 | 193 |
| **SSW** | 40 | 70 | 35 | 10 | 1 | 0 | 156 |
| **SW** | 30 | 41 | 22 | 15 | 5 | 6 | 119 |
| **WSW** | 17 | 33 | 29 | 44 | 21 | 3 | 147 |
| **W** | 26 | 64 | 61 | 70 | 15 | 2 | 238 |
| **WNW** | 14 | 76 | 115 | 104 | 12 | 0 | 321 |
| **NW** | 31 | 113 | 167 | 98 | 11 | 0 | 420 |
| **NNW** | 35 | 149 | 162 | 59 | 4 | 0 | 409 |
| **Total** | 552 | 1,183 | 957 | 506 | 85 | 11 | 3,294 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 2 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 288 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 3,294 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-34 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS50 | **Direction:** | WD50 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **G** | **Delta Temperature** | | Extremely Stable | | |
| **Wind Speed (mph)** | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 58 | 520 | 1,036 | 253 | 2 | 0 | 1,869 |
| **NNE** | 89 | 681 | 650 | 137 | 0 | 0 | 1,557 |
| **NE** | 79 | 212 | 24 | 4 | 0 | 0 | 319 |
| **ENE** | 59 | 33 | 1 | 0 | 0 | 0 | 93 |
| **E** | 50 | 18 | 0 | 0 | 0 | 0 | 68 |
| **ESE** | 54 | 20 | 0 | 0 | 0 | 0 | 74 |
| **SE** | 53 | 26 | 1 | 0 | 0 | 0 | 80 |
| **SSE** | 31 | 34 | 4 | 0 | 0 | 0 | 69 |
| **S** | 35 | 50 | 10 | 2 | 0 | 0 | 97 |
| **SSW** | 17 | 25 | 18 | 4 | 0 | 0 | 64 |
| **SW** | 16 | 17 | 4 | 5 | 1 | 0 | 43 |
| **WSW** | 11 | 17 | 7 | 5 | 2 | 0 | 42 |
| **W** | 11 | 22 | 25 | 3 | 2 | 0 | 63 |
| **WNW** | 18 | 46 | 45 | 21 | 0 | 0 | 130 |
| **NW** | 31 | 92 | 75 | 31 | 1 | 0 | 230 |
| **NNW** | 38 | 173 | 198 | 34 | 2 | 0 | 445 |
| **Total** | 650 | 1,986 | 2,098 | 499 | 10 | 0 | 5,243 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 2 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 288 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 5,243 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-35 Naughton Power Plant Joint Frequency Distribution 2019–2021, 50-Meter 3-Year Composite**

| Hours at Each Wind Speed and Direction | | | | | | | |
|---|---|---|---|---|---|---|---|
| Period of Record = 01/01/2019 00:00–12/31/2021 23:00 | | | | | | | |
| **Elevation:** | | **Speed:** | WS50 | **Direction:** | WD50 | **Lapse:** | DT50_10 |
| **Stability Class:** | | **ALL** | **Delta Temperature** | | | | |
| Wind Speed (mph) | | | | | | | |
| **Wind Direction (from)** | **0.7–3.5** | **3.6–7.5** | **7.6–12.5** | **12.6–18.5** | **18.6–24.5** | **> 24.6** | **Total** |
| **N** | 172 | 976 | 1,428 | 345 | 16 | 3 | 2,940 |
| **NNE** | 201 | 1,100 | 873 | 195 | 8 | 1 | 2,378 |
| **NE** | 182 | 390 | 112 | 94 | 27 | 4 | 809 |
| **ENE** | 135 | 143 | 93 | 109 | 78 | 40 | 598 |
| **E** | 132 | 149 | 152 | 135 | 61 | 39 | 668 |
| **ESE** | 134 | 191 | 168 | 68 | 10 | 7 | 578 |
| **SE** | 165 | 256 | 124 | 31 | 5 | 2 | 583 |
| **SSE** | 176 | 340 | 93 | 30 | 7 | 8 | 654 |
| **S** | 177 | 464 | 178 | 53 | 19 | 13 | 904 |
| **SSW** | 113 | 283 | 146 | 81 | 31 | 18 | 672 |
| **SW** | 93 | 168 | 116 | 139 | 158 | 123 | 797 |
| **WSW** | 49 | 160 | 233 | 530 | 453 | 202 | 1,627 |
| **W** | 65 | 234 | 479 | 898 | 895 | 677 | 3,248 |
| **WNW** | 66 | 322 | 930 | 1,834 | 1,326 | 768 | 5,246 |
| **NW** | 100 | 412 | 890 | 860 | 385 | 178 | 2,825 |
| **NNW** | 107 | 516 | 590 | 222 | 42 | 10 | 1,487 |
| **Total** | 2,067 | 6,104 | 6,605 | 5,624 | 3,521 | 2,093 | 26,014 |
| **Calm Hours not Included above for:** | | | | **Total Period** | | **All Hours** | 2 |
| **Variable Direction Hours for:** | | | | **Total Period** | | **All Hours** | 0 |
| **Invalid Hours for:** | | | | **Total Period** | | **All Hours** | 288 |
| **Number of Valid Hours for this Table:** | | | | **Total Period** | | **All Hours** | 26,014 |
| **Total Hours for the Period:** | | | | **Total Period** | | **All Hours** | 26,304 |


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**Table 2.4-36 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | WD10 | | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | A | | | **Delta Temperature** | | | Extremely Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 1 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **NE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 1 | 4 | 6 | 3 | 0 | 0 | 14 |
| **E** | 0 | 0 | 0 | 1 | 0 | 2 | 10 | 9 | 1 | 0 | 0 | 23 |
| **ESE** | 0 | 0 | 0 | 0 | 0 | 2 | 27 | 4 | 1 | 0 | 0 | 34 |
| **SE** | 0 | 0 | 0 | 0 | 0 | 4 | 5 | 1 | 0 | 0 | 0 | 10 |
| **SSE** | 0 | 0 | 0 | 1 | 0 | 7 | 6 | 3 | 0 | 0 | 0 | 17 |
| **S** | 0 | 0 | 0 | 0 | 0 | 3 | 9 | 6 | 0 | 0 | 0 | 18 |
| **SSW** | 0 | 0 | 0 | 0 | 1 | 2 | 8 | 3 | 4 | 0 | 0 | 18 |
| **SW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 3 | 29 | 16 | 0 | 48 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 9 | 6 | 1 | 17 |
| **W** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **NW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 3 | 0 | 2 | 5 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 0 | 1 | 4 |
| **Total** | 0 | 0 | 0 | 2 | 1 | 21 | 69 | 37 | 53 | 22 | 4 | 209 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 308 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 209 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-37 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS10 | | | **Direction:** WD10 | | | | **Lapse:** DT50_10 | | |
| **Stability Class:** | | | B | | | **Delta Temperature** | | | Moderately Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 0 | 1 | 0 | 0 | 3 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **NE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **ENE** | 0 | 0 | 0 | 0 | 1 | 1 | 3 | 1 | 0 | 0 | 0 | 6 |
| **E** | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 3 | 0 | 0 | 0 | 5 |
| **ESE** | 0 | 0 | 1 | 0 | 0 | 2 | 7 | 0 | 0 | 0 | 0 | 10 |
| **SE** | 0 | 0 | 0 | 0 | 0 | 4 | 3 | 0 | 0 | 0 | 0 | 7 |
| **SSE** | 0 | 0 | 0 | 0 | 1 | 2 | 0 | 0 | 0 | 1 | 0 | 4 |
| **S** | 0 | 0 | 0 | 0 | 0 | 2 | 4 | 1 | 1 | 0 | 0 | 8 |
| **SSW** | 0 | 0 | 0 | 0 | 0 | 1 | 1 | 2 | 0 | 1 | 0 | 5 |
| **SW** | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 2 | 7 | 0 | 0 | 10 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 3 | 2 | 1 | 6 |
| **W** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 3 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 1 | 4 |
| **NW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 1 | 1 | 1 | 4 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 0 | 0 | 1 |
| **Total** | 0 | 0 | 1 | 0 | 2 | 15 | 21 | 10 | 14 | 8 | 5 | 76 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 308 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 76 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-38 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | WD10 | | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | C | | | **Delta Temperature** | | | Slightly Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 0 | 1 | 0 | 0 | 4 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 3 | 0 | 0 | 1 | 0 | 0 | 4 |
| **NE** | 0 | 0 | 0 | 2 | 1 | 3 | 3 | 4 | 1 | 0 | 0 | 14 |
| **ENE** | 0 | 0 | 0 | 0 | 1 | 3 | 1 | 7 | 0 | 0 | 0 | 12 |
| **E** | 0 | 0 | 0 | 0 | 1 | 6 | 12 | 5 | 1 | 0 | 0 | 25 |
| **ESE** | 0 | 0 | 0 | 0 | 1 | 7 | 8 | 1 | 0 | 0 | 0 | 17 |
| **SE** | 1 | 0 | 0 | 4 | 4 | 7 | 6 | 0 | 0 | 0 | 0 | 22 |
| **SSE** | 0 | 0 | 0 | 2 | 7 | 6 | 9 | 2 | 0 | 0 | 0 | 26 |
| **S** | 0 | 0 | 0 | 0 | 6 | 8 | 8 | 3 | 2 | 0 | 0 | 27 |
| **SSW** | 0 | 0 | 0 | 1 | 5 | 5 | 4 | 1 | 2 | 2 | 0 | 20 |
| **SW** | 0 | 0 | 0 | 0 | 0 | 4 | 3 | 2 | 6 | 3 | 1 | 19 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 4 | 0 | 2 | 8 | 7 | 0 | 21 |
| **W** | 0 | 0 | 0 | 0 | 0 | 3 | 1 | 2 | 6 | 27 | 0 | 39 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 6 | 24 | 22 | 16 | 71 |
| **NW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 4 | 6 | 6 | 0 | 16 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 2 | 4 | 2 | 2 | 0 | 0 | 10 |
| **Total** | 1 | 0 | 0 | 9 | 26 | 63 | 63 | 41 | 60 | 67 | 17 | 347 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 308 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 347 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-39 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | | WD10 | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | D | | | **Delta Temperature** | | Neutral | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 1 | 12 | 13 | 34 | 26 | 3 | 3 | 0 | 0 | 92 |
| **NNE** | 0 | 1 | 3 | 8 | 9 | 16 | 18 | 4 | 1 | 0 | 0 | 60 |
| **NE** | 1 | 2 | 3 | 6 | 3 | 8 | 11 | 12 | 3 | 1 | 0 | 50 |
| **ENE** | 0 | 0 | 1 | 1 | 5 | 6 | 18 | 10 | 9 | 5 | 7 | 62 |
| **E** | 0 | 0 | 1 | 6 | 6 | 7 | 10 | 10 | 5 | 1 | 0 | 46 |
| **ESE** | 1 | 0 | 0 | 3 | 4 | 11 | 12 | 3 | 3 | 0 | 0 | 37 |
| **SE** | 1 | 1 | 6 | 13 | 11 | 8 | 19 | 1 | 0 | 0 | 0 | 60 |
| **SSE** | 0 | 1 | 2 | 17 | 13 | 16 | 12 | 3 | 0 | 0 | 0 | 64 |
| **S** | 2 | 0 | 3 | 17 | 21 | 32 | 23 | 3 | 4 | 0 | 0 | 105 |
| **SSW** | 0 | 0 | 3 | 8 | 6 | 14 | 6 | 6 | 5 | 2 | 0 | 50 |
| **SW** | 0 | 1 | 1 | 7 | 6 | 6 | 19 | 9 | 25 | 8 | 0 | 82 |
| **WSW** | 0 | 1 | 4 | 3 | 7 | 15 | 49 | 76 | 89 | 43 | 8 | 295 |
| **W** | 1 | 1 | 0 | 4 | 6 | 23 | 64 | 98 | 206 | 176 | 29 | 608 |
| **WNW** | 0 | 0 | 2 | 7 | 11 | 21 | 134 | 343 | 360 | 112 | 18 | 1,008 |
| **NW** | 0 | 0 | 1 | 5 | 8 | 26 | 69 | 92 | 71 | 17 | 3 | 292 |
| **NNW** | 0 | 0 | 2 | 3 | 9 | 17 | 16 | 8 | 3 | 0 | 0 | 58 |
| **Total** | 6 | 8 | 33 | 120 | 138 | 260 | 506 | 681 | 787 | 365 | 65 | 2969 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 308 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 2,969 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-40 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS10 | | | **Direction:** WD10 | | | | **Lapse:** DT50_10 | | |
| **Stability Class:** | | | E | | | **Delta Temperature** | | | Slightly Stable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 1 | 7 | 7 | 22 | 53 | 65 | 8 | 1 | 0 | 0 | 164 |
| **NNE** | 0 | 1 | 5 | 12 | 20 | 45 | 30 | 14 | 1 | 0 | 0 | 128 |
| **NE** | 0 | 2 | 2 | 12 | 8 | 11 | 11 | 3 | 2 | 0 | 0 | 51 |
| **ENE** | 0 | 1 | 3 | 12 | 7 | 4 | 6 | 5 | 5 | 1 | 4 | 48 |
| **E** | 0 | 1 | 4 | 7 | 7 | 5 | 7 | 1 | 3 | 1 | 0 | 36 |
| **ESE** | 0 | 0 | 5 | 8 | 9 | 4 | 7 | 0 | 0 | 0 | 0 | 33 |
| **SE** | 0 | 2 | 5 | 16 | 12 | 17 | 7 | 1 | 0 | 0 | 0 | 60 |
| **SSE** | 0 | 2 | 7 | 20 | 23 | 17 | 11 | 5 | 1 | 0 | 0 | 86 |
| **S** | 0 | 1 | 3 | 21 | 24 | 27 | 10 | 2 | 1 | 0 | 0 | 89 |
| **SSW** | 1 | 0 | 5 | 7 | 6 | 19 | 13 | 10 | 6 | 1 | 0 | 68 |
| **SW** | 0 | 0 | 2 | 2 | 3 | 7 | 15 | 11 | 24 | 19 | 5 | 88 |
| **WSW** | 0 | 0 | 0 | 4 | 2 | 14 | 30 | 72 | 48 | 21 | 7 | 198 |
| **W** | 0 | 0 | 2 | 7 | 3 | 12 | 27 | 61 | 83 | 26 | 8 | 229 |
| **WNW** | 0 | 2 | 0 | 11 | 8 | 13 | 57 | 106 | 129 | 50 | 14 | 390 |
| **NW** | 0 | 2 | 1 | 2 | 10 | 20 | 87 | 65 | 40 | 9 | 0 | 236 |
| **NNW** | 0 | 2 | 3 | 4 | 7 | 40 | 48 | 13 | 2 | 0 | 0 | 119 |
| **Total** | 1 | 17 | 54 | 152 | 171 | 308 | 431 | 377 | 346 | 128 | 38 | 2,023 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 308 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 2,023 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-41 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | | WD10 | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | F | | | **Delta Temperature** | | | Moderately Stable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 1 | 5 | 14 | 37 | 89 | 115 | 16 | 0 | 0 | 0 | 277 |
| **NNE** | 1 | 2 | 6 | 25 | 32 | 63 | 55 | 3 | 0 | 0 | 0 | 187 |
| **NE** | 0 | 0 | 7 | 16 | 14 | 25 | 8 | 1 | 0 | 0 | 0 | 71 |
| **ENE** | 0 | 0 | 5 | 15 | 10 | 6 | 0 | 1 | 0 | 0 | 0 | 37 |
| **E** | 0 | 1 | 6 | 13 | 13 | 4 | 0 | 0 | 0 | 0 | 0 | 37 |
| **ESE** | 0 | 2 | 4 | 8 | 8 | 3 | 2 | 1 | 0 | 0 | 0 | 28 |
| **SE** | 0 | 0 | 2 | 13 | 9 | 3 | 1 | 0 | 0 | 0 | 0 | 28 |
| **SSE** | 0 | 2 | 4 | 17 | 14 | 15 | 2 | 0 | 0 | 0 | 0 | 54 |
| **S** | 0 | 3 | 1 | 13 | 8 | 10 | 4 | 0 | 0 | 0 | 0 | 39 |
| **SSW** | 0 | 0 | 0 | 3 | 2 | 8 | 7 | 0 | 0 | 0 | 0 | 20 |
| **SW** | 0 | 0 | 2 | 2 | 2 | 1 | 5 | 7 | 4 | 0 | 0 | 23 |
| **WSW** | 1 | 0 | 0 | 2 | 1 | 8 | 6 | 15 | 3 | 0 | 0 | 36 |
| **W** | 0 | 0 | 1 | 1 | 1 | 4 | 10 | 11 | 6 | 0 | 0 | 34 |
| **WNW** | 0 | 0 | 2 | 2 | 0 | 3 | 25 | 11 | 2 | 0 | 0 | 45 |
| **NW** | 0 | 0 | 1 | 7 | 3 | 13 | 42 | 12 | 5 | 0 | 0 | 83 |
| **NNW** | 1 | 0 | 1 | 12 | 5 | 25 | 57 | 5 | 1 | 0 | 0 | 107 |
| **Total** | 3 | 11 | 47 | 163 | 159 | 280 | 339 | 83 | 21 | 0 | 0 | 1,106 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 308 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,106 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-42 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | | WD10 | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | G | | | **Delta Temperature** | | Extremely Stable | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 2 | 2 | 17 | 33 | 143 | 508 | 111 | 4 | 0 | 0 | 820 |
| **NNE** | 0 | 0 | 10 | 20 | 23 | 197 | 311 | 9 | 1 | 0 | 0 | 571 |
| **NE** | 0 | 0 | 6 | 7 | 6 | 20 | 12 | 0 | 0 | 0 | 0 | 51 |
| **ENE** | 0 | 0 | 3 | 9 | 5 | 5 | 0 | 0 | 0 | 0 | 0 | 22 |
| **E** | 0 | 1 | 1 | 9 | 7 | 4 | 0 | 0 | 0 | 0 | 0 | 22 |
| **ESE** | 0 | 1 | 1 | 3 | 3 | 1 | 0 | 0 | 0 | 0 | 0 | 9 |
| **SE** | 1 | 1 | 2 | 6 | 5 | 2 | 0 | 0 | 0 | 0 | 0 | 17 |
| **SSE** | 0 | 0 | 2 | 9 | 11 | 8 | 1 | 0 | 0 | 0 | 0 | 31 |
| **S** | 0 | 0 | 1 | 3 | 7 | 6 | 2 | 0 | 0 | 0 | 0 | 19 |
| **SSW** | 0 | 2 | 2 | 0 | 3 | 1 | 2 | 0 | 0 | 0 | 0 | 10 |
| **SW** | 0 | 0 | 0 | 0 | 2 | 2 | 2 | 0 | 0 | 0 | 0 | 6 |
| **WSW** | 0 | 0 | 0 | 1 | 2 | 1 | 0 | 1 | 0 | 0 | 0 | 5 |
| **W** | 0 | 0 | 1 | 1 | 3 | 1 | 2 | 0 | 0 | 0 | 0 | 8 |
| **WNW** | 0 | 0 | 0 | 2 | 2 | 3 | 4 | 1 | 0 | 0 | 0 | 12 |
| **NW** | 0 | 0 | 2 | 3 | 1 | 6 | 10 | 2 | 1 | 0 | 0 | 25 |
| **NNW** | 0 | 1 | 4 | 7 | 7 | 16 | 41 | 16 | 2 | 0 | 0 | 94 |
| **Total** | 1 | 8 | 37 | 97 | 120 | 416 | 895 | 140 | 8 | 0 | 0 | 1,722 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 308 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,722 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-43 Naughton Power Plant Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | | WD10 | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | ALL | | | **Delta Temperature** | | | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 4 | 15 | 50 | 105 | 322 | 716 | 138 | 11 | 0 | 0 | 1,361 |
| **NNE** | 1 | 4 | 24 | 65 | 84 | 324 | 414 | 30 | 4 | 0 | 0 | 950 |
| **NE** | 1 | 4 | 18 | 43 | 32 | 67 | 45 | 20 | 6 | 1 | 0 | 237 |
| **ENE** | 0 | 1 | 12 | 37 | 29 | 26 | 32 | 30 | 17 | 6 | 11 | 201 |
| **E** | 0 | 3 | 12 | 36 | 34 | 28 | 41 | 28 | 10 | 2 | 0 | 194 |
| **ESE** | 1 | 3 | 11 | 22 | 25 | 30 | 63 | 9 | 4 | 0 | 0 | 168 |
| **SE** | 3 | 4 | 15 | 52 | 41 | 45 | 41 | 3 | 0 | 0 | 0 | 204 |
| **SSE** | 0 | 5 | 15 | 66 | 69 | 71 | 41 | 13 | 1 | 1 | 0 | 282 |
| **S** | 2 | 4 | 8 | 54 | 66 | 88 | 60 | 15 | 8 | 0 | 0 | 305 |
| **SSW** | 1 | 2 | 10 | 19 | 23 | 50 | 41 | 22 | 17 | 6 | 0 | 191 |
| **SW** | 0 | 1 | 5 | 11 | 13 | 21 | 44 | 34 | 95 | 46 | 6 | 276 |
| **WSW** | 1 | 1 | 4 | 10 | 12 | 42 | 85 | 167 | 160 | 79 | 17 | 578 |
| **W** | 1 | 1 | 4 | 13 | 13 | 43 | 104 | 172 | 301 | 230 | 39 | 921 |
| **WNW** | 0 | 2 | 4 | 22 | 21 | 41 | 222 | 467 | 516 | 186 | 49 | 1,530 |
| **NW** | 0 | 2 | 5 | 17 | 22 | 65 | 208 | 176 | 127 | 33 | 6 | 661 |
| **NNW** | 1 | 3 | 10 | 26 | 28 | 100 | 167 | 45 | 12 | 0 | 1 | 393 |
| **Total** | 12 | 44 | 172 | 543 | 617 | 1,363 | 2,324 | 1,369 | 1,289 | 590 | 129 | 8,452 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 308 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 8,452 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-44 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS50 | | **Direction:** | WD50 | | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | A | | | **Delta Temperature** | | | Extremely Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 1 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **NE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 1 | 3 | 1 | 0 | 0 | 0 | 5 |
| **E** | 0 | 0 | 0 | 0 | 0 | 1 | 8 | 10 | 1 | 0 | 0 | 20 |
| **ESE** | 0 | 0 | 0 | 0 | 0 | 2 | 30 | 6 | 2 | 0 | 0 | 40 |
| **SE** | 0 | 0 | 0 | 0 | 0 | 3 | 6 | 3 | 0 | 0 | 0 | 12 |
| **SSE** | 0 | 0 | 0 | 0 | 0 | 4 | 10 | 3 | 1 | 0 | 1 | 19 |
| **S** | 0 | 0 | 0 | 0 | 1 | 4 | 8 | 9 | 0 | 1 | 5 | 28 |
| **SSW** | 0 | 0 | 0 | 0 | 0 | 1 | 6 | 5 | 3 | 3 | 4 | 22 |
| **SW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 23 | 11 | 13 | 48 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 10 | 5 | 5 | 20 |
| **W** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **NW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 1 | 2 | 5 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 0 | 1 | 4 |
| **Total** | 0 | 0 | 0 | 0 | 1 | 16 | 71 | 39 | 45 | 21 | 31 | 224 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 179 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 224 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-45 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS50 | | | **Direction:** WD50 | | | | **Lapse:** DT50_10 | | |
| **Stability Class:** | | | B | | | **Delta Temperature** | | | Moderately Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 0 | 1 | 0 | 0 | 3 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **NE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 0 | 0 | 1 |
| **E** | 0 | 0 | 0 | 0 | 1 | 1 | 2 | 3 | 0 | 0 | 0 | 7 |
| **ESE** | 0 | 0 | 0 | 0 | 0 | 1 | 7 | 0 | 0 | 0 | 0 | 8 |
| **SE** | 0 | 0 | 0 | 0 | 0 | 4 | 6 | 1 | 0 | 0 | 0 | 11 |
| **SSE** | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 0 | 0 | 0 | 1 | 3 |
| **S** | 0 | 0 | 0 | 0 | 0 | 4 | 5 | 1 | 0 | 1 | 0 | 11 |
| **SSW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 1 | 0 | 2 |
| **SW** | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 5 | 2 | 0 | 10 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 3 | 2 | 1 | 6 |
| **W** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 3 | 4 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 1 | 2 | 4 |
| **NW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 2 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 1 | 1 | 0 | 0 | 3 |
| **Total** | 0 | 0 | 0 | 0 | 1 | 14 | 23 | 9 | 11 | 8 | 9 | 75 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 179 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 75 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-46 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS50 | | **Direction:** | WD50 | | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | C | | | **Delta Temperature** | | | Slightly Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 2 | 2 | 0 | 1 | 0 | 0 | 5 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 4 | 1 | 0 | 0 | 1 | 0 | 6 |
| **NE** | 0 | 0 | 1 | 2 | 1 | 1 | 2 | 3 | 0 | 0 | 0 | 10 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 4 | 2 | 4 | 0 | 0 | 0 | 10 |
| **E** | 0 | 0 | 0 | 0 | 0 | 3 | 7 | 4 | 0 | 0 | 0 | 14 |
| **ESE** | 0 | 0 | 0 | 2 | 0 | 10 | 13 | 2 | 1 | 0 | 0 | 28 |
| **SE** | 0 | 0 | 0 | 1 | 5 | 11 | 7 | 0 | 0 | 0 | 0 | 24 |
| **SSE** | 0 | 0 | 0 | 0 | 5 | 11 | 6 | 3 | 0 | 0 | 0 | 25 |
| **S** | 0 | 0 | 0 | 0 | 4 | 9 | 7 | 5 | 1 | 1 | 1 | 28 |
| **SSW** | 0 | 0 | 1 | 0 | 2 | 7 | 5 | 1 | 4 | 0 | 1 | 21 |
| **SW** | 0 | 0 | 0 | 0 | 0 | 4 | 1 | 5 | 6 | 3 | 2 | 21 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 2 | 1 | 0 | 8 | 9 | 1 | 21 |
| **W** | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 2 | 4 | 22 | 4 | 34 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 1 | 1 | 3 | 18 | 24 | 21 | 70 |
| **NW** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 9 | 7 | 4 | 20 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 1 | 3 | 4 | 3 | 0 | 0 | 11 |
| **Total** | 0 | 0 | 2 | 5 | 17 | 70 | 60 | 36 | 55 | 67 | 34 | 348 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 179 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 348 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-47 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS50 | | **Direction:** | | WD50 | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | D | | | **Delta Temperature** | | Neutral | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 1 | 2 | 7 | 9 | 27 | 19 | 1 | 7 | 0 | 0 | 73 |
| **NNE** | 0 | 0 | 5 | 10 | 8 | 21 | 18 | 5 | 4 | 0 | 0 | 71 |
| **NE** | 0 | 0 | 2 | 9 | 5 | 10 | 9 | 9 | 8 | 0 | 0 | 52 |
| **ENE** | 0 | 0 | 3 | 3 | 7 | 5 | 13 | 10 | 11 | 1 | 11 | 65 |
| **E** | 0 | 0 | 1 | 6 | 3 | 9 | 10 | 7 | 5 | 1 | 0 | 42 |
| **ESE** | 0 | 0 | 1 | 8 | 4 | 15 | 13 | 5 | 1 | 0 | 0 | 47 |
| **SE** | 0 | 2 | 1 | 11 | 8 | 7 | 17 | 6 | 1 | 0 | 0 | 53 |
| **SSE** | 0 | 2 | 2 | 18 | 20 | 14 | 13 | 3 | 0 | 0 | 0 | 72 |
| **S** | 0 | 1 | 2 | 18 | 22 | 32 | 23 | 6 | 4 | 2 | 0 | 110 |
| **SSW** | 0 | 0 | 2 | 7 | 6 | 9 | 6 | 8 | 5 | 2 | 0 | 45 |
| **SW** | 0 | 0 | 0 | 5 | 3 | 8 | 18 | 6 | 24 | 12 | 6 | 82 |
| **WSW** | 0 | 0 | 0 | 1 | 8 | 9 | 30 | 54 | 69 | 54 | 13 | 238 |
| **W** | 0 | 0 | 0 | 6 | 7 | 21 | 47 | 87 | 181 | 185 | 61 | 595 |
| **WNW** | 0 | 0 | 1 | 4 | 5 | 14 | 102 | 272 | 408 | 191 | 32 | 1,030 |
| **NW** | 0 | 1 | 1 | 5 | 10 | 19 | 71 | 103 | 126 | 40 | 11 | 387 |
| **NNW** | 0 | 0 | 4 | 4 | 4 | 14 | 20 | 4 | 9 | 2 | 0 | 61 |
| **Total** | 0 | 7 | 27 | 122 | 129 | 234 | 429 | 586 | 863 | 490 | 134 | 3,023 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 179 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 3,023 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-48 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS50 | | **Direction:** | | WD50 | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | E | | | **Delta Temperature** | | Slightly Stable | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 2 | 4 | 10 | 11 | 23 | 50 | 23 | 4 | 0 | 0 | 127 |
| **NNE** | 1 | 2 | 4 | 14 | 9 | 26 | 25 | 8 | 4 | 0 | 0 | 93 |
| **NE** | 1 | 1 | 4 | 11 | 11 | 18 | 8 | 6 | 5 | 1 | 0 | 66 |
| **ENE** | 0 | 1 | 2 | 14 | 6 | 11 | 7 | 11 | 3 | 1 | 1 | 60 |
| **E** | 1 | 0 | 8 | 8 | 4 | 2 | 5 | 4 | 2 | 3 | 0 | 37 |
| **ESE** | 0 | 1 | 0 | 3 | 5 | 8 | 6 | 2 | 0 | 0 | 0 | 25 |
| **SE** | 1 | 2 | 6 | 2 | 9 | 13 | 19 | 3 | 1 | 0 | 0 | 56 |
| **SSE** | 0 | 3 | 6 | 14 | 13 | 16 | 11 | 2 | 4 | 1 | 0 | 70 |
| **S** | 0 | 0 | 6 | 14 | 24 | 36 | 16 | 6 | 3 | 0 | 0 | 105 |
| **SSW** | 0 | 0 | 5 | 11 | 9 | 14 | 19 | 6 | 8 | 4 | 0 | 76 |
| **SW** | 0 | 1 | 3 | 2 | 2 | 8 | 10 | 13 | 19 | 24 | 19 | 101 |
| **WSW** | 0 | 0 | 3 | 2 | 5 | 3 | 19 | 37 | 65 | 27 | 13 | 176 |
| **W** | 0 | 0 | 0 | 6 | 5 | 14 | 30 | 46 | 100 | 38 | 11 | 250 |
| **WNW** | 0 | 1 | 1 | 4 | 6 | 25 | 58 | 108 | 147 | 75 | 28 | 455 |
| **NW** | 0 | 0 | 2 | 3 | 4 | 23 | 52 | 67 | 57 | 38 | 10 | 256 |
| **NNW** | 1 | 0 | 3 | 5 | 5 | 27 | 34 | 20 | 8 | 2 | 0 | 105 |
| **Total** | 5 | 14 | 57 | 123 | 128 | 267 | 369 | 362 | 430 | 214 | 82 | 2,058 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 179 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 2,058 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-49 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS50 | | **Direction:** | | WD50 | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | F | | | **Delta Temperature** | | Moderately Stable | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 1 | 2 | 8 | 6 | 41 | 80 | 22 | 5 | 0 | 0 | 165 |
| **NNE** | 1 | 2 | 6 | 17 | 23 | 36 | 48 | 3 | 1 | 0 | 0 | 137 |
| **NE** | 0 | 2 | 5 | 15 | 16 | 11 | 6 | 0 | 1 | 0 | 0 | 56 |
| **ENE** | 2 | 0 | 7 | 5 | 4 | 4 | 5 | 2 | 0 | 0 | 0 | 29 |
| **E** | 1 | 5 | 0 | 5 | 6 | 5 | 4 | 0 | 1 | 0 | 0 | 27 |
| **ESE** | 0 | 2 | 3 | 6 | 8 | 1 | 2 | 2 | 1 | 0 | 0 | 25 |
| **SE** | 0 | 3 | 3 | 17 | 8 | 7 | 2 | 0 | 0 | 0 | 0 | 40 |
| **SSE** | 0 | 3 | 6 | 8 | 7 | 9 | 5 | 2 | 1 | 0 | 0 | 41 |
| **S** | 0 | 2 | 6 | 12 | 10 | 20 | 18 | 4 | 0 | 0 | 0 | 72 |
| **SSW** | 0 | 3 | 8 | 12 | 7 | 11 | 10 | 4 | 1 | 0 | 0 | 56 |
| **SW** | 0 | 5 | 5 | 7 | 4 | 4 | 5 | 5 | 9 | 4 | 0 | 48 |
| **WSW** | 1 | 2 | 1 | 3 | 1 | 4 | 6 | 7 | 8 | 3 | 0 | 36 |
| **W** | 0 | 3 | 0 | 7 | 6 | 6 | 9 | 19 | 19 | 0 | 0 | 69 |
| **WNW** | 1 | 3 | 1 | 2 | 3 | 11 | 29 | 30 | 4 | 0 | 0 | 84 |
| **NW** | 0 | 2 | 0 | 4 | 6 | 23 | 40 | 33 | 8 | 4 | 0 | 120 |
| **NNW** | 0 | 1 | 2 | 6 | 9 | 22 | 62 | 24 | 3 | 1 | 0 | 130 |
| **Total** | 6 | 39 | 55 | 134 | 124 | 215 | 331 | 157 | 62 | 12 | 0 | 1135 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 179 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,135 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-50 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS50 | | **Direction:** | | WD50 | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | G | | | **Delta Temperature** | | Extremely Stable | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 1 | 0 | 4 | 13 | 15 | 82 | 239 | 167 | 26 | 0 | 0 | 547 |
| **NNE** | 0 | 1 | 9 | 16 | 28 | 136 | 202 | 84 | 13 | 0 | 0 | 489 |
| **NE** | 0 | 3 | 7 | 14 | 21 | 42 | 10 | 4 | 1 | 0 | 0 | 102 |
| **ENE** | 1 | 1 | 5 | 8 | 11 | 8 | 1 | 0 | 0 | 0 | 0 | 35 |
| **E** | 0 | 1 | 4 | 18 | 7 | 4 | 2 | 0 | 0 | 0 | 0 | 36 |
| **ESE** | 2 | 2 | 11 | 7 | 4 | 3 | 0 | 0 | 0 | 0 | 0 | 29 |
| **SE** | 0 | 2 | 4 | 3 | 6 | 4 | 1 | 0 | 0 | 0 | 0 | 20 |
| **SSE** | 0 | 5 | 5 | 10 | 4 | 8 | 1 | 0 | 0 | 0 | 0 | 33 |
| **S** | 0 | 1 | 2 | 6 | 6 | 13 | 7 | 4 | 1 | 0 | 0 | 40 |
| **SSW** | 0 | 1 | 1 | 7 | 1 | 4 | 2 | 1 | 1 | 0 | 0 | 18 |
| **SW** | 0 | 1 | 1 | 4 | 2 | 7 | 1 | 1 | 2 | 0 | 0 | 19 |
| **WSW** | 0 | 0 | 1 | 5 | 1 | 4 | 6 | 1 | 1 | 0 | 0 | 19 |
| **W** | 0 | 1 | 0 | 3 | 1 | 3 | 6 | 4 | 0 | 0 | 0 | 18 |
| **WNW** | 0 | 0 | 0 | 8 | 4 | 6 | 9 | 3 | 2 | 0 | 0 | 32 |
| **NW** | 0 | 0 | 0 | 7 | 9 | 22 | 31 | 22 | 3 | 0 | 0 | 94 |
| **NNW** | 0 | 0 | 2 | 12 | 14 | 33 | 97 | 28 | 1 | 0 | 0 | 187 |
| **Total** | 4 | 19 | 56 | 141 | 134 | 379 | 615 | 319 | 51 | 0 | 0 | 1,718 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 179 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,718 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-51 Naughton Power Plant Joint Frequency Distribution 2022–2023, 50-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS50 | | **Direction:** | WD50 | | | **Lapse:** | DT50_10 | |
| **Stability Class:** | | | ALL | | | **Delta Temperature** | | | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 1 | 4 | 12 | 38 | 41 | 177 | 390 | 213 | 45 | 0 | 0 | 921 |
| **NNE** | 2 | 5 | 24 | 57 | 68 | 223 | 294 | 100 | 22 | 1 | 0 | 796 |
| **NE** | 1 | 6 | 19 | 51 | 54 | 82 | 35 | 22 | 15 | 1 | 0 | 286 |
| **ENE** | 3 | 2 | 17 | 30 | 28 | 33 | 32 | 28 | 14 | 2 | 12 | 205 |
| **E** | 2 | 6 | 13 | 37 | 21 | 25 | 38 | 28 | 9 | 4 | 0 | 183 |
| **ESE** | 2 | 5 | 15 | 26 | 21 | 40 | 71 | 17 | 5 | 0 | 0 | 202 |
| **SE** | 1 | 9 | 14 | 34 | 36 | 49 | 58 | 13 | 2 | 0 | 0 | 216 |
| **SSE** | 0 | 13 | 19 | 50 | 49 | 64 | 46 | 13 | 6 | 1 | 2 | 263 |
| **S** | 0 | 4 | 16 | 50 | 67 | 118 | 84 | 35 | 9 | 5 | 6 | 394 |
| **SSW** | 0 | 4 | 17 | 37 | 25 | 46 | 48 | 26 | 22 | 10 | 5 | 240 |
| **SW** | 0 | 7 | 9 | 18 | 11 | 31 | 36 | 33 | 88 | 56 | 40 | 329 |
| **WSW** | 1 | 2 | 5 | 11 | 15 | 22 | 62 | 99 | 164 | 100 | 33 | 516 |
| **W** | 0 | 4 | 0 | 22 | 19 | 44 | 94 | 158 | 304 | 246 | 79 | 970 |
| **WNW** | 1 | 4 | 3 | 18 | 18 | 57 | 199 | 416 | 580 | 291 | 83 | 1,675 |
| **NW** | 0 | 3 | 3 | 19 | 29 | 87 | 194 | 225 | 205 | 90 | 29 | 884 |
| **NNW** | 1 | 1 | 11 | 27 | 32 | 97 | 217 | 82 | 27 | 5 | 1 | 501 |
| **Total** | 15 | 79 | 197 | 525 | 534 | 1,195 | 1,898 | 1,508 | 1,517 | 812 | 290 | 8,581 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 179 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 8,581 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-52 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS10 | | | **Direction:** WD10 | | | | **Lapse:** DT60_10 | | |
| **Stability Class:** | | | A | | | **Delta Temperature** | | | Extremely Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 1 | 0 | 1 | 1 | 0 | 0 | 0 | 0 | 3 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 1 |
| **NE** | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 1 | 2 | 0 | 0 | 5 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 0 | 7 | 5 | 2 | 0 | 0 | 14 |
| **E** | 0 | 0 | 0 | 0 | 0 | 3 | 16 | 5 | 1 | 0 | 0 | 25 |
| **ESE** | 0 | 0 | 0 | 0 | 0 | 5 | 32 | 7 | 2 | 0 | 0 | 46 |
| **SE** | 0 | 0 | 0 | 0 | 1 | 3 | 4 | 1 | 0 | 0 | 0 | 9 |
| **SSE** | 0 | 0 | 0 | 1 | 0 | 0 | 3 | 1 | 1 | 2 | 2 | 10 |
| **S** | 0 | 0 | 1 | 0 | 4 | 2 | 5 | 3 | 4 | 3 | 1 | 23 |
| **SSW** | 0 | 0 | 0 | 1 | 2 | 2 | 2 | 3 | 14 | 6 | 3 | 33 |
| **SW** | 0 | 0 | 0 | 0 | 0 | 1 | 5 | 9 | 30 | 16 | 1 | 62 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 2 | 1 | 19 | 50 | 12 | 2 | 86 |
| **W** | 0 | 0 | 0 | 0 | 0 | 2 | 11 | 32 | 95 | 32 | 2 | 174 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 4 | 22 | 89 | 90 | 27 | 2 | 234 |
| **NW** | 0 | 0 | 0 | 1 | 0 | 1 | 8 | 41 | 49 | 16 | 6 | 122 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 1 | 3 | 4 | 0 | 1 | 0 | 9 |
| **Total** | 0 | 0 | 1 | 4 | 7 | 27 | 122 | 220 | 341 | 115 | 19 | 856 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 566 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 856 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-53 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS10 | | | **Direction:** WD10 | | | | **Lapse:** DT60_10 | | |
| **Stability Class:** | | | B | | | **Delta Temperature** | | | Moderately Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 1 | 1 | 0 | 4 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 0 | 0 | 0 | 0 | 3 |
| **NE** | 0 | 0 | 0 | 0 | 0 | 4 | 3 | 1 | 0 | 0 | 0 | 8 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 3 | 8 | 1 | 0 | 0 | 0 | 12 |
| **E** | 0 | 0 | 0 | 0 | 1 | 7 | 6 | 1 | 0 | 0 | 0 | 15 |
| **ESE** | 0 | 0 | 0 | 0 | 1 | 6 | 8 | 2 | 0 | 0 | 0 | 17 |
| **SE** | 0 | 0 | 0 | 1 | 1 | 6 | 2 | 1 | 0 | 0 | 0 | 11 |
| **SSE** | 0 | 0 | 0 | 0 | 1 | 4 | 2 | 1 | 0 | 0 | 0 | 8 |
| **S** | 0 | 0 | 0 | 1 | 4 | 25 | 11 | 2 | 0 | 0 | 0 | 43 |
| **SSW** | 0 | 0 | 0 | 0 | 1 | 4 | 4 | 4 | 5 | 1 | 0 | 19 |
| **SW** | 0 | 0 | 0 | 0 | 1 | 2 | 2 | 3 | 6 | 2 | 1 | 17 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 2 | 8 | 10 | 12 | 1 | 0 | 33 |
| **W** | 0 | 0 | 0 | 0 | 0 | 4 | 15 | 9 | 15 | 2 | 1 | 46 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 1 | 14 | 28 | 10 | 1 | 0 | 54 |
| **NW** | 0 | 0 | 0 | 0 | 0 | 0 | 18 | 16 | 14 | 1 | 0 | 49 |
| **NNW** | 0 | 0 | 0 | 0 | 1 | 4 | 5 | 1 | 1 | 0 | 0 | 12 |
| **Total** | 0 | 0 | 0 | 2 | 11 | 73 | 110 | 80 | 64 | 9 | 2 | 351 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 566 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 351 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-54 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS10 | | | **Direction:** WD10 | | | | **Lapse:** DT60_10 | | |
| **Stability Class:** | | | C | | | **Delta Temperature** | | | Slightly Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 4 | 1 | 0 | 1 | 0 | 0 | 6 |
| **NNE** | 0 | 0 | 0 | 0 | 3 | 1 | 2 | 0 | 0 | 0 | 0 | 6 |
| **NE** | 0 | 0 | 0 | 0 | 2 | 2 | 0 | 0 | 1 | 0 | 0 | 5 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 5 | 2 | 0 | 0 | 0 | 0 | 7 |
| **E** | 0 | 0 | 0 | 1 | 2 | 5 | 2 | 1 | 1 | 0 | 0 | 12 |
| **ESE** | 0 | 0 | 0 | 0 | 2 | 4 | 4 | 1 | 0 | 0 | 0 | 11 |
| **SE** | 0 | 0 | 0 | 1 | 4 | 6 | 5 | 0 | 0 | 0 | 0 | 16 |
| **SSE** | 0 | 0 | 0 | 1 | 5 | 5 | 9 | 1 | 0 | 0 | 0 | 21 |
| **S** | 0 | 0 | 0 | 5 | 2 | 12 | 11 | 1 | 2 | 0 | 0 | 33 |
| **SSW** | 0 | 0 | 0 | 1 | 1 | 8 | 1 | 3 | 1 | 0 | 0 | 15 |
| **SW** | 0 | 0 | 0 | 0 | 1 | 3 | 4 | 6 | 7 | 3 | 0 | 24 |
| **WSW** | 0 | 0 | 0 | 3 | 1 | 6 | 3 | 10 | 7 | 2 | 0 | 32 |
| **W** | 0 | 0 | 0 | 0 | 1 | 3 | 13 | 15 | 10 | 1 | 0 | 43 |
| **WNW** | 0 | 0 | 0 | 1 | 2 | 4 | 13 | 18 | 12 | 1 | 0 | 51 |
| **NW** | 0 | 0 | 0 | 1 | 0 | 8 | 19 | 18 | 12 | 4 | 0 | 62 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 11 | 7 | 1 | 0 | 0 | 0 | 19 |
| **Total** | 0 | 0 | 0 | 14 | 26 | 87 | 96 | 75 | 54 | 11 | 0 | 363 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 566 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 363 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-55 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | | WD10 | | **Lapse:** | DT60_10 | |
| **Stability Class:** | | | D | | | **Delta Temperature** | | Neutral | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 4 | 28 | 26 | 31 | 7 | 7 | 1 | 0 | 0 | 104 |
| **NNE** | 0 | 1 | 2 | 13 | 12 | 13 | 11 | 1 | 0 | 0 | 0 | 53 |
| **NE** | 0 | 0 | 0 | 8 | 5 | 4 | 21 | 9 | 11 | 0 | 1 | 59 |
| **ENE** | 0 | 3 | 2 | 7 | 2 | 8 | 17 | 21 | 11 | 4 | 13 | 88 |
| **E** | 0 | 0 | 2 | 3 | 5 | 6 | 9 | 11 | 7 | 1 | 0 | 44 |
| **ESE** | 1 | 1 | 1 | 4 | 5 | 3 | 15 | 6 | 6 | 0 | 0 | 42 |
| **SE** | 0 | 1 | 2 | 7 | 2 | 6 | 14 | 1 | 1 | 0 | 0 | 34 |
| **SSE** | 0 | 1 | 5 | 8 | 7 | 20 | 7 | 2 | 3 | 0 | 0 | 53 |
| **S** | 0 | 1 | 0 | 20 | 20 | 29 | 14 | 9 | 2 | 0 | 0 | 95 |
| **SSW** | 0 | 0 | 1 | 11 | 16 | 23 | 23 | 10 | 4 | 2 | 0 | 90 |
| **SW** | 0 | 1 | 1 | 8 | 6 | 12 | 29 | 31 | 35 | 16 | 2 | 141 |
| **WSW** | 0 | 0 | 4 | 3 | 4 | 14 | 32 | 54 | 52 | 16 | 0 | 179 |
| **W** | 0 | 1 | 2 | 7 | 6 | 17 | 73 | 79 | 70 | 5 | 3 | 263 |
| **WNW** | 0 | 0 | 1 | 8 | 8 | 19 | 83 | 85 | 78 | 16 | 0 | 298 |
| **NW** | 0 | 1 | 6 | 10 | 11 | 23 | 61 | 103 | 64 | 11 | 3 | 293 |
| **NNW** | 0 | 2 | 4 | 23 | 21 | 24 | 24 | 20 | 5 | 1 | 0 | 124 |
| **Total** | 1 | 13 | 37 | 168 | 156 | 252 | 440 | 449 | 350 | 72 | 22 | 1,960 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 566 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,960 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-56 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | | WD10 | | **Lapse:** | DT60_10 | |
| **Stability Class:** | | | E | | | **Delta Temperature** | | Slightly Stable | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 4 | 10 | 19 | 27 | 47 | 36 | 2 | 1 | 0 | 0 | 146 |
| **NNE** | 0 | 3 | 7 | 14 | 22 | 32 | 25 | 3 | 0 | 0 | 0 | 106 |
| **NE** | 0 | 2 | 2 | 8 | 10 | 13 | 7 | 2 | 6 | 2 | 0 | 52 |
| **ENE** | 0 | 1 | 4 | 7 | 2 | 6 | 10 | 12 | 9 | 0 | 2 | 53 |
| **E** | 0 | 0 | 1 | 4 | 1 | 2 | 4 | 2 | 2 | 0 | 0 | 16 |
| **ESE** | 0 | 2 | 4 | 4 | 5 | 2 | 9 | 1 | 0 | 0 | 0 | 27 |
| **SE** | 0 | 2 | 4 | 5 | 10 | 11 | 7 | 5 | 2 | 0 | 0 | 46 |
| **SSE** | 1 | 2 | 8 | 15 | 17 | 20 | 16 | 3 | 2 | 0 | 0 | 84 |
| **S** | 0 | 2 | 4 | 31 | 32 | 53 | 34 | 10 | 0 | 0 | 0 | 166 |
| **SSW** | 0 | 0 | 3 | 18 | 23 | 44 | 22 | 13 | 2 | 1 | 1 | 127 |
| **SW** | 0 | 2 | 3 | 12 | 14 | 24 | 40 | 26 | 17 | 8 | 1 | 147 |
| **WSW** | 0 | 0 | 5 | 2 | 4 | 27 | 68 | 41 | 23 | 12 | 2 | 184 |
| **W** | 0 | 0 | 4 | 8 | 3 | 20 | 60 | 52 | 46 | 14 | 1 | 208 |
| **WNW** | 0 | 0 | 0 | 4 | 7 | 21 | 65 | 42 | 39 | 3 | 0 | 181 |
| **NW** | 0 | 3 | 4 | 12 | 11 | 33 | 51 | 31 | 22 | 1 | 0 | 168 |
| **NNW** | 0 | 2 | 7 | 15 | 24 | 41 | 40 | 17 | 2 | 0 | 0 | 148 |
| **Total** | 1 | 25 | 70 | 178 | 212 | 396 | 494 | 262 | 173 | 41 | 7 | 1,859 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 566 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,859 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-57 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS10 | | | **Direction:** WD10 | | | | **Lapse:** DT60_10 | | |
| **Stability Class:** | | | F | | | **Delta Temperature** | | | Moderately Stable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 4 | 20 | 26 | 31 | 57 | 25 | 0 | 0 | 0 | 0 | 163 |
| **NNE** | 0 | 2 | 12 | 34 | 23 | 30 | 21 | 4 | 0 | 0 | 0 | 126 |
| **NE** | 0 | 1 | 3 | 7 | 11 | 5 | 2 | 2 | 0 | 0 | 0 | 31 |
| **ENE** | 0 | 6 | 2 | 14 | 0 | 0 | 1 | 2 | 1 | 0 | 0 | 26 |
| **E** | 0 | 0 | 6 | 9 | 1 | 3 | 0 | 0 | 0 | 0 | 0 | 19 |
| **ESE** | 0 | 0 | 4 | 7 | 4 | 2 | 2 | 0 | 0 | 0 | 0 | 19 |
| **SE** | 0 | 0 | 4 | 15 | 6 | 5 | 2 | 1 | 0 | 0 | 0 | 33 |
| **SSE** | 0 | 4 | 5 | 13 | 13 | 13 | 3 | 0 | 0 | 0 | 0 | 51 |
| **S** | 0 | 1 | 9 | 31 | 18 | 31 | 17 | 2 | 0 | 0 | 0 | 109 |
| **SSW** | 0 | 0 | 4 | 16 | 18 | 17 | 15 | 1 | 1 | 0 | 0 | 72 |
| **SW** | 0 | 2 | 1 | 6 | 5 | 11 | 17 | 15 | 0 | 0 | 0 | 57 |
| **WSW** | 0 | 0 | 0 | 7 | 0 | 5 | 15 | 13 | 1 | 0 | 0 | 41 |
| **W** | 0 | 1 | 4 | 6 | 1 | 5 | 23 | 9 | 0 | 0 | 0 | 49 |
| **WNW** | 0 | 1 | 3 | 7 | 4 | 10 | 23 | 5 | 0 | 0 | 0 | 53 |
| **NW** | 0 | 3 | 9 | 9 | 20 | 24 | 11 | 4 | 1 | 0 | 0 | 81 |
| **NNW** | 0 | 3 | 10 | 37 | 29 | 42 | 27 | 4 | 1 | 0 | 0 | 153 |
| **Total** | 0 | 28 | 96 | 244 | 184 | 260 | 204 | 62 | 5 | 0 | 0 | 1,083 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 566 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,083 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-58 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS10 | | **Direction:** | | WD10 | | **Lapse:** | DT60_10 | |
| **Stability Class:** | | | G | | | **Delta Temperature** | | Extremely Stable | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 1 | 13 | 51 | 161 | 104 | 84 | 12 | 1 | 0 | 0 | 0 | 427 |
| **NNE** | 0 | 5 | 9 | 30 | 18 | 16 | 9 | 0 | 0 | 0 | 0 | 87 |
| **NE** | 0 | 9 | 15 | 23 | 8 | 2 | 1 | 0 | 0 | 0 | 0 | 58 |
| **ENE** | 0 | 2 | 8 | 6 | 1 | 3 | 0 | 0 | 0 | 0 | 0 | 20 |
| **E** | 0 | 2 | 9 | 12 | 1 | 1 | 0 | 0 | 0 | 0 | 0 | 25 |
| **ESE** | 0 | 2 | 9 | 9 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 23 |
| **SE** | 2 | 0 | 8 | 16 | 6 | 4 | 0 | 0 | 0 | 0 | 0 | 36 |
| **SSE** | 0 | 1 | 10 | 11 | 16 | 9 | 2 | 0 | 0 | 0 | 0 | 49 |
| **S** | 1 | 4 | 15 | 34 | 22 | 19 | 7 | 0 | 1 | 0 | 0 | 103 |
| **SSW** | 0 | 1 | 10 | 12 | 8 | 9 | 2 | 0 | 0 | 0 | 0 | 42 |
| **SW** | 0 | 2 | 4 | 4 | 6 | 5 | 4 | 0 | 0 | 0 | 0 | 25 |
| **WSW** | 0 | 5 | 3 | 4 | 3 | 1 | 5 | 0 | 0 | 0 | 0 | 21 |
| **W** | 0 | 4 | 5 | 8 | 1 | 2 | 2 | 1 | 0 | 0 | 0 | 23 |
| **WNW** | 0 | 6 | 8 | 11 | 6 | 5 | 2 | 1 | 0 | 0 | 0 | 39 |
| **NW** | 1 | 7 | 15 | 52 | 27 | 12 | 4 | 1 | 0 | 0 | 0 | 119 |
| **NNW** | 0 | 13 | 62 | 219 | 168 | 142 | 20 | 1 | 0 | 0 | 0 | 625 |
| **Total** | 5 | 76 | 241 | 612 | 398 | 314 | 70 | 5 | 1 | 0 | 0 | 1,722 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 566 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,722 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-59 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 10-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | **Speed:** | WS10 | | **Direction:** | WD10 | | **Lapse:** | DT60_10 | | | |
| **Stability Class:** | | ALL | | | **Delta Temperature** | | | | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 1 | 21 | 85 | 235 | 188 | 224 | 84 | 10 | 4 | 1 | 0 | 853 |
| **NNE** | 0 | 11 | 30 | 91 | 78 | 93 | 70 | 8 | 1 | 0 | 0 | 382 |
| **NE** | 0 | 12 | 20 | 46 | 36 | 30 | 36 | 15 | 20 | 2 | 1 | 218 |
| **ENE** | 0 | 12 | 16 | 34 | 5 | 25 | 45 | 41 | 23 | 4 | 15 | 220 |
| **E** | 0 | 2 | 18 | 29 | 11 | 27 | 37 | 20 | 11 | 1 | 0 | 156 |
| **ESE** | 1 | 5 | 18 | 24 | 20 | 22 | 70 | 17 | 8 | 0 | 0 | 185 |
| **SE** | 2 | 3 | 18 | 45 | 30 | 41 | 34 | 9 | 3 | 0 | 0 | 185 |
| **SSE** | 1 | 8 | 28 | 49 | 59 | 71 | 42 | 8 | 6 | 2 | 2 | 276 |
| **S** | 1 | 8 | 29 | 122 | 102 | 171 | 99 | 27 | 9 | 3 | 1 | 572 |
| **SSW** | 0 | 1 | 18 | 59 | 69 | 107 | 69 | 34 | 27 | 10 | 4 | 398 |
| **SW** | 0 | 7 | 9 | 30 | 33 | 58 | 101 | 90 | 95 | 45 | 5 | 473 |
| **WSW** | 0 | 5 | 12 | 19 | 12 | 57 | 132 | 147 | 145 | 43 | 4 | 576 |
| **W** | 0 | 6 | 15 | 29 | 12 | 53 | 197 | 197 | 236 | 54 | 7 | 806 |
| **WNW** | 0 | 7 | 12 | 31 | 27 | 64 | 222 | 268 | 229 | 48 | 2 | 910 |
| **NW** | 1 | 14 | 34 | 85 | 69 | 101 | 172 | 214 | 162 | 33 | 9 | 894 |
| **NNW** | 0 | 20 | 83 | 294 | 243 | 265 | 126 | 48 | 9 | 2 | 0 | 1,090 |
| **Total** | 7 | 142 | 445 | 1,222 | 994 | 1,409 | 1,536 | 1,153 | 988 | 248 | 50 | 8,194 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 566 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 8,194 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-60 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS60 | | | **Direction:** WD60 | | | | **Lapse:** DT60_10 | | |
| **Stability Class:** | | | A | | | **Delta Temperature** | | | Extremely Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 1 | 0 | 0 | 1 | 1 | 0 | 0 | 0 | 0 | 3 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 1 |
| **NE** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2 | 0 | 0 | 2 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 0 | 4 | 8 | 2 | 0 | 0 | 14 |
| **E** | 0 | 0 | 1 | 0 | 0 | 1 | 11 | 9 | 0 | 0 | 0 | 22 |
| **ESE** | 0 | 0 | 0 | 0 | 0 | 4 | 28 | 14 | 3 | 0 | 0 | 49 |
| **SE** | 0 | 0 | 0 | 0 | 0 | 3 | 7 | 1 | 0 | 0 | 0 | 11 |
| **SSE** | 0 | 0 | 0 | 0 | 0 | 2 | 4 | 1 | 1 | 0 | 4 | 12 |
| **S** | 0 | 0 | 0 | 1 | 2 | 0 | 3 | 4 | 3 | 2 | 2 | 17 |
| **SSW** | 0 | 0 | 0 | 0 | 3 | 1 | 2 | 3 | 9 | 5 | 6 | 29 |
| **SW** | 0 | 0 | 0 | 0 | 0 | 2 | 3 | 6 | 22 | 18 | 11 | 62 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 1 | 0 | 11 | 38 | 28 | 11 | 90 |
| **W** | 0 | 0 | 0 | 0 | 0 | 3 | 8 | 25 | 53 | 70 | 18 | 177 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 1 | 10 | 54 | 101 | 50 | 11 | 227 |
| **NW** | 0 | 0 | 0 | 0 | 1 | 1 | 7 | 32 | 52 | 22 | 13 | 129 |
| **NNW** | 0 | 0 | 0 | 0 | 0 | 1 | 3 | 3 | 2 | 1 | 1 | 11 |
| **Total** | 0 | 0 | 2 | 1 | 6 | 21 | 91 | 171 | 289 | 196 | 77 | 856 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 570 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 856 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-61 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS60 | | | **Direction:** WD60 | | | | **Lapse:** DT60_10 | | |
| **Stability Class:** | | | B | | | **Delta Temperature** | | | Moderately Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 0 | 3 | 0 | 0 | 1 | 1 | 5 |
| **NNE** | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 0 | 0 | 0 | 0 | 3 |
| **NE** | 0 | 0 | 0 | 0 | 1 | 2 | 3 | 0 | 0 | 0 | 0 | 6 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 1 | 8 | 2 | 1 | 0 | 0 | 12 |
| **E** | 0 | 0 | 0 | 0 | 0 | 4 | 8 | 2 | 0 | 0 | 0 | 14 |
| **ESE** | 0 | 0 | 0 | 0 | 0 | 4 | 9 | 1 | 1 | 0 | 0 | 15 |
| **SE** | 0 | 0 | 0 | 0 | 2 | 5 | 4 | 0 | 0 | 0 | 0 | 11 |
| **SSE** | 0 | 0 | 0 | 1 | 0 | 3 | 0 | 1 | 1 | 0 | 0 | 6 |
| **S** | 0 | 0 | 0 | 1 | 1 | 20 | 15 | 2 | 0 | 0 | 0 | 39 |
| **SSW** | 0 | 0 | 0 | 0 | 0 | 9 | 8 | 2 | 2 | 1 | 0 | 22 |
| **SW** | 0 | 0 | 1 | 0 | 1 | 2 | 2 | 2 | 2 | 5 | 3 | 18 |
| **WSW** | 0 | 0 | 0 | 0 | 0 | 3 | 5 | 8 | 10 | 10 | 1 | 37 |
| **W** | 0 | 0 | 0 | 0 | 0 | 3 | 11 | 9 | 10 | 11 | 1 | 46 |
| **WNW** | 0 | 0 | 0 | 0 | 0 | 0 | 8 | 26 | 18 | 3 | 0 | 55 |
| **NW** | 0 | 0 | 0 | 0 | 0 | 1 | 9 | 17 | 20 | 4 | 0 | 51 |
| **NNW** | 0 | 0 | 0 | 0 | 1 | 2 | 4 | 2 | 2 | 0 | 0 | 11 |
| **Total** | 0 | 0 | 1 | 2 | 6 | 60 | 99 | 74 | 67 | 35 | 6 | 351 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 570 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 351 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-62 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS60 | | | **Direction:** WD60 | | | | **Lapse:** DT60_10 | | |
| **Stability Class:** | | | C | | | **Delta Temperature** | | | Slightly Unstable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 0 | 0 | 0 | 0 | 1 | 2 | 0 | 1 | 0 | 0 | 4 |
| **NNE** | 0 | 0 | 0 | 0 | 2 | 2 | 1 | 1 | 0 | 0 | 0 | 6 |
| **NE** | 0 | 0 | 0 | 0 | 0 | 4 | 0 | 0 | 1 | 0 | 0 | 5 |
| **ENE** | 0 | 0 | 0 | 0 | 0 | 2 | 2 | 0 | 0 | 0 | 0 | 4 |
| **E** | 0 | 0 | 0 | 1 | 2 | 3 | 3 | 2 | 1 | 0 | 0 | 12 |
| **ESE** | 0 | 0 | 0 | 1 | 0 | 6 | 3 | 1 | 0 | 0 | 0 | 11 |
| **SE** | 0 | 0 | 0 | 0 | 5 | 6 | 5 | 0 | 0 | 0 | 0 | 16 |
| **SSE** | 0 | 0 | 0 | 2 | 1 | 7 | 8 | 2 | 0 | 0 | 0 | 20 |
| **S** | 0 | 0 | 0 | 0 | 0 | 11 | 15 | 2 | 1 | 1 | 0 | 30 |
| **SSW** | 0 | 0 | 0 | 5 | 0 | 5 | 4 | 4 | 2 | 0 | 0 | 20 |
| **SW** | 0 | 0 | 0 | 2 | 2 | 4 | 3 | 4 | 2 | 5 | 1 | 23 |
| **WSW** | 0 | 0 | 0 | 1 | 0 | 4 | 4 | 6 | 10 | 4 | 1 | 30 |
| **W** | 0 | 0 | 0 | 0 | 2 | 2 | 10 | 11 | 12 | 6 | 4 | 47 |
| **WNW** | 0 | 0 | 0 | 0 | 1 | 1 | 8 | 16 | 15 | 3 | 1 | 45 |
| **NW** | 0 | 0 | 0 | 0 | 1 | 6 | 19 | 15 | 16 | 7 | 2 | 66 |
| **NNW** | 0 | 0 | 0 | 0 | 1 | 5 | 13 | 3 | 2 | 0 | 0 | 24 |
| **Total** | 0 | 0 | 0 | 12 | 17 | 69 | 100 | 67 | 63 | 26 | 9 | 363 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 570 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 363 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-63 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS60 | | **Direction:** | | WD60 | | **Lapse:** | DT60_10 | |
| **Stability Class:** | | | D | | | **Delta Temperature** | | Neutral | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 1 | 7 | 27 | 21 | 40 | 12 | 2 | 6 | 1 | 0 | 117 |
| **NNE** | 0 | 1 | 3 | 17 | 10 | 11 | 5 | 3 | 0 | 0 | 0 | 50 |
| **NE** | 1 | 0 | 2 | 3 | 4 | 6 | 13 | 15 | 7 | 6 | 1 | 58 |
| **ENE** | 0 | 0 | 0 | 10 | 5 | 9 | 15 | 20 | 15 | 10 | 12 | 97 |
| **E** | 0 | 0 | 0 | 1 | 4 | 6 | 9 | 13 | 8 | 3 | 0 | 44 |
| **ESE** | 0 | 1 | 1 | 5 | 5 | 3 | 14 | 2 | 10 | 3 | 0 | 44 |
| **SE** | 1 | 1 | 2 | 0 | 2 | 5 | 13 | 3 | 0 | 0 | 0 | 27 |
| **SSE** | 0 | 0 | 4 | 9 | 4 | 12 | 10 | 4 | 3 | 1 | 0 | 47 |
| **S** | 0 | 2 | 1 | 13 | 11 | 25 | 15 | 8 | 3 | 0 | 0 | 78 |
| **SSW** | 0 | 1 | 3 | 13 | 8 | 20 | 17 | 12 | 6 | 0 | 2 | 82 |
| **SW** | 0 | 3 | 3 | 11 | 6 | 10 | 16 | 27 | 25 | 30 | 15 | 146 |
| **WSW** | 0 | 1 | 0 | 7 | 7 | 11 | 25 | 36 | 58 | 31 | 14 | 190 |
| **W** | 0 | 1 | 4 | 2 | 4 | 5 | 28 | 67 | 80 | 60 | 8 | 260 |
| **WNW** | 0 | 1 | 0 | 2 | 5 | 7 | 58 | 102 | 92 | 48 | 16 | 331 |
| **NW** | 0 | 1 | 3 | 12 | 8 | 22 | 37 | 71 | 102 | 35 | 12 | 304 |
| **NNW** | 0 | 0 | 5 | 9 | 9 | 14 | 18 | 13 | 14 | 3 | 0 | 85 |
| **Total** | 2 | 14 | 38 | 141 | 113 | 206 | 305 | 398 | 429 | 231 | 80 | 1,960 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 570 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,960 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-64 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS60 | | **Direction:** | | WD60 | | **Lapse:** | DT60_10 | |
| **Stability Class:** | | | E | | | **Delta Temperature** | | Slightly Stable | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 1 | 4 | 3 | 17 | 16 | 39 | 57 | 16 | 1 | 0 | 0 | 154 |
| **NNE** | 1 | 1 | 8 | 10 | 13 | 25 | 24 | 11 | 3 | 0 | 0 | 96 |
| **NE** | 1 | 3 | 4 | 6 | 2 | 8 | 16 | 4 | 1 | 3 | 0 | 48 |
| **ENE** | 1 | 5 | 4 | 4 | 4 | 5 | 14 | 9 | 14 | 6 | 2 | 68 |
| **E** | 0 | 1 | 3 | 6 | 3 | 4 | 11 | 3 | 4 | 1 | 0 | 36 |
| **ESE** | 0 | 0 | 3 | 4 | 1 | 3 | 7 | 7 | 3 | 0 | 0 | 28 |
| **SE** | 0 | 1 | 2 | 6 | 4 | 6 | 5 | 2 | 5 | 0 | 0 | 31 |
| **SSE** | 0 | 1 | 1 | 11 | 5 | 9 | 5 | 4 | 2 | 3 | 0 | 41 |
| **S** | 0 | 3 | 6 | 22 | 28 | 31 | 27 | 10 | 5 | 0 | 0 | 132 |
| **SSW** | 0 | 1 | 3 | 10 | 23 | 22 | 30 | 12 | 14 | 2 | 0 | 117 |
| **SW** | 0 | 2 | 2 | 8 | 3 | 10 | 34 | 26 | 32 | 14 | 10 | 142 |
| **WSW** | 0 | 5 | 2 | 3 | 3 | 13 | 30 | 50 | 53 | 17 | 11 | 189 |
| **W** | 0 | 1 | 0 | 6 | 9 | 16 | 33 | 44 | 61 | 43 | 24 | 237 |
| **WNW** | 0 | 2 | 2 | 5 | 4 | 13 | 59 | 62 | 61 | 27 | 5 | 240 |
| **NW** | 0 | 1 | 1 | 1 | 4 | 14 | 58 | 52 | 43 | 21 | 4 | 199 |
| **NNW** | 0 | 0 | 3 | 11 | 6 | 31 | 17 | 19 | 10 | 3 | 0 | 100 |
| **Total** | 4 | 31 | 47 | 130 | 128 | 249 | 427 | 331 | 312 | 140 | 56 | 1,858 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 570 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,858 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-65 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS60 | | | **Direction:** WD60 | | | | **Lapse:** DT60_10 | | |
| **Stability Class:** | | | F | | | **Delta Temperature** | | | Moderately Stable | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 2 | 3 | 16 | 21 | 65 | 75 | 23 | 1 | 0 | 0 | 206 |
| **NNE** | 0 | 4 | 4 | 12 | 18 | 33 | 33 | 4 | 1 | 0 | 0 | 109 |
| **NE** | 0 | 3 | 3 | 6 | 9 | 6 | 5 | 2 | 3 | 0 | 0 | 37 |
| **ENE** | 0 | 0 | 3 | 8 | 5 | 4 | 2 | 1 | 2 | 0 | 0 | 25 |
| **E** | 0 | 2 | 4 | 2 | 2 | 3 | 2 | 0 | 0 | 0 | 0 | 15 |
| **ESE** | 0 | 4 | 0 | 5 | 0 | 4 | 1 | 1 | 0 | 0 | 0 | 15 |
| **SE** | 0 | 0 | 4 | 4 | 3 | 2 | 3 | 0 | 1 | 0 | 0 | 17 |
| **SSE** | 0 | 1 | 2 | 8 | 7 | 7 | 1 | 0 | 1 | 0 | 0 | 27 |
| **S** | 0 | 1 | 9 | 10 | 20 | 31 | 17 | 3 | 1 | 0 | 0 | 92 |
| **SSW** | 0 | 3 | 4 | 13 | 7 | 10 | 12 | 5 | 6 | 1 | 0 | 61 |
| **SW** | 1 | 2 | 3 | 9 | 4 | 7 | 8 | 4 | 15 | 0 | 0 | 53 |
| **WSW** | 0 | 1 | 4 | 7 | 8 | 7 | 12 | 21 | 16 | 5 | 0 | 81 |
| **W** | 0 | 2 | 1 | 3 | 4 | 7 | 10 | 19 | 14 | 4 | 0 | 64 |
| **WNW** | 0 | 1 | 2 | 2 | 9 | 8 | 22 | 21 | 6 | 0 | 0 | 71 |
| **NW** | 0 | 0 | 4 | 4 | 10 | 15 | 40 | 18 | 11 | 4 | 0 | 106 |
| **NNW** | 0 | 3 | 6 | 5 | 8 | 28 | 36 | 10 | 2 | 0 | 0 | 98 |
| **Total** | 1 | 29 | 56 | 114 | 135 | 237 | 279 | 132 | 80 | 14 | 0 | 1,077 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 570 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,077 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-66 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual**

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** WS60 | | | **Direction:** WD60 | | | **Lapse:** DT60_10 | | | |
| **Stability Class:** | | | G | | | **Delta Temperature** | | Extremely Stable | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 0 | 1 | 16 | 45 | 62 | 152 | 188 | 27 | 0 | 0 | 0 | 491 |
| **NNE** | 1 | 6 | 11 | 59 | 55 | 152 | 133 | 10 | 0 | 0 | 0 | 427 |
| **NE** | 2 | 5 | 12 | 26 | 24 | 14 | 2 | 0 | 0 | 0 | 0 | 85 |
| **ENE** | 1 | 4 | 6 | 7 | 5 | 7 | 3 | 0 | 0 | 0 | 0 | 33 |
| **E** | 0 | 7 | 8 | 9 | 6 | 5 | 2 | 0 | 0 | 0 | 0 | 37 |
| **ESE** | 1 | 3 | 10 | 9 | 4 | 6 | 0 | 0 | 0 | 0 | 0 | 33 |
| **SE** | 1 | 5 | 13 | 7 | 4 | 0 | 1 | 0 | 0 | 0 | 0 | 31 |
| **SSE** | 2 | 2 | 8 | 21 | 9 | 7 | 2 | 1 | 1 | 0 | 0 | 53 |
| **S** | 0 | 2 | 11 | 28 | 29 | 27 | 20 | 6 | 0 | 0 | 0 | 123 |
| **SSW** | 2 | 6 | 8 | 14 | 13 | 14 | 9 | 5 | 0 | 1 | 0 | 72 |
| **SW** | 0 | 6 | 7 | 11 | 8 | 9 | 8 | 2 | 3 | 0 | 0 | 54 |
| **WSW** | 0 | 6 | 4 | 3 | 8 | 9 | 7 | 8 | 3 | 0 | 0 | 48 |
| **W** | 0 | 2 | 6 | 10 | 7 | 3 | 3 | 3 | 3 | 0 | 0 | 37 |
| **WNW** | 0 | 6 | 4 | 5 | 4 | 5 | 8 | 9 | 1 | 0 | 0 | 42 |
| **NW** | 3 | 6 | 7 | 10 | 9 | 9 | 11 | 5 | 4 | 0 | 0 | 64 |
| **NNW** | 0 | 7 | 12 | 26 | 10 | 19 | 19 | 1 | 1 | 0 | 0 | 95 |
| **Total** | 13 | 74 | 143 | 290 | 257 | 438 | 416 | 77 | 16 | 1 | 0 | 1,725 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 570 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 1,725 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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## Table 2.4-67 Kemmerer Unit 1 Joint Frequency Distribution 2022–2023, 60-Meter Annual

| Hours at Each Wind Speed and Direction | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Period of Record = 04/09/2022 01:00-04/09/2023 00:00 | | | | | | | | | | | | |
| **Elevation:** | | | **Speed:** | WS60 | | **Direction:** | | WD60 | | **Lapse:** | DT60_10 | |
| **Stability Class:** | | | ALL | | | **Delta Temperature** | | | | | | |
| Wind Speed (mph) | | | | | | | | | | | | |
| **Wind Direction (from)** | **0.22-0.5** | **0.51-0.75** | **0.76-1** | **1.10-1.5** | **1.60-2** | **2.10-3** | **3.10-5** | **5.10-7** | **7.10-10** | **10.10-13** | **>13.10** | **Total** |
| **N** | 1 | 8 | 30 | 105 | 120 | 298 | 338 | 68 | 9 | 2 | 1 | 980 |
| **NNE** | 2 | 12 | 26 | 98 | 98 | 224 | 198 | 29 | 5 | 0 | 0 | 692 |
| **NE** | 4 | 11 | 21 | 41 | 40 | 40 | 39 | 21 | 14 | 9 | 1 | 241 |
| **ENE** | 2 | 9 | 13 | 29 | 19 | 28 | 48 | 40 | 34 | 16 | 14 | 253 |
| **E** | 0 | 10 | 16 | 19 | 17 | 26 | 46 | 29 | 13 | 4 | 0 | 180 |
| **ESE** | 1 | 8 | 14 | 24 | 10 | 30 | 62 | 26 | 17 | 3 | 0 | 195 |
| **SE** | 2 | 7 | 21 | 17 | 20 | 27 | 38 | 6 | 6 | 0 | 0 | 144 |
| **SSE** | 2 | 4 | 15 | 52 | 26 | 47 | 30 | 13 | 9 | 4 | 4 | 206 |
| **S** | 0 | 8 | 27 | 75 | 91 | 145 | 112 | 35 | 13 | 3 | 2 | 511 |
| **SSW** | 2 | 11 | 18 | 55 | 54 | 81 | 82 | 43 | 39 | 10 | 8 | 403 |
| **SW** | 1 | 13 | 16 | 41 | 24 | 44 | 74 | 71 | 101 | 72 | 40 | 498 |
| **WSW** | 0 | 13 | 10 | 21 | 26 | 48 | 83 | 140 | 188 | 95 | 38 | 665 |
| **W** | 0 | 6 | 11 | 21 | 26 | 39 | 103 | 178 | 233 | 194 | 55 | 868 |
| **WNW** | 0 | 10 | 8 | 14 | 23 | 35 | 173 | 290 | 294 | 131 | 33 | 1,011 |
| **NW** | 3 | 8 | 15 | 27 | 33 | 68 | 181 | 210 | 248 | 93 | 31 | 919 |
| **NNW** | 0 | 10 | 26 | 51 | 35 | 100 | 110 | 51 | 33 | 7 | 1 | 424 |
| **Total** | 20 | 148 | 287 | 690 | 662 | 1,280 | 1,717 | 1,250 | 1,256 | 643 | 228 | 8,190 |
| **Calm Hours not Included above for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Variable Direction Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 0 |
| **Invalid Hours for:** | | | | | | **Total Period** | | | | **All Hours** | | 570 |
| **Number of Valid Hours for this Table:** | | | | | | **Total Period** | | | | **All Hours** | | 8,190 |
| **Total Hours for the Period:** | | | | | | **Total Period** | | | | **All Hours** | | 8,760 |


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**Table 2.4-68 Kemmerer Unit 1 and Naughton Power Plant Stability Class Distribution for April 9, 2022–April 8, 2023, and January 1, 2019–December 31, 2021 (Naughton Power Plant Only)**

| | Kemmerer Unit 1 4/9/2022–4/8/2023 (%) | Naughton Power Plant 4/9/2023–4/8/2023 (%) | Naughton Power Plant 1/1/2019–12/31/2021 (%) |
|---|---|---|---|
| A | 10.5 | 2.5 | 2.0 |
| B | 4.3 | 0.9 | 1.9 |
| C | 4.4 | 4.1 | 4.4 |
| D | 23.9 | 35.1 | 38.5 |
| E | 22.7 | 23.9 | 20.6 |
| F | 13.2 | 13.1 | 12.6 |
| G | 21.0 | 20.4 | 20.0 |
| Based on Valid Hours | 8,194 | 8,452 | 25,867 |



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**Table 2.4-69 Naughton Power Plant 33-Foot (10-Meter) Annual Stability Persistence Summary for Jan 2019–Dec 2021**

| Stability Persistence (hours)/Percent | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Stability** | **1** | **2** | **3** | **4** | **5** | **6** | **7** | **8** | **9** | **10** | **11** | **12** | **13** | **14** | **15** | **16** | **17** | **18** | **19** | **20** | **21** | **22** | **23** | **24** | **GT.24** | **Total** |
| A | 107 | 43 | 25 | 13 | 14 | 4 | 10 | 3 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 220 |
| | 49 | 68 | 80 | 85 | 92 | 94 | 98 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| B | 286 | 72 | 14 | 5 | 2 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 379 |
| | 75 | 94 | 98 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| C | 536 | 141 | 39 | 26 | 11 | 2 | 1 | 1 | 0 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 758 |
| | 71 | 89 | 94 | 98 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| D | 598 | 348 | 222 | 157 | 160 | 124 | 95 | 83 | 66 | 67 | 47 | 57 | 29 | 13 | 15 | 6 | 7 | 6 | 9 | 5 | 3 | 2 | 0 | 0 | 16 | 2135 |
| | 28 | 44 | 55 | 62 | 70 | 75 | 80 | 84 | 87 | 90 | 92 | 95 | 96 | 97 | 97 | 98 | 98 | 98 | 99 | 99 | 99 | 99 | 99 | 99 | 100 | |
| E | 1160 | 461 | 266 | 146 | 79 | 56 | 50 | 26 | 15 | 7 | 4 | 3 | 5 | 6 | 1 | 3 | 3 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2292 |
| | 51 | 71 | 82 | 89 | 92 | 95 | 97 | 98 | 99 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| F | 914 | 386 | 190 | 101 | 53 | 28 | 14 | 5 | 5 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1696 |
| | 54 | 77 | 88 | 94 | 97 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| G | 252 | 132 | 82 | 71 | 57 | 46 | 53 | 54 | 57 | 60 | 52 | 31 | 11 | 16 | 9 | 9 | 1 | 1 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 995 |
| | 25 | 39 | 47 | 54 | 60 | 64 | 70 | 75 | 81 | 87 | 92 | 95 | 96 | 98 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| TOTAL | 3853 | 1583 | 838 | 519 | 376 | 260 | 223 | 172 | 144 | 134 | 104 | 91 | 45 | 35 | 25 | 18 | 11 | 8 | 10 | 5 | 3 | 2 | 0 | 0 | 16 | 8475 |


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**Table 2.4-70 Naughton Power Plant 164-Foot (50-Meter) Annual Stability Persistence Summary for Jan 2019–Dec 2021**

| Stability Persistence (hours)/Percent | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Stability** | **1** | **2** | **3** | **4** | **5** | **6** | **7** | **8** | **9** | **10** | **11** | **12** | **13** | **14** | **15** | **16** | **17** | **18** | **19** | **20** | **21** | **22** | **23** | **24** | **GT.24** | **Total** |
| A | 107 | 44 | 25 | 14 | 14 | 4 | 9 | 3 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 221 |
| | 48 | 68 | 80 | 86 | 92 | 94 | 98 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| B | 287 | 72 | 13 | 5 | 2 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 379 |
| | 76 | 95 | 98 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| C | 536 | 140 | 39 | 26 | 11 | 2 | 1 | 1 | 0 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 757 |
| | 71 | 89 | 94 | 98 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| D | 601 | 349 | 224 | 157 | 162 | 124 | 95 | 84 | 63 | 68 | 49 | 56 | 30 | 13 | 15 | 6 | 7 | 6 | 7 | 5 | 2 | 3 | 0 | 0 | 16 | 2142 |
| | 28 | 44 | 55 | 62 | 70 | 75 | 80 | 84 | 87 | 90 | 92 | 95 | 96 | 97 | 98 | 98 | 98 | 98 | 99 | 99 | 99 | 99 | 99 | 99 | 100 | |
| E | 1165 | 454 | 268 | 150 | 79 | 59 | 47 | 27 | 16 | 7 | 4 | 3 | 5 | 6 | 1 | 3 | 3 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 2298 |
| | 51 | 70 | 82 | 89 | 92 | 95 | 97 | 98 | 99 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| F | 915 | 386 | 190 | 104 | 55 | 27 | 14 | 4 | 5 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1700 |
| | 54 | 77 | 88 | 94 | 97 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| G | 253 | 133 | 88 | 68 | 56 | 44 | 52 | 57 | 53 | 59 | 47 | 30 | 14 | 17 | 11 | 12 | 0 | 1 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 996 |
| | 25 | 39 | 48 | 54 | 60 | 64 | 70 | 75 | 81 | 87 | 91 | 94 | 96 | 97 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| TOTAL | 3864 | 1578 | 847 | 524 | 379 | 260 | 218 | 176 | 138 | 134 | 101 | 89 | 49 | 36 | 27 | 21 | 10 | 8 | 8 | 5 | 2 | 3 | 0 | 0 | 16 | 8475 |


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**Table 2.4-71 Kemmerer Unit 1 33-Foot (10-Meter) Annual Stability Persistence Summary for April 2022–April 2023**

| Stability Persistence (hours)/Percent | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Stability** | **1** | **2** | **3** | **4** | **5** | **6** | **7** | **8** | **9** | **10** | **11** | **12** | **13** | **14** | **15** | **16** | **17** | **18** | **19** | **20** | **21** | **22** | **23** | **24** | **GT.24** | **Total** |
| A | 48 | 32 | 12 | 18 | 17 | 25 | 11 | 17 | 7 | 4 | 2 | 3 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 199 |
| | 24 | 40 | 46 | 55 | 64 | 76 | 82 | 90 | 94 | 96 | 97 | 98 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| B | 155 | 61 | 14 | 3 | 1 | 1 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 236 |
| | 66 | 92 | 97 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| C | 206 | 50 | 13 | 4 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 274 |
| | 75 | 93 | 98 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| D | 271 | 146 | 63 | 47 | 21 | 25 | 13 | 8 | 9 | 3 | 6 | 4 | 1 | 2 | 3 | 1 | 3 | 0 | 2 | 0 | 1 | 0 | 0 | 0 | 5 | 634 |
| | 43 | 66 | 76 | 83 | 86 | 90 | 92 | 94 | 95 | 96 | 97 | 97 | 97 | 98 | 98 | 98 | 99 | 99 | 99 | 99 | 99 | 99 | 99 | 99 | 100 | |
| E | 326 | 145 | 71 | 52 | 35 | 22 | 15 | 12 | 10 | 10 | 2 | 3 | 1 | 1 | 2 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 708 |
| | 46 | 66 | 77 | 84 | 89 | 92 | 94 | 96 | 97 | 98 | 98 | 99 | 99 | 99 | 99 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | |
| F | 331 | 138 | 70 | 30 | 15 | 9 | 5 | 1 | 1 | 2 | 1 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 604 |
| | 55 | 78 | 89 | 94 | 97 | 98 | 99 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| G | 78 | 42 | 25 | 26 | 25 | 14 | 16 | 19 | 20 | 18 | 10 | 6 | 5 | 4 | 3 | 2 | 1 | 4 | 0 | 1 | 1 | 1 | 0 | 0 | 0 | 321 |
| | 24 | 37 | 45 | 53 | 61 | 65 | 70 | 76 | 83 | 88 | 91 | 93 | 95 | 96 | 97 | 97 | 98 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | |
| TOTAL | 1415 | 614 | 268 | 180 | 114 | 97 | 61 | 57 | 47 | 37 | 21 | 17 | 10 | 7 | 8 | 3 | 4 | 4 | 2 | 2 | 2 | 1 | 0 | 0 | 5 | 2976 |


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**Table 2.4-72 Kemmerer Unit 1 197-Foot (60-Meter) Annual Stability Persistence Summary for April 2022–April 2023**

| Stability Persistence (hours)/Percent | | | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Stability** | **1** | **2** | **3** | **4** | **5** | **6** | **7** | **8** | **9** | **10** | **11** | **12** | **13** | **14** | **15** | **16** | **17** | **18** | **19** | **20** | **21** | **22** | **23** | **24** | **GT.24** | **Total** |
| A | 48 | 32 | 12 | 18 | 17 | 25 | 11 | 17 | 7 | 4 | 2 | 3 | 3 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 199 |
| | 24 | 40 | 46 | 55 | 64 | 76 | 82 | 90 | 94 | 96 | 97 | 98 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| B | 155 | 61 | 14 | 3 | 1 | 1 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 236 |
| | 66 | 92 | 97 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| C | 206 | 50 | 13 | 4 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 274 |
| | 75 | 93 | 98 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| D | 271 | 146 | 63 | 47 | 21 | 25 | 13 | 8 | 9 | 3 | 6 | 4 | 1 | 2 | 3 | 1 | 3 | 0 | 2 | 0 | 1 | 0 | 0 | 0 | 5 | 634 |
| | 43 | 66 | 76 | 83 | 86 | 90 | 92 | 94 | 95 | 96 | 97 | 97 | 97 | 98 | 98 | 98 | 99 | 99 | 99 | 99 | 99 | 99 | 99 | 99 | 100 | |
| E | 326 | 145 | 71 | 52 | 35 | 22 | 15 | 12 | 10 | 10 | 2 | 3 | 1 | 1 | 2 | 0 | 0 | 0 | 0 | 1 | 0 | 0 | 0 | 0 | 0 | 708 |
| | 46 | 66 | 77 | 84 | 89 | 92 | 94 | 96 | 97 | 98 | 98 | 99 | 99 | 99 | 99 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | |
| F | 331 | 138 | 70 | 30 | 15 | 9 | 5 | 1 | 1 | 2 | 1 | 1 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 604 |
| | 55 | 78 | 89 | 94 | 97 | 98 | 99 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| G | 78 | 42 | 25 | 26 | 25 | 14 | 16 | 19 | 20 | 18 | 10 | 6 | 5 | 4 | 3 | 2 | 1 | 4 | 0 | 1 | 1 | 1 | 0 | 0 | 0 | 321 |
| | 24 | 37 | 45 | 53 | 61 | 65 | 70 | 76 | 83 | 88 | 91 | 93 | 95 | 96 | 97 | 97 | 98 | 99 | 99 | 99 | 99 | 100 | 100 | 100 | 100 | |
| TOTAL | 1415 | 614 | 268 | 180 | 114 | 97 | 61 | 57 | 47 | 37 | 21 | 17 | 10 | 7 | 8 | 3 | 4 | 4 | 2 | 2 | 2 | 1 | 0 | 0 | 5 | 2976 |


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**Table 2.4-73 Temperature Inversion Frequency and Persistence, April 2022–April 2023**

| Number of Hours of Inversion Persistence | Number of Observations | Percent Probability |
|---|---|---|
| 1 | 57 | 12.31 |
| 2 | 57 | 24.62 |
| 3 | 26 | 30.24 |
| 4 | 38 | 38.44 |
| 5 | 21 | 42.98 |
| 6 | 15 | 46.22 |
| 7 | 19 | 50.32 |
| 8 | 7 | 51.84 |
| 9 | 12 | 54.43 |
| 10 | 17 | 58.10 |
| 11 | 33 | 65.23 |
| 12 | 52 | 76.46 |
| 13 | 19 | 80.56 |
| 14 | 28 | 86.61 |
| 15 | 8 | 88.34 |
| 16 | 18 | 92.22 |
| 17 | 7 | 93.74 |
| 18 | 7 | 95.25 |
| 19 | 6 | 96.54 |
| 20 | 5 | 97.62 |
| 21 | 2 | 98.06 |
| 22 | 3 | 98.70 |
| 23 | 0 | 98.70 |
| 24 | 0 | 98.70 |
| >24 | 6 | 100.00 |

[1] The longest inversion lasted 82 hours and started on 1/2/2023 at 17:00.


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**Table 2.4-74 Monthly Mean Temperatures (°F) at Kemmerer Unit 1 and Surrounding Sites**

| | Kemmerer Unit 1 | Naughton Power Plant | | Big Piney, WY | | Evanston-Uinta County Airport – Burns Field | Kemmerer, WY |
|---|---|---|---|---|---|---|---|
| | 4/9/22–4/8/23 | 4/9/22–4/8/23 | 2019–2021 | 4/9/22–4/8/23 | 1981–2010[1] | 1991–2020 | 1981–2010[1] |
| **Jan** | 14.3 | 15.2 | 19.7 | 5.7 | 11.0 | 21.7 | 15.1 |
| **Feb** | 12.0 | 13.8 | 18.8 | 3.9 | 14.6 | 23.1 | 16.8 |
| **Mar** | 16.6 | 17.1 | 27.2 | 9.9 | 27.2 | 31.4 | 26.5 |
| **Apr** | 31.3 | 31.4 | 37.7 | 30.2 | 37.1 | 39.1 | 36.3 |
| **May** | 45.6 | 44.5 | 46.3 | 43.0 | 45.8 | 48.7 | 45.6 |
| **Jun** | 58.6 | 57.5 | 58.0 | 54.5 | 54.9 | 58.1 | 54.2 |
| **Jul** | 69.1 | 68.6 | 66.2 | 65.3 | 61.9 | 67.5 | 61.1 |
| **Aug** | 65.5 | 65.1 | 64.4 | 63.8 | 59.6 | 65.6 | 59.7 |
| **Sep** | 57.7 | 58.1 | 54.4 | 55.8 | 49.8 | 56.3 | 50.2 |
| **Oct** | 40.7 | 43.1 | 39.1 | 40.9 | 38.2 | 43.9 | 39.0 |
| **Nov** | 23.1 | 23.0 | 31.8 | 19.7 | 23.8 | 31.1 | 25.7 |
| **Dec** | 18.0 | 18.3 | 21.0 | 10.2 | 12.6 | 21.8 | 15.4 |
| **Annual** | 37.5 | 38.1 | 40.4 | 33.6 | 36.5 | 42.4 | 37.2 |

[1] 1981-2010 Normals for Wyoming Station, Western and Central Wyoming Weather Forecast Office (Reference 2.4-11)


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**Table 2.4-75 Monthly Mean and Maximum Temperatures (°F) at Kemmerer Unit 1 and Surrounding Sites**

| | Kemmerer Unit 1 | Naughton Power Plant | | Big Piney, WY | | Evanston-Uinta County Airport – Burns Field | Kemmerer, WY |
|---|---|---|---|---|---|---|---|
| | 4/9/22–4/8/23 | 4/9/22–4/8/23 | 2019–2021 | 4/9/22–4/8/23 | 1981–2010[1] | 1991–2020 | 1981–2010[1] |
| **Jan** | 20.5 | 20.6 | 26.2 | 16.4 | 26.0 | 29.8 | 28.2 |
| **Feb** | 21.0 | 21.3 | 25.3 | 17.3 | 29.0 | 31.5 | 30.5 |
| **Mar** | 24.0 | 24.1 | 35.0 | 23.7 | 41.2 | 40.7 | 39.6 |
| **Apr** | 41.6 | 40.3 | 46.7 | 43.4 | 52.4 | 50.2 | 50.0 |
| **May** | 54.9 | 53.6 | 56.5 | 58.2 | 61.6 | 60.8 | 60.4 |
| **Jun** | 71.9 | 69.3 | 69.0 | 73.1 | 72.2 | 71.8 | 70.8 |
| **Jul** | 82.5 | 80.7 | 78.7 | 85.5 | 80.6 | 82.0 | 79.3 |
| **Aug** | 78.5 | 76.8 | 77.0 | 81.2 | 79.0 | 80.1 | 78.0 |
| **Sep** | 72.8 | 71.1 | 66.9 | 74.3 | 68.1 | 69.9 | 67.9 |
| **Oct** | 52.7 | 54.5 | 49.5 | 58.6 | 55.2 | 55.7 | 54.6 |
| **Nov** | 31.5 | 30.7 | 41.0 | 33.5 | 37.8 | 40.6 | 38.3 |
| **Dec** | 26.2 | 25.6 | 28.2 | 24.3 | 26.5 | 29.7 | 28.4 |
| **Annual** | 48.1 | 47.6 | 50.1 | 49.1 | 52.6 | 53.6 | 52.3 |

[1] 1981-2010 Normals for Wyoming Station, Western and Central Wyoming Weather Forecast Office (Reference 2.4-11)


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**Table 2.4-76 Monthly, Mean, and Minimum Temperatures (°F) at Kemmerer Unit 1 and Surrounding Sites**

| | Kemmerer Unit 1 | Naughton Power Plant | | Big Piney, WY | | Evanston-Uinta County Airport – Burns Field | Kemmerer, WY |
|---|---|---|---|---|---|---|---|
| | 4/9/22–4/8/23 | 4/9/22–4/8/23 | 2019–2021 | 4/9/22–4/8/23 | 1981–2010[1] | 1991–2020 | 1981–2010[1] |
| **Jan** | 7.1 | 9.2 | 12.1 | -4.9 | -4.0 | 13.6 | 2.0 |
| **Feb** | 3.6 | 6.4 | 10.9 | -9.5 | 0.3 | 14.8 | 3.0 |
| **Mar** | 8.2 | 9.5 | 18.8 | -3.9 | 13.2 | 22.0 | 13.4 |
| **Apr** | 21.2 | 22.0 | 27.4 | 16.5 | 21.9 | 28.0 | 22.5 |
| **May** | 32.3 | 33.8 | 34.9 | 27.8 | 30.0 | 36.6 | 30.8 |
| **Jun** | 42.0 | 42.9 | 43.9 | 35.9 | 37.6 | 44.4 | 37.6 |
| **Jul** | 52.0 | 52.8 | 50.5 | 45.0 | 43.3 | 52.9 | 42.9 |
| **Aug** | 51.8 | 52.5 | 48.8 | 46.4 | 40.3 | 51.1 | 41.3 |
| **Sep** | 41.5 | 43.8 | 39.7 | 37.2 | 31.4 | 42.8 | 32.6 |
| **Oct** | 28.0 | 31.1 | 28.1 | 23.1 | 21.3 | 32.1 | 23.4 |
| **Nov** | 14.9 | 15.6 | 22.5 | 5.9 | 9.9 | 21.6 | 13.0 |
| **Dec** | 8.7 | 9.9 | 12.9 | -3.9 | -1.3 | 13.8 | 2.4 |
| **Annual** | 26.0 | 27.6 | 29.3 | 18.0 | 20.4 | 31.1 | 22.2 |

[1] 1981-2010 Normals for Wyoming Station, Western and Central Wyoming Weather Forecast Office (Reference 2.4-11)


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**Table 2.4-77 National Ambient Air Quality Standards**
**(Sheet 1 of 2)**

| Pollutant | Primary/Secondary | Averaging Time | Level | Form |
|---|---|---|---|---|
| Carbon Monoxide (CO) | Primary | 8 Hours | 9 ppm | Not to be exceeded more than once per year |
| | | 1 Hour | 35 ppm | |
| Lead (Pb) | Primary and Secondary | Rolling 3 Month Average | 0.15 ($\mu g/m^3$) [1] | Not to be exceeded |
| Nitrogen Dioxide ($NO_2$) | Primary | 1 Hour | 100 ppb | 98$^{th}$ percentile of 1-hour daily maximum concentrations, averaged over 3 years |
| | Primary and Secondary | 1 Year | 53 ppb [2] | Annual mean |
| Ozone ($O_3$) | Primary and Secondary | 8 Hours | 0.070 ppm [3] | Annual fourth-highest daily maximum 8-hour concentration, averaged over 3 years |
| Particle Pollution (PM) | $PM_{2.5}$ | Primary | 1 Year | 12.0 $\mu g/m^3$ |
| | | Secondary | 1 Year | 15.0 $\mu g/m^3$ |
| | | Primary and Secondary | 24 Hours | 35 $\mu g/m^3$ |
| | $PM_{10}$ | Primary and Secondary | 24 Hours | 150 $\mu g/m^3$ |


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**Table 2.4-77 National Ambient Air Quality Standards**
**(Sheet 2 of 2)**

| Pollutant | Primary/Secondary | Averaging Time | Level | Form |
|---|---|---|---|---|
| Sulfur Dioxide ($SO_2$) | Primary | 1 Hour | 75 ppb [4] | 99th percentile of 1-hour daily maximum concentrations, averaged over 3 years |
| | Secondary | 3 Hours | 0.5 ppm | Not to be exceeded more than once per year |

[1] 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 (2008) standards have not been submitted and approved, the previous standards (1.5 $\mu g/m^3$ as a calendar quarter average) also remain in effect.

[2] 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 1-hour standard level.

[3] 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 1-hour (1979) and 8-hour (1997) $O_3$ standards.

[4] The previous $SO_2$ standards (0.14 ppm 24-hour and 0.03 ppm annual) will additionally remain in effect in certain areas: (1) any area for which it is not yet 1 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 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 State Implementation Plan to demonstrate attainment of the required NAAQS.


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**Table 2.4-78 ARCON 95th Percentile X/Q Values for the On-Site Control Room Evaluation for the Average Male**

| Release Point | 0-2 hours | 2-8 Hours | 8-24 Hours | 1-4 Days | 4-30 Days |
|---|---|---|---|---|---|
| **RXB** | 3.45E-03 | 3.01E-03 | 7.53E-04 | 9.41E-04 | 8.09E-04 |
| **RAB** | 3.97E-03 | 3.51E-03 | 9.02E-04 | 1.11E-03 | 9.38E-04 |
| **FHB** | 2.77E-03 | 2.28E-03 | 4.73E-04 | 6.36E-04 | 5.56E-04 |


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**Table 2.4-79 ARCON 99.5th Percentile X/Q Values at the EAB**

| Direction | 1 hour | 2 hours | 4 hours | 8hours | 12 hours | 24 hours | 96 hours | 168 hours | 360 hours | 720 hours |
|---|---|---|---|---|---|---|---|---|---|---|
| **East** | 1.15E-04 | 1.04E-04 | 9.10E-05 | 8.66E-05 | 7.07E-05 | 5.22E-05 | 2.60E-05 | 2.37E-05 | 1.78E-05 | 1.14E-05 |
| **East-Southeast** | 1.10E-04 | 9.82E-05 | 8.95E-05 | 8.05E-05 | 6.51E-05 | 4.74E-05 | 2.36E-05 | 2.08E-05 | 1.49E-05 | 9.65E-06 |
| **Southeast** | 1.09E-04 | 9.70E-05 | 8.87E-05 | 7.52E-05 | 5.94E-05 | 4.16E-05 | 2.05E-05 | 1.64E-05 | 1.18E-05 | 9.77E-06 |
| **South-Southeast** | 1.10E-04 | 9.89E-05 | 8.86E-05 | 7.50E-05 | 5.87E-05 | 3.77E-05 | 1.90E-05 | 1.55E-05 | 1.11E-05 | 9.07E-06 |
| **South** | 1.17E-04 | 1.09E-04 | 9.14E-05 | 7.78E-05 | 6.11E-05 | 4.34E-05 | 2.28E-05 | 1.73E-05 | 1.23E-05 | 1.05E-05 |
| **South-Southwest** | 1.27E-04 | 1.18E-04 | 1.07E-04 | 9.19E-05 | 7.50E-05 | 5.22E-05 | 3.12E-05 | 2.38E-05 | 1.74E-05 | 1.47E-05 |
| **Southwest** | 1.35E-04 | 1.31E-04 | 1.25E-04 | 1.17E-04 | 9.55E-05 | 7.08E-05 | 4.46E-05 | 3.90E-05 | 2.79E-05 | 2.21E-05 |
| **West-Southwest** | 1.46E-04 | 1.36E-04 | 1.29E-04 | 1.22E-04 | 1.03E-04 | 7.63E-05 | 5.61E-05 | 4.97E-05 | 3.90E-05 | 3.23E-05 |
| **West** | 1.53E-04 | 1.42E-04 | 1.31E-04 | 1.25E-04 | 1.05E-04 | 7.93E-05 | 5.89E-05 | 5.34E-05 | 4.58E-05 | 3.85E-05 |
| **West-Northwest** | 1.56E-04 | 1.44E-04 | 1.31E-04 | 1.23E-04 | 1.01E-04 | 7.64E-05 | 5.71E-05 | 5.20E-05 | 4.35E-05 | 3.91E-05 |
| **Northwest** | 1.70E-04 | 1.60E-04 | 1.42E-04 | 1.24E-04 | 9.93E-05 | 7.53E-05 | 5.39E-05 | 4.75E-05 | 3.96E-05 | 3.61E-05 |
| **North-Northwest** | 1.73E-04 | 1.69E-04 | 1.60E-04 | 1.49E-04 | 1.20E-04 | 8.71E-05 | 6.21E-05 | 5.38E-05 | 4.64E-05 | 4.29E-05 |
| **North** | 1.72E-04 | 1.69E-04 | 1.60E-04 | 1.52E-04 | 1.22E-04 | 8.40E-05 | 5.86E-05 | 5.09E-05 | 4.14E-05 | 3.80E-05 |
| **North-Northeast** | 1.72E-04 | 1.68E-04 | 1.58E-04 | 1.51E-04 | 1.20E-04 | 8.23E-05 | 5.47E-05 | 4.67E-05 | 3.78E-05 | 3.58E-05 |
| **Northeast** | 1.58E-04 | 1.54E-04 | 1.43E-04 | 1.31E-04 | 1.02E-04 | 6.33E-05 | 4.12E-05 | 3.45E-05 | 2.83E-05 | 2.58E-05 |
| **East-Northeast** | 1.31E-04 | 1.15E-04 | 1.01E-04 | 9.10E-05 | 7.47E-05 | 5.41E-05 | 2.81E-05 | 2.27E-05 | 1.65E-05 | 1.25E-05 |


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**Table 2.4-80 XOQDOQ Distances from RXB to Nearest Receptors Per Directional Sector**

| Direction | NI Release[1] | | | | | | EI Release[2] | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | EAB[4] | | TFF | | Nearest Residence[3,6] | | EAB | | TFF | | Nearest Residence[3,5] | |
| | *m* | *mi* | *m* | *mi* | *m* | *mi* | *m* | *mi* | *m* | *mi* | *m* | *mi* |
| **S** | 300 | 0.19 | | | 4,406 | 2.73 | 193 | 0.12 | | | 4,337 | 2.69 |
| **SSW** | 300 | 0.19 | | | | | 212 | 0.13 | | | | |
| **SW** | 300 | 0.19 | | | | | 250 | 0.16 | | | | |
| **WSW** | 300 | 0.19 | | | | | 308 | 0.19 | | | | |
| **W** | 300 | 0.19 | | | | | 384 | 0.24 | | | | |
| **WNW** | 300 | 0.19 | | | | | 466 | 0.29 | | | | |
| **NW** | 300 | 0.19 | | | | | 532 | 0.33 | | | | |
| **NNW** | 300 | 0.19 | | | | | 566 | 0.35 | | | | |
| **N** | 300 | 0.19 | 130 | 0.08 | | | 559 | 0.35 | 394 | 0.24 | | |
| **NNE** | 300 | 0.19 | | | | | 512 | 0.32 | | | | |
| **NE** | 300 | 0.19 | | | | | 438 | 0.27 | | | | |
| **ENE** | 300 | 0.19 | | | | | 357 | 0.22 | | | | |
| **E** | 300 | 0.19 | | | | | 286 | 0.18 | | | | |
| **ESE** | 300 | 0.19 | | | | | 234 | 0.15 | | | | |
| **SE** | 300 | 0.19 | | | | | 204 | 0.13 | | | | |
| **SSE** | 300 | 0.19 | | | | | 191 | 0.12 | | | | |

Notes:

1 The NI release point is a 100 m radius circle around the centerpoint of the reactor. REMP distances are not measured from this circle.

2 EI release is modeled at the cold salt storage tank.

3 The nearest dairy cow and vegetable garden are located at the nearest residence.

4 The EAB is modeled as a 400 m radius circle centered on the reactor.

5 The reactor centerpoint and the EI are 556 ft apart, this is subtracted from the distance to the nearest residence.

6 The nearest residence is 2.8 mi from the reactor centerpoint, so the 100 m is subtracted.


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## Table 2.4-81 XOQ and DOQ at Locations of Interest

| Location | From NI | | | | | | From EI | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Distance (mi) | Distance (m) | Undecayed Undepleted ($s/m^3$) | Decayed Undepleted ($s/m^3$) | Decayed Depleted ($s/m^3$) | Deposited ($m^{-2}$) | Distance (mi) | Distance (m) | Undecayed Undepleted ($s/m^3$) | Decayed Undepleted ($s/m^3$) | Decayed Depleted ($s/m^3$) | Deposited ($m^{-2}$) |
| EAB | 0.19 S | 300 S | 5.128E-05 | 5.126E-05 | 4.908E-05 | 1.513E-07 | 0.12 S | 193 S | 1.896E-04 | 1.895E-04 | 1.838E-04 | 2.841E-07 |
| Nearest Property | 2.74 S | 4406 S | 1.380E-06 | 1.370E-06 | 1.105E-06 | 2.740E-09 | 2.69 S | 4337 S | 1.505E-06 | 1.495E-06 | 1.207E-06 | 2.842E-09 |
| TFF | 0.08 N | 130 N | 2.755E-05 | 2.754E-05 | 2.692E-05 | 8.013E-08 | 0.24 N | 394 N | 6.092E-06 | 6.085E-06 | 5.769E-06 | 1.662E-08 |



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**Table 2.4-82 NI 50-mile X/Q and D/Q Values, No Decay/Undepleted**

| | Segment Boundary (miles) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| **Direction** | **.5-1** | **1-2** | **2-3** | **3-4** | **4-5** | **5-10** | **10-20** | **20-30** | **30-40** | **40-50** |
| **S** | 2.28E-05 | 5.36E-06 | 1.70E-06 | 8.81E-07 | 5.60E-07 | 2.59E-07 | 1.00E-07 | 5.23E-08 | 3.45E-08 | 2.53E-08 |
| **SSW** | 1.74E-05 | 4.12E-06 | 1.31E-06 | 6.79E-07 | 4.32E-07 | 2.00E-07 | 7.74E-08 | 4.05E-08 | 2.67E-08 | 1.97E-08 |
| **SW** | 4.99E-06 | 1.14E-06 | 3.55E-07 | 1.82E-07 | 1.14E-07 | 5.20E-08 | 1.96E-08 | 1.00E-08 | 6.54E-09 | 4.77E-09 |
| **WSW** | 3.14E-06 | 7.06E-07 | 2.15E-07 | 1.09E-07 | 6.81E-08 | 3.07E-08 | 1.13E-08 | 5.74E-09 | 3.71E-09 | 2.69E-09 |
| **W** | 2.92E-06 | 6.46E-07 | 1.94E-07 | 9.73E-08 | 6.04E-08 | 2.69E-08 | 9.78E-09 | 4.89E-09 | 3.14E-09 | 2.26E-09 |
| **WNW** | 2.97E-06 | 6.56E-07 | 1.97E-07 | 9.87E-08 | 6.13E-08 | 2.73E-08 | 9.90E-09 | 4.95E-09 | 3.17E-09 | 2.28E-09 |
| **NW** | 2.99E-06 | 6.62E-07 | 1.99E-07 | 9.99E-08 | 6.21E-08 | 2.77E-08 | 1.01E-08 | 5.03E-09 | 3.23E-09 | 2.33E-09 |
| **NNW** | 3.76E-06 | 8.32E-07 | 2.50E-07 | 1.26E-07 | 7.81E-08 | 3.49E-08 | 1.27E-08 | 6.38E-09 | 4.10E-09 | 2.95E-09 |
| **N** | 3.01E-06 | 6.56E-07 | 1.95E-07 | 9.71E-08 | 6.00E-08 | 2.65E-08 | 9.52E-09 | 4.72E-09 | 3.02E-09 | 2.17E-09 |
| **NNE** | 1.77E-06 | 3.86E-07 | 1.14E-07 | 5.67E-08 | 3.50E-08 | 1.54E-08 | 5.49E-09 | 2.71E-09 | 1.72E-09 | 1.23E-09 |
| **NE** | 1.60E-06 | 3.44E-07 | 1.01E-07 | 5.00E-08 | 3.07E-08 | 1.34E-08 | 4.73E-09 | 2.31E-09 | 1.46E-09 | 1.04E-09 |
| **ENE** | 2.88E-06 | 6.17E-07 | 1.79E-07 | 8.79E-08 | 5.36E-08 | 2.32E-08 | 7.97E-09 | 3.81E-09 | 2.37E-09 | 1.67E-09 |
| **E** | 4.71E-06 | 1.01E-06 | 2.93E-07 | 1.44E-07 | 8.75E-08 | 3.76E-08 | 1.29E-08 | 6.10E-09 | 3.78E-09 | 2.65E-09 |
| **ESE** | 7.06E-06 | 1.52E-06 | 4.41E-07 | 2.17E-07 | 1.32E-07 | 5.69E-08 | 1.96E-08 | 9.33E-09 | 5.80E-09 | 4.09E-09 |
| **SE** | 4.99E-06 | 1.10E-06 | 3.27E-07 | 1.64E-07 | 1.01E-07 | 4.47E-08 | 1.60E-08 | 7.89E-09 | 5.02E-09 | 3.59E-09 |
| **SSE** | 5.94E-06 | 1.35E-06 | 4.13E-07 | 2.10E-07 | 1.32E-07 | 5.95E-08 | 2.21E-08 | 1.13E-08 | 7.29E-09 | 5.29E-09 |


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**Table 2.4-83 NI 50-mile X/Q and D/Q Values, Decayed/Undepleted**

| | Segment Boundary (miles) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| **Direction** | **.5-1** | **1-2** | **2-3** | **3-4** | **4-5** | **5-10** | **10-20** | **20-30** | **30-40** | **40-50** |
| **S** | 2.27E-05 | 5.34E-06 | 1.69E-06 | 8.74E-07 | 5.54E-07 | 2.54E-07 | 9.65E-08 | 4.93E-08 | 3.18E-08 | 2.28E-08 |
| **SSW** | 1.74E-05 | 4.10E-06 | 1.30E-06 | 6.72E-07 | 4.26E-07 | 1.96E-07 | 7.40E-08 | 3.76E-08 | 2.41E-08 | 1.72E-08 |
| **SW** | 4.97E-06 | 1.14E-06 | 3.50E-07 | 1.78E-07 | 1.12E-07 | 5.02E-08 | 1.82E-08 | 8.92E-09 | 5.55E-09 | 3.87E-09 |
| **WSW** | 3.13E-06 | 7.01E-07 | 2.12E-07 | 1.07E-07 | 6.65E-08 | 2.95E-08 | 1.05E-08 | 5.03E-09 | 3.09E-09 | 2.13E-09 |
| **W** | 2.91E-06 | 6.41E-07 | 1.91E-07 | 9.55E-08 | 5.90E-08 | 2.59E-08 | 9.05E-09 | 4.30E-09 | 2.63E-09 | 1.80E-09 |
| **WNW** | 2.96E-06 | 6.51E-07 | 1.94E-07 | 9.69E-08 | 5.98E-08 | 2.62E-08 | 9.16E-09 | 4.34E-09 | 2.64E-09 | 1.81E-09 |
| **NW** | 2.98E-06 | 6.57E-07 | 1.96E-07 | 9.80E-08 | 6.06E-08 | 2.66E-08 | 9.31E-09 | 4.42E-09 | 2.69E-09 | 1.85E-09 |
| **NNW** | 3.74E-06 | 8.26E-07 | 2.47E-07 | 1.24E-07 | 7.64E-08 | 3.36E-08 | 1.18E-08 | 5.63E-09 | 3.45E-09 | 2.38E-09 |
| **N** | 3.00E-06 | 6.52E-07 | 1.93E-07 | 9.56E-08 | 5.88E-08 | 2.57E-08 | 8.93E-09 | 4.24E-09 | 2.60E-09 | 1.79E-09 |
| **NNE** | 1.77E-06 | 3.83E-07 | 1.13E-07 | 5.58E-08 | 3.42E-08 | 1.49E-08 | 5.13E-09 | 2.41E-09 | 1.47E-09 | 1.01E-09 |
| **NE** | 1.59E-06 | 3.43E-07 | 1.00E-07 | 4.94E-08 | 3.02E-08 | 1.31E-08 | 4.49E-09 | 2.12E-09 | 1.29E-09 | 8.94E-10 |
| **ENE** | 2.87E-06 | 6.15E-07 | 1.78E-07 | 8.73E-08 | 5.31E-08 | 2.28E-08 | 7.72E-09 | 3.61E-09 | 2.20E-09 | 1.52E-09 |
| **E** | 4.71E-06 | 1.01E-06 | 2.92E-07 | 1.43E-07 | 8.67E-08 | 3.71E-08 | 1.25E-08 | 5.77E-09 | 3.50E-09 | 2.41E-09 |
| **ESE** | 7.05E-06 | 1.51E-06 | 4.39E-07 | 2.15E-07 | 1.31E-07 | 5.61E-08 | 1.90E-08 | 8.87E-09 | 5.41E-09 | 3.74E-09 |
| **SE** | 4.98E-06 | 1.09E-06 | 3.25E-07 | 1.62E-07 | 9.98E-08 | 4.37E-08 | 1.53E-08 | 7.34E-09 | 4.54E-09 | 3.16E-09 |
| **SSE** | 5.92E-06 | 1.34E-06 | 4.09E-07 | 2.08E-07 | 1.30E-07 | 5.80E-08 | 2.10E-08 | 1.03E-08 | 6.47E-09 | 4.55E-09 |


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**Table 2.4-84 NI 50-mile X/Q and D/Q Values, Decayed/Depleted**

| | Segment Boundary (miles) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| **Direction** | **.5-1** | **1-2** | **2-3** | **3-4** | **4-5** | **5-10** | **10-20** | **20-30** | **30-40** | **40-50** |
| **S** | 2.04E-05 | 4.59E-06 | 1.38E-06 | 6.87E-07 | 4.23E-07 | 1.83E-07 | 6.22E-08 | 2.86E-08 | 1.70E-08 | 1.15E-08 |
| **SSW** | 1.56E-05 | 3.53E-06 | 1.06E-06 | 5.30E-07 | 3.26E-07 | 1.41E-07 | 4.80E-08 | 2.21E-08 | 1.31E-08 | 8.85E-09 |
| **SW** | 4.47E-06 | 9.79E-07 | 2.87E-07 | 1.41E-07 | 8.59E-08 | 3.67E-08 | 1.21E-08 | 5.40E-09 | 3.16E-09 | 2.10E-09 |
| **WSW** | 2.82E-06 | 6.05E-07 | 1.74E-07 | 8.48E-08 | 5.12E-08 | 2.16E-08 | 6.98E-09 | 3.07E-09 | 1.78E-09 | 1.17E-09 |
| **W** | 2.62E-06 | 5.53E-07 | 1.57E-07 | 7.57E-08 | 4.54E-08 | 1.90E-08 | 6.02E-09 | 2.62E-09 | 1.51E-09 | 9.90E-10 |
| **WNW** | 2.66E-06 | 5.61E-07 | 1.59E-07 | 7.68E-08 | 4.61E-08 | 1.92E-08 | 6.10E-09 | 2.65E-09 | 1.52E-09 | 9.98E-10 |
| **NW** | 2.68E-06 | 5.67E-07 | 1.61E-07 | 7.77E-08 | 4.67E-08 | 1.95E-08 | 6.19E-09 | 2.69E-09 | 1.55E-09 | 1.02E-09 |
| **NNW** | 3.37E-06 | 7.12E-07 | 2.03E-07 | 9.78E-08 | 5.88E-08 | 2.46E-08 | 7.84E-09 | 3.42E-09 | 1.97E-09 | 1.30E-09 |
| **N** | 2.70E-06 | 5.62E-07 | 1.58E-07 | 7.56E-08 | 4.51E-08 | 1.87E-08 | 5.89E-09 | 2.55E-09 | 1.46E-09 | 9.58E-10 |
| **NNE** | 1.59E-06 | 3.30E-07 | 9.24E-08 | 4.42E-08 | 2.63E-08 | 1.09E-08 | 3.39E-09 | 1.46E-09 | 8.32E-10 | 5.43E-10 |
| **NE** | 1.43E-06 | 2.95E-07 | 8.20E-08 | 3.90E-08 | 2.32E-08 | 9.50E-09 | 2.94E-09 | 1.25E-09 | 7.12E-10 | 4.64E-10 |
| **ENE** | 2.58E-06 | 5.29E-07 | 1.45E-07 | 6.86E-08 | 4.05E-08 | 1.64E-08 | 4.98E-09 | 2.09E-09 | 1.18E-09 | 7.61E-10 |
| **E** | 4.23E-06 | 8.67E-07 | 2.38E-07 | 1.12E-07 | 6.61E-08 | 2.67E-08 | 8.03E-09 | 3.34E-09 | 1.87E-09 | 1.21E-09 |
| **ESE** | 6.34E-06 | 1.30E-06 | 3.58E-07 | 1.69E-07 | 9.97E-08 | 4.04E-08 | 1.22E-08 | 5.12E-09 | 2.88E-09 | 1.86E-09 |
| **SE** | 4.48E-06 | 9.41E-07 | 2.66E-07 | 1.28E-07 | 7.63E-08 | 3.16E-08 | 9.95E-09 | 4.30E-09 | 2.47E-09 | 1.62E-09 |
| **SSE** | 5.32E-06 | 1.15E-06 | 3.35E-07 | 1.64E-07 | 9.92E-08 | 4.21E-08 | 1.37E-08 | 6.10E-09 | 3.56E-09 | 2.37E-09 |


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**Table 2.4-85 NI 50-mile X/Q and D/Q Values, Deposited**

| | Segment Boundary (miles) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| **Direction** | **.5-1** | **1-2** | **2-3** | **3-4** | **4-5** | **5-10** | **10-20** | **20-30** | **30-40** | **40-50** |
| **S** | 6.64E-08 | 1.36E-08 | 3.55E-09 | 1.60E-09 | 9.02E-10 | 3.47E-10 | 1.00E-10 | 3.98E-11 | 2.12E-11 | 1.32E-11 |
| **SSW** | 4.02E-08 | 8.23E-09 | 2.15E-09 | 9.65E-10 | 5.46E-10 | 2.10E-10 | 6.08E-11 | 2.41E-11 | 1.29E-11 | 7.96E-12 |
| **SW** | 1.10E-08 | 2.26E-09 | 5.89E-10 | 2.64E-10 | 1.50E-10 | 5.75E-11 | 1.66E-11 | 6.59E-12 | 3.52E-12 | 2.18E-12 |
| **WSW** | 9.30E-09 | 1.90E-09 | 4.97E-10 | 2.23E-10 | 1.26E-10 | 4.86E-11 | 1.41E-11 | 5.57E-12 | 2.97E-12 | 1.84E-12 |
| **W** | 1.03E-08 | 2.10E-09 | 5.49E-10 | 2.47E-10 | 1.40E-10 | 5.37E-11 | 1.55E-11 | 6.15E-12 | 3.29E-12 | 2.03E-12 |
| **WNW** | 8.62E-09 | 1.77E-09 | 4.61E-10 | 2.07E-10 | 1.17E-10 | 4.50E-11 | 1.30E-11 | 5.16E-12 | 2.76E-12 | 1.71E-12 |
| **NW** | 8.10E-09 | 1.66E-09 | 4.33E-10 | 1.95E-10 | 1.10E-10 | 4.23E-11 | 1.22E-11 | 4.85E-12 | 2.59E-12 | 1.60E-12 |
| **NNW** | 1.08E-08 | 2.21E-09 | 5.77E-10 | 2.59E-10 | 1.47E-10 | 5.64E-11 | 1.63E-11 | 6.47E-12 | 3.46E-12 | 2.14E-12 |
| **N** | 1.09E-08 | 2.24E-09 | 5.84E-10 | 2.63E-10 | 1.49E-10 | 5.71E-11 | 1.65E-11 | 6.55E-12 | 3.50E-12 | 2.16E-12 |
| **NNE** | 7.43E-09 | 1.52E-09 | 3.98E-10 | 1.79E-10 | 1.01E-10 | 3.88E-11 | 1.12E-11 | 4.45E-12 | 2.38E-12 | 1.47E-12 |
| **NE** | 1.12E-08 | 2.30E-09 | 6.01E-10 | 2.70E-10 | 1.53E-10 | 5.87E-11 | 1.70E-11 | 6.73E-12 | 3.60E-12 | 2.23E-12 |
| **ENE** | 2.81E-08 | 5.76E-09 | 1.50E-09 | 6.76E-10 | 3.82E-10 | 1.47E-10 | 4.25E-11 | 1.69E-11 | 9.00E-12 | 5.57E-12 |
| **E** | 5.15E-08 | 1.06E-08 | 2.75E-09 | 1.24E-09 | 7.00E-10 | 2.69E-10 | 7.79E-11 | 3.09E-11 | 1.65E-11 | 1.02E-11 |
| **ESE** | 7.05E-08 | 1.44E-08 | 3.77E-09 | 1.69E-09 | 9.58E-10 | 3.68E-10 | 1.07E-10 | 4.22E-11 | 2.26E-11 | 1.40E-11 |
| **SE** | 2.88E-08 | 5.91E-09 | 1.54E-09 | 6.93E-10 | 3.92E-10 | 1.51E-10 | 4.36E-11 | 1.73E-11 | 9.23E-12 | 5.71E-12 |
| **SSE** | 1.98E-08 | 4.05E-09 | 1.06E-09 | 4.75E-10 | 2.69E-10 | 1.03E-10 | 2.99E-11 | 1.19E-11 | 6.33E-12 | 3.92E-12 |


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**Table 2.4-86 El 50-mile X/Q and D/Q Values, No Decay/Undepleted**

| | Segment Boundary (miles) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| **Direction** | **.5-1** | **1-2** | **2-3** | **3-4** | **4-5** | **5-10** | **10-20** | **20-30** | **30-40** | **40-50** |
| **S** | 2.85E-05 | 5.97E-06 | 1.81E-06 | 9.25E-07 | 5.83E-07 | 2.67E-07 | 1.02E-07 | 5.32E-08 | 3.50E-08 | 2.57E-08 |
| **SSW** | 2.19E-05 | 4.60E-06 | 1.39E-06 | 7.13E-07 | 4.50E-07 | 2.07E-07 | 7.91E-08 | 4.12E-08 | 2.71E-08 | 1.99E-08 |
| **SW** | 5.93E-06 | 1.24E-06 | 3.71E-07 | 1.88E-07 | 1.18E-07 | 5.32E-08 | 1.99E-08 | 1.02E-08 | 6.61E-09 | 4.81E-09 |
| **WSW** | 3.62E-06 | 7.55E-07 | 2.24E-07 | 1.12E-07 | 6.98E-08 | 3.13E-08 | 1.15E-08 | 5.80E-09 | 3.74E-09 | 2.71E-09 |
| **W** | 3.29E-06 | 6.82E-07 | 2.00E-07 | 9.96E-08 | 6.16E-08 | 2.73E-08 | 9.89E-09 | 4.93E-09 | 3.16E-09 | 2.28E-09 |
| **WNW** | 3.33E-06 | 6.91E-07 | 2.03E-07 | 1.01E-07 | 6.24E-08 | 2.77E-08 | 1.00E-08 | 4.99E-09 | 3.19E-09 | 2.30E-09 |
| **NW** | 3.37E-06 | 6.99E-07 | 2.05E-07 | 1.02E-07 | 6.33E-08 | 2.81E-08 | 1.02E-08 | 5.07E-09 | 3.25E-09 | 2.34E-09 |
| **NNW** | 4.24E-06 | 8.80E-07 | 2.58E-07 | 1.29E-07 | 7.97E-08 | 3.54E-08 | 1.29E-08 | 6.43E-09 | 4.13E-09 | 2.97E-09 |
| **N** | 3.33E-06 | 6.88E-07 | 2.00E-07 | 9.91E-08 | 6.10E-08 | 2.68E-08 | 9.61E-09 | 4.76E-09 | 3.03E-09 | 2.18E-09 |
| **NNE** | 1.95E-06 | 4.03E-07 | 1.17E-07 | 5.78E-08 | 3.55E-08 | 1.56E-08 | 5.54E-09 | 2.73E-09 | 1.73E-09 | 1.24E-09 |
| **NE** | 1.73E-06 | 3.57E-07 | 1.03E-07 | 5.08E-08 | 3.11E-08 | 1.36E-08 | 4.76E-09 | 2.32E-09 | 1.46E-09 | 1.04E-09 |
| **ENE** | 3.07E-06 | 6.34E-07 | 1.82E-07 | 8.89E-08 | 5.41E-08 | 2.33E-08 | 8.00E-09 | 3.82E-09 | 2.38E-09 | 1.68E-09 |
| **E** | 5.02E-06 | 1.04E-06 | 2.98E-07 | 1.45E-07 | 8.83E-08 | 3.79E-08 | 1.29E-08 | 6.12E-09 | 3.79E-09 | 2.66E-09 |
| **ESE** | 7.57E-06 | 1.57E-06 | 4.48E-07 | 2.19E-07 | 1.33E-07 | 5.74E-08 | 1.97E-08 | 9.37E-09 | 5.82E-09 | 4.10E-09 |
| **SE** | 5.54E-06 | 1.15E-06 | 3.36E-07 | 1.67E-07 | 1.03E-07 | 4.53E-08 | 1.61E-08 | 7.95E-09 | 5.05E-09 | 3.61E-09 |
| **SSE** | 6.92E-06 | 1.45E-06 | 4.30E-07 | 2.17E-07 | 1.35E-07 | 6.08E-08 | 2.25E-08 | 1.14E-08 | 7.36E-09 | 5.34E-09 |


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**Table 2.4-87 EI 50-mile X/Q and D/Q Values, Decayed/Undepleted**

| | Segment Boundary (miles) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| **Direction** | **.5-1** | **1-2** | **2-3** | **3-4** | **4-5** | **5-10** | **10-20** | **20-30** | **30-40** | **40-50** |
| **S** | 2.84E-05 | 5.95E-06 | 1.80E-06 | 9.17E-07 | 5.77E-07 | 2.63E-07 | 9.87E-08 | 5.01E-08 | 3.22E-08 | 2.31E-08 |
| **SSW** | 2.19E-05 | 4.58E-06 | 1.38E-06 | 7.06E-07 | 4.44E-07 | 2.02E-07 | 7.57E-08 | 3.83E-08 | 2.45E-08 | 1.75E-08 |
| **SW** | 5.91E-06 | 1.23E-06 | 3.67E-07 | 1.85E-07 | 1.15E-07 | 5.14E-08 | 1.85E-08 | 9.04E-09 | 5.61E-09 | 3.91E-09 |
| **WSW** | 3.61E-06 | 7.50E-07 | 2.21E-07 | 1.10E-07 | 6.81E-08 | 3.01E-08 | 1.06E-08 | 5.08E-09 | 3.11E-09 | 2.14E-09 |
| **W** | 3.28E-06 | 6.78E-07 | 1.97E-07 | 9.78E-08 | 6.01E-08 | 2.63E-08 | 9.15E-09 | 4.34E-09 | 2.65E-09 | 1.82E-09 |
| **WNW** | 3.32E-06 | 6.86E-07 | 2.00E-07 | 9.90E-08 | 6.09E-08 | 2.66E-08 | 9.24E-09 | 4.37E-09 | 2.66E-09 | 1.82E-09 |
| **NW** | 3.35E-06 | 6.94E-07 | 2.02E-07 | 1.00E-07 | 6.18E-08 | 2.70E-08 | 9.40E-09 | 4.45E-09 | 2.71E-09 | 1.86E-09 |
| **NNW** | 4.22E-06 | 8.74E-07 | 2.55E-07 | 1.27E-07 | 7.79E-08 | 3.41E-08 | 1.19E-08 | 5.68E-09 | 3.47E-09 | 2.39E-09 |
| **N** | 3.32E-06 | 6.84E-07 | 1.98E-07 | 9.75E-08 | 5.98E-08 | 2.60E-08 | 9.01E-09 | 4.27E-09 | 2.61E-09 | 1.80E-09 |
| **NNE** | 1.94E-06 | 4.01E-07 | 1.16E-07 | 5.69E-08 | 3.48E-08 | 1.51E-08 | 5.17E-09 | 2.43E-09 | 1.48E-09 | 1.01E-09 |
| **NE** | 1.73E-06 | 3.56E-07 | 1.02E-07 | 5.02E-08 | 3.06E-08 | 1.32E-08 | 4.52E-09 | 2.13E-09 | 1.30E-09 | 8.97E-10 |
| **ENE** | 3.06E-06 | 6.33E-07 | 1.81E-07 | 8.82E-08 | 5.36E-08 | 2.30E-08 | 7.76E-09 | 3.62E-09 | 2.21E-09 | 1.53E-09 |
| **E** | 5.02E-06 | 1.04E-06 | 2.96E-07 | 1.44E-07 | 8.75E-08 | 3.73E-08 | 1.25E-08 | 5.79E-09 | 3.51E-09 | 2.41E-09 |
| **ESE** | 7.56E-06 | 1.56E-06 | 4.46E-07 | 2.18E-07 | 1.32E-07 | 5.66E-08 | 1.91E-08 | 8.91E-09 | 5.43E-09 | 3.75E-09 |
| **SE** | 5.53E-06 | 1.15E-06 | 3.34E-07 | 1.65E-07 | 1.02E-07 | 4.43E-08 | 1.55E-08 | 7.39E-09 | 4.56E-09 | 3.18E-09 |
| **SSE** | 6.91E-06 | 1.44E-06 | 4.27E-07 | 2.14E-07 | 1.33E-07 | 5.92E-08 | 2.13E-08 | 1.05E-08 | 6.54E-09 | 4.59E-09 |


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**Table 2.4-88 EI 50-mile X/Q and D/Q Values, Decayed/Depleted**

| | Segment Boundary (miles) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| **Direction** | **.5-1** | **1-2** | **2-3** | **3-4** | **4-5** | **5-10** | **10-20** | **20-30** | **30-40** | **40-50** |
| **S** | 2.56E-05 | 5.12E-06 | 1.47E-06 | 7.21E-07 | 4.40E-07 | 1.89E-07 | 6.36E-08 | 2.91E-08 | 1.73E-08 | 1.17E-08 |
| **SSW** | 1.97E-05 | 3.94E-06 | 1.13E-06 | 5.56E-07 | 3.39E-07 | 1.46E-07 | 4.91E-08 | 2.24E-08 | 1.33E-08 | 8.97E-09 |
| **SW** | 5.32E-06 | 1.06E-06 | 3.01E-07 | 1.46E-07 | 8.85E-08 | 3.75E-08 | 1.23E-08 | 5.46E-09 | 3.19E-09 | 2.12E-09 |
| **WSW** | 3.25E-06 | 6.47E-07 | 1.81E-07 | 8.73E-08 | 5.25E-08 | 2.20E-08 | 7.07E-09 | 3.10E-09 | 1.79E-09 | 1.18E-09 |
| **W** | 2.95E-06 | 5.85E-07 | 1.62E-07 | 7.75E-08 | 4.63E-08 | 1.93E-08 | 6.09E-09 | 2.64E-09 | 1.52E-09 | 9.96E-10 |
| **WNW** | 2.99E-06 | 5.92E-07 | 1.64E-07 | 7.85E-08 | 4.69E-08 | 1.95E-08 | 6.16E-09 | 2.67E-09 | 1.53E-09 | 1.00E-09 |
| **NW** | 3.02E-06 | 5.99E-07 | 1.66E-07 | 7.96E-08 | 4.76E-08 | 1.98E-08 | 6.26E-09 | 2.72E-09 | 1.56E-09 | 1.02E-09 |
| **NNW** | 3.80E-06 | 7.54E-07 | 2.09E-07 | 1.00E-07 | 6.00E-08 | 2.50E-08 | 7.92E-09 | 3.45E-09 | 1.99E-09 | 1.30E-09 |
| **N** | 2.99E-06 | 5.90E-07 | 1.62E-07 | 7.71E-08 | 4.59E-08 | 1.90E-08 | 5.94E-09 | 2.56E-09 | 1.47E-09 | 9.63E-10 |
| **NNE** | 1.75E-06 | 3.46E-07 | 9.48E-08 | 4.50E-08 | 2.67E-08 | 1.10E-08 | 3.42E-09 | 1.47E-09 | 8.37E-10 | 5.46E-10 |
| **NE** | 1.55E-06 | 3.06E-07 | 8.37E-08 | 3.96E-08 | 2.34E-08 | 9.59E-09 | 2.95E-09 | 1.26E-09 | 7.15E-10 | 4.66E-10 |
| **ENE** | 2.75E-06 | 5.44E-07 | 1.48E-07 | 6.94E-08 | 4.09E-08 | 1.66E-08 | 5.00E-09 | 2.09E-09 | 1.18E-09 | 7.62E-10 |
| **E** | 4.51E-06 | 8.92E-07 | 2.42E-07 | 1.13E-07 | 6.67E-08 | 2.69E-08 | 8.07E-09 | 3.35E-09 | 1.88E-09 | 1.21E-09 |
| **ESE** | 6.79E-06 | 1.34E-06 | 3.64E-07 | 1.71E-07 | 1.01E-07 | 4.08E-08 | 1.23E-08 | 5.14E-09 | 2.89E-09 | 1.87E-09 |
| **SE** | 4.97E-06 | 9.88E-07 | 2.73E-07 | 1.30E-07 | 7.76E-08 | 3.21E-08 | 1.00E-08 | 4.33E-09 | 2.48E-09 | 1.63E-09 |
| **SSE** | 6.21E-06 | 1.24E-06 | 3.49E-07 | 1.69E-07 | 1.02E-07 | 4.30E-08 | 1.39E-08 | 6.17E-09 | 3.60E-09 | 2.39E-09 |


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**Table 2.4-89 EI 50-mile X/Q and D/Q Values, Deposited**

| | Segment Boundary (miles) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| **Direction** | **.5-1** | **1-2** | **2-3** | **3-4** | **4-5** | **5-10** | **10-20** | **20-30** | **30-40** | **40-50** |
| **S** | 6.64E-08 | 1.36E-08 | 3.55E-09 | 1.60E-09 | 9.02E-10 | 3.47E-10 | 1.00E-10 | 3.98E-11 | 2.12E-11 | 1.32E-11 |
| **SSW** | 4.02E-08 | 8.23E-09 | 2.15E-09 | 9.65E-10 | 5.46E-10 | 2.10E-10 | 6.08E-11 | 2.41E-11 | 1.29E-11 | 7.96E-12 |
| **SW** | 1.10E-08 | 2.26E-09 | 5.89E-10 | 2.64E-10 | 1.50E-10 | 5.75E-11 | 1.66E-11 | 6.59E-12 | 3.52E-12 | 2.18E-12 |
| **WSW** | 9.30E-09 | 1.90E-09 | 4.97E-10 | 2.23E-10 | 1.26E-10 | 4.86E-11 | 1.41E-11 | 5.57E-12 | 2.97E-12 | 1.84E-12 |
| **W** | 1.03E-08 | 2.10E-09 | 5.49E-10 | 2.47E-10 | 1.40E-10 | 5.37E-11 | 1.55E-11 | 6.15E-12 | 3.29E-12 | 2.03E-12 |
| **WNW** | 8.62E-09 | 1.77E-09 | 4.61E-10 | 2.07E-10 | 1.17E-10 | 4.50E-11 | 1.30E-11 | 5.16E-12 | 2.76E-12 | 1.71E-12 |
| **NW** | 8.10E-09 | 1.66E-09 | 4.33E-10 | 1.95E-10 | 1.10E-10 | 4.23E-11 | 1.22E-11 | 4.85E-12 | 2.59E-12 | 1.60E-12 |
| **NNW** | 1.08E-08 | 2.21E-09 | 5.77E-10 | 2.59E-10 | 1.47E-10 | 5.64E-11 | 1.63E-11 | 6.47E-12 | 3.46E-12 | 2.14E-12 |
| **N** | 1.09E-08 | 2.24E-09 | 5.84E-10 | 2.63E-10 | 1.49E-10 | 5.71E-11 | 1.65E-11 | 6.55E-12 | 3.50E-12 | 2.16E-12 |
| **NNE** | 7.43E-09 | 1.52E-09 | 3.98E-10 | 1.79E-10 | 1.01E-10 | 3.88E-11 | 1.12E-11 | 4.45E-12 | 2.38E-12 | 1.47E-12 |
| **NE** | 1.12E-08 | 2.30E-09 | 6.01E-10 | 2.70E-10 | 1.53E-10 | 5.87E-11 | 1.70E-11 | 6.73E-12 | 3.60E-12 | 2.23E-12 |
| **ENE** | 2.81E-08 | 5.76E-09 | 1.50E-09 | 6.76E-10 | 3.82E-10 | 1.47E-10 | 4.25E-11 | 1.69E-11 | 9.00E-12 | 5.57E-12 |
| **E** | 5.15E-08 | 1.06E-08 | 2.75E-09 | 1.24E-09 | 7.00E-10 | 2.69E-10 | 7.79E-11 | 3.09E-11 | 1.65E-11 | 1.02E-11 |
| **ESE** | 7.05E-08 | 1.44E-08 | 3.77E-09 | 1.69E-09 | 9.58E-10 | 3.68E-10 | 1.07E-10 | 4.22E-11 | 2.26E-11 | 1.40E-11 |
| **SE** | 2.88E-08 | 5.91E-09 | 1.54E-09 | 6.93E-10 | 3.92E-10 | 1.51E-10 | 4.36E-11 | 1.73E-11 | 9.23E-12 | 5.71E-12 |
| **SSE** | 1.98E-08 | 4.05E-09 | 1.06E-09 | 4.75E-10 | 2.69E-10 | 1.03E-10 | 2.99E-11 | 1.19E-11 | 6.33E-12 | 3.92E-12 |


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**Table 2.4-90 Tower Instrument Specifications and Accuracies for Meteorological Monitoring Program (Preoperational and Operational)**

| Characteristics | Requirements[1] | Specifications |
|---|---|---|
| **Wind Speed Sensor** | | |
| Accuracy | ±0.2 m/s (±0.45 mph)<br>OR<br>±5% of observed wind speed | ±5% |
| Resolution | 0.1 m/s (0.1 mph) | 0.1 mph |
| Wind Direction Sensor | | |
| Accuracy | ±5 degrees | ±5 degrees |
| Resolution | 1.0 degree | 1.0 degree |
| **Temperature Sensors** | | |
| Accuracy (ambient) | ±0.5°C (±0.9°F) | ±0.9°F |
| Resolution (ambient) | 0.1°C (0.1°F) | 0.1°F |
| Accuracy (vertical temperature difference) | ±0.1°C (±0.18°F) | ±0.18°F |
| Resolution (vertical temperature difference) | 0.01°C (0.01°F) | 0.01°F |
| **Relative Humidity** | | |
| Accuracy | ±4% | ±4% |
| Resolution | 0.1% | 0.1% |
| **Precipitation Sensor** | | |
| Accuracy | ±10% for a volume equivalent to 2.54 mm (0.1 in.) of precipitation at a rate < 50 mm/hr (< 2 in/hr) | ±10% |
| Resolution | 0.25 mm (0.01 in.) | 0.25 mm |
| **Time** | | |
| Accuracy | ± 5 min | ± 5 min |
| Resolution | 1 min | 1 min |
| **Solar Radiation** | | |
| Accuracy | N/A | 10 $W/m^2$ |
| Resolution | N/A | 1 |

[1] Accuracy and resolution criteria from Regulatory Guide 1.23, Revision 1.


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**Table 2.4-91 Kemmerer Unit 1 Meteorological Percent Data Recovery**

| Parameter | Kemmerer Unit 1 4/9/2022–4/8/2023 (%) | Naughton Power Plant 4/9/2022–4/8/2023 (%) | Naughton Power Plant 1/1/2019–12/31/2021 (%) |
|---|---|---|---|
| Wind Speed 10-meter level[1] | 94.9 | 96.9 | 98.8 |
| Wind Speed 60-meter level[1] | 94.0 | 98.2 | 98.7 |
| Wind Direction 10-meter level[1] | 93.8 | 96.5 | 99.2 |
| Wind Direction 60-meter level[1] | 93.8 | 99.6 | 98.7 |
| Delta Temperature 60-10 meter[1] | 94.6 | 99.8 | 99.8 |
| Temperature 10-meter level[1] | 94.9 | 99.8 | 99.8 |
| Temperature 60-meter level[1] | 94.9 | 99.8 | 99.8 |
| Dew Point Temperature 10-meter level | 66.4 | 100.0 | 100.0 |
| Relative Humidity 10-meter level | 89.9 | 100.0 | 100.0 |
| Solar Radiation 8-meter level | 93.9 | 100.0 | 100.0 |
| Precipitation | 94.9 | 100.0 | 100.0 |
| *Composite* | | | |
| 10 meter Wind Speed and Direction, Delta Temperature 60-10 meter | 93.8 | 96.6 | 98.4 |
| 60 meter Wind Speed and Direction, Delta Temperature 60-10 meter | 93.8 | 98.0 | 98.5 |

[1] Data used in further meteorological analysis.


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**Table 2.4-92 List of Wildfires in Nearby Region of Site (January 1996-December 31st, 2023) (Sheet 1 of 2)**

| Wildfire Event ID | Wildfire Location | Begin Date of Wildfire Event | End Date of Wildfire Event | Source of Data |
|---|---|---|---|---|
| 5172294 | LINCOLN CO. | 8/15/2000 | 8/31/2000 | NEWSPAPER |
| 5157172 | LINCOLN CO. | 9/1/2000 | 9/15/2000 | NEWSPAPER |
| 5258155 | LINCOLN CO. | 7/4/2001 | 7/12/2001 | NEWSPAPER |
| 5263400 | LINCOLN CO. | 8/18/2001 | 8/23/2001 | NEWSPAPER |
| 5316474 | LINCOLN CO. | 9/1/2002 | 9/8/2002 | PARK/FOREST SERVICE |
| 5379614 | LINCOLN CO. | 7/12/2003 | 7/22/2003 | PARK/FOREST SERVICE |
| 5529577 | TETON & GROS VENTRE MOUNTAINS (ZONE) | 8/4/2006 | 8/15/2006 | OTHER FEDERAL AGENCY |
| 389641 | SALT RIVER & WYOMING RANGES (ZONE) | 6/24/2012 | 6/30/2012 | Park/Forest Service |
| 396234 | SALT RIVER & WYOMING RANGES (ZONE) | 7/1/2012 | 7/31/2012 | Park/Forest Service |
| 411819 | TETON & GROS VENTRE MOUNTAINS (ZONE) | 9/8/2012 | 9/28/2012 | Park/Forest Service |
| 653950 | TETON & GROS VENTRE MOUNTAINS (ZONE) | 7/17/2016 | 7/31/2016 | Park/Forest Service |
| 654088 | UPPER GREEN RIVER BASIN (ZONE) | 7/17/2016 | 7/31/2016 | Park/Forest Service |
| 654087 | SALT RIVER & WYOMING RANGES (ZONE) | 7/17/2016 | 7/31/2016 | Park/Forest Service |
| 657060 | TETON & GROS VENTRE MOUNTAINS (ZONE) | 8/1/2016 | 8/5/2016 | Park/Forest Service |
| 657061 | SALT RIVER & WYOMING RANGES (ZONE) | 8/1/2016 | 8/5/2016 | Park/Forest Service |
| 657062 | UPPER GREEN RIVER BASIN (ZONE) | 8/1/2016 | 8/5/2016 | Park/Forest Service |
| 657578 | JACKSON HOLE (ZONE) | 8/19/2016 | 8/31/2016 | Park/Forest Service |
| 657068 | TETON & GROS VENTRE MOUNTAINS (ZONE) | 8/19/2016 | 8/31/2016 | Park/Forest Service |
| 657602 | TETON & GROS VENTRE MOUNTAINS (ZONE) | 9/1/2016 | 9/17/2016 | Park/Forest Service |
| 657603 | JACKSON HOLE (ZONE) | 9/1/2016 | 9/17/2016 | Park/Forest Service |
| 785533 | SALT RIVER & WYOMING RANGES (ZONE) | 9/15/2018 | 9/30/2018 | Park/Forest Service |
| 785554 | SALT RIVER & WYOMING RANGES (ZONE) | 10/1/2018 | 10/5/2018 | Park/Forest Service |


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**Table 2.4-92 List of Wildfires in Nearby Region of Site (January 1996-December 31st, 2023) (Sheet 2 of 2)**

| Wildfire Event ID | Wildfire Location | Begin Date of Wildfire Event | End Date of Wildfire Event | Source of Data |
|---|---|---|---|---|
| 855982 | JACKSON HOLE (ZONE) | 9/1/2019 | 9/4/2019 | Fire Department/ Rescue |
| 920079 | JACKSON HOLE (ZONE) | 9/6/2020 | 9/9/2020 | Law Enforcement |



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## Figure 2.4-1 Meteorological Observation Stations and Climate Observing Stations

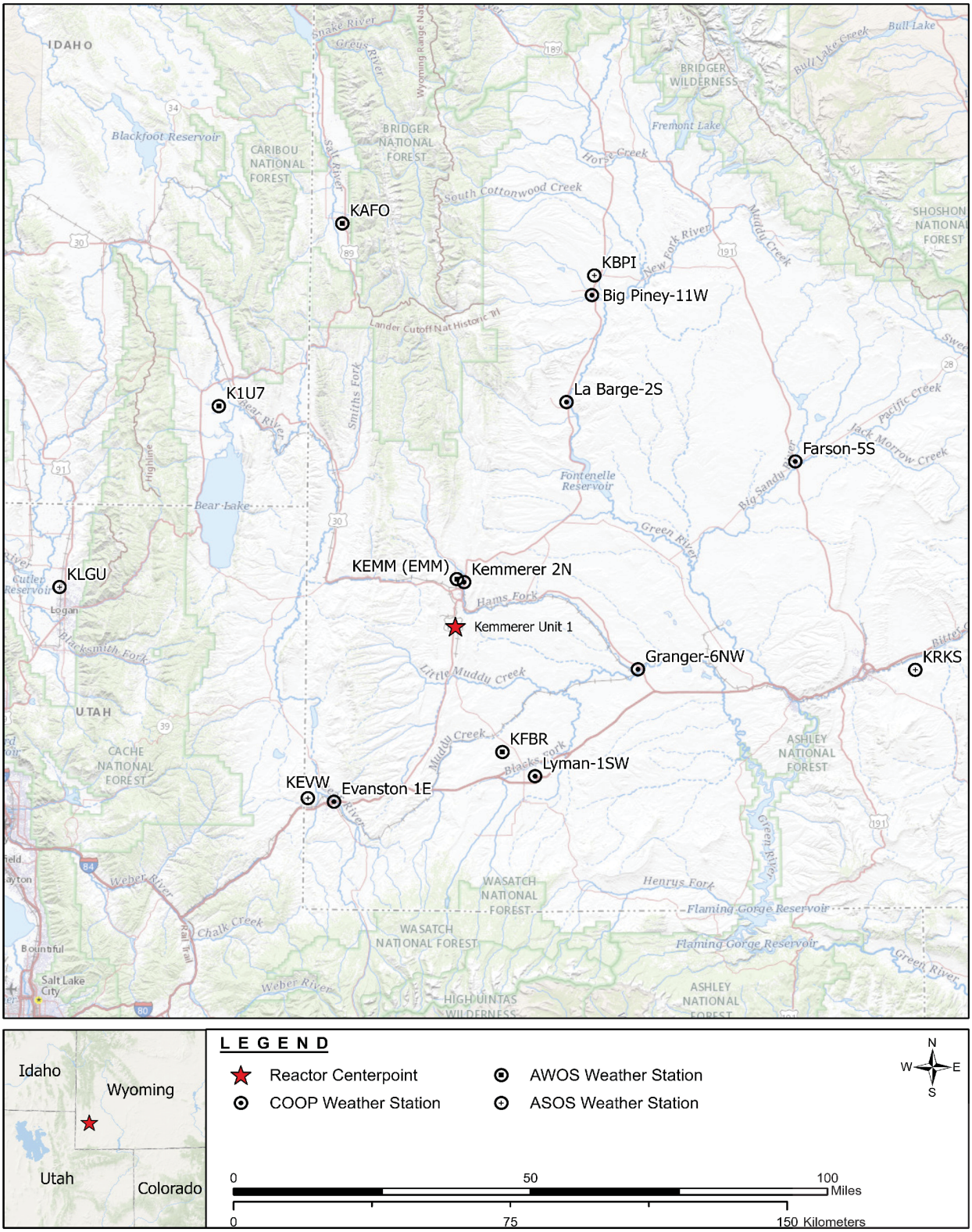



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**Figure 2.4-2 Naughton Power Plant Annual Precipitation Wind Rose 2019–2021, 10 Meter All**

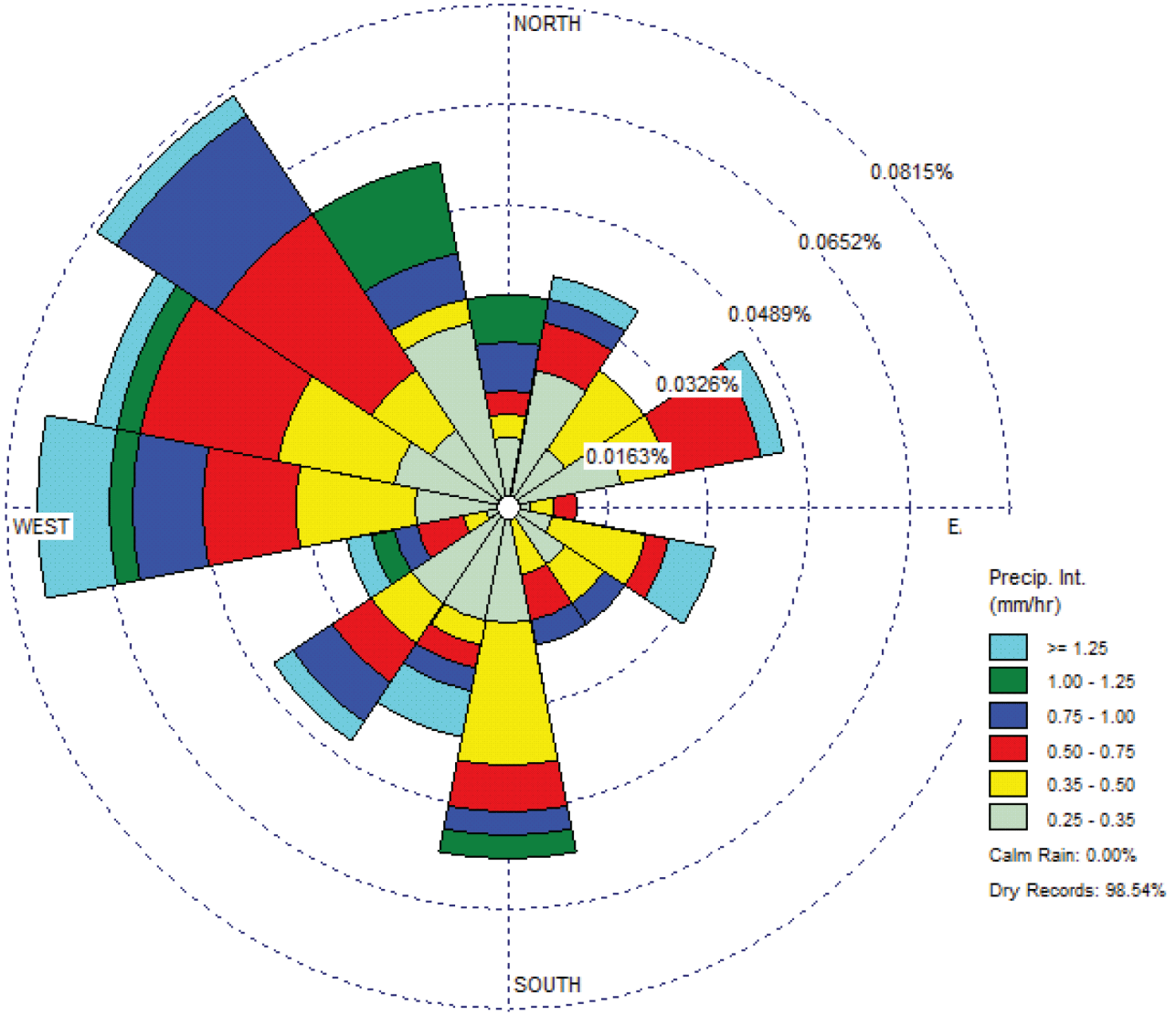






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**Figure 2.4-3 Naughton Power Plant Annual Precipitation Wind Rose 2019–2021, 50 Meter All**

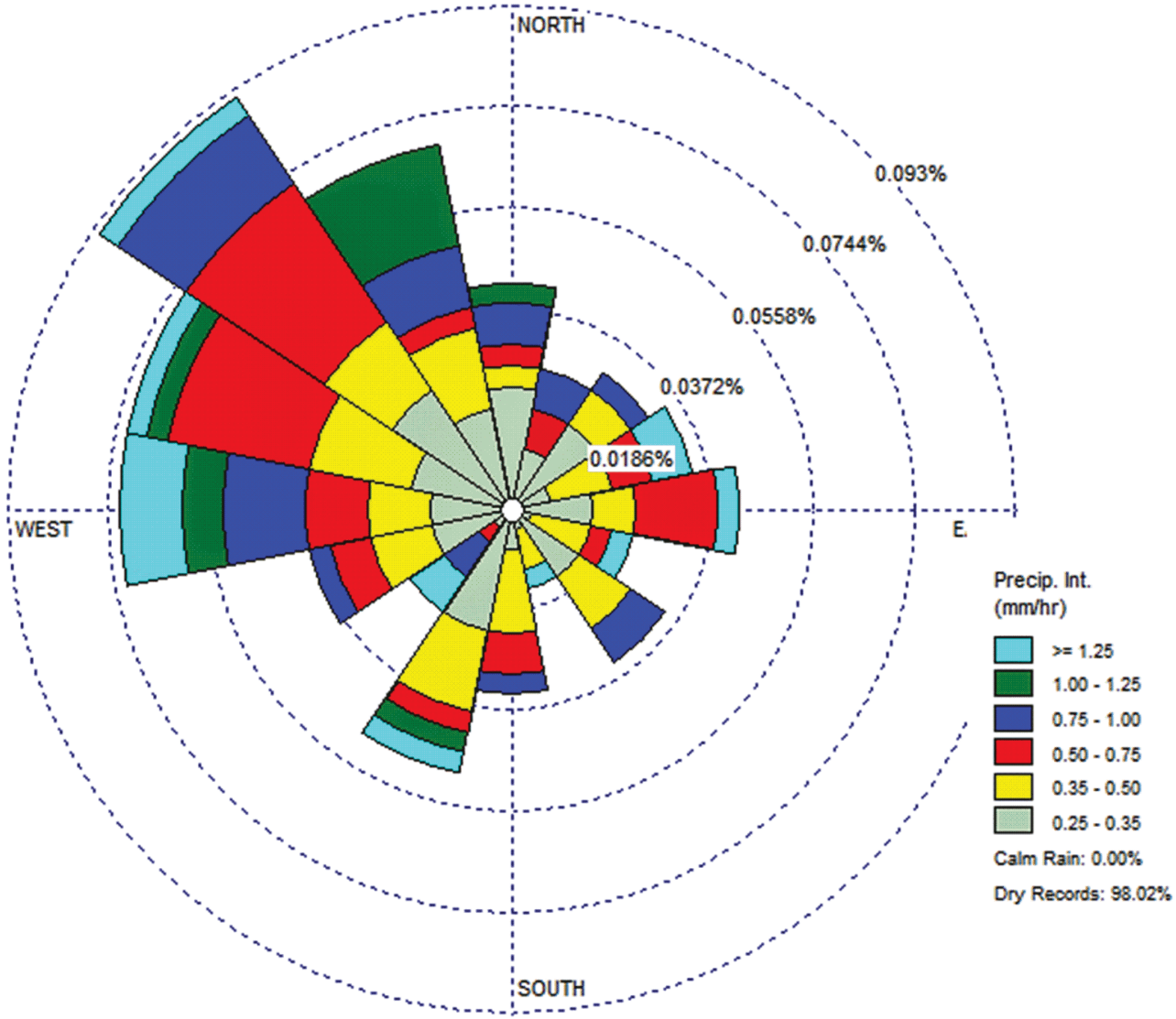






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**Figure 2.4-4 Naughton Power Plant Annual Precipitation Wind Rose 2022–2023, 10-Meter All**

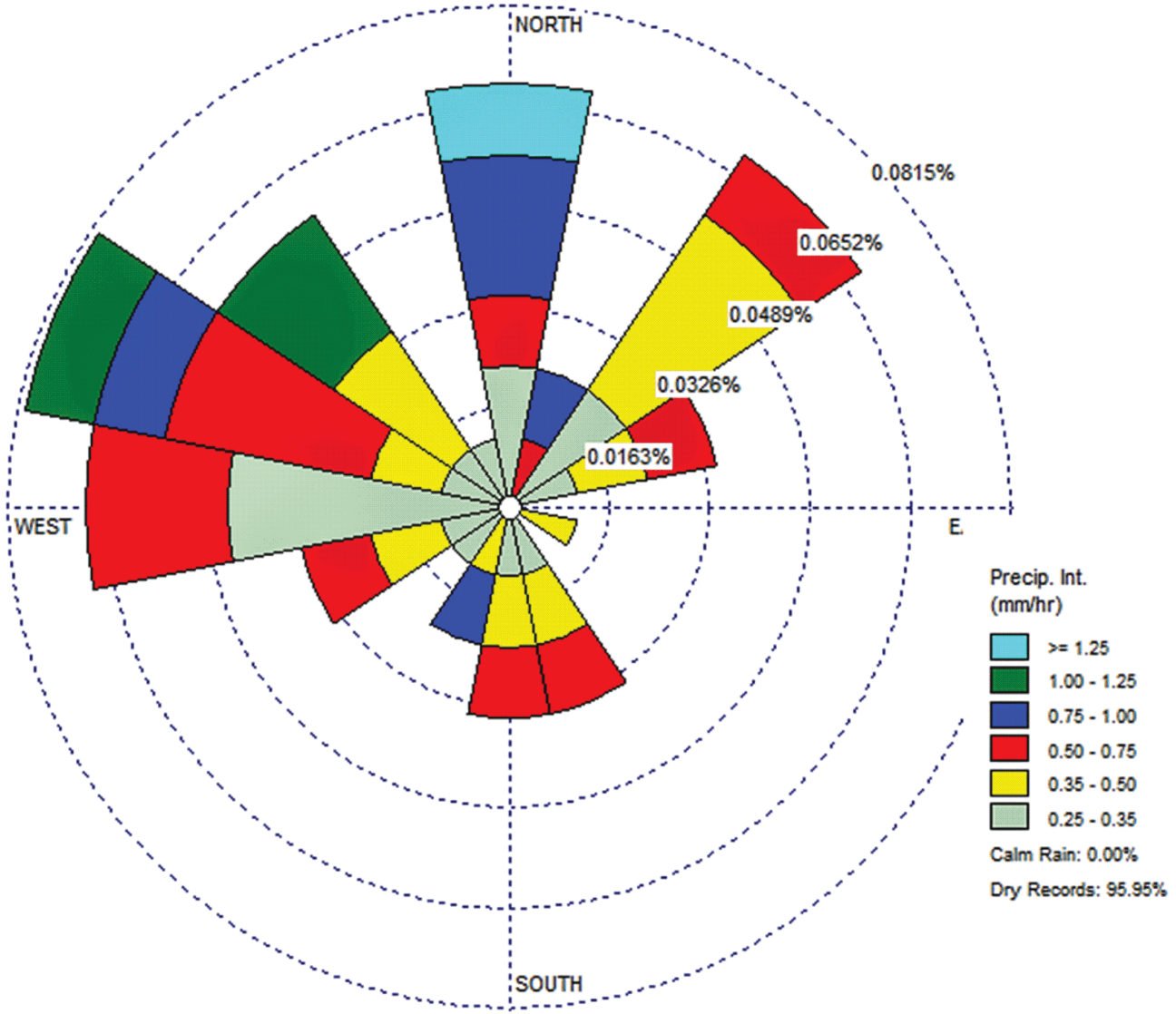


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**Figure 2.4-5 Naughton Power Plant Annual Precipitation Wind Rose 2022–2023, 50-Meter All**

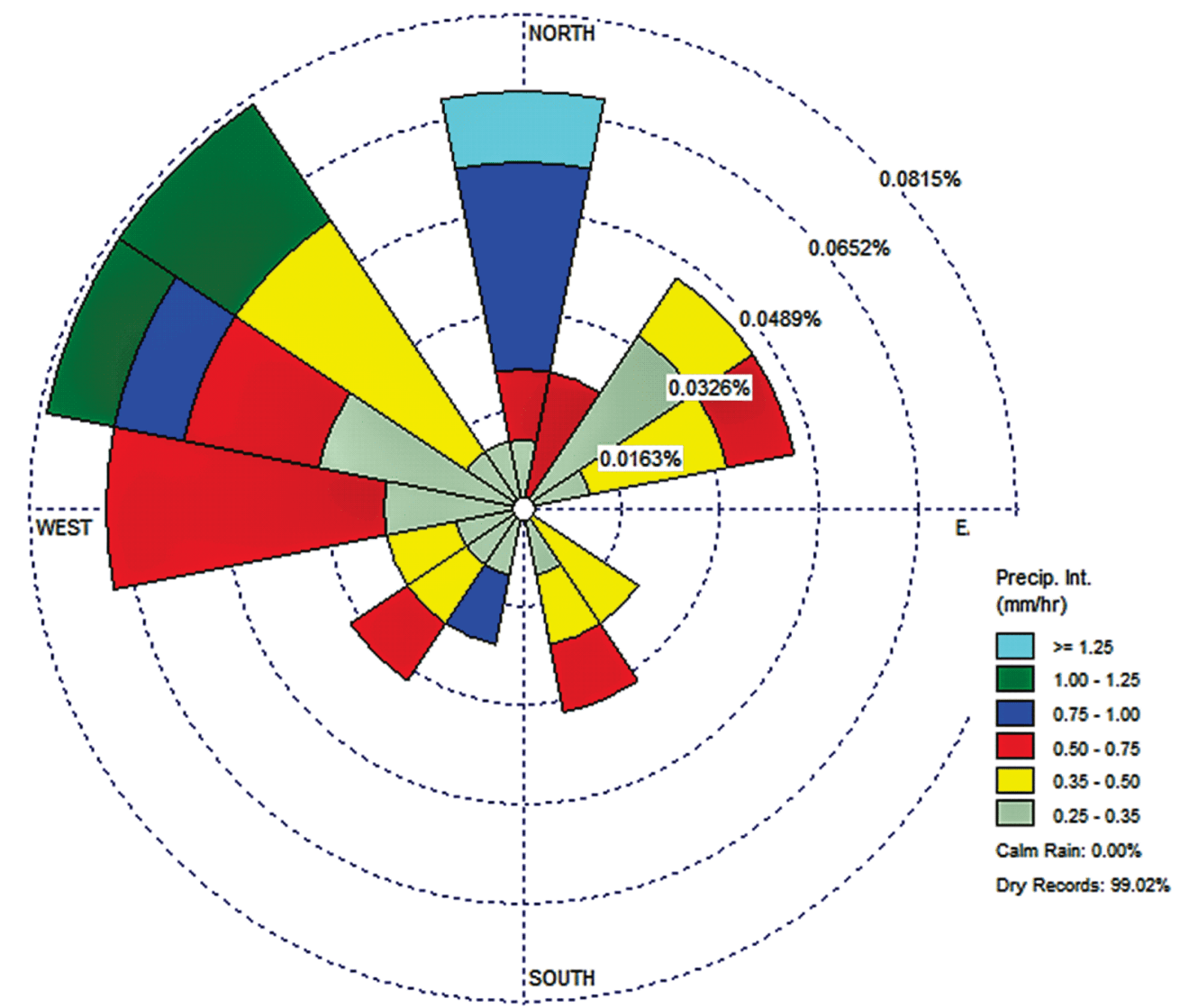






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**Figure 2.4-6 Kemmerer Unit 1 Annual Precipitation Wind Rose 2022–2023, 10 m**

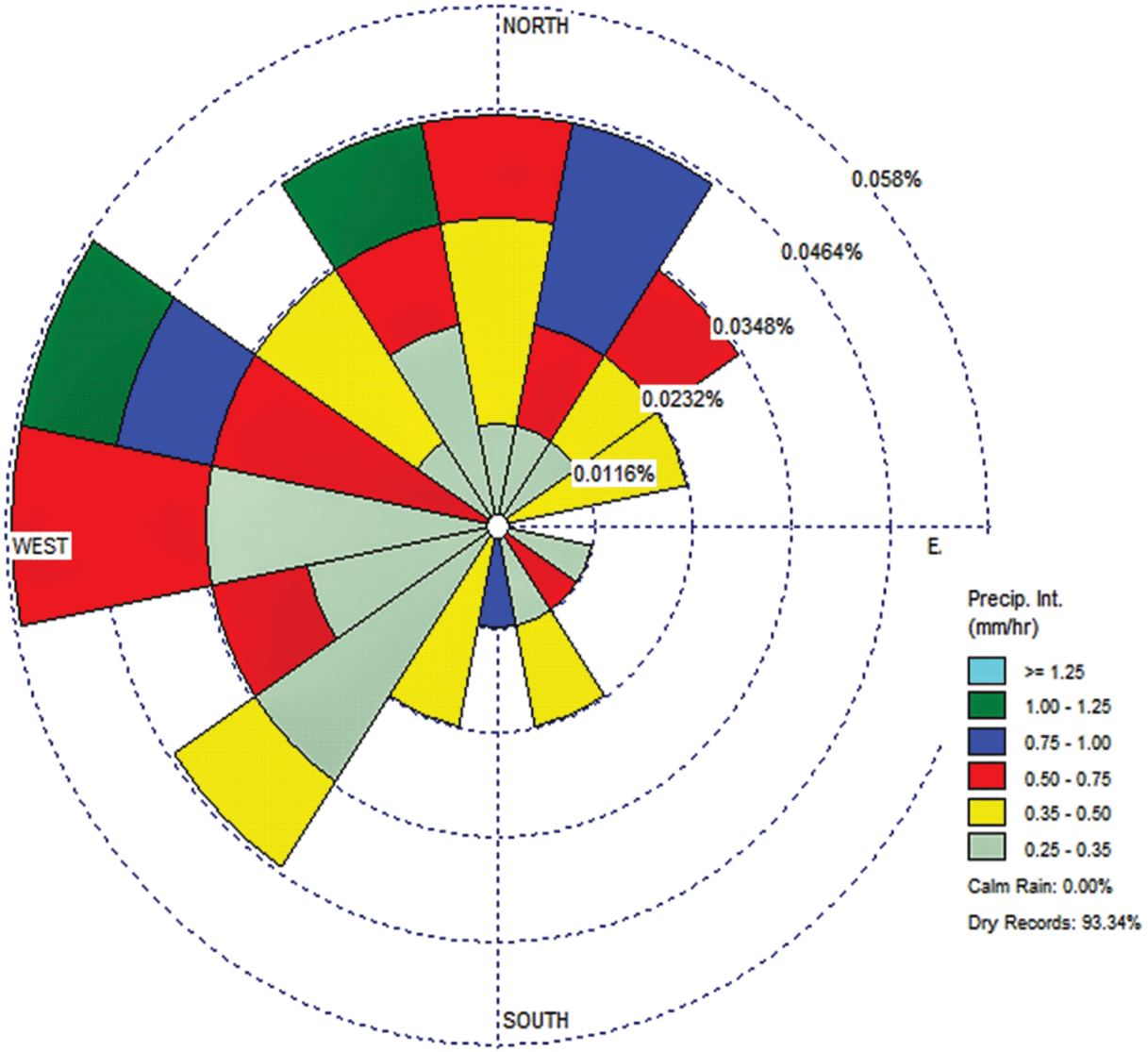






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**Figure 2.4-7 Kemmerer Unit 1 Annual Precipitation Wind Rose 2022–2023, 60 m**

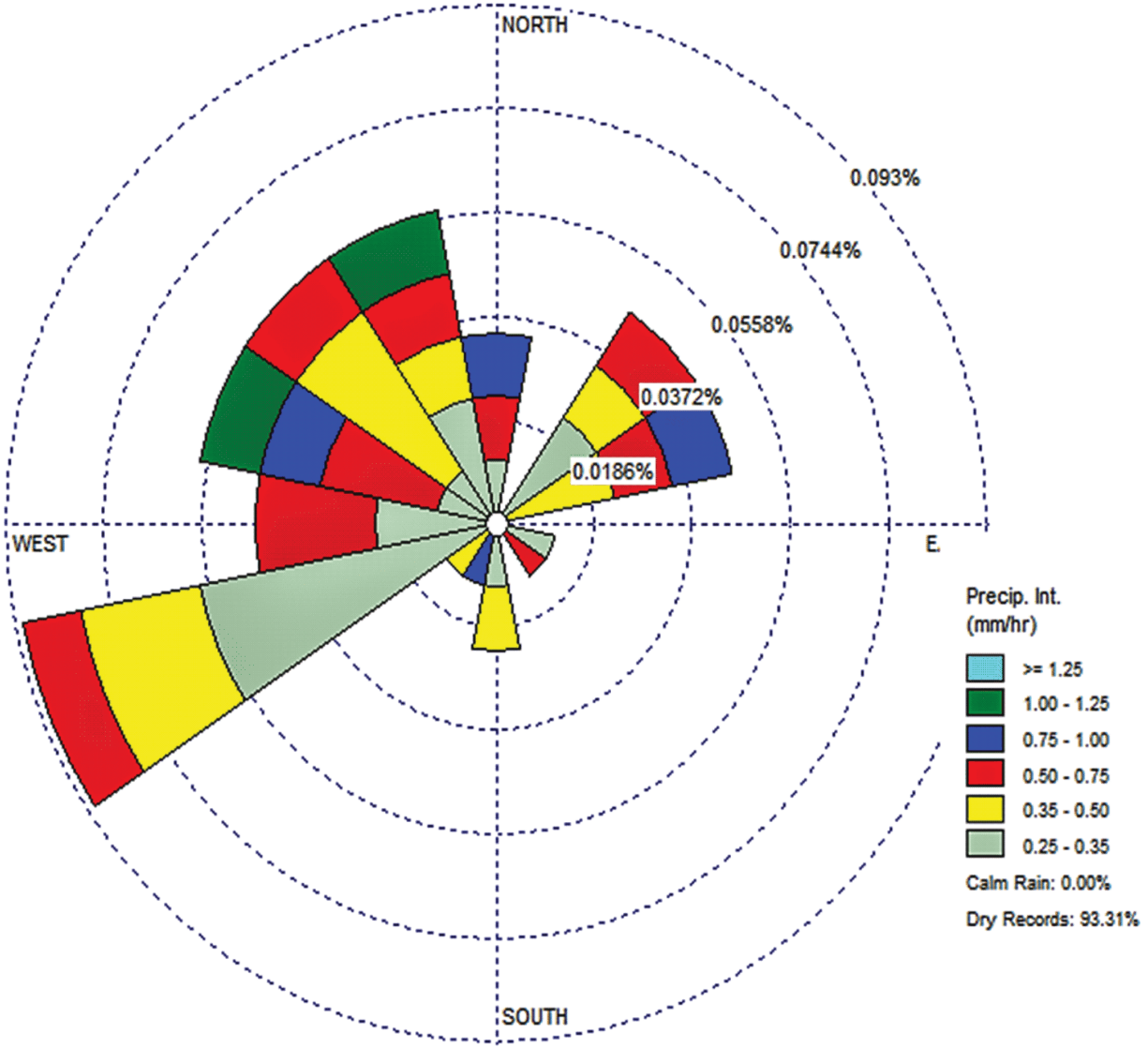






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**Figure 2.4-8 Naughton Power Plant Annual 3 Year Composite Wind Rose 2019–2021, 10 m**

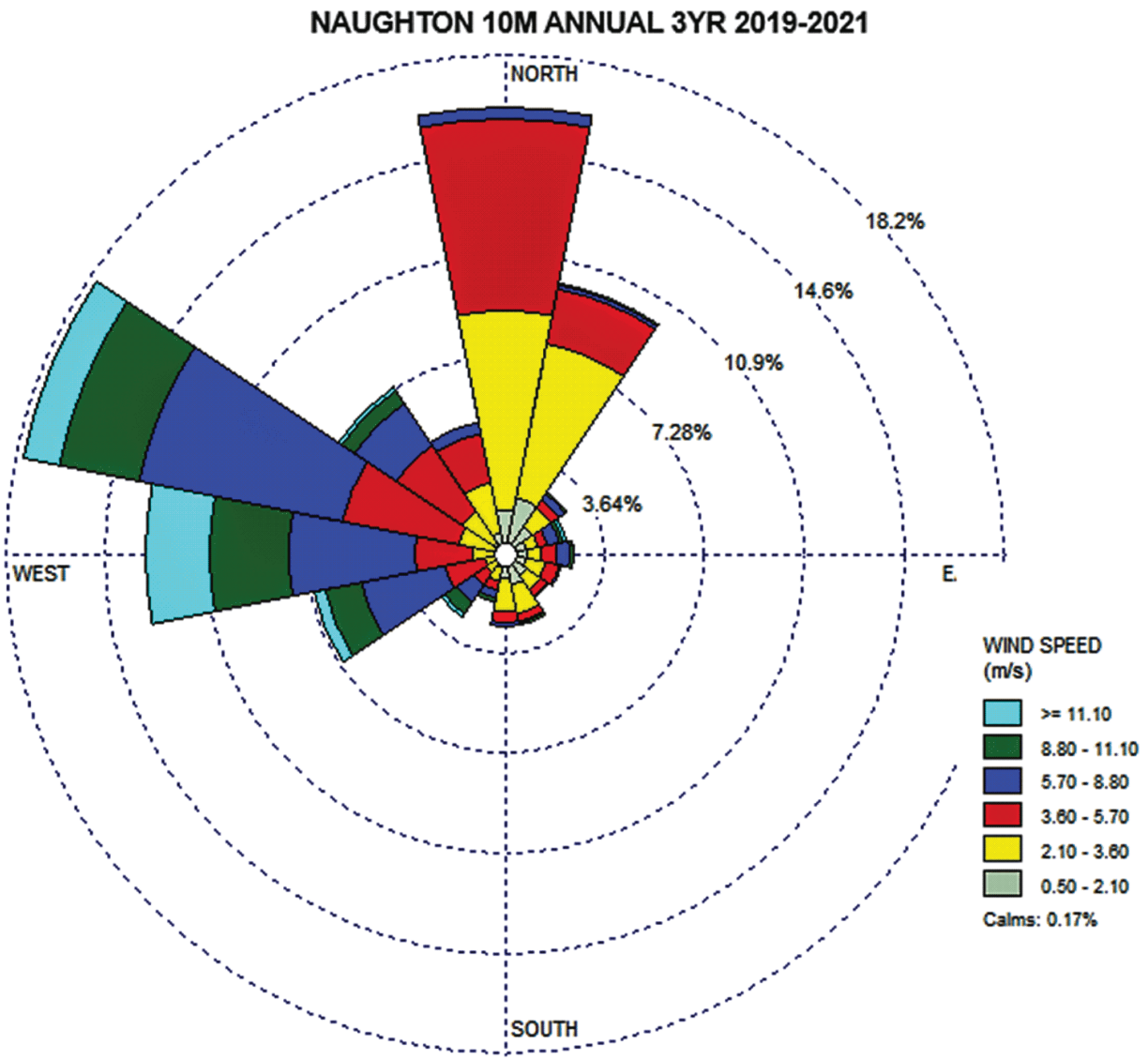






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**Figure 2.4-9 Naughton Power Plant Annual 3 Year Composite Wind Rose 2019–2021, 50 m**

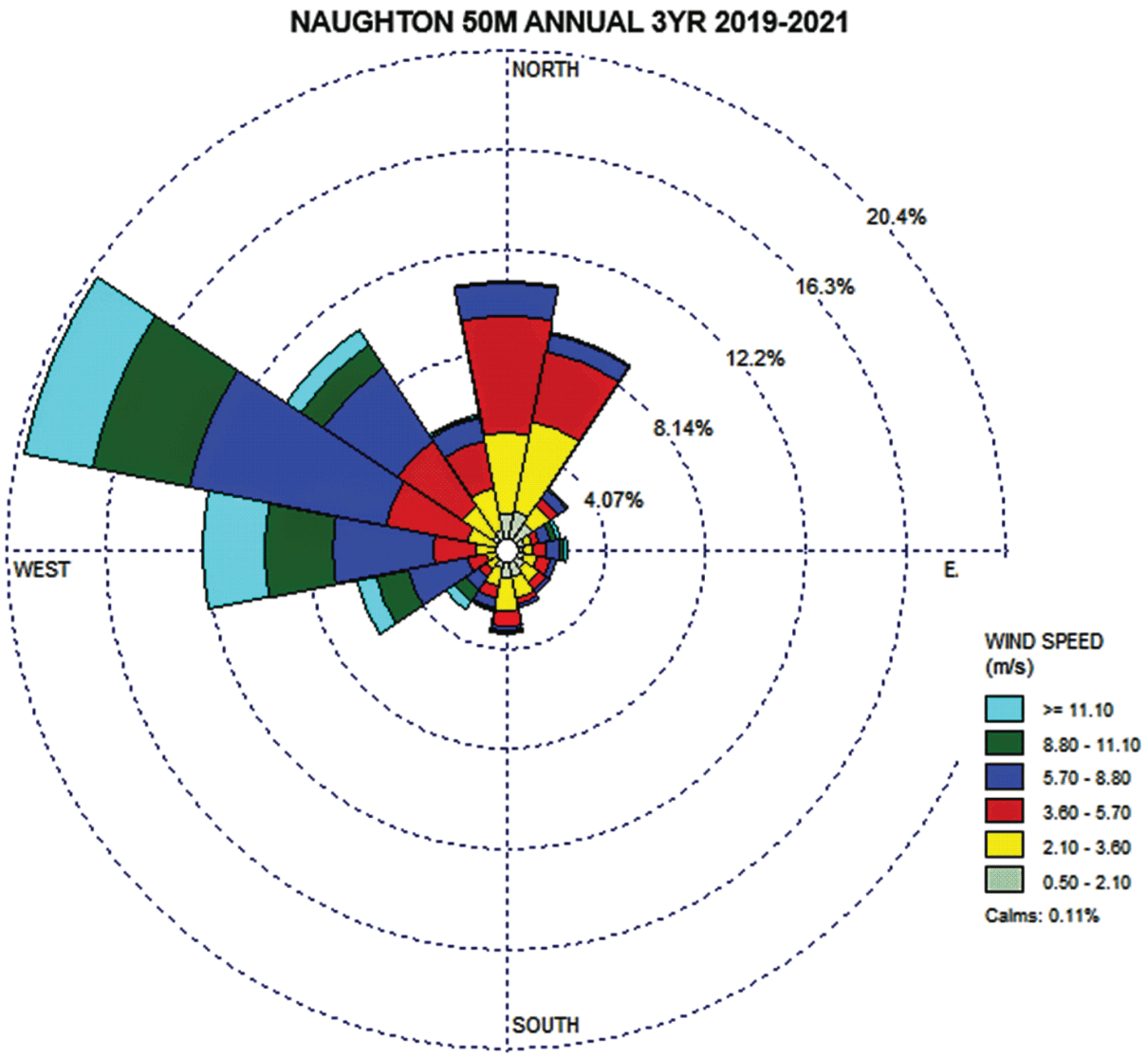






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**Figure 2.4-10 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

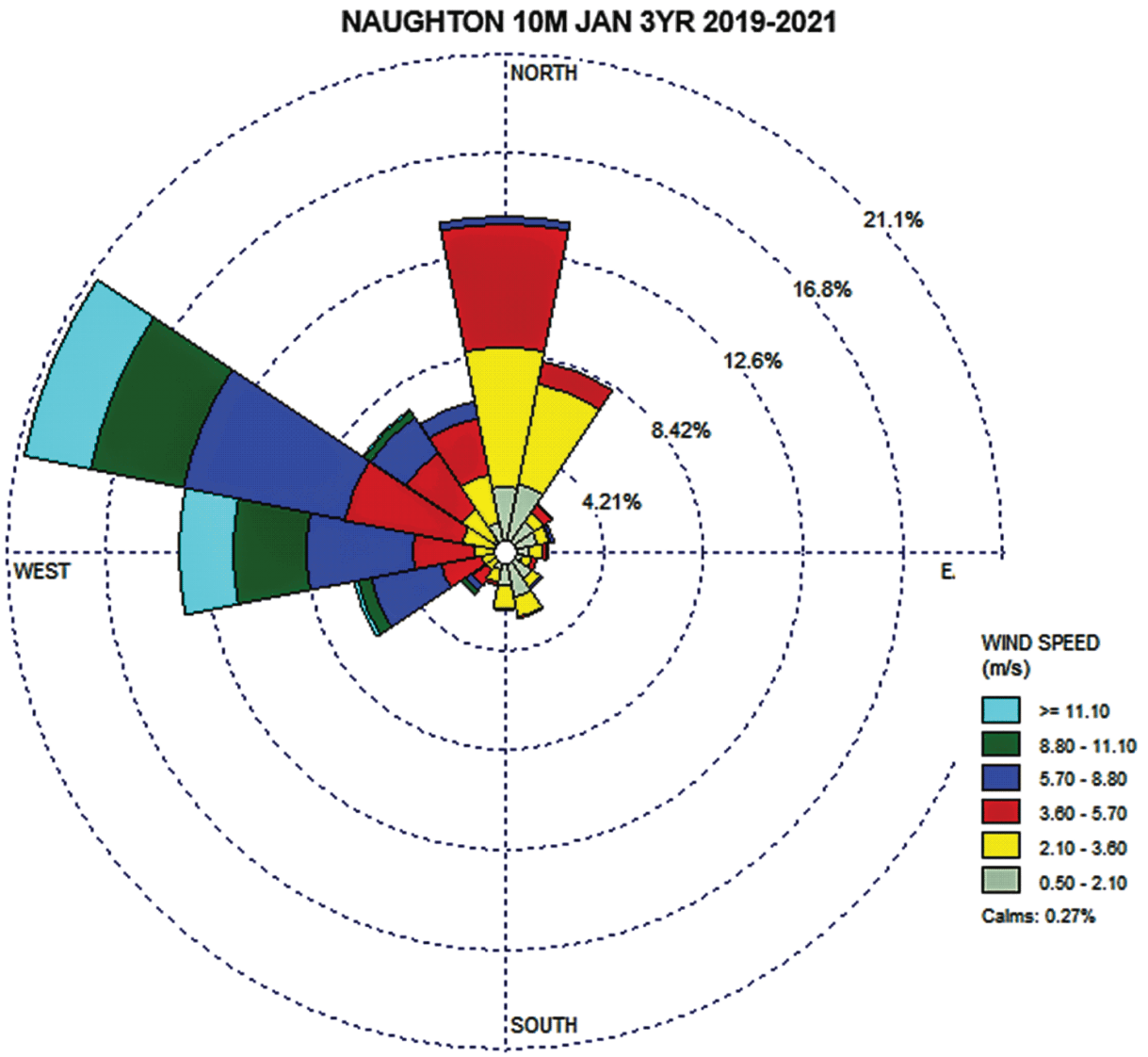






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**Figure 2.4-11 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

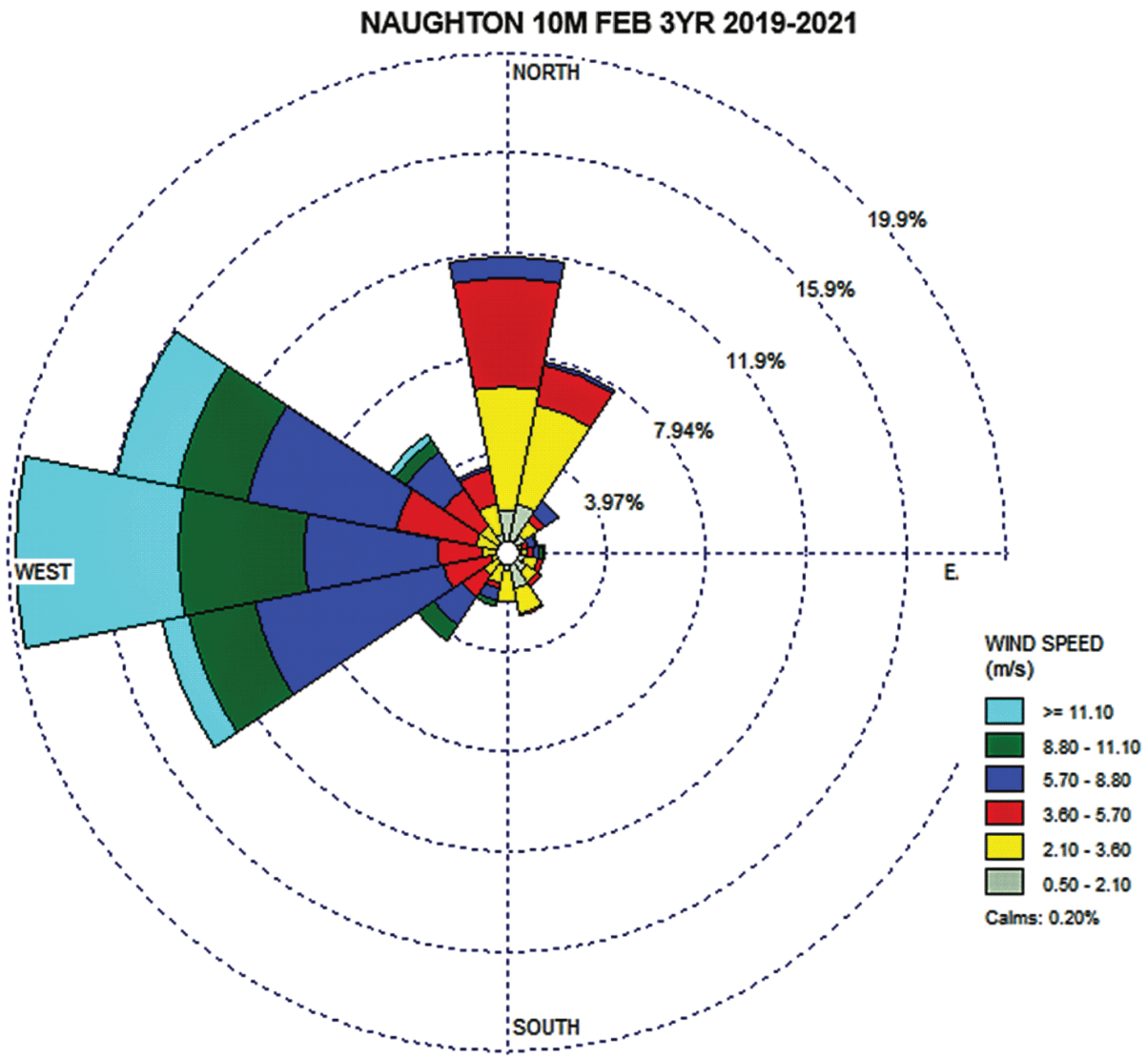






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**Figure 2.4-12 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

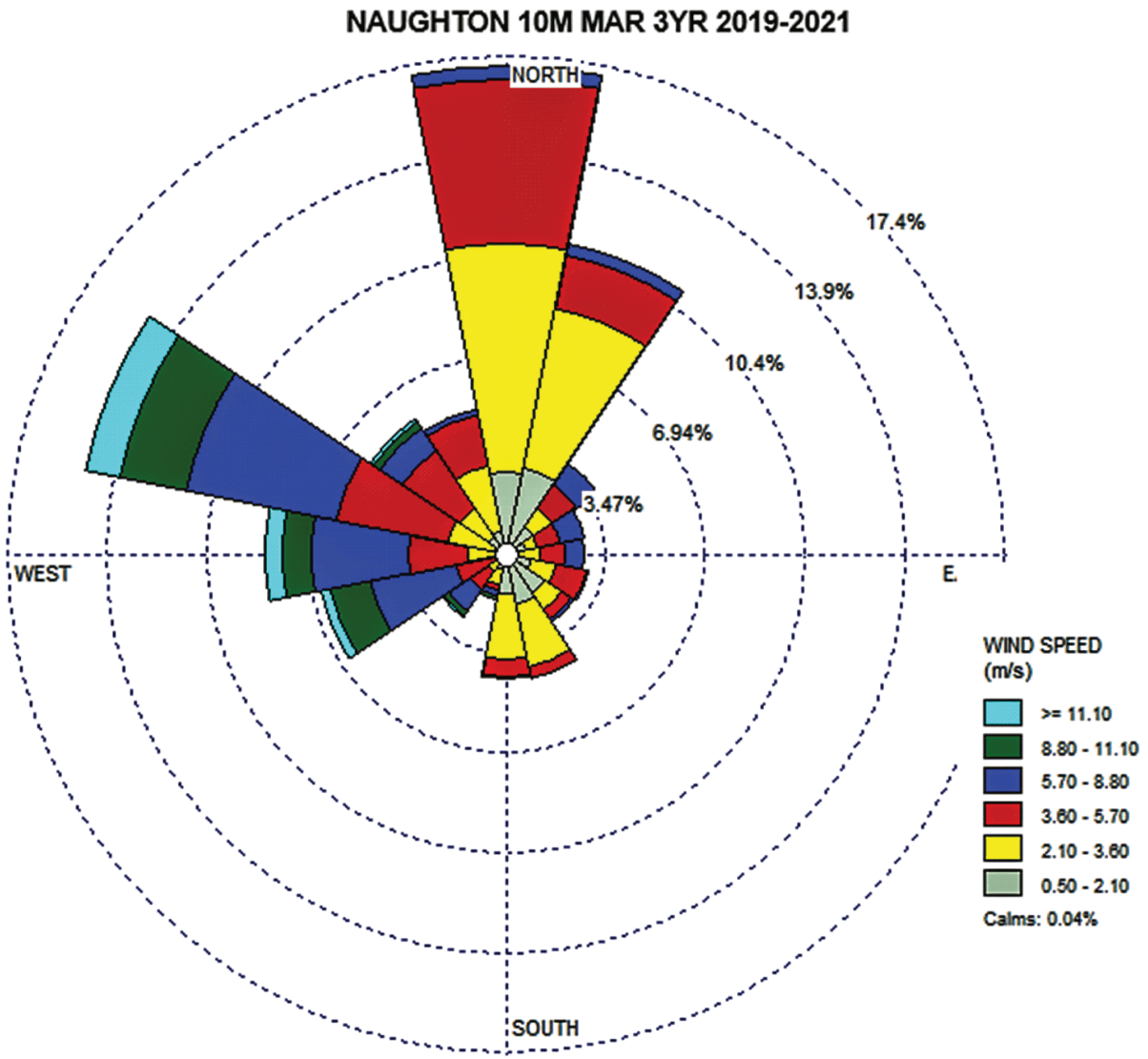






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**Figure 2.4-13 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

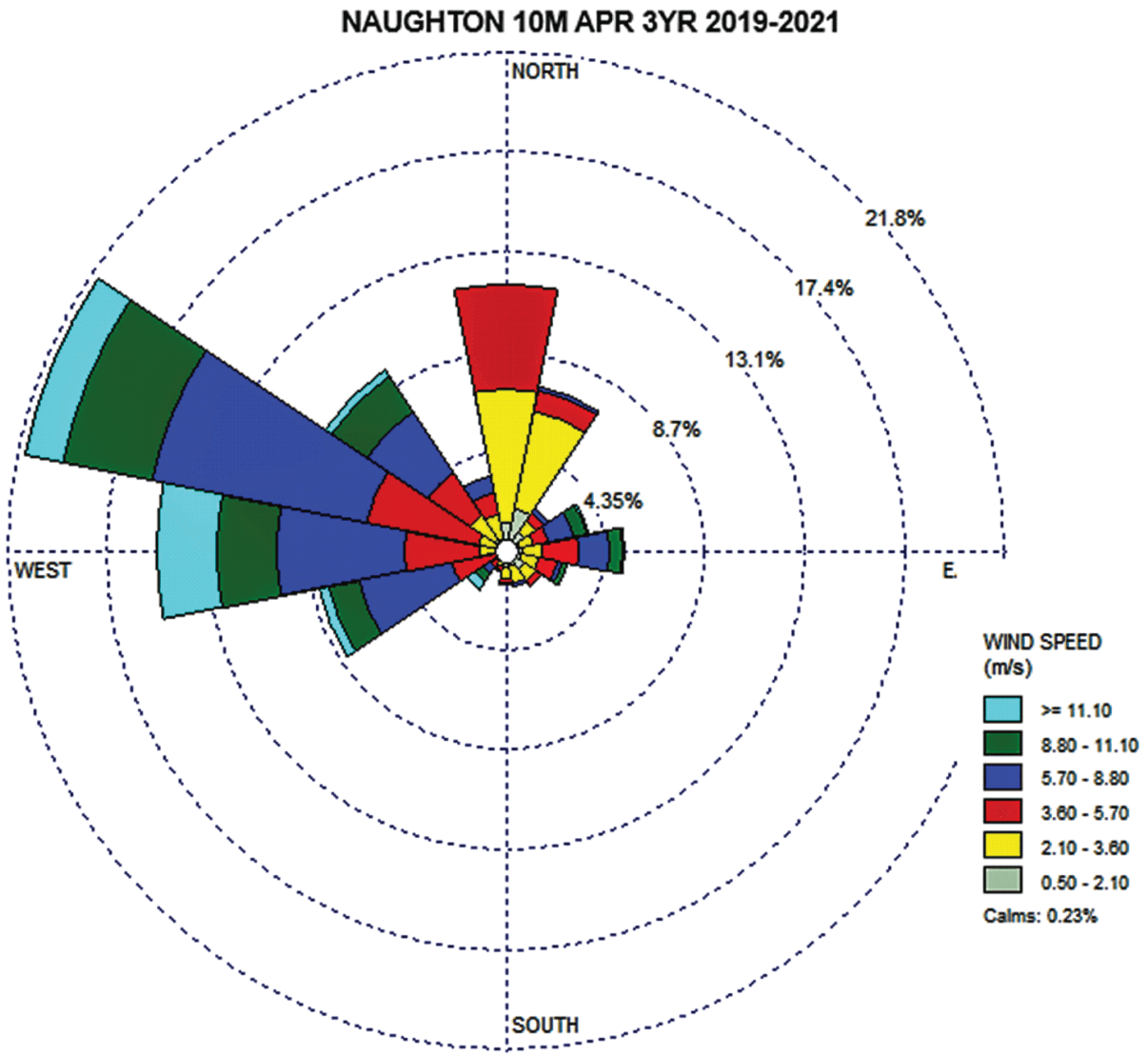






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**Figure 2.4-14 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

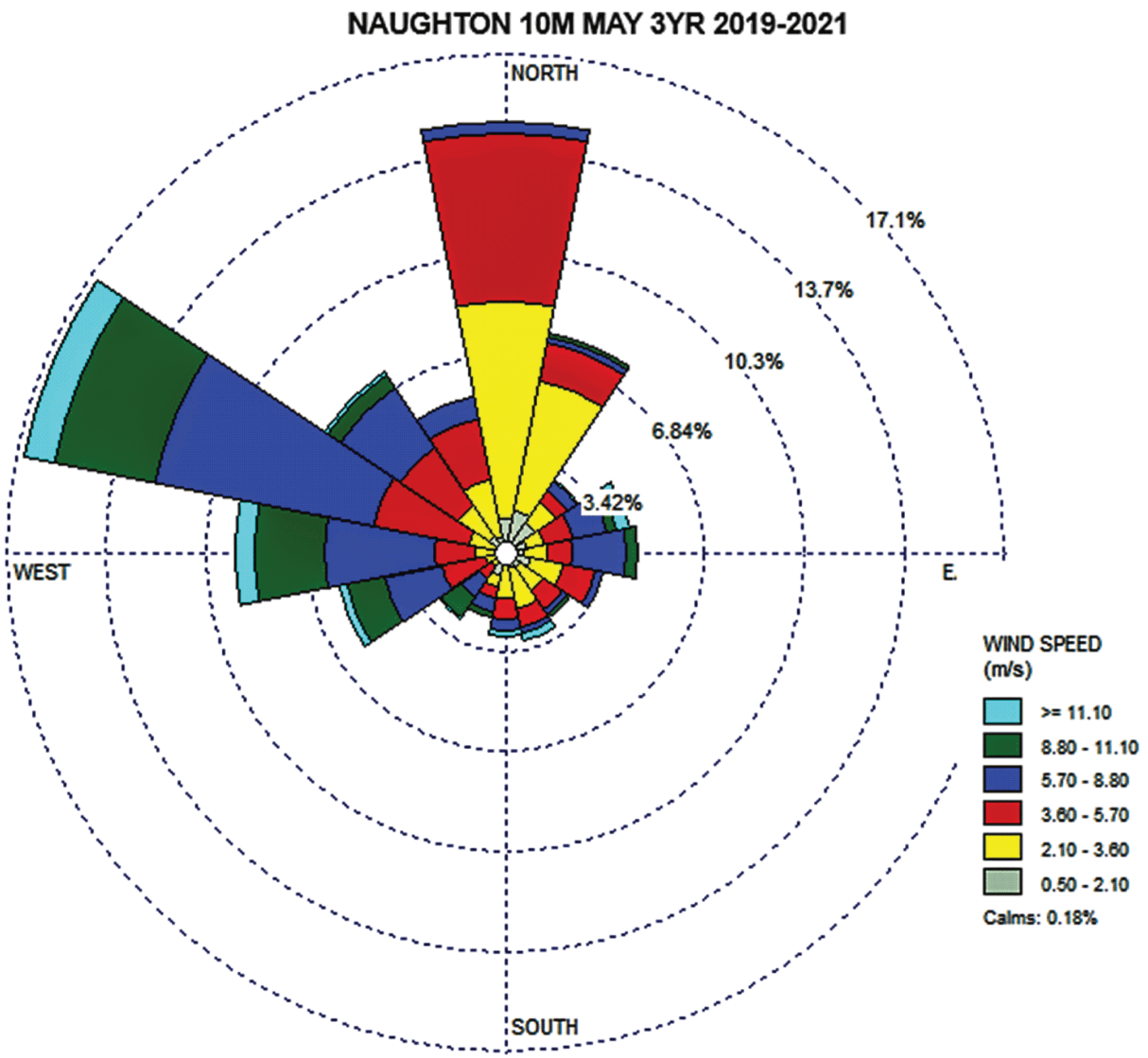




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**Figure 2.4-15 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

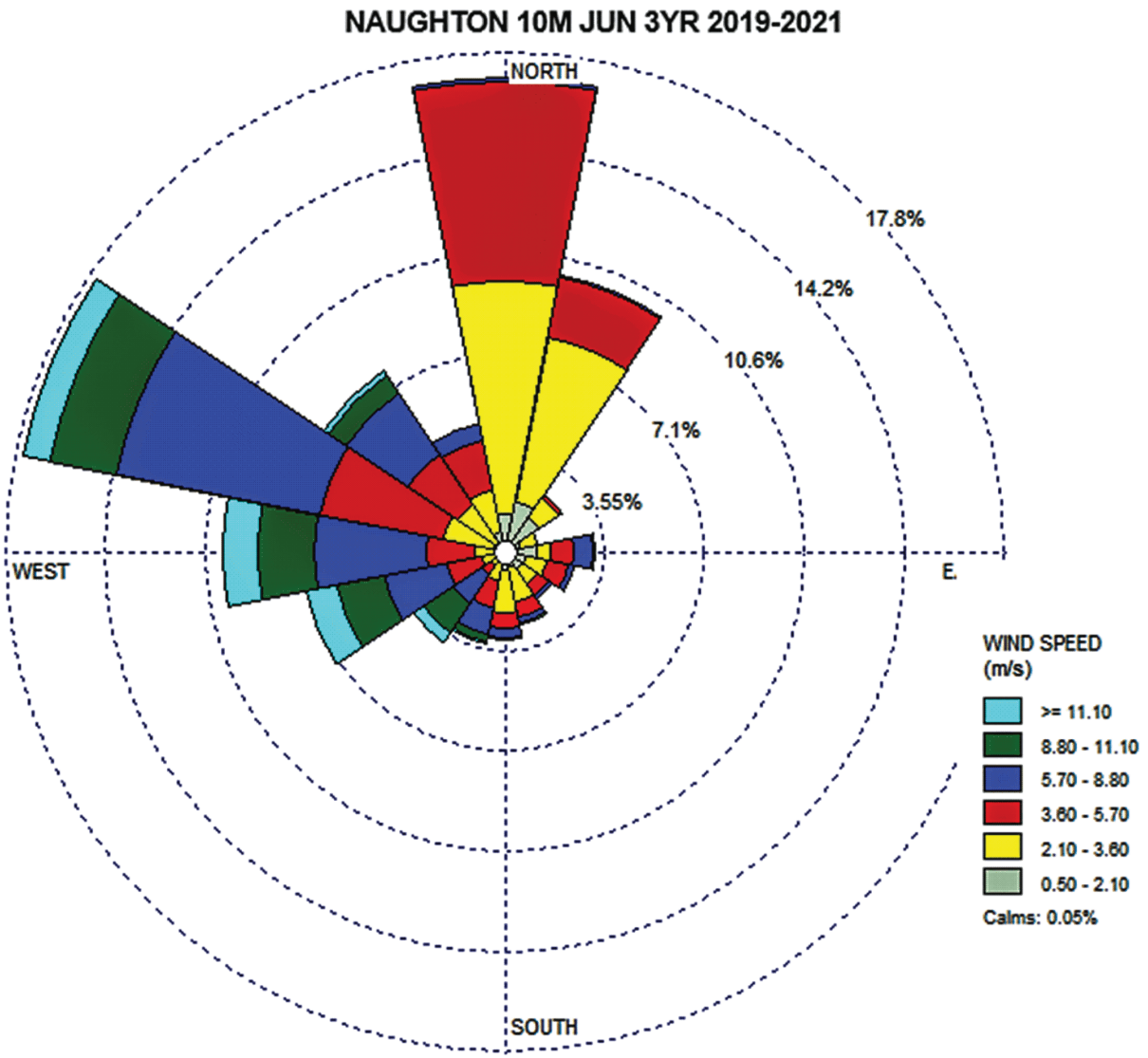






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**Figure 2.4-16 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

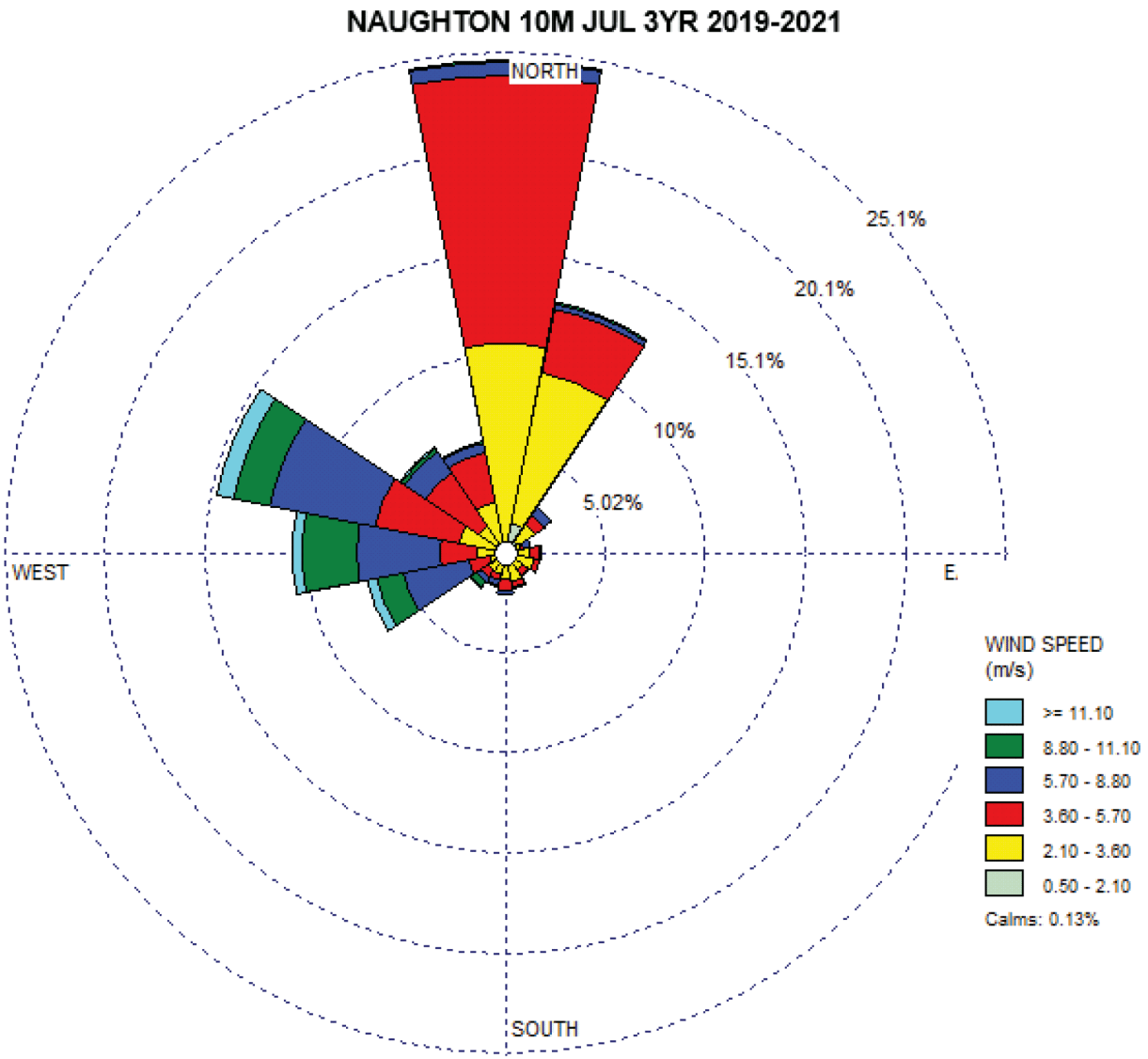






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**Figure 2.4-17 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

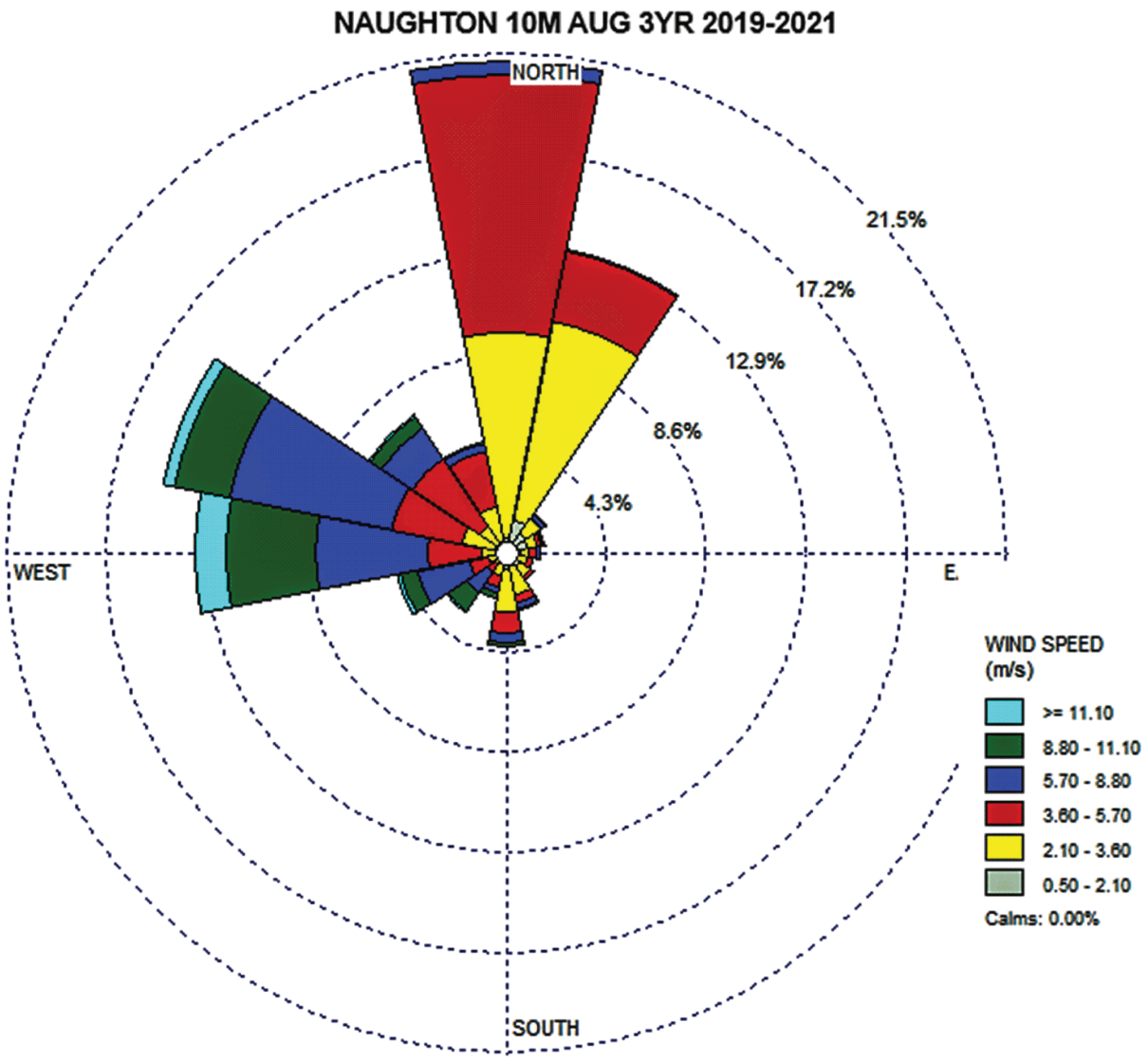






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**Figure 2.4-18 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

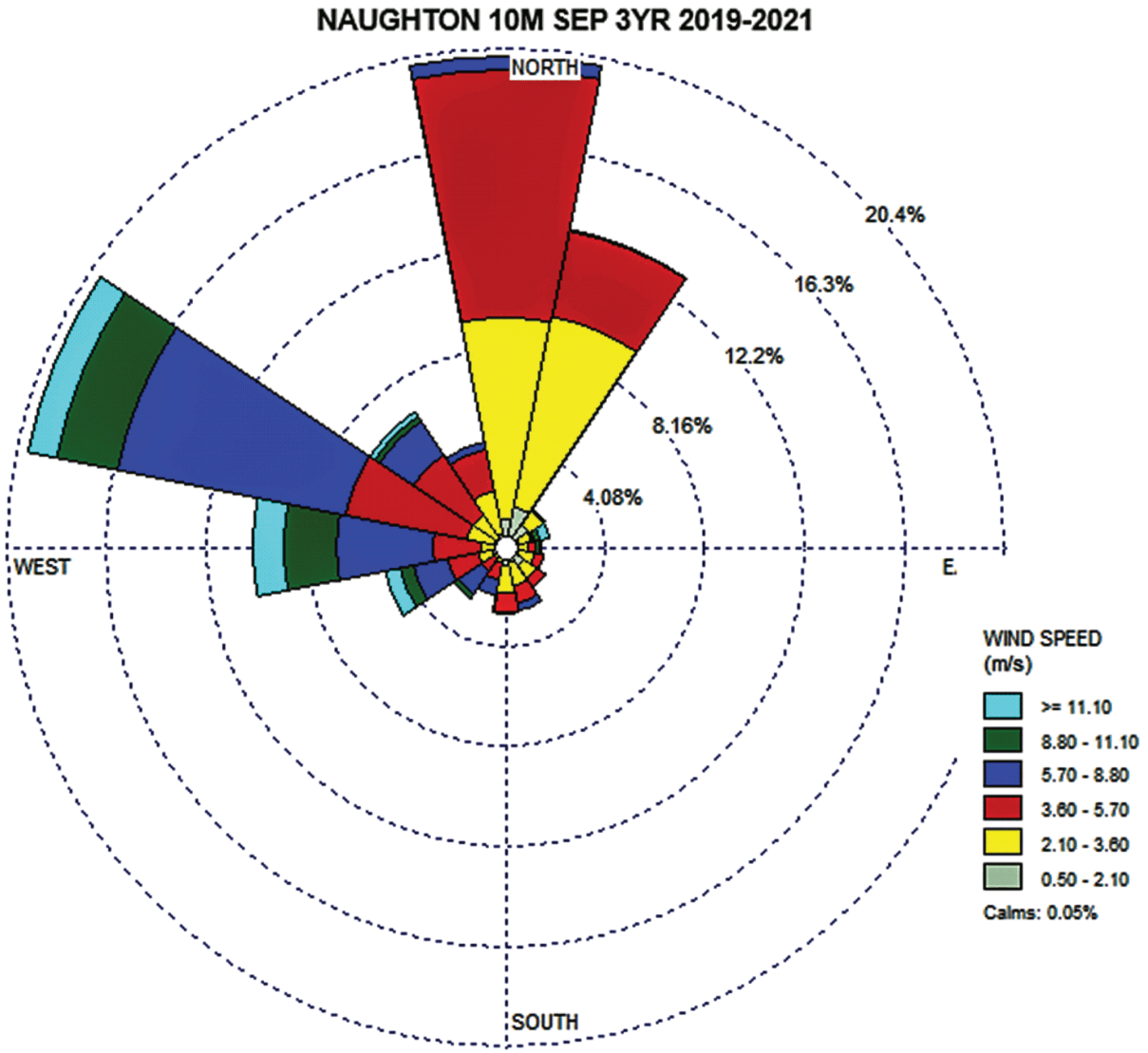


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**Figure 2.4-19 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

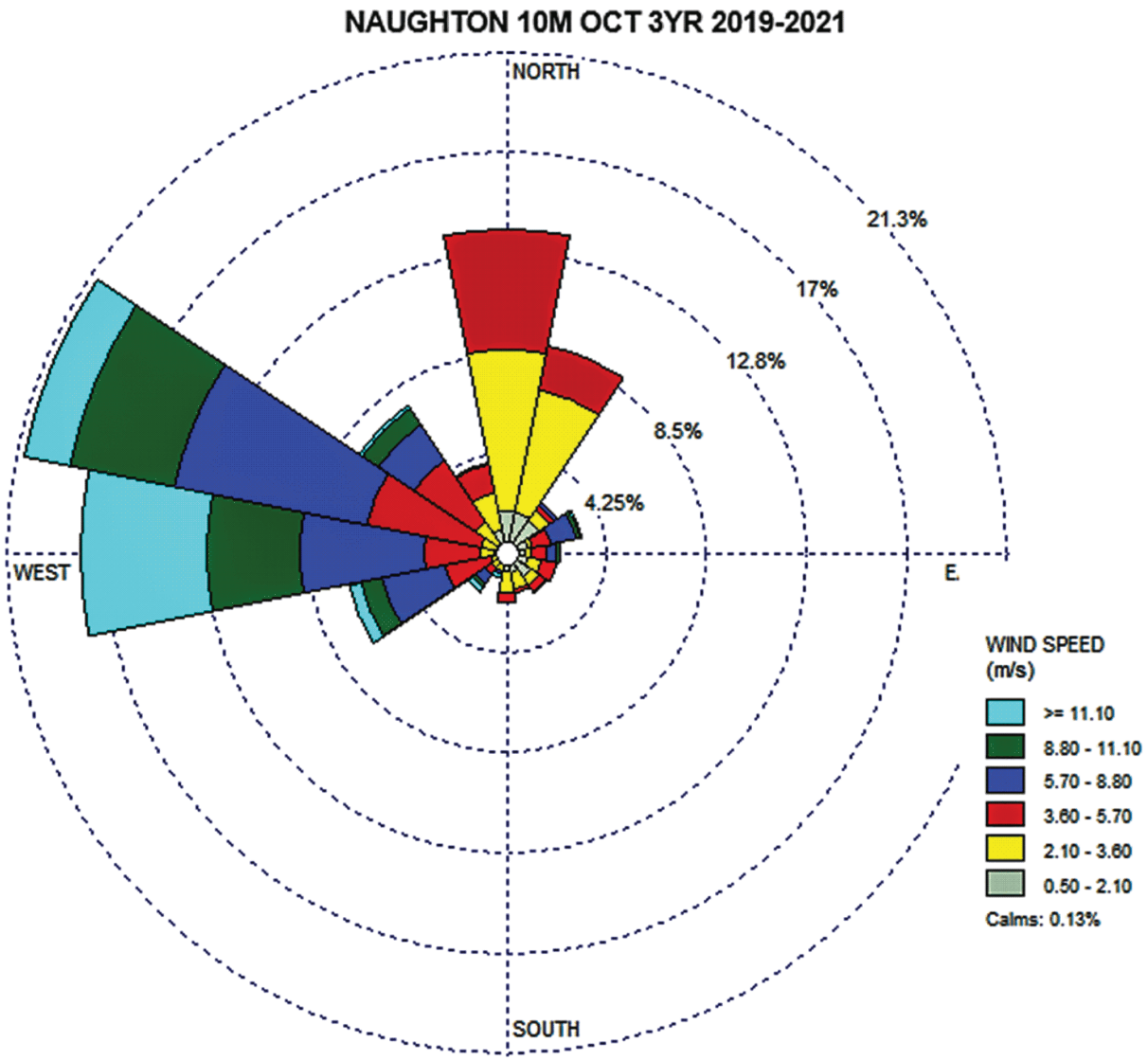






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**Figure 2.4-20 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

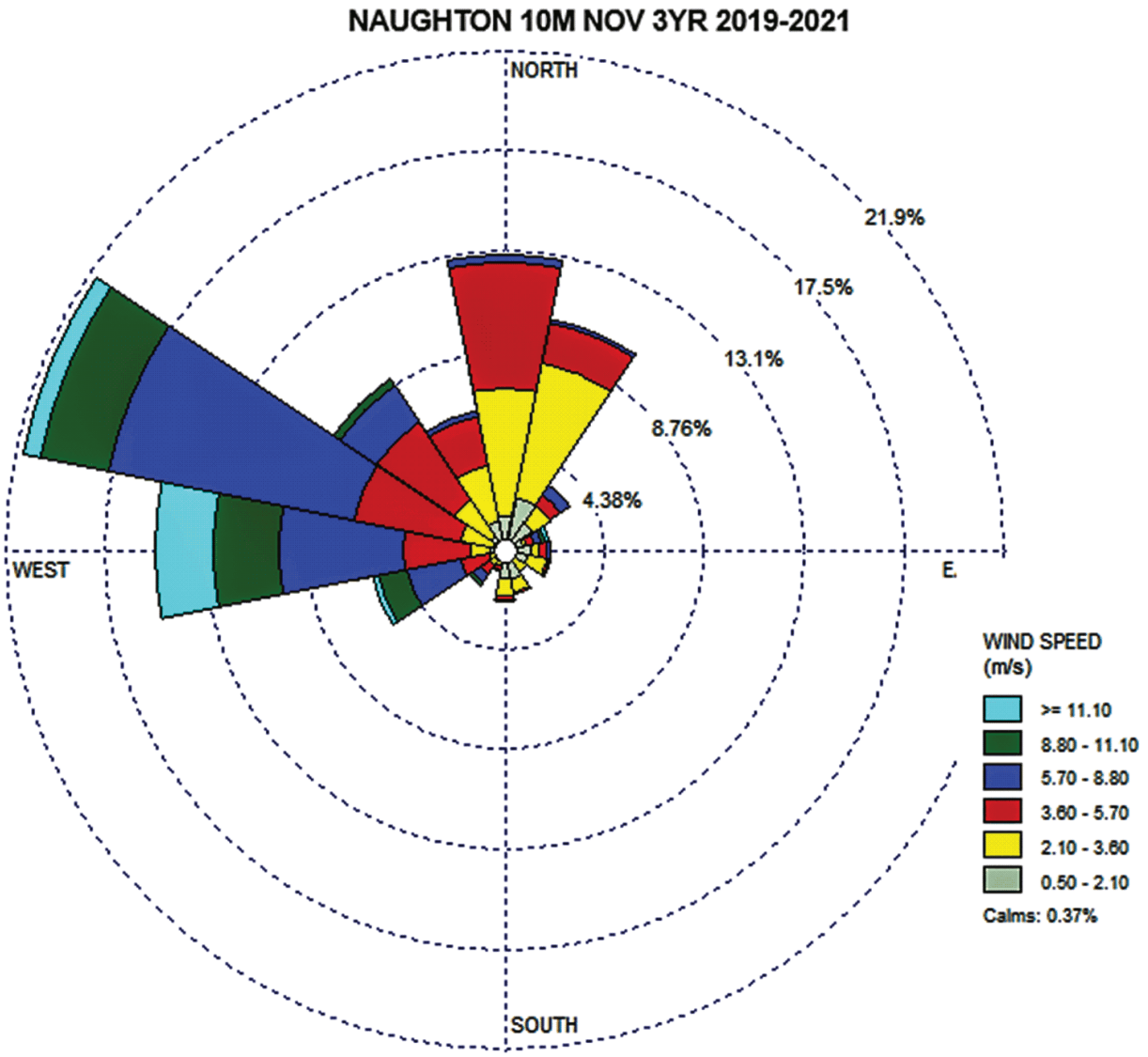


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**Figure 2.4-21 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 10 m**

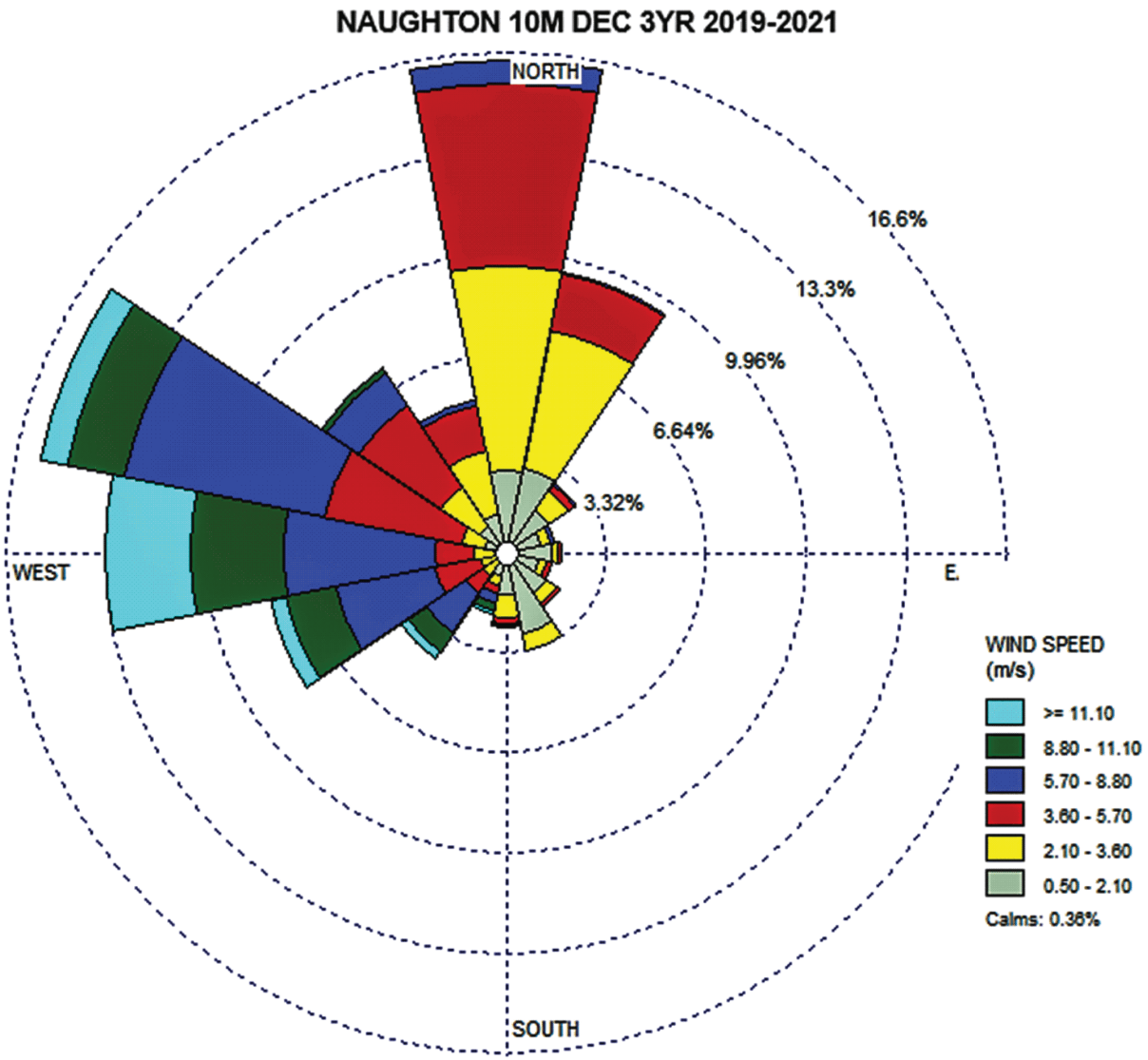






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**Figure 2.4-22 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

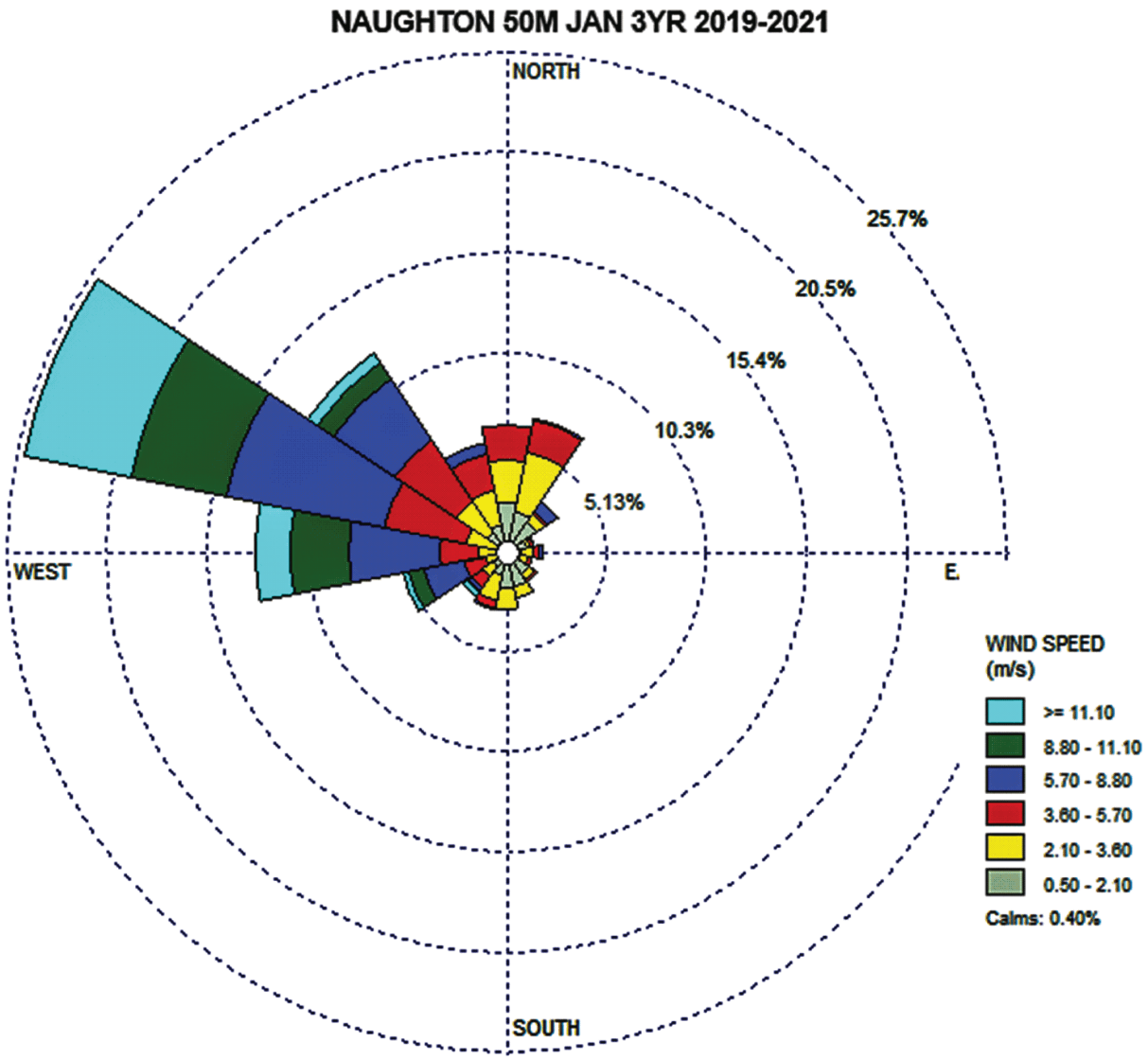






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**Figure 2.4-23 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

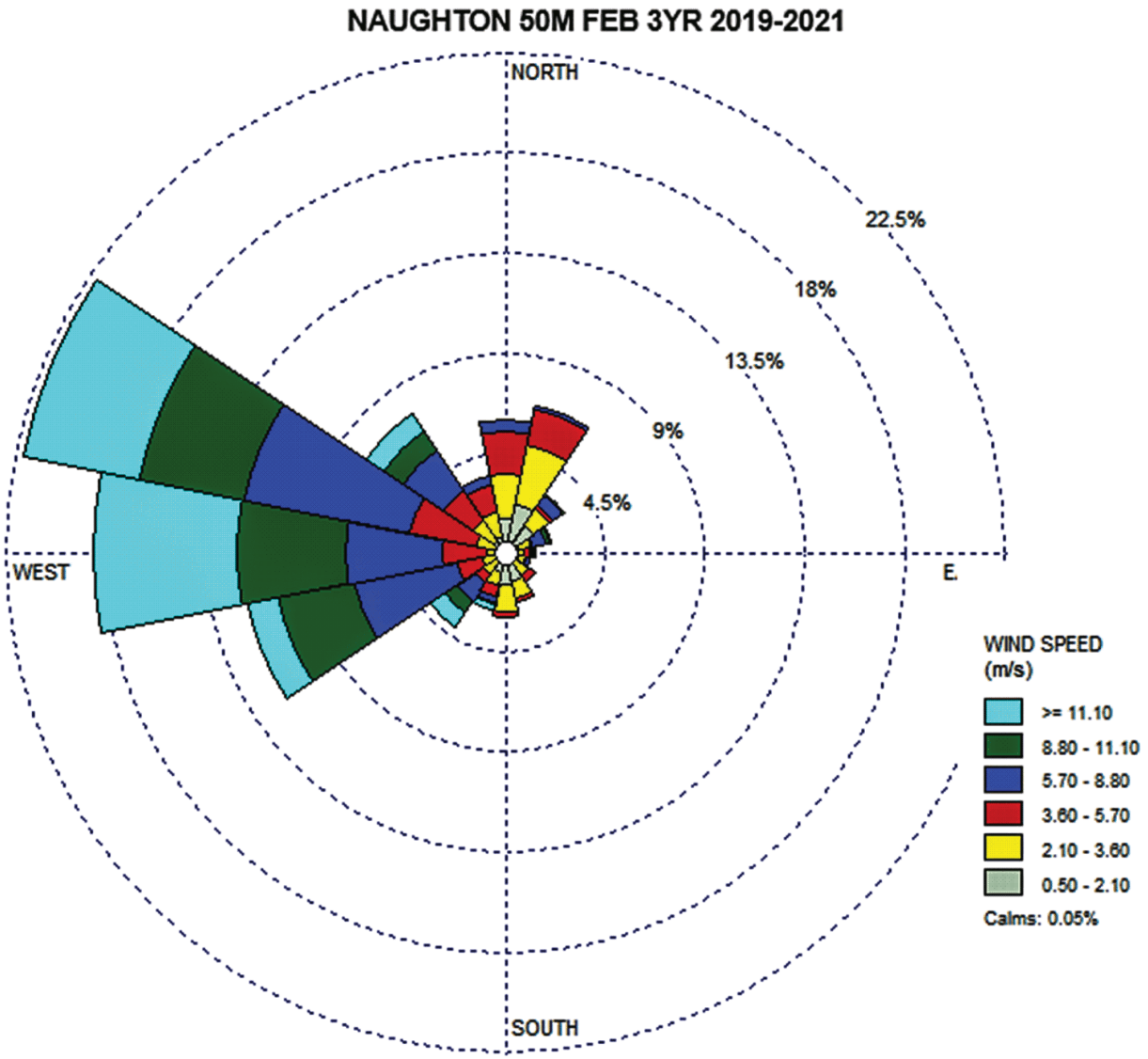






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**Figure 2.4-24 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

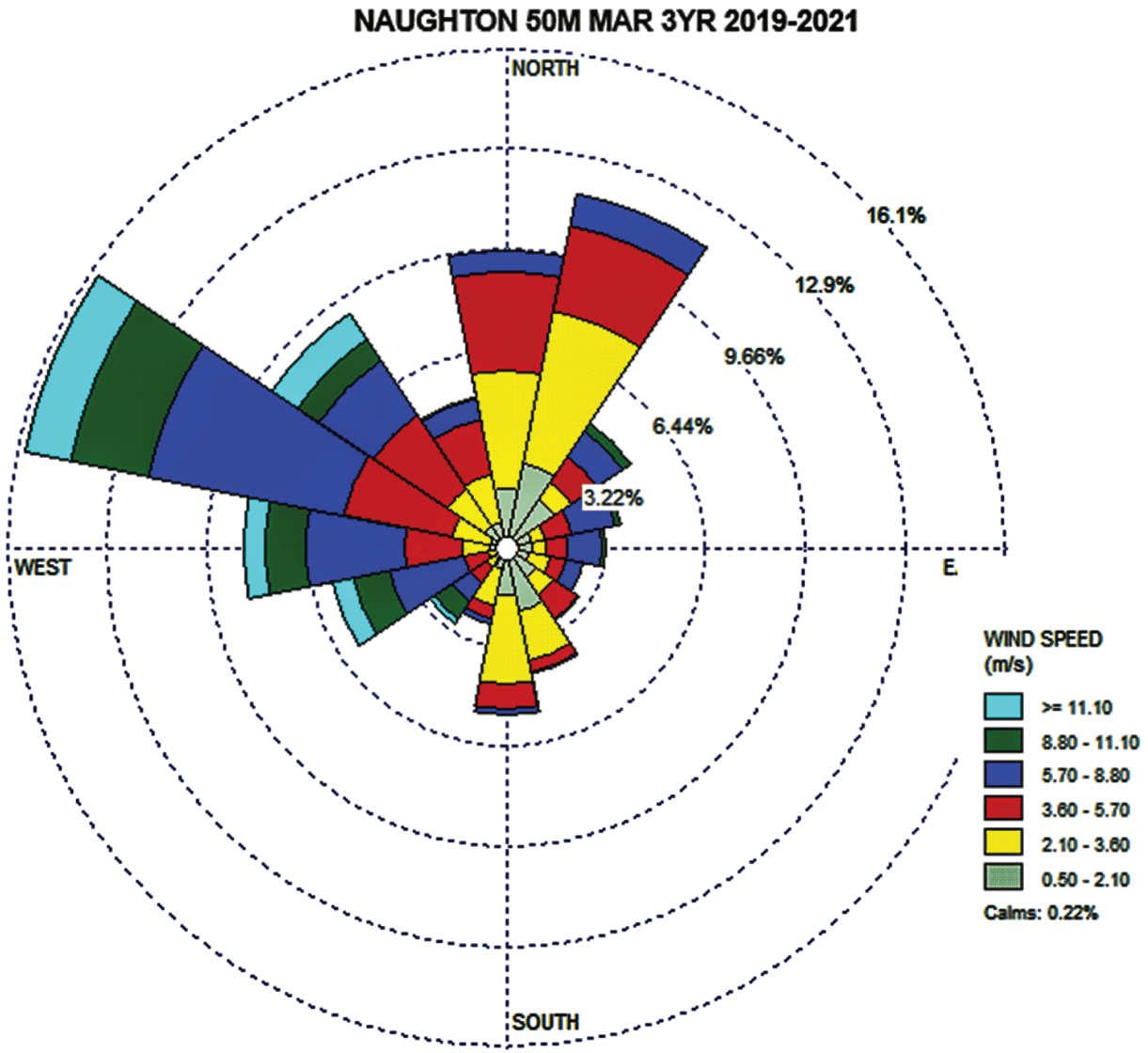






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**Figure 2.4-25 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

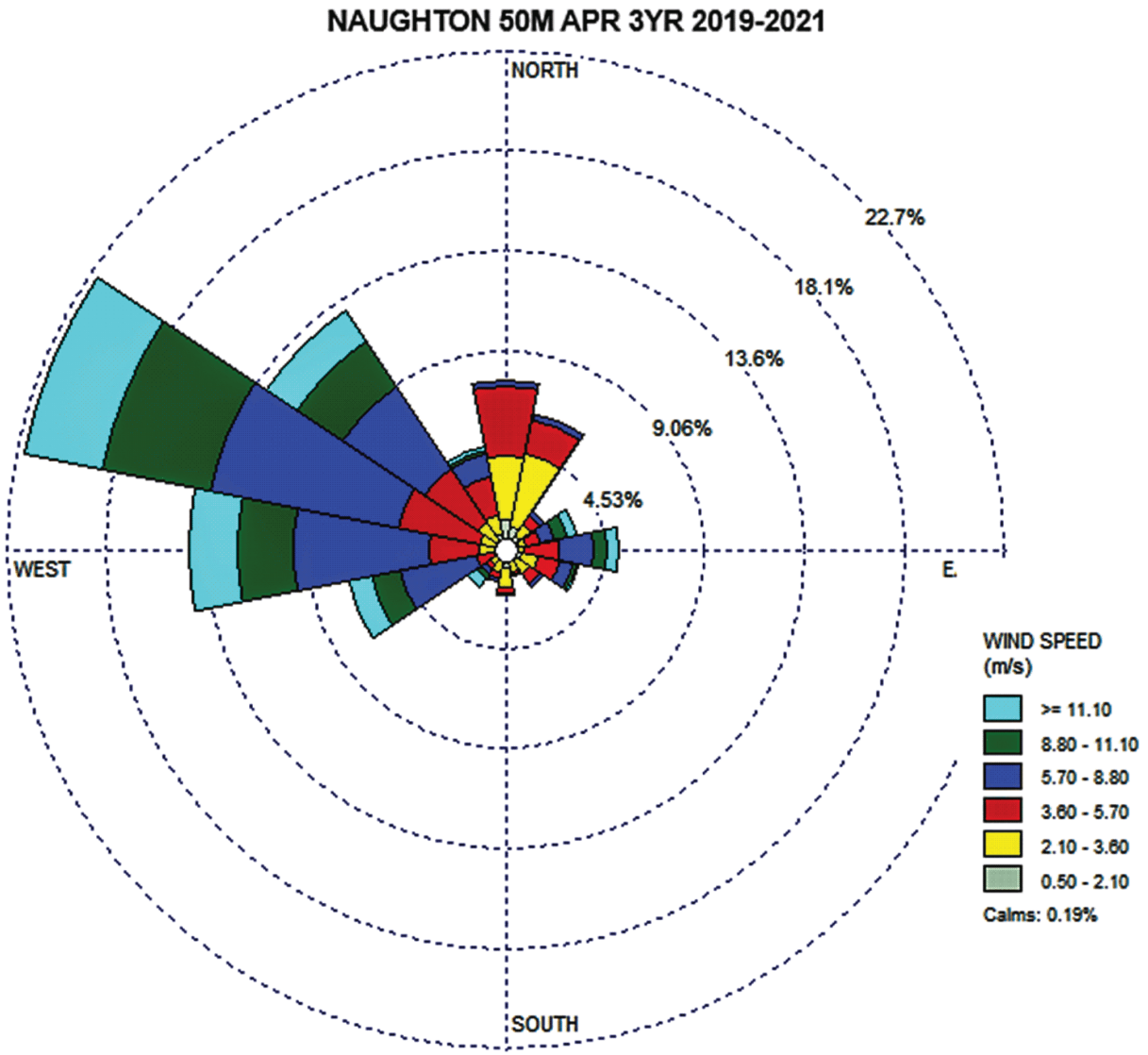






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**Figure 2.4-26 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

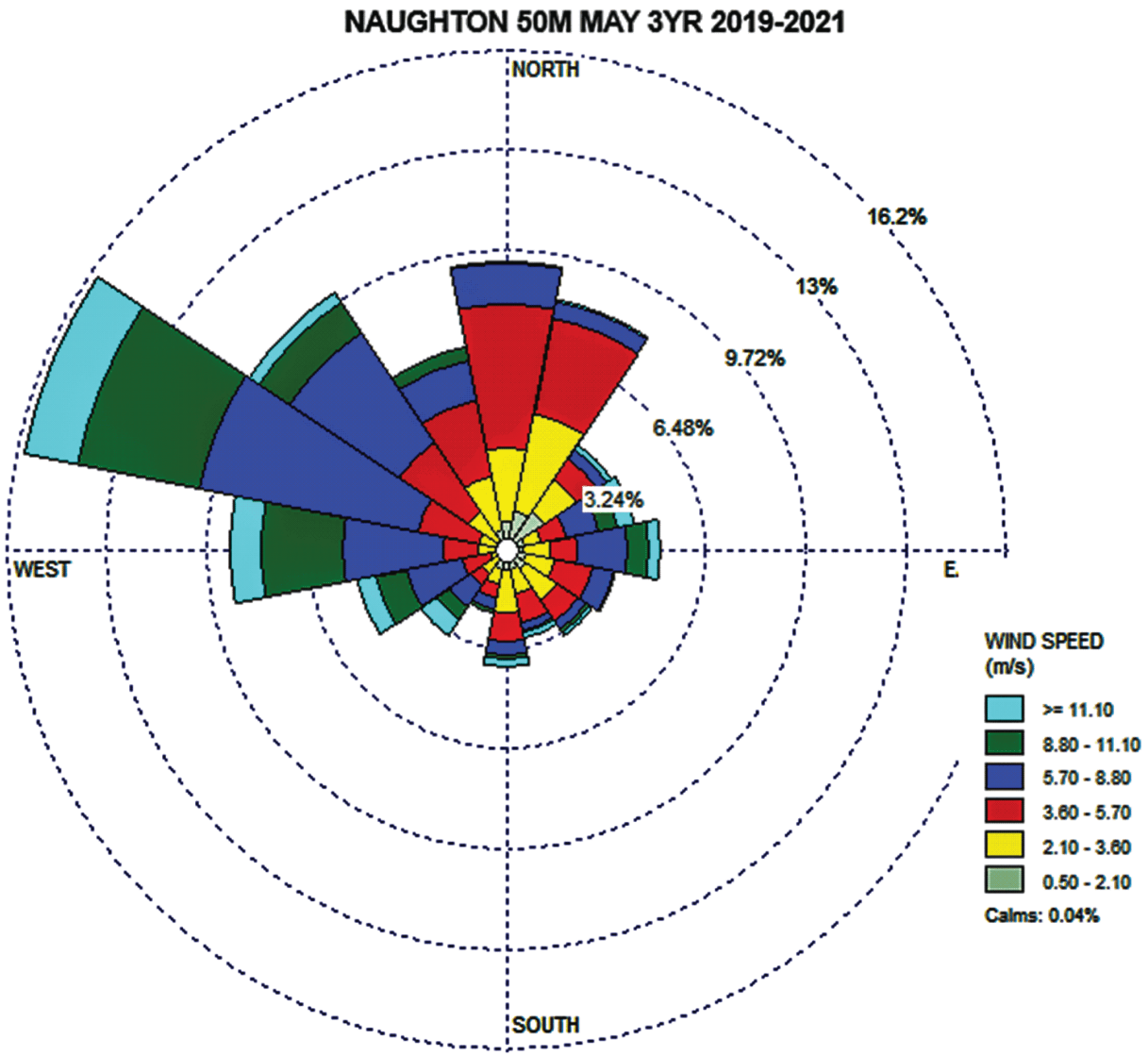






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**Figure 2.4-27 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

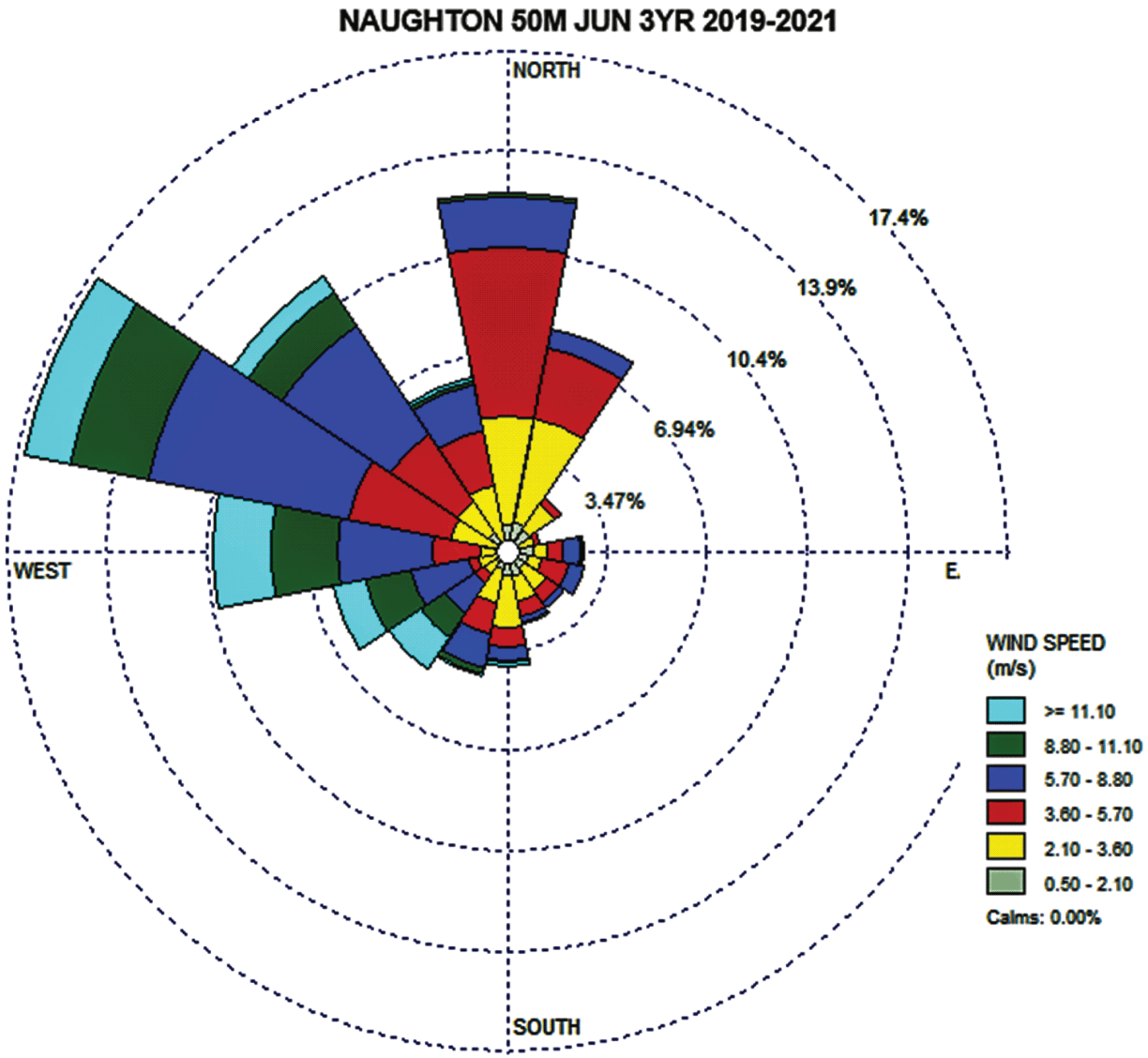






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**Figure 2.4-28 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

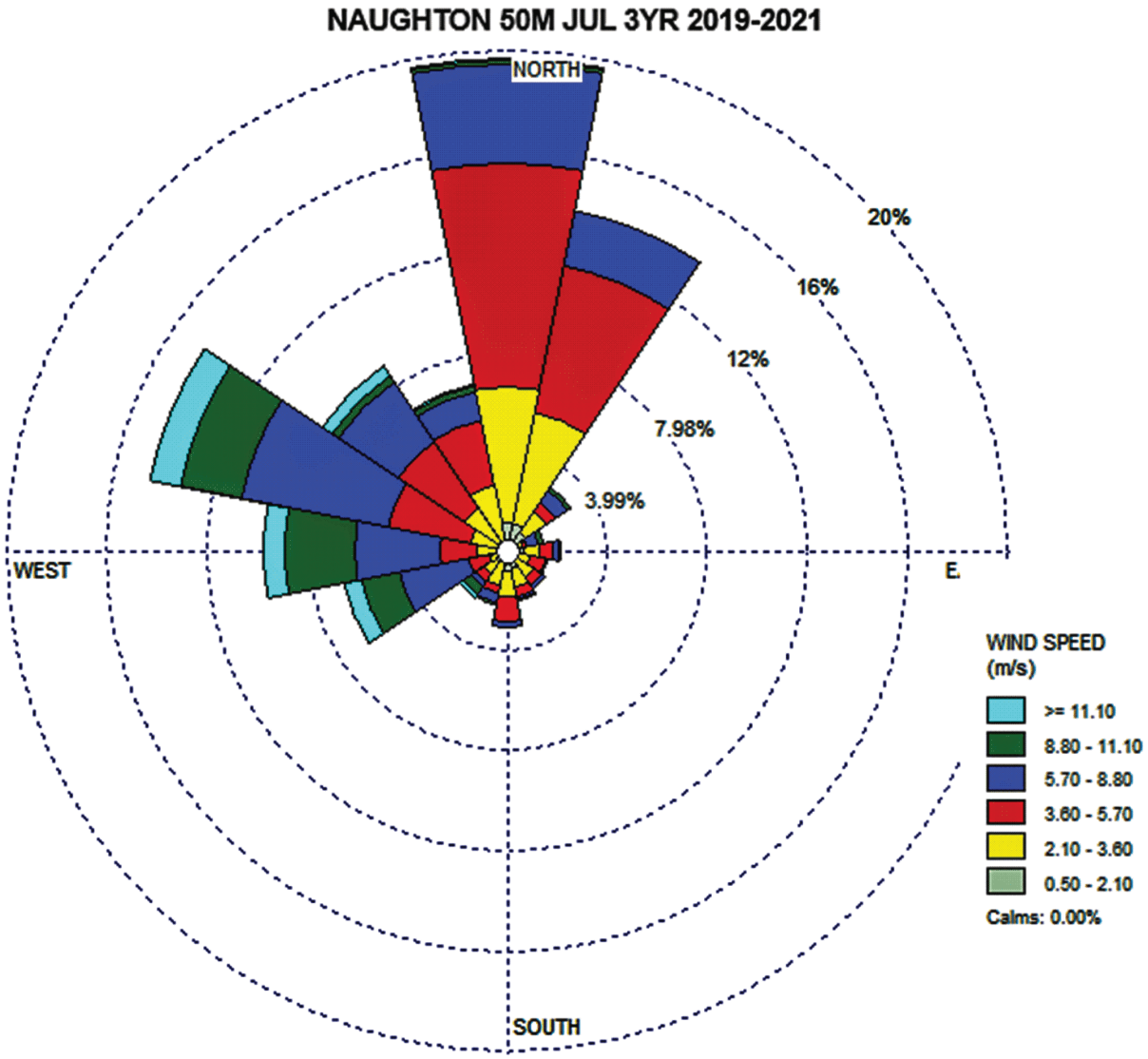






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**Figure 2.4-29 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

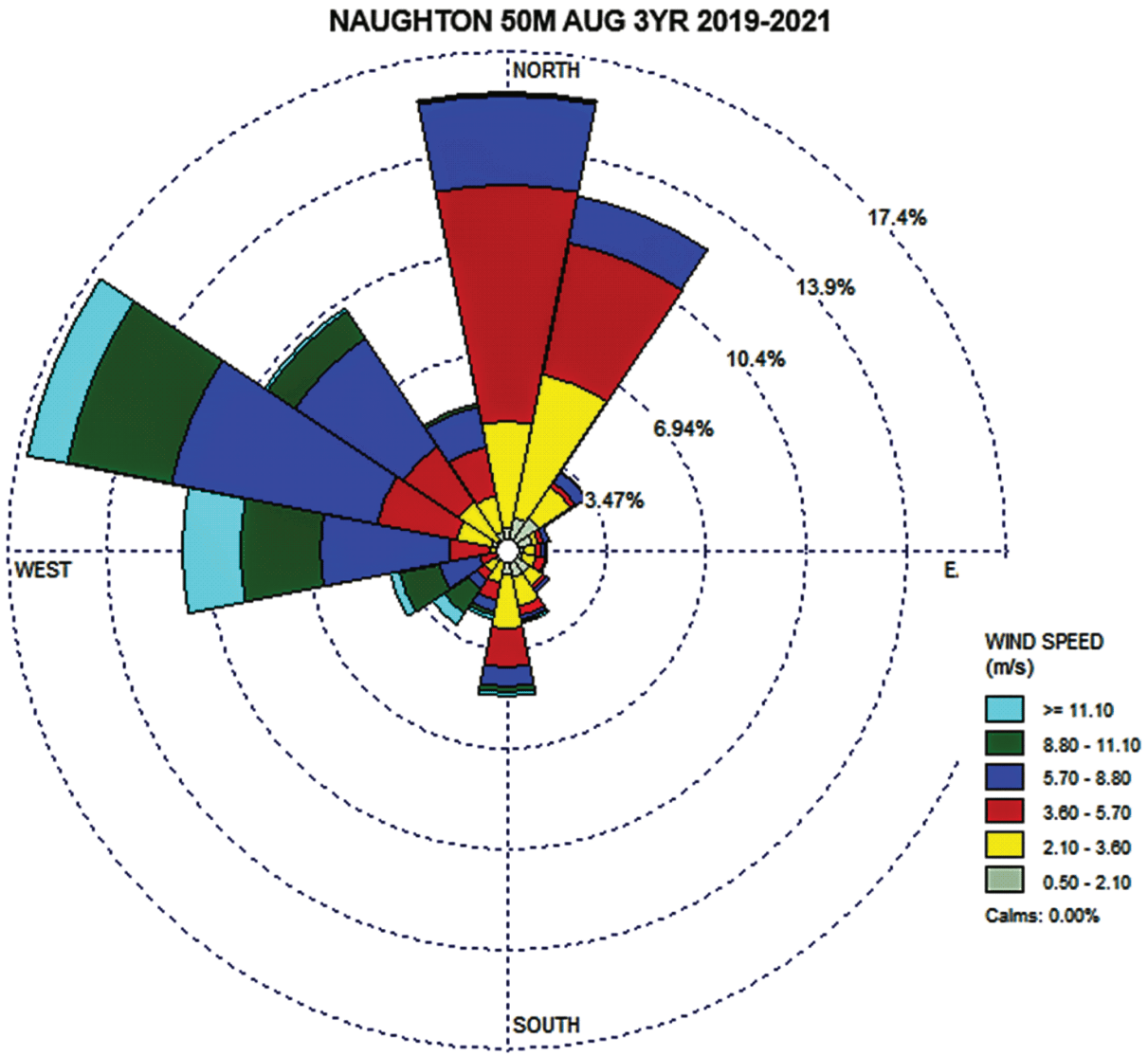






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**Figure 2.4-30 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

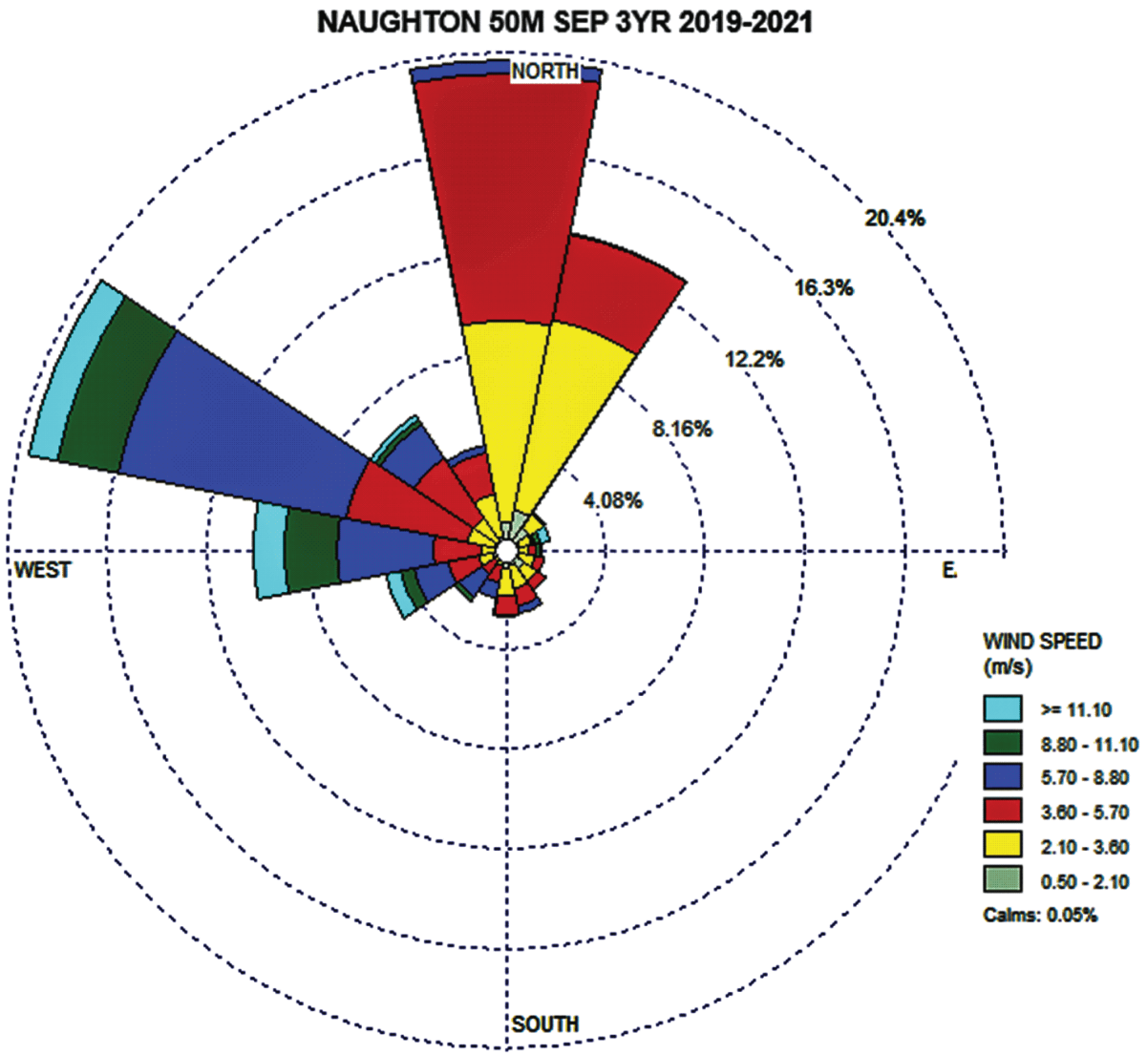






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**Figure 2.4-31 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

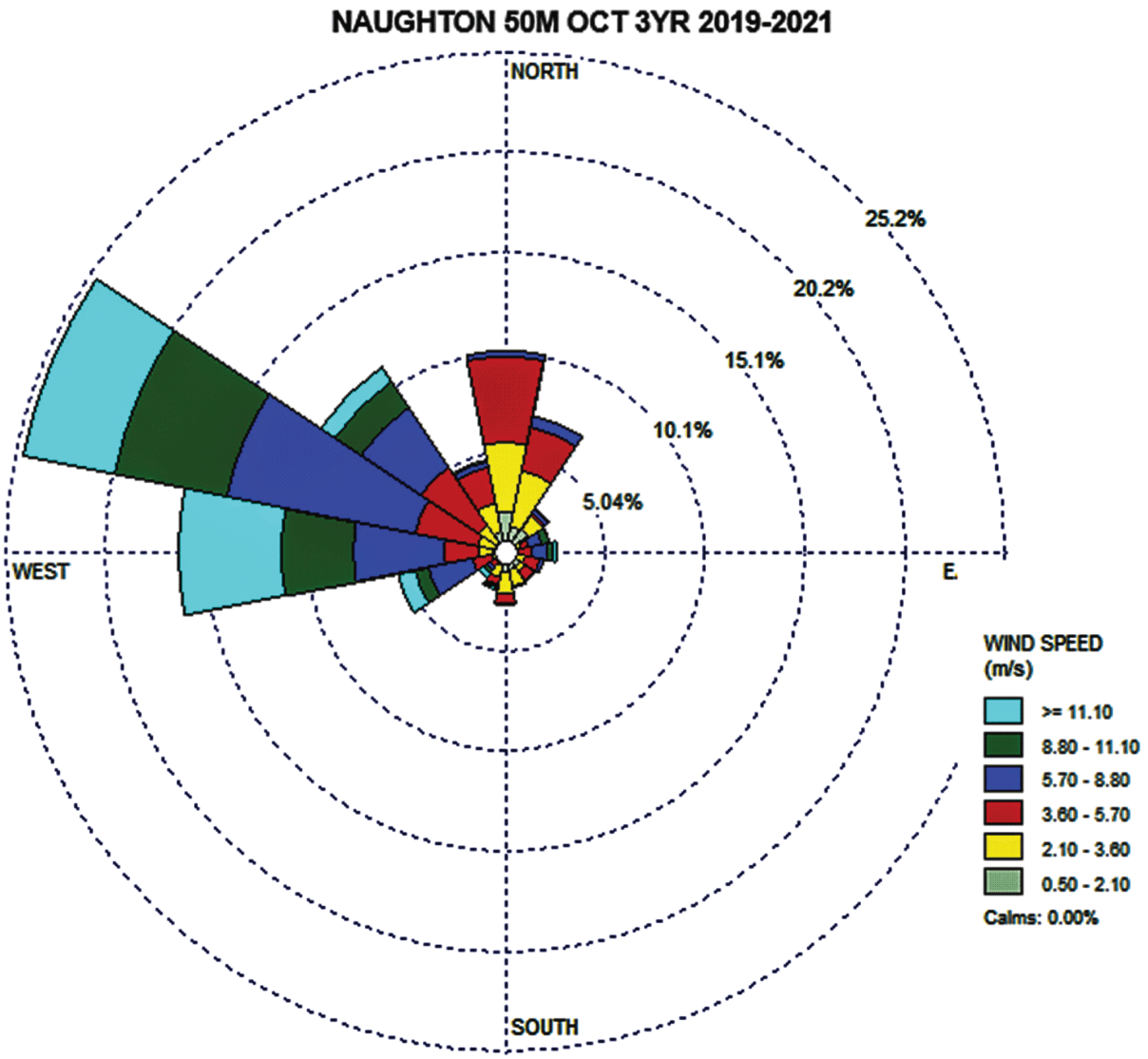






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**Figure 2.4-32 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

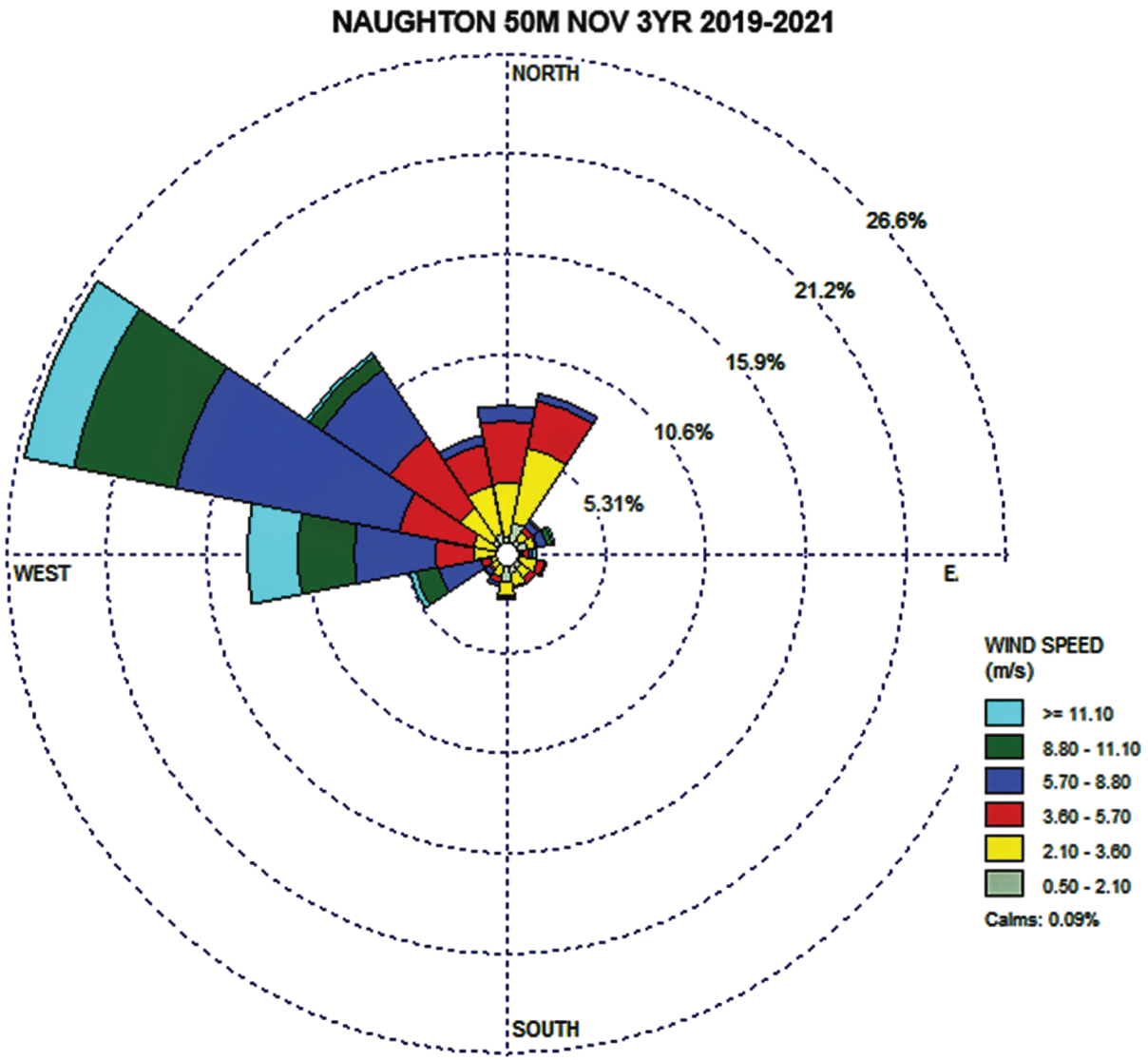






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**Figure 2.4-33 Naughton Power Plant Monthly Composite Wind Rose 2019–2021, 50 m**

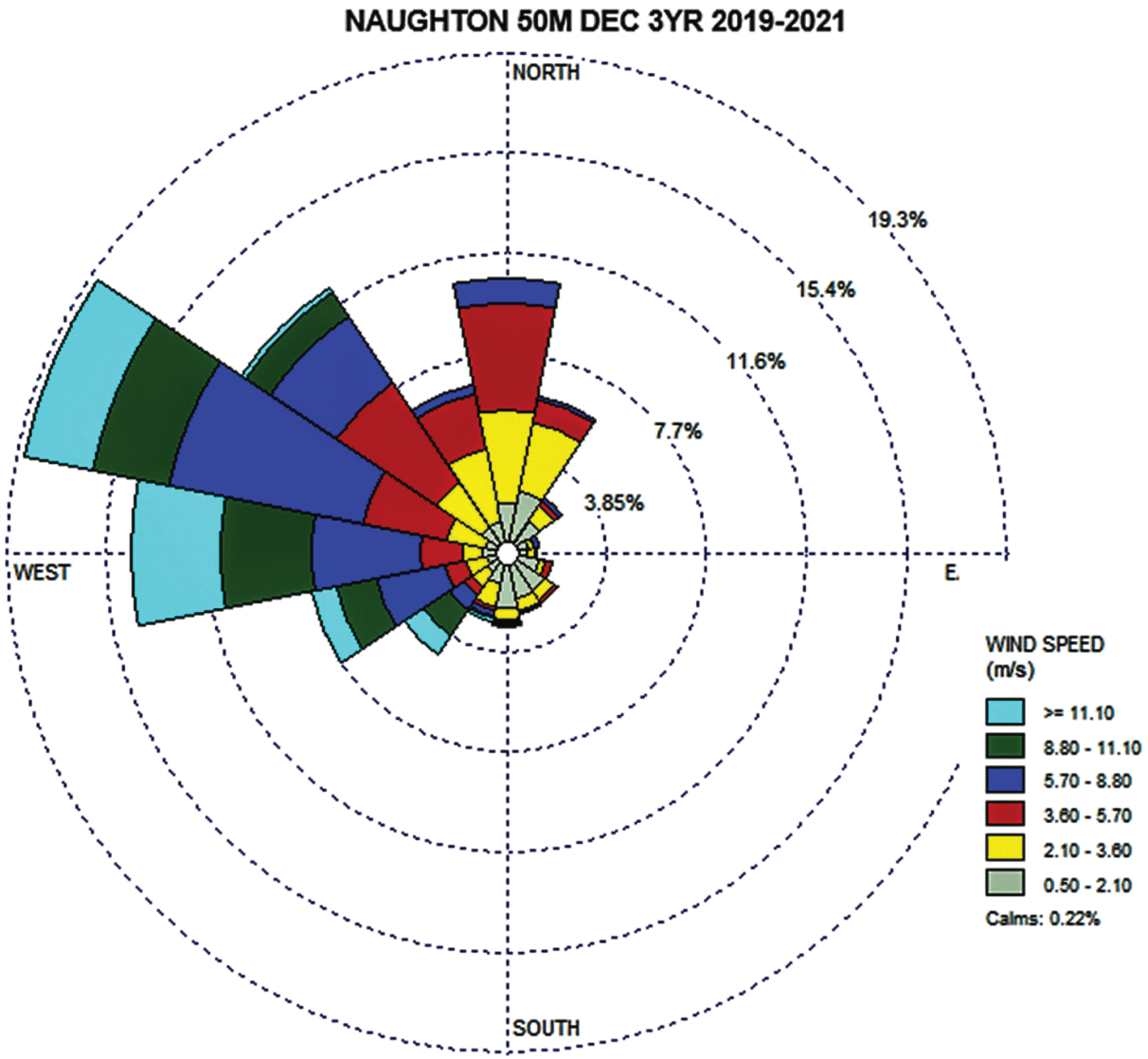






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**Figure 2.4-34 Naughton Power Plant Annual Average Wind Rose 2022–2023, 10 m**

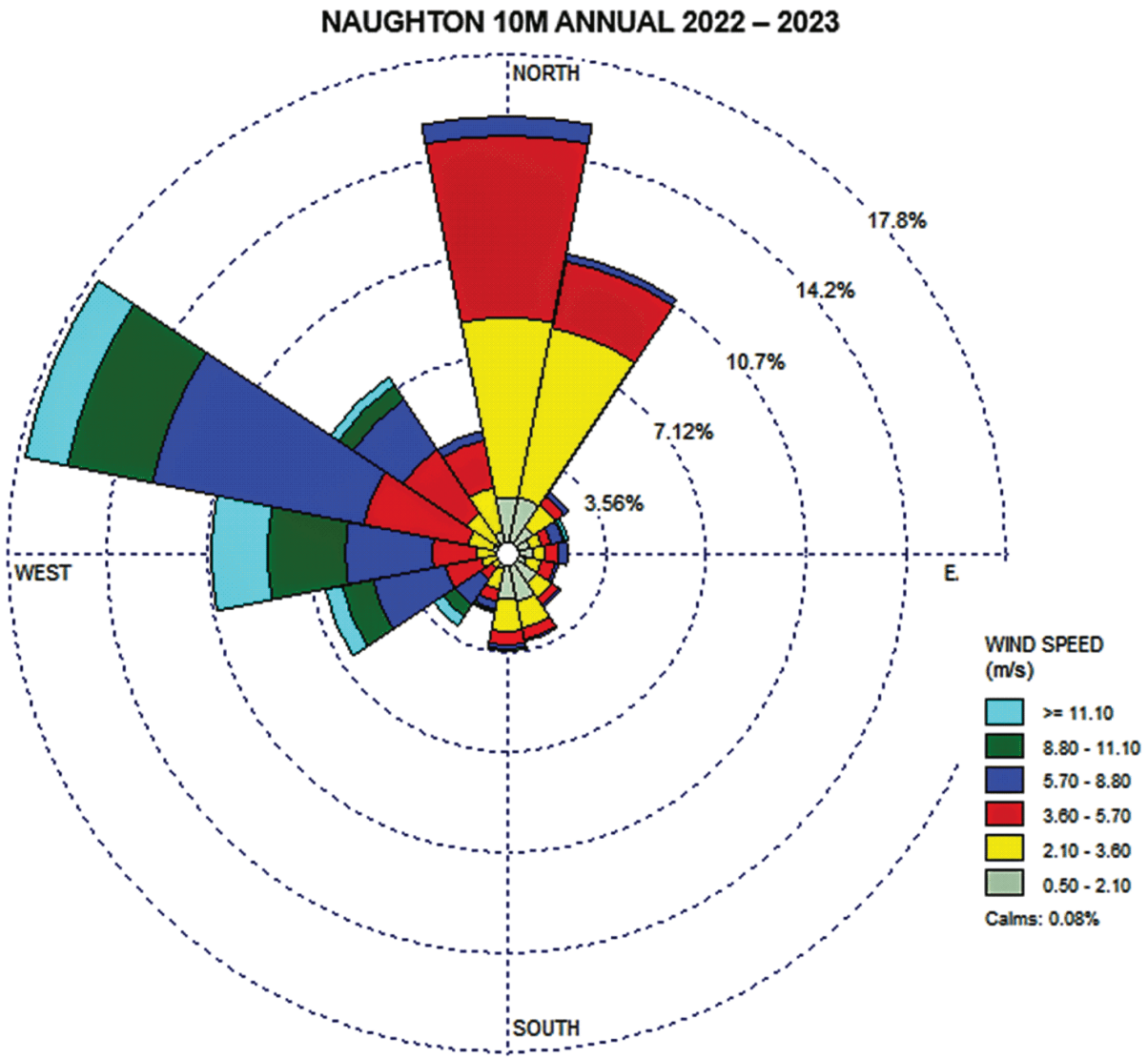






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**Figure 2.4-35 Naughton Power Plant Annual Average Wind Rose 2022–2023, 50 m**

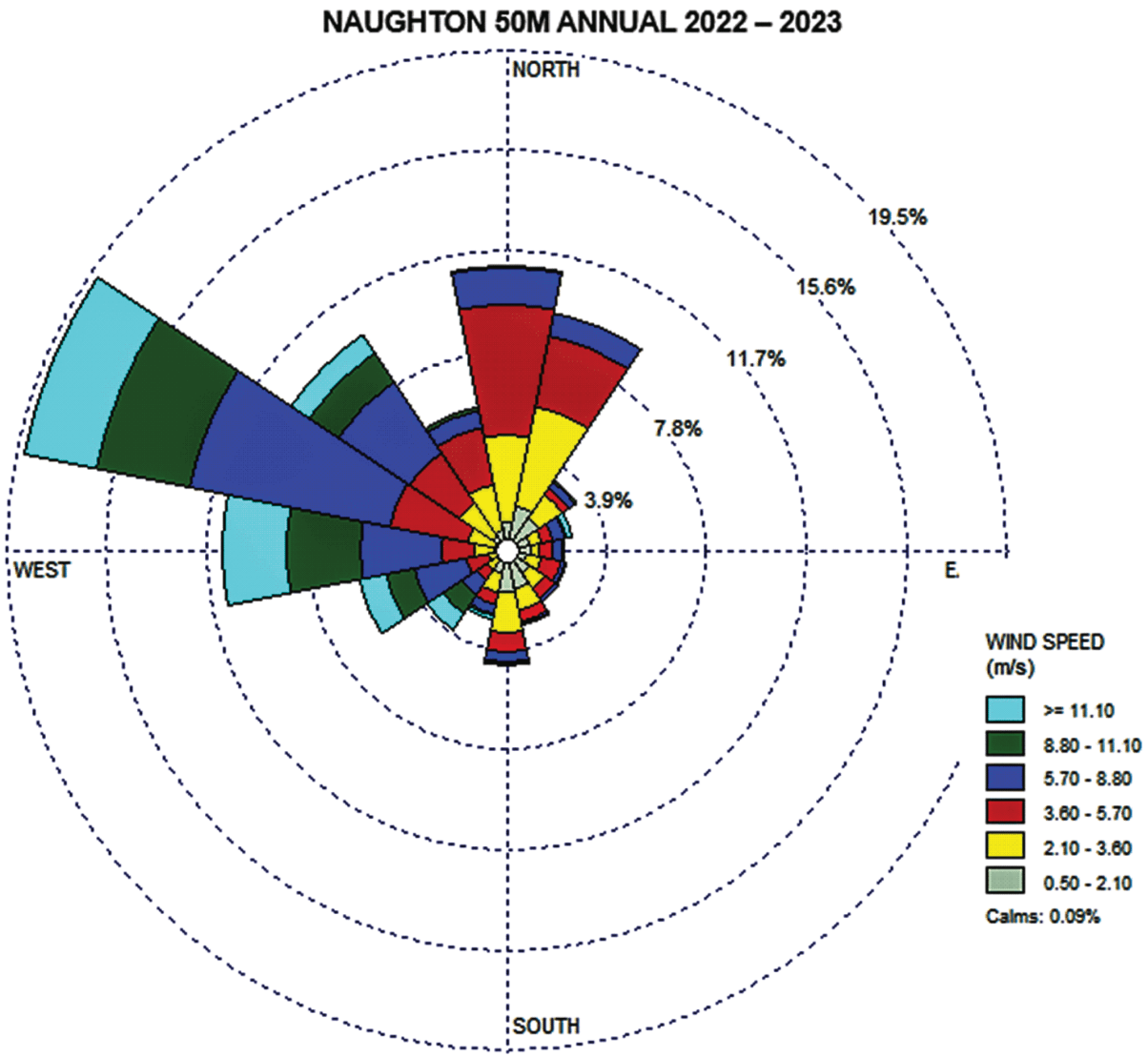






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**Figure 2.4-36 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 10 m**

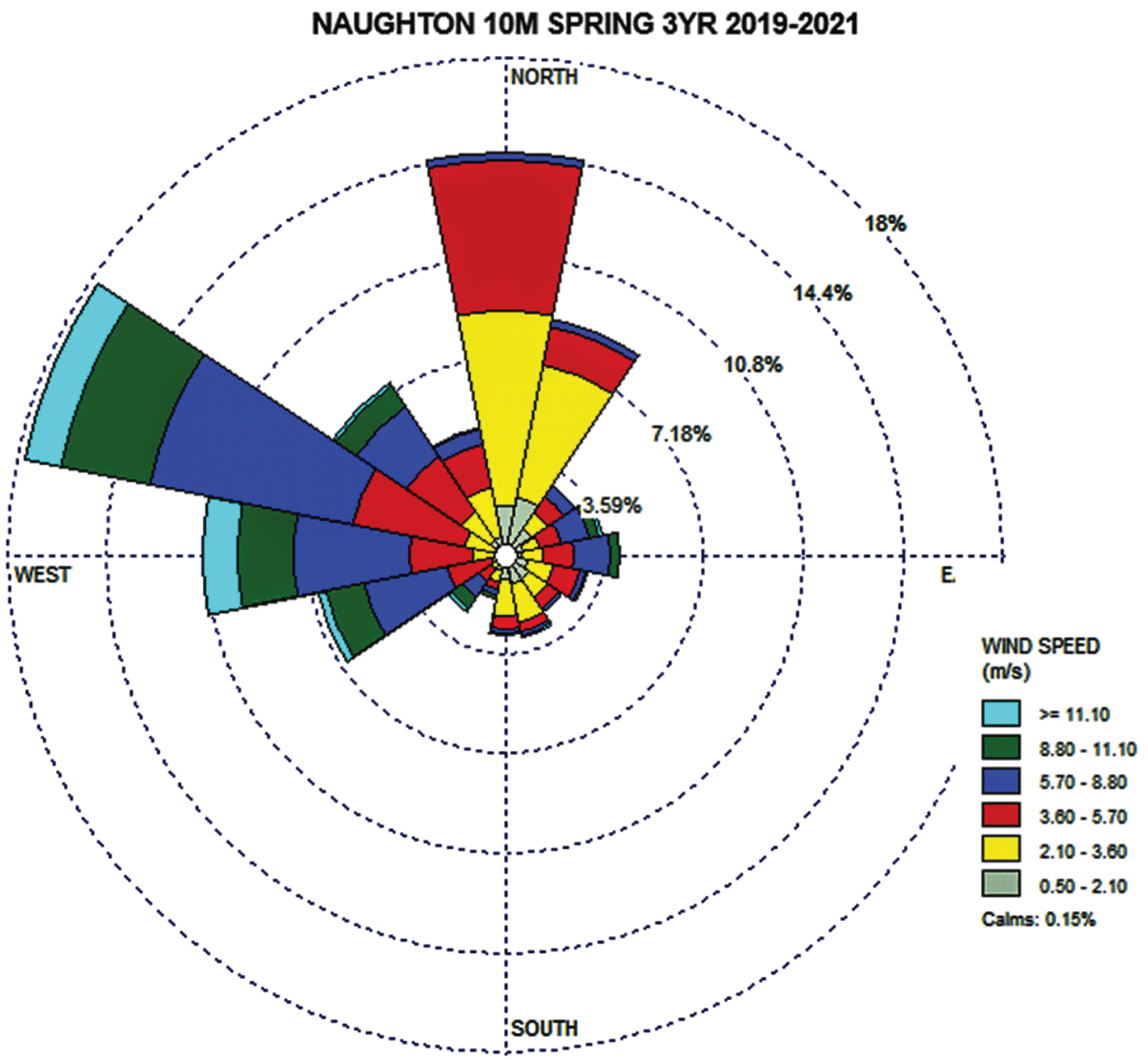






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**Figure 2.4-37 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 10 m**

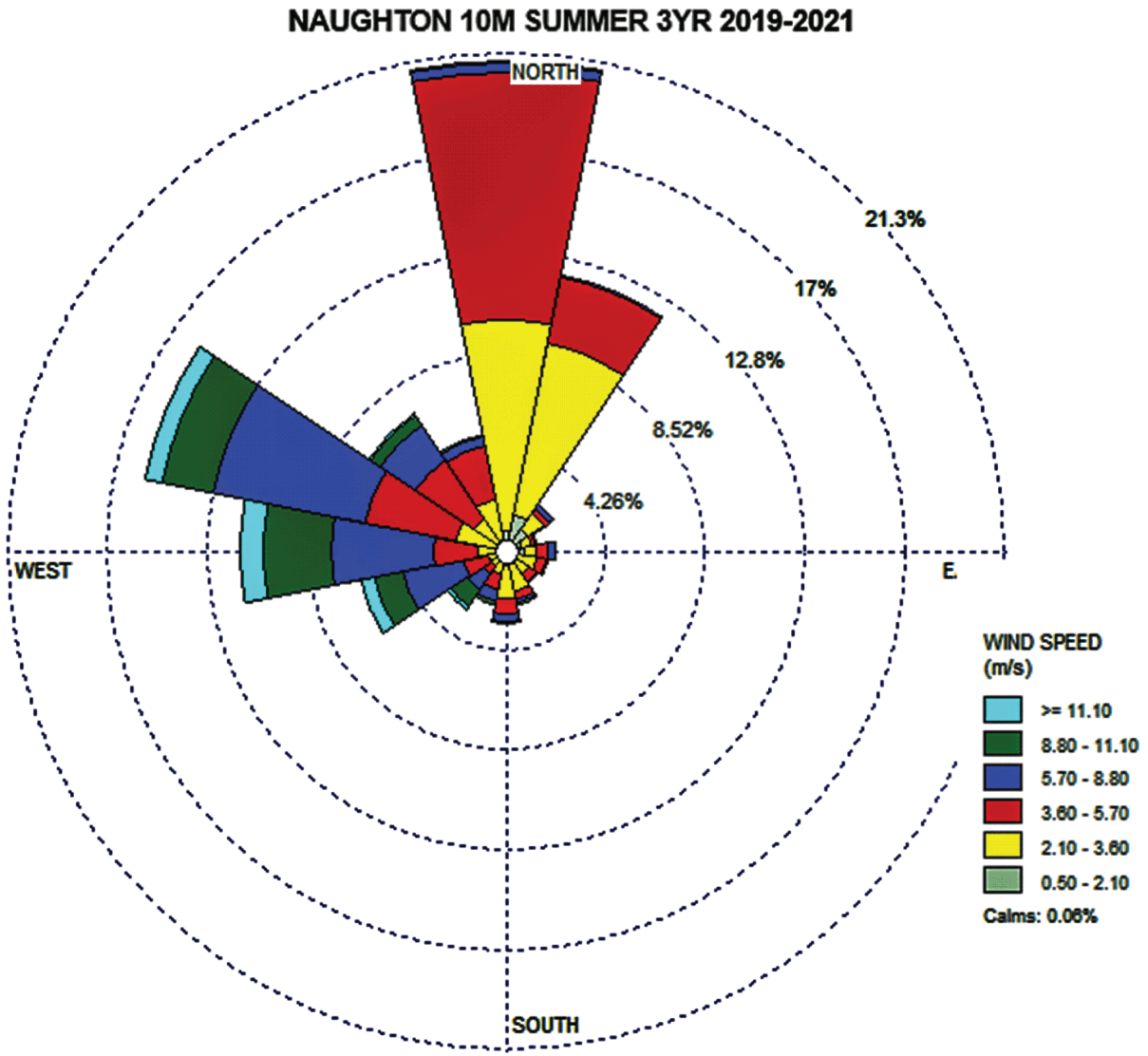






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**Figure 2.4-38 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 10 m**

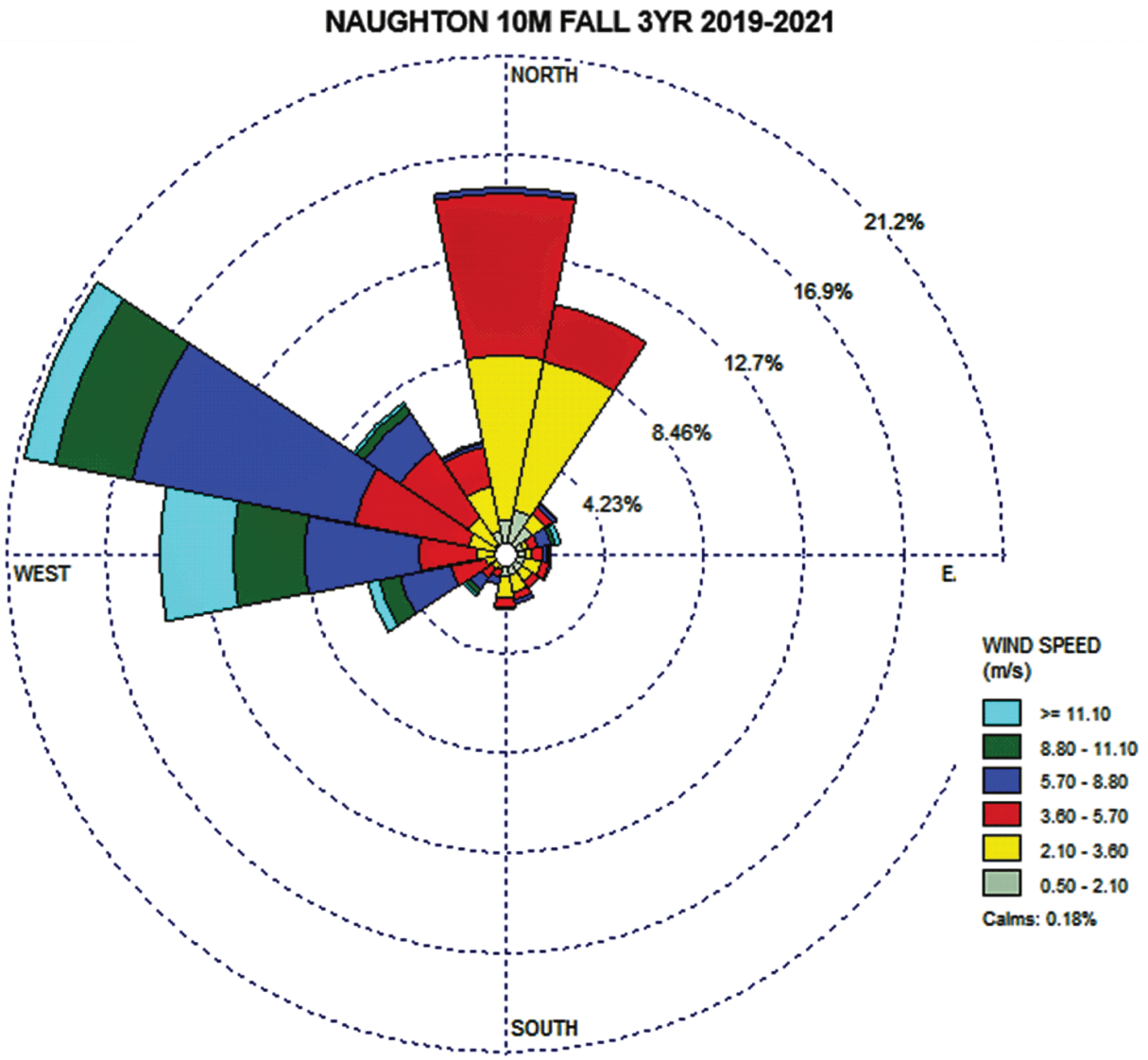






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**Figure 2.4-39 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 10 m**

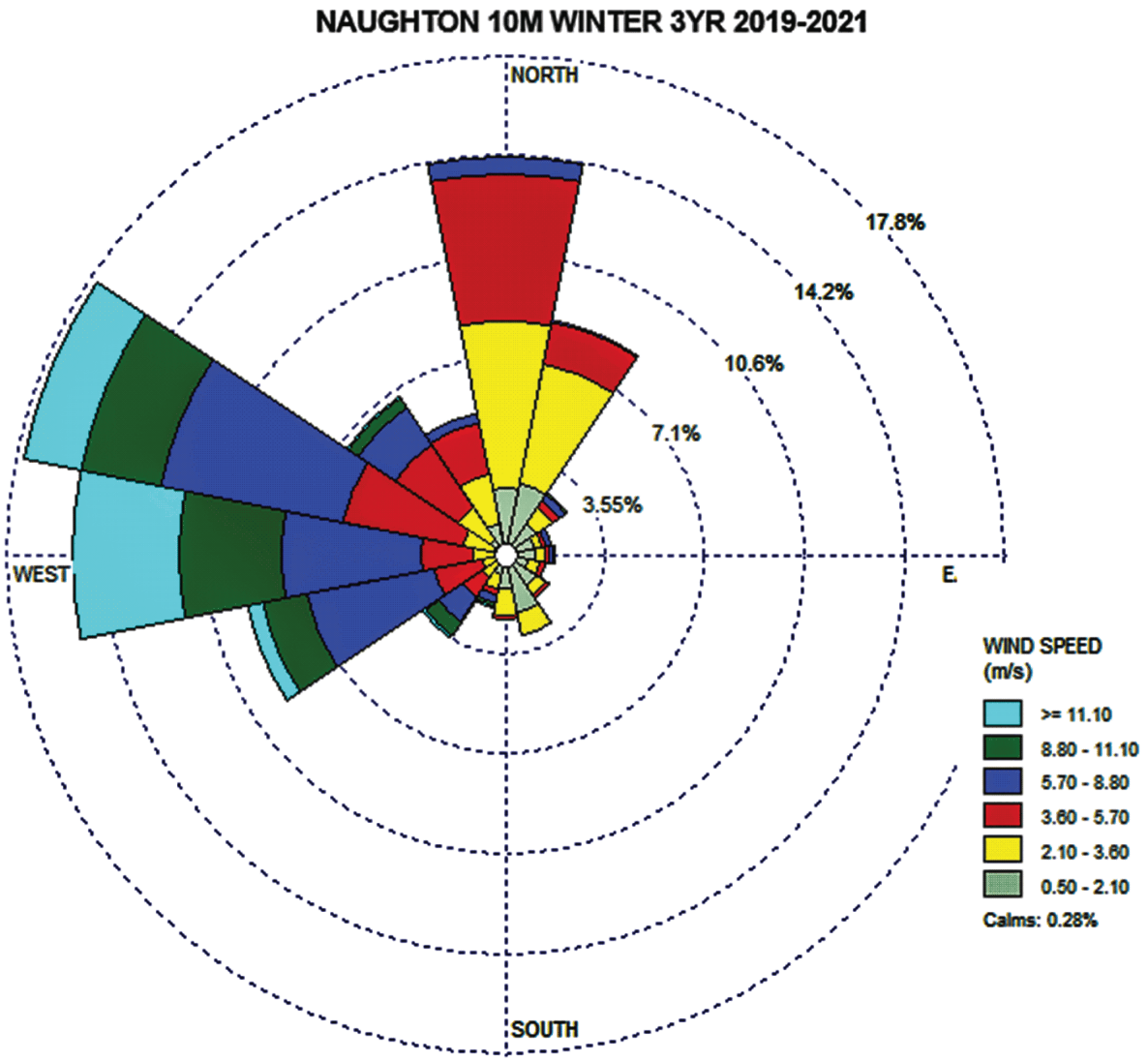






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**Figure 2.4-40 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 50 m**

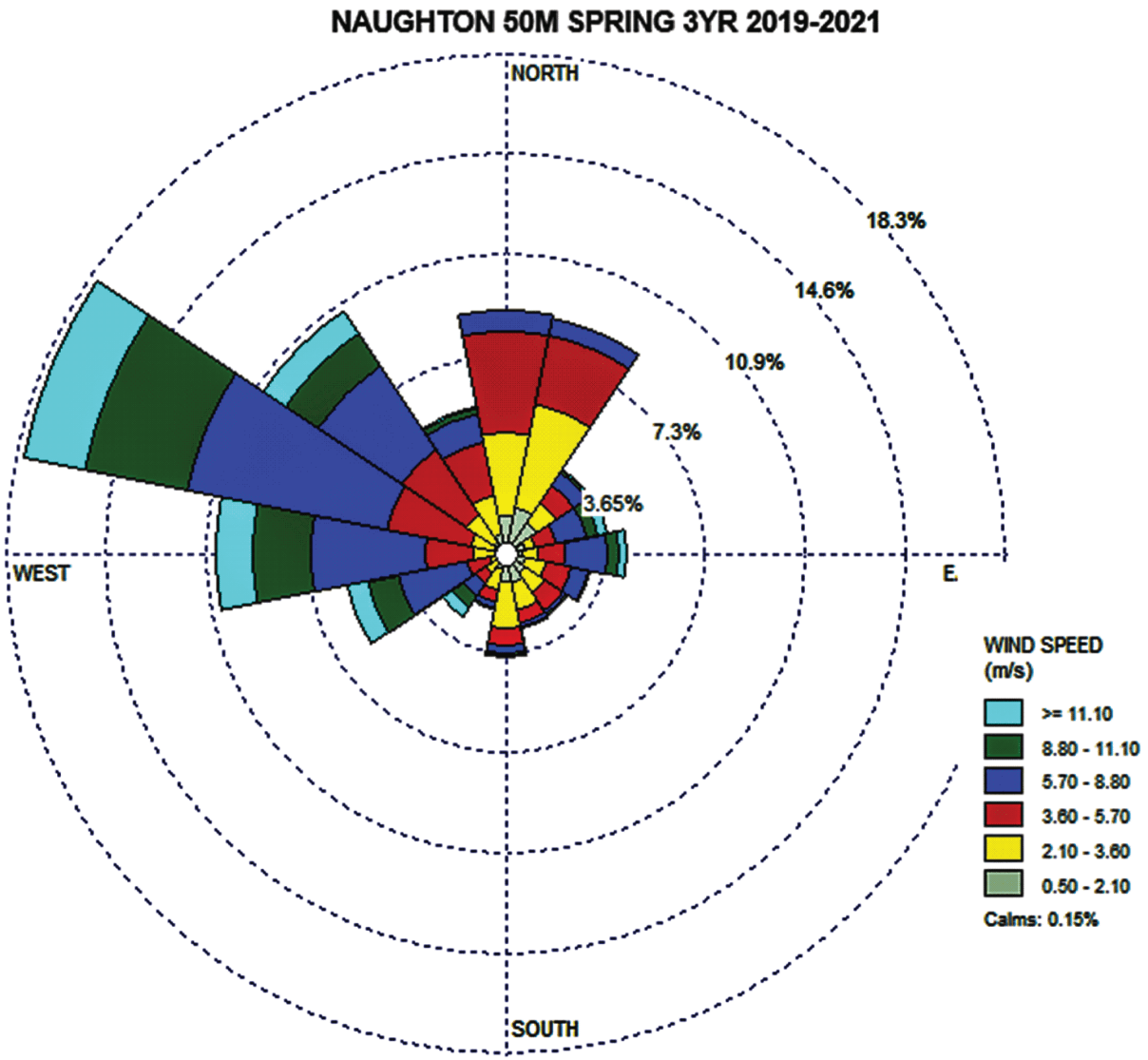






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**Figure 2.4-41 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 50 m**

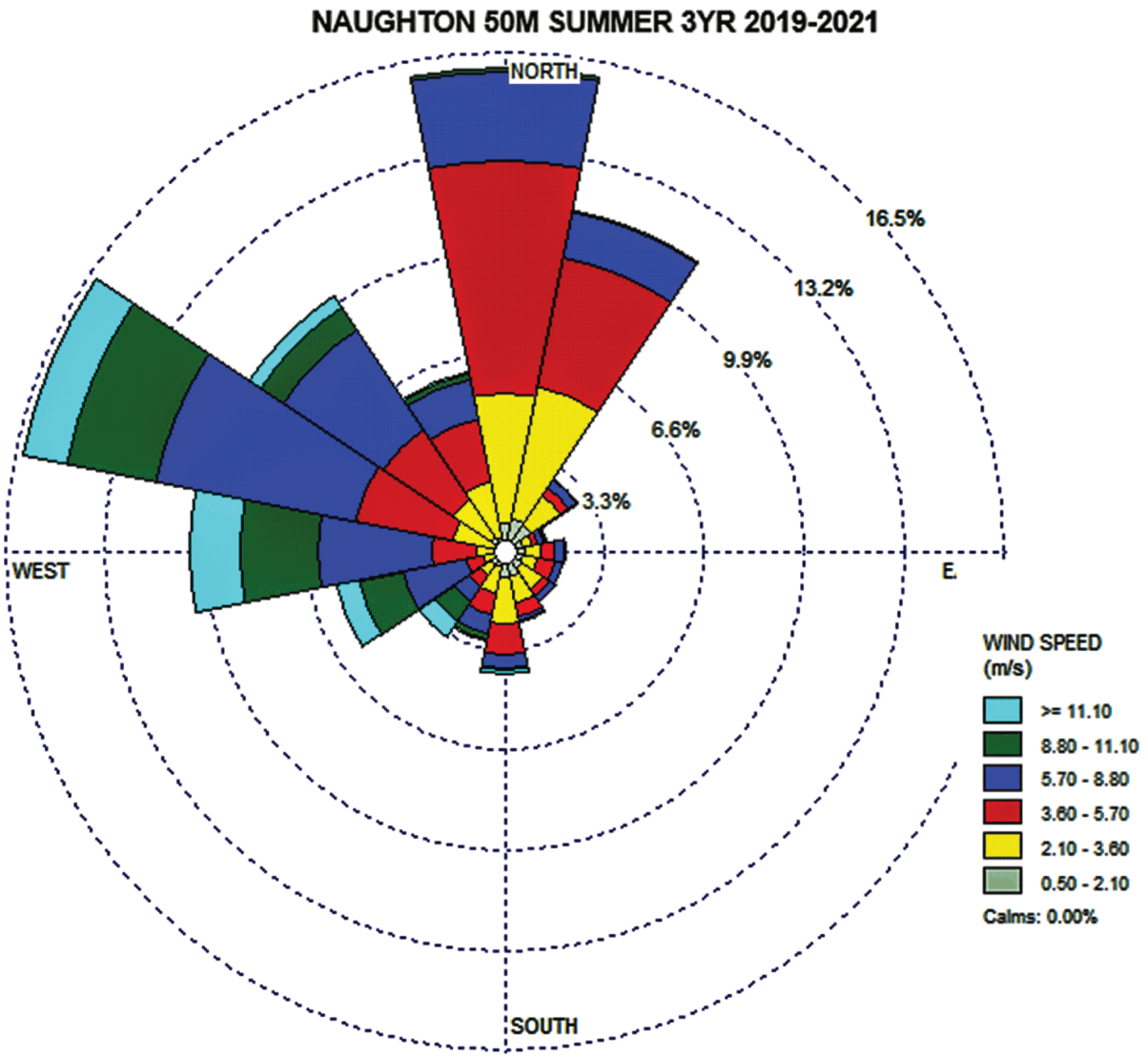






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**Figure 2.4-42 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 50 m**

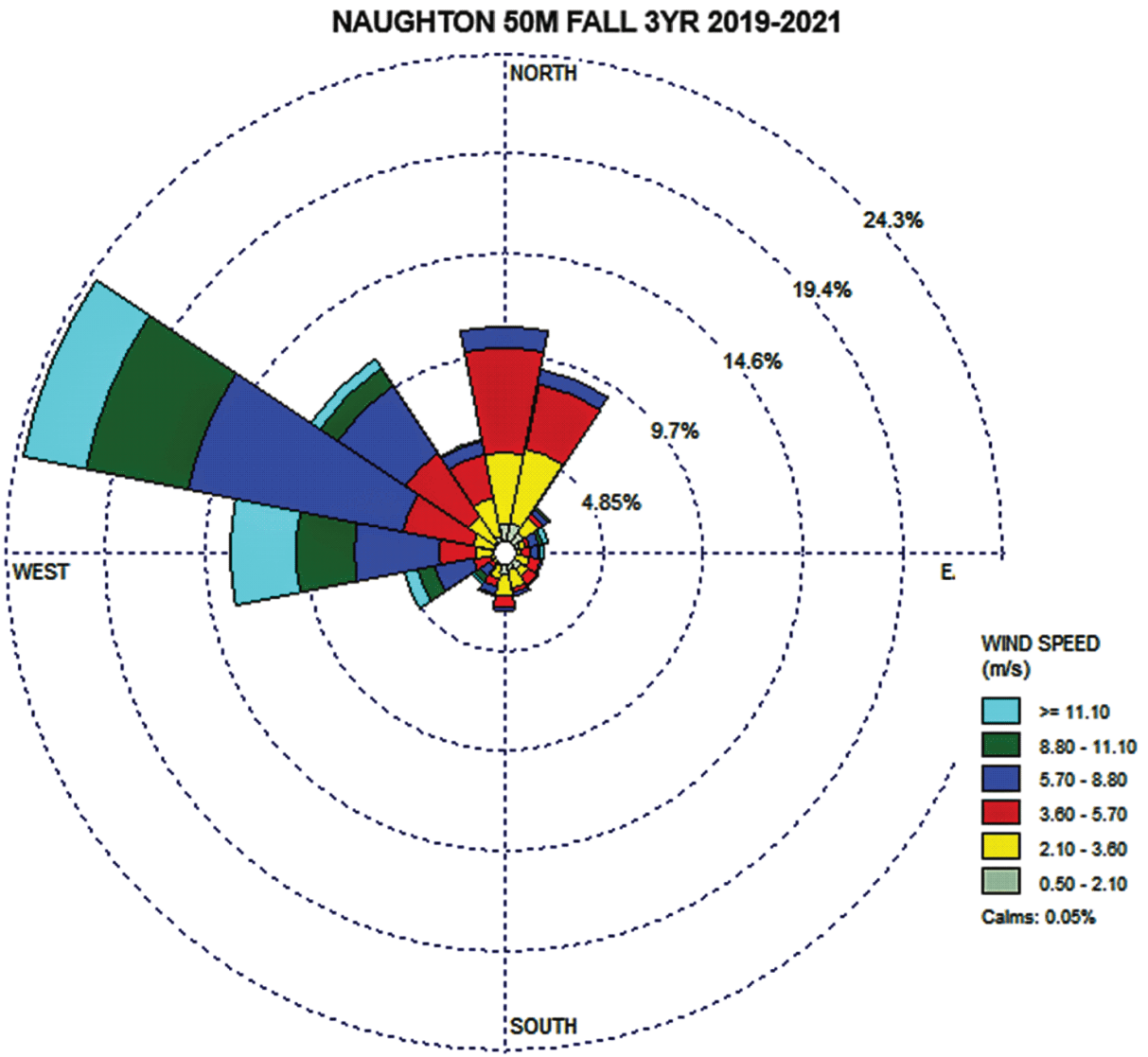






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**Figure 2.4-43 Naughton Power Plant Seasonal Composite Wind Rose 2019–2021, 50 m**

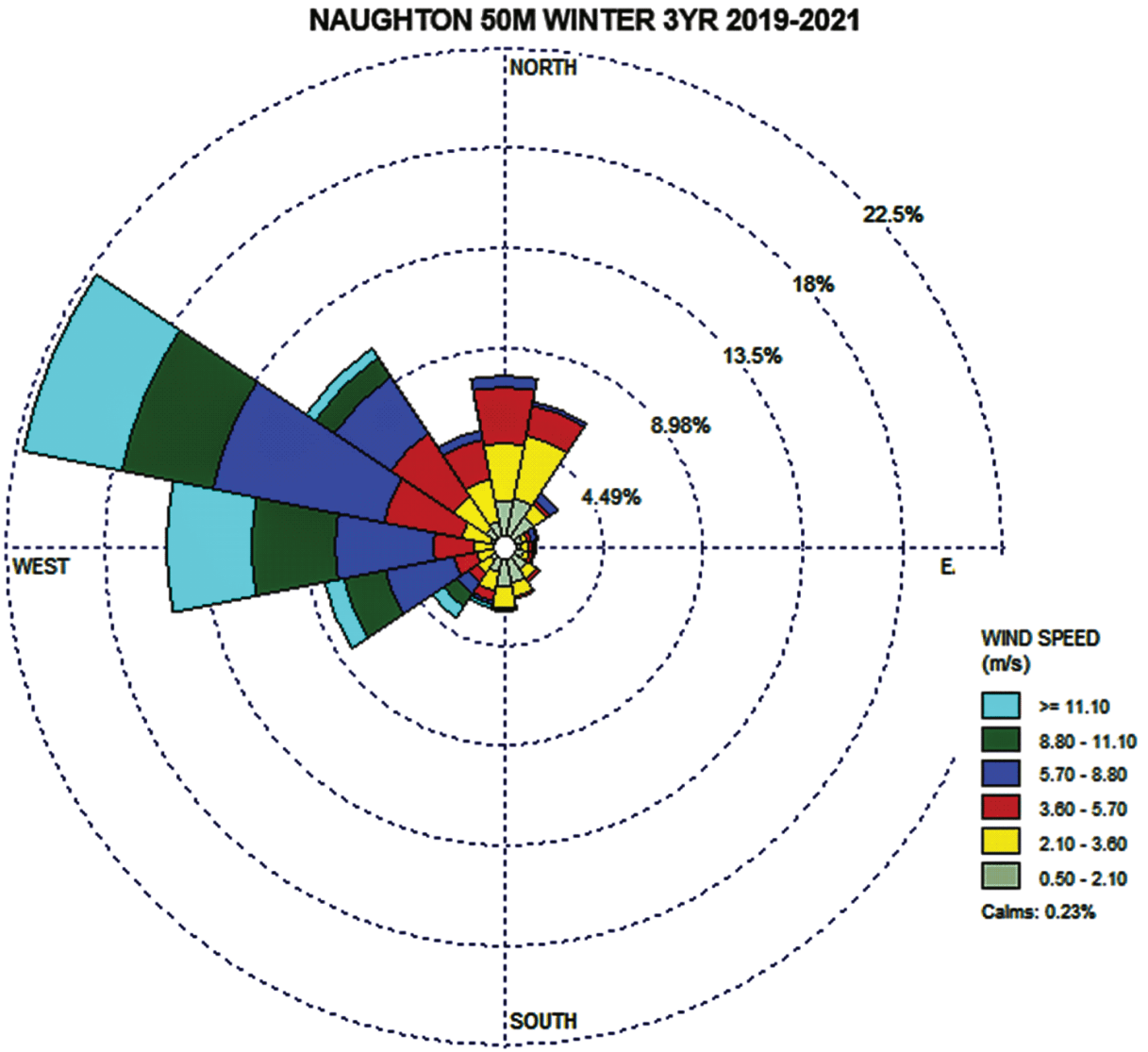




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**Figure 2.4-44 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 10 m**

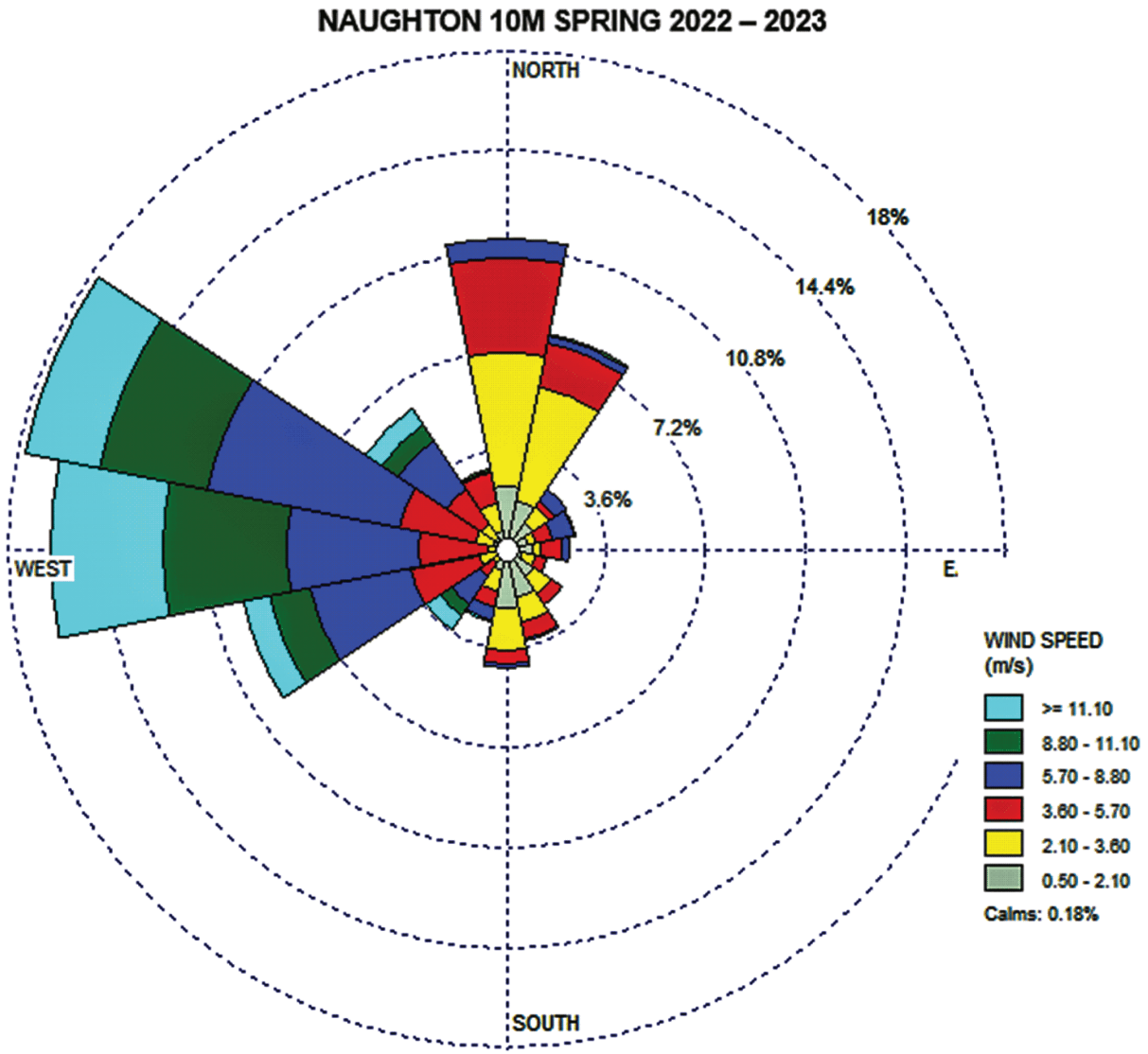






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**Figure 2.4-45 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 10 m**

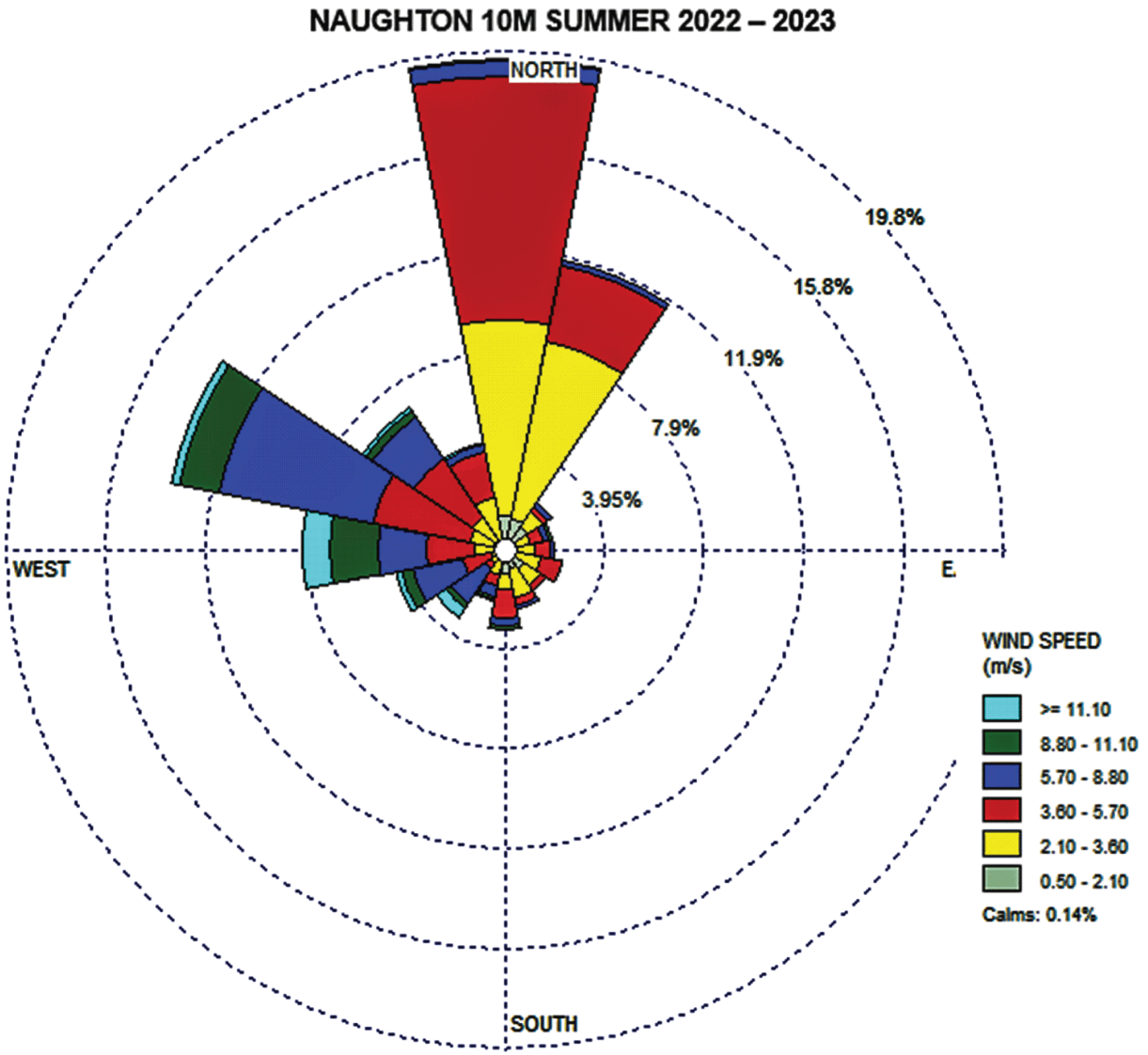






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**Figure 2.4-46 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 10 m**

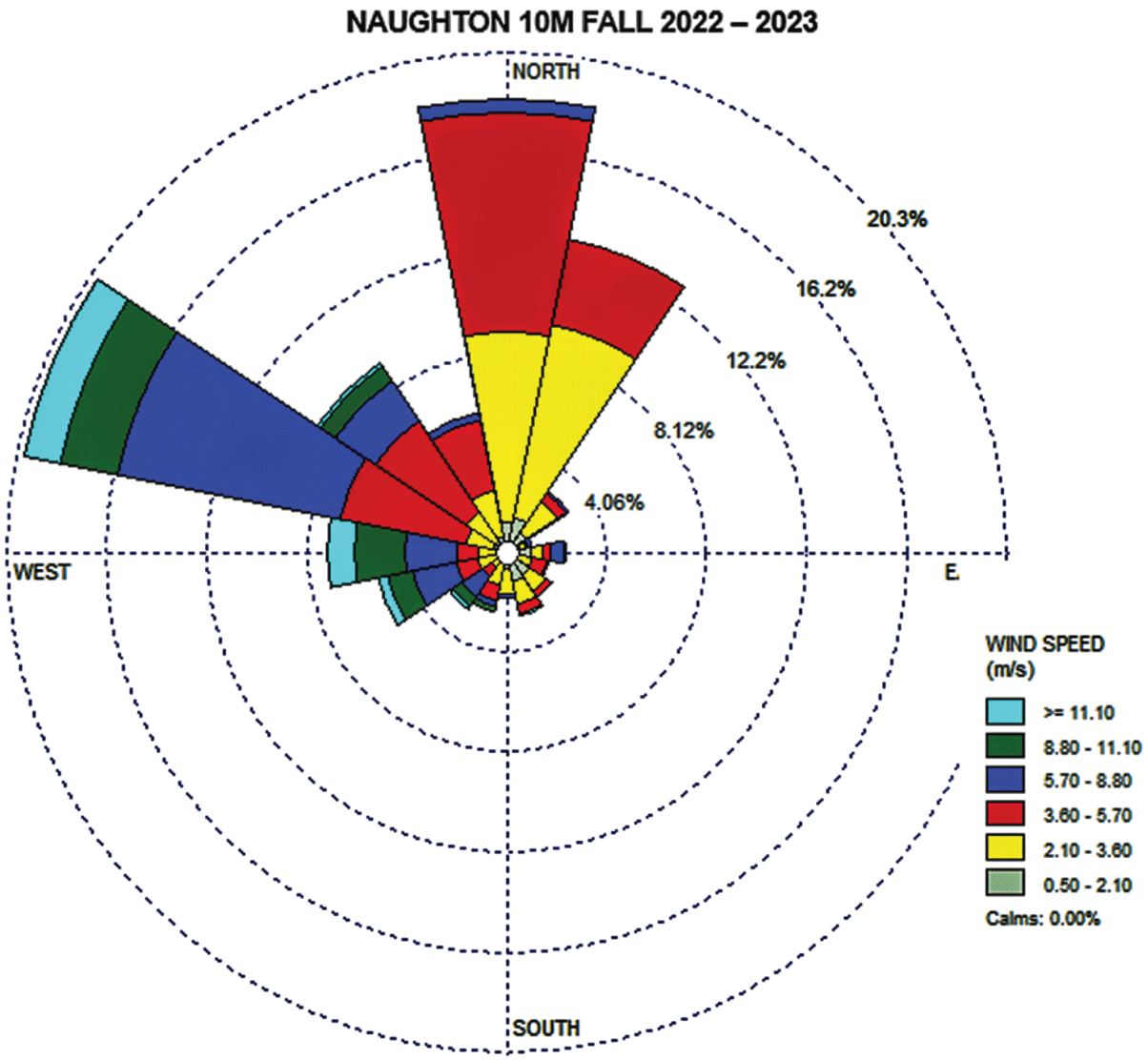






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**Figure 2.4-47 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 10 m**

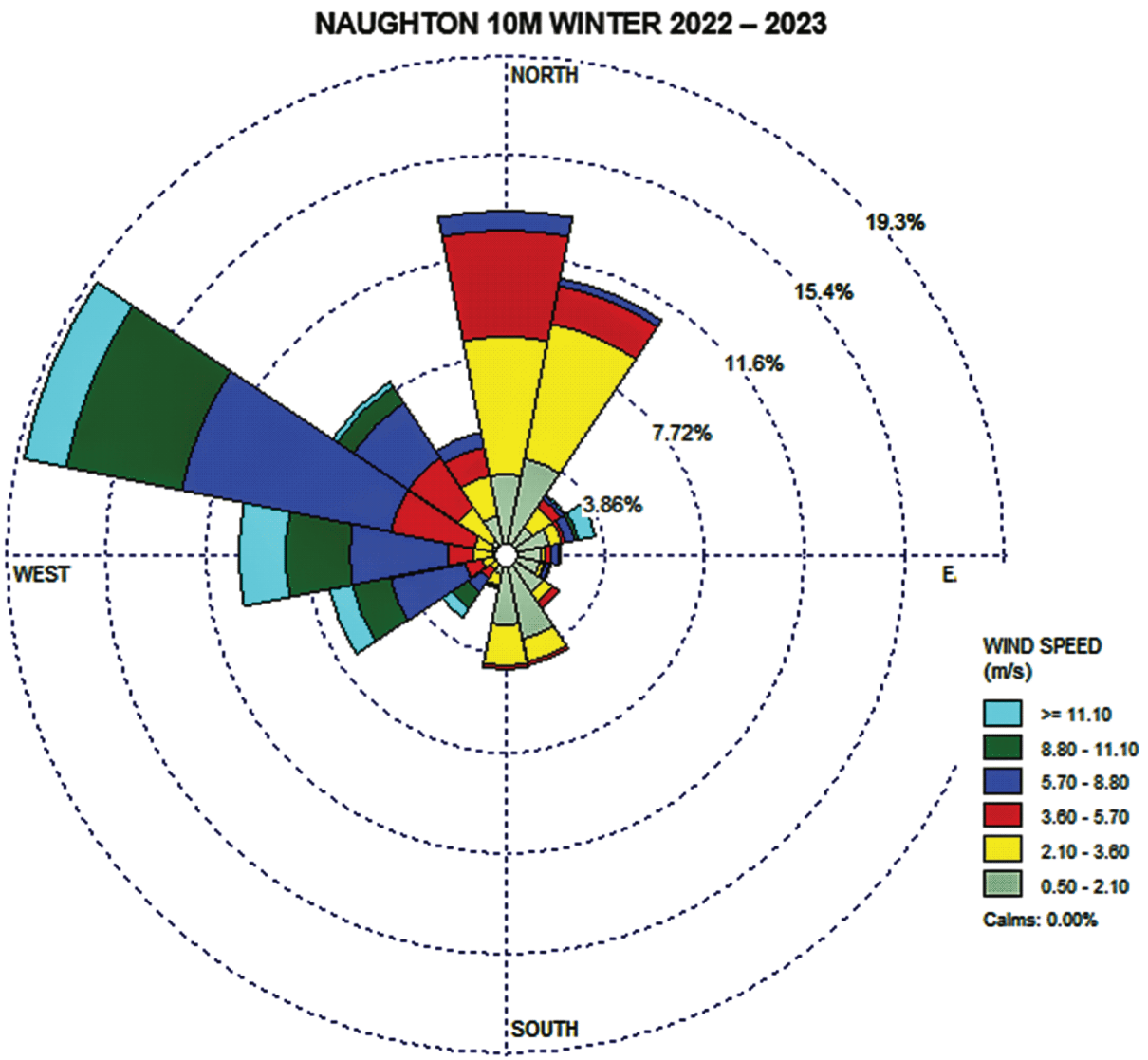






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**Figure 2.4-48 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 50 m**

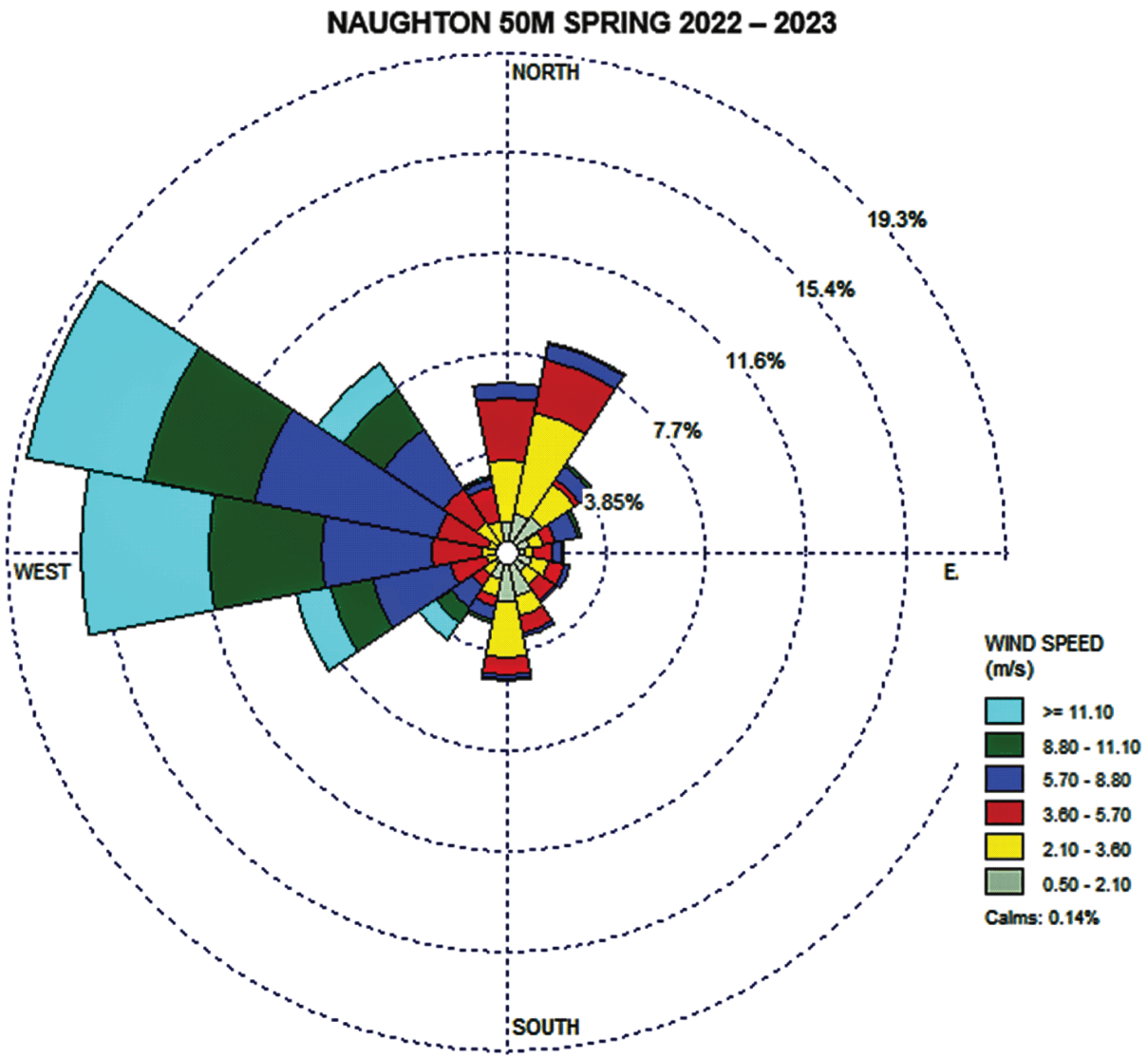






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**Figure 2.4-49 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 50 m**

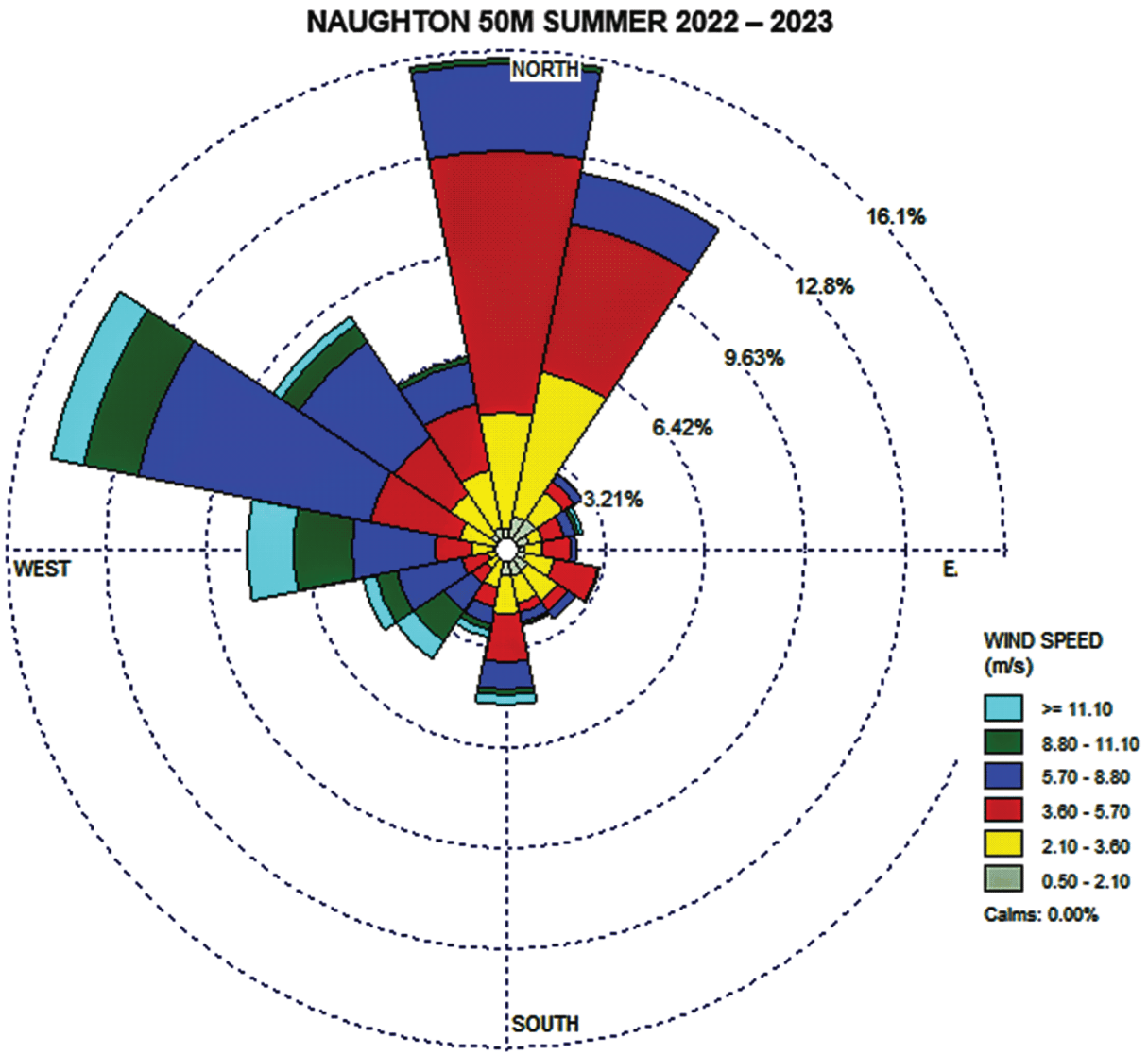






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**Figure 2.4-50 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 50 m**

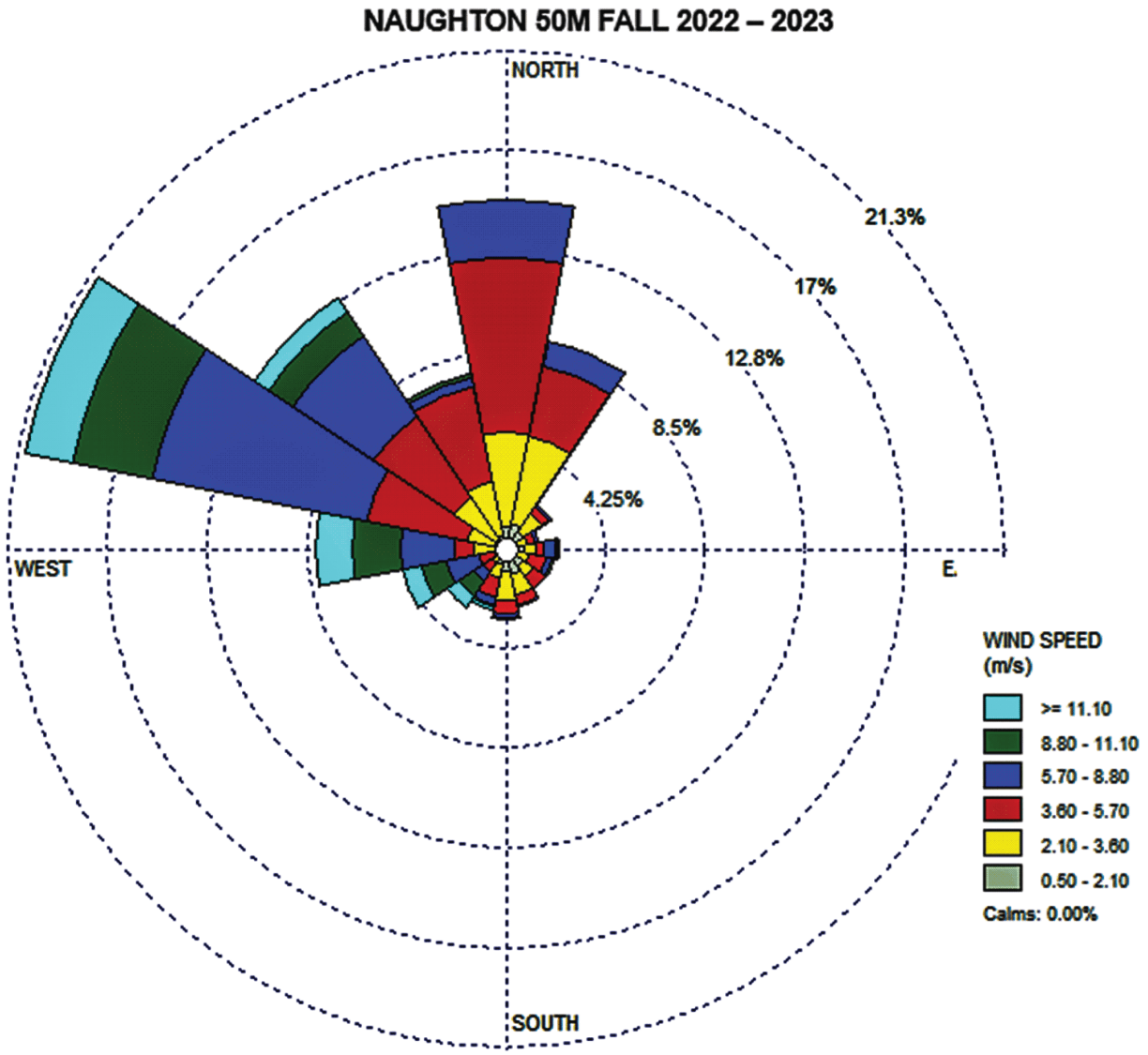






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**Figure 2.4-51 Naughton Power Plant Seasonal Composite Wind Rose 2022–2023, 50 m**

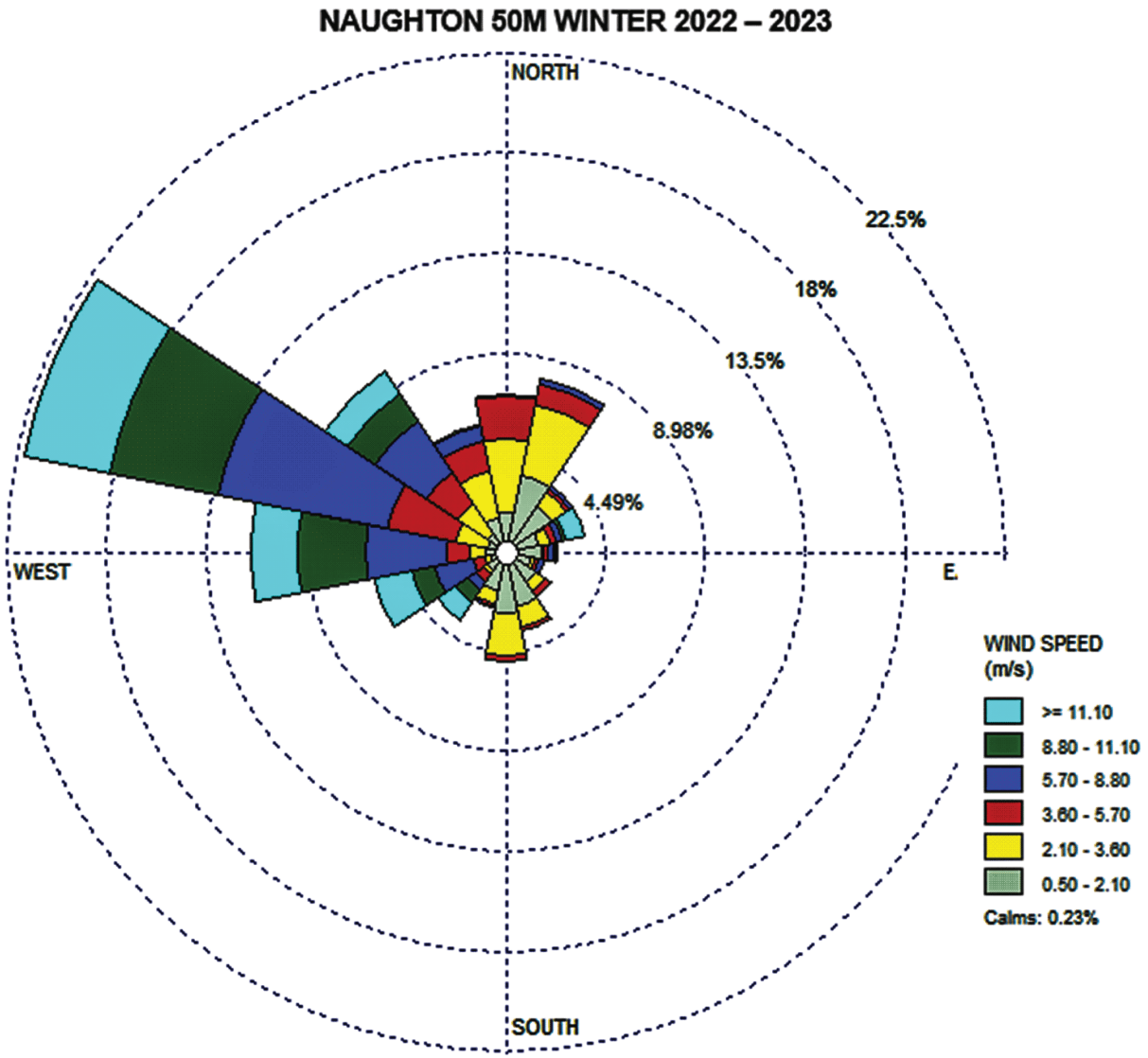






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**Figure 2.4-52 Kemmerer Unit 1 Annual Wind Rose 2022–2023, 10 m**

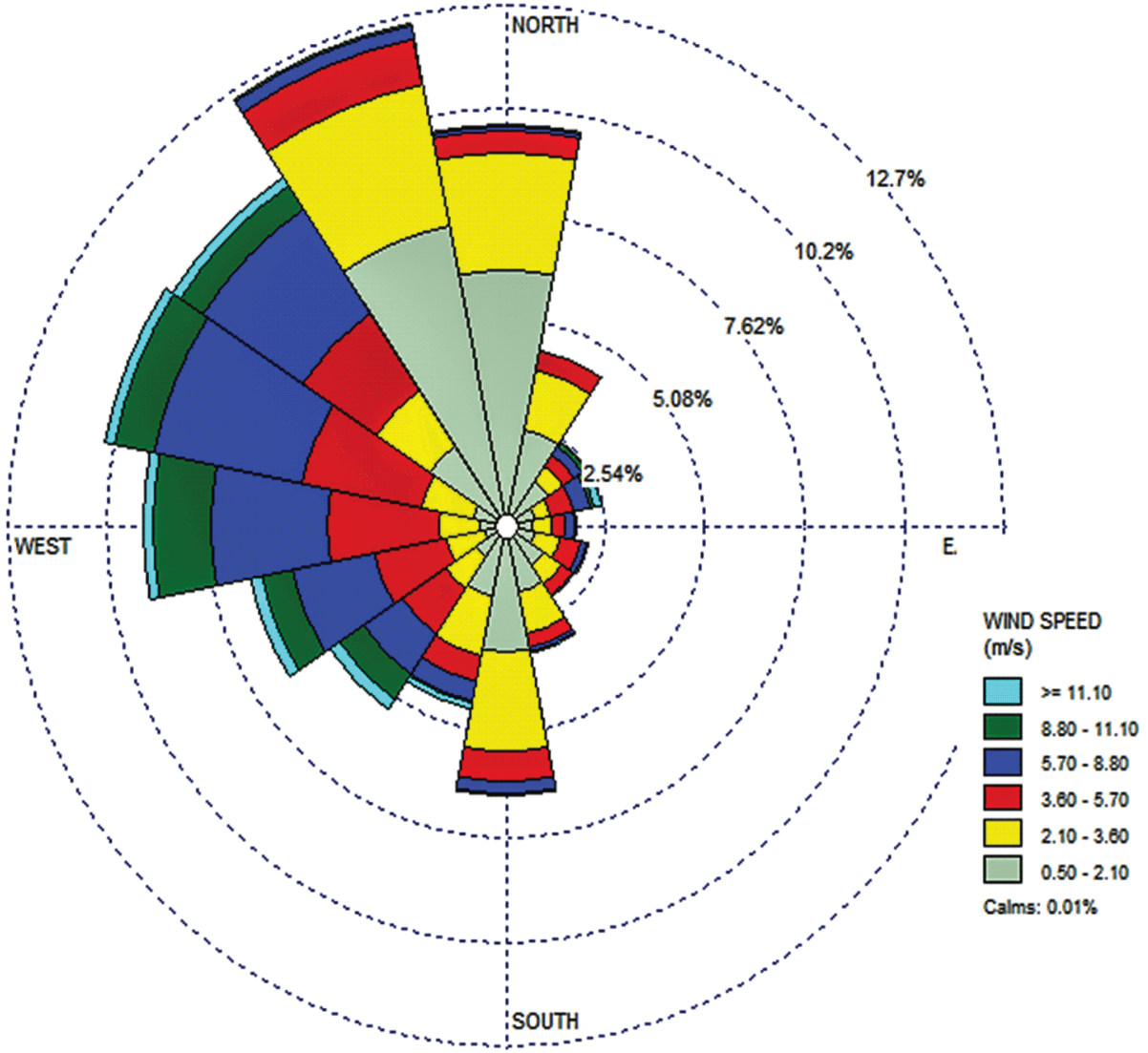






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**Figure 2.4-53 Kemmerer Unit 1 Annual Wind Rose 2022–2023, 60 m**

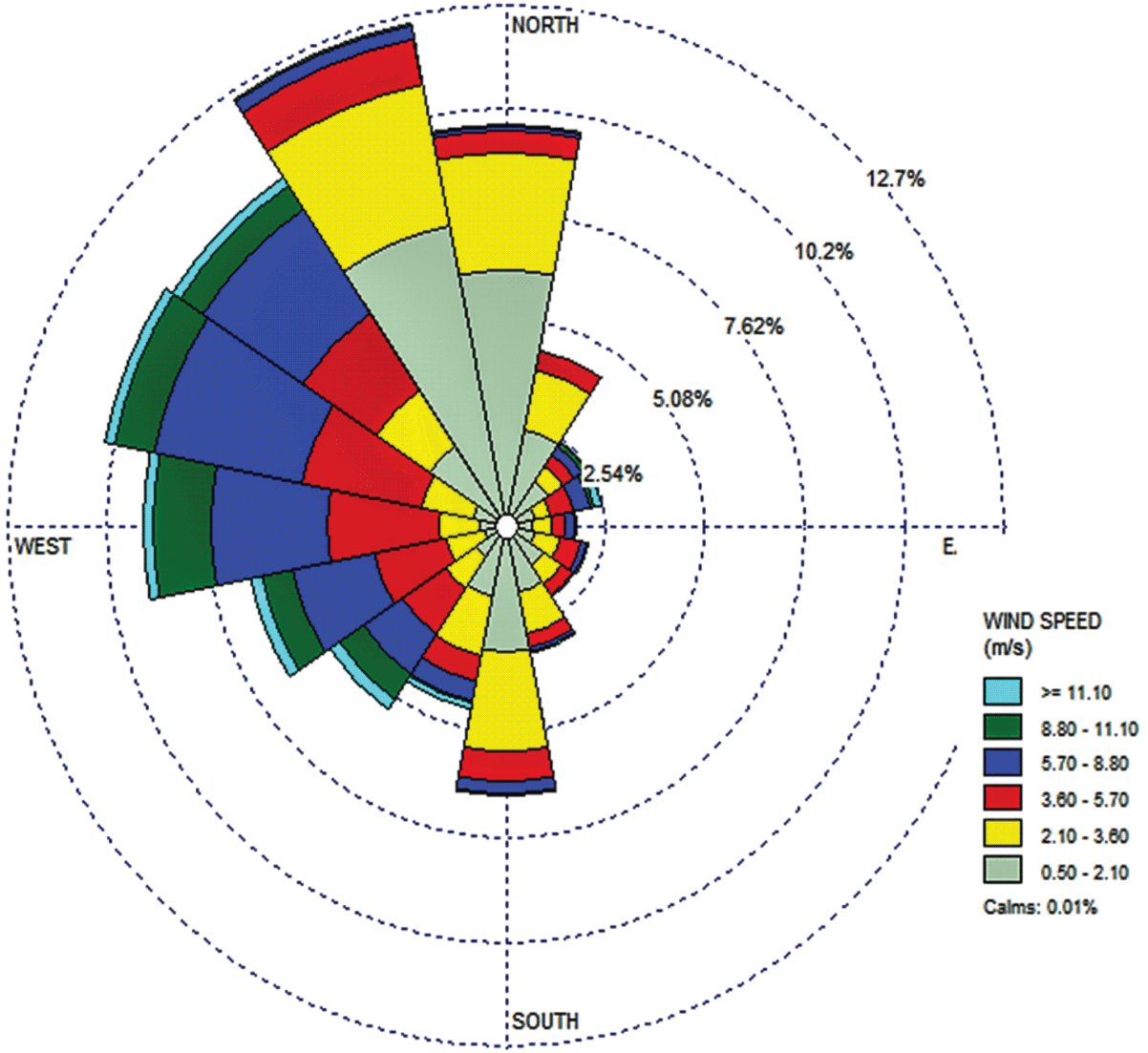






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**Figure 2.4-54 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 10 m**

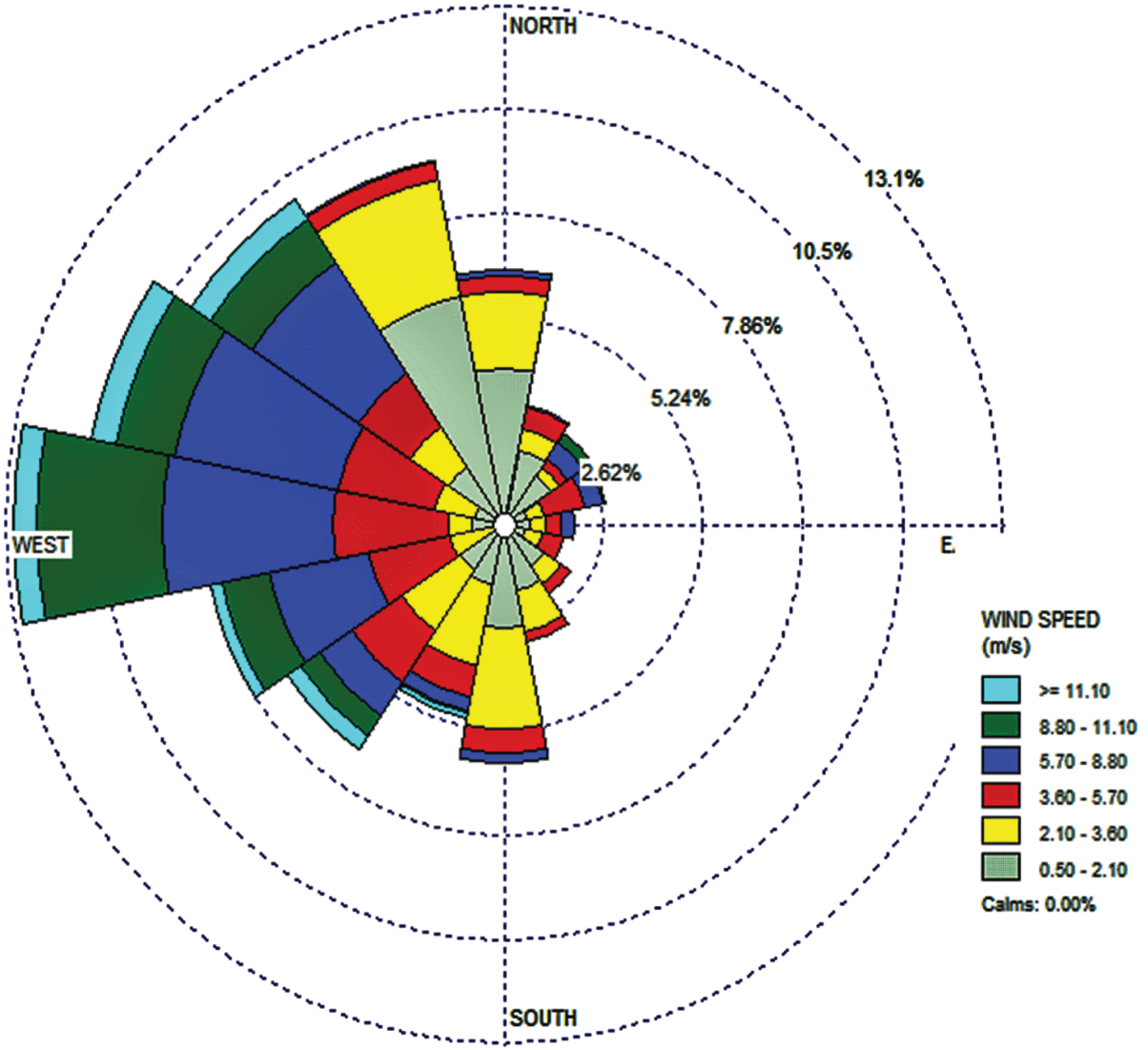






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**Figure 2.4-55 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 10 m**

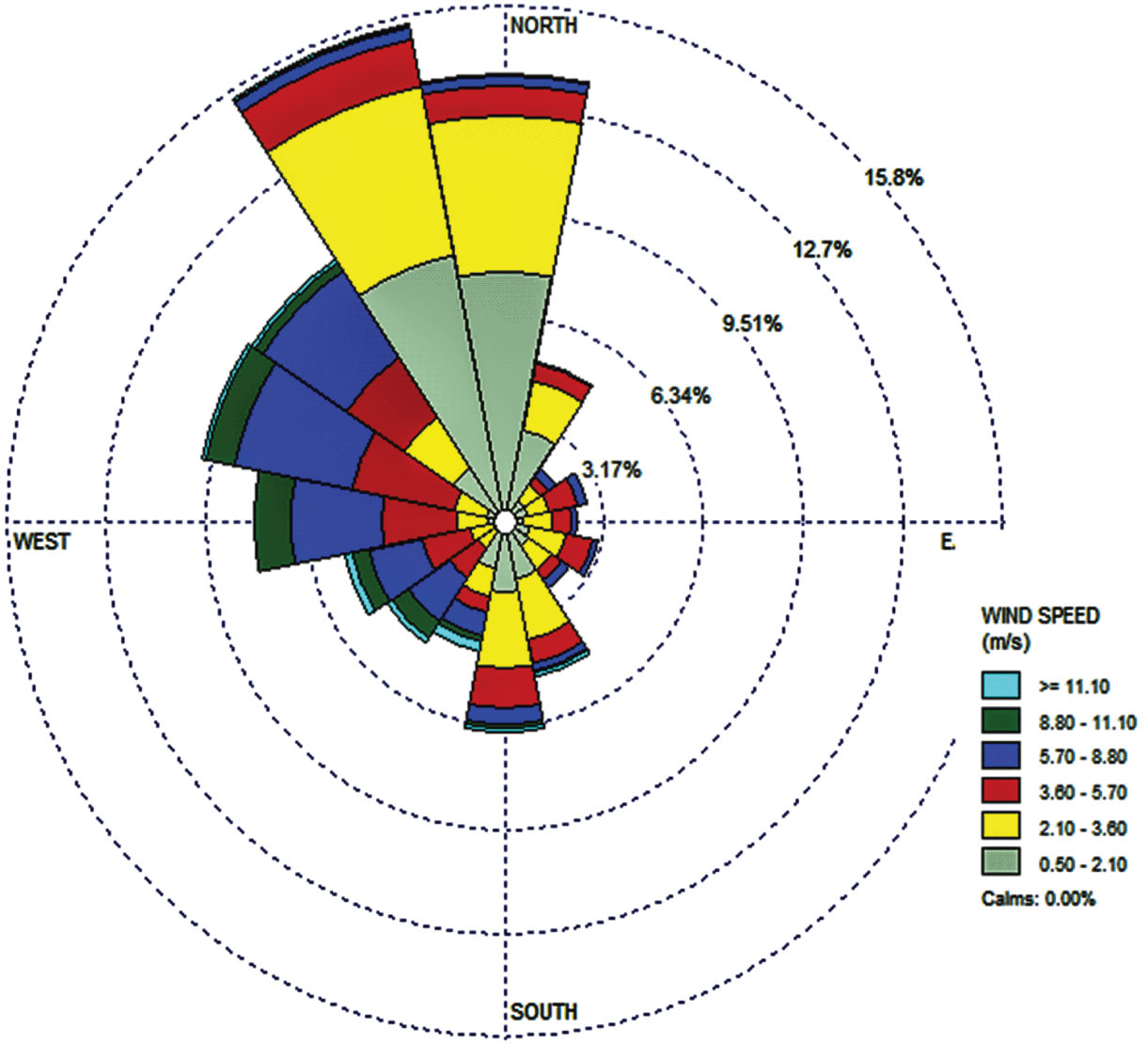






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**Figure 2.4-56 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 10 m**

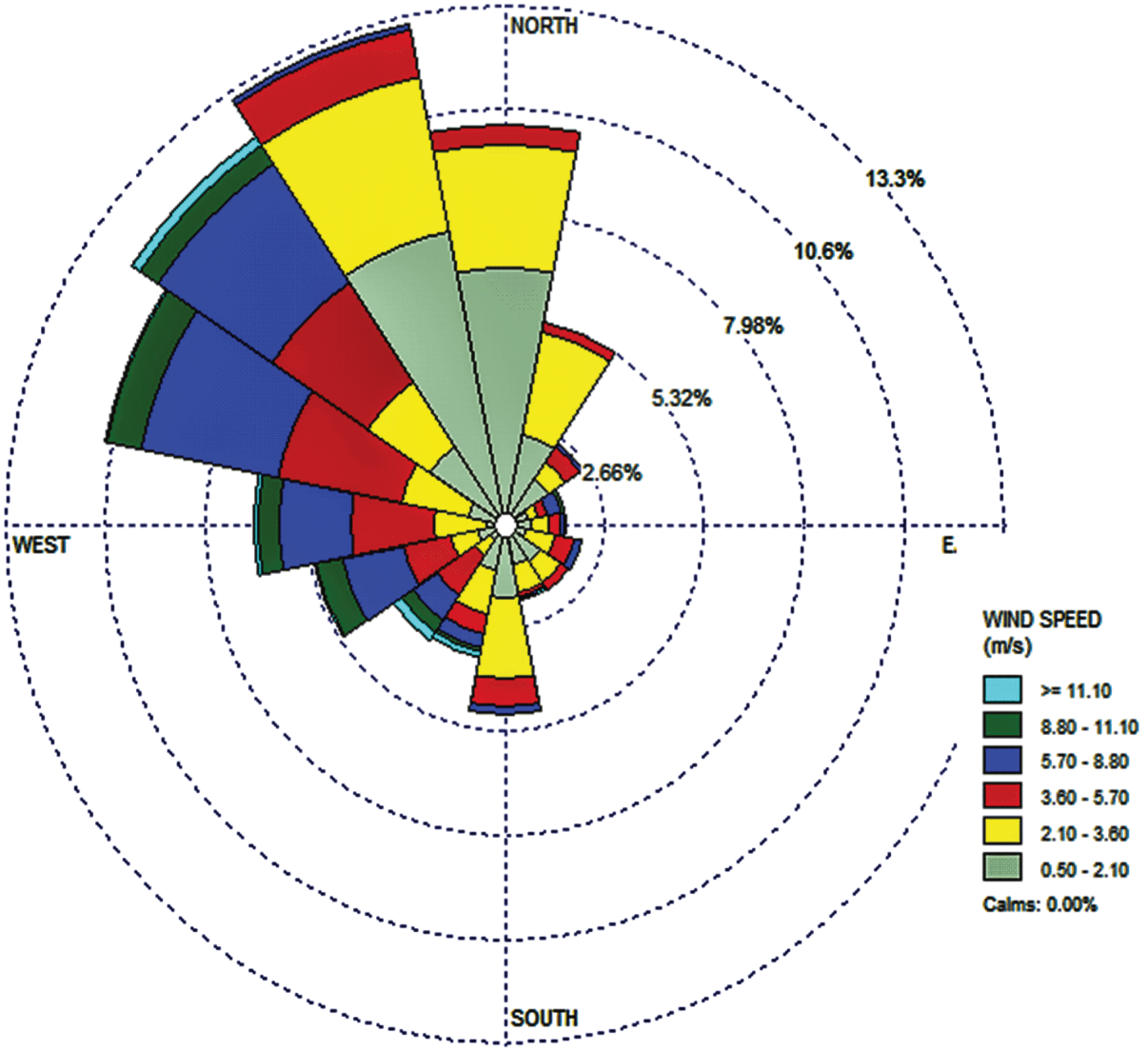






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**Figure 2.4-57 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 10 m**

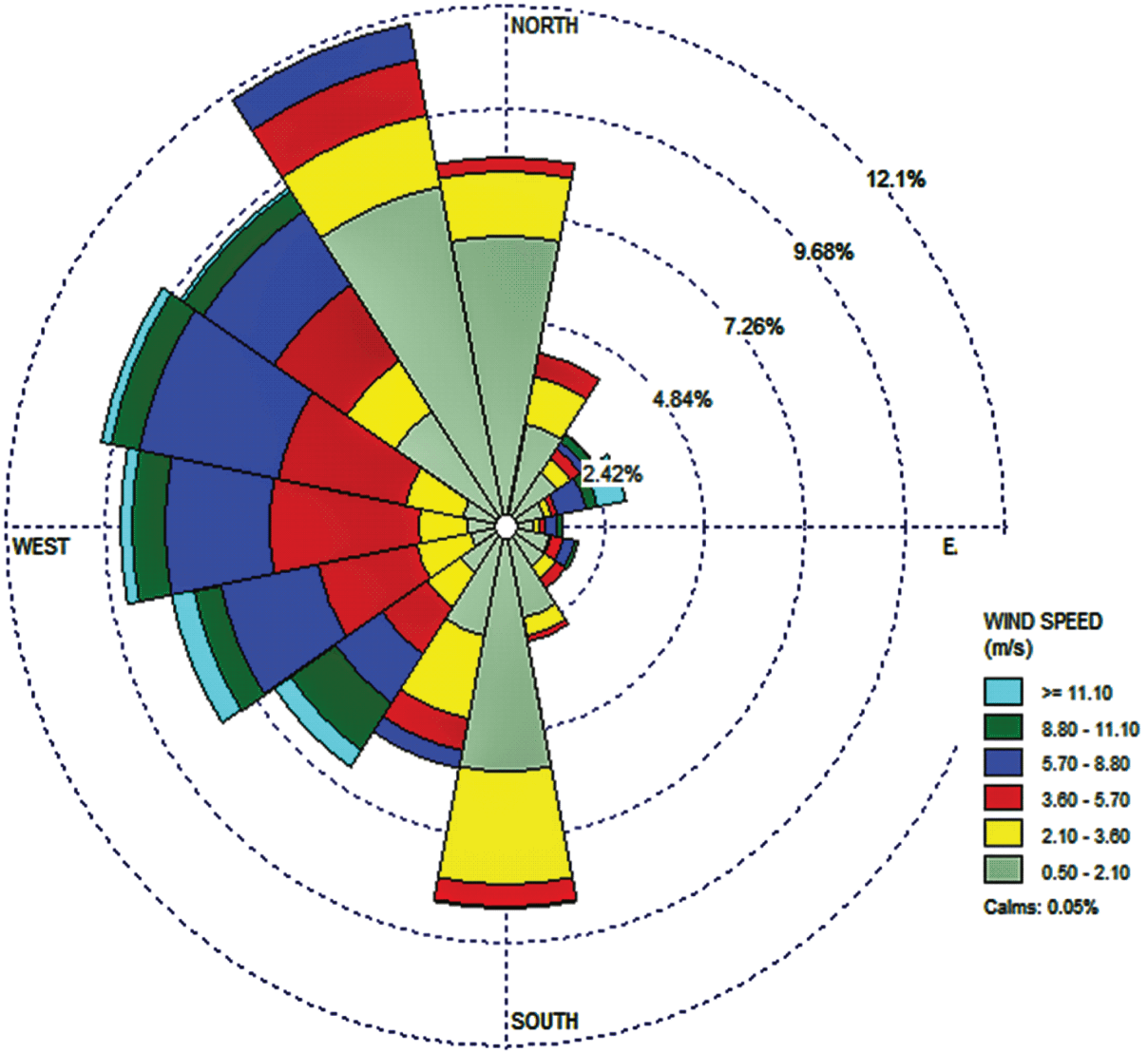






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**Figure 2.4-58 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 60 m**

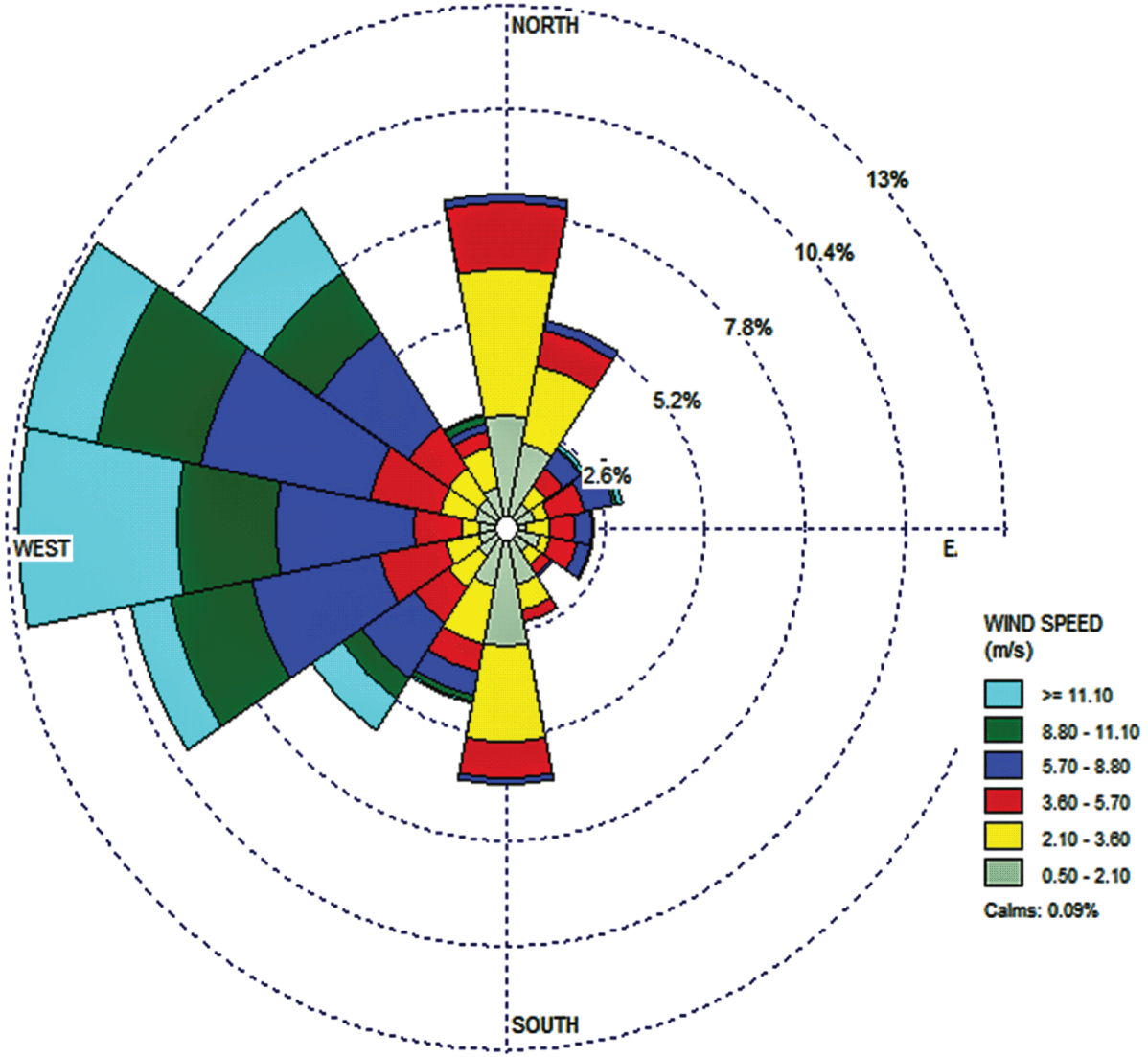






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**Figure 2.4-59 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 60 m**

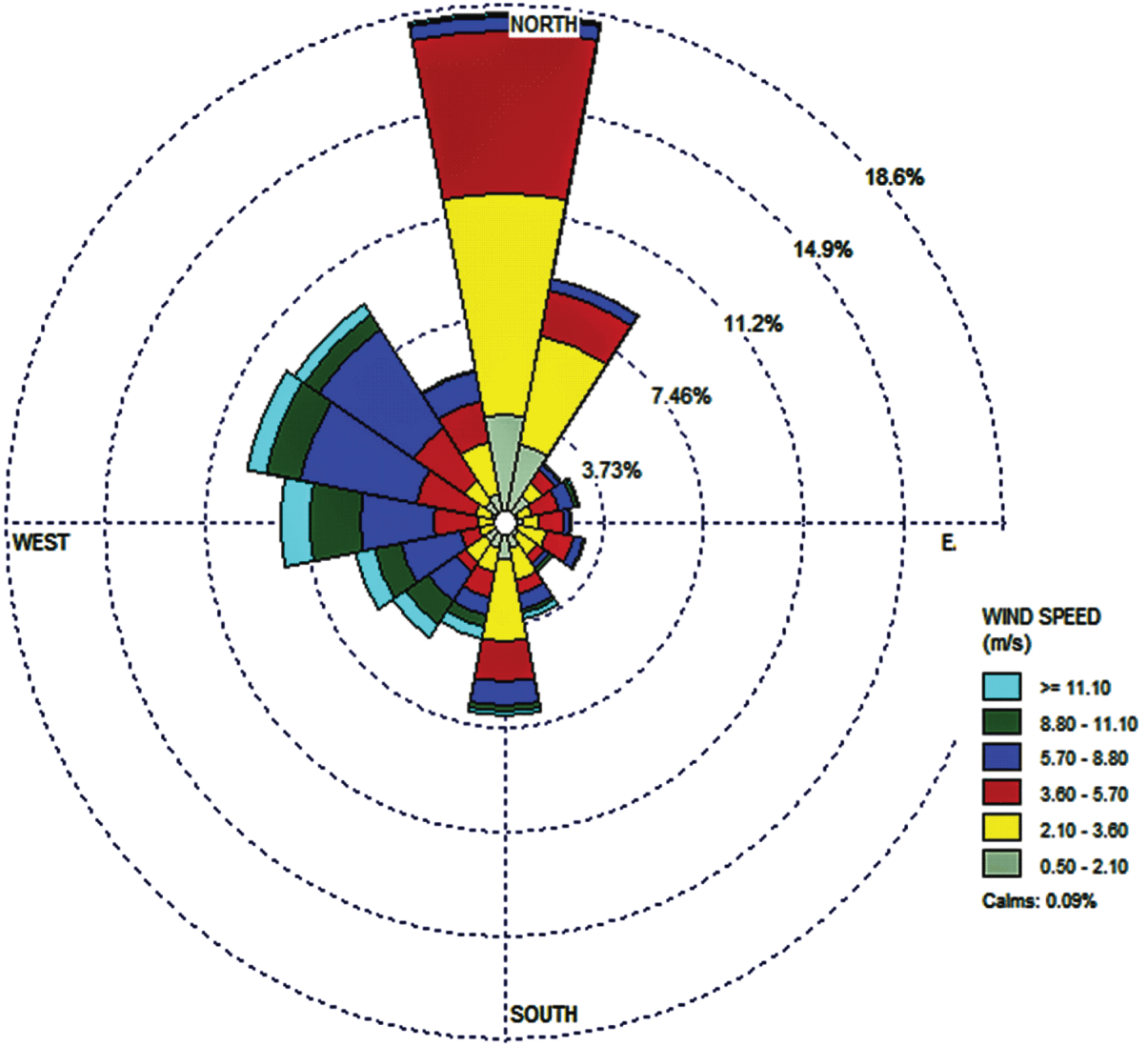






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**Figure 2.4-60 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 60 m**

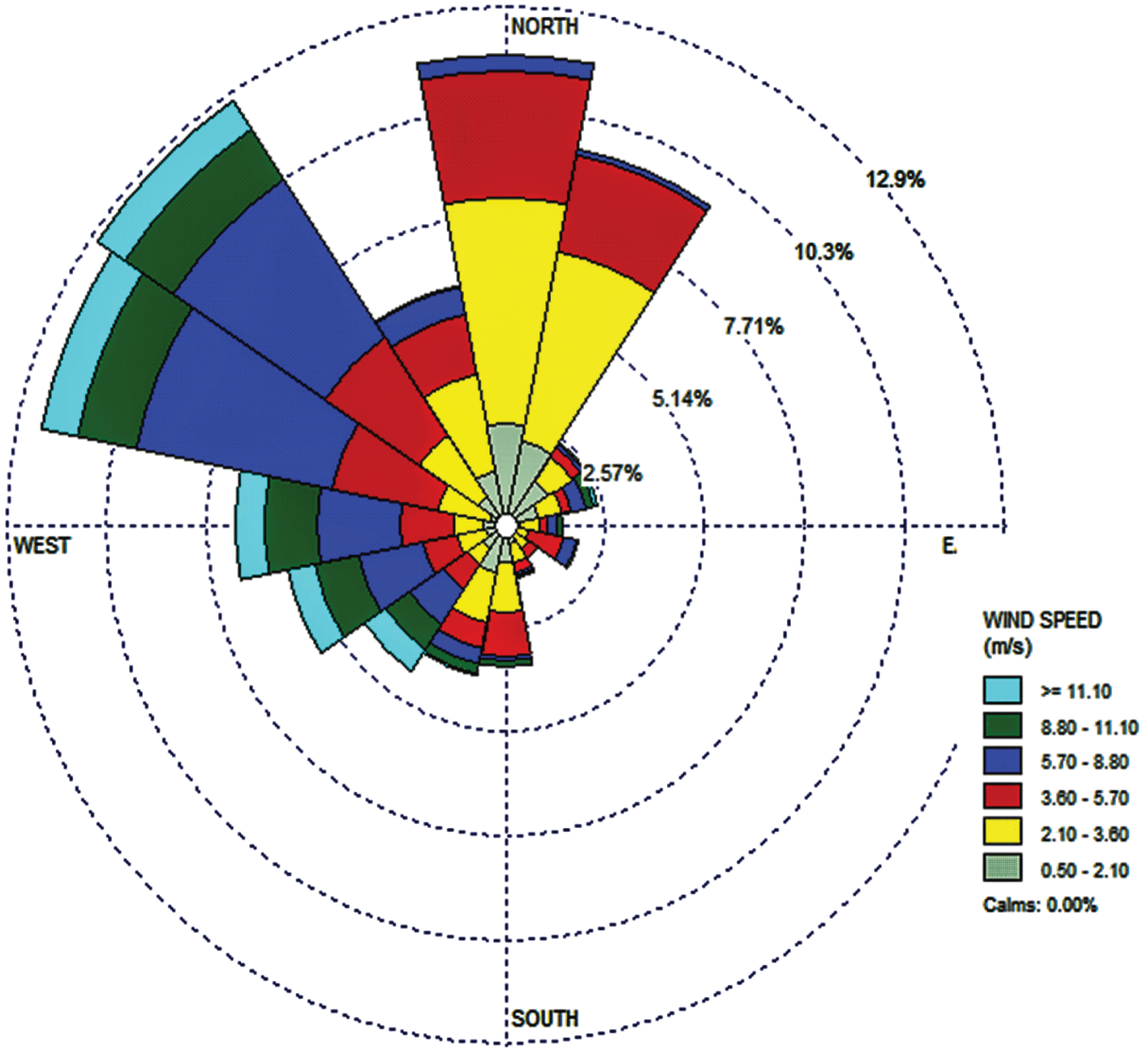






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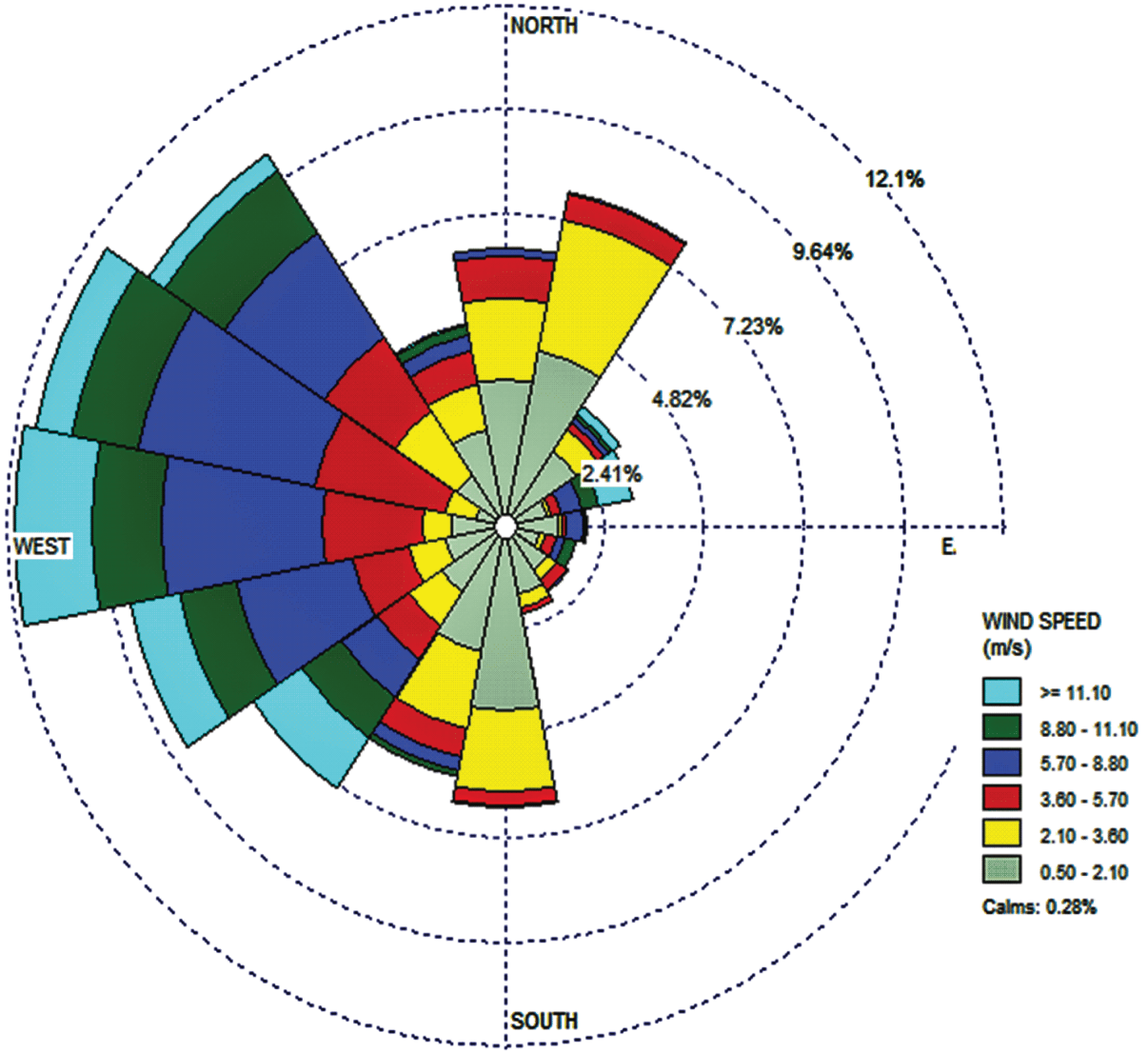
**Figure 2.4-61 Kemmerer Unit 1 Seasonal Composite Wind Rose 2022–2023, 60 m**





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## Figure 2.4-62 UGRB Ozone Non-Attainment Area and Nearest Class 1 Federal Area

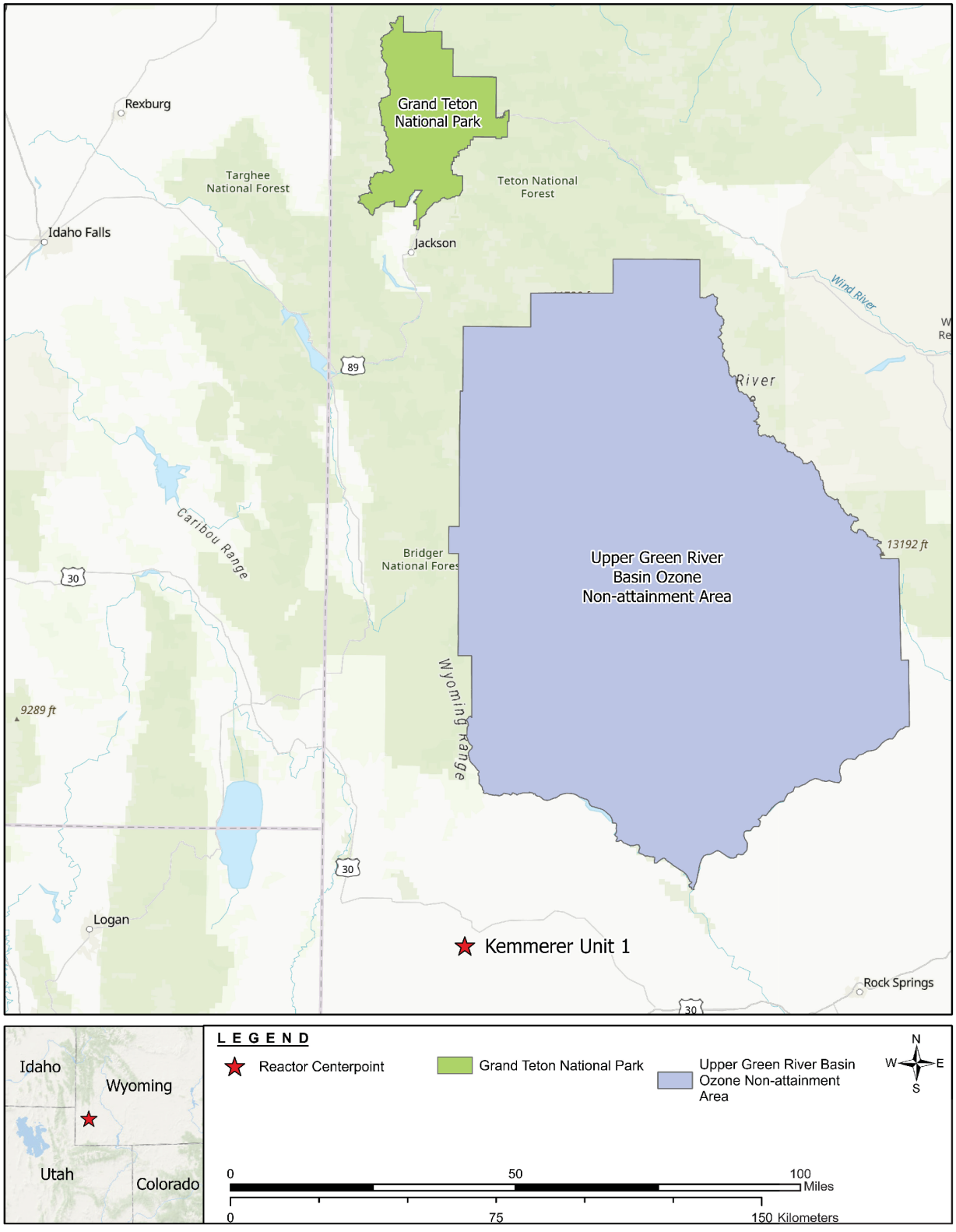



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**Figure 2.4-63 Topographic Map of Kemmerer Unit 1 with Location of Meteorological Tower**

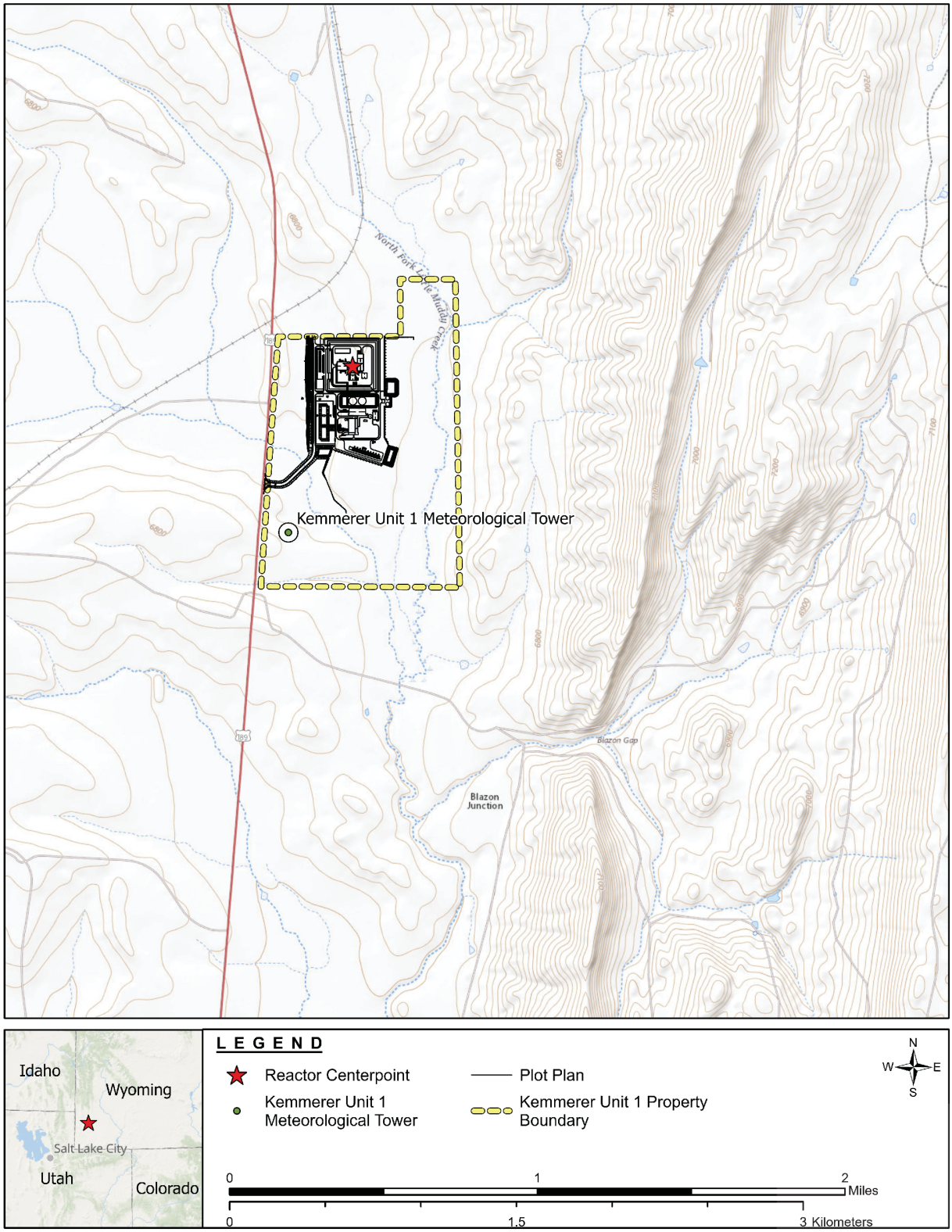



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**Figure 2.4-64 Topographic Map of Kemmerer Unit 1 with Location Out to 50 Miles**

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## 2.5 Hydrologic Description

The Kemmerer Unit 1 site is located in Lincoln County, Wyoming about seven miles (11 kilometers) south of the City of Kemmerer. Figure 2.5-1 shows the general location of the site on the east side of US Highway 189, about four miles (6.4 kilometers) southeast of the existing Naughton Power Plant. The site is within the Upper North Fork Little Muddy Creek watershed within the Greater Green River Basin and is bounded by the North Fork Little Muddy Creek on the east side and by an unnamed tributary on the southwest and south sides. The confluence of the unnamed tributary with the North Fork Little Muddy Creek is located just south of the site. The regional climate is considered semi-arid with an annual precipitation of about 10 inches (250 millimeters) (Reference 2.5-1). The Naughton Power Plant has a meteorological station which is located in the North Fork Little Muddy Creek watershed. The nearest meteorological gauge outside the North Fork Little Muddy Creek watershed is in the Hams Fork River watershed in the City of Kemmerer. Stream and reservoir water level gauges on the Hams Fork River watershed are discussed in Section 2.5.4.

Long-term droughts are a common occurrence in Wyoming. Seven statewide droughts lasting three years or longer occurred between 1900 and 2004 (Reference 2.5-2). The site is located in the Bear/Green climate division, 1 of 10 divisions used by the Wyoming State Climate Office to characterize the climate in different parts of the state. During this period (1900 to 2004), the greatest drought in terms of average annual precipitation deficit in the Bear/Green climate division occurred from 1988 to 1990, while the greatest drought in terms of total precipitation deficit occurred from 1900 to 1903 (Reference 2.5-2).

At the Kemmerer Unit 1 site location, the North Fork Little Muddy Creek is perennial, having a continuous flow. However, most of the time, the flow in the creek is composed solely of plant discharge from the Naughton Power Plant. Natural stream flow occurs only during heavy rainstorms and during the spring snowmelt. The creek has a wide floodplain of at least 1,000 feet and a gentle longitudinal slope near the site. The unnamed tributary on the southwest and south sides of the site is considered an ephemeral stream that is dry most of the time with flows occurring only during heavy rainstorms and during the spring snowmelt. Many of the smaller streams and creeks in the area are also ephemeral, flowing only in response to rainfall or snowmelt. Additional information about the watersheds for the North Fork Little Muddy Creek and the unnamed tributary can be found in Section 2.5.1.2.

Hams Fork River, a tributary to Blacks Fork River, is also within the Greater Green River Basin and flows through the City of Kemmerer north of the site. Hams Fork River flows east outside the North Fork Little Muddy Creek watershed.

Two reservoirs are located on Hams Fork River, Viva Naughton Reservoir and Kemmerer Reservoir. These reservoirs are not in the same watershed as Kemmerer Unit 1. Figure 2.5-33 shows the location of these reservoirs with respect to the site. Viva Naughton Reservoir, owned and operated by PacifiCorp, is located about 14 miles (23 kilometers) northwest of the City of Kemmerer and serves as the primary water supply for Kemmerer Unit 1. This reservoir provides raw water supply to the Naughton Power Plant via an intake structure on Hams Fork River that pumps river water to the Raw Water Settling Basin at the Naughton Power Plant. Viva Naughton Reservoir is also a source of non-safety-related with no special treatment (NST) plant water for


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Kemmerer Unit 1, in that water is diverted from Naughton Power Plant's Raw Water Settling Basin to the plant. Kemmerer Unit 1 does not rely on a surface water bodyto perform safety-related (SR) functions. Kemmerer Reservoir, located about one mile (1.6 kilometers) south and downstream of the Viva Naughton Reservoir, is used primarily for recreational purposes and does not have a regulated spillway. The outflow is approximately the same as the inflow. Because Kemmerer Reservoir is located in a different watershed and is not regulated, it does not impact water supply or contribute to flooding of Kemmerer Unit 1. Information on Viva Naughton Reservoir is found in Section 2.5.4. Average monthly flow rates of Hams Fork River as well as surface water uses are also described in Section 2.5.4. Information on ground water use is provided in Section 2.5.3.

Several ponds with earth dam embankments associated with the Naughton Power Plant are located upstream of the Kemmerer Unit 1 site within the North Fork Little Muddy Creek watershed. Information on these ponds and their potential flooding impact on the site are provided in Section 2.5.1.3.

Flood hazards from external events and mechanisms including probable maximum precipitation on streams and rivers, local intense precipitation, hydrologically and seismically induced dam breaches, probable maximum surge and seiche, probable maximum tsunami, ice effects, and channel diversions are evaluated. Their potential impacts on SR structures, systems, or components (SSCs) are assessed in Section 2.5.1.1 through Section 2.5.1.7.

## 2.5.1 Floods

### 2.5.1.1 Floods

This section examines historical flooding in the vicinity of Kemmerer Unit 1 and summarizes the different types, and combinations of, flood-producing phenomena considered in establishing the flood design basis. The impacts of local intense precipitation (LIP) are also described in this section.

#### 2.5.1.1.1 Flood History

Lincoln County, Wyoming historical flood records (Reference 2.5-3, Reference 2.5-4, and Reference 2.5-5) do not contain reports of major floods near the Kemmerer Unit 1 site.

#### 2.5.1.1.2 Flood Design Considerations

Probable flood-causing mechanisms at the site are evaluated and the postulated flood scenarios are summarized below:

Flooding from Rivers and Streams

The maximum water level at the site resulting from the Probable Maximum Flood (PMF), including climate change effects is 6,752.7 feet NAVD 88, based on simulated flood levels in the North Folk Little Muddy Creek watershed. For the PMF, an antecedent 40 percent Probable Maximum Precipitation (PMP) event is postulated to occur within three to five dry days prior, leaving the ground saturated. The PMF analysis included coincidental wind-wave actions, and a


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20 percent increase in the PMP values due to climate change. At the Nuclear Island (NI), the top of embankment slope is at elevation 6,756.0 feet NAVD 88 and the safety-significant structures have a minimum top of concrete elevation at floor level entrances of 6,758.0 feet NAVD 88, which are 3.3 feet and 5.3 feet, respectively above the PMF water level, including wind wave effects. Further details on the PMF assessment are presented in Section 2.5.1.2.

Flooding from Dam Breaches

Water impoundments located upstream of the site are investigated to determine potential flood hazards to the plant as a result of breaching of dams. Credible failure mechanisms include hydrologic (overtopping), seismic (non-overtopping) and sunny day (non-overtopping) failure scenarios combined with different coincidental conditions. As described in Section 2.5.1.3, Lake Arambel, Flue Gas Desulfurization (FGD) 4, and FGD 5 of the Naughton Power Plant are potential sources of dam breach flood risk to the plant. For hydrologic failures, the impoundments behind the dams are at their maximum operating water levels coincident with a PMP event. For seismic failures, the simultaneous breaching of all 3 dams coincidental with a 500-year precipitation event is postulated. Sunny day failure is not a controlling flood scenario because it is not accompanied by a coincidental extreme precipitation event. Climate change effects are included in the assessment. The water levels resulting from the dam breaches are combined with 2-year wind setup and wave runup to estimate the maximum flood levels at the site. The controlling dam breach event is a hydrologic dam overtopping failure scenario coincidental with a PMF flood event. The maximum dam breach flood level for the site is at 6,755.4 feet NAVD 88 occurring at the northeast corner of the NI, which is 0.6 feet below the top of embankment slope and 2.6 feet below the top of concrete at floor level entrances of safety-significant structures. The flood flow velocities during the dam break flood peak are mild, and of short duration. Channel velocities in the North Fork Little Muddy Creek, which is about 800 feet to the east of the NI embankment, are of the order of 9 to 11 feet per second. Overbank flow velocities adjacent to the NI are in the range of five to six feet per second during the peak. There are no downstream dams whose failure would cause flooding at Kemmerer Unit 1. Flooding due to dam breach is the design basis event for external flood hazards (other than LIP). Details of the dam breach flooding assessment are presented in Section 2.5.1.3.

Flooding from Surges and Seiches

The site is located inland, far from any seacoast and lake shore. In addition, there are no large bodies of open water such as reservoirs or ponds in the immediate vicinity. The North Fork Little Muddy Creek which runs in a generally north to south direction to the east of the site is very shallow with a normal water depth on the order of a couple of feet. Therefore, an evaluation of flooding due to surge and seiche is not required for Kemmerer Unit 1, as described in Section 2.5.1.4.

Flooding from Tsunami

Kemmerer Unit 1 is located approximately 700 miles from the nearest seacoast (Pacific Coast) with the top of embankment slope in the NI at 6,756 feet NAVD 88. Due to the long distance from the coast and the high ground elevation, it is not credible that the site would be impacted by


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tsunami events originated from the ocean. Flood risk due to hypothetical tsunami events originated in nearby water impoundments leading to overtopping or breaching of dams is bounded by the upstream dam break flooding scenarios as described in Section 2.5.1.5.

Flooding from Snow Melt and Ice Jams

Based on a comparison of the general winter storm PMP in addition to snow-water equivalent depths for the seasonal winter snowpack and the local storm PMP depths, it is concluded that the winter PMP on snow event is not a controlling flooding mechanism as described in Section 2.5.1.2.

The potential of ice jam events occurring on North Fork Little Muddy Creek is assessed in Section 2.5.1.6. Based on the assessment, flood risks affecting the safety functions of Kemmerer Unit 1 due to ice jam events are precluded.

Flooding from Channel Diversion

Based on a review of hydrologic, hydraulic, climatic, topographic and geologic evidence, and anthropogenic impacts near the site, channel diversions do not cause flooding hazards or water supply concerns that would affect the safety functions of Kemmerer Unit 1. Details are provided in Section 2.5.1.7.

Additional Considerations

Kemmerer Unit 1 does not require water for SR cooling. Therefore, there is no impact on SR functions or defense-in-depth measures associated with cooling water sources from any blockages or landslides due to natural events. In addition, low water and drought effects do not affect the ability to perform safety functions as described in Section 2.5.4, which also evaluates natural events that may reduce or limit the available raw water supply for the plant's NST cooling water needs.

The precipitation in all the flooding events is increased by 20 percent to account for future increase due to climate change. An evaluation of the climate change adjustment factor for precipitation is included in Section 2.5.1.2.

Assessments of erosion and sedimentation as a result of extreme flood events are described in Section 2.5.1.2, Section 2.5.1.3, and Section 2.5.2.

Dilution and dispersion of accidental releases to the hydrosphere are addressed in Section 2.5.3.2.

#### 2.5.1.1.3 Effects of Local Intense Precipitation

A flooding hazard assessment due to LIP events is performed. The PMP is used to establish the most severe LIP event. The PMP is defined as the upper limit of a theoretical measure of extreme rainfall at a given location. The NI layout and site grading facilitate positive drainage, directing runoff to North Fork Little Muddy Creek on the east side and to a drainage channel on the west side, ensuring that the plant is safe during LIP events. Details of the flooding hazard assessment due to LIP are provided below.


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#### 2.5.1.1.4 Probable Maximum Precipitation Depths

Estimates of the PMP depths for the site are obtained from Hydrometeorological Report Number (HMR) 49 (Reference 2.5-6) published by the National Oceanic and Atmospheric Administration (NOAA). Guidance is provided in HMR 49 for determining both a local and a general PMP for a 6-hour storm duration. The drainage area of the site for LIP is about 11 acres, with a time of concentration as low as 5 minutes. The use of a 6-hour storm duration to evaluate the flooding response of the site to precipitation is deemed appropriate. The larger, more conservative, precipitation depth of the two storm types (local and general storms) is selected for the LIP analysis. For the local storm PMP, the 1-hour 1-square mile value is determined from HMR 49 and adjusted for the elevation and size of the drainage area, and for the storm duration in accordance with HMR 49 methodology. A similar procedure is followed for the general storm PMP, starting with the 24-hour 10-square mile PMP value. The local PMP depth for a 6-hour storm for Kemmerer Unit 1 is estimated to be 10.7 inches, greater than the general 6-hour PMP depth of 4.4 inches and is therefore used to represent the LIP in the flood assessment.

HMR 49 does not provide an estimate for 5-minute rainfall duration. Due to the small drainage area of the NI, short-duration rainfall intensities are expected to be more critical for the determination of flood levels at the safety-significant SSCs. The 5-min PMP rainfall depth is estimated based on the 15-minute to 5-minute rainfall depth ratio derived from NOAA Atlas 14 (Reference 2.5-7), which provides precipitation frequency data for most of the U.S. However, NOAA Atlas 14 does not yet cover the State of Wyoming. Three proxy locations, covered by NOAA Atlas 14, from the neighboring State of Utah with similar topographic and orographic characteristics as Kemmerer Unit 1 are identified: Randloph, Utah (latitude: 41.6278 degrees, longitude: -111.0591 degrees, elevation: 6,909 feet NAVD 88), Woodruff, Utah (latitude: 41.3522 degrees, longitude: -111.0618 degrees, elevation: 6,862 feet NAVD 88), and Coalville, Utah (latitude: 41.2247 degrees, longitude: -111.0579 degrees, elevation: 6,968 feet NAVD 88). The 15-min to 5-minute rainfall depth ratio for extreme storms (100-, 200-, 500-, and 1000-year return periods) is consistently at 0.53 for all three locations and is therefore used to estimate the 5-minute PMP depth. The 6-hour PMP depths at 5-minute increments are given in Figure 2.5-2. The distribution of the 5-minute time increments are determined following HMR 49 guidance, which includes investigating two different storm distributions to determine which produces the highest water levels. The front-loaded storm has the peak intensities occurring during the second hour of the storm and the back-loaded storm has the peak occurring during the third hour. The results indicate no difference in the maximum water levels near safety-significant structures between the two rainfall distributions. For this analysis, the front-loaded distribution is used.

A climate change adjustment factor of 1.2, which is discussed in Section 2.5.1.2.3, is included in the precipitation depths shown in Figure 2.5-2.

#### 2.5.1.1.5 Site Description

There are four safety-significant structures within the NI: the Reactor Building (RXB), Reactor Auxiliary Building (RAB), NI Control Building (NCB), and Fuel Handling Building (FHB). There are also the SR Reactor Air Cooling System stacks, connected to the RXB. The Reactor Air Cooling System stacks have no openings at grade level and therefore there are no external flooding impacts. The NI site grade is at a minimum of 6,756.0 feet NAVD 88, and the top of concrete


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elevation at floor level entrances of non-safety-related with special treatment (NSRST) structures above or having connections to below-grade SR structures is at 6,758.0 feet NAVD 88, both above the design basis flood elevation. The site topography and the location of the safety-significant structures are depicted in Figure 2.5-3.

There are no safety-significant structures in the Sodium Test and Fill Facility (TFF) site located north of the NI, or in the Energy Island (EI) located south of the NI. The site grade of the TFF site is about the same as the NI. The northern part of the TFF site is in a cut area and a portion of the area outside of the TFF site drains towards the site.

For typical storm events, runoff in the NI is collected via ditches, catch basins, and storm drains, and conveyed to stormwater detention ponds and subsequently discharged offsite. However, for the LIP flooding, all storm drains, culverts, and catch basins are assumed clogged. Thus, the storm drain collection system is assumed to be inoperable with no conveyance capacity. All stormwater flow is either overland or directed through surface ditches.

#### 2.5.1.1.6 Flood Elevations

The LIP flood level at the site is estimated using a two-dimensional surface flow model. The hydraulic model, which includes the NI, TFF site, part of the EI, and the surrounding area north of the TFF site, is developed using HEC-RAS (Reference 2.5-8). The EI does not contribute flood flow to the NI because site grading directs stormwater away from the NI. A portion of the EI adjacent to the NI is included in the model to validate this flow pattern. The flood modelling in HEC-RAS is accomplished by first constructing a two-dimensional flow model domain (i.e., the extent of the area of interest where surface flow is to be modelled). A nominal grid size of 5 feet x 5 feet is used for modeling the LIP flood levels around the safety-significant structures.

Flow resistance due to surface roughness is represented by Manning's n. A Manning's n value of 0.03 is assigned for the gravel areas. For paved areas such as roads, parking lots and roofs, a Manning's n value of 0.011 and 0.013 are simulated (Reference 2.5-9). These areas are depicted in Figure 2.5-3. The maximum flood levels at the safety-significant structures are results of the model cases with a higher Manning's n value (0.013) for the paved areas.

The NI, EI, and much of the eastern and western portions of the TFF site are located on top of a fill embankment. The embankment has a side slope of 3 Horizontal to 1 Vertical. The model domain extends to the drainage divide (ridge line) north of the TFF site to include an area where runoff drains to the TFF site. On the west and east side of the computational grid, the model boundary for the NI is specified near the top of the embankment slope. The NI site grade is higher than the design basis flood level of 6,755.4 feet NAVD 88 (not including LIP). The flood level within the NI will not be subjected to backwater effect from the watershed during the design basis flood event. Therefore, a normal flow condition can realistically be applied at these model boundaries. The normal flow boundary condition is established using the average friction slopes.

The LIP rainfall time series (with no interception or infiltration conservatively assumed) is specified over the whole model domain as an unsteady flow boundary condition. Considering 100 percent runoff with no losses is conservative and represents a saturated ground condition.


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The maximum water surface elevation results are presented in Figure 2.5-5 and in Table 2.5-1. The location of the maximum water surface elevation for each structure is indicated in Figure 2.5-5 with a red star. All four locations are in narrow passageways between structures. The modeling cases with a Manning's n value of 0.013 for the paved area produce slightly higher flood levels at the safety-significant structures.

The HEC-RAS modeling results show that the maximum water surface elevation during a LIP event is below the top of the concrete slab, which is at elevation 6,758 feet NAVD 88. The maximum water surface elevation at the safety-significant structures is 6,757.8 ft NAVD 88 (around RXB), which is 0.2 feet below the top of the concrete slab.

LIP water levels are below the top of concrete elevation at safety-significant structures, therefore no safety-significant structures are impacted due to flooding from an LIP event. No above-grade flood protection measures are needed to ensure that there is no adverse flood impact at the safety-significant structures.

#### 2.5.1.1.7 Design Basis Flood Event

The LIP generates the design basis flood level for the safety-significant structures in the NI. The LIP event produces a maximum water elevation of 6,757.8 feet NAVD 88, which is 0.2 feet below the top of the concrete elevation of the safety-significant structures.

Based on the external flooding hazard assessments performed in response to each of the credible flood-causing mechanisms other than LIP, the design basis flood event for the site is the postulated hydrologic dam failure due to the PMP event with a coincidental PMF flood event. This combined event produces a design basis flood level of 6,755.4 feet NAVD 88 for Kemmerer Unit 1, including effects of climate change and coincident wind setup and wave runup. The NI grade elevation at the top of embankment slope is 0.6 feet higher at elevation 6,756.0 feet NAVD 88.

### 2.5.1.2 Probable Maximum Flood on Streams and Rivers

This section describes the assessment of flood hazard from streams and rivers as a result of the PMP over the watershed contributing to the Kemmerer Unit 1 Site. Historical flooding in the region is described in Section 2.5.1.1. Runoff and drainage from roofs of NSRST structures are also discussed in Section 2.5.1.1. Sedimentation and erosion effects are evaluated by considering the hydrological, topographical, and geological properties of the watershed. No adverse risk to the safety of the site as a result of the PMF event is predicted. Erosion and scour protection measures at the site are discussed in Section 2.5.2.

Methodologies described in HMR 49 are used to determine the PMP for the site. A PMP study was developed in 2014 for the Wyoming Water Development Office that covers the State of Wyoming. The estimated 6-hour local storm PMP value for the site based on the Probable Maximum Precipitation Study for Wyoming (Reference 2.5-10) is lower than the value estimated by HMR 49. The PMP estimates from HMR 49, which are more conservative, are therefore adopted in this PMF assessment. The computer program HEC-HMS (Reference 2.5-11), developed by the U.S. Army Corps of Engineers, is used to estimate flood flow and peak discharges in North Fork Little Muddy Creek and tributaries adjacent to the site as a result of the


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PMP. The rainfall-runoff process is modeled using the methodologies in Urban Hydrology for Small Watersheds (Reference 2.5-12) to generate the PMF flow. The U.S. Army Corps of Engineers computer program, HEC-RAS, is used to estimate the maximum water surface elevations at the site during the PMP event. The postulated PMP rainfall event for the PMF is based on the recommendation described in NUREG/CR-7046 (Reference 2.5-13), RG 1.59, “Design Basis Floods for Nuclear Power Plants,” Revision 2 and DG 1290. Combined events are evaluated to ensure that the most severe flood at the plant site is determined.

The development of the PMP rainfall for the PMF event assumes an antecedent rainfall event equivalent to the 40 percent PMP, with three to five dry days between the events leaving the ground highly saturated. Water level increases due to coincidental wind setup and wave runup associated with a 2-year wind speed are considered in conjunction with the PMP event. In addition, the impact of climate change is evaluated as part of the PMF assessment.

For the rainfall-runoff analysis, the watershed is divided into four sub-basins: West, Laydown, East and Confluence, as shown in Figure 2.5-6. The drainage areas of the sub-basins are listed in Table 2.5-2.

#### 2.5.1.2.1 Probable Maximum Precipitation

Estimation of the PMP depths for the watershed contributing to the site considers both local storm and general storm types. Local storm PMP depths (for small drainage areas) and the PMP depths for a general storm (for drainage area of any size) are evaluated in accordance with Reference 2.5-6. The larger, more conservative, depth between the two storm types is used for the PMF assessment. For the local storm, the 1-hour 1-$mi^2$ PMP value is determined from Reference 2.5-6 and adjusted for the elevation and size of the drainage area, and for the storm duration. A similar procedure is used for the general storm, starting with the 24-hour 10-$mi^2$ PMP value and adjusting for elevation and size of the drainage area, and for storm duration. The PMP depth for a 6-hour local storm for the site watershed is estimated to be 9.7 inches, greater than the 6-hour general storm PMP depth estimate of 4.3 inches. The 6-hour local storm PMP depth is therefore used to generate the PMF flood discharge at the site. Figure 2.5-7 shows the hyetograph of the PMP and the 40 percent PMP.

#### 2.5.1.2.2 Hydrologic Properties and Precipitation Losses

Precipitation losses for the watershed are estimated using the runoff methodology in Urban Hydrology for Small Watersheds. A runoff curve number (CN) is estimated for each drainage sub-basin in the watershed. Representative CN values for various land cover and soil conditions are presented in Urban Hydrology for Small Watersheds.

The CN value of each sub-basin is estimated based on its hydrologic soil grouping, land covers, and antecedent moisture condition. The land cover for the site watershed consists of a mixture of grass, weeds, and low-growing brush, with brush the minor element in a semiarid region. This corresponds to a ‘Herbaceous’ land cover in Urban Hydrology for Small Watersheds. Aerial imagery and observations made during site visits indicate that the ground cover condition is 'Fair', with 30 percent to 70 percent coverage of grass, weeds, and brush. The hydrologic soil group designations of different areas within the watershed are taken from the Natural Resource


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Conservation Service (NRCS) Web Soil Survey website (Reference 2.5-14). The FGD ponds, ash ponds, and water supply ponds located in the Sub-basin East are assigned a CN of 98, equivalent to that of impervious surfaces. A CN of 98 is also assigned to the developed areas at the site. The composite curve numbers for the four sub-basins, accounting for the percentage of the developed areas of the site, are summarized in Table 2.5-3.

The NRCS Web Soil Survey (Reference 2.5-15) also provides estimates of the erodibility factor, K, as a measure of susceptibility of a soil to sheet and rill erosion by water. The K factor is used in the Universal and Revised Universal Soil Loss Equation (USLE and RUSLE) to predict average annual soil loss rate. It is empirically determined based primarily on content of silt, clay, sand, and organic matter, soil structure, and saturated hydraulic conductivity. The K factor has a range of 0.02 to 0.69, where a higher K factor indicates higher sheet and rill erosion susceptibility.

The NRCS Web Soil Survey provides a qualitative correlation of the K factor to soil detachment and erosion potential:

- Soils high in clay have low K values, about 0.05 to 0.15, due to their resistance to detachment.
- Coarse textured soils, such as sandy soils, have low K values, about 0.05 to 0.2, because of their low runoff potential even though these soils are easily detached.
- Medium textured soils, such as the silt loam soils, have a moderate K value, about 0.25 to 0.4, because they are moderately susceptible to detachment and produce moderate runoff.
- Soils with a high silt content are the most erodible of all soils and tend to have high K values, greater than 0.4, because they are easily detached, tend to crust, and produce high runoff.

According to the soil classification on the NRCS Web Soil Survey website, the site watershed consists of four dominant soil map units whose erodibility factor varies. Over 41 percent of the watershed belongs to soil unit Cumberhill-Sandbranch-Poposhia complex of 4 to 15 percent slopes, with a K factor rating of 0.1 to 0.15. The second and fourth most dominant soil units are Blazon-Chaperton-Fola complex of 5 to 30 percent slopes and Sandbranch-Monte complex of 0 to 3 percent slopes, with K factor ratings of 0.32 and 0.24, respectively. Together, they represent about 28 percent of the watershed area. The third largest unit is classified as dumps, mine-pits, mine complex. It occupies 16.5 percent of the watershed but no K factor rating is established.

The range of K factors reflects low erosion susceptibility in over 40 percent of the watershed and intermediate susceptibility in about 28 percent of the watershed. Areas of high erosion or unrated susceptibility are located near the boundary of the watershed and at a distance away from the site.


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2.5.1.2.3 Climate Change Assessment

An assessment was performed for the potential impact of climate change on the predicted PMP depths. Probable Maximum Precipitation and Climate Change (Reference 2.5-16) indicates that PMP estimates for the U.S. and other locations around the world may increase by 20 to 30 percent by the years 2071-2100, relative to the period 1971-2000, due to increases in atmospheric moisture. The EPA has an online map showing climate projections over the U.S. (CREAT Climate Scenarios Projection Map [Reference 2.5-17] and Version 3.0 Methodology Guide [Reference 2.5-18]). For the site, an approximate 20 percent increase in the 100-year daily precipitation intensity is projected for the year 2060 (obtained by averaging the “Stormy” and “Not as stormy” projections for 2060).

According to the report of the Fourth National Climate Assessment (Reference 2.5-19), the frequency and intensity of heavy precipitation events across the U.S. have increased more than the average precipitation and the increase is expected to continue over the coming century, with greater increases occurring under a higher emissions scenario. Observed trends and model projections of increases in heavy precipitation are supported by physical relationships between temperature and humidity. These trends are consistent with a warmer world, as increased evaporation rates lead to higher levels of water vapor in the atmosphere, which leads to more frequent and intense precipitation extremes. For the Representative Concentration Pathway 4.5, the intermediate emission scenario in the Intergovernmental Panel on Climate Change (IPCC) Fifth Assessment Report, the region where the site is located appears to be in the 10 to 19 percent range, in terms of the increase in total annual precipitation falling in the heaviest 1 percent of events, by the end of the 21st century. Impacts, Risks, and Adaptation in the United States: Fourth National Climate Assessment (Reference 2.5-19) gives also an estimate of precipitation increase for the region of the project in the range of 20 to 29 percent in the heaviest 1 percent of events for Representative Concentration Pathway 8.6, the high emission scenario in the IPCC Fifth Assessment Report, for the period 2070-2099 relative to 1986-2015.

Based on the above estimates from the literature, a climate change adjustment factor of +20 percent for the PMP depths is used for the PMF assessment of the site.

2.5.1.2.4 Lag Time

The NRCS Unit Hydrograph method is used for the HEC-HMS simulations to transform precipitation excess to runoff flows. Equation 2.5-1 is used to estimate the lag time for the flow path of each sub-basin presented in Figure 2.5-6 (Reference 2.5-20). The estimated lag times are summarized in Table 2.5-4.

$$Lag = (0.00236)\left(\frac{L}{\sqrt{S}}\right)^{0.64} \quad \text{(Equation 2.5-1)}$$


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where:

$Lag$ = Lag time (hrs)

$L$ = Distance along main channel (ft)

$S$ = Weighted average main channel slope, expressed as a percent.

In accordance with guidance in Flood-Runoff Analysis (Reference 2.5-21, page 13-8), the computed lag time values are reduced for PMP events to account for the nonlinear response of the watershed during extreme precipitation events. The rainfall-runoff process is nonlinear, and the response of a drainage basin during an extreme event is expected to be different from that assumed in the derivation of the unit hydrographs. The approach most widely adopted is to apply "peaking factors" reducing the time to peak discharge or increasing the peak discharge. Reference 2.5-13 recommends about 5 to 20 percent increase for the peak discharge and a 33 percent reduction in the lag time. Similarly, to address non-linearity effects, Reference 2.5-21 recommends that the unit hydrograph be adjusted to account for anticipated shorter travel times for large events. In general, a lag time equal to about one-fifth to one-third of the time-to-peak of the unit hydrograph is appropriate. Reducing the lag time has the effect of increasing the peak discharge and decreasing the time to peak for the synthetic unit hydrograph. The lag times for the four sub-basins estimated with Equation 2.5-1 are decreased by 25 percent for the PMP event at the site. Table 2.5-4 shows the estimated reduced lag times for the four sub-basins.

#### 2.5.1.2.5 Peak Discharges during the Postulated Probable Maximum Precipitation Event

To determine the PMF peak discharges at site, the drainage area, lag time, CN values, and PMP depths (with a 20 percent increase for climate change) for the sub-basins are used as input to the HEC-HMS computer model. The flood flow hydrograph for each sub-basin is simulated for a 6-hour PMP event with an antecedent 40 percent PMP event starting 3.75 days prior, leaving a dry period of 3.5 days between the two storms. Figures 2.5-8 through 2.5-11 present the hyetographs and simulated PMF flood hydrographs for the four sub-basins. A summary of the peak discharges, including the climate change effect, for the four sub-basins and at the outlet of the model is shown in Table 2.5-5.

#### 2.5.1.2.6 Maximum Flood Water Levels during the Probable Maximum Precipitation Event

To determine the maximum water levels at the site during the postulated PMP event, the peak discharges estimated from the HEC-HMS model are input to a one-dimensional steady-state HEC-RAS model along with physical data describing the channel and overbank areas of the streams around the site. The model includes the part of the Sub-basin East that extends to Lake Arambel and the two ash ponds, FGD 4 and FGD 5, of the Naughton Power Plant. The locations of the HEC-RAS model cross sections around the site are shown in Figure 2.5-12. The site is partially on fill material with an embankment slope of 3:1 (horizontal: vertical). Roadway and railroad crossings, including a new plant access road to the site are modeled as inline structures. Culverts under the plant access road and all other road crossings are assumed to be blocked in the HEC-RAS model.


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Manning's roughness coefficients (n) for both the stream channel and overbank (floodplain) areas are estimated based on Reference 2.5-22. From aerial imagery and photos taken in the field, the main channel has the characteristics of a mountain stream with brush on the banks, and gravel and cobbles at the bottom of the channel. The floodplains have scattered brush with weeds. The estimated n values for the channel and floodplain areas are 0.04 and 0.05, respectively. Laydown areas and embankment slopes with riprap protection are assigned a conservative n value of 0.07.

The estimated maximum water levels during the PMP event along the Laydown, East Lower (North Fork Little Muddy Creek), West, West Lower and Confluence reaches are presented in Table 2.5-6. These water levels include the climate change adjustment factor (20 percent increase on the precipitation). The maximum water level for the NI is at elevation 6,751.6 feet NAVD 88, reported in the East Lower reach at cross section 8, located at the Northeast corner of the NI. The PMF level requires the addition of wind setup and wave runup generated from a coincidental wind event, which are discussed in Section 2.5.1.2.7.

The simulated maximum water profiles along the East Lower (North Folk Little Muddy Creek), West Lower, Laydown and Confluence reaches are presented in Figure 2.5-13 and Figure 2.5-14.

#### 2.5.1.2.7 Coincident Wind Wave Activity

The wave height and runup elevation are determined for a 2-year wind speed coincidental with the PMP event, using the methodologies described in Reference 2.5-23 and Reference 2.5-24. Following the guidance in Reference 2.5-23, the maximum wave runup is estimated as a function of the significant wave height and the surf similarity parameter. To estimate the significant wave height caused by the 2-year wind, the 2-year mean recurrence interval annual fastest mile wind speed at 30 feet above the ground at the site is obtained from Reference 2.5-25, and adjusted for duration, wind speed above water, and fetch length.

The significant wave height and spectral peak period are calculated using the deep-water equations based on the fetch length (Reference 2.5-23), and accounting for duration, fetch and depth limiting wave growth conditions. The validity of the deep-water wave equations is confirmed by checking the spectral peak period as a function of the water depth against the limiting spectral peak period for deep water waves (4.07 seconds).

The wind setup is estimated as a function of the wind speed, fetch, and water depth using the one-dimensional equation recommended in Introduction to Coastal Engineering and Management (Reference 2.5-26).

Figure 2.5-15 shows the maximum fetch towards the edge of the site measured at the 6,751.6 feet NAVD 88 elevation contour, the maximum water level for the site during the postulated PMP event. The length of this fetch is 2,100 feet, over an average water depth of about 5.57 feet and about 3.95 feet over the half fetch. The maximum wave runup and wind setup are estimated as 1.1 feet.


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2.5.1.2.8 Probable Maximum Flood Water Level

The maximum water level at the Kemmerer Unit 1 NI during the postulated PMP event is estimated as elevation 6,751.6 feet NAVD 88. With the coincidental wind setup and wave runup of 1.1 feet as described in Section 2.5.1.2.7, the maximum PMF water level for the NI of Kemmerer Unit 1 is at elevation 6,752.7 feet NAVD 88.

At-grade and above-grade openings of NSRST structures above or having connections to below-grade SR structures are located above the estimated maximum PMF flood level to prevent water ingress impacting the safety functions of the plant.

2.5.1.2.9 Sedimentation and Erosion Impact

Erosion and sedimentation as a result of the PMF event will not impact the safety of the site. This assessment considers the properties of the contributing watershed and flow paths, including the small size of the basin, surface soils of primarily low erodibility, fair ground cover, gentle channel slopes of 0.2 to 0.3 percent near the site, and an open floodplain. In addition, the flood flow velocities in channels and overbanks are in the low range and the flood peaks recess within a short duration. As can be seen in Table 2.5-6, during the peak of the PMF event, channel velocities in North Fork Little Muddy Creek adjacent to the site are of the order of 8 to 9 feet/second; velocities in the same creek upstream of the site are of the order 9 to 14 feet/second; velocities in the unnamed tributary west of the site, part of which will be channelized, are smaller ranging from 0.3 to 5.3 feet/second. Overbank velocities on the right bank of North Fork Little Muddy Creek adjacent to the site are in the range of 4.1 to 5.0 feet/second and in the unnamed tributary west of the site range from less than 0.5 feet/second to 2.4 feet/second. Nominal level of erosion and sedimentation in the watershed are expected to occur during extreme flood events. However, the hydrological, topographical, and geological properties of the watershed do not contribute to an environment susceptible to large scale mass wasting events. Considering that the NI is at a higher elevation than North Fork Little Muddy Creek, the main flood path, by over 15 feet, and above the PMF flood level by more than 3 feet on an embankment slope that is protected by riprap (Section 2.5.2), erosion and sedimentation will not adversely impact safety-significant SSCs, which are located at a minimum distance of 300 feet from the flood path.

Sediments and debris may be mobilized by erosion and scour during a major flood. Upstream of the Kemmerer Unit 1 site, North Fork Little Muddy Creek and its tributaries pass under the embankments of the railroad and Highway 189 through culverts and an opening for animal crossing. As a result of the constrictions at these crossings, the flow velocities would be much reduced upstream of the embankments and coarser materials would be in part settled and trapped upstream of Highway 189 and the railroad. During the peak of the flood, flow velocities would be higher and sufficient to carry finer sediments past these two embankments and continue downstream beyond the Kemmerer Unit 1 site. High velocities during peak flows may even cause some local scour in the channel of the creek. After the flood peak, as the flow velocities subside, small amounts of sediments may deposit. However, considering that the floodplain extends laterally more than 1000 feet beyond the channel of the creek, any sediment deposits would be spread over a broad area upstream, adjacent to, and downstream of the Kemmerer Unit 1 site. As a result, sediment buildup near the Kemmerer Unit 1 site would be


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minimal. Finally, because such deposition would be occurring after the peak flows had already passed, and that the flood peak subsides rapidly, it is not expected to have an impact on the maximum flood levels.

#### 2.5.1.3 Potential Dam Failures

This section presents an assessment of flooding hazards induced by potential dam failures to Kemmerer Unit 1. The analysis considers flood waves induced by potential dam failures caused by seismic activities, local or distant, and hydrologic events such as the PMF that could affect the site. The release of flood waves from the ponds upstream of the site as a result of dam failure from human activities or other non-natural events would have the same physical behavior and is therefore implicitly addressed in this assessment.

Guidance for the dam breach flooding analysis is provided in RG 1.59, Revision 2, “Design Basis Floods for Nuclear Power Plants,” and Draft Regulatory Guide DG-1290, Proposed Revision to RG 1.59, “Design Basis Flood for Nuclear Power Plants.”

As described in Section 2.5, Kemmerer Unit 1 does not rely on a surface water body as the ultimate heat sink or for other SR water supply. Therefore, flooding and associated impacts from potential dam failures would have no impact on the safety of the plant water supply.

The general hydrologic setting of the site is described in Section 2.5. As explained there, Hams Fork River runs through the City of Kemmerer and is the source of NST raw water to Kemmerer Unit 1. This river has two reservoirs upstream of the City of Kemmerer, Viva Naughton Reservoir and Kemmerer Reservoir. The site is located in the North Fork Little Muddy Creek watershed, outside of the Hams Fork River watershed and is not affected by potential dam breaches on Hams Fork River.

Within the North Fork Little Muddy Creek watershed, the Naughton Power Plant has two ash pond complexes (North Ash Pond [NAP] and South Ash Pond [SAP]), two FGD ponds, (FGD 4 and FGD 5), and one Raw Water Settling Basin. The NAP complex consists of the ash pond itself and a clear water pond referred to as Lake Arambel. PacifiCorp, the owner of the Naughton Power Plant, has submitted a closure plan (Reference 2.5-27) for the ash pond portion of the NAP complex. Only the Lake Arambel portion of the NAP complex is considered for possible dam breach in this analysis. PacifiCorp has also submitted a closure plan (Reference 2.5-28) for the SAP complex including both the ash pond and its associated clear water pond with both these ponds being filled in. The dewatering will also be completed prior to the planned operational date of Kemmerer Unit 1. Therefore, the SAP complex is not considered for possible dam breach in this analysis.

The Naughton Power Plant previously had two additional FGD ponds (FGD 1 and FGD 2). These ponds are closed and backfilled with earth, and therefore are not considered a source of flood risk from dam breaching. The remaining ponds (FGD 4, FGD 5, Raw Water Settling Basin and Lake Arambel) are constructed using earth-fill dams. If any of the embankment dams breaches, the discharge from those ponds would reach North Fork Little Muddy Creek and the Kemmerer Unit 1 Site. The Raw Water Settling Basin is located directly upstream of Lake Arambel; the flood


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wave from its dam breach would reach Lake Arambel first and then travel towards North Fork Little Muddy Creek after combining with the flood wave from the breaching of the Lake Arambel dam.

The dams located upstream of the Kemmerer Unit 1 Site with breaching possibilities are investigated for both hydrologic (overtopping) and seismic (non-overtopping) failures. For the hydrologic failures, the reservoirs behind the dams are assumed to be at the maximum operating water level coincident with a 72-hour PMP event. If such an event results in overtopping, the dam is postulated to breach with an overtopping failure. For the seismic failures, the reservoirs are assumed to be at the maximum water level when a seismically-induced piping failure in the dam is postulated. As described in Section 2.5.1.3.3, the grade elevation for the Kemmerer Unit 1 Site is above the maximum water level from either the hydrologic or seismic dam failure with coincidental wind wave effects; therefore, the breaching of upstream dams under the worst combination of conditions would not have adverse impacts on the safety of the plant. A sunny-day failure of any of the dams would release a smaller flood wave than a hydrologic overtopping event, or a simultaneous failure of the dams during a seismic event with a coincident flood, and is not evaluated further. The dam breach analysis also considers wind-induced wave action (wind setup and wave runup).

The dam breach option in conjunction with the unsteady flow option of the U.S. Army Corp of Engineers (USACE) computer program HEC-RAS is used to determine the breaching flood flow hydrographs and water levels at the site. Breach parameters are estimated using industry standard equations. Coincident flood flow hydrographs used in combination with the dam breach events are determined using the USACE computer program HEC-HMS.

Probable Maximum Precipitation Depths

Two different PMP storms are used in the dam breach analysis. To evaluate the potential of overtopping of upstream dams, a 72-hour PMP storm is used to maximize the precipitation depth and hence the volume of water that each pond would receive during a hydrologic event. A 6-hour PMP storm duration is used to establish the coincident PMF discharge hydrographs in the stream courses during the hydrologic breach event to establish the flow condition downstream of the dams. The shorter storm duration is appropriate because the drainage area of the streams near the site is relatively small with a hydrologic lag time of less than 2.5 hours.

PMP depths for the 6-hour storm have been developed in accordance with HMR 49 for the site as described in Section 2.5.1.2. The 72-hour, 10 square-mile general storm PMP depth obtained from HMR 49 is 10.8 inches. In addition to HMR 49, the Probable Maximum Precipitation Study for Wyoming used data from the National Climatic Data Center (NCDC) through 2012. The Probable Maximum Precipitation Study for Wyoming, Table 11.7 provides a comparison of depths for the shorter duration one-hour, one-square-mile local PMP storm and indicates that the study values near the site are 30 percent lower, or less conservative, than those presented in HMR 49. Thus, the HMR 49 values are used to estimate PMF discharge hydrographs. However, the Probable Maximum Precipitation Study for Wyoming, Table 11.8 provides a comparison of the PMP depths for the 10-square-mile, 24-hour general storm and indicates that the precipitation depths near the site are 50 percent larger than the HMR 49 values. Further, the Probable Maximum Precipitation Study for Wyoming 72-hour PMP depth is 16.1 inches greater than the


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72-hour HMR 49 PMP of 10.8 inches, and it would produce a greater, i.e., more conservative rainfall volume. Thus, to assess pond capacity for potential of overtopping, the Probable Maximum Precipitation Study for Wyoming 72-hour PMP depth of 16.1 inches is used. When performing the hydrologic breach assessment, the local 6-hour storm as determined from HMR 49 and used for the site's PMF in Section 2.5.1.2 is applied to simulate the coincidental flooding in the watershed.

Climate Change Assessment

Climate change impact on precipitation is evaluated in Section 2.5.1.2. Based on this evaluation, a factor of 1.2 is applied to all precipitation depths used in the dam breach analysis.

Drainage Areas

The drainage areas for the ponds are obtained from the hydrologic inflow design reports (Reference 2.5-29, Reference 2.5-30, and Reference 2.5-31) and summarized in Table 2.5-7. The hydraulic capacity report for the NAP complex (Reference 2.5-31) indicates that the combined drainage area for the NAP complex is 765 acres. There is a diversion channel constructed to divert the runoff from the Westmoreland Mine drainage area of 137 acres around the NAP complex. The design capacity of this diversion channel is not documented. Therefore, the full drainage area to the NAP complex of 765 acres is conservatively used in this assessment.

The Raw Water Settling Basin has a drainage area of 107 acres (Reference 2.5-31). It is assumed that the Raw Water Settling Basin would be breached coincidentally with the breaching of Lake Arambel and its entire water volume would be added to Lake Arambel's storage. The drainage areas for FGD 4 and FGD 5 consist of their pond surface areas plus part of their earth embankment.

Dam Embankment Data

The embankment properties and water levels for the ponds are obtained from their Hydrologic and Hydraulic Capacity and Inflow Design Flood Control System Plans (Reference 2.5-29, Reference 2.5-30 and Reference 2.5-31) and as-built construction reports for FGD 5 and FGD 4 (Reference 2.5-32 and Reference 2.5-33) as presented in Table 2.5-8. Each pond has side slopes of 3 horizontal to 1 vertical.

Pond Stage-Storage Volumes

The stage-storage information for the ponds is obtained from the as-built construction plans for FGD 5 (Reference 2.5-32) and FGD 4 (Reference 2.5-33) and the Hydrologic and Hydraulic Capacity and Inflow Design Flood Control System Plan for the NAP Complex (Reference 2.5-31). The storage capacity of the Raw Water Settling Basin is 75.58 acre-feet at its maximum operating water level. This volume is added to Lake Arambel during the hydrologic failure event.


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Cross-Section and Pond Topography

Cross-section geometry used in the HEC-RAS model setup is obtained from the United States Geological Survey (USGS) 3DEP LiDAR data (Reference 2.5-34 and Reference 2.5-35). The cross-section locations are shown in Figure 2.5-16 and Figure 2.5-17. Cross-sections upstream of the pond embankments are taken from topography provided in the pond Hydrologic and Hydraulic Capacity and Inflow Design Flood Control System Plans (Reference 2.5-29, Reference 2.5-30 and Reference 2.5-31).

Manning's Roughness Coefficients

Manning's roughness coefficients (n) for both the stream channel and overbank (floodplain) areas are estimated based on information presented in the HEC-RAS Hydraulic Reference Manual (Reference 2.5-36). From aerial imagery and visual inspections from a site visit, the main channel is a mountain stream with brush on the banks, and gravels and cobbles at the bottom of the channel. The floodplains have scattered brush with weeds. The n values for all stream reaches for the channel and floodplain areas are estimated to be 0.04 and 0.05, respectively (Reference 2.5-36). The site is surrounded by a laydown area on the east, west, and south sides. Portions of this laydown area are in the floodplains of both North Fork Little Muddy Creek and the unnamed tributary. For conservatism, the n value for the laydown area is selected as 0.07 to account for surfaces that are lined with gravels and ripraps.

Channels downstream of FGD 5 and FGD 4 pond embankments rarely have active flows. The overbanks n value of 0.05 is used.

#### 2.5.1.3.1 Seismic Failure

For the seismic failure scenario, each dam is postulated to breach as a result of an internal piping failure caused by a seismic event. A large seismic event could affect all dams at the same time and could result in a simultaneous breach of all upstream dams, as adopted in this assessment.

Two storm flooding event alternatives are considered to coincide with seismic dam failure floods:

Alternative 1

- If the dam fails under the $10^{-4}$ annual exceedance probability (AEP) seismic hazard ground motion peak ground acceleration, assume that failure coincides with the peak water level from a 25-year flood in the watershed above the dam.
- 2-year wind speed applied in the critical direction.

Alternative 2

- If the dam fails under half of the $10^{-4}$ AEP seismic hazard, assume that failure coincides with the peak water level from a 500-year flood (or half PMF, whichever is less) in the watershed above the dam.
- 2-year wind speed applied in the critical direction.


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This assessment considers the possibility that the dams can fail under both the $10^{-4}$ AEP and half of the $10^{-4}$ AEP seismic hazards. With the dams assumed to fail regardless of the seismic hazard AEP, it follows that both Alternatives 1 and 2 would apply. Alternative 2 with the 500-year or half PMF (whichever is less) coincident flood would generate a more severe flood hazard to the site than the 25-year coincident flood event specified in Alternative 1, and therefore is evaluated in detail.

500-Year Flood Flow

An HEC-HMS rainfall-runoff model is developed to determine the 500-year peak discharges in North Fork Little Muddy Creek and the unnamed tributary at the site.

For most of the United States, the NOAA Atlas 14 provides precipitation frequency data. However, NOAA Atlas 14 does not yet cover the State of Wyoming. The current source for precipitation frequency data is NOAA Atlas 2 (Reference 2.5-37) which includes maps of isopluvials for the 6-hour and 24-hour precipitation. Because Atlas 2 does not provide any estimates of the 500-year precipitation, a location is sought in the neighboring State of Utah, covered by Atlas 14, with similar topographic and orographic characteristics and about the same 24-hour precipitation as the Kemmerer Unit 1 Site. The location that meets these criteria is at latitude: 41.2247 degrees, longitude: -111.0579 degrees, elevation: 6,968 feet NAVD 88, and it is approximately 67 miles southwest of the site. Table 2.5-9 lists the corresponding 500-year point precipitation depths for durations from 5 minutes to one day, and these values are increased by 20 percent to account for the effect of climate change.

Because the drainage areas of the four sub-basins contributing runoff to North Fork Little Muddy Creek are small, the point precipitation values for a one-square-mile area as shown in Table 2.5-9 are applied without an area reduction factor.

The lag times for the PMP and the 500-year events are reduced by 25 percent to account for the nonlinear response of the watershed during extreme events. The 500-year, 24-hour storm peak discharges from the HEC-HMS model are summarized in Table 2.5-10.

In addition, the water level at the initiation of a seismic dam failure of each pond takes into account the 500-year, 24-hour inflow volume from the drainage area and storage at its maximum operating water level.

The runoff volume for the drainage area is computed using the NRCS formula (Reference 2.5-38). Section 2.5.1.2 indicates that most of the drainage area upstream of Lake Arambel has a CN of 81, with the exception of the water surface area which has a CN of 98. The surface area of the lake at the maximum operating water level is 2,778,795 square feet (63.8 acres) (Reference 2.5-31), which is 8.3 percent of the drainage area (765 acres), giving a composite CN of 82.4 for the entire drainage area.


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Using a precipitation depth of 3.95 inches and a CN of 82.4, the runoff depth over the Sub-basin Lake Arambel watershed is 2.19 inches. The drainage areas for FGD 5 and FGD 4 are only slightly larger than the pond surface area at the top of the embankments. Thus, for FGD 5 and FGD 4 no precipitation losses are considered, and the full 3.95 inches precipitation depth is used to compute the runoff volumes.

FGD 5

The 500-year 24-hour runoff volume is 17.22 acre-feet. Adding to the pond volume at the maximum operating water level of 6,862.06 feet NAVD 88 gives a water volume of 1,498.22 acre-feet, corresponding to a pond water level of 6,862.42 feet NAVD 88.

FGD 4

The 500-year 24-hour runoff volume is 12.71 acre-feet. Adding to the pond volume at the maximum operating water level of 6,869.56 feet NAVD 88 gives a water volume of 1,379.71 acre-feet, corresponding to a pond water level of 6,869.90 feet NAVD 88.

Lake Arambel

The 500-year 24-hour volume is 139.90 acre-feet. Adding this volume and the volume of the Raw Water Settling Basin (75.58 acre-feet) to the pond volume at the maximum operating water level of 6,890.56 feet NAVD 88 gives a volume of 1,606.53 acre-feet. This produces a water level of 6,893.84 feet NAVD 88.

Dam Breach Parameters

Breach parameters, which consist of the breach width, breach side slopes, depth of breach and time to full breach, are determined using empirical equations.

The HEC-RAS computer program estimates breach parameters using 5 different methodologies, MacDonald et al, Froehlich (1995), Froehlich (2008), Von Thun & Gillette, and Xu & Zhang described in the HEC-RAS Hydraulic Reference Manual. The pond and dam properties relevant to determining breach parameters are summarized in Table 2.5-11. Three of the five methods produce breach parameters that are very similar. The Froehlich 2008 method produces results which are realistic and conservative. Therefore, this method is selected for use in this analysis. The breach parameters used in the HEC-RAS model for each dam are summarized in Table 2.5-12. For the Lake Arambel dam, two sets of parameter values are given, one for a failure due to piping and another for failure due to overtopping. The parameters for failure due to piping apply to the case of seismic failure, while the parameters for failure due to overtopping apply to the case of hydrologic failure.


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Unsteady Flow Analysis of Seismic Dam Failures

Figure 2.5-18 shows a schematic of the HEC-RAS model used for the dam breach analysis. Figure 2.5-19 and Figure 2.5-20 show the upstream and the downstream parts of the model in greater detail. The cross sections of the streams defined based on the topography are shown in these figures as dark green lines. Interpolated cross sections, not shown in the figures, are used in the model to improve the stability of the numerical solution.

Junction 3 between the East Upper and the East Middle Reach is where the stream from FGD 4, represented by the SAP Reach, joins North Fork Little Muddy Creek. Junction 2, between the East Middle and the East Lower Reach, is where the stream from Lake Arambel, represented by the NAP Reach, joins North Fork Little Muddy Creek. The FGD 4 pond is at the upstream end of the SAP Reach, Lake Arambel is at the upstream end of the NAP Reach and the FGD 5 pond is at the upstream end of the East Upper Reach. Junction 1 is at the point where the East Lower and the West Lower Reaches join**.**

To maintain stability in the HEC-RAS unsteady flow computations during periods of low flow before the dam breach flood wave arrives, a constant inflow of 150 cubic feet per second is specified at the HEC-RAS storage area elements FGD 5, FGD 4 and Lake Arambel and at cross sections immediately downstream of these dams.

To account for the 500-year peak flow in the North Fork Little Muddy Creek, inflow hydrographs with constant flows of 9,310 cubic feet per second, 5,670 cubic feet per second, 320 cubic feet per second, and 1,500 cubic feet per second (from Table 2.5-10) are added to the East 21, West 16, Laydown 7, and Confluence 6 cross sections, respectively.

HEC-RAS Model Results

In the simulation it is assumed that all three dams breach simultaneously one hour after the start of the simulation. The simulation shows that the flood waves from each of the three breaches arrive at roughly the same time. The peak discharge at the downstream end of the streams carrying the flow from the three breached dams (East Upper Reach downstream of FGD 5, SAP Reach downstream of FGD 4 and NAP Reach downstream of Lake Arambel) occur at 1:30, 1:37 and 1:43 hours after the start of the simulation, respectively, or 30, 37 and 43 minutes after the breach of the dams.

The site is west of the East Lower Reach between Sections 2 to 8 (the location of these HEC-RAS sections is shown in Figure 2.5-16). The NI is between Sections East Lower 8 and 5 in the HEC-RAS model. The maximum water level for the NI is at elevation 6,754.1 feet NAVD 88 at Section East Lower 8, with a discharge of 73,619 cubic feet per second at the time of maximum stage. The maximum discharge at this section is 74,679 cubic feet per second, occurring 3 minutes before the maximum stage. Figure 2.5-21 shows the stage and flow hydrograph at Section East Lower 8. The maximum water level occurs 2:02 hours after the start of the simulation, or 1:02 hours after the breach of the three dams.


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Figure 2.5-22 shows the maximum water surface elevation along the streams carrying the flood waves from the three breached dams past the site. Table 2.5-13 lists the maximum water surface elevation and corresponding discharge at these sections of the East Lower Reach that are close to the site.

2.5.1.3.2 Hydrologic Dam Failure

For hydrologic dam failure, each of the three pond dams is investigated for potential overtopping during a 72-hour PMP event. The 72-hour PMP for the dam locations is 16.1 inches. Accounting for a 20 percent increase due to climate change, this becomes 19.3 inches. When estimating runoff from a PMP event, it is assumed that the ground has been saturated by prior storms and there are no precipitation losses. The effect of the PMP runoff on the water level and potential overtopping of each dam is investigated.

FGD 5

The drainage area of FGD 5 is 52.3 acres. With 100 percent runoff, the volume entering FGD 5 during a PMP event would be 84.20 acre-feet. Adding this volume to the pond volume at the maximum operating water level at elevation 6,862.06 feet NAVD 88 of 1481 acre-feet yields a volume of 1,565.20 acre-feet. The water level associated with this volume is elevation 6,863.80 feet NAVD 88, which is 1.26 feet below the top of the embankment. The analysis of the wind-induced wave heights discussed in Section 2.5.1.3.3 suggests that the maximum 2-year wind setup and wave runup in FGD 5 is 1.51 feet. Added to the PMP water level above, the maximum wave induced water level is at elevation 6,865.31 feet NAVD 88, which is 0.25 feet above the top of the embankment at elevation 6,865.06 feet NAVD 88, which suggests that the FGD 5 dam may be overtopped during the PMP.

HEC-HMS is used to estimate the time of overtopping of FGD 5 and Lake Arambel by simulating the 72-hour PMP in the sub-basins draining in these reservoirs. For FGD 5, its sub-basin is the pond itself, since no other area drains into it. The simulation started for FGD 5 with the water level at the maximum operation level plus the wind setup and wave runup, which is 6,863.57 feet NAVD 88, and for Lake Arambel at the maximum operation level 6,890.56 feet NAVD 88. The simulation of the 72-hour PMP showed that the dam crest level is reached in FGD 5 40 hours after the start of the PMP, and the dam crest level in Lake Arambel is reached 19 hours after the start of the PMP, a 21-hour difference between the overtopping of the two dams. The worst scenario for the breach of Lake Arambel is if the flood wave released from the dam arrives at the site at the same time as the peak of the PMF from the Sub-basin East. The site is exposed to a lower dam breach flood water level if the flood wave from a breach of the FGD 5 dam arrives after the peak of the PMF occurs. Therefore, the overtopping breach of FGD 5 is not simulated.

FGD 4

The drainage area to FGD 4 is 38.6 acres. With 100 percent runoff, the volume entering FGD 4 during a PMP event would be 62.15 acre-feet. Adding this volume to the pond volume at the maximum operating water level at Elevation 6,869.56 feet NAVD 88 of 1,367 acre-feet yields a volume of 1,429.15 acre-feet. The water level associated with this volume is elevation 6,871.22 feet NAVD 88, which is 3.34 feet below the top of the embankment. The analysis of the wind-induced wave heights discussed in Section 2.5.1.3.3 suggests that the maximum 2-year


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wind setup and wave runup in FGD 4 is 1.49 feet. Added to the PMP water level above, the maximum wave-induced water level in the pond is at elevation 6,872.71 feet NAVD 88, which is 1.85 feet below the top of the embankment. Because this dam will not be overtopped during the PMP event, an overtopping breach analysis for this dam is not required.

Lake Arambel

The drainage area to Lake Arambel is 765 acres. With a CN of 82.4, the runoff volume entering Lake Arambel during a PMP event would be 1,082.25 acre-feet. Adding this volume and the volume of the Raw Water Settling Basin (75.58 acre-feet) to the pond volume at the maximum operating water level at elevation 6,890.56 feet NAVD 88 of 1,391.05 acre-feet yields a total volume of 2,548.88 acre-feet. This volume is much greater than 1,689.27 acre-feet, the storage capacity of Lake Arambel at the level of its dam crest at elevation 6,895.06 feet NAVD 88. This means that the Lake Arambel dam will be overtopped during a PMP event.

PMF HEC-HMS Model

A HEC-HMS model is developed to simulate the Lake Arambel watershed with a drainage area of 765 acres or 1.2 square miles with a CN of 82.4.

The lag time flow path for the Lake Arambel watershed is shown in Figure 2.5-23.

The Chow Lag Formula, used in Section 2.5.1.2 is used to estimate the lag time for the Lake Arambel watershed and is summarized in Table 2.5-14. As described in Section 2.5.1.2, the lag time is reduced by 25 percent to account for the nonlinear watershed response to extreme precipitation events.

The results of the Lake Arambel PMF HEC-HMS analysis are summarized in Table 2.5-15. The flow hydrographs for Lake Arambel and the Sub-basin East are shown in Figure 2.5-24 and Figure 2.5-21. The flow hydrographs in Figure 2.5-24 and Figure 2.5-25 also show the 40 percent PMP flow hydrographs 3 days prior to the full PMP storm. For the HEC-RAS model, only the full PMP portion of the hydrographs is input into the model. Although 40 percent PMP hydrograph is not simulated, the saturated ground effects from the antecedent storm are represented.

The hydrographs from the Lake Arambel PMF HEC-HMS model are used as input into the overtopping HEC-RAS model.

The water level in Lake Arambel at the start of the PMP event is set at the maximum operating level of Elevation 6,890.56 feet NAVD 88. A sensitivity analysis is conducted to determine the dam failure condition that produces the maximum water level at Section East Lower 8 next to the NI, which occurs when the peak of the PMF and the peak of the wave from the failure of the Lake Arambel dam arrrive at the same time. The sensitivity analysis suggests that the maximum water level near the NI is reached when the failure of the Lake Arambel dam occurs after the water level in the lake has reached its maximum level (Elevation 6,896.03 feet NAVD 88), specifically 4:15 hours after the start of the PMP. Because the water level at the time of the breach is higher


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than the seismic failure level, the breach parameters are different than those for the seismic failure of the dam. The breach parameters used to simulate the breach of the dam due to overtopping are provided in Table 2.5-12.

HEC-RAS Model Results

The HEC-RAS simulation of the overtopping of the Lake Arambel dam used the PMP hydrographs obtained with HEC-HMS for the Lake Arambel, East, West, and Confluence sub-basins. The hydrograph for the Lake Arambel sub-basin is added as input to the Lake Arambel storage area, and the hydrographs for the East, West, Laydown, and Confluence sub-basins are added to the East Lower 21, West 16, and Confluence 6 cross sections, respectively.

As in the simulation of the seismic failure case, to maintain stability in the HEC-RAS unsteady flow computations during periods of low flow before the dam breach flood wave arrives, a constant inflow of 150 cubic feet per second is added at the HEC-RAS storage area elements FGD 5, FGD 4, and Lake Arambel and at sections East Upper 26, SAP 9 and NAP 8.

The maximum water surface level at the site occurs at Section East Lower 8 in the vicinity of the NI at elevation 6,754.4 feet NAVD 88, with a discharge of 76,448 cubic feet per second at the time of maximum stage. The maximum discharge at this section is 76,831 cubic feet per second, occurring 2 minutes before the maximum stage. Figure 2.5-26 shows the stage and flow hydrograph at Section East Lower 8. The maximum water level occurs 5:15 hours after the start of the simulation, which is the start of the 6-hour PMP. The maximum discharge at the same location occurs earlier, 5:13 hours after the start of the simulation.

Figure 2.5-27 shows the maximum water surface elevations along the streams carrying the flood wave from the breached Lake Arambel dam past the site.

Table 2.5-16 lists the maximum water surface elevation and corresponding discharge at those sections of the East Lower Reach that are close to the site.

Maximum channel velocities on the east side of the site (East Lower Reach, Sections 4 to 10) range from 9.3 to 11.0 feet per second. The right bank velocities in the same area are in the range of 5.2 to 6.3 feet per second.

A comparison of the results in Table 2.5-13 and Table 2.5-16 suggests that the hydrologic failure of the Lake Arambel dam produces higher maximum water levels at the site, with the maximum water level at the NI being at elevation 6,754.4 feet NAVD 88, compared with a maximum water level of 6,754.1 feet NAVD 88 for the case of simultaneous seismic failure of the three dams.

#### 2.5.1.3.3 Wave Runup and Wind Setup

The wave height and wind runup (including setup) elevation for both the hydrologic (precipitation) and seismic dam failures are determined for a 2-year wind speed with appropriately adjusted duration, in conjunction with the dam breach flood event. Methodologies described in Reference 2.5-42 and in Reference 2.5-43 are used to determine the wave height and runup and setup elevation at the site. The approach to estimate the wave runup and wind setup for the


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dambreak analysis is the same as that used for the wave runup and wind setup in Section 2.5.1.2 for the PMF flood level at the site. Wave runup and setup are estimated for FGD 4 and FGD 5 to support the overtopping assessment.

Figure 2.5-28 shows the assumed fetch for the site and the simulated maximum water surface levels at different HEC-RAS sections along the East Lower Reach for the dam breach case, which results in the highest water levels. The water level along the fetch is assumed to vary linearly between its northeast end elevation of 6,758 feet and its southwest end near Section East Lower 8 elevation of 6,754.4 feet NAVD 88. The length of the fetch is 2,130 feet from the contour line of 6,758 feet NAVD 88 (red line) to the edge of the site. Based on this fetch, the wave runup is estimated to be 0.88 feet and the wind setup 0.11 feet, resulting in a combined total of 0.99 feet.

The fetch length for FGD 5 is based on the maximum length at the top of the embankment dike (Figure 2.5-29). The length is measured to be 2,275 feet. The average depth of the FGD 5 pond along the fetch is 39.24 feet. The wave runup in FGD 5 is estimated to be 1.49 feet and the wind setup 0.02 feet, resulting in a combined total of 1.51 feet.

Adding the wave runup and setup to the PMP water level at elevation 6,863.80 feet NAVD 88 equals a wave runup and wind setup elevation of 6,865.31 feet NAVD 88, which is 0.25 feet above the crest of the dam, suggesting that the FGD 5 dam may overtopped during the PMP.

The fetch length for FGD 4 is based on the maximum length at the top of the embankment dike. The length is measured to be 2,245 feet (Figure 2.5-30). The average depth of the FGD 4 pond along the length of the fetch is 49.66 feet. The wave runup in FGD 5 is estimated to be 1.48 feet and the wind setup 0.01 feet resulting in a combined total of 1.49 feet.

Adding the wave runup and wind setup to the PMP water level at elevation 6,871.22 feet NAVD 88 equals a wave runup and wind setup elevation of 6,872.71 feet NAVD 88 which is 1.85 feet below the top of the embankment at elevation 6,874.56 feet NAVD 88, suggesting that the FGD 4 dam will not be overtopped during the PMP.

#### 2.5.1.3.4 Erosion and Sedimentation

Kemmerer Unit 1 does not rely on a surface water body as the ultimate heat sink, therefore an evaluation of the effects of sediment deposition and erosion due to dam failures on the ultimate heat sink is not applicable.

Assessment of erosion and sedimentation as a result of extreme flood events is described in Section 2.5.1.2. Hydrologic, topographic, and geologic properties of the watershed are considered. Similar to the PMF flood, the predicted channel and overbank velocities resulting from the upstream dam breach events are in the low range and the peak of the flood flow recesses after a short duration. Channel velocities in North Fork Little Muddy Creek adjacent to the site are of the order of 9 to 11 feet per second. Overbank velocities on the right bank of North Fork Little Muddy Creek adjacent to the site (East Lower Reach, Sections 4 to 10) are in the range of 5.2 to 6.3 feet per second. Nominal erosion and sedimentation in the watershed are expected to occur during extreme flood events. However, the watershed is not susceptible to large-scale mass wasting events. Considering that the NI of Kemmerer Unit 1 at a higher


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elevation than North Fork Little Muddy Creek by over 15 feet, and above the estimated dam breach flood level on an embankment slope that is protected by riprap, erosion and sedimentation will not adversely impact the functioning of safety-significant SSCs, all located at a minimum distance of 300 feet from the flood path.

Section 2.5.1.2.9 describes the sediment mobilization and deposition processes during a major flood in the North Fork Little Muddy Creek watershed of Kemmerer Unit 1. During the design basis dam breach event, additional sediment and debris loads are expected from the breach of the Lake Arambel dam coinciding with the PMF. However, the downstream transport and deposition of sediments and debris on North Fork Little Muddy Creek and its floodplain would be similar to that experienced during a major flood. Coarse materials mobilized upstream by erosion and scour would be in part settled and trapped behind US 189 and the railroad embankments, which create flow restrictions on the flood path. During the peak of the flood, the higher flow velocities could carry finer sediments past the embankments and continue downstream beyond the site. The high velocities during peak flows could cause some local scour in the channel of the creek. As the flood peak subsides, sediment deposition could occur. With a broad floodplain of more than a 1,000-feet wide, sediment buildup is expected to be small. Finally, because such deposition would be occurring after the peak flows had already passed, and that the flood peak recedes rapidly as shown in Figure 2.5-26, no negative impact on the maximum flood levels is expected from sedimentation.

#### 2.5.1.3.5 Maximum Dam Breach Water Level at the Kemmerer Unit 1 Site

The results from Section 2.5.1.3.2 indicate that the maximum dam breach flood elevation at the NI is elevation 6,754.4 feet NAVD 88. Adding the 2-year wind induced wave runup and wind setup depth of 1.0 feet, the maximum water level is elevation 6,755.4 feet NAVD 88. This water level is 0.6 feet below the minimum NI top of embankment slope elevation (6,756.0 feet NAVD 88) and 2.6 feet below the top of concrete elevation of NSRST structures (6,758.0 feet NAVD 88). No adverse impact to the safety of the plant systems will be caused by the postulated upstream dam failure events. Scour protection assessment is described in Section 2.5.4.

### 2.5.1.4 Probable Maximum Surge and Seiche Flooding

The hydrologic setting of Kemmerer Unit 1 is described in Section 2.5. The site is not located adjacent to seacoasts or large bodies of open water such as lakes and reservoirs. North Fork Little Muddy Creek is located east of the site as depicted on Figure 2.5-1 and has a shallow water depth on the order of two feet during normal flow conditions. There are a few shallow ephemeral tributaries nearby the site that are normally dry. Therefore, a detailed evaluation of flooding due to surge and seiche is not required for Kemmerer Unit 1.

The Naughton Power Plant, approximately four miles northwest of Kemmerer Unit 1, contains two operating FGD ponds (FGD 4 and FGD 5), two ash pond complexes (NAP and SAP), and a Raw Water Settling Basin. PacifiCorp, the owner of the Naughton Power Plant, has submitted a closure plan (Reference 2.5-44) of the NAP complex, committing to complete closure and filling in of the ash pond and only Lake Arambel of the NAP complex will remain operational. PacifiCorp has also submitted a closure plan (Reference 2.5-45) of the SAP complex, with a commitment to fill in both the ash pond and its associated clear water pond. The remaining ponds that could still


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be in operation during construction of Kemmerer Unit 1 or have the potential to be impounding water when the plant is in operation are FGD 4, FGD 5, Lake Arambel, and the Raw Water Settling Basin, which have limited storage volumes as shown in Table 2.5-17.

In the unlikely scenario that an extreme surge or seiche event causes water to spill over the ponds, the discharge flow over the embankments would be small because of the limited storage capacities of the ponds. The overtopping flow would attenuate before reaching Kemmerer Unit 1, about three miles away. The water impoundments at Naughton Power Plant are formed by embankments with top elevations above ground-level. The flood wave resulting from a breach of these embankments could present a credible flood risk at the site and is assessed in Section 2.5.1.3.

### 2.5.1.5 Probable Maximum Tsunami Hazards

Kemmerer Unit 1 is located more than 700 miles from the nearest seacoast (Pacific Coast) and the site grade of the NI is at a minimum elevation of 6,756 feet NAVD 88. Due to the distance from the coast and the high ground elevation, it is not credible that the site would be impacted by tsunami events originated from the ocean. No historical tsunami data is found for the site and nearby region. Potential flooding of the site by a tsunami event originated in nearby water bodies (two FGD ponds, Lake Arambel, and the Raw Water Settling Basin) is bounded by the postulated upstream dam break flooding events discussed in Section 2.5.1.3 because the volume of water released from the nearby water bodies from a tsunami event is less than that assumed in the dam breach events. Therefore, a detailed tsunami hazard evaluation is not required.

### 2.5.1.6 Ice Effects

Ice effects are considered in the evaluation of hydrological features, external flood hazards and flood protection requirements that are relevant to the safety of Kemmerer Unit 1. Reviews of historical data and projection of the most severe ice formation condition, accounting for climate change effects, conclude that ice formation has no adverse flooding impact on SR SSCs. Ice effects on water supply systems are also evaluated. The plant does not rely on a SR surface water supply. Icing events, therefore, do not impact the safety of the plant relative to surface water supply systems.

The following surface water bodies are included in the assessment:

- Hams Fork River, where the existing river water intake for the Naughton Power Plant (Naughton Cooling Water Intake Structure [CWIS]) is located. Hams Fork River supplies water for Kemmerer Unit 1.
- Raw Water Settling Basin at the Naughton Power Plant, which is used for interim storage of river water withdrawn from the Naughton CWIS on Hams Fork River to support plant operation. Raw water transfer pumps are used to deliver water from the Settling Basin to Kemmerer Unit 1.
- Viva Naughton Reservoir, which releases water to Hams Fork River through an outlet structure at the dam.
- North Fork Little Muddy Creek, which is located on the east side of the site as depicted in Figure 2.5-31.


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2.5.1.6.1 Historical Ice Accumulation

The potential for ice accumulation in the vicinity of the site is evaluated by examining the historical hydrometeorological data and by searching the Ice Jam Database (Reference 2.5-50) developed by the USACE. The Ice Jam Database is a compilation of information on known historical ice jams and ice events, and is maintained by the Ice Engineering Research Division of USACE, Cold Regions Research and Engineering Laboratory.

The Ice Jam Database search focuses on three stream courses in the nearby areas of Kemmerer Unit 1: Hams Fork River, Green River, and Bear River. There are eight historical ice jam events at Hams Fork River recorded in the database, seven events at Green River, and two events at Bear River, as shown in Table 2.5-18, Table 2.5-19 and Table 2.5-20, respectively. There is one ice jamming incident in a section of the Salt River in December 2005 reported in Reference 2.5-51. The record of historical ice jam events in the region indicates that the potential for ice jam formation exists around the site.

Maximum ice thickness is estimated as described in Section 2.5.1.6.3 using long-term historical air temperature measurements at the following nearby meteorological stations obtained from the NCDC of the NOAA (Reference 2.5-52):

- ‘KEMMERER 2 N WY US’ station, approximately five miles north-east of the site, with air temperature data from 6/1/1933 to 5/31/2022.
- ‘FOSSIL BUTTE, WY US’ station, approximately 11.5 miles north-west of the site, with air temperature data from 8/1/1990 to 5/31/2022.

Locations of the two stations are shown in Figure 2.5-31.

2.5.1.6.2 High and Low Water Levels

Historical air temperature data at the two meteorological stations indicates the mean daily temperatures at the site fell below freezing in the winter months. This introduces the possibility of blockage of stormwater drainage systems such as catch basins and culverts. In the evaluation of design basis flooding hazards for the SR SSCs at the site, including flooding due to LIP event, all culverts, catch basins, and storm drains are assumed blocked by ice formation or debris accumulations. The site grade elevation is above the maximum water levels from all postulated external flood events and has a positive ground slope that allows runoff from the LIP event to drain away from SR SSCs as overland flow, without relying on the conveyance of any drainage features (see Section 2.5.1.1). Therefore, complete blockages of the storm drain system by ice formation do not affect the safe operation of the plant.

The possibility of an ice jam formation at North Fork Little Muddy Creek on the east side of Kemmerer Unit 1 (as seen in Figure 2.5-31), is assessed, and determined to have no impact on plant safety as described below.

An ice jam formed upstream of the site could create an ice dam, causing water to pond on its upstream side. When the ice dam breaks, a flood wave from the impounded water is released downstream. Backwater levels recorded during historical ice jam events (Table 2.5-18, Table 2.5-19 and Table 2.5-20) show that the buildup of some ice jams exceeded the bankfull


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level, but only by a small amount, due to the wide floodplain. North Fork Little Muddy Creek is a relatively small and shallow stream with an average depth of approximately 2 feet. The site grade elevation is more than 15 feet above the creek bank, and a flood wave released from the breaching of a hypothetical ice jam on North Fork Little Muddy Creek, upstream of the plant will not flood the plant. Therefore, upstream ice jams do not present a flood risk, affecting the safety of the plant.

An ice jam formed downstream of the site could also create an ice dam. Streamflow impounded by the ice dam will raise the water level on its upstream side. However, water level increase near the site is negligible and will not flood the plant because North Fork Little Muddy Creek is shallow with a wide floodplain and at least 15 feet lower than the site grade.

#### 2.5.1.6.3 Ice Sheet Formation

Because the region experiences subfreezing air temperatures regularly during the winter months, ice formation on surface water bodies, including streams and impoundments, is a common occurrence. Formation of surface ice sheets results in the reduction of usable water volume in the impoundments, and along with frazil and anchor ice, potential blockage of water intakes and reservoir outlet works.

The maximum surface ice thickness that could form at the site is estimated using observed air temperature data for the period of 1934 through 2022 (water years) from the nearby 'KEMMERER 2 N WY US' meteorological station and for the period of 1991 to 2022 (water years) from 'FOSSIL BUTTE, WY US' meteorological station. Surface ice thickness is estimated as a function of accumulated freezing degree days (AFDD) using the modified Stefan equation (Reference 2.5-53). Table 2.5-21 and Table 2.5-22 summarize the estimated yearly peak AFDDs and corresponding ice thickness estimates at the two meteorological stations. As indicated in Table 2.5-21 for 'KEMMERER 2 N WY US', the maximum AFDD, for the water years of 1934 through 2022, is 2,531.5 degree-days occurring on April 27, 2008, with the corresponding ice thickness conservatively estimated at 40.3 inches. Table 2.5-22 for 'FOSSIL BUTTE, WY US' indicates that, for the water years of 1991 through 2022, the maximum AFDD is 2,228 degree-days occurring on April 12, 2008, with the corresponding ice thickness conservatively estimated at 37.3 inches.

The Naughton Power Plant Raw Water Settling Basin supplies plant water to Kemmerer Unit 1. Ice sheet formation in the Raw Water Settling Basin is expected, which results in a reduction of usable water volume. As seen in Figure 2.5-32, the lowest contour of the stage-storage table for the Raw Water Settling Basin (referred to as the Settling Pond) is at elevation 6,922 feet NGVD 29 and the maximum allowable operating level is at elevation 6,938 feet NGVD 29 with a capacity of 75.58 acre-feet. With an estimated maximum surface ice thickness of 40.3 inches, the available capacity of the Raw Water Settling Basin would be reduced to about 56.8 acre-feet based on the design storage capacity curve of the pond. The raw water transfer pumps have a design capacity to deliver 5,270 gpm, equivalent to 23.3 acre-feet per day of flow to meet the water demand of Kemmerer Unit 1. Including the plant water use from Naughton Power Plant, the total water demand is 68.5 acre-feet per day. During an icing event that produces the maximum estimated ice thickness, the Raw Water Settling Basin would hold about 0.8 days of operating flow volume for the plant, assuming the worst-case condition when the Naughton


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CWIS at Hams Fork River is not functioning and not pumping water to replenish the inventory in the pond. The transfer pumps downstream of the Raw Water Settling Basin, that deliver raw water to Kemmerer Unit 1, have design provisions to reduce potential service disruptions due to ice formation. This includes managing surface and frazil ice that are common winter phenomena in the region. The Raw Water Settling Basin does not perform SR functions and ice sheet formation on the surface of the Raw Water Settling Basin does not affect SR SSCs.

Viva Naughton Reservoir is depicted in Reference 2.5-60. The lowest contour on the reservoir capacity curve for Viva Naughton Reservoir is at elevation 7,170 feet mean sea level (MSL) and the normal pool is at elevation 7,242 feet MSL. The estimated maximum surface ice thickness of 40.3 inches does not have a significant adverse effect on the reservoir usable water volume. The outlet structure of Viva Naughton Reservoir is at an invert elevation of 7179.5 feet MSL, 63.5 feet below the normal pool. Surface ice sheets will not reach the outlet structure. Frazil ice may form on the grating protecting the outlet structure, leading to blockage. Blockage of the grating at the Viva Naughton Reservoir outlet structure reduces flow in Hams Fork River and interrupts the water withdrawal service of the Naughton CWIS, thus requiring the plant to use the Raw Water Settling Basin. The Raw Water Settling Basin can sustain operation of Kemmerer Unit 1 for about 0.8 days or less, under the maximum estimated ice thickness condition. Viva Naughton Reservoir does not perform SR functions and ice formation does not affect SR SSCs.

The Naughton CWIS at Hams Fork River (location shown in Figure 2.5-31) could be impacted by surface ice formation. The intake is used, and upgraded as needed, to supply surface water to the Raw Water Settling Basin, which provides interim storage and also functions as a sediment basin. From there, raw water is transferred to Kemmerer Unit 1 for plant use. As shown in Figure 2-6 of Reference 2.5-61, the design low water level of the Naughton CWIS is at elevation 6,969.5 feet. River flow from Hams Fork River is withdrawn through an intake tunnel with a seven foot vertical opening and the top of the tunnel is at elevation 6,970.0 feet. With the maximum surface ice thickness estimate of 40.3 inches, the ice layer could extend below the top of the intake tunnel during low water but it would not lead to a complete blockage of the tunnel opening. Some of the existing equipment in the river intake will be replaced or upgraded to support the operation of Kemmerer Unit 1. The upgrades include provisions such as heating elements and continuous rotation of the traveling water screens to minimize ice formation that would interrupt intake operation.

Ice sheet formation in the cooling tower basin during plant operation is not anticipated due to the waste heat content in the return flow. Ice sheet formation could occur if the plant is shut down for an extended period that coincides with subfreezing air temperatures. The cooling tower basin is not SR, therefore adverse impact to the safety of the plant as a result of interruptions of cooling tower operation because of ice formation is not expected. Similarly, frazil and anchor ice formation will not impact plant operation. No allowance has been made for a reduction in cooling basin storage capacity due to ice sheet formation.

#### 2.5.1.6.4 Ice-Induced Forces and Blockages

The site has no SR SSCs that are subject to ice-induced forces or blockages from either sheet ice or frazil ice formation on open water bodies.


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2.5.1.6.5 Consideration of Other Site-Related Evaluation Criteria

There is no additional site information that indicates an icing scenario more severe than those evaluated as described above.

2.5.1.6.6 Climate Change Assessment

An assessment is performed on the potential impact of climate change on the estimated ice thickness. Based on the report from the Fourth National Climate Assessment (Reference 2.5-54), the annual average temperature over the contiguous United States has increased by 1.2 degrees Fahrenheit (0.7 degrees Celsius) for the period from 1986 to 2015 and by 1.8 degrees Fahrenheit (1 degrees Celsius) relative to the beginning of the 20$^{th}$ century. Additional increases in annual average temperature of about 2.5 degrees Fahrenheit (1.4 degrees Celsius) are expected by the mid-21$^{st}$ century regardless of future emissions, and increases ranging from 3 degrees Fahrenheit to 12 degrees Fahrenheit (1.6 to 6.6 degrees Celsius) are expected by the end of the 21$^{st}$ century, with proportionally greater changes in high temperature extremes compared to low temperature extremes.

Over the past two decades, the number of high temperature records in the United States far exceeds the number of low temperature records. The length of the frost-free season, from the last freeze in spring to the first freeze of autumn, has increased for all regions since the early 1900s. The frequency of cold waves has decreased since the early 1900s, and the frequency of heat waves has increased since the mid-1960s.

High temperatures are projected to increase more than average temperatures. Cold waves are projected to become less intense and heat waves more intense. The number of days below freezing is projected to decline, while the number of days above 90 degrees Fahrenheit (32.2 degrees Celsius) is projected to rise. As indicated in Reference 2.5-55, the climate warming is faster in the high latitudes compared to the tropics, and during winter compared to summer. In addition, high mountain regions are warming relatively more than regions that are low-lying (for reference, Kemmerer Unit 1 is at an elevation of approximately 6,750 feet).

The EPA has created an online map showing climate projections over the U.S. (Reference 2.5-56 and Reference 2.5-57) for different variables (temperature, precipitation). An approximate increase in the average annual temperature projected for the year 2060 is about 5.3 to 6.7 degrees Fahrenheit (2.9 to 3.7 degrees Celsius) for the three scenarios used by the EPA. It is noted that the increase is not specific for the winter months, however it is expected that the average winter temperature will also increase.

On a regional basis for the contiguous United States, the annual extreme temperatures (both cold and warm extremes) are consistently projected to rise faster than annual averages of cold and warm temperatures (Reference 2.5-55).


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Based on Reference 2.5-58, a study at the Peace River in Canada, the ice thickness formation may decrease in the future as a result of climatic warming. Future projections show that under the highest emission scenario, average thickness decreases by about 7.9 inches (0.2 meters) between the 1980s and the 2050s; and by an additional 3.9 inches (0.1 meters) by the 2080s (Reference 2.5-58).

Based on the above assessment, the temperature in the winter months is projected to increase which will produce smaller ice thickness and reduce ice-related impact on the site.

### 2.5.1.7 Channel Diversions

This section describes the effects of channel diversions as potential hazards to Kemmerer Unit 1. Based on a review of hydrologic, hydraulic, climatic, topographic, and geologic evidence, and anthropogenic impacts near the site, as described below, channel diversions are not expected to cause flooding hazards or water supply concerns that would affect the safety functions of the plant.

#### 2.5.1.7.1 Historical Channel Diversions

The Kemmerer Unit 1 Site is bounded by the North Fork Little Muddy Creek on the east side, by an unnamed stream on the southwest and south sides, and by US 189 on the west side (as shown in Figure 2.5-33). South of the site, the unnamed stream joins the North Fork Little Muddy Creek, which continue to flow south for about eight miles where it joins Little Muddy Creek, which flows east to join Hams Fork River, a tributary to Blacks Fork and eventually the Green River, the largest tributary of the Colorado River (Reference 2.5-63). Hams Fork River flows through the City of Kemmerer, north of the site, and then turns to the east away from the site. Many of the smaller streams and creeks in the area are intermittent or ephemeral, flowing only in response to rainfall or snowmelt.

Based on a review of historical maps, there have been no significant changes in the course of streams in the general area of the site over the last 150 years. Figure 2.5-34 includes excerpts from three nineteenth century maps of Wyoming that show the approximate location of Hams Fork River and Little Muddy Creek. The first of these maps is from Hall and Lesley's 1857 geologic map (Reference 2.5-83) and shows Little Muddy Creek but not its North Fork. The second excerpt is from Mitchell's 1880 State, County and Township Map of Wyoming Territory (Reference 2.5-84) and shows the course of Little Muddy Creek and its North Fork, but not labeled. The third is from Rand, McNally & Co.'s 1897 State Map of Wyoming (Reference 2.5-84) and it shows Little Muddy Creek and its North Fork, but labels only Little Muddy Creek. Topographic maps from the USGS are available beginning in 1954 that show the historical course of North Fork Little Muddy Creek and other tributaries near the site. As shown in Figure 2.5-35, Figure 2.5-36, Figure 2.5-37, and Figure 2.5-38 (for years 1962, 1979, 2012 and 2021, respectively) the course of North Fork Little Muddy Creek and its tributaries near the site have remained unchanged between 1962 and 2021, with no evidence of river channel diversions. The same can be seen in four satellite images of the vicinity of the site from 1985, 1994, 2006 and 2020 shown in Figure 2.5-39, Figure 2.5-40, Figure 2.5-41, and Figure 2.5-42


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respectively. Even though there is the potential for landslides in the area, as discussed in Section 2.5.1.7.2, there are no reports of landslides along North Fork Little Muddy Creek since the settlement of this area.

In terms of flooding hazards, Wyoming has experienced some major floods in the past (Reference 2.5-68, Reference 2.5-69), but there were relatively few reported in Lincoln County (Reference 2.5-70) and none of them near the site. A search of publicly available sources did not produce any reports of significant flooding that caused changes in the course of natural channels in the general area of the site.

The nearest streamflow gage is USGS station 09223385 on Hams Fork below the Viva Naughton Reservoir, which has recorded discharge data for nine years, from August 1, 2007 through June 30, 2016. These data are shown in Figure 2.5-43. The maximum instantaneous flow during this period was 1,150 cubic feet per second (Reference 2.5-71), which did not cause any reported flooding. Based on data from 1946 to 1972, the maximum instantaneous discharge on Hams Fork near Frontier (Lat 41° 51\' 26”, Long 110°33\'45") was 2,450 cubic feet per second on May 19, 1950 (Reference 2.5-72). No flooding was reported that day.

#### 2.5.1.7.2 Regional Topographic Evidence

The topography and geology of the site area, which is located near the boundary between the Wyoming Basin and the Northern Rocky Mountains physiographic provinces (Reference 2.5-81), is characterized by north-south trending sub-parallel basins and mountain ranges. The basins and ranges were formed by thrust faulting and folding, normal faulting, and erosion of Paleozoic sedimentary strata. The ranges occur along erosion-resistant uplifted strata, and basins are developed in more erodible down-dropped strata. Small streams have developed along the basin axes with tributary streams ranging from dendritic pattern to trellis pattern nearly perpendicular to the range axes. Major rivers are incised and older than the present basins and ranges within the site region uplifting slowly enough that the rivers were able to maintain their courses, cutting through ridges as they developed. Some faults in the site vicinity are active and still undergoing movement. Section 2.6.1 provides further discussion on faults and other relevant geologic features.

Figure 2.5-44 depicts the Wyoming landslide incidence map for the site. Landslides have occurred hundreds of years ago along the east slope above North Fork Little Muddy Creek. The nearest landslide occurred about 1.7 miles south of the site. The toe of this landslide is about 400 feet from the North Fork Little Muddy Creek and its area is about 3.7 acres. Several smaller landslides occurred over a distance of about 1.7 miles to the south. The toe of most of these is farther from the creek, at a distance of the order of 1,400 feet. The largest of these landslides, located about 3.4 miles south of the site, has an area of about 11 acres and its foot is about 300 feet from North Fork Little Muddy Creek. A large landslide at about 4.5 miles north of the site occurred along the east slope above Hams Fork River. These landslides correspond to locations with slopes containing west-dipping sedimentary rock formations.


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The site is located in an area where the valley is wide. If any new landslides occur in this general area, deposited material from such landslides, because of their distance from the North Fork Little Muddy Creek, would not block its flow in any significant way to cause a backwater or diversions that would flood the site. A landslide along the east slope of the Hams Fork River would not impact the site, as explained in Section 2.5.1.7.5.

Figure 2.5-45 shows the Wyoming deep-seated landslide susceptibility map for the site area. Deep-seated landslides typically involve bedrock, and do not include debris flows or other shallow landslides. Deep-seated landslide susceptibility is classified using strength based on geologic unit lithology and on slope angle. The site has the least susceptibility for deep-seated landslides (class 0). Slopes east of North Fork Little Muddy Creek have moderate to most susceptibility (classes 5 to 10). Slope failures east of North Fork Little Muddy Creek may block one of its tributaries and cause a local diversion of its channel, but this would not have any effect on the site because this location is more than a mile upstream. Similarly, the slope immediately east of Hams Fork River also has moderate to most susceptibility (classes 5 to 9). A landslide along Hams Fork River will not impact the site as explained in Section 2.5.1.7.5.

#### 2.5.1.7.3 Channel Diversions Due to Ice Causes

Ice jams along rivers have occurred in Wyoming, some causing flooding, scouring, and damage in adjacent areas. There are few reports of ice jams in Lincoln County. One reported incident was the ice jamming in a section of the Salt River in December 2005 that caused water flow to be diverted to its old channel, threatening homes in the area (Reference 2.5-70). Another incident on March 15, 2017 caused flooding upstream of Granger along Hams Fork River (Reference 2.5-73). However, no ice jam-induced damage was reported in either case. Backwater effects were reported at the USGS gaging station 09223000, Hams Fork River below Pole Creek, Near Frontier, WY (Lat 42°06\'39.2", Long 110°42\'33.9"), on April 3, 1959, December 31, 2004, March 14, 2007, April 13, 2010, and on March 29, 2012 (Reference 2.5-82). More discussion on ice effects is presented in Section 2.5.1.6. There are no reports of ice formation in North Fork Little Muddy Creek.

There are no permafrost soils near the site. Permafrost soils limit infiltration causing higher runoff. Repeated flooding can thaw permafrost soils, making them more susceptible to erosion. Many high basins of Wyoming had permafrost environments during the Late Pleistocene. In present times, only small areas at high elevations in the alpine range have a mean annual soil temperature below 32 degrees Fahrenheit that would make the occurrence of permafrost possible (Reference 2.5-74). The watershed of the Kemmerer Unit 1 Site is below the alpine range. Therefore, there is no permafrost in this area.

Channel diversions near the site due to ice jams, other ice causes, or permafrost are not expected.

#### 2.5.1.7.4 Flooding of the Site Due to Channel Diversions

As indicated above, there is no credible evidence that flooding as a result of channel diversions could occur near the site or affect plant safety. Nominal erosion and sedimentation take place in natural streams as part of the stormwater runoff process. Depositions may build up over time, leading to changes in the course of a channel. Such changes would be gradual, localized, and of


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small scale if they were to occur because of the small size of the Kemmerer Unit 1 watershed. Because the floodplain of North Fork Little Muddy Creek is wide, disruptions to flow and associated flood level increase as a result of small local channel diversions would be minor and would dissipate within a short distance. Furthermore, the elevation of the NI is greater than 15 feet above North Fork Little Muddy Creek, protecting SR SSCs from flood risk associated with the channel. No landslides have occurred along North Fork Little Muddy Creek upstream of the site. The area east of a small tributary to North Fork Little Muddy Creek is characterized by the Wyoming State Geological Survey as having moderate susceptibility to landslide (Figure 2.5-45), given a score of 5 on a scale from 0 to 10, where 10 indicates the areas that are most susceptible to landslides and 0 indicates areas least susceptible. If such a landslide were to occur in part of the channel of the small eastern tributary to North Fork Little Muddy Creek, located more than a mile upstream of the site, it would have only a local effect and not affect flow conditions along the North Fork Little Muddy Creek because of the small drainage area of the tributary in question.

The historic landslides south of the site are between 300 and 2,000 feet from North Fork Little Muddy Creek. Further movement of these landslides or additional similar landslides should not impact its channel because of the distance of the deposited materials from the North Fork Little Muddy Creek.

An evaluation of the topographic and geological characteristics and historical landslide fields mapped in the area up and down river of North Fork Little Muddy Creek near the site showed that channel diversions leading to flooding of the site due to mass wasting activities are not considered credible.

#### 2.5.1.7.5 Human-Induced Causes of Channel Diversions

The Viva Naughton Dam on Hams Fork River located approximately 15 miles upstream of the City of Kemmerer is a potential human-induced cause of channel diversions in the vicinity of the plant. The reservoir stores 45,465 acre-feet of water (Reference 2.5-75). The primary purpose of the reservoir is to store water for use by the Naughton Power Plant. The Kemmerer Reservoir, which has minimal storage and is used primarily for recreation, is immediately downstream of the Viva Naughton Reservoir.

The worst-case flooding hazard associated with channel disruption on Hams Fork River would involve a failure of the Viva Naughton Dam combined with a landslide near Kemmerer that would block the flow of water further downstream. A map of historical landslides in Wyoming (Reference 2.5-77) does not show any landslides near the dam. However, a surficial soils map (Reference 2.5-78, Figure 2.5-46) shows a landslide area on the left (east) bank of Hams Fork River, about two miles downstream of Kemmerer. Figure 2.5-48 shows a cross section across Hams Fork River near the location of this landslide area. The location of this cross section, which extends into the watershed of the Kemmerer Unit 1 Site, is shown in Figure 2.5-47. The elevation difference between Hams Fork River and the basin boundary with North Fork Little Muddy Creek watershed where the plant is located is about 200 feet (Figure 2.5-48). If the Viva Naughton and the Kemmerer dams fail and the entire volume of their reservoirs is impounded by landslide deposits blocking Hams Fork River, the water surface behind the landslide will be well below the basin boundary of the two watersheds. Therefore, it would not have any impact on North Fork


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Little Muddy Creek. For the same reason, local changes in Hams Fork River due to erosion or deposition of materials following a failure of the Viva Naughton and Kemmerer dams would have no effect on plant safety.

It is concluded that channel diversions along Hams Fork River caused by anthropogenic activities, such as the operation of dams and reservoirs, would not have any impact on the site.

#### 2.5.1.7.6 Alternative Water Sources

The source of water supply for the plant is Hams Fork River, a tributary to the Green River, northeast of the site. However, because the plant does not require an SR water supply, loss of water supply, even in the highly unlikely event of a channel diversion along the Hams Fork River, does not compromise the plant safety.

Plant water supply and low water considerations are described in detail in Section 2.5.4.

#### 2.5.1.7.7 Other Considerations

About a mile south of the site, North Fork Little Muddy Creek is joined by an ephemeral stream draining the watershed upstream of Blazon Gap (Figure 2.5-49). Three streams, from the north, northeast and south, and join at the Blazon Gap and continue as a single tributary to the North Fork Little Muddy Creek. Gravel, cobbles, and boulders of different sizes fallen from steep slopes of the Blazon Gap watershed can be seen on the banks and bed of these streams. The drainage area upstream of Blazon Gap is relatively small, approximately 3.9 square miles, which is about 15 percent of the area of site watershed (26 square miles). Topographic information indicates that the valley floor is wide downstream of the gap. The main channel that drains the Blazon Gap watershed is an ephemeral stream, incised and of gentle grade that continues outside the gap towards North Fork Little Muddy Creek. The slope of the channel at and downstream of the Blazon Gap is about 1.3 percent. Upstream of the Gap, there are three tributaries to the main channel, two having slopes of about 2 percent and the other having a slope of about 3 percent. The upper reaches of these tributaries are steeper near the Blazon Gap watershed divide. It is expected that debris from the upper slopes inside Blazon Gap would mostly be deposited on the floodplain of the incised channel with some finer materials carried outside the gap towards North Fork Little Muddy Creek. Finer materials carried by major flood flows when the water level rises over the banks of the channel would be spread and deposited over a large area without causing significant obstruction in North Fork Little Muddy Creek. The deposits in the area immediately downstream of the Blazon Gap watershed resemble that of an inactive alluvial fan with a small surface slope and appear to have been formed by gradual deposition of sediments rather than sudden debris flows carrying large volumes of solids. Figure 2.5-50 shows the left (east) bank of North Fork Little Muddy Creek at the edge of this inactive alluvial fan. In parts of the bank, interbedded layers of silt and gravel deposits are visible (see Figure 2.5-51), likely deposited by occasional debris flow over time. No boulders, cobbles, or large angular gravels are visible at the surface of the fan or on the floodplains of nearby stream courses, including North Fork Little Muddy Creek. In addition, the channel of the stream coming from Blazon Gap is well incised with no evidence of having carried large quantities of debris recently, suggesting that this area has not been depositionally active in modern times. Cobbles and gravel can be seen on the bed of this stream (Figure 2.5-52), most likely carried during floods or fallen from its banks. However, there


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is no evidence of material deposition where this stream joins North Fork Little Muddy Creek. Therefore, potential large debris flow from Blazon Gap is not expected to cause a backflow that would affect the design basis flood level at the Kemmerer Unit 1 Site.

#### 2.5.1.7.8 Conclusions

A review of hydrologic, hydraulic, climatic, topographic and geologic evidence, and anthropogenic impacts on North Fork Little Muddy Creek and the unnamed tributaries near the Kemmerer Unit 1 Site, and on the Hams Fork River indicates that channel diversions are not expected during the operating life of the plant. Moreover, in the unlikely event that a channel diversion were to occur, it would be localized, of small scale, occur gradually over time, and have no adverse impact on the safety of the plant.

### 2.5.2 Flooding Protection

The design basis flood elevation at Kemmerer Unit 1 is established at 6755.4 feet NAVD 88, as presented in Section 2.5.1.3. The design basis flood elevation is a result of a hydrologic (overtopping) failure of the Lake Arambel dam coincident with a PMP event as detailed in Section 2.5.1.3. This flood elevation, which also includes the wind setup and wave runup caused by a two-year return period wind speed applied in the critical direction, would be reached along the northeast boundary of the NI. Figure 2.5-1 shows the location of the Kemmerer Unit 1 Site and surrounding streams. The NI site grade is at a minimum of 6,756.0 feet NAVD 88, and the top of concrete elevation at floor level entrances of NSRST structures above or having connections to below-grade SR structures is at 6,758.0 feet NAVD 88, both above the design basis flood elevation. The safety functions of the plant are not affected by the design basis flood event.

The flood hazard assessment resulting from LIP events is described in Section 2.5.1.1. The NI layout and grading facilitate positive drainage, directing runoff to North Fork Little Muddy Creek on the east side and to a drainage channel on the west side, ensuring that the plant is safe during LIP events. The detailed flood model confirms that there is no adverse impact on the NI safety-significant SSCs during the postulated LIP flood event.

As discussed in Section 2.5.3.1, the maximum groundwater elevations in the NI are predicted to be in the range of 6,737 to 6,738 feet NAVD 88, 18 to 19 feet below the minimum site grade elevation (6,756.0 feet NAVD 88). The backfill material used to raise the east and west sides of the NI is estimated to have a hydraulic conductivity of about 28 feet per day (0.01 centimeters per second). This hydraulic conductivity would limit the lateral distance that flood waters could travel in the subsurface zone through the riprap revetment and embankment fill material before the flood peaks subside. As the duration of elevated water levels during extreme flooding conditions, including the design basis flood event would be short, a few hours or less, the migration of groundwater could not reach the safety-significant structures located about 300 and 400 feet from the embankments on the east and west sides. Finally, below-grade concrete foundation walls of structures are damp proofed and waterproofed as discussed in Section 6.1.1.2.


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Because no safety-significant SSCs are adversely affected by inundation from any of the postulated flood events, no above-grade flood protection measures are required within the NI to prevent stormwater runoff from impacting the safety of the plant. Additionally, no technical specifications or emergency procedures to implement flood protection activities are required.

The design basis snow and probable maximum winter precipitation load on the roofs of NSRST structures is presented in Section 2.4.1.3.5.

The natural terrain near the center of the NI in the north-south direction exceeds the minimum site grade elevation (6,756.0 feet NAVD 88), as shown in Figure 2.5-53. However, the natural terrain of the east and west sides is lower, and therefore is raised to at least the minimum site grade elevation. The NI is bordered on the east and west by embankments having 3H:1V (horizontal to vertical) side slopes with the top of slope elevation at 6,756.0 feet NAVD 88. To the north of the NI is the TFF, which has the same site grade as the NI. To the south is the EI, which has a site grade three feet lower. The toe of the embankment on the east side of the NI is above the 100-year floodplain of North Fork Little Muddy Creek. The boundary of this floodplain, per the FEMA flood insurance rate map of this area (Reference 2.5-85), lies approximately 500 feet to the east of the embankment toe.

Although outside the 100-year floodplain of the nearby water courses including North Fork Little Muddy Creek, the toe of the embankment where the NI is situated is expected to be inundated during extreme floods such as the postulated events of PMF and the failure of the Lake Arambel dam coincidental with a PMP. Based on the flooding analysis results described in Section 2.5.1.1, the maximum flow velocity along North Fork Little Muddy Creek near the embankment toe is mild, ranging from five to six feet per second, and of short duration, even during the most severe of all credible flood events. Riprap is selected to provide scour and erosion protection of the embankment slope and toe.

The east embankment is protected from scour and erosion by a rock riprap lining. Figure 2.5-53 shows a schematic map of the linear extent of the revetments along the east side of the NI. As indicated in Section 2.5.1.3, the maximum flow velocity at the embankment during the worst postulated flood event ranges from five to six feet per second along the east side of the plant. The flood wave resulting from the dam breach scenario that defines the worst postulated flood event at the plant is of short duration and would rise, crest, and fall within a couple of hours. The nominal median (equivalent diameter) rock size in the riprap layer for the east embankment is 15 inches. The safety-significant structures are located approximately 300 feet or more from the east embankment. Any displacement of the riprap layer during a severe flood event is expected to be localized and can be repaired readily without affecting the safety of the plant.

The east embankment toe is protected from undermining by scour in the North Fork Little Muddy Creek floodplain during extreme flooding, including the design basis flood event. Potential scour during such extreme events is estimated to be of the order of two to three feet, based on an analysis of general scour using several estimation methods. The relatively low scour estimates obtained from this assessment are consistent with the characterization of the study area as having low to moderate susceptibility to erosion indicated in Section 2.5.1.2 and Section 2.5.1.3.


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Along the west side of the NI, the maximum flood elevation resulting from the worst postulated flood scenario is 6,748.5 feet NAVD 88, occurring at its northwest corner and resulting from the PMF. This is lower than the maximum flood elevation along the east side and the minimum NI site grade of 6,756 feet NAVD 88. The maximum flow velocity during the PMF event is about two feet per second at the embankment and about five feet per second in the channel near it, as indicated in Section 2.5.1.2. The PMF has a low risk of occurrence, and if it were to happen, the flood flows would overtop the channel banks for only a short duration of an hour or less. Therefore, any risk of scouring that could affect the integrity of the embankment is highly unlikely. Nevertheless, the embankment slope is vegetated to improve its resistance against localized surface erosion. Any erosion or scouring of the vegetated embankment, which is over 400 feet from the nearest safety-significant SSC, during a severe flood is expected to be localized and can be repaired readily without affecting the safety of the plant.

## 2.5.3 Groundwater

### 2.5.3.1 Groundwater Characterization

This section describes the regional and local groundwater characteristics and groundwater use at and in the vicinity of the Kemmerer Unit 1 site. The site is in southwestern Wyoming approximately 5 miles south of the City of Kemmerer in Lincoln County (Figure 2.5-54). This section provides the following information:

- Regional and local groundwater aquifers, sources, and sinks (Section 2.5.3.1.1)
- Regional and local groundwater use (Section 2.5.3.1.2)
- Present and known and likely future withdrawals (Section 2.5.3.1.3)
- Subsurface properties that affect movement of contaminants in groundwater (Section 2.5.3.1.4)
- Geochemical characteristics (Section 2.5.3.1.5)
- Groundwater levels including their seasonal and climatic fluctuations (Section 2.5.3.1.6)
- Groundwater flow paths, gradients, and velocities (Section 2.5.3.1.7)
- Groundwater monitoring and protection requirements (Section 2.5.3.1.8)
- Subsurface hydrostatic loading (Section 2.5.3.1.9)
- Reliability of groundwater resources for safety-related purposes (Section 2.5.3.1.10)
- Reliability of dewatering systems (Section 2.5.3.1.11)

#### 2.5.3.1.1 Regional and Local Groundwater Aquifers, Sources, and Sinks

Geologic Setting

The site is in the Middle Rocky Mountains physiographic province near the eastern edge of the North American Cordilleran Thrust Belt's Wyoming salient. A major portion of the Middle Rocky Mountains province resides in Wyoming and is bordered to the east by the western margin of the Wyoming Basin physiographic province (Figure 2.5-55). The site vicinity is characterized by a series of thrust sheets, dipping westward, from the Cretaceous Sevier orogeny.


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The Kemmerer Unit 1 site is located within the western extremity of the Greater Green River Basin (GGRB) drainage boundary (Figure 2.5-56). Geologic features within the GGRB are shown in Figure 2.5-57. The site is located east of the Absaroka thrust fault and the geologic units are of Cenozoic and Mesozoic age (Figure 2.6-8). The site is underlain by the Hilliard Shale of Upper Cretaceous age, and alluvium and colluvium of Holocene to Pleistocene age (Figure 2.6-8). Approximately 3.7 miles to the west lies the Elkol strip mine in the Adaville Formation, which is a coal-bearing formation of Upper Cretaceous age (Figure 2.6-8). To the east, approximately one mile from the Kemmerer Unit 1 site, lies the north-south trending Oyster Ridge, which is part of the Oyster Ridge Sandstone Member of Upper Cretaceous age. Unconsolidated gravel deposits are distributed within the Cumberland Flats, located between the Kemmerer mine to the west and Oyster Ridge to the east (Figure 2.6-8). The geological section (Figure 2.6-18) is located 3 miles south of the Kemmerer Unit 1 site.

The Hilliard Shale is a dark olive-gray marine shale, siltstone, and sandy shale containing thin, tan to light-gray sandstone and limestone interbeds. Approximately 3 miles north of Kemmerer Unit 1, the thickness of the Hilliard Shale varies from 2,500 to 3,000 feet. Below the Hilliard Shale is the westerly dipping (25 degrees to 30 degrees) coal-bearing Frontier Formation of approximately 2,000 feet in thickness and Upper Cretaceous age. Below the Frontier Formation is the westerly dipping Aspen Shale of approximately 600 feet in thickness and Lower Cretaceous age. Below the Aspen Shale is the westerly dipping Bear River Formation of approximately 1,000 feet in thickness and Lower Cretaceous age. Below the Bear River Formation lies the westerly dipping Gannett Group (Lower Cretaceous) and Stump Sandstone and Preuss Redbeds (Upper Jurassic) of approximately 500 feet in thickness. The bottoms of the Stump Sandstone and Preuss Beds are likely below sea level elevation at the site. The Elkol strip mine is located at the eastern edge of the asymmetric Lazeart syncline with the eastern edge of the limb dipping at an angle of 30 degrees. The western limb of the syncline is much steeper at about 55 degrees. The Absaroka fault truncates the western limb of the Lazeart syncline, and the axial trace of the syncline parallels the Absaroka thrust fault.

Hydrogeologic Setting

Quaternary alluvium and colluvium deposits occur along the major drainage basins of the GGRB. They are composed of interbedded and unconsolidated mixtures of clay, silt, sand, and gravel deposits located along stream and river channels. They are generally less than 50 feet thick but can exceed 100 feet in thickness locally. The Quaternary alluvium and colluvium are unconsolidated deposits. The Wyoming Framework Water Plan classifies the alluvial deposits a major aquifer where the deposits are sufficiently permeable and water saturated. The bedrock formations also contain aquifers with flow dominantly through weathered bedrock and fractures or joints. Shallow groundwater flow is controlled by the topography and stream or river drainage patterns. Shallow groundwater within 300 to 500 feet below ground surface occurs as heterogeneous and anisotropic flow at both local and regional scales. Within the bedrock formations, shallow groundwater flow generally is unconfined; however, deeper in the structural basins, the groundwater flow is confined by the low permeability strata between the highly permeable layers of sandstone, coal, limestone, and dolomite beds. Artesian (confined) groundwater flow is observed in some areas of the GGRB where the confined piezometric head is greater than the ground surface elevation.


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Groundwater recharge occurs along the outcrops (i.e., along the margins of the structural basins) and flows downward towards the structural basins within the GGRB (Figure 2.5-61). Potential recharge by direct infiltration of precipitation to Mesozoic hydrogeologic units in the GGRB is shown in Figure 2.5-62. Recharge to the aquifers at the site is low (i.e., winter precipitation is less than 8 inches as shown in Figure 2.5-62). The Hilliard Shale lies underneath the Kemmerer Unit 1 site below the alluvial soil and is classified as an aquitard with a highly weathered section (less than 30 feet) and unweathered rock with less fracture frequency as depth increases. Groundwater flows primarily along the weathered section of the Hilliard Shale with significantly less flow along the deeper part of the Hilliard Shale where groundwater flows through isolated fractures.

Groundwater discharge occurs along stream drainages as springs or as subcrop flow into overlying geological units. Groundwater and surface water interactions are dominant along the streams and river channels and recharge to the groundwater occurs from surface water sources in these areas. Table 2.5-23 shows the mapped outcrop area percentages of hydrogeologic units in the Wyoming GGRB. As shown in Table 2.5-23, the Cenozoic hydrogeologic unit outcrop accounts for 83 percent of the GGRB area.

Figure 2.5-63 shows hydrogeologic nomenclature for Mesozoic rocks in the GGRB in Wyoming. The Hilliard Shale, which underlies the Kemmerer Unit 1 site, is classified as a major aquitard while the deeper Frontier Formation is classified as a minor aquifer.

The mean annual precipitation for 1951–1980 ranged from about 8 to 60 inches within Lincoln County (1996 Water-Resources Investigations Report; Reference 2.5-88), with a mean value of approximately 8 inches at the Kemmerer Unit 1 site (Figure 2.5-64).

#### 2.5.3.1.2 Regional and Local Groundwater Use

Regional Use

Estimated groundwater usage for Lincoln County in 1993 was 7.3 billion gallons, where irrigation and public water supply accounted for the largest amount of the groundwater usage (approximately 71 percent for irrigation and 26 percent for public water supply). The remaining approximately 3 percent of the groundwater usage was for domestic systems, livestock, industrial, and mining (1996 Water-Resources Investigations Report, Table 5). The Kemmerer and Diamondville public water supply, which is the public water supply nearest to the Kemmerer Unit 1 site, was supplied solely by surface water from the Hams Fork River in 1993.

Based on the 1996 Water-Resources Investigations Report by Eddy-Miller et al., only six towns (Figure 2.5-65) within the GGRB received their primary water supply from groundwater. These towns, the aquifer formations, and distances from the Kemmerer Unit 1 site, are as follows:

- Town of Bairoil (Battle Springs Formation) – 159 miles
- Town of Big Piney (Wasatch Formation) – 62 miles
- Town of Marbleton (Wasatch Formation) – 63 miles
- Town of Opal (Green River Formation) – 13 miles
- Town of Superior (Erickson Sandstone) – 83 miles


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- Town of Wamsutter (Wasatch Formation) – 134 miles

Table 2.5-24 shows the municipal groundwater users, population, number of wells, depth of wells, and amount of groundwater used in 2005.

Local Use

Based on data provided by the 2023 Wyoming State Geological Survey (Reference 2.5-93), there are a total of 326 permitted water wells within a 5-mile radius of the Kemmerer Unit 1 site. The wells are categorized according to permitted use and include domestic, stock watering, industrial, irrigation, monitoring, miscellaneous, and other. Monitoring wells account for 76 percent of all permitted water wells within the 5-mile radius and are clustered mostly northwest and southwest of the Kemmerer Unit 1 site. Figure 2.5-66 shows only those permitted water wells which have reported values for extraction rates (although rates are not shown on this figure). Domestic water wells constitute 13 percent of the permitted wells within the five mile radius and are located to the northeast of the Kemmerer Unit 1 site, close to the City of Kemmerer. Miscellaneous and stock watering wells account for approximately 6 percent and 3 percent, respectively. The remaining 2 percent of the permitted water wells are permitted as irrigation (one well), industrial (four wells), and other well (one well). The closest permitted water well is about 1.3 miles northwest of the site and recorded as an abandoned monitoring well. The closest active downgradient well is about 1.6 miles south of the site and is permitted for monitoring by Chevron U.S., Inc. There are six permitted springs within the 5-mile radius of the Kemmerer Unit 1 site (Figure 2.5-66). The closest spring is about 1.7 miles southeast of the site and it is permitted to Blazon Coal Co. Inc. for domestic, industrial, municipal, and steam use.

According to the database, there are no permitted water wells or springs within a 1-mile radius of the Kemmerer Unit 1 site. The majority of the 1-mile radius is classified as a major aquitard, which is a poor producer of groundwater with poor groundwater quality (Figure 2.5-63).

#### 2.5.3.1.3 Present and Known and Likely Future Withdrawals

Regionally, future groundwater use projections for the six towns in the GGRB, described in Section 2.5.3.1.2, have been developed for low, medium, and high population growth rates for the years 2015, 2035, and 2055. Table 2.5-25, Table 2.5-26, and Table 2.5-27 summarize the projected use for low, moderate, and high population growth rates, respectively.

Within a 1-mile radius of the site, no such groundwater use projections have been published. However, because the Hilliard Shale is considered an aquitard, it is unlikely that water-supply wells will be developed in this formation in the future. Even if water-supply wells were developed in the vicinity of the Kemmerer Unit 1 site, they would be low-yielding and would have limited potential for reversing or altering groundwater flow directions on the site.

#### 2.5.3.1.4 Subsurface Properties that Affect Movement of Contaminants in Groundwater

The primary subsurface properties that effect groundwater and contaminant movement at the Kemmerer Unit 1 site include the hydraulic conductivity, effective porosity, and density of the bedrock. These properties were characterized as part of the Kemmerer Unit 1 site subsurface investigation, which included drilling and logging 160 borings (includes offset borings),


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conducting 28 packer tests, slug testing, sampling, and monitoring 52 groundwater observation wells, and conducting laboratory tests on soil, rock, and groundwater. Additional information on the subsurface investigation program is presented in Section 2.6.4. Results of the packer testing, slug testing, and laboratory rock testing are provided below. Groundwater geochemical sampling results are discussed in Section 2.5.3.1.5 and groundwater level monitoring results are presented in Section 2.5.3.1.6.

Borehole Packer Testing

A borehole packer test is a constant head test of an isolated interval in a borehole to determine the hydraulic conductivity. For the Kemmerer Unit 1 site investigation, a straddle packer arrangement was used to isolate the test zone in 7.5-foot intervals. A total of 28 packer tests were performed in 15 open boreholes during the field investigation. The locations of the tested boreholes are shown on Figure 2.5-67. Testing depths ranged from approximately 20 to 50 feet below ground surface, all in the Hilliard Shale. Of these tests, four were not able to achieve any flow, while in one test the flow rate did not vary with pressure; consequently, it was not possible to estimate hydraulic conductivities for these five tests.

The tests were performed and interpreted in accordance with USACE (Reference 2.5-91), Method 381-80. Table 2.5-28 presents the test results organized by midpoint depth interval (i.e., 20 to 30 feet below ground surface, 30 to 40 feet below ground surface, and 40 to 50 feet below ground surface). Summary statistics for the three midpoint depth intervals and the complete data set are provided in Table 2.5-29. The results in Table 2.5-29 indicate that the hydraulic conductivity of the Hilliard Shale decreases as depth increases. The decrease of hydraulic conductivity with depth corresponds with observed decreased rock weathering and fracture frequency.

Observation Well Slug Testing

The slug test method involves creating a sudden water level displacement in the observation well and observing the water level recover as it returns to the pre-test level. A total of 52 observation wells with 4-inch diameter slotted screens were installed in 2022 as part of the Kemmerer Unit 1 site subsurface investigation. Observation well locations are shown on Figure 2.5-71. The wells are screened within the Hilliard Shale bedrock. Rising head and falling head slug tests were completed in 48 of the wells using either pneumatic slugs, solid slugs, or water slugs. In some wells, multiple rising head and falling head tests were conducted to obtain data with sufficient initial head change and to demonstrate sufficiently reliable response for analysis. Slug testing could not be performed in OW-119-U, OW-127-U, OW-136-M, and OW-168-U due to low water level in the wells at the time of testing.

Slug testing was performed following Section 8 of ASTM D4044/4044m-15 (Reference 2.5-86). Hydraulic conductivity was estimated from the resulting test data using the Bouwer and Rice Slug Test method (Reference 2.5-87). Results are summarized in Table 2.5-30 for both weathered rock and fresh rock. In some instances, water levels in observation wells had not recovered from well installation and development, even after several months had elapsed (characteristic of observation wells completed in formations with very low hydraulic conductivities). In these cases, the hydraulic conductivity was estimated from long-term recovery water level measurements


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acquired over many months. The geometric mean of the weathered rock was estimated to be 6.2E-01 feet per day (2.2E-04 centimeters per second), whereas the geometric mean of the fresh rock was substantially lower and estimated to be 1.7E-03 feet per day (5.9E-07 centimeters per second).

Rock Properties

Laboratory rock testing included determination of porosity and specific gravity (numerically equivalent to density, assuming a water density of 1 gram per cubic centimeter). Porosity and specific gravity testing were conducted in accordance with The Complete International Society for Rock Mechanics Suggested Methods for Rock Characterization, Testing and Monitoring: 1974-2006 (Reference 2.5-89). Completed test results are included in the 2023 Subsurface Investigation Data Report (Table 14-1). Table 2.5-31 provides a summary (porosity and specific gravity were not determined for all test intervals shown in Table 14-1 of the 2023 Subsurface Investigation Data Report). A review of the data in Table 2.5-31, along with the summary statistics, indicates that the porosity tends to be higher at depths less of than 50 feet below ground surface than for greater depths, while the converse is true for specific gravity. These results suggest that the rock properties in the upper 50 feet below ground surface differ from those at greater depths.

Subsurface Property Summary

Hydraulic conductivity data acquired from borehole packer testing and observation well slug testing of the Hilliard Shale are aggregated and plotted as a function of depth in Figure 2.5-68. These results indicate that the formation exhibits extreme heterogeneity, with hydraulic conductivity values spanning nine orders of magnitude and ranging from 1.0E-07 to 1.0E+2 feet per day. The results also indicate a trend of decreasing hydraulic conductivity with increasing depth, which is generally consistent with the transition from highly weathered rock to fresh rock with increasing depth.

Similarly, the porosity data obtained from laboratory rock testing are plotted as a function of depth in Figure 2.5-69. The results show the porosity to be relatively heterogeneous in the upper 50 feet below ground surface, ranging from 2.6 percent to 8.9 percent in this depth interval, and then decreasing with depth and averaging 3.0 percent at depths greater than 50 feet below ground surface.

Lastly, the bulk specific gravity data resulting from laboratory rock testing, numerically equivalent to rock density, are plotted as a function of depth in Figure 2.5-70. The data reflect a decreasing trend in bulk specific gravity with increasing depth, with the bulk specific gravity becoming relatively constant with an average of 2.506 for depths greater than 50 feet below ground surface.

#### 2.5.3.1.5 Hydrogeochemical Characteristics

Groundwater samples were obtained from 22 observation wells for the purpose of geochemical and environmental characterization of groundwater at the site. Two surface water samples were also collected from North Fork of Little Muddy Creek for environmental characterization. Groundwater and surface water samples are listed with field parameter measurements in


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Table 2.5-32. A summary of groundwater and surface water geochemical, environmental, and radiometric analytical results is provided in Table 2.5-33a through Table 2.5-33e. Analytical results for chloride, sulfate, and pH are provided in Table 2.5-34.

The results shown in Table 2.5-32 indicate the groundwater generally has a neutral pH, low dissolved oxygen, high electrical conductivity, and negative oxidation-reduction potential. Results in Table 2.5-33a through Table 2.5-33e and Table 2.5-34 show high concentrations of the major anions and cations and high total dissolved solids concentrations, consistent with the high electrical conductivity field measurements.

#### 2.5.3.1.6 Groundwater Levels Including Their Seasonal and Climatic Fluctuations

Groundwater level measurements were collected from a network of 52 observation wells installed at 24 site locations, with 4 of the locations consisting of 3-well clusters and 20 of the locations consisting of 2-well clusters. Each well within each cluster is identified with a suffix of either U, M, or L. The three-well clusters include the U, M, and L series of wells, whereas the two-well clusters include the U and M series of wells. The U suffix identifies the upper series of wells, which are the shallowest wells, predominately screened within the highly weathered and fractured siltstone and ranging in depth from 20.0 to 50.0 feet below ground surface, except OW-187-U screened from 49.3 to 59.3 feet below ground surface. The M suffix identifies the next lower series of wells that are screened to depths varying from 40.0 to 85.0 feet below ground surface, except OW-119-M screened from 35.2 to 45.2 feet below ground surface. Similarly, the L suffix identifies the next deeper series of wells that are screened at depth intervals varying from 95.9 to 150.0 feet below ground surface. The locations of the observation wells are depicted on Figure 2.5-71. Table 2.5-35 provides construction details for the observation wells.

Two piezometers with pressure transducers, one measuring the water level in the creek (PZSW) and the other within the creek sediment bottom (PZGW), were installed at the North Fork Little Muddy Creek (Figure 2.5-71). PZGW was installed approximately 2 feet within the creek bed sediment. A staff gauge was also installed to measure the creek water level; however, hydrographs are plotted based on the PZSW data as it provided continuous data. Transducer data from PZSW and PZGW were collected from July 20, 2022 through December 29, 2022 at 15 minute intervals. Data after December 29, 2022 are not available because the staff gauge, PZSW, and PZGW stations were destroyed by severe winter weather.

The observation well network described above was used to characterize the temporal fluctuations in groundwater level. These data were acquired by two means: (1) manual groundwater level measurements from the 52 observation wells, collected once a month; and (2) continuous (hourly) groundwater level measurements using pressure transducers installed in 20 selected observation wells. The data include nine months of monthly manual water level data collected between August 17, 2022 and April 26, 2023, and transducer water level readings recorded from observation wells between July 20, 2022 and March 30, 2023. The monthly manually-measured groundwater levels are included in Table 2.5-32 and their hydrographs are plotted in Figure 2.5-72 through Figure 2.5-96. Site precipitation data from April 9, 2022 to April 19, 2023, collected hourly, are plotted on the hydrographs in units of inches for the purpose of assessing groundwater level response to precipitation.


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The following is inferred from the hydrographs and precipitation data:

- Observation wells OW-108-M, OW-109-M, OW-117-M, OW-117-L, OW-121-M(A), OW-127-M, OW-149-M, OW-153-M, and OW-167-M are still recovering from well development, which was completed in July 2022.
- There is no clear relation between measured groundwater level and precipitation data during this monitoring period.
- Observation wells OW-106-U and M, OW-109-U, OW-117-U, OW-119-U and M, OW-121-U, OW-122-U and M, OW-136-U, OW-137-U and M, OW-140-U and M, OW-149-U, OW-164-U and M, OW-168-M, and OW-179-U and M show a decreasing groundwater level during this monitoring period.
- Observation wells OW-108-M, OW-109-M, OW-117-M and L, OW-121-M, OW-122-L, OW-127-M, OW-136-M, OW-149-M, OW-153-U and M and L, OW-167-M, OW-185-U and M, OW-189-M, and OW-191-U and M show an increasing groundwater level during this monitoring period.
- Observation wells OW-108-U, OW-127-U, OW-167-U and L, OW-168-U, OW-186-U and M, OW-187-U and M, OW-188-U and M, OW-189-U, and OW-190-U and M show very little change in groundwater level during this monitoring period.
- The maximum groundwater elevation observed at the site during the monitoring period is 6,734.92 feet NAVD 88 at OW-153-L, which is 2.5 feet above ground surface. OW-153-L is located approximately 1,150 feet southwest of the NI (OW-122 well cluster).
- In general, observation well clusters show higher groundwater levels in the U series wells near the central area of the site and show higher elevation in the M series wells near the periphery of the site.
- The groundwater level difference between the U and M series wells is smallest (less than 1 foot) at well clusters OW-119-U and M, OW-122-U and M, OW-137-U and M, OW-140-U and M, OW-164-U and M, OW-168-U and M, OW-179-U and M, OW-187-U and M, OW-189-U and M, OW-190-U and M, and OW-191-U and M.
- The water level differences observed for PZSW (surface water) and PZGW (streambed sediments) in the North Fork Little Muddy Creek indicate a downward vertical gradient, suggesting that the stream is losing water to the groundwater.

#### 2.5.3.1.7 Groundwater Flow Paths, Gradients, and Velocities

Groundwater Level Contour Maps

Groundwater level contour maps were prepared using the manually measured groundwater levels included in Table 2.5-36 for the purpose of characterizing groundwater flow paths and gradients. It was assumed that the U, M, and L series wells monitor separate hydrogeologic units within the Hilliard Shale. Maps for the upper unit are based on water levels measured in 24 U series wells, while maps for the middle unit are based on measurements acquired from 24 M series wells. For the lower unit, measurements are available for only 4 L series wells and insufficient for creating contour maps; therefore, maps were not prepared for the lower unit. Groundwater level contours for the upper and middle units were manually interpreted and drawn


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for the period from August 2022 through April 2023. Figure 2.5-97 through Figure 2.5-114 display the resulting groundwater level contour maps and groundwater flow paths across the Kemmerer Unit 1 site. These results indicate that groundwater flows from the west, north, and east, converging near the center of the site, and then flows south towards the southerly site boundary. These groundwater flow paths did not exhibit significant variation for the August 2022 through April 2023 period.

Horizontal Hydraulic Gradients

The groundwater level maps were used to estimate horizontal hydraulic gradients at the Kemmerer Unit 1 site. For each map, horizontal hydraulic gradients were estimated for selected groundwater flow paths using the formula:

$$i_h = \Delta h / L \qquad \text{(Equation 2.5-2)}$$

where: $i_h$ = horizontal hydraulic gradient; $\Delta h$ = water level difference between observation wells along the flow path; and $L$ = the horizontal distance between the wells. Table 2.5-37 presents the results from the upper and middle series of wells along easterly (OW-187-U to OW-137-U and OW-187-M to OW-137-M), westerly (OW-185-U to OW-186-U and OW-185-M to OW-186-M), and southerly (OW-106-U to OW-164-U and OW-106-M to OW-164-M) flow directions as shown in the groundwater level maps. The horizontal hydraulic gradients for the upper and middle series of wells vary from 0.002 to 0.015 and from 0.003 to 0.019, respectively. For the upper series of wells, the average hydraulic gradients along the easterly, westerly, and southerly flow directions are 0.004, 0.014, and 0.003, respectively. For the middle series of wells, the average hydraulic gradients along the easterly, westerly, and southerly flow directions are 0.004, 0.019, and 0.004, respectively. Thus, the average horizontal gradients between the upper and middle series of wells are similar.

Vertical Hydraulic Gradients

The vertical hydraulic gradient ($i_v$) is calculated by dividing the difference in groundwater level ($\Delta$h) between well pairs by the vertical distance between midpoint screen elevations ($\Delta$z) of the U, M, and L series wells. Table 2.5-38 presents the calculated vertical hydraulic gradients between the cluster well pairs from August 2022 through April 2023 and provides a time-averaged vertical gradient for each of the well pairs. Figure 2.5-115 shows the spatial variation of the time-averaged vertical gradient for each well pair.

Water level data from the U and M well pairs generally indicate a downward gradient between upper and middle well pairs in the central area of the site (except OW-106-U and M, OW-137-U and M, and OW-140-U and M), and an upward gradient along the periphery of the site (OW-185-U and M, OW-187-U and M, OW-188-U and M, OW-189-U and M, and OW-191-U and M). The three-well clusters OW-122 and OW-167, located in the central and south part of the site (respectively), exhibit a downward gradient between upper, middle, and lower well pairs. Three-well cluster OW-117 exhibits an upward vertical gradient between middle and lower well pairs. OW-117-L has been recovering since July 2022 (when well development ended). As a result, the vertical gradient between upper and lower well pairs reversed from


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downward to upward in January 2023. The three-well cluster OW-153 exhibits a downward gradient between upper and middle well pairs, and an upward vertical gradient between both upper-lower and middle-lower well pairs. The average vertical gradient between well pairs OW-191-U and M, OW-187-U and M, OW-164-U and M, and OW-179-U and M is small (less than 0.004), suggesting the water level is similar between the wells.

For all the wells, the average downward gradient varies from 0.0002 to 1.6259, whereas the upward gradient averages between -0.0020 and -1.7165. Deeper groundwater flows upward near well clusters OW-117 and OW-153, resulting in an upward vertical gradient (between upper-lower and middle-lower well pairs) in these wells.

Groundwater Velocities

Based on the subsurface properties described in Section 2.5.3.1.4 and the groundwater flow paths and gradients described above, contaminants released to the saturated zone at the Kemmerer Unit 1 site would be transported in the southerly direction within the upper, most conductive interval of the Hilliard Shale. A conservative estimate of the groundwater velocity is provided in Section 2.5.3.2.3.2 for the purpose of assessing an accidental release of liquid effluent to the groundwater.

### 2.5.3.1.8 Groundwater Monitoring and Protection Requirements

Consistent with the NEI 2007 groundwater initiatives (Reference 2.5-90), a groundwater monitoring program will be developed as part of detailed design activities for the Kemmerer Unit 1 site prior to fuel load.

### 2.5.3.1.9 Subsurface Hydrostatic Loading

Hydrostatic uplift pressures for plant foundations and other safety-significant SSCs were calculated from the following equation:

$$p_w = (h - z)\gamma_w \qquad \text{(Equation 2.5-3)}$$

where: $p_w$ = hydrostatic uplift pressure (pounds per square foot); $h$ = groundwater head (feet); $z$ = elevation of foundation base (feet); and $\gamma_w$ = specific weight of water (62.40 pounds per cubic foot). In applying this equation, the maximum groundwater levels were obtained from a three-dimensional groundwater model developed for the Kemmerer Unit 1 site that accounts for the alteration of the hydraulic properties associated with the subsurface portions of the plant and fill used during construction. Conservative parameter values were used to estimate the maximum groundwater levels. Results using Equation 2.5-2 are summarized in Table 2.5-39 and show that the uplift pressure ranges from 811 pounds per square foot for the NCB to 6,178 pounds per square foot for the reactor shaft, which is the deepest structure at the site.


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2.5.3.1.10 Reliability of Groundwater Resources for SR Functions

The Kemmerer Unit 1 design does not rely on groundwater to perform SR functions or as a source of cooling water, potable water, or other plant needs. Plant water needs are described in Section 2.5.4.

2.5.3.1.11 Reliability of Dewatering Systems

The Kemmerer Unit 1 design does not rely upon a permanent dewatering system to maintain groundwater levels at or below the design basis groundwater level during plant operation.

During plant construction, dewatering is required for plant foundations excavations that extend below the water table. To estimate construction dewatering rates, a three-dimensional groundwater model was developed for the Kemmerer Unit 1 site. Excavations for the NI were incorporated into the model and the groundwater level maintained below the base of the excavations to allow for dry working conditions. Using this approach, the maximum dewatering rate is expected to be 50 gpm. This is a conservative estimate, which assumes all excavations have been opened simultaneously.

2.5.3.2 Accidental Releases of Liquid Effluents to Ground and Surface Waters

This section provides a conservative analysis of a postulated accidental release of radioactive liquid effluents to the environment at the Kemmerer Unit 1 site. The postulated failure mechanism and source term are used in the methodology to evaluate radionuclide transport in groundwater to potential receptors. The radionuclide concentrations to which a water user might be exposed and the associated dose are compared against regulatory limits.

The analysis meets the requirements of PDC 2 as it relates to 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.

In addition, the analysis meets the requirements of 10 CFR 100 as it relates to identifying and evaluating hydrological features of the site in accordance with 10 CFR 100.23(d) and 10 CFR 100.20(c), as applicable.

2.5.3.2.1 Failure Mechanism and Radioactivity Releases

It is postulated that a Liquid Radwaste Processing System tank outside of containment ruptures with its contents released to the environment. A simplified, conservative release scenario is assumed for which 80 percent of the tank volume is transferred instantaneously to the groundwater at a point within the NI and no credit is taken for the time required for radionuclides to travel from the ruptured tank to the saturated zone and for radioactive decay and other attenuation mechanisms to occur during unsaturated zone transport.


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Based on the observed groundwater levels and contour maps discussed in Section 2.5.3.1.6 and Section 2.5.3.1.7, respectively, the radioactive liquid tank contents, once released, would enter the saturated zone underlying the NI and migrate in a southerly direction towards the south boundary of the Kemmerer Unit 1 site. This groundwater pathway is illustrated in Figure 2.5-116.

Figure 2.5-117 illustrates the conceptual model used to evaluate an accidental release of radioactive liquid tank contents to groundwater. During saturated zone transport, radionuclide concentrations of the liquid released to the groundwater are reduced by the processes of adsorption, hydrodynamic dispersion, and radioactive decay.

The Liquid Radwaste Processing System is a zero-liquid release design. There are no effluents, or release pathways outside of any building. Spills or leakage will go into the building sump pits and be processed. The only potential pathway is from a catastrophic tank failure that overfills the sump pits and fills the plant floor beyond the lowest floor berm. Therefore, an accidental release of liquid effluents to surface waters is not considered credible and has not been evaluated.

#### 2.5.3.2.2 Radioactive Source Term

The Liquid Radwaste Processing System Waste Holdup Tank contains the largest inventory of radioactivity outside of the RXB. Kemmerer Unit 1 uses two 100 percent capacity, atmospheric pressure tanks located in the FHB that collect all liquid radioactive waste prior to processing. The redundant tank allows for maintenance and surge capacity. The tanks receive water from various sources in the FHB and the FAB. Each tank has a capacity of 4,000 gallons. As described above, it is assumed that 80 percent of one tank is transferred instantaneously to the groundwater. In addition, tank contents are processed when they are at 80 percent of capacity. Therefore, 3,200 gallons of liquid are assumed to be released.

The radionuclides expected to be present in a Waste Holdup Tank and their activity concentrations for commercial fuel are listed in Table 2.5-40, along with their radioactive half-lives. The progeny retained for analysis are limited to those considered necessary for dosimetric purposes. The effluent concentration limits (ECLs), as given in 10 CFR 20, Appendix B, Table 2, Column 2, are included in Table 2.5-40 for reference.

#### 2.5.3.2.3 Transport in Groundwater

The radionuclide transport analysis estimates the radionuclide concentrations that might result in exposure to a member of the public using groundwater obtained from a hypothetical water-supply well located at the southern boundary of the Kemmerer Unit 1 site. It is postulated that a Waste Holdup Tank fails such that its contents migrate vertically downward through the unsaturated zone and instantaneously enter the shallow groundwater system that occurs in the upper portions of the weathered siltstone underlying the site. Once in the shallow groundwater system, radionuclides are expected to migrate horizontally in the southerly direction towards the southern site boundary.


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2.5.3.2.3.1 Methodology

Radionuclide concentrations in the groundwater at the site boundary, to which a member of the public could be exposed via a domestic well, are estimated using a two-dimensional analytical model described by Codell and Duguid (Reference 2.5-95). This model allows calculation of solute concentrations in an aquifer at any point in the horizontal plane downgradient of a release as a function of time and accounts for advection, hydrodynamic dispersion, retardation, and radioactive decay. For the case of an instantaneous point source, vertically-averaged over the aquifer thickness, the concentration is estimated using Equation 4.29 of Codell and Duguid (Reference 2.5-95):

$$\frac{C}{C_0} = \frac{V}{n_e R} X_1 Y_1 Z_2 \qquad \text{(Equation 2.5-4)}$$

where

$C = C(x, y, t)$ = radionuclide concentration (microcurie per cubic centimeter)

$C_0$ = radionuclide concentration in the liquid source (microcurie per cubic centimeter),

$V$ = volume of liquid release (cubic feet),

$n_e$ = effective porosity,

$R$ = retardation factor, and

$X, Y, Z$ = Green's functions in the x, y, z coordinate directions, respectively.

Green's functions are defined as follows:

$$X_1 = \frac{1}{\sqrt{4\pi D_x t/R}} exp\left[-\frac{x - Ut/R^2}{4D_x t/R} - \lambda t\right] \qquad \text{(Equation 2.5-5)}$$

$$Y_1 = \frac{1}{\sqrt{4\pi D_y t/R}} exp\left[-\frac{y^2}{4D_y t/R}\right] \qquad \text{(Equation 2.5-6)}$$

$$Z_2 = \frac{1}{b} \qquad \text{(Equation 2.5-7)}$$

in which

$x$ = longitudinal distance from source (feet),


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$y$ = transverse distance from source (feet),

$t$ = time (day),

$b$ = aquifer thickness (feet),

$D_x$ = longitudinal coefficient of hydrodynamic dispersion (square feet per day),

$D_y$ = transverse coefficient of hydrodynamic dispersion (square feet per day),

$U$ = average linear velocity (feet per day),

$\lambda = ln(2)/t_{1/2}$ = decay rate (1/day), and

$t_{1/2}$ = radioactive half life (day).

Equation 2.5-4 determines the exposure concentration time history for a hypothetical receptor by setting x equal to the distance from the point of release to the site boundary and by setting y = 0. Equations 2.5-5 through 2.5-7 conservatively assume that a domestic well withdraws groundwater from the centerline of the plume for which radionuclide concentrations would be highest.

2.5.3.2.3.2 Parameter Assignment

Equation 2.5-4 includes several parameters that characterize the transport of radionuclides in groundwater. The basis for assigning values to each of the parameters is provided below.

Aquifer Thickness

An aquifer thickness of b = 10 feet is used. As described in Section 2.5.3.1.1, this is the approximate thickness of the completely and highly weathered siltstone, which extends from 20 to 30 feet below ground surface.

Average Linear Velocity

The average linear velocity, $U$, is computed by dividing the specific discharge, calculated from Darcy's law, by the effective porosity, that is,

$$U = \frac{K}{n_e}\frac{dh}{dx} \qquad \text{(Equation 2.5-8)}$$

where: $dh/dx$ = hydraulic gradient; $K$ = hydraulic conductivity (feet per day); and $n_e$ = effective porosity. Values for these quantities have been established in Section 2.5.3.1.4 and Section 2.5.3.1.7 and are taken as follows for this analysis:


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- $K$ = 17 feet per day, the maximum hydraulic conductivity observed in the weathered rock,
- $n$ = 0.059, the average total porosity n observed in the upper 50 feet,
- $n_e$ = 0.047, the effective porosity is 80 percent of n (Reference 2.5-96, Figure 2.17), and
- $dh/dx$ = 0.004, the maximum horizontal hydraulic gradient observed for the southerly flow path observed for the upper and middle wells (described in Section 2.5.3.1.6).

Using the above values, an average linear velocity of $U$ = 1.44 feet per day is obtained.

Groundwater Transport Distance and Advective Travel Time

An accidental release of radioactive liquid in groundwater would occur only in the NI. Current head measurements indicate that such a release would migrate in a southerly direction. The distance from the NI to the southern site boundary, where a receptor could be exposed to radioactivity in the groundwater, is about 3,785 feet. Therefore, a transport distance of L = 3,785 feet is assumed (Figure 2.5-116). Based on an average linear velocity of U = 1.44 feet per day, an advective travel time of $t_a$ = L/U = 3,785 feet / 1.44 feet per day = 2,627 days is calculated.

Coefficients of Hydrodynamic Dispersion

The longitudinal dispersivity is estimated using the relation between dispersivity and groundwater transport distance given by Xu and Eckstein (Reference 2.5-97, Equation 14b). This equation was chosen because it provides a higher weight to measured data points that are reliable (Reference 2.5-97). This equation is given as

$$\alpha_L = 0.83(\log_{10}L)^{2.414} \quad \text{(Equation 2.5-9)}$$

The length scale, L, and longitudinal dispersivity, $\alpha_L$, in the above equation are in meters. The length scale used in the above equation is L = 1,154 meters (3,785 feet) yielding $\alpha_L$ = 12.37 meters (40.58 feet). The transverse dispersivity, $\alpha_T$, is assumed to be 10 percent of the longitudinal dispersivity (Reference 2.5-98, page 12); therefore, $\alpha_T$ = 4.06 feet.

The longitudinal ($D_x$) and transverse ($D_y$) coefficients of hydrodynamic dispersion are calculated using the following relationships (Reference 2.5-98, Equations 20 and 21):

$$D_x = \alpha_L U + D_m \quad \text{(Equation 2.5-10)}$$

$$D_y = \alpha_T U + D_m \quad \text{(Equation 2.5-11)}$$


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where $D_m$ is the molecular diffusion coefficient. Considering that the diffusive flux is negligible relative to the dispersive flux, values of $D_x$ = 58.46 square feet per day and $D_y$ = 5.85 square feet per day are estimated.

Retardation Factors

The retardation factor is calculated in accordance with Codell and Duguid (Reference 2.5-95, Equation 4.14), that is,

$$R = 1 + \frac{\rho_b K_d}{n_e} \quad \text{(Equation 2.5-12)}$$

where: $\rho_b$ = bulk density; and $K_d$ = distribution coefficient. The bulk density is estimated from the equation:

$$\rho_b = (1 - n)\rho_s \quad \text{(Equation 2.5-13)}$$

where $\rho_s$ = 2.353 grams per cubic centimeter = the minimum solid phase density observed for all rock samples (Section 2.5.3.1.1).

The $K_d$ values used in this assessment are summarized in Table 2.5-40. Site-specific distribution coefficients for Sr, Y, Ru, Sb, Te, I, Cs, Ce, Pu, and Am were determined by Argonne National Laboratory. The testing was performed using 24 rock samples and site groundwater. The rock samples were obtained from ten borehole locations and analyzed in both millimeter- and centimeter-size fragments. Statistical analyses were performed for the centimeter-size fragments, the millimeter-size fragments, and all fragments. For Te, Ce, Pu and Am, $K_d$ values are presented as “greater than” values because of analytical testing limitations; the “greater than” value has been conservatively adopted for statistical analysis.

Distribution coefficients typically follow a lognormal distribution such that the geometric mean characterizes the central tendency of the data. Although it was anticipated that the $K_d$ values for millimeter-size fragments would be greater than for centimeter-size fragments, with the former having more surface area for adsorption, the geometric means for the two sizes do not show a consistent trend with fragment size. Therefore, the results for the millimeter- and centimeter-size fragments were pooled, and the geometric means calculated for the entire population. These values are included in the last column of Table 2.5-40.

Organic or inorganic complexants can greatly alter the sorption characteristics normally associated with radionuclides, in some cases making them freely mobile in the groundwater system. For the Natrium design, the use of complexants is not anticipated and the $K_d$ values described above are used to characterize radionuclide mobility in groundwater. This will be confirmed when the Liquid Radwaste Processing System water chemistry is better known.


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Release Volume

As described in Section 2.5.3.2.3, a single 4,000-gallon capacity Waste Holdup Tank is assumed to fail with 80 percent of its contents being transferred instantaneously to the groundwater. Therefore, the release volume is taken to be V = 3,200 gallons.

2.5.3.2.3.3 Source Term Analysis

The radionuclides listed in Table 2.5-40 include several with short half-lives that are likely to undergo significant radioactive decay during groundwater transport and prior to exposing a hypothetical receptor at the site boundary. Therefore, a source term analysis is used wherein the parent radionuclides in the source term are assumed to undergo radioactive decay for a period of time equal to the advective travel time from the point of release within the NI to the southern site boundary ($t_a$ = 2,627 days as described in Section 2.5.3.2.3.2). This approach is conservative as it ignores the processes of adsorption and hydrodynamic dispersion that further attenuate and dilute radionuclide concentrations. The calculation methodology and results are described below.

The activity concentrations for the parent, first progeny, and second progeny radionuclides are calculated using the Bateman equations (Reference 2.5-99, Equations B.41, B.43, and B.54, respectively):

$$C_1 = C_{10} exp(-\lambda_1 t) \quad \text{(Equation 2.5-14)}$$

$$C_2 = K_{21} exp(-\lambda_1 t) + K_{22} exp(-\lambda_2 t) \quad \text{(Equation 2.5-15)}$$

$$C_3 = K_{31} exp(-\lambda_1 t) + K_{32} exp(-\lambda_2 t) + K_{33} exp(-\lambda_3 t) \quad \text{(Equation 2.5-16)}$$

where: $C_1$, $C_2$, $C_3$ = activity concentrations of the parent, first progeny and second progeny radionuclides, respectively; $C_{10}$, $C_{20}$, $C_{30}$ = initial activity concentrations of the parent, first progeny and second progeny radionuclides, respectively; and $\lambda_1$, $\lambda_2$, $\lambda_3$ = radioactive decay constants for the parent, first progeny and second progeny radionuclides, respectively. For Equation 2.5-15, $K_{21}$ and $K_{22}$ are defined as:

$$K_{21} = \frac{d_{12}\lambda_2 C_{10}}{\lambda_2 - \lambda_1} \quad \text{(Equation 2.5-17)}$$

$$K_{22} = C_{20} - \frac{d_{12}\lambda_2 C_{10}}{\lambda_2 - \lambda_1} \quad \text{(Equation 2.5-18)}$$


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For Equation 2.5-16, $K_{31}$, $K_{32}$ and $K_{33}$ are calculated from:

$$K_{31} = \frac{d_{13}\lambda_3 C_{10}}{\lambda_3 - \lambda_1} + \frac{d_{23}\lambda_2 d_{12}\lambda_3 C_{10}}{(\lambda_3 - \lambda_1)(\lambda_2 - \lambda_1)} \quad \text{(Equation 2.5-19)}$$

$$K_{32} = \frac{d_{23}\lambda_3 C_{20}}{\lambda_3 - \lambda_2} + \frac{d_{23}\lambda_2 d_{12}\lambda_3 C_{10}}{(\lambda_3 - \lambda_2)(\lambda_2 - \lambda_1)} \quad \text{(Equation 2.5-20)}$$

$$K_{33} = C_{30} - \frac{d_{13}\lambda_3 C_{10}}{\lambda_3 - \lambda_1} - \frac{d_{23}\lambda_3 C_{20}}{\lambda_3 - \lambda_2} + \frac{d_{23}\lambda_2 d_{12}\lambda_3 C_{10}}{(\lambda_3 - \lambda_1)(\lambda_3 - \lambda_2)} \quad \text{(Equation 2.5-21)}$$

In the above equations, $d_{12}$ = fraction of parent radionuclide transitions that result in the production of first progeny radionuclide; $d_{13}$ = fraction of parent radionuclide transitions that result in production of second progeny radionuclide; and $d_{23}$ = fraction of first progeny radionuclide transitions that result in production of second progeny radionuclide. Values for $d_{12}$, $d_{13}$, and $d_{23}$ are obtained from Reference 2.5-94.

The equations described above are used to estimate activity concentrations for the radionuclides listed in Table 2.5-40 for $t_a$ = 2,627 days, the advective groundwater travel time from the point of release within the NI to the southern site boundary. A qualitative summary of the results is provided below.

- Based on the expected advective travel time, short-lived radionuclides undergo significant radioactive decay. They were eliminated from consideration if they decayed to less than 1E-06 of their effluent concentration limit (ECL). As noted above, eliminating radionuclides on this basis is conservative because the mechanisms of adsorption and hydrodynamic dispersion, which attenuate concentrations, are not considered. The source term analysis eliminated Te-129m and Te-129 from consideration.
- Several progeny radionuclides will be in secular equilibrium with their respective parents (i.e., those progeny with short half-lives relative to their parent) and have the same activity concentration as their parent. These include the following: Y-90 in equilibrium with Sr-90, Rh-106 in equilibrium with Ru-106, Te-125m in equilibrium with Sb-125, Ba-137m in equilibrium with Cs-137, and Pr-144m and Pr-144 in equilibrium with Ce-144.
- I-129 will not be in secular equilibrium with parents Te-129m and Te-129. The parents decay rapidly, while I-129 undergoes minor in-growth and very slow decay due to its long half-life. Similarly, Am-241 will not be in secular equilibrium with its Pu-241 parent. As Pu-241 decays there will be in-growth of Am-241.

2.5.3.2.3.4 Exposure Concentrations

Activity concentrations of the radionuclides of interest are calculated using the methodology, parameters, and results described above. Exposure concentrations were calculated in one of four ways:


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1. For radionuclides eliminated in the source term analysis (see Section 2.5.3.2.3.3), exposure concentrations are taken to be zero.

2. Exposure concentrations for the remaining parent radionuclides are calculated using Equation 2.5-4 using the parameters defined in Section 2.5.3.2.3.2 and their Waste Holdup Tank activity concentrations given in Table 2.5-40. The maximum value, $C_{max}$, and the time at which the maximum value occurs, $t_{max}$, are obtained by trial and error.

3. For progeny radionuclides in secular equilibrium with their respective parents, exposure concentrations are be taken to be the same as their parent, corrected by $d_{12}$, $d_{13}$, or $d_{23}$ as applicable.

4. For progeny radionuclides not in secular equilibrium with their parent, exposure concentrations are estimated by adopting a conservative value for $C_0$ in Equation 2.5-4. For I-129, $C_0$ is assigned a value of 1.40E-10 microcurie per cubic centimeter, which is the I-129 activity concentration specified for the Waste Holdup Tank. For Am-241, $C_0$ is estimated by applying Equation 2.5-15 with t = 50 years, which results in value of 1.67E-10 microcurie per cubic centimeter.

The results summarized in Table 2.5-41 conservatively represent the activity concentrations in water pumped from a well located on the southern site boundary of the Kemmerer Unit 1 site that could expose a member of the public (see Section 2.5.3.2.4 for additional discussion of the exposure scenario). The results in Table 2.5-41 demonstrate compliance with the “unity rule” of 10 CFR Part 20 Appendix B, which states that the sum of the $C_{max}$/ECL ratios should be less than 1 (the calculated sum is 4.68E-09). The values in Table 2.5-41 serve as input to the dose calculation described in Section 2.5.3.2.5.

### 2.5.3.2.4 Exposure Scenarios

The following exposure scenarios are evaluated:

A. Direct water use - This scenario assumes that members of the public would consume drinking water withdrawn near or at the point of entry in an unrestricted area. The sources of water include a groundwater well, or from a surface water body or river.

B. Indirect water use - This scenario assumes the use of water in indirect human consumption. Such scenarios may include livestock watering or irrigation of grazing pastures, consumption of animal products (meat and milk products), fish and invertebrate consumption, crop irrigation and consumption of such crops, or water used as an ingredient in food products or in food processing.

C. Combined use of water - This scenario assumes both direct and indirect water use. In such an instance, the scenarios would be modeled separately, and the resulting doses would be summed and compared to the acceptance criteria.


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For the Kemmerer Unit 1 site, the point of entry in an unrestricted area is assumed to be a hypothetical domestic well located on the southern site boundary and used by a member of the public. This exposure scenario is considered conservative because the hypothetical well would be located in what is considered a major aquitard that is a poor producer of groundwater with poor groundwater quality (Sections 2.5.3.1.1 and 2.5.3.1.5). Because of these hydrogeological and geochemical characteristics, it is unlikely that a member of the public would, in the future, install a water-supply well in this formation and use the water for domestic or agricultural purposes.

#### 2.5.3.2.5 Dose

The radionuclide activity concentrations presented in Table 2.5-41 are converted to an annual activity using an arbitrary flow rate of 1 cubic foot per second. The resultant activity is input to LADTAP. As Branch Technical Position (BTP) 11-6 does not specify a time for the dose limit to be attained within, one year is assumed, consistent with 10 CFR 20.1301. BTP 11-6 also does not specify if the dose limit is in total effective dose equivalent (TEDE). As TEDE is more conservative, it is assumed that the 100 millirem dose limit is 100 millirem per year TEDE. Thus, the resultant meat, leafy vegetable, vegetable, milk, and drinking water doses from LADTAP are converted to TEDE doses utilizing the maximum ratio of the Federal Guidance Report dose conversion factor to the LADTAP dose conversion factor for all isotopes considered. The ingestion doses are summed and it is found that the maximum exposed individual is an adult receiving a TEDE dose of 5.60E-03 millirem per year. This dose is within the 10 CFR 20.1301 limit of 100 millirem.

### 2.5.4 Low Water Considerations

Kemmerer Unit 1 is liquid sodium cooled and does not require water for SR cooling. This section investigates natural events that may reduce or limit the available raw water supply for the plant, ensuring that NST cooling water needs are met.

#### 2.5.4.1 Low Water in Rivers and Streams

The NST raw water needs are supplied by the Naughton Raw Water Pond, which also supplies water to the Naughton Power Plant Units 1, 2, and 3. The Naughton Raw Water Pond receives water through two 7-mile long pipelines from the Naughton CWIS at Hams Fork. Hams Fork is fed by the Viva Naughton Reservoir through an outlet structure located at the dam. The Naughton CWIS was originally constructed to provide raw water for Naughton Units 1 and 2, with a second intake bay later added to the structure for Unit 3 when additional raw water was required. As part of the Naughton CWIS original design, there are two radial gates spanning across the river width, helping divert the flow toward the intake and maintain sufficient water level for intake operation. Based on information provided by Reference 2.5-102, the low water level at the Naughton CWIS is at an elevation of 6,969.5 ft. This water level provides adequate submergence to the pumps at the Naughton CWIS to operate under their design flow and supply water to the Naughton Raw Water Pond.


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### 2.5.4.2 Low Water from Surges, Seiches, Tsunamis, or Ice Effects

The SR systems do not rely on surface water sources and are not affected by low water due to surge, seiche, or tsunami. The effects of ice on water supply are discussed in Section 2.5.1.6.

### 2.5.4.3 Historical Low Water

The historical water level measurements at Viva Naughton Reservoir are used to estimate the 100-year low water level and provide an evaluation for water availability for Kemmerer Unit 1. The historical water level measurements at Viva Naughton Reservoir are maintained by the Naughton Power Plant and are provided as end of the month measurements from December 1960 to July 2022. A frequency analysis is conducted on reservoir level measurements from February 1982 to July 2022. Prior to 1982, there is a considerable amount of missing data in the record. A total of six probability distribution functions, typically applied in low flow analyses, are tested, and Chi-square ($\chi^2$) and Kolmogorov-Smirnov (K-S) tests are used to measure their goodness of fit. Pearson Type 3 distribution provides the most satisfactory fit to the observed reservoir level measurements. However, as seen in Figure 2.5-118, the Pearson Type 3 distribution overestimates the low water levels for return periods higher than 10 years. A best-fit line is fitted across the data and the estimated 100-year low water level at Viva Naughton Reservoir is 7,200.0 ft (Figure 2.5-118), which corresponds to a reservoir volume of 6,800 ac-ft.

After taking into consideration the required submergence at the outlet pipe of the Viva Naughton Reservoir, the water demands for the Naughton Power Plant and the Kemmerer Unit 1 site (68.5 ac-ft/day, Section 2.5.4.5), and the future water demand for the City of Kemmerer (an additional 14.1 ac-ft/day), it is estimated that there is water available at the Viva Naughton Reservoir for about 54 days under 100-year low water level at the reservoir and with no inflow.

### 2.5.4.4 Future Controls

The SR systems do not rely on surface water sources and are not affected by drought conditions. The cooling tower system is only for normal cooling, and it is an NST system. No future controls are necessary to mitigate low water conditions.

### 2.5.4.5 Plant Requirements

The raw water transfer pumps that deliver water from the Naughton Raw Water Pond to Kemmerer Unit 1 are designed to have a rated capacity of 5,270 gpm, which is for NST functions. The Naughton CWIS at Hams Fork has a total design intake flow of 19,500 gpm (Reference 2.5-102). Units 1 and 2 intake bay uses three intake pumps with a rated design capacity of 2,500 gpm each, and a total design intake flow of 7,500 gpm. Unit 3 intake bay uses two intake pumps with a rated design flow of 6,000 gpm each, and a total design intake flow of 12,000 gpm.

Reference 2.5-102 provides information with regard to the average monthly withdrawal at the Naughton CWIS at Hams Fork for Units 1, 2, and 3 and the average monthly river flow at USGS Station 09223385, Hams Fork below Viva Naughton Reservoir. The average monthly withdrawals were estimated using available intake flow measurements from June 2016 to June 2017 (Reference 2.5-102), which range from a minimum value of 11.7 cfs to a maximum of 20.8 cfs


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with average of 15.2 cfs. The ratio of maximum to average withdrawal flow is about 1.4, and a conservative value of 1.5 is selected in assessing the ability of the Viva Naughton Reservoir to meet the plant water demand during a 100-year low water condition in Section 2.5.4.3.

According to Reference 2.5-102, the average actual intake flow for the Naughton Power Plant from June 2016 to June 2017 is 3,230 gpm for Units 1 and 2 and 3,594 gpm for Unit 3. The maximum actual intake flow (on a monthly basis) is used to represent the water demand of the Naughton Power Plant and is approximated by applying a factor 1.5 to the average actual intake flow, which results in 4,845 gpm for Units 1 and 2, and 5,391 gpm for Unit 3. Table 2.5-42 summarizes the water demands from Kemmerer Unit 1 and the Naughton Power Plant Units 1, 2 and 3. A comparison of the average monthly historical river discharge and the total water demand of Naughton Power Plant and Kemmerer Unit 1 is shown in Table 2.5-43.The intake structure of the City of Kemmerer is located downstream of the Naughton CWIS. Its withdrawal flow rate is not included in this comparison because it will not affect the assessment of availability of Hams Fork river discharge to meet the expected combined intake flow of Kemmerer Unit 1 and Naughton Power Plant. As the discharge in Hams Fork is controlled by flow releases from Viva Naughton Reservoir, the source water, the intake flow of the City of Kemmerer is included in the assessment of source water supply availability during a 100-year drought as described in Section 2.5.4.3.

The average monthly flow at Hams Fork is estimated in two additional ways as summarized in Table 2.5-44:

- Daily USGS data (2007 to 2016) for USGS station 09223385, Hams Fork below Viva Naughton Reservoir (Reference 2.5-103).
- Daily outflow record (1995 to 2021) from the Viva Naughton Reservoir provided by Naughton Power Plant.

Based on Table 2.5-43 and Table 2.5-44, there is sufficient flow in Hams Fork to meet the water demands of Naughton Units 1, 2, and 3 and Kemmerer Unit 1, assuming that the future water demands of Naughton Units 1, 2, and 3 will not change. The most recent Integrated Resource Plan from PacifiCorp (Reference 2.5-101), indicates that Naughton Units 1, 2, and 3 will be retired at the end of 2036. Availability of water will further improve for Kemmerer Unit 1 after 2036.

The Hams Fork Water Users Association (HFWUA) and PacifiCorp have a water use agreement since the early 2000’s, relating to the storage and release of water in Viva Naughton Reservoir (Reference 2.5-104). Based on Reference 2.5-105, the water use agreement between HFWUA and PacifiCorp was still in effect in 2021. The water use agreement, as described in Reference 2.5-104, was approved on April 5, 2013. It allows HFWUA to use up to 80 cfs of direct flows in Hams Fork released from Viva Naughton Reservoir commencing on May 1 and continuing through May 14 of each year. From May 15 through the end of runoff of each calendar year, PacifiCorp would release up to 141 cfs from Viva Naughton Reservoir for use by HFWUA. End of runoff is typically July 1$^{st}$ (Reference 2.5-104). HFWUA might request additional releases from the reservoir, but any additional releases would come from an irrigation pool of 6,000 ac-ft. PacifiCorp holds the 6,000 ac-ft irrigation pool for the benefit of HFWUA which would be available from May 1 through September 30 of each year. There is no carryover of the irrigation


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pool from year to year. Furthermore, the agreement called for HFWUA to waive its rights to demand or otherwise call for any additional flows from Hams Fork or discharges from the Viva Naughton Reservoir over and above the minimum flows or irrigation pool. As seen from Table 2.5-43 and Table 2.5-44, there would be adequate flow at Hams Fork in the months of May and June to satisfy the water use agreement with HFWUA without any disruption to the supply of water to Kemmerer Unit 1. In addition, based on Reference 2.5-106, HFWUA purchased 638 ac-ft of water in 2018 and 254 ac-ft of water in 2019 from PacifiCorp, which are small quantities compared to the overall capacity of the Viva Naughton Reservoir, and would not significantly impact the amount of water available to Kemmerer Unit 1.

Future outlook of water usage and water availability is examined. Based on currently available information, no other future users are expected to consume water from the Viva Naughton Reservoir. In addition, the IPCC report for 2021 (Reference 2.5-107) indicates that the annual average precipitation in the region of Kemmerer Unit 1 is expected to increase in the future due to climate change. This would likely improve the average long-term flow discharge in Hams Fork. For instance, Figure 4.32 of Reference 2.5-107, indicates a projected increase in the annual average precipitation on the order of 10 percent for North America depending on the projected temperature increase. A similar pattern is predicted for the long-term change of seasonal mean precipitation (Figure 4.24 of Reference 2.5-107), with a projected increase on the order of 10 percent for the region where Kemmerer Unit 1 is located.

### 2.5.4.6 Heat Sink Dependability Requirements

Kemmerer Unit 1 does not rely on surface water to support SR functions and there are no heat sink dependability requirements.

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**Table 2.5-1 Maximum LIP Water Levels Surrounding At-Grade NSRST Structures**

| Structure | FHB | NCB | RXB | RAB |
|---|---|---|---|---|
| Maximum WSE, ft NAVD 88 | 6,757.7 | 6,757.6 | 6,757.8 | 6,757.6 |



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**Table 2.5-2 Sub-Basin Drainage Areas**

| Sub-Basin | Drainage Area (ac) | Drainage Area ($mi^2$) |
|---|---|---|
| East | 11,755 | 18.4 |
| West | 4,698 | 7.4 |
| Laydown | 125 | 0.2 |
| Confluence | 727 | 1.2 |


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**Table 2.5-3 Composite Curve Numbers for Each Sub-Basin**

| Sub-Basin | Composite Runoff Curve Number |
|---|---|
| East | 85.2 |
| West | 82.6 |
| Laydown | 90.6 |
| Confluence | 85.8 |



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**Table 2.5-4 Lag Times for Each Sub-Basin**

| **Sub-Basin** | **Channel Length (ft)** | **Weighted Average Channel Slope** (%) | **Lag Time** (min) | **Reduced Lag Time** (min) |
|---|---|---|---|---|
| West | 24,570 | 1.83 | 76 | 57 |
| East | 45,190 | 0.77 | 147 | 110 |
| Laydown | 5,220 | 0.71 | 38 | 28 |
| Confluence | 9,040 | 1.16 | 46 | 34 |



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**Table 2.5-5 Probable Maximum Precipitation Peak Discharges at the Kemmerer Unit 1 Site**

| Hydrologic Element | Drainage Area | Peak Discharge * | | Time of Peak ** |
|---|---|---|---|---|
| | | 40%PMP+PMP | 40%PMP+PMP+CC | |
| | ($mi^2$) | (cfs) | (cfs) | (hr) |
| Sub-Basin East | 18.4 | 35,570 | 43,110 | 23:00 |
| Sub-Basin West | 7.4 | 22,520 | 27,370 | 22:15 |
| Sub-basin Laydown | 0.2 | 970 | 1,170 | 21:30 |
| Sub-Basin Confluence | 1.2 | 5,210 | 6,310 | 21:45 |
| Junction Outlet | 27.2 | 53,310 | 64,700 | 22:30 |

Notes

* The peak discharge values are rounded to the nearest 10 cfs

** The times of peak listed are hours of the fourth day of the simulation. The 6-hour PMP storm starts at 18:00 of the 4th day of the simulation.


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**Table 2.5-6 Maximum Water Levels during the Design Probable Maximum Precipitation Event (including a 20 percent climate change adjustment factor) (Sheet 1 of 2)**

| Reach | Cross Section | Peak Discharge | Water Level | Flow Velocity | | | Location Reference |
|---|---|---|---|---|---|---|---|
| | | | | Channel | Left Bank | Right Bank | |
| | | (cfs) | (ft) | (ft/s) | (ft/s) | (ft/s) | |
| Laydown | 7 | 1,170 | 6,761.6 | 9.2 | No overbank flow | | |
| | 6 | | 6,751.8 | 5.0 | | | |
| | 5 | | 6,748.0 | 0.8 | | | Close to TFF |
| | 4 | | 6,747.4 | 5.3 | 2.4 | 0.7 | Next to NI |
| | 3 | | 6,746.0 | 5.0 | No overbank flow | | Next to NI |
| | 2 | | 6,745.3 | 3.8 | 0.4 | No flow | Next to EI |
| | 1 | | 6,745.1 | 0.3 | 0.2 | 0.3 | Next to EI |
| | 0 | | 6,745.1 | 0.1 | 0.1 | 0.1 | |
| East Lower | 22.3 | 43,110 | 6,789.8 | 4.6 | 4.3 | 1.9 | |
| | 22 | | 6,789.5 | 6.3 | 5.5 | 4.2 | |
| | 21 | | 6,782.2 | 8.5 | 3.8 | 4.9 | |
| | 20 | | 6,776.7 | 14.1 | 6.1 | 5.3 | |
| | 19 | | 6,772.8 | 14.0 | 6.1 | 5.3 | |
| | 17 | | 6,765.3 | 8.3 | 4.3 | 4.5 | |
| | 16 | | 6,762.8 | 7.8 | 4.1 | 3.9 | |
| | 15 | | 6,761.6 | 8.0 | 4.1 | 4.1 | |
| | 14 | | 6,760.7 | 8.6 | 5.0 | 4.9 | |
| | 13 | | 6,758.0 | 10.9 | 6.1 | 5.6 | |
| | 12 | | 6,755.0 | 6.8 | 3.9 | 3.3 | |
| | 11 | | 6,754.0 | 9.1 | 2.9 | 4.5 | |
| | 10 | | 6,753.0 | 9.1 | 3.7 | 4.1 | Next to TFF |
| | 9 | | 6,752.3 | 8.5 | 4.1 | 5.0 | Next to TFF |
| | 8 | | 6,751.6 | 8.6 | 3.0 | 4.4 | Next to NI |
| | 7 | | 6,750.5 | 7.9 | 3.4 | 4.1 | Next to NI |
| | 6 | | 6,749.4 | 9.1 | 5.0 | 4.9 | Next to NI |
| | 5 | | 6,748.3 | 8.6 | 4.8 | 4.5 | Next to EI |
| | 4 | | 6,747.3 | 8.3 | 4.7 | 4.4 | Next to EI |
| | 3 | | 6,746.0 | 8.3 | 3.6 | 4.6 | |
| | 2 | | 6,745.5 | 7.2 | 4.1 | 4.3 | |
| | 1 | | 6,744.5 | 7.8 | 4.4 | 3.5 | |


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**Table 2.5-6 Maximum Water Levels during the Design Probable Maximum Precipitation Event (including a 20 percent climate change adjustment factor) (Sheet 2 of 2)**

| Reach | Cross Section | Peak Discharge | Water Level | Flow Velocity | | | Location Reference |
|---|---|---|---|---|---|---|---|
| | | | | Channel | Left Bank | Right Bank | |
| | | (cfs) | (ft) | (ft/s) | (ft/s) | (ft/s) | |
| West | 16 | 27,370 | 6,746.9 | 1.9 | 1.5 | 1.5 | |
| | 15 | | 6,746.9 | 1.7 | 1.3 | 1.3 | |
| | 14.5 | | Inline Structure | | | | Railroad |
| | 14 | | 6,746.8 | 1.8 | 1.3 | 1.4 | |
| | 13 | | 6,746.8 | 1.7 | 1.2 | 1.3 | |
| | 12 | | 6,746.7 | 1.6 | 1.2 | 1.3 | |
| | 11.5 | | Inline Structure | | | | Skull Road |
| | 11 | | 6,746.5 | 1.7 | 1.2 | 1.3 | |
| | 10 | | 6,746.4 | 2.0 | 1.4 | 1.5 | |
| | 9 | | 6,746.4 | 2.5 | 1.6 | 1.7 | |
| | 8 | | 6,746.3 | 2.6 | 1.4 | 2.1 | |
| | 7.5 | | Inline Structure | | | | US Highway 189 |
| | 7 | | 6,745.1 | 3.3 | 0.0 | 2.6 | |
| | 6 | | 6,745.1 | 3.0 | 2.3 | 2.3 | |
| | 5 | | 6,745.0 | 2.7 | 2.0 | 2.1 | |
| West Lower | 4 | 27,900 | 6,745.0 | 3.5 | 2.1 | 2.7 | |
| | 3.5 | | 6,744.9 | 3.7 | 2.5 | 2.8 | |
| | 3.25 | | Inline Structure | | | | |
| | 3 | | 6,744.6 | 3.8 | 2.6 | 2.9 | Site Access Road |
| | 2.5 | | 6,744.5 | 4.1 | 2.8 | 3.1 | |
| | 2 | | 6,744.5 | 4.0 | 2.5 | 2.7 | |
| | 1 | | 6,744.4 | 3.1 | 2.0 | 2.0 | |
| | 0.2 | | 6,744.2 | 3.3 | 2.4 | 1.6 | |
| | 0.1 | | 6,744.2 | 3.2 | 2.4 | 2.1 | |
| Confluence | 7 | 64,700 | 6,743.9 | 4.0 | 2.3 | 2.1 | |
| | 6 | | 6,743.7 | 4.7 | 2.7 | 1.6 | |
| | 5 | | 6,743.6 | 3.8 | 1.3 | 1.9 | |
| | 4 | | 6,743.4 | 3.7 | 2.0 | 2.1 | |


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**Table 2.5-7 Pond Drainage Areas**

| Pond | Drainage Area (acres) |
|---|---|
| FGD 5 | 52.3 |
| FGD 4 | 38.6 |
| Raw Water | 107[1] |
| Lake Arambel | 765 |

[1] Included in the Lake Arambel drainage area.


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**Table 2.5-8 Dam Embankment Data**

| Pond | Crest Elevation, ft (NAVD 88) | Max Operating Level, ft (NAVD 88) | Bottom of Pond Elevation, ft (NAVD 88) | Crest Width, ft |
|---|---|---|---|---|
| FGD 5 | 6,865.06 | 6,862.06 | 6812.53 | 25 |
| FGD 4 | 6,874.56 | 6,869.56 | 6818.56 | 25 |
| Lake Arambel | 6,895.06 | 6,890.56 | 6844.06 | 25 |



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**Table 2.5-9 500-year Point Precipitation Depth**

| Duration | 500-year Precipitation Depth, in | |
|---|---|---|
| | From NOAA Atlas 14 | Increased by 20% for Climate Change |
| 5-min | 0.748 | 0.898 |
| 15-min | 1.41 | 1.69 |
| 1-hour | 2.35 | 2.82 |
| 2-hours | 2.68 | 3.22 |
| 3-hours | 2.71 | 3.25 |
| 6-hours | 2.75 | 3.30 |
| 12-hours | 2.88 | 3.46 |
| 1-day | 3.29 | 3.95 |


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**Table 2.5-10 HEC-HMS 500-year Peak Discharges**

| Hydrologic Element | Drainage Area | Time Lag | CN | Peak Discharge |
|---|---|---|---|---|
| | $mi^2$ | min | | cfs |
| Sub-basin East | 18.4 | 110 | 85.2 | 9,310 |
| Sub-basin West | 7.4 | 57 | 82.6 | 5,670 |
| Sub-basin Laydown | 0.2 | 28 | 90.6 | 320 |
| Junction West | 7.6 | - | - | 5,830 |
| Sub-basin Confluence | 1.2 | 34 | 85.8 | 1,500 |
| Junction Outlet | 27.2 | - | - | 13,580 |



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**Table 2.5-11 Input for Breach Parameters**

| Pond | Top of Dam Elevation* (ft) | Breach Bottom Elevation* (ft) | Pool Elevation* at Failure (ft) | Pool Volume at Failure (ac-ft) | Crest Width (ft) | U/S Face Slope (H:1V) | D/S Face Slope (H:1V) | Dam Erodibility |
|---|---|---|---|---|---|---|---|---|
| FGD 4 | 6,874.56 | 6,819.56 | 6,869.73 | 1,374.7 | 25 | 3 | 3 | Medium** |
| FGD 5 | 6,865.06 | 6,816.06 | 6,862.26 | 1,490.08 | 25 | 3 | 3 | Medium*** |
| Lake Arambel | 6,895.56 | 6,845.06 | 6,891.53 | 1,453.55 | 25 | 3 | 3 | Medium**** |

* Elevation in NAVD 88

** The FGD 4 dam is an earth embankment (Reference 2.5-30)

*** The FGD 5 dam is an earth embankment constructed using sandy lean clay and claystone excavated from within the pond limits and an adjacent borrow area (Reference 2.5-29)

**** The Lake Arambel dam is an earth embankment (Reference 2.5-31)


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**Table 2.5-12 Selected HEC-RAS Breach Parameters**

| Breach Parameter | FGD 4 | FGD 5 | Lake Arambel | |
|---|---|---|---|---|
| **Failure Mode** | **Piping** | **Piping** | **Piping** | **Overtopping** |
| Final Bottom Width, ft | 59 | 66 | 69 | 87 |
| Specified Final Bottom Elevation, ft NAVD 88 | 6,819.56 | 6,816.06 | 6,845.06 | 6,845.06 |
| Breach Side Slopes, horizontal:vertical | 0.7 | 0.7 | 0.7 | 1.0 |
| Breach Weir Coefficient | 2.6 | 2.6 | 2.6 | 2.6 |
| Breach Formation Time, hrs | 0.44 | 0.51 | 0.53 | 0.54 |
| Piping Coefficient | 0.5 | 0.5 | 0.5 | - |
| Initial Piping Elevation, ft NAVD 88 | 6,829.56 | 6,824.56 | 6,854.56 | - |
| Failure Start Time, hr | 1:00 | 1:00 | 1:00 | - |
| Starting Water Surface Elevation, ft NAVD 88 | - | - | - | 6,895.56 |


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**Table 2.5-13 Seismic Dam Breach + 500-year Flood HEC-RAS Results**

| Reach | HEC-RAS Cross Section | Peak Discharge | Max WS El | Velocity | | | Comment |
|---|---|---|---|---|---|---|---|
| | | | | Channel | Left Bank | Right Bank | |
| | | (cfs) | (ft) | (ft/s) | (ft/s) | (ft/s) | |
| East Lower | 14 | 78,344 | 6,763.2 | 10.5 | 6.7 | 6.6 | |
| | 13 | 77,935 | 6,760.5 | 13.1 | 7.8 | 7.2 | |
| | 12 | 76,290 | 6,757.3 | 8.8 | 5.1 | 4.8 | |
| | 11 | 75,626 | 6,756.5 | 10.3 | 4.4 | 5.6 | |
| | 10 | 74,895 | 6,755.4 | 10.8 | 4.8 | 5.3 | Next to the TFF |
| | 9 | 74,148 | 6,754.7 | 9.3 | 4.7 | 6.0 | Next to the TFF |
| | 8 | 73,461 | 6,754.1 | 9.4 | 3.5 | 5.3 | Next to NI |
| | 7 | 73,374 | 6,752.9 | 9.5 | 4.7 | 5.3 | Next to NI |
| | 6 | 72,852 | 6,751.7 | 10.9 | 6.2 | 6.3 | Next to NI |
| | 5 | 72,378 | 6,750.6 | 10.5 | 6.0 | 5.9 | Next to EI |
| | 4 | 72,291 | 6,749.4 | 10.5 | 6.1 | 6.0 | Next to EI |
| | 3 | 71,903 | 6,747.6 | 11.1 | 5.0 | 6.5 | |
| | 2 | 71,528 | 6,746.6 | 10.3 | 5.9 | 6.2 | |
| | 1 | 66,178 | 6,744.9 | 11.0 | 6.4 | 5.0 | |


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**Table 2.5-14 Lake Arambel Lag Time**

| Sub-Basin | Channel Length | Average Channel Slope | Lag Time | Reduced Lag Time | Reduced Lag Time |
|---|---|---|---|---|---|
| | ft | % | hr | hr | min |
| Lake Arambel | 8,340 | 1.26 | 0.71 | 0.53 | 32 |


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**Table 2.5-15 PMF Peak Discharges**

| Hydrologic Element | Drainage Area | Time Lag | Peak Discharge | Time* of Peak |
|---|---|---|---|---|
| | $mi^2$ | min | cfs | hr:min |
| Sub-basin East | 17.2 | 110 | 40,330 | 5:00 |
| Lake Arambel | 1.2 | 32 | 6,460 | 3:45 |
| East Section 21 | 18.4 | - | 41,400 | 5:00 |
| Sub-basin West | 7.4 | 57 | 27,370 | 4:15 |
| Sub-basin Laydown | 0.2 | 28 | 1,170 | 3:45 |
| Sub-basin Confluence | 1.2 | 34 | 6,310 | 3:45 |
| Junction Outlet | 27.2 | - | 64,630 | 4:30 |

* Time is from the start of the 6-hour PMP


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**Table 2.5-16 Hydrologic failure of the Lake Arambel dam during PMP HEC-RAS Results**

| Reach | HEC-RAS Cross Section | Q | Max WS El | Velocity | | | Comment |
|---|---|---|---|---|---|---|---|
| | | | | Channel | Left Bank | Right Bank | |
| | | (cfs) | (ft) | (ft/s) | (ft/s) | (ft/s) | |
| East Lower | 14 | 78,818 | 6,763.3 | 10.5 | 6.7 | 6.6 | |
| | 13 | 78,564 | 6,760.6 | 13.0 | 7.8 | 7.2 | |
| | 12 | 77,731 | 6,757.4 | 8.8 | 5.1 | 4.8 | |
| | 11 | 77,328 | 6,756.6 | 10.3 | 4.4 | 5.6 | |
| | 10 | 76,896 | 6,755.6 | 10.9 | 4.9 | 5.2 | Next to TFF |
| | 9 | 76,662 | 6,754.9 | 9.3 | 4.7 | 6.0 | Next to the TFF |
| | 8 | 76,448 | 6,754.4 | 9.3 | 3.5 | 5.3 | Next to NI |
| | 7 | 76,192 | 6,753.2 | 9.6 | 4.5 | 5.4 | Next to NI |
| | 6 | 75,978 | 6,752.1 | 11.0 | 6.2 | 6.3 | Next to NI |
| | 5 | 75,838 | 6,751.0 | 10.6 | 6.1 | 6.0 | Next to NI |
| | 4 | 75,713 | 6,749.8 | 10.5 | 6.1 | 6.0 | Next to EI |
| | 3 | 75,426 | 6,748.1 | 11.0 | 5.0 | 6.5 | Next to EI |
| | 2 | 75,051 | 6,747.2 | 10.1 | 5.8 | 6.1 | |
| | 1 | 73,443 | 6,746.0 | 10.3 | 6.0 | 4.9 | |


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**Table 2.5-17 Maximum Capacity of FGD 4, FGD 5, Lake Arambel, and Raw Water Settling Basin**

| POND | MAXIMUM CAPACITY[1] (acre-ft) |
|---|---|
| FGD 4[2] | 1,367 |
| FGD 5[3] | 1,481 |
| Lake Arambel[4] | 1,391 |
| Raw Water Settling Basin[5] | 76 |

Note 1: Capacity at the maximum operating water level

Note 2: Reference 2.5-46

Note 3: Reference 2.5-47

Note 4: Reference 2.5-48

Note 5: Reference 2.5-49


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**Table 2.5-18 Historical Ice Jams Events at Hams Fork River**
**(Sheet 1 of 2)**

| Index Number in Ice Jam Database | Location | Date | Notes from Ice Jam Database |
|---|---|---|---|
| 20170321163035 | Hams Fork River near Granger, WY | 03/15/2017 | The ice jam would likely have broken and caused minor flooding downstream from the confluence of the Hams Fork River and Blacks Fork River in Granger, to Little America along the Blacks Fork River. |
| 20131210132250 | Hams Fork River near Frontier, WY | 03/29/2012 | USGS Water Resources Data for Wyoming WY 2012 reported an annual maximum peak stage of 4.93 ft on 03/29/2012 due to backwater from ice at USGS gaging station 09223000 Frontier, WY. The average daily discharge was estimated at 60 cfs. |
| 20120810162210 | Hams Fork River near Frontier, WY | 04/13/2010 | USGS Water Resources Data for Wyoming WY 2010 reported an annual maximum peak stage of 4.82 ft on 04/13/2010 due to backwater from ice at USGS gaging station 09223000 Frontier, WY. The average daily discharge was 45 cfs. |
| 20090630074203 | Hams Fork River near Frontier, WY | 03/14/2007 | USGS Water Resources Data for Wyoming WY 2007 reported an annual maximum peak stage of 4.16 ft on 03/14/2007 due to backwater from ice at USGS gaging station 09223000 Hams Fork River below Pole Creek, near Frontier, WY. The average daily discharge was 39 cfs. |
| 20090309084813 | Hams Fork River near Frontier, WY | 12/31/2004 | USGS Water Resources Data for Wyoming WY 2005 reported an annual maximum peak stage of 5.81 ft on 12/31/2004 due to backwater from ice at USGS gaging station Hams Fork River below Pole Creek, near Frontier, WY. The average daily discharge was 10 cfs. |


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**Table 2.5-18 Historical Ice Jams Events at Hams Fork River**
**(Sheet 2 of 2)**

| Index Number in Ice Jam Database | Location | Date | Notes from Ice Jam Database |
|---|---|---|---|
| 20071227113950 | Hams Fork River near Frontier, WY | 12/31/2004 | USGS Water Resources Data for Wyoming WY 2005 reported an annual maximum peak stage of 5.81 ft due to backwater from ice on 12/31/2004 at USGS gaging station 09223000 Hams Fork River below Pole Creek, near Frontier, WY. The discharge was estimated to be 10 cfs. |
| 11351 | Blacks and Hams Fork River near Granger, WY | 03/15/1997 | An ice jam along the Black's Fork in Western Sweetwater County has affected the Black's Fork/I-80 Underpass in Uinta County. Additional runoff or break-up of this dam will raise water levels near Granger and points further downstream. |
| 11350 | Hams Fork River near Frontier, WY | 04/03/1959 | Maximum annual gage height of 4.53 ft, affected by backwater from ice, reported at USGS gage Hams Fork River near Frontier, on 04/03/1959. Bankfull stage 4 ft. |


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**Table 2.5-19 Historical Ice Jams Events at Green River**

| Index Number in Ice Jam Database | Location | Date | Notes from Ice Jam Database |
|---|---|---|---|
| 20181218144621 | Jamestown, WY | 12/12/2018 | N/A |
| 20131210132630 | Green River, WY | 12/4/2011 | USGS Water Resources Data for Wyoming WY 2012 reported an annual maximum peak stage of 4.95 ft on 4-DEC-2011 due to backwater from ice at USGS gaging station 09217000 Green River, WY. The average daily discharge was estimated at 1,170 cfs. |
| 20120813094407 | Green River, WY | 11/25/2009 | USGS Water Resources Data for Wyoming WY 2010 reported an annual maximum peak stage of 3.29 ft on 25-NOV-2009 due to backwater from ice at USGS gaging station 09217000 Green River, WY. The average daily discharge was 1100 cfs. |
| 20090908122137 | Green River, WY | 2/6/2008 | USGS Water Resources Data for Wyoming WY 2008 reported an annual maximum peak stage of 4.34 ft on 06-FEB-2008 due to backwater from ice at USGS gaging station 09217000 Green River near Green River, WY. The average daily discharge was 670 cfs. |
| 20090630072525 | Green River, WY | 2/2/2007 | USGS Water Resources Data for Wyoming WY 2007 reported an annual maximum peak stage of 4.20 ft on 02-FEB-2007 due to backwater from ice at USGS gaging station 09217000 Green River near Green River, WY. The average daily discharge was 880 cfs. |
| 20090610141436 | Green River, WY | 12/2/2005 | USGS Water Resources Data for Wyoming WY 2006 reported an annual maximum peak stage of 5.19 ft on 02-DEC-2005 due to backwater from ice at USGS gaging station 09217000 Green River near Green River, WY. The average daily discharge was 910 cfs. |
| 20031014095253 | Green River, WY | 12/3/1923 | The USDA Weather Bureau reported an ice gorge on the Green River at Green River, WY on December 3, 1923. |


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**Table 2.5-20 Historical Ice Jams Events at Bear River**

| Index Number in Ice Jam Database | Location | Date | Notes from Ice Jam Database |
|---|---|---|---|
| 71 | Evanston, WY | 1/1/1985 | Construction activities have led to increased deposition in a slower meandering reach some four miles downstream of town. Ice jam formed in early January flooding fields and the county road which is the primary access to the community. County removed ice from about a half-mile reach for $22K, relieving the problem. |
| 20030529081258 | Evanston, WY | 1/1/1984 | Response to CRREL Ice Jam Survey reported an ice jam on the Bear River at Evanston which caused agricultural flooding and damage to pavement, buildings, and bridges. Restriction at the bridge caused water to back up. No other information available. |


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**Table 2.5-21 Peak AFDD and Estimated Ice Thickness for 'KEMMERER 2 N WY US' Station (Sheet 1 of 3)**

| Water Year | Peak AFDD | | Estimated Ice Thickness |
|---|---|---|---|
| | °F days | Date | in. |
| 1934 | 472.5 | 2/28/1934 | 17.4 |
| 1935 | 1009 | 3/28/1935 | 25.4 |
| 1936 | 1282 | 2/29/1936 | 28.6 |
| 1937 | 1808 | 4/9/1937 | 34.0 |
| 1938 | 1083.5 | 3/31/1938 | 26.3 |
| 1939 | 1664.5 | 3/16/1939 | 32.6 |
| 1940 | 923 | 3/14/1940 | 24.3 |
| 1941 | 596 | 12/31/1940 | 19.5 |
| 1942 | N/A | N/A | N/A |
| 1943 | N/A | N/A | N/A |
| 1944 | N/A | N/A | N/A |
| 1945 | N/A | N/A | N/A |
| 1946 | N/A | N/A | N/A |
| 1947 | N/A | N/A | N/A |
| 1948 | N/A | N/A | N/A |
| 1949 | 2185 | 4/2/1949 | 37.4 |
| 1950 | 854 | 1/27/1950 | 23.4 |
| 1951 | 278 | 1/12/1951 | 13.3 |
| 1952 | 586.5 | 3/20/1952 | 19.4 |
| 1953 | 145 | 2/14/1953 | 9.6 |
| 1954 | 426 | 12/30/1953 | 16.5 |
| 1955 | 1479 | 4/7/1955 | 30.8 |
| 1956 | 1808 | 3/19/1956 | 34.0 |
| 1957 | 1444.5 | 3/27/1957 | 30.4 |
| 1958 | 1775.5 | 4/12/1958 | 33.7 |
| 1959 | 1058.5 | 3/28/1959 | 26.0 |
| 1960 | 1280.5 | 3/19/1960 | 28.6 |
| 1961 | 1006.5 | 3/12/1961 | 25.4 |
| 1962 | 1205.5 | 3/31/1962 | 27.8 |
| 1963 | 1012 | 3/19/1963 | 25.4 |
| 1964 | 2008 | 4/8/1964 | 35.8 |
| 1965 | 1745.5 | 3/30/1965 | 33.4 |
| 1966 | 2042 | 3/26/1966 | 36.2 |
| 1967 | 1486 | 4/2/1967 | 30.8 |
| 1968 | 1485.5 | 3/21/1968 | 30.8 |


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**Table 2.5-21 Peak AFDD and Estimated Ice Thickness for 'KEMMERER 2 N WY US' Station (Sheet 2 of 3)**

| Water Year | Peak AFDD | | Estimated Ice Thickness |
|---|---|---|---|
| | °F days | Date | in. |
| 1969 | 1918 | 3/29/1969 | 35.0 |
| 1970 | 1092.5 | 4/4/1970 | 26.4 |
| 1971 | 1413 | 4/4/1971 | 30.1 |
| 1972 | 1690.5 | 3/4/1972 | 32.9 |
| 1973 | 783.5 | 12/31/1972 | 22.4 |
| 1974 | 1255 | 3/24/1974 | 28.3 |
| 1975 | 1097 | 4/20/1975 | 26.5 |
| 1976 | 1082 | 3/30/1976 | 26.3 |
| 1977 | 933 | 3/31/1977 | 24.4 |
| 1978 | 215 | 1/28/1978 | 11.7 |
| 1979 | 274.5 | 12/31/1978 | 13.3 |
| 1980 | 1624.5 | 4/13/1980 | 32.2 |
| 1981 | 827.5 | 4/8/1981 | 23.0 |
| 1982 | 1754.5 | 4/21/1982 | 33.5 |
| 1983 | 1795 | 4/16/1983 | 33.9 |
| 1984 | 2446 | 4/14/1984 | 39.6 |
| 1985 | 2457 | 3/31/1985 | 39.7 |
| 1986 | 1450.5 | 2/15/1986 | 30.5 |
| 1987 | 1354.5 | 3/30/1987 | 29.4 |
| 1988 | 1801.5 | 4/1/1988 | 34.0 |
| 1989 | 1901.5 | 3/6/1989 | 34.9 |
| 1990 | 1344.5 | 3/16/1990 | 29.3 |
| 1991 | 1850.5 | 3/31/1991 | 34.4 |
| 1992 | 1743 | 2/29/1992 | 33.4 |
| 1993 | 2429 | 3/14/1993 | 39.4 |
| 1994 | 1899.5 | 3/30/1994 | 34.9 |
| 1995 | 1734 | 4/1/1995 | 33.3 |
| 1996 | 1472.5 | 3/31/1996 | 30.7 |
| 1997 | 1746.5 | 3/30/1997 | 33.4 |
| 1998 | 1865.5 | 3/21/1998 | 34.6 |
| 1999 | 1461 | 4/12/1999 | 30.6 |
| 2000 | 1222 | 4/1/2000 | 28.0 |
| 2001 | 1159.5 | 2/5/2001 | 27.2 |
| 2002 | 2264 | 3/27/2002 | 38.1 |
| 2003 | 1133.5 | 2/28/2003 | 26.9 |


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**Table 2.5-21 Peak AFDD and Estimated Ice Thickness for 'KEMMERER 2 N WY US' Station (Sheet 3 of 3)**

| Water Year | Peak AFDD | | Estimated Ice Thickness |
|---|---|---|---|
| | °F days | Date | in. |
| 2004 | 2051.5 | 3/14/2004 | 36.2 |
| 2005 | 976.5 | 1/31/2005 | 25.0 |
| 2006 | 2010.5 | 4/3/2006 | 35.9 |
| 2007 | 1810 | 3/12/2007 | 34.0 |
| 2008 | 2531.5 | 4/27/2008 | 40.3 |
| 2009 | 1619.5 | 4/6/2009 | 32.2 |
| 2010 | 1104.5 | 4/9/2010 | 26.6 |
| 2011 | 2141 | 5/1/2011 | 37.0 |
| 2012 | 1767.5 | 3/9/2012 | 33.6 |
| 2013 | 1612.5 | 4/18/2013 | 32.1 |
| 2014 | 1288 | 4/4/2014 | 28.7 |
| 2015 | 897 | 3/6/2015 | 24.0 |
| 2016 | 1502 | 4/1/2016 | 31.0 |
| 2017 | 1599.5 | 3/7/2017 | 32.0 |
| 2018 | 938 | 3/29/2018 | 24.5 |
| 2019 | 1978 | 3/31/2019 | 35.6 |
| 2020 | 1736.5 | 4/4/2020 | 33.3 |
| 2021 | 1532.5 | 4/1/2021 | 31.3 |
| 2022 | 1429 | 3/23/2022 | 30.2 |


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**Table 2.5-22 Peak AFDD and Estimated Ice Thickness for 'FOSSIL BUTTE, WY US' Station**

| Water Year | Peak AFDD | | Estimated Ice Thickness |
|---|---|---|---|
| | °F days | Date | in. |
| 1991 | 1618.5 | 3/30/1991 | 32.2 |
| 1992 | 1676.5 | 2/29/1992 | 32.8 |
| 1993 | 1357.5 | 3/15/1993 | 29.5 |
| 1994 | 1635 | 3/12/1994 | 32.3 |
| 1995 | 1364 | 3/8/1995 | 29.5 |
| 1996 | 1105.5 | 3/27/1996 | 26.6 |
| 1997 | 1607.5 | 4/13/1997 | 32.1 |
| 1998 | 1643.5 | 3/22/1998 | 32.4 |
| 1999 | 1191 | 4/5/1999 | 27.6 |
| 2000 | 897 | 2/13/2000 | 24.0 |
| 2001 | 1512 | 3/17/2001 | 31.1 |
| 2002 | 1514 | 3/26/2002 | 31.1 |
| 2003 | 1127 | 3/6/2003 | 26.9 |
| 2004 | 1276 | 3/14/2004 | 28.6 |
| 2005 | 1632 | 4/1/2005 | 32.3 |
| 2006 | 1085 | 3/28/2006 | 26.4 |
| 2007 | 917.5 | 3/7/2007 | 24.2 |
| 2008 | 2228 | 4/12/2008 | 37.8 |
| 2009 | 938 | 4/6/2009 | 24.5 |
| 2010 | 1526.5 | 4/8/2010 | 31.3 |
| 2011 | 1766.5 | 3/30/2011 | 33.6 |
| 2012 | 1309.5 | 2/21/2012 | 28.9 |
| 2013 | 1573 | 3/26/2013 | 31.7 |
| 2014 | 1239.5 | 2/27/2014 | 28.2 |
| 2015 | 655.5 | 3/6/2015 | 20.5 |
| 2016 | 1455.5 | 3/19/2016 | 30.5 |
| 2017 | 1491 | 3/13/2017 | 30.9 |
| 2018 | 843 | 3/8/2018 | 23.2 |
| 2019 | 1339 | 4/1/2019 | 29.3 |
| 2020 | 1850 | 4/4/2020 | 34.4 |
| 2021 | 1410 | 3/31/2021 | 30.0 |
| 2022 | 1464 | 3/23/2022 | 30.6 |


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**Table 2.5-23 Mapped Outcrop Area Percentages of Hydrogeologic Units in the Greater Green River Basin**

| Unit | Area (square miles) | Area (percentage) |
|---|---|---|
| Surface water and ice | 104.0 | 0.50 |
| Volcanic rocks (Mesozoic and Cenozoic) | 8.4 | 0.04 |
| Cenozoic hydrogeologic units | 17,177.1 | 82.62 |
| Quaternary hydrogeologic units | 4,190.6 | 20.16 |
| Upper Tertiary hydrogeologic units | 774.0 | 3.72 |
| Lower Tertiary hydrogeologic units | 12,212.5 | 58.74 |
| Mesozoic hydrogeologic units | 2,504.3 | 12.04 |
| Upper Cretaceous hydrogeologic units | 1,962.3 | 9.44 |
| Lower Cretaceous hydrogeologic units | 179.5 | 0.86 |
| Jurassic hydrogeologic units | 181.8 | 0.87 |
| Triassic hydrogeologic units | 180.7 | 0.87 |
| Paleozoic hydrogeologic units | 181.6 | 0.87 |
| Precambrian hydrogeologic units | 816.2 | 3.93 |


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**Table 2.5-24 Municipal Groundwater Use for 2005 in the Greater Green River Basin**

| City/Town | Wells | Depth (ft) | Population | GPCPD | Total (ac-ft/yr) |
|---|---|---|---|---|---|
| Bairoil | 5 | 35-51 | 96 | 350 | 38 |
| Big Piney | 7 | 90-900 | 455 | 90 | 46 |
| Marbleton | 8 | 580-830 | 811 | 787 | 715 |
| Opal | 3 | 400-480 | 99 | 150 | 17 |
| Superior | 3 | 1,700 | 239 | 146 | 39 |
| Wamsutter | 3 | 1,365-1,905 | 265 | 100 | 30 |
| **Total** | **29** | | **1,965** | **402** | **884** |


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**Table 2.5-25 Greater Green River Basin Municipal Service Area Groundwater Use Projections, Low Growth**

| City/Town Service Area | Current Population | Current Use (GPCPD) | Current Use (ac-ft/yr) | 2015 Pop | 2015 Use (ac-ft/yr) | 2035 Pop | 2035 Use (ac-ft/yr) | 2055 Pop | 2055 Use (ac-ft/yr) |
|---|---|---|---|---|---|---|---|---|---|
| Bairoil | 96 | 350 | 38 | 97 | 38 | 89 | 35 | 83 | 33 |
| Big Piney | 455 | 90 | 46 | 625 | 63 | 913 | 92 | 1,232 | 124 |
| Marbleton | 811 | 787 | 715 | 1,109 | 978 | 1,622 | 1,430 | 2,189 | 1,930 |
| Opal | 99 | 150 | 17 | 112 | 19 | 122 | 20 | 134 | 23 |
| Superior | 239 | 146 | 39 | 242 | 40 | 224 | 37 | 207 | 34 |
| Wamsutter | 265 | 100 | 30 | 267 | 30 | 251 | 28 | 239 | 27 |
| TOTAL | 1,965 | 347 | 884 | 2,452 | 1,167 | 3,221 | 1,642 | 4,084 | 2,170 |


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**Table 2.5-26 Greater Green River Basin Municipal Service Area Groundwater Use Projections, Moderate Growth**

| City/Town Service Area | Current Population | Current Use (GPCPD) | Current Use (ac-ft/yr) | 2015 Pop | 2015 Use (ac-ft/yr) | 2035 Pop | 2035 Use (ac-ft/yr) | 2055 Pop | 2055 Use (ac-ft/yr) |
|---|---|---|---|---|---|---|---|---|---|
| Bairoil | 96 | 350 | 38 | 109 | 43 | 122 | 48 | 130 | 51 |
| Big Piney | 455 | 90 | 46 | 703 | 71 | 1,244 | 125 | 1,932 | 195 |
| Marbleton | 811 | 787 | 715 | 1,247 | 1,099 | 2,210 | 1,948 | 3,433 | 3,026 |
| Opal | 99 | 150 | 17 | 126 | 21 | 166 | 28 | 211 | 35 |
| Superior | 239 | 146 | 39 | 272 | 44 | 305 | 50 | 325 | 53 |
| Wamsutter | 265 | 100 | 30 | 300 | 34 | 342 | 38 | 374 | 42 |
| TOTAL | 1,965 | 347 | 884 | 2,757 | 1,312 | 4,389 | 2,238 | 6,405 | 3,403 |


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**Table 2.5-27 Greater Green River Basin Municipal Service Area Groundwater Use Projections, High Growth**

| City/Town Service Area | Current Population | Current Use (GPCPD) | Current Use (ac-ft/yr) | 2015 Pop | 2015 Use (ac-ft/yr) | 2035 Pop | 2035 Use (ac-ft/yr) | 2055 Pop | 2055 Use (ac-ft/yr) |
|---|---|---|---|---|---|---|---|---|---|
| Bairoil | 96 | 350 | 38 | 109 | 43 | 164 | 64 | 168 | 66 |
| Big Piney | 455 | 90 | 46 | 703 | 71 | 1,672 | 169 | 2,488 | 251 |
| Marbleton | 811 | 787 | 715 | 1,247 | 1,099 | 2,970 | 2,618 | 4,421 | 3,897 |
| Opal | 99 | 150 | 17 | 126 | 21 | 223 | 37 | 271 | 46 |
| Superior | 239 | 146 | 39 | 272 | 44 | 409 | 67 | 419 | 69 |
| Wamsutter | 265 | 100 | 30 | 300 | 34 | 459 | 51 | 482 | 54 |
| TOTAL | 1,965 | 347 | 884 | 2,757 | 1,312 | 5,897 | 3,007 | 8,249 | 4,382 |


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**Table 2.5-28 Borehole Packer Test Results**
**(Sheet 1 of 3)**

| Boring | Zone | Depth Interval (ft bgs) | Midpoint Depth (ft bgs) | Hydraulic Conductivity (ft/day) | Analysis Notes |
|---|---|---|---|---|---|
| **Midpoint Depth 20 to 30 ft bgs** | | | | | |
| B-115-PT | Z1 | 23.5 to 31.0 | 27.25 | 6.5E+00 | Top transducer above upper packer response to injected water is likely due to hydraulic connection through formation. High flow rate (max ~10 gpm). |
| B-139-PT | Z1 | 24.0 to 31.5 | 27.75 | 4.5E+00 | Very minor response (top transducer) of injected water above the top packer likely due to hydraulic connection through formation. High flow rate (max ~7 gpm). |
| **Midpoint Depth 30 to 40 ft bgs** | | | | | |
| B-115-PT | Z2 | 31.5 to 39.0 | 35.25 | 6.1E-02 | Max flow ~0.15 gpm |
| B-116-PT | Z1 | 29.0 to 36.5 | 32.75 | 1.1E+00 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~2.75 gpm |
| B-124-PT | Z1 | 35.5 to 43.0 | 39.25 | 7.6E-01 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~2.3 gpm |
| B-132-PT | Z1 | 30.5 to 38.0 | 34.25 | 6.7E-01 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~1.60 gpm |
| B-139-PT | Z2 | 31.5 to 39.0 | 35.25 | NA | No flow into the formation |
| B-142-PT | Z1 | 29.0 to 36.5 | 32.75 | 2.7E-01 | Max flow ~0.5 gpm |
| B-144-PT | Z1 | 34.0 to 41.5 | 37.75 | NA | No flow into the formation |
| B-171-PT | Z1 | 35.5 to 43.0 | 39.25 | 1.6E-01 | Max flow ~0.40 gpm |
| B-177-PT | Z1 | 33.0 to 40.5 | 36.75 | 6.2E-01 | Max flow ~1.6 gpm |
| B-181-PT | Z1 | 28.4 to 35.9 | 32.15 | 3.1E+00 | Top transducer above upper packer response to injected water is likely due to hydraulic connection through formation. High flow rate (max ~7 gpm). |


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**Table 2.5-28 Borehole Packer Test Results**
**(Sheet 2 of 3)**

| Boring | Zone | Depth Interval (ft bgs) | Midpoint Depth (ft bgs) | Hydraulic Conductivity (ft/day) | Analysis Notes |
|---|---|---|---|---|---|
| B-181-PT | Z2 | 36.0 to 43.5 | 39.75 | 1.3E+00 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~3.0 gpm |
| B-182-PT | Z1 | 35.0 to 42.5 | 38.75 | 9.4E-01 | Max flow ~1.8 gpm |
| **Midpoint Depth 40 to 50 ft bgs** | | | | | |
| B-115-PT | Z3 | 38.6 to 46.1 | 42.35 | 5.2E-02 | Overall intake was very low. Sequences C, D and E were discarded for K analysis. Even though, sequences A, B were used - intake was very low and in many cases were zero. The depth interval represents near impermeable. |
| B-116-PT | Z2 | 38.0 to 45.5 | 41.75 | 1.9E+00 | Permeable zone. Flow rate of ~5 gpm. Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. |
| B-126-PT | Z1 | 39.0 to 46.5 | 42.75 | NA | No flow into the formation. Flow < 0.1 gpm. |
| B-132-PT | Z2 | 38.0 to 45.5 | 41.75 | 3.5E-01 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~1.0 gpm |
| B-139-PT | Z3 | 39.0 to 46.5 | 42.75 | 1.4E-01 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. The high pressure in lower transducer is likely due to large pressure was applied during inflating of the lower packer. Max flow ~0.25 gpm |


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**Table 2.5-28 Borehole Packer Test Results**
**(Sheet 3 of 3)**

| Boring | Zone | Depth Interval (ft bgs) | Midpoint Depth (ft bgs) | Hydraulic Conductivity (ft/day) | Analysis Notes |
|---|---|---|---|---|---|
| B-142-PT | Z2 | 36.3 to 43.8 | 40.05 | 5.3E-02 | Very low flow rates (max flow ~0.1 gpm). Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. |
| B-144-PT | Z2 | 42.0 to 49.5 | 45.75 | NA | Flow rates do not vary with excess pressure and thus K estimate was not made. Max flow ~0.1 gpm |
| B-150-PT | Z1 | 37.8 to 45.3 | 41.55 | NA | No flow into the formation |
| B-157-PT | Z1 | 39.5 to 47.0 | 43.25 | 9.3E-01 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~2.5 gpm |
| B-160-PT | Z1 | 38.0 to 45.5 | 41.75 | 1.6E+00 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~4.5 gpm |
| B-171-PT | Z2 | 43.5 to 51.0 | 47.25 | 1.3E-01 | Max flow ~0.3 gpm |
| B-177-PT | Z2 | 41.0 to 48.5 | 44.75 | 6.9E-01 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~1.2 gpm |
| B-181-PT | Z3 | 43.6 to 51.1 | 47.35 | 8.4E-01 | Max flow ~2.5 gpm |
| B-182-PT | Z2 | 44.0 to 51.5 | 47.75 | 6.5E-02 | Lower transducer below lower packer showed response to injected water due to likely hydraulic connection through the formation. Max flow ~0.1 gpm |


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**Table 2.5-29 Borehole Packer Test Summary Statistics**

| Midpoint Depth Interval (ft bgs) | No. of Tests | Minimum Hydraulic Conductivity (ft/d) | Maximum Hydraulic conductivity (ft/d) | Geometric Mean Hydraulic Conductivity (ft/d) |
|---|---|---|---|---|
| 20 to 30 | 2 | 4.5E+00 | 6.5E+00 | 5.4E+00 |
| 30 to 40 | 10 | 6.1E-02 | 3.1E+00 | 5.7E-01 |
| 40 to 50 | 11 | 5.2E-02 | 1.9E+00 | 3.0E-01 |
| 20 to 50 | 23 | 5.2E-02 | 6.5E+00 | 5.1E-01 |



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**Table 2.5-30 Hydraulic Conductivity Results for Slug Tests and Long-Term Recovery (Sheet 1 of 3)**

| Observation Well | Rock Weathering | Well Screen Midpoint (ft bgs) | Type of Slug Test or Recovery Data | Hydraulic Conductivity (ft/d) | Quality of Analysis |
|---|---|---|---|---|---|
| **Weathered Rock - Slug Tests** | | | | | |
| OW-106U | Slightly weathered | 35 | Pneumatic slug | 1.7E+01 | A |
| OW-108U | Slightly weathered | 35 | Water slug | 1.4E-01 | B |
| OW-109U | Slightly weathered | 35 | Solid slug | 1.5E+00 | B |
| OW-117U | Slightly weathered | 35.2 | Pneumatic slug | 7.1E-01 | B |
| OW-121U | Completely to moderately weathered | 29.6 | Water slug | 1.1E+00 | B |
| OW-122U | Moderately to slightly weathered | 44.5 | Pneumatic slug | 2.2E-01 | B |
| OW-136U | Highly to slightly weathered | 28.1 | Pneumatic slug | 4.3E+00 | A |
| OW-137U | Moderately weathered to fresh | 29.4 | Solid slug | 7.1E-03 | C |
| OW-140U | Moderately weathered to fresh | 35 | Pneumatic slug | 4.0E-02 | B |
| OW-149U | Highly to slightly weathered | 35 | Water slug | 3.1E+00 | A |
| OW-167U | Slightly weathered | 40 | Solid slug | 1.1E-01 | C |
| OW-179U | Completely to slightly weathered | 30 | Water slug | 4.3E-01 | B |
| OW-185U | Moderately weathered | 41 | Pneumatic slug | 1.4E+00 | C |
| OW-186U | Slightly weathered to fresh | 39.8 | Pneumatic slug | 4.3E-01 | B |
| OW-187U | Moderately weathered to fresh | 54.3 | Pneumatic slug | 1.3E+01 | B |
| OW-188U | Completely to slightly weathered | 30 | Pneumatic slug | 2.1E+00 | B |
| OW-189U | Highly to slightly weathered | 39.9 | Pneumatic slug | 1.6E+00 | A |
| OW-190U | Highly to slightly weathered | 40 | Pneumatic slug | 5.1E+00 | B |
| OW-191U | Highly to slightly weathered | 45 | Pneumatic slug | 8.5E+00 | A |
| OW-106M | Slightly weathered | 55 | Insufficient data | NA | D |


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**Table 2.5-30 Hydraulic Conductivity Results for Slug Tests and Long-Term Recovery (Sheet 2 of 3)**

| Observation Well | Rock Weathering | Well Screen Midpoint (ft bgs) | Type of Slug Test or Recovery Data | Hydraulic Conductivity (ft/d) | Quality of Analysis |
|---|---|---|---|---|---|
| OW-119M | Slightly weathered to fresh | 40.2 | Pneumatic slug | 1.3E+00 | B |
| OW-137M | Slightly weathered to fresh | 75 | Pneumatic slug | 9.4E-02 | C |
| OW-188M | Slightly weathered to fresh | 50 | Pneumatic slug | 9.6E-01 | C |
| OW-191M | Slightly weathered to fresh | 60 | Pneumatic slug | 3.7E+00 | B |
| **Weathered Rock - Long-Term Recovery** | | | | | |
| OW-119U | Highly to slightly weathered | 25 | Insufficient data | NA | D |
| OW-127U | Completely to slightly weathered | 29.8 | Insufficient data | NA | D |
| OW-153U | Slightly weathered | 35 | Long-term recovery | 1.8E-04 | A |
| OW-168U | Completely to moderately weathered | 30.2 | Insufficient data | NA | D |
| **Fresh Rock - Slug Tests** | | | | | |
| OW-164U | Fresh | 40 | Pneumatic slug | 6.0E-01 | A |
| OW-122M | Fresh | 60 | Pneumatic slug | 3.4E-01 | B |
| OW-140M | Fresh | 59.5 | Pneumatic slug | 2.5E-01 | C |
| OW-164M | Fresh | 59.6 | Pneumatic slug | 6.2E+00 | B |
| OW-168M(A) | Fresh | 50 | Pneumatic slug | 1.5E+00 | B |
| OW-179M | Fresh | 45 | Pneumatic slug | 7.9E-01 | B |
| OW-185M | Fresh | 70 | Insufficient data | NA | D |
| OW-186M | Fresh | 59.9 | Pneumatic slug | 2.3E-01 | C |
| OW-187M | Fresh | 69.8 | Pneumatic slug | 2.8E+00 | A |
| OW-189M | Fresh | 60.1 | Pneumatic slug | 9.6E+01 | A |
| OW-190M | Fresh | 55 | Pneumatic slug | 1.7E+00 | C |
| **Fresh Rock - Long-Term Recovery** | | | | | |
| OW-108M | Fresh | 75 | Long-term recovery | 2.3E-06 | C |
| OW-109M | Fresh | 60 | Long-term recovery | 2.1E-05 | C |
| OW-117M | Fresh | 74.9 | Long-term recovery | 1.9E-06 | C |
| OW-121M(A) | Fresh | 60 | Long-term recovery | 6.5E-06 | C |
| OW-127M | Fresh | 49.6 | Long-term recovery | 4.3E-06 | C |



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**Table 2.5-30 Hydraulic Conductivity Results for Slug Tests and Long-Term Recovery (Sheet 3 of 3)**

| Observation Well | Rock Weathering | Well Screen Midpoint (ft bgs) | Type of Slug Test or Recovery Data | Hydraulic Conductivity (ft/d) | Quality of Analysis |
|---|---|---|---|---|---|
| OW-136M | Fresh | 70.1 | Long-term recovery | 3.7E-07 | C |
| OW-149M | Fresh | 60 | Long-term recovery | 4.8E-06 | C |
| OW-153M | Fresh | 64.5 | Long-term recovery | 3.4E-06 | C |
| OW-167M | Fresh | 80 | Long-term recovery | 2.8E-06 | C |
| OW-117L | Fresh | 110.1 | Long-term recovery | 9.4E-05 | A |
| OW-122L | Fresh | 105 | Long-term recovery | 7.9E-07 | C |
| OW-153L | Fresh | 100.9 | Long-term recovery | 5.7E-04 | B |
| OW-167L | Fresh | 145 | Insufficient data | NA | D |
| **Geometric Mean** | | | | | |
| Weathered Rock | | | | 6.2E-01 | |
| Fresh Rock | | | | 1.7E-03 | |

Quality of analysis:

A = Good (majority of data show log-linear response and data available to H/Ho = 0.1, i.e., 90% recovery).

B = Fair (log-linear response at least for H/Ho = 0.2 to 0.3; may be non-linear above or below 0.2 to 0.3).

C = Poor (poor log-linear response for H/Ho = 0.2 to 0.3, or data not available for H/Ho = 0.2 to 0.3).

D = Insufficient data


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**Table 2.5-31 Porosity and Specific Gravity Test Results**
**(Sheet 1 of 2)**

| Boring | Upper Interval (ft bgs) | Lower Interval (ft bgs) | Midpoint (ft bgs) | Porosity (%) | Specific Gravity |
|---|---|---|---|---|---|
| B-117 | 28.7 | 29.3 | 29.0 | 2.7 | |
| B-117 | 38.0 | 44.3 | 41.2 | 2.6 | 2.473 |
| B-117 | 47.5 | 50.4 | 49.0 | 3.0 | |
| B-117 | 55.4 | 63.2 | 59.3 | 2.6 | 2.505 |
| B-117 | 72.5 | 75.0 | 73.8 | 2.9 | |
| B-117 | 87.3 | 93.4 | 90.4 | 2.2 | 2.491 |
| B-117 | 103.6 | 108.0 | 105.8 | 2.2 | |
| B-117 | 117.0 | 122.9 | 120.0 | 2.2 | 2.504 |
| B-117 | 135.4 | 137.9 | 136.7 | 2.2 | |
| B-117 | 150.4 | 153.5 | 152.0 | 2.1 | 2.516 |
| B-122 | 18.9 | 21.4 | 20.2 | | 2.374 |
| B-122 | 28.6 | 29.1 | 28.9 | 4.5 | |
| B-122 | 55.3 | 64.4 | 59.9 | | 2.462 |
| B-122 | 68.2 | 70.5 | 69.4 | 3.6 | |
| B-122 | 84.8 | 90.7 | 87.8 | 3.4 | 2.486 |
| B-122 | 95.7 | 105.1 | 100.4 | 2.3 | 2.504 |
| B-122 | 113.1 | 115.7 | 114.4 | 2.6 | |
| B-122 | 126.7 | 131.6 | 129.2 | 2.5 | 2.528 |
| B-122 | 142.8 | 145.6 | 144.2 | 3.5 | |
| B-122 | 157.8 | 160.6 | 159.2 | 2.3 | 2.511 |
| B-122 | 176.2 | 179.3 | 177.8 | 3.2 | |
| B-122 | 192.5 | 194.4 | 193.5 | 2.4 | 2.479 |
| B-122 | 202.4 | 207.3 | 204.9 | 2.6 | |
| B-127 | 18.7 | 25.0 | 21.9 | | 2.374 |
| B-127 | 30.3 | 37.1 | 33.7 | 8.9 | |
| B-127 | 39.1 | 46.7 | 42.9 | 6.2 | 2.457 |
| B-127 | 50.4 | 51.4 | 50.9 | 2.7 | |
| B-127 | 56.1 | 65.6 | 60.9 | 4.0 | 2.495 |
| B-127 | 73.9 | 76.6 | 75.3 | 2.5 | |
| B-127 | 87.9 | 94.6 | 91.3 | 3.4 | 2.513 |
| B-127 | 102.1 | 109.0 | 105.6 | 3.9 | |
| B-127 | 118.3 | 121.0 | 119.7 | 3.8 | 2.517 |
| B-127 | 134.0 | 136.5 | 135.3 | 4.3 | |
| B-127 | 148.9 | 151.9 | 150.4 | 3.3 | 2.494 |
| B-129 | 28.5 | 29.1 | 28.8 | 6.1 | |
| B-129 | 32.8 | 33.8 | 33.3 | 7.6 | |


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**Table 2.5-31 Porosity and Specific Gravity Test Results**
**(Sheet 2 of 2)**

| Boring | Upper Interval (ft bgs) | Lower Interval (ft bgs) | Midpoint (ft bgs) | Porosity (%) | Specific Gravity |
|---|---|---|---|---|---|
| B-130 | 24.7 | 25.5 | 25.1 | 5.4 | |
| B-130 | 28.3 | 29.2 | 28.8 | 6.3 | |
| B-134 | 29.3 | 29.9 | 29.6 | 8.5 | |
| B-138 | 33.3 | 34.4 | 33.9 | 7.3 | |
| B-149 | 31.8 | 32.3 | 32.1 | 7.5 | 2.443 |
| B-149 | 60.3 | 62.0 | 61.2 | 2.5 | 2.515 |
| B-149 | 90.3 | 92.3 | 91.3 | 2.8 | 2.530 |
| B-165 | 121.5 | 123.5 | 122.5 | 3.5 | 2.531 |
| B-165 | 148.2 | 150.1 | 149.2 | 4.3 | 2.534 |
| B-169 | 31.3 | 32.1 | 31.7 | 6.4 | 2.353 |
| B-169 | 61.8 | 63.3 | 62.6 | 3.9 | 2.504 |
| B-169 | 88.7 | 90.7 | 89.7 | 4.6 | 2.504 |
| **Summary Statistics - Less than 50 ft bgs** | | | | | |
| Count | | | | 14 | 6 |
| Minimum | | | | 2.6 | 2.353 |
| Median | | | | 6.3 | 2.409 |
| Maximum | | | | 8.9 | 2.473 |
| Average | | | | 5.9 | 2.412 |
| Geometric Mean | | | | 5.5 | 2.412 |
| **Summary Statistics - Greater than 50 ft bgs** | | | | | |
| Count | | | | 31 | 20 |
| Minimum | | | | 2.1 | 2.462 |
| Median | | | | 2.8 | 2.505 |
| Maximum | | | | 4.6 | 2.534 |
| Average | | | | 3.0 | 2.506 |
| Geometric Mean | | | | 3.0 | 2.506 |


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**Table 2.5-32 Water Sampling Field Parameters**
**(Sheet 1 of 2)**

| Well/Sample ID | Date | Parameter[1] | | | | | |
|---|---|---|---|---|---|---|---|
| | | pH (std units) | Electrical Conductivity (mS/cm) | Turbidity (NTU) | Dissolved Oxygen (mg/L)[4] | Temperature (°C) | Redox Potential (± mv) |
| OW-117-U | 7/8/2022 | 7.37 | 11.723 | 3.35 | 0.02 | 11.6 | -130.0 |
| OW-119-M | 7/13/2022 | 6.89 | 9.579 | 6.28 | 0.02 | 9.6 | -184.3 |
| OW-121-U | 7/12/2022 | 7.02 | 5.750 | 1.60 | -0.03 | 11.0 | -145.1 |
| OW-122-M [2] | 7/15/2022 | 6.96 | 6.024 | 38.2 | 3.10 | 9.7 | -28.3 |
| OW-122-M [2][6] | 7/19/2022 | 6.91 | 9.009 | 67.4 | 0.96 | 11.1 | -23.4 |
| OW-122-U [2] | 7/18/2022 | 6.96 | 6.677 | 5.00 | 0.19 | 11.8 | -87.1 |
| OW-136-U | 7/12/2022 | 6.88 | 23.352 | 7.87 | 0.00 | 9.0 | 13.5 |
| OW-137-M [2][3] | 7/15/2022 | 7.34 | 10.728 | 83.8 | 0.86 | 11.4 | -127.5 |
| OW-140-U | 7/12/2022 | 7.01 | 16.725 | 17.2 | 1.53 | 10.4 | 47.1 |
| OW-153-L [2] | 7/18/2022 | 8.62 | 11.876 | 40.3 | 1.58 | 9.3 | 14.4 |
| OW-164-M | 7/15/2022 | 7.10 | 9.310 | 6.34 | -0.07 | 12.2 | -188.5 |
| OW-164-U | 7/15/2022 | 7.10 | 8.549 | 6.58 | -0.01 | 12.0 | -204.7 |
| OW-168-M(A) | 7/18/2022 | 6.94 | 6.482 | 6.31 | 0.07 | 13.1 | -48.2 |
| OW-185-M | 7/14/2022 | 8.53 | 7.490 | 16.1 | 1.87 | 18.7 | -112.8 |
| OW-185-U | 7/13/2022 | 6.85 | 17.797 | 8.25 | -0.16 | 13.1 | -67.0 |
| OW-186-M | 7/15/2022 | 7.27 | 9.992 | 25.3 | 0.01 | 10.2 | -230.5 |
| OW-187-M | 7/13/2022 | 7.49 | 19.302 | 6.01 | -0.02 | 9.0 | -88.0 |
| OW-187-U | 7/11/2022 | 7.27 | 20.650 | 1.10 | -0.04 | 8.7 | -26.3 |
| OW-188-M | 7/14/2022 | 7.04 | 8.935 | 0.65 | 0.30 | 8.4 | -145.9 |
| OW-188-U | 7/13/2022 | 7.00 | 16.570 | 9.79 | 0.91 | 9.2 | 46.7 |
| OW-189-U | 7/14/2022 | 7.29 | 12.438 | 7.08 | 0.13 | 11.4 | -47.7 |
| OW-190-M | 7/14/2022 | 7.03 | 14.224 | 0.54 | -0.01 | 7.7 | -78.0 |


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**Table 2.5-32 Water Sampling Field Parameters**
**(Sheet 2 of 2)**

| Well/Sample ID | Date | Parameter[1] | | | | | |
|---|---|---|---|---|---|---|---|
| | | pH (std units) | Electrical Conductivity (mS/cm) | Turbidity (NTU) | Dissolved Oxygen (mg/L)[4] | Temperature (°C) | Redox Potential (± mv) |
| OW-191-M | 7/12/2022 | 6.91 | 20.534 | 2.88 | -0.08 | 12.7 | -24.3 |
| OW-191-U | 7/11/2022 | 6.77 | 16.220 | 0.63 | -0.06 | 10.5 | -66.0 |
| Upstream SW [5] | 7/21/2022 | 8.00 | 1.448 | 162.0 | 3.89 | 24.1 | 79.3 |
| Downstream SW [5] | 7/21/2022 | 7.60 | 1.678 | 28.4 | 2.79 | 18.1 | 74.3 |

Notes:

[1] Parameter values shown are at time of sampling.

[2] Well purged dry; water quality reading taken after sufficient water recharged for sampling.

[3] OW-137-M not submitted for laboratory analysis due to low volume.

[4] Negative values are within calibration acceptance criteria (+/- 0.5 mg/L) and represent measurements at or near 0 mg/L.

[5] Surface water (SW) sample collected from North Fork of Little Muddy Creek.

[6] OW-122-M sampled on 07/19/2022 for Kd analysis only.


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**Table 2.5-33a Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 1 of 3)**

| | | | Location: | OW-117-U | | | OW-119-M | | | OW-121-U | | | OW-122-M | | | OW-122-U | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | Sample ID: | OW-117-U | | | OW-119-M | | | OW-121-U | | | OW-122-M | | | OW-122-U | | |
| | | | Sample Date: | 7/8/2022 | | | 7/13/2022 | | | 7/12/2022 | | | 7/15/2022 | | | 7/18/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| N | Calculation | Dissolved Organic Nitrogen | mg/L | 0.7 | J | | 1 | | | 1 | U | | 1 | U | | 1 | UJ | |
| N | EPA 200.7 | Aluminum, Total | mg/L | 0.09 | J+ | | 0.2 | | | 0.5 | U | | | | | 0.2 | | |
| N | EPA 200.7 | Arsenic, Total | mg/L | 0.009 | J | | 0.05 | U | | 0.5 | U | | | | | 0.02 | J | |
| N | EPA 200.7 | Barium, Total | mg/L | 0.012 | | | 0.009 | | | 0.017 | J | | | | | 0.007 | J | |
| N | EPA 200.7 | Calcium, Total | mg/L | 136 | | | 346 | | | 410 | | | | | | 347 | | |
| N | EPA 200.7 | Copper, Total | mg/L | 0.005 | U | | 0.005 | U | | 0.05 | U | | | | | 0.021 | | |
| N | EPA 200.7 | Iron, Total | mg/L | 0.84 | | | 1.94 | | | 1.89 | | | 2.46 | | | 0.85 | | |
| N | EPA 200.7 | Lead, Total | mg/L | 0.02 | U | | 0.005 | J | | 0.2 | U | | | | | 0.04 | U | |
| N | EPA 200.7 | Magnesium, Total | mg/L | 73.8 | | | 262 | | | 362 | | | | | | 296 | | |
| N | EPA 200.7 | Manganese, Total | mg/L | 0.045 | J+ | | 0.278 | | | 2.6 | | | 0.495 | | | 1.64 | | |
| N | EPA 200.7 | Phosphorus, Total as P | mg/L | 0.05 | J+ | | 0.04 | J+ | | 0.1 | U | | 0.01 | J | | 0.02 | | |
| N | EPA 200.7 | Potassium, Total | mg/L | 8 | | | 11.5 | | | 17.5 | | | | | | 14.8 | | |
| N | EPA 200.7 | Silica, (as $SiO_2$) Total | mg/L | 9.4 | | | 10.9 | | | 11.9 | | | 10.4 | | | 9.6 | | |
| N | EPA 200.7 | Sodium, Total | mg/L | 2670 | | | 1830 | | | 1260 | | | | | | 1270 | | |
| N | EPA 200.7 | Strontium, Total | mg/L | 8.29 | | | 8.99 | | | 7.84 | | | | | | 6.99 | | |
| N | EPA 200.7 | Vanadium, Total | mg/L | 0.01 | U | | 0.01 | U | | 0.1 | U | | | | | 0.02 | U | |
| N | EPA 200.7 | Zinc, Total | mg/L | 0.01 | U | | 0.01 | U | | 0.06 | J | | | | | 0.02 | U | |
| N | EPA 245.1 | Mercury, Total | mg/L | 0.00015 | UJ | | 0.00015 | U | | 0.00015 | U | | | | | 0.00015 | U | |
| N | EPA 300.0 | Fluoride | mg/L | 0.06 | J | | 0.08 | J | | 5.47 | | | 0.1 | U | | 0.1 | UJ | |
| N | EPA 300.0 | Nitrate as N | mg/L | 1 | U | | 5 | U | | 5 | U | | 0.1 | U | | 0.1 | UJ | |
| N | Hach 8000 | Chemical Oxygen Demand | mg/L | 34 | J+ | | 37 | | | 130 | | | 17 | | | 25 | J | |
| N | SM 2120 B | Color | Color Units | 5 | | | 10 | | | 10 | | | 60 | | | 10 | J | |
| N | SM 2320 B | Alkalinity - Bicarbonate (as $CaCO_3$) | mg/L | 566 | | | 456 | | | 400 | | | 474 | | | 448 | J | |
| N | SM 2320 B | Alkalinity - Carbonate (as $CaCO_3$) | mg/L | 1 | U | | 1 | U | | 1 | U | | 1 | U | | 1 | UJ | |
| N | SM 2320 B | Alkalinity - Hydroxide (as $CaCO_3$) | mg/L | 1 | U | | 1 | U | | 1 | U | | 1 | U | | 1 | UJ | |
| N | SM 2320 B | Alkalinity - Total (as $CaCO_3$) | mg/L | 566 | | | 456 | | | 400 | | | 474 | | | 448 | J | |
| N | SM 2340 B | Hardness, Soluble as $CaCO_3$ | mg/L | | | | | | | | | | 2020 | | | | | |
| N | SM 2340 B | Hardness, Total as $CaCO_3$ | mg/L | 643 | | | 1940 | | | 2510 | | | | | | 2080 | | |
| N | SM 2540 C | Total Dissolved Solids (TDS) | mg/L | 8410 | | | 7950 | | | 7130 | | | 7700 | | | 7940 | J | |
| N | SM 2540 D | Total Suspended Solids (TSS) | mg/L | 4 | J+ | | 6 | | | 6 | | | 17 | | | 5 | J | |
| N | SM 4500 NH3 H | Ammonia as N | mg/L | 2.8 | | | 1.5 | | | 0.2 | U | | 1.1 | | | 0.2 | UJ | |
| N | SM 4500 NH3-D | Ammonia, Dissolved | mg/L | 2.5 | | | 1.4 | | | 0.2 | U | | 1.1 | | | 0.09 | J | |


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**Table 2.5-33a Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 2 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-117-U | | | OW-119-M | | | OW-121-U | | | OW-122-M | | | OW-122-U | | |
| | | | **Sample ID:** | OW-117-U | | | OW-119-M | | | OW-121-U | | | OW-122-M | | | OW-122-U | | |
| | | | **Sample Date:** | 7/8/2022 | | | 7/13/2022 | | | 7/12/2022 | | | 7/15/2022 | | | 7/18/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| N | SM 4500 NO2-B | Nitrite as N | mg/L | 0.1 | U | | 0.1 | U | | 0.1 | U | | 0.006 | J | | 0.1 | UJ | |
| N | SM 4500 Norg | Dissolved Kjeldahl Nitrogen | mg/L | 3.2 | | | 2.4 | | | 1 | U | | 1.6 | | | 0.6 | J | |
| N | SM 4500 P-E | Phosphate, ortho as P | mg/L | 0.01 | U | | 0.007 | J | | 0.01 | | | 0.01 | U | | 0.01 | UJ | |
| N | SM 4500 S2-D | Sulfide | mg/L | 0.2 | J- | | 0.5 | | | 0.2 | J- | | 0.1 | U | | 0.1 | UJ | |
| N | SM 5210 B | Biochemical Oxygen Demand | mg/L | 5 | U | | 5 | U | | 62 | J+ | | 6 | | | 7 | J | |
| N | SM 5310 C | Total Organic Carbon | mg/L | 1.1 | J | | 2.2 | J- | | 20.4 | | | 2.3 | J | | 2.2 | J | |
| Y | EPA 200.7 | Aluminum, Dissolved | mg/L | | | | | | | | | | 0.3 | | | | | |
| Y | EPA 200.7 | Arsenic, Dissolved | mg/L | | | | | | | | | | 0.02 | J | | | | |
| Y | EPA 200.7 | Barium, Dissolved | mg/L | | | | | | | | | | 0.012 | J | | | | |
| Y | EPA 200.7 | Calcium, Dissolved | mg/L | | | | | | | | | | 362 | | | | | |
| Y | EPA 200.7 | Copper, Dissolved | mg/L | | | | | | | | | | 0.025 | U | | | | |
| Y | EPA 200.7 | Iron, Dissolved | mg/L | 0.73 | | | 2.15 | | | 2.14 | | | 2.28 | | | 0.99 | | |
| Y | EPA 200.7 | Lead, Dissolved | mg/L | | | | | | | | | | 0.1 | U | | | | |
| Y | EPA 200.7 | Magnesium, Dissolved | mg/L | | | | | | | | | | 272 | | | | | |
| Y | EPA 200.7 | Manganese, Dissolved | mg/L | 0.042 | | | 0.262 | | | 2.89 | | | 0.542 | | | 1.87 | | |
| Y | EPA 200.7 | Potassium, Dissolved | mg/L | | | | | | | | | | 11.6 | | | | | |
| Y | EPA 200.7 | Silica, (as $SiO_2$) Dissolved | mg/L | 8.7 | | | 10.4 | | | 11.7 | | | 10.4 | | | 9.6 | | |
| Y | EPA 200.7 | Sodium, Dissolved | mg/L | | | | | | | | | | 1430 | | | | | |
| Y | EPA 200.7 | Strontium, Dissolved | mg/L | | | | | | | | | | 8.73 | | | | | |
| Y | EPA 200.7 | Vanadium, Dissolved | mg/L | | | | | | | | | | 0.05 | U | | | | |
| Y | EPA 200.7 | Zinc, Dissolved | mg/L | | | | | | | | | | 0.05 | U | | | | |
| Y | EPA 245.1 | Mercury, Dissolved | mg/L | | | | | | | | | | 0.00015 | U | | | | |

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**Table 2.5-33a Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 3 of 3)**

| | | | Location: | OW-117-U | | | OW-119-M | | | OW-121-U | | | OW-122-M | | | OW-122-U | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | Sample ID: | OW-117-U | | | OW-119-M | | | OW-121-U | | | OW-122-M | | | OW-122-U | | |
| | | | Sample Date: | 7/8/2022 | | | 7/13/2022 | | | 7/12/2022 | | | 7/15/2022 | | | 7/18/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| **Radiometric Parameters** | | | | | | | | | | | | | | | | | | |
| N | 900.0 | Gross Alpha | pCi/L | 3.19 | U | 40.9 | 14.7 | U | 36 | 42 | U | 41.9 | -5.53 | U | 17.7 | 17.9 | U | 46.8 |
| N | 900.0 | Gross Beta | pCi/L | 20.9 | U | 16.7 | 9.02 | U | 14.8 | 11.4 | U | 12 | 8.63 | U | 11 | 29.3 | | 17.5 |
| N | 903.0 | Radium-226 | pCi/L | 0.366 | | 0.116 | 0.709 | | 0.181 | 0.731 | | 0.178 | 0.392 | | 0.138 | 0.545 | | 0.152 |
| N | 904.0 | Radium-228 | pCi/L | 1.16 | | 0.381 | 1.12 | | 0.452 | 0.527 | U | 0.439 | 1.04 | | 0.384 | 5.29 | | 2.35 |
| N | 903 and 904 | Radium 226 and 228 | pCi/L | 1.53 | | 0.398 | 1.83 | | 0.487 | 1.26 | | 0.474 | 1.43 | | 0.408 | 5.84 | | 2.35 |
| N | EPA 200.8 | Uranium, Total | mg/L | 0.0002 | J | | 0.0009 | J- | | 0.02 | | | | | | 0.0089 | | |
| Y | EPA 200.8 | Uranium, Dissolved | mg/L | | | | | | | | | | 0.005 | | | | | |

Notes:

EPA = Environmental Protection Agency

HEM = n-Hexane extractable material

J = Estimated qualification based on associated QC data

J+ = Estimated qualification; possibly biased high based on associated QC data

J- = Estimated qualification; possibly biased low based on associated QC data

mg/L = milligrams per liter

pCi/L = picocuries per liter

QC = Quality control

Qual = Qualifier

R = Rejected based on associated QC Data

SM = Standard method

U = Undetected

UJ = Undetected; the reported detection limit is considered estimated

Unc = Uncertainty for Methods 900, 903, and 904


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**Table 2.5-33b Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 1 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-136-U | | | OW-140-U | | | OW-153-L | | | OW-164-M | | | OW-164-U | | |
| | | | **Sample ID:** | OW-136-U | | | OW-140-U | | | OW-153-L | | | OW-164-M | | | OW-164-U | | |
| | | | **Sample Date:** | 7/12/2022 | | | 7/12/2022 | | | 7/18/2022 | | | 7/15/2022 | | | 7/15/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| N | Calculation | Dissolved Organic Nitrogen | mg/L | 1.4 | | | 9.3 | | | 1 | U | | 1 | U | | 1 | U | |
| N | EPA 200.7 | Aluminum, Total | mg/L | 0.5 | U | | | | | | | | 0.2 | | | 0.2 | | |
| N | EPA 200.7 | Arsenic, Total | mg/L | 0.5 | U | | | | | | | | 0.1 | U | | 0.01 | J | |
| N | EPA 200.7 | Barium, Total | mg/L | 0.011 | J | | | | | | | | 0.004 | J | | 0.011 | | |
| N | EPA 200.7 | Calcium, Total | mg/L | 433 | | | | | | | | | 321 | | | 349 | | |
| N | EPA 200.7 | Copper, Total | mg/L | 0.05 | U | | | | | | | | 0.01 | U | | 0.01 | U | |
| N | EPA 200.7 | Iron, Total | mg/L | 0.24 | | | 0.39 | | | 0.15 | J | | 0.93 | | | 1.32 | J | |
| N | EPA 200.7 | Lead, Total | mg/L | 0.2 | U | | | | | | | | 0.04 | U | | 0.04 | U | |
| N | EPA 200.7 | Magnesium, Total | mg/L | 1300 | | | | | | | | | 223 | | | 277 | | |
| N | EPA 200.7 | Manganese, Total | mg/L | 1.49 | | | 0.821 | | | 0.018 | | | 0.111 | | | 0.15 | J | |
| N | EPA 200.7 | Phosphorus, Total as P | mg/L | 0.1 | U | | 0.1 | U | | 0.05 | | | 0.03 | | | 0.04 | J | |
| N | EPA 200.7 | Potassium, Total | mg/L | 35.6 | | | | | | | | | 9.1 | | | 10.8 | | |
| N | EPA 200.7 | Silica, (as $SiO_2$) Total | mg/L | 10.1 | | | 9.7 | | | 7.2 | J- | | 9.8 | | | 11.7 | | |
| N | EPA 200.7 | Sodium, Total | mg/L | 4630 | | | | | | | | | 1870 | | | 1670 | | |
| N | EPA 200.7 | Strontium, Total | mg/L | 11.9 | | | | | | | | | 6.9 | | | 7.79 | | |
| N | EPA 200.7 | Vanadium, Total | mg/L | 0.1 | U | | | | | | | | 0.02 | U | | 0.02 | U | |
| N | EPA 200.7 | Zinc, Total | mg/L | 0.84 | | | | | | | | | 0.01 | J | | 0.02 | | |
| N | EPA 245.1 | Mercury, Total | mg/L | 0.00015 | U | | | | | | | | 0.00015 | U | | 0.00015 | U | |
| N | EPA 300.0 | Fluoride | mg/L | 5 | U | | 5 | U | | 0.59 | | | 0.1 | U | | 0.35 | J | |
| N | EPA 300.0 | Nitrate as N | mg/L | 5 | U | | 5 | U | | 0.1 | U | | 0.1 | U | | 0.1 | U | |
| N | Hach 8000 | Chemical Oxygen Demand | mg/L | 136 | | | 43 | | | 90 | | | 22 | | | 46 | | |
| N | SM 2120 B | Color | Color Units | 50 | | | 15 | | | 5 | | | 5 | | | 15 | | |
| N | SM 2320 B | Alkalinity - Bicarbonate (as $CaCO_3$) | mg/L | 777 | | | 578 | | | 433 | | | 462 | | | 437 | | |
| N | SM 2320 B | Alkalinity - Carbonate (as $CaCO_3$) | mg/L | 1 | U | | 1 | U | | 1.2 | J | | 1 | U | | 1 | U | |
| N | SM 2320 B | Alkalinity - Hydroxide (as $CaCO_3$) | mg/L | 1 | U | | 1 | U | | 1 | U | | 1 | U | | 1 | U | |
| N | SM 2320 B | Alkalinity - Total (as $CaCO_3$) | mg/L | 777 | | | 578 | | | 434 | | | 462 | | | 437 | | |
| N | SM 2340 B | Hardness, Soluble as $CaCO_3$ | mg/L | | | | 3490 | | | 84.7 | | | | | | | | |
| N | SM 2340 B | Hardness, Total as $CaCO_3$ | mg/L | 6450 | | | | | | | | | 1720 | | | 2010 | | |
| N | SM 2540 C | Total Dissolved Solids (TDS) | mg/L | 21600 | | | 14800 | | | 6330 | | | 8970 | | | 8280 | | |
| N | SM 2540 D | Total Suspended Solids (TSS) | mg/L | 11 | | | 17 | | | 10 | | | 8 | | | 6 | | |
| N | SM 4500 NH3 H | Ammonia as N | mg/L | 0.3 | J+ | | 1 | | | 2.4 | | | 2 | | | 0.9 | | |
| N | SM 4500 NH3-D | Ammonia, Dissolved | mg/L | 0.2 | U | | 1 | | | 2.3 | | | 2.2 | | | 0.9 | J | |

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**Table 2.5-33b Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 2 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-136-U | | | OW-140-U | | | OW-153-L | | | OW-164-M | | | OW-164-U | | |
| | | | **Sample ID:** | OW-136-U | | | OW-140-U | | | OW-153-L | | | OW-164-M | | | OW-164-U | | |
| | | | **Sample Date:** | 7/12/2022 | | | 7/12/2022 | | | 7/18/2022 | | | 7/15/2022 | | | 7/15/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| N | SM 4500 NO2-B | Nitrite as N | mg/L | 0.1 | U | | 0.1 | U | | 0.1 | U | | 0.008 | J | | 0.01 | J | |
| N | SM 4500 Norg | Dissolved Kjeldahl Nitrogen | mg/L | 1.5 | | | 1.3 | | | 2.6 | | | 2.5 | | | 1.4 | | |
| N | SM 4500 P-E | Phosphate, ortho as P | mg/L | 0.01 | U | | 0.01 | | | 0.02 | | | 0.01 | | | 0.008 | J | |
| N | SM 4500 S2-D | Sulfide | mg/L | 0.1 | U | | 0.1 | U | | | R | | 0.7 | | | 3.1 | J | |
| N | SM 5210 B | Biochemical Oxygen Demand | mg/L | 5 | U | | 5 | U | | 5 | U | | 6 | | | 13 | J | |
| N | SM 5310 C | Total Organic Carbon | mg/L | 8 | | | 3.4 | | | 0.2 | J- | | 2.1 | | | 4 | | |
| Y | EPA 200.7 | Aluminum, Dissolved | mg/L | | | | 0.2 | J | | 1 | J+ | | | | | | | |
| Y | EPA 200.7 | Arsenic, Dissolved | mg/L | | | | 0.5 | UJ | | 0.5 | U | | | | | | | |
| Y | EPA 200.7 | Barium, Dissolved | mg/L | | | | 0.05 | U | | 0.733 | | | | | | | | |
| Y | EPA 200.7 | Calcium, Dissolved | mg/L | | | | 367 | | | 19 | | | | | | | | |
| Y | EPA 200.7 | Copper, Dissolved | mg/L | | | | 0.05 | U | | 0.05 | U | | | | | | | |
| Y | EPA 200.7 | Iron, Dissolved | mg/L | 0.4 | U | | 0.25 | | | 0.12 | J | | 1.01 | | | 1.15 | | |
| Y | EPA 200.7 | Lead, Dissolved | mg/L | | | | 0.2 | U | | 0.2 | U | | | | | | | |
| Y | EPA 200.7 | Magnesium, Dissolved | mg/L | | | | 626 | | | 9.1 | | | | | | | | |
| Y | EPA 200.7 | Manganese, Dissolved | mg/L | 1.4 | | | 0.779 | | | 0.012 | J | | 0.118 | | | 0.149 | | |
| Y | EPA 200.7 | Potassium, Dissolved | mg/L | | | | 20.5 | | | 10.6 | | | | | | | | |
| Y | EPA 200.7 | Silica, (as $SiO_2$) Dissolved | mg/L | 9.3 | | | 8.3 | | | 7.4 | | | 9.8 | | | 10.6 | | |
| Y | EPA 200.7 | Sodium, Dissolved | mg/L | | | | 3210 | | | 2200 | | | | | | | | |
| Y | EPA 200.7 | Strontium, Dissolved | mg/L | | | | 7.97 | | | 2.14 | | | | | | | | |
| Y | EPA 200.7 | Vanadium, Dissolved | mg/L | | | | 0.1 | U | | 0.1 | U | | | | | | | |
| Y | EPA 200.7 | Zinc, Dissolved | mg/L | | | | 0.1 | U | | 0.05 | J | | | | | | | |
| Y | EPA 245.1 | Mercury, Dissolved | mg/L | | | | 0.00015 | U | | | R | | | | | | | |


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**Table 2.5-33b Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 3 of 3)**

| | | | Location: | OW-136-U | | | OW-140-U | | | OW-153-L | | | OW-164-M | | | OW-164-U | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | Sample ID: | OW-136-U | | | OW-140-U | | | OW-153-L | | | OW-164-M | | | OW-164-U | | |
| | | | Sample Date: | 7/12/2022 | | | 7/12/2022 | | | 7/18/2022 | | | 7/15/2022 | | | 7/15/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| **Radiometric Parameters** | | | | | | | | | | | | | | | | | | |
| N | 900.0 | Gross Alpha | pCi/L | 143 | U | 146 | 57.9 | U | 86.5 | 22.9 | U | 43 | 11.7 | U | 32.5 | 31 | U | 31.5 |
| N | 900.0 | Gross Beta | pCi/L | 73.3 | U | 55.5 | -4.14 | U | 20.2 | 19.4 | U | 15.2 | 3.21 | U | 15.6 | 7.98 | U | 12.5 |
| N | 903.0 | Radium-226 | pCi/L | 1.27 | | 0.236 | 0.792 | | 0.178 | 0.972 | | 0.194 | 0.36 | | 0.125 | 0.356 | | 0.125 |
| N | 904.0 | Radium-228 | pCi/L | 2.87 | | 0.75 | 2.25 | | 0.694 | 6.47 | | 2.66 | 0.911 | | 0.363 | 0.492 | J | 0.271 |
| N | 903 and 904 | Radium 226 and 228 | pCi/L | 4.14 | | 0.786 | 3.05 | | 0.716 | 7.45 | | 2.67 | 1.27 | | 0.384 | 0.848 | J | 0.298 |
| N | EPA 200.8 | Uranium, Total | mg/L | 0.0935 | | | | | | | | | 0.0025 | U | | 0.0005 | J | |
| Y | EPA 200.8 | Uranium, Dissolved | mg/L | | | | 0.0056 | | | 0.0002 | J | | | | | | | |

Notes:

EPA = Environmental Protection Agency

HEM = n-Hexane extractable material

J = Estimated qualification based on associated QC data

J+ = Estimated qualification; possibly biased high based on associated QC data

J- = Estimated qualification; possibly biased low based on associated QC data

mg/L = milligrams per liter

pCi/L = picocuries per liter

QC = Quality control

Qual = Qualifier

R = Rejected based on associated QC Data

SM = Standard method

U = Undetected

UJ = Undetected; the reported detection limit is considered estimated

Unc = Uncertainty for Methods 900, 903, and 904


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**Table 2.5-33c Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 1 of 3)**

| | | | Location: | OW-164-U | | | OW-168-M (A) | | | OW-185-M | | | OW-185-U | | | OW-186-M | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | Sample ID: | Duplicate 1 | | | OW-168-M (A) | | | OW-185-M | | | OW-185-U | | | OW-186-M | | |
| | | | Sample Date: | 7/15/2022 | | | 7/18/2022 | | | 7/14/2022 | | | 7/13/2022 | | | 7/15/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| N | Calculation | Dissolved Organic Nitrogen | mg/L | 1 | U | | 1 | UJ | | 1 | U | | 1 | U | | 1 | U | |
| N | EPA 200.7 | Aluminum, Total | mg/L | 5 | U | | 0.3 | | | | | | 0.2 | | | | | |
| N | EPA 200.7 | Arsenic, Total | mg/L | 5 | U | | 0.02 | J | | | | | 0.05 | U | | | | |
| N | EPA 200.7 | Barium, Total | mg/L | 0.5 | U | | 0.009 | J | | | | | 0.009 | | | | | |
| N | EPA 200.7 | Calcium, Total | mg/L | 372 | | | 355 | | | | | | 377 | | | | | |
| N | EPA 200.7 | Copper, Total | mg/L | 0.5 | U | | 0.01 | U | | | | | 0.005 | U | | | | |
| N | EPA 200.7 | Iron, Total | mg/L | 0.97 | J | | 1.48 | | | 0.1 | | | 2.14 | | | 0.56 | | |
| N | EPA 200.7 | Lead, Total | mg/L | 2 | U | | 0.04 | U | | | | | 0.02 | U | | | | |
| N | EPA 200.7 | Magnesium, Total | mg/L | 289 | | | 262 | | | | | | 662 | | | | | |
| N | EPA 200.7 | Manganese, Total | mg/L | 0.2 | J | | 0.968 | | | 0.25 | U | | 3.15 | | | 0.05 | | |
| N | EPA 200.7 | Phosphorus, Total as P | mg/L | 0.4 | J | | 0.02 | | | 0.05 | | | 0.03 | J+ | | 0.07 | | |
| N | EPA 200.7 | Potassium, Total | mg/L | 11.7 | J | | 10.2 | | | | | | 23.6 | | | | | |
| N | EPA 200.7 | Silica, (as $SiO_2$) Total | mg/L | 12.2 | | | 9.6 | | | 9.3 | | | 12 | | | 15.1 | | |
| N | EPA 200.7 | Sodium, Total | mg/L | 1890 | | | 934 | | | | | | 4070 | | | | | |
| N | EPA 200.7 | Strontium, Total | mg/L | 8.62 | | | 7.83 | | | | | | 8.4 | | | | | |
| N | EPA 200.7 | Vanadium, Total | mg/L | 1 | U | | 0.02 | U | | | | | 0.002 | J | | | | |
| N | EPA 200.7 | Zinc, Total | mg/L | 1 | U | | 0.02 | | | | | | 0.01 | U | | | | |
| N | EPA 245.1 | Mercury, Total | mg/L | 0.00015 | U | | 0.00015 | U | | | | | 0.00015 | U | | | | |
| N | EPA 300.0 | Fluoride | mg/L | 0.82 | J | | 0.13 | J | | 0.68 | | | 0.1 | U | | 4.51 | | |
| N | EPA 300.0 | Nitrate as N | mg/L | 0.1 | U | | 0.1 | UJ | | 5 | U | | 5 | U | | 0.1 | U | |
| N | EPA 447.0M | Chlorophyll a | ug/L | | | | | | | | | | | | | | | |
| N | Hach 8000 | Chemical Oxygen Demand | mg/L | 40 | | | 26 | J | | 39 | | | 63 | | | 46 | | |
| N | SM 2120 B | Color | Color Units | 15 | | | 20 | J | | 5 | | | 15 | | | 15 | | |
| N | SM 2320 B | Alkalinity - Bicarbonate (as $CaCO_3$) | mg/L | 436 | | | 433 | J | | 740 | | | 892 | | | 793 | | |
| N | SM 2320 B | Alkalinity - Carbonate (as $CaCO_3$) | mg/L | 1 | U | | 1 | UJ | | 1 | U | | 1 | U | | 1 | U | |
| N | SM 2320 B | Alkalinity - Hydroxide (as $CaCO_3$) | mg/L | 1 | U | | 1 | UJ | | 1 | U | | 1 | U | | 1 | U | |
| N | SM 2320 B | Alkalinity - Total (as $CaCO_3$) | mg/L | 436 | | | 433 | J | | 740 | | | 892 | | | 793 | | |
| N | SM 2340 B | Hardness, Soluble as $CaCO_3$ | mg/L | | | | | | | 70.3 | | | | | | 531 | | |
| N | SM 2340 B | Hardness, Total as $CaCO_3$ | mg/L | 2120 | | | 1970 | | | | | | 3670 | | | | | |
| N | SM 2540 C | Total Dissolved Solids (TDS) | mg/L | 8280 | | | 5950 | J | | 4420 | | | 17100 | | | 7700 | | |
| N | SM 2540 D | Total Suspended Solids (TSS) | mg/L | 6 | | | 333 | J | | 5 | | | 11 | | | 12 | | |
| N | SM 4500 NH3 H | Ammonia as N | mg/L | 0.9 | | | 0.6 | J | | 1.4 | | | 1.4 | | | 3.1 | | |


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**Table 2.5-33c Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 2 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-164-U | | | OW-168-M (A) | | | OW-185-M | | | OW-185-U | | | OW-186-M | | |
| | | | **Sample ID:** | Duplicate 1 | | | OW-168-M (A) | | | OW-185-M | | | OW-185-U | | | OW-186-M | | |
| | | | **Sample Date:** | 7/15/2022 | | | 7/18/2022 | | | 7/14/2022 | | | 7/13/2022 | | | 7/15/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| N | SM 4500 NH3-D | Ammonia, Dissolved | mg/L | 1.5 | J | | 0.8 | J | | 1.4 | | | 1.3 | | | 3 | | |
| N | SM 4500 NO2-B | Nitrite as N | mg/L | 0.01 | J | | 0.1 | UJ | | 0.1 | U | | 0.1 | U | | 0.02 | J | |
| N | SM 4500 Norg | Dissolved Kjeldahl Nitrogen | mg/L | 1.4 | | | 1.1 | J | | 1.5 | | | 1.8 | | | 3.1 | | |
| N | SM 4500 P-E | Phosphate, ortho as P | mg/L | 0.009 | J | | 0.01 | UJ | | 0.02 | | | 0.04 | | | 0.02 | | |
| N | SM 4500 S2-D | Sulfide | mg/L | 2.1 | J | | | R | | 0.1 | U | | 0.1 | U | | 3.9 | | |
| N | SM 5210 B | Biochemical Oxygen Demand | mg/L | 17 | J | | 5 | UJ | | 4 | J | | 5 | U | | 23 | | |
| N | SM 5310 C | Total Organic Carbon | mg/L | 4 | | | 1.8 | J- | | 1.1 | | | 5 | | | 4.4 | | |
| Y | EPA 200.7 | Aluminum, Dissolved | mg/L | | | | | | | 0.2 | U | | | | | 0.2 | J | |
| Y | EPA 200.7 | Arsenic, Dissolved | mg/L | | | | | | | 0.25 | U | | | | | 0.5 | U | |
| Y | EPA 200.7 | Barium, Dissolved | mg/L | | | | | | | 0.096 | | | | | | 0.05 | U | |
| Y | EPA 200.7 | Calcium, Dissolved | mg/L | | | | | | | 13.8 | | | | | | 119 | | |
| Y | EPA 200.7 | Copper, Dissolved | mg/L | | | | | | | 0.025 | U | | | | | 0.05 | U | |
| Y | EPA 200.7 | Iron, Dissolved | mg/L | 1.14 | | | 1.59 | | | 0.02 | J | | 5.02 | | | 0.24 | | |
| Y | EPA 200.7 | Lead, Dissolved | mg/L | | | | | | | 0.1 | U | | | | | 0.2 | U | |
| Y | EPA 200.7 | Magnesium, Dissolved | mg/L | | | | | | | 8.7 | | | | | | 56.7 | | |
| Y | EPA 200.7 | Manganese, Dissolved | mg/L | 0.148 | | | 1.03 | | | 0.013 | J | | 2.18 | | | 0.05 | U | |
| Y | EPA 200.7 | Potassium, Dissolved | mg/L | | | | | | | 7.4 | | | | | | 6.5 | | |
| Y | EPA 200.7 | Silica, (as $SiO_2$) Dissolved | mg/L | 10.8 | | | 9 | | | 7.8 | | | 11.1 | | | 11.1 | | |
| Y | EPA 200.7 | Sodium, Dissolved | mg/L | | | | | | | 1530 | | | | | | 2120 | | |
| Y | EPA 200.7 | Strontium, Dissolved | mg/L | | | | | | | 1.32 | | | | | | 7.45 | | |
| Y | EPA 200.7 | Vanadium, Dissolved | mg/L | | | | | | | 0.02 | J | | | | | 0.1 | U | |
| Y | EPA 200.7 | Zinc, Dissolved | mg/L | | | | | | | 0.05 | U | | | | | 0.1 | U | |
| Y | EPA 245.1 | Mercury, Dissolved | mg/L | | | | | | | 0.00015 | U | | | | | 0.00015 | U | |


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**Table 2.5-33c Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 3 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-164-U | | | OW-168-M (A) | | | OW-185-M | | | OW-185-U | | | OW-186-M | | |
| | | | **Sample ID:** | Duplicate 1 | | | OW-168-M (A) | | | OW-185-M | | | OW-185-U | | | OW-186-M | | |
| | | | **Sample Date:** | 7/15/2022 | | | 7/18/2022 | | | 7/14/2022 | | | 7/13/2022 | | | 7/15/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| **Radiometric Parameters** | | | | | | | | | | | | | | | | | | |
| N | 900.0 | Gross Alpha | pCi/L | 5.91 | U | 40.5 | -2.89 | U | 26.5 | -14.7 | U | 15.2 | 83 | U | 108 | 13.8 | U | 34.2 |
| N | 900.0 | Gross Beta | pCi/L | -7.16 | U | 13.1 | 8.51 | U | 10.6 | 10.7 | U | 7.09 | -5.96 | U | 32.9 | 29.6 | | 17.5 |
| N | 903.0 | Radium-226 | pCi/L | 0.404 | | 0.146 | 0.726 | | 0.222 | 0.0934 | U | 0.088 | 0.548 | | 0.167 | 0.213 | | 0.101 |
| N | 904.0 | Radium-228 | pCi/L | 0.97 | J | 0.37 | 1.59 | | 0.583 | 0.243 | U | 0.322 | 1.89 | | 0.629 | 0.658 | | 0.321 |
| N | 903 and 904 | Radium 226 and 228 | pCi/L | 1.37 | J | 0.398 | 2.31 | | 0.624 | 0.336 | U | 0.334 | 2.44 | | 0.651 | 0.871 | | 0.337 |
| N | EPA 200.8 | Uranium, Total | mg/L | 0.0005 | J | | 0.0116 | | | | | | 0.0337 | | | | | |
| Y | EPA 200.8 | Uranium, Dissolved | mg/L | | | | | | | 0.0018 | J | | | | | 0.0002 | J | |

Notes:

EPA = Environmental Protection Agency

HEM = n-Hexane extractable material

J = Estimated qualification based on associated QC data

J+ = Estimated qualification; possibly biased high based on associated QC data

J- = Estimated qualification; possibly biased low based on associated QC data

mg/L = milligrams per liter

pCi/L = picocuries per liter

QC = Quality control

Qual = Qualifier

R = Rejected based on associated QC Data

SM = Standard method

U = Undetected

UJ = Undetected; the reported detection limit is considered estimated

Unc = Uncertainty for Methods 900, 903, and 904


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**Table 2.5-33d Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 1 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-187-M | | | OW-187-U | | | OW-188-M | | | OW-188-U | | | OW-189-U | | |
| | | | **Sample ID:** | OW-187-M | | | OW-187-U | | | OW-188-M | | | OW-188-U | | | OW-189-U | | |
| | | | **Sample Date:** | 7/13/2022 | | | 7/11/2022 | | | 7/14/2022 | | | 7/13/2022 | | | 7/14/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| N | Calculation | Dissolved Organic Nitrogen | mg/L | 1.1 | | | 1.6 | | | 1 | U | | 1 | U | | 1.7 | | |
| N | EPA 200.7 | Aluminum, Total | mg/L | 2 | | | 0.1 | J+ | | 0.3 | | | 0.6 | | | 2.5 | U | |
| N | EPA 200.7 | Arsenic, Total | mg/L | 0.05 | U | | 0.006 | J | | 2.5 | UJ | | 0.006 | J | | 2.5 | U | |
| N | EPA 200.7 | Barium, Total | mg/L | 0.012 | | | 0.008 | | | 0.25 | U | | 0.01 | | | 0.25 | U | |
| N | EPA 200.7 | Calcium, Total | mg/L | 117 | | | 191 | | | 387 | | | 370 | | | 380 | | |
| N | EPA 200.7 | Copper, Total | mg/L | 0.005 | | | 0.005 | U | | 0.025 | U | | 0.072 | | | 0.025 | U | |
| N | EPA 200.7 | Iron, Total | mg/L | 1.29 | | | 1.09 | | | 0.97 | J | | 0.49 | | | 0.47 | | |
| N | EPA 200.7 | Lead, Total | mg/L | 0.02 | U | | 0.02 | U | | 0.02 | J | | 0.02 | U | | 0.1 | U | |
| N | EPA 200.7 | Magnesium, Total | mg/L | 73.3 | | | 186 | | | 389 | | | 878 | | | 603 | | |
| N | EPA 200.7 | Manganese, Total | mg/L | 0.041 | | | 0.065 | | | 0.332 | | | 1.71 | | | 1.72 | | |
| N | EPA 200.7 | Phosphorus, Total as P | mg/L | 0.1 | | | 0.1 | | | 0.07 | | | 0.02 | J+ | | 0.06 | J+ | |
| N | EPA 200.7 | Potassium, Total | mg/L | 10.6 | | | 13.6 | | | 12.9 | | | 38 | | | 23.9 | J | |
| N | EPA 200.7 | Silica, (as $SiO_2$) Total | mg/L | 19.6 | | | 8.5 | | | 8.4 | J- | | 12.1 | | | 9.1 | | |
| N | EPA 200.7 | Sodium, Total | mg/L | 4960 | | | 5040 | | | 2600 | | | 3590 | | | 2540 | | |
| N | EPA 200.7 | Strontium, Total | mg/L | 11.7 | | | 14.4 | | | 8.7 | | | 7.97 | | | 8.37 | | |
| N | EPA 200.7 | Vanadium, Total | mg/L | 0.004 | J | | 0.002 | J | | 0.5 | U | | 0.001 | J | | 0.05 | U | |
| N | EPA 200.7 | Zinc, Total | mg/L | 0.01 | | | 0.01 | U | | 0.5 | U | | 0.01 | U | | 0.5 | U | |
| N | EPA 245.1 | Mercury, Total | mg/L | 0.00015 | U | | 0.00015 | U | | 0.00015 | U | | 0.00015 | U | | 0.00015 | U | |
| N | EPA 300.0 | Fluoride | mg/L | 0.1 | U | | 0.1 | U | | 0.1 | U | | 0.1 | UJ | | 0.74 | | |
| N | EPA 300.0 | Nitrate as N | mg/L | 5 | U | | 5 | UJ | | 5 | UJ | | 5 | UJ | | 5 | U | |
| N | Hach 8000 | Chemical Oxygen Demand | mg/L | 83 | | | 73 | | | 24 | | | 44 | | | 67 | | |
| N | SM 2120 B | Color | Color Units | 25 | | | 20 | | | 10 | | | 25 | | | 20 | | |
| N | SM 2320 B | Alkalinity - Bicarbonate (as $CaCO_3$) | mg/L | 695 | | | 722 | | | 480 | | | 583 | | | 396 | | |
| N | SM 2320 B | Alkalinity - Carbonate (as $CaCO_3$) | mg/L | 1 | U | | 1 | U | | 1 | U | | 1 | U | | 1 | U | |
| N | SM 2320 B | Alkalinity - Hydroxide (as $CaCO_3$) | mg/L | 1 | U | | 1 | U | | 1 | U | | 1 | U | | 1 | U | |
| N | SM 2320 B | Alkalinity - Total (as $CaCO_3$) | mg/L | 695 | | | 722 | | | 480 | | | 583 | | | 396 | | |
| N | SM 2340 B | Hardness, Total as $CaCO_3$ | mg/L | 593 | | | 1240 | | | 2570 | | | 4540 | | | 3430 | | |
| N | SM 2540 C | Total Dissolved Solids (TDS) | mg/L | 15000 | | | 16700 | | | 12100 | | | 16800 | | | 12200 | | |


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**Table 2.5-33d Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 2 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-187-M | | | OW-187-U | | | OW-188-M | | | OW-188-U | | | OW-189-U | | |
| | | | **Sample ID:** | OW-187-M | | | OW-187-U | | | OW-188-M | | | OW-188-U | | | OW-189-U | | |
| | | | **Sample Date:** | 7/13/2022 | | | 7/11/2022 | | | 7/14/2022 | | | 7/13/2022 | | | 7/14/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |
| N | SM 2540 D | Total Suspended Solids (TSS) | mg/L | 50 | | | 3 | J | | 7 | J | | 18 | | | 7 | | |
| N | SM 4500 NH3 H | Ammonia as N | mg/L | 6.1 | | | 7 | | | 2.5 | | | 0.3 | J+ | | 0.5 | J+ | |
| N | SM 4500 NH3-D | Ammonia, Dissolved | mg/L | 5.7 | | | 6.5 | | | 2.5 | | | 0.2 | U | | 0.3 | J+ | |
| N | SM 4500 NO2-B | Nitrite as N | mg/L | 0.1 | U | | 0.1 | U | | 0.1 | U | | 0.1 | U | | 0.1 | U | |
| N | SM 4500 Norg | Dissolved Kjeldahl Nitrogen | mg/L | 6.8 | | | 8.1 | | | 3 | | | 0.7 | J | | 2 | | |
| N | SM 4500 P-E | Phosphate, ortho as P | mg/L | 0.01 | | | 0.01 | U | | 0.01 | U | | 0.005 | J | | 0.01 | U | |
| N | SM 4500 S2-D | Sulfide | mg/L | 0.1 | U | | 0.1 | UJ | | 0.1 | U | | 0.1 | U | | 0.1 | U | |
| N | SM 5210 B | Biochemical Oxygen Demand | mg/L | 11 | | | 5 | U | | 4 | J | | 28 | J+ | | 6 | | |
| N | SM 5310 C | Total Organic Carbon | mg/L | 4.1 | | | 4 | J- | | 2.6 | J | | 4.5 | | | 4.4 | | |
| Y | EPA 200.7 | Iron, Dissolved | mg/L | 0.55 | | | 1.15 | | | 2.42 | | | 0.2 | U | | 0.57 | | |
| Y | EPA 200.7 | Manganese, Dissolved | mg/L | 0.029 | J | | 0.066 | | | 0.336 | | | 1.69 | | | 1.66 | | |
| Y | EPA 200.7 | Silica, (as $SiO_2$) Dissolved | mg/L | 7.9 | | | 7.9 | | | 8.4 | | | 9 | | | 9 | | |
| **Radiometric Parameters** | | | | | | | | | | | | | | | | | | |
| N | 900.0 | Gross Alpha | pCi/L | 55.2 | U | 73.6 | 28.3 | U | 63.6 | -16.8 | U | 42.4 | 82.4 | U | 106 | 25.4 | U | 59.1 |
| N | 900.0 | Gross Beta | pCi/L | -29.5 | U | 26.2 | 63.2 | | 31.8 | 7.62 | U | 19.1 | 30.8 | U | 32.6 | 24 | U | 20.5 |
| N | 903.0 | Radium-226 | pCi/L | 0.448 | | 0.142 | 0.577 | | 0.155 | 0.135 | | 0.096 | 0.711 | | 0.181 | 0.658 | | 0.189 |
| N | 904.0 | Radium-228 | pCi/L | 1.98 | | 0.828 | 1 | | 0.388 | 0.621 | | 0.363 | 2.24 | | 0.764 | 1.62 | | 0.566 |
| N | 903 and 904 | Radium 226 and 228 | pCi/L | 2.42 | | 0.84 | 1.58 | | 0.418 | 0.756 | | 0.375 | 2.95 | | 0.785 | 2.28 | | 0.597 |
| N | EPA 200.8 | Uranium, Total | mg/L | 0.0006 | J | | 0.0015 | | | 0.0148 | | | 0.0435 | | | 0.0238 | | |



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**Table 2.5-33d Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 3 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-187-M | | | OW-187-U | | | OW-188-M | | | OW-188-U | | | OW-189-U | | |
| | | | **Sample ID:** | OW-187-M | | | OW-187-U | | | OW-188-M | | | OW-188-U | | | OW-189-U | | |
| | | | **Sample Date:** | 7/13/2022 | | | 7/11/2022 | | | 7/14/2022 | | | 7/13/2022 | | | 7/14/2022 | | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** |

Notes:

EPA = Environmental Protection Agency

HEM = n-Hexane extractable material

J = Estimated qualification based on associated QC data

J+ = Estimated qualification; possibly biased high based on associated QC data

J- = Estimated qualification; possibly biased low based on associated QC data

mg/L = milligrams per liter

pCi/L = picocuries per liter

QC = Quality control

Qual = Qualifier

R = Rejected based on associated QC Data

SM = Standard method

U = Undetected

UJ = Undetected; the reported detection limit is considered estimated

Unc = Uncertainty for Methods 900, 903, and 904


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**Table 2.5-33e Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 1 of 3)**

| | | | Location: | OW-190-M | | | OW-191-M | | | OW-191-U | | | Upstream | | Downstream | | Downstream | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | Sample ID: | OW-190-M | | | OW-191-M | | | OW-191-U | | | Upstream | | Downstream | | Duplicate - 2 | |
| | | | Sample Date: | 7/14/2022 | | | 7/12/2022 | | | 7/11/2022 | | | 7/21/2022 | | 7/21/2022 | | 7/21/2022 | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Result** | **Qual** | **Result** | **Qual** |
| N | Calculation | Dissolved Organic Nitrogen | mg/L | 1 | U | | 1 | U | | 1 | U | | | | | | | |
| N | EPA 1664A | Oil and Grease (HEM) | mg/L | | | | | | | | | | 6 | J+ | 5 | UJ | 5 | J+ |
| N | EPA 200.7 | Aluminum, Total | mg/L | 0.1 | J+ | | 0.5 | U | | 0.1 | J | | | | | | | |
| N | EPA 200.7 | Arsenic, Total | mg/L | 0.01 | J | | 0.5 | U | | 0.25 | U | | | | | | | |
| N | EPA 200.7 | Barium, Total | mg/L | 0.005 | J | | 0.05 | U | | 0.007 | J | | | | | | | |
| N | EPA 200.7 | Calcium, Total | mg/L | 187 | | | 470 | | | 415 | | | | | | | | |
| N | EPA 200.7 | Copper, Total | mg/L | 0.01 | U | | 0.05 | U | | 0.025 | U | | | | | | | |
| N | EPA 200.7 | Iron, Total | mg/L | 1.79 | | | 3.4 | | | 4.57 | | | | | | | | |
| N | EPA 200.7 | Lead, Total | mg/L | 0.04 | U | | 0.2 | U | | 0.1 | U | | | | | | | |
| N | EPA 200.7 | Magnesium, Total | mg/L | 158 | | | 724 | | | 709 | | | | | | | | |
| N | EPA 200.7 | Manganese, Total | mg/L | 0.04 | J+ | | 1.86 | | | 1.67 | | | | | | | | |
| N | EPA 200.7 | Phosphorus, Total as P | mg/L | 0.05 | J+ | | 0.1 | U | | 0.05 | U | | | | | | | |
| N | EPA 200.7 | Potassium, Total | mg/L | 9.5 | | | 25.1 | | | 25.9 | | | | | | | | |
| N | EPA 200.7 | Silica, (as $SiO_2$) Total | mg/L | 8.5 | J- | | 10.2 | | | 9.1 | | | | | | | | |
| N | EPA 200.7 | Sodium, Total | mg/L | 3160 | | | 5060 | | | 4650 | | | | | | | | |
| N | EPA 200.7 | Strontium, Total | mg/L | 6.88 | | | 13.2 | | | 11.7 | | | | | | | | |
| N | EPA 200.7 | Vanadium, Total | mg/L | 0.02 | U | | 0.1 | U | | 0.05 | U | | | | | | | |
| N | EPA 200.7 | Zinc, Total | mg/L | 0.02 | U | | 0.05 | J | | 0.05 | | | | | | | | |
| N | EPA 245.1 | Mercury, Total | mg/L | 0.00015 | U | | 0.00015 | U | | 0.00015 | U | | | | | | | |
| N | EPA 300.0 | Fluoride | mg/L | 0.11 | | | 5 | U | | 0.1 | U | | | | | | | |
| N | EPA 300.0 | Nitrate as N | mg/L | 5 | U | | 5 | U | | 5 | U | | | | | | | |
| N | EPA 447.0M | Chlorophyll a | ug/L | | | | | | | | | | 4.3 | J | 4.2 | J | 11.7 | J |
| N | Hach 8000 | Chemical Oxygen Demand | mg/L | 26 | | | 59 | | | 65 | | | | | | | | |
| N | SM 2120 B | Color | Color Units | 10 | | | 20 | | | 15 | | | | | | | | |
| N | SM 2320 B | Alkalinity - Bicarbonate (as $CaCO_3$) | mg/L | 886 | | | 885 | | | 890 | | | | | | | | |
| N | SM 2320 B | Alkalinity - Carbonate (as $CaCO_3$) | mg/L | 1 | U | | 1 | U | | 1 | U | | | | | | | |
| N | SM 2320 B | Alkalinity - Hydroxide (as $CaCO_3$) | mg/L | 1 | U | | 1 | U | | 1 | U | | | | | | | |
| N | SM 2320 B | Alkalinity - Total (as $CaCO_3$) | mg/L | 886 | | | 885 | | | 890 | | | | | | | | |

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**Table 2.5-33e Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 2 of 3)**

| | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-190-M | | | OW-191-M | | | OW-191-U | | | Upstream | | Downstream | | Downstream | |
| | | | **Sample ID:** | OW-190-M | | | OW-191-M | | | OW-191-U | | | Upstream | | Downstream | | Duplicate - 2 | |
| | | | **Sample Date:** | 7/14/2022 | | | 7/12/2022 | | | 7/11/2022 | | | 7/21/2022 | | 7/21/2022 | | 7/21/2022 | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Result** | **Qual** | **Result** | **Qual** |
| N | SM 2340 B | Hardness, Total as $CaCO_3$ | mg/L | 1120 | | | 4160 | | | 3960 | | | | | | | | |
| N | SM 2540 C | Total Dissolved Solids (TDS) | mg/L | 11700 | | | 19600 | | | 20000 | | | | | | | | |
| N | SM 2540 D | Total Suspended Solids (TSS) | mg/L | 8 | | | 13 | | | 12 | | | | | | | | |
| N | SM 4500 NH3 H | Ammonia as N | mg/L | 5.3 | | | 2.9 | | | 2.2 | | | | | | | | |
| N | SM 4500 NH3-D | Ammonia, Dissolved | mg/L | 5 | | | 2.9 | | | 2.1 | | | | | | | | |
| N | SM 4500 NO2-B | Nitrite as N | mg/L | 0.1 | U | | 0.1 | U | | 0.1 | U | | | | | | | |
| N | SM 4500 Norg | Dissolved Kjeldahl Nitrogen | mg/L | 5.1 | | | 3 | | | 2.6 | | | | | | | | |
| N | SM 4500 P-E | Phosphate, ortho as P | mg/L | 0.01 | U | | 0.02 | | | 0.01 | U | | | | | | | |
| N | SM 4500 S2-D | Sulfide | mg/L | 0.1 | U | | 0.1 | U | | 0.1 | U | | | | | | | |
| N | SM 5210 B | Biochemical Oxygen Demand | mg/L | 5 | U | | 5 | U | | 5 | U | | | | | | | |
| N | SM 5310 C | Total Organic Carbon | mg/L | 1.9 | J+ | | 2.5 | | | 2.7 | | | | | | | | |
| Y | EPA 200.7 | Iron, Dissolved | mg/L | 2.09 | | | 4.95 | | | 4.74 | | | | | | | | |
| Y | EPA 200.7 | Manganese, Dissolved | mg/L | 0.05 | | | 1.69 | | | 1.61 | | | | | | | | |
| Y | EPA 200.7 | Potassium, Dissolved | mg/L | | | | | | | | | | | | | | | |
| Y | EPA 200.7 | Silica, (as $SiO_2$) Dissolved | mg/L | 9.2 | | | 9.1 | | | 137 | | | | | | | | |
| **Radiometric Parameters** | | | | | | | | | | | | | | | | | | |
| N | 900.0 | Gross Alpha | pCi/L | 25.9 | U | 71.2 | 14.7 | U | 96.4 | 71 | U | 78.6 | | | | | | |
| N | 900.0 | Gross Beta | pCi/L | 2.97 | U | 19.4 | 2.47 | U | 29.5 | 51.5 | U | 36.1 | | | | | | |
| N | 903.0 | Radium-226 | pCi/L | 0.41 | | 0.152 | 0.402 | | 0.135 | 0.542 | | 0.146 | | | | | | |
| N | 904.0 | Radium-228 | pCi/L | 1.11 | | 0.417 | 1.02 | | 0.422 | 2.42 | | 0.724 | | | | | | |
| N | 903 and 904 | Radium 226 and 228 | pCi/L | 1.52 | | 0.444 | 1.42 | | 0.443 | 2.97 | | 0.739 | | | | | | |
| N | EPA 200.8 | Uranium, Total | mg/L | 0.0025 | U | | 0.0097 | | | 0.0098 | | | | | | | | |


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**Table 2.5-33e Geochemical, Environmental, and Radiometric Analytical Results**
**(Sheet 3 of 3)**

| | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | **Location:** | OW-190-M | | | OW-191-M | | | OW-191-U | | | Upstream | | Downstream | | Downstream | |
| | | | **Sample ID:** | OW-190-M | | | OW-191-M | | | OW-191-U | | | Upstream | | Downstream | | Duplicate - 2 | |
| | | | **Sample Date:** | 7/14/2022 | | | 7/12/2022 | | | 7/11/2022 | | | 7/21/2022 | | 7/21/2022 | | 7/21/2022 | |
| **Field Filtered (Y/N)** | **Method** | **Parameter** | **Units** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Unc** | **Result** | **Qual** | **Result** | **Qual** | **Result** | **Qual** |

Notes:

EPA = Environmental Protection Agency

HEM = n-Hexane extractable material

J = Estimated qualification based on associated QC data

J+ = Estimated qualification; possibly biased high based on associated QC data

J- = Estimated qualification; possibly biased low based on associated QC data

mg/L = milligrams per liter

pCi/L = picocuries per liter

QC = Quality control

Qual = Qualifier

R = Rejected based on associated QC Data

SM = Standard method

U = Undetected

UJ = Undetected; the reported detection limit is considered estimated

Unc = Uncertainty for Methods 900, 903, and 904


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**Table 2.5-34 Chloride, Sulfate, and pH Analytical Results**

| | | Location: | OW-117-U | OW-119-M | OW-121-U | OW-122-M | OW-122-U | OW-136-U | OW-140-U | OW-153-L |
|---|---|---|---|---|---|---|---|---|---|---|
| | | Sample Date: | 7/8/2022 | 7/13/2022 | 7/12/2022 | 7/15/2022 | 7/18/2022 | 7/12/2022 | 7/12/2022 | 7/18/2022 |
| **Method** | **Parameter** | **Units** | Result | Result | Result | Result | Result | Result | Result | Result |
| EPA 300.0 | Chloride | mg/L | 1500 | 500 | 390 | 420 | 520 | 2800 | 1400 | 3800 |
| EPA 300.0 | Sulfate | mg/L | 3900 | 4800 | 4400 | 4600 | 4900 | 12000 | 9100 | 3.6 |
| Field Probe | pH | Std Units | 7.37 | 6.89 | 7.02 | 6.96 | 6.96 | 6.88 | 7.01 | 8.62 |

| | | Location: | OW-164-M | OW-164-U | OW-164-U | OW-168-M(A) | OW-185-M | OW-185-U | OW-186-M | OW-187-M |
|---|---|---|---|---|---|---|---|---|---|---|
| | | Sample Date: | 7/15/2022 | 7/15/2022 | 7/15/2022 | 7/18/2022 | 7/14/2022 | 7/13/2022 | 7/15/2022 | 7/13/2022 |
| **Method** | **Parameter** | **Units** | Result | Result | Result | Result | Result | Result | Result | Result |
| EPA 300.0 | Chloride | mg/L | 510 | 480 | 480 | 440 | 1800 | 890 | 410 | 2100 |
| EPA 300.0 | Sulfate | mg/L | 5400 | 4900 | 5300 | 3900 | 670 | 11000 | 4400 | 8000 |
| Field Probe | pH | Std Units | 7.10 | 7.10 | 7.10 | 6.94 | 8.53 | 6.85 | 7.27 | 7.49 |

| | | Location: | OW-187-U | OW-188-M | OW-188-U | OW-189-U | OW-190-M | OW-191-M | OW-191-U |
|---|---|---|---|---|---|---|---|---|---|
| | | Sample Date: | 7/11/2022 | 7/14/2022 | 7/13/2022 | 7/14/2022 | 7/14/2022 | 7/12/2022 | 7/11/2022 |
| **Method** | **Parameter** | Units | Result | Result | Result | Result | Result | Result | Result |
| EPA 300.0 | Chloride | mg/L | 2200 | 640 | 510 | 1000 | 670 | 3200 | 3100 |
| EPA 300.0 | Sulfate | mg/L | 8900 | 7700 | 11000 | 7200 | 6900 | 9900 | 10000 |
| Field Probe | pH | Std Units | 7.27 | 7.04 | 7.00 | 7.29 | 7.03 | 6.91 | 6.77 |

Notes:

mg/L = milligrams per liter

J = Estimated qualification based on associated QC data


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**Table 2.5-35 Construction Information for Observation Wells**

| Well Number | Well Construction | | | | | | | As Built Coordinates | |
|---|---|---|---|---|---|---|---|---|---|
| | Riser / Screen Diameter (in) | Elevation Top of Casing (ft)(1) | Elevation waterd Surface (ft)(1) | Depth of Well (feet below ground surface)(2) | Depth to Top of Screen (ft)(2) | Elevation Top of Screen (ft) | Elevation Bottom of Screen (ft) | Northing (ft) | Easting (ft) |
| OW-106-M | 4 | 6745.45 | 6742.8 | 60.4 | 50.0 | 6692.8 | 6682.8 | 768008.9 | 2494157.6 |
| OW-106-U | 4 | 6745.45 | 6742.8 | 41.2 | 30.8 | 6712.0 | 6702.0 | 768019.1 | 2494151.3 |
| OW-108-M | 4 | 6759.21 | 6756.6 | 80.4 | 70.0 | 6686.6 | 6676.6 | 768016.4 | 2494537.9 |
| OW-108-U | 4 | 6759.58 | 6757.0 | 40.4 | 30.0 | 6727.0 | 6717.0 | 768025.5 | 2494530.6 |
| OW-109-M | 4 | 6758.02 | 6755.5 | 65.4 | 55.0 | 6700.5 | 6690.5 | 767974.9 | 2494400.0 |
| OW-109-U | 4 | 6758.37 | 6755.8 | 40.4 | 30.0 | 6725.8 | 6715.8 | 767986.2 | 2494405.1 |
| OW-117-L | 4 | 6742.64 | 6740.2 | 115.7 | 105.3 | 6634.9 | 6624.9 | 767768.7 | 2494185.2 |
| OW-117-M | 4 | 6743.70 | 6740.9 | 80.5 | 70.1 | 6670.8 | 6660.8 | 767780.9 | 2494194.6 |
| OW-117-U | 4 | 6743.14 | 6740.5 | 40.6 | 30.2 | 6710.3 | 6700.3 | 767789.5 | 2494185.2 |
| OW-119-M | 4 | 6753.82 | 6751.3 | 46.1 | 35.7 | 6715.6 | 6705.6 | 767735.0 | 2494314.7 |
| OW-119-U | 4 | 6753.74 | 6751.3 | 30.4 | 20.0 | 6731.3 | 6721.3 | 767724.1 | 2494314.9 |
| OW-121-M(A) | 4 | 6760.55 | 6757.9 | 65.4 | 55.0 | 6702.9 | 6692.9 | 767811.6 | 2494419.3 |
| OW-121-U | 4 | 6760.32 | 6757.9 | 35.0 | 24.6 | 6733.3 | 6723.3 | 767820.7 | 2494415.4 |
| OW-122-L | 4 | 6759.01 | 6756.2 | 110.4 | 100.0 | 6656.2 | 6646.2 | 767762.9 | 2494370.0 |
| OW-122-M | 4 | 6759.10 | 6756.3 | 65.4 | 55.0 | 6701.3 | 6691.3 | 767782.0 | 2494371.4 |
| OW-122-U | 4 | 6759.10 | 6756.3 | 47.1 | 41.7 | 6714.6 | 6709.6 | 767770.5 | 2494380.3 |
| OW-127-M | 4 | 6759.37 | 6756.6 | 55.0 | 44.6 | 6712.0 | 6702.0 | 767807.2 | 2494504.1 |
| OW-127-U | 4 | 6759.32 | 6756.8 | 35.4 | 25.0 | 6731.8 | 6721.8 | 767797.3 | 2494503.3 |
| OW-136-M | 4 | 6740.26 | 6737.6 | 75.5 | 65.1 | 6672.5 | 6662.5 | 767402.4 | 2493834.4 |
| OW-136-U | 4 | 6740.91 | 6737.8 | 33.5 | 23.1 | 6714.7 | 6704.7 | 767412.3 | 2493832.7 |
| OW-137-M | 4 | 6750.89 | 6748.4 | 80.7 | 70.3 | 6678.1 | 6668.1 | 767486.2 | 2494261.0 |
| OW-137-U | 4 | 6751.22 | 6748.1 | 34.6 | 24.2 | 6723.9 | 6713.9 | 767496.4 | 2494263.9 |
| OW-140-M | 4 | 6743.05 | 6740.3 | 65.3 | 54.9 | 6685.4 | 6675.4 | 767831.3 | 2493814.2 |
| OW-140-U | 4 | 6742.63 | 6739.9 | 40.4 | 30.0 | 6709.9 | 6699.9 | 767838.9 | 2493824.3 |
| OW-149-M | 4 | 6759.81 | 6757.4 | 65.4 | 55.0 | 6702.4 | 6692.4 | 767222.0 | 2494414.3 |
| OW-149-U | 4 | 6759.91 | 6757.3 | 40.2 | 29.8 | 6727.5 | 6717.5 | 767210.2 | 2494413.8 |
| OW-153-L | 4 | 6735.17 | 6732.4 | 106.3 | 95.9 | 6636.5 | 6626.5 | 766755.7 | 2493825.3 |
| OW-153-M | 4 | 6735.12 | 6732.3 | 70.9 | 60.5 | 6671.8 | 6661.8 | 766770.3 | 2493824.9 |
| OW-153-U | 4 | 6735.04 | 6732.5 | 40.4 | 30.0 | 6702.5 | 6692.5 | 766771.0 | 2493809.3 |
| OW-164-M | 4 | 6748.59 | 6746.2 | 65.0 | 54.6 | 6691.6 | 6681.6 | 766652.2 | 2494168.2 |
| OW-164-U | 4 | 6748.54 | 6746.0 | 45.4 | 35.0 | 6711.0 | 6701.0 | 766672.2 | 2494170.3 |
| OW-167-L | 4 | 6756.63 | 6754.0 | 150.4 | 140.0 | 6614.0 | 6604.0 | 766884.1 | 2494566.5 |
| OW-167-M | 4 | 6756.50 | 6753.6 | 85.4 | 75.0 | 6678.6 | 6668.6 | 766864.3 | 2494563.4 |
| OW-167-U | 4 | 6756.38 | 6753.6 | 45.0 | 34.6 | 6719.0 | 6709.0 | 766873.0 | 2494574.7 |
| OW-168-M(A) | 4 | 6758.34 | 6756.0 | 55.7 | 45.3 | 6710.7 | 6700.7 | 766911.9 | 2494358.8 |
| OW-168-U | 4 | 6758.51 | 6756.0 | 35.9 | 25.5 | 6730.5 | 6720.5 | 766902.3 | 2494361.7 |
| OW-179-M | 4 | 6754.86 | 6752.4 | 50.4 | 40.0 | 6712.4 | 6702.4 | 766384.7 | 2494356.3 |
| OW-179-U | 4 | 6754.60 | 6752.3 | 35.6 | 25.2 | 6727.1 | 6717.1 | 766401.8 | 2494357.5 |
| OW-185-M | 4 | 6743.85 | 6741.1 | 75.4 | 65.0 | 6676.1 | 6666.1 | 767177.5 | 2495628.0 |
| OW-185-U | 4 | 6743.47 | 6741.1 | 46.4 | 36.0 | 6705.1 | 6695.1 | 767168.8 | 2495618.3 |
| OW-186-M | 4 | 6751.09 | 6748.7 | 65.3 | 54.9 | 6693.8 | 6683.8 | 767146.2 | 2494888.6 |
| OW-186-U | 4 | 6751.03 | 6748.6 | 45.2 | 34.8 | 6713.8 | 6703.8 | 767160.0 | 2494892.8 |
| OW-187-M | 4 | 6735.46 | 6732.7 | 75.4 | 65.0 | 6667.7 | 6657.7 | 767922.5 | 2492457.8 |
| OW-187-U | 4 | 6735.87 | 6733.1 | 59.9 | 49.5 | 6683.6 | 6673.6 | 767914.2 | 2492466.8 |
| OW-188-M | 4 | 6747.48 | 6744.9 | 55.4 | 45.0 | 6699.9 | 6689.9 | 768283.5 | 2495294.3 |
| OW-188-U | 4 | 6747.54 | 6744.9 | 35.4 | 25.0 | 6719.9 | 6709.9 | 768295.5 | 2495285.1 |
| OW-189-M | 4 | 6748.43 | 6746.0 | 65.7 | 55.3 | 6690.7 | 6680.7 | 768821.9 | 2493729.7 |
| OW-189-U | 4 | 6748.97 | 6746.4 | 45.3 | 34.9 | 6711.5 | 6701.5 | 768838.0 | 2493743.5 |
| OW-190-M | 4 | 6732.86 | 6730.3 | 60.7 | 50.3 | 6680.0 | 6670.0 | 766459.3 | 2493061.1 |
| OW-190-U | 4 | 6732.99 | 6730.3 | 45.1 | 34.7 | 6695.6 | 6685.6 | 766470.7 | 2493050.8 |
| OW-191-M | 4 | 6740.56 | 6737.4 | 64.9 | 54.5 | 6682.9 | 6672.9 | 765988.3 | 2495598.6 |
| OW-191-U | 4 | 6740.47 | 6737.6 | 50.4 | 40.0 | 6697.6 | 6687.6 | 765977.3 | 2495603.1 |
| PZSW** | NA | 6738.20 | NA | NA | NA | NA | NA | 767163.1 | 2495771.0 |
| PZGW** | NA | 6736.90 | NA | Estimated 1.5 ft | 4.0 | 6732.90 | 6730.90 | 767161.4 | 2495772.4 |

Notes:

1) Horizontal/Vertical Datum: State Plane Coordinates (Wyoming West [FIPS 4904]) NAD 1983/NAVD 1988.

2) Well bottom and screen depths referenced to ground surface.

3) Piezometer assembly in creek bed estimated at seven feet length


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**Table 2.5-36 Monthly Manual Groundwater Levels**

| Observation Well ID | August 17, 2022 | September 20-21, 2022 | October 26-27, 2022 | November 29-30, 2022 | December 28-29, 2022 | January 24-25, 2023 | February 27-28, 2023 and March 1, 2023 | March 29-30, 2023 | April 25-26, 2023 | Groundwater Elevation Difference between Maximum and Minimum |
|---|---|---|---|---|---|---|---|---|---|---|
| OW-106-M | 6723.67 | 6723.60 | 6723.51 | 6723.45 | 6723.36 | 6723.34 | 6723.28 | 6723.22 | 6723.17 | 0.50 |
| OW-106-U | 6722.20 | 6722.03 | 6721.92 | 6721.82 | 6721.72 | 6721.60 | 6721.52 | 6721.41 | 6721.27 | 0.93 |
| OW-108-M | 6680.83 | 6681.12 | 6681.38 | 6681.66 | 6681.89 | 6682.09 | 6682.37 | 6682.59 | 6682.77 | 1.94 |
| OW-108-U | 6722.98 | 6722.95 | 6722.87 | 6722.85 | 6722.90 | 6722.81 | 6722.83 | 6722.84 | 6722.77 | 0.21 |
| OW-109-M | 6696.82 | 6698.12 | 6699.51 | 6702.77 | 6702.46 | 6704.63 | 6705.23 | 6706.51 | 6708.48 | 11.66 |
| OW-109-U | 6725.28 | 6724.72 | 6724.16 | 6723.78 | 6723.51 | 6723.27 | 6723.07 | 6722.92 | 6722.77 | 2.51 |
| OW-117-L | 6686.27 | 6700.75 | 6710.85 | 6716.76 | 6720.40 | 6723.02 | 6725.51 | 6727.20 | 6728.47 | 42.20 |
| OW-117-M | 6666.73 | 6667.43 | 6668.08 | 6668.56 | 6668.92 | 6669.20 | 6669.51 | 6669.74 | 6669.91 | 3.18 |
| OW-117-U | 6721.44 | 6721.37 | 6720.97 | 6720.77 | 6720.68 | 6720.49 | 6720.38 | 6720.26 | 6720.06 | 1.38 |
| OW-119-M | 6721.62 | 6721.43 | 6721.15 | 6720.99 | 6720.88 | 6720.72 | 6720.62 | 6720.55 | 6720.40 | 1.22 |
| OW-119-U | 6721.51 | 6721.42 | 6721.26 | 6721.16 | 6721.06 | 6721.04 | 6721.02 | 6720.98 | 6721.03 | 0.53 |
| OW-121-M(A) | 6698.28 | 6698.84 | 6699.34 | NM | 6700.14 | 6700.45 | 6700.85 | 6701.17 | 6701.50 | 3.22 |
| OW-121-U | 6724.70 | 6724.22 | 6723.74 | NM | 6723.12 | 6722.90 | 6722.90 | 6722.90 | 6722.88 | 1.82 |
| OW-122-L | 6652.86 | 6653.14 | 6653.27 | 6653.36 | 6653.45 | 6653.53 | 6653.62 | 6653.68 | 6653.73 | 0.87 |
| OW-122-M | 6723.51 | 6723.05 | 6722.63 | 6722.39 | 6722.21 | 6722.00 | 6721.88 | 6721.75 | 6721.59 | 1.92 |
| OW-122-U | 6723.48 | 6723.05 | 6722.73 | 6722.48 | 6722.33 | 6722.13 | 6721.99 | 6721.88 | 6721.76 | 1.72 |
| OW-127-M | 6707.76 | 6707.99 | 6708.18 | 6708.37 | 6708.53 | 6708.67 | 6708.87 | 6709.03 | 6709.15 | 1.39 |
| OW-127-U | 6722.56 | 6722.55 | 6722.49 | 6722.49 | 6722.49 | 6722.40 | 6722.44 | 6722.42 | 6722.35 | 0.21 |
| OW-136-M | 6663.53 | 6663.62 | 6663.69 | 6663.77 | 6663.82 | 6663.85 | 6663.93 | 6663.98 | 6664.02 | 0.49 |
| OW-136-U | 6721.90 | 6721.94 | 6722.07 | 6721.95 | 6721.85 | 6721.49 | 6721.56 | 6721.38 | 6721.23 | 0.84 |
| OW-137-M | 6721.98 | 6721.67 | 6721.39 | 6721.25 | 6721.15 | 6720.98 | 6720.89 | 6720.83 | 6720.67 | 1.31 |
| OW-137-U | 6721.08 | 6720.93 | 6720.74 | 6720.63 | 6720.52 | 6720.35 | 6720.27 | 6720.15 | 6720.02 | 1.06 |
| OW-140-M | 6722.61 | 6722.61 | 6722.70 | 6722.68 | 6722.57 | 6722.13 | 6722.15 | 6722.00 | 6721.82 | 0.88 |
| OW-140-U | 6722.49 | 6722.51 | 6722.58 | 6722.56 | 6722.26 | 6721.96 | 6722.02 | 6721.88 | 6721.70 | 0.88 |
| OW-149-M | 6696.98 | 6697.34 | 6697.67 | 6698.04 | 6698.31 | 6698.59 | 6698.91 | 6699.16 | 6699.37 | 2.39 |
| OW-149-U | 6720.44 | 6720.34 | 6720.24 | 6720.16 | 6720.13 | 6720.07 | 6720.07 | 6720.03 | 6719.97 | 0.47 |
| OW-153-L | 6728.55 | 6732.44 | 6733.96 | 6734.73 | 6734.92 | NM | NM | NM | 6734.27 | 6.37 |
| OW-153-M | 6668.13 | 6668.86 | 6669.61 | 6670.44 | 6671.40 | 6672.77 | 6674.24 | 6675.50 | 6676.61 | 8.48 |
| OW-153-U | 6718.11 | 6720.18 | 6720.69 | 6720.57 | 6720.56 | 6720.63 | 6720.62 | 6720.45 | 6720.35 | 2.58 |
| OW-164-M | 6718.53 | 6718.39 | 6718.30 | 6718.27 | 6718.33 | 6718.27 | 6718.31 | 6718.19 | 6718.11 | 0.42 |
| OW-164-U | 6718.56 | 6718.39 | 6718.33 | 6718.31 | 6718.32 | 6718.29 | 6718.20 | 6718.20 | 6718.11 | 0.45 |
| OW-167-L | 6609.81 | 6609.93 | 6609.93 | 6609.98 | 6609.98 | 6609.99 | 6610.06 | 6610.09 | 6610.12 | 0.31 |
| OW-167-M | 6673.26 | 6673.72 | 6674.12 | 6674.50 | 6674.79 | 6675.05 | 6675.40 | 6675.67 | 6675.90 | 2.64 |
| OW-167-U | 6720.19 | 6720.19 | 6720.20 | 6720.12 | 6720.16 | 6720.02 | 6720.09 | 6720.06 | 6719.99 | 0.21 |
| OW-168-M(A) | 6719.89 | 6719.75 | 6719.64 | 6719.59 | 6719.59 | 6719.51 | 6719.55 | 6719.51 | 6719.46 | 0.43 |
| OW-168-U | 6720.14 | 6720.17 | 6720.16 | 6720.18 | 6720.18 | 6720.16 | 6720.19 | 6720.17 | 6720.18 | 0.05 |
| OW-179-M | 6718.57 | 6718.52 | 6718.38 | 6718.36 | 6718.36 | 6718.34 | 6718.37 | 6718.33 | 6718.30 | 0.27 |
| OW-179-U | 6718.64 | 6718.54 | 6718.45 | 6718.44 | 6718.43 | 6718.40 | 6718.44 | 6718.39 | 6718.35 | 0.29 |
| OW-185-M | 6733.16 | 6733.22 | 6733.50 | 6733.66 | 6733.39 | 6733.28 | 6733.61 | 6733.52 | 6733.64 | 0.50 |
| OW-185-U | 6731.59 | 6731.68 | 6732.12 | 6732.32 | 6732.47 | 6732.62 | 6733.22 | 6733.04 | 6732.99 | 1.63 |
| OW-186-M | 6719.46 | 6719.47 | 6719.49 | 6719.56 | 6719.56 | 6719.47 | 6719.57 | 6719.47 | 6719.42 | 0.15 |
| OW-186-U | 6722.40 | 6722.52 | 6722.55 | 6722.41 | 6722.42 | 6722.34 | 6722.43 | 6722.40 | 6722.26 | 0.29 |
| OW-187-M | 6728.38 | 6728.42 | 6728.58 | 6728.56 | 6728.54 | 6728.36 | 6728.59 | 6728.43 | 6728.48 | 0.23 |
| OW-187-U | 6728.32 | 6728.38 | 6728.56 | 6728.52 | 6728.51 | 6728.38 | 6728.54 | 6728.38 | 6728.46 | 0.24 |
| OW-188-M | 6726.56 | 6726.51 | 6726.62 | 6726.61 | 6726.66 | 6726.55 | 6726.78 | 6726.79 | 6726.69 | 0.28 |
| OW-188-U | 6723.44 | 6723.39 | 6723.45 | 6723.44 | 6723.43 | 6723.40 | 6723.47 | 6723.45 | 6723.34 | 0.13 |
| OW-189-M | 6725.87 | 6725.79 | 6725.70 | 6725.73 | 6725.88 | 6726.04 | 6726.18 | 6726.19 | 6726.19 | 0.49 |
| OW-189-U | 6725.67 | 6725.68 | 6725.57 | 6725.55 | 6725.54 | 6725.46 | 6725.50 | 6725.47 | 6725.30 | 0.38 |
| OW-190-M | 6724.24 | 6724.41 | 6724.38 | 6724.29 | 6724.30 | 6724.14 | 6724.31 | 6724.26 | 6724.18 | 0.27 |
| OW-190-U | 6724.94 | 6724.98 | 6725.14 | 6725.03 | 6725.04 | 6724.77 | 6725.02 | 6724.92 | 6724.85 | 0.37 |
| OW-191-M | 6726.55 | 6726.59 | 6726.84 | 6727.03 | 6727.18 | 6727.17 | 6727.51 | 6727.32 | 6727.24 | 0.96 |
| OW-191-U | 6726.53 | 6726.57 | 6726.81 | 6726.98 | 6727.12 | 6727.12 | 6727.45 | 6727.26 | 6727.18 | 0.92 |

Note:

1) Groundwater level in NAVD 1988, feet.

Observation well still recovering after well development.

Maximum groundwater level for each observation well.

NM = Water level not measured.


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 1 of 9)**

| August 17, 2022 | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-97 | Easterly (OW-187-U to OW-137-U) | 6728.32 | 6721.08 | 7.24 | 1845.04 | 0.004 |
| | | Westerly (OW-185-U to OW-186-U) | 6731.59 | 6722.40 | 9.19 | 725.64 | 0.013 |
| | | Southerly (OW-106-U to OW-164-U) | 6722.20 | 6718.56 | 3.64 | 1347.05 | 0.003 |
| Middle (M) well series | Figure 2.5-98 | Easterly (OW-187-M to OW-137-M) | 6728.38 | 6721.98 | 6.40 | 1855.17 | 0.003 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.16 | 6719.46 | 13.70 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.67 | 6718.53 | 5.14 | 1356.75 | 0.004 |


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 2 of 9)**

| **September 20-21, 2022** | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-99 | Easterly (OW-187-U to OW-137-U) | 6728.38 | 6720.93 | 7.45 | 1845.04 | 0.004 |
| | | Westerly (OW-185-U to OW-186-U) | 6731.68 | 6722.52 | 9.16 | 725.64 | 0.013 |
| | | Southerly (OW-106-U to OW-164-U) | 6722.03 | 6718.39 | 3.64 | 1347.05 | 0.003 |
| Middle (M) well series | Figure 2.5-100 | Easterly (OW-187-M to OW-137-M) | 6728.42 | 6721.67 | 6.75 | 1855.17 | 0.004 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.22 | 6719.47 | 13.75 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.60 | 6718.39 | 5.21 | 1356.75 | 0.004 |


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 3 of 9)**

| October 26-27, 2022 | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-101 | Easterly (OW-187-U to OW-137-U) | 6728.56 | 6720.74 | 7.82 | 1845.04 | 0.004 |
| | | Westerly (OW-185-U to OW-186-U) | 6732.12 | 6722.55 | 9.57 | 725.64 | 0.013 |
| | | Southerly (OW-106-U to OW-164-U) | 6721.92 | 6718.33 | 3.59 | 1347.05 | 0.003 |
| Middle (M) well series | Figure 2.5-102 | Easterly (OW-187-M to OW-137-M) | 6728.58 | 6721.39 | 7.19 | 1855.17 | 0.004 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.50 | 6719.49 | 14.01 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.51 | 6718.30 | 5.21 | 1356.75 | 0.004 |


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 4 of 9)**

| November 29-30, 2022 | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-103 | Easterly (OW-187-U to OW-137-U) | 6728.52 | 6720.63 | 7.89 | 1845.04 | 0.004 |
| | | Westerly (OW-185-U to OW-186-U) | 6732.32 | 6722.41 | 9.91 | 725.64 | 0.014 |
| | | Southerly (OW-106-U to OW-164-U) | 6721.82 | 6718.31 | 3.51 | 1347.05 | 0.003 |
| Middle (M) well series | Figure 2.5-104 | Easterly (OW-187-M to OW-137-M) | 6728.56 | 6721.25 | 7.31 | 1855.17 | 0.004 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.66 | 6719.56 | 14.10 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.45 | 6718.27 | 5.18 | 1356.75 | 0.004 |


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 5 of 9)**

| December 28-29, 2022 | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-105 | Easterly (OW-187-U to OW-137-U) | 6728.51 | 6720.52 | 7.99 | 1845.04 | 0.004 |
| | | Westerly (OW-185-U to OW-186-U) | 6732.47 | 6722.42 | 10.05 | 725.64 | 0.014 |
| | | Southerly (OW-106-U to OW-164-U) | 6721.72 | 6718.32 | 3.40 | 1347.05 | 0.003 |
| Middle (M) well series | Figure 2.5-106 | Easterly (OW-187-M to OW-137-M) | 6728.54 | 6721.15 | 7.39 | 1855.17 | 0.004 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.39 | 6719.56 | 13.83 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.36 | 6718.33 | 5.03 | 1356.75 | 0.004 |


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 6 of 9)**

| **January 24-25, 2023** | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-107 | Easterly (OW-187-U to OW-137-U) | 6728.38 | 6720.35 | 8.03 | 1845.04 | 0.004 |
| | | Westerly (OW-185-U to OW-186-U) | 6732.62 | 6722.34 | 10.28 | 725.64 | 0.014 |
| | | Southerly (OW-106-U to OW-164-U) | 6721.60 | 6718.29 | 3.31 | 1347.05 | 0.002 |
| Middle (M) well series | Figure 2.5-108 | Easterly (OW-187-M to OW-137-M) | 6728.36 | 6720.98 | 7.38 | 1855.17 | 0.004 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.28 | 6719.47 | 13.81 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.34 | 6718.27 | 5.07 | 1356.75 | 0.004 |


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 7 of 9)**

| February 27-28 and March 1, 2023 | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-109 | Easterly (OW-187-U to OW-137-U) | 6728.54 | 6720.27 | 8.27 | 1845.04 | 0.004 |
| | | Westerly (OW-185-U to OW-186-U) | 6733.22 | 6722.43 | 10.79 | 725.64 | 0.015 |
| | | Southerly (OW-106-U to OW-164-U) | 6721.52 | 6718.20 | 3.32 | 1347.05 | 0.002 |
| Middle (M) well series | Figure 2.5-110 | Easterly (OW-187-M to OW-137-M) | 6728.59 | 6720.89 | 7.70 | 1855.17 | 0.004 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.61 | 6719.57 | 14.04 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.28 | 6718.31 | 4.97 | 1356.75 | 0.004 |


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 8 of 9)**

| **March 29-30, 2023** | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-111 | Easterly (OW-187-U to OW-137-U) | 6728.38 | 6720.15 | 8.23 | 1845.04 | 0.004 |
| | | Westerly (OW-185-U to OW-186-U) | 6733.04 | 6722.40 | 10.64 | 725.64 | 0.015 |
| | | Southerly (OW-106-U to OW-164-U) | 6721.41 | 6718.20 | 3.21 | 1347.05 | 0.002 |
| Middle (M) well series | Figure 2.5-112 | Easterly (OW-187-M to OW-137-M) | 6728.43 | 6720.83 | 7.60 | 1855.17 | 0.004 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.52 | 6719.47 | 14.05 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.22 | 6718.19 | 5.03 | 1356.75 | 0.004 |


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**Table 2.5-37 Monthly Horizontal Hydraulic Gradients**
**(Sheet 9 of 9)**

| **April 25-26, 2023** | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Well series** | **Associated Figure** | **Flow Path** | **Groundwater elevation at upgradient location 1 ($h_1$) (ft NAVD 88)** | **Groundwater elevation at downgradient location 2 ($h_2$) (ft NAVD 88)** | **Head Difference (Δh) (ft)** | **Distance between location 1 and 2 along the flow path (L) (ft)** | **Horizontal Hydraulic Gradient ($i_h$)** |
| Upper (U) well series | Figure 2.5-113 | Easterly (OW-187-U to OW-137-U) | 6728.46 | 6720.02 | 8.44 | 1845.04 | 0.005 |
| | | Westerly (OW-185-U to OW-186-U) | 6732.99 | 6722.26 | 10.73 | 725.64 | 0.015 |
| | | Southerly (OW-106-U to OW-164-U) | 6721.27 | 6718.11 | 3.16 | 1347.05 | 0.002 |
| Middle (M) well series | Figure 2.5-114 | Easterly (OW-187-M to OW-137-M) | 6728.48 | 6720.67 | 7.81 | 1855.17 | 0.004 |
| | | Westerly (OW-185-M to OW-186-M) | 6733.64 | 6719.42 | 14.22 | 740.11 | 0.019 |
| | | Southerly (OW-106-M to OW-164-M) | 6723.17 | 6718.11 | 5.06 | 1356.75 | 0.004 |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 1 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-106-U/OW-106-M | 8/17/2022 | 6712.80 | 6702.80 | 6707.80 | 6722.20 | 6692.80 | 6682.80 | 6687.80 | 6723.67 | -- | -- | -- | -- | 20.0 | -1.48 | -0.0738 | **-0.0843** |
| | 9/20/2022 | 6712.80 | 6702.80 | 6707.80 | 6722.03 | 6692.80 | 6682.80 | 6687.80 | 6723.60 | -- | -- | -- | -- | 20.0 | -1.58 | -0.0788 | |
| | 10/27/2022 | 6712.80 | 6702.80 | 6707.80 | 6721.92 | 6692.80 | 6682.80 | 6687.80 | 6723.51 | -- | -- | -- | -- | 20.0 | -1.60 | -0.0798 | |
| | 11/29/2022 | 6712.80 | 6702.80 | 6707.80 | 6721.82 | 6692.80 | 6682.80 | 6687.80 | 6723.45 | -- | -- | -- | -- | 20.0 | -1.64 | -0.0818 | |
| | 12/28/2022 | 6712.80 | 6702.80 | 6707.80 | 6721.72 | 6692.80 | 6682.80 | 6687.80 | 6723.36 | -- | -- | -- | -- | 20.0 | -1.65 | -0.0823 | |
| | 1/24/2023 | 6712.80 | 6702.80 | 6707.80 | 6721.60 | 6692.80 | 6682.80 | 6687.80 | 6723.34 | -- | -- | -- | -- | 20.0 | -1.75 | -0.0873 | |
| | 2/27/2023 | 6712.80 | 6702.80 | 6707.80 | 6721.52 | 6692.80 | 6682.80 | 6687.80 | 6723.28 | -- | -- | -- | -- | 20.0 | -1.77 | -0.0883 | |
| | 3/30/2023 | 6712.80 | 6702.80 | 6707.80 | 6721.41 | 6692.80 | 6682.80 | 6687.80 | 6723.22 | -- | -- | -- | -- | 20.0 | -1.82 | -0.0908 | |
| | 4/26/2023 | 6712.80 | 6702.80 | 6707.80 | 6721.27 | 6692.80 | 6682.80 | 6687.80 | 6723.17 | -- | -- | -- | -- | 20.0 | -1.91 | -0.0953 | |
| OW-108-U/OW-108-M | 8/17/2022 | 6727.00 | 6717.00 | 6722.00 | 6722.98 | 6686.60 | 6676.60 | 6681.60 | 6680.83 | -- | -- | -- | -- | 40.4 | 42.15 | 1.0433 | **1.0152** |
| | 9/20/2022 | 6727.00 | 6717.00 | 6722.00 | 6722.95 | 6686.60 | 6676.60 | 6681.60 | 6681.12 | -- | -- | -- | -- | 40.4 | 41.83 | 1.0354 | |
| | 10/27/2022 | 6727.00 | 6717.00 | 6722.00 | 6722.87 | 6686.60 | 6676.60 | 6681.60 | 6681.38 | -- | -- | -- | -- | 40.4 | 41.49 | 1.0270 | |
| | 11/29/2022 | 6727.00 | 6717.00 | 6722.00 | 6722.85 | 6686.60 | 6676.60 | 6681.60 | 6681.66 | -- | -- | -- | -- | 40.4 | 41.19 | 1.0196 | |
| | 12/28/2022 | 6727.00 | 6717.00 | 6722.00 | 6722.90 | 6686.60 | 6676.60 | 6681.60 | 6681.89 | -- | -- | -- | -- | 40.4 | 41.01 | 1.0151 | |
| | 1/24/2023 | 6727.00 | 6717.00 | 6722.00 | 6722.81 | 6686.60 | 6676.60 | 6681.60 | 6682.09 | -- | -- | -- | -- | 40.4 | 40.72 | 1.0080 | |
| | 2/28/2023 | 6727.00 | 6717.00 | 6722.00 | 6722.83 | 6686.60 | 6676.60 | 6681.60 | 6682.37 | -- | -- | -- | -- | 40.4 | 40.46 | 1.0015 | |
| | 3/30/2023 | 6727.00 | 6717.00 | 6722.00 | 6722.84 | 6686.60 | 6676.60 | 6681.60 | 6682.59 | -- | -- | -- | -- | 40.4 | 40.25 | 0.9963 | |
| | 4/25/2023 | 6727.00 | 6717.00 | 6722.00 | 6722.77 | 6686.60 | 6676.60 | 6681.60 | 6682.77 | -- | -- | -- | -- | 40.4 | 40.00 | 0.9901 | |
| OW-109-U/OW-109-M | 8/17/2022 | 6725.80 | 6715.80 | 6720.80 | 6725.28 | 6700.50 | 6690.50 | 6695.50 | 6696.82 | -- | -- | -- | -- | 25.3 | 28.46 | 1.1249 | **0.8298** |
| | 9/20/2022 | 6725.80 | 6715.80 | 6720.80 | 6724.72 | 6700.50 | 6690.50 | 6695.50 | 6698.12 | -- | -- | -- | -- | 25.3 | 26.60 | 1.0514 | |
| | 10/27/2022 | 6725.80 | 6715.80 | 6720.80 | 6724.16 | 6700.50 | 6690.50 | 6695.50 | 6699.51 | -- | -- | -- | -- | 25.3 | 24.65 | 0.9743 | |
| | 11/29/2022 | 6725.80 | 6715.80 | 6720.80 | 6723.78 | 6700.50 | 6690.50 | 6695.50 | 6702.77 | -- | -- | -- | -- | 25.3 | 21.01 | 0.8305 | |
| | 12/28/2022 | 6725.80 | 6715.80 | 6720.80 | 6723.51 | 6700.50 | 6690.50 | 6695.50 | 6702.46 | -- | -- | -- | -- | 25.3 | 21.05 | 0.8320 | |
| | 1/24/2023 | 6725.80 | 6715.80 | 6720.80 | 6723.27 | 6700.50 | 6690.50 | 6695.50 | 6704.63 | -- | -- | -- | -- | 25.3 | 18.64 | 0.7368 | |
| | 2/28/2023 | 6725.80 | 6715.80 | 6720.80 | 6723.07 | 6700.50 | 6690.50 | 6695.50 | 6705.23 | -- | -- | -- | -- | 25.3 | 17.84 | 0.7052 | |
| | 3/30/2023 | 6725.80 | 6715.80 | 6720.80 | 6722.92 | 6700.50 | 6690.50 | 6695.50 | 6706.51 | -- | -- | -- | -- | 25.3 | 16.41 | 0.6486 | |
| | 4/25/2023 | 6725.80 | 6715.80 | 6720.80 | 6722.77 | 6700.50 | 6690.50 | 6695.50 | 6708.48 | -- | -- | -- | -- | 25.3 | 14.29 | 0.5648 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 2 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-117-U/OW-117-M | 8/17/2022 | 6710.30 | 6700.30 | 6705.30 | 6721.44 | 6671.00 | 6661.00 | 6666.00 | 6666.73 | -- | -- | -- | -- | 39.3 | 54.71 | 1.3921 | **1.3241** |
| | 9/20/2022 | 6710.30 | 6700.30 | 6705.30 | 6721.37 | 6671.00 | 6661.00 | 6666.00 | 6667.43 | -- | -- | -- | -- | 39.3 | 53.94 | 1.3725 | |
| | 10/27/2022 | 6710.30 | 6700.30 | 6705.30 | 6720.97 | 6671.00 | 6661.00 | 6666.00 | 6668.08 | -- | -- | -- | -- | 39.3 | 52.89 | 1.3458 | |
| | 11/29/2022 | 6710.30 | 6700.30 | 6705.30 | 6720.77 | 6671.00 | 6661.00 | 6666.00 | 6668.56 | -- | -- | -- | -- | 39.3 | 52.21 | 1.3285 | |
| | 12/28/2022 | 6710.30 | 6700.30 | 6705.30 | 6720.68 | 6671.00 | 6661.00 | 6666.00 | 6668.92 | -- | -- | -- | -- | 39.3 | 51.76 | 1.3170 | |
| | 1/24/2023 | 6710.30 | 6700.30 | 6705.30 | 6720.49 | 6671.00 | 6661.00 | 6666.00 | 6669.20 | -- | -- | -- | -- | 39.3 | 51.29 | 1.3051 | |
| | 2/27/2023 | 6710.30 | 6700.30 | 6705.30 | 6720.38 | 6671.00 | 6661.00 | 6666.00 | 6669.51 | -- | -- | -- | -- | 39.3 | 50.87 | 1.2944 | |
| | 3/29/2023 | 6710.30 | 6700.30 | 6705.30 | 6720.26 | 6671.00 | 6661.00 | 6666.00 | 6669.74 | -- | -- | -- | -- | 39.3 | 50.52 | 1.2855 | |
| | 4/26/2023 | 6710.30 | 6700.30 | 6705.30 | 6720.06 | 6671.00 | 6661.00 | 6666.00 | 6669.91 | -- | -- | -- | -- | 39.3 | 50.15 | 1.2761 | |
| OW-117-U/OW-117-L | 8/17/2022 | 6710.30 | 6700.30 | 6705.30 | 6721.44 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6686.27 | 75.2 | 35.16 | 0.4676 | **0.0696** |
| | 9/20/2022 | 6710.30 | 6700.30 | 6705.30 | 6721.37 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6700.75 | 75.2 | 20.61 | 0.2741 | |
| | 10/27/2022 | 6710.30 | 6700.30 | 6705.30 | 6720.97 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6710.85 | 75.2 | 10.11 | 0.1345 | |
| | 11/29/2022 | 6710.30 | 6700.30 | 6705.30 | 6720.77 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6716.76 | 75.2 | 4.00 | 0.0532 | |
| | 12/28/2022 | 6710.30 | 6700.30 | 6705.30 | 6720.68 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6720.40 | 75.2 | 0.27 | 0.0036 | |
| | 1/24/2023 | 6710.30 | 6700.30 | 6705.30 | 6720.49 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6723.02 | 75.2 | -2.54 | -0.0337 | |
| | 2/27/2023 | 6710.30 | 6700.30 | 6705.30 | 6720.38 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6725.51 | 75.2 | -5.14 | -0.0683 | |
| | 3/29/2023 | 6710.30 | 6700.30 | 6705.30 | 6720.26 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6727.20 | 75.2 | -6.95 | -0.0924 | |
| | 4/26/2023 | 6710.30 | 6700.30 | 6705.30 | 6720.06 | -- | -- | -- | -- | 6635.10 | 6625.10 | 6630.10 | 6728.47 | 75.2 | -8.42 | -0.1119 | |
| OW-117-M/OW-117-L | 8/17/2022 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6666.73 | 6635.10 | 6625.10 | 6630.10 | 6686.27 | 35.9 | -19.55 | -0.5444 | **-1.3036** |
| | 9/20/2022 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6667.43 | 6635.10 | 6625.10 | 6630.10 | 6700.75 | 35.9 | -33.33 | -0.9283 | |
| | 10/27/2022 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6668.08 | 6635.10 | 6625.10 | 6630.10 | 6710.85 | 35.9 | -42.78 | -1.1915 | |
| | 11/29/2022 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6668.56 | 6635.10 | 6625.10 | 6630.10 | 6716.76 | 35.9 | -48.21 | -1.3428 | |
| | 12/28/2022 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6668.92 | 6635.10 | 6625.10 | 6630.10 | 6720.40 | 35.9 | -51.49 | -1.4341 | |
| | 1/24/2023 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6669.20 | 6635.10 | 6625.10 | 6630.10 | 6723.02 | 35.9 | -53.83 | -1.4993 | |
| | 2/27/2023 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6669.51 | 6635.10 | 6625.10 | 6630.10 | 6725.51 | 35.9 | -56.01 | -1.5600 | |
| | 3/29/2023 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6669.74 | 6635.10 | 6625.10 | 6630.10 | 6727.20 | 35.9 | -57.47 | -1.6007 | |
| | 4/26/2023 | -- | -- | -- | -- | 6671.00 | 6661.00 | 6666.00 | 6669.91 | 6635.10 | 6625.10 | 6630.10 | 6728.47 | 35.9 | -58.57 | -1.6314 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 3 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-119-U/OW-119-M | 8/17/2022 | 6731.30 | 6721.30 | 6726.30 | 6721.51 | 6716.10 | 6706.10 | 6711.10 | 6721.62 | -- | -- | -- | -- | 15.2 | -0.11 | -0.0075 | **0.0152** |
| | 9/20/2022 | 6731.30 | 6721.30 | 6726.30 | 6721.42 | 6716.10 | 6706.10 | 6711.10 | 6721.43 | -- | -- | -- | -- | 15.2 | -0.01 | -0.0010 | |
| | 10/27/2022 | 6731.30 | 6721.30 | 6726.30 | 6721.26 | 6716.10 | 6706.10 | 6711.10 | 6721.15 | -- | -- | -- | -- | 15.2 | 0.11 | 0.0069 | |
| | 11/29/2022 | 6731.30 | 6721.30 | 6726.30 | 6721.16 | 6716.10 | 6706.10 | 6711.10 | 6720.99 | -- | -- | -- | -- | 15.2 | 0.17 | 0.0109 | |
| | 12/28/2022 | 6731.30 | 6721.30 | 6726.30 | 6721.06 | 6716.10 | 6706.10 | 6711.10 | 6720.88 | -- | -- | -- | -- | 15.2 | 0.18 | 0.0115 | |
| | 1/24/2023 | 6731.30 | 6721.30 | 6726.30 | 6721.04 | 6716.10 | 6706.10 | 6711.10 | 6720.72 | -- | -- | -- | -- | 15.2 | 0.32 | 0.0207 | |
| | 2/28/2023 | 6731.30 | 6721.30 | 6726.30 | 6721.02 | 6716.10 | 6706.10 | 6711.10 | 6720.62 | -- | -- | -- | -- | 15.2 | 0.40 | 0.0260 | |
| | 3/30/2023 | 6731.30 | 6721.30 | 6726.30 | 6720.98 | 6716.10 | 6706.10 | 6711.10 | 6720.55 | -- | -- | -- | -- | 15.2 | 0.43 | 0.0280 | |
| | 4/26/2023 | 6731.30 | 6721.30 | 6726.30 | 6721.03 | 6716.10 | 6706.10 | 6711.10 | 6720.40 | -- | -- | -- | -- | 15.2 | 0.63 | 0.0411 | |
| OW-121-U/OW-121-M(A) | 8/17/2022 | 6733.30 | 6723.30 | 6728.30 | 6724.70 | 6702.90 | 6692.90 | 6697.90 | 6698.28 | -- | -- | -- | -- | 30.4 | 26.42 | 0.8692 | **0.7682** |
| | 9/20/2022 | 6733.30 | 6723.30 | 6728.30 | 6724.22 | 6702.90 | 6692.90 | 6697.90 | 6698.84 | -- | -- | -- | -- | 30.4 | 25.38 | 0.8350 | |
| | 10/27/2022 | 6733.30 | 6723.30 | 6728.30 | 6723.74 | 6702.90 | 6692.90 | 6697.90 | 6699.34 | -- | -- | -- | -- | 30.4 | 24.40 | 0.8027 | |
| | 11/29/2022 | 6733.30 | 6723.30 | 6728.30 | NM | 6702.90 | 6692.90 | 6697.90 | NM | -- | -- | -- | -- | 30.4 | NA | NA | |
| | 12/28/2022 | 6733.30 | 6723.30 | 6728.30 | 6723.12 | 6702.90 | 6692.90 | 6697.90 | 6700.14 | -- | -- | -- | -- | 30.4 | 22.98 | 0.7560 | |
| | 1/24/2023 | 6733.30 | 6723.30 | 6728.30 | 6722.90 | 6702.90 | 6692.90 | 6697.90 | 6700.45 | -- | -- | -- | -- | 30.4 | 22.45 | 0.7386 | |
| | 2/28/2023 | 6733.30 | 6723.30 | 6728.30 | 6722.90 | 6702.90 | 6692.90 | 6697.90 | 6700.85 | -- | -- | -- | -- | 30.4 | 22.05 | 0.7254 | |
| | 3/30/2023 | 6733.30 | 6723.30 | 6728.30 | 6722.90 | 6702.90 | 6692.90 | 6697.90 | 6701.17 | -- | -- | -- | -- | 30.4 | 21.73 | 0.7149 | |
| | 4/25/2023 | 6733.30 | 6723.30 | 6728.30 | 6722.88 | 6702.90 | 6692.90 | 6697.90 | 6701.50 | -- | -- | -- | -- | 30.4 | 21.38 | 0.7034 | |
| OW-122-U/OW-122-M | 8/17/2022 | 6714.30 | 6709.30 | 6711.80 | 6723.48 | 6701.30 | 6691.30 | 6696.30 | 6723.51 | -- | -- | -- | -- | 15.5 | -0.03 | -0.0021 | **0.0057** |
| | 9/20/2022 | 6714.30 | 6709.30 | 6711.80 | 6723.05 | 6701.30 | 6691.30 | 6696.30 | 6723.05 | -- | -- | -- | -- | 15.5 | 0.00 | -0.0001 | |
| | 10/27/2022 | 6714.30 | 6709.30 | 6711.80 | 6722.73 | 6701.30 | 6691.30 | 6696.30 | 6722.63 | -- | -- | -- | -- | 15.5 | 0.10 | 0.0063 | |
| | 11/29/2022 | 6714.30 | 6709.30 | 6711.80 | 6722.48 | 6701.30 | 6691.30 | 6696.30 | 6722.39 | -- | -- | -- | -- | 15.5 | 0.09 | 0.0057 | |
| | 12/28/2022 | 6714.30 | 6709.30 | 6711.80 | 6722.33 | 6701.30 | 6691.30 | 6696.30 | 6722.21 | -- | -- | -- | -- | 15.5 | 0.12 | 0.0076 | |
| | 1/24/2023 | 6714.30 | 6709.30 | 6711.80 | 6722.13 | 6701.30 | 6691.30 | 6696.30 | 6722.00 | -- | -- | -- | -- | 15.5 | 0.13 | 0.0083 | |
| | 2/28/2023 | 6714.30 | 6709.30 | 6711.80 | 6721.99 | 6701.30 | 6691.30 | 6696.30 | 6721.88 | -- | -- | -- | -- | 15.5 | 0.11 | 0.0070 | |
| | 3/30/2023 | 6714.30 | 6709.30 | 6711.80 | 6721.88 | 6701.30 | 6691.30 | 6696.30 | 6721.75 | -- | -- | -- | -- | 15.5 | 0.13 | 0.0083 | |
| | 4/25/2023 | 6714.30 | 6709.30 | 6711.80 | 6721.76 | 6701.30 | 6691.30 | 6696.30 | 6721.59 | -- | -- | -- | -- | 15.5 | 0.17 | 0.0108 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 4 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-122-U/OW-122-L | 8/17/2022 | 6714.30 | 6709.30 | 6711.80 | 6723.48 | 6701.30 | 6691.30 | 6696.30 | 6723.51 | -- | -- | -- | -- | 15.5 | -0.03 | -0.0021 | **1.1389** |
| | 9/20/2022 | 6714.30 | 6709.30 | 6711.80 | 6723.05 | 6701.30 | 6691.30 | 6696.30 | 6723.05 | -- | -- | -- | -- | 15.5 | 0.00 | -0.0001 | |
| | 10/27/2022 | 6714.30 | 6709.30 | 6711.80 | 6722.73 | 6701.30 | 6691.30 | 6696.30 | 6722.63 | -- | -- | -- | -- | 15.5 | 0.10 | 0.0063 | |
| | 11/29/2022 | 6714.30 | 6709.30 | 6711.80 | 6722.48 | 6701.30 | 6691.30 | 6696.30 | 6722.39 | -- | -- | -- | -- | 15.5 | 0.09 | 0.0057 | |
| | 12/28/2022 | 6714.30 | 6709.30 | 6711.80 | 6722.33 | 6701.30 | 6691.30 | 6696.30 | 6722.21 | -- | -- | -- | -- | 15.5 | 0.12 | 0.0076 | |
| | 1/24/2023 | 6714.30 | 6709.30 | 6711.80 | 6722.13 | 6701.30 | 6691.30 | 6696.30 | 6722.00 | -- | -- | -- | -- | 15.5 | 0.13 | 0.0083 | |
| | 2/28/2023 | 6714.30 | 6709.30 | 6711.80 | 6721.99 | 6701.30 | 6691.30 | 6696.30 | 6721.88 | -- | -- | -- | -- | 15.5 | 0.11 | 0.0070 | |
| | 3/30/2023 | 6714.30 | 6709.30 | 6711.80 | 6721.88 | 6701.30 | 6691.30 | 6696.30 | 6721.75 | -- | -- | -- | -- | 15.5 | 0.13 | 0.0083 | |
| | 4/25/2023 | 6714.30 | 6709.30 | 6711.80 | 6721.76 | 6701.30 | 6691.30 | 6696.30 | 6721.59 | -- | -- | -- | -- | 15.5 | 0.17 | 0.0108 | |
| OW-122-M/OW-122-L | 8/17/2022 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6723.51 | 6656.20 | 6646.20 | 6651.20 | 6652.86 | 45.1 | 70.65 | 1.5665 | **1.5284** |
| | 9/20/2022 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6723.05 | 6656.20 | 6646.20 | 6651.20 | 6653.14 | 45.1 | 69.91 | 1.5501 | |
| | 10/27/2022 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6722.63 | 6656.20 | 6646.20 | 6651.20 | 6653.27 | 45.1 | 69.36 | 1.5379 | |
| | 11/29/2022 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6722.39 | 6656.20 | 6646.20 | 6651.20 | 6653.36 | 45.1 | 69.03 | 1.5306 | |
| | 12/28/2022 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6722.21 | 6656.20 | 6646.20 | 6651.20 | 6653.45 | 45.1 | 68.76 | 1.5246 | |
| | 1/24/2023 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6722.00 | 6656.20 | 6646.20 | 6651.20 | 6653.53 | 45.1 | 68.47 | 1.5182 | |
| | 2/28/2023 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6721.88 | 6656.20 | 6646.20 | 6651.20 | 6653.62 | 45.1 | 68.26 | 1.5135 | |
| | 3/30/2023 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6721.75 | 6656.20 | 6646.20 | 6651.20 | 6653.68 | 45.1 | 68.07 | 1.5093 | |
| | 4/25/2023 | -- | -- | -- | -- | 6701.30 | 6691.30 | 6696.30 | 6721.59 | 6656.20 | 6646.20 | 6651.20 | 6653.73 | 45.1 | 67.86 | 1.5047 | |
| OW-127-U/OW-127-M | 8/17/2022 | 6732.00 | 6722.00 | 6727.00 | 6722.56 | 6712.00 | 6702.00 | 6707.00 | 6707.76 | -- | -- | -- | -- | 20.0 | 14.80 | 0.7399 | **0.6979** |
| | 9/20/2022 | 6732.00 | 6722.00 | 6727.00 | 6722.55 | 6712.00 | 6702.00 | 6707.00 | 6707.99 | -- | -- | -- | -- | 20.0 | 14.56 | 0.7279 | |
| | 10/27/2022 | 6732.00 | 6722.00 | 6727.00 | 6722.49 | 6712.00 | 6702.00 | 6707.00 | 6708.18 | -- | -- | -- | -- | 20.0 | 14.31 | 0.7154 | |
| | 11/29/2022 | 6732.00 | 6722.00 | 6727.00 | 6722.49 | 6712.00 | 6702.00 | 6707.00 | 6708.37 | -- | -- | -- | -- | 20.0 | 14.12 | 0.7059 | |
| | 12/28/2022 | 6732.00 | 6722.00 | 6727.00 | 6722.49 | 6712.00 | 6702.00 | 6707.00 | 6708.53 | -- | -- | -- | -- | 20.0 | 13.96 | 0.6979 | |
| | 1/24/2023 | 6732.00 | 6722.00 | 6727.00 | 6722.40 | 6712.00 | 6702.00 | 6707.00 | 6708.67 | -- | -- | -- | -- | 20.0 | 13.73 | 0.6864 | |
| | 2/28/2023 | 6732.00 | 6722.00 | 6727.00 | 6722.44 | 6712.00 | 6702.00 | 6707.00 | 6708.87 | -- | -- | -- | -- | 20.0 | 13.57 | 0.6784 | |
| | 3/30/2023 | 6732.00 | 6722.00 | 6727.00 | 6722.42 | 6712.00 | 6702.00 | 6707.00 | 6709.03 | -- | -- | -- | -- | 20.0 | 13.39 | 0.6694 | |
| | 4/25/2023 | 6732.00 | 6722.00 | 6727.00 | 6722.35 | 6712.00 | 6702.00 | 6707.00 | 6709.15 | -- | -- | -- | -- | 20.0 | 13.20 | 0.6599 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 5 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-136-U/OW-136-M | 8/17/2022 | 6714.70 | 6704.70 | 6709.70 | 6721.90 | 6672.50 | 6662.50 | 6667.50 | 6663.53 | -- | -- | -- | -- | 42.2 | 58.37 | 1.3831 | **1.3721** |
| | 9/20/2022 | 6714.70 | 6704.70 | 6709.70 | 6721.94 | 6672.50 | 6662.50 | 6667.50 | 6663.62 | -- | -- | -- | -- | 42.2 | 58.32 | 1.3819 | |
| | 10/26/2022 | 6714.70 | 6704.70 | 6709.70 | 6722.07 | 6672.50 | 6662.50 | 6667.50 | 6663.69 | -- | -- | -- | -- | 42.2 | 58.38 | 1.3833 | |
| | 11/29/2022 | 6714.70 | 6704.70 | 6709.70 | 6721.95 | 6672.50 | 6662.50 | 6667.50 | 6663.77 | -- | -- | -- | -- | 42.2 | 58.18 | 1.3786 | |
| | 12/29/2022 | 6714.70 | 6704.70 | 6709.70 | 6721.85 | 6672.50 | 6662.50 | 6667.50 | 6663.82 | -- | -- | -- | -- | 42.2 | 58.03 | 1.3750 | |
| | 1/24/2023 | 6714.70 | 6704.70 | 6709.70 | 6721.49 | 6672.50 | 6662.50 | 6667.50 | 6663.85 | -- | -- | -- | -- | 42.2 | 57.64 | 1.3658 | |
| | 2/27/2023 | 6714.70 | 6704.70 | 6709.70 | 6721.56 | 6672.50 | 6662.50 | 6667.50 | 6663.93 | -- | -- | -- | -- | 42.2 | 57.63 | 1.3656 | |
| | 3/29/2023 | 6714.70 | 6704.70 | 6709.70 | 6721.38 | 6672.50 | 6662.50 | 6667.50 | 6663.98 | -- | -- | -- | -- | 42.2 | 57.40 | 1.3601 | |
| | 4/25/2023 | 6714.70 | 6704.70 | 6709.70 | 6721.23 | 6672.50 | 6662.50 | 6667.50 | 6664.02 | -- | -- | -- | -- | 42.2 | 57.21 | 1.3556 | |
| OW-137-U/OW-137-M | 8/17/2022 | 6723.70 | 6713.70 | 6718.70 | 6721.08 | 6678.40 | 6668.40 | 6673.40 | 6721.98 | -- | -- | -- | -- | 45.3 | -0.90 | -0.0199 | **-0.0150** |
| | 9/20/2022 | 6723.70 | 6713.70 | 6718.70 | 6720.93 | 6678.40 | 6668.40 | 6673.40 | 6721.67 | -- | -- | -- | -- | 45.3 | -0.74 | -0.0163 | |
| | 10/27/2022 | 6723.70 | 6713.70 | 6718.70 | 6720.74 | 6678.40 | 6668.40 | 6673.40 | 6721.39 | -- | -- | -- | -- | 45.3 | -0.65 | -0.0143 | |
| | 11/29/2022 | 6723.70 | 6713.70 | 6718.70 | 6720.63 | 6678.40 | 6668.40 | 6673.40 | 6721.25 | -- | -- | -- | -- | 45.3 | -0.62 | -0.0137 | |
| | 12/28/2022 | 6723.70 | 6713.70 | 6718.70 | 6720.52 | 6678.40 | 6668.40 | 6673.40 | 6721.15 | -- | -- | -- | -- | 45.3 | -0.63 | -0.0139 | |
| | 1/24/2023 | 6723.70 | 6713.70 | 6718.70 | 6720.35 | 6678.40 | 6668.40 | 6673.40 | 6720.98 | -- | -- | -- | -- | 45.3 | -0.63 | -0.0139 | |
| | 2/28/2023 | 6723.70 | 6713.70 | 6718.70 | 6720.27 | 6678.40 | 6668.40 | 6673.40 | 6720.89 | -- | -- | -- | -- | 45.3 | -0.62 | -0.0137 | |
| | 3/30/2023 | 6723.70 | 6713.70 | 6718.70 | 6720.15 | 6678.40 | 6668.40 | 6673.40 | 6720.83 | -- | -- | -- | -- | 45.3 | -0.68 | -0.0150 | |
| | 4/26/2023 | 6723.70 | 6713.70 | 6718.70 | 6720.02 | 6678.40 | 6668.40 | 6673.40 | 6720.67 | -- | -- | -- | -- | 45.3 | -0.65 | -0.0143 | |
| OW-140-U/OW-140-M | 8/17/2022 | 6709.90 | 6699.90 | 6704.90 | 6722.49 | 6685.80 | 6675.80 | 6680.80 | 6722.61 | -- | -- | -- | -- | 24.1 | -0.11 | -0.0046 | **0.0057** |
| | 9/20/2022 | 6709.90 | 6699.90 | 6704.90 | 6722.51 | 6685.80 | 6675.80 | 6680.80 | 6722.61 | -- | -- | -- | -- | 24.1 | -0.09 | -0.0038 | |
| | 10/26/2022 | 6709.90 | 6699.90 | 6704.90 | 6722.58 | 6685.80 | 6675.80 | 6680.80 | 6722.70 | -- | -- | -- | -- | 24.1 | -0.11 | -0.0046 | |
| | 11/29/2022 | 6709.90 | 6699.90 | 6704.90 | 6722.56 | 6685.80 | 6675.80 | 6680.80 | 6722.68 | -- | -- | -- | -- | 24.1 | -0.11 | -0.0046 | |
| | 12/29/2022 | 6709.90 | 6699.90 | 6704.90 | 6722.26 | 6685.80 | 6675.80 | 6680.80 | 6722.57 | -- | -- | -- | -- | 24.1 | -0.30 | -0.0125 | |
| | 1/24/2023 | 6709.90 | 6699.90 | 6704.90 | 6721.96 | 6685.80 | 6675.80 | 6680.80 | 6722.13 | -- | -- | -- | -- | 24.1 | -0.16 | -0.0067 | |
| | 2/27/2023 | 6709.90 | 6699.90 | 6704.90 | 6722.02 | 6685.80 | 6675.80 | 6680.80 | 6722.15 | -- | -- | -- | -- | 24.1 | -0.12 | -0.0050 | |
| | 3/29/2023 | 6709.90 | 6699.90 | 6704.90 | 6721.88 | 6685.80 | 6675.80 | 6680.80 | 6722.00 | -- | -- | -- | -- | 24.1 | -0.11 | -0.0046 | |
| | 4/25/2023 | 6709.90 | 6699.90 | 6704.90 | 6721.70 | 6685.80 | 6675.80 | 6680.80 | 6721.82 | -- | -- | -- | -- | 24.1 | -0.11 | -0.0046 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 6 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-149-U/OW-149-M | 8/17/2022 | 6727.30 | 6717.30 | 6722.30 | 6720.44 | 6702.40 | 6692.40 | 6697.40 | 6696.98 | -- | -- | -- | -- | 24.9 | 23.45 | 0.9419 | **0.8792** |
| | 9/20/2022 | 6727.30 | 6717.30 | 6722.30 | 6720.34 | 6702.40 | 6692.40 | 6697.40 | 6697.34 | -- | -- | -- | -- | 24.9 | 22.99 | 0.9234 | |
| | 10/26/2022 | 6727.30 | 6717.30 | 6722.30 | 6720.24 | 6702.40 | 6692.40 | 6697.40 | 6697.67 | -- | -- | -- | -- | 24.9 | 22.56 | 0.9062 | |
| | 11/29/2022 | 6727.30 | 6717.30 | 6722.30 | 6720.16 | 6702.40 | 6692.40 | 6697.40 | 6698.04 | -- | -- | -- | -- | 24.9 | 22.11 | 0.8881 | |
| | 12/28/2022 | 6727.30 | 6717.30 | 6722.30 | 6720.13 | 6702.40 | 6692.40 | 6697.40 | 6698.31 | -- | -- | -- | -- | 24.9 | 21.81 | 0.8760 | |
| | 1/24/2023 | 6727.30 | 6717.30 | 6722.30 | 6720.07 | 6702.40 | 6692.40 | 6697.40 | 6698.59 | -- | -- | -- | -- | 24.9 | 21.47 | 0.8624 | |
| | 2/28/2023 | 6727.30 | 6717.30 | 6722.30 | 6720.07 | 6702.40 | 6692.40 | 6697.40 | 6698.91 | -- | -- | -- | -- | 24.9 | 21.15 | 0.8495 | |
| | 3/29/2023 | 6727.30 | 6717.30 | 6722.30 | 6720.03 | 6702.40 | 6692.40 | 6697.40 | 6699.16 | -- | -- | -- | -- | 24.9 | 20.86 | 0.8379 | |
| | 4/25/2023 | 6727.30 | 6717.30 | 6722.30 | 6719.97 | 6702.40 | 6692.40 | 6697.40 | 6699.37 | -- | -- | -- | -- | 24.9 | 20.59 | 0.8270 | |
| OW-153-U/OW-153-M | 8/17/2022 | 6702.50 | 6692.50 | 6697.50 | 6718.11 | 6672.80 | 6662.80 | 6667.80 | 6668.13 | -- | -- | -- | -- | 29.7 | 49.98 | 1.6828 | **1.6259** |
| | 9/20/2022 | 6702.50 | 6692.50 | 6697.50 | 6720.18 | 6672.80 | 6662.80 | 6667.80 | 6668.86 | -- | -- | -- | -- | 29.7 | 51.32 | 1.7279 | |
| | 10/26/2022 | 6702.50 | 6692.50 | 6697.50 | 6720.69 | 6672.80 | 6662.80 | 6667.80 | 6669.61 | -- | -- | -- | -- | 29.7 | 51.08 | 1.7198 | |
| | 11/29/2022 | 6702.50 | 6692.50 | 6697.50 | 6720.57 | 6672.80 | 6662.80 | 6667.80 | 6670.44 | -- | -- | -- | -- | 29.7 | 50.13 | 1.6879 | |
| | 12/29/2022 | 6702.50 | 6692.50 | 6697.50 | 6720.56 | 6672.80 | 6662.80 | 6667.80 | 6671.40 | -- | -- | -- | -- | 29.7 | 49.16 | 1.6552 | |
| | 1/24/2023 | 6702.50 | 6692.50 | 6697.50 | 6720.63 | 6672.80 | 6662.80 | 6667.80 | 6672.77 | -- | -- | -- | -- | 29.7 | 47.86 | 1.6114 | |
| | 2/27/2023 | 6702.50 | 6692.50 | 6697.50 | 6720.62 | 6672.80 | 6662.80 | 6667.80 | 6674.24 | -- | -- | -- | -- | 29.7 | 46.38 | 1.5616 | |
| | 3/29/2023 | 6702.50 | 6692.50 | 6697.50 | 6720.45 | 6672.80 | 6662.80 | 6667.80 | 6675.50 | -- | -- | -- | -- | 29.7 | 44.95 | 1.5134 | |
| | 4/25/2023 | 6702.50 | 6692.50 | 6697.50 | 6720.35 | 6672.80 | 6662.80 | 6667.80 | 6676.61 | -- | -- | -- | -- | 29.7 | 43.74 | 1.4727 | |
| OW-153-U/OW-153-L | 8/17/2022 | 6702.50 | 6692.50 | 6697.50 | 6718.11 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | 6728.55 | 66.0 | -10.44 | -0.1583 | **-0.1981** |
| | 9/20/2022 | 6702.50 | 6692.50 | 6697.50 | 6720.18 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | 6732.44 | 66.0 | -12.26 | -0.1858 | |
| | 10/26/2022 | 6702.50 | 6692.50 | 6697.50 | 6720.69 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | 6733.96 | 66.0 | -13.27 | -0.2011 | |
| | 11/29/2022 | 6702.50 | 6692.50 | 6697.50 | 6720.57 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | 6734.73 | 66.0 | -14.16 | -0.2146 | |
| | 12/29/2022 | 6702.50 | 6692.50 | 6697.50 | 6720.56 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | 6734.92 | 66.0 | -14.36 | -0.2176 | |
| | 1/24/2023 | 6702.50 | 6692.50 | 6697.50 | 6720.63 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | NM | 66.0 | NA | NA | |
| | 2/27/2023 | 6702.50 | 6692.50 | 6697.50 | 6720.62 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | NM | 66.0 | NA | NA | |
| | 3/29/2023 | 6702.50 | 6692.50 | 6697.50 | 6720.45 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | NM | 66.0 | NA | NA | |
| | 4/25/2023 | 6702.50 | 6692.50 | 6697.50 | 6720.35 | -- | -- | -- | -- | 6636.50 | 6626.50 | 6631.50 | 6734.27 | 66.0 | -13.92 | -0.2110 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 7 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-153-M/OW-153-L | 8/17/2022 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6668.13 | 6636.50 | 6626.50 | 6631.50 | 6728.55 | 36.3 | -60.42 | -1.6646 | **-1.7165** |
| | 9/20/2022 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6668.86 | 6636.50 | 6626.50 | 6631.50 | 6732.44 | 36.3 | -63.58 | -1.7516 | |
| | 10/26/2022 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6669.61 | 6636.50 | 6626.50 | 6631.50 | 6733.96 | 36.3 | -64.35 | -1.7728 | |
| | 11/29/2022 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6670.44 | 6636.50 | 6626.50 | 6631.50 | 6734.73 | 36.3 | -64.29 | -1.7712 | |
| | 12/29/2022 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6671.40 | 6636.50 | 6626.50 | 6631.50 | 6734.92 | 36.3 | -63.52 | -1.7500 | |
| | 1/24/2023 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6672.77 | 6636.50 | 6626.50 | 6631.50 | NM | 36.3 | NA | NA | |
| | 2/27/2023 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6674.24 | 6636.50 | 6626.50 | 6631.50 | NM | 36.3 | NA | NA | |
| | 3/29/2023 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6675.50 | 6636.50 | 6626.50 | 6631.50 | NM | 36.3 | NA | NA | |
| | 4/25/2023 | -- | -- | -- | -- | 6672.80 | 6662.80 | 6667.80 | 6676.61 | 6636.50 | 6626.50 | 6631.50 | 6734.27 | 36.3 | -57.66 | -1.5885 | |
| OW-164-U/OW-164-M | 8/17/2022 | 6711.00 | 6701.00 | 6706.00 | 6718.56 | 6691.60 | 6681.60 | 6686.60 | 6718.53 | -- | -- | -- | -- | 19.4 | 0.03 | 0.0017 | **0.00002** |
| | 9/20/2022 | 6711.00 | 6701.00 | 6706.00 | 6718.39 | 6691.60 | 6681.60 | 6686.60 | 6718.39 | -- | -- | -- | -- | 19.4 | 0.00 | 0.0001 | |
| | 10/26/2022 | 6711.00 | 6701.00 | 6706.00 | 6718.33 | 6691.60 | 6681.60 | 6686.60 | 6718.30 | -- | -- | -- | -- | 19.4 | 0.03 | 0.0017 | |
| | 11/29/2022 | 6711.00 | 6701.00 | 6706.00 | 6718.31 | 6691.60 | 6681.60 | 6686.60 | 6718.27 | -- | -- | -- | -- | 19.4 | 0.04 | 0.0022 | |
| | 12/28/2022 | 6711.00 | 6701.00 | 6706.00 | 6718.32 | 6691.60 | 6681.60 | 6686.60 | 6718.33 | -- | -- | -- | -- | 19.4 | -0.01 | -0.0004 | |
| | 1/24/2023 | 6711.00 | 6701.00 | 6706.00 | 6718.29 | 6691.60 | 6681.60 | 6686.60 | 6718.27 | -- | -- | -- | -- | 19.4 | 0.02 | 0.0012 | |
| | 2/27/2023 | 6711.00 | 6701.00 | 6706.00 | 6718.20 | 6691.60 | 6681.60 | 6686.60 | 6718.31 | -- | -- | -- | -- | 19.4 | -0.11 | -0.0055 | |
| | 3/29/2023 | 6711.00 | 6701.00 | 6706.00 | 6718.20 | 6691.60 | 6681.60 | 6686.60 | 6718.19 | -- | -- | -- | -- | 19.4 | 0.01 | 0.0007 | |
| | 4/25/2023 | 6711.00 | 6701.00 | 6706.00 | 6718.11 | 6691.60 | 6681.60 | 6686.60 | 6718.11 | -- | -- | -- | -- | 19.4 | 0.00 | 0.0001 | |
| OW-167-U/OW-167-M | 8/17/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.19 | 6678.60 | 6668.60 | 6673.60 | 6673.26 | -- | -- | -- | -- | 40.0 | 46.93 | 1.1733 | **1.1351** |
| | 9/20/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.19 | 6678.60 | 6668.60 | 6673.60 | 6673.72 | -- | -- | -- | -- | 40.0 | 46.47 | 1.1618 | |
| | 10/26/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.20 | 6678.60 | 6668.60 | 6673.60 | 6674.12 | -- | -- | -- | -- | 40.0 | 46.08 | 1.1521 | |
| | 11/29/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.12 | 6678.60 | 6668.60 | 6673.60 | 6674.50 | -- | -- | -- | -- | 40.0 | 45.62 | 1.1406 | |
| | 12/28/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.16 | 6678.60 | 6668.60 | 6673.60 | 6674.79 | -- | -- | -- | -- | 40.0 | 45.37 | 1.1343 | |
| | 1/24/2023 | 6718.60 | 6708.60 | 6713.60 | 6720.02 | 6678.60 | 6668.60 | 6673.60 | 6675.05 | -- | -- | -- | -- | 40.0 | 44.97 | 1.1243 | |
| | 2/27/2023 | 6718.60 | 6708.60 | 6713.60 | 6720.09 | 6678.60 | 6668.60 | 6673.60 | 6675.40 | -- | -- | -- | -- | 40.0 | 44.69 | 1.1173 | |
| | 3/29/2023 | 6718.60 | 6708.60 | 6713.60 | 6720.06 | 6678.60 | 6668.60 | 6673.60 | 6675.67 | -- | -- | -- | -- | 40.0 | 44.39 | 1.1098 | |
| | 4/25/2023 | 6718.60 | 6708.60 | 6713.60 | 6719.99 | 6678.60 | 6668.60 | 6673.60 | 6675.90 | -- | -- | -- | -- | 40.0 | 44.09 | 1.1023 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 8 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-167-U/OW-167-L | 8/17/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.19 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6609.81 | 104.6 | 110.38 | 1.0552 | **1.0528** |
| | 9/20/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.19 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6609.93 | 104.6 | 110.26 | 1.0541 | |
| | 10/26/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.20 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6609.93 | 104.6 | 110.27 | 1.0542 | |
| | 11/29/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.12 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6609.98 | 104.6 | 110.14 | 1.0529 | |
| | 12/28/2022 | 6718.60 | 6708.60 | 6713.60 | 6720.16 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6609.98 | 104.6 | 110.18 | 1.0533 | |
| | 1/24/2023 | 6718.60 | 6708.60 | 6713.60 | 6720.02 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6609.99 | 104.6 | 110.03 | 1.0519 | |
| | 2/27/2023 | 6718.60 | 6708.60 | 6713.60 | 6720.09 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6610.06 | 104.6 | 110.03 | 1.0519 | |
| | 3/29/2023 | 6718.60 | 6708.60 | 6713.60 | 6720.06 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6610.09 | 104.6 | 109.97 | 1.0513 | |
| | 4/25/2023 | 6718.60 | 6708.60 | 6713.60 | 6719.99 | -- | -- | -- | -- | 6614.00 | 6604.00 | 6609.00 | 6610.12 | 104.6 | 109.87 | 1.0504 | |
| OW-167-M/OW-167-L | 8/17/2022 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6673.26 | 6614.00 | 6604.00 | 6609.00 | 6609.81 | 64.6 | 63.44 | 0.9821 | **1.0018** |
| | 9/20/2022 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6673.72 | 6614.00 | 6604.00 | 6609.00 | 6609.93 | 64.6 | 63.78 | 0.9874 | |
| | 10/26/2022 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6674.12 | 6614.00 | 6604.00 | 6609.00 | 6609.93 | 64.6 | 64.18 | 0.9936 | |
| | 11/29/2022 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6674.50 | 6614.00 | 6604.00 | 6609.00 | 6609.98 | 64.6 | 64.51 | 0.9987 | |
| | 12/28/2022 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6674.79 | 6614.00 | 6604.00 | 6609.00 | 6609.98 | 64.6 | 64.80 | 1.0032 | |
| | 1/24/2023 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6675.05 | 6614.00 | 6604.00 | 6609.00 | 6609.99 | 64.6 | 65.05 | 1.0070 | |
| | 2/27/2023 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6675.40 | 6614.00 | 6604.00 | 6609.00 | 6610.06 | 64.6 | 65.33 | 1.0114 | |
| | 3/29/2023 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6675.67 | 6614.00 | 6604.00 | 6609.00 | 6610.09 | 64.6 | 65.57 | 1.0151 | |
| | 4/25/2023 | -- | -- | -- | -- | 6678.60 | 6668.60 | 6673.60 | 6675.90 | 6614.00 | 6604.00 | 6609.00 | 6610.12 | 64.6 | 65.77 | 1.0182 | |
| OW-168-U/OW-168-M(A) | 8/17/2022 | 6730.80 | 6720.80 | 6725.80 | 6720.14 | 6711.00 | 6701.00 | 6706.00 | 6719.89 | -- | -- | -- | -- | 19.8 | 0.24 | 0.0123 | **0.0280** |
| | 9/20/2022 | 6730.80 | 6720.80 | 6725.80 | 6720.17 | 6711.00 | 6701.00 | 6706.00 | 6719.75 | -- | -- | -- | -- | 19.8 | 0.41 | 0.0209 | |
| | 10/26/2022 | 6730.80 | 6720.80 | 6725.80 | 6720.16 | 6711.00 | 6701.00 | 6706.00 | 6719.64 | -- | -- | -- | -- | 19.8 | 0.51 | 0.0260 | |
| | 11/29/2022 | 6730.80 | 6720.80 | 6725.80 | 6720.18 | 6711.00 | 6701.00 | 6706.00 | 6719.59 | -- | -- | -- | -- | 19.8 | 0.58 | 0.0295 | |
| | 12/28/2022 | 6730.80 | 6720.80 | 6725.80 | 6720.18 | 6711.00 | 6701.00 | 6706.00 | 6719.59 | -- | -- | -- | -- | 19.8 | 0.58 | 0.0295 | |
| | 1/24/2023 | 6730.80 | 6720.80 | 6725.80 | 6720.16 | 6711.00 | 6701.00 | 6706.00 | 6719.51 | -- | -- | -- | -- | 19.8 | 0.64 | 0.0325 | |
| | 2/27/2023 | 6730.80 | 6720.80 | 6725.80 | 6720.19 | 6711.00 | 6701.00 | 6706.00 | 6719.55 | -- | -- | -- | -- | 19.8 | 0.63 | 0.0320 | |
| | 3/29/2023 | 6730.80 | 6720.80 | 6725.80 | 6720.17 | 6711.00 | 6701.00 | 6706.00 | 6719.51 | -- | -- | -- | -- | 19.8 | 0.65 | 0.0330 | |
| | 4/25/2023 | 6730.80 | 6720.80 | 6725.80 | 6720.18 | 6711.00 | 6701.00 | 6706.00 | 6719.46 | -- | -- | -- | -- | 19.8 | 0.71 | 0.0361 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 9 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-179-U/OW-179-M | 8/17/2022 | 6727.30 | 6717.30 | 6722.30 | 6718.64 | 6712.40 | 6702.40 | 6707.40 | 6718.57 | -- | -- | -- | -- | 14.9 | 0.07 | 0.0050 | **0.0044** |
| | 9/20/2022 | 6727.30 | 6717.30 | 6722.30 | 6718.54 | 6712.40 | 6702.40 | 6707.40 | 6718.52 | -- | -- | -- | -- | 14.9 | 0.02 | 0.0016 | |
| | 10/26/2022 | 6727.30 | 6717.30 | 6722.30 | 6718.45 | 6712.40 | 6702.40 | 6707.40 | 6718.38 | -- | -- | -- | -- | 14.9 | 0.07 | 0.0050 | |
| | 11/29/2022 | 6727.30 | 6717.30 | 6722.30 | 6718.44 | 6712.40 | 6702.40 | 6707.40 | 6718.36 | -- | -- | -- | -- | 14.9 | 0.08 | 0.0057 | |
| | 12/28/2022 | 6727.30 | 6717.30 | 6722.30 | 6718.43 | 6712.40 | 6702.40 | 6707.40 | 6718.36 | -- | -- | -- | -- | 14.9 | 0.07 | 0.0050 | |
| | 1/24/2023 | 6727.30 | 6717.30 | 6722.30 | 6718.40 | 6712.40 | 6702.40 | 6707.40 | 6718.34 | -- | -- | -- | -- | 14.9 | 0.06 | 0.0043 | |
| | 2/27/2023 | 6727.30 | 6717.30 | 6722.30 | 6718.44 | 6712.40 | 6702.40 | 6707.40 | 6718.37 | -- | -- | -- | -- | 14.9 | 0.07 | 0.0050 | |
| | 3/29/2023 | 6727.30 | 6717.30 | 6722.30 | 6718.39 | 6712.40 | 6702.40 | 6707.40 | 6718.33 | -- | -- | -- | -- | 14.9 | 0.06 | 0.0043 | |
| | 4/25/2023 | 6727.30 | 6717.30 | 6722.30 | 6718.35 | 6712.40 | 6702.40 | 6707.40 | 6718.30 | -- | -- | -- | -- | 14.9 | 0.05 | 0.0037 | |
| OW-185-U/OW-185-M | 8/17/2022 | 6705.10 | 6695.10 | 6700.10 | 6731.59 | 6676.10 | 6666.10 | 6671.10 | 6733.16 | -- | -- | -- | -- | 29.0 | -1.57 | -0.0540 | **-0.0341** |
| | 9/21/2022 | 6705.10 | 6695.10 | 6700.10 | 6731.68 | 6676.10 | 6666.10 | 6671.10 | 6733.22 | -- | -- | -- | -- | 29.0 | -1.54 | -0.0530 | |
| | 10/26/2022 | 6705.10 | 6695.10 | 6700.10 | 6732.12 | 6676.10 | 6666.10 | 6671.10 | 6733.50 | -- | -- | -- | -- | 29.0 | -1.38 | -0.0475 | |
| | 11/30/2022 | 6705.10 | 6695.10 | 6700.10 | 6732.32 | 6676.10 | 6666.10 | 6671.10 | 6733.66 | -- | -- | -- | -- | 29.0 | -1.34 | -0.0461 | |
| | 12/29/2022 | 6705.10 | 6695.10 | 6700.10 | 6732.47 | 6676.10 | 6666.10 | 6671.10 | 6733.39 | -- | -- | -- | -- | 29.0 | -0.92 | -0.0316 | |
| | 1/25/2023 | 6705.10 | 6695.10 | 6700.10 | 6732.62 | 6676.10 | 6666.10 | 6671.10 | 6733.28 | -- | -- | -- | -- | 29.0 | -0.66 | -0.0226 | |
| | 3/1/2023 | 6705.10 | 6695.10 | 6700.10 | 6733.22 | 6676.10 | 6666.10 | 6671.10 | 6733.61 | -- | -- | -- | -- | 29.0 | -0.39 | -0.0133 | |
| | 3/29/2023 | 6705.10 | 6695.10 | 6700.10 | 6733.04 | 6676.10 | 6666.10 | 6671.10 | 6733.52 | -- | -- | -- | -- | 29.0 | -0.48 | -0.0164 | |
| | 4/26/2023 | 6705.10 | 6695.10 | 6700.10 | 6732.99 | 6676.10 | 6666.10 | 6671.10 | 6733.64 | -- | -- | -- | -- | 29.0 | -0.65 | -0.0223 | |
| OW-186-U/OW-186-M | 8/17/2022 | 6713.80 | 6703.80 | 6708.80 | 6722.40 | 6693.80 | 6683.80 | 6688.80 | 6719.46 | -- | -- | -- | -- | 20.0 | 2.94 | 0.1469 | **0.1457** |
| | 9/20/2022 | 6713.80 | 6703.80 | 6708.80 | 6722.52 | 6693.80 | 6683.80 | 6688.80 | 6719.47 | -- | -- | -- | -- | 20.0 | 3.05 | 0.1524 | |
| | 10/26/2022 | 6713.80 | 6703.80 | 6708.80 | 6722.55 | 6693.80 | 6683.80 | 6688.80 | 6719.49 | -- | -- | -- | -- | 20.0 | 3.06 | 0.1529 | |
| | 11/29/2022 | 6713.80 | 6703.80 | 6708.80 | 6722.41 | 6693.80 | 6683.80 | 6688.80 | 6719.56 | -- | -- | -- | -- | 20.0 | 2.85 | 0.1424 | |
| | 12/29/2022 | 6713.80 | 6703.80 | 6708.80 | 6722.42 | 6693.80 | 6683.80 | 6688.80 | 6719.56 | -- | -- | -- | -- | 20.0 | 2.86 | 0.1429 | |
| | 1/24/2023 | 6713.80 | 6703.80 | 6708.80 | 6722.34 | 6693.80 | 6683.80 | 6688.80 | 6719.47 | -- | -- | -- | -- | 20.0 | 2.87 | 0.1434 | |
| | 3/1/2023 | 6713.80 | 6703.80 | 6708.80 | 6722.43 | 6693.80 | 6683.80 | 6688.80 | 6719.57 | -- | -- | -- | -- | 20.0 | 2.86 | 0.1429 | |
| | 3/29/2023 | 6713.80 | 6703.80 | 6708.80 | 6722.40 | 6693.80 | 6683.80 | 6688.80 | 6719.47 | -- | -- | -- | -- | 20.0 | 2.93 | 0.1464 | |
| | 4/25/2023 | 6713.80 | 6703.80 | 6708.80 | 6722.26 | 6693.80 | 6683.80 | 6688.80 | 6719.42 | -- | -- | -- | -- | 20.0 | 2.84 | 0.1419 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 10 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-187-U/OW-187-M | 8/17/2022 | 6683.80 | 6673.80 | 6678.80 | 6728.32 | 6667.90 | 6657.90 | 6662.90 | 6728.38 | -- | -- | -- | -- | 15.9 | -0.06 | -0.0037 | **-0.0020** |
| | 9/21/2022 | 6683.80 | 6673.80 | 6678.80 | 6728.38 | 6667.90 | 6657.90 | 6662.90 | 6728.42 | -- | -- | -- | -- | 15.9 | -0.04 | -0.0025 | |
| | 10/26/2022 | 6683.80 | 6673.80 | 6678.80 | 6728.56 | 6667.90 | 6657.90 | 6662.90 | 6728.58 | -- | -- | -- | -- | 15.9 | -0.02 | -0.0012 | |
| | 11/30/2022 | 6683.80 | 6673.80 | 6678.80 | 6728.52 | 6667.90 | 6657.90 | 6662.90 | 6728.56 | -- | -- | -- | -- | 15.9 | -0.04 | -0.0025 | |
| | 12/29/2022 | 6683.80 | 6673.80 | 6678.80 | 6728.51 | 6667.90 | 6657.90 | 6662.90 | 6728.54 | -- | -- | -- | -- | 15.9 | -0.03 | -0.0019 | |
| | 1/24/2023 | 6683.80 | 6673.80 | 6678.80 | 6728.38 | 6667.90 | 6657.90 | 6662.90 | 6728.36 | -- | -- | -- | -- | 15.9 | 0.02 | 0.0013 | |
| | 2/27/2023 | 6683.80 | 6673.80 | 6678.80 | 6728.54 | 6667.90 | 6657.90 | 6662.90 | 6728.59 | -- | -- | -- | -- | 15.9 | -0.05 | -0.0031 | |
| | 3/29/2023 | 6683.80 | 6673.80 | 6678.80 | 6728.38 | 6667.90 | 6657.90 | 6662.90 | 6728.43 | -- | -- | -- | -- | 15.9 | -0.05 | -0.0031 | |
| | 4/26/2023 | 6683.80 | 6673.80 | 6678.80 | 6728.46 | 6667.90 | 6657.90 | 6662.90 | 6728.48 | -- | -- | -- | -- | 15.9 | -0.02 | -0.0012 | |
| OW-188-U/OW-188-M | 8/17/2022 | 6719.90 | 6709.90 | 6714.90 | 6723.44 | 6699.90 | 6689.90 | 6694.90 | 6726.56 | -- | -- | -- | -- | 20.0 | -3.12 | -0.1560 | **-0.1609** |
| | 9/21/2022 | 6719.90 | 6709.90 | 6714.90 | 6723.39 | 6699.90 | 6689.90 | 6694.90 | 6726.51 | -- | -- | -- | -- | 20.0 | -3.12 | -0.1560 | |
| | 10/26/2022 | 6719.90 | 6709.90 | 6714.90 | 6723.45 | 6699.90 | 6689.90 | 6694.90 | 6726.62 | -- | -- | -- | -- | 20.0 | -3.17 | -0.1585 | |
| | 11/30/2022 | 6719.90 | 6709.90 | 6714.90 | 6723.44 | 6699.90 | 6689.90 | 6694.90 | 6726.61 | -- | -- | -- | -- | 20.0 | -3.17 | -0.1585 | |
| | 12/29/2022 | 6719.90 | 6709.90 | 6714.90 | 6723.43 | 6699.90 | 6689.90 | 6694.90 | 6726.66 | -- | -- | -- | -- | 20.0 | -3.23 | -0.1615 | |
| | 1/25/2023 | 6719.90 | 6709.90 | 6714.90 | 6723.40 | 6699.90 | 6689.90 | 6694.90 | 6726.55 | -- | -- | -- | -- | 20.0 | -3.15 | -0.1575 | |
| | 3/1/2023 | 6719.90 | 6709.90 | 6714.90 | 6723.47 | 6699.90 | 6689.90 | 6694.90 | 6726.78 | -- | -- | -- | -- | 20.0 | -3.31 | -0.1655 | |
| | 3/29/2023 | 6719.90 | 6709.90 | 6714.90 | 6723.45 | 6699.90 | 6689.90 | 6694.90 | 6726.79 | -- | -- | -- | -- | 20.0 | -3.34 | -0.1670 | |
| | 4/26/2023 | 6719.90 | 6709.90 | 6714.90 | 6723.34 | 6699.90 | 6689.90 | 6694.90 | 6726.69 | -- | -- | -- | -- | 20.0 | -3.35 | -0.1675 | |
| OW-189-U/OW-189-M | 8/17/2022 | 6711.50 | 6701.50 | 6706.50 | 6725.67 | 6690.90 | 6680.90 | 6685.90 | 6725.87 | -- | -- | -- | -- | 20.6 | -0.20 | -0.0098 | **-0.0207** |
| | 9/20/2022 | 6711.50 | 6701.50 | 6706.50 | 6725.68 | 6690.90 | 6680.90 | 6685.90 | 6725.79 | -- | -- | -- | -- | 20.6 | -0.11 | -0.0054 | |
| | 10/27/2022 | 6711.50 | 6701.50 | 6706.50 | 6725.57 | 6690.90 | 6680.90 | 6685.90 | 6725.70 | -- | -- | -- | -- | 20.6 | -0.13 | -0.0064 | |
| | 11/29/2022 | 6711.50 | 6701.50 | 6706.50 | 6725.55 | 6690.90 | 6680.90 | 6685.90 | 6725.73 | -- | -- | -- | -- | 20.6 | -0.18 | -0.0088 | |
| | 12/29/2022 | 6711.50 | 6701.50 | 6706.50 | 6725.54 | 6690.90 | 6680.90 | 6685.90 | 6725.88 | -- | -- | -- | -- | 20.6 | -0.34 | -0.0166 | |
| | 1/24/2023 | 6711.50 | 6701.50 | 6706.50 | 6725.46 | 6690.90 | 6680.90 | 6685.90 | 6726.04 | -- | -- | -- | -- | 20.6 | -0.58 | -0.0282 | |
| | 2/27/2023 | 6711.50 | 6701.50 | 6706.50 | 6725.50 | 6690.90 | 6680.90 | 6685.90 | 6726.18 | -- | -- | -- | -- | 20.6 | -0.68 | -0.0331 | |
| | 3/29/2023 | 6711.50 | 6701.50 | 6706.50 | 6725.47 | 6690.90 | 6680.90 | 6685.90 | 6726.19 | -- | -- | -- | -- | 20.6 | -0.72 | -0.0350 | |
| | 4/25/2023 | 6711.50 | 6701.50 | 6706.50 | 6725.30 | 6690.90 | 6680.90 | 6685.90 | 6726.19 | -- | -- | -- | -- | 20.6 | -0.89 | -0.0433 | |


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**Table 2.5-38 Monthly Vertical Hydraulic Gradients**
**(Sheet 11 of 11)**

| Well Pair | Date | Upper Zone | | | | Middle Zone | | | | Lower Zone | | | | Δz (ft)" | Δh (ft)" | iv (ft/ft)" | Average iv (ft/ft) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | Top of screen (NAVD88) | Bottom of screen (NAVD88) | Midpoint (NAVD88) | Elevation of Water (NAVD88) | | | | |
| OW-190-U/OW-190-M | 8/17/2022 | 6695.30 | 6685.30 | 6690.30 | 6724.94 | 6680.30 | 6670.30 | 6675.30 | 6724.24 | -- | -- | -- | -- | 15.0 | 0.70 | 0.0464 | **0.0455** |
| | 9/20/2022 | 6695.30 | 6685.30 | 6690.30 | 6724.98 | 6680.30 | 6670.30 | 6675.30 | 6724.41 | -- | -- | -- | -- | 15.0 | 0.57 | 0.0377 | |
| | 10/26/2022 | 6695.30 | 6685.30 | 6690.30 | 6725.14 | 6680.30 | 6670.30 | 6675.30 | 6724.38 | -- | -- | -- | -- | 15.0 | 0.76 | 0.0504 | |
| | 11/29/2022 | 6695.30 | 6685.30 | 6690.30 | 6725.03 | 6680.30 | 6670.30 | 6675.30 | 6724.29 | -- | -- | -- | -- | 15.0 | 0.74 | 0.0491 | |
| | 12/29/2022 | 6695.30 | 6685.30 | 6690.30 | 6725.04 | 6680.30 | 6670.30 | 6675.30 | 6724.30 | -- | -- | -- | -- | 15.0 | 0.74 | 0.0491 | |
| | 1/25/2023 | 6695.30 | 6685.30 | 6690.30 | 6724.77 | 6680.30 | 6670.30 | 6675.30 | 6724.14 | -- | -- | -- | -- | 15.0 | 0.63 | 0.0417 | |
| | 2/27/2023 | 6695.30 | 6685.30 | 6690.30 | 6725.02 | 6680.30 | 6670.30 | 6675.30 | 6724.31 | -- | -- | -- | -- | 15.0 | 0.71 | 0.0471 | |
| | 3/29/2023 | 6695.30 | 6685.30 | 6690.30 | 6724.92 | 6680.30 | 6670.30 | 6675.30 | 6724.26 | -- | -- | -- | -- | 15.0 | 0.66 | 0.0437 | |
| | 4/25/2023 | 6695.30 | 6685.30 | 6690.30 | 6724.85 | 6680.30 | 6670.30 | 6675.30 | 6724.18 | -- | -- | -- | -- | 15.0 | 0.67 | 0.0444 | |
| OW-191-U/OW-191-M | 8/17/2022 | 6697.60 | 6687.60 | 6692.60 | 6726.53 | 6682.40 | 6672.40 | 6677.40 | 6726.55 | -- | -- | -- | -- | 15.2 | -0.01 | -0.0009 | **-0.0026** |
| | 9/21/2022 | 6697.60 | 6687.60 | 6692.60 | 6726.57 | 6682.40 | 6672.40 | 6677.40 | 6726.59 | -- | -- | -- | -- | 15.2 | -0.01 | -0.0009 | |
| | 10/26/2022 | 6697.60 | 6687.60 | 6692.60 | 6726.81 | 6682.40 | 6672.40 | 6677.40 | 6726.84 | -- | -- | -- | -- | 15.2 | -0.02 | -0.0016 | |
| | 11/30/2022 | 6697.60 | 6687.60 | 6692.60 | 6726.98 | 6682.40 | 6672.40 | 6677.40 | 6727.03 | -- | -- | -- | -- | 15.2 | -0.04 | -0.0029 | |
| | 12/29/2022 | 6697.60 | 6687.60 | 6692.60 | 6727.12 | 6682.40 | 6672.40 | 6677.40 | 6727.18 | -- | -- | -- | -- | 15.2 | -0.05 | -0.0036 | |
| | 1/25/2023 | 6697.60 | 6687.60 | 6692.60 | 6727.12 | 6682.40 | 6672.40 | 6677.40 | 6727.17 | -- | -- | -- | -- | 15.2 | -0.04 | -0.0029 | |
| | 3/1/2023 | 6697.60 | 6687.60 | 6692.60 | 6727.45 | 6682.40 | 6672.40 | 6677.40 | 6727.51 | -- | -- | -- | -- | 15.2 | -0.05 | -0.0036 | |
| | 3/29/2023 | 6697.60 | 6687.60 | 6692.60 | 6727.26 | 6682.40 | 6672.40 | 6677.40 | 6727.32 | -- | -- | -- | -- | 15.2 | -0.05 | -0.0036 | |
| | 4/26/2023 | 6697.60 | 6687.60 | 6692.60 | 6727.18 | 6682.40 | 6672.40 | 6677.40 | 6727.24 | -- | -- | -- | -- | 15.2 | -0.05 | -0.0036 | |


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**Table 2.5-39 Hydrostatic Uplift Pressures for Plant Foundations and Other Safety-Significant SSCs**

| Plant Foundation or SSC | Maximum Groundwater Head, h (ft) | Foundation/SSC Base Elevation, z (ft) | Hydrostatic Uplift Pressure, $p_w$ (lb/ft$^2$) |
|---|---|---|---|
| **Safety-Related** | | | |
| Reactor Building (RXB) | 6737 | 6702 | 2184 |
| Reactor Shaft (RXB Shaft) | 6738 | 6639 | 6178 |
| Fuel Handling Building (FHB North) | 6738 | 670 | 2122 |
| Fuel Handling Building (FHB South) | 6738 | 6710 | 1747 |
| Nuclear Island Control Building (NCB) | 6737 | 6724 | 811 |
| **Non-Safety-Related with Special Treatment (NSRST)** | | | |
| Reactor Auxiliary Building (RAB) | 6737 | 6723 | 874 |
| Reactor Auxiliary Building Tanks (RAB Tanks) | 6737 | 6693 | 2746 |

Notes:

Groundwater density assumed to be 62.40 pcf.


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**Table 2.5-40 Radionuclides of Interest**

| Parent Radionuclide | Progeny Radionuclide | Radioactive Half-Life[1], $t_{1/2}$ | Waste Holdup Tank Activity Concentration, $C_0$ (µCi/cm$^3$) | Effluent Concentration Limit, ECL (µCi/cm$^3$) | Distribution Coefficient, $K_d$ (g/cm$^3$) |
|---|---|---|---|---|---|
| Sr-90 | | 28.79 y | 4.78E-08 | 5.00E-07 | 3.09 |
| | Y-90 | 64.1 h | 4.78E-08 | 7.00E-06 | 1309 |
| Ru-106 | | 373.59 d | 5.52E-09 | 3.00E-06 | 10.43 |
| | Rh-106 | 29.8 s | 5.53E-09 | NA | ND |
| Sb-125 | | 2.75856 y | 1.52E-07 | 3.00E-05 | 32.37 |
| | Te-125m | 57.4 d | 2.13E-09 | 2.00E-05 | 1298 |
| Te-129m | | 33.6 d | 9.39E-11 | 7.00E-06 | 1298 |
| | Te-129 | 69.6 m | 8.53E-14 | 4.00E-04 | 1298 |
| | I-129 | 1.57E+07 y | 1.40E-10 | 2.00E-07 | 4.71 |
| Cs-134 | | 2.0648 y | 3.14E-05 | 9.00E-07 | 883 |
| Cs-137 | | 30.1671 y | 3.30E-04 | 1.00E-06 | 883 |
| | Ba-137m | 2.552 m | 6.25E-08 | NA | ND |
| Ce-144 | | 284.91 d | 6.50E-09 | 3.00E-06 | 2817 |
| | Pr-144m | 7.2 m | 6.19E-11 | NA | ND |
| | Pr-144 | 17.28 m | 6.50E-09 | 6.00E-04 | ND |
| Pu-238 | | 87.7 y | 2.04E-09 | 2.00E-08 | 632 |
| Pu-239 | | 2.41E+04 y | 7.15E-10 | 2.00E-08 | 632 |
| Pu-240 | | 6564 y | 2.84E-10 | 2.00E-08 | 632 |
| Pu-241 | | 14.35 y | 4.57E-09 | 1.00E-06 | 632 |
| | Am-241 | 432.2 y | 3.95E-11 | 2.00E-08 | 113 |

NA = not available

ND = not determined

[1]Obtained from Reference 2.5-94



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**Table 2.5-41 Radionuclide Activity Concentrations in Groundwater**

| Parent | Progeny | $C_{max}$ ($\mu Ci/cm^3$) | $t_{max}$ (y) | ECL ($\mu Ci/cm^3$) | $C_{max}$/ECL | Remarks |
|---|---|---|---|---|---|---|
| Sr-90 | | 7.2E-22 | 700 | 5.0E-07 | 1.4E-15 | Equation 2.5.2.3-1 |
| | Y-90 | 7.2E-22 | | 7.0E-06 | 1.0E-16 | Secular equilibrium with Sr-90 |
| Ru-106 | | 2.3E-198 | 350 | 3.0E-06 | 7.7E-193 | Equation 2.5.2.3-1 |
| | Rh-106 | 2.3E-198 | | NA | --- | Secular equilibrium with Ru-106 |
| Sb-125 | | 5.3E-213 | 1,000 | 3.0E-05 | 1.8E-208 | Equation 2.5.2.3-1 |
| | Te-125m | 1.2E-213 | | 2.0E-05 | 6.2E-209 | Secular equilibrium with Sb-125 (corrected by $d_{12}$) |
| Te-129m | | --- | | 7.0E-06 | --- | Eliminated in the source term analysis |
| | Te-129 | --- | | 4.0E-04 | --- | Eliminated in the source term analysis |
| | I-129 | 9.3E-16 | 1,600 | 2.0E-07 | 4.7E-09 | Equation 2.5.2.3-1 |
| Cs-134 | | 0.0E+00 | | 9.0E-07 | 0.0E+00 | Equation 2.5.2.3-1 |
| Cs-137 | | 0.0E+00 | | 1.0E-06 | 0.0E+00 | Equation 2.5.2.3-1 |
| | Ba-137m | 0.0E+00 | | NA | --- | Secular equilibrium with Cs-137 (corrected by d12) |
| Ce-144 | | 0.0E+00 | | 3.0E-06 | 0.0E+00 | Equation 2.5.2.3-1 |
| | Pr-144m | 0.0E+00 | | NA | --- | Secular equilibrium with Ce-144 (corrected by $d_{12}$) |
| | Pr-144 | 0.0E+00 | | 6.0E-04 | 0.0E+00 | Secular equilibrium with Ce-144 (corrected by $d_{13}$) |
| Pu-238 | | 6.4E-169 | 25,000 | 2.0E-08 | 3.2E-161 | Equation 2.5.2.3-1 |
| Pu-239 | | 1.3E-19 | 187,500 | 2.0E-08 | 6.3E-12 | Equation 2.5.2.3-1 |
| Pu-240 | | 1.4E-25 | 150,000 | 2.0E-08 | 7.2E-18 | Equation 2.5.2.3-1 |
| Pu-241 | | 0.0E+00 | | 1.0E-06 | 0.0E+00 | Equation 2.5.2.3-1 |
| | Am-241 | 4.5E-35 | 20,000 | 2.0E-08 | 2.3E-27 | Equation 2.5.2.3-1 |
| | | | | | 4.68E-09 | Sum of Fractions |

NA = not available


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**Table 2.5-42 Summary of Water Demand for Naughton Units 1, 2, and 3 and Kemmerer Unit 1**

| Facility | Water Demand | | |
|---|---|---|---|
| | gpm | cfs | Ac-ft / day |
| Kemmerer Unit 1 | 5270 | 11.7 | 23.3 |
| Naughton Units 1 and 2[1] | 4845 | 10.8 | 21.4 |
| Naughton Unit 3[1] | 5391 | 12.0 | 23.8 |
| Totals | 15,506 | 34.5 | 68.5 |

Note 1: From Reference 2.5-102. A ratio of 1.5 is applied to obtain the maximum monthly value from the average monthly value.



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**Table 2.5-43 Comparison of Total Water Demand from Kemmerer Unit 1 and Naughton Power Plant with Average Monthly Discharge at Hams Fork**

| Month | Monthly Withdrawal at Naughton CWIS (cfs)[1] | Kemmerer Unit 1 Demand (cfs)[2] | Total Water Demand of Kemmerer Unit 1 and Naughton Power Plant (cfs) | Average Monthly Hams Fork Discharge (cfs)[1] | Total Water Demand as Percentage of Hams Fork Discharge |
|---|---|---|---|---|---|
| January | 13.3 | 11.7 | 25.0 | 35.7 | 70% |
| February | 12.6 | 11.7 | 24.3 | 30.9 | 79% |
| March | 13.1 | 11.7 | 24.8 | 29.8 | 83% |
| April | 12.2 | 11.7 | 23.9 | 89.4 | 27% |
| May | 11.7 | 11.7 | 23.4 | 358.3 | 7% |
| June | 17.6 | 11.7 | 29.3 | 407.5 | 7% |
| July | 20.8 | 11.7 | 32.5 | 152.4 | 21% |
| August | 19.2 | 11.7 | 30.9 | 81.6 | 38% |
| September | 18.7 | 11.7 | 30.4 | 97.9 | 31% |
| October | 15.9 | 11.7 | 27.6 | 51.8 | 53% |
| November | 13.9 | 11.7 | 25.6 | 38.7 | 66% |
| December | 13.3 | 11.7 | 25.0 | 36.7 | 68% |

Note 1: From Reference 2.5-102

Note 2: From Table 2.5-42


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**Table 2.5-44 Comparison of Total Water Demand from Kemmerer Unit 1 and Naughton Power Plant with Average Monthly Hams Fork Discharge from USGS Daily Data and Daily Outflow at Viva Naughton Reservoir**

| Month | Total Water Demand of Kemmerer Unit 1 and Naughton Power Plant (cfs)[1] | USGS Stream Gage Data | | Viva Naughton Reservoir Outflow | |
|---|---|---|---|---|---|
| | | Average Monthly Hams Fork Discharge (cfs)[2] | Total Water Demand as a Percentage of Hams Fork Discharge (%) | Average Monthly Hams Fork Discharge (cfs)[3] | Total Water Demand as a Percentage of Hams Fork Discharge (%) |
| January | 25.0 | 38.4 | 65% | 41.2 | 61% |
| February | 24.3 | 33.5 | 73% | 41.3 | 59% |
| March | 24.8 | 32.3 | 77% | 57.7 | 43% |
| April | 23.9 | 93.9 | 25% | 149.3 | 16% |
| May | 23.4 | 364.1 | 6% | 371.7 | 6% |
| June | 29.3 | 413.4 | 7% | 374.5 | 8% |
| July | 32.5 | 159.6 | 20% | 145.5 | 22% |
| August | 30.9 | 79.9 | 39% | 87.4 | 35% |
| September | 30.4 | 100.4 | 30% | 83.8 | 36% |
| October | 27.6 | 52.1 | 53% | 57.1 | 48% |
| November | 25.6 | 39.7 | 65% | 46.0 | 56% |
| December | 25.0 | 39.2 | 64% | 43.7 | 57% |

Note 1: From Table 2.5-102 and Table 2.5-42.

Note 2: From USGS Hams Fork Below Viva Naughton Reservoir (Table 2.5-103)

Note 3: From data provided by Naughton Power Plant covering the period from 1995-2021 for outflow from Viva Naughton Reservoir.


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**Figure 2.5-1 General Location of the Kemmerer Unit 1 Site and Surrounding Streams**

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**Figure 2.5-2 6-Hour Local PMP Depths in 5-minute Increments (including 20 percent increase due to climate change)**

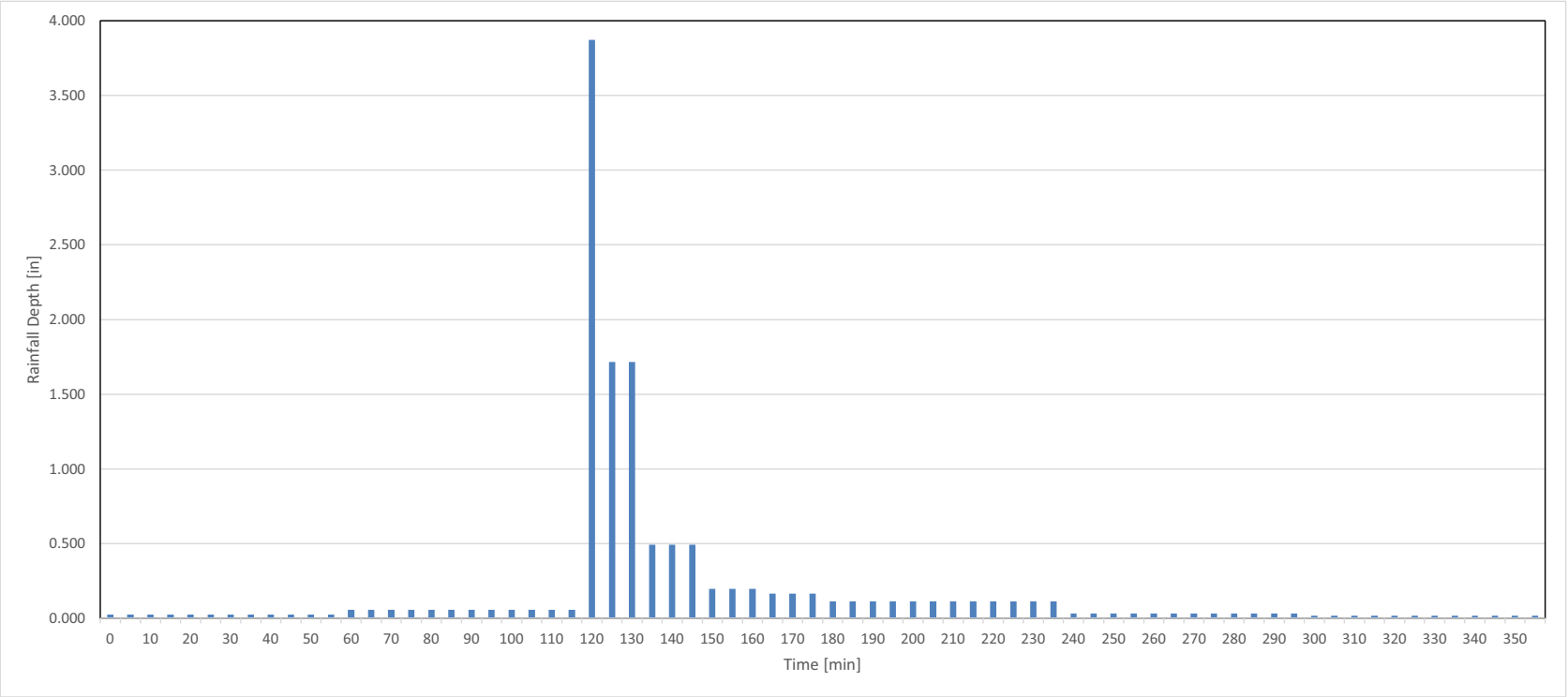



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**Figure 2.5-3 Finish Grade and Safety-Significant Structures**

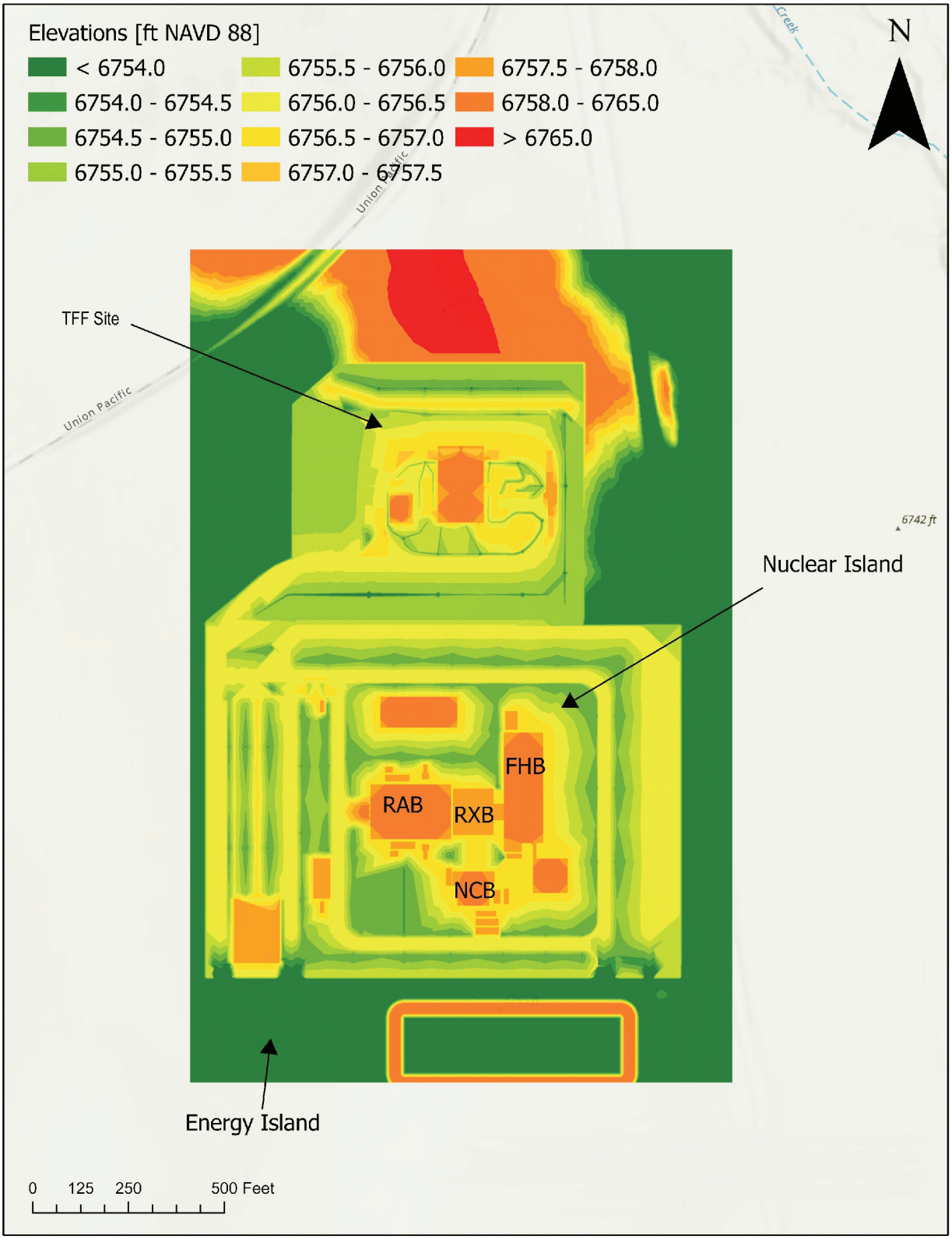



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**Figure 2.5-4 Gravel (Blue colored areas) and Building Roofs, Roads and Parking Lots (Grey colored areas)**

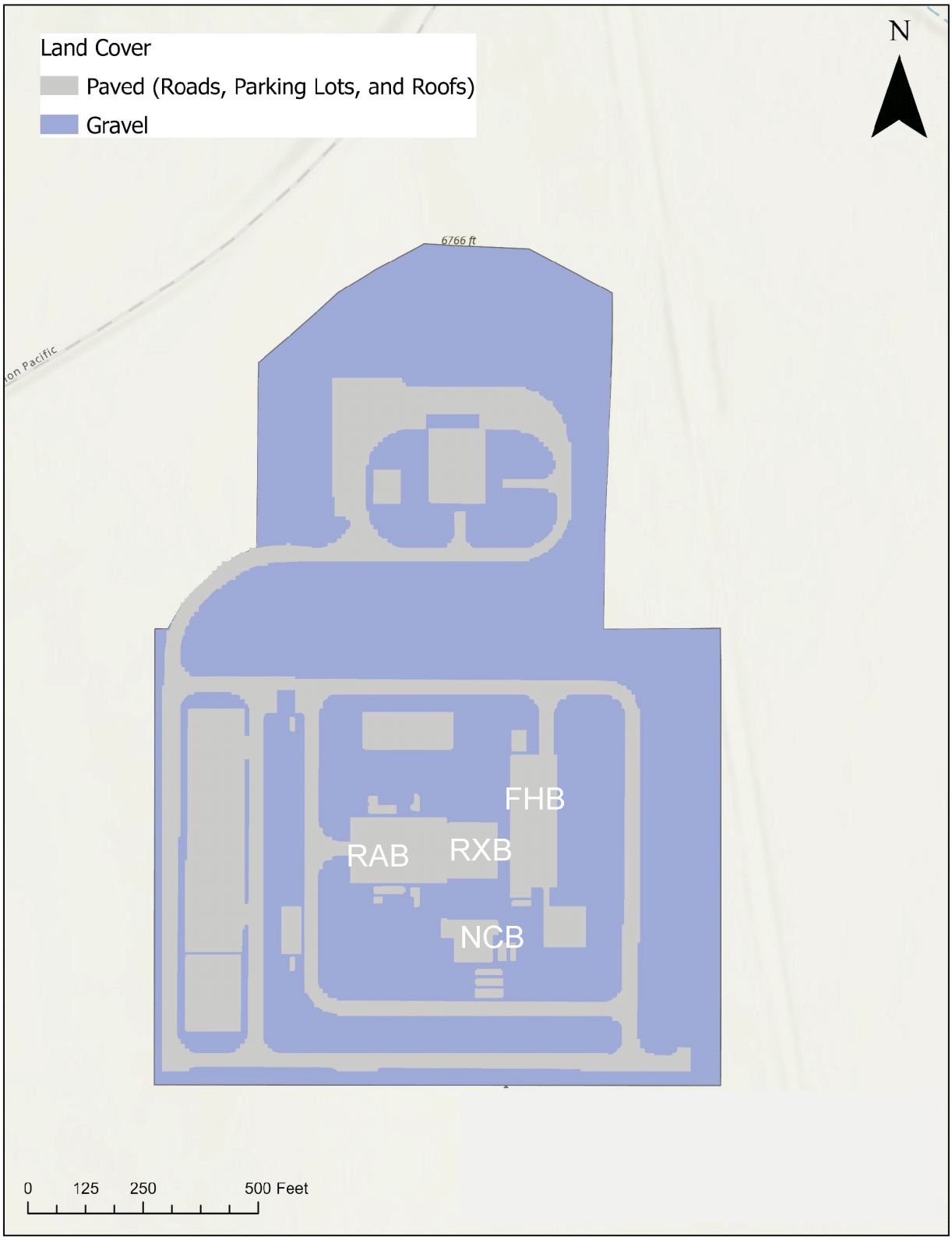



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**Figure 2.5-5 Estimated Maximum Water Surface Elevation (in feet NAVD 88) for LIP with Climate Change Adjustment. Red stars are the locations of the Maximum Water Surface Elevation for each Structure.**

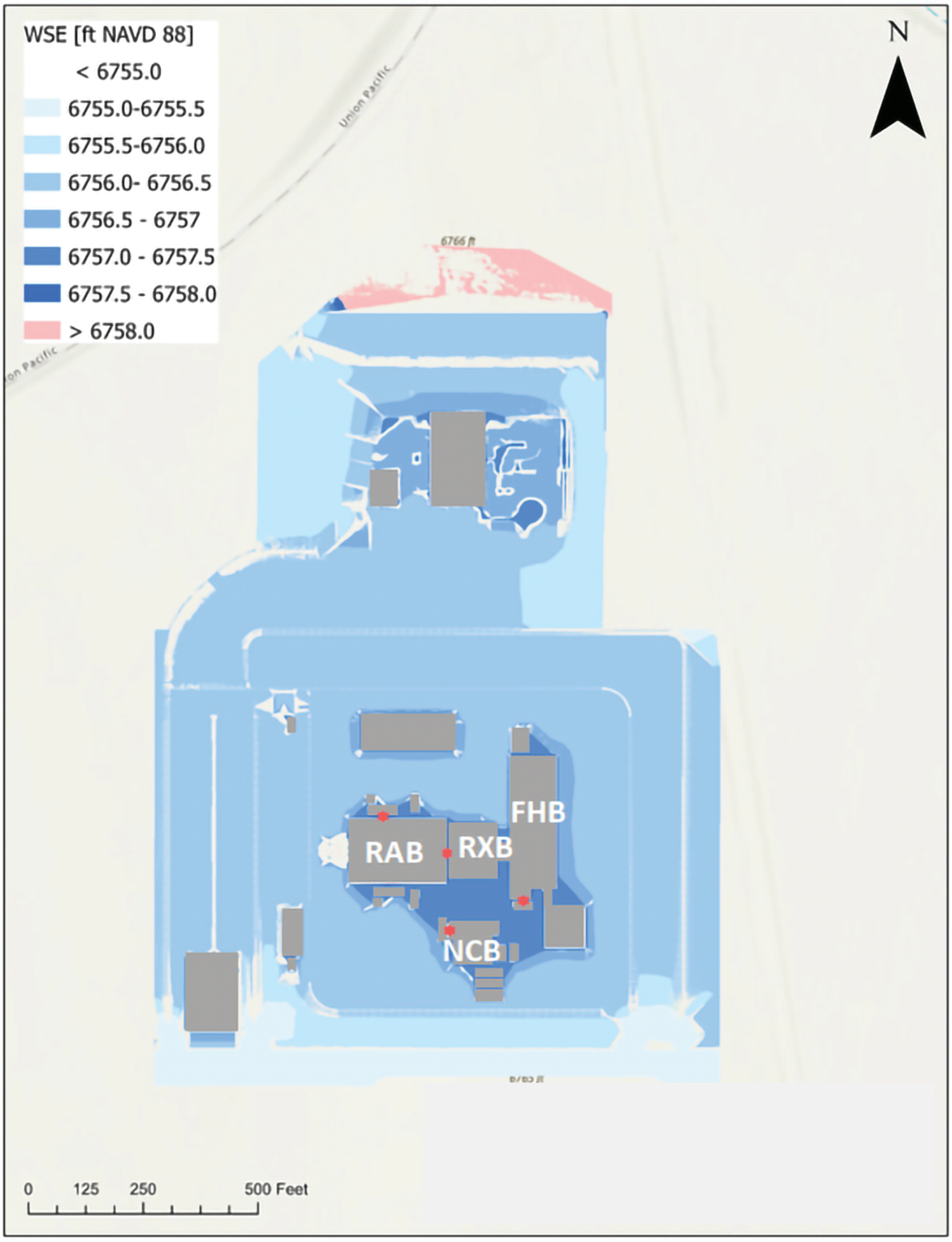



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**Figure 2.5-6 Sub-basins, Streams and Flow Paths at the Kemmerer Unit 1 Site**

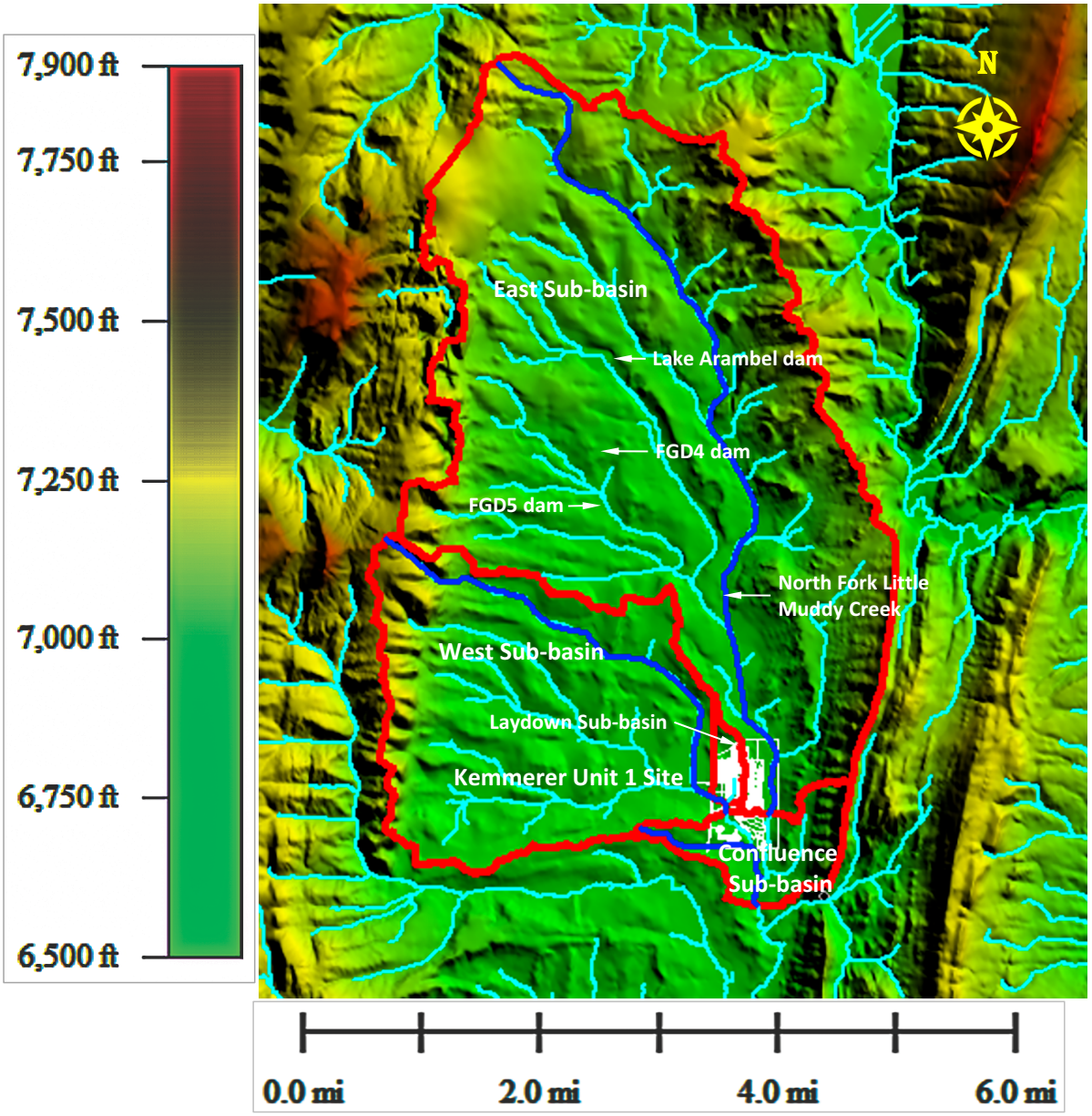



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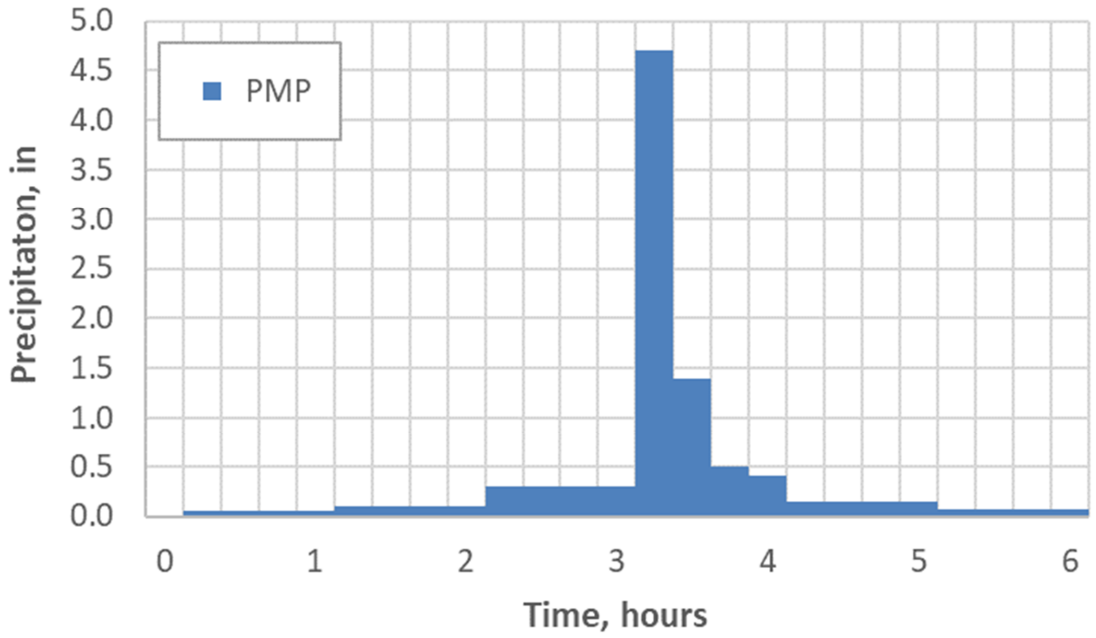
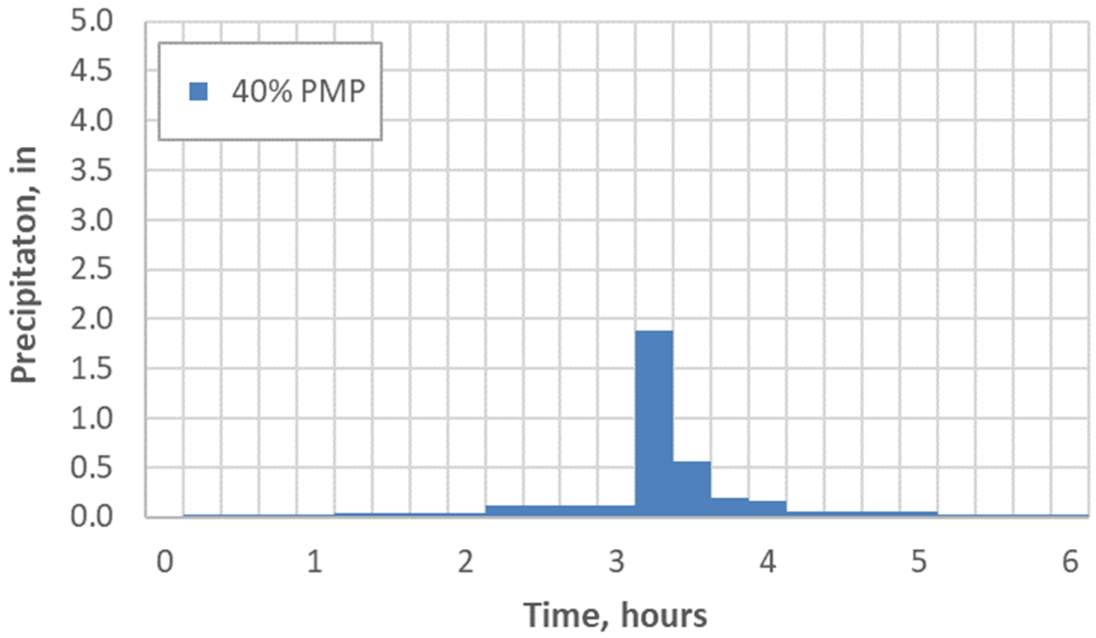
**Figure 2.5-7 Hyetograph of (a) the PMP and (b) the 40% PMP**




(a)




(b)



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**Figure 2.5-8 Sub-basin East Precipitation and Discharge**

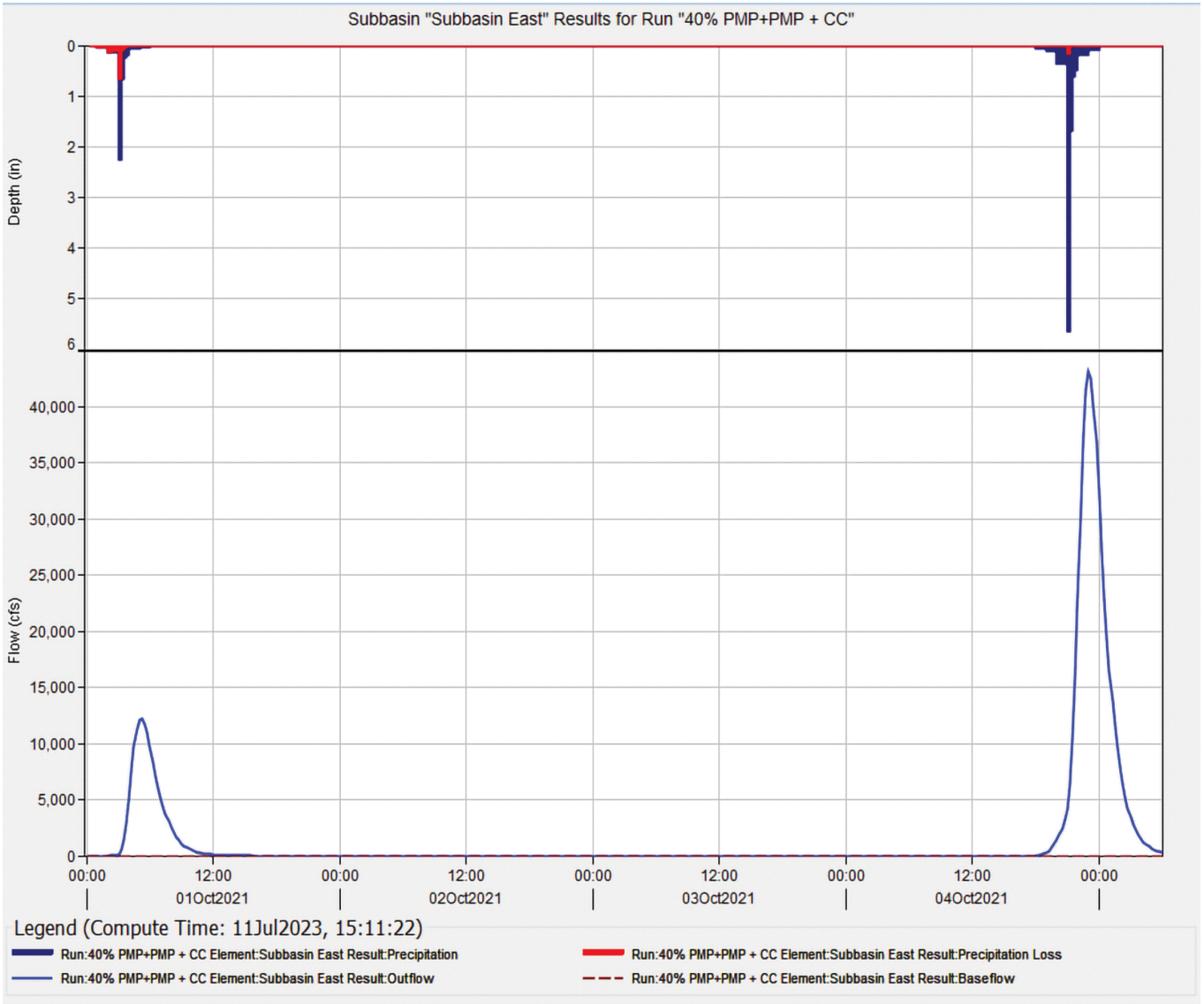



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**Figure 2.5-9 Sub-basin West Precipitation and Discharge**

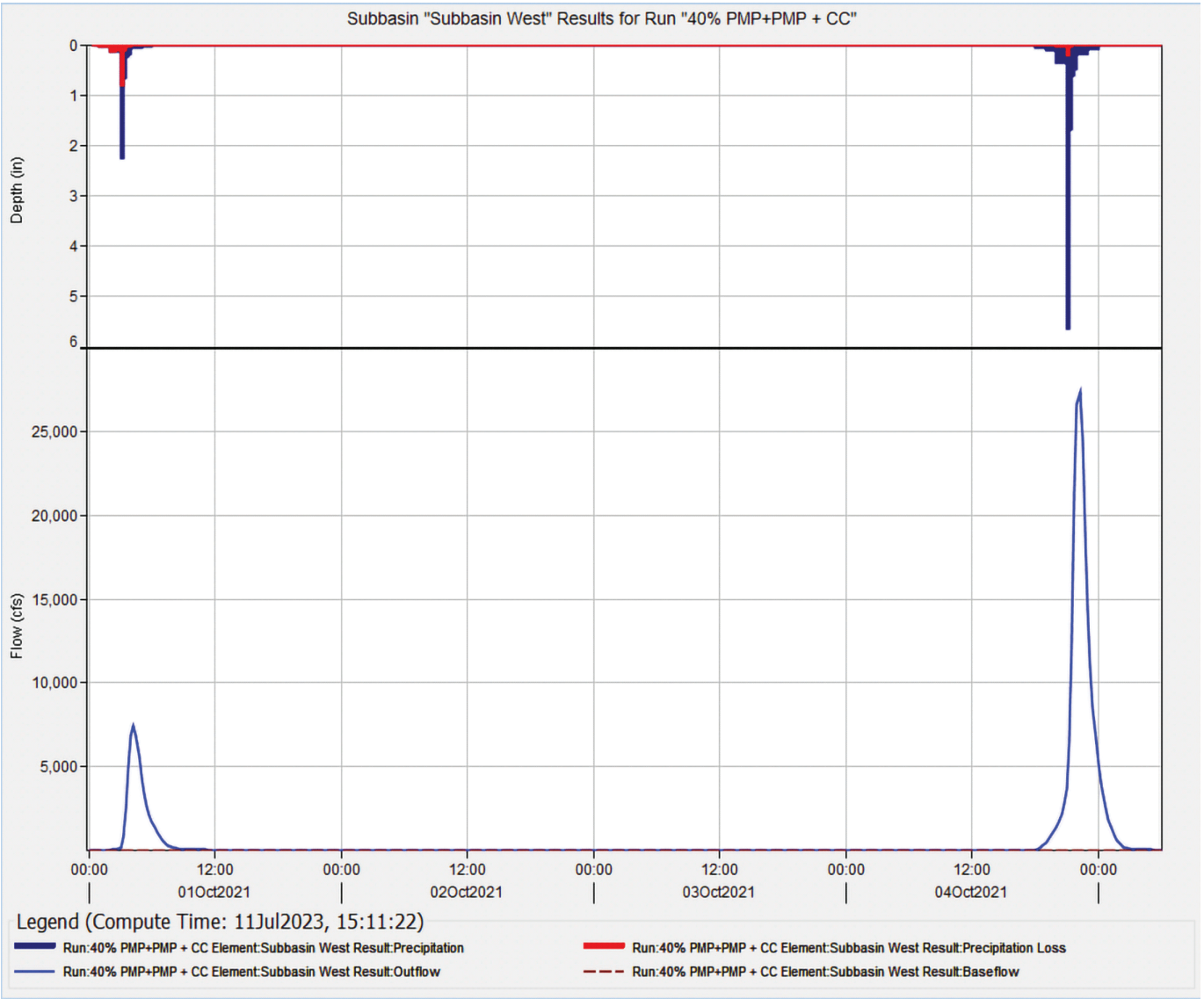




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**Figure 2.5-10 Sub-basin Laydown Precipitation and Discharge**

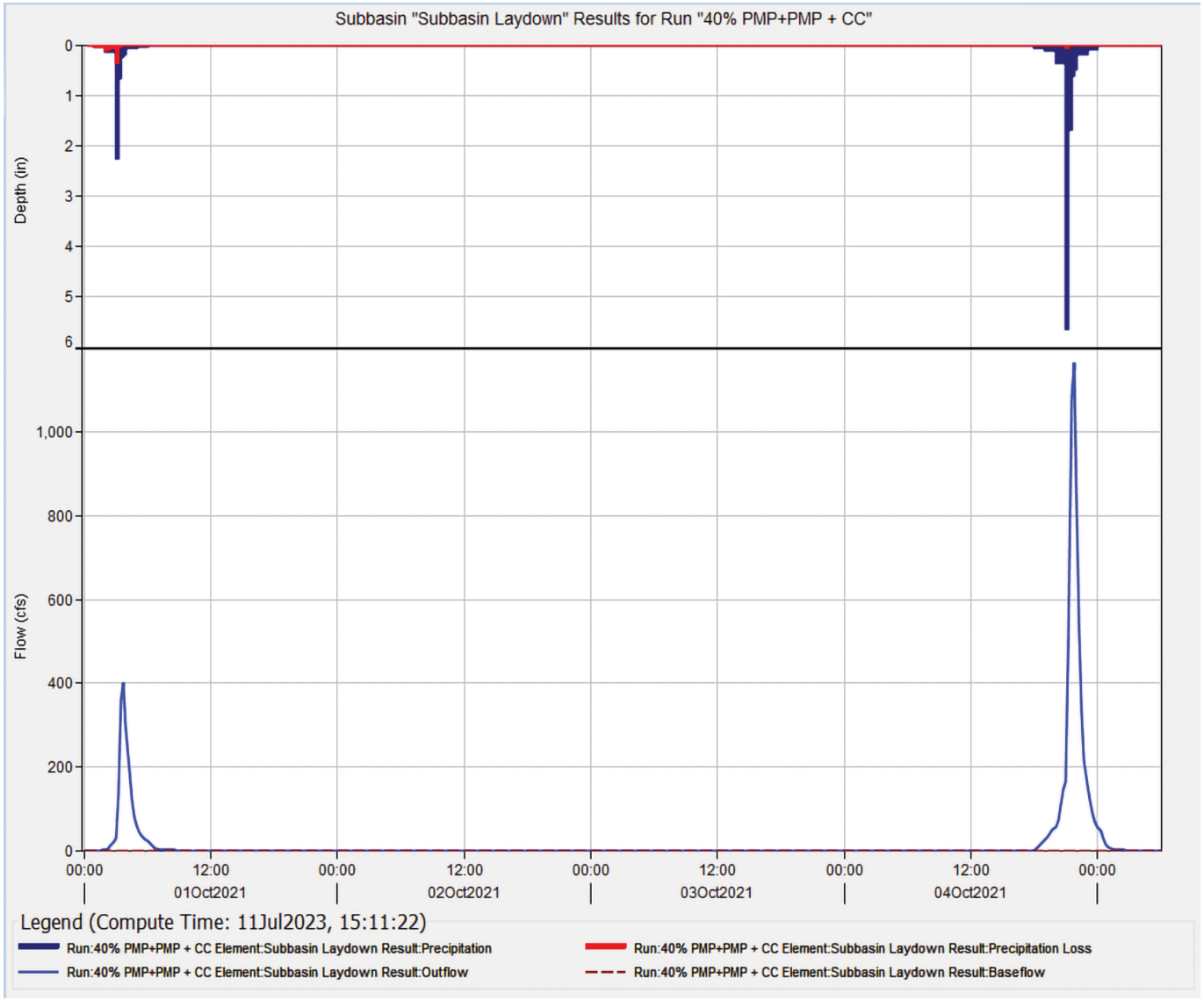



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**Figure 2.5-11 Sub-basin Confluence Precipitation and Discharge**

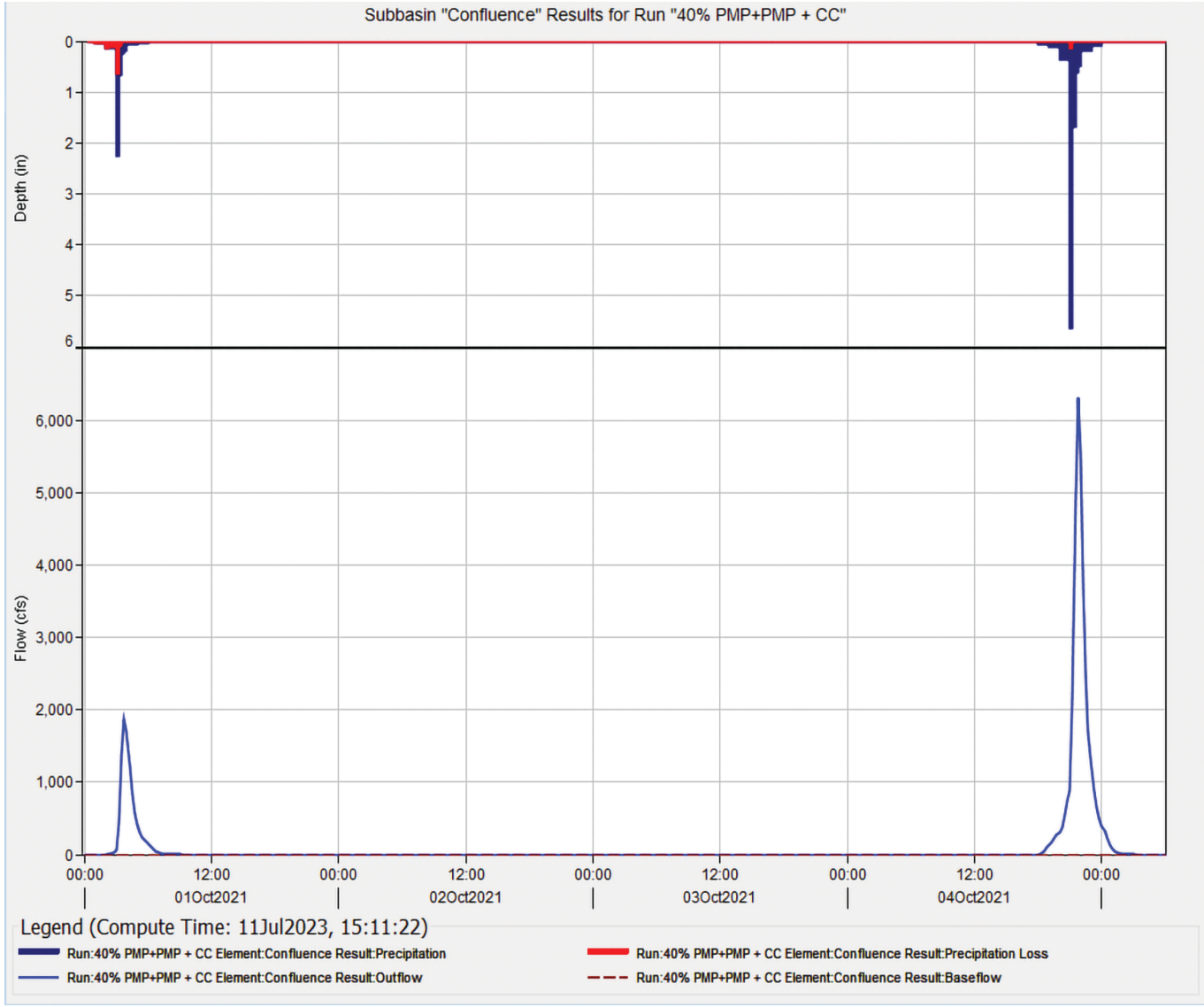



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**Figure 2.5-12 HEC-RAS Model Cross Sections**

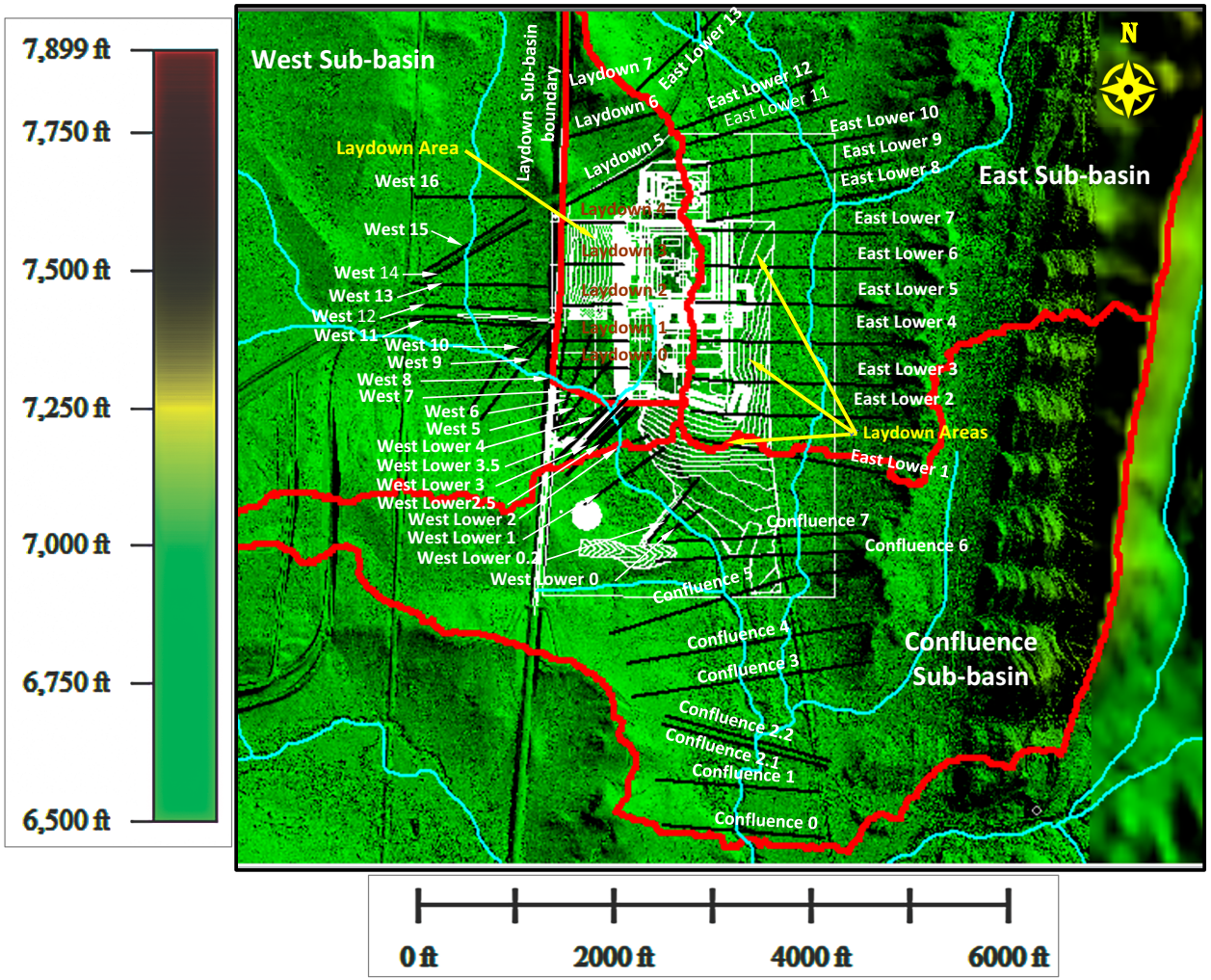



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**Figure 2.5-13 Maximum Water Surface Profile Along East Lower (North Folk Little Muddy Creek) and Confluence Reaches with Climate Change (40%PMP+PMP+CC)**

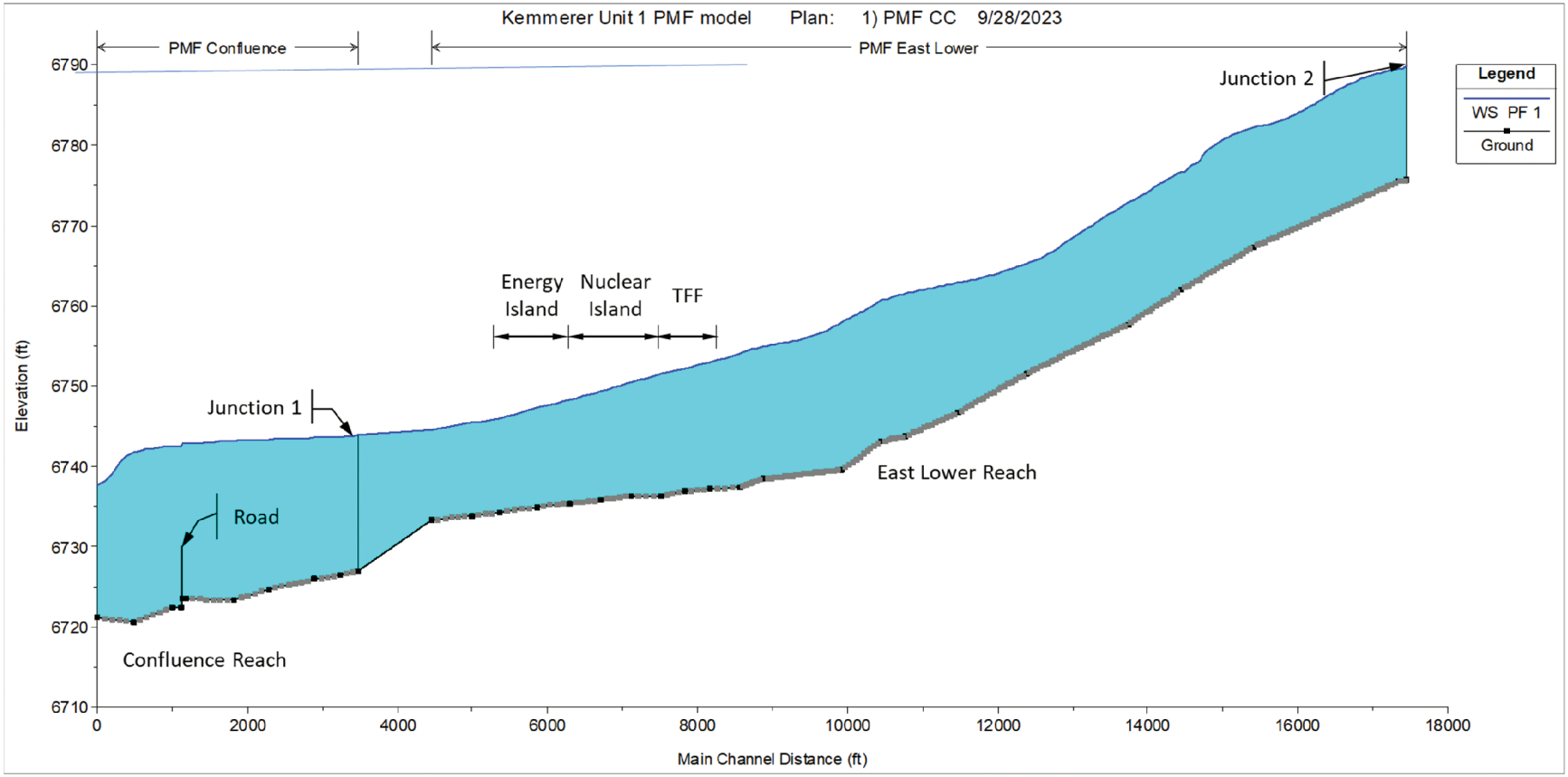



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**Figure 2.5-14 Maximum Water Surface Profile along Laydown, West Lower and Confluence Reaches with Climate Change (40%PMP+PMP+CC)**

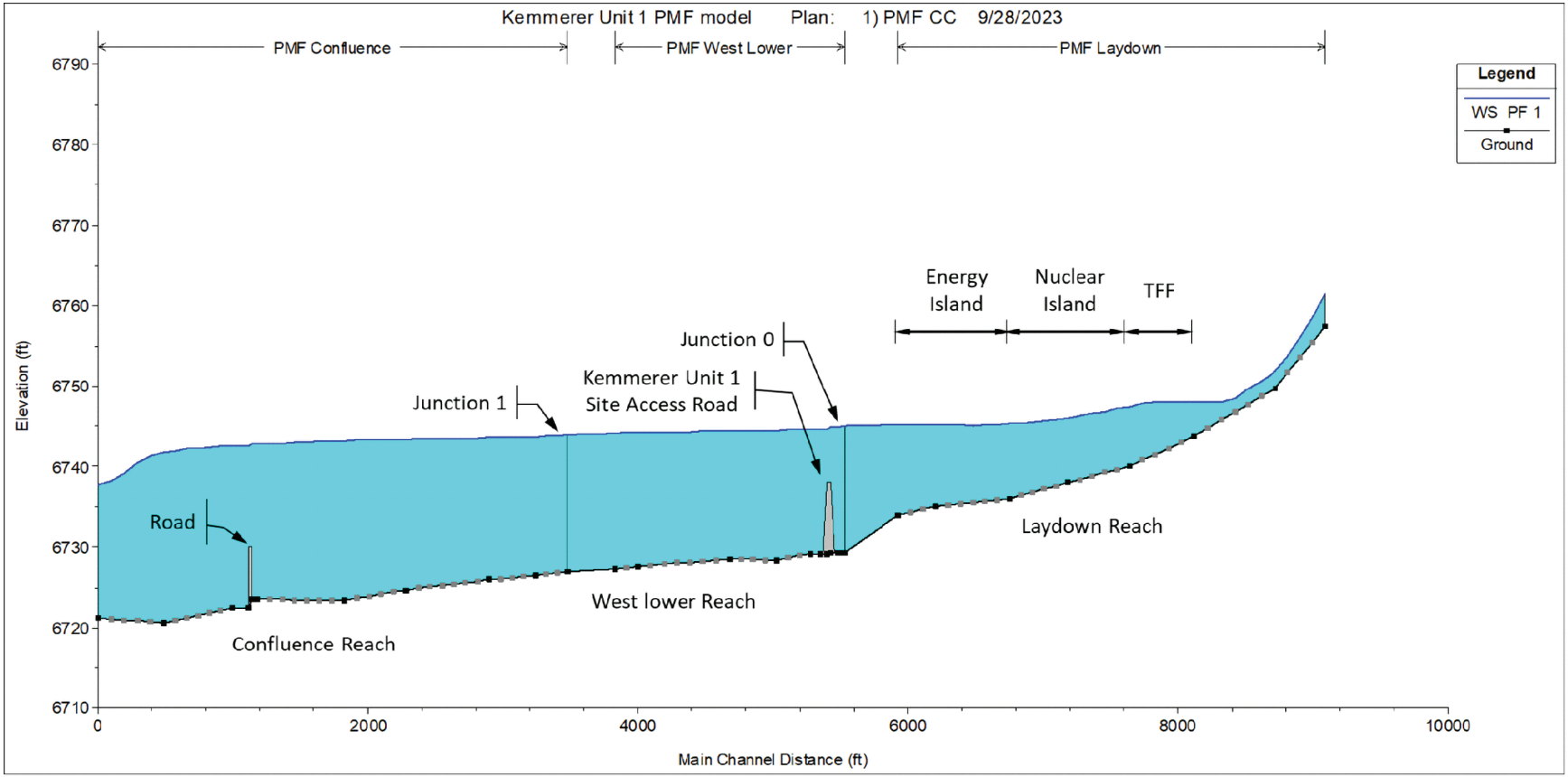



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**Figure 2.5-15 Fetch Length (Yellow Line) at the Kemmerer Unit 1 Site**

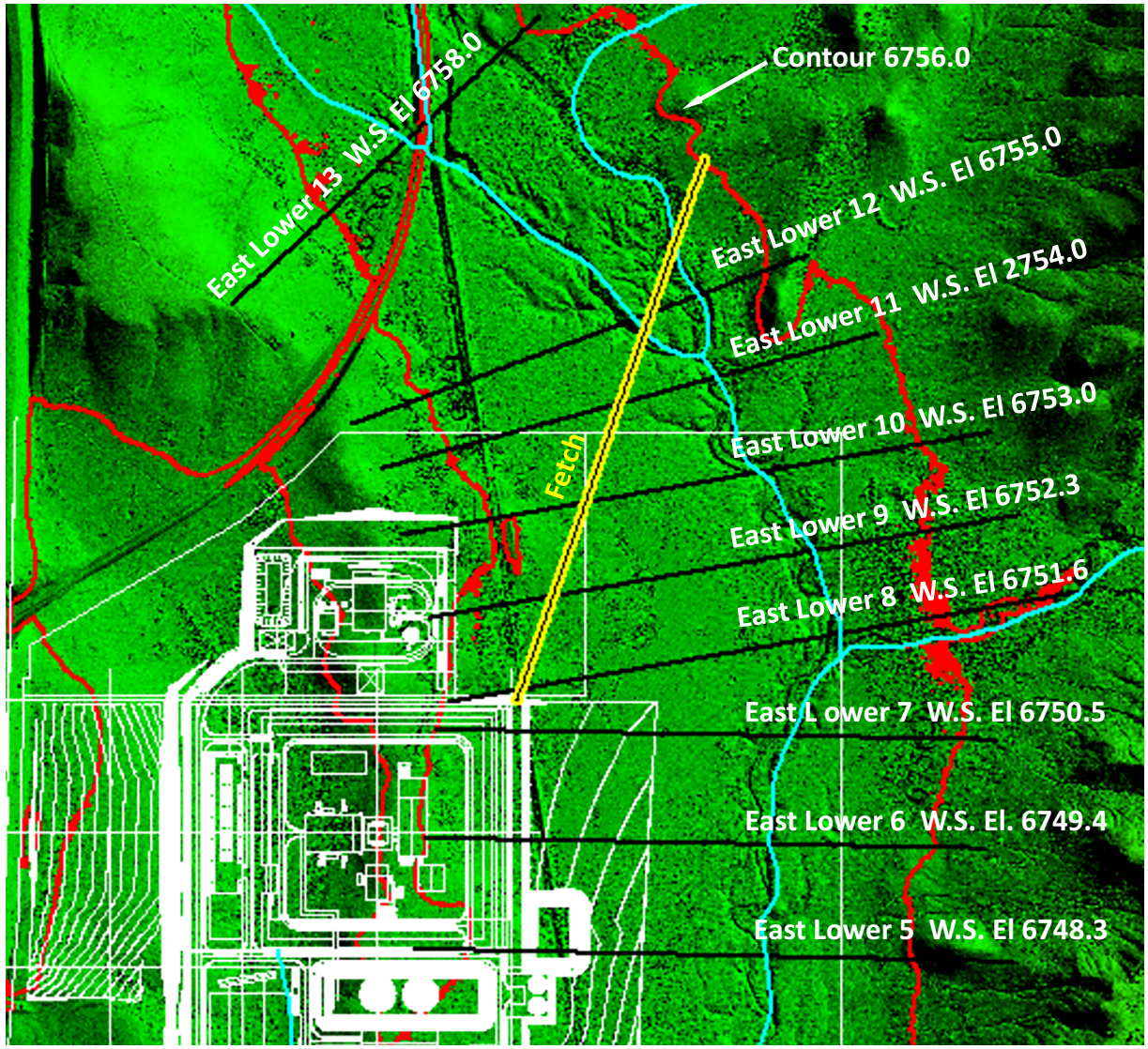




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**Figure 2.5-16 HEC-RAS Cross-Sections at Kemmerer Unit 1 Site**

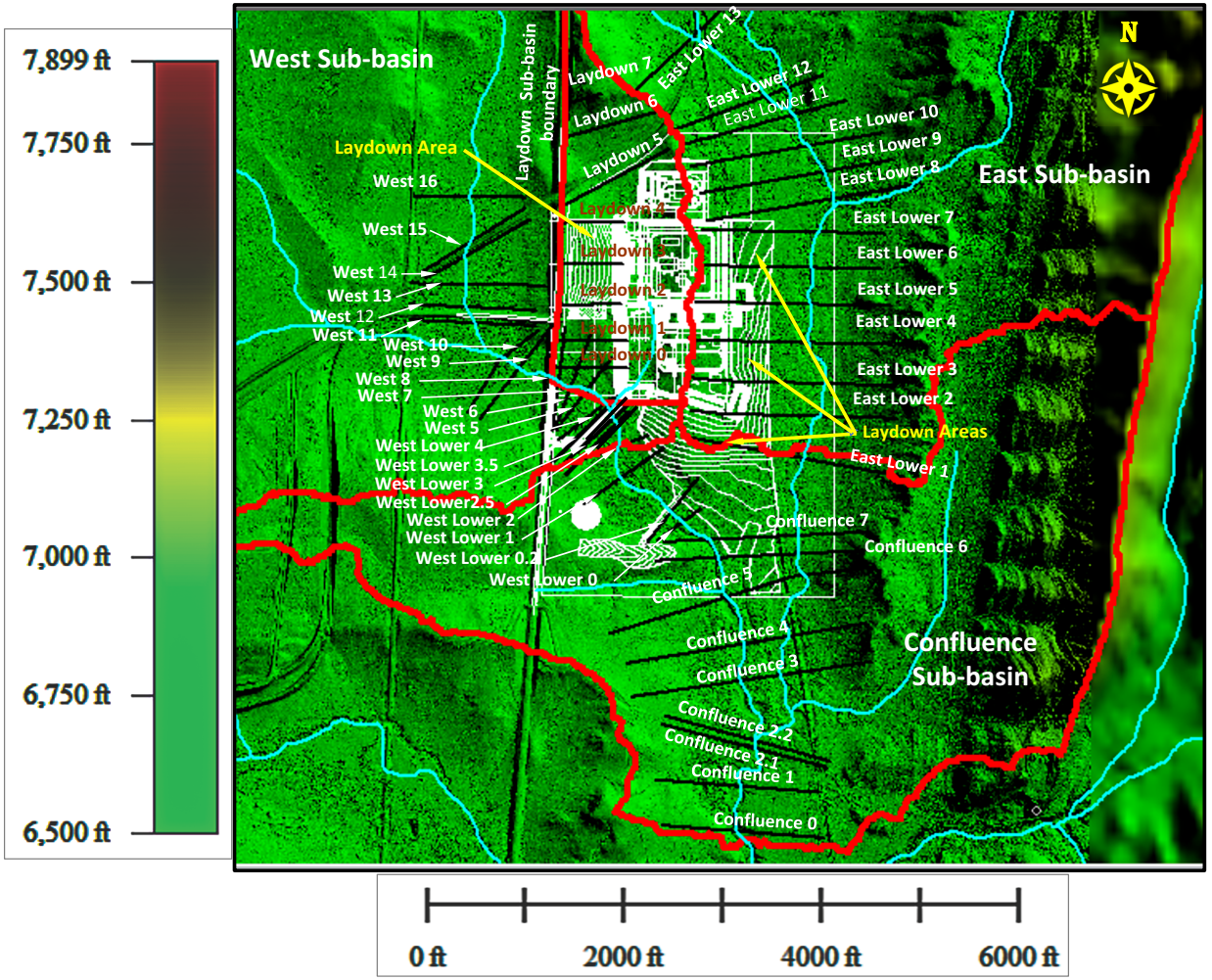



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**Figure 2.5-17 HEC-RAS Cross-Section Locations Between Ponds and Kemmerer Unit 1 Site**

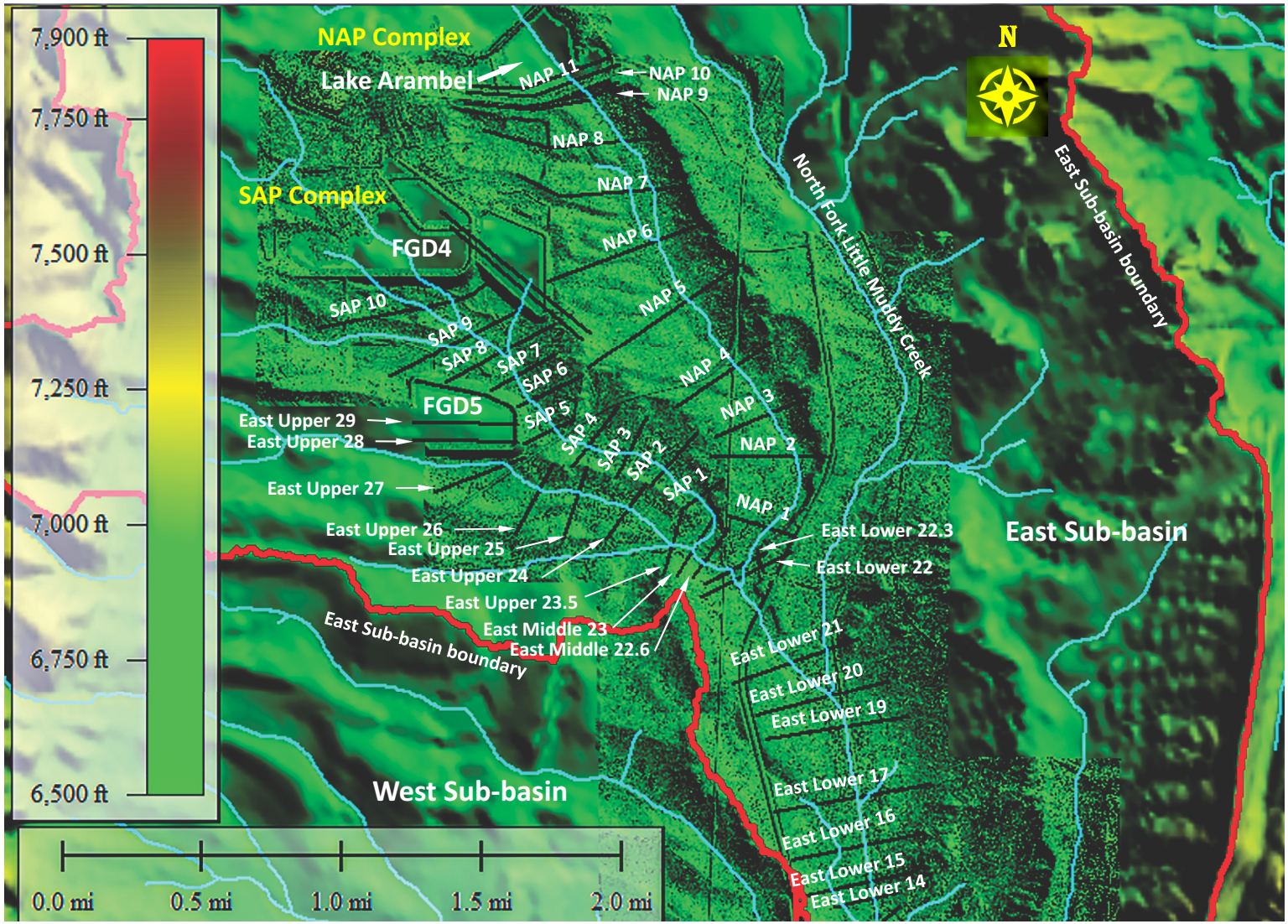



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**Figure 2.5-18 Schematic Representation of the HEC-RAS model**

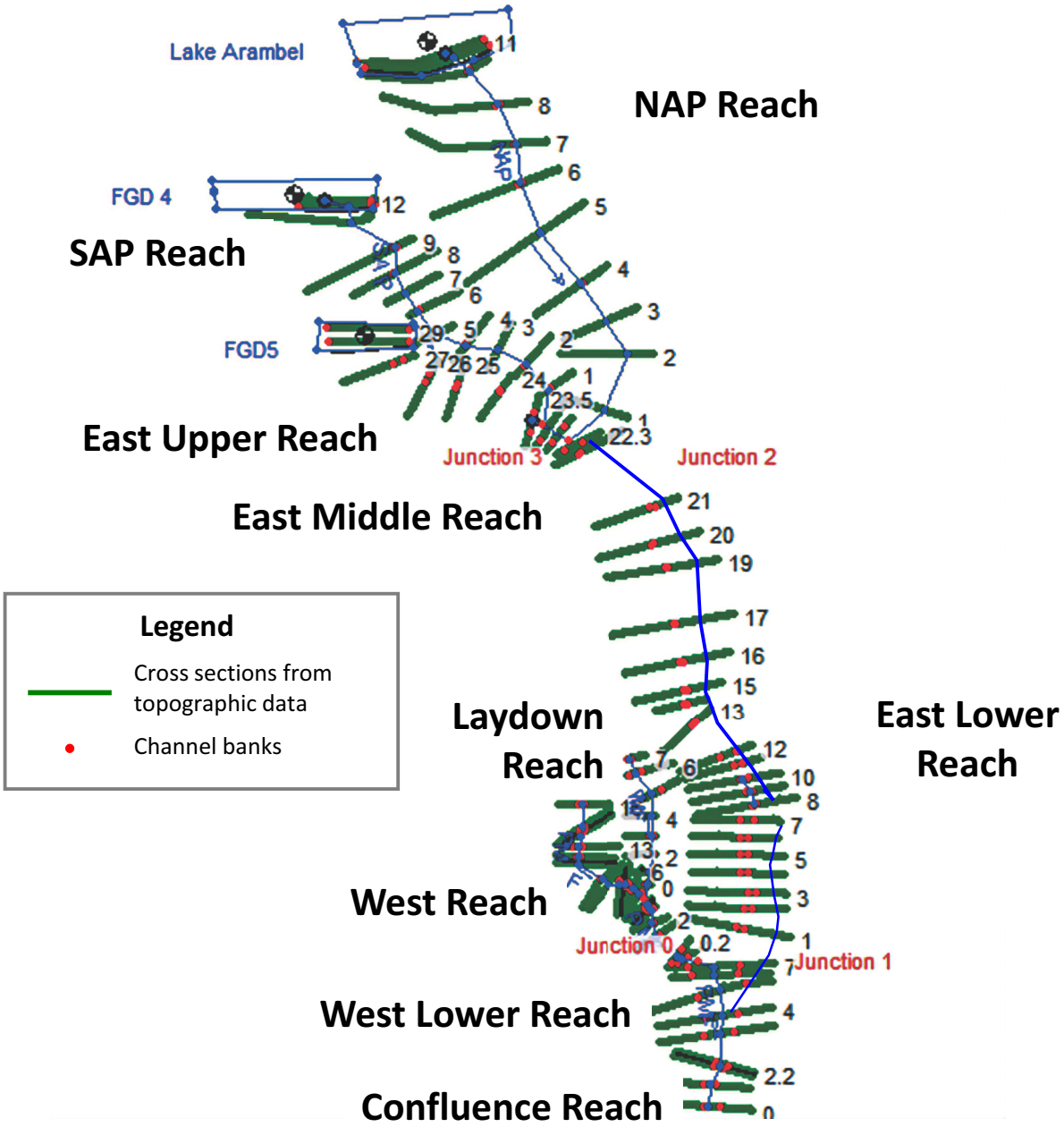



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**Figure 2.5-19 Schematic Representation of the Downstream Part of the HEC-RAS Model Shown in Figure 2.5-18**

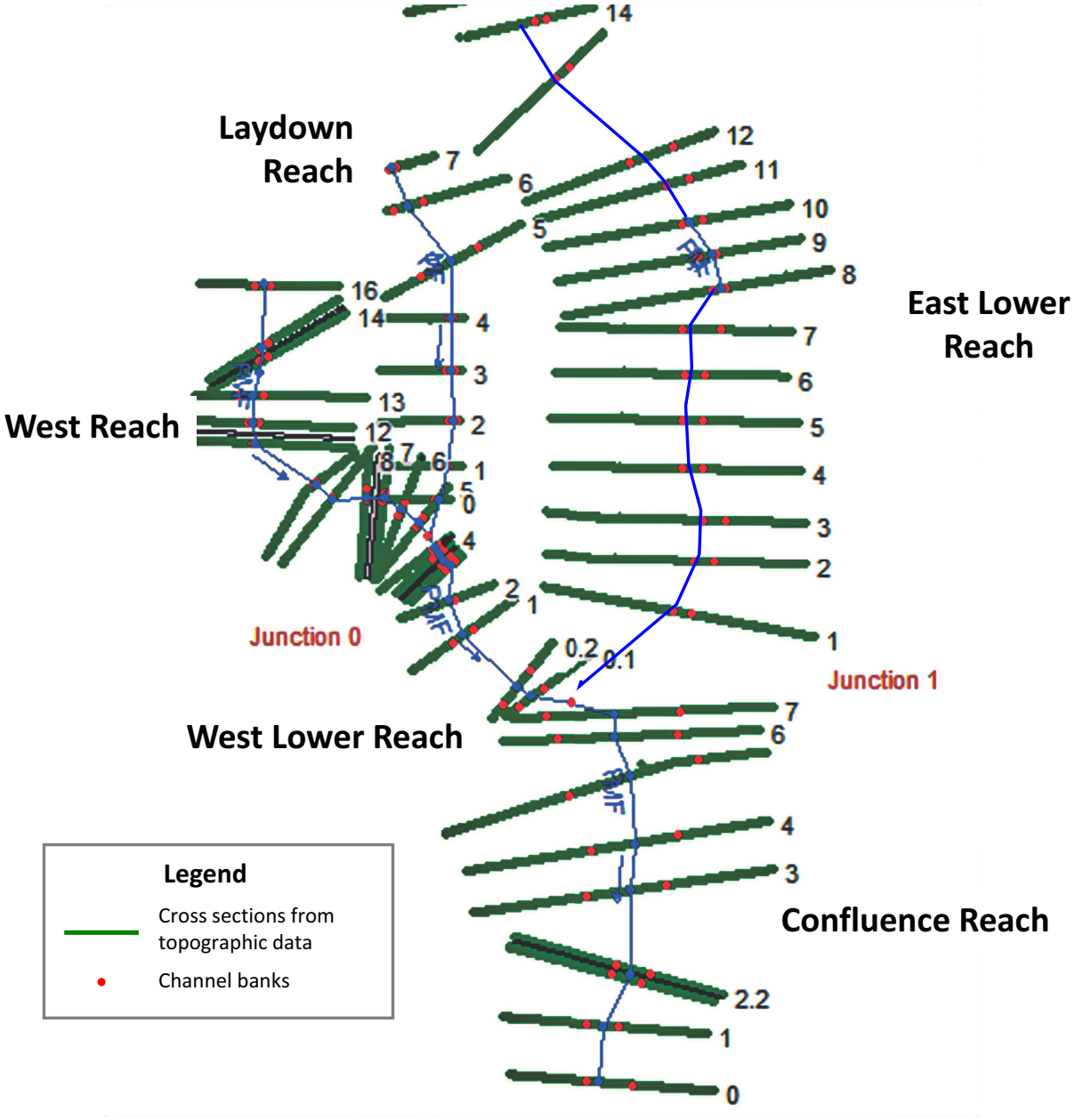



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**Figure 2.5-20 Schematic Representation of the Upstream Part of the HEC-RAS Model Shown in Figure 2.5-18**

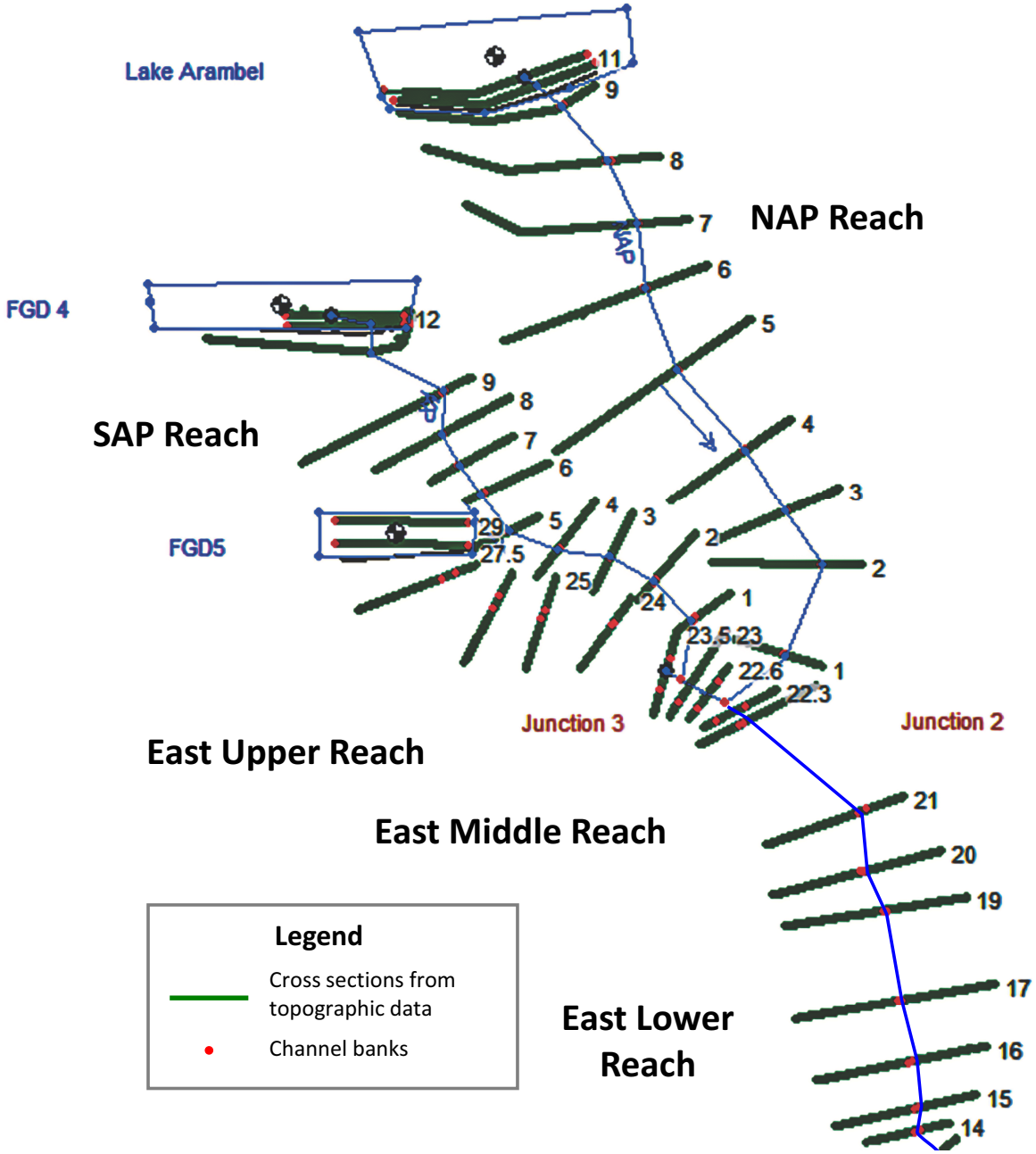



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**Figure 2.5-21 Simultaneous Dam Breach of all Three Dams plus 500-Year Flood Peak Discharge. Stage and Flow Hydrograph at Section East Lower 8, Next to the Nuclear Island**

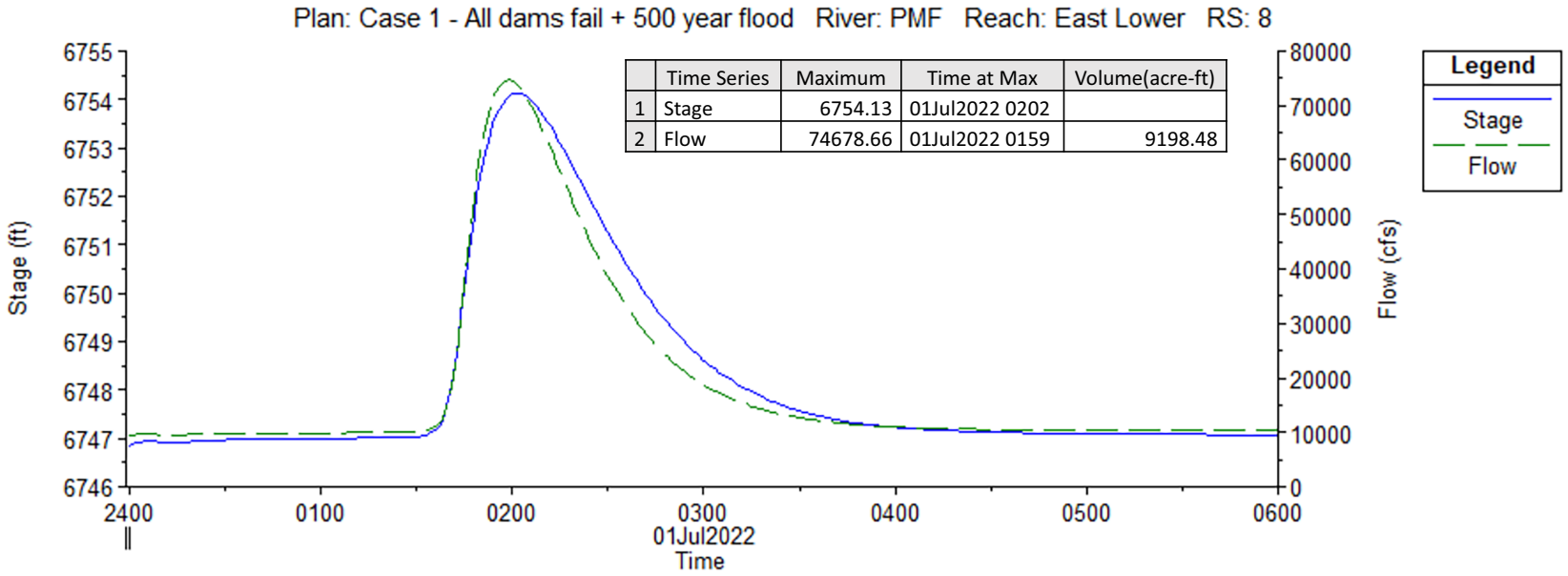



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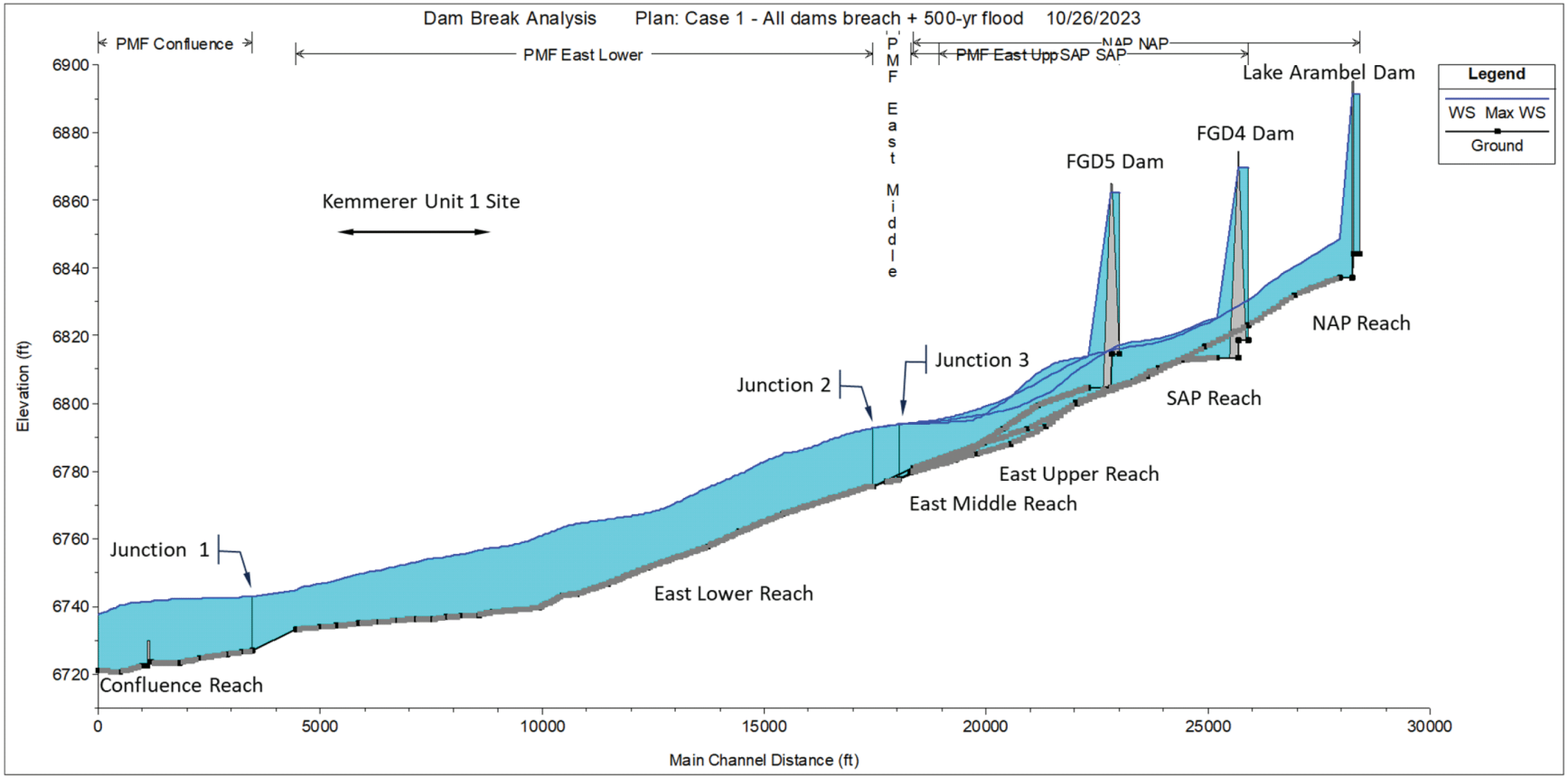
**Figure 2.5-22 Simultaneous Dam Breach of all Three Dams Plus 500-year Flood Peak Discharge. Maximum Water Surface Profile**





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## Figure 2.5-23 Lake Arambel Watershed Lag Flow Path

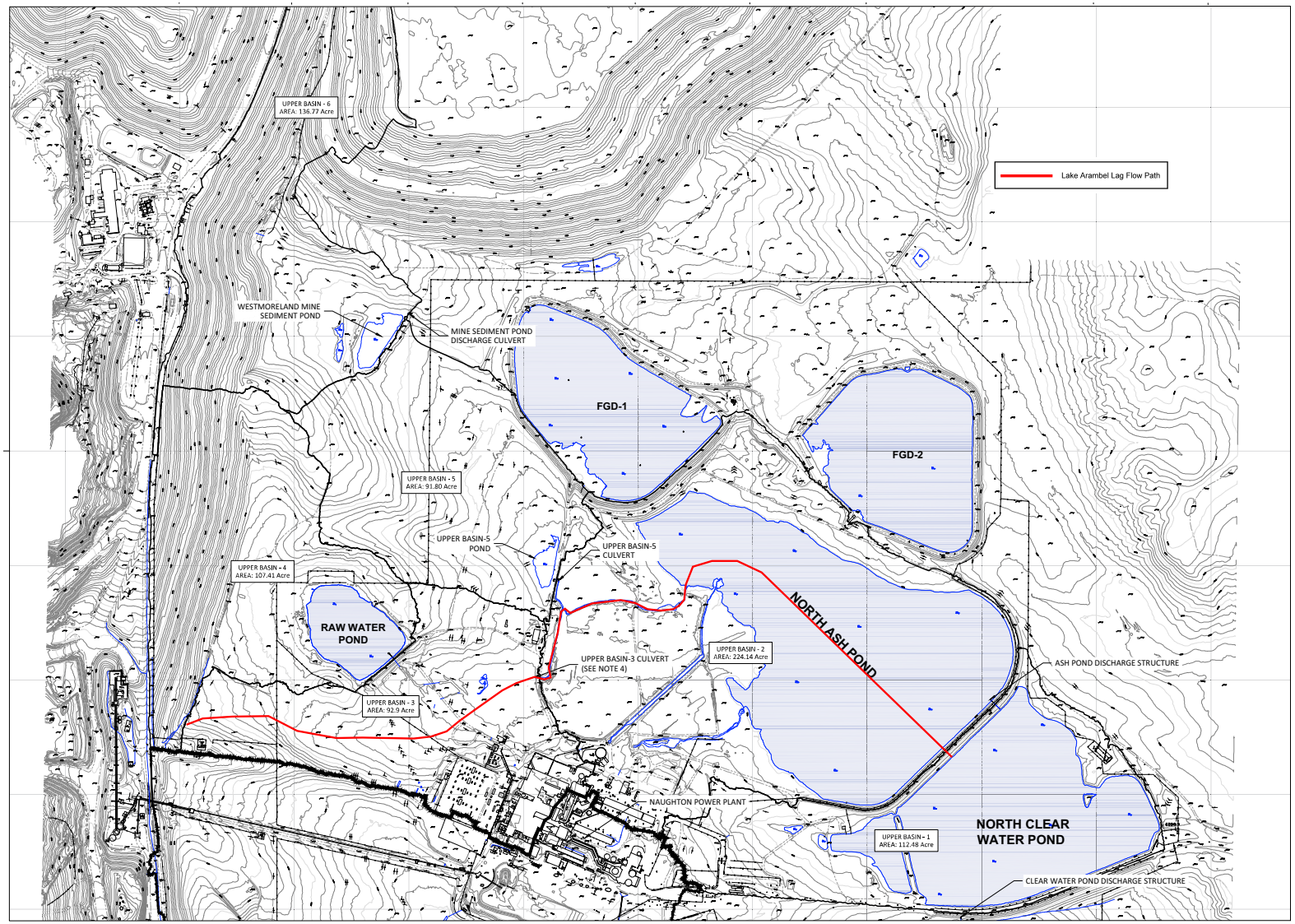



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**Figure 2.5-24 Lake Arambel PMF Runoff Hydrograph**

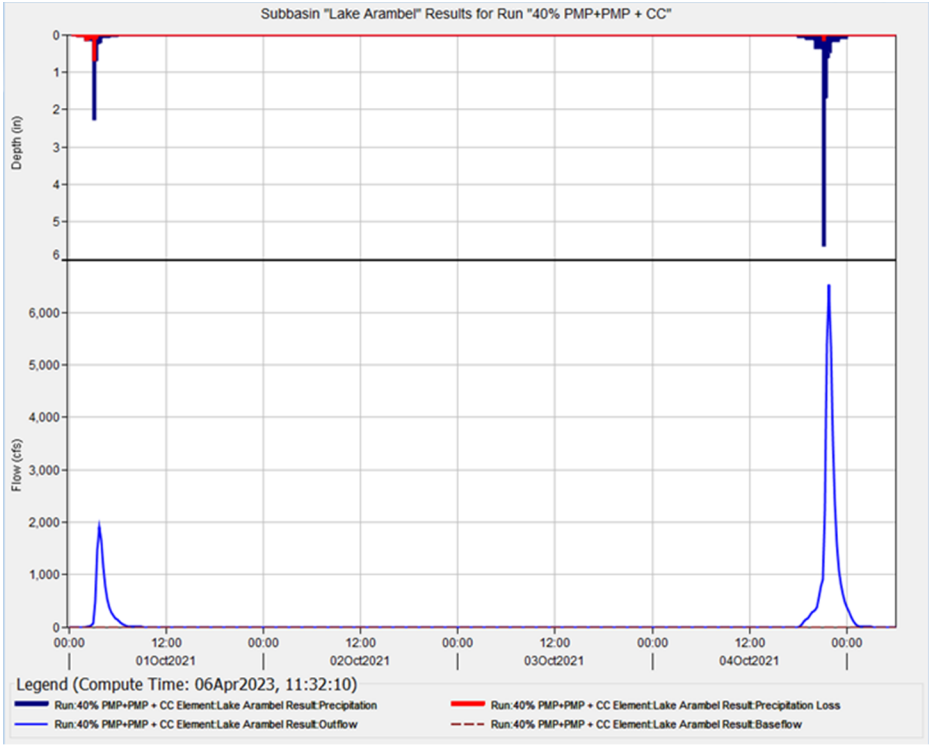



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**Figure 2.5-25 North Fork Little Muddy Creek (Sub-basin East) PMF Runoff Hydrograph**

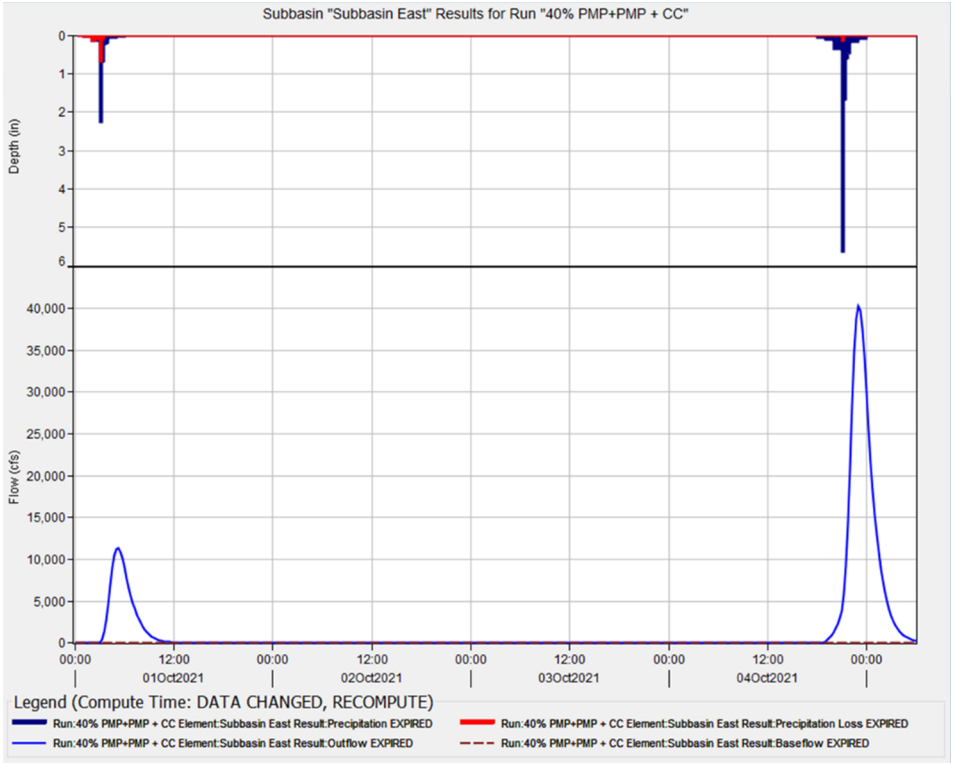



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**Figure 2.5-26 Hydrologic Failure (Overtopping) of the Lake Arambel Dam during the PMP. Stage and Flow Hydrograph at Section East Lower 8, Next to the Nuclear Island**

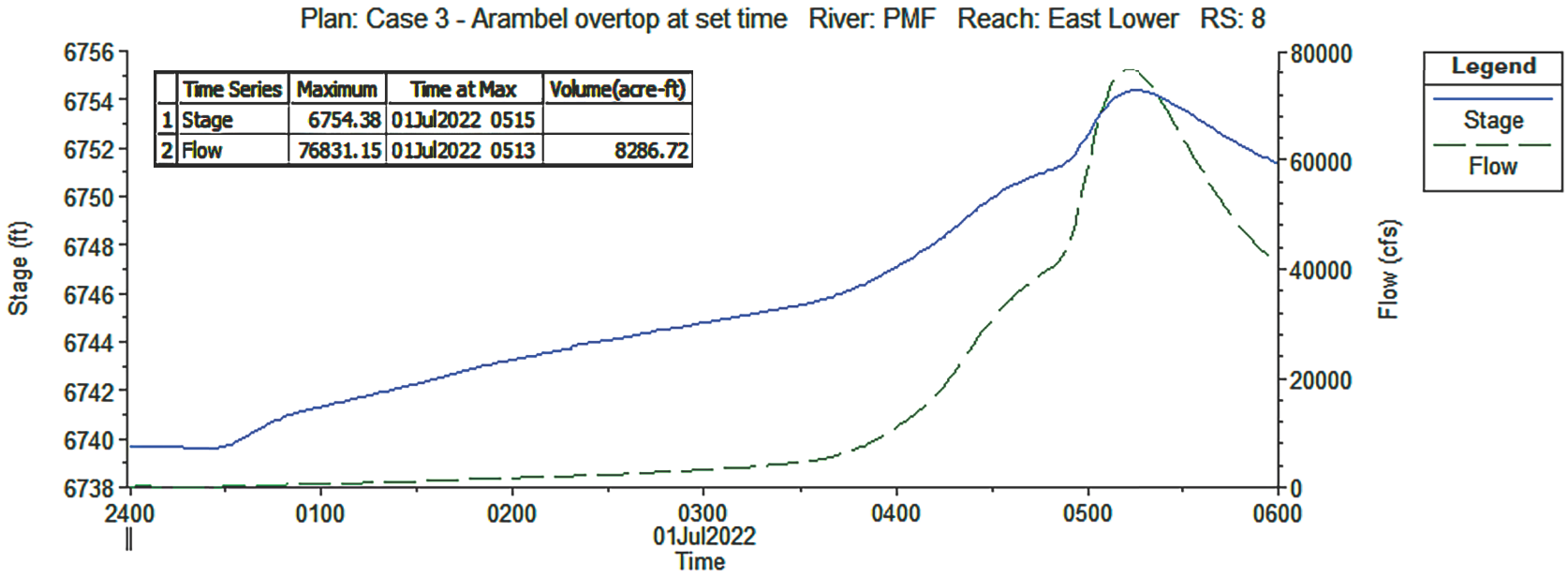



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**Figure 2.5-27 Maximum Water Surface Profile. Hydrologic Failure (Overtopping) of the Lake Arambel Dam during the PMP**

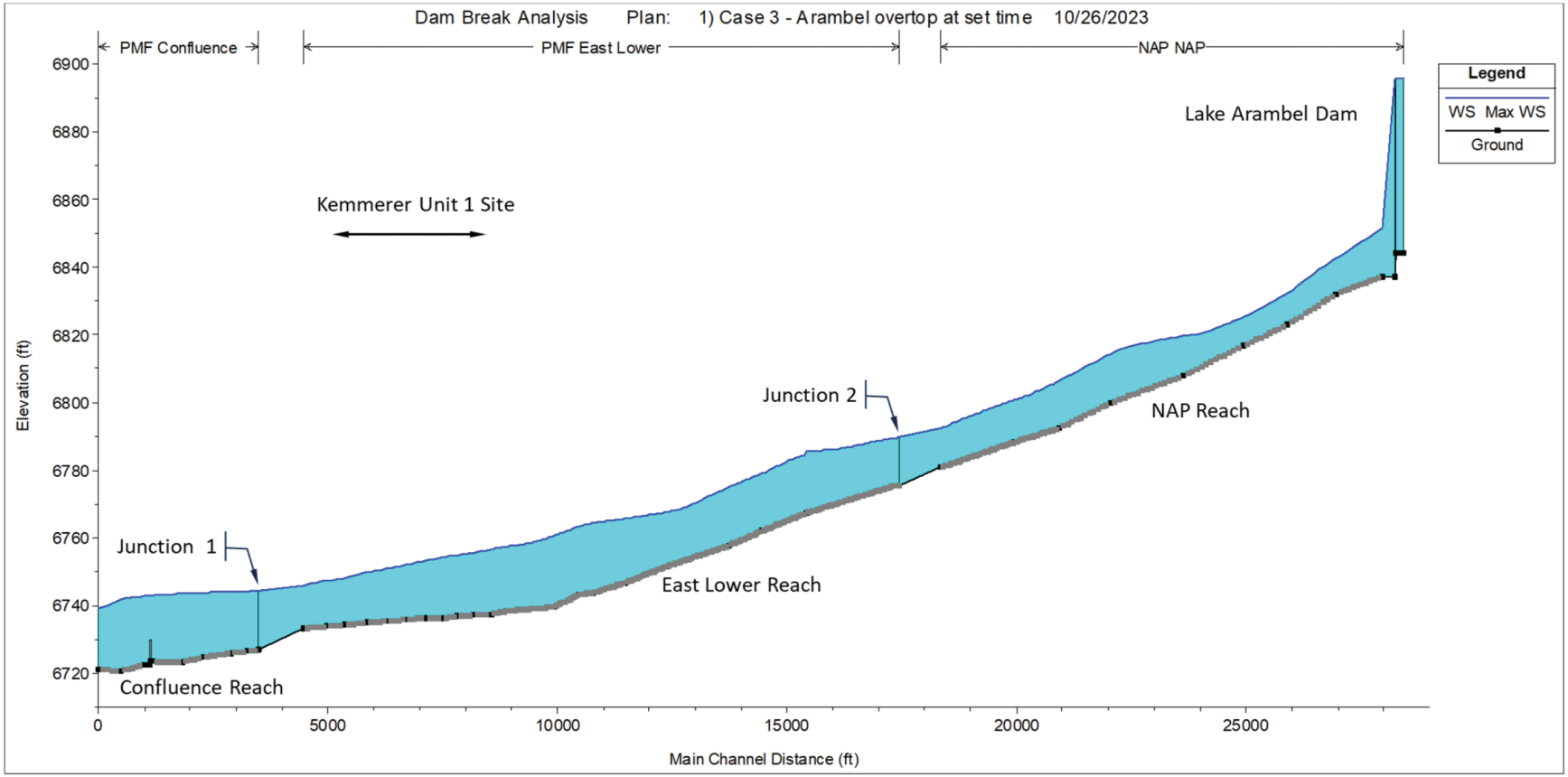



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**Figure 2.5-28 Dam Breach Fetch Length (yellow line)**

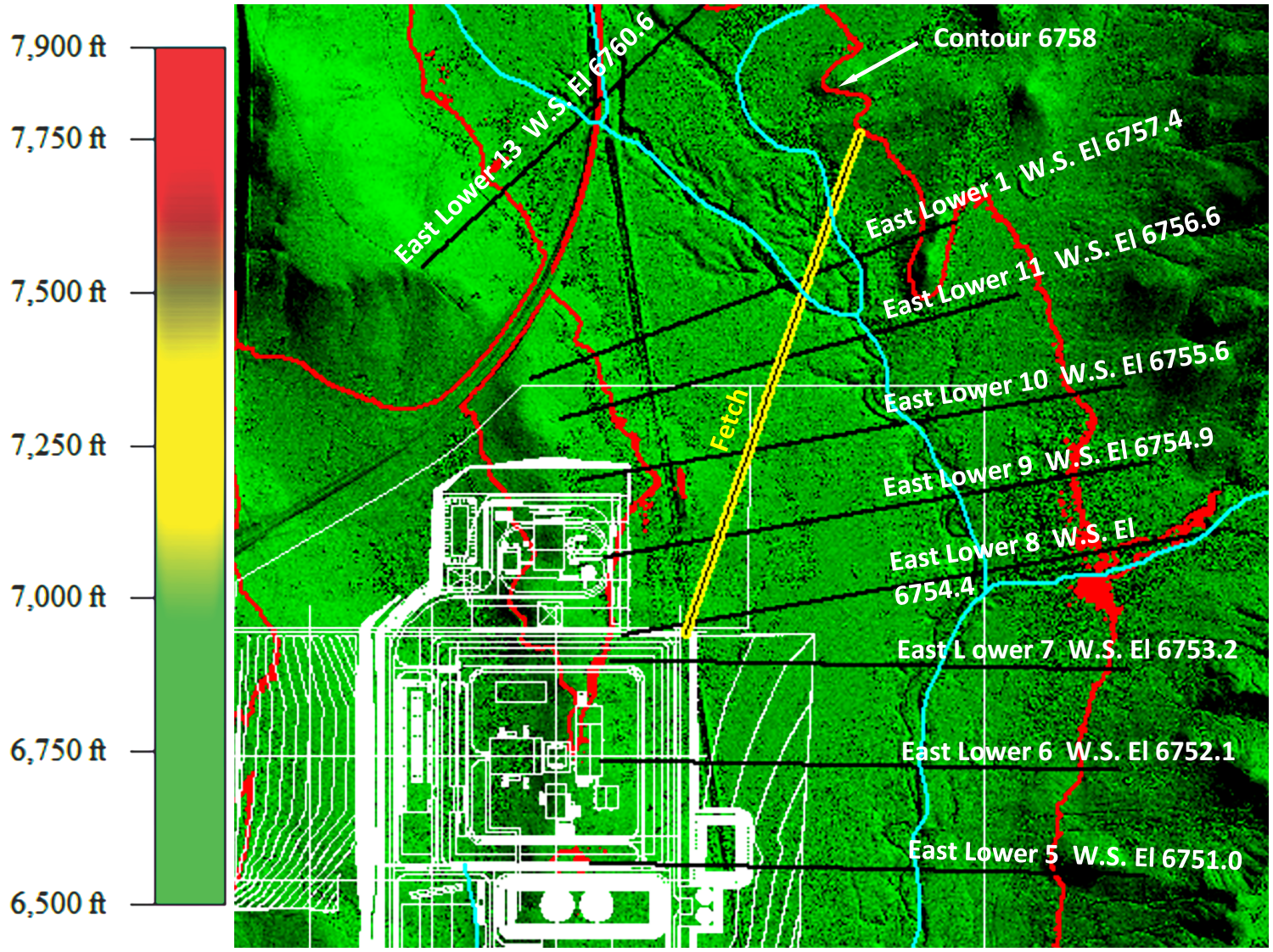



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**Figure 2.5-29 FGD 5 Fetch Length (yellow line)**

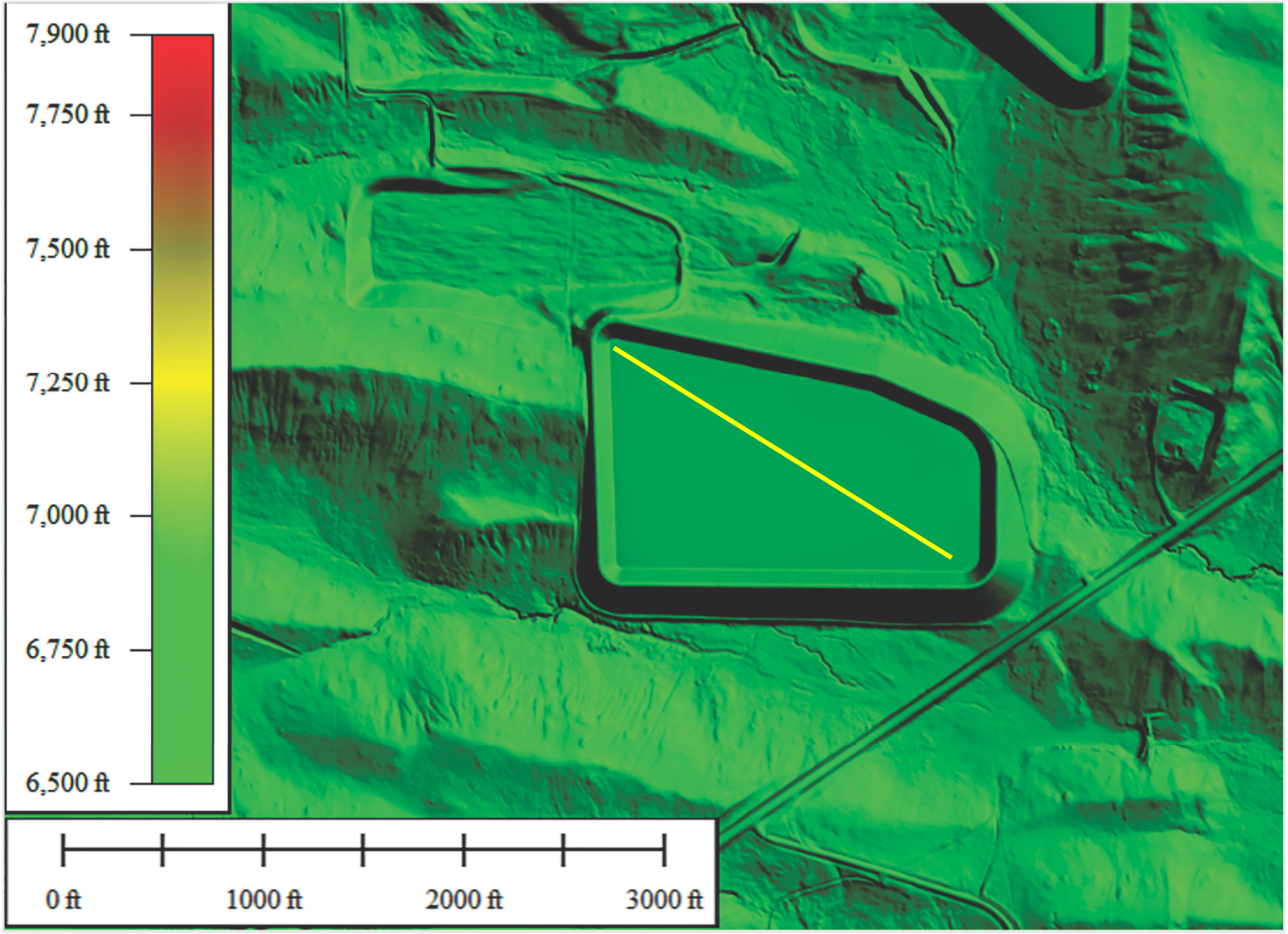






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**Figure 2.5-30 FGD 4 Fetch Length (yellow line)**

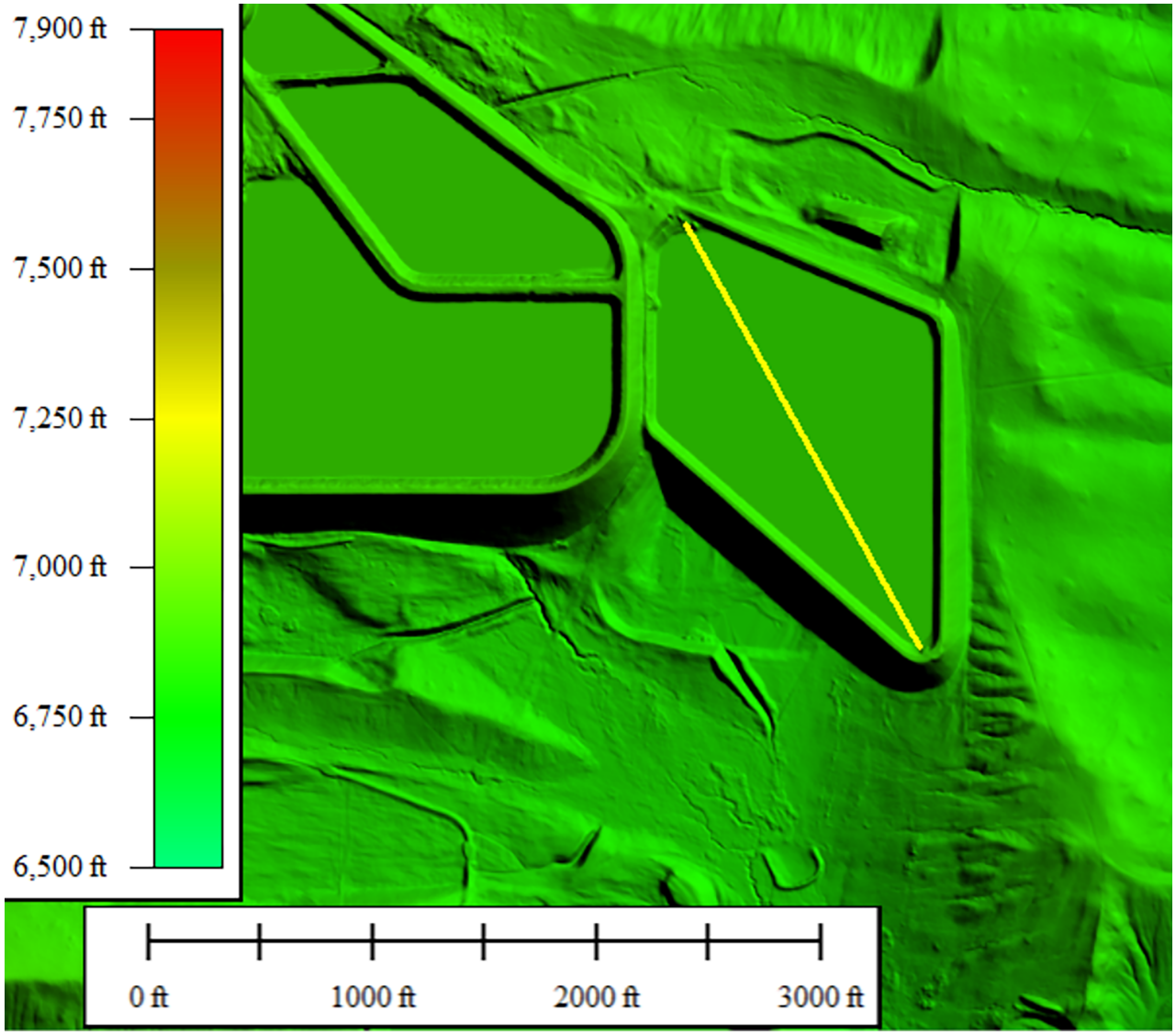






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**Figure 2.5-31 Kemmerer Unit 1 Site Location and NCDC Stations**

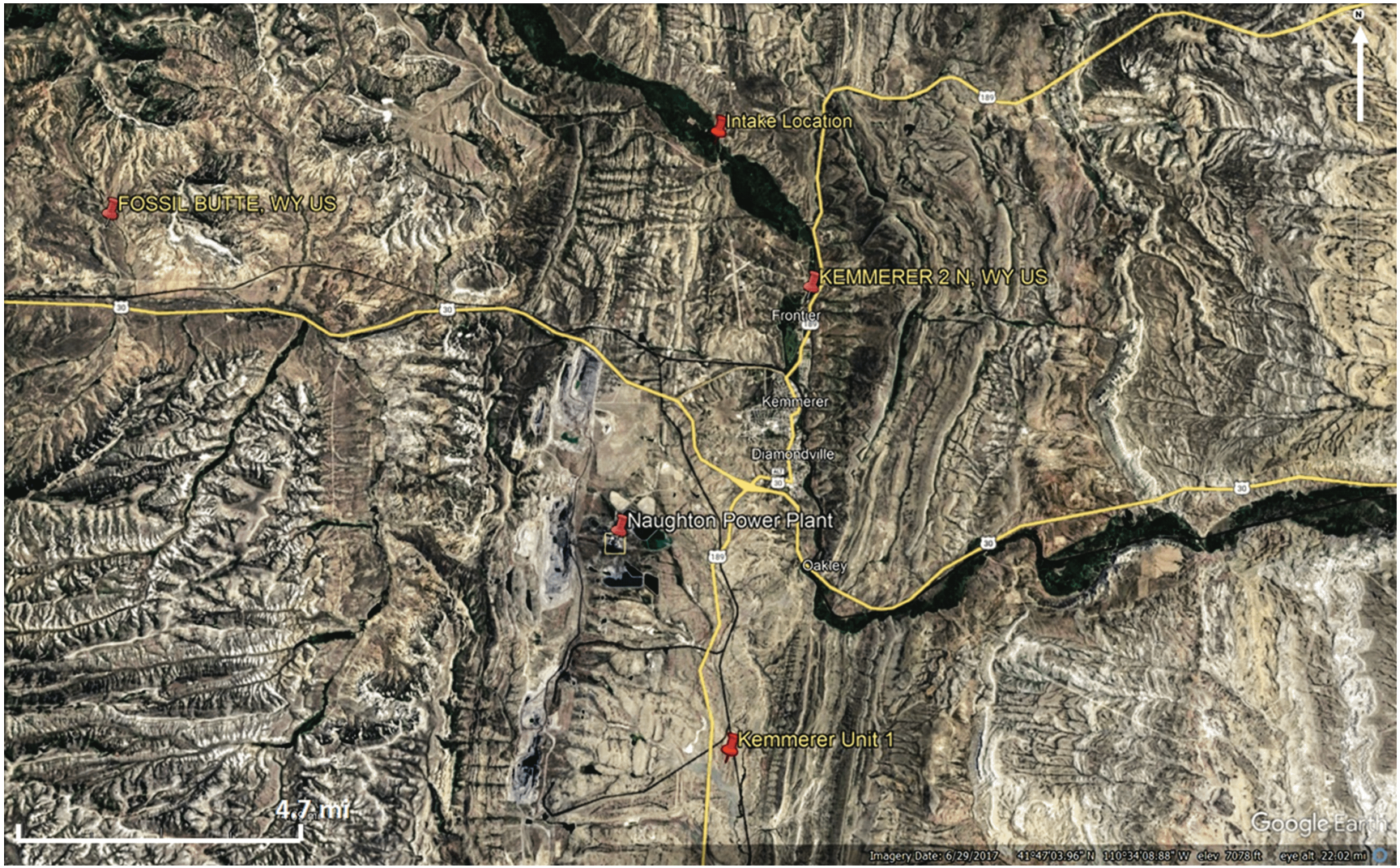



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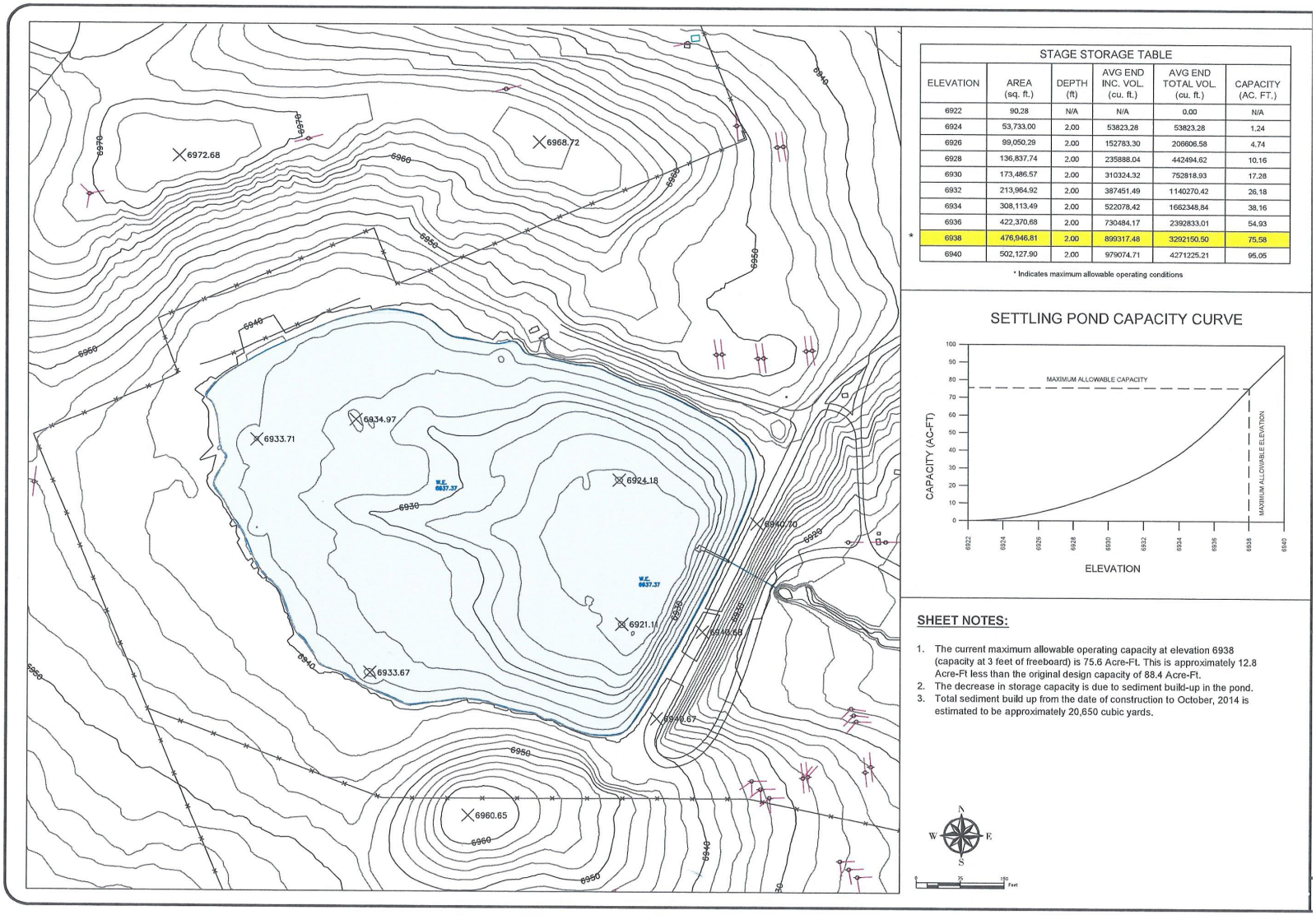
**Figure 2.5-32 Existing Raw Water Settling Basin (Settling Pond) at Naughton Power Plant (Reference 2.5-62)**




STAGE STORAGE TABLE

| ELEVATION | AREA (sq. ft.) | DEPTH (ft) | AVG END INC. VOL. (cu. ft.) | AVG END TOTAL VOL. (cu. ft.) | CAPACITY (AC. FT.) |
|---|---|---|---|---|---|
| 6922 | 90.28 | N/A | N/A | 0.00 | N/A |
| 6924 | 53,733.00 | 2.00 | 53823.28 | 53823.28 | 1.24 |
| 6926 | 99,050.29 | 2.00 | 152783.30 | 206606.58 | 4.74 |
| 6928 | 136,837.74 | 2.00 | 235888.04 | 442494.62 | 10.16 |
| 6930 | 173,486.57 | 2.00 | 310324.32 | 752818.93 | 17.28 |
| 6932 | 213,964.92 | 2.00 | 387451.49 | 1140270.42 | 26.18 |
| 6934 | 308,113.49 | 2.00 | 522078.42 | 1662348.84 | 38.16 |
| 6936 | 422,370.68 | 2.00 | 730484.17 | 2392833.01 | 54.93 |
| * 6938 | 476,946.81 | 2.00 | 899317.48 | 3292150.50 | 75.58 |
| 6940 | 502,127.90 | 2.00 | 979074.71 | 4271225.21 | 95.05 |

* Indicates maximum allowable operating conditions

SETTLING POND CAPACITY CURVE




**SHEET NOTES:**

1. The current maximum allowable operating capacity at elevation 6938 (capacity at 3 feet of freeboard) is 75.6 Acre-Ft. This is approximately 12.8 Acre-Ft less than the original design capacity of 88.4 Acre-Ft.
2. The decrease in storage capacity is due to sediment build-up in the pond.
3. Total sediment build up from the date of construction to October, 2014 is estimated to be approximately 20,650 cubic yards.





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**Figure 2.5-33 Kemmerer Unit 1 Site and Key Features in its Vicinity**

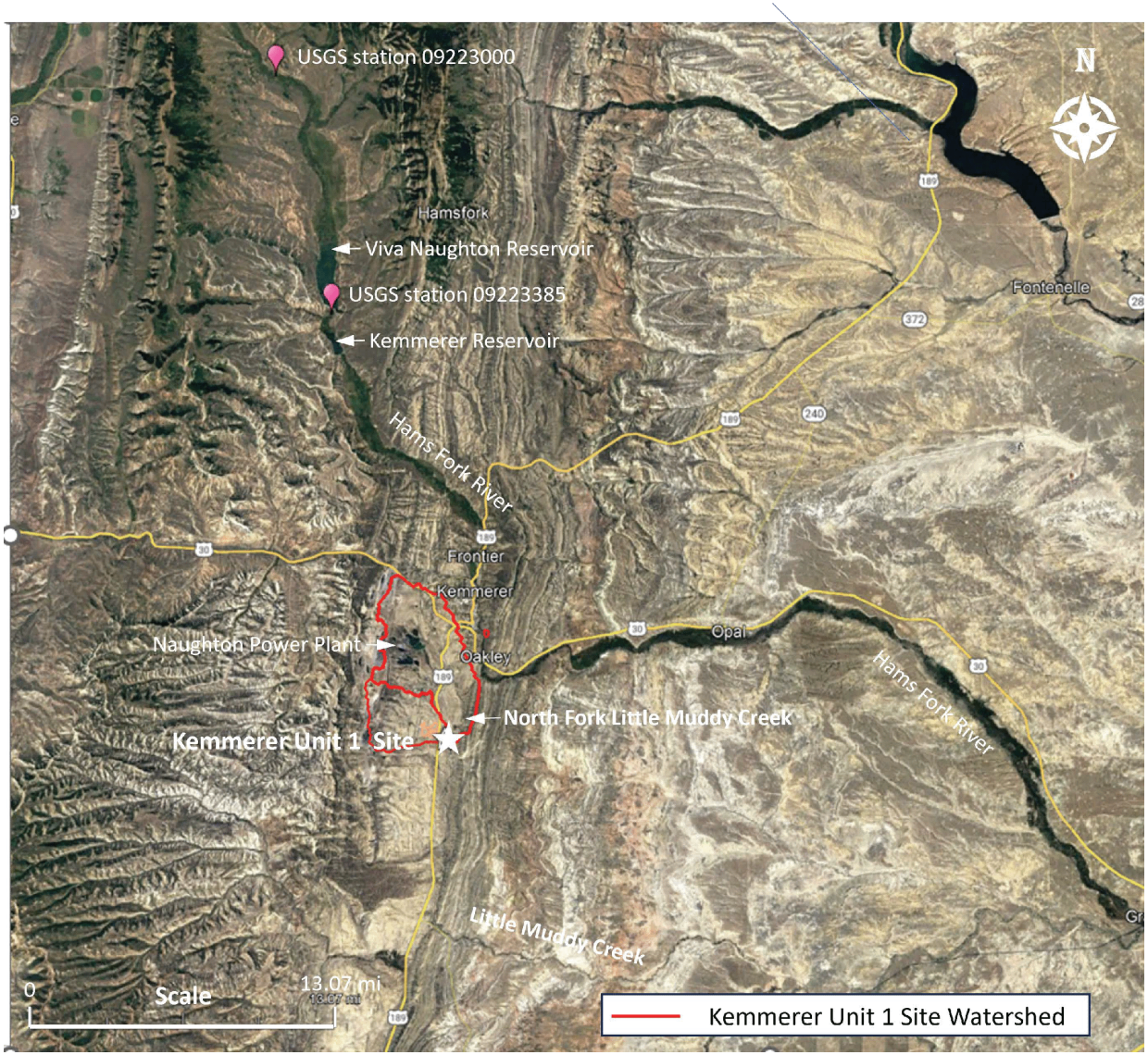



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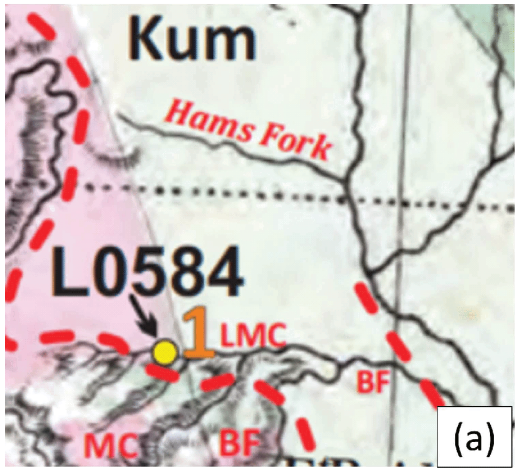
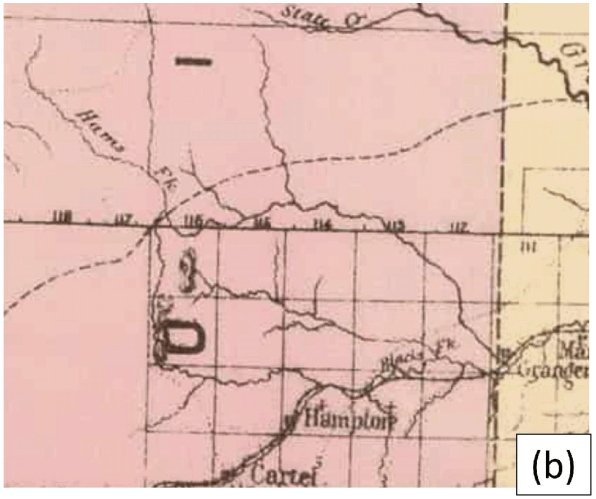
**Figure 2.5-34 Excerpts from Historical Maps: (a) Hall and Lesley's 1857 Map, (b) Mitchell's 1880 State, County and Township Map of Wyoming Territory, (c) Rand, McNally & Co.'s 1897 State Map of Wyoming**



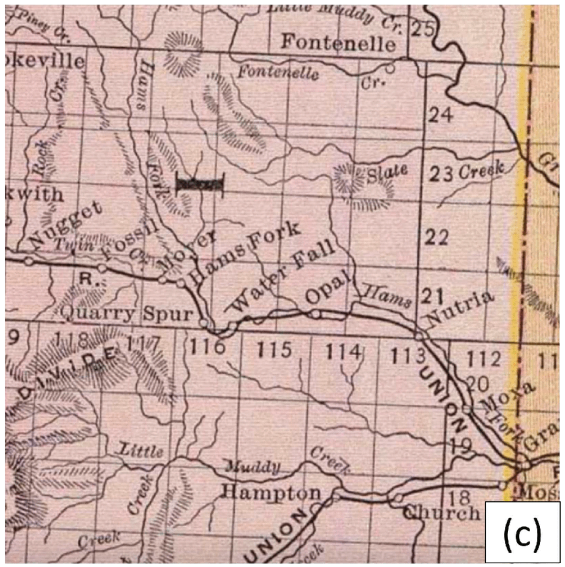





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**Figure 2.5-35 Excerpt from USGS Topographic Map Showing the Course of North Fork Little Muddy Creek and Tributaries in 1962 (Sheet Elkol, Wyoming, 1:24 000)**

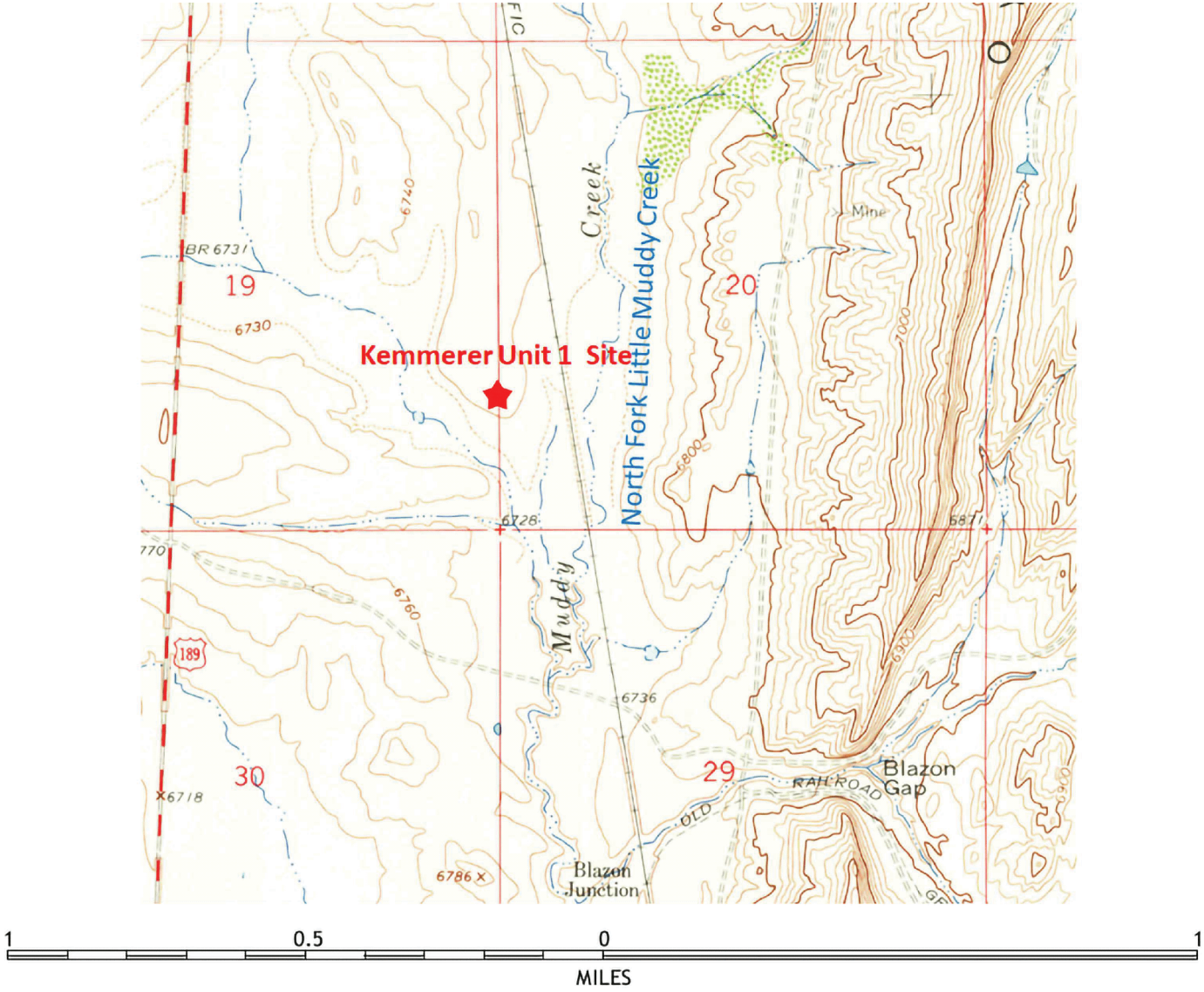



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**Figure 2.5-36 Excerpt from USGS Topographic Map Showing the Course of North Fork Little Muddy Creek and Tributaries in 1979 (Sheet Kemmerer, Wyoming, 1:100 000)**

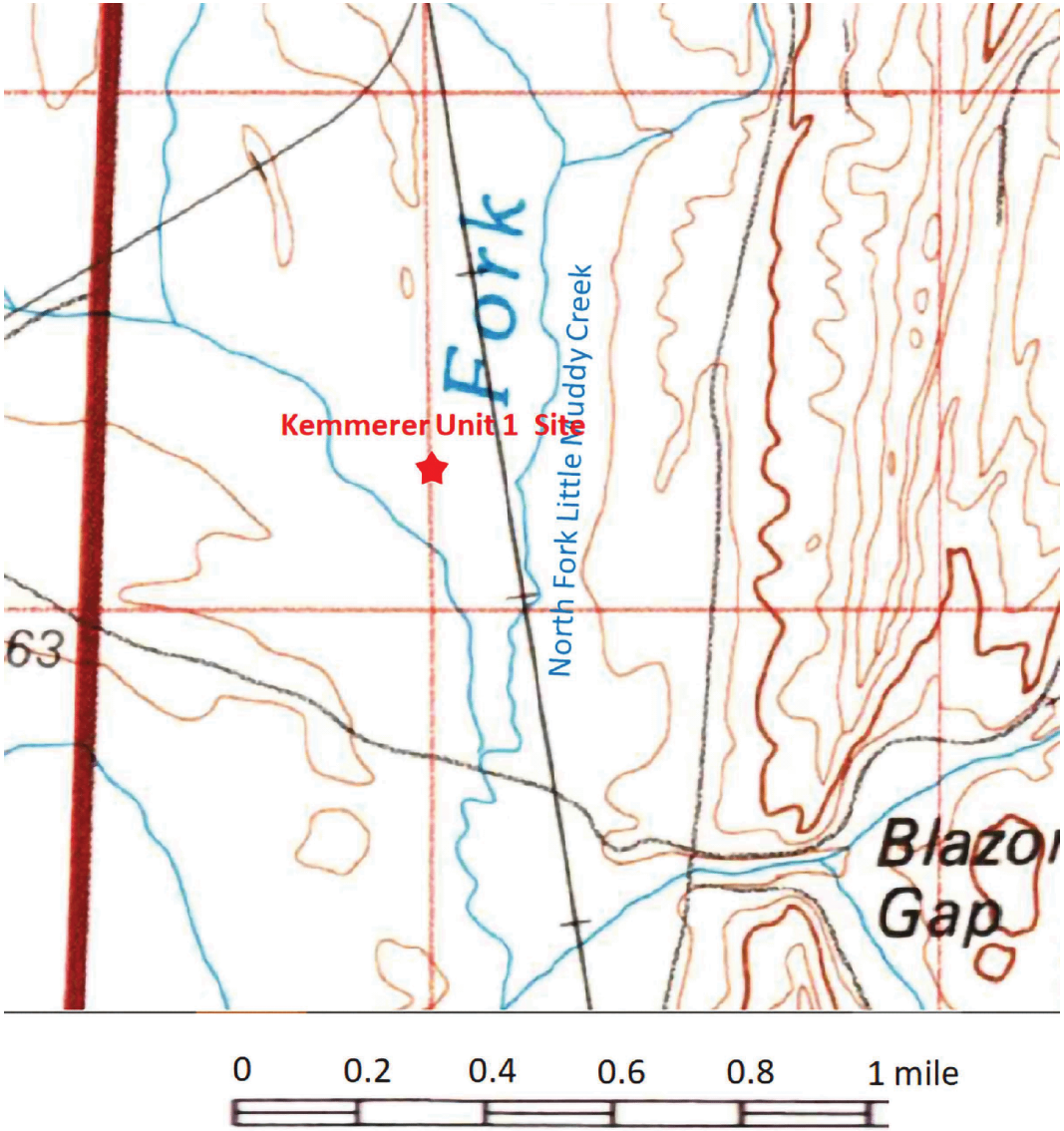






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**Figure 2.5-37 Excerpt from USGS Topographic Map Showing the Course of North Fork Little Muddy Creek and Tributaries in 2012 (Sheet Elkol, Wyoming, 1:24 000)**

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**Figure 2.5-38 Excerpt from USGS Topographic Map Showing the Course of North Fork Little Muddy Creek and Tributaries in 2021 (Sheet Elkol, Wyoming, 1:24 000)**

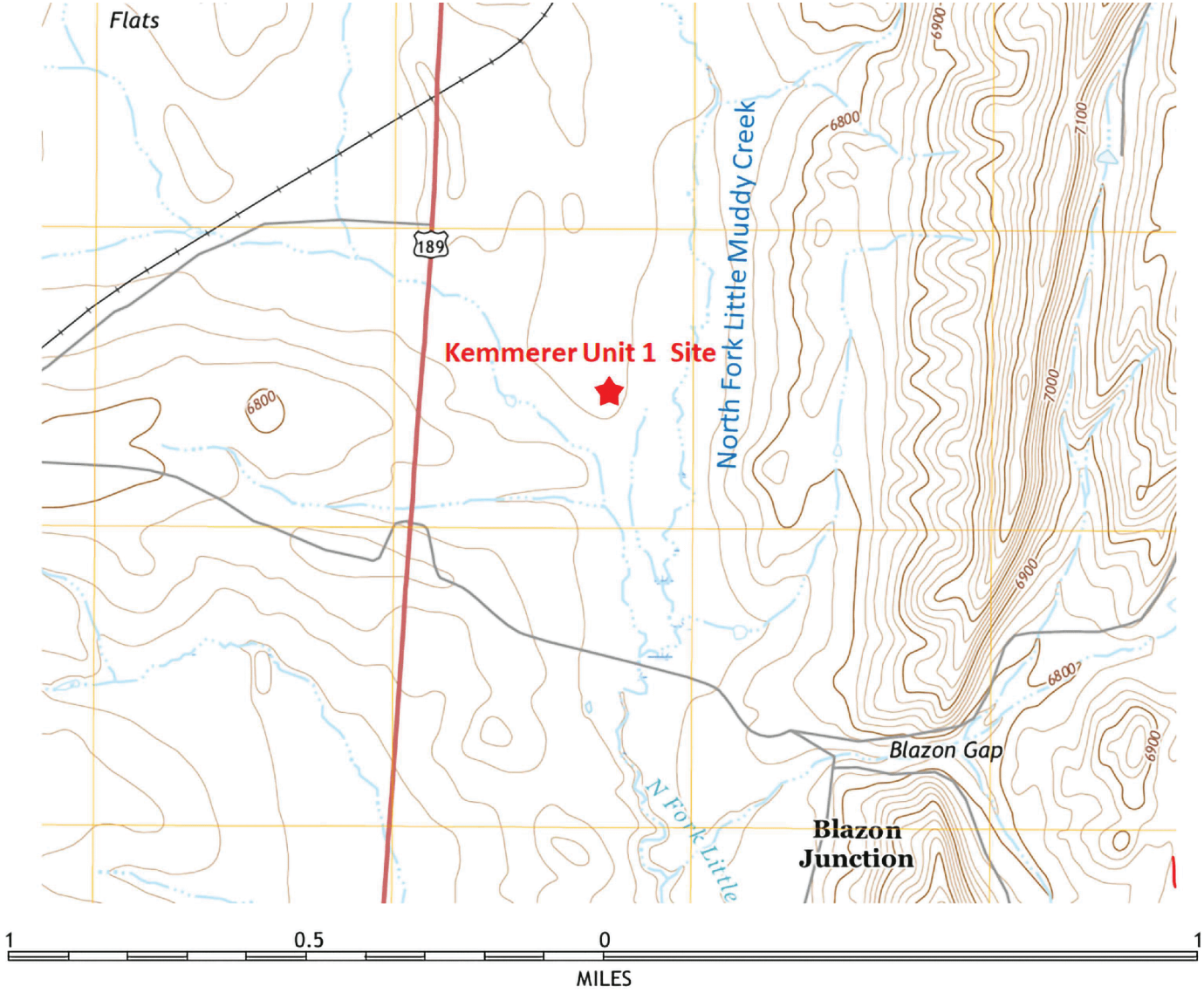



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**Figure 2.5-39 Satellite Image of the Vicinity of the Kemmerer Unit 1 Site on December 30, 1985 (source: Google Earth; Light Blue Lines Indicate Streams based on the Current Topography)**

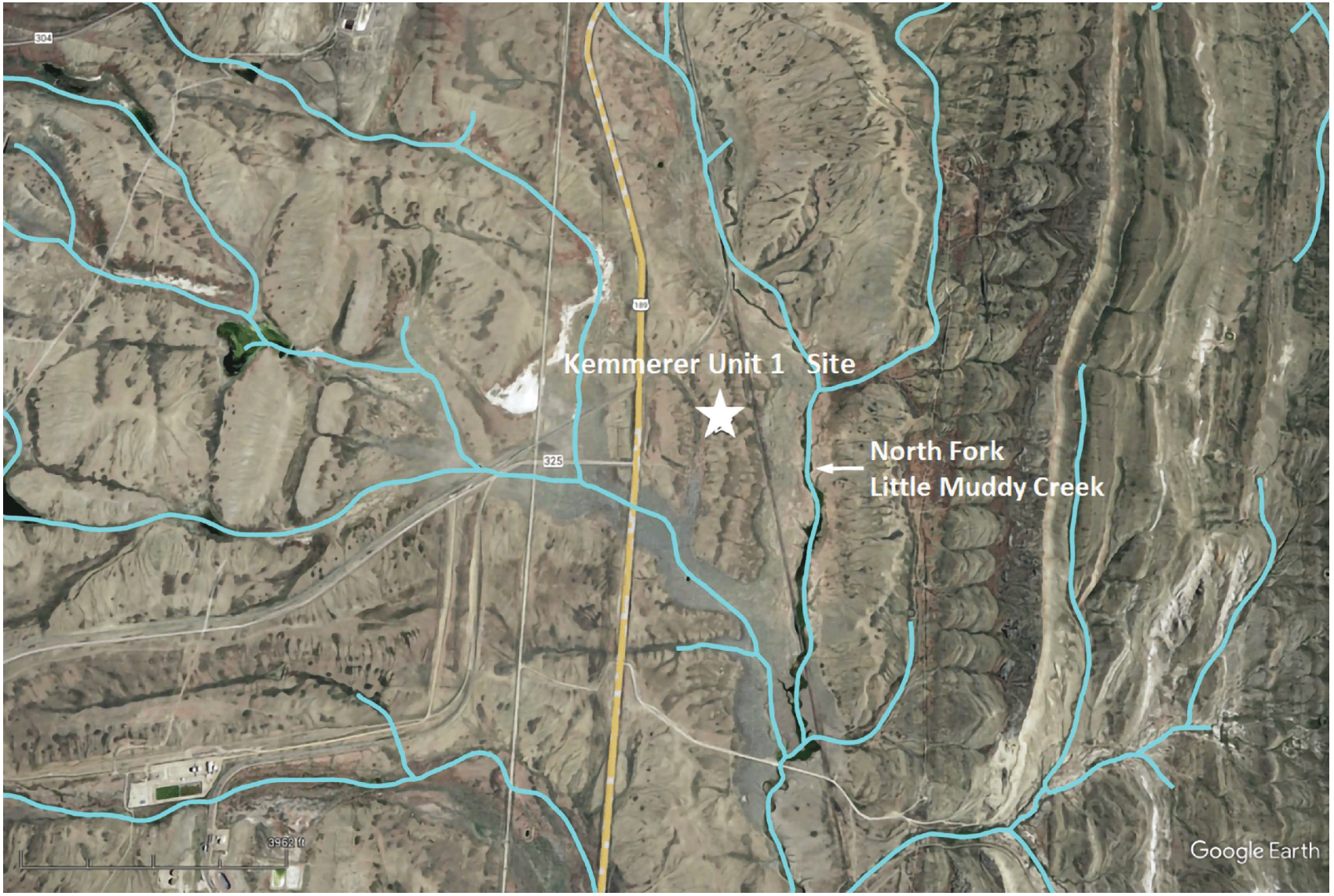






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**Figure 2.5-40 Satellite Image of the Vicinity of the Kemmerer Unit 1 Site on August 24, 1994 (source: Google Earth; Light Blue Lines Indicate Streams based on the Current Topography)**

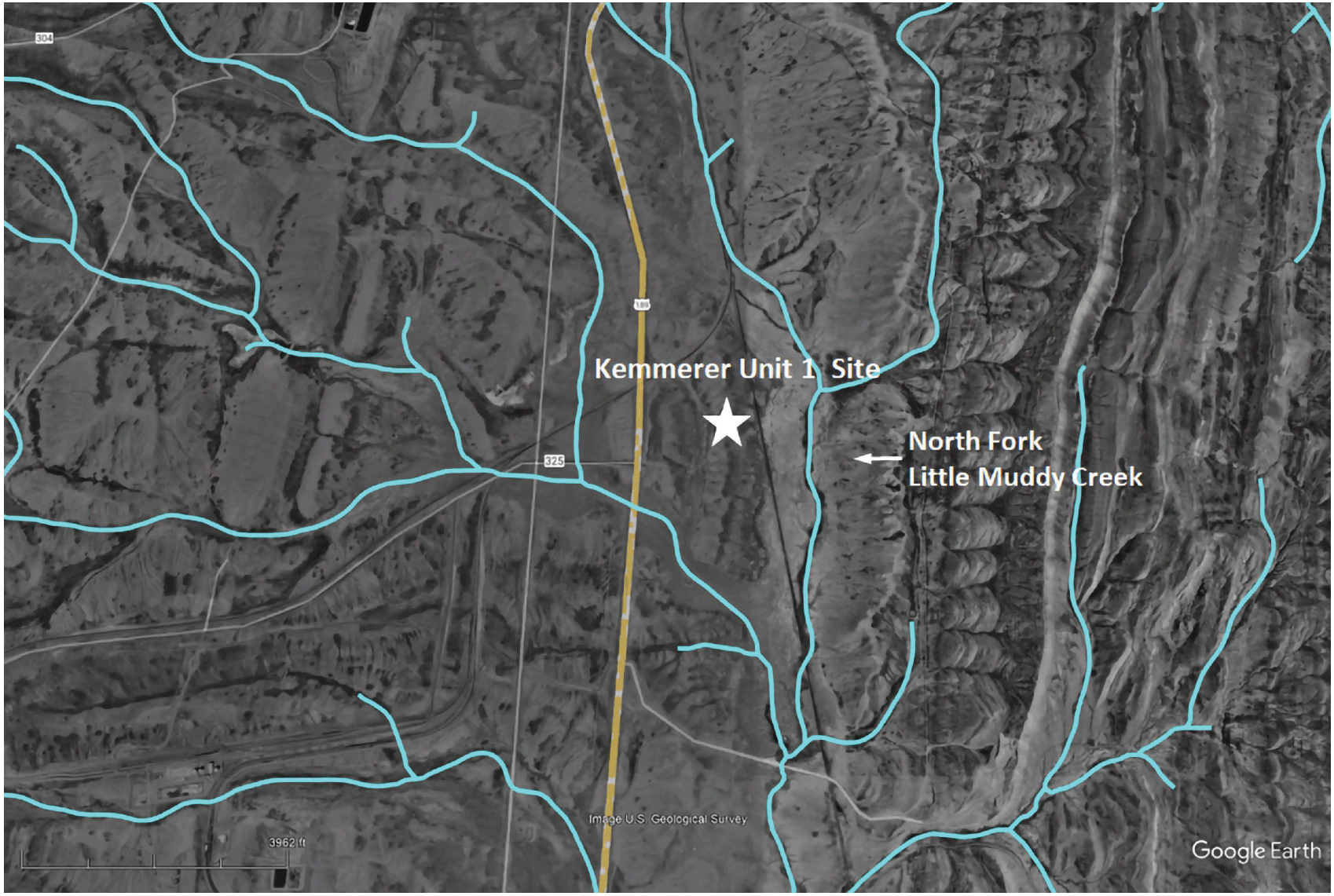



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**Figure 2.5-41 Satellite Image of the Vicinity of the Kemmerer Unit 1 Site on July 10, 2006 (source: Google Earth; Light Blue Lines Indicate Streams based on the Current Topography)**

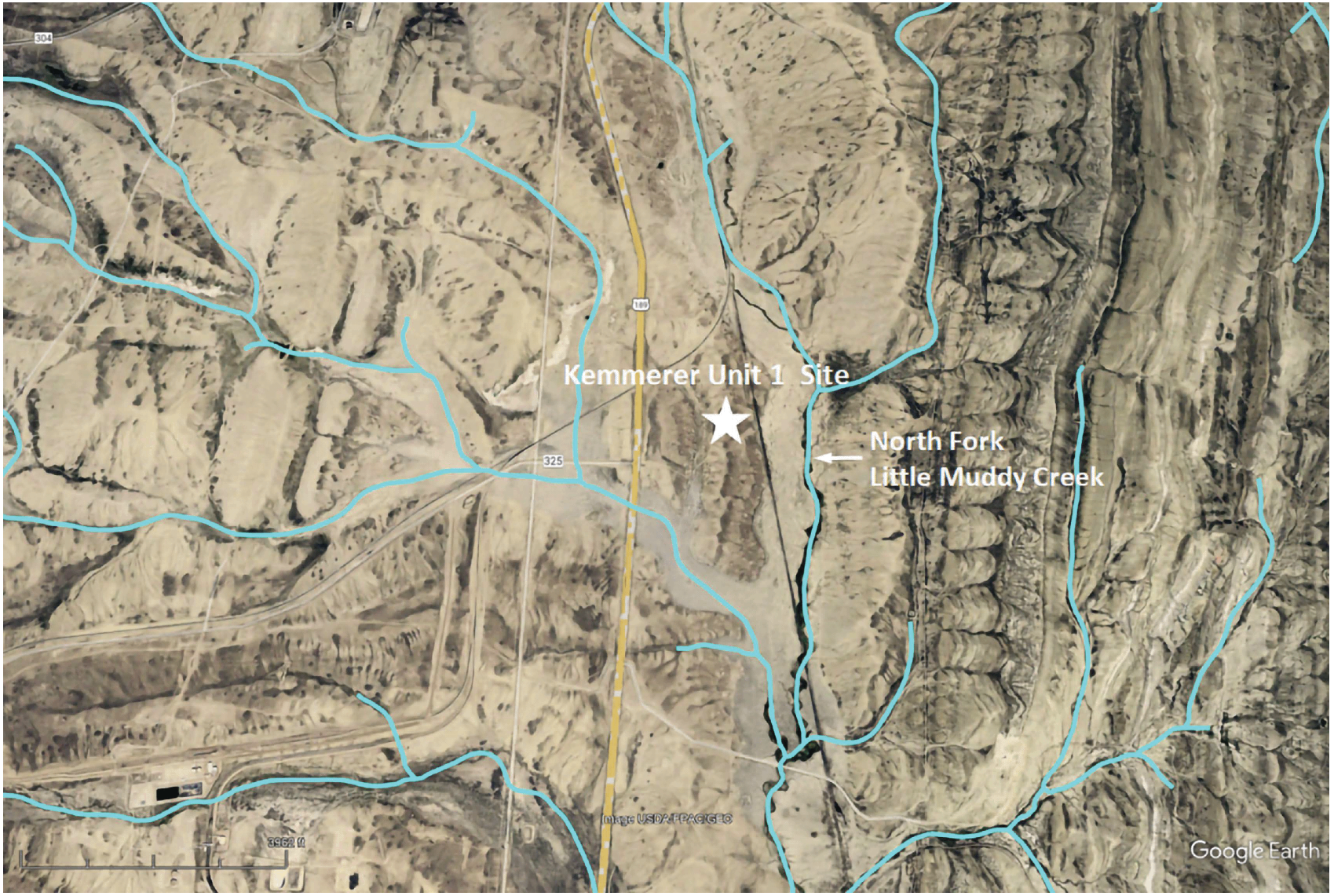



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**Figure 2.5-42 Satellite Image of the Vicinity of the Kemmerer Unit 1 Site on November 4, 2020 (source: Google Earth; Light Blue Lines Indicate Streams based on the Current Topography)**

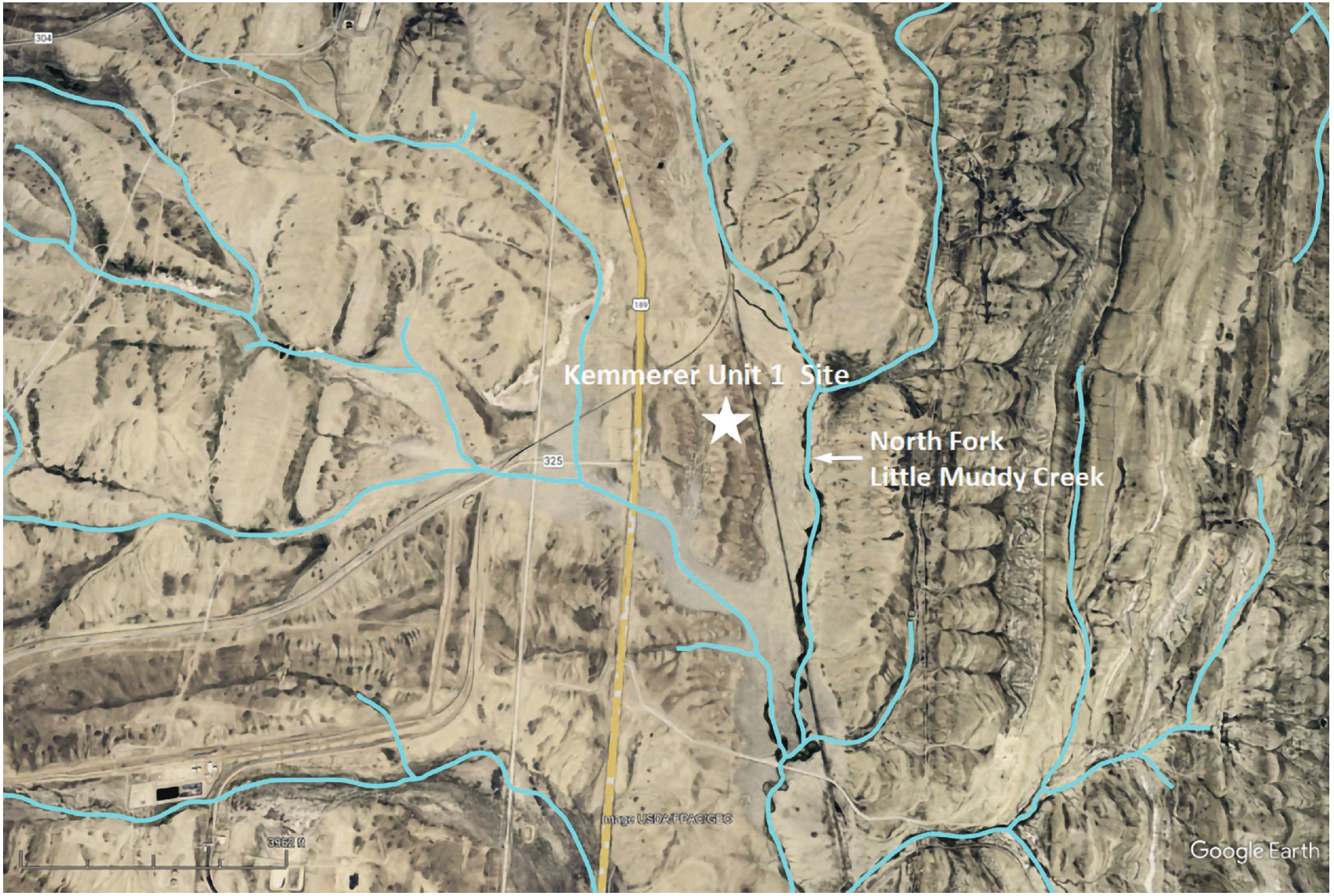



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**Figure 2.5-43 Daily Discharge on Hams Fork River below the Viva Naughton Reservoir**

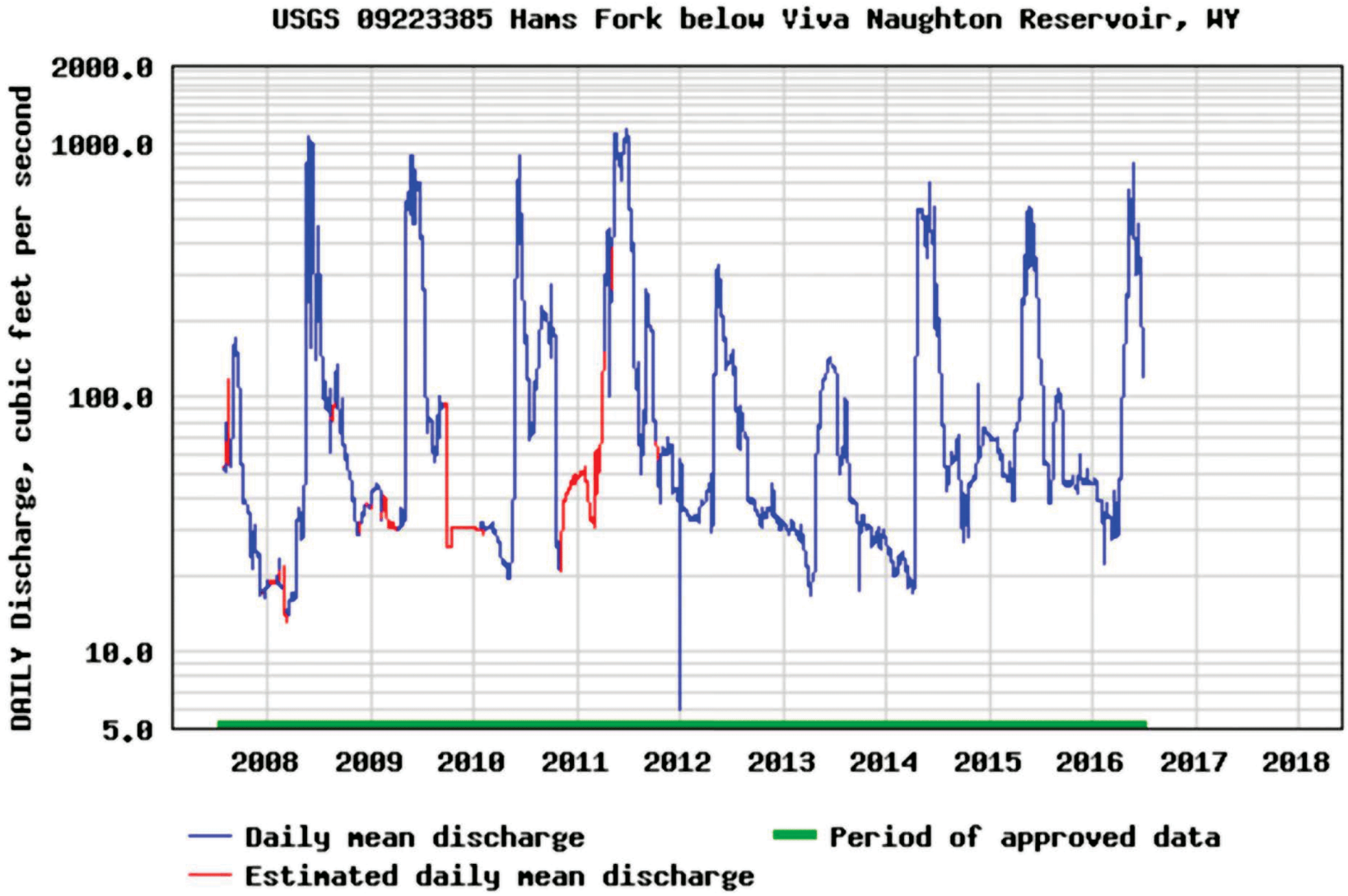






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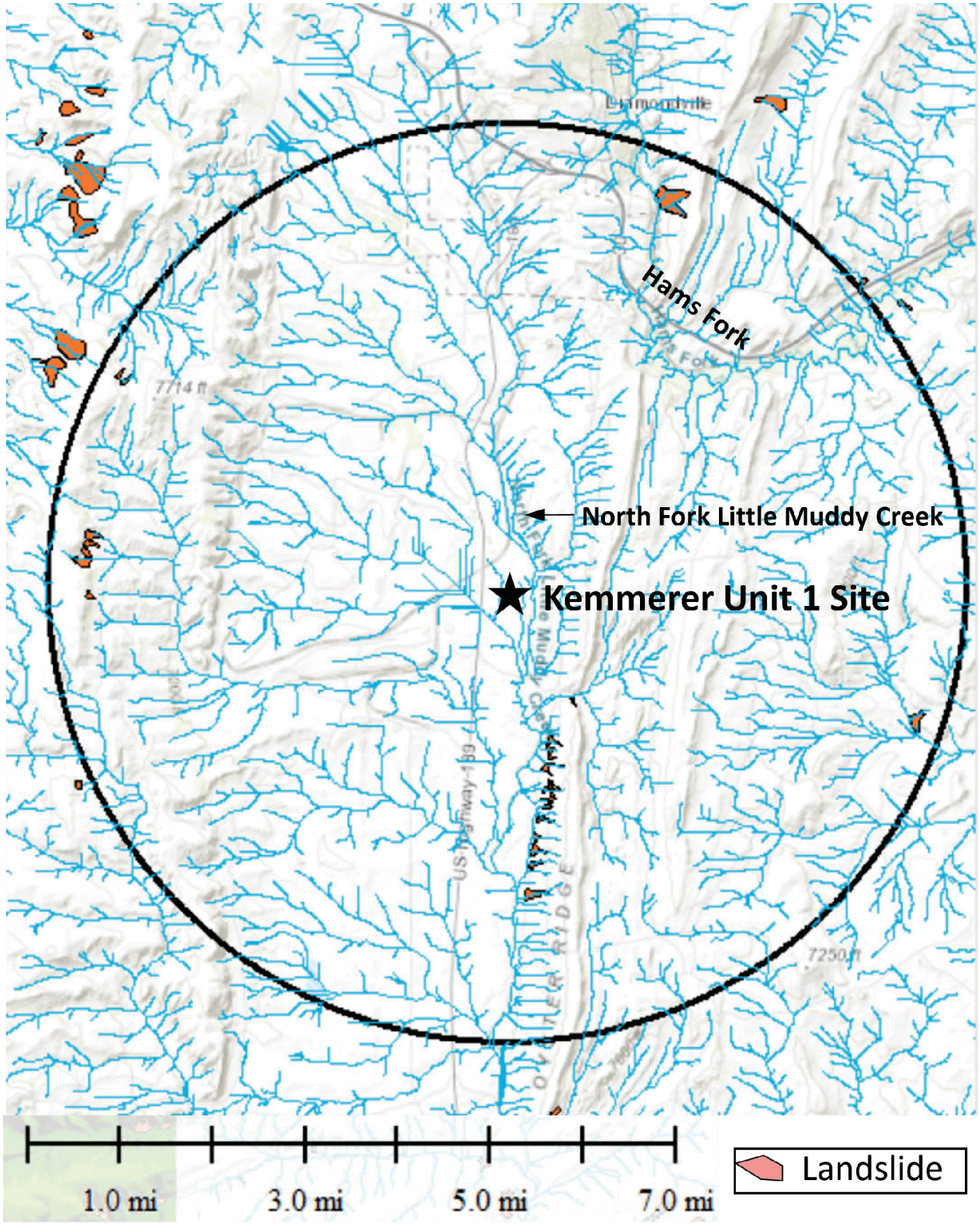
**Figure 2.5-44 Landslide Incidence Map (Reference 2.5-79). The Circle Encloses an Area of 5-mile Radius.**




Note: The circle encloses an area of 5-mile radius


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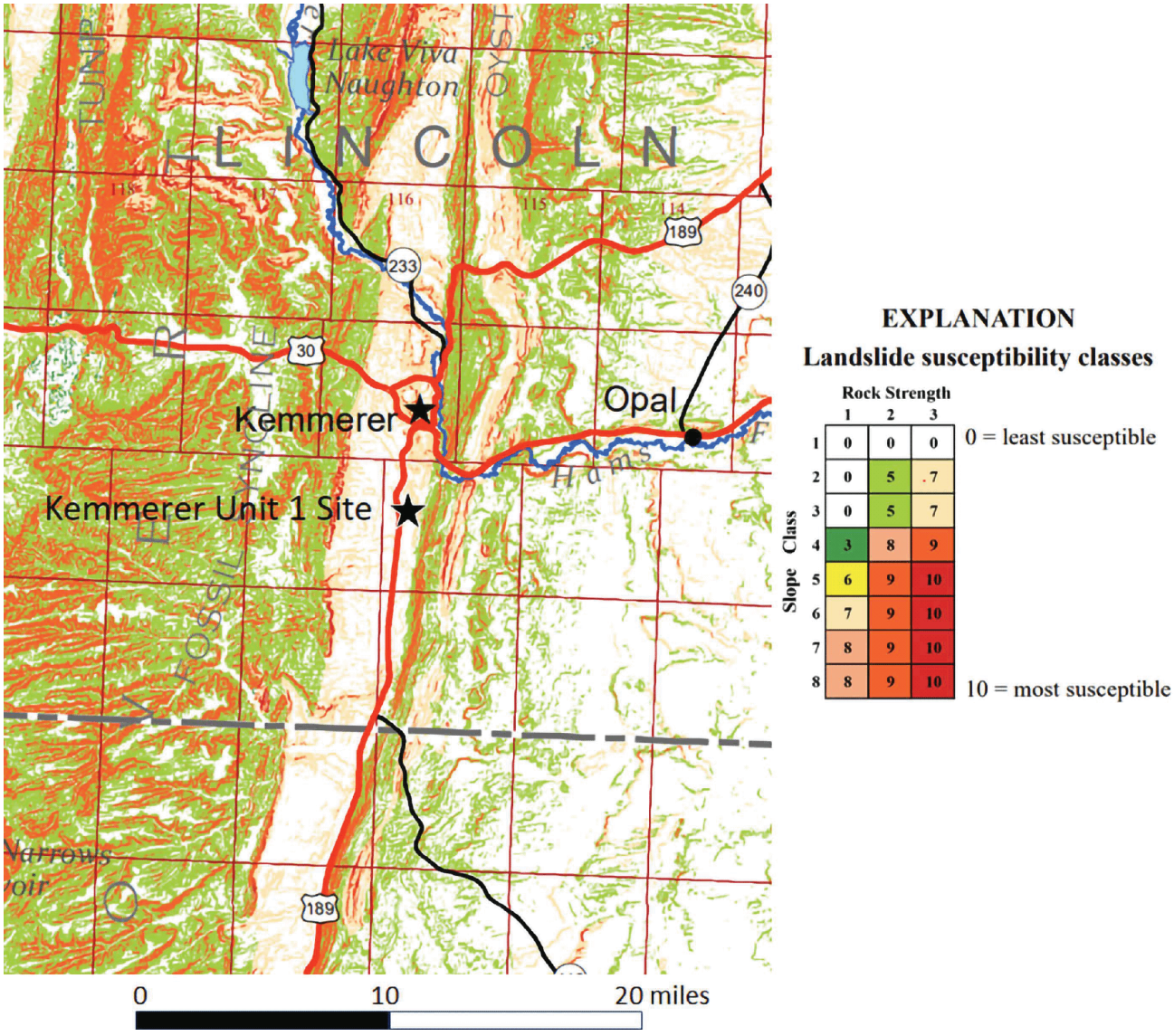
**Figure 2.5-45 Deep-Seated Landslide Susceptibility Map (Reference 2.5-76)**




| Slope Class | Rock Strength 1 | 2 | 3 |
|---|---|---|---|
| 1 | 0 | 0 | 0 |
| 2 | 0 | 5 | .7 |
| 3 | 0 | 5 | 7 |
| 4 | 3 | 8 | 9 |
| 5 | 6 | 9 | 10 |
| 6 | 7 | 9 | 10 |
| 7 | 8 | 9 | 10 |
| 8 | 8 | 9 | 10 |


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**Figure 2.5-46 Surficial Materials Map Showing a Landslide Area along Hams Fork River near Kemmerer (excerpt from Reference 2.5-78)**

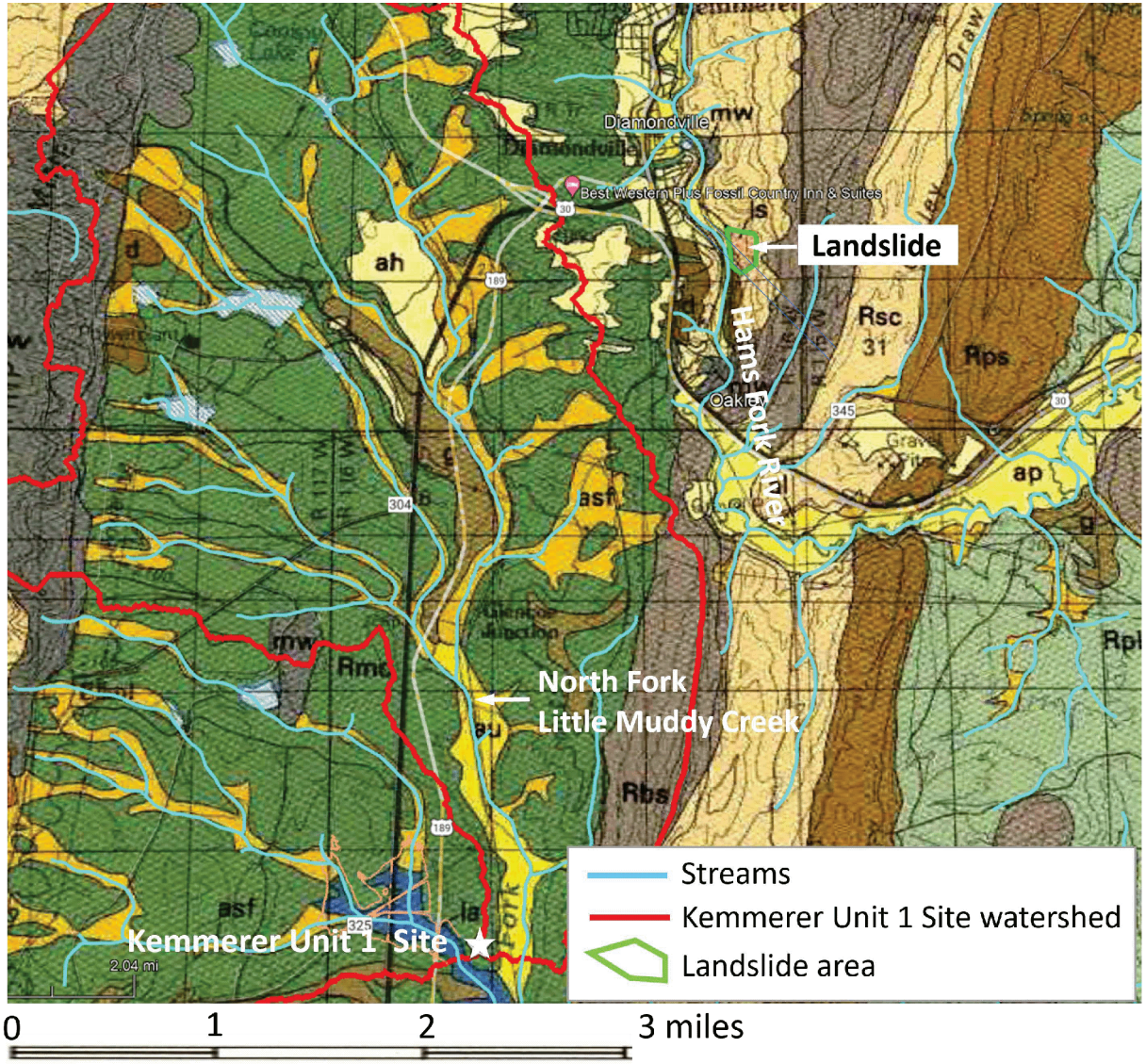



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**Figure 2.5-47 Kemmerer Unit 1 Site Watershed**

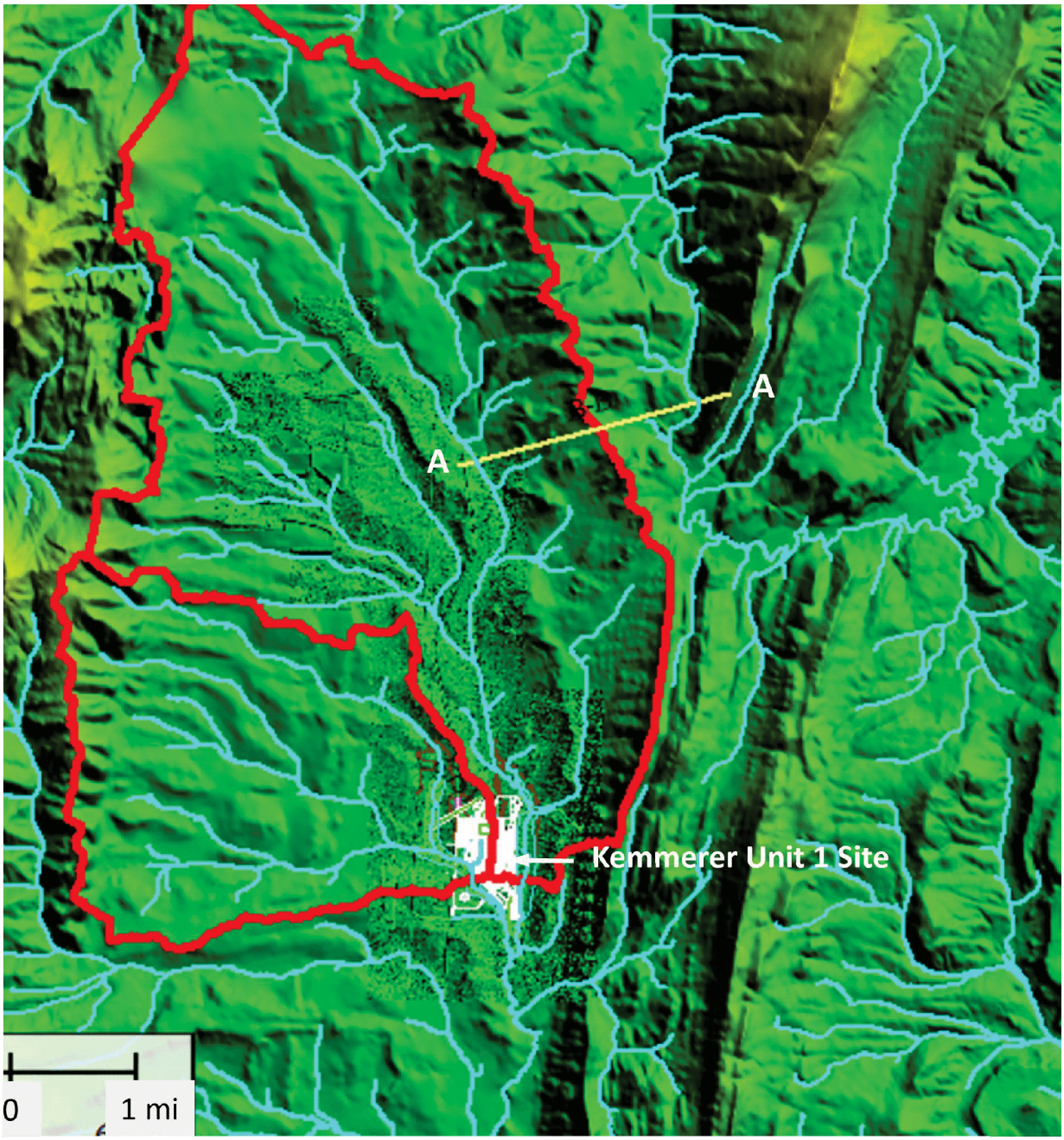




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**Figure 2.5-48 Cross Section A-A Defined in Figure 2.5-47**

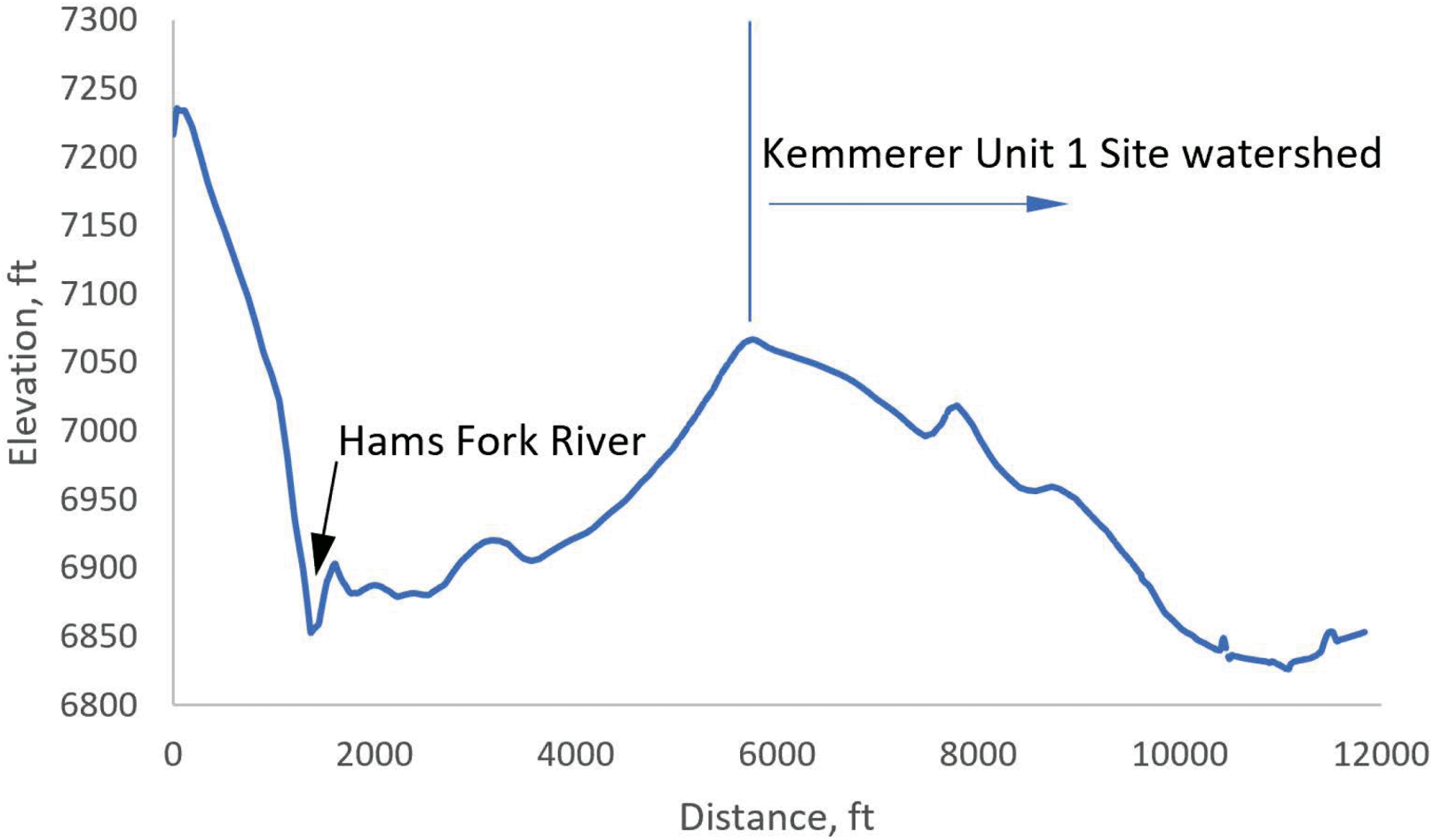






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**Figure 2.5-49 The Blazon Gap Watershed**

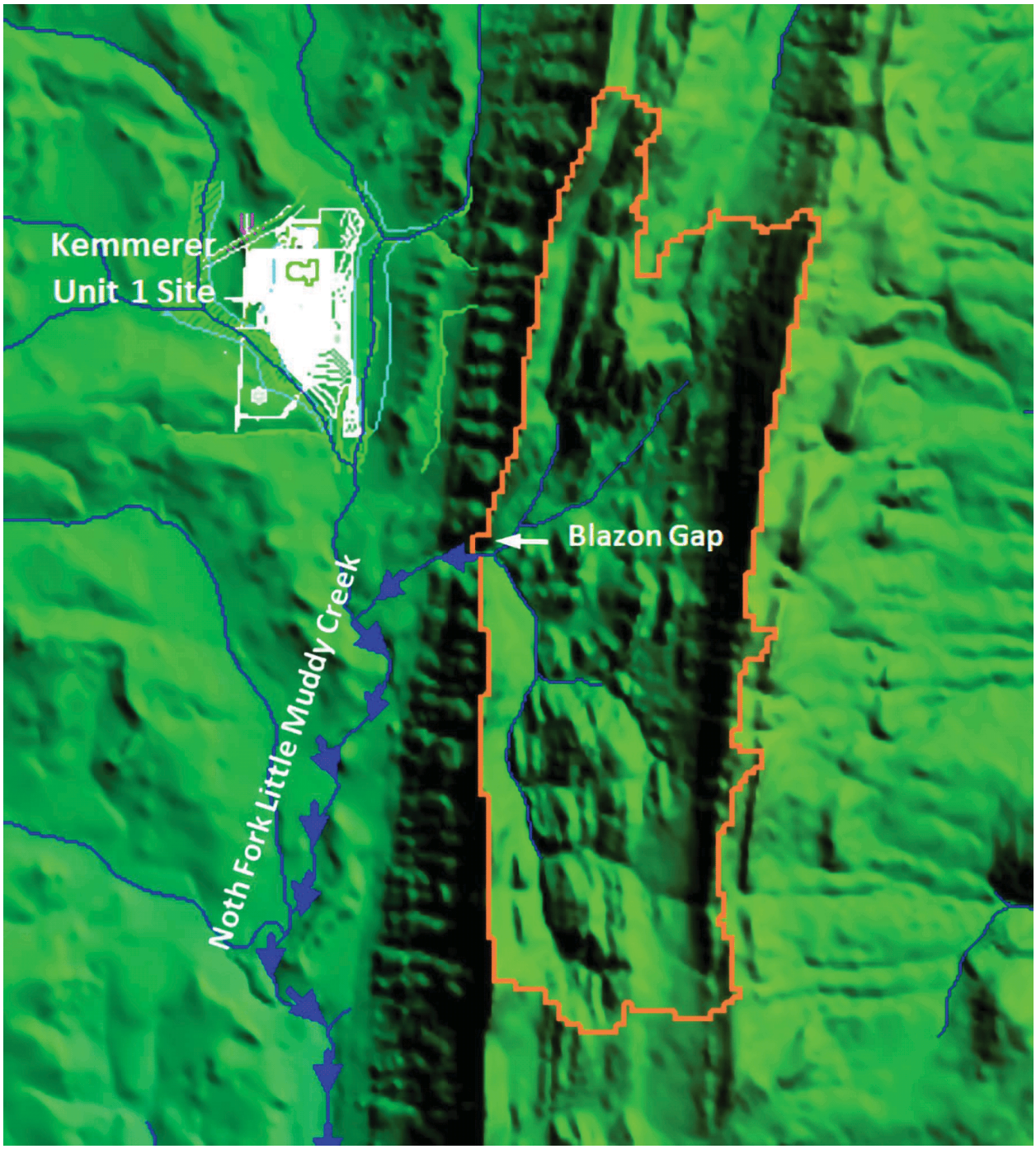



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**Figure 2.5-50 The North Fork Little Muddy Creek Close to Blazon Gap**

Note: The right side of the photo shows the left (east) bank of the creek with deposits that most likely have come from the Blazon Gap watershed.


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**Figure 2.5-51 Closeup of the East Bank Face of the North Fork Little Muddy Creek showing Interbedded Silt and Gravel Deposits**

Note: This is at the location of the photograph of Figure 2.5.1.7-r


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**Figure 2.5-52 The Stream Draining the Blazon Gap Watershed**

Note: This photograph was taken at a location about halfway between Blazon Gap and the North Fork Little Muddy Creek


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**Figure 2.5-53 Schematic Map Showing the Linear Extent of Riprap Revetment along the East Side of the Nuclear Island**

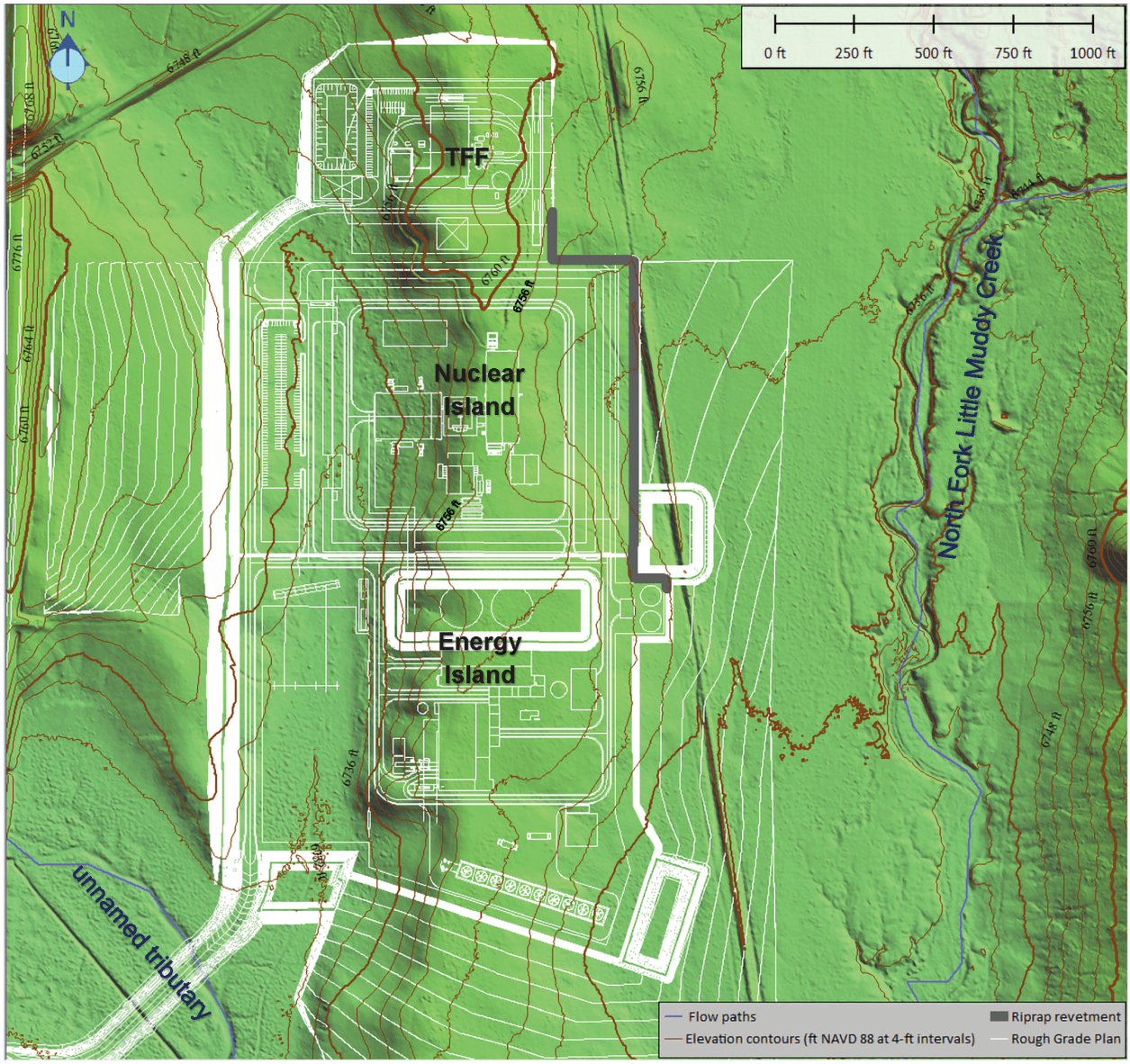



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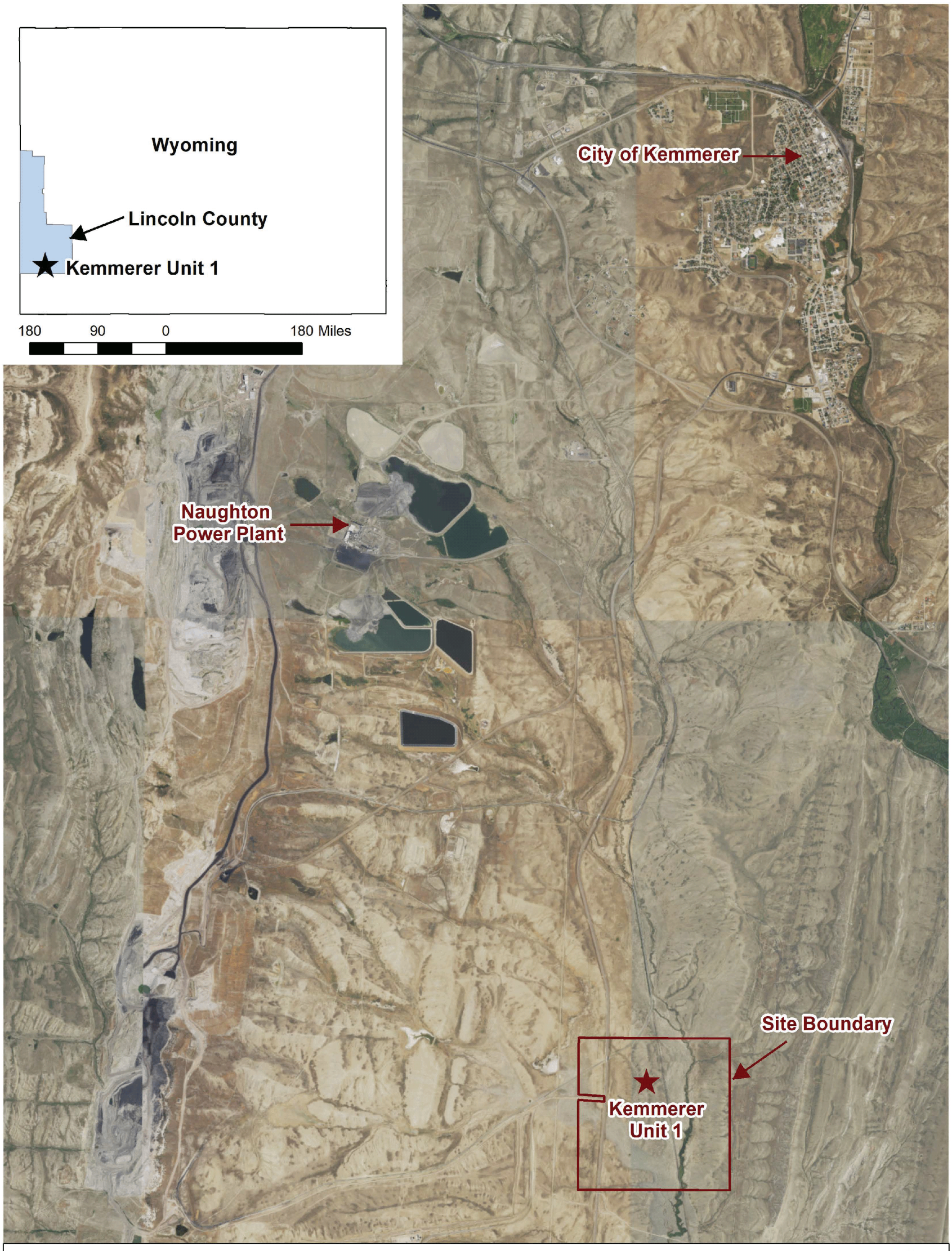
**Figure 2.5-54 Location Map for Kemmerer Unit 1 Site**




MAP SOURCE INFORMATION

Map Generation Date and Location - 9/11/2023 \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\PSAR \Groundwater\Figures\GIS\Location map for Kemmerer Unit 1 site.mxd.
State boundary shapefile downloaded from USGS: https://www.sciencebase.gov/catalog/item/52c78623e4b060b9ebca5be5
County boundary shapefile downloaded from ArcGIS Hub: https://hub.arcgis.com/datasets/esri::usa-counties/explore?location=37.859987%2C-71.124312%2C3.90
Site Boundary shapefile: \\amers.ibechtel.com\power\Projects\Natrium\Environmental\GIS\Site Selection \SiteSelectionData.gdb\Parcels1920_1
Image Source: USGS The National Map: Orthoimagery. Data refreshed December, 2021.

**Location map for Kemmerer Unit 1 site**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 2.5.3.1-1**


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**Figure 2.5-55 Regional Physiographic Provinces in the Kemmerer Unit 1 Site Vicinity**

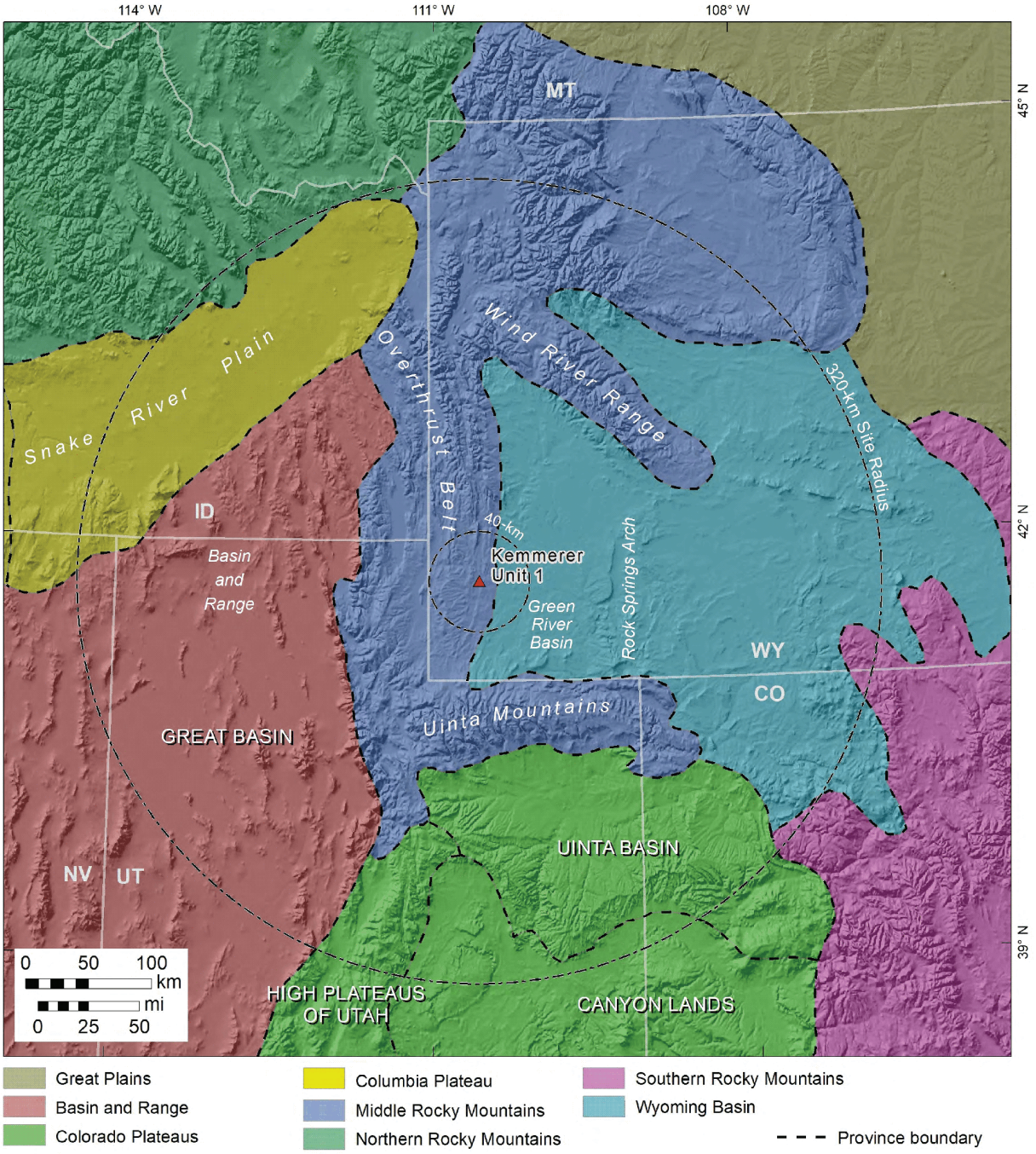



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**Figure 2.5-56 Greater Green River Basin and the Kemmerer Unit 1 Site**

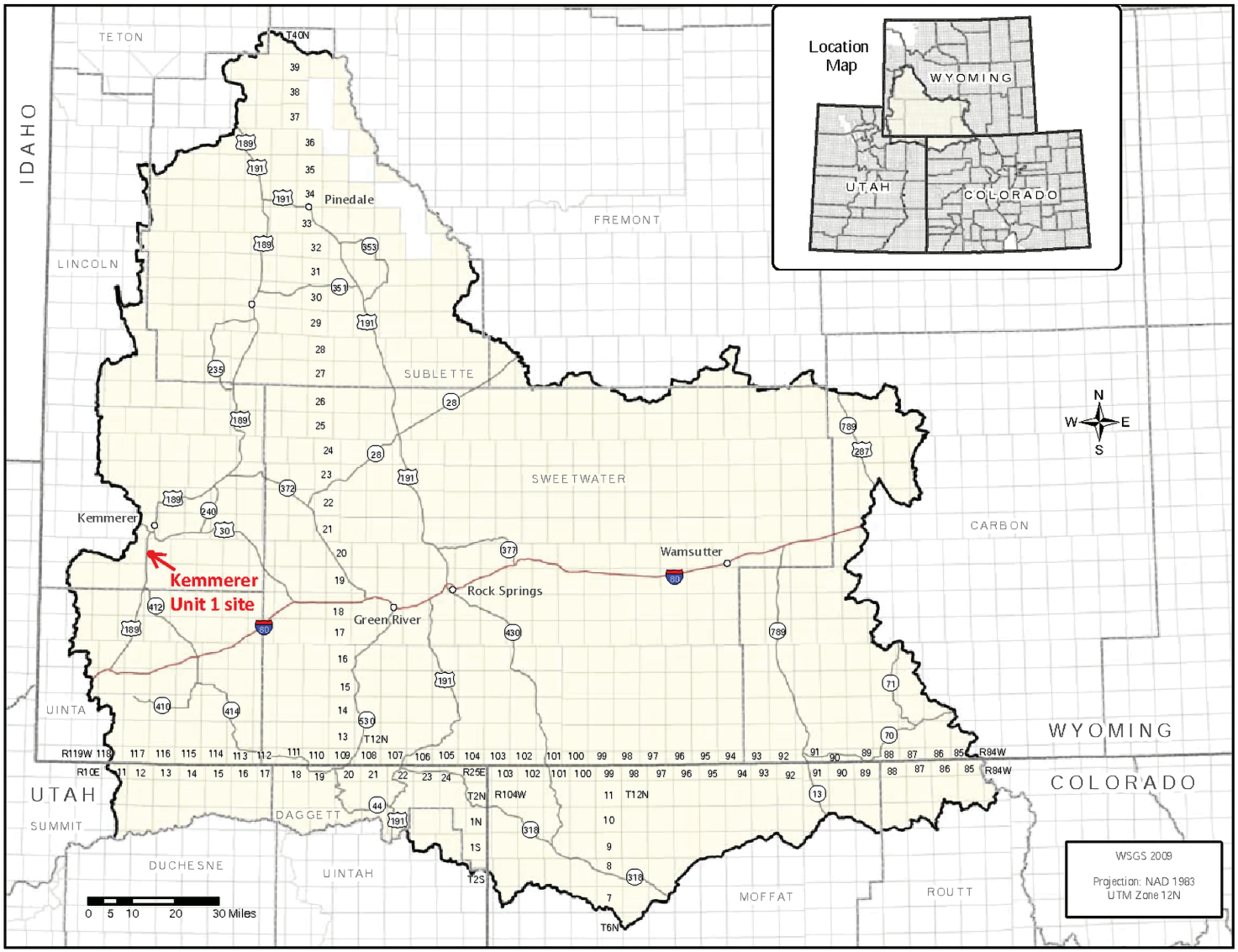




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**Figure 2.5-57 Geologic Features within the Greater Green River Basin**

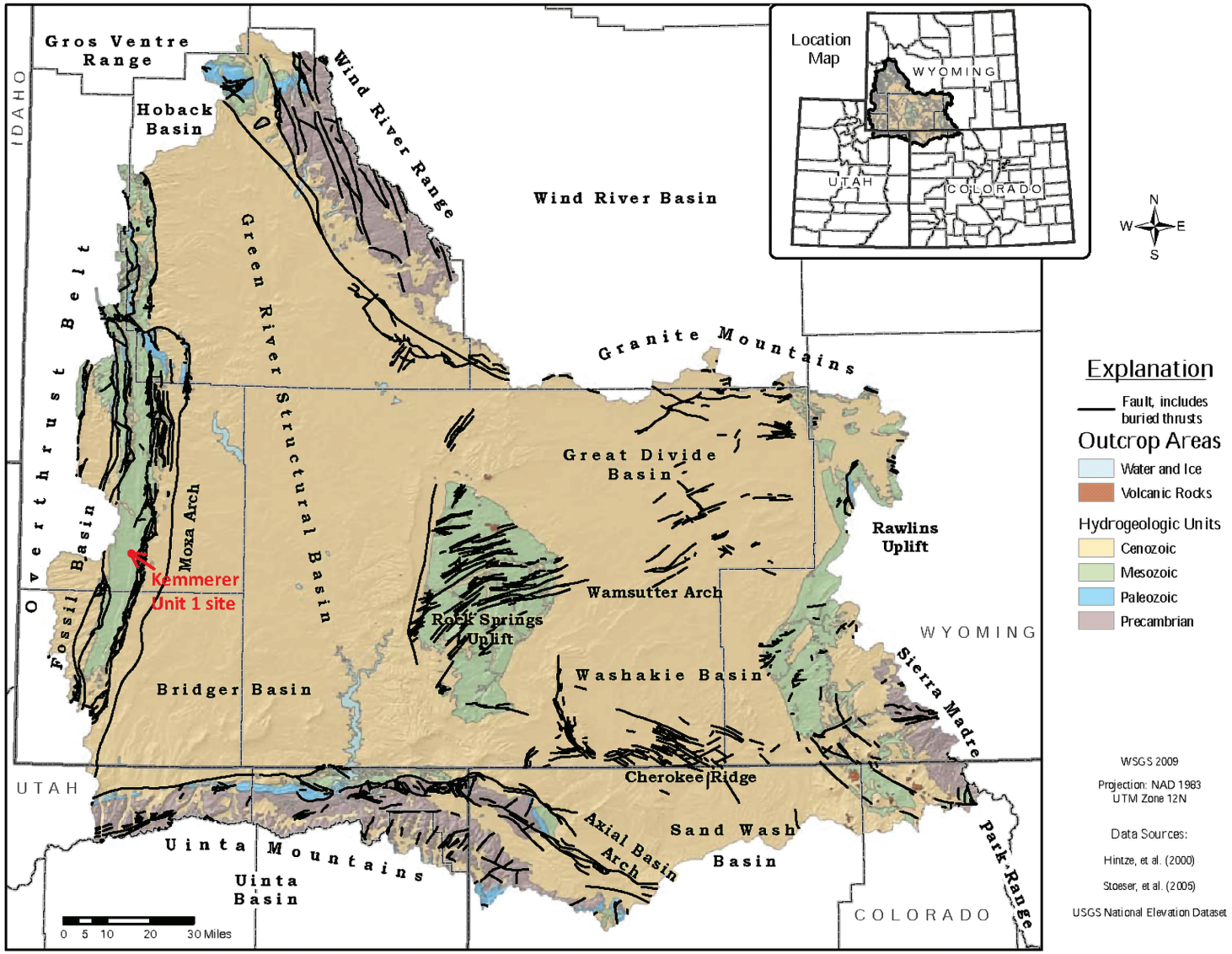



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**Figure 2.5-58 Not Used**



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**Figure 2.5-59 Not Used**



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**Figure 2.5-60 Not Used**


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**Figure 2.5-61 Outcrop Areas and Hydrogeologic Units by Geologic Era in Greater Green River Basin**

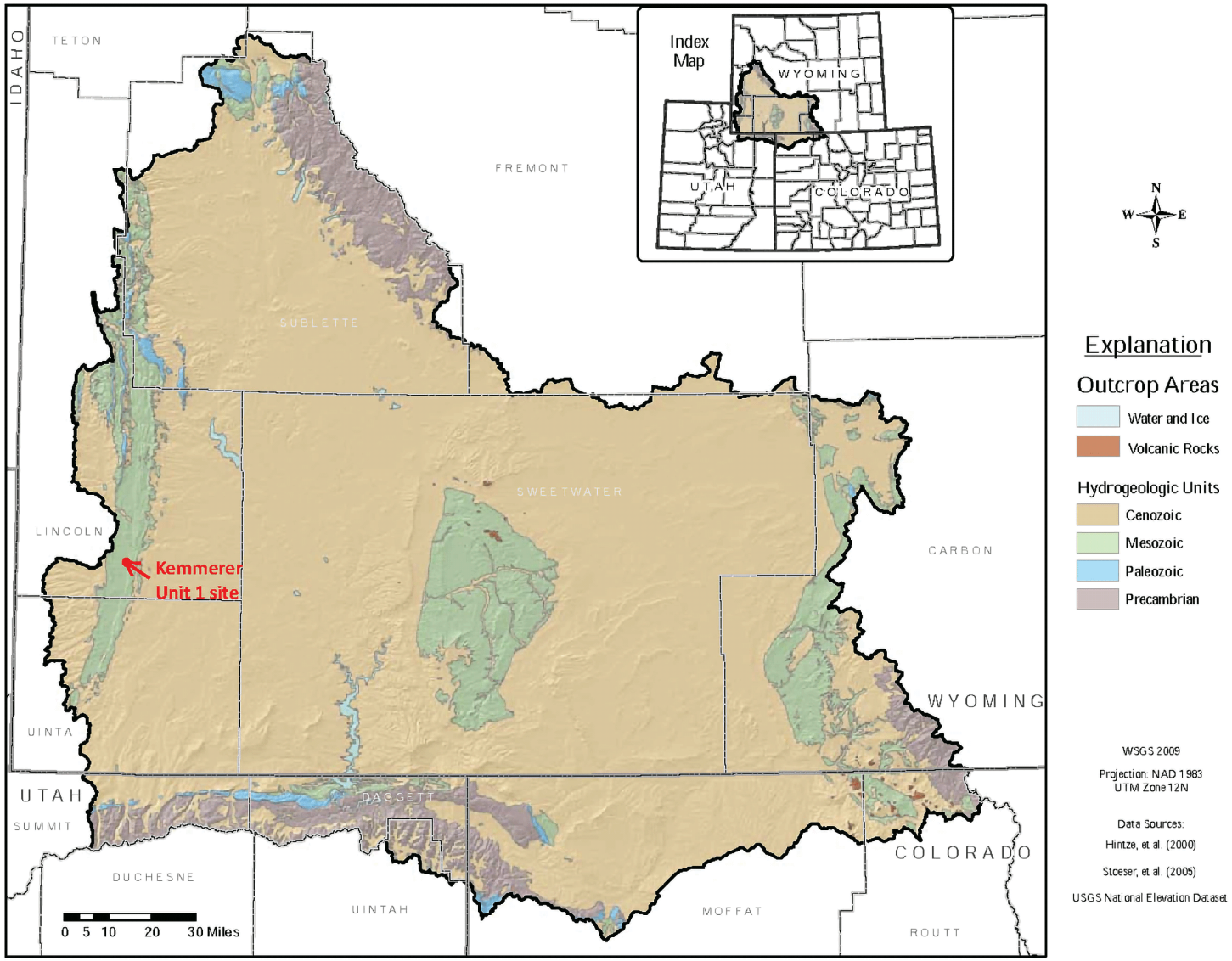



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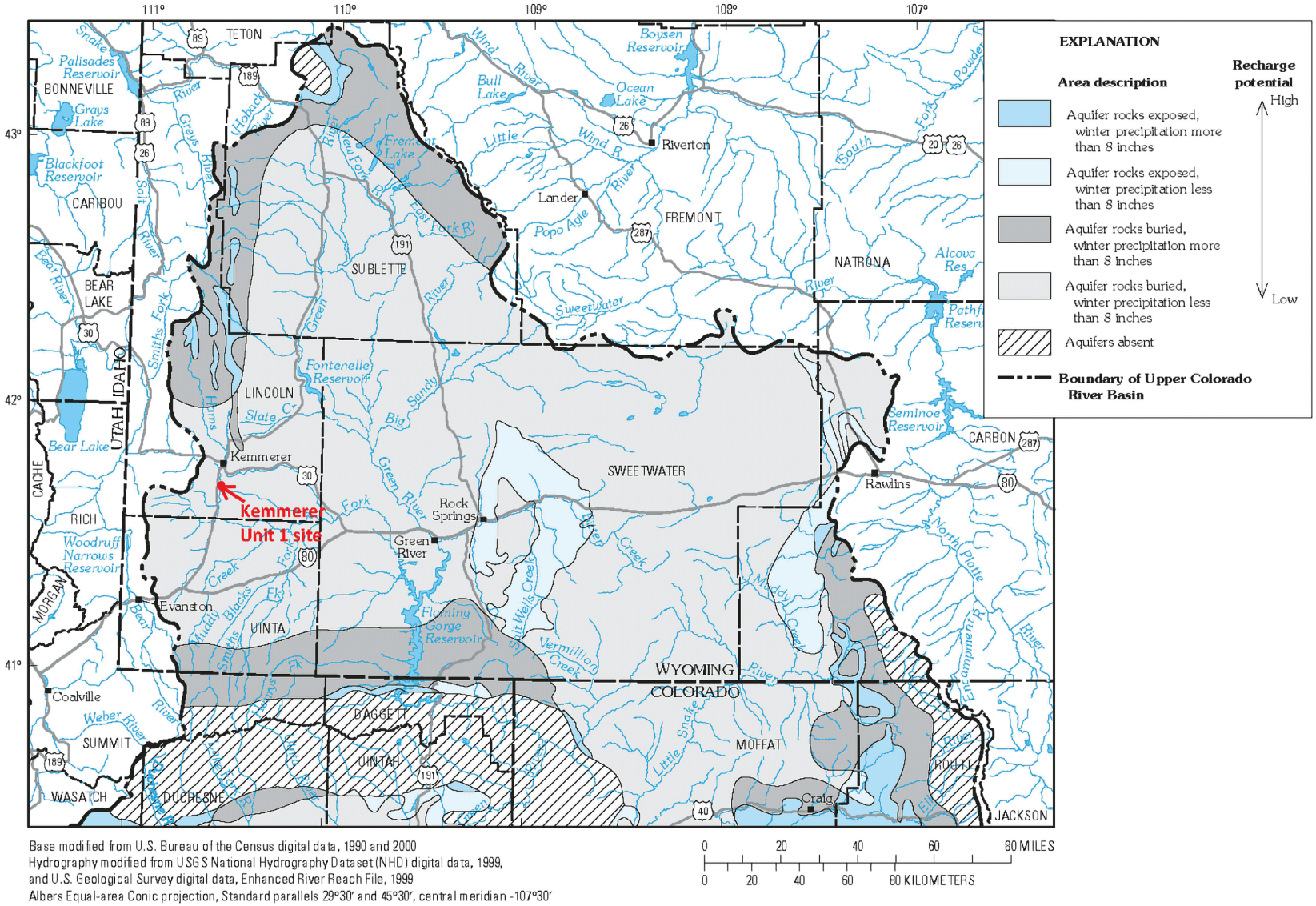
**Figure 2.5-62 Potential for Recharge by Direct Infiltration of Precipitation to Mesozoic Hydrogeologic Units in the Greater Green River Basin**





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## Figure 2.5-63 Hydrogeologic Nomenclature for Mesozoic Rocks in the Greater Green River Basin in Wyoming

<table>
<tr><th>Era</th><th>System</th><th>Component geologic units in Wyoming</th><th colspan="2">Hydrogeologic role/unit of Ahern et al. (1981) for Green River Basin and Overthrust Belt</th><th>Hydrogeologic role/unit of Collentine et al. (1981) for Great Divide and Washakie Basins</th><th>Hydrogeologic unit of Taylor et al. (1986)</th><th>Hydrogeologic unit of Freethey and Cordy (1991)</th><th colspan="2">Hydrogeologic unit used in this report</th></tr>
<tr><td rowspan="11">MESOZOIC</td><td rowspan="5">Cretaceous</td><td>Lance Formation and Fox Hills Sandstone</td><td colspan="2">Minor aquifer (part of Tertiary aquifer system)</td><td>Minor aquifer (part of Tertiary aquifer system)</td><td rowspan="6">Upper Mesozoic confining layers and aquifers</td><td rowspan="3">Mesaverde aquifer</td><td>Lance-Fox Hills aquifer</td><td rowspan="3">Mesaverde aquifer</td></tr>
<tr><td>Lewis Shale</td><td colspan="2">Aquitard</td><td>Major aquitard</td><td>Lewis confining unit</td></tr>
<tr><td>Mesaverde Group/ Formation, Lance Formation, Fox Hills Sandstone, and Adaville Formation</td><td colspan="2">Major aquifer-Mesaverde-Adaville aquifer</td><td>Major aquifer-Mesaverde aquifer</td><td>Mesaverde aquifer</td></tr>
<tr><td rowspan="2">Aspen Shale, Baxter Shale, Mowry Shale, Steele Shale, Cody Shale, Thermopolis Shale, Hilliard Shale, Muddy Sandstone, Frontier Formation, Niobrara Formation, and Blind Bull Formation</td><td colspan="2">Baxter and Hilliard Shales-Major aquitard</td><td>Baxter Shale and equivalents-Major aquitard</td><td>Mancos confining unit</td><td colspan="2">Baxter-Mowry confining unit</td></tr>
<tr><td>Frontier Formation-Minor aquifer (Frontier aquifer)</td><td rowspan="2">Upper Jurassic-Lower Cretaceous aquifers</td><td>Frontier Formation-Minor aquifer (Frontier aquifer)</td><td colspan="3"></td></tr>
<tr><td rowspan="6">Jurassic</td><td>Cloverly Formation, Bear River Formation, and Gannett Group</td><td colspan="2">Aspen Shale, Bear River Formation, and Gannett Group-Discontinuous aquifers with local confining beds</td><td>Minor aquifer (Cloverly aquifer)</td><td>Dakota aquifer</td><td colspan="2">Cloverly aquifer</td></tr>
<tr><td>Morrison Formation</td><td colspan="2">Not present in area</td><td>Aquitard</td><td rowspan="5">Middle Mesozoic aquifers</td><td>Morrison confining unit and aquifer</td><td colspan="2">Morrison confining unit</td></tr>
<tr><td>Sundance Formation and Stump Formation</td><td colspan="2" rowspan="2">Stump Formation and Preuss Sandstone-Aquitard</td><td rowspan="3">Sundance Formation and Nugget Sandstone-Minor aquifer (Sundance-Nugget aquifer)</td><td>Curtis-Stump confining unit</td><td colspan="2">Sundance confining unit</td></tr>
<tr><td>Sundance Formation and Preuss Sandstone</td><td>Entrada-Preuss aquifer</td><td colspan="2">Sundance aquifer</td></tr>
<tr><td>Gypsum Spring Formation and Twin Creek Limestone</td><td colspan="2">Twin Creek Limestone-Minor aquifer</td><td>Carmel-Twin Creek confining unit</td><td colspan="2">Gypsum Spring confining unit</td></tr>
<tr><td>Nugget Sandstone</td><td rowspan="2">Nugget Sandstone-Major aquifer Ankareh Formation-Minor aquifer Thaynes Limestone-Major aquifer</td><td rowspan="2">Nugget aquifer system</td><td>Navajo-Nugget aquifer</td><td colspan="2">Nugget aquifer</td></tr>
<tr><td>Triassic</td><td>Dinwoody Formation, Ankareh Formation, Chugwater Group/ Formation, Goose Egg Formation, Thaynes Limestone, and Woodside Shale</td><td>Chugwater and Phosphoria Formations-aquitard</td><td>Lower Mesozoic confining layers</td><td>Chinle-Moenkopi confining unit</td><td colspan="2">Chugwater-Dinwoody confining unit</td></tr>
<tr><td colspan="3">Dinwoody Formation and Woodside Shale-aquitard</td></tr>
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**Figure 2.5-64 Mean Annual Precipitation for Lincoln County, Wyoming, 1951-80**

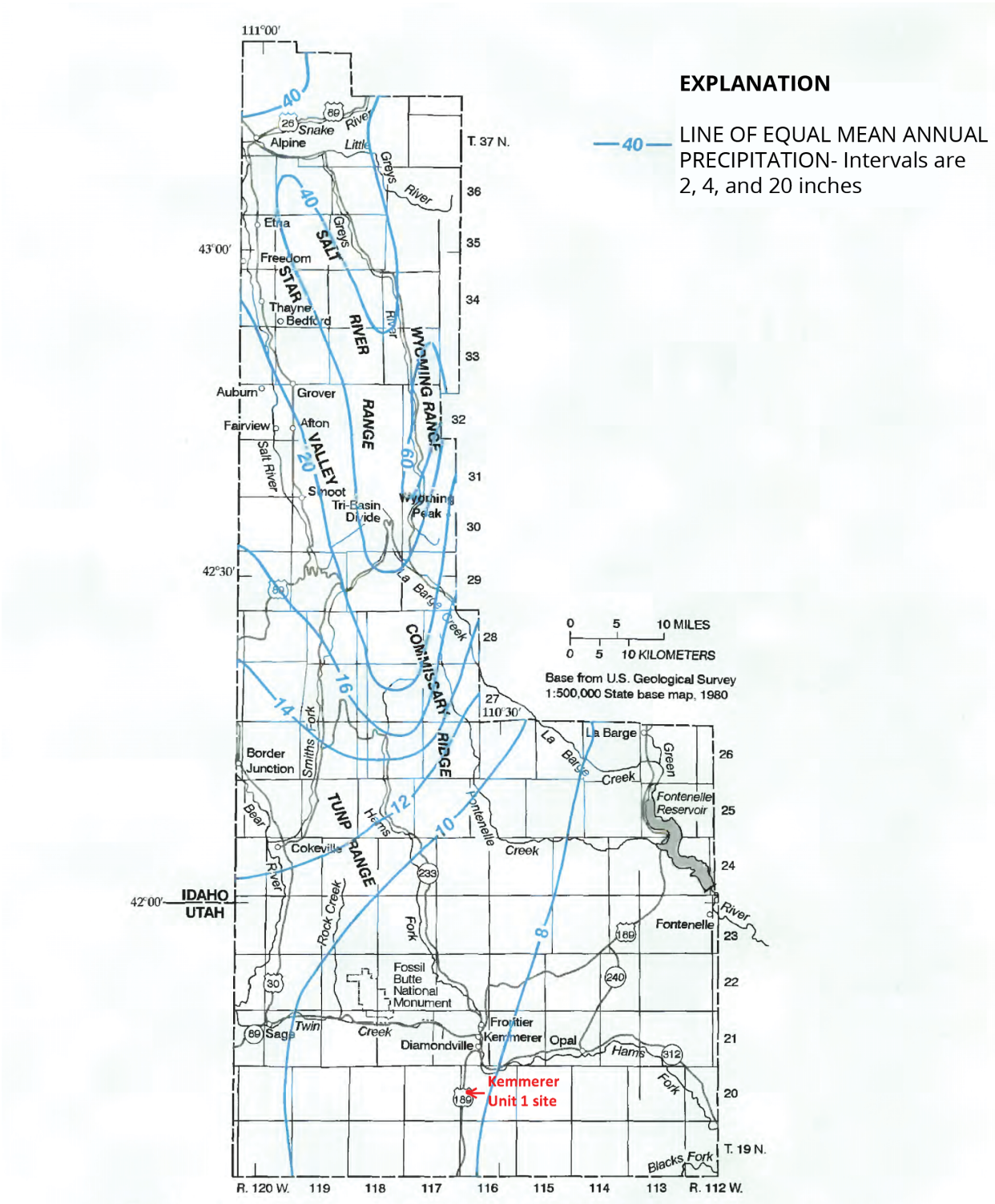


Figure 2. Mean annual precipitation for Lincoln County, Wyoming, 1951-80 (modified from Martner, 1986, fig.6.1)


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**Figure 2.5-65 Location of Groundwater-Dependent Towns with Relative to the Kemmerer Unit 1 Site**

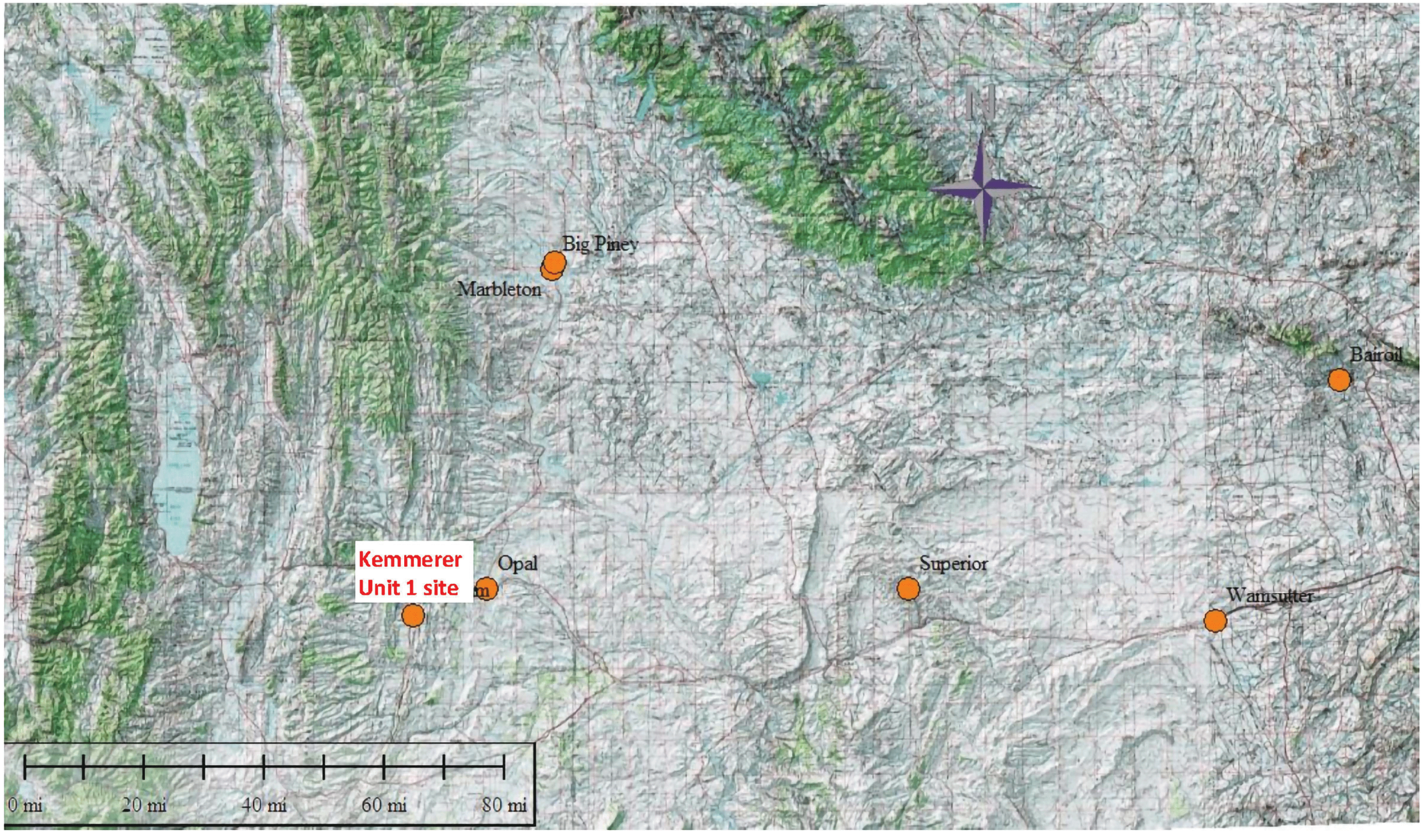






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**Figure 2.5-66 Permitted Groundwater Extraction Wells within a 5-Mile Radius of the Kemmerer Unit 1 Site (DOM_GW = domestic groundwater; STK = livestock; IND_GW = industrial groundwater; MIS = miscellaneous)**

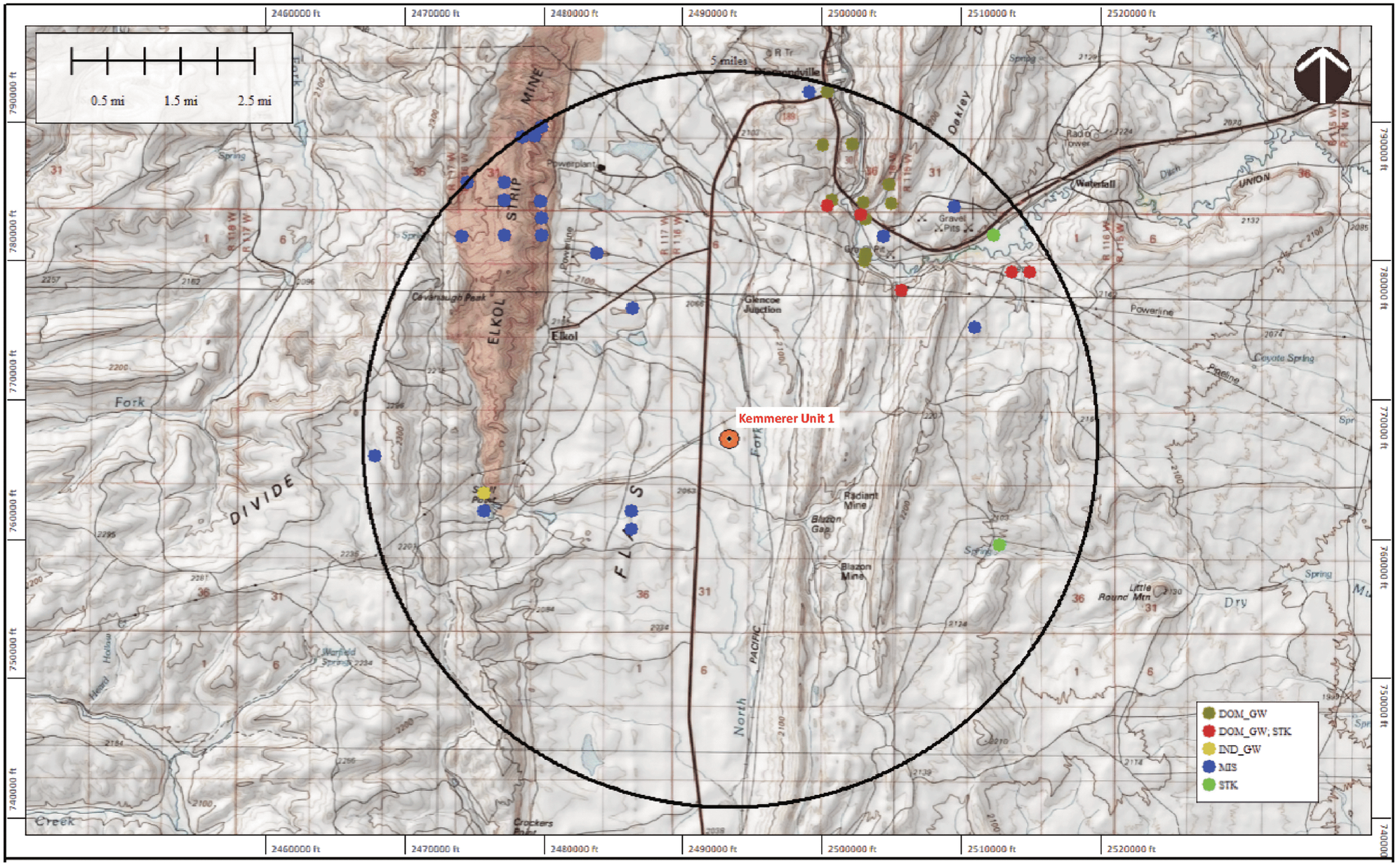



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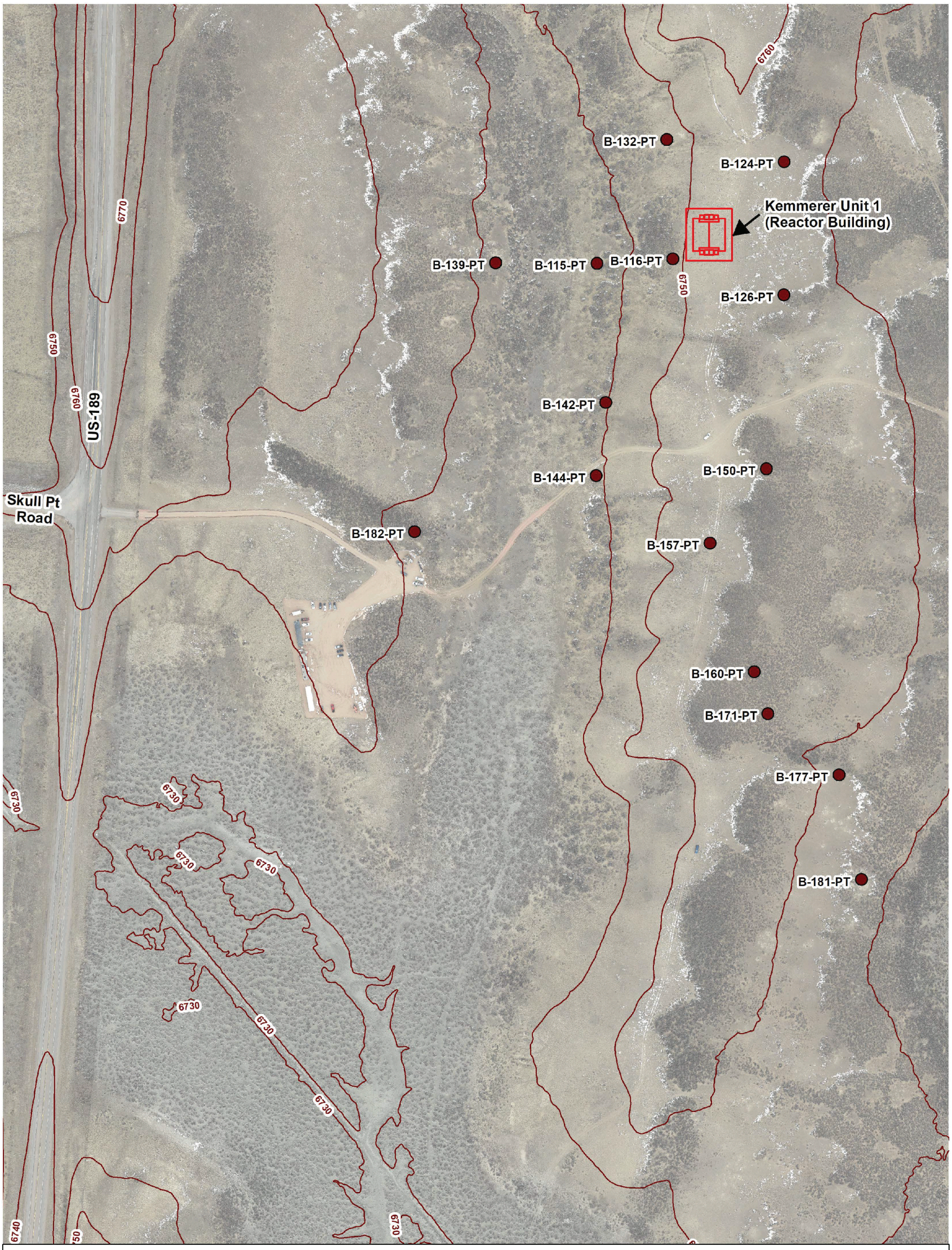
**Figure 2.5-67 Locations of Borehole Packer Tests**




MAP SOURCE INFORMATION

Map Generation Date and Location - 9/11/2023 \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\PSAR\Groundwater\Figures\GIS\Locations of borehole packer tests.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane)
Shapefile for Packer Test Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Packer -Calculation\shapefile\packer.shp.
Reactor Building shapefile: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\KMZ\Plot Plan Rev 1 and grading.gdb

**Locations of borehole packer tests**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure**

**2.5.3.1-14**


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**Figure 2.5-68 Rock Hydraulic Conductivity as a Function of Depth Below Ground Surface**

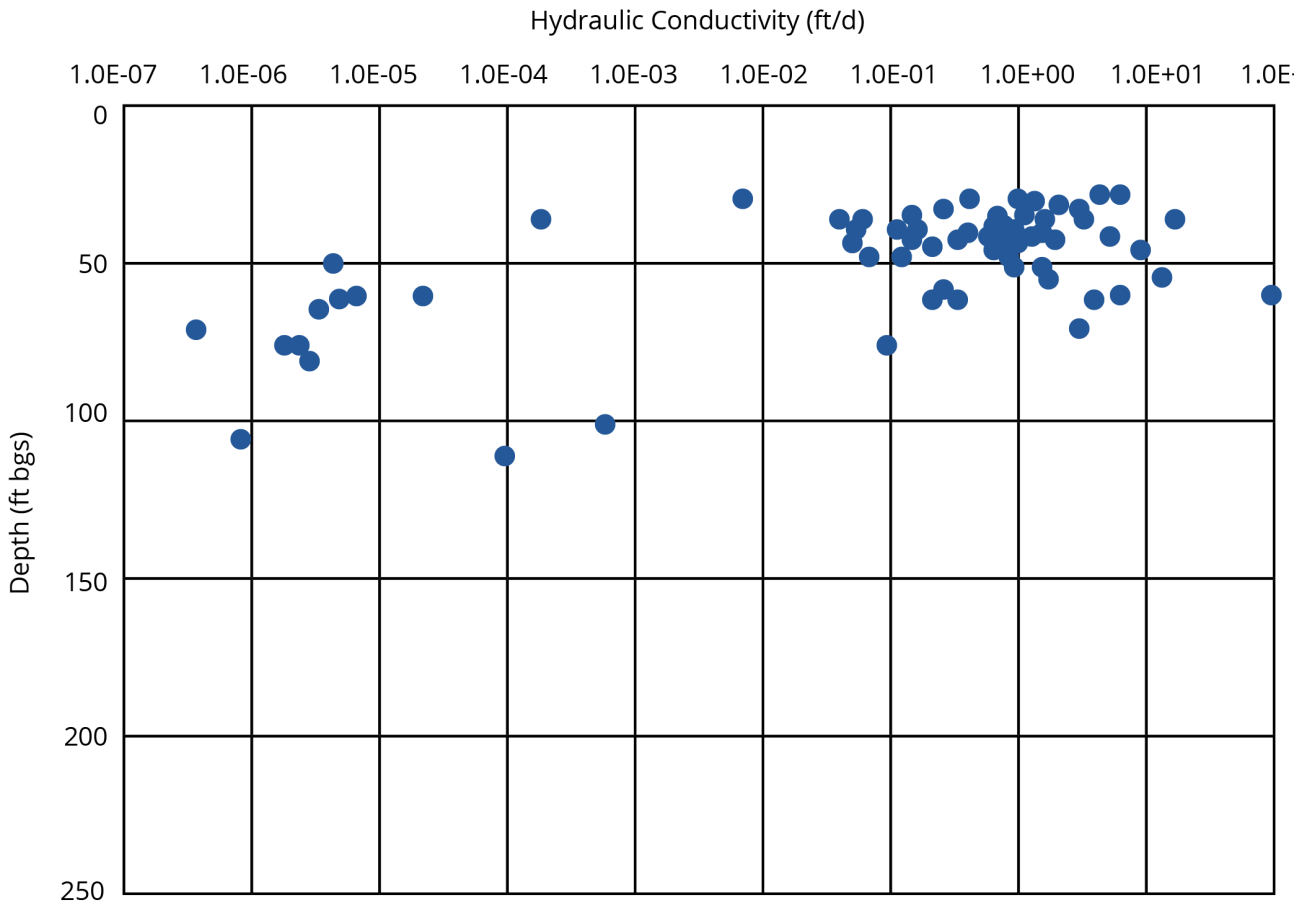






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**Figure 2.5-69 Rock Porosity as a Function of Depth Below Ground Surface**

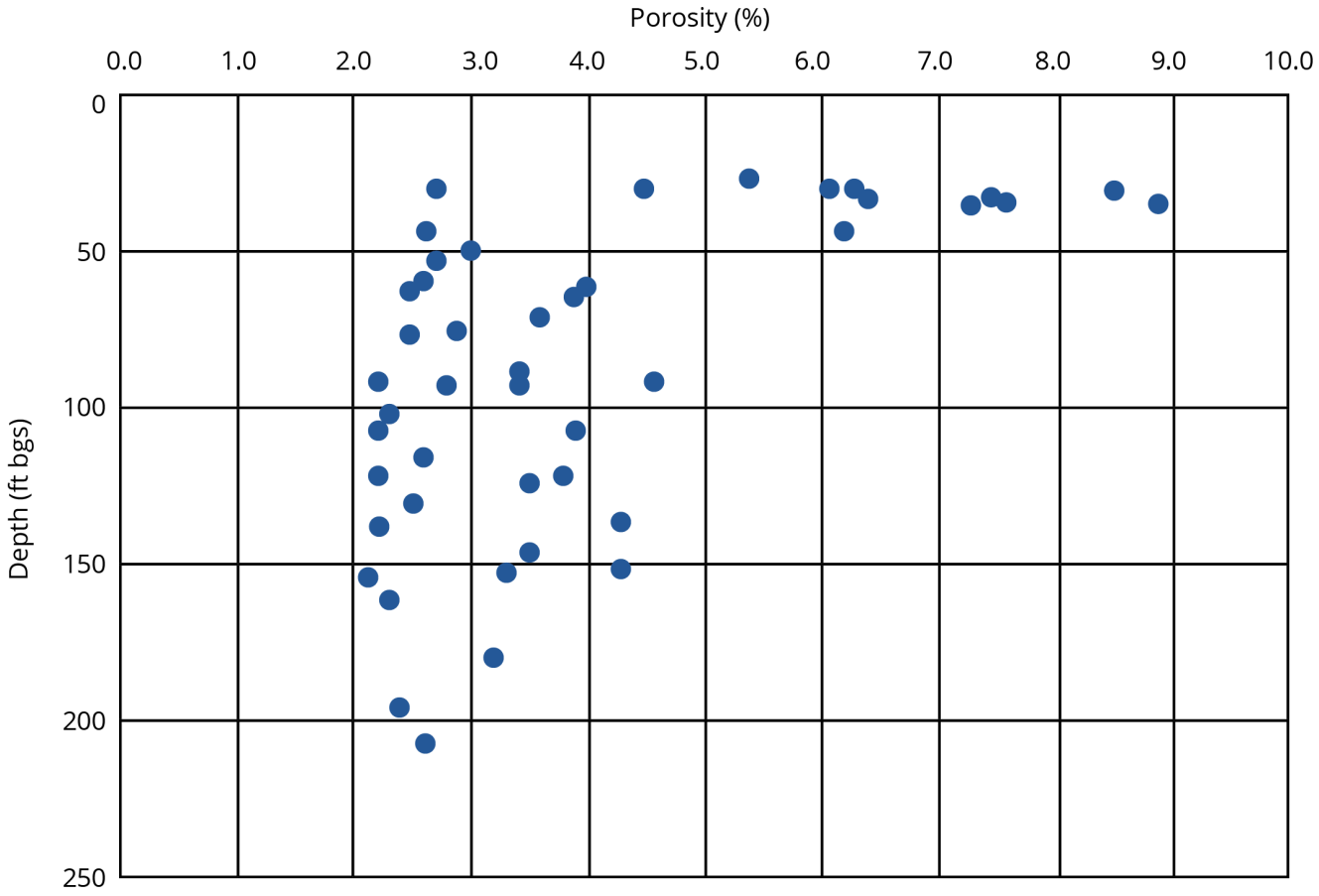






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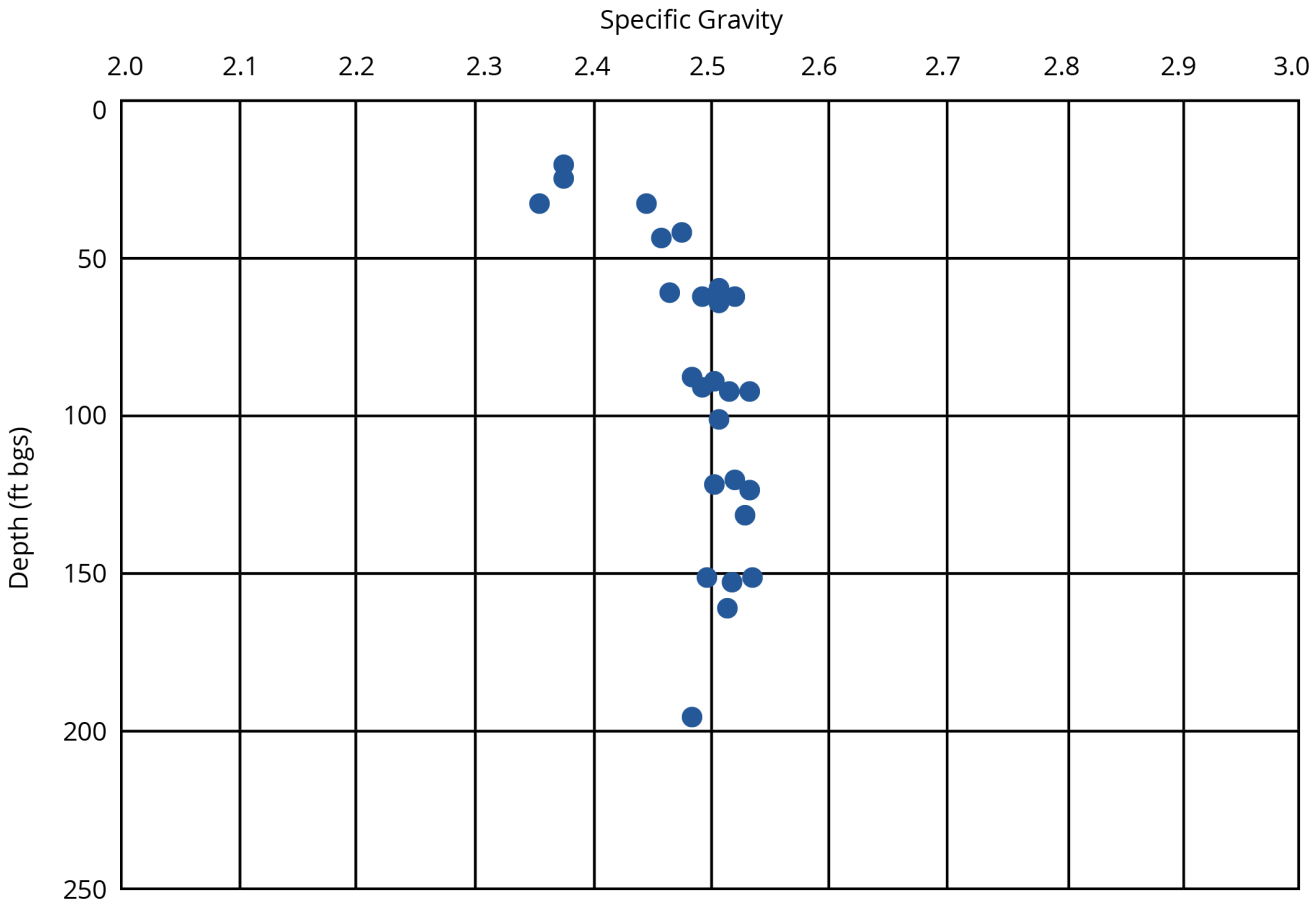
**Figure 2.5-70 Rock Specific Gravity as a Function of Depth Below Ground Surface**



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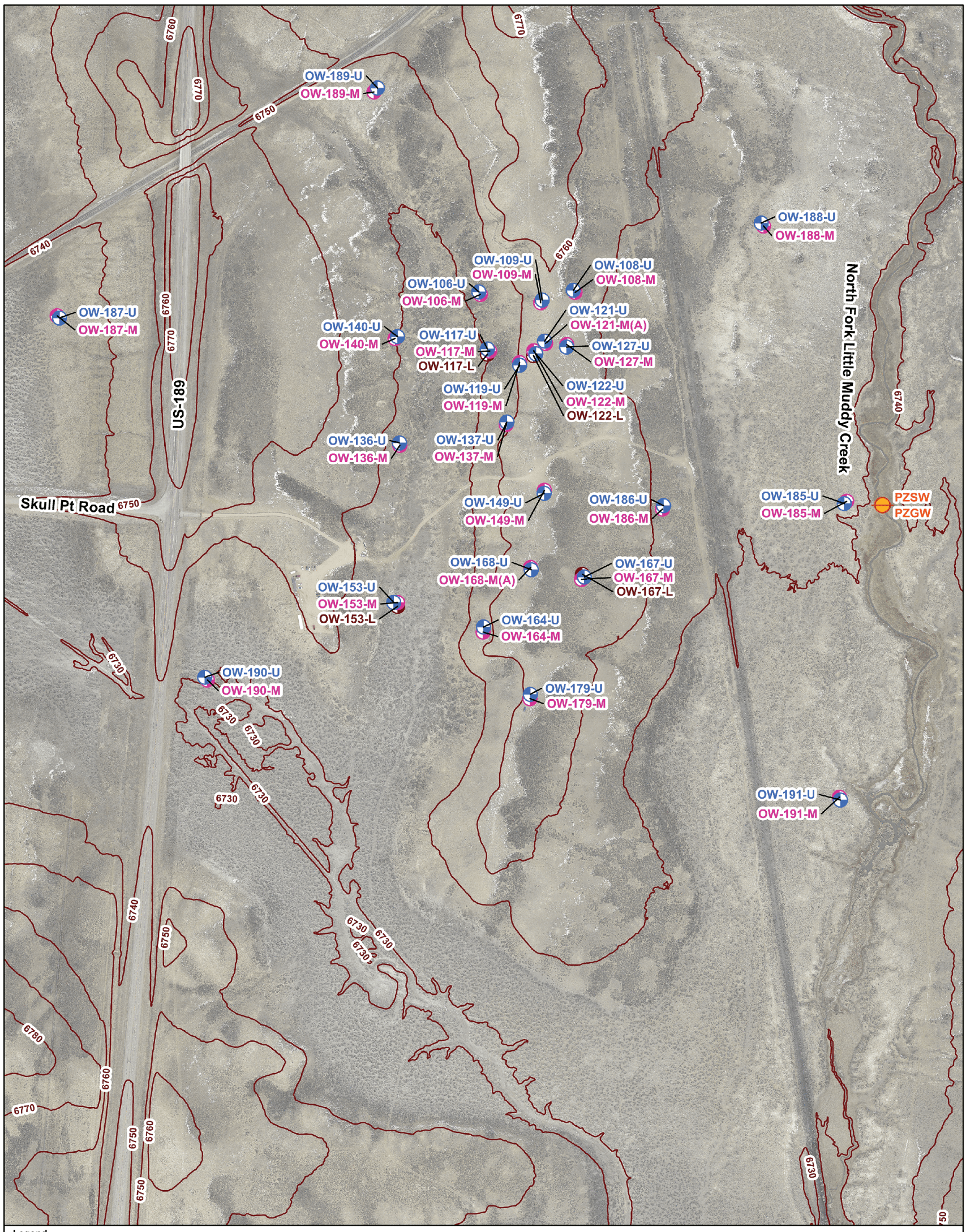
## Figure 2.5-71 Observation Well Locations




**Legend**

- Shallow Observation Well
- Intermediate Observation Well
- Deep Observation Well
- Piezometer
- LiDAR Topographic Contour Line (ft)

N W E S

0 175 350 700 Feet

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/9/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Surface Water and Groundwater Level Observation Location Map.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Location (Well, Piezometer, and Surface Water): \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_March_2023.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988

**Location Map**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 1**


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**Figure 2.5-72 Hydrographs for Observation Wells OW-106-U and M**

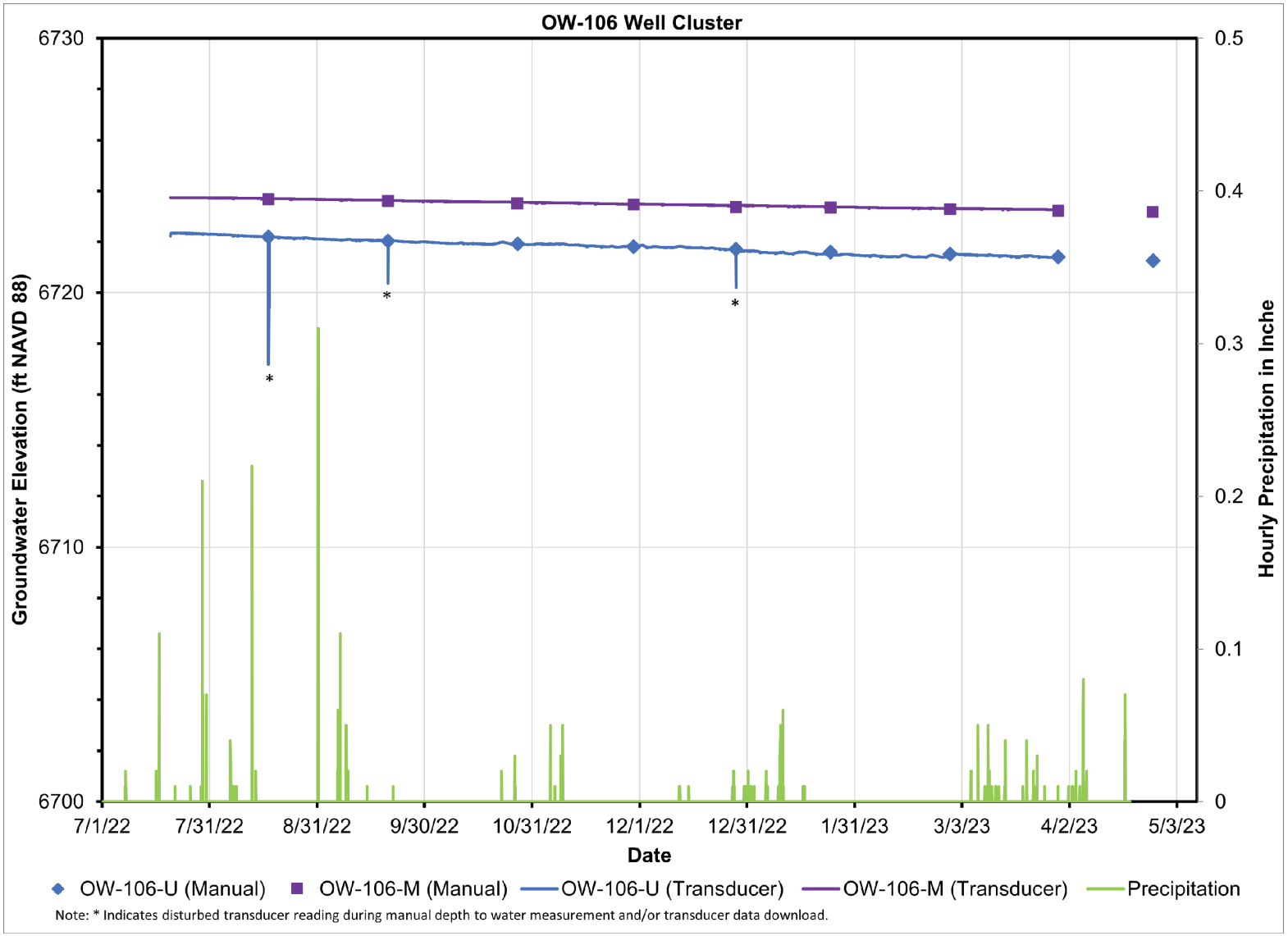




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**Figure 2.5-73 Hydrographs for Observation Wells OW-108-U and M**

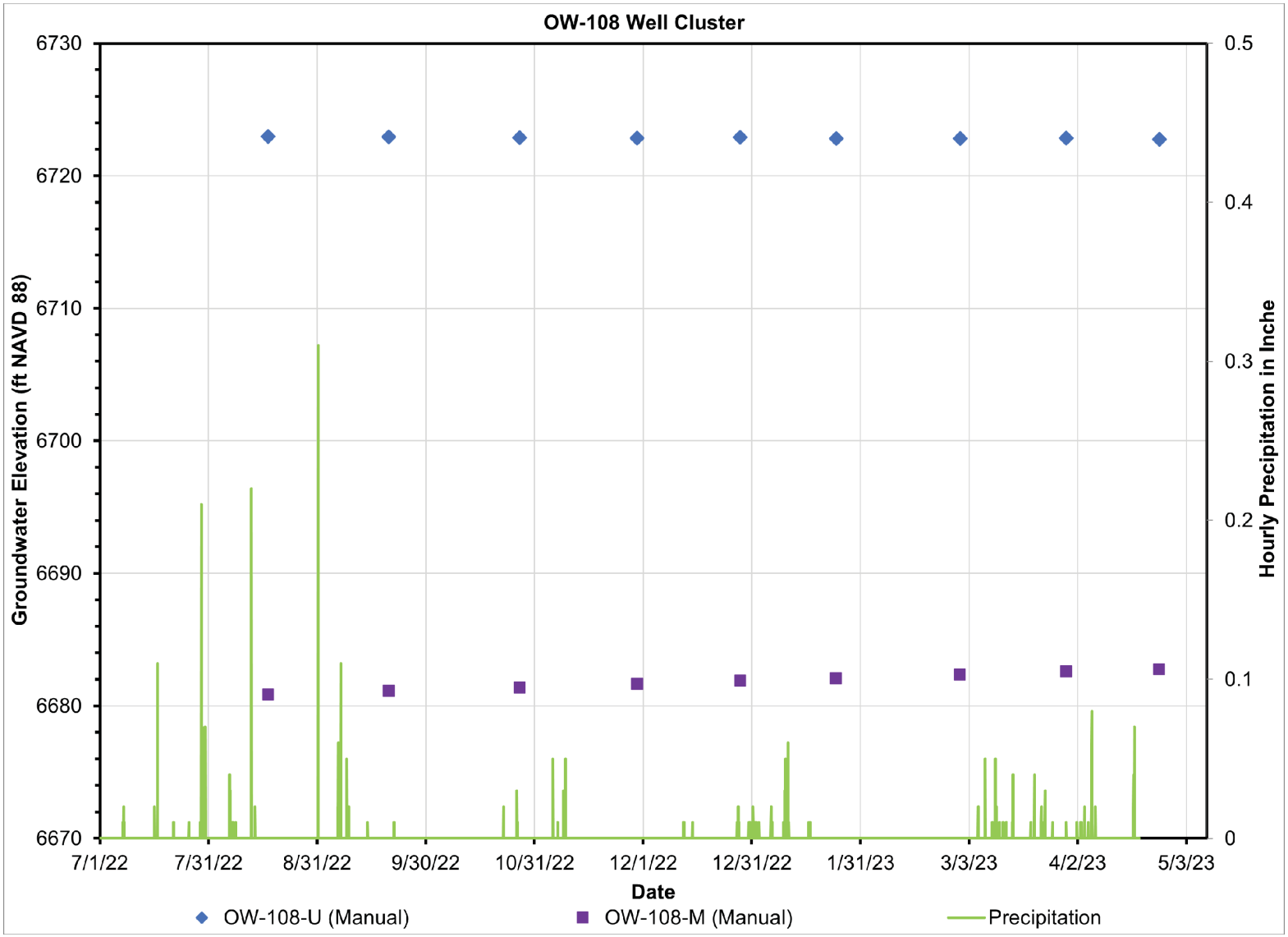




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**Figure 2.5-74 Hydrographs for Observation Wells OW-109-U and M**

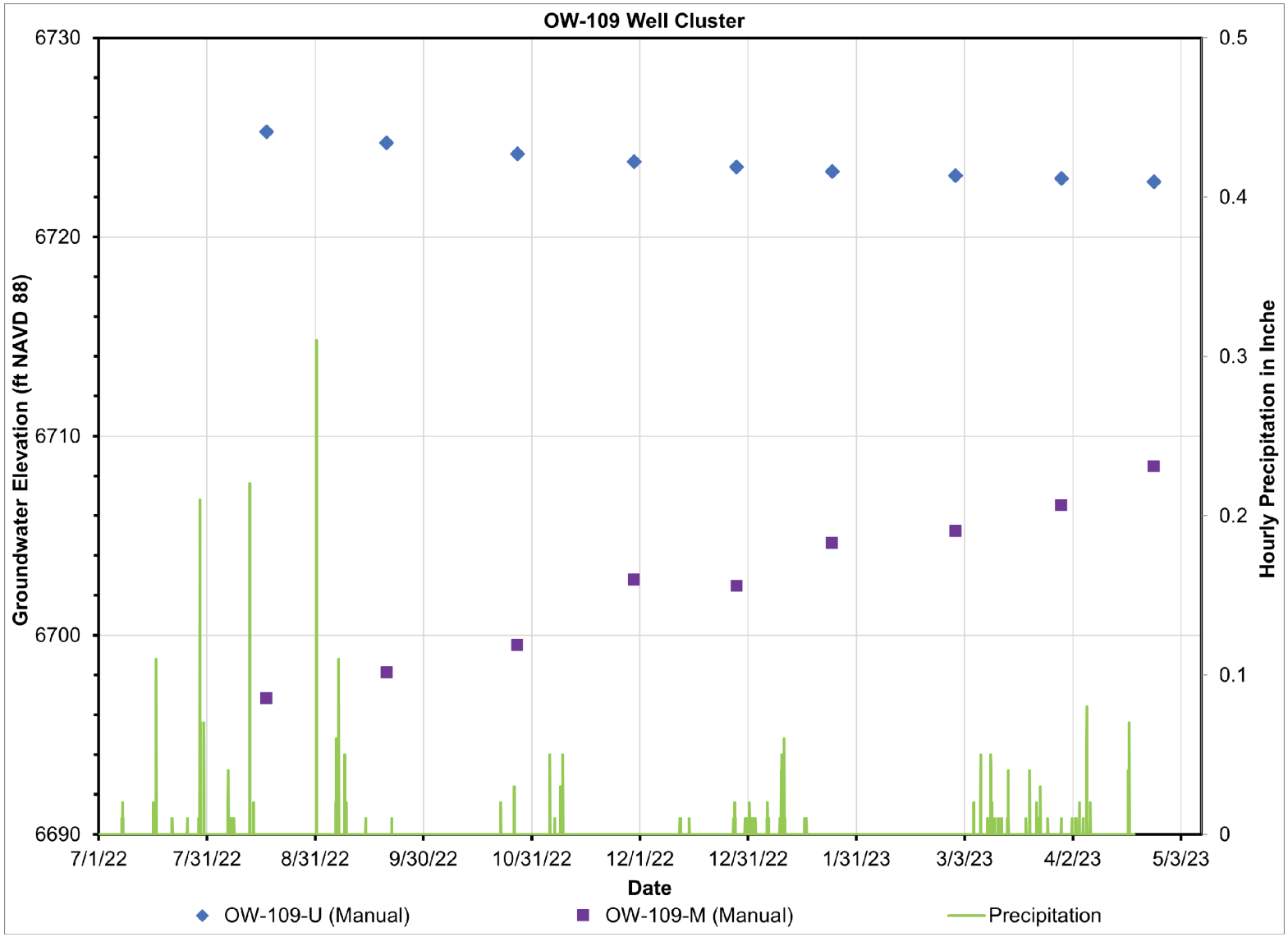






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**Figure 2.5-75 Hydrographs for Observation Wells OW-117-U, M, and L (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download)**

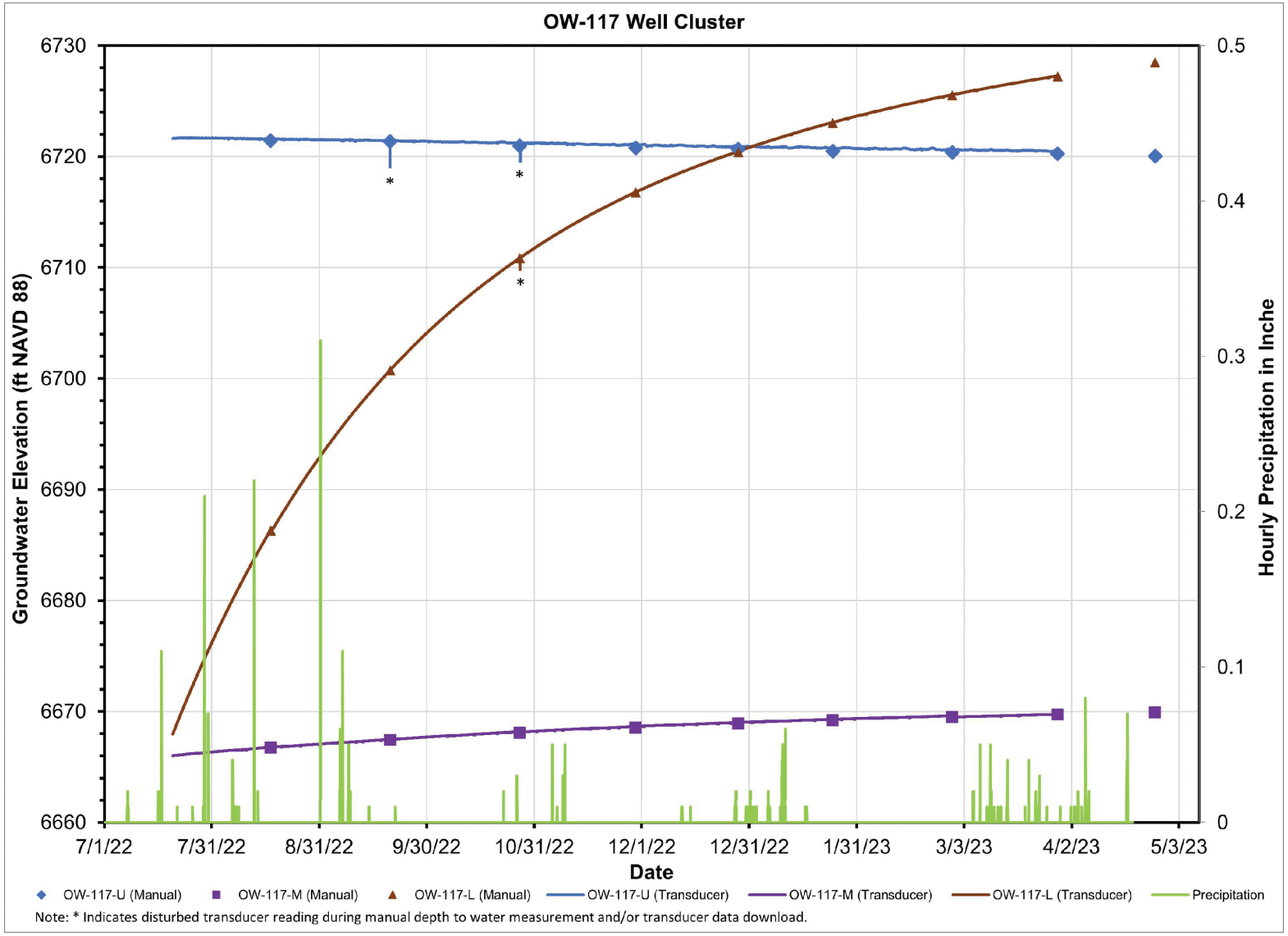




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**Figure 2.5-76 Hydrographs for Observation Wells OW-119-U and M**

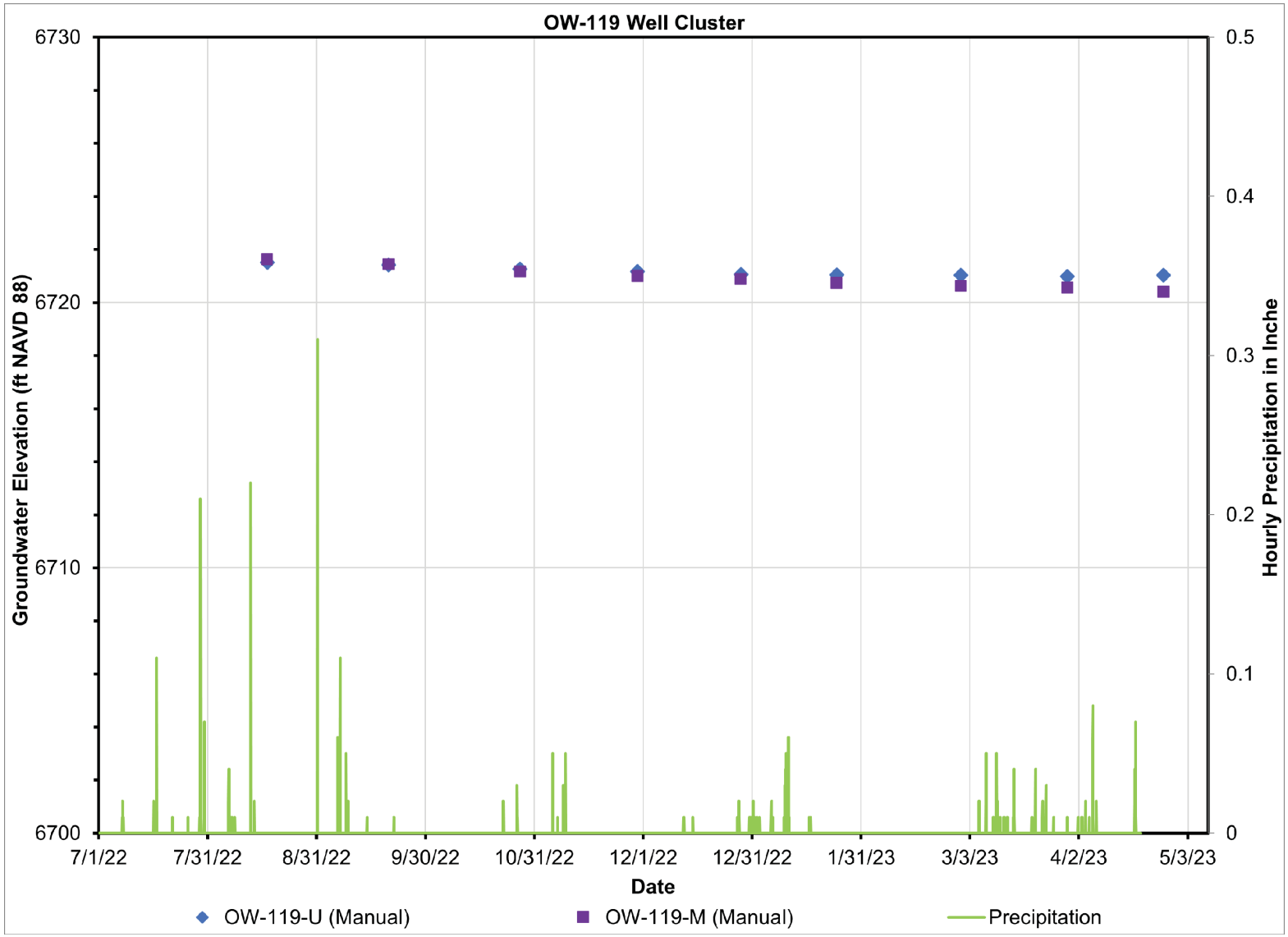






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**Figure 2.5-77 Hydrographs for Observation Wells OW-121-U and M**

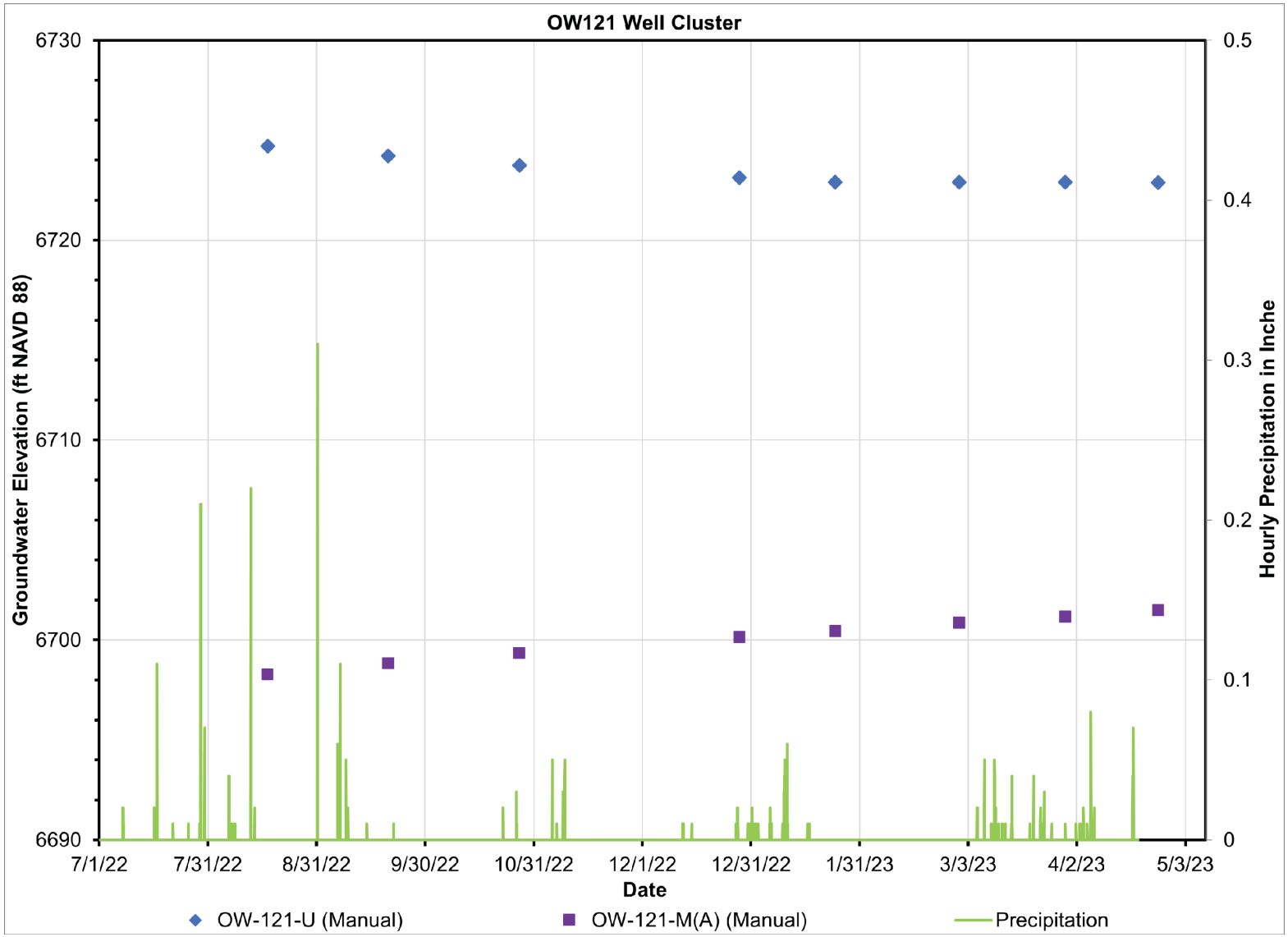






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**Figure 2.5-78 Hydrographs for Observation Wells OW-122-U, M, and L (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download)**

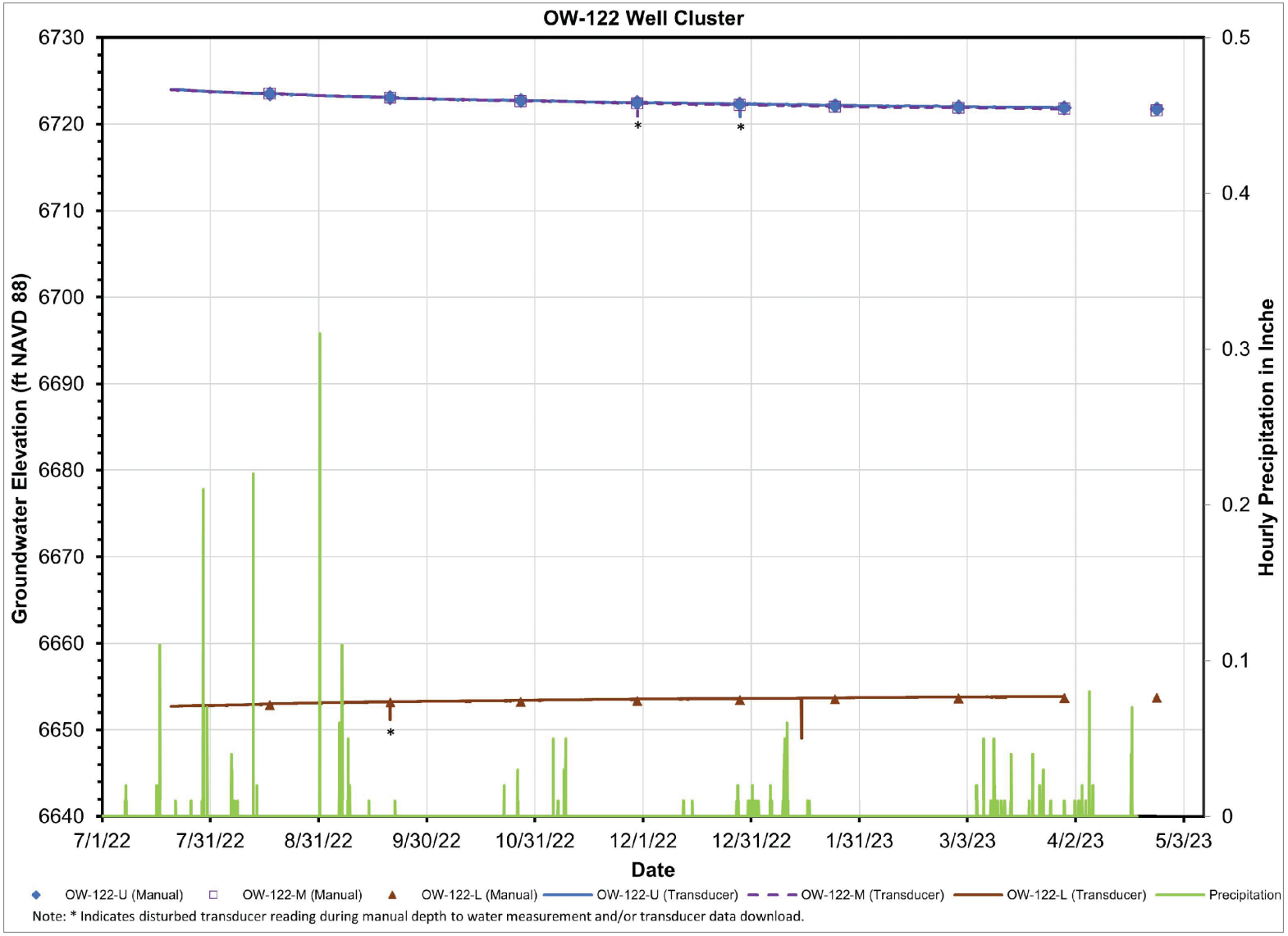




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**Figure 2.5-79 Hydrographs for Observation Wells OW-127-U and M**

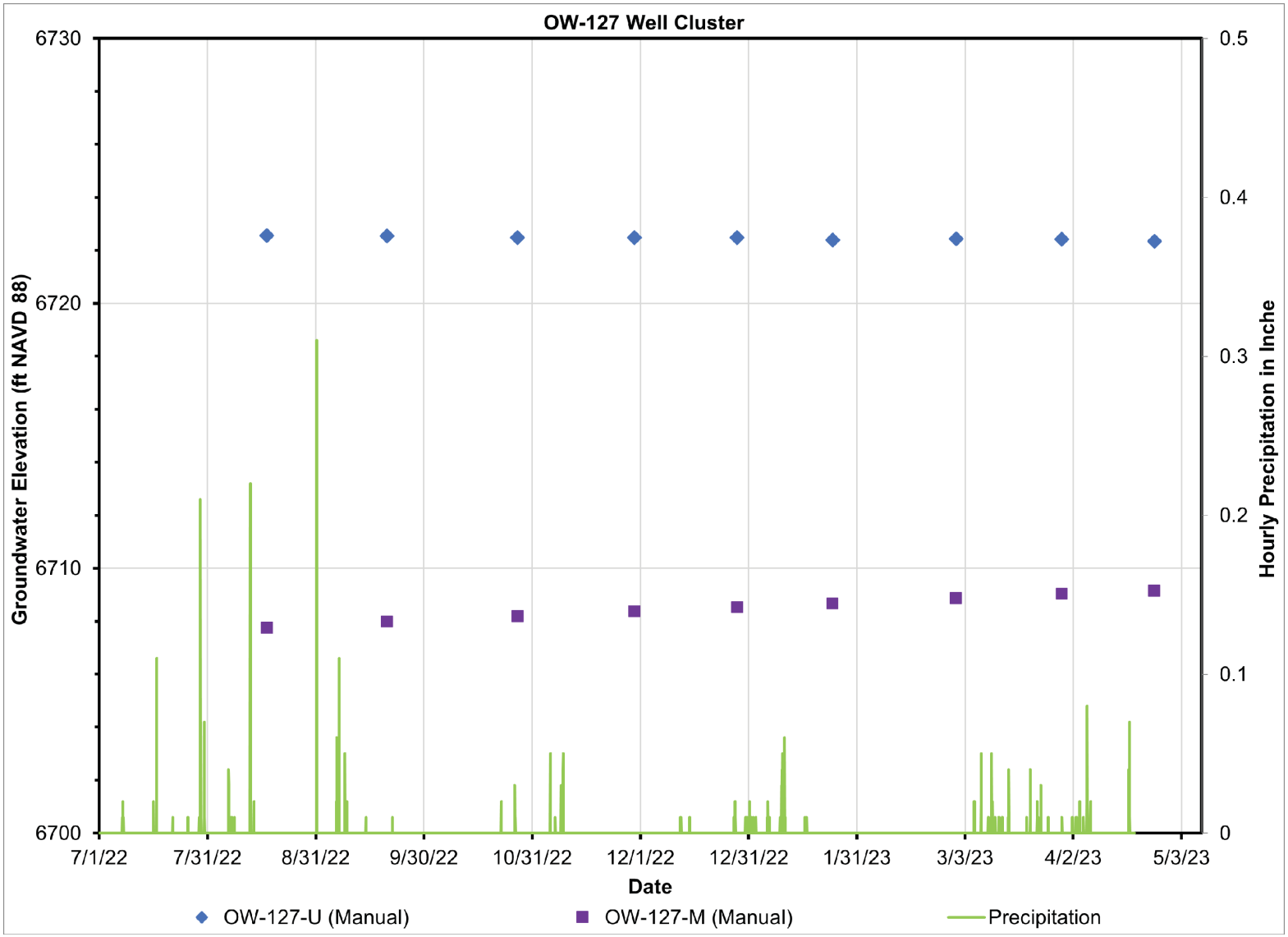




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**Figure 2.5-80 Hydrographs for Observation Wells OW-136-U and M**

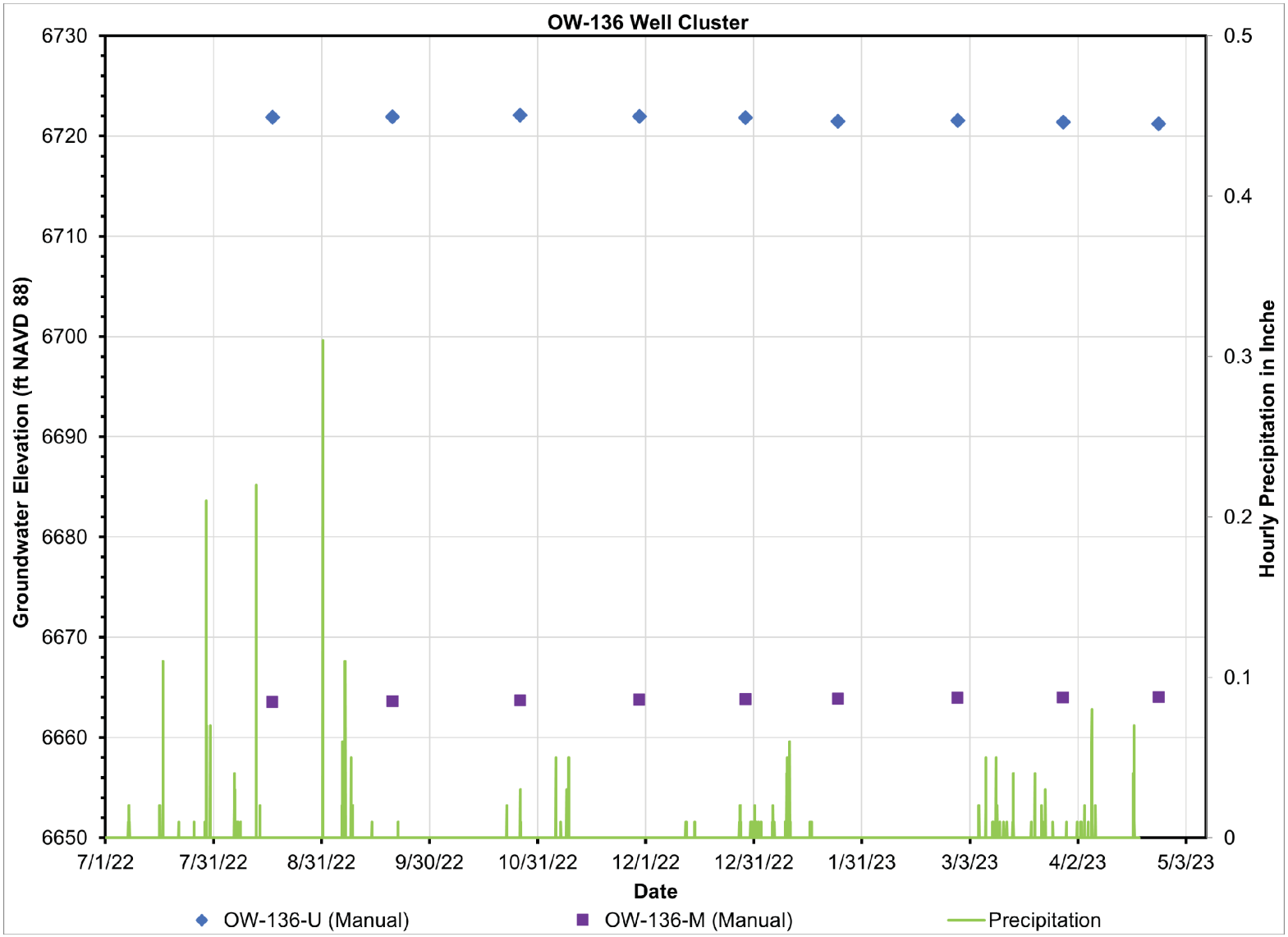






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**Figure 2.5-81 Hydrographs for Observation Wells OW-137-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download)**

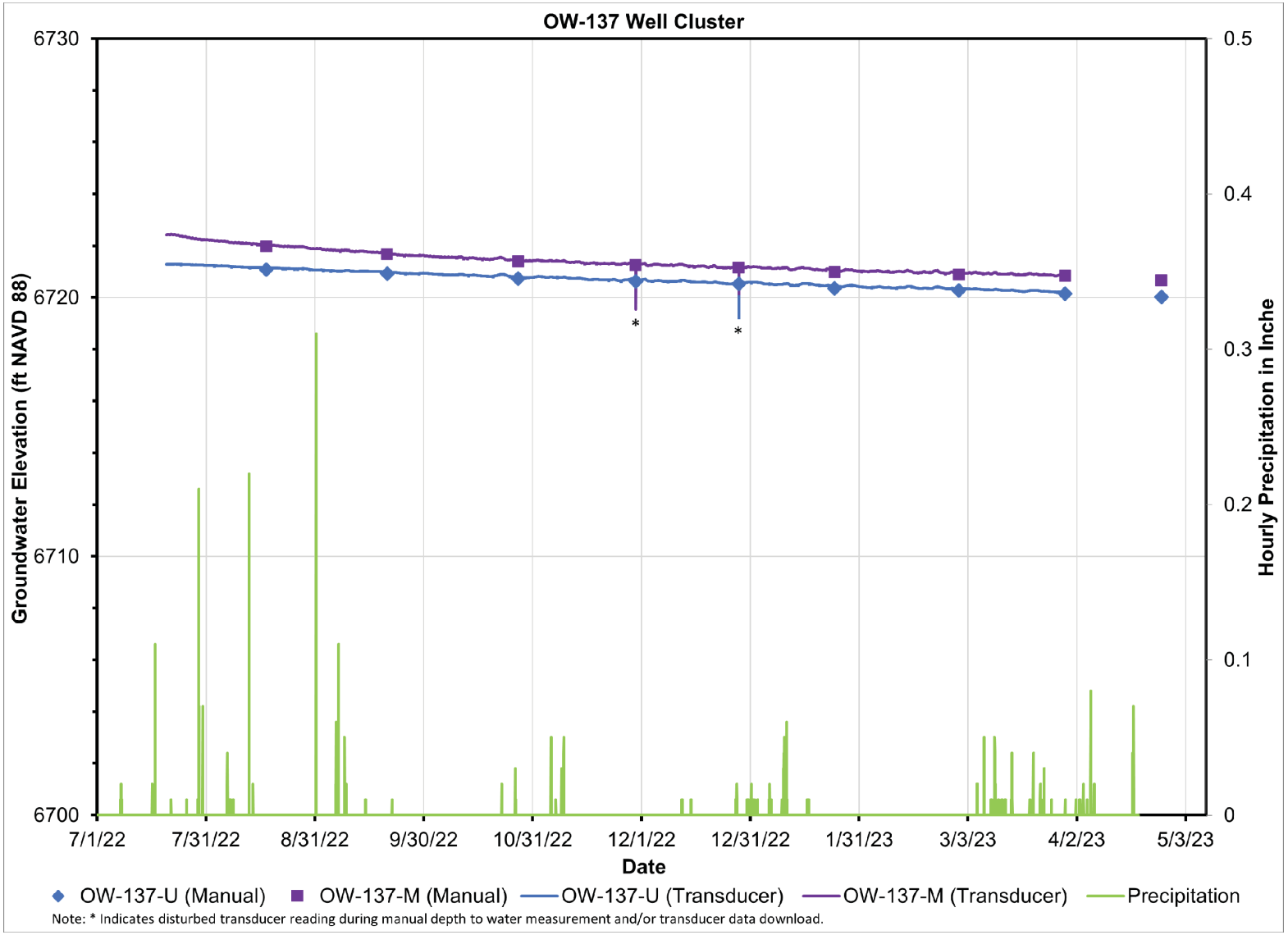



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**Figure 2.5-82 Hydrographs for Observation Wells OW-140-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download)**

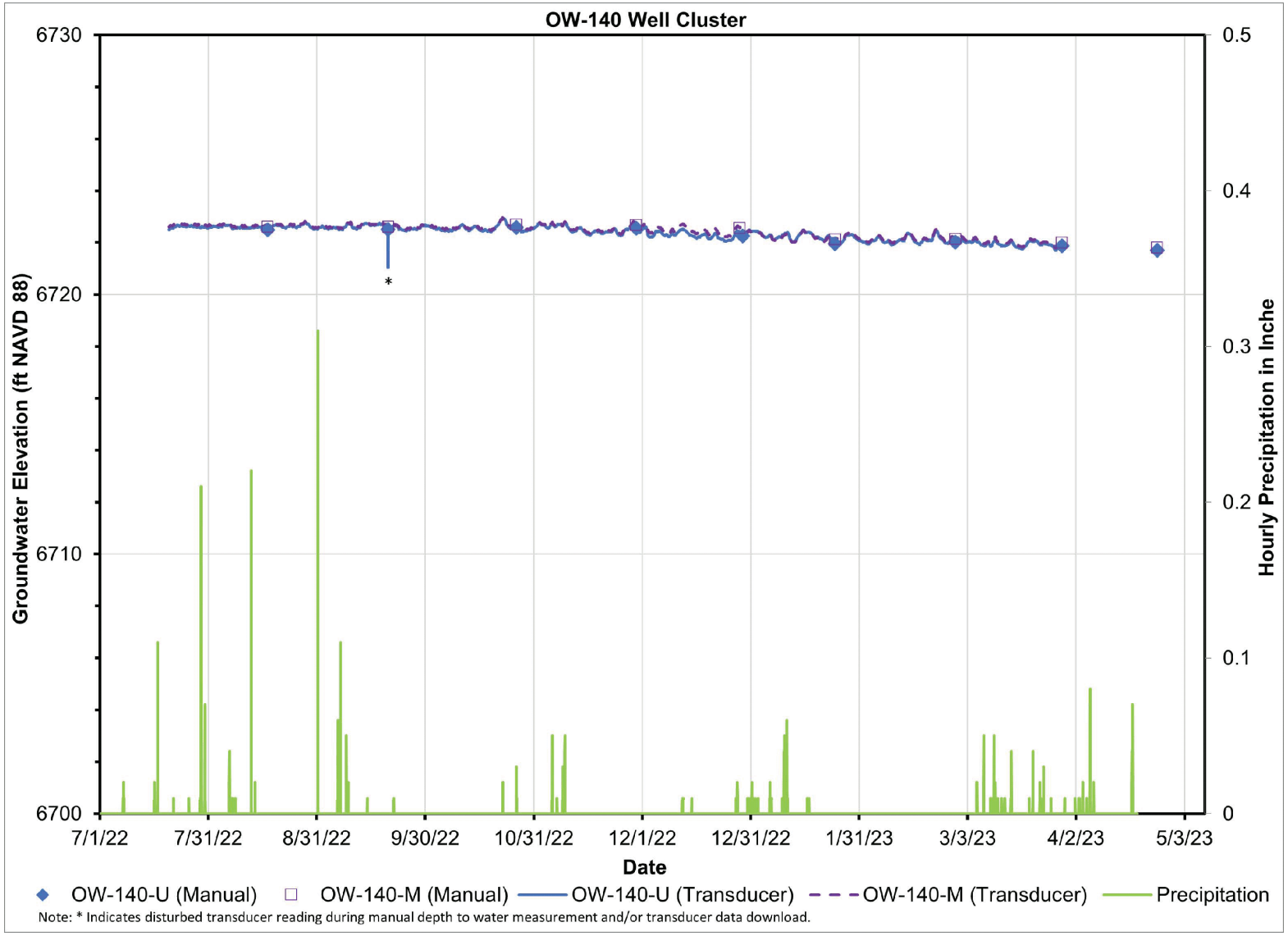



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**Figure 2.5-83 Hydrographs for Observation Wells OW-149-U and M**

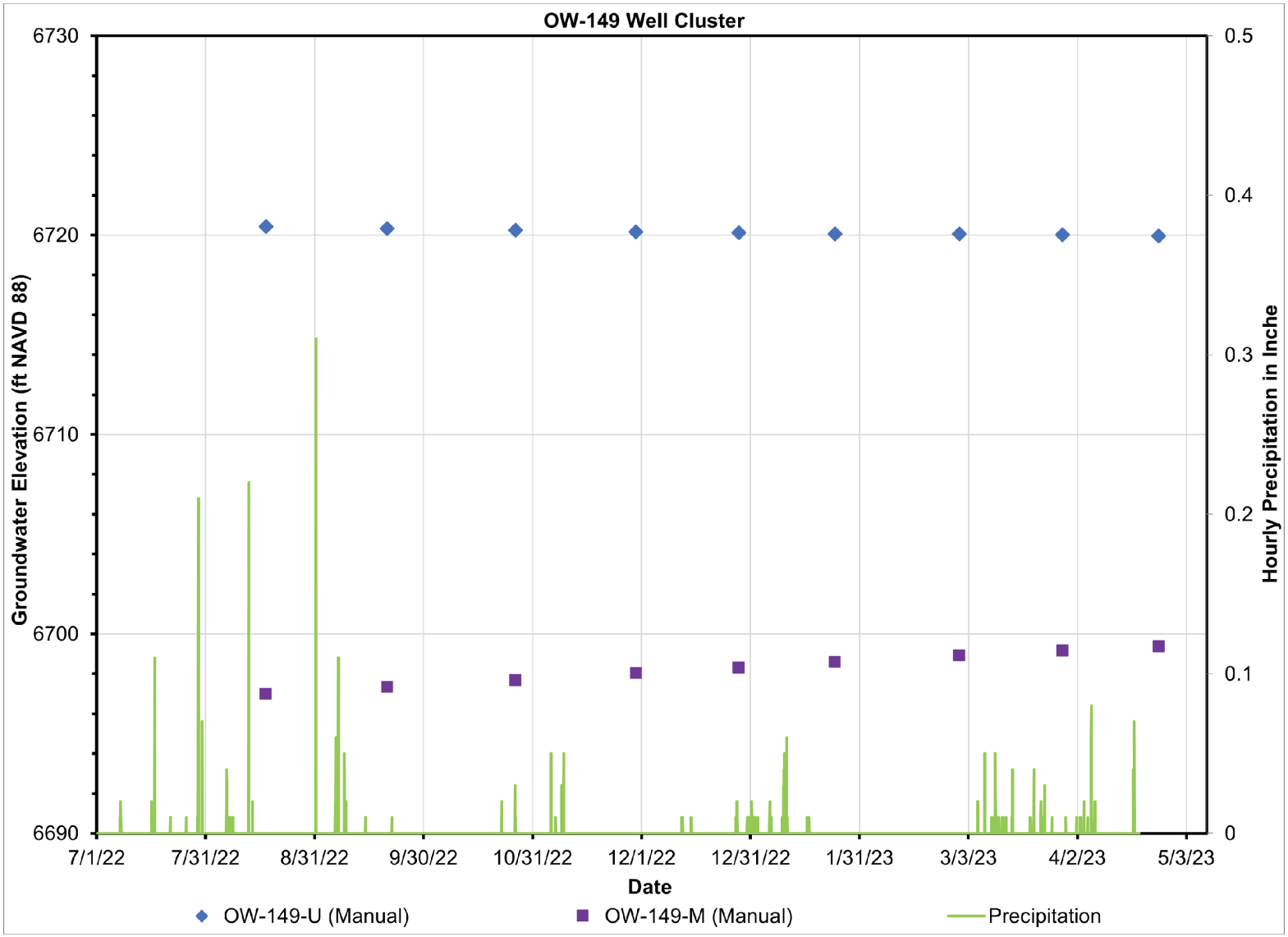






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**Figure 2.5-84 Hydrographs for Observation Wells OW-153-U, M, and L**

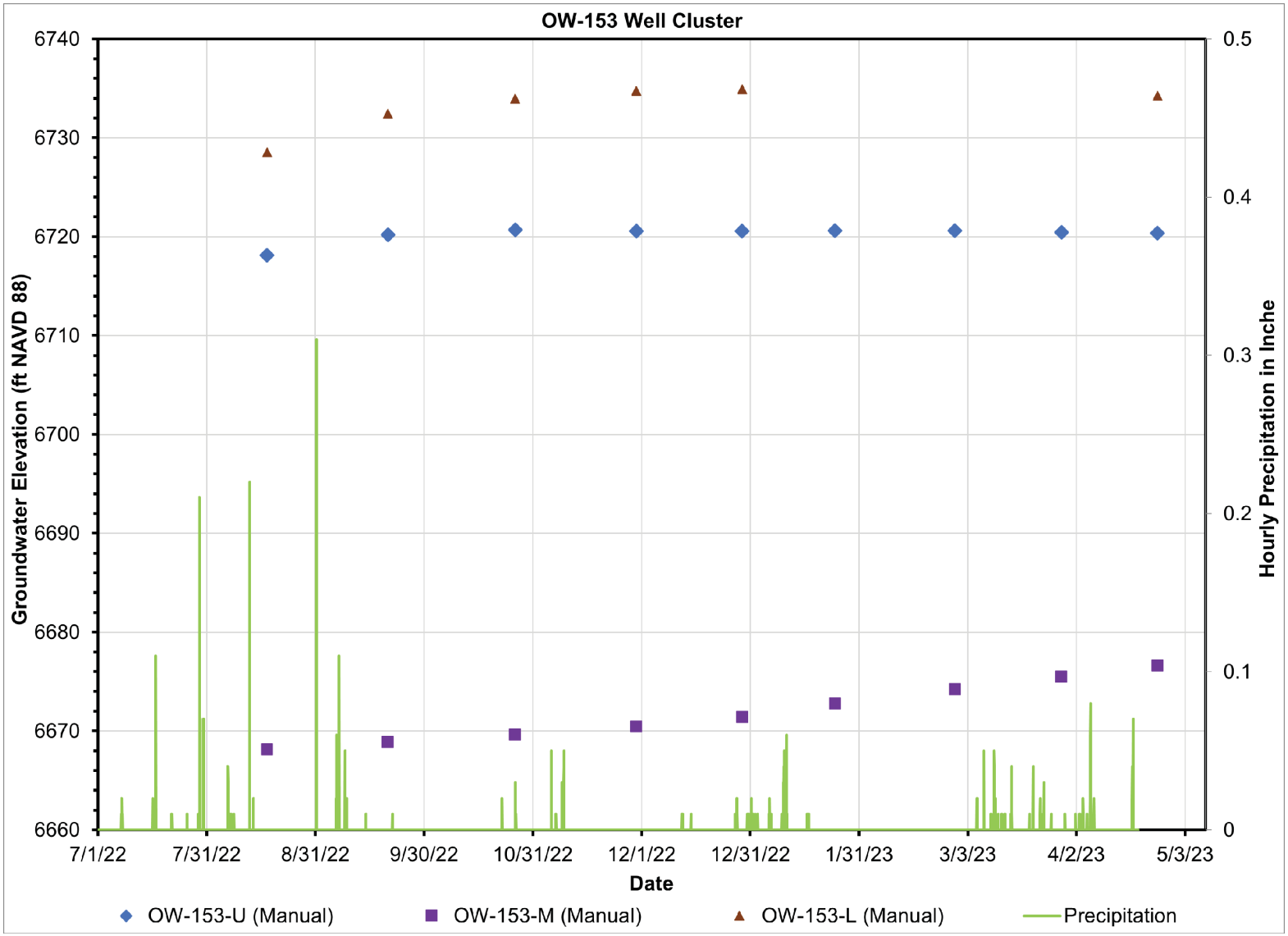




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**Figure 2.5-85 Hydrographs for Observation Wells OW-164-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download)**

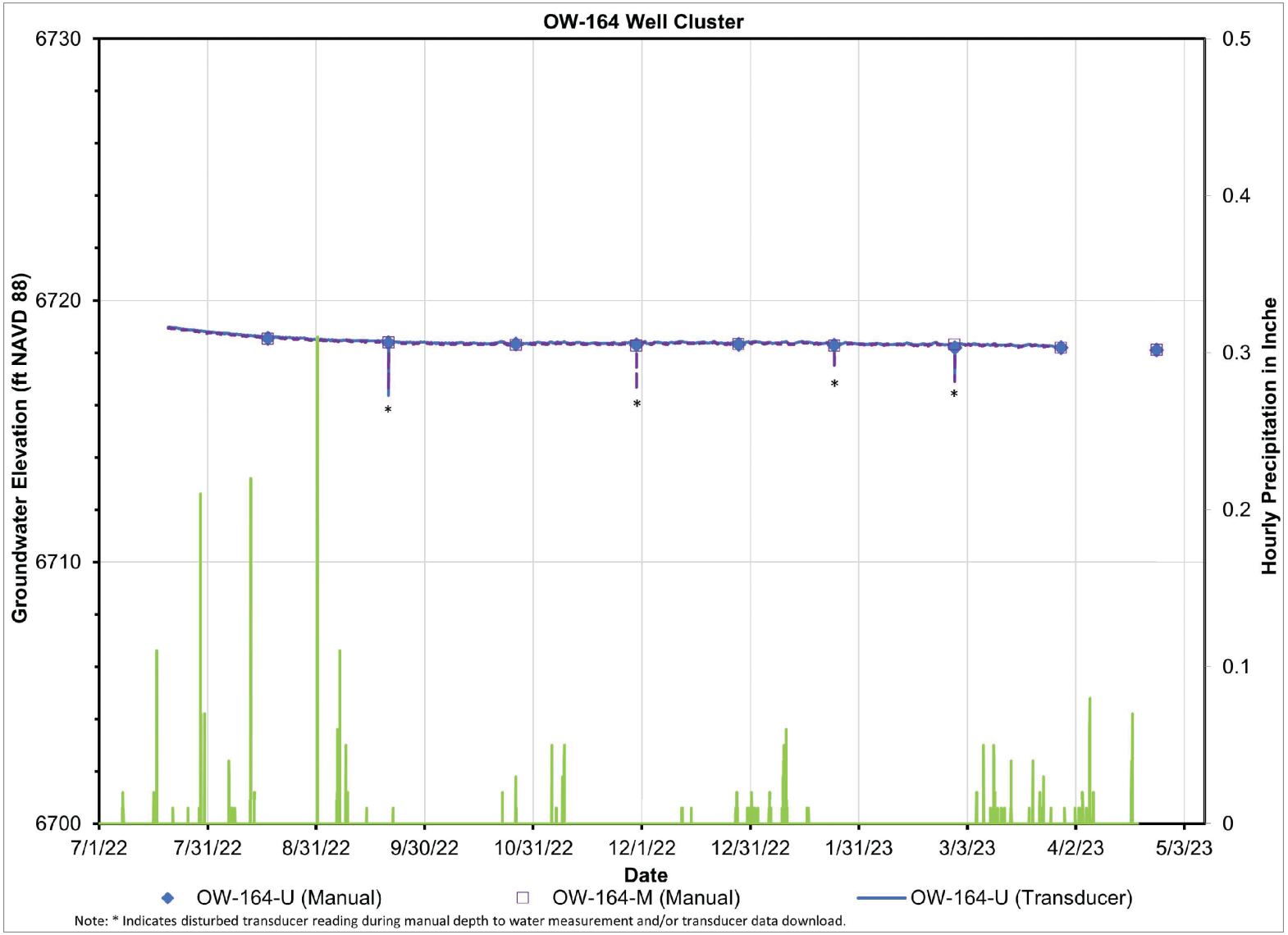




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**Figure 2.5-86 Hydrographs for Observation Wells OW-167-U, M, and L**

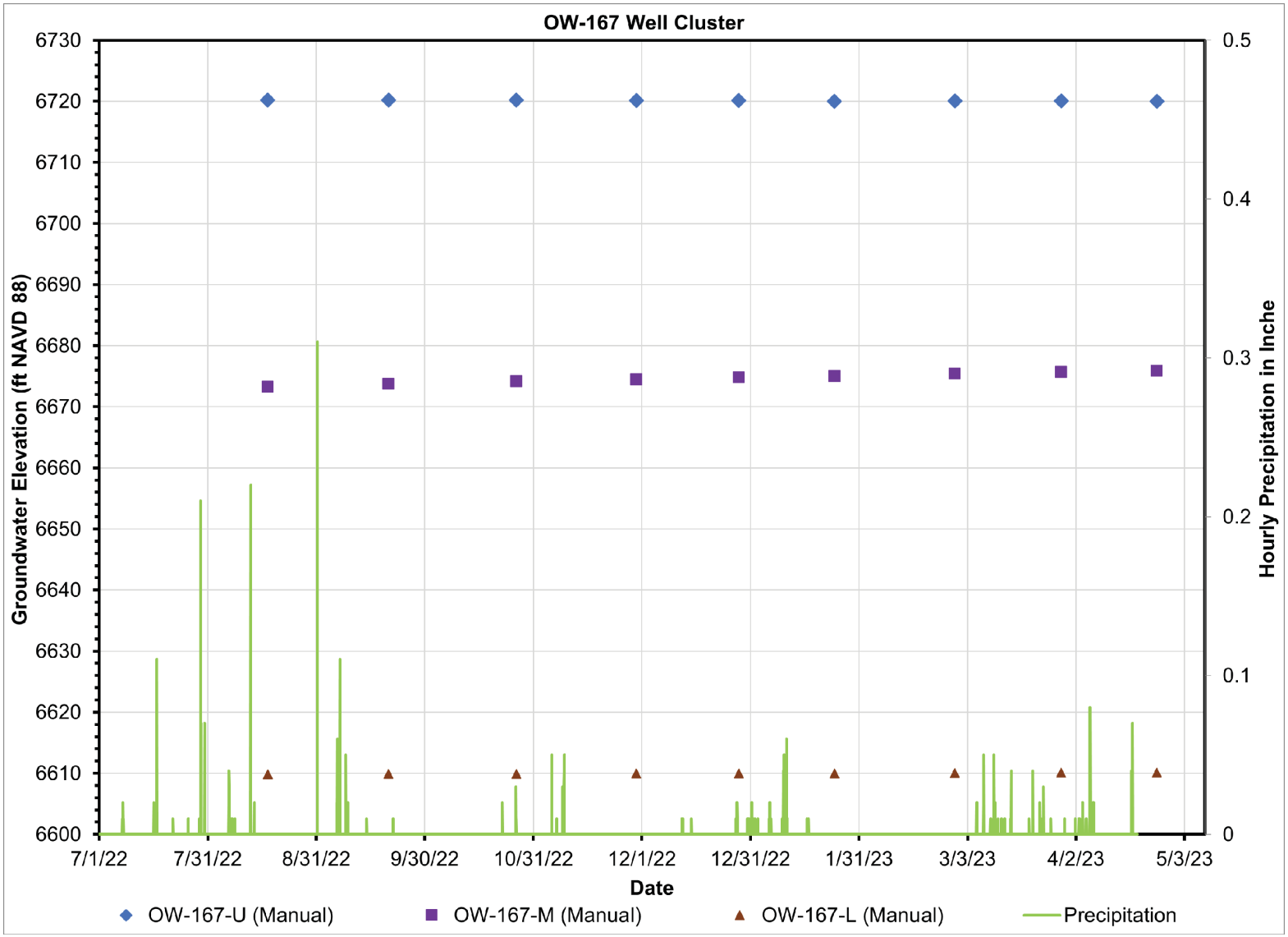




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**Figure 2.5-87 Hydrographs for Observation Wells OW-168-U and M**

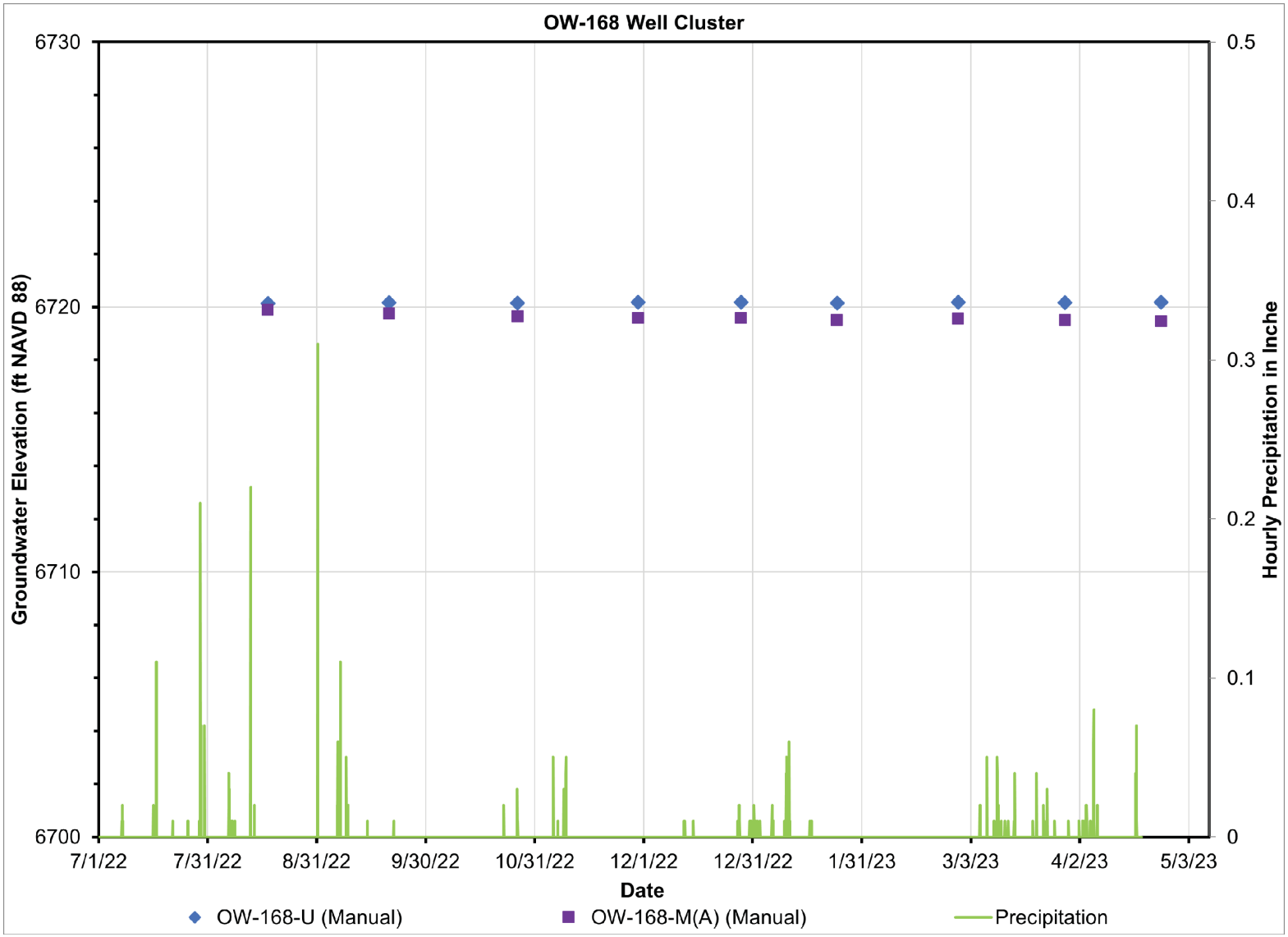




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**Figure 2.5-88 Hydrographs for Observation Wells OW-179-U and M**

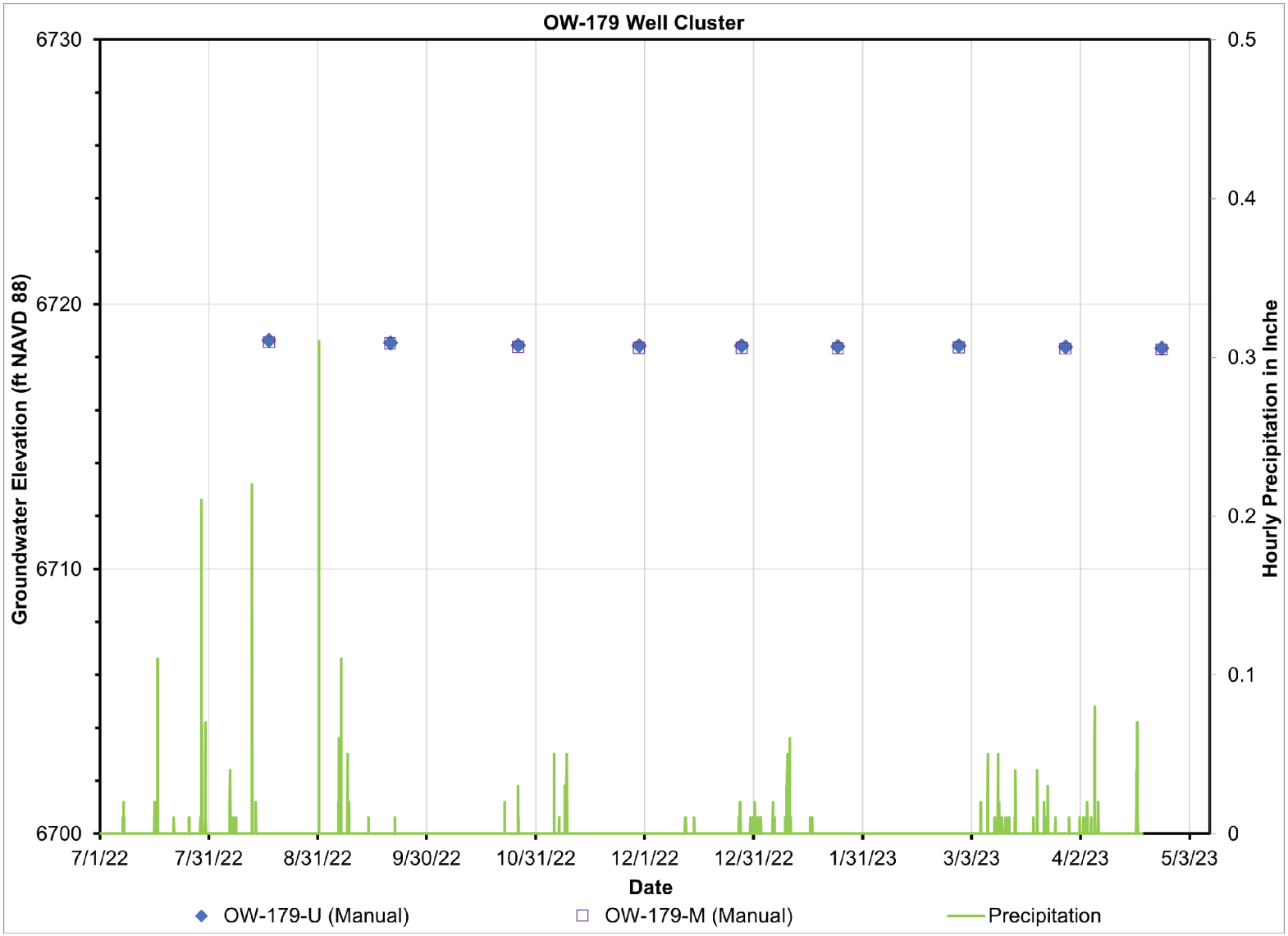






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**Figure 2.5-89 Hydrographs for Observation Wells OW-185-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download)**

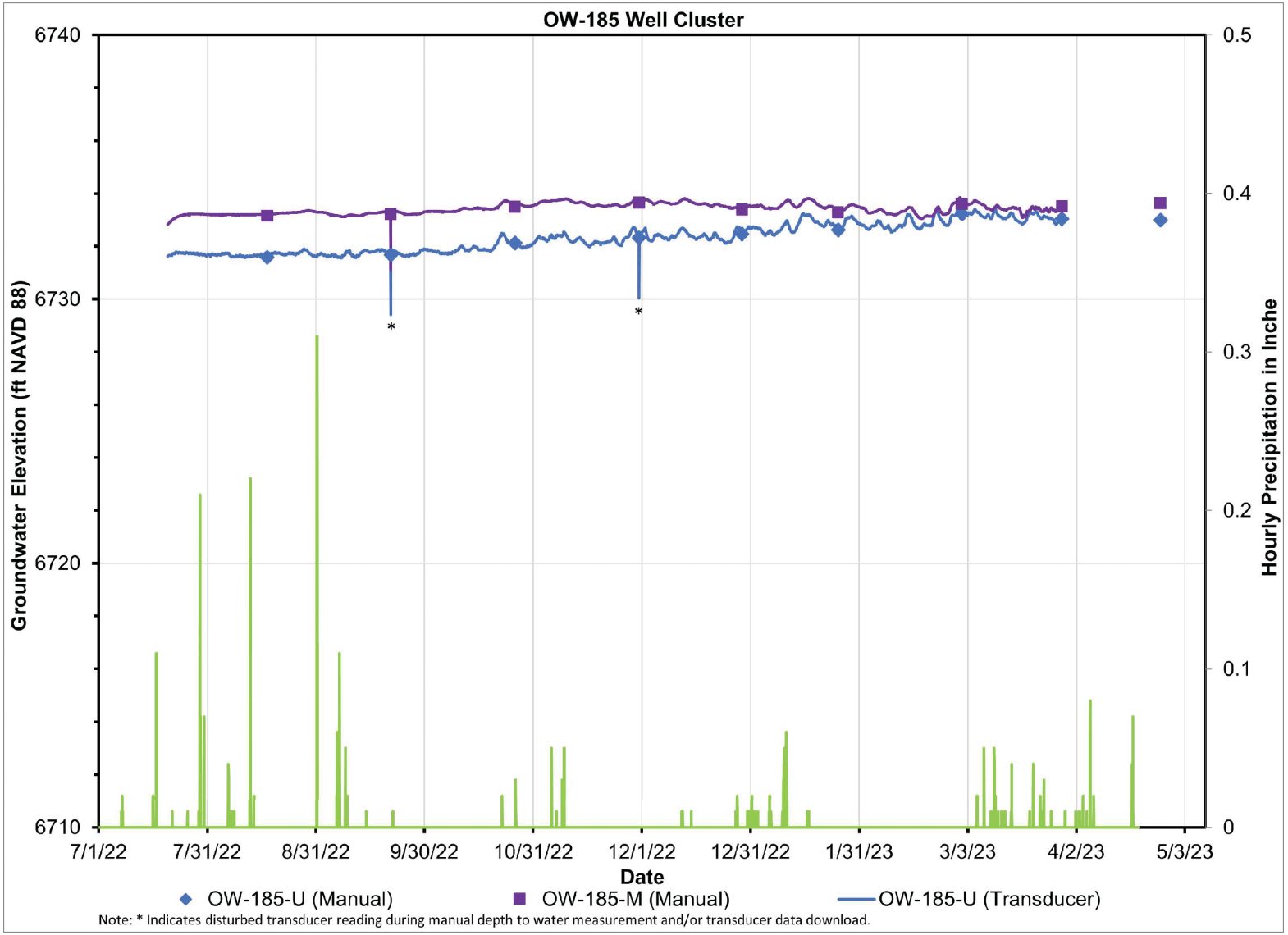




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**Figure 2.5-90 Hydrographs for observation wells OW-186-U and M (* indicates disturbed transducer reading during manual depth to water measurement and/or transducer data download)**

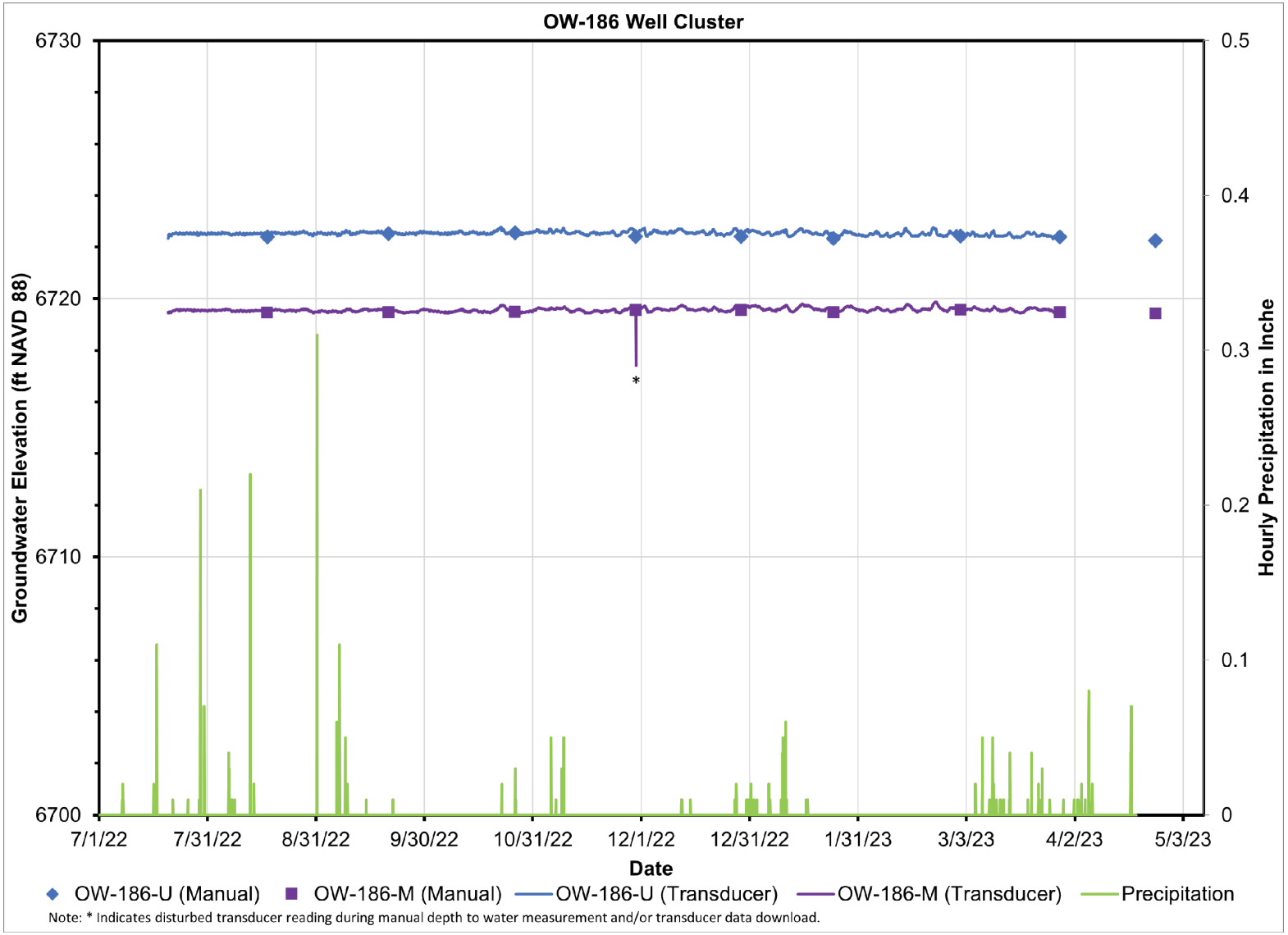



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**Figure 2.5-91 Hydrographs for Observation Wells OW-187-U and M**

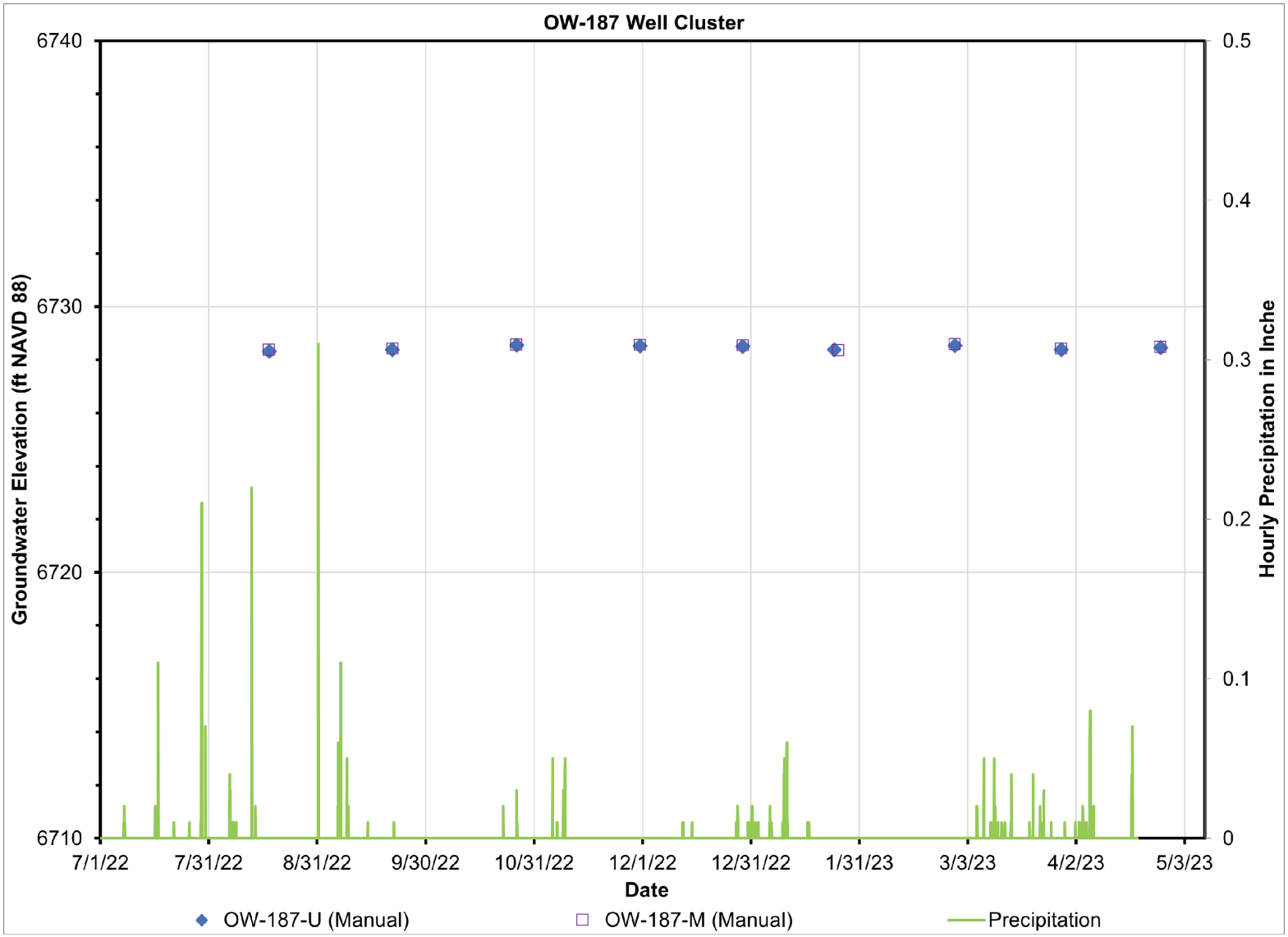






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**Figure 2.5-92 Hydrographs for Observation Wells OW-188-U and M**

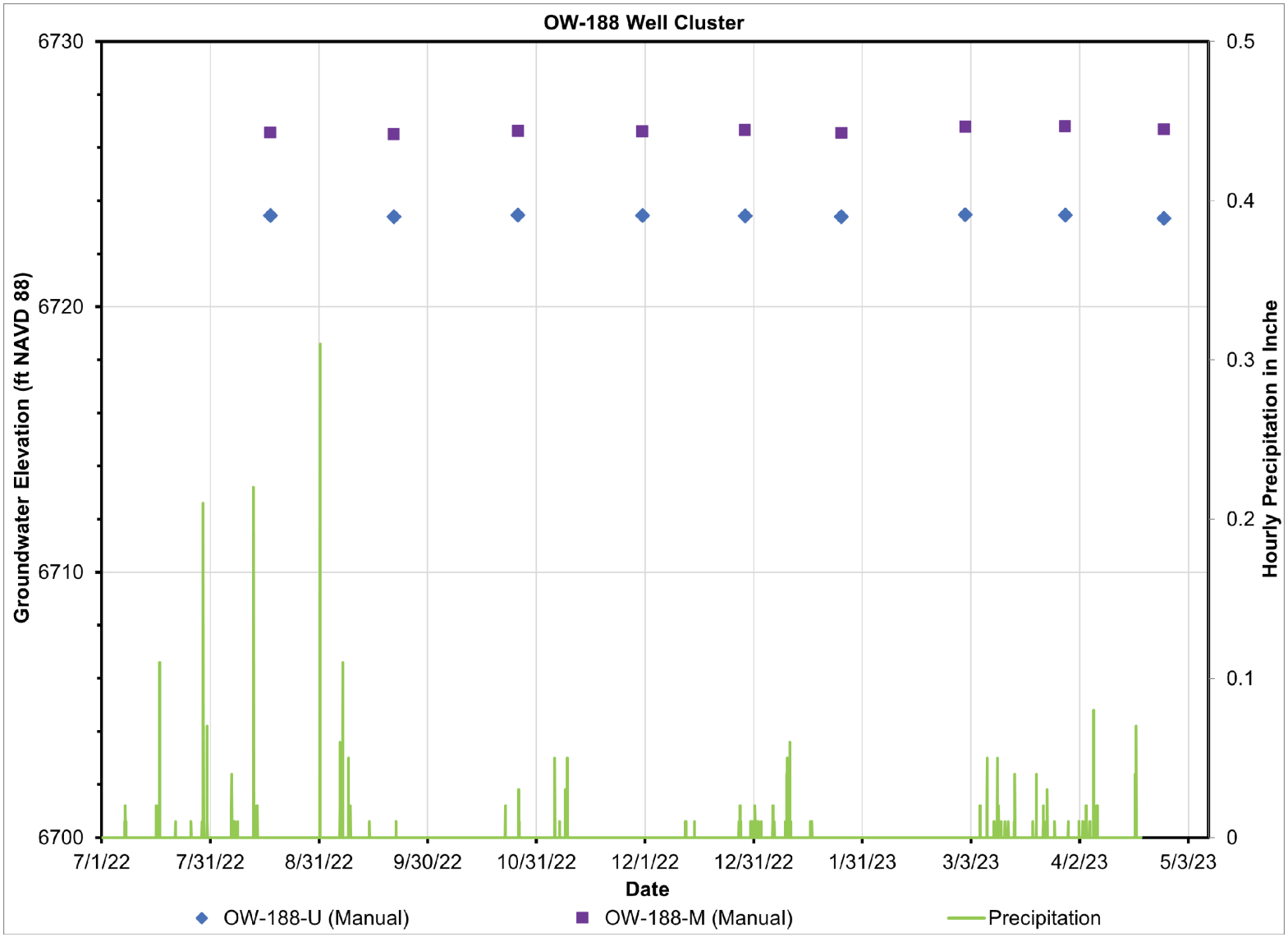




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**Figure 2.5-93 Hydrographs for Observation Wells OW-189-U and M**

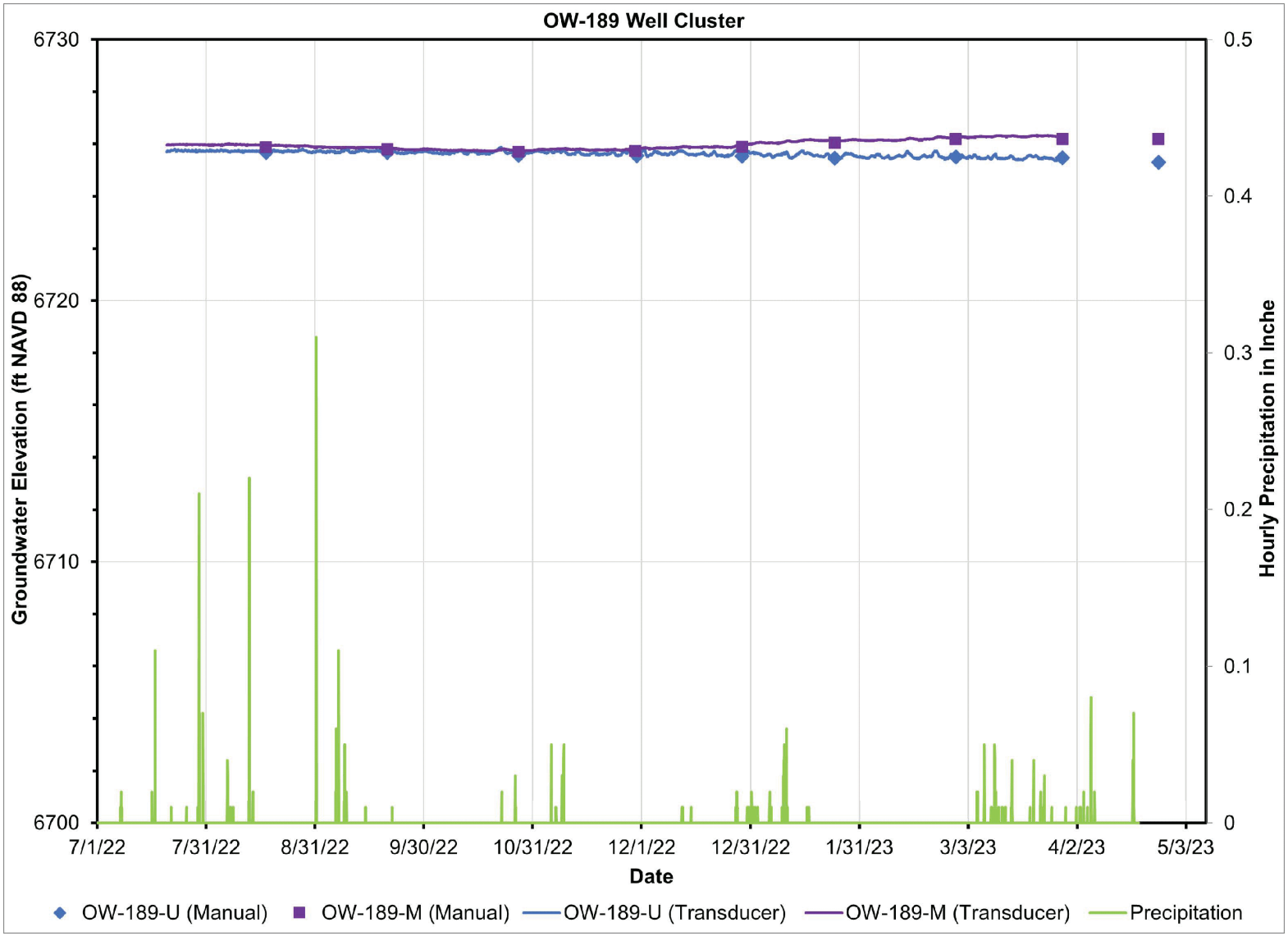




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**Figure 2.5-94 Hydrographs for Observation Wells OW-190-U and M**

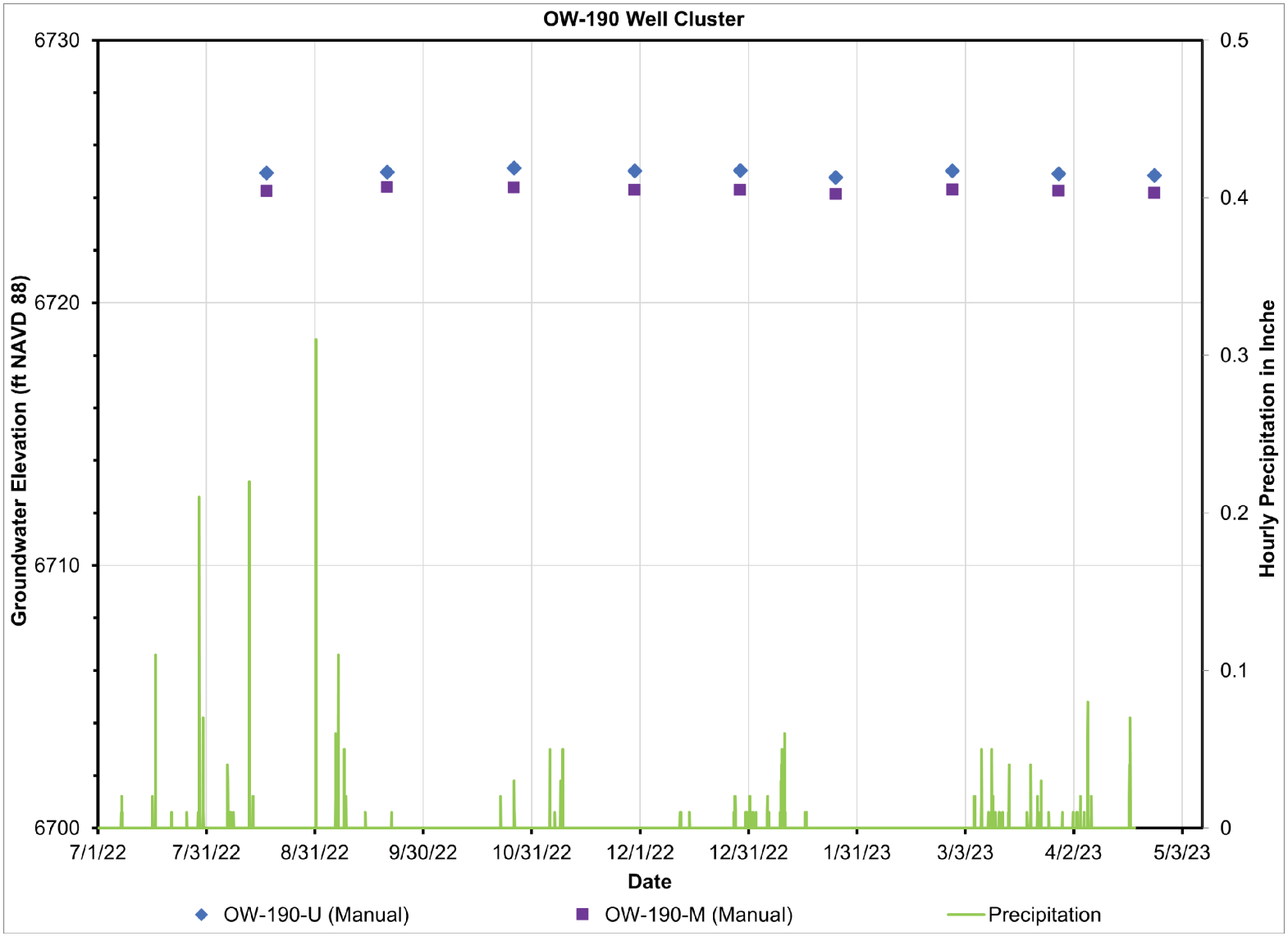






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**Figure 2.5-95 Hydrographs for Observation Wells OW-191-U and M**

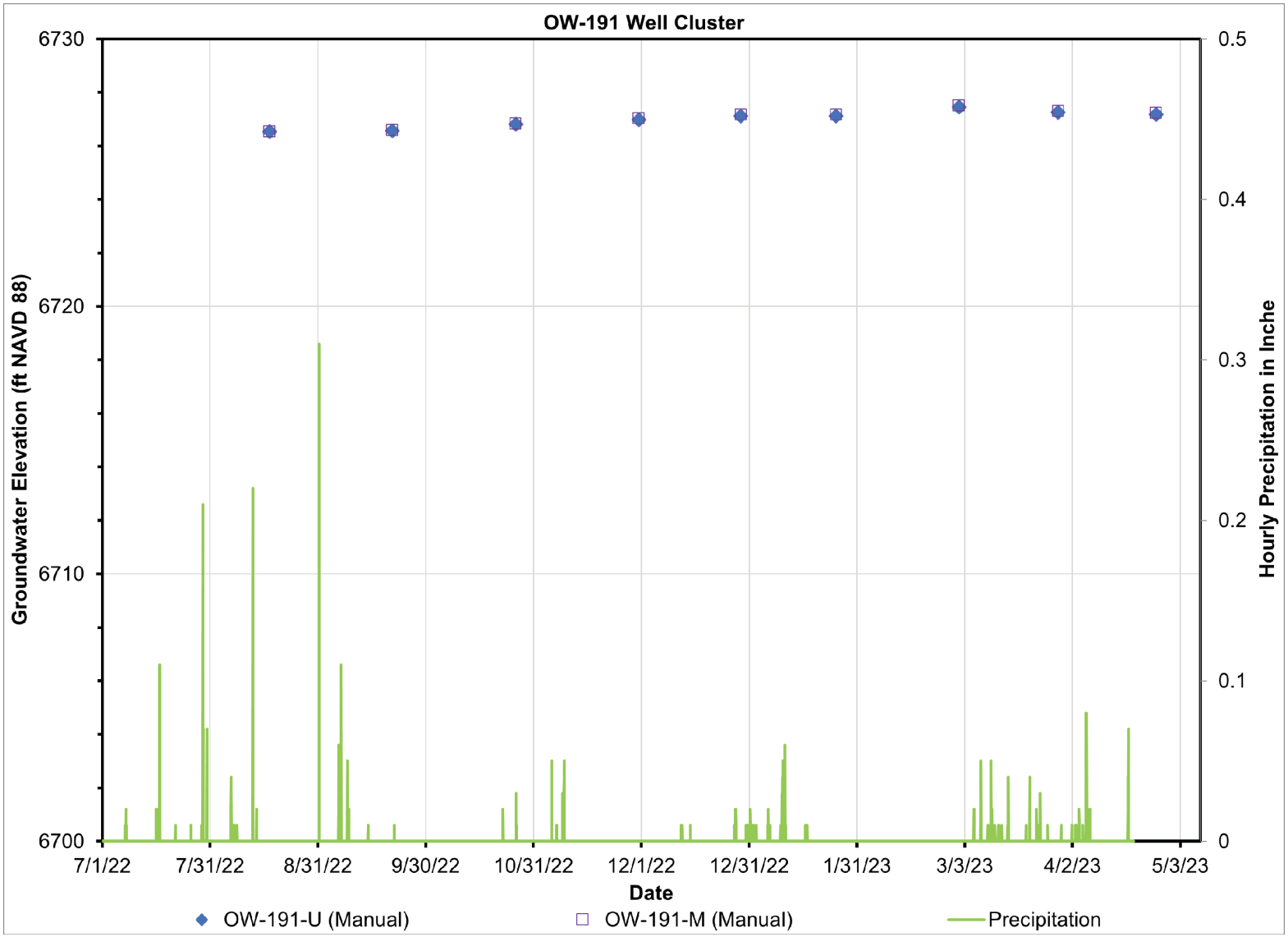






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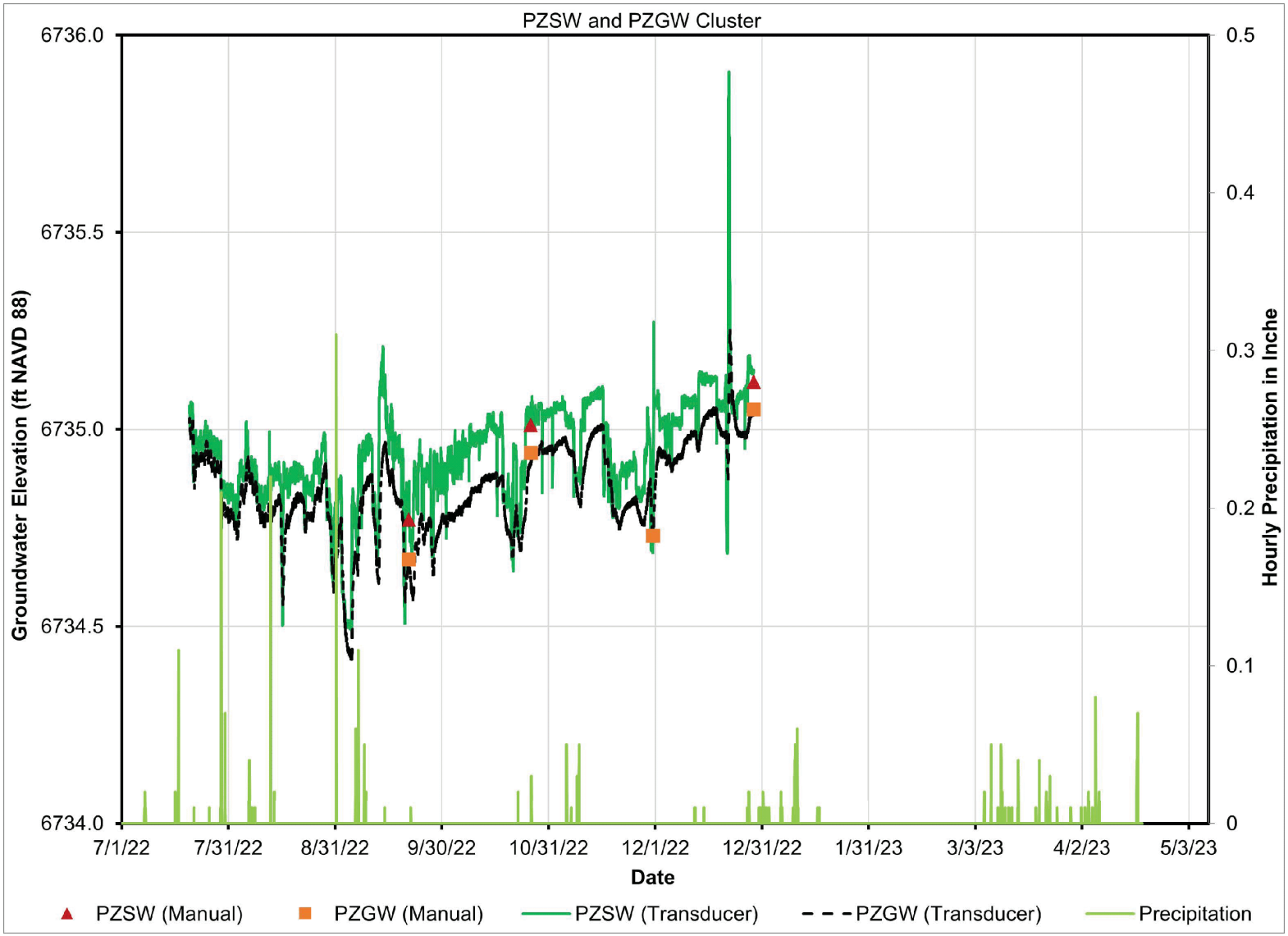
**Figure 2.5-96 Hydrographs for PZSW and PZGW**





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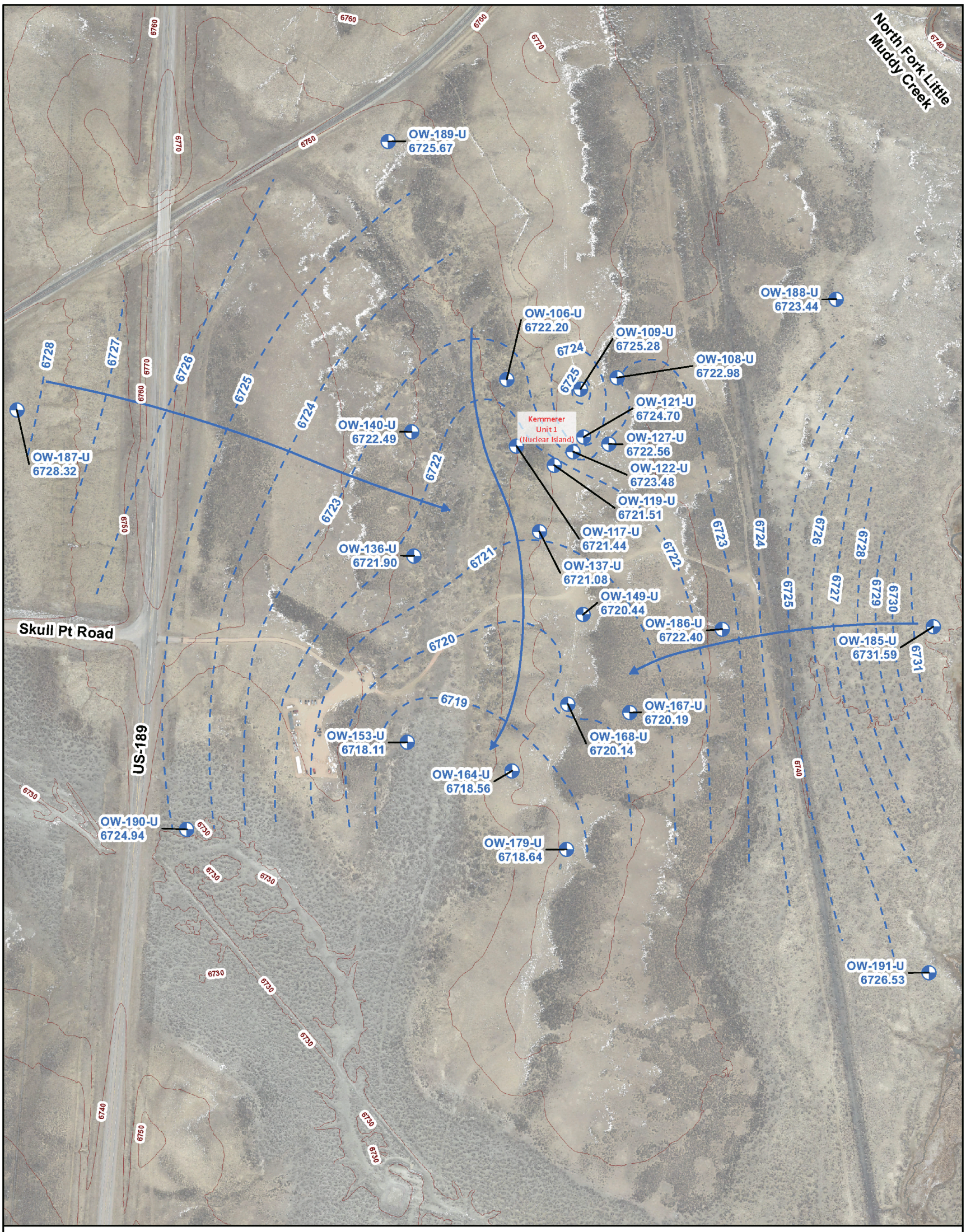
## Figure 2.5-97 Groundwater Level Contour Map and Flow Paths for August 17, 2022 (U Series Wells)




MAP SOURCE INFORMATION

Map Generation Date and Location - 5/26/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_August2022_Shallow.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_August_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_August2022_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_August2022_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**August 17, 2022**
**U (Upper) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 3A**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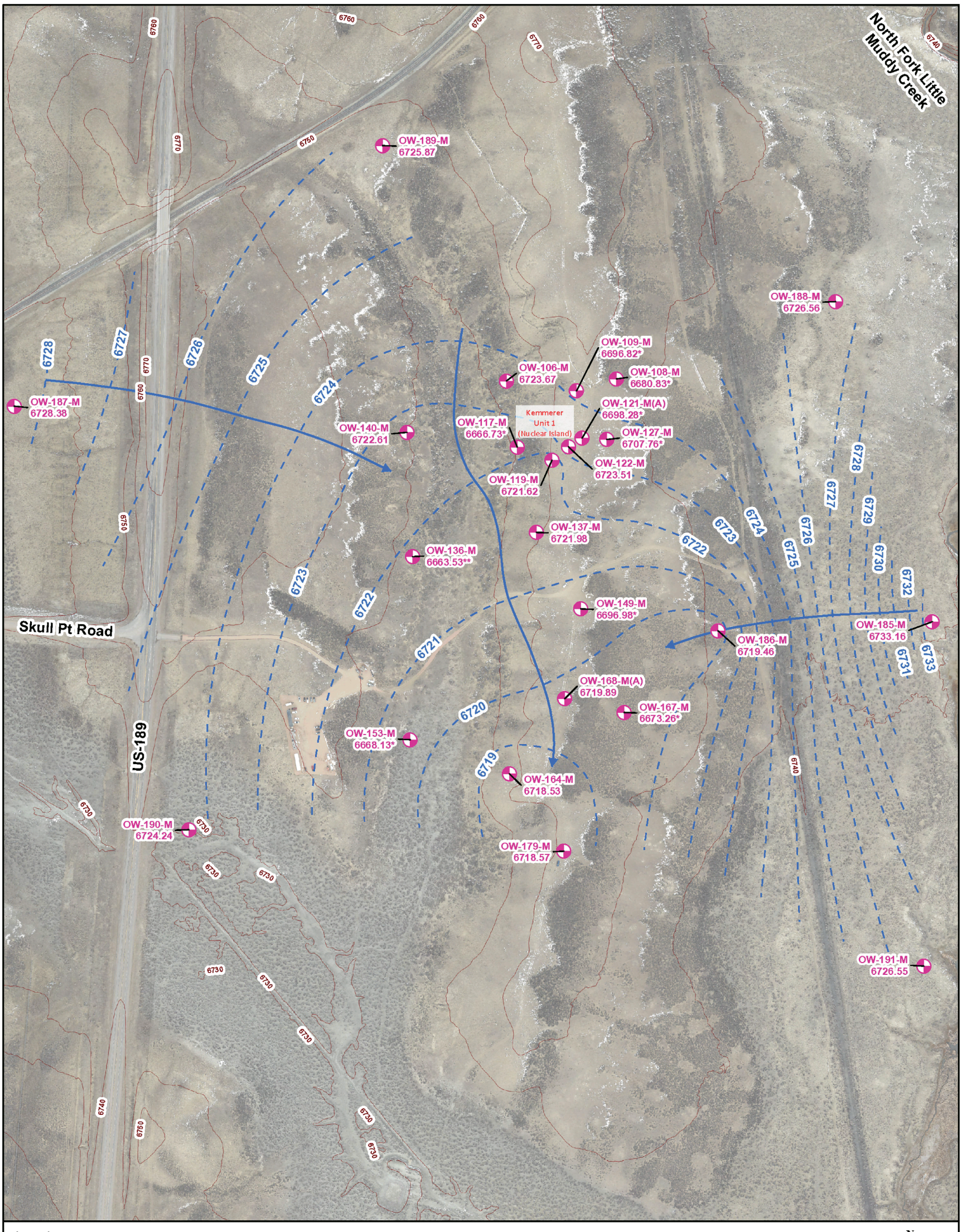
## Figure 2.5-98 Groundwater Level Contour Map and Flow Paths for August 17, 2022 (M Series Wells)




**Legend**

- Intermediate Observation Well
- Groundwater Elevation Contour in ft (August 17, 2022)
- Approximate Groundwater Flow Path
- LiDAR Topographic Contour Line (ft)

Notes:
* - Groundwater elevations from observation wells OW-108-M, OW-109-M, OW-117-M, OW-121-M(A), OW-127-M, OW-149-M, OW-153-M, and OW-167-M were not used to make the groundwater contours because these wells are still recovering (since end of well development) and were deemed not representative of the static water level at the time of measurement.
** - Groundwater elevation from observation well OW-136-M was not used to make the groundwater contours because the groundwater elevation at OW-136-M was significantly lower than surrounding wells.

N W E S

0 150 300 600 Feet

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/26/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_August2022_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_August_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_August2022_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_August2022_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**August 17, 2022**
**M (Middle) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 3B**


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**Figure 2.5-99 Groundwater Level Contour Map and Flow Paths for September 20-21, 2022 (U Series Wells)**

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/22/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_September2022_Shallow.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_September_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_September2022_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_September2022_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**September 20-21, 2022**
**U (Upper) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 4A**


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**Figure 2.5-100 Groundwater Level Contour Map and Flow Paths for September 20-21, 2022 (M Series Wells)**

**Legend**

- Intermediate Observation Well
- Groundwater Elevation Contour in ft (September 20-21, 2022)
- Approximate Groundwater Flow Path
- LiDAR Topographic Contour Line (ft)

Notes:
* - Groundwater elevations from observation wells OW-108-M, OW-109-M, OW-117-M, OW-121-M(A), OW-127-M, OW-149-M, OW-153-M, and OW-167-M were not used to make the groundwater contours because these wells are still recovering (since end of well development) and were deemed not representative of the static water level at the time of measurement.
** - Groundwater elevation from observation well OW-136-M was not used to make the groundwater contours because the groundwater elevation at OW-136-M was significantly lower than surrounding wells.

N, W, E, S

0 150 300 600 Feet

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/22/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_September2022_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_September_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_September2022_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_September2022_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**September 20-21, 2022**
**M (Middle) Series Wells**
Natrium Demonstration Project
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Project/Job No.26396

**Figure 4B**


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**Figure 2.5-101 Groundwater Level Contour Map and Flow Paths for October 26-27, 2022 (U Series Wells)**

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/18/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_October2022_Shallow.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_October_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_October2022_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_October2022_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**October 26-27, 2022**
**U (Upper) Series Wells**
Natrium Demonstration Project
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Project/Job No.26396

**Figure 5A**


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**Figure 2.5-102 Groundwater Level Contour Map and Flow Paths for October 26-27, 2022 (M Series Wells)**

**Legend**

- Intermediate Observation Well
- Groundwater Elevation Contour in ft (October 26-27, 2022)
- Approximate Groundwater Flow Path
- LiDAR Topographic Contour Line (ft)

Notes:
* - Groundwater elevations from observation wells OW-108-M, OW-109-M, OW-117-M, OW-121-M(A), OW-127-M, OW-149-M, OW-153-M, and OW-167-M were not used to make the groundwater contours because these wells are still recovering (since end of well development) and were deemed not representative of the static water level at the time of measurement.
** - Groundwater elevation from observation well OW-136-M was not used to make the groundwater contours because the groundwater elevation at OW-136-M was significantly lower than surrounding wells.

0 150 300 600 Feet

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/18/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_October2022_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_October_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_October2022_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_October2022_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**October 26-27, 2022**
**M (Middle) Series Wells**
Natrium Demonstration Project
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Project/Job No.26396

**Figure 5B**


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**Figure 2.5-103 Groundwater Level Contour Map and Flow Paths for November 29-30, 2022 (U Series Wells)**

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Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_November_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_November2022_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_November2022_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**November 29-30, 2022**
**U (Upper) Series Wells**
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Project/Job No.26396

**Figure 6A**


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**Figure 2.5-104 Groundwater Level Contour Map and Flow Paths for November 29-30, 2022 (M Series Wells)**

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Map Generation Date and Location - 5/18/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_November2022_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_November_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_November2022_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_November2022_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**November 29-30, 2022**
**M (Middle) Series Wells**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure**

**6B**


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**Figure 2.5-105 Groundwater Level Contour Map and Flow Paths for December 28-29, 2022 (U Series Wells)**

**Legend**

- Shallow Observation Well
- Groundwater Elevation Contour in ft (December 28-29, 2022)
- Approximate Groundwater Flow Path
- LiDAR Topographic Contour Line (ft)

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/17/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_December2022_Shallow.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_December_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_December2022_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_December2022_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**December 28-29, 2022**
**U (Upper) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 7A**


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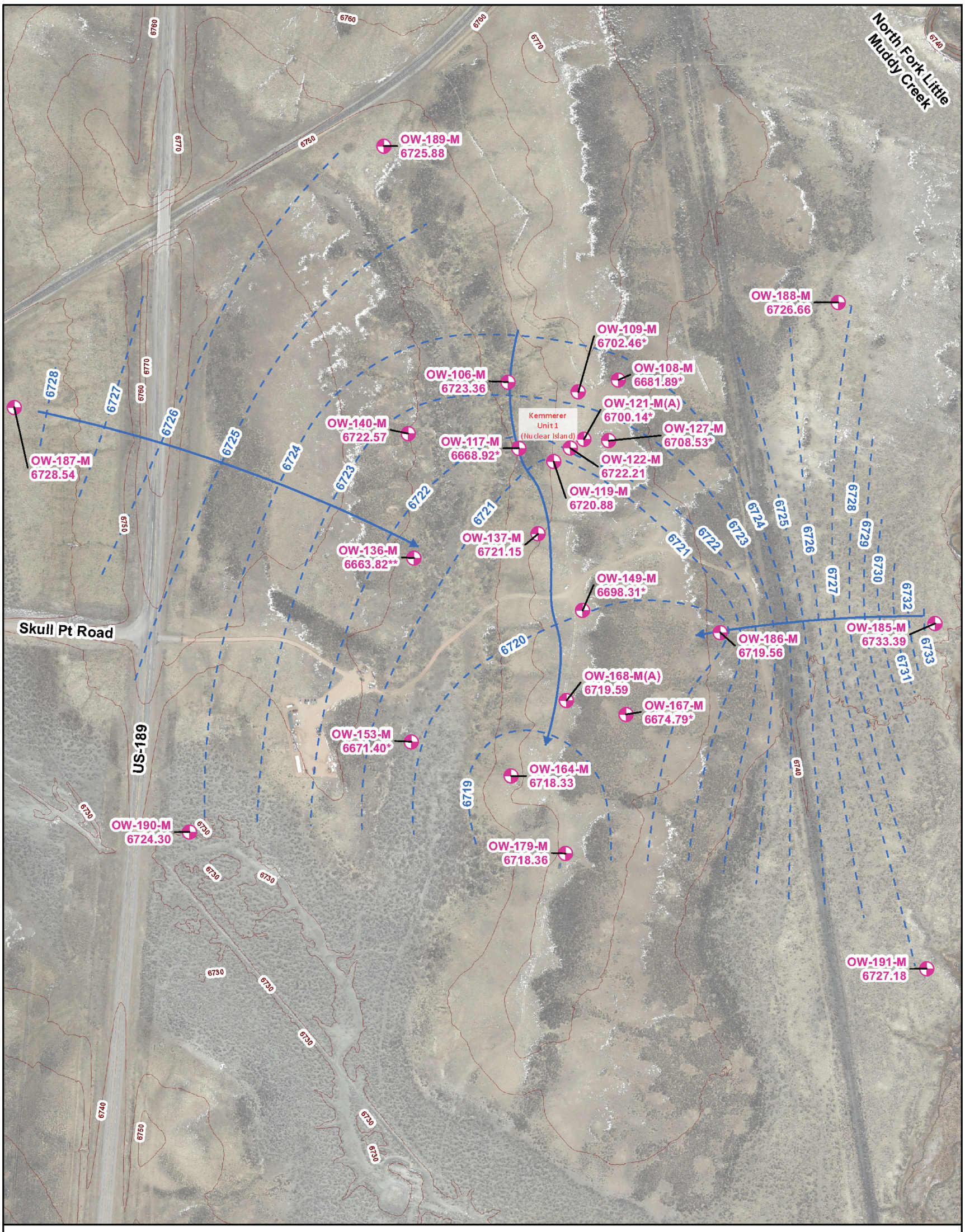
**Figure 2.5-106 Groundwater Level Contour Map and Flow Paths for December 28-29, 2022 (M Series Wells)**




**Legend**

- Intermediate Observation Well
- Groundwater Elevation Contour in ft (December 28-29, 2022)
- Approximate Groundwater Flow Path
- LiDAR Topographic Contour Line (ft)

Notes:
* - Groundwater elevations from observation wells OW-108-M, OW-109-M, OW-117-M, OW-121-M(A), OW-127-M, OW-149-M, OW-153-M, and OW-167-M were not used to make the groundwater contours because these wells are still recovering (since end of well development) and were deemed not representative of the static water level at the time of measurement.
** - Groundwater elevation from observation well OW-136-M was not used to make the groundwater contours because the groundwater elevation at OW-136-M was significantly lower than surrounding wells.

N, W, E, S

0 150 300 600 Feet

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/17/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_December2022_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_December_2022.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_December2022_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_December2022_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**December 28-29, 2022**
**M (Middle) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 7B**


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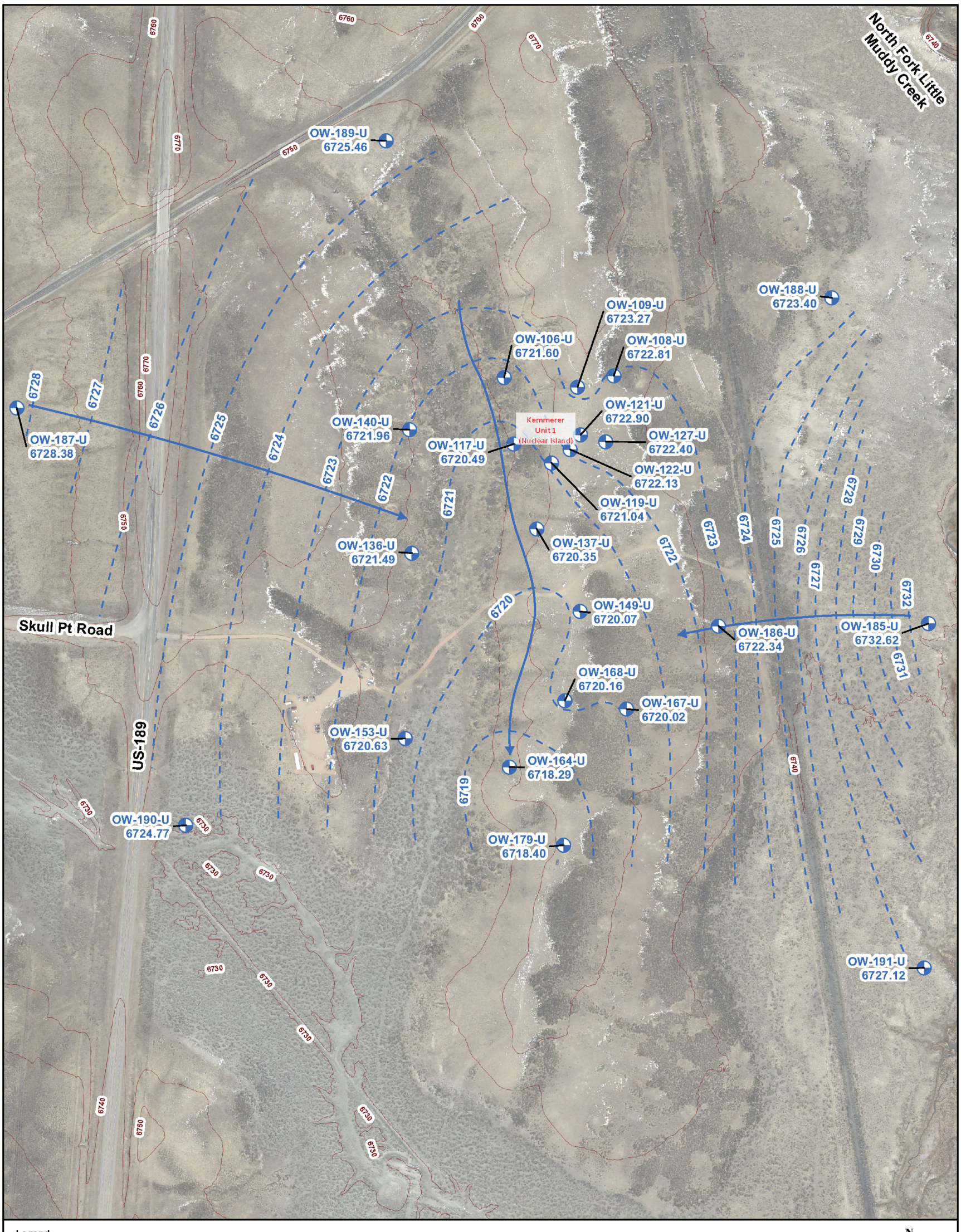
**Figure 2.5-107 Groundwater Level Contour Map and Flow Paths for January 24-25, 2023 (U Series Wells)**




**Legend**

- Shallow Observation Well
- Groundwater Elevation Contour in ft (January 24-25, 2023)
- Approximate Groundwater Flow Path
- LiDAR Topographic Contour Line (ft)

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/16/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_January2023_Shallow.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_January_2023.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_January2023_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_January2023_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**January 24-25, 2023**
**U (Upper) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 8A**


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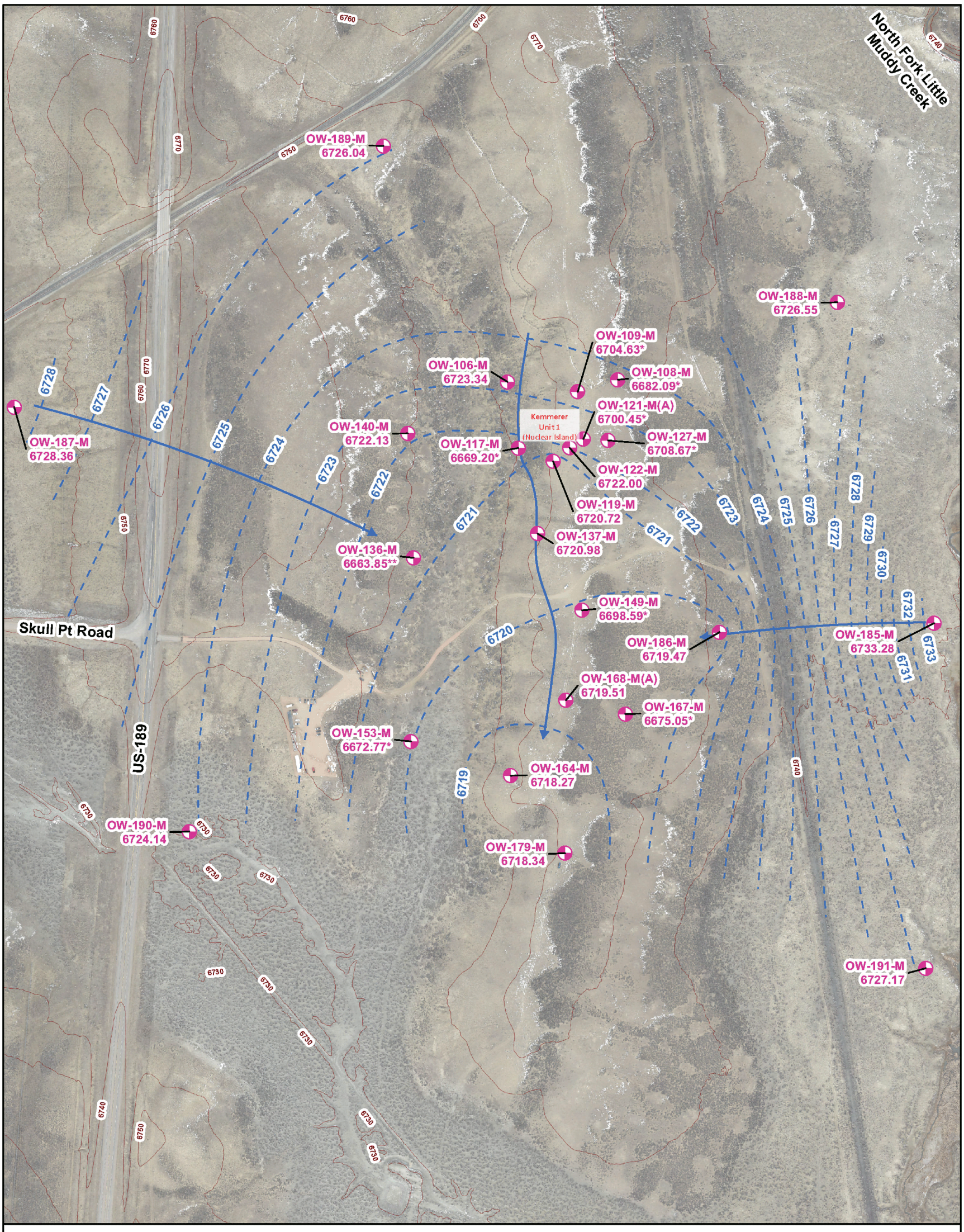
**Figure 2.5-108 Groundwater Level Contour Map and Flow Paths for January 24-25, 2023 (M Series Wells)**




MAP SOURCE INFORMATION

Map Generation Date and Location - 5/11/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_January2023_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_January_2023.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_January2023_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_January2023_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**January 24-25, 2023**
**M (Middle) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 8B**


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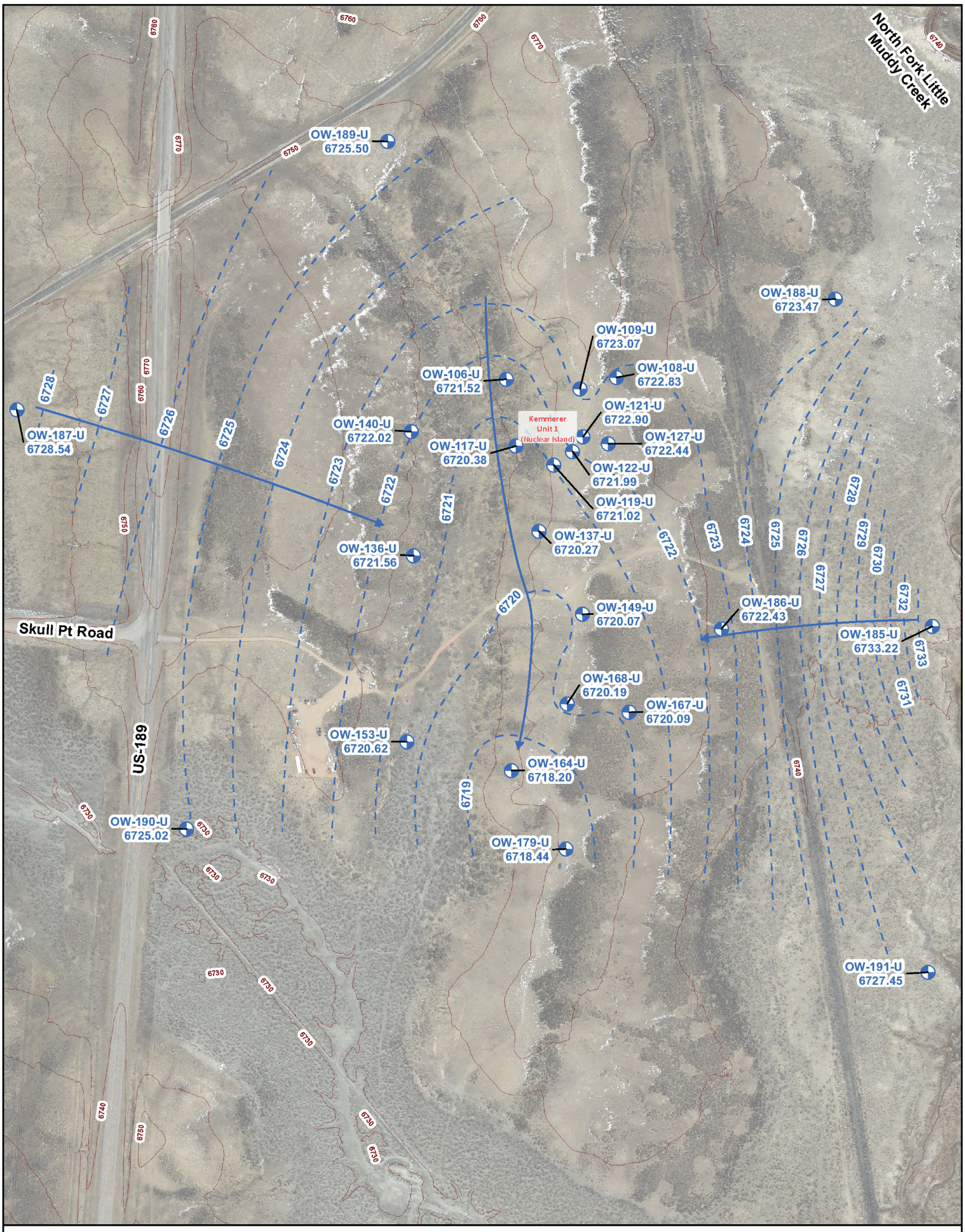
**Figure 2.5-109 Groundwater Level Contour Map and Flow Paths for February 27-28 and March 1, 2023 (U Series Wells)**




MAP SOURCE INFORMATION

Map Generation Date and Location - 5/10/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_February2023_Shallow.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_February_2023.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_February2023_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_February2023_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**February 27-28 and March 1 2023**
**U (Upper) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 9A**


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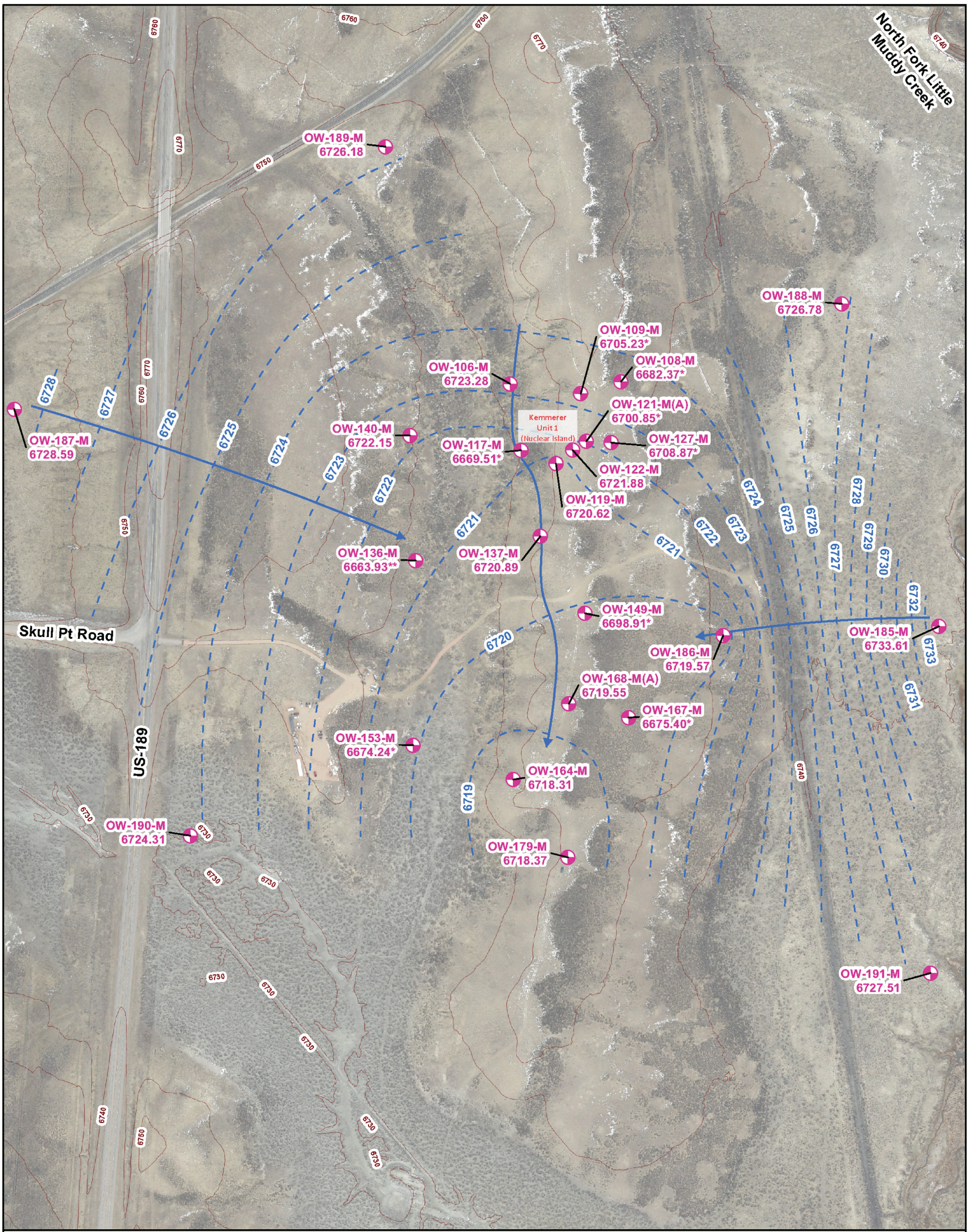
**Figure 2.5-110 Groundwater Level Contour Map and Flow Paths for February 27-28 and March 1, 2023 (M Series Wells)**




MAP SOURCE INFORMATION

Map Generation Date and Location - 5/16/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_February2023_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_February_2023.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_February2023_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_February2023_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**February 27-28 and March 1 2023**
**M (Middle) Series Wells**
Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 9B**


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**Figure 2.5-111 Groundwater Level Contour Map and Flow Paths for March 29-30, 2023 (U Series Wells)**

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/9/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_March2023_Shallow.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_March_2023.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_March2023_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_March2023_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988

**Groundwater Elevation Contour Map**
**29-30 March 2023**
**U (Upper) Series Wells**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 10A**


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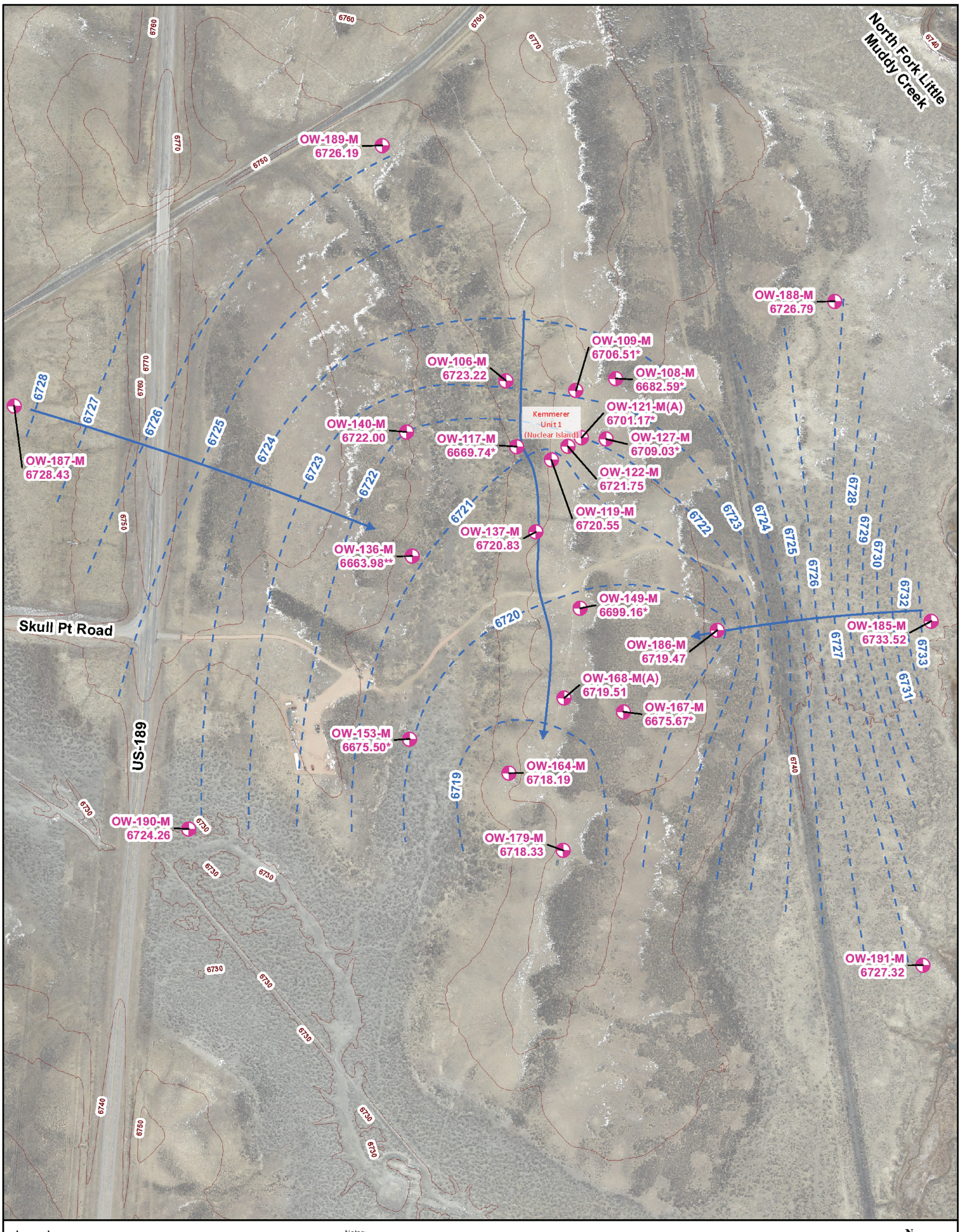
**Figure 2.5-112 Groundwater Level Contour Map and Flow Paths for March 29-30, 2023 (M Series Wells)**




**Legend**

- Intermediate Observation Well
- Groundwater Elevation Contour in ft (29-30 March 2023)
- GWFlow_Direction_March2023_Intermediate
- LiDAR Topographic Contour Line (ft)

Notes:
* - Groundwater elevations from observation wells OW-108-M, OW-109-M, OW-117-M, OW-121-M(A), OW-127-M, OW-149-M, OW-153-M, and OW-167-M were not used to make the groundwater contours because these wells are still recovering (since end of well development) and were deemed not representative of the static water level at the time of measurement.
** - Groundwater elevation from observation well OW-136-M was not used to make the groundwater contours because the groundwater elevation at OW-136-M was significantly lower than surrounding wells.

N W E S

0 150 300 600 Feet

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/9/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_March2023_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_March_2023.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_March2023_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_March2023_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988

**Groundwater Elevation Contour Map**
**29-30 March 2023**
**M (Middle) Series Wells**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure**

**10B**


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**Figure 2.5-113 Groundwater Level Contour Map and Flow Paths for April 25-26, 2023 (U Series Wells)**

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/10/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_April2023_Shallow.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_April_2023.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_April2023_Shallow.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_April2023_Shallow.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**25-26 April 2023**
**U (Upper) Series Wells**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 11A**


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**Figure 2.5-114 Groundwater Level Contour Map and Flow Paths for April 25-26, 2023 (M Series Wells)**

**Legend**

- Intermediate Observation Well
- Groundwater Elevation Contour in ft (25-26 April 2023)
- Approximate Groundwater Flow Path
- LiDAR Topographic Contour Line (ft)

Notes:
* - Groundwater elevations from observation wells OW-108-M, OW-109-M, OW-117-M, OW-121-M(A), OW-127-M, OW-149-M, OW-153-M, and OW-167-M were not used to make the groundwater contours because these wells are still recovering (since end of well development) and were deemed not representative of the static water level at the time of measurement.
** - Groundwater elevation from observation well OW-136-M was not used to make the groundwater contours because the groundwater elevation at OW-136-M was significantly lower than surrounding wells.

N W E S

0 150 300 600 Feet

MAP SOURCE INFORMATION

Map Generation Date and Location - 5/15/2023 and \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\MXD\Groundwater Elevation Contours Map_April2023_Intermediate.mxd.
Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Well Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_April_2023.shp.
Shapefile for Groundwater Elevation Contour: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWE_Contour_April2023_Intermediate.shp.
Shapefile for Approximate Groundwater Flow Path: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\GWFlow_Direction_April2023_Intermediate.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988.

**Groundwater Elevation Contour Map**
**25-26 April 2023**
**M (Middle) Series Wells**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 11B**


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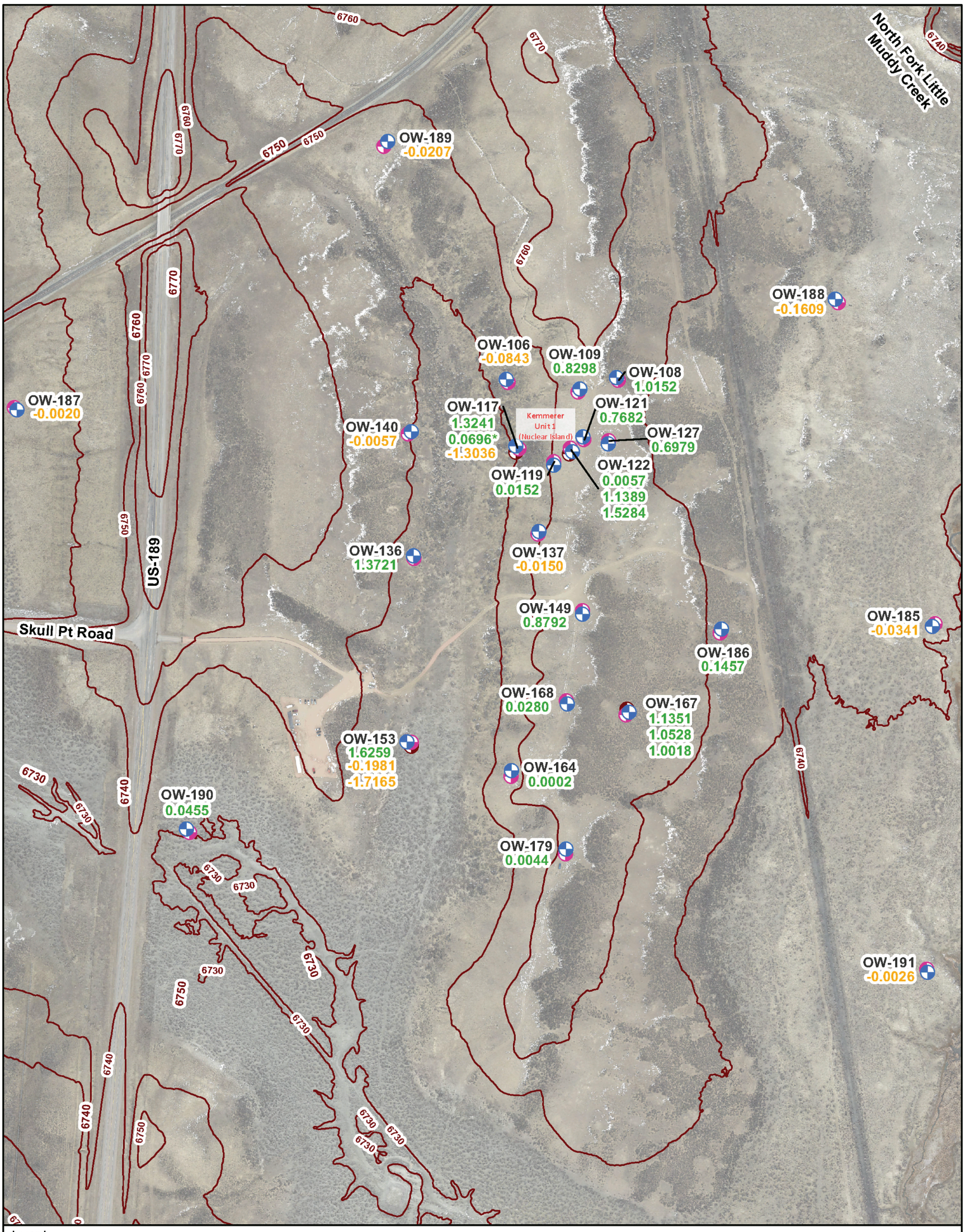
**Figure 2.5-115 Average Vertical Hydraulic Gradients for the U, M, and L Series Wells**




MAP SOURCE INFORMATION

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Image Source: PacifiCorp (2021) LiDAR State Plane (\\amers.ibechtel.com\power\Projects\Natrium\G&HES\PacifiCorp (2021) LiDAR State Plane).
Shapefile for Observation Location: \\amers.ibechtel.com\power\Projects\Natrium\G&HES\Geology\Groundwater Contour Maps\Shapefile\Groundwater_Elevation_March_2023.shp.
Horizontal/Vertical Datum: State Plane Coordinates NAD 1983 (Wyoming West FIPS 4904)/NAVD 1988

**Average Vertical Hydraulic Gradient Between U, M, and L Series Wells**

Natrium Demonstration Project
Kemmerer, Wyoming
Project/Job No.26396

**Figure 12**


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## Figure 2.5-116 Kemmerer Unit 1 Site Plan

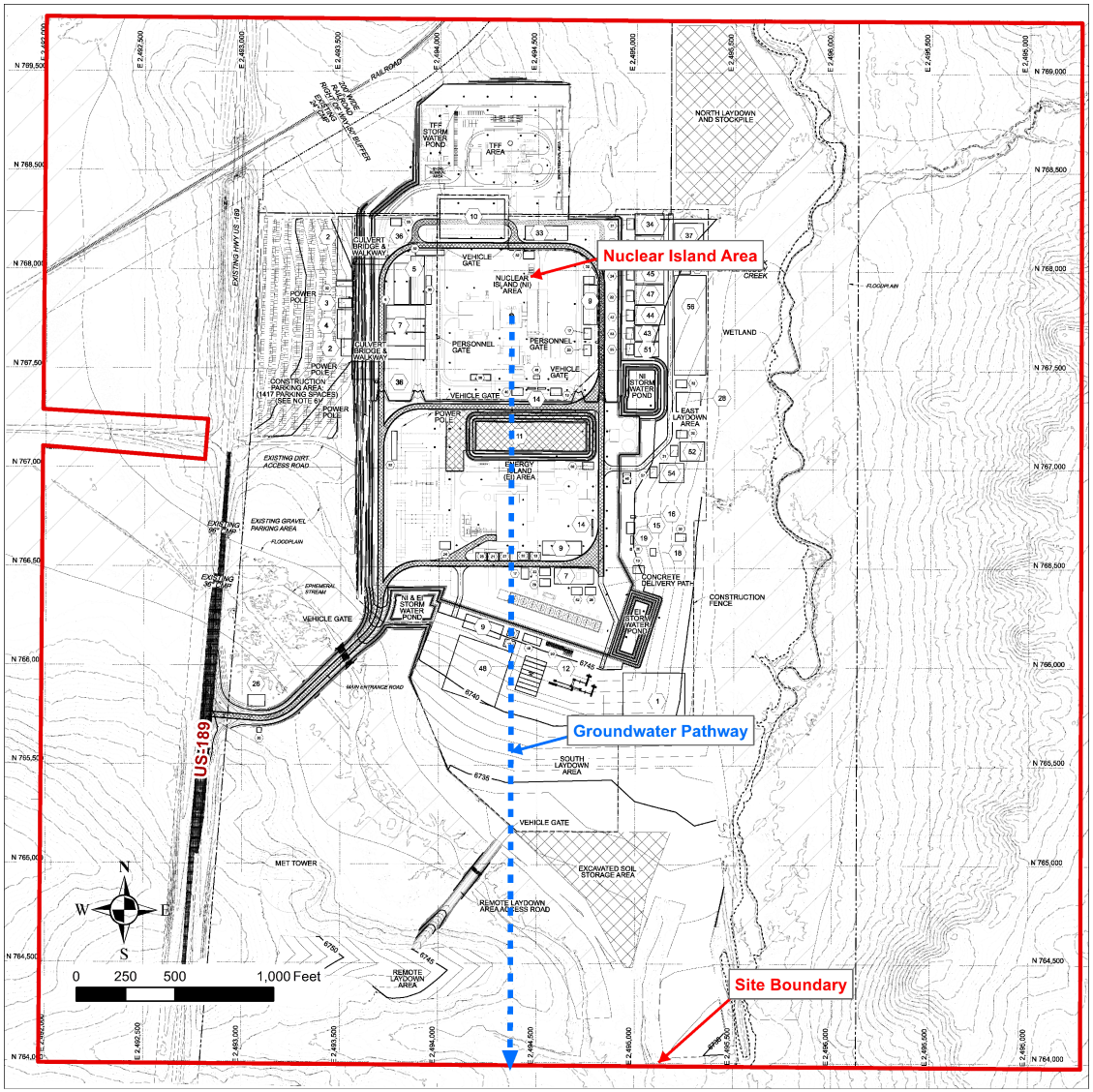



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**Figure 2.5-117 Conceptual Model for Radionuclide Transport in Groundwater**

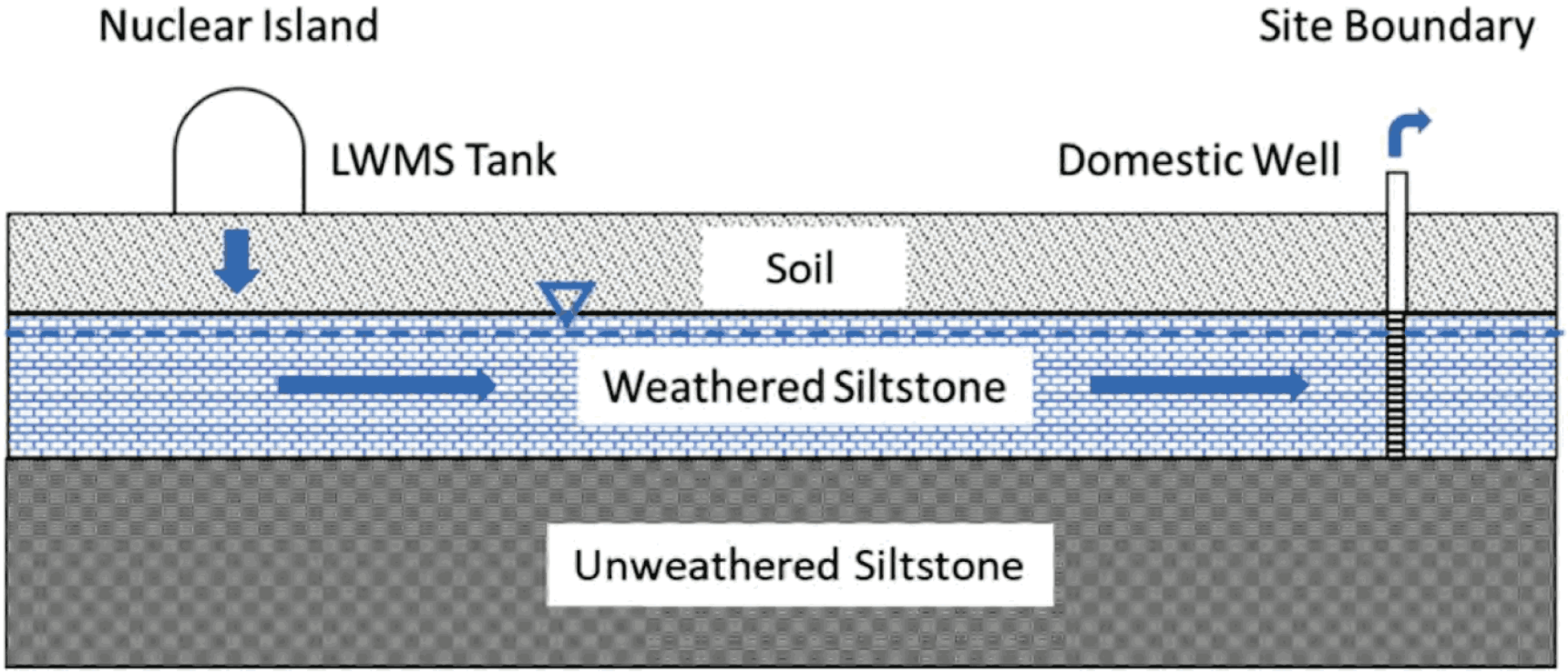



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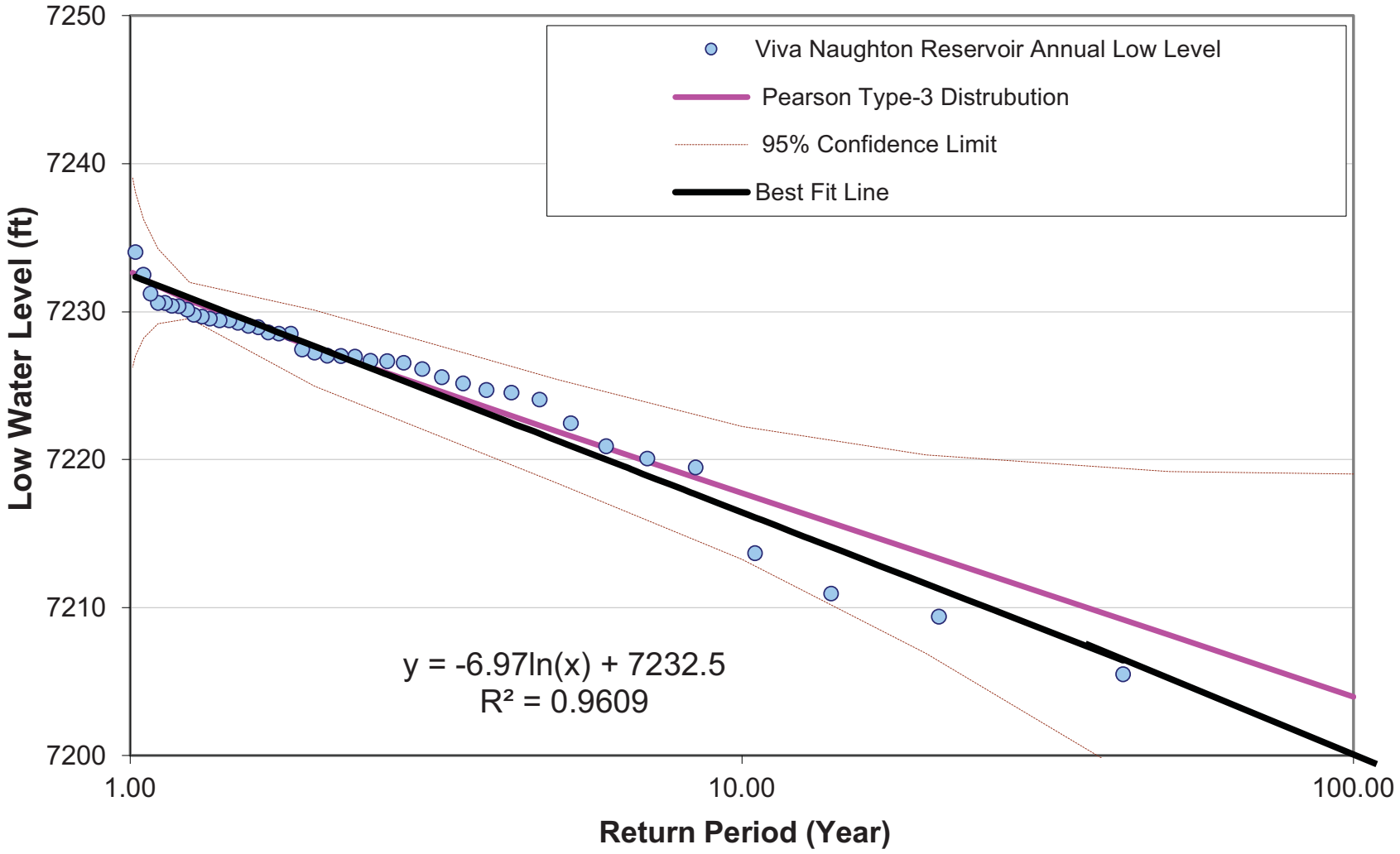
**Figure 2.5-118 Pearson Type 3 Distribution Probability Density Function for Viva Naughton Reservoir Annual Low Level and Best Fit Line**





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# 2.6 Geology, Seismology, and Geotechnical Engineering

This section provides the seismological and geological characteristics of the site and surrounding region to support the evaluation for load-bearing capability and seismic design. This includes derivation of the site-specific Ground Motion Response Spectrum (GMRS) and safe shutdown earthquake (SSE), in compliance with 10 CFR 100.21(d) and 10 CFR 100.23, “Geologic and seismic siting criteria” and 10 CFR 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants.” The results of site geologic, geotechnical, and seismic investigations support analysis and evaluation of seismic effects on safety-significant structures, systems, and components at the site. The size of the region that was investigated and the data obtained is in conformance with Regulatory Guide (RG) 1.208, Revision 0, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” and was determined based on the evaluation of regional seismic hazards and their potential impact on the facility. A summary of the relevant studies describing the site, the investigations performed, and the investigation results and conclusions are provided in this section. Additional detailed geological, geotechnical, and seismological information is documented in separate reports maintained in accordance with the records retention requirements of the quality assurance program.

Section 2.6.1, Geologic Hazards, provides a summary of the site region, site vicinity, site area, and site location geology in Section 2.6.1.1, Section 2.6.1.2, Section 2.6.1.3, and Section 2.6.1.4, respectively. The complete results of the geological investigations are documented in the detailed Geologic Field Reconnaissance Report, that is maintained in accordance with the records retention requirements of the quality assurance program. The report contains the geologic field observations, geologic mapping, and associated literature review. The geologic field reconnaissance was performed to (1) collect data for the development of geologic maps of the site vicinity, site area, and site location in accordance with the criteria described in Appendix C of RG 1.208 and (2) evaluate the Quaternary geology and geomorphology of the site area for evidence of the presence of non-tectonic surface deformation.

Section 2.6.1.5 provides the summary of the development and results of the Seismic Source Characterization (SSC) model applicable to the Kemmerer Unit 1 site. A Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 study (described below in SSHAC Level 3 Study) was performed to develop the SSC model for the project region.

Section 2.6.2, Vibratory Ground Motion, provides a summary of the development and results of the Ground Motion Characterization (GMC) model applicable to the Kemmerer Unit 1 site in Section 2.6.2.1. The development and results of the site-specific Probabilistic Seismic Hazard Analysis (PSHA), that includes the site response analysis and PSHA control point hazard, are provided in Section 2.6.2.2 and Section 2.6.2.3 and were developed as part of the SSHAC Level 3 study (described below in SSHAC Level 3 Study).

Section 2.6.2.4 includes a summary of the development and results of the GMRS and the SSE for the Kemmerer Unit 1 site, documented in separate calculation reports maintained in accordance with the records retention requirements of the quality assurance program.

Section 2.6.3, Surface Deformation, provides a description of the evaluation of non-seismic geologic hazards.


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Section 2.6.4, Stability of Subsurface Materials and Foundations, provides the site characterization evaluation and results of geotechnical studies of subsurface materials related to stability of foundations for safety-significant structures at the site in accordance with the criteria in RG 1.132, Revision 3, “Geologic and Geotechnical Site Characterization Investigations for Nuclear Power Plants.”

Section 2.6.5, Stability of Slopes, provides the results of the evaluation of slope characteristics and stability at the site.

SSHAC Level 3 Study

A SSHAC Level 3 study was performed to develop SSC and GMC models for a regional area that encompasses the greater Wyoming (WY) area, which includes the Kemmerer Unit 1 site region. The site-specific PSHA and site response analysis elements of the study were developed specifically for the Kemmerer Unit 1 site.

The SSHAC Level 3 study was performed following the process and guidance provided in NUREG-2213 (Reference 2.6-1) to develop the SSC and GMC models for the project area.

In accordance with the guidance provided in NUREG-2213, the SSHAC Level 3 study was reviewed by a participatory peer review panel. The participatory peer review involved continual participatory review of the study from start to completion and consisted of a panel of industry experts that reviewed the methodology, technical assessments, incorporation of uncertainty, and documentation of the SSC and GMC models for use in the PSHA. The participatory peer review panel also reviewed the PSHA performed to develop the PSHA control point hazard for the Kemmerer Unit 1 site. The participatory peer review panel issued a final closure letter documenting successful completion of the SSHAC Level 3 study, which concluded that the process, technical assessments, and documentation fully meets the accepted guidance and expectations for a SSHAC Level 3 study.

The PSHA calculations to develop the control point hazard were performed in accordance with “Approach 3” of NUREG/CR-6728 (Reference 2.6-2) and quality assurance procedures for safety-related (SR) work, and the PSHA control point hazard results were used as input to site response analyses to develop the GMRS and SSE for the site following the guidance provided in RG 1.208 in separate calculations.

Complete documentation of the SSHAC process and the technical assessments performed as part of the SSHAC study is included in the Natrium Demonstration Project SSHAC Level 3 Probabilistic Seismic Hazard Analysis report (referred to as the SSHAC Level 3 Study report). The SSHAC Level 3 Study report also includes the project plan, supporting appendices, project subcontractor reports, Workshop and Working Meeting summaries, participatory peer review panel written correspondence, and participatory peer review panel endorsement letter. The SSHAC Level 3 study report provides a detailed and thorough description of the work performed and fully documents the technical assessments and judgments made by the SSHAC. The methods and results of the SSHAC Level 3 study are summarized in Section 2.6.1 and Section 2.6.2, and the separate detailed SSHAC Level 3 Study report is maintained in accordance with the records retention requirements of the quality assurance program.


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## 2.6.1 Geologic Hazards

This section summarizes the geology within 320 km radius site region (Section 2.6.1.1), the 40 km radius site vicinity (Section 2.6.1.2), the 5 km radius site area (Section 2.6.1.3), and the 1 km radius site location (Section 2.6.1.4). A more detailed description of the geologic history, tectonic evolution, and stratigraphic units within these sections is presented in the geologic field reconnaissance report. The SSC model for the site is summarized in Section 2.6.1.5. The geologic characteristics of the Kemmerer Unit 1 Site (Section 2.6.1.1 through Section 2.6.1.4) form the basis of the discussion of seismic sources. Other geologic hazards relevant to the site are described in Section 2.6.2 and Section 2.6.3.

### 2.6.1.1 Regional Geologic and Seismologic Setting

This section addresses the regional geology within the site region (320 km radius) with particular attention to the site vicinity (40 km radius), site area (5 km radius), and site location (1 km radius) of the site in southwest Wyoming. Literature pertaining to the geologic, geomorphic, and seismic setting of the site region was reviewed, including examination of topics pertinent to the geologic and structural setting of the site.

#### 2.6.1.1.1 Regional Physiography and Geomorphology

The site is located in southwest Wyoming, with the site region encompassing parts of eastern Idaho and Utah along with northwest Colorado (Figure 2.6-1). The site is located in the Middle Rocky Mountains physiographic province, which extends for about 676 km from south-central Montana to north-central Utah, while the site region extends into neighboring physiographic provinces (Figure 2.6-2; Reference 2.6-3). Most of the province consists of deeply dissected uplands of resistant intrusive rocks and Quaternary to older sedimentary deposits (Reference 2.6-3). The general features (Reference 2.6-4) that characterize this province are mountainous terrain, stream valleys, alluvial basins, sharp ridge lines, dunes, U-shaped valleys, glacial lakes, and glacial moraines. In the site vicinity, resistant sandstone beds underlie ridges while more easily erodible shale and siltstone commonly underlies low-relief areas. Linear ranges make up the Overthrust Belt, which borders the Middle Rocky Mountains province to the east in Wyoming and forms the western margin of the Wyoming Basin physiographic province (Figure 2.6-2). A linear band of thick sedimentary rocks that were folded and faulted characterizes the Overthrust Belt, which is part of the Cordilleran orogenic belt (Reference 2.6-5).

The eastern margin of the site region is located within the Wyoming Basin physiographic province, which extends from central Wyoming to northern Colorado (Figure 2.6-2; Reference 2.6-3). The Wyoming Basin province is a group of small to large uplifts and intermontane basins, which lies between the Middle and Southern Rocky Mountains physiographic provinces. The Wyoming Basin is one of many intermontane basins that formed during the uplift of the Rocky Mountains and subsequent weathering and erosion resulted in the deposition of layers of sediment on the order of thousands of feet thick in the basin (Reference 2.6-6). North and west of the Rock Springs Arch is the Green River Basin, which is a large structural depression (Figure 2.6-2). The floor of the Green River Basin ranges in elevation from 1,800 to 2,400 meters and has undergone large-scale erosion resulting in broad dissected


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lowlands interspersed with numerous ridges, mesas, and buttes (Reference 2.6-7). Characteristic features of the Wyoming Basin include hogbacks, cuestas, permanent and playa lakes, buttes and mesas, and numerous broad intermontane basins.

#### 2.6.1.1.2 Regional Geologic History and Tectonic Evolution

The regional geologic history and tectonic evolution of the site region is described in detail in the geologic field reconnaissance report. In brief, the site includes several geologic provinces that record an extensive geologic history of the site region going back over 2 billion years (Reference 2.6-8, Reference 2.6-9, and Reference 2.6-10). The geologic history includes the early development of the North American craton or Laurentia, its inclusion and rifting from the supercontinent Rodinia, and a subsequent sequence of accretionary events associated with the subduction of the Farallon plate, referred to as the North American Cordillera. This sequence of accretionary events provides the basis for the creation and juxtaposition of bedrock units as well as the mountain building events (orogenies) in the site region (Figure 2.6-3 and Figure 2.6-4) and site vicinity (Figure 2.6-4 and Figure 2.6-5). These accretionary events include from oldest to youngest, the Antler, Sevier, and Laramide orogenies that helped create the current geological setting of the site region (Figure 2.6-4).

### 2.6.1.2 Site Vicinity Geologic Setting

The geology of the site vicinity primarily consists of Cretaceous synorogenic sedimentary rocks deposited in Western Interior Seaway, the foreland basin of the Sevier Orogeny. Further Sevier thrusting deformed these rocks, which are today part of the Sevier thin skinned fold and thrust belt. Following the end of the Sevier orogeny in the early Cenozoic, a basin containing the ephemeral Lake Gosiute formed and a wide variety of siltstones, claystones, and sandstones were deposited in the basin.

#### 2.6.1.2.1 Site Vicinity Stratigraphy and Lithology

Geologic formations exposed within the site vicinity consist of Paleozoic to Eocene sedimentary rocks (Figure 2.6-5). West dipping thrust faults juxtapose sequences of west to east dipping Paleozoic to Eocene age sedimentary rocks in the Middle Rocky Mountains portion of the site vicinity. Nearly horizontal, predominantly Eocene aged sedimentary rocks underly the east portion of the site vicinity in the Green River Basin, part of the Wyoming Basin physiographic province (Figure 2.6-2).

Figure 2.6-5 is a compilation of the best available geologic maps within the site vicinity. The 1:62,500 scale United States Geological Survey (USGS) map of the Cokeville 30-minute quadrangle (Reference 2.6-11) and the 1:100,000 scale maps of the Kemmerer, Evanston, and Logan 30 minute x 60 minute quadrangles of the Wyoming State Geological Survey and Utah Geological Survey (Reference 2.6-12, Reference 2.6-13, and Reference 2.6-14) were used to characterize the geology of the areas within the site vicinity lacking 1:24,000 scale mapping. Wherever possible a uniform stratigraphic nomenclature and symbology was applied to the map compilation. Boundary conflicts among maps of varying authorship, scale, and vintage were resolved wherever possible in the site vicinity geologic map (Figure 2.6-5).


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The stratigraphy within the site vicinity (Figure 2.6-6 and Figure 2.6-7) is largely based on the description of the geologic units presented in Reference 2.6-12. The major geological units (grouped and un-grouped formations) found in the site vicinity are shown in Figure 2.6-7, Figure 2.6-8, and Figure 2.6-9.

2.6.1.2.2 Site Vicinity Structural Geology and Tectonic Setting

The site vicinity spans the eastern Sevier fold-and-thrust belt and the transition to the foreland basin (Figure 2.6-4 and Figure 2.6-10). The Sevier belt grew from west to east in Cretaceous to early Cenozoic time by formation and propagation of major west-dipping, east-vergent thrust faults (Reference 2.6-15 and Reference 2.6-16). The thrust faults are grouped into five major systems at the latitude of the site, each known by the name of the structurally lowest thrust fault. From west to east, the major thrust systems and their periods of activity include: the Paris-Willard thrust system (possibly Aptian to Albian); Meade-Laketown thrust system (Cenomanian to Turonian); Crawford thrust system (Coniacian through Santonian); Absaroka thrust system (Santonian-early Paleocene); and the Darby-Hogsback system (Paleocene to early Eocene) (Reference 2.6-15; Figure 2.6-10 and Figure 2.6-11). The site vicinity encompasses the eastern part of the Absaroka thrust system and the entire Darby-Hogsback system. The leading edge of the Crawford thrust system is just outside the western boundary of the site vicinity radius (Figure 2.6-5).

Cenozoic extensional deformation and normal faulting have to varying degrees overprinted many of the Sevier structures in Western Wyoming (Reference 2.6-15 and Reference 2.6-17). In detail, many of the normal faults bounding early Cenozoic extensional basins in the Sevier belt are parallel to and in the hanging walls of thrust faults, suggesting that the normal faults cut down through the hanging wall and sole into the thrusts at relatively shallow depths. Examples of these structures in the western part of the site vicinity include a series of north-northeast trending Paleogene graben and associated basin fill in the hanging walls of the Medicine Butte thrust and Tunp thrust in the Absaroka thrust sheet (Reference 2.6-15 and Reference 2.6-17). In another documented mode of extensional reactivation of Sevier thrust structures, normal faults and graben have developed above ramp-flat transitions in the major thrust sheets. In these cases, younger normal faults extend downward and sole into the top of the Sevier ramp, such that a graben forms above the blind ramp-flat transition. Williams and Dixon (Reference 2.6-18) present a seismic reflection profile that likely images this structural relationship between a Cenozoic normal fault and the top of the ramp in the Darby-Hogsback thrust fault (although the authors interpret all faults in the seismic section as thrust faults). A more definitive example is the narrow north-south-trending graben containing Eocene strata that is located 5 to 6 km east of the site (Figure 2.6-5; Reference 2.6-19). This graben is located directly above the ramp-flat transition in the underlying Darby-Hogsback thrust fault. Based on interpretation of a reprocessed east-west seismic reflection profile, the normal fault bounding the eastern margin of the graben is likely the primary fault and it intersects the ramp in the Darby-Hogsback thrust at a depth of about 3 km. The antithetic normal fault on the western margin of the graben intersects the primary west-dipping fault at a depth of about 1.5 km, consistent with the relatively narrow (1 to 1.5 km) surface width of the graben that M'Gonigle and Dover (Reference 2.6-19) mapped.


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Constenius (Reference 2.6-17) interpreted the extensional structures in the examples above to have formed shortly after the cessation of Sevier-Laramide shortening in Eocene time. They accommodated extension or “collapse” of the ancestral Cordilleran orogen during roll-back or foundering of the flat Farallon slab. Both styles of deformation suggest that the kinematics of Paleogene extension consisted primarily of reversal of slip on the contractional Sevier-Laramide structures (Reference 2.6-17). Some late Cenozoic normal faults, which presumably are genetically associated with the younger Neogene-Quaternary phase of extension that led to the formation of the modern Basin and Range topography in the Cordillera (Reference 2.6-20), also are localized along Sevier thrust faults. For example, the normal, late Quaternary Rock Creek fault, located about 25 km northwest of the site, is parallel to and in the hanging wall of the Tunp thrust fault, and may be partially or wholly utilizing the older fault surface at depth to accommodate active west-down extension in the modern tectonic setting. Similarly, the surface trace of the late Quaternary Bear River fault zone about 70 km south-southwest of the site overlies a ramp-flat transition in the Darby-Hogsback thrust fault. West (Reference 2.6-21) interpreted that the Bear River fault zone soles into the ramp of the underlying Darby-Hogsback thrust, reactivating it as a normal fault.

### 2.6.1.3 Site Area Geology and Structure

The bedrock geology of the site area is predominantly composed of Cretaceous sedimentary rocks. These were deposited in the foreland basin of the Sevier Orogeny and were subsequently deformed in a thin-skinned fold and thrust belt when the Sevier Orogeny propagated eastward. The site area has been tectonically quiescent since the cessation of Sevier deformation in the early Cenozoic, with the exception of a post-Eocene period of normal faulting that produced a horst and graben structure.

#### 2.6.1.3.1 Site Area Bedrock Stratigraphy

The stratigraphic nomenclature conforms to that used in the M’Gonigle and Dover (Reference 2.6-12) Geologic Map of the Kemmerer 30 minute x 60 minute Quadrangle, Lincoln, Uinta, and Sweetwater Counties, Wyoming and the M’Gonigle (Reference 2.6-22) Geologic Map of the Elkol Quadrangle, Southwestern Wyoming.

The bedrock stratigraphy exposed in the site area (Figure 2.6-8) consists of mainly Cretaceous age sedimentary rocks deposited during Sevier thrusting, sedimentation, and Western Interior Seaway transgression and regression. Eocene age sedimentary rocks deposited as sediments were shed from the Sevier topographic highs into the Green River Basin. Outside of the site area, Mesozoic sedimentary rocks are found in the hanging walls of Sevier thrusts (Section 2.6.1.1.2). Stratigraphic nomenclature is uniform across the site area. Detailed descriptions of the bedrock formations present within the site area are presented below and are based on field observations, unit descriptions (Reference 2.6-12), and boring logs from the site.

**Frontier Formation**

The Upper Cretaceous Frontier Formation consists of an interbedded marine and nonmarine sequence of sandstone, siltstone, and carbonaceous shale with interbeds of coal and, locally, porcellanite and conglomerate (Figure 2.6-7 and Figure 2.6-8). It is subdivided into the Chalk Creek Member, Coalville Member, Allen Hollow Member, Oyster Ridge Sandstone Member, and


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Dry Hollow Member (Reference 2.6-12 and Reference 2.6-22). Outcrops of sandstone and coal of the Frontier Formation are well-exposed at the ground surface in the Blazon Gap with several fining upward sequences and coal seams along the west dipping bedding (Figure 2.6-12 and Figure 2.6-13). The Frontier Formation is generally well-exposed throughout the site area (Figure 2.6-13), especially where it crosses Hams Fork and is locally overturned on the west limb of the Lazeart Syncline (Figure 2.6-13).

*Chalk Creek Member*

The Chalk Creek Member (Figure 2.6-7 and Figure 2.6-8) is composed of gray, greenish-gray, and brown nonmarine shale, bentonite shale, tuff and tuffaceous sandstone, carbonaceous shale, coal, and tan fine to coarse grained platy to crossbedded sandstone with local bioturbation that is interpreted to have formed in a deltaic environment (Reference 2.6-12 and Reference 2.6-23). The Chalk Creek member contains mollusk fauna and is 300 to 430 meters thick. The Chalk Creek Member was lumped into the Lower Frontier Formation (undivided) unit (Figure 2.6-5).

*Coalville Member*

The Coalville Member consists of dark greenish-gray shale and tan fine grained bioturbated sandstone with crossbeds and ripples. The Coalville Member contains brackish to marine fauna and is 24 to 46 meters thick. It is thought to have been deposited along deltaic and nearshore environments during a regional transgression of the Western Interior Seaway (Reference 2.6-12 and Reference 2.6-23). The Coalville Member was incorporated into the Lower Frontier Formation (undivided) unit (Figure 2.6-5).

*Allen Hollow Member*

The Allen Hollow Member is primarily composed of dark gray to greenish-brown marine shale with sandstone and siltstone beds present in the upper part of the member (Figure 2.6-7 and Figure 2.6-8). This unit has a total thickness of 92 meters and is largely covered. The Allen Hollow Member was incorporated into the Lower Frontier Formation (undivided) unit (Figure 2.6-5).

*Oyster Ridge Sandstone Member*

The Oyster Ridge Sandstone Member contains predominantly light tan to white fine to medium grained cliff forming sandstone with cross-bedding and some brown shale layers (Figure 2.6-7, Figure 2.6-12, and Figure 2.6-8). There is minor channeling in the upper part of the member. The Oyster Ridge Sandstone is interpreted to have been deposited in a prograding nearshore marine setting and is 15 to 60 meters thick (Reference 2.6-12 and Reference 2.6-23).

*Dry Hollow Member*

The Dry Hollow Member is 100 to 130 meters thick and represents the uppermost member of the Frontier Formation (Figure 2.6-7 and Figure 2.6-8). It has gray, greenish-gray, and tan deltaic shale and siltstone with tan and brown fine to medium grained platy to crossbedded and locally bioturbated thin sandstone interbeds and coal seams (Reference 2.6-12 and Reference 2.6-23).


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In the upper part of the Dry Hollow Member are marine fauna, which directly overly the coal rich beds of the locally named “Kemmerer coal zone.” Locally, the lower portion of the Member has a pebbly conglomerate that forms channels 1 to 3 meters thick cutting into the Oyster Ridge Sandstone. Boring B-122 at the site was advanced into the underlying Dry Hollow Member of the Frontier Formation. Drilling cores from the Dry Hollow Member included sandstone, siltstone, shale, carbonaceous shale, and a coal bed with a drilled thickness of approximately 28 feet. The Dry Hollow Member was incorporated into the Upper Frontier Formation (undivided) unit on Figure 2.6-5.

**Hilliard Shale**

The Upper Cretaceous Hilliard Shale consists of dark olive gray marine shale, siltstone, and sandy shale with thin interbeds of limestone and tan to light gray sandstone, especially in the upper part of the unit (Figure 2.6-7 and Figure 2.6-8; Reference 2.6-12). The Hilliard Shale sampled during drilling consisted predominantly of siltstone, sandy siltstone, interbedded sandstones, and minor bentonitic intervals up to approximately 0.2 meters thick (Figure 2.6-14). The claystone and siltstone are weak, poorly exposed, and thinly bedded and make up the bulk of the unit. M’Gonigle and Dover (Reference 2.6-12) subdivided the Hilliard Shale into the Conglomerate of Little Muddy Creek, Shurtleff Sandstone Member, and the Hinshaw Member. Only the Hinshaw Member is located within the site area. A marker bed of fossiliferous marl (limey mudstone) of the Hilliard Shale with shells and shell casts, strong reaction to hydrochloric acid, and parallel to regional strike are present at the ground surface in the site area (Figure 2.6-15).

*Hinshaw Member*

The Hinshaw Member is located in the upper portion of the Hilliard Shale and represents a transitional sequence from the main body of the Hilliard Shale into the overlying Lazeart Sandstone member of the Adaville Formation (Figure 2.6-8). It consists of marine shale interbedded with 0.3 to 10 meters thick sandstone beds. The sandstone is gray to tan, fine grained, contains feldspar and chert grains, and has sharp basal contacts. Sedimentary structures include burrows, hummocky bedding, trough cross stratification, large-scale ball and pillow structures, and difficult to discern bedding mainly 0.2 to 0.5 meters in thickness. The thickness of the Hinshaw Member ranges from 260 to 305 meters thick. The Hinshaw Member is exposed in the site area along the Lazeart Syncline (Figure 2.6-8 and Figure 2.6-13).

**Adaville Formation**

The Upper Cretaceous Adaville Formation consists of interbedded, primarily nonmarine, gray, brown, and tan shale and siltstone, brown and gray, medium to fine grained and locally bioturbated sandstone that is platy to cross-bedded, and carbonaceous shale (Figure 2.6-8). The bedded sandstone is calcareous, and partly conglomeratic in the upper part of the formation. This unit contains numerous coal beds that are 5 to 35 meters thick. Within the Adaville Formation is the Lazeart Sandstone Member (Reference 2.6-12 and Reference 2.6-22). Outcrops of coal seams of the Adaville Formation are well-exposed at a roadcut along


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Highway 30 (Figure 2.6-13 and Figure 2.6-16) and elsewhere in the site vicinity (Figure 2.6-13). Sandstone of the Adaville Formation is generally clean, fine-grained, and locally cross-bedded in the site area (Figure 2.6-13).

*Lazeart Sandstone Member*

The Lazeart Sandstone Member is a very light gray, yellow-brown, and tan, fine to medium grained “salt and pepper” sandstone that forms thick cliffs and ledges with brownish-gray shale and coal beds in the intervening slopes (Figure 2.6-8). It includes marine fossils and bedding suggesting progradational beach sequences which overlap to the south. The unit is approximately 180 meters thick; however, it thins and pinches out north of Hams Fork (Reference 2.6-12; Figure 2.6-8 and Figure 2.6-13).

**Hams Fork Conglomerate**

The Upper Cretaceous Hams Fork Conglomerate consists of poorly to moderately consolidated cobble and boulder conglomerate beds containing sandstone and siltstone matrix interstratified with sandstone and mudstone beds (Figure 2.6-8). Cobble and boulder clasts include quartzite, chert, and limestone and are well-rounded and approximately 15 cm on average (up to 0.5 meters) in diameter. Clasts are derived from formations such as the Worm Creek Quartzite Member of the St. Charles Limestone and the Brigham Quartzite of southeastern Idaho (Reference 2.6-24). Where exposed, this unit forms hogbacks at the surface, but generally forms poor outcrops with bedding frequently marked with trains of loose boulders with calcium carbonate rinds or caliche (Figure 2.6-13 and Figure 2.6-17). The Hams Fork Conglomerate is concordant and folded with the underlying Adaville Formation within the Lazeart Syncline (Reference 2.6-12).

2.6.1.3.2 Site Area Structural Geology

The review of the structural setting is based on existing published information supplemented with observations from the site geologic field reconnaissance. New data include geologic mapping and bedding attitudes measured from rock outcrops. The geologic mapping of the site area is presented in Figure 2.6-8 and in site area cross sections (Figure 2.6-11 and Figure 2.6-18). Bedding generally strikes to the north and dips moderately (20 to 30 degrees) to the west.

West dipping bedding characterizes the bedrock structure of the site area, especially within the Cumberland Flats and along Oyster Ridge (Figure 2.6-8). At the site, strata of the Hilliard Shale generally strike about 7 degrees with dips of about 20 degrees west.

The Darby-Hogsback thrust system underlies the site area and represents the eastern extent of the Sevier belt (Figure 2.6-10 and Figure 2.6-11). The Darby-Hogsback thrust fault breaks upward from the basal decollement approximately beneath the western edge of the Cumberland Flats. Bedding in the hanging wall of the Darby-Hogsback fault dips uniformly west above the ramp, generating the west-dipping homocline of Cretaceous strata that underlies the Cumberland Flats and the site (Reference 2.6-22 and Reference 2.6-25). At approximately 6 km depth, the ramp of the Darby-Hogsback thrust overlies a repeated section of Triassic and Paleozoic strata for about 2 km, which overlies Precambrian crystalline basement of the basal decollement at


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about 8 km depth (Figure 2.6-6 and Figure 2.6-11; Reference 2.6-15). The Darby-Hogsback thrust flattens in Upper Cretaceous strata east of Cumberland Gap and extends for about 9 km east as a blind thrust past the eastern boundary of the site area radius (Reference 2.6-15).

Steeply dipping normal faults are common in the eastern portion of the site area (Figure 2.6-8) within both Cretaceous and Eocene deposits. These normal faults create a horst and graben structure with a narrow north-south-trending graben containing Eocene strata that is located 5 to 6 km east of the site (Figure 2.6-8 and Figure 2.6-18). This graben is located directly above the ramp-flat transition in the underlying Darby-Hogsback thrust fault. The normal fault bounding the eastern margin of the graben is likely the dominant structure, and it intersects the ramp in the Darby-Hogsback thrust at a depth of about 3 km. East of this graben the normal faults are generally downthrown to the east. The antithetic normal fault on the western margin of the graben intersects the primary west-dipping fault at a depth of about 1.5 km, consistent with the relatively narrow (1 to 1.5 km) surface width of the graben as mapped (Reference 2.6-19).

#### 2.6.1.3.3 Site Area Geomorphology and Surficial Geology

The site is located within the Cumberland Flats (Figure 2.6-8) which is the broad, relatively flat-lying, north-south trending valley produced from weathering of the Hilliard Shale. The broad Cumberland Flats valley, containing long, roughly north-south trending resistant ridges associated with coarser grained, more indurated facies within the Hilliard Shale, characterizes the site area topography. Oyster Ridge, which bounds the eastern side of the valley, rises 183 meters above the Cumberland Flats. A second unnamed ridge to the west, which includes Skull Point and Crockers Point, lies along the western boundary of the Cumberland Flats and rises over 150 meters. This western ridge contains the coal rich Adaville Formation and the open pit Kemmerer coal mine in the northwest portion of the site area which consists of large, graded areas and stockpiles of mine tailings (Figure 2.6-8). Several smaller open pit mines as well as mine adits were observed in the site area but none within the site location (Figure 2.6-8 and Figure 2.6-9).

Hams Fork, a tributary of the Green River, flows easterly across the northern portion of the site area and cuts through Oyster Ridge (Figure 2.6-8 and Figure 2.6-9). The North Fork of Little Muddy Creek provides the principal drainage for the site area and flows southerly through the center of the site area following Oyster Ridge with an incised water gap at Blazon Gap where it crosses the Oyster Ridge to the east (Figure 2.6-8). The Skull Point Reservoir lies in the northwestern portion of the site area near the Naughton Power Plant and several other small lakes and ponds occur within the Cumberland Flats in the site area (Figure 2.6-8 and Figure 2.6-19).

The Quaternary surficial units in the site area consist primarily of Holocene to Upper Pleistocene alluvial deposits (Qal) within the stream valleys and tributaries of the North Fork of Muddy Creek and Hams Fork (Figure 2.6-9). Unit Qal includes channel-fill and floodplain deposits as well as some alluvial fan, terrace deposits, valley side colluvium or talus, and sediments deposited in bogs, lakes, or deltas, locally, that are undivided at the scale of mapping. The unit is composed of clay, silt, and sand, with gravel present in varying amounts dependent on the local bedrock parent material.


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Holocene and Pleistocene Colluvium (Qc) deposits consist of weathered residuum transported by hillslope processes including slopewash, landslides, and slope creep. Colluvium is deposited at the toe of hillslopes and in hollows on the hillsides. Colluvium mapped in the site area is Holocene to Pleistocene in age. The thickness and areal extent of colluvium deposits varies depending on the subsurface bedrock unit. The Adaville Formation, which erodes primarily by mechanical weathering, produces abundant colluvial deposits which blanket the lower angle slopes underlain by stratigraphically adjacent units. Few distinct landslides were observed within the site area (Figure 2.6-8), and none were observed at the site location (Figure 2.6-9).

Two gravel deposits are present within the site area (Figure 2.6-8). Holocene and Pleistocene gravel (Qg) consists of an unconsolidated, well-rounded, gravel, cobble, and boulder pavement deposit common on pediment surfaces and commonly located on prominent topographically higher benches (Figure 2.6-20). This unit is commonly comprised of quartzite and agate from erosion of nearby conglomerate rich bedrock formations and typically consists of larger cobbles and boulders with some calcareous varnish. Pleistocene and Pliocene high-level terrace gravel (QTg) consist of these same gravels on terraces and pediments more than 75 meters above present major stream drainages. These boulder and cobble sized clasts have thick calcium carbonate rinds or caliche with relict carbonate staining that occurred prior to their current deposition and indicates they are being transferred locally from the Cretaceous Hams Fork Conglomerate of the Evanston Formation (Figure 2.6-5).

### 2.6.1.4 Site Location Geologic Mapping

Original mapping within the site location was conducted based on field reconnaissance and Light Detection and Ranging (LiDAR) analyses (Figure 2.6-9). A series of topographic highs, primarily north-south oriented, with intervening lows containing relatively small drainages, characterize the site location. There is a prominent ridge, also oriented north-south, located to the east that forms the topographic highpoint of the site location. While in the field, it was verified that Hilliard Shale exclusively underlies the site location, and that mapping differs from previous mapping only for the Quaternary units. Previous Quaternary mapping was modified to reflect the geomorphic and topographic characteristics of the site location (Figure 2.6-9). Mapping of gravel (Qg) extended along topographic highs to match field observations. Gravel, bedrock (Kh), colluvium (Qc), and fan material (Qf) were mapped along the slope of the eastern ridge to characterize slope processes in detail (Section 2.6.3.1.1). The extent of alluvial materials (Qal) and Holocene active channels (Qch) was also mapped within topographic lows. In addition, anthropogenic features such as mine tailings, stockponds, and historic railroad tracks were mapped as artificial fill (af); (Figure 2.6-9). Geologic mapping is described in more detail in Section 2.6.3.

### 2.6.1.5 Seismic Source Characterization of Kemmerer Unit 1

The geologic history, structure, and tectonic evolution of the site described in Section 2.6.1.1 through Section 2.6.1.4, contributes to the current tectonic setting defined by the seismic sources. The seismic source characterization study conducted as part of the regional SSHAC Level 3 study (Section 2.6) was initiated to adequately characterize the site region for seismic hazards applicable to the site. The SSHAC Level 3 reporting is summarized in this section. Figure 2.6-21 shows the project region included in the SSHAC Level 3 study, which envelops a 320 km radius site region around the Kemmerer Unit 1 site.


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#### 2.6.1.5.1 Regional Earthquake Catalog

The regional earthquake catalog is a critical input to the SSC model. The catalog is used to help characterize the seismotectonic setting of the site region as well as to develop recurrence parameters for the seismic source zones. Following regulatory guidelines, the regional earthquake catalog was compiled for a region that extends at least 320 km around each of four proposed sites (Figure 2.6-22) and envelopes the Kemmerer Unit 1 site region. The catalog extends from a latitude of 38.50 degrees north to 47.50 degrees north and from a longitude of 115.55 degrees west to 100.90 degrees west. The catalog extent was partially determined by the extents of the primary source catalogs used in the regional earthquake catalog compilation. The final regional catalog spans the period of 1850 through December 31, 2020.

The regional earthquake catalog was primarily assembled from three previous catalog compilations within the region:

- Updated Central and Eastern United States (CEUS) (Reference 2.6-26)
- Idaho National Laboratory (INL) (Reference 2.6-27)
- Working Group on Utah Earthquake Probabilities (WGUEP) (Reference 2.6-28)

Other compilations used include:

- USGS Advanced National Seismic Network Comprehensive Earthquake Catalog (Reference 2.6-29)
- National Earthquake Information Center Preliminary Determination of Epicenters (Reference 2.6-30)
- Global Centroid Moment Tensor (Reference 2.6-31)
- Saint Louis University North America Moment Tensor (Reference 2.6-32)
- 2018 National Seismic Hazard Mapping Program catalogs (Reference 2.6-33)

After removal of duplicate earthquakes and inclusion of an additional 568 non-tectonic earthquakes, the regional earthquake catalog contains 22,973 earthquakes of magnitudes ranging from Expected Value of Moment Magnitude (E[**M**]) 2.3 to 7.2, Earthquake Equivalent Count (N*) values ranging from 1.002 to 2.592, and includes earthquakes from 1850 through December 31, 2020 (Figure 2.6-22).

#### 2.6.1.5.2 Seismic Source Characterization Model Implementation

The SSC model defines the seismogenic potential, locations, magnitudes, and rates of future earthquakes which can affect the site region. In combination with the site profile and the GMC model described in Section 2.6.2.1, the SSC model was developed as input to the site-specific PSHA described in Section 2.6.2.2.


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## Overview of SSC Model

The SSC model has two types of seismic sources: seismic source zones (sometimes called "areal" or "background" sources) and fault sources, as shown in the master logic tree (Figure 2.6-23). Logic trees are used to represent the alternative models and parameter values for the key components of the SSC model. For example, the range of magnitudes for a given source are shown on different branches of the logic tree, and then weighted.

Seismic source zones represent non-specific sources of future earthquakes from unrecognized faults in the crust. Seismic source zones are characterized with a defined geometry, crustal thickness, rate of earthquake occurrence, Earthquake Maximum Magnitude ($M_{MAX}$), and Magnitude-Frequency Distribution function. Potential future earthquakes in the seismic source zones are treated as "point" sources and are modeled with rupture characteristics such as geometry, rate, and slip sense. Eleven seismic source zones affect the site were developed and are illustrated (Figure 2.6-24).

Fault sources are specific planar sources of earthquakes that are attributed to well-defined, seismogenic or potentially seismogenic geologic faults or fault zones. The tectonic features that are treated as fault sources are sufficiently active in the current tectonic regime and have been considered as specific sources of seismic hazard that are separate and distinct from adjacent areas (i.e., seismic source zones). Mapped location, geometry, depth, slip sense, slip rate, magnitude-frequency distribution shape, and probability of the occurrence of an earthquake in a given time period characterize fault sources. Figure 2.6-24 shows the fault sources in the site region that are included in the SSC model. Figure 2.6-25 shows distant sources beyond the site region that can affect the site and are included in the SSC model.

## Seismic Source Zone Characterizations

Seismic source zones for the site region (blue lines, Figure 2.6-26) were developed by reviewing the SSC model for the INL (Reference 2.6-27) PSHA, which focused on large areas of Idaho, northern Utah, western Wyoming, and southwestern Montana in and around the INL site in the Eastern Snake River Plain. The boundaries of many INL source zones (pink lines, Figure 2.6-26) are generally adopted with some revisions. For areas of Wyoming, Montana, Colorado, and the Dakotas that minimally overlap or lie entirely to the east of the INL study region, new seismic source zones were defined based on evaluation of province-scale geodynamic relationships such as variations in lithospheric structure, heat flow, and the state of stress that are interpreted to influence the kinematics and magnitude of future earthquake ruptures.

Parameters that define the likely range of geometry and sense of slip of future earthquake ruptures characterize the seismic source zones. As summarized in Table 2.6-1 and Figure 2.6-27, the parameters include seismogenic thickness, fault kinematics (which encompass the likely range of geometry and sense of slip of future ruptures), and $M_{MAX}$. Literature on the lithospheric structure and geodynamics in the site region and focal mechanisms of small earthquakes were reviewed to evaluate the kinematics of seismogenic deformation in selected areas. The seismotectonic analysis included inversions of groups of focal mechanisms for components of a reduced incremental strain tensor using a micropolar continuum model (Reference 2.6-34 and Reference 2.6-35). The inversion approach employed is analogous to


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methods for deriving a reduced stress tensor for a group of focal mechanisms, but the micropolar model that forms the basis for the inversion assumes deformation boundary conditions with additional kinematic degrees of freedom (rather than stress), and it permits direct comparison of seismogenic strain with independent geodetic estimates of strain. The micropolar approach previously was used to evaluate seismogenic deformation in source zones for the INL investigation (Reference 2.6-36), and its application here provides for continuity in seismotectonic characterization between the two studies

The following background source zones were defined as part of the SSC model. The Intermountain Seismic Belt (ISB) and Colorado Plateau-Wyoming Craton (CP-WC) contribute significantly to hazard at the site. The remaining background sources each contribute less than 1% to the total hazard (Table 2.6-11).

- Intermountain Seismic Belt
- Colorado Plateau-Wyoming Craton
- Trona Triggered Seismicity Zone
- Yellowstone Geoid Anomaly
- Eastern Snake River Plain
- Basin and Range
- Northern Rio Grande Rift
- Yellowstone Caldera
- Centennial Tectonic Belt
- Idaho Batholith
- Great Plains

Intermountain Seismic Belt

The ISB (Figure 2.6-26) is a zone of transtensional deformation south and southwest of Yellowstone that accommodates oblique left-lateral translation of the Eastern Snake River Plain as an approximately rigid block relative to western Wyoming, southeastern Idaho, and northern Utah (Reference 2.6-27). Analysis of associated Global Positioning System (GPS) data indicated distributed left-lateral transtensional shear across an approximately 150 km wide, northeast-trending region east of the Eastern Snake River Plain that extends into western Wyoming; this zone encompasses the ISB and its major Quaternary-active structures such as the Wasatch, Bear Lake, and Grand Valley normal faults. The zone of distributed deformation is expressed as an eastward-decreasing velocity gradient in east-west velocity profiles across the ISB.

The boundaries of the ISB seismic source zone adopted in this study are the same as those of INL (Reference 2.6-27) with slight modifications (Figure 2.6-26). The western boundary of the ISB zone with the Basin and Range zone is modified to follow the western margin of a gradient in the base of seismicity that Zeng and others (Reference 2.6-38) modeled to deepen eastward, toward the Wyoming Craton and Colorado Plateau (Figure 2.6-28). The eastern boundary of the ISB source zone is defined to lie at the eastern end of the GPS velocity gradient between the Eastern Snake River Plain and relatively tectonically stable crust in western Wyoming.


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The maximum tensile principal stress in the ISB is interpreted to be approximately east-northeast-west-southwest to east-west (Reference 2.6-39), at a high angle to major Quaternary-active normal faults. Kinematic analysis of focal mechanisms from the ISB indicates the seismogenic deformation accommodates approximately east-west horizontal extension (Reference 2.6-36). Major normal faults of the ISB (e.g., Wasatch, Bear Lake, and Grand Valley faults) strike north-south to north-northwest-south-southeast, and exhibit a right-stepping, en echelon geometry consistent with regional left-lateral wrench tectonics (Reference 2.6-27). The transtensional style of deformation is reflected in a mix of normal and strike-slip focal mechanisms from small earthquakes in the ISB source zone (Reference 2.6-27; Table 2.6-1).

Colorado Plateau – Wyoming Craton

The CP-WC seismic source zone lies east of the Basin and Range, ISB, and Yellowstone hotspot (Figure 2.6-26). Lithosphere that has been relatively stable tectonically since the end of the Laramide orogeny and was not subject to distributed Neogene extension that deformed the adjacent ISB and Northern Rio Grande Rift underlies the CP-WC seismic source zone. Relatively colder and less buoyant upper mantle (Figure 2.6-29) and lower heat flow (Figure 2.6-30) than in surrounding source zones to the west, northwest, and southeast characterize the CP-WC source zone. Based on limited seismicity and the Zeng and others (Reference 2.6-38) analysis of heat flow data east of the ISB, the CP-WC source zone is assessed to have thicker seismogenic crust than the adjacent, more tectonically active regions (Figure 2.6-28; Table 2.6-1).

**Seismic Source Characterization Fault Sources**

This section lists the individual fault sources that make up the SSC model and describes in more detail only the characteristics of the Rock Creek fault which is located nearest to the site and is the most hazard significant.

The identification and characterization of fault sources is based in part on information from:

- USGS Quaternary Fault and Fold Database (QFFD) faults in the site region (Figure 2.6-31)
- National Seismic Hazard Model 2023 (NSHM23) fault sources in the site region (Figure 2.6-32)
- INL SSHAC Level 3 fault sources (Reference 2.6-27)
- WGUEP fault sources (Reference 2.6-28)
- Other regional PSHA and seismic hazard studies
- Evaluations of published and unpublished literature and maps
- Evaluations of LiDAR data

To facilitate the organization and presentation of the numerous fault sources in the model, this discussion is subdivided into five different regional classifications of faults, shown in Figure 2.6-33 and listed in Table 2.6-2. These classes of faults are simply based on geographic location, but two groups (Clustered Faults and Wasatch Fault System) are also grouped together based on fault behavior (Clustered) and potential linkage (Wasatch). Descriptions of fault sources within each class or region are ordered by distance from the site.


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The final SSC model includes fault sources in Wyoming, Idaho, and Utah (Figure 2.6-31). Closest site-to-source distance to the site is listed in Table 2.6-3. Fault source parameters are listed in the SSC Hazard Input Document. A master logic tree illustrating how fault sources are characterized is presented in Figure 2.6-34 and applies to most fault sources with minor differences in the recurrence portion of the tree based on the data available to constrain recurrence. The three clustered faults (Rock Creek, Bear River, and Greys River) include more detailed logic trees to include a behavior mode that models whether a fault is currently operating as either in or out of a temporal cluster of earthquake activity. The decision on whether clustered behavior has ended or will continue is a hazard-significant branch in the logic tree that is more important to hazard than the selection of temporal recurrence models (e.g., Poisson, renewal) (Reference 2.6-40 and Reference 2.6-41).

Given the potential hazard contributions at long periods (e.g., 10 sec), two distant sources, the Cascadia subduction zone interface and the New Madrid repeated, large-magnitude earthquake, are included in the SSC model. No other distant sources contributed to the seismic hazard at the site.

**Clustered Fault Sources**

Three faults along the eastern extent of active normal faulting in the Wyoming Salient region exhibit pronounced temporally clustered behavior with high strain release rates in the Holocene that appear to greatly exceed the long-term strain accumulation rate which modern low geodetic rates and the absence of long-term geomorphic features in the landscape (e.g., mountain fronts). Under the headings below, these clustered fault sources are presented in order of increasing distance from the site (Figure 2.6-35) and include the following:

- Rock Creek fault
- Bear River fault
- Greys River fault

The earliest paleoseismic work on the Rock Creek (Reference 2.6-42), Bear River (Reference 2.6-43 and Reference 2.6-44), and Greys River faults (Reference 2.6-42 and Reference 2.6-45) revealed that these faults each ruptured twice in the mid-to-late Holocene and this high rate of recent activity was significantly greater than their long-term behavior. McCalpin (Reference 2.6-42) also showed that not only do these three faults individually exhibit clustered behavior, but regional faults as a group appear to be clustered in time (Figure 2.6-36). More recent work of Hecker and others (Reference 2.6-46) has documented an additional late Holocene event on the Bear River fault (Figure 2.6-37). An additional late Holocene rupture of the Rock Creek fault is included in the reinterpretation of the Cook Canyon trench log. Hecker and others (Reference 2.6-46) updated the McCalpin (Reference 2.6-42) diagram (Figure 2.6-36) to include the additional Bear River rupture (Figure 2.6-37).

The Rock Creek, Bear River, and Greys River faults are broken out from other faults in the Wyoming Salient region and characterized differently because of their extremely temporally clustered behavior with a high rate (approximately 1 millimeter per year) of Holocene strain release, an order of magnitude lower GPS rates, and an apparent lack of geomorphic or geologic evidence that could justify this high rate had persisted for any significant time in the past. Other


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faults in the region that have ruptured in the mid- to late-Holocene and also appear to be slipping at a relatively high rate exhibit geomorphic and geologic evidence of this behavior occurring over much of the Quaternary. For example, the Eastern Bear Lake and Star Valley section of the Grand Valley fault exhibit scarps in Holocene and latest Pleistocene alluvial deposits and also form Class 1 range fronts (Reference 2.6-47) and alluvial-filled valleys on the hanging wall. These basic characteristics document that persistent base level erosion has occurred during the Quaternary across the Eastern Bear Lake and Star Valley faults, as opposed to the three clustered fault sources.

The Rock Creek fault represents the most hazard significant fault source to the site and its characteristics are described below in detail.

**Rock Creek Fault**

The Holocene-active Rock Creek fault lies about 23 km west-northwest of the site and represents the most hazard significant fault source to the site. Maps of the Rock Creek fault and source are shown in Figure 2.6-38. The characterization of the Rock Creek fault generally follows the master fault logic tree (Figure 2.6-34), with the exception of the recurrence portion of the tree (Figure 2.6-39, Figure 2.6-40a, Figure 2.6-40b, and Figure 2.6-41).

From the south, the Rock Creek fault lies east of and parallels the south-flowing Rock Creek drainage, crosses northward into the Sublette Creek and Pine Creek drainages, and then crosses over another drainage divide into the West Fork drainage in the Tunp Range. The fault is not associated with a range front unlike many other normal faults in the ISB. For most of its length, the fault traverses bedrock terrain in the hanging wall of the Tunp Thrust, a west-dipping Sevier fold and thrust belt structure (Figure 2.6-38).

In the U.S. Geological Survey (USGS) QFFD (Reference 2.6-48), the Rock Creek fault (#729) is characterized as a 41 km long, Class A, high-angle, down-to-the-west, normal fault (Reference 2.6-49). The slip rate category of 0.2 to 1.0 millimeter per yearr was assigned to the fault. Two Holocene paleoearthquakes were identified at the Cook Canyon trench site at 3.6 ± 0.3 thousands of years before present (ka) and about 4.6 ± 0.2 ka and an absence of events was interpreted for the period 4.6 to 15 ka (Reference 2.6-42; Reference 2.6-50). This suggests recurrence intervals of greater than 3.3 thousands of years (kyr) (open interval since most-recent earthquake [MRE]), 0.6 to 1.5 kyr (closed interval between MRE and penultimate earthquake), and greater than 10 kyr (open interval between penultimate earthquake and 15 ka). Witkind (Reference 2.6-51) inferred historical (less than 100 years) movement on the fault, but this inference is unclear and appears unfounded (Reference 2.6-49). Based on a colluvial wedge thickness of 2.5 meters, McCalpin and Warren (Reference 2.6-50) estimated 4 to 5 meters of displacement in the MRE. McCalpin and Haller (Reference 2.6-49) noted the evidence for an earthquake cluster in the middle Holocene and recognized that the Rock Creek fault “may have considerably faster slip rates over short intervals of geologic time (several thousand years).”

In the NSHM23, the Rock Creek fault is characterized as a 41 km long, west-dipping normal fault (Reference 2.6-52).


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Numerous scarp profiles were developed from LiDAR data along the Rock Creek fault. Figure 2.6-42 shows along-strike distribution of vertical scarp-height measurements for the Rock Creek fault and suggests that the MRE may have ruptured the entire fault length.

The Cook Canyon trench site provides the only paleoseismic trench data for the Rock Creek fault. The alluvial surface in Rock Creek exhibits the largest scarp height (10 meters) observed in alluvial deposits along the entire fault (Figure 2.6-43). The Cook Canyon scarp height of 10 meters (Figure 2.6-44) is very similar to four nearby bedrock scarp heights that range in height from 9.25 to 10 meters (Figure 2.6-43). These bedrock scarp heights strongly suggest that the bedrock has experienced a similar number of events (and cumulative slip) that the Cook Canyon alluvial surface has experienced. The fact that scarp heights in bedrock are not greater than the Cook Canyon alluvial scarp height implies a long period of quiescence prior to the current latest Pleistocene to Holocene earthquake cluster. The time required to erode bedrock scarps through mass wasting, landsliding, and periglacial conditions is difficult to estimate, but would likely require significant time suggesting the period of quiescence may be much greater than the duration of the earthquake cluster.

Rock Creek Fault Logic Tree

The Rock Creek fault is the nearest fault source to the site and the most hazard significant. As such, the Rock Creek fault is characterized in detail and this is reflected in its recurrence logic tree (Figure 2.6-39, Figure 2.6-40a, Figure 2.6-40b, and Figure 2.6-41).

The first three nodes of the recurrence characterization for the Rock Creek fault source (Figure 2.6-39) include “behavior mode”, “type of data constraining rate”, and “recurrence approach or model.” Branch weights and values are shown in more detailed trees (Figure 2.6-40a, Figure 2.6-40b, and Figure 2.6-41). The second node indicates the type of data used to constrain slip rate and illustrates that paleoseismic and geomorphic data were used for the In-Cluster branch. The Out-of-Cluster branch used geodetic, landscape (long-term geomorphology), and the lowest USGS QFFD slip rate bin to estimate rate. The third node indicates the recurrence approaches applied to each of the different types of data in the second node.

The first node describes whether the fault will continue with clustered behavior or if the Holocene cluster has ended. This is the most hazard significant node of the logic tree and is the most important decision in modeling the fault’s recurrence. The weight of 0.8 assigned to the “In-Cluster” branch reflects that the evidence for multiple surface-rupturing earthquakes during the Holocene indicates the Rock Creek fault is currently releasing long-accumulated strain in a clustered mode, and not yet out of a clustered mode. Given the evidence for geologically recent clustered behavior, arguments for why the cluster may now be done and the fault has entered a period of tectonic quiescence were considered, consistent with the low geodetic slip rate and geomorphic evidence that the high Holocene slip rate is not characteristic of the apparently slower, longer-term late Pleistocene average rate. The primary consideration is that steady Pleistocene strain accumulation at the geodetic rate implies quiescent periods of 100,000 years or longer to accumulate enough slip to release in the observed Holocene cluster. Continental glacial cycles modulate strain, it is possible that strain in excess of the observed 11 meters of Holocene slip was stored on the fault since the previous episode of clustered strain release.


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Given the uncertainty in the processes modulating clustered behavior on the fault, and thus uncertainty in the maximum accumulated strain available to be released in the Holocene cluster, it was concluded that the observed Holocene behavior will likely continue. Figure 2.6-40a and Figure 2.6-40b illustrate the recurrence characterization of the In-Cluster behavior.

One of the challenges in modeling earthquake recurrence on the three clustered faults is that the paleoseismic data are limited to such a brief record of only two or three events in the mid- to late-Holocene. The In-Cluster branch includes different recurrence approaches, observation periods, and geologic offsets to capture the center, body, and range of the technically defensible interpretations (Figure 2.6-39). The second node indicates that the data used to constrain rates are from paleoseismic and geomorphic information. The third node indicates whether the recurrence approach used recurrence intervals or slip rates. The slip rate branch is weighted 0.8 because this branch included four different geologic offsets or models to constrain rates, and spans not only the Holocene but late Pleistocene. The recurrence interval branch is weighted 0.2 (Figure 2.6-39).

The observation period node on the slip rate branch favored the Holocene 0.6 over the late Pleistocene 0.4. Although more heavily weighing slip rates developed for longer time periods is preferred, a slightly higher weight was assigned to the Holocene branch to reflect a higher confidence in age constraints of the deposits used to develop Holocene rates.

On the fifth node (Figure 2.6-40a), four slip rates are developed with two each spanning the Holocene and late Pleistocene. An equal weight of 0.5 was assigned to the “offset marsh deposits” and “slip from trench reconstructions” on the Holocene branch. On the late Pleistocene branch, rates derived from “estimated minimum bedrock offset” were given a slightly higher weight of 0.6 than the 0.4 weight assigned to the rates derived from “scarp profile” given that age estimates for this branch are considered not as well constrained.

Both geologic offset distributions on the Holocene branch are assigned weights of 0.2, 0.6, and 0.2 (Figure 2.6-40a). On the latest Pleistocene branch, the weights of minimum bedrock offset slip rates are symmetric, but flatter to reflect a lower confidence in the central value 0.3, 0.4, and 0.3. The scarp profile slip rate weights of 0.1, 0.5, and 0.4 reflect the lower confidence in the low rate given that scarp height typically underestimates fault slip.

The end-branch slip rate values for In-Cluster behavior range from 0.7 to 1.7 millimeter per year (Figure 2.6-40b). Weighted end branch slip rate distributions for the Rock Creek fault are plotted (Figure 2.6-45) with the In-Cluster rates shown in red and the Out-of-Cluster rates shown in blue.

The recurrence interval branch (Figure 2.6-40b) includes epistemic uncertainty in the number of earthquakes documented in the trench at Cook Canyon and alternative on time-independent (Poisson) or time-dependent Brownian Passage Time models.

The Out-of-Cluster recurrence assessment (Figure 2.6-41) represents much lower rates than the In-Cluster branch (Figure 2.6-40a and Figure 2.6-40b). Out-of-Cluster slip rate estimates range from 0.001 to 0.39 millimeter per year. The second node on the Out-of-Cluster branch includes three types of data used in constraining slip rate estimates. The data types and weights include geodetic [0.7], landscape [0.1], and USGS less than 0.2 millimeter per year slip rate bin [0.2]. The geodetic branch was assigned the greatest rate, as it was assessed to have higher-quality


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estimates of the long-term out-of-cluster rate even though the geodetic rates are based on measurements over decadal time scales. The landscape branch attempts to quantify the long-term rate by assuming maximum scarp height in bedrock of 15 meters (larger than any observed along Rock Creek or Greys River faults) over assumed Pleistocene time intervals of 50 kyr, 120 kyr, and 240 kyr. The lowest weight was applied to the landscape branch, given the loosely constrained age estimates. The USGS slip rate bin of less than 0.2 millimeter per year was modeled using the NSHM23 Distribution 5 slip rate estimate applied to most of low slip rate faults in the site region that had no independent slip rate information. A 3-point distribution of 0.005 [0.35], 0.05 [0.6], and 0.19 millimeter per year [0.05] represented the continuous Distribution 5 (Reference 2.6-52) shown in Figure 2.6-41.

**Bear River Fault**

The logic tree for the Bear River fault is shown in Figure 2.6-46.

**Greys River Fault**

Maps and logic tree for the Greys River fault are shown in Figure 2.6-47 and Figure 2.6-48, respectively.

**Fault Sources in the Wyoming Salient Region**

Fault sources in the Wyoming Salient region are illustrated in Figure 2.6-49. The following fault sources are presented in order of increasing distance from the site:

- Spring Creek – North Bridger Creek fault
- Sublette Flat fault (North and South)
- Rykman Canyon – The Pinnacle fault
- Hogsback fault
- Eastern Bear Valley fault (North and South)
- Crawford Mountains – Saleratus Creek fault
- Whitney Canyon fault
- Porcupine Mountains fault
- Eastern and Western Bear Lake faults
- Bear River Range North and South faults
- Grand Valley fault (Star Valley, Grand Valley, Swan Valley sections)

The East Cache, West Cache, East Gem Valley, and East Dayton – Oxford faults exhibited insignificant contributions to seismic hazard at the site (<1%).


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### Fault Sources in the Teton Region

Fault sources in the Teton region are illustrated in Figure 2.6-50. Fault sources are presented in order of increasing distance from the site.

- Hoback fault
- Teton fault
- Upper Yellowstone Valley (West and East) faults
- East Mount Sheridan fault
- Eagle Bay–Buffalo Fork fault

The Hoback and Upper Yellowstone Valley (West and East) faults exhibit insignificant contributions to hazard at the site.

### Fault Sources in the Central Wyoming Region

Fault sources in the central Wyoming region are illustrated in Figure 2.6-51. Fault sources are presented in order of increasing distance from the site:

- Continental fault
- Leckie fault
- Flattop fault
- South Granite Mountains fault
- Chicken Springs fault
- North Granite Mountains fault
- Stagner Creek fault
- Cedar Ridge fault
- Saratoga (West and East) faults

### Wasatch Fault System Sources

For use in the PSHA, the characterization of the Wasatch Fault system with some simplifications was adopted from the WGUEP (Reference 2.6-28). Logic trees for the Wasatch sources are shown in Figure 2.6-52, Figure 2.6-53, Figure 2.6-54, and Figure 2.6-55. In the WGUEP characterization, the central segments (shown in red on Figure 2.6-56) are allowed to rupture in complex combinations; the northern and southern segments (blue on Figure 2.6-56) rupture in simple scenarios.

The WGUEP characterization of the Wasatch Fault system is a consensus model. Based on the model and literature and given its relatively low contributions to hazard at the site, the WGUEP characterization of the Wasatch fault system was used without updates in this project.


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The implementation of the WGUEP characterization included the following simplifications:

- No listric fault geometry
- No synchronous rupture of the Salt Lake City segment and the West Valley fault
- Only one Salt Lake City downtown fault geometry model (“Model A”)

**Distant Fault Sources (Cascadia and New Madrid)**

The SSC model includes distant sources of large earthquakes from the New Madrid seismic zone and the Cascadia subduction interface that can contribute to long-period hazard at the site.

New Madrid Seismic Zone

The New Madid seismic source produced the 1811 to 1812 earthquake sequence, which includes the three largest historical earthquakes in the central and eastern U.S. The New Madrid seismic zone is located approximately 1,900 km from the site (Figure 2.6-23).

For use in the PSHA, the characterization of the New Madrid repeated, large-magnitude earthquake source from the SSHAC Level 3 CEUS SSC model (Reference 2.6-53) is adopted. This model represents Earthquake Characteristic Magnitude ($M_{CHAR}$) events of **M** 6.7 to 7.9 with in-cluster [0.9] high recurrence rate from paleoliquefaction data, and out-of-cluster [0.1] low rates.

Cascadia Subduction Interface

The Cascadia subduction interface is located approximately 1,000 km from the site (Figure 2.6-23). For use in the PSHA, the characterization of the Cascadia subduction interface source from the INL SSHAC Level 3 PSHA (Reference 2.6-27; Figure 2.6-57) is adopted. This model represents mega-earthquake ruptures on the interface between the overriding North America and subducting Juan de Fuca and Gorda plates.

### 2.6.1.6 Conclusions

Section 2.6.1 describes the geology, structure, and temporal relations of the site region, site vicinity, and site area for the Kemmerer Unit 1 site. These characteristics help define the areal and fault sources used in the SSC model. Results of the SSC model are used in conjunction with the GMC model to calculate the PSHA for use at the site as described in Section 2.6.2.

## 2.6.2 Vibratory Ground Motion

This section describes the methods used to develop the GMC model, PSHA, and site-specific seismic hazard and presents the results for use in the design and evaluation of the seismic effects on safety-significant structures, systems, and components at the Kemmerer Unit 1 site.

The GMC model was developed using the SSHAC Level 3 process in accordance with the guidance in NUREG-2213 (Reference 2.6-68) for the site region and is documented in the detailed SSHAC Level 3 Study report as described in Section 2.6. A summary of the development of the GMC model, including evaluation of the data, models, and methods, as well


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as the estimates of uncertainty relevant to the region, is provided in Section 2.6.2.1. The GMC model is consistent with the range of earthquake scenarios specified in the SSC model described in Section 2.6.1.5. The GMC model provides input to the PSHA.

Section 2.6.2.2 summarizes the site-specific PSHA calculations and provides the results at the top of a generic rock profile (the “reference rock profile” with time-averaged shear-wave velocity in the upper 30 meters ($V_{s30}$) of 2,500 feet [ft] per second [s] [ft/s]) that is consistent with the GMC model for crustal earthquakes described in Section 2.6.2.1. The calculations of hazard for the PSHA control point (located at a depth of 130 feet below the ground surface elevation at the nuclear island of 6756.0 feet NAVD 88]), including vertical motions, are summarized in Section 2.6.2.3. and incorporate site response calculations between the top of the reference rock profile and the position within the rock portion of the site profile, that is, the PSHA control point.

Another set of site-specific site response calculations, which consider the softer materials above the PSHA control point, are performed to convert the motions from the PSHA control point to the GMRS horizon, located at a depth of 15 feet below the ground surface. The latter results are used to develop the GMRS and the corresponding SSE for the site. The GMRS and SSE are presented in Section 2.6.2.4 and a concluding statement is presented in Section 2.6.2.5.

### 2.6.2.1 Ground Motion Characterization Model Overview

Consistent with the SSHAC methodology provided in NUREG-2213 (Reference 2.6-68), the GMC model captures the center, body, and range of technically defensible interpretations (center body and range of the technically defensible interpretations) for ground motions at the site including all epistemic uncertainty in the median ground motion and the associated aleatory uncertainty. The GMC model consists of three separate models, all of which predict motions at the top of the same reference rock profile [$V_{s30}$ 2,500 ft/s]. The first model considers ground motions from crustal earthquakes in the faults and area sources within 320 kilometers (km) of the site. The second and third models consider motions from the Cascadia subduction zone and the New Madrid region. Three separate ground motion models are required because these three kinds of earthquakes have substantially different source and path characteristics. Epistemic uncertainty in reference rock ground motions is captured by constructing a logic tree for each one of these models. The structure and weights in these logic trees is informed by review of available models, input from experts, and analysis of recordings described in the ground motion database, and discussions as part of the SSHAC process.

The GMC model for crustal earthquakes is developed using the “backbone” approach, which considers the adjustments of a Ground-Motion Prediction Model (GMPM) from a “host” region to a “target” region, and the epistemic uncertainty in those adjustments (Reference 2.6-62 and Reference 2.6-63). The adjustments are developed by comparing regional ground-motion data to the predictions of the backbone model. The crustal GMC logic tree is populated by modified versions of a single GMPM. Bommer and Stafford (Reference 2.6-64) identified the Next-Generation Attenuation Relationships for Western U.S. Enhancement model of Chiou and Youngs (Reference 2.6-65) as the most adaptable GMPM for shallow crustal seismicity because its formulation is most consistent with seismological theory. The Chiou and Youngs model (Reference 2.6-65) was chosen as the backbone model for crustal earthquakes in the region


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around the site. In the final logic tree for crustal earthquakes, there are different adjustments for the western and eastern portions of the region, which regionally also exhibit different source and path characteristics.

For subduction and New Madrid earthquakes, which are treated using modifications to existing models from the INL SSHAC Level 3 PSHA (Reference 2.6-66) and the Next Generation Attenuation Relationships for Central and Eastern North America SSHAC Level 3 study (Reference 2.6-76), respectively, the model functional forms are modified to obtain better agreement with available recordings at the distances of interest. For the subduction zone model, the associated GMC logic tree is also expanded, adding another node or level to the tree. This node controls the extent of the modification to the existing models.

The GMC model is consistent with the range of hazard-significant earthquake magnitudes and distances included in the SSC model as summarized in Section 2.6.1.5, and how they are parameterized. The model for reference-rock motions and the site response model are also consistent.

### 2.6.2.2 PSHA Calculations for Reference Profile

The PSHA calculations are divided into three main steps.

- The calculation of seismic hazard at the top of the reference rock profile using as inputs the SSC and GMC models (Section 2.6.2.2).
- Calculation of SAFs for the site, considering the PSHA control point located at a depth of 130 feet, using as inputs the reference rock hazard results and the properties of the soil profile at the site (Section 2.6.2.3).
- Calculation of PSHA control point hazard results by convolving the reference rock hazard curves with the amplification factors (Section 2.6.2.3).

Addressing the first bullet, this section summarizes the SSC and GMC inputs and presents the results in the form of mean hazard, hazard fractiles, uniform hazard spectra, and deaggregation.

#### 2.6.2.2.1 Seismic Source Characteristics Inputs

SSC inputs for the reference-rock PSHA calculations include seismic source zones and faults sources in the study region. They also include distant sources in the New Madrid Fault Source (NMFS) and Cascadia, which can generate very large earthquakes and affect the hazard for long structural periods.

Parameters for source zones include the source geometry (i.e., coordinates of the boundary polygon), behavior of ruptures at the boundary (leaky or strict), seismogenic thickness and top of ruptures, style of faulting, strike of ruptures, dip and dip direction of ruptures, maximum earthquake magnitude, and earthquake recurrence parameters (rate and *b*-value, which are specified separately for each “cell” comprising the source zone). Earthquakes within source zones are modeled by software using virtual ruptures in three dimensions, so that the rupture distance and other rupture characteristics required by the GMC can be calculated for each earthquake.


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As discussed in Section 2.6.1.5, parameters for faults include the fault surface coordinates, seismogenic probability, fault style, fault dip, seismogenic thickness, rupture model (i.e., possibility of segmentation), rupture length, characteristic magnitude, earthquake recurrence model, and slip rate or recurrence interval.

The models for sources which are distant and well investigated were adopted from recent studies, as discussed in Section 2.6.1.5. The model for the Wasatch fault was adopted from Working Group on Utah Earthquake Probabilities (Reference 2.6-67). The model for New Madrid source was adopted from the SSHAC Level 3 Central and Eastern United States SSC model (Reference 2.6-79). The model for the Cascadia subduction interface is adopted from the INL SSHAC Level 3 PSHA (Reference 2.6-66).

As discussed in Section 2.6.1.5, each source is modeled according to alternative interpretations and conceptual models. Alternatives are included by using local and global logic trees that represent the range of defensible interpretations, their relative weights, and any correlation. Source characteristics (e.g., multiple fault orientations, dips, and seismogenic depths) are explicitly modeled without simplification.

#### 2.6.2.2.2 Ground Motion Characterization Inputs

The GMC model predicts pseudo-spectral acceleration for a common reference rock condition [$V_{s30}$ 2,500 ft/s] that corresponds to the host-region of the crustal backbone GMC. Each individual model consists of an expression for the median pseudo-spectral acceleration as a function of magnitude, distance, and other rupture characteristics, and a distribution of aleatory residuals, which are used to calculate the probability that the ground motion amplitude from an earthquake of given magnitude, distance, and other characteristics exceeds any amplitude of interest. Partially non-ergodic aleatory variability models (i.e., using single-station sigma) are used. This is consistent with subsequent site response analysis at the PSHA control point. This model consists of three different GMC logic trees: one for shallow crustal seismic sources, one for the New Madrid fault zone, and one for subduction interface sources (Section 2.6.1.5).

#### 2.6.2.2.3 PSHA Results for Reference Rock

Mean Hazard

Mean hazard curves for the 20 spectral periods considered are provided in Figure 2.6-58 and Table 2.6-4. The curves in the tables are interpolated to a common set of amplitudes. Some curves do not reach the smallest of these common amplitudes, and others fall well below 1E-9 per year by the largest amplitudes; these entries are populated in Table 2.6-4 with a dash mark (-).

Figure 2.6-59, Figure 2.6-60, Figure 2.6-61, and Figure 2.6-62 plot total mean hazard as well as the sum of hazard for periods of 0.01, 0.1, 1 and 10 seconds from 5 different source types:

- “Faults,” which includes Groups 1-4, Layered, and Clustered faults from the SSC model;
- “Background,” which includes seismic source zones from the SSC model;
- “Wasatch,” which includes the faults in the SSC model from the Wasatch fault system;


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- "NMFS," which includes the faults in the SSC model from the New Madrid Fault System; and
- "Interface," which includes the megathrust interface faults in the SSC model from the Cascadia subduction zone.

These plots provide a more aggregate view of how the different source types contribute to hazard across a broader range of periods. The contributions of these types are similar at lower periods (higher frequencies) of 0.01 and 0.1 seconds: hazard is dominated by the closer faults and background sources, and neither Wasatch, NMFS, nor interface sources have a significant contribution. The contributions of those more distant sources (Wasatch, NMFS, and interface) increase as the spectral period increases. For 1 second hazard they are still less significant than the closer faults and background sources, but for 10 second hazard both NMFS and Wasatch contribute more than background sources to 1E-4 per year hazard. This is because those distant sources can generate large magnitude earthquakes, combined with the relative GMC attenuation characteristics of long-period ground motions that allow them to maintain higher amplitudes over greater distance compared to short-period ground motions.

Hazard Fractiles

Figure 2.6-63, Figure 2.6-64, Figure 2.6-65, and Figure 2.6-66 and Table 2.6-5, Table 2.6-6, Table 2.6-7, and Table 2.6-8 show fractiles of total hazard for periods of 0.01, 0.1, 1 and 10 seconds. For ground motion motions corresponding to total hazard of 1E-4, the range between the 0.15 fractile and 0.85 fractile is about a factor of 10 in annual frequency for shorter spectral periods, but is much greater for 10 seconds. This difference represents a range of approximately ±1 standard deviation of uncertainty in the PSHA (for instance, uncertainty from alternative ground motion models or maximum magnitudes). These ranges are also typical of a reference-rock PSHA performed in the United States.

Uniform Hazard Response Spectra

Figure 2.6-67 and Table 2.6-9 present mean Uniform Hazard Response Spectra (UHRS) for 11 Annual Frequency of Exceedance (AFE) interpolated from mean hazard curves. Mean UHRS ordinates are reported at all 20 spectral periods provided by the GMC. The peak ground acceleration (PGA), analogous to 0.01 seconds in this calculation, is 0.221g (force of gravity) at the 1E-4 per year AFE, 0.507g at 1E-5 per year, and 1.07g at 1E-6 per year.

Deaggregation

Table 2.6-10 contains mean deaggregated reference-rock magnitude (M), distance in km (R), and the number of standard deviations above or below the median ground motion for a given scenario (ε) for periods of 0.01, 0.1, 1 and 10 seconds at mean ground motion amplitudes corresponding to AFEs of the UHRS in Table 2.6-9. For a given AFE, shorter period ground motions are controlled by smaller, closer earthquakes than longer period ground motions, especially for smaller AFEs (larger ground motions). This reflects the increasing importance of distant sources capable of generating large events for longer periods and lower ground motions.


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Bold values in Table 2.6-10 for low-frequency (LF) (1 second) deaggregated M and R correspond to calculations considering only distances greater than or equal to 100 km, per RG 1.208. This is only required for the first four AFEs: 1E-2, 4E-3, 1E-3, and 4E-4 per year, and captures the importance of fault sources (Figure 2.6-61) at the edge of the site region. Background sources (Colorado Plateau-Wyoming Craton and Intermountain Seismic Belt, per Table 2.6-11) are the most important sources at large amplitudes, so the RG 1.208 procedure does not screen them out.

Table 2.6-10 provides modal deaggregated M and R for periods of 0.01, 0.1, 1 and 10 seconds at mean ground motion amplitudes corresponding to AFEs of the UHRS in Table 2.6-9. Table 2.6-9 indicates the most likely M, R pair that will result in a particular ground motion. The influence of NMFS on 10 second hazard is clear in Table 2.6-12 with the M, R pair of M7.9 at 1,825 km.

The contributions of M, R pairs to mean hazard for 0.1 second and 1 second at AFEs of 1E-4, 1E-5, and 1E-6 per year are shown in Figure 2.6-68, Figure 2.6-69, Figure 2.6-70, Figure 2.6-71, Figure 2.6-72, and Figure 2.6-73.

### 2.6.2.3 PSHA Calculation at PSHA Control Point

This section summarizes the site response calculations for the PSHA control point, which is located at the top of an generic rock profile (the “reference rock profile” with $V_{s30}$ of 2,500 ft/s) that is consistent with the GMC model for crustal earthquakes described in Section 2.6.2.1, and the convolution of the site response with the reference rock hazard to obtain the hazard at the PSHA control point.

The site response calculations used the One-Step approach in NRC Research Information Letter 2021-15 (Reference 2.6-77) and a site-specific profile. The calculated site adjustment factor (SAF) converts ground motion amplitudes from the top of the reference profile to the PSHA control point in the site-specific profile. Epistemic uncertainty in the site response is captured by using a logic tree that considers alternative shear-wave velocity ($V_S$) profiles and alternative values of other model parameters for the site-specific profile. These alternative $V_S$ profiles are constructed using multiple site-specific measurement techniques for the shallow portion (Figure 2.6-87), and regional geophysical measurements for the deep portion, as described in Section 2.6.1.5. Aleatory uncertainty is captured by generating 60 randomized profiles for each branch of the logic tree.

The site response calculations consider all logic-tree branches and all randomized profiles, using inputs with multiple amplitudes of shaking representing a broad range of AFEs and spectral shapes. All site response calculations utilized an equivalent-linear approach using random vibration theory. The epistemic and aleatory uncertainties in the SAFs from all the site response runs are then condensed into seven SAFs, with their associated weights, and an associated aleatory sigma. Figure 2.6-74 shows the seven SAFs and the aleatory sigma for one input amplitude. For each spectral period, the result of these calculations is seven amplitude-dependent SAFs, with their associated weights, and an amplitude-dependent aleatory sigma. These results are combined with the reference-rock hazard results summarized in Section 2.6.2.2 to obtain the horizontal PSHA control-point hazard.


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The vertical motions are quantified by multiplying the horizontal spectra by the ratio of vertical to horizontal motion (V/H) associated with that hazard level. The selected V/H model is a well established one and is appropriate for the site region. Details are provided in Section 2.6.2.3.2.

#### 2.6.2.3.1 PSHA Control Point Horizontal Hazard

This section describes the combination of reference-rock hazard curves summarized in Section 2.6.2.2 and the SAFs shown in Figure 2.6-74 using Approach 3 of NUREG/CR-6728 (Reference 2.6-75). The result of this calculation are hazard curves and UHRS that are applicable to the PSHA control point (Figure 2.6-87).

Mean horizontal UHRS are developed by interpolating mean hazard curves to 11 AFEs and are calculated for all spectral periods in the PSHA. The PGA, analogous to 0.01 seconds, is 0.157g at the 1E-4 per year AFE, 0.368g at 1E-5 per year, and 0.771g at 1E-6 per year. The UHRS are presented in Figure 2.6-75 and Table 2.6-13.

#### 2.6.2.3.2 PSHA Control Point Vertical Hazard

Vertical UHRS are developed from the V/H ground motion ratios of Gulerce and Abrahamson (Reference 2.6-69) for AFEs of 1E-4, 1E-5, and 1E-6 per year. The logarithmic mean and standard deviation of V/H from Reference 2.6-69 depend on magnitude, rupture distance, style of faulting, $V_{S30}$, and PGA1100 (the PGA associated with the same magnitude-distance-style scenario and $V_{S30}$ equal to 1,100 meters per second.

The magnitude and rupture distance used are the mean values obtained from the reference-rock deaggregation for 10 hertz (Hz) at AFEs of interest (Table 2.6-10). The style of faulting is normal, which corresponds to the seismic sources that dominate 10 Hz hazard for the AFEs of interest.

Figure 2.6-76 illustrates the 1E-4 per year median V/H ratio and its ±1 uncertainty range. The mean is also shown; it is calculated under the assumption of a lognormal distribution. While the median V/H ratio peaks at 0.05 seconds, the relatively large uncertainty at long periods elevates the mean ratio at 10 seconds above the mean value at 0.05 seconds. Figure 2.6-77 shows the mean V/H ratios for AFEs of 1E-4, 1E-5, and 1E-6 per year. Figure 2.6-78 shows the vertical UHRS along with their horizontal counterparts. Table 2.6-14 lists the mean V/H ratios and vertical UHRS.

### 2.6.2.4 Site-Specific Ground Motion Response Spectra and Safe Shutdown Earthquake

This section summarizes the development of the GMRS and SSE and compatible soil property profiles for the site. The GMRS and SSE are calculated at the GMRS horizon, which is located at the top of competent material, 15 feet below the ground surface elevation of 6756.0 feet NAVD 88. This depth corresponds to the uppermost in-situ competent material ($V_s$ greater than 1,000 feet per second).

The design spectra and their corresponding strain-compatible soil property profiles are computed via multiple Probabilistic Site Response Analyses (PSRA), which convert the motions from the PSHA control point to the GMRS horizon. The approach used to develop the amplification factors from PSHA control point to GMRS horizon utilizes an equivalent-linear model and random


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vibrations, and is consistent with Approach 1 in NUREG-CR/6728 (Reference 2.6-75) to develop the hazard-consistent site-specific motions at or near the ground surface. Approach 1 consists of multiplying the PSHA control point spectra by the calculated median amplification factors between the PSHA control point and the GMRS horizon. These amplification factors and their corresponding strain-compatible soil property profiles are computed via multiple PSRAs. Each PSRA combines the results of 600 SRAs considering the epistemic and aleatory uncertainties in the shear wave velocity profiles and nonlinear soil properties. The resulting 1E-4 and 1E-5 UHRS spectra are used to develop the GMRS and corresponding SSE response spectra for the site (Figure 2.6-79).

#### 2.6.2.4.1 Ground Motion Response Spectrum

The GMRS is calculated using the calculation method in ASCE 43-19 (Reference 2.6-78) for Seismic Design Category 5. The horizontal GMRS at PGA is equal to 0.297g. The calculated horizontal GMRS for the site is shown in Figure 2.6-79 and listed in Table 2.6-15. The randomized damping and shear modulus reduction curves in the PSRAs related to the GMRS calculation are shown in Figure 2.6-80. The maximum shear strain profiles, strain-compatible shear wave velocity profiles, and strain-compatible damping ratio profiles obtained from the related PSRAs are shown in Figure 2.6-82, Figure 2.6-83, and Figure 2.6-84, respectively. The strain-compatible shear wave velocity and damping ratio profiles are also listed in Table 2.6-16.

#### 2.6.2.4.2 Safe Shutdown Earthquake

Since the PGA in the calculated horizontal GMRS is greater than 0.1g, and the horizontal GMRS spectrum has a sufficiently smooth and broad spectral shape, the GMRS is selected as the site SSE, consistent with 10 CFR 50, Appendix S and RG 1.208. The V/H ratios applied to the horizontal spectrum to calculate the vertical spectrum are listed in Table 2.6-17 and shown in Figure 2.6-85. They are calculated using the same Gulerce-Abrahamson (Reference 2.6-69) approach used in Section 2.6.2.3.2, using the appropriate $Vs_{30}$ and PGA1100 for the GMRS horizon. The calculated horizontal and vertical acceleration response spectra representing the site SSE are shown in Figure 2.6-86 and summarized in Table 2.6-15.

### 2.6.2.5 Conclusions

Section 2.6.2 summarizes the GMC and the results associated with the PSHA and Site Response Analysis to determine the GMRS and SSE for the site.

## 2.6.3 Surface Deformation

### 2.6.3.1 Regional and Local Non-Seismic Geologic Hazards

Potential non-seismic geologic hazards that can contribute to surface deformation include landsliding, mining or mine collapse, karst, and liquefaction. Each of these hazards presents a unique challenge to development within the site area and are discussed below. Seismic and fault rupture hazard are discussed in Section 2.6.1.5. There is no evidence for active faulting within the site area. A discussion of the Rock Creek fault, which is the only recognized active fault within the site vicinity, is provided in Section 2.6.1.5. Paleoliquefaction is addressed in Section 2.6.3.1.3, although the triggering event for this geological hazard is a seismic event.


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Methodology

Two primary methodologies, geologic field reconnaissance and desktop Light Detection and Ranging (LiDAR) analysis, were utilized to evaluate the potential for surface deformation in the site area. Field reconnaissance consisted of mapping of bedrock and Quaternary geology for the purpose of creating and updating existing geologic mapping. LiDAR analysis consisted of topographic analysis of LiDAR and derivative products to increase understanding of the geology and seismic and non-seismic hazards.

Geologic Field Reconnaissance

The geologic field reconnaissance investigation included compilation of data including previous investigations, publications pertaining to the geologic and tectonic setting of the Kemmerer Unit 1 Site, and best available published or in-press geologic mapping within the site vicinity (40-kilometer [km] radius). Field reconnaissance efforts focused on characterization of the bedrock stratigraphy, structure, geomorphology, and surficial geology of the site area (8-km radius) as well as verification and augmentation of existing geologic mapping in the site vicinity (Figure 2.6-8). Anthropogenic activities (i.e., active and abandoned mines, talus piles, and stockponds) were mapped and evaluated as part of the geologic field reconnaissance. The site location (1-km radius) was mapped in detail as part of the field reconnaissance efforts (Figure 2.6-9).

Geologic field reconnaissance was performed in two phases. The initial phase focused on geologic mapping of bedrock and Quaternary deposits, and characterization of geologic structures in the site vicinity. This phase also included evaluation of select rock cores from previous subsurface drilling conducted at the site. The second phase included continued geologic mapping of bedrock and Quaternary deposits, characterization of geologic structures, as well as field traverses to evaluate potential faults within the site area. This phase also included detailed mapping of the site evaluating the geology and geomorphology with particular focus on non-tectonic hazards such as the potential for landsliding, liquefaction, karst, diapirism, and mine-collapse. The field reconnaissance activities are summarized in the following sections.

Data Sources

The primary data sources used to accomplish the objectives of the geologic field reconnaissance include: (1) previous geological investigations of the site, (2) published and draft geologic mapping of the site area and site vicinity, and (3) high resolution topographic data from LiDAR surveys.

Previous Investigations

An extensive subsurface investigation and laboratory testing program was conducted to obtain information on subsurface materials and site conditions for use in preparation of a subsurface investigation report for the site. Work at the site commenced in November 2021 and was completed in July 2022.


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A total of 160 boreholes were drilled at the site, consisting of 111 primary borings along with 49 offset borings. Based on these 160 boring logs and associated laboratory testing, surficial soils at the site consist of alluvium or colluvium and residual soils. Based on the site subsurface investigations, the alluvium or colluvium is undifferentiated reworked sediments derived from weathering of the underlying residual soils and bedrock units that have been locally transported and deposited at the site through streamflow, slope wash, gravity, and wind or similar geomorphic processes. Residual soils at the site are derived from the in-situ weathering and decomposition of the underlying bedrock formations. The site is entirely underlain at the ground surface by bedrock associated with the Upper Cretaceous Hilliard Shale, a marine shale comprised of claystone, siltstone, sandstone, and bentonite. The Hilliard Shale sampled during drilling consisted predominantly of siltstone, shale, sandy siltstone, interbedded sandstones, and minor thin to moderately bedded bentonitic intervals. The borings, with the exception of boring B-122, were advanced within the Hilliard Shale. Boring B-122 was advanced through the lower Hilliard Shale and into the underlying Dry Hollow Member of the Upper Cretaceous Frontier Formation. The Dry Hollow Member includes nonmarine shale, siltstone, sandstone, and coal units. Drilling cores from the Dry Hollow Member included sandstone, siltstone, shale, carbonaceous shale, and a coal bed with a drilled thickness of approximately 9 m. The stratigraphically conformable Hilliard Shale and Frontier Formation units appear to dip between approximately 10 degrees and 25 degrees in the core samples which matches the regional structural mapping of approximately 20 degrees west.

Geologic Mapping

Comprehensive geologic reconnaissance mapping was performed at progressively greater detail as the location of investigations neared the site. The following zones of investigation are defined in terms of their radial distances from the site based on RG 1.208, Revision 0, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion:”

- Site region—within 320 km – Figure 2.6-3
- Site vicinity—within 40 km – Figure 2.6-5
- Site area—within 8 km – Figure 2.6-8
- Site location—within 1 km – Figure 2.6-9

The site vicinity was investigated through geologic field reconnaissance and compilation of available geologic literature. Figure 2.6-13 shows the tracks that the field reconnaissance team produced, and Figure 2.6-5 shows the final map produced from field reconnaissance and review of existing literature. The best available published geologic maps within the distances listed were compiled into a Geographic Information System geologic mapping database that included digital versions of the geologic maps, LiDAR (see subsequent paragraphs) and other digital topographic data, aerial photography, and geographic data that included roads and municipal facilities. Less than one-fourth of the site vicinity has 1:24,000 scale geologic mapping published by the USGS with publication dates ranging from 1979 to 1992. The 1:62,500 scale USGS map of the Cokeville quadrangle (Reference 2.6-80) and the 1:100,000 scale maps of the Kemmerer, Evanston, and Logan quadrangles of the Wyoming State Geological Survey and Utah Geological Survey (Reference 2.6-81; Reference 2.6-82; Reference 2.6-83) were used to characterize the geology of the areas within the site vicinity lacking 1:24,000 scale mapping.


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The site region generalized geologic map was made from published regional geologic maps with minor editing related to resolving map unit border issues as shown in Figure 2.6-3. In addition to reviewing and compiling geologic maps and literature materials within the geologic mapping database, detailed original geologic mapping and review of Quaternary and bedrock units was performed within the site area, depicted in Figure 2.6-8, and site location shown in Figure 2.6-9, as part of the geologic field reconnaissance.

LiDAR Products

The LiDAR coverage of the 40-km site area is from the USGS 3D Elevation Program Project WY Southwest 2020 D20, WY Southwest 4 2020 Work Unit (Reference 2.6-84). Two data deliverables were used to create four derivative products for use in geologic mapping efforts including digital elevation model, hillshade, slope, and contour maps. The first data deliverable used was the GeoTIFF format hydro-flattened bare earth digital elevation model tiles. The 40-km site area 1-m cell size LiDAR digital elevation model was created from 5,188 1 x 1 km GeoTIFF format tiles. These tiles are hydro-flattened bare earth, which means that the ground surface model will reliably represent hydrologic features (i.e., drainage channels, tributary streams and rivers) and the ground surface does not show elevations above the ground surface (i.e., bushes, trees and buildings). The individual tiles were combined and used to calculate three derivative data products: hillshade, slope, and contours.

The second data deliverable from USGS 3D Elevation Program Project WY Southwest 2020 D20, WY Southwest 4 2020 Work Unit (Reference 2.6-84) used to create data derivatives for map products were LiDAR point tiles. Seventy-two tiles covering approximately a third of the 8-km site area were downloaded and processed to calculate a high resolution hillshade raster. Data were collected between July 4 and August 25, 2020, and consist of high-resolution topographic mapping with 13 centimeter horizontal accuracy and 5 centimeter vertical accuracy. Resolution of the data is 2 points per square meter and 0.71-meter point spacing. LiDAR derived products were used during geologic reconnaissance to map bedrock and surficial geologic units based on their topographic expression.

LiDAR Analysis

Desktop analyses of LiDAR data were performed for select areas within the site region primarily to identify and characterize seismic sources; however, hazards associated with landsliding, mining, karst, liquefaction, and diapirism were also investigated.

#### 2.6.3.1.1 Landslide Hazards

Landsliding in the site vicinity is manifest by the downslope movement of rock or debris (Reference 2.6-85) at varying velocities. Landslides are a result of a number of complex factors including but not limited to bedrock competency and strength, slope aspect, slope gradient, soil depth and type and cohesion, vegetation, drainage patterns, and precipitation. (Reference 2.6-85). Landslides are frequently observed within the site vicinity (Figure 2.6-5) but are nonexistent in the site area (Figure 2.6-8). Landslides in the site vicinity are located mostly to the north and northwest of the site location and include shallow debris slumps to larger rotational and translational bedrock landsliding. These landslide features commonly occur on high, steep slopes where weaker Eocene sedimentary rocks, further described in the geologic field


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reconnaissance report, often form ridges atop uplifted Cretaceous bedrock. These Eocene siltstones and mudstones are more susceptible to landsliding due to their weaker rock strength and the presence of discontinuities within the steep slopes they overlay.

In contrast, the site area is composed almost entirely of more competent Cretaceous rocks. Geomorphologically, the site area is located in the low-lying Cumberland Flats valley (Figure 2.6-8). Oyster Ridge to the east of the site area contains the steepest slopes in the site area. These slopes are underlain by the Oyster Ridge sandstone of the Frontier Formation, a highly competent ridge-forming sandstone. No landsliding was mapped along Oyster Ridge; however, small alluvial fans were mapped near the base of slopes that are the result of shallow colluvial and alluvial processes (Figure 2.6-9). No evidence of debris flows or long runout debris slides such as lateral levees, channel avulsion, or lobate or hummocky topography was identified.

While there are numerous landslides in the site vicinity, there are no significant landslides in the site area. Topographic, geologic, and geomorphic conditions that create landslides in the site vicinity are also absent in the site area. As such, the potential landslide hazard to the site area is very low.

#### 2.6.3.1.2 Anthropogenic and Mining Hazards

The bedrock geologic units in the site vicinity are often coal rich and there are many coal mines in the area, both active and abandoned. Most notably, the active open pit Kemmerer Coal Mine is located approximately 5 km west across the Cumberland Flats from the site (Figure 2.6-8). In addition, there are numerous abandoned coal mines, including several located in the site area that do not extend to depth. Many abandoned coal mines in the Kemmerer region are located in the coal rich Dry Hollow member of the Frontier Formation, including an abandoned mine just outside the eastern border of the site location (Figure 2.6-8 and Figure 2.6-9).

The closest abandoned mine to the site is located approximately 1 km east of the site location on the western margin of Oyster Ridge. This mine appears to be relatively small with minimal talus debris, on the order of a few cubic meters, and a limited areal extent of a few tens of meters. This relatively small mine is typical of the abandoned mines identified in the site area. These mines are not within 1 km of, and do not underlie, the site. Subsurface drilling of the site vicinity did not find evidence of mine activity extending under the site and investigation of nearby mining activity indicates that there are no mine shafts that extend under the site. No evidence for anthropogenic activities in the site area that would present any hazard to the site, such as mine collapse, were found.

#### 2.6.3.1.3 Karst and Dissolution Hazards

Karst is a terrain with distinctive hydrology and morphology created through the dissolution of soluble rock (Reference 2.6-86). The process of dissolution, acting over long periods of time, creates an underground drainage system and associated karst landscape features. The list of karst landscape features is extensive and the characteristics of these features vary with the geologic, hydrologic, tectonic, and climatic setting. Sinkholes, dolines, caves, tower karst, cone karst, rillenkarren, blind valleys, springs, underground drainage, and sinking streams are described from karst regions around the globe. The fundamental feature common to karst areas


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is the presence of an underground drainage system created through the dissolution of soluble rock. Geologic hazards in karst terrains can include ground subsidence or collapse, rapid underground drainage, irregular soil-bedrock contact, and deep weathering along joints. Non-tectonic surface faulting can be caused by collapse into subsurface dissolution features.

There is no evidence to support any of the described types of karst in the site vicinity. There is only a small amount of limestone in the site vicinity (Figure 2.6-5), all associated with the Jurassic-Triassic undivided unit. The closest limestone is a variety of small outcrops of the Jurassic Twin Creek Limestone located 10–15 km from the site. No sinkholes, caves, sinking streams, or other karst features were observed either in the field or on LiDAR.

#### 2.6.3.1.4 Paleoliquefaction

Liquefaction occurs when saturated sediment loses its strength due to the ground shaking induced by an earthquake. This causes an increase in pore-water pressure that causes the sediment to liquefy and flow towards the surface. Liquefaction results in characteristic sedimentary structures, such as sand blows, dikes, and sills (Reference 2.6-87). No evidence of paleoliquefaction or sedimentary structures caused by liquefaction in the site vicinity was found during field reconnaissance or on LiDAR. In addition, the site is underlain by fine-grained shale bedrock and non-susceptible soils which are planned to be excavated. The potential for liquefaction at the site is addressed in Section 2.6.4.8.

#### 2.6.3.1.5 Diapirism

Diapirism occurs when salt or mud begins to pierce its overburden, often producing normal faulting as overburden thins above the propagating diapir (Reference 2.6-88). While there is normal faulting in the site vicinity, fault patterns do not match those modeled by Reference 2.6-88 and the site is not located in a tectonic setting that commonly has diapirs, such as a passive continental margin. No evidence of diapirism in the site vicinity was found and any normal faulting is considered to be tectonic in origin.

#### 2.6.3.1.6 Oil and Gas Wells

Within the site area there are two active vertical gas wells, one is located about 5 km southwest of the site and the other is located 8 km to the west of the site (Reference 2.6-89). There are also five abandoned vertical gas wells and one abandoned vertical oil well located within the site area (Reference 2.6-89 and Reference 2.6-90). The closest abandoned well, a gas well, is located about 4 km southeast of the site (Reference 2.6-89).

#### 2.6.3.1.7 Fracking and Waste Water Wells

No evidence for fracking or waste water injection wells were noted during geologic field reconnaissance. According to the Wyoming Oil and Gas Commission and the Wyoming Department of Environmental Quality Underground Injection Control, there are no fracking or waste water wells in the site area (Reference 2.6-89 and Reference 2.6-91).


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### 2.6.3.2 Conclusions

In summary, based on evaluations conducted, the threat of non-seismic geologic hazards at the site is negligible and the site does not have the potential for surface deformation.

## 2.6.4 Stability of Subsurface Materials and Foundations

This section presents information on the stability of subsurface materials and foundations at the Kemmerer Unit 1 Site, hereafter referred to as the site. The geological, geophysical and geotechnical information presented is used as the basis to evaluate the stability of subsurface materials and the stability of foundations for SR structures at the site.

Information presented in this section is developed from a subsurface investigation performed between December 2021 and July 2022 (see Section 2.6.4.3). The subsurface investigation was performed over a substantial portion of the site but predominantly within the footprint of the Nuclear Island (NI) and Energy Island (EI) areas.

Note that the elevation datum referenced herein is the North American Vertical Datum of 1988, Geoid 18. The horizontal datum for all exploratory point locations is State Plane North American Datum of 1983, Wyoming West Zone. All references to depth are below the existing ground surface unless stated otherwise.

This section addresses the following topics:

- Section 2.6.4.1 - Geologic Features
- Section 2.6.4.2 - Properties of Subsurface Materials
- Section 2.6.4.3 - Foundation Interfaces
- Section 2.6.4.4 - Geophysical Surveys
- Section 2.6.4.5 - Excavation and Backfill
- Section 2.6.4.6 - Groundwater Conditions
- Section 2.6.4.7 - Response of Soil and Rock to Dynamic Loading
- Section 2.6.4.8 - Liquefaction Potential
- Section 2.6.4.9 - Earthquake Design Bases
- Section 2.6.4.10 - Foundation Stability
- Section 2.6.4.11 - Design Criteria
- Section 2.6.4.12 - Techniques to Improve Subsurface Conditions

### 2.6.4.1 Geologic Features

This section provides a summary of the geologic features at the site. More detailed descriptions of the geologic conditions at the site and in the vicinity are presented in Section 2.6.1. The information is taken from the subsurface investigation program performed at the site (discussed in Section 2.6.4.2.1) and a geologic reconnaissance of the site and site vicinity (discussed in Section 2.6.3).


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This section addresses the following topics:

- Section 2.6.4.1.1 - Geologic Setting
- Section 2.6.4.1.2 - Site Geologic Features

### 2.6.4.1.1 Geologic Setting

The site is located in southwest Wyoming, in the Middle Rocky Mountain physiographic province. Features that characterize this province include mountainous terrain, stream valleys, alluvial basins, and sharp ridge lines. In the vicinity of the site, ridges are underlain by more resistant sandstone beds and low-relief areas are commonly underlain by more easily erodible shale formations.

The bedrock geology of the site area is predominantly composed of Cretaceous sedimentary rocks. These were deposited in the foreland basin of the Sevier Orogeny and were subsequently deformed in a thin-skinned fold and thrust belt when the Sevier Orogeny propagated eastward. The site area has been tectonically quiescent since the cessation of Sevier deformation in the early Cenozoic, with the exception of a post-Eocene period of normal faulting that produced a horst and graben structure.

The bedrock stratigraphy exposed at the site area (Figure 2.6-88) consists of mainly Cretaceous age sedimentary rocks deposited during Sevier thrusting, sedimentation, and Western Interior Seaway transgression and regression. These sedimentary rocks consist of the Hilliard Shale underlain by the Frontier Formation. The bedrock structure is characterized by west dipping bedding. At the site, strata of the Hilliard Shale generally strike about N7°E with dips of approximately 20 degrees. Figure 2.6-89 illustrates a geologic section through the Cumberland Flats, including the site location, showing the underlying bedrock formations and west dipping bedding. Figure 2.6-90 provides a stratigraphic column through these formations.

The Hilliard Shale consists of dark olive gray marine shale, siltstone, and sandy shale with thin interbeds of limestone and tan to light gray sandstone, especially in the upper part of the unit. The Hilliard Shale sampled during the recent subsurface exploration consisted predominantly of siltstone, sandy siltstone, interbedded sandstones, and minor thin to moderately bedded bentonitic intervals. The base of the Hillard Shale was encountered at an approximate depth of 1,255 feet in boring B-122, equivalent to an elevation of 5,501.5 feet. The Hilliard Shale is subdivided, from youngest to oldest, into the Hinshaw Member, Shurtleff Sandstone Member, and the Conglomerate of Little Muddy Creek.

The Frontier Formation consists of an interbedded marine and nonmarine sequence of sandstone, siltstone, and carbonaceous shale with interbeds of coal and, locally, porcellanite and conglomerate. It is subdivided, from youngest to oldest, into the Dry Hollow Member, Oyster Ridge Sandstone Member, Allen Hollow Member, Coalville Member, and Chalk Creek Member. Boring B-122 at the site was advanced into the underlying Dry Hollow Member. Drilled cores from the Dry Hollow Member included sandstone, siltstone, shale, carbonaceous shale, and a coal bed with a drilled thickness of approximately 28 feet.

Detailed descriptions of the geologic history, structure, and stratigraphy are provided in Section 2.6.1.


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#### 2.6.4.1.2 Site Geologic Features

The site is located within the Cumberland Flats (Figure 2.6-89) which is a broad, relatively flat-lying, north-south trending valley produced by weathering of the underlying Hilliard Shale. The topography of the Cumberland Flats valley contains long, roughly north-south trending resistant ridges associated with coarser grained, more indurated facies within the Hilliard Shale. The topography of the site includes a north/south oriented ridge in the central portion of the site with a north/south oriented drainage way to the west and the floodplain of the North Fork Little Muddy Creek to the east. Existing site elevations along the ridge range from 6,750 to 6,760 feet and from 6,730 to 6,740 feet in the drainage way. The finished plant grade elevation in the NI is set at 6,757 feet. The finished plant grade elevation in the EI, immediately south of the NI, is set at 6,754 feet. The greatest foundation embedment in the NI is for the Reactor Building (RXB) at an approximate depth of 118 feet below finished grade, at elevation 6,639 feet.

The recent geologic mapping verified that the Hilliard Shale exclusively underlies the site and confirmed the previous mapping of Quaternary surficial units as well as Holocene and Pleistocene gravel deposits. The Quaternary surficial deposits at the site consist primarily of Holocene to Upper Pleistocene alluvial deposits within the stream valley and tributaries of the North Fork of Muddy Creek. Alluvial deposits include channel-fill and floodplain deposits as well as some alluvial fan, terrace deposits, valley side colluvium or talus, and sediments deposited in deltas. The unit is composed of clay, silt, and sand, with gravel present in varying amounts depending on the local bedrock parent material.

Two previously mapped gravel deposits are present within the site area (Figure 2.6-88). These Holocene and Pleistocene gravels consist of an unconsolidated, well-rounded, gravel-cobble-boulder pavement deposit commonly found on pediment surfaces and frequently located on prominent topographically high benches. This unit includes primarily gravel-cobble-boulder detrital and lag deposits, commonly composed of quartzite and agate, from erosion of nearby conglomerate-rich bedrock formations. It typically consists of larger cobbles and boulders with some calcareous varnish.

The recent geologic mapping extended the mapping of gravel along topographic highs and included gravel, bedrock, colluvium, and alluvial fan material along the slope of the eastern ridge. The extent of alluvial materials and Holocene active channel materials was also mapped within topographic lows. In addition, anthropogenic features such as mine tailings, stockponds, and historic railroad tracks were mapped as artificial fill. Details of site geologic features and observations of recent geologic mapping are provided in Section 2.6.1 and Section 2.6.3.

### 2.6.4.2 Properties of Subsurface Materials

This section describes the static and dynamic engineering properties of the subsurface materials encountered at the site and addresses the following topics:

- Section 2.6.4.2.1 - Subsurface Investigation Program
- Section 2.6.4.2.2 - Laboratory Test Program
- Section 2.6.4.2.3 through Section 2.6.4.2.10 - Engineering Properties


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#### 2.6.4.2.1 Subsurface Investigation Program

The site is undeveloped and partially vegetated with sagebrush and saltbrush. No subsurface data were available previously for the site. Subsurface data from the nearby Naughton Coal Power Plant and publicly available data from regional exploratory wells were reviewed to aid in developing the site-specific subsurface investigation program. The data from the Naughton Coal Power Plant were taken at shallow depths and used for the design of plant structures. The regional exploratory well data were taken at deeper depths, on the order of 5,000 feet. These data were used for natural resource exploration. Published data included sonic velocity logs.

Two approximately 100-foot deep borings with soil sampling and rock coring were initially drilled at the site. Data from these borings, along with the results of laboratory testing conducted on selected soil and rock samples from the borings, were used to establish the subsurface investigation program. This program included soil borings, rock coring, downhole and surface geophysical testing, test pit explorations, in-situ pressuremeter testing, groundwater wells, and in-situ permeability testing. Details of the subsurface investigation are described in Section 2.6.4.3. A summary of the field work performed for this investigation is presented in Table 2.6-18. Details and results of this work are documented in a subsurface investigation data report. The development and execution of the subsurface investigation program was performed in accordance with guidance presented in RG 1.132. The program was conducted in accordance with applicable codes and standards as identified in RG 1.132 and ASME NQA-1 requirements as described in Section 8.1 to meet 10 CFR 50.55a.

#### 2.6.4.2.2 Laboratory Testing Program

A laboratory testing program was conducted on soil and rock specimens obtained from the subsurface investigation. The tests performed on the soil specimens focused primarily on obtaining the index properties (grain size, natural moisture content, specific gravity, and plasticity) and the shear strength, settlement, and compaction characteristics. The tests performed on the rock core specimens focused on obtaining the index properties (unit weight and specific gravity), compressive strength, shear and elastic moduli, Poisson’s ratio, slake durability, and calcium carbonate content. Tests to determine the damping and modulus degradation characteristics of soil and rock specimens were also conducted. Petrographic analyses, X-ray diffraction and scanning electron microscopy were conducted on rock specimens.

The laboratory testing program was developed and performed in accordance with the guidance presented in RG 1.138, “Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants,” Revision 3. The type of test, numbers of tests, and testing agency are summarized in Table 2.6-19. Test data and results of the laboratory testing are documented in the subsurface investigation data report.

#### 2.6.4.2.3 Engineering Properties

Results of the recent subsurface investigation program disclose a generalized subsurface profile at the site consisting of approximately 20 feet of overburden soil, underlain by 30 feet of weathered rock followed by fresh rock below a depth of approximately 50 feet. The weathered rock grades from completely weathered to slightly weathered. Fresh rock exhibits no signs of


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weathering and is characterized as very strong. The degree of weathering was assessed during drilling by the rig geologist following the rock classification guidelines as documented in the subsurface investigation data report.

The static engineering properties for the overburden soil, proposed granular backfill, weathered rock, fresh rock, and rock mass are derived from the recent subsurface investigation and laboratory testing programs and are provided in Table 2.6-20. The engineering properties are developed to evaluate the stability of the foundation materials. The following sections briefly describe the sources and methods used to develop these properties.

2.6.4.2.4 Engineering Properties of Overburden Soil

The following paragraphs describe the properties of the overburden soil. The properties are developed based on the results of in-situ test results and laboratory testing documented in the subsurface investigation data report.

Field measured Standard Penetration Test (SPT) blow counts (N-values) were obtained during standard penetration testing in the overburden soil using calibrated automatic hammers in accordance ASTM D1586 (Reference 2.6-95). Field measured N-values are adjusted with correction factors to improve their repeatability and account for variations in test procedures (Reference 2.6-96).

These correction factors are as follows:

- ECF - energy correction factor = ETR (Energy Transfer Ratio) / 60 percent
- $C_B$ - borehole diameter correction
- $C_S$ - sampler correction factor for soil sampler using a liner
- $C_R$ - rod length correction

These correction factors are applied to the field measured N-value resulting in an $N_{60}$-value. Table 2.6-21 provides a summary of the field measured N-values and calculated $N_{60}$-values. The recommended SPT $N_{60}$-value is the median $N_{60}$-value from Table 2.6-21.

Generally, the overburden soils are silty/sandy lean clay, according to the USCS (ASTM D2487, Reference 2.6-97). Index tests (natural moisture content, grain size [percentage passing No. 200 sieve], and Atterberg limits [liquid limit and plasticity index]) were conducted on representative specimens of overburden soil. A summary of the moisture content and Atterberg limits test results is presented in Table 2.6-22.

Laboratory test results to determine total unit weight and specific gravity were conducted on selected intact samples from the overburden soil. Unit weight values were also obtained during other laboratory tests including UU and CU triaxial compression, swell, and consolidation tests. A summary of the total unit weight and specific gravity test results is presented in Table 2.6-23.

Strength of the overburden soil was evaluated considering the test results from 36 UU triaxial tests, 7 CU triaxial tests and 1 direct shear test conducted on selected intact samples. These test results are documented in the subsurface investigation data report and are scattered even


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among samples taken from the same sample tube. In general, the UU tests indicate much lower undrained shear strength ($S_u$) than the interpreted values from the SPT-N values and pocket-penetrometer or Torvane readings. This is attributed to sample disturbance during transportation of the intact samples or preparation of the test samples at the laboratory; the samples were very dry when taken in the field and extruded in the laboratory. The low UU strengths also indicate that the soil was acting more as a granular material, in line with its high silt content. It is more reasonable to base the $S_u$ of the in-situ overburden soil on the correlation with the SPT-N values.

The $S_u$ is calculated using the SPT-N value (Reference 2.6-98):

$S_u = N_{60}/8$ kilo-pounds (kips) per square foot (ksf)

Thus, using a mean $N_{60} = 59$ from Table 2.6-21, with a standard deviation of 27, the

$S_u = (59 - 27)/8 = 4$ ksf.

The value for high strain elastic modulus ($E_H$) for fine grained soils is determined from the relationship with $S_u$ (Reference 2.6-99):

$E_H = 600 \times S_u$

The high strain shear modulus ($G_H$) is a function of the modulus of elasticity (Reference 2.6-98):

$G_H = E_H / [2*(1+\mu)]$

Where, $\mu$ = Poisson's ratio = 0.35 from Table 2.6-20

For the overburden soil in the area of the SR structures,

- $E_H = 600 \times 4$ ksf = 2,400 ksf
- $G_H = 2{,}400/[2 \times (1+0.35)] \sim 900$ ksf.

One-dimensional consolidations tests were conducted on selected intact samples. These samples were taken from areas outside of the NI where the SR structures are located. The samples were taken in the softer soils in low-lying areas and do not reflect the stiff to hard in-situ soils in the higher (ridge) areas where the SR structures are located. Intact sampling in stiff to hard soils is difficult and results in disturbed samples for which laboratory test results are not indicative of in-situ conditions. Results of consolidation testing are documented in the subsurface investigation data report.

#### 2.6.4.2.5 Engineering Properties of Engineered Fill Material

Engineered fill materials will be used during construction of the SR structures. Compacted granular backfill is planned as side fill at the RXB, Fuel Handling Building (FHB), and NI Control Building (NCB). Compacted granular backfill is planned to support the mat foundation of the FHB. Cementitious fill (lean concrete or controlled low strength material [CLSM] or "flowable fill") is planned as backfill in the annulus between fresh rock and the foundation wall of the RXB shaft.


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This annulus is expected to be small, thus the presence of cementitious fill is not expected to alter the dynamic behavior of the structure. In an alternative plan, the RXB shaft wall will be constructed directly against the excavated rock face. In this plan, there is no annulus and thus no backfill.

The following engineering properties are assigned to compacted granular backfill. These properties will be confirmed once a borrow source has been identified.

- $N_{60}$ = 30 blows per foot
- Internal friction angle (φ) = 35º
- Total unit weight (g) = 130 pcf
- $G_s$ = 2.7
- $E_H$ = 1,100 ksf
- $G_H$ = 425 ksf
- Poisson's ratio (n) = 0.30
- Shear and compression wave velocity ($V_s$ and $V_p$) (see values in Table 2.6-32)
- Shear modulus and material damping versus shear strain (see values in Table 2.6-33)

#### 2.6.4.2.6 Engineering Properties of Weathered and Fresh Rock

Generally, rock coring was initiated in a borehole after two consecutive blow counts exceeding 50 blows per 6 inches of penetration. Rock quality designation (RQD) values were not recorded by the rig geologist where highly weathered rock was encountered. RQD and core recovery data are provided on the coring logs documented in the subsurface investigation data report. Figure 2.6-91 shows RQD plotted against elevation for 1,965 core runs to an elevation of 6,500 feet. In fresh rock, the mean RQD is high, exceeding 90 percent. Figure 2.6-92 shows core recovery plotted against elevation for 2,234 core runs to an elevation of 6,500 feet. In fresh rock, the mean core recovery is high, exceeding 95 percent.

Mineralogic analyses were conducted on rock core specimens by the University of Wyoming using x-ray diffraction, petrography, and scanning electron microscopy techniques. The specimens were calcareous mudstones and siltstones. Specimens contained abundant clay minerals, quartz, potassium feldspar, plagioclase feldspar, calcite, and dolomite. Clay minerals identified using x-ray diffraction include smectite, illite, and kaolinite. Results are documented in the subsurface investigation data report.

Unit weight, specific gravity, moisture content, porosity, and absorption tests were conducted on specimens of weathered and fresh rock. A summary of results is presented in Table 2.6-24.

Unconfined compressive strength tests were conducted primarily on specimens of fresh rock. A summary of results along with the recommended unconfined compressive strength value is presented in Table 2.6-25. Young's modulus values were measured on 27 unconfined compressive strength tests. For each test, the Young's modulus and Poisson's ratio values were determined for three stress ranges:


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- 400-3,050 psi – average Young's modulus = 7.4 x $10^5$ psi, Poisson's ratio = 0.17
- 3,050-5,700 psi – average Young's modulus = 8.9 x $10^5$psi, Poisson's ratio = 0.30
- 5,700-8,350 psi – average Young's modulus = 8.7 x $10^5$ psi, Poisson's ratio = 0.47

Slake durability index tests were performed as part of the laboratory testing program. Slake durability index values provide an indication of the susceptibility of the rock to slaking. Testing was primarily limited to specimens of fresh rock. Eleven specimens of fresh rock were tested with a mean and median slake durability index value of 97 percent and a standard deviation (SD) of 0.78 percent.

Calcium carbonate content tests (calcite equivalent, percent) were conducted as part of the laboratory testing program. Sixty-nine samples of weathered and fresh rock were tested with a mean and median calcium carbonate content value of 12 percent and an SD of 4.1 percent.

Estimates of the Hoek-Brown intact rock material constant ($m_i$) were derived from triaxial testing and are shown in Figure 2.6-93 for all weathering grades. A value of 3 with a range of 2 to 5 is used.

Specimens of siltstone and clay interbeds were analyzed for the swelling minerals. The mineral found in the largest quantity was smectite. A median value of 7.4 percent and range of 1.4 to 59.4 percent was measured.

#### 2.6.4.2.7 Discontinuity Properties

Discontinuity orientations were measured from acoustic televiewer surveys performed in eight boreholes. The primary discontinuity set has a dip of 20 degrees and a dip direction of 285 degrees and corresponds to bedding. In addition, acoustic televiewer surveys show many randomly-oriented joints. Section 2.6.1 provides a discussion of folds and faults.

Tilt tests were performed on seven specimens of saw-cut siltstone surfaces to estimate the basic friction angle. The results of these tests ranged from 27 degrees to 33 degrees, with a median value of 29 degrees. The value of 29 degrees is used for all weathering grades of siltstone.

Measurements of joint roughness ($J_r$) and joint alteration ($J_a$) were estimated as part of the rock core logging process. Guidance for estimating these parameters is provided in Reference 2.6-100. Numerical values for these two parameters were averaged for 10-foot increments and are plotted in Figure 2.6-94.

Discontinuity spacing was measured during logging of the core run after drilling. A total of 1,576 measurements of discontinuity spacing were made from all boreholes core runs where discontinuities were noted. The following discontinuity spacings are used:

- Highly to completely weathered rock – 0.7 feet
- Slightly to moderately weathered rock – 3 feet
- Fresh rock – 15 feet


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No measurements were made of the wall strength of discontinuities. Wall strength is taken as 50 percent of the intact rock strength. The following values for wall strength of discontinuities are used:

- Highly to completely weathered rock – 750 psi
- Slightly to moderately weathered rock – 1,500 psi
- Fresh rock – 3,500 psi

The FI or frequency is a measure of the number of fractures occurring within a unit length. It was calculated by dividing the number of natural fractures by the core length and subsequently reported as fractures per foot. The following values for FI are used:

- Highly to completely weathered rock – 1.0
- Slightly to moderately weathered rock – 0.3
- Fresh rock – 0.02

The Barton-Bandis joint strength model (Reference 2.6-103) describes the relationship between the shear strength and normal stress for discontinuity surfaces. It can be used in stability analyses when failure is discontinuity driven as opposed to a rock mass failure. The relationship incorporates normal stress, residual friction angle, joint roughness coefficient (JRC) and joint wall compressive strength. Measured $J_r$ and estimated JRC indicate that the majority of discontinuity surfaces are relatively smooth with a modal value of 2.5 for the JRC. For stability analyses using the Barton-Bandis joint strength model, a value of 2 for JRC is used.

#### 2.6.4.2.8 Rock Mass Properties

Rock mass properties account for discontinuities, weathered and fractured zones and shear zones. The following paragraphs describe the rock mass strength and deformation properties. The application of the geologic strength index (GSI) classification and Hoek-Brown failure criterion assumes that the rock mass contains several sets of discontinuities that are closely spaced relative to the proposed structure, such that it behaves as a homogeneous and isotropic mass and that a predetermined failure plane does not exist. In other words, while the behavior of the rock mass is controlled by the movement and rotation of the rock blocks separated by intersecting discontinuities, there are no preferred failure directions. The size of the excavation is expected to be much larger than the rock blocks that make up the rock mass at the site.

The GSI is calculated indirectly from estimates of $J_r$ and $J_a$ combined with RQD measurements using the methodology by Hoek et al. (Reference 2.6-126).

Values for GSI are:

- Highly to completely weathered rock – 25
- Slightly to moderately weathered rock – 60
- Fresh rock – 80


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The Hoek-Brown criterion uses a disturbance factor (D) to account for damage from blasting or other excavation methods, as well as stress relief from removal of overburden. Values may range from 0 to 1, with a value of 0 assigned for no disturbance and a value of 1 assigned to poor or mine production blasting. This D has been reported as being limited to the first few feet of rock for civil engineering rock slope applications (Reference 2.6-101). Based on guidance provide by Hoek (Reference 2.6-102), a value of 0.7 is used for the first 5 feet of rock beyond the excavation surface.

Rock mass deformation modulus ($E_m$) values were developed for each of the rock weathering zones from seven empirical equations that incorporate the intact rock modulus. Table 2.6-34 provides a summary of the results from these equations along with a best estimate value.

The Generalized Hoek-Brown failure criterion is used to estimate rock mass strength. This is an empirical criterion which relates strength in terms of the major and minor principal stresses and assumes that failure is controlled by interlocked blocks and pieces of intact rock, with no preferred failure plane. The equations presented by Hoek et. al (Reference 2.6-128), were used to determine the rock mass strength for each of the rock weathering zones using the Hoek-Brown criterion along with GSI, intact uniaxial compressive strength ($\sigma_{ci}$), $m_i$ and D. The Hoek-Brown parameters and rock mass strength are shown in Table 2.6-26.

#### 2.6.4.2.9 Dynamic Properties

The following dynamic properties of the subsurface materials are evaluated from data collected during the recent subsurface investigation:

- $V_s$ and $V_p$
- Poisson’s ratio
- Low strain shear modulus ($G_{max}$) and material damping
- Shear modulus and material damping versus shear strain

$V_s$ and $V_p$ measurements were made using P-S suspension logging in eight boreholes located in the NI and EI areas, and the downhole (DH) method in three boreholes located in the NI area. P-S logging data were collected to a depth of approximately 1,500 feet while DH data were collected to approximately 600 feet. $V_s$ and $V_p$ data were collected in the overburden soil, weathered rock, and fresh rock. The velocity profiles developed from these data are presented in Figure 2.6-95 for $V_s$ and Figure 2.6-96 for $V_p$.

$V_s$ profiles were also determined using SASW along 13 arrays across the site. These arrays ranged in length from 400 to 5,200 feet. The $V_s$ profiles developed from these data are presented in Figure 2.6-97, illustrating the short arrays, and Figure 2.6-98 illustrating the long, sitewide arrays.

The summary of combined $V_s$ profiles from these data is presented in Figure 2.6-99. Figure 2.6-100 shows the mean plus one standard deviation of the SASW data set and the combined P-S/DH data set. Section 2.6.4.7.2 provides a discussion on the differences in these data sets.


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Dynamic (low-strain) Poisson's ratios have been computed from the P-S and DH velocity data. These results show that, in general, Poisson's ratio in the overburden soil and weathered rock ranges from 0.35 to 0.38 and in the intact rock (below elevation 6,655 ft) is approximately 0.33.

$G_{max}$ is determined using the relationship provided in Reference 2.6-104:

$$G_{max} = \rho V_s^2 \text{ or } G_{max} = \frac{\gamma}{g} V_s^2$$

Where, $\rho$ is the soil/rock mass density, $\gamma$ is the soil/rock unit weight, and $g$ is the acceleration of gravity. $G_{max}$ values are calculated from $V_s$ (P-S data) and are shown in Table 2.6-27 where the overburden soils extend from 0 to 20 feet, weathered rock extends from 20 to 50 feet, and fresh rock is below 50 feet.

Low strain material damping is evaluated from laboratory tests using the free-free resonant column test. Testing of fresh rock specimens from depths of 50 to 200 feet provide a mean material damping value of approximately 2.3 percent.

The strain dependent characteristics of shear modulus and damping for in-situ soil, weathered rock and fresh rock are evaluated based on empirical relationships, test results on similar materials, and test results on site rock core samples. The shear modulus and damping relationships for in-situ soil are assigned based on clay soil with a PI of 16, interpolated from EPRI 1993 (Reference 2.6-126). A range of values (upper and lower bound) for shear modulus and damping relationships for weathered rock are assigned based on dynamic testing on weathered and unweathered shale by Stokoe et al. (Reference 2.6-127). The assigned values for in-situ soil and weathered rock are provided in Table 2.6-33.

A range of values, shown in Figure 2.6-101, is used to bound the shear modulus and damping relationships for fresh rock at the site. Laboratory dynamic tests conducted on three rock core specimens from the site are documented in an addendum to the subsurface investigation data report. With one exception, these results are in agreement with the relationships for fresh rock presented in Figure 2.6-101.

#### 2.6.4.2.10 Corrosion Potential

The degree of corrosiveness of in-situ materials towards buried metal is measured by pH, chloride content, and electrical resistivity. The aggressiveness of in-situ materials towards buried reinforced concrete is measured by sulfate content. Laboratory testing to measure pH, chloride, and sulfate was conducted on 8 soil samples, 10 rock samples, and 23 groundwater samples. Laboratory testing to measure electrical resistivity was conducted on five soil samples from shallow test pits. Test results disclosed the following:

- pH test results are in the mildly corrosive range for steel corrosiveness
- Chloride test results are in the mildly corrosive range for steel corrosiveness
- Electrical resistivity results are in the mildly to corrosive range for steel corrosiveness
- Sulfate test results indicate severe attack for concrete exposure


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### 2.6.4.3 Exploration and Foundation Interfaces

This section discusses the recent subsurface investigation program and reviews the layout of SR structures. Field work was preformed between December 2021 and July 2022 and extended across the entire site. The scope of the field program is summarized in Table 2.6-18. The components of the field work are described in the following paragraphs.

The subsurface investigation program was performed following the guidelines in RG 1.132. A summary of the associated laboratory testing program along with test results is described in Section 2.6.4.2. Field work was performed under an audited and approved quality program with work procedures developed specifically for the project. Soil and rock core samples were stored in a temperature-controlled storage facility at the nearby Naughton Coal Power Plant during the subsurface investigation. The samples were transported to and stored in the sample storage facility following the methods in ASTM D4220 (Reference 2.6-105).

The subsurface investigation program was developed to characterize the entire site with particular focus on the NI and EI areas based on the site layout/plot plan at the time of the exploration. Exploration points and methods were distributed across SR structures following the guidelines in RG 1.132. Exploration points and methods were distributed across the remainder of the site in an effort to fully characterize the subsurface conditions. Some aspects of the site layout have changed since the completion of field work and the distribution of exploration points at SR structures does not meet the guidelines in RG 1.132 in some areas. Nevertheless, the subsurface investigation program shows that the site is relatively uniform and the high density of borings in the NI is sufficient to characterize the subsurface conditions associated with SR structures.

The as-built boring location plan is provided as Figure 2.6-102 and Figure 2.6-103. Figure 2.6-102 shows as-built exploration point locations in the NI an EI areas. Figure 2.6-103 shows as-built exploration point locations on the perimeter of the site and as-built coordinates, elevations, and explorations depths. Cross sections that show the subsurface profile beneath SR structures are provided in Figure 2.6-104 and Figure 2.6-105.

Section 2.6.4.3.1 through Section 2.6.4.3.8 provide a summary of the field work conducted for the subsurface investigation program. Geophysical surveys are described in Section 2.6.4.4. Details and results of the subsurface exploration program are provided in the subsurface investigation data report.

#### 2.6.4.3.1 Borings

The 109 borings drilled at the site ranged in depth from approximately 12 to 325 feet with the exception of B-122, at the center of the RXB, which was drilled to a depth of 1,520 feet. The depth of the borings associated with the SR structures ranged from 175 to 1,520 feet. The shallow borings (12 to 25 feet) were drilled in roadway and parking areas. A total of 13,765 linear feet of drilling, excluding offset borings for intact sampling and pressuremeter testing, was conducted at the site.


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The borings were advanced in soil using rotary wash drilling techniques until SPT refusal occurred, at which point rock coring was started. The soil was sampled using an SPT sampler. The SPTs were performed with automatic hammers in accordance with ASTM D1586 (Reference 2.6-95). The automatic hammers were calibrated in accordance with ASTM D4633 (Reference 2.6-106). SPT energy measurements are documented in the subsurface investigation data report. SPT refusal was defined as an SPT N-value of 50 blow counts resulting in 6 inches or less of penetration.

The recovered soil samples were visually described and classified by the rig geologist. Each sample was photographed. A selected portion of the soil sample was placed in a glass sample jar with a moisture-proof lid. The sample jars were labeled, placed in boxes and transported to the onsite storage area. Where cohesive soil was identified for intact sampling, samples were taken at the selected depth in offset borings. Samples were retrieved using a three-inch inside diameter thin-walled tube sampler in accordance with ASTM D1587 (Reference 2.6-107), sealed with microcrystalline wax, and protected with plastic caps to prevent moisture loss.

The borings were advanced in rock using wireline rock coring equipment consisting of five-foot-long HQ3 core barrels with a split inner barrel. At the greater depths in B-122, 10-foot-long core barrels were used. Rock coring was performed in accordance with ASTM D2113 (Reference 2.6-108). A total of 10,583 linear feet of rock coring was conducted. After removal of rock cores from the split inner barrel, the recovered rock core was photographed and placed in wooden core boxes. The onsite geologist visually examined the core, noting the presence of joints and fractures and distinguishing natural breaks from mechanical breaks, and recorded this information on the field log. The geologist also computed the percentage recovery and RQD. Filled core boxes were transported to an onsite sample storage facility where a senior geologist reviewed the field log against the rock core.

The borings were tremie-backfilled with cement-bentonite slurry once all testing and sampling was completed. Boring logs and photographs of the soil samples and rock cores are documented in the subsurface investigation data report. Boring depths are summarized in Table 2.6-28.

#### 2.6.4.3.2 Test Pits

Eight test pits were excavated to depths of four feet at the site to obtain bulk samples and to conduct in-situ thermal resistivity testing. The excavated soil was visually examined and classified by the onsite geologist and recorded on a field log. Bulk samples of representative soil types were taken and stored for future testing. Upon completion of the excavation, each test pit was backfilled with the excavated soil. The test pit logs are contained in the subsurface investigation data report.

#### 2.6.4.3.3 Observation Wells, Field Permeability Testing, and Long-Term Groundwater Level Monitoring

Fifty-two groundwater observation wells were installed across the site to monitor groundwater levels. Well locations were selected to provide a comprehensive representation of the site. The wells were installed in the weathered and fresh rock as well-clusters with two or three wells in the cluster installed at different depths. Screen intervals were determined from the adjacent geotechnical boring. The boreholes for the observation wells were advanced using a rotary


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air-percussion drill rig. Wells were constructed with PVC well screen and casing, filter pack, bentonite seal, and topped with a grout plug. Each well was capped with a protective locked steel cover and surrounded with a concrete pad. Each well was developed by pumping and surging. Field permeability testing was conducted in the weathered rock and fresh rock. Slug tests were conducted in the well borehole while packer tests were conducted in offset borings.

Long-term groundwater level monitoring is being conducted to evaluate the seasonal fluctuation of the groundwater levels. The 24-month program began in August 2022. The monitoring includes taking monthly manual water level measurements using an electronic water level meter and downloading transducer data from the wells so equipped. Twenty observation wells were equipped with vented pressure transducers to provide nearly continuous groundwater level measurements.

Section 2.6.4.6 provides further discussions of observations wells, field permeability testing, and long-term groundwater monitoring.

#### 2.6.4.3.4 Downhole Geophysical Testing

DH geophysical testing was conducted in boreholes in the NI and EI areas. This testing program is summarized in Section 2.6.4.4.2. Details of the field methods and results are documented in the subsurface investigation data report.

#### 2.6.4.3.5 Electrical Resistivity Testing

Field electrical resistivity testing was conducted at eight array locations in both the NI and EI. This testing program is summarized in Section 2.6.4.4.1. Details of the field methods and results are documented in the subsurface investigation data report.

#### 2.6.4.3.6 Thermal Resistivity Testing

Near-surface geophysical testing in the form of in-situ thermal resistivity testing was conducted in accordance with approved subcontractor procedures and ASTM D5334 (Reference 2.6-109). Testing was performed in eight test pits at one-foot depth intervals to a maximum depth of four feet. Three test pits were located in the NI and five were located in the EI area. Details of field methods and results are documented in the subsurface investigation data report.

#### 2.6.4.3.7 Pressuremeter Testing

Pressuremeter testing was conducted in accordance with approved subcontractor procedures and ASTM D4719 (Reference 2.6-110). Tests were conducted on weathered rock to determine the in-situ value of elastic modulus. Offset borings were drilled to the depth of interest. Fourteen tests were taken at five borehole locations. Boreholes were located in the EI area. A calibrated, approximately three-inch diameter pressuremeter was used. Details of field methods and results are documented in the subsurface investigation data report.


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#### 2.6.4.3.8 Spectral Analysis of Surface Waves

Surface geophysical testing using the Spectral Analysis of Surface Waves (SASW) method was conducted at the site to evaluate the $V_s$ profile and assess the variability of $V_s$ across the site. Short arrays, which target shallower depths, were conducted in and around the NI while the longer arrays, targeting greater depths, extended across the site. This testing program is summarized in Section 2.6.4.4.1. Details of field methods and results are documented in the subsurface investigation data report.

### 2.6.4.4 Geophysical Surveys

Both surface and downhole geophysical testing methods were used to collect site-specific data. Surface testing consisted of electrical resistivity testing to develop electrical resistivity profiles, and SASW testing to develop $V_S$ data and DH methods were used to collect the following data:

- Natural gamma
- Dual induction (conductivity)
- Long and short normal resistivity
- Spontaneous potential
- Magnetic resonance
- Caliper
- Acoustic televiewer with deviation
- $V_S$ and $V_P$ velocity

Details of the methods, equipment, and results of the surface and DH geophysical testing are documented in the subsurface investigation data report.

#### 2.6.4.4.1 Surface and Near Surface Geophysical Testing

SASW testing was conducted at the University of Texas. This non-invasive test method measures Rayleigh-type surface waves that are generated from a vibratory source. High-frequency surface waves propagate with a short wavelength through shallow materials, and low-frequency surface waves propagate with a longer wavelength through deeper materials. Through spectral analysis, a dispersion curve was developed as an equivalent $V_S$ profile.

Thirteen SASW arrays were conducted to characterize the site, with locations shown in Figure 2.6-103. Eight of the arrays were concentrated in the NI area and had total lengths of approximately 400 to 680 feet each. Four longer arrays, two roughly north/south and two roughly east/west, were also conducted and had total lengths ranging from approximately 1,900 to 4,000 feet. One nearly mile-long (5,200 feet) roughly north/south array was also conducted. Hammers and metal plates were used to generate surface waves for the shorter wavelengths and a large mobile vibrator called “Liquidator” was used for longer wavelengths. The $V_S$ profiles interpreted from the results of the SASW testing are shown in Figure 2.6-97 and Figure 2.6-98.


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Field electrical resistivity testing was conducted in accordance with ASTM G57 (Reference 2.6-111) using four-pin Wenner resistivity arrays to determine the in-situ electrical resistivity of the subsurface materials. Testing consisted of seven short arrays with electrode spacing up to 30-feet and one long array with electrode spacing up to 300 feet. At each location, there were two intersecting arrays (approximately north/south and east/west). Two short arrays were located in the NI while the remaining five short arrays were located west of the NI or in the EI. The long array was located in the switchyard area.

2.6.4.4.2 Downhole Geophysical Testing

DH geophysical testing to measure $V_s$ and $V_p$ was conducted in eight uncased and three cased boreholes using P-S suspension logging and DH seismic testing. These boreholes were located across the site including four in the NI and four in the EI. Borehole depths ranged from 170 feet (B-134) to 1,520 feet (B-122). Boring B-122 was located at the center of the RXB. The cased boreholes, used for DH seismic testing, were co-located with three of the uncased boreholes in the NI, namely B-117, B-122, and B-127.

DH seismic testing was conducted in accordance with ASTM D7400 (Reference 2.6-112). DH seismic testing measures $V_S$ and $V_P$ data using an excitation source on the ground surface and DH receivers. Cased and grouted boreholes were used for seismic DH testing. P-S suspension logging was also conducted. This method measures $V_S$ and $V_P$ data using a DH excitation source and downhole receivers. Uncased boreholes were used for P-S suspension logging. For both methods, the boreholes were filled with water during the test. While DH seismic testing measures travel time from the surface to the receiver, P-S suspension logging measures travel time between the downhole source and two downhole receivers. DH seismic testing was conducted to a depth of approximately 600 feet while P-S suspension logging was conducted to a depth of approximately 1,500 feet. Table 2.6-29 provides a summary of the boreholes and the DH geophysical methods conducted in each. Measured $V_S$ and $V_P$ data from both methods are documented in the subsurface investigation data report. Results of these tests are discussed in Section 2.6.4.2.9 and are shown on Figure 2.6-95 and Figure 2.6-96.

Varieties of downhole geophysical testing methods were also used to assist in characterizing the geologic units encountered at the site. These are summarized Table 2.6-29 and are listed at the beginning of Section 2.6.4.4. Natural gamma and dual induction (conductivity) were conducted in accordance with ASTM D6274 (Reference 2.6-113) and ASTM D6726 (Reference 2.6-114), respectively. Long and short normal resistivity and spontaneous potential were conducted in accordance with ASTM D5753 (Reference 2.6-115). Caliper testing was conducted in accordance with ASTM D6167 (Reference 2.6-116). Natural gamma, dual induction, and long and short resistivity logging aid in characterizing the lithology of the borehole. Spontaneous potential and magnetic resonance logging aid in characterizing the porosity of the geologic unit and flow of groundwater through the unit. Caliper logging measures the diameter of the borehole and identifies any anomalous portions of the borehole wall.

Acoustic televiewer logging was conducted in accordance with ASTM D5753 (Reference 2.6-115) at each of the uncased boreholes. The acoustic televiewer obtained a continuous and oriented 360-degree view of the borehole wall from which the character, relation, and orientation of lithologic and structural planar features such as discontinuity orientations, dip


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directions, aperture widths, and fracture frequency can be evaluated. Borehole inclination and deviation from vertical were also measured using the acoustic televiewer. In the three cased boreholes, the acoustic televiewer was used to provide images of the grout interface at the outside of the PVC casing to verify the grout bond.

The acoustic televiewer data were processed to produce sinusoidal projections of planar and semi-planar discontinuities over the televiewer images. The sinusoidal projections were processed to calculate an apparent dip angle using the nominal boring diameter for each boring. The acoustic televiewer images and the dip and azimuth of the dip data are produced on multi-page logs documented in the subsurface investigation data report. Depths on the logs are referenced to ground surface. Rose diagrams of the discontinuity dip azimuths are also included in the subsurface investigation data report.

#### 2.6.4.5 Excavation and Backfill

##### 2.6.4.5.1 Excavation

Excavation in soil will be made with conventional earthmoving equipment with cuts in the side slopes made at inclinations of two (horizontal) to one (vertical). Excavation will adhere to regulations from Occupational Safety and Health Administration (OSHA), in accordance with 29 CFR 1926, “Safety and Health Regulations for Construction.” Where vertical cuts are required, such as for adjacent construction, these cuts will be supported with tied-back sheet piles or soldier pile and lagging walls.

The weathered rock below the overburden soil transitions from completely weathered rock to slightly weathered rock. Vertical cuts in the weathered rock will be achieved using conventional excavating equipment. Support of excavation will be made using rock dowels, rock bolts, and shotcrete. Excavations in completely to highly weathered rock require grouted rock dowels and a layer of shotcrete. The excavations in moderately to slightly weathered rock require tensioned rock anchors and a layer of shotcrete.

As described in Section 2.5.3.1, groundwater is encountered above the top of fresh rock, within the weathered rock. Dewatering as described in Section 2.6.4.6 is required during deep excavation and for support of excavation. In some cases, horizontal drains may be required.

A neat excavation in fresh rock will be made using controlled blasting techniques with near-vertical rock slopes. Methods include production and perimeter blasting. Production blasting will remove the bulk of the excavation, and perimeter blasting will form the finished excavation face and foundation grade. Support of excavation will be made with tensioned rock bolts and a layer of shotcrete, as required.

Geologic mapping of the excavation, in conformance with RG 1.132, will be performed before placement of shotcrete, backfill materials, dental concrete, or foundation concrete. The geologic mapping program will include photographic documentation of the exposed surface and documentation of geologic features.


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2.6.4.5.2 Backfill

SR structures are located along the ridge line, the elevation of which is approximately the rough grade elevation of the NI. Figure 2.6-108 and Figure 2.6-109 illustrate the extent of the excavations for NI structures. Where structures are embedded in fresh rock, cementitious backfill will be placed in the annulus between the structure and excavation walls from the bottom of excavation to the top of fresh rock. With a neat excavation in fresh rock, the annulus around the sides of the excavation is expected to be small. An alternative plan calls for no annulus where the structure wall is constructed against the excavation; thus, no backfill. Compacted granular backfill will be placed above fresh rock, in the weathered rock zone. Compacted granular backfill will also be used as sidefill from the base of the excavation to finish grade. Compacted granular backfill will be used to support the grade slab portions of the FHB. The deeper portions of the FHB are founded on weathered and fresh rock. Founding conditions are discussed in Section 2.6.4.10.1.

Cementitious backfill in the annulus will be placed in lifts, allowing the preceding lift to set before placing additional material. Compacted granular backfill will be placed in lifts and compacted to at least 95 percent of the modified Proctor (ASTM D1557, Reference 2.6-117) maximum dry density.

2.6.4.6 Groundwater Conditions

Groundwater conditions, field permeability testing, chemical properties and construction dewatering are discussed in the following sections. More detailed discussions of groundwater at and in the vicinity of the site are provided in Section 2.5.3.1.

Groundwater conditions across the site were explored with the installation of 52 observation wells, the locations of which are shown on Figure 2.6-102. These wells included 24 upper-level (U) wells, 24 mid-level (M) wells, and 4 lower-level (L) wells. The U wells were screened within 50 feet below ground surface. The M wells were screened between 50 and 100 feet below ground surface and the L wells were screened between 100 and 150 feet below ground surface. Table 2.6-30 provides a summary of these wells. Typically, 10-foot screen lengths were used. The well installation logs and well development data are documented in the subsurface investigation data report.

A long-term (24 months) groundwater monitoring program was initiated in August 2022 to evaluate the seasonal variation of groundwater levels at the site. For this program, vented pressure transducers were installed in 20 wells to provide nearly continuous water level readings. This program includes monthly manual water level readings in all 52 wells and downloading of transducer data from the 20 wells. Observation wells with vented pressure transducers installed are noted in Table 2.6-30.

Regionally, the Hilliard Shale is classified as a major aquitard. Observations made during the subsurface investigation indicated that groundwater was primarily flowing through the weathered material. Groundwater flow within the fresh siltstone (Hilliard Shale) is through discontinuities; the fracture density significantly drops below depths of 50 to 60 feet.


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Field permeability testing was conducted to determine hydraulic conductivity and transmissivity values for the weathered and fresh rock at the site. A total of 28 packer tests adjacent to boreholes and 48 slug tests were performed. Long-term well recovery data were also evaluated. Slug tests were conducted in both weathered and fresh rock, while packer tests were conducted only in weathered rock. Test data and results are provided in the subsurface investigation data report. These tests spatially cover the site with increasing density of the slug and packer tests at and around the NI and EI areas. Table 2.6-31 summarizes the wells and borings in which these tests were taken. A discussion of the results from slug test and packer testing is presented in Section 2.5.3.1.

A three-dimensional groundwater model was developed for the site that accounts for monthly water level readings from the 52 wells and alterations of the hydraulic properties of the subsurface materials. Model results indicate that groundwater flows from the west, north, and east, converging near the center of the site, and then flows south towards the southerly site boundary. Figure 2.6-106 displays the base model with resulting groundwater level contour and flow paths across the site.

The horizontal hydraulic gradients for the upper and middle series of wells vary from 0.002 to 0.015 and from 0.003 to 0.019, respectively. For the upper series of wells, the average hydraulic gradients along the easterly, westerly, and southerly flow directions are 0.004, 0.014, and 0.003, respectively. For the middle series of wells, the average hydraulic gradients along the easterly, westerly, and southerly flow directions are 0.004, 0.019, and 0.004, respectively. Thus, the average horizontal gradients between the upper and middle series of wells are similar.

The vertical hydraulic gradients generally indicate a downward gradient between upper and middle well pairs in the central area of the site and an upward gradient along the periphery of the site. For all the wells, the average downward gradient varies from 0.0002 to 1.6259, whereas the upward gradient averages between -0.0020 and -1.7165.

Groundwater samples were obtained from 22 observation wells for the purpose of geochemical and environmental characterization of groundwater at the site. Results show that the groundwater generally has a neutral pH, low dissolved oxygen, high electrical conductivity, and negative oxidation-reduction potential. Results show high concentrations of the major anions and cations and high total dissolved solids concentrations, consistent with the high electrical conductivity field measurements. A detailed discussion of chemical characteristics of the groundwater is provided in Section 2.5.3.1.5.

Temporary dewatering of the foundation excavations extending below the water table during construction is required. Dewatering rates range from approximately 35 to 50 gallons per minute depending on the depth, size, and number of opened planned excavations. Dewatering is performed in a manner that minimizes drawdown effects on the surrounding environment. Dewatering for excavations is achieved by gravity-type systems. Groundwater is extracted by pumping from sumps in the lowest working levels of the excavation and transferred to an impoundment facility. Horizontal relief wells are installed in the rock excavation walls to prevent hydrostatic pressure buildup behind the walls and the water is collected in sumps from which it is pumped. The response to groundwater extraction is assessed using the established network of observation wells installed at the site.


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### 2.6.4.7 Response of Soil and Rock to Dynamic Loading

This section presents descriptions of the response of soil and rock to dynamic loading and addresses the following topics:

- Section 2.6.4.7.1 – Effects of Prior Earthquakes
- Section 2.6.4.7.2 – Shear and Compression Wave Velocity Profiles
- Section 2.6.4.7.3 – Strain-Dependent Modulus Reduction and Damping Properties

#### 2.6.4.7.1 Effects of Prior Earthquakes

The historical earthquake events are described in Section 2.6.2. The effects of prior earthquakes including potential paleoseismic features at the site are described in Section 2.6.1. No evidence of liquefaction was observed at the site as described in Section 2.6.3.1.4.

#### 2.6.4.7.2 Shear and Compression Wave Velocity Profiles

Various geophysical surveys, both surface and DH, were conducted at the site to characterize the in-situ dynamic properties of the soil and rock as described in Section 2.6.4.2 and Section 2.6.4.4. P-S Suspension logging and DH seismic methods were used to collect $V_s$ and $V_p$ measurements in the soil and rock. P-S logging was conducted in four boreholes in each of the NI and EI areas. DH seismic was conducted in three boreholes in the NI area. These data sets show agreement both between the NI and EI and between P-S logging and DH seismic. Figure 2.6-95 and Figure 2.6-96 present $V_s$ and $V_p$ profiles collected at the site. The mean $V_s$ profile from the P-S and DH data along with the mean +/- one standard deviation are plotted in Figure 2.6-100.

The surface geophysical survey, SASW, was used to evaluate the $V_s$ profile within the NI and across the site. Short arrays were used in the NI and long arrays were used for site-wide data. These results are plotted in Figure 2.6-97 and Figure 2.6-98. These data show agreement.

Figure 2.6-99 is a combined plot of all DH and surface-generated Vs profiles along with a plot of the mean and +/- one standard deviation of the SASW data and P-S/DH data. This figure shows agreement of data in the shallow and deep portions of the profile. In the central portion of the profile, SASW and P-S/DH data diverge, attributed to the effects of sampling (wavelength) differences between the surface and downhole methods. The figure suggests that epistemic uncertainty in the data sets is generally captured by the +/- one standard deviation profiles. Figure 2.6-100 shows only a plot of the mean and +/- one standard deviation for the SASW data and the P-S/DH data. When evaluating site response, both data sets (SASW and P-S/DH) will be considered as possible solutions for the site $V_s$ profile rather than an average between the data sets.

#### 2.6.4.7.3 Strain-Dependent Modulus Reduction and Damping Properties

The variation of damping and shear modulus in soil and rock under dynamic loading is evaluated in this section. Section 2.6.4.2.5 provides the dynamic engineering properties for compacted granular backfill. As discussed in Section 2.6.4.2.9, the relationship between shear modulus and


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damping with shear strain for in-situ soil, weathered rock, and fresh rock was evaluated based on empirical relationships, test results on similar materials and test results on site rock core specimens. Values for compacted granular backfill, in-situ soil, and weathered rock are presented in Table 2.6-33. A range of values for the modulus reduction and damping for fresh rock (Hilliard Shale) are shown in Figure 2.6-101. Note that damping values are typically capped at 15 percent when performing site response analyses.

#### 2.6.4.8 Liquefaction Potential

This section presents the evaluation of the liquefaction potential of the materials adjacent to and under SR structures at the site and conforms to the guidelines in RG 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites,” Revision 0. Soil liquefaction is a process by which loose, saturated, granular deposits lose a significant portion of their shear strength due to pore-water pressure buildup resulting from cyclic loading, such as that caused by an earthquake. Soil liquefaction can lead to foundation bearing failures and excessive settlements.

Youd et al. (Reference 2.6-119) defines liquefaction as the transformation of a granular material from solid to liquefied state as a consequence of increased pore-water pressure and reduced effective stress. Increased pore-water pressure is induced by the tendency of granular materials to compact when subjected to cyclic shear deformations. The change of state occurs most readily in loose to moderately dense granular soils with poor drainage, such as silty sands or sands and gravels capped by or containing seams of impermeable sediment. As liquefaction occurs, the soil stratum softens, allowing large cyclic deformations to occur.

The SR structures at the site are embedded at various depths. Figure 2.6-108 and Figure 2.6-109 illustrate the extent of these excavations in the subsurface materials. With the exception of the grade slabs for the FHB, the SR structures are founded on weathered rock and fresh rock. The grade slabs for the FHB are expected to be founded on compacted granular backfill. Founding conditions are discussed in Section 2.6.4.10.1.

The RXB Head Access Area (HAA) slab is founded 55 feet below finished grade and the RXB reactor cavity slab is founded approximately 118 feet below finished grade with both structures bearing on fresh rock. If any dental work is required to prepare the foundation surface, dental concrete is used. Liquefaction cannot occur in fresh rock or concrete; therefore, there is no potential for liquefaction in the foundation materials of the RXB shaft.

The NCB is founded 33 feet below finished grade. The foundation material at this depth is weathered rock, which is not susceptible to liquefaction.

The deeper portions of the FHB are founded at depths of 32, 47, and 53 feet below finished grade. The foundation material at a depth of 32 feet is weathered rock and at 47 and 53 feet is fresh rock. As previously stated, the weathered rock and fresh rock are not susceptible to liquefaction. The shallower portions of the FHB are founded at frost depth, 3.5 feet below finished grade, and are founded on compacted granular backfill. The granular backfill is compacted to at least 95 percent of the modified Proctor value. Based on the material type and degree of compaction, this backfill is not susceptible to liquefaction.


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Section 2.6.4.5 discusses backfilling around SR structures. Concrete or cementitious backfill is used from the foundation level to the top of fresh rock if an annular space is present. Liquefaction cannot occur in concrete or cementitious backfill. Granular backfill is used around the structures from the top of fresh rock to finished grade. The granular backfill, as noted above, is not susceptible to liquefaction.

The liquefaction potential of overburden soil at the site was also evaluated. Since liquefaction occurs in granular materials, the standard methods for determining liquefaction resistance apply to soils that classify as sands or gravels. Since very little of the overburden soil at the site classifies as sand and none classifies as gravel, as described in Section 2.6.4.2.4, the SPT methods available to quantify resistance to liquefaction are not applicable. Fine-grained soils, and particularly cohesive soils, as encountered on site (Section 2.6.4.2.4) are not susceptible to liquefaction using the methods provided in RG 1.198. Nevertheless, the overburden soils at the site were evaluated using criteria for fine-grained soils proposed by Polito (Reference 2.6-120) and Seed et al. (Reference 2.6-121).

Atterberg limit test results from representative soils are plotted on Figure 2.6-107 along with the Polito criteria for liquefiable soils (Reference 2.6-120), whereby plasticity is the key factor. Where the Polito criteria, whether measured in terms of liquid limit or PI, are met, the soil is not considered susceptible to liquefaction. A PI of 10 and a liquid limit of 30 are threshold values (Reference 2.6-120). All of the site soils plotted on Figure 2.6-107 fall outside the zone of liquefiable soils and therefore are not susceptible to liquefaction by the Polito criteria.

In summary, SR structures are primarily founded on rock (weathered or fresh). Portions of the FHB are founded on compacted granular backfill. The sidewalls of SR structures are backfilled with concrete or cementitious backfill and compacted granular backfill. Neither the founding materials nor the backfill materials are susceptible to liquefaction. The overburden soils at the site consist of cohesive fine-grained materials that are not susceptible to liquefaction.

### 2.6.4.9 Earthquake Design Basis

The development of the site-specific GMRS is described in Section 2.6.2. The associated vertical GMRS is computed with V/H ratios as described in Section 2.6.2.

### 2.6.4.10 Static and Dynamic Stability

This section presents the evaluation of static and dynamic stability of SR structures and includes bearing capacity, settlement, and lateral earth pressures at the site. Description of subsurface materials, including the engineering properties of soil, compacted granular backfill, weathered rock, and fresh rock are described in Section 2.6.4.2. Figure 2.6-104 and Figure 2.6-105 present cross sections through the excavations and subsurface materials at the SR structure locations.

#### 2.6.4.10.1 Bearing Capacity

Figure 2.6-108 and Figure 2.6-109 illustrate cross sections through the excavations and subsurface materials at the SR structure locations, indicating the planned excavation support method. Section C of Figure 2.6-108 provides an east/west cross section through the FHB, RXB, and Reactor Auxiliary Building (RAB). This figure shows that the deeper basements of the FHB


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and the RXB reactor cavity basemat are founded in fresh rock. Section C of Figure 2.6-109 provides a north/south cross section through the RXB and NCB. This figure shows the NCB is founded on weathered rock. Section D of Figure 2.6-109 provides a north/south cross section through the FHB. This figure shows the basements of the FHB are founded on weathered rock and fresh rock while the mat foundation is founded in backfill.

The $q_{ult}$ for foundations supported on in-situ (cohesive) soil is calculated from Skempton's Equation in Reference 2.6-122. The $q_{ult}$ for foundations supported on compacted granular backfill is calculated from Vesic in Winterkorn & Fang (Reference 2.6-123). A factor of safety of three, applied to $q_{ult}$, is used to calculate the $q_{all}$ for foundations supported on in-situ soil or compacted granular backfill.

The $q_{ult}$ for foundations supported on rock (weathered and fresh), which is modeled as a Hoek-Brown material, is obtained using the analytical method developed by Serrano, Olalla, and Gonzalez (Reference 2.6-124).

A partial factor of safety is applied to $q_{ult}$ to calculate $q_{all}$ for foundations supported on rock as recommended by Serrano and Olalla (Reference 2.6-118). The partial safety factor is selected based on the variation in the rock properties (rock mass rating [RMR] or GSI), and unconfined compressive strength. The RMR can be calculated from the GSI using guidance in Hoek and Brown (Reference 2.6-92).

The RXB is founded at two levels. The HAA slab is founded 55 feet below finished grade at elevation 6,702 feet, at the transition from weathered to fresh rock. The lower portion, the reactor cavity basemat, is founded approximately 118 feet below finished grade at elevation 6,639 feet, in fresh rock. A $q_{all}$ of 72.5 ksf is used for the HAA slab and a $q_{all}$ of 76 ksf is used for the reactor cavity basemat.

The FHB foundation consists of a shallow mat and three basement levels as shown on Figure 2.6-109. The mat is founded 3.5 feet below finished grade at elevation 6,753.5 feet with the shallower basement level at elevation 6,725 feet and the two deeper basement levels at elevation 6,710 feet and 6,704 feet. The mat foundation is founded on compacted granular backfilll, the shallow basement is founded on weathered rock and the two deeper basements are founded on fresh rock. A $q_{all}$ of 6 ksf is used for shallow foundations on compacted granular backfill. A $q_{all}$ of 18.5 ksf is calculated for the shallow basement founded on weathered rock. A $q_{all}$ of 72.5 ksf is calculated for the deeper basements founded on fresh rock.

The NCB basement foundation, shown in Figure 2.6-109, is founded 33 feet below finished grade at elevation 6,724 feet. The basement is founded on weathered rock with a calculated $q_{all}$ of 19 ksf.

No increase in allowable bearing capacity for seismic loading is provided for the subsurface materials supporting SR structures, and thus the dynamic bearing capacity is the same as the static $q_{all}$.


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2.6.4.10.2 Settlement

Settlement of structures founded on compacted granular backfill is calculated considering the material as linearly elastic. Settlement of structures founded on weathered rock and fresh rock is calculated similarly except the elastic modulus of the soil is replaced by the rock mass deformation modulus. Settlement is calculated from Poulos & Davis (Reference 2.6-93) as follows:

$$d = \Sigma(\Delta p_i \times \Delta h_i) / E_i$$

where:

$i$ = 1 to n, where n is the number of layers

$p_i$ = vertical applied pressure at center of layer i

$h_i$ = thickness of layer i

$E_i$ = elastic modulus of layer i

The stress distribution below rectangular foundations is calculated from a Boussinesq-type distribution provided by Poulos & Davis (Reference 2.6-93) for flexible foundations with settlement calculated at the center and edge of these foundations. Differential settlement is calculated as the difference in settlement between the center and edge of foundation.

The coefficient of subgrade reaction is calculated by dividing the foundation load by the calculated settlement.

At the RXB, the average settlement of the HAA slab, with a foundation load of 6,200 psf is calculated as 0.07 inches. The coefficient of subgrade reaction for the HAA slab is 1,000 pcf. The average settlement of the reactor cavity basemat, with a foundation load of about 2,250 psf, is calculated as less than 0.02 inches. Differential settlements are small, about half the value of the average settlement. The coefficient of subgrade reaction for the reactor cavity slab is 1,980 kcf.

Settlement of the FHB foundations is evaluated considering the various foundation levels with a foundation load of 2,000 psf. The average settlement of the shallow mat foundations is calculated to range from approximately 0.1 to 0.2 inches. The coefficient of subgrade reaction for these shallow mats ranges from 120 to 240 kcf. The average settlement of the FHB basements is calculated to range from approximately 0.02 to 0.03 inches. Differential settlements are small, about half the value of the average settlement. The coefficient of subgrade reaction for the FHB basement slabs ranges from 700 to 1,500 kcf.

The average settlement of the NCB basement, with a foundation load of 1,600 psf is calculated as 0.04 inches with a differential settlement about half this value. The coefficient of subgrade reaction for the NCB basement slab is 500 kcf.


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Calculated settlements, both total and differential, of the SR structure foundations are within the design criteria established for these structures.

#### 2.6.4.10.3 Lateral Earth Pressure

The foundations at the RXB, FHB and NCB will include permanent below-grade walls. The lateral earth pressure exerted on below-grade walls is determined by considering the state of stress of the soil/rock behind the wall which develops as a result of wall movement. Due to the depth and rigidity of the below-grade walls associated with these structures, they will behave as non-yielding walls in an at-rest condition. The below-grade walls embedded in soil are backfilled with compacted granular backfill and will experience at-rest pressures due to the backfill. The below-grade walls embedded in weathered rock will experience at-rest pressures due to the weathered rock. The at-rest earth pressure coefficient ($K_0$) is calculated using the Jaky derivation discussed in Bowles (Reference 2.6-98). The below-grade walls embedded in fresh rock are backfilled with cementitious material such as lean concrete and the lateral pressure on the wall becomes negligible after the lean concrete hardens.

The calculation of lateral earth pressure behind walls backfilled with compacted granular backfill or walls in weathered rock also includes surcharge, hydrostatic, and seismic components. A design acceleration of 0.333 g is used to calculate the seismic component following methods provided in ASCE 4-16 (Reference 2.6-94). Figure 2.6-110 provides at-rest lateral earth pressure diagrams for the maximum wall depth in granular backfill and weathered rock, illustrating the individual components of lateral earth pressure and the total (combined) at-rest pressure diagram.

### 2.6.4.11 Design Criteria

RG 1.198 provides that cohesive soils with fines content greater than 30 percent that are either classified as clays or have a PI greater than 30 percent should generally not be considered susceptible to liquefaction. The liquefaction potential of the founding materials for SR structures and the overburden soils at the site is evaluated in Section 2.6.4.8. The SR structures are founded on compacted granular backfill, weathered rock, and fresh rock. These materials are not susceptible to liquefaction. The overburden soils are fine-grained and are not susceptible to liquefaction. Methods to evaluate the liquefaction potential of fine-grained soils (Reference 2.6-120 and Reference 2.6-121) confirm this conclusion.

The bearing capacity and settlement of SR structures are described in Section 2.6.4.10. Both shallow grade foundations and deep mat foundations are used. Computed allowable bearing capacity ranges from 6 ksf to 76 ksf, significantly more than the maximum applied bearing pressures. Industry-accepted design criteria for settlement provides 1 inch of settlement for shallow foundations and 2 inches of settlement for mat foundations (Reference 2.6-98). The structural design criteria for NI structures provides values of acceptable differential settlement across mat foundations. The calculated settlement for shallow foundations for SR structures on compacted granular backfill range from 0.1 to 0.2 inches, well within the industry accepted and project-specific criteria. The calculated settlement for deep mat foundations of SR structures on weathered and fresh rock range from 0.02 to 0.07 inches, within the industry-accepted criteria.


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### 2.6.4.12 Techniques to Improve Subsurface Conditions

The site is underlain with residual soil overlying weathered rock which overlays fresh rock. Fresh rock is generally encountered at depths of about 50 feet below existing grade. The overlying soils and weathered rock are stiff to hard. These conditions are suitable for the planned construction without improvement to the subsurface conditions.

## 2.6.5 Stability of Slopes

The information presented in this section is based on the results of the site-specific subsurface investigation performed at the Kemmerer Unit 1 site as described in Section 2.6.4. Note that the elevation datum referenced herein is the North American Vertical Datum of 1988, Geoid 18 and horizontal coordinates are referenced to the Wyoming State Plane Coordinate System, WY West (4904) North American Datum of 1983.

The Kemmerer Unit 1 Site is located in southwest Wyoming, in the Middle Rocky Mountain physiographic province. The site lies on the eastern side of Cumberland Flats, a broad, relatively flat-lying north-south trending valley. In the vicinity of the site, the valley is approximately 3.5 miles wide as shown in Figure 2.6-111. The valley is bounded on the west by an unnamed ridge which includes the open pit Kemmerer coal mine that rises approximately 470 feet. The valley is bounded on the east by Oyster Ridge that rises approximately 600 feet. The site is located approximately 0.8 miles from the toe of Oyster Ridge. The topography of the valley is characterized with long, roughly north-south trending resistant ridges.

Site topography includes a north-south trending ridge, on which the RXB, NCB, and FHB are located. The elevation on top of the ridge varies from approximately 6,760 feet to 6,750 feet, decreasing to the south as shown in Figure 2.6-112. The eastern shoulder of the ridge slopes gently to an elevation of about 6,745 feet while the western shoulder slopes more steeply to an elevation of about 6,737 feet. The RAB is located on the western shoulder of the ridge. Other topographic features include drainage ways on the west and south boundaries of the ridge, an abandoned rail bed trending northwest-southeast located east of the ridge, and North Fork Little Muddy Creek located east of the NI.

A variety of slopes, both temporary and permanent, consisting of soil and rock will be constructed at the site. Temporary slopes in soil and rock will be excavated for underground construction. These slopes will be backfilled during construction. Permanent slopes are limited to soil slopes.

### 2.6.5.1 Slope Characteristics

There are no planned permanent slopes in proximity to safety-significant structures in the NI. The NI area will generally be graded level to gently sloping away from the structures for drainage purposes as shown in Figure 2.6-113. Permanent slopes are located on the perimeter of the NI. Temporary slopes are planned for the NI excavations during construction.

Permanent fill slopes are located on the east, south, and west sides of the NI. A 3(H):1(V) down slope with a maximum height of 9 feet is located about 360 feet east of the FHB. A 3(H):1(V) down slope with a maximum height of 3 feet, is located about 190 feet south of the NCB, and a 3(H):1(V) down slope with a maximum height of 10 feet is located about 440 feet west of the


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RAB. Permanent slopes, associated with the NI stormwater pond, will have a height on the order of 17 feet or less with slope inclinations of 3(H):1(V). This pond is located outside of the NI, about 460 feet southeast of the FHB.

During construction, temporary excavation cut slopes will be made in the NI to construct the foundations for the RXB, RAB, NCB and FHB. These excavations will be made in the overburden soil, weathered rock, and fresh rock. The excavation will include vertical cuts with support systems, sloped faces, and sloped ramps for construction access.

Groundwater conditions in the NI and at the excavations, including seepage, high and low groundwater levels, and groundwater control are discussed in Section 2.5.3.1.

### 2.6.5.2 Design Criteria and Analyses

Slopes are evaluated under short-term and long-term loading conditions as well as under seismic loading. Recommended minimum required factor of safety values under these loading conditions are provided in Reference 2.6-134.

The stability of temporary soil slopes is evaluated under short-term loading conditions primarily using the Spencer method, based on a circular slip failure mechanism as described in Reference 2.6-134. The stability of temporary rock slopes is evaluated using a finite element analysis. Temporary slopes are used during construction for excavation of the foundations for safety-significant structures. Results of the stability analyses of soil and rock slopes is used to develop a preliminary excavation model. This model includes overburden soils excavated at a slope inclination of 2(H):1(V) and weathered rock and fresh rock excavated with vertical cuts, supported with shotcrete along with rock dowels or rock anchors. Additional discussions on these excavations are presented in Section 2.6.4.5.

The stability of permanent soil slopes is evaluated primarily using the Spencer method. The block failure method is also used to evaluate slope stability. Results show that the factors of safety of the permanent slopes exceed the recommended minimum values. In addition, the stability of the safety-significant structures is not influenced by the permanent slopes due to their configuration (height of slopes and their distances from safety-significant NI structures). Thus, there are no planned permanent slopes that will adversely affect, either directly or indirectly, safety-significant structures in the NI.

### 2.6.5.3 Results of the Investigation

A detailed subsurface investigation and laboratory testing program, as described in Section 2.6.4, encompassing the areas of existing and planned slopes, was conducted to characterize soil, rock, and groundwater conditions. These data are used for design of temporary soil and rock cut slopes and permanent slopes. There are no slopes that will adversely affect, either directly or indirectly, the safety-significant structures.

### 2.6.5.4 Properties of Borrow Material

The properties of borrow materials for use in constructing fill slopes are discussed in Section 2.6.4.


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**Table 2.6-1 Seismic Source Zone Characteristics**
**(Sheet 1 of 2)**

| | Basin and Range | Colorado Plateau-Wyoming Craton | Centennial Tectonic Belt | Eastern Snake River Plain | Great Plains | Idaho Batholith | ISB | Northern Rio Grande Rift | Trona Triggered Seismicity Zone | YS[1] Caldera | YS Geoid Anomaly |
|---|---|---|---|---|---|---|---|---|---|---|---|
| **Seismogenic Thickness** | 12 km [0.2]<br>13 km [0.6]<br>15 km [0.2] | 15 km [0.2]<br>17 km [0.5]<br>20 km [0.3] | 12 km [0.2]<br>13 km [0.6]<br>15 km [0.2] | 10 km [0.2]<br>16 km [0.2]<br>17 km [0.4]<br>20 km [0.2] | 17 km [0.2]<br>19 km [0.5]<br>21 km [0.3] | 12 km [0.2]<br>13 km [0.6]<br>15 km [0.2] | 12 km [0.2]<br>15 km [0.6]<br>17 km [0.2] | 11 km [0.3]<br>14 km [0.5]<br>17 km [0.2] | 15 km [0.2]<br>17 km [0.5]<br>20 km [0.3] | 8 km [0.2]<br>9 km [0.6]<br>14 km [0.2] | 15 km [0.3]<br>17 km [0.4]<br>20 km [0.3] |
| **Style of Faulting** | Normal (75%)<br>Strike-slip (25%) | Normal (75%)<br>Strike-slip (25%) | Normal (50%)<br>Reverse (25%)<br>Strike-slip (25%) | Normal (70%)<br>Strike-slip (30%) | Normal (75%)<br>Strike-slip (25%) | Normal (50%)<br>Strike-slip (50%) | Normal (85%)<br>Strike-slip (15%) | Normal (75%)<br>Strike-slip (25%) | Normal (75%)<br>Strike-slip (25%) | Normal (100%) | Normal (75%)<br>Strike-slip (25%) |
| **Strike of Ruptures** | 0° (50%)<br>30° (50%) | 70° (10%)<br>90° (10%)<br>290° (10%)<br>310° (10%)<br>330° (10%)<br>Uniform aleatory (50%) | Normal+Reverse<br>280° (10%)<br>300° (40%)<br>330° (40%)<br>350° (10%)<br>Strike-slip<br>315° (50%)<br>45° (50%) | 220° (10%)<br>285° (20%)<br>325° (50%)<br>340° (20%) | 0° (10%)<br>45° (10%)<br>90° (10%)<br>315° (20%)<br>Uniform aleatory (50%) | Uniform aleatory | 0° (60%)<br>30° (25%)<br>310° (15%) | 0° (20%)<br>30° (10%)<br>315° (20%)<br>Uniform aleatory (50%) | 70° (10%)<br>90° (10%)<br>290° (10%)<br>310° (10%)<br>330° (10%)<br>Uniform aleatory (50%) | Uniform aleatory | 0° (50%)<br>45° (25%)<br>315° (25%) |
| **Dip of Ruptures** | Normal<br>35° (25%)<br>50° (50%)<br>65° (25%)<br>Strike-slip<br>75° (25%)<br>90° (75%) | Normal<br>35° (25%)<br>50° (50%)<br>65° (25%)<br>Strike-slip<br>75° (25%)<br>90° (75%) | Normal+Reverse<br>30° (20%)<br>55° (40%)<br>75° (40%)<br>Strike-slip<br>50° (20%)<br>75° (40%)<br>90° (40%) | Normal<br>50° (20%)<br>70° (40%)<br>90° (40%)<br>Strike-slip<br>70° (50%)<br>90° (50%) | Normal<br>35° (25%)<br>50° (50%)<br>65° (25%)<br>Strike-slip<br>75° (25%)<br>90° (75%) | Normal<br>30° (20%)<br>55° (40%)<br>75° (40%)<br>Strike-slip<br>90° (100%) | Normal<br>30° (10%)<br>55° (60%)<br>75° (30%)<br>Strike-slip<br>75° (25%)<br>90° (75%) | Normal<br>35° (20%)<br>50° (60%)<br>65° (20%)<br>Strike-slip<br>90° (100%) | Normal<br>35° (25%)<br>50° (50%)<br>65° (25%)<br>Strike-slip<br>75° (25%)<br>90° (75%) | 35° (25%)<br>50° (50%)<br>65° (25%) | Normal<br>35° (25%)<br>50° (50%)<br>65° (25%)<br>Strike-slip<br>75° (25%)<br>90° (75%) |
| **Dip Direction** | E (50%)<br>W (50%) | Uniform aleatory | E (50%)<br>W (50%) | NE (50%)<br>SW (50%) | Uniform aleatory | Uniform aleatory | E (25%)<br>W (75%) | Uniform aleatory | Uniform aleatory | Uniform aleatory | Uniform aleatory |


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**Table 2.6-1 Seismic Source Zone Characteristics**
**(Sheet 2 of 2)**

| | Basin and Range | Colorado Plateau-Wyoming Craton | Centennial Tectonic Belt | Eastern Snake River Plain | Great Plains | Idaho Batholith | ISB | Northern Rio Grande Rift | Trona Triggered Seismicity Zone | YS[1] Caldera | YS Geoid Anomaly |
|---|---|---|---|---|---|---|---|---|---|---|---|
| **Mmax** | 6.75 [0.25]<br>7.0 [0.5]<br>7.25 [0.25] | 6.75 [0.25]<br>7.0 [0.5]<br>7.25 [0.25] | 6.75 [0.25]<br>7.0 [0.5]<br>7.25 [0.25] | 5.5 [0.4]<br>6.0 [0.4]<br>6.5 [0.2] | Default [0.5]<br>6.75 [0.25]<br>7.0 [0.5]<br>7.25 [0.25]<br>CEUS SSC [0.5]<br>5.6 [0.101]<br>6.1 [0.244]<br>6.6 [0.310]<br>7.2 [0.244]<br>8.0 [0.101] | 6.75 [0.25]<br>7.0 [0.5]<br>7.25 [0.25] | 6.75 [0.25]<br>7.0 [0.5]<br>7.25 [0.25] | 6.75 [0.25]<br>7.0 [0.5]<br>7.25 [0.25] | 5.0 [0.4]<br>6.0 [0.4]<br>7.0 [0.2] | 6.5 [0.25]<br>6.75 [0.5]<br>7.0 [0.25] | 6.75 [0.25]<br>7.0 [0.5]<br>7.25 [0.25] |
| **Mmax (obs)** | E[**M**] 5.92008 | E[**M**] 5.61959 | E[**M**] 6.11947 | Mw 6.5 worldwide analog | E[**M**] 4.81964 | E[**M**] 6.42020 | E[**M**] 6.41934 | E[**M**] 4.41882 | E[**M**] 3.71994 | E[**M**] 4.81964 | E[**M**] 5.71959 |
| **Spatial Variation of Recurrence** | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] | Smoothing [1.0] |
| **Source Term** | WUS[2] | CEUS | WUS | WUS | CEUS | WUS | WUS | WUS | CEUS | WUS | WUS |

1 Yellowstone

2 Western U.S.


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**Table 2.6-2 Geographic Regions Used to Help Organize the Discussion of Fault Sources**

| Region or Class | Description |
|---|---|
| Clustered faults | Three fault sources exhibiting clustered behavior (Rock Creek, Bear River, and Greys River) located along the eastern margin of the Wyoming Salient. |
| Wyoming Salient region | General region of the Wyoming Salient of the Sevier fold and thrust belt spanning portions of Wyoming, Idaho, and Utah between the Teton Range on the north and Uinta Range on the south. The greatest number of fault sources in the SSC model belong to this region. |
| Teton region | Five fault sources including the Teton - Yellowstone region, including the Teton fault. |
| Central Wyoming region | Six fault sources, all in the Colorado Plateau-Wyoming Craton seismic source zone. |
| Wasatch Fault System | Includes principal fault sources belonging to the Wasatch fault system along the west flank of the Wasatch Range. |


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**Table 2.6-3 Closest Fault Source-to-Site Distances**
**(Sheet 1 of 2)**

| | | | Site-To-Source Distance (km) |
|---|---|---|---|
| **Fault Source** | **Group** | **Region/Class** | **Kemmerer Unit 1** |
| Rock Creek | Clustered | Clustered | 23 |
| Spring Creek-North Bridger Creek | 1 | WY Salient | 27 |
| Sublette Flat South | 1 | WY Salient | 33 |
| Rykman Canyon-The Pinnacle | 1 | WY Salient | 35 |
| Hogsback | 3 | WY Salient | 37 |
| Eastern Bear Valley South | 1 | WY Salient | 38 |
| Sublette Flat North | 1 | WY Salient | 41 |
| Crawford Mountains-Saleratus Creek | 4 | WY Salient | 41 |
| Whitney Canyon | 1 | WY Salient | 43 |
| Eastern Bear Valley North | 1 | WY Salient | 52 |
| Porcupine Mountains | 4 | WY Salient | 58 |
| Bear River | Clustered | Clustered | 59 |
| Eastern Bear Lake | Layered | WY Salient | 63 |
| Western Bear Lake | 4 | WY Salient | 69 |
| Bear River Range North | 1 | WY Salient | 83 |
| Bear River Range South | 1 | WY Salient | 84 |
| Greys River | Clustered | Clustered | 89 |
| Grand Valley, Star Valley Section | 4 | WY Salient | 95 |
| East Cache | Layered | WY Salient | 101 |
| West Cache Full | Layered | WY Salient | 110 |
| East Gem Valley | 4 | WY Salient | 118 |
| Wasatch, Brigham City Segment | Wasatch | Wasatch | 121 |
| East Dayton-Oxford | 2 | WY Salient | 122 |
| Wasatch, Weber Segment | Wasatch | Wasatch | 122 |
| Wasatch, Collinston Segment | Wasatch | Wasatch | 124 |
| Continental | 1 | Central WY | 133 |
| Wasatch, Clarkston Mtn Segment | Wasatch | Wasatch | 134 |
| Leckie | 2 | Central WY | 139 |
| Wasatch, Malad City Segment | Wasatch | Wasatch | 142 |
| Grand Valley, Grand and Swan Sections | 4 | WY Salient | 146 |
| Wasatch, Salt Lake City Segment | Wasatch | Wasatch | 147 |
| Wasatch, Provo Segment | Wasatch | Wasatch | 169 |
| Hoback | 4 | Teton | 181 |


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**Table 2.6-3 Closest Fault Source-to-Site Distances**
**(Sheet 2 of 2)**

| | | | Site-To-Source Distance (km) |
|---|---|---|---|
| **Fault Source** | **Group** | **Region/Class** | **Kemmerer Unit 1** |
| Flattop | 2 | Central WY | 187 |
| Teton | 4 | Teton | 199 |
| Wasatch, Nephi Segment | Wasatch | Wasatch | 210 |
| South Granite Mountains | Layered | Central WY | 212 |
| Chicken Springs | Layered | Central WY | 214 |
| Wasatch, Levan Segment | Wasatch | Wasatch | 247 |
| North Granite Mountains | 2 | Central WY | 254 |
| Stagner Creek | 4 | Central WY | 262 |
| Upper Yellowstone Valley West | 3 | Teton | 267 |
| East Mount Sheridan | 4 | Teton | 269 |
| Eagle Bay-Buffalo Fork | 4 | Teton | 269 |
| Upper Yellowstone Valley East | 3 | Teton | 277 |
| Cedar Ridge | 1 | Central WY | 280 |
| Wasatch, Fayette Segment | Wasatch | Wasatch | 283 |
| Saratoga West | 2 | Central WY | 291 |
| Saratoga East | 1 | Central WY | 308 |


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**Table 2.6-4 Mean Reference-Rock Hazard Curves (AFE) by Spectral Period at the site (Sheet 1 of 3)**

| Amp. (g) | 0.01 sec | 0.02 sec | 0.03 sec | 0.04 sec | 0.05 sec | 0.075 sec |
|---|---|---|---|---|---|---|
| 0.0005 | - | - | - | - | - | - |
| 0.001 | - | - | - | - | - | - |
| 0.005 | - | - | - | - | - | - |
| 0.01 | 2.77E-02 | 2.77E-02 | 2.94E-02 | 3.19E-02 | 3.45E-02 | 4.09E-02 |
| 0.015 | 1.63E-02 | 1.63E-02 | 1.77E-02 | 1.96E-02 | 2.17E-02 | 2.67E-02 |
| 0.03 | 5.65E-03 | 5.70E-03 | 6.41E-03 | 7.48E-03 | 8.65E-03 | 1.15E-02 |
| 0.05 | 2.32E-03 | 2.35E-03 | 2.74E-03 | 3.30E-03 | 3.95E-03 | 5.54E-03 |
| 0.075 | 1.10E-03 | 1.12E-03 | 1.34E-03 | 1.66E-03 | 2.04E-03 | 2.97E-03 |
| 0.1 | 6.16E-04 | 6.28E-04 | 7.68E-04 | 9.78E-04 | 1.23E-03 | 1.86E-03 |
| 0.15 | 2.55E-04 | 2.61E-04 | 3.32E-04 | 4.40E-04 | 5.77E-04 | 9.26E-04 |
| 0.3 | 4.44E-05 | 4.57E-05 | 6.17E-05 | 8.85E-05 | 1.26E-04 | 2.34E-04 |
| 0.5 | 1.04E-05 | 1.08E-05 | 1.50E-05 | 2.24E-05 | 3.34E-05 | 6.85E-05 |
| 0.75 | 3.08E-06 | 3.19E-06 | 4.53E-06 | 6.96E-06 | 1.07E-05 | 2.32E-05 |
| 1 | 1.24E-06 | 1.29E-06 | 1.87E-06 | 2.94E-06 | 4.63E-06 | 1.04E-05 |
| 1.5 | 3.19E-07 | 3.33E-07 | 5.00E-07 | 8.19E-07 | 1.34E-06 | 3.22E-06 |
| 3 | 2.07E-08 | 2.18E-08 | 3.54E-08 | 6.44E-08 | 1.17E-07 | 3.37E-07 |
| 5 | 2.03E-09 | 2.15E-09 | 3.72E-09 | 7.35E-09 | 1.46E-08 | 4.95E-08 |
| 7.5 | 2.61E-10 | 2.78E-10 | 5.08E-10 | 1.07E-09 | 2.29E-09 | 8.89E-09 |
| 10 | - | - | - | 2.48E-10 | 5.48E-10 | 2.34E-09 |
| 15 | - | - | - | - | - | 3.01E-10 |


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**Table 2.6-4 Mean Reference-Rock Hazard Curves (AFE) by Spectral Period at the site (Sheet 2 of 3)**

| Amp. (g) | 0.15 sec | 0.2 sec | 0.3 sec | 0.4 sec | 0.5 sec | 0.75 sec |
|---|---|---|---|---|---|---|
| 0.0005 | - | - | - | - | - | - |
| 0.001 | - | - | - | - | - | - |
| 0.005 | - | - | - | - | - | - |
| 0.01 | 5.19E-02 | 5.64E-02 | 5.67E-02 | 5.23E-02 | 4.68E-02 | 3.34E-02 |
| 0.015 | 3.45E-02 | 3.69E-02 | 3.60E-02 | 3.23E-02 | 2.82E-02 | 1.91E-02 |
| 0.03 | 1.53E-02 | 1.60E-02 | 1.47E-02 | 1.24E-02 | 1.03E-02 | 6.26E-03 |
| 0.05 | 7.57E-03 | 7.77E-03 | 6.75E-03 | 5.43E-03 | 4.32E-03 | 2.42E-03 |
| 0.075 | 4.12E-03 | 4.15E-03 | 3.46E-03 | 2.68E-03 | 2.06E-03 | 1.08E-03 |
| 0.1 | 2.60E-03 | 2.59E-03 | 2.09E-03 | 1.57E-03 | 1.19E-03 | 5.90E-04 |
| 0.15 | 1.33E-03 | 1.30E-03 | 1.00E-03 | 7.24E-04 | 5.25E-04 | 2.39E-04 |
| 0.3 | 3.67E-04 | 3.50E-04 | 2.46E-04 | 1.63E-04 | 1.09E-04 | 4.14E-05 |
| 0.5 | 1.16E-04 | 1.09E-04 | 7.19E-05 | 4.50E-05 | 2.86E-05 | 9.76E-06 |
| 0.75 | 4.07E-05 | 3.81E-05 | 2.43E-05 | 1.47E-05 | 9.13E-06 | 2.91E-06 |
| 1 | 1.87E-05 | 1.74E-05 | 1.09E-05 | 6.47E-06 | 3.95E-06 | 1.20E-06 |
| 1.5 | 5.93E-06 | 5.48E-06 | 3.35E-06 | 1.95E-06 | 1.16E-06 | 3.22E-07 |
| 3 | 6.85E-07 | 6.26E-07 | 3.60E-07 | 1.97E-07 | 1.10E-07 | 2.51E-08 |
| 5 | 1.12E-07 | 1.02E-07 | 5.57E-08 | 2.91E-08 | 1.54E-08 | 3.01E-09 |
| 7.5 | 2.21E-08 | 2.02E-08 | 1.06E-08 | 5.37E-09 | 2.73E-09 | 4.74E-10 |
| 10 | 6.26E-09 | 5.72E-09 | 2.96E-09 | 1.47E-09 | 7.28E-10 | - |
| 15 | 8.91E-10 | 8.22E-10 | 4.16E-10 | 2.02E-10 | 9.70E-11 | - |


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**Table 2.6-4 Mean Reference-Rock Hazard Curves (AFE) by Spectral Period at the site (Sheet 3 of 3)**

| Amp. (g) | 1.5 sec | 2 sec | 3 sec | 5 sec | 7.5 sec | 10 sec |
|---|---|---|---|---|---|---|
| 0.0005 | - | - | - | 7.01E-02 | 3.73E-02 | 2.34E-02 |
| 0.001 | 1.69E-01 | 1.25E-01 | 8.13E-02 | 3.63E-02 | 1.72E-02 | 1.01E-02 |
| 0.005 | 3.49E-02 | 2.36E-02 | 1.20E-02 | 3.33E-03 | 1.14E-03 | 5.55E-04 |
| 0.01 | 1.34E-02 | 8.21E-03 | 3.44E-03 | 7.73E-04 | 2.26E-04 | 9.39E-05 |
| 0.015 | 6.71E-03 | 3.83E-03 | 1.43E-03 | 2.89E-04 | 7.38E-05 | 2.72E-05 |
| 0.03 | 1.68E-03 | 8.56E-04 | 2.65E-04 | 3.87E-05 | 7.25E-06 | 2.17E-06 |
| 0.05 | 5.26E-04 | 2.41E-04 | 6.12E-05 | 6.74E-06 | 1.00E-06 | 2.56E-07 |
| 0.075 | 1.94E-04 | 7.95E-05 | 1.67E-05 | 1.45E-06 | 1.79E-07 | 4.01E-08 |
| 0.1 | 8.89E-05 | 3.32E-05 | 6.11E-06 | 4.39E-07 | 4.70E-08 | 9.60E-09 |
| 0.15 | 2.76E-05 | 9.15E-06 | 1.40E-06 | 7.77E-08 | 6.74E-09 | 1.20E-09 |
| 0.3 | 3.20E-06 | 8.78E-07 | 9.17E-08 | 2.65E-09 | 1.52E-10 | 2.10E-11 |
| 0.5 | 5.91E-07 | 1.36E-07 | 9.72E-09 | 1.67E-10 | - | - |
| 0.75 | 1.39E-07 | 2.69E-08 | 1.35E-09 | - | - | - |
| 1 | 4.64E-08 | 7.77E-09 | 3.01E-10 | - | - | - |
| 1.5 | 7.86E-09 | 1.05E-09 | - | - | - | - |
| 3 | 2.87E-10 | 2.56E-11 | - | - | - | - |
| 5 | - | - | - | - | - | - |
| 7.5 | - | - | - | - | - | - |
| 10 | - | - | - | - | - | - |
| 15 | - | - | - | - | - | - |


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**Table 2.6-5 Mean and Fractile Hazard Curves (AFE) for 0.01 second Reference Rock Hazard at the site**

| Amp. (g) | Mean | 0.05 | 0.16 | 0.50 | 0.84 | 0.95 |
|---|---|---|---|---|---|---|
| 0.0100 | 2.77E-02 | 1.08E-02 | 1.61E-02 | 2.73E-02 | 4.26E-02 | 5.43E-02 |
| 0.0237 | 8.34E-03 | 2.19E-03 | 3.66E-03 | 7.25E-03 | 1.36E-02 | 2.03E-02 |
| 0.0562 | 1.89E-03 | 3.58E-04 | 6.36E-04 | 1.45E-03 | 3.29E-03 | 4.96E-03 |
| 0.133 | 3.42E-04 | 3.48E-05 | 7.67E-05 | 2.28E-04 | 6.13E-04 | 1.01E-03 |
| 0.316 | 3.89E-05 | 2.08E-06 | 5.96E-06 | 2.15E-05 | 6.92E-05 | 1.39E-04 |
| 0.750 | 3.12E-06 | 5.22E-08 | 2.07E-07 | 1.24E-06 | 5.67E-06 | 1.30E-05 |
| 1.78 | 1.72E-07 | 2.95E-10 | 2.10E-09 | 3.03E-08 | 2.75E-07 | 8.79E-07 |
| 4.22 | 4.64E-09 | - | 1.14E-10 | 2.74E-10 | 4.75E-09 | 2.18E-08 |
| 10.0 | 5.52E-11 | - | - | - | 1.33E-10 | 2.75E-10 |


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**Table 2.6-6 Mean and Fractile Hazard Curves (AFE) for 0.1 second Reference Rock Hazard at the site**

| Amp. (g) | Mean | 0.05 | 0.16 | 0.50 | 0.84 | 0.95 |
|---|---|---|---|---|---|---|
| 0.0100 | 4.53E-02 | 2.27E-02 | 3.13E-02 | 4.63E-02 | 6.65E-02 | 8.10E-02 |
| 0.0259 | 1.61E-02 | 6.06E-03 | 8.72E-03 | 1.57E-02 | 2.47E-02 | 3.25E-02 |
| 0.0669 | 4.25E-03 | 1.12E-03 | 1.88E-03 | 3.61E-03 | 6.93E-03 | 1.02E-02 |
| 0.173 | 9.03E-04 | 1.55E-04 | 2.89E-04 | 7.03E-04 | 1.56E-03 | 2.44E-03 |
| 0.447 | 1.26E-04 | 1.04E-05 | 2.36E-05 | 8.05E-05 | 2.24E-04 | 4.14E-04 |
| 1.16 | 9.82E-06 | 3.47E-07 | 1.13E-06 | 4.81E-06 | 1.79E-05 | 3.45E-05 |
| 2.99 | 5.42E-07 | 2.04E-09 | 1.48E-08 | 1.57E-07 | 8.95E-07 | 2.42E-06 |
| 7.73 | 1.40E-08 | 1.00E-12 | 1.27E-10 | 9.77E-10 | 1.68E-08 | 7.32E-08 |
| 20.0 | 1.35E-10 | - | - | - | 1.76E-10 | 5.25E-10 |


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**Table 2.6-7 Mean and Fractile Hazard Curves (AFE) for 1 second Reference Rock Hazard at the site**

| Amp. (g) | Mean | 0.05 | 0.16 | 0.50 | 0.84 | 0.95 |
|---|---|---|---|---|---|---|
| 0.0100 | 2.37E-02 | 9.01E-03 | 1.39E-02 | 2.39E-02 | 3.73E-02 | 4.72E-02 |
| 0.0237 | 5.98E-03 | 1.48E-03 | 2.52E-03 | 5.17E-03 | 9.87E-03 | 1.49E-02 |
| 0.0562 | 1.07E-03 | 1.75E-04 | 3.30E-04 | 8.33E-04 | 1.80E-03 | 2.95E-03 |
| 0.133 | 1.50E-04 | 1.48E-05 | 3.26E-05 | 9.61E-05 | 2.57E-04 | 4.64E-04 |
| 0.316 | 1.39E-05 | 7.68E-07 | 1.96E-06 | 7.15E-06 | 2.31E-05 | 5.25E-05 |
| 0.750 | 9.80E-07 | 1.62E-08 | 6.67E-08 | 4.05E-07 | 1.67E-06 | 4.11E-06 |
| 1.78 | 4.80E-08 | 1.75E-10 | 8.18E-10 | 9.85E-09 | 7.86E-08 | 2.22E-07 |
| 4.22 | 1.28E-09 | - | - | 1.74E-10 | 1.42E-09 | 5.93E-09 |
| 10.0 | 1.72E-11 | - | - | - | 1.13E-10 | 1.66E-10 |


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**Table 2.6-8 Mean and Fractile Hazard Curves (AFE) for 10 second Reference Rock Hazard at the site**

| Amp. (g) | Mean | 0.05 | 0.16 | 0.50 | 0.84 | 0.95 |
|---|---|---|---|---|---|---|
| 0.000100 | 9.34E-02 | 6.58E-02 | 7.40E-02 | 1.02E-01 | 1.36E-01 | 1.48E-01 |
| 0.00029 | 4.08E-02 | 2.19E-02 | 2.70E-02 | 4.23E-02 | 6.27E-02 | 7.03E-02 |
| 0.000841 | 1.29E-02 | 4.22E-03 | 6.14E-03 | 1.21E-02 | 2.15E-02 | 2.59E-02 |
| 0.00244 | 2.47E-03 | 3.30E-04 | 6.60E-04 | 2.05E-03 | 4.47E-03 | 6.21E-03 |
| 0.00707 | 2.51E-04 | 6.38E-06 | 1.96E-05 | 1.24E-04 | 4.58E-04 | 8.63E-04 |
| 0.0205 | 9.57E-06 | 3.69E-08 | 1.51E-07 | 1.59E-06 | 1.43E-05 | 4.28E-05 |
| 0.0595 | 1.20E-07 | 1.22E-10 | 2.91E-10 | 4.56E-09 | 7.62E-08 | 3.96E-07 |
| 0.172 | 5.69E-10 | - | - | 1.12E-10 | 1.97E-10 | 8.95E-10 |
| 0.500 | - | - | - | - | - | - |


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**Table 2.6-9 Mean Reference Rock UHRS (g) for 11 AFEs at the site**

| T (sec) | 1E-2/yr | 3E-3/yr | 1E-3/yr | 4E-4/yr | 1E-4/yr | 4E-5/yr | 1E-5/yr | 4E-6/yr | 1E-6/yr | 1E-7/yr | 1E-8/yr |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 0.01 | 2.09E-02 | 4.34E-02 | 7.87E-02 | 1.40E-01 | 2.21E-01 | 3.46E-01 | 5.07E-01 | 7.56E-01 | 1.07E+00 | 2.04E+00 | 3.55E+00 |
| 0.02 | 2.10E-02 | 4.37E-02 | 7.93E-02 | 1.41E-01 | 2.24E-01 | 3.50E-01 | 5.13E-01 | 7.65E-01 | 1.08E+00 | 2.06E+00 | 3.59E+00 |
| 0.03 | 2.24E-02 | 4.74E-02 | 8.72E-02 | 1.57E-01 | 2.49E-01 | 3.91E-01 | 5.76E-01 | 8.61E-01 | 1.21E+00 | 2.32E+00 | 4.05E+00 |
| 0.04 | 2.47E-02 | 5.29E-02 | 9.88E-02 | 1.80E-01 | 2.86E-01 | 4.49E-01 | 6.62E-01 | 9.94E-01 | 1.41E+00 | 2.69E+00 | 4.67E+00 |
| 0.05 | 2.72E-02 | 5.92E-02 | 1.13E-01 | 2.08E-01 | 3.31E-01 | 5.20E-01 | 7.68E-01 | 1.15E+00 | 1.64E+00 | 3.13E+00 | 5.46E+00 |
| 0.075 | 3.31E-02 | 7.46E-02 | 1.44E-01 | 2.67E-01 | 4.29E-01 | 6.82E-01 | 1.01E+00 | 1.53E+00 | 2.17E+00 | 4.20E+00 | 7.30E+00 |
| 0.1 | 3.70E-02 | 8.35E-02 | 1.62E-01 | 3.04E-01 | 4.87E-01 | 7.74E-01 | 1.15E+00 | 1.74E+00 | 2.48E+00 | 4.76E+00 | 8.33E+00 |
| 0.15 | 4.11E-02 | 9.15E-02 | 1.77E-01 | 3.32E-01 | 5.32E-01 | 8.41E-01 | 1.25E+00 | 1.88E+00 | 2.68E+00 | 5.15E+00 | 9.02E+00 |
| 0.2 | 4.21E-02 | 9.14E-02 | 1.74E-01 | 3.24E-01 | 5.18E-01 | 8.20E-01 | 1.21E+00 | 1.83E+00 | 2.61E+00 | 5.02E+00 | 8.84E+00 |
| 0.3 | 3.88E-02 | 8.13E-02 | 1.50E-01 | 2.73E-01 | 4.38E-01 | 6.94E-01 | 1.03E+00 | 1.55E+00 | 2.21E+00 | 4.30E+00 | 7.61E+00 |
| 0.4 | 3.43E-02 | 7.03E-02 | 1.28E-01 | 2.30E-01 | 3.67E-01 | 5.82E-01 | 8.61E-01 | 1.30E+00 | 1.85E+00 | 3.63E+00 | 6.50E+00 |
| 0.5 | 3.05E-02 | 6.11E-02 | 1.09E-01 | 1.96E-01 | 3.11E-01 | 4.91E-01 | 7.27E-01 | 1.09E+00 | 1.57E+00 | 3.08E+00 | 5.56E+00 |
| 0.75 | 2.27E-02 | 4.48E-02 | 7.79E-02 | 1.36E-01 | 2.15E-01 | 3.38E-01 | 4.96E-01 | 7.42E-01 | 1.06E+00 | 2.08E+00 | 3.78E+00 |
| 1 | 1.76E-02 | 3.39E-02 | 5.79E-02 | 9.93E-02 | 1.55E-01 | 2.41E-01 | 3.53E-01 | 5.25E-01 | 7.42E-01 | 1.46E+00 | 2.63E+00 |
| 1.5 | 1.20E-02 | 2.27E-02 | 3.79E-02 | 6.29E-02 | 9.58E-02 | 1.46E-01 | 2.09E-01 | 3.06E-01 | 4.27E-01 | 8.21E-01 | 1.43E+00 |
| 2 | 8.87E-03 | 1.70E-02 | 2.81E-02 | 4.61E-02 | 6.91E-02 | 1.03E-01 | 1.46E-01 | 2.10E-01 | 2.89E-01 | 5.40E-01 | 9.44E-01 |
| 3 | 5.60E-03 | 1.07E-02 | 1.76E-02 | 2.87E-02 | 4.25E-02 | 6.26E-02 | 8.69E-02 | 1.22E-01 | 1.64E-01 | 2.94E-01 | 4.97E-01 |
| 5 | 2.67E-03 | 5.28E-03 | 8.93E-03 | 1.48E-02 | 2.22E-02 | 3.24E-02 | 4.51E-02 | 6.19E-02 | 8.24E-02 | 1.42E-01 | 2.30E-01 |
| 7.5 | 1.48E-03 | 3.07E-03 | 5.33E-03 | 8.96E-03 | 1.36E-02 | 2.00E-02 | 2.75E-02 | 3.80E-02 | 5.00E-02 | 8.54E-02 | 1.38E-01 |
| 10 | 1.01E-03 | 2.18E-03 | 3.87E-03 | 6.54E-03 | 9.78E-03 | 1.46E-02 | 2.00E-02 | 2.77E-02 | 3.63E-02 | 6.14E-02 | 9.92E-02 |


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**Table 2.6-10 Mean Reference M, R, and ε Deaggregation of Reference Rock Hazard for 11 AFEs at the site (Bold Text in LF Row Indicates that Contributions from Distances less than 100 km are Ignored)**

| T (sec) | Value | 1E-2 | 4E-3 | 1E-3 | 4E-4 | 1E-4 | 4E-5 | 1E-5 | 4E-6 | 1E-6 | 1E-7 | 1E-8 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 0.01 | M | 6.5 | 6.6 | 6.7 | 6.7 | 6.7 | 6.6 | 6.5 | 6.5 | 6.4 | 6.5 | 6.5 |
| | R (km) | 84 | 58 | 40 | 28 | 22 | 17 | 13 | 9 | 7 | 5 | 4 |
| | ε | -0.3 | 0.0 | 0.2 | 0.5 | 0.8 | 1.0 | 1.1 | 1.3 | 1.4 | 1.9 | 2.4 |
| 0.1 | M | 6.4 | 6.5 | 6.6 | 6.7 | 6.6 | 6.5 | 6.5 | 6.4 | 6.4 | 6.5 | 6.5 |
| | R (km) | 72 | 49 | 34 | 25 | 21 | 16 | 12 | 9 | 7 | 5 | 4 |
| | ε | -0.5 | -0.2 | 0.0 | 0.4 | 0.7 | 0.9 | 1.1 | 1.2 | 1.4 | 1.9 | 2.4 |
| 1 | M | 6.7 | 6.8 | 6.9 | 6.9 | 6.9 | 6.9 | 6.8 | 6.8 | 6.8 | 6.8 | 6.8 |
| | R (km) | 150 | 120 | 86 | 56 | 39 | 28 | 20 | 14 | 10 | 6 | 4 |
| | ε | 0.0 | 0.3 | 0.5 | 0.7 | 1.0 | 1.2 | 1.3 | 1.4 | 1.5 | 1.9 | 2.4 |
| 10 | M | 7.2 | 7.4 | 7.5 | 7.5 | 7.6 | 7.6 | 7.6 | 7.6 | 7.6 | 7.7 | 7.8 |
| | R (km) | 394 | 528 | 650 | 734 | 757 | 762 | 780 | 827 | 888 | 1060 | 1230 |
| | ε | -0.4 | 0.0 | 0.4 | 0.7 | 1.1 | 1.4 | 1.7 | 2.1 | 2.3 | 2.9 | 3.4 |
| LF | M | **6.8** | **7.0** | **7.0** | **7.1** | 6.9 | 6.9 | 6.8 | 6.8 | 6.8 | 6.8 | 6.8 |
| | R (km) | **250** | **276** | **308** | **351** | 39 | 28 | 20 | 14 | 10 | 6 | 4 |
| | ε | 0.0 | 0.3 | 0.5 | 0.7 | 1.0 | 1.2 | 1.3 | 1.4 | 1.5 | 1.9 | 2.4 |


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**Table 2.6-11 Percent Contribution of Seismic Sources to 1 Sec Mean Reference-Rock Hazard for Three AFEs at the site**

(Included Sources Capture 99% of Total Hazard, See Figure 2.6-61)

| Seismic Source Name | Source Type | Contribution to Total Hazard |
|---|---|---|
| ***0.155g at Total AFE 1E-4/yr*** | | |
| Rock Creek | Fault | 58% |
| Colorado Plateau-Wyoming Craton | Background | 16% |
| Intermountain Seismic Belt | Background | 15% |
| Bear River | Fault | 2% |
| Wasatch | Fault | 1.4% |
| Eastern Bear Lake Central plus South | Fault | 1.4% |
| Spring Creek-North Bridger | Fault | 0.9% |
| Crawford Mountains - Saleratus Creek | Fault | 0.9% |
| Greys River | Fault | 0.9% |
| NMFS | Fault System | 0.6% |
| Grand Valley Star | Fault | 0.4% |
| Eastern Bear Lake Full | Fault | 0.3% |
| Trona | Background | 0.3% |
| Sublette Flat North | Fault | 0.3% |
| Rykman Creek-Pinnacle | Fault | 0.2% |
| Eastern Bear Valley South | Fault | 0.2% |
| Whitney Canyon | Fault | 0.2% |
| Hogsback | Fault | 0.2% |
| Sublette Flat South | Fault | 0.2% |
| Porcupine Mountains | Fault | 0.1% |
| ***0.353g at Total AFE 1E-5/yr*** | | |
| Rock Creek | Fault | 43% |
| Colorado Plateau-Wyoming Craton | Background | 29% |
| Intermountain Seismic Belt | Background | 26% |
| Bear River | Fault | 0.4% |
| Spring Creek-North Bridger | Fault | 0.4% |
| Trona | Background | 0.4% |
| Eastern Bear Lake Central plus South | Fault | 0.3% |
| Crawford Mountains - Saleratus Creek | Fault | 0.3% |
| Wasatch | Fault | 0.3% |
| Greys River | Fault | 0.2% |
| ***0.742g at Total AFE 1E-6/yr*** | | |
| Colorado Plateau-Wyoming Craton | Background | 47% |
| Intermountain Seismic Belt | Background | 32% |
| Rock Creek | Fault | 19% |
| Trona | Background | 0.6% |


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**Table 2.6-12 Modal M and R Deaggregation of Reference-Rock Hazard for 11 AFEs at the site**

| T (sec) | Value | 1E-2 | 4E-3 | 1E-3 | 4E-4 | 1E-4 | 4E-5 | 1E-5 | 4E-6 | 1E-6 | 1E-7 | 1E-8 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 0.01 | M | 6.6 | 7.1 | 7.1 | 7.1 | 7.1 | 7.1 | 7.1 | 6.6 | 6.6 | 6.6 | 6.6 |
| | R (km) | 110 | 22.5 | 22.5 | 22.5 | 22.5 | 22.5 | 22.5 | 2.5 | 2.5 | 2.5 | 2.5 |
| 0.1 | M | 6.6 | 7.1 | 7.1 | 7.1 | 7.1 | 7.1 | 7.1 | 6.6 | 6.6 | 6.6 | 6.6 |
| | R (km) | 110 | 22.5 | 22.5 | 22.5 | 22.5 | 22.5 | 22.5 | 2.5 | 2.5 | 2.5 | 2.5 |
| 1 | M | 6.6 | 7.1 | 7.1 | 7.1 | 7.1 | 7.1 | 7.1 | 6.6 | 6.6 | 6.6 | 6.6 |
| | R (km) | 110 | 22.5 | 22.5 | 22.5 | 22.5 | 22.5 | 22.5 | 2.5 | 2.5 | 2.5 | 2.5 |
| 10 | M | 7.1 | 7.9 | 7.9 | 7.9 | 7.9 | 7.9 | 7.9 | 7.9 | 7.9 | 7.9 | 7.9 |
| | R (km) | 130 | 1825 | 1825 | 1825 | 1825 | 1825 | 1825 | 1825 | 1825 | 1825 | 1825 |


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**Table 2.6-13 Mean PSHA Control Point UHRS (g) for 11 AFEs at the site**

| T (sec) | 1E-2/yr | 3E-3/yr | 1E-3/yr | 4E-4/yr | 1E-4/yr | 4E-5/yr | 1E-5/yr | 4E-6/yr | 1E-6/yr | 1E-7/yr | 1E-8/yr |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 0.01 | 1.44E-02 | 3.05E-02 | 5.48E-02 | 9.78E-02 | 1.57E-01 | 2.49E-01 | 3.68E-01 | 5.51E-01 | 7.71E-01 | 1.47E+00 | 2.50E+00 |
| 0.02 | 1.45E-02 | 3.11E-02 | 5.60E-02 | 1.00E-01 | 1.61E-01 | 2.56E-01 | 3.77E-01 | 5.63E-01 | 7.86E-01 | 1.51E+00 | 2.57E+00 |
| 0.03 | 1.59E-02 | 3.50E-02 | 6.43E-02 | 1.16E-01 | 1.88E-01 | 2.99E-01 | 4.41E-01 | 6.57E-01 | 9.27E-01 | 1.74E+00 | 2.97E+00 |
| 0.04 | 1.86E-02 | 4.18E-02 | 7.85E-02 | 1.41E-01 | 2.32E-01 | 3.75E-01 | 5.61E-01 | 8.36E-01 | 1.18E+00 | 2.22E+00 | 3.77E+00 |
| 0.05 | 2.09E-02 | 4.86E-02 | 9.30E-02 | 1.72E-01 | 2.85E-01 | 4.69E-01 | 7.01E-01 | 1.05E+00 | 1.49E+00 | 2.84E+00 | 4.79E+00 |
| 0.075 | 2.27E-02 | 5.41E-02 | 1.06E-01 | 2.00E-01 | 3.37E-01 | 5.53E-01 | 8.28E-01 | 1.25E+00 | 1.78E+00 | 3.44E+00 | 5.87E+00 |
| 0.1 | 2.27E-02 | 5.36E-02 | 1.04E-01 | 1.98E-01 | 3.25E-01 | 5.32E-01 | 7.97E-01 | 1.21E+00 | 1.72E+00 | 3.31E+00 | 5.79E+00 |
| 0.15 | 2.54E-02 | 5.88E-02 | 1.13E-01 | 2.11E-01 | 3.48E-01 | 5.65E-01 | 8.43E-01 | 1.26E+00 | 1.76E+00 | 3.33E+00 | 5.71E+00 |
| 0.2 | 2.82E-02 | 6.36E-02 | 1.20E-01 | 2.21E-01 | 3.61E-01 | 5.84E-01 | 8.61E-01 | 1.29E+00 | 1.81E+00 | 3.46E+00 | 6.00E+00 |
| 0.3 | 2.70E-02 | 5.66E-02 | 1.05E-01 | 1.91E-01 | 3.09E-01 | 4.94E-01 | 7.37E-01 | 1.11E+00 | 1.59E+00 | 3.11E+00 | 5.55E+00 |
| 0.4 | 2.25E-02 | 4.71E-02 | 8.52E-02 | 1.53E-01 | 2.47E-01 | 3.94E-01 | 5.87E-01 | 8.95E-01 | 1.28E+00 | 2.54E+00 | 4.70E+00 |
| 0.5 | 1.99E-02 | 4.00E-02 | 7.16E-02 | 1.28E-01 | 2.06E-01 | 3.28E-01 | 4.92E-01 | 7.39E-01 | 1.05E+00 | 2.10E+00 | 3.94E+00 |
| 0.75 | 1.58E-02 | 3.15E-02 | 5.43E-02 | 9.48E-02 | 1.52E-01 | 2.42E-01 | 3.59E-01 | 5.40E-01 | 7.65E-01 | 1.52E+00 | 2.80E+00 |
| 1 | 1.26E-02 | 2.44E-02 | 4.23E-02 | 7.28E-02 | 1.14E-01 | 1.82E-01 | 2.63E-01 | 3.87E-01 | 5.49E-01 | 1.10E+00 | 2.01E+00 |
| 1.5 | 9.76E-03 | 1.88E-02 | 3.13E-02 | 5.14E-02 | 7.84E-02 | 1.21E-01 | 1.74E-01 | 2.53E-01 | 3.55E-01 | 6.97E-01 | 1.23E+00 |
| 2 | 6.72E-03 | 1.30E-02 | 2.17E-02 | 3.58E-02 | 5.40E-02 | 8.15E-02 | 1.16E-01 | 1.66E-01 | 2.29E-01 | 4.31E-01 | 7.56E-01 |
| 3 | 4.11E-03 | 7.83E-03 | 1.28E-02 | 2.09E-02 | 3.16E-02 | 4.66E-02 | 6.54E-02 | 9.18E-02 | 1.23E-01 | 2.21E-01 | 3.75E-01 |
| 5 | 2.19E-03 | 4.41E-03 | 7.44E-03 | 1.24E-02 | 1.83E-02 | 2.73E-02 | 3.78E-02 | 5.26E-02 | 7.00E-02 | 1.22E-01 | 1.99E-01 |
| 7.5 | 1.34E-03 | 2.80E-03 | 4.86E-03 | 8.26E-03 | 1.28E-02 | 1.88E-02 | 2.62E-02 | 3.63E-02 | 4.82E-02 | 8.29E-02 | 1.36E-01 |
| 10 | 9.83E-04 | 2.13E-03 | 3.78E-03 | 6.36E-03 | 9.71E-03 | 1.47E-02 | 2.04E-02 | 2.83E-02 | 3.76E-02 | 6.48E-02 | 1.05E-01 |


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**Table 2.6-14 Mean V/H Ratios and PSHA Control Point Vertical UHRS for Three AFEs at the site**

| T (sec) | Mean V/H Ratios | | | Mean Vertical UHRS (g) | | |
|---|---|---|---|---|---|---|
| | 1E-4/yr | 1E-5/yr | 1E-6/yr | 1E-4/yr | 1E-5/yr | 1E-6/yr |
| 0.01 | 0.838 | 0.888 | 0.919 | 1.31E-01 | 3.27E-01 | 7.09E-01 |
| 0.02 | 0.838 | 0.888 | 0.919 | 1.35E-01 | 3.35E-01 | 7.23E-01 |
| 0.03 | 0.921 | 0.988 | 1.031 | 1.73E-01 | 4.36E-01 | 9.56E-01 |
| 0.04 | 0.946 | 1.034 | 1.091 | 2.19E-01 | 5.80E-01 | 1.28E+00 |
| 0.05 | 0.959 | 1.063 | 1.130 | 2.73E-01 | 7.45E-01 | 1.69E+00 |
| 0.075 | 0.879 | 0.959 | 1.008 | 2.96E-01 | 7.93E-01 | 1.80E+00 |
| 0.1 | 0.809 | 0.857 | 0.884 | 2.63E-01 | 6.83E-01 | 1.52E+00 |
| 0.15 | 0.754 | 0.768 | 0.774 | 2.62E-01 | 6.48E-01 | 1.36E+00 |
| 0.2 | 0.744 | 0.738 | 0.732 | 2.69E-01 | 6.35E-01 | 1.32E+00 |
| 0.3 | 0.821 | 0.806 | 0.797 | 2.54E-01 | 5.94E-01 | 1.26E+00 |
| 0.4 | 0.892 | 0.871 | 0.859 | 2.20E-01 | 5.11E-01 | 1.10E+00 |
| 0.5 | 0.935 | 0.909 | 0.895 | 1.93E-01 | 4.47E-01 | 9.41E-01 |
| 0.75 | 0.916 | 0.886 | 0.869 | 1.39E-01 | 3.18E-01 | 6.65E-01 |
| 1 | 0.906 | 0.876 | 0.859 | 1.04E-01 | 2.31E-01 | 4.72E-01 |
| 1.5 | 0.909 | 0.879 | 0.862 | 7.12E-02 | 1.53E-01 | 3.06E-01 |
| 2 | 0.900 | 0.870 | 0.854 | 4.86E-02 | 1.01E-01 | 1.95E-01 |
| 3 | 0.921 | 0.891 | 0.874 | 2.91E-02 | 5.82E-02 | 1.08E-01 |
| 5 | 1.009 | 0.976 | 0.957 | 1.85E-02 | 3.69E-02 | 6.70E-02 |
| 7.5 | 1.030 | 0.996 | 0.977 | 1.32E-02 | 2.61E-02 | 4.71E-02 |
| 10 | 1.046 | 1.012 | 0.992 | 1.02E-02 | 2.06E-02 | 3.74E-02 |


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**Table 2.6-15 Horizontal and Vertical Spectral Accelerations for GMRS Horizon and SSE**
(see Figure 2.6-86)

| f (Hz) | $SA_H$ (g) | $SA_V$ (g) | f (Hz) | $SA_H$ (g) | $SA_V$ (g) | f (Hz) | $SA_H$ (g) | $SA_V$ (g) |
|---|---|---|---|---|---|---|---|---|
| 0.10 | 0.011 | 0.008 | 3.60 | 0.479 | 0.284 | 15.00 | 0.729 | 0.611 |
| 0.20 | 0.020 | 0.016 | 3.80 | 0.502 | 0.295 | 16.00 | 0.716 | 0.608 |
| 0.30 | 0.030 | 0.023 | 4.00 | 0.523 | 0.308 | 17.00 | 0.695 | 0.596 |
| 0.40 | 0.044 | 0.033 | 4.20 | 0.544 | 0.321 | 18.00 | 0.671 | 0.580 |
| 0.50 | 0.061 | 0.046 | 4.40 | 0.562 | 0.332 | 20.00 | 0.611 | 0.537 |
| 0.60 | 0.080 | 0.062 | 4.60 | 0.580 | 0.343 | 22.00 | 0.550 | 0.475 |
| 0.70 | 0.096 | 0.076 | 4.80 | 0.595 | 0.352 | 25.00 | 0.476 | 0.402 |
| 0.80 | 0.110 | 0.087 | 5.00 | 0.610 | 0.361 | 28.00 | 0.422 | 0.346 |
| 0.90 | 0.124 | 0.098 | 5.25 | 0.625 | 0.374 | 31.00 | 0.384 | 0.307 |
| 1.00 | 0.139 | 0.110 | 5.50 | 0.639 | 0.385 | 34.00 | 0.356 | 0.279 |
| 1.10 | 0.155 | 0.118 | 5.75 | 0.650 | 0.395 | 37.00 | 0.339 | 0.258 |
| 1.20 | 0.171 | 0.127 | 6.00 | 0.660 | 0.403 | 40.00 | 0.327 | 0.244 |
| 1.30 | 0.186 | 0.134 | 6.25 | 0.668 | 0.411 | 43.00 | 0.320 | 0.234 |
| 1.40 | 0.200 | 0.141 | 6.50 | 0.675 | 0.417 | 46.00 | 0.315 | 0.226 |
| 1.50 | 0.212 | 0.146 | 6.75 | 0.680 | 0.425 | 49.00 | 0.312 | 0.220 |
| 1.60 | 0.224 | 0.151 | 7.00 | 0.684 | 0.434 | 52.00 | 0.309 | 0.217 |
| 1.70 | 0.235 | 0.156 | 7.25 | 0.687 | 0.443 | 55.00 | 0.307 | 0.216 |
| 1.80 | 0.246 | 0.161 | 7.50 | 0.689 | 0.451 | 58.00 | 0.306 | 0.215 |
| 1.90 | 0.258 | 0.166 | 7.75 | 0.691 | 0.458 | 61.00 | 0.305 | 0.214 |
| 2.00 | 0.269 | 0.171 | 8.00 | 0.692 | 0.465 | 64.00 | 0.304 | 0.213 |
| 2.10 | 0.281 | 0.178 | 8.50 | 0.695 | 0.477 | 67.00 | 0.303 | 0.213 |
| 2.20 | 0.293 | 0.184 | 9.00 | 0.698 | 0.488 | 70.00 | 0.303 | 0.212 |
| 2.30 | 0.306 | 0.192 | 9.50 | 0.701 | 0.499 | 73.00 | 0.302 | 0.212 |
| 2.40 | 0.319 | 0.199 | 10.00 | 0.705 | 0.510 | 76.00 | 0.302 | 0.212 |
| 2.50 | 0.332 | 0.206 | 10.50 | 0.711 | 0.526 | 79.00 | 0.302 | 0.212 |
| 2.60 | 0.346 | 0.214 | 11.00 | 0.716 | 0.542 | 82.00 | 0.302 | 0.212 |
| 2.70 | 0.360 | 0.221 | 11.50 | 0.721 | 0.557 | 85.00 | 0.302 | 0.212 |
| 2.80 | 0.373 | 0.228 | 12.00 | 0.726 | 0.571 | 88.00 | 0.301 | 0.212 |
| 2.90 | 0.387 | 0.236 | 12.50 | 0.730 | 0.584 | 91.00 | 0.301 | 0.211 |
| 3.00 | 0.401 | 0.243 | 13.00 | 0.733 | 0.595 | 94.00 | 0.300 | 0.211 |
| 3.15 | 0.421 | 0.254 | 13.50 | 0.734 | 0.603 | 97.00 | 0.299 | 0.210 |
| 3.30 | 0.441 | 0.264 | 14.00 | 0.734 | 0.607 | 100.00 | 0.297 | 0.209 |
| 3.45 | 0.460 | 0.274 | 14.50 | 0.732 | 0.610 | - | - | - |


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**Table 2.6-16 Median and Logarithmic Standard Deviation of Strain-Compatible Shear Wave Velocity and Damping Profiles Corresponding to GMRS (Sheet 1 of 2)**

| Top Depth (ft) | Median $V_s$ (ft/sec) | $\sigma_{lnVs}$[1] | Median D (%) | $\sigma_{lnD}$ | Top Depth (ft) | Median $V_s$ (ft/sec) | $\sigma_{lnVs}$ | Median D (%) | $\sigma_{lnD}$ |
|---|---|---|---|---|---|---|---|---|---|
| 0 | 1176.2 | 0.30 | 3.81 | 0.99 | 85 | 4203.5 | 0.26 | 0.82 | 1.10 |
| 1 | 1175.8 | 0.30 | 3.81 | 0.99 | 90 | 4606.7 | 0.27 | 0.80 | 1.09 |
| 2 | 1182.7 | 0.33 | 3.43 | 1.01 | 95 | 4799.3 | 0.25 | 0.79 | 1.09 |
| 3 | 1182.7 | 0.33 | 3.43 | 1.01 | 100 | 5498.3 | 0.15 | 0.77 | 1.09 |
| 4 | 1236.9 | 0.39 | 3.41 | 1.02 | 105 | 5457.4 | 0.15 | 0.77 | 1.09 |
| 5 | 1236.9 | 0.39 | 3.41 | 1.02 | 110 | 5485.7 | 0.16 | 0.77 | 1.09 |
| 6 | 1200.5 | 0.41 | 3.66 | 1.02 | 115 | 5533.3 | 0.16 | 0.77 | 1.09 |
| 7 | 1200.5 | 0.41 | 3.66 | 1.02 | 120 | 5559.9 | 0.16 | 0.78 | 1.09 |
| 8 | 1175.3 | 0.43 | 3.88 | 1.02 | 125 | 5595.6 | 0.15 | 0.78 | 1.09 |
| 9 | 1175.3 | 0.43 | 3.88 | 1.02 | 130 | 5616.6 | 0.16 | 0.78 | 1.08 |
| 10 | 1319.5 | 0.50 | 3.56 | 1.03 | 135 | 5579.6 | 0.15 | 0.78 | 1.08 |
| 11 | 1319.5 | 0.50 | 3.56 | 1.03 | 140 | 5584.4 | 0.13 | 0.78 | 1.09 |
| 12 | 1820.9 | 0.33 | 2.14 | 1.18 | 145 | 5585.8 | 0.15 | 0.79 | 1.09 |
| 13 | 1820.9 | 0.33 | 2.14 | 1.18 | 150 | 5455.5 | 0.15 | 0.80 | 1.08 |
| 14 | 2033.3 | 0.37 | 1.32 | 1.31 | 155 | 5462.6 | 0.16 | 0.80 | 1.08 |
| 15 | 2033.3 | 0.37 | 1.32 | 1.31 | 160 | 5445.6 | 0.16 | 0.80 | 1.08 |
| 16 | 2318.5 | 0.29 | 1.27 | 1.29 | 170 | 5447.9 | 0.15 | 0.80 | 1.09 |
| 17 | 2318.5 | 0.29 | 1.27 | 1.29 | 180 | 5437.5 | 0.15 | 0.81 | 1.09 |
| 18 | 2331.9 | 0.30 | 1.29 | 1.30 | 190 | 5488.4 | 0.18 | 0.81 | 1.09 |
| 19 | 2331.9 | 0.30 | 1.29 | 1.30 | 200 | 5480.6 | 0.15 | 0.82 | 1.08 |
| 20 | 2369.9 | 0.32 | 1.31 | 1.31 | 210 | 5573.3 | 0.15 | 0.81 | 1.08 |
| 22 | 2350.8 | 0.32 | 1.33 | 1.32 | 220 | 5502.0 | 0.17 | 0.82 | 1.08 |
| 24 | 2335.1 | 0.32 | 1.35 | 1.32 | 230 | 5391.0 | 0.17 | 0.83 | 1.08 |
| 26 | 2348.7 | 0.30 | 1.36 | 1.32 | 240 | 5313.2 | 0.16 | 0.84 | 1.08 |
| 28 | 2422.9 | 0.35 | 1.38 | 1.34 | 250 | 5480.7 | 0.17 | 0.83 | 1.08 |
| 30 | 2470.7 | 0.37 | 1.38 | 1.34 | 260 | 5511.5 | 0.16 | 0.83 | 1.08 |
| 32 | 2489.1 | 0.37 | 1.39 | 1.35 | 270 | 5499.6 | 0.15 | 0.83 | 1.09 |
| 34 | 2745.5 | 0.43 | 1.38 | 1.36 | 280 | 5466.2 | 0.15 | 0.84 | 1.09 |
| 36 | 2766.4 | 0.45 | 1.39 | 1.37 | 290 | 5401.2 | 0.16 | 0.84 | 1.08 |
| 38 | 2800.4 | 0.45 | 1.40 | 1.37 | 300 | 5356.8 | 0.15 | 0.85 | 1.09 |
| 40 | 3546.2 | 0.36 | 1.04 | 1.23 | 310 | 5302.9 | 0.14 | 0.85 | 1.09 |
| 42 | 3600.6 | 0.36 | 1.04 | 1.23 | 320 | 5376.6 | 0.16 | 0.85 | 1.09 |
| 44 | 3663.2 | 0.36 | 1.04 | 1.23 | 330 | 5378.7 | 0.16 | 0.85 | 1.08 |
| 46 | 3646.7 | 0.37 | 1.05 | 1.23 | 340 | 5363.5 | 0.17 | 0.86 | 1.08 |



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**Table 2.6-16 Median and Logarithmic Standard Deviation of Strain-Compatible Shear Wave Velocity and Damping Profiles Corresponding to GMRS (Sheet 2 of 2)**

| Top Depth (ft) | Median $V_s$ (ft/sec) | $\sigma_{lnVs}$[1] | Median D (%) | $\sigma_{lnD}$ | Top Depth (ft) | Median $V_s$ (ft/sec) | $\sigma_{lnVs}$ | Median D (%) | $\sigma_{lnD}$ |
|---|---|---|---|---|---|---|---|---|---|
| 48 | 3649.2 | 0.36 | 1.05 | 1.23 | 350 | 5393.3 | 0.16 | 0.86 | 1.09 |
| 50 | 4031.8 | 0.23 | 0.78 | 1.09 | 360 | 5322.3 | 0.16 | 0.86 | 1.08 |
| 52 | 4083.1 | 0.24 | 0.78 | 1.09 | 380 | 5316.0 | 0.15 | 0.86 | 1.09 |
| 54 | 4118.4 | 0.24 | 0.79 | 1.09 | 400 | 5788.2 | 0.18 | 0.85 | 1.09 |
| 56 | 4176.8 | 0.24 | 0.79 | 1.09 | 420 | 5775.3 | 0.17 | 0.85 | 1.09 |
| 58 | 4175.5 | 0.23 | 0.79 | 1.09 | 440 | 5818.7 | 0.17 | 0.85 | 1.09 |
| 60 | 4138.3 | 0.23 | 0.79 | 1.09 | 460 | 5868.0 | 0.16 | 0.85 | 1.09 |
| 65 | 4125.4 | 0.23 | 0.80 | 1.09 | 480 | 5885.9 | 0.17 | 0.85 | 1.09 |
| 70 | 4125.0 | 0.25 | 0.81 | 1.09 | 500 | 5530.8 | 0.15 | 0.87 | 1.09 |
| 75 | 4075.5 | 0.26 | 0.81 | 1.09 | 520 | 5648.1 | 0.15 | 0.87 | 1.08 |
| 80 | 4151.4 | 0.26 | 0.81 | 1.10 | 540 | 5686.4 | 0.13 | 0.95 | 0.70 |

[1] σ is defined as Aleatory Uncertainty


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**Table 2.6-17 Mean GMRS Horizon V/H Ratios Calculated per Gülerce and Abrahamson**
(Reference 2.6-69)

| T (sec.) | V/H Ratio |
|---|---|
| | GMRS and SSE |
| PGA | 0.702 |
| PGV | 0.704 |
| 0.01 | 0.702 |
| 0.02 | 0.702 |
| 0.03 | 0.787 |
| 0.04 | 0.844 |
| 0.05 | 0.878 |
| 0.08 | 0.819 |
| 0.10 | 0.722 |
| 0.15 | 0.621 |
| 0.20 | 0.592 |
| 0.26 | 0.588 |
| 0.30 | 0.598 |
| 0.40 | 0.622 |
| 0.50 | 0.635 |
| 0.75 | 0.716 |
| 1.00 | 0.790 |
| 1.50 | 0.786 |
| 2.00 | 0.758 |
| 3.00 | 0.741 |
| 4.00 | 0.771 |
| 5.00 | 0.782 |
| 7.50 | 0.782 |
| 10.00 | 0.782 |


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**Table 2.6-18 Summary of Subsurface Investigation Field Work**

| Field Activity/Test | Quantity |
|---|---|
| Boreholes, including soil borings and rock coring | 109 |
| Borehole drilling, excluding offset borings for intact sampling and pressuremeter testing | 13,765 linear feet |
| Standard penetration test sampling | 587 |
| Intact soil samples | 38 |
| Rock core sampling | 10,583 linear feet |
| Test pits | 8 |
| Pressuremeter tests | 12 |
| Groundwater observation wells | 52 |
| Slug tests | 48 |
| Field permeability (Packer tests) | 28 |
| Wells equipped with vented pressure transducers for long-term water level monitoring | 20 |
| Vented pressure transducers installed in stream gage | 2 |
| P-S suspension logging (no. of boreholes) | 8 |
| DH seismic, with magnetic resonance (no. of boreholes) | 3 |
| DH geophysical testing (no. of boreholes)<br>• Natural Gamma<br>• Dual Induction<br>• Long and Short Normal Resistivity<br>• Spontaneous Potential<br>• Caliper<br>• Acoustic Televiewer/Deviation | 8 |
| SASW | 8 short arrays<br>5 long arrays |
| Electrical resistivity (N-S and E-W arrays) | 7 short<br>1 long |
| Thermal resistivity | 4 in each of 8 test pits |


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**Table 2.6-19 Laboratory Testing Program**
**(Sheet 1 of 2)**

| Soil Test | Quantity | Rock Test | Quantity |
|---|---|---|---|
| Moisture content<br>ASTM D2216 | 123 | Moisture content<br>ISRM 2007 | 21 |
| Unit weight<br>ASTM D7263 | 72 | Unit weight<br>ISRM 2007 | 148 |
| Specific gravity<br>ASTM D854 | 11 | Specific gravity<br>ISRM 2007 | 26 |
| Sieve analysis<br>ASTM D6913 | 77 | Absorption<br>ISRM 2007 | 25 |
| Sieve analysis with hydrometer<br>ASTM D422/D7928 | 23 | Porosity<br>ISRM 2007 | 45 |
| Atterberg limits<br>ASTM D4318 | 104 | Petrographic analysis (thin section)<br>ASTM D2216 | 18 |
| 1-D swell<br>ASTM D4546 | 8 | Slake durability<br>ASTM D4644 | 12 |
| Chemical analysis (pH, chloride, and sulfate)<br>ASTM D | 19 | Unconfined free-free resonant column<br>Univ. of Texas | 11 |
| Organic content<br>ASTM D2974 | 16 | X-ray diffraction<br>Univ. of Wyoming | 39 |
| X-ray radiography of soil samples<br>ASTM D4452 | 13 | Scanning Electron Microscopy (SEM)<br>Univ. of Wyoming | 19 |
| Unconsolidated-undrained triaxial compression<br>ASTM D2850 | 36 | Swelling rocks<br>ISRM 2007 | 14 |
| Consolidated-undrained triaxial compression<br>ASTM D4767 | 7 | Calcium carbonate<br>ASTM D4373 | 69 |
| Direct shear<br>ASTM D3080 | 1 | Point load index<br>ASTM D5731 | 104 |
| Consolidation<br>ASTM D2435/D4186 | 9 | Unconfined compression<br>ASTM D7012/ISRM 2007 | 79 |
| Modified Proctor test<br>ASTM D1557 | 6 | Unconfined compression with stress-strain<br>ASTM D7012/ISRM 2007 | 69 |
| CBR<br>ASTM D1883 | 3 | Multi-stage triaxial compression<br>ASTM D7012 | 13 |


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**Table 2.6-19 Laboratory Testing Program**
**(Sheet 2 of 2)**

| Soil Test | Quantity | Rock Test | Quantity |
|---|---|---|---|
| Electrical Resistivity<br>ASTM G187 | 5 | Direct shear (along joint or bedding plane)<br>ASTM D5607 | 7 |
| Resonant column torsional shear (RCTS)<br>ASTM D | 1 | Basic friction angle of saw-cut rock surface<br>Alejano et al. 2018, Reference 2.6-125 | 7 |
| Thermal Resistivity<br>ASTM D5334 | 6 | Resonant column torsional shear (RCTS)<br>Univ. of Texas | 3 |
| **Testing Agencies**<br>Chemtech-Ford, Salt Lake City, UT<br>CTL Group, Skokie, IL<br>Eurofins TestAmerica, St Louis, MO<br>GeoTesting Express, Acton, MA<br>RESPEC, Rapid City, SD<br>University of Texas, Austin, TX<br>University of Wyoming, Laramie, WY<br>Wood/WSP, Atlanta, GA | | | |



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**Table 2.6-20 Summary of Static Engineering Properties**

| Property | In-situ Soil | Compacted Granular Backfill | Weathered Rock | | Fresh Rock |
|---|---|---|---|---|---|
| | High Area | | CW / HW | MW / SW | |
| Material / USCS symbol[1] | **CL**, ML, SM, GC | GW, SW | Siltstone | | |
| Total unit weight, γ (pcf) | 125 | 130 | 147 | 148 | 156 |
| Specific gravity, $G_s$ | 2.70 | 2.70 | 2.37 | 2.37 | 2.50 |
| Natural moisture content (%) | 9.8 | -- | 4.4 | 3.4 | 2.3 |
| Liquid limit, LL (%) | 35 | -- | -- | -- | -- |
| Plasticity index, PI (%) | 16 | -- | -- | -- | -- |
| SPT $N_{60}$-value (bpf) | 59 | 30 | Refusal | -- | -- |
| Undrained shear strength, $S_u$ (psf) | 4,000 | -- | -- | -- | -- |
| Internal friction angle, (degree) | 0 | 35 | -- | -- | -- |
| Drained shear strength, (degree) | 29 | 35 | -- | -- | -- |
| Drained shear strength, (psf) | 0 | 0 | -- | -- | -- |
| Unconfined compressive strength, (psi) | -- | -- | 1,500 | 3,000 | 7,000 |
| Static Poisson's ratio | 0.35 | 0.3 | -- | -- | 0.17 to 0.47, depending on stress range |
| High-strain elastic modulus, $E_H$ (ksf) | 2,400 | 1,100 | 4,500 [2] | 12,672 [2],[3] | 86,400 [2], [3] |
| High-strain shear modulus, $G_H$ (ksf) | 900 | 425 | -- | -- | -- |

NOTES:

[1] Primary soil type is in **Bold** font.

[2] Rock mass deformation modulus taken from Table 2.6-34.

[3] For reduced modulus values due to disturbance during construction/excavation.


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**Table 2.6-21 Summary of SPT N-Values**

| Location | Count | N-Value | Mean | Median | SD | Min | Max |
|---|---|---|---|---|---|---|---|
| Sitewide | 587 | N | 36 | 33 | 20 | 2 | 92 |
| | | $N_{60}$ | 52 | 47 | 29 | 3 | 195 |
| High Area | 284 | N | 42 | 38 | 19 | 8 | 92 |
| | | $N_{60}$ | 59 | 53 | 27 | 10 | 144 |


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**Table 2.6-22 Summary of Index Test Results**

| Location | Index Test | Count | Mean | Median | SD | Min | Max |
|---|---|---|---|---|---|---|---|
| Sitewide | Moisture Content, % | 123 | 12.6 | 11.3 | 5.3 | 3.7 | 25.7 |
| | Liquid Limit | 104 | 37 | 35 | 9 | 29 | 77 |
| | Plasticity Index | 104 | 19 | 17 | 7 | 10 | 52 |
| High Area | Moisture Content, % | 74 | 10.0 | 9.8 | 3.5 | 3.7 | 23.0 |
| | Liquid Limit | 66 | 35 | 35 | 3 | 29 | 47 |
| | Plasticity Index | 66 | 17 | 16 | 3 | 10 | 28 |


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**Table 2.6-23 Summary of Total Unit Weight and Specific Gravity Test Results**

| Index Test | Count | Mean | Median | SD | Minimum | Maximum |
|---|---|---|---|---|---|---|
| Total Unit Weight, pcf | 72 | 124.8 | 127.4 | 5.46 | 112.2 | 140.8 |
| Specific Gravity | 11 | 2.70 | 2.68 | 0.08 | 2.57 | 2.81 |


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**Table 2.6-24 Summary of Rock Testing**

| Degree of Weathering | Unit weight, pcf | Specific Gravity | Moisture Content, % | Porosity, % | Absorption, % |
|---|---|---|---|---|---|
| Completely to highly | 147 | 2.37 | 4.4 | 6.8 | 2.7 |
| Moderately to slightly | 148 | 2.37 | 3.4 | 6.2 | 2.3 |
| Fresh | 156 | 2.50 | 2.3 | 2.8 | 1.2 |



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**Table 2.6-25 Summary of Unconfined Compressive Strength Testing of Rock**

| | | Unconfined Compressive Strength, psi | | | | | |
|---|---|---|---|---|---|---|---|
| Degree of Weathering | Count | Mean | Median | SD | Min | Max | Recommended Value |
| Completely to highly | 1 | 3,090 | 3,090 | 3,090 | 3,090 | 3,090 | 1,500 |
| Moderately to slightly | 6 | 3,555 | 3,210 | 1,880 | 1,675 | 6,765 | 3,000 |
| Fresh | 72 | 6,928 | 7,066 | 2,582 | 1,770 | 13,460 | 7,000 |


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**Table 2.6-26 Hoek-Brown Parameters and Rock Mass Strength**

| Weathering Grade | D | Hoek-Brown Parameters | | | Rock Mass Strength (psi) | | |
|---|---|---|---|---|---|---|---|
| | | $m_b$ | s | a | $\sigma_t$ | $\sigma_c$ | $\sigma_{cm}$ |
| Completely to highly | 0 | 0.206 | 0.000 | 0.531 | -1.75 | 18 | 80 |
| | 0.7 | 0.049 | 0.000 | 0.531 | -0.59 | 4.7 | 35 |
| Moderately to slightly | 0 | 0.719 | 0.012 | 0.503 | -49 | 320 | 410 |
| | 0.7 | 0.333 | 0.003 | 0.503 | -27 | 160 | 260 |
| Fresh | 0 | 1.469 | 0.108 | 0.501 | -520 | 2,300 | 2,200 |
| | 0.7 | 1.000 | 0.055 | 0.501 | -390 | 1,600 | 1,600 |

Abbreviations:

D = Disturbance factor

$m_b$ = Hoek-Brown material constant (reduced)

s = Hoek-Brown constant

a = Hoek-Brown constant

$\sigma_t$ = rock mass tensile strength

$\sigma_c$ = rock mass uniaxial compressive strength

$\sigma_{cm}$ = rock mass global strength



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**Table 2.6-27 Low Strain Shear Modulus**

| Elevation Range (ft) | Depth Range (ft) | $V_s$ (fps) | G (ksf) |
|---|---|---|---|
| 6,755 - 6,735 | 0 - 20 | 926 | 3,330 |
| 6,735 - 6,705 | 20 - 50 | 2,867 | 37,644 |
| 6,705 - 6,655 | 50 - 100 | 5,274 | 134,739 |
| 6,655 - 6,605 | 100 - 150 | 5,805 | 163,229 |
| 6,605 - 6,555 | 150 - 200 | 5,757 | 160,590 |
| 6,555 - 6,455 | 200 - 300 | 5,715 | 158,239 |
| 6,455 - 6,355 | 300 - 400 | 5,874 | 167,142 |
| 6,355 - 6,255 | 400 - 500 | 6,578 | 209,602 |
| 6,255 - 6,155 | 500 - 600 | 5,824 | 164,338 |
| 6,155 - 6,055 | 600 - 700 | 5,870 | 166,920 |
| 6,055 - 5,955 | 700 - 800 | 5,927 | 170,190 |
| 5,955 - 5,855 | 800 - 900 | 5,775 | 161,556 |
| 5,855 - 5,755 | 900 - 1,000 | 6,977 | 235,809 |
| 5,755 - 5,655 | 1,000 - 1,100 | 6,485 | 203,736 |
| 5,655 - 5,555 | 1,100 - 1,200 | 7,222 | 252,702 |
| 5,555 - 5,455 | 1,200 - 1,300 | 7,517 | 273,750 |
| 5,455 - 5,355 | 1,300 - 1,400 | 5,945 | 171,227 |
| 5,355 - 5,305 | 1,400 - 1,452 | 5,453 | 144,062 |


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**Table 2.6-28 Summary of Exploration Depths of Borings**

| Boring ID | Drill Depth, ft | Boring ID | Drill Depth, ft | Boring ID | Drill Depth, ft | Boring ID | Drill Depth, ft |
|---|---|---|---|---|---|---|---|
| B-101 | 25.3 | B-128A | 24.0 | B-155 | 100.5 | B-184 | 51.7 |
| B-102 | 50.1 | B-129 | 50.3 | B-156 | 104.0 | B-185 | 101.5 |
| B-103 | 23.6 | B-130 | 200.1 | B-157 | 101.5 | B-186 | 102.2 |
| B-104 | 50.5 | B-131 | 102.1 | B-158/ 158A | 25.0/30.8 | B-187 | 100.2 |
| B-105 | 25.1 | B-132 | 51.6 | B-159 | 100.0 | B-188 | 100.4 |
| B-106 | 100.5 | B-133 | 101.8 | B-160 | 170.0 | B-189 | 100.8 |
| B-107 | 100.6 | B-134 | 170.0 | B-161 | 101.0 | B-190 | 100.0 |
| B-108 | 200.5 | B-135/ 135A | 19.5/102.1 | B-162 | 50.0 | B-191 | 100.1 |
| B-109 | 100.4 | B-136 | 100.4 | B-163 | 50.5 | B-192 | 25.0 |
| B-110 | 101.3 | B-137 | 101.0 | B-164 | 100.6 | B-193 | 15.0 |
| B-111 | 100.0 | B-138 | 50.0 | B-165 | 150.6 | B-194 | 16.1 |
| B-112 | 200.7 | B-139 | 50.0 | B-166 | 152.0 | B-195 | 15.2 |
| B-113 | 201.8 | B-140 | 101.0 | B-167 | 151.5 | B-196 | 19.9 |
| B-114/ 114A | 12.0/200.3 | B-141 | 25.0 | B-168 | 220.4 | B-197 | 20.0 |
| B-115 | 200.4 | B-142 | 50.3 | B-169 | 120.0 | B-198 | 15.0 |
| B-116 | 200.2 | B-142A | 12.0 | B-170 | 100.3 | B-199 | 15.0 |
| B-117 | 320.4 | B-143 | 50.7 | B-171 | 102.0 | B-200 | 14.9 |
| B-118 | 300.3 | B-144 | 50.0 | B-172 | 26.3 | B-201 | 17.4 |
| B-119 | 301.9 | B-145 | 100.3 | B-173 | 26.5 | B-202 | 17.2 |
| B-120 | 300.6 | B-146 | 100.3 | B-174 | 100.6 | B-203 | 30.4 |
| B-121 | 300.6 | B-147 | 50.0 | B-175 | 100.9 | B-204 | 33.8 |
| B-122 | 1519.5 | B-148 | 219.9 | B-176 | 50.4 | B-204A | 20.0 |
| B-123 | 186.1 | B-149 | 100.3 | B-177 | 51.5 | B-205 | 30.3 |
| B-123A | 200.3 | B-150 | 220.0 | B-178 | 50.6 | B-206 | 15.2 |
| B-124 | 200.9 | B-151 | 100.3 | B-179 | 152.3 | B-207 | 15.0 |
| B-125 | 201.5 | B-152 | 100.7 | B-180 | 50.7 | B-208 | 104.8 |
| B-126 | 200.7 | B-153 | 111.5 | B-181 | 102.2 | B-209 | 100.0 |
| B-127 | 324.7 | B-153A | 150.2 | B-182 | 50.3 | | |
| B-128 | 101.6 | B-154/ 154A | 23.8/50.0 | B-183 | 24.3 | | |

Notes: Summary does not include two 100 ft exploratory borings, and offset borings for intact sampling and pressuremeter testing.


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**Table 2.6-29 Summary of Borehole Depths DH Geophysical Surveys**

| Natural Gamma<br>Dual Induction<br>Long and Short Normal Resistivity<br>Spontaneous Potential<br>Caliper<br>Acoustic Televiewer/Deviation<br>P-S Suspension Logging | | Acoustic Televiewer/Deviation<br>Magnetic Resonance<br>DH Seismic Testing | |
|---|---|---|---|
| **Uncased Borehole** | **Depth Below Ground Surface (ft)** | **Cased Borehole** | **Depth Below Ground Surface (ft)** |
| B-117 | 320.4 | B-117-CB | 300.8 |
| B-122 | 1,519.5 | B-122-CB | 606.1 |
| B-127 | 324.7 | B-127-CB | 303.0 |
| B-134 | 170.0 | | |
| B-148 | 219.9 | | |
| B-150 | 220.0 | | |
| B-160 | 170.0 | | |
| B-168 | 220.4 | | |


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**Table 2.6-30 Summary of Observation Wells**

| | | | |
|---|---|---|---|
| **OW-106U** | **OW-122U** | OW-153U | **OW-185M** |
| **OW-106M** | **OW-122M** | OW-153M | **OW-186U** |
| OW-108U | **OW-122L** | OW-153L | **OW-186M** |
| OW-108M | OW-127U | **OW-164U** | OW-187U |
| OW-109U | OW-127M | **OW-164M** | OW-187M |
| OW-109M | OW-136U | OW-167U | OW-188U |
| **OW-117U** | OW-136M | OW-167M | OW-188M |
| **OW-117M** | **OW-137U** | OW-167L | **OW-189U** |
| **OW-117L** | **OW-137M** | OW-168U | **OW-189M** |
| OW-119U | **OW-140U** | OW-168M(A) | OW-190U |
| OW-119M | **OW-140M** | OW-179U | OW-190M |
| OW-121U | OW-149U | OW-179M | OW-191U |
| OW-121M(A) | OW-149M | **OW-185U** | OW-191M |

Notes:

1) Wells in Bold font are equipped with vented pressure transducers.

2) U indicates upper level well, M indicates mid level well and L indicates lower level well

3) (A) indicates an offset location



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**Table 2.6-31 Summary of Permeability Test Locations**

| Well ID of Slug Test in Weathered Rock | | Well ID of Slug Test in Fresh Rock |
|---|---|---|
| OW-106U | OW-185U | OW-164U |
| OW-108U | OW-186U | OW-122M |
| OW-109U | OW-187U | OW-140M |
| OW-117U | OW-188U | OW-164M |
| OW-121U | OW-189U | OW-168M(A) |
| OW-122U | OW-190U | OW-179M |
| OW-136U | OW-191U | OW-185M |
| OW-137U | OW-106M | OW-186M |
| OW-140U | OW-119M | OW-187M |
| OW-149U | OW-137M | OW-189M |
| OW-167U | OW-188M | OW-190M |
| OW-179U | OW-191M | |
| **Packer Test Locations (adjacent to corresponding borehole)** | | |
| B-115PT(3) | B-139PT(3) | B-160PT(1) |
| B-116PT(2) | B-142PT(2) | B-171PT(2) |
| B-124PT(1) | B-144PT(2) | B-177PT(2) |
| B-126PT(1) | B-150PT(1) | B-181PT(3) |
| B-132PT(2) | B-157PT(1) | B-182PT(2) |
| (X) - indicates number of depth intervals tested | | |


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**Table 2.6-32 $V_s$ and $V_p$ Profiles of Compacted Granular Backfill**

| Depth | Overburden Pressure, psf | SPT N-Value. bpf | $V_s$, fps | $V_p$, fps |
|---|---|---|---|---|
| 0 | 0 | 5 | 471 | 880 |
| 5 | 650 | 17 | 769 | 1,438 |
| 10 | 1,300 | 24 | 885 | 1,655 |
| 15 | 1,950 | 29 | 956 | 1,789 |
| 20 | 2,288 | 31 | 983 | 1,838 |
| 25 | 2,626 | 33 | 1,008 | 1,886 |
| 30 | 2,964 | 36 | 1,045 | 1,955 |

Notes: Groundwater depth taken at 15 feet


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**Table 2.6-33 Modulus Reduction and Damping Relationships for Compacted Granular Backfill, In-Situ Soil, and Weathered Rock**
**(Sheet 1 of 2)**

| Shear Strain (%) | Modulus Reduction ($G/G_{max}$) | Damping, % |
|---|---|---|
| Compacted Granular Backfill | | |
| 0.0001 | 1 | 0.8 |
| 0.0003 | 0.96 | 0.9 |
| 0.001 | 0.87 | 1.8 |
| 0.003 | 0.72 | 3.0 |
| 0.01 | 0.54 | 5.5 |
| 0.03 | 0.37 | 9.5 |
| 0.1 | 0.21 | 15.5 |
| 0.3 | 0.1 | 21.0 |
| 1 | 0.05 | 24.6 |
| In-Situ Soil | | |
| 0.0001 | 1 | 1.2 |
| 0.0003 | 1 | 1.2 |
| 0.001 | 0.985 | 1.35 |
| 0.003 | 0.93 | 2.07 |
| 0.01 | 0.78 | 3.7 |
| 0.03 | 0.545 | 7.05 |
| 0.1 | 0.28 | 12.85 |
| 0.3 | 0.125 | 18.4 |
| 0.6 | 0.065 | 19.8 |
| 1 | 0.045 | 19.6 |
| Weathered Rock - Lower Bound | | |
| 0.000001 | 1.00 | 3.04 |
| 0.000005 | 1.00 | 3.04 |
| 0.00001 | 1.00 | 3.05 |
| 0.00005 | 1.00 | 3.10 |
| 0.0001 | 0.99 | 3.16 |
| 0.00017 | 0.98 | 3.25 |
| 0.00034 | 0.97 | 3.46 |
| 0.0005 | 0.95 | 3.65 |
| 0.001 | 0.91 | 4.23 |
| 0.002 | 0.83 | 5.29 |
| 0.003 | 0.77 | 6.25 |
| 0.005 | 0.67 | 7.91 |
| 0.007 | 0.59 | 9.32 |
| 0.01 | 0.50 | 11.05 |


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**Table 2.6-33 Modulus Reduction and Damping Relationships for Compacted Granular Backfill, In-Situ Soil, and Weathered Rock**
**(Sheet 2 of 2)**

| Shear Strain (%) | Modulus Reduction ($G/G_{max}$) | Damping, % |
|---|---|---|
| Weathered Rock - Upper Bound | | |
| 0.000001 | 1.00 | 1.00 |
| 0.00001 | 1.00 | 1.00 |
| 0.0001 | 1.00 | 1.01 |
| 0.001 | 1.00 | 1.06 |
| 0.003 | 0.99 | 1.19 |
| 0.005 | 0.98 | 1.31 |
| 0.01 | 0.95 | 1.61 |
| 0.02 | 0.91 | 2.19 |
| 0.03 | 0.87 | 2.73 |
| 0.04 | 0.83 | 3.25 |
| 0.05 | 0.80 | 3.74 |
| 0.06 | 0.77 | 4.21 |
| 0.07 | 0.74 | 4.66 |
| 0.08 | 0.71 | 5.08 |
| 0.085 | 0.70 | 5.29 |
| 0.1 | 0.67 | 5.88 |
| 0.15 | 0.57 | 7.59 |
| 0.2 | 0.50 | 9.02 |


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**Table 2.6-34 Summary of Rock Mass Deformation Modulus Estimates**

| Siltstone | Input Parameters | | | Empirical Estimates of Rock Mass Modulus (psi) | | | | |
|---|---|---|---|---|---|---|---|---|
| | GSI | Intact Rock Modulus $E_i$ (psi) | D | Min | Max | Average | Median | Best Estimate |
| Completely to highly weathered | 25 | 180,000 | 0 | 2,500 | 31,000 | 12,000 | 11,000 | 31,000 |
| | 25 | 180,000 | 0.7 | 5,400 | 18,000 | 12,000 | 12,000 | 26,000 |
| Slightly to moderately weathered | 60 | 300,000 | 0 | 40,000 | 160,000 | 95,000 | 82,000 | 88,000 |
| | 60 | 300,000 | 0.7 | 60,000 | 93,000 | 77,000 | 77,000 | 77,000 |
| Fresh | 80 | 920,000 | 0 | 390,000 | 810,000 | 600,000 | 590,000 | 600,000 |
| | 80 | 920,000 | 0.7 | 440,000 | 510,000 | 480,000 | 480,000 | 480,000 |


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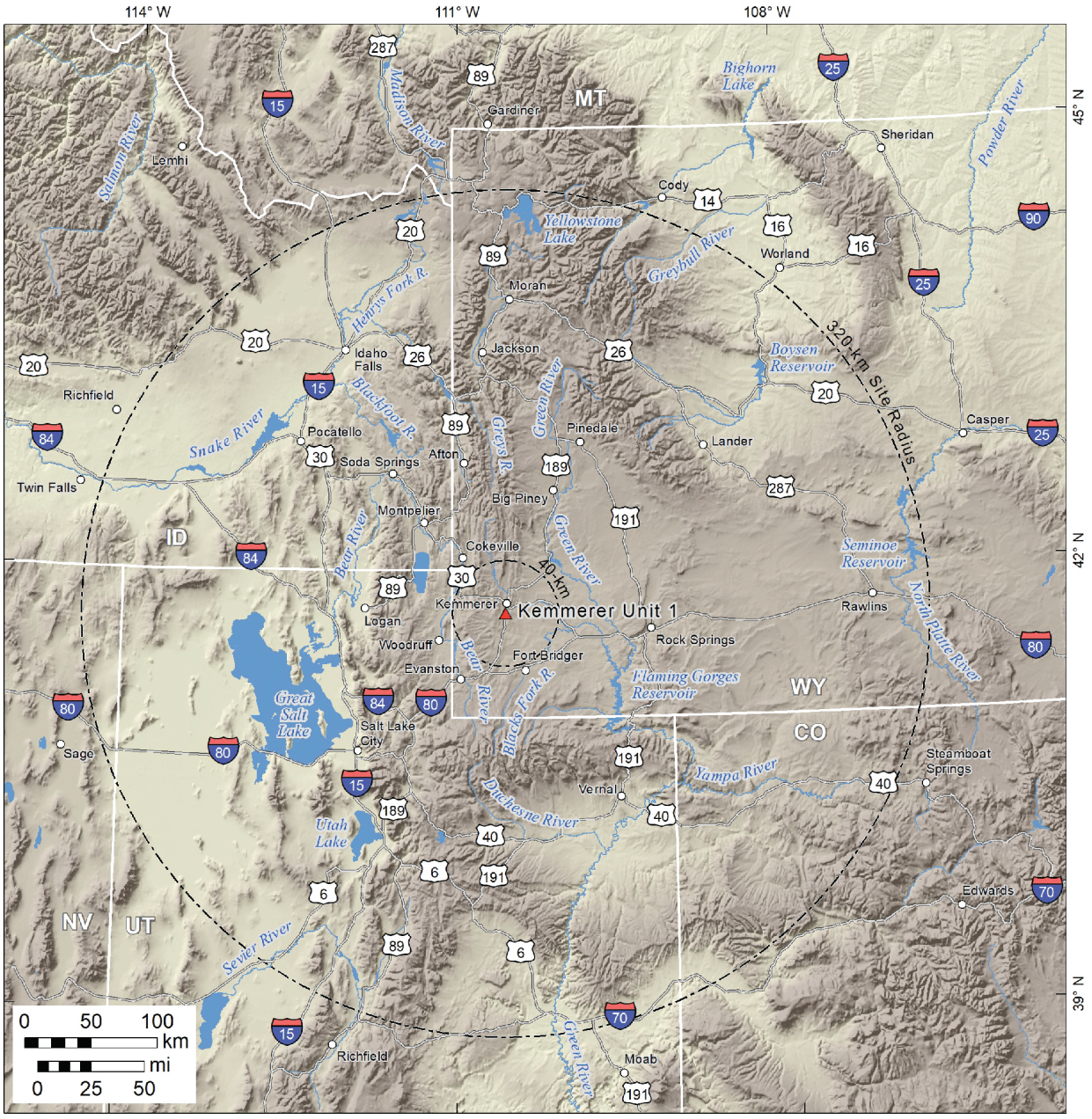
## Figure 2.6-1 Kemmerer Unit 1 Site Location Map





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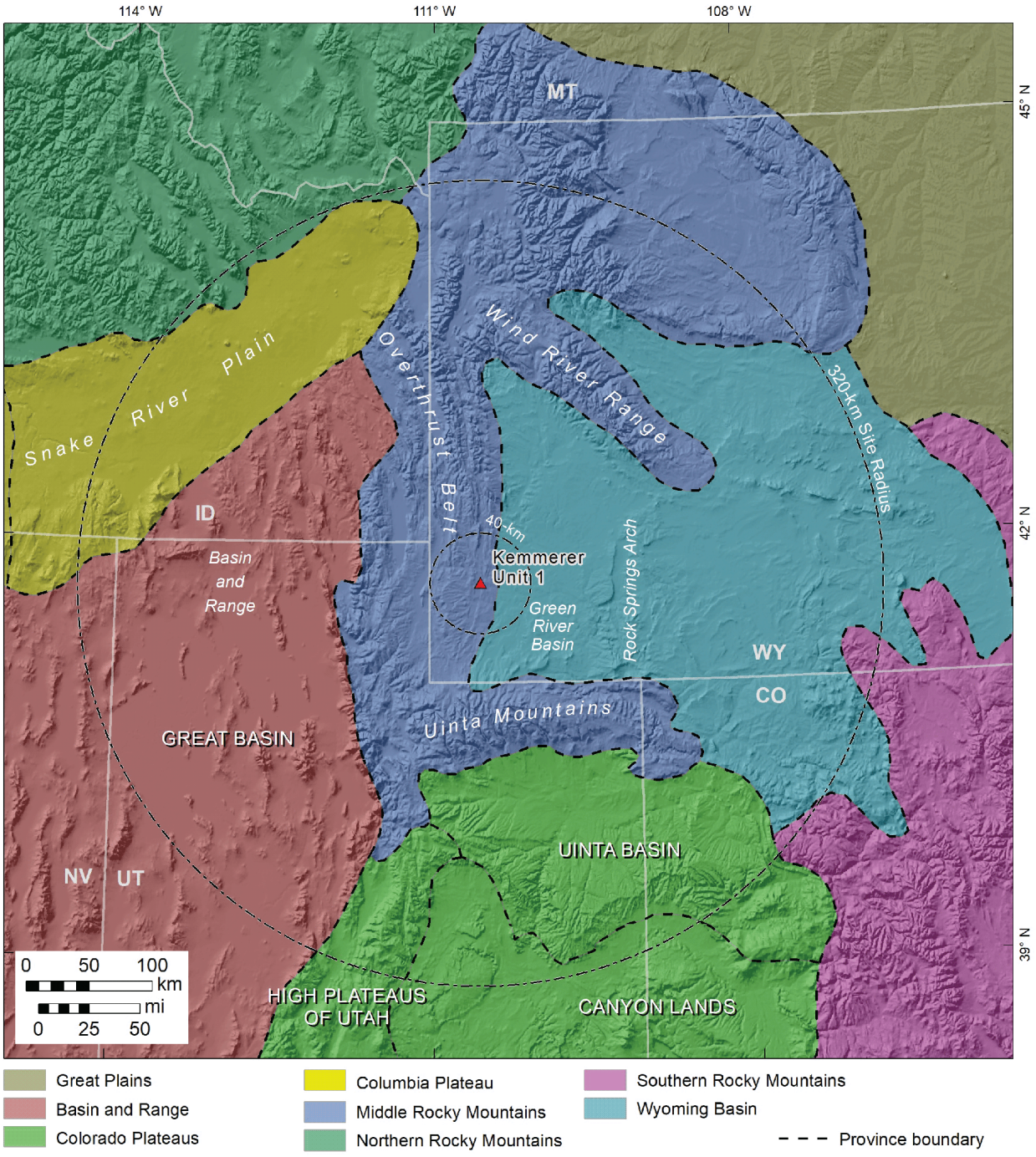
## Figure 2.6-2 Map of Regional Physiographic Provinces





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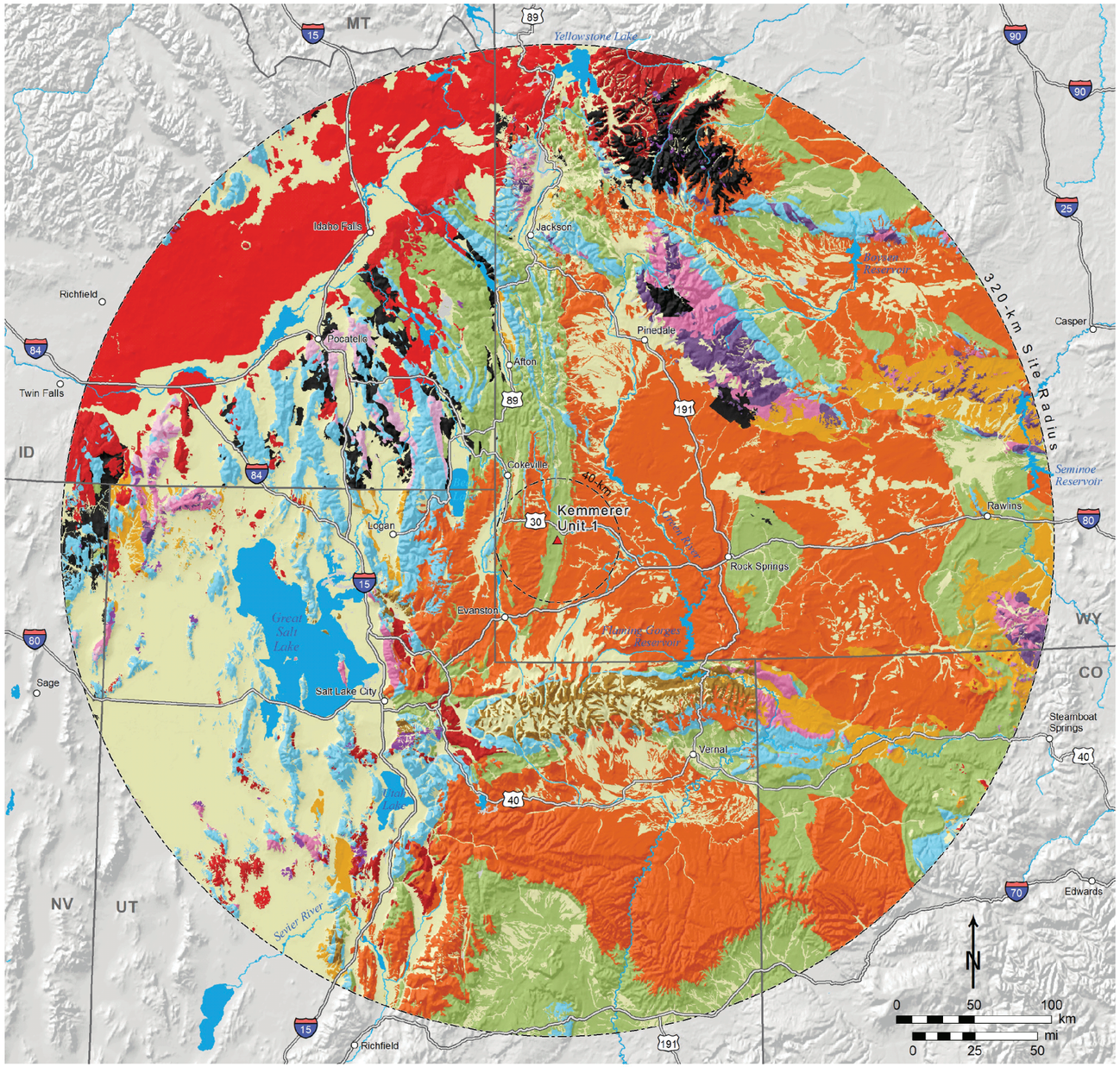
## Figure 2.6-3 Generalized Geologic Map of the Site Region (320 km Radius)




**Generalized Geologic Units**

- Unconsolidated, Quaternary
- Volcanic, Quaternary
- Unconsolidated / Sedimentary, Neogene
- Plutonic, Neogene
- Volcanic, Neogene
- Plutonic, Paleogene
- Volcanic, Paleogene
- Unconsolidated / Sedimentary, Paleogene
- Plutonic, Mesozoic
- Sedimentary, Mesozoic
- Metamorphic, Paleozoic
- Sedimentary, Paleozoic
- Sedimentary, Precambrian
- Metamorphic, Precambrian
- Plutonic, Precambrian
- Igneous - Undifferentiated

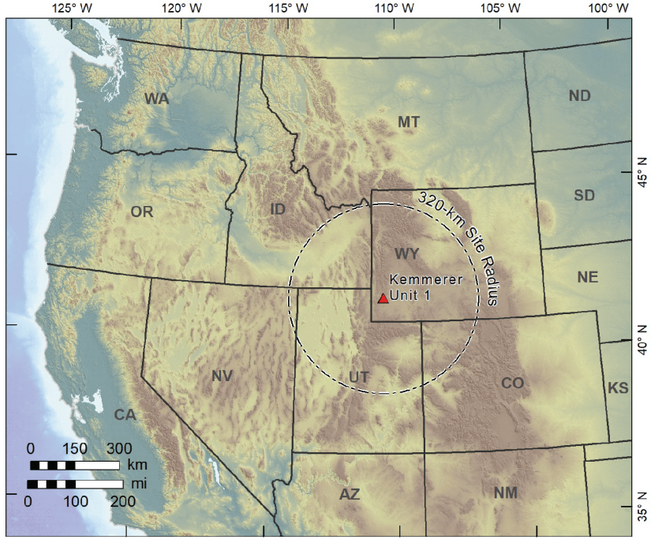







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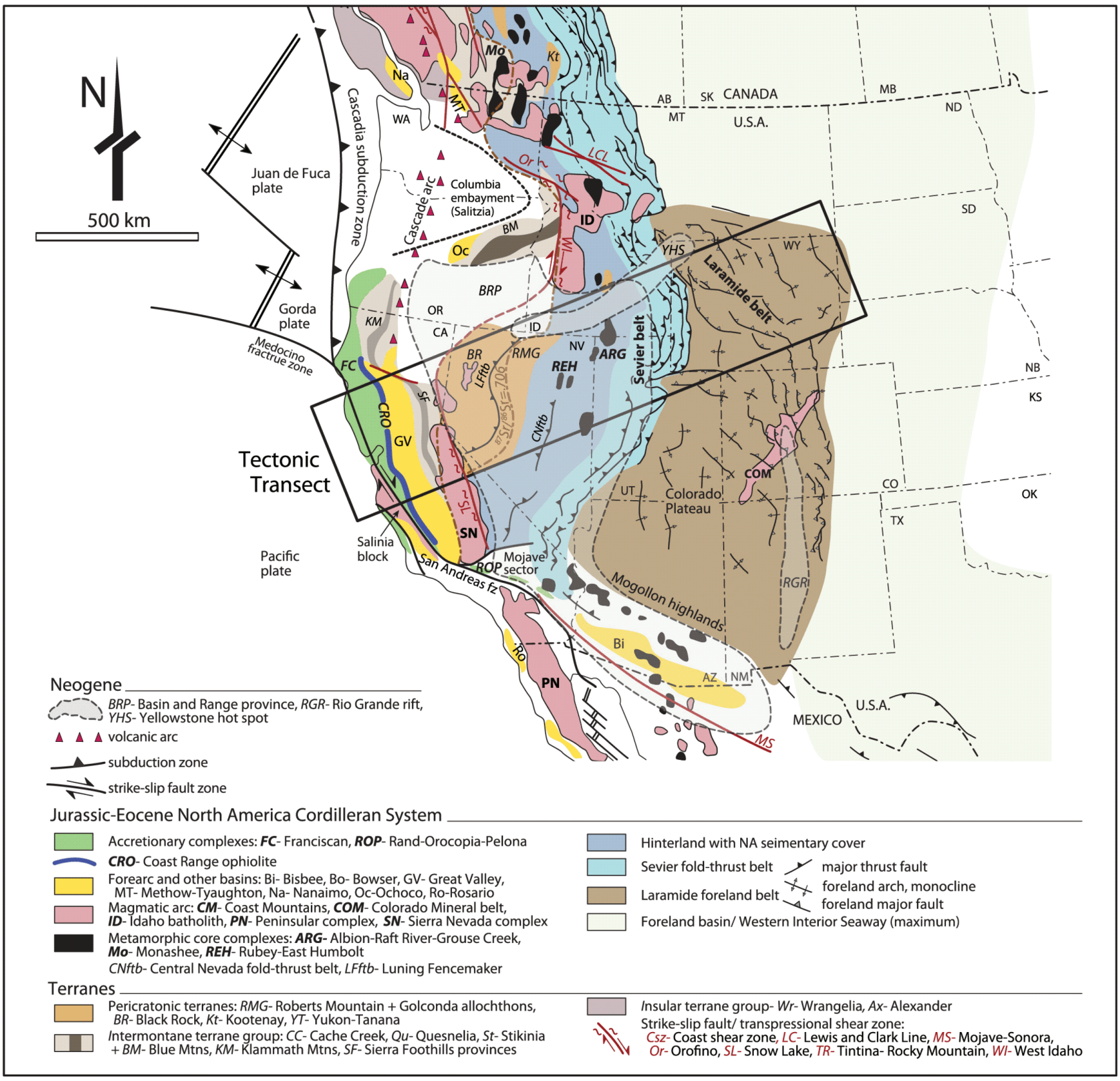
**Figure 2.6-4 Generalized Map of North American Cordillera**





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## Figure 2.6-5 Generalized Geologic Map of Site Vicinity (40 km radius)

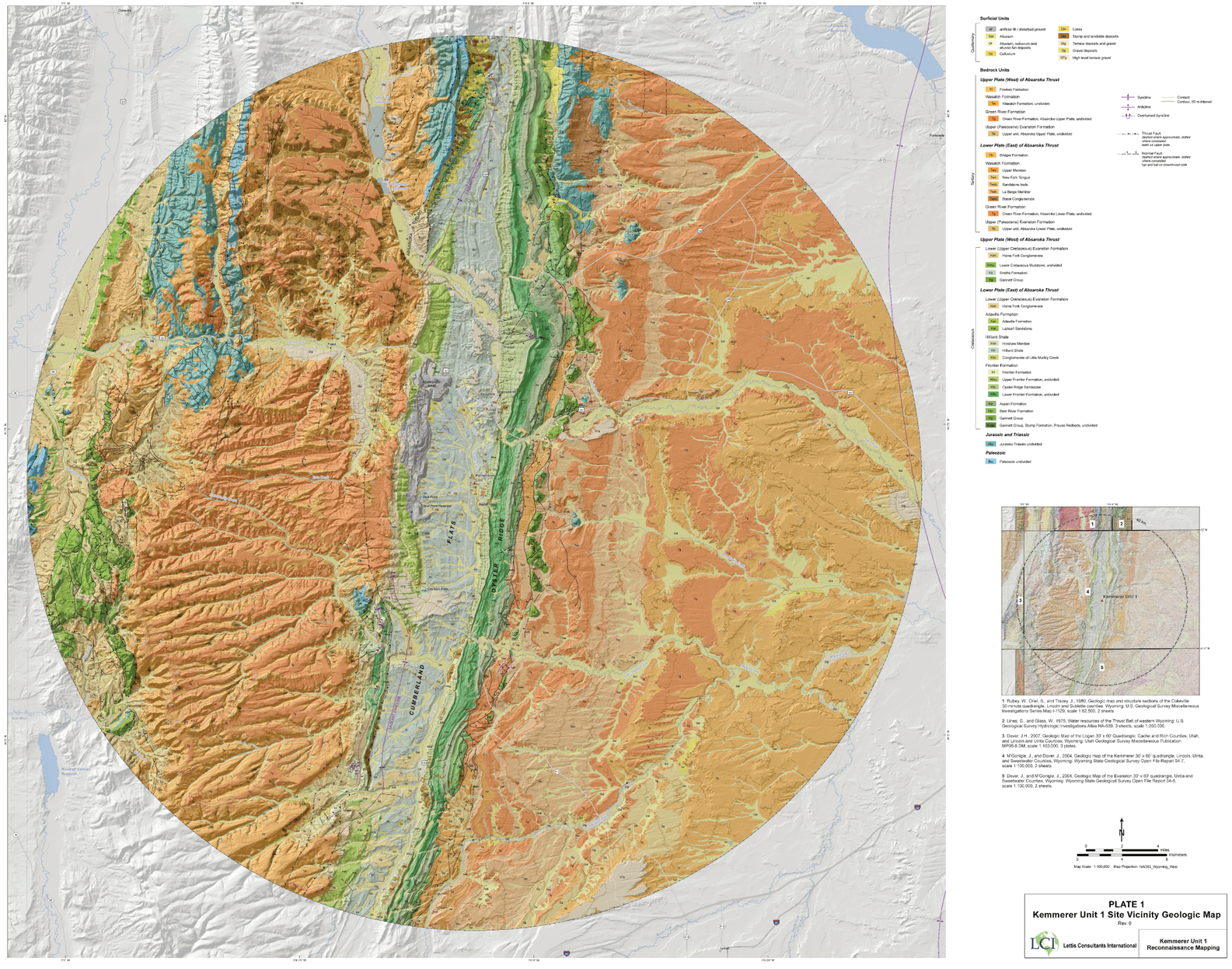



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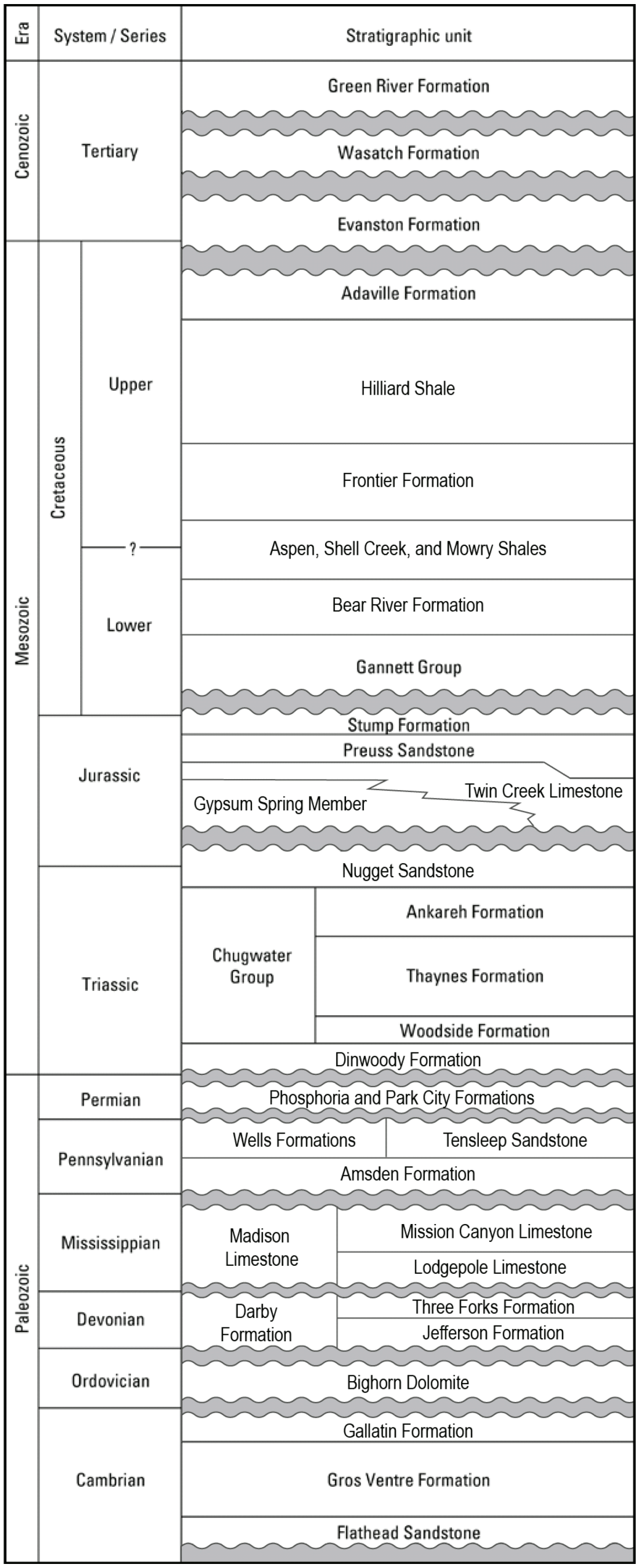
**Figure 2.6-6 Generalized Stratigraphic Columns of Wyoming-Idaho-Utah Thrust Belt**





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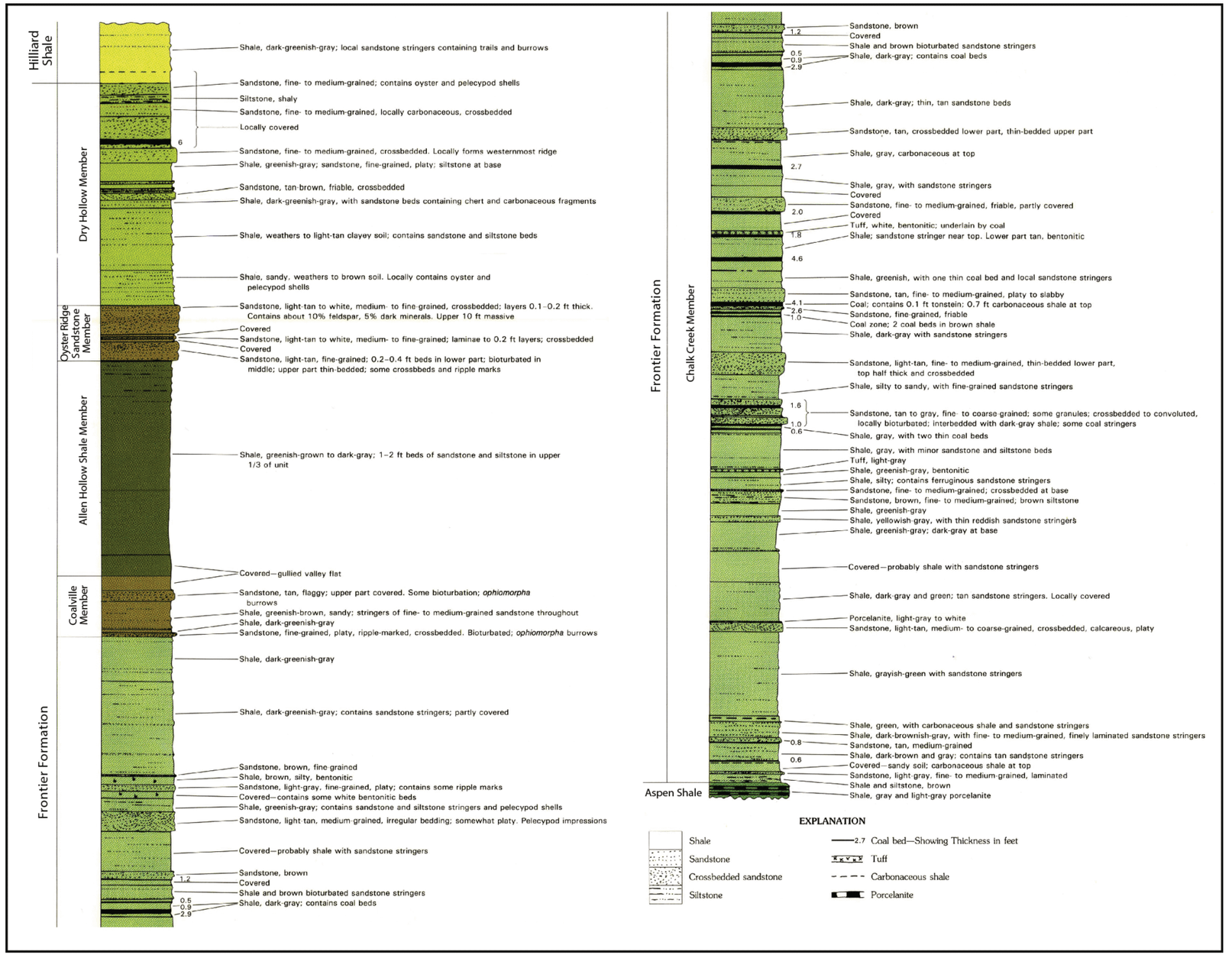
## Figure 2.6-7 Stratigraphic Column of Elkol Quadrangle





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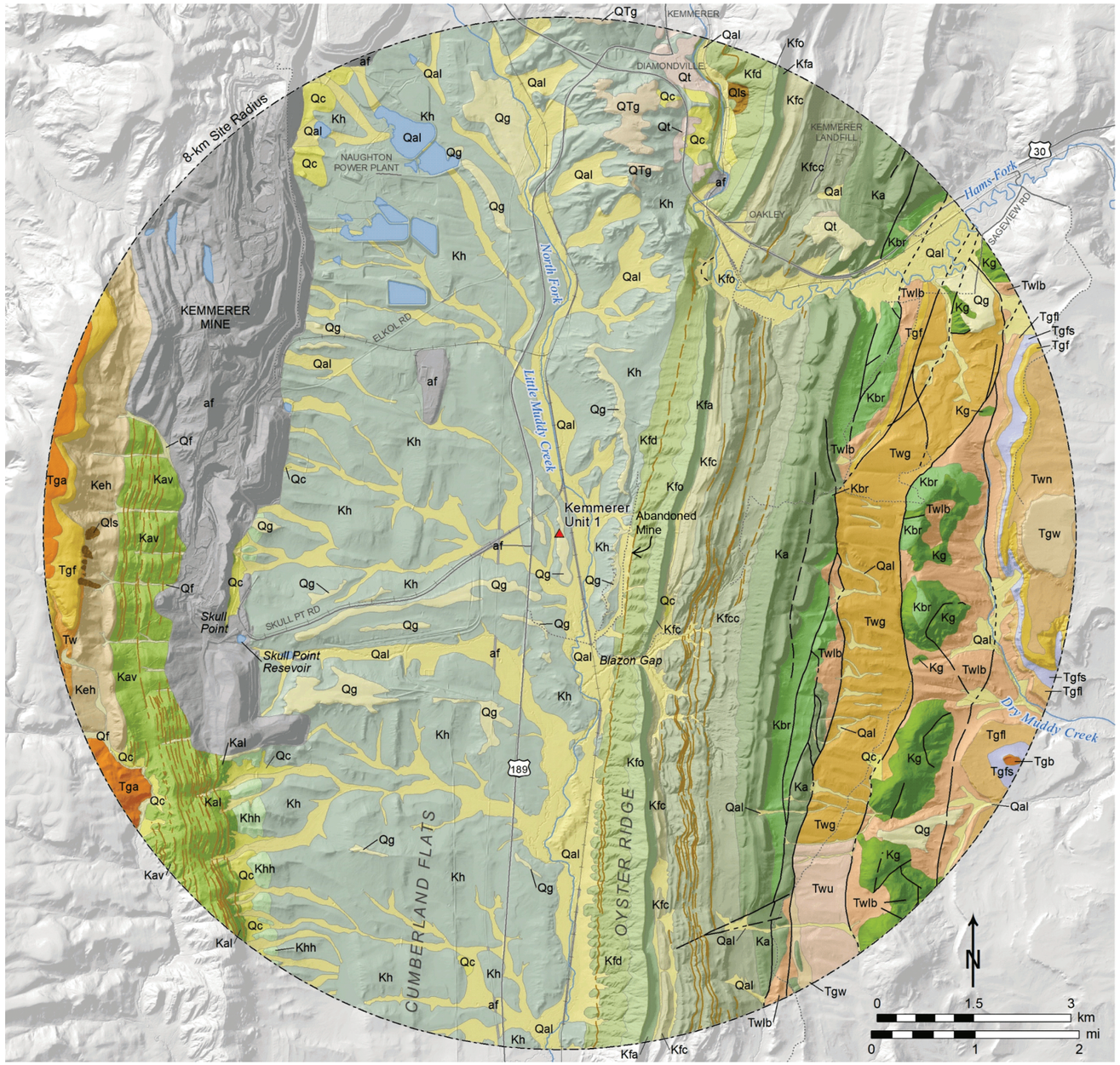
## Figure 2.6-8 Geologic Map of Site Area (8 km Radius)




**Quaternary**

**Surficial Units**

| | | | |
|---|---|---|---|
| af | artificial fill / disturbed ground | Qf | Alluvium, colluvium and alluvial-fan deposits |
| Qal | Alluvium | Qls | Slump and landslide deposits |
| Qc | Colluvium | Qt | Terrace deposits |
| Qg | Gravel deposits | QTg | High terrace deposits |

**Eocene**

**Wasatch Formation**

| | |
|---|---|
| Twu | Upper Member |
| Twn | New Fork Tongue |
| Twlb | La Barge Member |
| Tw | Main body |

**Green River Formation**

| | |
|---|---|
| Tga | Angelo Member |
| Tgb | Fossil Buttle Member |
| Tgw | Wilkins Peak Member |
| Tgf | Fontenelle Tongue |
| Tgfs | Middle Part Fontenelle Tongue |
| Tgfl | Lower Part Fontenelle Tongue |
| Twg | Wasatch and Green River Fms undivided |

**Evanston Formation**

| | |
|---|---|
| Keh | Hams Fork Conglomerate |

**Adaville Formation**

| | |
|---|---|
| Kav | Adaville Formation |
| Kal | Lazeart Sandstone |

**Cretaceous**

**Hilliard Shale**

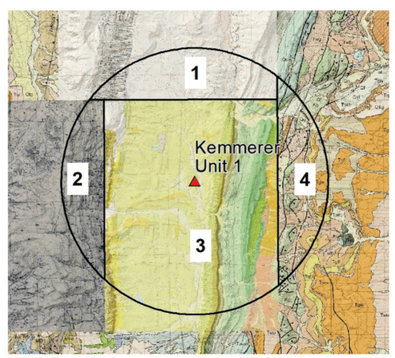
| | |
|---|---|
| Khh | Hinshaw Member |
| Kh | Hilliard Shale |

**Frontier Formation**

| | |
|---|---|
| Kfd | Dry Hollow Member |
| Kfo | Oyster Ridge Sandstone |
| Kfa | Allen Hollow Member |
| Kfc | Coalville Member |
| Kfcc | Chalk Creek Member |
| Ka | Aspen Formation |
| Kbr | Bear River Formation |
| Kg | Gannet Group |

- coal, dashed
- coal, dotted
- coal, solid
- contact, solid
- fault, dashed
- fault, dotted
- fault, solid

1 Weaver and M'Gonigle (1987)
2 M'Gonigle (1979)
3 M'Gonigle (1992)
4 M'Gonigle and Dover (2004)





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## Figure 2.6-9 Geologic Map of Site Location (1 km Radius)

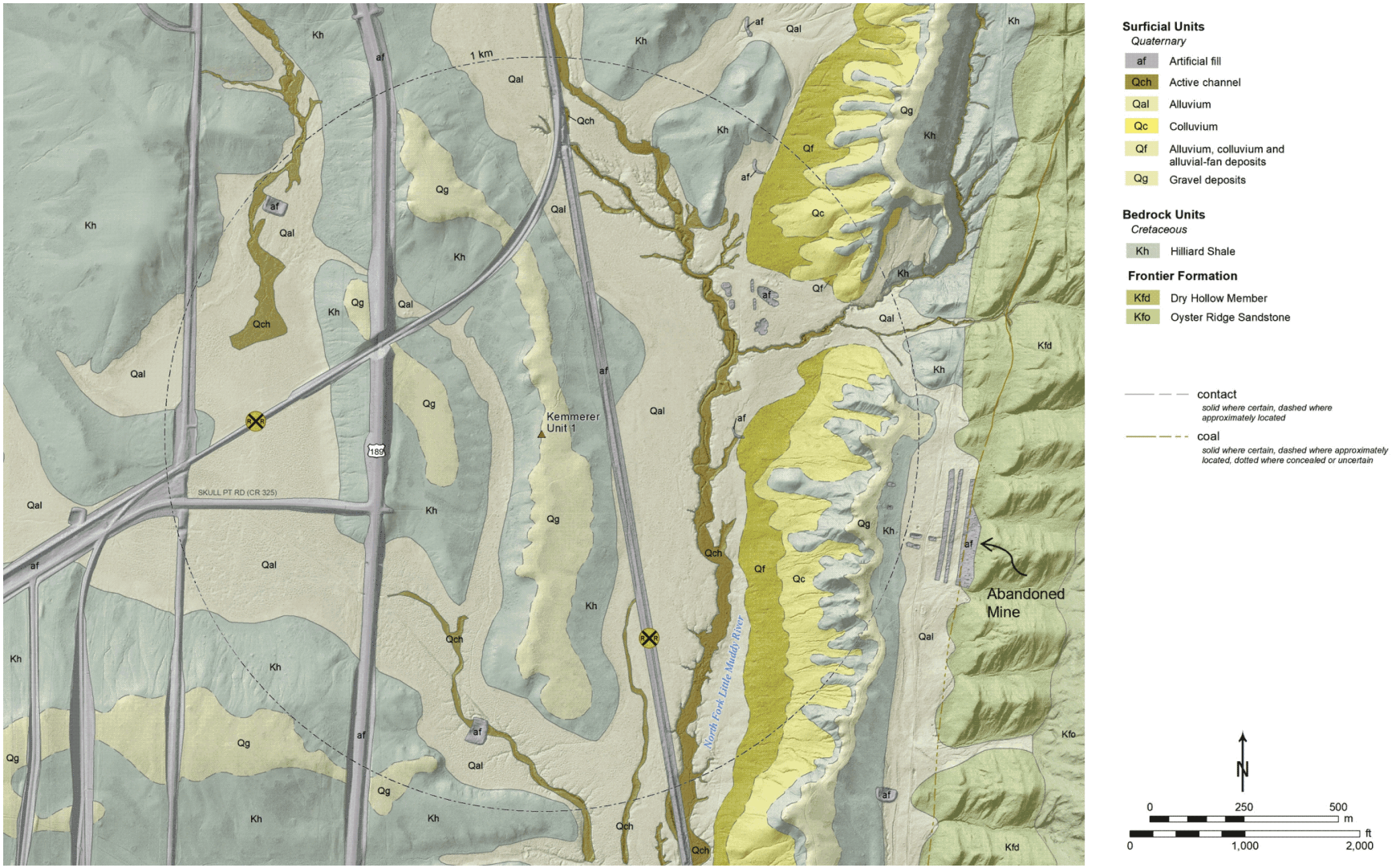






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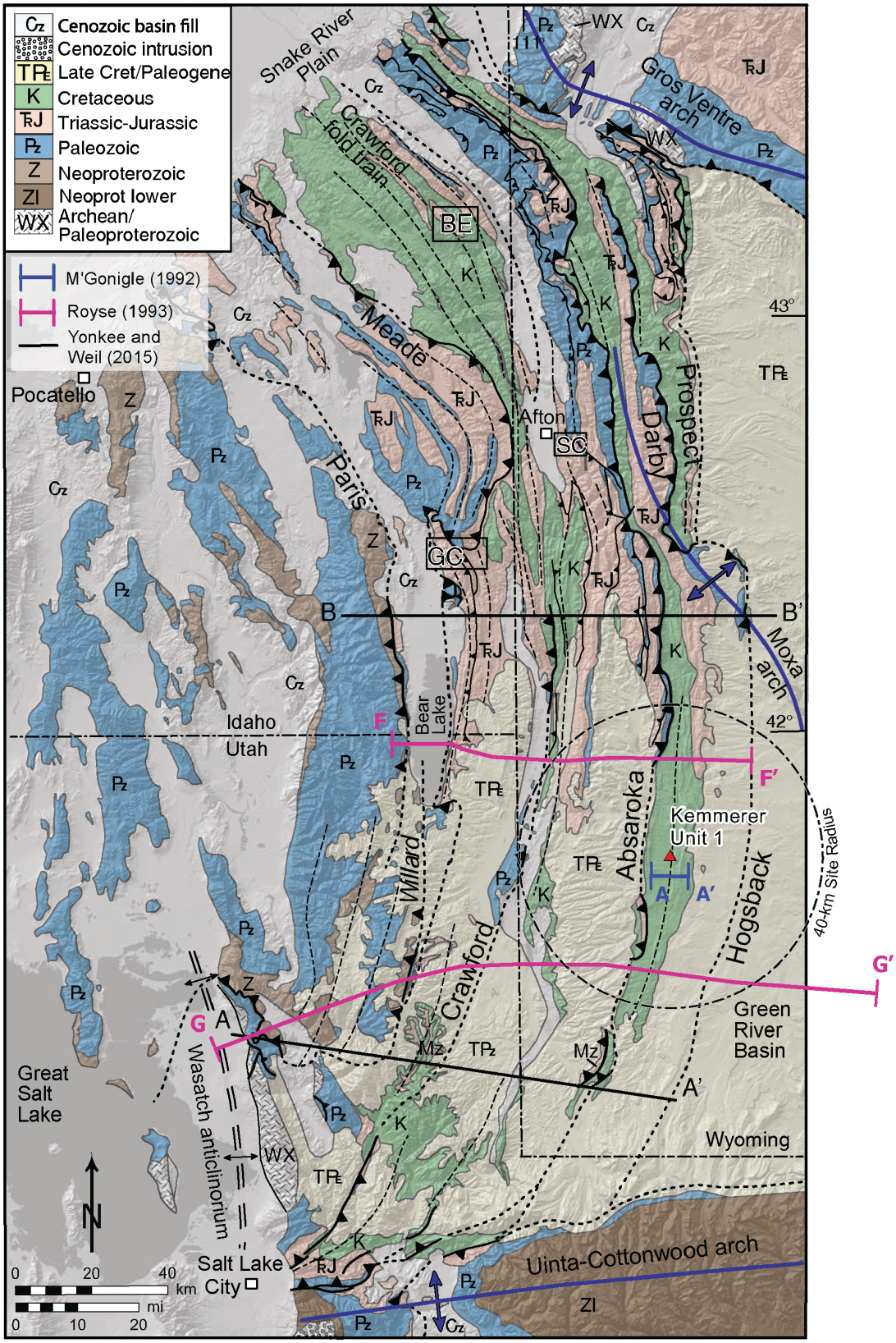
**Figure 2.6-10 Geologic Map of Wyoming Salient of the Sevier Orogenic Belt**





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## Figure 2.6-11 Cross Sections of the Sevier Orogeny in Wyoming

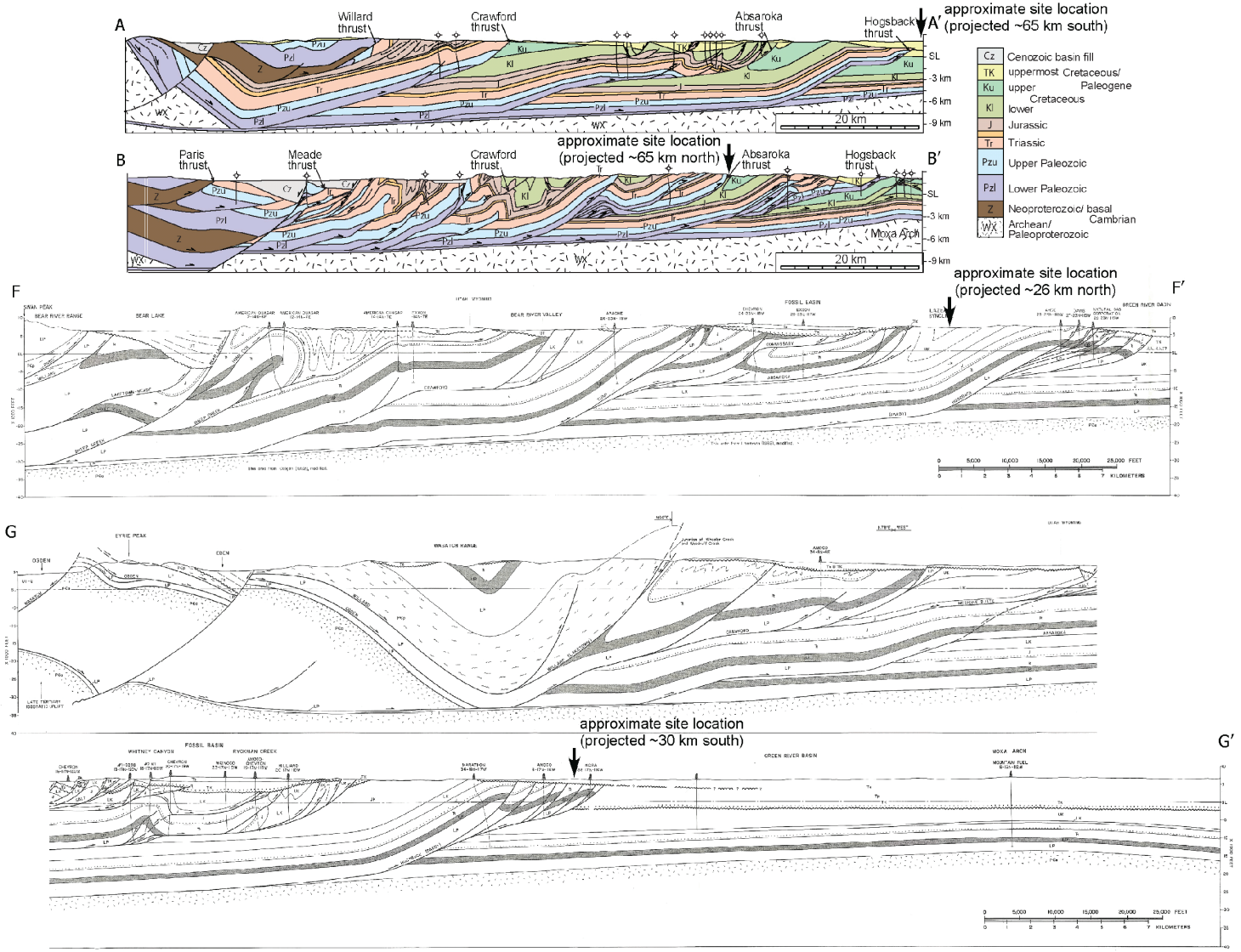



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**Figure 2.6-12 Photograph of Frontier Formation**

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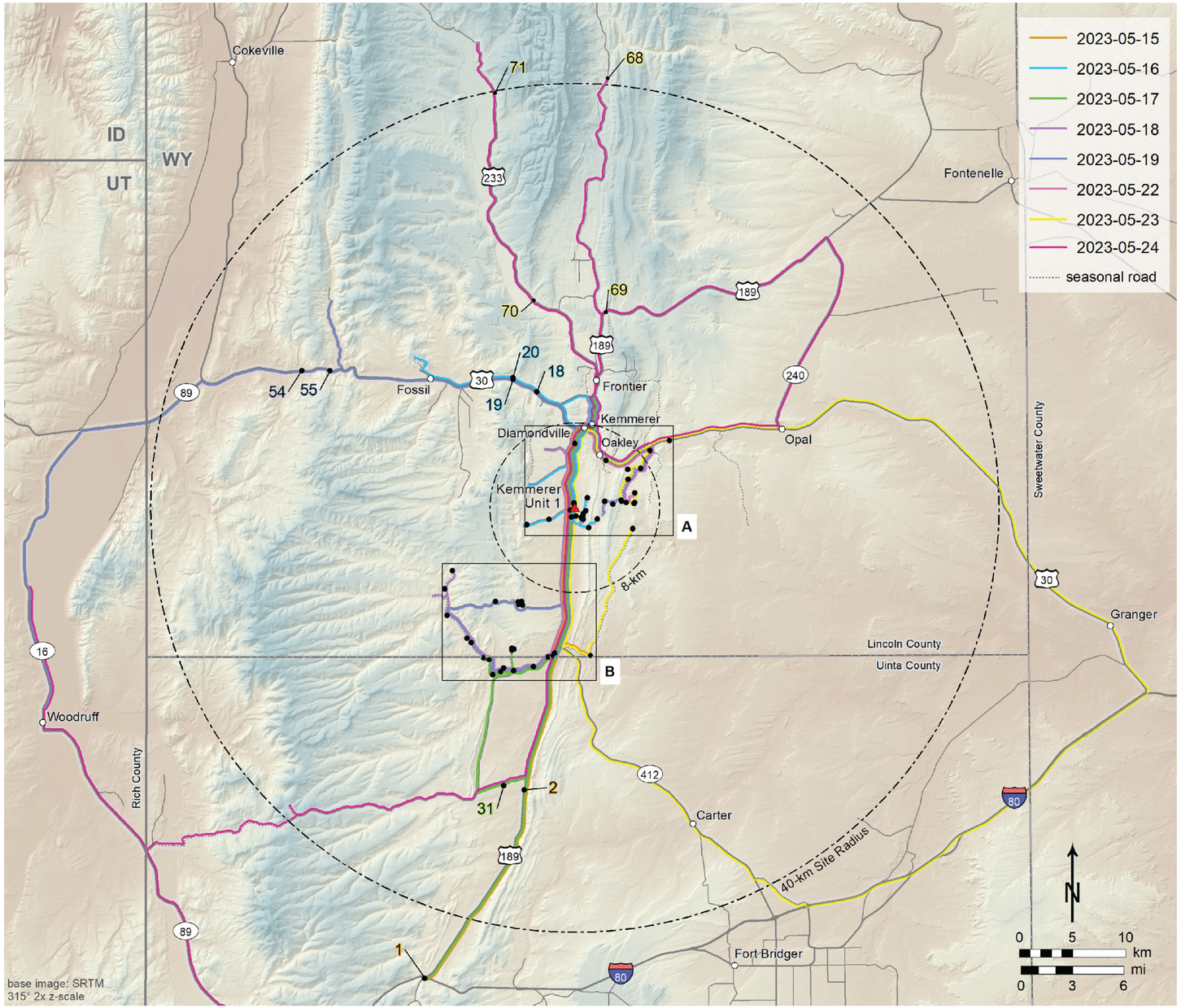
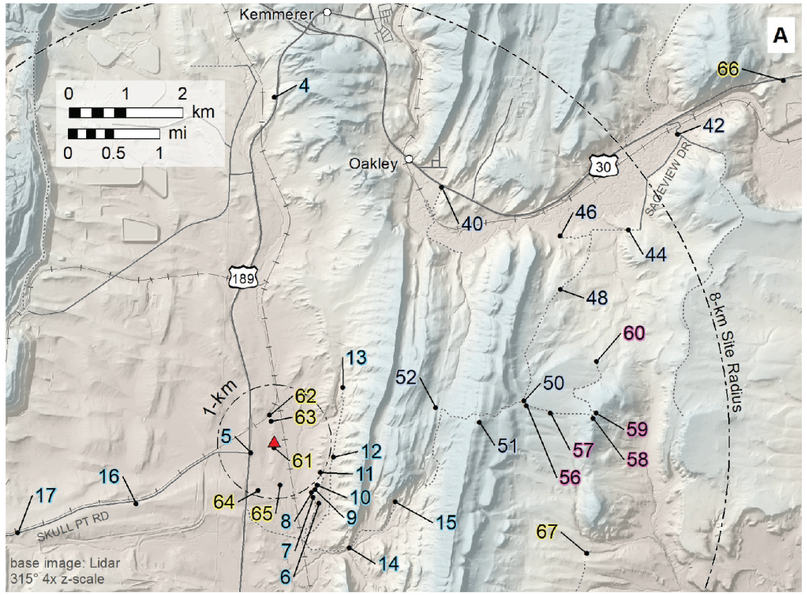
**Figure 2.6-13 GPS Tracks and Field Stations**



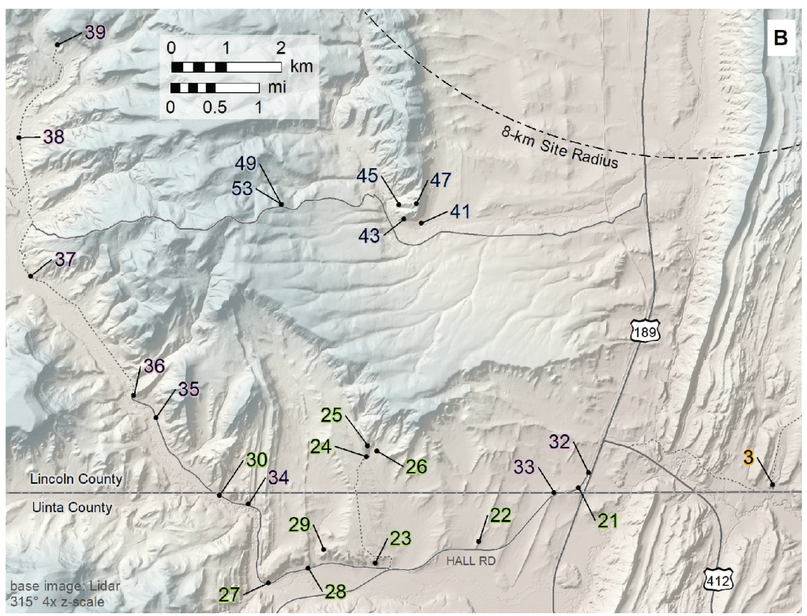






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**Figure 2.6-14 Rock Core**

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**Figure 2.6-15 Photograph of Kemmerer Unit 1 Site**

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**Figure 2.6-16 Coal Beds in Adaville Formation**

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**Figure 2.6-17 Photograph of Hams Fork Conglomerate**

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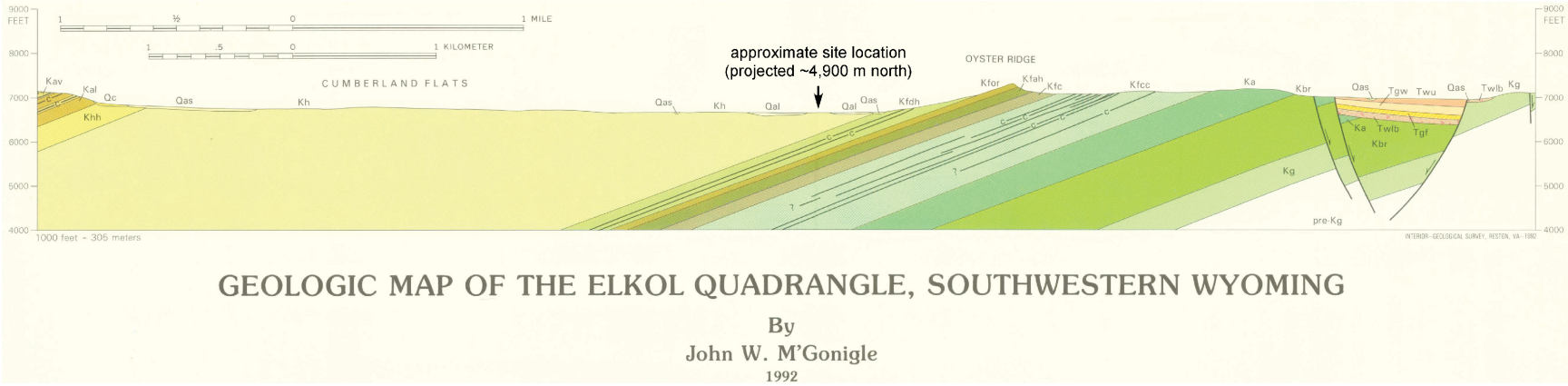
**Figure 2.6-18 Cross Section of the Elkol Quadrangle**




GEOLOGIC MAP OF THE ELKOL QUADRANGLE, SOUTHWESTERN WYOMING

By

John W. M'Gonigle

1992


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**Figure 2.6-19 Looking Across Cumberland Flats**

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**Figure 2.6-20 Photograph of Qg**

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**Figure 2.6-21 SSHAC Project Region Showing 320 km radii**

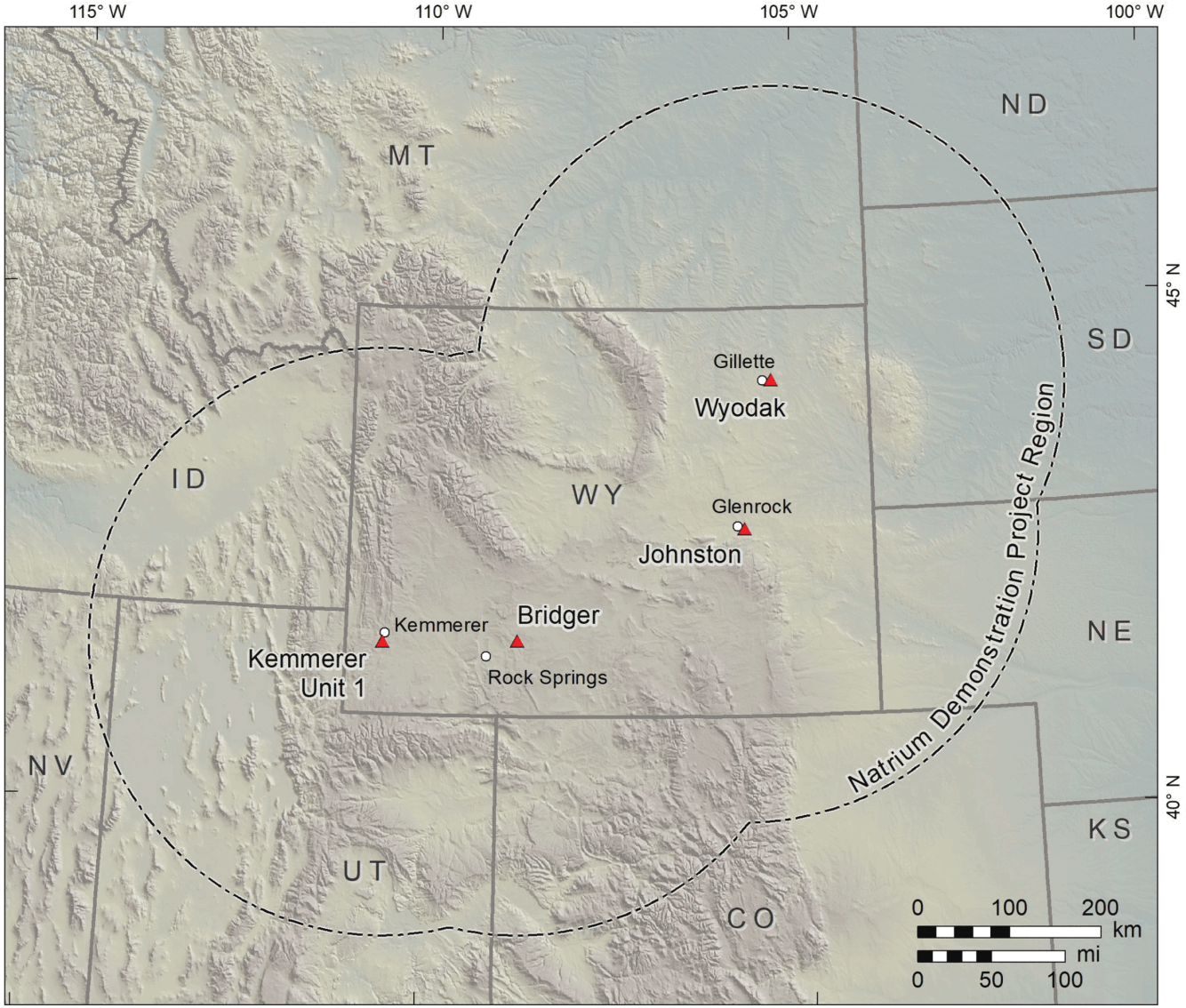



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**Figure 2.6-22 Regional Earthquake Catalog for Magnitude E[M] ≥ 2.3**

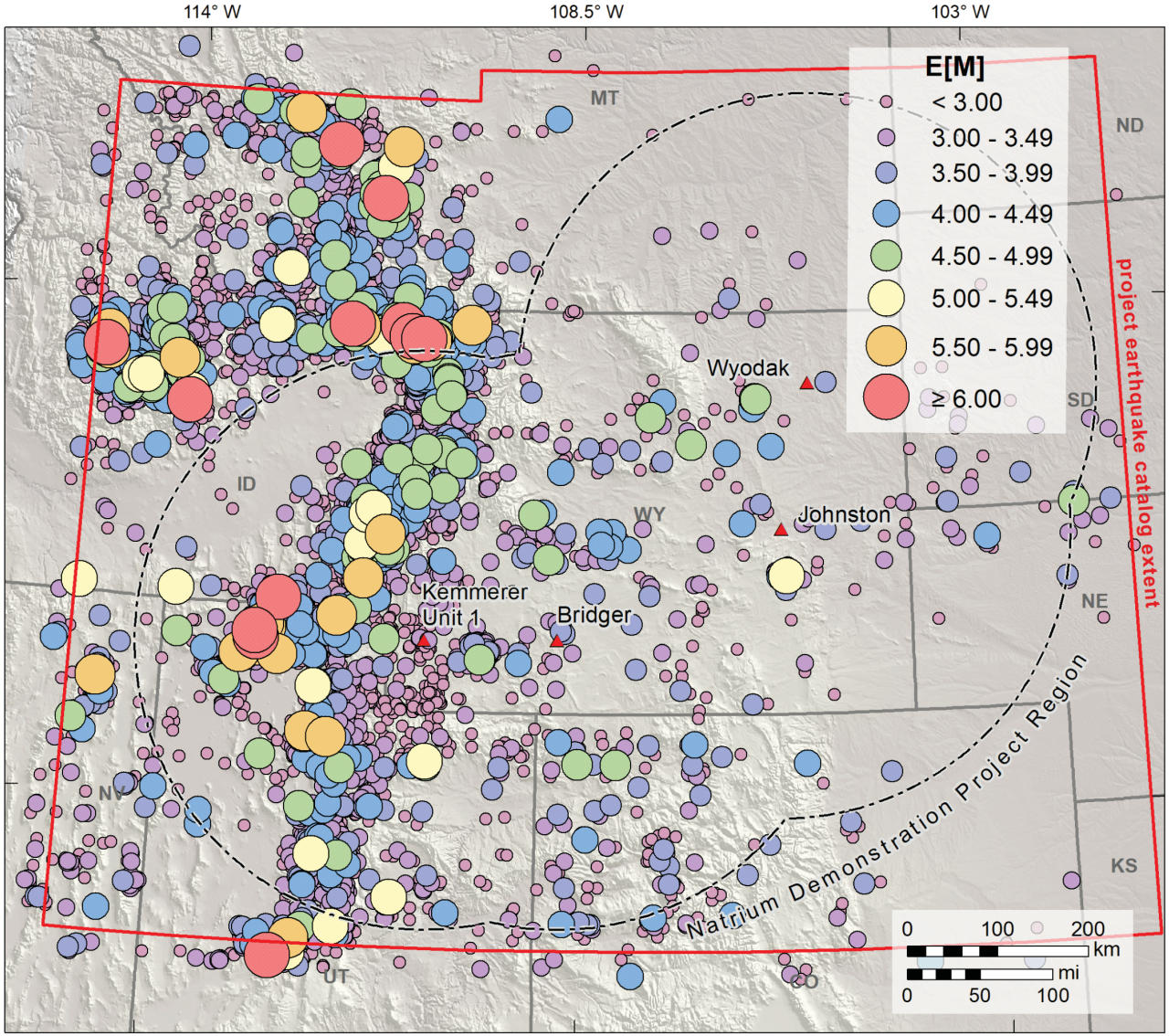



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**Figure 2.6-23 Master Logic Tree for the SSC Model**

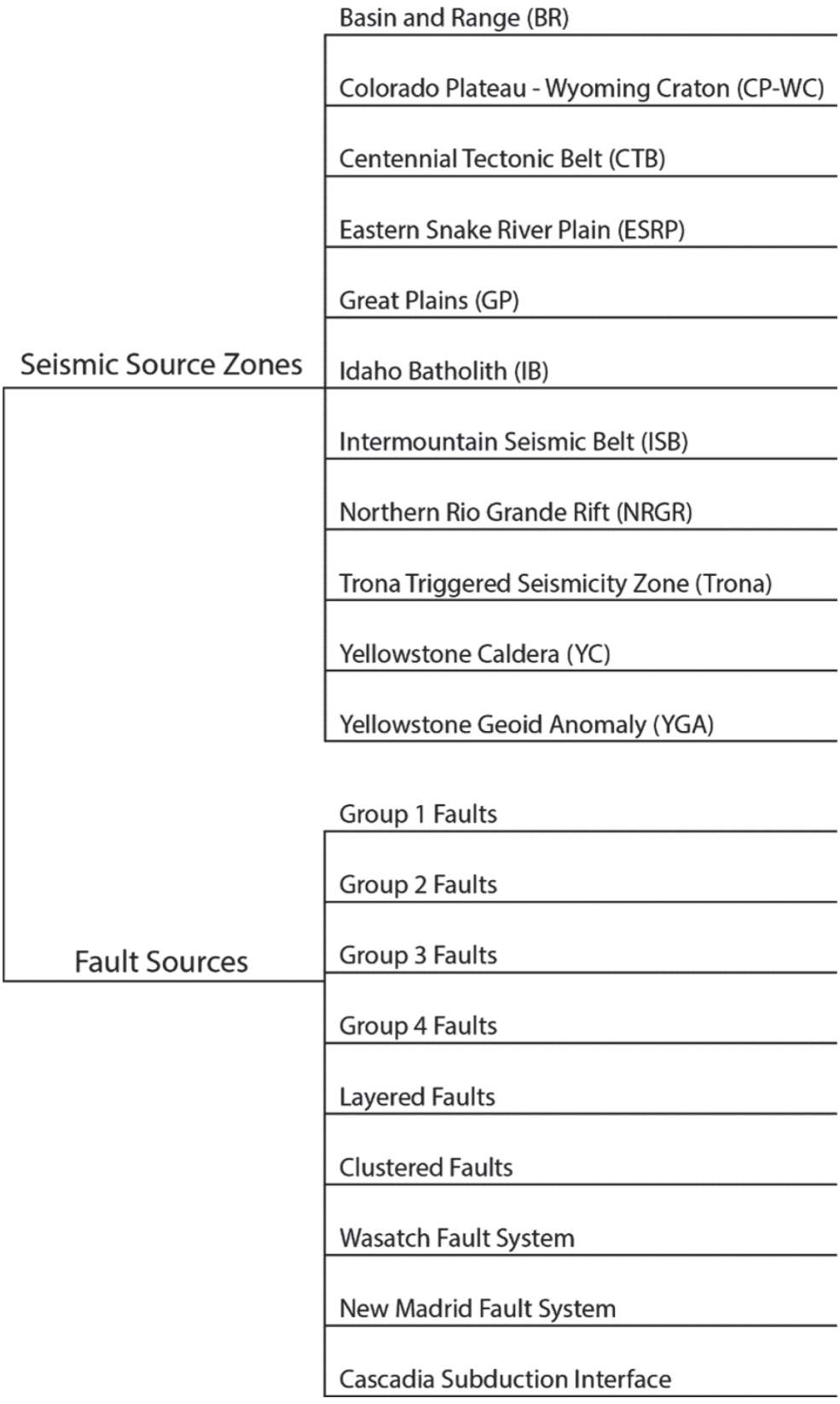



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**Figure 2.6-24 Map of Seismic Source Zones and Fault Sources in the Site Region**

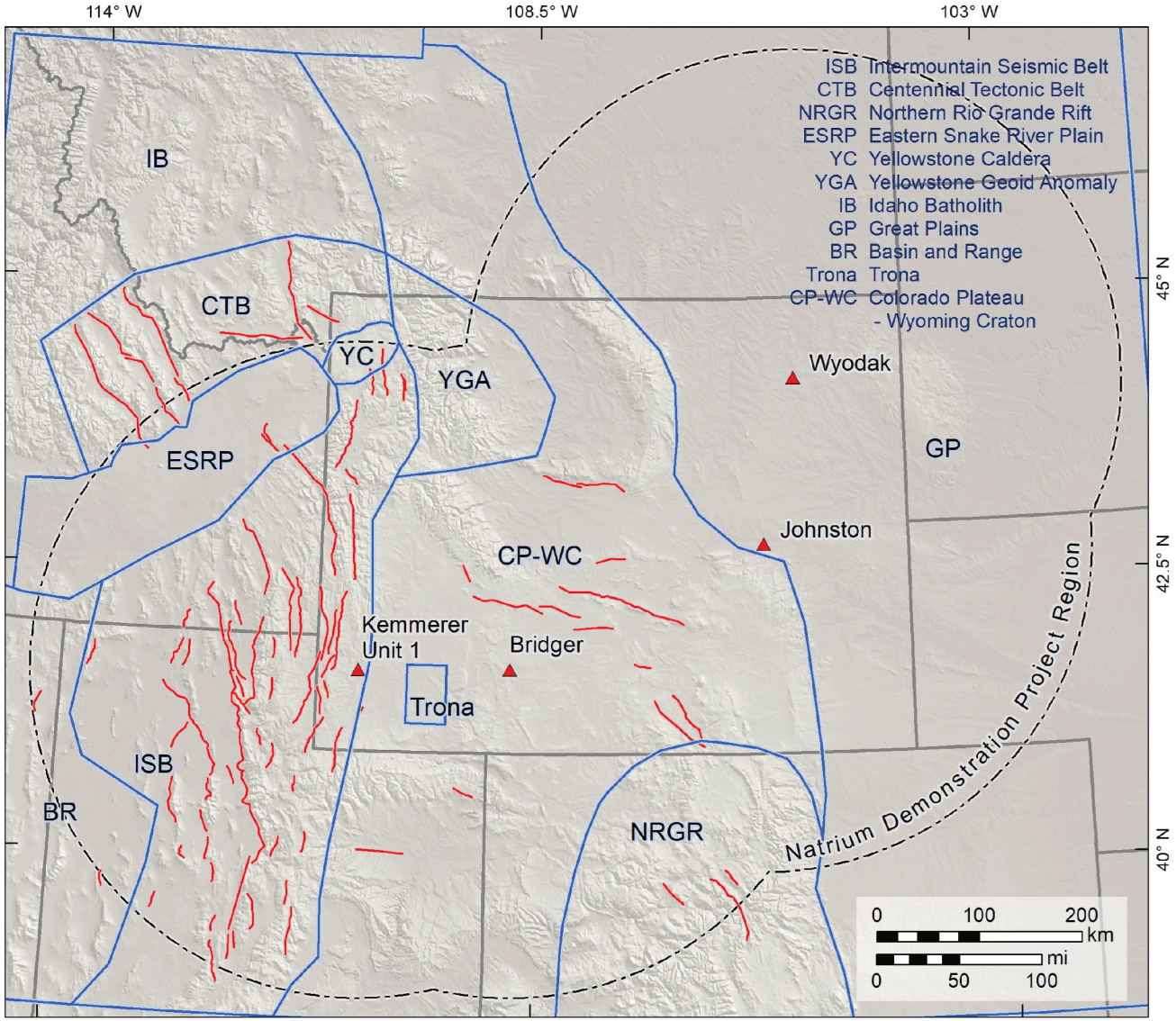



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**Figure 2.6-25 Map of Distant Seismic Sources Beyond the Site Region showing Distances from the Kemmerer Unit 1 Site**

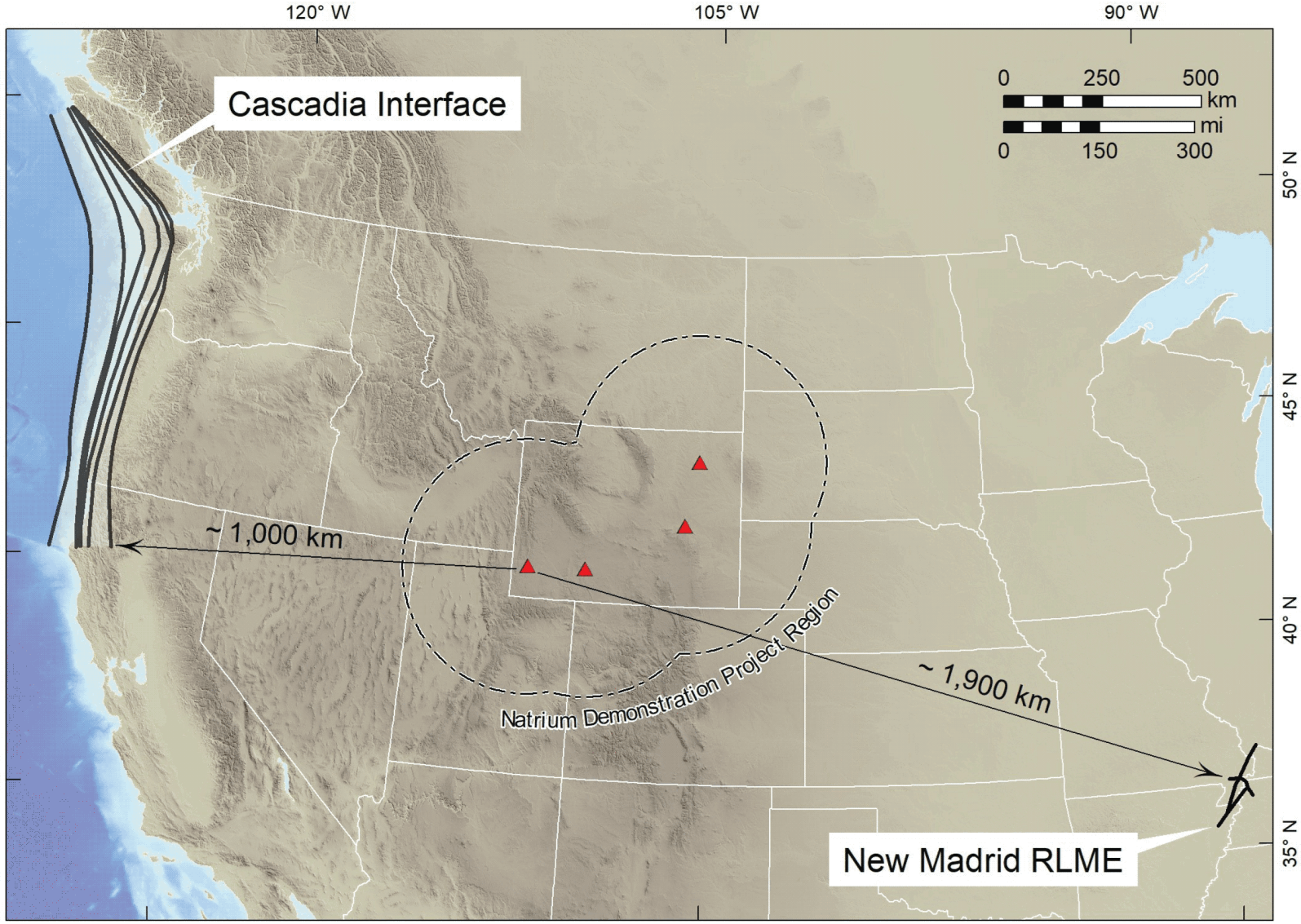






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**Figure 2.6-26 Seismic Source Zones Defined for the Site Region**

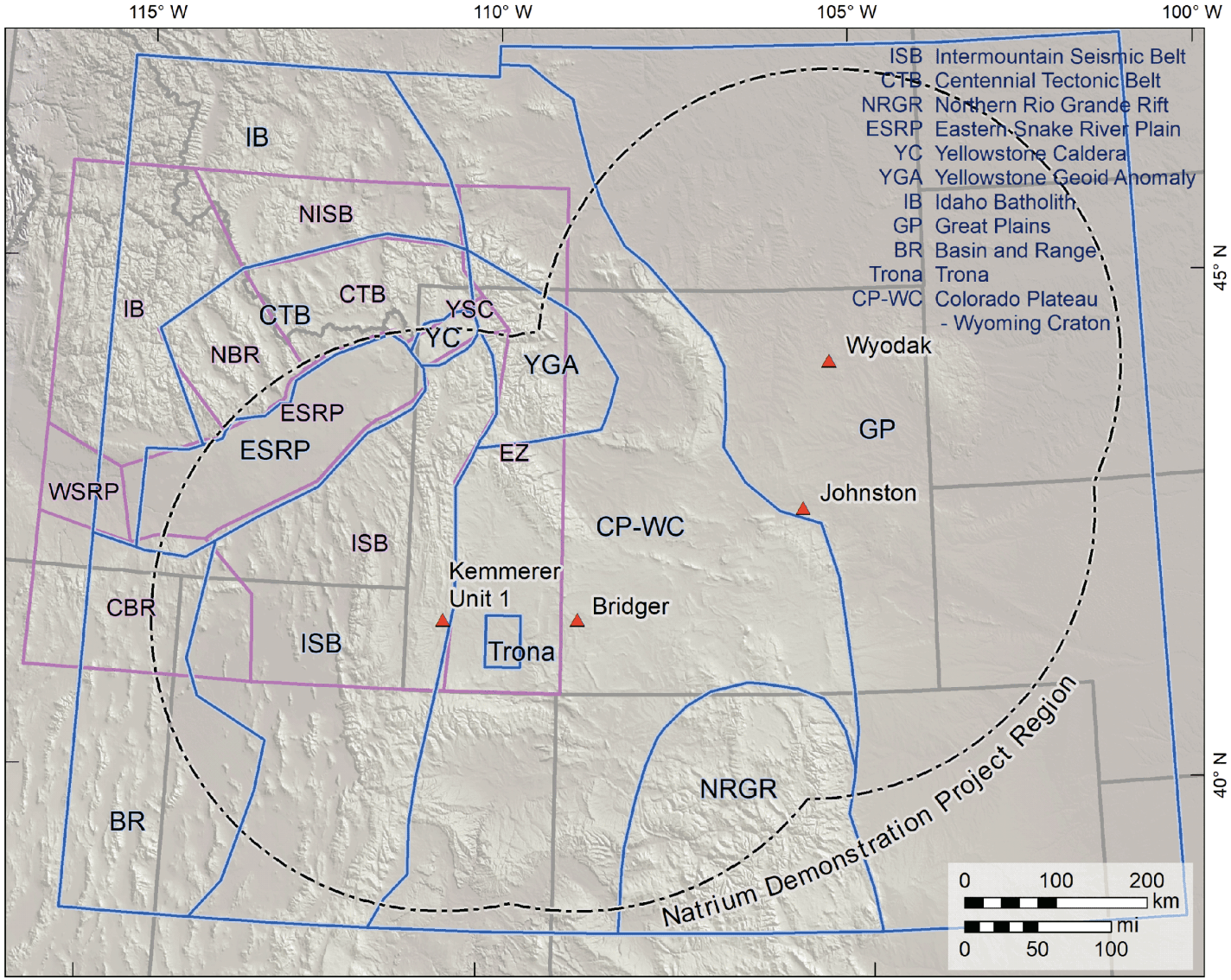



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**Figure 2.6-27 Master Logic Tree for Seismic Source Zones**

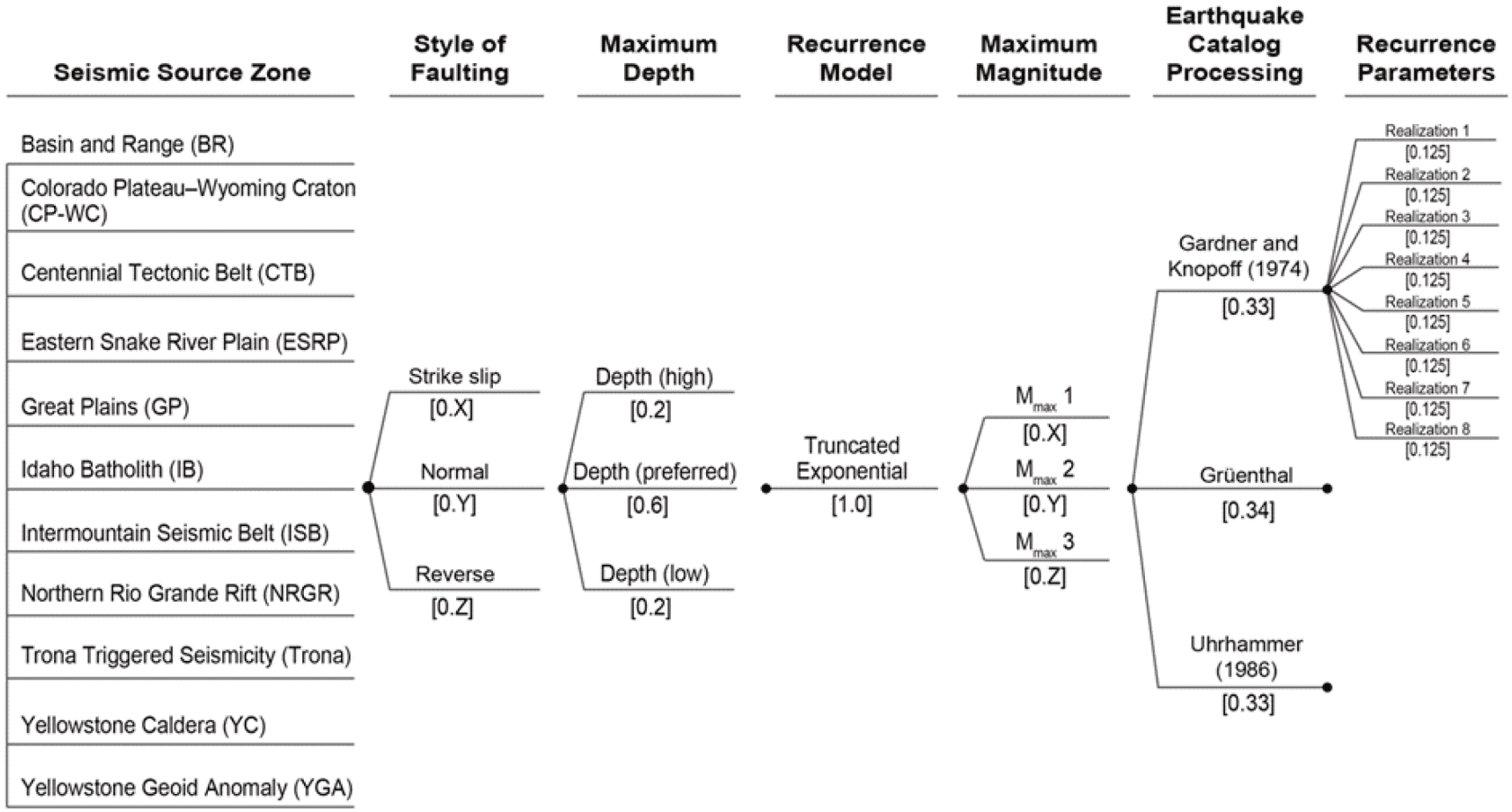



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**Figure 2.6-28 Contours of Seismogenic Thickness in the Site Region**

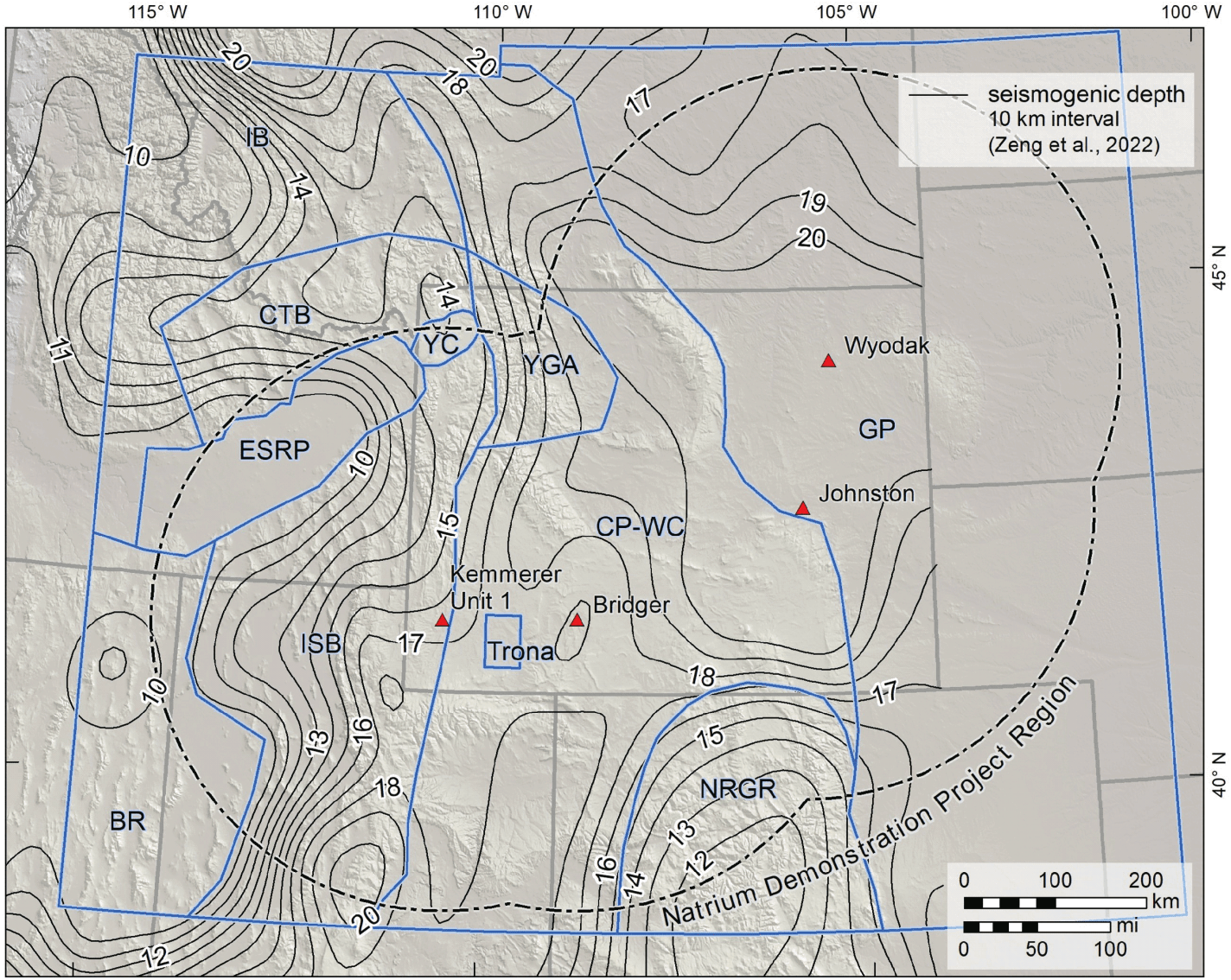



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**Figure 2.6-29 Contribution of Buoyancy in the Upper Mantle to Average Topographic Elevation**

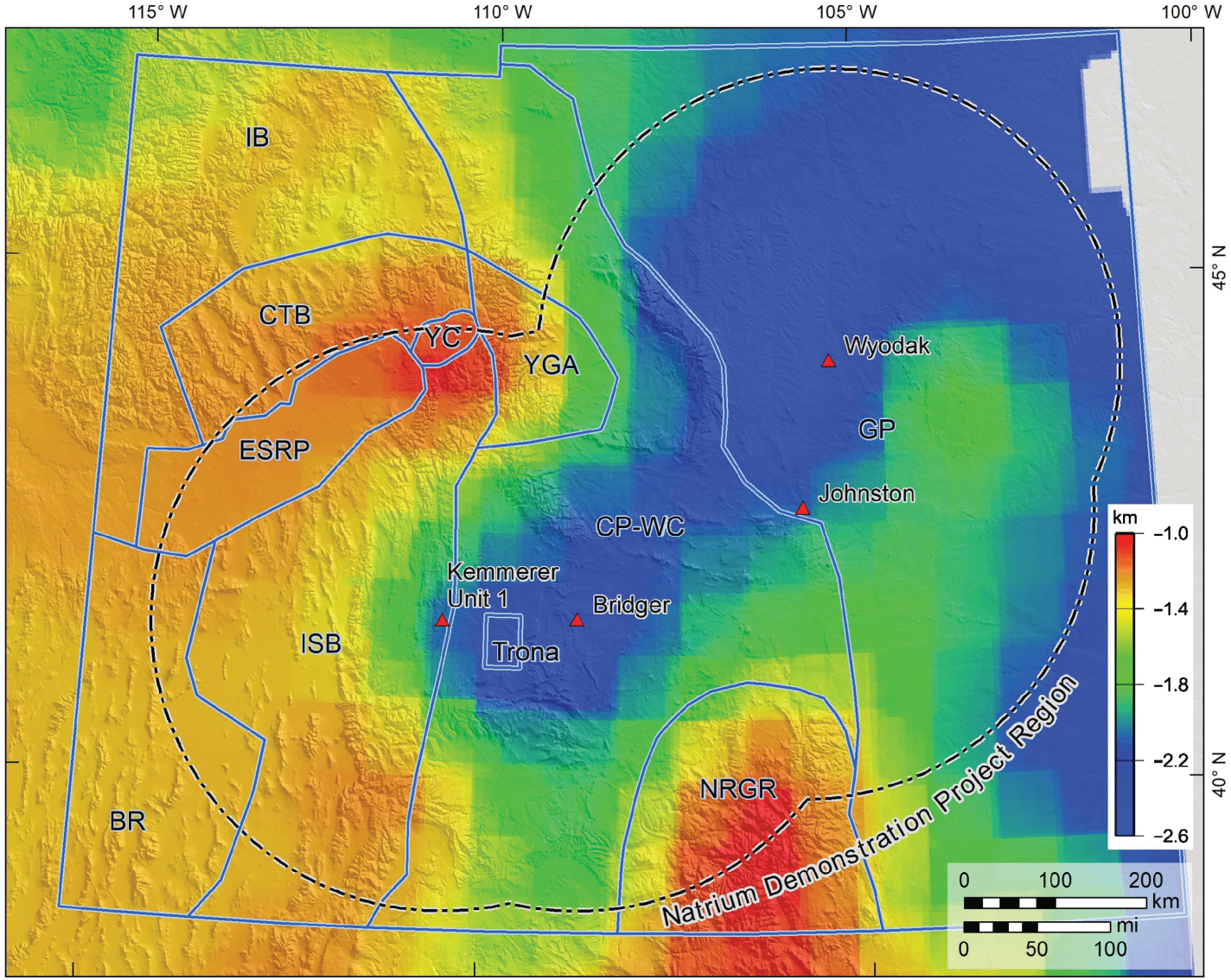



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**Figure 2.6-30 Heat Flow in the Site Region**

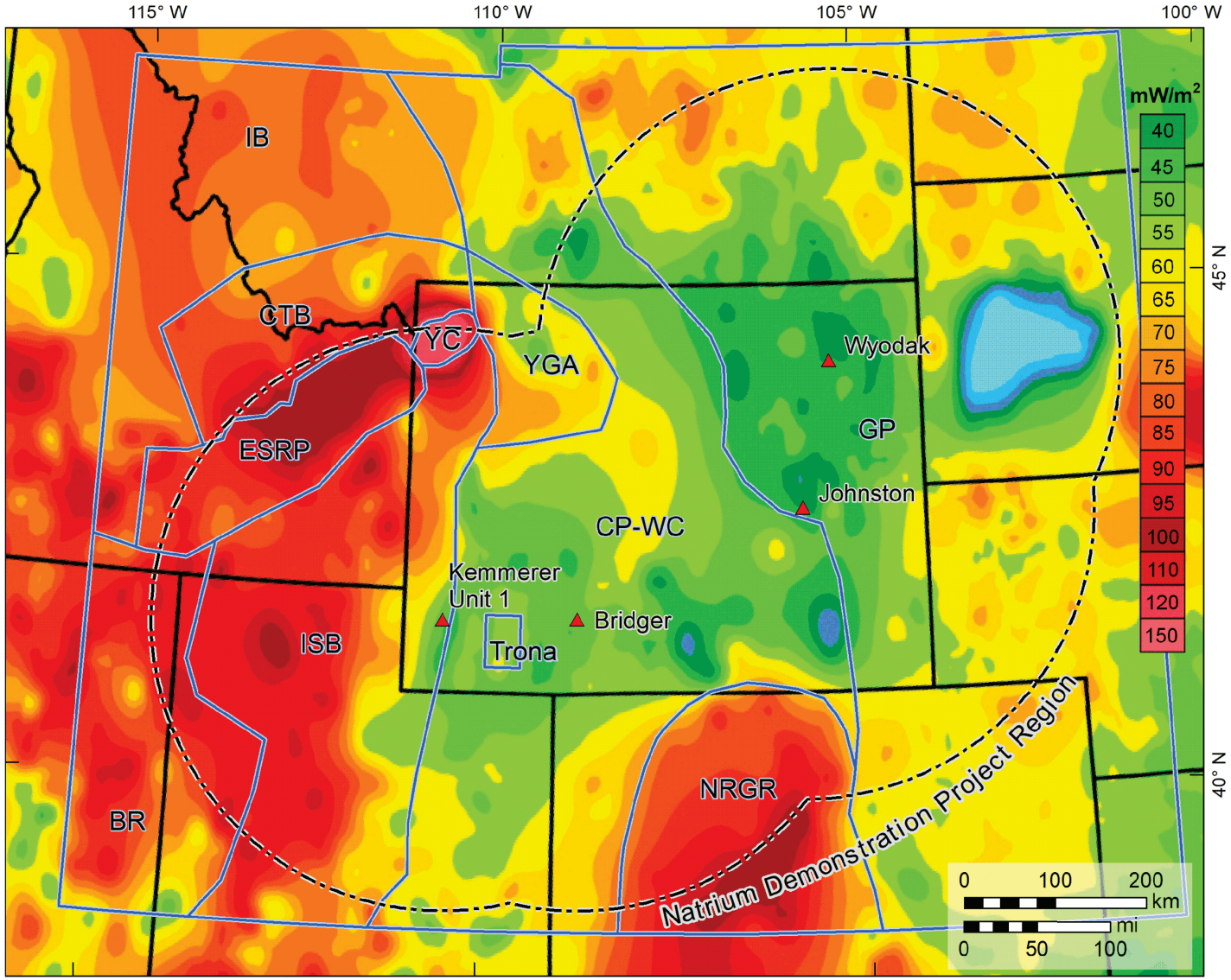






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**Figure 2.6-31 Faults in the Site Region from the USGS QFFD**

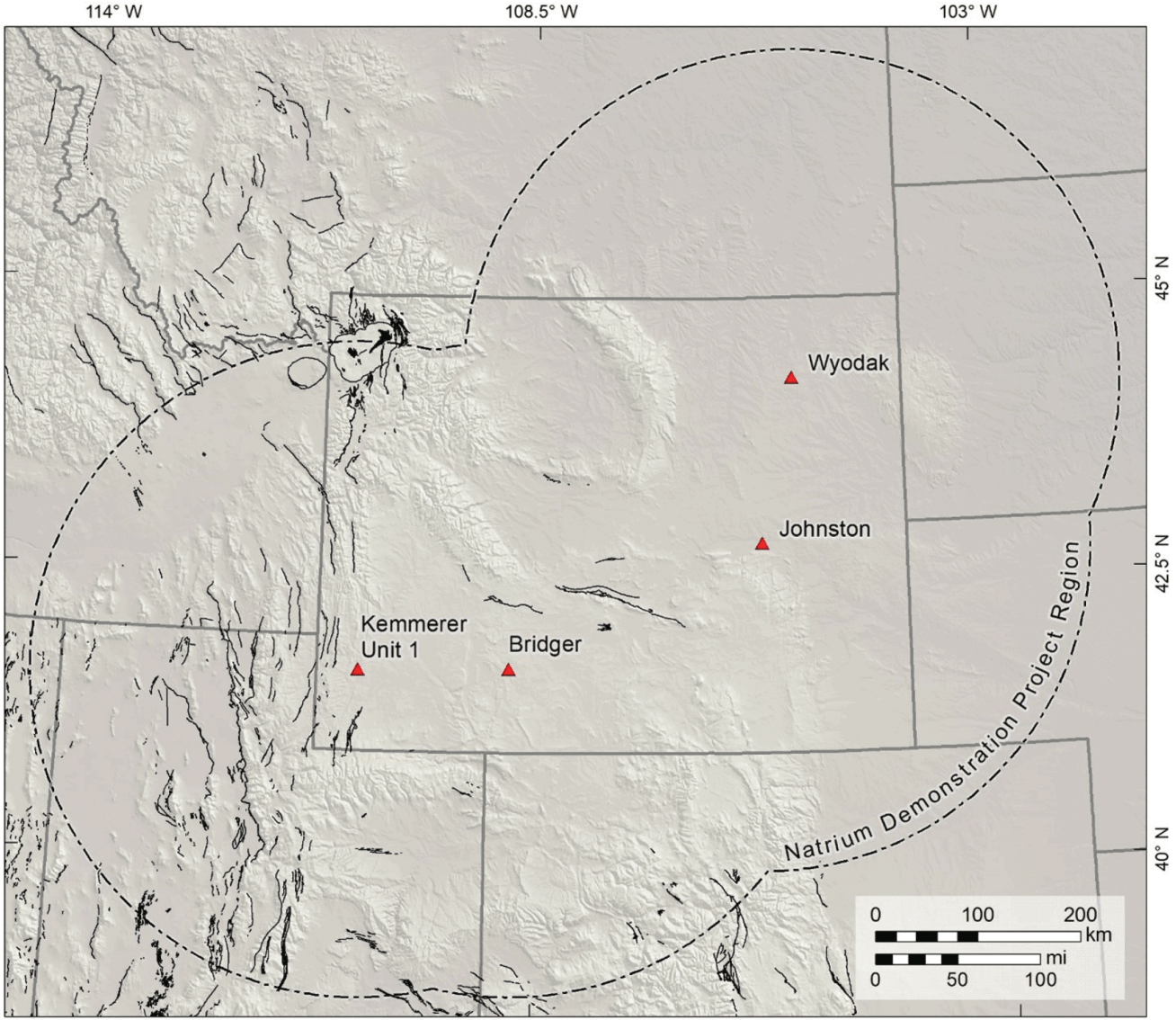






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**Figure 2.6-32 Fault Sources in the Site Region from the USGS NSHM23**

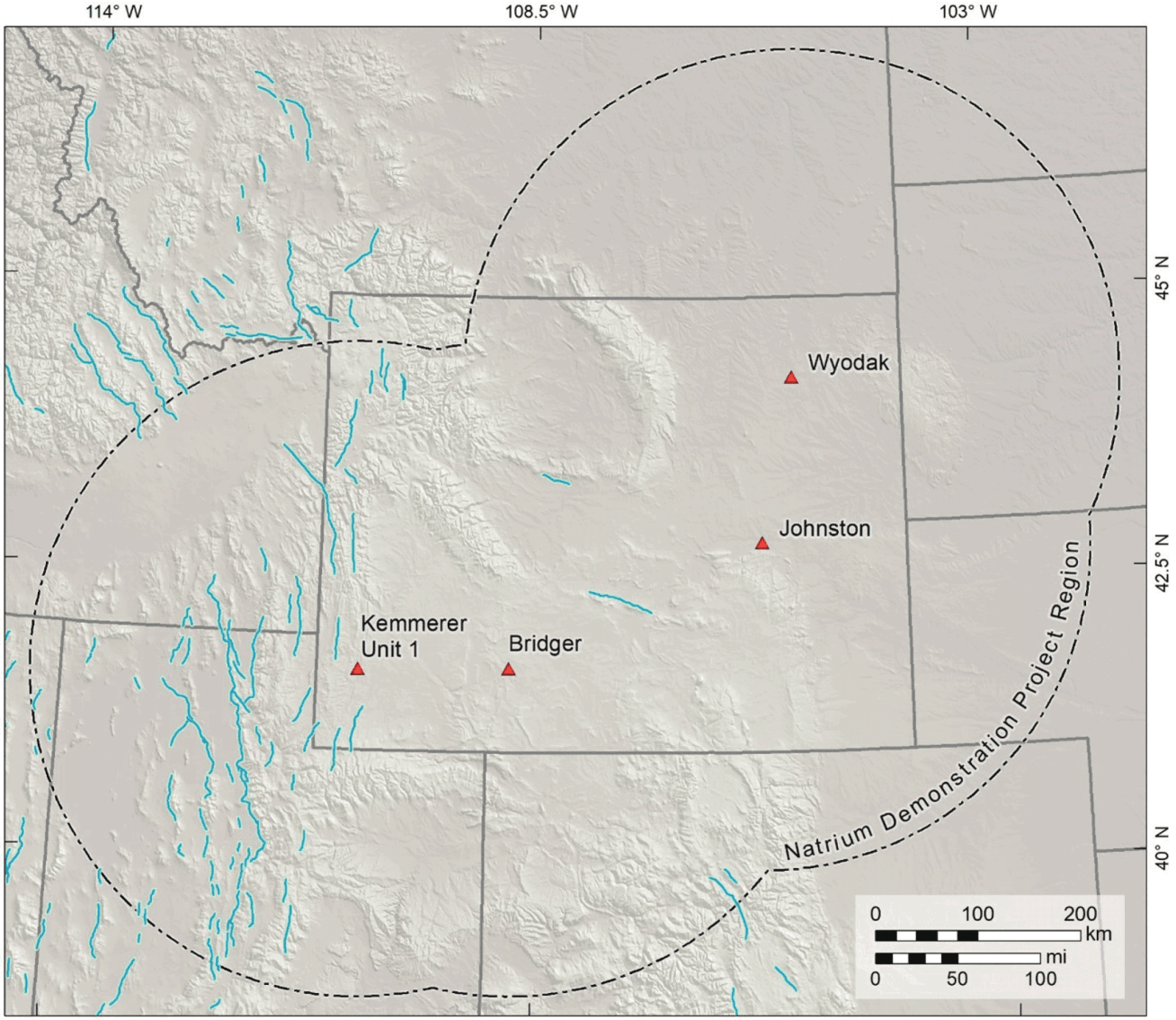






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**Figure 2.6-33 Fault Sources in the Site Region Subdivided by Group or Region**

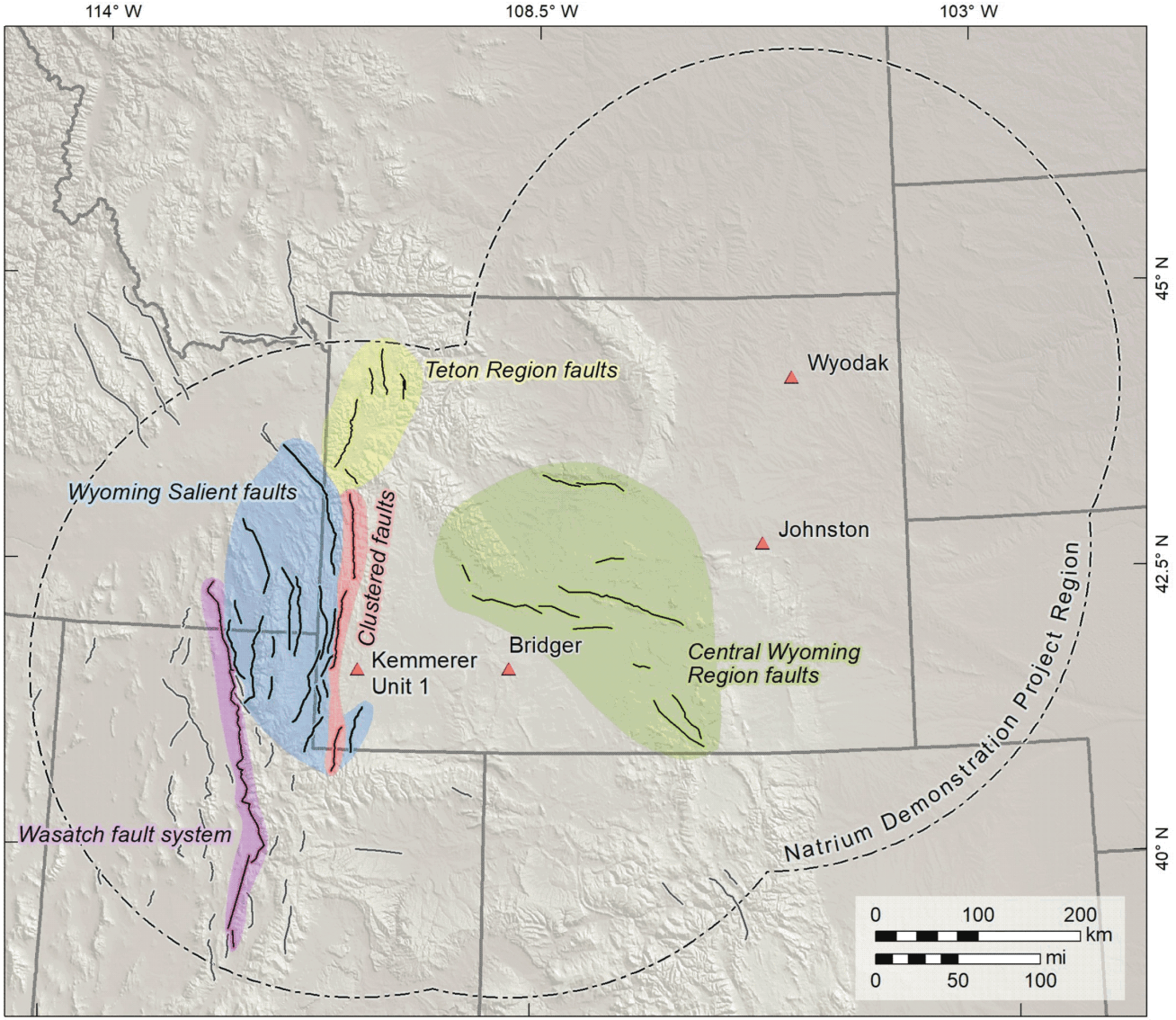



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**Figure 2.6-34 Master Logic Tree for Fault Sources**

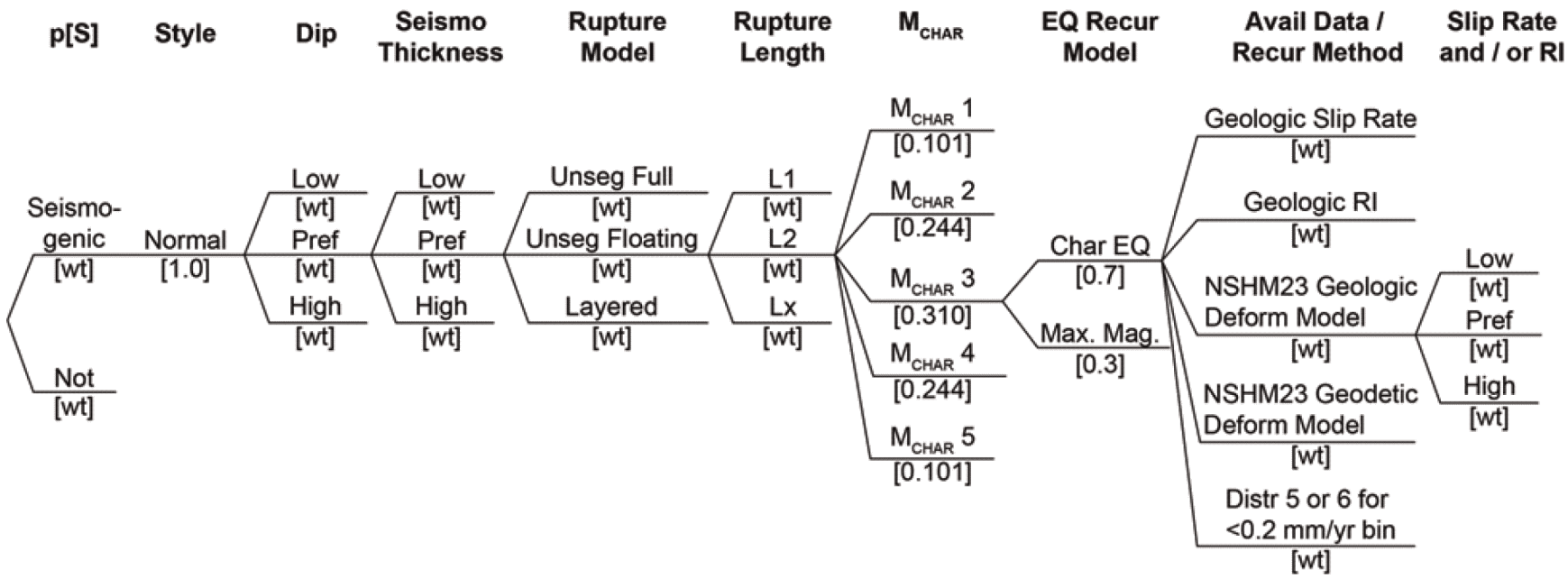



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**Figure 2.6-35 Clustered Fault Sources (Red); Other Fault Sources (Grey)**

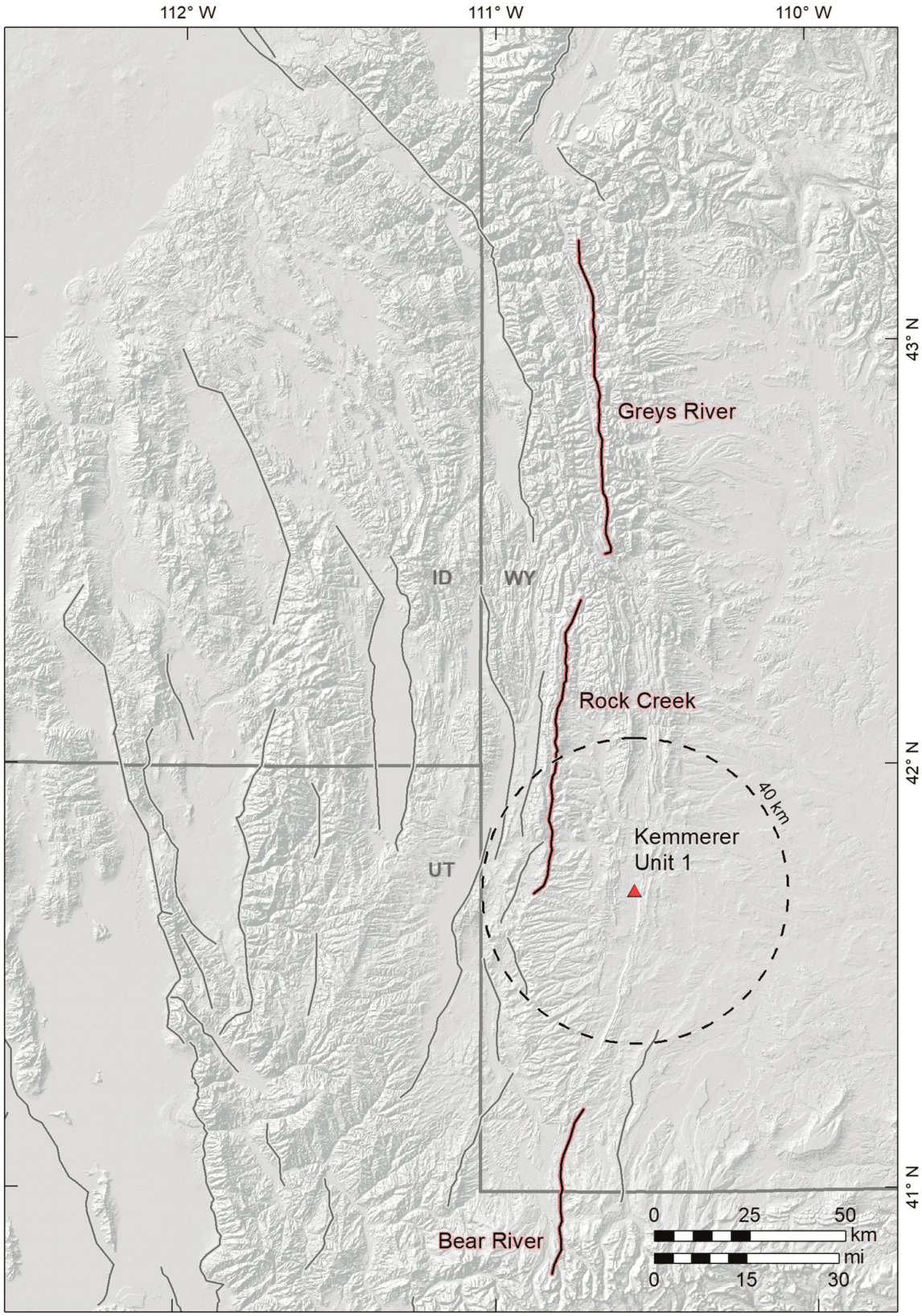



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**Figure 2.6-36 Temporal Distribution of Surface-rupturing Earthquakes on Faults in the Wyoming Salient region, showing apparent cluster of Earthquakes in mid- to late Holocene**

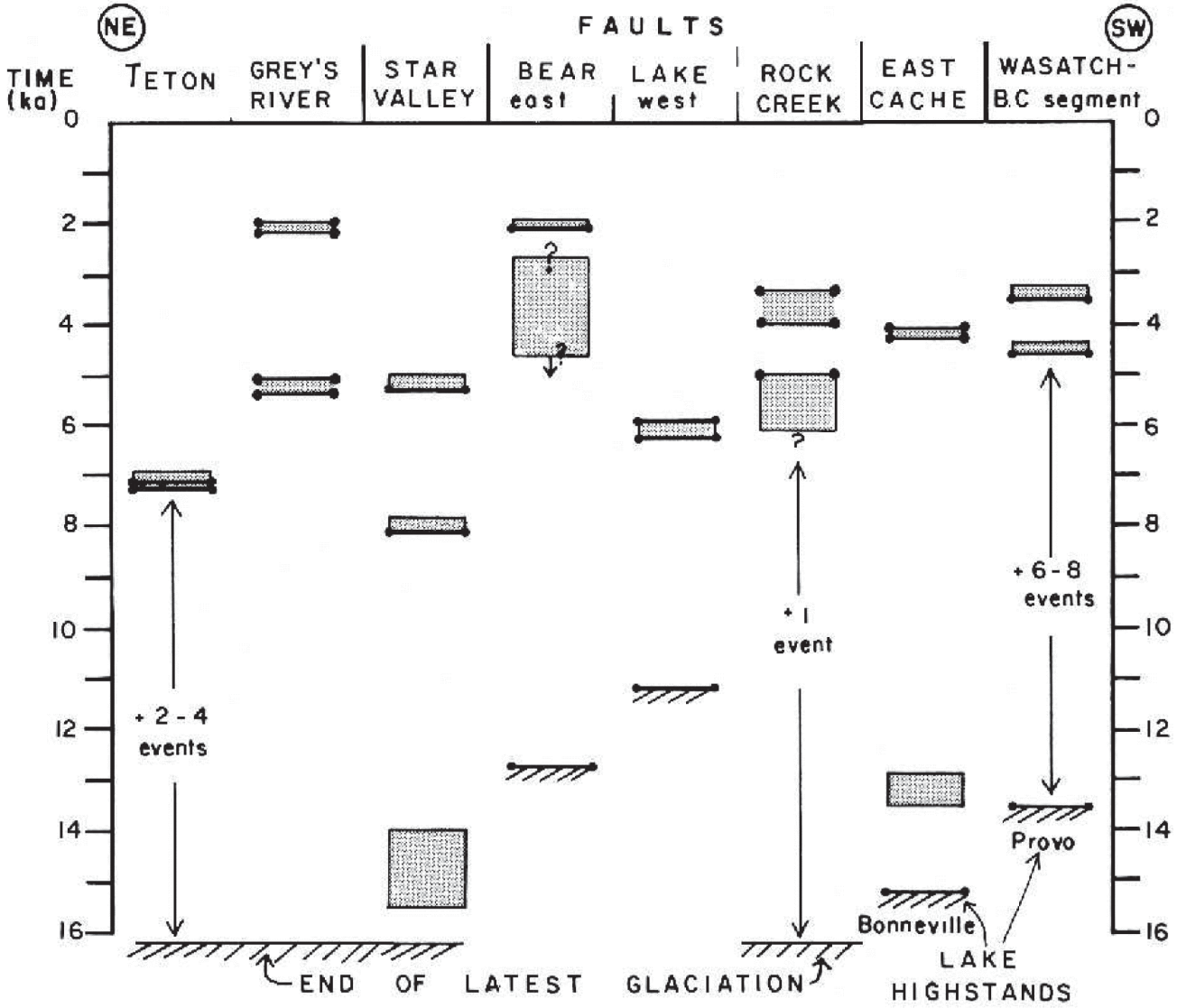



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**Figure 2.6-37 Temporal Distribution of Surface-rupturing Earthquakes on Faults in the Wyoming Salient region, showing apparent cluster of Earthquakes in mid- to late Holocene**

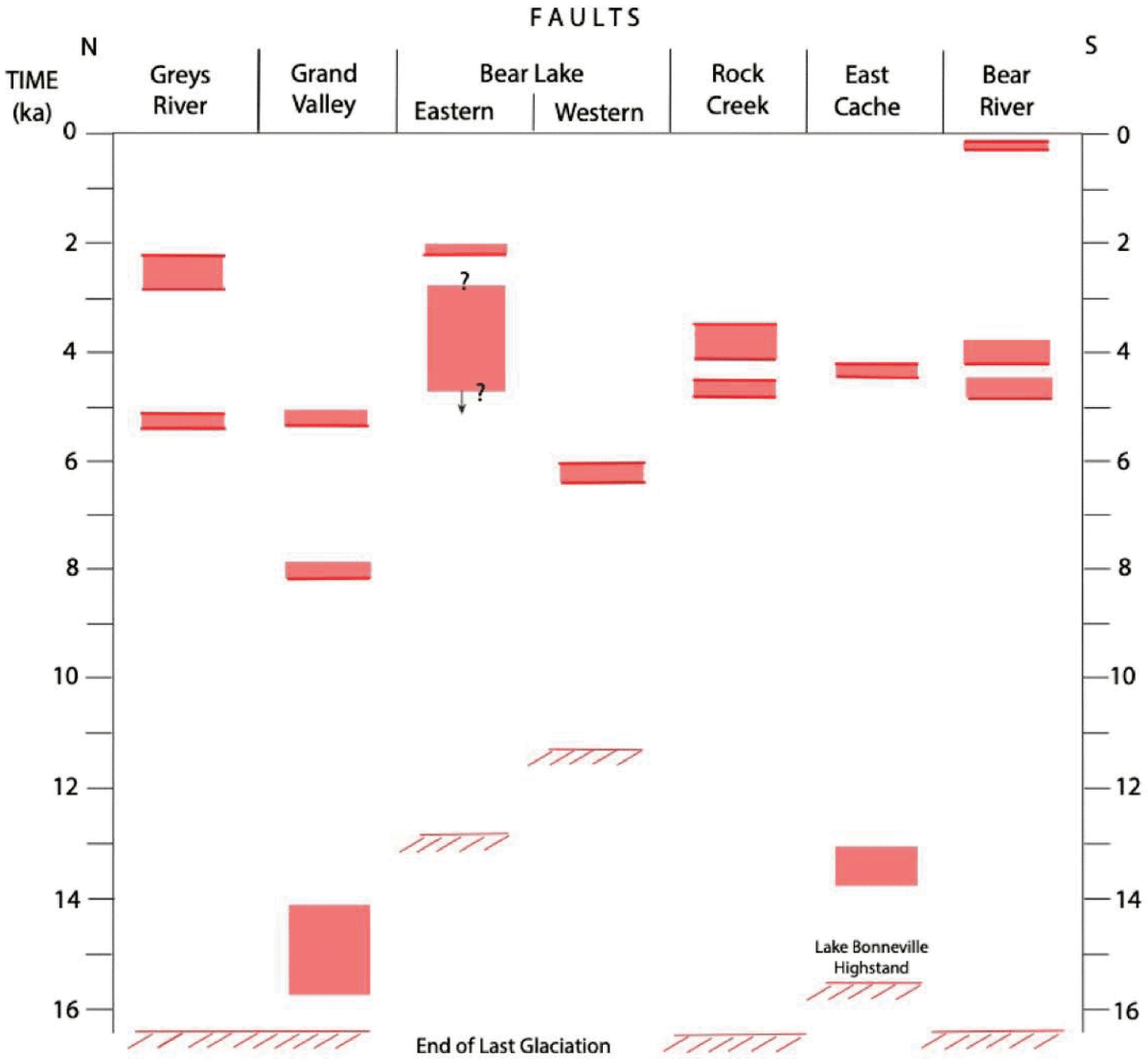






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**Figure 2.6-38 Rock Creek Fault Mapping and Fault Source**

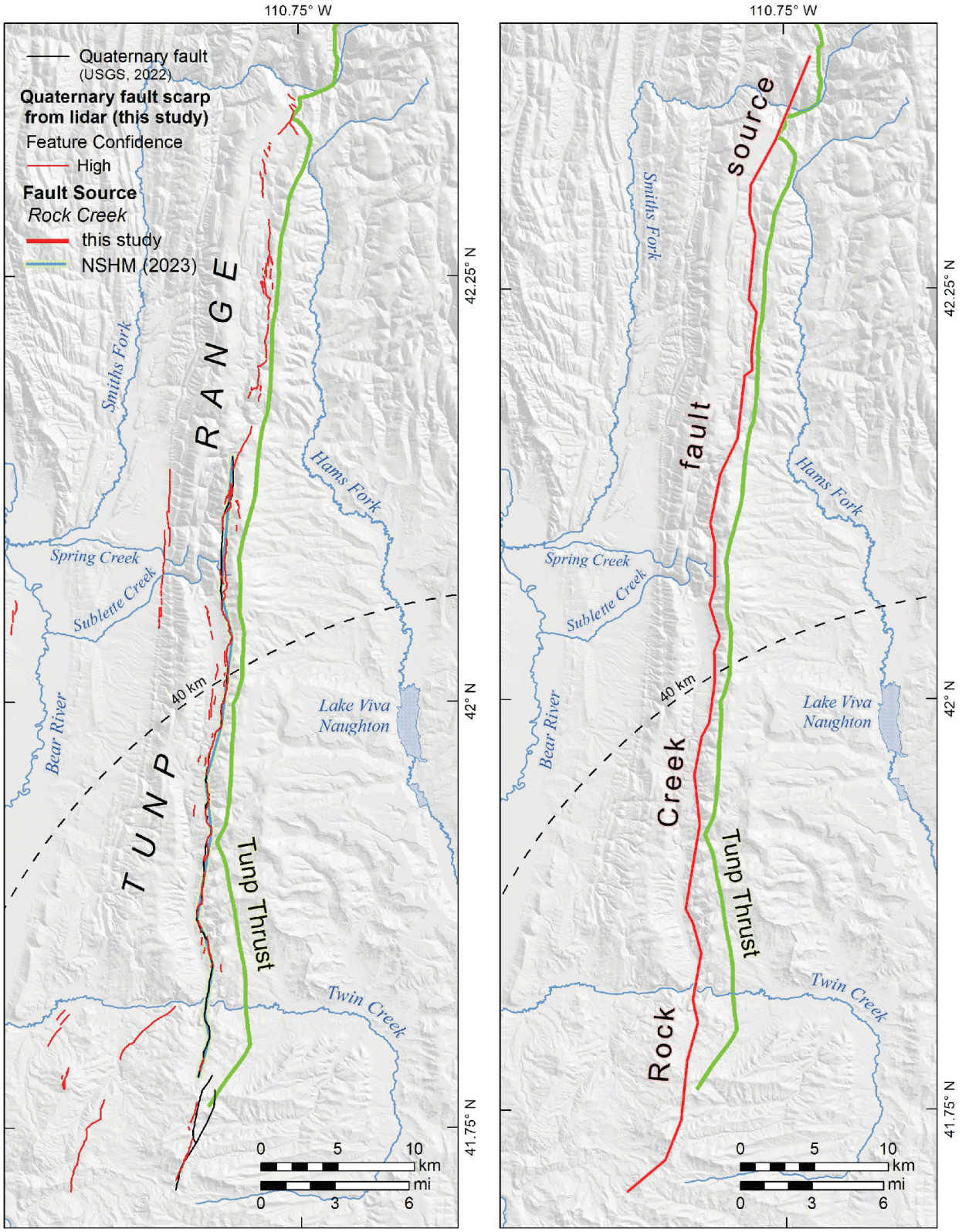



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**Figure 2.6-39 Master Logic Tree for the Rock Creek Fault Source**

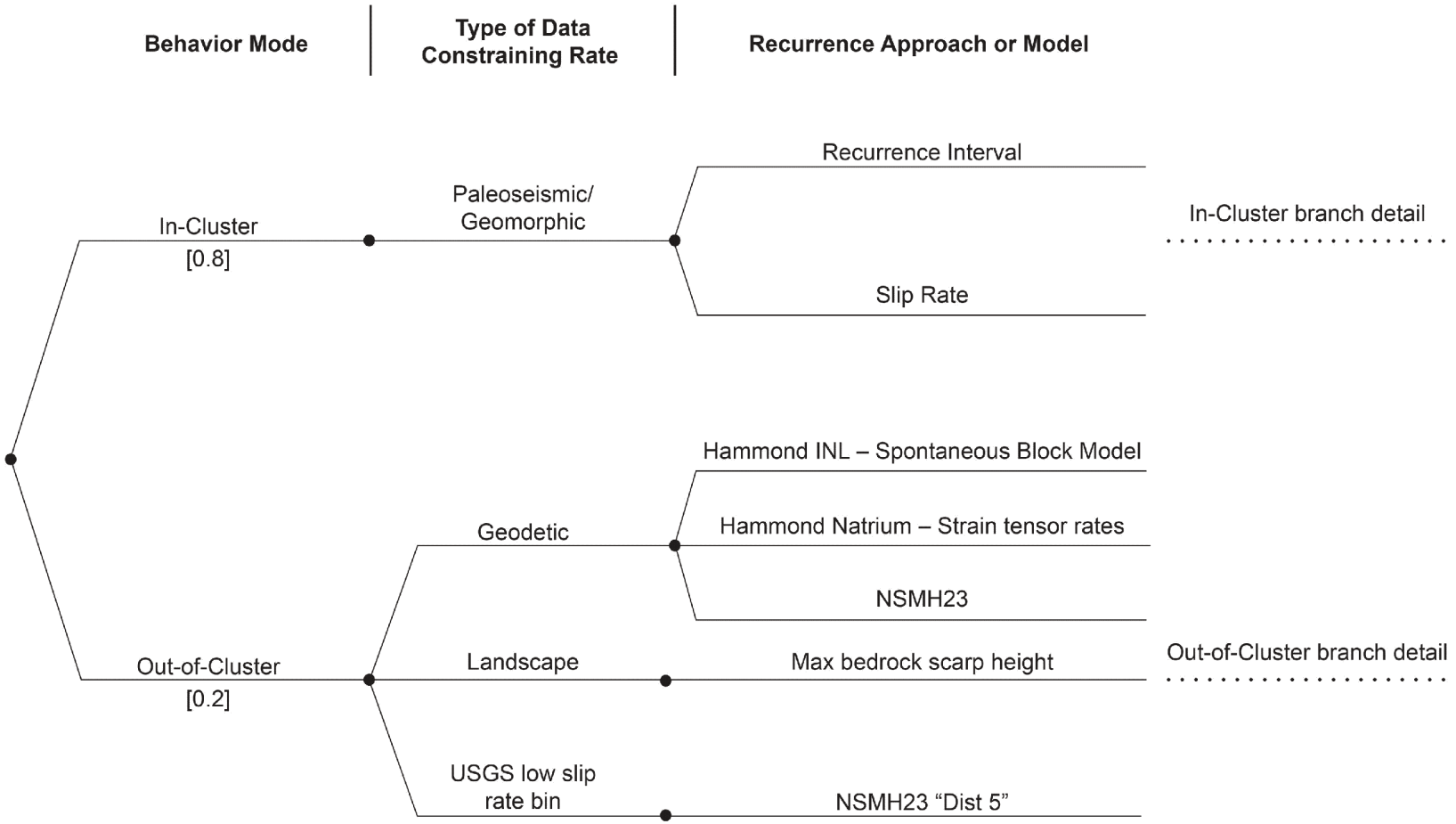



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**Figure 2.6-40a In-Cluster Logic Tree for the Rock Creek Fault Source**

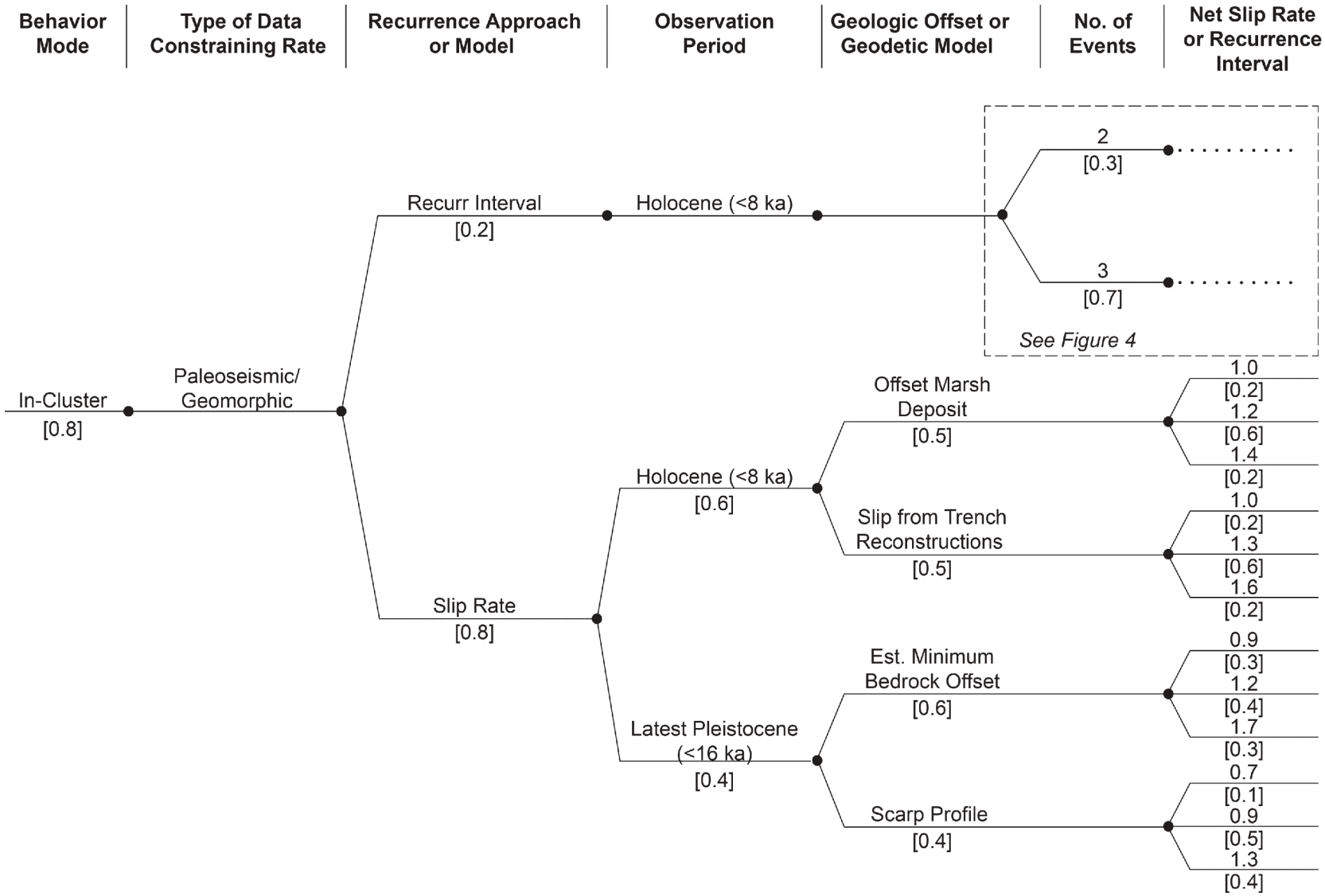




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**Figure 2.6-40b In-Cluster Logic Tree for the Rock Creek Fault Source, Detail**

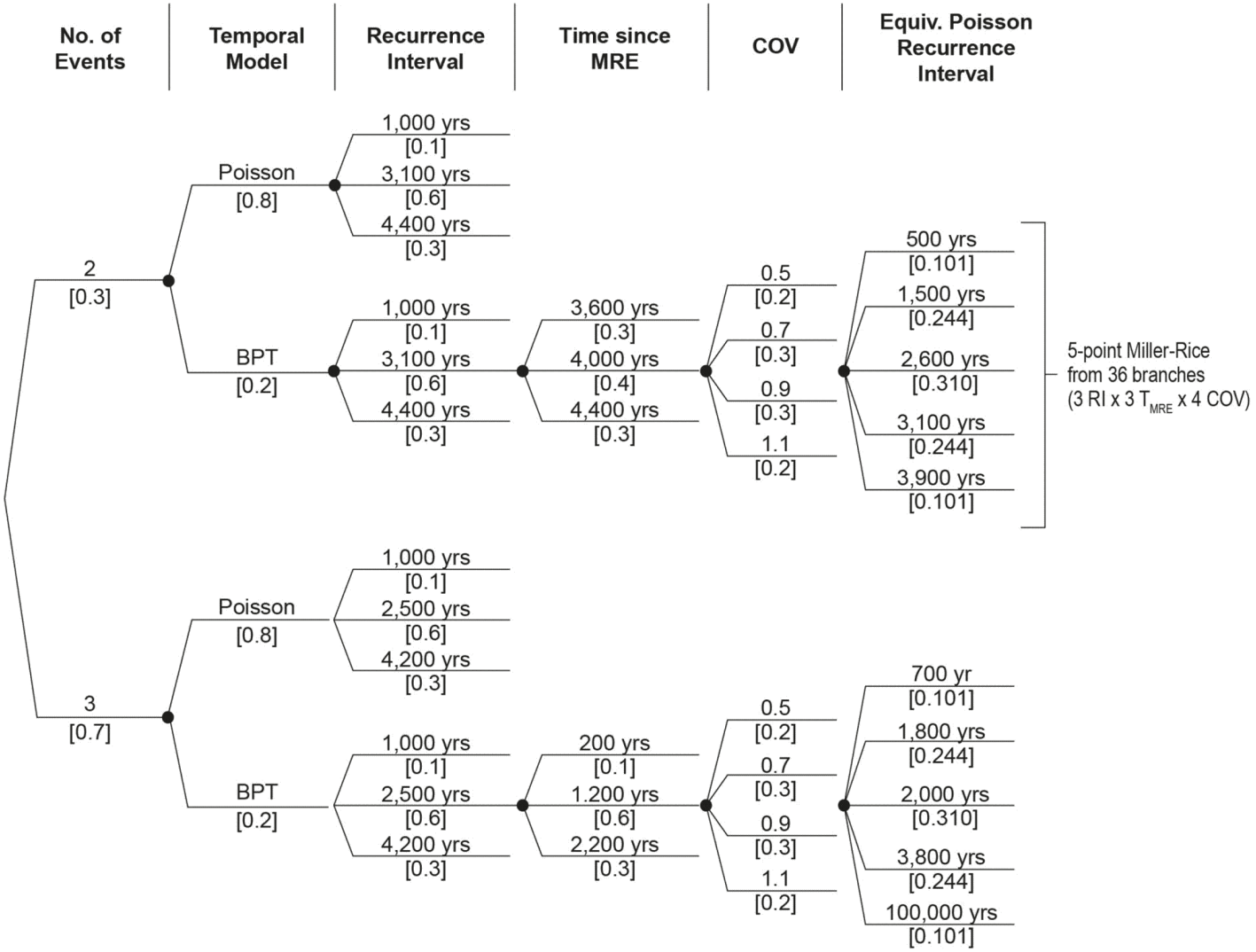






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**Figure 2.6-41 Out-of-Cluster Logic Tree for the Rock Creek Fault Source**

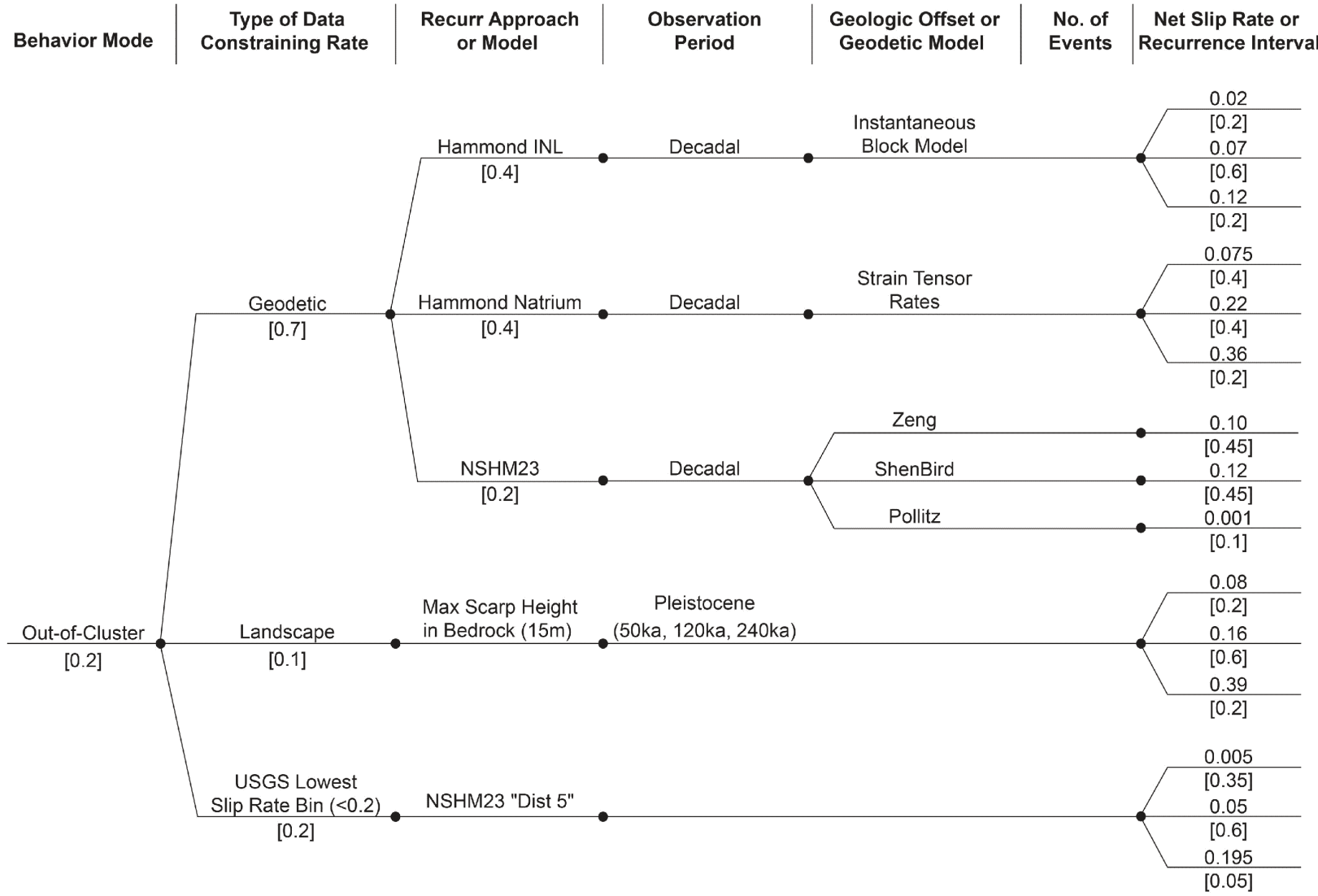




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**Figure 2.6-42 Along-strike Distribution of Vertical Separations for the Rock Creek Fault from Interpretation of LiDAR Scarp Profiles**

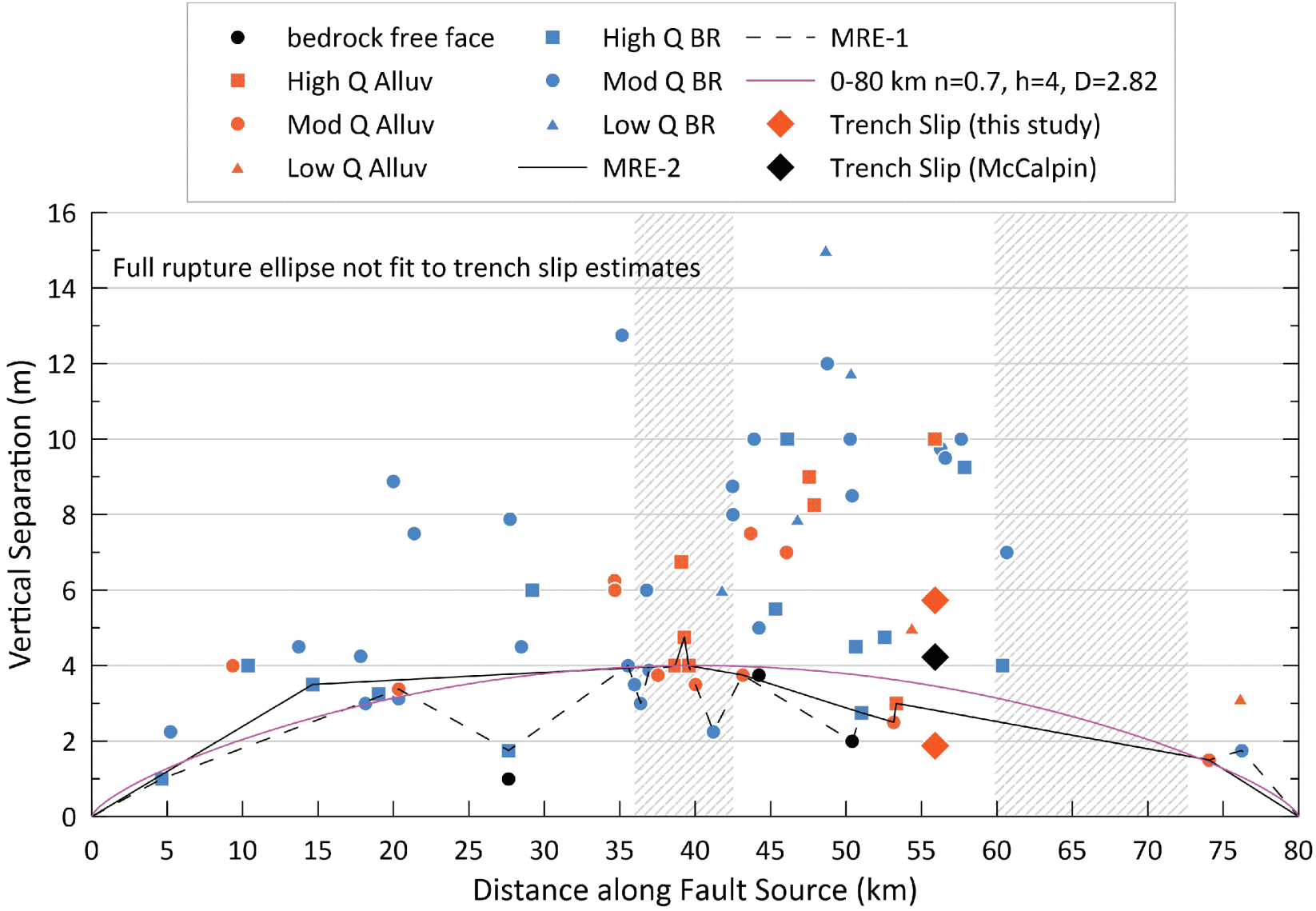



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**Figure 2.6-43 LiDAR-Based Estimates of Vertical Surface Displacement (in m) Across the Southern Rock Creek Fault (Red) from LiDAR-Based Topographic Profiles 57-61**

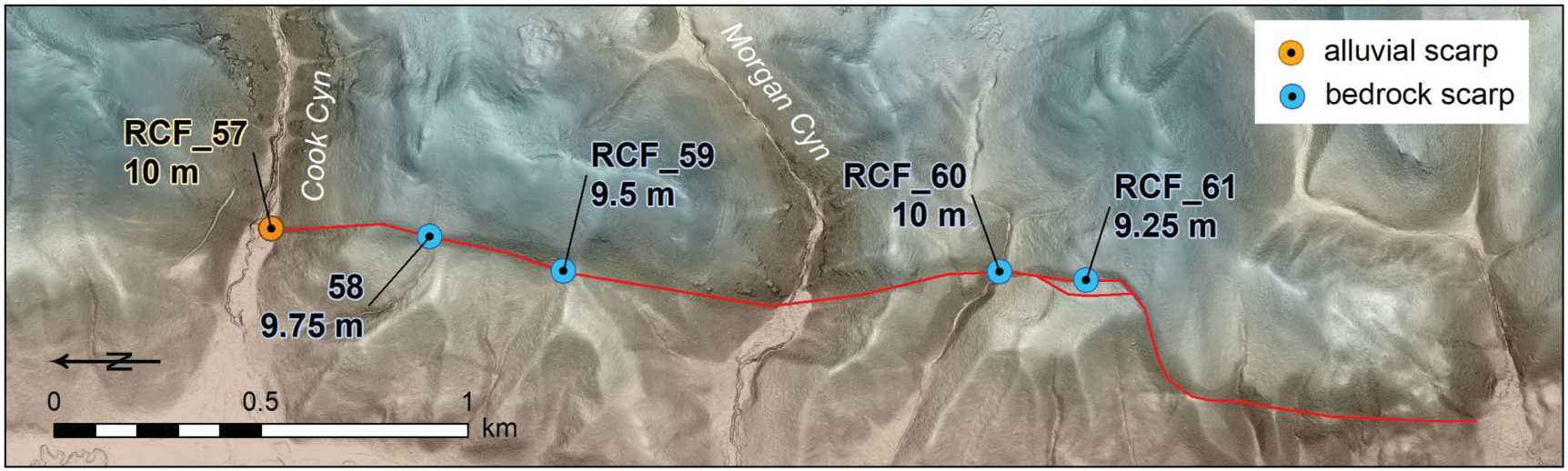



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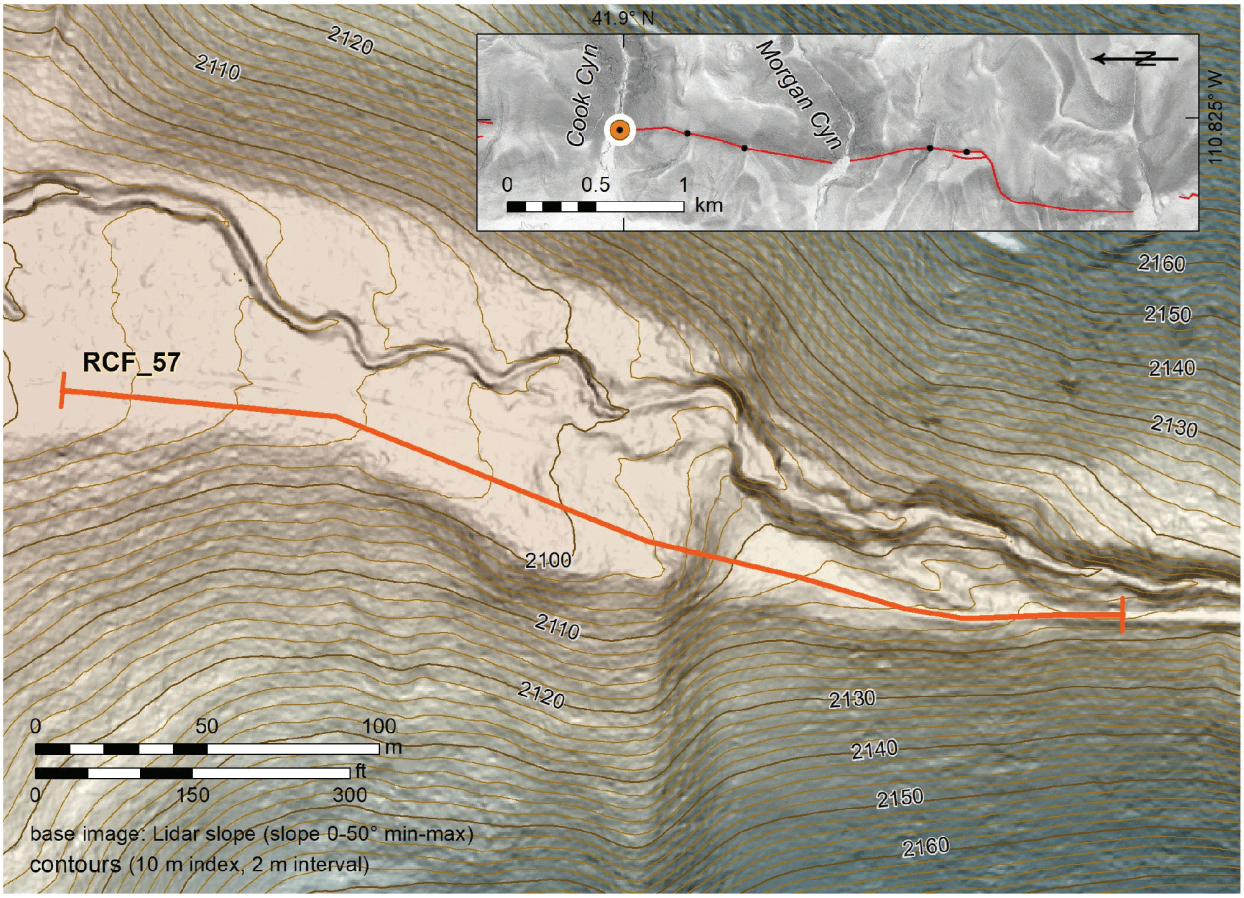
**Figure 2.6-44 LiDAR-Based Mapping of the Rock Creek Fault at Cook Canyon**



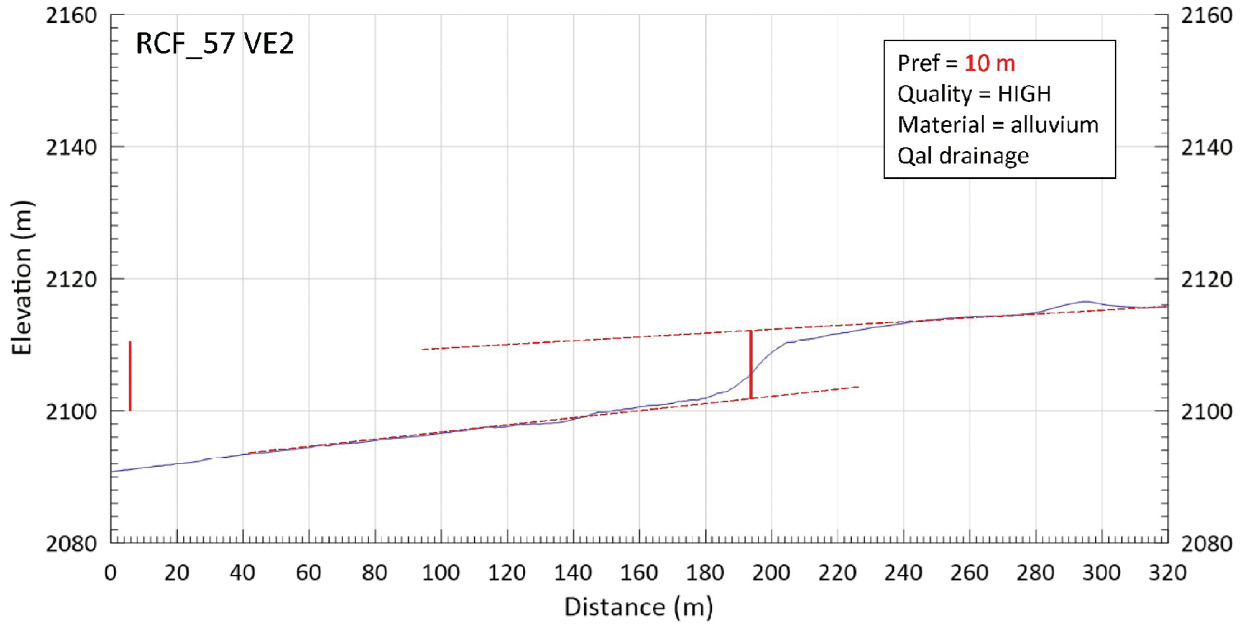





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**Figure 2.6-45 Rock Creek Fault Source Slip Rates Bar Chart**

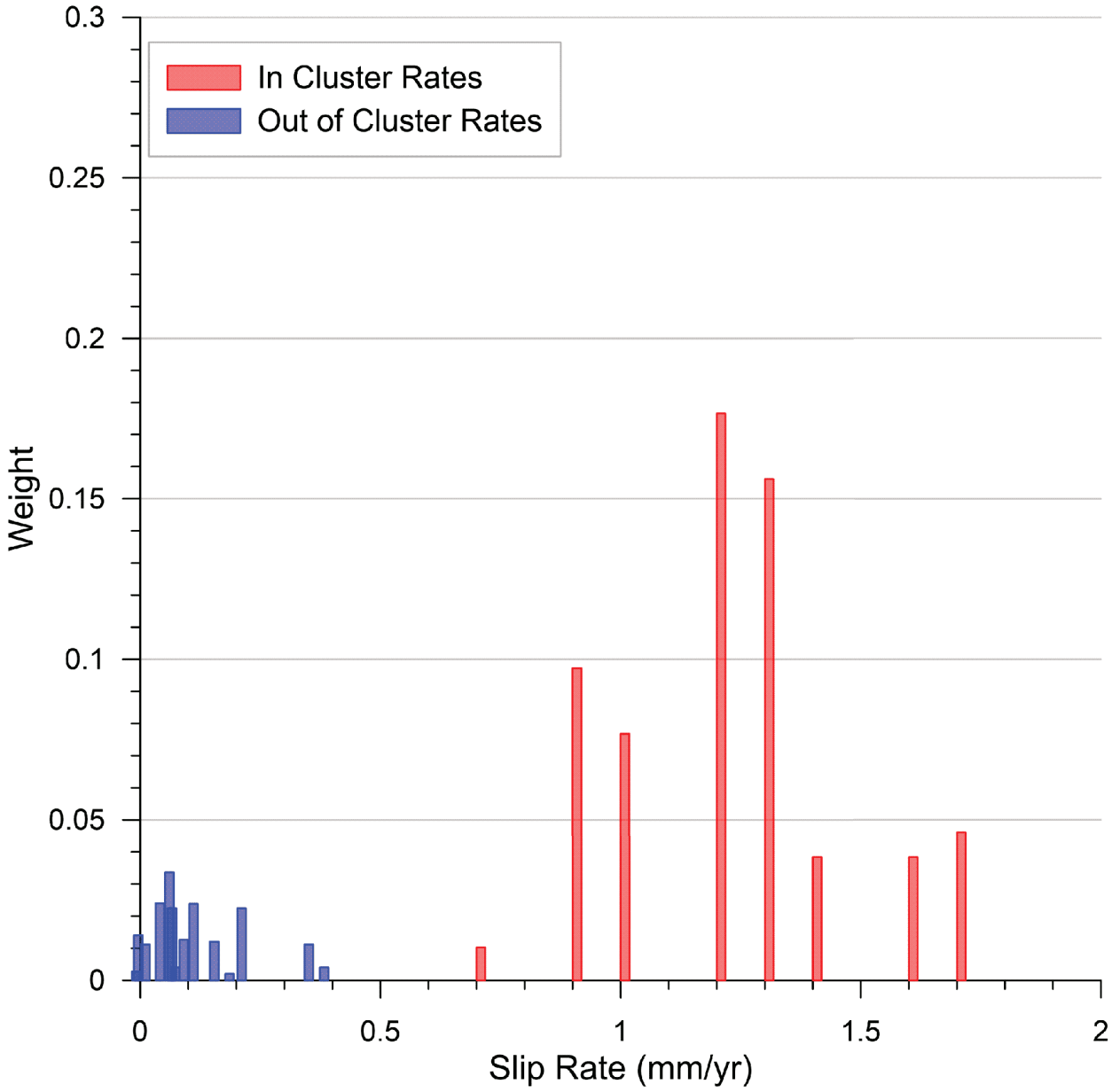



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**Figure 2.6-46 Logic Tree for the Bear River Fault Source**

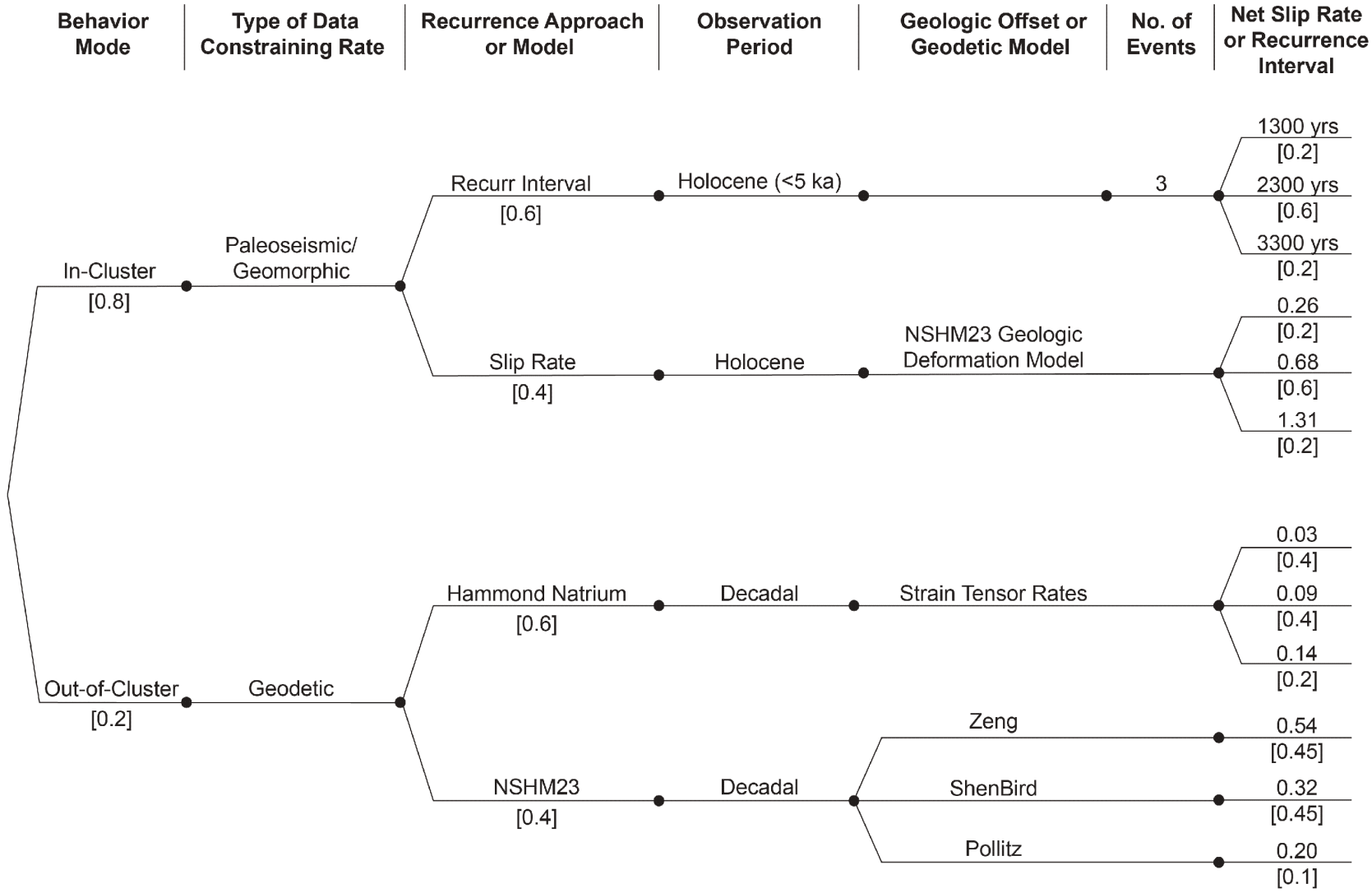




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**Figure 2.6-47 Greys River Fault Mapping and Fault Source**

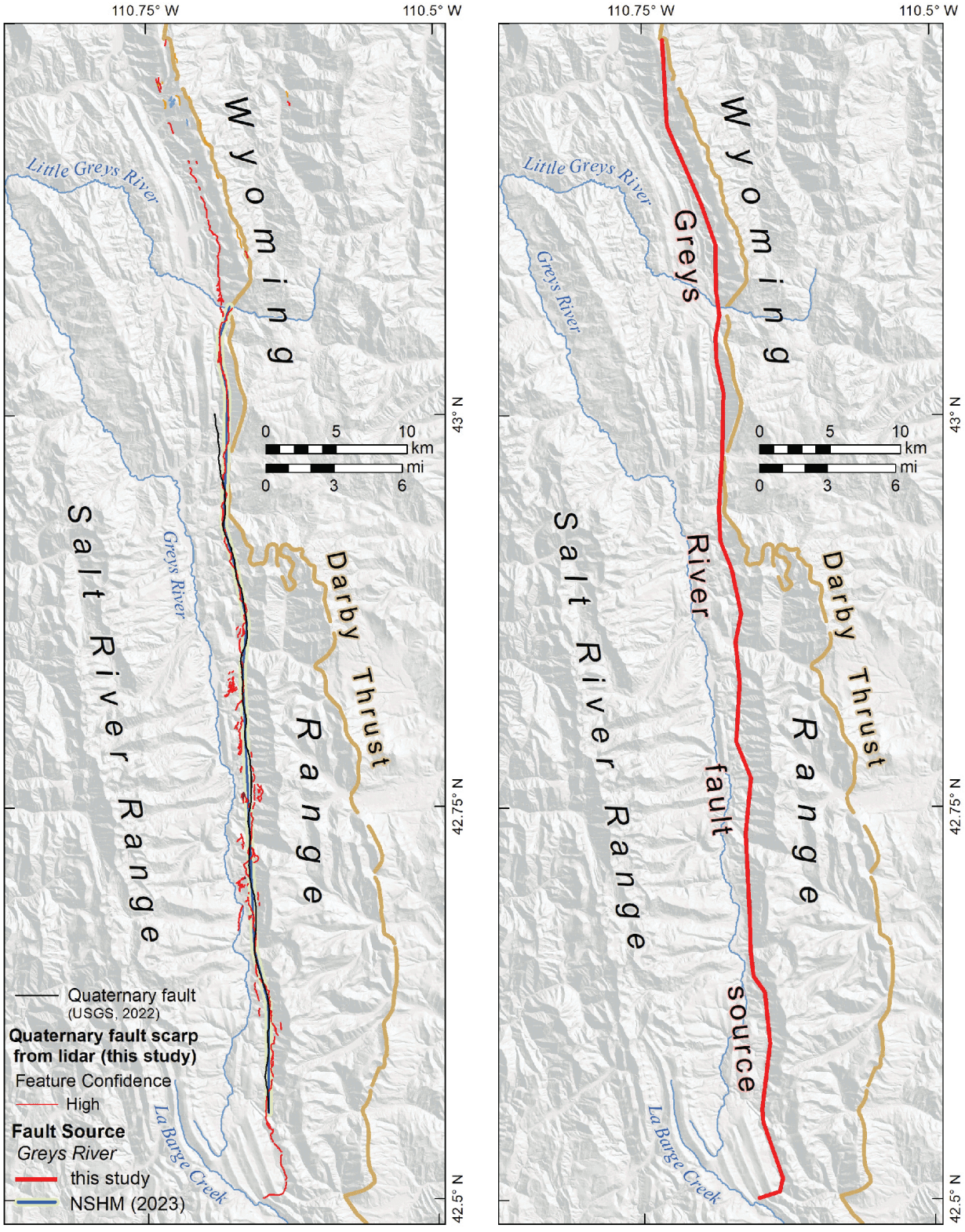



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**Figure 2.6-48 Logic Tree for the Greys River Fault Source**

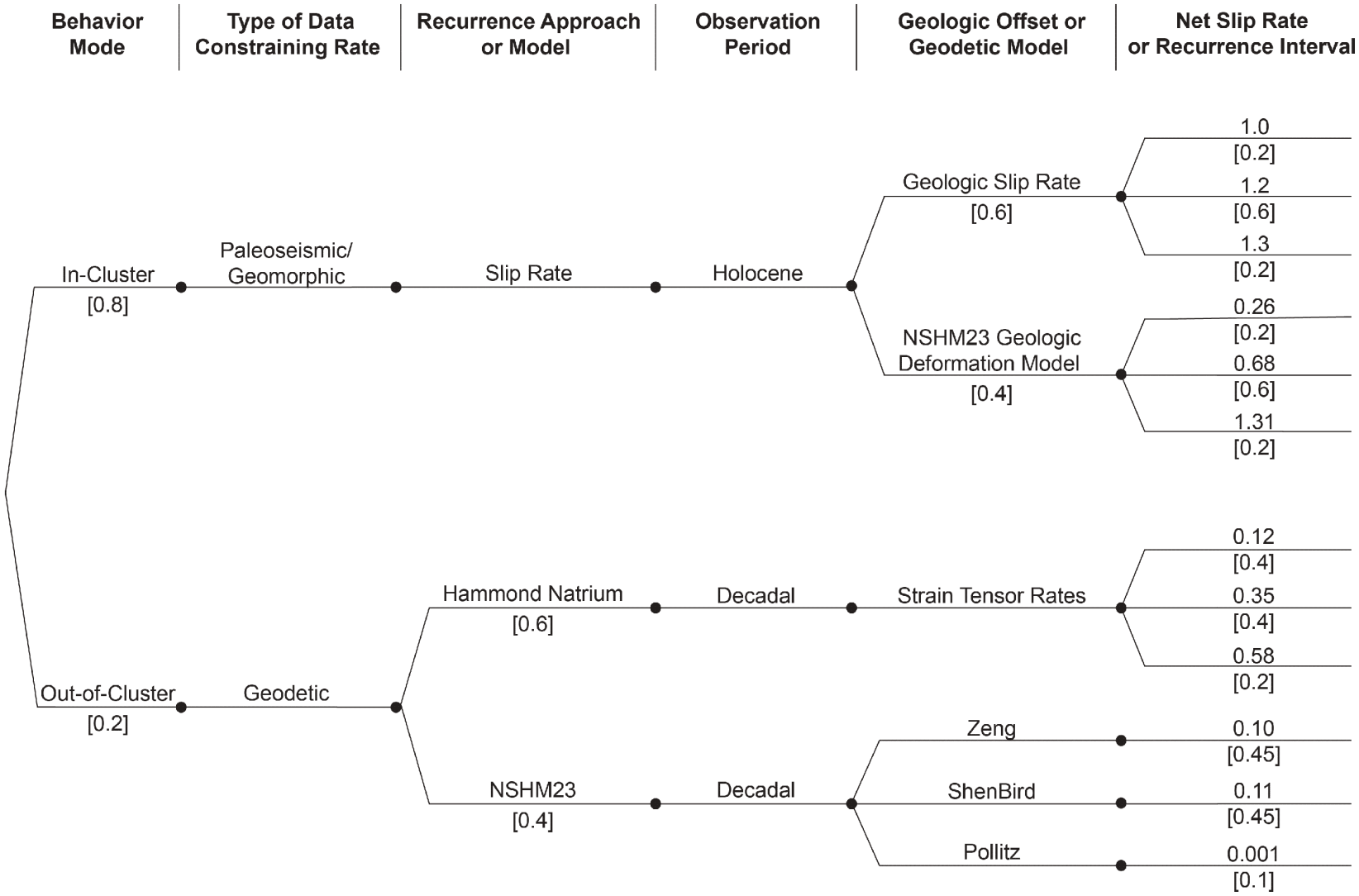




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**Figure 2.6-49 Fault sources in the Wyoming Salient region (blue); other fault sources (grey)**

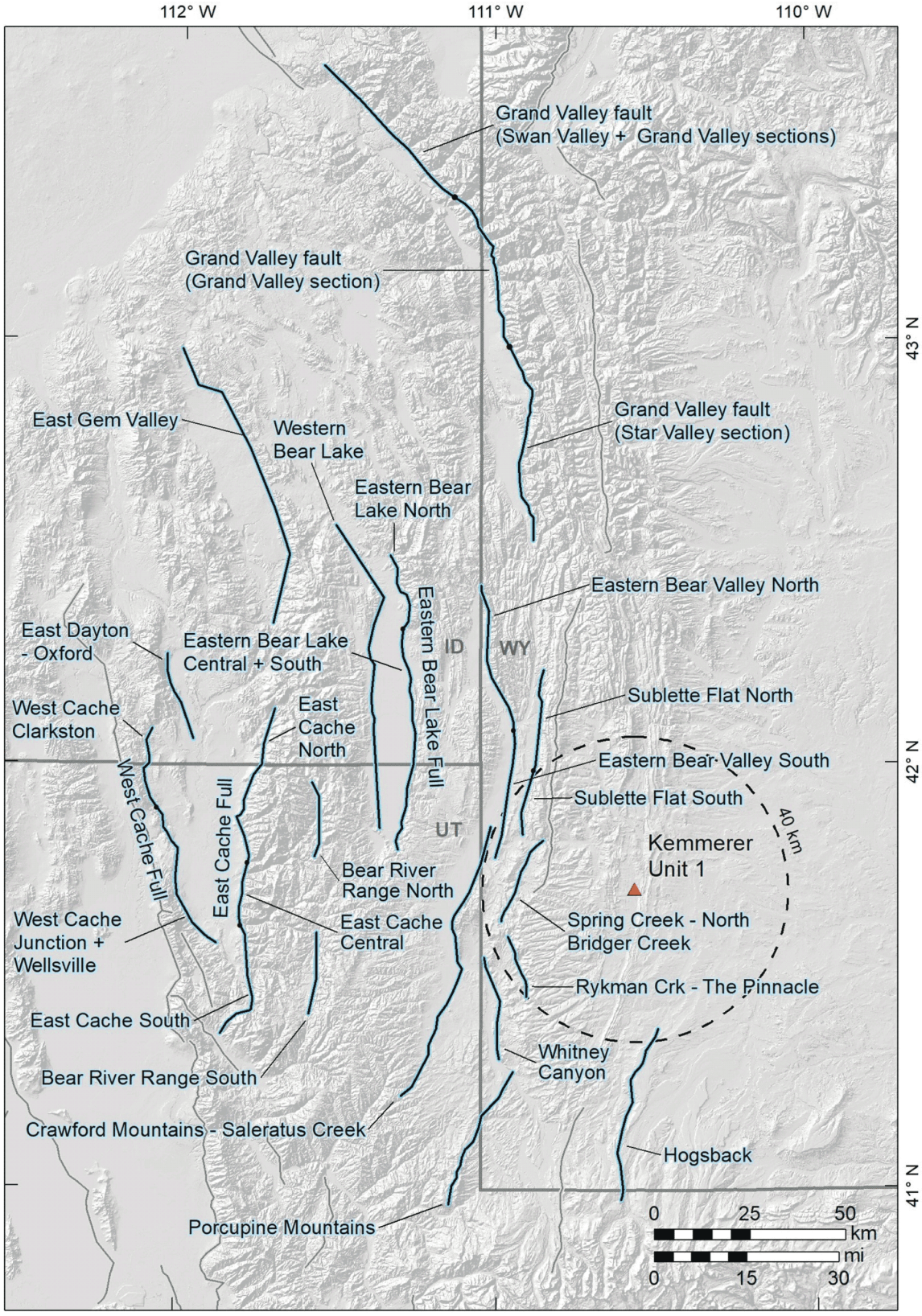



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**Figure 2.6-50 Fault sources in the Teton region (yellow); other fault sources (grey)**

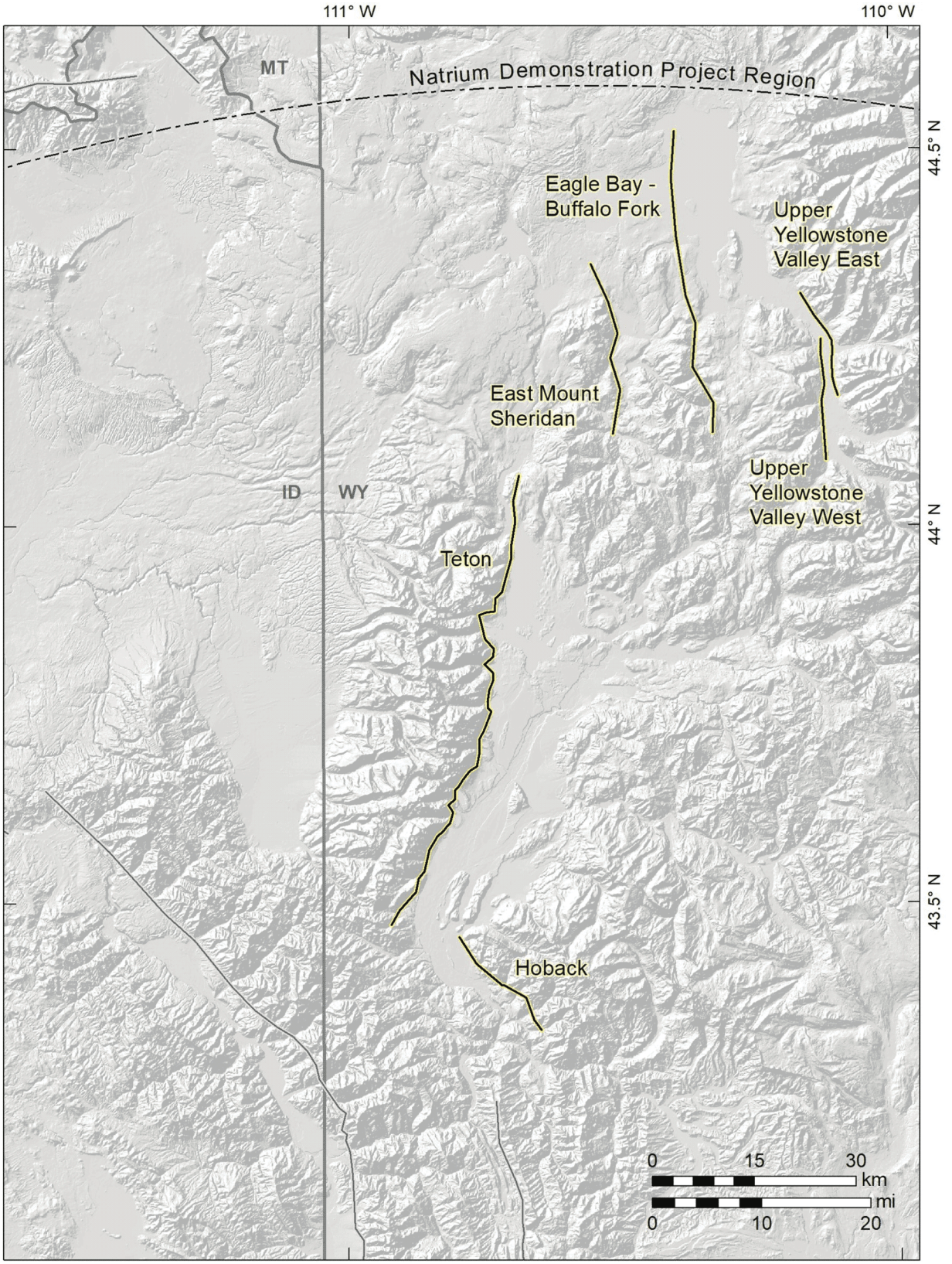



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**Figure 2.6-51 Fault sources in the central Wyoming region (green); other fault sources (grey)**

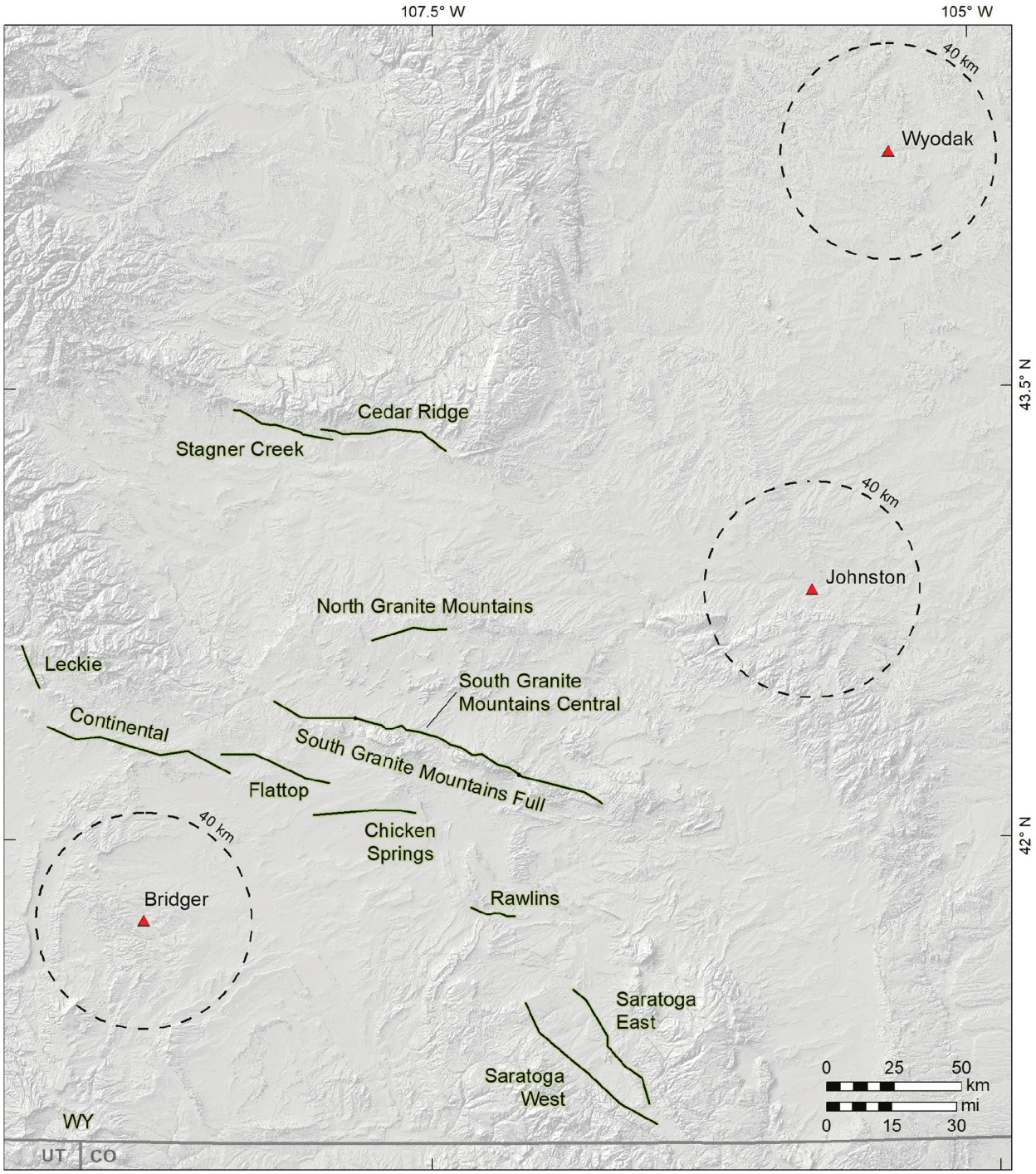



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**Figure 2.6-52 Wasatch fault system logic tree for rupture models and rupture sources**

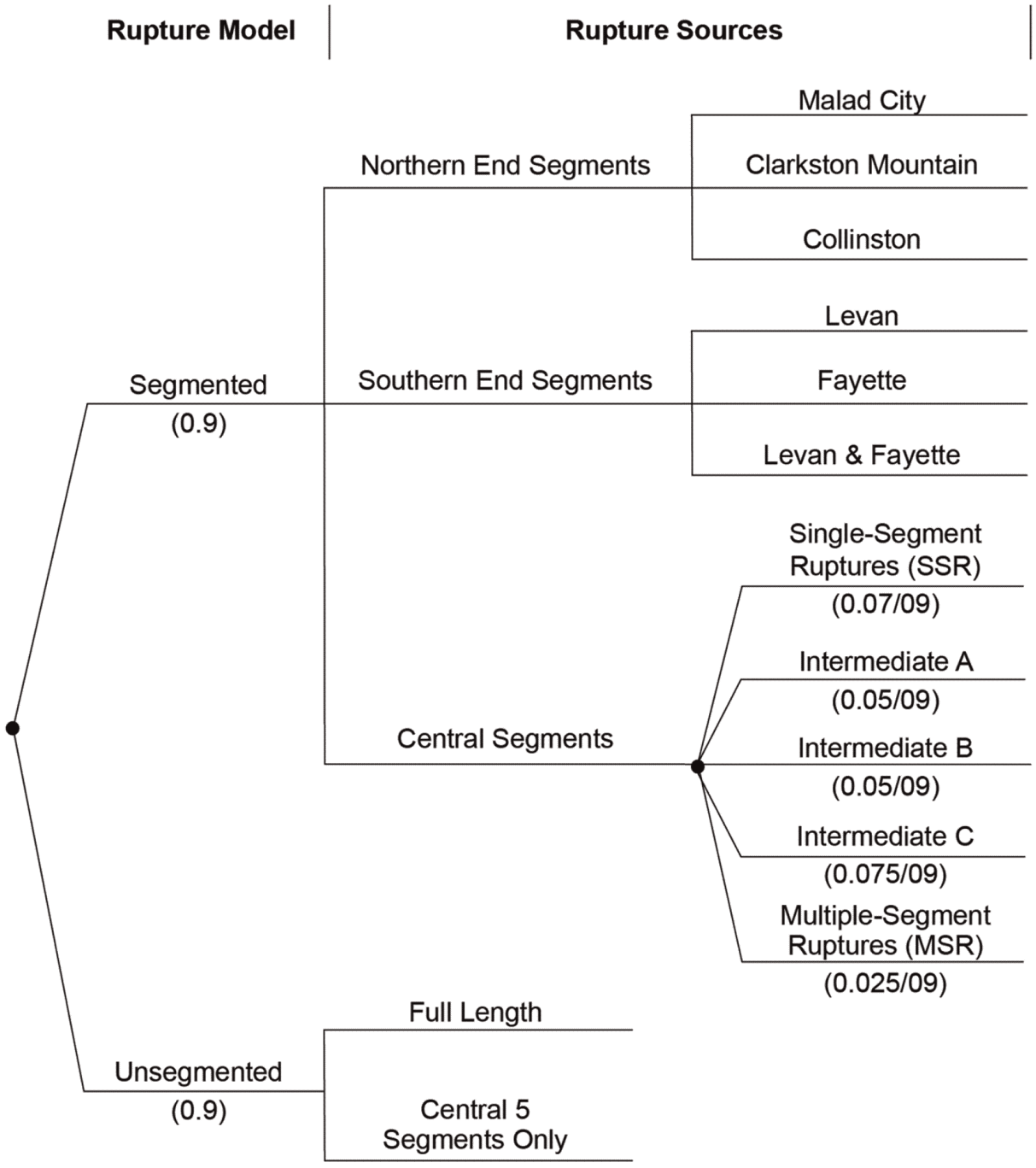



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**Figure 2.6-53 Wasatch fault system logic tree for rupture models and rupture sources for central segments**

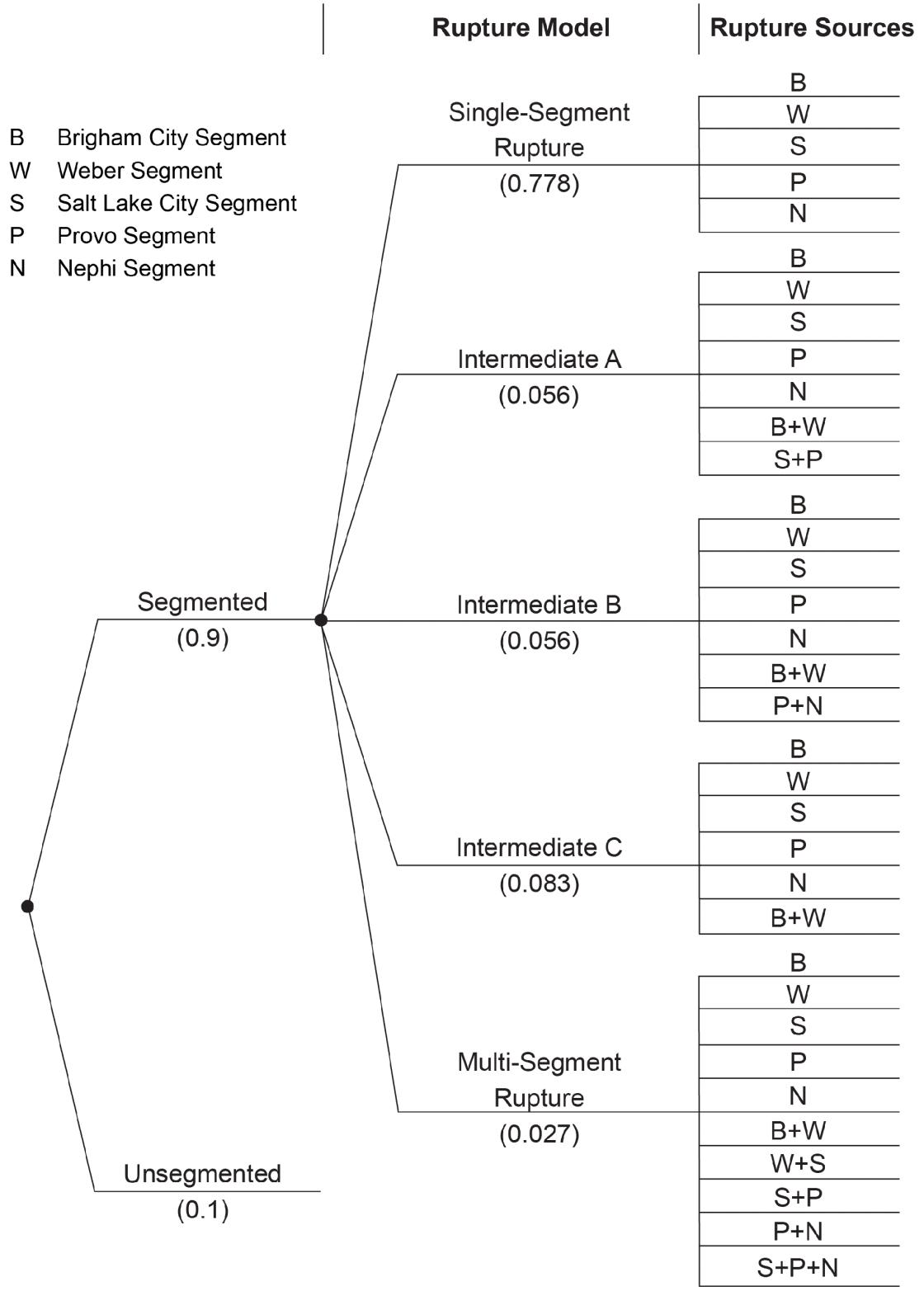



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**Figure 2.6-54 Wasatch fault system logic tree for recurrence, slip rates, and time dependence**

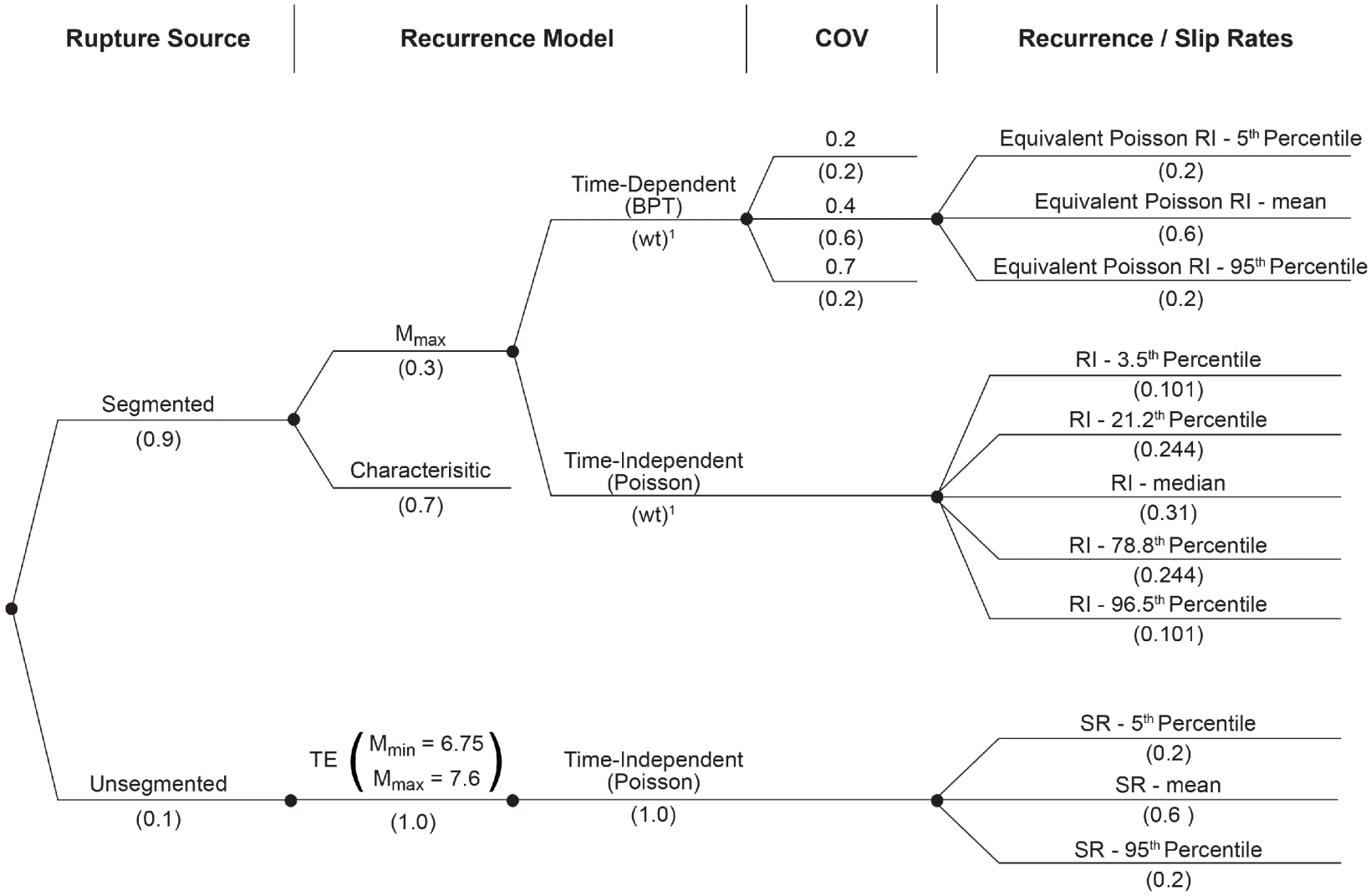



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**Figure 2.6-55 Wasatch logic tree for $M_{CHAR}$ for segmented rupture models**

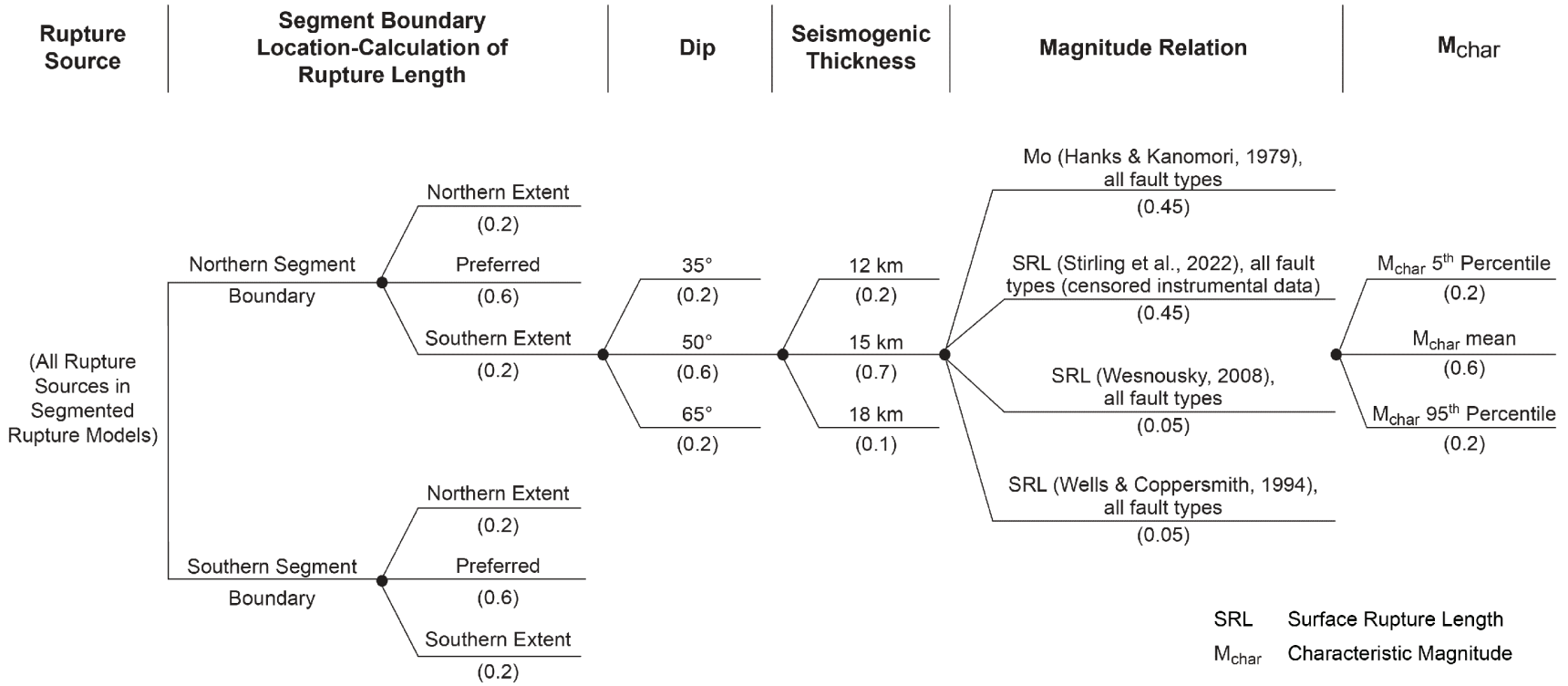



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**Figure 2.6-56 Map of Wasatch fault system sources**

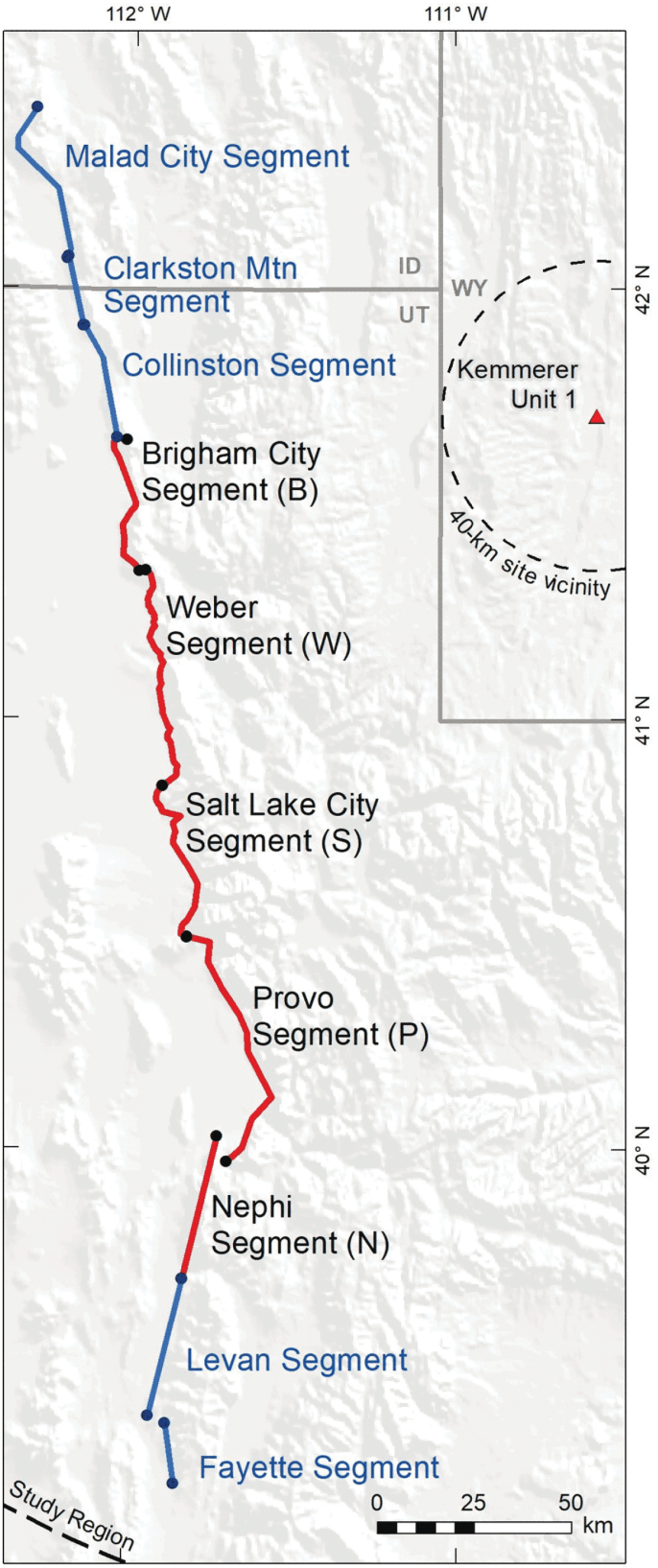



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**Figure 2.6-57 Cascadia subduction interface source from INL SSHAC 2022 (Reference 2.6-27)**

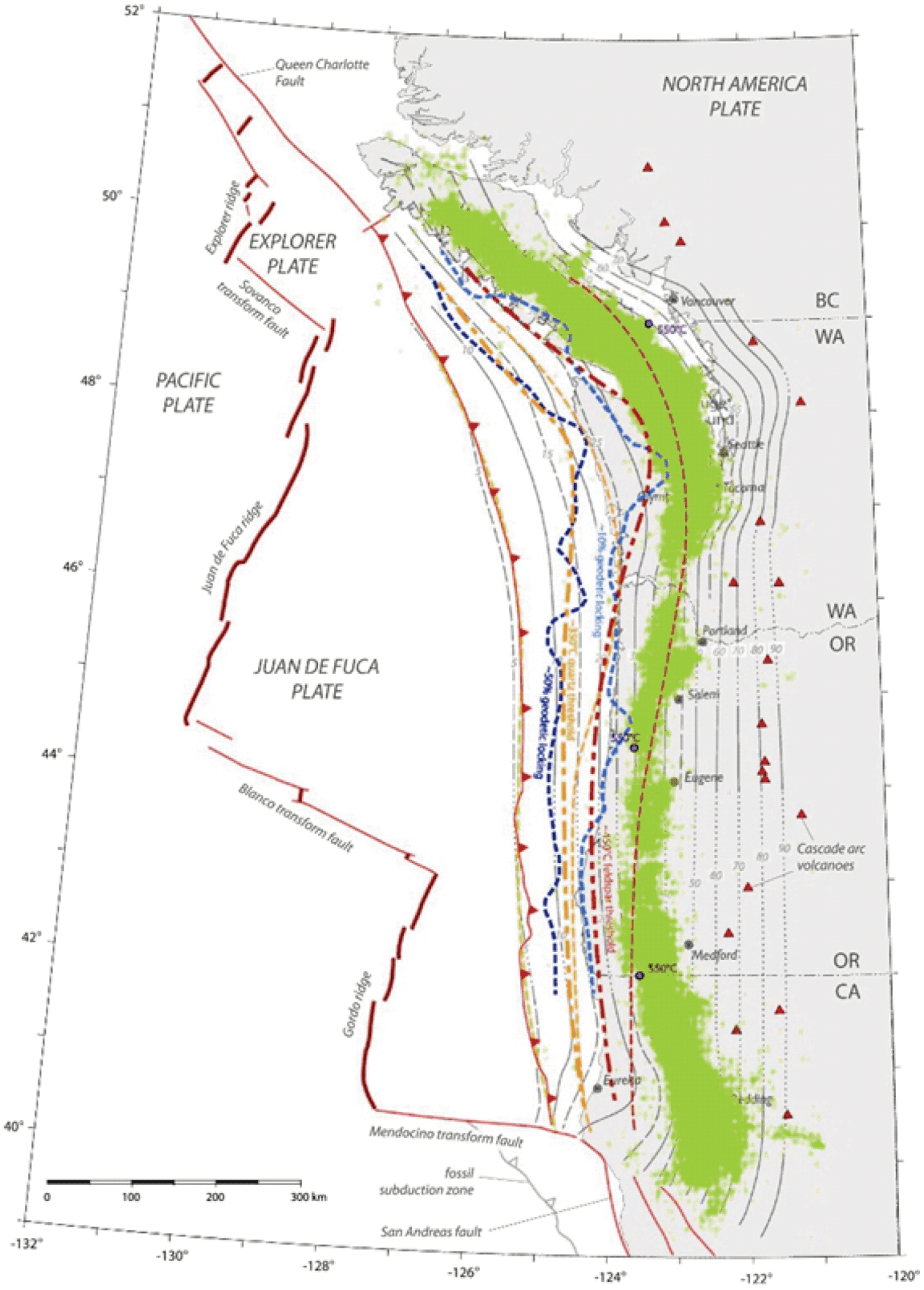



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**Figure 2.6-58 Mean reference rock hazard curves by spectral period at the site**

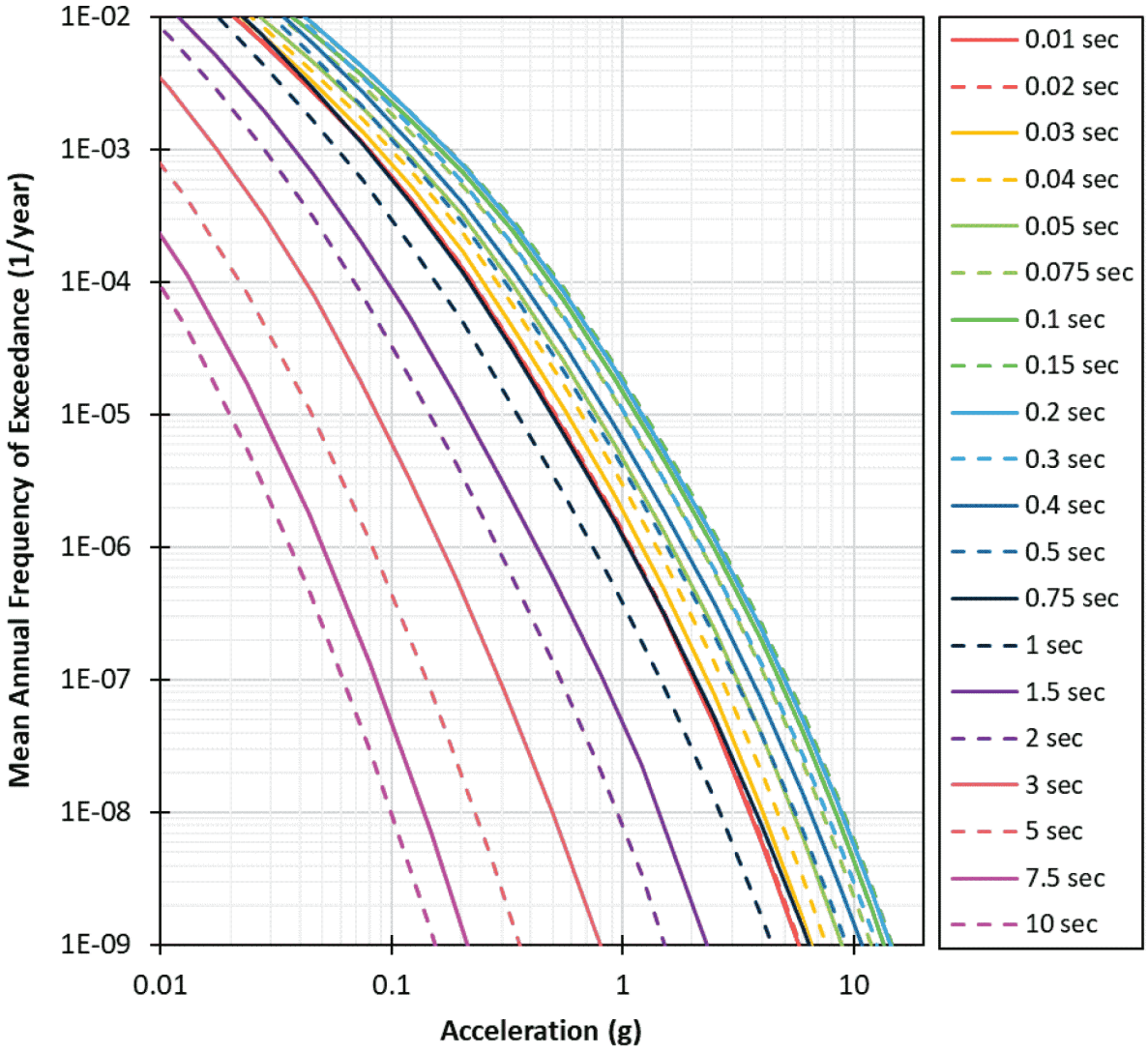




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**Figure 2.6-59 Contribution of source types to 0.01 second mean reference rock hazard at the site**

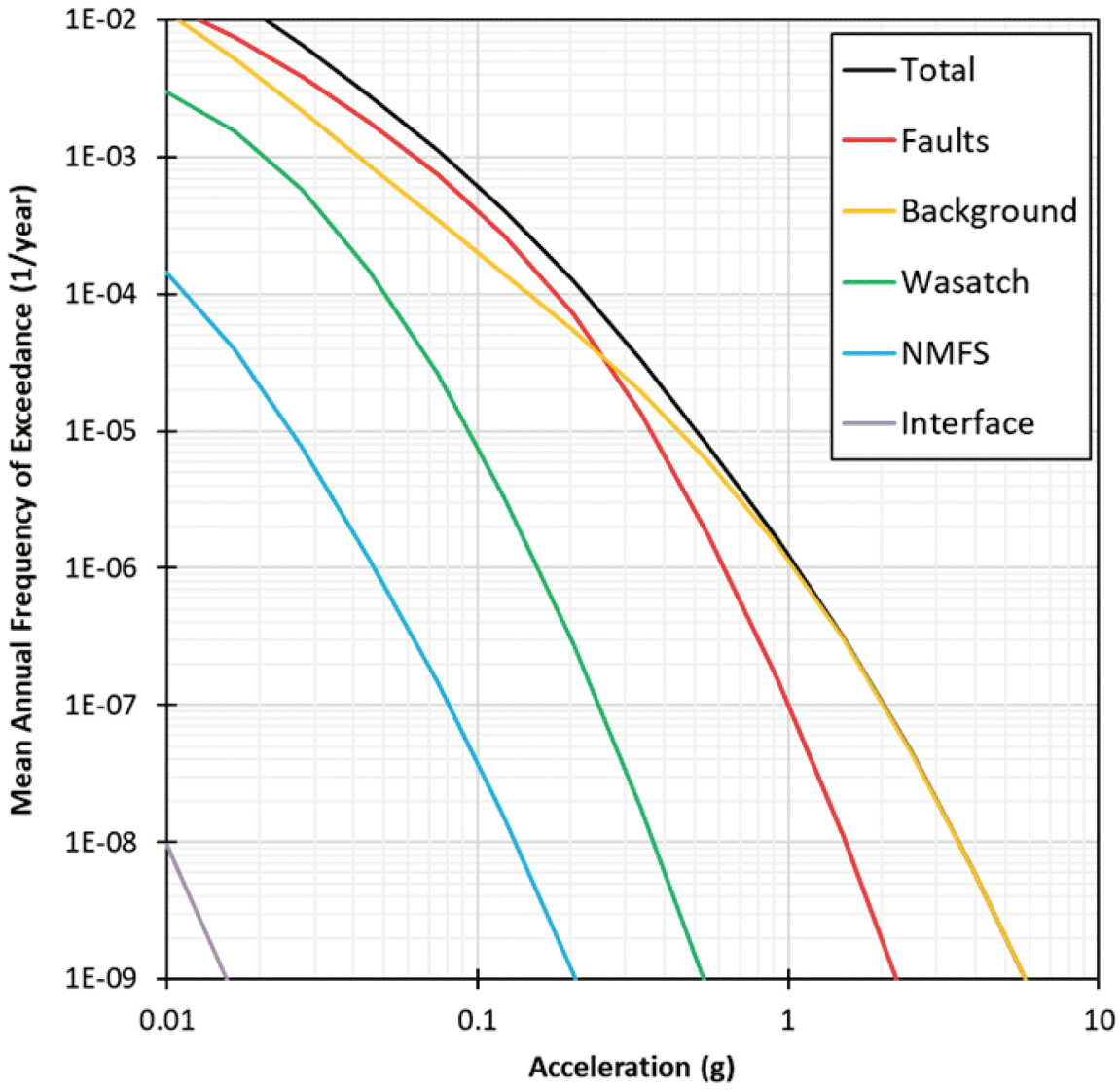



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**Figure 2.6-60 Contribution of source types to 0.1 second mean reference rock hazard at the site**

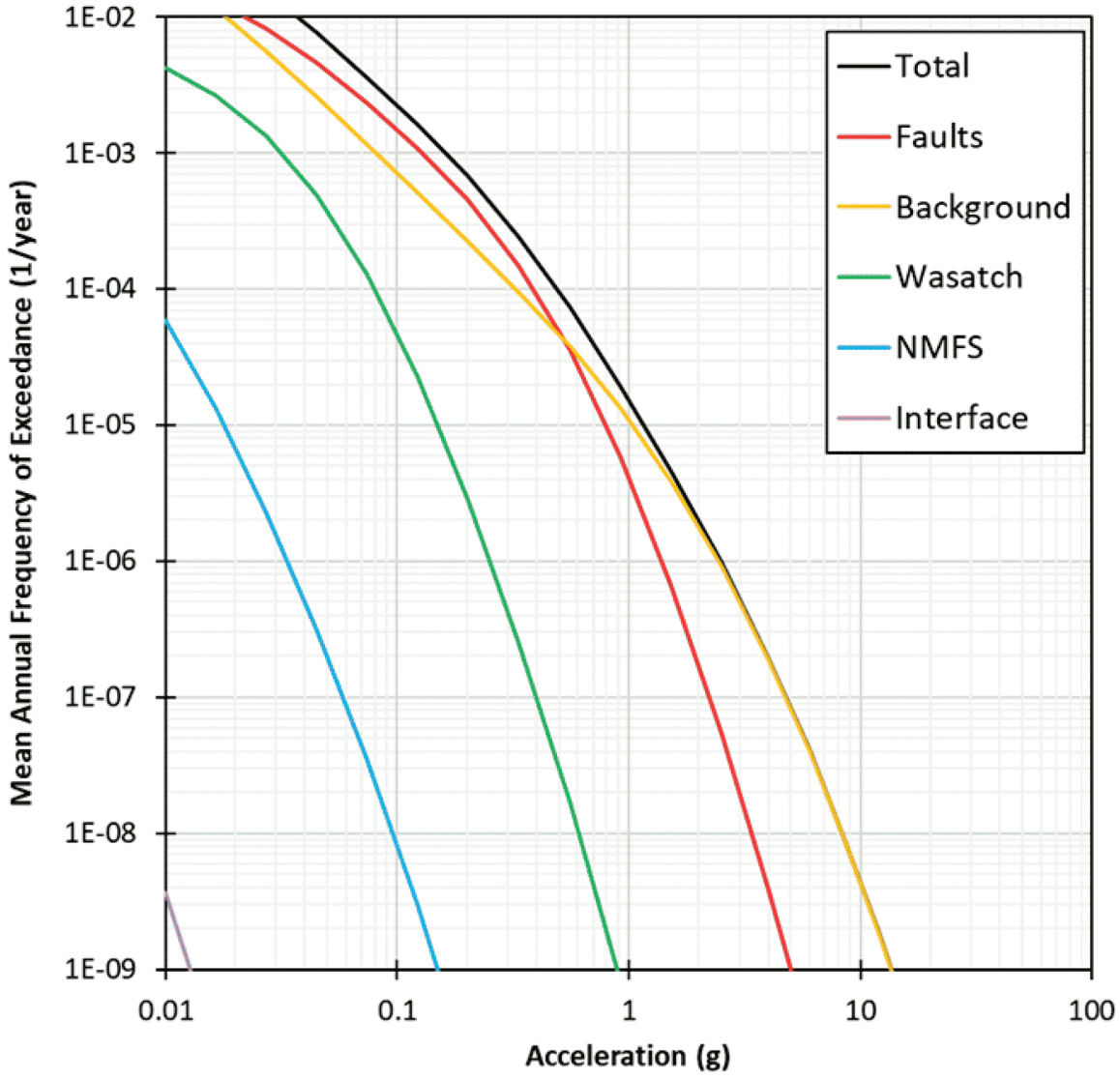




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**Figure 2.6-61 Contribution of source types to 1 second mean reference rock hazard at the site**

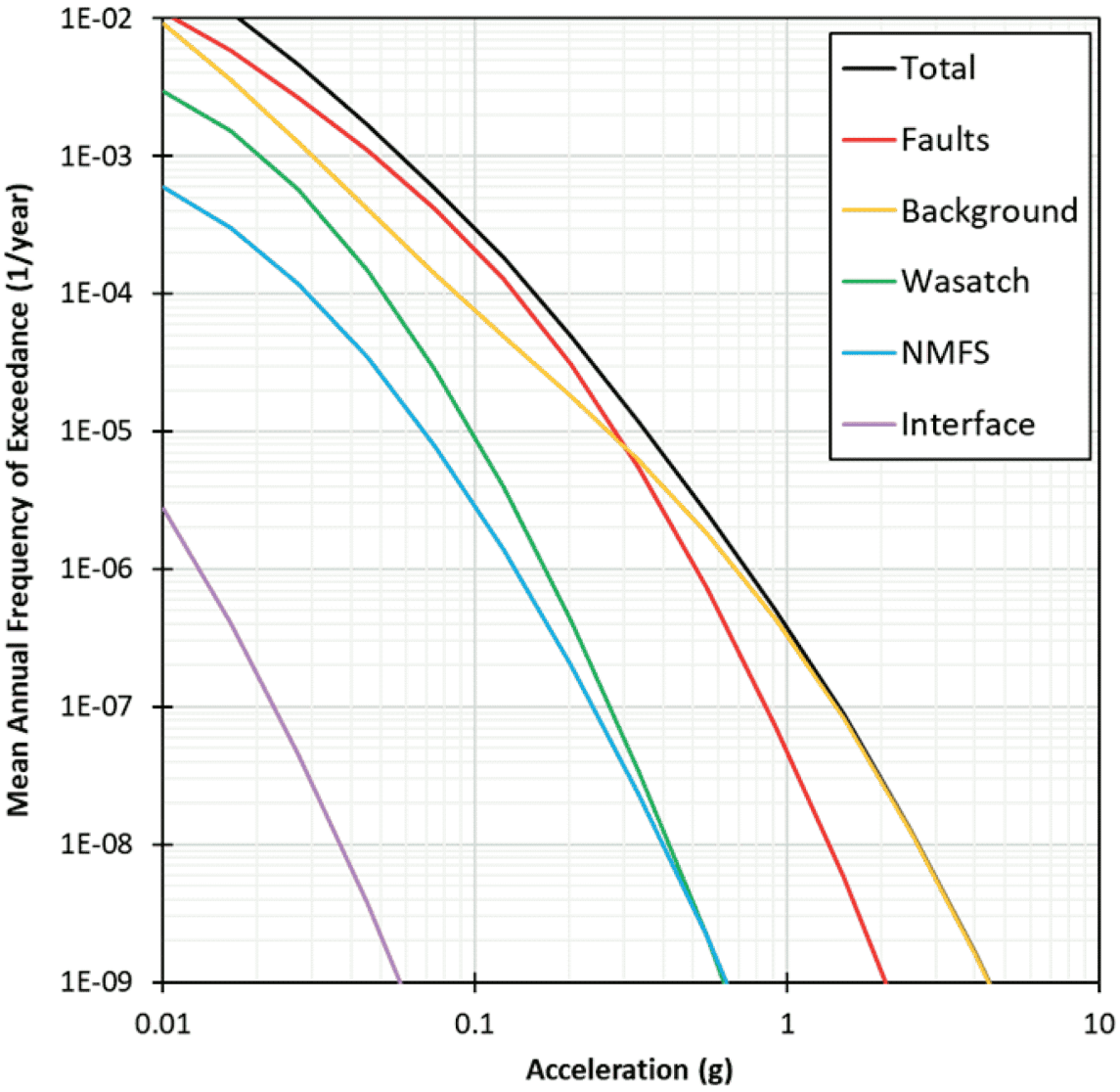




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**Figure 2.6-62 Contribution of source types to 10 second mean reference rock hazard at the site**

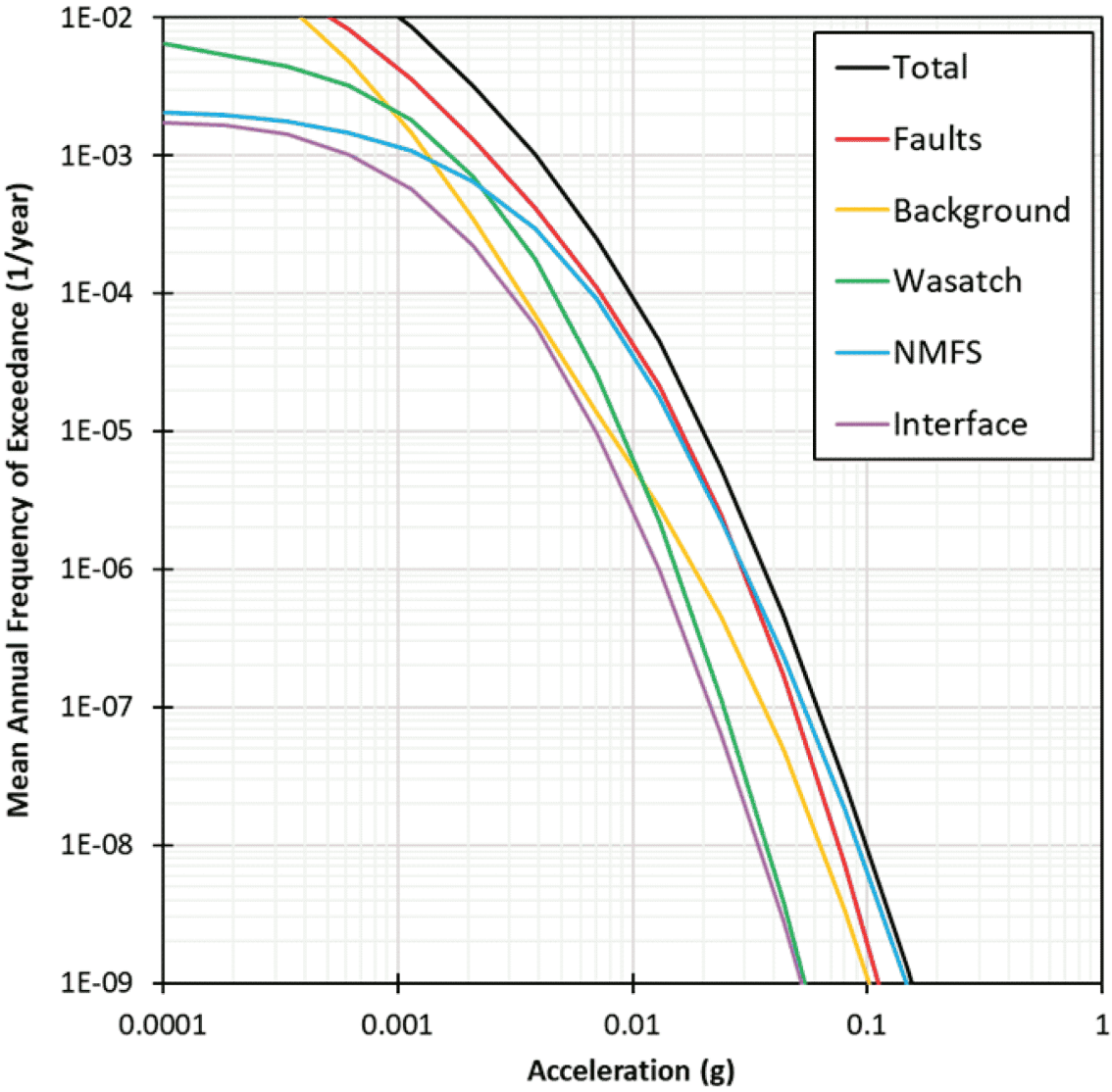



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**Figure 2.6-63 Mean and fractiles of 0.01 second total reference rock hazard at the site**

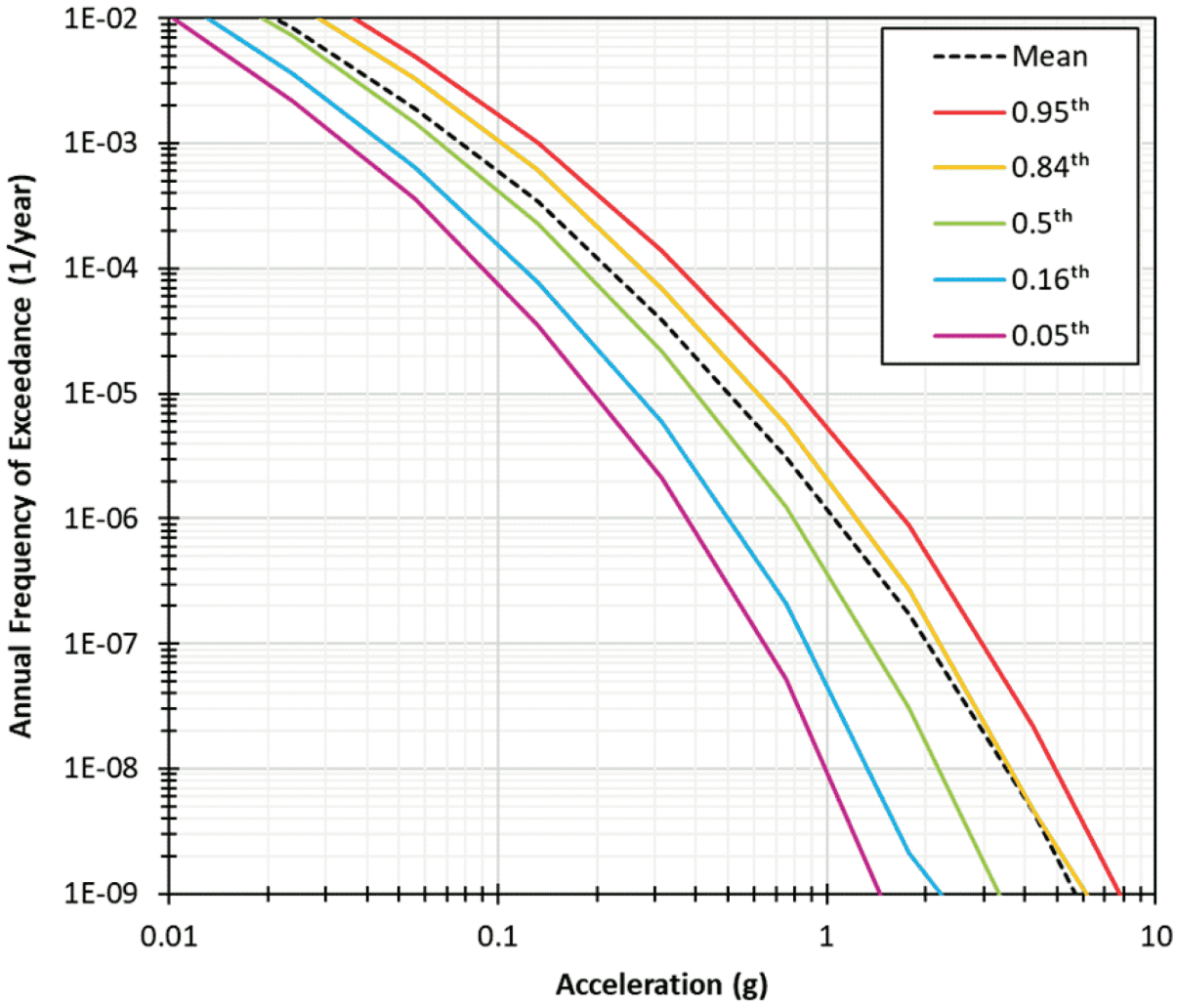




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**Figure 2.6-64 Mean and fractiles of 0.1 second total reference rock hazard at the site**

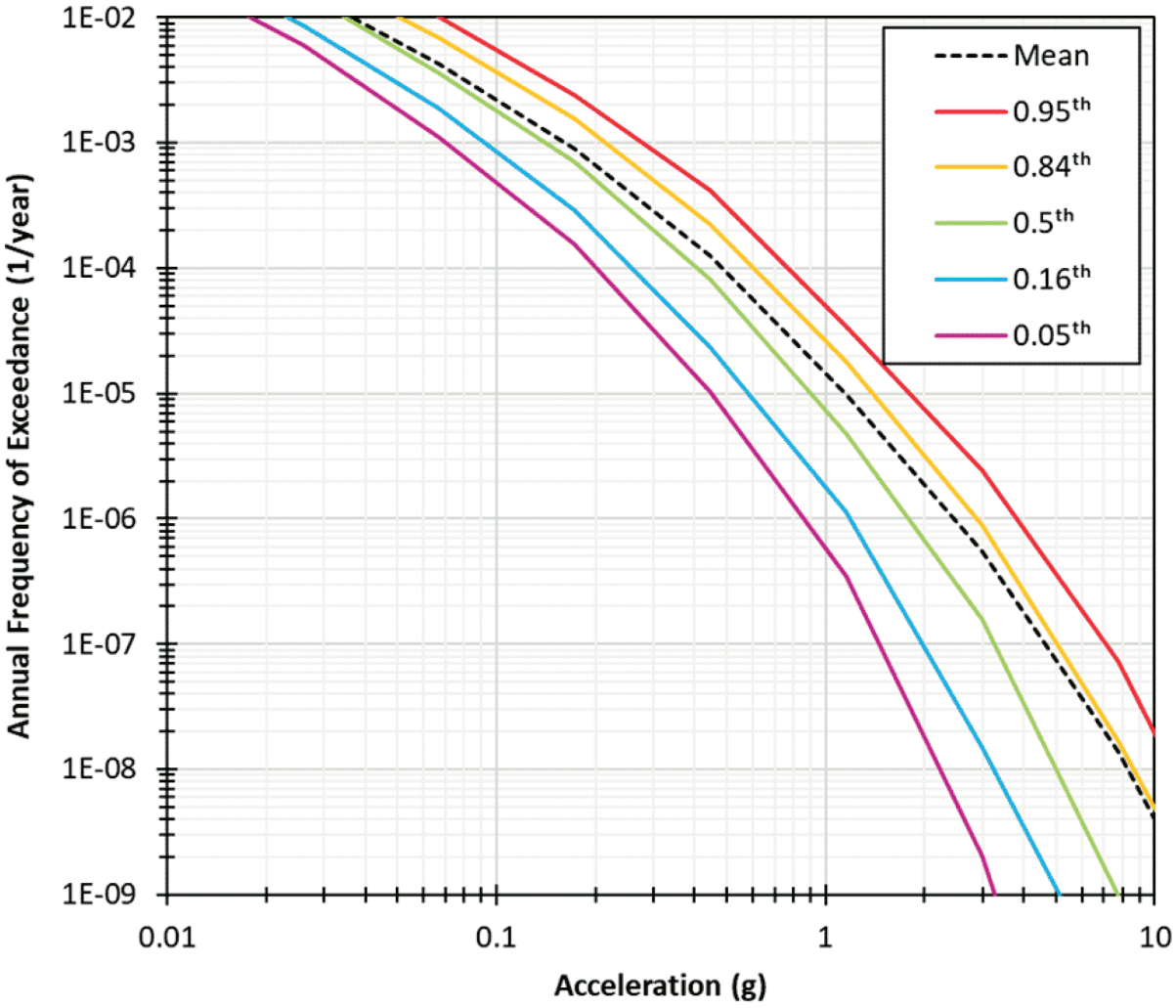



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**Figure 2.6-65 Mean and fractiles of 1 second total reference rock hazard at the site**

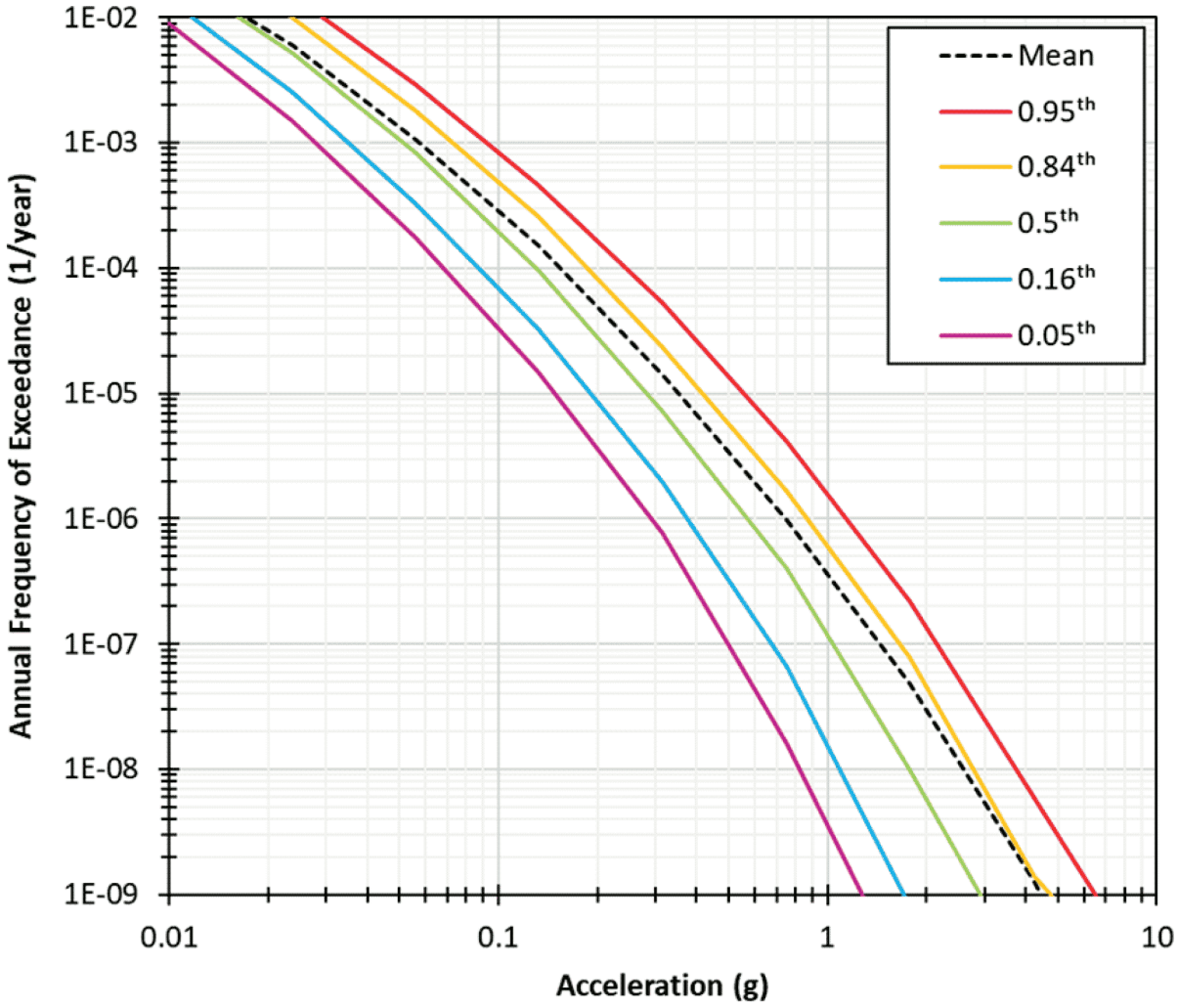



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**Figure 2.6-66 Mean and fractiles of 10 second total reference rock hazard at the site**

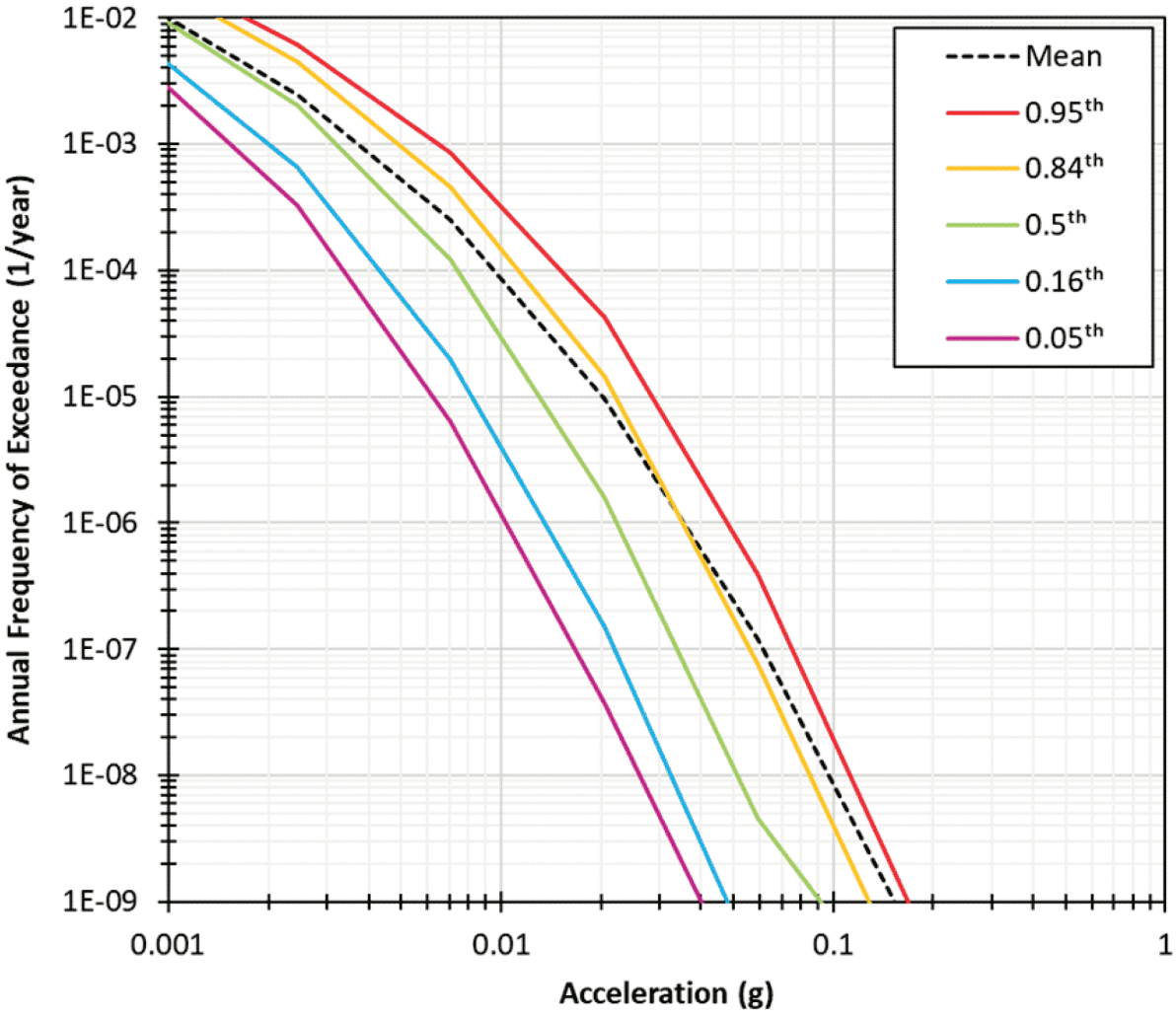



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**Figure 2.6-67 Mean reference rock UHRS for eleven AFEs at the site**

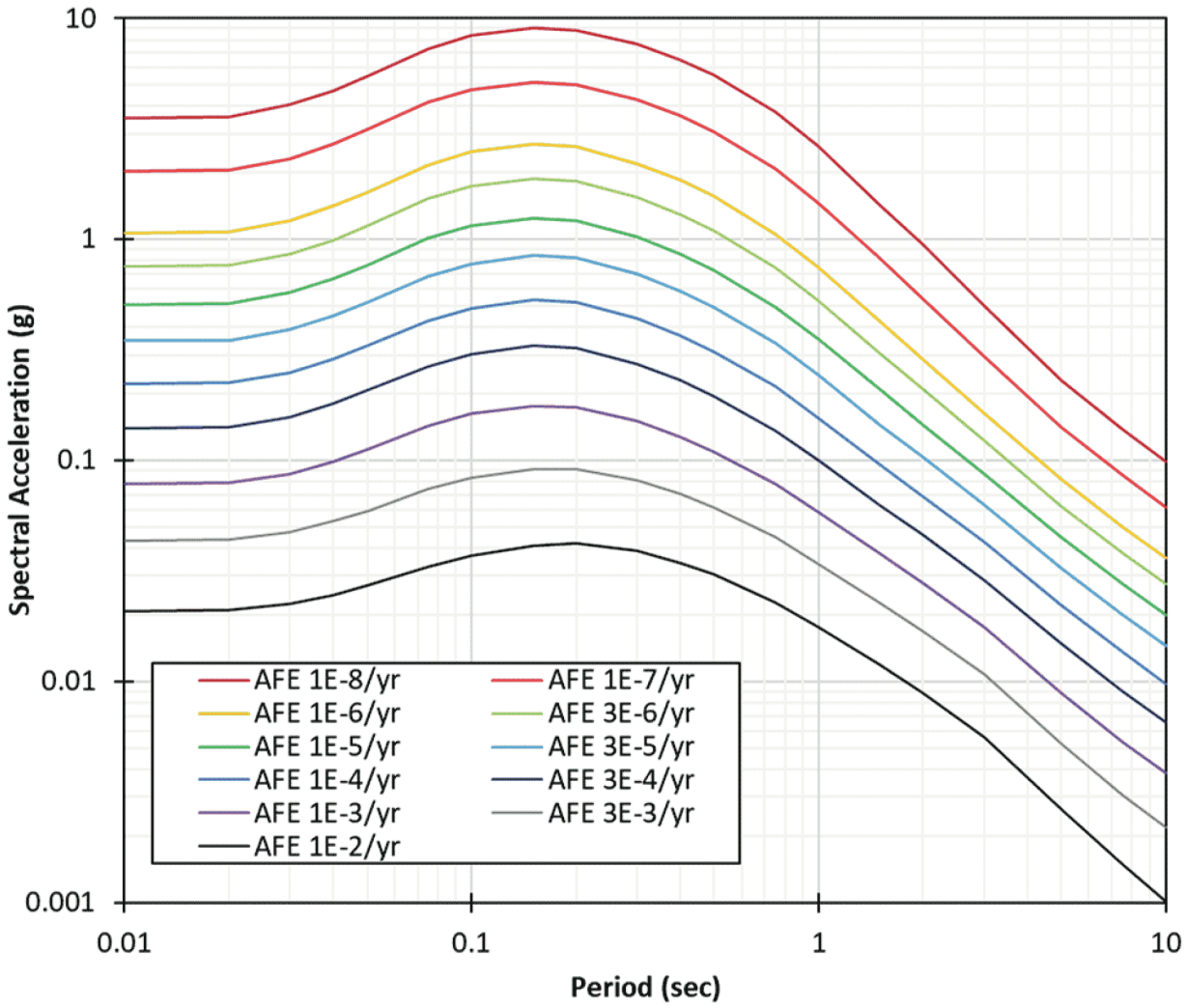




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**Figure 2.6-68 Contribution of M, R Pairs to Mean 1E-4/yr 0.1 second Reference Rock Hazard at the site**

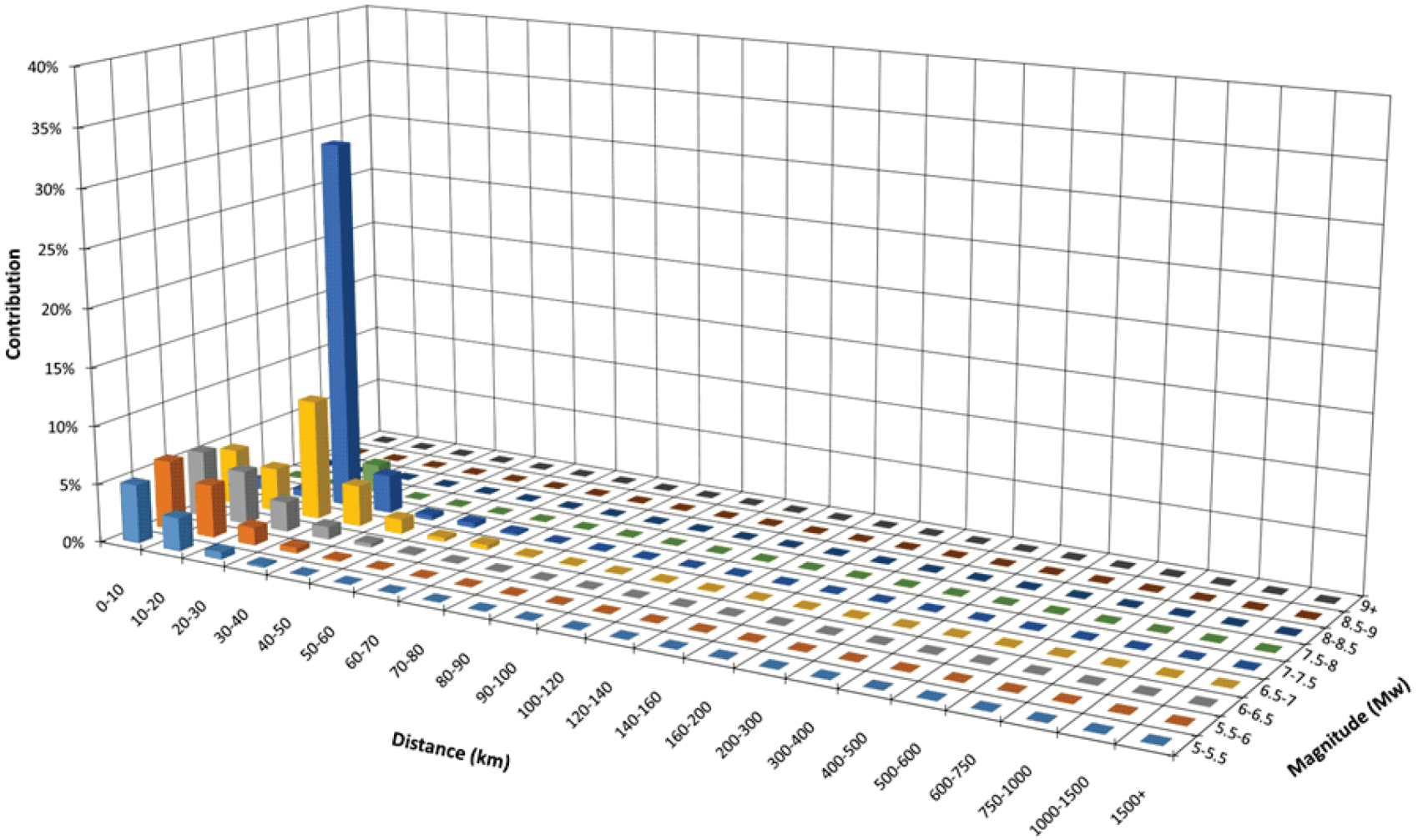



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**Figure 2.6-69 Contribution of M, R Pairs to Mean 1E-4/yr 1 second Reference Rock Hazard at the site**

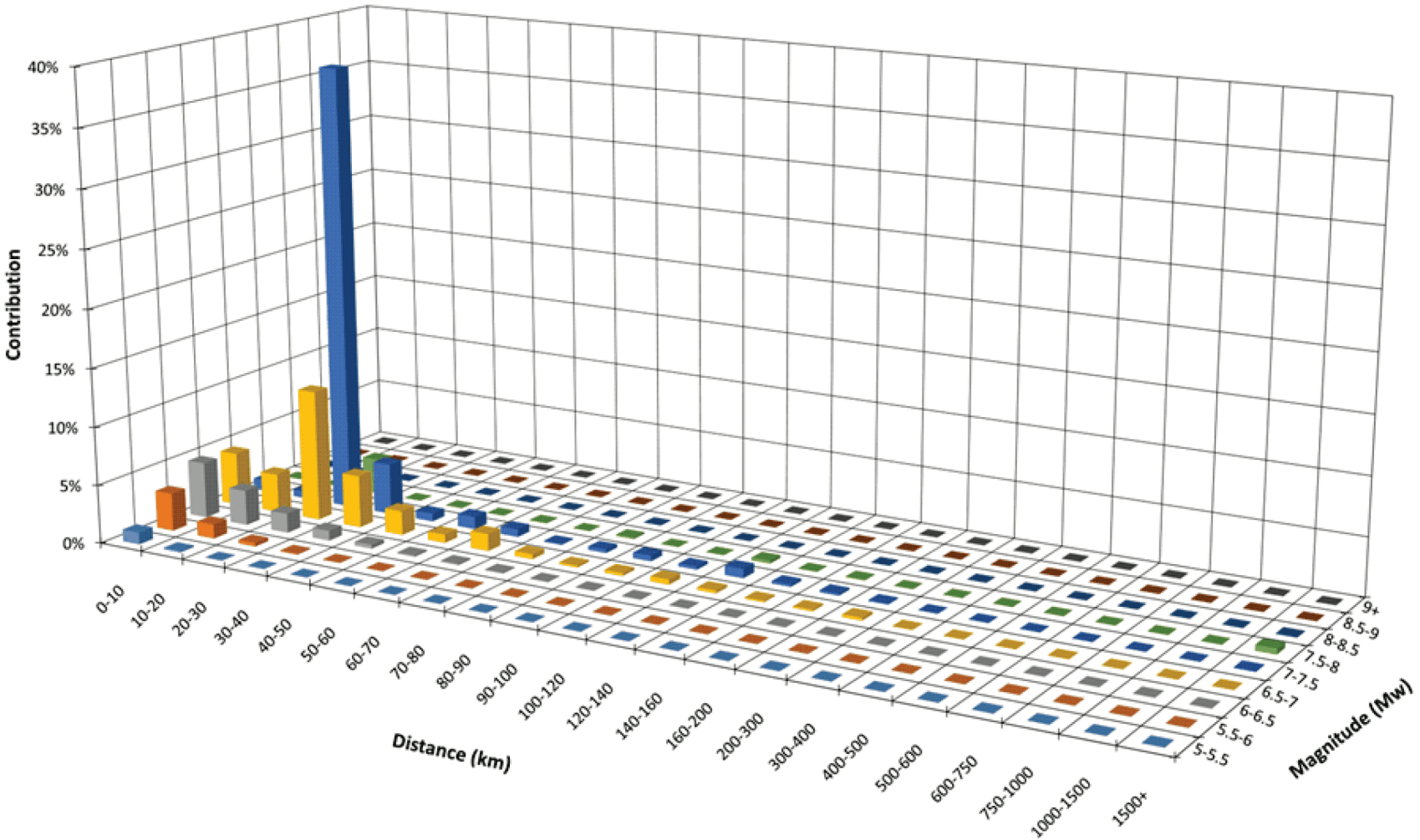



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**Figure 2.6-70 Contribution of M, R Pairs to Mean 1E-5/yr 0.1 second Reference Rock Hazard at the site**

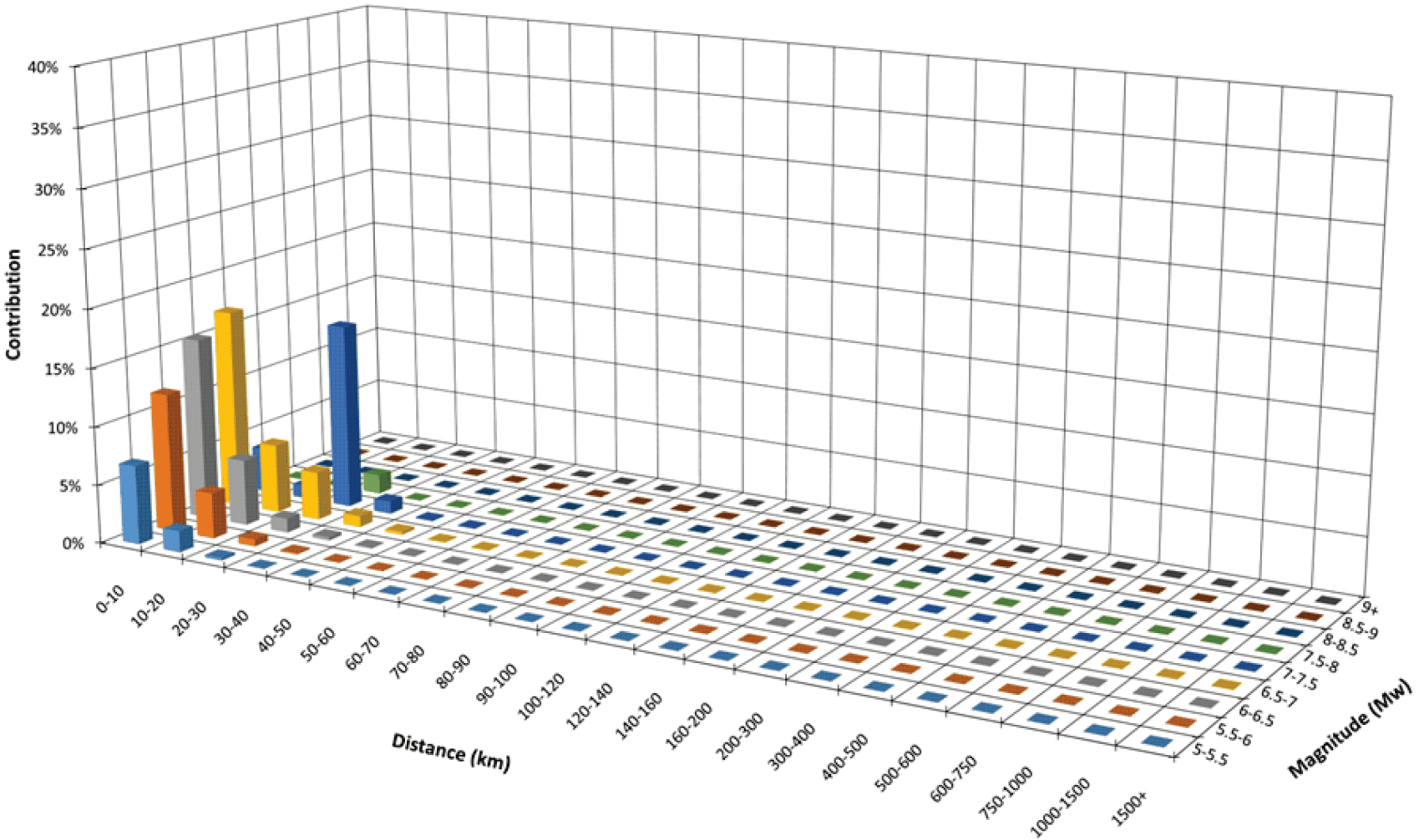



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**Figure 2.6-71 Contribution of M, R Pairs to Mean 1E-5/yr 1 second Reference Rock Hazard at the site**

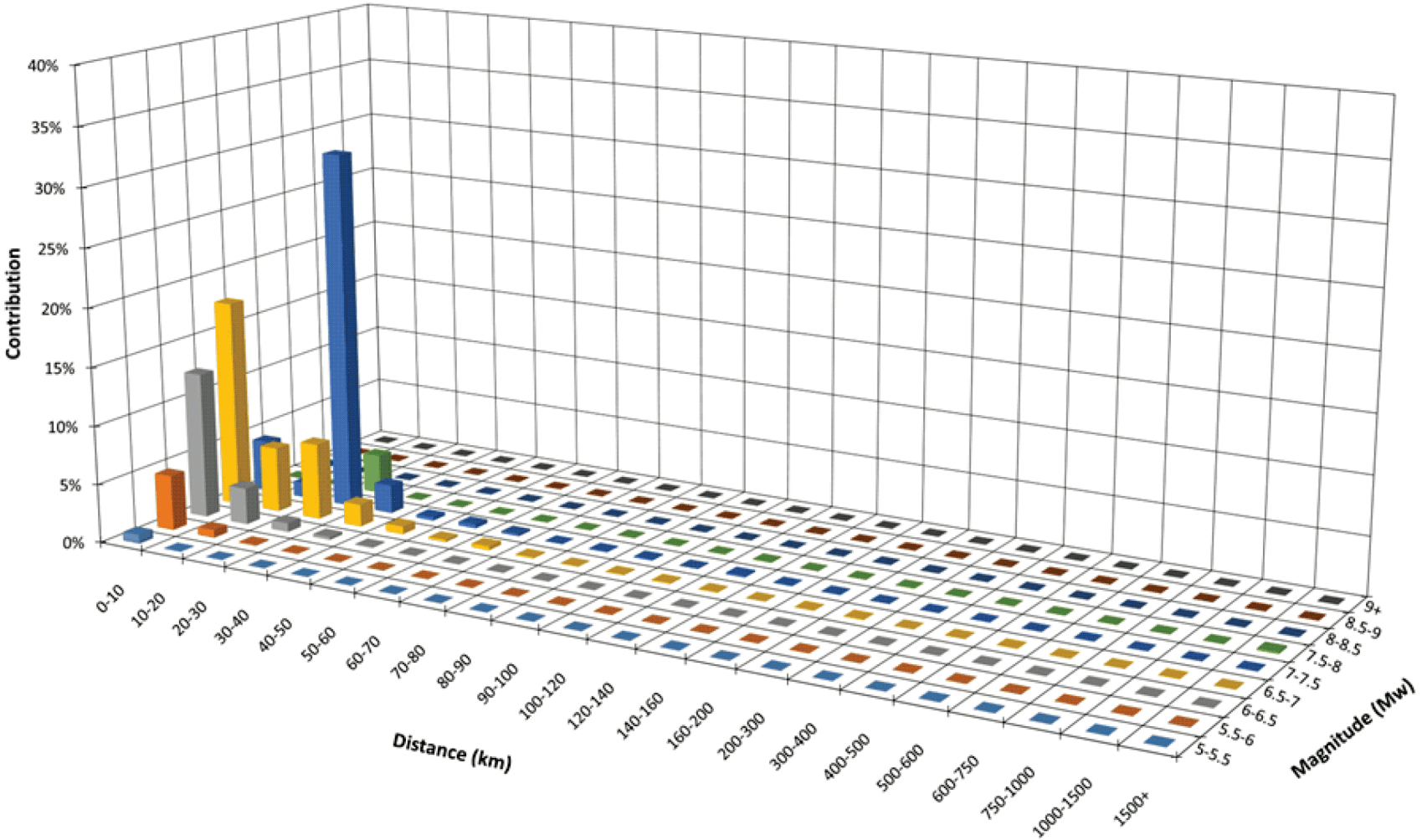



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**Figure 2.6-72 Contribution of M, R Pairs to Mean 1E-6/yr 0.1 second Reference Rock Hazard at the site**

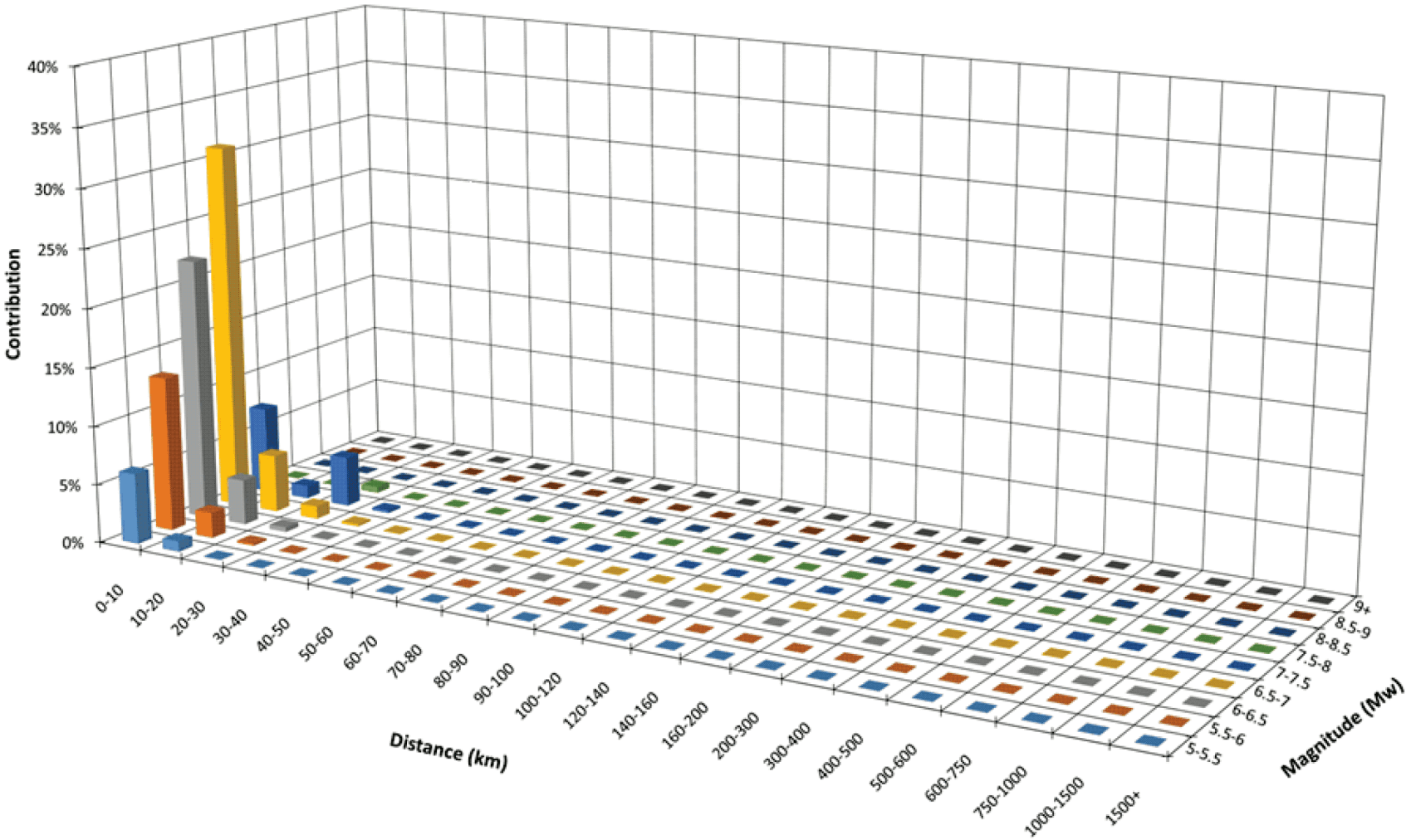



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**Figure 2.6-73 Contribution of M, R Pairs to Mean 1E-6/yr 1 second Reference Rock Hazard at the site**

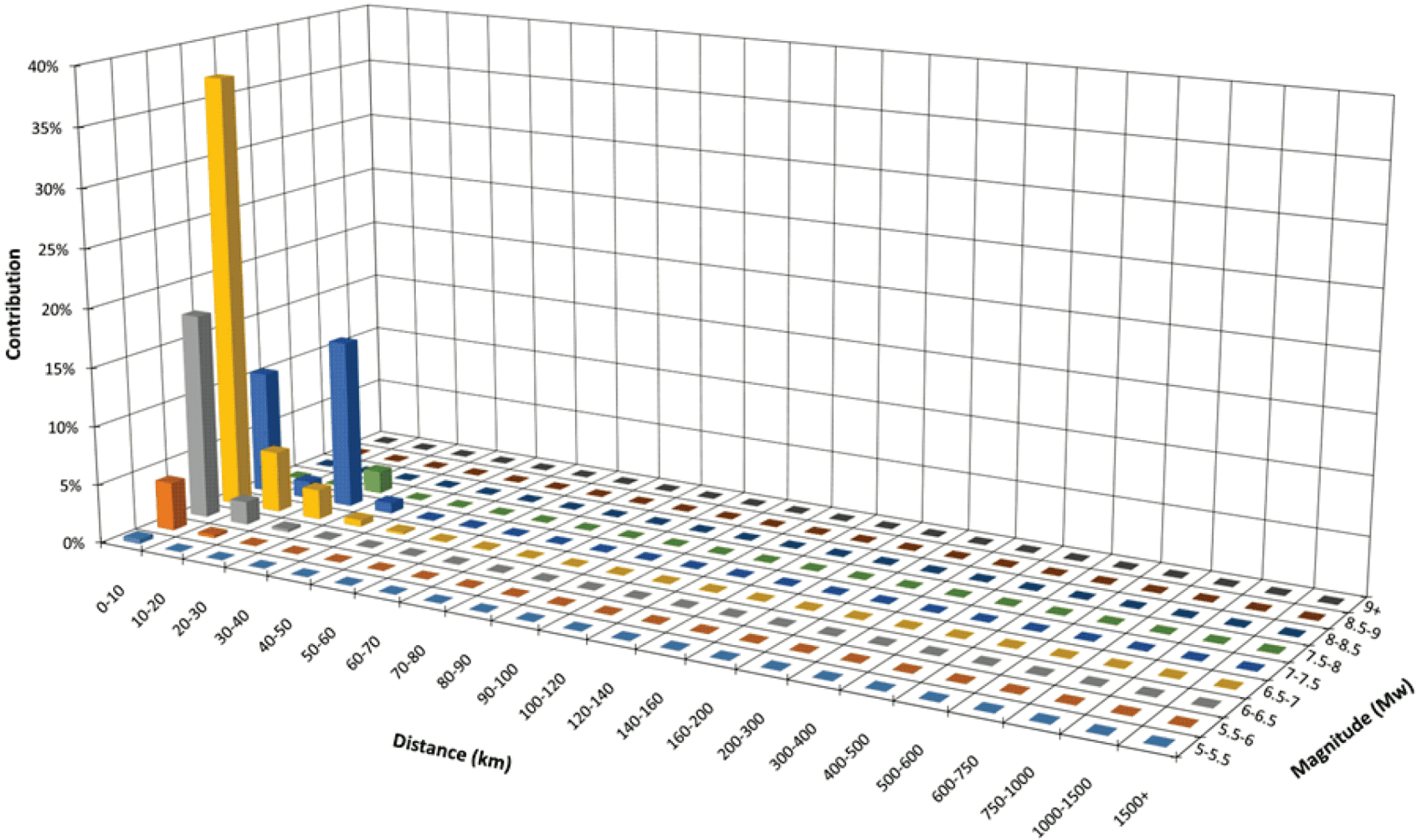



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**Figure 2.6-74 Seven Median SAF Models Corresponding to Each of the Discretized Fractiles of the Distribution of Epistemic Uncertainty for Input Level 5 corresponding to an AFE of 1E-5**

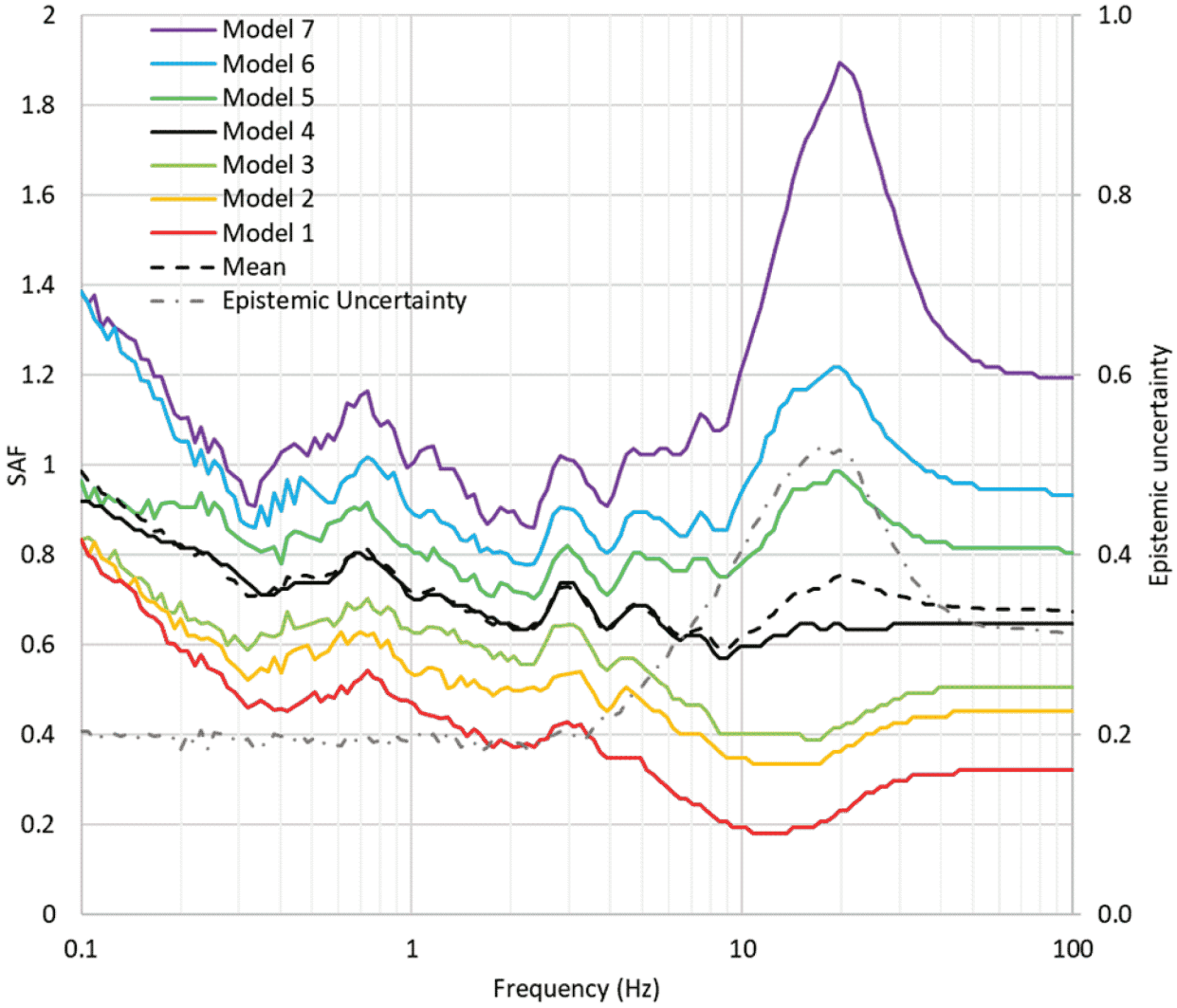



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**Figure 2.6-75 Mean Horizontal UHRS at PSHA Control Point for 11 AFEs at the site**

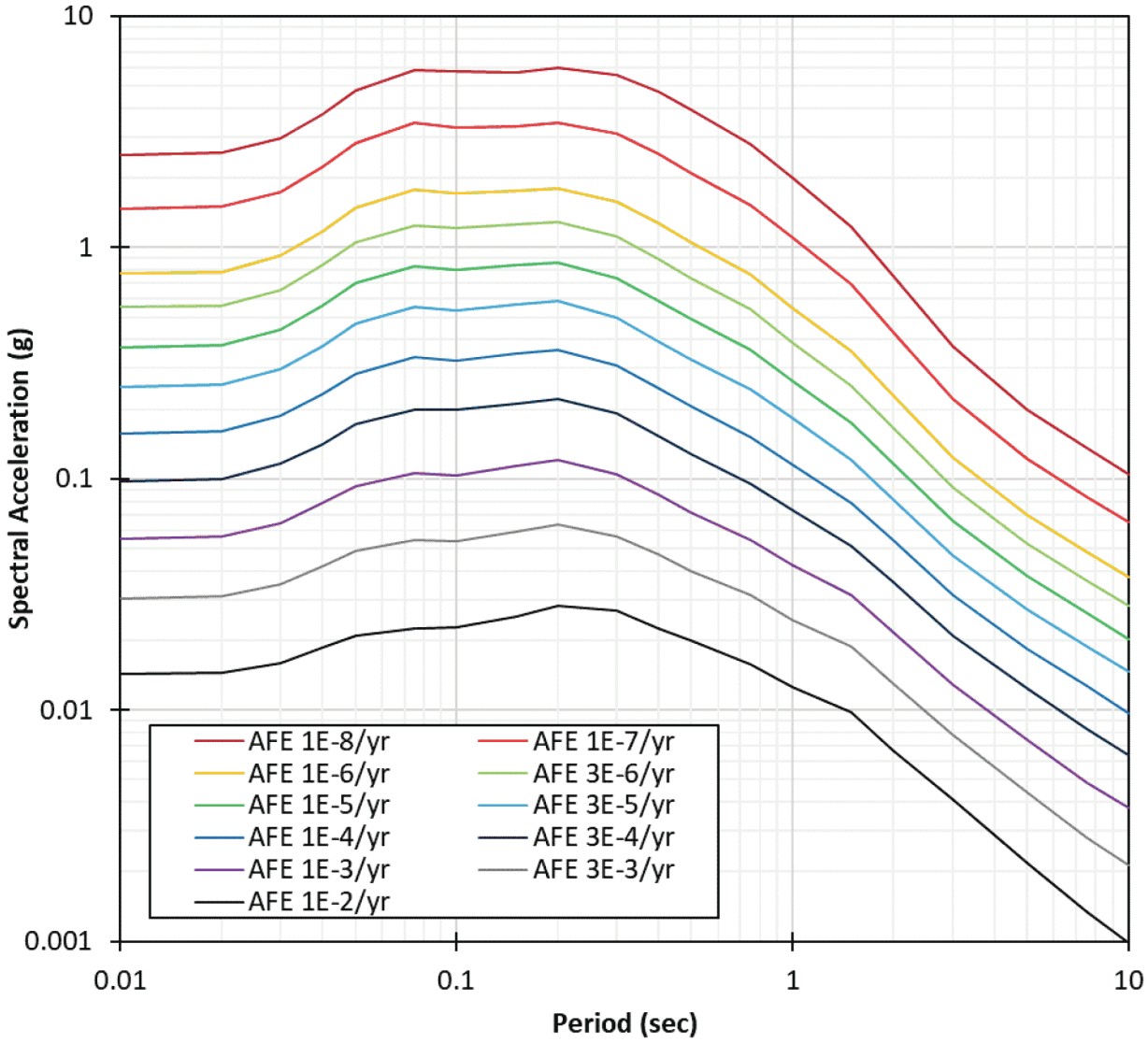



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**Figure 2.6-76 GA11 V/H Ratios at PSHA Control Point for an AFE of 1E-4/yr at the site**

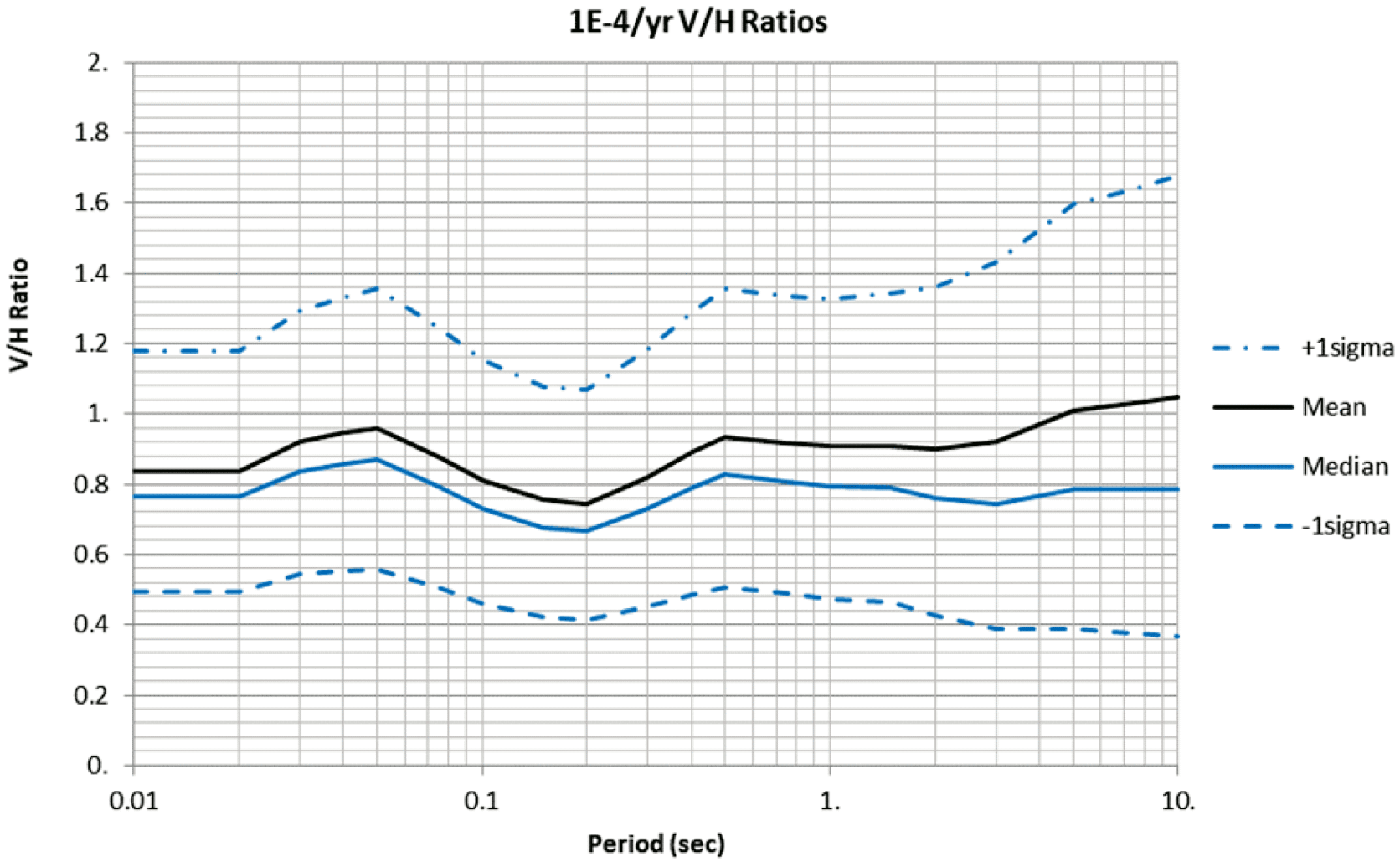






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**Figure 2.6-77 GA11 Mean V/H Ratios at PSHA Control Point for AFEs of 1E-4, 1E-5, and 1E 6/yr at the site**

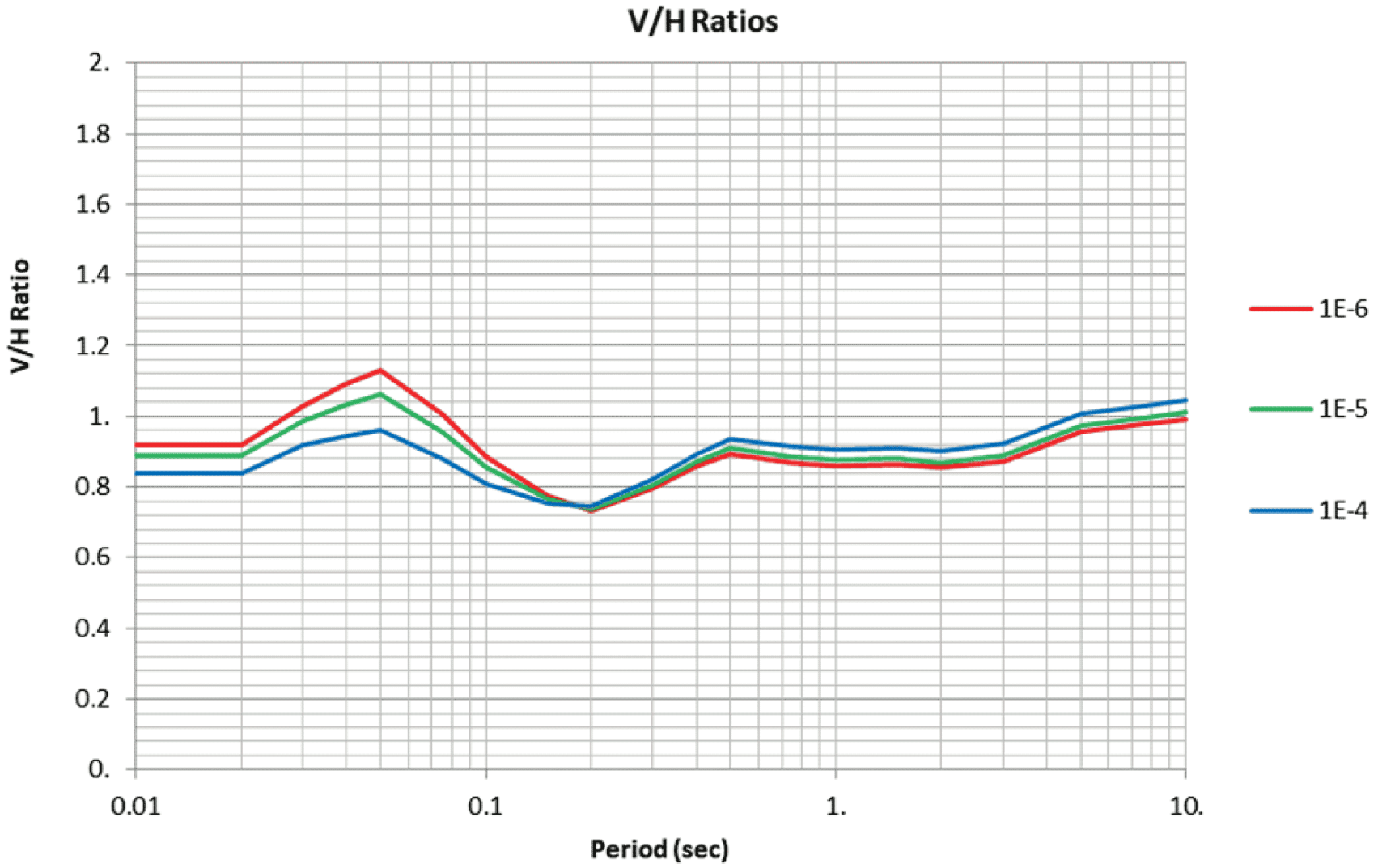


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**Figure 2.6-78 Mean Horizontal and Vertical UHRS at PSHA Control Point for AFEs of 1E-4, 1E-5, and 1E-6/yr at the site**

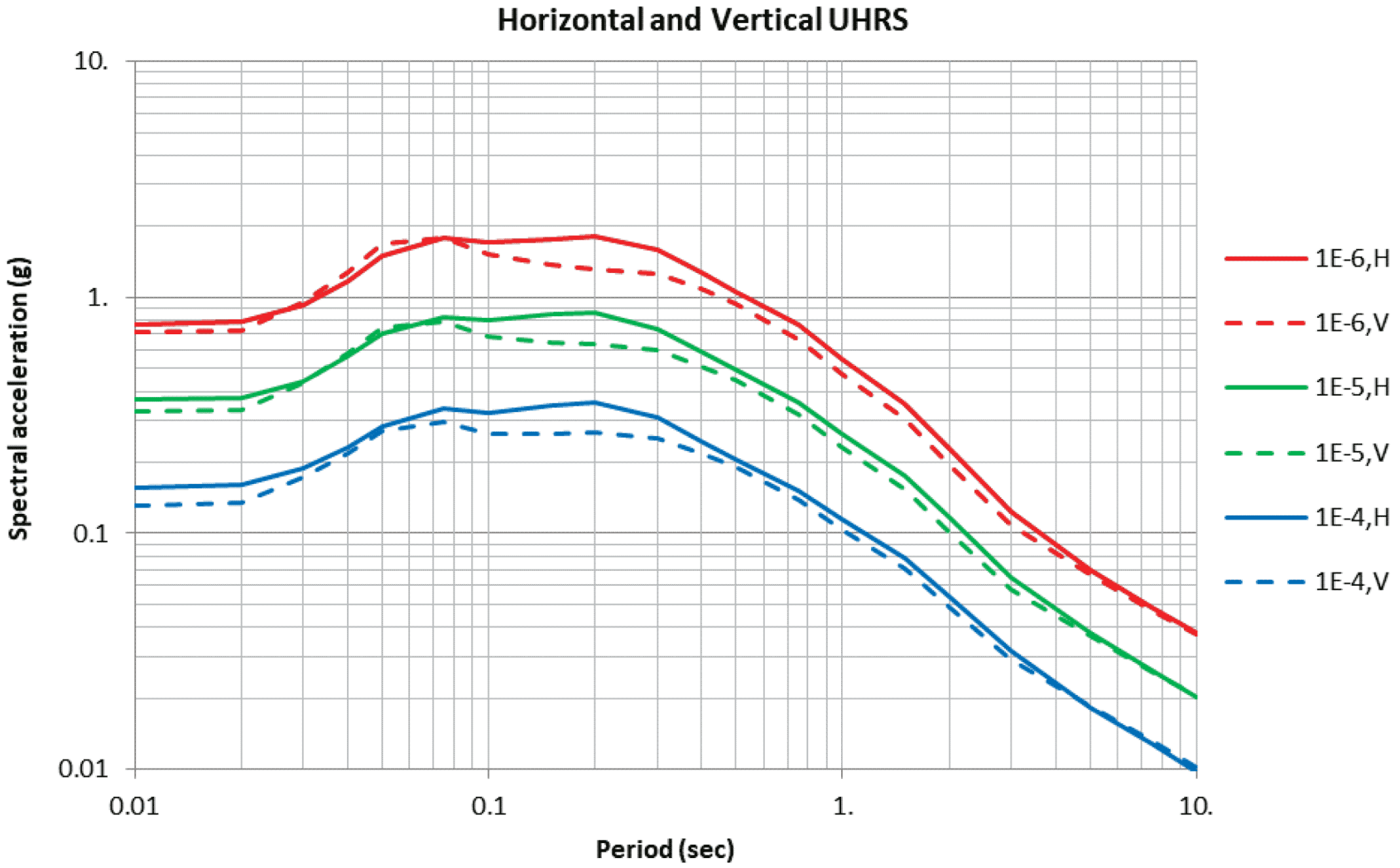






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**Figure 2.6-79 Site Horizontal GMRS and UHRS at GMRS Horizon**

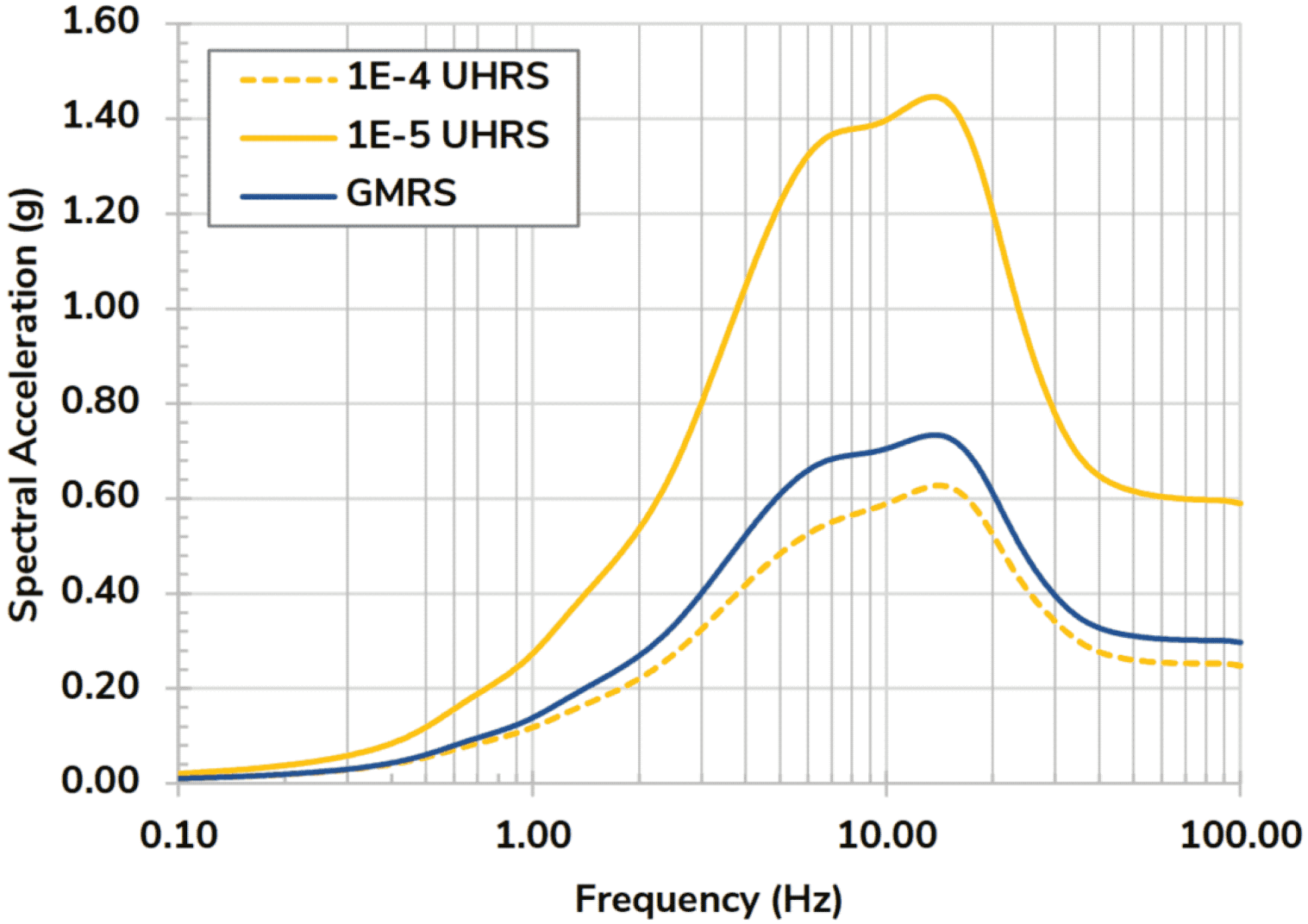



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**Figure 2.6-80 Damping Ratio and Shear Modulus Reduction Curves for GMRS Calculation**

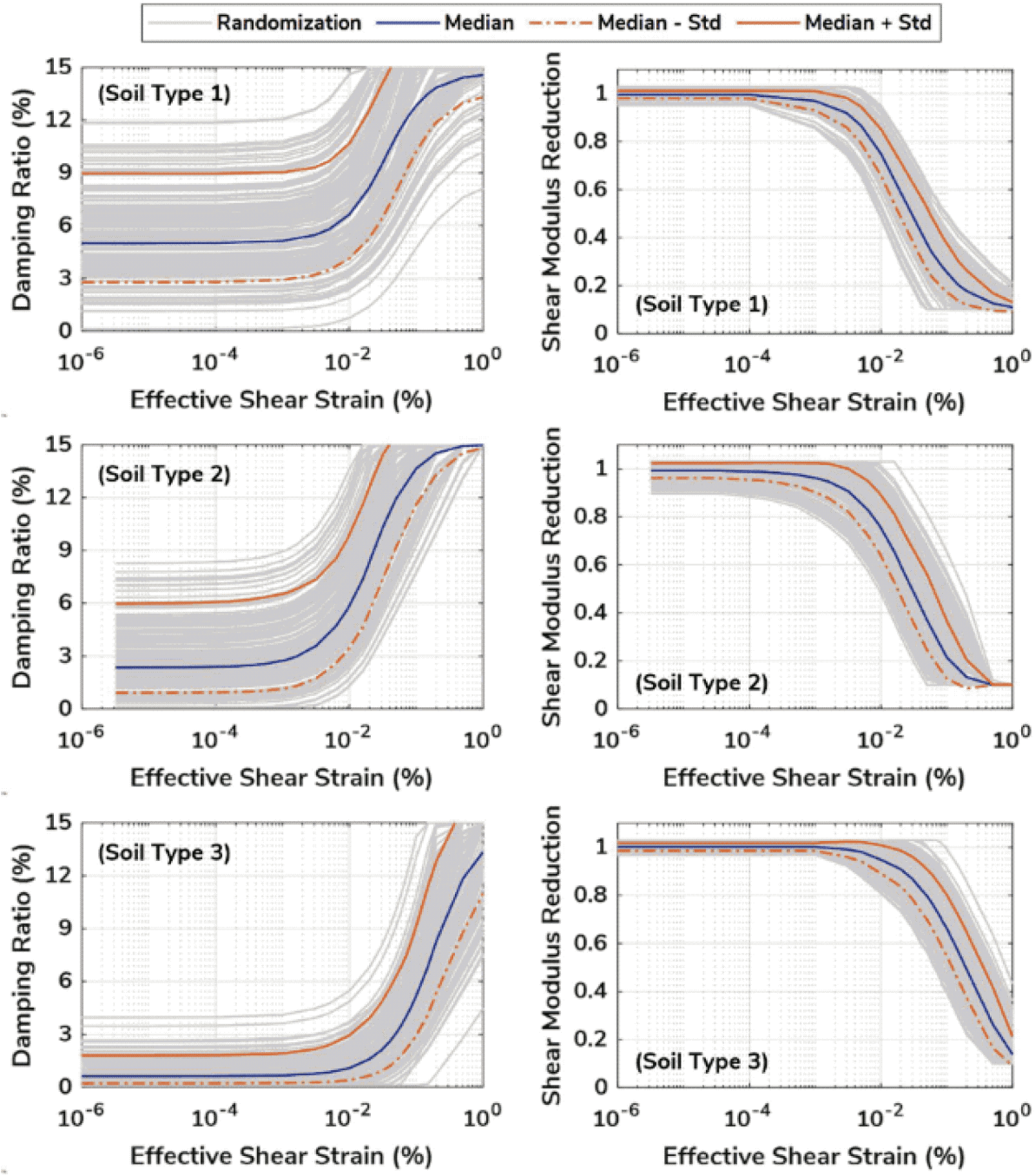



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**Figure 2.6-81 Not Used**


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**Figure 2.6-82 Maximum Shear Strain Profiles Corresponding to 1E-4 and 1E-5 AFE UHRS at GMRS Horizon**

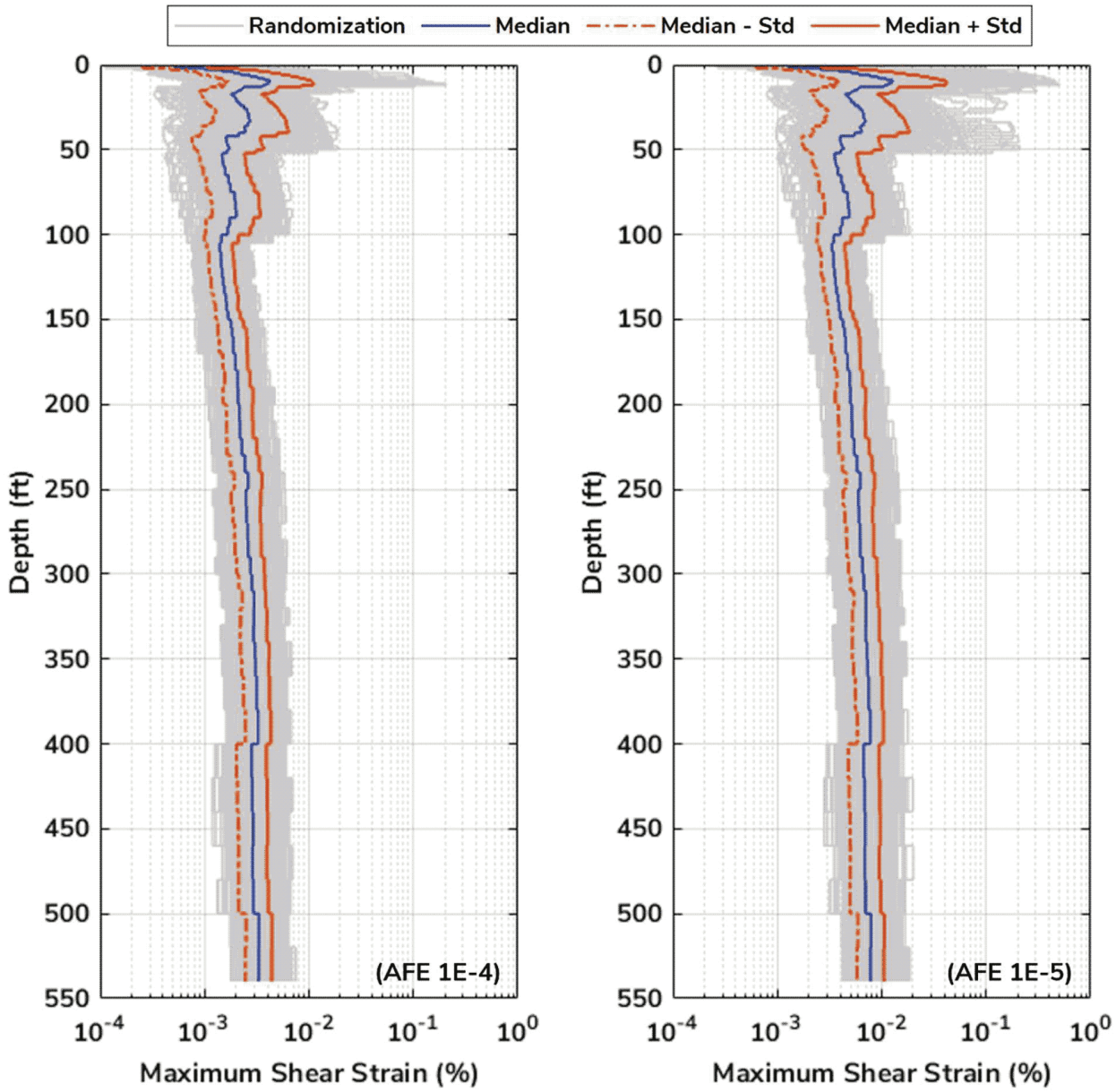



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**Figure 2.6-83 Strain-Compatible Shear Wave Velocity Profiles Corresponding to 1E-4 and 1E-5 AFE UHRS at GMRS Horizon**

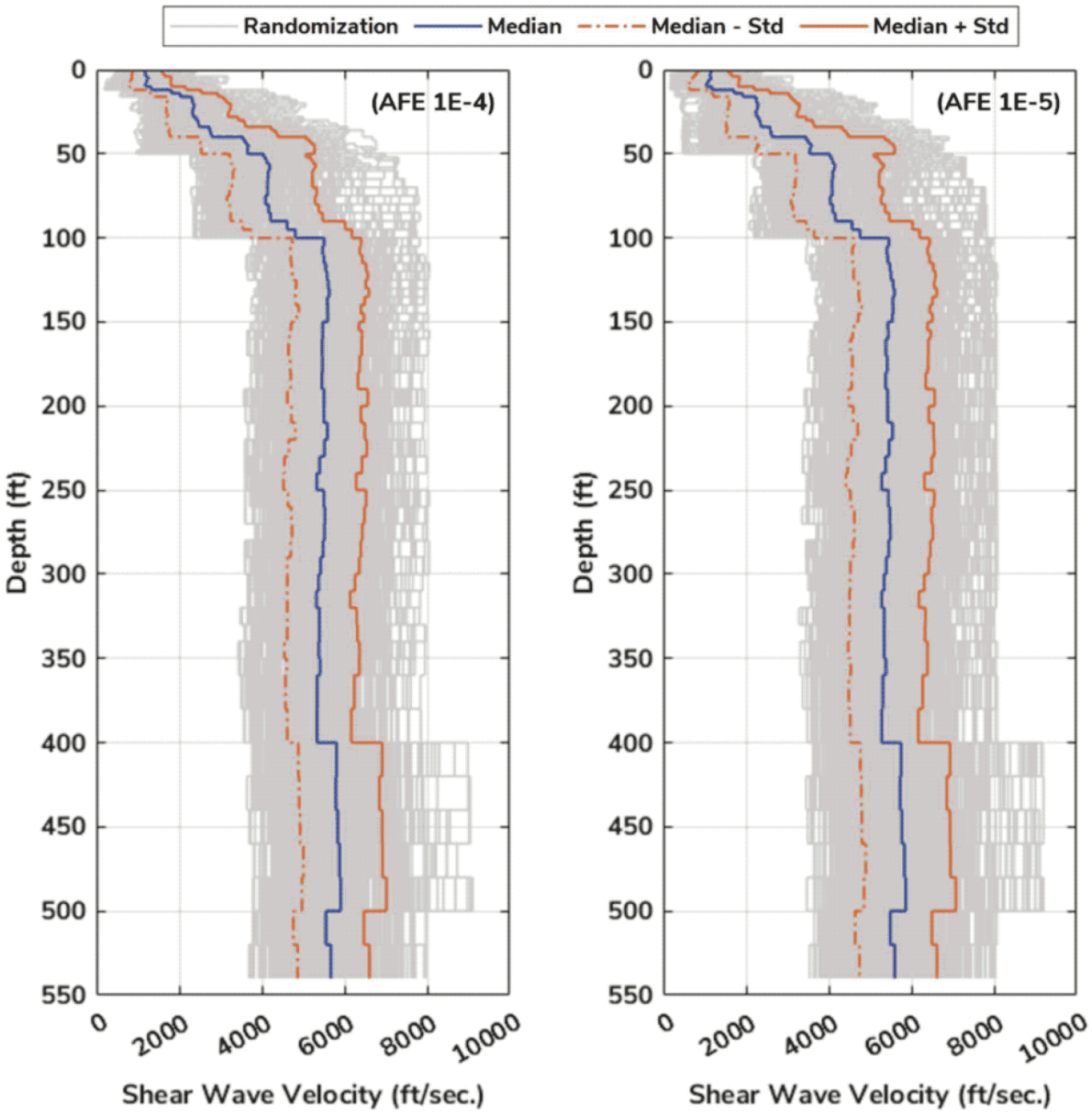



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**Figure 2.6-84 Strain-Compatible Damping Ratio Profiles Corresponding to 1E-4 and 1E-5 AFE UHRS at GMRS Horizon**

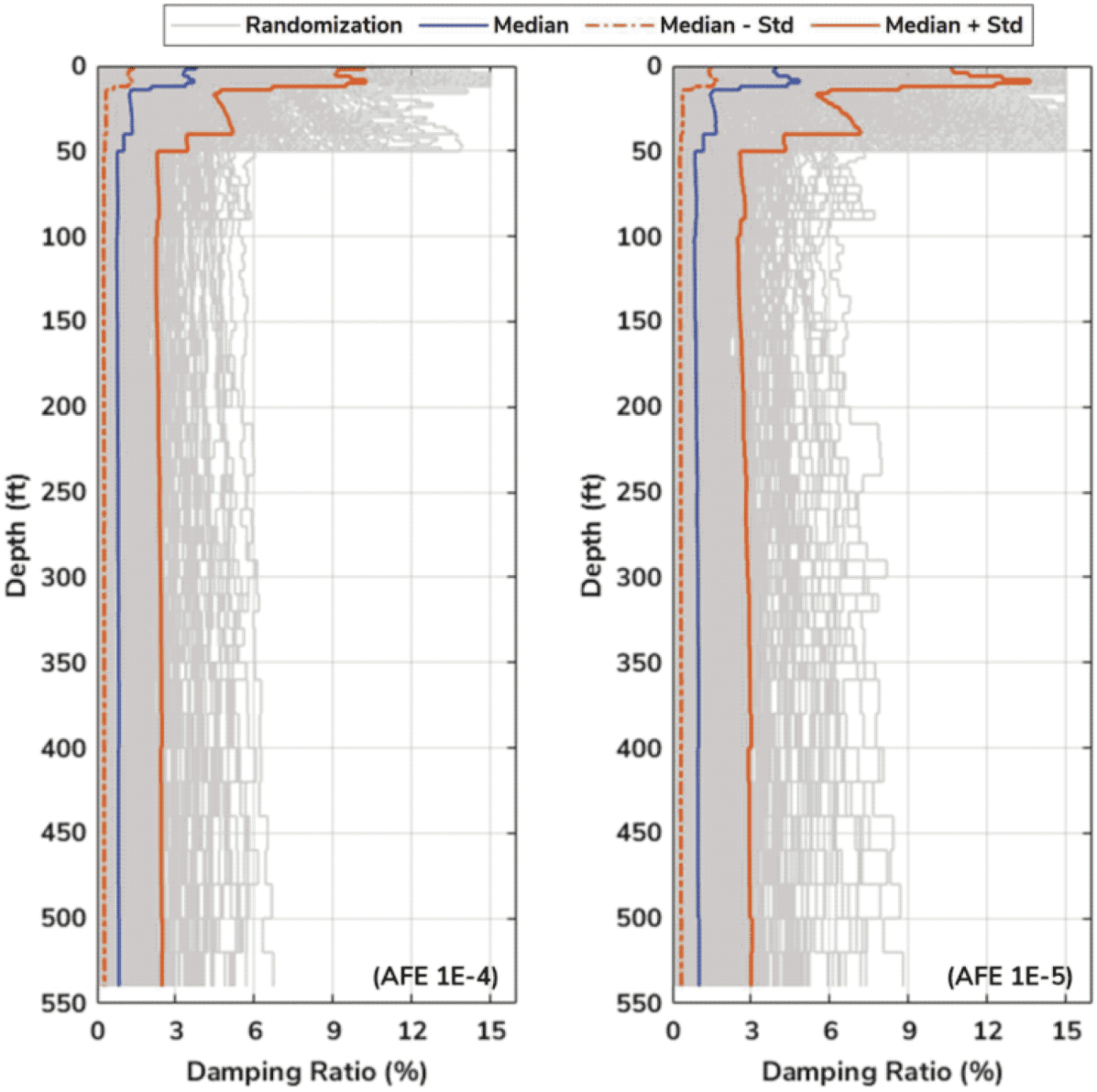



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**Figure 2.6-85 Mean V/H Ratios for Vertical GMRS and SSE Calculation**

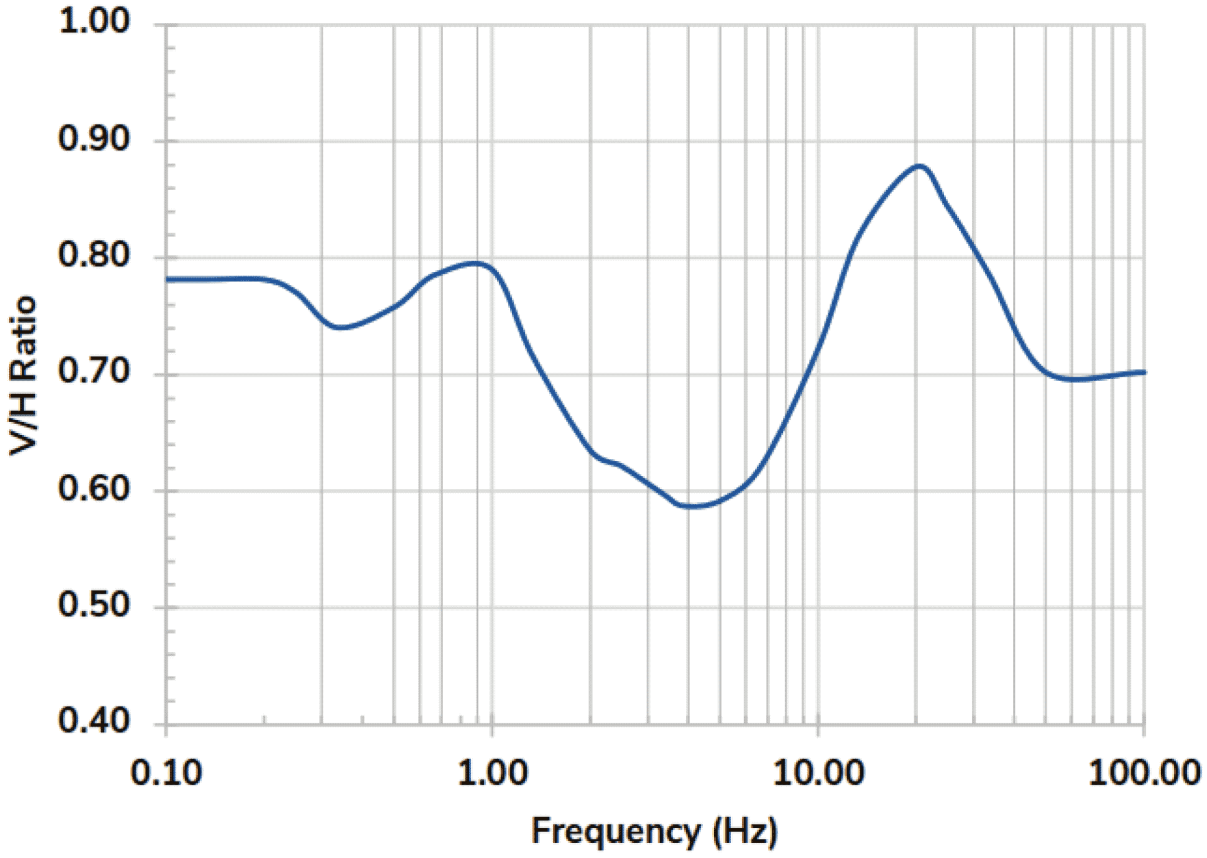



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**Figure 2.6-86 Site Horizontal and Vertical GMRS and SSE**

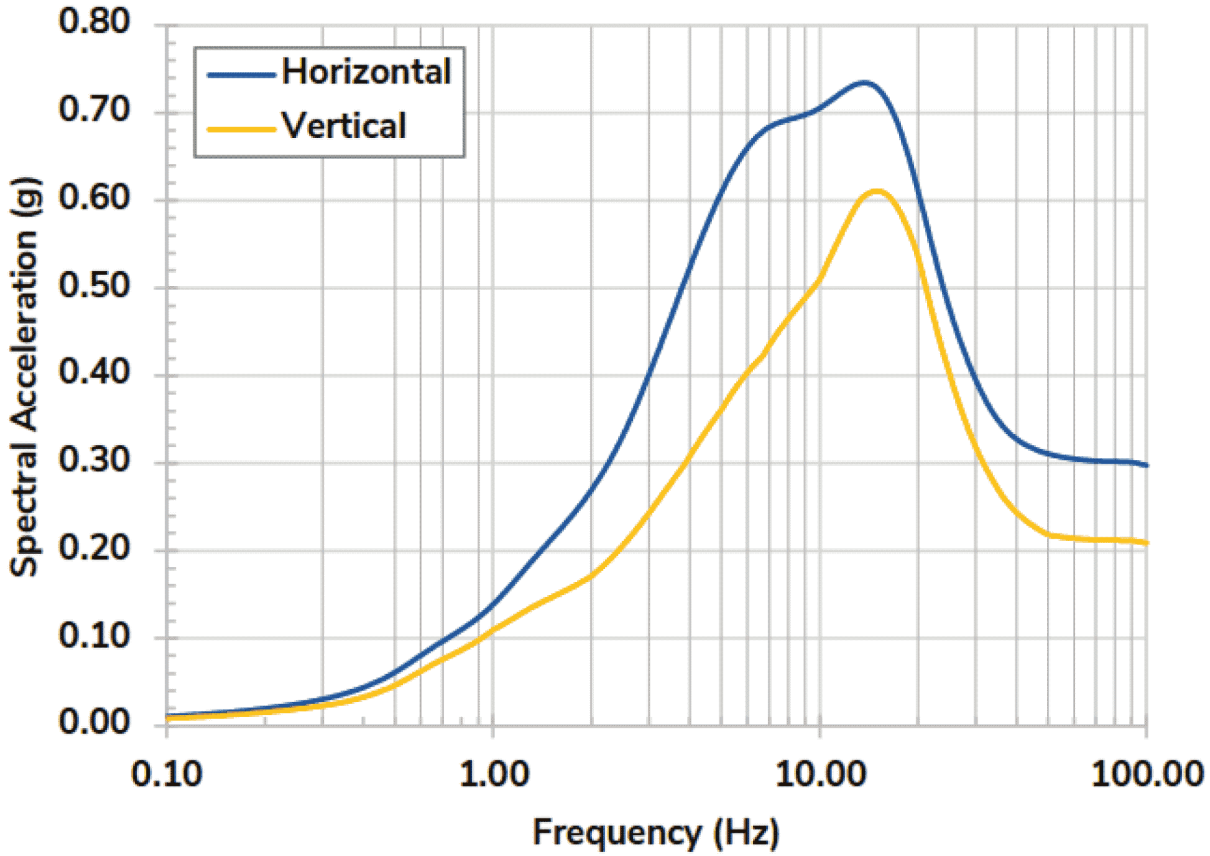



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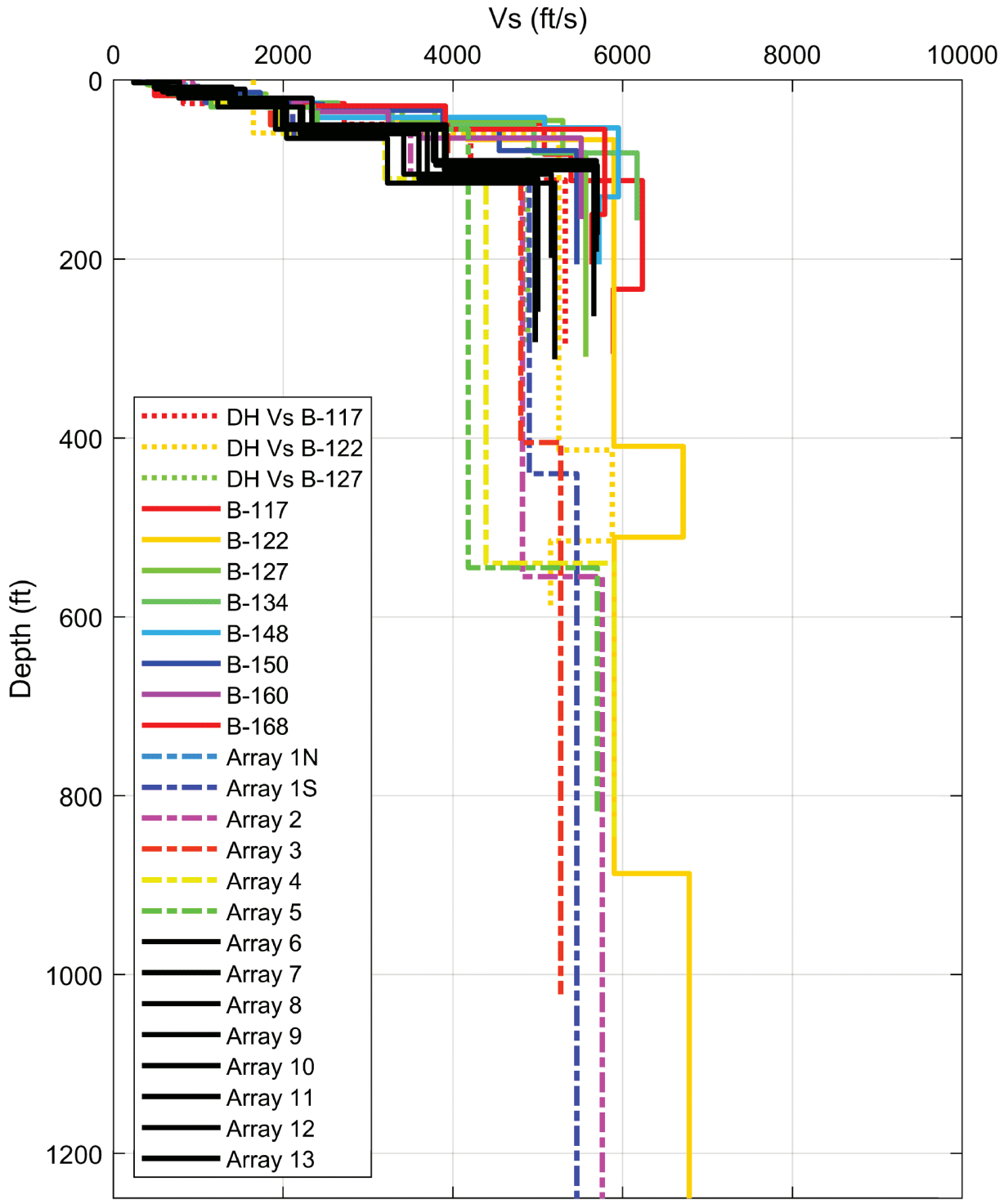
**Figure 2.6-87 All VS profiles from Downhole (DH), Suspension logging (B) and SASW measurements (Array)**





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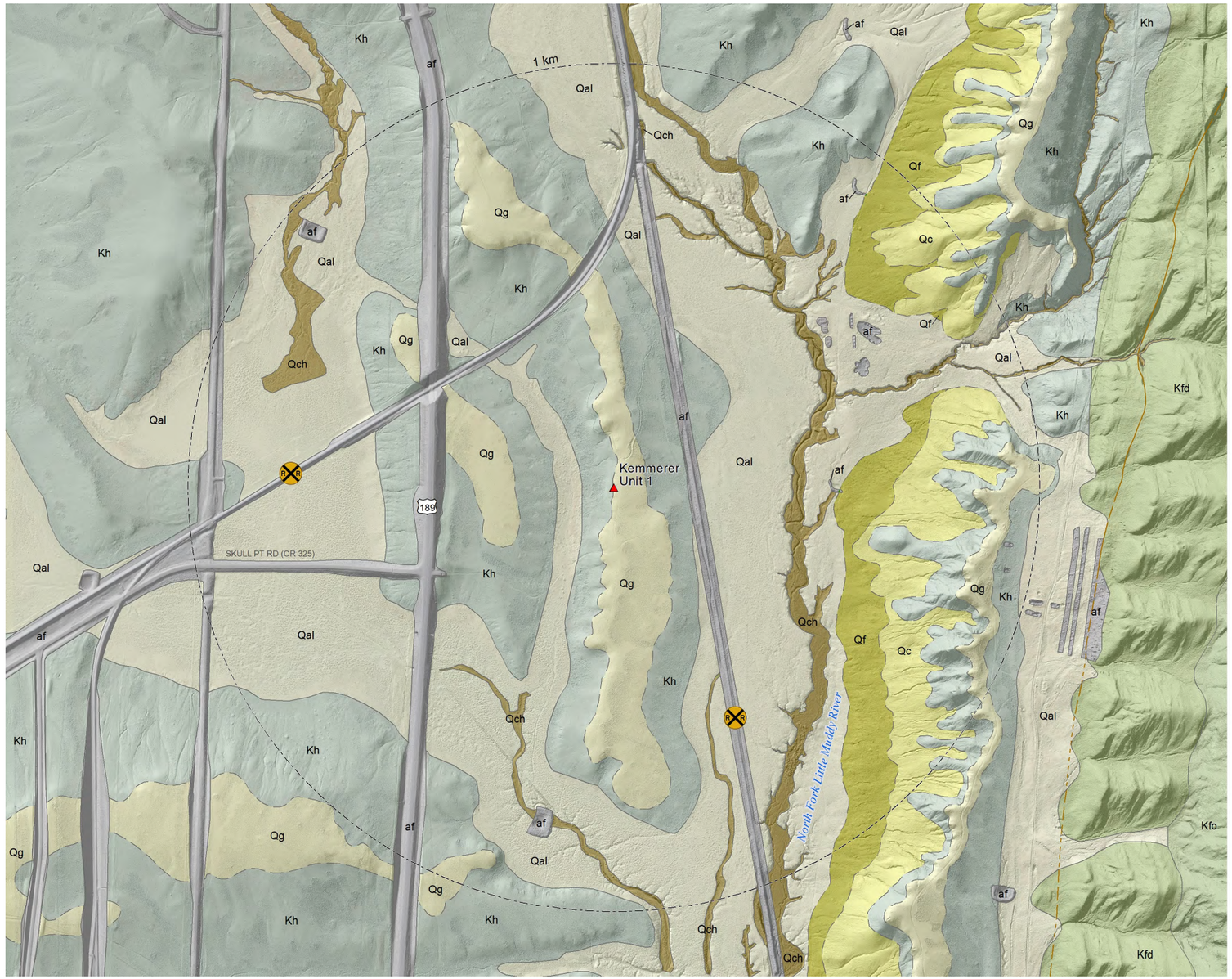
## Figure 2.6-88 Geologic Map of Site Location




**Surficial Units**

*Quaternary*

- af — Artificial fill
- Qch — Active channel
- Qal — Alluvium
- Qc — Colluvium
- Qf — Alluvium, colluvium and alluvial-fan deposits
- Qg — Gravel deposits

**Bedrock Units**

*Cretaceous*

- Kh — Hilliard Shale

**Frontier Formation**

- Kfd — Dry Hollow Member
- Kfo — Oyster Ridge Sandstone

- contact — *solid where certain, dashed where approximately located*
- coal — *solid where certain, dashed where approximately located, dotted where concealed or uncertain*

N

0 250 500 m

0 1,000 2,000 ft


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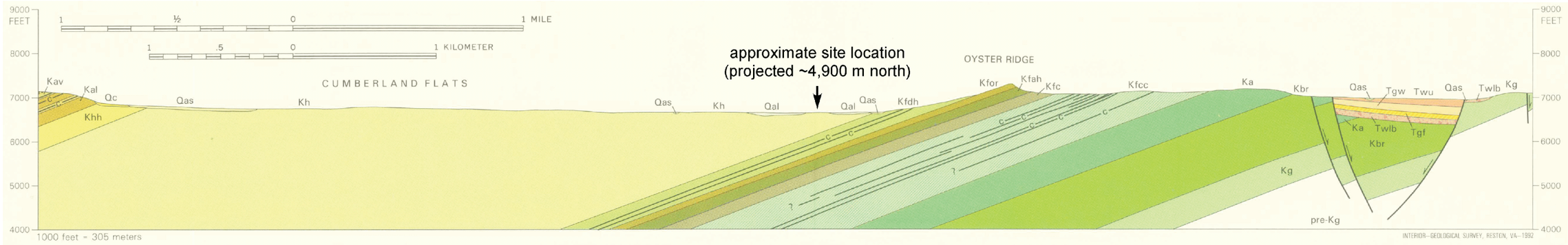
**Figure 2.6-89 Geologic Section of Elkol Quadrangle**




# GEOLOGIC MAP OF THE ELKOL QUADRANGLE, SOUTHWESTERN WYOMING

By

John W. M'Gonigle

1992



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## Figure 2.6-90 Stratigraphic Column of Elkol Quadrangle

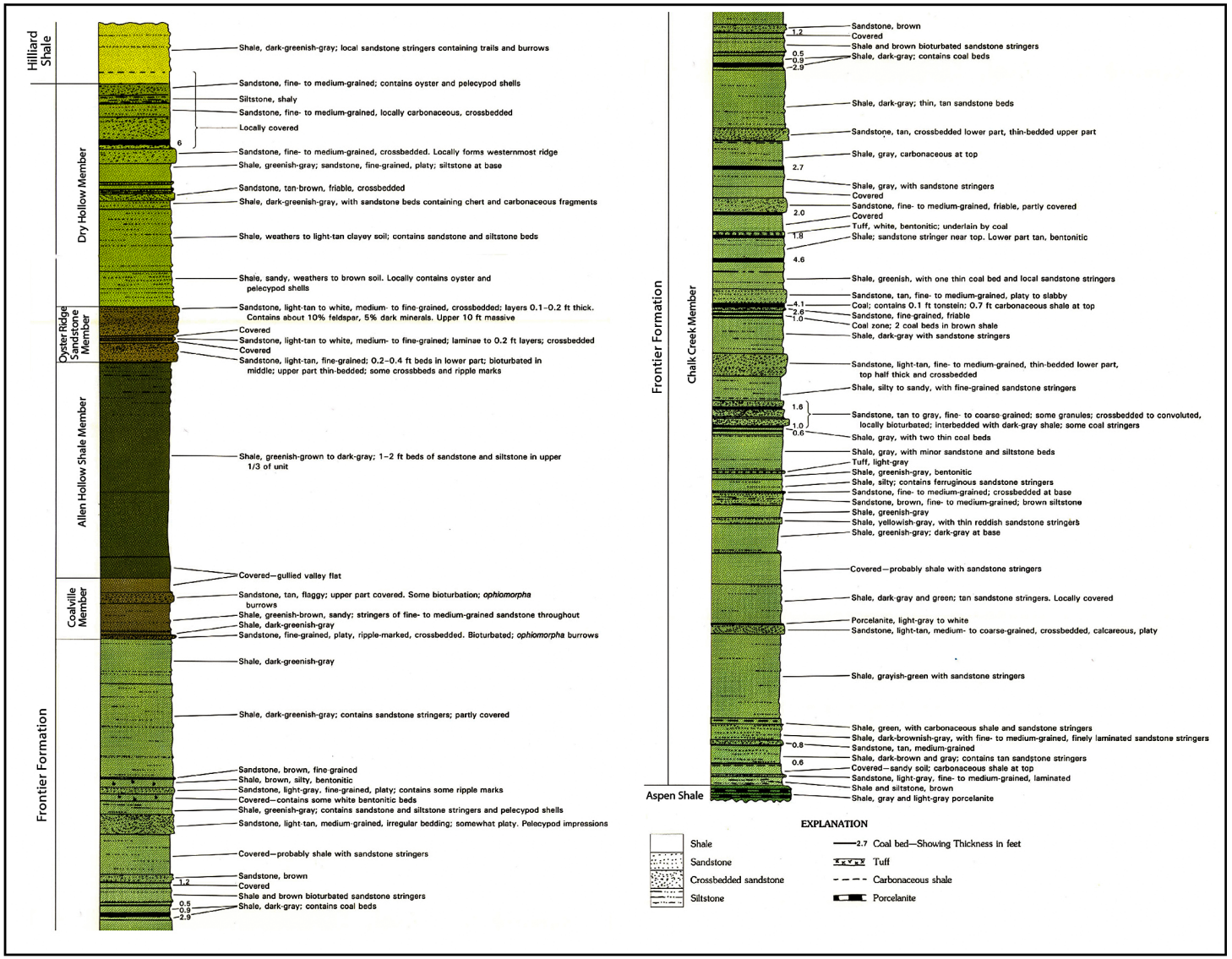



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**Figure 2.6-91 Summary of RQD Values**

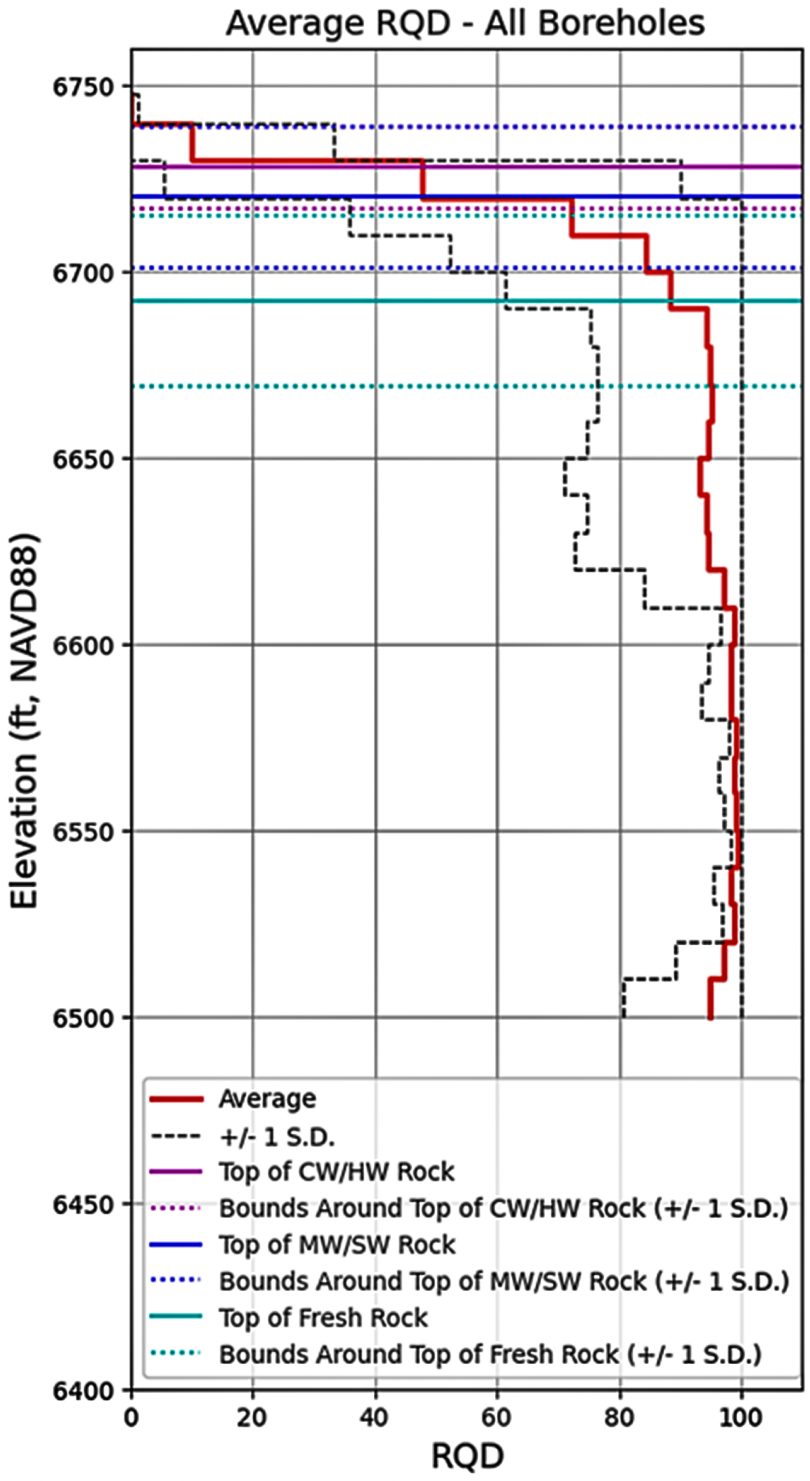



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**Figure 2.6-92 Summary of Rock Core Recovery**

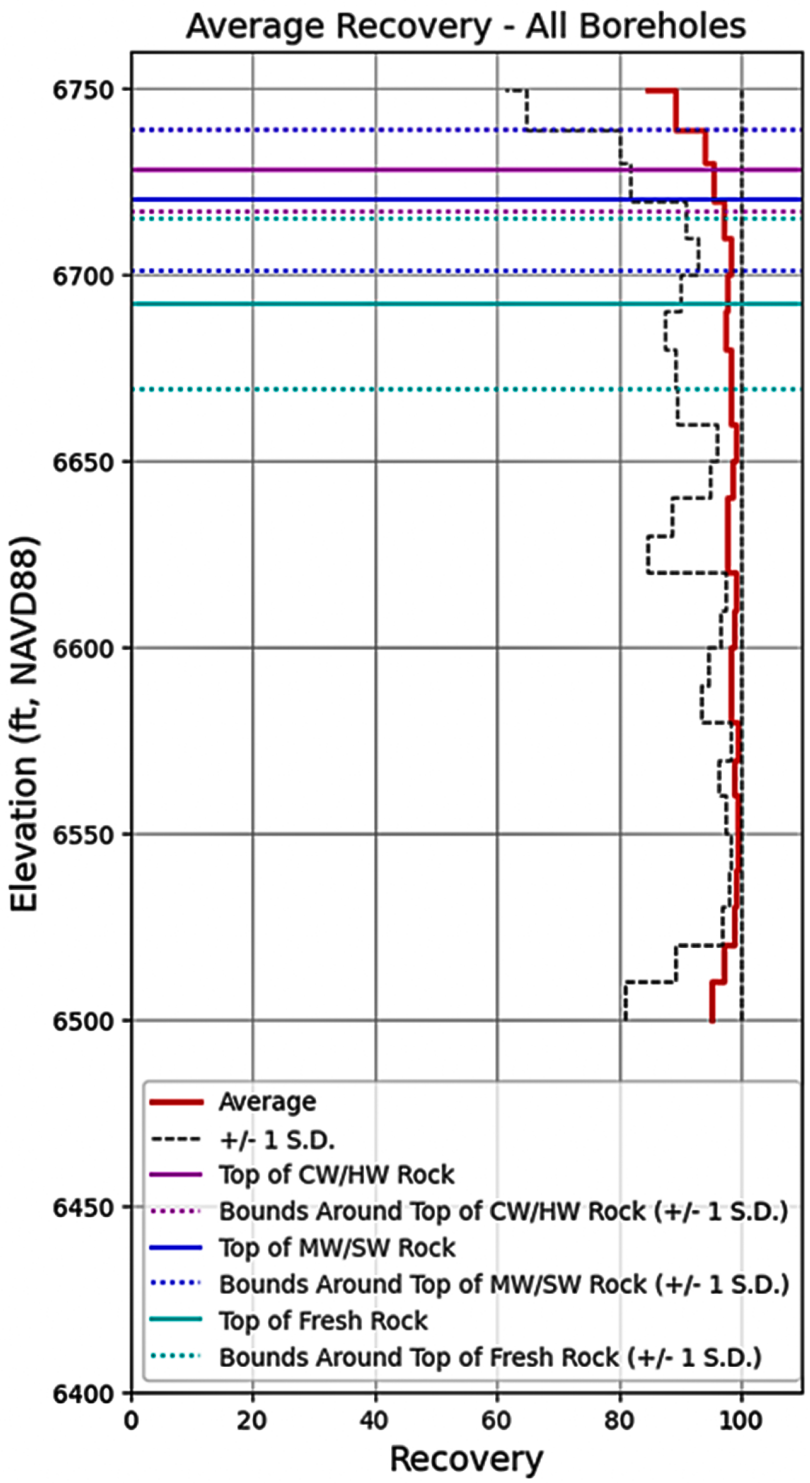



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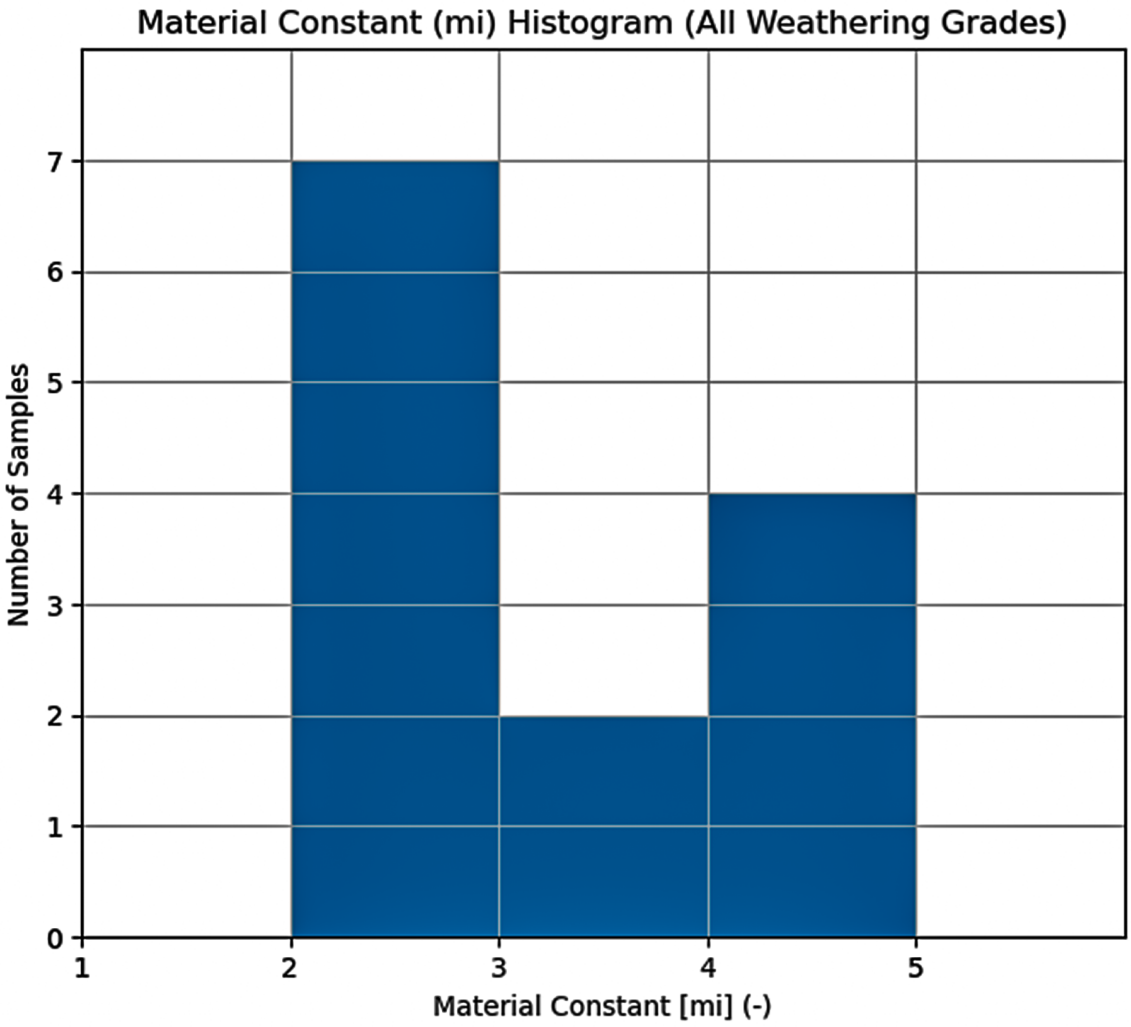
**Figure 2.6-93 Summary of Hoek-Brown Material Constant**




| | |
|---|---|
| count | 13.00 |
| mean | 3.29 |
| std | 1.04 |
| min | 2.05 |
| 25% | 2.52 |
| 50% | 2.83 |
| 75% | 4.15 |
| max | 4.87 |


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**Figure 2.6-94 Summary Joint Roughness and Joint Alteration**

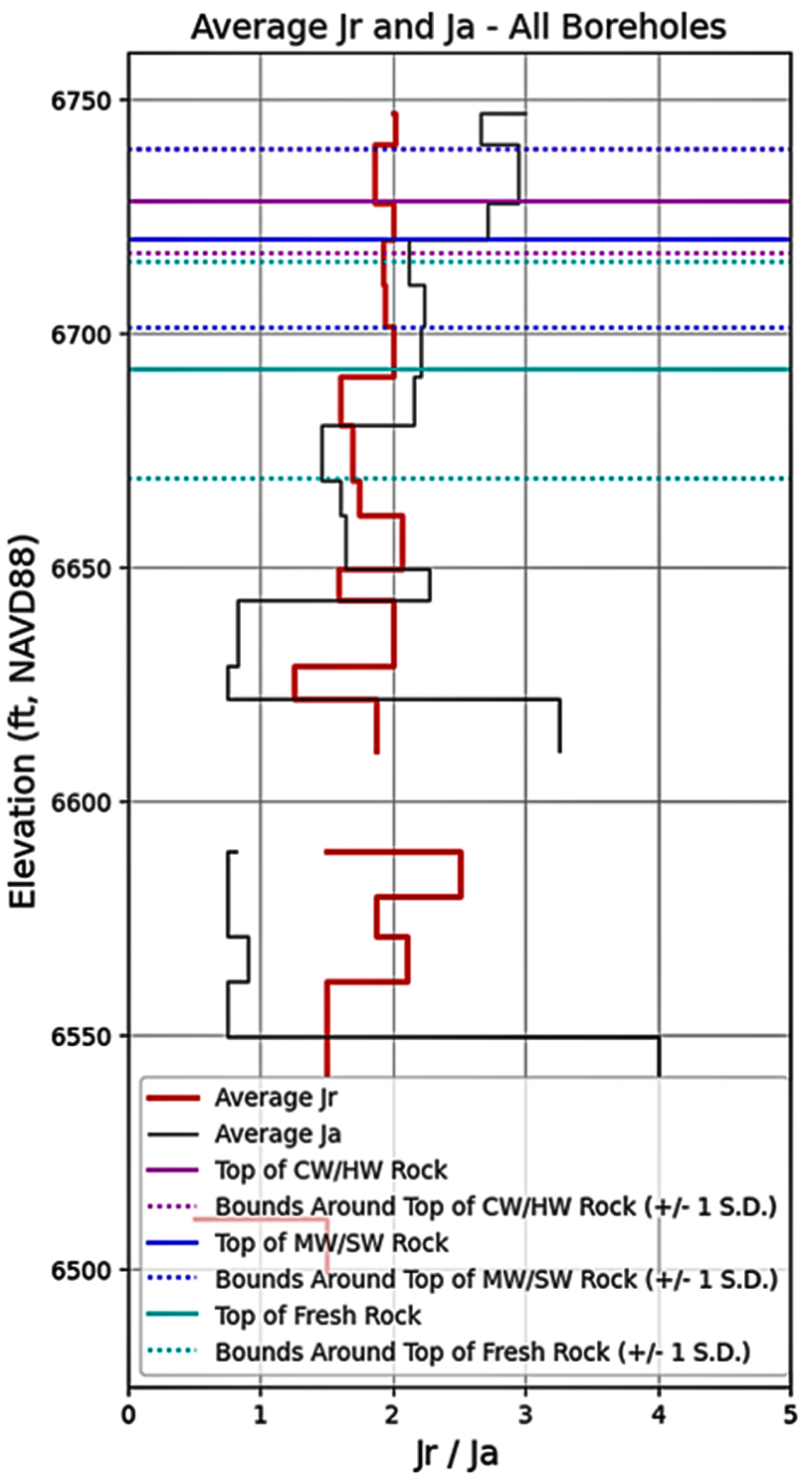




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**Figure 2.6-95 P-S and DH $V_s$ Profiles**

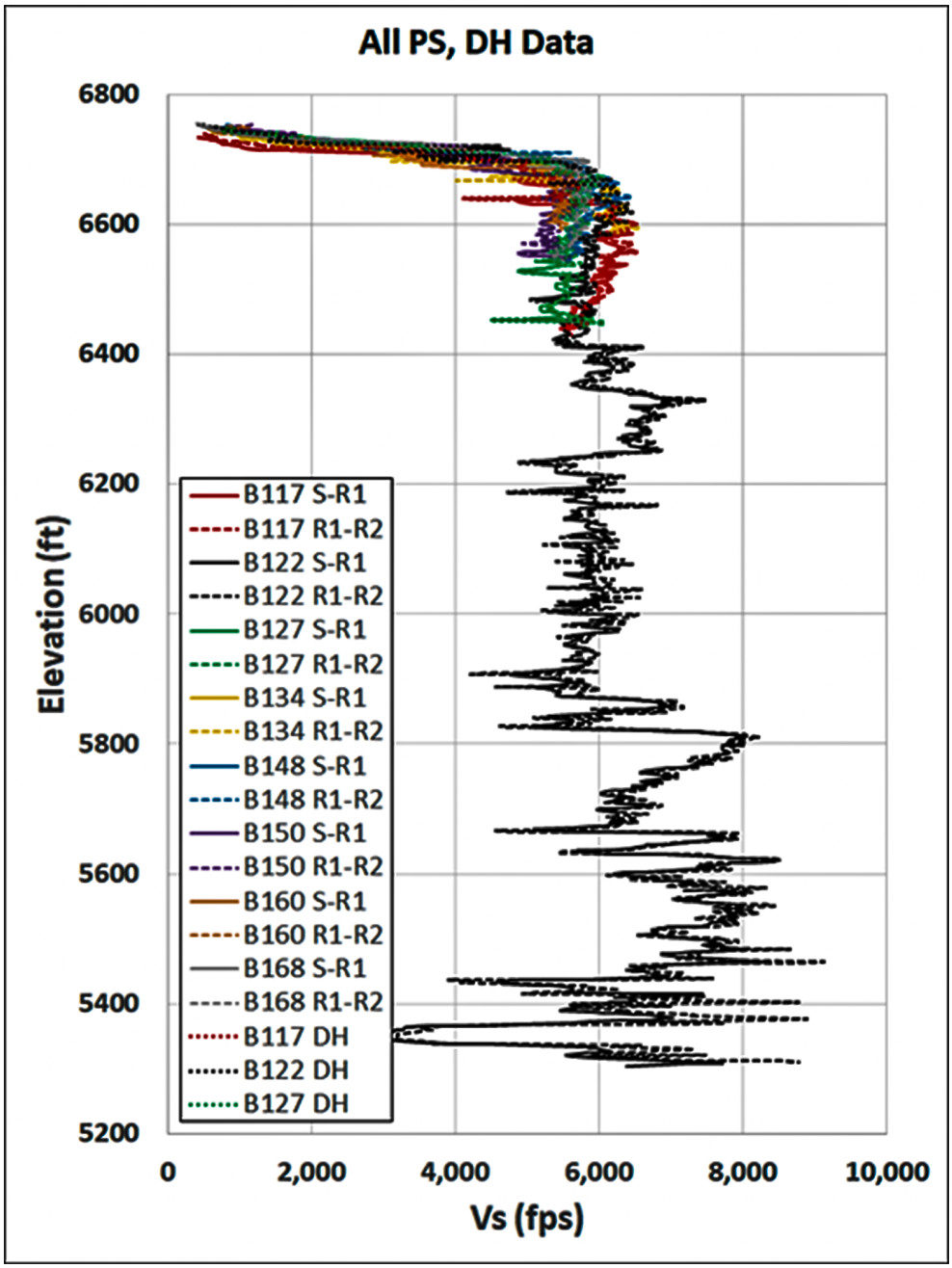




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**Figure 2.6-96 P-S and DH $V_p$ Profiles**

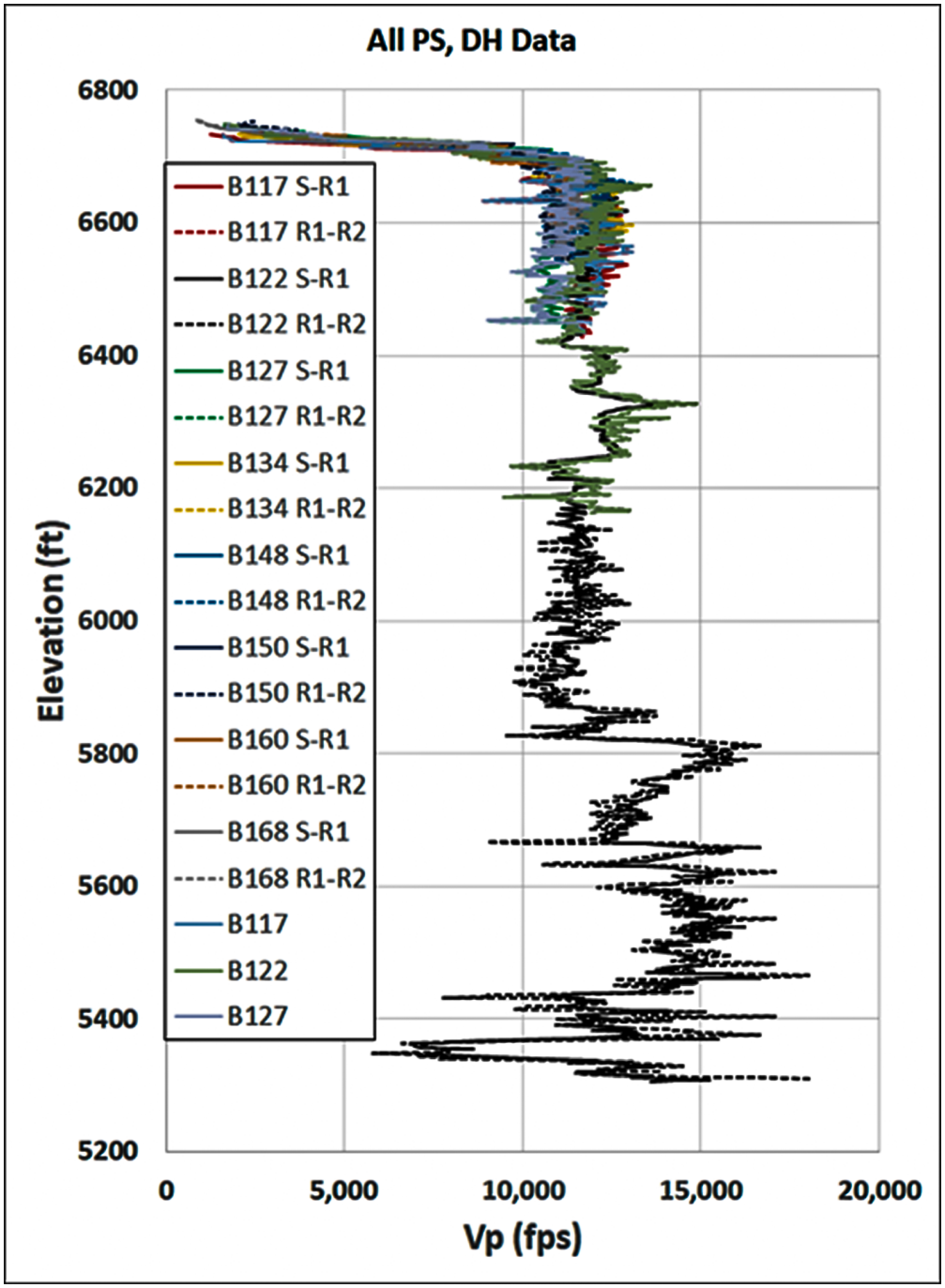




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**Figure 2.6-97 Short SASW Arrays**

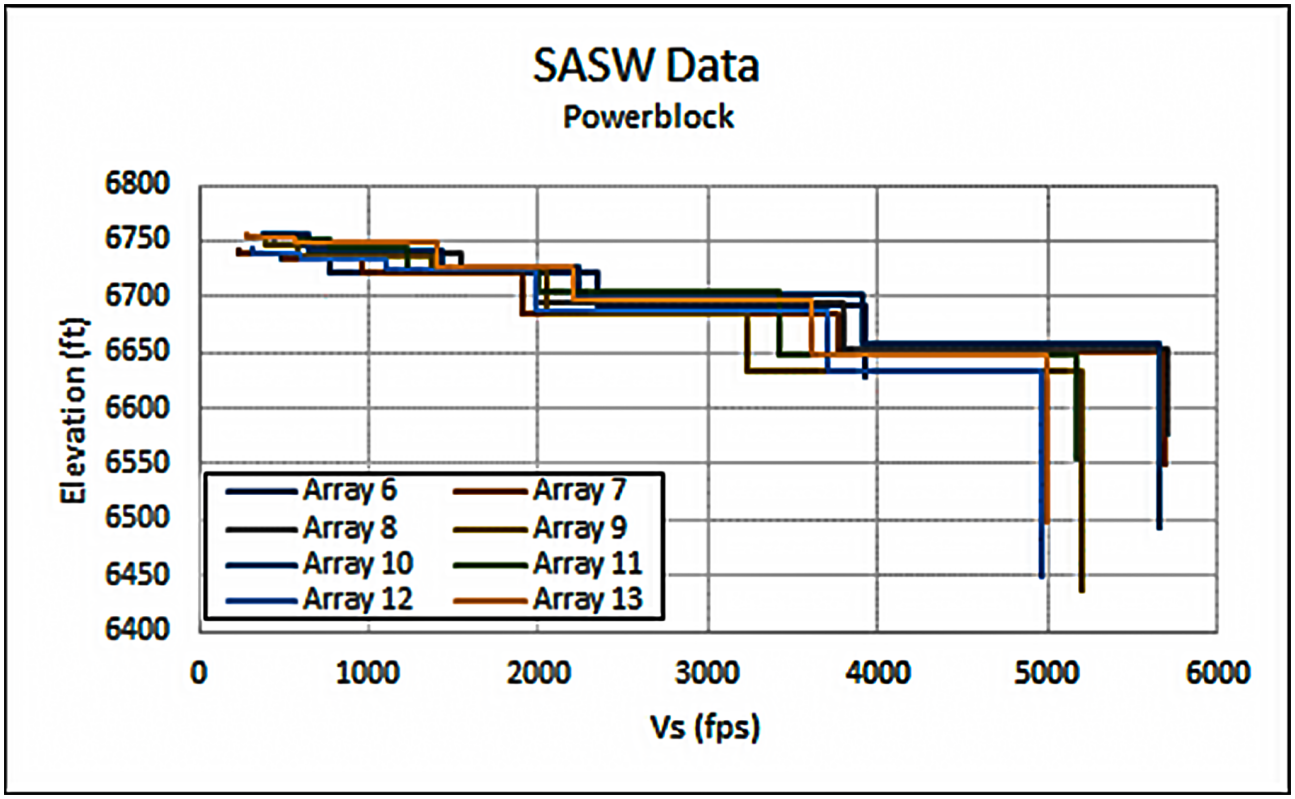



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**Figure 2.6-98 Long, Sitewide SASW Arrays**

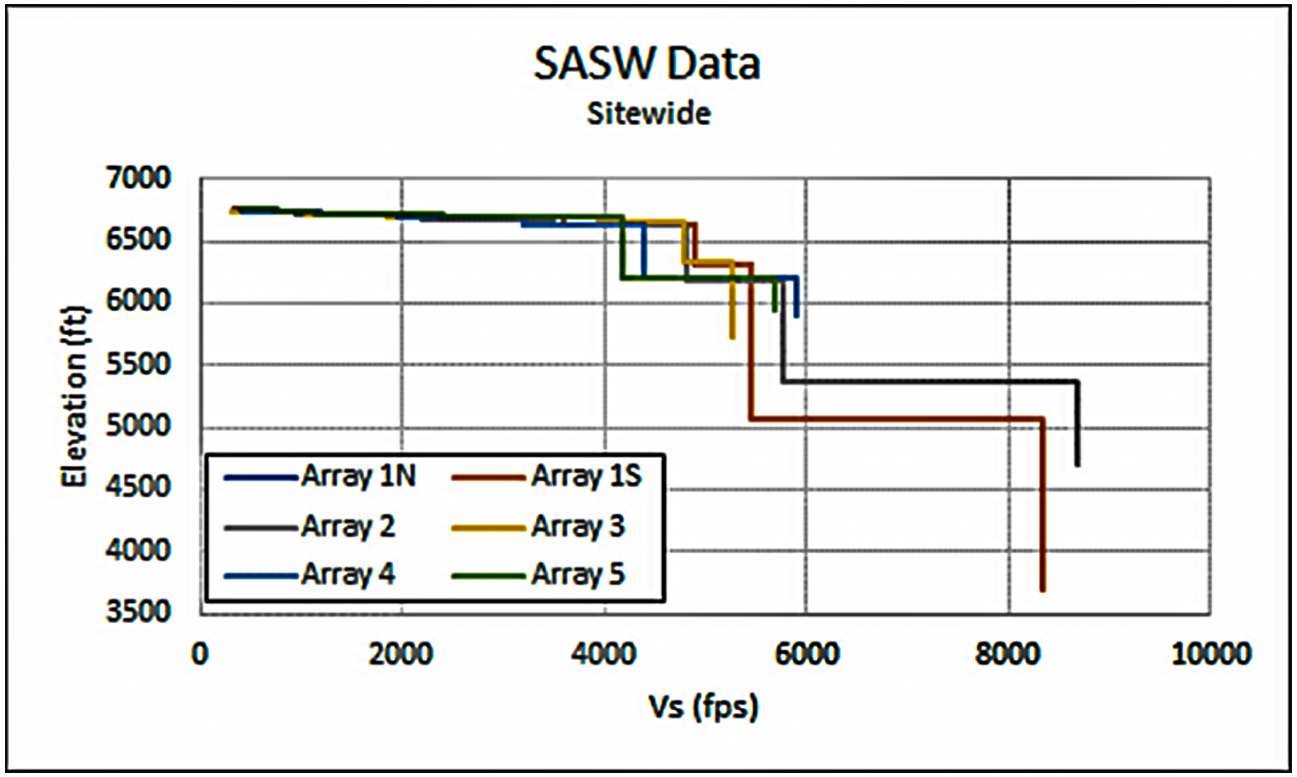




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Figure 2.6-99 Combined $V_s$ Profiles

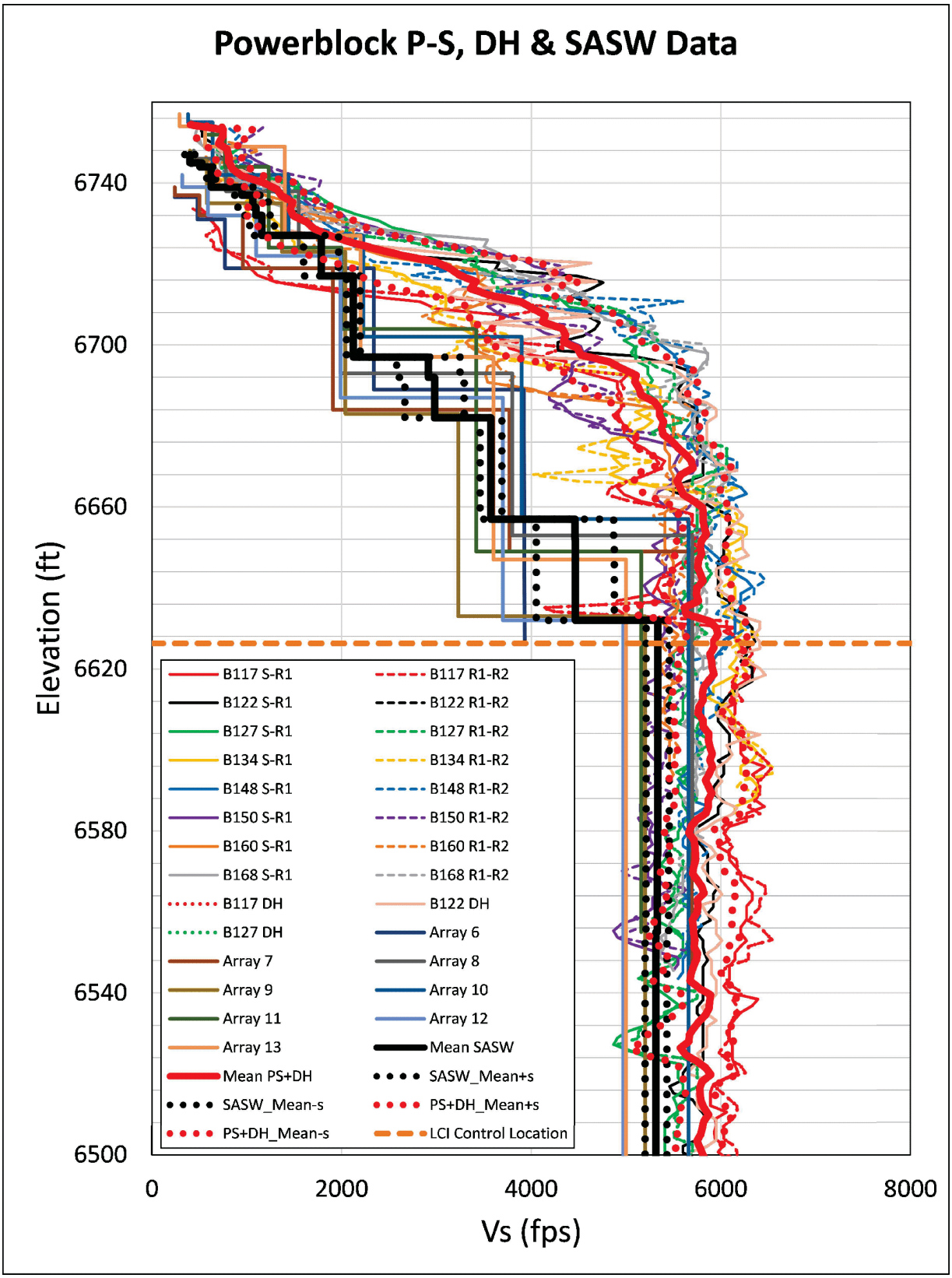



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Figure 2.6-100 Mean $V_s$ Profiles

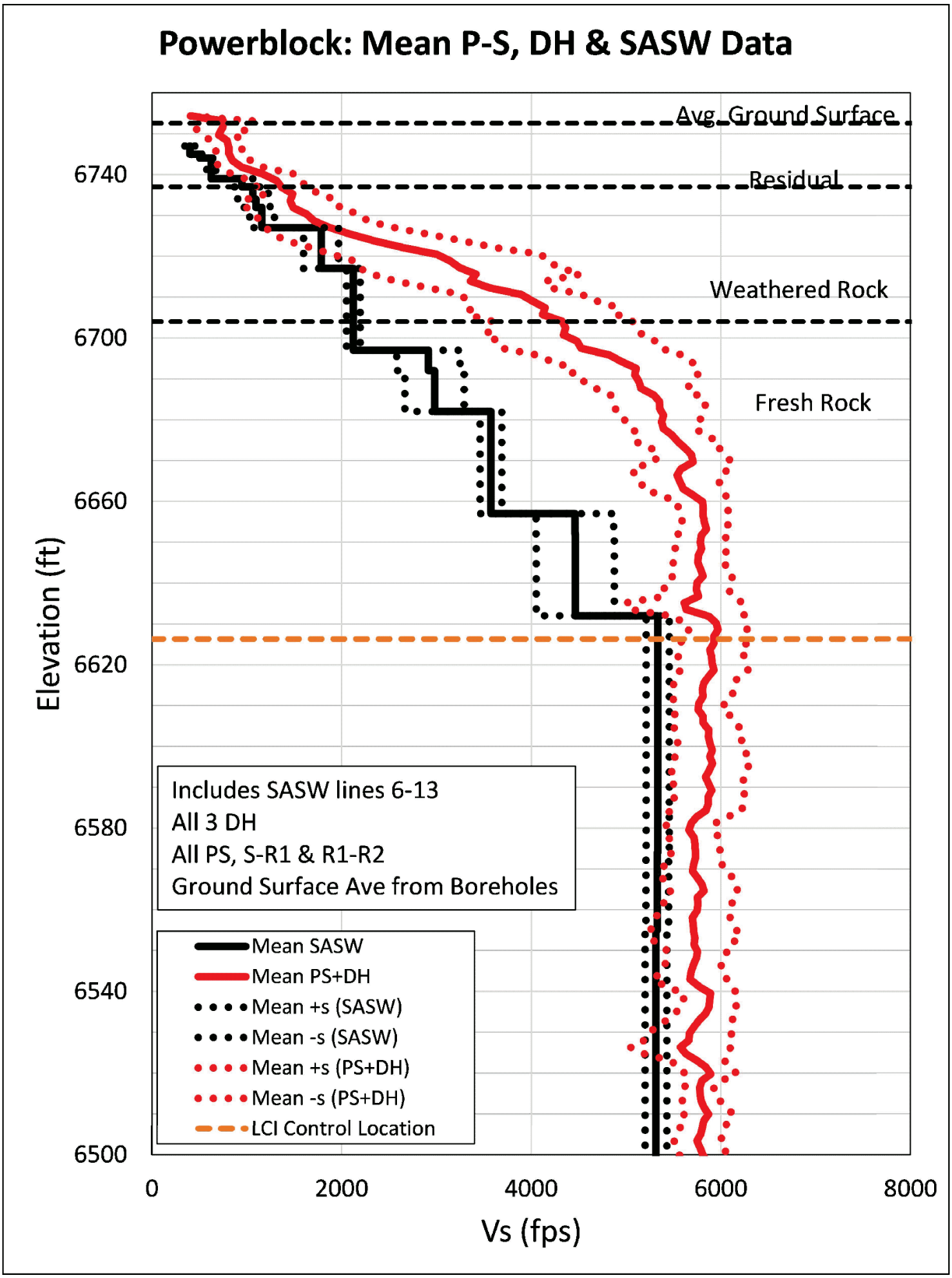



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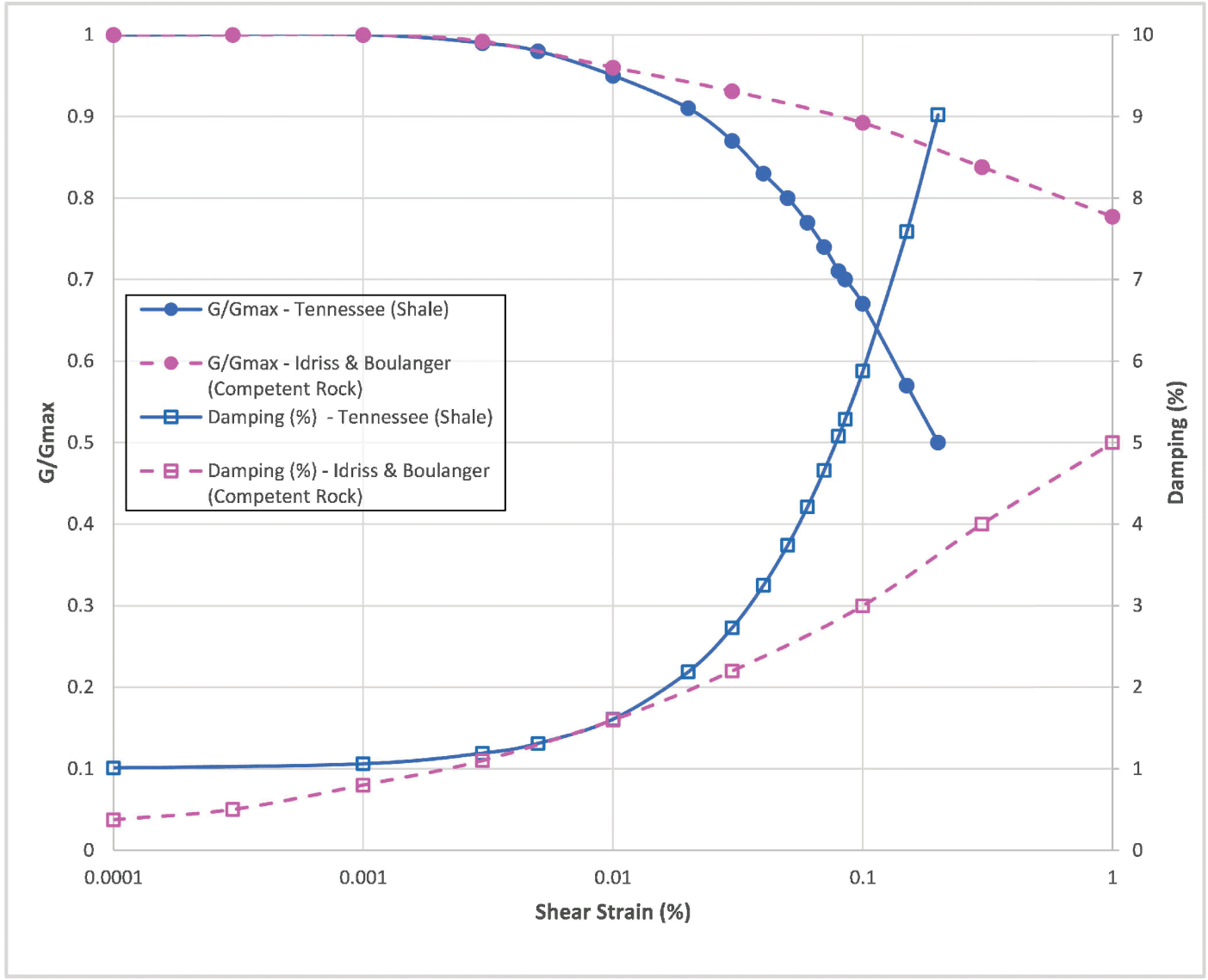
**Figure 2.6-101 Modulus Reduction and Damping – Fresh Rock**





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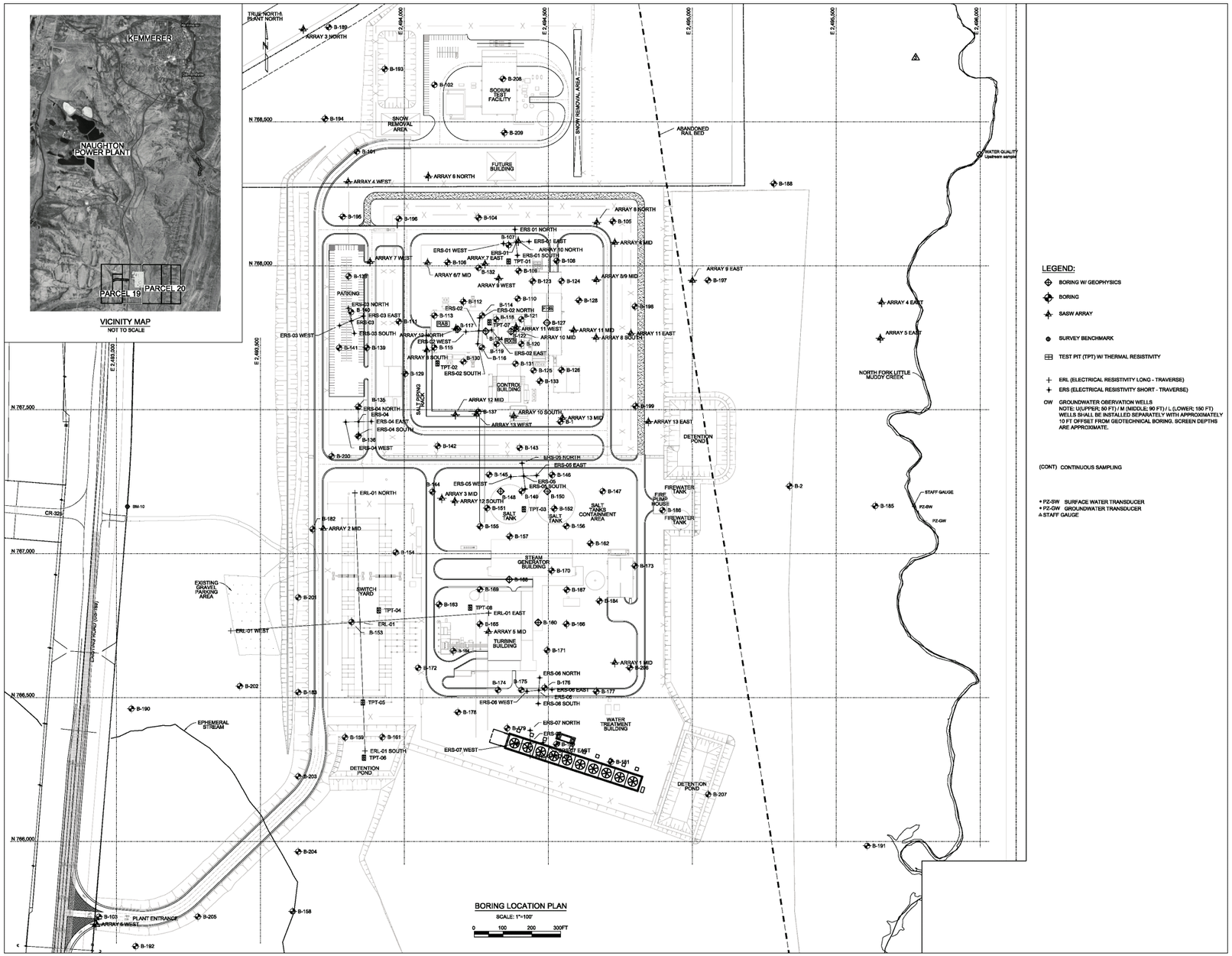
## Figure 2.6-102 Boring Location Plan





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# Figure 2.6-103 Boring Location Plan Exploration Schedule

**BORINGS**

| Feature ID | Northing (US survey feet) | Easting (US survey feet) | Ground Surface Elevation (ft) | Drilled Depth (ft) | Notes |
|---|---|---|---|---|---|
| B-1 | 767458.4 | 2494542.6 | 6757.1 | 100.5 | Exploratory boring, NST |
| B-2 | 767234.9 | 2495338.3 | 6741.3 | 100.4 | Exploratory boring, NST |
| B-101 | 768397.8 | 2493835.3 | 6740.1 | 25.3 | |
| B-102 | 768628.9 | 2494104.7 | 6752.0 | 50.1 | |
| B-103 | 765738.0 | 2492939.3 | 6733.0 | 23.6 | |
| B-104 | 768167.2 | 2494257.8 | 6752.9 | 50.3 | |
| B-105 | 768154.1 | 2494724.4 | 6748.1 | 25.1 | |
| B-106 | 768012.8 | 2494150.7 | 6742.6 | 100.5 | OW-10A-U, OW-106-M |
| B-107 | 768072.2 | 2494362.4 | 6736.0 | 100.6 | CONT |
| B-108 | 768017.4 | 2494529.3 | 6757.0 | 200.5 | OW-10B-U, OW-108-M, CONT |
| B-109 | 767982.7 | 2494397.4 | 6755.3 | 100.4 | OW-10A-U, OW-109-M, B-195A |
| B-110 | 767886.4 | 2494394.0 | 6736.4 | 101.3 | |
| B-111 | 767806.1 | 2493980.9 | 6736.8 | 100.0 | B-111A |
| B-112 | 767675.4 | 2494205.0 | 6742.8 | 200.7 | |
| B-113 | 767827.2 | 2494104.6 | 6738.0 | 201.8 | |
| B-114 | 767827.7 | 2494270.4 | 6745.5 | 12.0 | B-114A(200.3 ft) |
| B-115 | 767718.1 | 2494104.1 | 6736.8 | 200.4 | B-115A, B-115B, B-115-PT |
| B-116 | 767716.0 | 2494270.0 | 6746.2 | 200.2 | B-116-IT |
| B-117 | 767779.7 | 2494185.1 | 6740.1 | 320.4 | OW-11F-U, OW-117-M, OW-117-L, B-117A, B-117B,3-117-CB, CONT |
| B-118 | 767813.4 | 2494319.5 | 6751.0 | 300.3 | |
| B-119 | 767728.2 | 2494321.0 | 6751.8 | 301.9 | OW-113-U, OW-119-M |
| B-120 | 767728.9 | 2494406.3 | 6757.1 | 300.6 | |
| B-121 | 767814.4 | 2494406.5 | 6757.5 | 300.6 | OW-12J-U, OW-121-M(A) |
| B-122 | 767772.2 | 2494370.4 | 6756.3 | 1519.5 | OW-12C-U, OW-122-M, OW-122-L, B-122-CB, CONT |
| B-123 | 767946.5 | 2494446.0 | 6758.3 | 186.1 | B-123A |
| B-124 | 767947.1 | 2494546.1 | 6755.4 | 200.9 | B-124-IT |
| B-125 | 767636.4 | 2494449.1 | 6757.0 | 201.5 | B-125-IMT, B-125-PMT(A) |
| B-126 | 767638.0 | 2494546.0 | 6754.8 | 200.7 | B-126-IT |
| B-127 | 767802.1 | 2494495.5 | 6756.8 | 324.7 | OW-12F-U, OW-127U, B-12F-CB, CONT |
| B-128 | 767979.4 | 2494606.7 | 6758.3 | 101.6 | B-128A |
| B-129 | 767625.1 | 2494003.4 | 6735.6 | 50.3 | |
| B-130 | 767667.1 | 2494205.5 | 6742.5 | 200.1 | |
| B-131 | 767660.0 | 2494386.3 | 6756.6 | 102.1 | |
| B-132 | 767993.1 | 2494258.1 | 6747.1 | 51.6 | B-132-IT |
| B-133 | 767599.4 | 2494471.2 | 6756.3 | 101.8 | |
| B-134 | 767772.1 | 2494282.7 | 6746.7 | 170.0 | |
| B-135 | 767737.7 | 2493994.2 | 6738.7 | 19.3 | B-135A(102.1 FT) |
| B-136 | 767408.4 | 2493840.2 | 6737.5 | 100.4 | OW-13A-U, OW-136-M |
| B-137 | 767493.1 | 2494256.7 | 6747.8 | 101.0 | OW-13F-U, OW-137-M |
| B-138 | 767963.7 | 2493806.2 | 6740.8 | 50.0 | B-138-IMT |
| B-139 | 767715.4 | 2493869.9 | 6739.0 | 50.0 | B-139-IT |
| B-140 | 767839.1 | 2493815.7 | 6739.8 | 101.0 | OW-14B-U, OW-140M, CONT |
| B-141 | 767715.2 | 2493773.9 | 6741.9 | 25.0 | |
| B-142 | 767374.1 | 2494116.0 | 6739.0 | 50.3 | B-142A B-142-PT |
| B-143 | 767367.1 | 2494400.7 | 6758.2 | 50.7 | |
| B-144 | 767215.1 | 2494097.5 | 6738.1 | 50.0 | B-144A B-144-PT |
| B-145 | 767274.7 | 2494294.8 | 6753.8 | 100.3 | |
| B-146 | 767273.7 | 2494513.6 | 6757.2 | 100.3 | |
| B-147 | 767217.4 | 2494689.1 | 6755.3 | 50.0 | |
| B-148 | 767216.7 | 2494344.6 | 6750.0 | 239.9 | |
| B-149 | 767217.4 | 2494407.9 | 6757.2 | 100.3 | OW-14H-U, OW-149-M |
| B-150 | 767216.7 | 2494496.4 | 6757.0 | 220.0 | B-150-IT, B-150-PMT, B-156-PMT(A) |
| B-151 | 767158.4 | 2494287.8 | 6750.7 | 100.3 | |
| B-152 | 767156.4 | 2494520.9 | 6756.3 | 100.7 | |
| B-153 | 766762.4 | 2493816.9 | 6732.3 | 111.5 | OW-15I-U, OW-153-M, OW-153-L, B-153A, B-153B |
| B-154 | 767030.1 | 2493920.6 | 6737.4 | 23.8 | B-154A(50.0 FT) |
| B-155 | 767095.1 | 2494263.2 | 6751.5 | 100.5 | |
| B-156 | 767096.4 | 2494562.9 | 6755.4 | 104.0 | |
| B-157 | 767060.1 | 2494366.1 | 6756.0 | 101.5 | B-157-IT |
| B-158 | 765757.1 | 2493612.4 | 6728.1 | 25.0 | B-158A |
| B-159 | 766361.1 | 2493794.4 | 6731.8 | 100.0 | B-159A B & C |
| B-160 | 766760.1 | 2494465.0 | 6753.0 | 170.0 | B-160-IT |
| B-161 | 766960.7 | 2495923.6 | 6731.3 | 101.0 | B-161A |
| B-162 | 767035.1 | 2494646.6 | 6754.3 | 50.0 | |
| B-163 | 766823.1 | 2494120.8 | 6740.0 | 50.5 | |
| B-164 | 766662.1 | 2494170.2 | 6746.0 | 100.6 | OW-16I-U, OW-164-M |
| B-165 | 766755.1 | 2494262.6 | 6753.2 | 150.6 | |
| B-166 | 766755.1 | 2494563.1 | 6752.8 | 152.0 | |
| B-167 | 766874.1 | 2494564.4 | 6753.7 | 151.5 | OW-16T-U, OW-167-M, OW-167-L |
| B-168 | 766910.1 | 2494364.6 | 6756.0 | 220.4 | OW-16I-U, OW-168-M(A), CONT |
| B-169 | 766875.1 | 2494263.0 | 6753.9 | 120.0 | B-169-IMT, B-169-PMT(A) |
| B-170 | 766941.1 | 2494511.8 | 6754.7 | 100.3 | B-170-IMT |
| B-171 | 766664.1 | 2494496.1 | 6752.1 | 102.0 | B-171-IT |
| B-172 | 766603.1 | 2494048.2 | 6737.2 | 26.3 | |
| B-173 | 766957.7 | 2494801.1 | 6751.0 | 26.5 | |
| B-174 | 766634.1 | 2494319.7 | 6751.1 | 100.6 | |
| B-175 | 766526.1 | 2494406.4 | 6752.6 | 100.9 | |
| B-176 | 766534.1 | 2494487.4 | 6751.7 | 50.4 | B-176A |
| B-177 | 766521.1 | 2494667.0 | 6748.1 | 51.5 | B-177-IT |
| B-178 | 766449.1 | 2494186.2 | 6740.3 | 50.6 | |
| B-179 | 766393.1 | 2494357.5 | 6752.2 | 152.3 | OW-17I-U, OW-179-M |
| B-180 | 766338.1 | 2494528.1 | 6751.1 | 50.7 | |
| B-181 | 766276.7 | 2494717.9 | 6746.3 | 102.2 | B-181-IT |
| B-182 | 767085.4 | 2493681.5 | 6737.4 | 50.3 | B-182-IT |
| B-183 | 766518.4 | 2493631.9 | 6737.8 | 24.3 | |
| B-184 | 766835.4 | 2494678.1 | 6753.1 | 51.7 | |
| B-185 | 767166.4 | 2495635.7 | 6740.9 | 101.5 | OW-18I-U, OW-185-M |
| B-186 | 767151.1 | 2494896.7 | 6748.5 | 102.2 | OW-18I-U, OW-186-M |
| B-187 | 767912.1 | 2492457.1 | 6732.7 | 100.2 | OW-18T-U, OW-187-M |
| B-188 | 768285.1 | 2495283.5 | 6745.0 | 100.4 | OW-18I-U, OW-188-M |
| B-189 | 768829.1 | 2493737.4 | 6745.9 | 100.8 | OW-18I-U, OW-189-M |
| B-190 | 766461.1 | 2493052.2 | 6730.1 | 100.0 | OW-19I-U, OW-190-M |
| B-191 | 765985.1 | 2495607.8 | 6737.6 | 100.1 | OW-19J-U, OW-191-M |
| B-192 | 765636.4 | 2493066.8 | 6736.3 | 25.0 | |
| B-193 | 768683.4 | 2493932.2 | 6747.3 | 15.0 | |
| B-194 | 768311.1 | 2493724.0 | 6741.3 | 10.1 | |
| B-195 | 768171.4 | 2493786.1 | 6741.1 | 15.2 | |
| B-196 | 768163.4 | 2493981.2 | 6738.7 | 19.9 | B-196A |
| B-197 | 767949.2 | 2495055.3 | 6741.5 | 20.0 | |
| B-198 | 767858.1 | 2494801.2 | 6744.1 | 15.0 | |
| B-199 | 767513.4 | 2494802.9 | 6748.7 | 15.0 | |
| B-200 | 767338.4 | 2493747.7 | 6740.3 | 14.9 | |
| B-201 | 766948.1 | 2493631.8 | 6740.3 | 17.4 | |
| B-202 | 766540.4 | 2493428.7 | 6731.9 | 17.2 | |
| B-203 | 766226.1 | 2493631.7 | 6731.1 | 30.4 | |
| B-204 | 765965.4 | 2493631.7 | 6730.1 | 33.8 | B-204A |
| B-205 | 765739.1 | 2493283.1 | 6730.4 | 30.3 | |
| B-206 | 766603.4 | 2494783.5 | 6747.5 | 15.2 | |
| B-207 | 766163.1 | 2495955.6 | 6736.0 | 15.0 | |
| B-208 | 768649.4 | 2494342.6 | 6764.0 | 104.8 | |
| B-209 | 768462.1 | 2494347.8 | 6765.4 | 100.0 | |

Note :
1. Coordinates for offset borings and obse vations wells are found in Subsurface Investigation Data Report for Natrium Demonstration Projec, Rev. 2, by WSP USA Environmental & Instucture Inc.
2. Cont = continuous sampling

**TEST PITS**

| Feature IE | Northing (US survey foot) | Easting (US survey foot) | Ground Surface Elevation (ft) | Exploration Depth (ft) | Notes |
|---|---|---|---|---|---|
| TPT-01 | 768015.4 | 2494362.4 | 6753.1 | 3.8 | TPT-01A |
| TPT-02 | 767661.7 | 2494115.6 | 6737.0 | 3.8 | TPT-02A |
| TPT-03 | 767154.6 | 2494415.6 | 6756.7 | 3.8 | TPT-03A |
| TPT-04 | 766900.3 | 2493931.0 | 6731.8 | 3.0 | TPT-04A |
| TPT-05 | 766483.2 | 2493855.9 | 6731.6 | 3.8 | TPT-05A |
| TPT-06 | 766291.1 | 2493860.2 | 6731.0 | 3.8 | TPT-06A |
| TPT-07 | 767694.3 | 2494296.2 | 6747.9 | 3.8 | TPT-07A |
| TPT-08 | 766812.4 | 2494228.6 | 6749.0 | 3.8 | TPT-08A |

Note: Coordinates for test pits are found in Subsuface Investigation Data Repor: for Natrium Demonstration Project, Rev. 2, by WSP USA Environmental & Instructure Inc.

**REFERENCE STATIONS**

| Feature ID | Northing (US survey foot) | Easting (US survey foot) | Reference Elevation (ft) | Notes |
|---|---|---|---|---|
| BM-10 | 767163.8 | 2493038.3 | 6754.9 | WYDOT Highway Benchmark Brass Cap in Concrete |
| BM-100 | 764920.7 | 2493349.5 | 6751.6 | Aluminum Pipe and Cap |
| BM-101 | 769115.9 | 2494165.5 | 6766.6 | Aluminum Pipe and Cap |
| DOWNSTREAM STA. | 764543.9 | 2495609.7 | 6729.6 | Surface water sampling |
| UPSTREAM STA. | 768386.8 | 2495997.9 | 6736.3 | Surface water sampling |
| PZ-SW | 767163.1 | 2495771 | 6738.7 | Reference mark |
| PZ-GW | 767161.4 | 2495772.4 | 6736.9 | Reference mark |
| STAFF GAUGE | 767166.8 | 2495769.6 | 6737.2 | Reference mark (15-ft) |

**ELECTRICAL RESISITY ARRAY POINTS**

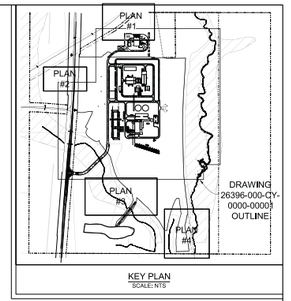
| Feature I0 | Northing (US survey foot) | Easting (US survey foot) | Ground Surface Elevation (ft) | Notes |
|---|---|---|---|---|
| ERL-01 CEVTER | 766763.9 | 2493817.0 | 5731.8 | |
| ERL-01 EAST | 766793.8 | 2494293.1 | 5753.8 | |
| ERL-01 NORTH | 767211.3 | 2493828.2 | 5735.5 | |
| ERL-01 SOUTH | 766314.5 | 2493863.6 | 5731.7 | |
| ERL-01 WIST | 766732.0 | 2493397.2 | 5741.0 | |
| ERS-01 CEVTER | 768081.6 | 2494388.8 | 5758.3 | |
| ERS-01 EAST | 768084.2 | 2494433.3 | 5759.4 | |
| ERS-01 NORTH | 768126.2 | 2494384.3 | 5757.9 | |
| ERS-01 SOUTH | 768036.5 | 2494392.9 | 5756.1 | |
| ERS-01 WIST | 768077.5 | 2494344.2 | 5754.7 | |
| ERS-02 CEVTER | 767774.1 | 2494258.6 | 5744.7 | |
| ERS-02 EAST | 767777.1 | 2494303.3 | 5749.0 | |
| ERS-02 NORTH | 767819.0 | 2494261.7 | 5744.7 | |
| ERS-02 SOUTH | 767729.2 | 2494255.2 | 5744.2 | |
| ERS-02 WIST | 767771.2 | 2494214.0 | 5741.5 | |
| ERS-03 CEVTER | 767808.8 | 2493816.2 | 5739.9 | |
| ERS-03 EAST | 767825.5 | 2493857.9 | 5738.5 | |
| ERS-03 NORTH | 767852.5 | 2493805.6 | 5740.1 | |
| ERS-03 SOUTH | 767765.1 | 2493826.4 | 5739.9 | |
| ERS-03 WIST | 767792.4 | 2493774.9 | 5741.8 | |
| ERS-04 CEVTER | 767458.4 | 2493841.0 | 5737.1 | |
| ERS-04 EAST | 767459.2 | 2493885.8 | 5735.6 | |
| ERS-04 NORTH | 767503.4 | 2493840.0 | 5738.4 | |
| ERS-04 SOUTH | 767413.5 | 2493841.8 | 5737.0 | |
| ERS-04 WIST | 767457.5 | 2493796.2 | 5738.7 | |
| ERS-05 CEVTER | 767269.7 | 2494414.9 | 5757.9 | |
| ERS-05 EAST | 767271.6 | 2494459.5 | 5757.5 | |
| ERS-05 NORTH | 767314.3 | 2494409.0 | 5758.4 | |
| ERS-05 SOUTH | 767225.0 | 2494419.9 | 5757.2 | |
| ERS-05 WIST | 767267.3 | 2494369.8 | 5757.2 | |
| ERS-06 CEVTER | 766525.4 | 2494667.4 | 5751.6 | |
| ERS-06 EAST | 766527.5 | 2494512.7 | 5751.5 | |
| ERS-06 NORTH | 766569.3 | 2494470.3 | 5751.5 | |
| ERS-06 SOUTH | 766479.3 | 2494465.3 | 5752.0 | |
| ERS-06 WIST | 766522.1 | 2494426.0 | 5752.1 | |
| ERS-07 CEVTER | 766341.7 | 2494437.1 | 5752.4 | |
| ERS-07 EAST | 766343.5 | 2494482.0 | 5751.7 | |
| ERS-07 NORTH | 766386.2 | 2494437.3 | 5752.6 | |
| ERS-07 SOUTH | 766296.6 | 2494437.0 | 5752.3 | |
| ERS-07 WIST | 766339.8 | 2494392.6 | 5752.8 | |

**SASW ARRLY POINTS**

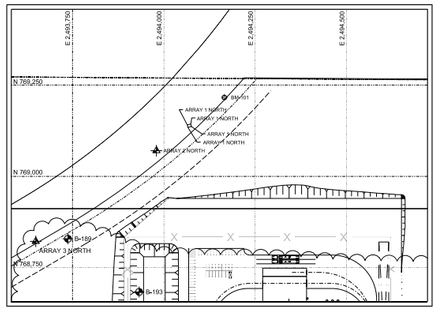
| Feature ID | Northing (US survey foot) | Easting (US survey foot) | Ground Surface Elevation (ft) | Notes |
|---|---|---|---|---|
| ARRAY 1 SOUTH | 764108.1 | 2491892.2 | 6735.1 | |
| ARRAY 1 MID | 766621.5 | 2493731.3 | 6749.6 | |
| ARRAY 1 NORTH | 769136.5 | 2494075.3 | 6760.5 | TOP OF CONCRETE PAD |
| ARRAY 1 NORTH | 769143.3 | 2494073.6 | 6760.5 | TOP OF CONCRETE PAD |
| ARRAY 1 NORTH | 769140.9 | 2494064.0 | 6760.5 | TOP OF CONCRETE PAD |
| ARRAY 1 NORTH | 769134.1 | 2494065.6 | 6760.5 | TOP OF CONCRETE PAD |
| ARRAY 2 NORTH | 769070.4 | 2493979.5 | 6754.7 | |
| ARRAY 2 SOUTH | 765104.8 | 2493457.7 | 6731.8 | |
| ARRAY 2 MID | 767087.4 | 2493718.9 | 6735.9 | |
| ARRAY 3 NORTH | 768822.2 | 2493649.4 | 6745.2 | |
| ARRAY 3 SOUTH | 765563.0 | 2493612.9 | 6743.7 | |
| ARRAY 3 MID | 767192.3 | 2493130.4 | 6740.7 | |
| ARRAY 4 EAST | 767873.0 | 2495657.9 | 6742.8 | |
| ARRAY 4 WEST | 768202.1 | 2495398.8 | 6737.2 | |
| ARRAY 4 MID | 768081.1 | 2493731.2 | 6748.8 | |
| ARRAY 5 WEST | 767713.0 | 2492929.3 | 6733.9 | |
| ARRAY 5 EAST | 767748.1 | 2495653.3 | 6742.3 | |
| ARRAY 5 MID | 766729.4 | 2494292.3 | 6754.2 | |
| ARRAY 6 SOUTH | 767706.8 | 2494078.4 | 6736.2 | |
| ARRAY 6 NORTH | 768310.9 | 2493083.4 | 6743.9 | |
| ARRAY 6/7 MID | 768010.9 | 2493080.7 | 6739.3 | |
| ARRAY 7 EAST | 768006.9 | 2493280.2 | 6748.1 | |
| ARRAY 7 EWEST | 768014.3 | 2493881.2 | 6739.3 | |
| ARRAY 8 SOUTH | 767750.9 | 2493666.2 | 6749.3 | |
| ARRAY 8 NORTH | 768150.7 | 2493668.8 | 6751.6 | |
| ARRAY 8/9 MID | 767951.0 | 2493667.3 | 6748.2 | |
| ARRAY 9 EAST | 767559.4 | 2493999.7 | 6753.2 | |
| ARRAY 9 WEST | 767956.3 | 2493277.9 | 6750.2 | |
| ARRAY 10 MID | 767777.8 | 2493888.6 | 6756.5 | |
| ARRAY 10 NORTH | 768085.7 | 2493895.8 | 6758.7 | |
| ARRAY 10 SOUTH | 767478.4 | 2493880.3 | 6756.2 | |
| ARRAY 11 WEST | 767787.8 | 2493888.3 | 6756.3 | |
| ARRAY 11 EAST | 767765.5 | 2493787.2 | 6744.3 | |
| ARRAY 11 MID | 767776.5 | 2493588.6 | 6753.7 | |
| ARRAY 12 SOUTH | 767182.5 | 2493175.3 | 6743.7 | |
| ARRAY 12 NORTH | 767781.9 | 2493179.4 | 6739.8 | |
| ARRAY 12 MID | 767482.4 | 2493177.6 | 6742.5 | |
| ARRAY 13 WEST | 767485.6 | 2493250.8 | 6747.3 | |
| ARRAY 13 EAST | 767458.7 | 2493847.9 | 6746.7 | |
| ARRAY 13 MID | 767472.3 | 2493549.2 | 6756.7 | |

NOTES:

1. WORK THIS DRAWING WITH THE BORING LOCATION PLAN ON DRAWING 26396-000-CY-0000-00001.
2. COORDINATES SHOWN ON THIS DRAWING ARE BASED ON THE WYOMING STATE PLANE COORDINATE SYSTEM, WY, WEST (4904) NAD 83. VERTICAL DATUM IS BASED ON NAVD 88. BOTH ARE EXPRESSED IN US SURVEY FEET.



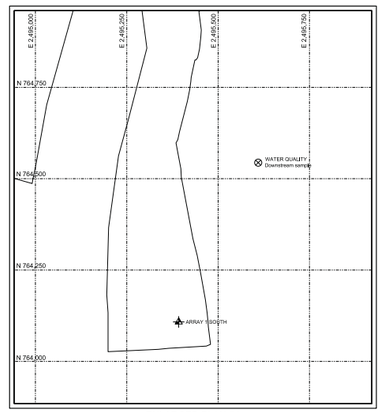

KEY PLAN
SCALE: NTS



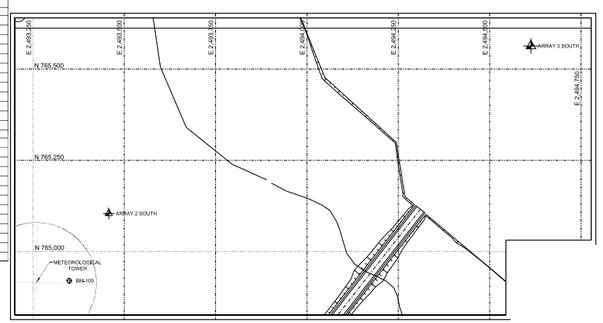

ENLARGED PLAN #1
SCALE: NTS




ENLARGED PLAN #2
SCALE: NTS




ENLARGED PLAN #4
SCALE: NTS




ENLARGED PLAN #3
SCALE: NTS


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**Figure 2.6-104 East/West Geologic Cross Section through RXB**

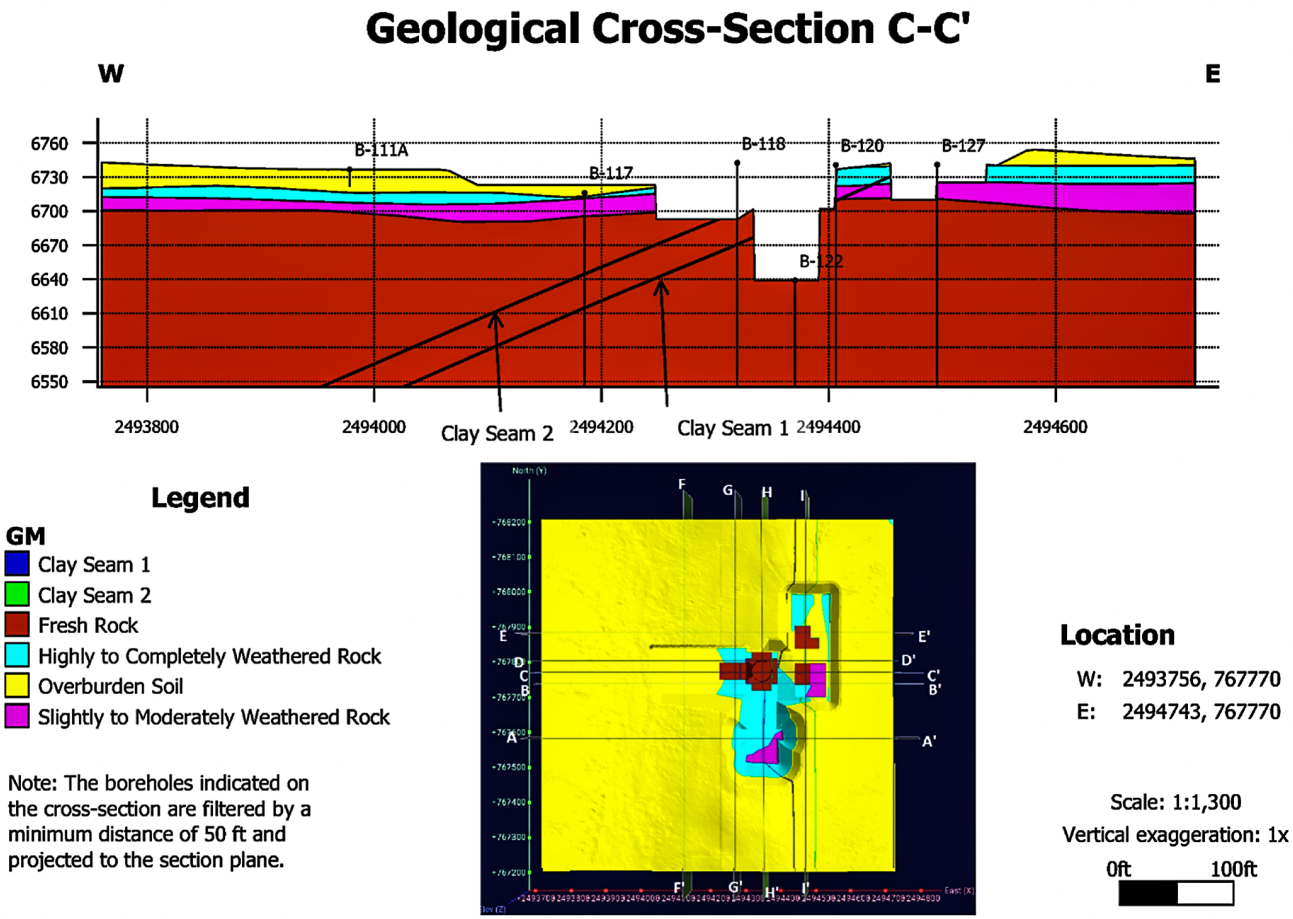



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**Figure 2.6-105 North/South Cross Geologic Cross Section – RXB/NCB**

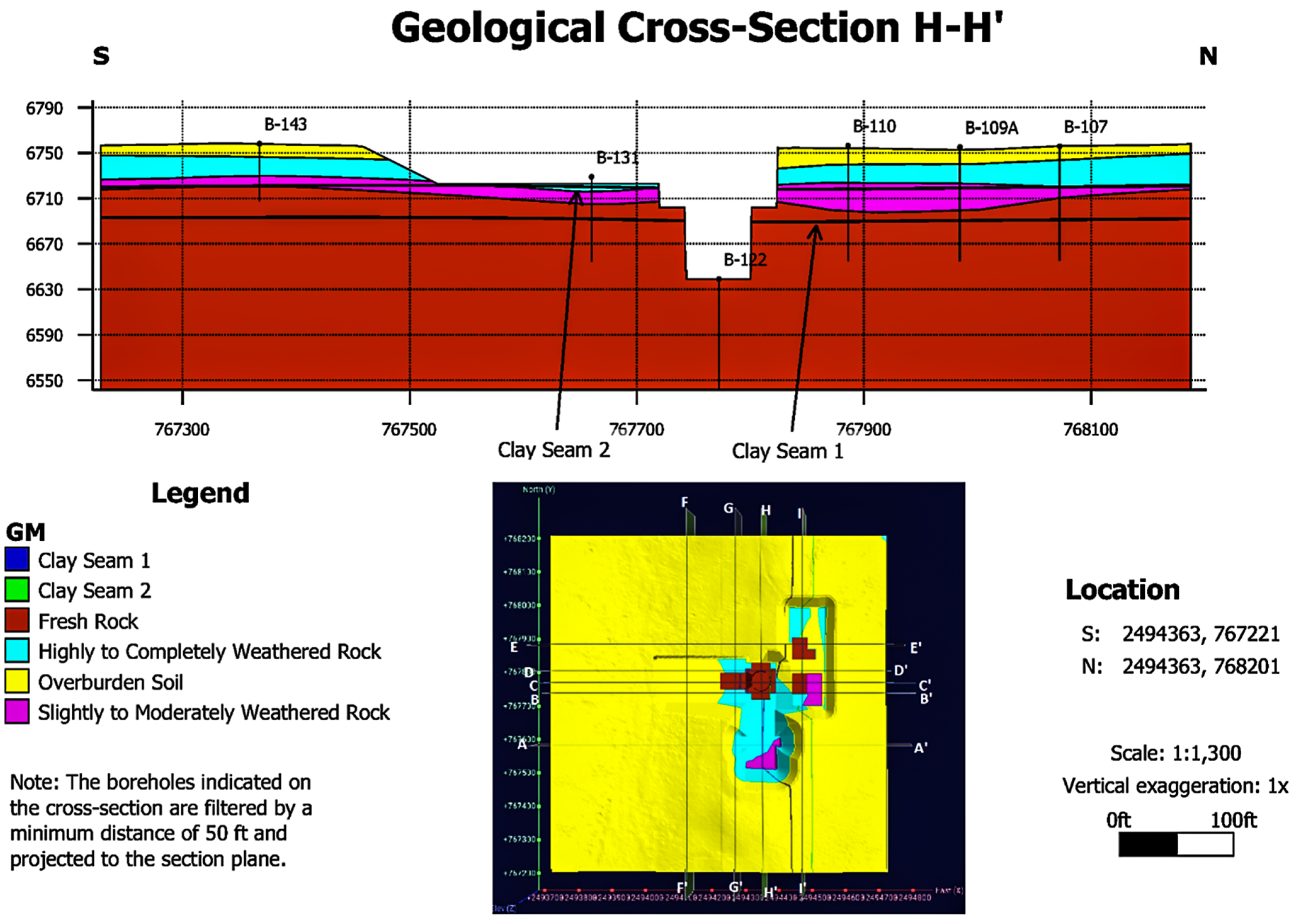



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**Figure 2.6-106 Post-Construction Groundwater Contour Map, Base Model**

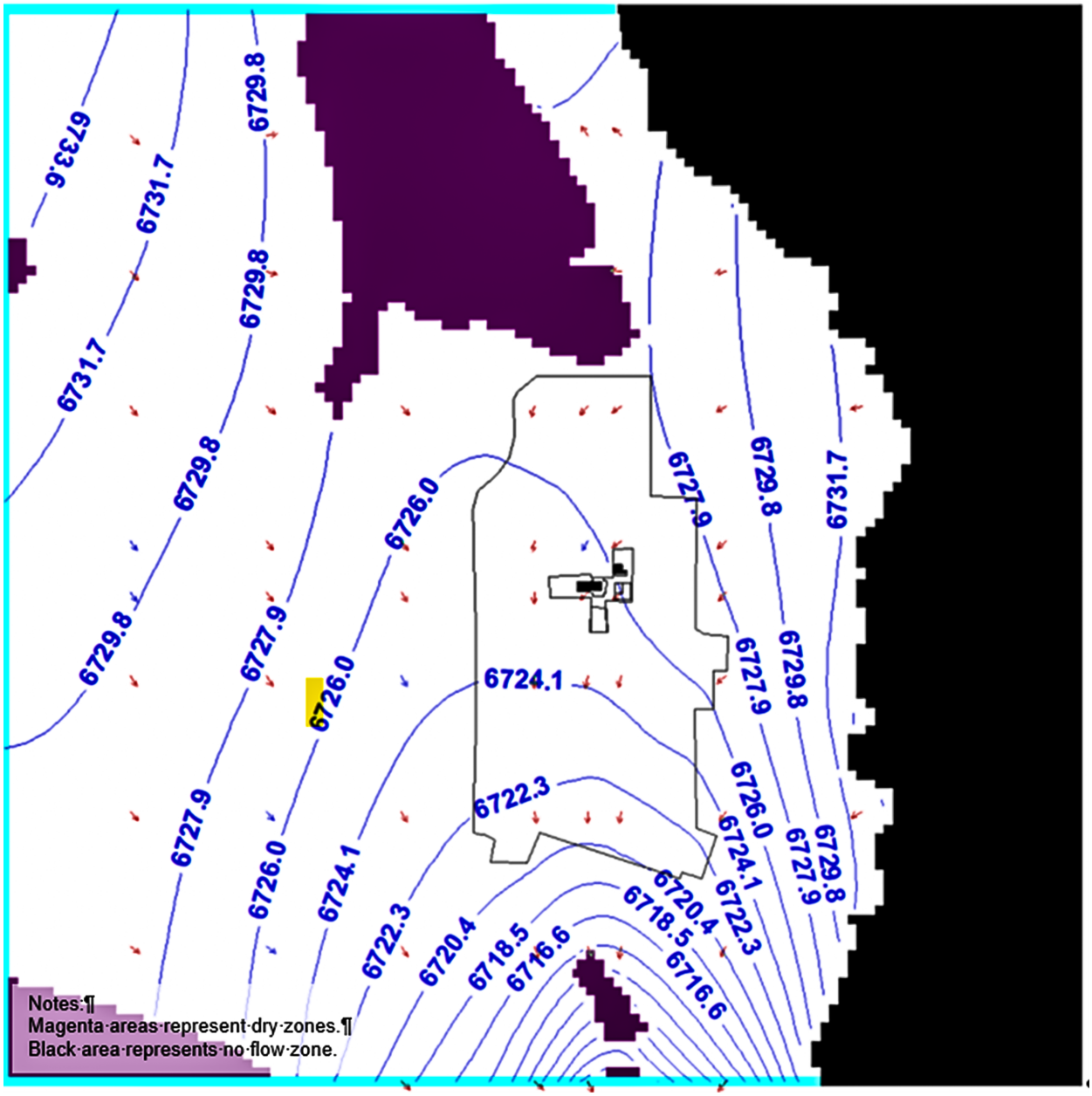



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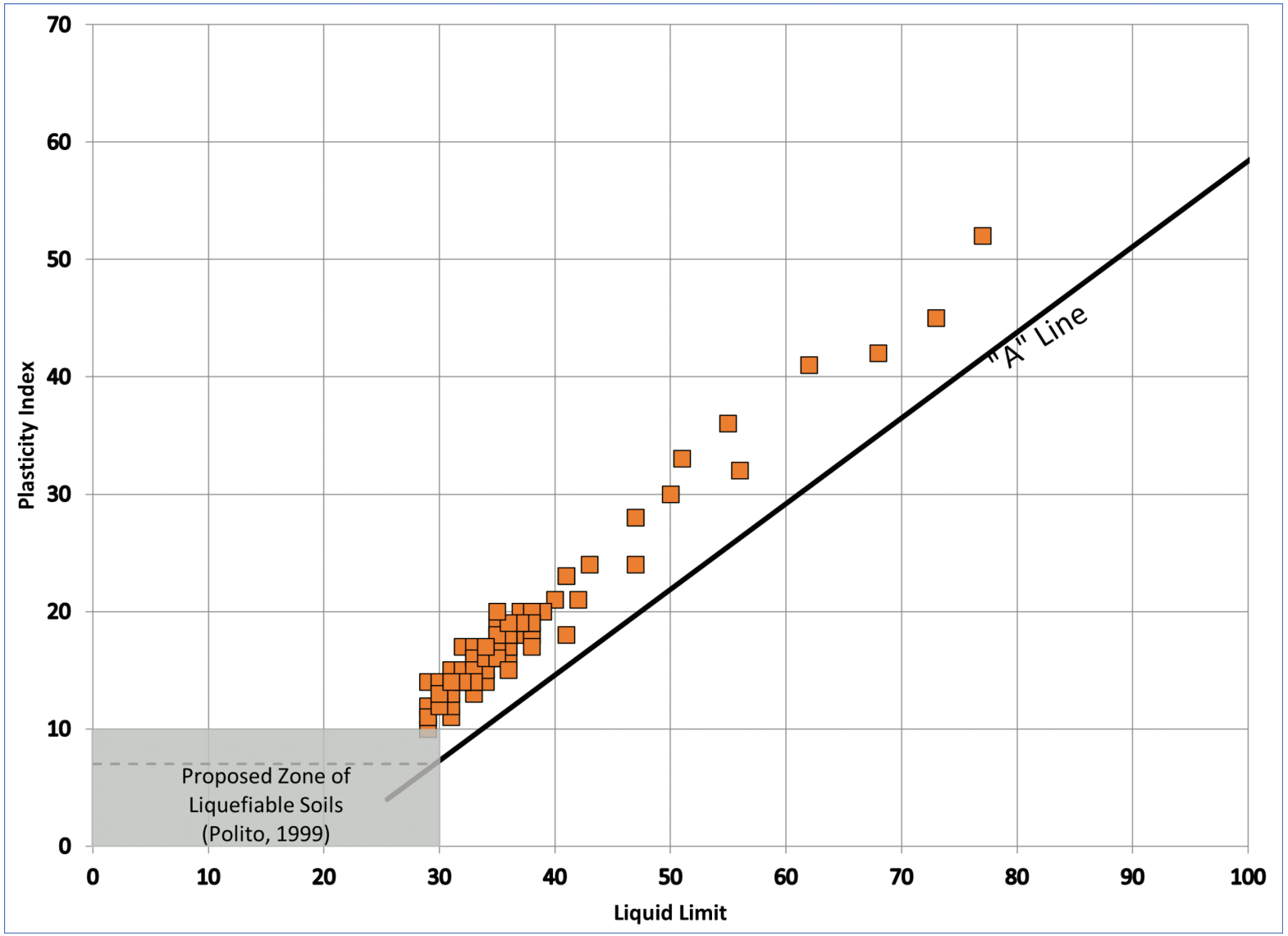
**Figure 2.6-107 Liquefaction Assessment – Polito**





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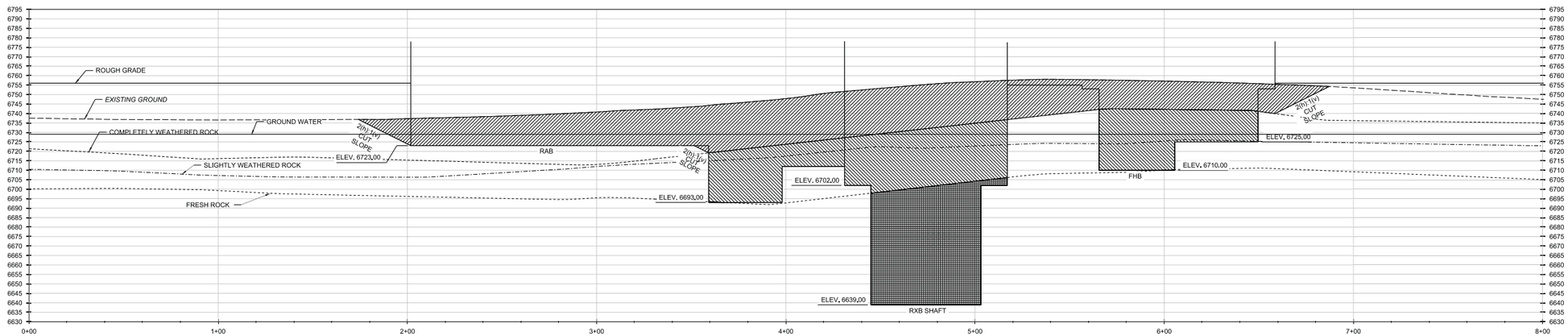
# Figure 2.6-108 Excavation Sections – E/W through FHB/RXB/RAB




SECTION A
CEK-1000-0001

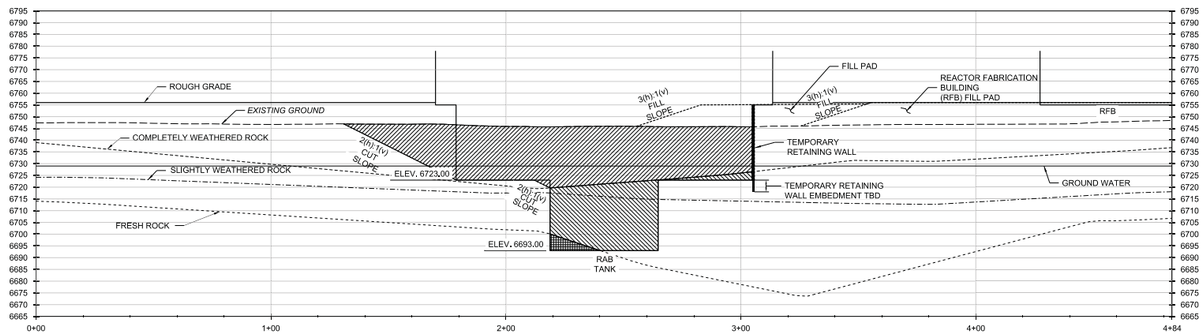
0 20 40 60FT
HORIZONTAL SCALE: 1"=20'
0 20 40 60FT
VERTICAL SCALE: 1"=20'

NOTES:

1. THE GROUNDWATER LEVEL SHOWN ON THIS DRAWING IS THE MAXIMUM GROUNDWATER LEVEL (ELEV. 6729.00) BASED ON THE LONG-TERM GROUNDWATER MEASUREMENT DATA AT THE NATRIUM SITE FROM MAY/JUNE 2022 TILL APRIL 2023.

LEGEND:

- CONVENTIONAL EXCAVATION REMOVE SOIL
- VERTICAL CUT WITH SUPPORTS IN WEATHERED ROCK
- VERTICAL CUT WITH SUPPORTS IN FRESH ROCK




SECTION B
CEK-1000-0001

0 20 40 60FT
HORIZONTAL SCALE: 1"=20'
0 20 40 60FT
VERTICAL SCALE: 1"=20'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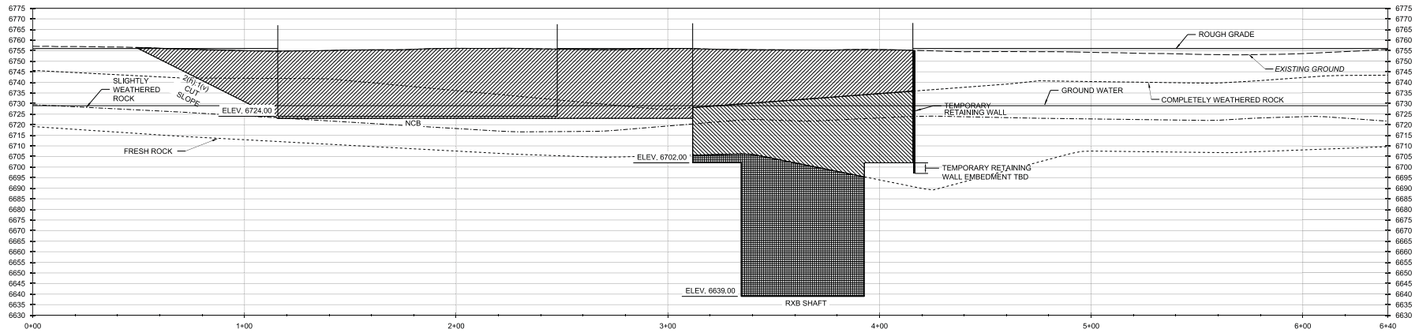
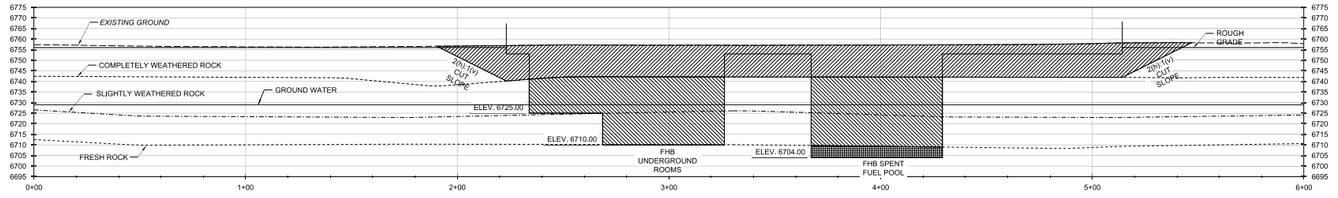
# Figure 2.6-109 Excavation Sections – N/S through RXB/NCB and FHB



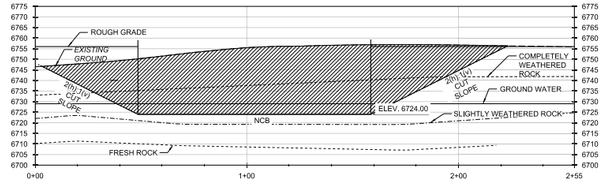

SECTION C
CEK-1000-0001

HORIZONTAL SCALE: 1"=20'
VERTICAL SCALE: 1"=20'




SECTION D
CEK-1000-0001

HORIZONTAL SCALE: 1"=20'
VERTICAL SCALE: 1"=20'




SECTION E
CEK-1000-0001

HORIZONTAL SCALE: 1"=20'
VERTICAL SCALE: 1"=20'

NOTES:

1. THE GROUNDWATER LEVEL SHOWN ON THIS DRAWING IS THE MAXIMUM GROUNDWATER LEVEL (ELEV. 6729.00) BASED ON THE LONG-TERM GROUNDWATER MEASUREMENT DATA AT THE NATRIUM SITE FROM MAY/JUNE 2022 TILL APRIL 2023.

LEGEND:

- CONVENTIONAL EXCAVATION REMOVE SOIL
- VERTICAL CUT WITH SUPPORTS IN WEATHERED ROCK
- VERTICAL CUT WITH SUPPORTS IN FRESH ROCK


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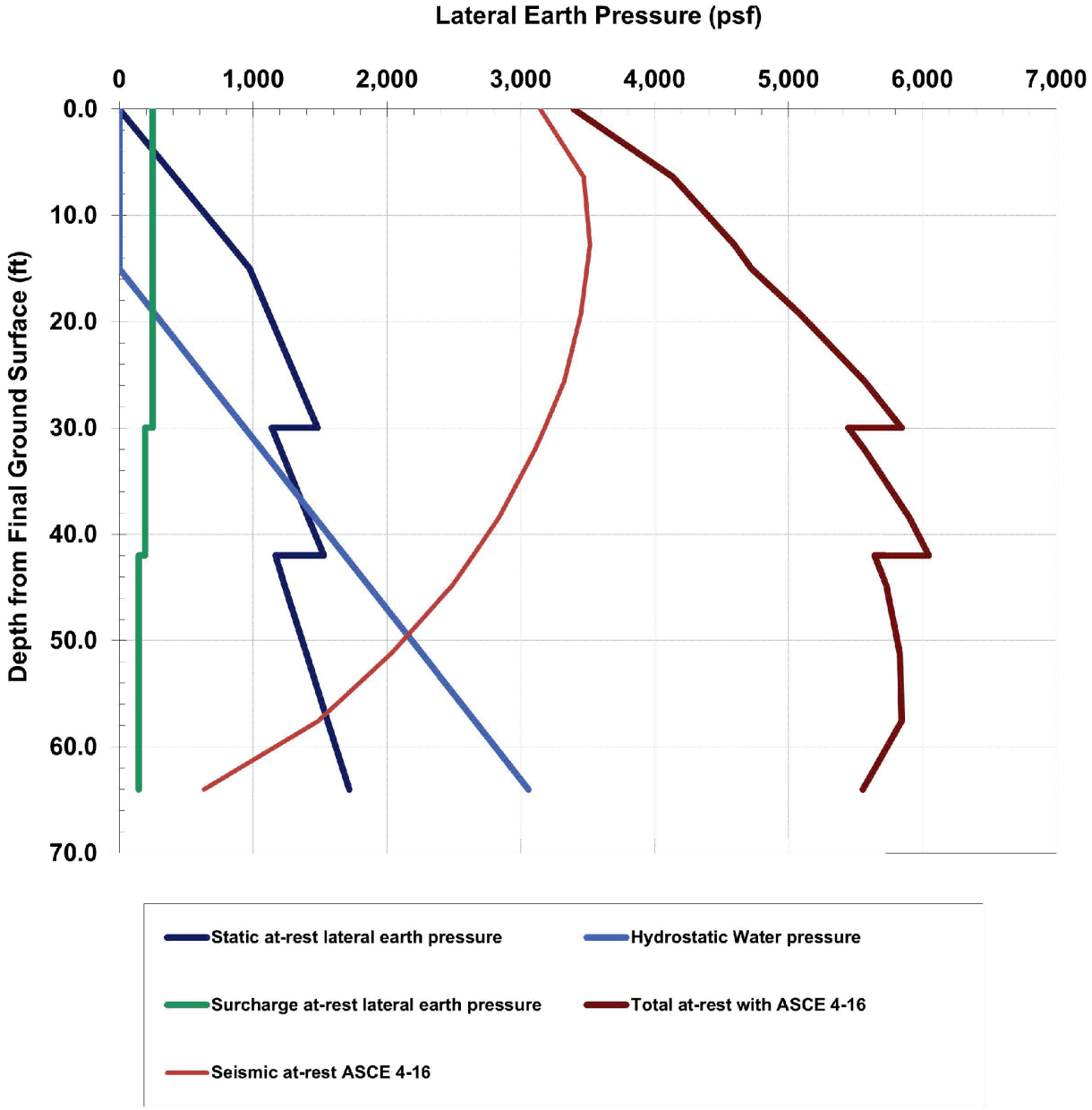
**Figure 2.6-110 Combined At-Rest Lateral Earth Pressures on Below Grade Walls**





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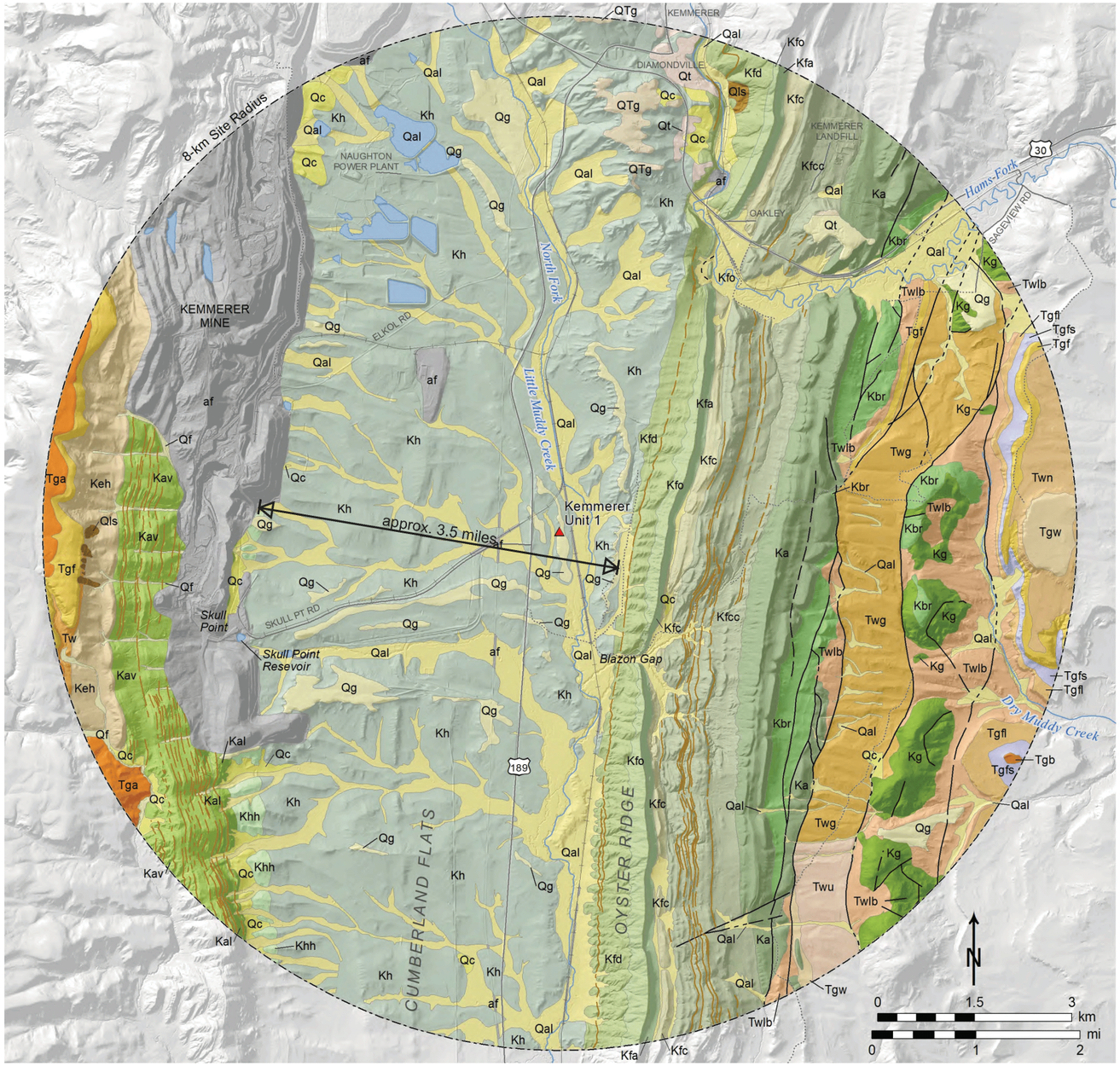
## Figure 2.6-111 Geologic Map of the Site Area




**Quaternary**

**Surficial Units**

| | |
|---|---|
| af | artificial fill / disturbed ground |
| Qal | Alluvium |
| Qc | Colluvium |
| Qg | Gravel deposits |
| Qf | Alluvium, colluvium and alluvial-fan deposits |
| Qls | Slump and landslide deposits |
| Qt | Terrace deposits |
| QTg | High terrace deposits |

**Eocene**

**Wasatch Formation**

| | |
|---|---|
| Twu | Upper Member |
| Twn | New Fork Tongue |
| Twlb | La Barge Member |
| Tw | Main body |

**Green River Formation**

| | |
|---|---|
| Tga | Angelo Member |
| Tgb | Fossil Buttle Member |
| Tgw | Wilkins Peak Member |
| Tgf | Fontenelle Tongue |
| Tgfs | Middle Part Fontenelle Tongue |
| Tgfl | Lower Part Fontenelle Tongue |
| Twg | Wasatch and Green River Fms undivided |

**Evanston Formation**

| | |
|---|---|
| Keh | Hams Fork Conglomerate |

**Adaville Formation**

| | |
|---|---|
| Kav | Adaville Formation |
| Kal | Lazeart Sandstone |

**Hilliard Shale**

| | |
|---|---|
| Khh | Hinshaw Member |
| Kh | Hilliard Shale |

**Frontier Formation**

| | |
|---|---|
| Kfd | Dry Hollow Member |
| Kfo | Oyster Ridge Sandstone |
| Kfa | Allen Hollow Member |
| Kfc | Coalville Member |
| Kfcc | Chalk Creek Member |

**Cretaceous**

| | |
|---|---|
| Ka | Aspen Formation |
| Kbr | Bear River Formation |
| Kg | Gannet Group |

| | |
|---|---|
| — – coal, dashed | — – fault, dashed |
| - - - coal, dotted | - - - fault, dotted |
| —— coal, solid | —— fault, solid |
| contact, solid | |

Sources:
1. Reference 2.6.5-d
2. Reference 2.6.5-a
3. Reference 2.6.5-b
4. Reference 2.6.5-c

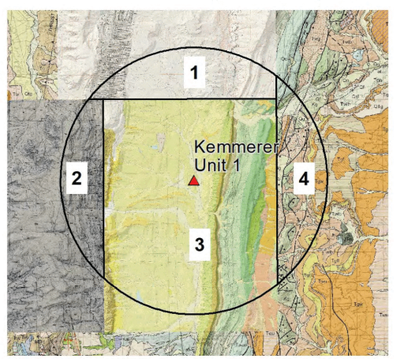





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**Figure 2.6-112 Existing Site Topography**

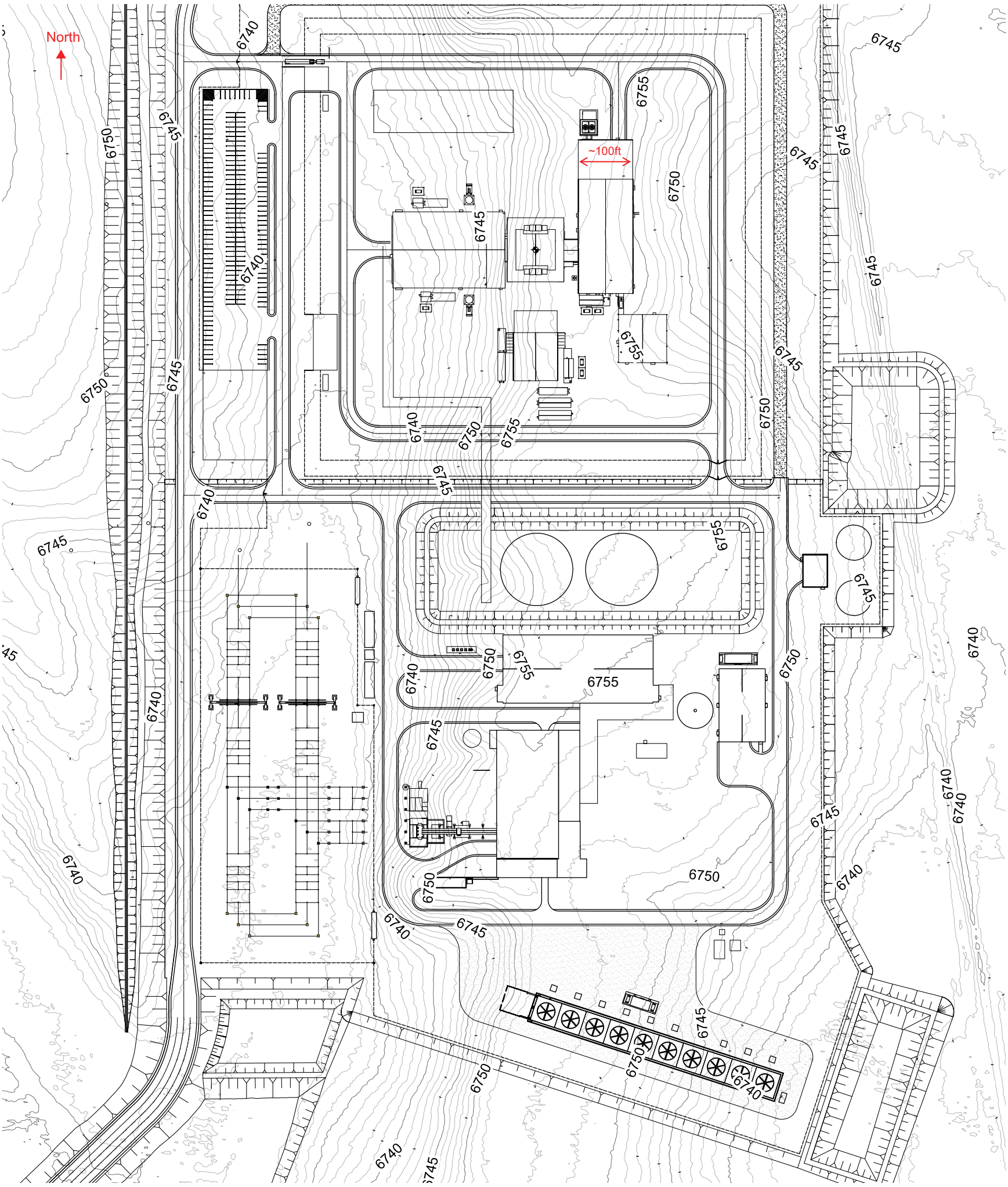


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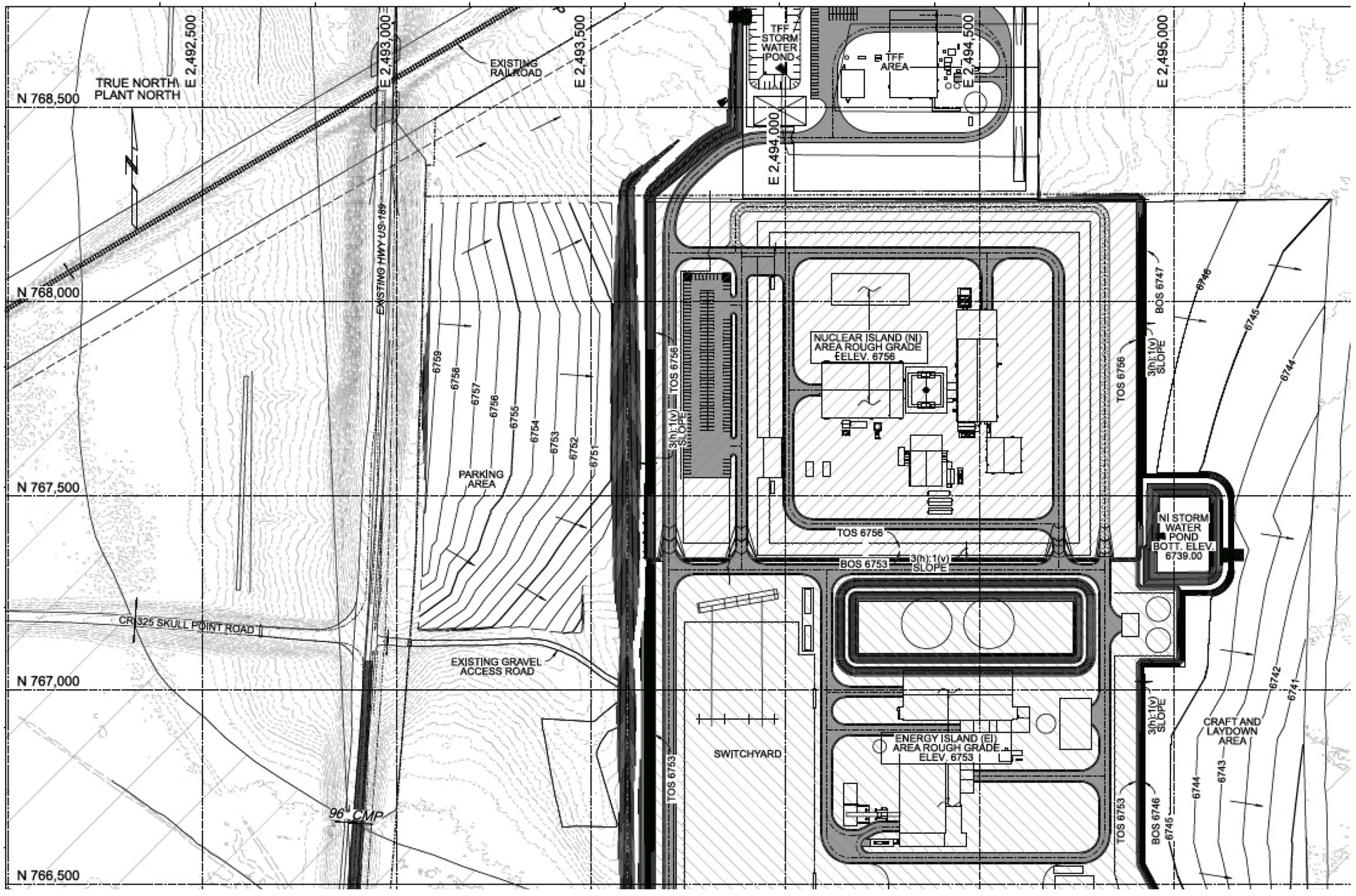
**Figure 2.6-113 NI Rough Grade Plan**





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## 2.7 Volcanic Hazards

Quaternary volcanoes are present within 320 kilometers of the Kemmerer Unit 1 site, so a volcanic hazards assessment was performed in accordance with NRC Regulatory Guide 4.26, Revision 0, “Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites,” to evaluate the potential for future volcanic eruptions in the region, to characterize the types of volcanic phenomena those potential eruptions might produce, and to determine if those phenomena could present an external event hazard at the site. The methodology used for screening of potential volcanic hazards, results of the hazards assessment, and retained volcanic hazard characteristics are documented in Sections 4-9 of NAT-3226, Revision 0A (Reference 2.7-1), which are incorporated by reference into the SAR. See Section 1.4.2 for a list of documents incorporated by reference into the SAR.

The volcanic hazards assessment concludes that the site is located at sufficient distances from sources of potential future volcanic eruptions to preclude the following hazard phenomena from reaching the site:

- Potential for the opening of a new volcanic vent
- Proximal hazards
- Lava flows
- Pyroclastic density flows

The following hazard phenomena have the potential to reach the site and screened in for further evaluation:

- Tephra-falls, including volcanic ash (tephra less than 2 milimeters in diameter)
- Debris flows

Tephra-fall characteristics are described in Sections 8 and 9 of Reference 2.7-1. Volcanic tephra-falls at the site are a beyond design basis event because the likelihood of tephra-fall occurring at the site is less than $1x10^{-5}$ per year.

Debris flows would not create flooding conditions at the site as described in Section 8 of Reference 2.7-1.

References

2.7-1 TerraPower, LLC, “An Analysis of Potential Volcanic Hazards at the Proposed Natrium Site near Kemmerer, Wyoming,” NAT-3226, Revision 0A.


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## 2.8 Summary of External Hazards

The design basis external hazards were established using the site characterization information described in Chapter 2. A summary of the design basis external hazards is provided in Section 2.1.



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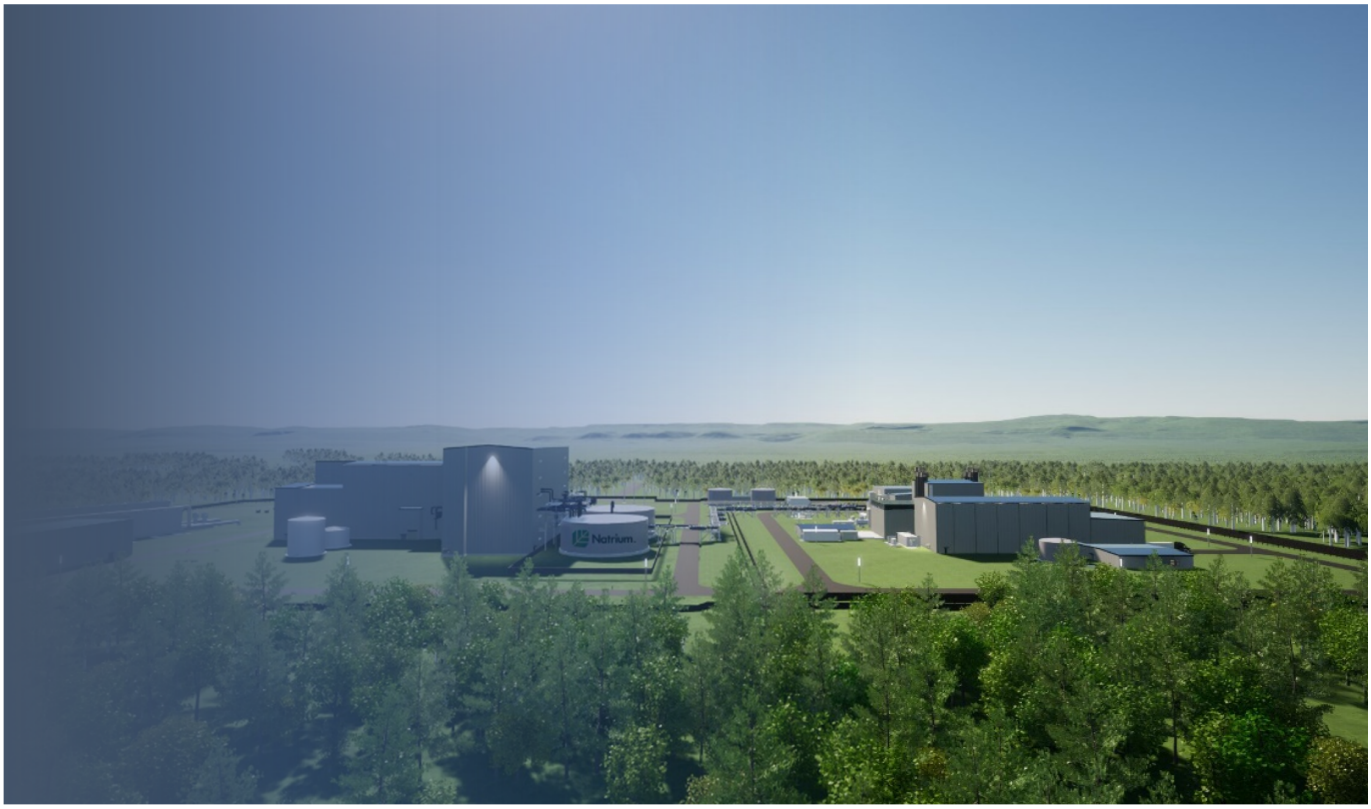
Natrium™

a TerraPower & GE-Hitachi technology



# Kemmerer Power Station Unit 1
# PSAR, Chapter 3

US SFR Owner, LLC, a wholly owned subsidiary of TerraPower, LLC

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## TABLE OF CONTENTS

**Chapter 3 Licensing Basis Events . . . . . 3.1-1**

3.1 Probabilistic Risk Assessment . . . . . 3.1-1

3.1.1 Overview of PRA . . . . . 3.1-1

3.1.2 Summary of PRA Results Essential to the Licensing Basis . . . . . 3.1-4

3.2 Licensing Methodology for Mechanistic Source Term . . . . . 3.2-1

3.2.1 Mechanistic Source Term Summary Tables . . . . . 3.2-2

3.3 Licensing Methods for Evaluation of Licensing Basis Events . . . . . 3.3-1

3.3.1 Licensing Methodology for Design Basis Accidents . . . . . 3.3-1

3.3.2 Licensing Methodology for AOO, DBE, and BDBE . . . . . 3.3-2

3.3.3 Licensing Methodology for Major Accident . . . . . 3.3-3

3.3.4 LBE Plant Response and Analysis Overview . . . . . 3.3-3

3.4 Licensing Basis Event Selection Methodology . . . . . 3.4-1

3.5 LBE Summary . . . . . 3.5-1

3.5.1 Summary Evaluation of AOOs, DBEs, and BDBEs . . . . . 3.5-1

3.5.2 Summary Evaluation of DBAs . . . . . 3.5-1

3.6 Anticipated Operational Occurrences . . . . . 3.6-1

3.6.1 Loss of Primary Flow AOOs . . . . . 3.6-1

3.6.2 Increase or Decrease in Heat Removal AOOs . . . . . 3.6-2

3.6.3 Reactivity AOOs . . . . . 3.6-3

3.6.4 Fuel Handling AOOs . . . . . 3.6-3

3.7 Design Basis Events . . . . . 3.7-1

3.7.1 Loss of Primary Flow DBEs . . . . . 3.7-1

3.7.2 Increase or Decrease in Heat Removal DBEs . . . . . 3.7-2

3.7.3 Release from Ex-Vessel Systems DBEs . . . . . 3.7-5

3.7.4 Fuel Handling DBEs . . . . . 3.7-10


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## TABLE OF CONTENTS


3.7.5 Local Fuel Fault DBEs . . . . . 3.7-15

3.8 Beyond Design Basis Events . . . . . 3.8-1

3.8.1 Loss of Primary Flow BDBEs . . . . . 3.8-1

3.8.2 Increase or Decrease in Heat Removal BDBEs . . . . . 3.8-5

3.8.3 Reactivity BDBEs . . . . . 3.8-8

3.8.4 Release from Ex-Vessel Systems BDBEs . . . . . 3.8-11

3.8.5 Fuel Handling BDBEs . . . . . 3.8-12

3.8.6 Local Fuel Fault BDBEs . . . . . 3.8-23

3.8.7 Other BDBEs . . . . . 3.8-25

3.9 Design Basis Accidents . . . . . 3.9-1

3.9.1 Loss of Primary Flow DBAs . . . . . 3.9-1

3.9.2 Increase or Decrease in Heat Removal DBAs . . . . . 3.9-2

3.9.3 Reactivity DBAs . . . . . 3.9-4

3.9.4 Release from Ex-Vessel Systems DBAs . . . . . 3.9-4

3.9.5 Fuel Handling DBAs . . . . . 3.9-7

3.9.6 Local Fuel Fault DBAs . . . . . 3.9-10

3.10 10 CFR 50.155 Mitigation Strategies . . . . . 3.10-1

3.11 Fuel System Design . . . . . 3.11-1

3.12 Nuclear Design . . . . . 3.12-1

3.12.1 Nuclear Design Summary . . . . . 3.12-1

3.12.2 Design Basis . . . . . 3.12-3

3.12.3 Codes Used in System Evaluation . . . . . 3.12-4

3.13 Thermal and Hydraulic Design . . . . . 3.13-1

3.13.1 Description . . . . . 3.13-1

3.13.2 Design Basis . . . . . 3.13-1



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## TABLE OF CONTENTS

3.13.3 Codes Used in System Evaluation . . . . . 3.13-2

3.13.4 Testing . . . . . 3.13-2

3.14 Criticality Safety of Fresh and Spent Fuel . . . . . 3.14-1

3.14.1 Design Basis . . . . . 3.14-1

3.14.2 Methodology . . . . . 3.14-1


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## LIST OF TABLES


Table 3.1-1 Potentially More-Than-Minor PRA Assumptions . . . . . 3.1-6

Table 3.1-2 Hazard Screening Summary . . . . . 3.1-8

Table 3.1-3 Hazards That Proceed to Analysis . . . . . 3.1-12

Table 3.2-1 Protected Loss of Flow or Loss of Offsite Power . . . . . 3.2-3

Table 3.2-2 Local Fault Event . . . . . 3.2-4

Table 3.2-3 In-vessel Fuel Drop . . . . . 3.2-5

Table 3.2-4 Local Fault Event DBA . . . . . 3.2-6

Table 3.2-5 In-vessel Fuel Drop DBA, 1 Assembly . . . . . 3.2-8

Table 3.2-6 In-vessel Fuel Drop DBA, 2 Assemblies . . . . . 3.2-9

Table 3.2-7 Spent Fuel Release inside EVHM . . . . . 3.2-10

Table 3.2-8 Low Decay (LTA/LDA) Fuel Release inside EVHM . . . . . 3.2-11

Table 3.2-9 Spent Fuel Drop in Spent Fuel Pool, 1 Assembly . . . . . 3.2-12

Table 3.2-10 Primary Sodium Processing Leak . . . . . 3.2-13

Table 3.2-11 Primary Sodium Processing Leak DBA . . . . . 3.2-14

Table 3.2-12 Cover Gas Release, Confinement Success . . . . . 3.2-15

Table 3.2-13 Cover Gas Release, Confinement Bypass . . . . . 3.2-16

Table 3.2-14 Cover Gas Release DBA . . . . . 3.2-17

Table 3.2-15 Release from RWG Holdup Tank, Release via FHB exhaust . . . . . 3.2-18

Table 3.2-16 Release from RWG Holdup Tank, Release via RWG . . . . . 3.2-19

Table 3.2-17 Release from RWG Holdup Tank, Release via RWG DBA . . . . . 3.2-20

Table 3.2-18 Release of Tritium from SPS-I Cold Trap . . . . . 3.2-21

Table 3.2-19 Unprotected Loss of Heat Sink, Full Core and IVS Failure . . . . . 3.2-22

Table 3.5-1 Summary of AOOs . . . . . 3.5-2

Table 3.5-2 Summary of DBEs . . . . . 3.5-3

Table 3.5-3 Summary of BDBEs . . . . . 3.5-5

Table 3.5-4 Summary of DBAs . . . . . 3.5-8



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## LIST OF FIGURES

Figure 3.5-1 F-C Chart for LBEs with Uncertainty Bands . . . . . 3.5-10


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# Chapter 3 Licensing Basis Events

## 3.1 Probabilistic Risk Assessment

The probabilistic risk assessment (PRA) is the plant model that provides an integrated assessment of risk to the public from the nuclear power plant. Implementing the NEI 18-04 (Reference 3.1-1) framework requires a technically adequate PRA. Section 3.1.1 summarizes the scope, methodology, and technical adequacy of the PRA. Section 3.1.2 identifies the chapters that describe the remaining PRA insights and results that impact the licensing basis.

### 3.1.1 Overview of PRA

#### 3.1.1.1 PRA Methodology

The PRA is being developed in accordance with the applicable requirements of ASME/ANS RA-S-1.4-2021, “Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants” (Reference 3.1-2) as endorsed by Regulatory Guide (RG) 1.247, trial revision, “Acceptability of Probabilistic Risk Assessment Results for Non-Light Water Reactor Risk-Informed Activities,” without deviations. The intent of the PRA is to implement the NEI 18-04 framework as endorsed by RG 1.233, Revision 0, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors.” The PRA is maintained and documented under the Quality Assurance Program described in Section 8.1.

#### 3.1.1.2 PRA Technical Adequacy

The PRA technical adequacy for the construction permit (CP) application (CPA) is established by conducting a self-assessment of the internal events at-power and low-power shutdown PRA following the guidance provided in NEI 20-09 (Reference 3.1-3). The self-assessment includes a review of the following PRA elements from ASME/ANS RA-S-1.4-2021.

- Plant Operating State Analysis
- Initiating Event Analysis
- Event Sequence Analysis
- Success Criteria Analysis
- Systems Analysis
- Human Reliability Analysis
- Data Analysis
- Hazard Screening
- Event Sequence Quantification Analysis
- Mechanistic Source Term Analysis
- Radiological Consequence Analysis
- Risk Integration


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The self-assessment results were compared to the supporting requirements identified in Table B-2 and Table B-3 of DG-1404 (Reference 3.1-4) and concluded that the PRA is acceptable to use for the CPA. PRA assumptions are documented and assumptions that are evaluated to have potential more-than-minor impacts are described in Table 3.1-1. Proposed PRA model changes are documented and describe the proposed change and potential impact on the model. The PRA model changes are tracked for completion. Additionally, an evaluation comparing differences from the CP stage design to the CP stage PRA determined that the CP stage PRA reasonably reflects the CP stage design and the identified differences have a minimal impact on PRA results and key insights. As the plant design evolves, configuration and design control processes will ensure the PRA is reviewed and updated to reflect the as-designed, as-to-be-constructed, and as-to-be-operated plant design and margin to Quantitative Health Objectives are maintained.

In accordance with RG 1.247, prior to the OLA, a full-scope peer review of the PRA, reflecting the operating license (OL) plant design, will be performed against the applicable requirements of ASME/ANS RA-S-1.4-2021 following the guidance provided in NEI 20-09.

#### 3.1.1.3 Sources of Radionuclides

Sources of radionuclides were included by identifying events that could potentially impact the reactor core, components within the Reactor Enclosure System (RES), core assemblies being removed from the RES to the Ex-Vessel Storage Tank (EVST) during a refueling outage, spent fuel processing activities, and radionuclide sources that could result in a release greater than 100 mrem (30-day Total Effective Dose Equivalent) at the exclusion area boundary. The following sources of radionuclides are evaluated by the PRA.

- RES, including spent fuel in In-vessel Storage
- Primary Sodium Processing System and Sodium Cover Gas System (SCG)
- Intermediate Sodium Processing System
- Gaseous Radiological Waste
- Ex-Vessel Handling Machine (EVHM)
- EVST
- Spent Fuel Processing, including bottom loading transfer cask, Pin Removal Cell, Pool Immersion Cell, and Dry Cask loading
- Spent Fuel Pool

#### 3.1.1.4 Multi-Reactor Scenarios

Multi-reactor scenarios are not applicable, as Kemmerer Unit 1 is a single reactor site.

#### 3.1.1.5 Internal and External Hazards

A hazard screening analysis has been performed for all applicable hazards, which includes all hazards identified in Tables B-1 and B-2 of RG 1.247. The hazard screening for at-power operation and low-power shutdown uses the generic screening criteria specified in ASME/ANS RA-S-1.4-2021 and examines coexistent hazards and cliff-edge effects. Table 3.1-2


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summarizes the hazards that screened out of the at-power operation PRA and low-power shutdown PRA. Table 3.1-3 summarizes the hazards that are addressed by detailed PRA or design basis hazard levels (DBHLs).

#### 3.1.1.6 Plant Operating States

The scope of plant operating states (POS) focuses on changes in plant operations that could affect performance of one or more of the radionuclide release barriers or plant safety functions. The analysis determines that none of the POSs are screened out. The following POSs are included in the scope of the PRA.

- At-Power Operation
    - Peak Power
    - Minimum Power
- Hot Standby
- Unlocked Core
- Reactor Shutdown
- Preparation for Refueling Operations
- Refueling Activities
- Preparation for Reactor Operation
- Refuel Reactor Startup (hot standby)
- Fuel resides in the EVST
- Fuel movement in the EVST
- Intermediate Sodium Processing System Cold Trap Leak
- Spent Fuel Processing
- Fuel resides in Spent Fuel Pool

#### 3.1.1.7 Initiating Event Identification

Internal Initiating Events (IE) are considered for all sources of radionuclides and POSs that are identified in Section 3.1.1.3 and Section 3.1.1.6, respectively. IEs were initially identified by reviewing published IE lists for light water reactors and non-light water reactors. The plant event fault list was then reviewed to identify plant-specific events. Lastly, the plant system list was reviewed to identify special initiators or events caused by interactions between systems or trains. After IE identification was complete, the IEs were grouped based on expected mitigation response, timing, and other factors. The grouped IEs were combined into a single IE whose frequency is the sum of the individual IE frequencies. The individual event frequencies were developed using appropriate and available generic data sources in combination with available design-specific information and fault tree modeling as appropriate.

#### 3.1.1.8 PRA Model Integration

Event sequences identify combinations of POSs, IEs, safety functions, system failures and successes, and end-states. The end-states result in either a safe, stable plant state or a radionuclide release. The radionuclide releases are referred to as release categories and are


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grouped together based on similar timing and severity of release. An appropriate mechanistic source term is assigned to each release category to calculate the radiological consequences needed to evaluate tasks 7a and 7b of NEI 18-04.

The integrated evaluations calculated in Section 4.1 adjust the release category sequence tags to reflect the Quantitative Health Objective radiological consequence data. The total frequency of releases calculated are the values identified in Section 4.1.1 to Section 4.1.3.

#### 3.1.1.9 Software and Analytical Tools

Electric Power Research Institute Phoenix Architect 2.0 (Reference 3.1-5), which includes CAFTA, PRAQuant, UNCERT, and FRANX, and FTREX Version 1.8 (Reference 3.1-6), are used to perform the event sequence modeling and quantification. The parametric uncertainty analysis used UNCERT to calculate the 5th and 95th percentile values for licensing basis event (LBE) frequencies. Phoenix Architect and FTREX are industry standard codes that have been demonstrated to produce reliable results. No method-specific limitations have been identified with regard to the software tools or the methodology implemented to quantify the model.

The software and analytical tools that are used to determine the mechanistic source terms and evaluate the radiological consequences are described in Section 3.2 and Section 3.3, respectively.

#### 3.1.1.10 Site Characteristics

No adjustments are made to the PRA and uncertainty assessments to address the bounding site characterizations and structure, system, or component (SSC) fragilities based on the DBHLs described in Section 6.1.1 as the PRA at the CP stage is internal events at-power and low-power shutdown only.

### 3.1.2 Summary of PRA Results Essential to the Licensing Basis

While PRA results and risk insights in the preliminary safety analysis report are preliminary relative to the PRA results and risk insights that will be presented in the final safety analysis report, the following chapters discuss identified gaps and address how the PRA is used during the design and construction of the plant.

Chapter 3 presents the LBEs which are derived from the PRA and informed by the PRA event sequence families. Chapter 3 includes a brief description of the LBEs and a plot of the frequencies, consequences, and uncertainties of these LBEs in comparison to the F-C target presented in Figure 3-1 of NEI 18-04. The event descriptions include initial plant conditions, radionuclide sources, IEs covered in the event sequence family, plant response of SSCs, and operator actions that perform PRA safety functions.

Chapter 4 presents the integrated risk results across all of the anticipated operational occurrences, design basis events, and beyond design basis events, and compares the integrated risk results to the three cumulative plant performance metrics contained in NEI 18-04. Chapter 4


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also describes the defense-in-depth adequacy evaluation to ensure proper defense-in-depth protective measures have been incorporated to alleviate uncertainties, assumptions, and limitations in the PRA.

Chapter 5 presents the preliminary determination of PRA safety functions addressed by safety-related SSCs and non-safety-related with special treatment (NSRST) SSCs. Chapter 5 includes the documentation of risk-significant SSCs that are required to meet the F-C target and cumulative risk metrics.

Chapter 6 describes the reliability and capability targets for safety-related SSCs and NSRST SSCs. The PRA results and insights provide input to the targets and the resulting special treatments.

Chapter 8 describes the plant programs that are used to support the design and construction phase. The descriptions include how the programs support and receive insights from the PRA.

**References**

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3.1-2 ASME/ANS, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” ASME/ANS RA-S-1.4-2021, February, 2021.

3.1-3 Nuclear Energy Institute, “Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard,” NEI Technical Report 20-09, Revision 1, August 2020, [ADAMS Accession No. ML20302A115].

3.1-4 U.S. Nuclear Regulatory Commission, “Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water-Reactors,” Revision 1, August 2023, [ADAMS Accession No. ML23194A194].

3.1-5 Electric Power Research Institute, “Phoenix Architect,” Version 2.0, May 2022.

3.1-6 Electric Power Research Institute, “Fault Tree Reliability Evaluation Expert (FTREX),” Version 1.8, June 2015.

3.1-7 TerraPower LLC, “An Analysis of Potential Volcanic Hazards at the Proposed Natrium™ Site near Kemmerer, Wyoming,” NAT-3226, Revision 0A.


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**Table 3.1-1 Potentially More-Than-Minor PRA Assumptions**
**(Sheet 1 of 2)**

| Assumption | Description |
|---|---|
| Local core blockage | It is assumed that local core faults have an initiating event frequency of $1x10^{-3}$ per year and that the frequency of a loss of hydraulic hold down is included in this value. Further, it is assumed that this event is lognormally distributed with an error factor of 10.<br>The generic data on which this frequency is based show a range of seven orders of magnitude. This assumption picks the mid-range. |
| Testing frequency for each complete channel of the Nuclear Island Control System (NIC) system | It is assumed that each complete channel of the NIC system is tested once per quarter and the testing checks all instruments and functions on the channel. Each test takes about two hours and the channel is unavailable and not functional during the testing. This is a simplifying assumption that is not inconsistent with the very limited design information available. |
| Software common cause factor of the NIC system | It is assumed that common cause factor of the NIC system is bounded by a single digital common cause factor. This failure has a $1x10^{-4}$ failure probability and fails NIC system functions. Therefore, no common cause factors of individual NIC system components are modeled. |
| EVHM Passive Failure | In general, the fundamental failure mode for passive cooling of the EVHM is insufficient heat transfer. This part of the system is not fully designed and thus additional failure modes could exist. A passive system failure rate of $1x10^{-4}$ per demand is assumed. |
| SCG Filter Location | It is assumed that the cell for the SCG filters and traps is located inside the Reactor Building and that the cell is sealed against leakage to the surrounding Reactor Building areas. |
| Consequential fuel damage | It is assumed that inadvertent motion of any partially raised core assembly during the refueling process does not result in damage that could result in radionuclide release of any adjacent assembly.<br>This assumption is reasonable given what is known about the current design. |
| Molten Salt Data | Reliability data for components in molten salt systems generally is not available. Therefore, the reliability data for analogous sodium systems is used. |
| Control Rod Insertion Failure Rate | The control rod insertion/binding most closely resembles the pressurized water reactor gravity rod design. The pressurized water reactor data is reasonable for use for the rod binding portion of the failure rate. The drive-in function consists of a motor and a threaded shaft which shares many characteristics with a motor operated valve so motor operated valve data from NUREG/CR-6928 was used. |


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**Table 3.1-1 Potentially More-Than-Minor PRA Assumptions**
**(Sheet 2 of 2)**

| Assumption | Description |
|---|---|
| EVHM active cooling is sufficient for a hot fuel element greater than design | Because design details are not available, based on discussion with design staff, it is expected that active cooling mode will be able to cool any fuel assembly removed from the vessel, thus if a fuel handling error removes a fuel assembly that has not had time to decay sufficient cooling will be available with active cooling. |
| EVST/HVAC dampers | It is assumed the dampers for the Fuel Handling Building HVAC and the passive dampers for the EVST vault are different types such that common cause is not applicable. |
| Different geometry for the two groups of Control Rod Assemblies (CRA) | Primary CRAs have a different geometry from secondary CRAs. Therefore, the two groups of CRAs are considered as different common cause failure groups for failure to insert on scram.<br>This assumption is consistent with the best available design information. |
| Assumed leak rate from vessel head | Assumed leak rate from reactor head to containment (i.e., the Head Access Area [HAA]) is 1% volume per day when the head is successfully isolated. Failure of head isolation results in a leakage rate of 100% volume per day. |
| Assumed leak rate from containment (i.e., the HAA). | Assumed leak rate from containment (i.e., the HAA) to the environment is 10% volume per day when the containment is successfully isolated. Failure of containment isolation results in a leakage rate of 100% volume per day. |


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**Table 3.1-2 Hazard Screening Summary**
**(Sheet 1 of 4)**

| Screening Category | Hazard |
|---|---|
| Individual Hazards - Qualitatively Screened Out | Abrasive Dust and Sand Storms |
| | Air Pollution |
| | Aircraft Crash, Aircraft Impact, Aviation Accident, Jet Impact |
| | Animals |
| | Argon Gas Leak |
| | Avalanche |
| | Biological Agents |
| | Biological Events, Biological phenomena in water (plants, marine growth), including biological contamination, Organic Material in Water |
| | Biological phenomena in air (insects, leaves), including biological contamination |
| | Blast (pressure wave, not a chemical explosion) inside building |
| | Chemical Release after Pipeline Accident |
| | Chemical Release after Pipeline Accident Within the Site |
| | Chemical Release after Transportation Accident |
| | Chemical Release Outside Site |
| | Chemical Release to Water |
| | Chemical Release Within the Site, Release of Chemicals from On-Site Storage |
| | Collision of ships and floating debris with water intakes |
| | Corrosion (e.g., from salt water) |
| | Direct Impact from Ship Collision |
| | Dropped/Collapsing/Falling Loads |
| | Drought |
| | Eddy currents into ground |
| | Electromagnetic interference from off-site and on-site / Electromagnetic disturbance / Magnetic Disturbance |
| | Excavation Work |
| | Explosion After Pipeline Accident |
| | Explosion after Pipeline Accident Within the Site |
| | Explosion Outside Plant |
| | External Fire |
| | External Flooding |
| | Extreme Hail |
| | Extreme Rain / Intense Precipitation |
| | Extreme Snow (including snowstorm) |


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**Table 3.1-2 Hazard Screening Summary**
**(Sheet 2 of 4)**

| Screening Category | Hazard |
|---|---|
| Individual Hazards - Qualitatively Screened Out (continued) | Extreme snowpack |
| | Failure of a Dam Upstream of the Plant |
| | Fire generated from off-site sources |
| | Fog |
| | Forest Fire |
| | Frazil Ice |
| | Grass Fire (Off-Site) |
| | Grass Fire (On-Site) |
| | Ground Contamination |
| | Ground Washout |
| | Groundwater related effects |
| | High Air Temperature |
| | High Sea/River Water Level (e.g. high tide) |
| | High Sea/River Water Temperature |
| | High Tide |
| | Ice Barriers |
| | Ice Storm / sub-cooled rain |
| | Industrial or Military Facility Accident |
| | Internal Fire Spreading from Other Units on the Site |
| | Internal Flood and Harsh Environment Spreading from Other Units on the Site |
| | Internal missiles (from explosions, ruptures, collapses, dropping, high energy rotating machinery) |
| | Land Rise |
| | Landslide (above water) |
| | Lightning |
| | Low Air Temperature |
| | Low Sea/River/Lake Water Level |
| | Low Sea/River/Lake Water Temperature |
| | Marine Accidents, Ship Impact |
| | Missiles from Military Activity |
| | Missiles from other installations; Externally generated missiles |
| | Missiles from Other Units on the Site |
| | Mist |
| | Multi-Unit Issues |


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**Table 3.1-2 Hazard Screening Summary**
**(Sheet 3 of 4)**

| Screening Category | Hazard |
|---|---|
| Individual Hazards - Qualitatively Screened Out (continued) | Radioactive releases |
| | Railroad Accident |
| | Release of Corrosive Gases or Liquids from Off-Site Storage |
| | Release of Hazardous Gas (Asphyxiant, Toxic) from Off-Site Storage |
| | Release of Hazardous Gas (Asphyxiant, Toxic) from On-Site Storage, Toxic Gas |
| | Release of Hazardous/Corrosive Gases or Liquids from On-Site Storage |
| | Release of toxic gas inside plant |
| | River Diversion |
| | River transported material |
| | Salt Storm |
| | Seiche |
| | Shockwave from Transformer Explosion |
| | Sinkholes, Karsts |
| | Snow (drift) |
| | Soil Erosion, Coastal Erosion |
| | Soil Frost |
| | Soil shrink-swell consolidation |
| | Solid or Fluid (non-gaseous) Impurities from Ship Release |
| | Storm Surge |
| | Strong Water Current (underwater erosion) |
| | Structural Collapse |
| | Surface Ice, Ice Cover |
| | Truck Accident (Off-Site), Vehicle Impact (Off-Site) |
| | Tsunami |
| | Turbine-Generated Missiles |
| | Underwater Landslide |
| | Vibrations, Ground vibration (e.g., due to nearby explosions) |
| | Water currents during a Flood |
| | Waves, Other Extraordinary Waves |
| | White Frost |


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**Table 3.1-2 Hazard Screening Summary**
**(Sheet 4 of 4)**

| Screening Category | Hazard |
|---|---|
| Coexistent Hazards - Qualitatively Screened Out | Chemical release and Contamination from chemicals |
| | Drought and External fire |
| | Drought and Low water level |
| | Explosion after pipeline accident and Chemical release after pipeline accident |
| | Explosion after transportation accident and Chemical release after transportation accident |
| | Explosion outside plant and Chemical release outside or inside site |
| | Extreme rain and Lightning |
| | Extreme snow (including snowstorm) and Earthquake |
| | High air temperature and High water temperature |
| | Low air temperature and Low water temperature |
| Individual Hazards - Quantitatively Screened Out | Direct Impact of Heavy Transportation Within the Site |
| | Excavation Work Within the Site Area |
| | Meteorite |
| | Orbital Debris |
| | Truck Explosions |
| | Vehicle Impact (On-Site), Truck Accident (On-Site) |
| | Vehicle/Ship Explosion (On-Site) |
| | Volcanic Phenomena - Small Volcano Eruption |


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**Table 3.1-3 Hazards That Proceed to Analysis**

| Hazard | Analysis |
|---|---|
| Volcanic Phenomena | Volcanic hazards are assessed in Section 4-9 of NAT-3226, Revision 0A (Reference 3.1-7), which are incorporated by reference into Section 2.7. |
| Volcanic Phenomena and Extreme Snow (inlcuding snowstorm) | |
| Volcanic Phenomena and Earthquake | |
| Earthquakes, Seismic Events/Hazards | The seismic DBHL is credited for the CPA, but seismic hazards are selected for seismic PRA development. Efficacy of the DBHL will be confirmed with the seismic PRA at the OL stage |
| Soil Liquefaction | |
| Internal Fire | The internal fire hazards are selected for fire PRA development. The fire PRA will be available at the OL stage. |
| Explosion Within Plant | |
| Explosion Within the Site | |
| Explosive Electrical Faults | |
| Fire from Cask Transporter, on site materials | |
| Non-Safety Building Fire | |
| Sodium Fire | |
| Standby Diesel Generator on site | |
| Extreme Winds | The high winds DBHL is credited for the CPA, but high winds hazards are selected for high winds PRA development. Efficacy of the DBHL will be confirmed with the high winds PRA at the OL stage. |
| Strong Wind | |
| Cyclones (hurricanes, tornadoes, and tropical typhoons) | |
| Extreme Air Pressure (High/Low Gradient) | |
| Strong Winds and Extreme air pressure | |
| Strong Winds and Ice barriers | |
| Strong Winds and Extreme rain | |
| Tornadoes and Extreme hail | |
| Internal Flood | The internal flooding hazards are selected for internal flooding PRA development. The internal flooding PRA will be available at the OL stage. |
| Pipe Failure Effects | |
| Pipe Whip | |
| Steam Release | |
| Spray | |
| Jet Effects | |


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## 3.2 Licensing Methodology for Mechanistic Source Term

The mechanistic source term (MST) is defined as the characteristics of a radionuclide release at a particular location, including the physical and chemical properties of released material, release magnitude, heat content (or energy) of the carrier fluid, and location relative to local obstacles that would affect transport away from the release point and the temporal variations in these parameters (e.g., time of release duration) that are calculated using models and supporting scientific data that simulate the physical and chemical processes that describe the radionuclide inventories and the time-dependent radionuclide transport mechanisms that are necessary and sufficient to predict the source term. The characteristics and attributes needed to achieve the objectives of an MST analysis are described by Regulatory Guide (RG) 1.247, trial revision, "Acceptability of Probabilistic Risk Assessment Results for Non-Light Water Reactor Risk-Informed Activities," as follows:

- Radionuclide releases are grouped into smaller subsets of representative source terms or release categories.
- Radionuclide releases are assessed for each release category, including consideration of timing, location, amount released, and the radionuclide transport barriers and transport mechanisms.
- Radiological source terms are calculated using appropriate methods or codes.
- Uncertainties in the MSTs and associated transport phenomena are identified, characterized, and quantified to the extent practical.
- Documentation of the MST analysis shall provide traceability of the work.

The methodology used to develop MSTs for evaluation of potential radiological consequences in anticipated operational occurrences (AOOs), design basis events (DBEs), beyond design basis events (BDBEs), and design basis accident (DBA) events is documented in TP-LIC-RPT-0003, "Radiological Source Term Methodology Report," Revision 1 (Reference 3.2-1). Sections 1 through 8 of TP-LIC-RPT-0003 are incorporated by reference. See Section 1.4.2 for a list of documents incorporated by reference.

The types of events with potential for radionuclide release, either from fuel failure or loss of integrity of a system carrying primary coolant, and the resultant release of dissolved isotopes include the following phenomena:

- Significant reduction in flow through the reactor core
- Localized high power-to-flow conditions within core assemblies
- Physical damage to the fuel (e.g., physical damage due to stochastic cladding failures during normal operation and potential impact damage due to scenarios like drop events)
- Reactivity insertion such as uncontrolled control rod withdrawal

These phenomena and their related events and models are discussed in more detail in Sections 2 and 4 of TP-LIC-RPT-0003.



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Assessments to evaluate model fidelity and accuracy, scalability of the models, and acceptance of the models are in process, and are captured in Section 5 of TP-LIC-RPT-0003. Requirements to determine model biases and uncertainties are also discussed in Section 5 of TP-LIC-RPT-0003. These and other items identified for development in Sections 3.2, 3.3, and 5 of TP-LIC-RPT-0003 will be available by the operating license stage.

Detailed discussion of the codes selected for use in this method and other codes supporting their use are discussed in Section 4 of TP-LIC-RPT-0003.

### 3.2.1 Mechanistic Source Term Summary Tables

Table 3.2-1 through Table 3.2-19 summarize the source terms developed for the LBEs described in Anticipated Operational Occurrences (Section 3.6), Design Basis Events (Section 3.7), Beyond Design Basis Events (Section 3.8), and Design Basis Accidents (Section 3.9). These tables include a list of the isotopes which contribute to at least 95 percent of the total dose, and the cumulative activity release of each isotope in curies.

## References

3.2-1 TerraPower, LLC “Radiological Source Term Methodology Report,” TP-LIC-RPT-0003, Revision 1


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**Table 3.2-1 Protected Loss of Flow or Loss of Offsite Power**

| Cumulative Activity Release to Environment (Ci) | | | | | |
|---|---|---|---|---|---|
| **Isotope** | **2hr** | **8hr** | **24hr** | **96hr** | **720hr** |
| **Xe-133** | 1.21E+02 | 1.88E+03 | 1.56E+04 | 1.74E+05 | 1.03E+06 |
| **Xe-135** | 1.10E+02 | 1.30E+03 | 5.54E+03 | 9.82E+03 | 9.87E+03 |
| **I-131** | 1.95E-01 | 7.04E-01 | 2.56E+00 | 1.99E+01 | 1.55E+02 |
| **Kr-88** | 3.83E+01 | 2.55E+02 | 4.27E+02 | 4.35E+02 | 4.35E+02 |
| **Cs-137** | 8.65E-02 | 4.28E-01 | 1.65E+00 | 7.60E+00 | 5.39E+01 |
| **Cs-134** | 3.28E-02 | 1.62E-01 | 6.25E-01 | 2.87E+00 | 2.02E+01 |


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**Table 3.2-2 Local Fault Event**

| Cumulative Activity Release to Environment (Ci) | | | | | |
|---|---|---|---|---|---|
| **Isotope** | **2hr** | **8hr** | **24hr** | **96hr** | **720hr** |
| **Xe-133** | 1.12E+00 | 1.74E+01 | 1.44E+02 | 1.62E+03 | 9.56E+03 |
| **Cs-137** | 1.06E-03 | 5.46E-03 | 2.52E-02 | 2.54E-01 | 2.83E+00 |
| **Cs-134** | 5.89E-04 | 3.02E-03 | 1.40E-02 | 1.41E-01 | 1.55E+00 |
| **I-131** | 1.91E-03 | 7.02E-03 | 2.63E-02 | 2.22E-01 | 1.65E+00 |
| **Xe-135** | 1.03E+00 | 1.21E+01 | 5.19E+01 | 9.20E+01 | 9.24E+01 |
| **Kr-88** | 3.21E-01 | 2.14E+00 | 3.57E+00 | 3.64E+00 | 3.64E+00 |
| **Cs-136** | 6.61E-04 | 3.37E-03 | 1.52E-02 | 1.39E-01 | 8.47E-01 |



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**Table 3.2-3 In-vessel Fuel Drop**

| Cumulative Activity Release to Environment (Ci) | | | | | |
|---|---|---|---|---|---|
| **Isotope** | **2hr** | **8hr** | **24hr** | **96hr** | **720hr** |
| **Xe-135** | 1.09E+03 | 1.16E+04 | 3.99E+04 | 5.51E+04 | 5.51E+04 |
| **Xe-133** | 1.21E+03 | 1.67E+04 | 1.04E+05 | 4.25E+05 | 5.10E+05 |
| **Kr-88** | 3.75E+02 | 2.28E+03 | 3.50E+03 | 3.54E+03 | 3.54E+03 |
| **I-131** | 2.69E+00 | 7.73E+00 | 2.46E+01 | 7.76E+01 | 9.31E+01 |
| **Cs-137** | 1.46E+00 | 4.89E+00 | 1.68E+01 | 3.94E+01 | 4.74E+01 |
| **Cs-134** | 6.38E-01 | 2.14E+00 | 7.36E+00 | 1.72E+01 | 2.07E+01 |
| **Sr-90** | 6.35E-01 | 9.34E-01 | 9.76E-01 | 9.78E-01 | 9.78E-01 |
| **Kr-87** | 1.95E+02 | 5.74E+02 | 6.08E+02 | 6.08E+02 | 6.08E+02 |
| **I-133** | 5.27E+00 | 1.41E+01 | 3.52E+01 | 6.10E+01 | 6.18E+01 |
| **Kr-85m** | 1.66E+02 | 1.34E+03 | 2.88E+03 | 3.08E+03 | 3.08E+03 |
| **Sr-89** | 8.81E+00 | 1.30E+01 | 1.35E+01 | 1.36E+01 | 1.36E+01 |
| **Ba-140** | 1.36E+01 | 1.99E+01 | 2.08E+01 | 2.08E+01 | 2.08E+01 |
| **Cs-136** | 8.61E-01 | 2.87E+00 | 9.70E+00 | 2.18E+01 | 2.51E+01 |


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**Table 3.2-4 Local Fault Event DBA**
**(Sheet 1 of 2)**

| Plenum Release | | | | | | |
|---|---|---|---|---|---|---|
| **Cumulative Activity Release to Environment (Ci)** | | | | | | |
| **Isotope** | **2hr** | **8hr** | **24hr** | **96hr** | **360hr** | **720hr** |
| **I-131** | 6.00E-02 | 8.73E-01 | 6.20E+00 | 4.35E+01 | 1.27E+02 | 1.60E+02 |
| **Xe-133** | 9.17E+00 | 1.33E+02 | 9.25E+02 | 6.07E+03 | 1.49E+04 | 1.70E+04 |
| **I-133** | 1.16E-01 | 1.51E+00 | 8.08E+00 | 2.30E+01 | 2.48E+01 | 2.48E+01 |
| **Xe-135** | 8.38E+00 | 9.29E+01 | 3.50E+02 | 5.36E+02 | 5.37E+02 | 5.37E+02 |
| **Pu-239** | 4.16E-06 | 3.40E-05 | 9.80E-05 | 1.63E-04 | 1.66E-04 | 1.66E-04 |
| **Kr-88** | 2.85E+00 | 1.80E+01 | 2.84E+01 | 2.88E+01 | 2.88E+01 | 2.88E+01 |
| **Eu-154** | 3.18E-03 | 2.60E-02 | 7.49E-02 | 1.24E-01 | 1.27E-01 | 1.27E-01 |
| **Pu-238** | 2.56E-06 | 2.09E-05 | 6.02E-05 | 1.00E-04 | 1.02E-04 | 1.02E-04 |
| **Eu-156** | 4.65E-02 | 3.77E-01 | 1.07E+00 | 1.74E+00 | 1.77E+00 | 1.77E+00 |
| **Ru-106** | 1.28E-03 | 1.05E-02 | 3.01E-02 | 5.01E-02 | 5.11E-02 | 5.11E-02 |
| **Cs-137** | 1.86E-02 | 1.52E-01 | 4.37E-01 | 7.27E-01 | 7.43E-01 | 7.43E-01 |
| **Eu-155** | 1.36E-02 | 1.11E-01 | 3.19E-01 | 5.31E-01 | 5.42E-01 | 5.42E-01 |
| **Rb-88** | 2.74E+00 | 1.95E+01 | 3.14E+01 | 3.18E+01 | 3.18E+01 | 3.18E+01 |
| **Cs-134** | 8.13E-03 | 6.63E-02 | 1.91E-01 | 3.18E-01 | 3.25E-01 | 3.25E-01 |
| **Y-91** | 7.09E-03 | 5.77E-02 | 1.66E-01 | 2.74E-01 | 2.80E-01 | 2.80E-01 |
| **I-135** | 9.87E-02 | 9.82E-01 | 2.98E+00 | 3.75E+00 | 3.75E+00 | 3.75E+00 |
| **Non-Plenum Release** | | | | | | |
| | **Cumulative Activity Release to Environment (Ci)** | | | | | |
| **Isotope** | **2hr** | **8hr** | **24hr** | **96hr** | **360hr** | **720hr** |
| **Ce-144** | 2.38E-02 | 1.97E-01 | 5.67E-01 | 9.43E-01 | 9.63E-01 | 9.63E-01 |
| **I-131** | 2.58E-01 | 2.11E+00 | 5.94E+00 | 9.46E+00 | 9.60E+00 | 9.60E+00 |
| **Cs-137** | 9.35E-02 | 7.71E-01 | 2.23E+00 | 3.71E+00 | 3.79E+00 | 3.79E+00 |
| **Ru-106** | 4.55E-03 | 3.75E-02 | 1.08E-01 | 1.80E-01 | 1.84E-01 | 1.84E-01 |
| **Cs-134** | 4.09E-02 | 3.37E-01 | 9.75E-01 | 1.62E+00 | 1.66E+00 | 1.66E+00 |
| **I-133** | 5.02E-01 | 3.71E+00 | 8.70E+00 | 1.09E+01 | 1.09E+01 | 1.09E+01 |
| **Sr-90** | 7.58E-03 | 6.25E-02 | 1.81E-01 | 3.01E-01 | 3.07E-01 | 3.07E-01 |
| **Ba-140** | 4.85E-01 | 3.97E+00 | 1.13E+01 | 1.83E+01 | 1.86E+01 | 1.86E+01 |
| **Y-91** | 2.52E-02 | 2.08E-01 | 6.02E-01 | 9.98E-01 | 1.02E+00 | 1.02E+00 |
| **Xe-133** | 3.02E+00 | 4.44E+01 | 3.15E+02 | 2.15E+03 | 5.42E+03 | 6.19E+03 |
| **La-140** | 3.94E-02 | 5.30E-01 | 2.58E+00 | 6.53E+00 | 6.82E+00 | 6.82E+00 |
| **Xe-135** | 2.82E+00 | 3.38E+01 | 1.41E+02 | 2.37E+02 | 2.37E+02 | 2.37E+02 |
| **Sr-89** | 1.05E-01 | 8.66E-01 | 2.49E+00 | 4.12E+00 | 4.20E+00 | 4.20E+00 |
| **Zr-95** | 3.02E-02 | 2.49E-01 | 7.18E-01 | 1.19E+00 | 1.21E+00 | 1.21E+00 |
| **Cs-136** | 5.52E-02 | 4.52E-01 | 1.29E+00 | 2.08E+00 | 2.12E+00 | 2.12E+00 |
| **Kr-88** | 9.32E-01 | 5.95E+00 | 9.44E+00 | 9.57E+00 | 9.57E+00 | 9.57E+00 |



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**Table 3.2-4 Local Fault Event DBA**
**(Sheet 2 of 2)**

| Non-Plenum Release | | | | | | |
|---|---|---|---|---|---|---|
| | Cumulative Activity Release to Environment (Ci) | | | | | |
| **Isotope** | **2hr** | **8hr** | **24hr** | **96hr** | **360hr** | **720hr** |
| **I-135** | 4.27E-01 | 2.52E+00 | 4.23E+00 | 4.42E+00 | 4.42E+00 | 4.42E+00 |
| **Ce-141** | 3.20E-02 | 2.64E-01 | 7.59E-01 | 1.25E+00 | 1.27E+00 | 1.27E+00 |
| **Pr-143** | 2.69E-02 | 2.22E-01 | 6.41E-01 | 1.06E+00 | 1.08E+00 | 1.08E+00 |
| **Te-132** | 2.62E-02 | 2.10E-01 | 5.72E-01 | 8.57E-01 | 8.65E-01 | 8.65E-01 |
| **Ru-103** | 2.16E-02 | 1.78E-01 | 5.11E-01 | 8.43E-01 | 8.59E-01 | 8.59E-01 |
| **Nb-95** | 3.01E-02 | 2.49E-01 | 7.19E-01 | 1.20E+00 | 1.22E+00 | 1.22E+00 |
| **Pm-147** | 4.69E-03 | 3.87E-02 | 1.12E-01 | 1.86E-01 | 1.90E-01 | 1.90E-01 |
| **Rb-88** | 9.72E-01 | 6.54E+00 | 1.05E+01 | 1.06E+01 | 1.06E+01 | 1.06E+01 |


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**Table 3.2-5 In-vessel Fuel Drop DBA, 1 Assembly**

| Plenum Release | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Cumulative Activity Release to Environment (Ci) | | | | | | | | |
| **Isotope** | **2hr** | **4hr** | **8hr** | **24hr** | **120hr** | **320hr** | **600hr** | **720hr** |
| **I-131** | 1.22E+00 | 2.44E+00 | 4.83E+00 | 1.40E+01 | 5.85E+01 | 1.11E+02 | 1.39E+02 | 1.44E+02 |
| **Xe-133** | 1.90E+02 | 3.78E+02 | 7.48E+02 | 2.14E+03 | 8.27E+03 | 1.39E+04 | 1.59E+04 | 1.61E+04 |
| **Pu-239** | 9.13E-05 | 1.33E-04 | 1.60E-04 | 1.67E-04 | 1.67E-04 | 1.67E-04 | 1.67E-04 | 1.67E-04 |
| **Eu-154** | 6.95E-02 | 1.01E-01 | 1.22E-01 | 1.27E-01 | 1.27E-01 | 1.27E-01 | 1.27E-01 | 1.27E-01 |
| **Pu-238** | 5.63E-05 | 8.18E-05 | 9.85E-05 | 1.03E-04 | 1.03E-04 | 1.03E-04 | 1.03E-04 | 1.03E-04 |
| **Eu-156** | 9.27E-01 | 1.34E+00 | 1.62E+00 | 1.69E+00 | 1.69E+00 | 1.69E+00 | 1.69E+00 | 1.69E+00 |
| **Ru-106** | 2.79E-02 | 4.05E-02 | 4.88E-02 | 5.10E-02 | 5.10E-02 | 5.10E-02 | 5.10E-02 | 5.10E-02 |
| **Cs-137** | 4.06E-01 | 5.90E-01 | 7.11E-01 | 7.42E-01 | 7.43E-01 | 7.43E-01 | 7.43E-01 | 7.43E-01 |
| **Eu-155** | 2.97E-01 | 4.31E-01 | 5.19E-01 | 5.42E-01 | 5.42E-01 | 5.42E-01 | 5.42E-01 | 5.42E-01 |
| **I-133** | 4.03E-01 | 7.80E-01 | 1.46E+00 | 3.43E+00 | 6.07E+00 | 6.18E+00 | 6.18E+00 | 6.18E+00 |
| **Cs-134** | 1.77E-01 | 2.58E-01 | 3.11E-01 | 3.24E-01 | 3.24E-01 | 3.24E-01 | 3.24E-01 | 3.24E-01 |
| **Y-91** | 1.52E-01 | 2.20E-01 | 2.65E-01 | 2.77E-01 | 2.77E-01 | 2.77E-01 | 2.77E-01 | 2.77E-01 |
| **Pu-240** | 1.24E-05 | 1.80E-05 | 2.17E-05 | 2.26E-05 | 2.26E-05 | 2.26E-05 | 2.26E-05 | 2.26E-05 |
| **Non-Plenum Release** | | | | | | | | |
| | Cumulative Activity Release to Environment (Ci) | | | | | | | |
| **Isotope** | 2hr | 4hr | 8hr | 24hr | 120hr | 320hr | 600hr | 720hr |
| **Ce-144** | 5.22E-01 | 7.61E-01 | 9.19E-01 | 9.59E-01 | 9.59E-01 | 9.59E-01 | 9.59E-01 | 9.59E-01 |
| **I-131** | 4.71E+00 | 6.85E+00 | 8.24E+00 | 8.59E+00 | 8.59E+00 | 8.59E+00 | 8.59E+00 | 8.59E+00 |
| **Cs-137** | 2.06E+00 | 3.01E+00 | 3.63E+00 | 3.79E+00 | 3.79E+00 | 3.79E+00 | 3.79E+00 | 3.79E+00 |
| **Ru-106** | 9.99E-02 | 1.46E-01 | 1.76E-01 | 1.83E-01 | 1.83E-01 | 1.83E-01 | 1.83E-01 | 1.83E-01 |
| **Cs-134** | 9.00E-01 | 1.31E+00 | 1.58E+00 | 1.65E+00 | 1.65E+00 | 1.65E+00 | 1.65E+00 | 1.65E+00 |
| **Sr-90** | 1.67E-01 | 2.44E-01 | 2.94E-01 | 3.07E-01 | 3.07E-01 | 3.07E-01 | 3.07E-01 | 3.07E-01 |
| **Ba-140** | 9.35E+00 | 1.36E+01 | 1.64E+01 | 1.71E+01 | 1.71E+01 | 1.71E+01 | 1.71E+01 | 1.71E+01 |
| **Y-91** | 5.43E-01 | 7.91E-01 | 9.54E-01 | 9.96E-01 | 9.96E-01 | 9.96E-01 | 9.96E-01 | 9.96E-01 |
| **Sr-89** | 2.24E+00 | 3.27E+00 | 3.94E+00 | 4.12E+00 | 4.12E+00 | 4.12E+00 | 4.12E+00 | 4.12E+00 |
| **Xe-133** | 6.29E+01 | 1.26E+02 | 2.49E+02 | 7.18E+02 | 2.79E+03 | 4.69E+03 | 5.36E+03 | 5.45E+03 |
| **Zr-95** | 6.50E-01 | 9.47E-01 | 1.14E+00 | 1.19E+00 | 1.19E+00 | 1.19E+00 | 1.19E+00 | 1.19E+00 |
| **I-133** | 1.56E+00 | 2.22E+00 | 2.63E+00 | 2.71E+00 | 2.71E+00 | 2.71E+00 | 2.71E+00 | 2.71E+00 |
| **Cs-136** | 1.07E+00 | 1.56E+00 | 1.87E+00 | 1.96E+00 | 1.96E+00 | 1.96E+00 | 1.96E+00 | 1.96E+00 |
| **Ce-141** | 6.73E-01 | 9.81E-01 | 1.18E+00 | 1.23E+00 | 1.23E+00 | 1.23E+00 | 1.23E+00 | 1.23E+00 |
| **La-140** | 7.45E-01 | 1.22E+00 | 1.64E+00 | 1.80E+00 | 1.80E+00 | 1.80E+00 | 1.80E+00 | 1.80E+00 |
| **I-132** | 3.91E+00 | 4.96E+00 | 5.34E+00 | 5.38E+00 | 5.38E+00 | 5.38E+00 | 5.38E+00 | 5.38E+00 |
| **Pr-143** | 5.62E-01 | 8.19E-01 | 9.87E-01 | 1.03E+00 | 1.03E+00 | 1.03E+00 | 1.03E+00 | 1.03E+00 |
| **Ru-103** | 4.56E-01 | 6.65E-01 | 8.02E-01 | 8.37E-01 | 8.37E-01 | 8.37E-01 | 8.37E-01 | 8.37E-01 |


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**Table 3.2-6 In-vessel Fuel Drop DBA, 2 Assemblies**

| Plenum Release | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Cumulative Activity Release to Environment (Ci) | | | | | | | | |
| **Isotope** | **2hr** | **4hr** | **8hr** | **24hr** | **120hr** | **320hr** | **600hr** | **720hr** |
| **I-131** | 2.44E+00 | 4.87E+00 | 9.66E+00 | 2.81E+01 | 1.17E+02 | 2.21E+02 | 2.78E+02 | 2.89E+02 |
| **Xe-133** | 3.81E+02 | 7.57E+02 | 1.50E+03 | 4.29E+03 | 1.65E+04 | 2.77E+04 | 3.17E+04 | 3.22E+04 |
| **Pu-239** | 1.83E-04 | 2.65E-04 | 3.20E-04 | 3.34E-04 | 3.34E-04 | 3.34E-04 | 3.34E-04 | 3.34E-04 |
| **Eu-154** | 1.39E-01 | 2.02E-01 | 2.43E-01 | 2.54E-01 | 2.54E-01 | 2.54E-01 | 2.54E-01 | 2.54E-01 |
| **Pu-238** | 1.13E-04 | 1.64E-04 | 1.97E-04 | 2.06E-04 | 2.06E-04 | 2.06E-04 | 2.06E-04 | 2.06E-04 |
| **Eu-156** | 1.85E+00 | 2.69E+00 | 3.24E+00 | 3.38E+00 | 3.38E+00 | 3.38E+00 | 3.38E+00 | 3.38E+00 |
| **Ru-106** | 5.58E-02 | 8.11E-02 | 9.77E-02 | 1.02E-01 | 1.02E-01 | 1.02E-01 | 1.02E-01 | 1.02E-01 |
| **Cs-137** | 8.13E-01 | 1.18E+00 | 1.42E+00 | 1.48E+00 | 1.49E+00 | 1.49E+00 | 1.49E+00 | 1.49E+00 |
| **Eu-155** | 5.93E-01 | 8.62E-01 | 1.04E+00 | 1.08E+00 | 1.08E+00 | 1.08E+00 | 1.08E+00 | 1.08E+00 |
| **I-133** | 8.06E-01 | 1.56E+00 | 2.92E+00 | 6.86E+00 | 1.21E+01 | 1.24E+01 | 1.24E+01 | 1.24E+01 |
| **Cs-134** | 3.55E-01 | 5.15E-01 | 6.21E-01 | 6.48E-01 | 6.48E-01 | 6.48E-01 | 6.48E-01 | 6.48E-01 |
| **Y-91** | 3.03E-01 | 4.40E-01 | 5.30E-01 | 5.53E-01 | 5.53E-01 | 5.53E-01 | 5.53E-01 | 5.53E-01 |
| **Pu-240** | 2.48E-05 | 3.60E-05 | 4.34E-05 | 4.53E-05 | 4.53E-05 | 4.53E-05 | 4.53E-05 | 4.53E-05 |
| **Non-Plenum Release** | | | | | | | | |
| | **Cumulative Activity Release to Environment (Ci)** | | | | | | | |
| **Isotope** | **2hr** | **4hr** | **8hr** | **24hr** | **120hr** | **320hr** | **600hr** | **720hr** |
| **Ce-144** | 1.04E+00 | 1.52E+00 | 1.84E+00 | 1.92E+00 | 1.92E+00 | 1.92E+00 | 1.92E+00 | 1.92E+00 |
| **I-131** | 9.41E+00 | 1.37E+01 | 1.65E+01 | 1.72E+01 | 1.72E+01 | 1.72E+01 | 1.72E+01 | 1.72E+01 |
| **Cs-137** | 4.12E+00 | 6.01E+00 | 7.25E+00 | 7.57E+00 | 7.57E+00 | 7.57E+00 | 7.57E+00 | 7.57E+00 |
| **Ru-106** | 2.00E-01 | 2.91E-01 | 3.51E-01 | 3.67E-01 | 3.67E-01 | 3.67E-01 | 3.67E-01 | 3.67E-01 |
| **Cs-134** | 1.80E+00 | 2.62E+00 | 3.17E+00 | 3.31E+00 | 3.31E+00 | 3.31E+00 | 3.31E+00 | 3.31E+00 |
| **Sr-90** | 3.34E-01 | 4.87E-01 | 5.88E-01 | 6.14E-01 | 6.14E-01 | 6.14E-01 | 6.14E-01 | 6.14E-01 |
| **Ba-140** | 1.87E+01 | 2.72E+01 | 3.28E+01 | 3.42E+01 | 3.42E+01 | 3.42E+01 | 3.42E+01 | 3.42E+01 |
| **Y-91** | 1.09E+00 | 1.58E+00 | 1.91E+00 | 1.99E+00 | 1.99E+00 | 1.99E+00 | 1.99E+00 | 1.99E+00 |
| **Sr-89** | 4.48E+00 | 6.54E+00 | 7.89E+00 | 8.23E+00 | 8.23E+00 | 8.23E+00 | 8.23E+00 | 8.23E+00 |
| **Xe-133** | 1.26E+02 | 2.51E+02 | 4.99E+02 | 1.44E+03 | 5.58E+03 | 9.37E+03 | 1.07E+04 | 1.09E+04 |
| **Zr-95** | 1.30E+00 | 1.89E+00 | 2.28E+00 | 2.38E+00 | 2.38E+00 | 2.38E+00 | 2.38E+00 | 2.38E+00 |
| **I-133** | 3.11E+00 | 4.45E+00 | 5.25E+00 | 5.43E+00 | 5.43E+00 | 5.43E+00 | 5.43E+00 | 5.43E+00 |
| **Cs-136** | 2.14E+00 | 3.11E+00 | 3.75E+00 | 3.91E+00 | 3.91E+00 | 3.91E+00 | 3.91E+00 | 3.91E+00 |
| **Ce-141** | 1.35E+00 | 1.96E+00 | 2.37E+00 | 2.47E+00 | 2.47E+00 | 2.47E+00 | 2.47E+00 | 2.47E+00 |
| **La-140** | 1.49E+00 | 2.44E+00 | 3.28E+00 | 3.59E+00 | 3.59E+00 | 3.59E+00 | 3.59E+00 | 3.59E+00 |
| **I-132** | 7.81E+00 | 9.92E+00 | 1.07E+01 | 1.08E+01 | 1.08E+01 | 1.08E+01 | 1.08E+01 | 1.08E+01 |
| **Pr-143** | 1.12E+00 | 1.64E+00 | 1.97E+00 | 2.06E+00 | 2.06E+00 | 2.06E+00 | 2.06E+00 | 2.06E+00 |
| **Ru-103** | 9.12E-01 | 1.33E+00 | 1.60E+00 | 1.67E+00 | 1.67E+00 | 1.67E+00 | 1.67E+00 | 1.67E+00 |


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**Table 3.2-7 Spent Fuel Release inside EVHM**

| Cumulative Activity Release to Environment (Ci) | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **Isotope** | **2hr** | **4hr** | **8hr** | **24hr** | **120hr** | **320hr** | **600hr** | **720hr** |
| **Ce-144** | 4.12E-02 | 8.25E-02 | 1.65E-01 | 4.95E-01 | 2.46E+00 | 6.50E+00 | 1.20E+01 | 1.43E+01 |
| **Cs-137** | 6.26E-02 | 1.25E-01 | 2.51E-01 | 7.53E-01 | 3.76E+00 | 1.00E+01 | 1.88E+01 | 2.26E+01 |
| **Cs-134** | 2.29E-02 | 4.58E-02 | 9.17E-02 | 2.75E-01 | 1.37E+00 | 3.65E+00 | 6.80E+00 | 8.13E+00 |
| **Sr-90** | 4.31E-03 | 8.63E-03 | 1.73E-02 | 5.18E-02 | 2.59E-01 | 6.91E-01 | 1.29E+00 | 1.55E+00 |
| **Ru-106** | 1.97E-03 | 3.95E-03 | 7.90E-03 | 2.37E-02 | 1.18E-01 | 3.12E-01 | 5.79E-01 | 6.91E-01 |
| **Pu-239** | 0.00E+00 | 1.91E-06 | 3.81E-06 | 1.14E-05 | 5.72E-05 | 1.53E-04 | 2.86E-04 | 3.43E-04 |
| **Eu-154** | 8.30E-04 | 1.66E-03 | 3.32E-03 | 9.97E-03 | 4.98E-02 | 1.33E-01 | 2.48E-01 | 2.98E-01 |
| **Pu-238** | 0.00E+00 | 1.17E-06 | 2.35E-06 | 7.05E-06 | 3.52E-05 | 9.39E-05 | 1.76E-04 | 2.11E-04 |
| **Eu-155** | 3.41E-03 | 6.83E-03 | 1.37E-02 | 4.10E-02 | 2.05E-01 | 5.45E-01 | 1.02E+00 | 1.22E+00 |


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**Table 3.2-8 Low Decay (LTA/LDA) Fuel Release inside EVHM**

| Cumulative Activity Release to Environment (Ci) | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **Isotope** | **2hr** | **4hr** | **8hr** | **24hr** | **120hr** | **320hr** | **600hr** | **720hr** |
| **Ce-144** | 6.83E-02 | 1.37E-01 | 2.74E-01 | 8.20E-01 | 4.08E+00 | 1.08E+01 | 1.99E+01 | 2.37E+01 |
| **I-131** | 1.76E-01 | 3.51E-01 | 6.97E-01 | 2.03E+00 | 8.61E+00 | 1.68E+01 | 2.17E+01 | 2.27E+01 |
| **Cs-137** | 6.35E-02 | 1.27E-01 | 2.54E-01 | 7.63E-01 | 3.81E+00 | 1.02E+01 | 1.90E+01 | 2.28E+01 |
| **Ru-106** | 2.90E-03 | 5.81E-03 | 1.16E-02 | 3.49E-02 | 1.74E-01 | 4.60E-01 | 8.52E-01 | 1.02E+00 |
| **Cs-134** | 2.77E-02 | 5.55E-02 | 1.11E-01 | 3.33E-01 | 1.66E+00 | 4.41E+00 | 8.22E+00 | 9.85E+00 |
| **Sr-90** | 4.37E-03 | 8.75E-03 | 1.75E-02 | 5.26E-02 | 2.63E-01 | 7.00E-01 | 1.31E+00 | 1.57E+00 |
| **Y-91** | 1.58E-02 | 3.16E-02 | 6.31E-02 | 1.89E-01 | 9.21E-01 | 2.34E+00 | 4.10E+00 | 4.79E+00 |
| **Ce-141** | 8.80E-02 | 1.76E-01 | 3.52E-01 | 1.05E+00 | 5.02E+00 | 1.23E+01 | 2.05E+01 | 2.35E+01 |
| **La-140** | 2.03E-02 | 4.55E-02 | 1.10E-01 | 5.24E-01 | 5.58E+00 | 1.65E+01 | 2.60E+01 | 2.85E+01 |
| **Ba-140** | 1.63E-01 | 3.26E-01 | 6.49E-01 | 1.91E+00 | 8.60E+00 | 1.86E+01 | 2.68E+01 | 2.90E+01 |
| **Sr-89** | 5.87E-02 | 1.17E-01 | 2.35E-01 | 7.01E-01 | 3.41E+00 | 8.60E+00 | 1.49E+01 | 1.73E+01 |
| **Pu-239** | 0.00E+00 | 1.90E-06 | 3.81E-06 | 1.14E-05 | 5.72E-05 | 1.53E-04 | 2.86E-04 | 3.43E-04 |
| **Zr-95** | 1.89E-02 | 3.78E-02 | 7.55E-02 | 2.26E-01 | 1.11E+00 | 2.82E+00 | 4.97E+00 | 5.81E+00 |
| **Eu-154** | 8.69E-04 | 1.74E-03 | 3.48E-03 | 1.04E-02 | 5.22E-02 | 1.39E-01 | 2.60E-01 | 3.12E-01 |
| **Pu-238** | 0.00E+00 | 1.17E-06 | 2.35E-06 | 7.04E-06 | 3.52E-05 | 9.39E-05 | 1.76E-04 | 2.11E-04 |


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**Table 3.2-9 Spent Fuel Drop in Spent Fuel Pool, 1 Assembly**

| Cumulative Activity Release to Environment (Ci) | | | | |
|---|---|---|---|---|
| **Isotope** | **0.25hr** | **0.5hr** | **1hr** | **2hr** |
| **Ce-144** | 5.40E+00 | 5.40E+00 | 5.40E+00 | 5.40E+00 |
| **Cs-137** | 4.76E+01 | 4.76E+01 | 4.76E+01 | 4.76E+01 |
| **Sr-90** | 3.58E+00 | 3.58E+00 | 3.58E+00 | 3.58E+00 |
| **Cs-134** | 1.60E+01 | 1.60E+01 | 1.60E+01 | 1.60E+01 |
| **Pu-239** | 7.95E-04 | 7.95E-04 | 7.95E-04 | 7.95E-04 |
| **Eu-154** | 6.21E-01 | 6.21E-01 | 6.21E-01 | 6.21E-01 |
| **Pu-238** | 4.88E-04 | 4.88E-04 | 4.88E-04 | 4.88E-04 |
| **Eu-155** | 2.51E+00 | 2.51E+00 | 2.51E+00 | 2.51E+00 |
| **Ru-106** | 1.37E-01 | 1.37E-01 | 1.37E-01 | 1.37E-01 |


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**Table 3.2-10 Primary Sodium Processing Leak**

| Cumulative Activity Release to Environment (Ci) | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Isotope** | **2hr** | **8hr** | **24hr** | **96hr** | **120hr** | **360hr** | **720hr** |
| **Na-22** | 1.01E-03 | 4.82E-03 | 1.45E-02 | 5.09E-02 | 6.08E-02 | 1.20E-01 | 1.47E-01 |
| **Na-24** | 4.96E+00 | 2.07E+01 | 4.53E+01 | 6.48E+01 | 6.51E+01 | 6.53E+01 | 6.53E+01 |
| **Cs-134** | 3.95E-05 | 1.88E-04 | 5.66E-04 | 1.99E-03 | 2.37E-03 | 4.68E-03 | 5.71E-03 |
| **Cs-137** | 1.18E-04 | 5.64E-04 | 1.70E-03 | 5.97E-03 | 7.13E-03 | 1.41E-02 | 1.73E-02 |


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**Table 3.2-11 Primary Sodium Processing Leak DBA**

| Cumulative Activity Release to Environment (Ci) | | | | | | | |
|---|---|---|---|---|---|---|---|
| **Isotope** | **2hr** | **8hr** | **24hr** | **96hr** | **120hr** | **360hr** | **720hr** |
| **Na-22** | 6.32E-03 | 2.38E-02 | 6.10E-02 | 1.34E-01 | 1.42E-01 | 1.55E-01 | 1.55E-01 |
| **Na-24** | 3.13E+01 | 1.04E+02 | 2.00E+02 | 2.50E+02 | 2.50E+02 | 2.50E+02 | 2.50E+02 |
| **Cs-134** | 2.47E-04 | 9.29E-04 | 2.38E-03 | 5.23E-03 | 5.55E-03 | 6.04E-03 | 6.05E-03 |
| **Cs-137** | 7.40E-04 | 2.79E-03 | 7.15E-03 | 1.57E-02 | 1.67E-02 | 1.82E-02 | 1.82E-02 |


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**Table 3.2-12 Cover Gas Release, Confinement Success**

| Cumulative Activity Release to Environment (Ci) | | | | | | |
|---|---|---|---|---|---|---|
| **Isotope** | **1hr** | **2hr** | **8hr** | **24hr** | **48hr** | **96hr** |
| **Ar-41** | 8.82E+02 | 8.87E+02 | 9.10E+02 | 9.15E+02 | 9.15E+02 | 9.15E+02 |
| **Kr-88** | 8.32E+01 | 8.37E+01 | 8.77E+01 | 8.98E+01 | 8.99E+01 | 8.99E+01 |
| **Xe-135** | 2.98E+02 | 3.01E+02 | 3.33E+02 | 3.99E+02 | 4.24E+02 | 4.28E+02 |
| **Xe-138** | 7.18E+01 | 7.19E+01 | 7.19E+01 | 7.19E+01 | 7.19E+01 | 7.19E+01 |
| **Cs-138** | 1.92E+01 | 1.92E+01 | 1.92E+01 | 1.92E+01 | 1.92E+01 | 1.92E+01 |
| **Kr-87** | 5.60E+01 | 5.63E+01 | 5.70E+01 | 5.71E+01 | 5.71E+01 | 5.71E+01 |
| **Rb-88** | 3.20E+01 | 3.20E+01 | 3.23E+01 | 3.24E+01 | 3.24E+01 | 3.24E+01 |


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**Table 3.2-13 Cover Gas Release, Confinement Bypass**

| Cumulative Activity Release to Environment (Ci) | | | | | | |
|---|---|---|---|---|---|---|
| **Isotope** | **1hr** | **2hr** | **4hr** | **8hr** | **12hr** | **24hr** |
| **Ar-41** | 1.55E+03 | 2.46E+03 | 3.30E+03 | 3.68E+03 | 3.73E+03 | 3.73E+03 |
| **Kr-88** | 1.51E+02 | 2.52E+02 | 3.64E+02 | 4.37E+02 | 4.51E+02 | 4.55E+02 |
| **Xe-135** | 5.63E+02 | 1.01E+03 | 1.64E+03 | 2.28E+03 | 2.53E+03 | 2.68E+03 |
| **Kr-87** | 9.54E+01 | 1.43E+02 | 1.77E+02 | 1.88E+02 | 1.89E+02 | 1.89E+02 |
| **Xe-138** | 8.70E+01 | 9.09E+01 | 9.11E+01 | 9.11E+01 | 9.11E+01 | 9.11E+01 |
| **Xe-133** | 5.80E+02 | 1.07E+03 | 1.84E+03 | 2.80E+03 | 3.29E+03 | 3.74E+03 |


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**Table 3.2-14 Cover Gas Release DBA**

| Cumulative Activity Release to Environment (Ci) | | | | |
|---|---|---|---|---|
| **Isotope** | **0.5hr** | **1hr** | **2hr** | **4hr** |
| **Ar-41** | 7.97E+03 | 1.01E+04 | 1.07E+04 | 1.08E+04 |
| **Kr-88** | 7.47E+02 | 9.56E+02 | 1.03E+03 | 1.04E+03 |
| **Xe-135** | 2.66E+03 | 3.47E+03 | 3.79E+03 | 3.83E+03 |
| **Xe-138** | 7.18E+02 | 7.70E+02 | 7.74E+02 | 7.74E+02 |
| **Cs-138** | 1.47E+02 | 2.24E+02 | 2.43E+02 | 2.44E+02 |
| **Kr-87** | 5.09E+02 | 6.32E+02 | 6.69E+02 | 6.71E+02 |
| **Rb-88** | 2.39E+02 | 4.17E+02 | 4.96E+02 | 5.03E+02 |


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**Table 3.2-15 Release from RWG Holdup Tank, Release via FHB exhaust**

| Cumulative Activity Release to Environment (Ci) | | | | | |
|---|---|---|---|---|---|
| **Isotope** | **0.5hr** | **1hr** | **2hr** | **4hr** | **8hr** |
| **Xe-135** | 2.10E+02 | 2.99E+02 | 3.52E+02 | 3.63E+02 | 3.63E+02 |
| **Xe-133** | 1.32E+03 | 1.90E+03 | 2.26E+03 | 2.34E+03 | 2.34E+03 |
| **Kr-88** | 1.12E+01 | 1.55E+01 | 1.79E+01 | 1.83E+01 | 1.83E+01 |
| **Ar-41** | 3.22E+00 | 4.39E+00 | 4.97E+00 | 5.05E+00 | 5.05E+00 |
| **Kr-87** | 3.26E+00 | 4.35E+00 | 4.83E+00 | 4.89E+00 | 4.89E+00 |
| **Kr-85m** | 1.55E+01 | 2.18E+01 | 2.54E+01 | 2.61E+01 | 2.62E+01 |
| **Xe-133m** | 4.31E+01 | 6.18E+01 | 7.36E+01 | 7.62E+01 | 7.63E+01 |

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**Table 3.2-16 Release from RWG Holdup Tank, Release via RWG**

| Cumulative Activity Release to Environment (Ci) | | | | | |
|---|---|---|---|---|---|
| **Isotope** | **0.125hr** | **0.5hr** | **1hr** | **1.5hr** | **2hr** |
| **Xe-135** | 2.00E+02 | 3.57E+02 | 3.74E+02 | 3.75E+02 | 3.75E+02 |
| **Xe-133** | 1.24E+03 | 2.23E+03 | 2.34E+03 | 2.35E+03 | 2.35E+03 |
| **Kr-88** | 1.09E+01 | 1.93E+01 | 2.01E+01 | 2.02E+01 | 2.02E+01 |
| **Ar-41** | 3.21E+00 | 5.60E+00 | 5.83E+00 | 5.84E+00 | 5.84E+00 |
| **Kr-87** | 3.33E+00 | 5.74E+00 | 5.95E+00 | 5.96E+00 | 5.96E+00 |
| **Kr-85m** | 1.50E+01 | 2.66E+01 | 2.78E+01 | 2.79E+01 | 2.79E+01 |
| **Xe-133m** | 4.06E+01 | 7.29E+01 | 7.65E+01 | 7.67E+01 | 7.67E+01 |
| **Xe-138** | 7.29E-01 | 1.07E+00 | 1.08E+00 | 1.08E+00 | 1.08E+00 |


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**Table 3.2-17 Release from RWG Holdup Tank, Release via RWG DBA**

| Cumulative Activity Release to Environment (Ci) | | | | | |
|---|---|---|---|---|---|
| **Isotope** | **0.125hr** | **0.5hr** | **1hr** | **1.5hr** | **2hr** |
| **Xe-135** | 2.00E+02 | 3.57E+02 | 3.74E+02 | 3.75E+02 | 3.75E+02 |
| **Xe-133** | 1.24E+03 | 2.23E+03 | 2.34E+03 | 2.35E+03 | 2.35E+03 |
| **Kr-88** | 1.09E+01 | 1.93E+01 | 2.01E+01 | 2.02E+01 | 2.02E+01 |
| **Rb-88** | 1.10E+01 | 1.98E+01 | 2.07E+01 | 2.07E+01 | 2.07E+01 |
| **Ar-41** | 3.21E+00 | 5.60E+00 | 5.83E+00 | 5.84E+00 | 5.84E+00 |
| **Kr-87** | 3.33E+00 | 5.74E+00 | 5.95E+00 | 5.96E+00 | 5.96E+00 |
| **Kr-85m** | 1.50E+01 | 2.66E+01 | 2.78E+01 | 2.79E+01 | 2.79E+01 |
| **Cs-138** | 8.64E-01 | 1.50E+00 | 1.55E+00 | 1.55E+00 | 1.55E+00 |
| **Xe-133m** | 4.06E+01 | 7.29E+01 | 7.65E+01 | 7.67E+01 | 7.67E+01 |


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**Table 3.2-18 Release of Tritium from SPS-I Cold Trap**

| Cumulative Activity Release to Environment (Ci) | | | | | |
|---|---|---|---|---|---|
| **Isotope** | **4hr** | **8hr** | **12hr** | **24hr** | **30hr** |
| **H-3** | 4.15E+03 | 8.87E+03 | 1.36E+04 | 2.77E+04 | 2.83E+04 |



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**Table 3.2-19 Unprotected Loss of Heat Sink, Full Core and IVS Failure**

| Cumulative Activity Release to Environment (Ci) | | | | | | |
|---|---|---|---|---|---|---|
| **Isotope** | **12hr** | **24hr** | **96hr** | **168hr** | **504hr** | **720hr** |
| **Xe-133** | 0.00E+00 | 2.64E+03 | 1.64E+05 | 4.48E+05 | 1.29E+06 | 1.43E+06 |
| **I-131** | 0.00E+00 | 6.55E-01 | 1.49E+01 | 3.87E+01 | 1.33E+02 | 1.60E+02 |
| **Cs-137** | 0.00E+00 | 7.13E-01 | 7.83E+00 | 1.49E+01 | 4.59E+01 | 6.42E+01 |
| **Xe-135** | 0.00E+00 | 5.99E+02 | 3.52E+03 | 3.57E+03 | 3.57E+03 | 3.57E+03 |
| **Cs-134** | 0.00E+00 | 2.88E-01 | 2.88E+00 | 5.41E+00 | 1.64E+01 | 2.28E+01 |


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## 3.3 Licensing Methods for Evaluation of Licensing Basis Events

The methodologies used in the evaluation of licensing basis events (LBE) include a description of the computer programs and the calculation framework selected for use with a specific set of transients or accidents, and provide information such as the mathematical models used, assumptions included in the programs, and procedures for treating program inputs and outputs. The methodologies also include required assumptions about plant equipment availability and any other information necessary to specify the calculation procedure.

The methodologies discussed in this section are preliminary. Specific areas requiring further development are described in more detail in the respective sections.

### 3.3.1 Licensing Methodology for Design Basis Accidents

Bounding parameters are selected for use in evaluating design basis accidents (DBAs), with preliminary sensitivity analyses and analytical results to show the chosen parameters and assumptions to be conservative. Final parameters and results will be provided at the operating license stage.

#### 3.3.1.1 In-Vessel DBAs without Radiological Release

The methodology used to evaluate DBAs which affect core assemblies and which screen out for potential for radiological release is documented TP-LIC-RPT-0004, “Design Basis Accident Methodology for In-Vessel Events without Radiological Release,” Revision 0 (Reference 3.3-1). Sections 1 through 8 of TP-LIC-RPT-0004 are incorporated by reference. See Section 1.4.2 for a list of documents incorporated by reference.

Assessments to evaluate model fidelity and accuracy, scalability of the models, and acceptance of the models are in process, and are captured in Section 5 of TP-LIC-RPT-0004.

Requirements to determine overall model adequacy following these assessments are discussed in Sections 5 and 7 of TP-LIC-RPT-0004.

Detailed discussion of the codes selected for use in this method and other codes supporting their use are captured in Section 4 of TP-LIC-RPT-0004.

#### 3.3.1.2 Ex-Vessel DBAs and DBAs with Potential Radiological Release

The methodology used to evaluate DBAs which affect SSCs outside of the reactor core is documented in TP-LIC-RPT-0007, “Design Basis Accident Methodology Report for Potential Fuel Failure and Release,” Revision 0 (Reference 3.3-2). Sections 1 through 7 of TP-LIC-RPT-0007 are incorporated by reference into the SAR. This methodology also builds upon the models described in Section 3.3.1.1 to evaluate in-vessel DBAs which screen in for potential radiological release.

Assessments to evaluate model fidelity and accuracy, scalability of the models, and acceptance of the models are in process, and are captured in Sections 5 and 6 of TP-LIC-RPT-0007.


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Requirements to determine overall model adequacy following these assessments are also discussed in Section 6 of TP-LIC-RPT-0007.

Detailed discussion of the codes selected for use in this method and other codes supporting their use are captured in Sections 5 and 6 of TP-LIC-RPT-0007.

#### 3.3.1.3 Partial Flow Blockage DBAs

A partial flow blockage in a fuel assembly is an event selected for evaluation based on the design and operating characteristics of the assemblies in a sodium-cooled fast reactor, namely the tight spacing of fuel pins, high power density, and high burnup. A partial flow blockage may be initiated due to an accumulation of debris circulated in the primary sodium, a failure of wire-wrapped spacers, and from swelling or bowing of the fuel pins.

The methodology used to evaluate partial flow blockage DBAs is documented in TP-LIC-RPT-0008, “Partial Flow Blockage Methodology,” Revision 0 (Reference 3.3-3). Sections 1 through 7 of TP-LIC-RPT-0008 are incorporated by reference.

The methodology described in TP-LIC-RPT-0008 has been developed to apply to all partial flow blockage LBEs. DBA specific considerations are discussed in Section 5 of TP-LIC-RPT-0008.

Assessments to evaluate model fidelity and accuracy, scalability of the models, and acceptance of the models are in process, and are captured in Section 5.4 of TP-LIC-RPT-0008.

Requirements to determine overall model adequacy following these assessments are discussed in Section 6 of TP-LIC-RPT-0008.

Detailed discussion of the codes selected for use in this method and other codes supporting their use are captured in Section 5.3 of TP-LIC-RPT-0008.

#### 3.3.1.4 Radiological Consequences

The methodology used to evaluate radiological consequences of DBAs is documented in TP-LIC-RPT-0005, “Radiological Release Consequences Methodology,” Revision 0 (Reference 3.3-4). Sections 1 through 6 of TP-LIC-RPT-0005 are incorporated by reference. Detailed discussion of the methodology developed for determining the dose consequences of a DBA is captured in Section 4 of TP-LIC-RPT-0005.

### 3.3.2 Licensing Methodology for AOO, DBE, and BDBE

#### 3.3.2.1 AOOs, DBEs, and BDBEs

The licensing methodologies used to evaluate anticipated operational occurrences (AOOs), design basis events (DBEs), and beyond design basis events (BDBEs) are similar to the methods described in Section 3.3.1. Where only safety-related SSCs are considered for mitigation in DBA analyses, AOO, DBE, and BDBE analyses take into account the expected response of all SSCs within the plant regardless of safety classification. Additionally, when


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analyzing AOO, DBE, and BDBE transients, either the same conservative approach used for DBAs will be used, or a best-estimate plus uncertainty approach may be used as an alternative method to evaluate those events.

The codes selected for use in these methodologies match those described in the related DBA methodology for Section 3.3.1.

#### 3.3.2.2 Radiological Consequences

The methodology used to evaluate radiological consequences of AOOs, DBEs, and BDBEs is documented in TP-LIC-RPT-0005. Detailed discussion of the methodology developed for determining the dose consequences of these events, and the related uncertainty treatment, is captured in Section 3 of TP-LIC-RPT-0005.

### 3.3.3 Licensing Methodology for Major Accident

The Major Accident is a deterministic analysis that meets the requirement of 10 CFR 50.34(a)(1)(ii)(D). A fission product release from the reactor core into functional containment is analyzed deterministically with consideration for the expected demonstrable leak rate, fission product cleanup systems that mitigate the consequences of the accident, and applicable site characteristics and meteorology. The offsite radiological consequences comply with 10 CFR 50.34 dose limits. The type of containment is a functional containment as described in Section 1.3.2.1.

The Major Accident is postulated from considerations of possible accidental events that would release an appreciable quantity of fission products into the functional containment. Because no DBA will result in release of an appreciable quantity of fission products, a major accident event was developed as described in TP-LIC-RPT-0010, “Major Accident Methodology,” Revision 0 (Reference 3.3-5). Sections 1 through 6 of TP-LIC-RPT-0010 are incorporated by reference. The postulated event results in an unprotected loss of heat sink, with the resulting fuel damage leading to the highest core wide risk release while maintaining an intact functional containment.

#### 3.3.3.1 Radiological Consequences

Evaluation of the event is captured in Section 4.3 of TP-LIC-RPT-0010, with a resulting consequence at the 5th percentile, 95th percentile, and mean 30-day total effective dose equivalent at the exclusion area boundary (EAB) being 1.21E-01 rem, 5.22E-01 rem, and 1.79E-01 rem respectively. The mechanistic source term is summarized in Table 3.2-19.

### 3.3.4 LBE Plant Response and Analysis Overview

This section summarizes the general plant response to LBE initiating events, the initial conditions assumed in LBE analyses, and the LBE analysis end states.


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#### 3.3.4.1 LBE General Plant Response

LBEs are presented in groups based on similar types of initiating events found in Section 3.6 through Section 3.9. The general plant response to these types of initiating events is described in this section. The LBE descriptions in PSAR Section 3.6 through Section 3.9 describe successes and failures of PRA safety functions. The assumption that a function fails in the LBE description does not necessarily mean the failure was a result of the LBE initiating event.

##### 3.3.4.1.1 Increase in Heat Removal

An increase in heat removal from the primary system results in a decrease in primary sodium temperature, which causes an increase in reactor power. Reactor power continues to increase until either a reactor power runback or scram occurs. If a reactor power runback successfully completes and prevents a scram setpoint from being reached, the event is not classified as an LBE. The PSPs and ISPs may reduce speed due to a reactor power runback, and the PSPs and ISPs will reduce speed if a scram occurs. The decay heat removal is provided by RAC and available modes of IAC. If the high high primary sodium temperature PSP and ISP trip setpoints are reached following the scram, the operating PSPs and ISPs trip. If PSPs trip, the primary sodium temperature may increase after the PSP coast down is complete as the core transitions to natural circulation cooling, then decreases after natural circulation cooling is established and decay heat decreases.

##### 3.3.4.1.2 Decrease in Heat Removal

A decrease in heat removal from the primary system results in an increase in primary sodium temperature, which causes a decrease in reactor power. The primary sodium temperature continues to increase until either a reactor power runback or scram occurs. If a reactor power runback successfully completes and prevents a scram setpoint from being reached, the event is not classified as an LBE. The PSPs and ISPs may reduce speed due to a reactor power runback, and the PSPs and ISPs will reduce speed if a scram occurs. The decay heat removal is provided by RAC and available modes of IAC. If the high high primary sodium temperature PSP and ISP trip setpoints are reached, the operating PSPs and ISPs trip. If PSPs trip, the primary sodium temperature may increase after the PSP coast down is complete as the core transitions to natural circulation cooling, then decreases after natural circulation cooling is established and decay heat decreases.

##### 3.3.4.1.3 Reactivity

A malfunction of the reactivity control system, such as the inadvertent withdrawal of a control rod, results in a positive reactivity excursion and a scram on neutron flux, power-to-flow ratio, or flux rate-of-change. Spurious scrams and reactivity insertion due to the safe shutdown earthquake are also included in this group. The PSPs and ISPs will reduce speed if a scram occurs. The decay heat removal is provided by RAC and available modes of IAC. If the high high primary sodium temperature PSP and ISP trip setpoints are reached, the operating PSPs and ISPs trip. If the PSPs trip the primary sodium temperature may increase after the PSP coast down is complete as the core transitions to natural circulation cooling, then decreases after natural circulation cooling is established and decay heat decreases.


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#### 3.3.4.1.4 Loss of Primary Flow

The loss of one or both PSPs causes a reduction in the primary sodium flow that results in a scram on loss of flow or high high power-to-flow ratio. The primary sodium temperature increases due to the core flow reduction and decreases after the reactor scrams. The PSPs and ISPs will reduce speed if a scram occurs. The decay heat removal is provided by RAC and available modes of IAC. If the high high primary sodium temperature PSP and ISP trip setpoints are reached, the remaining operating PSPs and ISPs trip. The primary sodium temperature may increase after the PSP coast down is complete as the core transitions to natural circulation cooling, then decreases after natural circulation cooling is established and decay heat decreases.

#### 3.3.4.1.5 Local Fuel Fault

A localized fault, such as a localized blockage of fuel subchannels, occurs in the reactor core during full power operating conditions. The localized fault results in fuel damage limited to one fuel assembly but does not cause the primary sodium temperature to increase outside of nominal operating conditions. Operators may scram the reactor if required due to failed fuel activity, but a scram is not required to maintain the LBE consequences within analyzed limits. The PSPs and ISPs will reduce speed if a scram occurs. The assumption that an entire fuel assembly failed due to the event is conservative and deterministic with no detailed transient core response analysis performed for the local fuel fault events.

#### 3.3.4.1.6 Release From Ex-Vessel Systems

These LBEs evaluate radionuclide release from the SCG, SPS, and RWG.

#### 3.3.4.1.7 Fuel Handling

These LBEs evaluate loss of heat removal from systems and vessels containing spent fuel assemblies and fuel assembly drops in the reactor vessel, SFP, EVST, and other fuel handling vessels. Releases from vessels containing damaged spent fuel assemblies in the FHB are typically modeled as releases from the EVHM. A release modeled from the EVHM is bounding because the EVHM has the smallest free volume and the EVHM assumed leakage rate is conservative compared to the expected leakage rates from the BLTC, PRC, EVST, and PIC.

### 3.3.4.2 LBE Analysis Initial Conditions

The core models were developed based on the following steady-state conditions, which define the plant conditions assumed for low power and full power operation used in the LBE analyses:

| Plant Condition | Reactor Power (%) | Core Flow (%) | Core Inlet Temperature (°C) |
|---|---|---|---|
| Low Power | 5 | 15 | 309.5 |
| Full Power | 100 | 100 | 360.1 |

For LBEs that occur during multiple initial conditions, full power or low power initial conditions were analyzed, with shutdown and refueling initial conditions being covered by the analyses performed at low power initial conditions.


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Where LBEs are evaluated at shutdown or refueling initial conditions, the reactor is assumed to be subcritical and at zero power, and the configuration of functional containment is captured per event.

3.3.4.3 LBE Analysis End States

LBE analysis results in either safe shutdown conditions or safe and stable conditions. The safe shutdown LBE end state is defined by the analysis termination criteria below:

- Peak Cladding Temperature value below 1157°F (625°C) and is stable or decreasing
- K-effective 1.0 (for BDBE only) or less and core power stable or reducing

For transients that rely on passive long-term cooling to meet the acceptance criteria (not applicable for short-term transients), the end state shall also include the additional criterion below:

- Heat removal exceeds total heat produced (fission, decay, and equipment produced)

If the transient does not result in a reactor scram the safe and stable end state is used. The safe and stable LBE end state is defined by the analysis termination criteria below:

- K-effective 1.0
- Heat removal is equal to heat production

**References**

3.3-1 TerraPower, LLC “Design Basis Accident Methodology for In-Vessel Events without Radiological Release,” TP-LIC-RPT-0004, Revision 0

3.3-2 TerraPower, LLC “Design Basis Accident Methodology Report for Potential Fuel Failure and Release,” TP-LIC-RPT-0007, Revision 0

3.3-3 TerraPower, LLC “Partial Flow Blockage Methodology,” TP-LIC-RPT-0008, Revision 0

3.3-4 TerraPower, LLC “Radiological Release Consequences Methodology,” TP-LIC-RPT-0005, Revision 0

3.3-5 TerraPower, LLC “Major Accident Methodology,” TP-LIC-RPT-0010, Revision 0


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## 3.4 Licensing Basis Event Selection Methodology

The NEI 18-04 methodology, as endorsed by Regulatory Guide 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” Revision 0, is implemented without deviation for licensing basis event (LBE) selection and evaluation. The selection of LBEs includes the systematic definition, categorization, and evaluation of event sequences. The phenomena caused by the design basis seismic hazard are not covered by other DBA evaluations, so a seismic DBA was developed.


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# 3.5 LBE Summary

## 3.5.1 Summary Evaluation of AOOs, DBEs, and BDBEs

Table 3.5-1, Table 3.5-2, and Table 3.5-3 present summaries of the anticipated operational occurrences (AOOs), design basis events (DBEs), and beyond design basis events (BDBEs), respectively. Descriptions of the AOO, DBE, and BDBE evaluations, identification of radionuclide sources, risk-significant licensing basis events (LBEs), high-consequence BDBEs, and the event end states are provided in Sections 3.6 through 3.8. LBEs are classified as risk-significant if the LBE site boundary dose exceeds 2.5 mrem over 30 days and the frequency of the dose is within one percent of the F-C target. BDBEs are classified as high-consequence if the dose exceeds 10 CFR 50.34 criteria. High-consequence BDBEs are identified in Table 3.5-3.

Plots of the frequencies, consequences, and uncertainties of the LBEs, with comparison to the NEI 18-04 F-C target, are shown in Figure 3.5-1.

## 3.5.2 Summary Evaluation of DBAs

Table 3.5-4 presents a summary of the design basis accidents (DBAs). The table provides a mapping of DBEs to DBAs and the dose consequence of each DBA. The calculated dose consequences of the DBAs are within the dose limits defined in 10 CFR 50.34(a)(1)(ii)(D).

Descriptions of DBA evaluations are included in Section 3.9.


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**Table 3.5-1 Summary of AOOs**

| AOO ID | AOO Title | Risk Significant |
|---|---|---|
| DHP-L1PP-BL | Loss of One Primary Sodium Pump with Non-Passive IAC | No |
| DHP-LOOP-BL | Loss of Offsite Power with Non-Passive IAC | No |
| DHS-ISTL-BL | Loss of Heat Sink with Non-Passive IAC | No |
| OTH-LMAC-BL | Loss of a Single Medium Voltage AC Bus with Non-Passive IAC | No |
| RFH-LSPC-BL | Loss of SFP Cooling with Cooling Restored | No |
| RFH-LTCA-BL | Loss of EVST Active Cooling While Storing Fuel Assemblies with Passive Cooling | No |
| RFH-OERC-BL | Fuel Damage While Handling an LTA or LDA Test Pin with PRC Barrier Successfully Retains Release | No |
| RPD-CW1ACS-BL | Control Rod-Induced Transient Overpower with Non-Passive IAC | No |
| RPD-SS-BL | Reactor Scram or Spurious Scram with Non-Passive IAC | No |
| SUD-IACA-BL | Loss of One Train of IAC While Shutdown with Non-Passive IAC | No |
| SUD-LOOP-BL | Loss of Offsite Power While at Low Power with Non-Passive IAC | No |


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**Table 3.5-2 Summary of DBEs**
**(Sheet 1 of 2)**

| DBE ID | DBE Title | Risk Significant |
|---|---|---|
| DHP-L1PP-2 | Loss of One Primary Sodium Pump with RAC | No |
| DHP-LOOP-1 | Loss of Offsite Power with Passive IAC | No |
| DHS-ISTL-1 | Loss of Heat Sink with Passive IAC | No |
| DHS-ISTL-2 | Loss of Heat Sink with RAC | No |
| DHS-RNBK-1 | Energy Island Transient without Reactor Power Runback, with Non-Passive IAC | No |
| DHS-RNBK-3 | Energy Island Transient without Reactor Power Runback, with RAC | No |
| IPI-IHEL-BL | Intermediate Heat Exchanger Secondary-to-Primary Leak with Non-Passive IAC | No |
| LFF-SAO-BL | Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Successfully Retains Release (Vessel Head Success) | No |
| RFH-ESWR-BL | Excessive Sodium-Water Reaction in the PIC | No |
| RFH-ESWR-1 | Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Successfully Retains Release | No |
| RFH-FDIV-BL | Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with No Damage | No |
| RFH-FDIV-1 | Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with a Single Assembly Failed and Functional Containment Barriers Successfully Retain Release | Yes |
| RFH-FDIV-3 | Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with Two Assemblies Failed and Functional Containment Barriers Successfully Retain Release | Yes |
| RFH-FDSP-1 | Fuel Handling Event Occurs While Moving Fuel Assembly in the SFP with a Single Assembly Failed and FHB Barrier Fails to Retain Release | Yes |
| RFH-LBCA-BL | Loss of BLTC Active Cooling While Handling Fuel Assembly with Passive Cooling | No |
| RFH-LMCA-BL | Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA with Passive Cooling | No |
| RFH-LTCA-1 | Loss of EVST Active Cooling While Storing Fuel Assemblies without Passive Cooling | No |
| RFH-OEFH-BL | Removal of Fuel Assembly Prior to Decay with Active Cooling | No |
| RRS-CGR-BL | SCG Leak Inside the HAA where RXB Substructure Successfully Retains Release | No |


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**Table 3.5-2 Summary of DBEs**
**(Sheet 2 of 2)**

| DBE ID | DBE Title | Risk Significant |
|---|---|---|
| RRS-CGR-1 | SCG Leak Downstream the SCG Cell where RXB Substructure Successfully Retains Release | No |
| RRS-ISPL-BL | SPS-I Leak at the Cold Trap with Reactor Auxiliary Building Barrier Fails to Retain Release | No |
| RRS-SPLX-BL | SPS-P Leak in the RXB | No |
| RRS-SPLA-BL | SPS-P Leak in the Reactor Auxiliary Building | No |
| RRS-RWG-1 | RWG Leak from the Holdup Tank with FHB Barrier Fails to Retain Release | No |
| RRS-RWG-2 | RWG Leaks to the Stack to the Environment | No |
| SUD-IACA-1 | Loss of One Train of IAC While Shutdown with Passive IAC | No |
| SUD-IHEL-BL | Intermediate Heat Exchanger Secondary-to-Primary Leak While Shutdown with Non-Passive IAC | No |
| SUD-LOOP-1 | Loss of Offsite Power While at Low Power with Passive IAC | No |


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**Table 3.5-3 Summary of BDBEs**
**(Sheet 1 of 3)**

| BDBE ID | BDBE Title | Risk Significant |
|---|---|---|
| DHP-L1PP-1 | Loss of One Primary Sodium Pump with Passive IAC | No |
| DHP-L1PP-3 | Loss of One Primary Sodium Pump with Scram Motor Drive-In | No |
| DHP-L1PP-4 | Loss of One Primary Sodium Pump with Alternative Shunt Trip | No |
| DHP-LAPP-BL | Loss of All Primary Pumps with Non-Passive IAC | No |
| DHP-LOOP-2 | Loss of Offsite Power with RAC | No |
| DHP-LOOP-3 | Loss of Offsite Power with Scram Motor Drive-In | No |
| DHP-LOOP-4 | Loss of Offsite Power with Alternative Shunt Trip | No |
| DHS-ISTL-3 | Loss of Heat Sink with Scram Motor Drive-In | No |
| DHS-ISTL-4 | Loss of Heat Sink with Alternative Shunt Trip | No |
| DHS-RNBK-2 | Energy Island Transient without Reactor Power Runback, with Passive IAC | No |
| IPI-IHEL-1 | Intermediate Heat Exchanger Secondary-to-Primary Leak with Passive IAC | No |
| IPI-IHEL-2 | Intermediate heat Exchanger Secondary-to-Primary Leak with RAC | No |
| LFF-SAO-1 | Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Fails to Retain Release (Vessel Head Success) | No |
| LFF-SAO-2 | Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Successfully Retains Release (Vessel Head Failed) | No |
| OTH-LMAC-1 | Loss of a Single Medium Voltage AC Bus with Passive IAC | No |
| OTH-LMAC-2 | Loss of a Single Medium Voltage AC Bus with RAC | No |
| OTH-LMAC-3 | Loss of a Single Medium Voltage AC Bus with Scram Motor Drive-In | No |
| OTH-LMAC-4 | Loss of a Single Medium Voltage AC Bus with Alternative Shunt Trip | No |
| RFH-ESWR-2 | Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Fails to Retain Release (FHB Credited) | Yes |
| RFH-ESWR-3 | Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Fails to Retain Release (FHB Not Credited) | Yes |
| RFH-FDBL-1 | Fuel Handling Event Occurs While Moving Fuel Assembly in the BLTC with BLTC Barrier Successfully Retains Release | No |
| RFH-FDBL-2 | Fuel Handling Event Occurs While Moving Fuel Assembly in the BLTC with BLTC Barrier Fails to Retain Release | Yes |


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**Table 3.5-3 Summary of BDBEs**
**(Sheet 2 of 3)**

| BDBE ID | BDBE Title | Risk Significant |
|---|---|---|
| RFH-FDEM-1 | Fuel Handling Event Occurs While Moving Fuel Assembly or an LTA or LDA in the EVHM with EVHM Barrier Successfully Retains Release | No |
| RFH-FDEM-2* | Fuel Handling Event Occurs While Moving Fuel Assembly or an LTA or LDA in the EVHM with EVHM Barrier Fails to Retain Release | Yes |
| RFH-FDET-BL | Fuel Handling Event Occurs While Moving Fuel Assembly in the EVST with No Damage | No |
| RFH-FDET-1 | Fuel Handling Event Occurs While Moving Fuel Assembly in the EVST with a Single Assembly Failed and EVST Barrier Successfully Retains Release | No |
| RFH-FDIV-2 | Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with a Single Assembly Failed and Functional Containment Barriers Fail to Retain Release | Yes |
| RFH-FDIV-4 | Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with Two Assemblies Failed and Functional Containment Barriers Fail to Retain Release | No |
| RFH-FDPI-1 | Fuel Handling Event Occurs While Moving Fuel Assembly in the PIC with PIC Barrier Fails to Retain Release | Yes |
| RFH-FDPI-BL | Fuel Handling Event Occurs While Moving Fuel Assembly in the PIC with PIC Barrier Successfully Retains Release | No |
| RFH-FDSP-2 | Fuel Handling Event Occurs While Moving Fuel Assembly in the SFP with Two Assemblies Failed and FHB Barrier Fails to Retain Release | Yes |
| RFH-FDRC-1 | Fuel Handling Event Occurs While Moving Fuel Assembly in the Pin Removal Cell with Pin Removal Cell Barrier Successfully Retains Release | No |
| RFH-LMCA-1 | Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA without Passive Cooling, EVHM Barrier Successfully Retains Release | No |
| RFH-LMCA-2* | Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA without Passive Cooling, EVHM Barrier Fails to Retain Release | Yes |
| RFH-LSPC-1 | Loss of SFP Cooling with Makeup Restored | No |
| RPD-CW1ACS-1 | Control Rod-Induced Transient Overpower with Passive IAC | No |
| RPD-CW1ACS-2 | Control Rod-Induced Transient Overpower with RAC | No |
| RPD-CW1ACS-3 | Control Rod-Induced Transient Overpower with Scram Motor Drive-In | No |


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**Table 3.5-3 Summary of BDBEs**
**(Sheet 3 of 3)**

| BDBE ID | BDBE Title | Risk Significant |
|---|---|---|
| RPD-CW1ACS-4 | Control Rod-Induced Transient Overpower with Alternative Shunt Trip | No |
| RPD-SS-1 | Reactor Scram or Spurious Scram with Passive IAC | No |
| RPD-SS-2 | Reactor Scram or Spurious Scram with RAC | No |
| RPD-SS-3 | Reactor Scram or Spurious Scram with Scram Motor Drive-in | No |
| RPD-SS-4 | Reactor Scram or Spurious Scram with Alternative Shunt Trip | No |
| SUD-CGR-1 | SCG Leak Inside the Head Access Area where RXB Substructure Fails to Retain Release While Shutdown | No |
| SUD-CGR-2 | SCG Leak Downstream the SCG Cell where RXB Substructure Fails to Retain Release While Shutdown | No |
| SUD-IACA-2 | Loss of One Train of IAC While Shutdown with RAC | No |
| SUD-IHEL-1 | Intermediate Heat Exchanger Secondary-to-Primary Leak While Shutdown with Passive IAC | No |

*Denotes high-consequence BDBE


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**Table 3.5-4 Summary of DBAs**
**(Sheet 1 of 2)**

| DBE ID | DBA ID | DBA Title | DBA 2-hour Exclusion Area Boundary TEDE (rem) | DBA 30-day Low Population Zone Boundary TEDE (rem) |
|---|---|---|---|---|
| DHP-L1PP-2 | DHP-L1PP-CN | Loss of One Primary Sodium Pump | No Release | No Release |
| DHP-LOOP-1 | DHP-LOOP-CN | Loss of Offsite Power | No Release | No Release |
| DHS-ISTL-1 | DHS-ISTL-CN | Loss of Heat Sink | No Release | No Release |
| DHS-ISTL-2 | | | | |
| DHS-RNBK-1 | DHS-RNBK-CN | Energy Island Transient | No Release | No Release |
| DHS-RNBK-3 | | | | |
| IPI-IHEL-BL | IPI-IHEL-CN | Intermediate Heat Exchanger Secondary-to-Primary Leak | No Release | No Release |
| LFF-SAO-BL | LFF-SAO-CN | Core Blockage and Local Faults | 1.33E-01 | 9.60E-01 |
| RFH-ESWR-BL | RFH-ESWR-CN | Excessive Sodium-Water Reaction in the PIC | 2.01E-02 | 6.09E-01 |
| RFH-ESWR-1 | | | | |
| RFH-FDIV-BL | RFH-FDIV-CN | Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel | 1.74E+00 | 3.19E+00 |
| RFH-FDIV-1 | | | | |
| RFH-FDIV-3 | | | | |
| RFH-FDSP-1 | RFH-FDSP-CN | Fuel Handling Event Occurs While Moving Fuel Assembly in the SFP | 7.33E+00 | 7.33E+00 |


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**Table 3.5-4 Summary of DBAs**
**(Sheet 2 of 2)**

| DBE ID | DBA ID | DBA Title | DBA 2-hour Exclusion Area Boundary TEDE (rem) | DBA 30-day Low Population Zone Boundary TEDE (rem) |
|---|---|---|---|---|
| RFH-LBCA-BL | RFH-LBCA-CN | Loss of BLTC Active Cooling While Handling Fuel Assembly | No Release | No Release |
| RFH-LMCA-BL | RFH-LMCA-CN | Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA | No Release | No Release |
| RFH-LTCA-1 | RFH-LTCA-CN | Loss of EVST Cooling While Storing Fuel Assemblies | No Release | No Release |
| RRS-CGR-BL | RRS-CGR-CN | SCG Leak Downstream the SCG Cell | 7.75E+00 | 7.77E+00 |
| RRS-CGR-1 | | | | |
| RRS-ISPL-BL | RRS-ISPL-CN | SPS-I Leak at the Cold Trap | 1.34E-01 | 5.30E-01 |
| RRS-RWG-1 | RRS-RWG-CN | RWG Leak | 9.59E-02 | 9.59E-02 |
| RRS-RWG-2 | | | | |
| RRS-SPLA-BL | RRS-SPLA-CN | SPS-P Leak in the RAB | 8.13E-02 | 2.78E-01 |
| RRS-SPLX-BL | RRS-SPLX-CN | SPS-P Leak in the RXB | 8.13E-02 | 2.78E-01 |
| N/A | SEV-SSE-CN | Seismic Event | No Release | No Release |
| SUD-IACA-1 | SUD-IACA-CN | Loss of One Train of IAC While Shutdown | No Release | No Release |
| SUD-IHEL-BL | SUD-IHEL-CN | Intermediate Heat Exchanger Secondary-to-Primary Leak While Shutdown | No Release | No Release |
| SUD-LOOP-1 | SUD-LOOP-CN | Loss of Offsite Power While at Low Power | No Release | No Release |


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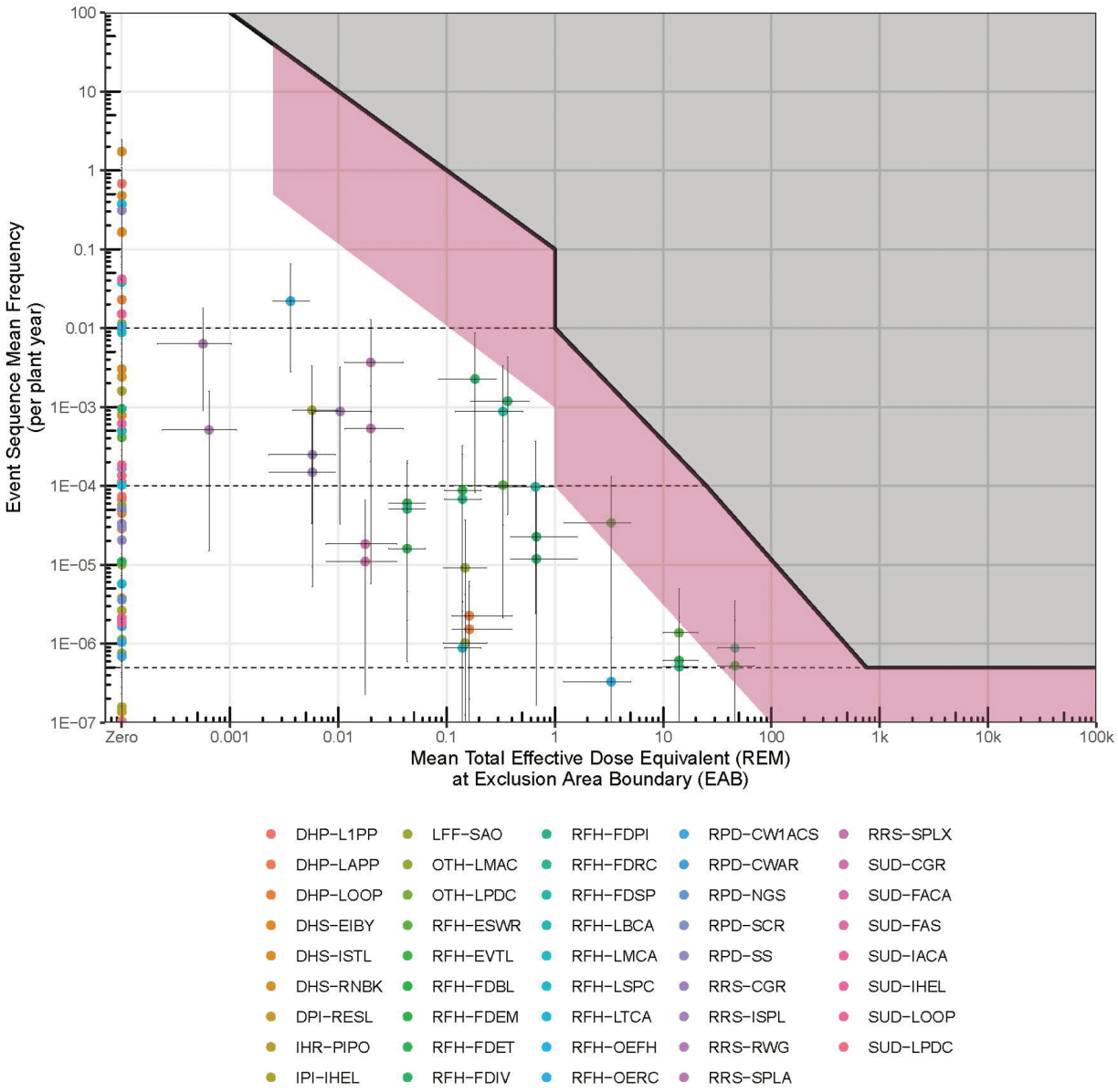
**Figure 3.5-1 F-C Chart for LBEs with Uncertainty Bands**



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# 3.6 Anticipated Operational Occurrences

Anticipated operational occurrences (AOOs) are anticipated event sequences expected to occur one or more times during the life of the plant. Event sequences with mean frequencies of $1x10^{-2}$ per plant-year and greater are classified as AOOs. AOOs take into account the expected responses of all structures, systems, or components within the plant, regardless of safety classification. Operator actions are not required to mitigate AOOs unless otherwise noted in the AOO description.

Mechanistic source term summary tables are provided in Section 3.2.

## 3.6.1 Loss of Primary Flow AOOs

### 3.6.1.1 Loss of One Primary Sodium Pump with Non-Passive IAC (DHP-L1PP-BL)

#### 3.6.1.1.1 LBE Description

The postulated initiating event is a partial loss of primary flow due to the loss of a single Primary Sodium Pump during full power operating conditions. A reactor scram on high high power-to-flow ratio occurs. The final conditions result in the reactor in safe shutdown with the Intermediate Air Cooling System (IAC) in non-passive mode providing decay heat removal.

#### 3.6.1.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this licensing basis event (LBE).

### 3.6.1.2 Loss of Offsite Power with Non-Passive IAC (DHP-LOOP-BL)

#### 3.6.1.2.1 LBE Description

The postulated initiating event is a loss of offsite power during full power operating conditions, resulting in a reactor scram on high high power-to-flow ratio. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

#### 3.6.1.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.6.1.3 Loss of Offsite Power While at Low Power with Non-Passive IAC (SUD-LOOP-BL)

#### 3.6.1.3.1 LBE Description

The postulated initiating event is a loss of offsite power during low power operating conditions. The reactor scrams due to the loss of primary flow. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.


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3.6.1.3.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.6.1.4 Loss of a Single Medium Voltage AC Bus with Non-Passive IAC (OTH-LMAC-BL)

3.6.1.4.1 LBE Description

The postulated initiating event is the loss of a primary 6.9 kilovolt bus during full power operating conditions. The loss of the bus results in the loss of one Primary Sodium Pump and one Intermediate Sodium Pump. A reactor scram on high high power-to-flow ratio occurs. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

3.6.1.4.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.6.2 Increase or Decrease in Heat Removal AOOs

3.6.2.1 Loss of Heat Sink with Non-Passive IAC (DHS-ISTL-BL)

3.6.2.1.1 LBE Description

The postulated initiating event is a loss of heat removal from one intermediate sodium loop during full power operating conditions. A reactor scram on high high cold pool temperature occurs. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

3.6.2.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.6.2.2 Loss of One Train of IAC While Shutdown with Non-Passive IAC (SUD-IACA-BL)

3.6.2.2.1 LBE Description

The postulated initiating event is the loss of one train of IAC during low power or shutdown operating conditions. The final conditions result in the reactor safe and stable at low power or in safe shutdown with decay heat removal provided by IAC in non-passive mode.

3.6.2.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.6.3 Reactivity AOOs

#### 3.6.3.1 Control Rod-Induced Transient Overpower with Non-Passive IAC (RPD-CW1ACS-BL)

##### 3.6.3.1.1 LBE Description

The postulated initiating event is the continuous withdrawal of a control rod during full power operating conditions. A reactor scram on high high power range neutron flux or high high power-to-flow ratio occurs. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

##### 3.6.3.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

#### 3.6.3.2 Reactor Scram or Spurious Scram with Non-Passive IAC (RPD-SS-BL)

##### 3.6.3.2.1 LBE Description

The postulated initiating event is a reactor scram or spurious scram during full power operating conditions. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

##### 3.6.3.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.6.4 Fuel Handling AOOs

#### 3.6.4.1 Fuel Damage While Handling an LTA or LDA Test Pin with PRC Barrier Successfully Retains Release (RFH-OERC-BL)

##### 3.6.4.1.1 LBE Description

The postulated initiating event is a fuel handling event while handling a Lead Test Assembly or Lead Demonstration Assembly in the Pin Removal Cell during pin removal activities. The Pin Removal Cell boundary is not failed. Fuel cladding is damaged, and a radionuclide release occurs.

##### 3.6.4.1.2 Mechanistic Source Term

The EVHM source term is applied to this LBE because the postulated release from the EVHM leads to more conservative results than a postulated release from the PRC. The analysis assumes the rupture of the cladding of seven pins in one high burnup LTA or LDA fuel assembly in the EVHM. The analysis assumes seven pins are damaged to account for the pin being removed from the assembly and the six surrounding pins. The event is assumed to occur


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2.5 days after the reactor is shutdown from full power operation. The design leakage through the EVHM is released to the FHB and the FHB volume is assumed to be released instantaneously to the environment. Holdup and decay in the FHB is not considered. The mechanistic source term is summarized in Table 3.2-8.

3.6.4.1.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day Total Effective Dose Equivalent (TEDE) at the Exclusion Area Boundary (EAB) for this AOO are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 2.46E-03 | 3.62E-03 | 5.40E-03 |

The 5th percentile, 95th percentile, and mean frequency for this AOO are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 2.78E-03 | 2.20E-02 | 6.49E-02 |

3.6.4.2 Loss of Spent Fuel Pool Cooling with Cooling Restored (RFH-LSPC-BL)

3.6.4.2.1 LBE Description

The postulated initiating event is a loss of active spent fuel pool cooling during full power, low power, shutdown, or refueling operating conditions. The Spent Fuel Pool cooling system is restored prior to Spent Fuel Pool boiling and no fuel damage occurs.

3.6.4.2.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.

3.6.4.3 Loss of EVST Active Cooling While Storing Fuel Assemblies with Passive Cooling (RFH-LTCA-BL)

3.6.4.3.1 LBE Description

The postulated initiating event is a loss of active Ex-Vessel Storage Tank cooling during full power, low power, shutdown, or refueling operating conditions. Ex-Vessel Storage Tank passive cooling is available, and no fuel damage occurs.

3.6.4.3.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.


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## 3.7 Design Basis Events

Design basis events (DBEs) are infrequent event sequences that are less likely than anticipated operational occurrences and not expected to occur in the life of the plant. Event sequences with mean frequencies of $1x10^{-4}$ per plant-year to $1x10^{-2}$ per plant-year are classified as DBEs. DBEs take into account the expected responses of all structures, systems, or components within the plant regardless of safety classification. Operator actions are not required to mitigate DBEs unless otherwise noted in the DBE description.

Mechanistic source term summary tables are provided in Section 3.2.

### 3.7.1 Loss of Primary Flow DBEs

#### 3.7.1.1 Loss of One Primary Sodium Pump with RAC (DHP-L1PP-2)

##### 3.7.1.1.1 LBE Description

The postulated initiating event is a partial loss of primary flow due to the loss of a single Primary Sodium Pump (PSP) during full power operating conditions. A reactor scram on high high power-to-flow ratio occurs. The unaffected PSP and Intermediate Sodium Pumps (ISPs) trip on high high primary sodium temperature. Non-passive and passive Intermediate Air Cooling System (IAC) cooling fail on both trains. The final conditions result in the reactor in safe shutdown with the Reactor Air Cooling System (RAC) providing decay heat removal.

This is the most limiting DBE that mapped to the DHP-L1PP-CN design basis accident (DBA). The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 4.15E-05 | 1.11E-04 | 2.40E-04 |

##### 3.7.1.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this licensing basis event (LBE).

#### 3.7.1.2 Loss of Offsite Power with Passive IAC (DHP-LOOP-1)

##### 3.7.1.2.1 LBE Description

The postulated initiating event is a loss of offsite power during full power operating conditions, resulting in a reactor scram on high high power-to-flow ratio. Non-passive IAC cooling fails on both trains. The final conditions result in the reactor in safe shutdown with the IAC passive mode providing decay heat removal.

This is the most limiting DBE that mapped to the DHP-LOOP-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.


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| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.01E-05 | 1.80E-04 | 5.11E-04 |

3.7.1.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.7.1.3 Loss of Offsite Power While at Low Power with Passive IAC (SUD-LOOP-1)

3.7.1.3.1 LBE Description

The postulated initiating event is a loss of offsite power during low power operating conditions. The reactor scrams due to the loss of primary flow. Non-passive IAC cooling fails on both trains. The final conditions result in the reactor in safe shutdown with the IAC passive mode providing decay heat removal.

This is the most limiting DBE that mapped to the SUD-LOOP-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 2.09E-05 | 1.35E-04 | 2.87E-04 |

3.7.1.3.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.7.2 Increase or Decrease in Heat Removal DBEs

3.7.2.1 Loss of Heat Sink with Passive IAC (DHS-ISTL-1)

3.7.2.1.1 LBE Description

The postulated initiating event is a loss of heat removal from one intermediate sodium loop during full power operating conditions. A reactor scram on high high cold pool temperature occurs. Non-passive IAC cooling fails on both trains. PSPs and ISPs trip on high high primary sodium temperature. The final conditions result in the reactor in safe shutdown with the IAC passive mode providing decay heat removal.

3.7.2.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.7.2.2 Loss of Heat Sink with RAC (DHS-ISTL-2)

#### 3.7.2.2.1 LBE Description

The postulated initiating event is a loss of heat removal from one intermediate sodium loop during full power operating conditions. A reactor scram on high high cold pool temperature occurs. PSPs and ISPs trip on high high primary sodium temperature. Non-passive and passive IAC cooling fails on both trains. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

This is the most limiting DBE that mapped to the DHS-ISTL-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.05E-03 | 3.02E-03 | 7.45E-03 |

#### 3.7.2.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.7.2.3 Energy Island Transient without Reactor Power Runback, with Non-Passive IAC (DHS-RNBK-1)

#### 3.7.2.3.1 LBE Description

The postulated initiating event is a reactor power runback demand initiated by a decrease or increase in heat removal from the NSS during full power operating conditions. This event is also assumed to bound other events that generate reactor power runback signals. A reactor power runback may be initiated, but a reactor scram occurs before the reactor power runback is complete. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.

#### 3.7.2.3.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.7.2.4 Energy Island Transient without Reactor Power Runback, with RAC (DHS-RNBK-3)

#### 3.7.2.4.1 LBE Description

The postulated initiating event is a reactor power runback demand initiated by a decrease or increase in heat removal from the NSS during full power operating conditions. This event is also assumed to bound other events that generate reactor power runback signals. A reactor power runback may be initiated, but a reactor scram occurs before the reactor power runback is complete. PSPs and ISPs trip on high high primary sodium temperature. Non-passive and passive IAC cooling fails on both trains. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.


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This is the most limiting DBE that mapped to the DHS-RNBK-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 4.39E-05 | 1.84E-04 | 4.62E-04 |

3.7.2.4.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.7.2.5 Loss of One Train of IAC While Shutdown with Passive IAC (SUD-IACA-1)

3.7.2.5.1 LBE Description

The postulated initiating event is a loss of one IAC train during low power or shutdown operating conditions. Non-passive IAC cooling fails on the unaffected IAC train. The final conditions result in the reactor safe and stable at low power or in safe shutdown with decay heat removal provided by the unaffected IAC train in passive mode.

This is the most limiting DBE that mapped to the SUD-IACA-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.40E-04 | 6.14E-04 | 1.49E-03 |

3.7.2.5.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.7.2.6 Intermediate Heat Exchanger Secondary-to-Primary Leak with Non-Passive IAC (IPI-IHEL-BL)

3.7.2.6.1 LBE Description

The postulated initiating event is an Intermediate Heat Exchanger tube break during full power operating conditions, resulting in intermediate sodium leakage into the Primary Heat Transport System and loss of heat removal from one intermediate sodium loop. A reactor scram on high high primary sodium level occurs and the ISPs trip on high high primary sodium level. Intermediate sodium volume that can drain into the PHT is limited by the IHT physical arrangement as described in Section 7.1.4 and the reactor vessel is not overfilled with sodium. The final conditions result in the reactor in safe shutdown with decay heat removal provided by the unaffected IAC train in non-passive mode.

This is the most limiting DBE that mapped to the IPI-IHEL-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.


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| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.46E-04 | 1.59E-03 | 5.18E-03 |

3.7.2.6.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.7.2.7 Intermediate Heat Exchanger Secondary-to-Primary Leak While Shutdown with Non-Passive IAC (SUD-IHEL-BL)

3.7.2.7.1 LBE Description

The postulated initiating event is an Intermediate Heat Exchanger tube break during low power or shutdown operating conditions, resulting in intermediate sodium leakage into the PHT and loss of heat removal from one intermediate sodium loop. If the event occurs at low power, a reactor scram on high high primary sodium level occurs. Intermediate sodium volume that can drain into the PHT is limited by the IHT physical arrangement as described in Section 7.1.4 and the reactor vessel is not overfilled with sodium. The final conditions result in the reactor in safe shutdown with decay heat removal provided by the unaffected IAC train in non-passive mode.

This is the most limiting DBE that mapped to the SUD-IHEL-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.17E-05 | 1.81E-04 | 5.81E-04 |

3.7.2.7.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.7.3 Release from Ex-Vessel Systems DBEs

3.7.3.1 SCG Leak Inside the HAA where RXB Substructure Successfully Retains Release (RRS-CGR-BL)

3.7.3.1.1 LBE Description

A Sodium Cover Gas System (SCG) leak occurs inside the Head Access Area (HAA) during full power, low power, or shutdown operating conditions. Functional containment barriers successfully perform their radionuclide retention functions. Fuel cladding is not failed, but a radionuclide release occurs.

3.7.3.1.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from the cover gas volume in the reactor vessel into the HAA compartment at the initiation of the event. For the first 30 minutes of the event, the HAA is slowly exhausted to the


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environment through the Nuclear Island Heating, Ventilation, and Air Conditioning System (NHV) at a rate of 400 cfm with NHV filtration. At 30 minutes, the HAA NHV system is isolated and the HAA begins to leak to the RXB at a rate of 10 percent volume per day. The RXB is exhausted to the environment at a rate of 1500 cfm with NHV filtration. The mechanistic source term is summarized in Table 3.2-12.

3.7.3.1.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day Total Effective Dose Equivalent (TEDE) at the Exclusion Area Boundary (EAB) for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 2.27E-03 | 5.75E-03 | 9.37E-03 |

The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 9.47E-06 | 2.48E-04 | 8.69E-04 |

3.7.3.2 SCG Leak Downstream the SCG Cell where RXB Substructure Successfully Retains Release (RRS-CGR-1)

3.7.3.2.1 LBE Description

An SCG leak occurs downstream of the HAA primary functional containment boundary isolation valve during full power, low power, or shutdown operating conditions. Functional containment barriers successfully perform their radionuclide retention functions. Fuel cladding is not failed, but a radionuclide release from the SCG occurs.

3.7.3.2.2 Mechanistic Source Term

Postulated SCG leaks outside the HAA boundary are in the vapor trap cell compartment. The vapor trap cell compartment has a lower leakage rate than the HAA, so the mechanistic source term and radiological consequence results from RRS-CGR-BL are conservatively applied to this LBE. The mechanistic source term is summarized in Table 3.2-12.

3.7.3.2.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 2.27E-03 | 5.75E-03 | 9.37E-03 |


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This is the most limiting DBE that mapped to the RRS-CGR-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 5.23E-06 | 1.49E-04 | 5.19E-04 |

3.7.3.3 SPS Leak at the Cold Trap with RAB Barrier Successfully Retains Release (RRS-ISPL-BL)

3.7.3.3.1 LBE Description

The postulated initiating event is a Sodium Processing System (SPS) leak during full power, low power, or shutdown operating conditions that releases precipitated tritium from the Intermediate SPS (SPS-I) cold trap. Other functional containment barriers successfully perform their radionuclide retention functions. Fuel cladding is not failed, but a radionuclide release from the SPS occurs.

3.7.3.3.2 Mechanistic Source Term

A release of precipitated tritium from one SPS-I cold trap is assumed to occur over the course of 24 hours. The RAB is conservatively assumed to be released to the environment instantaneously. The mechanistic source term is summarized in Table 3.2-18.

3.7.3.3.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 5.93E-03 | 1.04E-02 | 2.04E-02 |

This is the most limiting DBE that mapped to the RRS-ISPL-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 3.29E-05 | 8.77E-04 | 3.19E-03 |

3.7.3.4 SPS-P Leak in the RXB (RRS-SPLX-BL)

3.7.3.4.1 LBE Description

The postulated initiating event is a leak in the SPS system inboard of the SPS isolation valves within the HAA during full power, low power, or shutdown operating conditions. The SPS pump trips on low primary sodium level, terminating the loss of primary sodium inventory. Functional containment barriers successfully perform their radionuclide retention functions.


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3.7.3.4.2 Mechanistic Source Term

The HAA compartment is expected to have a similar leakage rate as the SPS cell in the RAB and the HAA compartment has a larger volume than the SPS cell in the RAB, so the mechanistic source term and radiological consequence results from RRS-SPLA-BL are applied to this LBE. The mechanistic source term is summarized in Table 3.2-10.

3.7.3.4.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.15E-02 | 1.99E-02 | 3.98E-02 |

This is the most limiting DBE that mapped to the RRS-SPLX-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 5.74E-06 | 5.33E-04 | 1.85E-03 |

3.7.3.5 SPS-P Leak in the RAB (RRS-SPLA-BL)

3.7.3.5.1 LBE Description

The postulated initiating event is a leak in the SPS system within the RAB during full power, low power, or shutdown operating conditions. The SPS pump trips on low primary sodium level, terminating the loss of primary sodium inventory. Functional containment barriers successfully perform their radionuclide retention functions.

3.7.3.5.2 Mechanistic Source Term

The SPS leak is a break in the SPS piping leading to the cold trap that drains into the SPS cell. The SPS piping is designed such that most of the piping is sloped down towards the reactor, minimizing the potential leakage volume from a break at the end of the pipe segment. The sodium volume released from this break is assumed to be 20 gallons. The SPS cell leaks at 10 percent volume per day to the RAB and the RAB exhausts to the environment fully every two hours. The mechanistic source term is summarized in Table 3.2-10.

3.7.3.5.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.


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| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.15E-02 | 1.99E-02 | 3.98E-02 |

This is the most limiting DBE that mapped to the RRS-SPLA-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 2.03E-04 | 3.68E-03 | 1.28E-02 |

### 3.7.3.6 RWG Leak from the Holdup Tank with FHB Barrier Fails to Retain Release (RRS-RWG-1)

#### 3.7.3.6.1 LBE Description

The postulated initiating event is a leak from a radioactive waste gas holdup tank into the Fuel Handling Building (FHB) during full power, low power, shutdown, or refueling operating conditions. The FHB fails to perform its radionuclide retention function and a radionuclide release occurs.

#### 3.7.3.6.2 Mechanistic Source Term

The postulated mechanistic source term is a release of the radioactive waste gas holdup tank contents to the tank vault. The vault provides confinement of radionuclides. The holdup tank vault is discharged to the environment at a rate of 500 cfm with NHV filtration. The mechanistic source term is summarized in Table 3.2-15.

#### 3.7.3.6.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 2.14E-04 | 5.62E-04 | 1.03E-03 |

The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 8.89E-04 | 6.33E-03 | 1.83E-02 |


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### 3.7.3.7 RWG Leak to the Stack to the Environment (RRS-RWG-2)

#### 3.7.3.7.1 LBE Description

The postulated initiating event is a leak from a radioactive waste gas holdup tank to the plant stack during full power, low power, shutdown, or refueling operating conditions. All released fission products are discharged via the plant stack.

#### 3.7.3.7.2 Mechanistic Source Term

The postulated mechanistic source term is a release of the radioactive waste gas holdup tank contents to downstream Gaseous Radwaste Processing System (RWG) piping. The RWG provides confinement of radionuclides that are released via the vent stack to the environment with filtration over a 10-minute period. The mechanistic source term is summarized in Table 3.2-16.

#### 3.7.3.7.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 2.38E-04 | 6.39E-04 | 1.15E-03 |

This is the most limiting DBE that mapped to the RRS-RWG-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.52E-05 | 5.11E-04 | 1.58E-03 |

## 3.7.4 Fuel Handling DBEs

### 3.7.4.1 Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with No Damage (RFH-FDIV-BL)

#### 3.7.4.1.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the reactor vessel during in-vessel fuel moves. The dropped fuel assembly is not damaged and there is no damage to fuel assemblies impacted by the dropped fuel assembly.

#### 3.7.4.1.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.


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### 3.7.4.2 Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with a Single Assembly Failed and Functional Containment Barriers Successfully Retain Release (RFH-FDIV-1)

#### 3.7.4.2.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the reactor vessel during in-vessel fuel moves. The dropped fuel assembly is damaged, and a radionuclide release occurs, but there is no damage to fuel assemblies impacted by the dropped fuel assembly. Functional containment barriers successfully perform their radionuclide retention functions.

#### 3.7.4.2.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in one high burnup fuel assembly. Radionuclides are released from the primary functional containment boundary at an assumed rate of one percent of cover gas volume per day to the environment. The mechanistic source term is summarized in Table 3.2-5.

#### 3.7.4.2.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 8.38E-02 | 1.83E-01 | 2.90E-01 |

The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 8.32E-05 | 2.26E-03 | 8.73E-03 |

### 3.7.4.3 Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with Two Assemblies Failed and Functional Containment Barriers Successfully Retain Release (RFH-FDIV-3)

#### 3.7.4.3.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the reactor vessel during in-vessel fuel moves. The dropped fuel assembly and the impacted fuel assembly are damaged, and a radionuclide release occurs. Functional containment barriers successfully perform their radionuclide retention functions.


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3.7.4.3.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in two high burnup fuel assemblies. Radionuclides are released from the primary functional containment boundary at an assumed rate of one percent of cover gas volume per day to the environment. The mechanistic source term is summarized in Table 3.2-6.

3.7.4.3.3 Radiological Consequences Results

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.67E-01 | 3.66E-01 | 5.82E-01 |

This is the most limiting DBE that mapped to the RFH-FDIV-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 4.33E-05 | 1.19E-03 | 4.32E-03 |

3.7.4.4 Fuel Handling Event Occurs While Moving Fuel Assembly in the SFP with a Single Assembly Failed and FHB Barrier Fails to Retain Release (RFH-FDSP-1)

3.7.4.4.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the spent fuel pool during fuel moves. The dropped fuel assembly is damaged, and a radionuclide release occurs, but there is no damage to fuel assemblies impacted by the dropped fuel assembly. The FHB fails to perform its radionuclide retention function.

3.7.4.4.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in one high burnup fuel assembly. The fuel assembly drop is assumed to occur 310 days after the fuel assembly is stored in the In-Vessel Storage (IVS) after removal from the active core, and prior to placement in the SFP.

When the fuel cladding is damaged, the fuel pin sodium bond material reacts with water in the spent fuel pool in an exothermic gas-producing reaction. The nature of this reaction is being studied for further characterization of the radionuclide retention in the spent fuel pool. The preliminary mechanistic source term analysis assumptions for radionuclide retention in the spent fuel pool are informed by RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Revision 0, with a relatively low decontamination factor to address the fuel pin sodium bond material and water reaction.


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The fission product particulate released into the spent fuel pool from the solid fuel matrix is assumed to behave essentially the same as the particulate released from the fuel during a comparable light-water reactor fuel handling accident. Therefore, this portion of the particulate is assumed to be completely retained within the spent fuel pool water inventory in accordance with the guidance for fuel handling accidents in RG 1.183 Appendix B. The portion of the radionuclide particulate that had migrated into the fuel pin plenum is involved with the reaction with the sodium bond material displaced into the fuel pin plenum as it contacts the spent fuel pool water. The analysis assumes a reduced spent fuel pool decontamination factor from the decontamination factor in RG 1.183 for the fission product particulate assigned to the aerosol removal efficiency to address the impact of the sodium bond material and water reaction. The FHB volume is assumed to exhaust to the environment instantaneously with no credit for NHV filtration. The mechanistic source term is summarized in Table 3.2-9.

3.7.4.4.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.20E-01 | 3.31E-01 | 5.05E-01 |

This is the most limiting DBE that mapped to the RFH-FDSP-CN DBA The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 3.24E-05 | 8.77E-04 | 3.36E-03 |

3.7.4.5 Excessive Sodium-Water Reaction in the PIC (RFH-ESWR-BL)

3.7.4.5.1 LBE Description

The postulated initiating event is a sodium-water reaction in the Pool Immersion Cell (PIC) during fuel handling operations. The fuel cladding integrity is maintained.

3.7.4.5.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.

3.7.4.6 Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Successfully Retains Release (RFH-ESWR-1)

3.7.4.6.1 LBE Description

The postulated initiating event is a sodium-water reaction in the PIC while the BLTC is mated to the PIC that causes fuel cladding failure. The combined PIC and BLTC boundary is not failed, but a radionuclide release occurs.


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3.7.4.6.2 Mechanistic Source Term

The mechanistic source term analysis assumes the combined PIC and BLTC boundary design leakage rate is released to the FHB and the FHB volume is instantaneously released to the environment with no NHV filtration. The credit of the BLTC in the event provides a low leakage barrier to mitigate the release. The mechanistic source term assumes partial confinement or retention of radionuclides and conservatively applies the dose consequences of RFH-FDSP-1. The mechanistic source term is summarized in Table 3.2-9.

3.7.4.6.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.20E-01 | 3.31E-01 | 5.05E-01 |

This is the most limiting DBE that mapped to the RFH-ESWR-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 2.44E-06 | 1.03E-04 | 4.02E-04 |

3.7.4.7 Loss of EVST Active Cooling While Storing Fuel Assembly without Passive Cooling (RFH-LTCA-1)

3.7.4.7.1 LBE Description

The postulated initiating event is a loss of Ex-Vessel Storage Tank (EVST) cooling with spent fuel in the EVST. EVST passive cooling is unavailable or provides inadequate cooling to the assemblies contained in the EVST. The EVST boundary is not failed. Analysis demonstrates that no fuel damage occurs for at least 72 hours following the loss of cooling. Longer term degraded heat removal conditions require further assessment. Fuel cladding is not damaged, and no radionuclide release occurs.

This is the most limiting DBE that mapped to the RFH-LTCA-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 5.12E-06 | 1.02E-04 | 3.72E-04 |

3.7.4.7.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.


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### 3.7.4.8 Loss of BLTC Active Cooling While Handling Fuel Assembly with Passive Cooling (RFH-LBCA-BL)

#### 3.7.4.8.1 LBE Description

The postulated initiating event is a loss of bottom loading transfer cask active cooling with spent fuel in the bottom loading transfer cask. Bottom loading transfer cask passive cooling is available, and no fuel damage occurs.

This is the most limiting DBE that mapped to the RFH-LBCA-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.95E-05 | 5.02E-04 | 1.72E-03 |

#### 3.7.4.8.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.

### 3.7.4.9 Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA with Passive Cooling (RFH-LMCA-BL)

#### 3.7.4.9.1 LBE Description

The postulated initiating event is a loss of Ex-Vessel Handling Machine (EVHM) active cooling with spent fuel in the EVHM. EVHM passive cooling is available, and no fuel damage occurs.

This is the most limiting DBE that mapped to the RFH-LMCA-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 7.51E-04 | 8.82E-03 | 2.46E-02 |

#### 3.7.4.9.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.

## 3.7.5 Local Fuel Fault DBEs

### 3.7.5.1 Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Successfully Retains Release (Vessel Head Success) (LFF-SAO-BL)

#### 3.7.5.1.1 LBE Description

The postulated initiating event is a blockage of fuel subchannels or other localized faults within the reactor core during full power operating conditions, resulting in fuel damage. Operators manually shutdown the reactor due to exceeding failed fuel activity limits. Functional containment


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barriers successfully perform their radionuclide retention functions, but a radionuclide release occurs. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.

3.7.5.1.2 Mechanistic Source Term

This LBE is modeled as an instantaneous release of the radionuclide inventory from all pins in one high burnup fuel assembly into the primary sodium. Eighty percent of released non-noble gas radionuclides are postulated to become entrained in bubbles and enter the cover gas without primary sodium pool scrubbing. Radionuclides are released from the primary functional containment boundary at an assumed rate of 1 percent of the cover gas volume per day, and subsequently released from the HAA to the environment at an assumed rate of 10 percent of the HAA volume per day. The mechanistic source term is summarized in Table 3.2-2.

3.7.5.1.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this DBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 3.82E-03 | 5.70E-03 | 9.87E-03 |

This is the most limiting DBE that mapped to the LFF-SAO-CN DBA. The 5th percentile, 95th percentile, and mean frequency for this DBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 3.32E-05 | 9.10E-04 | 3.36E-03 |


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# 3.8 Beyond Design Basis Events

Beyond Design Basis Events (BDBEs) are rare event sequences that are less likely than design basis events and are not expected to occur in the life of a nuclear power plant. Event sequences with mean frequencies of $5x10^{-7}$ per plant-year to $1x10^{-4}$ per plant-year are classified as BDBEs. BDBEs take into account the expected response of all structures, systems, or components within the plant regardless of safety classification.

Mechanistic source term summary tables are provided in Section 3.2.

## 3.8.1 Loss of Primary Flow BDBEs

### 3.8.1.1 Loss of One Primary Sodium Pump with Passive IAC (DHP-L1PP-1)

#### 3.8.1.1.1 LBE Description

The postulated initiating event is a partial loss of primary flow due to the loss of a single Primary Sodium Pump (PSP) during full power operating conditions. A reactor scram on high high power-to-flow ratio occurs. The unaffected PSP and Intermediate Sodium Pumps (ISPs) trip on high high primary sodium temperature. Non-passive Intermediate Air Cooling System (IAC) cooling fails on both trains. The final conditions result in the reactor in safe shutdown with the IAC passive mode providing decay heat removal.

#### 3.8.1.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this licensing basis event (LBE).

### 3.8.1.2 Loss of One Primary Sodium Pump with Scram Motor Drive-In (DHP-L1PP-3)

#### 3.8.1.2.1 LBE Description

The postulated initiating event is a partial loss of primary flow due to the loss of a single PSP during full power operating conditions. The reactor scram on high high power-to-flow ratio fails. A Control Rod Drive System (CRD) driveline scram follow occurs, resulting in control rods being driven into the core. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.

#### 3.8.1.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.8.1.3 Loss of One Primary Sodium Pump with Alternative Shunt Trip (DHP-L1PP-4)

#### 3.8.1.3.1 LBE Description

The postulated initiating event is a partial loss of primary flow due to the loss of a single PSP during full power operating conditions. The normal reactor scram on high high power-to-flow ratio fails. The alternative shunt trip circuit scrams the reactor. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.

#### 3.8.1.3.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.1.4 Loss of All Primary Sodium Pumps with Non-Passive IAC (DHP-LAPP-BL)

#### 3.8.1.4.1 LBE Description

The postulated initiating event is a total loss of primary flow during full power operating conditions. A reactor scram on high high power-to-flow ratio occurs. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.

#### 3.8.1.4.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.1.5 Loss of Offsite Power with RAC (DHP-LOOP-2)

#### 3.8.1.5.1 LBE Description

The postulated initiating event is a loss of offsite power during full power operating conditions, resulting in a reactor scram on high high power-to-flow ratio. Non-passive and passive IAC cooling fails on both trains. The final conditions result in the reactor in safe shutdown with the Reactor Air Cooling System (RAC) providing decay heat removal.

#### 3.8.1.5.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.1.6 Loss of Offsite Power with Scram Motor Drive-In (DHP-LOOP-3)

#### 3.8.1.6.1 LBE Description

The postulated initiating event is a loss of offsite power during full power operating conditions. The reactor scram on high high power-to-flow ratio fails. A CRD driveline scram follow occurs, resulting in control rods being driven into the core. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.


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The fuel performance acceptance criteria are not met for this event. Fuel cladding is failed, and a radionuclide release occurs.

3.8.1.6.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the estimated radionuclide inventory due to creep rupture of two-thirds of the core with no fuel melt. The leakage rate through the primary functional containment boundary to the HAA is assumed to be one percent per day of the cover gas region volume. The leakage rate out of the Head Access Area (HAA) is assumed to be 10 percent of the HAA volume per day. The mechanistic source term is summarized in Table 3.2-1.

3.8.1.6.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day Total Effective Dose Equivalent (TEDE) at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.12E-01 | 1.62E-01 | 4.06E-01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.99E-07 | 2.26E-06 | 6.17E-06 |

3.8.1.7 Loss of Offsite Power with Alternative Shunt Trip (DHP-LOOP-4)

3.8.1.7.1 LBE Description

The postulated initiating event is a loss of offsite power during full power operating conditions. The normal reactor scram on high high power-to-flow ratio fails. The alternative shunt trip circuit scrams the reactor. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

The fuel performance acceptance criteria are not met for this event. Fuel cladding is failed, and a radionuclide release occurs.

3.8.1.7.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the estimated radionuclide inventory due to creep rupture of two-thirds of the core with no fuel melt. The leakage rate through the primary functional containment boundary to the HAA is assumed to be 1 percent per day of the cover gas region volume. The leakage rate out of the HAA is assumed to be 10 percent of the HAA volume per day. The mechanistic source term is summarized in Table 3.2-1.


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3.8.1.7.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.12E-01 | 1.62E-01 | 4.06E-01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 6.79E-08 | 1.52E-06 | 5.33E-06 |

3.8.1.8 Loss of a Single Medium Voltage AC Bus with Passive IAC (OTH-LMAC-1)

3.8.1.8.1 LBE Description

The postulated initiating event is the loss of a primary 6.9 kilovolt bus during full power operating conditions. A reactor scram on high high power-to-flow ratio occurs. Non-passive IAC fails on both trains. The unaffected PSP and ISP trip on high high primary sodium temperature. The final conditions result in the reactor in safe shutdown with the IAC passive mode providing decay heat removal.

3.8.1.8.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.8.1.9 Loss of a Single Medium Voltage AC Bus with RAC (OTH-LMAC-2)

3.8.1.9.1 LBE Description

The postulated initiating event is the loss of a primary 6.9 kilovolt bus during full power operating conditions. A reactor scram on high high power-to-flow ratio occurs. Passive and non-passive IAC fails on both trains. The unaffected PSP and ISP trip on high high primary sodium temperature. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

3.8.1.9.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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3.8.1.10 Loss of a Single Medium Voltage AC Bus with Scram Motor Drive-In (OTH-LMAC-3)

3.8.1.10.1 LBE Description

The postulated initiating event is the loss of a primary 6.9 kilovolt bus during full power operating conditions. The reactor scram on high high power-to-flow ratio fails. A CRD driveline scram follow occurs, resulting in control rods being driven into the core. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

3.8.1.10.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.8.1.11 Loss of a Single Medium Voltage AC Bus with Alternative Shunt Trip (OTH-LMAC-4)

3.8.1.11.1 LBE Description

The postulated initiating event is the loss of a primary 6.9 kilovolt bus during full power operating conditions. The normal reactor scram on high high power-to-flow ratio fails. The alternative shunt trip circuit scrams the reactor. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.

3.8.1.11.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.8.2 Increase or Decrease in Heat Removal BDBEs

3.8.2.1 Loss of Heat Sink with Scram Motor Drive-In (DHS-ISTL-3)

3.8.2.1.1 LBE Description

The postulated initiating event is a loss of heat removal from one intermediate sodium loop during full power operating conditions. The reactor scram on high high cold pool temperature fails. A CRD driveline scram follow occurs, resulting in control rods being driven into the core. The final conditions result in the reactor in safe shutdown with the unaffected IAC train in non-passive mode providing decay heat removal.

3.8.2.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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3.8.2.2 Loss of Heat Sink with Alternative Shunt Trip (DHS-ISTL-4)

3.8.2.2.1 LBE Description

The postulated initiating event is a loss of heat removal from one intermediate sodium loop during full power operating conditions. The normal reactor scram on high high cold pool temperature fails. The alternative shunt trip circuit scrams the reactor. The final conditions result in the reactor in safe shutdown with the unaffected IAC train in non-passive mode providing decay heat removal.

3.8.2.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.8.2.3 Energy Island Transient without Reactor Power Runback, with Passive IAC (DHS-RNBK-2)

3.8.2.3.1 LBE Description

The postulated initiating event is a reactor power runback demand initiated by a decrease or increase in heat removal from the NSS during full power operating conditions. This event is also assumed to bound other events that generate reactor power runback signals. A reactor power runback may be initiated, but a reactor scram occurs before the reactor power runback is complete. PSPs and ISPs trip on high high primary sodium temperature. Non-passive IAC cooling fails on both trains. The final conditions result in the reactor in safe shutdown with the IAC in passive mode providing decay heat removal.

3.8.2.3.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

3.8.2.4 Loss of One Train of IAC While Shutdown with RAC (SUD-IACA-2)

3.8.2.4.1 LBE Description

The postulated initiating event is a loss of an IAC train during low power or shutdown operating conditions. Non-passive IAC cooling fails on the unaffected train and passive IAC cooling fails on both trains. The final conditions result in the reactor safe and stable at low power or in safe shutdown with the RAC providing decay heat removal.

3.8.2.4.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.8.2.5 Intermediate Heat Exchanger Secondary-to-Primary Leak with Passive IAC, (IPI-IHEL-1)

#### 3.8.2.5.1 LBE Description

The postulated initiating event is an Intermediate Heat Exchanger tube break during full power operating conditions, resulting in intermediate sodium leakage into the Primary Heat Transport System (PHT) and loss of heat removal from one intermediate sodium loop. A reactor scram on high high primary sodium level occurs and ISPs trip on high high primary sodium level. Intermediate sodium volume that can drain into the PHT is limited by the IHT physical arrangement as described in Section 7.1.4 and the reactor vessel is not overfilled with sodium. The unaffected train of non-passive IAC fails. The final conditions result in the reactor in safe shutdown with decay heat removal provided by the unaffected IAC train in passive mode.

#### 3.8.2.5.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.2.6 Intermediate Heat Exchanger Secondary-to-Primary Leak with RAC (IPI-IHEL-2)

#### 3.8.2.6.1 LBE Description

The postulated initiating event is an Intermediate Heat Exchanger tube break during full power operating conditions, resulting in intermediate sodium leakage into the PHT and loss of heat removal from one intermediate sodium loop. A reactor scram on high high primary sodium level occurs and ISPs trip on high high primary sodium level. Intermediate sodium volume that can drain into the PHT is limited by the IHT physical arrangement as described in Section 7.1.4 and the reactor vessel is not overfilled with sodium. The unaffected train of passive and non-passive IAC fails. PSPs trip on high high primary sodium temperature and coast down. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

#### 3.8.2.6.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.2.7 Intermediate Heat Exchanger Secondary-to-Primary Leak While Shutdown with Passive IAC (SUD-IHEL-1)

#### 3.8.2.7.1 LBE Description

The postulated initiating event is an Intermediate Heat Exchanger tube break during low power or shutdown operating conditions, resulting in intermediate sodium leakage into the PHT and loss of heat removal from one intermediate sodium loop. If the event occurs at low power a reactor scram on high high primary sodium level occurs. Intermediate sodium volume that can drain into the PHT is limited by the IHT physical arrangement as described in Section 7.1.4 and the reactor


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vessel is not overfilled with sodium. The unaffected train of non-passive IAC fails. The final conditions result in the reactor in safe shutdown with decay heat removal provided by the unaffected IAC train in passive mode.

#### 3.8.2.7.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.3 Reactivity BDBEs

#### 3.8.3.1 Control Rod-Induced Transient Overpower with Passive IAC (RPD-CW1ACS-1)

##### 3.8.3.1.1 LBE Description

The postulated initiating event is the continuous withdrawal of a control rod during full power operating conditions. A reactor scram on high high power range neutron flux or high high power-to-flow ratio occurs. Non-passive IAC cooling fails on both trains. The PSPs and ISPs trip on high high primary sodium temperature. The final conditions result in the reactor in safe shutdown with the IAC passive mode providing decay heat removal.

##### 3.8.3.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

#### 3.8.3.2 Control Rod-Induced Transient Overpower with RAC (RPD-CW1ACS-2)

##### 3.8.3.2.1 LBE Description

The postulated initiating event is the continuous withdrawal of a control rod during full power operating conditions. A reactor scram on high high power range neutron flux or high high power-to-flow ratio occurs. Non-passive and passive IAC cooling fails on both trains. The PSPs and ISPs trip on high high primary sodium temperature. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

##### 3.8.3.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.8.3.3 Control Rod-Induced Transient Overpower with Scram Motor Drive-In (RPD-CW1ACS-3)

#### 3.8.3.3.1 LBE Description

The postulated initiating event is the continuous withdrawal of a control rod during full power operating conditions. A reactor scram signal occurs on high high power range neutron flux or high high power-to-flow ratio, but the reactor does not scram. A CRD driveline scram follow occurs, resulting in control rods being driven into the core. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

#### 3.8.3.3.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.3.4 Control Rod-Induced Transient Overpower with Alternative Shunt Trip (RPD-CW1ACS-4)

#### 3.8.3.4.1 LBE Description

The postulated initiating event is the continuous withdrawal of a control rod during full power operating conditions. The normal reactor scram fails. The alternative shunt trip circuit scrams the reactor. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

#### 3.8.3.4.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.3.5 Reactor Scram or Spurious Scram with Passive IAC (RPD-SS-1)

#### 3.8.3.5.1 LBE Description

The postulated initiating event is a reactor scram or spurious scram during full power operating conditions. Non-passive IAC cooling fails on both trains. The PSPs and ISPs trip on high high primary sodium temperature. The final conditions result in the reactor in safe shutdown with the IAC in passive mode providing decay heat removal.

#### 3.8.3.5.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.8.3.6 Reactor Scram or Spurious Scram with RAC (RPD-SS-2)

#### 3.8.3.6.1 LBE Description

The postulated initiating event is a reactor scram or spurious scram during full power operating conditions. Non-passive and passive IAC cooling fails on both trains. The PSPs and ISPs trip on high high primary sodium temperature. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

#### 3.8.3.6.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.3.7 Reactor Scram or Spurious Scram with Scram Motor Drive-In (RPD-SS-3)

#### 3.8.3.7.1 LBE Description

The postulated initiating event is a reactor scram or spurious scram during full power operating conditions, but the reactor scram does not occur. A CRD driveline scram follow occurs, resulting in control rods being driven into the core. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

#### 3.8.3.7.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.8.3.8 Reactor Scram or Spurious Scram with Alternative Shunt Trip (RPD-SS-4)

#### 3.8.3.8.1 LBE Description

The postulated initiating event is a Nuclear Island transient that requires a scram during full power operating conditions. The reactor scram fails. The alternative shunt trip circuit scrams the reactor. The final conditions result in the reactor in safe shutdown with the IAC in non-passive mode providing decay heat removal.

#### 3.8.3.8.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.8.4 Release from Ex-Vessel Systems BDBEs

#### 3.8.4.1 SCG Leak Inside the HAA where RXB Substructure Fails to Retain Release While Shutdown (SUD-CGR-1)

##### 3.8.4.1.1 LBE Description

A Sodium Cover Gas System (SCG) leak occurs inside the HAA during low power or shutdown operating conditions. Functional containment barriers fail to perform their radionuclide retention functions. Fuel cladding is not failed, but a radionuclide release occurs.

##### 3.8.4.1.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from the cover gas volume contained in the reactor vessel to the HAA at the initiation of the event. The HAA is exhausted to the environment through the Nuclear Island Heating, Ventilation, and Air Conditioning System (NHV) at a rate of 400 cubic feet per minute with NHV filtration. The mechanistic source term is summarized in Table 3.2-13.

##### 3.8.4.1.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 7.73E-03 | 1.77E-02 | 3.47E-02 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 3.44E-07 | 1.83E-05 | 6.70E-05 |

#### 3.8.4.2 SCG Leak Downstream the SCG Cell where RXB Substructure Fails to Retain Release While Shutdown (SUD-CGR-2)

##### 3.8.4.2.1 LBE Description

An SCG leak occurs downstream of the SCG vapor trap cell during low power or shutdown operating conditions. Functional containment barriers fail to perform their radionuclide retention functions. Fuel cladding is not failed, but a radionuclide release occurs.

##### 3.8.4.2.2 Mechanistic Source Term

SCG leaks outside the HAA boundary are located in the vapor trap cell compartment. The vapor trap cell compartment has a lower leakage rate than the HAA, so the mechanistic source term and radiological consequence results from SUG-CGR-1 are applied to this LBE. The mechanistic source term is summarized in Table 3.2-13.


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3.8.4.2.3 Radiological Consequences Results

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 7.73E-03 | 1.77E-02 | 3.47E-02 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 2.25E-07 | 1.10E-05 | 4.20E-05 |

3.8.5 Fuel Handling BDBEs

3.8.5.1 Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with a Single Assembly Failed and Functional Containment Barriers Fail to Retain Release (RFH-FDIV-2)

3.8.5.1.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the reactor vessel during in-vessel fuel moves. The dropped fuel assembly is damaged, but there is no damage to other fuel assemblies impacted by the dropped fuel assembly. Functional containment barriers fail to perform their radionuclide retention functions.

3.8.5.1.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in two high burnup fuel assemblies. This is conservative because only one fuel assembly is assumed to be damaged in this LBE. Radionuclides are released from the primary functional containment boundary at a degraded rate of 100 percent of cover gas volume per day, and subsequently released from the Reactor Building (RXB) at a degraded rate of 100 percent of the RXB volume per day. The mechanistic source term is summarized in Table 3.2-3.

3.8.5.1.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 3.88E-01 | 6.74E-01 | 1.62E+00 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.


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| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 3.06E-07 | 2.27E-05 | 8.48E-05 |

3.8.5.2 Fuel Handling Event Occurs While Moving Fuel Assembly in the Reactor Vessel with Two Assemblies Failed and Functional Containment Barriers Fails to Retain Release (RFH-FDIV-4)

3.8.5.2.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the reactor vessel during in-vessel fuel moves. The dropped fuel assembly and the impacted fuel assembly are damaged. Functional containment barriers fail to perform their radionuclide retention functions.

3.8.5.2.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in two high burnup fuel assemblies. Radionuclides are released from the primary functional containment boundary at a degraded rate of 100 percent of cover gas volume per day, and subsequently released from the RXB at a degraded rate of 100 percent of the RXB volume per day. The mechanistic source term is summarized in Table 3.2-3.

3.8.5.2.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 3.88E-01 | 6.74E-01 | 1.62E+00 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.66E-07 | 1.19E-05 | 4.41E-05 |

3.8.5.3 Fuel Handling Event Occurs While Moving Fuel Assembly in the SFP with Two Assemblies Failed and FHB Barrier Fails to Retain Release (RFH-FDSP-2)

3.8.5.3.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the spent fuel pool during fuel moves. The dropped fuel assembly and an impacted fuel assembly are damaged. The Fuel Handling Building (FHB) fails to perform its radionuclide retention function. The fuel cladding is failed within this scenario, and a radionuclide release occurs.


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3.8.5.3.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in two high burnup fuel assemblies. The radiological consequences are increased by a factor of two from those identified in RFH-FDSP-1 to account for two damaged fuel assemblies. The mechanistic source term is summarized in Table 3.2-9.

3.8.5.3.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 2.40E-01 | 6.62E-01 | 1.01E+00 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 2.41E-06 | 9.74E-05 | 3.72E-04 |

3.8.5.4 Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Fails to Retain Release (FHB Credited) (RFH-ESWR-2)

3.8.5.4.1 LBE Description

The postulated initiating event is a sodium-water reaction in the Pool Immersion Cell (PIC) that causes fuel cladding failure. The FHB boundary is not failed and the PIC to BLTC boundary is assumed to fail.

3.8.5.4.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in one high burnup fuel assembly into the FHB. The FHB volume is released to the environment through FHB NHV filtration. The radiological consequences calculated for RFH-FDSP-1 are increased to scale the SFP decontamination factor to an effective FHB NHV filtration removal efficiency. The mechanistic source term is summarized in Table 3.2-9.

3.8.5.4.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.20E+00 | 3.31E+00 | 5.05E+00 |


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The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 4.90E-07 | 3.40E-05 | 1.61E-04 |

3.8.5.5 Excessive Sodium-Water Reaction in the PIC with PIC and BLTC Barrier Fails to Retain Release (FHB Not Credited) (RFH-ESWR-3)

3.8.5.5.1 LBE Description

The postulated initiating event is a sodium water reaction in the PIC that causes fuel cladding failure. The FHB boundary and the PIC and BLTC boundary are assumed to fail.

3.8.5.5.2 Mechanistic Source Term

The mechanistic source term is modeled as the failure of all fuel pins in one high burnup fuel assembly in the PIC while the BLTC is mated to the PIC. The radionuclide release is modeled as an instantaneous release from the FHB to the environment with no FHB NHV filtration. The radiological consequences of RFH-FDBL-1 are scaled to account for the larger leakage rates from the functional containment boundaries and the smaller release fractions. The mechanistic source term is summarized in Table 3.2-7.

3.8.5.5.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean dose for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE Dose [rem] | Mean 30-day EAB TEDE Dose [rem] | 95th Percentile 30-day EAB TEDE Dose [rem] |
|---|---|---|
| 1.00E+01 | 1.40E+01 | 2.10E+01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 3.12E-08 | 1.38E-06 | 4.94E-06 |

3.8.5.6 Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA without Passive Cooling, EVHM Barrier Successfully Retains Release (RFH-LMCA-1)

3.8.5.6.1 LBE Description

The postulated initiating event is a loss of EVHM active and passive cooling during refueling while handling a fuel assembly or a Lead Test Assembly (LTA) or Lead Demonstration Assembly (LDA). The EVHM boundary provides radionuclide retention. Fuel cladding is damaged, and a radionuclide release occurs.


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3.8.5.6.2 Mechanistic Source Term

The mechanistic source term analysis assumes an instantaneous release of the radionuclide inventory from all pins in one high burnup LTA or LDA in the EVHM. The fuel damage is assumed to occur 2.5 days after the LTA or LDA is removed from full power operation. LTAs will not be subjected to high burnup conditions, but the analysis conservatively assumes a high burnup assembly inventory for the LTA. The design leakage through the EVHM is released to the FHB and the FHB volume is assumed to be released instantaneously to the environment. Holdup and decay in the FHB is not considered. The mechanistic source term is summarized in Table 3.2-8.

3.8.5.6.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 9.53E-02 | 1.40E-01 | 2.09E-01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.04E-08 | 8.82E-07 | 3.43E-06 |

3.8.5.7 Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA without Passive Cooling, EVHM Barrier Fails to Retain Release (RFH-LMCA-2)

3.8.5.7.1 LBE Description

The postulated initiating event is a loss of EVHM active and passive cooling during refueling while handling a fuel assembly or an LTA or LDA. The EVHM fails to provide radionuclide retention. Fuel cladding damage and a radionuclide release occurs.

3.8.5.7.2 Mechanistic Source Term

The mechanistic source term analysis assumes an instantaneous release of the radionuclide inventory from all pins in one high burnup LTA or LDA in the EVHM. The EVHM source term was scaled for this LBE to account for the larger leakage rates from the functional containment boundaries and the smaller release fractions. The fuel drop is assumed to occur 2.5 days after the LTA or LDA is removed from full power operation. LTAs will not be subjected to high burnup conditions, but the analysis conservatively assumes a high burnup assembly inventory for the LTA. The assumed degraded leakage from the EVHM is released to the FHB and the FHB volume is assumed to be released instantaneously to the environment. Holdup and decay in the FHB is not considered. The mechanistic source term is summarized in Table 3.2-8.


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3.8.5.7.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 3.20E+01 | 4.60E+01 | 7.00E+01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 9.72E-09 | 8.82E-07 | 3.48E-06 |

3.8.5.8 Fuel Handling Event Occurs While Moving Fuel Assembly in the BLTC with BLTC Barrier Successfully Retains Release (RFH-FDBL-1)

3.8.5.8.1 LBE Description

The postulated initiating event is a fuel handling event while moving fuel in the bottom loading transfer cask (BLTC). The BLTC boundary is not failed. Fuel cladding is damaged, and a radionuclide release occurs.

3.8.5.8.2 Mechanistic Source Term

The mechanistic source term modeled as the failure of all fuel pins in one high burnup fuel assembly in the EVHM is used because a fuel assembly failure in the EVHM will result in more severe consequences than a failure in the BLTC. Radionuclides are released from the EVHM at the design EVHM leakage rate, and subsequently released from the FHB to the environment with no credit for the FHB radionuclide retention. The analysis assumes that time after shutdown from power for the fuel assembly in the EVHM is a minimum of 210 days. The mechanistic source term is summarized in Table 3.2-7.

3.8.5.8.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 2.91E-02 | 4.32E-02 | 6.36E-02 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 4.63E-08 | 6.03E-05 | 2.09E-04 |


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### 3.8.5.9 Fuel Handling Event Occurs While Moving Fuel Assembly in the BLTC with BLTC Barrier Fails to Retain Release (RFH-FDBL-2)

#### 3.8.5.9.1 LBE Description

The postulated initiating event is a fuel handling event while moving fuel in the BLTC. The BLTC boundary is failed. Fuel cladding is damaged, and a radionuclide release occurs.

#### 3.8.5.9.2 Mechanistic Source Term

The mechanistic source term modeled as the failure of all fuel pins in one high burnup fuel assembly in the EVHM is used because a fuel assembly failure in the EVHM will result in more severe consequences than a failure in the BLTC. The EVHM source term was scaled for this LBE to account for the larger leakage rates from the functional containment boundaries and the smaller release fractions. The analysis assumes that time after shutdown from power for the fuel assembly in the EVHM is a minimum of 210 days. The assumed degraded leakage from the EVHM is released to the FHB and the FHB volume is assumed to be released instantaneously to the environment. Holdup and decay in the FHB is not considered. The mechanistic source term is summarized in Table 3.2-7.

#### 3.8.5.9.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean dose for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE Dose [rem] | Mean 30-day EAB TEDE Dose [rem] | 95th Percentile 30-day EAB TEDE Dose [rem] |
|---|---|---|
| 1.00E+01 | 1.40E+01 | 2.10E+01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 2.41E-08 | 6.10E-07 | 2.33E-06 |

### 3.8.5.10 Fuel Handling Event Occurs While Moving Fuel Assembly or an LTA or LDA in the EVHM with EVHM Barrier Successfully Retains Release (RFH-FDEM-1)

#### 3.8.5.10.1 LBE Description

The postulated initiating event is a fuel handling event while moving fuel or an LTA or LDA fuel assembly in the EVHM. The EVHM boundary is not failed. Fuel cladding is damaged, and a radionuclide release occurs.

#### 3.8.5.10.2 Mechanistic Source Term

The mechanistic source term analysis assumes an instantaneous release of the radionuclide inventory from all pins in one high burnup LTA or LDA fuel assembly in the EVHM. The fuel drop is assumed to occur 2.5 days after the LTA or LDA assembly is removed from full power


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operation. LTAs will not be subjected to high burnup conditions, but the analysis conservatively assumes a high burnup inventory for the LTA. The mechanistic source term is summarized in Table 3.2-8.

3.8.5.10.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 9.53E-02 | 1.40E-01 | 2.09E-01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 3.34E-06 | 8.80E-05 | 3.27E-04 |

3.8.5.11 Fuel Handling Event Occurs While Moving Fuel Assembly or an LTA or LDA in the EVHM with EVHM Barrier Fails to Retain Release (RFH-FDEM-2)

3.8.5.11.1 LBE Description

The postulated initiating event is a fuel handling event while moving fuel or an LTA or LDA in the EVHM. The EVHM boundary is assumed to fail. Fuel cladding is damaged, and a radionuclide release occurs.

3.8.5.11.2 Mechanistic Source Term

The mechanistic source term analysis assumes an instantaneous release of the radionuclide inventory from all pins in one high burnup LTA or LDA in the EVHM. The fuel drop is assumed to occur 2.5 days after the LTA or LDA is removed from full power operation. LTAs will not be subjected to high burnup conditions, but the analysis conservatively assumes a high burnup assembly inventory for the LTA. The mechanistic source term and radiological consequence results are scaled to assume EVHM leakage is 100 percent of the EVHM volume per day. The assumed degraded leakage from the EVHM is released to the FHB and the FHB volume is assumed to be released instantaneously to the environment. Holdup and decay in the FHB is not considered. The mechanistic source term is summarized in Table 3.2-8.

3.8.5.11.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 3.20E+01 | 4.60E+01 | 7.00E+01 |


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The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 7.28E-09 | 5.25E-07 | 1.99E-06 |

3.8.5.12 Fuel Handling Event Occurs While Moving Fuel Assembly in the EVST with No Damage (RFH-FDET-BL)

3.8.5.12.1 LBE Description

The postulated initiating event is a fuel handling event while moving fuel in the Ex-Vessel Storage Tank (EVST). The EVST boundary is not failed. Fuel cladding is not damaged, and no radionuclide release occurs.

3.8.5.12.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.

3.8.5.13 Fuel Handling Event Occurs While Moving Fuel Assembly in the EVST with a Single Assembly Failed and EVST Barrier Successfully Retains Release (RFH-FDET-1)

3.8.5.13.1 LBE Description

The postulated initiating event is a fuel handling event while moving fuel in the EVST. The EVST boundary is not failed. Fuel cladding is damaged, and a radionuclide release occurs.

3.8.5.13.2 Mechanistic Source Term

The mechanistic source term is modeled as the failure of all fuel pins in a single high burnup fuel assembly in the EVHM. The EVHM boundary is not failed, but no credit is taken for the FHB radionuclide retention. The fuel assembly drop is assumed to occur 210 days after the fuel assembly is removed from the active core. The EVHM source term is used because a fuel assembly failure in the EVHM will result in more severe consequences than a failure in the EVST. The mechanistic source term is summarized in Table 3.2-7.

3.8.5.13.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 2.91E-02 | 4.32E-02 | 6.36E-02 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 5.92E-07 | 1.61E-05 | 6.01E-05 |


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### 3.8.5.14 Fuel Handling Event Occurs While Moving Fuel Assembly in the PIC with PIC Barrier Successfully Retains Release (RFH-FDPI-BL)

#### 3.8.5.14.1 LBE Description

The postulated initiating event is a fuel handling event while moving a fuel assembly in the PIC. The PIC and BLTC boundary is not failed. Fuel cladding is damaged, and a radionuclide release occurs.

#### 3.8.5.14.2 Mechanistic Source Term

The EVHM source term is used because a fuel assembly failure in the EVHM will result in more severe consequences than a failure in the PIC. The mechanistic source term is modeled as the failure of all fuel pins in one high burnup fuel assembly in the EVHM. Radionuclides are released from the EVHM at the normal EVHM leakage rate, and subsequently released from the FHB to the environment with no credit for the FHB radionuclide retention. The analysis assumes that time after shutdown from power for the fuel assembly in the EVHM is a minimum of 210 days. The mechanistic source term is summarized in Table 3.2-7.

#### 3.8.5.14.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean dose for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE Dose [rem] | Mean 30-day EAB TEDE Dose [rem] | 95th Percentile 30-day EAB TEDE Dose [rem] |
|---|---|---|
| 2.91E-02 | 4.32E-02 | 6.36E-02 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.98E-06 | 5.10E-05 | 1.93E-04 |

### 3.8.5.15 Fuel Handling Event Occurs While Moving Fuel Assembly in the PIC with PIC Barrier Fails to Retain Release (RFH-FDPI-1)

#### 3.8.5.15.1 LBE Description

The postulated initiating event is a fuel handling event while moving a fuel assembly in the PIC. The PIC and BLTC boundary is assumed to fail. Fuel cladding is damaged, and a radionuclide release occurs.

#### 3.8.5.15.2 Mechanistic Source Term

The mechanistic source term modeled as the failure of all fuel pins in one high burnup fuel assembly in the EVHM is used because a fuel assembly failure in the EVHM will result in more severe consequences than a failure in the PIC. The source term was scaled to account for the larger leakage rates from the functional containment boundaries and the smaller release


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fractions. The analysis assumes that time after shutdown from power for the fuel assembly in the EVHM is a minimum of 210 days. The assumed degraded leakage from the EVHM is released to the FHB and the FHB volume is assumed to be released instantaneously to the environment. Holdup and decay in the FHB is not considered. The mechanistic source term is summarized in Table 3.2-7.

3.8.5.15.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 1.00E+01 | 1.40E+01 | 2.10E+01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.98E-08 | 5.10E-07 | 1.93E-06 |

3.8.5.16 Fuel Handling Event Occurs While Moving Fuel Assembly in the PRC with PRC Barrier Successfully Retains Release (RFH-FDRC-1)

3.8.5.16.1 LBE Description

The postulated initiating event is an LTA or LDA fuel assembly dropped inside the Pin Removal Cell. The Pin Removal Cell boundary is not failed. Fuel cladding is damaged, and a radionuclide release occurs.

3.8.5.16.2 Mechanistic Source Term

The mechanistic source term analysis assumes an instantaneous release of the radionuclide inventory from all pins in one high burnup LTA or LDA fuel assembly in the EVHM resulting in the rupture of the cladding of all pins within that assembly. Radionuclides are released from the EVHM at the normal EVHM leakage rate, and subsequently released from the FHB to the environment with no credit for the FHB radionuclide retention. The assumption that the event occurs in the EVHM is acceptable because the leakage rate from the PRC is less than that assumed from the EVHM, and the remaining flow path to the environment is the same. The fuel drop is assumed to occur 2.5 days after the LTA or LDA is removed from full power operation. LTAs will not be subjected to high burnup conditions, but the analysis conservatively assumes a high burnup assembly inventory for the LTA. The mechanistic source term is summarized in Table 3.2-8.

3.8.5.16.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.


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| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 9.53E-02 | 1.40E-01 | 2.09E-01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 2.59E-06 | 6.79E-05 | 2.50E-04 |

3.8.6 Local Fuel Fault BDBEs

3.8.6.1 Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Fails to Retain Release (Vessel Head Success) (LFF-SAO-1)

3.8.6.1.1 LBE Description

The postulated initiating event is a blockage of fuel subchannels or other localized faults within the reactor core during full power operating conditions, resulting in fuel damage. Operators manually shutdown the reactor due to exceeding failed fuel activity limits. The reactor head boundary is not failed, but the RXB fails to perform its radionuclide retention function and a radionuclide release occurs. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.

3.8.6.1.2 Mechanistic Source Term

The mechanistic source term analysis assumes that all fuel pins in a single high burnup fuel assembly are failed. The event is postulated to be a rapid transient where the power to flow imbalance immediately results in a rapid heat up from nominal conditions, at which point the cladding of all pins is assumed to fail releasing fission products into the pool.

The analysis assumes particulates are released from the fuel pins directly into the hot pool region. The leakage rate through the primary functional containment boundary to the HAA is assumed to be 1 percent per day of the cover gas region volume. The leakage rate out of the HAA is assumed to be 100 percent of the HAA volume per day. The analysis assumes leakage from the HAA is released directly to the environment. Holdup and decay in the RXB are not considered. The mechanistic source term is summarized in Table 3.2-4.

3.8.6.1.3 Radiological Consequences

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 9.29E-02 | 1.48E-01 | 2.37E-01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.


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| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.19E-08 | 1.02E-06 | 4.21E-06 |

3.8.6.2 Core Blockage and Local Faults with Non-Passive IAC and Functional Containment Barrier Successfully Retains Release (Vessel Head Failed) (LFF-SAO-2)

3.8.6.2.1 LBE Description

The postulated initiating event is a blockage of fuel subchannels or other localized faults within the reactor core during full power operating conditions, resulting in fuel damage. Operators manually shutdown the reactor due to exceeding failed fuel activity limits. The reactor head fails to perform its radionuclide retention function, the RXB successfully performs its radionuclide function, and a radionuclide release occurs. The final conditions result in the reactor in safe shutdown with the IAC non-passive mode providing decay heat removal.

3.8.6.2.2 Mechanistic Source Term

The mechanistic source term analysis assumes that all fuel pins in a single high burnup fuel assembly are failed. The event is postulated to be a rapid transient where the power-to-flow imbalance immediately results in a rapid heat up from nominal conditions, at which point the cladding of all pins is assumed to fail releasing fission products into the pool.

The analysis assumes particulates are released from the fuel pins directly into the hot pool region. The leakage rate through the primary functional containment boundary to the HAA is assumed to be 1 percent per day of the cover gas region volume. Although the HAA integrity is maintained during the postulated event, the leakage rate out of the HAA is assumed to be 100 percent of the HAA volume per day. The analysis assumes leakage from the HAA is released directly to the environment. Holdup and decay in the RXB are not considered. The mechanistic source term is summarized in Table 3.2-4.

3.8.6.2.3 Radiological Consequence Results

The 5th percentile, 95th percentile, and mean 30-day TEDE at the EAB for this BDBE are listed below.

| 5th Percentile 30-day EAB TEDE [rem] | Mean 30-day EAB TEDE [rem] | 95th Percentile 30-day EAB TEDE [rem] |
|---|---|---|
| 9.29E-02 | 1.48E-01 | 2.37E-01 |

The 5th percentile, 95th percentile, and mean frequency for this BDBE are listed below.

| 5th Percentile Frequency | Mean Frequency | 95th Percentile Frequency |
|---|---|---|
| 1.26E-07 | 9.13E-06 | 3.71E-05 |


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## 3.8.7 Other BDBEs

### 3.8.7.1 Loss of SFP Cooling with Makeup Restored (RFH-LSPC-1)

#### 3.8.7.1.1 LBE Description

The postulated initiating event is a loss of spent fuel pool cooling during full power, low power, shutdown, or refueling operating conditions. The normal spent fuel pool cooling system is not restored, but emergency makeup is successfully implemented. Makeup to spent fuel pool is adequate for spent fuel assembly cooling and fuel cladding is not damaged.

#### 3.8.7.1.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.


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# 3.9 Design Basis Accidents

Design basis accidents (DBAs) are postulated accidents that are used to set design criteria and performance objectives for the design of safety-related structures, systems, or components. DBAs are derived from design basis events based on the capabilities and reliabilities of safety-related structures, systems, or components needed to mitigate and prevent accidents, respectively. DBAs are derived from the design basis events by prescriptively assuming that only safety-related structures, systems, or components are available to mitigate postulated accident consequences to within the 10 CFR 50.34 dose limits.

DBAs are analyzed in accordance with the methodologies in Section 3.3.1. Mechanistic source term summary tables are provided in Section 3.2.

## 3.9.1 Loss of Primary Flow DBAs

### 3.9.1.1 Loss of One Primary Sodium Pump (DHP-L1PP-CN)

#### 3.9.1.1.1 LBE Description

The postulated initiating event is a partial loss of primary flow due to the loss of a single Primary Sodium Pump (PSP) during full power operating conditions. A reactor scram on high high power-to-flow ratio occurs. The unaffected PSP and Intermediate Sodium Pumps (ISPs) trip on high high primary sodium temperature, and the PSP coasts down. The final conditions result in the reactor in safe shutdown with decay heat removal provided by the Reactor Air Cooling System (RAC).

#### 3.9.1.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.9.1.2 Loss of Offsite Power (DHP-LOOP-CN)

#### 3.9.1.2.1 LBE Description

The postulated initiating event is a loss of offsite power during full power operating conditions, resulting in a reactor scram on high high power-to-flow ratio. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

#### 3.9.1.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.9.1.3 Loss of Offsite Power While at Low Power (SUD-LOOP-CN)

#### 3.9.1.3.1 LBE Description

The postulated initiating event is a loss of offsite power during low power operating conditions. The reactor scrams due to the loss of primary flow. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

#### 3.9.1.3.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

## 3.9.2 Increase or Decrease in Heat Removal DBAs

### 3.9.2.1 Loss of Heat Sink (DHS-ISTL-CN)

#### 3.9.2.1.1 LBE Description

The postulated initiating event is a loss of heat removal from one intermediate sodium loop during full power operating conditions. A reactor scram on high high cold pool temperature occurs. PSPs and ISPs trip on high high primary sodium temperature, and the PSPs coast down. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

#### 3.9.2.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.9.2.2 Energy Island Transient (DHS-RNBK-CN)

#### 3.9.2.2.1 LBE Description

The postulated initiating event is a reactor power runback demand initiated by a decrease or increase in heat removal from the NSS during full power operating conditions. This event is also assumed to bound other events that generate reactor power runback signals. A reactor power runback may be initiated, but a reactor scram occurs before the reactor power runback is complete. PSPs and ISPs trip on high high primary sodium temperature, and the PSPs coast down. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

#### 3.9.2.2.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.9.2.3 Loss of One Train of IAC While Shutdown (SUD-IACA-CN)

#### 3.9.2.3.1 LBE Description

The postulated initiating event is the loss of an Intermediate Air Cooling System (IAC) train during low power or shutdown operating conditions. The final conditions result in the reactor safe and stable at low power or in safe shutdown with the RAC providing decay heat removal.

#### 3.9.2.3.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.9.2.4 IHX Tube Break (IPI-IHEL-CN)

#### 3.9.2.4.1 LBE Description

The postulated initiating event is an Intermediate Heat Exchanger tube break during full power operating conditions, resulting in intermediate sodium leakage into the Primary Heat Transport System (PHT) and loss of heat removal from one intermediate sodium loop. A reactor scram on high high primary sodium level occurs. Intermediate sodium volume that can drain into the PHT is limited by the Intermediate Heat Transport System (IHT) physical arrangement as described in Section 7.1.4, and the reactor vessel is not overfilled with sodium. PSPs and ISPs trip on high high primary sodium temperature, and PSPs coast down. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

#### 3.9.2.4.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.9.2.5 Intermediate Heat Exchanger Secondary-to-Primary Leak While Shutdown (SUD-IHEL-CN)

#### 3.9.2.5.1 LBE Description

The postulated initiating event is an Intermediate Heat Exchanger tube break during low power or shutdown operating conditions, resulting in intermediate sodium leakage into the PHT and loss of heat removal from one intermediate sodium loop. If the event occurs at low power, a reactor scram on high high primary sodium level occurs. Intermediate sodium volume that can drain into the PHT is limited by the IHT physical arrangement as described in Section 7.1.4, and the reactor vessel is not overfilled with sodium. The final conditions result in the reactor in safe shutdown with decay heat removal provided by the RAC.

#### 3.9.2.5.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.


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### 3.9.3 Reactivity DBAs

#### 3.9.3.1 Seismic Event (SEV-SSE-CN)

##### 3.9.3.1.1 LBE Description

The postulated initiating event is a safe shutdown earthquake during full power, low power, shutdown, or refueling operating conditions.

The seismic reactivity insertion DBA is analyzed at full power operating conditions using a sine wave over a range of frequencies that encapsulates the combined safe shutdown earthquake lateral and vertical reactivity time history. This ensures a plausible spectrum of profiles are evaluated and the DBA is conservatively modeled. The peak reactivity insertion for the safe shutdown earthquake is approximately $0.30.

The sensitivity analysis for the seismic event considers three scenarios: nominal, PSP and ISP rampdown failure during reactor scram, and loss of offsite power. The loss of offsite power analysis results in the most limiting short term response. PSPs and ISPs trip due to loss of power and the PSPs coast down. The reactor scrams on high high neutron flux rate or high high power-to-flow ratio. The control rod drop time is assumed to be degraded due to the seismic event. The final conditions result in the reactor in safe shutdown with the RAC providing decay heat removal.

##### 3.9.3.1.2 Radiological Consequences

The fuel performance acceptance criteria are satisfied. Therefore, fuel integrity is maintained for the peak fuel pin and there is no radiological release for this LBE.

### 3.9.4 Release from Ex-Vessel Systems DBAs

#### 3.9.4.1 SCG Leak Downstream the SCG Cell (RRS-CGR-CN)

##### 3.9.4.1.1 LBE Description

A Sodium Cover Gas System (SCG) leak occurs downstream of the Head Access Area (HAA) primary functional containment boundary isolation valve during full power, low power, or shutdown operating conditions. Fuel cladding is not failed, but a radionuclide release occurs.

##### 3.9.4.1.2 Mechanistic Source Term

The vapor trap cell compartment has a lower leakage rate than the HAA, so this analysis conservatively uses the HAA leakage rate. The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from the cover gas volume contained in the reactor vessel into the HAA compartment at the initiation of the event. The HAA releases directly to the environment through the Nuclear Island Heating, Ventilation, and Air Conditioning System (NHV) with no credit for filtration, and a two-hour full compartment exhaust time. The mechanistic source term is summarized in Table 3.2-14.


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3.9.4.1.3 Radiological Consequences

The enveloping 2-hour Exclusion Area Boundary (EAB) Total Effective Dose Equivalent (TEDE) and 30-day Low Population Zone (LPZ) TEDE for this LBE are listed below.

| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 7.75E+00 | 7.77E+00 |

3.9.4.2 SPS-I Leak at the Cold Trap (RRS-ISPL-CN)

3.9.4.2.1 LBE Description

The postulated initiating event is a Sodium Processing System (SPS) leak at the cold trap during full power, low power, shutdown, or refueling operating conditions. A release of precipitated tritium from the Intermediate SPS (SPS-I) cold trap occurs. Fuel cladding is not failed, but a radionuclide release from the SPS occurs.

3.9.4.2.2 Mechanistic Source Term

A release of precipitated tritium from one SPS-I cold trap is assumed to occur over the course of 24 hours. The radionuclide activity is assumed to be released directly to the environment with no credit taken for holdup in the Reactor Auxiliary Building (RAB). The mechanistic source term is summarized in Table 3.2-18.

3.9.4.2.3 Radiological Consequences

The enveloping 2-hour EAB TEDE and 30-day LPZ TEDE for this LBE are listed below.

| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 1.34E-01 | 5.30E-01 |

3.9.4.3 SPS-P Leak in the RXB (RRS-SPLX-CN)

3.9.4.3.1 LBE Description

The postulated initiating event is a leak in the SPS system inboard of the SPS isolation valves within the HAA during full power, low power, or shutdown operating conditions. The SPS pump trips on low low primary sodium level, terminating the loss of primary sodium inventory. Fuel cladding is not failed, but a radionuclide release from the SPS occurs.

3.9.4.3.2 Mechanistic Source Term

The SPS leak is modeled as a break in the SPS piping leading to the cold trap that drains into the SPS cell. The SPS piping is designed such that most of the piping is sloped down towards the reactor, minimizing the potential leakage volume from a break at the end of the pipe segment. The sodium volume released from this break is assumed to be 20 gallons. The SPS cell leaks


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with a degraded condition of 50 percent volume per day to the RAB. The RAB instantaneously exhausts to the environment with no credit for NHV filtration. The mechanistic source term is summarized in Table 3.2-11.

3.9.4.3.3 Radiological Consequences

The enveloping 2-hour EAB TEDE and 30-day LPZ TEDE for this LBE are listed below.

| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 8.13E-02 | 2.78E-01 |

3.9.4.4 SPS-P Leak in the RAB (RRS-SPLA-CN)

3.9.4.4.1 LBE Description

The postulated initiating event is a leak in the SPS system within the RAB during full power, low power, or shutdown operating conditions. The SPS pump trips on low low primary sodium level, terminating the loss of primary sodium inventory. Fuel cladding is not failed, but a radionuclide release from the SPS occurs.

3.9.4.4.2 Mechanistic Source Term

The SPS leak is modeled as a break in the SPS piping leading to the cold trap that drains into the SPS cell. The SPS piping is designed such that most of the piping is sloped down towards the reactor, minimizing the potential leakage volume from a break at the end of the pipe segment. The sodium volume released from this break is assumed to be 20 gallons. The SPS cell leaks with a degraded condition of 50 percent volume per day to the RAB. The RAB instantaneously exhausts to the environment with no credit for NHV filtration. The mechanistic source term is summarized in Table 3.2-11.

3.9.4.4.3 Radiological Consequences

The enveloping 2-hour EAB TEDE and 30-day LPZ TEDE for this LBE are listed below.

| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 8.13E-02 | 2.78E-01 |

3.9.4.5 RWG Leak (RRS-RWG-CN)

3.9.4.5.1 LBE Description

The postulated initiating event is a leak from a radioactive waste gas holdup tank during full power, low power, shutdown, or refueling operating conditions. A radionuclide release occurs.


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3.9.4.5.2 Mechanistic Source Term

The mechanistic source term is modeled as a release of the holdup tank contents to downstream Gaseous Radwaste Processing System (RWG) piping and the plant vent stack that occurs over a 10-minute period. The analysis assumes no credit for the FHB to contain the radionuclides released from the holdup tank. The mechanistic source term is summarized in Table 3.2-17.

3.9.4.5.3 Radiological Consequences

The enveloping 2-hour EAB TEDE and 30-day LPZ TEDE for this LBE are listed below.

| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 9.59E-02 | 9.59E-02 |

3.9.5 Fuel Handling DBAs

3.9.5.1 Fuel Handling Event Occurs while Moving Fuel Assembly in the Reactor Vessel (RFH-FDIV-CN)

3.9.5.1.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the reactor vessel during in-vessel fuel moves. The dropped fuel assembly and the impacted fuel assembly are damaged. Functional containment barriers successfully perform their radionuclide retention functions.

3.9.5.1.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in two high burnup fuel assemblies. The movement of fuel is assumed not to occur for at least 2.5 days after the reactor is shutdown from full power operation.

The analysis assumes particulates are released from the fuel pins directly into the hot pool region. The cover gas region, core assembly transfer tube, and Ex-Vessel Handling Machine (EVHM) are assumed connected during the release. The leakage rate through the primary functional containment boundary is assumed to be one percent per day of the connected cover gas region, core assembly transfer tube, and EVHM volume. The leakage rate out of the HAA cover gas, core assembly transfer tube, and EVHM are assumed to be 100 percent of the HAA this region volume per day. The analysis assumes leakage from the HAA is released directly to the environment. Holdup and decay in the RXB are not considered. The mechanistic source term is summarized in Table 3.2-6.

3.9.5.1.3 Radiological Consequences

The enveloping 2-hour EAB TEDE and 30-day LPZ TEDE for this LBE are listed below.

| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 1.74E+00 | 3.19E+00 |


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### 3.9.5.2 Fuel Handling Event Occurs while Moving Fuel Assembly in the SFP (RFH-FDSP-CN)

#### 3.9.5.2.1 LBE Description

The postulated initiating event is the drop of a fuel assembly in the spent fuel pool during fuel moves. The dropped fuel assembly is damaged, but there is no damage to fuel assemblies impacted by the dropped fuel assembly.

#### 3.9.5.2.2 Mechanistic Source Term

The mechanistic source term is modeled as an instantaneous release of the radionuclide inventory from all pins in one high burnup fuel assembly. The analysis for this DBA does not credit filtration by the NHV or the hold-up of radionuclides within the FHB to allow for decay and possible deposition prior to release.

Spent fuel assemblies are stored in the Ex-Vessel Storage Tank (EVST) after removal from the reactor, so the minimum time after shutdown from power for the handling of any spent fuel assembly in the spent fuel pool is assumed to be 310 days. Since radioactive iodine and most noble gasses have a short half-life, this post power decay time of 310 days means iodine and noble gas release have a minor impact on dose results.

When the fuel cladding is damaged, the fuel pin sodium bond material reacts with water in the spent fuel pool in an exothermic gas-producing reaction. The nature of this reaction is being studied for further characterization of the radionuclide retention in the spent fuel pool. The preliminary mechanistic source term analysis assumptions for radionuclide retention in the spent fuel pool are informed by RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Revision 0, with a relatively low decontamination factor to address the fuel pin sodium bond material and water reaction.

The fission product particulate that was released into the spent fuel pool from the solid fuel matrix is assumed to behave essentially the same as the particulate released from the fuel during a comparable LWR fuel handling accident. Therefore, this portion of the particulate is assumed to be completely retained within the spent fuel pool water inventory in accordance with the guidance for a fuel handling accident in RG 1.183 Appendix B. The portion of the radionuclide particulate that had migrated into the fuel pin plenum is involved in the reaction with the sodium bond material displaced into the fuel pin plenum as it contacts the spent fuel pool water. The analysis assumes a reduced spent fuel pool decontamination factor from the decontamination factor in RG 1.183 for the fission product particulate assigned to the aerosol removal efficiency to address the impact of the sodium bond material and water reaction.

Radionuclides released from the spent fuel pool are transported to the FHB. The FHB volume releases instantaneously to the environment with no credit for NHV filtration. The mechanistic source term is summarized in Table 3.2-9.

#### 3.9.5.2.3 Radiological Consequences Results

The enveloping 2-hour EAB TEDE and 30-day LPZ TEDE for this LBE are listed below.


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| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 7.33E+00 | 7.33E+00 |

3.9.5.3 Excessive Sodium-Water Reaction in the PIC (RFH-ESWR-CN)

3.9.5.3.1 LBE Description

The postulated initiating event is a sodium-water reaction in the Pool Immersion Cell (PIC) during fuel handling activities that causes fuel cladding failure. The BLTC boundary successfully retains the radionuclide release following the sodium-water reaction, and the PIC boundary is not credited.

3.9.5.3.2 Mechanistic Source Term

The mechanistic source term modeled as the failure of all fuel pins in one high burnup fuel assembly in the EVHM is used because a fuel assembly failure in the EVHM will result in more severe consequences than a failure in the PIC. The fuel assembly damage is assumed to occur 210 days after reactor shutdown. No credit is taken for FHB radionuclide retention. The mechanistic source term is summarized in Table 3.2-7.

3.9.5.3.3 Radiological Consequences

The enveloping 2-hour EAB TEDE and 30-day LPZ TEDE for this LBE are listed below.

| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 2.01E-02 | 6.09E-01 |

3.9.5.4 Loss of EVST Active Cooling While Storing Fuel Assemblies (RFH-LTCA-CN)

3.9.5.4.1 LBE Description

The postulated initiating event is a loss of EVST cooling with spent fuel in the EVST. EVST passive cooling is unavailable or provides inadequate cooling to the assemblies contained in the EVST. The EVST boundary is not failed. Analysis demonstrates that no fuel damage occurs for at least 72 hours following the loss of cooling. Longer term degraded heat removal conditions require further assessment. Fuel cladding is not damaged, and no radionuclide release occurs.

3.9.5.4.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.

3.9.5.5 Loss of BLTC Active Cooling While Handling Fuel Assembly (RFH-LBCA-CN)

3.9.5.5.1 LBE Description

The postulated initiating event is a loss of bottom loading transfer cask active cooling with spent fuel in the bottom loading transfer cask. Passive cooling is maintained during this event. Fuel cladding is not damaged, and no radionuclide release occurs.


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3.9.5.5.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.

3.9.5.6 Loss of EVHM Active Cooling While Handling Fuel Assembly or an LTA or LDA (RFH-LMCA-CN)

3.9.5.6.1 LBE Description

The postulated initiating event is a loss of EVHM active cooling with spent fuel or a Lead Test Assembly (LTA) or Lead Demonstration Assembly (LDA) in the EVHM. Passive cooling is maintained during this event. Fuel cladding is not damaged, and no radionuclide release occurs.

3.9.5.6.2 Radiological Consequences

Fuel cladding integrity is maintained and there is no radiological release for this LBE.

3.9.6 Local Fuel Fault DBAs

3.9.6.1 Core Blockage and Local Faults (LFF-SAO-CN)

The postulated initiating event is a blockage of fuel subchannels or other localized faults within the reactor core during full power operating conditions, resulting in fuel damage. Operators fail to shut down the reactor and the reactor continues operating at full power. The affected fuel assembly fails, and a radionuclide release occurs.

3.9.6.1.1 Mechanistic Source Term

The mechanistic source term analysis assumes that all fuel pins in a single high burnup fuel assembly fail. The event is postulated to be a rapid transient where the power-to-flow imbalance immediately results in a rapid heat up from nominal conditions, at which point the cladding of all pins is assumed to fail releasing fission products into the pool.

Particulates are assumed to be released from the fuel pins directly into the hot pool region. The leakage rate through the primary functional containment boundary to the HAA is assumed to be one percent per day of the cover gas region volume. The leakage rate out of the HAA is assumed to be 100 percent of the HAA volume per day. The analysis assumes leakage from the HAA is released directly to the environment. Holdup and decay in the RXB are not considered. The mechanistic source term is summarized in Table 3.2-4.

3.9.6.1.2 Radiological Consequences

The enveloping 2-hour EAB TEDE and 30-day LPZ TEDE for this LBE are listed below.

| Enveloping 2-Hour EAB TEDE [rem] | Enveloping 30-day LPZ TEDE [rem] |
|---|---|
| 1.33E-01 | 9.60E-01 |


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# 3.10 10 CFR 50.155 Mitigation Strategies

Mitigation of beyond design basis events will be addressed at the operating license stage in accordance with 10 CFR 50.155.


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# 3.11 Fuel System Design

The fuel qualification plan and preliminary design are included in NAT-2806, “TerraPower, LLC (TerraPower) Natrium Topical Report: Fuel and Control Assembly Qualification,” Revision 0 (Reference 3.11-1), which also identifies acceptance criteria and preliminary qualification results. The report is incorporated by reference. Final results, including radionuclide retention requirements and associated fuel criteria to ensure conformance with Principal Design Criterion 10, will be provided at the operating license stage.

## References

3.11-1 NAT-2806, “TerraPower, LLC (TerraPower) Natrium Topical Report: Fuel and Control Assembly Qualification”, Revision 0 (proprietary information, not publicly available)



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# 3.12 Nuclear Design

## 3.12.1 Nuclear Design Summary

### 3.12.1.1 Description

The nuclear design of the reactor core is inherently coupled with the Reactor Core System design (Section 7.1.1), the Reactor Enclosure System design (Section 7.1.2), and the Primary Heat Transport System design (Section 7.1.3). The scope of this section is focused on the nuclear design of the core, specifically, the determination and evaluation of reactivity coefficients, power distributions, and shutdown margin. Core design parameters are summarized in Table 7.1.1-1.

TP-LIC-RPT-0011, “Core Design and Thermal Hydraulic Technical Report,” Revision 0 (Reference 3.12-1), provides a description of the evaluation models, computer codes, and analyses that support the nuclear design of the core. Sections 1 through 8 of TP-LIC-RPT-0011 are incorporated by reference. See Section 1.4.2 for a list of documents incorporated by reference. This Section describes the interrelationship between the design and analyses presented here and in TP-LIC-RP-0011.

The core design is evaluated with initial reactor core loading conditions, specifically, fuel loading conditions at beginning of life and end of first cycle. The core analysis applies fuel and control rod manipulations to achieve an equilibrium fuel cycle, which is evaluated at the beginning of equilibrium cycle, middle of equilibrium cycle, and end of equilibrium cycle. The core design and analysis will support the development of the Core Operating Limits Report at the operating license stage.

### 3.12.1.2 Reactivity Coefficients

The following reactivity coefficients are important for Natrium core design and evaluation:

- Fuel Doppler constant
- Fuel density coefficient
- Coolant density coefficient
- Cladding density coefficient
- Structure density coefficient
- Radial expansion coefficients

The fuel Doppler constant and fuel density coefficients reflect the change in reactivity due to a change in fuel temperature. The coolant density coefficient is calculated based on the change in coolant void reactivity worth. The cladding density coefficient is the change in reactivity due to the change in temperature. Structure density coefficient and the radial expansion coefficients reflect the change in reactivity due to the expansion of the lower grid support structure and of the core assembly bowing that results from different power and flow conditions. The four coefficients which act in a rapid enough time window for consideration in prompt feedback characteristics are the fuel Doppler constant, the fuel density coefficient to capture changes in metallic fuel volume,


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the coolant density coefficient which captures temperature changes to sodium coolant density, and radial expansion coefficients which capture the changes in fuel movement with respect to the radial direction.

The methodology for developing reactivity coefficients is discussed in additional detail in Section 4.1.1.3 of TP-LIC-RPT-0011.

#### 3.12.1.3 Power Distribution

Power distribution calculations are discussed in Section 4.1.1.2 of TP-LIC-RPT-0011. Fuel assembly reconstructed pin power peaking factors are analyzed over all cycles and time nodes and the maximum power peaking factor values are provided along with the location of the assembly with the peaking fuel pin power.

The maximum enthalpy rise hot power peaking factor, $F_{\Delta H}$, is defined as the ratio of the maximum integrated reconstructed fuel rod power within the core at a given time-step to the average fuel rod power at the same time-step.

The axial peaking factor, $F_Z$, is defined as the maximum local reconstructed fuel rod power at any axial point in a fuel rod, divided by the average axial segment power of that same fuel rod.

The radial rod peaking factor, $F_{XY}$, is the maximum ratio of a fuel rod local reconstructed power to the average fuel rod local reconstructed power in the horizontal plane within the core.

#### 3.12.1.4 Shutdown Margin

Shutdown margin is the amount of reactivity by which the reactor is subcritical, or would be subcritical from a given condition, assuming that all control rods are inserted with the exception of at least the highest worth control rod, which is assumed to be fully withdrawn.

The shutdown margin calculation accounts for the following factors:

- Positive reactivity inserted from a reduction in temperature
- Cycle excess reactivity
- Margin for uncertainties

The methodology for determining shutdown margin is described in further detail in Section 4.1.1.5 of TP-LIC-RPT-0011.

#### 3.12.1.5 Decay Heat Generation Rate

The decay heat generation rate methodology predicts the heat produced due to the decay of radioactive isotopes within the reactor. These include fission products, activated non-fissionable materials, and actinides produced within the fuel. This rate is used in downstream analyses to ensure sufficient cooling of the reactor and individual assemblies is maintained after shutdown.

The methodology for determining decay heat generation rate is described in further detail in Section 4.1.1.6 of TP-LIC-RPT-0011.


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### 3.12.1.6 Reactor Stability

TP-LIC-RPT-0006, “Stability Methodology Topical Report,” Revision 0 (Reference 3.12-2), provides a description of the method used in evaluating and demonstrating compliance with Principal Design Criterion (PDC) 12. Sections 1 through 8 of TP-LIC-RPT-0006 are incorporated by reference. See Section 1.4.2 for a list of documents incorporated by reference.

In the demonstration of this methodology using preliminary design parameters, no instabilities are identified to occur over the range of power evaluated. Analytical results confirming compliance with PDC 12 will be provided at the operating license stage.

### 3.12.1.7 Neutron Fluence

Neutron fluence and the resulting displacements per atom resulting from the fast fluence passing through the core and reactor vessel has been evaluated for the preliminary design based on models using stainless steel material. Results based on the final design and material selection will be available at the operating license stage.

The methodology for determining fluence and displacements per atom is described in further detail in Section 4.3 of TP-LIC-RPT-0011.

## 3.12.2 Design Basis

Consistent with PDC 10, the reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure 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 PDC 11, the reactor core and associated systems that contribute to reactivity feedback are 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.

Consistent with PDC 12, the reactor core; associated structures; and associated coolant, control, and protection systems are 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.

Consistent with PDC 26, the nuclear design analysis is performed to confirm that minimum of two reactivity control systems or means shall provide:

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


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2. A means which is independent and diverse from the others, is capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the specified acceptable system radionuclide release design limits for the fission product barriers are not exceeded.

3. 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 core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.

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

### 3.12.3 Codes Used in System Evaluation

Detailed discussion of the codes selected for use and the core neutronic evaluation models are discussed in Section 3 of TP-LIC-RPT-0011. Plans for the Validation and Verification of the codes used in these evaluation models is discussed in Section 5.1 of TP-LIC-RPT-0011.

## References

3.12-1 TerraPower, LLC “Core Design and Thermal Hydraulic Technical Report,” TP-LIC-RPT-0011, Revision 0

3.12-2 TerraPower, LLC “Stability Methodology Topical Report,” TP-LIC-RPT-0006, Revision 0


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# 3.13 Thermal and Hydraulic Design

## 3.13.1 Description

The thermal hydraulic design of the reactor core is inherently coupled with Reactor Core System design (Section 7.1.1), Reactor Enclosure System design (Section 7.1.2), and Primary Heat Transport System design (Section 7.1.3). The scope of this section is focused on the thermal hydraulic design of the core, specifically, heat transport from the reactor core to the primary sodium coolant.

TP-LIC-RPT-0011, “Core Design and Thermal Hydraulic Technical Report,” Revision 0 (Reference 3.13-1), provides a description of the evaluation models, computer codes, and analyses that support the thermal hydraulic design of the core. The discussion below describes the interrelationship between the design and analyses presented in Sections 7.1.1, 7.1.2, and 7.1.3 and in TP-LIC-RPT-0011.

### 3.13.1.1 Core Geometry

The core geometry is supported and constrained by the core barrel structures within the Reactor Enclosure System, which surrounds the removable core assemblies of the Reactor Core System and separates the Reactor Core System from the In-vessel Storage. The core design includes 162 fuel assemblies contained in hexagonal ducts. The fuel assemblies are surrounded radially by a region of reflector assemblies and then a region of shield assemblies before reaching the reactor core barrel.

There are multiple flow zones within the core, and these are mapped based on the combination of the inlet nozzles of the core assemblies, named universal stack nozzles, and the receptacle inlet where they are placed. There are up to five orifice zones per nozzle, with the combination of nozzle and receptacle mapping to different flow requirements. These universal stack nozzles are split into four types: an inner fuel nozzle and an outer fuel nozzle, a reflector assembly nozzle, and a shield assembly nozzle.

Additional core geometry and flow path details are described in Sections 7.1.1 and 7.1.2, and in Sections 2 and 4.1.2 of TP-LIC-RPT-0011.

## 3.13.2 Design Basis

Consistent with Principal Design Criterion 10, the thermal hydraulic design provides adequate transfer of heat from the fuel to the coolant to ensure that the Specified Acceptable System Radionuclide Release Design Limits will not be exceeded during normal operation and unplanned transients.

Consistent with Principal Design Criterion 15, the thermal hydraulic design provides adequate margin such that critical parameters of the safety-significant elements of the primary coolant boundary will not be exceeded during normal operation and anticipated operational occurrences.


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Consistent with Principal Design Criterion 34, the thermal hydraulic design provides for the removal of residual heat. Fission product decay heat and other residual heat is transferred from the reactor core at a rate such that Specified Acceptable System Radionuclide Release Design Limits and the design conditions of the primary coolant boundary are not exceeded during normal operation and anticipated operational occurrences.

A summary discussion of Specified Acceptable System Radionuclide Release Design Limits is provided in Section 1.3.2 and specific design criteria compliance with the above Principal Design Criteria is described in Chapter 7.

### 3.13.3 Codes Used in System Evaluation

Detailed discussion of the codes selected for use and the thermal hydraulic evaluation models are discussed in Section 3.2 of TP-LIC-RPT-0011. Plans for the Validation and Verification of the codes used in these evaluation models is discussed in Section 5.2 of TP-LIC-RPT-0011.

### 3.13.4 Testing

Testing is planned to provide additional characterization for the behavior of sodium through the core assemblies and heat transfer within the Reactor Core System. Testing is also planned to support validation of the codes selected to model thermal hydraulic performance of the core. Results of this testing will be reflected in the operating license stage.

Planned testing includes characterization of temperature distributions within the assemblies, flow distribution from the high pressure inlet and through the core assemblies, pressure drop loss coefficients, and hold-down forces for the core assemblies. Long-duration testing is also planned to provide information on hydraulic performance of the orifice plates from high-velocity sodium flow, and to determine velocity limits based on the results of erosion testing.

Additional detail is described in Section 5.2 of TP-LIC-RPT-0011.

## References

3.13-1 TerraPower, LLC “Core Design and Thermal Hydraulic Technical Report,” TP-LIC-RPT-0011, Revision 0


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# 3.14 Criticality Safety of Fresh and Spent Fuel

This section demonstrates conformance with Principal Design Criterion 62, “Prevention of Criticality in Fuel Storage and Handling” (as described in Section 5.3) and 10 CFR 50.68, except for the U-235 enrichment limit defined in 10 CFR 50.68(b)(7).

Special nuclear material, other than fuel onsite, is less than the quantity necessary for a critical mass. Radiation monitoring in the Fuel Handling Building to detect excessive radiation levels is described in Section 7.6.6. A nuclear material control and accounting program will be provided at operating license stage.

## 3.14.1 Design Basis

Fuel is stored and handled by the following Structures, Systems, and Components:

- Water Pool Fuel Handling System described in Section 7.3.1
- Ex-Vessel Fuel Handling System described in Section 7.3.2
- In-Vessel Fuel Handling System described in Section 7.3.3
- Reactor Enclosure System described in Section 7.1.2

The criticality analysis of the Spent Fuel Pool (SFP) conforms to Regulatory Guide (RG) 1.240, “Fresh and Spent Fuel Pool Criticality Analysis,” Revision 0, that endorses Nuclear Energy Institute (NEI) 12-16, “Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants,” Revision 4 (Reference 3.14-1) to provide guidelines for criticality analysis of fuel storage areas. RG 1.240 is used as guidance to establish the methodology for criticality analysis in New Assembly Preconditioning Station (NAPS), Ex-Vessel Storage Tank (EVST), and In-Vessel Storage (IVS).

Structures, Systems, and Components that form the fuel storage locations and the handling equipment associated with the transfer of core assemblies are designed in accordance with Principal Design Criterion 62. The design prevents inadvertent criticality by using geometrically safe configurations and utilizing fixed borated neutron absorbers.

## 3.14.2 Methodology

The methodology used to evaluate criticality safety of fresh and spent fuel is described below. Subcriticality is determined by the calculation of $k_{eff}$ and assuring it is below the Upper Subcritical Limit (USL), while considering additional margin for uncertainty.

The USLs are developed based on applicable benchmarks neutronically similar to the fuel storage systems. The USL varies based on system conditions. For example, the USL for the IVS and EVST is different than the USL for the SFP because fuel in the IVS and EVST is moderated by sodium and fuel in the SFP is moderated by water.


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The computed $k_{eff}$ for the system plus twice the Monte Carlo uncertainty ($\sigma$) must be less than the USL:

$$k_{eff} + 2\sigma < \text{USL} \quad \text{(Equation 3.14-1)}$$

This ensures that the USL is met at a 95 percent probability and 95 percent confidence level.

The USL is defined as:

$$\text{USL} = 1 + \text{bias} - \sigma_{\text{bias}} - \Delta_{\text{SM}} - \Delta_{\text{AOA}} \quad \text{(Equation 3.14-2)}$$

Where,

- Bias is calculated as the difference between the calculated $k_{eff}$ and the critical experiment model. Because a positive bias can be non-conservative, the bias is set to zero when the calculated average $k_{eff}$ is greater than 1.0.
- $\sigma_{\text{bias}}$ is the uncertainty in the bias.
- $\Delta_{\text{SM}}$ is an administrative margin, typically 0.05.
- $\Delta_{\text{AOA}}$ is an additional subcritical margin to account for extensions in the area of applicability if the available benchmarks are not sufficiently similar to the fuel storage systems.

The USLs are computed using either methodologies outlined in NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” July 2001 (Reference 3.14-2) or Whisper-1.1 program described in LA-UR-14-26558, “Whisper: Sensitivity/Uncertainty-Based Computational Methods and Software for Determining Baseline Upper Subcritical Limits,” (Reference 3.14-3).

The methodology outlined in NUREG/CR-6698 selects applicable benchmark experiments similar to the plant system. Key parameters used to determine applicability include system geometry, U-235 enrichment, fissile material composition, neutron energy spectrum, coolant type, reflector material, fuel density, and temperature. Trends in the $k_{eff}$ results are examined as a function of the system parameters selected.

Experiments are selected from the NEA/NSC/DOC(95)03, “International Handbook of Evaluated Criticality Safety Benchmark Experiments” (Reference 3.14-4). These experiments can also be supplemented with key liquid metal fast reactor experiments from the NEA/NSC/DOC(2006)01, “International Handbook of Evaluated Reactor Physics Benchmark Experiments” (Reference 3.14-5).

After developing a list of applicable benchmarks, an MCNP 6.2 benchmark is developed for each experiment using LA-UR-17-29981, “MCNP User's Manual, Code Version 6.2,” 2017 (Reference 3.14-6). Bias, $\sigma_{\text{bias}}$, and $\Delta_{\text{AOA}}$ are computed using the guidance provided in NUREG/CR-6698.


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An alternate method of USL computation utilizes more advanced sensitivity and uncertainty methods and the Whisper-1.1 program (Reference 3.14-3). Whisper is developed by Los Alamos National Laboratory for criticality safety applications and is distributed with MCNP 6.2. Whisper includes approximately 1,100 MCNP benchmarks from the International Handbook of Evaluated Criticality Safety Benchmark Experiments that cover a wide range of criticality safety problems. Whisper uses sensitivity profile data, as computed by MCNP, along with nuclear data covariance files to calculate the bias, $\sigma_{bias}$, and $\Delta_{AOA}$ for an input application. Approximately 100 Whisper benchmarks will be selected.

Whisper also provides parameters to account for MCNP software errors and uncertainties in cross-section data. However, these computed parameters are generally much less than the standard $\Delta_{SM}$ value of 0.05 applied as a general administrative margin. Therefore, these parameters as provided by Whisper are not directly incorporated into the USL.

The following sections describe the methodology utilized to perform the criticality analysis for each spent fuel storage location. The final methodology, analyses, and results will be provided at the operating license stage. The use of either method for USL development is at the discretion of the organization performing the criticality analysis.

The procedures prohibiting the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under adverse moderation conditions by unborated water will be provided at the operating license stage.

#### 3.14.2.1 Spent Fuel Pool Storage

The SFP design, codes considered in the design of the SFP, controls for operation of the fuel handling equipment and fuel storage, layout, and associated subsystems and equipment are described in Section 7.3.1.

The SFP is used for storage of spent fuel assemblies; fresh fuel assemblies are not stored in the SFP. The spent fuel is stored in two types of racks as described in Section 7.3.1. Failed fuel is canned in a Failed Fuel Canister and inerted. Because each Failed Fuel Canister is certified as leak-tight, the criticality analysis of the SFP will credit moderator exclusion within each Failed Fuel Canister.

Borated stainless steel is used as a neutron absorber material in the racks; 90 percent of the boron is credited in the borated stainless steel neutron absorber plates.

The SFP analysis follows the guidelines of RG 1.240 and NEI 12-16.The SFP does not contain soluble boron. The racks for intact and failed fuel are modeled in sufficient detail to capture the physics of the system and contain conservative reactivity assumptions.


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The following models are used:

- Conservative fuel assembly model – the following fuel assembly parameters are varied to determine the most reactive condition: enrichment, duct thickness, pellet density, pellet diameter, cladding thickness, and pin pitch. Simplifications to the model are justified by the analysis. The fuel is modeled as fresh fuel with no burnup.
- Conservative rack model – evaluates the most reactive placement of fuel within the SFP (e.g., lateral shifting within compartments), water density, and rack dimensions.
- Cases that bound credible abnormal events – evaluates temperatures beyond normal operating range, dropped and mispositioned assemblies, design basis seismic events, and low water density.

#### 3.14.2.2 Ex-Vessel Storage Tank

The EVST design, codes considered in the design of the EVST, operation of the fuel handling equipment and fuel storage, layout, and associated subsystems are described in Section 7.3.2.

The EVST is used as a staging area for both fresh and spent fuel assemblies. The EVST is modeled in sufficient detail to capture the physics of the system and contain conservative reactivity assumptions.

The following models are used:

- Conservative fuel assembly model – the following fuel assembly parameters are modeled to determine the most reactive condition: enrichment, duct thickness, pellet density, pellet diameter, cladding thickness, and pin pitch. Items may be conservatively replaced with sodium, such as the inlet nozzle, wire wrap, and handling socket. Simplifications to the model will be justified by the analysis. The fuel is modeled as fresh fuel with no credit for burnup.
- Conservative EVST model – Evaluates the most reactive placement of fuel within the EVST (i.e., lateral shifting within compartments), sodium density, and EVST dimensions.
- Cases that bound credible abnormal events – Evaluates temperatures beyond normal operating range, dropped and mispositioned assemblies, design basis seismic events, fuel pins flooded with sodium from assumed breach, sodium at solid density, and moderator treated as a void.

#### 3.14.2.3 In-Vessel Storage

The IVS design, layout, and codes considered in the design are described in Section 7.1.2 The operation of the fuel handling equipment and associated subsystems are described in Section 7.3.3.

The IVS is used to store spent fuel assemblies. The IVS is located outside the core between the inner and outer core barrels.The model does not assume fuel assembly burnup for conservatism. Fuel is modeled as fresh at the maximum enrichment.


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A combined reactor core and IVS criticality model reports the $k_{eff}$ of the reactor core. The combined model of the reactor core and IVS reports $k_{eff}$ approximately equal to one and does not provide any information about the IVS. The exchange of neutrons between the reactor core and IVS is largely eliminated due to the distance between the core and IVS and the reflector and boron carbide shield assemblies along the perimeter of the reactor core.

In the IVS models, the core is removed and a white or reflective boundary is placed on the inner surface of the IVS. This boundary condition allows $k_{eff}$ for the IVS to be computed separate from the reactor core. Modeling the inner surface as white or reflective is more reactive than other modeling options, such as modeling the core as sodium.

The IVS is modeled in sufficient detail to capture the physics of the system and contain conservative reactivity assumptions.

The following models are used:

- Conservative fuel assembly model – the following fuel assembly parameters are varied to determine the most reactive condition: enrichment, duct thickness, pellet density, active fuel height, pellet diameter, cladding thickness, and pin pitch. Items may be conservatively replaced with sodium, such as the inlet nozzle, wire wrap, and handling socket. Simplifications to the model will be justified by the analysis. The fuel is modeled as fresh fuel with no credit for burnup.
- Conservative IVS model – Evaluates the most reactive placement of fuel within the IVS (i.e., lateral shifting within compartments), sodium density, IVS dimensions, and core barrel dimensions.
- Cases that bound credible abnormal events – Evaluates temperatures beyond normal operating range, dropped and mispositioned assemblies, design basis seismic events, fuel pins flooded with sodium from assumed breach, sodium at solid density, and moderator treated as a void.

#### 3.14.2.4 New Assembly Preconditioning Station

The NAPS design, layout, and codes considered in the design are described in Section 7.3.2. The NAPS is a preconditioning station that stores fresh fuel assemblies in a gas. The gas heats the fuel prior to insertion into the EVST. The analysis conservatively assumes moderation by unborated water. NAPS is modeled in sufficient detail to capture the physics of the system and contain conservative reactivity assumptions.

The following models are used:

- Conservative fuel assembly model - the following fuel assembly parameters will be modeled at the most reactive condition: enrichment, duct thickness, pellet density, pellet diameter, cladding thickness, and pin pitch. Simplifications will be justified by the analysis. The fuel will be modeled as fresh fuel.
- Conservative NAPS model - Evaluates the most reactive placement of fuel within the NAPS (i.e., lateral shifting within compartments), water density, and NAPS dimensions.


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- Cases that bound credible abnormal events - Evaluates temperatures beyond normal operating range, dropped and misplaced assemblies, and design basis seismic events.

## References


3.14-1 NEI, NEI 12-16, “Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants,” Revision 4.

3.14-2 NRC, NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” January 2001.

3.14-3 B. Kiedrowski, LA-UR-14-26558, “Whisper: Sensitivity/Uncertainty-Based Computational Methods and Software for Determining Baseline Upper Subcritical Limits,” Los Alamos National Laboratory.

3.14-4 OECD, NEA/NSC/DOC(95)03, “International Handbook of Evaluated Criticality Safety Benchmark Experiments”.

3.14-5 OECD, NEA/NSC/DOC(2006)01, “International Handbook of Evaluated Reactor Physics Benchmark Experiments,” 2019.

3.14-6 Los Alamos National Laboratory, LA-UR-17-29981, “MCNP User's Manual, Code Version 6.2,” 2017.




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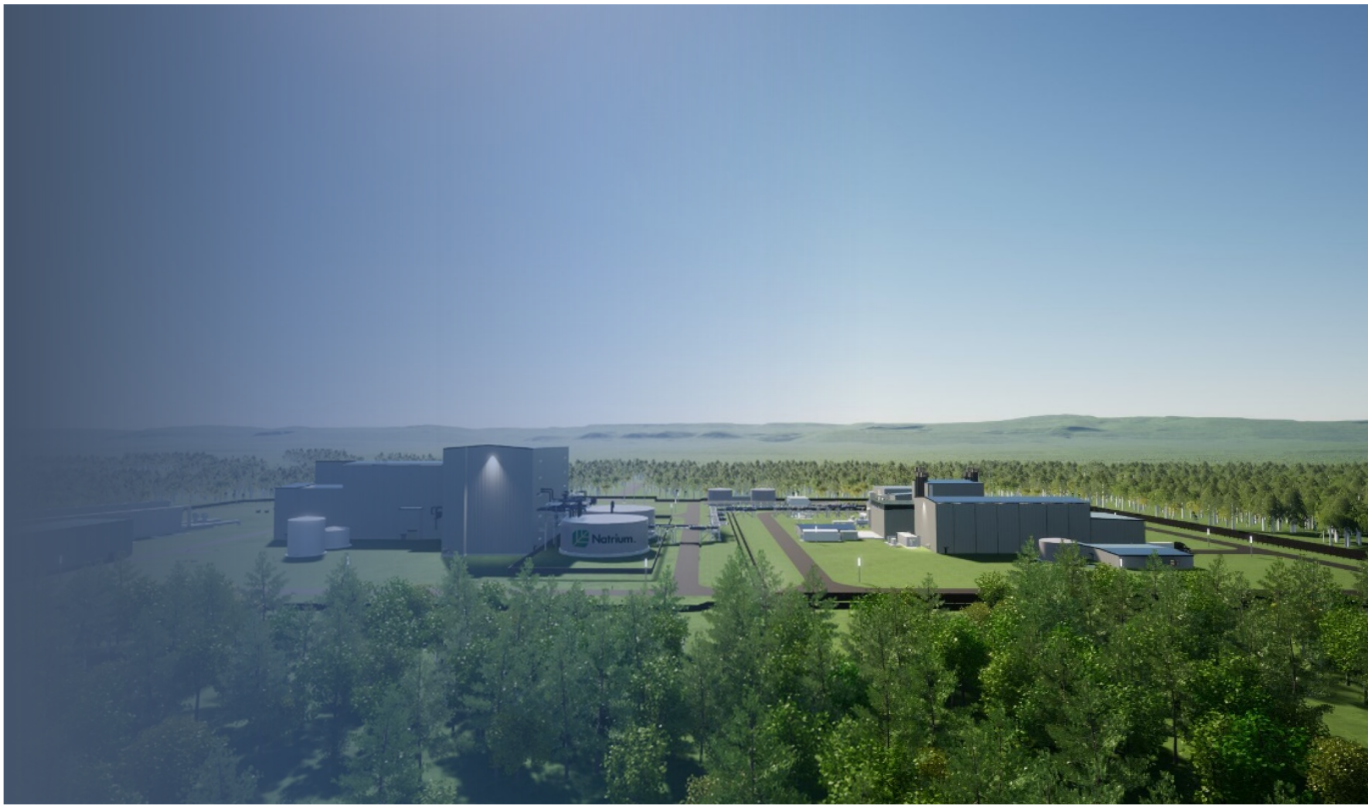

# Kemmerer Power Station Unit 1
# PSAR, Chapter 4

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## TABLE OF CONTENTS

**Chapter 4 Integrated Evaluations . . . . . 4.1-1**

4.1 Overall Plant Risk Performance Summary . . . . . 4.1-1

4.1.1 Site Boundary Dose . . . . . 4.1-2

4.1.2 Exclusion Area Boundary Early Fatality Risk . . . . . 4.1-2

4.1.3 Latent Cancer Risk . . . . . 4.1-2

4.2 Defense-in-Depth . . . . . 4.2-1

4.2.1 Plant Capability Summary . . . . . 4.2-1

4.2.2 Programmatic DID Summary . . . . . 4.2-3

4.2.3 Integrated DID Evaluation . . . . . 4.2-4


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## LIST OF TABLES

Table 4.2-1 Mean Risk Margins for Risk-Significant LBEs . . . . . 4.2-5

Table 4.2-2 95th Percentile Risk Margins for Risk-Significant LBEs . . . . . 4.2-6

Table 4.2-3 Evaluation of DL Functions for Risk-Significant DBEs . . . . . 4.2-7

Table 4.2-4 Evaluation of DL Functions for Risk-Significant BDBEs . . . . . 4.2-8

Table 4.2-5 DBA with Single Feature Reliance . . . . . 4.2-9


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# Chapter 4 Integrated Evaluations

## 4.1 Overall Plant Risk Performance Summary

The overall integrated risk performance of the plant as captured within the scope of the probabilistic risk assessment (PRA) developed for the PSAR, has been evaluated against the three cumulative plant risk performance metrics contained in NEI 18-04, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Revision 1 (Reference 4.1-1), Section 3.2.2, Task 7b, for risk to the public from radiation. These metrics are summarized as follows:

- Site Boundary Risk: The total mean frequency of exceeding a site boundary dose of 100 mrem from all licensing basis events (LBEs) and other quantified events (OQEs) should not exceed one per plant-year to comply with the annual cumulative exposure limits in 10 CFR 20.
- Exclusion Area Boundary (EAB) Early Fatality Risk: The average individual risk of early fatality within 1 mile of the EAB from all LBEs and OQEs based on mean estimates of frequencies and consequences shall not exceed $5\times10^{-7}$ per plant-year to ensure the NRC safety goal Quantitative Health Objective for early fatality risk is met.
- Latent Cancer Risk: The average individual risk of latent cancer fatalities within 10 miles of the EAB from all LBEs and OQEs based on mean estimates of frequencies and consequences shall not exceed $2\times10^{-6}$ per plant-year to ensure the NRC safety goal Quantitative Health Objective for latent cancer fatality risk is met.

For the integrated risk evaluation, “all LBEs” include anticipated operational occurrences (AOOs), design basis events (DBEs), and beyond design basis events (BDBEs); as design basis accidents are addressed deterministically, they are not included in the integrated risk evaluation. OQEs are events that are quantified outside of the LBE frequency cutoff. The results are based on mean values, without uncertainties.

The integrated performance assessment uses the same considerations and methodologies from the Chapter 3 assessments regarding site characteristics, locations of individual members of the public, source of dose, analysis method, key modeling assumptions, modes of operation, consideration of other radiological sources on the site, and uncertainty and sensitivity. There is an additional key assumption made that only impacts the cumulative QHO results and does not impact the Chapter 3 assessments. The local boiling source term used for the applicable OQEs assumes that bulk boiling of sodium does not occur. This assumption may underestimate the calculated QHOs; however, the QHO results documented in PSAR Sections 4.1.1, 4.1.2, and 4.1.3 remain conservative as OQEs are not required to be included in the QHO results.

The software and analytical tools that were used in performing the integrated risk evaluation are the same as those described in Chapter 3.

The limitations in the scope and level of detail of the current PRA are discussed in Section 3.1.1.


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### 4.1.1 Site Boundary Dose

Based on the current results of the PRA used for the PSAR, the total mean frequency, without uncertainties, of exceeding a site boundary dose of 100 mrem (1 mSv) from all AOOs, DBEs, BDBEs, and OQEs is $3.12\times10^{-3}$ per plant-year. Therefore, the site boundary dose integrated risk metric is satisfied with margin to the target that the total mean frequency of exceeding a site boundary dose of 100 mrem (1 mSv) from all AOOs, DBEs, BDBEs, and OQEs should not exceed one per plant-year.

### 4.1.2 Exclusion Area Boundary Early Fatality Risk

Based on the current results of the PRA used for the PSAR, the mean frequency, without uncertainties, of the cumulative average individual risk of early fatality within 1 mile of the EAB, summed over all AOOs, DBEs, BDBEs, and OQEs, is 0 per plant-year. Therefore, the EAB early fatality integrated risk metric target of $5\times10^{-7}$ per plant-year is satisfied with margin.

### 4.1.3 Latent Cancer Risk

Based on the current results of the PRA used for the PSAR, the mean frequency, without uncertainties, of the cumulative average individual risk of latent cancer within 10 miles of the EAB, summed over all AOO, DBEs, BDBEs, and OQEs, is $3.86\times10^{-9}$ per plant-year. Therefore, the latent cancer integrated risk metric target of $2\times10^{-6}$ per plant-year is satisfied with margin.

## References

4.1-1 NEI, “Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Technical Report 18-04, Revision 1, August 2019 [ADAMS Accession No. ML19241A472].


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## 4.2 Defense-in-Depth

An overview of methodology and results of the defense-in-depth (DID) adequacy evaluation are provided in Section 4.2.1, Section 4.2.2, and Section 4.2.3. The DID adequacy evaluation follows the guidance in NEI 18-04 (Reference 4.2-1), Chapter 5. There are three key elements of the process for evaluating DID adequacy that are included in Section 4.2.1, Section 4.2.2, and Section 4.2.3 as follows.

Section 4.2.1 presents the plant capability DID, which is used to select functions, structures, systems, or components (SSCs), and their bounding design capabilities to assure safety adequacy. The summary of plant capability DID includes the following:

- Evaluation of licensing basis event (LBE) Margin
- Layers of Defense Evaluation
- Evaluation of Single Feature Reliance
- Evaluation of Prevention-Mitigation Balance

Section 4.2.2 presents the programmatic DID, which involves incorporating special treatment while designing, manufacturing, constructing, operating, maintaining, testing, and inspecting the plant and the associated processes to ensure there is reasonable assurance that the predicted performance can be achieved throughout the lifetime of the plant. The summary of programmatic DID includes the following:

- Evaluation of Significant Uncertainties
- Input for Programs Required for safety-related SSC Performance Monitoring
- Input for Programs Required for non-safety-related with special treatment (NSRST) SSC Performance Monitoring

Section 4.2.3 presents the integrated evaluation of DID, which provides a systematic and comprehensive process for examining the DID adequacy achieved by the combination of plant capability and programmatic elements.

The remaining portions of the DID adequacy evaluation that are not discussed in this Chapter are provided in Chapters 3, 5, 6, 7, and 8.

### 4.2.1 Plant Capability Summary

Plant capability DID includes plant functional capability DID and plant physical capability DID. Plant functional capability DID is introduced through systems and features designed to prevent occurrence of LBEs or mitigate the consequences of LBEs. Plant physical capability DID is introduced through SSC robustness and physical barriers to prevent or limit radionuclide release. The quantitative and qualitative guidelines for plant capability attributes provided in NEI 18-04 Table 5-2 have been evaluated and confirmed. The five layers of defense included in these guidelines are:

- DL1: Prevent off-normal operation and anticipated operational occurrences


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- DL2: Control abnormal operation, detect failures, and prevent design basis events (DBEs)
- DL3: Control DBEs within the analyzed design basis conditions and prevent beyond design basis events (BDBEs)
- DL4: Control severe plant conditions and mitigate consequences of BDBEs
- DL5: Deploy adequate offsite protective actions and prevent adverse impact on public health and safety

Section 4.2.1.1 to Section 4.2.1.4 summarize the review of plant capability DID, which includes a review of the following from NEI 18-04 Section 5:

- Sufficient margin is established between the frequencies and consequences of individual LBEs to the frequency-consequence (F-C) target.
- The frequency of all DBEs and BDBEs is maintained below $1x10^{-2}$ per plant-year and $1x10^{-4}$ per plant-year, respectively.
- The individual risks are maintained below the Quantitative Health Objectives (QHOs).
- Sufficient independence exists between Defense Lines (DLs).
- Assurance against over-reliance on any single feature across DLs.
- Sufficient balance exists between prevention and mitigation in the DLs.

The SSCs that are categorized as NSRST as a result of the DID evaluations are documented in Chapter 5 and their performance criteria and special treatments are documented in Chapters 6 and 7.

#### 4.2.1.1 LBE Margin

Table 4.2-1 and Table 4.2-2 present a summary of the preliminary margins between the frequencies and consequences of the risk-significant LBEs and the F-C target at both the mean and 95th percentile values. Only those LBEs identified as risk-significant using mean values are presented in Table 4.2-1. Table 4.2-2 presents all risk-significant LBEs. See Chapter 3 for the LBE description associated with the LBE identifications. The DID baseline to support plant operation will be established at the operating license (OL) stage.

#### 4.2.1.2 Layers of Defense Evaluation

The qualitative guidance for plant capability DID in Table 5-2 of NEI 18-04 is satisfied for each LBE. Eight risk-significant LBEs are identified. A summary identification of the layers of defense for the risk-significant LBEs is presented in this section. The evaluation summarized in this section demonstrates that there is no over-reliance on a single layer of defense for risk-significant LBEs. No programmatic actions have been added as a result of a significant dependence on any single layer attribute at the construction permit stage.

DL1 prevents off-normal operation and anticipated operational occurrences by implementing programmatic elements and robust design features. The programmatic elements assure that quality, reliability, and conservatism are present in the design, construction, and operation of the plant. Design features reduce the likelihood of initiating events and help assure that SSCs


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performing DL functions can reliably operate. Evaluation of programmatic DID will be provided at the OL stage. Therefore, the preliminary DL1 is limited to design features that promote reliable operation and does not yet include programmatic elements. Section 4.2.2 discusses programmatic DID.

DL2 controls abnormal operation, detects failures, and prevents DBEs. No risk-significant anticipated operational occurrences are identified.

DL3 controls DBEs within the analyzed design basis conditions and prevents risk-significant BDBEs. Table 4.2-3 summarizes the evaluation of DL functions for risk-significant DBEs. For each risk-significant DBE, functions to maintain the frequency of all BDBEs below $1x10^{-4}$ per plant-year are listed. An indication of whether a single design or operational feature is relied upon to meet the quantitative DID objectives is also provided.

DL4 and DL5 control severe plant conditions, mitigate consequences of BDBEs, deploy adequate offsite protective actions, and prevent adverse impact on public health and safety. Table 4.2-4 identifies functions relied upon to maintain individual risks below the QHOs for the risk-significant BDBEs. An indication of whether a single barrier or plant feature is relied upon to meet the quantitative DID objectives is also provided.

#### 4.2.1.3 Single Feature Reliance

An evaluation of the dependence on a single feature across the layers of defense was performed for all risk-significant LBEs and design basis accidents (DBAs). The DL functions were identified for each risk-significant LBE and DBA to determine if a single design or operational feature is relied on to satisfy all five DLs. There is no dependence on a single feature across the layers of defense for the risk-significant LBEs. One DBA was identified to have dependence on a single barrier, and is discussed further in Table 4.2-5. No special treatments were added to provide assurance against over-reliance on any single feature across multiple layers of defense for the risk-significant LBEs at the construction permit stage.

#### 4.2.1.4 Prevention-Mitigation Balance

An evaluation of all risk-significant LBEs and DBAs was performed to ensure both prevention and mitigation DL functions exist for each LBE. This was performed by identifying DLs that are available to prevent or mitigate each risk-significant LBE and DBA. Since DBAs are deterministic in nature, DL1 features are identified that ensure the reliability and availability of the DL3 functions that are utilized to mitigate the event. No plant capability or programmatic actions were added as a result of this evaluation at the construction permit stage.

### 4.2.2 Programmatic DID Summary

The guidelines for programmatic DID adequacy outlined in NEI 18-04, Section 5.8, including the programmatic DID attributes and considerations in NEI 18-04, Table 5-5 and 5-6, are evaluated and included in the design and plant programs development in Chapters 6, 7, and 8.

Any additional special treatments identified from the integrated DID adequacy evaluation described in this chapter will be provided at the OL stage.


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4.2.2.1 Evaluation of Significant Uncertainties

The guidelines for evaluation of significant uncertainties or assumptions in the PRA described in NEI 18-04, Section 5.9.4, are included in the Integrated Decision-Making Process.

The results of the evaluation will be provided at the OL stage.

4.2.2.2 Programs Required for safety-related SSC Performance Monitoring

Chapter 7 identifies the plant-specific programs used to perform monitoring of safety-related SSCs. A summary of additions to or modifications of the programmatic controls provided to account for and manage risk-significant uncertainties as a result of cross-cutting DID evaluations will be provided at the OL stage.

4.2.2.3 Programs Required for NSRST SSC Performance Monitoring

Chapter 7 identifies the plant-specific programs used to perform monitoring of NSRST SSCs. A summary of additions to or modifications of the programmatic controls provided to account for and manage safety-significant uncertainties as a result of cross-cutting DID evaluations will be provided at the OL stage.

4.2.3 Integrated DID Evaluation

The results of the integrated DID evaluation, the establishment of the baseline evaluation of DID, and discussion on how changes to the design are assessed for possible effects on the baseline DID will be provided at the OL stage.

**References**

4.2-1 Nuclear Energy Institute (NEI) 18-04, “Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” Revision 1, August 2019. [ADAMS Accession No. ML19241A472].


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**Table 4.2-1 Mean Risk Margins for Risk-Significant LBEs**

| LBE Category | LBE ID | Mean Frequency/ plant-yr | Mean Dose (rem) | F-C Target Frequency at LBE Dose/ plant-yr[a] | Mean Frequency Margin[b] | F-C Target Dose at LBE Frequency (rem)[c] | Mean Dose Margin[d] |
|---|---|---|---|---|---|---|---|
| BDBE | RFH-ESWR-2 | $3.40 \times 10^{-5}$ | 3.31 | $1.81 \times 10^{-3}$ | 53.1 | 50.0 | 15.1 |
| BDBE | RFH-FDEM-2 | $5.25 \times 10^{-7}$ | 46.0 | $3.88 \times 10^{-5}$ | 73.8 | 727 | 15.8 |
| BDBE | RFH-LMCA-2 | $8.82 \times 10^{-7}$ | 46.0 | $3.88 \times 10^{-5}$ | 44.0 | 522 | 11.3 |

Notes:

[a] Frequency value measured at the LBE mean dose level from the F-C target

[b] Ratio of the frequency in Note [a] to the LBE mean frequency (Mean Frequency Margin)

[c] Dose value measured at the LBE mean frequency from the F-C target

[d] Ratio of the dose in Note [c] to the LBE mean dose (Mean Dose Margin)


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**Table 4.2-2 95th Percentile Risk Margins for Risk-Significant LBEs**

| LBE Name | LBE Category | 95th Perc. Freq. [/ plant-yr] | 95th Perc. Dose [rem] | F-C Target Freq. at LBE Dose [/ plant-yr][a] | Mean Frequency Margin[b] | F-C Target Dose at LBE Freq. [rem][c] | Mean Dose Margin[d] |
|---|---|---|---|---|---|---|---|
| RFH-FDIV-1 | DBE | $8.73 \times 10^{-3}$ | 0.290 | $3.45 \times 10^{-1}$ | 39.5 | 1.10 | 3.79 |
| RFH-FDIV-3 | DBE | $4.32 \times 10^{-3}$ | 0.582 | $1.72 \times 10^{-1}$ | 39.8 | 1.80 | 3.09 |
| RFH-FDSP-1 | DBE | $3.36 \times 10^{-3}$ | 0.505 | $1.98 \times 10^{-1}$ | 59.0 | 2.14 | 4.25 |
| RFH-FDSP-2 | BDBE | $3.72 \times 10^{-4}$ | 1.01 | $9.86 \times 10^{-3}$ | 26.5 | 9.99 | 9.90 |
| RFH-ESWR-2 | BDBE | $1.31 \times 10^{-4}$ | 5.05 | $9.87 \times 10^{-4}$ | 7.53 | 20.7 | 4.10 |
| RFH-FDIV-2 | BDBE | $8.48 \times 10^{-5}$ | 1.62 | $5.03 \times 10^{-3}$ | 59.2 | 27.8 | 17.2 |
| RFH-ESWR-3 | BDBE | $4.94 \times 10^{-6}$ | 21.0 | $1.29 \times 10^{-4}$ | 26.1 | 173 | 8.22 |
| RFH-LMCA-2 | BDBE | $3.48 \times 10^{-6}$ | 70.0 | $2.02 \times 10^{-5}$ | 5.78 | 216 | 3.08 |
| RFH-FDBL-2 | BDBE | $2.33 \times 10^{-6}$ | 21.0 | $1.29 \times 10^{-4}$ | 55.3 | 280 | 13.3 |
| RFH-FDEM-2 | BDBE | $1.99 \times 10^{-6}$ | 70.0 | $2.02 \times 10^{-5}$ | 10.1 | 310 | 4.42 |
| RFH-FDPI-1 | BDBE | $1.93 \times 10^{-6}$ | 21.0 | $1.29 \times 10^{-4}$ | 66.6 | 316 | 15.0 |

Notes:

[a] Frequency value measured at the LBE mean dose level from the F-C target

[b] Ratio of the frequency in Note [a] to the LBE mean frequency (Mean Frequency Margin)

[c] Dose value measured at the LBE mean frequency from the F-C target

[d] Ratio of the dose in Note [c] to the LBE mean dose (Mean Dose Margin)


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**Table 4.2-3 Evaluation of DL Functions for Risk-Significant DBEs**

| DBE ID | Maintain Frequency of all BDBEs < $1x10^{-4}$/plant-year | | Any single design or operation feature relied upon to meet quantitative objectives for DBE? |
|---|---|---|---|
| | DL Function | Description | |
| RFH-FDIV-1 | DL3-RR1 | Primary System Boundary including RES Barrier | No. There is no single design or operational feature relied upon for this LBE family. |
| | DL3-RR3a | Ex-Vessel Handling Machine to Head Barrier Function | |
| RFH-FDIV-3 | DL3-RR1 | Primary System Boundary including RES Barrier | No. There is no single design or operational feature relied upon for this LBE family. |
| | DL3-RR3a | Ex-Vessel Handling Machine to Head Barrier Function | |
| RFH-FDSP-1 | None. | The only difference between the DBE, RFH-FDSP-1, and BDBE, RFH-FDSP-2, events is that the BDBE event considers failure of an additional fuel assembly due to the fuel handling event. The DBE and BDBE events rely on the same set of DL functions but have different consequences. | No. There is no single design or operational feature relied upon for this LBE family. |


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**Table 4.2-4 Evaluation of DL Functions for Risk-Significant BDBEs**

| BDBE ID | Functions to Maintain Individual Risks < QHOs | | Any single design or operational feature relied upon? |
|---|---|---|---|
| | DL Function | Description | |
| RFH-ESWR-2 | DL4-RR7 | Fuel Handling Building Barrier | No. There is no single design or operational feature relied upon for this LBE family. |
| RFH-ESWR-3 | None. | All available functions are failed for this LBE. | No. There is no single design or operational feature relied upon for this LBE family. |
| RFH-FDBL-2 | None. | All available functions are failed for this LBE. | No. There is no single design or operational feature relied upon for this LBE family. |
| RFH-FDEM-2 | None. | All available functions are failed for this LBE. | No. There is no single design or operational feature relied upon for this LBE family. |
| RFH-FDIV-2 | None. | All available functions are failed for this LBE. | No. There is no single design or operational feature relied upon for this LBE family. |
| RFH-FDPI-1 | None. | All available functions are failed for this LBE. | No. There is no single design or operational feature relied upon for this LBE family. |
| RFH-FDSP-2 | None. | All available functions are failed for this LBE. | No. There is no single design or operational feature relied upon for this LBE family. |
| RFH-LMCA-2 | None. | All available functions are failed for this LBE. | No. There is no single design or operational feature relied upon for this LBE family. |


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**Table 4.2-5 DBA with Single Feature Reliance**

| DBA ID | Any single feature/barrier relied upon? | Comment |
|---|---|---|
| RRS-ISPL-CN | Yes | The PRA evaluation is based on the worst-case scenario which models 40 years of accumulated tritium released with no containment credited.<br>The Intermediate Sodium Processing System is located above grade in the Reactor Auxiliary Building. The total accumulated tritium cannot exceed the DBA dose even in the worst case scenario. |


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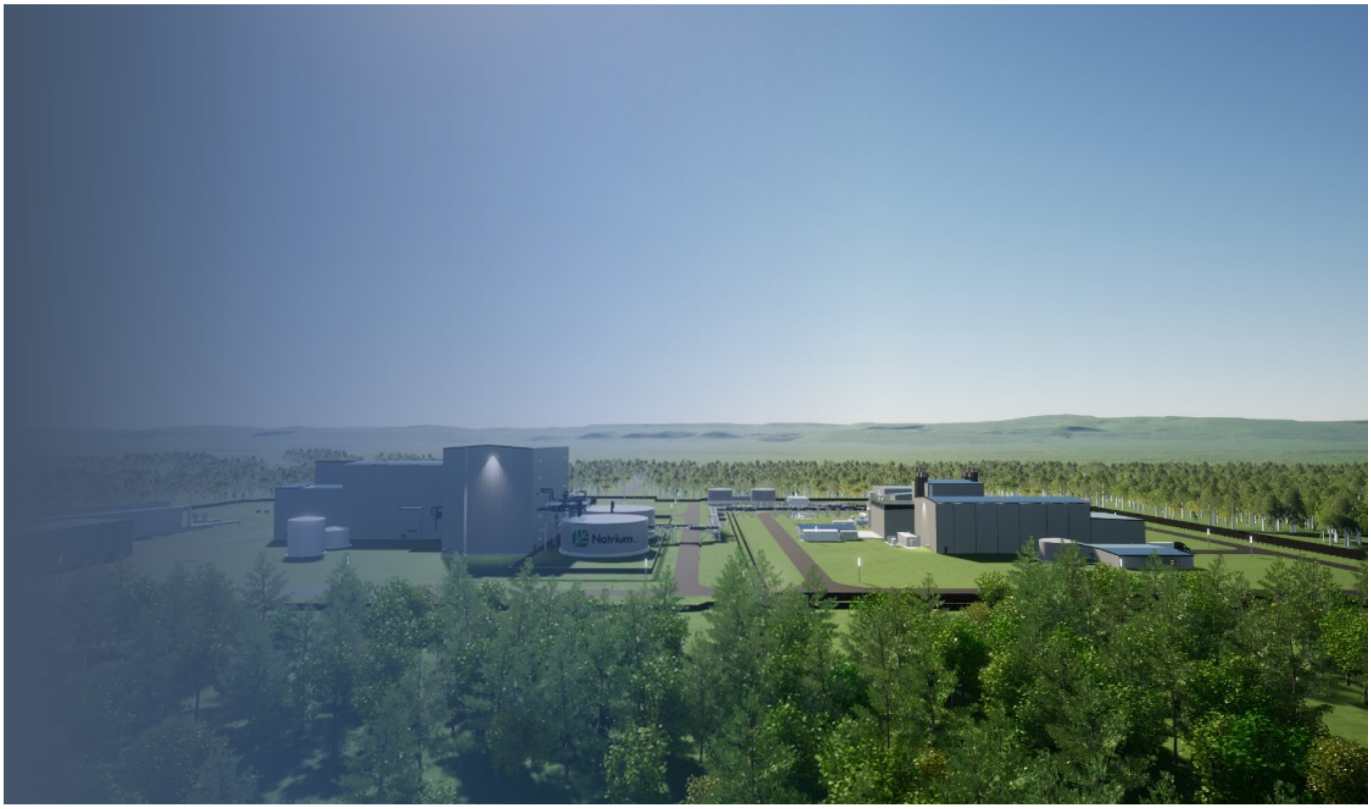

# Kemmerer Power Station Unit 1
# PSAR, Chapter 5

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## TABLE OF CONTENTS

**Chapter 5 Safety Functions, Design Criteria, and SSC Safety Classification . . . . . 5.1-1**

5.1 Safety Classification of SSCs . . . . . 5.1-1

5.2 Safety-Significant PRA Safety Functions . . . . . 5.2-1

5.2.1 SR Functions . . . . . 5.2-1

5.2.2 NSRST Functions . . . . . 5.2-1

5.2.3 NST Functions . . . . . 5.2-1

5.3 Principal Design Criteria . . . . . 5.3-1

5.3.1 Overall Requirements . . . . . 5.3-1

5.3.2 Multiple Barriers . . . . . 5.3-4

5.3.3 Reactivity Control . . . . . 5.3-10

5.3.4 Fluid and Heat Transport Systems . . . . . 5.3-16

5.3.5 Fuel and Reactivity Control . . . . . 5.3-21

5.3.6 Additional PDC . . . . . 5.3-24

5.4 Safety-Related Structures, Systems, and Components . . . . . 5.4-1

5.5 Non-Safety-Related with Special Treatments Structures, Systems, and Components . . . . . 5.5-1


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## LIST OF TABLES

Table 5.2-1 SR Functions Supporting Control of Heat Generation . . . . . 5.2-2

Table 5.2-2 SR Functions Supporting Control of Heat Removal . . . . . 5.2-4

Table 5.2-3 SR Functions Supporting Retaining Radionuclides . . . . . 5.2-8

Table 5.2-4 NSRST Functions. . . . . 5.2-12


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# Chapter 5 Safety Functions, Design Criteria, and SSC Safety Classification

## 5.1 Safety Classification of SSCs

Safety classification of structures, systems, or components (SSCs) is performed in accordance with NEI 18-04 (Reference 5.1-1), as endorsed by Regulatory Guide 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors,” Revision 0. Probabilistic risk assessment (PRA) Safety Functions are initially assigned to systems. PRA Safety Functions are determined and a structured approach is used to identify SSCs that perform the PRA Safety Functions. The safety significance of a PRA Safety Function determines the safety classification of the SSCs credited to perform the function as defined in NEI 18-04.

## References

5.1-1 NEI Technical Report 18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” [ADAMS Accession No. ML19241A472], August 2019.


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# 5.2 Safety-Significant PRA Safety Functions

## 5.2.1 SR Functions

Safety-related (SR) functions are the probabilistic risk assessment (PRA) safety functions (PSF) credited to maintain the consequence of one or more design basis events (DBE) or the frequency of one or more high-consequence beyond design basis events (BDBE) inside the F-C Target, and to conservatively ensure that 10 CFR 50.34 dose requirements can be met. The term SR function is synonymous with the term required safety function defined in NEI 18-04 (Reference 5.2-1).

Table 5.2-1, Table 5.2-2, and Table 5.2-3 list the PSF identification numbers, the SR function descriptions, the anticipated operational occurrences (AOO), DBEs, and BDBEs the SR functions are credited in, the structures, systems, or components (SSC) credited to perform the SR functions, and the safety-related design criteria (SRDC) for the SR SSCs. SR functions with “None” in the “Associated LBE” column are conservatively identified for classification as an SR function. Safety analysis evaluations will be performed to support all SR function classifications at the operating license stage.

See Section 6.1.2 for a description of SRDC.

## 5.2.2 NSRST Functions

Non-safety-related with special treatment (NSRST) functions are the non-SR PSFs that are risk-significant or are required for defense-in-depth (DID) adequacy.

Table 5.2-4 lists the PSF identification numbers, the NSRST function descriptions, the SSC safety significance criteria satisfied (risk-significant or DID), the AOOs, DBEs, and BDBEs the NSRST functions are credited in, and the SSCs credited to perform the NSRST functions. SSCs are classified as risk-significant if the SSC function is necessary to keep any LBEs inside the F-C Target, or if the total frequency of licensing basis events (LBE) with the SSCs failed is within one percent of any of the three cumulative risk targets. Risk-significant PSFs with “None” in the “Associated LBE” column are conservatively identified for classification as a risk-significant NSRST function. Safety analysis evaluations will be performed to support all risk-significant SSC designations at the operating license stage.

## 5.2.3 NST Functions

Non-safety-related with no special treatment (NST) functions are PSFs that are not SR functions or NSRST functions. NST functions are not safety-significant and are not listed in this chapter.

**References**

5.2-1 Nuclear Energy Institute, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” NEI Technical Report 18-04, Revision 1, August 2019, [ADAMS Accession No. ML19241A472]


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**Table 5.2-1 SR Functions Supporting Control of Heat Generation**
**(Sheet 1 of 2)**

| PSF ID | SR Function Description | Associated LBE(s) | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-RC1 | Scram - Gravity Driven Absorber Insertion by Latch Release | DHP-L1PP-BL | AOO | Control Rod Drive System (CRD) | Upon receipt of a scram signal from RPS, the Control Rod Drive System shall release the control rod assemblies in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | DHP-LOOP-BL | AOO | | |
| | | DHS-ISTL-BL | AOO | | |
| | | OTH-LMAC-BL | AOO | Reactor Core System (RCC) | Upon CRD rod release, control rod assemblies shall insert into the core to a depth and within a time limit to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | RPD-CW1ACS-BL | AOO | | |
| | | RPD-SS-BL | AOO | | |
| | | DHP-L1PP-2 | DBE | Reactor Enclosure System (RES) | Upon CRD rod release, RES shall provide structural support and position control preventing binding of CRDL and ensure control rod release is uninhibited to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | DHP-LOOP-1 | DBE | | |
| | | DHS-ISTL-1 | DBE | | |
| | | DHS-ISTL-2 | DBE | | |
| | | DHS-RNBK-1 | DBE | | |
| | | DHS-RNBK-3 | DBE | Reactor Protection System (RPS) | The RPS shall receive monitored signals and generate a scram signal upon exceeding a scram setpoint in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | IPI-IHEL-BL | DBE | | |
| | | DHP-L1PP-1 | BDBE | | |
| | | DHP-L1PP-4 | BDBE | Nuclear Instrumentation System (XIS) | XIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | DHP-LAPP-BL | BDBE | | |
| | | DHP-LOOP-2 | BDBE | | |
| | | DHP-LOOP-4 | BDBE | Reactor Instrumentation System (RIS) | RIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | DHS-ISTL-4 | BDBE | | |
| | | DHS-RNBK-2 | BDBE | | |
| | | IPI-IHEL-1 | BDBE | | |
| | | IPI-IHEL-2 | BDBE | | |
| | | OTH-LMAC-1 | BDBE | | |
| | | OTH-LMAC-2 | BDBE | | |
| | | OTH-LMAC-4 | BDBE | | |
| | | RPD-CW1ACS-1 | BDBE | | |
| | | RPD-CW1ACS-2 | BDBE | | |
| | | RPD-CW1ACS-4 | BDBE | | |
| | | RPD-SS-1 | BDBE | | |
| | | RPD-SS-2 | BDBE | | |
| | | RPD-SS-4 | BDBE | | |


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**Table 5.2-1 SR Functions Supporting Control of Heat Generation**
**(Sheet 2 of 2)**

| PSF ID | SR Function Description | Associated LBE(s) | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-RC2 | Reactor Scram on Loss of Power | None | N/A | Control Rod Drive System (CRD) | Upon receipt of a scram signal from RPS, the Control Rod Drive System shall release the control rod assemblies in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Protection System (RPS) | Upon power loss, the RPS shall fail safe by generating a scram signal in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |

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**Table 5.2-2 SR Functions Supporting Control of Heat Removal**
**(Sheet 1 of 4)**

| PSF ID | SR Function Description | Associated LBE | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-HR1 | PSP Coastdown | DHP-L1PP-2<br>DHS-ISTL-1<br>DHS-ISTL-2<br>DHS-RNBK-3<br>DHP-L1PP-1<br>DHS-RNBK-2<br>IPI-IHEL-1<br>IPI-IHEL-2<br>OTH-LMAC-1<br>OTH-LMAC-2<br>RPD-CW1ACS-1<br>RPD-CW1ACS-2<br>RPD-SS-1<br>RPD-SS-2 | DBE<br>DBE<br>DBE<br>DBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE | Primary Heat Transport System (PHT) | The Primary Sodium Pump (PSP) shall coast down upon a pump trip to support transition to natural circulation, without flow reversal within the fuel channels, to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-HR2 | PSP Trip on High High Primary Sodium Temperature | DHP-L1PP-2<br>DHS-ISTL-1<br>DHS-ISTL-2<br>DHS-RNBK-3<br>DHP-L1PP-1<br>DHS-RNBK-2<br>IPI-IHEL-1<br>IPI-IHEL-2<br>OTH-LMAC-1<br>OTH-LMAC-2<br>RPD-CW1ACS-1<br>RPD-CW1ACS-2<br>RPD-SS-1<br>RPD-SS-2 | DBE<br>DBE<br>DBE<br>DBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE | Primary Heat Transport System (PHT) | Upon receiving a trip signal from the RPS, the PSPs shall trip to eliminate heat generation to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Protection System (RPS) | The RPS shall receive monitored signals and generate a PSP trip signal upon exceeding a high temperature limit with a low flux signal in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Nuclear Instrumentation System (XIS)<br>Reactor Instrumentation System (RIS) | XIS and RIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |


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**Table 5.2-2 SR Functions Supporting Control of Heat Removal**
**(Sheet 2 of 4)**

| PSF ID | SR Function Description | Associated LBE | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-HR3 | ISP Trip on High High Primary Sodium Temperature | DHP-L1PP-2<br>DHS-ISTL-1<br>DHS-ISTL-2<br>DHS-RNBK-3<br>DHP-L1PP-1<br>DHS-RNBK-2<br>OTH-LMAC-1<br>OTH-LMAC-2<br>RPD-CW1ACS-1<br>RPD-CW1ACS-2<br>RPD-SS-1<br>RPD-SS-2 | DBE<br>DBE<br>DBE<br>DBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE | Intermediate Heat Transport System (IHT) | Upon receiving a trip signal from the RPS, the Intermediate Sodium Pumps (ISP) shall trip to eliminate heat generation to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Protection System (RPS) | The RPS shall receive monitored signals and generate an ISP trip signal upon exceeding a high temperature limit with a low flux signal in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Nuclear Instrumentation System (XIS)<br>Reactor Instrumentation System (RIS) | XIS and RIS shall send monitored signals to the RPS in time to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-HR4 | Inherent - RAC Operation | DHP-L1PP-2<br>DHS-ISTL-2<br>DHS-RNBK-3<br>DHP-LOOP-2<br>IPI-IHEL-2<br>OTH-LMAC-2<br>RPD-CW1ACS-2<br>RPD-SS-2<br>SUD-IACA-2 | DBE<br>DBE<br>DBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE | Reactor Air Cooling System (RAC)<br>Reactor Enclosure System (RES)<br>Reactor Building (RXB) | RAC shall continuously transfer heat to the atmosphere via natural circulation at a rate during accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-HR5 | Natural Circulation of Sodium in Primary System | DHP-L1PP-2<br>DHS-ISTL-2<br>DHS-RNBK-3<br>DHP-LOOP-2<br>IPI-IHEL-2<br>OTH-LMAC-2<br>RPD-CW1ACS-2<br>RPD-SS-2<br>SUD-IACA-2 | DBE<br>DBE<br>DBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE<br>BDBE | Primary Heat Transport System (PHT)<br>Reactor Core System (RCC)<br>Reactor Enclosure System (RES) | The primary flow circuit elevations and geometries shall promote natural circulation to remove heat from the core at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |


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**Table 5.2-2 SR Functions Supporting Control of Heat Removal**
**(Sheet 3 of 4)**

| PSF ID | SR Function Description | Associated LBE | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-HR6 | Passive Heat Removal in EVHM | RFH-LMCA-BL | DBE | Ex-Vessel Fuel Handling System (FHE) | The Ex-Vessel Handling Machine (EVHM) shall passively remove decay heat from spent fuel assemblies at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Core System (RCC) | Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-HR7 | Passive Heat Removal in EVST | RFH-LTCA-BL | AOO | Ex-Vessel Fuel Handling System (FHE) | The Ex-Vessel Storage Tank (EVST) shall passively remove decay heat from spent fuel assemblies at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Core System (RCC) | Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-HR8 | Passive Heat Removal in BLTC | RFH-LBCA-BL | DBE | Ex-Vessel Fuel Handling System (FHE) | The Bottom Loading Transfer Cask (BLTC) shall passively remove decay heat from spent fuel at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Core System (RCC) | Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-HR9 | Passive Heat Removal in PRC | None | N/A | Ex-Vessel Fuel Handling System (FHE) | The Pin Removal Cell (PRC) shall passively remove decay heat from spent fuel at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Core System (RCC) | Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-HR10 | Passive Heat Removal in FHP | RFH-LSPC-BL<br>RFH-LSPC-1 | AOO<br>BDBE | Water Fuel Pool Handling System (FHP) | The spent fuel pool shall passively remove decay heat from spent fuel at a rate to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Core System (RCC) | Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |


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**Table 5.2-2 SR Functions Supporting Control of Heat Removal**
**(Sheet 4 of 4)**

| PSF ID | SR Function Description | Associated LBE | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-HR11 | SPS Pump Trip on Low Low Primary Sodium Level | RRS-SPLA-BL<br>RRS-SPLX-BL | DBE<br>DBE | Sodium Processing System (SPS) | Upon receiving a trip signal from the RPS, the SPS pumps shall trip in time to maintain reactor vessel primary sodium level high enough to establish reasonable assurance that natural circulation in the primary sodium flow circuit can develop. |
| | | | | Reactor Instrumentation System (RIS) | RIS shall send monitored signals to the RPS in time to maintain reactor vessel primary sodium level necessary to establish reasonable assurance that natural circulation in the primary sodium flow circuit can develop. |
| | | | | Reactor Protection System (RPS) | The RPS shall monitor received signals, and if primary sodium level exceeds the setpoint after a scram shutdown, the RPS shall trip the SPS pumps in time to maintain reactor vessel primary sodium level high enough to establish reasonable assurance that natural circulation in the primary sodium flow circuit can develop. |
| DL3-HR12 | ISP Pump Trip on High High Primary Sodium Level | IPI-IHEL-BL<br>IPI-IHEL-1<br>IPI-IHEL-2 | DBE<br>BDBE<br>BDBE | Intermediate Heat Transport System (IHT) | Upon receiving a trip signal from the RPS, the ISPs shall trip in time to prevent over-pressurization and failure of the primary boundary, to establish reasonable assurance primary inventory is maintained, and establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Instrumentation System (RIS)<br>Nuclear Instrumentation System (XIS) | XIS and RIS shall send monitored signals to the RPS in time for the plant to respond to prevent over-pressurization and failure of the primary boundary, resulting in a safe shutdown condition. The functions will provide reasonable assurance primary inventory is maintained and any associated radiological dose calculation results are maintained under the 10 CFR 50.34 dose criteria. |
| | | | | Reactor Protection System (RPS) | The RPS shall monitor received parameters, and if primary level exceeds the setpoint after a scram shutdown, the RPS shall trip the ISPs. This function shall occur in time to prevent over-pressurization and failure of the primary boundary, which provides reasonable assurance the primary inventory is maintained, and radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-HR13 | Passive Heat Removal in the Failed Fuel Canister | None | N/A | Water Fuel Pool Handling System (FHP) | The Failed Fuel Canister shall passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that the radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Reactor Core System (RCC) | Fuel assemblies shall be designed to passively remove decay heat at a rate through natural circulation and conduction to establish reasonable assurance that radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |


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**Table 5.2-3 SR Functions Supporting Retaining Radionuclides**
**(Sheet 1 of 4)**

| PSF-ID | SR Function Description | Associated LBE | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-RR1 | Primary Coolant Boundary including RES Barrier | LFF-SAO-BL<br>RFH-FDIV-1<br>RFH-FDIV-3<br>DHP-LOOP-3<br>DHP-LOOP-4<br>LFF-SAO-1 | DBE<br>DBE<br>DBE<br>BDBE<br>BDBE<br>BDBE | Reactor Enclosure System (RES) | The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Primary Heat Transport System (PHT) | The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Sodium Processing System (SPS) | The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Sodium Cover Gas System (SCG) | The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Control Rod Drive System (CRD) | The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | In-Vessel Fuel Handling System (FHI) | The primary system boundary shall have low leakage in postulated accident conditions to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |


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**Table 5.2-3 SR Functions Supporting Retaining Radionuclides**
**(Sheet 2 of 4)**

| PSF-ID | SR Function Description | Associated LBE | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-RR2 | Cladding barrier | DHP-L1PP-BL | AOO | Reactor Core System (RCC) | During accident conditions, the fuel cladding shall retain radionuclide fission products to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | DHP-LOOP-BL | AOO | | |
| | | DHS-ISTL-BL | AOO | | |
| | | OTH-LMAC-BL | AOO | | |
| | | RFH-LSPC-BL | AOO | | |
| | | RFH-LTCA-BL | AOO | | |
| | | RPD-CW1ACS-BL | AOO | | |
| | | RPD-SS-BL | AOO | | |
| | | SUD-IACA-BL | AOO | | |
| | | SUD-LOOP-BL | AOO | | |
| | | DHP-L1PP-2 | DBE | | |
| | | DHP-LOOP-1 | DBE | | |
| | | DHS-ISTL-1 | DBE | | |
| | | DHS-ISTL-2 | DBE | | |
| | | DHS-RNBK-1 | DBE | | |
| | | DHS-RNBK-3 | DBE | | |
| | | IPI-IHEL-BL | DBE | | |
| | | RFH-ESWR-BL | DBE | | |
| | | RFH-FDIV-BL | DBE | | |
| | | RFH-LBCA-BL | DBE | | |
| | | RFH-LMCA-BL | DBE | | |
| | | RFH-LTCA-1 | DBE | | |
| | | SUD-IACA-1 | DBE | | |
| | | SUD-IHEL-BL | DBE | | |
| | | SUD-LOOP-1 | DBE | | |
| | | DHP-L1PP-1 | BDBE | | |
| | | DHP-L1PP-3 | BDBE | | |
| | | DHP-L1PP-4 | BDBE | | |
| | | DHP-LAPP-BL | BDBE | | |
| | | DHP-LOOP-2 | BDBE | | |
| | | DHS-ISTL-3 | BDBE | | |
| | | DHS-ISTL-4 | BDBE | | |
| | | DHS-RNBK-2 | BDBE | | |


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**Table 5.2-3 SR Functions Supporting Retaining Radionuclides**
**(Sheet 3 of 4)**

| PSF-ID | SR Function Description | Associated LBE | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-RR2 (continued) | Cladding barrier (continued) | IPI-IHEL-1 | BDBE | | |
| | | IPI-IHEL-2 | BDBE | | |
| | | OTH-LMAC-1 | BDBE | | |
| | | OTH-LMAC-2 | BDBE | | |
| | | OTH-LMAC-3 | BDBE | | |
| | | OTH-LMAC-4 | BDBE | | |
| | | RFH-FDET-BL | BDBE | | |
| | | RFH-LSPC-1 | BDBE | | |
| | | RPD-CW1ACS-1 | BDBE | | |
| | | RPD-CW1ACS-2 | BDBE | | |
| | | RPD-CW1ACS-3 | BDBE | | |
| | | RPD-CW1ACS-4 | BDBE | | |
| | | RPD-SS-1 | BDBE | | |
| | | RPD-SS-2 | BDBE | | |
| | | RPD-SS-3 | BDBE | | |
| | | RPD-SS-4 | BDBE | | |
| | | SUD-IACA-2 | BDBE | | |
| | | SUD-IHEL-1 | BDBE | | |
| DL3-RR3 | EVHM Cask barrier | RFH-FDIV-1<br>RFH-FDIV-3<br>RFH-FDEM-1<br>RFH-LMCA-1 | DBE<br>DBE<br>BDBE<br>BDBE | Ex-Vessel Fuel Handling System (FHE) | During postulated accidents, the EVHM, in a standalone state, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | | During postulated accidents, when the EVHM is joined with the reactor head, the combined barrier shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | | During postulated accidents, when the EVHM is joined with the EVST, the combined barrier shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |


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**Table 5.2-3 SR Functions Supporting Retaining Radionuclides**
**(Sheet 4 of 4)**

| PSF-ID | SR Function Description | Associated LBE | Type | SR SSC(s) | SRDC |
|---|---|---|---|---|---|
| DL3-RR4 | EVST Barrier | RFH-FDET-BL<br>RFH-FDET-1 | BDBE<br>BDBE | Ex-Vessel Fuel Handling System (FHE)<br>Sodium Processing System (SPS) | During postulated accidents the EVST, in a standalone state, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | | During postulated accidents, when the EVST is joined with the BLTC, the combined barrier shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | | The EVST Auxiliary Systems, up to and including the isolation valves, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-RR5 | BLTC barrier | RFH-ESWR-1<br>RFH-FDBL-1<br>RFH-FDPI-BL | DBE<br>BDBE<br>BDBE | Ex-Vessel Fuel Handling System (FHE) | During postulated accidents the BLTC, in a standalone state, shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | | During postulated accidents, when the BLTC is joined with the Pool Immersion Cell (PIC), the combined barrier shall contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| | | | | Ex-Vessel Fuel Handling System (FHE) | During postulated accidents, during all fuel movements in and out of the BLTC, |
| | | | | Ex-Vessel Fuel Handling System (FHE)<br>Water Pool Fuel Handling System (FHP) | During postulated accidents, during all fuel movements in and out of the BLTC, the installed equipment shall temporarily contain radionuclides from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-RR6 | PRC Cell barrier | RFH-OERC-BL<br>RFH-FDRC-1 | AOO<br>BDBE | Ex-Vessel Fuel Handling System (FHE) | The PRC shall contain radionuclide releases from the fuel to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-RR7 | RES Pressure Relief | None | N/A | Sodium Cover Gas System (SCG)<br>Reactor Enclosure System (RES) | Prior to reaching the RES and supporting system functional containment design pressure, SCG shall actuate the primary pressure relief valve to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |
| DL3-RR8 | Failed Fuel Canister Barrier | None | N/A | Water Pool Fuel Handling System (FHP) | During accident conditions, the Failed Fuel Canister containing failed fuel shall retain radionuclide fission products to establish reasonable assurance radionuclide release results in calculated radiological dose under the 10 CFR 50.34 dose criteria at a safe shutdown condition. |


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**Table 5.2-4 NSRST Functions**
**(Sheet 1 of 4)**

| PSF ID | PSF Description | NSRST SSC Safety Significance Criteria (Risk-Significant or DID) | Associated LBE ID | | SSC(s) |
|---|---|---|---|---|---|
| DL4-RC1 | Reactor Scram - Manual | Required for DID Adequacy | None | N/A | Reactor Protection System (RPS) |
| DL4-RC3 | CRD Driveline Scram Follow | Risk-Significant Function | DHP-L1PP-3 | BDBE | Control Rod Drive System (CRD)<br>Nuclear Island (NI) DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>Reactor Core System (RCC)<br>Reactor Protection System (RPS) |
| | | | DHP-LOOP-3 | BDBE | |
| | | | DHS-ISTL-3 | BDBE | |
| | | | OTH-LMAC-3 | BDBE | |
| | | | RPD-CW1ACS-3 | BDBE | |
| | | | RPD-SS-3 | BDBE | |
| DL4-RC4 | Automatic Seismic Trip | Required for DID Adequacy | None | N/A | Anticipatory Automatic Seismic Trip System (AST)<br>NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>Reactor Protection System (RPS) |
| DL4-RC6 | Alternative Shunt Trip | Risk-Significant Function | DHP-L1PP-4 | BDBE | See Note 1 |
| | | | DHP-LOOP-4 | BDBE | |
| | | | DHS-ISTL-4 | BDBE | |
| | | | OTH-LMAC-4 | BDBE | |
| | | | RPD-CW1ACS-4 | BDBE | |
| | | | RPD-SS-4 | BDBE | |
| DL4-RC6a | Alternative Shunt Trip on High Core Exit Temperature | Risk-Significant Function | RPD-CW1ACS-4 | BDBE | See Note 1 |
| DL4-RC6b | Alternative Shunt Trip on Low Primary Sodium Pump Outlet Pressure | Risk-Significant Function | DHP-L1PP-4 | BDBE | See Note 1 |
| | | | DHP-LOOP-4 | BDBE | |
| | | | OTH-LMAC-4 | BDBE | |
| DL4-RC6c | Alternative Shunt Trip on High High IHT level | Risk-Significant Function | None | N/A | See Note 1 |
| DL4-RC6d | Alternative Shunt Trip on Low Low IHT level | Risk-Significant Function | None | N/A | See Note 1 |
| DL2-HR2 | ISP Trip on Low IHT Level | Required for DID Adequacy | None | N/A | Coolant Temperature Monitoring and Control System (CTC)<br>Intermediate Heat Transport System (IHT)<br>NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP) |


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**Table 5.2-4 NSRST Functions**
**(Sheet 2 of 4)**

| PSF ID | PSF Description | NSRST SSC Safety Significance Criteria (Risk-Significant or DID) | Associated LBE ID | | SSC(s) |
|---|---|---|---|---|---|
| DL4-HR1 | IAC Passive Mode Operation | Required for DID Adequacy | DHP-LOOP-1 | DBE | Intermediate Air Cooling System (IAC)<br>Intermediate Heat Transport System (IHT)<br>Primary Heat Transport System (PHT)<br>Sodium Cover Gas System (SCG)<br>Sodium Processing System (SPS) |
| | | | DHS-ISTL-1 | DBE | |
| | | | SUD-IACA-1 | DBE | |
| | | | SUD-LOOP-1 | DBE | |
| | | | DHP-L1PP-1 | BDBE | |
| | | | DHS-RNBK-2 | BDBE | |
| | | | IPI-IHEL-1 | BDBE | |
| | | | OTH-LMAC-1 | BDBE | |
| | | | RPD-CW1ACS-1 | BDBE | |
| | | | RPD-SS-1 | BDBE | |
| | | | SUD-IHEL-1 | BDBE | |
| DL4-HR2 | PSP Trip Automatic Backup | Required for DID Adequacy | None | N/A | Coolant Temperature Monitoring and Control System (CTC)<br>NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>Primary Heat Transport System (PHT)<br>Reactor Instrumentation System (RIS) |
| DL4-HR3 | ISP Trip Automatic Backup | Required for DID Adequacy | None | N/A | Coolant Temperature Monitoring and Control System (CTC)<br>NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>Intermediate Heat Transport System (IHT)<br>Reactor Instrumentation System (RIS) |
| DL4-HR4 | FHP Emergency Makeup | Risk-Significant Function | RFH-LSPC-1 | BDBE | Water Pool Fuel Handling System (FHP) |
| DL4-HR6 | Manual PSP Trip | Required for DID Adequacy | None | N/A | NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>Primary Heat Transport System (PHT)<br>Reactor Protection System (RPS) |
| DL4-HR7 | Manual ISP Trip | Required for DID Adequacy | None | N/A | NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>Intermediate Heat Transport System (IHT)<br>Reactor Protection System (RPS) |
| DL2-RR2 | PIC Radionuclide Boundary | Required for DID Adequacy | RFH-ESWR-1 | DBE | Water Pool Fuel Handling System (FHP) |
| DL2-RR4 | Gaseous Radwaste Barrier | Required for DID Adequacy | None | N/A | Gaseous Radwaste Processing System (RWG) |
| DL2-RR7 | Primary SPS Barrier | Required for DID Adequacy | None | N/A | Sodium Processing System (SPS) |
| DL2-RR8 | Intermediate Cold Trap SPS Barrier | Required for DID Adequacy | None | N/A | Sodium Processing System (SPS) |
| DL2-RR10 | Primary SCG Barrier | Required for DID Adequacy | None | N/A | Sodium Cover Gas System (SCG) |


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**Table 5.2-4 NSRST Functions**
**(Sheet 3 of 4)**

| PSF ID | PSF Description | NSRST SSC Safety Significance Criteria (Risk-Significant or DID) | Associated LBE ID | | SSC(s) |
|---|---|---|---|---|---|
| DL4-RR1 | Ex-RES Functional Containment Barrier | Required for DID Adequacy | RRS-SPLX-BL | DBE | NI HVAC System (NHV)<br>Reactor Building (RXB) |
| | | | RRS-CGR-BL | DBE | |
| | | | RRS-CGR-1 | DBE | |
| | | | LFF-SAO-BL | DBE | |
| | | | LFF-SAO-2 | BDBE | |
| DL4-RR1a | HAA barrier | Required for DID Adequacy | LFF-SAO-BL | DBE | NI HVAC System (NHV)<br>Reactor Building (RXB)<br>Reactor Enclosure System (RES) |
| | | | RRS-CGR-1 | DBE | |
| | | | RRS-CGR-BL | DBE | |
| | | | LFF-SAO-2 | BDBE | |
| DL4-RR1b | Guard Vessel Leak Prevention Function | Required for DID Adequacy | None | N/A | Reactor Enclosure System (RES) |
| DL4-RR1c | HAA HVAC Operations Following Postulated Release | Required for DID Adequacy | RRS-CGR-BL | DBE | NI HVAC System (NHV)<br>Radiation Monitoring System (RMS)<br>Utility Monitoring and Control System (UMC) |
| | | | RRS-CGR-1 | DBE | |
| DL4-RR3 | SPS Cells barrier | Required for DID Adequacy | RRS-SPLA-BL | DBE | NI HVAC System (NHV)<br>Reactor Auxiliary Building (RAB)<br>Sodium Processing System (SPS) |
| DL4-RR3a | SPS Supply Valve Isolation on Leak Detection | Required for DID Adequacy | None | N/A | Auxiliary Monitoring and Control System (AMC)<br>NI Fire Protection System (NFP)<br>Sodium Processing System (SPS) |
| DL4-RR3b | SPS Pumps Trip on Leak Detection | Required for DID Adequacy | None | N/A | Auxiliary Monitoring and Control System (AMC)<br>NI Fire Protection System (NFP)<br>Sodium Processing System (SPS)<br>NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP) |
| DL4-RR3c | SPS Cells Barrier Isolation on Leak Detection | Required for DID Adequacy | None | N/A | NI HVAC System (NHV)<br>NI Fire Protection System (NFP)<br>Sodium Processing System (SPS)<br>Utility Monitoring and Control System (UMC) |
| DL4-RR4 | SCG Cells barriers | Required for DID Adequacy | None | N/A | NI HVAC System (NHV)<br>Sodium Cover Gas System (SCG) |
| DL4-RR4a | Automatically Close SCG Isolation Valves on Leak Detection | Required for DID Adequacy | None | N/A | Auxiliary Monitoring and Control System (AMC)<br>Sodium Cover Gas System (SCG)<br>Radiation Monitoring System (RMS) |


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**Table 5.2-4 NSRST Functions**
**(Sheet 4 of 4)**

| PSF ID | PSF Description | NSRST SSC Safety Significance Criteria (Risk-Significant or DID) | Associated LBE ID | | SSC(s) |
|---|---|---|---|---|---|
| DL4-RR4b | SCG Cells Barrier Isolation on Leak Detection | Required for DID Adequacy | None | N/A | NI HVAC System (NHV)<br>Radiation Monitoring System (RMS)<br>Utility Monitoring and Control System (UMC) |
| DL4-RR4c | Vapor Trap Cell Isolation on Overpressure | Required for DID Adequacy | None | N/A | NI HVAC System (NHV)<br>NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>Sodium Cover Gas System (SCG)<br>Utility Monitoring and Control System (UMC) |
| DL4-RR5 | EVST Guard Tank Leak Prevention Function | Required for DID Adequacy | None | N/A | Ex-Vessel Fuel Handling System (FHE) |
| DL4-RR7 | Fuel Handling Building Barrier | Required for DID Adequacy | RFH-ESWR-2 | BDBE | Fuel Handling Building (FHB)<br>NI HVAC System (NHV)<br>NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP) |
| DL4-RR8 | Manual SPS Pump Trip on Leak Indication | Required for DID Adequacy | None | N/A | NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>Reactor Instrumentation System (RIS)<br>Reactor Protection System (RPS)<br>Sodium Processing System (SPS) |
| DL4-DID1 | Intermediate Leak Guard Piping | Required for DID Adequacy | None | N/A | Intermediate Heat Transport System (IHT) |
| DL5-PAM1 | Post Accident Monitoring | Required for DID Adequacy | None | N/A | Ex-Vessel Fuel Handling System (FHE)<br>Water Fuel Pool Handling System (FHP)<br>NI DC Power Supply System (NDC)<br>NI Uninterruptible AC Power Supply System (NUP)<br>NI HVAC System (NHV)<br>Reactor Air Cooling System (RAC)<br>Reactor Instrumentation System (RIS)<br>Radiation Monitoring System (RMS)<br>Reactor Protection System (RPS)<br>Sodium Cover Gas System (SCG)<br>Sodium Processing System (SPS)<br>Nuclear Instrumentation System (XIS) |

Note:

[1] DL4-RC6(a-d) is identified as a defense line function in the most recent revision of the PRA. The requirement has not yet been integrated into the design and is therefore not discussed further in the construction permit application.


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# 5.3 Principal Design Criteria

Principal Design Criteria (PDC) were developed based on the SFR-DC in Regulatory Guide (RG) 1.232, Revision 0, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” as informed by design features specific to the Natrium reactor. Topical report NATD-LIC-RPRT-0002, Revision 0 (Reference 5.3-1), is incorporated by reference into the SAR in its entirety. This section includes the PDC, a summary of the PDC implementation, and references to relevant PSAR chapters and sections containing additional information regarding PDC implementation.

The PDC apply to normal operations, anticipated operational occurrences, and postulated design basis accidents as described in RG 1.232.

## 5.3.1 Overall Requirements

### 5.3.1.1 PDC 1 - Quality Standards and Records

Safety-significant structures, systems, and components shall be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the 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 shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of safety-significant structures, systems, and components shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Implementation Summary

Codes and standards for design, fabrication, erection, and testing are applied to safety-significant structures, systems, and components (SSC) in accordance with their safety significance. Codes and standards are identified in Section 1.4.4 and discussed in other sections where the pertinent subject is described.

The quality assurance program provides adequate assurance that safety-significant SSCs will satisfactorily perform their safety functions is described in Section 8.1. Records of the design, fabrication, erection, and testing of safety-significant SSCs are maintained in accordance with the quality assurance program.

Relevant PSAR Chapters and Sections

Section 1.4.4, Section 8.1

### 5.3.1.2 PDC 2 - Design Bases for Protection Against Natural Phenomena

Safety-significant structures, systems, and components shall be 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



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structures, systems, and components 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 safety significance of the functions to be performed.

Implementation Summary

Natural phenomena considered in the design of safety-significant SSCs are characterized with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area with sufficient margin to account for uncertainties. Characterization of natural phenomena for the site is described in Chapter 2.

Design basis hazard levels are established using the site characterization information. Safety-related (SR) SSCs are designed to withstand the design basis external hazards with no adverse impact to their ability to perform their SR functions. The design basis hazard parameters and their bases are summarized in Section 6.1.1. Non-safety-related with special treatment (NSRST) SSCs are designed to withstand external hazards consistent with the safety significance of the functions they perform using a graded approach through the application of special treatments. NSRST SSCs are protected from external hazards at a different level of severity or intensity than the design basis hazards. Consideration of external hazards in the design of safety-significant SSCs is described in Section 6.1.3, Section 6.4, and Chapter 7.

Relevant PSAR Chapters and Sections

Chapter 2, Section 6.1.1, Section 6.1.3, Section 6.4, Chapter 7

#### 5.3.1.3 PDC 3 - Fire Protection

Safety-significant structures, systems, and components 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 unit, particularly in locations with safety-significant structures, systems, or components. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on safety-significant structures, systems, and components. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Implementation Summary

The plant is designed in accordance with RG 1.189, Revision 5, “Fire Protection for Nuclear Power Plants,” to the extent noted in Section 1.4.1, to minimize the probability and effects of fires and explosions. Noncombustible materials are used where practical. The Nuclear Island (NI) and Energy Island (EI) are physically separate and have separate fire protection systems to minimize the potential for fires originating in the EI from adversely impacting the NI. The NI Fire Protection System (NFP) is designed to have appropriate capacity and capability to minimize the adverse


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effects of fires on safety-significant SSCs. The NFP is also designed such that the rupture or inadvertent operation of the fire protection system does not significantly impair the ability of safety-significant SSCs to perform their safety-significant functions. There are no water-based fire suppression systems in the Reactor Building (RXB), Reactor Auxiliary Building (RAB), or Fuel Handling Building (FHB) areas containing sodium systems.

The general approach to mitigation of sodium leakage is to reduce the contact of sodium with materials that sodium reacts with such as air, water, and concrete. This is accomplished through leakage collection and confinement systems such as guard piping, pipe jackets, and covering concrete surfaces where the potential for contact with sodium leakage exists. Guard piping and pipe jackets have sodium leakage detection and collection systems.

Relevant PSAR Chapters and Sections

Section 7.5.2

#### 5.3.1.4 PDC 4 - Environmental and Dynamic Effects Design Bases

Safety-significant structures, systems, and components shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may 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.

Chemical consequences of accidents, such as sodium leakage, shall be appropriately considered for the design of safety-significant structures, systems, and components, which must be protected.

Implementation Summary

SR SSCs are designed to withstand credible dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids without adverse impact on their ability to perform their SR functions. SR SSCs are designed to withstand the effects of, or are protected from, sodium leaks. Sodium-containing SSCs external to the reactor vessel include a means of collecting and monitoring leakage from moderate energy cracks, where required. Methods include guard pipe, jackets, lined cells, or catch pans.

NSRST SSCs are designed to withstand environmental and dynamic effects in accordance with the safety significance of the functions performed by the SSCs.


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Relevant PSAR Chapters and Sections

Section 6.1.1, Chapter 7

5.3.1.5 PDC 5 - Sharing of Structures, Systems, and Components

Safety-significant structures, systems, and components shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, the ability to achieve and maintain safe shutdown of the remaining units.

Implementation Summary

Kemmerer Unit 1 is a single unit site. Safety-significant SSCs are not shared among units.

Relevant PSAR Chapters and Sections

Section 1.1

5.3.2 Multiple Barriers

5.3.2.1 PDC 10 - Reactor Design

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

Implementation Summary

The Reactor Core System, Reactor Enclosure System (RES), Primary Heat Transport System (PHT), Reactor Protection System (RPS), and NI Control System (NIC) are designed with appropriate margin to assure the Specified Acceptable System Radionuclide Release Design Limits (SARRDL) are not exceeded during normal operation or AOOs. Nuclear design is described in Section 3.12 and the thermal hydraulic design of the reactor core is described in Section 3.13.

Relevant PSAR Chapters and Sections

Section 3.12, Section 3.13, Section 7.1.1, Section 7.1.2, Section 7.1.3, Section 7.1.4, Section 7.2.2, Section 7.6.2, Section 7.6.3

5.3.2.2 PDC 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.


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Implementation Summary

The primary reactivity feedback mechanisms in increasing order of response time are:

1. Doppler – temperature changes in the fuel alter associated neutron cross-sections.
2. Metallic fuel volume – fuel contraction or expansion in the axial direction along the length of the fuel.
3. Sodium coolant density – temperature changes cause the coolant density to change thereby impacting neutron spectrum and leakage.
4. Radial distortion feedback – core assembly displacement and distortion with respect to the radial direction resulting from core load pad restraint expansion and contraction, and grid plate expansion and contraction.

The summation of the prompt acting components of the reactivity feedback mechanisms are net negative in the power operating range and tend to compensate for a rapid increase in reactivity.

Relevant PSAR Chapters and Sections

Section 3.12, Section 7.1.1, Section 7.1.2

### 5.3.2.3 PDC 12 - Suppression of Reactor Power Oscillations

The reactor core; associated structures; and associated coolant, 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.

Implementation Summary

The reactor stability is analyzed as described in Section 7.1.1 to demonstrate that power oscillations can be reliably and readily detected and do not result in conditions exceeding the SARRDLs.

Relevant PSAR Chapters and Sections

Section 3.12, Section 7.1.1, Section 7.1.2, Section 7.2.5

### 5.3.2.4 PDC 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, safety-significant elements of the primary coolant boundary, and functional containment. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.


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Implementation Summary

Instrumentation and controls are provided to monitor and control variables and systems during normal operation, AOOs, design basis events (DBE) and design basis accidents (DBA).

The NIC provides the capability to reliably monitor and control the plant systems during normal power operations and plant startup and shutdown, including steady state and anticipated plant transients. Appropriate controls are provided to maintain associated variables and systems within prescribed operating ranges.

The RPS accepts input signals from plant instrumentation, applies necessary logic, and automatically generates output signals to initiate safety functions including scram and Engineered Safety Feature (ESF) functions when required. The RPS also displays critical information to operators in the Main Control Room (MCR) and Remote Shutdown Complex (RSC) during normal and postulated post-accident conditions.

The Nuclear Instrumentation System (XIS) monitors the neutron flux outside the core during fuel movement, reactor startup, power operations, shutdown, and accident conditions. XIS neutron flux signals are provided to the RPS in support of SR protective functions. The XIS also provides isolated, hard-wired signals to NIC, Coolant Temperature Monitoring and Control System, and Rod Monitoring and Control System (RMC) for rod and coolant temperature control and display in the MCR and the Fuel Handling Control Room, which is used during refueling operations.

The Reactor Instrumentation System provides instrumentation to monitor variables in or near the reactor vessel and transmits corresponding signals to the RPS and NIC for use in protective, control, accident monitoring, and surveillance functions.

The Radiation Monitoring System (RMS) monitors the status of functional containment by providing radiation activity indications for systems, buildings, and radioactive effluent release pathways throughout the plant for use in the protection of plant personnel and the surrounding environment during all normal conditions, accident conditions, and post-accident conditions.

Relevant PSAR Chapters and Sections

Section 7.6

5.3.2.5 PDC 14 - Primary Coolant Boundary

The safety-significant elements of the primary coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Implementation Summary

During Modes 1-3, the primary coolant boundary is composed of the components which retain the primary sodium coolant and the primary cover gas inventories, as necessary to accomplish core heat removal functions and maintain fuel cladding integrity. During Modes 1-3, primary coolant boundary integrity is required to retain primary sodium within the reactor vessel, maintain primary sodium system connection boundaries up to and including the first isolation valve, and


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retain primary cover gas inventory within allowable leakage specifications up to the first isolation valve. Primary coolant boundary components are designed to maintain their boundary integrity functions within design limits up to the specified design conditions and load combinations.

During Mode 4, the primary coolant boundary is composed of the components which directly retain primary sodium coolant as necessary to accomplish core heat removal functions and maintain fuel cladding integrity. Primary cover gas retaining components are not included in the Mode 4 definition of primary coolant boundary because the reactor is placed in a safe and cool shutdown condition. Appropriate functional containment boundaries are available to allow for maintenance on primary coolant boundary components that do not directly retain primary sodium coolant during Mode 4.

The safety-significant elements of the primary coolant boundary are fabricated, erected, and tested in accordance with appropriate codes and standards so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Relevant PSAR Chapters and Sections

Section 6.4.3, Section 7.1.2, Section 7.1.3, Section 7.2.3, Section 7.2.4, Section 7.2.5

#### 5.3.2.6 PDC 15 - Primary Coolant System Design

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

Implementation Summary

Protection and control setpoints are established to ensure the design conditions of the safety-significant elements of the primary coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. The primary coolant boundary is designed and constructed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of the ASME Code as described in applicable sections of Chapter 7. The Sodium Cover Gas System (SCG) provides primary coolant boundary overpressure protection in accordance with the requirements of the ASME Code.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.2.3, Section 7.2.4, Section 7.2.5, Section 7.6

#### 5.3.2.7 PDC 16 - Containment Design

A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactivity to the environment and to ensure that the functional containment safety-significant design conditions are not exceeded for as long as postulated accident conditions require.


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Implementation Summary

Radionuclide retention is implemented using a functional containment strategy, which consists of multiple barriers that control the release of radioactivity to the environment. SSCs that are a part of the functional containment strategy are designed such that their safety-significant design conditions are not exceeded for as long as postulated accident conditions require.

Relevant PSAR Chapters and Sections

Section 1.3.2.1, Chapter 7

5.3.2.8 PDC 17 - Electric Power Systems

Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each power system shall be to provide sufficient capacity and capability to ensure that (1) specified acceptable system radionuclide release design limits are not exceeded as a result of anticipated operational occurrences and (2) safety functions that rely on electric power are maintained in the event of postulated accidents.

The electric power systems shall include an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its safety function.

If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for safety-significant functions is provided.

Implementation Summary

Kemmerer Unit 1 is a passive design that does not rely on offsite or onsite electrical power to perform SR functions. Failures of the electric power system do not adversely affect the performance of SR functions.

Electric power is not required to ensure that SARRDLs are not exceeded as a result of AOOs, and SR functions for reactor scram and engineered safety features are achieved without reliance on electrical power. While AOO descriptions capture the plant response including the expected behavior of NSRST and non-safety-related with no special treatment (NST) SSCs, AOOs can be successfully mitigated without exceeding SARRDLs by only SR SSCs without reliance on electrical power. SR functions used to demonstrate mitigation of AOOs and DBAs use passive forces such as natural circulation and gravity. Functions that require NSRST electrical power are supplied through the NSRST electrical systems.

Relevant PSAR Chapters and Sections

Section 7.7.1


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5.3.2.9 PDC 18 - Inspection and Testing of Electric Power Systems

Safety-significant electric power systems shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among systems.

Implementation Summary

The safety-significant electrical systems are designed with a capability to periodically test the operability and functional performance of the components of the systems and the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among systems. Inspection and testing of electrical systems do not prevent safety-related functions of other systems from being performed.

Relevant PSAR Chapters and Sections

Section 7.7.1

5.3.2.10 PDC 19 - Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent, as defined in § 50.2, for the duration of the accident.

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

Adequate protection against sodium aerosols shall be provided to permit access and occupancy of the control room under accident conditions.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during shutdown, and (2) with a potential capability for subsequent safe shutdown of the reactor through the use of suitable procedures.

Implementation Summary

The MCR contains instrumentation and controls to operate the reactor safely under normal conditions and to maintain the reactor in a safe condition under accident conditions.


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The MCR design allows habitability during normal conditions, AOOs and DBAs as defined below:

- During normal conditions, heating, ventilation, and air conditioning (HVAC) is provided to the MCR by the NI HVAC System (NHV). Aerosol detectors, smoke detectors, and radiation detectors are provided in the MCR outside air duct.
- If high levels of radiation or aerosols (including sodium aerosols) are detected, the MCR HVAC system is isolated and placed in recirculation.
- In the event of an MCR evacuation, operators scram the reactor, perform any required manual actions and evacuate to the RSC. Safe shutdown can also be achieved outside the MCR from equipment provided near the RSC. The RSC provides the ability to monitor plant parameters outside the MCR. Radiological dose calculations for MCR occupants during DBAs are evaluated using DBE source terms which credit the existence of buildings and FHB HVAC equipment including high-efficiency particulate absorbing filtration prior to release to the environment. Conservative MCR dispersion factors, developed following RG 1.194, Revision 0, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” maximize radionuclide concentrations at the MCR receptor from the released source term. Considering dose mitigation from NSRST SSCs and the use of DBE source terms for MCR dose assessments is considered acceptable because there are no operator actions performed from the MCR required for the successful performance of SR functions. The use of MCR HVAC system isolation is not required to meet the PDC 19 dose requirement for DBAs.

Relevant PSAR Chapters and Sections

Section 7.5.1, Section 7.6.7, Section 7.8.4

5.3.3 Reactivity Control

5.3.3.1 PDC 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 the specified acceptable system radionuclide release design limits is not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of safety-significant systems and components.

Implementation Summary

The RPS is designed to accept input signals from instruments to monitor and control plant conditions during postulated accidents and reliably initiate the SR scram and engineered safety feature functions as necessary to ensure reactor safety. The RPS is also designed to ensure SARRDL are not exceeded as a result of AOOs.


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The portions of the RPS which support SR functions are arranged into four redundant divisions each with physical, electrical, and functional independence from the other redundant divisions. Each process variable used by the RPS to perform a safety function is measured by four redundant sensing instruments or groups of instruments which are also divided into four divisions.

Relevant PSAR Chapters and Sections

Section 7.6.3, Section 7.6.4, Section 7.6.5

5.3.3.2 PDC 21 - Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Implementation Summary

Conformance with industry standards ensures that the RPS is designed for high functional reliability, redundancy, and inservice testability commensurate with the safety functions to be performed.

The SR portions of the RPS are arranged into four redundant and independent divisions. Each process variable input used by the RPS to perform a safety function is measured by four redundant sensing instruments or groups of instruments which are also divided into four redundant and independent divisions. The RPS actuates reactor scram and ESF functions when at least 2 out of the 4 divisions independently determine that such action is required. The divisional arrangement and 2-out-of-4 voting design features allow inservice testability. The RPS design allows for any division to be taken out of service for testing at any time without loss of any safety function and without loss of minimum required redundancy.

Relevant PSAR Chapters and Sections

Section 7.6.3, Section 7.6.4, Section 7.6.5

5.3.3.3 PDC 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


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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.

Implementation Summary

Conformance with industry standards ensures that the RPS is 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.

The RPS cabinets are contained in the NI Control Building substructure. The SR NI Control Building substructure is designed to provide appropriate seismic protection and protection from natural phenomena such that there is not a loss of the RPS safety function. The RPS vault rooms are designed to meet fire protection requirements.

Relevant PSAR Chapters and Sections

Section 7.6.3, Section 7.6.4, Section 7.6.5

#### 5.3.3.4 PDC 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 (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, sodium and sodium reaction products, pressure, steam, water, and radiation) are experienced.

Implementation Summary

The RPS, associated divisions, and the reactor trip breakers and ESF breakers fail safe on a loss of power. A shunt trip is also provided in addition to the undervoltage trip to increase the reactor trip breaker and the ESF breaker reliability and diversity. The RPS equipment is environmentally and seismically qualified to remain operable during anticipated adverse conditions. The RPS performs self-diagnostics and places outputs in a safe state if a critical fault is detected.

Relevant PSAR Chapters and Sections

Section 7.6.3, Section 7.6.4

#### 5.3.3.5 PDC 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.


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Implementation Summary

Conformance with industry standards ensures that the RPS is separated from control systems to the extent that failure of any single control system component or channel, 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 RPS.

There is no direct communication from NI control systems to the RPS. The RPS is isolated and functionally independent of the non-safety control systems and uses input signals from independent instrument channels. One-way communication from the RPS to control systems is utilized via a data diode or hardwired connection. RPS components such as shared sensors are isolated from control systems using SR isolation hardware.

Relevant PSAR Chapters and Sections

Section 7.6.2, Section 7.6.3, Section 7.6.4, Section 7.6.5, Section 7.6.8

5.3.3.6 PDC 25 - Protection System Requirements for Reactivity Control Malfunctions

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

Implementation Summary

The RPS reactor scram and ESF functions are designed to ensure that SARRDLs are not exceeded as a result of anticipated operational occurrences even when a single failure of the reactivity control system has occurred.

The RPS is designed with four independent divisions. Failure of one division does not preclude completion of safety functions. XIS provides flux signals to RPS in support of SR protective functions, and the four XIS divisions correspond to the four RPS divisions. In addition, the NIC provides reactivity control during normal operations and under AOOs through power runback by RMC and Coolant Temperature Monitoring and Control System. The NIC is independent of the RPS and failure of the NIC does not impact RPS safety and reactivity control functions.

Relevant PSAR Chapters and Sections

Section 7.6.2, Section 7.6.3, Section 7.6.4

5.3.3.7 PDC 26 - Reactivity Control Systems

A minimum of two reactivity control systems or means shall provide:

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


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2. A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the specified acceptable system radionuclide release design limits are not exceeded.

3. 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 core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.

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

Implementation Summary

Two reactivity control means, as well as a primary and secondary set of rods, are provided. The two reactivity control means are rod de-latch and rod motor drive. The RPS initiates rod de-latch to drop both the primary and secondary set of rods into the core using gravity (scram). After rod de-latch, the RPS also sends a sends a signal to the rod motor drives to fully insert the rods into the core. There are no common cause failures between the rod de-latch and rod motor drive. The bundle geometry is also diverse with respect to the primary and secondary rod sets to eliminate common cause failure from effecting all rods.

Each part of PDC 26 is met as follows:

1. The RPS initiates the de-latch of the Primary and Secondary sets of rods to insert negative reactivity at an amount and rate to maintain conditions less than the specified acceptable system radionuclide release limits design during normal operation, including anticipated operational occurrences. The reactor is safely shutdown including accounting for the appropriate malfunction. This consideration of malfunctions is accounted for in two different types of scenarios. In the first scenario the highest worth rod is assumed to be held in the fully withdrawn position as the malfunction with the other rods succeeding to drop by gravity, and achieves hot standby with margin. The second scenario considers either the primary or secondary set failing to insert as the malfunction with the other set succeeding to drop by gravity, and achieves safe shutdown following the event.

2. The rod motor drive provides a means independent and diverse from de-latch by inserting sufficient negative reactivity at the rate necessary that the SARRDLs are not exceeded from planned, normal power changes. Additionally, the primary and secondary control rod bundle geometric designs are unique and provide protection against a common mode failure of rod binding.

3. The RPS initiates the de-latch of the Primary and Secondary sets of rods to insert negative reactivity at a rate and amount to assure safe shutdown following a postulated accident, while maintaining the capability to cool the core. The reactor is safely shutdown including accounting for the appropriate malfunction. This consideration of malfunctions is accounted for in two different types of scenarios. In the first scenario the highest worth rod is assumed to be held in the fully withdrawn position as the malfunction with the other


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rods succeeding to drop by gravity, and achieves hot standby with margin. The second scenario considers either the primary or secondary set failing to insert as the malfunction with the other set succeeding to drop by gravity, and achieves safe shutdown following the event.

4. Fully inserted primary and secondary control rods, along with administrative controls that preclude removal of more than one rod bundle form the reactor core, maintain the reactor shutdown under conditions which allow for fuel loading, inspection, and repair. Administrative controls include reloading control rods first during a refueling evolution.

Relevant PSAR Chapters and Sections

Section 7.1.1, Section 7.1.2, Section 7.2.5, Section 7.6.2, Section 7.6.3

5.3.3.8 PDC 28 - Reactivity Limits

The reactivity control systems 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 safety-significant elements of the primary coolant 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 core.

Implementation Summary

The design of the RCC, RMC, CRD, and RPS ensure that postulated reactivity AOOs, DBEs, and DBAs do not result in damage to the safety-significant elements of the primary coolant boundary greater than limited local yielding, nor result in disturbance of the core, its support structures, or other reactor vessel internals that significantly impair the capability to cool the core. The RES supports reactivity control functions by providing structural support to Control Rod Drive Mechanisms and drivelines, and maintaining control rod driveline alignment with control assemblies to limit the potential amount and rate of reactivity increase during postulated reactivity events. RES contributes to structurally supporting and maintaining the geometry of reactor core assemblies, to limit fuel assembly deflections, and provide adequate coolant flow and heat removal. The RMC and CRD control the maximum control rod withdrawal speed to limit the reactivity worth insertion rate. A CRD control interlock prevents withdrawal of more than one control rod at a time. The Control Rod Drive Mechanism also includes a brake to hold the control rod assembly at any selected position within the stroke of travel and prevent it from inadvertent insertion or withdrawal.

Relevant PSAR Chapters and Sections

Section 7.1.1, Section 7.1.2, Section 7.2.5, Section 7.6.2, Section 7.6.3

5.3.3.9 PDC 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.


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Implementation Summary

The design of the RCC, CRD, NIC, RPS, XIS, and Reactor Instrumentation System assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

Relevant PSAR Chapters and Sections

Section 7.1.1, Section 7.2.5, Section 7.6.2, Section 7.6.3, Section 7.6.4, Section 7.6.5

5.3.4 Fluid and Heat Transport Systems

5.3.4.1 PDC 30 - Quality of Primary Coolant Boundary

Components that are part of the primary coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of primary coolant leakage.

Implementation Summary

Codes and standards for the design, fabrication, erection, and testing of primary coolant boundary components are described in Chapter 7. Codes and standards are selected commensurate with the safety significance of the functions performed by the primary coolant boundary components. Monitoring equipment in the SCG is capable of determining if a leak from the reactor vessel to the guard vessel exists. Sodium Processing System (SPS) piping containing primary coolant is also monitored for leakage.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.2.3, Section 7.2.4, Section 7.2.5

5.3.4.2 PDC 31 - Fracture Prevention of Primary Coolant Boundary

Safety-significant components that are part of the primary coolant boundary shall be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture 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 under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

Implementation Summary

Material selection and design of safety-significant components that are part of the primary coolant boundary comply with ASME Code as defined in Chapter 7 which ensures such components behave in a nonbrittle manner, are compatible with thermal effects, and are


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compatible with liquid sodium and its aerosols, vapors, and oxidation products to minimize the probability of rapidly propagating failure. Safety-significant RES SSCs that are part of the primary coolant boundary are constructed with austenitic stainless steel to minimize irradiation embrittlement and fracture from thermal stresses.

Relevant PSAR Chapters and Sections

Section 6.4.3, Section 7.1.2, Section 7.1.3, Section 7.2.3, Section 7.2.4, Section 7.2.5

#### 5.3.4.3 PDC 32 - Inspection of Primary Coolant Boundary

Safety-significant components that are part of the primary coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.

Implementation Summary

Strategies for monitoring and non-destructive examination pre-service inspection and in-service inspection of the primary coolant boundary are implemented by the Reliability and Integrity Management (RIM) Program to ensure SSCs affecting plant safety are monitored and examined at appropriate frequencies for acceptable performance. Access is provided to implement the RIM Program for all primary coolant boundary components.

The RIM Program and assessment of available testing and service data provide an appropriate material surveillance program for the reactor vessel without the use of material surveillance coupons as described in Section 7.1.2.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.2.3, Section 7.2.4, Section 7.2.5, Chapter 8

#### 5.3.4.4 PDC 33 - Primary Coolant Inventory Maintenance

A system to maintain primary coolant inventory for protection against small breaks in the primary coolant boundary shall be provided as necessary to ensure that specified acceptable system radionuclide release design limits are not exceeded as a result of primary coolant inventory loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that the system safety function can be accomplished using the piping, pumps, and valves used to maintain primary coolant inventory during normal reactor operation.

Implementation Summary

The primary coolant inventory is protected against small breaks in primary coolant boundary components by the SPS pump trip. If the low low primary sodium level setpoint is reached due to leakage of primary coolant through the SPS, the SPS pumps are tripped by the RPS or the Auxiliary Monitoring and Control System.


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The design of the RES prevents primary coolant inventory loss that results in exceeding the SARRDLs. There are no penetrations through the reactor vessel and the primary coolant boundary penetrations through the reactor head are above the operating level of primary sodium in the cover gas space. In the unlikely event of reactor vessel boundary leakage, the surrounding guard vessel contains leaked sodium. The annular space between the reactor vessel and guard vessel is sized such that the equilibrium coolant level remains sufficient to remove reactor core heat.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.2.4, Section 7.6.3

#### 5.3.4.5 PDC 34 - Residual Heat Removal

A 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 reactor core at a rate such that specified acceptable system radionuclide release design limits and the design conditions of the primary coolant boundary are not exceeded.

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

Implementation Summary

The RES, PHT, IHT, and IAC transfer fission product decay heat and other residual heat from the reactor core during normal operation and AOOs at a rate such that SARRDLs and the design conditions of the primary coolant boundary are not exceeded. Two intermediate loops are provided, each with passive and active heat removal modes. The combination of passive and active residual heat removal functions provides defense-in-depth for residual heat removal.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.1.4, Section 7.2.2

#### 5.3.4.6 PDC 35 - Emergency Core Cooling

A system to assure sufficient core 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 reactor core during and following postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.


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Implementation Summary

The RES establishes a natural circulation circuit via the PHT to passively transfer decay heat from the reactor core to the Reactor Air Cooling System (RAC) such that fuel and clad damage that could interfere with continued effective core cooling is prevented. The RAC is reliable, passive, continuously in operation, and has no isolation. RAC is configured with four inlets and four outlets with two of each on opposite ends of the RXB to provide separation in the event of external hazards. The PSPs and ISPs are tripped when excessive temperatures are reached in the primary sodium.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.1.4, Section 7.2.1

5.3.4.7 PDC 36 - Inspection of Emergency Core Cooling System

A system that provides emergency core cooling shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.

Implementation Summary

The RES, PHT, and RAC are designed to permit access for appropriate periodic inspection of important components in accordance with the RIM Program to ensure the integrity and capability of the systems.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.2.1, Chapter 8

5.3.4.8 PDC 37 - Testing of Emergency Core Cooling System

A system that provides emergency core cooling shall be designed to permit appropriate periodic functional testing to ensure (1) the structural 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 any associated systems and interfaces necessary to transfer decay heat to the ultimate heat sink.

Implementation Summary

The RES, PHT, and RAC components that provide decay heat removal are included in the scope of the RIM Program. The performance capability and operability of the RES, PHT, and RAC natural circulation flow path is monitored through reactor instrumentation and RAC instrumentation.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.2.1


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5.3.4.9 PDC 44 - Structural and Equipment Cooling

A system to transfer heat from safety-significant structures, systems, and components to an ultimate heat sink shall be provided, as necessary, to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

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

Implementation Summary

The NST NHV cooling function removes heat from safety-significant SSCs to facilitate their operation during normal operating conditions. Active structural and equipment cooling systems are not required to support the function of safety-significant SSCs during DBAs.
Safety-significant SSCs are cooled passively during DBAs.

Relevant PSAR Chapters and Sections

Section 7.5.1

5.3.4.10 PDC 45 - Inspection of Structural and Equipment Cooling Systems

The structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the systems.

Implementation Summary

The NHV system is designed to permit appropriate periodic inspection of its NST heat removal functions that support normal operation of safety-significant SSCs. The condition of passive heat sinks will be monitored for degradation against critical parameters assumed in the analyses.

Relevant PSAR Chapters and Sections

Section 7.5.1

5.3.4.11 PDC 46 - Testing of Structural and Equipment Cooling Systems

The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural integrity of their components, (2) the operability and performance of the system components, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring the systems into operation for reactor shutdown and postulated accidents, including the operation of associated systems.


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Implementation Summary

The NHV system is designed to permit appropriate periodic inspection of its NST heat removal functions that support normal operation of safety-significant SSCs. The condition of passive heat sinks will be monitored for degradation against critical parameters assumed in the analyses.

Relevant PSAR Chapters and Sections

Section 7.5.1

5.3.5 Fuel and Reactivity Control

5.3.5.1 PDC 60 - Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to suitably control 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.

Implementation Summary

The Gaseous Radwaste Processing System (RWG), Liquid Radwaste Processing System (RWL), and Solid Radwaste Processing System (RWS) control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation and AOOs. The RMS and NIC provide monitoring and control for the RWG, RWS, and RWL radioactive waste processing functions. The SCG controls and filters exhaust gas from the reactor vessel and guard vessel to the RWG.

Relevant PSAR Chapters and Sections

Section 7.2.3, Section 7.4.1, Section 7.6.2, Section 7.6.6, Section 9.1, Section 9.3

5.3.5.2 PDC 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure 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-significant components, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the safety significance of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.


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Implementation Summary

Fuel storage and handling, radioactive waste, and other systems that contain radioactivity are designed to support safety-significant functions, as needed, to ensure adequate safety under normal and postulated accident conditions. The SSCs within the scope of PDC 61 are:

- RES In-Vessel Storage
- In-Vessel Fuel Handling System (FHI)
- Ex-Vessel Fuel Handling System (FHE)
- Water Pool Fuel Handling System (FHP)
- RWS
- RWL
- RWG
- SPS

The SSC descriptions in Chapter 7 include identification of applicable design attributes needed to ensure adequate safety under normal and postulated accident conditions. These design attributes include the following, as applicable:

- The capability for inspection and testing of components as appropriate based on safety-significant functions of the components (see also Testing Program, RIM Program, Design Reliability Assurance Program, Inservice Testing Program, and Post-Construction Inspection Testing and Analysis Program, as described in Chapter 8).
- Suitable shielding for radiation protection.
- Containment, confinement, and filtering systems, as appropriate based on safety-significant functions of the components.
- Residual heat removal capability having reliability and testability that reflects the safety significance of heat removal.
- Fuel storage cooling availability under accident conditions.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.2.4, Section 7.3.1, Section 7.3.2, Section 7.3.3, Section 7.4.1, Section 7.6.2, Chapter 8, Section 9.1, Section 9.3

### 5.3.5.3 PDC 62 - Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by the physical systems or processes, preferably by use of geometrically safe configurations.

Implementation Summary

The design of the RES In-Vessel Storage, FHI, FHE, and FHP fuel storage and handling systems prevents inadvertent criticality by using design attributes such as geometrically safe configurations and fixed neutron absorbers.


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Relevant PSAR Chapters and Sections

Section 3.14, Section 7.1.2, Section 7.3

5.3.5.4 PDC 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 residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Implementation Summary

The systems subject to PDC 63 requirements are FHE, FHP, NIC, SPS, SCG, RWS, RWL, and RWG.

Fuel handling systems provide heat removal capabilities, including detecting the loss of heat removal capabilities, as needed to support safety-significant functions. Fuel handling systems provide functions such as temperature monitoring, water level monitoring, flow monitoring, equipment alarms, leak detection, and radiation monitoring to allow operators to act, as needed, to address potential loss of heat removal capability or excessive radiation levels. Radioactive waste systems include instruments that detect excessive radiation levels to allow initiation of appropriate actions. The RMS, NIC, SCG, and SPS monitor normal operational radiation levels and potential off-normal conditions, including area radiation monitoring, process radiation monitoring, and effluent radiation monitoring.

Relevant PSAR Chapters and Sections

Section 7.2.3, Section 7.2.4, Section 7.3, Section 7.4.1, Section 7.6.2, Section 7.6.6, Chapter 9

5.3.5.5 PDC 64 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor building atmosphere, spaces containing components for primary system sodium and cover gas cleanup and processing, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Implementation Summary

Exhaust from the RXB, FHB, and RAB pass through a combined stack for effluent monitoring. Means are provided for monitoring the RAC outlet flow to quantify radioactive effluent discharge. The RMS and Plant Monitoring and Control System monitor the RXB atmosphere, NI building spaces containing components for primary system sodium and cover gas cleanup and processing, gaseous effluent discharge paths, and the plant environs for radiological release.

Relevant PSAR Chapters and Sections

Section 7.2.1, Section 7.4.1, Section 7.5.1, Section 7.6.2, Section 7.6.6


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## 5.3.6 Additional PDC

### 5.3.6.1 PDC 70 - Intermediate Coolant System

If an intermediate cooling system is provided, then the intermediate coolant system shall be designed with sufficient margin to assure that (1) the design conditions of the intermediate coolant boundary are not exceeded during normal operations, including anticipated operational occurrences, and (2) the integrity of the primary coolant boundary is maintained during postulated accidents.

Implementation Summary

The IHT, IAC, SCG, and SPS intermediate coolant boundary components are designed to appropriate industry codes to assure the design conditions of the intermediate coolant boundary are maintained during normal operations, AOOs and DBAs and the integrity of the primary coolant boundary is maintained during DBAs. The intermediate SCG provides intermediate coolant boundary overpressure protection in accordance with the requirements of the ASME Code. The effects of the sodium-salt reaction due to a postulated leak in the SHX is the subject of research and development as described in Chapter 13.

Relevant PSAR Chapters and Sections

Section 7.1.4, Section 7.2.2, Section 7.2.3, Section 7.2.4

### 5.3.6.2 PDC 71 - Primary Coolant and Cover Gas Purity Control

Systems shall be provided as necessary to maintain the purity of primary coolant sodium and cover gas within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radionuclide concentrations, and (4) air or moisture ingress as a result of a leak of cover gas.

Implementation Summary

Primary SPS removes contaminants from the primary coolant. Primary SCG provides argon gas to various components on the reactor head and the reactor vessel between the sodium coolant and the head. The SCG maintains a positive pressure in the reactor vessel, provides a minimum sweep flow to remove any fission gases and contaminants which can accumulate, and provides a gas stream to the cover gas monitoring system.

Barrier gas seals are used for static seals to ensure that argon gas leaks into associated equipment and prevents radiological effluent from being released. The SCG monitors for impurities including oxygen, hydrogen, methane, and nitrogen.

Relevant PSAR Chapters and Sections

Section 7.1.3, Section 7.2.3, Section 7.2.4


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### 5.3.6.3 PDC 72 - Sodium Heating Systems

Heating systems shall be provided for safety-significant systems and components that contain or could be required to contain sodium. 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 sodium are maintained within design limits assuming a single failure. If plugging of any cover gas line due to condensation or plate out of sodium aerosol or vapor could prevent accomplishing a safety function, the temperature control and the relevant corrective measures associated with that line shall be considered safety-significant.

Implementation Summary

Safety-significant sodium containing systems include the PHT, IHT, IAC, CRD, RES, SPS, SCG, and FHE. During normal operation, decay heat prevents sodium freezing. When decay heat is insufficient to maintain adequate sodium temperatures, operation of the Primary Sodium Pumps and Intermediate Sodium Pumps provides sufficient heat to the primary and intermediate sodium to prevent freezing. Heat trace systems, process heaters, and associated controls are designed to prevent adverse temperature distributions in sodium-containing systems.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.1.4, Section 7.2.2, Section 7.2.3, Section 7.2.4, Section 7.3.2, Section 7.6.2

### 5.3.6.4 PDC 73 - Sodium Leakage Detection and Reaction Prevention and Mitigation

Means to detect and identify sodium leakage as practical and to limit and control the extent of sodium-air and sodium-concrete reactions and to mitigate the effects of fires resulting from these sodium-air and sodium-concrete reactions shall be provided to ensure that the safety functions of safety-significant structures, systems, and components are maintained. Systems from which sodium leakage constitutes a significant safety hazard shall include measures for protection, such as inserted enclosures or guard vessels.

Implementation Summary

The PHT, IHT, IAC, RES, SPS, SCG, and FHE include a means of detecting or limiting sodium-air and sodium-concrete reactions as necessary to ensure that the safety functions of safety-significant structures, systems, and components are maintained. Methods include use of:

- seals and inert gases to prevent oxygen ingress into sodium containing vessels
- leak containing features such as cells, guard pipes, or clamshell enclosures to limit the interaction between leaked sodium and oxygen or concrete
- leak jackets and catch pans to collect and redirect leakage to limit the sodium-oxygen and sodium-concrete interaction
- inerted guard vessels surrounding the Reactor Vessel and Ex-Vessel Storage Tank
- leakage detection equipment


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Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.4, Section 7.2.2, Section 7.2.3, Section 7.2.4, Section 7.3.2, Section 7.5.2

### 5.3.6.5 PDC 74 - Sodium/Water Reaction Prevention/Mitigation

Structures, systems, and components containing sodium shall be designed and located to avoid contact between sodium and water and to limit the adverse effects of chemical reactions between sodium and water on the capability of any structure, system, or component to perform any of its intended safety functions. If steam-water is used for energy conversion, to prevent loss of any plant safety function, the sodium-steam generator system shall be designed to detect and contain sodium-water reactions and limit the effects of the energy and reaction products released by such reactions, including mitigation of the effects of any resulting fire involving sodium.

Implementation Summary

In general, water-containing systems and components are not located in proximity to sodium-containing systems and components. There are no water-based fire suppression systems in the RXB or RAB. Water-based fire suppression systems in the FHB are separated from sodium-containing systems and components. There are no sodium-water steam generators; the NI Salt System transfers heat to the EI.

The PIC uses an inert gas stream to remove external sodium from core assemblies prior to immersion in the SFP. PIC sodium removal is performed in a physically isolated cleaning vessel and uses a combination of control system permissives, software integrity level, and passive engineered controls to preclude excess sodium-water reactions.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.1.4, Section 7.2.2, Section 7.2.3, Section 7.2.4, Section 7.3.1, Section 7.3.2, Section 7.5.2

### 5.3.6.6 PDC 75 - Quality of the Intermediate Coolant Boundary

Components that are part of the intermediate coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed.

Implementation Summary

The intermediate coolant boundary is composed of portions of the IHT, IAC, SPS, and SCG. Intermediate coolant boundary components are designed and constructed in accordance with ASME codes and standards to ensure that design conditions of the intermediate coolant boundary and primary coolant boundary are not exceeded. Codes and standards for the design, fabrication, erection, and testing of intermediate coolant boundary components are described in Chapter 7. Codes and standards are selected commensurate with the safety significance of the functions performed by the intermediate coolant boundary components.


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Relevant PSAR Chapters and Sections

Section 7.1.4, Section 7.2.2, Section 7.2.3, Section 7.2.4

5.3.6.7 PDC 76 - Fracture Prevention of the Intermediate Coolant Boundary

Safety-significant components that are part of intermediate coolant boundary shall be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

Implementation Summary

Material selection and design of the safety-significant components part of the intermediate coolant boundary are performed in accordance with the codes and standards described in Chapter 7, which ensures the components behave in a nonbrittle manner and minimizes the probability of a rapidly propagating failure.

Relevant PSAR Chapters and Sections

Section 7.1.4, Section 7.2.2, Section 7.2.3, Section 7.2.4

5.3.6.8 PDC 77 - Inspection of the Intermediate Coolant Boundary

Components that are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity commensurate with the system's safety significance, and (2) an appropriate material surveillance program for the intermediate coolant boundary.

Implementation Summary

The IHT, IAC, SCG, and SPS intermediate coolant boundary components are designed to permit access for appropriate periodic inspection and functional testing of important components in accordance with the RIM Program to ensure the integrity and capability of the systems.

Relevant PSAR Chapters and Sections

Section 7.1.4, Section 7.2.2, Section 7.2.3, Section 7.2.4, Chapter 8

5.3.6.9 PDC 78 - Primary Coolant System Interfaces

When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically incompatible with the primary coolant, the interface location shall be designed to ensure that the primary coolant is separated from the chemically incompatible fluid by two redundant, passive barriers.

When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically compatible with the primary coolant, then the interface location is allowed to be a single passive barrier provided that the following conditions are met:


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(1) postulated leakage at the interface location does not result in failure of the intended safety functions of safety-significant structures, systems or components or result in exceeding the specified acceptable system radionuclide release design limits (2) the fluid contained in the structure, system, or component is maintained at a higher pressure than the primary coolant during normal operation, anticipated operational occurrences, shutdown, and accident conditions.

Implementation Summary

The interface between the IHT and primary coolant system is a single passive barrier since both contain sodium fluids. The IHT is maintained at a higher pressure than the primary coolant by static head. The higher pressure requirement is to ensure any leakage in the interface between the two systems does not result in a release of radioactive primary coolant to the nonradioactive part of the heat transport system. The primary and intermediate sodium coolant interface in the IHX is described in Chapter 7.

The SPS system cold traps and SPS and SCG pressure boundary cells are single passive barriers with nitrogen as the interfacing fluid. Nitrogen is chemically compatible with the sodium coolant. The cold traps and cells are isolable from the primary coolant system to limit loss of primary system inventory resulting from postulated leakage. A pressure differential between the nitrogen and the primary coolant system in these systems is not required because leakage will not result in contaminating the nonradioactive part of the heat transport system, and leakage across the interface in either direction does not result in failure of the intended SR and NSRST functions.

The SCG directly interfaces with the primary coolant; there is no boundary interface location between the primary coolant and the SCG. Refer to the discussions provided for PDC 71 and PDC 79 on maintaining the argon interface with the primary coolant.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.1.3, Section 7.1.4

5.3.6.10 PDC 79 - Cover Gas Inventory Maintenance

A system to maintain cover gas inventory shall be provided as necessary to ensure that the primary coolant sodium design limits are not exceeded as a result of cover gas loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the primary coolant boundary.

Implementation Summary

Cover gas inventory in the reactor vessel is maintained by the SCG and cover gas makeup is provided at a sufficient rate to ensure the design limits of the primary coolant sodium are not exceeded due to postulated primary coolant boundary leakage.

Relevant PSAR Chapters and Sections

Section 7.1.2, Section 7.2.3


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5.3.6.11 PDC 80 - Reactor Vessel and Reactor System Structural Design Basis

The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.

Implementation Summary

The reactor vessel, reactor head, and reactor internals are designed in accordance with ASME BPVC Section III, Division 5 to ensure their integrity is maintained during postulated accidents. The structural design of the reactor vessel, reactor head, and reactor internals allow for sufficient passive removal of residual heat and insertion of the neutron absorbers during DBAs.

Relevant PSAR Chapters and Sections

Section 7.1.1, Section 7.1.2

5.3.6.12 PDC 81 - Reactor Building Design Basis

The design of the reactor building shall be such that, during postulated accidents, it structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink.

Implementation Summary

The RXB structurally supports and protects the RES and RAC such that the geometry for passive removal of residual heat from the reactor core is maintained during DBAs.

Relevant PSAR Chapters and Sections

Section 7.8.1

5.3.6.13 PDC 82 - Provisions for Periodic Reactor Building Inspection

The reactor building shall be designed to permit (1) appropriate periodic inspection of all important structural areas, and (2) an appropriate surveillance program.

Implementation Summary

The RXB is designed to permit periodic structural inspections of important areas. The cylindrical reactor cavity is accessible by remotely operated inspection equipment.

Relevant PSAR Chapters and Sections

Section 7.8.1


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## References

5.3-1 TerraPower, “Principal Design Criteria for the Natrium Advanced Reactor,” NATD-LIC-RPRT-0002, Revision 0


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## 5.4 Safety-Related Structures, Systems, and Components

The methodology for the safety classification of structures, systems, and components (SSCs) is summarized in Section 5.1. Safety-related (SR) SSCs, SR functions performed by SR SSCs, and the anticipated operational occurrences, design basis events, and beyond design basis events the SR functions are credited in are summarized in Section 5.2. SR SSCs, non-safety-related with special treatment (NSRST) functions performed by SR SSCs, and the anticipated operational occurrences, design basis events, and beyond design basis events the NSRST functions are credited in are summarized in Section 5.2.



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# 5.5 Non-Safety-Related with Special Treatments Structures, Systems, and Components

The methodology for the safety classification of structures, systems, and components (SSCs) is summarized in Section 5.1. Non-safety-related with special treatment (NSRST) SSCs and the safety-significance criteria that establish the NSRST SSC safety classifications are summarized in Section 5.2. NSRST functions performed by NSRST SSCs and the anticipated operational occurrences, design basis events, and beyond design basis events the NSRST functions are credited in are also summarized in Section 5.2.



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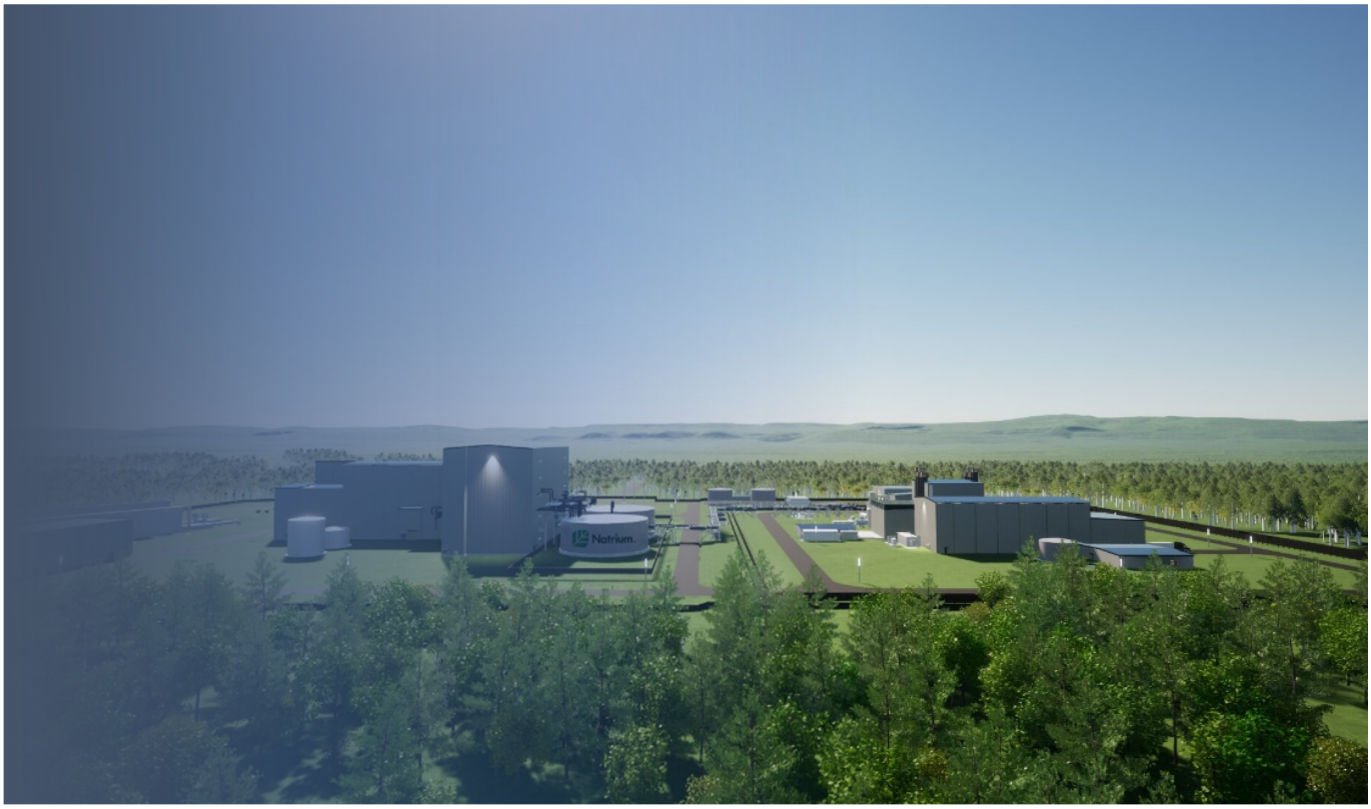

# Kemmerer Power Station Unit 1
# PSAR, Chapter 6

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## TABLE OF CONTENTS

**Chapter 6 Safety-Significant SSC Criteria and Capabilities . . . . . 6.1-1**

6.1 Design Requirements for Safety-Related SSCs. . . . . 6.1-1

6.1.1 Design Basis Hazard Levels . . . . . 6.1-1

6.1.2 Summary of SRDC. . . . . 6.1-4

6.1.3 Summary of DBHL-Related Requirements for Non-Safety-Related SSCs . . . . . 6.1-5

6.2 Reliability and Capability Targets for Safety-Significant SSCs. . . . . 6.2-1

6.3 Special Treatment Requirements for Safety-Significant SSCs . . . . . 6.3-1

6.4 Design of Safety-Significant SSCs. . . . . 6.4-1

6.4.1 Seismic Design. . . . . 6.4-1

6.4.2 Design of Safety-Significant Structures . . . . . 6.4-10

6.4.3 Mechanical Systems and Components . . . . . 6.4-24

6.4.4 ASME BPVC, Section III Piping Systems, Piping Components, and Associated Supports . . . . . 6.4-33


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## LIST OF TABLES


Table 6.1-1 Design Basis Hazard Levels . . . . . 6.1-7

Table 6.4-1 NSRST SSC Seismic Classifications and Seismic Criteria . . . . . 6.4-37

Table 6.4-2 Design Codes and Standards for Safety-Significant Structures . . . . . 6.4-38

Table 6.4-3 Applicability of Structure Primary Design Codes Based on Safety Classification . . . . . 6.4-40

Table 6.4-4 Applicability of Regulatory Guides to Safety-Significant Structures . . . . 6.4-41

Table 6.4-5 Load Combinations and Load Factors for Concrete SR Substructures . . . . . 6.4-43

Table 6.4-6 Load Combinations and Load Factors for Structural Steel of SR Substructures . . . . . 6.4-45

Table 6.4-7 Load Combinations for Extraordinary Events on NSRST Substructure Analysis of Concrete Structures . . . . . 6.4-47

Table 6.4-8 Load Combinations for Extraordinary Events on NSRST Substructure Analysis of Steel Structures . . . . . 6.4-49

Table 6.4-9 Load Factors and Factors of Safety for SR Structural Stability . . . . . 6.4-51

Table 6.4-10 Material Properties of Uncracked and Cracked Concrete . . . . . 6.4-52

Table 6.4-11 Load Combinations for Extraordinary Events on NSRST Superstructure Analysis of Concrete Structures . . . . . 6.4-53

Table 6.4-12 Load Combinations for Extraordinary Events on NSRST Superstructure Analysis of Steel Structures . . . . . 6.4-55

Table 6.4-13 Design Basis Tornado Parameters . . . . . 6.4-57

Table 6.4-14 Design Basis Tornado Missile Spectrum and Maximum Speed . . . . . 6.4-58

Table 6.4-15 Applicability of Tornado and Extreme Winter Precipitation Events to Safety-Significant SSCs . . . . . 6.4-59

Table 6.4-16 Summary of Design Transients . . . . . 6.4-60

Table 6.4-17 Applicable Loads for ASME Code Class A and B Components . . . . . 6.4-61

Table 6.4-18 Load Combinations and Service Stress Limits for ASME Code Class A and B Components . . . . . 6.4-62



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# Chapter 6 Safety-Significant SSC Criteria and Capabilities

## 6.1 Design Requirements for Safety-Related SSCs

Section 6.1 describes design requirements for safety-related (SR) structures, systems, or components (SSCs). Section 6.1.1 describes the DBHLs and their bases. Section 6.1.2 describes a summary of the Safety Related Design Criteria (SRDC). Section 6.1.3 describes design-basis hazard level (DBHL)-related requirements for non-safety-related SSCs.

### 6.1.1 Design Basis Hazard Levels

DBHLs are a design specification of the level of severity or intensity of hazards for which the SR SSCs are designed to withstand with no adverse impact on their capability to perform their SR functions. The DBHLs and their bases are summarized in this section and Table 6.1-1. The methodologies used to transform DBHLs into loads for design of SR structures are described in Section 6.4.2. Internal hazards will be evaluated at the operating license (OL) stage.

DBHL-related requirements may be applied to non-safety-related with special treatment (NSRST) and non-safety-related with no special treatment (NST) SSCs to prevent interactions with SR SSCs that would adversely impact the ability of an SR SSC to perform an SR function as described in Section 6.1.3.

Special treatments for protection from hazards are applied to NSRST SSCs in accordance with the safety significance of the SSCs. NSRST SSCs may be protected from hazards at a different level of severity or intensity than the DBHL because the NSRST SSCs do not perform SR functions. The application of special treatments is described in Section 6.3. Hazard protection for SSCs is described in Chapter 7.

#### 6.1.1.1 Wind and Tornado

The design basis wind and tornado hazard parameters and their bases are described in Table 2.1-1.

#### 6.1.1.2 External Flood

The design basis external flood hazard parameters and their bases are described in Table 2.1-1.

SR structures are located below grade and are damp-proofed and waterproofed to minimize potential groundwater intrusion. To prevent external flooding from entering an NSRST structure and propagating to an SR structure, top of concrete elevation at NSRST structures is approximately 6,758 feet (2,060 meters), which is above the design basis flood level described in Table 2.1-1. Additionally, NSRST structure exterior doors are located above the ponding level due to local intense precipitation and the site is graded away from the exterior doors of NSRST structures. Therefore, the design basis external flood hazards cannot flood SR SSCs and no specific flood protection is required.


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6.1.1.3 Internal Flood

SR SSCs are designed to withstand the design basis internal flooding with no adverse impact on their capability to perform their SR functions.

The design basis internal flooding analysis assumes, for each analyzed area, the single, worst-case rupture of a non-safety-related sodium, molten salt, or water tank or pipe that has no seismic special treatment or is not protected from the design basis tornado hazard. The worst-case pipe rupture is assumed to be a circumferential break.

The Energy Island (EI) and Nuclear Island (NI) are physically separated and the NI is at a higher elevation than the EI, so flooding from fluid systems in the EI does not affect SSCs in the NI.

6.1.1.4 Missile Protection

6.1.1.4.1 Missile Selection

The design basis missile sources and their bases are described in this section.

- Pressurized systems and components. The design basis missiles from pressurized systems and components, including tanks and cylinders containing compressed gas, are determined by probability analysis. If the probability of a missile occurring, impacting an SR target, and resulting in adverse impact to the ability of an SR target to perform an SR function is less than $1x10^{-7}$ per year, the missile is not considered a design basis missile.
- Turbine missiles. Turbine missiles are considered as a part of the site evaluation as described in Table 2.1-1.
- Rotating equipment. Missiles from rotating equipment will be considered.
- Secondary missiles generated as a result of primary missile impacts. Secondary missiles are considered in the evaluation of the impacts of the design basis missiles.
- Wind and tornado generated missiles. The design basis wind and tornado missiles and their bases are described in Table 2.1-1 and Section 6.1.1.1.
- Site proximity missiles (except aircraft). Site proximity missiles are considered as a part of the site evaluation as described in Table 2.1-1.

6.1.1.4.2 Missile Protection

SR SSCs are protected from, or designed to withstand impacts from, a single design basis missile without loss of capability to perform their SR functions. Design basis missiles are addressed by one of the following methods:

1. Physically separating or orienting potential design basis missile sources such that missiles will not impact SR SSCs.

2. Demonstrating that the missile impact does not adversely impact the ability of SR SSCs to perform their SR functions.


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3. Providing missile barriers. If missile barriers are used, missile barrier design procedures will be provided at the OL stage.

### 6.1.1.5 Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping

SR SSCs are designed such that the design basis pipe ruptures will have no adverse impact on their capability to perform their SR functions. The design basis pipe ruptures are addressed by one of the following methods:

1. Piping systems subject to postulated design basis pipe ruptures are physically separated from SR SSCs where feasible.

2. Providing guard piping, barriers, or fluid collection systems to mitigate the effects of the design basis pipe ruptures on SR SSCs.

3. Demonstrating that the design basis pipe rupture does not adversely impact the ability of SR SSCs to perform their SR functions.

The NI structures contain water, molten salt, and sodium piping systems. The EI and NI are physically separated and the NI is at a higher elevation than the EI, so pipe ruptures in fluid systems in the EI do not affect SSCs in the NI.

Moderate energy water piping systems are defined as systems that, during normal operating conditions, are either in operation or maintained pressurized where both of the following temperature-pressure conditions are met:

1. The maximum operating temperature is 200 degrees Fahrenheit (95 degrees Celsius) or less.

2. The maximum operating pressure is 275 pounds per square inch gauge (1900 kilopascals) or less.

Moderate energy molten salt piping systems are defined as systems that, during normal operating conditions, are either in operation or maintained pressurized where both of the following temperature-pressure conditions are met:

1. The maximum operating temperature is less than the boiling temperature of the molten salt at the site elevation.

2. The maximum operating pressure is 275 pounds per square inch gauge (1900 kilopascals) or less.

Moderate energy sodium piping systems are defined as systems that, during normal operating conditions, are either in operation or maintained pressurized where both of the following temperature-pressure conditions are met:

1. The maximum operating temperature is less than the boiling temperature of the sodium at the site elevation.


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2. The maximum operating pressure is 275 pounds per square inch gauge (1900 kilopascals) or less.

High energy piping systems are piping systems that exceed one or both of the moderate energy temperature-pressure conditions during normal operating conditions. Piping systems that exceed the moderate energy temperature-pressure conditions for about two percent or less of the time during which the system is in operation are considered moderate energy.

High energy liquid piping systems are not planned to be installed in safety-significant structures.

#### 6.1.1.6 Seismic Hazard

The design basis earthquake is the safe shutdown earthquake (SSE). The SSE and the basis for the SSE are described in Table 2.1-1. SR SSCs are designed to withstand the SSE with no adverse impact on their capability to perform their SR functions. Seismic analysis and design are described in Section 6.4.1.

#### 6.1.1.7 Extreme Winter Precipitation

The design basis extreme winter precipitation hazard parameters and their bases are described in Table 2.1-1.

#### 6.1.1.8 Internal Fire

The design basis internal fire hazard will be evaluated at the OL stage.

### 6.1.2 Summary of SRDC

The SRDC are design criteria that are necessary to fulfill the Required Functional Design Criteria for those SSCs selected to perform SR functions. The Required Functional Design Criteria are defined and developed in accordance with NEI 18-04 (Reference 6.1-1) as endorsed by Regulatory Guide 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," Revision 0. The Required Functional Design Criteria are compared to the Principal Design Criteria (PDC) in Section 5.3 to confirm that the PDC are complete for the design. The SRDC include parameters that are the subject of capability targets for a given SSC. The SRDC are incorporated into the design of SSCs described in Chapter 7. The SR functions are described in Section 5.2. The SRDC are provided in Tables 5.2-1, 5.2-2, and 5.2-3 to show their connection to the SR functions. Capability targets are discussed in Section 6.2.


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### 6.1.3 Summary of DBHL-Related Requirements for Non-Safety-Related SSCs

NSRST and NST SSCs are designed such that the DBHLs will not cause an NSRST or NST SSC to adversely impact the ability of an SR SSC to perform an SR function. DBHL-related interactions between NSRST or NST SSCs and SR SSCs are prevented by at least one of the following criteria:

1. The failure of the NSRST or NST SSC does not cause the NSRST or NST structure to strike an SR SSC.

2. The failure of the NSRST or NST SSC does not adversely impact the ability of an SR SSC to perform an SR function.

3. The NSRST or NST SSC is analyzed and designed to prevent its failure under DBHL conditions. Design requirements implemented for the prevention of DBHL-related interactions are described in Section 6.1.3.1 and Section 6.1.3.2.

#### 6.1.3.1 Seismic Interaction

NSRST and NST SSCs designed to prevent seismic DBHL-related interactions are designed in accordance with ASCE 7-16 (Reference 6.1-2). Design evaluations demonstrate that the NSRST or NST structure will withstand the SSE with an acceptance criterion of moderate inelastic deformation of the structure's primary seismic force-resisting system. The design evaluations apply response parameters that correspond to ASCE 43-19 (Reference 6.1-3) Limit State B.

In addition to an acceptance criterion of moderate inelastic deformation, design evaluations demonstrate that seismic separation between an NSRST or NST structure and adjacent SR SSCs will exceed the required seismic separation for the SSE. The methods of ASCE 43-19 Section 7.3 are applied for evaluation of seismic separation. Per ASCE 43-19 Section 7.3, elastically computed displacements are increased by a factor to account for the inelastic response. Construction tolerances are considered when determining the required minimum as-designed seismic separation.

#### 6.1.3.2 Tornado Interaction

NSRST and NST SSCs designed to prevent tornado DBHL-related interactions are designed in accordance with the wind loads and provisions of ASCE 7-16 using the design basis tornado hazard wind speed, atmospheric pressure drop, and tornado missiles identified in Table 2.1-1.


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## References

6.1-1 NEI Technical Report 18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” [ADAMS Accession No. ML19241A472], August 2019.

6.1-2 American Society of Civil Engineers, Minimum Design Loads and Associated Criteria for Buildings and Other Structures, ASCE 7-16.

6.1-3 American Society of Civil Engineers, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, ASCE 43-19.


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**Table 6.1-1 Design Basis Hazard Levels**

| Hazard | Design Basis Hazard Level | Summary of Basis for Parameters |
|---|---|---|
| Wind Loadings | See Section 2.1 | See Section 2.1 |
| Tornadoes | See Section 2.1 | See Section 2.1 |
| External Flooding | See Section 2.1 | See Section 2.1 |
| Internal Flooding | See Section 6.1.1.3 | See Section 6.1.1.3 |
| Missiles | See Section 6.1.1.4 | See Section 6.1.1.4 |
| Rupture of Piping | See Section 6.1.1.5 | See Section 6.1.1.5 |
| Seismic | See Section 6.1.1.6 | See Section 6.1.1.6 |
| Extreme Winter Precipitation | See Section 2.1 | See Section 2.1 |
| Internal Fire | See Section 6.1.1.8 | See Section 6.1.1.8 |



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## 6.2 Reliability and Capability Targets for Safety-Significant SSCs

Assurance that a structure, system, or component (SSC) performs its probabilistic risk assessment (PRA) safety function is accomplished by achieving and monitoring the levels of reliability and capability of the SSC that are assessed in the PRA. The PRA is described in Section 3.1, licensing basis events are described in Sections 3.4 through 3.9, and PRA safety functions are described in Section 5.2.

A reliability target is a performance goal established for the probability that an SSC will complete its specified function to achieve plant-level risk and reliability goals. Quantitative reliability targets are the output of the Design Reliability Assurance Program, which is described in Section 8.0. The reliability targets for safety-related and non-safety-related with special treatment SSCs have not yet been confirmed. The Design Reliability Assurance Program will confirm, at the operating license (OL) stage, that the reliability performance targets assumed in the PRA have been met.

A capability target is a performance goal focused on the ability of an SSC to perform its PRA safety function with adequate margins. Lifetime degradation is considered as part of the capability.

Specific operational performance criteria are included as part of the profile of the success of a PRA safety function. These quantitative targets are based on analytical models used to demonstrate the capability of the plant to respond to hazards. The capability targets for safety-related and non-safety-related with special treatment SSCs have not yet been confirmed. The output of analytical models will confirm, at the OL stage, that the capability targets assumed in the PRA have been met.

For SSCs classified as safety-related or non-safety-related with special treatment, the process for using the reliability and capability to develop special treatment requirements is described in Section 6.3.


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## 6.3 Special Treatment Requirements for Safety-Significant SSCs

Special treatments are those requirements that provide increased assurance beyond normal industrial practices that safety-significant structures, systems, or components (SSCs) perform their design basis functions. Special treatments are developed to provide reasonable confidence that safety-significant SSCs meet the reliability and capability targets. The method for confirming specific reliability and capability targets for SSCs at the operating license (OL) stage is described in Section 6.2.

Special treatments applied to safety-significant SSCs are identified in Chapter 7. The special treatments identified in Chapter 7 are preliminary and are selected using a systematic process considering the PRA safety functions performed by the SSC and the safety-significance, risk significance, and equipment type of the SSC. After the reliability and capability targets are confirmed, special treatments are finalized and applied to safety-significant SSCs to provide reasonable confidence that the levels of reliability, availability, and capabilities of functions of the SSCs assessed in the probabilistic risk assessment, or determined to be necessary for defense-in-depth, are met. The probabilistic risk assessment is described in Section 3.1.

Some of the special treatments applied to safety-significant SSCs are related to seismic classification and plant programs. Seismic classifications and seismic special treatments are defined in Section 6.4.1. Plant programs that are identified as special treatments are summarized in Section 8.0. Once special treatments are finalized, plant programs identified in Section 8.0 will be assessed to see if any changes are warranted. Updated programs will be provided at the OL stage.


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# 6.4 Design of Safety-Significant SSCs

## 6.4.1 Seismic Design

Safety-significant structures, systems, or components (SSCs) are designed to withstand the effects of natural phenomena including earthquakes in accordance with Principal Design Criterion (PDC) 2. The seismic design bases for SSCs are established by their seismic classifications and applicable special treatments that are consistent with the safety classification. Seismic design and analysis methodology and parameters presented in this section are applicable to safety-related (SR) and non-safety-related with special treatment (NSRST) SSCs.

### 6.4.1.1 Seismic Classification of SSCs

Seismic classifications are defined to support graded application of seismic design requirements to SSCs and are initially assigned to SSCs based on their safety classification and the probabilistic risk assessment safety functions performed by the SSCs. Special treatments that establish SSC seismic design requirements are applied to SSCs based on their seismic classification. The seismic classifications are SCS1 for SR seismic risk significant SSCs, SCS2 for SR non-seismic risk significant SSCs, and SCN1, SCN2, and SCN3 for NSRST SSCs based on seismic risk significance.

SR SSCs are designed to withstand the effects of the safe shutdown earthquake (SSE) and remain functional, consistent with the requirements for Seismic Category I SSCs described in Regulatory Guide (RG) 1.29 Revision 6, “Seismic Design Classification for Nuclear Power Plants”. SR SSCs are assigned a seismic classification of SCS1 or SCS2 to provide graded levels of seismic special treatment based on seismic risk significance. SCS2 imposes reduced seismic qualification requirements relative to SCS1 commensurate with the seismic risk significance of the SSC function.

NSRST SSCs are designed using commercial standards with additional seismic special treatments applied to provide confidence of functional performance following the SSE. Seismic special treatments for NSRST SSCs are developed based on application of ASCE/SEI 7-16 (Reference 6.4-2) to develop seismic demands and design requirements using the SSE as the maximum credible earthquake.

SCN1, SCN2, and SCN3 seismic classifications are assigned to NSRST SSCs to provide graded levels of seismic special treatment and associated seismic performance. SCN1 and SCN2 impose seismic qualification requirements and SCN1 is initially assigned to components that perform an NSRST function after the SSE. SCN3 is initially assigned to components that are required to maintain structural integrity after the SSE. SCN2 is not used in initial assignments.

Seismic classifications and the application of seismic special treatments to SSCs are developed with the consideration of the SSC contribution to seismic risk.


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6.4.1.2 Seismic Design Basis

6.4.1.2.1 Design Ground Motion for SR and NSRST SSCs

The seismic design is based on the SSE and the Operating Basis Earthquake (OBE) ground motion magnitudes. The SSE design response spectra are described in Section 2.6.2. The OBE ground motion magnitude is set at one-third that of the SSE. SR SSCs are designed to withstand the effects of the seismic Design Basis Hazard Level without loss of capability to perform their SR functions. Section 6.1.1 provides information related to the Design Basis Hazard Level.

6.4.1.2.2 Development of FIRS

Separate Foundation Input Response Spectra (FIRS) are developed for the Reactor Building (RXB), the Fuel Handling Building (FHB), and the Nuclear Island (NI) Control Building (NCB). The full column method is used to develop the FIRS because the structures of interest have embedded foundations. The full column method includes the effects from the soil layers above the foundation level in the respective site response analysis. FIRS are developed as free-field outcrop motions at the respective foundation level using methods consistent with the development of the SSE. Section 2.6.2 provides information related to how the SSE is developed. Vertical-to-horizontal ratios, developed using methods consistent with the development of the SSE vertical spectrum, are used to calculate the vertical spectra representing the FIRS of the three different structures.

6.4.1.2.3 Development of Maximum Considered Earthquake Ground Motion

The determination of the seismic response spectra for NSRST SSCs is conducted in accordance with ASCE/SEI 7-16. The response spectra for SCN1, SCN2, and SCN3 use the SSE or SSE-based response motion in place of Maximum Considered Earthquake Ground Motion and the applicable Seismic Risk Category as described in Table 6.4-1.

6.4.1.3 Seismic Design and Analysis of SR SSCs

6.4.1.3.1 Seismic Analysis Methods

The seismic analysis methods applied to SR SSCs are based on use of either a dynamic analysis method or an equivalent static load analysis method. Analysis methods incorporating inelastic and nonlinear considerations are used based on the specific systems being evaluated.

Dynamic Analysis Methods

For the calculation of seismic responses of SSCs, three options for dynamic analysis methods are used:

- Response Spectrum Analysis Method,
- Time History Analysis Method, or
- SSI analysis using the computer program System for Analysis of Soil Structure Interaction with the Random Vibration Theory based method and the complex frequency response analysis method.


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Response Spectrum Analysis Method

For the response spectrum modal analysis of SSCs, significant modes are captured in the total mass of the structure in each direction. The modal responses are combined consistent with RG 1.92 Revision 3, “Combining Modal Responses and Spatial Components in Seismic Response Analysis”.

Time History Analysis Method

The time history analysis method is based on guidance from RG 1.92. The representative maximum response of interest of the SSC is obtained either by performing separate analysis for each of the three components of earthquake motion, or by performing a single analysis with all three components of earthquake motion applied simultaneously.

Random Vibration Theory

The Random Vibration Theory-based approach, described in ASCE/SEI 4-16 (Reference 6.4-1), uses an acceleration response spectra as input, and is used for Soil-Structure Interaction (SSI) analysis. Time histories for subsystem analysis are generated from the in-structure response spectra as needed.

Dynamic Analysis Considerations

The following items are considered in the dynamic analysis:

- For SSI analysis of SR structures, the seismic analysis is performed using three dimensional models of the structures with input motion consisting of two horizontal and one vertical components.
- Torsional, rocking, and translational responses of structures and their foundations are evaluated, and if not negligible, are used as input in the analysis.
- The number of discrete mass degrees of freedom in dynamic modeling is used in accordance with ASCE/SEI 4-16.
- The frequency cutoff meets the requirements outlined in ASCE/SEI 4-16.
- For SR distributed systems with multiple support locations within the structure, seismic anchor motions, in terms of maximum relative displacement of the SSC support points with respect to one of the support points, are computed.
- Modeling of structures is performed in accordance with ASCE/SEI 4-16.

Equivalent Static Load Method

An equivalent static load method in accordance with ASCE/SEI 4-16 for seismic analysis of SR SSCs is performed if:


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- Justification is provided that the SSC can be realistically represented by a simple model and the method produces conservative results in terms of responses. In some cases, typical examples or published results for similar structures are considered acceptable in support of the use of the simplified method if it is demonstrated that the example has similar configuration, structural system, and dynamic properties as the SSC of interest.
- The equivalent static analysis method accounts for the relative motion between points of support.
- A dynamic amplification factor, α, is applied to the peak spectral acceleration of the applicable ground or floor response spectrum to obtain an equivalent static load for the SSC that can be represented by a simple model. The dynamic amplification factor in ASCE/SEI 4-16, Section 4.5.1, is set equal to:
    - for systems with single supports, and
    - for systems with multiple supports, with 1.0 applied for the individual supports.

Soil-Structure Interaction

SSI analyses are performed for the RXB, FHB, and NCB substructures, each with a minimum of three site profiles covering the range of soil and rock properties.

SSI analyses are performed on linear models that assume isotropic elastic material properties for the soil and structure. Unless it can be justified that fixed-base analysis is applicable, SSI analysis methods in Chapter 5 of ASCE/SEI 4-16 are used.

Due to embedment of the SR building structures, the method selected for SSI analysis considers the effects of site ground motion amplification within the embedment depth. In addition, the methodology considers the impedance and flexibility of the embedded foundation in terms of foundation stiffness and foundation damping in the analysis. The method also provides all required seismic responses needed for structural design.

SSSI Effects

Structure-Soil-Structure Interaction (SSSI) effects are evaluated and addressed in accordance with the guidance from ASCE/SEI 4-16 Chapter 5.1.5 for building structures in close proximity.

SSSI effects between the RXB substructure and the adjacent Reactor Auxiliary Building (RAB) structure are assessed. An SSI analysis containing the RXB structural model and a simplified RAB structural model is used to evaluate the significance of SSSI effects on member forces and ISRS. SSSI effects are incorporated into the foundation input motion development, and considered for the design and analysis of the RXB and RAB structures, if determined to be significant.

SSSI effects between the RXB substructure and the adjacent RXB superstructure foundation are accounted for in the SSI analysis. The SSI analysis includes the RXB substructure model and RXB superstructure foundation model and provides seismic responses for structural design. Similarly, SSSI effects between the FHB substructure and the separated north wing structure are accounted for in the SSI analysis, which includes the FHB substructure and north wing structural models.


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Three Components of Earthquake Motion

The methods for combining the responses due to three spatial components of design ground motion, for both the response spectrum method and the time history method are consistent with the methods provided in RG 1.92.

Supporting Media for SR Substructures

SR substructures are embedded below the ground surface. The seismic SSI responses of the embedded structures are influenced by kinematic effects, with added contribution from inertial effects. The site profiles have a controlling effect on the kinematic effects.

For each SR substructure, the site profile representing the building location, including the excavation of the in-situ layers replaced by engineered backfill, is modeled in the SSI model. In order to consider the range of soil and rock properties in the analysis, three or more site profiles are used to cover the range of lower bound, best estimate, and upper bound properties. The site profiles are developed from the geotechnical data collected at the site as described in Section 2.6.4.

Damping Values in Dynamic Analysis

The damping values used in the dynamic analysis of SR SSCs, expressed as a percentage of critical damping, are consistent with the guidance in RG 1.61.

For SR structures, the damping for uncracked concrete properties is consistent with Table 2 of RG 1.61 Revision 1, “Damping Values for Seismic Design of Nuclear Power Plants” (i.e., OBE damping values) and damping for cracked concrete properties is consistent with Table 1 of RG 1.61 (i.e., SSE damping values). The material damping for the site-specific soil and rock profiles are developed from the geotechnical data collected at the site and the maximum damping values for the conduct of site response analyses are limited to 15 percent of critical damping as described in Section 2.6.4.

Analysis Procedures for Damping

For structural stiffness and damping, uncracked properties are used. The damping values for uncracked properties are consistent with Table 2 of RG 1.61. If the stress results from analysis show the cracking condition for the main load carrying concrete walls and slabs, the analysis is revised using cracked stiffness and damping properties.

6.4.1.3.2 Procedures Used for Analytical Modeling

The provisions of Chapters 3, 4, and 5 of ASCE/SEI 4-16 are followed for seismic modeling and linear seismic analysis of SR structures considering the structural layout and embedment condition of the structures.

Stiffness properties of the SR structures are modeled following the recommendations of ASCE/SEI 4-16, Section 3.3. An effective stiffness is assigned to the reinforced concrete members following the provisions of ASCE/SEI 4-16, Section 3.3 and Table 3-2.


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Mass inertia properties assigned to the dynamic models meet the requirements set in ASCE/SEI 43-19 and the provisions of ASCE/SEI 4-16, Section 3.4.2 are used for determination of the dynamic mass. The mass is discretized in the dynamic model according to the requirements of ASCE/SEI 4-16, Section 3.4.1.

Three-dimensional finite element models and the discretization of the models for the seismic response analyses of SR structures are developed following the applicable provisions of ASCE/SEI 4-16, Section 3.1.3 and Section 3.8. The one-step seismic response analyses are performed on refined finite element models that meet the requirements of ASCE/SEI 4-16, Section 3.1.2.2. Less refined models that meet the requirements of ASCE/SEI 4-16, Section 3.1.2.1(a), are used to calculate seismic demands for a two-step analysis approach that is followed with a seismic analysis using a refined finite element model to obtain seismic responses. Lumped-mass stick models may be used provided that the provisions of ASCE/SEI 4-16, Section 3.8.1.3 are implemented.

For assessment of the effect of nearby NSRST structures on SR structures and development of input motion for NSRST structures supported directly by SR structures, the same provisions used for modeling the SR structures are used.

Multiply-Supported Equipment and Components with Distinct Inputs

For the effects of dynamic coupling of the primary SR structure with major equipment, systems, and components, the recommendation of Section 3.7 of ASCE/SEI 4-16 is used.

The seismic anchor motion is the horizontal and vertical movement, horizontal movement only, or vertical movement only of the structural support for a system or component due to differential movement from a seismic event. The seismic anchor motion is considered in the design of commodities and supports that span multiple levels.

The reactor is seismically isolated from the RXB substructure as described in Section 7.1.2. This seismic isolation boundary represents a specific design consideration for umbilicals, which are distributed systems that are supported on both the reactor and RXB substructure. The relative displacements between the reactor and the RXB substructure that result from the displacement of the seismic isolation system are considered in the analysis of the umbilicals and are accounted for through the application of seismic anchor motions.

#### 6.4.1.3.3 Combination of Individual Modal Responses

The combination of modal responses is consistent with RG 1.92 using the square root of the sum of squares methodology or complete quadratic combination method. For closely spaced modes and high-frequency modes, the complete quadratic combination method is used.


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6.4.1.3.4 Natural Frequencies and Responses

The SR structures are embedded below the finished grade with multiple foundations at depth. The fixed base natural frequencies of these structures, ignoring the support of soil and rock, are not considered realistic; therefore, the SSI results that consider the geometry and the embedment of the structures are used to show the natural frequencies and response of the structures.

6.4.1.3.5 Development of In-Structure Floor Response Spectra

Effects of Parameter Variations on Floor Responses

The In-Structure Response Spectra (ISRS) are generated from seismic analysis for equipment design and qualification. The analysis is performed in accordance with the requirements of Section 6.2 of ASCE/SEI 4-16. The maximum frequency for the ISRS calculation is set at 100 hertz (Hz) with frequency intervals of 100 frequency points uniformly distributed over logarithmic cycles of 0.10 Hz to 1.0 Hz, 1.0 Hz to 10 Hz, and 10 Hz to 100 Hz. The ISRS calculation is consistent with RG 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” Revision 1.

In computing the ISRS results, the following steps are used to obtain the ISRS for subsystem analysis:

- Determine the required spectral damping values for the ISRS for subsystem analysis. ISRS results are calculated for the same set of spectral damping values and frequencies.
- Select the locations in the structure and the corresponding nodes in the SSI model where ISRS are required. The node set is selected to ensure the final ISRS bounds the full range of responses (dominant direction responses, rigid responses, as well as the local flexural responses) and is applicable for subsequent analysis of the SSC supported by the selected nodes.
- Perform seismic analyses for every soil case and input motion in the three directions of x, y, and z.
- Combine the responses in X, Y, Z response directions due to x, y, z input motion using square root of the sum of squares of the ISRS, consistent with Section 6.2.1.2 of ASCE/SEI 4-16. For each soil case, the co-directional responses are combined into three ISRS in the main directions.
- Envelope the square root of the sum of squares combined ISRS for the respective soil cases for each analysis case. If there are gaps between the peaks of the acceleration response spectra among the soil cases caused by the soil SSI frequency, the gaps are filled to cover soil cases between the soil cases selected for SSI analysis.
- Smooth and broaden the ISRS curves to obtain the final ISRS results.


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Time History Generation for Subsystem Analysis

Time histories are generated for subsystem analysis from the computed ISRS. For a given subsystem in-structure location, the enveloping ISRS for the three directional components of motion are the target spectra for time history generation. Five sets of 3-component time histories for subsystem analysis are generated in accordance with ASCE/SEI 4-16. Criteria and guidance from ASCE/SEI 43-19 are applied to verify the adequacy of the resulting time histories.

#### 6.4.1.3.6 Methods Used to Account for Torsional Effects

Accidental torsion is the torsion resulting from deviations in design, construction and loading and is accounted for in the structural analysis and design of SR structures along with other seismic loading conditions.

The actual eccentricity is considered by the three-dimensional structural model that includes the distribution of mass and stiffness in the structure. Accidental torsional effects are determined in accordance with provisions in Section 3.1 and Section 4.5.1 of ASCE/SEI 4-16. For embedded SR structures, a simplified approach is used which increases the shear forces in the shear walls and other lateral load carrying members by a factor that amounts to the torsional moment. An increase of 5 percent in seismic shear forces is considered for design. The effect of shear force from accidental torsional moment is not used to reduce the in-plane shear forces.

#### 6.4.1.3.7 Seismic Stability

Seismic stability for SR SSCs is evaluated per ASCE/SEI 43-19 (Reference 6.4-3). Based on the recommendation in Section 7.2.1 of ASCE/SEI 43-19, once it is shown that the center of gravity of a structure is below the ground surface, an assessment of the stability is not required.

### 6.4.1.4 Modeling and Analysis of Underground SR Structures Directly Supporting NSRST Superstructures

A coupled dynamic analysis is performed for each underground SR structure that directly supports an NSRST superstructure. The NSRST superstructure is represented with a lumped mass model and connected to a detailed model of the underground SR structure. A sensitivity study is performed to confirm that the superstructure lumped mass model adequately represents the major coupling effects between the underground SR structure and NSRST superstructure.

### 6.4.1.5 Interaction of NSRST and NST SSCs with SR SSCs

Interaction of NSRST and non-safety-related with no special treatment (NST) SSCs with SR SSCs is described in Section 6.1.3.

### 6.4.1.6 Seismic Analysis Methods and Procedures Used for Analytical Modeling for NSRST SSCs

NSRST SSCs are designed using commercial codes. The seismic analysis method used is ASCE/SEI 7-16. For NSRST SSCs, the maximum considered earthquake is the SSE, adjusted based on seismic risk category and importance factor.


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Damping

NSRST SSCs analyses use 5 percent of critical damping as discussed in Section C11.4.2 in ASCE/SEI 7-16.

Seismic Load Effect and Combinations

The seismic load effect in the horizontal and vertical directions is determined in accordance with Section 12.4.2 of ASCE/SEI 7-16. The load combinations for seismic design consider the strength design, including overstrength and allowable stress design, that are in accordance with ASCE/SEI 7-16 Sections 2.3.6 and 2.4.5, and with Chapters 12, 13, and 15.

Direction of Loading

The design seismic forces are applied independently in each of two orthogonal horizontal directions, and the orthogonal interaction effects are neglected consistent with Section 12.5 of ASCE/SEI 7-16.

Equivalent Lateral Force Procedure

Seismic base shear and seismic response coefficients are calculated in accordance with Section 12.8 of ASCE/SEI 7-16.

Modal Response Spectrum Analysis

For the structural modal analysis, the number of modes is chosen to have a combined mass participation factor of 100 percent. Alternatively, the analysis is permitted to include a minimum number of modes to obtain a combined modal mass participation of at least 90 percent of the actual mass in each orthogonal horizontal direction of response considered in the model.

Modal Response Parameters

In accordance with ASCE/SEI 7-16, the design response spectra is divided by the $R/I_e$ factor for force-related design parameters. The displacements are computed using the response spectrum modified with the $R/I_e$ factor and are multiplied by a $C_d/I_e$ factor to produce the expected inelastic displacements.

Where:

- $C_d$ is the deflection amplification factor as defined in Tables 12.2-1, 15.4-1 and 15.4-2 of ASCE/SEI 7-16
- $I_e$ is the seismic importance factor
- R is the response modification coefficient as defined in Tables 12.2-1, 12.14-1, 15.4-1, and 15.4-2 of ASCE/SEI 7-16


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Combined Response Parameters

The seismic analysis uses the square root of the sum of squares or complete quadratic combination method for combining the response parameters; the complete quadratic combination method is used for closely spaced modes.

Scaling Design Values of Combined Response

If the combined response for the modal base shear ($V_t$) is less than 100 percent of the calculated base shear (V), the equivalent lateral force procedure is used and the forces are multiplied by $V/V_t$. If the combined response for the modal base shear ($V_t$) is less than $C_s W$, drifts are multiplied by $(C_s W)/V_t$.

Where:

- $C_s$ is the seismic response coefficient defined Section 12.8.1.1.or 19.3.1 of ASCE/SEI 7-16
- W is the effective seismic weight defined in Sections 19.2 and 19.3 of ASCE/SEI 7-16

Methods Used to Account for Torsional Effect

For the determination of accidental torsion in NSRST structures, the provisions of Sections 12.8.4.2 and 12.9.2.2.2 of ASCE/SEI 7-16 are used in the analysis. Amplification of accidental torsion is considered in accordance with Section 12.8.4.3 of ASCE/SEI 7-16 when the equivalent lateral force procedure is used.

Generation of Loads for Embedded NSRST Structures

Design and analysis procedures for the NSRST RAB substructure is discussed in Section 7.8.3.1.3.

### 6.4.2 Design of Safety-Significant Structures

The safety-significant structures, NCB, RXB, FHB, and RAB, consist of an embedded reinforced concrete substructure located below the finished plant grade elevation and a steel-framed superstructure located above the finished plant grade elevation. The site layout of the safety-significant structures is shown on Figure 1.2-1.

The RXB, FHB, and NCB substructures are classified as SR structures and the safety classification of the RAB substructure is NSRST. The FHB, NCB, and RAB superstructures are classified as NSRST. The RXB superstructure is NST.

#### 6.4.2.1 Applicable Codes, Standards, Specifications, and Regulatory Guidance

The codes, standards, and specifications applicable for the design, fabrication, construction, testing and inspection of safety-significant structures are listed in Table 6.4-2. The primary design codes for the SR substructures are ACI 349-13 (Reference 6.4-4) for concrete design and ANSI/AISC N690-18 (Reference 6.4-5) for structural steel design. The primary design codes


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for the NSRST RAB substructure are ACI 318-19 (Reference 6.4-6) for concrete design and ANSI/AISC 360-16 (Reference 6.4-7) for structural steel design. The primary design codes for the NSRST superstructures are ACI 318-19 (Reference 6.4-8) for concrete design and ANSI/AISC 360-16 for structural steel design.

The primary design codes and standards for SR and NSRST structures are identified in Table 6.4-3.

Applicable regulatory guidance documents are listed in Section 1.4.1. The RGs applicable to the design and construction of safety-significant structures are listed in Table 6.4-4.

6.4.2.2 Loads and Loading Combinations

Safety-significant structures are designed and analyzed for appropriate combinations of design loads and loading combinations, including normal loads, severe environmental loads, extreme environmental loads, and abnormal loads. The applied design loads and associated symbols are described in the following sections.

6.4.2.2.1 Normal Loads

Dead Loads (D)

Dead loads include the self-weight of the safety-significant structure, such as the walls, roof, slab, steel beams, and columns, including permanent plant equipment, and mechanical and electrical distribution systems. Dead loads are determined in accordance with ASCE 7-16 (Reference 6.4-2). The normal weight of reinforced concrete and steel is based on a design density of 150 pounds per cubic foot and 490 pounds per cubic foot, respectively. In addition, significant loads associated with support loads of distribution systems, such as piping, cable trays, and NI Heating, Ventilation, and Air Conditioning System components, are also included as dead loads.

A uniform dead load of 20 pounds per square foot (psf) is applied to the concrete walls to represent penetration sleeves or wall-supported systems and components. Contained liquid (e.g., tanks, Spent Fuel Pool) is treated as a dead load, as well as the weight of NI Cranes and Hoists crane trolleys and bridges.

Live Loads (L)

Live loads are loads produced by the use and occupancy of the building that does not include construction loads or environmental loads, such as wind, snow, rain, earthquake, flood, or dead load. Live load refers to a normal load that may vary with intensity or point of application. Live loads include moveable equipment or equipment that is likely to be moved as a non-permanent weight.

For floor beams, girders, and slabs at roof and elevated floor levels, a 4,000 pound-force concentrated load is used in locations that maximize moment and shear without carrying out to other members.


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For equipment supports, live loads also consider contributory loads due to associated equipment operation and movement. Live loads are applied, removed, or shifted in location as necessary to obtain the worst-case loading conditions.

Floor area live loads are omitted from areas occupied by permanent plant equipment whose weights are specifically included in the design dead loads except where access is available. Live loads associated with elevators and machinery are increased from 20 percent to 100 percent in accordance with ASCE 7-16 to account for impact.

Temporary equipment laydown loads are evaluated, and the maximum of the actual laydown load is utilized. Uniform live loads are the maximum loads expected by the intended use.

Roof Live Loads ($L_r$)

A uniform load of 20 psf is used for the roof live load of the NSRST superstructures, with allowed reductions in accordance with ASCE 7-16.

Snow Loads (S)

The NSRST superstructures are designed for a ground snow load of 35 psf. Roofs are evaluated to confirm they possess adequate stiffness to preclude progressive deflection as meltwater is created from the snow load or from rainfall on the surface.

Ice Loads (I)

Ice loading for NSRST superstructure are considered as 0.25-inch nominal ice thickness with 50 miles per hour concurrent wind speed in accordance with ASCE 7-16 with an ice density not less than 56 pounds per cubic foot for the NSRST superstructures.

Soil Pressure Loads (H)

Soil lateral earth pressure loads resulting from soil bearing pressure and hydrostatic pressure are applied to exterior walls up to the specified finished grade elevation of safety-significant substructures. Lateral earth pressure is based upon the appropriate lateral earth pressure coefficient, soil density, and includes the effects of groundwater. Submerged soil pressures include the effect of the buoyant soil plus the hydrostatic loads due to normal ground water or flooding.

Vertical pressure due to buoyancy effects from design groundwater level on the underside of the foundation are also considered. During construction, the effects by crane and vehicle surcharge loads are added to the lateral soil pressure. Dynamic seismic-induced soil pressures on embedded below-grade walls are calculated directly as part of the seismic analysis of the structures and included as a component of SSE loads for SR substructures.


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Rain Loads (R)

Snow loads for NSRST superstructures are considered to bound the rain loads for NSRST superstructures. NSRST superstructure roofs are evaluated to confirm they possess adequate stiffness to preclude progressive deflection as meltwater is created from the snow load or from rainfall on the surface.

Buoyant Forces (F')

A groundwater level of 15 feet below the finished plant grade is used for structural stability checks. Soil friction resisting buoyancy is used for the evaluation of embedded structures. The normal force due to soil pressure is calculated using soil properties as described in Section 2.6.4. Soil friction is not used when waterproofing is provided on the exterior surface of the below grade substructures.

Hydrostatic Loads (F)

Hydrostatic loads are the result of fluid pressures acting on the internal surface of compartments, such as the Spent Fuel Pool and liquid radioactive water processing tanks. The compartments may have constant or fluctuating liquid levels.

Loads resulting from groundwater pressure based on the groundwater and external flooding are included as part of the soil pressure loads. The external flooding from the design basis flood does not impact the NI buildings as described in Section 2.5.1.1. Impulsive hydrodynamic loads resulting from the seismic excitation of fluids (e.g., Spent Fuel Pool) are included as a component of the SSE load.

Thermal Loads ($T_o$)

Thermal loads consist of thermally induced forces and moments resulting from normal plant operation, start-up, shutdown, or environmental conditions. Thermal loads due to axial restraints and thermal gradients as well as self-relieving nature of thermal loads are considered.

Pipe Reaction Loads ($R_o$)

Pipe reactions coming from piping system supports during normal operating, start-up, or shutdown conditions are based on the critical transient or steady-state condition, excluding dead load and earthquake reactions. Appropriate dynamic load factors are used when applying transient loads.

Crane Loads ($C_{cr}$)

The NI Cranes and Hoists crane self-weights are included as dead loads. The crane live loads are defined as the rated capacity of the crane and are used for the design of the runway's beams, connections, and crane supports. The crane live loads consider the maximum wheel loads including vibration forces as well as vertical impact, lateral, and longitudinal forces induced by a moving crane. Further details of the NI Cranes and Hoists are provided in Section 1.1.4.3.13.


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The maximum crane wheel loads are increased by 25 percent to consider the effects of impact, vibration, and unexpected force. The longitudinal and lateral forces are calculated as 10 percent and 20 percent of the sum of the rated capacity and applied on the traction surface of the crane girder. The crane test load is considered in the crane's support design.

6.4.2.2.2 Severe Environmental Loads

Operating Basis Earthquake ($E_o$)

The OBE is defined as one-third of the safe shutdown earthquake. Earthquake loads from the operating basis earthquake are evaluated in accordance with Appendix S of 10 CFR Part 50.

Wind Loads (W)

The basic wind speed is 110 miles per hour (180 kilometers per hour) for Risk Category III buildings (NCB) and is 115 miles per hour (190 kilometers per hour) for Risk Category IV (RAB, RXB, FHB) buildings in accordance with ASCE 7-16. Normal wind loading (not including tornado winds) is applied in accordance with ASCE 7-16 and wind load calculations follow procedures outlined in ASCE 7-16. Wind loads on components and cladding are also applied in accordance with ASCE 7-16.

6.4.2.2.3 Extreme Environmental Loads

Safe Shutdown Earthquake ($E_{ss}$)

The SSE for the evaluation of safety-significant structures is described in Section 2.6.2.

Tornado Wind Pressure ($W_{tw}$)

Tornado loads include tornado wind pressure, tornado-generated differential pressure, and tornado-generated missiles. The maximum wind speed used is 160 miles per hour (260 kilometers per hour) and tornado effects are combined with other loads as prescribed in applicable codes and regulatory guidance.

Tornado wind pressure parameters listed in Table 6.4-13 reflect geographic Region III as defined in Table 1 of RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1.

Tornado - Generated Differential Pressure ($W_{tp}$)

Tornado-generated differential pressure drop is 0.6 pounds per square inch (4 kilopascals) and the rate of pressure drop is 0.2 pounds per square inch per second (1 kilopascal per second) rate as defined in Table 1 of RG 1.76 for geographic Region III and as listed in Table 6.4-13.


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Tornado - Generated Missiles ($W_{tm}$)

The tornado-generated missile parameters reflect geographic Region III as defined in Table 2 of RG 1.76 and as listed in Table 6.4-14. The automobile missile is considered to impact elevations less than 30 feet above all grade levels. Vertical velocities are taken as 0.67 of the corresponding horizontal velocities. Missiles are considered as striking in all directions.

Hurricanes

The site is located in Wyoming, is inland from coastal areas, and is located outside of the regions identified in RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” Revision 0. Due to the site location, there are no direct impacts from hurricane loading conditions at the site.

Extreme Winter Precipitation Event ($S_E$)

The NSRST superstructures are designed for the extreme winter precipitation load of 75 psf in accordance with Interim Staff Guidance DC/COL-ISG-007. The extreme winter precipitation load is considered as a ground snow load per ASCE 7-16.

6.4.2.2.4 Abnormal Loads

Abnormal loadings are considered as applicable for the following cases:

- Internal Flooding Loads ($F_a$)
- Accident Pressure Loads ($P_a$)
- Accident Thermal Loads ($T_a$)
- Accident Pipe Reaction Loads ($R_a$)
- Pipe Break Loads, subcategorized as:
    - Pipe Break Reaction Loads ($Y_r$)
    - Pipe Break Jet Impingement Loads ($Y_j$)
    - Pipe Break Missile Impact Loads ($Y_m$)

6.4.2.2.5 Loading Combinations

SR Substructure Loading Combinations

The loading combinations used for the design of concrete and structural steel of SR substructures, primarily based on ANSI/AISC N690-18 and ACI 349-13, are provided in Table 6.4-11 and Table 6.4-12.

NSRST Structure Loading Combinations

The loading combinations used for the design of the NSRST RAB substructure and the NSRST superstructures are in accordance with IBC-2021 load combinations for strength design or Load and Resistance Factor Design (Section 1605.1 of IBC-2021) as shown below:


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1.4 (D + F)

1.2 (D + F) + 1.6 (L + H) + 0.5 ($L_r$ or S or R)

1.2 (D + F) + 1.6 ($L_r$ or S or R) + 1.6 H + ($f_1$ L or 0.5 W)

1.2 (D + F) + 1.0 W + $f_1$ L + 1.6 H + 0.5 ($L_r$ or S or R)

1.2 (D + F) + 1.0E + $f_1$ L + 1.6 H + $f_2$ S

0.9 D + 1.0 W + 1.6 H

0.9 (D + F) + 1.0 E + 1.6 H

Where:

D – Dead Load

E – determined from ASCE 7-16 or $E_{ss}$ – Combined effect of horizontal and vertical earthquake induced forces as defined in Section 2.3.6 of ASCE 7-16

$E_{ss}$ – Safe Shutdown Earthquake

F – Load due to fluids with well-defined pressures and maximum heights

H – Load due to lateral earth pressures, ground water pressure or pressure of bulk materials

L – Live Load

$L_r$ – Roof Live Load

S – Snow Load

R – Rain Load

W – Wind Load

$f_1$ = 1 (for places of public assembly live loads in excess of 100 psf, and parking garages and 0.5 for other live loads)

$f_2$ = 0.7 (for roof configurations that do not shed snow off the structure and 0.2 for other roof configurations)

ACI 318-19 and AISC 360-16 load combinations are used if determined to be more stringent.

NSRST structures are evaluated for extraordinary events such as extreme winter precipitation, tornado missiles, and accidents per building's functional requirements using load combinations per Table 6.4-7, Table 6.4-8, Table 6.4-11 and Table 6.4-12.


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6.4.2.2.6 Load Factors for Structural Stability

SR Substructure Foundations

The required factors of safety and load combinations for structural stability, such as overturning moments, sliding, and floatation of SR substructure foundations are addressed in Table 6.4-9.

NSRST Substructure Foundations

The following minimum factors of safety for structural stability of NSRST substructure are:

- Overturning 1.50
- Sliding 1.10
- Buoyancy 1.10

6.4.2.3 Design and Analysis Procedures

6.4.2.3.1 Design of Safety-Significant Substructures

The safety-significant substructures are designed as reinforced concrete structures, including concrete slabs, walls, and foundations. The SR concrete substructures are designed by the limit state design method in accordance with ACI 349-13 and RG 1.142 Revision 3, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments).” The NSRST RAB concrete substructure is designed by the limit state design method in accordance with ACI 318-08.

Concrete Structure Design

The reinforced concrete SR RXB substructure is structurally isolated from the steel NST RXB superstructure as described in Section 7.8.1.1. The reinforced concrete SR NCB and FHB substructures provide support for the NSRST superstructure as described in Section 7.8.4 and Section 7.8.2 respectively. The reinforced concrete NSRST RAB substructure provides support for the NSRST RAB superstructure as described in Section 7.8.3.

Steel Structure Design

Structural steel shapes are not used as principal structural members in SR and NSRST substructures. Stay-in-place formwork is used as a prefabricated modular support frame to support concrete pouring and curing.

Bolted connections are designed based on provisions of ANSI/AISC N690-18 for SR structures, AISC 360-16 for NSRST structures, and AISC 348-14 (Reference 6.4-12).

Welded connections meet the criteria provided in ANSI/AISC N690-18 for SR structures, AISC 360-16 for NSRST structures, AWS D1.1-15 (Reference 6.4-13) for structural steel, and AWS D1.6-07 (Reference 6.4-14) for stainless steel.


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Missile Protection Design

Safety-significant substructures are designed for tornado wind loading and missiles as shown in Table 6.4-15.

Concrete Anchors and Embedment Plate Design

The SR concrete anchors and steel embedments that transmit forces to the concrete structure by bearing, shear, tension, or a combination thereof are designed according to ACI 349-13, Appendix D, with the exceptions noted in RG 1.199, “Anchoring Components and Structural Supports in Concrete,” Revision 1. Concrete anchors in NSRST substructures are designed in accordance with ACI 318-19.

SR and NSRST substructures comprised of concrete are analyzed by modeling the structures in GT STRUDLVersion 40 finite element model program utilizing the appropriate load combinations from Table 6.4-5 and Table 6.4-7.

6.4.2.3.2 Analysis of Safety-Significant Substructures

FE Model Geometry

The GT STRUDL is used to develop a finite element (FE) structural model of the safety-significant substructures. GT STRUDL is a structural engineering finite element software capable of performing structural analysis and design. The geometry of FE model is created using shell elements to represent the walls and slabs.

Dynamic Mass

The dynamic mass of inertia is defined prior to performing dynamic analysis, in GT STRUDL. There are three ways to define inertia: 1) automatic computation, 2) direct definition, and 3) definition based on static loading conditions. All mass specified in the ADDITIONS mode is accumulated by the number of inertia definitions it may be given. The inertia of the SR NI substructure model is defined by the first and third approach. The automatic computation of inertia approach uses the weight densities defined for elements. Inertia is defined based on static loading conditions. This approach converts the absolute value of forces to their translational mass equivalents by dividing the force components by gravitational acceleration.

In the FE model for safety-significant substructures, the following loads are defined as dynamic mass:

- Dead weight plus 25 percent of live load
- Superstructure loads included its dead weight, 25 percent of live load, and 75 percent of snow load
- Equipment weights


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The dead weight of the substructure is defined by weight densities. Other applicable dead and live loads are defined by element surface loads. The superstructure loads are defined by point loads at the column base supports, area loads, and line loads, as appropriate. Equipment loads are defined by joint loads.

Element and Material Properties

Reinforced concrete elements are modeled using best estimate stiffness properties depending on the stress state of the concrete under seismic load combinations. In lieu of detailed stiffness evaluations, Table 3-2 of ASCE 4-16 (Reference 6.4-1) permits the use of effective stiffness of reinforced concrete elements.

For out-of-plane response of walls, slabs, and diaphragms, it is acceptable to consider cracked section properties in flexure, and uncracked section in shear and axial. Considering a cracked section flexural rigidity of $0.5EI_g$, shear rigidity of GA, axial rigidity of EA, while maintaining the total structural weight, the following modified properties are used for the cracked sections indicated with prime (') symbol:

$t' = 0.707 \times t$

$E' = 1.414 \times E$

$G' = 1.414 \times G$

$\gamma' = 1.414 \times \gamma$

Where

$E'$ = Modulus of elasticity of concrete for cracked section

$E$ = Modulus of elasticity of concrete for uncracked section

$G'$ = Shear modulus of concrete for cracked section

$G$ = Shear modulus of concrete for uncracked section

$\gamma'$ = Weight density of concrete for cracked section

$\gamma$ = Weight density of concrete for uncracked section

$t'$ = Thickness of cracked wall or slab section

$t$ = Thickness of uncracked wall or slab section

$I_g$ = Gross moment of inertia

$A$ = Gross area of concrete section


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Element Type and Mesh Size

The walls and slabs are modeled using GT STRUDL SBMITC elements. The element is a stretching and bending quadrilateral element with six degrees-of-freedom at each of the four corner nodes. The SBMITC is a versatile element for general use in both thin and moderately thick plate-shell problems. A rectangular shape with aspect ratio not exceeding 2:1 is used for most of the elements. Triangular elements are not used in the FE model.

For seismic analysis, the finite element model produces response parameters of interest critical to design. The mesh size must be capable of transmitting the entire frequency range of interest. At least five elements are needed for the smallest wavelength which must be first calculated by dividing the lowest velocity by the highest frequency of interest as follows:

$$\lambda = \frac{V}{f}$$

Where

$\lambda$ = The wavelength

$V$ = The lowest shear wave velocity

$f$ = The highest frequency of interest (i.e., cutoff frequency)

The lowest shear wave velocity is approximately 600 feet per second (200 meters per second) for soft soil or backfill. A conservative lowest shear wave velocity of 6,400 feet per second (2,000 meters per second) is used for hard rock. The associated cutoff frequencies are approximately 14 Hz and 70 Hz. Therefore, the maximum mesh sizes (L) to capture response up to the respective cutoff frequencies are:

For soft soil and backfill: = [(600 ft/sec) / (14 Hz)] / 5 = 8.6 feet (2.6 meters)

For hard rock: = [(6,400 ft/sec) / (70 Hz)] / 5 = 18.3 feet (5.58 meters)

Material Properties

The nominal concrete compressive strength is 5,000 pounds per square inch (30,000 kilopascals). Young's modulus, shear modulus, Poisson's ratio, and weight density for cracked and uncracked concrete are listed in Table 6.4-10.

It is acceptable to model the stiffness assuming the sections are cracked in flexure. Concrete sections are considered cracked in flexure if they are not directly supported by soil or rock, and uncracked if they are.

In addition, to account for concrete cracking in flexure while maintaining the total weight (mass), the thickness of the cracked section is adjusted.


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Load for Dynamic Mass

Dynamic mass for seismic analysis includes structural dead weight, dead weight of major equipment, weight of superstructure, 25 percent of live loads, 75 percent of snow load (when applicable), and hydrodynamic load.

Superstructure Reactions

The superstructure is not explicitly modeled together with the substructure. However, its weight and associated loads are included in the substructure FE model as independent load cases to represent the dynamic mass. The loads are calculated in a separate superstructure FE model. The superstructure reactions are calculated for dead loads, 25 percent of live loads, and 75 percent of snow loads.

Superstructure column locations are modeled at the center of each column with joints that are connected to the slab or wall elements using rigid beams. It is intended to connect each column joint to four nodes of the corresponding one finite element which is the closest and mostly centered with respect to the column location. The column location and the center of the finite element may not coincide due to the mesh size, but it is considered reasonable and acceptable to capture the seismic mass effects of the superstructure for seismic analysis and critical section design. In scenarios where more accurate analysis is warranted, the column and slab connections are modeled in a refined model.

Fixed Base Analysis and Results

The fixed base modal or frequency analysis and stiffness analysis (i.e., static) are performed with total weight including self-weight, dead and live loads, superstructure loads, major equipment loads, hydrodynamic loads, and crane load. The fixed base is implemented at the lowest elevation. The purpose of the analysis is to strictly check the geometry and the load application (e.g., by examining deformed shape and input and output equilibrium).

Dynamic Soil Pressures on Walls of NSRST Substructures

The dynamic soil pressure as a seismic lateral earth pressure is calculated using the simplified method for non-yielding walls with no significant structure to structure interaction in accordance with Section 8.2 of ASCE 4-16.

#### 6.4.2.3.3 Design of NSRST Superstructures

The NSRST superstructures are designed as steel structures. Specific details on the design of the NSRST superstructures are provided in Section 7.8 for each of the NSRST superstructures.

Concrete Structure Design

Concrete located within NSRST superstructures is designed by the limit state design method in accordance with ACI 318-19. Required reinforcing for the concrete slabs and radiation protection shield walls are determined in accordance with applicable code provisions.


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Steel Structure Design

The steel structure design of NSRST superstructures is based on the Load and Resistance Factor Design methods in accordance with AISC 325-17 (Reference 6.4-9). The NSRST structural steel members are designed, and code checked using provisions of AISC 360-16 and AISC 341-16 (Reference 6.4-10) as applicable per ASCE 7-16 and IBC-2021 (Reference 6.4-11).

Bolted connections are designed based on provisions of AISC 360-16 for NSRST structures, and AISC 348-14 (Reference 6.4-12).

Welded connections meet the criteria provided in ANSI/AISC N690-18 for SR structures, AISC 360-16 for NSRST structures, AWS D1.1-15 (Reference 6.4-13) for structural steel, and AWS D1.6-07 (Reference 6.4-14) for stainless steel.

Missile Protection Design

Each of the NSRST superstructures are designed to withstand applicable tornado-induced missiles as described in shown in Table 6.4-15.

Concrete Anchors and Embedment Plate Design

The NSRST concrete anchors and steel embedments that transmit forces to the concrete structure by bearing, shear, tension, or a combination thereof are designed according to Chapter 17 of ACI 318-08. Concrete anchors used in NSRST anchors in SR concrete are designed according to Appendix D of ACI 349-13.

6.4.2.3.4 Analysis of NSRST Superstructures

The analysis of the NSRST superstructure is performed using the structural analysis software GT STRUDL as the primary analysis tool based on three-dimensional finite element models of the structures. The analysis output covers the structural member design, deflection check, support reactions, dynamic response characteristics, and structural steel quantities.

Geometry and Dimensions

The structural model for steel superstructure is manually created based on coordination with architectural and general arrangement layout drawings.

Structural Modeling

The GT STRUDL model is a representation of the primary Lateral Force Resisting System and gravity load system within the superstructure that contribute to the stability of the structure. Member releases are set based on the expected frame behavior for the corresponding Lateral Force Resisting System.


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The model includes walls, slabs, major structural beams, and columns, except secondary members. Secondary members including girt systems, roof purlins, and construction aids are not modeled because they do not contribute to stability of the structure while an equivalent member load is applied to the model. These secondary members are checked independently using the applicable design loads.

Load Input

The self-weight dead loads of the superstructure model are based on the weight densities of the modeled analytical members. Static loads calculated in building design load calculation including equipment weights are applied to the analytical models based on tributary widths of the members. These include dead, live, wind, and snow loads.

Consideration of the applicable loads and loads effects of construction methods, equipment operation, and sequence of construction are included in the load combinations for analysis and design. Design load during construction for buildings and other structures are developed in accordance with ASCE 37-14 (Reference 6.4-15).

Seismic Analysis

The seismic analysis of NSRST superstructures follow ASCE 7-16; further details are provided in Section 6.4.1.

Structure Analysis

The structural analysis is performed in accordance with the load combinations identified in Table 6.4-12.

6.4.2.4 Materials, Quality Control, and Special Construction Techniques

6.4.2.4.1 Materials

The major construction materials for the safety-significant structures are concrete, reinforcing steel, structural steel, stainless steel, bolts, and weld electrodes.

Concrete

The structural concrete work applied in safety-significant structures is performed in accordance with ACI 301-16 (Reference 6.4-16). The structural concrete has a minimum compressive strength ($f'_c$) of 5,000 pounds per square inch (30,000 kilopascals) measured at 28 days.

Reinforcing Steel

Reinforcing bars meet the requirements of ASTM A615-09b (Reference 6.4-17) or ASTM A706-09b (Reference 6.4-18), Grade 60. Headed reinforcing bars conform to ASTM A970-07 (Reference 6.4-19).


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Structural Steel

The SR structural steel shapes conform to Section NA3 of ANSI/AISC N690-18. For NSRST structures, the structural steel shapes conform to Section A3 of AISC 360-16.

Stainless Steel

For the SR Spent Fuel Pool in the FHB, the stainless steel material for the liner plate meets the requirements of ASTM A240-16a (Reference 6.4-20) listed in Table 6.4-2.

6.4.2.4.2 Quality Control

Quality control measures and procedures for materials and procurement documents associated with safety-significant structures are consistent with guidance provided in the Quality Assurance Program Description as described in Section 8.1.

6.4.2.4.3 Special Construction Techniques

No special construction techniques are used for the construction of safety-significant structures.

6.4.3 Mechanical Systems and Components

This section addresses the design and analysis of ASME Code Section III, Division 5 mechanical components, supports, and core support structures. The plant transient events and number of occurrences used in fatigue analyses are described in Section 6.4.3.1.1.

Seismic analysis of mechanical systems and components is described in Section 6.4.3.2. The structural integrity of components, component supports, and core support structures is ensured by satisfying the ASME Section III, Division 5 service stress limits for specified loads and load combinations. Stress levels for load combinations, in the mechanical components and supports, are calculated using the methods and equations in the ASME Code, Section III. Load combinations and service stress limits are given in Section 6.4.3.3 for mechanical components and associated supports.

6.4.3.1 Special Topics for Mechanical Components

This section addresses methods of analysis for components, supports, and core supports designated as ASME Code, Section III Class A and B.

6.4.3.1.1 Design Transients

The design transient conditions are derived from the licensing basis events that generate changes in thermal hydraulic conditions in the nuclear island. The licensing basis events are described in Section 3.6 through 3.9.

The following service and test loading categories, as defined in the ASME Code Section III, Division 5 (Reference 6.4-23), are considered in the design of ASME Code Section III, Division 5 components.


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Level A Service Loadings

Level A Service Loadings are loadings arising from system startup, hot standby operation in the design power range, hot standby, and system shutdown, excluding loadings covered by Level B, C, and D Service Loadings or Test Loading.

Level B Service Loadings

These are deviations from Level A Service Loadings that are anticipated to occur often enough that the design should include a capability to withstand the loadings without operational impairment. The events which cause Level B Service Loadings include those transients which result from single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. These events include abnormal incidents not resulting in a forced outage and forced outages for which the corrective action does not include any repair of mechanical damage resulting from the transient progression. The estimated duration of a Level B Service Loading is included in the design specifications.

Level C Service Loadings

These are deviations from Level A Service Loadings that require a shutdown for correction of the loadings or repair of damage in the system. The conditions have a low probability of occurrence, but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events may not exceed 25. If more than 25 are expected, then some types of events must be evaluated by the more stringent requirements of the Level B Service Limits.

Level D Service Loadings

These are combinations of loadings associated with postulated events of extremely low probability whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that only considerations of public health and safety are involved.

Testing Loadings

These are pressure loadings that occur during hydrostatic tests, pneumatic tests, and leak tests. Other types of tests shall be classified under Service Level A or B loading categories. If any elevated temperature tests are specified as test loadings for a component, then these loadings are considered as part of the Service Level B loadings for the component.

Transients used in the design of ASME Section III components are described in this section. The transients and specified number of occurrences of each transient are listed in Table 6.4-16.


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Service Level A Conditions

Service Level A Transient 1 – Cold Shutdown to Hot Standby

This transient covers the heatup of the plant from cold shutdown to hot zero power. The transient begins with the plant at cold shutdown. The primary sodium is heated up to hot standby conditions, IAC is operated along with the Primary Sodium Pump (PSP) and Intermediate Sodium Pump (ISP) to maintain hot standby conditions.

Service Level A Transient 2 – Hot Standby to Full Power

The reactor is taken critical and the plant is heated up to full power operating conditions as the power ascent occurs. PSP and ISP flow is increased and the Intermediate Air Cooling System (IAC) continues to operate to maintain target temperatures. NI Salt System (NSS) and Energy Island Salt Heat Transport System (ESS) are brought online.

The PSP and ISP flow are adjusted to the normal operating value for the power level. Cooling is transitioned from the IAC to the NSS and ESS. This transient ends with the reactor at full power.

Service Level A Transient 3 – Full Power to Cold Shutdown

This transient begins with the reactor operating at full power. Power descent is performed to transition from Mode 1 to Mode 2 operation. Power reduction is achieved by inserting control rods. Flow in the Primary Heat Transport System (PHT), Intermediate Heat Transport System (IHT), NSS and ESS are adjusted as reactor power decreases. Cooling is transitioned from the NSS and ESS to the IAC. The plant enters Mode 2 once cooling is provided by IAC and the NSS isolation valves are closed. The reactor is shutdown and further cooled down to cold shutdown conditions.

Service Level A Transient 4 – Hot Standby to Cold Shutdown

This transient consists of a cooldown from hot standby to cold shutdown. During the cooldown, heat removal is provided by the IAC.

Service Level A Transient 5 – Decrease in Reactor Power

This transient begins with the reactor operating at full power. Power is then reduced to the minimum power level for automatic control. Flow in the PHT, IHT, NSS and ESS are adjusted as reactor power decreases. The lower core power results in the PHT and IHT temperatures decreasing.

Service Level A Transient 6 – Increase in Reactor Power

The reactor is initially at the minimum power level for automatic control and operating at steady state. Power is increased to full power using the control rods. Flow in the PHT, IHT, NSS and ESS are adjusted as reactor power increases. The PHT and IHT temperatures are increased as a result of increased power.


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Service Level A Transient 7 – Partial Decrease in Reactor Power

This transient begins with the reactor operating at full power. Power is reduced to 50 percent power. Flow in the PHT, IHT, NSS and ESS are adjusted as reactor power decreases. The PHT and IHT temperatures decrease as a result of the decreased power.

Service Level A Transient 8 – Partial Increase in Reactor Power

The reactor is initially operating at 50 percent reactor power. Power is increased to full power. Flow in the PHT, IHT, NSS and ESS are adjusted as reactor power increases. The PHT and IHT temperatures are increased as a result of increased power.

Service Level A Transient 9 – Runback Recovery

This event transitions the plant from a low power operating condition with IAC cooling to full reactor power. To bring NSS and ESS online, PSP and ISP flows are increased. Once NSS and ESS are brought online, the PSP and ISP flows are adjusted down. Reactor power is increased and cooling is transitioned from the IAC to the NSS and ESS. IAC cooling is secured and the reactor power is increased to full power. The transient ends with the reactor operating at full power.

Service Level A Transient 10 – Steady State Temperature Fluctuations

Minor fluctuations, within the steady state operating range, in the normal sodium temperatures are specified in the nuclear island. Temperature fluctuations can occur due to drifts in plant sensors that produce a change in a plant parameter affecting temperatures, minor changes made by operators or control systems when the plant is otherwise operating at steady state, small variations in power, or other unplanned changes in temperatures.

Service Level A Transient 11 – Steady State Pressure Fluctuations

Minor fluctuations, within the steady state operating range, in reactor vessel and intermediate loop pressures are specified to bound the pressure changes expected due to normal Sodium Cover Gas System operations. Pressure fluctuations can occur due to capacity or operation of the cover gas flow pressure regulator, minor changes in plant parameters affecting pressure, or other unplanned changes in system pressures.

Service Level B Conditions

Service Level B Transient 1 – Normal Reactor Scram

The reactor is initially operating at full power when the reactor scram signal occurs and the core is shut down by insertion of control rod assemblies. PSP and ISP flows are reduced. NSS and ESS flows are secured and heat removal is provided by the IAC. The plant is further cooled down to hot standby.


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Service Level B Transient 2 – Loss of One Primary Sodium Pump

The reactor is initially operating at full power when flow from a single PSP is lost and the PSP coasts down. A reactor scram occurs. The operating PSP and ISP flows are then reduced. NSS and ESS flows are secured and heat removal is provided by the IAC. The plant is further cooled down to hot standby.

Service Level B Transient 3 – Loss of One Intermediate Sodium Pump

The reactor is initially operating at full power when all heat removal from one IHT loop is lost. A reactor scram occurs. NSS and ESS flow is secured and cooling transitions to IAC in the operational loop. The plant is further cooled down to hot standby.

Service Level B Transient 4 – Control Rod Induced Transient Overpower

With the reactor operating at full power, a control rod induced transient overpower occurs resulting in a reactor scram. The NSS and ESS flow is secured and cooling transitions to IAC. The plant is further cooled down to hot standby.

Service Level B Transient 5 – Loss of Offsite Power

With the reactor operating at full power, a loss of offsite power occurs resulting in a reactor scram. Due to the loss of AC power, the PSPs and ISPs trip and coast down and NSS isolation valves close. When IAC cooling is established, the plant is cooled down to hot standby. The transient concludes with the plant in hot standby.

Service Level B Transient 6 - Reactor Power Runback

This transient initiates from full power and represents a perturbation in NSS that increases or decreases heat removal capabilities and requires a power runback. When the power runback completes, NSS and ESS are isolated and heat removal is provided by the IAC. The transient concludes with reactor power and temperatures stabilized at low reactor power operating conditions.

Service Level B Transient 7 – Cover Gas Overpressure

The reactor is initially operating at full power. The cover gas regulators are assumed to fail closed coincident with an operational transient that causes significant increase in PHT temperatures. As a result, pressure in the Reactor Enclosure System (RES) increases. Overpressure protection is provided for the RES by a rupture disc and relief valve.

Service Level B Transient 8 – Primary Sodium Pump Overspeed

One PSP has an overspeed event that results in a high flow rate through the PHT. Increased sodium flow decreases sodium temperature within the core, causing a minor increase in reactor power. This transient concludes with reactor power returned to nominal full power conditions.


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Service Level B Transient 9 – Intermediate Sodium Pump Overspeed

One or both ISPs have an overspeed event that results in a high flow rate through the IHT. The primary cold pool temperature decreases, which increases reactor power. This transient concludes with the reactor power returned to nominal full power conditions.

Service Level C Conditions

Service Level C Transient 1 – Loss of All Primary Sodium Pumps

The reactor is initially operating at full power when the flow from both PSPs is lost and both PSPs coast down. A reactor scram occurs. The ISPs continue to operate following the scram. NSS and ESS flows are secured and heat removal is provided by the IAC. The plant is cooled down to cold shutdown.

Service Level C Transient 2 – Intermediate Heat Exchanger Leak

The reactor is initially operating at full power when a single Intermediate Heat Exchanger tube experiences a rupture, resulting in sodium flow from the intermediate to the primary system. Primary sodium level increases, which results in a reactor scram. Flow in the leaking IHT loop is secured to reduce the flow into the reactor vessel. The ISPs trip and the PSP flow is reduced. NSS and ESS flows are secured and heat removal is provided by the IAC in the unaffected IHT loop. The plant is further cooled down to cold shutdown.

Service Level D Conditions

Service Level D Transient 1 – Extended Duration Intermediate Air Cooling

This transient initiates with a reactor scram. For this faulted event, passive IAC cooling is relied upon to cool the reactor and the Reactor Air Cooling System is assumed to be unavailable. Cooldown is modeled until the temperatures are maintained below a temperature that can be sustained indefinitely.

Service Level D Transient 2 – Extended Duration Reactor Air Cooling

This transient initiates with a reactor scram. For this faulted event, the Reactor Air Cooling System is relied upon to cool the reactor. Active and passive IAC cooling are assumed to be unavailable. Cooldown is modeled until the temperatures are maintained below a temperature that can be sustained indefinitely.

Test Conditions

Test 1 – Hydrostatic or Pneumatic Test

Hydrostatic or pneumatic testing of the reactor head and other portions of the reactor enclosure occurs during fabrication.


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Test 2 – Nuclear Island Dry Heatup and Fill

The primary system is heated up dry from ambient temperature to the normal refueling temperature and the primary system is filled with sodium. Temporary systems are used to heat, fill, and achieve sodium purity levels.

Test 3 – Nuclear Island Drain and Cooldown to Ambient

The nuclear island is initially at cold shutdown conditions and sodium is drained using temporary systems. Subsequently, cooldown to ambient conditions occurs by natural heat transfer to the surroundings.

Test 4 – Intermediate System Dry Heatup and Fill

The intermediate system is heated up dry from ambient temperature to the normal refueling temperature. During IHT filling, differential temperatures between the sodium and metal are maintained to prevent sodium freezing.

Test 5 – Intermediate System Drain and Cooldown to Ambient

The intermediate system is initially at cold shutdown conditions and sodium is drained. Subsequently, cooldown to ambient conditions occurs by natural heat transfer to the surroundings.

Test 6 – Intermediate System Freezing and Melting

This transient covers scenarios in which a portion of the intermediate sodium is overcooled and freezes, potentially resulting in a loss of intermediate system flow. Freezing of the sodium in the Intermediate Heat Exchangers is not credible due to the location of the Intermediate Heat Exchangers within the PHT. A melt out procedure is performed to remove the frozen intermediate sodium using trace heating.

#### 6.4.3.1.2 Computer Programs Used in Analyses

The computer programs used in the static and dynamic analysis of mechanical equipment include:

- ANSYS: general purpose finite element analysis program used for static and dynamic analysis of mechanical components.
- TP RELAP5-3D: system thermal-hydraulic analysis code used to develop the thermal-hydraulic conditions for the ASME design transients.
- STAR-CCM+: general purpose computational fluid dynamics software used for detailed thermal hydraulic analyses.

### 6.4.3.2 Seismic Analysis for Mechanical Components

Seismic analysis of mechanical systems and components follows the methodology summarized in Section 6.4.1, with the following RES specific attributes:


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Determination of Number of Earthquake Cycles

Fatigue analyses evaluate the equivalent of 10 cycles at peak stress levels from one SSE and five OBEs. Analyses may be performed at just one of these stress levels. For SSE-level evaluations, 20 cycles are applied. Alternatively, for OBE-level evaluations, cycle numbers are adjusted according to IEEE/IEC 60980-344 (Reference 6.4-21).

Basis for Selection of Frequencies

The fundamental frequency of the RES seismic isolation is designed to be smaller than one half the dominant frequency of the major RES SR SSC. However, when sufficient separation cannot be provided, the RES and the RES seismic isolation are designed to meet the applicable code limits and applied loads. The seismic isolation of the RES is described in Section 7.1.2.

Analysis Procedure for Damping

Damping in the RES seismic model is applied uniformly as frequency dependent Rayleigh damping at a target level of four percent.

When modal damping is used in a response spectrum or mode-superposition time-history analysis, the application of damping level is in accordance with ASCE 4-16, Section 3.5. When mode superposition is used in a response spectrum analysis for seismic qualification of RES SSCs, the analysis methods and application of damping are in accordance with RG 1.92.

Use of Equivalent Vertical Static Factors

The vertical component of the safe shutdown earthquake is considered to occur simultaneously with the two horizontal components in the RES seismic model. Therefore, constant vertical static factors are not directly used for the design of the seismic RES SR SSCs. However, when separate dynamic analyses are performed on the RES subsystems, static vertical load factors can be used to calculate vertical response loads if the SSC is rigid in the vertical direction.

An equivalent static load method is acceptable if:

1. Justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses.

2. The simplified static analysis method accounts for the relative motion between all points of support.

3. To obtain an equivalent static load for an SSC that can be represented by a simple model, a factor of 1.5 is applied to the peak spectral acceleration of the applicable ground or floor response spectrum. A factor less than 1.5 may be used, if adequate justification is provided.

The in-structure response spectrum is obtained from the RES seismic model at the support location of the subsystem.


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### 6.4.3.3 ASME Code Class A and B Components, Component Supports, and Core Support Structures

This section discusses the structural integrity of the components, component supports, and core support structures designed and constructed according to the rules of the ASME Code Section III.

#### 6.4.3.3.1 Loading Combinations, System Operating Transients, and Stress Limits

The loads considered for the design of Code Class A and B components, component supports, and core support structures are summarized in Table 6.4-17.

Pressure loading is identified as either the design pressure, an operating pressure associated with a Service Level A, B, C, or D event, or a hydrostatic or pneumatic test pressure. Pressure may be internal or external to the component. For core support structures, design pressure differences and operating pressure differences associated with a Service Level A, B, C, or D event are considered.

The plant transient events are defined in Section 6.4.3.1.1. The variations in pressure, temperature, and flow rate during the transients are developed and used as inputs to the applicable component's stress analysis.

Buoyancy loads are applicable to the components submerged in fluid. Buoyancy forces are calculated based on the weight of the fluid displaced.

Seismic loadings are due to the vibratory ground motion associated with the SSE. The OBE is set at 1/3 of the SSE which eliminates the need for explicit response or design analysis per 10 CFR 50 Appendix S; however, the cyclic effects of OBE are included in the fatigue analysis as described in Section 6.4.3.2. Seismic loads can consist of both inertial loads and anchor motions, as applicable, where a component is supported at multiple locations.

Code Class A or B components designed in accordance with HBB or HCB are subjected to static pressure testing loads in accordance with HBB-6000 or HCB-6000 as applicable. Component supports consider the weight of the pressure test medium where applicable.

The Code Class A and B design and service load combinations and the applicable service stress limits are provided in Table 6.4-18. Static pressure tests are evaluated against the Test limits. The other test events described in Section 6.4.3.1.1 are evaluated against Level B Service Limits per HBB-3113.7 (Reference 6.4-23).

Where functionality of components and component supports cannot be ensured by satisfaction of the service stress limits alone, functional limits are established to support the required safety functions of components and component supports.


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6.4.3.3.2 Design and Installation of Pressure Relief Devices

The reactions loading due to discharge of a relief device is analyzed using the guidance of ASME Code Section III, Appendix O (Reference 6.4-22). Reaction forces and moments are based on a static analysis with a dynamic load factor of 2.0 unless justification for a lower dynamic load factor is provided or dynamic analysis is performed. The reaction loads are included in the applicable load combination shown in Table 6.4-18.

## 6.4.4 ASME BPVC, Section III Piping Systems, Piping Components, and Associated Supports

### 6.4.4.1 Introduction

This section addresses design of safety-significant piping systems and piping supports.

Design of piping systems, including piping components and associated supports is preliminary and informs development of equipment layout and building arrangements. Design of piping systems, piping components, and associated supports, consistent with the requirements of ASME Boiler and Pressure Vessel Code (BPVC) Section III, will be performed in support of the operating license stage.

### 6.4.4.2 Safety Design Basis

#### 6.4.4.2.1 ASME BPVC

Application of ASME BPVC, Section III, Division 1 (Reference 6.4-24, Reference 6.4-25, and Reference 6.4-26) to piping is based on 10 CFR 50.55a conditions of use.

Design of ASME BPVC, Section III, Division 1 piping and supports is performed as follows:

- Class 1 piping per the requirements of BPVC Section III, Subarticle NB-3600
- Class 2 piping per the requirements of BPVC Section III, Subarticle NC-3600
- Class 3 piping per the requirements of BPVC Section III, Subarticle ND-3600
- Class 1, 2, 3 piping supports per the design requirements of BPVC Section III, Subarticle NF

Where safety-significant piping systems are subjected to metal temperatures exceeding the temperature limits of ASME BPVC Section III, Division 1, design is performed in accordance with ASME BPVC Section III, Division 5 (Reference 6.4-23) as follows:

- Class A piping per the design requirements of ASME BPVC, Section III, Division 5, Subarticle HBB-3600
- Class B piping per the design requirements of ASME BPVC, Section III, Division 5, Subarticle HCB-3600
- Class A and Class B metallic supports in low temperature service per the design requirements of ASME BPVC, Section III, Division 5, Subsection HF


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6.4.4.2.2 ASME Code Cases

ASME Code Cases may be used if they are, either conditionally or unconditionally, approved in RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” Revision 39.

6.4.4.2.3 Design Specification

Design Specifications required for ASME BPVC, Section III piping, piping components and associated supports are prepared during final design with conformance documented in Design Reports.

Where ASME BPVC, Section III, Division 5 is applicable, Design Specifications prepared during final design address mechanical loadings, stress intensification reductions, and reductions in yield strength with conformance documented in Design Reports.

6.4.4.3 Piping Analysis Methods

Flexibility

Where possible, piping flexibility is obtained through “natural” pipe routing with the inclusion of expansion loops when required.

**References**

6.4-1 American Society of Civil Engineers/Structural Engineering Institute, Seismic Analysis of Safety-Related Nuclear Structures, ASCE/SEI 4-16.

6.4-2 American Society of Civil Engineers/Structural Engineering Institute, Minimum Design Loads and Associated Criteria for Buildings and Other Structures, ASCE/SEI 7-16 w/Supplement 1.

6.4-3 American Society of Civil Engineers/Structural Engineering Institute, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, ASCE/SEI 43-19.

6.4-4 American Concrete Institute, Building Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, ACI 349-13.

6.4-5 American National Standards Institute/American Institute of Steel Construction, Specification for Safety-Related Steel Structures for Nuclear Facilities, ANSI/AISC N690-18.

6.4-6 American Concrete Institute, Building Code Requirements for Structural Concrete, ACI 318-08.

6.4-7 American National Standards Institute/American Institute of Steel Construction, Specification for Structural Steel Buildings, ANSI/AISC 360-16.


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6.4-8 American Concrete Institute, Building Code Requirements for Structural Concrete, Commentary on Building Code Requirements for Structural Concrete (ACI 318R-19), ACI 318-19.

6.4-9 American Institute of Steel Construction, AISC Steel Construction Manual, 15th Edition, AISC 325-17.

6.4-10 American Institute of Steel Construction, Seismic Provisions for Structural Steel Buildings, AISC 341-16.

6.4-11 International Building Code, 2021 International Building Code, IBC 2021.

6.4-12 American Institute of Steel Construction, Specification for Structural Joints Using High Strength Bolts, AISC 348-14.

6.4-13 American Welding Society, Structural Welding Code – Steel, AWS D1.1-15.

6.4-14 American Welding Society, Structural Welding Code – Stainless Steel, AWS D1.6-07.

6.4-15 American Society of Civil Engineers, Design Loads on Structures During Construction, ASCE 37-14.

6.4-16 American Concrete Institute, Specifications for Structural Concrete, ACI 301-16.

6.4-17 American Society for Testing and Materials, Standard Specification for Deformed and Plain Carbon-Steel Bars for Concrete Reinforcement, ASTM A615-09b.

6.4-18 American Society for Testing and Materials, Standard Specification for Deformed and Plain Low-Alloy Steel Bars for Concrete Reinforcement, ASTM A706-09b.

6.4-19 American Society for Testing and Materials, Standard Specification for Headed Steel Bars for Concrete Reinforcement, ASTM A970-07.

6.4-20 American Society for Testing and Materials, Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessel and for General Applications, ASTM A240-16a.

6.4-21 Institute of Electrical and Electronics Engineers, “IEEE/IEC 60980-344-2020: Nuclear facilities – Equipment important to safety – Seismic qualification,” Piscataway, NJ, 2020.

6.4-22 ASME Boiler and Pressure Vessel Code, Section III, Appendices, Rules for Construction of Nuclear Facility Components, 2017 Edition.

6.4-23 ASME Boiler and Pressure Vessel Code, Section III, Division 5, Rules for Construction of Nuclear Facility Components – High Temperature Reactors, 2017 Edition.


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6.4-24 ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components, Division 1 – Subsection NB, Class 1 Components, 2017 Edition.

6.4-25 ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components, Division 1 – Subsection NC, Class 2 Components, 2017 Edition.

6.4-26 ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components, Division 1 – Subsection ND, Class 3 Components, 2017 Edition.



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**Table 6.4-1 NSRST SSC Seismic Classifications and Seismic Criteria**

| Seismic Classification | Seismic Design Parameters |
|---|---|
| SCN1 | • Seismic Design Parameters:<br>- Maximum Considered Earthquake Ground Motion = SSE<br>- Seismic Risk Category IV<br>- Importance Factor, I > 1.0 ($I_e$ = 1.5 for structures)<br>• Structural Design: International Building Code 2021<br>• Non-structural components:<br>- Commercial design codes applicable to SSC type<br>- Designated seismic system requiring qualification by testing |
| SCN2 | • Seismic Design Parameters:<br>- Maximum Considered Earthquake Ground Motion = SSE<br>- Seismic Risk Category III<br>- Importance Factor, I > 1.0 ($I_e$ = 1.25 for structures)<br>• Structural Design: International Building Code 2021<br>• Non-structural components:<br>- Commercial design codes applicable to SSC type<br>- Designated seismic system requiring qualification by testing |
| SCN3 | • Seismic Design Parameters:<br>- Maximum Considered Earthquake Ground Motion = SSE<br>- Seismic Risk Category II<br>- Importance Factor, I = 1.0 ($I_e$ = 1.0 for structures)<br>• Structural Design: International Building Code 2021<br>• Non-structural components:<br>- Commercial design codes applicable to SSC type |


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**Table 6.4-2 Design Codes and Standards for Safety-Significant Structures (Sheet 1 of 2)**

| Design Code | Title |
|---|---|
| ACI 301-05 | Specifications for Structural Concrete |
| ACI 301-16 | Specifications for Structural Concrete |
| ACI 318-08 | Building Code Requirements for Structural Concrete and Commentary |
| ACI 318-19 | Building Code Requirements for Structural Concrete (ACI 318-19) Commentary on Building Code Requirements for Structural Concrete (ACI 318R-19) |
| ACI 347R-14 | Guide to Formwork for Concrete |
| ACI 349-13 | Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary |
| AISC 325-17 | Steel Construction Manual, 15th Edition |
| AISC 341-16 | Seismic Provisions for Structural Steel Buildings |
| AISC 348-14 | Specification for Structural Joints Using High-Strength Bolts |
| AISC 360-16 | Specification for Structural Steel Buildings |
| AISC N690-18 | Specification for Safety-Related Steel Structures for Nuclear Facilities |
| ASCE/SEI 4-16 | Seismic Analysis of Safety-Related Nuclear Structures |
| ASCE/SEI 7-16 | Minimum Design Loads and Associated Criteria for Buildings and Other Structures |
| ASCE 37-14 | Design Loads on Structures During Construction |
| ASCE 43-19 | Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities |
| ASTM A36-20 | Standard Specification for Carbon Steel |
| ASTM A240-16a | Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessel and for General Applications |
| ASTM A325-14 | Standard Specification For Structural Bolts, Steel, Heat Treated, 120/105 ksi Minimum Tensile Strength |
| ASTM A490-14 | Standard Specification for Structural Bolts, Alloy Steel, Heat Treated, 150ksi Minimum Tensile Strength |
| ASTM A500-99 | Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes |
| ASTM A572-12 | Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel |
| ASTM A615-09b | Standard Specification for Deformed and Plain Carbon-Steel Bars for Concrete Reinforcement |
| ASTM A659-18 | Standard Specification for Commercial Steel (CS), Sheet and Strip, Carbon (0.16 Maximum to 0.25 Maximum Percent), Hot-Rolled |
| ASTM A706-09b | Standard Specification for Deformed and Plain Low-Alloy Steel Bars for Concrete Reinforcement |


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**Table 6.4-2 Design Codes and Standards for Safety-Significant Structures (Sheet 2 of 2)**

| Design Code | Title |
|---|---|
| ASTM A913-19 | Standard Specification for High-Strength Low-Alloy Steel Shapes of Structural Quality, Produced by Quenching and Self-Tempering Process (QST) |
| ASTM A970-07 | Standard Specification for Headed Steel Bars for Concrete Reinforcement |
| ASTM A992-20 | Standard Specification for Structural Steel Shapes |
| ASTM A1011-18a | Standard Specification for Steel, Sheet, and Strip, Hot-Rolled, Carbon, Structural, High-Strength Low-Alloy, High-Strength Low-Alloy with Improved Formability, and Ultra-High Strength |
| ASTM F1554-20 | Standard Specification for Anchor Bolts, Steel, 36, 55, and 105-ksi Yield Strength |
| ASTM F3125-22 | Standard Specification for High Strength Structural Bolts, and Assemblies, Steel and Alloy Steel, Heat Treated, Inch Dimensions 120ksi and 150ksi Minimum Tensile Strength, and Metric Dimensions 830MPa and 1040MPa Minimum Tensile Strength |
| ASTM F3148-17a | Standard Specification for High Strength Structural Bolt Assemblies, Steel and Alloy Steel, Heat Treated, 144ksi Minimum Tensile Strength, Inch Dimensions |
| AWS D1.1-15 | Structural Welding Code - Steel |
| AWS D1.6-07 | Structural Welding Code - Stainless Steel |
| IBC-2021 | International Building Code |


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**Table 6.4-3 Applicability of Structure Primary Design Codes Based on Safety Classification**

| Code | Applicability | Safety Classification | |
|---|---|---|---|
| | | SR | NSRST |
| ACI 301-05 | Structural Concrete Specification | X | |
| ACI 301-16 | Structural Concrete Specification | | X |
| ACI 318-08 | Concrete Structure – Limit State Design Method (in conjunction with ACI349-13 and ACI 301-05) | X | |
| ACI 318-19 | Concrete Structure – Limit State Design Method (in conjunction with ACI301-16) | | X |
| ACI 349-13 | Concrete Structure – Limit State Design Method | X | |
| AISC 341-16 | Structural Steel Seismic Design Provisions | X | X |
| AISC 348-14 | Specification for Structural Joints Using High-Strength Bolts | X | X |
| AISC 360-16 | Structural Steel – Load and Resistance Factor Design (in conjunction with AISC348-14) | | X |
| AISC N690-18 | Structural Steel – Load and Resistance Factor Design ( in conjunction with AISC360-16) | X | |
| AISC 325-17 | Structural Steel Design – Shapes Only | X | X |
| ASCE 7-16 | General Design Basis Loads | X | X |
| ASCE 7-16 | Seismic Design Methodology | X | X |
| AWS D1.1-2015 | Steel Welding Design | X | X |
| AWS D1.6-2007 | Stainless Steel Welding Design | X | X |
| IBC 2021 | Building Design Code for State of Wyoming | | X |


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**Table 6.4-4 Applicability of Regulatory Guides to Safety-Significant Structures**
**(Sheet 1 of 2)**

| RG | Revision | Title | Conformance | Basis |
|---|---|---|---|---|
| RG 1.13 | 2 | Spent Fuel Storage Facility Design Basis | Partial | Provides guidance regarding the design of spent fuel storage facilities that is applicable to structural-related aspects for the Spent Fuel Pool and Fuel Handling Building. Partial conformance for the Fuel Handling Building superstructure is related to the structural design approach for tornado-generated missiles as described in Section 7.8.2. |
| RG 1.69 | Rev. 1 | Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants | Full | Provides guidance related to minimum concrete thickness, loading requirements, and concrete design for radiation shielding in safety-significant structures. |
| RG 1.76 | Rev. 1 | Design Basis Tornado for Nuclear Power Plants | Full | Provides parameters for design basis tornado characteristics and tornado-generated missiles used for the design and analysis safety-significant structures. |
| RG 1.102 | Rev. 1 | Flood Protection for Nuclear Power Plants | Full | Provides guidance for flood protection of safety-significant structures. |
| RG 1.142 | Rev. 3 | Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) | Full | Provides guidance for the design of safety-related concrete structures. |


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**Table 6.4-4 Applicability of Regulatory Guides to Safety-Significant Structures (Sheet 2 of 2)**

| RG | Revision | Title | Conformance | Basis |
|---|---|---|---|---|
| RG 1.143 | Rev. 2 | Design Guidance for Radioactive Waste Management Systems, Structures, And Components Installed in Light-Water-Cooled Nuclear Power Plants | Partial | Provides guidance for the design of radwaste systems and structural-related aspects of buildings that house radwaste systems. Partial conformance for the Fuel Handling Building superstructure design is related to use of applicable load combinations for extraordinary events based on the building's functional requirements as discussed in Section 6.4.2.2.5 for natural phenomena and man-induced hazards. In addition, updated codes and standards are used for the structural design instead of the versions cited in RG 1.143. |
| RG 1.199 | Rev. 1 | Anchoring Components and Structural Supports in Concrete | Full | Provides acceptable method for design, installation, testing, evaluation, and quality assurance of anchors used for components and structural supports in safety-related concrete structures. |
| RG 1.243 | Rev. 0 | Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments | Full | Provides updated regulatory guidance for the design, fabrication, and erection of safety-related steel structures and endorses, with exceptions and clarifications, the procedures and standards in ANSI/AISC-N690-18. |



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**Table 6.4-5 Load Combinations and Load Factors for Concrete SR Substructures**
**(Sheet 1 of 2)**

| | | Normal and Severe | | | | | | | | | Extreme and Abnormal | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Design Load Cases[1][5]** | | C-1 | C-2 | C-3 | C-4 | C-5 | C-6 | C-7 | C-8 | C-9 | C-9[6] | C-10 | C-11 | C-12 | C-13 | C-16 | C-17 |
| **Load Combination Equation No. in ACI349-13** | | 9-1 | 9-2 | 9-2 | 9-2 | 9-3 | 9-3 | 9-3 | 9-4 | 9-5 | 9-6 | 9-6 | 9-7 | 9-7 | 9-7 | 9-8 | 9-9 |
| Dead | D [3] | 1.4 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Live | L | | 1.6 | 1.6 | 1.6 | 0.8 | 0.8 | 0.8 | 1.6 | 1.6 | 1.0[16] | 1.0[16] | 1.0[16] | 1.0[16] | 1.0[16] | 1.0[16] | 1.0[16] |
| Roof Live | $L_r$ | | 0.5 | | | 1.6 | | | | | | | | | | | |
| Snow | S | | | 0.5 | | | 1.6 | | | | | | | | | | |
| Rain | R | | | | 0.5 | | | 1.6 | | | | | | | | | |
| Hydrostatic | F | 1.4 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Soil | H | | 1.6 | 1.6 | 1.6 | 0.8 | 0.8 | 0.8 | 1.6 | 1.6 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Pipe Reactions | $R_o$ | 1.0[9] | 1.6[10] | 1.6[10] | 1.6[10] | 0.8[12] | 0.8[12] | 0.8[12] | 1.6[14] | 1.6 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Thermal | $T_o$ | 1.0 | 1.6 | 1.6[11] | 1.6[11] | 0.8[13] | 0.8[13] | 0.8[13] | | | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Crane | $C_{cr}$ | | 1.4 | 1.4 | 1.4 | 1.4 | 1.4 | 1.4 | 1.4[15] | | 1.0 | 1.0 | | | | 1.0 | |
| Wind | W [2] | | | | | | | | | 1.0[17] | | | | | | | |
| Extreme Winter Precipitation Event | $S_E$ | | | | | | | | | | 1.0 | | | | | | |
| Operating Earthquake | $E_o$ | | | | | | | | 1.6 | | | | | | | | |
| Safe Shutdown Earthquake | $E_{ss}$ | | | | | | | | | | | 1.0 | | | | | 1.0 |
| Tornado Wind Pressure | $W_{tw}$ [2] | | | | | | | | | | | | | 1.0 | | | |
| Tornado - Generated Differential Pressure | $W_{tp}$[2] | | | | | | | | | | | | 1.0 | 0.5 | 0.5 | | |
| Tornado - Generated Missiles | $W_{tm}$ [2] | | | | | | | | | | | | | 1.0 | | | |
| Accident Internal Flooding | $F_a$ | | | | | | | | | | | | | | | 1.0 | 1.0 |
| Accident Pressure | $P_a$ | | | | | | | | | | | | | | | 1.4 [4] | 1.0 |
| Accident Thermal | $T_a$ | | | | | | | | | | | | | | | 1.0 | 1.0 |
| Accident Pipe Reactions | $R_a$ | | | | | | | | | | | | | | | 1.0 | 1.0 |
| Accident Pipe Break Reactions | $Y_r$ | | | | | | | | | | | | | | | | 1.0 |


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**Table 6.4-5 Load Combinations and Load Factors for Concrete SR Substructures**
**(Sheet 2 of 2)**

| | | Normal and Severe | | | | | | | | | Extreme and Abnormal | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Design Load Cases[1][5]** | | C-1 | C-2 | C-3 | C-4 | C-5 | C-6 | C-7 | C-8 | C-9 | C-9[6] | C-10 | C-11 | C-12 | C-13 | C-16 | C-17 |
| **Load Combination Equation No. in ACI349-13** | | 9-1 | 9-2 | 9-2 | 9-2 | 9-3 | 9-3 | 9-3 | 9-4 | 9-5 | 9-6 | 9-6 | 9-7 | 9-7 | 9-7 | 9-8 | 9-9 |
| Accident Pipe Break Jet Impingement | $Y_j$ [8] | | | | | | | | | | | | | | | | 1.0 |
| Accident Pipe Break Missile Impact | $Y_m$ | | | | | | | | | | | | | | | | 1.0 |

Notes

(1) Refer to Section 6.4.2.2 for descriptions of individual Design Loads.

(2) Refer to Section 6.4.2.2.3 for guidance on consideration of directional effects due to wind pressure and missiles.

(3) Load combinations presented above are evaluated with a 0.9 factor on the dead load component to assess the adverse effects of the reduced dead load. For any other load (e.g., Live) if the load reduces the effects of other loads, the corresponding factor for that load is 0.9 of the assigned factors, if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the factor for that load is zero.

(4) The load factor is taken as 1.4 (in lieu of 1.2 in ACI 349-13) in accordance with Regulatory Position 5 in RG 1.142, and 1.2 in RG 1.199.

(5) The load combinations presented in this table are more severe than or in direct agreement with ACI 349-13.

(6) This load combination and its associated load factors are derived from ACI 349-13, with amendments in accordance with Interim Staff Guidance DC/COL-ISG-007.

(7) Not Used

(8) In accordance with Regulatory Position 5 in RG 1.142, factored and combined as per $Y_j$.

(9) The load factor is taken as 1.0 (in lieu of 1.4 in ACI 349-13) in accordance with Regulatory Position 5 in RG 1.142.

(10) The load factor is taken as 1.6 (in lieu of 1.2 in ACI 349-13) in accordance with Regulatory Position 5 in RG 1.142.

(11) The load factor of 1.6 To is added in accordance with Regulatory Position 5 in RG 1.142.

(12) The load factor is taken as 0.8 (in lieu of 1.2 in ACI 349-13) in accordance with Regulatory Position 5 in RG 1.142.

(13) The load factor of 0.8 To is added in accordance with Regulatory Position 5 in RG 1.142.

(14) The load factor is taken as 1.6 (in lieu of 1.2 in ACI 349-13) in accordance with Regulatory Position 5 in RG 1.142.

(15) The load factor of 1.4 $C_{cr}$ To is added in accordance with Regulatory Position 5 in RG 1.142.

(16) The load factor is taken as 1.0 (in lieu of 0.8 in ACI 349-13) in accordance with Regulatory Position 5 in RG 1.142.

(17) In ASCE/SEI 7-16, wind loads are consistent with strength-level design; a wind load factor of 1.0 is appropriate (in lieu of 1.6 in ACI 349-13). The basic wind speed for Risk Category III and IV structures in ASCE 7-16 is based on a 1700-year return period and, therefore, does not require the load factor of 1.6


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**Table 6.4-6 Load Combinations and Load Factors for Structural Steel of SR Substructures**
**(Sheet 1 of 2)**

| | | Normal and Severe | | | | | | | | | | | | | | | | | Extreme and Abnormal | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Design Load Cases[1]** | | **S-1** | **S-2** | **S-3** | **S-4** | **S-5(5)** | **S-6** | **S-7** | **S-8** | **S-9[5]** | **S-10** | **S-11** | **S-12** | **S-13[5]** | **S-14** | **S-15** | **S-16** | **S-17[6]** | **S-18** | **S-19[8]** | **S-20[8]** | **S-23** | **S-24** | **S-25** |
| **Load Combination Equation in AISC N690-18** | | **NB2-1** | **NB2-2** | **NB2-2** | **NB2-2** | **See Note (5)** | **NB2-3** | **NB2-3** | **NB2-3** | **NB2-3** | **NB2-4** | **NB2-4** | **NB2-4** | **See Note (5)** | **NB2-5** | **NB2-5** | **NB2-5** | **NB2-6** | **NB2-6** | **NB2-7** | **NB2-7** | **NB2-8** | **NB2-9** | **NB2-9** |
| Dead | $D^{(3)(4)}$ | 1.4 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Live | L | | 1.6 | 1.6 | 1.6 | 1.6 | 0.8 | 0.8 | 0.8 | 0.8 | $1.6^{(10)}$ | $1.6^{(10)}$ | $1.6^{(10)}$ | $1.6^{(10)}$ | $1.6^{(10)}$ | $1.6^{(10)}$ | $1.6^{(10)}$ | 0.8 | 0.8 | 0.8 | 0.8 | 0.8 | 0.8 | 0.8 |
| Roof Live | $L_r$ | | 0.5 | | | | 1.6 | | | | 0.5 | | | | 0.2 | | | | | | | | | |
| Snow | S | | | 0.5 | | 0.5 | | 1.6 | | 1.6 | | 0.5 | | 0.5 | | 0.2 | | | | | | | | |
| Rain | R | | | | 0.5 | | | | 1.6 | | | | 0.5 | | | | 0.2 | | | | | | | | |
| Atmospheric Ice, Self-weight | $D_i$ | | | | | 0.2 | | | | 0.7 | | | | 1.0 | | | | | | | | | | |
| Atmospheric Ice, Concurrent Wind Loading | $W_i^{(2)}$ | | | | | | | | | | | | | 1.0 | | | | | | | | | | |
| Hydrostatic | F | 1.4 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Soil (not applicable for steel structures) | H | | 1.6 | 1.6 | 1.6 | | 0.8 | 0.8 | 0.8 | 0.8 | 1.6 | 1.6 | 1.6 | | 1.6 | 1.6 | 1.6 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Pipe Reactions | $R_o$ | 1.4 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.0 | 1.0 | 1.0 | 1.0 | | | |
| Thermal | $T_o$ | 1.0 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.2 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | | | |
| Crane | $C_{cr}$ | 1.0 | 1.4 | 1.4 | 1.4 | 1.4 | 1.4 | 1.4 | 1.4 | 1.4 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | | | 1.0 | | |
| Wind | $W^{(3)}$ | | | | | | | | | | 1.0 | 1.0 | 1.0 | | | | | | | | | | | |
| Extreme Winter Precipitation Event | $S_E$ | | | | | | | | | | | | | | | | | 1.0 | | | | | | |
| Safe Shutdown Earthquake | $E_{ss}$ | | | | | | | | | | | | | | | | | | 1.0 | | | | 0.7 | 0.7 |
| Operating Earthquake | $E_o$ | | | | | | | | | | | | | | 1.6 | 1.6 | 1.6 | | | | | | | |
| Tornado Wind Pressure | $W_{tw}^{(2)}$ | | | | | | | | | | | | | | | | | | | 1.0 | | | | |
| Tornado - Generated Differential Pressure | $W_{tp}^{(2)}$ | | | | | | | | | | | | | | | | | | | 0.5 | 1.0 | | | |
| Tornado - Generated Missiles | $W_{tm}^{(2)}$ | | | | | | | | | | | | | | | | | | | 1.0 | | | | |
| Accident Internal Flooding | $F_a$ | | | | | | | | | | | | | | | | | | | | | 1.0 | 1.0 | 1.0 |
| Accident Pressure | $P_a$ | | | | | | | | | | | | | | | | | | | | | $1.4^{(11)}$ | 1.0 | 1.0 |
| Accident Thermal | $T_a$ | | | | | | | | | | | | | | | | | | | | | 1.0 | 1.0 | 1.0 |
| Accident Pipe Reactions | $R_a$ | | | | | | | | | | | | | | | | | | | | | 1.0 | 1.0 | 1.0 |
| Accident Pipe Break Reactions | $Y_r$ | | | | | | | | | | | | | | | | | | | | | | | 1.0 |


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**Table 6.4-6 Load Combinations and Load Factors for Structural Steel of SR Substructures**
**(Sheet 2 of 2)**

| | | Normal and Severe | | | | | | | | | | | | | | | | | Extreme and Abnormal | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Design Load Cases[1]** | | **S-1** | **S-2** | **S-3** | **S-4** | **S-5(5)** | **S-6** | **S-7** | **S-8** | **S-9[5]** | **S-10** | **S-11** | **S-12** | **S-13[5]** | **S-14** | **S-15** | **S-16** | **S-17[6]** | **S-18** | **S-19[8]** | **S-20[8]** | **S-23** | **S-24** | **S-25** |
| **Load Combination Equation in AISC N690-18** | | **NB2-1** | **NB2-2** | **NB2-2** | **NB2-2** | **See Note (5)** | **NB2-3** | **NB2-3** | **NB2-3** | **NB2-3** | **NB2-4** | **NB2-4** | **NB2-4** | **See Note (5)** | **NB2-5** | **NB2-5** | **NB2-5** | **NB2-6** | **NB2-6** | **NB2-7** | **NB2-7** | **NB2-8** | **NB2-9** | **NB2-9** |
| Accident Pipe Break Jet Impingement | $Y_i$ [7] | | | | | | | | | | | | | | | | | | | | | | | 1.0 |
| Accident Pipe Break Missile Impact | $Y_m$ | | | | | | | | | | | | | | | | | | | | | | | 1.0 |

Notes:

(1) Refer to Section 6.4.2.2 for descriptions of individual Design Loads.

(2) Refer to Seismic Design Criteria for guidance on consideration of seismic orthogonal effects and inclusion of portions of live and snow load in seismic mass.

(3) Refer to Section 6.4.2.2.3 for guidance on consideration of directional effects due to wind pressure and missiles.

(4) If the Dead Load acts to stabilize the structure against the destabilizing effects of lateral force or uplift, the load factor on Dead Load shall be 0.90 of the assigned factors and that on other gravity loads (e.g., L, $L_r$, S, $C_{cr}$) shall be zero.

(5) The load combinations presented in this table are more severe than or in direct agreement with AISC N690-18.

(6) This load combination and its associated load factors are derived from AISC N690-18, with amendments per ASCE 7, Section 2.3.3.

(7) This load combination and its associated load factors are derived from AISC N690-18, with amendments in accordance with Interim Staff Guidance DC/COL-ISG-007.

(8) In accordance with Regulatory Position 5 in RG 1.142, other (non-seismic) hydrodynamic loads are taken as per $Y_j$.

(9) Not Used

(10) This load combination is per AISC N690-18, NB2.5d.(6).

(11) The load factor is taken as 1.6 (in lieu of 0.8 in ANSI/AISC N690-18) in accordance with RG 1.243.

(12) The load factor is taken as 1.4 (in lieu of 1.2 in ANSI/AISC N690-18) in accordance with RG 1.243.

(13) The load combinations are in accordance with RG 1.221


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**Table 6.4-7 Load Combinations for Extraordinary Events on NSRST Substructure Analysis of Concrete Structures (Sheet 1 of 2)**

| **Design Load Cases** [1][5] | | **C-9**[6] | **C-11** | **C-12** | **C-13** | **C-16** | **C-17** |
|---|---|---|---|---|---|---|---|
| **Load Combination Equation No. in ACI 349-13** | | **9-6** | **9-7** | **9-7** | **9-7** | **9-8** | **9-9** |
| Dead | D [3] | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Live | L | 1.0[10] | 1.0[10] | 1.0[10] | 1.0[10] | 1.0[10] | 1.0[10] |
| Hydrostatic | F | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Soil | H | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Pipe Reactions | $R_o$ | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Thermal | $T_o$ | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Crane | $C_{cr}$ | 1.0 | | | | 1.0 | |
| Extreme Winter Precipitation Event | $S_E$ | 1.0 | | | | | |
| Safe Shutdown Earthquake | $E_{ss}$ | | | | | | 1.0 |
| Tornado Wind Pressure | $W_{tw}$ [2] | | | 1.0 | | | |
| Tornado - Generated Differential Pressure | $W_{tp}$[2] | | 1.0 | 0.5 | 0.5 | | |
| Tornado - Generated Missiles | $W_{tm}$ [2] | | | 1.0 | | | |
| Accident Internal Flooding | $F_a$ | | | | | 1.0 | 1.0 |
| Accident Pressure | $P_a$ | | | | | 1.4 [4] | 1.0 |
| Accident Thermal | $T_a$ | | | | | 1.0 | 1.0 |
| Accident Pipe Reactions | $R_a$ | | | | | 1.0 | 1.0 |
| Accident Pipe Break Reactions | $Y_r$ | | | | | | 1.0 |


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**Table 6.4-7 Load Combinations for Extraordinary Events on NSRST Substructure Analysis of Concrete Structures (Sheet 2 of 2)**

| **Design Load Cases** [1][5] | | **C-9**[6] | **C-11** | **C-12** | **C-13** | **C-16** | **C-17** |
|---|---|---|---|---|---|---|---|
| **Load Combination Equation No. in ACI 349-13** | | **9-6** | **9-7** | **9-7** | **9-7** | **9-8** | **9-9** |
| Accident Pipe Break Jet Impingement | $Y_j$ [9] | | | | | | 1.0 |
| Accident Pipe Break Missile Impact | $Y_m$ | | | | | | 1.0 |

Notes:

(1) Refer to Section 6.4.2.2 for description of individual design loads.

(2) Refer to Section 6.4.2.2.3 for discussion of directional effects due to wind pressure and missiles.

(3) Load combinations presented above are evaluated with a 0.9 factor on the dead load component to assess the adverse effects of the reduced dead load. For any other load (e.g., live) if the load reduces the effects of other loads, the corresponding factor for that load is 0.9 of the assigned factors, if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the factor for that load is zero.

(4) The load factor is taken as 1.4 (in lieu of 1.2 in ACI 349-13) in accordance with Regulatory Position 5 in RG 1.142, and Section 1.2 in RG 1.199.

(5) Load combinations presented in this table are more severe than or in direct agreement with ACI 349-13.

(6) This load combination and its associated load factors are derived from ACI 349-13, with amendments per Interim Staff Guidance DC/COL-ISG-007.

(7) Not Used

(8) The load combinations are in accordance with RG 1.221.

(9) In accordance with Regulatory Position 5 Table in RG 1.142, factored and combined as per $Y_j$.

(10) The load factor is taken as 1.0 (in lieu of 0.8 in ACI 349-13) in accordance with Regulatory Position 5 in RG 1.142.


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**Table 6.4-8 Load Combinations for Extraordinary Events on NSRST Substructure Analysis of Steel Structures (Sheet 1 of 2)**

| **Design Load Cases** [1] | | **S-17**[5] | **S-19** | **S-20** | **S-23** | **S-24**[8] | **S-25** |
|---|---|---|---|---|---|---|---|
| **Load Combination Equation in AISC N690-18** | | **NB2-6** | **NB2-7** | **NB2-7** | **NB2-8** | **NB2-9** | **NB2-9** |
| Dead | D [3][4] | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Live | L | 0.8 | 0.8 | 0.8 | 0.8 | 0.8 | 0.8 |
| Hydrostatic | F | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Soil (not applicable for steel structures) | H | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Pipe Reactions | $R_o$ | 1.0 | 1.0 | 1.0 | | | |
| Thermal | $T_o$ | 1.0 | 1.0 | 1.0 | | | |
| Crane | $C_{cr}$ | 1.0 | | | 1.0 | | |
| Extreme Winter Precipitation Event | $S_E$ | 1.0 | | | | | |
| Safe Shutdown Earthquake | $E_{ss}$ | | | | | 0.7 | 0.7 |
| Tornado Wind Pressure | $W_{tw}$ [2] | | 1.0 | | | | |
| Tornado - Generated Differential Pressure | $W_{tp}$[2] | | 0.5 | 1.0 | | | |
| Tornado - Generated Missiles | $W_{tm}$ [2] | | 1.0 | | | | |
| Accident Internal Flooding | $F_a$ | | | | 1.0 | 1.0 | 1.0 |
| Accident Pressure | $P_a$ | | | | 1.4 [9] | 1.0 | 1.0 |


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**Table 6.4-8 Load Combinations for Extraordinary Events on NSRST Substructure Analysis of Steel Structures (Sheet 2 of 2)**

| **Design Load Cases** [1] | | **S-17**[5] | **S-19** | **S-20** | **S-23** | **S-24**[8] | **S-25** |
|---|---|---|---|---|---|---|---|
| **Load Combination Equation in AISC N690-18** | | **NB2-6** | **NB2-7** | **NB2-7** | **NB2-8** | **NB2-9** | **NB2-9** |
| Accident Thermal | | | | | | | |
| Accident Pipe Reactions | $R_a$ | | | | 1.0 | 1.0 | 1.0 |
| Accident Pipe Break Reactions | $Y_r$ | | | | | | 1.0 |
| Accident Pipe Break Jet Impingement | $Y_j$ (6) | | | | | | 1.0 |

Notes:

(1) Refer to Section 6.4.2.2 for descriptions of individual Design Loads.

(2) Refer to Section 6.4.2.2.3 for discussion of directional effects due to wind pressure and missiles.

(3) If the dead load acts to stabilize the structure against the destabilizing effects of lateral force or uplift, the load factor on dead load is 0.90 of the assigned factors and for other gravity loads (e.g., L, $L_r$, S, $C_{cr}$) is zero.

(4) Load combinations presented in this table are more severe than or in direct agreement with AISC N690-18.

(5) This load combination and its associated load factors are derived from AISC N690-18, with amendments in accordance with Interim Staff Guidance DC/COL-ISG-007.

(6) In accordance with Regulatory Position 5 in RG 1.142, other (non-seismic) hydrodynamic loads are taken as $Y_j$.

(7) Not Used

(8) Load combination is in accordance with AISC N690-18, NB2.5d.(6).

(9) The load factor is taken as 1.4 (in lieu of 1.2 in AISC N690-18) in accordance with RG 1.243.


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**Table 6.4-9 Load Factors and Factors of Safety for SR Structural Stability**

| **Design Load Cases** [1] | | **St-1** | **St-2** | **St-3** | **St-4** | **St-7** |
|---|---|---|---|---|---|---|
| Dead | D | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Soil | H | 1.0 | 1.0 | 1.0 | 1.0 | |
| Wind | W[2] | | 1.0 | | | |
| Safe Shutdown Earthquake | $E_{ss}$ | 1.0 | | | | 1.0[5] |
| Tornado Wind Pressure | $W_{tw}$[2] | | | 1.0 | 1.0 | |
| Tornado - Generated Differential Pressure | $W_{tp}$ | | | 0.5 | | |
| Tornado - Generated Missiles | $W_{tm(2)}$ | | | 1.0 | | |
| Buoyancy | F’ | | | | | 1.0 |
| **Minimum Factors of Safety** | | **St-1[4]** | **St-2** | **St-3** | **St-4** | **St-7** |
| Overturning Factor of Safety | 1.1 | 1.5 | 1.1 | 1.1 | -- | |
| Sliding Factor of Safety | 1.1 | 1.5 | 1.1 | 1.1 | -- | |
| Floatation Factor of Safety | -- | -- | -- | -- | 1.1 | |

Notes:

(1) Refer to Section 6.4.2.2 for descriptions of individual design loads.

(2) Refer to Section 6.4.2.2.3 for discussion of directional effects due to wind pressure and missiles.

(3) Not Used

(4) The overturning factor of safety, sliding factor of safety and Floatation Factor of Safety is 1.5 when OBE is used and 1.1 when SSE is used.

(5) Vertical earthquake uplift is added to buoyancy using SSE.


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**Table 6.4-10 Material Properties of Uncracked and Cracked Concrete**

| | Uncracked Concrete | Cracked Concrete |
|---|---|---|
| Young's Modulus (E) | 580,392 ksf | 820,670 ksf |
| Shear Modulus (G) | 247,968 ksf | 350,626 ksf |
| Poisson's Ratio | 0.17 | 0.17 |
| Density | 0.150 kcf | 0.212 kcf |



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**Table 6.4-11 Load Combinations for Extraordinary Events on NSRST Superstructure Analysis of Concrete Structures (Sheet 1 of 2)**

| **Design Load Cases** [1][5] | | **C-9**[6] | **C-11** | **C-12** | **C-13** | **C-16** | **C-17** |
|---|---|---|---|---|---|---|---|
| **Load Combination Equation No. in ACI 349-13** | | 9-6 | 9-7 | 9-7 | 9-7 | 9-8 | 9-9 |
| Dead | $D$[3] | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Live | $L$ | 1.0[9] | 1.0[9] | 1.0[9] | 1.0[9] | 1.0[9] | 1.0[9] |
| Hydrostatic | $F$ | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Soil | $H$ | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Pipe Reactions | $R_o$ | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Thermal | $T_o$ | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Crane | $C_{cr}$ | 1.0 | | | | 1.0 | |
| Extreme Winter Precipitation Event | $S_E$ | 1.0 | | | | | |
| Safe Shutdown Earthquake | $E_{ss}$ | | | | | | 1.0 |
| Tornado Wind Pressure | $W_{tw}$ [2] | | | 1.0 | | | |
| Tornado - Generated Differential Pressure | $W_{tp}$[2] | | 1.0 | 0.5 | 0.5 | | |
| Tornado - Generated Missiles | $W_{tm}$ [2] | | | 1.0 | | | |
| Accident Internal Flooding | $F_a$ | | | | | 1.0 | 1.0 |
| Accident Pressure | $P_a$ | | | | | 1.4 [4] | 1.0 |
| Accident Thermal | $T_a$ | | | | | 1.0 | 1.0 |
| Accident Pipe Reactions | $R_a$ | | | | | 1.0 | 1.0 |


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**Table 6.4-11 Load Combinations for Extraordinary Events on NSRST Superstructure Analysis of Concrete Structures (Sheet 2 of 2)**

| **Design Load Cases** [1][5] | | **C-9**[6] | **C-11** | **C-12** | **C-13** | **C-16** | **C-17** |
|---|---|---|---|---|---|---|---|
| **Load Combination Equation No. in ACI 349-13** | | 9-6 | 9-7 | 9-7 | 9-7 | 9-8 | 9-9 |
| Accident Pipe Break Reactions | $Y_r$ | | | | | | 1.0 |
| Accident Pipe Break Jet Impingement | $Y_j$ [8] | | | | | | 1.0 |
| Accident Pipe Break Missile Impact | $Y_m$ | | | | | | 1.0 |

Notes:

(1) Refer to Section 6.4.2.2 for descriptions of individual design loads.

(2) Refer to Section 6.4.2.2.3 for guidance on consideration of directional effects due to wind pressure and missiles.

(3) Load combinations presented are evaluated with a 0.9 factor on the dead load component to assess the adverse effects of the reduced dead load. For any other load (e.g., live) if the load reduces the effects of other loads, the corresponding factor for that load is 0.9 of the assigned factors, if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the factor for that load is zero.

(4) Load factor is 1.4 (in lieu of 1.2 in ACI 349-13) are in accordance with Regulatory Position 5 in RG 1.142, and Section 1.2 in RG 1.199.

(5) Load combinations presented in this table are more severe than or in direct agreement with ACI 349-13.

(6) This load combination and associated load factors are derived from ACI 349-13, with amendments per Interim Staff Guidance DC/COL-ISG-007.

(7) Not Used

(8) Per Regulatory Position 5 Table in RG 1.142, factored and combined as per Yj.

(9) Load factor is 1.0 (in lieu of 0.8 in ACI 349-13) are in accordance with Regulatory Position 5 in RG 1.142.


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**Table 6.4-12 Load Combinations for Extraordinary Events on NSRST Superstructure Analysis of Steel Structures (Sheet 1 of 2)**

| **Design Load Cases** [1] | | **S-17**[5] | **S-19** | **S-20** | **S-23** | **S-24**[8] | **S-25** |
|---|---|---|---|---|---|---|---|
| **Load Combination Equation in AISC N690-18** | | NB2-6 | NB2-7 | NB2-7 | NB2-8 | NB2-9 | NB2-9 |
| Dead | $D^{(3)(4)}$ | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Live | L | 0.8 | 0.8 | 0.8 | 0.8 | 0.8 | 0.8 |
| Hydrostatic | F | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Soil (not applicable for steel structures) | H | 1.0 | 1.0 | 1.0 | 1.0 | | |
| Pipe Reactions | $R_o$ | 1.0 | 1.0 | 1.0 | | | |
| Thermal | $T_o$ | 1.0 | 1.0 | 1.0 | | | |
| Crane | $C_{cr}$ | 1.0 | | | 1.0 | | |
| Extreme Winter Precipitation Event | $S_E$ | 1.0 | | | | | |
| Safe Shutdown Earthquake | $E_{ss}$ | | | | | 0.7 | 0.7 |
| Tornado Wind Pressure | $W_{tw}$ [2] | | 1.0 | | | | |
| Tornado - Generated Differential Pressure | $W_{tp}$[2] | | 0.5 | 1.0 | | | |
| Tornado - Generated Missiles | $W_{tm}$ [2] | | 1.0 | | | | |
| Accident Internal Flooding | $F_a$ | | | | 1.0 | 1.0 | 1.0 |
| Accident Pressure | $P_a$ | | | | 1.4 [9] | 1.0 | 1.0 |
| Accident Thermal | $T_a$ | | | | 1.0 | 1.0 | 1.0 |
| Accident Pipe Reactions | $R_a$ | | | | 1.0 | 1.0 | 1.0 |


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**Table 6.4-12 Load Combinations for Extraordinary Events on NSRST Superstructure Analysis of Steel Structures (Sheet 2 of 2)**

| **Design Load Cases** [1] | | **S-17**[5] | **S-19** | **S-20** | **S-23** | **S-24**[8] | **S-25** |
|---|---|---|---|---|---|---|---|
| **Load Combination Equation in AISC N690-18** | | NB2-6 | NB2-7 | NB2-7 | NB2-8 | NB2-9 | NB2-9 |
| Accident Pipe Break Reactions | $Y_r$ | | | | | | 1.0 |
| Accident Pipe Break Jet Impingement | $Y_j$ [6] | | | | | | 1.0 |
| Accident Pipe Break Missile Impact | $Y_m$ | | | | | | 1.0 |

Notes:

(1) Descriptions of individual design loads are provided in Section 6.4.2.2.

(2) Refer to Section 6.4.2.2.3 for guidance on consideration of directional effects due to wind pressure and missiles.

(3) If the dead load acts to stabilize the structure against the destabilizing effects of lateral force or uplift, the load factor on dead load is 0.90 of the assigned factors and that on other gravity loads (e.g., L, $L_r$, S, $C_{cr}$) are zero.

(4) Load combinations presented in this table are more severe than or in accordance with AISC N690-18.

(5) Load combination and associated load factors are derived from AISC N690-18, with amendments in accordance with Interim Staff Guidance DC/COL ISG-007.

(6) In accordance with Regulatory Position 5 in RG 1.142, other (non-seismic) hydrodynamic loads are $Y_j$.

(7) Not Used

(8) This load combination is per AISC N690-18, NB2.5d.(6).

(9) Load factor is 1.4 (in lieu of 1.2 in AISC N690-18) are in accordance with RG 1.243.


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**Table 6.4-13 Design Basis Tornado Parameters**

| Tornado Parameters | Values | Unit |
|---|---|---|
| Maximum Wind Speed | 160 | miles per hour |
| Translational Speed | 32 | miles per hour |
| Maximum Rotational Speed | 128 | miles per hour |
| Radius of Maximum Rotational Speed | 150 | feet |
| Pressure Drop | 0.6 | pounds per square inch |
| Rate of Pressure Drop | 0.2 | pounds per square inch per second |


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**Table 6.4-14 Design Basis Tornado Missile Spectrum and Maximum Speed**

| Missile | Dimension | Mass (pounds) | Horizontal Velocity (feet per second) | Vertical Velocity (feet per second) |
|---|---|---|---|---|
| Schedule 40 Pipe | 6.625 in diameter × 15 ft long | 287 | 79 | 53 |
| Automobile | 14.9ft × 5.6ft × 4.9ft | 2,595 | 79 | 53 |
| Solid Steel Sphere | 1 in diameter | 0.147 | 20 | 14 |



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**Table 6.4-15 Applicability of Tornado and Extreme Winter Precipitation Events to Safety-Significant SSCs**

| SSC | | | Tornado Wind[3] | Small Tornado Missile Protection[2] | Automobile Tornado Missile Protection[2] | Extreme Winter Precipitation Event |
|---|---|---|---|---|---|---|
| RXB | Substructure | | Yes | Yes | Yes | N/A |
| | Superstructure (NST) | Column | Yes | Yes | Yes | Yes |
| | | Building Envelope | No | No | No | Yes |
| | Reactor Air Cooling System | | Yes | Yes[1] | Yes[1] | Yes |
| FHB[2] | Substructure | | Yes | No | No | N/A |
| | Superstructure | Column | Yes | Yes | Yes | Yes |
| | | Building Envelope | No | No | No | Yes |
| RAB[2] | Substructure | | N/A | No | No | N/A |
| | Superstructure | Column | Yes | Yes | Yes | Yes |
| | | Building Envelope | No | No | Yes | Yes |
| NCB | Substructure | | Yes | Yes | Yes | N/A |
| | Superstructure | Column | Yes | Yes | Yes | Yes |
| | | Building Envelope | No | No | No | Yes |

Notes:

1. RAC stacks and supports are permitted to sustain localized damage due to tornado missile impacts provided: (1) airflow through RAC remains at a sufficient level to maintain cooling requirements, and (2) analysis shows the undamaged portions of the RAC can still withstand tornado winds.
2. Solid radioactive waste processing, liquid radioactive waste processing, and gaseous radioactive waste processing equipment located within the FHB are appropriately protected as needed from external missiles. The RAB structure is designed to protect the sodium piping and associated systems and components containing sodium from automobile missiles.
3. The Building Envelope is designed for normal wind loading as the building is considered as fully enclosed during normal design wind event. The Building Envelope is not credited to protect the building interior from tornado loading conditions.


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**Table 6.4-16 Summary of Design Transients**

| Event ID | Event Name |
|---|---|
| A.1 | Cold Shutdown to Hot Standby |
| A.2 | Hot Standby to Full Power |
| A.3 | Full Power to Cold Shutdown |
| A.4 | Hot Standby to Cold Shutdown |
| A.5 | Decrease in Reactor Power |
| A.6 | Increase in Reactor Power |
| A.7 | Partial Decrease in Reactor Power |
| A.8 | Partial Increase in Reactor Power |
| A. 9 | Runback Recovery |
| A.10 | Steady State Temperature Fluctuations |
| A.11 | Steady State Pressure Fluctuations |
| B.1 | Normal Reactor Scram |
| B.2 | Loss of One Primary Sodium Pump |
| B.3 | Loss of One Intermediate Sodium Loop |
| B.4 | Control Rod Induced Transient Overpower |
| B.5 | Loss of Offsite Power |
| B.6 | Reactor Power Runback |
| B.7 | Cover Gas Overpressure |
| B.8 | Primary Sodium Pump Overspeed |
| B.9 | Intermediate Sodium Pump Overspeed |
| C.1 | Loss of All Primary Sodium Pumps |
| C.2 | Intermediate Heat Exchanger Leak |
| D.1 | Extended Duration Intermediate Air Cooling |
| D.2 | Extended Duration Reactor Air Cooling |
| Test 1 | Hydrostatic or Pneumatic Test |
| Test 2 | Nuclear Island Dry Heatup and Fill |
| Test 3 | Nuclear Island Drain and Cooldown to Ambient |
| Test 4 | Intermediate System Dry Heatup and Fill |
| Test 5 | Intermediate System Drain and Cooldown to Ambient |
| Test 6 | Intermediate System Freezing and Melting |


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**Table 6.4-17 Applicable Loads for ASME Code Class A and B Components**

| Load | Description |
|---|---|
| $T_{design}$ | Design Temperature |
| $P_{design}$ | Design Pressure or Design Pressure Difference |
| Plant Transients | Transient temperatures, pressures, and flow rate as applicable |
| Safe Shutdown Earthquake | Seismic loads due to safe shutdown earthquake |
| Operating Basis Earthquake | Defined as 1/3 SSE. Considered for fatigue evaluation only |
| Dead Weight | Self-weight of component |
| Weight of Contents | Weight of sodium coolant or other process fluid |
| Buoyancy | Buoyancy forces |
| Preload | Loads due to bolt preload |
| Connected Piping | Mechanical and thermal loads due to attached piping |
| Mechanical | Mechanical reaction loads (including deadweight) induced by components or equipment other than piping. |
| Vibration | Vibratory loadings resulting from flow induced vibration or radiated vibration from equipment such as pumps. |
| Static Pressure Test | Hydrostatic or pneumatic test pressure in accordance with the requirements of HBB-6000 or HCB-6000 as applicable, and the weight of the test medium. |
| Support Motions | Loading applied to components supported at multiple locations resulting from differential thermal movement of their supports (thermal anchor motions). |


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**Table 6.4-18 Load Combinations and Service Stress Limits for ASME Code Class A and B Components**

| Condition | Load Combinations | Service Stress Limit |
|---|---|---|
| Design | $P_{design}$ + $T_{design}$(1) + DW + BOLT + WC + B + CP + MECH | Design |
| Service Level A Transients | PT + DW + BOLT + WC + B + CP + MECH + VIBR(2) + DISP | A |
| Service Level B Transients | PT + DW + BOLT + WC + B + CP + MECH + VIBR(2) + DISP | B |
| Service Level B OBE | PT(3) + DW + BOLT + WC + B + CP + MECH + VIBR(2) +/- OBE(4) + DISP | B |
| Service Level C Transients | PT + DW + BOLT + WC + B + CP + MECH + VIBR(2) + DISP | C |
| Service Level D Transients | PT + DW + BOLT + WC + B + CP + MECH | D |
| Service Level D SSE | PT(3) + DW + BOLT + WC + B + CP + MECH +/- SSE | D |
| Static Pressure Test | PTEST + DW + BOLT +CP + MECH | Test |
| Test (Other than static pressure tests) | PT + DW + BOLT + WC + B + CP + MECH | B |

Notes:

1. Design Temperature is for determination of material properties only.
2. Vibration loads are only applicable for fatigue analysis.
3. The PT conditions for seismic events (SSE and OBE) are steady state 100% power.
4. OBE loads are only applicable for fatigue analysis when OBE is less than or equal to one-third of SSE.


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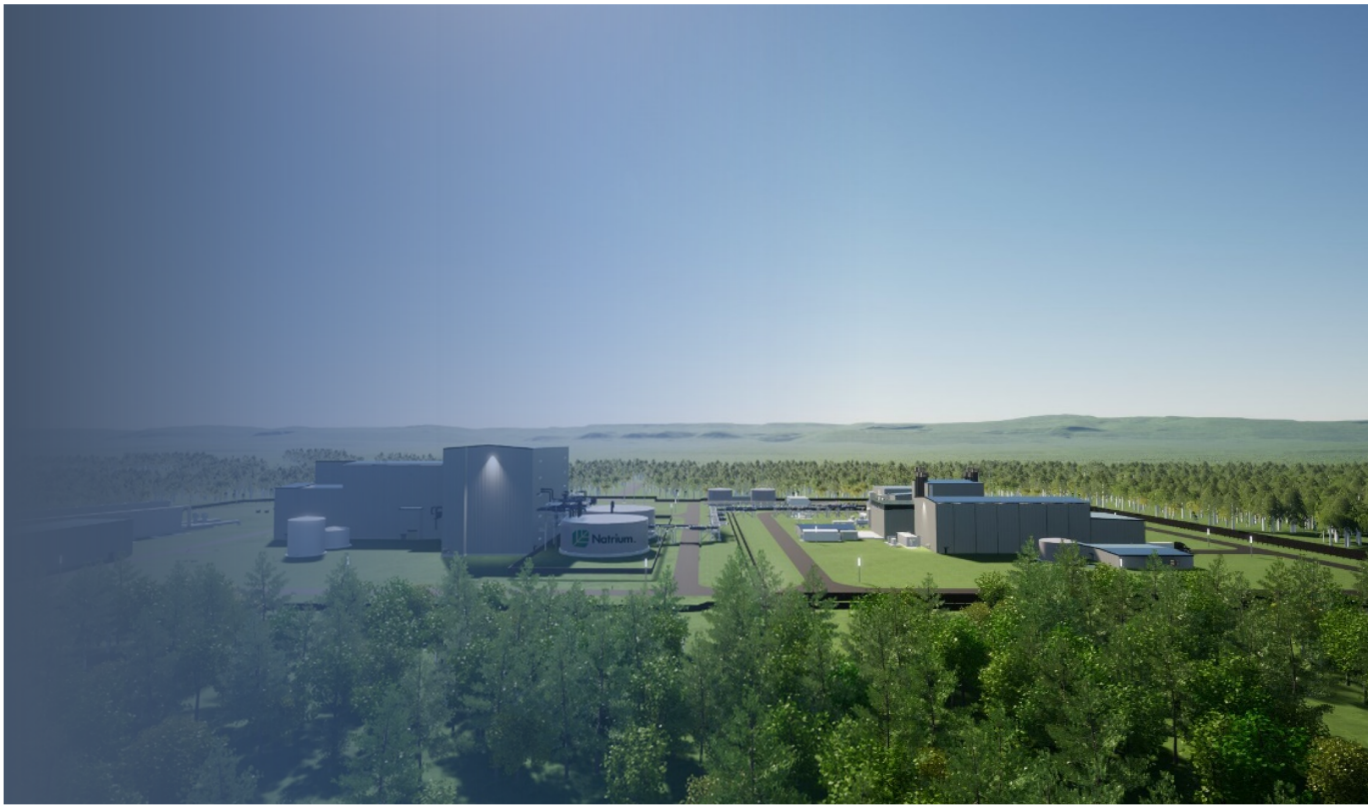

# Kemmerer Power Station Unit 1
# PSAR, Chapter 7

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## TABLE OF CONTENTS

**Chapter 7 Descriptions for Safety-Significant SSCs . . . . . 7.1-1**

7.1 Nuclear Heat Supply Systems . . . . . 7.1-1

7.1.1 Reactor Core System . . . . . 7.1-1

7.1.2 Reactor Enclosure System . . . . . 7.1-14

7.1.3 Primary Heat Transport System . . . . . 7.1-48

7.1.4 Intermediate Heat Transport System . . . . . 7.1-73

7.2 Reactor Auxiliary Systems . . . . . 7.2-1

7.2.1 Reactor Air Cooling System . . . . . 7.2-1

7.2.2 Intermediate Air Cooling System . . . . . 7.2-13

7.2.3 Sodium Cover Gas System . . . . . 7.2-22

7.2.4 Sodium Processing System . . . . . 7.2-41

7.2.5 Control Rod Drive System . . . . . 7.2-57

7.3 Fuel Handling Systems . . . . . 7.3-1

7.3.1 Water Pool Fuel Handling System . . . . . 7.3-1

7.3.2 Ex-Vessel Fuel Handling System . . . . . 7.3-16

7.3.3 In-Vessel Fuel Handling System . . . . . 7.3-38

7.4 Radwaste Systems . . . . . 7.4-1

7.4.1 Gaseous Radwaste Processing System . . . . . 7.4-1

7.5 Ancillary Systems . . . . . 7.5-1

7.5.1 NI HVAC System . . . . . 7.5-1

7.5.2 Nuclear Island Fire Protection System . . . . . 7.5-10

7.6 Integrated Controls Systems . . . . . 7.6-1

7.6.1 Instrumentation and Control Systems Introduction . . . . . 7.6-1

7.6.2 Nuclear Island Control System . . . . . 7.6-6

7.6.3 Reactor Protection System . . . . . 7.6-18

7.6.4 Nuclear Instrumentation System . . . . . 7.6-36


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## TABLE OF CONTENTS

7.6.5 Reactor Instrumentation System . . . . . 7.6-48
7.6.6 Radiation Monitoring System . . . . . 7.6-60
7.6.7 Control Room and Indications . . . . . 7.6-69
7.6.8 Anticipatory Automatic Seismic Trip System . . . . . 7.6-75
7.7 Electrical Systems . . . . . 7.7-1
7.7.1 Nuclear Island Auxiliary Electrical Systems . . . . . 7.7-1
7.8 Buildings and Structures . . . . . 7.8-1
7.8.1 Reactor Building . . . . . 7.8-1
7.8.2 Fuel Handling Building . . . . . 7.8-8
7.8.3 Reactor Auxiliary Building . . . . . 7.8-18
7.8.4 Nuclear Island Control Building . . . . . 7.8-27


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## LIST OF TABLES


Table 7.1.1-1 Reactor Core Design Parameters . . . . . 7.1-11

Table 7.1.2-1 RES Safety Classification and ASME Code Applicability . . . . . 7.1-33

Table 7.1.2-2 Applicable Codes and Standards . . . . . 7.1-34

Table 7.1.2-3 Preliminary RES Dimensions . . . . . 7.1-35

Table 7.1.2-4 Preliminary RES Design Temperature, Service Environment, and Materials of Construction . . . . . 7.1-36

Table 7.1.3-1 Monitored PHT Parameters . . . . . 7.1-63

Table 7.1.3-2 PSP Major Components Safety Classifications . . . . . 7.1-64

Table 7.1.3-3 PSP Preliminary Design Performance Characteristics . . . . . 7.1-65

Table 7.1.3-4 IHX Major Components Classifications . . . . . 7.1-66

Table 7.1.3-5 IHX Preliminary Design Performance Characteristics . . . . . 7.1-67

Table 7.1.3-6 Preliminary PHT Design Temperature, Service Environment, and Materials of Construction . . . . . 7.1-68

Table 7.1.4-1 IHT Safety and Seismic Classifications . . . . . 7.1-81

Table 7.1.4-2 Applicable Codes and Standards . . . . . 7.1-82

Table 7.2.1-1 RAC Safety Classifications . . . . . 7.2-9

Table 7.2.1-2 Applicable Codes and Standards . . . . . 7.2-10

Table 7.2.1-3 Preliminary Material Selection and Design Temperatures for RAC Components . . . . . 7.2-11

Table 7.2.2-1 IAC Safety and Seismic Classifications . . . . . 7.2-19

Table 7.2.2-2 Applicable Codes and Standards . . . . . 7.2-20

Table 7.2.5-1 CRD Safety and Seismic Classifications . . . . . 7.2-66

Table 7.2.5-2 Applicable Codes and Standards . . . . . 7.2-67

Table 7.2.5-3 Preliminary CRD Design Conditions . . . . . 7.2-68

Table 7.2.5-4 Preliminary CRD Materials of Construction . . . . . 7.2-69

Table 7.3.1-1 Primary FHP and Component Classifications . . . . . 7.3-12

Table 7.3.1-2 Codes and Standards . . . . . 7.3-13



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## LIST OF TABLES

Table 7.3.2-1 FHE Major Component Classifications . . . . . 7.3-33
Table 7.3.2-2 Codes and Standards . . . . . 7.3-34
Table 7.3.3-1 FHI Safety-Significant Components Classifications . . . . . 7.3-47
Table 7.3.3-2 Codes and Standards . . . . . 7.3-48
Table 7.5.1-1 Codes and Standards Applicable to Nuclear Island HVAC Systems . . . . 7.5-9
Table 7.5.2-1 NFP Design Codes . . . . . 7.5-18
Table 7.6.1-1 I&C System Classification and DL Functions . . . . . 7.6-5
Table 7.6.3-1 RPS Equipment and Classifications . . . . . 7.6-35
Table 7.6.5-1 RIS Variable Summary . . . . . 7.6-58


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## LIST OF FIGURES


Figure 7.1.1-1 Core Assembly Configuration . . . . . 7.1-12

Figure 7.1.1-2 Illustration of Reactor Core Loading Pattern . . . . . 7.1-13

Figure 7.1.2-1 Simplified Reactor Enclosure System General Arrangement . . . . . 7.1-37

Figure 7.1.2-2 Simplified Reactor Head Configuration . . . . . 7.1-38

Figure 7.1.2-3 Simplified Rotatable Plug Assembly General Arrangement . . . . . 7.1-39

Figure 7.1.2-4 Simplified Reactor Fixed Internals General Arrangement . . . . . 7.1-40

Figure 7.1.2-5 Simplified Core Barrel Structures General Arrangement . . . . . 7.1-41

Figure 7.1.2-6 Simplified Guard Vessel General Arrangement . . . . . 7.1-42

Figure 7.1.2-7 Simplified Reactor Support Structure Arrangement . . . . . 7.1-43

Figure 7.1.2-8 Primary Coolant Flowpath . . . . . 7.1-44

Figure 7.1.2-9 Sodium Pool Regions . . . . . 7.1-45

Figure 7.1.2-10 Simplified Diagram of Well Seal and HAA Seal . . . . . 7.1-46

Figure 7.1.3-1 Simplified PHT Flow Diagram . . . . . 7.1-69

Figure 7.1.3-2 Simplified RAC and PHT Process Flow Diagram . . . . . 7.1-70

Figure 7.1.3-3 Sectional View of Installed Primary Sodium Pump . . . . . 7.1-71

Figure 7.1.3-4 Intermediate Heat Exchangers . . . . . 7.1-72

Figure 7.1.4-1 Intermediate Loop Simplified Diagram . . . . . 7.1-83

Figure 7.2.1-1 Simplified RAC Airflow Diagram . . . . . 7.2-12

Figure 7.2.2-1 IAC Simplified Arrangement . . . . . 7.2-21

Figure 7.2.3-1 Primary Cover Gas System Block Diagram . . . . . 7.2-39

Figure 7.2.3-2 Intermediate Cover Gas System Block Diagram . . . . . 7.2-40

Figure 7.2.4-1 Block Flow Diagram of the Primary SPS . . . . . 7.2-54

Figure 7.2.4-2 Block Flow Diagram of the Intermediate SPS . . . . . 7.2-55

Figure 7.2.4-3 Block Flow Diagram of the EVST SPS . . . . . 7.2-56

Figure 7.2.5-1 CRD/RPA Cross-Sectional Layout . . . . . 7.2-70

Figure 7.3.1-1 FHP Top-Down View . . . . . 7.3-14




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## LIST OF FIGURES

Figure 7.3.1-2 SFP Side View . . . . . 7.3-15
Figure 7.3.2-1 Ex-Vessel Handling Machine Primary Components. . . . . 7.3-35
Figure 7.3.2-2 Bottom Loading Transfer Cask Primary Components . . . . . 7.3-36
Figure 7.3.2-3 Ex-Vessel Storage Tank Primary Components . . . . . 7.3-37
Figure 7.3.3-1 FHI General Arrangement within Reactor Vessel . . . . . 7.3-49
Figure 7.3.3-2 IVTM Primary Components and Motions . . . . . 7.3-50
Figure 7.3.3-3 FTL Primary Components . . . . . 7.3-51
Figure 7.6.2-1 Nuclear Island Control System Block Diagram . . . . . 7.6-17
Figure 7.6.4-1 XIS Overview . . . . . 7.6-47
Figure 7.6.5-1 Reactor Instrumentation System Block Diagram . . . . . 7.6-59
Figure 7.6.6-1 RMS Interface Diagram . . . . . 7.6-68
Figure 7.6.7-1 MCR Layout . . . . . 7.6-74
Figure 7.6.8-1 AST Signals and Interface . . . . . 7.6-80


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# Chapter 7 Descriptions for Safety-Significant SSCs

## 7.1 Nuclear Heat Supply Systems

Nuclear heat supply systems consist of the Reactor Core System (RCC), Reactor Enclosure System (RES), Primary Heat Transport System (PHT), Intermediate Heat Transport System (IHT), and Nuclear Island Salt System (NSS). These systems are involved in the generation of nuclear energy and transfer of heat from the reactor core to the Energy Island and are described in Section 7.1.1 through Section 7.1.4, except the NSS, which is described in Section 1.1.4.

### 7.1.1 Reactor Core System

#### 7.1.1.1 Summary Description

The RCC consists of replaceable core assemblies. The RCC is contained within the Reactor Vessel (RV) and is supported and constrained by the Core Barrel Structures (CBS). The RV and CBS are part of the RES, described in Section 7.1.2. Figure 7.1.2-1 provides a representative arrangement of the RCC within the RES.

The reactor is a pool-type, sodium-cooled, fast reactor which operates at near-atmospheric pressure, circulating liquid sodium coolant through its core, with heat transferred from the primary sodium coolant to an intermediate sodium loop via heat exchangers. The reactor core design is based on decades of U.S.-based Sodium Fast Reactor design and operational experience from the Experimental Breeder Reactor II, General Electric Hitachi Power Reactor Innovative Small Module, and the Fast Flux Test Facility in addition to over a decade of technology development from the TerraPower® TWR® technology and the Clinch River Breeder Reactor Program. The initial fuel for the core is sodium-bonded uranium - 10 weight percent Zirconium (U-10Zr) that is encapsulated by HT-9 ferritic-martensitic steel cladding. Historically, metallic fuel has been successfully demonstrated in the Experimental Breeder Reactor II program and metal fuel testing was conducted in the Fast Flux Test Facility reactor core.

The RCC is designed to sustain a nuclear chain reaction for fission heat generation. Thermal power and reactivity are controlled by a combination of fuel management operations on a cycle-by-cycle basis and vertical positioning of the control rods throughout the operating cycle.

When operating, the reactor core generates and transfers heat to the primary sodium coolant in the PHT, described in Section 7.1.3. The PHT primary sodium pumps supply sodium coolant flow to the core inlet plenum located at the bottom of the reactor core. The primary sodium coolant enters the core assemblies through their inlet nozzles, which interface with the receptacles located in the RES inlet plenum. The core assembly inlet nozzles, named universal stack nozzles, contain orifice plates that control the primary sodium coolant flow through each core assembly to satisfy cooling requirements, which vary by assembly. Some of the flow is designed to bypass the assemblies, providing cooling of core assemblies stored in the RES In-Vessel Storage. As the sodium coolant exits each core assembly, it mixes in the hot pool volume within the RV and flows past the RES Upper Internal Structure. The primary sodium then flows into the PHT Intermediate Heat Exchanger, where heat is transferred to the IHT (described in Section 7.1.4). The flow circuit is completed as the cooled primary sodium at the outlet of the


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Intermediate Heat Exchangers enters the Primary Sodium Pumps and is discharged back into the RES inlet plenum. Section 7.1.3 provides greater detail on the primary sodium coolant flowpath.

Preliminary reactor core design parameters are summarized in Table 7.1.1-1.

When the reactor is shut down for refueling, core assemblies are handled using the In-Vessel Transfer Machine described in Section 7.3.3. Fuel assemblies that have reached their burnup targets and failed fuel assemblies (if any) are placed in the RES In-Vessel Storage for cooling prior to removing from the RV for longer-term storage.

There are various core assembly types, positioned within the core based on the assembly function. Each assembly is removable to allow for relocation and replacement, and contains unique identification. The assembly types have similar shape and external configurations. Common features of core assemblies include an inlet nozzle, a duct tube, load pads, and a handling socket. The assembly duct and its load pad features, in conjunction with the core restraint system, described in Section 7.1.1.3, are designed to enable the core restraint function as needed for proper core reactivity response while also providing adequate inter-assembly clearances for fuel management during refueling operations. Assembly internal configurations are specific to the particular assembly type. A representative core assembly configuration is shown in Figure 7.1.1-1. The preliminary design arrangement of core assemblies within the core is depicted in Figure 7.1.1-2.

Control assemblies are described in Section 7.1.1.3.1 and NAT-2806, Revision 0, “Natrium Topical Report: Fuel and Control Assembly Qualification” (Reference 7.1.1-1). NAT-2806 is incorporated by reference in its entirety into the SAR. See Section 1.4.2 for a list of documents incorporated by reference into the SAR. Other core assembly types are described below.

Fuel Assembly

The reactor fuel is contained within fuel pins arranged in bundles and housed in fuel assemblies. The fuel assembly contains the fuel, produces heat, provides the neutron flux environment, and can be removed from or shuffled in the reactor core during refueling. The fuel assembly is classified safety-related (SR) and designed to:

- Position the fuel properly in the reactor core for controlled nuclear reaction and generation of thermal power
- Provide passages to guide and control the primary sodium coolant for heat removal
- Provide shielding to protect components outside of the core from excessive fluence
- Provide features for proper interfacing with other core assemblies, the CBS, the In-Vessel Storage, and shipping and handling equipment
- Provide physical barriers from fission products to primary coolant, serving as part of the functional containment described in Section 1.3.2.1

The fuel assembly contains a fuel pin bundle with its attachments. Figure 5-3 in NAT-2806 provides a representative depiction of the features of a fuel assembly.


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The initial fuel assembly uses a sodium-bonded metal fuel, which is described in NAT-2806. The pin bundle is arranged in a tight, hex-packed configuration, with each pin separated from adjacent pins by a wire wrap. The fuel pin comprises shield slugs at the bottom, a sodium-bonded fuel column in the center, and a gas plenum (containing a tag gas capsule) at the top to capture fission gas, contained within an HT9 alloy steel cladding. The internal shield slugs mitigate radiation dose to the core support structure. The fuel pin cladding provides a functional containment barrier and provides mechanical support to resist plenum gas pressure and fuel swelling due to fission product generation under irradiation. Tag gas is used to identify assemblies containing failed fuel pins using an online fuel monitoring system as described in Section 8 of NAT-2806.

NAT-2806 includes additional details of the fuel assembly and acceptance criteria that when satisfied support a finding that the fuel is qualified for use. Compliance with these acceptance criteria will be described at the operating license (OL) stage.

Reflector Assembly

The reflector assemblies are located radially around the fuel assemblies within the core as shown in Figure 7.1.1-2.

The purposes of the reflector assemblies are to:

- Reflect neutrons into the core's fueled region, which reduces core reactivity loss and limits radiation damage to permanent reactor structures
- Provide passages to guide and control the flow of primary sodium coolant
- Provide features for proper interfacing with other core assemblies, the CBS, In-Vessel Transfer Machine, and shipping and handling equipment

Structural components within the reflector assemblies are classified SR. Non-structural components within the reflector assemblies do not perform safety-significant functions and are classified non-safety-related with no special treatment (NST).

Shield Assembly

Two rows of shield assemblies are located in the outermost rows of the reactor core as shown in Figure 7.1.1-2.

The purposes of the shield assemblies are to:

- Absorb radial neutron leakage outside of the reflector assemblies to limit activation of intermediate sodium coolant and reduce irradiation of Structures, Systems and Components outside of the reactor core
- Provide passages to guide and control the flow of primary sodium coolant
- Provide features for proper interfacing with other core assemblies, the CBS, In-Vessel Transfer Machine, and shipping and handling equipment


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Structural components within the shield assemblies are classified SR. Non-structural components within the shield assemblies do not perform safety-significant functions and are classified NST.

Material Surveillance Assembly

The purpose of the material surveillance assemblies is to provide enclosure and support for the material specimens. Material surveillance requirements are determined by the Reliability and Integrity Management Program, described in Chapter 8. Placement of the material surveillance assemblies within the core is based on input from the surveillance program plan and the requirements for temperature and fluence.

Structural components within the material surveillance assemblies are classified SR. Non-structural components within the material surveillance assemblies do not perform safety-significant functions and are classified NST.

Neutron Source Assembly

The neutron source assemblies produce neutrons, which allows for reactor startup and measurement of source range signals when the reactor core is in a subcritical configuration.

Structural components within the neutron source assemblies are classified SR. Non-structural components within the neutron source assemblies do not perform safety-significant functions and are classified NST.

Simulated Core Assembly

The simulated core assemblies are used to support primary sodium coolant fill and non-power initial operations, including hot functional testing of the reactor system. Simulated core assemblies are also used to assist in recovering a stuck fuel assembly that cannot be removed by the maximum pull force allowed by the In-Vessel Transfer Machine. Multiple configurations of simulated core assemblies are available depending on the intended use.

Structural components within the simulated core assemblies are classified SR. Non-structural components within the simulated core assemblies do not perform safety-significant functions and are classified NST.

Bi-Metallic Assembly

The bi-metallic assemblies support the core restraint function. The bi-metallic assemblies are thermally activated and enhance the core lock-up behavior. Design of the bi-metallic assemblies and their locations are under research and development and described in Section 13.2.

Structural components within the bi-metallic assemblies are classified SR. Non-structural components within the bi-metallic assemblies do not perform safety-significant functions and are classified NST.


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Lead Demonstration Assembly

The lead demonstration assemblies are used primarily to support fuel surveillance as described in NAT-2806. Further description of the lead demonstration assemblies is provided in NAT-2806.

The lead demonstration assemblies are classified SR.

Lead Test Assembly

The lead test assemblies are used to gain operating experience with and aid in the transition to advanced fuel designs. Further description of the lead test assemblies is provided in NAT-2806.

The lead test assemblies are classified SR.

7.1.1.2 Design Basis

Compliance with regulatory criteria and identification of design criteria, codes and standards that define the RCC design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are described in Section 5.3. Compliance with PDC that are applicable to the RCC is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the RCC is protected against natural phenomena by its location within the reactor vessel, which is located within, and protected by, the reinforced concrete RXB substructure. The location within the RXB provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions as described in Section 7.8.1. The RCC is designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1. The seismic response of the reactor core assemblies is analyzed to predict the overall structural response and reactivity insertion due to assembly displacements. The RCC is designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the RCC is compatible with and capable of accommodating the effects of the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents. The selection of materials of construction and fabrication methods for core assembly components provides assurance of compatibility with the liquid sodium operating environment, including oxidation products, throughout the design lifetime. Dynamic analysis of the reactor core provides assurance of structural and functional integrity under vibratory loadings.

Consistent with PDC 10, *Reactor design*, the core is designed such that the specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded during normal operation and anticipated operational occurrences (AOOs). Transient and accident analyses, as described


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in Section 3.3, are performed to assure margin to SARRDLs for normal operation and AOOs. Fuel cladding temperature limits are set to provide significant margin to cladding failure for assurance of fuel pin reliability and to maintain a coolable core geometry.

Consistent with PDC 11, *Reactor inherent protection*, the reactivity feedback mechanisms include fuel temperature (Doppler) effects, fuel axial expansion and contraction, primary sodium coolant density, control rod driveline axial expansion, and radial distortion feedbacks. The reactor core is designed such that the net effect of the prompt acting components of these feedback mechanisms is negative in the power operating range. These feedback mechanisms tend to compensate for a rapid increase in reactivity. Reactivity coefficients are described further in Section 3.12.

Consistent with PDC 12, *Suppression of reactor power oscillations*, the reactor core is designed to prevent power oscillations that could result in conditions exceeding SARRDLs. The core stability analysis is further described in Section 3.12.

Consistent with PDC 26, *Reactivity control systems*, the design provides two reactivity control means of different design principles to insert the control rods, as described in Section 7.2.5. The primary and secondary control rods are designed to satisfy PDC 26 as described in Section 5.3. Preliminary design details on diversity of the primary and secondary control rod assemblies is described in NAT-2806, Sections 5.1, 5.2, and 5.4. Additional information on reactivity control is provided in Section 7.1.1.3.1.

Consistent with PDC 28, *Reactivity limits*, the analysis and design of the control rods, including the rod control withdrawal limitations, interlocks, and control features described in Sections 7.2.5 and 7.6.2 and the scram actuations described in Section 7.6.3, provides assurance that the effects of postulated reactivity accidents do not result in damage to the primary coolant boundary greater than limited local yielding or impair the capability to cool the core.

Consistent with PDC 29, *Protection against anticipated operational occurrences*, the reactor and reactivity control systems are designed and tested to assure an extremely high probability that their safety functions will be accomplished during AOOs. The CRA design includes features to limit the occurrence of common cause failures as described in Section 7.1.1.3.

Consistent with PDC 80, *Reactor vessel and reactor system structural design basis*, the RCC is designed to maintain its geometry to support the natural circulation flowpath of primary coolant through the reactor core during postulated accidents and to allow for control rod insertion for reactor shutdown during postulated accident conditions. The reactor core assemblies are structurally supported by the core restraint system, which transfers loads to the core barrel structures described in Section 7.1.2. The core restraint system maintains the core configuration to support primary coolant flow for passive heat removal and to support reactor shutdown by sufficient insertion of neutron absorbers within the CRAs.


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Functional Requirements and Design Criteria

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The RCC supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classification of RCC components relied upon to perform the function:

| Function ID | Function Description | RCC Functional Support |
|---|---|---|
| DL3-RC1 | Scram - Gravity Driven Absorber Insertion by Latch Release (SR) | Movement or deflection of RCC components, relative to CRD components, is maintained within limits required for control rod release and insertion during plant operation and Licensing Basis Events. The RCC supports all scram functions in the same manner. |
| DL3-RR2 | Cladding Barrier (SR) | The fuel assemblies maintain natural circulation capability and coolable geometry by limiting their structural deformations within acceptable ranges and maintaining the fuel cladding within its thermal-mechanical constraints. Cladding temperature limits are set to ensure fuel cladding reliability. During accident conditions, the fuel cladding retains radionuclide fission products to remain below SARRDLs. |
| DL3-HR5 | Natural Circulation of Sodium in Primary System (SR) | The heat generated by the RCC drives the natural circulation of sodium in the PHT. The RCC provides passages to guide and control the flow of sodium coolant. |
| DL3-HR6 | Passive Heat Removal in the Ex-Vessel Handling Machine (SR) | The fuel assemblies maintain natural circulation capability and coolable geometry by limiting their structural deformations within acceptable ranges. |
| DL3-HR7 | Passive Heat Removal in the Ex-Vessel Storage Tank (SR) | The fuel assemblies maintain natural circulation capability and coolable geometry by limiting their structural deformations within acceptable ranges. |
| DL3-HR8 | Passive Heat Removal in the Bottom Loading Transfer Cask (SR) | The fuel assemblies maintain natural circulation capability and coolable geometry by limiting their structural deformations within acceptable ranges. |
| DL3-HR9 | Passive Heat Removal in Pin Removal Cell (SR) | The fuel assemblies maintain natural circulation capability and coolable geometry by limiting their structural deformations within acceptable ranges. |


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| Function ID | Function Description | RCC Functional Support |
|---|---|---|
| DL3-HR10 | Passive Heat Removal in the Water Pool Fuel Handling System (SR) | The fuel assemblies maintain natural circulation capability and coolable geometry by limiting their structural deformations within acceptable ranges. |
| DL3-HR13 | Passive Heat Removal in the Failed Fuel Canister (SR) | The fuel assemblies maintain natural circulation capability and coolable geometry by limiting their structural deformations within acceptable ranges. |
| DL4-RC3 | Control Rod Drive System Driveline Scram Follow (NSRST) | Movement or deflection of RCC components, relative to CRD components, is maintained within limits required for control rod insertion to provide adequate defense-in-depth capability. |

The RCC also provides a SR structural support load path function for SR components as described in Section 7.1.1.3.

Safety Related Design Criteria corresponding to the SR functions applicable to the RCC are derived as described in Section 6.1.2 and identified in Section 5.2. The Safety Related Design Criteria are incorporated into the design of the RCC as described in Section 7.1.1.3.

Seismic Classification and Design Codes

Seismic classifications for RCC components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The SR components of the RCC are classified SCS1. Seismic design, analysis, and qualification is performed as described in Section 6.4. RCC components are qualified to withstand seismic loads associated with the safe shutdown earthquake (described in Section 2.6.2) without loss of the capability to perform safety-significant functions.

### 7.1.1.3 Design and Performance Evaluation

The core assemblies provide design features that include the ability to maintain coolability, retain radionuclides, and control in-core reactivity, ex-core fluence, and activation exposure. This is accomplished by a core design that uses liquid sodium coolant, a metallic uranium fuel system, steel cladding and duct structures, and two independent and diverse sets of control rods (further described in Section 7.1.1.3.1).

Reactor power level, start-up, and shut-down is controlled by operator action. However, no operator action is required to perform the DL3 scram functions listed above or to initiate natural circulation of the primary coolant for emergency decay heat removal.

Materials of fabrication and construction methods of the core assemblies are selected to be compatible with the operating environment and anticipated structural loading conditions, including consideration of normal operating conditions and Licensing Basis Events.


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Some components within the core assemblies, including the load pads, duct tubes, handling sockets, and inlet nozzles, perform the SR structural support function listed in Section 7.1.1.2.

The seismic response of the reactor core assemblies is analyzed to predict the overall structural response and reactivity changes due to assembly displacements. The method uses input loadings from the RES seismic analysis. The outputs of the seismic response analysis are used for evaluation of reactivity changes, control rod drop times, component level stresses, and other downstream analyses.

The RES CBS includes a strainer at the Core Inlet Plenum that reduces the potential for debris that could impact RCC functions from entering the core and creating flow blockages. Air flow testing and inspection of core assemblies also reduce the probability of debris inclusions.

The preliminary list of qualification testing of the RCC is described in NAT-2806.

During operation of a fast reactor, assemblies are subjected to temperature and fluence gradients, inducing movement and displacement of the assemblies, which can cause changes in reactivity and core power. A core restraint system is incorporated in the design to achieve determinate and predictable configurations of core assemblies. The core restraint system design is under research and development as described in Section 13.2.

#### 7.1.1.3.1 Reactivity Control

There are nine primary control rod assemblies and four secondary control rod assemblies which function in conjunction with the CRD to position neutron absorber material to appropriately control and terminate the nuclear reaction. The control rod drive mechanism, control rod driveline, and control assembly are directly coupled during normal operation. The connection coupling between the control rod and the driveline is illustrated in NAT-2806, Figure 5-17.

The control assemblies are the basic nuclear control components of the reactor, and include the control rods that contain absorber pins, as shown in NAT-2806, Figure 5-12. The absorber pins contain cladded, helium-bonded, Boron Carbide absorber pellets. The control rods can be adjusted in the axial direction during operation by the Control Rod Drive Mechanisms.

The control rods (located within dedicated hexagonal control assemblies in the core) are driven by the Control Rod Drive Mechanism to move and position absorber material vertically within the core to control core reactivity and power and maintain fuel within design limits. The control rods control core reactivity changes during normal reactor operation and accident conditions. The control rods provide scram insertion capability with sufficient reactivity worth to shut down the reactor and maintain it in safe shutdown even if the highest worth rod is stuck in the withdrawn position.

To provide design diversity to limit common cause failures in compliance with PDC 26, the design includes both primary and secondary control assemblies. The secondary control assemblies provide a regulating function to manage small reactivity changes. NAT-2806 provides additional detail on the control assemblies.


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7.1.1.3.2 Methods and Analyses

Methods and analyses related to the RCC are described in Sections 3.11-3.13.

7.1.1.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. The following programmatic special treatments are preliminarily identified as applicable to the RCC to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Equipment Qualification Program
- Post-Construction Inspection, Testing, and Analysis Program (Chapter 12)

Additional special treatments include qualification and testing of fuel and control assemblies as described in NAT-2806.

7.1.1.5 Instrumentation Requirements

The Nuclear Instrumentation System (Section 7.6.4) provides instrumentation to sense the neutron flux leaking from the core to provide indication of reactor power.

The Reactor Instrumentation System (Section 7.6.5) provides instrumentation and signals for monitoring core temperature, core neutron flux, core coolant flow rate, and fuel failure (via tag gas or coolant activity analysis), as well as other parameters of systems that interface with the RCC, allowing for monitoring of the reactor core.

The Radiation Monitoring System (Section 7.6.6) ensures that radiation release from the reactor core can be detected, providing information on the state of the reactor core.

References

7.1.1-1 TerraPower, “TerraPower, LLC (TerraPower) Natrium Topical Report: Fuel and Control Assembly Qualification”, NAT-2806, Revision 0 (proprietary information, not publicly available).


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**Table 7.1.1-1 Reactor Core Design Parameters**

| Parameter | Value |
|---|---|
| Core Power | 840 MWt |
| Inlet Temperature at 100% power and flow | 680°F (360°C) |
| Outlet Temperature at 100% power and flow | 950°F (510°C) |
| Core Maximum Neutron Flux | $3.98 \times 10^{15}$ n/cm$^2$-s |
| Core Average Fast Flux Fraction (End of Equilibrium Cycle) | 0.67 |
| Core Maximum Power Density (Beginning of Life) | 5,473 W/in$^3$ (334 W/cm$^3$) |



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**Figure 7.1.1-1 Core Assembly Configuration**

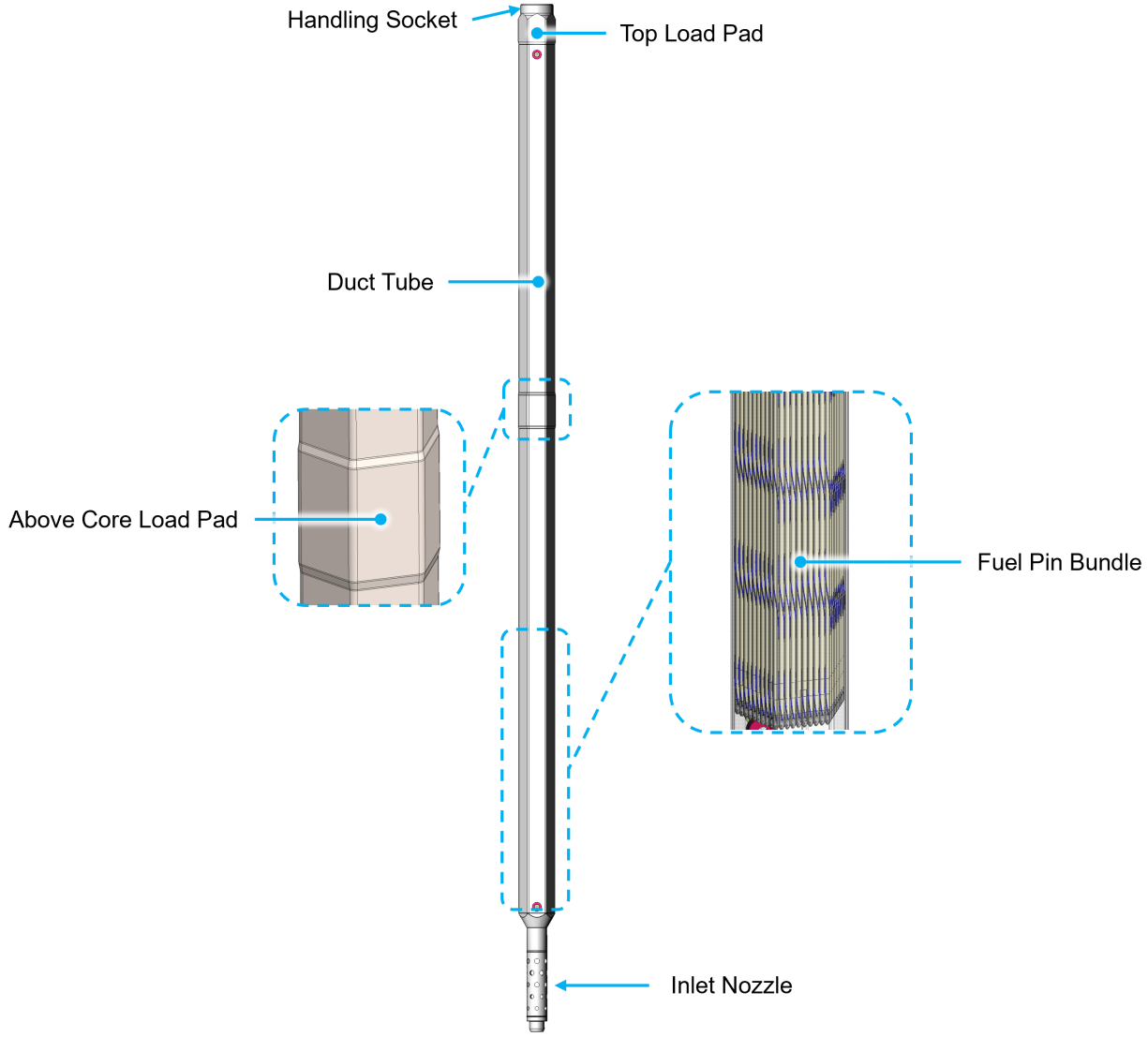



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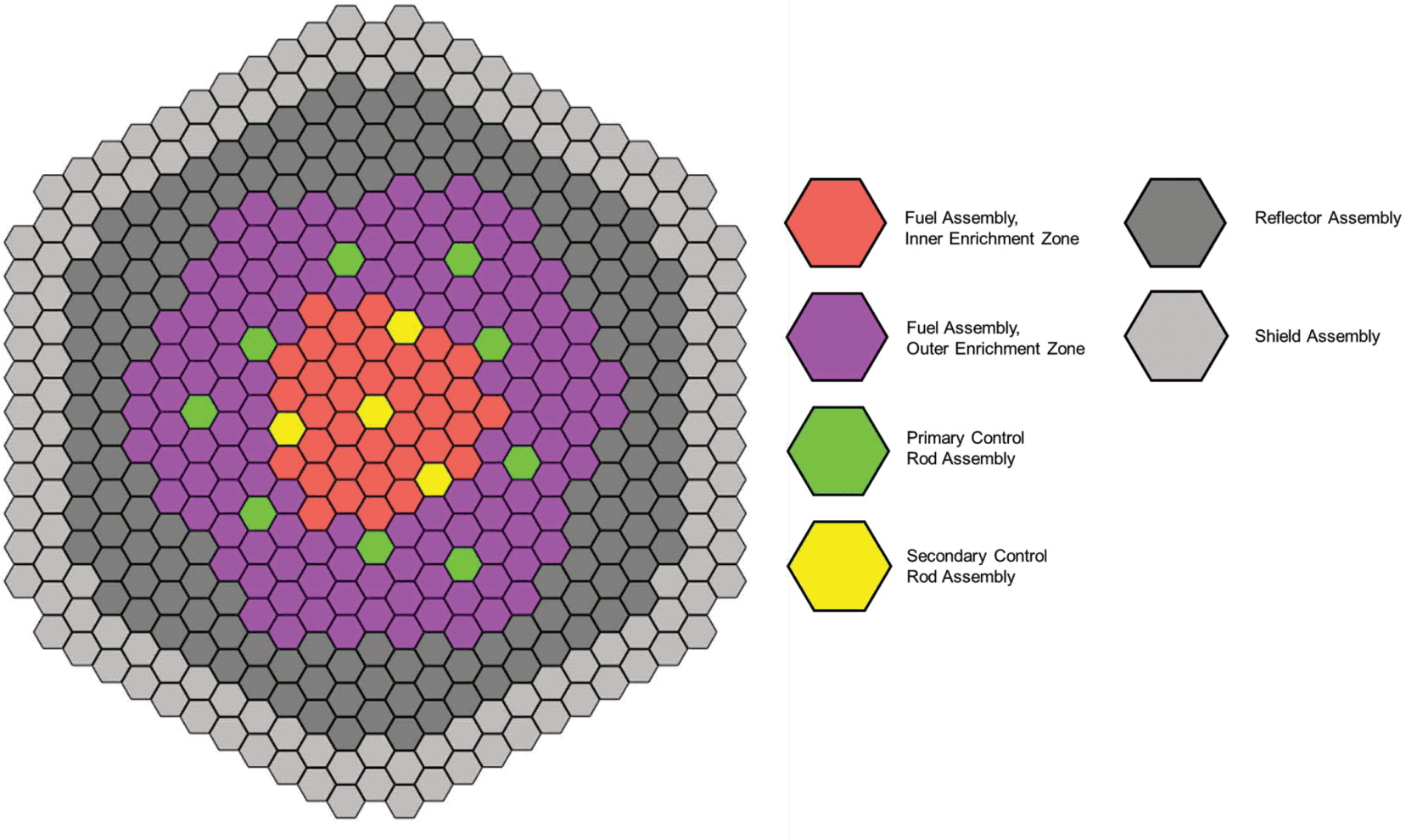
**Figure 7.1.1-2 Illustration of Reactor Core Loading Pattern**



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## 7.1.2 Reactor Enclosure System

### 7.1.2.1 Summary Description

The Reactor Enclosure System (RES) includes the reactor vessel and head, the reactor internal structures, the guard vessel, and the reactor support structures. The RES supports the reactor core and contains the primary sodium coolant. Figure 7.1.2-1 provides the preliminary general arrangement of the RES. The RES is located within the reinforced concrete Reactor Building (RXB) substructure described in Section 7.8.1.1.

The RES design incorporates enhanced safety features, including:

- a low-pressure, pool-type reactor design with no boundary penetrations below the primary coolant operating level
- an integral primary heat transport system that is contained within the Reactor Vessel (RV)
- an earthquake-resistant design incorporating reactor seismic isolation from the supporting structure
- location within a subgrade reinforced concrete structure for enhanced external hazard protection
- passive emergency core cooling by natural circulation of primary coolant and natural convection air cooling for heat rejection to the atmosphere

The RES provides radionuclide retention, reactivity control, and reactor heat removal functions during normal and off-normal conditions. The RES supports radionuclide retention by providing functional containment barriers and primary coolant boundary components. The RES supports reactivity control by establishing the location of reactor control assemblies relative to control rod drive components to allow control rod positioning and insertion within the reactor core. The RES supports reactor heat removal by maintaining the geometry needed for primary coolant circulation within the RV and supports heat transfer to heat removal systems.

The RES forms a portion of the primary coolant boundary, which includes portions of systems enclosing the primary coolant volume and primary coolant cover gas volume as described in Section 5.3.2.5. The RES primary coolant boundary components are the RV and reactor head (including head seals, welds, penetrations, and plugs), Rotatable Plug Assembly (RPA), RPA seals, and the reactor instrumentation drywells. These components also form a portion of the functional containment boundary described in Section 1.3.2.1. Primary coolant boundary components are designed to provide an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The metallic components of the RES primary coolant boundary are designed and constructed in accordance with the ASME BPVC Section III, Division 5 (Reference 7.1.2-1).

The high boiling point of the primary sodium coolant allows reactor operation at near-atmospheric pressure, with a significant margin to coolant boiling, resulting in a low-pressure primary coolant boundary design. The pool-type reactor operates with a net positive pressure inert cover gas provided by the Sodium Cover Gas System (SCG) described in Section 7.2.3.


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The RES interfaces with the RXB (Section 7.8.1), Reactor Core System (Section 7.1.1), Primary Heat Transport System (PHT) (Section 7.1.3), Control Rod Drive System (CRD) (Section 7.2.5), Reactor Air Cooling System (RAC) (Section 7.2.1), Fuel Handling Systems (Section 7.3), SCG (Section 7.2.3), Sodium Processing System (SPS) (Section 7.2.4), and I&C systems (Section 7.6).

#### 7.1.2.1.1 Reactor Vessel and Head

The RV is a vertical, right-circular cylindrical shell design with an integrally-welded, torispherical bottom head. The RV is supported by the flat, disc-shaped reactor head, which is attached to the RV through a pressure-retaining structural weld. The RV and reactor head are constructed of austenitic stainless steel for compatibility with the liquid sodium operating environment. The RV and reactor head are designed and fabricated in accordance with ASME BPVC Section III, Division 5.

The RV is supported through a structural weld between the top of the shell and the reactor head. The reactor head extends beyond the diameter of the RV. The reactor head is supported through the intervening Guard Vessel (GV) flange described in Section 7.1.2.1.3, by the Reactor Support Structure (RSS) described in Section 7.1.2.1.4. The RV provides structural support to reactor internal structures as described in Section 7.1.2.1.2. The RV provides a heat transfer surface that supports reactor decay heat removal to the RAC, as described in Section 7.1.2.3.

The reactor head, including the RPA, forms the top head closure for the RV. The reactor head provides access to the internal areas of the RV through head penetrations. Penetrations through the RES primary coolant boundary are located in the reactor head above the primary coolant level. A simplified representation of the preliminary reactor head configuration is provided in Figure 7.1.2-2. The reactor head is an austenitic stainless steel component that is designed to accommodate the loading requirements associated with the penetrations, provide structural support for components mounted to the reactor head, and provide structural support for the RV and its contents. The reactor head is designed to provide sealing of the head penetrations to limit leakage to within acceptable limits associated with functional containment requirements. The reactor head material thickness provides radiological shielding to the Head Access Area (HAA). Insulation on the exterior of the reactor head and through-head nozzles reduces heat loss through the reactor head to the HAA. The reactor head provides the top closure of the annular space between the RV and the GV described in Section 7.1.2.1.3. Access ports are provided to support inspections within the RV-GV annular space.

The RPA interfaces with and supports fuel handling equipment described in Section 7.3 to provide core assembly insertion and extraction from the vessel, and in-vessel placement capabilities, during refueling outage periods. The RPA consists of the rotatable plug and the Upper Internal Structure (UIS). The RPA includes features for mounting and sealing the RPA to the reactor head and features to allow rotation of the rotatable plug to support refueling operations. The rotatable plug includes equipment penetrations, and mounting and sealing features associated with the penetrations. The RPA is mounted to a large, eccentrically-located penetration through the reactor head.


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Figure 7.1.2-3 illustrates the preliminary general arrangement of the components associated with the RPA.

#### 7.1.2.1.2 Reactor Internal Structures

Internal structures within the RV provide structural support for reactor core assemblies and primary heat transport components, guide primary coolant flow through the core, define sodium pool regions (hot pool, cold pool, and warm pool), and provide in-vessel core assembly storage. The reactor internal structures consist of the UIS, Reactor Fixed Internals (RFI), and Core Barrel Structures (CBS).

The UIS is a vertical, cylindrical assembly mounted to the bottom of the rotatable plug (Figure 7.1.2-3). The UIS is located above the reactor core and provides shroud tubes as guides to locate control rod drive mechanism (CRDM) drivelines for alignment with the control rod assemblies within the reactor and drywells to control placement of core exit temperature monitoring instrumentation.

With the RPA locked and mounted to the reactor head in the plant operation position, the vertical axis of the UIS is aligned with the reactor core central vertical axis and the UIS is stationary. With the RPA in the refueling position, and the CRDM drivelines unlatched and fully retracted, the UIS rotates with the rotatable plug. The cylindrical geometry of the UIS is interrupted to provide a space (cutout) for installation of the In-Vessel Transfer Machine (IVTM). The UIS provides vertical and lateral support to the IVTM during refueling operations. The location of the IVTM in the cutout space in the UIS, along with the eccentric center location of the rotatable plug in the reactor head, allows the IVTM to reach assembly locations within the core and the In-Vessel Storage (IVS).

The RFI structures provide support for internal reactor components and direct the flow of primary sodium coolant in the heat removal circuit. The RFI provide the separation between the hot, cold, and warm pools and provide neutron shielding to limit activation of intermediate coolant within the Intermediate Heat Exchanger (IHX). The general arrangement of the RFI is shown in Figure 7.1.2-4.

The CBS support core assemblies installed in the reactor core and IVS locations, and provide separation between the hot and cold pools. The CBS also provide a load path from RFI structures and fixed in-vessel shielding to the RV. The CBS consist of the inner and outer core barrels, the core former ring, the core inlet plenum including upper and lower grid plate, core inlet receptacles, core strainer, support gussets, and the IVS. The CBS transmits loads from the reactor core, permanent shielding, and the IVS to the RV to sustain deadweight and dynamic loads. The preliminary general arrangement of the CBS is illustrated in Figure 7.1.2-5.

#### 7.1.2.1.3 Guard Vessel

The RV is surrounded by a GV, which provides a defense-in-depth coolant and functional containment boundary in the unlikely occurrence of RV leakage. The GV is a vertical, right-circular cylindrical shell design with an integrally-welded, torispherical bottom head similar in shape to the RV. There are no penetrations through the GV wall or bottom head. The GV is supported by the RSS through an integral flange at the top of the shell. The GV is larger in


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diameter than the RV, forming an annular space between the GV and RV. The annular space is inerted with argon gas minimizing the potential for sodium-air reaction. The geometry of the annular space is maintained by the Reactor Support Blocks (RSBs) described in Section 7.1.2.1.4. The GV provides a heat transfer surface that supports reactor decay heat removal to the RAC, as further described in Section 7.1.2.3. The preliminary general arrangement of the GV is illustrated in Figure 7.1.2-6.

#### 7.1.2.1.4 Reactor Support Structure

The RES is supported by the RSS, which transmits loads to the RXB substructure. The RSS structural load path consists of RSBs, a Modular Isolated Reactor Support Structure (MIRSS), and seismic isolators. The preliminary configuration of the RSS is illustrated in Figure 7.1.2-7.

### 7.1.2.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the RES design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the RES is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the RES is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels are described in Section 6.1.1. The RES is located within, and protected by, the reinforced concrete RXB substructure. The location within the RXB provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions as described in Section 7.8.1. The RES is designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1. RES components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions.

Consistent with PDC 3, *Fire protection*, the design of the RES minimizes the probability and effect of fires through design features to minimize the potential for sodium leakage to the surroundings and through the selection of non-combustible and non-fire sustaining materials, to the extent practical. The potential for sodium-air reactions are minimized by the use of an inert cover gas over the primary coolant free surface and the GV surrounding the RV that includes an inert gas-filled annular space. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the RES is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences (AOOs), and postulated accidents. Materials selection for primary coolant boundary components consistent with the requirements of ASME BPVC Section III, Division 5 provides assurance of


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compatibility with the sodium environment, including sodium aerosols, vapors, and oxidation products, throughout the design lifetime. Equipment qualification for harsh environmental conditions resulting from accident conditions is provided by the RES component quality, design, and qualification requirements in accordance with the Equipment Qualification Program described in Chapter 8.

The location of the RES within the RXB substructure provides protection from external hazards caused by transportation incidents and offsite industrial and chemical facilities.

The RES is protected from dynamic effects, including missiles and discharging fluids, that could result from equipment failures, and from events and conditions outside the nuclear island, in accordance with the methods described in Section 6.1.1.

Consistent with PDC 10, *Reactor design*, the design of the RES provides appropriate margin to assure specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded during normal operation or AOOs. The pool-type reactor operates at near-atmospheric pressure with a low pressure inert cover gas. The reactor internal structures maintain alignment of control rod drivelines with control assemblies to provide absorber insertion for reactivity control. The reactor internal structures maintain core support and configuration for adequate coolant flow and heat removal under forced flow and natural circulation conditions to provide margin to SARRDLs during normal operation or AOOs. The RV is designed with no penetrations through the primary coolant boundary and the reactor head penetrations are located above the primary coolant level and sealed to limit cover gas leakage to within limits associated with functional containment requirements.

Consistent with PDC 11, *Reactor inherent protection*, the design of the RES structurally supports and maintains the geometry of reactor core assemblies to limit fuel assembly deflections and, in conjunction with the core restraint system described in Section 7.1.1, contributes to assurance that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Consistent with PDC 12, *Suppression of reactor power oscillations*, the design of the RES contributes to limiting power oscillations that could result in conditions exceeding SARRDLs by restraining core assembly lateral deflection, distortion, or excessive movements that could occur during normal operation or off-normal events, such as earthquakes. The RES limits power oscillations caused by entrainment of gas and gas voids by limiting local primary coolant flow velocities and establishing an adequate level of primary sodium to prevent a dynamic ingestion of sodium cover gas.

Consistent with PDC 14, *Primary coolant boundary*, the RES primary coolant boundary components are designed, fabricated, erected, and tested to provide an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The primary coolant boundary operates at low pressure, and metallic boundary components are designed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5. Provisions are provided for leak testing


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primary coolant boundary components in accordance with applicable requirements of the ASME Code. Over-pressure protection of the primary coolant boundary is provided per the requirements of the ASME Code.

Radiation and thermal exposure to primary coolant boundary components is limited by the design configuration of RES internal components, and the location of in-vessel radiation and thermal shielding, to maintain a low probability of material degradation that could result in rapidly propagating failure or gross rupture. The primary coolant boundary is constructed of austenitic stainless steel materials, which further limits the potential for material fractures caused by radiation and thermal stresses.

Penetrations through the primary coolant boundary are located in the reactor head and are sealed to limit leakage during normal and off-normal operations to within specified allowable limits associated with functional containment requirements. Non-metallic components are designed, analyzed, tested, verified, and qualified in accordance with the Equipment Qualification Program. At mating surfaces of primary coolant boundary penetrations, redundant sealing methods are provided. Sealed penetrations include provisions for testing and inspection to provide assurance that allowable leakage limits associated with functional containment requirements are maintained.

Consistent with PDC 15, *Primary coolant system design*, the design of the RES provides sufficient margin such that the design conditions of the primary coolant boundary are not exceeded during conditions of normal operation, including AOOs. The primary coolant boundary operates at low pressure and is designed and constructed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5. Sufficient margin to design conditions is provided by the design requirements established as described in Section 6.4.3. Over-pressure protection of the primary coolant boundary is provided per the requirements of the ASME Code.

Consistent with PDC 16, *Containment design*, the design of the primary coolant boundary supports functional containment to control the release of radioactivity to the environment and provide assurance that functional containment design conditions are not exceeded.

Consistent with PDC 26, *Reactivity control systems*, the design of the RES supports reactivity control by providing structural support to CRDMs and drivelines, and maintaining control rod driveline alignment with control assemblies to allow insertion of negative reactivity at a sufficient rate to provide a safe shutdown condition.

Consistent with PDC 28, *Reactivity limits*, the design of the RES supports reactivity control functions by providing structural support to CRDMs and drivelines, and maintaining control rod driveline alignment with control assemblies to limit the potential amount and rate of reactivity increase during postulated reactivity events. RES contributes to structurally supporting and maintaining the geometry of reactor core assemblies, to limit fuel assembly deflections, and provide adequate coolant flow and heat removal. Analysis of effects of reactivity events is described in Chapter 3.


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Consistent with PDC 29, *Protection against anticipated operational occurrences*, the design of the RES supports reactivity control systems functions by providing structural support to CRDMs and drivelines, and maintaining control rod driveline alignment with control assemblies, to provide assurance of an extremely high probability of accomplishing their safety functions in the event of AOOs.

Consistent with PDC 30, *Quality of primary coolant boundary*, the safety-related primary coolant boundary is designed, fabricated, erected, and tested to quality standards established as described in the Quality Assurance Program Description (Section 8.1). The primary coolant boundary components are designed and constructed in accordance with the requirements of ASME BPVC, Section III, Division 5. Means are provided for detecting and, to the extent practical, identifying the location of the source of primary coolant leakage. Pre-service and in-service monitoring and non-destructive examination (MANDE) requirements implemented by the Reliability and Integrity Management Program (described in Chapter 8) for the primary coolant boundary provide assurance that quality requirements are met.

Consistent with PDC 31, *Fracture prevention of primary coolant boundary*, the primary coolant boundary components of the RES are designed with sufficient margin to provide assurance that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. RES primary coolant boundary metallic components are designed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5 for Class A components, as applicable. Selection of RES primary coolant boundary materials of construction considers material compatibility with liquid sodium and its aerosols, vapors, and oxidation products. The RES primary coolant boundary component design and material selection is made with consideration of service temperatures, service degradation of material properties, and applicable degradation mechanisms. Metallic primary coolant boundary components are constructed of austenitic stainless steel to minimize the effects of irradiation embrittlement and fracture from thermal stresses. The Degradation Mechanism Assessment is performed in accordance with the Reliability and Integrity Management Program, which meets the requirements of ASME BPVC Section XI, Division 2 (Reference 7.1.2-2) as endorsed with exceptions and clarifications by Regulatory Guide 1.246, Acceptability of ASME Code, Section XI, Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,” for Non-Light Water Reactors, Revision 0.

Consistent with PDC 32, *Inspection of primary coolant boundary*, the design of the RES permits periodic inspection and functional testing of important areas and features to assess structural and leaktight integrity through implementation of the MANDE pre-service and in-service inspection requirements of the Reliability and Integrity Management Program. Access for RV and GV internal MANDE is provided through reactor head penetrations, and for external reactor head MANDE by insulation and other equipment removal, as needed.

RV material surveillance is provided by MANDE and Degradation Mechanism Assessment through the Reliability and Integrity Management Program. Key considerations of the Degradation Mechanism Assessment include RV materials of construction, the environment to which the RV materials are exposed, limits of impurities for RV materials and primary sodium,


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and RV fabrication techniques. Available service degradation data for RV materials under anticipated environmental conditions from published testing results from operating and test reactors, and specific testing performed to qualify the RV materials, is evaluated to support the basis for RV material selection. The Reliability and Integrity Management Program, along with the assessment of available testing and service data, provide an appropriate material surveillance program for the RV.

Consistent with PDC 33, *Primary coolant inventory maintenance*, the design of the RES provides assurance that primary coolant inventory is maintained such that SARRDLs are not exceeded as a result of primary coolant inventory loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the boundary. There are no penetrations through the RV and the primary coolant boundary penetrations through the reactor head are above the operating level of primary sodium in the cover gas space. In the unlikely event of RV boundary leakage, the surrounding GV contains leaked sodium. The annular space between the RV and GV is sized such that the equilibrium coolant level remains sufficient to remove reactor core heat. The SPS includes piping that penetrates the reactor head and extends into the sodium coolant. The SPS pumps are tripped on a low low primary sodium level by the RPS to protect against primary coolant inventory loss in the event of SPS pump discharge piping leakage.

Consistent with PDC 34, *Residual heat removal*, the RES design supports residual heat removal during normal operations and AOOs by establishing and maintaining the flowpath for primary coolant forced flow and natural circulation within the RV. Reactor core decay heat and other residual heat is transferred to the intermediate system through the IHX, as described in Section 7.1.4, at a sufficient rate that SARRDLs and the RES design conditions are not exceeded.

Consistent with PDC 35, *Emergency core cooling*, the design of the RES supports sufficient removal of residual heat during postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented. The RES design establishes and maintains natural circulation of the primary coolant through the reactor core in the event of a loss of forced coolant flow. The natural circulation flowpath directs sodium coolant through the core where heat is absorbed, then circulates the hot sodium to transfer heat to the RV and GV walls and to the RAC for rejection to the atmosphere. The RV and GV heat transfer surfaces provide sufficient surface area, and are enhanced for improved heat transfer through surface treatment, to support adequate emergency core cooling. The transfer of core decay heat to the RAC occurs passively, without reliance on actuation systems, active components, or manual operator actions such that no credible single failure prevents the RES function in support of emergency core cooling.

Consistent with PDC 36, *Inspection of emergency core cooling system*, the design of the RES permits appropriate periodic inspection of components to provide assurance of integrity and capability for the support of the emergency core cooling function through implementation of the Reliability and Integrity Management Program as described for the compliance with PDC 32.


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Consistent with PDC 37, *Testing of emergency core cooling system*, the design of the RES permits appropriate periodic functional testing to assure structural integrity, operability, and performance of RES components relied on for emergency core cooling. Structural integrity is assured through implementation of the Reliability and Integrity Management Program. The operability and performance capability of the RES primary coolant circulation flowpath is monitored during plant operation by installed reactor instrumentation, described for compliance with PDC 13.

Consistent with PDC 61, *Fuel storage and handling and radioactivity control*, the design of the RES includes in-vessel storage for initial decay and cooling of core assemblies prior to removal and placement in ex-vessel storage and supports in-vessel core assembly handling operations. The IVS is located within the RV. Stored fuel is cooled by primary coolant flow through the IVS as part of the normal and emergency core cooling flowpaths. The IVS is inspected and tested consistent with other reactor vessel internal structures and in accordance with the Reliability and Integrity Management Program as described for PDC 32. In-vessel core assembly handling features are described in Section 7.3.3.

Consistent with PDC 62, *Prevention of criticality in fuel storage and handling*, the design of the RES IVS prevents criticality within the stored fuel assemblies by providing a geometrically safe storage configuration. The criticality methodology for IVS is described in Section 3.14.

Consistent with PDC 72, *Sodium heating systems*, heat is added to the primary coolant volume from core decay heat or core power generation and circulated by the PHT within the RV, preventing trapped hold up volumes and stagnation where localized freezing might occur. When sufficient decay heat is unavailable, heat is added to the primary coolant volume from the Intermediate Heat Transport System (IHT) through the IHX. In the event of a single failure, a single Primary Sodium Pump (PSP) or intermediate sodium loop adds sufficient heat to the primary coolant volume for freeze protection.

Consistent with PDC 73, *Sodium leakage detection and reaction prevention and mitigation*, the design of the RES provides means to detect sodium leakage, to limit and control the extent of sodium-air and sodium-concrete reactions, and to mitigate the effects of fires resulting from sodium-air reactions. The GV surrounding the RV is provided to contain potential sodium leakage and prevent sodium-concrete reactions. The annular RV-GV space is filled with argon gas to provide an inert atmosphere to prevent sodium-air reactions and associated fires. The GV includes sodium leakage detection features to identify potential RV leakage. The reactor head penetrations are provided with seals and are located in the cover gas space to minimize the potential for sodium leakage.

Consistent with PDC 74, *Sodium/water reaction prevention/mitigation*, the RES is located within the RXB substructure, which does not house water-filled systems to limit the potential for contact between primary sodium and water. Protection from environmental conditions that could result in water intrusion into the RXB is provided as described in Section 7.8.1.

Consistent with PDC 78, *Primary Coolant System Interfaces*, the design of the RES excludes interfaces with SSCs containing fluid that is chemically incompatible with the primary coolant. Interface with the IHT, which contains chemically compatible fluid, includes a single passive


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barrier within the IHX. The IHT is maintained at a higher pressure than the primary coolant during normal operation, AOOs, shutdown, and accident conditions. The cover gas space accommodates increased primary coolant inventory due to IHX leakage and the Reactor Protection System initiates reactor scram and Intermediate Sodium Pump trip on a high primary coolant level condition such that RES safety-significant functions are maintained.

Consistent with PDC 79, *Cover gas inventory maintenance*, the SCG provides sufficient argon gas inventory to maintain cover gas overpressure accounting for limited leakage from the sealed penetrations in the primary coolant boundary and potential rupture of small piping or other small components that are part of the primary coolant boundary such that the primary coolant sodium design limits are not exceeded.

Consistent with PDC 80, *Reactor vessel and reactor system structural design basis*, the RV is designed to maintain the geometry of reactor vessel internal structures to support the natural circulation flowpath of primary coolant through the reactor core and to heat transfer surfaces for passive heat removal to the IHT and the RAC during postulated accident conditions. The reactor internal structures support the reactor core and CRDM drivelines alignment to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown during postulated accident conditions. The RV, reactor head, and reactor internals structures are designed and constructed in accordance with the requirements of ASME BPVC Section III, Division 5.

Functional Requirements and Design Criteria

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as safety-related (SR) functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The RES supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classification of RES components relied upon to perform the function:

| Function ID | Function Description | RES Functional Support |
|---|---|---|
| DL3-RC1 | Scram – Gravity driven absorber insertion by latch release (SR) | Movement or deflection of RES supported CRD components, relative to RCC components, is maintained within limits required for control rod release and insertion during plant operation and licensing-basis events. |
| DL3-HR4 | Inherent – RAC Operation (SR) | The RES provides a natural circulation flowpath of primary coolant through the reactor core and heat transfer to the RAC for decay heat removal. |
| DL3-HR5 | Natural Circulation of Sodium in Primary System (SR) | The RES provides a natural circulation flowpath of primary coolant through the reactor core for decay heat removal. |
| DL3-RR1 | Primary Coolant Boundary including RES Barrier (SR) | The RES provides a limited-leakage primary system boundary and functional containment barrier. |


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| Function ID | Function Description | RES Functional Support |
|---|---|---|
| DL4-RR1 | HAA Barrier, GV Leak Prevention Function (NSRST) | The GV, and the HAA and Well seals, provide a defense-in-depth functional containment boundary and the GV provides a backup primary coolant boundary. |

Safety classifications for RES structures and components are listed in Table 7.1.2-1. Safety-Related Design Criteria corresponding to the SR functions applicable to the RES are derived as described in Section 6.1.2 and identified in Section 5.2. The Safety-Related Design Criteria are incorporated into the design of the RES as described in Section 7.1.2.3.

The RES provides a structural support load path function for SR and NSRST components as described in Section 7.1.2.3.

Regulatory Guidance

Regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the RES, along with a description of the use and conformance to the guidance, includes:

RG 1.20, Revision 4, *Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing*

Partial Conformance. Reactor vessel internal components are classified as “prototype” in accordance with Section C.1 as “unique, first-of-a-kind, within the US domestic nuclear power plant fleet for which no previous valid prototype can be referenced.” Reactor vessel internal components are screened and analyzed for effects due to potential excitation mechanisms in accordance with Section C.2.

RG 1.31, Revision 4, *Control of Ferrite Content in Stainless Steel Weld Metal*

Partial Conformance. Limits are imposed on the maximum amount of delta ferrite when welding austenitic stainless steel ASME BPVC Section III, Division 5, Class A components with design temperatures exceeding 425 degrees Celsius to prevent degradation of material properties over time.

RG 1.36, Revision 1, *Nonmetallic Thermal Insulation for Austenitic Stainless Steel*

Full Conformance. Methods and procedures are used for selection of the reactor head non-metallic insulation to prevent stress-corrosion cracking of stainless steel by minimizing leachable chlorides and fluorides to the lowest practicable levels.

RG 1.44, Revision 1, *Control of the Processing and Use of Stainless Steel*

Partial Conformance. Regulatory Position C.1 criterion for cleanliness is implemented in fabrication specifications for safety-significant, austenitic stainless steel RES components.


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RG 1.69, Revision 1, *Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants*

Full Conformance. The MIRSS design shielding analysis, including steel plates and internal concrete shielding, is conducted per RG 1.69, Subsection 1.a.

RG 1.87, Revision 2, *Acceptability of ASME Code, Section III, Division 5, High Temperature Reactors*

Full Conformance. Applicability, limitations, and restrictions regarding the use of ASME BPVC Section III, Division 5 are conformed to for Code pressure boundary components and core support structures, including the listed acceptable (N872) and conditionally acceptable (N-861, N862, N-898) ASME Code Cases.

RG 1.124, Revision 3, *Service Limits and Loading Combinations for Linear-Type Supports*

Full Conformance. The levels of service limits and combination of loadings are applied for the RSS designed in accordance with ASME BPVC Section III, Division I – NF Class 1 linear-type component and piping supports.

RG 1.130, Revision 3, *Service Limits and Loading Combinations for Class 1 Plate-and-Shell Type Component Supports*

Full Conformance. The levels of service limits and combination of loadings are applied for the RSS designed in accordance with ASME BPVC Section III, Division I – NF Class 1 plate-and-shell type component and piping supports.

Seismic Classification and Design Codes

Seismic classifications for RES components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The seismic classification of RES components is SCS1. Seismic design, analysis, and qualification is performed as described in Section 6.4. RES components are qualified to withstand seismic loads associated with the safe shutdown earthquake described in Section 2.6.2 without loss of the capability to perform SR functions.

Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of RES components as special treatments, as described in Section 6.3. Codes and standards that are applicable to the RES are listed in Table 7.1.2-2.

### 7.1.2.3 System Design and Performance Evaluation

The RES is designed to meet design basis requirements described in Section 7.1.2.2 and evaluated to provide assurance that associated performance requirements are met. Design features are provided by the RES design to support the identified safety-significant functions and to provide assurance of compliance with applicable regulatory criteria.


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The RES is protected from design-basis hazards and associated design basis hazard level described in Section 6.1.1. Protection from tornadoes, high-winds and wind-driven missiles, extreme climatic conditions, and external fires is provided by the location within the hardened RXB. External flooding and precipitation protection for the site SR structures, including the RES location within the RXB, is described in Section 6.1.1. Protection from volcanic hazards (described in Section 2.7) potential for adversely affecting heat transfer surfaces important for RAC function is provided by RAC performance monitoring and ashfall vacuuming as described in Section 7.2.1. The location within the RXB substructure also provides RES protection from local transportation incident-related hazards.

The RES components internal to the RV that are within the primary coolant flowpath are analyzed for the effects of flow-induced vibration to provide assurance that the design is not susceptible to degradation that could result in significant component damage or failure, or adversely affect equipment service life. The RES is included within the Comprehensive Vibration Assessment Program, which is applied as a programmatic special treatment as described in Section 7.1.2.4.

Table 7.1.2-3 provides preliminary approximate dimensions for selected RES components.

Reactor Vessel and Head

The RV contains the necessary inventory of primary coolant such that the reactor core remains covered and cooled during normal operation and licensing-basis events. The RV and reactor head, including the pressure retaining assembly weld, form a significant portion of the primary coolant boundary and support the functional containment as a boundary against the release of radionuclides to the HAA. The RV, reactor head, and metallic metallic components of the RPA are designed and constructed in accordance with the requirements of the ASME BPVC, Section III, Division 5, Class A. Code stress analysis methods, along with identification of loads and load combinations, for metallic boundary components are described in Section 6.4.3.

There are no penetrations through the RV wall or bottom head. The penetrations through the primary coolant boundary are in the reactor head and are located above the sodium level in the cover gas space, thereby minimizing the potential for loss of primary coolant. Penetrations through the reactor head are provided with limited leakage seal features that limit leakage to within cover gas makeup limits and limits associated with the functional containment leakage requirements.

The RV and reactor head assembly contains the pool-type reactor design that operates at near atmospheric pressure. The thin-walled vessel design promotes conduction heat transfer between the inner and outer RV surface, and surface treatment of the outer surface enhances thermal radiation heat transfer from the RV to support reactor cooling by the RAC. The surface treatment for heat transfer enhancement is the subject of research and development as described in Section 13.2.1.

Materials of construction for primary coolant boundary components are selected to be consistent with the requirements of ASME BPVC Section III, Division 5 to provide adequate material quality requirements and assure compatibility with the high-temperature sodium operating environment. Materials are selected based on fracture toughness, creep strength and resistance, corrosion


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resistance, thermal aging resistance, and neutron embrittlement resistance considerations. The selection of austenitic stainless steel for fabrication of metallic pressure boundary components provides assurance of adequate resistance to stress and thermal factors in the sodium operating environment, as well as resistance to neutron embrittlement in the fast-neutron fluence environment. Fast-neutron fluence calculations are expected to demonstrate that materials remain within degradation limits with margin. Based on the expected results of the evaluations of the effects on the metallic materials of neutron fluence and the high-temperature sodium operating conditions, there is margin to material degradation limits throughout the design lifetime such that an in-vessel material surveillance program is not required, as further described in Section 7.1.2.2 for compliance with PDC 32.

Preliminary design temperature, service environment, and selected materials of construction for the RV and reactor head are provided in Table 7.1.2-4.

The RES provides the flow path for primary coolant to transfer heat from the reactor core to the IHT during normal power operation and shutdown. In the event that forced primary coolant flow is not available, the RES configuration is designed to passively establish and maintain natural circulation flow through the reactor core. The forced flow and natural circulation flowpaths for reactor heat removal are the same. The coolant flowpath is shown schematically in Figure 7.1.2-8. Natural circulation flow is driven by heat transfer to the RV wall (and within the IHX to the IHT, when available), which cools the hot pool sodium and increases its density. The higher density sodium flows to the cold pool region in the lower elevation of the RV and into the core inlet plenum, where it is drawn into the core by the upward flow of sodium heated by core decay heat. Reactor core decay heat is transferred to the IHT (described in Section 7.1.4) and the Intermediate Air Cooling System (described in Section 7.2.2) through the PHT IHXs and rejected to the atmosphere as the preferred method for residual heat removal.

In the event that IHT and Intermediate Air Cooling System heat removal is unavailable, emergency decay heat removal is provided by natural convection air flow through the RAC. Decay heat from the reactor is transferred to the primary sodium coolant within the core. Heat is transferred from the hot sodium coolant to the RV liner and vessel wall by conduction, then from the RV to the GV by thermal radiation, thereby raising the temperature of the GV. Heat is transferred from the GV to the RAC by thermal radiation and convection heat transfer as described in Section 7.2.1. For conditions when heat transfer from the IHX is not available, the hot sodium pool coolant temperature increases, which significantly increases the thermal radiation heat transfer rate from the RV and GV for heat rejection to the atmosphere by the RAC. This passive heat transfer to the RAC is continuously in service and does not require operator-initiated or automatic actions to provide emergency heat removal.

The RPA, in conjunction with refueling equipment described in Section 7.3.3, provides core assembly handling capabilities. The RPA design also provides structural support for the CRD and instrument drywells. In addition, the RPA provides support for the UIS (described in Section 7.1.2.1.2), which is mounted to the underside of the rotatable plug. The rotatable plug is located and supported by a ledge formed on the inner diameter of the reactor head penetration. Thermal shield plates suspended from the underside of the rotatable plug reduce heat transferred to the RPA from the primary sodium hot pool region.


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During plant operation, the RPA is locked in place to the reactor head mounting flange and is not rotatable. The fixed position aligns the UIS and CRDMs to support control rod positioning within the reactor control assemblies. The RPA incorporates a diverse set of seals to limit leakage to within acceptable limits associated with primary coolant boundary and functional containment requirements.

During refueling operations, the RPA is disconnected from the reactor head mounting flange and elevated to allow rotation of the rotatable plug. The IVTM is installed to the rotatable plug during Mode 4, and removed and stored following refueling operations. Rotation of the rotatable plug is facilitated by a rotatable plug drive mechanism, as described in Section 7.3.3, and the RPA bearing assembly. The eccentricity of the center of the rotatable plug relative to the reactor head center and the configuration of the UIS supports positioning the IVTM to reach core assemblies in the core and in the IVS. In the refueling configuration, the RPA provides a seal arrangement to limit leakage from the primary coolant cover gas space to within allowable limits associated with functional containment requirements.

Reactor Internal Structures

The reactor internal structures consists of the UIS (Figure 7.1.2-3), the RFI (Figure 7.1.2-4) and the CBS (Figure 7.1.2-5). The reactor internal structures are designed to maintain a primary coolant flowpath through the reactor core and to the heat removal systems under normal operating conditions and licensing-basis events. The reactor internal structures support the reactor core to assure a coolable core geometry and sufficient control assembly alignment to provide control absorber insertion and safe shutdowns.

The reactor internal structures are designed, fabricated, erected, and tested in accordance with ASME BPVC Section III, Division 5. Code stress analysis methods, along with identification of loads and load combinations, for reactor internal structures are described in Section 6.4.3. The reactor internal structures are fabricated from austenitic stainless steel for compatibility with the high-temperature sodium operating environment.

Preliminary design temperature, service environment, and selected materials of construction for the reactor internal structures are provided in Table 7.1.2-4.

The UIS is structurally mounted to, and supported by, the rotatable plug within the RPA, which is supported by the reactor head. The UIS is designed to provide alignment between the CRDMs mounted to the top of the rotatable plug and the associated drivelines with the control assemblies in the reactor core to assure the ability to insert control rod absorbers upon a scram signal from the Reactor Protection System. The design limits deflection and distortion of the UIS to maintain alignment between the CRDM drivelines and the control assemblies.

The RFI structures include the hot pool liner and the vessel liner, along with the seal plate and the thermal baffle plate, as shown in Figure 7.1.2-4. The hot pool liner is a welded, cylindrical structure that separates the hot and cold pools of the primary sodium coolant and directs hot primary sodium coolant from the core toward the IHX inlets. The hot pool liner is rigidly connected to and supports the seal plate and thermal baffle plate. Together with the vessel liner, these components form a thermally insulated cavity of liquid sodium between the hot and cold pools referred to as the warm sodium pool region. The resulting stratification of the hot and cold


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pools by this configuration minimizes heat transfer between pools, increases thermal efficiency, and reduces thermal gradients on RES structural members and PHT equipment. The hot, cold, and warm pool sodium regions are illustrated in Figure 7.1.2-9.

The hot pool liner provides support for the vessel liner and the PSP pump tanks. The PSP pump tanks route cold sodium to the PSP inlets and direct the pump discharge flow to the core inlet plenum through the discharge pipe. Lateral loads from the IHX and PSP are transmitted to the hot pool liner through the seal plate and the thermal baffle plate. The seal plate provides separation between the warm and cold sodium pool regions.

The boron carbide fixed in-vessel shielding surrounds the reactor core and minimizes intermediate sodium activation. The fixed in-vessel shielding is supported by the hot pool liner through the seal plate and thermal baffle plate, and by the CBS through the upper grid plate.

RFI drywells are provided for insertion of instrumentation within the RV and to prevent instrument contact with sodium coolant. The drywells are designed to accommodate instrumentation for measurement of hot pool sodium temperature and level, cold pool sodium temperature, core exit temperature, and core power. The drywells form a part of the primary coolant boundary.

The RFI includes the SPS guide tube, which houses the SPS pump and main heat exchanger as described in Section 7.2.4, and extends from the penetration in the reactor head to the cold pool.

Reactor head thermal shield plates, located in the sodium cover gas space and supported by the RV, reduce heat load to the reactor head from the hot pool and, in conjunction with the reactor head, provide radiological shielding to limit dose rates in the HAA. The location and configuration of the shield plates reduces the migration of sodium aerosols to the upper cover gas space.

The CBS are designed to support the reactor core and core assemblies stored in the IVS. The CBS structures are shown in Figure 7.1.2-5. The reactor core is located internal to, and is supported by, the core barrel. The core former ring, located at the top of the core barrel and surrounding the core assemblies, provides lateral support for core assemblies in response to assembly bowing and other operational distortions as further described in Section 7.1.1. The inner core barrel provides a flow boundary for the core. The outer core barrel defines the boundary between the hot sodium pool region and the cold pool region, and transfers loads from the core to the other members of the core support structure and to the reactor vessel wall. The core barrels configuration provides separation of coolant flow between the active core and the IVS (located between the inner and outer core barrel) and limits the flow through the IVS to that needed to remove stored assembly decay heat.

The core inlet plenum receives cold coolant flow from the PSP discharge pipe. The upper and lower grid plates locate and support the core inlet receptacles. The receptacles receive the core assembly inlet nozzles described in Section 7.1.1 and provide lower end lateral and vertical support for the assemblies. The configuration of the coolant flowpath through the receptacle provides a hydraulic down force that assists in holding the assembly nozzle (termed hydraulic hold-down) in the receptacle during operation. The receptacles include discriminatory features to support proper placement of assemblies during refueling operations. The core strainer surrounds the core inlet receptacles and limits the entrance of debris into the receptacles and core assembly flow channels to limit the potential for core flow blockages.


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The IVS structure consists of storage guide tubes with receptacles and a top plate to locate and support the guide tubes. The IVS is designed to accept fuel assemblies for interim storage and initial decay heat removal, before moving assemblies to the spent fuel pool described in Section 7.3.1. The IVS design includes storage for intact fuel assemblies and failed fuel assemblies. Sufficient fuel assembly geometry is maintained to prevent criticality within the IVS, as described in Section 3.14.

Guard Vessel

The RES design includes a GV surrounding the RV. The GV provides defense-in-depth protection in the unlikely event of RV leakage. The GV is designed, fabricated, erected, and tested in accordance with ASME BPVC Section VIII, Division 1 (Reference 7.1.2-3). Code stress analysis methods, along with identification of loads and load combinations, for the GV are described in Section 6.4.3. The GV is fabricated from alloy steel. The annular space between the GV and RV is inerted with argon gas supplied from the SCG, and the external surface of the GV is exposed to the RAC air flowpath. The inert RV-GV annulus atmosphere minimizes the potential for sodium-air reaction in the event of RV leakage.

Preliminary design temperature, service environment, and selected materials of construction for the GV are provided in Table 7.1.2-4.

The RV-GV annulus volume is designed to maintain adequate primary coolant inventory within the RV for reactor core cooling in the unlikely event of a RV primary coolant boundary leak. In this case, the equilibrium RV and GV sodium level remains above the level of the IHX inlets such that reactor heat is transferred to the intermediate system or to the RAC.

The GV minimizes the potential for sodium contact with RXB concrete and the surrounding atmosphere. Sealing of the interface between the GV flange and the reactor head is provided by a welded metallic omega seal to limit leakage of potentially radioactive gases to the HAA.

The GV shell inner and outer surfaces are designed to enhance heat transfer from the RV to the RAC. Non-weld areas of the shell receive surface treatments to increase the emissivity of the surface and enhance thermal radiation heat transfer characteristics. The surface treatment for heat transfer enhancement is the subject of research and development as described in Section 13.2.1.

Guide tubes are located within the RV-GV annulus space to support installation of sodium leak detection instrumentation. Liquid sodium sensors are located near the bottom of the GV to provide detection of RV leakage.

Reactor Support Structure

The RES is supported by the RSS. The RSS transfers static and dynamic loads from the RES to the RXB substructure. The RSS is designed, fabricated, erected, and tested in accordance with ASME BPVC Section III, Division 5. Code stress analysis methods, along with identification of loads and load combinations, for the RSS design are described in Section 6.4.3.


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A series of RSBs are located evenly-spaced around the perimeter of the GV-to-reactor head interface to transfer loads to the MIRSS. The RSB consists of an uplift block, bearing assemblies, guide blocks, and a lower block. The RSB assembly is bolted to the MIRSS and provides uplift restraint for seismic events and allows radial movement to accommodate reactor head and GV thermal expansion. The RSBs maintain alignment of the GV-to-reactor head interface. Shielding blocks installed between the RSB locations provide radiation shielding for the HAA.

The MIRSS is a circular, welded steel plate-and-shell support structure that surrounds the GV and transfers loads from the rigidly-attached RSBs to the RXB HAA basemat through seismic isolators. The MIRSS also supports the RAC CCA and forms a part of the RAC air flowpath through ducting internal to the structure. Non-structural concrete surrounds the internal ducting for biological shielding. The annular gaps between the MIRSS and the HAA floor, and between the MIRSS and the GV, are sealed by the HAA Seal and the Well Seal, respectively, to preserve the integrity of the RAC air flowpath and to limit leakage to the HAA. These seals are designed to meet the applicable requirements of ASME AG-1 (Reference 7.1.2-6). The configuration of the seals is provided in Figure 7.1.2-10.

Seismic isolation of the RES is provided to limit the transmission and frequency of RXB horizontal and vertical displacements associated with severe seismic events. A series of seismic isolators are located evenly-spaced around the perimeter of the MIRSS. Load is transferred from the MIRSS to the RXB basemat through the seismic isolators. Details of the seismic isolation system qualification methodology is provided in NAT-8922 (Reference 7.1.2-4), which is incorporated by reference into the SAR. See Section 1.4.2 for a list of documents incorporated by reference into the SAR. Additional details of the seismic isolation system design will be provided at the operating license stage.

### 7.1.2.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the RES is described in Section 7.1.2.2. The following programmatic special treatments are preliminarily identified as applicable to the RES to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- Equipment Qualification Program
- Comprehensive Vibration Assessment Program
- Post-Construction Inspection, Testing, and Analysis Program (described in Chapter 12)

### 7.1.2.5 Instrumentation Requirements

The RES physically supports and locates reactor and primary coolant monitoring instrumentation by providing drywells and thermowells, penetrations, and penetration sealing surfaces. The RES also provides access and support for sodium leakage monitoring instrumentation.


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## References

7.1.2-1 American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Facility Components, Division 5, High Temperature Reactors,” ASME, 2017

7.1.2-2 American Society of Mechanical Engineers “Boiler and Pressure Vessel Code, Section XI: Rules for Inservice Inspection of Nuclear Reactor Facility Components – Division 2: Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Reactor Facilities,” ASME, 2019

7.1.2-3 American Society of Mechanical Engineers “Boiler and Pressure Vessel Code, Section VIII: Rules for Construction of Pressure Vessels – Division 1,” ASME, 2021

7.1.2-4 TerraPower, “Reactor Seismic Isolation System Qualification Topical Report,” NAT-8922, Revision 0

7.1.2-5 American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code, Section II, Part A, Ferrous Material Specifications,” ASME

7.1.2-6 American Society of Mechanical Engineers AG-1, “Code on Nuclear Air and Gas Treatment,” ASME, 2019



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**Table 7.1.2-1 RES Safety Classification and ASME Code Applicability**

| Component | Safety Classification | ASME BPVC Code |
|---|---|---|
| Reactor Vessel | SR | Section III, Division 5 - HBB |
| Reactor Head, Rotatable Plug, Penetration Nozzles and Flanges, Plugs, Cover Plates, Thermowells | SR | Section III, Division 5 - HBB |
| RPA Riser | SR | Section III, Division 5 - HBA |
| Reactor Head Instrumentation Penetration Assemblies | SR | Section III, Division 1 - NB |
| Guard Vessel | Note 1 | Section VIII, Division 1 |
| Reactor Internal Structures | SR | Section III, Division 5 - HGB |
| Reactor Support Structure | SR | Section III, Division 5 - HFA |
| HAA Seal, Well Seal | SR | None |

Note 1: The Guard Vessel is classified as SR for heat removal functions. The ASME Section VIII, Division 1 code classification corresponds to the Guard Vessel pressure boundary in support of the NSRST defense-in-depth radionuclide retention function.


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**Table 7.1.2-2 Applicable Codes and Standards**

| Standard | Edition | Title |
|---|---|---|
| ASME Boiler and Pressure Vessel Code, Section III, Division 5 | 2017 | Rules for Constructions of Nuclear Facility Components - High Temperature Reactors |
| ASME Boiler and Pressure Vessel Code, Section VIII, Division 1 | 2021 | Rules for Construction of Pressure Vessels |
| ASME AG-1 | 2019 | Code on Nuclear Air and Gas Treatment |



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**Table 7.1.2-3 Preliminary RES Dimensions**

| Dimension | Value (in) |
|---|---|
| RV outside diameter | 415.5 |
| Reactor Head outside diameter | 453 |
| GV outside diameter | 433.5 |
| RV and Reactor Head Assembly height | 797 |
| RV wall thickness | 2 |
| GV wall thickness | 1 |
| Reactor Head thickness | 12 |


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**Table 7.1.2-4 Preliminary RES Design Temperature, Service Environment, and Materials of Construction**

| Component | Base Material* | Weld Material** | Design Temperature | Service Environment | Notes |
|---|---|---|---|---|---|
| Reactor Vessel | SA-240 316H | Table HBB-I-14.1(b) | 1050°F | Sodium | Primary coolant boundary component |
| Reactor Head | SA-965 F316H | Table HBB-I-14.1(b) | 800°F | Argon gas, sodium vapor and aerosols | Primary coolant boundary component |
| Rotatable Plug | SA-965 F316H | Table HBB-I-14.1(b) | 800°F | Argon gas, sodium vapor and aerosols | Primary coolant boundary component |
| Instrument Drywells | SA-213 TP316H | Table HBB-I-14.1(b) | 1050°F | Sodium, air | Primary coolant boundary component |
| RFI | SA-240 316H | Table HBB-I-14.1(b) | 1050°F | Sodium; Argon gas, sodium vapor and aerosols | |
| CBS | SA-240 316H | Table HBB-I-14.1(b) | 1050°F | Sodium | |
| UIS | SA-240 316H | Table HBB-I-14.1(b) | 1050°F | Sodium; Argon gas, sodium vapor and aerosols | |
| GV | SA-387 Gr 22, Cl 1, 2-1/4Cr-1Mo | - | 650°F | Argon gas, air | |
| GV Flange | SA-182 Gr F22 Cl 1, 2-1/4Cr-1Mo | - | 650°F | Argon gas, air | |

*ASME BPVC Section II, Part A, “Ferrous Material Specifications” (Reference 7.1.2-5)

**ASME BPVC Section III, Division 5 “High Temperature Reactors,” 2017, as endorsed by RG 1.87 with limitations and restrictions (see Section 7.1.2.2)


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**Figure 7.1.2-1 Simplified Reactor Enclosure System General Arrangement**

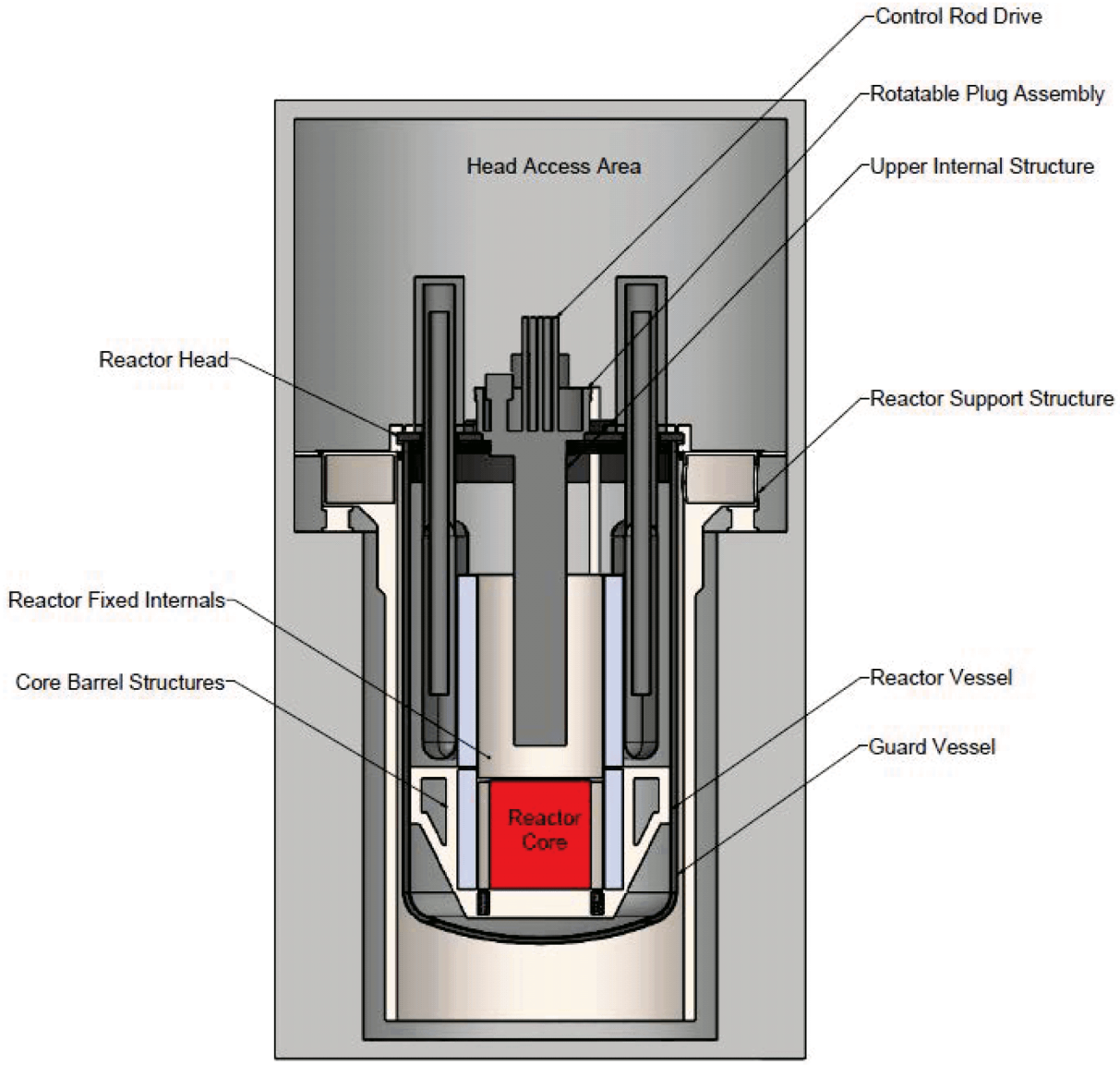




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**Figure 7.1.2-2 Simplified Reactor Head Configuration**

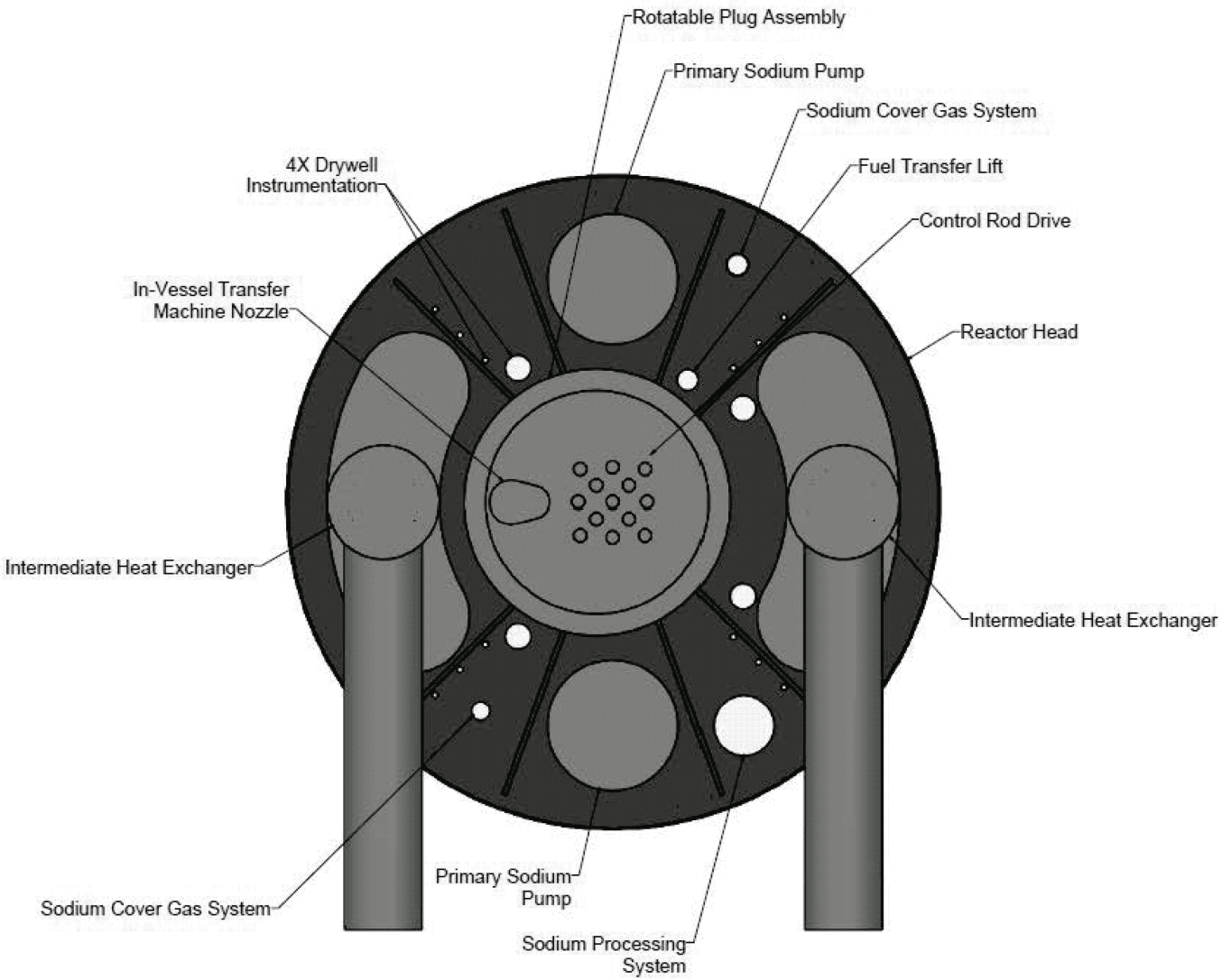


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**Figure 7.1.2-3 Simplified Rotatable Plug Assembly General Arrangement**

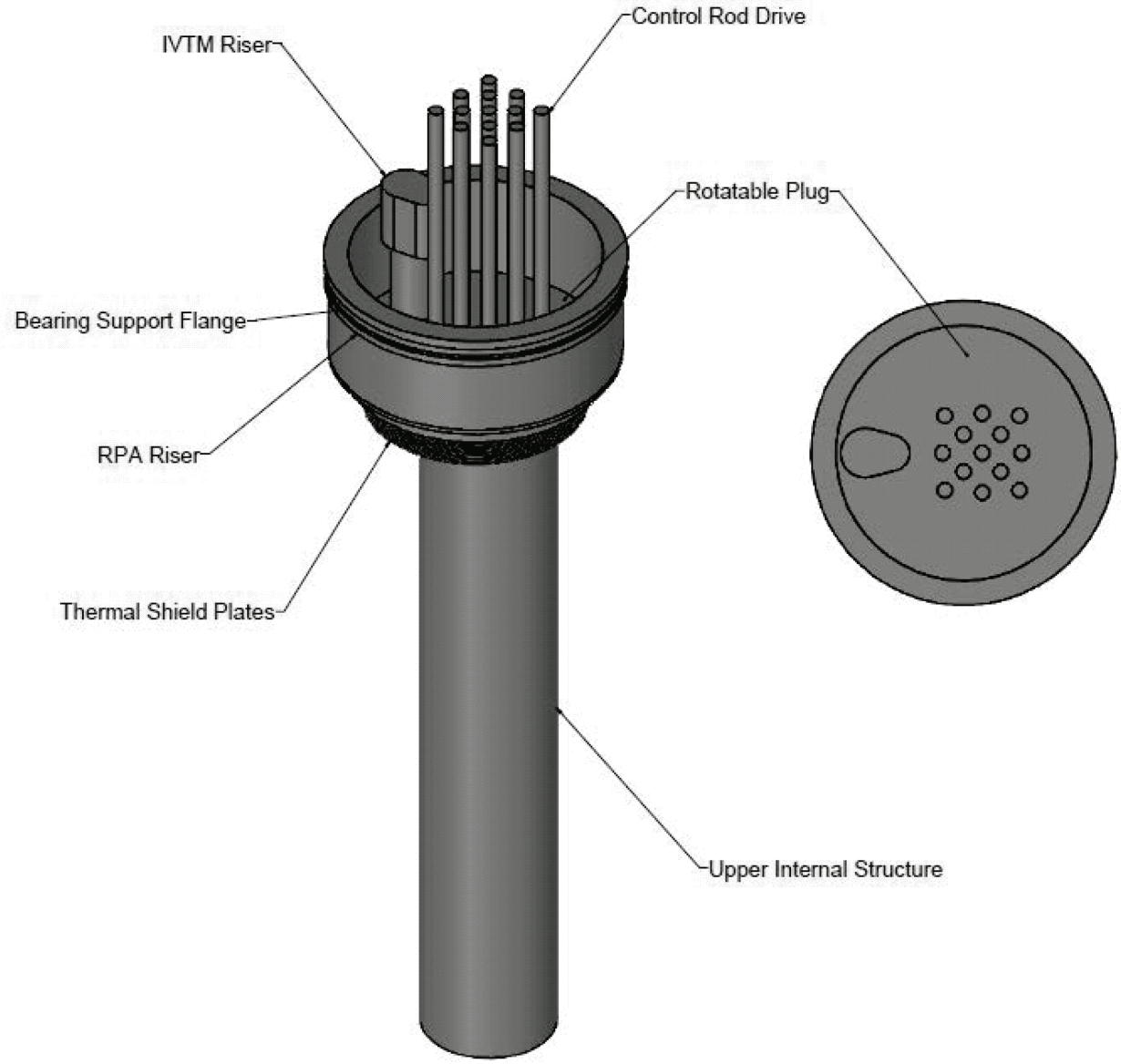



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**Figure 7.1.2-4 Simplified Reactor Fixed Internals General Arrangement**

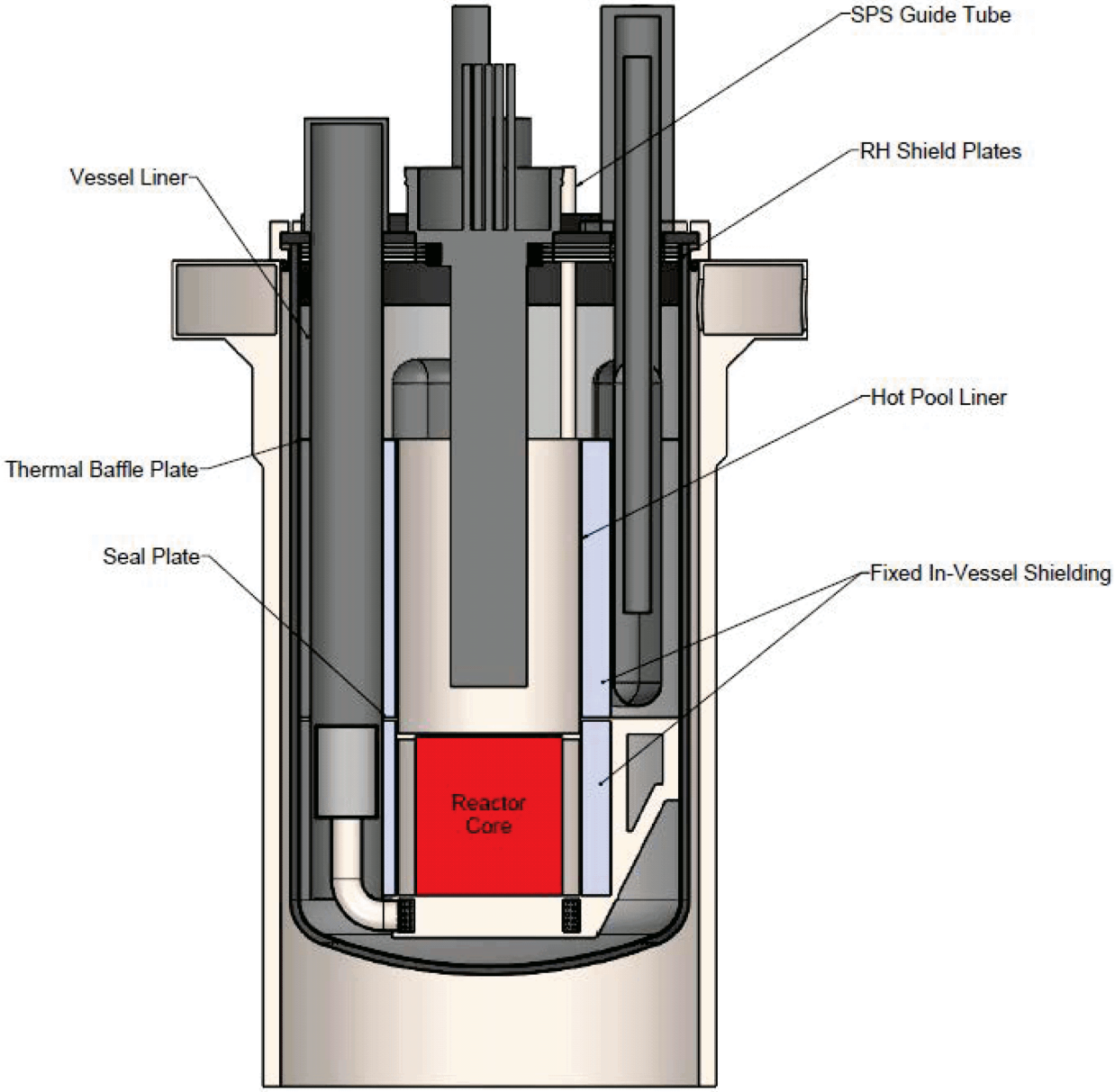



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**Figure 7.1.2-5 Simplified Core Barrel Structures General Arrangement**

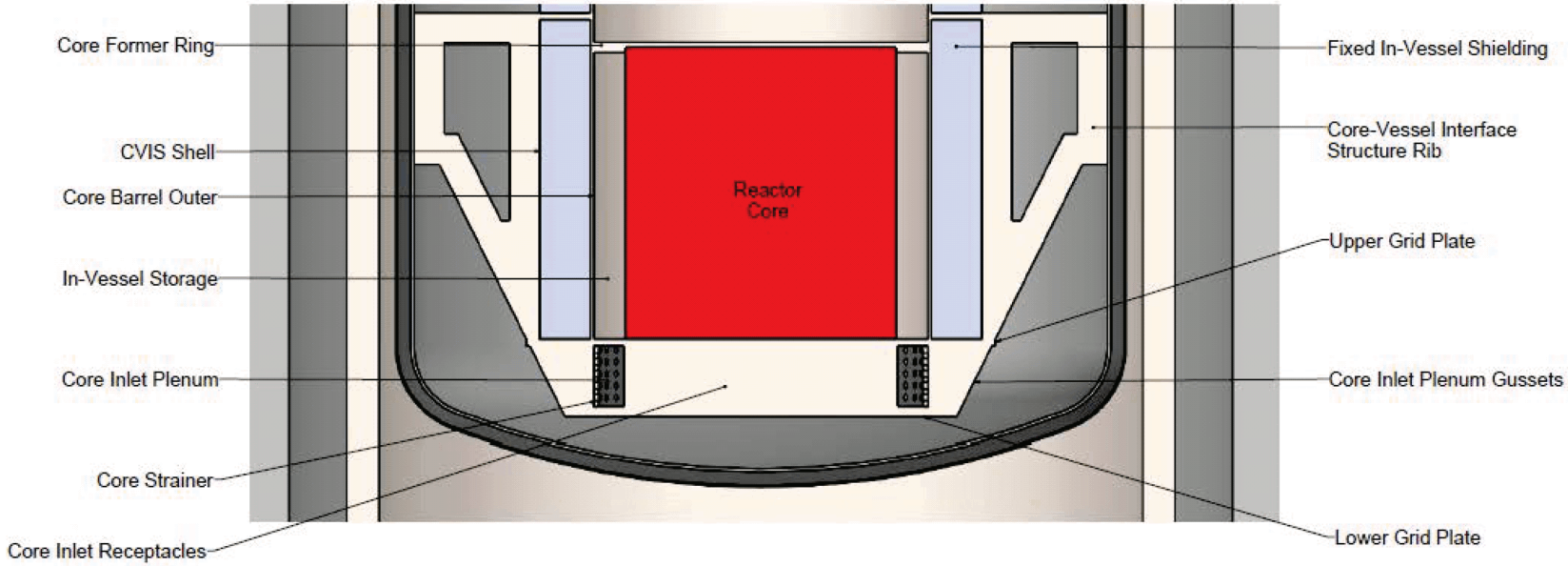




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**Figure 7.1.2-6 Simplified Guard Vessel General Arrangement**

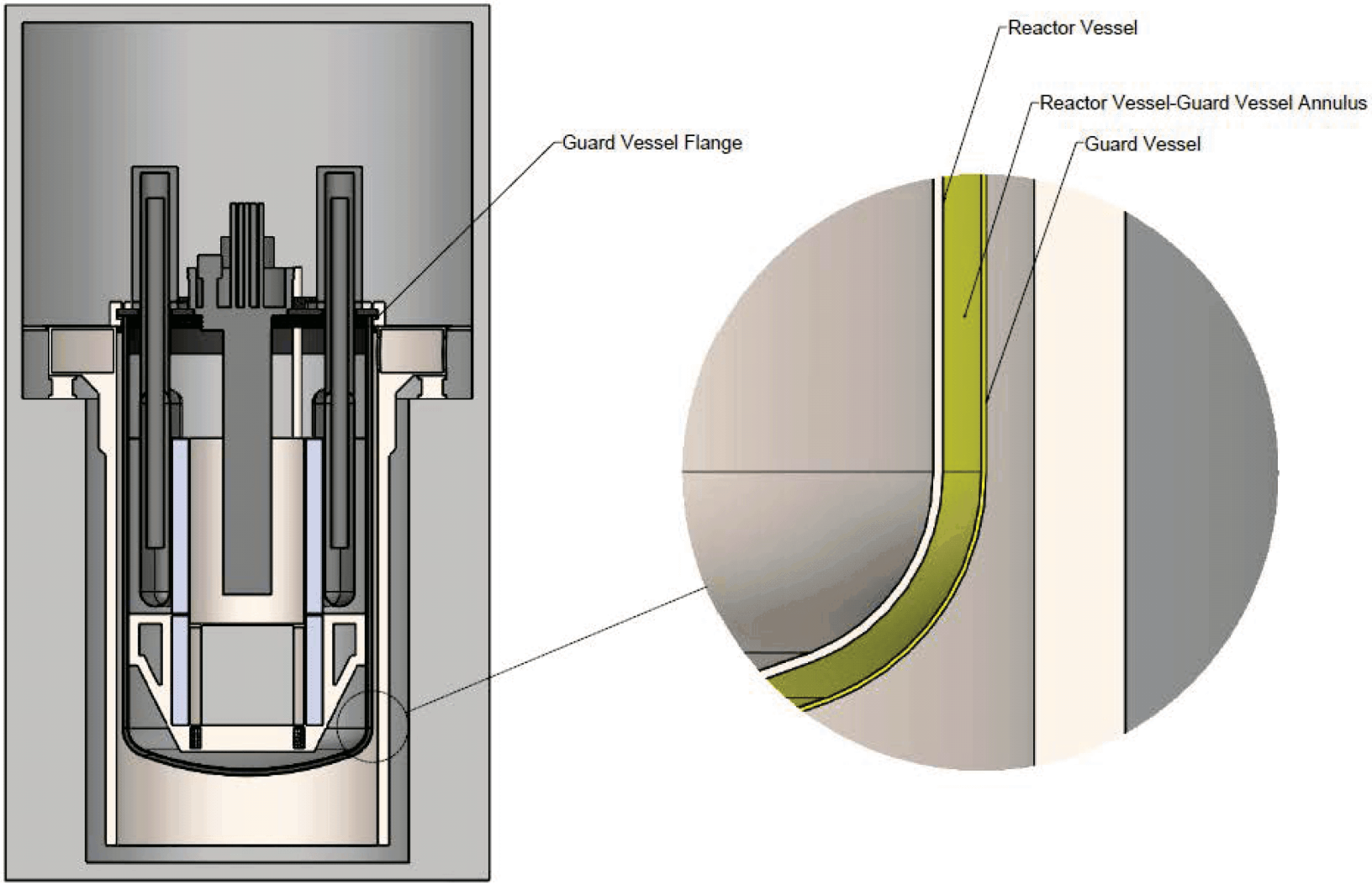






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**Figure 7.1.2-7 Simplified Reactor Support Structure Arrangement**

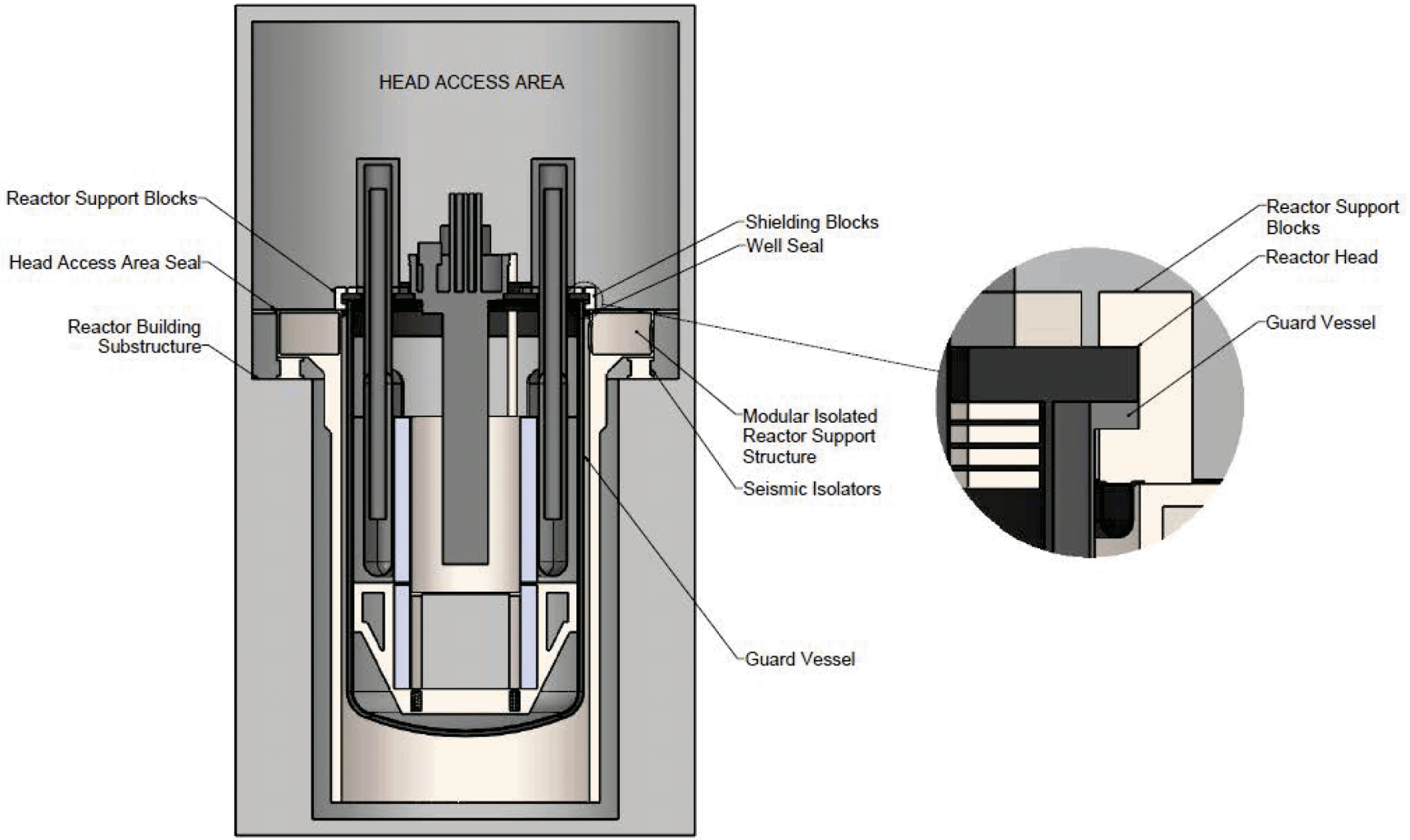



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**Figure 7.1.2-8 Primary Coolant Flowpath**

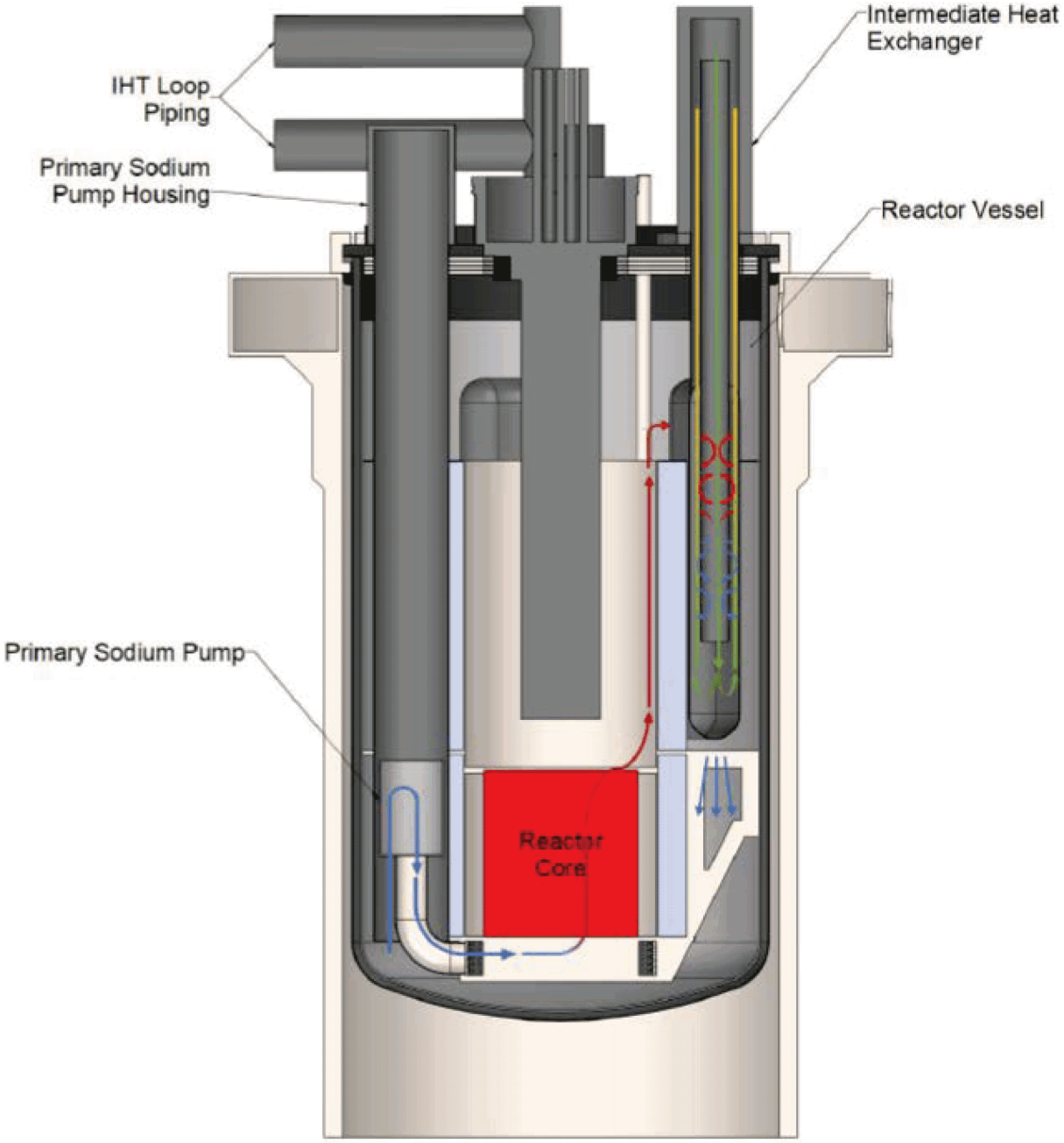



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**Figure 7.1.2-9 Sodium Pool Regions**

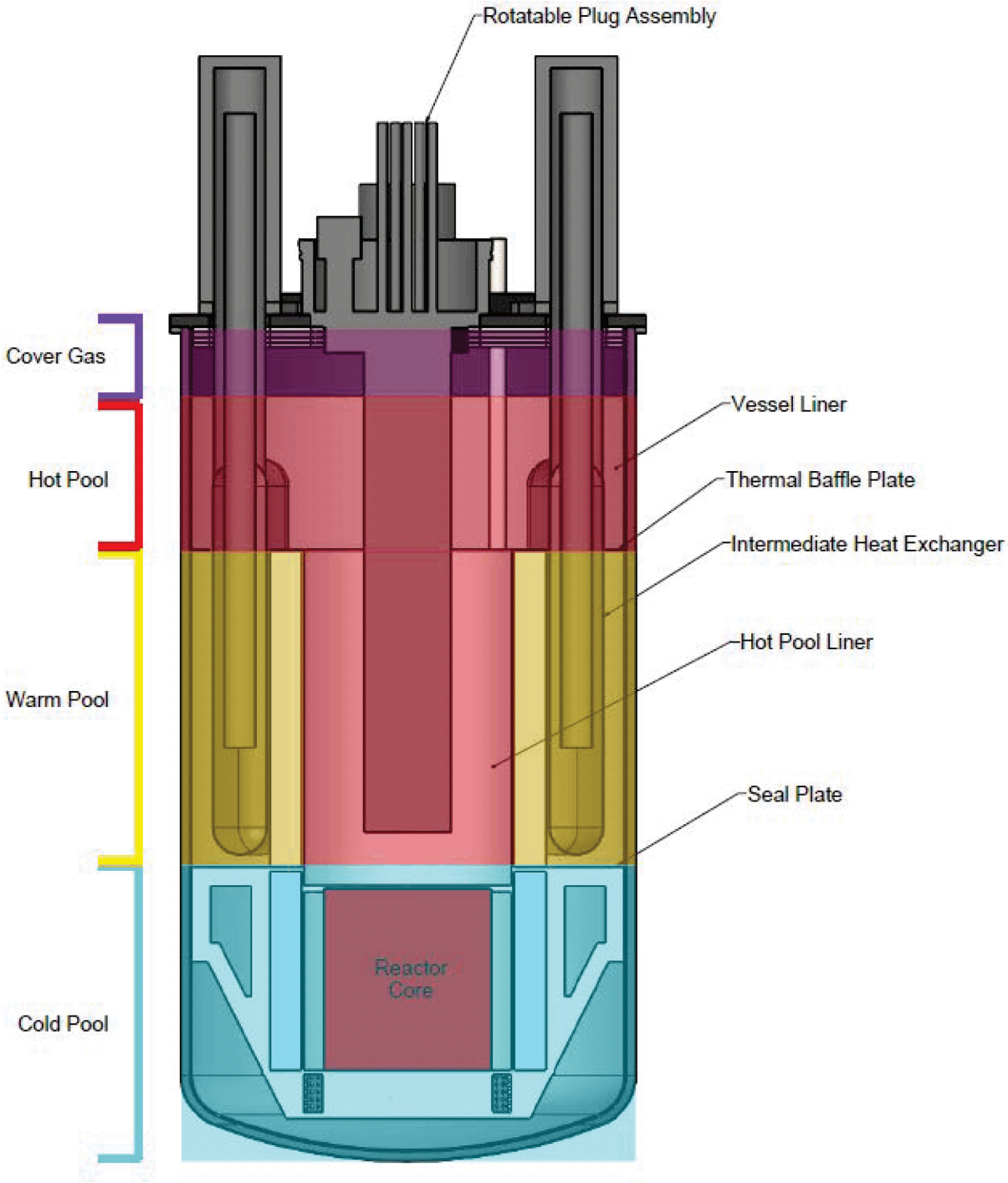




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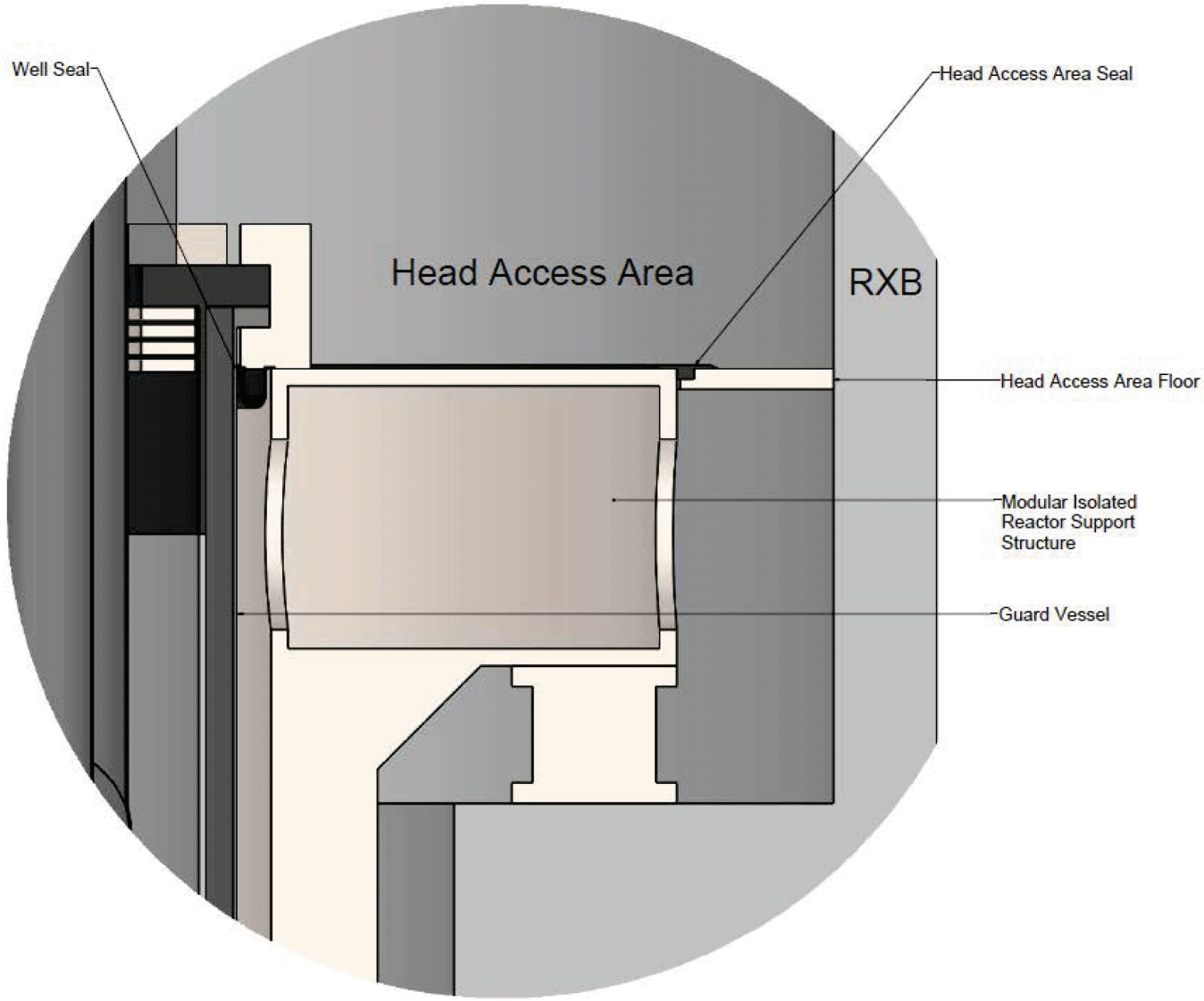
**Figure 7.1.2-10 Simplified Diagram of Well Seal and HAA Seal**





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## 7.1.3 Primary Heat Transport System

### 7.1.3.1 Summary Description

The Primary Heat Transport System (PHT) includes two Intermediate Heat Exchangers (IHXs), two Primary Sodium Pumps (PSPs), and three sodium pools within the Reactor Vessel (RV) (hot pool, cold pool, and warm pool), all located in the Reactor Enclosure System (RES) (see Section 7.1.2). The PSP flanges, IHX flanges, and PSP motors are located above the reactor head. The PSP isolation circuit breakers are located in the Nuclear Island Control Building. The PSP adjustable speed drives are located in the Nuclear Island Electrical Building. The primary function of the PHT is to transport heat from the reactor core to the Intermediate Heat Transport System (IHT) for eventual use in power generation. A safety-significant function for the PHT is to maintain primary sodium temperatures within design limits that preclude damage to the reactor, fuel, or reactor internals. During normal operation, plant start-up, shutdown, refueling, and during certain licensing-basis events (LBEs), two PSPs drive primary sodium flow through the PHT circuit. Two IHXs facilitate heat transfer from primary sodium to intermediate sodium, which ultimately transfers heat to the energy island for electricity production. During certain LBEs the PHT passively facilitates natural circulation of the primary sodium in order to transfer heat from the reactor core to the RV walls, where heat is exchanged to the Guard Vessel (GV) and to the Reactor Air Cooling System (RAC) (see Section 7.2.1). A simplified PHT flow diagram is provided in Figure 7.1.3-1.

The primary coolant inventory (primary sodium) is maintained within the RV. The primary coolant inventory includes the primary sodium volumes of the hot pool, cold pool, and warm pool, along with the primary sodium volumes in the PSPs, IHXs, and Reactor Core System (see Section 7.1.1). The void area between the hot pool and the reactor head is inerted using argon gas provided by the Sodium Cover Gas System (SCG) (see Section 7.2.3). The Sodium Processing System (SPS) draws primary sodium from the reactor, processes it via SPS purification and monitoring equipment, then returns purified sodium to the reactor (see Section 7.2.4). The PHT is a safety-related (SR) system. SR subsystems and components of PHT are described in further detail in Section 7.1.3.2 (PSP) and Section 7.1.3.3 (IHX).

The primary coolant flows in a loop through the reactor core, hot pool, IHXs, cold pool, and PSPs. The same flow path is applicable during normal operations and during natural circulation conditions when the PSPs are unavailable. Cold primary sodium enters the PSP at the PSP tank inlet (below the cold pool seal plate) and travels upward through the pump tank. The sodium then flows through the PSP inlet, downward through the impellers, and exits the pump at the PSP discharge pipe. Sodium travels through the discharge pipe and enters the core inlet plenum. The core inlet plenum distributes sodium up into the core receptacles, which in turn feeds the individual core assemblies via their inlet nozzles. Sodium passes up through the fuel region of the core, cooling the core and heating the coolant, and exits the core into the hot pool. Sodium is drawn into each IHX perforated shell inlet. Sodium travels down through a series of staggered baffles within the IHX. Heat is transferred from the primary sodium to the intermediate sodium as the primary sodium travels downward through the IHXs. Intermediate sodium from the IHT flows upward through the tube side of the IHX. The cooled primary sodium exits the IHX into the cold


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pool, completing the sodium flow cycle. The flow path represents a closed cycle driven by the PSPs during normal operation, or driven by natural circulation between the hot and cold pools when the PSPs are not operating.

Heat generated by the reactor core is transferred to the IHT (see Section 7.1.4) during normal operations via primary coolant circulation flow through the IHXs driven by the PSPs. PHT passively transfers heat to RAC during normal conditions and LBEs. During LBEs, heat generated by the reactor core is transferred to the RAC (see Section 7.2.1) via natural circulation (see Figure 7.1.3-2). Heat transfer to IHT occurs in the IHXs. Heat transfer to RAC occurs along the RV walls. During natural circulation the flow path is unchanged, and flow is driven by the natural convection of primary sodium exchanging heat through the IHXs to IHT and RES to RAC (if heat transfer to IHT is not available, all decay heat is transferred to RAC).

The RES structures, systems, or components (SSCs) (see Section 7.1.2) provide the flow path for the primary coolant to transfer heat from the reactor core to IHT and RAC. In the event forced primary coolant flow is not available from the PSPs, the RES configuration is designed to maintain natural circulation flow through the reactor core.

#### 7.1.3.1.1 Design Basis

Compliance with regulatory criteria, identification of design criteria, and codes and standards that define the PHT safety design basis are described below. Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the PHT is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the PHT is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels are described in Section 6.1.1. The PHT is located within, and protected by, the reinforced concrete RXB substructure. The location within the RXB provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions as described in Section 7.8.1. The PHT is designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1. PHT components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions.

Consistent with PDC 3, *Fire protection*, the design of the PHT minimizes the probability and effect of fires through design features to minimize the potential for sodium leakage to the surroundings and through the selection of non-combustible and non-fire sustaining materials, to the extent practical. The potential for sodium-air reactions are minimized by the use of an inert cover gas over the primary coolant free surface. The PSP includes an oil collection system designed to contain leaked lubricating oil as a fire prevention feature. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the PHT safety-significant SSCs are designed to address environmental effects including sodium environment, thermal conditions, radiation fluence, and dynamic effects of fluid during normal operating conditions, maintenance conditions, testing operations, and postulated accidents. PHT


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safety-significant SSCs that form part of the primary coolant boundary are designed per the requirements of ASME BPVC Section III, Division 5 (Reference 7.1.3-2) (as endorsed with exceptions and clarifications in Regulatory Guide 1.87). PHT safety-significant SSCs within the primary coolant boundary are designed to ensure material compatibility with liquid sodium and its aerosols, vapors, and oxidation products. Proper material selection ensures PHT safety-significant SSCs fulfill design life requirements while accommodating thermal effects, such as creep, thermal expansion, and thermal striping. Requirements for protecting pressure boundary integrity of safety-significant SSCs are performed under the Reliability and Integrity Management (RIM) Program (See Chapter 8), per ASME BPVC Section XI, Division 2 (Reference 7.1.3-1) as endorsed with exceptions and clarifications by Regulatory Guide 1.246. Safety-related PHT SSCs are designed, analyzed, tested, verified, and qualified for form, fit, and function per the Equipment Qualification Program (See Chapter 8). In accordance with Regulatory Guide 1.20, the Comprehensive Vibration Assessment Program (see Chapter 8) ensures reactor internals and any secondary systems that may transfer vibrations to the reactor internals are designed to accommodate dynamic fluid effects such as flow-induced vibration, sloshing, and erosion. The low operating pressure of the design precludes high energy rupture and dynamic pipe breaks. PHT primary sodium containing piping is located within the RV.

Consistent with PDC 10, *Reactor design*, the PHT is designed, with appropriate margin, to prevent exceeding specified acceptable system radionuclide release design limits (SARRDLs) by ensuring a forced circulation of primary coolant during normal operations, as well as a passive primary coolant circuit during off normal conditions, providing adequate heat transfer from the Reactor Core System to the PHT during any condition. Radionuclide release due to primary sodium leakage is minimized by the integrity of the primary coolant boundary SSCs. PHT safety-significant SSCs that form part of the primary coolant boundary are designed per the requirements of ASME BPVC Section III, Division 5. ASME SSCs are designed to satisfy ASME code stress limits under service level conditions (including radiation effects) and bounding load combinations (including temperature and pressure).

Consistent with PDC 11, *Reactor inherent protection*, the PHT is designed to maintain a stable and uniform core inlet temperature, thereby ensuring that inherent nuclear feedback characteristics of the reactor core design tend to counteract any rapid increases in reactivity in the power operating range.

Consistent with PDC 12, *Suppression of reactor power oscillations*, the PHT PSPs maintain the flow of coolant, which helps to ensure uniform temperature distribution within the reactor core to prevent the development of hot spots which could lead to power oscillations. The IHXs facilitate efficient removal of heat from the primary sodium during normal operations, ensuring the primary sodium is at the desired temperature.

Consistent with PDC 13, *Instrumentation and control*, the PHT process parameters such as pump speed, temperature, and sodium level are monitored by instruments within the RV. Depending on the safety classification, the signals are processed by either the Nuclear Island Control System (NIC) or Reactor Protection System (RPS).


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Consistent with PDC 14, *Primary coolant boundary*, the primary coolant boundary components are designed, fabricated, erected, and tested to provide an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The primary coolant boundary operates at low pressure, and metallic boundary components are designed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5. Provisions are provided for leak testing primary coolant boundary components in accordance with applicable requirements of the ASME Code. Over-pressure protection of the primary coolant boundary is provided per the requirements of the ASME Code.

Consistent with PDC 15, *Primary coolant system design*, the design of the PHT provides sufficient margin such that the design conditions of the primary coolant boundary are not exceeded during conditions of normal operation, including AOOs. The primary coolant boundary operates at low pressure and is designed and constructed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5. Sufficient margin to design conditions is provided by the design requirements established as described in Section 6.4.3. Over-pressure protection of the primary coolant boundary is provided per the requirements of the ASME Code.

Consistent with PDC 16, *Containment design*, the design of the primary coolant boundary supports functional containment to control the release of radioactivity to the environment and provide assurance that functional containment design conditions are not exceeded.

Consistent with PDC 30, *Quality of primary coolant boundary*, the safety-related primary coolant boundary is designed, fabricated, erected, and tested to quality standards established as described in the Quality Assurance Program Description (Section 8.1). The primary coolant boundary components are designed and constructed in accordance with the requirements of ASME BPVC, Section III, Division 5. Means are provided for detecting and, to the extent practical, identifying the location of the source of primary coolant leakage. Pre-service and in-service monitoring and non-destructive examination (MANDE) requirements implemented by the Reliability and Integrity Management Program (described in Chapter 8) for the primary coolant boundary provide assurance that quality requirements are met.

Consistent with PDC 31, *Fracture prevention of primary coolant boundary*, the primary coolant boundary components of the PHT are designed with sufficient margin to provide assurance that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. PHT primary coolant boundary metallic components are designed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5 for Class A components, as applicable. Selection of PHT primary coolant boundary materials of construction considers material compatibility with liquid sodium and its aerosols, vapors, and oxidation products. The PHT primary coolant boundary component design and material selection is made with consideration of service temperatures, service degradation of material properties, and applicable degradation mechanisms. Metallic primary coolant boundary components are constructed of austenitic stainless steel to minimize the effects of irradiation embrittlement and fracture from thermal stresses. The Degradation Mechanism Assessment is performed in accordance with the Reliability and Integrity


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Management Program, which meets the requirements of ASME BPVC Section XI, Division 2 (Reference 7.1.2-2) as endorsed with exceptions and clarifications by Regulatory Guide 1.246, Revision 0.

Consistent with PDC 32, *Inspection of primary coolant boundary*, the design of the primary coolant boundary permits periodic inspection and functional testing of important areas and features to assess structural and leaktight integrity through implementation of the MANDE pre-service and in-service inspection requirements of the Reliability and Integrity Management Program. The PHT provides access for implementing the RIM Program for all primary coolant boundary components. Access for inspections and inspection equipment is provided at the reactor head interfaces with adjacent SSCs. The PHT also supports access to SSCs, including access ports on the IHXs. The PHT facilitates the removal of insulation or other components as needed to accommodate inspection activities, and the reactor head is designed for complete removal of PHT equipment. PHT is designed to allow periodic testing of primary coolant boundary SSCs.

Consistent with PDC 33, *Primary coolant inventory maintenance*, the pool-type configuration of the reactor, with no penetrations below the primary sodium, minimizes the possibility of a primary system leak or rupture. As such, PHT SSCs do not perform an inventory maintenance function.

Consistent with PDC 34, *Residual heat removal*, the PHT supports residual heat removal by providing flow to remove decay heat and other residual heat from the reactor core and transfer it to IHT via the IHXs. The PHT can provide flow with use of the PSPs or by way of natural circulation through the system. Depending on the operating mode and conditions, either the IHT or Intermediate Air Cooling System (IAC) are used to transfer the heat from the IHX.

Consistent with PDC 35, *Emergency core cooling*, in the event the PSPs or the IHXs are not functional, the PHT supports emergency core cooling by maintaining sufficient primary sodium inventory and providing passive flow to the core for heat removal via natural circulation. The natural circulation circuit includes the RES RV and vessel internals. The heat removed from the reactor core is exchanged with the RV wall, where it is transferred into the RV-GV annulus space and passively removed via convection and radiation by RAC. RAC provides passive decay heat removal through natural convection of air upon loss of other means of cooling. The PSPs include isolation circuit breakers which are tripped when excessive temperatures are reached in the primary sodium. This establishes the initial conditions for emergency core cooling (natural circulation) and eliminates non-decay heat addition to the primary system. These pump trips ensure SARRDLs are not exceeded as a result of anticipated operational occurrences.

Consistent with PDC 36, *Inspection of emergency core cooling system*, the SSCs of the primary natural circulation path fall under the RIM Program per ASME BPVC Section XI, Division 2. The RIM Program assigns monitoring and non-destructive examination (MANDE) strategies such that the reliability targets are maintained over the life of the plant. The PHT design provides access for implementing monitoring and non-destructive examination strategies. The PSP isolation circuit breakers are designed to permit periodic inspection.


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Consistent with PDC 37, *Testing of emergency core cooling system*, the PHT SSCs contributing to the natural circulation circuit are passive. The monitoring and non-destructive examination strategies are implemented by the RIM Program for passive components that may impact safety. See also PDC 36. The PSP isolation circuit breakers are designed to permit periodic testing. There is capability to test periodically the operability and functional performance of these breakers, and the operability of the PSP trips as a whole.

Consistent with PDC 71, *Primary coolant and cover gas purity control*, primary sodium purity and cover gas purity are managed by SPS and SCG respectively. Interfaces to these systems are provided by the RES.

Consistent with PDC 72, *Sodium heating systems*, the PHT maintains primary sodium above the freezing temperature of liquid sodium with adequate margin. When decay heat is unavailable or insufficient to maintain adequate sodium temperatures, the PHT SSCs can add heat to the primary sodium via kinetic energy of the PSPs and friction losses throughout PHT.

Consistent with PDC 74, *Sodium/water reaction prevention/mitigation*, the PHT is located within the RXB substructure, which does not house water-filled systems to limit the potential for contact between primary sodium and water. Protection from environmental conditions that could result in water intrusion into the RXB is provided as described in Section 7.8.1.

Consistent with PDC 78, *Primary coolant system interfaces*, the intermediate sodium is compatible with the primary coolant and the interface in the IHX consists of a single barrier. The intermediate sodium is maintained at a higher pressure than primary sodium during normal operation, anticipated operational occurrences, and accident conditions such that potential leakage within the IHX flows into the primary system to prevent loss of core cooling capability. The Reactor Vessel cover gas space accommodates increased primary coolant inventory due to IHX leakage and the Reactor Protection System initiates reactor scram and intermediate system pump trip on a high primary coolant level condition such that primary coolant system safety-significant functions are maintained. The PSP lube oil system is designed such that a double barrier exists between lube oil to primary sodium.

Functional Requirements and Design Criteria

Probabilistic risk assessment safety functions (PSFs) are determined through the plant-specific probabilistic risk assessment described in Section 3.1. Safety-significant PSFs are identified as SR functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The PHT supports the following SR and NSRST functions:


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| Function ID | Function Description | PHT Functional Support |
|---|---|---|
| DL3-HR1 | PSP Coastdown (SR) | In the event of PSP trip, the PSPs have sufficient rotational inertia to provide coastdown flow for a smooth transition to natural circulation without flow reversal in the fuel channels of the reactor core. PSP coastdown assures that fuel design limits are not exceeded during DBAs (SR function) and minimizes the thermal transient experienced by the reactor vessel and its internals (NSRST function). |
| DL3-HR2 | PSP Trip on High High Primary Sodium Temperature (SR) | The PSP circuit breakers trip on a combination of a low neutron flux signal, existence of a scram signal, and cold pool temperature at the high high temperature setpoint. This function reduces additional heat load created by continued pump operation. |
| DL3-HR5 | Natural Circulation of Sodium in Primary System (SR) | Natural circulation of the primary sodium flow path is achieved and maintained, following pump coastdown, with sufficient flowrate to ensure adequate core cooling. PHT components within the primary sodium flow path remain unobstructed to allow natural circulation. |
| DL3-RR1a | IHX Primary System Barrier (SR) | IHX components that form part of the primary coolant boundary have sufficiently low leakage to assure that SARRDLs are not exceeded. |
| DL3-RR1d | Primary Sodium Pump Seal (SR) | PSP components that form part of the primary coolant boundary have sufficiently low leakage to assure that SARRDLs are not exceeded. |
| DL4-HR1 | IAC Passive Mode Operation (NSRST) | Natural circulation of the primary sodium flow path is achieved and maintained, following pump coastdown, with sufficient flowrate to ensure adequate core cooling. PHT components within the primary sodium flow path remain unobstructed to allow natural circulation. |
| DL4-HR2 | PSP Trip Automatic Backup (NSRST) | The PSPs are automatically run to minimum flow or tripped on logic that is diverse from the SR PSP trip function DL3-HR2. This function provides alternative means of preventing pump heat addition if RPS fails to trip the pumps. |
| DL4-HR6 | Manual PSP Trip (NSRST) | As an alternate means of preventing pump heat addition, the PSPs allow for manual trip. This function is available to support long term response during transients (e.g., if the pumps fail to reduce flow in a runback or after a scram). |

The PHT primary coolant boundary components are designed to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The PHT primary coolant boundary components, include:

- IHX mounting plate and seal interface of the IHX with the reactor head


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- PSP support flange, PSP shaft seal, and seal interface of the PSP with the reactor head
- IHX tube bundle and tubesheets
- IHX coaxial intermediate sodium supply and return piping within the reactor
- IHX shell above upper tubesheet
- IHX shell below lower tubesheet

The PHT primary coolant boundary components are designed to ASME BPVC Section III, Division 5, “High Temperature Reactors” (Reference 7.1.3-2) subject to the limitations of Regulatory Guide 1.87, as applicable.

An analysis and evaluation of the need for high point vents in accordance with the requirements of 10 CFR 50.34(a)(4), 10 CFR 50.34(f)(2)(vi), and 10 CFR 50.46a has been performed. High point vents are not required to maintain core cooling because the accumulation of noncondensible gases does not cause the loss of function of emergency core cooling systems.

Instrumentation is available to monitor PHT parameters, including indication of coolant temperatures and indication of coolant level in the reactor in accordance with 10 CFR 50.34(f)(2)(xviii). Instrumentation for primary sodium signals meet the requirements of 10 CFR 50.55a(h), as applicable. Table 7.1.3-1 provides a summary of monitored parameters within PHT (specific instrumentation are not all PHT SSCs).

SSC safety classifications are performed as described in Section 5.1. The PSPs, IHXs, and PSP isolation circuit breakers are classified as SR SSCs. The classifications of other major PHT components are described in Section 7.1.3.2 and Section 7.1.3.3. The safety-related design criteria corresponding to the SR functions are derived as described in Section 6.1.2 and identified in Section 5.2. Seismic design, analysis, and qualification are performed as described in Section 6.4. Seismic classifications of SSCs are determined in accordance with the methods described in Section 6.4.1. SSCs that are classified SCS1 are qualified to withstand seismic loads associated with the safe shutdown earthquake (described in Section 2.6.2) without loss of the capability to perform SR functions. Seismic special treatments are applied to PHT SSCs as applicable based on seismic classification.

Regulatory Guidance

Regulatory Guide 1.20, Rev. 4, 2017 “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Startup Testing,” partial conformance. PHT SSCs within the RV are screened and analyzed for effects due to potential excitation mechanisms in accordance with Section C.2. (Includes Flow Induced Vibration, Acoustic Resonance, Acoustic Induced Vibration, and Mechanically Induced Vibration).

Regulatory Guide 1.61, Rev. 1, 2007 “Damping Values for Seismic Design of Nuclear Power Plants,” full conformance. This guidance provides acceptable component damping factors for PHT SSC seismic response analyses.


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Regulatory Guide 1.87, Rev. 2, 2023 “Acceptability of ASME Code, Section III, Division 5, “High Temperature Reactors,” full conformance. This guidance provides code acceptability, limitations, and restrictions regarding the use of ASME BPVC Section III, Division 5, and is applicable to the design of PHT primary coolant boundary SSCs.

Regulatory Guide 1.100, Rev. 4, 2020 “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” partial conformance. This guidance is used for the functional qualification of active mechanical equipment and is applied to PSPs, as applicable.

Regulatory Guide 1.189, Rev. 4, 2021 “Fire Protection for Nuclear Power Plants,” partial conformance. PHT sodium-containing equipment and components are designed with non-combustible and non-fire sustaining materials.

Regulatory Guide 1.192, Rev. 4, 2021 “Operation and Maintenance Code Case Acceptability, ASME OM Code,” full conformance. This guidance is applicable to inform in-service inspection of PHT components.

#### 7.1.3.1.2 System Design and Performance Evaluation

The PHT is located in the RV (with the exception of IHX and PSP components above the reactor head), in contact with primary sodium and intermediate sodium. No operator actions are required to implement PHT SR functions. PSP and IHX preliminary performance characteristics are described in Section 7.1.3.2 and Section 7.1.3.3.

During normal operation, in all modes of operation (Modes 1, 2, 3, and 4), the PSPs provide motive force for coolant flow within the reactor from the cold pool, through the core to the hot pool, and through the IHX to the cold pool.

During LBEs, the PHT responds to events using various defense line functions. The PHT supports power runback, reactor scram, and loss of offsite power as described below.

A power runback event initiates an orderly reduction in power of the reactor at controlled pace (e.g., not a sudden reduction in power like for a reactor scram) to reduce the impact of the transient on plant equipment. During power runback events, the PHT flow control is initiated (this signal is coincident with a power runback signal but follows reactor power). The power-to-flow ratio is kept less than or equal to one during the power runback. Both PSPs gradually reduce flow; PSP flow is automatically controlled to follow reactor power. Reactor power and PSP flow are continually reduced until desired power is reached. Sodium continues to cool down by active and passive means. The IAC (See Section 7.2.2) is transitioned to active mode. Primary sodium also cools via RAC (see Section 7.2.1). The PHT power runback functions are not safety-significant functions.

A reactor scram involves the automatic shutting down of the reactor by the RPS via rapid insertion of control rods. During a reactor scram the PSP ramp down functions are initiated (the ramp down signal is coincident with a reactor scram signal) and the PSP flow is reduced using a predefined ramp down curve to reach the predefined minimum flow. If a pump fails to ramp down, the PSPs trip, pump coastdown is initiated on high high primary sodium temperature signal, and


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a reduction in PSP flow occurs as part of the coastdown sequence. After the PSPs complete their ramp down or coastdown sequence, the pumps will be restarted when available. Sodium continues to cool down by active and passive means. The IAC is transitioned to active mode.

A loss of offsite power event has similar PHT response as a reactor scram event, with the exception that when power is lost to the PSPs, the pumps transition to coastdown. After the PSPs complete their coastdown sequence, the pumps transition to a minimum flow if power is restored via the standby diesel generators. If power is not restored, flow reduces to natural circulation flow rates. Sodium continues to cool down by passive means (or active means if available). Sodium continues to cool down passively via RAC natural circulation and IAC natural circulation. If power is restored by the standby diesel generators, active IAC cooling is also available.

The design of the PHT accounts for the effect of radiation on material properties to accommodate the expected radiation environment over the SSC operating life. PHT components that are part of the primary coolant boundary are designed to ensure that the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. Preliminary design temperature, service environment, and selected materials of construction for the PHT are provided in Table 7.1.3-6.

The PHT provides instrumentation for primary sodium signals to the RPS and NIC to inform and enable safe control of the reactor during normal conditions and LBEs. See Table 7.1.3-1 for a summary of monitored parameters within PHT (specific instrumentation are not all PHT SSCs).

#### 7.1.3.1.3 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the PHT include ASME BPVC Section III, Division 5 and ASME BPVC Section XI, Division 2 as discussed above and in Section 7.1.3.2 and Section 7.1.3.3. The following programmatic special treatments are preliminarily identified as applicable to the PHT to meet quality, design, reliability, and performance requirements (summary program descriptions are provided in Chapter 8):

- Quality Assurance Program
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- Equipment Qualification Program
- Comprehensive Vibration Assessment Program
- Post-construction Inspection, Testing, and Analysis Program (see Chapter 12)

#### 7.1.3.1.4 Instrumentation Requirements

No operator actions are required to implement PHT SR functions. See Table 7.1.3-1 for a summary of monitored parameters within PHT (specific instrumentation are not all PHT SSCs). The Reactor Instrumentation System (see Section 7.6.5) includes instrumentation for PHT


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parameters which are processed by the RPS (see Section 7.6.3). The Coolant Temperature Monitoring and Control System, part of the NIC (see Section 7.6.2), monitors and controls PHT flow functions. Controls and displays are described in Section 7.6.7.

#### 7.1.3.2 Primary Sodium Pump

##### 7.1.3.2.1 Summary Description

The two PSPs transport heat from the reactor core to the IHT for use in power generation (see Section 7.1.4). During normal operations, plant startup and shutdown, and during certain LBEs, the PSPs provide the necessary head to drive primary sodium flow.

The PSPs are two identical vertically-oriented centrifugal pumps located on opposite sides of the RV. The PSPs provide the primary sodium flow in the reactor by drawing sodium from the cold pool and discharging into the reactor core inlet plenum. The PSPs are installed in the RES and housed within a pump tank (one per pump). The inlet of the pump tank is the cold pool. The pump tank, which forms a part of the boundary between the hot pool and cold pool, allows flow around the exterior of the PSPs to the pump inlet, directing the flow downward to the pump impellers. The pump shaft is driven by a motor mounted above the reactor head. The shaft passes through a seal mounted to the reactor head and enters the pump tank in the cover gas region. The shaft is enclosed in the support column within the pump tank. The shaft terminates at its connection to the impellers.

The PSPs are anchored to and supported by the reactor head. Below the reactor head, each PSP pump tank is laterally, vertically, and rotationally anchored by the seal plate. The pump shaft is sealed near the support flange to prevent sodium and cover gas leakage to the air outside of the RES. The seal also limits leakage of both air and moisture into the RV. The support flange and the non-metallic shaft seal form part of the primary coolant boundary. The pumps have seals at the pump discharge interface to the piping that directs flow towards the core inlet plenum. The seals are designed to allow the pumps to be inserted into the RV from above and provide for relative vertical motion between the pumps and discharge pipe.

Sodium coolant is drawn up from the cold pool into the intake on the pump case. The intake turns the coolant downward, and directs it to the impellers. The discharged sodium exits the PSP through the pump discharge piping into the reactor core inlet plenum. The sodium flow rate is controlled by varying the pump motor speed, which varies the impeller speed.

The PSPs provide sufficient coastdown momentum during reactor scram, PSP trips, and loss of offsite power to establish natural circulation flow sufficient to assure that fuel design limits are not exceeded.

Each PSP is supported by a lube oil system that provides lubricant to the mechanical bearings of the pumps. The mechanical bearings and lube oil system are both located outside of the primary coolant boundary.


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A simplified sectional view of a PSP is provided in Figure 7.1.3-3. The safety classification of the PSPs is SR, and the safety classifications for major PSP components are provided in Table 7.1.3-2. The PSP lube oil system, motor, and adjustable speed drive are all classified as NST.

7.1.3.2.2 Design Basis

The primary coolant boundary components of the PSPs are designed and constructed to ASME BPVC Section III, Division 5, Class A standards (Reference 7.1.3-2), subject to the limitations of Regulatory Guide 1.87 as applicable.

Seismic classifications for PSP components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The overall seismic classification of PSP is SCS1. Seismic design, analysis, and qualification is performed as described in Section 6.4. PSP SCS1 components are qualified to withstand seismic loads associated with the safe shutdown earthquake (described in Section 2.6.2) without loss of the capability to perform SR functions.

The PHT includes circuit breakers that are used to trip the PSPs on demand and automatic trip signals. IEEE 344-2013 (Reference 7.1.3-3) is used for the seismic qualification of the isolation circuit breakers of the PSPs.

7.1.3.2.3 System Design and Performance Evaluation

Preliminary design performance characteristics for the PSPs are identified in Table 7.1.3-3.

The PSPs are the only PHT components that require operator inputs to change state according to the desired conditions during normal operation.

During normal operations, the PSPs are continuously in operation. While at full power, the PSPs operate at full flow. During refueling and shutdown, the PSPs operate at low flow, circulating primary sodium throughout PHT. The PSPs are designed to operate at variable flow rates during extended shutdowns where decay heat is no longer available as needed to provide heat to maintain the sodium inventory in a liquid state.

During startup of the reactor, the PSPs raise the primary sodium temperature to the hot zero power temperature. After hot zero power temperature is reached, flow is reduced to a minimum. As power is slowly increased in the reactor, the PSPs increase PHT flow. Upon reaching full power, the PSPs continuously operate at full flow unless called to do otherwise by operator action, the NIC, or the RPS.

During LBEs, the PSPs are automatically controlled by the NIC or the RPS. Defense line functions involving the PHT are characterized as power runbacks, reactor scrams, or loss of offsite power.

During a power runback, the reactor power and PSP flow are ramped down in an orderly fashion in response to a plant event to prevent an automatic reactor scram.


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During reactor scrams, the PSPs are ramped down using a predefined ramp down curve to minimum flow. However, under certain conditions after a reactor scram, the PSPs can automatically trip resulting in coastdown. After completion of ramp down or coastdown, the PSPs transition to minimum flow.

During loss of offsite power, a sequence of events similar to a reactor scram follows, except coastdown is initiated immediately when power is lost to the PSPs.

The PSPs are designed such that a natural flow path through them exists when the shaft is stopped.

The PSP case and support flange are designed with sodium-compatible metallic materials per ASME BPVC Section III, Division 5, including weld materials, with the restrictions and limitations identified in Regulatory Guide 1.87. The PSP metallic components, including PSP primary coolant boundary components as well as the support column, pump shaft, and impellers, are constructed from 304 or 316 stainless steel. The materials for other PSP components such as discharge nozzle seals, shaft seals, and shaft bearings are designed using sodium-compatible materials.

The PSP design accounts for the effects of radiation on the material properties of the components. The PSPs are designed to accommodate the expected radiation environment over their operating life. The PSPs form part of the primary coolant boundary and are designed to ensure that the boundary behaves in a non-brittle manner, and the probability of rapidly propagating fracture is minimized.

#### 7.1.3.2.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the PSPs include ASME BPVC Section III Division 5 (Reference 7.1.3-2), ASME BPVC Section XI Division 2 (Reference 7.1.3-1), and ASME QME-1 (Reference 7.1.3-4).

#### 7.1.3.2.5 Instrumentation Requirements

The PSPs are provided with various instruments to ensure proper operation of the pumps. The Reactor Instrumentation System (Section 7.6.5) provides instrumentation for primary sodium signals associated with the PHT parameters to monitor primary sodium conditions, as well as equipment conditions. See Table 7.1.3-1 for a summary of the PHT related instrumentation parameters.

### 7.1.3.3 Intermediate Heat Exchanger

#### 7.1.3.3.1 Summary Description

There are two IHXs in the PHT that transfer heat from the primary sodium of the PHT to the intermediate sodium of IHT (see Section 7.1.4). During normal operations, the heat transported to the IHT is ultimately transferred to the EI for power generation. The IHXs are vertical, counter-flow, single-pass, straight-tube type heat exchangers, having a vertical single-pass shell


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and a floating lower head. Each IHX exchanges heat between two liquid sodium circuits: the primary sodium, and the intermediate sodium. Each IHX penetrates the reactor head, with the shell and tubes fully submerged below the primary sodium surface, and are supported from the top by the reactor head at an IHX mounting plate. The IHT intermediate sodium cold and hot leg piping connects to the IHX above the reactor head.

Flow in the IHX includes the primary sodium on the shell side and the intermediate sodium on the tube side. Primary sodium enters the IHX through shell perforations below the upper plenum and flows downward over a series of horizontal baffles and through the tube bundles. This stream then exits the IHX into the cold pool region through two outlet nozzles at the bottom. The cold intermediate sodium stream enters the cold leg downcomer at the top of the IHX (via the coaxial intermediate sodium inlet pipe), flows vertically downward to the bottom head, then turns upward to be evenly distributed to the tubes by the floating lower plenum. As the intermediate sodium stream flows upward through the IHX tubes, it is heated by the primary sodium which is flowing past the tubes in the opposite direction. The intermediate sodium exits the individual tubes at the upper plenum and flows upward to exit the IHX via the coaxial intermediate sodium outlet pipe.

The two IHXs have identical geometry. The IHXs are supported at the mounting plates which maintain the position of the IHX bodies to preserve the primary sodium flow path through the IHXs and the interfacing RV internal structure. The IHX primary sodium side is connected directly to the PHT hot and cold pools.

A simplified diagram of the IHX is provided in Figure 7.1.3-4. The safety classification of the IHX is SR, and the safety classifications for major sub-components of IHX are described in Table 7.1.3-4.

#### 7.1.3.3.2 Design Basis

The primary coolant boundary components of the IHXs are designed and constructed to ASME BPVC Section III, Division 5 Class A standards (Reference 7.1.3-2), subject to the limitations of Regulatory Guide 1.87, as applicable. Non-tubing components of the IHX are constructed from 304 or 304H stainless steel material and tubing components are constructed from 304H stainless steel. Materials selection conforms with the requirements of ASME BPVC Section III, Division 5, including weld materials, with the restrictions and limitations identified in Regulatory Guide 1.87.

#### 7.1.3.3.3 System Design and Performance Evaluation

The IHX primary function during all modes of operation is to transfer heat from PHT to IHT. The IHX supports the PHT SR function to facilitate natural circulation of the primary sodium to transfer heat from the reactor core to the RV walls, where the heat is exchanged to the guard vessel and to the RAC. PHT components, including the IHX, allow a flow path for natural circulation of primary sodium through the RV. No operator actions are required to implement IHX SR functions.

Any potential leakage of IHX components that provide primary to intermediate sodium boundary occurs from the intermediate sodium into the primary sodium. The intermediate sodium of IHT is maintained at higher pressure than the primary sodium of PHT. Leakage between IHT and PHT


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via the IHX tubes does not result in failure of the intended safety functions of the IHX. Leakage of intermediate sodium into primary sodium would be detected by IHT expansion tank level decrease (see Section 7.1.4) and PHT hot pool level increase.

The IHX design accounts for the effect of elevated temperatures in a sodium environment. The IHXs are contained within the RV (with the exception of the hot and cold leg intermediate sodium piping that is above the reactor head) and are in contact with primary sodium and intermediate sodium. The IHX is designed to accommodate the effects of, and to be compatible with, liquid sodium and its aerosols and oxidation products. IHX components are designed to accommodate the effects of thermal expansion and thermal striping as applicable. Preliminary design performance characteristics for the IHXs are identified in Table 7.1.3-5.

The IHX design accounts for the effects of radiation on the material properties of the components. The IHX is designed to accommodate the expected radiation environment over its operating life. The IHX forms part of the primary coolant boundary and is designed to ensure that the boundary behaves in a non-brittle manner, and the probability of rapidly propagating fracture is minimized. The RES provides shielding to the IHXs to limit the activation of intermediate sodium.

There are no active interface requirements to implement IHX SR functions.

#### 7.1.3.3.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the IHX include ASME BPVC Section III, Division 5 and ASME BPVC Section XI, Division 2.

#### 7.1.3.3.5 Instrumentation Requirements

No controls or displays are required to accomplish IHX SR functions because the IHX SR functions are passive functions. No instruments are required to implement IHX SR functions.

## References

7.1.3-1 ASME, “Rules for Inservice Inspection of Nuclear Reactor Facility Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Reactor Facilities,” ASME BPVC Section XI, Division 2, 2019.

7.1.3-2 ASME, “Rules for Construction of Nuclear Facility Components - Division 5 - High Temperature Reactors,” ASME BPVC Section III, Division 5, 2017.

7.1.3-3 Institute of Electrical and Electronics Engineers, “IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations,” IEEE 344-2013.

7.1.3-4 ASME, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities, ASME QME-1,” ASME, 2023.

7.1.3-5 ASME, “Boiler and Pressure Vessel Code, Section II, Part A, Ferrous Material Specifications,” ASME


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**Table 7.1.3-1 Monitored PHT Parameters**

| Parameter | Safety-Significant Signal Function |
|---|---|
| Cold Pool Primary Sodium Temperature | Yes |
| Hot Pool Primary Sodium Temperature | Yes |
| Hot Pool Primary Sodium Level | Yes |
| PSP Speed (Primary Sodium Flow) | Yes |
| PSP Current (Primary Sodium Flow) | Yes |
| PSP Voltage (Primary Sodium Flow) | No |
| PSP Frequency (Primary Sodium Flow) | No |
| Pump Tank Level (Primary Sodium Flow) | No |
| PSP Bypass Pressure | No |
| PSP Bypass Temperature | No |


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**Table 7.1.3-2 PSP Major Components Safety Classifications**

| Component | Safety Classification |
|---|---|
| PSP | SR |
| PSP Support Flange | SR |
| PSP Isolation Circuit Breakers | SR |


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**Table 7.1.3-3 PSP Preliminary Design Performance Characteristics**

| Parameter | Nominal Value |
|---|---|
| Flow @ Full Power Operating Point | 40,000 gpm (150 $m^3$/s) |
| Pump Head @ Full Power Operating Point | 300 ft (90 m) |
| Minimum Hydraulic Efficiency at @ Full Power Operating Point | 80% |
| Coastdown Halving Time | 3.5 - 5 s |
| Design Temperature | 1,050°F (566 °C) |


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**Table 7.1.3-4 IHX Major Components Classifications**

| SSC Name | Safety Classification | Seismic Classification |
|---|---|---|
| IHX Mounting Plate | SR | SCS1 |
| IHX Outer Shell | SR | SCS1 |
| IHX Tube Bundle | SR | SCS1 |


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**Table 7.1.3-5 IHX Preliminary Design Performance Characteristics**

| Parameter | Nominal Value |
|---|---|
| Rated capacity | 1,450 MBtu/hr (425 MWth) |
| Nominal Primary Sodium Inlet Temperature | 950°F (510°C) |
| Nominal Primary Sodium Outlet Temperature | 680°F (360°C) |
| Nominal Intermediate Sodium Inlet Temperature | 620°F (330°C) |
| Nominal Intermediate Sodium Outlet Temperature | 910°F (490°C) |


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**Table 7.1.3-6 Preliminary PHT Design Temperature, Service Environment, and Materials of Construction**

| Component | Base Material* | Weld Material** | Design Temperature | Service Environment | Notes |
|---|---|---|---|---|---|
| PSP Support Flange and Pump Case (pressure boundary) | Type 304 or 316 Stainless Steel | Table HBB-I-14.1(b) | 1050°F | Argon gas, sodium vapor and aerosols | Primary coolant boundary component |
| IHX Non-Tubing Components - Tubesheet, Riser, Mounting Plate | SA-965 F304H<br>SA-240 304H | Table HBB-I-14.1(b) | 1050°F | Sodium | Primary coolant boundary component |
| IHX – Tubing | SA-213 TP304H | Table HBB-I-14.1(b) | 1050°F | Sodium | Primary coolant boundary component |

*ASME BPVC Section II, Part A, “Ferrous Material Specifications” (Reference 7.1.3-5)

**ASME BPVC Section III, Division 5 “High Temperature Reactors,” 2017, as endorsed by RG 1.87 with limitations and restrictions (see Section 7.1.3.2)


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**Figure 7.1.3-1 Simplified PHT Flow Diagram**

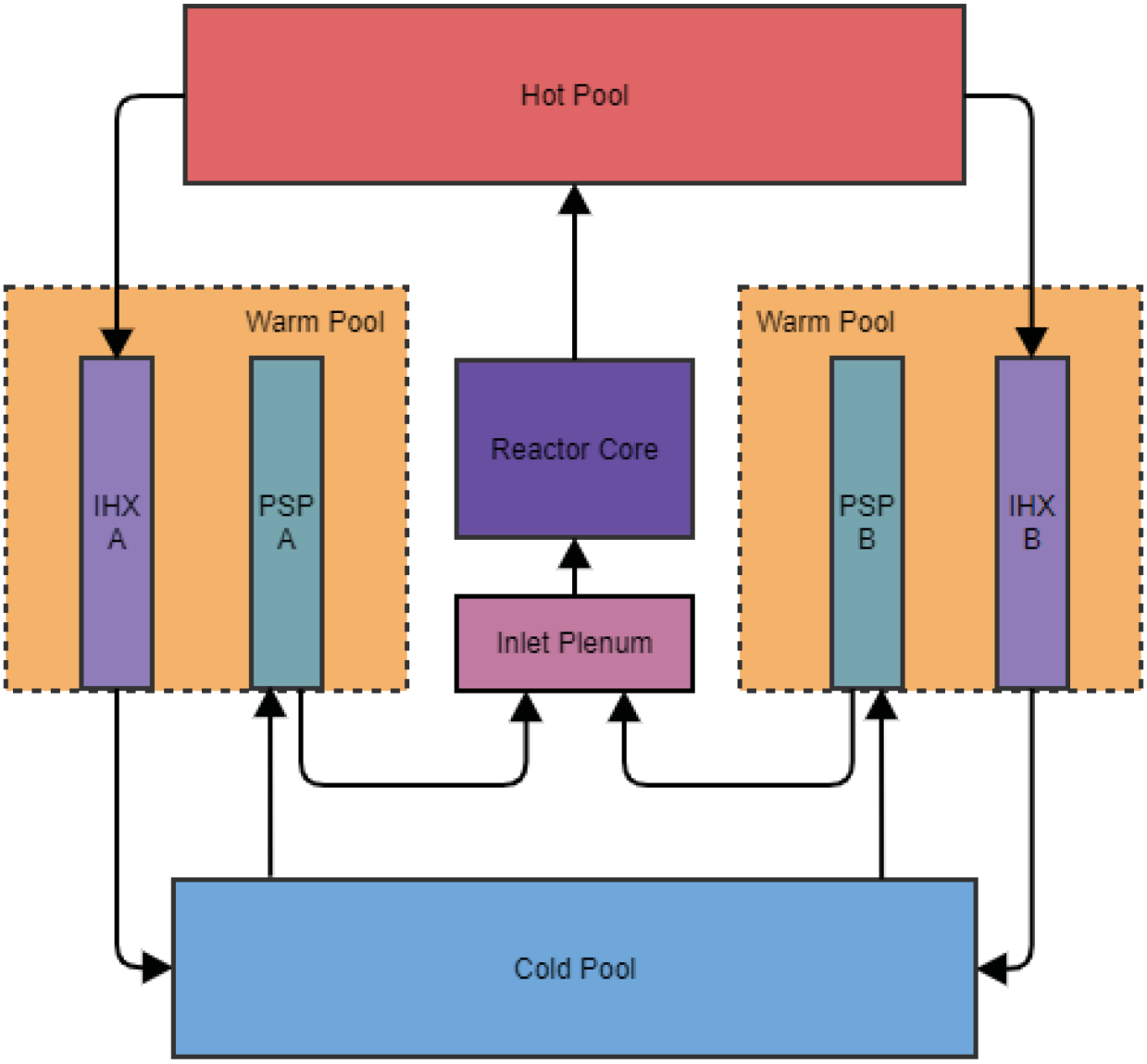



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**Figure 7.1.3-2 Simplified RAC and PHT Process Flow Diagram**

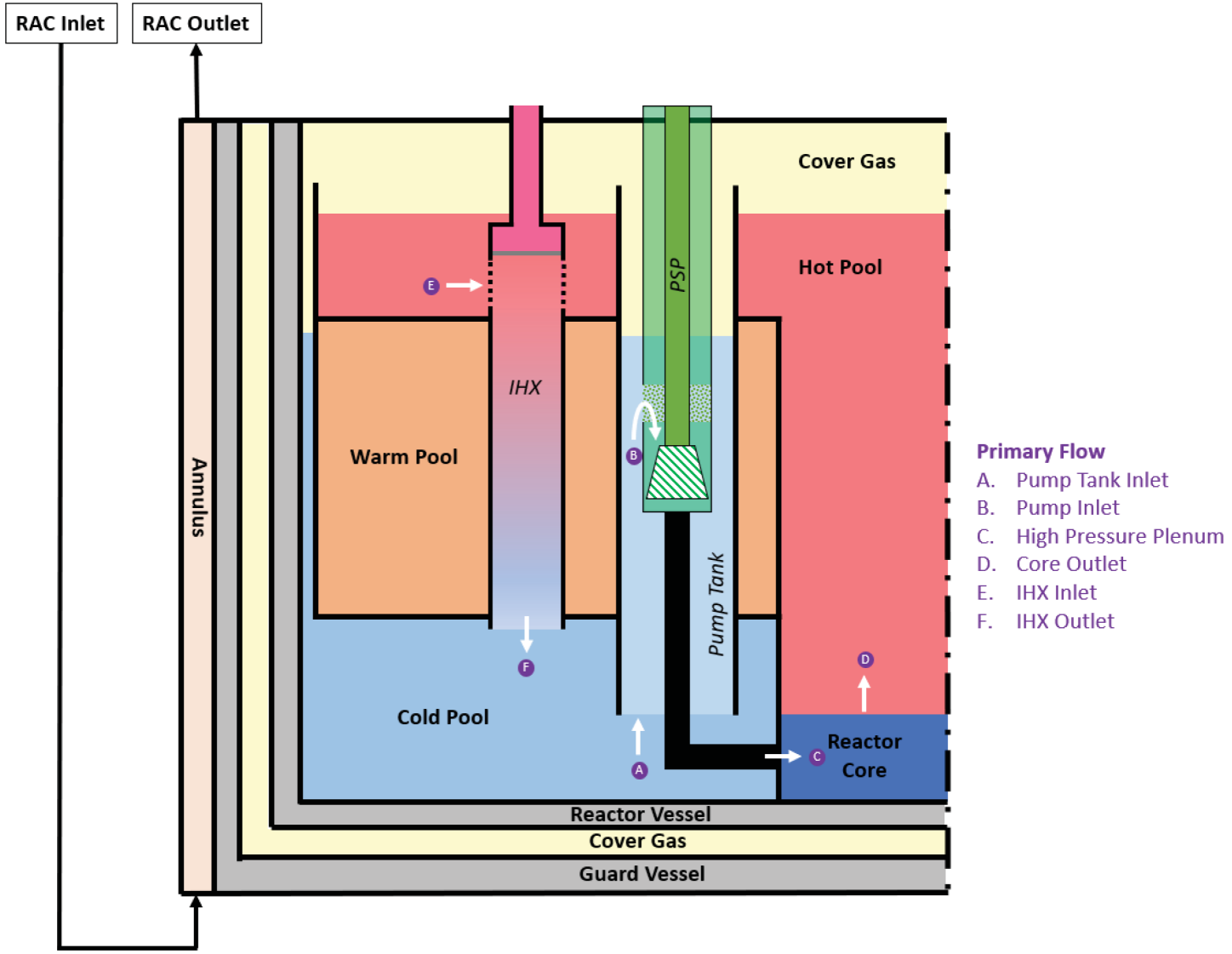



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**Figure 7.1.3-3 Sectional View of Installed Primary Sodium Pump**

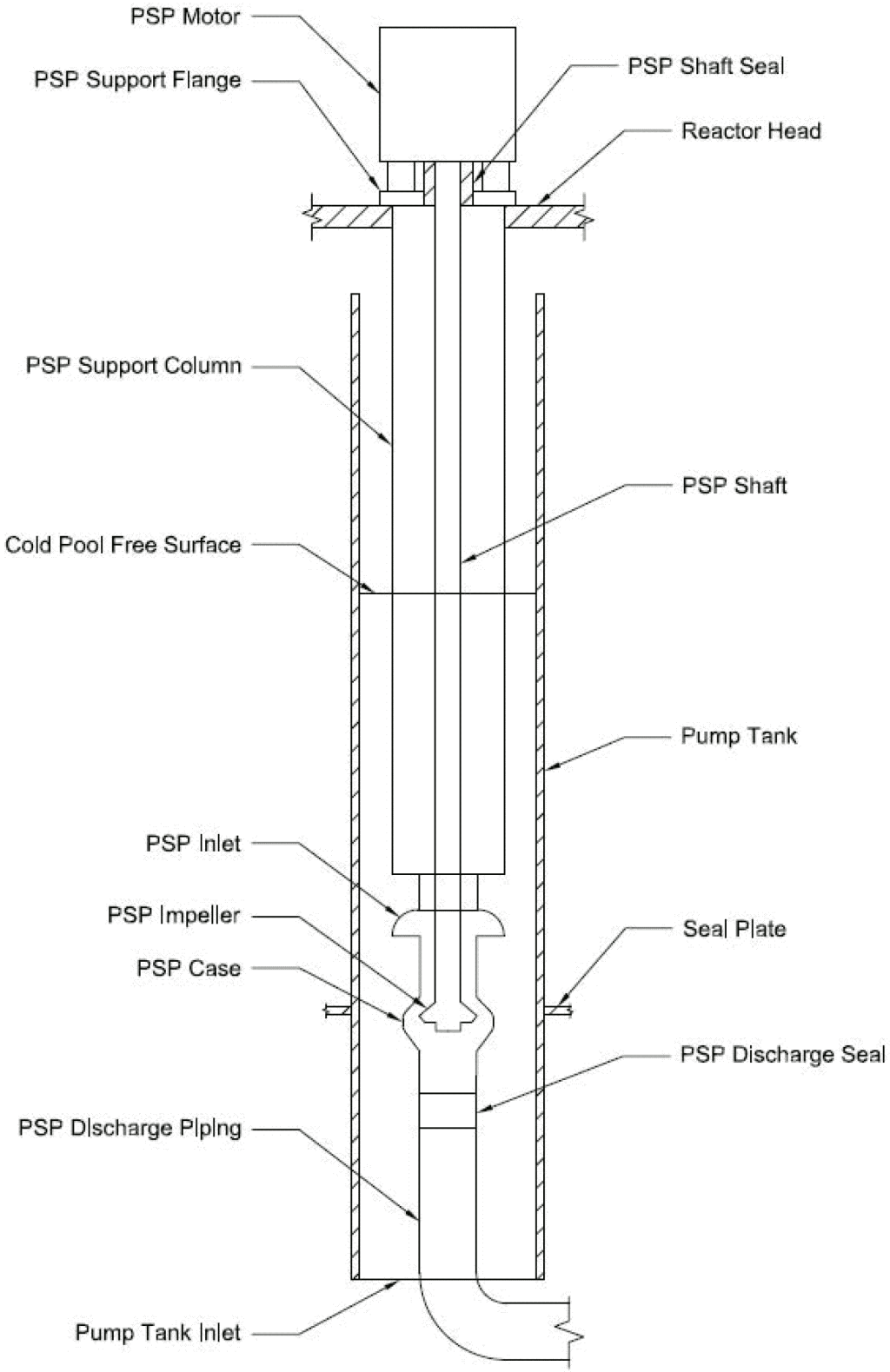




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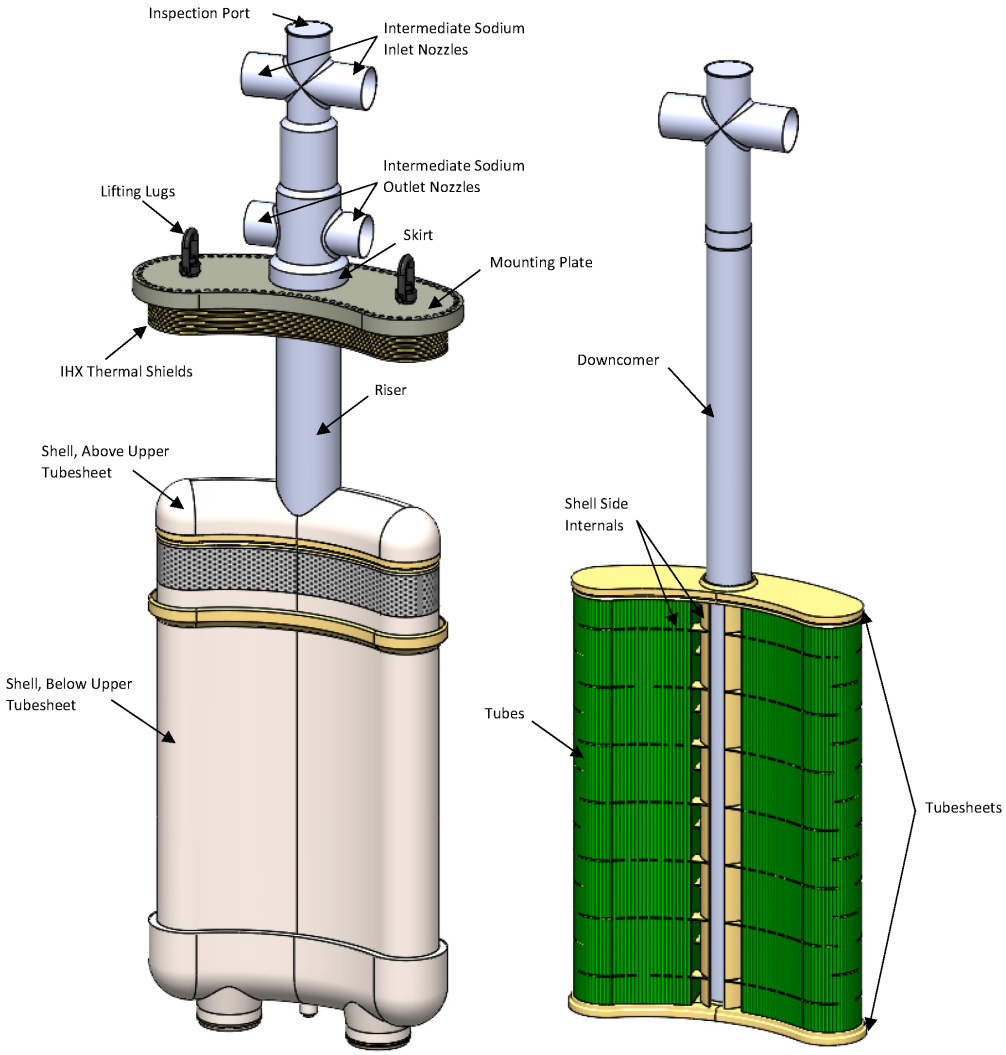
**Figure 7.1.3-4 Intermediate Heat Exchangers**





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## 7.1.4 Intermediate Heat Transport System

### 7.1.4.1 Summary Description

The Intermediate Heat Transport System (IHT) transports heat from the primary coolant to the Nuclear Island Salt System (NSS) for electrical power generation and energy storage during normal reactor power operation, and to the Intermediate Air Cooling System (IAC) for rejection to the atmosphere during reactor start-up and shutdown operation as the normal and preferred means of providing reactor cooling. In the event of a loss of pumped flow, the IHT is designed to provide natural circulation of intermediate coolant for reactor decay heat removal through the IAC Sodium-Air Heat Exchangers (AHXs).

The IHT includes equipment essential to the heat transport function within the two intermediate sodium loops. Each loop includes an Intermediate Sodium Pump (ISP), an array of Sodium-Salt Heat Exchangers (SHXs), and interconnecting piping. Each intermediate loop connects to one Primary Heat Transport System (PHT) Intermediate Heat Exchanger (IHX) (described in Section 7.1.3) and one IAC AHX (described in Section 7.2.2). The IHT supporting equipment includes expansion tanks and a drain subsystem. The system loop piping and components are located within the Reactor Auxiliary Building (described in Section 7.8.3), with the exception of piping in the Reactor Building (described in Section 7.8.1) that connects to the IHX nozzles and the piping in the yard area that connects to the IAC AHXs. Figure 7.1.4-1 provides the preliminary simplified diagram of a single intermediate loop.

The ISPs are motor-driven pumps. Intermediate loop sodium flow rate is controlled by controlling ISP speed, which is accomplished with an adjustable speed drive unit that varies frequency and voltage of the input power to the pump motors. The ISPs are powered by the Nuclear Island AC Electrical Power - Medium Voltage System (described in Section 1.1.4.3.6). In the event that normal power is unavailable, the ISPs are powered for low flow operation by the standby diesel generators. The ISP power supply includes two circuit breakers in series for pump emergency shut-off.

Thermal expansion of intermediate coolant is accommodated by an expansion tank located at the highest point in each intermediate loop. The IHT piping is sloped to allow for gravity drainage to drain tanks connected to piping low points. The IHT piping and components are provided with supplemental heating to prevent sodium from freezing.

Intermediate sodium is maintained at a higher pressure than primary sodium coolant, such that sodium flows into the PHT in the unlikely event of IHX tube leakage. The intermediate sodium is chemically compatible with the primary sodium coolant.

The IHT piping includes features to detect and contain potential sodium leakage to limit sodium interactions with the surroundings. Leakage is directed away from safety-related (SR) structures, systems, and components in the Head Access Area and into catch pans by the leakage containment equipment. The IHT piping includes features to limit heat loss to the surroundings and protect against hazards associated with high surface temperatures.


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### 7.1.4.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the IHT design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the IHT is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, safety-significant IHT components are designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels are described in Section 6.1.1. Safety-significant IHT components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions. The SR IHT components required to support emergency shutdown of the ISPs are located below-grade within, and protected by, the reinforced concrete Nuclear Island Control Building substructure (described in Section 7.8.4). The Nuclear Island Control Building provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions. Safety-significant IHT components are designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1.

Consistent with PDC 3, *Fire protection*, the design of IHT piping includes containment features for potential sodium leakage, which allows timely detection of leakage in order to minimize possibility of fire due to sodium interactions with the surroundings. Design of the IHT provides for the selection of non-combustible and non-fire sustaining materials to the extent practical. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the IHT is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences (AOOs), and postulated accidents. Material selection for sodium-containing components provides assurance of compatibility with the sodium environment, including sodium aerosols, vapors, and oxidation products, throughout the design lifetime. Equipment qualification for environmental service conditions is provided by the IHT component quality, design, and qualification requirements in accordance with the Equipment Qualification Program (described in Chapter 8).

SR IHT components required to support emergency shutdown of the ISPs are located below-grade within the reinforced concrete Nuclear Island Control Building substructure for protection from the effects from missiles and discharging fluids. The IHT piping located within the Reactor Building substructure is protected from external hazards by the structure.

Consistent with PDC 10, *Reactor design*, the design of the IHT supports the reactor core heat removal function that maintains sufficient margin to Specified Acceptable System Radionuclide Release Design Limits during normal operation and shutdown conditions. Safety-significant features support ISP trip to limit heat input to the primary coolant during AOOs.


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Consistent with PDC 34, *Residual heat removal*, the IHT design provides residual heat removal during normal operations and AOOs. Reactor core decay heat and other residual heat is transferred to the intermediate sodium through the IHX and to the atmosphere through the IAC AHX, as described in Section 7.2.2, at a sufficient rate that Specified Acceptable System Radionuclide Release Design Limits and the primary coolant boundary design conditions are not exceeded. Two intermediate loops are provided such that a single failure in one loop does not prevent the residual heat removal function.

Consistent with PDC 35, *Emergency core cooling*, the IHT supports emergency core cooling by providing ISP emergency shutdown to limit heat input to the primary coolant system such that the Reactor Air Cooling System (RAC) heat removal capacity is not exceeded for postulated accidents. The ISP emergency shutdown design includes two series trip circuit breakers for protection against single failure of a breaker to trip. The design includes redundant shutdown signals from the protection system.

Consistent with PDC 70, *Intermediate coolant system*, the design of the IHT provides sufficient margin such that the design conditions of the intermediate coolant boundary are not exceeded during conditions of normal operation, including AOOs. The intermediate coolant boundary operates at low pressure and is designed and constructed to satisfy applicable stress limits under anticipated operating conditions with sufficient margin to design conditions. Over-pressure protection of the intermediate coolant boundary is provided by the Sodium Cover Gas System, as described in Section 7.2.3, such that the integrity of the primary coolant boundary at the PHT IHX is maintained.

Consistent with PDC 72, *Sodium heating systems*, the IHT maintains primary sodium above the freezing temperature of liquid sodium with adequate margin. When decay heat is unavailable or insufficient to maintain adequate sodium temperatures, the IHT SSCs can add heat to the primary sodium via kinetic energy of the ISPs.

Consistent with PDC 73, *Sodium leakage detection and reaction prevention and mitigation*, the design of the IHT provides means to detect and identify sodium leakage, to limit and control the extent of sodium-air and sodium-concrete reactions, and to mitigate the effects of fires resulting from sodium-air reactions. Leakage containment and detection features are provided to contain and identify potential sodium leakage from IHT piping.

Consistent with PDC 74, *Sodium/water reaction prevention/mitigation*, the IHT is designed and located to avoid contact between sodium and water. Water-containing systems are not located in areas of the Reactor Building and Reactor Auxiliary Building where potential sodium leakage from IHT components could result in sodium-water reactions. The IHT piping located in the yard area is protected to prevent potential sodium leakage exposure to water from atmospheric conditions including precipitation.

Consistent with PDC 75, *Quality of the intermediate coolant boundary*, the intermediate coolant boundary is designed, fabricated, erected, and tested to quality standards commensurate with its safety-significant function in accordance with the applicable design and construction code requirements.


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Consistent with PDC 76, *Fracture prevention of the intermediate coolant boundary*, the intermediate coolant boundary is designed with sufficient margin to provide assurance that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. Material selection, design, and fabrication of intermediate coolant boundary components considers the effects of elevated operating temperatures, fatigue, loss of ductility, and chemical compatibility to provide assurance that the material behaves in a non-brittle manner and provides resistance to rapidly propagating fracture. The intermediate coolant boundary operates at low pressure and is designed and constructed to satisfy applicable stress limits under anticipated operating conditions with sufficient margin to design conditions. Over-pressure protection of the intermediate coolant boundary is provided by the Sodium Cover Gas System as described in Section 7.2.3.

Consistent with PDC 77, *Inspection of the intermediate coolant boundary*, the intermediate coolant boundary is designed to permit periodic inspection and functional testing of important areas and features to assess structural and leaktight integrity. The intermediate coolant boundary is within the scope of pre-service and in-service monitoring and non-destructive examination performed under the Reliability and Integrity Management Program described in Chapter 8.

Consistent with PDC 78, *Primary Coolant System Interfaces*, the intermediate sodium is chemically compatible with the primary coolant. The intermediate loop interface with the primary coolant system consists of a single passive barrier within the IHX. The intermediate loop is maintained at a higher pressure than the primary coolant during normal operation, AOOs, shutdown, and accident conditions such that postulated leakage flows into the primary system to prevent loss of core cooling capability. The Reactor Vessel cover gas space accommodates increased primary coolant inventory due to IHX leakage and the Reactor Protection System initiates reactor scram and ISP trip on a high primary coolant level condition such that primary coolant system safety-significant functions are maintained and Specified Acceptable System Radionuclide Release Design Limits are not exceeded.

Functional Requirements and Design Criteria

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The IHT supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classification of IHT components relied upon to perform the function:



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| Function ID | Function Description | IHT Functional Support |
|---|---|---|
| DL2-HR2 | ISP Trip on Low IHT level (NSRST) | The ISP is automatically tripped to limit loss of intermediate sodium due to leakage. |
| DL4-HR7 | Manual ISP Trip (NSRST) | Operator manual trip of ISP as backup to SR automatic trip to limit heat addition to primary coolant. |
| DL3-HR3 | ISP Trip on High High Primary Sodium Temperature (SR) | The ISP is automatically tripped to limit heat addition to the primary coolant. |
| DL3-HR12 | ISP Trip on High High Primary Sodium Level (SR) | The ISP is automatically tripped to limit intermediate sodium flow to the primary system through IHX tube leakage. |
| DL4-HR1 | IAC Passive Mode Operation (NSRST) | The IHT provides natural circulation of intermediate sodium coolant to transport heat from the PHT IHX to the IAC AHX for licensing basis events when the ISP and RAC are unavailable. |
| DL4-HR3 | ISP Trip Automatic Backup (NSRST) | The ISP automatic speed reduction to minimum or trip of the ISP as backup to the SR automatic trip to limit heat addition to primary coolant. |
| DL4-DID1 | Intermediate Leak Guard Piping (NSRST) | The IHT provides sodium leakage detection and containment for IHT piping located within the Reactor Building. |

Safety classifications for IHT components are listed in Table 7.1.4-1. The Safety-Related Design Criteria corresponding to the SR functions applicable to the IHT are derived as described in Section 6.1.2 and identified in Section 5.2. The Safety-Related Design Criteria are incorporated into the design of the IHT as described in Section 7.1.4.3.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the IHT, along with a description of the use and conformance to the guidance, includes:

RG 1.22, Revision 0, *Periodic Testing of Protection System Actuation Functions (Safety Guide 22)*

Full Conformance. The IHT ISP circuit breakers conform to references of “actuated device” and “actuated equipment.”

RG 1.36, Revision 1, *Nonmetallic Thermal Insulation for Austenitic Stainless Steel*

Full Conformance. Methods and procedures are used for selection of non-metallic insulation to prevent stress-corrosion cracking of stainless steel by minimizing leachable chlorides and fluorides to the lowest practicable levels.


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Seismic Classification and Design Codes

Seismic classifications for IHT components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The seismic classification of IHT components is listed in Table 7.1.4-1. Seismic design, analysis, and qualification is performed as described in Section 6.4. The IHT components are qualified to withstand seismic loads associated with the safe shutdown earthquake (described in Section 2.6.2) without loss of the capability to perform safety-significant functions.

Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of IHT components as special treatments, as described in Section 6.3. The safety-significant IHT piping and components are designed in conformance with the industry consensus codes and standards indicated in Table 7.1.4-2.

### 7.1.4.3 System Design and Performance Evaluation

The IHT supports electrical power generation and energy storage by transporting heat from the PHT IHXs to the NSS via the SHXs under normal operating conditions. The IHT provides low power and shutdown reactor core heat removal by circulating intermediate sodium coolant through the IAC AHXs, where heat is rejected to the atmosphere. The IHT provides for emergency shutdown of the ISPs to limit heat input to the PHT during limiting LBEs requiring the RAC operation, described in Section 7.2.1, and to limit intermediate sodium flow into the PHT in the unlikely event of IHX tube leakage.

The IHT supports heat rejection via the IAC by circulating intermediate sodium through the inline AHX for reactor heat removal in low power operating modes and reactor shutdown conditions. In IAC Active Mode (described in Section 7.2.2), intermediate sodium is circulated by the ISPs. When the ISPs are not available to provide forced circulation, the IHT is designed to support IAC heat rejection in Blower Mode and Passive Mode through natural circulation of intermediate sodium. The elevation difference between the installation location of the IHX, where sodium is heated reducing its density, and the elevated AHX, where sodium is cooled increasing its density, provides the potential energy to drive natural circulation flow. The elevation difference between the IHX and the AHX is designed to provide the necessary natural circulation flow rate to meet specified heat removal requirements based on the limiting case without reliance on RAC heat rejection with flow in two intermediate loops. No manual operator actions or automatic actuations are required for the IHT to perform its passive heat removal safety-significant function.

The IHT is configured to maintain intermediate sodium at a higher pressure than primary sodium whenever the intermediate loop contains liquid sodium. This design feature, combined with limitations on the primary and intermediate cover gas pressures, provides assurance that sodium flows into the primary system in the event of IHX tube leakage and prevents loss of primary coolant level within the Reactor Vessel that could affect core cooling or result in the release of primary coolant radionuclides to the IHT. Siphoning of intermediate coolant into the primary system that could result in Reactor Vessel overfill is prevented by cover gas flow into the intermediate loop from the expansion tank when a low sodium level condition in the expansion tank is reached.


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During normal operation, IHT pressure is maintained at a lower pressure than NSS so that, in the event of an SHX leak, NSS coolant flows from the NSS to the intermediate loop. The effects of the sodium-salt reaction due to a postulated leak in the SHX is the subject of research and development as described in Section 13.2.3.

The expansion tank interfaces with the Sodium Cover Gas System to provide an inert cover gas within the void space, as described in Section 7.2.3, to prevent sodium-air interaction and to provide IHT over-pressure protection. The expansion tank interfaces with the Sodium Processing System, described in Section 7.2.4, which provides intermediate sodium processing to limit contaminants and maintain sodium chemistry within limits. The IHT maintains a continuous flow of intermediate sodium through the expansion tank to circulate processed sodium through each intermediate loop and limit thermal stratification within the tank.

Emergency shutdown features are provided for the ISPs. Two trip circuit breakers are installed in series, along with associated trip mechanisms, in the power supply circuit to each ISP motor. The trip circuit equipment is classified SR and is installed in the SR Nuclear Island Control Building substructure.

Emergency shutdown of the ISPs is initiated in response to the following abnormal conditions sensed within the IHT or PHT:

- Low level in an IHT expansion tank (trips associated ISP)
- High-High primary sodium level (trips both ISPs)
- High-High primary sodium temperature (trips both ISPs)

An automatic emergency shutdown signal actuates the SR trip mechanism and opens the trip circuit breakers. Backup to the emergency shutdown signal is provided by local manual breaker trip and by reducing pump speed to the minimum flow setpoint.

The ISP trip initiated by abnormal level conditions in the IHT expansion tank or in the PHT sodium pool provides protection against Reactor Vessel overfill in the unlikely event of leakage within the IHX.

The ISP trip initiated by abnormal temperature condition in the PHT cold pool reduces the heat load to the PHT caused by ISP operation in order to maintain heat removal requirements for the RAC within limits.

IHT components are exposed to an internal environment of high temperature and liquid sodium. Externally, IHT components are exposed to the Reactor Auxiliary Building and Reactor Building internal environment, except for the portion of piping routed to the IAC outside of the Reactor Auxiliary Building. Material selections for IHT components are based on compatibility with the applicable design internal and external environments, as well as design loadings associated with plant operational events in accordance with applicable design code requirements.

The IHT piping is designed to meet ASME B31.1 (Reference 7.1.4-1) requirements. IHT expansion tanks, drain tanks, SHXs, and ISP pump tanks are designed, fabricated, and tested in accordance with ASME BPVC Section VIII (Reference 7.1.4-2).


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IHT piping is welded to interfacing components, including the IHXs, SHXs, AHXs, and drain valves to minimize the potential for sodium leakage at equipment connections. The IHT piping within the Reactor Building HAA includes a continuous leak guard enclosing the pipe as a secondary barrier for leak protection. The leak guard contains and redirects sodium leaks toward dedicated drains and containment features to prevent interaction with SR equipment in the HAA.

#### 7.1.4.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the IHT is described in Section 7.1.4.2. The following programmatic special treatments are preliminarily identified as applicable to the IHT to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- Equipment Qualification Program
- Post-Construction Inspection, Testing, and Analysis Program (described in Chapter 12)

#### 7.1.4.5 Instrumentation Requirements

The IHT provides instrumentation required for expansion tank level signals and ISP trip circuit breaker status.

**References**

7.1.4-1 American Society of Mechanical Engineers, “Power Piping, B31.1,” ASME, 2022

7.1.4-2 American Society of Mechanical Engineers “Boiler and Pressure Vessel Code, Section VIII: Rules for Construction of Pressure Vessels,” ASME, 2021

7.1.4-3 American Society of Mechanical Engineers, “Stainless Steel Pipe, B36.19M,” ASME, 2004

7.1.4-4 IEEE, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations,” IEEE 308-2001

7.1.4-5 IEEE, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” IEEE 323-2016

7.1.4-6 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-1991



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**Table 7.1.4-1 IHT Safety and Seismic Classifications**

| Component | Safety Classification | Seismic Classification |
|---|---|---|
| Piping, Boundary Valves, Expansion Tanks | NSRST | SCN3 |
| ISP Tank | NSRST | SCN3 |
| SHX | NSRST | SCN3 |
| Piping (HAA), Leakage Enclosure | NSRST | SCN1 |
| ISP Trip Circuit Breaker, Trip Actuator | SR | SCS1 |


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**Table 7.1.4-2 Applicable Codes and Standards**

| Standard | Edition | Title |
|---|---|---|
| ASME Boiler and Pressure Vessel Code, Section VIII | 2021 | Rules for Construction of Pressure Vessels |
| ASME B31.1 | 2022 | Power Piping |
| ASME B36.19M (Reference 7.1.4-3) | 2004 | Stainless Steel Pipe |
| IEEE 308 (Reference 7.1.4-4) | 2001 | IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations |
| IEEE 323 (Reference 7.1.4-5) | 2016 | IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations |
| IEEE 603 (Reference 7.1.4-6) | 1991 | IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations |


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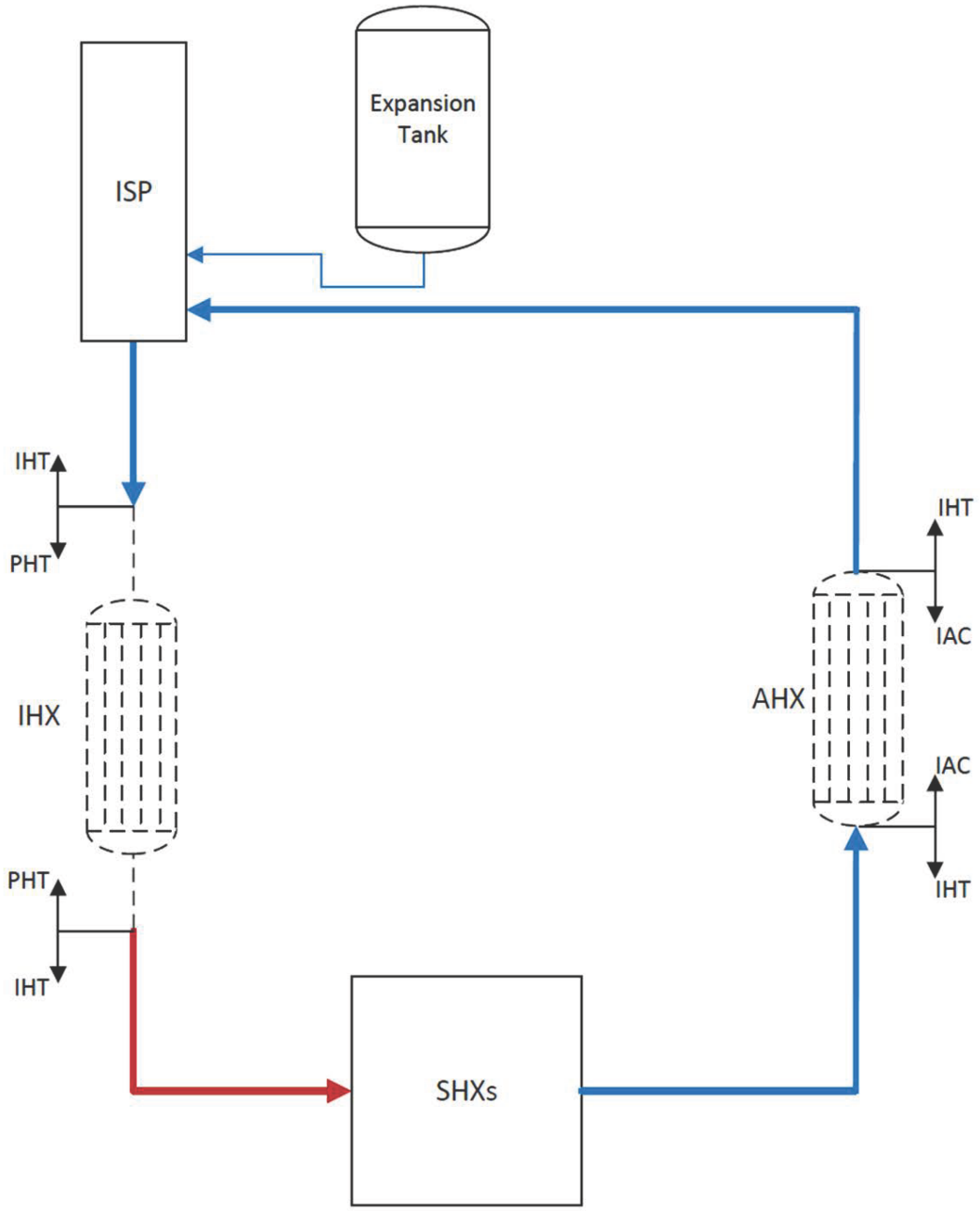
**Figure 7.1.4-1 Intermediate Loop Simplified Diagram**



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## 7.2 Reactor Auxiliary Systems

Reactor Auxiliary Systems consist of the Reactor Air Cooling System (RAC), Intermediate Air Cooling System (IAC), Sodium Cover Gas System (SCG), Sodium Processing System (SPS), and Control Rod Drive System (CRD) which are described in Section 7.2.1 through Section 7.2.5.

### 7.2.1 Reactor Air Cooling System

#### 7.2.1.1 Summary Description

The Reactor Air Cooling System (RAC) supports passive reactor decay heat removal by natural convection air flow and heat rejection to the atmosphere. The RAC, in conjunction with the Primary Heat Transport System (PHT) (Section 7.1.3) and the Reactor Enclosure System (RES) (Section 7.1.2), provides the long-term emergency core cooling function. The RAC is continuously in operation and requires no automatic or manual actuations or operator action to perform the heat removal safety function.

The RAC consists of four inlet and four outlet stacks, a Collector Cylinder Assembly (CCA), and associated inlet and outlet ducting. The RAC is integrated into the Reactor Building (RXB) substructure design as described in Section 7.8.1. The reinforced concrete RXB substructure houses the CCA and the inlet and outlet ducting and includes RAC enclosure structures that house the lower portion of the inlet and outlet stacks. Figure 7.2.1-1 provides a simplified representation of the RAC configuration for a single inlet and outlet flow path.

The steel RAC inlet and outlet stacks are each straight, single flue design. The RXB RAC enclosures provide the foundation and structural support for the stacks. The stacks are free-standing above the RXB substructure and are located adjacent to the north and south sides of the RXB superstructure, with two inlet and two outlet stacks on each side. The stacks are not supported by the RXB superstructure. The outlet stacks terminate above the roofline of the RXB superstructure. The inlet stacks terminate a minimum of 30 feet above grade. The stack outlets are at a higher elevation than the stack inlets to provide separation of the air flow streams to limit the potential for recirculation. The inlet stacks inlet openings are in a horizontal orientation and protected by louver assemblies configured to prevent wildlife ingress and to minimize precipitation and debris entry. The design of the inlet louvers includes provisions for temporary installation of fine screens in the event of severe small debris or dust conditions. Stack outlets are vertically oriented and protected by a chimney cowl configured to mitigate wind effects and prevent rain, large wind-blown debris, and animal ingress. Drains at the base of the inlet stacks allow drainage of water resulting from precipitation entry. Instrumentation is provided to monitor inlet temperature, outlet temperature, and outlet stack mass flowrate.

The CCA consists of an inner cylindrical, steel collector cylinder and an outer cylindrical, steel inlet flow divider plate with an integral supporting structure and insulation between the cylinders. The CCA surrounds the RES Guard Vessel (GV) and is structurally supported at the top of the assembly by the Reactor Support Structure described in Section 7.1.2. The CCA provides separation of inlet and outlet air flow paths. Inlet ducting provides an air flow path from each inlet stack to a common inlet plenum within the Head Access Area (HAA). An inlet air downcomer annulus, formed by the inlet flow divider plate and the RXB reactor cavity, extends downward


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from the inlet plenum to near the bottom of the reactor cavity. A riser annulus, formed by the collector cylinder and the GV, extends to ducting within the RES Modular Isolated Reactor Support Structure (MIRSS) described in Section 7.1.2. The MIRSS ducting provides a flow path to the RAC outlet ducting, which provides a flow path to the base of the outlet stack. The CCA is insulated within the void space between the collector cylinder and the inlet flow divider plate to limit heating of the reactor cavity concrete walls. The RAC inlet ducting is insulated to limit heating of the incoming air form the surrounding ambient air. The RAC outlet ducting is insulated to limit heating of the internal HAA and RAC enclosure environment and the RXB concrete.

Natural convection air flow within the RAC is induced by the buoyancy forces resulting from heating the air within the riser annulus. As the air temperature increases, its density is reduced, and the heated air naturally rises in the annulus. The rising air draws ambient air in through the inlet stack openings and ducting, into the inlet plenum, and downward through the downcomer annulus. The cool air is drawn around the bottom of the CCA and up into the riser annulus to be heated. Heated air flows to the outlet stacks via common header ducts within the MIRSS and outlet ducting. From the outlet ducting, the heated air enters and rises within the outlet stacks to exit the outlet opening, thereby rejecting the heat to the atmosphere. Figure 7.2.1-1 illustrates the air flow path within the RAC.

Reactor core heat is transferred to the primary coolant, and from the coolant to the GV, as described in Sections 7.1.2 and 7.1.3, resulting in an elevated temperature of the GV outer surface. The collector cylinder is heated by thermal radiation heat transfer from the GV outer surface. Heat is transferred to the air flowing within the riser annulus by convection heat transfer from the elevated temperature GV outer surface and collector cylinder inner surface.

#### 7.2.1.2 Design Basis

Compliance with regulatory criteria, identification of design criteria, and codes and standards that define the RAC design basis are described below.

Regulatory Criteria

Principal design criteria are identified in Section 5.3. Compliance with PDC that are applicable to the RAC is described below:

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the RAC is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels are described in Section 6.1.1. With the exception of the upper portions of the RAC stacks, the RAC is located in and protected by the reinforced concrete RXB substructure. The location within the RXB provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions as described in Section 7.8.1. The upper portion of the RAC stacks are designed to withstand the effects of tornadoes and extreme climate without loss of capability to perform safety-significant functions. The RAC inlet louver assemblies and outlet chimney cowls are configured to reduce precipitation entering the system. The RAC is designed to withstand the effects of earthquakes in accordance with the methods described in


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Section 6.4.1. RAC components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform their safety-significant functions.

Consistent with PDC 3, *Fire protection*, the RAC is designed and located to minimize the probability and effects of fire and explosions. The RAC design uses non-combustible and non-fire sustaining materials to the extent practical. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the RAC is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences (AOOs), and postulated accidents, including climatic conditions of minimum and maximum air temperature and relative humidity. The RAC is protected from dynamic effects associated with equipment failures including missiles, pipe ruptures, and explosions.

Consistent with PDC 13, *Instrumentation and control*, the RAC inlet temperature, outlet temperature, and outlet flow rate are monitored to provide assessment of system performance across anticipated ranges including during normal operations, AOOs, and accident conditions.

Consistent with PDC 35, *Emergency core cooling*, the RAC is designed in conjunction with the natural circulation of primary coolant within the reactor vessel (RV) and heat transfer from the RV to the GV to transfer heat from the reactor core during and following postulated accidents to provide assurance that sufficient core cooling is maintained.

The RAC is reliable, passive, continuously in operation, and has no isolation. The RAC is configured with four inlets and four outlets with two of each on opposite ends of the RXB to provide separation in the event of external hazards. The RAC air flow path is interconnected through common air regions, including the region surrounding the GV, to compensate for potential blockages and maintain air flow across the GV surface.

Consistent with PDC 36, *Inspection of emergency core cooling system*, monitoring of the RAC parameters provides assurance during normal operation that the system is functioning properly. The RAC design permits periodic inspection of important components to ensure the integrity and capability of the system. Access is provided for inspection of ducts and stacks, and the riser annulus between the GV and CCA. Inspections of the RAC flow passages are performed under the Reliability and Integrity Management (RIM) Program to ensure the integrity and capability of the system.

Consistent with PDC 37, *Testing of emergency core cooling system*, the RAC design permits assessment of the structural integrity of RAC components and the operability and performance of the RAC through implementation of the RIM Program, including the associated MANDE and DMA. The RAC provides passive heat removal continuously and requires no operational sequence to bring the system into operation. Flow is driven by natural convection and the RAC heat removal rate increases with an increase in reactor temperature. Instrumentation is provided to measure inlet and outlet air temperatures and outlet air flow rate to provide status of the heat


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removal in all operating modes and to assure RAC operability and adequate system performance. The monitored parameters will provide early indication of RAC performance degradation.

Consistent with PDC 64, *Monitoring Radioactivity Releases*, means are provided for monitoring radioactivity releases from the RAC. Activation of air within the RAC flow path results in N-16 and Ar-41 gaseous effluents from RAC. The RAC contribution to overall plant gaseous effluents is described in Section 9.1. RAC outlet flow is monitored to quantify radioactive effluent discharge.

Functional Requirements and Design Criteria

Probabilistic risk assessment safety functions (PSFs) are determined through the plant-specific probabilistic risk assessment (PRA) described in Section 3.1. The safety-significant PSFs are identified as SR functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The RAC supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classification of RAC components relied upon to perform the function:

| Function ID | Function Description | RAC Functional Support |
|---|---|---|
| DL3-HR4 | Inherent - RAC Operation (SR) | The RAC provides passive emergency core cooling capability and operates continuously. |
| DL5-PAM1 | Post-Accident Monitoring (NSRST) | RAC performance variables are monitored to provide data for Post-Accident Monitoring (PAM). |

Safety classifications for RAC structures and components are listed in Table 7.2.1-1. Safety related design criteria (SRDC) corresponding to the SR function applicable to the RAC are derived as described in Section 6.1.2 and identified in Section 5.2. The SRDC are incorporated into the design of the RAC as described in Section 7.2.1.3.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the RAC, along with a description of the use and conformance to the guidance, includes:

RG 1.76, Revision 1, *Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants*

Full conformance. The RAC withstands pressure drops due to wind, and the RAC above ground stacks are designed to withstand wind generated missiles.

RG 1.97, Revision 5, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*

Full conformance. The RAC contains Type B and D variables with redundant instrumentation adhering to IEEE 497-2016 (Reference 7.2.1-1).


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RG 1.100, Revision 4, *Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants*

Partial conformance. The RAC adheres to IEEE 344-2020 (Reference 7.2.1-2) for instrumentation and has no active mechanical equipment.

RG 1.243, Revision 0, *Safety-Related Steel Structures and Steel-Plate Composite Walls for other than Reactor Vessels and Containments*

Full Conformance. The RAC CCA adheres to increased load combinations.

Seismic Classification and Design Codes

Seismic classifications for the RAC components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The seismic classification of RAC components is SCS1. The seismic classification of instrumentation supporting the PAM function is SCN3. Seismic design, analysis, and qualification are performed as described in Section 6.4. RAC components are qualified to withstand seismic loads associated with the safe shutdown earthquake, described in Section 2.6.2, without loss of the capability to perform the safety-significant function.

Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of RAC components as special treatments, as described in Section 6.3. Codes and standards that are applicable to the RAC are listed in Table 7.2.1-2.

### 7.2.1.3 System Design and Performance Evaluation

The RAC is designed to meet design basis requirements described in Section 7.2.1.2 and evaluated to provide assurance that associated performance requirements are met. Design features are provided by the RAC design to support the identified safety-significant functions and to provide assurance of compliance with applicable regulatory criteria.

The RAC inlet and outlet stacks are designed to meet ASME STS-1 (Reference 7.2.1-3) requirements. The stacks, inlet louver housings, chimney cowls, and the CCA are designed in accordance with the structural requirements of AISC N690 (Reference 7.2.1-4). The inlet and outlet ducting and the stack inlet louvers and outlet cowls are designed in accordance with ASME AG-1 (Reference 7.2.1-5). The outlet ducting is designed to provide sufficient flexibility to accommodate thermal expansion and seismic movement.

Preliminary material selections and design temperatures for the RAC components are identified in Table 7.2.1-3. Materials of construction for the RAC components are selected to be compatible with the operating environment during normal and accident conditions. The RAC components design includes adequate corrosion allowance to provide assurance that the safety function will be maintained over the design life.


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The RAC is designed as a passive system with no active components in order to provide adequate reliability. The RAC includes redundant inlet and outlet stacks with common inlet and outlet plenums to compensate for potential blockage in a stack inlet or outlet opening. Continuous operation and monitoring of RAC operating parameters provides confirmation of performance capability during normal plant operation.

Fixed inlet louvers on the inlet stacks allow intake of air and reduce penetration of wind-driven rain and missiles. The large inlet louver openings allow for low airflow velocities and low pressure drop, reducing the entrainment of precipitation while minimizing airflow restriction. The inlet louver assembly is accessible for inspection and cleaning if required.

Chimney cowls are designed to provide protection against ingress of precipitation and debris, and to provide sufficient outlet airflow to meet safety function requirements. The cowl design limits the effects high winds on outlet air flowrate.

The RAC design includes sufficient capacity margin to tolerate penetration of dust and sand without significant performance degradation. Continuous RAC operation and operating parameter monitoring provides indication of RAC performance degradation. Access to internal surfaces provides for periodic inspection and debris removal, if necessary. The RAC design includes features for remote vacuuming of the internal air flow path during plant outage periods.

The RAC components are protected from design-basis hazards and the associated design-basis hazard levels described in Section 6.1.1. The RAC air flow path is designed to withstand the atmospheric pressure transient effects from tornadoes. The RAC inlets and outlets are elevated and protected from significant precipitation intrusion by inlet louver assemblies and outlet chimney cowls. The RAC stacks are designed to withstand loading conditions due to seismic events, high winds, accumulation of snow, tornado missiles, and tornado-induced pressure transients without loss of safety function capability. The minimum RAC inlet elevation of 30 feet protects the inlet louvers from the external large missile (automobile) hazard (Section 6.4.2.2.3). Separation of stacks on either side of the RXB superstructure provides enhanced missile protection.

The RAC inlet and outlet stacks, ducting, and other portions of the air flow path are designed to provide sufficient air flowrate to remove the design basis heat load under the limiting external environmental conditions specified in Section 2.1. The HAA Seal and the Well Seal are provided to minimize air leakage from the RAC air flow path as described in Section 7.1.2. The CCA is constructed as a sealed assembly to prevent bypass air flow between the inlet and outlet flow paths.

The RAC removes heat from the primary coolant, as described in Section 7.2.1.1, continuously when sufficient reactor heat is produced. The rate of heat transfer is dependent on the temperature of the GV wall and increases significantly with increasing GV temperature due to the relationship of the thermal radiation heat transfer rate with surface temperature. Higher GV surface temperature results in increased thermal radiation heat transfer to the collector cylinder and increased temperature of the collector cylinder surface, which increases convection heat transfer to the air within the RAC riser annulus. Thermal radiation heat transfer is enhanced by


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collector cylinder surface treatment to increase the emissivity of the surface. The surface treatment for heat transfer enhancement is the subject of research and development as described in Section 13.2.1.

Under abnormal conditions of natural circulation of primary coolant within the Reactor Vessel (RV), and without heat removal from the PHT to the Intermediate Heat Transport System (IHT) described in Section 7.1.4, the coolant temperature near the RV wall increases, which increases the temperature of the GV. The resulting higher temperature GV outer surface significantly increases thermal radiation heat transfer to the collector cylinder and increases the heat transfer rate to the air flowing within the riser, which further increases its temperature and flow rate, increasing overall heat rejection to the atmosphere. In this manner, RAC heat removal capacity increases or decreases based on the heat removal requirements from the primary coolant and the reactor core.

Under normal conditions where reactor heat removal is provided by heat transfer from the PHT to the IHT as described in Section 7.1.3, the GV surface temperature is lower, and the heat removal by the RAC is reduced.

#### 7.2.1.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the RAC is described in Section 7.2.1.2. The following programmatic special treatments are preliminarily identified as applicable to the RAC to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8. The Post-Construction Inspection, Testing, and Analysis Program is described in Chapter 12.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- Equipment Qualification Program
- Post-Construction Inspection, Testing, and Analysis Program

#### 7.2.1.5 Instrumentation Requirements

The RAC inlet temperature, outlet temperature, and outlet mass flow rate are monitored to indicate heat removal performance. The instruments are installed on the RAC vertical stacks within the RXB substructure. This location protects the instrumentation from external hazards and allows instrument access during outages. The measured parameters are IEEE 497-2016 variable Types B and D for PAM.

Each inlet stack contains a temperature instrument and each outlet stack contains temperature and mass flow instruments. Stacks contain redundant instrumentation for reliability. Signal output is routed directly to the Reactor Protection System, as described in Section 7.6.3, with redundant instruments on a separate division.


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## References

7.2.1-1 IEEE, “Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497, 2016.

7.2.1-2 IEEE, “Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations,” IEEE 344, 2020.

7.2.1-3 ASME, “Steel Stacks,” ASME STS-1, 2021.

7.2.1-4 ANSI/AISC, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” ANSI/AISC N690, 2018.

7.2.1-5 ASME, “Code on Nuclear Air and Gas Treatment,” ASME AG-1, 2019.



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**Table 7.2.1-1 RAC Safety Classifications**

| Component | Safety Classification |
|---|---|
| CCA | SR |
| Inlet stacks | SR |
| Outlet stacks | SR |
| Inlet ducts | SR |
| Outlet ducts | SR |
| Inlet temperature instrumentation | NSRST |
| Outlet temperature instrumentation | NSRST |
| Outlet mass flow instrumentation | NSRST |
| Inlet louver assemblies | SR |
| Outlet chimney cowls | SR |


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**Table 7.2.1-2 Applicable Codes and Standards**

| Standard | Edition | Title |
|---|---|---|
| ANSI/AISC N690 | 2018 | Specification for Safety-Related Steel Structures for Nuclear Facilities |
| ASME AG-1 | 2019 | Code on Nuclear Air and Gas Treatment |
| ASME STS-1 | 2021 | Steel Stacks |
| IEEE 497 | 2016 | IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations |
| IEEE 344 | 2020 | IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations |


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**Table 7.2.1-3 Preliminary Material Selection and Design Temperatures for RAC Components**

| SSC Component | Material | Design Temperature (°F) |
|---|---|---|
| CCA collector cylinder | Chromium-molybdenum alloy, ASTM A387 Grade 22 | 525 |
| CCA inlet flow divider plate, support structure | Carbon steel, ASTM A36 | 525 |
| Inlet and outlet stacks | Carbon steel, ASTM A36 | 150 (inlet stack)<br>525 (outlet stack) |
| Inlet and outlet ducts | Carbon steel, ASTM A36 | 150 (inlet duct)<br>525 (outlet duct) |
| Inlet louver assembly - louvers | Aluminium | 150 |
| Inlet louver assembly - structure | Stainless steel | 150 |
| Outlet chimney cowls | Stainless steel or steel | 525 |


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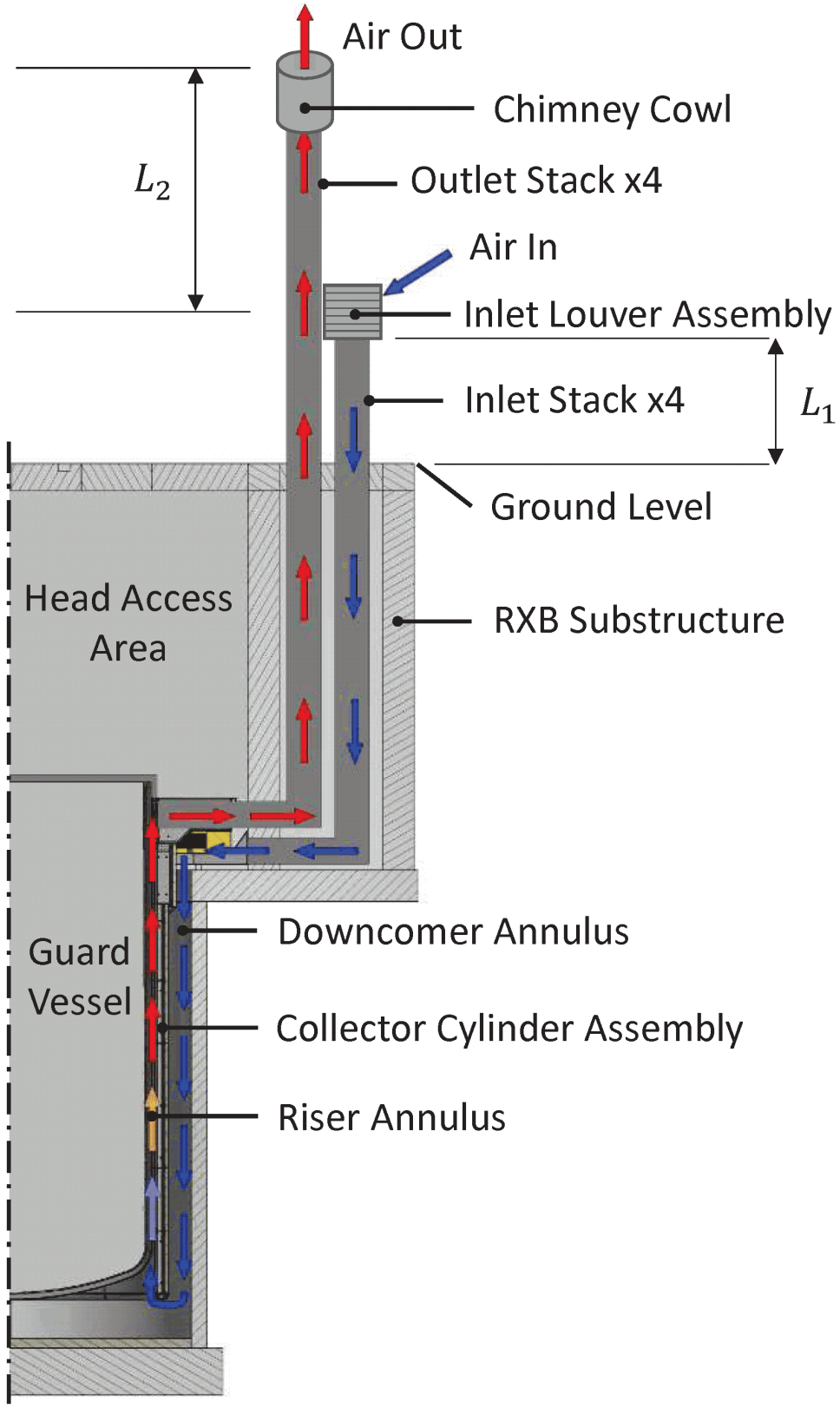
**Figure 7.2.1-1 Simplified RAC Airflow Diagram**



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## 7.2.2 Intermediate Air Cooling System

### 7.2.2.1 Summary Description

The Intermediate Air Cooling System (IAC) transfers heat from the intermediate sodium (described in Section 7.1.4) to the atmosphere. The IAC is the primary and preferred means of reactor heat removal during low power and shutdown conditions.

The IAC consists of two equal capacity trains that each contain a Sodium-Air Heat Exchanger (AHX), chimney structure (air stack), air blower, dampers, and air heater housed within a supporting Air Stack Structures and Equipment (ASE) structure. The two ASEs are located in the yard area on either side of the Reactor Auxiliary Building. An AHX is connected in series within each of the two intermediate loops. Intermediate sodium flows on the tube-side of the AHX with airflow on the shell-side. Each ASE includes a blower for forced airflow and remotely operated dampers to control airflow and heat transfer from the intermediate sodium to the atmosphere. The air dampers above and below the AHXs are normally closed during Mode-1 operation to limit heat transferred from the intermediate sodium when the IAC is not required. The dampers fail open on loss of power. The blower discharge dampers are normally open and fail open. Screens are provided to cover the air inlet and outlet to prevent large debris and other unwanted intrusion. A cap on the air stack outlet minimizes entry of precipitation and large debris.

The ASEs include provisions for sodium catch pans and leak detection below the AHXs. Further description of these Nuclear Island Fire Protection System (NFP) features is included in Section 7.5.2.

A preliminary simplified diagram of the IAC is shown in Figure 7.2.2-1.

### 7.2.2.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the IAC design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the IAC is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the IAC is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels are described in Section 6.1.1. IAC components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions consistent with their safety significance. Safety-significant IAC components are designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1.


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Consistent with PDC 3, *Fire protection*, the design of the IAC minimizes the probability and effect of fires through design features to minimize the potential for sodium leakage to the surroundings and the selection of non-combustible and non-fire sustaining materials to the extent practical. The location of the ASE is separate from other structures, which minimizes the potential for fires to impact safety-significant SSCs. There are no water-based suppression systems and no fire water piping in the ASE where the sodium-containing AHX is located. The ASE is designed to minimize fires in the event of sodium leakage from the AHX by diverting leakage to collection points and minimizing the surface area of leaked sodium exposed to air. Concrete surfaces that have the potential to interact with sodium are covered with an appropriate material to minimize sodium-concrete interactions. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the IAC is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences (AOOs), and postulated accidents. Material selection for sodium-containing components provides assurance of compatibility with the sodium environment, including sodium aerosols, vapors, and oxidation products, throughout the design lifetime. The IAC is protected from external hazards caused by transportation incidents and offsite industrial and chemical facilities.

Consistent with PDC 10, *Reactor design*, the design of the IAC supports appropriate margin to assure Specified Acceptable Radionuclide Release Limits (SARRDLs) are not exceeded during normal operation, including AOOs. The IAC supports the reactor core decay heat removal function that maintains sufficient margin to SARRDLs during low power and shutdown conditions.

Consistent with PDC 34, *Residual heat removal*, the IAC design provides residual heat removal during normal operations and AOOs. Reactor core decay heat and other residual heat is transferred to the intermediate system through the Intermediate Heat Exchanger (IHX), as described in Section 7.1.4, and to the atmosphere through the IAC AHX, at a sufficient rate that SARRDLs and the primary coolant boundary design conditions are not exceeded. Two redundant intermediate loops are provided such that a single failure that disables one loop does not prevent the residual heat removal function when IAC is operated in Active Mode or Blower Mode described in Section 7.2.2.3.

Consistent with PDC 70, *Intermediate coolant system*, the design of the IAC provides sufficient margin such that the design conditions of the intermediate coolant boundary are not exceeded during conditions of normal operation, including AOOs. The intermediate coolant boundary within the IAC AHX operates at low pressure and is designed and constructed to satisfy applicable stress limits under anticipated operating conditions with sufficient margin to design conditions. Over-pressure protection of the intermediate coolant boundary is provided by the Sodium Cover Gas (SCG) as described in Section 7.2.3.

Consistent with PDC 72, *Sodium heating systems*, the design of the IAC provides heaters within the ASE to prevent sodium freezing in the AHX. Heating systems and their controls are appropriately designed to ensure that the temperature distribution and rate of change of temperature in the intermediate sodium loop are maintained within design limits.


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Consistent with PDC 73, *Sodium leakage detection and reaction prevention and mitigation*, the design of the IAC provides means to detect and identify sodium leakage, to limit and control the extent of sodium-air and sodium-concrete reactions, and to mitigate the effects of fires resulting from sodium-air reactions. Catch pans and leak detection are provided within the ASE to contain and identify potential sodium leakage from the IAC AHX.

Consistent with PDC 74, *Sodium/water reaction prevention/mitigation*, the IAC is designed and located to avoid contact between sodium and water. The ASE air intakes and exhausts are provided with weather caps to limit water intrusion into the ASE. The ASEs are not located within proximity of water-containing systems. Sodium containment features are provided within the ASE to limit exposure of potential sodium leakage to the surrounding environment.

Consistent with PDC 75, *Quality of the intermediate coolant boundary*, the portion of the IAC AHX that forms a part of the intermediate coolant boundary is designed, fabricated, erected, and tested to quality standards commensurate with its safety-significant function in accordance with applicable design and construction code requirements.

Consistent with PDC 76, *Fracture prevention of the intermediate coolant boundary*, the intermediate coolant boundary portion of the IAC AHX is designed with sufficient margin to provide assurance that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. The AHX is not subject to irradiation embrittlement effects due to its installation location relative to the reactor core. Material selection, design, and fabrication of the AHX considers the effects of elevated operating temperatures, fatigue, loss of ductility, and chemical compatibility to provide assurance that the material behaves in a non-brittle manner and provides resistance to rapidly propagating fracture. The intermediate coolant boundary within the IAC AHX operates at low pressure and is designed and constructed to satisfy applicable stress limits under anticipated operating conditions with sufficient margin to design conditions. Over-pressure protection of the intermediate coolant boundary is provided by the SCG as described in Section 7.2.3.

Consistent with PDC 77, *Inspection of the intermediate coolant boundary*, the intermediate coolant boundary portion of the IAC AHX is designed to permit periodic inspection and functional testing to assess structural and leaktight integrity. The intermediate coolant boundary is within the scope of pre-service and in-service monitoring and non-destructive examination (MANDE) performed under the Reliability and Integrity Management Program described in Chapter 8.

Functional Requirements and Design Criteria

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as safety-related (SR) functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The IAC supports the following safety-significant function, which is designated as NSRST to indicate the safety classification of IAC components relied upon to perform the function:


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| Function ID | Function Description | IAC Functional Support |
|---|---|---|
| DL4-HR1 | IAC Passive Mode Operation (NSRST) | The IAC transfers heat from the IHT by natural convection airflow to the atmosphere for licensing-basis events when the Intermediate Sodium Pump (ISP) and Reactor Air Cooling System are unavailable. |

Safety classifications for IAC components are listed in Table 7.2.2-1.

Regulatory Guidance

Regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the IAC, along with a description of the use and conformance to the guidance, includes:

RG 1.36, Revision 1, *Nonmetallic Thermal Insulation for Austenitic Stainless Steel*

Full Conformance. Methods and procedures are used for selection of non-metallic insulation to prevent stress-corrosion cracking of stainless steel by minimizing leachable chlorides and fluorides to the lowest practicable levels.

Seismic Classification and Design Codes

Seismic classifications for IAC components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The seismic classification of IAC components are listed in Table 7.2.2-1. Seismic design, analysis, and qualification is performed as described in Section 6.4.

Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of IAC components as special treatments, as described in Section 6.3. The safety-significant IAC components are designed in conformance with the industry consensus codes and standards indicated in Table 7.2.2-2.

### 7.2.2.3 System Design and Performance Evaluation

The IAC is designed to provide reactor core decay heat removal capability. The system also provides low reactor power heat removal capability in support of plant start-up and shutdown for transition to and from Nuclear Island Salt System cooling, respectively. Reactor heat is rejected to the atmosphere by airflow across the AHX within the ASE. The IAC is shut down and available in standby while the plant is operating in Mode-1.

IAC Active Mode provides normal operation cooling to the intermediate loop with forced circulation of the intermediate sodium provided by operation of the ISPs and forced airflow across the AHXs from operation of the IAC blowers. A single train of IAC operating in Active Mode provides the required heat removal for the limiting decay heat load.


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IAC Blower Mode provides off-normal operation cooling to the intermediate loop with natural circulation of the intermediate sodium and forced airflow across the AHXs from operation of the IAC blowers when the ISPs are unavailable. A single train of IAC operating in Blower Mode provides the required heat removal for the limiting decay heat load.

IAC Passive Mode provides emergency operation cooling to the intermediate loop with natural circulation of the intermediate sodium and natural convection airflow across the AHX tubes. This IAC operating mode provides the safety-significant defense-in-depth backup to the SR Reactor Air Cooling System heat removal. No automatic actuations or manual operator actions are required for the IAC to perform its passive heat removal function in Passive Mode. No electrical power is required to initiate natural convection airflow across the AHX. The AHXs are located at an elevation above the Primary Heat Transport System Intermediate Heat Exchanger that is sufficient to provide the necessary pressure potential to drive natural circulation in the intermediate loop as described in Section 7.1.4. Natural convection airflow is induced in the air stack by the decreased density of the heated air in the AHX. Ambient air is drawn in through the low-elevation air intake and the heated air exits through the elevated air stack exhaust openings. In Passive Mode, the IAC provides the required heat removal for the limiting decay heat load with two trains in operation.

IAC components are exposed to operating environments of high temperature liquid sodium, ambient air, and outdoor weather conditions. Material selections for IAC components are based on compatibility with the applicable design internal and external environments, as well as design loadings associated with plant operational events in accordance with applicable design code requirements. The AHX assembly is designed and analyzed in accordance with the requirements of ASME BPVC Section VIII (Reference 7.2.2-1).

#### 7.2.2.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the IAC is described in Section 7.2.2.2. The following programmatic special treatments are preliminarily identified as applicable to the IAC to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- Equipment Qualification Program
- Post-Construction Inspection, Testing, and Analysis Program

#### 7.2.2.5 Instrumentation Requirements

IAC system and equipment parameters are monitored by the Nuclear Island Control System (NIC) (described in Section 7.6.2). System process and equipment parameters are also monitored via local control panels or local gauges.


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## References

7.2.2-1 American Society of Mechanical Engineers “Boiler and Pressure Vessel Code, Section VIII: Rules for Construction of Pressure Vessels,” ASME, 2021

7.2.2-2 American Society of Mechanical Engineers STS-1, “Steel Stacks,” ASME, 2021

7.2.2-3 Sheet Metal and Air Conditioning Contractors' National Association, “Accepted Industry Practice for Industrial Duct Construction,” SMACNA 1108, 2008



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**Table 7.2.2-1 IAC Safety and Seismic Classifications**

| Component | Safety Classification | Seismic Classification |
|---|---|---|
| ASE, Air Stack | NSRST | SCN3 |
| AHX | NSRST | SCN3 |
| Blower (flowpath integrity), Heater (flowpath integrity), Dampers, Screens, Stack Cap | NSRST | SCN3 |


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**Table 7.2.2-2 Applicable Codes and Standards**

| Standard | Edition | Title |
|---|---|---|
| ASME Boiler and Pressure Vessel Code, Section VIII | 2021 | Rules for Construction of Pressure Vessels |
| ASME STS-1 (Reference 7.2.2-2) | 2021 | Steel Stacks |
| SMACNA 1108 (Reference 7.2.2-3) | 2008 | Accepted Industry Practice for Industrial Duct Construction |



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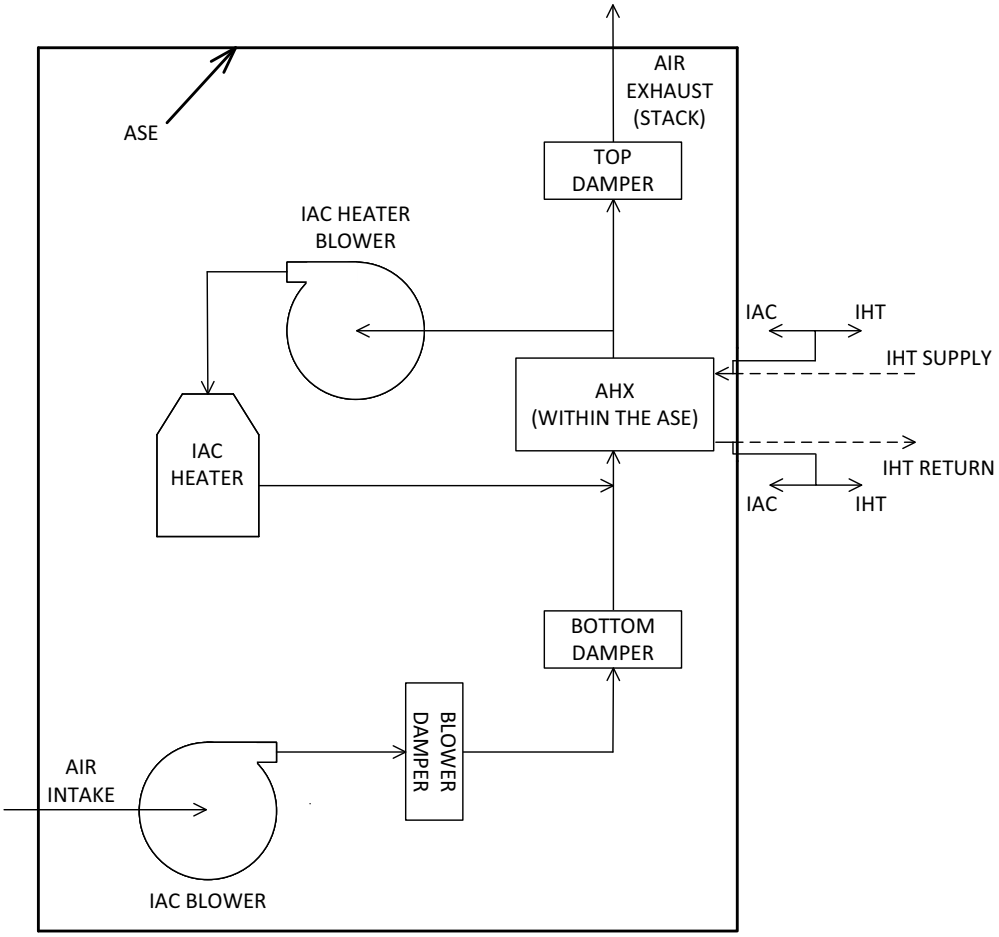
**Figure 7.2.2-1 IAC Simplified Arrangement**



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## 7.2.3 Sodium Cover Gas System

### 7.2.3.1 Summary Description

The Sodium Cover Gas System (SCG) controls, monitors, and supplies inert argon gas to various systems and components throughout the Reactor Building (RXB), Reactor Auxiliary Building (RAB), and Fuel Handling Building (FHB). The SCG comprises the Primary SCG and the Intermediate SCG. The Primary SCG maintains argon gas supply to the Reactor Vessel (RV) feed line, RV-Guard Vessel (GV) annulus, seals and gas barriers on the reactor head, Primary Sodium Pump (PSP) lube oil and gas barrier sealing, and purge gas to various SSCs. The Intermediate SCG maintains argon gas supply for gas spaces in the Intermediate Heat Transport System (IHT), Intermediate Sodium Pump (ISP) lube oil and gas barrier sealing, IHT tank bubbler tubes, and purge gas to various SSCs.

#### 7.2.3.1.1 Primary Sodium Cover Gas

The Primary SCG consists of gas distribution, monitoring, and filtering and is located in the RXB and FHB. It receives high pressure, purified argon from the Nuclear Island Argon Gas Distribution and Storage System (NGA) into the gas distribution modules in the primary supply control skid. Local argon accumulators are located directly downstream of the NGA supply and provide additional inventory in the event of a disruption of NGA supply. The modules adjust gas pressure and control the flow to the various seals, volumes, and penetrations connected to the Reactor Enclosure System (RES) according to the specific needs of the interface. These include the RV, RV-GV annulus, and the various reactor head seals, purges, and volumes which include the Fuel Transfer Lift (FTL), In-Vessel Transfer Machine (IVTM), Rotatable Plug Assembly (RPA), Control Rod Drive Mechanisms (CRDMs), and PSPs.

All gas entering the RV is exhausted through a single port during normal operation. The exhaust gas passes through the vapor trap cell wall and into the primary filter skid. In the skid, the gas passes through the vapor trap which consists of a preheater and vapor condenser, then through an aerosol filter to remove sodium vapors and aerosols. If cesium is present the gas is routed through a cesium vapor filter before being exhausted to the Gaseous Radwaste Processing System (RWG) located in the FHB. A separate exhaust path used to support failed assembly lift test by sipping and for refueling conditions is taken from the FTL, routed through filters, and exhausted directly to RWG. Gas in the RV-GV annulus is generally recycled in a closed loop and only exhausts to relieve high pressure. This gas exhausts directly to RWG or Nuclear Island Heating, Ventilation, and Air Conditioning System (NHV), depending on activity.

The Primary SCG provides monitoring for the primary cover gas, RV-GV annulus gas, and Primary SCG enclosures. The functions of primary cover gas monitoring include impurity monitoring, gas sampling, radiation monitoring, and tag gas monitoring. Monitoring of the sampling and monitoring cabinet enclosure atmospheres provides leak detection for associated cover gas piping. The RV-GV annulus gas monitoring provides sodium and cover gas leak indication through oxygen monitoring, radiation monitoring, and filter differential pressure monitoring.

A simplified block flow diagram of the Primary SCG is shown in Figure 7.2.3-1.


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#### 7.2.3.1.2 Intermediate Sodium Cover Gas

The Intermediate SCG receives purified argon gas from the NGA within the RAB, adjusts pressure, and controls flow as required by the downstream processes. Local argon accumulators are located directly downstream of the NGA supply and provide additional inventory in the event of a disruption of NGA supply. The Intermediate SCG distributes the gas to the required locations:

- Intermediate Sodium Pump (ISP) tanks
- Intermediate Heat Transport System (IHT) drain tanks
- IHT tank bubbler tubes
- ISP gas barriers
- ISP lube oil skids
- Aerosol filter purges and sample line purges

Gas is introduced through the ISP gas barrier to prevent sodium deposition. Additional gas is introduced directly into the ISP tank as needed to maintain a minimum gas flow for monitoring and pressure control. The gas from the ISP tanks and ISP gas barriers passes to the expansion tanks and then to the drain tanks via equalization lines. A portion of the flow within the equalization lines is diverted to the intermediate sampling and monitoring through a set of filters. The remaining gas then passes through the drain tanks, which allows some aerosol settling, and then through aerosol filters, which remove the remainder of the sodium vapors and aerosols. The gas is then discharged through an oil bubbler to NHV. The oil bubbler provides a minimum back-pressure and additional gas filtering.

A simplified block flow diagram of the Intermediate SCG is shown in Figure 7.2.3-2.

### 7.2.3.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the SCG safety design basis are described below.

#### 7.2.3.2.1 Compliance with Principal Design Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the SCG is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, safety-significant SCG SSCs maintain their capability to perform their safety-significant functions during adverse conditions of natural phenomena. Safety-significant SCG SSCs are designed for protection against natural phenomena commensurate with their safety significance. SCG SSCs are protected from most adverse natural phenomena conditions by RXB (Section 7.8.1) and RAB (Section 7.8.3) structures. SCG SSCs that perform safety-significant functions are primarily located below grade (all SR SCG SSCs are located in the RXB below grade). Safety-significant SSCs that are above grade are designed to a hazard level commensurate with the safety significance of their functions. SCG SSC loads are supported by the RXB and RAB structures. These structures are designed to accommodate SCG normal loads, dead loads, live loads, and


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loads from natural phenomena. SCG SSCs are designed and analyzed according to site seismic response spectra and RAB and RXB structure designs. SR SSCs are designed to withstand the effects of a seismic event, such as during and after a safe shutdown earthquake (SSE). Non-safety-significant SCG SSCs whose failure could reduce the functioning of safety-related (SR) SSC to an unacceptable level are designed to withstand an SSE.

Consistent with PDC 3, *Fire protection*, safety-significant SCG SSCs are located within the RAB and RXB to avoid damage from fires and explosions. Components that contain sodium are located in inert environments, or enclosed within pipe chases or secondary enclosures to mitigate the effects of sodium leakage. SCG SSCs are designed and constructed of noncombustible and fire-resistant materials.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, safety-significant SCG SSCs are protected from environmental conditions, dynamic transients, and chemical accidents. The SCG is designed to be protected from the effects of liquid sodium and its aerosols and oxidation products. Piping that contains sodium aerosols is provided with the means to collect and monitor leakage.

Consistent with PDC 10, *Reactor design*, SCG leakage is maintained below acceptance limits during normal operation and anticipated operational occurrences (AOOs). The SCG provides positive pressure at slightly above atmospheric pressure to the RV and the GV. The SCG provides argon gas to reactor head seals to prevent leakage of cover gas to the environment.

Consistent with PDC 13, *Instrumentation and control*, the SCG provides instrumentation and monitoring for the RV pressure, GV pressure, primary coolant boundary and SCG passive barrier isolation valve positions, RV relief valve positions, and cover gas radiation monitoring. These SCG SSCs provide appropriate signals to instrumentation and control systems according to their safety significance.

Consistent with PDC 14, *Primary coolant boundary*, the portions of the SCG that constitute the primary coolant boundary are designed, fabricated, erected, and tested to provide an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The primary coolant boundary operates at low pressure, and metallic boundary components are designed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5 (Reference 7.2.3-1). Provisions are provided for leak testing primary coolant boundary components in accordance with applicable requirements of the ASME Code. Over-pressure protection of the primary coolant boundary is provided per the requirements of the ASME Code.

Consistent with PDC 15, *Primary coolant system design*, the design of the SCG provides sufficient margin such that the design conditions of the primary coolant boundary are not exceeded during conditions of normal operation, including AOOs. The primary coolant boundary operates at low pressure and is designed and constructed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5. Sufficient margin to design conditions is provided by the design requirements established as described in Section 6.4.3. Over-pressure protection of the primary coolant boundary is provided per the requirements of the ASME Code.


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Consistent with PDC 16, *Containment design*, the SCG provides multiple layers of functional containment to limit the release of radionuclides during postulated accident conditions and to ensure that design conditions are not exceeded. The portion of the SCG from the RV up to and including the first isolation valve are part of the primary coolant boundary and are designed and constructed according to ASME Section III, Division 5 to ensure the radionuclide retention function is maintained. The SCG utilizes secondary enclosures that prevent potential releases from exceeding SARRDLs as applicable. Radionuclide sources include activated primary sodium, cover gas activity, and activated argon. Additional SCG functional containment support includes:

- The vapor trap cell is an enveloping barrier that prevents leaks from the exhaust piping, vapor condenser, aerosol filter, and cesium filters from being released to accessible and general areas. This functional containment barrier utilizes a steel liner assembly that is sealed to prevent leakage and is installed within a concrete biological shield. Penetrations and floor plugs are mated to the liner assembly surfaces to limit leakage out of the cell.
- SCG exhaust gas monitoring cabinets outside of the HAA are contained within secondary enclosures that are held at slight vacuum pressures to prevent possible leakage of radionuclides in the event of a component or pipe rupture. Leak rates are maintained at levels that prevent exceeding SARRDLs.
- The SCG provides argon flow and pressure control to the primary barrier purge and barrier gas seals on the RES to ensure that the radionuclide retention function is achieved. Supported SSCs include Control Rod Drive Mechanism (CRDM), RPA, FTL, PSP, and IVTM. Positive pressure argon gas ensures that potential leaks result in argon gas entering the RV.

Consistent with PDC 30, *Quality of primary coolant boundary*, the portions of the SCG that constitute the primary coolant boundary are SR to ensure a low probability of abnormal leakage, rapidly propagating failures, and gross ruptures. The SCG provides cover gas to the RV at a pressure that is slightly above site atmospheric conditions. SCG primary coolant boundary SSCs are designed, fabricated, constructed, and tested according to ASME Section III, Division 5 to ensure the integrity of the system is maintained during all postulated accidents. The SCG SSCs are designed to satisfy ASME code stress limits under applicable service level conditions and load combinations. SCG SSCs that are part of the primary coolant boundary are designed, analyzed, tested, verified, and qualified according to the Equipment Qualification Program. Monitoring and non-destructive examination (MANDE) is performed during pre-service and in-service phases under the reliability and integrity management (RIM) Program (as described in Chapter 8). The RIM Program follows ASME Section XI, Division 2 (Reference 7.2.3-2) as endorsed by Regulatory Guide 1.246, Acceptability of ASME Code, Section XI, Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,” for Non-Light Water Reactors, Revision 0. The SCG circulates argon gas to the RV-GV annulus for leakage monitoring from the RV into the GV. Monitoring equipment in the SCG is capable of determining if a leak exists by trending radiation levels, oxygen concentrations, and absolute filter differential pressures.


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Consistent with PDC 31, *Fracture Prevention of primary coolant boundary*, SCG primary coolant boundary SSCs are designed for environmental effects including sodium environment, thermal conditions, radiation fluence, and dynamic effects of fluid during all normal operating conditions, maintenance conditions, testing operations, and postulated accidents. The SCG provides overpressure protection for the primary coolant boundary in order to prevent exceeding its design pressure. SCG provides an inert cover gas and monitors gas composition to mitigate primary coolant contamination. SCG provides high purity cover gas at slightly above atmospheric pressure to prevent contaminants, such as oxygen, from interacting with the primary coolant which can cause adverse effects such as accelerated corrosion and plugging of small orifices. SCG primary coolant boundary SSCs are SR, ASME Section III, Division 5, to provide reliability and ensure integrity under all operating, maintenance, testing, and postulated accident conditions. The SCG SSCs are designed to satisfy ASME code stress limits under applicable service level conditions and load combinations. SCG primary coolant boundary SSCs are designed, analyzed, tested, verified, and qualified according to the Equipment Qualification Program. The RIM Program uses Degradation Mechanism Assessment (DMA) for safety-significant SSCs in the SCG per ASME Section XI, Division 2.

Consistent with PDC 32, *Inspection of primary coolant boundary*, MANDE is performed during pre-service and in-service phases under the RIM Program. The DMA identifies the appropriate MANDE methods to apply to each SSC that forms a portion of the primary coolant boundary. The DMA includes SSC materials of construction, the environment in which the SSC operates, limits of impurities, and fabrication techniques. The assessment of each applicable SCG SSC ensures the SSC continues to function with an acceptable level of reliability against all identified degradation phenomenon. SCG piping and equipment up to and including the isolation valves form a part of the primary coolant boundary; however, actuation of primary coolant boundary isolation valves to close is not an SR function and is not required to satisfy functional containment requirements. As discussed under PDC 31, SCG SSC materials are chosen to prevent and mitigate the effects of degradation within the system; austenitic stainless steels are used to mitigate the effects of neutron embrittlement and to protect the primary coolant boundary from fracture and failure. The SCG is designed to permit periodic inspection and functional testing of important areas and features to assess their structural and leak tight integrity.

Consistent with PDC 60, *Control of releases of radioactive materials to the environment*, the SCG controls exhaust gas from the RV and GV to the RWG for radiological effluent handling. During normal operations, radiological effluent exists in the SCG exhaust pathway due to activated argon, fission products, and activated sodium. Effluent that is not filtered out in the vapor condenser, aerosol filter, or cesium filter are routed to the RWG for final holdup and disposal. SCG exhaust gas from the RV-GV annulus normally exhausts to the NHV as needed but will discharge to the RWG during abnormal conditions in which radiological effluent exists at higher concentrations. During a potential overpressure event, the SCG relief valves open and discharge sodium vapor/aerosols, argon, and other effluents to a holdup volume. This volume is designed for sufficient holdup capacity for maximum overpressure conditions and discharge. The volume holds up the radiological effluent until maintenance can be performed at safe exposure rates.


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Consistent with PDC 63, *Monitoring fuel and waste storage*, the SCG interfaces with the Radiation Monitoring System (RMS) to monitor radiation levels and to detect when excessive levels are reached in the primary cover gas that may be indicative of issues with fuel stored in the In-Vessel Storage (IVS).

Consistent with PDC 70, *Intermediate coolant system*, applicable SCG SSCs are designed and constructed in accordance with ASME B31.3 (Reference 7.2.3-3) and ASME Section VIII (Reference 7.2.3-4) to ensure that design conditions of the intermediate coolant boundary are not exceeded and that the integrity of the primary coolant boundary is maintained. The Intermediate SCG is operated at pressures slightly above site atmospheric conditions and contains overpressure protection equipment that prevents external overpressurization of the primary system. Intermediate SCG relief valves and rupture disks are designed and fabricated in accordance with ASME Section VIII.

Consistent with PDC 71, *Primary coolant and cover gas purity control*, the SCG receives ultra-high purity argon gas from the NGA which is then distributed to the RV, RV seals, purges, and PSP. Positive pressure gas enters the RV cover gas region and sweeps across to the lower pressure regions. Gas is exhausted from the RV through SCG equipment to remove metal vapors and particulates prior to sampling the stream to monitor for gas composition and exhausting to RWG. Argon gas is supplied to reactor head penetrations. Purge gas is used to minimize migration of sodium vapors and radioactive gases into the penetration annuli. Barrier gas seals are used for static seals to ensure that argon gas leaks into associated equipment and prevents radiological effluent from being released. The SCG exhaust from the RV is sampled and monitored for impurities and radiological effluent. The SCG monitors for oxygen, hydrogen, methane, and nitrogen which can be used to determine the source of a leak in the system.

Consistent with PDC 72, *Sodium heating systems*, the safety-significant portions of the SCG that contain or could contain sodium are equipped with heating systems to prevent freezing. The overpressure protection discharge piping to the vapor trap cell is heated to ensure that freezing and plugging do not occur during an overpressure event. The Primary SCG supply line is equipped with heat tracing between the process heater and the reactor head connection, including the overpressure protection skid, to ensure that sodium diffusion back towards argon supply SSCs will not freeze if gas flow is stopped. During an overpressure event, sufficient heating is provided by the process gas to maintain adequate temperatures and therefore the trace heaters on the overpressure relief lines are non-safety-related with no special treatment (NST). The SCG exhaust gas from the RV contains significant sodium content which is removed by the vapor condenser and aerosol filters in the vapor trap cell. During periods when these SSCs are unavailable, heat tracing maintains the sodium content at sufficient temperatures to prevent freezing and plugging of the condenser, aerosol filters, and interconnecting piping.


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Consistent with PDC 73, *Sodium leakage detection and reaction prevention and mitigation*, sodium leak detection and sodium reaction prevention and mitigation applicable to SCG includes:

- The Primary SCG provides detection capabilities for sodium leakage from the RV into the GV. Argon gas is supplied to the RV-GV annulus to provide an inert environment that mitigates the effects of a sodium leak. Annulus gas is recirculated and monitored for sodium buildup via absolute filter differential pressure changes, oxygen concentrations, and radiation levels.
- The Intermediate SCG provides monitoring for impurities, particularly nitrogen gas.
- The vapor trap cell contains the primary filter skid, which includes the sodium vapor condenser, aerosol filter, cesium filter, instruments, and interconnected piping. The cell is inerted with nitrogen and maintained at slight vacuum conditions to mitigate the effects of a leak from any of the SSCs contained within the cell. Oxygen content is maintained at low levels to prevent significant reactions for leaked sodium while protecting the stainless steel from nitriding.

Consistent with PDC 74, *Sodium water reaction prevention and mitigation*, water sources and water containing systems are separated from the sodium containing portions of SCG to preclude direct water-sodium interaction. The SCG detects sodium-salt reactions between the Nuclear Island Salt System (NSS) and the IHT by continuously monitoring for reaction byproducts, specifically nitrogen gas, in the Intermediate SCG.

Consistent with PDC 75, *Quality of the intermediate coolant boundary*, applicable SCG SSCs are designed and constructed in accordance with ASME B31.3 to ensure that design conditions of the intermediate coolant boundary are not exceeded, and that the integrity of the primary coolant boundary is maintained during postulated accidents. The Intermediate SCG is operated at pressures slightly above site atmospheric conditions and contains overpressure protection equipment that prevents external over pressurization of the primary system. Intermediate SCG relief valves and rupture disks are designed and fabricated in accordance with ASME Section VIII.

Consistent with PDC 76, *Fracture prevention of the intermediate coolant boundary*, applicable SCG SSCs are designed according to ASME B31.3 to ensure that SSC materials behave in a non-brittle manner and the probability of rapidly propagating fracture is minimized when stressed under operating, maintenance, testing, and accident conditions. SCG pipe and component materials are austenitic stainless steel. This material type is resilient to degradation, swelling, irradiation creep, fatigue, loss of ductility, and dimensional changes. The SCG provides pressure control for applicable Intermediate SCG and IHT SSCs. Pressure is controlled through a series of regulating and pressure controlling valves. Argon gas supply cabinets are provided with relief valves for overpressure protection to prevent exceeding design limits. The SCG provides the IHT with drain tank and ISP expansion tank overpressure protection skids that further prevent exceeding design pressure limits.

Consistent with PDC 77, *Inspection of the intermediate coolant boundary*, MANDE is performed during pre-service and in-service phases under the RIM Program. The RIM Program follows ASME Section XI, Division 2 and is used to monitor and examine SSCs that affect plant safety at


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appropriate frequencies for acceptable performance. The SCG is designed to permit periodic inspection and functional testing of important areas and features to assess their structural and leak tight integrity.

Consistent with PDC 78, *Primary coolant system interfaces*, the SCG directly interfaces with the primary coolant; there is no boundary interface location between the primary coolant and the SCG. Refer to the discussions provided for PDC 71 and PDC 79 on maintaining the argon interface with the primary coolant.

Consistent with PDC 79, *Cover gas inventory maintenance*, the SCG provides ultra-pure argon gas to the RV cover gas region to sweep contaminants and to provide a buffer between the hot pool and reactor head. Two sets of redundant supply pressure regulators ensure that purified argon gas supplied by the NGA to the RV and the RV-GV annulus is reduced to pressures that are slightly above atmospheric conditions. One set of backpressure control valves modulates to ensure that RV pressure is maintained at nominal conditions. SCG instruments monitor supply and exhaust pressure and flow. The SCG provides additional cover gas inventory functions, including:

- A flow control valve in the SCG supply ensures that a nominal minimum flow rate is maintained during all modes of operation. Local argon accumulators provide supplies during a temporary loss or disruption of the NGA argon supply. During postulated accidents in which the RV rapidly depressurizes as coolant decreases in temperature, two supply pressure control valves act in parallel to provide sufficient makeup argon to the RV. Flow instruments provide indication of abnormal conditions in which flow may cease or significantly increase for events such as inadvertent isolation or a line break, respectively.
- Reactor head seals are provided with positive argon gas from the SCG to prevent and mitigate the effects of a leak. Positive pressure on the seals ensures that cover gas in the RV is maintained and that potential leakage will only occur into the RV. Cover gas inventory is further maintained by primary coolant boundary isolation valves located on the supply and exhaust pathways.
- The SCG performs online sampling and monitoring of the exhaust gas from the RV; impurities, such as oxygen, are potential indicators of a leak in the primary coolant boundary.


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7.2.3.2.2 Functional Requirements and Design Criteria

Probabilistic risk assessment safety functions (PSFs) are determined through the plant-specific probabilistic risk assessment (PRA) described in Section 3.1. Safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The SCG supports the following SR and NSRST functions:

| Function ID | Function Description | SCG Functional Support |
|---|---|---|
| DL3-RR1c | SCG Primary System Barrier (SR) | The SCG supply and exhaust piping, from the RES to the SCG primary coolant boundary isolation valves, forms a portion of the primary coolant boundary. Closure of the isolation valves is not an SR primary coolant boundary function. |
| DL3-RR7 | RES Pressure Relief (SR) | The SCG provides overpressure protection for the RV. This includes RV internal overpressure protection as well as RV external overpressure protection (i.e., overpressure within the RV-GV annulus and CRDM housings). |
| DL4-HR1 | IAC Passive Mode Operation (NSRST) | The un-isolable SSCs of the Intermediate SCG are designed to maintain their pressure retaining functions during all modes and LBEs where IAC operability is necessary. These Intermediate SCG SSCs communicate with the IHT gas space and are included as a portion of functional containment. The Intermediate SCG provides overpressure protection for IAC to ensure that the Sodium-Air Heat Exchanger (AHX) heat removal capabilities are maintained. |
| DL2-RR10 | Primary SCG Barrier (NSRST) | SCG SSCs provide a radionuclide barrier beyond the primary coolant boundary isolation valves. |
| DL4-RR4 | SCG Cells Barriers (NSRST) | The SCG vapor trap cell is designed to provide a low leakage barrier for postulated SCG releases; the primary overpressure protection equipment is designed to discharge into the vapor trap cell. SCG components for which a leak can exceed SARRDL are designed with an enveloping barrier. |
| DL4-RR4a | Automatically Close SCG Isolation Valves on Leak Detection (NSRST) | The SCG isolation valves close upon leak detection. |


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| Function ID | Function Description | SCG Functional Support |
|---|---|---|
| DL4-RR4c | Vapor Trap Cell Isolation on Overpressure (NSRST) | The primary coolant overpressure relief valves discharge into the vapor trap cell. The vapor trap cell is isolated on a primary coolant overpressure relief valve open indication. The vapor trap cell is a low leakage barrier for postulated releases. |
| DL5-PAM1 | Post Accident Monitoring (NSRST) | The SCG provides components for monitoring the following PAM variables:<br>• Reactor Vessel Pressure<br>• Guard Vessel (GV) Pressure<br>• SCG Isolation Valve Position<br>• Cover Gas Radiation<br>• SCG Relief Valve Position |

Portions of the SCG form part of the primary coolant boundary. The primary coolant boundary is described in Section 5.3. SCG piping that penetrates the RES, up to and including the first isolation valve, is part of the primary coolant boundary.

SSC safety classifications are performed as described in Section 5.1. SRDC corresponding to the SR functions are derived as described in Section 6.1.2 and identified in Section 5.2. The Safety Related Design Criteria (SRDC) are incorporated into the design of the SCG as described in Section 7.2.3.1 and Section 7.2.3.3. Seismic design, analysis, and qualification are performed as described in Section 6.4. Seismic classifications of SSCs are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. SR SCG SSCs are classified SCS1. SSCs that are classified SCS1 are qualified to withstand seismic loads associated with the SSE (described in Section 2.6.2) without loss of the capability to perform SR functions. NSRST SCG SSCs are classified SCN1 or SCN3. Seismic special treatments are applied to NSRST portions of the SCG, as applicable based on seismic classification, to provide assurance that risk-significant functions or functions required for adequate defense-in-depth are capable of being performed. Major equipment safety classifications for Primary SCG and Intermediate SCG are identified in Figure 7.2.3-1 and Figure 7.2.3-2, respectively. These figures provide identification of general SSC classifications for SCG equipment; individual SSCs are classified as described in Section 5.1.

#### 7.2.3.2.3 Regulatory Guidance

Regulatory Guide 1.11 Rev. 1, 2010, “Instrument Lines Penetrating the Primary Reactor Containment,” partial conformance. This guidance is not directly applicable to functional containment SSCs. The SCG design includes process supply lines to reactor head seals that are analogous to instrument lines that penetrate containment as described in Section C, however, the functional containment differs from the containment of a traditional light water reactor (LWR) described in the Regulatory Guide.


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Regulatory Guide 1.36 Rev. 1, 2015, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,” partial conformance. Applicable SCG SSC insulation is controlled to prevent stress-corrosion cracking due to leachable contaminants. As applicable, SCG insulation minimizes the degree of leachable chlorides and flourides to the lowest practicable levels. This is accomplished by procuring insulation that reduces the possibility of external contamination and through qualification testing. Portions of the SCG that fall outside of Division 1 design temperature limits shall conform with Division 5, which is not covered under this guidance.

Regulatory Guide 1.44 Rev. 1, 2011, “Control of the Processing and Use of Stainless Steel,” full conformance. The SCG design includes austenitic stainless steel in the form of Type 304 or Type 316 steel. This includes portions of the SCG that comprise the primary system boundary and portions that perform a function in the shutdown of the reactor. Applicable SCG SSCs conform with the criteria outlined in Section C as applicable.

Regulatory Guide 1.45 Rev. 1, 2008, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” partial conformance. This guidance is not directly applicable to the SCG design. SCG instruments monitor for cover gas leakage from seals, plugs, and other interfaces with the reactor head as well as leaks from within the SCG. SCG instruments and mechanical components that perform RV-GV annulus gas monitoring provide indication of potential RV to GV leaks, including leaks that may involve primary sodium.

Regulatory Guide 1.61 Rev. 1, 2007, “Damping Values for Seismic Design of Nuclear Power Plants,” full conformance. Applicable SCG SSCs are analyzed to the acceptable damping values used in elastic dynamic seismic analysis. As applicable, piping damping shall conform to Section 2 and mechanical and electrical component damping shall conform to Section 5 for SSEs.

Regulatory Guide 1.68.3 Rev. 1, 2012, “Preoperational Testing of Instrument and Control Air Systems,” partial conformance. Applicable SCG SSCs that use instrument air are tested to verify that all components function properly at normal pressures and following possible pressure increases and that the system responds as designed to a loss-of-air-pressure event. Exception is taken to Item 1 of Section C regarding Regulatory Guide 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants, Revision 4, as it pertains to test programs for LWRs.

Regulatory Guide 1.84, Rev. 39, 2021, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” full conformance. SCG design and construction shall comply with the Code Case tables provided in the RG without exception.

Regulatory Guide 1.87 Rev. 2, 2021, “Acceptability of ASME Section III, Division 5, High Temperature Reactors,” full conformance. The safety-significant SCG SSCs that constitute the primary coolant boundary include design, construction, testing, and quality assurance that conform to the endorsed portions of ASME Section III, Division 5 and ASME Section II Part D (Reference 7.2.3-5) with applicable exceptions as noted in the Regulatory Guide.


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Regulatory Guide 1.97 Rev. 5, 2019, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” full conformance. Applicable SCG instruments that support Post-Accident Monitoring (PAM) functionality follow IEEE Std. 497-2016 (Reference 7.2.3-6) for applicable PAM variable types. Passive components, such as piping, are not required to adhere to redundancy requirements set forth in IEEE Std. 497-2016.

Regulatory Guide 1.100 Rev. 1, 2020, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” partial conformance. Applicable SCG SR SSCs conform to the Regulatory Guide pertaining to IEEE 344 (Reference 7.2.3-7) and ASME QME-1 (Reference 7.2.3-8) for electrical and mechanical equipment. SCG NSRST SSCs will apply this Regulatory Guide on a case-by-case basis commensurate with the safety significance and function being performed.

Regulatory Guide 1.124 Rev. 3, 2013, “Service Limits and Loading Combinations for Class 1 Linear-Type Supports,” full conformance. The SCG is designed, constructed, and tested to ASME BPVC III Division 5-2017. SCG SSC supports are designed according to this Code, however, Section HFA references and directs the usage of ASME BPVC III Division 1 Section NF. For those sections that use AMSE BPVC III Division 1, this RG shall be followed according to Section C.

Regulatory Guide 1.141 Rev. 1, 2010, “Containment Isolation Provisions for Fluid Systems,” partial conformance. This guidance is not directly applicable to functional containment SSCs. The SCG contains primary coolant boundary isolation valves that form a portion of the functional containment. Exception is taken to containment and containment isolation valves as defined for LWRs and to references of general design criteria (GDC) 54, 55, 56, and 57 as they apply to LWRs.

Regulatory Guide 1.147 Rev. 20, 2021, “In-service Inspection Code Case Acceptability, ASME Section XI, Division 1,” partial conformance. Code cases that are applicable to the SCG design are consistent with the pre-service and in-service program requirements.

Regulatory Guide 1.153 Rev. 1, 1996, “Criteria for Safety Systems (12/85),” partial conformance. As applicable, SCG instruments shall comply with the requirements established in IEEE Std. 603. Exception is taken to the code year used. SCG instruments shall use IEEE Std. 603-2018 (Reference 7.2.3-9) instead of the 1991 edition included in Section C.

Regulatory Guide 1.199 Rev. 1, 2020, “Anchoring Components and Structural Supports in Concrete,” full conformance. As applicable, SCG SSC supports anchors are designed, installed, tested, and evaluated according to ACI 355.2-07 (Reference 7.2.3-10) and ASTM E488/E488M-15 (Reference 7.2.3-11). All exceptions apply as noted in the Regulatory Guide.


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### 7.2.3.3 System Design and Performance Evaluation

The SCG controls, monitors, and supplies inert argon gas to various systems and components and includes the Primary SCG and Intermediate SCG. The major process functions of the Primary SCG include:

- Maintaining coolant purity by preventing oxygen, water, nitrogen, and other contaminants from entering the sodium coolant.
- Maintaining a slight positive pressure within the RV cover gas relative to the HAA.
- Providing overpressure protection for the RV.
- Retaining RV overpressure discharge.
- Providing a boundary to primary coolant leakage under normal and off normal conditions.
- Preventing solid sodium deposition on movable reactor head components.
- Removing sodium vapor and aerosols from the cover gas effluent.
- Removing cesium vapor from the cover gas effluent.
- Monitoring primary cover gas for general radiological conditions and fission products indicative of fuel failures.
- Providing isolation of the Primary SCG.
- Maintaining the RV cover gas activity below specified limits using a purge flow.
- Providing constant pressure and leak detection to seals in the FTL, IVTM, and RPA.
- Providing constant pressure and leak detection to the CRDMs for scram cylinders, drivelines, and housings.
- Providing supply and exhaust connections for temporary maintenance equipment.
- Providing constant flow to purge the CRDM, FTL, IVTM, and RPA annuli.
- Providing overpressure protection for the CRDM housings and drivelines.
- Providing constant pressure gas to the PSP gas barriers.
- Providing pressure-controlled gas to the PSP lube oil systems.
- Providing overpressure protection for the GV.
- Maintaining GV pressure above HAA pressure and below reactor pressure.
- Circulating gas through the RV-GV annulus to monitor for RV leak indicators such as drop in oxygen, increase in filter pressure drop, and/or increased radioactivity.
- Providing purge gas to aerosol filters for backflushing.
- Monitoring for impurities and general radiological conditions.
- Providing a grab sample point for primary cover gas.
- Analyzing isotopic tag gas ratios following fuel failure to identify specific failed fuel assemblies.


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The Primary SCG monitoring consists of five sealed cabinets located on grade level in the RXB and services both primary cover gas and RV-GV gas. Each of these cabinets performs a separate monitoring function of the effluent streams. The functions of primary cover gas monitoring include:

- Gas filtration.
- Impurity monitoring for hydrogen, oxygen, nitrogen, and methane.
- Gas sampling.
- Gamma spectroscopy and radiation monitoring for failed fuel monitoring.
- Tag gas monitoring for failed fuel localization.

A sample line draws a portion of the Primary SCG gas into the gas filtration impurities monitoring and sampling (GFIMS) cabinet after it has passed through the filtration equipment in the primary filter skid. After the sampled gas is further filtered, a portion of the sample gas is routed into the failed fuel monitoring cabinet which contains redundant equipment for radiation monitoring for general activity and for the presence of failed fuel. The gamma spectroscopy analysis cabinet contains a gamma spectroscopy unit that is used for periodic spectral analysis of the cover gas to identify isotopes indicative of failed fuel. The cover gas exhaust stream is monitored for fission products to determine if a fuel cladding breach has occurred. If no fission gas is detected, the gas is discharged directly to the RWG vacuum tank. If an increase in fission gas is detected, indicating a new breach, the gas stream is routed to the tag gas analysis cabinet where xenon and krypton isotopes in the gas stream are stripped out and concentrated. The tag gas analysis cabinet analyzes the xenon and krypton isotopic ratios present with a mass spectrometer, and the specific fuel assembly, or fuel assembly candidates, which contains the breach can be identified. The balance of gas is discharged to the RWG vacuum tank.

The functions of RV-GV annulus gas monitoring include:

- Oxygen monitoring for sodium leak indication.
- Radiation monitoring for sodium or primary cover gas leak indication.
- Absolute filter differential pressure monitoring for sodium leak indication.

The RV-GV annulus module initially adjusts the RV-GV annulus gas oxygen content to the several part per million range while being recirculated via the GV leakage detection and recirculation skid blower. The RV-GV annulus exhaust is routed to a portion of the primary GFIMS cabinet that is dedicated to oxygen and radiation monitoring. The argon gas is monitored continuously during operation for oxygen content, radioactivity, and pressure drop across a filter as it is circulated to determine if a potential RV leak exists. If a leak occurs, the diatomic oxygen content is expected to drop as sodium or sodium vapor oxidizes. Additionally, activity in the annulus may increase if Argon-41, Neon-23, fission products, and/or sodium radioisotopes leak into the RV-GV annulus. Finally, an increase in pressure drop across a filter may also indicate a leak as sodium aerosols and/or sodium oxide accumulates on it.


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The major functions of the Intermediate SCG include the following:

- Maintaining coolant purity by preventing oxygen, water, nitrogen, and other contaminants from entering the intermediate sodium loop via the gas stream.
- Maintaining the ISP tank, IHT expansion tank, and IHT drain tank cover gas pressures above atmospheric pressure.
- Providing constant pressure gas to the ISP gas barriers.
- Providing pressure-controlled gas to the ISP lube oil systems.
- Providing flow-controlled gas to IHT expansion tank and drain tank bubbler tubes for level indication.
- Providing overpressure protection for each IHT tank.
- Providing overpressure protection for each IHT loop.
- Providing a boundary to intermediate coolant leakage under normal and off normal conditions.
- Removing sodium vapor and aerosols from the intermediate cover gas effluent.
- Providing isolation of the Intermediate SCG during plant events.
- Providing gas transfer from drain tanks to expansion tanks during loop drain.
- Providing gas pressure to the drain tanks for pneumatic loop fill.
- Monitoring for SHX salt leakage into the sodium by monitoring the IHT expansion tank cover gas nitrogen levels. Nitrogen is a byproduct of the salt-sodium reaction.
- Monitoring for gas impurities.
- Providing a grab sample point for intermediate cover gas.

The intermediate gas monitoring system, located in the RAB, provides dedicated, identical monitoring systems for each intermediate loop. Each loop provides a gas stream drawn from the equalization lines that is passed into an intermediate GFIMS cabinet. In the cabinet, it is drawn through a HEPA filter before being routed through gas monitoring instruments. Dedicated instruments for the measurement of methane, oxygen, hydrogen, and nitrogen are included. Grab samples can also be taken in the intermediate GFIMS cabinets.

#### 7.2.3.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the SCG are applied as described in Section 7.2.3.2.1. Codes and standards applied to various SCG SSCs include:

- ASME Section III, Division 5
- ASME B31.3
- ASME Section VIII, Division 1
- ASME BPVC V (Reference 7.2.3-12)
- ASME BPVC III Division 1 (Reference 7.2.3-13)
- AISC N690 (Reference 7.2.3-14)
- ASME AG-1 (Reference 7.2.3-15)


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- ASME B16.34 (Reference 7.2.3-16)
- ASME PTC-25 (Reference 7.2.3-17)
- IEEE 382 (Reference 7.2.3-18)
- IEEE 497 (Reference 7.2.3-19)

The following programmatic special treatments are preliminarily identified as relevant to SCG SR and NSRST SSCs as applicable to meet quality, design, reliability, and performance requirements (summary program descriptions are provided in Chapter 8):

- Quality Assurance Program
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- In-service Testing Program
- Equipment Qualification Program
- Post-construction Inspection, Testing, and Analysis Program (see Chapter 12)

7.2.3.5 Instrumentation Requirements

The SCG provides instrumentation and monitoring for RV pressure, GV pressure, primary coolant boundary valve positions, passive barrier valve positions, RV relief valve positions, and cover gas radiation monitoring during normal operations and LBEs. These SCG SSCs provide signals to instrumentation and control systems (See Section 7.6) according to safety significance.

**References**

7.2.3-1 ASME, “Rules for Construction of Nuclear Facility Components - Division 5 - High Temperature Reactors,” ASME Section III, Division 5, 2017.

7.2.3-2 ASME, “Rules for In-service Inspection of Nuclear Reactor Facility Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Reactor Facilities,” ASME Section XI, Division 2, 2019

7.2.3-3 ASME, “Process Piping,” ASME B31.3, 2020.

7.2.3-4 ASME, “Rules for Construction of Pressure Vessels Division 1,” ASME Section VIII, Division 1, 2021.

7.2.3-5 ASME, “Boiler and Pressure Vessel Code, Section II: Materials,” ASME II Part D, 2019

7.2.3-6 IEEE, “Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE Std. 497, 2016

7.2.3-7 IEEE, “IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations,” IEEE 344, 2020


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7.2.3-8 ASME, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” ASME QME-1, 2023

7.2.3-9 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603, 2018

7.2.3-10 ACI, “Qualification of Post-Installed Mechanical Anchors in Concrete & Commentary,” ACI 355.2-07, 2007

7.2.3-11 ASTM, “Standard Test Methods for Strength of Anchors in Concrete Elements,” ASTM E488/E488M-15, 2015

7.2.3-12 ASME, “Nondestructive Examination,” ASME BPVC V, 2021

7.2.3-13 ASME, “Boiler and Pressure Vessel Code, Section III: Rules for Construction of Nuclear Power Plant Components, Subsection NCA: Class 2 Components,” ASME BPVC III Division 1, 2017

7.2.3-14 ANSI/AISC, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” ANSI/AISC N690, 2018

7.2.3-15 ASME, “Code on Nuclear Air and Gas Treatment,” ASME AG-1, 2019

7.2.3-16 ASME, “Valves – Flanged, Threaded and Welding End,” ASME B16.34, 2020

7.2.3-17 ASME, “Pressure Relief Devices,” ASME PTC-25, 2014

7.2.3-18 IEEE, “IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations and Other Nuclear Facilities,” IEEE 382, 2006

7.2.3-19 IEEE, “Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497, 2016


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**Figure 7.2.3-1 Primary Cover Gas System Block Diagram**

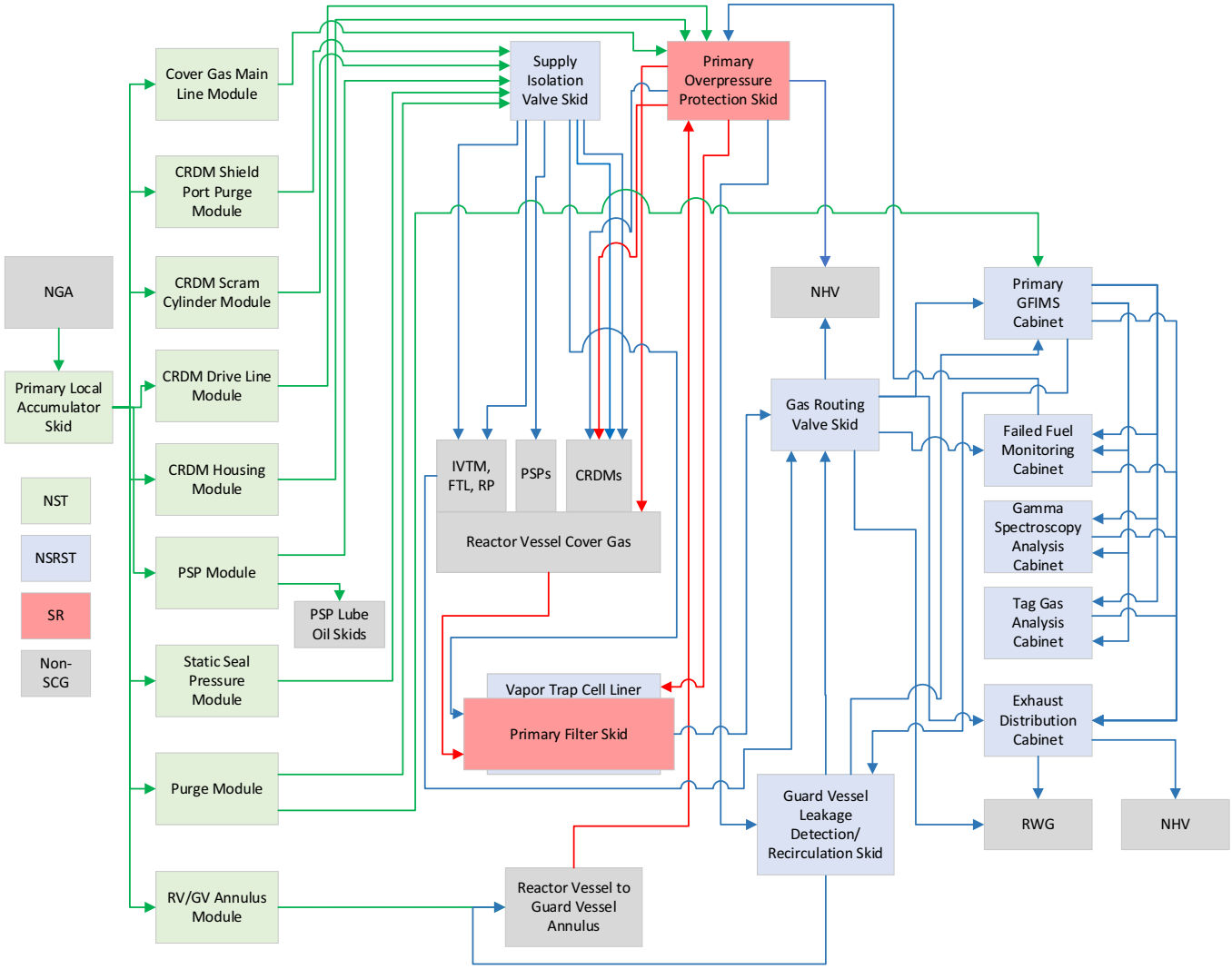




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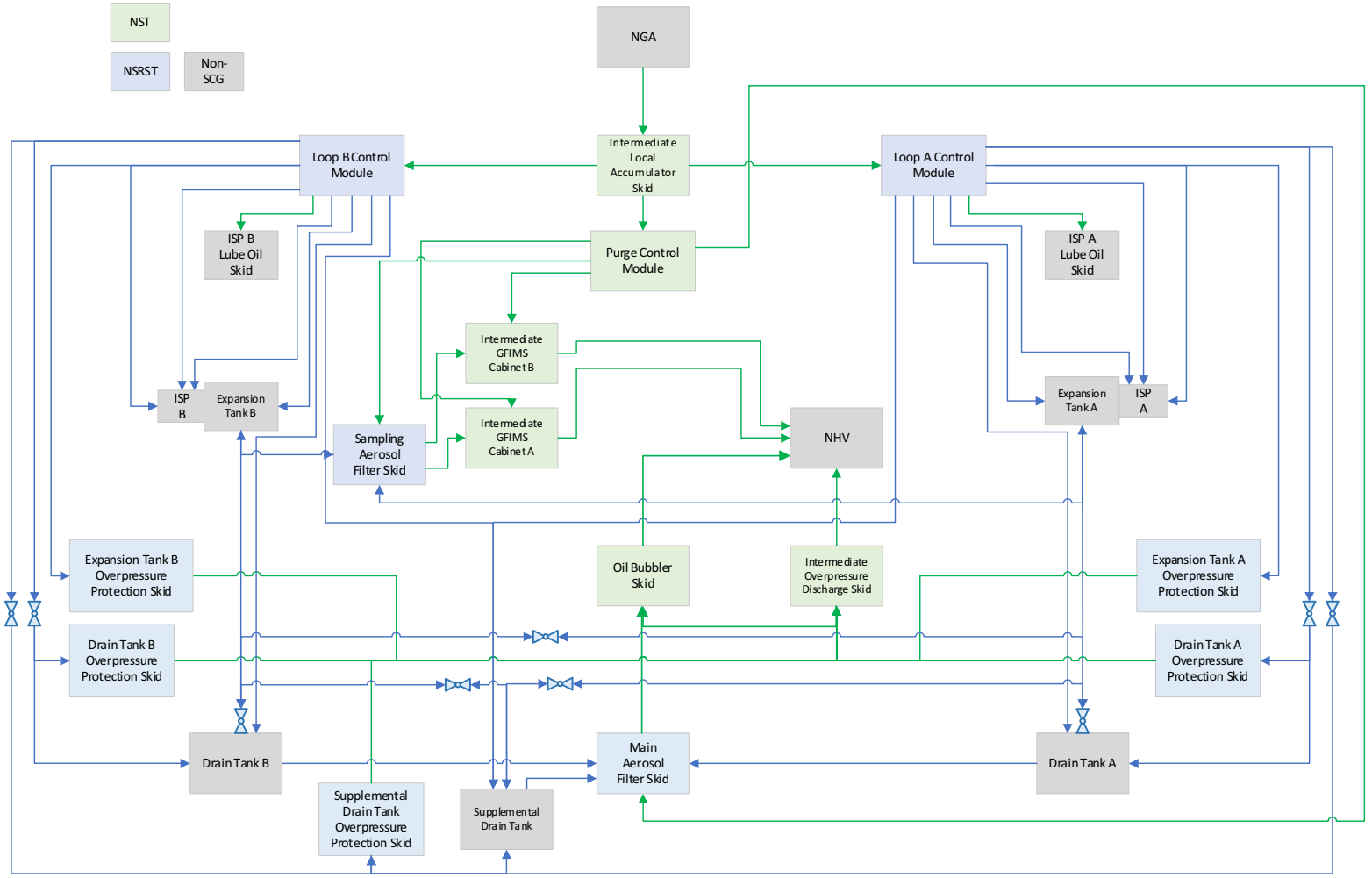
**Figure 7.2.3-2 Intermediate Cover Gas System Block Diagram**





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## 7.2.4 Sodium Processing System

### 7.2.4.1 Summary Description

The Sodium Processing System (SPS) monitors and removes the impurities of the liquid sodium inventories of the primary sodium (see Section 7.1.3), intermediate sodium (see Section 7.1.4), and Ex-Vessel Storage Tank (EVST) sodium (see Section 7.3.2). Impurities include cesium, oxygen compounds, hydrogen compounds, debris, and corrosion products.

The Primary SPS (SPS-P) loop draws primary sodium from the Reactor Vessel (RV), using a submersible electromagnetic (EM) pump within the RV, and delivers it to the sodium monitoring and purification equipment located below grade in the Reactor Auxiliary Building (RAB). The purification equipment consists of cold traps, which remove oxide and hydride impurities, and a cesium trap, which removes cesium from the sodium. Sodium is monitored for oxygen and hydrogen using a plugging temperature indicator (PTI), a multi-purpose sampler (MPS), and direct-measurement oxygen and hydrogen sensors. Cesium concentration is monitored using a radiation monitor. Sampling for sodium chemistry analysis is provided by the MPS. After analysis and purification, sodium flows back into the RV. Isolation valves are located on the supply and return lines.

The Intermediate SPS (SPS-I) loop draws intermediate sodium from each Intermediate Heat Transport System (IHT) loop, or from the IHT drain tanks, and delivers the sodium to its respective sodium monitoring and purification equipment located on grade level in the RAB. The SPS-I contains similar equipment and functions as the SPS-P with the exception that the SPS-I does not contain a cesium trap or associated radiological monitoring.

The EVST SPS (SPS-E) loop draws sodium from the EVST and delivers it to sodium monitoring and purification equipment located below grade in the Fuel Handling Building (FHB). The SPS-E contains similar equipment and functions as the SPS-I.

The monitoring and impurity removal functions of SPS are non-safety-related with no special treatment (NST) functions. Process block flow diagrams for the SPS-P, SPS-I, and SPS-E are provided in Figure 7.2.4-1, Figure 7.2.4-2, and Figure 7.2.4-3, respectively.

### 7.2.4.2 Design Basis

Compliance with regulatory criteria, identification of design criteria, and codes and standards that define the SPS safety design basis are described below.

#### 7.2.4.2.1 Compliance with Principal Design Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the SPS is described below.

Consistent with PDC 2, *Design Bases for Protection Against Natural Phenomena,* safety-significant SPS structures, systems, or components (SSCs) maintain their capability to perform their safety-significant functions during adverse conditions of natural phenomena. Safety-significant SPS SSCs are designed for protection against natural phenomena


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commensurate with their safety significance. SPS SSCs are primarily protected from adverse natural phenomena conditions by Reactor Building (RXB) (Section 7.8.1), RAB (Section 7.8.3), and FHB (Section 7.8.2) structures. Safety-significant SSCs are designed to a hazard level commensurate with the safety significance of their functions. SPS safety-significant SSCs are designed using the site-specific seismic accelerations as described in Section 6.4.1.

Consistent with PDC 3, *Fire Protection,* SPS sodium-containing equipment and components, including non-metallic components in contact with the sodium environment, are designed with non-combustible and non-fire sustaining materials. SPS fire protection is provided by the Nuclear Island Fire Protection System (NFP) (see Section 7.5.2). The SPS design incorporates features to assist in fire protection by isolating sodium from air and water. SPS isolates sodium from air and water by confining sodium in piping. SSCs designated as safety-related (SR) are designed per ASME Section III, Division 5 (Reference 7.2.4-1) to maintain isolation. SSCs designated as non-safety-related with special treatment (NSRST) are designed per ASME B.31.1 (Reference 7.2.4-2) to maintain isolation. SPS piping containing sodium in the Reactor Auxiliary Building (RAB) and FHB utilize pipe jackets with leak detection and a means to collect sodium.

Consistent with PDC 4, *Environmental and Dynamic Effects Design Bases,* the nuclear island separation, RXB, RAB, and FHB structures, and the below grade elevation of safety-significant SSCs protect the SPS from applicable external hazards caused by transportation incidents, offsite industrial and chemical facilities, and external turbine missiles.

Consistent with PDC 14, *Primary Coolant Boundary,* the primary coolant boundary components are designed, fabricated, erected, and tested to provide an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The primary coolant boundary operates at low pressure, and metallic boundary components are designed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5. Provisions are provided for leak testing primary coolant boundary components in accordance with applicable requirements of the ASME Code. Over-pressure protection of the primary coolant boundary is provided per the requirements of the ASME Code.

Consistent with PDC 15, *Primary Coolant System Design,* the design of the SPS provides sufficient margin such that the design conditions of the primary coolant boundary are not exceeded during conditions of normal operation, including AOOs. The primary coolant boundary operates at low pressure and is designed and constructed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5. Sufficient margin to design conditions is provided by the design requirements established as described in Section 6.4.3.

Consistent with PDC 16, *Containment Design,* SPS-P equipment is located below grade and within enclosures that can mitigate the radiological consequences of leakage. SPS-P piping up to and including the first isolation valve are part of the primary coolant boundary and utilize a secondary containment. SPS isolation valves are designed to shut as needed for primary containment. SPS circulation may be halted and isolated in the event of a leak to limit the inventory of sodium that can be released. SPS-I is located within the RAB. The SPS-I cold traps


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make up part of the functional containment for the control of potential radionuclides. SPS-E is located within the FHB. Portions of the SPS-E piping from the EVST up to and including the isolation valves are part of the EVST radionuclide barrier.

Consistent with PDC 30, *Quality of Primary Coolant Boundary,* the safety-related primary coolant boundary is designed, fabricated, erected, and tested to quality standards established as described in the Quality Assurance Program Description (Section 8.1). The primary coolant boundary components are designed and constructed in accordance with the requirements of ASME BPVC, Section III, Division 5. Means are provided for detecting and, to the extent practical, identifying the location of the source of primary coolant leakage. Pre-service and in-service monitoring and non-destructive examination (MANDE) requirements implemented by the Reliability and Integrity Management Program (described in Chapter 8) for the primary coolant boundary provide assurance that quality requirements are met.

Consistent with PDC 31, *Fracture Prevention of Primary Coolant Boundary,* safety-significant portions of SPS that make up parts of the primary coolant boundary are designed per the requirements of ASME Section III, Division 5. Primary coolant boundary SSCs are constructed to prevent material fractures caused by radiation embrittlement and fracture from thermal stresses. SPS SSCs maintain coolant composition within established limits by removing oxygen and hydrogen. All SPS SR boundaries, including non-ASME SSCs and non-metallic SSCs, are tested, verified, and qualified per the Equipment Qualification Program described in Chapter 8.

Consistent with PDC 32, *Inspection of Primary Coolant Boundary,* portions of SPS that make up the primary coolant boundary are subject to testing conducted per ASME Section XI, Division 2 (Reference 7.2.4-3) as invoked by ASME Section III and implemented by the Reliability and Integrity Management (RIM) Program described in Chapter 8. The SPS design supports necessary testing, inspection, and maintenance including adequate access and working space or remote means as necessary. Remote means of performing maintenance are utilized where access is limited. SPS vaults are equipped with viewing windows or cameras as needed. SPS primary coolant boundaries are monitored for sodium leaks to confirm integrity.

Consistent with PDC 33, *Primary Coolant Inventory Maintenance,* safety-significant portions of SPS that make up parts of the primary coolant boundary are designed per the requirements of ASME Section III, Division 5. Primary coolant boundary SSCs are constructed to prevent material fractures caused by radiation and thermal stresses. All SPS SR boundaries, including non-ASME SSCs and non-metallic SSCs, are designed, analyzed, tested, verified, and qualified per the Equipment Qualification Program described in Chapter 8. Coolant inventory in the RV is protected by the RPS and the Auxiliary Monitoring and Control System (AMC) (see Section 7.6), depending on the actuating signal. In the event of a low reactor sodium level, the SPS-P pump is tripped; the Reactor Protection System (RPS) provides this SR function by interrupting power to the pump. In the event of leakage in SPS-P, the AMC will automatically trip the SPS-P pump. Once the SPS-P pump trips, sodium flow stops even if the isolation valves are not closed. Because the entirety of the Primary SPS is above the primary coolant level, no siphon can form that would lower reactor level below the safe level.


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Consistent with PDC 61, *Fuel Storage and Handling and Radioactivity Control,* the SPS-E maintains the EVST sodium inventory within established limits of oxygen and hydrogen concentration by circulating the sodium through purification equipment. The SPS-E is located below grade to reduce personnel exposure in the off-normal event that the EVST sodium contain radionuclides. The SPS-E includes features to contain, collect, and detect sodium leaks, including leak detection. The SPS-E design supports necessary testing, inspection, and maintenance including adequate access and working space or remote means as necessary. Remote means of performing maintenance are utilized where access is limited. The SPS-E is not utilized to remove heat from the EVST. The SPS-E design prevents it from removing sodium below the minimum level required for EVST residual heat removal; the SPS-E is designed such that a leak in the SPS-E cannot drain, sluice or siphon the EVST below its safe level.

Consistent with PDC 63, *Monitoring Fuel and Waste Storage,* the SPS-E is designed such that a leak in the SPS-E cannot drain, sluice, or siphon the EVST below its safe level. Additionally, SPS-E isolation valves are designed to shut when they receive a control signal indicating a low EVST level. SPS is also monitored by leak detection to confirm leak tight integrity. Structural designs are seismically qualified with special treatments commensurate with the safety significance.

Consistent with PDC 70, *Intermediate Coolant System,* the SPS-I piping between the isolation valves and the IHT forms a portion of the intermediate coolant boundary. The design of the SPS-I intermediate coolant boundary components provides sufficient margin such that the design conditions of the intermediate coolant boundary are not exceeded during conditions of normal operation, including AOOs.

Consistent with PDC 71, *Primary Coolant and Cover Gas Purity Control,* the SPS removes contaminants from primary sodium to designed limits. Oxygen, hydrogen, and other dissolved species with limited low temperature sodium solubility, are removed in the cold traps. Cesium is removed in the cesium trap. Insoluble particulates are removed in a particulate filter. Sodium purity is monitored using a plugging temperature indicator. Oxygen and hydrogen are monitored using installed meters. Other impurities are monitored for by coolant sampling. Heat tracing is used to preclude sodium freezing and plugging. Interfaces with air systems are designed to prevent introduction of air into the primary coolant boundary. SPS is designed to preclude the introduction of debris and particulates from the SPS into the primary coolant sodium.

Consistent with PDC 72, *Sodium Heating Systems,* electric heat trace is used to maintain SPS sodium temperature within design limits without requiring other heat sources. SPS heating systems are designed with margin to prevent exceeding thermal design limits of SR portions of the SPS even with a single failure present. These limits include thermal stresses, heatup and cooldown limits, and thermal distribution limits. Due to limited access to some portions of SPS-P and SPS-E, redundant heat trace is used in some areas to improve reliability. Sodium freezing in this piping will not result in damage to the primary coolant boundary and therefore heat tracing on this section is not SR.

Consistent with PDC 73, *Sodium Leakage Detection and Reaction Prevention and Mitigation,* the SPS is monitored for leakage. Controls are in place to isolate SPS upon indication of leakage. SPS is designed to ASME standards to limit leakage. The impact of sodium-air and


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sodium-concrete interactions within SPS are minimized by locating SPS-P within inert cells or inert pipe enclosures. Additionally, SPS is designed such that it does not hinder the SR functions of any system during a design basis event.

Consistent with PDC 74, *Sodium/Water Reaction Prevention/Mitigation,* the SPS is monitored for leakage. Controls are in place to isolate SPS upon indication of leakage. SPS is designed to ASME standards to limit leakage. The impact of sodium-air and sodium-concrete interactions within SPS are minimized. Additionally, SPS is designed such that it does not hinder the SR functions of any system during a design basis event. Major components of SPS-P are located in inerted vaults and an inert cover gas is applied to limit contact between sodium and the moisture in air. Water sources and water containing systems are physically separated from the sodium containing portions of SPS to preclude the potential for direct water-sodium interaction.

Consistent with PDC 75, *Quality of the Intermediate Coolant Boundary,* the SPS-I piping between the isolation valves and the IHT forms a part of the intermediate coolant boundary. This portion of piping is designed in compliance with ASME codes and with seismic special treatments.

Consistent with PDC 76, *Fracture Prevention of the Intermediate Coolant Boundary,* the SPS-I is designed in compliance with ASME codes and seismic special treatments. SR portions of SPS are designed with margin to failure modes that could cause gross leakage. Additionally, SPS-I design ensures that connections to IHT do not introduce pressures, temperatures or other adverse conditions that could cause the IHT to exceed any design operating conditions.

Consistent with PDC 77, *Inspection of the Intermediate Coolant Boundary,* the SPS-I design supports necessary testing, inspection, and maintenance including adequate access and working space or remote means as necessary. Remote means of performing inspections and maintenance are utilized where access is limited. SPS is also monitored by leak detection to confirm leak tight integrity and structural designs are seismically qualified with special treatments commensurate with the safety significance.

Consistent with PDC 78, *Primary Coolant System Interfaces,* the SPS cold traps and cells are single passive barriers with nitrogen gas as the interfacing fluid. Nitrogen is chemically compatible with the sodium coolant. The cold traps and cells are isolable from the primary coolant system to limit loss of primary system inventory resulting from postulated leakage. A pressure differential between the nitrogen and the primary coolant system in these systems is not required because leakage will not result in contaminating the nonradioactive part of the heat transport system, and leakage across the interface in either direction does not result in failure of the intended SR and NSRST functions.


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7.2.4.2.2 Functional Requirements and Design Criteria

Probabilistic risk assessment safety functions (PSFs) are determined through the plant-specific probabilistic risk assessment (PRA) described in Section 3.1. Safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The SPS supports the following SR and NSRST functions:

| Function ID | Function Description | SPS Functional Support |
|---|---|---|
| DL3-HR11 | SPS Pump Trip on Low Primary Sodium Level (SR) | The SPS-P pump trips if primary sodium level in the hot pool reaches the low level setpoint. This function supports primary sodium heat removal by preventing the RV sodium from draining below its safe level. |
| DL3-RR1b | SPS Primary System Barrier (SR) | SPS-P SSCs provide a portion of the primary coolant boundary and act as a low leakage barrier to support radionuclide retention. This function is applicable to SPS components from the RES up to and including the isolation valves for SPS components that penetrate the RES. This function does not include isolation valve closure. |
| DL3-RR4 | EVST Barrier (SR) | SPS-E SSCs provide a barrier that supports radionuclide retention in the EVST. This function is applicable to the SPS-E SSCs exiting the EVST, up to and including the SPS isolation valves. This function does not include isolation valve closure. |
| DL4-HR1 | IAC Passive Mode Operation (NSRST) | The portions of the SPS-I piping connected to the IHT, between the SPS-I isolation valves and the IHT, are designed to prevent leakage or flow disruptions to ensure IAC heat removal remains available. |
| DL2-RR7 | Primary SPS Barrier (NSRST) | The SPS-P provides a passive barrier for primary sodium. This function is applicable to SPS SSCs that are not part of the primary coolant boundary but do contain primary sodium. This barrier mitigates leakage of primary sodium and retains radionuclides. |
| DL2-RR8 | Intermediate Cold Trap SPS Barrier (NSRST) | The SPS-I cold trap provides a barrier to retain sodium. This function ensures adequate containment for radionuclide accumulation in the SPS-I cold traps. |
| DL4-RR8 | Manual SPS Pump Trip on Low Primary Sodium Level (NSRST) | The SPS-P pump is manually tripped on indication of low primary sodium level as a backup to the SR trip DL3-HR11. |


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| Function ID | Function Description | SPS Functional Support |
|---|---|---|
| DL4-RR3 | SPS Cells Barrier (NSRST) | The SPS-P enclosures are designed as a low leakage barrier to contain radionuclides in the event of an SPS leak within the enclosure. |
| DL4-RR3a | SPS Supply Valve Isolation on Leak Detection (NSRST) | SPS-P piping and components are monitored for sodium leaks. On an SPS-P leak detection signal, the SPS-P supply isolation valve automatically closes. This function mitigates the severity of an SPS leak by limiting flow from the primary system. |
| DL4-RR3b | SPS Pump Trip on Leak Detection (NSRST) | SPS-P piping and components are monitored for sodium leaks. The SPS-P pumps trip on detection of a sodium leak in SPS-P. This function mitigates the severity of an SPS-P leak by limiting flow from the primary system. |
| DL4-RR3c | SPS Cells Barrier Isolation on Leak Detection (NSRST) | SPS-P piping and components are monitored for sodium leaks. On an SPS-P leak detection signal, ventilation of the SPS-P cells is automatically isolated. |
| DL5-PAM1 | Post Accident Monitoring (NSRST) | To support post accident monitoring, SPS-P isolation valves provide valve position indication and SPS-P pump breakers provide breaker position indication. |

Portions of the SPS-P SSCs are part of the primary coolant boundary. The primary coolant boundary is described in Section 5.3. The SPS ex-vessel piping that penetrates the RES, up to and including the first isolation valve, are part of the primary coolant boundary.

Sodium leak detection is provided by NFP as described in Section 7.5.2. The SPS includes sodium leak detection for SPS piping, SPS isolation valves, and SPS piping enclosures in the Head Access Area (HAA). SPS leak detection uses devices such as wire and cable, spark plugs, and mutual inductance type sensors.

SPS safety classifications are performed as described in Section 5.1. Safety-Related Design Criteria (SRDC) corresponding to the SR functions are described in Section 6.1.2 and identified in Section 5.2. The SRDC are incorporated into the design of the SPS as described in Section 7.2.4.1 and Section 7.2.4.3. Seismic design, analysis, and qualification is performed as described in Section 6.4. Seismic classifications of SSCs are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. SR SPS SSCs are classified SCS1. SSCs that are classified SCS1 are qualified to withstand seismic loads associated with the safe shutdown earthquake (SSE) (described in Section 2.6.2) without loss of the capability to perform SR functions. NSRST SPS SSCs are classified SCN1. Seismic special treatments are applied to NSRST portions of the SPS, as applicable based on seismic classification, to provide assurance that risk-significant functions or functions required for adequate defense-in-depth are capable of being performed. Major


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equipment safety classifications for SPS are identified in Figure 7.2.4-1, Figure 7.2.4-2, and Figure 7.2.4-3. These figures provide identification of general SSC classifications for SPS equipment; individual SSCs are classified as described in Section 5.1.

7.2.4.2.3 Conformance with Applicable Regulatory Guidance

Regulatory Guide 1.22, Rev. 0, 1972, “Periodic Testing of Protection System Actuation Functions (Safety Guide 22),” full conformance. The SPS-P pump breakers conform to “actuated device” and “actuated equipment” specifications as detailed in the regulatory guide.

Regulatory Guide 1.31, Rev. 4, 2013, “Control of Ferrite Content in Stainless Steel Weld Metal,” partial conformance with the exception that ASME Section III, Division 5 is used for high temperature conditions. Applicable SPS SSCs conform to this guidance which is used to supplement the use of ASME Section III.

Regulatory Guide 1.34, Rev. 1, 2011, “Control of Electroslag Weld Properties,” full conformance. This regulatory guide is applied to relevant SPS SSC welding practices when welding to ASME Section IX as invoked by ASME Section III.

Regulatory Guide 1.36, Rev. 1, 2015, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,” full conformance. This regulatory guide is applied to relevant SPS SSCs to prevent stress-corrosion cracking in austenitic stainless steel by minimizing contamination from non-metallic insulating materials.

Regulatory Guide 1.44, Rev. 1, 2011, “Control of the Processing and Use of Stainless Steel,” full conformance. This regulatory guidance is applied to relevant stainless steel components of SPS SSCs that make up: the primary coolant boundary, systems required for shutdown, systems required for emergency core cooling, and reactor internals needed for cooling.

Regulatory Guide 1.45, Rev. 1, 2008, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” partial conformance. This guidance is not directly applicable to the SPS design. Regulatory positions C.1.1, C.1.3, C.1.4, C.2.3, C.2.4 of this regulatory guide are utilized as reference information and guidance for SPS design.

Regulatory Guide 1.50, Rev. 1, 2011, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” full conformance. This regulatory guidance is applied to relevant SPS SSC welding practices when welding to ASME Section IX as invoked by ASME Section III.

Regulatory Guide 1.61, Rev. 1, 2007, “Damping Values for Seismic Design of Nuclear Power Plants,” full conformance. This regulatory guidance is applied to relevant SPS SSCs to provide acceptable damping values to be used for seismic analysis and design of SPS SSCs.

Regulatory Guide 1.84, Rev. 39, 2021, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” full conformance. This regulatory guidance is applied to relevant SPS SSCs to provide requirements that govern the use of ASME Section III code cases.


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Regulatory Guide 1.87, Rev. 2, 2023, “Acceptability of ASME Section III, Division 5, High Temperature Reactors,” full conformance. This regulatory guidance is applied to relevant SPS SSCs to provide amplifying requirements to supplement use of ASME Section III Division 5.

Regulatory Guide 1.89, Rev. 2 2023, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” full conformance. Special treatments applied to SPS-P pump breakers (i.e., Equipment Qualification Program as described in Chapter 8) conform to this regulatory guide.

Regulatory Guide 1.100, Rev. 4, 2020, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” partial conformance. SPS-P pump breakers conform to the requirements pertaining to IEEE 344 (Reference 7.2.4-4) and ASME QME-1 (Reference 7.2.4-5). Exception to the guidance is taken for components that are classified NSRST, for which codes and standards are applied in accordance with their designated special treatments. Exception is taken to the IEEE 344 revision; IEEE 344-2020 is used instead of IEEE 344-2013.

Regulatory Guide 1.141, Rev. 1, 2010, “Containment Isolation Provisions for Fluid Systems,” partial conformance. This regulatory guidance is applied to relevant SPS SSCs to provide supplemental guidance regarding SSCs that penetrate the primary confinement boundary. Portions of this regulatory guidance that supplement ANSI N271-1976 (Reference 7.2.4-6) are relevant to applicable SPS SSCs.

Regulatory Guide 1.153, Rev. 1, 1996, “Criteria for Safety Systems (12/85),” full conformance. SPS-P pump breakers comply with the requirements established in IEEE 603-1991 (Reference 7.2.4-7).

Regulatory Guide 1.189, Rev. 5, 2023, “Fire Protection for Nuclear Power Plants,” partial conformance. This regulatory guidance is applied to relevant SPS SSCs to provide instructions on the development of a fire hazard analysis, fire protection program, and safe shutdown analysis. The SPS will conform with relevant aspects of this guidance; only section 4.1 of this regulatory guide is directly applicable to the SPS design.

Regulatory Guide 1.193, Rev. 7, 2021, “ASME Code Cases Not Approved for Use,” full conformance. This regulatory guidance is applied to relevant SPS SSCs to restrict the use of ASME Section III code cases that the NRC does not endorse.

Regulatory Guide 4.21, Rev. 0, 2008, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” partial conformance. This regulatory guidance is applied to relevant SPS SSCs to provide requirements for minimizing facility contamination, minimizing contamination of the environment, facilitating decommissioning, and minimizing the generation of waste. Portions of sections 1.1-1.5, 3.1, and 4 that provide guidance on design aspects are applicable to the SPS design.


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Regulatory Guide 8.19, Rev. 1, 1979, “Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants -- Design Stage Man-Rem Estimates,” full conformance. This regulatory guidance is applied to provide amplifying instructions for estimating personnel exposure during normal operations and anticipated operational occurrences (AOOs) which can be used to inform the design of applicable SPS SSCs.

7.2.4.3 System Design and Performance Evaluation

The SPS may be operated continuously whenever monitoring of liquid sodium for contaminants or processing is required. Operation of the SPS-P and SPS-I may occur in each of the four reactor modes. Similarly, the SPS-E is capable of operating independent of the reactor modes when sodium monitoring or processing is necessary in the EVST.

The SPS monitors and controls oxygen concentration in liquid sodium inventories. Oxides can precipitate from the sodium on cold surfaces and cause plugging of small orifices. Additionally, high oxygen concentrations in sodium can cause accelerated corrosion at elevated temperatures. Oxygen may enter the sodium in the Primary Heat Transport System (PHT), IHT, and EVST by dissolution of the native oxides from steel surfaces, as well as by transport from air and moisture ingress. Oxygen may enter the IHT through in-leakage from Nuclear Island Salt System (NSS). Oxygen is removed from the system using cold traps where the sodium is cooled down to precipitate sodium oxide from solution onto high surface area packing media.

The SPS monitors and controls hydrogen concentrations in liquid sodium inventories. Sodium hydride can precipitate from solution onto cold surfaces and plug small orifices. Hydrogen may enter the sodium in the PHT, IHT, and EVST by diffusing out of steel components in contact with liquid sodium or sodium vapor, diffusion through the steel structures, and reaction with hydrogen bearing cover gas impurities such as hydrogen, water and hydrocarbons.

The SPS also reduces tritium concentrations in liquid sodium inventories. Tritium is generated from fission in the fuel and from neutron capture in the control rods. Tritium levels in the primary coolant are reduced, along with hydrogen, using the primary cold traps. Tritium from the primary coolant diffuses through the intermediate heat exchanger to the intermediate sodium. Tritium concentrations are reduced from the intermediate coolant, along with hydrogen, using the intermediate cold traps.

The SPS-P monitors and controls cesium specific activity in the primary sodium to prevent evaporation into cover gas and subsequent migration. Control of cesium specific activity within the primary sodium helps to minimize the potential for radiological exposure. Primary sodium may contain radioactive cesium isotopes from failed fuel. Cesium is removed from the primary coolant using cesium traps filled with reticulated vitreous carbon.

Maintaining liquid sodium above minimum temperatures, as needed based on operating conditions, is accomplished using heat tracing on the SPS piping.

The SPS monitoring equipment includes plugging temperature indicators, oxygen and hydrogen meters, process radiation monitors, and multipurpose samplers. The SPS impurity removal equipment includes filtration, cold traps, and cesium traps.


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The SPS-P ex-vessel piping between the RES and the first SPS isolation valve are part of the primary coolant boundary and are ASME Section III, Division 5 components. The SPS-P SSCs within the RES are not part of the primary coolant boundary. The SPS-P SSCs within the RES that are not part of the primary coolant boundary are ASME Section VIII (Reference 7.2.4-8) and ASME B31.1 components.

#### 7.2.4.3.1 Plugging Temperature Indicator

The key process indicator for sodium purity is the saturation temperature which correlates directly with impurity concentration. The plugging temperature indicator (PTI) determines the saturation temperature at which impurities, such as oxygen and hydrogen, fall out of solution in the process sodium. The SPS operates with the sodium above this saturation temperature to avoid plugging passageways in the system. The saturation temperature is also used in conjunction with the traps to adjust processing temperatures for impurity removal.

#### 7.2.4.3.2 Oxygen and Hydrogen Meters

The SPS uses oxygen meters and hydrogen meters to monitor the concentrations of oxygen and hydrogen within the liquid sodium inventories. Oxygen electrochemical meters, or a similar technology, will be used in the sodium processing systems to monitor the oxygen concentration. Dissolved hydrogen sensors which employ a probe and an ion pump, or a similar technology, will be used to monitor the hydrogen concentration.

#### 7.2.4.3.3 Process Radiation Monitors

Process radiation monitors only apply to SPS-P. The SPS-P deploys two adjacent-to-line gamma spectrometers to identify and measure radioactive isotopes in the primary sodium: one spectrometer on the main bypass line and one spectrometer on the sodium return line downstream of the cesium trap. The monitor on the bypass line measures the cesium content of the process sodium, indicating whether the cesium trap may need to be brought online to lower the cesium content in the sodium. The gamma spectrometer downstream of the cesium trap has the same measurement functionality and is used to monitor the effectiveness of the cesium trap.

#### 7.2.4.3.4 Multipurpose Sampler

The multi purpose sampler (MPS) collects sodium samples that can be sent to a laboratory for analysis of impurities. The MPS can accommodate different sampling techniques such as overflow sampling, equilibration sampling, and filtration sampling.

#### 7.2.4.3.5 SPS Filtration

The SPS uses filtration, in the form of sintered metal filters, to remove potential solid particulate matter from the sodium inventory.

#### 7.2.4.3.6 Cold Traps

Cold traps purify sodium by decreasing the oxygen and hydrogen impurity content that has been introduced into the sodium. The cold trap removes hydrogen and oxygen from the sodium via precipitation at reduced temperature.


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#### 7.2.4.3.7 Cesium Traps

Primary sodium may contain radioactive cesium isotopes from failed fuel. Cesium is removed from the primary coolant using reticulated vitreous carbon traps as part of the SPS-P process equipment.

### 7.2.4.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the SPS are applied as described in Section 7.2.3.2, Section 7.2.4.3, and Section 7.2.4.4; these include:

- ASME Section III, Division 5, 2017
- ASME Section VIII, 2021
- ASME Section XI, Division 2, 2019
- ASME B31.1, 2022
- ASME QME-1, 2023
- ANSI N271-1976
- IEEE 344-2020
- IEEE 603-1991

The following programmatic special treatments are preliminarily identified as relevant to SPS SR and NSRST SSCs, as applicable, to meet quality, design, reliability, and performance requirements (summary program descriptions are provided in Chapter 8):

- Quality Assurance Program
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- In-service Testing Program
- Equipment Qualification Program
- Post-construction Inspection, Testing, and Analysis Program (see Chapter 12)

### 7.2.4.5 Instrumentation Requirements

All SPS SR functions are passive or are executed on a loss of power such as pumps securing, and valves failing shut. No operator actions are required to support SPS SR functions. Major SPS instrumentation components are described above in Section 7.2.4.3.

## References

7.2.4-1 ASME, “Rules for Construction of Nuclear Facility Components - Division 5 - High Temperature Reactors,” ASME BPVC Section III Division 5, 2017.

7.2.4-2 ASME, “Power Piping Standard,” ASME B31.1, 2020.


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7.2.4-3 ASME, “Rules for In-service Inspection of Nuclear Reactor Facility Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Reactor Facilities,” ASME Section XI, Division 2, 2019.

7.2.4-4 IEEE, “Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations,” IEEE 344-2020.

7.2.4-5 ASME, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” ASME QME-1, 2023

7.2.4-6 ANSI, “Containment Isolation Provisions for Fluid Systems,” ANSI N271-1976.

7.2.4-7 IEEE, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-1991

7.2.4-8 ASME, “Rules for Construction of Pressure Vessels,” ASME BPVC-VIII, 2021.



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## Figure 7.2.4-1 Block Flow Diagram of the Primary SPS

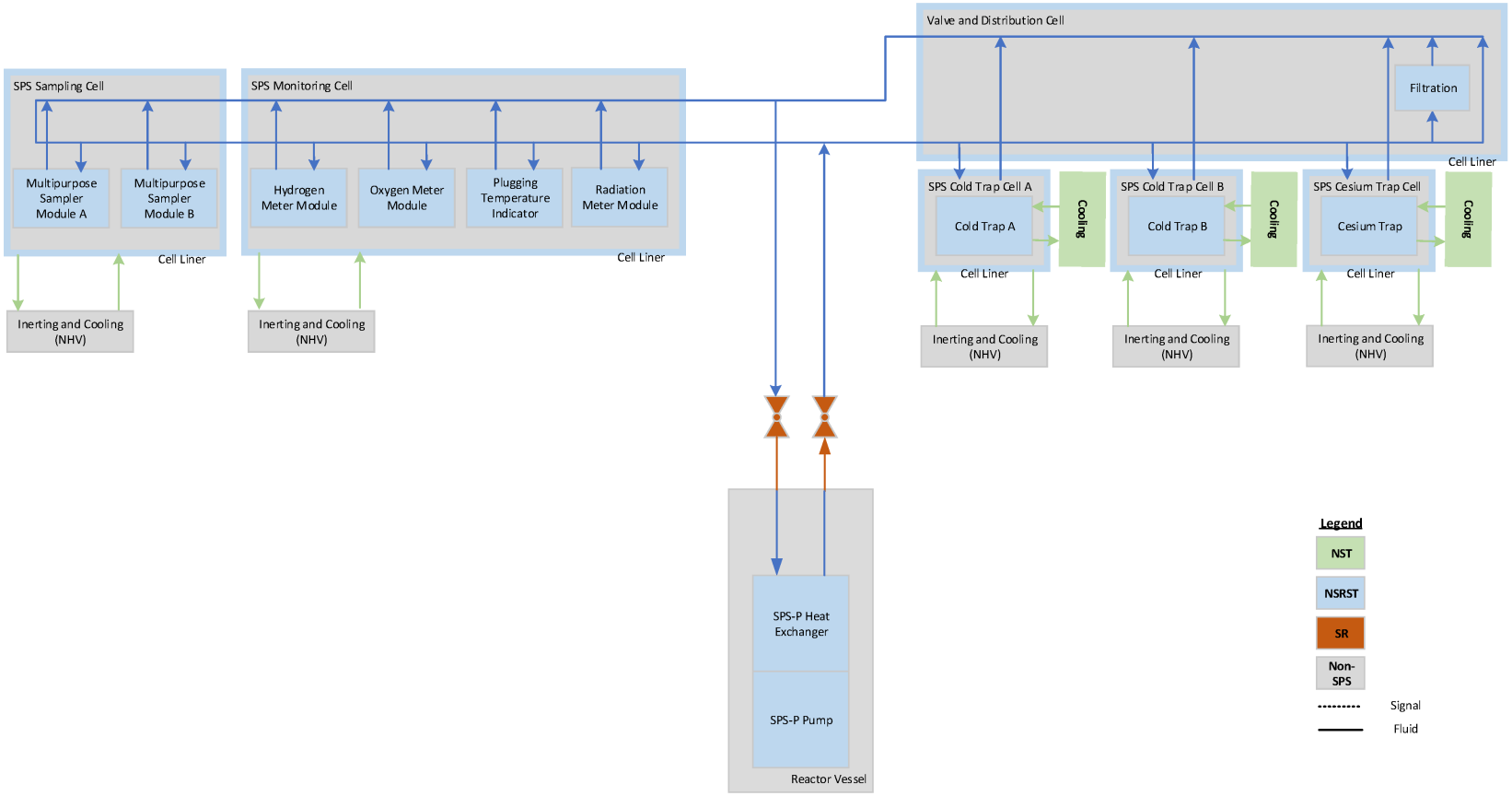



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**Figure 7.2.4-2 Block Flow Diagram of the Intermediate SPS**

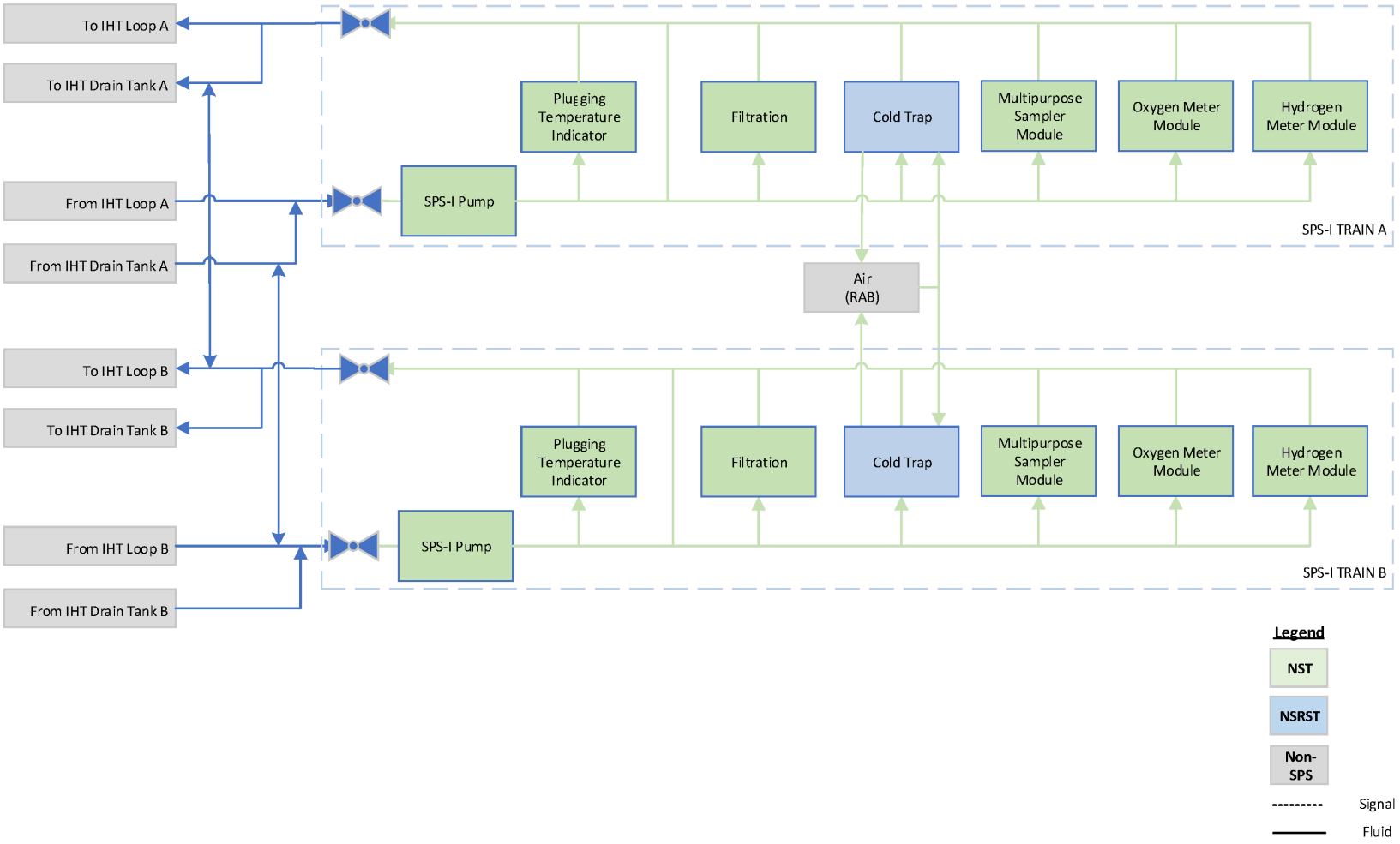



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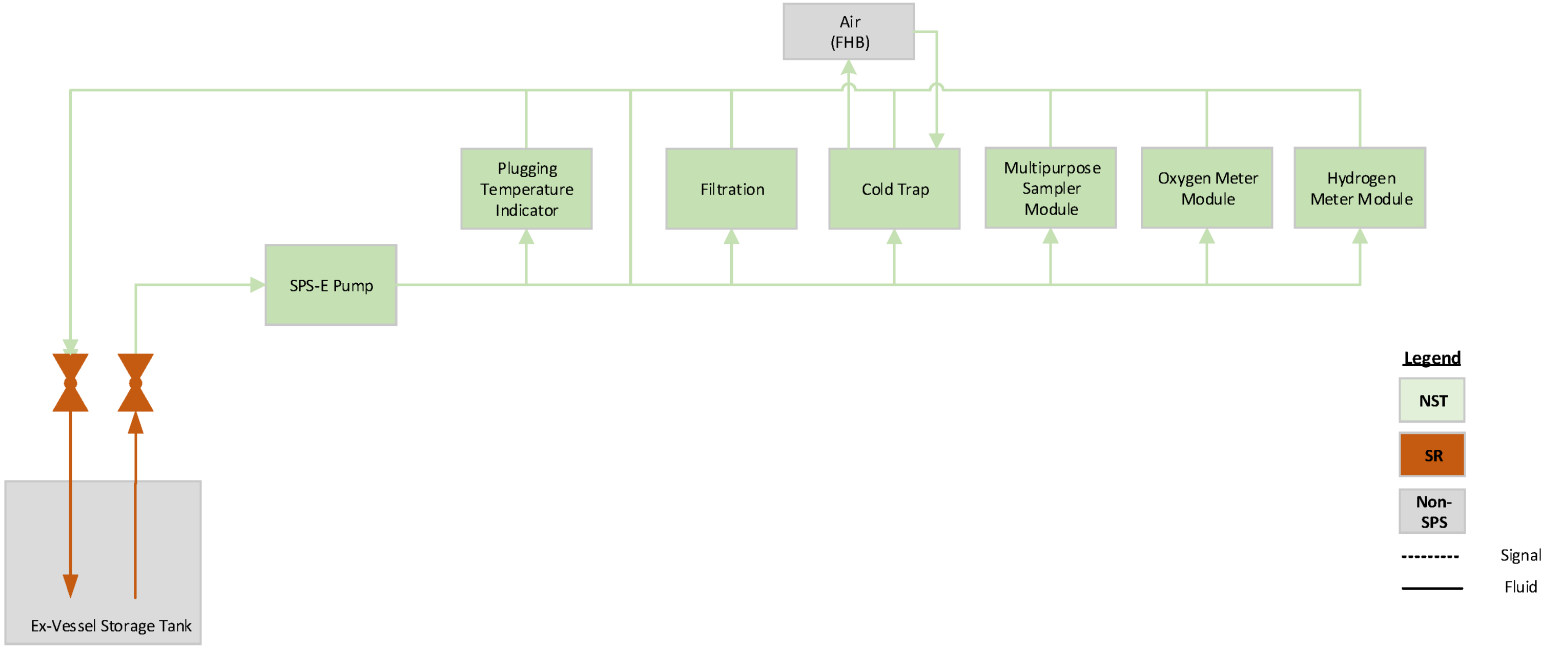
**Figure 7.2.4-3 Block Flow Diagram of the EVST SPS**





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## 7.2.5 Control Rod Drive System

### 7.2.5.1 Summary Description

The control rod drive (CRD) consists of 13 identical control rod drives, 9 of which are associated with the primary control rod assemblies and 4 associated with the secondary control rod assemblies described in Section 7.1.1. The CRD supports normal reactivity control and reactor scram functions. Portions of the CRD provide a primary coolant boundary function.

The control rod drive assembly includes the control rod drive mechanism (CRDM) and housing, motor drive assembly, motor controller, driveline assembly, scram valve and cylinder assembly, and seals. The CRD, with the exception of the motor controller, is located within the Head Access Area (HAA) of the Reactor Building (RXB) substructure described in Section 7.8.1. The motor controller is located in the Nuclear Island Control Building (NCB) substructure described in Section 7.8.4. Figure 7.2.5-1 provides the simplified CRD cross-sectional layout showing key components.

The CRD positions the control rod assemblies (CRAs) within the reactor core for reactivity control. The driveline is directly coupled to the CRA through a pneumatically-controlled latching mechanism during normal operation. The latching mechanism is designed to provide remote coupling and de-coupling of the CRA. The latching mechanism is engaged by pressurizing the scram valve and cylinder assembly with inert gas supplied by the Sodium Cover Gas System (SCG) as described in Section 7.2.3. The latching mechanism is released by venting gas pressure from the assembly. The scram valve and cylinder assembly consists of three electric solenoid-operated pilot valves that supply gas pressure to a piston and cylinder assembly. Piston motion resulting from pressurizing or depressurizing the cylinder causes the latching mechanism to engage or release the CRA. The scram valves are fail-safe, de-energize to vent, and are controlled by the Reactor Protection System (RPS) described in Section 7.6.3 Venting gas pressure de-couples the driveline from the CRA and allows CRA insertion into the core by gravity.

The CRD provides the capability to position the CRAs within the reactor core for reactor start-up, power control, power shaping, and shutdown based on demand signals from the Rod Monitoring and Control System (RMC), which is a subsystem of the Nuclear Island Control System (NIC) described in Section 7.6.2. The independent control rod drives allow positioning of individual CRAs and a fault within a single control rod drive does not affect operation of other control rod drives. Motor-driven positioning of the CRA is controlled by conversion of the rotational motion of the CRDM motor drive output shaft to linear motion of the driveline through a ball screw drive assembly within the CRDM housing. An electro-mechanical brake prevents driveline motion unless a demand signal is present. The CRDM provides control of CRA position for fine reactivity control and fast motor-driven insertion for rapid negative reactivity insertion. CRDM high speed motor-driven CRA insertion provides power run-back capability. The motor controller and drive is supplied power from the Nuclear Island Uninterruptible AC Power Supply System described in Section 7.7.1.


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The CRD supports emergency shutdown of the reactor through reactor scram capability in response to scram signals from the RPS. The RPS scram signal results in de-energizing the scram solenoid-operated valves (SOVs), which vent gas pressure within the scram cylinder and disengage the CRA from the driveline. The released CRA inserts into the core by gravity. The RPS scram signal also initiates a CRDM motor-driven driveline scram follow, whereby the CRDM motor drives the driveline downward at maximum insertion speed. Driveline scram follow provides assurance that the CRA is fully inserted and provides CRA insertion position indication.

The CRD provides indications of a coupled CRA position to the RMC through motor shaft and ball screw revolutions monitoring and a linear position transducer. CRA scram insertion is indicated by full driveline scram follow insertion position.

The CRDM housings are welded to, and structurally supported by, penetrations in the Rotatable Plug Assembly (RPA), described in Section 7.1.2. The CRDM housings form a portion of the primary coolant boundary. The CRDM housing consists of a lower housing flanged to an upper housing, and a top access plate. Metallic seals and bellows are provided to isolate reactor cover gas and aerosolized sodium from entering into the CRDM lower housing through the driveline passage. The housing is operated with an inert gas purge and static overpressure, supplied by the SCG, such that seal or bellows leakage flow is directed into the cover gas space. A continuous argon shield plug is applied to each CRD penetration through the RPA, preventing the escape of reactor cover gas. Purge gas flow monitors are provided in the SCG gas supply line to identify leakage. The top access plate provides support for the motor drive and is sealed where the motor drive shaft enters the CRDM upper housing.

#### 7.2.5.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the RES design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC applicable to the CRD is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena,* the CRD is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels (DBHLs) are described in Section 6.1.1. The CRD is located in, and protected by, the RXB substructure except for the CRDM motor controllers that are located in, and protected by, the NCB substructure. The location within these safety-related (SR) structures provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions. The CRD is designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1. CRD components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions.


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Consistent with PDC 3, *Fire protection,* the design of the CRD minimizes the probability and effect of fires and explosions. Sodium-air interactions that can result in fires and explosions are limited through the use of bellows, leak detection, and inert gas from the SCG. The CRD and its components are designed using either non-combustible or fire-resistant materials to the extent practical. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases,* the CRD is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences (AOOs), and postulated accidents. Materials selection for primary coolant boundary components consistent with the requirements of ASME BPVC Section III, Division 5 (Reference 7.2.5-1) provides assurance of compatibility with the sodium environment, including sodium aerosols, vapors, and oxidation products, throughout the design lifetime. Equipment qualification for harsh environmental conditions resulting from accident conditions is provided by the CRD component quality, design, and qualification requirements in accordance with the Equipment Qualification Program described in Chapter 8.

The location of the CRD within the RXB and NCB substructures provides protection from external hazards caused by transportation incidents and offsite industrial and chemical facilities. The location also provides protection from dynamic effects including missiles and events outside the nuclear island, in accordance with the methods described in Section 6.1.1.

Consistent with PDC 12, *Suppression of reactor power oscillations*, the design of the CRD contributes to limiting power oscillations that could result in conditions exceeding specified acceptable system radionuclide release design limits (SARRDLs). The CRD provides power oscillation control by positioning of CRAs in response to control signals from the RMC and insertion in response to scram signal from the RPS if reactor scram setpoints are exceeded.

Consistent with PDC 14, *Primary coolant boundary*, the CRD primary coolant boundary components are designed, fabricated, erected, and tested to provide an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The primary coolant boundary operates at low pressure, and boundary components are designed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5. Provisions are provided for leak testing primary coolant boundary components in accordance with applicable requirements of the ASME Code. The CRDM housing is constructed of austenitic stainless steel, which further limits the potential for material fractures caused by radiation and thermal stresses. Over-pressure protection of the primary coolant boundary is provided per the requirements of the ASME Code.

Consistent with PDC 15, *Primary coolant system design,* the design of the CRD provides sufficient margin such that the design conditions of the primary coolant boundary are not exceeded during conditions of normal operation, including AOOs. The primary coolant boundary operates at low pressure and is designed and constructed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5.


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Consistent with PDC 16, *Containment design,* portions of the CRD that are part of the primary coolant boundary provide a barrier to control the release of radioactivity to the environment. The CRDM housing, which is part of the primary system coolant boundary, is designed in accordance with applicable ASME BPVC Section III, Division 5 standards to assure high reliability of this barrier. Radiation and thermal shielding, the CRD shield plug, and bellows reduce radiation and temperature levels in the CRDM internals, as well as prevent reactor cover gas from entering the CRDM housing. A continuous argon shield plug purge flows from the CRDM into the cover gas region below the RPA, preventing the flow of argon, sodium vapor, and radionuclide gases up through the CRDM penetration.

Consistent with PDC 26, *Reactivity control system,* the CRD provides reactivity control by:

- Rapidly shutting down the reactor on a scram signal from the RPS during postulated accidents to assure safe shutdown is achieved. This is accomplished by the CRDMs releasing the CRAs for gravity-driven insertion into the core.
- Inserting and withdrawing CRAs as required for normal operations using the motor-driven function of the CRDM.
- Diverse means of reactivity control through the nine primary control rod assemblies and four secondary control rod assemblies. The primary control rods have a different pin bundle geometry than the secondary control rods to prevent a common mode failure of binding between the CRAs and the reactor core assembly during a gravity-driven scram.

The CRD inserts the CRAs either by gravity scram or via the motor-driven driveline scram follow function. Upon a valid scram signal, the RPS sends a signal to scram the CRAs and a separate signal to the motor controller for a driveline scram insert as a scram follow action to ensure the CRAs are inserted.

Consistent with PDC 28, *Reactivity limits,* the CRD scrams on a scram signal from the RPS using gravity to insert the CRAs to assure rapid shutdown of the reactor to prevent damage that could result in uncontrolled reactivity changes. The CRD limits the maximum rod withdrawal speed to control the reactivity insertion rate and provides braking features that allow the system to hold a CRA at a selected position within the stroke of travel, preventing inadvertent insertion or withdrawal. A CRD control interlock provided by the RMC prevents withdrawal of more than one rod at a time. The CRD performs a power run-back by motor-driven insertion of CRAs upon command from the RMC prior to reaching scram setpoints to limit thermal stresses on the plant.

Consistent with PDC 29, *Protection against anticipated operational occurrences,* redundant features of the CRD provide assurance that reactivity control systems accomplish their safety functions in the event of AOOs. The CRD has a total of 13 CRDMs that release the CRAs for gravity-driven insertion on a scram signal from the RPS. Each CRDM works independently to assure that the failure of one CRDM does not inhibit the safety function of another CRDM. Each CRDM has three scram SOVs. Only two of the three scram SOVs are required for scram function. Therefore, a failure of one scram SOV does not inhibit the safety function to scram, or lead to a spurious scram.


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Consistent with PDC 30, *Quality of primary coolant boundary,* the CRDM housing is part of the primary coolant boundary, and therefore is designed, fabricated, erected, and tested in accordance with quality standards commensurate with the safety significance of its performed function. CRD primary coolant boundary components are designed to ASME BPVC Section III, Division 5.

Consistent with PDC 31, *Fracture prevention of primary coolant boundary*, the primary coolant boundary components of the CRD are designed with sufficient margin to provide assurance that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. CRD primary coolant boundary metallic components are designed to satisfy code stress limits under service level conditions and bounding load combinations per the requirements of ASME BPVC Section III, Division 5, as applicable. Selection of CRD primary coolant boundary materials of construction considers material compatibility with a sodium environment. The CRD primary coolant boundary component design and material selection are made with consideration of service temperatures, service degradation of material properties, and applicable degradation mechanisms. Metallic primary coolant boundary components are constructed of austenitic stainless steel to minimize the effects of irradiation embrittlement and the potential for fracture from thermal stresses. The Degradation Mechanism Assessment is performed in accordance with the Reliability and Integrity Management (RIM) Program as described in Chapter 8, which meets the requirements of ASME BPVC Section XI, Division 2 (Reference 7.2.5-2) as endorsed with exceptions and clarifications by Regulatory Guide (RG) 1.246, Revision 0.

Consistent with PDC 32, *Inspection of primary coolant boundary*, the design of the CRD permits periodic inspection and functional testing of important areas and features to assess structural and leaktight integrity through implementation of the monitoring and non-destructive examination (MANDE) pre-service and in-service inspection requirements of the RIM Program.

Consistent with PDC 80, *Reactor vessel and reactor system structural design basis,* integrity of the CRD is maintained to assure that during postulated accidents, control rod absorbers are inserted to provide reactor shutdown. CRDMs are mounted to, and are structurally supported by, the RPA. The driveline penetrates the RPA such that the CRDM housing forms part of the primary coolant boundary. The CRDM housing is designed according to ASME BPVC, Section III, Division 5.

Functional Requirements and Design Criteria

Probabilistic risk assessment safety functions (PSFs) are determined through the plant-specific probabilistic risk assessment (PRA) described in Section 3.1. The safety-significant PSFs are identified as SR or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The CRD supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classifications of CRD components relied upon to perform the function.


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| Function ID | Function Description | CRD Functional Support |
|---|---|---|
| DL3-RC1, DL3-RC2 | Scram – Gravity-driven absorber insertion by latch release (SR) | The CRD releases the CRAs for gravity insertion on an automatic scram signal or loss of power. |
| DL4-RC3 | CRD Driveline Scram Follow (NSRST) | The CRD provides motor-driven CRA insertion on a driveline scram follow demand signal and provides indication of full insertion of CRAs. |
| DL3-RR1 | Primary Coolant Boundary (SR) | The CRDM housing forms part of the primary coolant boundary. |

Safety classifications for CRD SSCs are listed in Table 7.2.5-1. Safety-Related Design Criteria (SRDC) corresponding to the SR functions applicable to the CRD are derived as described in Section 6.1.2 and identified in Section 5.2. The SRDC are incorporated into the design of the CRD as described in Section 7.2.5.3.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the CRD, along with a description of the use and conformance of the guidance, includes:

RG 1.20, Revision 4, *Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing*

Partial Conformance. The CRD lower guide tube is within the scope of the Comprehensive Vibration Assessment Program (CVAP). SSCs within the reactor vessel are classified as prototypes in accordance with RG 1.20 Section C.1. SSCs within the reactor vessel are screened and analyzed for effects due to potential excitation mechanisms in accordance with RG 1.20 Section C.2.

RG 1.31, Revision 4, *Control of Ferrite Content in Stainless Steel Weld Metal*

Partial Conformance. Limits are imposed on the maximum amount of delta ferrite when welding austenitic stainless steel in ASME BPVC Section III, Division 5 SSCs with design temperatures exceeding 797 degrees F (425 degrees C) to prevent degradation of material properties of safety-significant CRD SSCs over time.

RG 1.44, Revision 1, *Control of the Processing and Use of Stainless Steel*

Partial Conformance. Regulatory Position C.1, criterion for cleanliness, is implemented in fabrication specifications for all CRD austenitic stainless steel, safety-significant SSCs.

RG 1.87, Revision 2, *Acceptability of ASME Code, Section III, Division 5, High Temperature Reactors*

Full Conformance. The CRD primary system coolant boundary components are designed, fabricated, and erected according to ASME BPVC Section III, Division 5 to provide a low probability of failure during normal operation or postulated accident conditions.


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RG 1.100, Revision 4, *Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants*

Partial Conformance. SR electrical and mechanical CRDM and CRD driveline assemblies are qualified in accordance with ASME QME-1 or IEEE 323-2003 (Reference 7.2.5-3).

Seismic Classification and Design Codes

Seismic classifications for CRD components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The CRD seismic classification for SR components is SCS1 and for NSRST components is SCN1. Seismic design, analysis, and qualification are performed as described in Section 6.4. CRD components classified SCS1 and SCN1 are qualified to withstand seismic loads associated with the safe shutdown earthquake, as described in Section 2.6.2, without loss of the capability to perform safety-significant functions.

Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of CRD components as special treatments, as described in Section 6.3. Codes and standards that are applicable to the CRD are listed in Table 7.2.5-2.

The methods used for development of CRDM loads, load combinations, and stress limits are described in Section 6.4.4.

### 7.2.5.3 System Design and Performance Evaluation

The CRD is designed to meet design basis requirements described in Section 7.2.5.2 and evaluated to provide assurance that associated performance requirements are met. Design features are provided by the CRD design to support the identified safety-significant functions and to provide assurance of compliance with applicable regulatory criteria.

The safety-significant portions of the CRD are protected from design basis hazards and associated DBHLs described in Section 6.1.1. Protection from tornadoes, high-winds and wind-driven missiles, extreme climatic conditions, and external fires is provided by the location within the hardened RXB and NCB substructures. External flooding and extreme precipitation protection for the site SR structures, including the CRD location within the RXB and NCB, is described in Section 6.1.1.

The CRD components are qualified over the range of service conditions for functionality during normal operations and accident conditions. The CRDM contains a series of bellows that serve to limit sodium cover gas, aerosolized sodium, and radionuclides entering the CRDM housing. SR components within the CRDM housing are maintained in an argon-filled, inert environment. Each CRD driveline penetration through the RPA is provided with a bellows assembly and seal configuration that reduces the operating temperature of the CRDM. Radiation shielding and thermal shielding within the CRDM protect the internal components, and also contribute to radiation and temperature reduction in the HAA. The CRDMs are cooled by the ambient environment of the HAA. Preliminary design conditions for the CRD components are provided in Table 7.2.5-3.


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Materials of construction for primary coolant boundary components are selected to be consistent with the requirements of ASME BPVC Section III, Division 5 to provide adequate material quality requirements and assure compatibility with the high-temperature operating environment. Materials are selected based on fracture toughness, creep strength and resistance, corrosion resistance, and thermal aging resistance considerations. The selection of austenitic stainless steel for fabrication of metallic pressure boundary components provides assurance of adequate resistance to stress and thermal factors in the high temperature operating environment. Preliminary selection of materials of construction for CRD components is provided in Table 7.2.5-4.

The SR CRD scram is a fail-safe function that is performed by de-energizing the scram valves, which vent the scram cylinder and release the CRAs to insert by gravity into the reactor core. The scram valve and cylinder assembly is located within the CRDM housing and is not exposed to the primary sodium environment. A scram is initiated upon receiving two out of four scram signals from the RPS. Actuation of two out of the three scram valves is sufficient to vent the scram cylinder gas pressure and release the CRA, such that failure of a single valve to vent does not prevent the scram function. The scram valves vent the scram cylinder on loss of power to the SOV.

### 7.2.5.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the CRD is described in Section 7.2.5.2. The following programmatic special treatments are preliminarily identified as applicable to the CRD to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8. The Post-Construction Inspection, Testing, and Analysis Program is described in Chapter 12.

- Quality Assurance Program
- Design Reliability Assurance Program
- Equipment Qualification Program
- Post-Construction Inspection, Testing, and Analysis Program
- Comprehensive Vibration Assessment Program
- Reliability and Integrity Management Program

### 7.2.5.5 Instrumentation Requirements

The CRDM has instrumentation to support equipment monitoring, predictive maintenance, and identification of failed components or assemblies. These include CRA position indication which interfaces with the RMC, and scram valve position indication and thermocouples which interface with the NIC.

## References

7.2.5-1 ASME, “Rules for Construction of Nuclear Facility Components – High Temperature Reactors,” ASME BPVC, Section III, Division 5, 2017.


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7.2.5-2 ASME, “Rules for Inservice Inspection of Nuclear Power Plant Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Reactor Facilities,” ASME BPVC Section XI, Division 2, 2019.

7.2.5-3 IEEE, “Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” IEEE 323, 2003.

7.2.5-4 IEEE, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603, 1991.

7.2.5-5 ASME, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” QME-1, 2023.


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**Table 7.2.5-1 CRD Safety and Seismic Classifications**

| Component | Safety Classification |
|---|---|
| Motor Drive Assembly | NSRST |
| Scram Valve and Cylinder Assembly | SR |
| Driveline Assembly | SR |
| CRDM Housing Assembly | SR |



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**Table 7.2.5-2 Applicable Codes and Standards**

| Standard | Edition | Title |
|---|---|---|
| ASME BPVC, Section III, Division 5 (Reference 7.2.5-1) | 2017 | Rules for Construction of Nuclear Facility Components - High Temperature Reactors |
| IEEE 323 (Reference 7.2.5-3) | 2003 | Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations |
| IEEE 603 (Reference 7.2.5-4) | 1991 | Standard Criteria for Safety Systems for Nuclear Power Generating Stations |
| QME-1 (Reference 7.2.5-5) | 2023 | Qualification of Active Mechanical Equipment Used in Nuclear Facilities |


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**Table 7.2.5-3 Preliminary CRD Design Conditions**

| Component | Design Temperature (°F) | Design Pressure (psig) |
|---|---|---|
| CRDM Housing | 150 (upper housing in HAA)<br>450 (near RPA penetration) | 30 (upper housing)<br>60 (lower housing) |
| Driveline Assembly | 1080 | 80 |
| Scram Valve and Cylinder Assembly | 400 | 300 |



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**Table 7.2.5-4 Preliminary CRD Materials of Construction**

| Component | Material |
|---|---|
| CRDM Housing | Type 316 Stainless Steel |
| Driveline Assembly | Type 316 Stainless Steel<br>Alloy 718 |
| Scram Valve and Cylinder Assembly | Type 304 Stainless Steel |



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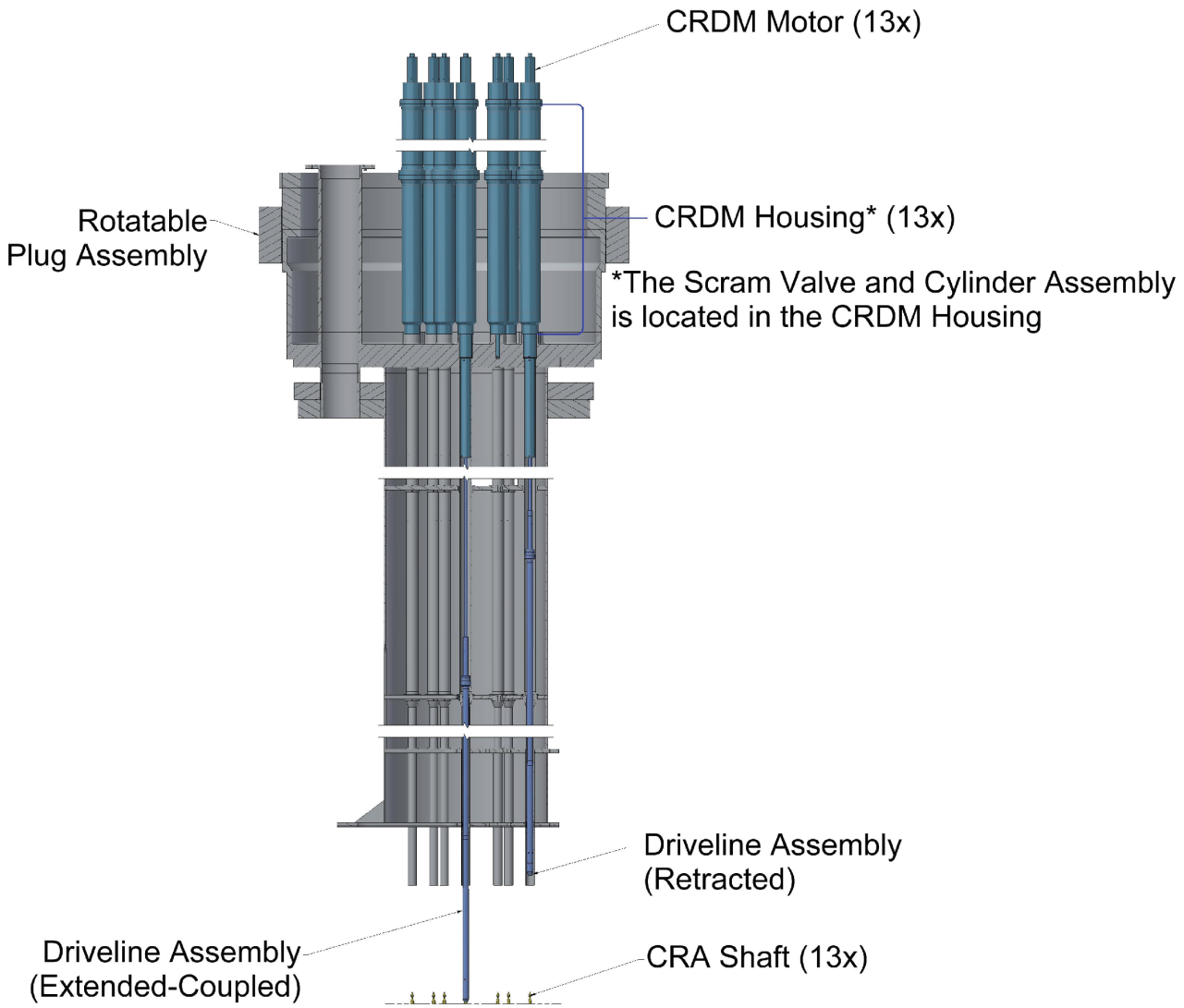
**Figure 7.2.5-1 CRD/RPA Cross-Sectional Layout**





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## 7.3 Fuel Handling Systems

Fuel handling activities are supported by the Ex-Vessel Fuel Handling System described in Section 7.3.2, the In-Vessel Fuel Handling System described in Section 7.3.3, and the Water Pool Fuel Handling System (FHP) described in this section. These systems encompass the fuel handling operations from the reactor core to the SFP.

### 7.3.1 Water Pool Fuel Handling System

#### 7.3.1.1 Summary Description

The FHP provides a water-based environment for the intermediate-term storage of spent fuel, and is used for the storage of non-fuel irradiated core assemblies prior to disposal. The FHP provides core assembly cooling, spent fuel reactivity control, fission product containment, and radiation shielding. The FHP is divided into four subsystems: Spent Fuel Pool (SFP), Pool Immersion Cell (PIC), Fuel Pool Cooling (FPC), and Fuel Pool Purification (FPP).

Spent Fuel Pool

The SFP is a water-filled, concrete pool located below-grade within the Fuel Handling Building (FHB) substructure that contains fuel storage racks for reactor core assembly storage. The concrete SFP structure is described in Section 7.8.2. The SFP includes a stainless steel liner to minimize leakage from the pool and a leak collection and detection system. The SFP design includes provisions for normal makeup water supply, emergency makeup supply via an exterior fill leg, pool cooling, and water purification.

The SFP has three operational zones as shown in Figure 7.3.1-1: the PIC cavity, the cask loading pit, and the fuel storage area. Figure 7.3.1-2 provides a representative elevation view of the SFP.

The fuel storage area contains seven storage racks for core assembly storage (intact fuel racks) and one storage rack for failed fuel canister (FFC) storage (failed fuel rack). The storage racks are designed to promote natural circulation of pool water to cool the core assemblies. Since the SFP does not employ soluble boron, the storage racks maintain stored spent fuel subcritical by controlling the relative spacing within the racks (geometry control) and the installation of fixed borated neutron absorbers within the assembly storage cells. The storage racks do not rely upon specific zoning of fuel assemblies to maintain subcriticality. The spent fuel pool criticality analysis is described in Section 3.14.

Core assembly handling within the SFP is performed with the pool handling machine (PHM) gantry-and-hoist style crane. The PHM can retrieve core assemblies from the cask loading pit and transfer each core assembly underwater to the corresponding rack in the fuel storage area. The PHM includes viewing for navigation, selection of core assemblies for manipulation, and features to maintain minimum shielding requirements. The PHM uses handling tools to manipulate core assemblies. The PHM is used to perform inspections and maintenance in the pool. The operational area of the PHM is the fuel storage area and the cask loading pit.


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Pool Immersion Cell

The PIC is the system that transitions core assemblies being stored in a sodium environment to a water-filled environment.

The Ex-Vessel Fuel Handling System transfers core assemblies into the PIC from the bottom loading transfer cask (BLTC). The PIC is the entrance to the SFP and is located in the corner of the SFP as shown in Figure 7.3.1-1. Core assemblies are loaded into a cleaning vessel that provides a controlled environment within the PIC to chemically clean and remove residual sodium coolant. The effluents from chemical processing are processed through the PIC processing skid where they are separated. Gaseous effluent is filtered within the processing skid and directed to the Nuclear Island (NI) Heating, Ventilation, and Air Conditioning System, as described in Section 7.5.1, while liquid effluent is transferred to the FPP. The cleaning vessel isolates core assemblies from the surrounding environment to prevent sodium-water interactions. During the cleaning process, heat is passively transferred through the cleaning vessel to the SFP water. Following cleaning, the core assembly is rinsed and the PIC transports the core assembly underwater to the cask loading pit area of the SFP.

The PIC transports core assemblies individually, in a vertical orientation, from the upper pool area to the pool floor, via an elevator. Once the core assembly is at the pool floor level, the PIC uses a horizontal trolley to transfer the core assembly upright to the cask loading pit area. From there, the core assembly is accessible to the PHM and is transported to the corresponding spent fuel storage rack.

An additional design feature of the PIC is to encapsulate a fuel assembly with indicated fuel pin cladding damage inside an FFC that is inerted with helium. The PIC is loaded with an FFC to receive the failed fuel assembly. No sodium cleaning occurs for a failed fuel assembly; the assembly is placed directly within the FFC from the BLTC. The FFC isolates the leaking assembly from the external environment. Once the canisterization process is complete, the FFC is transferred to the SFP and stored in the failed fuel rack. The FFC is designed to prevent metallic uranium-water interactions and sodium-water reactions that can occur due to cladding failure in the sodium-bonded fuel.

Fuel Pool Cooling

The FPC is classified as non-safety-related with no special treatment.

The FPC is the normal cooling system for the SFP. The FPC is designed to maintain the SFP water temperature within prescribed limits by transferring decay heat generated from core assemblies to the NI Water System. The FPC consists of two independent cooling trains that transfer heat to the NI Water System. Each train is sized to remove the entire heat load of the SFP. The suction and discharge lines of the FPC are located to prevent draining of the SFP below a minimum safe level in the event of a FPC leak or rupture.

Fuel Pool Purification

The FPP is classified as non-safety-related with no special treatment.


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Filtration and purification of the SFP water is provided by the FPP. The FPP is designed to purify SFP water directly from the SFP, and process the PIC effluent water prior to the return to the SFP. The FPP consists of a single purification train that uses a combination of anion and cation exchange resins to maintain pool chemistry within a specified range. The suction and discharge lines of the FPP are located to prevent draining of the SFP below a minimum safe level in the event of a FPP leak or rupture.

#### 7.3.1.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the FHP design basis are described below.

Regulatory Criteria

Principal design criteria are identified in Section 5.3. Compliance with Principal Design Criteria (PDC) that are applicable to the FHP is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the FHP structures, systems, and components (SSCs) are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of external hazards and natural phenomena, to maintain capability to perform safety-significant functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels are described in Section 6.1.1. The handling equipment, storage racks, and SFP structure are designed to withstand earthquakes using methods and criteria described in Section 6.4.1. The FHP is protected by the FHB, described in Section 7.8.2, from external flooding and precipitation hazards. The SFP is designed to account for the effects of extreme winds and wind-related missiles. The operation of safety-significant FHP equipment is administratively controlled to prevent operation during extreme wind events.

Consistent with PDC 3, *Fire Protection*, the FHP core assembly handling equipment and cleaning process are designed to contain assembly residual sodium to limit the potential for fires. The FHP is located within the FHB in dedicated spaces for core assembly handling and storage activities to minimize the potential impact to other safety-significant SSCs in the event of a fire. The FHP equipment is constructed of noncombustible and non-flammable materials to the extent practical. Fire detection and protection is provided by the NI Fire Protection System, as described in Section 7.5.2.

Consistent with PDC 4, *Environment and dynamic effects design bases*, the FHP SSCs are designed to accommodate the effects of, and to be compatible with applicable environmental conditions of spent fuel decay heat and radiation, liquid sodium and it's aerosols, vapors, and oxidation products. Equipment qualification for harsh environmental conditions is provided by the component quality, design, and qualification requirements in accordance with the Equipment Qualification Program described in Chapter 8. The FHP is protected against dynamic effects, including the effects of discharging fluids that may result from equipment failures and external events. The FHP SSCs are designed using materials that are selected to be capable of withstanding the operating environments to which they are exposed.


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Consistent with PDC 61, *Fuel storage and handling and radioactivity control*, the design of the FHP provides adequate SFP water inventory for stored fuel residual heat removal and provides suitable shielding for radiation protection under normal and postulated accident conditions. FHP fuel storage and handling equipment provides safe containment and confinement of core assemblies and is designed to permit periodic inspection and testing to assure safety-significant functional capability.

Consistent with PDC 62, *Prevention of criticality in fuel storage and handling*, criticality in the fuel storage and handling system is prevented in accordance with 10 CFR 50.68(b) and the criticality analysis is described in Section 3.14. Criticality in the SFP fuel storage area is prevented by fixed borated stainless steel neutron absorber plates in the fuel region of the storage racks and by having a subcritical geometry configuration that maintains criticality safety within the fuel storage areas during normal and accident conditions.

Consistent with PDC 63, *Monitoring fuel and waste storage*, SFP level and temperature instrumentation and FHB radiation monitoring is provided to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions.

Consistent with PDC 74, *Sodium and water reaction prevention and mitigation*, the FHP is designed to remove sodium from core assemblies in an inert environment within the PIC prior to immersion in the SFP.

Functional Requirements and Design Criteria

Probabilistic risk assessment (PRA) safety functions are determined through the plant-specific PRA described in Section 3.1. The safety-significant PRA safety functions are identified as SR functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The FHP supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classification of FHP components relied upon to perform the function:


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| Function ID | Function Description | FHP Functional Support |
|---|---|---|
| DL3-HR10 | Passive Heat Removal in FHP (SR) | Core assemblies (including FFCs in failed fuel racks) are normally cooled by FPC; however, if normal cooling is unavailable then core assemblies are passively cooled by natural circulation within the SFP (i.e., pool water is the medium that dissipates decay heat.) ensuring core assemblies are continuously cooled to maintain fuel integrity.<br><br>Core assemblies in the PIC are cooled via natural circulation of the SFP water. The PIC cleaning vessel is designed to provide passive heat transfer from a core assembly to the SFP water. |
| DL3-HR13 | Passive Heat Removal in the FFC (SR) | Core assemblies within an FFC are cooled passively via cold wall conduction from the FFC to the SFP. The water volume of the SFP acts as a heat sink to passively cool the FFCs in the PIC, during the transfer process to the failed fuel rack, and provide cooling to the FFCs stored within the failed fuel rack. |
| DL3-RR8 | Failed Fuel Canister Barrier (SR) | The FFC forms a radionuclide confinement boundary for an enclosed fuel assembly. Once sealed, the FFC prevents release of radionuclides during normal and off normal conditions. Additionally, the FFC prevents sodium-water interaction between the fuel assembly and the SFP water inventory. |
| DL3-RR5a | BLTC Transfer Barrier (SR) | During a core assembly transfer from the BLTC to the PIC, the BLTC barrier provides a containment barrier. The PIC provides a radionuclide boundary in concert with the BLTC during a fuel drop from the BLTC. The functional containment is maintained by the BLTC, the PIC Receiver, the cleaning vessel, and the Process Skid. |


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| Function ID | Function Description | FHP Functional Support |
|---|---|---|
| DL4-HR4 | FHP Emergency Makeup (NSRST) | Emergency makeup water for the SFP is provided from a connection external to the FHB. No powered SSCs are required to provide emergency makeup water via the exterior fill leg. The piping is designed with unobstructed flow to the SFP and allows water delivery methods to the SFP as described in Section 7.3.1.3. |
| DL2-RR2 | PIC Radionuclide Boundary (NSRST) | The PIC components (PIC Receiver, cleaning vessel, and process skid) form a radionuclide boundary and act as a functional containment between the core assembly and the FHB environment when a core assembly is in the PIC during anticipated operational occurrences. |
| DL5-PAM1 | Post Accident Monitoring (NSRST) | The FHP system provides instrument signals for use in PAM as described in Section 7.3.1.5. Instrument sensors that have been identified as PAM variables (i.e., SFP water level and temperature) receive additional treatment to ensure they are hardened and reliable. |

Structural Support (SR) – Load Path for SR SSCs

The intact and failed fuel racks are designed to provide structural support for core assemblies as described in Section 7.3.1.3. The load path from the intact and failed fuel rack continues through the SFP liner and into the SR FHB substructure as described in Section 7.8.2. The failed fuel canister provides support to failed fuel assemblies.

Temporary Load Path for SR SSCs (NSRST)

The PHM and PIC supports the weight of core assemblies and FFCs during fuel movement activities that are associated with DL3-RR2, DL3-RR5a, DL3-RR8, and DL3-HR13. The PHM and PIC provides structural support for SR SSCs and is described in Section 7.3.1.3.

External Hazard Protection

The intact fuel racks and failed fuel rack in conjunction with the SFP water inventory provide protection from wind related missiles for SR core assemblies stored in the SFP.

Safety classifications for FHP systems and components are listed in Table 7.3.1-1. SR design criteria corresponding to the SR functions applicable to the FHP are derived as described in Section 6.1.2 and identified in Section 5.2. The SR design criteria are incorporated into the design of the FHP as described in Section 7.3.1.3 and Section 7.3.1.1.


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Regulatory Guidance

FHP SSCs conform to the following regulatory guides:

RG 1.13, *Spent Fuel Storage Facility Design Basis*, Revision 2

Partial Conformance. The normal makeup is not Seismic Category I. The design load for the heat sink is not at least 0.3% reactor thermal power as the design precludes a full core offload into the SFP. Each exception is supported by analysis indicating the SFP can passively cool for more than 7 days before operator action is required.

RG 1.240, *Fresh and Spent Fuel Pool Criticality Analyses*, Revision 0

Full conformance. The criticality analysis, as described in Section 3.14, is in accordance with RG 1.240.

RG 1.97, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*, Revision 5

Partial Conformance. The guidance in RG 1.97 is used to fulfill the requirements set forth in 10 CFR 50.155(e). SFP level and temperature indications are identified as PAM variables.

Seismic Classification and Design Codes

Seismic classification for FHP components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied, based on the seismic classification, to provide assurance that risk-significant functions or functions required for adequate defense-in-depth are capable of being performed. The seismic classification of FHP components are tabulated in Table 7.3.1-1. Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of FHP components as special treatments, as described in Section 6.3. Codes and standards that are applicable to the FHP are listed in Table 7.3.1-2.

### 7.3.1.3 System Design and Performance Evaluation

The FHP is designed to meet design basis requirements described in Section 7.3.1.2 and evaluated to provide assurance that associated performance requirements are met. Design features are provided by the FHP design that support safety-significant functions and provide assurance of compliance with applicable regulatory criteria.

The FHP employs an active cooling loop attached to the SFP. The FHP SSCs with a handling or a storage function are immersed in the SFP and use the body of water as a heat sink. In the event that active cooling is lost, the SFP relies on evaporation to passively cool spent fuel assemblies and redundant makeup water systems to replace the SFP water inventory. The FHP shielding is inherent in most situations and is provided by the SFP water. Fuel handling activities are limited by physical and electronic interlocks to prevent a reduction in effective shielding. Where shielding with water is not practical, the FHP uses engineering biological shields, physical access controls, or administrative boundaries to protect workers from sources of radioactivity.


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Spent Fuel Pool

The SFP is sized to provide storage for ten years of spent fuel from normal operation as well as provide storage for other core assemblies.

The water volume of the SFP cools spent fuel for at least 7 days without makeup based on a fully loaded SFP heat load (i.e., approximately 225 kilowatts for a full SFP). The total SFP water inventory is approximately 574,000 gallons. Makeup water is provided from a connection to NI Water System during normal operations and from offsite sources using an external fill leg.

The primary components of the SFP include:

- SFP Structure – The concrete walls and floor provide the structural support for the SFP and is described in Section 7.8.2. The SFP is equipped with a stainless steel liner that prevents leakage of the pool water and is designed to the applicable requirements of ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Strutures and Commentary,” (Reference 7.3.1-3) and AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” (Reference 7.3.1-4). There are no pool wall penetrations below the elevation of the stored core assemblies to preclude the possibility of wall leakage uncovering spent fuel. The SFP is equipped with a leak chase system to monitor leakage from the steel liner.
- SFP Intact Fuel Racks – The intact fuel racks store intact core assemblies and are constructed from stainless steel. Intact fuel racks are designed to the applicable requirements of ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components, Division 1, Subsection NF,” (Reference 7.3.1-5), to provide structural integrity to the storage locations of fuel assemblies. The intact fuel racks use a combination of spacing and borated stainless steel plates in the fuel region to maintain criticality safety. A neutron absorber material monitoring program will be provided at the operating license stage. The spent fuel pool is equipped with seven free standing intact fuel racks.
- SFP Failed Fuel Racks - The failed fuel rack contains storage positions for failed fuel assemblies within FFCs. The failed fuel rack uses a combination of spacing and borated stainless steel plates installed in the fuel region to maintain subcriticality. The failed fuel rack is designed to the applicable requirements of ASME III, Division 1, Subsection NF, to provide structural integrity to the storage locations FFCs that contain fuel assemblies. Spare PIC cleaning vessels can be stored in the failed fuel rack due to the same external geometry as an FFC.
- Exterior Fill Leg - Backup makeup water to the SFP is provided by the SFP exterior fill leg; a dedicated fill pipe that runs from the exterior of the FHB directly into the SFP with no obstructions. The piping is open ended on the SFP side and is designed to allow tanker trucks, fire trucks, or other water delivery methods to provide a source of makeup water directly to the SFP. The exterior fill leg is designed to be functional during seismic and non-seismic design-basis events. The exterior fill leg is designed to the applicable requirements of ASME B31.1, “Power Piping,” (Reference 7.3.1-6).


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The SFP PHM is designed to applicable requirements of ANS 57.1, “Design Requirements for Light Water Reactor Fuel Handling Systems,” (Reference 7.3.1-7) for the design of the PHM to support the safety-significant functions. A mechanical positive control system ensures core assemblies cannot be dropped. The PHM is designed to the requirements of ASME NOG-1 Type 1 for seismic, single failure proof design features, and the PHM grappling mechanism is designed to the applicable requirements of ASME BTH-1, “Design of Below-the-Hook Lifting Devices,” (Reference 7.3.1-8).

The PHM is a temporary load path for FFCs and core assemblies. The load path between the core assembly or FFC and the machine’s rails is a temporary support during transfers. The PHM is not required to operate after a seismic event but does overhang the SFP and must not jeopardize the SFP safety functions, nor allow the load to drop. The PHM is designed to the requirements of ASME NOG-1, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) (Reference 7.3.1-1), Type 1 with seismic, single failure proof design features.

The PHM vertical motion limits and interlocks ensure that core assemblies have a minimum water height above the core assemblies to maintain shielding. The minimum water height is determined based on worst case expected radiological conditions of the core assemblies.

Pool Immersion Cell

The PIC provides a containment barrier function as described in Section 7.3.1.2. The confinement function is achieved by the PIC Receiver that mates to the PIC cleaning vessel by redundant inflatable seals, associated piping, and the PIC Processing Skid including isolation valves and radionuclide filtration equipment. The PIC Receiver and cleaning vessel are designed to the applicable requirements of ASME BPVC Section VIII, “Rules for Construction of Pressure Vessels, Division 1,” (Reference 7.3.1-9). The NI Air and Gas Distribution System supplies gas to the PIC to inflate the seals and for the chemical cleaning process.

The PIC processing filtration equipment is designed to the applicable requirements of ASME AG-1, “Code on Nuclear Air and Gas Treatment,” (Reference 7.3.1-10), and the piping is designed to ASME B31.3, “Process Piping,” (Reference 7.3.1-11). During a core assembly transfer from the BLTC to the PIC, the system is vented through the PIC processing skid to NI Heating, Ventilation, and Air Conditioning System, as described in Section 7.5.1. The PIC processing skid provides a controlled filtration path for radionuclides in the event a core assembly is dropped, while the BLTC and PIC provide a radionuclide confinement barrier. The PIC process skid is designed to the applicable requirements of ASME BPVC Section VIII.

The cleaning vessel passively transfers heat from the core assembly to the SFP. The structural components of the PIC act as part of the temporary load path for SR SSCs while core assemblies are within the PIC and are designed to the applicable requirements of AISC N690 to ensure the PIC can support the weight of core assemblies. The elevator and horizontal trolley act as part of the temporary load path for SR SSCs and structural support during the transfer of core assemblies from the PIC to the cask loading pit. The PIC transport components are designed to the applicable requirements of ASME NUM-1 to provide single failure proof design features. The PIC capture components are designed to the applicable requirements of ASME BTH-1.


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In the case where a spent fuel assembly fails within the SFP, the PIC includes design features to transport the failed fuel assembly back to the PIC for canisterization.

The FFCs are designed to the applicable requirements of ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components, Division 1, Subsection NB,” (Reference 7.3.1-12) to support the function to retain radionuclides of failed fuel assemblies. The FFC non-confinement components are designed to the applicable requirements of ASME III, Division 1, Subsection NF. After the failed fuel assembly is inserted and the canister is voided and back-filled, closure welds are performed while the FFC is contained in the PIC basket prior to transporting the FFC to the SFP. The FFC is tested to the applicable leakage criteria in ANSI N14.5, “Leakage Tests for Radioactive Materials,” (Reference 7.3.1-13). The canister is considered leak tight, preventing the ingress or egress of fluids such that it provides separation between damaged fuel assemblies and the SFP. Natural circulation within the SFP cools each FFC.

#### 7.3.1.4 Special Treatments

Special treatments are initially selected based on SSC safety classification as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the FHP to meet quality, design, reliability, and performance requirements are described in Section 7.3.1.2 and Section 7.3.1.3. The post-construction and start-up inspection and testing requirements applicable to the FHP are described in Chapter 12.

The programmatic special treatments, described in Section 8.0, that are applicable to FHP include:

- Quality Assurance Program
- Equipment Qualification Program
- Design Reliability Assurance Program
- Post Construction Inspection, Testing, and Analysis Program
- Reliability and Integrity Management Program

#### 7.3.1.5 Instrumentation Requirements

The SFP contains redundant level and temperature instruments in conformance with RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” Revision 5, and are designed with the applicable functional and design criteria established in IEEE 497, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” (Reference 7.3.1-14). The SFP water level sensors initiate SFP refill upon detection of low water level during normal operation, and perform a second function to monitor the SFP water level to maintain passive cooling of the spent fuel under adverse conditions to support PAM. The SFP temperature instruments are used to support PAM.

The Fuel Handling Supervisory Control System (FHC), described in Section 7.6.2, provides the hardware and software to monitor and control the fuel handling systems from the Fuel Handling Control Room. The FHC does not perform any safety-significant functions and is classified as non-safety-related with no special treatment. The FHC provides monitoring and controls for


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interfacing equipment operation. The FHC is integrated with the Plant Monitoring and Control System (PMC) and has workstations independent of the PMC located in the FHB. The PHM is operated locally and the equipment data is monitored by FHC.

## References


7.3.1-1 ASME NOG-1, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” ASME, 2020

7.3.1-2 ASME NUM-1, “Rules for Construction of Cranes, Monorails, and Hoists (With Bridge or Trolley or Hoist of the Underhung Type),” ASME, 2016

7.3.1-3 ACI 349, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” ACI, 2013

7.3.1-4 AISC N690, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” AISC, 2018

7.3.1-5 ASME, “BPVC, Section III: Rules for Construction of Nuclear Facility Components, Division 1, Subsection NF, Class 3,” ASME, 2017

7.3.1-6 ASME B31.1, “Power Piping,” ASME, 2022

7.3.1-7 ANSI/ANS 57.1, “Design Requirements for Light Water Reactor Fuel Handling Systems,” ANSI/ANS, 1992

7.3.1-8 ASME BTH-1, “Design of Below-the-Hook Lifting Devices,” ASME, 2017

7.3.1-9 ASME, “BPVC, Section VIII: Rules for Construction of Pressure Vessels, Division 1,” ASME, 2021

7.3.1-10 ASME AG-1, “Code on Nuclear Air and Gas Treatment,” ASME, 2009

7.3.1-11 ASME B31.3, “Process Piping,” ASME, 2020

7.3.1-12 ASME, “BPVC, Section III: Rules for Construction of Nuclear Facility Components, Division 1, Subsection NB, Class 1 Components,” ASME, 2017

7.3.1-13 ANSI N14.5, “Leakage Tests for Radioactive Materials,” ANSI, 2014

7.3.1-14 IEEE 497, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE, 2016




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**Table 7.3.1-1 Primary FHP and Component Classifications**

| Component | Safety Classification | Seismic Classification |
|---|---|---|
| SFP Intact Fuel Rack | SR | SCS1 |
| SFP Failed Fuel Rack | SR | SCS1 |
| SFP Liner | SR | SCS1 |
| SFP Exterior Fill Leg | NSRST | SCN3 |
| SFP Level Instruments | NSRST | SCN3 |
| SFP Temperature Instruments | NSRST | SCN3 |
| PHM | NSRST | SCN3 |
| PIC Cleaning Vessel | SR | SCS2 |
| PIC Failed Fuel Canister | SR | SCS1 |
| PIC Process Skid | SR | SCS2 |
| PIC Receiver | SR | SCS2 |



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**Table 7.3.1-2 Codes and Standards**

| Code | Version | Title |
|---|---|---|
| AISC N690 | 2018 | Specification for Safety-Related Steel Structures for Nuclear Facilities |
| ANSI/ANS 57.1 | 1992 | Design Requirements for Light Water Reactor Fuel Handling Systems |
| ANSI N14.5 | 2014 | Leakage Tests for Radioactive Materials |
| ASME BPVC Section III, Division 1, Subsection NF, Class 3 | 2017 | Rules for Construction of Nuclear Power Plant Components |
| ASME BPVC Section III, Division 1, Subsection NB | 2017 | Rules for Construction of Nuclear Power Plant Components |
| ASME AG-1 | 2009 | Code on Nuclear Air and Gas Treatment |
| ASME NOG-1 | 2020 | Rules for Construction of Overhead and Gantry Cranes |
| ASME NUM-1 | 2016 | Rules for Construction of Cranes, Monorails, and Hoists |
| ASME B31.1 | 2022 | Power Piping |
| ASME B31.3 | 2020 | Process Piping |
| ASME BPVC Section VIII, Division 1 | 2021 | Rules for Construction of Pressure Vessels |
| IEEE 497 | 2016 | IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations |
| ACI 349 | 2013 | Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary |


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**Figure 7.3.1-1 FHP Top-Down View**

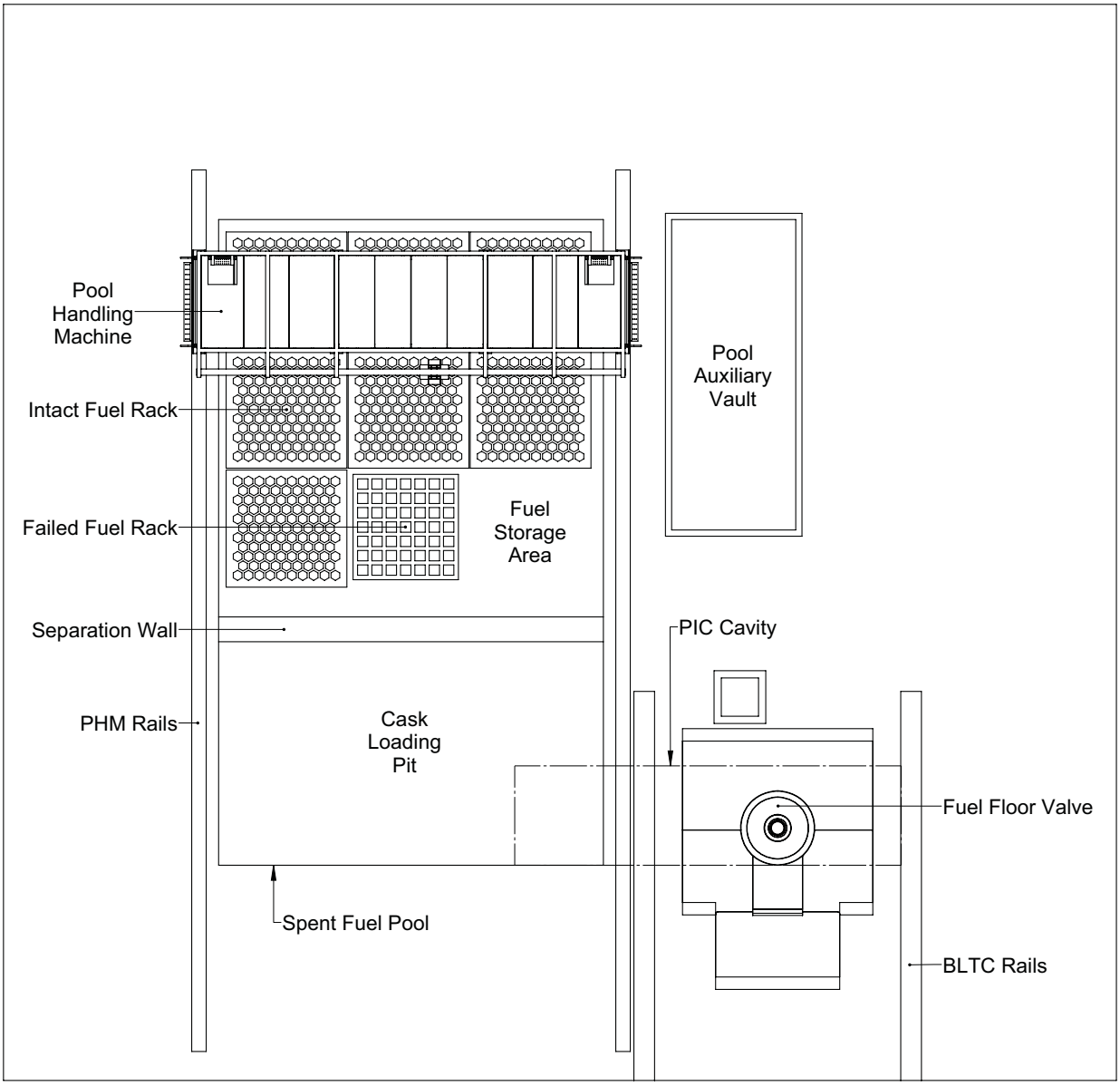




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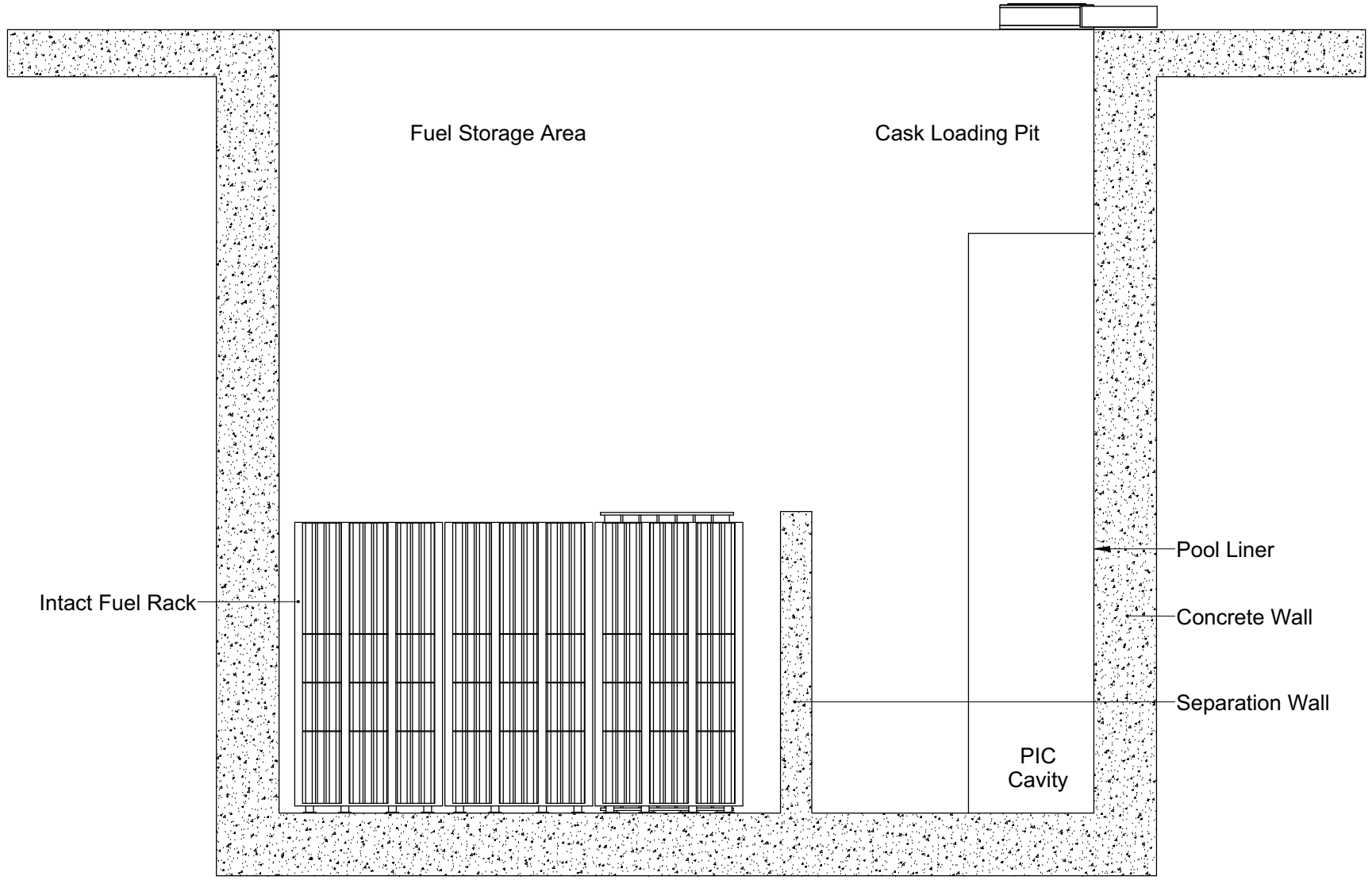
**Figure 7.3.1-2 SFP Side View**



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## 7.3.2 Ex-Vessel Fuel Handling System

The Ex-Vessel Fuel Handling System (FHE) includes equipment that facilitates the receipt, inspection, conditioning, storage, and removal from the core of core assemblies (Section 7.1.1). FHE equipment interfaces with the In-Vessel Fuel Handling System (FHI) as described in Section 7.3.3 for in-vessel handling operations and Water Pool Fuel Handling System (FHP) as described in Section 7.3.1 for fuel pool handling operations.

### 7.3.2.1 System Summary Description

FHE equipment transfers new core assemblies from the point of receipt through inspection and conditioning to the reactor vessel. The FHE is designed to convey a core assembly in an inert environment from the time it is placed in the New Assembly Preconditioning Station (NAPS) until it is transferred to the reactor vessel (Section 7.1.2), and subsequently from the time a core assembly is removed from the reactor vessel until it is placed in the Pool Immersion Cell and lowered into the spent fuel pool. Transfer of core assemblies in a shielded and inert environment provides radiation shielding, radionuclide retention, and a medium for removal of decay heat.

The FHE provides the following new core assembly handling functions:

- Pre-conditioning of new core assemblies in preparation for transfer to the reactor vessel.
- Storage of new core assemblies (after pre-conditioning in the NAPS).
- Transport of new core assemblies to the reactor vessel.

The FHE provides the following spent core assembly handling functions:

- Transportation of spent core assemblies from reactor vessel to Ex-Vessel Storage Tank (EVST).
- Transportation of spent core assemblies from EVST to the Water Pool Fuel Handling System.

The major components of the FHE are described below. Table 7.3.2-1 provides the FHE major component safety and seismic classifications.

#### 7.3.2.1.1 Ex-Vessel Handling Machine

The Ex-Vessel Handling Machine (EVHM) is a rail-mounted, shielded, sealable, vessel transport system that supports core assembly handling operations. It is used to transfer core assemblies (non-fuel, irradiated fuel, and non-irradiated fuel), contained within a liquid sodium filled core assembly pot (CAP), between the reactor vessel located in the Reactor Building (RXB) and the EVST located in the Fuel Handling Building (FHB). Core assemblies are transferred through the fuel transfer ports of the EVST and reactor vessel via an internal grapple and hoist system that raises and lowers a CAP containing core assemblies. The EVHM also transfers Lead Demonstration Assemblies (LDAs) and Lead Test Assemblies (LTAs) between the EVST and the Pin Removal Cell (PRC) after completion of refueling operations. In addition to providing shielding capabilities, the EVHM provides an inert atmosphere for the core assemblies along with heating or cooling as required. Figure 7.3.2-1 depicts the EVHM primary components.


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The EVHM consists of the following subsystems:

- Ex-Vessel Handling Machine Transporter

  The EVHM transporter is a rail-mounted structure that houses, transports, and positions the shielded cask assembly. This transporter enables the shielded cask assembly to transfer core assemblies between the EVST and the FTL, as well as between the EVST and the PRC. It is propelled by electrically driven wheels and is powered through floor service stations via a cable reel that pays out the power cable as it moves through the facility. In addition to housing and transporting the shielded cask assembly, the transporter also houses and provides access to all the systems and equipment needed for the shielded cask assembly and transporter to operate.

- Shielded Cask Assembly

  The shielded cask assembly has three primary functions; to provide the required shielding and pressure boundary management for safe handling of irradiated core assemblies, to maintain an argon inert environment, and to maintain fuel assemblies within their allowable design temperatures.

  Shielding and pressure boundary management prevents release of irradiated materials from the shielded cask assembly. The cask assembly is maintained inert to prevent the liquid sodium from reacting with oxygen and to prevent oxidation of sodium on the grapple, grapple hoist system, and other internal surfaces.

- Grapple and Hoist System

  The grapple and hoist system interfaces with a CAP and transfers it and its contents into and out of the shielded cask assembly. The grapple is located within the shielded cask module while the hoist system rests on top of the shielded cask module. The grapple does not directly handle any reactor core assemblies. The grapple provides the capability to interface with the drip pan within the drip pan module to allow for disposal of residual sodium through the MCV.

- Cask Heating and Cooling System

  During EVHM operations that require cooling, the exhaust blower fans continuously pull air through the annulus between the Main Shielding Cask and the cold wall. During EVHM operations that require heating, dampers contained within the air channels will shut and resistive heaters are energized. The dampers prevent air from circulating through the annulus between the MSC and the cold wall. The resistive heaters provide heat to the cold wall to avoid heat loss while the Argon System heaters are in operation for temperature control. If the EVHM loses its plant side power connection, the dampers fail open, allowing the EVHM to maintain cooling capability through natural convection. The cladding of a spent fuel assembly can be kept within acceptable temperature limits by passive cooling means e.g., radiative, and natural convection heat transfer. For a conservative maximum core assembly heating rate of 11 kW (normal 2.5 kW), and with maximum environmental condition of 120 degrees Fahrenheit (48 degrees Celsius),


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passive cooling maintains a steady state maximum cladding temperature (MCT) below 1100 degrees Fahrenheit. This provides a thermal margin of approximately 63 degrees Fahrenheit (35 degrees Celsius).

- Core Assembly Pot

  The CAP interfaces between each core assembly and the EVHM cask. Each CAP is filled with liquid sodium and acts as the heat sink and conduction path to assist in removing heat from the irradiated core assembly and preventing overheating of the fuel pins. The CAP is a heavy-walled tube, which is closed at the bottom, and provides grapple handling socket at the top. The upper part of the CAP also has a passive liquid sodium siphon feature that lowers the liquid sodium level in the CAP as it leaves the liquid sodium in the reactor vessel or the EVST. Lowering the liquid sodium level in the CAP minimizes spillage that can occur during liquid sodium temperature increases and fuel handling operations.

#### 7.3.2.1.2 Bottom Loading Transfer Cask

The Bottom Loading Transfer Cask (BLTC) is a rail-mounted, shielded, sealable vessel transport system that supports core assembly handling operations. It is used to transfer core assemblies (non-fuel, irradiated fuel, and non-irradiated fuel), between the NAPS, Pool Immersion Cell and the EVST located in the FHB. Core assemblies are transferred through the fuel transfer ports of the NAPS, EVST, and Pool Immersion Cell using the BLTC internal grapple and hoist system which raises and lowers the core assemblies.

In addition to providing shielding capabilities, the BLTC provides an inert atmosphere for the core assemblies along with heating and cooling as required. Figure 7.3.2-2 depicts the BLTC primary components.

The BLTC consists of the following subsystems:

- Bottom Loading Transfer Cask Transporter

  The BLTC transporter is a rail-mounted structure that houses, transports, and positions the shielded cask assembly. This transporter enables the shielded cask assembly to transfer core assemblies between the EVST, NAPS and the PIC. The BLTC transporter is propelled by electrically driven wheels and is powered through floor service stations via a cable reel that pays out the power cable as it moves through the facility. In addition to housing and transporting the shielded cask assembly, the transporter also houses and provides access to the systems and components needed for the shielded cask assembly and transporter to operate.

- Shielded Cask Assembly

  The shielded cask assembly has three primary functions; to provide the required shielding and pressure boundary management for safe handling of irradiated core assemblies, to maintain an argon inert environment, and to maintain fuel assemblies within their allowable design temperatures.


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Shielding and pressure boundary management prevents release of irradiated materials from the shielded cask assembly. The cask assembly is maintained inert to prevent the liquid sodium from reacting with oxygen and to prevent oxidation of sodium on the grapple, grapple hoist system, and other internal surfaces.

- Grapple and Hoist System

  The grapple and hoist system is responsible for interfacing with a core assembly and transferring it and its contents into and out of a shielded cask assembly. The grapple is located within the shielded cask module while the hoist system rests on top of the shielded cask module. The grapple provides the capability to interface with the drip pan within the drip pan module to allow for disposal of residual sodium through the movable closure valve module (MCV).

- Cask Heating and Cooling System

  At the bottom and top of the main shielding cask, channels in the shielding layers allow ambient air to be drawn into and up through the annulus between the internal diameter of the MSC and the cold wall. During BLTC operations that require cooling, the exhaust blower fans continuously pull air through these channels. During BLTC operations that require heating, dampers contained within the air channels shut and resistive heaters are energized. The resistive heaters provide heat to the cold wall to avoid heat loss while the argon system heaters are in operation for temperature control. If the BLTC losses plant power the dampers fail open, allowing the BLTC to maintain cooling capability through natural convection. The MCT of a core assembly can be kept within acceptable limits using passive cooling. For a core assembly with maximum decay heat of 1.2 kW and environmental conditions at 120 degrees Fahrenheit, the MCT remains below 1100 degrees Fahrenheit. The thermal margin is no less than 104 degrees Fahrenheit (58 degrees Celsius).

#### 7.3.2.1.3 Ex-Vessel Storage Tank

The EVST is a liquid sodium filled tank with a rotatable carousel for core assemblies. Its function is to provide a controlled environment for temporary storage for new components in route to the reactor, and irradiated core assemblies removed from the reactor. The EVST has the ability to actively and passively remove decay heat. Storage positions are provided for 100 core assemblies within CAPs. Figure 7.3.2-3 depicts the EVST primary components.

The EVST consists of the following subsystems:

- Striker Plate

  The striker plate protects the EVST's functional and confinement capabilities against accidentally dropped external loads onto the EVST. The striker plate, which forms the operating deck surface above the EVST, is a plate supported from the building structure. Penetrations through the striker plate are covered by access port covers which transfer all loads to the striker plate.


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- Rotating Carousel

  The rotating storage carousel is a vertical cylinder which houses and supports CAP. It provides orderly transient storage for radioactive core assemblies in an array with a subcritical configuration by geometry, consistent with safety and shielding requirements.

- Main Vessel

  The main vessel contains the liquid sodium coolant and confines the radioactive contaminants. The main vessel provides a gas-tight inert environment to assist in maintaining the liquid sodium purity. The main vessel houses the independently supported core assembly storage carousel.

- Guard Vessel

  In the event of a main vessel leak, the guard vessel captures the leaked liquid sodium while maintaining the sodium level in the main vessel above the fueled portion of the core assemblies.

- Fuel Transfer Adapter

  The fuel transfer adapter (FTA) provides an interface between the FFVs on the operating deck level and the transfer ports in the vessel head of the main vessel. The FTA maintains the inert environment during core assembly transfers and when a transfer port plug is removed.

- Ex-Vessel Storage Tank Sodium Cooling and Heating System

  The EVST requires heat removal systems capable of removing decay heat from a fully loaded EVST to maintain fuel cladding temperatures below design allowable temperatures. Passive cooling is provided during off-normal operations.

  When there is insufficient decay heat in the vessel, the sodium heating system maintains the sodium temperature within operating limits to overcome normal heat losses. Electrical trace heaters on the exterior of the guard vessel and trace heating of the sodium processing loop are used to provide sufficient heat to maintain the sodium temperature if no or few spent fuel assemblies are in the vessel.

#### 7.3.2.1.4 Ex-Vessel Storage Tank Cover Gas

The EVST cover gas supports the EVST by providing an argon gas inert barrier between the EVST vessel and the liquid sodium. The equipment supplies fresh argon into the EVST main and guard vessels and provides direct control of contaminants and pressure control within the main vessel. The equipment allows for radiological monitoring, sodium filtration, impurity monitoring, pressure relief, and constant pressure to the EVST buffer seals.


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#### 7.3.2.1.5 Fueling Floor Valve

The FFV is a temporary closure and opening device used during refueling and maintenance operations. The fueling floor valve is used at transfer ports of the storage cells and reactor vessel to permit the transfer of shield plugs, core assembly pots, and core assemblies while maintaining an inert containment boundary. The valve body provides a pressure boundary, shielding, and load bearing. Static and dynamic seals provide isolation between interfacing facilities, components, and exterior operating environment. The static and dynamic seals also isolate between internal valve cavities. The FFV has a normal operating internal temperature range of 250 degrees Fahrenheit to 400 degrees Fahrenheit and a normal operating internal pressure range of 0.3 pounds per square inch gauge to 0.4 pound per square inch gauge.

#### 7.3.2.1.6 Core Assembly Transfer Tube

The core assembly transfer tube (CATT) provides an inert passageway between the EVHM and reactor vessel for fuel transfers in support of refueling. The CATT is a long tubular structure that provides an inert boundary and shielding between the drive assembly of the Fuel Transfer Lift (FTL) and an FFV. The CATT is installed only during Mode 4, plant refueling operations. The CATT has redundant sealing capabilities.

Monitoring of conditions within the CATT is performed to ensure that core assembly integrity is maintained. The pressure of the internal cavity of the CATT is monitored to ensure there is not a leak of reactor atmosphere or argon gas to the RXB. Argon gas within the CATT maintains a positive pressure in the event of a leak which prevents atmosphere migration into the cavity.

The temperature of the internal cavity of the CATT is maintained to prevent solidification of the liquid sodium coolant and prevent overheating of the core assembly. The minimum temperature in the CATT internal cavity is 400 degrees Fahrenheit and the maximum temperature is 1100 degrees Fahrenheit.

#### 7.3.2.1.7 Pin Removal Cell

The Pin Removal Cell (PRC) is a hot cell located below grade along the EVHM travel path between the EVST and reactor vessel. Its function is to provide a designated area for the receipt of irradiated lead driver assemblies (LDA) and lead test assemblies (LTA) within CAPs from the EVST via the EVHM. The PRC supports the extraction and packaging of their removable pins. The PRC provides an inert atmosphere free of oxygen, moisture, and deleterious contaminants that allows for transfer of fuel assemblies back to the EVST and provides cooling to maintain required lead test pin (LTP) temperatures. Decay heat removal is performed passively through the sodium tank out to the tank shell then to the tank room environment via natural convection.

The pin extraction machine (PEM) is a remotely operated electro-mechanical handling device located in the PRC. The PEM provides a range of motion, required load carrying capacity, as well as a modular effector head to change out various grapples and inspection instruments as required.


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Working in conjunction with the pin extraction machine are manipulators which include a mix of master-slave and powered units. The manipulators have the reach to access all required areas of the cell.

A stainless-steel liquid sodium holding tank is mounted into the PRC hot cell floor underneath the FFV to receive an LDA or LTA in a CAP from the EVHM. Submerging the CAP containing the LDA or LTA in liquid sodium allows for effective decay heat removal, keeping the test assemblies within their design temperature range. Heat is transferred from the LDA or LTA, through the liquid sodium and out of the tank via free convection.

#### 7.3.2.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the FHE design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the FHE is described below.

Licensing management has decided to delete PDC 1 from all PSAR sections since it is addressed globally in PSAR Chapter 5.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, FHE safety-related (SR) components are designed and evaluated to withstand external hazards and natural phenomena. FHE components located in the FHB (Section 7.8.2) and RXB (Section 7.8.1.1) substructures are afforded protection against the effects of natural phenomena. In the event of adverse conditions due to natural phenomena SR FHE components located outside of the FHB and RXB substructures are designed to maintain the capability to perform their required safety functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels are described in Section 6.1.1. FHE components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions.

Consistent with PDC 3, *Fire protection*, the design of the FHE minimizes the probability and effect of fires through design features that minimize the potential for liquid sodium leakage to the surroundings as well as the selection of non-combustible and non-fire sustaining materials to the extent practical. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the FHE is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents. Materials selection for FHE components provides assurance of compatibility with the liquid sodium environment, including sodium aerosols, vapors, and oxidation products. Equipment qualification for harsh environmental conditions resulting from accident conditions is provided by the FHE component quality, design, and qualification requirements in accordance with the Equipment Qualification Program (described in Chapter 8).


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Consistent with PDC 16, *Containment Design*, radionuclide retention is implemented using a functional containment strategy, which consists of multiple barriers that control the release of radioactivity to the environment. The EVHM, FFV (located at the top of the CATT) and CATT gas retaining parts are a part of the functional containment strategy and are designed such that their safety-significant design conditions are not exceeded for as long as the postulated accident conditions require.

Consistent with PDC 61, *Fuel storage and handling and radioactivity control*, the FHE is designed to permit periodic inspection and testing of safety-significant core assembly handling and storage components. Fuel handling and storage equipment is designed to provide suitable shielding for radiation protection and to provide a functional containment barrier for appropriate confinement and containment of assemblies during storage and handling operations. The storage and handling equipment is designed to provide passive residual heat removal to assure adequate cooling under postulated accident conditions.

Consistent with PDC 62, *Prevention of criticality in fuel storage and handling*, FHE equipment is designed to provide a means to prevent criticality during storage and handling of fuel assemblies via a geometrically sub-critical configuration which limits $k_{eff} \leq 0.95$ based on the most reactive fuel configuration. Section 3.14 contains information pertaining to criticality analysis.

Consistent with PDC 63, *Monitoring fuel and waste storage*, the FHE provides for passive heat removal to maintain the fuel at or below the temperature design limit. The FHE design includes instrumentation to detect conditions that may result in loss of residual heat removal capability or excessive radiation levels, and to initiate appropriate safety actions.

Consistent with PDC 72, *Sodium Heating System*, the EVST provides a safety-significant functional containment function. The main vessel and process loops include heat tracing to prevent liquid sodium freezing when there is insufficient heat coming from contents inside of the EVST. Heat tracing is provided for normal operations.

Consistent with PDC 73, *Sodium leakage detection and reaction prevention and mitigation*, the EVST design provides a main and guard vessel to contain liquid sodium coolant and confine radioactive contaminants. In the event of a main vessel leak, the guard vessel captures the leaked liquid sodium while maintaining the liquid sodium level in the main vessel above the fueled portion of the core assemblies. The EVST guard vessel has leakage detection devices. The EVST vault is lined with stainless steel to protect concrete from liquid sodium reactions.

Consistent with PDC 74, *Sodium/water reaction prevention/mitigation*, the FHE liquid sodium containing equipment is designed to be sealed, inerted, and segregated to prevent contact with water.

Functional Requirements and Design Criteria

Probabilistic risk assessment (PRA) safety functions are determined through the plant-specific PRA described in Section 3.1. The safety-significant PRA safety functions are identified as SR functions or non-safety-related with special treatment (NSRST) functions as defined in


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Section 5.2. The FHE supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classification of FHE components relied upon to perform the function:

| Function ID | Function Description | FHE Functional Support |
|---|---|---|
| DL3-HR6 | Passive Heat Removal in EVHM (SR) | Passive heat removal in the EVHM prevents fuel damage. |
| DL3-HR7 | Passive Heat Removal in EVST (SR) | Passive heat removal in the EVST prevents fuel damage of the fuel stored there. This function is inclusive of all heat transfer from fuel inside the EVST to the ultimate heat sink. |
| DL3-HR8 | Passive Heat Removal in BLTC (SR) | Passive heat removal in the BLTC prevents fuel damage. |
| DL3-HR9 | Passive Heat Removal in PRC (SR) | Passive heat removal in the PRC prevents fuel damage. |
| DL3-RR3 | EVHM Cask Barrier (SR) | EVHM cask barrier provides a containment barrier when fuel is in the cask and being transported between stations. The EVHM performs a containment function when connected to either station and the drip pan, FFV, and movable closure valve are open (whether there is fuel in the cask or not). The EVHM extends the functional containment boundary when mated to reactor head, EVST and PRC. |
| DL3-RR3a | EVHM Transfer Barrier Function (SR) | Temporary contain function during refueling. |
| DL3-RR4 | EVST Barrier (SR) | The EVST barrier provides a radionuclide barrier up to and including the isolation valves leading to the EVST auxiliary systems. Valve closure is not needed to meet SR function. |
| DL3-RR5 | BLTC Barrier (SR) | The BLTC barrier provides a containment barrier while fuel is in cask. The BLTC barrier also provides temporary seals when mated to the EVST during fuel movements. |
| DL3-RR5a | BLTC Transfer Barrier (SR) | Temporary contain function provided by installed equipment during all fuel movements in and out of the BLTC. |


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| Function ID | Function Description | FHE Functional Support |
|---|---|---|
| DL3-RR6 | PRC Cell Barrier (SR) | PRC Hot Cell barrier provides a barrier for potential releases from fuel damaged during pin removal process. |
| DL4-RR5 | EVST Guard Tank Leak Prevention function (NSRST) | Defense in depth back up barrier to EVST |
| DL5-PAM1 | Post Accident Monitoring (NSRST) | The EVST provides a post-accident monitoring temperature indication instrument to allow operators to monitor the EVST internal bulk sodium temperatures to determine if the fuel cladding temperature limits are being approached. |

The FHE provides a temporary structural support load path for SR and NSRST SSCs that do not require continuous load path structural support. The temporary structural support load path is used during the transport of core assemblies. Information pertaining to the design of the FHE is provided in Section 7.3.2.3.

The FHE provides a structural support load path function for SR SSCs as described in Section 7.3.2.3.

FHE components located in the FHB and RXB substructures are afforded protection against the effects of natural phenomena. SR FHE components located outside of the FHB and RXB substructures are designed to maintain the capability to perform their required safety functions. Refer to Section 6.4.2 for additional information pertaining to the design of SR and NSRST structures.

Safety classifications for FHE structures and components are listed in Table 7.3.2-1. The SR design criteria corresponding to the SR functions applicable to the FHE are derived as described in Section 6.1.2 and identified in Section 5.2. The SR design criteria are incorporated into the design of the FHE as described in Section 7.3.2.3.

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the FHE, along with a description of the use and conformance to the guidance, is provided.

Regulatory Guide (RG) 1.31, Revision 4, *Control of Ferrite Content in Stainless Steel Weld Metal*

Partial Conformance, RG 1.31 references the latest consensus standards. The design complies with RG 1.31 with the exception of Regulatory Position C.4 that when design temperatures exceed 800 degrees Fahrenheit (427 degrees Celsius), the delta ferrite shall be limited to the range ferrite number 3 to 10 in accordance with the ASME Section III, Division 5 Code.

RG 1.87, Revision 2, *Acceptability of ASME Code, Section III, Division 5, High Temperature Reactors*


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Full Conformance, The RG 1.87 applicability, limitations, and restrictions regarding the use of ASME BPVC Section III, Division 5 are conformed to for FHE code pressure boundary components and core support structures, including the listed acceptable (N872) and conditionally acceptable (N-861, N862, N-898) ASME Code Cases.

RG 1.97, Revision 5, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*

Partial conformance, provides guidance with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants.

RG 1.244, Revision 0, *Control of Heavy Loads at Nuclear Power Plants*

Full Conformance, RG 1.244, provides guidance for the control of heavy loads. Guidance includes providing protection against equipment failure that could result in a heavy load drop.

Seismic Classification and Design Codes

Seismic classifications for FHE components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based upon seismic classification. The seismic classification of FHE components are listed in Table 7.3.2-1. Seismic design, analysis, and qualification is performed as described in Section 6.4. Seismic special treatments are applied to NSRST portions of the FHE, based on seismic classification, to provide assurance that risk-significant functions or functions required for adequate defense-in-depth are capable of being performed.

Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of FHE components as special treatments, as described in Section 6.3. Codes and standards that are applicable to the FHE are listed in Table 7.3.2-2.

### 7.3.2.3 System Design and Performance Evaluation

The FHE is designed to meet design basis requirements described in Section 7.3.2.2 and evaluated to provide assurance that associated performance requirements are met. Design features are provided by the FHE design that support safety-significant functions and provide assurance of compliance with applicable regulatory criteria.

The safety-significant portions of the FHE are protected from design-basis hazards and associated design-basis hazard levels described in Section 6.1.1. Protection from tornadoes, high-winds and wind-driven missiles, extreme climatic conditions, and external fires is provided by the RXB and FHB substructures as discussed in Sections 7.8.1.1 and 7.8.2 respectively. External flooding and precipitation protection for the site SR structures, including the FHE location within the RXB and FHB, is described in Section 6.1.1. The location of the FHE within the RXB and FHB substructures also provides protection from local transportation incident-related hazards, offsite industrial and chemical facilities, and turbine failure-related missiles initiating from the plant turbine-generator set.

The FHE location within the RXB and FHB substructure provides protection from accidental or intentional aircraft impact (Section 11.4).


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Ex Vessel Handling Machine

The EVHM uses a liquid sodium filled core assembly pot as well as argon gas to provide passive heat removal to maintain the fuel at or below temperature design limits. The liquid sodium filled core assembly pot provides cooling for a core assembly up to approximately 11 kilowatts (37,534 BTU/hr) decay heat without need for forced cooling during an anticipated operational occurrence, design basis event, or loss of offsite power event. For a core assembly heating rate of approximately 11 kilowatts, passive cooling is maintained at an approximate steady state maximum cladding temperature of 1,094 degrees Fahrenheit (590 degrees Celsius). This provides a thermal margin of approximately 63 degree Fahrenheit (35 degrees Celsius). The EVHM shielded cask is comprised of lead and B4C which augments passive heat removal and shielding capabilities while carrying a discharged core assembly. During transfer, the liquid sodium filled core assembly pot encompass a discharged core assembly. The liquid sodium filled core assembly pot resides at the center of the EVHM shielded cask cavity.

The applicable requirements of ASME BPVC Section III, Division 5, are applied in the design of the shielded cask assembly to ensure that heat is radiated passively and cooling is provided to maintain temperature limits as well as to ensure pressure boundary integrity remains intact to mitigate radionuclide release.

The EVHM cask barrier provides a containment barrier when fuel is in cask and being transported between stations. The EVHM performs a containment function when connected to either station and the drip pan, FFV, and movable closure valve are open (whether there is fuel in the cask or not). The EVHM extends the functional containment boundary when mated to the reactor head, EVST, and PRC.

The shielded cask assembly uses seals within the movable closure valve, drip pan module, and viewing module. Each seal is redundant, static, or dynamic and constitute a pressure boundary. The sealed and argon inerted environment limit the escape of radioactive gases to the RXB and FHB. Each inflatable seal is designed to remain inflated for 72 hours in the event of a loss of primary argon supply.

In Modes 1, 2, and 3, the functional containment boundary is coincident with the Primary Coolant Boundary. The EVHM to head barrier is designed to perform a temporary contain function during Mode 4 refueling. The FHE provides pressure and gas seals to maintain a sealed and inert environment to maintain containment barrier.

The EVHM Shielded Cask Assembly and argon supply, FFV, CATT Structure, EVST FTA, and PRC FTA are seismically qualified and are designed to remain a sealed mating surface during a design basis earthquake.

The FTA, CATT and FFV are designed to the applicable requirements of ASME BPVC Section III, Division 5, to ensure that heat is radiated passively and cooling is provided to maintain temperature limits.


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The requirements of ASME NOG-1, “Rules for Construction of Overhead and Gantry Cranes,” (Reference 7.3.2-2) are utilized to ensure the containment barrier is not compromised due to a seismic event. The EVHM is designed using rail-based, seismic restraints that limit uplifting or tipping of the cask when moving or parked.

The EVHM grapple and hoist are designed to meet the applicable requirements of ASME NOG-1 and ASME BTH-1 (Reference 7.3.2-5) as endorsed in RG 1.244 “Control of Heavy Loads at Nuclear Facilities,” Revision 0.

Ex-Vessel Storage Tank

The EVST provides passive heat removal of decay heat from spent fuel assemblies. The EVST is designed for approximately 100 storage locations for fuel assemblies and non-fuel core assemblies. The EVST is designed for a maximum decay heat load of 216 kW, with a normal decay heat load of 136 kW. The EVST transfers heat from the vessel to the atmosphere by natural convection. The EVST passive cooling system is designed to remove the maximum decay heat and provide cooling of the core assemblies. Passive cooling maintains the fuel assembly temperatures below the maximum pin cladding temperature of 1157 degrees Fahrenheit (625 degrees Celsius).

The EVST is equipped with a main and guard vessel used to contain liquid sodium coolant and confine radioactive contaminants. In the event of a main vessel leak, the guard vessel captures the leaked liquid sodium while maintaining the liquid sodium level in the main vessel above the core assemblies. The main vessel head is the gamma shield between the EVST and the operating floor. The EVST contains residual liquid sodium from CAPs and core assemblies. EVST main vessel is equipped with isolation valves that close upon a detection of a leak or a loss of power to contain the radionuclides within the EVST vessel.

The EVST Cover Gas system consists of the argon gas supply for the EVST main vessel and guard vessel. It provides monitoring and purification of the inert gas within the EVST as well as, pressure control for the EVST Main Vessel. The system provides for removal of impurities, fission products, and sodium vapor from the cover gas. The system also prevents ingress of air beyond the EVST head.

The applicable requirements of the ASME BPVC, Section III, Division 5 are applied in the design of the main pressure vessel and guard vessel to ensure that radioactive material is confined and adequate cooling is provided to maintain temperature limits.

The applicable requirements of ANSI/AISC N690 (Reference 7.3.2-3) are applied relative to the design of the EVST striker plate.

The EVST design utilizes sensors designed to the requirements of IEEE 497 (Reference 7.3.2-4) to monitor the temperature of the main pressure vessel. This allows operators to monitor the EVST internal bulk sodium temperatures to determine if the fuel cladding temperature limits are being approached.


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Bottom Loading Transfer Cask

The BLTC is designed to accommodate the maximum cladding temperature of a 1.2 kW fuel assembly. The BLTC handles discharged core assemblies following a specified storage time in the EVST. The storage time is of sufficient length to maintain fuel cladding temperatures below design allowable temperatures. The BLTC uses the argon gas system as well as cold wall blowers to provide active heat removal capability during normal operation and uses passive cooling to provide heat removal capability during a loss of active cooling event.

The applicable requirements of ASME BPVC Section III, Division 5, are applied to the design of the BLTC shielded cask assembly to ensure heat is radiated passively and adequate cooling is provided to maintain temperature limits.

The BLTC is designed to provide a containment barrier while core assemblies are in cask. The BLTC also provides temporary seals to the EVST during fuel movements.

The BLTC maintains core assemblies continuously sealed within an inert environment. The shielded cask assembly uses seals within the movable closure valve, drip pan module, and viewing module. Each seal is redundant, static, or a dynamic seal and constitutes a pressure boundary. The sealed and argon inerted environment limit the escape of radioactive gases to prevent airborne activity in the RXB and FHB. Each inflatable seal is designed to remain inflated for 72 hours in the event of a loss of primary argon supply.

To ensure the pressure boundary is not compromised due to a seismic event, the BLTC is designed using rail-based, seismic restraints that limit uplifting or tipping of the cask when moving or parked per ASME NOG-1.

The BLTC grapple and hoist are designed to meet the applicable requirements of ASME NOG-1 and ASME BTH-1 (Reference 7.3.2-5) as endorsed in RG 1.244 “Control of Heavy Loads at Nuclear Facilities,” Revision 0.

A temporary contain function is provided by installed equipment during fuel movements in and out of the BLTC. The FHE provides pressure and gas seals to maintain a sealed and inert environment to maintain containment barrier.

The BLTC shield cask assembly and argon supply, EVST FTA, and FFV are seismically qualified by design to remain a sealed mating surface during an SSE.

Pin Removal Cell

The PRC is designed to provide passive heat removal to prevent fuel damage. The PRC provides an internal liquid sodium holding tank as a cooling medium with accompanying ventilation system which maintains the temperature of the LDA and LTA inside the hot cell below design limits.

The PRC structure is designed to provide a confinement function for radionuclides within the hot cell. Isolation valves are provided for systems that penetrate the PRC structure. Openings to the fueling floor are equipped with FFVs or capped with port plugs at all times to maintain closure.


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The Pin Extraction Machine, which is a remotely operated mechanical handling device is used to remove LTPs from the LTA and transfer them to the previously staged canister within the PRC.

The applicable requirements of ASME BPVC Section III, Division 5, are applied to the design of the PRC liquid sodium holding tank to ensure heat is radiated passively and adequate cooling is afforded.

Materials of construction for FHE components are selected to be consistent with the requirements of ASME BPVC to provide adequate material quality requirements and assure compatibility with the high-temperature liquid sodium operating environment.

#### 7.3.2.4 Special Treatments

Special treatments are initially selected based on SSC safety classification as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the FHE is described in Section 7.3.2.3. The following programmatic special treatments are preliminarily identified as applicable to the FHE to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- Equipment Qualification Program
- Post-Construction Inspection, Testing, and Analysis Program (Chapter 12)

#### 7.3.2.5 Instrumentation Requirements

The FHE local instrumentation and controls systems contain monitoring and control capabilities to safely handle core assemblies. The handling of fuel during refueling is controlled by a series of interlocks. The EVHM and BLTC contain controls and interlocks that prevent load drops or handling accidents in conformance with the applicable requirements of ASME NOG-1.

Interlocks protect the EVST equipment and payloads.

- Carousel Rotation-The semi-automatic carousel positioning system moves the selected storage position to its respective transfer port. A system of interlocks prevents repositioning when damage would result. This system verifies automatically that all transfer port plugs are in place or, if not, that the BLTC or EVHM grapple is either in the fully retracted position or not at the port with the FFV closed.
- BLTC-EVHM Lockout-An interlock circuit verifies that the EVST or BLTC is in a safe condition to access. These interlocks require that the carousel is stopped, that the position agrees with the address requested (except in the MANUAL mode), that the carousel position produces an alignment between a storage location and an open port, and that the EVHM or BLTC is over the correct Floor Valve.


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Primary Operator interaction with the Fuel Handling Supervisory Control System (FHC) and Fuel Handling systems is through multiple human-machine interface units located in the FHB control room. The human-machine interface units are part of the Plant Monitoring and Control System. The FHC provides the interfaces needed for the fuel handling operators to monitor and control the various fuel handling SSCs remotely from the FHB control room. This includes process, alarm, setpoints and command data exchange with the Plant Monitoring and Control System's human-machine interfaces that are in the FHB Control Room. The Plant Monitoring and Control System's human-machine interfaces provide the operators the window to the process for the controlling and monitoring of the fuel handling SSCs from the FHB control room. The FHC provides supervisory, control and data acquisition functions for the fuel handling SSCs. The FHC is a non-safety related with no special treatment system.

The movement of core assemblies is accomplished through the coordination of individual machines and systems that are connected to the FHC to form an interconnected fuel handling system. The FHC provides supervisory, control and data acquisition functionality to orchestrate, monitor and control machine movements to support the initial fuel loading of the reactor followed by subsequent refueling of the reactor, placement of non-fuel core assemblies and for the movement of spent fuel and components from the reactor.

References


7.3.2-1 American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Facility Components, Division 5, High Temperature Reactors," ASME, 2017

7.3.2-2 American Society of Mechanical Engineers, NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," ASME 2020

7.3.2-3 American Institute of Steel Construction, "Specification for Safety-Related Steel Structures for Nuclear Facilities," ANSI/AISC N690, 2018

7.3.2-4 Institute of Electrical and Electronic Engineers, Standard 497-2016, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," IEEE, 2016

7.3.2-5 American Society of Mechanical Engineers, Standard BTH-1, "Design of Below-the-Hook Lifting Devices," ASME, March 15, 2017

7.3.2-6 American Society of Mechanical Engineers NML-1, Rules For The Movement Of Loads Using Overhead Handling Equipment In Nuclear Facilities, 2019.

7.3.2-7 American Society of Mechanical Engineers NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," ASME, 2015

7.3.2-8 American Society of Mechanical Engineers B31.1, "Power Piping," ASME, 2020

7.3.2-9 American Society of Mechanical Engineers AG-1, "Code on Nuclear Air and Gas Treatment," ASME, 2017



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7.3.2-10 Institute of Electrical and Electronics Engineers 1023, “IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities, Institute of Electrical and Electronics Engineers,” IEEE, 2020

7.3.2-11 Institute of Electrical and Electronics Engineers 1050, “Guide for Instrumentation and Control Equipment Grounding in Generating Stations,” IEEE, 2004

7.3.2-12 Military Standard 461G, “Department of Defense Interface Standard: Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment,” MIL-STD, 2015

7.3.2-13 National Fire Protection Association, “National Electrical Code,” NFPA, 2020

7.3.2-14 International Electrotechnical Commission 61000-3, “Electromagnetic Compatibility (EMC) - Part 3: Limit - ALL PARTS,” IEC, 2017



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**Table 7.3.2-1 FHE Major Component Classifications**

| Component | Safety Classification | Seismic Classification |
|---|---|---|
| EVHM | SR | SCS1 |
| BLTC | SR | SCS1 |
| EVST | SR | SCS1 |
| Fueling Floor Valve | SR | SCS1 |
| Core Assembly Transfer Tube | SR | SCS1 |
| Pin Removal Cell | SR | SCS1 |
| Core Assembly Pot | SR | SCS1 |
| EVHM | NSRST | SCN3 |
| BLTC | NSRST | SCN3 |
| EVST | NSRST | SCN3 |
| Pin Extraction Machine | NSRST | SCN3 |
| Pin Removal Cell | NSRST | SCN3 |
| Core Assembly Pot | NSRST | SCN3 |


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**Table 7.3.2-2 Codes and Standards**

| Code | Version | Title |
|---|---|---|
| ASME BPVC, Section III Division 5 | 2017 | Boiler and Pressure Vessel Code |
| ASME NOG-1 | 2020 | Rules for Construction of Overhead and Gantry Cranes |
| ASME NML-1 (Reference 7.3.2-6) | 2019 | Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities Nuclear Material Lifting |
| ASME BTH-1 (Reference 7.3.2-5) | 2017 | Design of Below-the-Hook Lifting Devices |
| ASME NQA-1 (Reference 7.3.2-7) | 2015 | Quality Assurance Requirements for Nuclear Facility Applications |
| ASME B31.1 (Reference 7.3.2-8) | 2020 | Power Piping |
| ASME AG-1 (Reference 7.3.2-9) | 2017 | Code on Nuclear Air and Gas Treatment |
| ANSI/AISC N690 | 2018 | Specification for Safety-Related Steel Structures for Nuclear Facilities |
| IEEE 1023 (Reference 7.3.2-10) | 2020 | IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities |
| IEEE 1050 (Reference 7.3.2-11) | 2004 | Guide for Instrumentation and Control Equipment Grounding in Generating Stations |
| IEEE 497 | 2016 | Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations |
| MIL-STD-461G (Reference 7.3.2-12) | 2015 | Department of Defense Interface Standard: Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment |
| NFPA 70 (Reference 7.3.2-13) | 2020 | National Electrical Code |
| IEC 61000-3 (Reference 7.3.2-14) | 2017 | Electromagnetic Compatibility (EMC) - Part 3: Limit - All Parts |


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**Figure 7.3.2-1 Ex-Vessel Handling Machine Primary Components**

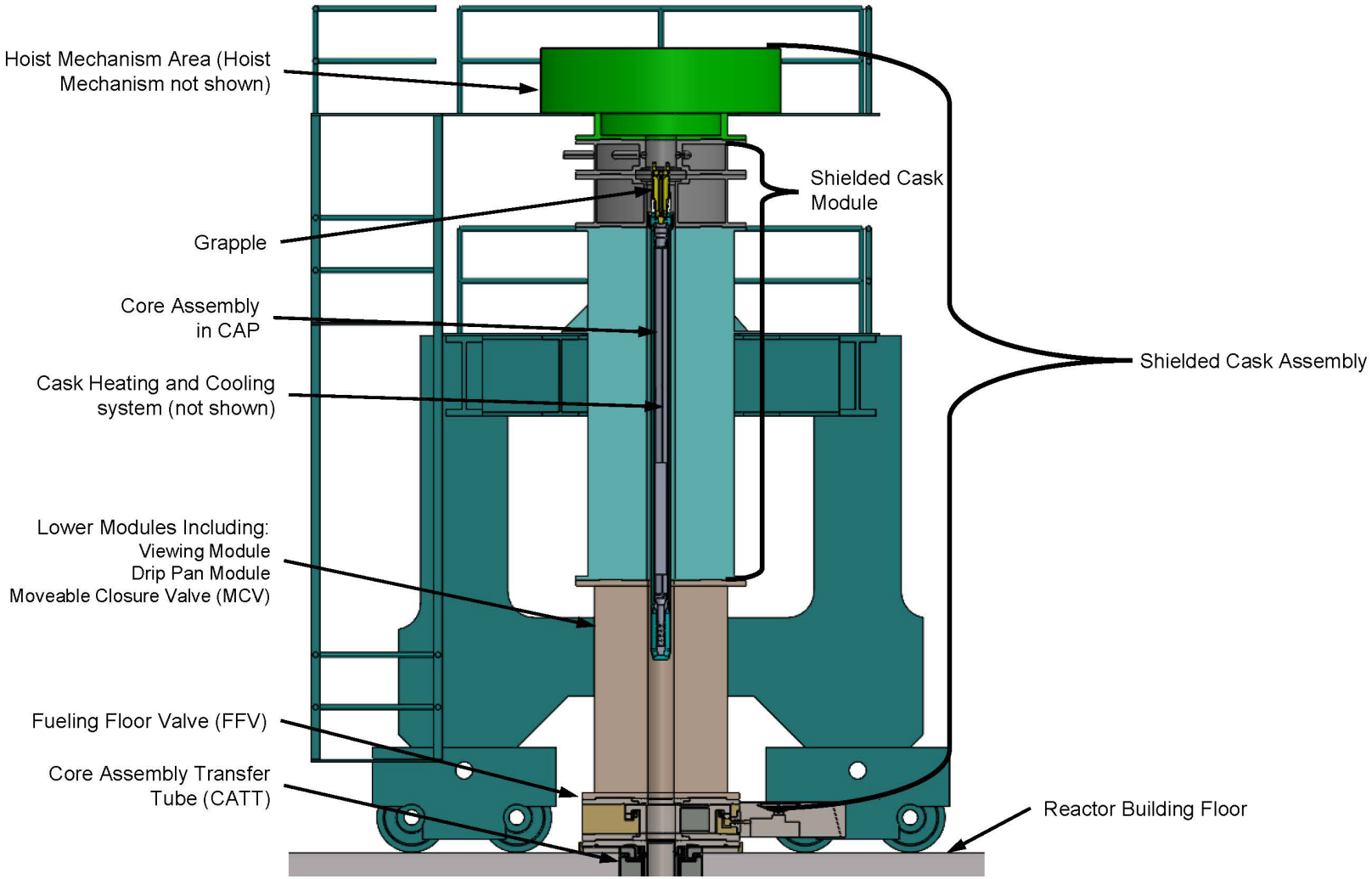




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**Figure 7.3.2-2 Bottom Loading Transfer Cask Primary Components**

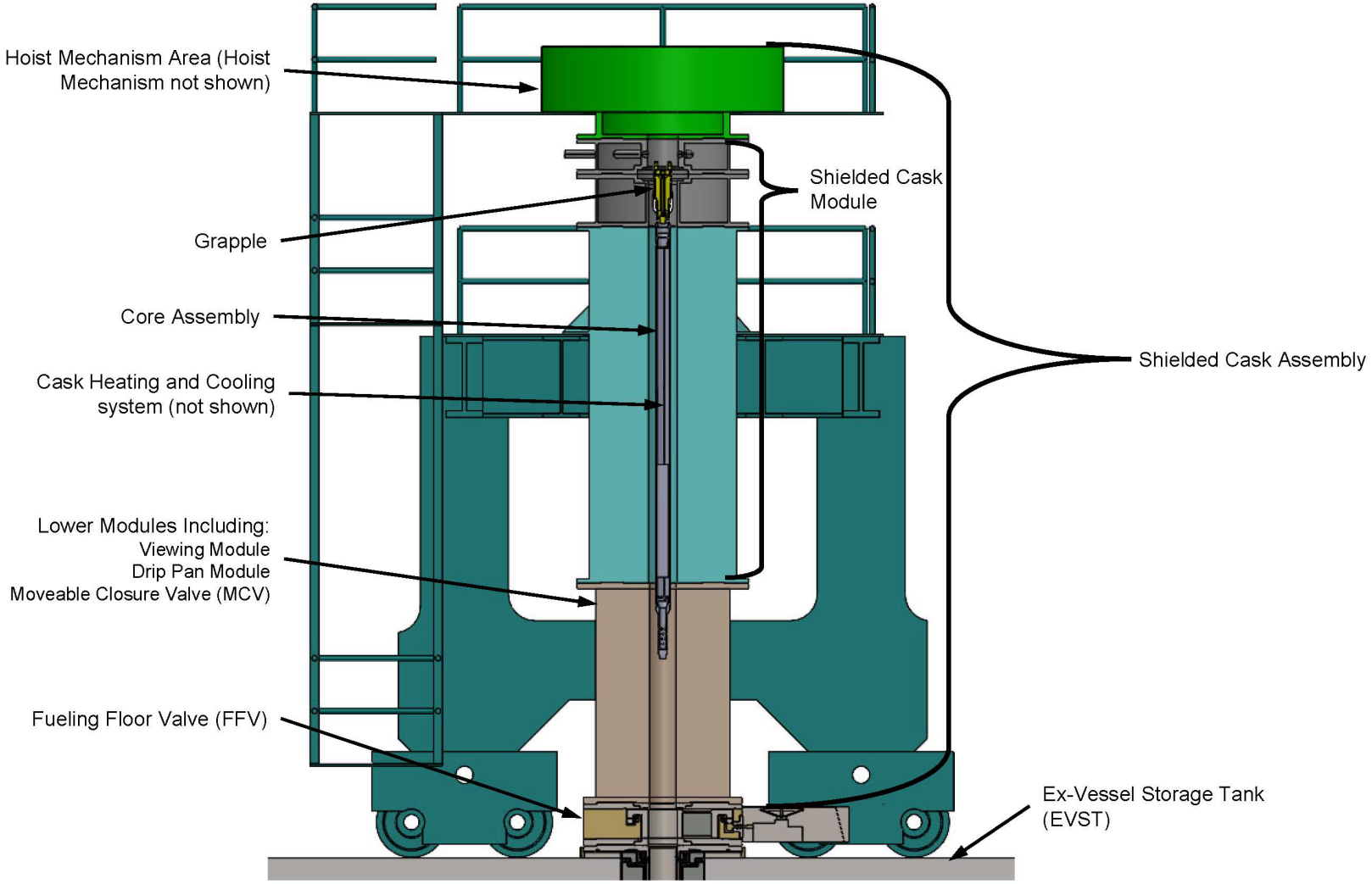



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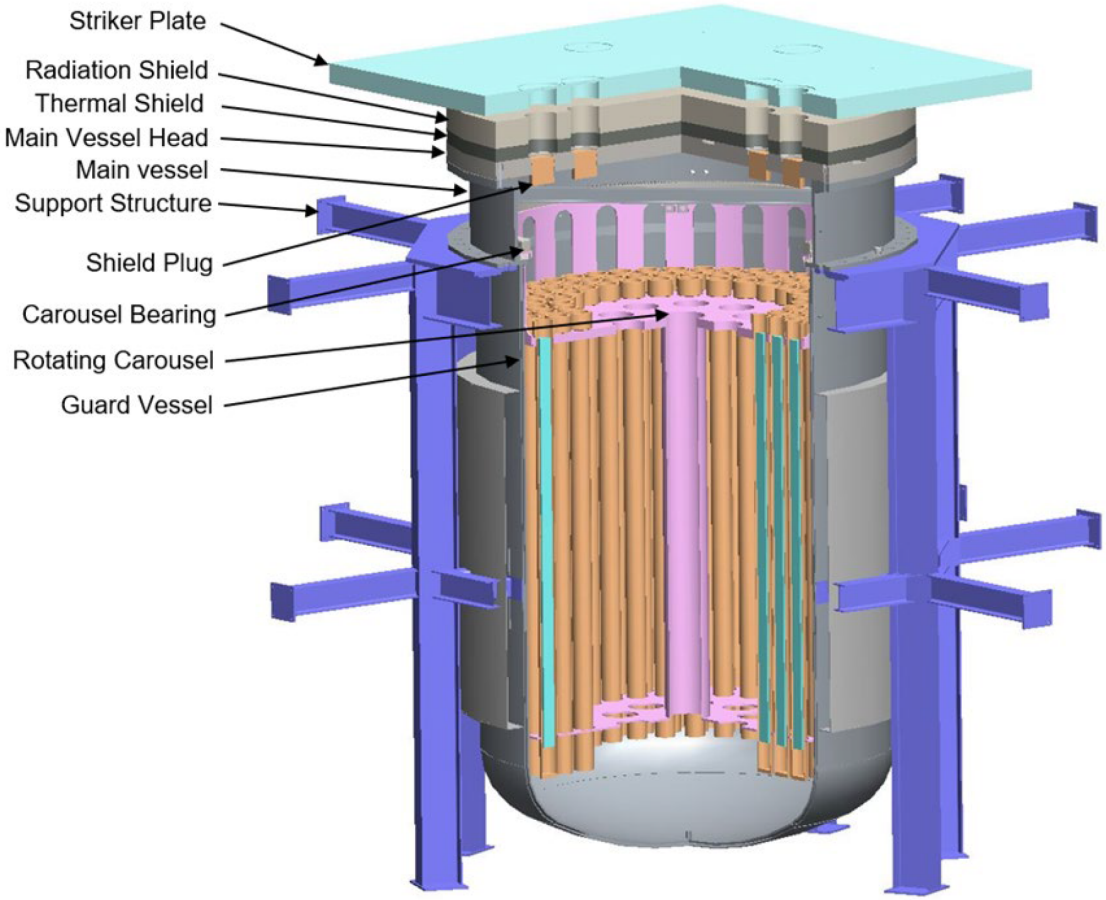
**Figure 7.3.2-3 Ex-Vessel Storage Tank Primary Components**



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## 7.3.3 In-Vessel Fuel Handling System

### 7.3.3.1 Summary Description

The In-Vessel Fuel Handling System (FHI) includes the equipment necessary to manipulate the positions of core assemblies (Section 7.1.1) during Mode 4 refueling operations and maintain the functional containment boundary when the equipment is installed. FHI equipment interfaces with the Ex-Vessel Fuel Handling System (FHE) (Section 7.3.2) to transfer core assemblies into and out of the reactor vessel. Refer to Section 7.1.2 for information pertaining to the reactor vessel.

The FHI design provides for the ability to:

- shuffle core assemblies within the reactor vessel (including core locations and In-Vessel Storage locations)
- move core assemblies to or from In-Vessel Storage
- receive core assemblies from the FHE for placement into the reactor vessel
- discharge core assemblies from the reactor vessel to the FHE

The FHI is comprised of six major components; In-Vessel Transfer Machine (IVTM), Rotatable Plug Drive (RPD), Rotatable Plug Jack (RPJ), Fuel Transfer Lift (FTL), Fuel Transfer Lift Shield Plug (FTLSP), and In-Vessel Drive Controller (IVDC). Figure 7.3.3-1 shows the primary FHI equipment positioned within the reactor vessel and on the reactor head.

In-Vessel Transfer Machine

The IVTM is a multi-axis pantograph type fuel handling machine, mounted to the Rotatable Plug Assembly (RPA). The IVTM in combination with the rotation of the RPA and motion control of the IVDC transfers core assemblies between the core, In-Vessel Storage, and the FTL within the reactor vessel.

The IVTM is equipped with a mechanical grapple assembly used to grip the handling socket at the top of a core assembly. The IVTM grapple and pantograph are located in the reactor vessel and are manipulated and monitored via control shafts. The control shafts extend from the IVTM drive assembly located in the HAA, where they are instrumented, into the reactor vessel where they operate various mechanical elements. The IVTM handles a single core assembly at a time and maintains a vertical orientation of the handled core assembly during withdrawal, transfers, and insertions. The IVTM grapple finger is part of the research and development program as presented in Chapter 13.

The IVTM is installed on the RPA at the beginning of a refueling outage once the plant enters Mode 4 and performs refueling operations per the refueling outage plan. The IVTM is removed at the end of a refueling outage while the plant is still in Mode 4 operation.

When the IVTM is installed it forms part of the functional containment boundary. When the IVTM is not installed in the reactor it is replaced with a shield plug.

Figure 7.3.3-2 depicts the IVTM Primary Components and motions.


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In-Vessel Drive Controller

The IVDC is a control system that integrates the motor controllers, field sensor data acquisition equipment, field control device signal output equipment, a local human-machine interface, and a programmable controller that provides local control for the IVTM and the RPD.

The Fuel Handling Supervisory Control System (FHC) (Section 7.6.2) communicates with the IVDC to provide commands for FHI fuel handling operations. The IVDC takes those commands and translates them into the required motions for the IVTM and RPD components to execute.

The IVDC controls and enforces interlocks and the dynamically updated safe operating zones of the IVTM based on the states communicated by the FHC, the current position of the IVTM, and the states of the various interlocks and safety systems.

The IVDC supports control of IVTM and RPD operations which range from remote automatic (which allows control remotely via the FHC from the control room) to manual operation in support of off-normal operations.

Fuel Transfer Lift

The FTL is a vertical lift used to raise and lower core assemblies, which are located inside of a Core Assembly Pot (CAP), into and out of the reactor. The FTL raises and lowers one core assembly into and out of the reactor at a time. The FTL has two elevation positions – a raised elevation position and a lowered elevation position. At the raised elevation, the FTL interfaces with the EVHM grapple. At the lowered elevation position, the FTL interfaces with the IVTM grapple. The core assembly pot and EVHM are part of the FHE and are discussed in Section 7.3.2. Figure 7.3.3-3 depicts the components that comprise the FTL.

The FTL is installed in the FTL nozzle at the beginning of a refueling outage once the plant enters Mode 4. The FTL is supported by the FTL nozzle on the reactor head. The FTL is removed at the end of a refueling outage while the plant is still in Mode 4 operation. The FTL is not installed during non-outage periods.

When the FTL is installed it forms part of the functional containment boundary. When the FTL is not installed in the reactor, it is replaced with a shield plug.

Rotatable Plug Jack

The RPJ is a jacking system which lifts the RPA off its static interface with the reactor head thereby transferring the weight onto the RPA bearing and allowing for rotation. The mechanical portion of the jacking assembly includes electro-mechanical jacks. The RPJ is mounted on and supported by the reactor head.

Rotatable Plug Drive

The RPD is a drive assembly which provides the motive force to rotate and restrict motion of the RPA in support of performing refueling maneuvers. The RPD is controlled by the IVDC and in conjunction with the IVTM positions the IVTM grapple within the reactor during refueling. The RPD is supported by the reactor head.


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Fuel Transfer Lift Shield Plug

The FTLSP is inserted into the FTL bore and comprises part of the functional containment boundary during installation and removal of the FTL when the core assembly transfer tube and fueling floor valve are not installed (Section 7.3.2). Refer to Figure 7.3.3-3 which depicts the location of the FTL bore. To maintain the functional containment boundary the FTLSP utilizes a redundant pair of static seals with argon barrier gas supplied between the seals. The FTLSP is removed during refueling to facilitate the insertion and removal of core assemblies from the reactor through the FTL.

### 7.3.3.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the FHI design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the FHI are described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, The FHI is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The FHI is within, and protected by, the reinforced concrete Reactor Building (RXB) substructure which is described in Section 7.8.1.1. The location of the FHI within the RXB provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions. The FHI is designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1. Design-basis hazard levels are presented in Section 6.1.1. FHI components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions.

Consistent with PDC 3, *Fire protection*, the design of the FHI minimizes the probability and effect of fires through design features that minimize the potential for liquid sodium leakage to the surroundings as well as the selection of non-combustible and non-fire sustaining materials to the extent practical. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the FHI is designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents. FHI equipment is designed using materials that are selected to be capable of withstanding the operating environments to which they are exposed. FHI safety-related structures, systems, or components (SSCs) within the ASME jurisdictional boundaries are designed per the requirements of ASME BPVC, Section III, Division 5, 2017, (Reference 7.3.3-1) as endorsed with exceptions and clarifications in Regulatory Guide (RG) 1.87, “Rules for Construction of Nuclear Facility Components, Division 5-High Temperature Reactors,” Revision 2, to ensure material compatibility with liquid sodium and its aerosols, vapors, and


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oxidation products. Equipment qualification for harsh environmental conditions resulting from accident conditions is provided by the FHI component quality, design, and qualification requirements in accordance with the Equipment Qualification Program (described in Chapter 8).

The location of the FHI within the RXB substructure provides protection from external hazards caused by transportation incidents (including external flammability and explosion toxicity hazards), offsite industrial and chemical facilities (including nearby manufacturing plants, chemical plants, refineries, storage facilities, mining and quarry operations, oil and gas pipelines, drilling operations, wells, and underground gas storage facilities).

The FHI is protected from the dynamic effects, including missiles and discharging fluids, that could result in equipment failures and from events and conditions outside the NI.

Consistent with PDC 13, *Instrumentation and control*, the design of the FHI locates, positions, and structurally supports instrumentation needed to monitor variables over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to ensure adequate safety. The design of the FHI instrumentation and controls in conjunction with the FHC, control the calibration of the IVTM, in-vessel motion, as well as core assembly identification which is obtained using the core assembly identification pawl. The FHI also monitors functional containment boundary seal barrier gas. Section 7.6.2 provides the description of the protection and control systems provided to maintain these variables within prescribed operating ranges.

Consistent with PDC 16, *Containment Design*, the design of the FHI supports functional containment (further described in Section 1.3.2.1) to control the release of radioactivity to the environment and provide assurance that functional containment design conditions are not exceeded for as long as postulated accident conditions require. The FHI SSCs form part of the functional containment boundary during Mode 4.

FHI functional containment seals are individually pressurized to allow each seal to be monitored and tested for seal integrity.

Consistent with PDC 61, *Fuel storage and handling and radioactivity control*, the FHI is designed to permit periodic inspection and testing of safety-significant components and is suitably shielded for radiation protection. The FHI design includes appropriate confinement and containment of core assemblies within the Core Assembly Pot.

The FHI is remotely operated during in-vessel refueling operations for radiation protection. The FHI is designed to limit radiation levels in the Head Access Area from core assembly handling operations by shielding integral to the FTL main structure and drive assembly. The FHI penetration and interface seals provide a functional containment boundary while the FTL and IVTM are installed to the reactor head during refueling operations. The FHI design provides residual heat removal from core assemblies within the liquid sodium filled Core Assembly Pot.

Consistent with PDC 62, *Prevention of criticality in fuel storage and handling*, criticality in the fuel storage and handling systems is prevented by physical systems or processes which include the use of geometrically safe storage configurations.The FHI core assembly manipulations involve the handling of a single core assembly at a time.


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Functional Requirements and Design Criteria

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as Safety-Related (SR) functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The FHI supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classification of FHI components relied upon to perform the function:

| Function ID | Function Description | FHI Functional Support |
|---|---|---|
| DL3-RR1f | FHI Seal (SR) | The FHI seals to the head perform part of the primary coolant boundary barrier function. |

The FHI provides a temporary structural support SR and NSRST load path function for the core assemblies being transferred, charged or discharged in the reactor vessel, that do not require continuous load path structural support. The FHI structures and components support the weight of core assemblies during fuel movement. Information pertaining to the design of the FHI is provided in Section 7.3.3.3.

Safety classifications for FHI structures and components are listed in Table 7.3.3-1.

Regulatory Guidance

Regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the FHI, along with a description of the use and conformance to the guidance, is provided below:

RG 1.100, Revision 4, *Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants*

Partial conformance, RG 1.100, describes methods for use in the seismic qualification of electrical and active mechanical equipment and the functional qualification of active mechanical equipment for nuclear power plants. This RG addresses revisions to IEEE Standard 344-2013, IEEE Standard C37.98-2013, and ASME QME-1-2017.

RG 1.244, Revision 0, *Control of Heavy Loads at Nuclear Facilities*

Full Conformance, RG 1.244, provides guidance for the control of heavy loads. Guidance includes providing protection against equipment failure that could result in a heavy load drop.

RG 1.87, Revision 2, *Acceptability of ASME Code Section III, Division 5, "High Temperature Reactors"*

Full Conformance, ASME Section III, Division 5 is implemented, as applicable, to FHI SSCs as endorsed with limitations and restrictions.

Seismic Classification and Design Codes

Seismic classifications for FHI components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based upon seismic classification. The seismic classification of FHI components are listed in Table 7.3.3-1. Seismic



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design, analysis, and qualification is performed as described in Section 6.4. SR seismic risk significant (SRS) FHI components are qualified to withstand seismic loads associated with the safe shutdown earthquake (Section 2.6.2) without loss of the capability to perform SR functions. Seismic special treatments are applied to NSRST portions of the FHI, based on seismic classification, to provide assurance that risk-significant functions or functions required for adequate defense-in-depth are capable of being performed.

Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of FHI components as special treatments, as described in Section 6.3. Codes and standards that are applicable to the FHI are listed in Table 7.3.2-2.

#### 7.3.3.3 System Design and Performance Evaluation

The FHI is designed to meet design basis requirements described in Section 7.3.3.2 and evaluated to provide assurance that associated performance requirements are met. Design features are provided by the FHI design to support the identified safety-significant functions and to provide assurance of compliance with applicable regulatory criteria.

The safety-significant portions of the FHI are protected from design-basis hazards. Design-basis hazard levels are presented in Section 6.1.1. Protection from flooding, tornadoes, high-winds and wind-driven missiles, extreme climatic conditions, and external fires is provided by the location below-grade within the RXB (Section 7.8.1.1).The location within the RXB substructure also affords FHI protection from local transportation incident-related hazards, offsite industrial and chemical facilities, and turbine failure-related missiles initiating from the plant turbine-generator set.

<u>In-Vessel Transfer Machine</u>

The IVTM main guide tube is the primary structural body of the in-vessel subassembly and transfers loading from the grapple and pantograph assembly to the RES structures, specifically the upper internal structure and the RPA. The bottom of the IVTM main guide tube is supported radially by a pin at the bottom of the upper internal structure. Some lateral refueling loads are transferred back to the upper internal structure through this support. The IVTM is installed in a penetration in the RPA and interfaces via a structural nozzle riser which supports the weight of the IVTM vertically and resolves vertical loads induced during refueling operations.

Although IVTM lifts do not meet the definition of critical lifts per ASME NML-1-2019, “Rules For The Movement Of Loads Using Overhead Handling Equipment In Nuclear Facilities,” (Reference 7.3.3-2), the IVTM lifting mechanisms are designed using ASME NML-1-2019 for nuclear safety critical lifts. This enhances operational safety.

The IVTM drive assembly contains motors, gearboxes, actuators, couplings, sensors, and any other hardware and instrumentation to provide the motive force and control required to support IVTM functions. The drive assembly is located outside of the reactor vessel, on top of the in-vessel subassembly, in the Head Access Area. The drive assembly receives power, signal, and inert gas supply from plant systems via a flexible overhead cable management system.


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The in-vessel subassembly includes all control shafts and in-vessel moving components located within the reactor functional containment boundary.

Fuel Transfer Lift

The FTL is mounted on and supported by the RES. The FTL is supported from the FTL nozzle and extends out of the reactor head. The top ring of the FTL nozzle supports the vertical weight of the FTL through the FTL head. The FTL extends into the reactor, below the top of the core. The penetration in the top of the In-Vessel Storage for the FTL transfers the lateral loads imparted onto the FTL.

FTL lifts do not meet the definition of critical lifts per ASME NML-1-2019 however the FTL lifting mechanisms are configured with enhanced safety features and increased safety factors associated with critical lifts using ASME NUM-1-2016, “Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type,” (Reference 7.3.3-3) requirements for Type 1 equipment per ASME NML-1-2019.

Functional Containment Boundary Seals

The IVTM and FTL make up part of the functional containment boundary during refueling operations. Components which form part of the functional containment boundary on the IVTM and FTL are classified as SR.

The IVTM and FTL utilize both static and dynamic seals, and structural housings, to maintain part of the functional containment boundary during refueling. Static seals are utilized between stationary components such as structural housings and nozzles, and dynamic seals are utilized between control shafts, position shafts, and their supporting structures. All seals are configured in a redundant sealing arrangement with a primary and secondary seal to provide increased assurance that FHI equipment maintains the reactor functional containment boundary. In addition to redundant sealing configurations, each volume between the primary and secondary seal is supplied with pressurized argon barrier gas from the SCG (Section 7.2.3). The pressurized argon barrier gas is maintained at a pressure slightly higher than the reactor cover gas system to ensure radionuclides do not escape in the event of a seal failure. The seal barrier gas is monitored by the FHC (Section 7.6.2) and IVDC.

Pressure testing is performed in accordance with the requirements of ASME BPVC Section III, Division 5.

Fuel Transfer Lift Shield Plug

The FTLSP is supported by the FTL bore that sits at the same level as the reactor head. Thermal and radiological shielding plates extend below the FTLSP.

The FTLSP is inserted into the FTL bore and makes up part of the functional containment boundary during installation and removal of the FTL. Refer to Figure 7.3.3-3 which depicts the location of the FTL bore.


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When the FTL is installed, the FTLSP is used to seal its inner bore. After the FTL is installed, the core assembly transfer tube and fueling floor valve are installed and the FTLSP is subsequently removed from the FTL. The core assembly transfer tube and fueling floor valve extend the functional containment boundary up to the refueling floor to facilitate refueling operations. After refueling is complete the FTLSP is reinstalled and the fueling floor valve and core assembly transfer tube are removed.

FHI safety-related SSCs within the ASME jurisdictional boundaries are designed per the requirements of ASME BPVC Section III, Division 5. Sodium wetted structural components are made from 316 Stainless Steel and Alloy 718 or other materials with properties to ensure material compatibility with liquid sodium and its aerosols, vapors, and oxidation products.

#### 7.3.3.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the FHI is described in Section 7.3.3.2 and Section 7.3.3.3. The following programmatic special treatments are preliminarily identified as applicable to the FHI to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Reliability and Integrity Management Program
- Equipment Qualification Program
- Post-Construction Inspection, Testing, and Analysis Program (Chapter 12)
- Comprehensive Vibrational Assessment Program

#### 7.3.3.5 Instrumentation Requirements

The IVTM and FTL contains multiple sensors to monitor and report position, load, and other status information to the FHC.

During normal operation the FHC, combined with the IVDC, provides interlocks and remote automatic control of the FHI equipment and sub-systems. In the event of off-normal operations, manual controls are available.

Primary Operator interaction with the FHC and Fuel Handling systems is through multiple human-machine interface units located in the FHB control room. The human-machine interface units are part of the PMC. The FHC provides the interfaces needed for the fuel handling operators to monitor and control the various fuel handling SSCs remotely from the FHB control room. This includes process, alarm, setpoints and command data exchange with the PMC system’s human-machine interfaces that are in the FHB Control Room. The PMC human-machine interfaces provide the operators the window to the process for the controlling


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and monitoring of the fuel handling SSCs from the FHB control room. The FHC provides supervisory, control and data acquisition functions for the fuel handling SSCs. The FHC is a non-safety related with no special treatment system.

The movement of core assemblies is accomplished through the coordination of individual machines and systems that are connected to the FHC to form an interconnected fuel handling system. The FHC provides supervisory, control and data acquisition functionality to orchestrate, monitor and control machine movements to support the initial fuel loading of the reactor followed by subsequent refueling of the reactor, placement of non-fuel core assemblies and for the movement of spent fuel and components from the reactor.

**References**


7.3.3-1 American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Facility Components, Division 5, High Temperature Reactors,” ASME, 2017

7.3.3-2 American Society of Mechanical Engineers, NML-1, Rules For The Movement Of Loads Using Overhead Handling Equipment In Nuclear Facilities, 2019.

7.3.3-3 American Society of Mechanical Engineers, NUM-1, Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type, 2016.

7.3.3-4 American Nuclear Society 57.1, “Design Requirements for Light Water Reactor Fuel Handling System,” ANS, 1992

7.3.3-5 American Society of Mechanical Engineers, “Rules for Inservice Inspection of Nuclear Power Plant Components,” Section XI, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,” ASME, 2019

7.3.3-6 International Electrotechnical Commission 61000-3, “Electromagnetic Compatibility (EMC) - Part 3: Limit - ALL PARTS,” IEC, 2017

7.3.3-7 Institute of Electrical and Electronics Engineers 1023, “IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities, Institute of Electrical and Electronics Engineers,” IEEE, 2020

7.3.3-8 Institute of Electrical and Electronics Engineers 1050, “Guide for Instrumentation and Control Equipment Grounding in Generating Stations,” IEEE, 2004



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**Table 7.3.3-1 FHI Safety-Significant Components Classifications**

| FHI SSC | Safety Class | Seismic Classification |
|---|---|---|
| IVTM | SR | SCS1 |
| FTL | SR | SCS1 |
| FTLSP | SR | SCS1 |
| IVTM | NSRST | SCN3 |
| FTL | NSRST | SCN3 |
| Rotatable Plug Jack (Chocks) | NSRST | SCN3 |



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**Table 7.3.3-2 Codes and Standards**

| Code | Version | Title |
|---|---|---|
| ANS 57.1 (Reference 7.3.3-4) | 1992 | Design Requirements for Light Water Reactor Fuel Handling System |
| ASME BPVC, Section XI, Division 2 (Reference 7.3.3-5) | 2019 | Rules for Inservice Inspection of Nuclear Power Plant Components, Section XI, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants |
| ASME BPVC, Section III, Division 5 | 2017 | Rules for Construction of Nuclear Power Plant Components, Division 5: High Temperature Reactors |
| ASME NML-1 | 2019 | Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities |
| ASME NUM-1 | 2016 | Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type) |
| ASME BTH-1 | 2017 | Design of Below-the-Hook Lifting Devices |
| IEC 61000-3 (Reference 7.3.3-6) | 2017 | Electromagnetic Compatibility (EMC) - Part 3: Limit - ALL PARTS |
| IEEE 1023 (Reference 7.3.3-7) | 2020 | IEEE Recommended Practice for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations and Other Nuclear Facilities, Institute of Electrical and Electronics Engineers |
| IEEE 1050 (Reference 7.3.3-8) | 2004 | Guide for Instrumentation and Control Equipment Grounding in Generating Stations |


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**Figure 7.3.3-1 FHI General Arrangement within Reactor Vessel**

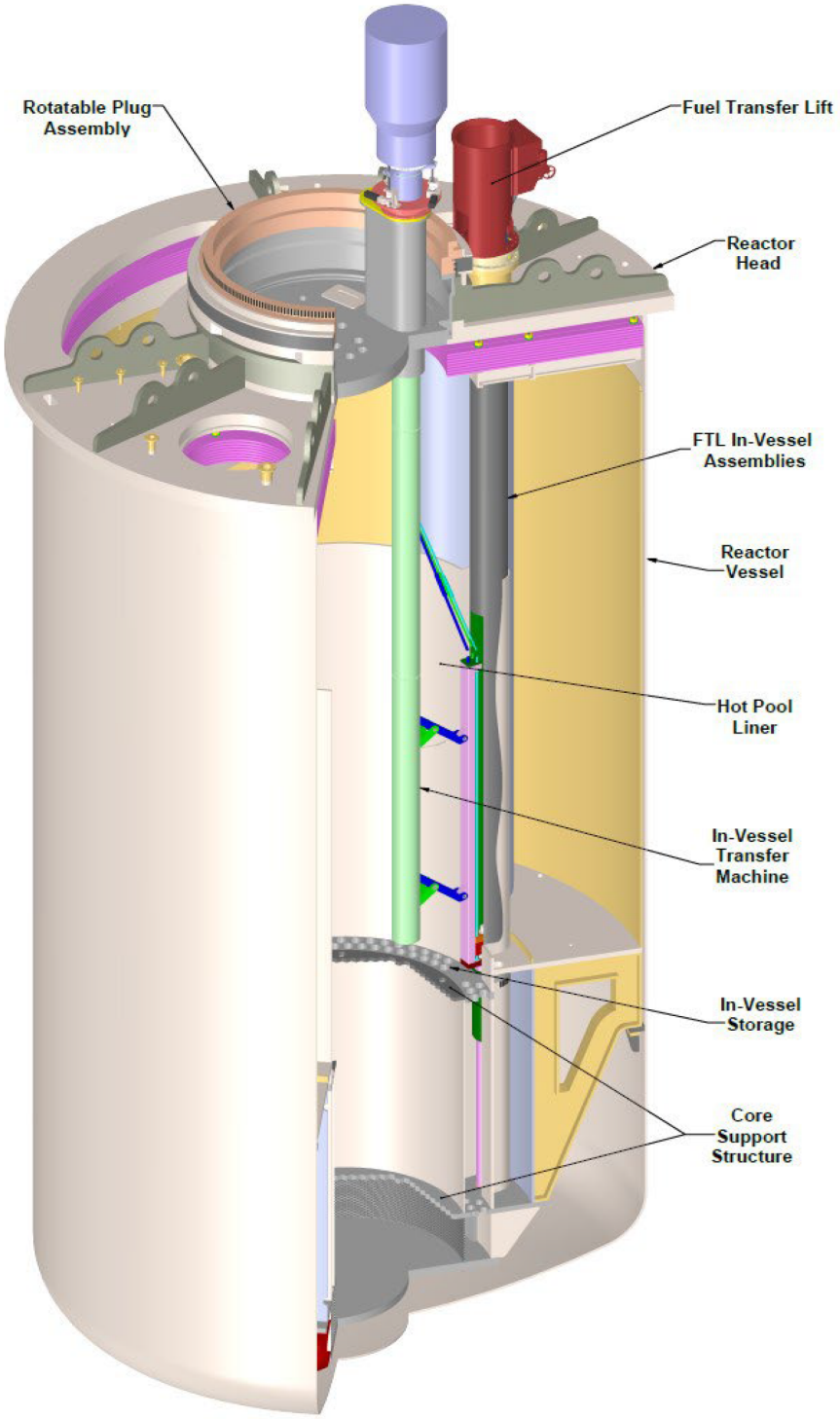




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**Figure 7.3.3-2 IVTM Primary Components and Motions**

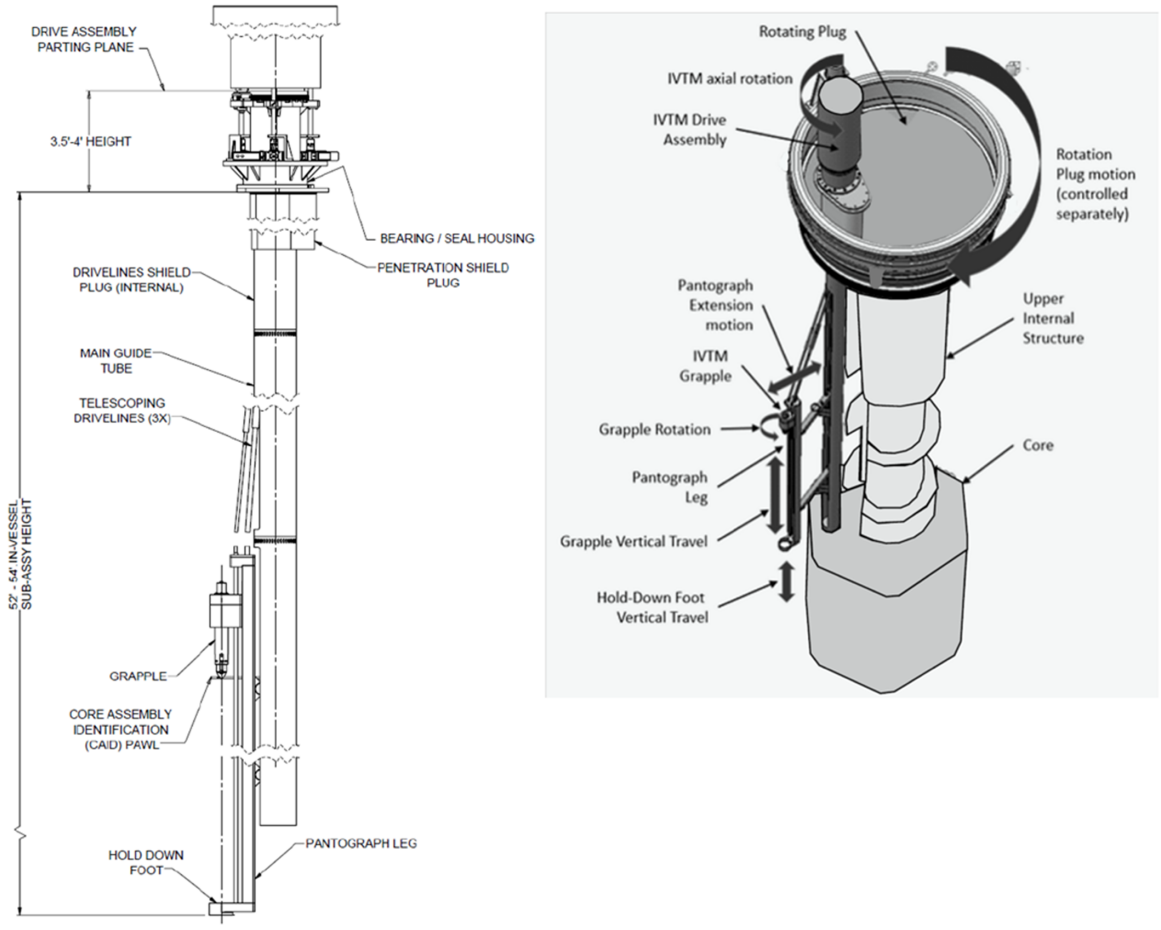



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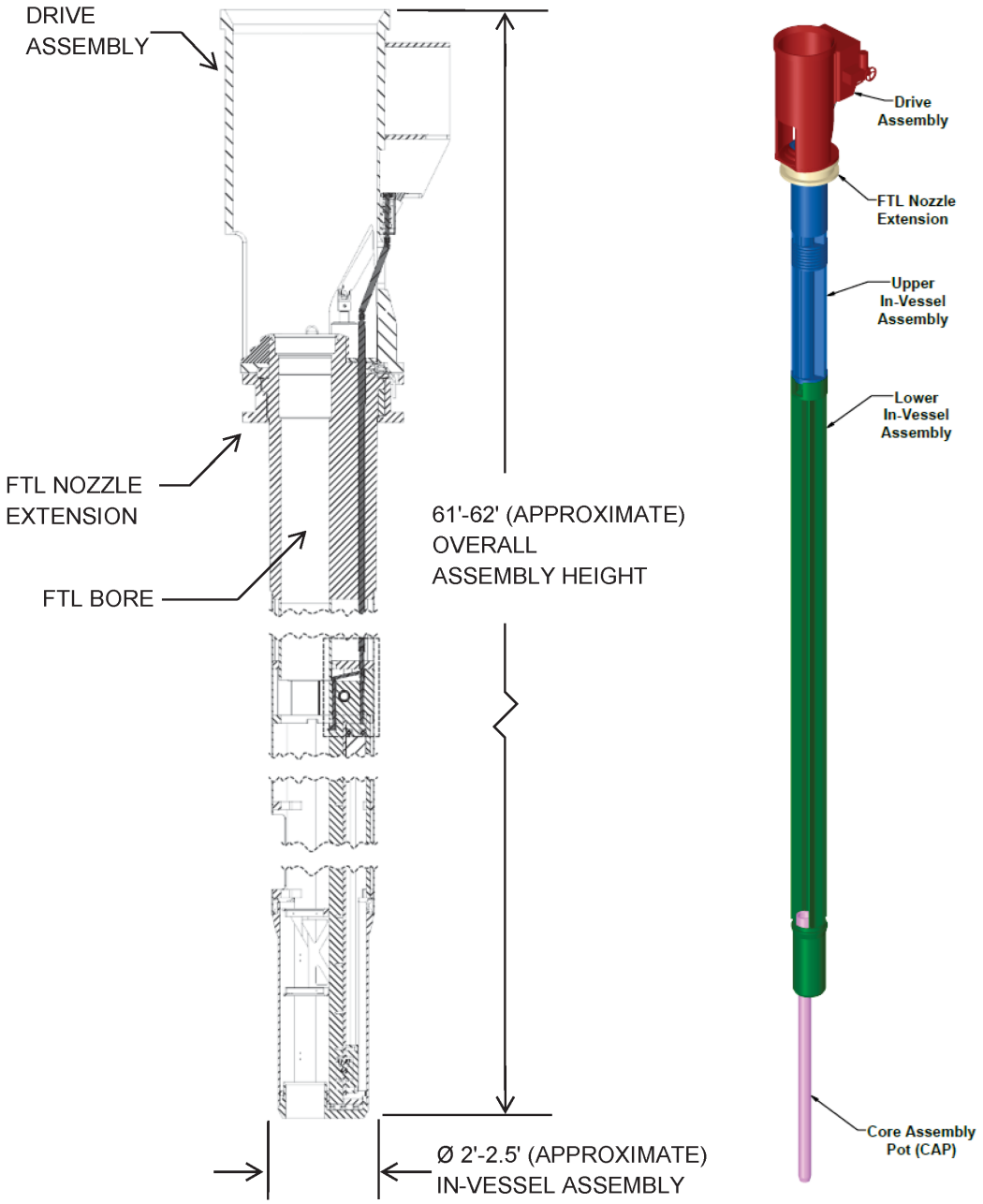
**Figure 7.3.3-3 FTL Primary Components**



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## 7.4 Radwaste Systems

The Kemmerer Unit 1 radioactive waste treatment systems are designed to meet the applicable guidance of RG 1.143, Rev. 2, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.”

The Gaseous Radwaste Processing System (RWG) is classified as non-safety-related with special treatment (NSRST) and is described in Section 7.4.1.

The Liquid Radwaste Processing System is classified as non-safety-related with no special treatment (NST) and is described in Section 9.1.

The Solid Radwaste Processing System is classified as NST and is described in Section 9.3.

### 7.4.1 Gaseous Radwaste Processing System

#### 7.4.1.1 Summary Description

The RWG processes the gaseous waste streams from the primary Sodium Cover Gas System and the Sodium Cover Gas System exhaust distribution cabinet as discussed in Sections 7.2.3 and 7.3.2. The system provides holdup for the decay of short-lived radioactive isotopes such as argon-41. Waste gas is transported through filtration medium and activated charcoal adsorption beds, which provide enhanced decay of longer lived isotopes like xenon and krypton prior to environmental discharge via the plant stack. RWG discharge to the plant stack is diluted by airflow from Nuclear Island Heating, Ventilation, and Air Conditioning System (NHV). The combined exhaust is monitored and recorded for compliance with administrative and federal dose limits.

The RWG is classified as NSRST because it supports the retain radionuclides fundamental safety function, which protects plant personnel and the environment, supports compliance with 10 CFR Part 20, and minimizes radioactive releases. Consistent with principal design criteria (PDC) requirements, as it relates to the RWG the releases of gaseous radioactive effluents to the environment are controlled, and adequate safety is ensured under normal and postulated accident conditions. The operation or the safe shutdown of the reactor does not require the RWG or its components to be operational. The failure of the RWG does not compromise any safety-related system or component and does not prevent the safe shutdown of the plant.

The RWG design is informed with the applicable requirements of 10 CFR 50 Appendix I, and designed to comply with 10 CFR 20.1101(b), 10 CFR 20.1302, and 10 CFR 20.1301, including the effluent concentrations limits of 40 CFR 190 as implemented under 10 CFR 20.1301(e).

A simplified block flow diagram of the RWG is shown in Figure 7.4.1-1.

#### 7.4.1.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the RWG design basis are described below.


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Regulatory Criteria

PDC are identified in Section 5.3. Compliance with PDC that are applicable to the RWG is described below.

Consistent with PDC 2, *Design Basis for Protection Against Natural Phenomena*, the RWG is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. RWG equipment and components are located within the Fuel Handling Building and are housed in concrete enclosures, which provide protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions as described in Section 7.8.2. The safety-significant portions of the RWG are designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1.

Consistent with PDC 3, *Fire Protectio*n, the RWG is designed to minimize the probability and effect of fires through the selection of non-combustible and non-fire sustaining materials, to the extent practical. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and Dynamic Effects Design Bases*, the RWG is designed, constructed, and tested to maintain radiological retention during the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences (AOOs), and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. In addition, RWG equipment and components are located within the Fuel Handling Building and housed in concrete enclosures protecting structure, system, or components from dynamic effects including the effects of missiles, pipe whipping, and discharging fluids that may result in equipment failures and from events and conditions outside the nuclear power unit.

Consistent with PDC 60, *Control of Releases of Radioactive Materials to the Environment*, the RWG is designed to control the release of gaseous effluents during normal reactor operation, including AOOs. The system provides holdup for the decay of short-lived radioactive isotopes. Waste gas is transported through filtration medium and activated charcoal adsorption beds, which provide enhanced decay of longer lived isotopes. The RWG effluent is combined with the NHV (described in Section 7.5.1) exhaust flow, which is monitored.

Consistent with PDC 61, *Fuel Storage and Handling and Radioactivity Control*, the RWG is designed to permit periodic inspection and testing of safety-significant equipment and components. Cubicles and rooms housing RWG equipment and components are shielded as necessary for radiation protection. The RWG is designed to retain radionuclides and to filter particulates from gaseous effluent during normal operation and AOOs.

Consistent with PDC 63, *Monitoring Fuel and Waste Storage*, the RWG design includes instrumentation to detect conditions that may result in excessive radiation levels and to initiate appropriate actions.

Consistent with PDC 64, *Monitoring Radioactive Releases*, the RWG effluent discharge path is continuously monitored.


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Functional Requirements and Design Criteria

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as safety-related (SR) functions or NSRST functions as defined in Section 5.2. The RWG supports the following safety-significant functions, which are designated as NSRST to indicate the safety classification of RWG components relied upon to perform the function:

| | | |
|---|---|---|
| DL2-RR4 | Retain Radionuclide - RWG Radionuclide Boundary (NSRST) | RWG components (including seals) form a radionuclide confinement boundary which prevents the uncontrolled release of radionuclides. Additionally, leak detection and isolation are provided by the system to limit the release of radionuclides. |

Regulatory Guidance

RWG structure, system, or components conform to the following regulatory guides:

RG 1.109, Rev. 1, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I*

Full Conformance. Gaseous pathway doses to the maximally exposed individual using the currently NRC-approved computer code GASPAR II (NUREG/CR-4653), at the nearest residence, garden, and meat animal and the exclusion area boundary meet the guidance of RG 1.109.

RG 1.110, Rev. 1, *Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors*

Full Conformance. A cost-benefit analysis is performed as required by 10 CFR 50.34a and 10 CFR 50, Appendix I, to demonstrate conformance with regulatory requirements. The cost-benefit analysis is performed using the guidance of RG 1.110.

RG 1.112, Rev. 1, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors*

Full Conformance. The calculation for RWG source terms for normal operation uses the base dynamic adsorption coefficients for xenon and krypton as provided in NUREG-0017. The calculation assumes no adsorption for any other gases. As a result, the source term methodology satisfies the requirements of RG 1.112.


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RG 1.140, Rev. 3, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants*

Full Conformance. The RWG is a power cycle waste offgas system that relies on activated carbon delay beds and high efficiency particulate air filtration; the criteria outlined in RG 1.140 are incorporated into the system's applicable design features.

RG 1.143, Rev. 2, *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants*

Partial Conformance. The RWG is designed in conformance with RG 1.143, sections C.2 and C.4 including the codes and standards referenced by those sections and the application of RG 8.8.

The methodologies described in Section 5.1 and Section 6.1.3 replace the methodologies of Regulatory Positions C.5 and C.6 respectively.

RG 8.8, Rev. 3, *Operating Philosophy for Maintaining Occupational and Public Radiation Exposures as low as reasonably achievable*

Full Conformance. The RWG is designed with features in accordance with RG 8.8, including shielded cubicles and remotely operated equipment to maintain worker radiation exposures as low as reasonably achievable during normal operation, inspection, testing and maintenance.

Seismic Classification and Design Codes

Seismic classifications for RWG components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The NSRST portions of RWG are designed to seismic classification SCN3. Seismic design, analysis, and qualification is performed as described in Section 6.4. Seismic special treatments are applied to NSRST portions of the RWG to provide assurance that risk-significant functions or functions required for adequate defense-in-depth are capable of being performed.

Codes and standards are applied for the design, fabrication, testing, inspection, and maintenance of RWG components as special treatments, as described in Section 6.3. Codes and standards applicable to the RWG are listed in Table 7.4.1-1.

### 7.4.1.3 System Design and Performance Evaluation

#### 7.4.1.3.1 System Description

The RWG is located below grade in the Fuel Handling Building, as described in Section 7.8.2, and is a once-through, ambient temperature charcoal adsorber delay system that receives radioactive gaseous waste. The RWG filters particulates, provides holdup and adsorption to allow radioactive decay of short and long lived isotopes, and discharges to the NHV plant exhaust stack for release to the environment as a monitored release.


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Waste gas inputs are received into the RWG vacuum tank. Radiation monitoring is provided at the inlet of the vacuum tank. Vacuum is created by one of two compressors, which are instrumented to automatically maintain the vacuum. The tank provides a short-term source of vacuum to allow continued flow of waste gas during an interruption of compressor service.

The compressors are arranged in parallel, and their controls are such that a single compressor starts when vacuum pressure reaches a setpoint. In the event of a compressor failure, the failed compressor is isolated and the alternate compressor starts, allowing maintenance as needed without shutting down the remainder of the RWG.

A particulate filter is located upstream of each RWG compressor. The vessels housing the filters are designed to ASME Section VIII (Reference 7.4.1-2). The filters are designed, constructed, and tested consistent with ASME AG-1b-2009 (Reference 7.4.1-3) using the guidance in RG 1.140. Individual radioactive gas filters, upstream of each compressor, can be isolated and replaced when a high differential pressure exists due to excessive particulate buildup.

Waste gas is transported from a compressor through a stainless steel shell-and-tube aftercooler. The waste gas stream (tube side) is cooled by chilled water (shell side) from Nuclear Island Water System. Each aftercooler is mated with its corresponding compressor as part of the compressor skid. If the temperature of the waste gas from the coolers exceeds a high setpoint, indicating inadequate cooling, an alarm will be triggered to alert the operator to the abnormal condition.

Waste gas is transported from the aftercooler to the RWG holdup tank, which is designed to delay flow for 24 hours allowing decay of argon-41. The outlet flow from the holdup tank is regulated by a pressure reducing valve to prevent high pressure within the charcoal adsorber beds. The RWG holdup tank is an ASME Section VIII stainless steel vessel located in an individual, shielded room. The tank is baffled to minimize gas mixing and maximize retention time. The tank supports variations in the incoming process flow and allows a steadier outlet flow to the downstream portion of the system.The pressure relief valve on the tank is designed to prevent tank overpressure from occurring under maximum flow conditions.

Waste gas is directed from the holdup tank to the charcoal adsorber beds. There are four charcoal adsorber beds configured in parallel. Flow rate into each charcoal adsorber bed is controlled by a flow control valve to maximize the efficiency of the charcoal. Entrance into, and exit from, a charcoal adsorber bed are designed to minimize the potential of charcoal loss. Each charcoal adsorber bed contains activated charcoal optimized for xenon and krypton retention. The charcoal adsorber beds contain differential pressure instrumentation and pressure relief valves, and their design includes the ability to replace the charcoal. Additionally, radiation monitors are located on the discharge piping of each charcoal adsorber bed vessel and flow is isolated in the event of high radiation. Each charcoal adsorber bed is housed within an ASME Section VIII stainless steel vessel. All four vessels are housed within a shielded room.

High efficiency particulate air filters are located upstream and downstream of each charcoal adsorber bed. The vessels containing the filters are designed to ASME Section VIII. The filters are designed, constructed, and tested consistent with ASME AG-1b-2009 using the guidance in RG 1.140 Rev 3.



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The RWG utilizes valves with fail-closed automatic isolation capability to establish a barrier and prevent the spread of contamination when undesirable conditions exist which could adversely affect system releases or equipment performance. Diaphragm valves are used throughout the RWG to minimize system leakage. Piping is seamless stainless steel with large radius bends and butt welds. The use of flanged connections is limited to certain components for ease of maintenance. Check valves are used to prevent backflow conditions and minimize the unintended spread of gaseous waste.

Processed gaseous waste is discharged to the NHV plant stack which provides the monitored effluent path to the environment.

Low voltage alternating current electrical power is provided to RWG equipment from the Nuclear Island AC Electrical Power Low Voltage System as discussed in Section 1.1.4.3.6. The motor control centers and alternating current panel boards provide power to RWG equipment. The RWG does not need power for any NSRST function.

#### 7.4.1.3.2 Modes of Operation

The RWG is operable in all modes of reactor operation. The RWG is NSRST and is not required to mitigate the consequences of design basis events or design basis accidents. The system is designed to operate during normal operations and AOOs.

#### 7.4.1.3.3 Materials and Construction

RWG equipment and components are designed, constructed, and tested to the requirements set forth in the applicable codes and standards listed in RG 1.143. Conformance with RG 1.143 ensures that RWG equipment and components are compatible with the chemical, physical, and radioactive environments expected during normal conditions and AOOs. RWG materials for pressure retaining components conform to the requirements of the specifications for materials listed in Section II of the ASME Boiler and Pressure Vessel Code (Reference 7.4.1-4), except that malleable, wrought, or cast-iron materials and plastic pipe are not used. Major components of the RWG are fabricated of stainless steel for corrosion protection.

### 7.4.1.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the RWG is described in Section 7.4.1.2. The following programmatic special treatments are preliminarily identified as applicable to the RWG to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Equipment Qualification Program
- Quality Assurance Program
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing, and Analysis Program
- Reliability and Integrity Management Program


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7.4.1.5 Instrumentation Requirements

The RWG includes the following instrumentation, controls, and displays.

- Equipment and components are designed to protect against over pressurization and to enable proper flow.
- Monitoring waste gas temperatures to ensure aftercoolers are operating properly and to enable identification of malfunctions.
- Discharge flow rate to enable adequate dispersion of gas and determination of radioactive release rates.
- Radioactivity concentrations to determine atmospheric release rates, holdup times, and equipment performance and provide for the automatic termination of releases to the atmosphere as necessary.
- Valves used for automatic termination of release are designed to fail closed.
- Effluent radiation monitoring devices continuously monitor and record gaseous radioactivity released to the atmosphere through normal release pathways.
- Radiation monitoring readouts are located in the main control room with additional readouts in specific controlled areas to facilitate system operation.

The Auxiliary Monitoring and Control System provides control signal and receives instrument data from the RWG for flow, pressure, and temperature as discussed in Section 7.6.2. The Radiation Monitoring System measures radiation levels and sends the data to the Auxiliary Monitoring and Control System and RWG as described in Section 7.6.6. Redundancy is provided where the loss of an instrument would have a significant adverse impact on the RWG.

The RWG receives control signals from the Auxiliary Monitoring and Control System. RWG controls do not require redundancy, Class 1E, or harsh environment qualification, but are seismically qualified.

**References**

7.4.1-1 ASME, “Quality Assurance Requirements for Nuclear Facility Applications”, ASME NQA-1, 2015.

7.4.1-2 ASME Boiler and Pressure Vessel Code, “Section VIII: Rules for Construction of Pressure Vessels Division 1”, ASME BPVC-VIII-1, 2021.

7.4.1-3 ASME, “Code on Nuclear Air and Gas Treatment (including 2010 Addenda 1a and 2011 Addenda 1b)”, ASME AG-1-2009.

7.4.1-4 ASME Boiler and Pressure Vessel Code, “Section II: Materials”, ASME BPVC-II, 2019.

7.4.1-5 ANSI/ANS 55.4, “Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants,” ANSI/ANS, 2007.


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7.4.1-6 ANSI/HPS N13.1, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities,” ANSI/HPS, 2021.

7.4.1-7 API 618, “Reciprocating Compressors for Petroleum, Chemical, and Gas Industry Services,” API, 2007.

7.4.1-8 American Society of Mechanical Engineers, “B31.3: Process Piping,” ASME, 2020.



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**Table 7.4.1-1 RWG Codes and Standards**

| Code/Standard | Title |
|---|---|
| ANSI/ANS 55.4, 2007 | Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants |
| ANSI/HPS N13.1, 2021 | Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities |
| API Standard 618, 2007 | Reciprocating Compressors for Petroleum, Chemical, and Gas Industry Services |
| ASME AG-1b, 2009 | Code on Nuclear Air and Gas Treatment |
| ASME NQA-1, 2015 | Quality Assurance Program Requirements for Nuclear Facilities |
| ASME B31.3, 2020 | Process Piping |
| ASME BPVC Section II, 2019 | Materials |
| ASME BPVC Section VIII, Division 1, 2021 | Rules for Construction of Pressure Vessels |


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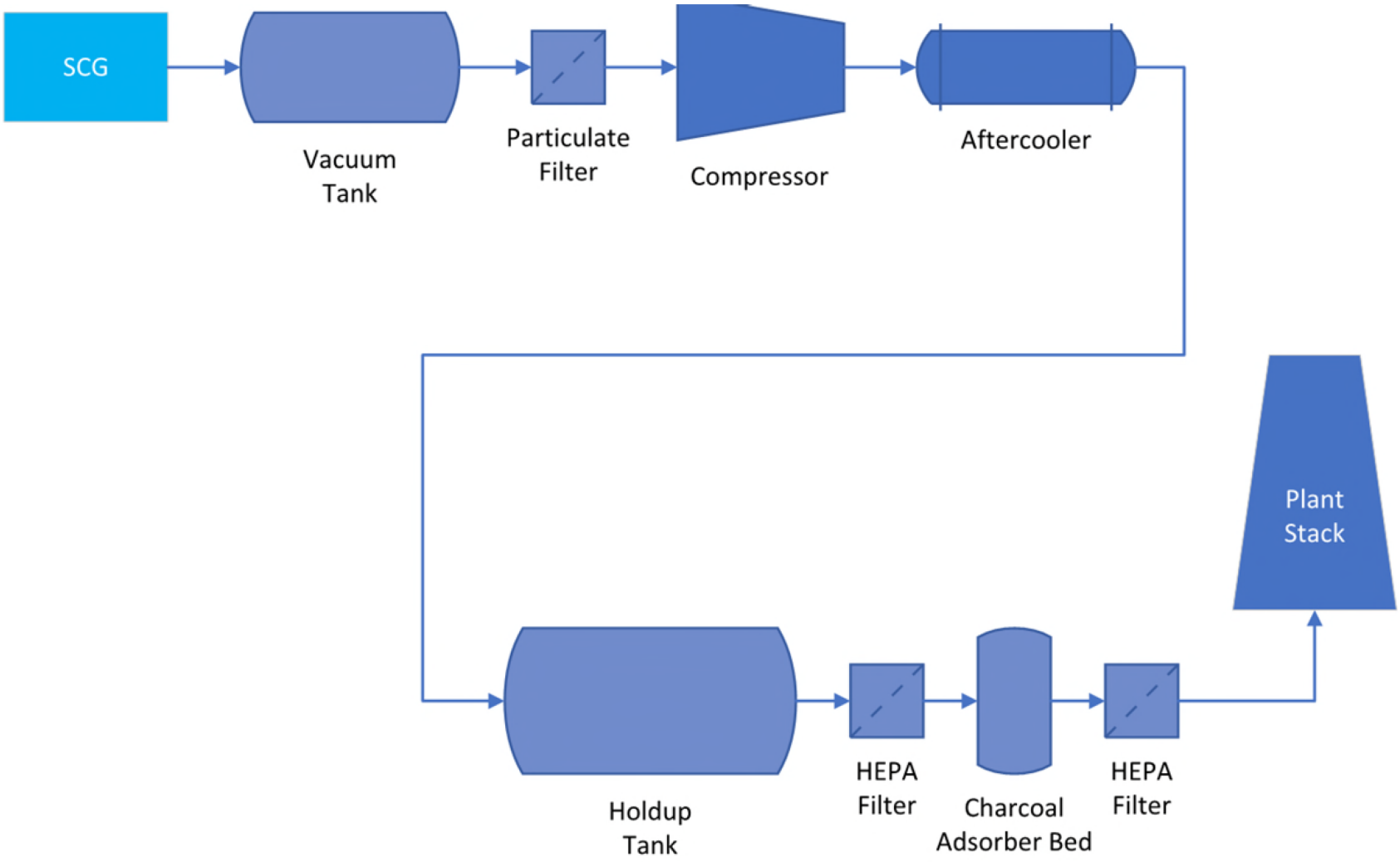
**Figure 7.4.1-1 Gaseous Radwaste Processing System Overview**





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# 7.5 Ancillary Systems

## 7.5.1 NI HVAC System

### 7.5.1.1 Summary Description

The Nuclear Island Heating, Ventilation, and Air Conditioning System (NHV) is allocated on a per building basis within the nuclear island. Each building includes unique design function requirements from the heating, ventilation and air conditioning (HVAC) system to regulate and move heated and cooled air throughout the specific building. The HVAC systems provide fresh air and control exhaust air from all nuclear island (NI) buildings.

The Reactor Auxiliary Building (RAB), Reactor Building (RXB), Fuel Handling Building (FHB) and Fuel Auxiliary Building (FAB) include radiologically controlled areas (RCAs) for which HVAC airflow is controlled from areas of lesser potential contamination to areas of higher potential contamination. The Nuclear Island Control Building (NCB) does not contain RCAs.

The NHV is classified as non-safety-related with special treatment (NSRST) to support radionuclide retentions functions described in Section 7.5.1.2 and Main Control Room (MCR) isolation as described in Section 7.5.1.3.

### 7.5.1.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the NHV design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to NHV are described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the NHV is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. The methods applied for the evaluation of natural phenomena and the determination of associated design basis hazard levels are described in Section 6.1.1. Safety-significant NHV equipment are located within structures that provide protection from natural phenomena as described in Section 7.8.

Consistent with PDC 3, *Fire protection*, NHV is designed to minimize the probability and effect of fires through the selection of non-combustible and non-fire sustaining materials, to the extent practical. NHV is designed to respond to signals received from the NFP system in response to fire and smoke in the plant to mitigate the effects of fires. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the NHV is designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operations, including anticipated operational occurrences, and postulated accidents.


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In areas with potential for airborne contaminants or sodium, appropriate compatible materials are selected or coatings are applied to mitigate adverse interactions which could compromise the ability to perform safety-significant functions.

Consistent with PDC 16, *Containment design*, the NHV is designed to provide isolation of functional containment barrier penetrations to support radionuclide retention. The NHV isolation dampers and valves are designed to fail closed on loss of power to provide the isolation function.

Consistent with PDC 19, *Control room*, the NHV design provides suitable environmental conditions for normal operation of the MCR. The NHV design for isolation ensures control room habitability is maintained for radiological, sodium aerosol, and chemical hazards associated with accident conditions.

Consistent with PDC 44, *Structural and equipment cooling*, the NHV provides cooling to remove heat from safety-significant SSCs to facilitate their operation during normal operating conditions. NHV structural and equipment cooling is not required to support the function of safety-significant SSCs during DBAs.

Consistent with PDC 64, *Monitoring radioactivity releases*, the NHV maintains the RCAs at a negative pressure to prevent the inadvertent release of airborne radioactive constituents. The NHV is designed such that the exhaust flows from areas of lesser potential contamination to areas of higher potential contamination by controlling the negative pressure distributions within buildings. Exhaust from RXB, FHB, RAB, and FAB is through the main plant stack for effluent monitoring by the Radiation Monitoring System described in Section 7.6.6.

Radionuclide retention, control of exhaust, and respiratory protection through MCR isolation ensures compliance with 10 CFR Part 20 requirements.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to NHV, along with a description of the use and conformance to the guidance is provided below:

RG 1.13, Rev 2, *Spent Fuel Storage Facility Design Basis*

Partial conformance. RG 1.13 is applicable to the NHV design to support “Limiting Offsite Release of Radioactivity” for the FHB ventilation systems.

RG 1.78, Rev 2, *Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release*

Partial conformance. RG 1.78 Regulatory Positions C.3 and C.4 contain guidance on control room HVAC and isolation systems. Regulatory Position C.4.1, Detection System, is used as guidance for NHV instruments and control systems.

Control room habitability as described in RG 1.78 is assured through the NSRST manual isolation design function of the MCR envelope upon indication of toxic hazard by the fresh air intake sensors.


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RG 1.97, Rev 5, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*

Partial conformance. Post accident monitoring is performed in the MCR or the RSC. RG 1.97 is applicable to NHV to provide an appropriate environment for monitoring parameters during accident and beyond-design-basis-events. Isolation of the MCR or utilization of the compressed breathable air bottles in the RSC as NSRST functions.

RG 1.140, Rev 3, *Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants*

Full conformance. The NHV is designed in accordance with the guidance of RG 1.140 to ensure the design function of isolation and controlled exhaust path are in accordance with approved codes and standards. RG 1.140 is fully applicable to NHV radiological control functions and maintenance of radiological control areas.

RG 1.189, Rev 5, *Fire Protection for Nuclear Power Plants*

Full conformance. RG 1.189 applicability for NHV is for control room habitability, RSC habitability for 72-hours, and interface with NFP control equipment for HVAC SSC operation to mitigate fire events.

RG 1.196, Rev 1, *Control Room Habitability at Light-Water Nuclear Power Reactors*

Partial conformance. RG 1.196 is applicable for compliance with PDC-19 for control room habitability. The NHV provides HVAC for control room habitability in accordance with RG 1.196 guidance.

RG 1.197, Rev 0, *Demonstrating Control Room Envelope Integrity at Nuclear Power Reactor*

Full conformance. RG 1.197 is applicable to NHV for performance and testing of MCR HVAC.

RG 4.21, Rev 0, *Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning*

Full conformance. RG 4.21 guidance is applicable to the NHV design and operation to minimize spread of radioactive constituents throughout the plant.

RG 8.8, Rev 3, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable*

Full conformance. RG 8.8 guidance is applicable to the NHV design and operation to ensure airflow in radiological areas is maintained to minimize personnel exposure to airborne contaminants.


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Functional Requirements and Design Criteria

Probabilistic risk assessment safety functions (PSFs) are determined through the plant-specific probabilistic risk assessment described in Section 3.1. The safety-significant PSFs are identified as non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The NHV supports the following safety-significant functions, which are designated as NSRST to indicate the safety classification of NHV components relied upon to perform the function:


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| | | |
|---|---|---|
| DL4-RR1 | Ex-RES Functional Containment Barrier (NSRST) | The NHV provides isolation of the HAA barrier by closure of the NHV penetration isolation valves. |
| DL4-RR1a | HAA Barrier (NSRST) | The NHV provides isolation of the HAA barrier by closure of the NHV penetration isolation valves. |
| DL4-RR1c | HAA HVAC Operations Following Postulated Release (NSRST) | The NHV provides isolation of the HAA barrier by closure of the NHV penetration isolation valves. |
| DL4-RR3 | SPS Cells Barrier (NSRST) | The NHV ensures the integrity of the SPS cells barrier by isolation of the NHV penetrations with closure of the NHV isolation valves. |
| DL4-RR3c | SPS Cells Barrier Isolation on Leak Detection (NSRST) | On a leak detection signal, isolation at the barrier of the SPS-P cells is automatically isolated. |
| DL4-RR4 | SCG Cells Barrier (NSRST) | The NHV ensures the integrity of the SCG cell barrier by isolation of the NHV penetrations with closure of the NHV isolation valves |
| DL4-RR4b | SCG Cells Barrier Isolation on Leak Detection (NSRST) | On a leak detection signal, NHV ensures the integrity of the SCG cell barrier by isolation of the NHV penetrations. |
| DL4-RR4c | Vapor Trap Cell Isolation on Overpressure (NSRST) | On a primary coolant overpressure relief valve open position indication, the vapor trap cells are automatically isolated. |
| DL4-RR7 | Fuel Handling Building Barrier (NSRST) | The NHV provides HEPA filtration of all exhaust air from the FHB prior to release through the monitored main plant stack. |
| DL5-PAM1 | Post Accident Monitoring (NSRST) | Post accident monitoring is accomplished in the MCR or the Remote Shutdown Complex (RSC). NHV provides conditioned, filtered air in the MCR to support post accident monitoring. In case the MCR personnel evacuate to the RSC, NHV provides a manually operated compressed breathable air bottle rack for use by plant personnel to provide a habitable environment for post accident monitoring in the RSC. |


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Seismic Classification and Design Codes

Seismic classifications for NHV components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The NSRST portions of NHV are designed to seismic classification SCN1. Seismic design, analysis, and qualification is performed as described in Section 6.4. Seismic special treatments are applied to NSRST portions of the NHV to provide assurance that risk-significant functions or functions required for adequate defense-in-depth are capable of being performed.

The NHV SSCs required to perform NSRST functions are designed in accordance with the codes and standards included in Table 7.5.1-1 to ensure their design function is accomplished.

7.5.1.3 System Design and Performance Evaluation

NSRST portions of NHV are designed in accordance with ASME AG-1 (Reference 7.5.1-1) or ASME B31.1 (Reference 7.5.1-2) to ensure isolation and retention or controlled release path design functions can be performed. The NSRST isolation dampers and valves are provided with NST power by the Nuclear Island AC Electrical Power Low Voltage System (NLV). The valves and dampers are designed to fail closed to ensure the radionuclide retention design function is accomplished when power is not available.

Reactor Building HVAC

The NSRST function to isolate the HAA for radionuclide retention is ensured by the double isolation arrangement of one inboard and one outboard air operated valve pair for each penetration. The NHV penetration through the HAA barrier is constructed from pipe and valves to ensure integrity of the barrier when isolation is required.

The below grade SCG vapor trap cell is maintained at slightly negative pressure with inerted atmosphere during operations and is monitored for leakage from the SCG equipment. The SCG trap cell is automatically isolated upon leakage indication in the MCR. The reactor overpressure protection release point is the SCG trap cell. The NHV cooling loop is designed with pipe and equipment to accommodate the reactor overpressure relief into the SCG trap cell. The cooling loop is automatically isolated based on the position indication of the reactor vessel relief valves. The isolation function is NSRST to support the radionuclide retention function for the SCG trap cell.

Reactor Auxiliary Building HVAC

Below grade SPS cells are maintained at slightly negative pressure with inerted atmosphere during operations. The SPS cells include sodium leak detectors which are connected to and monitored by an NFP sodium leak detection control panel (NACP). The NACP transmits the sodium leak detection signals to the nuclear island utility monitoring and control system for SPS cell isolation. This isolation function is NSRST to support the radionuclide retention function for the SPS cells.


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NI Control Building HVAC

Sodium monitors, toxic chemical sensors, and a radiation monitor continually sample the MCR fresh air intake. Should a sensor detect elevated levels of any hazard, an alarm in the MCR alerts operators to place the MCR HVAC units in recirculation mode and deenergize the below grade HVAC systems to prevent similar hazards in the remote shutdown facility. The isolation of the MCR envelope and the below grade portion of the NCB is an NSRST function. Recirculation mode of the MCR HVAC credits the NSRST high efficiency particulate air (HEPA) filters to minimize operator dose in the MCR. The filters are included in a maintenance program to ensure their operation with reliable power. See Section 7.6.7.1 for additional information on control room operations and Section 7.6.7.2 for RSC operations.

Fuel Handling Building HVAC

The FHB normal exhaust is provided through two 50 percent HEPA filtration units. Exhaust flow is controlled by differential pressure monitors installed on the main FHB working floor to maintain the FHB slightly negative. The exhaust flow from all areas of the FHB is through welded ductwork, HEPA filters and out the monitored plant stack. Indication of high radiation on the HEPA filter discharge alarms in the MCR for manual operator action.

Controlled Release Path

The controlled combined release path from the FHB, RXB, RAB, and FAB through the monitored plant stack is an NSRST function.

### 7.5.1.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes applicable to NHV is described in Section 7.5.1.2. The following programmatic special treatments for NHV NSRST SSCs are preliminarily identified as applicable to NHV to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8:

- Quality Assurance Program Description
- Reliability and Integrity Management Program
- Equipment Qualification Program
- Testing Program
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing, and Analysis Program

### 7.5.1.5 Instrumentation Requirements

Upon a high radiation condition following a postulated release, an indication is provided to the MCR operation personnel via the nuclear island control system displays to evaluate the condition and initiate a manual isolation of the HAA HVAC system, as necessary.


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Upon a high radiation condition within the SPS cells HVAC system, an indication is provided to the MCR operation personnel via the nuclear island control system displays. Operations personnel will evaluate the condition and initiate a manual isolation of the SPS cells, as necessary.

Upon a high radiation condition within the SCG trap cell HVAC system, an indication is provided to the MCR operation personnel via the nuclear island control system displays to evaluate the condition and initiate a manual isolation of the SCG trap cell, as necessary.

The nuclear island control system monitors the position of the reactor vessel pressure relief valve and automatically isolates the SCG trap cell if the relief valve opens.

In the event of a high radiation condition, toxic chemical detection, or sodium vapor detection in the fresh air intake of the MCR, an indication is provided to the MCR operation personnel via the nuclear island control system displays to evaluate the condition and initiate a manual isolation of the control room HVAC system, as necessary.

## References

7.5.1-1 American Society of Mechanical Engineers, “Code on Nuclear Air and Gas Treatment, Includes Addenda A (2010), Addenda B (2011),” ASME, 2009.

7.5.1-2 American Society of Mechanical Engineers, “B31.1: Power Piping,” ASME, 2022.


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**Table 7.5.1-1 Codes and Standards Applicable to Nuclear Island HVAC Systems**

| Author / Organization | Item | Year / Revision | Title | Applicability |
|---|---|---|---|---|
| ASME | AG-1 | 2009 | ASME AG-1-2009: Code on Nuclear Air and Gas Treatment (including 2010 Addenda 1a and 2011 Addenda 1b) | RXB, RAB, FHB, NCB |
| ASME | NQA-1 | 2015 | Quality Assurance Program Requirements for Nuclear Facilities | RXB, RAB, FHB, NCB |
| ASME | B31.1 | 2022 | Power Piping | RXB, underground |



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## 7.5.2 Nuclear Island Fire Protection System

### 7.5.2.1 Summary Description

The Nuclear Island Fire Protection System (NFP) is a collection of four systems located throughout the Nuclear Island (NI). The NFP incorporates dedicated systems and equipment located within the NI facilities and associated yard areas. The primary purpose of the NFP is to mitigate the effects of fire within and around the NI facilities.

NI Fire Water Distribution System

The NI Fire Water Distribution System (NFD) distributes fire water for fire hydrants and water-based fire suppression within NI buildings, facilities, structures, and yard area. The water supply to the NFD is from the Energy Island Fire Protection System (EFP).

NFD is non-safety-related with no special treatment (NST).

NI Fire Suppression System

The NI Fire Suppression System (NFS) suppresses and controls fire with water, clean agents, and chemical means in both automatic systems and manual fire-fighting methods.

NFS is NST.

NI Fire Detection and Alarm System

The NI Fire Detection and Alarm System (NFA) detects smoke and fire, monitors and controls suppression systems, and notifies personnel of fire emergency situations.

NFA is NST.

Sodium Leak Detection, Collection, and Containment System

The Sodium Leak Detection, Collection, and Containment System (NNA) detects, collects, and contains sodium leaks to mitigate sodium fires. The design basis strategy for NNA is to mitigate the effects of a sodium fire once a leak has occurred. Leak prevention is outside the scope of NNA and is addressed by other systems. NNA leak detection contributes to automatic and manual leak responses required to meet Defense Lines (DL).

NNA is non-safety-related with special treatment (NSRST).

### 7.5.2.2 Design Basis

Compliance with regulatory criteria and identification of design criteria and codes and standards that define the NFP design basis are described below.


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Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the NFP is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, NFP components are designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

Consistent with PDC 3, *Fire Protection*, NFP is designed and located to minimize the probability and effect of fire and explosions to safety-significant structures, systems, or components (SSCs). NFP materials are noncombustible and fire-resistant wherever possible. NFA, NFD, and NFS are designed and provided with appropriate capacity and capability to minimize the adverse effects of fires on safety significant SSCs. NFD and NFS are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, NFP systems and components are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. These systems and components are appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit.

Chemical consequences of accidents, such as sodium leakage, are considered for the design of NNA components.

Consistent with PDC 73, *Sodium leakage detection and reaction prevention and mitigation*, the NNA sodium leak detection system provides the means to detect and identify sodium leakage outside of the Head Access Area (HAA) as practical and to limit and control the extent of sodium-air and sodium-concrete reactions. The NNA leak jacket and drainage system mitigates the effects of fires resulting from sodium-air and sodium-concrete reactions to ensure the safety function of safety significant SSCs are maintained.

Consistent with PDC 74, *Sodium/water reaction prevention/mitigation*, structures housing sodium-containing systems (Reactor Building and Reactor Auxiliary Building) do not contain fire water suppression systems to avoid contact between sodium and water and to limit the adverse effects of chemical reactions between sodium and water on the capability of SSCs to perform safety-significant functions. Fire water suppression systems are not provided in areas with sodium-containing systems in the FHB.


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Functional Requirements and Design Criteria

Probabilistic risk assessment safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. Safety-significant PSFs are defined in Section 5.2. The NFP supports the following safety-significant functions, which are designated as NSRST to indicate the safety classification of NFP systems and components relied upon to perform the function:

| Function ID | Function Description | NFP Functional Support |
|---|---|---|
| DL4-RR3a | Sodium Processing System (SPS) Supply Valve Isolation on Leak Detection (NSRST) | NNA sends leak detection signal to Auxiliary Monitoring and Control System (AMC). AMC automatically closes Primary SPS (SPS-P) valve. |
| DL4-RR3b | SPS Pumps Trip on Leak Detection (NSRST) | NNA sends leak detection signal to AMC. AMC automatically trips SPS-P pumps. |
| DL4-RR3c | SPS Cells Barrier Isolation on Leak Detection (NSRST) | NNA sends leak detection signal to the Utility Monitoring and Control System (UMC). UMC automatically closes NI Heating, Ventilation, and Air Conditioning System (NHV) dampers. |

Regulatory Guide 1.189, Revision 5, *Fire Protection for Nuclear Power Plants*

Full Conformance. The NFP is designed to Regulatory Guide (RG) 1.189, Revision 5, “Fire Protection for Nuclear Power Plants,” which meets the regulatory requirements of 10 CFR 50.48(a). RG 1.189 contains guidance for fire protection of nuclear power plants, including the Fire Protection Program, Fire Hazards Analysis, building design, passive features, and fire protection system design and installation. The NFP design scope is limited to fire protection system design and installation, which includes NFD, NFS, and NFA systems only. There are no guidelines in RG 1.189 related to sodium leak detection, collection, or containment.

Seismic Classification and Design Codes

NNA has an initial seismic classification of SCN1 with an NSRST active function required after a seismic event of automatic valve closure upon sodium leak detection in the SPS-P.

Additional design requirements for fire protection are applied to the NFP using the design codes listed in Table 7.5.2-1.

### 7.5.2.3 System Design and Performance Evaluation

The Fire Protection Program and Fire Hazards Analysis will be provided at the operating license stage. The Fire Protection Program addresses those SSCs that could affect safety or the protection of licensed radioactive materials.


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#### 7.5.2.3.1 Nuclear Island Fire Water Distribution System

The NFD consists of fire hydrants and piping necessary to distribute fire water to the required locations around the NI. The system begins at the connections from the EFP, which provides fire water for the NFP. The water then flows through the underground piping system and is distributed throughout the yard for hydrants and buildings for sprinkler and standpipe systems.

The fire water supply from the EFP is in full conformance with guidance provided in RG 1.189. NFD materials are noncombustible and fire-resistant wherever possible. The EFP water supply starts at the two fire water storage tanks in accordance with NFPA 22 (Reference 7.5.2-1) with a minimum of 300,000 gallons each. Water supply capacity is capable of refilling either tank in 8 hours or less. There are two fire pumps in accordance with NFPA 20 (Reference 7.5.2-2): a primary electric motor pump and a backup diesel engine driven pump. A jockey pump is also installed in the system to ensure the required pressure is maintained during normal conditions before a fire pump is activated. The fire pumps are in a sprinklered fire pump house with a 3-hour fire rated separation between the pumps. The diesel fuel tank is located in the room with the diesel fire pump and is sized in accordance with NFPA 20. The fire water distribution loop around the Energy Island supplies NFD in two separate locations for reliability.

#### 7.5.2.3.2 Nuclear Island Fire Suppression System

The NFS consists of water-based fire suppression systems, such as sprinkler and standpipe systems, clean agent suppression systems, and fire extinguishers. NFS materials are noncombustible and fire-resistant wherever possible.

The water-based suppression systems include a series of valves and piping that control and distribute the water throughout the system coverage areas beginning where the fire water distribution system enters the building and ending at the sprinkler heads in a sprinkler system or hose connections in a standpipe system.

Fire protection valves are located within heated enclosures. Portions of the fire protection piping exposed to or potentially exposed to freezing temperatures are protected from freezing via insulation or heat tracing. Wet pipe sprinkler systems are not installed where temperatures fall below 40 degrees Fahrenheit (4.4 degrees Celsius).

The water-based suppression systems are located in buildings where sodium is not present. Buildings containing sodium that are not sprinklered and do not contain fire water are the Reactor Building and the Reactor Auxiliary Building. The FHB contains sodium but also includes separated areas that have sprinkler protection, such as radwaste handling areas and truck bays.

Preaction water-based suppression systems reduce the risk of accidental water damage and are used where water-filled piping is not desirable. Preaction systems rely on air or nitrogen to keep the system pressurized while a valve holds water back from the distribution piping. Water enters the pipe when a fire is detected and then flows from activated sprinkler heads. The Main Control Room (MCR) in the NI Control Building superstructure is equipped with a preaction suppression system.


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Clean agent fire suppression systems are located where fire suppression is required but water would be detrimental to the contents of the area. Clean agent suppression systems start at the pressurized agent cylinders. Upon receiving the designated signal from the NFA for activation, the pressurized gas flows through a piping distribution network and is released from nozzles at the end of the piping network.

There are four independent clean agent systems in the NI Control Building substructure where the Reactor Protection System is located. One system is provided for each Reactor Protection System division.

Fire extinguishers are located throughout the NI with firefighting contents appropriate for the identified hazard where they are installed. Class D fire extinguishers are provided in areas containing sodium in addition to fire extinguishers for typical hazard types.

#### 7.5.2.3.3 Nuclear Island Fire Detection and Alarm System

The NFA consists of initiation devices (e.g., smoke detectors, heat detectors, manual pull stations, and suppression system monitor modules), notification appliances (speakers and strobes), control equipment (fire alarm control panels and notification appliance panels), and connecting wiring. NFA materials are noncombustible and fire-resistant wherever possible.

The NFA is connected to the suppression systems within NFS and interfaces with other systems as necessary to meet code requirements. Signals are sent to the MCR NFA workstation for continuous centralized monitoring in accordance with NFPA 72 (Reference 7.5.2-3).

Electronic components have primary AC power and secondary power from the NI Alternating Current Electrical Power Low Voltage System. The NFA has internal system storage batteries to operate in the event that power is not available.

#### 7.5.2.3.4 Sodium Leak Detection, Collection, and Containment System

The NNA is designed to detect leaked sodium, collect it, and then drain it to strategically located catch pans for containment to mitigate the effects of a sodium fire while giving operators early notification of a leak so they can take necessary action. NNA is located in all areas containing sodium with the exception of the HAA. Sodium leak mitigation in the HAA is outside of NNA scope.

Sodium leak prevention is provided by the sodium system designs and is therefore not included in the NNA design. The intent of the NNA is to provide post leak sodium fire mitigation by detecting, collecting, and containing sodium in leak jackets and catch pans for prevention of fire spread to nearby SSCs. This containment of the sodium allows time for the operators to be notified of the leak and respond with manual Class D fire extinguishers to suppress the remaining reaction in the catch pan and redirect the leaking sodium system to the sodium system drain tank.


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NNA equipment and components consist of sodium leak detectors, Sodium Leak Detection Control Panels (NACPs), wiring, leak jackets, drain lines, and catch pans. Materials selected for NNA components that may contain or contact sodium or sodium reaction products are compatible with liquid sodium and its aerosols, vapors, and oxidation products. NNA materials are noncombustible and fire-resistant wherever possible.

The specific type of sodium leak detector is based on the needs of the installation location and configuration. Sodium leak detectors are located within pipe leak jackets, at vulnerable valves, and inside catch pans. The sodium leak detectors are connected to and monitored by an NACP. The NACP transmits sodium signals to the AMC for operator notification and DL contributions. SPS-P sodium leak detection signals are also transmitted to the UMC for the DL function of automatic NHV damper closure. There is no interface between the NNA and the NFA.

Single wall sodium piping outside of contained skids and cells are equipped with leak detection, leak jacketing, drain lines, and catch pans or other means to contain leaked sodium. The arrangement of NNA components and equipment is designed to prevent leaked sodium from interactions with concrete that would create a fire hazard. Prevention of sodium-concrete interactions that do not present a fire hazard is outside the scope of NNA and is discussed in Section 5.3.

Catch pans are open metal trays with a lip that contains the leaked sodium in a controlled manner to prevent fire spread to nearby SSCs while manual mitigation operations are completed. Catch pans are sized based on postulated leaks.

Electronic components have primary AC power and secondary power from the NI AC Electrical Power Low Voltage System. The NNA has internal system storage batteries to operate in the event that NI AC Electrical Power Low Voltage System power is not available.

#### 7.5.2.3.5 Operation

The NFP operation is the same throughout all phases of plant operation. Normal system operation is standby mode with all NFP system components ready to be activated. Passive components do not need to be activated to perform their functions.

Upon activation by fire, smoke, or heat, the NFA goes into alarm mode and alerts personnel of a fire condition. When smoke is detected in a duct, the associated NHV air handling unit is shut down and local smoke dampers (where installed) close to prevent the spread of smoke.

The NFA also has trouble and supervisory modes which alert personnel of an issue that requires attention within the fire alarm system or within a fire suppression system.

The NFS consists of both automatic and manual suppression systems. Automatic sprinklers activate individually when the heat activated sprinkler component reaches its designated temperature. Automatic clean agent suppression systems are activated upon receiving the designated signal from the NFA. Manual fire suppression is applied by trained personnel on an as needed basis.


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The sodium leak detectors activate an alarm on the NACP upon detection of sodium. The NACP transmits that signal to the AMC for notification of personnel in the MCR. The leak jackets surrounding the sodium pipe passively contain any sodium leaked from the pipe and direct it through a sloped drainage system into a catch pan.

7.5.2.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the NFP is provided in Table 7.5.2-1. The following programmatic special treatments are preliminarily identified as applicable to NFP to meet quality, design, reliability, and performance requirements. The associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program (satisfies RG 1.143 quality assurance requirements)
- Equipment Qualification Program
- Post-Construction Inspection, Testing, and Analysis Program (Chapter 12)
- Design Reliability Assurance Program
- Testing Program

7.5.2.5 Instrumentation Requirements

The status of the NFS, and NFA are monitored within the NFA. Each building has its own Fire Alarm Control Panel, and signals associated with fire protection systems in the building are monitored by the Fire Alarm Control Panel. Each Fire Alarm Control Panel is networked, and signals are sent to the NFA workstation in the MCR.

The sodium leak detectors are monitored within NNA. Leak detector activation generates an alarm signal at the NACP. The signals are then transmitted to the AMC for notification in the MCR and DL contributions. SPS-P sodium leak detection signals are also transmitted to the UMC for the DL function of automatic NHV damper closure.

**References**

7.5.2-1 NFPA 22, *Standard for the Installation of Stationary Pumps for Fire Protection*, 2022

7.5.2-2 NFPA 20, *Standard for Water Tanks for Private Fire Protection*, 2023

7.5.2-3 NFPA 72 ®, *National Fire Alarm and Signaling Code* ®, 2022

7.5.2-4 International Building Code (IBC), *International Building Code*, 2021

7.5.2-5 IEEE 515, *IEEE Standard for the Testing, Design, Installation, and Maintenance of Electrical Resistance Trace Heating for Industrial Applications*, 2020

7.5.2-6 IEEE 622, *Recommended Practice for the Design and Installation of Electric Heat Tracing Systems for Nuclear Power Generating Stations*, 1987

7.5.2-7 NFPA 10, *Standard for Portable Fire Extinguishers*, 2022


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7.5.2-8 NFPA 13, *Standard for the Installation of Sprinkler Systems*, 2022

7.5.2-9 NFPA 14, *Standard for the Installation of Standpipe and Hose Systems*, 2024

7.5.2-10 NFPA 24, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*, 2022

7.5.2-11 NFPA 25, *Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*, 2023

7.5.2-12 NFPA 70 ®, *National Electrical Code* ®, 2023

7.5.2-13 NFPA 75, *Standard for the Fire Protection of Information Technology Equipment*, 2020

7.5.2-14 NFPA 90A, *Standard for the Installation of Air-Conditioning and Ventilating Systems*, 2024

7.5.2-15 NFPA 90B, *Standard for the Installation of Warm Air Heating and Air-Conditioning Systems*, 2024

7.5.2-16 NFPA 1961, *Standard on Fire Hose*, 2020

7.5.2-17 NFPA 1962, *Standard for the Care, Use, Inspection, Service Testing, and Replacement of Fire Hose, Couplings, Nozzles, and Fire Hose Appliances*, 2018

7.5.2-18 NFPA 2001, *Standard on Clean Agent Fire Extinguishing Systems*, 2022



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**Table 7.5.2-1 NFP Design Codes**

| Document Number | Title |
|---|---|
| IBC | International Building Code 2021 |
| IEEE 515 | IEEE Standard for the Testing, Design, Installation, and Maintenance of Electrical Resistance Trace Heating for Industrial Applications |
| IEEE 622 | Recommended Practice for the Design and Installation of Electric Heat Tracing Systems for Nuclear Power Generating Stations |
| NFPA 10 | Standard for Portable Fire Extinguishers |
| NFPA 13 | Standard for the Installation of Sprinkler Systems |
| NFPA 14 | Standard for the Installation of Standpipe and Hose Systems |
| NFPA 20 | Standard for the Installation of Stationary Pumps for Fire Protection |
| NFPA 22 | Standard for Water Tanks for Private Fire Protection |
| NFPA 24 | Standard for the Installation of Private Fire Service Mains and Their Appurtenances |
| NFPA 25 | Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems |
| NFPA 70® | National Electrical Code® |
| NFPA 72® | National Fire Alarm and Signaling Code® |
| NFPA 75 | Standard for the Fire Protection of Information Technology Equipment |
| NFPA 90A | Standard for the Installation of Air-Conditioning and Ventilating Systems |
| NFPA 90B | Standard for the Installation of Warm Air Heating and Air-Conditioning Systems |
| NFPA 1961 | Standard on Fire Hose |
| NFPA 1962 | Standard for the Care, Use, Inspection, Service Testing, and Replacement of Fire Hose, Couplings, Nozzles, and Fire Hose Appliances |
| NFPA 2001 | Standard on Clean Agent Fire Extinguishing Systems |


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# 7.6 Integrated Controls Systems

## 7.6.1 Instrumentation and Control Systems Introduction

Instrumentation and control (I&C) systems provide automatic actuation in response to licensing-basis events, contribute to satisfying Principal Design Criteria (PDC), and provide means for manual actuation of components in support of defense-in-depth. The design basis of I&C systems is described in Section 7.6.1.1. The I&C system architecture is described in Section 7.6.1.2. Safety-significant I&C systems are described in Section 7.6.2 through Section 7.6.6 and Section 7.6.8. Control rooms and indications are described in Section 7.6.7.

The I&C systems are designed using IEEE 603-2018, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (Reference 7.6.1-1) as described in Sections 7.6.2 through 7.6.6 and Section 7.6.8. IEEE 603-2018 provides an equivalent or greater level of safety as IEEE 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (Reference 7.6.1-2) and is therefore sufficient to show compliance with regulation.

### 7.6.1.1 Instrumentation and Control Systems Design Basis

I&C systems are designed to perform assigned safety-related (SR), non-safety-related with special treatment (NSRST), and non-safety-related with no special treatment (NST) functions and to assist in satisfying PDC. Additional design requirements result from regulations, regulatory guidance, and industry codes and standards. The relationship of I&C systems to plant-level Defense Lines (DLs), the classification of I&C systems, compliance with regulations, and the assignment of special treatments are described in Section 7.6.1.1.1 through Section 7.6.1.1.4. A description of how specific safety-significant I&C systems support satisfying PDC is provided in Section 7.6.2 through Section 7.6.6 and Section 7.6.8.

#### 7.6.1.1.1 Relationship to Plant-Level Lines of Defense

The relationship of I&C systems to DLs is described in NAT-4950, “I&C Architecture and Design Basis Topical Report,” Revision 1 (Reference 7.6.1-3). NAT-4950, is incorporated by reference into the SAR. See Section 1.4.2 for a list of documents incorporated by reference into the SAR. The classification of I&C systems and associated DLs that the I&C systems support are shown in Table 7.6.1-1. Specific DL functions for the I&C systems are described in Section 7.6.2 through Section 7.6.6 and Section 7.6.8. The DLs are described in Section 4.2.1.2.

#### 7.6.1.1.2 Instrumentation and Control System Classification

An overview of the I&C system classification process is provided in NAT-4950. I&C system classification is described in Section 5.1.

#### 7.6.1.1.3 Regulatory Requirements and Industry Standards

Descriptions of compliance with regulatory requirements and conformance with regulatory guidance and industry codes and standards for I&C systems are provided in NAT-4950.


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7.6.1.1.4 Instrumentation and Control System Special Treatment

Special treatments are initially selected based on SSC safety classification as described in Section 6.3. Special treatments that have been assigned to particular safety-significant I&C SSCs are described in Section 7.6.2 through Section 7.6.6 and Section 7.6.8. The method for confirming specific reliability and capability targets for SSCs at the operating license stage is described in Section 6.2.

7.6.1.2 Instrumentation and Control System Architecture

The I&C system architecture incorporates the following fundamental I&C design principles consistent with Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews (Reference 7.6.1-4):

- Independence
- Communications and logical independence
- Redundancy
- Diversity
- System Integrity
- Reliable I&C
- Human Factors Engineering

The fundamental I&C design principles are described in NAT-4950.

Human factors are considered in the design of I&C systems. Human factors engineering is described in Section 11.2.

7.6.1.2.1 Instrumentation and Control Systems

The I&C system architecture, to include representative figures, is shown in NAT-4950.

Since I&C systems are classified as SR, NSRST, and NST, the I&C architecture implements the following communication strategy to support the fundamental I&C design principles:

- There are no direct digital data communications from either the NSRST and NST systems to the SR systems.
- One-way communication from SR I&C systems to the NSRST or NST systems is via isolated hardwire or through data-diode and gateways.
- The SR I&C systems do not have inter-divisional communications with the exception of the RPS voting.

Additional information relating to communications for the I&C architecture is provided in NAT-4950.


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SR I&C Systems

The Reactor Protection System (RPS) accepts input signals from plant instrumentation, applies required logic, and automatically generates output signals to initiate safety functions. The RPS also serves to display information to operators in the Main Control Room and Remote Shutdown Complex during normal and post-accident conditions. The RPS is described in Section 7.6.3. The Main Control Room and Remote Shutdown Complex are described in Section 7.6.7.1 and Section 7.6.7.2, respectively.

The Nuclear Instrumentation System provides instrumentation to sense neutron flux during fuel movement, reactor startup, power operations, reactor shutdown, and licensing-basis events. The Nuclear Instrumentation System amplifies and conditions the instrument signals and provides input signals to the RPS and Nuclear Island Control System (NIC) for use in protective, control, and monitoring functions. The Nuclear Instrumentation System is described in Section 7.6.4.

The Reactor Instrumentation System provides instrumentation to monitor selected parameters in or near the reactor vessel and transmits corresponding signals to the RPS and NIC for use in protective, control, and monitoring functions. The Reactor Instrumentation System is described in Section 7.6.5.

NSRST I&C Systems

The NIC is a group of I&C systems that perform NSRST and NST control and monitoring functions. The NIC is implemented on a Distributed Control System platform that provides a means for gathering information from field sensors, executing both manual and automatic logic, interfacing with field actuators, and providing a means for operator interface. The NIC also includes controls that are executed by control systems outside of the Distributed Control System that include interfaces for communication with the Distributed Control System. The NIC and associated I&C systems are described in Section 7.6.2.

The Radiation Monitoring System provides radiation activity indications for use in the protection of plant personnel and the surrounding environment during operating, refueling, and post-accident conditions. The Radiation Monitoring System also performs continuous radiation monitoring and sampling analysis of selected radioactive processes, as well as monitoring for the selected release points where radioactive effluents leave the site boundary. The Radiation Monitoring System is described in Section 7.6.6.

The Anticipatory Automatic Seismic Trip System senses ground motion and provides a reactor scram signal to the RPS in response to a seismic beyond design basis event. The Anticipatory Automatic Seismic Trip System is described in Section 7.6.8.

Post-Accident Monitoring functionality is provided for Type B, C, D, and F variables as defined by IEEE 497-2016, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” (Reference 7.6.1-5). There are no Type A Post-Accident Monitoring variables in the I&C design. The design of Post-Accident Monitoring capabilities is informed by IEEE 497-2016. A description of the conformance to IEEE 497-2016 as endorsed by Regulatory


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Guide 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” Revision 5, will be provided at the operating license stage. The Post-Accident Monitoring functionality is provided using the NIC and RPS systems.

## References


7.6.1-1 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-2018, December 7, 2018.

7.6.1-2 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-1991, December 31, 1991.

7.6.1-3 TerraPower, “I&C Architecture and Design Basis Topical Report,” NAT-4950 Revision 1, ML24068A187.

7.6.1-4 U.S. NRC, “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” February 2021, ML21011A140.

7.6.1-5 IEEE, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497-2016, July 29, 2016.




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**Table 7.6.1-1 I&C System Classification and DL Functions**

| System | Classification | Associated DL |
|---|---|---|
| Nuclear Island Control System | Not Applicable | The NIC is a collection of sub-systems and is described in Section 7.6.2. The NIC is not classified as a system and is not assigned DLs as a system. Sub-systems of the NIC are classified and assigned DLs. |
| Nuclear Island Rod Monitoring and Control System | NST | DL2, DL4, DL5 |
| Nuclear Island Coolant Temperature Control System | NSRST | DL2, DL4, DL5 |
| Nuclear Island Utility Monitoring Control System | NSRST | DL2, DL4, DL5 |
| Nuclear Island Auxiliary Monitoring and Control System | NSRST | DL2, DL4, DL5 |
| Nuclear Island Plant Monitoring and Control System | NST | DL2, DL4, DL5 |
| Nuclear Island Fuel Handling Supervisory Control System | NST | No DLs identified |
| Reactor Protection System | SR | DL3, DL4, DL5 |
| Nuclear Instrumentation System | SR | DL2, DL3, DL5 |
| Reactor Instrumentation System | SR | DL2, DL3, DL4,DL5 |
| Anticipatory Automatic Seismic Trip System | NSRST | DL4 |
| Radiation Monitoring System | NSRST | DL4, DL5 |


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## 7.6.2 Nuclear Island Control System

### 7.6.2.1 System Description, Architecture, and Equipment Locations

The Nuclear Island Control System (NIC) is a digital system with non-safety-related with special treatment (NSRST) and non-safety-related with no special treatment (NST) functions and is a group of instrumentation and control (I&C) subsystems. The NIC is implemented on a Nuclear Island Distributed Control System (DCS) which provides a means for gathering information from various sources, such as field sensors, human-system Interface (HSI), and data feeds from other I&C systems. The DCS also interfaces with various field actuators such that it can execute both manual and automatic logic based on the gathered information.

While most of the NIC is implemented on a DCS, it also includes controls that are executed outside of the DCS. These are referred to as “vendor supplied” control systems. Vendor supplied control systems include interfaces for communication with the DCS.

The following sub-systems are part of the NIC:

- Auxiliary Monitoring and Control System (AMC) – The AMC monitors and controls Nuclear Island (NI) auxiliary systems.
  - Gaseous Radwaste Processing System (Section 7.4.1)
  - Liquid Radwaste Processing System (Section 9.1.1)
  - Solid Radwaste Processing System (RWS) (Section 9.3.1)
  - Sodium Cover Gas (SCG) (Section 7.2.3)
  - Sodium Processing System (SPS) (Section 7.2.4)
  - Radiation Monitoring System (RMS) (Section 7.6.6) (monitoring only)
  - Nuclear Island Fire Protection System (NFP) (Section 7.5.2) (monitoring only)
- Coolant Temperature Monitoring and Control System (CTC) – The CTC monitors and controls
  - Primary Heat Transport System (Section 7.1.3)
  - Intermediate Heat Transport System (IHT) (Section 7.1.4)
  - NI Salt Heat Transport System (Section 1.1.4.2.1)
  - Intermediate Air Cooling System (Section 7.2.2)
  - Energy Island Salt System (Section 1.1.4.2.1) flow control functions.
- Fuel Handling Supervisory Control System (FHC) – The FHC monitors and controls the sub-systems and sub-components of the fuel handling systems (Section 7.3).
- Plant Monitoring and Control System (PMC) – The PMC includes plant information processing and display devices, storage devices, and operator manual command devices which are used to execute plant wide monitoring and operating functions.
- Rod Monitoring and Control System (RMC) – The RMC monitors and controls the sub-systems and sub-components of the control rod drive (CRD) (Section 7.2.5).


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- Utility Monitoring and Control System (UMC) – The UMC monitors and controls the various auxiliary electrical and ancillary systems.
  - Nuclear Island Auxiliary Electrical System (Section 7.7.1)
  - Nuclear Island DC Power Supply System (Section 7.7.1)
  - Nuclear Island Uninterruptible AC Power Supply System (Section 7.7.1)
  - Standby diesel generator (Section 7.7.1.3.1)
  - Nuclear Island Heating, Ventilation, and Air Conditioning System (NHV) (Section 7.5.1)
  - Nuclear Island Air and Gas Distribution System (Section 1.1.4.3.15)
  - NFP (Section 7.5.2) (monitoring only)
  - Seismic Monitoring System (Section 1.1.4.3.16) (monitoring only)
  - RMS (monitoring only)

Each NIC sub-system is integrated with the PMC to provide information displayed on the HSIs and Visual Display Units (VDUs).

The NIC is implemented on a DCS platform which provides a set of standard hardware and software that is configured and customized to meet the needs of the functions being implemented. The DCS is a platform that allows expansion as needed to implement the various aspects of the NIC. This allows for the different sub-systems of the NIC to use common hardware and software to simplify the design of the NIC.

The NIC performs the following system functions:

- Input-output (IO) – The NIC provides an interface between field sensors and actuators and the DCS controller and remote IO cabinets, both of which house various IO modules.
- Display of Information – The NIC displays information on the Main Control Room (Section 7.6.7.1) HSIs, local HSIs, and a plant overview display.
- Automatic and Manual Actuation – NIC controllers include logic to determine when automatic actuations are necessary and allow for manual actuations to be performed. Manual actuations are performed via operator commands initiated from various NIC HSIs or from hardwired switches.

#### 7.6.2.1.1 NIC Architecture

Figure 7.6.2-1 is a block diagram of the NIC and sub-systems showing connections between HSIs, VDUs, typical redundant controllers, remote IO, redundant Reactor Protection System (RPS) to NIC gateways, engineering workstations, and Supervisory, Control and Data Acquisition controllers and HSIs. For simplicity, the connections between the NIC and other systems and devices shown in Figure 7.6.2-1 are intended to represent a generic interface indicating data is shared to and from the NIC. The RPS is described in Section 7.6.3.


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The DCS controller and remote IO cabinets allow expansion and customization to include varying IO modules to interface with the different types of field devices. The DCS gathers information from various sources to make this data available on the plant data highway and the HSI bus so that this information may be displayed. These sources of information include IO, data from other I&C systems (e.g., RPS), and historian.

The NIC includes a data interface between the RPS and NIC. For this interface, the RPS sends data via a one-way data diode to the plant data highway via an RPS-NIC gateway. This gateway is used to perform data conversion into a format that is received by the NIC.

#### 7.6.2.1.2 Equipment Locations

NIC cabinets and components are located in the Main Control Room and Computer Room within the Nuclear Island Control Building (NCB). The IO modules are located near the modular electric rooms as well as distributed throughout the NI for greater flexibility in the placement of sensors and actuators. HSIs and VDUs are located within the Main Control Room and Fuel Handling Control Room; with local HSIs and VDUs available in selected areas of the NI.

### 7.6.2.2 NIC Design Bases and Associated NSRST Functions

Safety classification and associated NSRST functions are specific to each NIC sub-system and NIC component. The NIC as a whole is not classified. The process for classifying NIC sub-systems and components is described in Section 7.6.1.1.2.

The NIC plant data highway, HSI bus, DCS infrastructure, HSIs, and VDUs are NST components. NSRST functions do not rely on these NST components. There are two types of sub-systems for the NIC: sub-systems with both NST and NSRST functions, and sub-systems with only NST functions. The devices with NSRST and NST functions use similar hardware and software, however devices with NSRST functions have special treatments applied.

For sub-systems that only support NST functions, the equipment within the controller cabinets and remote IO cabinets are NST and no special treatments are applied. For sub-systems that support both NSRST and NST functions, the controllers, power supplies, and backplane and IO network are NSRST. The IO modules are a mix of both NST and NSRST, where the NSRST IO modules are used to support both NST and NSRST functions and the NST IO modules are limited to only supporting NST functions. The hardware devices used for NST and NSRST IO modules are the same.

The NIC conforms to RGs as follows:

RG 1.75, Revision 3, *Physical Independence of Electric Systems*

NIC sub-systems that are classified as NSRST partially conform to RG 1.75 Revision 3. Conformance is partial because IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018 (Reference 7.6.2-1) is used rather than the version of the standard endorsed by this RG.


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RG 1.89, Revision 2, *Environmental Qualification of Certain Electric Equipment Important to Safety For Nuclear Power Plants*

NIC sub-systems that are classified as NSRST fully conform to RG 1.89, Revision 2.

RG 1.100, Revision 4, *Seismic Qualification and Mechanical Equipment for Nuclear Power Plants*

NIC sub-systems that are classified as NSRST fully conform to RG 1.100, Revision 4.

RG 1.180, Revision 2, *Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems*

NIC sub-systems that are classified as NSRST fully conform to RG 1.180, Revision 4.

RG 1.204, Revision 1, *Guidelines for Lightning Protection for Production and Utilization Facilities*

NIC sub-systems that are classified as NSRST fully conform to RG 1.204, Revision 1.

7.6.2.2.1 Defense Line Functions

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as safety-related (SR) functions or NSRST functions as defined in Section 5.2. The NIC supports the following safety-significant functions, which are designated as NSRST to indicate the safety classification of NIC components relied upon to perform the function:

AMC Sub-System

The AMC is a digital NIC sub-system with allocated plant functions supporting DL2, DL4, and DL5, including the following DL4 NSRST functions:

| Function ID | Function Description | AMC Functional Support |
|---|---|---|
| DL4-RR3a | SPS supply valve isolation on leak detection | AMC receives sodium leak detection signals from the NFP and provides a signal to close the SPS-P pump supply side isolation valve on the detection of a sodium leak. |
| DL4-RR3b | SPS pump trip on leak Detection | AMC receives sodium leak detection signals from the NFP and provides a signal to trip the SPS-P pump on the detection of a sodium leak. |
| DL4-RR4a | Automatically close SCG isolation valves on leak detection | AMC receives radioactivity indication signals from the RMS, compares them to a setpoint, and provides a signal to close the SCG reactor vessel exhaust isolation valves that are downstream of the vapor cell. |


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CTC Sub-System

The CTC is a digital NIC sub-system with allocated plant functions supporting DL2, DL4, and DL5, including the following DL2 and DL4 NSRST functions:

| Function ID | Function Description | CTC Functional Support |
|---|---|---|
| DL2-HR2 | Intermediate sodium pump (ISP) trip on low IHT level | CTC receives IHT expansion tank level signals from the IHT and provides a signal to trip the ISPs if a low level setpoint is reached. |
| DL4-HR2 | Primary sodium pump (PSP) trip automatic backup | CTC receives primary sodium hot pool temperature signals from the Reactor Instrumentation System (RIS) and provides a signal to trip the PSPs if the high-high primary sodium temperature is exceeded. |
| DL4-HR3 | ISP pump trip automatic Backup | CTC receives primary sodium hot pool temperature signals from the RIS and provides a signal to trip the ISPs if the high-high primary sodium temperature is exceeded. |

FHC Sub-System

The FHC is a digital NIC sub-system with no allocated DL functions and does not provide any SR or NSRST functions, only NST functions.

PMC Sub-System

The PMC is a digital NIC sub-system with allocated plant functions supporting DL2, DL4, and DL5 and does not provide any SR or NSRST functions, only NST functions.

RMC Sub-System

The RMC is a digital NIC sub-system with allocated plant functions supporting DL2, DL4, and DL5 and does not provide any SR or NSRST functions, only NST functions.

UMC Sub-System

The UMC is a digital NIC sub-system with allocated plant functions supporting DL2, DL4, and DL5, including the following DL4 NSRST functions:

| Function ID | Function Description | UMC Functional Support |
|---|---|---|
| DL4-RR1c | Head Access Area (HAA) HVAC operations following postulated release | The UMC receives radioactivity indication signals from the RMS and sends signals to the NHV to close the HAA isolation valves if a setpoint is exceeded. |


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| Function ID | Function Description | UMC Functional Support |
|---|---|---|
| DL4-RR3c | SPS cells barrier isolation leak detection | The UMC receives sodium leak detection signals from the NFP and sends signals to close dampers in the NHV to isolate ventilation to the SPS-P cells. |
| DL4-RR4b | SCG cells barrier isolation leak detection | The UMC receives radioactivity indication signals from the RMS and sends signals to close valves to isolate the SCG vapor trap cells if a setpoint is exceeded. |
| DL4-RR4c | Vapor trap cell isolation on overpressure | The UMC receives poistion indication signals for the reactor vessel relief valves and sends a signal to close valves to isolate the SCG vapor trap cells if a relief valve is open. |

7.6.2.2.2 Principal Design Criteria

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the NSRST NIC SSCs are located either in the SR RXB or NCB substructures or the NSRST NCB superstructure, which is designed to withstand the effects of applicable external hazards without the loss of structural integrity. In this way, the design of the NIC ensures that the NSRST functions will be performed in the event of an earthquake, external flooding, high winds, and external climate.

Consistent with PDC 10, *Reactor design*, the NIC functions to support reactivity control and decay heat removal during all normal, anticipated operational occurrence (AOO), and postulated accident conditions. The NIC subsystems, CTC (classified NSRST), RMC (classified NST), and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to ensure reactor limits are not exceeded during all normal, AOO, and postulated accident conditions

Consistent with PDC 13, *Instrumentation and control*, the NIC provides the capability to reliably monitor and control the plant systems during normal power operations, plant startup and shutdown, including steady state and anticipated plant transients. Appropriate controls are provided to maintain associated variables and systems within prescribed operating ranges. The backbone of NIC is implemented on a DCS which provides a means for monitoring and gathering information from various sources, such as field sensors, HSI, and data feeds from other I&C components. This information is used by DCS to monitor plant variables and execute both manual and automatic control of plant systems to maintain these variables and systems within prescribed operating ranges.

Consistent with PDC 15, *Primary coolant system design*, the NIC functions to support reactivity control and decay heat removal during all normal, AOO, and postulated accident conditions. The NIC subsystems, CTC (classified NSRST), RMC (classified NST), and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to ensure primary coolant limits are not exceeded during all normal, AOO, and postulated accident conditions.

Consistent with PDC 24, *Separation of protection and control systems*, the RPS is separated from the NIC NSRST and NST 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


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system satisfying all reliability, redundancy, and independence requirements of RPS, as required by Clause 5.6 of IEEE 603-2018, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations” (Reference 7.6.2-2).

There is no direct communication from NIC systems to the RPS. The RPS is isolated and functionally independent of NSRST and NST control systems and uses input signals from independent instrument channels. One-way communication from RPS to NSRST and NST control systems is via a data diode or hardwired connection.

Consistent with PDC 25, *Protection system requirements for reactivity control malfunctions*, the NIC provides reactivity control during normal operations and during AOOs through power runback performed by RMC and CTC.

Consistent with PDC 26, *Reactivity control systems*, the NIC functions to support reactivity control and decay heat removal during all normal, AOO, and postulated accident conditions. The NIC subsystems, CTC (classified NSRST), RMC (classified NST), and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to ensure reactivity limits are not exceeded during all normal, AOO, and postulated accident conditions.

Consistent with PDC 28, *Reactivity Limits*, the RMC and Control Rod Drive System control the maximum rod withdrawal speed to limit the nuclear reactivity worth insertion rate. A Control Rod Drive System control interlock prevents withdrawal of more than one rod at a time. The Control Rod Drive System also includes a brake to hold the control rod assembly at any selected position within the stroke of travel and prevents it from inadvertent insertion or withdrawal during any non-accident, accident, post-accident, and seismic conditions.

Consistent with PDC 29, *Protection against anticipated operational occurrences*, the NIC functions to support reactivity control and decay heat removal during all normal, AOO, and postulated accident conditions. The NIC subsystems, CTC (classified NSRST), RMC (classified NST), and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to ensure reactivity limits are not exceeded during all normal, AOO, and postulated accident conditions.

Consistent with PDC 60, *Control of releases of radioactive materials to the environment*, the NIC subsystems, AMC (classified NSRST) and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to support control of releases of radioactive material to the environment during all normal, AOO, and postulated accident conditions.

Consistent with PDC 61, *Fuel storage and handling and radioactivity control*, the NIC functions to support control of releases of radioactive material to the environment during all normal, AOO, and postulated accident conditions. FHC provides interlocks and remote control of the In-Vessel Fuel Handling System equipment and sub-systems during normal operation. The interlocks prevent inadvertent operation of equipment and damage to system components and interfacing systems. In the event of control system malfunctions, manual controls provide redundancy.

The NIC subsystems, FHC (classified NST) and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to ensure residual heat removal capability.


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Consistent with PDC 63, *Monitoring fuel and waste storage*, the NIC subsystems, FHC (classified NST) and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to detect conditions that may result in loss of residual heat removal capability.

Consistent with PDC 64, *Monitoring radioactivity releases*, the NIC subsystems, AMC (classified NSRST), RMC (classified NST), and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to monitor the plant for radioactivity released from normal operations including AOO, and from postulated accidents. The Radiation Monitoring System and PMC monitor for radiological release through the exhaust air in the vent stack.

Consistent with PDC 72, *Sodium heating systems*, the NIC subsystems, RMC (classified NST), CTC (classified NSRST), and PMC (classified NST), are designed and operated with appropriate controls and setpoint margins to ensure that the temperature distribution and rate of change of temperature in systems and components containing sodium are maintained within design limits. The NIC controllers have the capability of being redundant to mitigate single system hardware failures.

#### 7.6.2.3 Fundamental Design Principles in the System Design

The NIC subsystems perform both NSRST and NST functions. When required to meet quality, design, reliability, and performance requirements, special treatments are applied to all NIC NSRST SSCs and also to NIC NST SSCs, as needed.

Special treatments are initially selected based on SSC safety classification as described in Section 6.3.

The following programmatic special treatments are preliminarily identified as applicable to the NSRST NIC subsystems (AMC, CTC, and UMC):

- Design Reliability Assurance Program, as described in Table 8.0-1
- Equipment Qualification Program, as described in Table 8.0-1
- Quality Assurance Program Description, as described in Section 8.1
- Post-construction Inspection, Testing, and Analysis Program, as described in Chapter 12

Additionally, the Human Factors Engineering Program Plan, described in Section 11.2, is identified as a programmatic special treatment applicable to the NIC DCS and all NIC subsystems.

Seismic classifications for NIC subsystem (AMC, CTC, and UMC) components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The seismic classification of NIC subsystem (AMC, CTC, and UMC) components are SCN3. Seismic design, analysis, and qualification is performed as described in Section 6.4.

NIC design includes elements of quality that minimize failures in the NSRST functions. These properties of qualification, reliability, robustness, security, and defense-in-depth features are discussed in Section 7.6.2.3.1 through Section 7.6.2.3.5.


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#### 7.6.2.3.1 Qualified Instrumentation and Control

The NIC meets the quality standards established in the Quality Assurance Program Description.

No part of the NIC or its sub-systems is required to operate during or after a seismic event. NIC equipment is mounted in cabinets using standard industrial practices to ensure that the equipment is sufficiently supported.

If NIC equipment is installed near equipment that is required to operate during or following a seismic event, the NIC equipment is designed to ensure that it will not fail in a way that would prevent another SSC from performing its safety-significant function. This is done by using robust cabinets with adequate anchoring to ensure that any NIC equipment failures only affect devices within the cabinets and do not affect any external components.

The seismic design of the NIC and its subsystems is described in Section 6.4.1. The Equipment Qualification Program described in Table 8.0-1 addresses the requirements, standards, and process for qualifying the NIC system equipment.

#### 7.6.2.3.2 Reliable Instrumentation and Control

The NIC is used by operators to determine the status of the plant and to operate NST and NSRST devices. Some NIC functions require higher levels of reliability. Achieving an acceptable level of reliability ensures that the system performs its functions when required.

In addition to the elements discussed in Section 7.6.2.3.1, redundancy contributes to the reliability and availability achieved within the NIC.

The NIC includes provisions for redundancy to increase system reliability for NSRST functions. Controllers and IO modules have the capability of being implemented with various levels of redundancy. With a redundant IO module and controller, the only common failure point is the physical termination of the field device. Except for the physical connection between the termination point and the IO modules, no single failures of any component within the NIC can cause a failure.

The HSI bus and plant data highway is implemented using redundant equipment such that no single failure in a device on those networks will cause a failure of the network. Multiple HSIs and VDUs are provided in various locations in the facility (i.e., in the Main Control Room and Remote Shutdown Complex) such that a complete failure of one of these devices does not affect the ability of operators to access information or perform operations. Additionally, the DCS includes application and database servers that perform different functions. These servers are also redundant such that a failure in one device does not result in a failure of the DCS. Connections between the VDUs, HSIs, and servers and the HSI bus and plant data highway are redundant.

NIC controller and remote IO cabinets contain redundant power supplies powered from separate sources such that a failure in a power supply, or in one of the power feeds, does not impact any NIC functions. When necessary, equipment with dual redundant power supplies is selected such that the loss of a single power supply or power feed will not result in a loss of functionality.


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The method for confirming specific reliability and capability targets for SSCs at the operating license stage is described in Section 6.2.

#### 7.6.2.3.3 Robust Instrumentation and Control

An important element of the I&C robustness is separation from other I&C systems and isolation of signals from the field and other systems. The NIC includes a data interface between the RPS and the NIC. For this interface, the RPS sends data via a one-way data diode to the plant data highway via an RPS-NIC gateway. This gateway is used to perform data conversion into a format that is received by the NIC and ensures separation of the NIC from the RPS.

The NIC also includes hardwired interfaces between the RPS and NIC. For these signals, an isolation device is provided to ensure that the NIC does not have an adverse impact on the RPS.

#### 7.6.2.3.4 Secure Instrumentation and Control

Security for I&C systems is addressed through the physical security and cyber security programs that are described in Section 11.6.

#### 7.6.2.3.5 Defense-in-Depth

Some plant parameters are monitored by both the SR, and the NSRST and NST portions of the I&C systems. When diversity between the SR and NSRST or NST functions is required, redundant sensors are utilized where one sensor provides a signal to the SR I&C system and another sensor is used to provide a signal to the NSRST or NST I&C system. In cases where a SR sensor is shared between SR and NSRST or NST I&C systems, the sensor information is routed through a signal splitter which then sends information to the SR I&C systems, as well as the NST or NSRST I&C system. This signal splitter provides isolation so that the NSRST and NST side of the splitter cannot have an adverse impact on the SR side of the splitter.

### 7.6.2.4 Operator Interface

The plant data highway, HSI bus, HSIs, VDUs, and other DCS servers are classified as NST. Display and control functions on the NIC are NST functions and the HSIs are used to perform those functions.

For those instances where the display of information is an NSRST function, a separate NSRST display device is provided. These NSRST display devices are placed within a loop that also includes a NIC IO module such that the information is acquired by both the NSRST display device and the IO module. This allows for this data to be displayed both on the NSRST display device as well as on an HSI.

For those instances where the manual control of a device is an NSRST function, the NSRST manual actuation is performed using a hardwired switch. For functions where an NSRST hardwired switch is provided for manual actuation of a function, operators can also perform a manual actuation from an Main Control Room HSI, although this is an NST function.


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Post Accident Monitoring / Safety Parameter Display System

The NIC performs data-acquisition and display of Post-Accident Monitoring (PAM) for Type D and E variables as defined in IEEE 497-2016, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations” (Reference 7.6.2-3). The credited display of all other PAM variable types is outside of the scope of the NIC; however, PAM variables from the RPS are sent to the NIC so that all PAM variables are displayed on the NIC.

The Safety Parameter Display System screens are used to consolidate safety parameters and PAM variables into a single device that is used to determine the status of the plant. The Safety Parameter Display System is a set of screens that are available from the NIC HSIs and is not a stand-alone system.

**References**

7.6.2-1 IEEE, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018, November 1, 2018.

7.6.2-2 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-2018, December 7, 2018.

7.6.2-3 IEEE, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497-2016, July 29, 2016.


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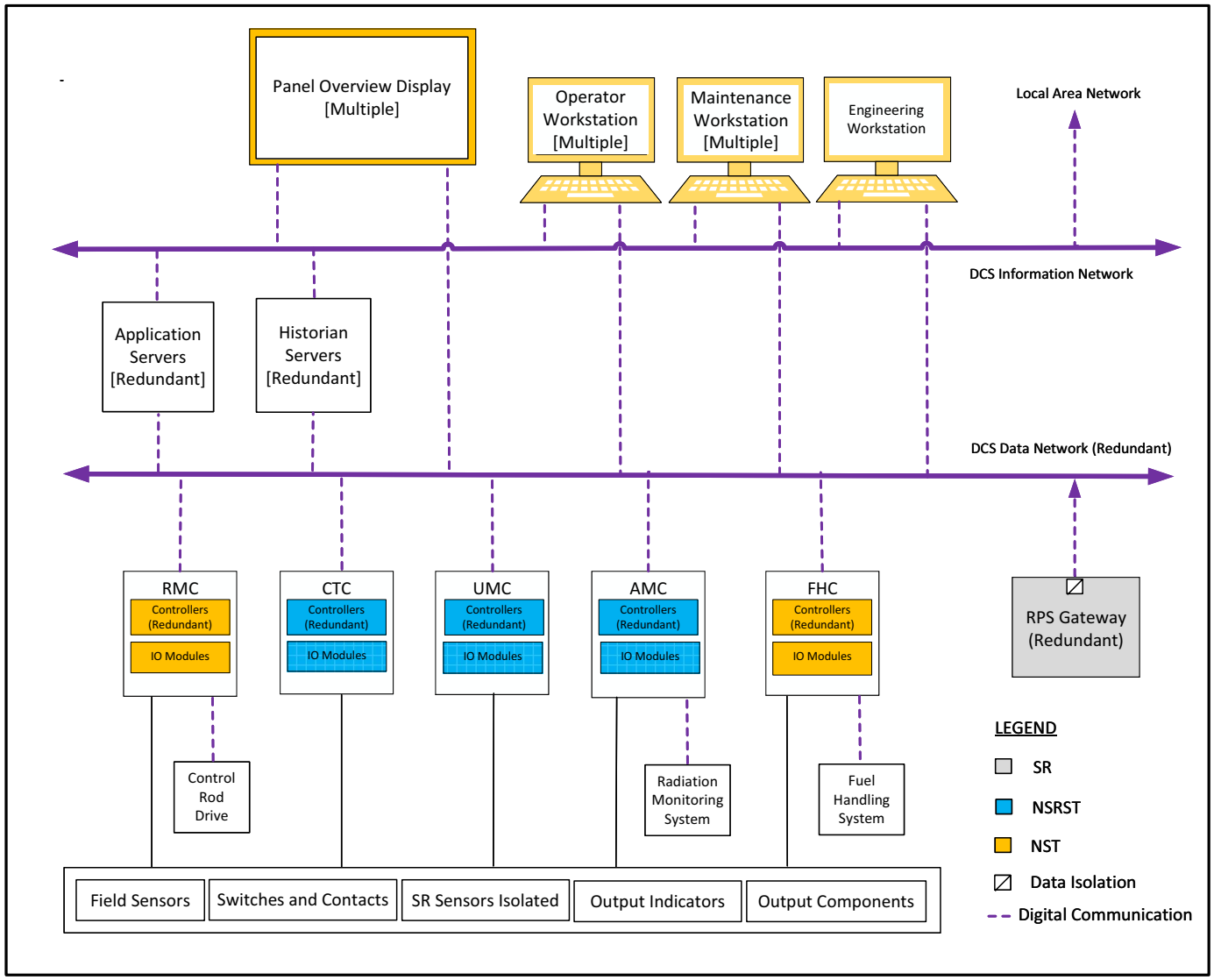
**Figure 7.6.2-1 Nuclear Island Control System Block Diagram**





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## 7.6.3 Reactor Protection System

### 7.6.3.1 Reactor Protection System Description, Architecture, and Equipment Locations

The Reactor Protection System (RPS) accepts input signals from plant instrumentation, applies required logic, and automatically generates output signals to initiate safety functions, including reactor scram and engineered safety feature (ESF) actuations. The portions of the RPS which support these safety functions are arranged into four redundant divisions (A, B, C, and D). Each process variable used by RPS to perform a safety function is measured by four redundant instruments or groups of instruments, which are also divided into four divisions. Each division of RPS receives analog input signals from one associated division of sensing instruments. The RPS is designed so that input signals are accepted, required logic is applied, and output signals are generated within the time assumed in the safety analysis.

Within each RPS division, the input signals are received by a chassis of electronics, referred to as a safety function chassis, which is dedicated to the performance of safety functions. Within the dedicated safety function chassis of each division, input signals are converted to digital form and required signal conditioning and processing logic are applied. Processed variables are compared to established setpoints within that division of RPS. If a variable exceeds the established setpoint within a given RPS division, then the RPS division asserts a divisional vote in favor of initiating the applicable reactor scram or ESF actuation.

Each division shares its division and parameter-specific vote with all other divisions. When a given RPS division senses that two or more of the four RPS divisions has voted in favor of a particular reactor scram or ESF actuation based on the same input parameter, then that division automatically generates an output signal to the appropriate reactor or ESF actuation interface circuit to initiate that reactor scram or ESF actuation function.

The RPS also provides the capability to manually initiate each reactor scram and ESF actuation function from the Main Control Room (MCR), although these manual functions are not credited in the safety analysis. These manual reactor scram and ESF control circuits are diverse from the RPS automatic logic.

Each of the four RPS divisions has a division-specific Special Video Display Unit (SVDU) in the MCR. Each divisional SVDU displays parameters and status of the associated RPS division only. The MCR is described in Section 7.6.7.1. Division A and Division B SVDUs are provided for the Remote Shutdown Complex as described in Section 7.6.7.2.

The RPS utilizes SVDUs to display Type B, C, and F Post-Accident Monitoring (PAM) variables, as defined by IEEE 497-2016 “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations” (Reference 7.6.3-1). The design of PAM capabilities is informed by IEEE 497-2016. A description of the conformance to IEEE 497-2016 as endorsed by Regulatory Guide (RG) 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” Revision 5, will be provided at the operating license (OL) stage.


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There are variables identified as PAM Type B, C, and F variables that are already provided to the RPS for the performance of safety functions. For these variables, the RPS displays the information that is input to the safety function chassis on the SVDUs to provide the PAM indication. This avoids the complexity of introducing an additional I&C system with the sole purpose of supporting PAM functions.

For variables required for PAM which are not already provided to the RPS for the performance of safety functions, there is a separate, PAM-only, chassis in RPS Divisions A and B dedicated to processing signals from these variables. The PAM-only chassis is functionally separate from the safety function chassis, is composed of safety-grade platform components of the same type as the safety function chassis, and is treated as part of the SR protection system for simplicity. Segregating these variables onto the dedicated PAM-only chassis avoids additional burden on the safety function chassis and preserves the simplicity in the design and performance of the safety functions. By using two PAM-only chassis, the single failure criteria required for PAM Type B and C variables is satisfied.

Inputs from sensors that monitor parameters needed for both RPS safety functions and PAM are processed by RPS Division A and B safety function chassis and displayed on RPS Division A and B SVDUs. There are some variables required for PAM which are not provided to RPS safety function chassis. For these variables, RPS Divisions A and B have a PAM-only chassis that is dedicated to processing input signals and is separate from the safety function chassis.

#### 7.6.3.1.1 Reactor Protection System Architecture

The RPS architecture is described in NAT-4950 “I&C Architecture and Design Basis Topical Report,” Revision 1 (Reference 7.6.3-2), which includes a representative figure. NAT-4950 is incorporated by reference into the SAR in its entirety. See Section 1.4.2 for a list of documents incorporated by reference into the SAR.

The RadICS platform as described in the RadICS Topical Report (Reference 7.6.3-3) is used for the RPS. The generic open items and plant-specific action items described in Sections 6.0 and 7.0 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001 (Reference 7.6.3-4) will be provided at the OL stage.

A list of RPS equipment and associated safety and seismic classifications is provided in Table 7.6.3-1. A description of the seismic classification process is provided in Section 6.4.1.

#### 7.6.3.1.2 Reactor Protection System Equipment Locations

RPS equipment is located in the four division-specific RPS vault, Nuclear Island Distributed Control System (DCS), and Control Rod Drive System (CRD) cabinet rooms in the Nuclear Island Control Building (NCB) substructure that are shown on Figure 1.1-12 and in the MCR. The four division-specific RPS vault, DCS, and CRD Cabinet Rooms contain RPS electronics cabinets and reactor trip breakers (RTBs). The Division A, B, C, and D SVDUs are located in the MCR. The Remote Shutdown Complex is provided with redundant SVDUs which are fed from Divisions A and B. The Remote Shutdown Complex is described in Section 7.6.7.2. The NCB substructure is described in Section 7.8.4.1.


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### 7.6.3.2 Reactor Protection System Design Bases and Associated Safety Functions

The RPS is designed to perform assigned safety-related (SR) and non-safety-related with special treatment (NSRST) functions and to assist in satisfying Principal Design Criteria (PDC). A description of the process for assigning DL functions to the RPS is provided in Section 7.6.1.1.1. A description of the process for assigning the safety classification of the RPS is provided in Section 7.6.1.2. A list of RPS equipment, the safety classification of the equipment, and the seismic classification of the equipment is provided in Table 7.6.3-1. A description of the seismic classification process is provided in Section 6.4.1.

Setpoints for the RPS are determined using a methodology that conforms to ANSI/ISA Standard 67.04.01-2018, “Setpoints for Nuclear Safety-Related Instrumentation,” (Reference 7.6.3-5) as endorsed by Regulatory Guide (RG) 1.105, “Setpoints for Safety-Related Instrumentation,” Revision 4, in order to ensure:

- Setpoints for SR instrumentation are established to protect plant safety and analytical limits
- Maintenance of instrument channels implementing these setpoints ensures they are functioning as required, consistent with the plant technical specifications

The setpoint control program is described in Section 2.0 of Table 11.5-1.

Special treatments are applied to safety-significant equipment in the RPS to meet quality, design, reliability and performance requirements. Special treatments are initially selected based on structure, system, or component (SSC) safety classification as described in Section 6.3. Special treatments specific to RPS fundamental design principles are described in Section 7.6.3.1 through Section 7.6.3.3. The following programmatic special treatments are identified as applicable to the RPS:

- Design Reliability Assurance Program (Table 8.0-1)
- Equipment Qualification Program (Table 8.0-1)
- Post-construction inspection, testing, and analysis program (Chapter 12)
- Quality Assurance Program Description (Section 8.1)
- Human Factors Engineering Program Plan (Section 11.2)

Codes, standards, and regulatory guides (RGs) are applied to RPS equipment that is part of the RadICS platform as described in the RadICS Topical Report. Application-specific elements of the RPS comply with newer codes, standards, and RGs identified for use as special treatment so long as they do not violate codes, standards, and RGs identified in the RadICS Topical Report or contradict the Safety Evaluation for Topical Report 2016-RPC003-TR-001.

The RPS conforms to RGs as follows:

RG 1.22, Revision 0, *Periodic Testing of Protection System Actuation Functions*

Application-specific elements of the RPS fully conform to RG 1.22, Revision 0.


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RG 1.47, Revision 1, *Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems*

Application-specific elements of the RPS fully conform to RG 1.47, Revision 1.

RG 1.53, Revision 2, *Application of the Single-Failure Criterion to Safety Systems*

Application-specific elements of the RPS partially conform to RG 1.53, Revision 2. Conformance is partial because IEEE 379-2014, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” (Reference 7.6.3-6) is used rather than the version of the standard endorsed by this RG.

RG 1.75, Revision 3, *Physical Independence of Electric Systems*

Application-specific elements of the RPS partially conform to RG 1.75 Revision 3. Conformance is partial because IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018 (Reference 7.6.3-7) is used rather than the version of the standard endorsed by this RG.

RG 1.97, Revision 5, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*

The RPS partially conforms to RG 1.97, Revision 5. The RPS is one of several systems which supports PAM functions and, alone, is not capable of fully conforming with all of the requirements of the RG. The extent of conformance to RG 1.97, Revision 5, will be provided at the OL stage.

RG 1.100, Revision 4, *Seismic Qualification and Mechanical Equipment for Nuclear Power Plants*

Application-specific elements of the RPS fully conform to RG 1.100, Revision 4. RadICS platform components used in the RPS conform to an alternate version of the RG in accordance with the RadICS Topical Report and the Safety Evaluation for Topical Report 2016-RPC003-TR-001.

RG 1.105, Revision 4, *Setpoints for Safety Related Instrumentation*

The RPS fully conforms to RG 1.105, Revision 4.

RG 1.118, Revision 3, *Periodic Testing of Electric Power and Protection Systems*

Application-specific elements of the RPS partially conform to RG 1.118, Revision 3. Conformance is partial because IEEE 338-2022, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” (Reference 7.6.3-8) is used rather than the version of the standard endorsed by this RG.

RG 1.152, Revision 4, *Criteria for Use of Computers in Safety Systems of Nuclear Power Plants*

Application-specific elements of the RPS fully conform to RG 1.152, Revision 4. RadICS platform components used in the RPS conform to an alternate version of the RG in accordance with the RadICS Topical Report and the Safety Evaluation for Topical Report 2016-RPC003-TR-001.


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RG 1.153, Revision 1, *Criteria for Safety Systems*

Application-specific elements of the RPS partially conform to RG 1.153, Revision 1. Conformance is partial because IEEE 603-2018, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (Reference 7.6.3-9) is used for the application-specific elements rather than the version of the standard endorsed by this RG. RadICS platform components conform to RG 1.153, Revision 1, in accordance with the RadICS Topical Report and the Safety Evaluation for Topical Report 2016-RPC003-TR-001.

RG 1.180, Revision 2, *Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems*

Application-specific elements of the RPS fully conform to RG 1.180, Revision 2. RadICS platform components used in the RPS conform to an alternate version of the RG in accordance with the RadICS Topical Report and the Safety Evaluation for Topical Report 2016-RPC003-TR-001.

#### 7.6.3.2.1 Defense Line Functions

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as safety-related (SR) functions or non-safety-related with special treatment (NSRST) functions as defined in Section 5.2. The RPS supports the following safety-significant functions, which are designated as SR (for DL3 functions) or NSRST (for DL4 and DL5 functions) to indicate the safety classification of RPS components relied upon to perform the function:



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| Function ID | Function Description | RPS Functional Support |
|---|---|---|
| DL3-RC1a | Reactor scram on high-high neutron flux | RPS continuously monitors neutron flux signal inputs from the XIS and compares them to a power range, high range setpoint. When a setpoint is exceeded, all control rod assemblies (CRAs) are inserted by gravity. |
| DL3-RC1b | Reactor scram on high-high hot pool temperature | RPS continuously monitors primary sodium hot pool temperature signal inputs from the RIS and compares Them to a setpoint. When a setpoint is exceeded, all CRAs are inserted by gravity. |
| DL3-RC1c | Reactor scram on high-high primary sodium level | RPS continuously monitors primary sodium hot pool level signal inputs from the RIS and compares them to a setpoint. When a setpoint is exceeded, all CRAs are inserted by gravity. |
| DL3-RC1d | Reactor scram on high-high power-to-flow ratio | RPS continuously monitors neutron flux signal inputs from the XIS. RPS continuously monitors primary sodium pump (PSP) speed and current inputs from the RIS and calculates primary sodium flow. When the ratio of neutron flux to primary sodium flow exceeds a setpoint, all CRAs are inserted by gravity. |
| DL3-RC1e | Reactor scram on high-high cold pool temperature | RPS continuously monitors primary sodium cold pool temperature signal inputs from the RIS and compares them to a setpoint. When a setpoint is exceeded, all CRAs are inserted by gravity. |
| DL3-RC1f | Reactor scram on high-high positive neutron flux rate | RPS continuously monitors neutron flux signal inputs from the XIS, calculates the rate of change of neutron flux, and compares them to a setpoint. When a setpoint is exceeded, all CRAs are inserted by gravity. |
| DL3-RC1g | Reactor scram on high-high negative neutron flux rate | RPS continuously monitors neutron flux signal inputs from the XIS, calculates the rate of change of neutron flux, and compares them to a setpoint. When a setpoint is exceeded, all CRAs are inserted by gravity. |
| DL3-RC1h | Reactor scram on low-low primary sodium level | RPS continuously monitors primary sodium hot pool level signal inputs from the RIS and compares them to a setpoint. When a setpoint is exceeded, all CRAs are inserted by gravity. |


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| Function ID | Function Description | RPS Functional Support |
|---|---|---|
| DL3-RC1i | Reactor scram on low power, high neutron flux | RPS continuously monitors neutron flux signal inputs from the XIS and compares them to a power range, low range setpoint. When a setpoint is exceeded, all CRAs are inserted by gravity. |
| DL3-RC1j | Reactor scram on loss of primary sodium flow | RPS continuously monitors PSP speed and current inputs from the RIS, calculates primary sodium flow, and performs a calculation to determine loss of primary sodium flow. When a loss of primary sodium flow is determined, all CRAs are inserted by gravity. |
| DL3-RC2 | Reactor scram on loss of power | RPS outputs fail in the safe state in the event of a loss of incoming power. |
| DL3- HR2 | PSP trip on high-high primary sodium temperature | RPS continuously monitors primary sodium cold pool temperature signal inputs from the RIS and neutron flux signal inputs from the XIS and compares them to setpoints. If a scram signal exists in the RPS, primary sodium cold pool temperature is above a setpoint, and neutron flux is below a setpoint, the PSPs are tripped. |
| DL3-HR3 | Intermediate sodium pump (ISP) trip on high-high primary sodium temperature | RPS continuously monitors primary sodium cold pool temperature signal inputs from the RIS and neutron flux signal inputs from the XIS and compares them to setpoints. If a scram signal exists in the RPS, primary sodium cold pool temperature is above a setpoint, and neutron flux is below a setpoint, the ISPs are tripped. |
| DL3-HR11 | Sodium processing system (SPS) pump trip on low primary sodium level | RPS continuously monitors primary sodium hot pool level signal inputs from the RIS and compares them to a setpoint. When a setpoint is exceeded, the SPS-P pump is tripped. |
| DL3-HR12 | ISP trip on high-high primary sodium level | RPS continuously monitors primary sodium hot pool level signal inputs from the RIS and neutron flux signal inputs from the XIS and compares them to setpoints. If a scram signal exists in the RPS, primary sodium hot pool level is above a setpoint, and neutron flux is below a setpoint, the ISPs are tripped. |
| DL4-RC1 | Reactor scram-manual | All CRAs are inserted by gravity drop on manual switch. |


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| Function ID | Function Description | RPS Functional Support |
|---|---|---|
| DL4-RC3 | CRD driveline scram follow | RPS issues a failsafe control signal to mechanically drive Control rods in when any automatic or manual scram occurs. The RPS output signals that initiate CRD driveline scram follow are independent of the RTB demand output signals from the RPS. |
| DL4-RC4 | Automatic seismic trip | The RPS opens the RTBs upon receipt of a seismic trip demand signal from the anticipatory automatic seismic trip system (AST). |
| DL4-HR6 | Manual PSP trip | PSP pumps are tripped on manual switch. |
| DL4-HR7 | Manual ISP trip | ISP pumps are tripped on manual switch. |
| DL4-RR8 | Manual SPS pump trip on low primary sodium level | RPS continuously monitors and displays primary sodium hot pool level from signal inputs from the RIS. SPS-P pump is tripped on manual switch. |
| DL5-PAM1 | Post-accident monitoring | The RPS supports the display of Type B, C, and F PAM variables. |

Safety-Related Design Criteria corresponding to the SR functions applicable to the RPS are derived as described in Section 6.1.2 and identified in Section 5.2. The Safety-Related Design Criteria are incorporated into the design of the RPS as described in Section 7.6.3.1.

The NSRST functions DL4-RC1, DL4-HR6, DL4-HR7, and DL4-RR8 are manual initiations of automatic safety functions. Based solely on the PRA model, these functions would be classified as non-safety-related with no special treatment (NST). However, they are classified as NSRST for defense-in-depth purposes and compliance with IEEE 603-1991 for manual initiation of an automatic safety function.

There are no non-safety-related with no special treatment DL functions allocated to the RPS. The CRAs are described in Section 7.1.1. The Primary Heat Transport System and Primary Sodium Pump are described in Section 7.1.3. The Intermediate Heat Transport System and Intermediate Sodium Pump are described in Section 7.1.4. The Sodium Processing System is described in Section 7.2.4. The CRD is described in Section 7.2.5. The Anticipatory Automatic Seismic Trip System is described in Section 7.6.8.

#### 7.6.3.2.2 Principal Design Criteria

PDC are identified in Section 5.3. The RPS conforms to PDC as described below.

Consistent with PDC 2, *Design based for protection against natural phenomena,* in the event of adverse conditions due to natural phenomena, all RPS safety-significant SSCs maintain the capability to perform their required functions.


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Earthquakes- SR portions of the RPS are located in the NCB substructure, which is designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1. SR portions of the RPS are seismically qualified to remain operable in the SR NCB substructure during and after postulated earthquakes.

External flooding- The RPS accounts for external flooding hazard by being located inside of the NCB, which is above the maximum flood level.

High winds- To address the design basis extreme wind hazard, SR portions of the RPS are located in the SR NCB substructure, which is designed for the extreme wind hazard.

Extreme climate- The RPS equipment is located in the climate-controlled NCB to mitigate temperature, humidity, and winter precipitation hazards.

Consistent with PDC 3, *Fire protection*, each SR RPS division is located in its own division-specific RPS vault in the SR NCB substructure. The RPS vault room walls are all 3-hour fire rated and the doors have at least a 3-hour fire rating. Locating each SR RPS division in its own 3-hour fire rated vault room minimizes the likelihood of a fire simultaneously affecting multiple divisions and minimizes the likelihood that fire will affect any RPS SR function.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the SR NCB substructure protects the SR portions of the RPS from external hazards caused by transportation incidents, incidents at offsite industrial and chemical facilities, and external missiles resulting from a turbine failure in the plant's turbine generator set. The SR NCB substructure provides a mild environment for SR portions of the RPS, which are qualified to remain operable in such an environment during postulated external hazards.

Consistent with PDC 10, *Reactor design*, the RPS DL3 reactor scram and ESF functions are design with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded as a result of anticipated operational occurrences (AOOs) which have not been mitigated by DL2 functions.

Consistent with PDC 13, *Instrumentation and control*, the RPS accepts input signals from plant instrumentation in order to monitor plant variables and automatically generate reactor scram and ESF actuation signals when required in order to mitigate AOOs and design-basis events within prescribed design basis accident limits. The RPS serves to display these input variables and others to operators in the control room and in the Remote Shutdown Complex during normal and postulated post-accident conditions. All input instruments, RPS signal conditioning, and RPS displays associated with reactor scram, ESF, and PAM functions are designed to monitor associated variables of their anticipated ranges for normal operation, AOOs, and accident conditions as appropriate.

Consistent with PDC 15, *Primary coolant system design*, the RPS DL3 reactor scram and ESF functions are designed with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded as a result of AOOs which have not been mitigated by DL2 functions.


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Consistent with PDC 20, *Protection system functions*, plant instrumentation continuously senses plant conditions and provides corresponding input signals to the RPS. The RPS is designed to continuously monitor those input signals, apply required logic, and automatically initiate DL3 reactor scram and ESF functions when the RPS senses an AOO which has not been mitigated by DL2 functions as designed, or when design basis event or accident conditions are sensed. The RPS is designed to perform these DL3 functions to ensure that analytical limits established in the safety analysis are not exceeded as a result of AOOs or design-basis events.

Consistent with PDC 21, *Protection system reliability and testability*, the SR portions of the RPS are arranged into four redundant and independent divisions. Each process variable input used by RPS to perform a safety function is measured by four redundant sensing instruments (or groups of instruments) which are also divided into four redundant and independent divisions. The RPS actuates reactor scram and ESF functions when at least 2 out of the 4 divisions independently determine that such action is required.

This divisional arrangement and 2-out-of-4 voting design features allow in-service testability. The system design allows for any division to be taken out of service for testing at any time without loss of any safety function and without loss of minimum required redundancy.

The divisional arrangement and 2-out-of-4 voting design features promote high system reliability. The RPS is designed to withstand a failure of any division without loss of any safety function and without loss of minimum required redundancy. In fact, the RPS is designed to withstand a single failure in any operating division, even when one division has been taken out of service for testing. The capability to periodically test divisions independently while in-service to determine failures and losses of redundancy that may have occurred further promotes high system reliability.

Consistent with PDC 22, *Protection system independence*, the SR portions of the RPS are arranged into four redundant and independent divisions. Each process variable input used by RPS to perform a safety function is measured by four redundant sensing instruments (or groups of instruments) which are also divided into four redundant and independent divisions. The RPS actuates reactor scram and ESF functions when at least 2 out of the 4 divisions independently determine that such action is required.

Each of the four redundant RPS divisions are independent and physically separated from the other redundant RPS divisions to the degree necessary to ensure accomplishment of safety functions.

All four redundant RPS divisions are independent and physically separated from the effects of design-basis events which they are designed to mitigate. The RPS equipment is located to minimize potential damage from hazards and is qualified to substantiate design basis performance requirements.

All four redundant RPS divisions are independent and physically separated or isolated from other systems, including interconnected systems and equipment in proximity, to the degree necessary to ensure accomplishment of safety functions.

The divisional arrangement and 2-out-of-4 voting design features allow for any division to be out of service for maintenance or testing without any loss of safety function.


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Consistent with PDC 23, *Protection system failure modes*, during normal operation (when reactor scram is not required), the RPS maintains voltage on the RTB undervoltage coils to keep RTBs in the closed position, thus maintaining power to the CRD solenoid operated valve coils which maintain argon pressure to hold the rods at controlled positions above or within the core. On loss of incoming power to RPS, the RPS voltage on the RTB undervoltage coils is lost, allowing the undervoltage release mechanism to open the RTBs, thus interrupting power to the CRD solenoid operated valve coils, venting CRD argon and allowing rods to drop fully into the core via gravity. In this way, the RPS reactor scram circuit is designed as “de-energize to trip”, consistent with PDC 23.

All RPS ESF functions execute the tripping of pump breakers to stop the associated pump, which is evaluated to be the safe state. During normal operation (when ESF actuation is not required), the RPS maintains normally open relays in the closed position in order to maintain control power on the pump breaker undervoltage coils, thus keeping pump breakers in the closed position, maintaining normal power to the pumps. On loss of incoming power to RPS, the RPS voltage on the normally open relays is removed, allowing those relay contacts to open, interrupting control power to the pump breaker undervoltage coils, tripping the pump breakers, and stopping pump rotation. In this way, the RPS ESF circuits are designed as “de-energize to trip”, consistent with PDC 23.

The SR RPS equipment is located in the mild environment of the SR NCB substructure. The SR RPS equipment is environmentally and seismically qualified to remain operable during anticipated adverse conditions at this location.

Additionally, the RPS electronics (logic and input-output) modules perform continuous self-diagnostics. In the event that self-diagnostics detect a critical fault in the electronics that could affect performance of a safety function, the system places all outputs to the safe (tripped) state.

The RPS also employs range-checking on analog input signals from system such as Nuclear Instrumentation System and Reactor Instrumentation System. Instrument channels are designed such that loss of power at the transmitting system is detected by RPS range-checking, in which case the RPS will place the associated circuit outputs in the safe state. The Nuclear Instrumentation System is described in Section 7.6.4, and the Reactor Instrumentation System is described in Section 7.6.5.

Consistent with PDC 24, *Separation of protection and control systems*, the RPS is designed in accordance IEEE Std 603-2018 Clause 5.1, Single-Failure Criterion, Clause 5.6, Independence, and Clause 6.3, Interaction Between the Sense and Command Features and Other Systems.

Accordingly, the RPS is designed with sufficient separation from control systems such that all reliability, redundancy, and independence requirements of the RPS remain intact, even when there is a single control system failure, or failure or removal from service of any component common to the RPS or any control system.

The standard listed above for PDC 24 is applied in a manner that does not violate codes, standards, and RGs identified in the RadICS Topical Report as described in Section 7.6.3.2.


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Consistent with PDC 25, *Protection system requirements for reactivity control malfunctions*, the RPS is designed in accordance IEEE Std 603-2018 Clause 5.1, Single-Failure Criterion, Clause 5.6, Independence, and Clause 6.3, Interaction Between the Sense and Command Features and Other Systems.

Accordingly, the RPS DL3 reactor scram and ESF functions are designed to ensure that analytical limits established in the safety analysis are not exceeded as a result of AOOs even when a single failure of the reactivity control system has occurred.

The standard listed above for PDC 25 is applied in a manner that does not violate codes, standards, and RGs identified in the RadICS Topical Report as described in Section 7.6.3.2.

Consistent with PDC 26, *Reactivity control systems*, the RPS DL3 reactor scram function is designed to insert sufficient negative reactivity promptly with appropriate margin to ensure analytical limits established in the safety analysis are not exceeded as a result of AOOs which have not been mitigated by DL2 functions as designed. The RPS DL3 reactor scram function is also designed to insert sufficient negative reactivity with appropriate margin to achieve and maintain a safe shutdown condition following a postulated accident.

The RPS is designed such that whenever any division generates a DL3 reactor scram demand output, that RPS division also generates an independent signal to the CRD system to demand a DL4 CRD driveline scram follow. The CRD system initiates DL4 CRD driveline scram follow when at least 2 of 4 RPS divisions send a demand signal.

Consistent with PDC 28, *Reactivity limits*, the RPS DL3 reactor scram function is designed with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded during postulated accidents.

Consistent with PDC 29, *Protection against AOOs*, the RPS DL3 reactor scram and ESF functions are designed with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded as a result of AOOs which have not been mitigated by DL2 functions.

### 7.6.3.3 Reactor Protection System Fundamental Design Principles

The RPS design minimizes the potential for failures to occur in SSCs that perform SR functions. Qualification, reliability, robustness, security, diversity, and defense-in-depth features are discussed in Section 7.6.3.3.1 through Section 7.6.3.3.4. The following industry codes and standards are applied as special treatments to the overall design of safety-significant, application-specific equipment in the RPS that is outside the scope of the RadICS platform:

- IEEE 603-2018
- IEEE 7-4.3.2-2016 (Reference 7.6.3-10)
- IEEE 379-2014
- IEEE 384-2018


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#### 7.6.3.3.1 Qualified Instrumentation and Control

The following industry codes, standards, and RGs are applied as special treatments for environmental, seismic, and electromagnetic capability qualification of safety-significant, application-specific equipment in the RPS that is outside the scope of the RadICS platform:

- IEEE/IEC 60780-323-2016, “International Standard- Nuclear Facilities- Electrical equipment important to safety-Qualification,” (Reference 7.6.3-11)
- IEEE/IEC 60980-344-2020 “International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification” (Reference 7.6.3-12)
- IEEE 420-2013, “IEEE Standard for the Decide and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Station” (Reference 7.6.3-13)
- RG 1.180, Revision 2, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems”

The safety-significant equipment in the RPS is assigned seismic special treatment commensurate with the seismic classifications shown in Table 7.6.3-1. Seismic special treatment is described in Section 6.4.1.

#### 7.6.3.3.2 Reliable Instrumentation and Control

The method for confirming specific reliability and capability targets for SSCs at the OL stage is described in Section 6.2.

Repair

The ability of the RPS to facilitate timely recognition, location, replacement, repair, and adjustment of malfunctioning equipment will be described as part of addressing the plant-specific action items from Section 7.0 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001 at the OL stage.

Identification

IEEE 603-2018 Clause 5.11, IEEE 384-2018, and IEEE 420-2013 are applied as special treatments to safety-significant, application-specific equipment in the RPS that is outside the scope of the RadICS platform in order to ensure equipment is distinctly identified for each redundant portion of the system. Identification of RPS equipment that is part of the RadICS platform is described in the RadICS Topical Report.

Human Factors

RPS equipment is designed in accordance with IEEE 420-2013, including requirements regarding human factors, and is reviewed per the Human Factors Engineering Program Plan that is described in Section 11.2.


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Reliability

The following industry codes and standards are applied as special treatments to the safety-significant equipment in the RPS to ensure that reliability and capability targets are met:

- IEEE 338-2022
- IEEE 603- 2018

The use of these codes and standards to satisfy plant-specific action items from Section 7.0 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001 will be provided at the OL stage. This will include a system-level evaluation of the reliability of the final application-specific RPS design to address plant-specific action item 7.6.

Capability for Testing and Calibration

The RPS is capable of testing and calibration during power operation without loss of required safety functions at the system level. The design of RPS supports checking each channel (including sensor and any electronics) required for a safety function by one of the following means:

- cross-checking between redundant channels with available readouts
- introducing and varying substitute input
- perturbing the monitored variable

Additional detail on the RPS capability for testing and calibration is included in Section 7.6.3.2.2 in the description of the RPS conformance to PDC 21.

The capability for testing and calibration to satisfy plant-specific action items from Section 7.0 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001 will be provided at the OL stage. This will include a system-level evaluation of system testing and surveillance of the final application-specific RPS design to address plant-specific action item 7.8.

Maintenance Bypass

A description of maintenance bypass features will be provided as part of addressing the plant-specific action items described in Section 7.0 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001 at the OL stage.


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#### 7.6.3.3.3 Robust Instrumentation and Control

Robustness of the RPS design is the degree to which it can perform its SR functions correctly in the presence of invalid inputs or stressful environmental conditions. The following elements are considered for system robustness:

Single Failure

Single failure is addressed by applying IEEE 379-2014 as a special treatment to the design of the RPS. A description of how single failures cannot adversely affect protection functions will be provided as part of addressing the plant-specific action items described in Section 7.0 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001 at the OL stage.

Completion of Protective Action

The RPS is designed so that, once initiated automatically or manually, the sequence of protective actions continue to completion. Deliberate operator action is required to return the RPS to normal.

System Integrity

The characteristics of the RPS to accomplish safety functions under the full range of design basis conditions are described in the RadICS Topical Report. Application-specific elements are designed and qualified to accomplish safety functions under the full range of design basis conditions. Plant-specific information relating to system integrity to satisfy plant-specific action items from Section 7.0 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001 will be provided at the OL stage. This will include the equipment environmental qualification of the final application-specific RPS design to address plant-specific action item 7.4.

Independence

A description of independence for the RPS is provided in Section 7.6.3.2.2 in the description of the RPS conformance to PDC 22.

Interaction Between Sense and Command Features and Other Systems

A description of the interaction between RPS sense and command features and other systems is provided in Section 7.6.3.2.2 in the description of RPS conformance to PDC 24.

Operating Bypasses

A description of the implementation of operating bypass capabilities will be provided at the OL stage.

Redundancy

The RPS consists of four redundant and independent divisions. Additional description of the reliability, testability, and independence of the RPS divisions is provided in Section 7.6.3.2.2 in the description of RPS conformance to PDC 21 and PDC 22.


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Diversity in Support of Defense-in-Depth to Address Common-Cause Failures

Diversity in support of defense-in-depth to address Common-Cause Failures is described in NAT-4950.

Predictable and Repeatable Behavior

The ability of the RPS to perform in a deterministic manner is described in the RadICS Topical Report and is evaluated in Section 3.7.2 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001. Additional detail of the ability of the RPS to operate in a predictable and repeatable manner will be provided at the OL stage.

7.6.3.3.4 Secure Instrumentation and Control

A Secure Development and Operational Environment is established for the RPS that satisfies the applicable regulatory evaluation criteria of RG 1.152 Rev 4. A description of the Secure Development and Operational Environment will be provided as part of addressing the plant-specific action items described in Section 7.0 of the Safety Evaluation for Topical Report 2016-RPC003-TR-001 at the OL stage.

Security for I&C systems is addressed through the physical security and cyber security programs that are described in Section 11.6.

**References**

7.6.3-1 IEEE, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497-2016, July 29, 2016.

7.6.3-2 TerraPower, “I&C Architecture and Design Basis Topical Report,” NAT-4950, Revision 1, ML24068A187.

7.6.3-3 RadICS LLC, “RadICS Topical Report,” 2016-RPC003-TR-001, Revision 2, NP-A, ML20202A030.

7.6.3-4 U.S. NRC, “Safety Evaluation for Topical Report 2016-RPC003-TR-001 RadICS Safety System Digital Platform,” August 2019, ML19134A193.

7.6.3-5 ANSI/ISI, “Setpoints for Nuclear Safety-Related Instrumentation,” ANSI/ISI Standard 67.04.01-2018, December 8, 2018.

7.6.3-6 IEEE, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” IEEE 379-2014, May 16, 2014.

7.6.3-7 IEEE, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018, November 1, 2018.

7.6.3-8 IEEE, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” IEEE 338-2022, December 3, 2022.


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7.6.3-9 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-2018, December 7, 2018.

7.6.3-10 IEEE, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” IEEE 7-4.3.2-2016, July 29, 2016.

7.6.3-11 IEEE/IEC, “International Standard - Nuclear facilities - Electrical equipment important to safety - Qualification,” IEEE/IEC 60780-323-2016.

7.6.3-12 IEEE/IEC, “International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification,” IEEE/IEC 60980-344-2020.

7.6.3-13 IEEE, "IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations," IEEE 420-2013, May 3, 2013.



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**Table 7.6.3-1 RPS Equipment and Classifications**

| General RPS Equipment Description | Component Description | Safety Classification | Seismic Classification |
|---|---|---|---|
| RPS Electronics Cabinets | RPS Division A Cabinet | SR | SCS1 |
| | RPS Division B Cabinet | SR | SCS1 |
| | RPS Division C Cabinet | SR | SCS1 |
| | RPS Division D Cabinet | SR | SCS1 |
| RPS Data Gateway | RPS Data Gateway Cabinet | NST | None |
| RPS Divisional SVDUs (MCR) | RPS Division A SVDU | NSRST | SCN1 |
| | RPS Division B SVDU | NSRST | SCN1 |
| | RPS Division C SVDU | NSRST | SCN1 |
| | RPS Division D SVDU | NSRST | SCN1 |
| RPS Manual Switches | RPS Division A Manual Switch Panel | NSRST | SCN1 |
| | RPS Division B Manual Switch Panel | NSRST | SCN1 |
| | RPS Division C Manual Switch Panel | NSRST | SCN1 |
| | RPS Division D Manual Switch Panel | NSRST | SCN1 |
| Remote Shutdown SVDUs | RPS Remote Shutdown SVDU A | NSRST | SCN1 |
| | RPS Remote Shutdown SVDU B | NSRST | SCN1 |
| Reactor Trip Breakers | RTB Enclosure Panel A1 | SR | SCS1 |
| | RTB Enclosure Panel A2 | SR | SCS1 |
| | RTB Enclosure Panel B1 | SR | SCS1 |
| | RTB Enclosure Panel B2 | SR | SCS1 |
| | RTB Enclosure Panel C1 | SR | SCS1 |
| | RTB Enclosure Panel C2 | SR | SCS1 |
| | RTB Enclosure Panel D1 | SR | SCS1 |
| | RTB Enclosure Panel D2 | SR | SCS1 |


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## 7.6.4 Nuclear Instrumentation System

### 7.6.4.1 System Description, Architecture, and Equipment Locations

The analog Nuclear Instrumentation System (XIS) provides instrumentation to sense neutron flux leakage from the core during fuel movement, reactor startup, power operations, shutdown, and accident conditions. Neutron leakage is proportional to reactor flux and thus can be converted into a power indication. The XIS uses a High Temperature Fission Chamber to measure the neutron flux in the reactor. The fission chamber utilizes a high voltage probe to determine the average flux inside the chamber. The neutron flux is sent as an electrical signal to the XIS enclosures and converted to source, wide, and power range signals. The XIS amplifies and conditions the instrument signals and provides corresponding input signals to the Reactor Protection System (RPS) (described in Section 7.6.3) for protective functions and the Nuclear Island (NI) Control System (NIC) (described in Section 7.6.2) for use in control, monitoring, and display functions. The XIS is designed to ensure that neutron flux is detected and signals are generated and sent to the RPS within the time assumed in the safety analysis.

There is one High Temperature Fission Chamber detector for each division (four total). Each detector is located within an individual XIS drywell. The XIS drywells, at the depth of the detectors, are located on the outside perimeter of the Core Barrel Structure outer barrel (described in Section 7.1.2). Each detector and associated electronics have the ability to provide three signal ranges, power range, wide range, and source range. The source range signal includes an audible count rate variable.

Each High Temperature Fission Chamber detector is wired to an XIS field amplifier enclosure which accepts the detector signal, performs required pre-amplification, and transmits corresponding instrument signals to the XIS electronics enclosures. These enclosures accept the signals from the field amplifier, perform signal conditioning, conversion, and amplification, and transmit corresponding signals to remote input-output cabinets associated with RPS, NIC, Fuel Handling Control Room (FHCR) (described in Section 7.3.2.5), and Main Control Room (MCR) (described in Section 7.6.7.1).

Each XIS division also provides isolated hardwired signals to ACR monitoring devices in the MCR and in the FHCR for use during refueling and startup operations.

A wide range neutron flux signal is sent to the RPS and NIC for display as a Type B and D Post-Accident Monitoring (PAM) variable, as defined by IEEE 497-2016, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” (Reference 7.6.4-1). The display of PAM variables by the RPS is described in Section 7.6.3.1.

#### 7.6.4.1.1 System Architecture

The XIS is comprised of four redundant divisions, each of which includes a division-specific flux detector and signal conditioning. Each XIS division is separate and independent from the other XIS divisions. XIS provides flux signals to RPS in support of safety-related (SR) protective functions, and the four XIS divisions correspond to the four RPS divisions. Each XIS division


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interfaces with the corresponding RPS division only, thus maintaining the required XIS divisional separation and RPS divisional separation. XIS signals sent to RPS are made available to the plant data bus via the RPS data gateway.

The XIS also provides signals to the NIC in support of control functions which are non-safety-related with no special treatment (NST). The NIC subsystems such as Rod Monitoring and Control System and NI Coolant Temperature Monitoring and Control System, which use the XIS signals in real-time control schemes, are provided with isolated hardwired signals from XIS.

Figure 7.6.4-1 provides an overview of the XIS.

#### 7.6.4.1.2 Equipment Locations

The XIS detectors are located in individual drywells external and adjacent to the core. The XIS detectors are wired to the XIS field amplifier enclosures located in the Reactor Building (RXB) Head Access Area. The amplified signals are wired to the XIS electronics enclosures located in the Nuclear Island Control Building (NCB) Substructure. The XIS Audible Count Rate monitors are located in both the MCR and FHCR.

### 7.6.4.2 System Design Bases and Associated Safety-Related Functions

The XIS is an SR system supporting SR, non-safety-related with special treatment (NSRST), and NST functions. The equipment used to provide the SR neutron flux measurements and functions are also used to support the NSRST and NST functions. A description of the process for assigning the safety classification of the XIS is provided in Section 7.6.1.1.2.

The XIS conforms to regulatory guides (RGs) as follows:

RG 1.47, Revision 1, *Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems*

The XIS fully conform to RG 1.47, Revision 1.

RG 1.53, Revision 2, *Application of the Single-Failure Criterion to Safety Systems*

The XIS partially conforms to RG 1.53, Revision 2. Conformance is partial because IEEE 379-2014, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” (Reference 7.6.4-2) is used rather than the version of the standard endorsed by this RG.

RG 1.75, Revision 3, *Physical Independence of Electric Systems*

Application-specific elements of the RPS partially conform to RG 1.75 Revision 3. Conformance is partial because IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018 (Reference 7.6.4-3) is used rather than the version of the standard endorsed by this RG.


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RG 1.97, Revision 5, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*

The XIS partially conforms to RG 1.97, Revision 5. The XIS is one of several systems which supports PAM functions and, alone, is not capable of fully conforming with all of the requirements of the RG. The extent of conformance to RG 1.97, Revision 5, will be provided at the operating license stage.

RG 1.100, Revision 4, *Seismic Qualification and Mechanical Equipment for Nuclear Power Plants*

The XIS fully conforms to RG 1.100, Revision 4.

RG 1.118, Revision 3, *Periodic Testing of Electric Power and Protection Systems*

The XIS partially conform to RG 1.118, Revision 3. Conformance is partial because IEEE 338-2022, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” (Reference 7.6.4-4) is used rather than the version of the standard endorsed by this RG.

RG 1.152, Revision 4, *Criteria for Use of Computers in Safety Systems of Nuclear Power Plants*

The XIS fully conforms to RG 1.152, Revision 4.

RG 1.153, Revision 1, *Criteria for Safety Systems*

The XIS partially conforms to RG 1.153, Revision 1. Conformance is partial because IEEE 603-2018, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (Reference 7.6.4-5) is used for the application-specific elements rather than the version of the standard endorsed by this RG.

RG 1.180, Revision 2, *Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems*

The XIS fully conforms to RG 1.180, Revision 2.

#### 7.6.4.2.1 Defense Line Functions

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as safety-related (SR) functions or NSRST functions as defined in Section 5.2. The XIS supports the following safety-significant functions, which are designated as SR (for DL3 functions) or NSRST (for the DL5 function) to indicate the safety classification of RPS components relied upon to perform the function:

| Function ID | Function Description | XIS Functional Support |
|---|---|---|
| DL3-RC1a | Reactor scram on high-high neutron flux | XIS includes sensors that monitor neutron flux and send signals to the RPS. |
| DL3-RC1d | Reactor scram on high-high power-to-flow Ratio | XIS includes sensors that monitor neutron flux and send signals to the RPS. |


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| Function ID | Function Description | XIS Functional Support |
|---|---|---|
| DL3-RC1f | Reactor scram on high-high positive neutron flux rate | XIS includes sensors that monitor neutron flux and send signals to the RPS. |
| DL3-RC1g | Reactor scram on high-high negative neutron flux rate | XIS includes sensors that monitor neutron flux and send signals to the RPS. |
| DL3-RC1i | Reactor scram on low power, high neutron | XIS includes sensors that monitor neutron flux and send signals to the RPS. |
| DL3- HR2 | PSP trip on high-high primary sodium temperature | XIS includes sensors that monitor neutron flux and send signals to the RPS. |
| DL3-HR3 | Intermediate Sodium Pump (ISP) trip on high-high primary sodium temperature | XIS includes sensors that monitor neutron flux and send signals to the RPS. |
| DL3-HR12 | ISP trip on high-high primary sodium level | XIS includes sensors that monitor neutron flux and send signals to the RPS. |
| DL5-PAM1 | Post-accident monitoring | XIS includes sensors that monitor neutron flux and send signals to the RPS and NIC for display as Type B and D PAM variables. |

Safety-Related Design Criteria corresponding to the SR functions applicable to the XIS are derived as described in Section 6.1.2 and identified in Section 5.2. The Safety-Related Design Criteria are incorporated into the design of the RPS part of the I&C architecture as described in Section 7.6.4.1.

The neutron flux measurement signals are replicated and isolated and are sent to the NI Coolant Temperature Monitoring and Control System and Rod Monitoring and Control System to support NST DL2 functions and the NST Audible Count Rate monitoring devices in the MCR and FHCR.

The neutron flux measurement signals are replicated and isolated and are sent to the NI Coolant Temperature Monitoring and Control System and Rod Monitoring and Control System to support NST DL2 functions and the NST Audible Count Rate monitoring devices in the MCR and FHCR.

#### 7.6.4.2.2 Principal Design Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the XIS is described below.

Consistent with PDC 2, *Design Bases for Protection Against Natural Phenomenon System Bases*, the XIS is designed so that in the event of adverse conditions due to natural phenomena, all XIS structures, systems, and components (SSCs) maintain the capability to perform their safety-significant functions.

Earthquakes – Safety-significant portions of the XIS are located in the SR RXB and NCB substructures which are designed to withstand earthquakes. SR portions of the XIS are seismically qualified to remain operable in the SR RXB and NCB substructures during and after earthquakes.


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External Flooding – The XIS accounts for external flooding hazard by being located inside of the RXB and NCB, which is located above the maximum flood level.

High Winds – To address the design basis extreme wind hazard (tornado and associated missiles), SR portions of XIS are located in the SR RXB and NCB substructures, which are designed for the extreme wind hazard.

Extreme Climate – XIS equipment is located in the climate-controlled RXB and NCB to mitigate temperature, humidity, and winter precipitation hazards.

Consistent with PDC 3, *Fire protection*, XIS equipment is located within the SR RXB and NCB substructure. The NCB walls and doors are 3-hour fire rated. XIS equipment in the NCB is located in a division-specific, 3-hour fire rated room, which minimized the likelihood of a fire affecting multiple XIS divisions at the same time. XIS equipment in the RXB is grouped by division and spaced apart to minimize the likelihood of a fire affecting multiple XIS divisions at the same time.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the SR RXB and NCB substructures protect the XIS from external hazards caused by transportation incidents, incidents at offsite industrial and chemical facilities, and external missiles resulting from a turbine failure in the plant's turbine generator set. The SR NCB substructure provides a mild environment for the SR portions of the XIS located there. Safety-significant portions of the XIS located in the RXB substructure are qualified for harsh environments.

Consistent with PDC 13, *Instrumentation and Control*, the XIS accepts input signals from instrumentation to monitor plant variables and interfaces with the RPS. All instrumentation and XIS signal conditioning associated with reactor scram, engineered safety feature (ESF), and PAM functions are designed to monitor associated variables within their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions as appropriate.

Consistent with PDC 15, *Primary Coolant System Design*, the XIS supporting DL3 reactor scram and ESF functions are designed with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded as a result of anticipated operational occurrences which have not been mitigated by DL2 functions.

Consistent with PDC 20, *Protection System Functions*, the XIS is designed to continuously monitor neutron flux and provide those signals to the RPS. The XIS is designed to support RPS, which performs these DL3 functions to ensure that analytical limits established in the safety analysis are not exceeded because of anticipated operational occurrences or design-basis events.

Consistent with PDC 21, *Protection System Reliability and Testability*, the SR portions of the XIS are arranged into four redundant and independent divisions. XIS provides process variable input to the RPS by four redundant sensing instruments which are also divided into four redundant and independent divisions.


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This divisional arrangement of XIS and RPS and 2-out-of-4 voting design within RPS allows in-service testability of the XIS. The system design allows for any division to be taken out of service for testing at any time without loss of any safety function and without loss of minimum required redundancy.The divisional arrangement of XIS and RPS and 2-out-of-4 voting design features within RPS promote high system reliability. The XIS is designed to withstand a failure of any division without loss of any safety function and without loss of minimum required redundancy. The XIS is designed to withstand a single failure in any operating division, even when one division has been taken out of service for testing. The capability to periodically test divisions independently while in-service to determine failures and losses of redundancy that may have occurred further promotes high system reliability.RG 1.75 R3.

Consistent with PDC 22, *Protection System Independence*, the SR portions of the XIS are arranged into four redundant and independent divisions. XIS provides process variable input to RPS to perform a safety function by four redundant sensing instruments which are also divided into four redundant and independent divisions. The RPS actuates reactor scram and ESF functions when at least 2 out of the 4 divisions independently determine that such action is required.

Each of the four redundant XIS divisions are independent and physically separated from the other redundant XIS divisions to the degree necessary to ensure accomplishment of safety functions.

All four redundant XIS divisions are independent and physically separated from the effects of design basis events which they are designed to mitigate. The XIS equipment is located to minimize potential damage from hazards and is qualified to substantiate design basis performance requirements.

All four redundant XIS divisions are independent and physically separated or isolated from other systems, including interconnected systems and equipment in proximity, to the degree necessary to ensure accomplishment of safety functions.

The divisional arrangement of XIS and RPS and 2-out-of-4 voting design features within RPS allow for any division to be out of service for maintenance or testing without any loss of safety function.

The XIS is 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 any safety function.

Consistent with PDC 23, *Protection System Failure Modes*, during normal operation (when neutron flux is within normal range, i.e., reactor scram is not required), the XIS maintains associated analog signals representing neutron flux. On loss of incoming power to XIS, the neutron flux signals to RPS are lost, thus RPS will place the associated circuit outputs in the safe state due to RPS employing range-checking on analog input signals from XIS.

The SR XIS equipment is located in mild and harsh environments of the SR NCB and RXB substructures, respectively. The SR XIS equipment is environmentally and seismically qualified to remain operable during anticipated adverse conditions at these locations.


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The XIS electronics modules perform continuous self-diagnostics. If self-diagnostics detect a critical fault in the electronics that could affect performance of a safety function, the system places all outputs to the safe state.

Consistent with PDC 24, *Separation of Protection and Control Systems*, the XIS is designed in accordance IEEE Std 603-2018 Clause 5.1 Single-Failure Criterion, Clause 5.6 Independence, and Clause 6.3 Interaction Between the Sense and Command Features and Other Systems.

The XIS is designed with sufficient separation from control systems such that all reliability, redundancy, and independence requirements of the XIS remain intact, even when there is a single control system failure, or failure or removal from service of any component common to the XIS or any control system.

Consistent with PDC 25, *Protection System Requirements for Reactivity Control Malfunctions*, the XIS is designed in accordance with IEEE Std 603-2018 Clause 5.1 Single-Failure Criterion, Clause 5.6 Independence, and Clause 6.3 Interaction Between the Sense and Command Features and Other Systems.

The RPS DL3 reactor scram and ESF functions supported by XIS neutron flux signals are designed to ensure that analytical limits established in the safety analysis are not exceeded as a result of anticipated operational occurrences even when a single failure of the reactivity control system has occurred.

Consistent with PDC 29, *Protection Against Anticipated Operational Occurrences*, the RPS DL3 reactor scram and ESF functions supported by XIS neutron flux signals are designed with appropriate margin to ensure that analytical limits established in the safety analysis are not exceeded as a result of anticipated operational occurrences which have not been mitigated by DL2 functions.

#### 7.6.4.3 Fundamental Design Principles in the System Design

The XIS performs both SR and NSRST functions and the special treatments are applied to all XIS SR SSCs and, as required by Clause 5.12 of IEEE 603-2018, these special treatments are also applied to all NSRST and NST SSCs.

Special treatments are initially selected as described in Section 6.3. The following programmatic special treatments are preliminarily identified as applicable to the XIS to meet quality, design, reliability, and performance requirements:

- Quality Assurance Program Description, as described in Section 8.1
- Human Factors Engineering Program Plan, as described in Section 11.2
- Equipment Qualification Program, as described in Table 8.0-1
- Post-Construction Inspection, Testing, and Acceptance Program, as described in Chapter 12
- Design Reliability Assurance Program, as described in Table 8.0-1


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SSC specific special treatments are preliminarily identified as applicable to the XIS to meet quality, design, reliability, and performance requirements.

Seismic classifications for XIS components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The seismic classification of XIS components is SCS1. Seismic design, analysis, and qualification is performed as described in Section 6.4. XIS components are qualified to withstand seismic loads associated with the safe shutdown earthquake (described in Section 2.6.2) without loss of the capability to perform safety-significant functions.

The following codes and standards are applied as special treatments to the XIS to meet quality, design, reliability, and performance requirements:

- IEEE 338-2022
- IEEE 379-2014
- IEEE 384-2018
- IEEE 420-2013, “IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations,” (Reference 7.6.4-6)
- IEEE 603-2018
- IEEE 7-4.3.2-2016 (Reference 7.6.4-7)

The XIS design includes elements of quality that minimize the potential for failures to occur that could impact SR and NSRST functions. These properties of qualification, reliability, robustness, security, and defense-in-depth features are discussed in Section 7.6.4.3.1 through Section 7.6.4.3.5.

#### 7.6.4.3.1 Qualified Instrumentation and Control

The XIS meets the quality standards established in the Quality Assurance Program Description.

The SR XIS equipment is located in mild and harsh environments of the SR NCB and RXB substructures, respectively. The SR XIS equipment is environmentally and seismically qualified to remain operable during anticipated adverse conditions at these locations.

The Equipment Qualification Program described in Section 8.0 addresses the requirements, standards, and process for qualifying the XIS equipment.

#### 7.6.4.3.2 Reliable Instrumentation and Control

The XIS is designed for high functional reliability and in-service testability commensurate with the safety functions performed.

The method for confirming specific reliability and capability targets for SSCs at the operating license stage is described in Section 6.2.


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The XIS is designed to permit periodic testing when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred. The XIS channels are included in the overall bypass schemes for placing SR channels into Maintenance Bypass to ensure that the correct functions are allocated to protection systems.

The goals of the XIS reliability include minimizing out-of-service time for repair and reducing the frequency or automation of surveillance testing. To support this, the XIS meets the following:

- The neutron flux channels are designed to accommodate periodic calibration and checks for associated protective and PAM functions as required by RG 1.118 and Clause 5.7 of IEEE 603-2018.
- XIS channels which support SR functions are designed to facilitate timely recognition, location, replacement, repair, and adjustment of malfunctioning equipment in accordance with Clause 5.10 of IEEE 603-2018.
- XIS channels which support Post-Accident Monitoring are designed to facilitate timely recognition, location, replacement, repair, and adjustment of malfunctioning equipment in accordance with Section 6.11 of IEEE 497-2016.

The power range and wide range channels require periodic calorimetric adjustments. These adjustments align the XIS readouts to match power levels determined by accurate calorimetric calculations using core thermal power as measured by temperature and coolant flow rate measurements (within a separate system). These adjustments are via individual range gain and offset adjustments locally at the XIS cabinets.

#### 7.6.4.3.3 Robust Instrumentation and Control

Robustness of the XIS design is the degree to which it can perform its SR functions in the presence of invalid inputs or abnormal environmental conditions. The robust instrumentation and control design for XIS reflects the use of design methods and adherence to engineering best practices to ensure its functions are achieved for all operational states and accident conditions. The instrumentation and control design requirements address the full range of operating environments associated with normal operation, transient, and accident conditions, as well as internal and external hazards.

The following elements address XIS system robustness:

- Single Failure Criterion - The XIS meets single failure criterion by complying with RG 1.53 and IEEE 379-2014 as required by Clause 5.1 of IEEE 603-2018.
- Completion of Protective Action - The XIS supports the SR functions of RPS as required by Clause 5.2 of IEEE 603-2018.
- System Integrity - The XIS is designed to accomplish its safety functions under the full range of applicable worst case environmental conditions, including design basis accident conditions, at installed locations as required by Clause 5.5 of IEEE 603-2018.


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- Independence - The XIS meets independence criteria between XIS channels and between the XIS SR and non-SR equipment by complying with RG 1.75 and IEEE 384-2018 as required by Clause 5.6 of IEEE 603-2018.
- Interaction between sense and command features and other systems - The XIS is powered from the NI Uninterruptible AC Power Supply System (NUP). The NUP power feeds to XIS are organized into four NSRST divisions, corresponding to the four divisions of XIS. Each of the four divisions of NUP maintains independence and separation from the other NUP divisions, as well as between the NSRST NUP divisions and NST equipment. Additionally, appropriate isolation is provided at the interconnection of the SR XIS and other NSRST systems such that no credible failure on the NSRST side of the isolation can prevent any portion of the SR XIS from performing its safety functions. These isolation device(s) are classified as part of the SR XIS as required by Clause 6.3 of IEEE 603-2018.
- Maintenance Bypasses - The XIS provides maintenance bypass capabilities as required by Clause 6.7 of IEEE 603-2018.
- Redundancy - The XIS contain sufficient redundancy to meet single failure criterion by complying with RG 1.53 and IEEE 379-2014 as required by Clause 5.1 of IEEE 603-2018.

#### 7.6.4.3.4 Secure Instrumentation and Control

Security for I&C systems is addressed through the physical security and cyber security programs that are described in Section 11.6.

#### 7.6.4.3.5 Defense in Depth

The XIS provides inputs to the DL2 control functions in the NI Coolant Temperature Monitoring and Control System and Rod Monitoring and Control System and the defense-in-depth DL3 RPS SR functions. The XIS is currently designed as an analog system that does not contain software to perform any SR functions. As such, there are no credible Common-Cause Failures or a potential to degrade or defeat the XIS safety functions as required by Clause 5.16 of IEEE 603-2018. In the event that procured components contain embedded digital devices, it is required that these devices conform with applicable industry standards and regulatory guidance. For embedded digital devices utilized to perform SR and NSRST functions, a Common-Cause Failure analysis is performed per the guidance in SECY-22-0076, “Expansion of Current Policy on Potential Common-Cause Failures in Digital Instrumentation and Control Systems,” and SRM-SECY-22-0076, “Staff Requirements – SECY-22-0076 – Expansion of Current Policy on Potential Common-Cause Failures in Digital Instrumentation and Control Systems.”

Manual control of the DL3 functions that rely on the XIS inputs is provided within the RPS as required by Clause 6.2 of IEEE 603-2018.

### 7.6.4.4 Operator Interface

The XIS operational aspects are displayed via the RPS and NIC. The NIC subsystem, Plant Monitoring and Control System, has access to XIS inputs, such as XIS flux measurements. XIS PAM information is transmitted to the RPS. This ensures XIS status and information is displayed


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as required by Clause 5.8 of IEEE 603-2018. NST audible count rate devices in the MCR and FHCR provide operators with source range indication via an audible signal which corresponds to count rate.

The XIS operator interfaces consider the guidance of NUREG-0700, “Human-System Interface Design Review Guidelines,” (Reference 7.6.4-8) and NUREG-0711, “Human Factors Engineering Program Review Model,” (Reference 7.6.4-9) as required by Clause 5.14 of IEEE 603-2018. XIS operator interfaces are addressed as part of the Human Factors Engineering Program Plan that is described in Section 11.2.

**References**

7.6.4-1 IEEE, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497-2016, July 29, 2016.

7.6.4-2 IEEE, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” IEEE 379-2014, May 16, 2014.

7.6.4-3 IEEE, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018, November 1, 2018.

7.6.4-4 IEEE, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” IEEE 338-2022, December 3, 2022.

7.6.4-5 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-2018, December 7, 2018.

7.6.4-6 IEEE, “IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations,” IEEE 420-2013, May 3, 2013.

7.6.4-7 IEEE, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” IEEE 7-4.3.2-2016, July 29, 2016.

7.6.4-8 U.S. NRC, “Human-System Interface Design Review Guidelines,” NUREG-0700, Revision 3.

7.6.4-9 U.S. NRC, “Human Factors Engineering Program Review Model,” NUREG-0711, Revision 3.


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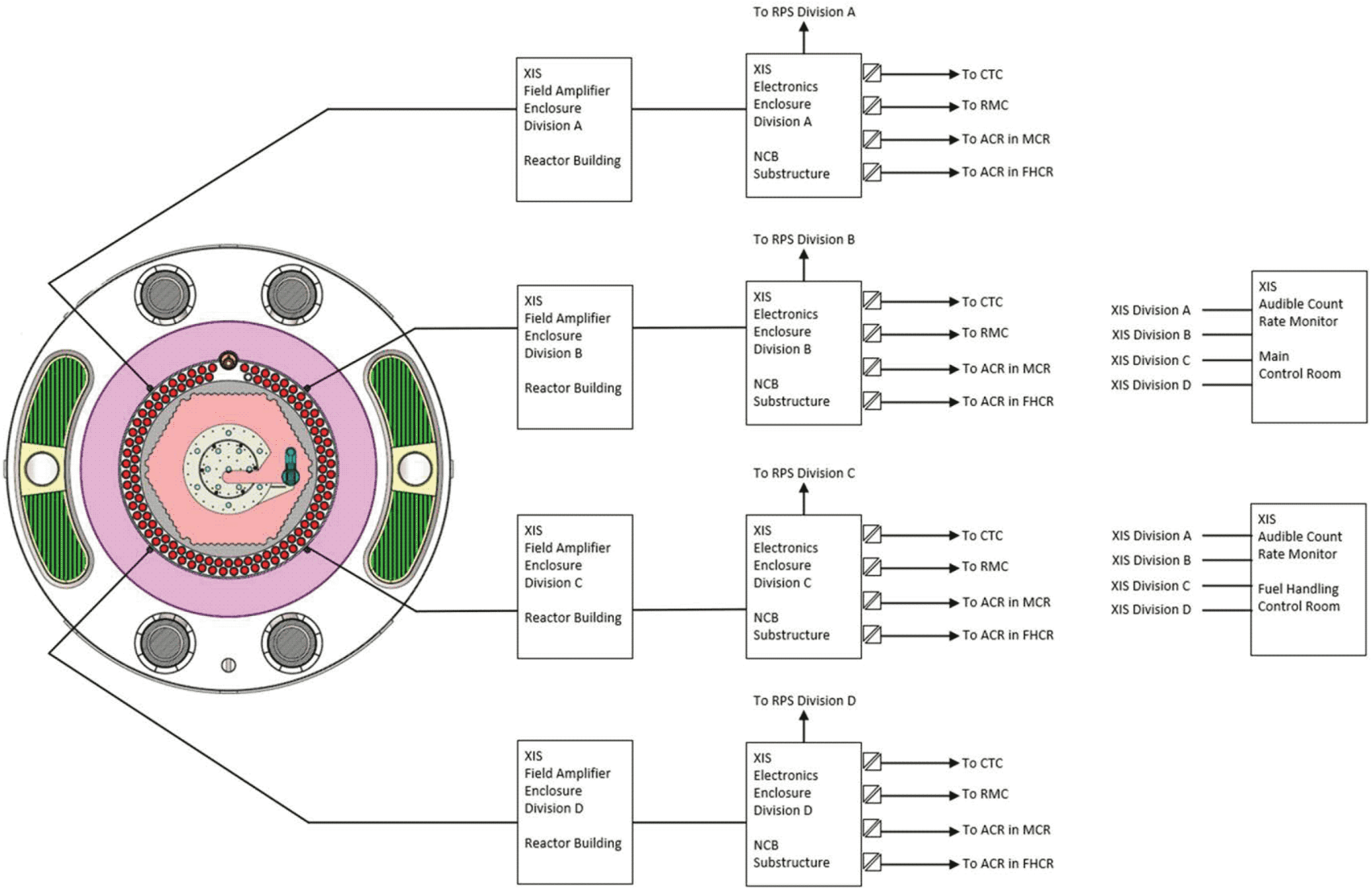
**Figure 7.6.4-1 XIS Overview**



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## 7.6.5 Reactor Instrumentation System

### 7.6.5.1 Reactor Instrumentation System Description, Architecture, and Equipment Locations

The Reactor Instrumentation System (RIS) contains instrumentation to measure selected parameters in or near the reactor vessel and transmit corresponding signals to the Reactor Protection System (RPS) and the Nuclear Island Control System (NIC) for use in protective, control, accident monitoring, and surveillance functions. The RIS is designed to ensure that SR parameters are measured and transmitted to the RPS within the time assumed in the safety analysis. The RIS is a collection of individual RIS instrument channels, where a channel is defined as a single sensor and its associated cabling and signal amplification electronics. An individual RIS instrument channel may support one or multiple functions which are classified as safety-related (SR), non-safety-related with special treatment (NSRST), or non-safety-related with no special treatment (NST). The RIS also includes cables which run from the sensors in or near the reactor vessel to the signal conditioning electronics cabinets, and from the signal conditioning electronics cabinets to the RPS and NIC terminal blocks. The NIC is described in Section 7.6.2 and the RPS in Section 7.6.3.

#### 7.6.5.1.1 Reactor Instrumentation System Architecture

The RIS has four redundant and independent divisions for parameters that provide SR inputs to the RPS via SR cabling. Signal splitters are used to duplicate the SR inputs and provide them to the NIC as NSRST and NST signals. A block diagram of the RIS is shown in Figure 7.6.5-1. A summary of variables monitored by the RIS, the number of instruments that monitor each variable, and the safety classification of the components is shown in Table 7.6.5-1.

#### 7.6.5.1.2 Reactor Instrumentation System Equipment Location

The RIS measurement sensors are located in or near the reactor vessel in the Reactor Building (RXB). Safety-significant portions of the RIS are located in the RXB and Nuclear Island Control Building (NCB) substructures. The RXB is described in Section 7.8.1. The NCB substructure is described in Section 7.8.4.

### 7.6.5.2 Reactor Instrumentation System Design Bases and Associated Safety Functions

The RIS is designed to perform assigned SR, NSRST, and NST functions and to assist in satisfying Principal Design Criteria (PDC). A description of the process for assigning Defense Line (DL) functions to the RIS is provided in Section 7.6.1.1.1. A description of the process for describing the safety classification of the RIS is provided in Section 7.6.1.1.2.

Special treatments are applied to safety-significant equipment in the RIS to meet quality, design, reliability, and performance requirements. Special treatments are initially selected based on structure, system, or component (SSC) safety classification as described in Section 6.3. The following programmatic special treatments are identified as applicable to the RIS:

- Design Reliability Assurance Program (Table 8.0-1)
- Equipment Qualification Program (Table 8.0-1)
- Post-construction inspection, testing, and analysis program (Chapter 12)


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- Quality Assurance Program Description (Section 8.1)
- Human Factors Engineering Program Plan (Section 11.2)

The RIS conforms to RGs as follows:

RG 1.22, Revision 0, *Periodic Testing of Protection System Actuation Functions*

The RIS fully conforms to RG 1.22, Revision 0.

RG 1.53, Revision 2, *Application of the Single-Failure Criterion to Safety Systems*

The RIS partially conforms to RG 1.53, Revision 2. Conformance is partial because IEEE 379-2014, "IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems," (Reference 7.6.5-1) is used rather than the version of the standard endorsed by this RG.

RG 1.75, Revision 3, *Physical Independence of Electric Systems*

The RIS partially conforms to RG 1.75, Revision 3. Conformance is partial because IEEE 384-2018, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," (Reference 7.6.5-2) is used rather than the version of the standard endorsed by this RG.

RG 1.97, Revision 5, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*

The RIS partially conforms to RG 1.97, Revision 5. The RIS is one of several systems which supports PAM functions and, alone, is not capable of fully conforming with all of the requirements of the RG. The extent of conformance to RG 1.97, Revision 5, will be provided at the OL stage.

RG 1.100, Revision 4, *Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants*

The RIS fully conforms to RG 1.100, Revision 4.

RG 1.118, Revision 3, *Periodic Testing of Electric Power and Protection Systems*

The RIS partially conforms to RG 1.118, Revision 3. Conformance is partial because IEEE 338-2022, "IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems," (Reference 7.6.5-3) is used rather than the version of the standard endorsed by this RG.

RG 1.153, Revision 1, *Criteria for Safety Systems*

The RIS partially conforms to RG 1.153, Revision 1. Conformance is partial because IEEE 603-2018, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," (Reference 7.6.5-4) is used rather than the version of the standard endorsed by this RG.

RG 1.180, Revision 2, *Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems*


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The RIS fully conforms to RG 1.180, Revision 2.

7.6.5.2.1 Defense Line Functions

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The RIS supports the following safety-significant functions, which are designated as SR (for DL3 functions) or NSRST (for DL 4 and DL 5 functions) to indicate the safety classification of RIS components relied upon to perform the function:

| Function ID | Function Description | RIS Functional Support |
|---|---|---|
| DL3-RC1b | Reactor scram on high-high hot pool temperature | RIS includes sensors that monitor primary sodium hot pool temperature and send signals from the sensors to the RPS. |
| DL3-RC1c | Reactor scram on high-high primary sodium level | RIS includes sensors that monitor primary sodium hot pool level and send signals from the sensors to the RPS. |
| DL3-RC1d | Reactor scram on high-high power-to-flow ratio | RIS includes sensors that monitor primary sodium pump (PSP) speed and PSP current and send signals from the sensors to the RPS. |
| DL3-RC1e | Reactor scram on high-high cold pool temperature | RIS includes sensors that monitor primary sodium cold pool temperature and send signals from the sensors to the RPS. |
| DL3-RC1h | Reactor scram on low-low primary sodium level | RIS includes sensors that monitor primary sodium hot pool level and send signals from the sensors to the RPS. |
| DL3-RC1j | Reactor scram on loss of primary sodium flow | RIS includes sensors that monitor PSP speed and PSP current and send signals from the sensors to the RPS. |
| DL3-HR2 | PSP trip on high-high primary sodium temperature | RIS includes sensors that monitor primary sodium cold pool temperature and send signals to the RPS. |
| DL3-HR3 | Intermediate sodium pump (ISP) trip on high-high primary sodium temperature | RIS includes sensors that monitor primary sodium cold pool temperature and send signals to the RPS. |
| DL3-HR11 | Sodium processing system (SPS) pump trip on low primary sodium level | RIS includes sensors that monitor primary sodium hot pool level and send signals to the RPS. |
| DL3-HR12 | ISP trip on high-high primary sodium level | RIS includes sensors that monitor primary sodium hot pool level and send signals to the RPS. |
| DL4-HR2 | PSP trip automatic backup | RIS includes sensors that monitor primary sodium hot pool temperature and send signals to the NIC. |
| DL4-HR3 | ISP trip automatic backup | RIS includes sensors that monitor primary sodium hot pool temperature and send signals to the NIC. |


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| Function ID | Function Description | RIS Functional Support |
|---|---|---|
| DL4-RR8 | Manual SPS pump trip on low primary sodium level | RIS includes sensors that monitor primary sodium hot pool level and send signals to the RPS. |
| DL5-PAM1 | Post-accident monitoring | RIS includes sensors that monitor Type B, C, and F PAM variables and send signals to the RPS. |

Safety-Related Design Criteria corresponding to the SR functions applicable to the RIS are derived as described in Section 6.1.2 and identified in Section 5.2. The Safety-Related Design Criteria are incorporated into the design of the RIS as described in Section 7.6.5.1.

The RIS provides signals to control Primary Heat Transport System flow, PSP flow, initiate power runback, and inhibit rod withdrawal in support of NST DL2 functions. The Primary Heat Transport System and PSP are described in Section 7.1.3. The Intermediate Heat Transport System and Intermediate Sodium Pump are described in Section 7.1.4. Sodium Cover Gas system is described in Section 7.2.3. Sodium Processing System is described in Section 7.2.4.

7.6.5.2.2 Principal Design Criteria

PDC are identified in Section 5.3. The RIS conforms to PDC as described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, in the event of adverse conditions due to natural phenomena, all RIS safety-significant SSCs maintain the capability to perform their safety functions.

Earthquakes- Safety-significant portions of the RIS are located in the RXB and NCB substructures, which are designed to withstand earthquakes. Safety-significant portions of the RIS are seismically qualified to remain operable in the RXB and NCB substructures during and after earthquakes.

External flooding- The RIS accounts for external flooding hazard by being located inside of the RXB and NCB, which is above the maximum flood level.

High winds- To address the design basis extreme wind hazard, safety-significant portions of the RIS are located in the SR RXB and NCB substructures, which are designed for the extreme wind hazard.

Extreme climate- The RIS equipment is located in the climate-controlled RXB and NCB to mitigate temperature, humidity, and winter precipitation hazards.

Consistent with PDC 3, *Fire protection*, RIS equipment is located within the SR RXB and NCB substructures. The NCB walls and doors are 3-hour fire rated. RIS equipment in the NCB is located in a division-specific, 3-hour fire rated room, which minimizes the likelihood of a fire affecting multiple RIS divisions at the same time. RIS equipment in the RXB is grouped by division and spaced apart to minimize the likelihood of a fire affecting multiple RIS divisions at the same time.


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Consistent with PDC 4, *Environmental and dynamic effects design bases*, the SR RXB and NCB substructures protect the RIS from external hazards caused by transportation incidents, incidents at offsite industrial and chemical facilities, and external missiles resulting from a turbine failure in the plant's turbine generator set. The SR NCB substructure provides a mild environment for the SR portions of the RIS located there. SR portions of the RIS located in the RXB substructure are qualified for harsh environments.

Consistent with PDC 13, *Instrumentation and control*, the RIS provides instrumentation to monitor variables in or near the reactor vessel and transmit corresponding signals to RPS and NIC for use in protective, control, accident monitoring, and surveillance functions during normal operations, anticipated operational occurrences, and postulated accidents to maintain variables and systems within prescribed operating ranges.

Consistent with PDC 15, *Primary coolant system design*, the parameters monitored by the RIS, and input to the RPS and NIC, are selected to support the design for sufficient margin to ensure that the design conditions of the safety-significant elements of the primary coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Consistent with PDC 20, *Protection system functions*, the RIS continuously monitors parameters necessary to sense accident conditions and provides signals to the RPS for the performance of protection functions.

Consistent with PDC 21, *Protection system reliability and testability*, the RIS is designed with redundant divisions for high functional reliability and inservice testability. The use of four independent divisions ensures a single failure does not result in a loss of the protection functions and ensures that removal from service of any component or division does not result in the loss of the required minimum redundancy.

Consistent with PDC 22, *Protection system independence*, the RIS is designed to assure that the effects of natural phenomena and the effects of normal operating, maintenance, testing, and postulated accident conditions on redundant divisions do not result in the loss of a safety function.

The SR portions of the RIS are arranged into four redundant and independent divisions. RIS provides process variable input to RPS with four redundant sensing instruments (or groups of instruments) which are divided into four redundant and independent divisions.

Each of the four redundant RIS divisions are independent and physically separated from the other RIS divisions to the degree necessary to ensure accomplishment of safety functions and allow maintenance and testing. The RIS equipment is located to minimize potential damage from hazards and is qualified to substantiate design basis performance requirements.

The divisional arrangement of RIS and RPS and the 2-out-of-4 voting design features within RPS allow for any division to be out of service for maintenance or testing without loss of a safety function.


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Consistent with PDC 24, *Separation of protection and control systems*, the RIS inputs to the NIC are electronically isolated and independent of the inputs to the RPS, which ensures that control system failures or removal of control system components will not impact RPS functionality. The RIS has redundant divisions, which ensure that removal of any RIS division will not impact the protection or control functions.

Consistent with PDC 29, *Protection against anticipated operational occurrences*, the design of protection and reactivity control systems assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. The RIS supports the RPS and NIC in satisfying this PDC by providing highly reliable inputs through qualification and multiple divisions.

#### 7.6.5.3 Reactor Instrumentation System Fundamental Design Principles

The RIS design includes elements of quality that minimize the potential for failures to occur in SSCs associated with SR functions. These properties of qualification, reliability, robustness, security, diversity, and defense-in-depth features are discussed in Section 7.6.5.3.1 through Section 7.6.5.3.4. The following industry codes and standards are applied as special treatments to the overall design of safety-significant equipment in the RIS:

- IEEE 603-2018
- IEEE 379-2014
- IEEE 384-2018

##### 7.6.5.3.1 Qualified Instrumentation and Control

The following industry codes and standards and RG are applied as special treatments to RIS for environmental, seismic, and electromagnetic compatibility qualification:

- IEEE/IEC 60780-323-2016 “International Standard - Nuclear facilities - Electrical equipment important to safety - Qualification” (Reference 7.6.5-5)
- IEEE/IEC 60980-344-2020, “International Standard-Nuclear Facilities-Equipment important to safety-Seismic qualification,” (Reference 7.6.5-6)
- IEEE 420-2013 “IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations” (Reference 7.6.5-7)
- RG 1.180, Revision 2, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems”

The safety-significant equipment in the RIS is assigned seismic special treatment commensurate with a seismic classification of SCS1. Seismic classification is described in Section 6.4.1.

##### 7.6.5.3.2 Reliable Instrumentation and Control

The method for confirming specific reliability and capability targets for SSCs at the operating license (OL) stage is described in Section 6.2.


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Repair

SR equipment in the RIS is designed to facilitate timely recognition, location, adjustment, repair, and replacement of malfunctioning equipment.

Identification

SR equipment in the RIS is distinctly identified for each redundant portion of the system, in accordance with the requirements of IEEE 384-2018 and IEEE 420-2013.

Auxiliary Features

Some equipment in the RIS provides signals to the RPS and NIC that support Post-Accident Monitoring functionality. The design of Post-Accident Monitoring capabilities is informed by IEEE 497-2016 “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations” (Reference 7.6.5-8). A description of the conformance to IEEE 497-2016, as endorsed by Regulatory Guide 1.97, Revision 5, will be provided at the OL stage.

Human Factors

RIS equipment is designed in accordance with IEEE 420-2013, including requirements regarding human factors. The display of information provided from the RIS to the RPS and NIC is reviewed per the Human Factors Engineering Program Plan described in Section 11.2.

Reliability

IEEE 338-2022 is applied as a special treatment to the safety-significant equipment in the RIS to ensure that reliability and capability targets are met.

Capability for Testing and Calibration

SR portions of the RIS are capable of testing and calibration during power operation without loss of SR functions at the system level. The design of RIS supports checking each channel (including sensor and any electronics) required for a safety function by one of the following means:

- cross-checking between redundant channels with available readouts
- introducing and varying substitute input
- perturbing the monitored variable

Instrument channels supporting Post-Accident Monitoring have testing capability to facilitate timely recognition, location, adjustment, and repair of malfunctioning equipment.

Maintenance Bypass

SR portions of the RIS are designed such that when one division (or a portion of one division) is bypassed for maintenance, there is no loss of safety function at the system level. A description of the implementation of maintenance bypass capabilities for the RIS will be provided at the OL stage.


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#### 7.6.5.3.3 Robust Instrumentation and Control

Robustness of the RIS design is the degree to which it can perform its SR functions correctly in the presence of invalid inputs or abnormal environmental conditions. The following elements (as appropriate) are considered for system robustness:

Single Failure

Single failure is addressed by applying IEEE 379-2014 as a special treatment to the design of RIS as described in Section 7.6.5.3.

Completion of Protective Action

The RIS instrumentation provides selected measurements in or near the reactor vessel and transmits corresponding signals to the RPS and the NIC for use in protective, control, accident monitoring, and surveillance functions.

System Integrity

Safety-significant equipment in the RIS is designed to perform DL functions provided in Section 7.6.5.2.1 under the full range of applicable conditions.

Independence

IEEE 384-2018, which provides detailed criteria for independence, is applied as a special treatment to safety-significant equipment in the RIS.

Information Displays

The RIS instrumentation provides selected measurements in or near the reactor vessel and transmits corresponding signals to the RPS and the NIC for use in protective, control, accident monitoring, and surveillance functions. No information displays are part of the RIS.

Automatic Control

The RIS instrumentation provides selected measurements in or near the reactor vessel and transmits corresponding signals to the RPS and the NIC for use in protective, control, accident monitoring, and surveillance functions. No automatic control is performed by the RIS.

Interaction Between Sense and Command Features and Other Systems

The RIS instrumentation provides selected measurements in or near the reactor vessel and transmits corresponding signals to the RPS and the NIC for use in protective, control, accident monitoring, and surveillance functions. The RIS does not have command features which interact with other systems.

Operating Bypasses

A description of the implementation of operating bypass capabilities for the RIS will be provided at the OL stage.


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Redundancy

The RIS includes instrument channels which support SR, NSRST, and NST sensing functions. The architecture of the RIS is such that appropriate channel redundancy is provided through the use of four redundant and independent divisions.

Diversity in Support of defense-in-depth to Address Common-Cause Failures

A description of diversity in support of defense-in-depth to address Common-Cause Failures is provided in NAT-4950, “I&C Architecture and Design Basis Topical Report” Revision 1 (Reference 7.6.5-9). NAT-4950 is incorporated by reference into the SAR. See Section 1.4.2 for a list of documents incorporated by reference into the SAR.

Predictable and Repeatable Behavior

SR channels of the RIS are designed so that their behavior is predictable, such that the same set of inputs will produce the same outputs within a known, guaranteed cycle time, and functions are consistently completed within the response time allocated by safety analysis. A description of the predictable and repeatable behavior of the RIS will be provided at the OL stage.

7.6.5.3.4 Secure Instrumentation and Control

Security for I&C systems is addressed through the physical security and cyber security programs that are described in Section 11.6.

References

7.6.5-1 IEEE, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” IEEE 379-2014, May 16, 2014.

7.6.5-2 IEEE, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018, November 1, 2018.

7.6.5-3 IEEE, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” IEEE 338-2022, December 3, 2022.

7.6.5-4 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-2018, December 7, 2018.

7.6.5-5 IEEE/IEC, “International Standard - Nuclear facilities - Electrical equipment important to safety - Qualification,” IEEE/IEC 60780-323-2016.

7.6.5-6 IEEE/IEC, “International Standard - Nuclear facilities - Equipment important to safety-Seismic qualification,” IEEE/IEC 60980-344-2020.

7.6.5-7 IEEE, “IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations,” IEEE 420-2013, May 3, 2013.


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7.6.5-8 IEEE, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497-2016, July 29, 2016.

7.6.5-9 TerraPower, “I&C Architecture and Design Basis Topical Report,” NAT-4950, Revision 1, ML24068A187.


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**Table 7.6.5-1 RIS Variable Summary**

| Variable | Number of Instruments | Safety Classification |
|---|---|---|
| Primary Sodium Hot Pool Temperature | 4 | SR |
| Primary Sodium Hot Pool Level | 4 | SR |
| Primary Sodium Cold Pool Temperature | 8 | SR |
| PSP Shaft Speed | 8 | SR |
| PSP Current Detector | 8 | SR |
| Reactor Vessel-Guard Vessel Liquid Sodium Detection | 3 | NSRST |
| Core Exit Temperature | 15 | NSRST |
| Reactor Head Temperature | 3 | NST |


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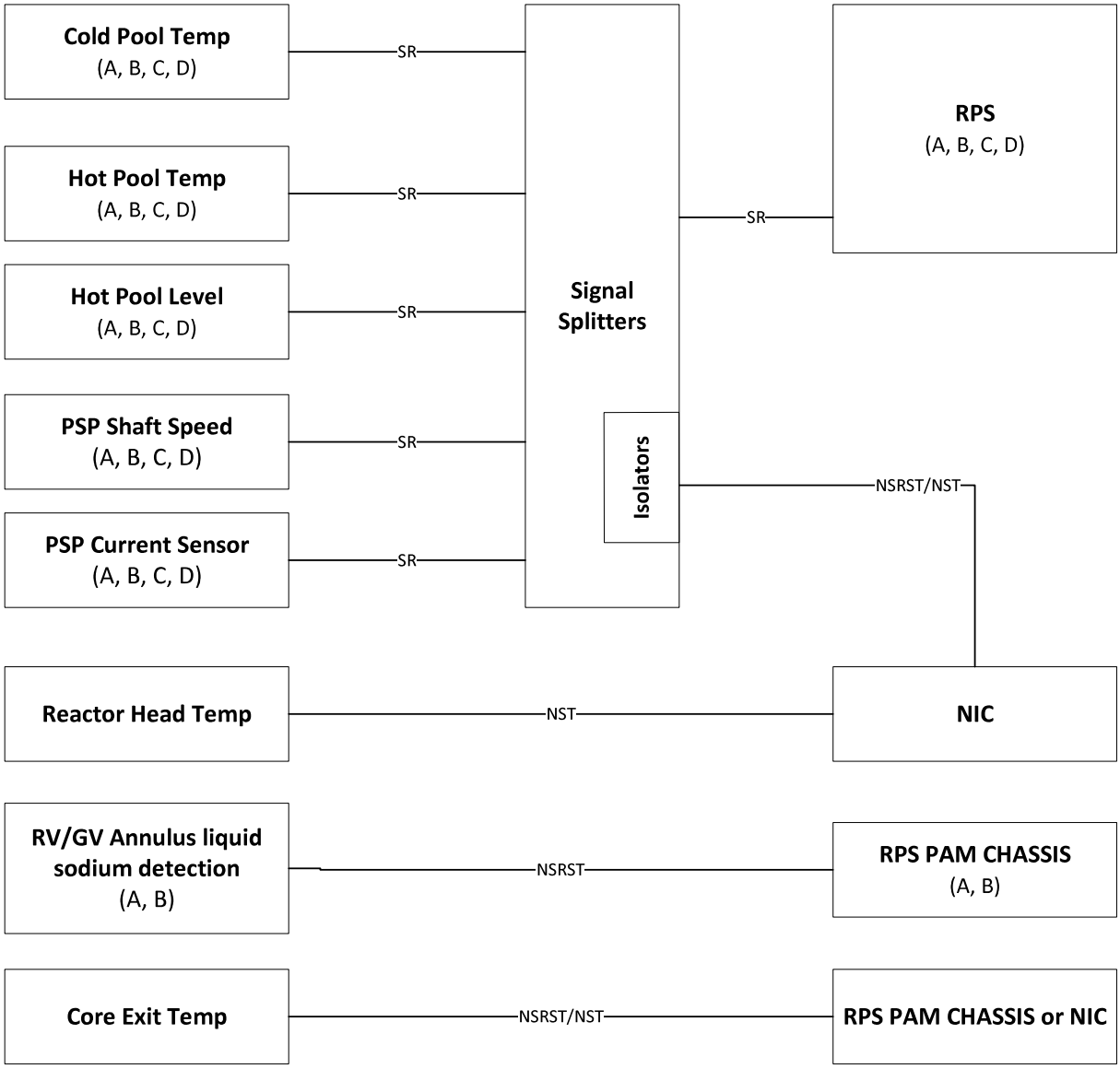
**Figure 7.6.5-1 Reactor Instrumentation System Block Diagram**





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## 7.6.6 Radiation Monitoring System

### 7.6.6.1 System Description, Architecture, and Equipment Locations

The Radiation Monitoring System (RMS) is a digital system with non-safety-related with special treatment (NSRST) and non-safety-related with no special treatment (NST) functions that provides radiation detection for systems, buildings, and release pathways throughout the plant at strategic interface points. Continuous radiation monitoring and sampling analysis of selected radioactive processes are performed, as well as monitoring for the selected release points where radioactive effluents can leave the site boundary.

The RMS operates continuously and provides adequate information and warning to ensure personnel exposure does not exceed 10 CFR 20 limits.

Each RMS monitor sends activity, monitor health, control, and status information to the Digital Radiation Monitor System (DRMS). The DRMS is a data acquisition computer system that receives information from each radiation monitor, regardless of system assignment, via an isolated digital communication loop. The DRMS provides a direct digital (via communication link) connection to the Auxiliary Monitoring and Control System (AMC) (described in Section 7.6.2) which provides the interface between the operator in the Main Control Room (MCR) and the DRMS.

Certain monitors from the RMS are credited for Post-Accident Monitoring (PAM) Type C, D, E, or F, as defined in IEEE 497-2016 “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” (Reference 7.6.6-1). Signals from RMS monitors credited for PAM Type C and F are sent to the Reactor Protection System via a parallel analog signal in addition to the signal to the DRMS. The processing and display of PAM information by the Reactor Protection System is described in Section 7.6.3. RMS monitors credited only for PAM Type D or E are sent only to the DRMS. A monitor may be credited for multiple combinations of PAM types.

The RMS consists of three main monitoring groups:

- Area Radiation Monitoring (ARM)
- Effluent Radiation Monitor (ERM)
- Process Radiation Monitoring (PRM)

The ARMs warn personnel of an existing radiation health hazard in a prescribed area of the plant, with select NST monitors providing PAM Type E signals to the AMC via the DRMS connection. In addition, the Head Access Area (HAA) ARMs are credited for PAM Type C, D, and E.

The ERMs provide early indication of potential health hazards and releases to the environment and the public and provide the assurance that personnel exposure does not exceed 10 CFR 20 limits. Select monitors also provide PAM Type E signals to the instrumentation and control (I&C) AMC via the DRMS connection. The ERMs are located on the monitored plant stack, which accepts input from Gaseous Radwaste Processing System and Nuclear Island Heating, Ventilation, and Air Conditioning Systems. These system outputs are also individually monitored


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prior to sending gaseous effluent to the monitored plant stack. In addition, there is a tritium sample skid located on the monitored plant stack to ensure no unmonitored tritium is released from the site. There are no liquid effluent radiation monitors in the RMS as there are no liquid effluents possible in the design as described in Section 9.1.

The PRMs provide an early indication of a potential health hazard or equipment malfunction, with select monitors providing PAM Type E signals to the I&C AMC via the DRMS connection. In addition, the Sodium Cover Gas (SCG) process gas PRMs are credited for PAM Type C, D, E, and F.

#### 7.6.6.1.1 System Architecture

The RMS provides local and control room indication of plant radiological conditions in areas that are required to be monitored for area radiation levels, airborne activity, and radioactive material in various process and effluent streams. Depending on the variable being monitored, the system output may be local or control room indication of area radiological conditions, an alarm warning, or an input to the AMC (described in Section 7.6.2) via the DRMS for NST display and control functions, as needed.

The individual monitor microprocessor communicates with the DRMS via direct communication. The information displays in the MCR on operator consoles via the DRMS to AMC digital communication interface. Additionally, radiation monitors that support NSRST functions utilize a stand-alone, isolated interface and display Radiation Indicator Controller that communicates directly with the monitor it is assigned to. These controllers are used to supplement the connection to the DRMS for monitors supporting NSRST functions.

The interface diagram shown in Figure 7.6.6-1 provides a high-level view of the RMS interfaces.

#### 7.6.6.1.2 Equipment Locations

RMS components are located in various areas within the plant. Each monitor detection system is located close to the detector location to minimize signal losses and to provide local indication and control of the radiation monitor. General monitor locations for RMS are:

- The areas, buildings, and rooms designated for ARM locations are Reactor Building (RXB) (Section 7.8.1), Reactor Auxiliary Building (RAB) (Section 7.8.3), Fuel Handling Building (FHB) (Section 7.8.2), Fuel Auxiliary Building (FAB) (Section 1.1.4.4.6), Fuel Transport and Storage System (Section 7.3), Nuclear Island Control Building (NCB) (Section 7.8.4), Solid Radwaste Processing System (Section 9.3.1), and Emergency Response Facility Primary (Section 7.6.7.3).
- The area designated for ERM locations are at the monitored plant stack. This includes a tritium sample skid.
- The processes and airborne locations designated for PRMs are Fuel Pool Purification (Section 7.3.1), Pool Immersion Cell (Section 7.3.1), MCR (airborne) (Section 7.6.7.1), RXB (airborne), Sodium Processing System Cell (airborne) (Section 7.2.4), FHB,


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(airborne), SCG Cell (airborne), RAB (airborne), FAB (airborne), Gaseous Radwaste Processing System, Liquid Radwaste Processing System, Solid Radwaste Processing System (Section 9.3.1), SCG (Section 7.2.3), and Sodium Processing System.

The DRMS is located within the MCR computer room (Figure 1.1-13). Operator access to each monitor is via the Visual Display Unit in the MCR by way of the AMC interface with the DRMS. The Type C and F PAM monitors are classified as NSRST and have individual Radiation Indicator Controllers located within the NCB cabinet room or the divisional Reactor Protection System cabinet room (Figure 1.1-12).

7.6.6.2 System Design Bases and Associated Safety-Significant Functions

The RMS provides radiation detection for systems, buildings, and release pathways throughout the plant at strategic interface points. The RMS is capable of supporting NSRST functions during all normal, anticipated operational occurrence (AOO), post-accident, and maintenance plant operations. The operating conditions for which each RMS instrument is designed to meet are based on the specific system or building the instrument is monitoring. A description of the process for assigning the safety classification of the RMS is provided in Section 7.6.1.1.2.

The RMS conforms to regulatory guides (RGs) as follows:

RG 1.21, Revision 3, *Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste*

The RMS fully conforms to RG 1.21, Revision 3.

RG 1.97, Revision 5, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*

The RMS partially conforms to RG 1.97, Revision 5. The RMS is one of several systems which supports PAM functions and, alone, is not capable of fully conforming with all of the requirements of the RG. The extent of conformance to RG 1.97, Revision 5, will be provided at the operating license stage.

RG 1.100, Revision 4, *Seismic Qualification and Mechanical Equipment for Nuclear Power Plants*

RMS components that are classified as NSRST fully conform to RG 1.100, Revision 4.

RG 1.180, Revision 2, *Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems*

RMS components that are classified as NSRST fully conform to RG 1.180, Revision 4.


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7.6.6.2.1 Defense Line Functions

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The RMS supports the following safety-significant functions, which are designated as NSRST to indicate the safety classification of RMS components relied upon to perform the function:

| Function ID | Function Description | RMS Functional Support |
|---|---|---|
| DL4-RR1c | Head Access Area (HAA)<br>HVAC operations following postulated release | The RMS detects radioactivity and sends signals to the UMC. |
| DL4-RR4a | Automatically close SCG isolation valves on leak detection | The RMS detects radioactivity and sends signals to the AMC. |
| DL4-RR4b | SCG cells barrier isolation<br>Leak detection | The RMS detects radioactivity and sends signals to the UMC. |
| DL5-PAM1 | Post-accident monitoring | The RMS detects radioactivity and sends signals to the RPS for display of Type C and F PAM variables. |

7.6.6.2.2 Principal Design Criteria

PDC are identified in Chapter 5.3. Compliance with PDC that are applicable to the RMS is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena,* NSRST RMS components are located in structures designed to withstand the effects of applicable external hazards without the loss of structural integrity. In this way, the design of the RMS ensures that the NSRST functions will be performed in the event of an earthquake, external flooding, high winds, and external climate.

Consistent with PDC 13, *Instrumentation and control*, the RMS monitors plant area, effluent, and process radiation variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions. The RMS provides radiological information to inform on the integrity of the reactor core, safety-significant elements of the primary coolant boundary, and functional containment. Controls are provided by RMS to maintain these variables and systems within prescribed operating ranges.

Consistent with PDC 60, *Control of release of radioactive materials to the environment,* the Gaseous Radwaste Processing System, Liquid Radwaste Processing System, Solid Radwaste Processing System SSCs suitably control the release of radioactive materials in gaseous and liquid effluents, handling radioactive solid wastes produced during normal reactor operation, including AOOs, and sufficient holdup capacity for retention of gaseous and liquid effluent


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containing radioactive materials. The RMS provides the monitoring component in support of these functions by indicating waste processing has provided sufficient radiological margin to allow for waste release (gaseous, solid) or the need for further processing and re-use (liquid).

Consistent with PDC 64, *Monitoring radioactivity release,* the RMS provides a means for monitoring all NI building atmospheres with the potential for radioactive release, spaces containing components for primary system sodium and cover gas cleanup and processing, effluent discharge paths, and the plant environments, via the Nuclear Island Heating, Ventilation, and Air Conditioning System, for radioactivity that may be released from normal operations, including AOOs, and from postulated accidents

#### 7.6.6.3 Fundamental Design Principles in the System Design

System performance of each RMS monitor meets the following goals:

- Selected instrument ranges envelope the complete minimum and maximum conditions over normal operations, AOOs, postulated accidents, and post-accident conditions, as applicable.
- Selected setpoints provide indication of monitored radiation levels exceeding nominal limits.
- No automatic control actions are provided by RMS; and, therefore, developed setpoints only indicate in the MCR and trigger alarms locally or remotely, as required.
- Response times to meet plant requirements.

Special treatments are initially selected based on SSC safety classification as described in Section 6.3.

The following programmatic special treatments are preliminarily identified as applicable to the RMS to meet quality, design, reliability, and performance requirements:

- Design Reliability Assurance Program, as described in Table 8.0-1
- Equipment Qualification Program, as described in Table 8.0-1
- Quality Assurance Program Description, as described in Section 8.1
- Post-Construction Inspection, Testing, and Analysis Program, as described in Chapter 12

Seismic classifications for RMS components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The seismic classification of RMS components is SCN1. Seismic design, analysis, and qualification is performed as described in Section 6.4.


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The following codes and standards are applied as special treatments to the RMS to meet quality, design, reliability, and performance requirements:

- IEEE 338-2022 “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” (Reference 7.6.6-2)
- IEEE 603-2018 “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (Reference 7.6.6-3)
- IEEE 379-2014 “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” (Reference 7.6.6-4)
- IEEE 497-2016

Additionally, the RMS design includes elements of quality that minimize the potential for failures to occur in the NSRST functions. These properties of qualification, reliability, robustness, security, and defense-in-depth features are discussed in Section 7.6.6.3.1 through Section 7.6.6.3.5.

#### 7.6.6.3.1 Qualified Instrumentation and Control

The RMS meets the quality standards established in the Quality Assurance Program Description (QAPD).

RMS components are designed to the environmental requirements specified for the location and for continuous operation within normal and abnormal minimum and maximum ambient conditions. The instrumentation and equipment used for NSRST functions meet seismic and environmental requirements per IEEE-497-2016.

The Equipment Qualification Program described in Table 8.0-1 addresses the requirements standards and process for qualifying the RMS system equipment.

#### 7.6.6.3.2 Reliable Instrumentation and Control

The RMS is designed to perform its functions in a reliable manner. The following design features provide RMS reliability:

- Redundant monitoring points based on area, process, and effluent component classifications
- Reliable sources of power provided by the Nuclear Island Uninterruptible AC Power Supply System
- Self-testing to detect individual monitor failures
- Quality design process for both hardware and software design which meets the QAPD

The method for confirming specific reliability and capability targets for SSCs at the operating license stage is described in Section 6.2.


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Radiation monitors credited with an NSRST function are under a surveillance testing program where there are two main types of testing activities that are performed on the monitors:

- Surveillance testing
- Periodic calibration

#### 7.6.6.3.3 Robust Instrumentation and Control

The robust RMS I&C design reflects the use of design methods and adherence to engineering best practices to ensure its functions are achieved for all operational states and accident conditions.

Redundancy

The RMS design, where necessary, uses overlapping area coverage in area applications and redundancy in process monitoring applications credited for NSRST functions, such as the HAA and SCG monitoring points which are implemented on redundant monitors. Effluent monitoring has inherent redundancy in that the combined plant stack has two dedicated wide range gas monitors. When coupled with the area and process monitoring, this provides for point of origin activity detection.

This design provides the following:

- Any single component fault does not prevent the RMS from performing the assigned NSRST functions.
- Redundant monitors are part of the RMS design where monitors support NSRST functions.

Power Supply

The RMS monitors are powered by the Nuclear Island Uninterruptible AC Power Supply System which provides uninterruptible AC power to each monitor. The Nuclear Island Uninterruptible AC Power Supply System provides 72-hour battery backup to the NSRST monitors for the HAA and the SCG locations. The Nuclear Island Uninterruptible AC Power Supply System is described in Section 7.7.1.

#### 7.6.6.3.4 Secure Instrumentation and Control

Security for I&C systems is addressed through the physical security and cyber security programs that are described in Section 11.6.

#### 7.6.6.3.5 Defense-in-Depth

Post accident monitoring is required for sufficient defense-in-depth since it is necessary for providing information following a postulated release that may be used to inform further response or mitigation.


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#### 7.6.6.4 Operator Interface

The DRMS provides signals to the AMC for operator use and display. The DRMS is capable of independent indication, alarming, and recording of radiation levels. The DRMS can also perform control functions on monitors such as source check, setpoint control, manipulation of valves and pumps, and database changes or queries. All monitor information is available for display in the MCR on operator consoles via the DRMS to AMC digital communication interface. The DRMS does not impose range or detection limits on any radiation monitor. The MCR alarm functions will be generated by the AMC when a radiation monitors activity signal, received via the DRMS, exceeds its setpoint value. The ARMs provide local alarms for the assigned areas. The ERM and PRM monitors can also alarm locally.

Operator access to each monitor is via the Visual Display Unit in the MCR by way of the AMC interface with the DRMS. The radiation monitors used for NSRST functions also have individual Radiation Indicator Controllers located either within the divisional RPS cabinet room or in the NCB cabinet room.

**References**

7.6.6-1 IEEE, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE 497-2016, July 29, 2016.

7.6.6-2 IEEE, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” IEEE 338-2022, December 3, 2022.

7.6.6-3 IEEE, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE 603-2018, December 7, 2018.

7.6.6-4 IEEE, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” IEEE 379-2000, March 9, 2001.



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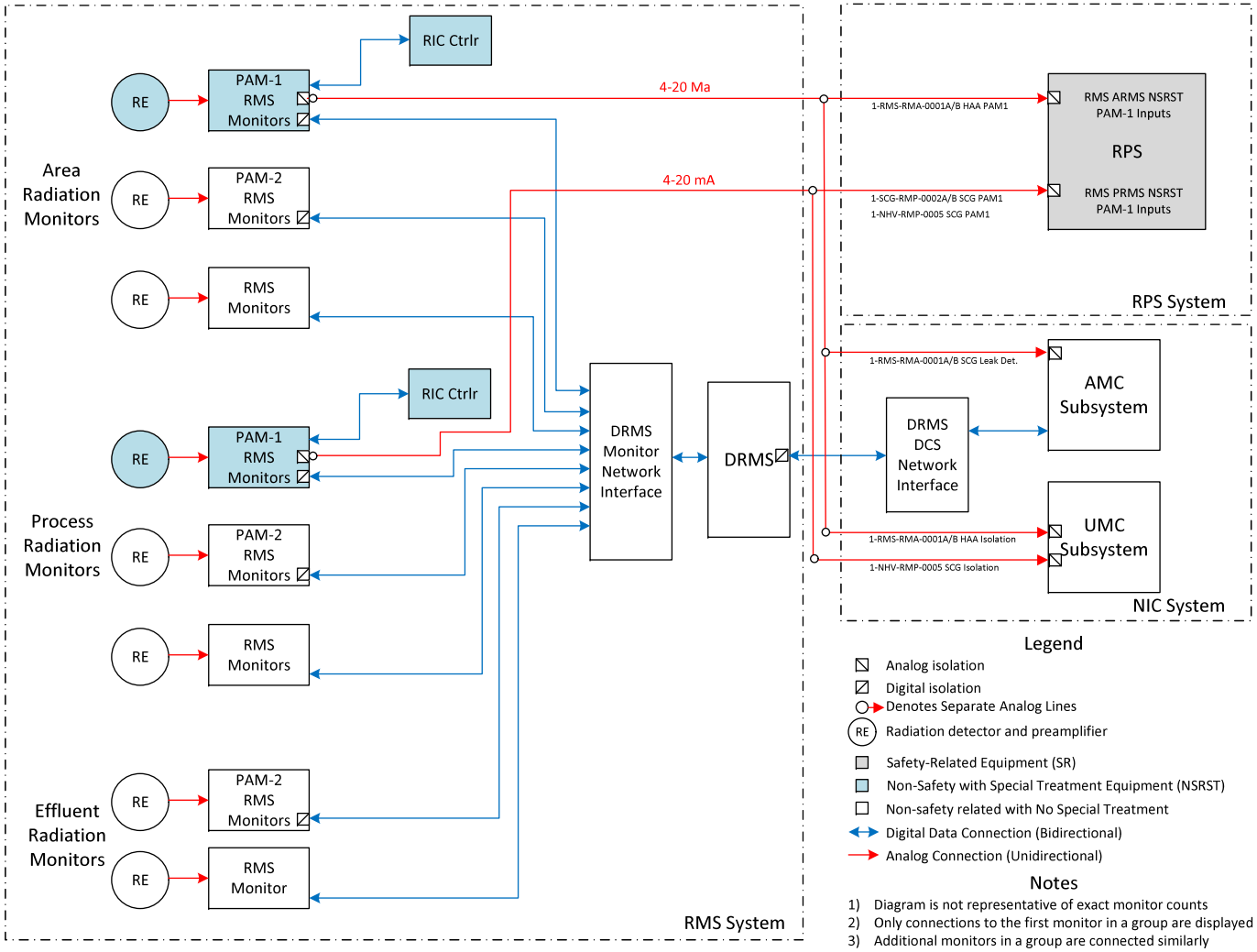
## Figure 7.6.6-1 RMS Interface Diagram





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## 7.6.7 Control Room and Indications

A control room and indications are provided that satisfy Principal Design Criterion (PDC) 19, Control Room, and the requirements of the emergency plan. The PDC are described in Section 5.3. The emergency plan is described in Section 11.3.

### 7.6.7.1 Main Control Room

The Main Control Room (MCR) is designed to satisfy requirements described in PDC 19, Control Room, (Section 5.3.2.10). Equipment necessary to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions is described in Section 7.6.7.1.1.

The MCR is located within the non-safety-related with special treatment (NSRST) Nuclear Island (NI) Control Building superstructure, as shown in Figures 1.1-13 through 1.1-15, and protected from external hazards as described in Section 7.8.4.2. There are no safety-related operator actions required.

The NI Heating, Ventilation, and Air Conditioning System controls the environmental conditions during normal operations and takes measures to ensure MCR habitability for radiological, sodium aerosol, and chemical hazards associated with accident conditions required by PDC 19. The NI Heating, Ventilation, and Air Conditioning System is described in Section 7.5.1.

#### 7.6.7.1.1 Plant Control

The MCR contains equipment that allows operators to initiate or take manual control of functions associated with the Reactor Protection System (RPS) and NI Control System (NIC) via human system interfaces located at the workstations described in Section 7.6.7.1.2. The MCR has equipment that allows the operators to maintain awareness of the status of the plant and automated functions.

The RPS provides the capability to manually initiate each reactor scram and event sequence family from the MCR. Each of the four RPS divisions has a division-specific Special Video Display Unit (SVDU) in the MCR that displays parameters and status of the associated RPS division only. The RPS is described in Section 7.6.3.

The NIC provides the capability to manually initiate or take manual control of each function for the NIC subsystems described in Section 7.6.2. A description of the manual control of NIC functions is provided in Section 7.6.7.1.2 in the discussion of NI and Energy Island (EI) operator workstations.

#### 7.6.7.1.2 MCR layout

The conceptual layout of the MCR is depicted in Figure 7.6.7-1. The MCR contains the following equipment for the RPS, NIC, and EI Integrated Control System (EIC):

- RPS workstation
- Group-View Display System (GVDS)


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- NI and EI operator workstations
- Supervisor workstation
- Communications workstation

RPS Workstation

The RPS workstation contains four SVDUs, one for each of the four RPS divisions. Each RPS divisional SVDU displays parameters and status of the associated RPS division only. The Division A and B SVDUs are also capable of displaying Post-Accident Monitoring (PAM) variables assigned to that division. PAM variables are described in Section 7.6.1.2. The RPS workstation contains controls necessary to manually initiate each reactor scram and event sequence family. The RPS divisional SVDUs and manual controls are classified as NSRST and are assigned seismic special treatments as part of NSRST Seismic Special Treatment Category Graded Seismic Risk Significant category A. Seismic special treatment is described in Section 6.4.1.

Group-View Display System

The GVDS is a wall-mounted array of Visual Display Units (VDUs) that present software-based information. The GVDS is used to support situation awareness and crew coordination by displaying key parameters and alarm information to aid understanding of plant status. The GVDS primarily consists of a set of pre-made displays that provide relevant overviews of plant component and parameter status including emergency operating procedure entry conditions.

The GVDS for the NI and the EI each contain a four VDU array that displays a consolidated overview of plant status and alarm information. Two of the VDUs on the GVDS contain pre-configured plant overview information displays. The remaining VDUs on the GVDS are user-configurable, including access to closed circuit televisions. GVDS VDUs do not have plant control functions, as these are designed for monitoring purposes only. GVDS display contents are managed at the supervisor workstation.

The GVDS includes alarm overview information. Additional information about alarms is accessible at the operator and supervisor workstations.

All information that is displayed on the GVDS is also accessible via the operator and supervisor workstation VDUs. The GVDS also contains a wall-mounted framework constructed to hold the VDUs intact in a seismic event. The GVDS framework design supports easy access and replacement of VDUs.

NI and EI Operator Workstations

The NI and EI operator workstations contain the controls and indications provided through the NIC and EIC needed by the NI and the EI operators to perform all of the tasks assigned to those roles. The human-system interface on the NI operator workstations is used to monitor and control the functions of the NIC and EIC. The human-system interface on the NI and EI operator workstations also include the integrated Safety Parameter Display System functions, which are


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described in Section 7.6.7.1.3. The NI and EI operator workstations are desk-style consoles designed for sit-down operation and contain software-based controls and information. The NI and EI operator workstations are designed based on modular, commercial desk components.

Two operator workstations are provided for the NI portion of the MCR and two operator workstations are provided for the EI portion of the MCR. Each workstation desk contains four commercial VDUs, one hardwired optical mouse, and one hardwired mechanical keyboard driven by one central processing unit.

Supervisor Workstation

The supervisor workstation is a desk-style console designed for sit-down operation and contains indications (no controls) provided through the NIC and EIC to support the information needs associated with the supervisor role. The supervisor workstation is an L-shaped design based on modular, commercial desk components. The supervisor workstation desk contains a total of two central processing units, four commercial VDUs, two hardwired optical mice, and two hardwired mechanical keyboards.

Communications Workstation

The communications workstation is a desk-style console designed for sit-down monitoring and contains indications (no controls) provided through the NIC and EIC. The communications workstation is designed based on modular, commercial desk components.

The communications workstation desk contains two commercial VDUs, one central processing unit, one hardwired optical mouse, and one hardwired mechanical keyboard.

One printer is provided in the MCR to support the operator's ability to print screens, log files, and associated information.

NIC human-system interfaces and VDUs are NST devices. All equipment associated with the EIC is classified as NST.

### 7.6.7.1.3 Conformance to Human Factors Engineering and Compliance with 10 CFR 50.34(f)

The Human Factors Engineering program described in Section 11.2 is applied to the design of the MCR and equipment described in Section 7.6.7.1.2 to satisfy 10 CFR 50.34(f)(2)(iii). The indications provided in the MCR comply with the technologically relevant portions of 10 CFR 50.34(f).

The Safety Parameter Display System functionality includes:

- Display of variables in the full range expected during normal operation and abnormal or emergency conditions
- Provisions for data trending
- An indication when limits are approached or exceeded using simple diagrams with well-presented safety margin


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- Display of safety system status

The Safety Parameter Display System includes displays that assist the plant staff in executing emergency procedures, severe accident procedures, and site emergency response to limit the effect of accident conditions.

Alarms are provided to indicate deviations from normal operational limits, loss of availability of safety systems, and unplanned unavailability of standby equipment.

The monitoring of in-plant radiation and airborne radioactivity is performed by the RMS as described in Section 7.6.6.

The display of PAM variables is described in Section 7.6.1.2.

#### 7.6.7.2 Remote Shutdown Complex

In the event that the MCR cannot be inhabited, operators are evacuated to the Remote Shutdown Complex (RSC). The RSC is located in the Division B RPS Vault, Distributed Control System, and control rod drive cabinet room as shown on Figure 1.1-12. The RSC is classified as NSRST and is located in the safety-related NI Control Building substructure as described in Section 7.8.4.1. The MCR and RSC are located in close proximity to support prompt staffing of the RSC in the event of an evacuation of the MCR.

The ventilation necessary to support operator habitability of the RSC is provided via the use of a bottled air injection system. The bottled air injection system is initiated manually to provide habitability for three operators and has a 72-hour mission time.

Self-contained breathing apparatus and hazardous materials suits are located in the RSC supporting personnel leaving the area to perform equipment manipulation, if required. An intermediate chamber sized for one individual with hazardous materials and self-contained breathing apparatus gear to enter and exit the RSC is provided. The intermediate chamber is capable of being pressurized and contains doors designed to facilitate entry and exit into the RSC once the bottled air injection system has been activated.

The RSC contains equipment that provides indications and controls necessary to maintain the unit in a safe condition during shutdown with a capability for subsequent safe shutdown of the reactor, as required by PDC 19. These indications, and the capabilities necessary for prompt shutdown of the reactor using local equipment and potential capability for subsequent safe shutdown of the reactor through suitable procedures, as required by PDC 19, Control Room, are described in Section 7.6.7.2.1.

##### 7.6.7.2.1 Indications and Capabilities

The RSC contains two RPS SVDUs capable of displaying parameters and status associated with RPS Divisions A and B. The Division A and B RPS SVDUs also display PAM variables assigned to that division. The RPS is described in Section 7.6.3. PAM variables are described in Section 7.6.1.2.


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Located in the Division B RPS Vault, Distributed Control System, and control rod drive cabinet room, the location of the RSC provides access to nearby equipment capable of tripping all Reactor Trip Breakers and event sequence family components. Means of tripping equipment necessary for prompt reactor shutdown and safe shutdown are located in the RPS Vaults, Distributed Control System, and control rod drive cabinet rooms, as well as the Primary Sodium Pump and Intermediate Sodium Pump trip breaker rooms, shown in Figure 1.1-12. This equipment is accessed from the RSC using self-contained breathing apparatus and hazardous materials suits, as described in Section 7.6.7.2 above.

7.6.7.2.2 Conformance to Human Factors Engineering

The Human Factors Engineering program described in Section 11.2 is applied to the design of the RSC and equipment described in Section 7.6.7.2.1.

7.6.7.3 Emergency Response Facilities

Emergency Response Facilities (ERF) capable of performing the functions of the Primary ERF and Backup ERF described in Section 11.3.9.1 and the methods of providing indications necessary to support these functions will be described at the operating license stage.

The Primary and Backup ERFs receive data from the NIC to provide indication of plant status as shown in NAT-4950, “I&C Architecture and Design Basis Topical Report,” Revision 1 (Reference 7.6.7-1). NAT-4950, Revision 1, is incorporated by reference into the SAR. See Section 1.4.2 for a list of documents incorporated by reference into the SAR.

**Reference**

7.6.7-1 TerraPower, “I&C Architecture and Design Basis Topical Report,” NAT-4950 Revision 1, ML24068A187.


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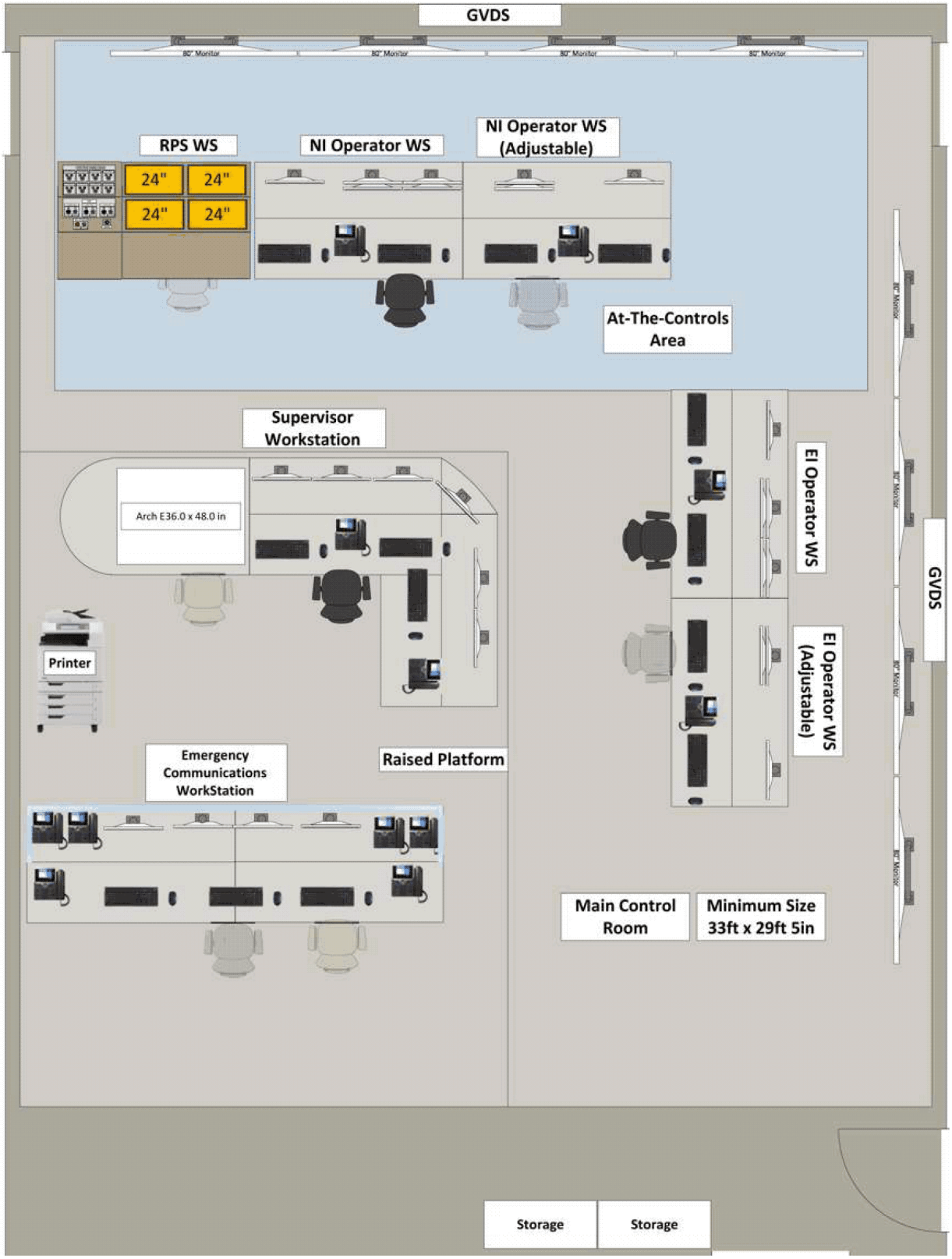
**Figure 7.6.7-1 MCR Layout**





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## 7.6.8 Anticipatory Automatic Seismic Trip System

### 7.6.8.1 System Description, Architecture, and Equipment Locations

The Anticipatory Automatic Seismic Trip System (AST) measures ground motion, applies required signal processing and logic, and automatically generates output actuation signals to the Reactor Trip Breakers (RTBs) through the interface device provided by the Reactor Protection System (RPS) when required by seismic conditions. The RTB interface device is located downstream of the RPS logic process, where an analog relay matrix is deployed as an isolation device to control the RTB power supply. The AST is used during seismic beyond design basis events greater than the safe shutdown earthquake to scram the reactor before the peak seismic wave reaches the facility. Seismic design is described in Section 6.4.1. The RPS is described in Section 7.6.3.

#### 7.6.8.1.1 System Architecture

The AST consists of four channels, each channel includes triaxial seismic accelerometers, a signal processing unit, and a bi-stable unit. The triaxial accelerometers are placed in four different quadrants equidistantly spaced around the Reactor Building (RXB). The RXB is described in Section 7.8.1.

Each signal processing unit receives input from the triaxial accelerometers and produces an output to the bistable unit. If the signal processing unit output exceeds the bi-stable unit setpoint, then the bi-stable unit ouputs an isolated signal to the shunt trip feature in the RTB. The RTB arrangement constitutes 2-out-of-4 logic for the reactor scram function. The RTB arrangement prevents spurious actuation caused by a failure of a single AST channel since two RTBs allocated to different AST channels are needed to cause a reactor trip.

The AST signals and interfaces are shown in Figure 7.6.8-1.

#### 7.6.8.1.2 Equipment Locations

The triaxial seismic accelerometers are located in boreholes at four locations at 90-degree angles around the RXB. The signal processing takes place in cabinets in the Nuclear Island Control Building substructure. The specific locations of cabinets and other AST structures, systems, or components (SSCs) will be described at the operating license (OL) stage.

### 7.6.8.2 System Design Bases and Associated Safety-Significant Functions

The AST is designed to perform a non-safety-related with special treatment (NSRST) function. A description of the process for assigning NSRST functions to the AST is provided in Section 7.6.1.1.1. A description of the process for describing the safety classification of the AST is provided in Section 7.6.1.2. The NSRST function assigned to the AST is described in Section 7.6.8.2.1. The application of PDC to the AST is described in Section 7.6.8.2.2.

Special treatments are applied to safety-significant AST SSCs to meet quality, design, reliability, and performance requirements. Special treatments are initially selected based on SSC safety classification as described in Section 6.3. Special treatments specific to AST fundamental design


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principles are described in Section 7.6.8.3. The following programmatic special treatments are identified as applicable to the AST and the associated summary program descriptions are provided:

- Design Reliability Assurance Program (Table 8.0-1)
- Equipment Qualification Program (Table 8.0-1)
- Post-construction inspection, testing, and analysis program (Chapter 12)
- Quality Assurance Program Description (Section 8.1)

The AST conforms to regulatory guides (RGs) as follows:

RG 1.53, Revision 2, *Application of the Single-Failure Criterion to Safety Systems*

The AST partially conform to RG 1.53, Revision 2. Conformance is partial because IEEE 379-2014, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” (Reference 7.6.8-1) is used rather than the version of the standard endorsed by this RG.

RG 1.100, Revision 4, *Seismic Qualification and Mechanical Equipment for Nuclear Power Plants*

The AST partially conforms to RG 1.100, Revision 4. Conformance is partial because IEEE/IEC 60980-344-2020, “Nuclear Facilities- Equipment Important to Safety-Seismic Qualification,” (Reference 7.6.8-2) is used rather than the version of the standard endorsed by this RG.

RG 1.118, Revision 3, *Periodic Testing of Electric Power and Protection Systems*

The AST partially conforms to RG 1.118, Revision 3. Conformance is partial because IEEE 338-2022, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” (Reference 7.6.8-3) is used rather than the version of the standard endorsed by this RG.

RG 1.180, Revision 2, *Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems*

The AST fully conforms to RG 1.180, Revision 2.

7.6.8.2.1 Defense Line Functions

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as safety-related (SR) functions or NSRST functions as defined in Section 5.2.


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| Function ID | Function Description | AST Functional Support |
|---|---|---|
| DL4-RC4 | Automatic seismic trip | The AST continuously monitors signal inputs from triaxial accelerometers and sends an isolated signal to the shunt trip feature in the RTBs when the signal inputs exceed a bi-stable setpoint. |

7.6.8.2.2 Principal Design Criteria

PDC are described in Section 5.3. The AST conforms to PDC as described below.

Consistent with PDC 2, *Design based for protection against natural phenomena*, the AST is seismically qualified for postulated earthquake. The AST SSCs are protected from external flooding, high winds, and extreme climate. Below grade SSCs are provided with subgrade waterproofing and evaluated for hydrologic loads if applicable. Additional details of the implementation of this PDC will be provided with equipment locations at the OL stage.

Consistent with PDC 3, *Fire protection*, the AST is protected from fire hazards based upon the location of its SSCs. Additional details of the implementation of this PDC will be provided with equipment locations at the OL stage.

Consistent with PDC 4, *Environmental and dynamic effects design basis*, the AST processing cabinets are located in and protected from external hazards by the Nuclear Island Control Building. The NCB is described in Section 7.8.4.1.

Consistent with PDC 13, *Instrumentation and control*, the AST monitors and measures ground motion, applies required processing and logic to calculate the seismic threat level, and automatically generates signals to open the RTBs. The AST monitors the anticipated range of seismic activities to ensure a reactor scram is initiated with adequate margin during a seismic event.

Consistent with PDC 24, *Separation of protection and control systems*, interconnection of the RPS and the AST is limited to the interface with the RTBs. The RPS is separated and isolated from the AST to the extent that any single failure of an AST component or channel, or failure or removal from service of a single protection system component or channel which is common to the AST and RPS, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the RPS.

7.6.8.3 AST Fundamental Design Principles

The AST design includes elements of quality that minimize the potential for failures to occur in SSCs associated with NSRST functions. These properties of qualification, reliability, robustness, security, diversity, and defense-in-depth features are discussed in Section 7.6.8.3.1 through Section 7.6.8.3.4. The following industry codes and standards are applied as special treatments to the overall design of safety-significant AST SSCs:

- IEEE 379-2014,


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- IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits” (Reference 7.6.8-4)

Seismic classifications for AST components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatments are applied based on the seismic classification. The seismic classification of AST components is SCN1. Seismic design, analysis, and qualification is performed as described in Section 6.4.

7.6.8.3.1 Qualified Instrumentation and Control

The AST is qualified per the Equipment Qualification Program described in Table 8.0-1.

7.6.8.3.2 Reliable Instrumentation and Control

The method for confirming specific reliability and capability targets for SSCs at the OL stage is described in Section 6.2.

Identification

IEEE 384-2018 is applied as a special treatment to the AST to ensure equipment is distinctly identified for each redundant portion of the system.

Human Factors

The AST equipment is designed and reviewed per the Human Factors Engineering Program Plan described in Section 11.2.

Reliability

Reliability of the AST is addressed through the Design Reliability Assurance Program described in Table 8.0-1.

Capability for Testing and Calibration

The AST conforms to IEEE 338-2022 for testing and calibration capabilities.

7.6.8.3.3 Robust Instrumentation and Control

Robustness of the AST design is the degree to which it can perform its NSRST function in the presence of invalid inputs or stressful environmental conditions. The following elements are considered for system robustness:

Single Failure

Single failure is addressed by applying IEEE 379-2014 as a special treatment to the design of the AST.


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Independence

IEEE 384-2018 is applied as a special treatment to the AST in order to meet independence requirements.

7.6.8.3.4 Secure Instrumentation and Control

Security for I&C systems is addressed through the physical security and cyber security programs that are described in Section 11.6.

**References**

7.6.8-1 IEEE, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” IEEE 379-2014, May 16, 2014.

7.6.8-2 IEEE/IEC, “International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification,” IEEE/IEC 60980-344-2020.

7.6.8-3 IEEE, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” IEEE 338-2022, December 3, 2022.

7.6.8-4 IEEE, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE 384-2018, November 1, 2018.



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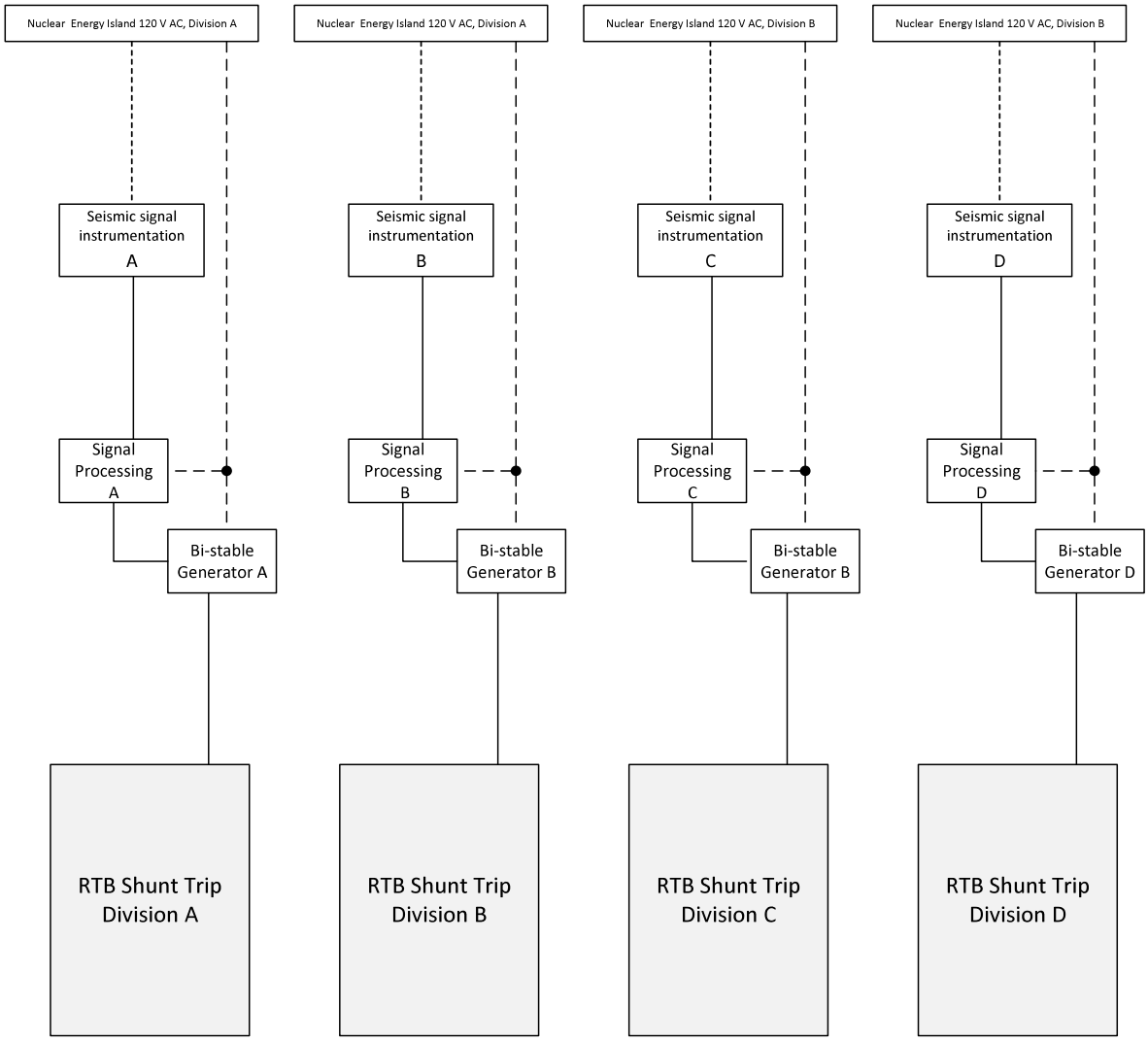
## Figure 7.6.8-1 AST Signals and Interface



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# 7.7 Electrical Systems

Electrical systems include the Nuclear Island (NI) Auxiliary Electrical Systems, NI Ancillary Electrical System, and Energy Island Auxiliary Electrical System. Section 1.1.4 contains overviews of these systems and associated subsystems. These systems are classified as non-safety-related with no special treatment (NST), except for portions of the NI Auxiliary Electrical Systems, which are classified as non-safety-related with special treatment (NSRST), and are described in this section.

## 7.7.1 Nuclear Island Auxiliary Electrical Systems

### 7.7.1.1 Summary Description

The NI auxiliary electrical systems comprise several systems, including the Nuclear Island Uninterruptible AC Power Supply System (NUP) and Nuclear Island DC Power Supply System (NDC), to distribute auxiliary electrical power to the NI. The NDC and NUP provide power to NI systems and components during all modes of plant operation. Failures of the NUP or NDC do not affect the ability of the plant to perform safety-related (SR) functions. Certain portions of the NUP and NDC (comprising Divisions A, B, C, and D) are classified as NSRST, as these systems provide power to support NSRST SSCs functional requirements. This section describes the NSRST portions of the NUP and NDC. The NDC and NUP each have an NST portion that provides power to NST breaker control and NST control systems.

The NUP provides uninterruptible AC power to NI loads, including the following**:**

- Reactor Protection System and Post-Accident Monitoring cabinets (see Section 7.6.3)
- Nuclear Instrumentation System (see Section 7.6.4)
- Radiation Monitoring System (see Section 7.6.6)
- Control rod drive motor controllers (for scram follow motor drive-in function) (see Section 7.2.5)
- Reactor Air Cooling System instrumentation (see Section 7.2.1)
- Main Control Room emergency operating lighting (see Section 1.1.4.3)
- Reactor Protection System equipment rooms (including Remote Shutdown Complex) emergency operating lighting (see Section 1.1.4.3)
- Battery monitoring system (see Section 7.7.1.3.2)

The NDC provides DC power to NI loads, including the following**:**

- NUP (see Section 7.7.1.3.1)
- Primary Sodium Pump isolation breaker backup trip power (see Section 7.1.3)
- Intermediate Sodium Pump isolation breaker backup trip power (see Section 7.1.4)
- Sodium Processing System pump isolation breaker backup trip power (see Section 7.2.4)

A simplified single line diagram of Division A and Division B of the NUP and NDC is depicted in Figure 7.7.1-1. A simplified single line diagram of Division C and Division D of the NUP and NDC is depicted in Figure 7.7.1-2.


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### 7.7.1.2 Design Basis

Compliance with regulatory criteria, identification of design criteria, and codes and standards that define the NDC and NUP safety design basis are described below.

Regulatory Criteria

PDC are identified in Section 5.3. Compliance with PDC that are applicable to the NDC and NUP is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the NDC and NUP are designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions. Safety-significant components are located within, and protected by, the reinforced concrete Nuclear Island Control Building (NCB) substructure. The location within the NCB provides protection from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions as described in Section 7.8.4. The safety-significant portions of the NDC and the NUP are separated by division, and are designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1. The NDC and the NUP components are designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions.

Consistent with PDC 3, *Fire protection*, NDC and NUP NSRST components are designed and located to minimize the probability and effect of fires through the selection of non-combustible and non-fire sustaining materials, to the extent practical. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, NDC and NUP NSRST components are designed to be compatible with the environmental conditions associated with normal operations, maintenance, testing, anticipated operational occurrences, and postulated accidents. These electrical systems and components are located in the NCB substructure and separated by division. The electrical systems and components are protected from dynamic effects, including the effects from missiles, pipe whipping, and discharge fluids that may result from equipment failures and from external hazards.

Consistent with PDC 17, Electric power systems, electric power is not required in order to meet SARRDLs for anticipated operational occurrences or postulated accidents. The SR scram and ESF functions described in Section 7.6 are accomplished without the need for electric power in response to DBAs. Sufficient power is provided for safety-significant functions by the design of the NSRST NUP and NDC, however, failures of the electric power system do not adversely affect the performance of SR functions. SSCs that require electrical power to perform NSRST functions are supplied through the NSRST electrical systems.

Consistent with PDC 18, *Inspection and testing of electric power systems*, the NSRST electrical systems are designed with capability to test periodically (1) the operability and functional performance of the components of the systems and (2) the operability of the systems as a whole


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and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among systems.

Functional Requirements and Design Criteria

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The NUP and NDC support the following safety-significant functions, which are designated as NSRST to indicate the safety classification of components relied upon to perform the functions:

| Function ID | Function Description | NDC and NUP Functional Support |
|---|---|---|
| DL4-RC3 | Control Rod Drive Driveline Scram Follow | The NDC and NUP provide electric power to support performance of these functions. |
| DL4-RC4 | Automatic Seismic Trip | |
| DL2-HR2 | Intermediate Sodium Pump Trip on Low Intermediate Heat Transport System Level | |
| DL4-HR2 | Primary Sodium Pump Trip Automatic Backup | |
| DL4-HR3 | Intermediate Sodium Pump Trip Automatic Backup | |
| DL4-HR6 | Manual Primary Sodium Pump Trip | |
| DL4-HR7 | Manual Intermediate Sodium Pump Trip | |
| DL4-RR3b | Sodium Processing System Pumps Trip on Leak Detection | |
| DL4-RR4c | Vapor Trap Cell Isolation on Overpressure | |
| DL4-RR7 | Fuel Handling Building Barrier | |
| DL4-RR8 | Manual Sodium Processing System Pump Trip on Leak Indication | |
| DL5-PAM1 | Post-Accident Monitoring | |

Seismic Classification and Design Codes

Seismic classifications for NDC and NUP components are determined in accordance with the methods described in Section 6.4.1. Seismic special treatment is applied based on the seismic classification. The NSRST components of the NDC and NUP are classified SCN1. Seismic design, analysis, and qualification is performed as described in Section 6.4. NSRST components of the NDC and NUP are qualified to withstand seismic loads associated with the safe shutdown earthquake (described in Section 2.6.2) without loss of the capability to perform safety-significant functions.

The NUP and NDC equipment is designed, tested, and qualified to commercial quality standards. Industry electrical equipment consensus standards are applied as special treatments to meet reliability and performance capability requirements as described in Section 7.7.1.4. Section 7.7.1.3 contains additional descriptions of conformance with industry standards.


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### 7.7.1.3 System Design and Performance Evaluation

#### 7.7.1.3.1 NUP

For system reliability and to ensure continuous and stable uninterrupted electrical power is provided to NI systems and components, the NUP comprises four independent divisions (A, B, C, and D), one for each Reactor Protection System division, consistent with IEEE 603-1991 (Reference 7.7.1-3) and IEEE 379-2000 (Reference 7.7.1-7).

Four independent divisions of 120 volts alternating current (VAC) single-phase power of the NUP provide regulated power to plant loads during normal operations and during a loss of AC power. Each division includes a 120 VAC single-phase inverter with an integrated static transfer switch (STS), a 480/120 VAC single-phase bypass Regulating Voltage Transformer (RVT), two 120 VAC panelboards (one uninterruptible power supply (UPS)-backed and one non-UPS-backed), and a Remote Manual Bypass Switch (RMBS) as outlined in Figures 7.7.1-1 and 7.7.1-2. The non-UPS-backed panelboard feeds the same loads as the UPS-backed panelboard as a backup.

During normal plant operations, each inverter receives 250 VDC from the respective division 250 VDC switchboard. The inverters convert 250 VDC to regulated single-phase 120 VAC to energize a 120 VAC UPS panelboard. The 120 VAC UPS panelboard provides power to the NSRST UPS loads, as discussed in Section 7.7.1.1. Each NUP division is single-point solidly grounded at the inverter.

Each division RVT receives 480 VAC power from the same NST Standby Diesel Generator (SDG)-backed Motor Control Center (MCC) as the normal battery charger of that division. The RVT transforms 480 VAC into regulated single-phase 120 VAC to energize a second 120 VAC (non-UPS) panelboard to provide redundant power to the UPS loads. For these loads, one power feed is from an inverter via the 120 VAC UPS panelboard, and the other is supplied from an RVT via the 120 VAC non-UPS panelboard.

Each RVT secondary winding also connects to an STS and RMBS to provide an inverter bypass power source when needed. In the event the inverter fails, the STS automatically transfers the 120 VAC UPS panelboard from the inverter to the RVT. When the inverter needs to be taken out of service for maintenance, repair, or testing, the 120 VAC UPS panelboard can be fed via the RVT by manually closing the RMBS.

During a loss of offsite power event, the NST Standby Diesel Generators (SDGs) (part of the Energy Island) are automatically started to power their associated MCCs to maintain AC input power to the battery chargers and RVTs to keep both the 120 VAC UPS panelboard and 120 VAC panelboard energized. The batteries supply uninterrupted power to the inverters and UPS panelboard loads until the SDGs come online.

In the event of a loss of all AC power, the inverters remain powered by the batteries for 72 hours (Divisions A and B) and 8 hours (Divisions C and D) to keep the 120 VAC UPS panelboard and NSRST loads energized. The RVTs and 120 VAC panelboards are lost until AC power is restored at the MCC. There are provisions for connection of a portable diesel generator for providing power to Post-Accident Monitoring loads beyond the 72-hour coping period of Division A and B batteries.


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In the event an inverter fails, or there is an overload or short-circuit on the inverter output, the inverter's STS initiates an automatic transfer of the AC UPS loads to the RVT. In this case, the RVT powers both the 120 VAC UPS panelboard and the 120 VAC panelboard.

When an inverter needs to be taken out of service for maintenance, repair, or testing, the loads can be transferred to the RVT using the RMBS, located outside the inverter enclosure, to isolate the inverter. In this case, the RVT will power both the 120 VAC UPS panelboard and the 120 VAC panelboard.

Independence of the redundant NSRST divisions is maintained by physical separation and electrical isolation between divisions, and between NST and NSRST SSCs, consistent with IEEE 384-1992 (Reference 7.7.1-4), ensuring a postulated single failure affects only a single division, consistent with IEEE 379-2000.

#### 7.7.1.3.2 NDC

For system reliability and to ensure continuous and stable uninterrupted electrical power is provided to NI systems and components, the NDC comprises four independent 250 VDC divisions - one for each Reactor Protection System division, consistent with IEEE 603-1991 and IEEE 379-2000.

Each division has one battery bank, a Battery Transfer Switch (BTS), one normal switchboard, and access to two battery chargers – one powered by an SDG-backed MCC and a swing charger which can be powered by an MCC on either SDG train via a manual transfer switch. The chargers maintain charge on the respective division batteries and supply NI UPS loads through 120 VAC inverters. NDC provides power to the NSRST DC loads, as discussed in Section 7.7.1.1.

Each battery charger is designed to supply its connected loads while simultaneously recharging its associated battery from the design minimum charge state to full charge within 24 hours.

The battery chargers are designed to supply the loads with the battery disconnected from the system for maintenance, testing, or equalization.

The battery chargers are equipped with features that prevent undercharging or overcharging (e.g., ambient temperature changes).

During a loss of offsite power event, NST SDGs (part of the Energy Island) are automatically started to energize their associated MCCs to maintain AC input power to the battery chargers.

Divisions A and C chargers are supplied power from separate 480 VAC SDG-A backed NST MCCs (in the Nuclear Island AC Electrical Power Low Voltage System (NLV)) while Divisions B and D are supplied power from separate 480 VAC SDG-B backed NST MCCs (in NLV). The loss of one NLV MCC results in a loss of function of one charger until the swing charger is brought on line. The battery supplies uninterruptible power to the system loads until the swing charger is operating.


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The swing chargers are shared between Divisions A and B DC switchboards, and between Divisions C and D DC switchboards. If a normal battery charger fails or requires maintenance, the swing battery charger can be aligned to serve the full load of the out-of-service normal battery charger. Each swing charger can be connected to either an SDG-A backed MCC, or an SDG-B backed MCC via an NST manual transfer switch.

Divisions A and B batteries are sized to supply the respective division loads for 72 hours.

Divisions C and D batteries are sized to supply the respective division loads for 8 hours.

The batteries are sealed Valve-Regulated Lead-Acid type, selected and designed per IEEE 1187-2013 (Reference 7.7.1-1) and IEEE 1189-2007 (Reference 7.7.1-2). Batteries are sized per IEEE 485-2010 (Reference 7.7.1-5).

Each normal charger and battery pair has a manually operated BTS to facilitate battery equalization and discharge testing by isolating the normal switchboard and allowing direct connection between battery and charger. When battery equalization is required, the normally closed charger output disconnect switch to the 250 VDC switchboard is opened, a second charger output disconnect switch to the BTS is closed through a mechanical interlock between the two output switches, and the BTS is placed in the equalize/test position.

A battery monitoring system continuously monitors key battery parameters such as cell temperature, cell resistance, charging current, float voltage, and battery room temperature, per IEEE 1491-2012 (Reference 7.7.1-6), to prompt corrective action prior to failure. The battery monitoring system guards against potential overcharging and magnitude and frequency of discharge cycles that may degrade battery performance. The battery monitoring system provides reasonable assurance that the batteries are not exposed to prolonged periods of high temperature.

The NDC is ungrounded. Ground detection relays are provided in the DC switchboards to monitor the presence of a grounded conductor and send a common trouble alarm to the Utility Monitoring and Control System (UMC) to prompt operator response to clear the faulted condition. The ungrounded NDC will remain operational with a single ground fault until a second fault is received to generate sufficient fault current to trip a protection device and isolate the faulted circuit.

A voltage surge suppressor is provided at the respective 480 VAC MCCs to minimize the potential for component damage in NDC equipment resulting from electrical transients.

Independence of the redundant divisions is maintained by physical separation and electrical isolation between divisions, and between NST and NSRST SSCs, consistent with IEEE 384-1992, ensuring a postulated single failure affects only a single division, consistent with IEEE 379-2000.

The battery rooms are temperature-controlled to mitigate accelerated cell aging from high operating temperatures and allow operation of the batteries within the temperature correction factor used for sizing the batteries.


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#### 7.7.1.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Industry codes and standards will be applied as special treatments as needed to meet reliability targets and performance requirements. These special treatments will be finalized and provided at the operating license stage.

The following programmatic special treatments are applied to the NDC and NUP to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Design Reliability Assurance Program
- Equipment Qualification Program
- Post Construction Inspection, Testing, and Analysis Program (Chapter 12)
- Quality Assurance Program Description

#### 7.7.1.5 Instrumentation Requirements

The NDC and NUP include integrated voltmeters and ammeters to provide continuous monitoring locally in the equipment rooms and remotely via the UMC.

Electrical protection and metering are provided for NUP equipment.

Protection and alarms are provided to identify RVT and inverter faults. Protective devices have the ability to isolate the faults.

Operation of a protection or monitoring device sends a common trouble signal to the UMC.

Electrical protection and metering are provided for NDC equipment, including adequate protection against overload and short circuit conditions under normal operation and off-normal operation. The statuses of the normal battery chargers, swing battery chargers, battery transfer switches, and DC switchboards are monitored through the UMC.

Protection and alarms are provided to identify normal and swing battery charger faults. Protective devices have the ability to isolate the faults. Operation of a protection or monitoring device sends a common trouble signal to the UMC.

The battery chargers serve as isolation devices between the AC input power from the NST NI AC electrical power system and NDC, consistent with IEEE 384-1992.

The battery chargers are designed so that they do not become a load on the battery due to power feedback during loss of AC power to the chargers.

The battery charger output current is limited for overload protection.

Each battery charger has one AC input circuit breaker and two DC output disconnect switches (normal output and output to BTS) for charger protection and isolation. The two output disconnect switches are mechanically interlocked to prevent the switches from being closed concurrently.


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Overvoltage protection coordination is provided between the battery chargers and the NUP inverters; the DC overvoltage protection in the battery charger shuts down the charger automatically before the inverter overvoltage protection is activated.

The battery charger design prevents propagation of AC input power temporary overvoltage/undervoltage to DC battery circuits, DC-connected loads, and to the inverter. When input voltage and frequency to the battery chargers are outside the operating limits, resulting in unacceptable output voltages, the battery chargers shut down and provide electrical isolation to the NDC. The battery chargers return to operation automatically when the input voltage and frequency are restored within the normal operating limits.

Protection and alarms for the DC switchboards are provided to identify faults. Protective devices have the ability to isolate the faults. Operation of a protection or monitoring device sends a common trouble signal to the UMC.

**References**

7.7.1-1 Institute of Electrical and Electronics Engineers, “IEEE Recommended Practice for Installation Design and Installation of Valve-Regulated Lead-Acid Batteries for Stationary Applications,” IEEE Standard 1187-2013.

7.7.1-2 Institute of Electrical and Electronics Engineers, “IEEE Guide for Selection of Valve-Regulated Lead-Acid (VRLA) Batteries for Stationary Applications,” IEEE Standard 1189-2007.

7.7.1-3 Institute of Electrical and Electronics Engineers, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE Standard 603-1991.

7.7.1-4 Institute of Electrical and Electronics Engineers, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE Standard 384-1992.

7.7.1-5 Institute of Electrical and Electronics Engineers, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications,” IEEE Standard 485-2010.

7.7.1-6 Institute of Electrical and Electronics Engineers, “IEEE Guide for Selection and Use of Battery Monitoring Equipment in Stationary Applications,” IEEE Standard 1491-2012.

7.7.1-7 Institute of Electrical and Electronics Engineers, “IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems,” IEEE Standard 379-2000.


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## Figure 7.7.1-1 Simplified NSRST Electrical Systems Single Line Diagram - Divisions A and B

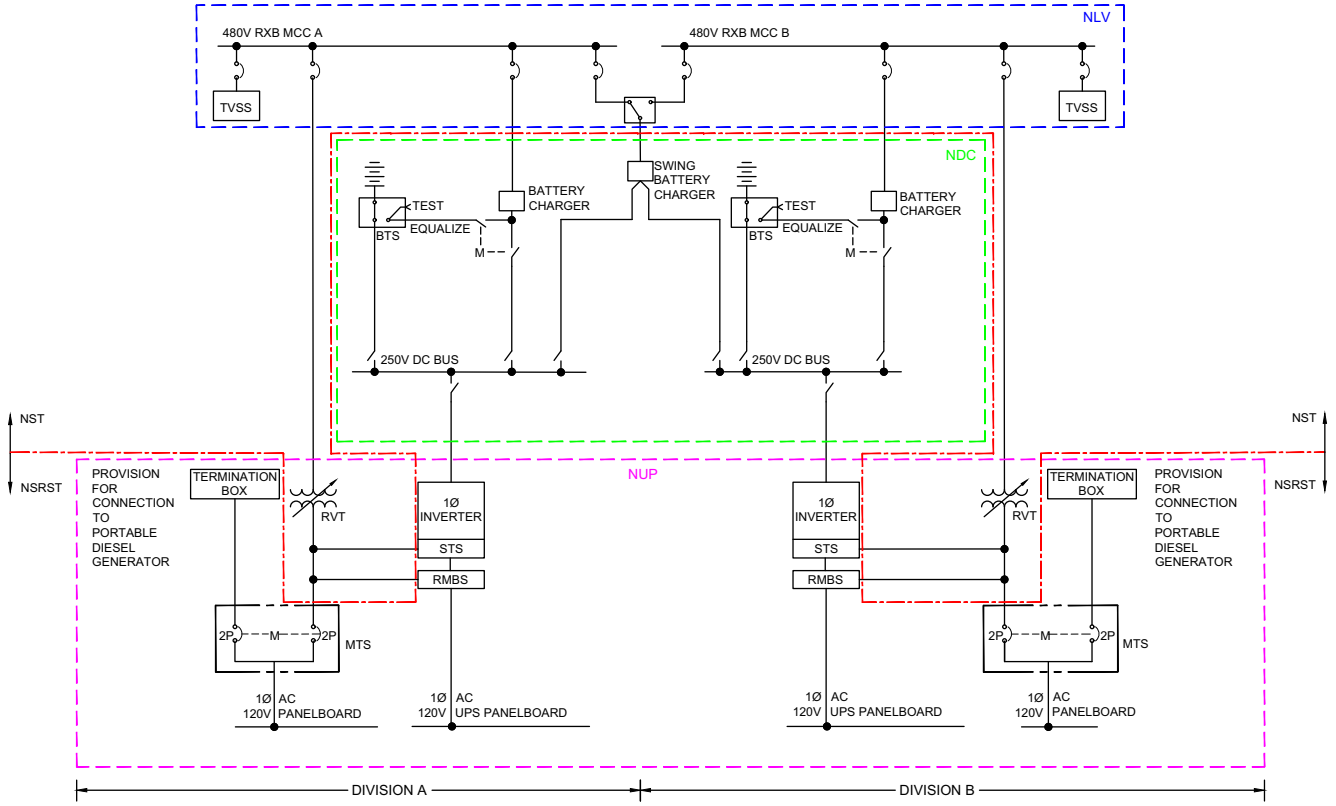



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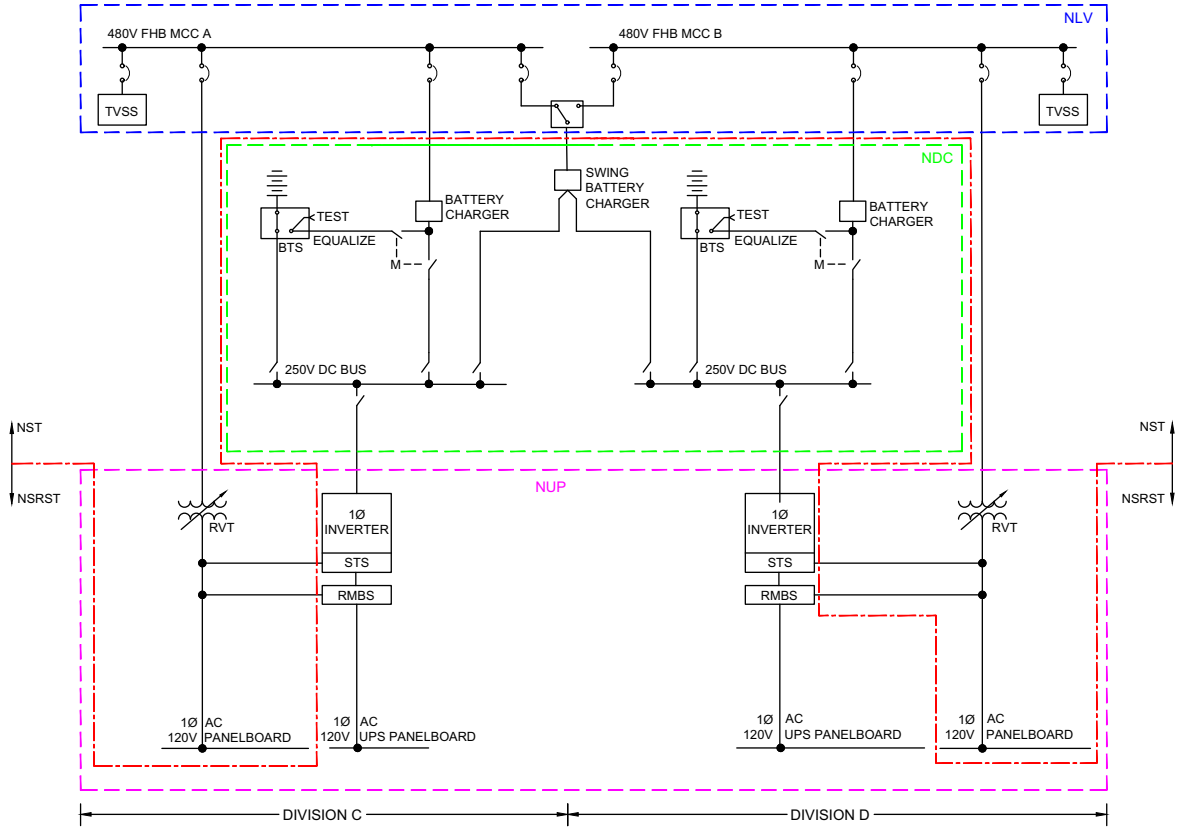
**Figure 7.7.1-2 Simplified NSRST Electrical Systems Single Line Diagram - Divisions C and D**





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## 7.8 Buildings and Structures

This section describes the safety-significant buildings and structures located on the Nuclear Island (NI), which includes the Reactor Building (RXB), Fuel Handling Building (FHB), Reactor Auxiliary Building (RAB), and Nuclear Island Control Building (NCB). Additional site buildings and structures are described in Section 1.1.4.4.

Refer to the site layout plan in Figure 1.2-1 for additional information pertaining to building location on the site.

### 7.8.1 Reactor Building

The Reactor Building (RXB) is composed of two main structural sections: a steel framed above-grade superstructure (described in Section 1.1.4.4.2) and a reinforced concrete and steel substructure below grade. The safety-related (SR) RXB substructure provides housing to support and protect SR systems and components, including the reactor, Reactor Enclosure System (RES), and the duct stacks for the Reactor Air Cooling System (RAC) that provide passive air cooling of the RES. The RAC is integrated into the RXB design and has above ground intake and discharge stacks outboard of the superstructure which exhaust above the RXB roof. The above grade RXB superstructure, which does not house any safety-significant equipment, is classified as non-safety-related with no special treatment (NST). The RXB superstructure is structurally designed to ensure there are no adverse impacts or interactions as described in PSAR Section 6.1.3, with the SR RXB substructure, SR RAC stacks, or any nearby SR or non-safety-related with special treatment (NSRST) NI structures. The RXB is normally unoccupied with access to substructure areas restricted during normal plant operation.

As shown on the site layout plan in Figure 1.2-1, the RXB is centrally located in the Nuclear Island (NI) and is surrounded by other structures providing safety-significant functions but is functionally separate from the other NI buildings. The Reactor Auxiliary Building (RAB), located to the west of the RXB, is functionally connected to the RXB through a connector which is structurally isolated at the RXB and RAB interface. To the east is the Fuel Handling Building (FHB) which is functionally connected to the RXB through the FHB connector and is seismically isolated at the RXB and FHB interface.

#### 7.8.1.1 Reactor Building Substructure

##### 7.8.1.1.1 Summary Description

The RXB substructure is decoupled from the RXB superstructure. Design information pertaining to safety-significant structures is presented in Section 6.4.2. Figures 1.1-5 and 1.1-6 provide general arrangement drawings of the RXB substructure.

The RXB substructure consists of a primary floor at grade level, a subgrade space that houses the seismic isolators that support the RES and Modular Isolated Reactor Support Structure (Section 7.1.2). A cylindrical cavity space provided below the rectangular Head Access Area (HAA) houses the RES and RAC components. The HAA structure allows access to the reactor


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head, forms part of the RXB substructure, and is classified as SR. The HAA contains the reactor head, associated penetrations, the Primary Sodium Pumps, and control rod drive assemblies. The RAC enclosure structures provide a foundation and load path for above grade RAC stacks.

The RXB substructure is approximately at grade level and below, and is designed to withstand the design basis external hazards identified in Section 6.1.1.

At an approximate elevation of 6,758 feet, a slab with an integral steel plate located on its underside covers the HAA and represents the top of the RXB substructure. Portions of the grade slab are removable to provide access to the HAA for maintenance and refueling activities. The removable concrete planks are supported by steel box girders. The removable planks at grade are approximately 48 inches thick whereas the rest of the grade slab is approximately 42 inches thick.

The base mat of the cylindrical cavity is at approximately elevation 6,639 feet; the bottom of the HAA is at approximately elevation 6,702 feet. The HAA slab is the primary RXB substructure load bearing foundation. It directly supports the embedded HAA shear walls which support the grade slab. Located on the north and south sides of the reactor are reinforced concrete enclosures for the RAC air ducts with base slab at the same elevation as the base of the HAA. The HAA and RAC Enclosures are accessible for maintenance via grade-level removable access plugs and personnel access ladders.

The NSRST RAB (Section 7.8.3) substructure is located approximately 16 feet, 6 inches west of the RXB substructure and is not considered in the RXB substructure design.

There is an embedded non-integral corridor connecting the RXB and RAB which houses the Intermediate Heat Transport System (Section 7.1.4). The Intermediate Heat Transport System loop piping and components are primarily located within the RAB (Section 7.8.3), with the exceptions of piping in the RXB at the connections to the Intermediate Heat Exchanger nozzles (see Figure 1.1-6).

The SR FHB (Section 7.8.2) substructure is located approximately 40 feet east of the RXB substructure and does not affect the RXB substructure design.

#### 7.8.1.1.2 Design Bases

Regulatory Criteria

Principal design criteria are identified in Section 5.3. Compliance with Principal Design Criteria (PDC) that are applicable to the RXB substructure are described below:

Consistent with PDC 2, *Design Bases for Protection Against Natural Phenomena,* the RXB substructure is designed in accordance with codes and standards identified in Section 6.4.2 to withstand the effects of design-basis hazard levels (Section 6.1.1) without the loss of structural integrity. The design bases for the RXB substructure considers the most severe natural phenomena and loading combinations outlined in Section 6.4.2 to provide assurance that safety-significant systems and components housed within the substructure are appropriately


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protected and not adversely impacted by natural phenomena. The RXB substructure is designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1.

Consistent with PDC 3, *Fire Protection,* the RXB substructure is designed with plant features that minimize the probability and effect of fires and explosions by the use of low combustible materials and physical separation. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and Dynamic Effects Design Bases*, The RXB substructure is designed in accordance with codes and standards and specific load combinations identified in Section 6.4.2 to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents without the loss of structural integrity.

Consistent with PDC 16, *Containment design*, the HAA, which is part of the RXB substructure, provides a functional containment boundary that performs a radionuclide retention function.

Consistent with PDC 35, *Emergency core cooling,* The RXB substructure houses, protects, and supports the below grade portion of the RAC (described in Section 7.2.1) within the RAC enclosures and the reactor cavity. The RXB substructure reactor cavity walls form a portion of the RAC inlet air downcomer. The walls of the RXB substructure below the operating deck are part of the air plenum for the RAC.

Consistent with PDC 36, *Inspection of emergency core cooling system,* The RXB substructure is designed to allow for access for the performance of monitoring and non-destructive examination activities for the RAC as a part of the RIM Program and to permit inspections of portions of the RXB that provide support for the RAC and form part of the RAC air flow path.

Consistent with PDC 37, *Testing of emergency core cooling system,* the RXB provides access to permit periodic functional testing of the RAC and confirm structural integrity of the portions of the RXB that support the RAC.

Consistent with PDC 81, *Reactor building design basis,* the RXB substructure is designed to structurally support and protect the RES and RAC such that the geometry for passive removal of residual heat from the reactor core is maintained during DBAs.

Consistent with PDC 82, *Provisions for periodic reactor building inspection,* the RXB substructure is designed to permit periodic structural inspections of important areas. The HAA and RAC Enclosures are accessible for inspection and surveillance via removable access plugs in the ground floor. The cylindrical reactor cavity is accessible by remotely operated inspection equipment.


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Functional Requirements and Design Criteria

The probabilistic risk assessment safety functions (PSF) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The RXB substructure supports the following safety-significant functions, which are designated as SR or NSRST to indicate the safety classification of RXB substructure components relied upon to perform the function:

| Function ID | Function Description | Functional Support |
|---|---|---|
| DL3-HR4 | Inherent RAC Operation (SR) | The RXB substructure maintains the outer boundary of the RAC air flow path. The HAA walls and slab of the RXB substructure below the operating deck and above the slab supporting the RES/MIRSS are part of the RAC air plenum. |
| DL4-RR1 | Ex-RES Functional Containment Barrier (NSRST) | The ex-RES functional containment barrier is the functional containment barrier that encompasses the primary system boundary, including the RES barrier. The ex-RES functional containment boundary includes the guard vessel, the HAA walls, and the HAA HVAC isolation. |
| DL4-RR1a | HAA Barrier (NSRST) | This functional requirement is a sub-function of the ex-RES functional containment barrier (DL4-RR1) that includes the HAA barrier. |

Load Path for SR SSCs (SR)

The RES is supported by the reactor support structure. The RSS transfers loads from the RES to the RXB substructure. Refer to Section 7.1.2 for a description of the RES load path. The seismic isolators are outboard of the cylindrical cavity such that the cavity is not part of the primary RES load path.

The RAC inlet and outlet stacks are freestanding above grade, straight, and single flue. The stacks are supported by the RXB substructure. Portions of the RXB, including the cavity and the lower portion of the HAA (below the operating level approximately aligned with the reactor head) provide a flow path for RAC operation. Additional details on the RAC are provided in Section 7.2.1.

Load Path for NSRST SSCs, (NSRST)

The SR RXB substructure performs the structural support load path for NSRST SSCs. Where required, the RXB substructure walls and slabs provide load path to support SSCs in the performance of their respective NSRST PSFs while inside the RXB substructure. Structural analysis and design special treatments related to the NSRST load path PSF of the RXB substructure are bounded by the special treatments associated with the SR load path PSF.


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Temporary Load Path for SR SSCs, (NSRST)

The SR RXB substructure performs the structural support PSF of temporary load path for SR SSCs. The RXB substructure grade slab, including the removable planks are in direct support of FHE SSCs and provide temporary load path to support the SSC's in the performance of their respective SR PSFs while in the RXB. Structural analysis and design special treatments related to the NSRST temporary load path PSF of the RXB substructure are bounded by the special treatments associated with the SR load path PSF.

External Hazard Protection for SR SSC's, (SR) and for NSRST SSC's (NSRST)

The RXB Substructure is located below grade. Mitigation of external hazards includes design of the RXB substructure in accordance with codes and standards identified in Section 6.4.2 to withstand the effects of design-basis hazard levels (Section 6.1.1) without the loss of structural integrity. The design of the RXB substructure provides assurance that SR and NSRST SSCs located within the confines of the RXB substructure are protected and not impacted by external hazards.

Regulatory Guides

The NRC Regulatory Guides (RGs) applied to safety-significant structures are identified in Section 6.4.2.

Seismic Classification and Design Codes

The SR RXB substructure seismic classification is SCS1. The RXB substructure is designed based on the design codes and standards described in Section 6.4.1 and Section 6.4.2.

Loads and Load Combinations

The RXB substructure and foundation are designed for loads for construction, normal plant start-up, operation, shutdown, and design basis accidents. The load combinations for the RXB substructure are provided in Section 6.4.2.

The load combinations for the concrete structural elements in the RXB substructure are based on ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary” (Reference 7.8.1-1), as shown in Section 6.4.2. The load combinations for steel structural elements in the RXB substructure are based on ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities” (Reference 7.8.1-2), as shown in Section 6.4.2.

7.8.1.1.3 Structure Design and Performance Evaluation

SR Structural Characteristics that Support SR Functions

The SR RXB substructure and associated foundation are designed in accordance with the codes and standards as identified in Section 6.4.2 to ensure that structural integrity is maintained in response to applicable design-basis hazard levels. Design-basis hazard levels are discussed in


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Section 6.1.1. The substructure design features ensure that systems and components housed within the substructure that perform SR functions are not adversely impacted by design basis hazards.

Materials and Construction Techniques

The materials, quality control, and construction techniques used for the RXB substructure are described in Section 6.4.2.

Design and Analysis Procedures

The RXB substructure design and analysis is performed for both non-seismic and seismic loads using 3-dimensional finite element analysis. Non-seismic loads are analyzed using commercial structural analysis software and seismic loads are analyzed with SASSI to consider Soil Structure Interaction (SSI) effects.

Section cuts in the finite element models are used to evaluate wall axial, shear, and flexure demands. Corresponding wall capacities are developed according to ACI 349-13 provisions. Demand to capacity ratios that include load factors in accordance with the information presented in Section 6.4.2 are then computed. Demand-to-capacity ratios do not exceed 0.80 for design-basis loads, or 1.00 for beyond design basis events.

During detailed design, seismic interactions of nearby structures upon the RXB substructure through Seismic Soil Structure Interaction effects are addressed based upon methodologies described in Section 6.4.1. Seismic soil structure interaction is not considered in preliminary design.

The soil properties and input motions used in the soil structure interaction analysis are based on preliminary probabilistic seismic hazards analysis results and will be updated at the operating license stage to reflect the final probabilistic seismic hazards analysis results. Three representations of soil profiles are analyzed: best estimate, lower bound, and upper bound, and each of these considers three spectrally-matched ground motion time-histories as inputs. The multi-case deterministic response analysis results in the generation of conservative response quantities, appropriate for preliminary design. The response analysis methodologies described in Section 6.4.1 are used during detailed design. In addition to the building forces used for structural design checks, in-structure response spectra, nodal displacements, and accelerations are generated for use in component design.

The preliminary substructure soil structure interaction model consists of shell elements for walls and slabs, beam elements for prominent steel portions of the structure and RES and RAC, spring elements for the seismic isolators that support the RES, and soil elements for the excavated soil volume. A typical shell element size is approximately 4 feet. Mass includes self-weight of structural elements, including steel liners, stay in place form work and consideration of permanent equipment loads and the appropriate part of the live load. Damping ratios of 7 percent (of critical damping) and 4 percent are applied to structural concrete and steel elements, respectively.


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Structural Acceptance Criteria

The acceptance criteria for reinforced concrete structures and components are in accordance with the limits defined in ACI 349/349R as modified by RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” Revision 3, for the load combinations, as applicable, as presented in Section 6.4.2 for SR reinforced concrete structures. The acceptance criteria for steel structures and components are in accordance with the limits defined in AISC N690 as modified by RG 1.142 for the load combinations, as applicable, as presented in Section 6.4.2 for SR steel structures. For the RXB substructure the load combinations put forth in Section 6.4.2 for the stability analysis of SR foundations takes into account the calculation of factors of safety.

#### 7.8.1.1.4 Special Treatments

Special treatments are initially selected based on SSC safety classification as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the RXB substructure are identified within the context of this section and Section 6.4.2. The following programmatic special treatments are preliminarily identified as applicable to the RXB substructure to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing and Analysis Program (see Chapter 12)

**References**

7.8.1-1 American Concrete Institute, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” ACI 349-13.

7.8.1-2 American Institute of Steel Construction, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” ANSI/AISC N690-18.


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### 7.8.2 Fuel Handling Building

The Fuel Handling Building (FHB) houses equipment related to fuel receipt, refueling, fuel storage, radioactive waste systems, and the Spent Fuel Pool (SFP). The rectangular-shaped FHB structure is approximately 103 feet wide by 293 feet long and is comprised of a reinforced concrete substructure and a steel-framed and reinforced-concrete superstructure with metal siding and a metal roof. The below-grade portion of the FHB, classified as safety-related (SR), provides structural support and shielding to SR and non-safety-related with special treatment (NSRST) systems housed within separated reinforced concrete enclosures.

The above-grade portion of the FHB, classified as NSRST, consists of a single-story main building, an adjoining north wing, and a single-story steel framed transfer corridor to the west that connects the Reactor Building to the FHB.

As shown on the site layout plan in Figure 1.2-1, the FHB is located approximately 28 feet to the east of the Reactor Building and approximately 40 feet to the north of the Fuel Auxiliary Building. Distances between adjacent buildings meet requirements in the International Building Code (IBC) 2021 (Reference 7.8.2-1) and address seismic interaction based on methodologies described in Section 6.4.1. Figure 1.1-7 through Figure 1.1-9 provide general arrangement drawings for the FHB.

#### 7.8.2.1 FHB Substructure

##### 7.8.2.1.1 Summary Description

The SR FHB substructure consists of several reinforced concrete slabs and walls that divide the substructure into several distinct areas each with varying elevations. The enclosed sub-grade areas are protected from design basis hazards by the three foot thick grade slab which also supports the Ex-Vessel Fuel Handling System (FHE) Ex-Vessel Handling Machine and Bottom Loading Transfer Cask at grade.

The sub-grade area of the FHB substructure that houses the safety-significant Water Pool Fuel Handling System (FHP), described in Section 7.3.1, includes the SFP subsystem. The SFP has 5 foot thick walls extending from the grade slab to a 4 foot thick foundation slab at an approximate depth of 51 feet from grade. The SFP, as discussed in Section 7.3.1, is equipped with a stainless steel liner and leak chase system utilized to monitor leakage from the liner. The liner and leak chase system are integrated into the FHB substructure design. Adjacent to the SFP is the Pool Auxiliary Vault which contains FHP system equipment and the SFP leak chase sump.

The FHB sub-grade area that supports the FHE Ex-Vessel Storage Tank, Gaseous Radwaste Processing System tanks, and associated equipment and components described in Sections 7.3.2 and 7.4.1 respectively, has 3 foot thick walls that extend approximately 30 feet to a 3 foot thick foundation slab. The FHE Pin Removal Cell is located in a separate FHB sub-grade area with 3 foot thick walls extending to a depth of approximately 45 feet to a 3 foot thick foundation slab.


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7.8.2.1.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the FHB substructure safety design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the FHB substructure is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena,* the FHB substructure is designed in accordance with codes and standards identified in Section 6.4.2 to withstand the effects of design-basis hazard levels (see Section 6.1.1) without the loss of structural integrity. The design bases for the FHB substructure considers the most severe natural phenomena and loading combinations outlined in Section 6.4.2 to provide assurance that safety-significant systems and components housed within the substructure are appropriately protected and not adversely impacted by natural phenomena.

Consistent with PDC 3, *Fire protection,* the FHB substructure is designed with plant features that minimize the probability and effects of fires and explosions, including the use of noncombustible or fire-resistive construction materials. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases,* the FHB substructure is designed in accordance with codes and standards and specific load combinations identified in Section 6.4.2 to accommodate the dynamic effects associated with internal and external missiles (see Section 6.1.1), and to withstand the effects of environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents without the loss of structural integrity.

Functional Requirements

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The FHB substructure supports the following safety-significant functions, which are designated as SR or NSRST:

External Hazard Protection for SR SSCs (SR)

The FHB substructure is designed to protect SR SSCs from the effects of design-basis hazard levels (Section 6.1.1) without the loss of structural integrity.

Load Path for SR SSCs (SR)

The FHB substructure provides the load path for structural support of SR SSCs in the performance of their respective SR PSFs.


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External Hazard Protection for NSRST SSCs (NSRST)

The FHB substructure is designed to protect NSRST SSCs from the effects of design basis hazard levels (Section 6.1.1) without the loss of structural integrity.

Load Path for NSRST SSCs (NSRST)

The FHB substructure provides the load path for structural support of NSRST SSCs in the performance of their respective NSRST PSFs.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the FHB substructure is provided in Section 6.4.2, along with a description of the use and conformance to the guidance.

Seismic Classification and Design Codes

The SR FHB substructure seismic classification is SCS1. The reinforced concrete portions of the FHB substructure, including the SFP, are designed to ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary” (Reference 7.8.2-2) as described in Section 6.4.2.3. The sub-grade portions of the FHB that contain Gaseous Radwaste System components are designed in accordance with guidance provided in RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Revision 2.

In addition, the SFP is designed to conform with the structural-related criteria in RG 1.13, “Spent Fuel Storage Facility Design Basis,” Revision 2. Structural steel portions of FHB substructure associated with the SFP liner and a leak chase system are designed to AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities” (Reference 7.8.2-3).

Loads and Load Combinations

The FHB substructure is designed for applicable design loads for construction, normal plant start-up, operation, shutdown, and design-basis hazard levels. The load combinations for the FHB substructure are provided in Section 6.4.2.2.

7.8.2.1.3 Structure Design and Performance Evaluation

SR Structural Characteristics that Support RSFs

The SR FHB substructure is designed in accordance with the codes and standards as identified in Table 6.4-2 such that structural integrity is maintained in response to applicable design-basis hazard levels, as outlined in Section 6.1.1. The design features of the substructure, including the SFP, provide a lateral resisting system for structural support of safety-significant systems and components housed within the substructure to provide assurance that functions performed by these systems and components are not adversely impacted by design basis hazards.


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Materials and Special Construction Techniques

The materials used for construction of the FHB substructure are described in Section 6.4.2.4. There are no special construction techniques used for the FHB substructure.

Design and Analysis Procedures

The design and analysis procedures for the SR FHB substructure are described in Section 6.4.2.3.

SFP Design

The reinforced concrete SFP is seismically designed as described in Section 6.4.1. Hydrodynamic loads, resulting from seismic events, induced by both the impulsive and convective masses of the fluid in the SFP are determined using the methodology provided in ACI 350.3-20, “Code Requirements for Seismic Analysis and Design of Liquid-Containing Concrete Structures and Commentary,” (Reference 7.8.2-10) for rectangular tanks.

Structural Acceptance Criteria

The acceptance criteria for the FHB reinforced concrete structures and the SFP are in accordance with the limits defined in ACI 349-13 as modified by RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” Revision 2, for the load combinations, as applicable, provided in Table 6.4-5 for SR reinforced concrete structures. The FHB substructure meets the Factor of Safety requirements for load combinations in Table 6.4-9 for the stability analysis of SR foundations.

The acceptance criteria for the SFP liner are in accordance with the limits defined in AISC N690-18 for the load combinations provided in Table 6.4-8 and requirements for fabrication and construction.

#### 7.8.2.1.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the FHB substructure is described in Section 7.8.2.1.3. The following programmatic special treatments are preliminarily identified as applicable to the FHB substructure to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing and Analysis Program (see Chapter 12)


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### 7.8.2.2 Fuel Handling Building Superstructure

#### 7.8.2.2.1 Summary Description

The FHB superstructure is approximately 76 feet above grade at its highest point. An overhead bridge crane is supported by the FHB superstructure for movement of spent fuel casks and equipment for maintenance within the facility. The superstructure also includes an enclosed corridor located on the west side of the FHB that aligns with an opening in the Reactor Building with tracks for the rail-mounted Ex-Vessel Handling Machine that supports core assembly handling operations from the Reactor Building to the Ex-Vessel Storage Tank in the FHB. The FHB main building is supported on the FHB substructure reinforced concrete slab and the transfer corridor and the FHB north wing are supported on separate concrete slabs.

The FHB superstructure north wing is approximately 50 feet above grade at its highest point and houses Liquid Radwaste Processing System and Solid Radioactive Waste System equipment, within shielded reinforced concrete enclosures, along with a truck bay for receipt of unirradiated fuel and removal of spent fuel and radwaste. The north wing of the superstructure has a second elevation that houses NI Water System and NI Heating, Ventilation, and Air Conditioning System equipment.

#### 7.8.2.2.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the FHB superstructure safety design basis are described below.

Regulatory Criteria

The PDC are identified in Section 5.3. Compliance with PDC that are applicable to the FHB superstructure is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena,* the FHB superstructure is designed in accordance with codes and standards and load combinations identified in Section 6.4.2 to withstand the effects of applicable external hazards without the loss of structural integrity; providing assurance that systems and components, including the NI Cranes and Hoists crane, housed within the superstructure are appropriately protected and not adversely impacted by natural phenomena.

Consistent with PDC 3, *Fire protection,* the FHB superstructure is designed with plant features that minimize the probability and effects of fires and explosions, including the use of noncombustible or fire-resistive construction materials. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases,* the FHB superstructure is designed in accordance with codes and standards and load combinations identified in Section 6.4.2 to withstand the effects of applicable dynamic effects and hazards that result from environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents.


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Consistent with PDC 16, *Containment design,* the FHB superstructure is designed as an enveloping barrier to support maintaining a negative pressure for radionuclide retention as described in Section 7.5.1.

Functional Requirements

The PSFs are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR or NSRST functions as defined in Section 5.2. The FHB superstructure supports the following NSRST functions:

| Function ID | Function Description | Functional Support |
|---|---|---|
| DL4-RR7 | Fuel Handling Building Barrier | The FHB superstructure is designed for a slightly negative pressure as part of the enveloping barrier to reduce the amount of radionuclides escaping for fuel drop events. This function is credited in maintaining main control room habitability for a postulated fuel drop in the spent fuel pool. |

Temporary Load Path for SR SSCs

The FHB superstructure, specifically the transfer corridor grade slab, provides the load path for temporary structural support for the EVHM as it performs its SR PSFs.

External Hazard Protection for NSRST SSCs

The FHB superstructure is designed to protect NSRST SSCs from the effects of applicable external hazards as defined in Section 6.4.2 without the loss of structural integrity.

Load Path for NSRST SSCs

The FHB superstructure provides the load path for structural support of NSRST SSCs in the performance of their respective NSRST PSFs.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the FHB superstructure is provided in Section 6.4.2, along with a description of the use and conformance to the guidance.

Seismic Classification and Design Codes

The superstructure of the FHB is classified as NSRST and seismic class SCN3. The primary design codes for the steel superstructure are AISC 360-16, “Specification for Structural Steel Buildings,” (Reference 7.8.2-4) and ACI 318-19, “Building Code Requirements for Structural Concrete and Commentary,” (Reference 7.8.2-8) as discussed in Section 6.4.2. The


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above-grade portions of the FHB that contain Liquid Radwaste Processing System and Solid Radwaste Processing System system components are designed in accordance with guidance provided in RG 1.143.

Loads and Load Combinations

The FHB superstructure is designed for applicable loads related to construction, normal plant start-up, operation, shutdown, seismic interaction effects and tornado effects. The load combinations for concrete structures and steel structural elements in the FHB superstructure are provided in Table 6.4-11 and Table 6.4-12, respectively.

7.8.2.2.3 Structure Design and Performance Evaluation

NSRST Structural Characteristics that Support Risk-Significant or Defense-in-Depth Functions

The NSRST FHB superstructure is designed in accordance with the codes and standards as identified in Table 6.4-2 to provide assurance that structural integrity is maintained in response to applicable external hazards. The superstructure design features, including the design of the crane rail support steel is in accordance with ASCE 7-16, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” (Reference 7.8.2-5), such that risk-significant and defense-in-depth functions performed by systems and components housed within the superstructure are not adversely impacted by external hazards. In addition, the FHB superstructure is designed for the negative pressure associated with the enveloping barrier requirements for DL4-RR7.

Materials and Special Construction Techniques

The materials used for construction of the FHB superstructure are described in Section 6.4.2.4. There are no special construction techniques used for the FHB substructure.

Design and Analysis Procedures

The FHB superstructure is divided into two separate models to represent the main building and the north wing. For both models GT STRUDL® Version 40 (Reference 7.8.2-6) is used as the primary analysis tool.

Building Geometry

The GT STRUDL analytical model is a representation of the primary lateral force resisting system and gravity load resisting systems that contribute to the stability of the FHB structure. The primary members include columns, girders, beams, and horizontal and vertical braces that are critical to the stability and load path of the structure. Secondary members, that do not contribute to stability of the structure, including girt systems, roof purlins, and construction aids are not explicitly modeled. Instead, equivalent member loads are applied to the explicitly modeled structural members to account for their weight.


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Lateral Force Resisting System

The main building of the FHB superstructure is comprised of steel moment frame in both the north-south and east-west directions with a roof diaphragm composed of horizontal trusses. The north wing of the FHB superstructure is comprised of concrete shear walls for the Liquid Radwaste Processing System and Gaseous Radwaste Processing System rooms and braced frames supporting the steel superstructure with horizontal roof bracing serving as the roof diaphragm.

Application of Loads

The FHB superstructure is analyzed for normal, severe and extreme load conditions as described in Section 6.4.2. The loadings considered include site-specific snow, wind, and seismic environmental conditions in combination with dead and live loads that exist as part of normal activities within the FHB. The code requirements of ASCE 7-16 are followed for the development of crane, wind, snow, and seismic loadings. The static loads calculated in the FHB design load calculation are applied to the analytical model based on tributary widths of the analytical members. For analytical members within GT STRUDL the self-weight is calculated based on densities and section properties of the analytical members.

FHB Superstructure Seismic Analysis

The FHB superstructure seismic classification is SCN3 as described in Section 6.4.1. The following design parameters are used for developing seismic demands of SCN3 structures using ASCE 7-16 modal response spectrum analysis procedures:

- Since the FHB superstructure is founded on a SR substructure, the risk targeted maximum considered earthquake is taken as the in-structure floor response spectra developed by the method described in Section 6.4.1.
- Seismic risk category is set to II, and seismic importance factor is set to 1.0 from Table 1.5-2 of ASCE 7-16.
- Per FHB occupancy considerations, Chapter 12 of ASCE 7-16 is used to establish seismic design requirements treating FHB superstructure as a “Building Structure.”

The enveloping response spectrum curves are input into GT STRUDL for both horizontal directions. The vertical component of the seismic effect is determined by scaling the dead load in accordance with Equation 12.4-4a of ASCE 7-16.

Modal response spectrum analysis is performed per Section 12.9.1 of ASCE 7-16 for seismic loading. Both seismic and non-seismic loads are analyzed using the direct analysis method from AISC 360-16, Chapter C. For load combinations comprised of exclusively of non-seismic loads GT STRUDL’s built-in nonlinear (second-order) analysis function is used to meet the analysis and design requirements of Chapter C of AISC 360-16 for the direct analysis method. For load combinations that include seismic loads, the approximate second-order analysis procedure from Appendix 8 of AISC 360-16 is used to meet the analysis and design requirements of Chapter C of AISC 360-16 for the direct analysis method.


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FHB Superstructure Design

The FHB superstructure members are designed by using the load and resistance factor design of AISC 360-16 for steel and limit state method of ACI 318-19 for concrete.

As described in Section 6.4.2, in addition members subjected to tornado-generated missile impacts are designed based on the energy balance method to obtain an upper limit estimate of structural elasto-plastic response. The strain energy of the target at maximum response is used to balance the residual kinetic energy of the target (or target-missile combination) resulting from missile impact.

Structural members are shown to possess adequate ductility and deflection capacity to absorb the strain energy due to impact of the external hazard tornado-generated missiles. Permissible ductility and displacements for steel structures are obtained from Table NB3.1 of AISC N690-18 and permissible ductility and displacements limits for concrete structures are obtained from Appendix F of ACI 349-13.

The FHB main building superstructure provides support for the NI Cranes and Hoists overhead bridge crane and the crane rails and column supports are designed to the maximum wheel loads as described in Section 4.9 of ASCE 7-16. The maximum vertical wheel load is increased by an impact factor of 25 percent per Section 4.9.3 of ASCE 7-16 to account for the remotely operated powered bridge crane.

For the FHB substructure crane rail support design, additional crane test loads and load combinations are included in accordance with Section 4140 of ASME NOG-1-2020, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” (Reference 7.8.2-7).

Structural Acceptance Criteria

The acceptance criteria for the concrete portions of the FHB superstructure are in accordance with the limits defined in ACI 318-19 for the load combinations, as applicable, in Table 6.4-11 for NSRST concrete structures. The acceptance criteria for the FHB steel superstructure are in accordance with the limits defined in AISC 360-16 for the load combinations, as applicable, in Table 6.4-12 for NSRST steel structures.

#### 7.8.2.2.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the FHB superstructure is described in Section 7.8.2.2.2. The following programmatic special treatments are preliminarily identified as applicable to the FHB superstructure to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing and Analysis Program (see Chapter 12)


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## References


7.8.2-1 International Building Code, IBC 2021.

7.8.2-2 American Concrete Institute, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” ACI 349-13.

7.8.2-3 American Institute of Steel Construction, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” ANSI/AISC N690-18.

7.8.2-4 American Institute of Steel Construction, “Specification for Structural Steel Buildings,” ANSI/AISC 360-16.

7.8.2-5 American Society of Civil Engineers, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” ASCE/SEI 7-16.

7.8.2-6 GT STRUDL Version 40, November 2021, Hexagon PPM, USA.

7.8.2-7 American Society of Mechanical Engineers, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” ASME NOG-1-2020.

7.8.2-8 American Concrete Institute, “Building Code Requirements for Structural Concrete and Commentary,” ACI 318/318R-19.

7.8.2-9 American Institute of Steel Construction, “Seismic Provisions for Structural Steel Buildings,” ANSI/AISC 341-16.

7.8.2-10 American Concrete Institute, “Code Requirements for Seismic Analysis and Design of Liquid-Containing Concrete Structures and Commentary,” ACI 350.3-20.




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### 7.8.3 Reactor Auxiliary Building

The Reactor Auxiliary Building (RAB) is a rectangular three-level structure that houses components that receive the intermediate sodium from the Reactor Building and transfers the heat to the Nuclear Island (NI) Salt System. There are two reinforced concrete RAB substructure levels and a steel-framed RAB superstructure with metal siding and metal roof. Both the below-grade and above-grade portions of the RAB are classified as non-safety-related with special treatment (NSRST) as the structure houses the following NSRST systems:

- Intermediate Heat Transport System (IHT) piping loops and components, including the Sodium-Salt Heat Exchangers, Intermediate Sodium Pumps, and IHT expansion tanks described in Section 7.1.4
- Sodium Cover Gas (SCG) equipment discussed in Section 7.2.3
- Sodium Processing System (SPS) components described in Section 7.2.4

The Intermediate Air Cooling System (IAC) Air Stack Structures and Equipment (ASE), discussed in Section 7.2.2, are located adjacent to the RAB and are connected to equipment within the RAB. In addition, the RAB houses NI Air and Gas Distribution System and NI Salt System equipment. Electrical equipment is located inside of non-safety-related with no special treatment electrical-modules that are located north and south of the RAB.

As shown in Figure 1.2-1, the RAB superstructure is located 5 feet to the west of the Reactor Building superstructure with an underground pipe chase between the two buildings and an above ground personnel passageway that connects the two buildings. The distance between the RAB substructure and Reactor Building substructure is approximately 16.5 feet. Distance between adjacent buildings meets International Building Code (IBC) 2021 (Reference 7.8.3-1) requirements. Seismic interactions between the RAB and Reactor Building are addressed based on methodologies described in Section 6.4.1. Figures 1.1-10 through 1.1-11 show the general arrangements for various RAB elevations.

#### 7.8.3.1 RAB Substructure

##### 7.8.3.1.1 Summary Description

The RAB substructure consists of reinforced concrete walls and slabs that divide the substructure into two different elevations. The upper level of the substructure, which houses the IHT drain tanks, SPS equipment, and the NI Salt System drain vessel, is approximately 119 feet wide, 126 feet long, and 30 feet deep with a foundation slab thickness of 5 feet. The lower RAB substructure level is approximately 55 feet wide, 40 feet long, and 27.5 feet deep and has a foundation slab thickness of 3 feet. This lower substructure level, which houses the IHT supplemental tank is 62 feet below grade, and is inaccessible by personnel from adjacent spaces. The RAB substructure exterior walls are 4.5 feet thick and the substructure roof at grade is 3 feet thick.

##### 7.8.3.1.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the RAB safety design basis are described below.


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Regulatory Criteria (RAB Substructure and Superstructure)

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the RAB is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the RAB substructure and superstructure are designed in accordance with codes and standards and load combinations identified in Section 6.4.2 to withstand the effects of applicable external hazards without the loss of structural integrity providing assurance that systems and components housed within the RAB are appropriately protected and not adversely impacted by natural phenomena.

Consistent with PDC 3, *Fire protection*, the RAB substructure and superstructure are designed with plant features that minimize the probability and effects of fires and explosions, including the use of noncombustible or fire-resistive construction materials. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the RAB substructure and superstructure is designed in accordance with codes and standards and load combinations identified in Section 6.4.2 to withstand the effects of applicable dynamic effects and hazards that result from environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents.

Consistent with PDC 16, *Containment design*, the RAB substructure SPS Cells are designed as an enveloping boundary for functional containment for radionuclide retention.

Functional Requirements

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR or NSRST functions as defined in Section 5.2. The RAB substructure supports the following NSRST functions:

| Function ID | Function Description | RAB Functional Support |
|---|---|---|
| DL4-RR3 | SPS Cells Barrier | The RAB substructure provides an enveloping boundary for radionuclide retention in the SPS cells. |



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External Hazard Protection for NSRST SSCs

The RAB substructure is designed to protect NSRST SSCs from the effects of applicable external hazards as defined in Section 6.4.2 without the loss of structural integrity.

Load Path for NSRST SSCs

The RAB substructure provides the load path for structural support of NSRST SSCs in the performance of their respective NSRST PSFs.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the RAB substructure is provided in Section 6.4.2, along with a description of the use and conformance to the guidance.

Seismic Classification and Design Codes

The substructure of the RAB is classified as NSRST with a seismic classification of SCN1. The primary design code is ACI 318-19, “Building Code Requirements for Structural Concrete and Commentary,” (Reference 7.8.3-3) for the reinforced concrete design as described and Section 6.4.2.1. Other codes and standards applicable to the RAB substructure are identified in Tables 6.4-2 and 6.4-3.

Loads and Load Combinations

The RAB substructure is designed for applicable loads related to construction, normal plant start-up, operation, shutdown, and design basis events. The load combinations for the RAB substructure are provided in Section 6.4.2.2.

7.8.3.1.3 Structure Design and Performance Evaluation

NSRST Structural Characteristics that Support Safety-Significant Functions

The NSRST RAB substructure is designed in accordance with the codes and standards as identified in Table 6.4-2 such that structural integrity is maintained in response to applicable hazards, as outlined in Section 6.1.3. The substructure design features, which includes reinforced concrete walls and foundation slabs, provide a lateral resisting system for structural support of NSRST systems and components housed within the substructure to provide assurance that functions performed by these systems and components are not adversely impacted by applicable hazards. In addition, the RAB substructure SPS Cells are designed for the slight negative pressure associated with the enveloping boundary requirements for DL4-RR3.

Materials and Special Construction Techniques

The materials used for construction of the RAB substructure are described in Section 6.4.2.4. There are no special construction techniques for the RAB substructure.


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Design and Analysis Procedures

The RAB substructure finite element analysis model is developed using GT STRUDL® Version 40 (Reference 7.8.3-4) as the primary analysis tool. The geometry and dimensions of the substructure are derived using nodal coordinates. The finite element type, mesh size, and material properties of the RAB substructure GT STRUDL model are described in Section 6.4.2.3.2. For the boundary conditions, soil springs are calculated using the ASCE 4-16, "Seismic Analysis of Safety-Related Nuclear Structures," (Reference 7.8.3-5) simple spring method that are applied to the bottom of the RAB substructure slabs in contact with soil.

The dead weight of the RAB substructure is defined by weight densities. Other applicable dead and live loads are defined by element surface loads that are applied as a force-unit area. Lateral forces applied to the exterior walls of the substructure for hydrostatic and soil pressure and seismic lateral earth pressure are developed and applied as surface loads to the model elements. The seismic lateral earth pressures are developed utilizing methodologies described in Section 8.2.2 of ASCE 4-16 for non-yielding walls with no significant structure to structure interaction.

Superstructure loads are defined by concentrated loads and moments at the column joints. To account for the difference in ASCE 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," (Reference 7.8.3-6) risk categories between the superstructure and substructure, a scale factor of 1.5 is applied to the superstructure seismic loads.

The RAB substructure considers three categories of loads for the seismic mass:

- Dead weight plus 25 percent of live load
- Superstructure loads, which includes its dead weight, 25 percent of live load, and 20 percent of snow load
- Equipment weights

All degrees of freedoms are specified for the direction of the inertia.

The seismic inertia from the substructure is calculated using provisions in Chapter 15 of ASCE 7-16 for rigid non-building structures. The demands from the seismic and non-seismic loads are combined with proper load factors to form load combinations based on IBC 2021 and ASCE 7-16 with additional NSRST load combinations as described in Section 6.4.3. A static analysis is performed for the load combinations including seismic, and results are used for the design of the substructure structural members.

The RAB substructure stability is evaluated for overturning, sliding, and buoyancy utilizing the load combinations shown on Section 6.4.2.2.5 and the factors of safety described in Section 6.4.2.2.6.

Structural Acceptance Criteria

The acceptance criteria for NSRST reinforced concrete structures are in accordance with the limits defined in ACI 318-19 for the load combinations, provided in Table 6.4-7 for NSRST concrete structures.


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#### 7.8.3.1.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the RAB substructure is described in Section 7.8.3.1.2. The following programmatic special treatments are preliminarily identified as applicable to the RAB substructure to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing and Analysis Program (see Chapter 12)

### 7.8.3.2 RAB Superstructure

#### 7.8.3.2.1 Summary Description

The steel-framed RAB superstructure is 144 feet wide by 214 feet long and 70 feet above grade at its highest point. The RAB superstructure main floor at grade level supports the IHT Sodium-Salt Heat Exchangers that is part of the substructure analytical model. The superstructure includes two equipment platform levels that provide structural support to components, including SCG equipment, IHT Intermediate Sodium Pumps, NI Air and Gas Distribution System equipment, and IHT expansion tanks. The east end of the RAB superstructure roof includes two removable roofing panels allowing crane access to remove Intermediate Sodium Pumps. The RAB superstructure roof also supports NI Heating, Ventilation, and Air Conditioning System equipment described in Section 7.5.1.

#### 7.8.3.2.2 Design Basis

Regulatory Criteria

The RAB superstructure is designed to ensure that NSRST systems and components are protected from the effects of applicable hazards defined in Section 6.1.3 and is structurally designed to meet PDC 1, PDC 2, PDC 3, and PDC 4 as described in Section 7.8.3.1.2.

Functional Requirements

The PSFs are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as NSRST functions as defined in Section 5.2. The RAB superstructure supports the following safety-significant functions, which are designated as NSRST to indicate the safety classification of the RAB superstructure relied upon to perform the function.

External Hazard Protection for NSRST SSCs

The RAB superstructure is designed to protect NSRST SSCs from the effects of applicable external hazards as defined in Section 6.4.2 without the loss of structural integrity.


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Load Path for NSRST SSCs

The RAB superstructure provides the load path for structural support of NSRST SSCs in the performance of their respective NSRST PSFs.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the RAB substructure is provided in Section 6.4.2, along with a description of the use and conformance to the guidance.

Seismic Classification and Design Codes

The RAB superstructure seismic classification is SCN3. The primary design code for the structural design of the RAB superstructure is ANSI/AISC 360-16, “Specification for Structural Steel Buildings,” (Reference 7.8.3-7). Other design codes are provided in Table 6.4-3.

Loads and Load Combinations

Load combinations and load factors for the RAB superstructure structural steel are provided in Table 6.4-8.

7.8.3.2.3 Structure Design and Performance Evaluation

NSRST Structural Characteristics that Support Risk-Significant or DID Functions

The NSRST RAB superstructure is designed in accordance with the codes and standards identified in Section 6.4.2 such that structural integrity is maintained in response to applicable hazards, as outlined in Section 6.1.3. The superstructure has a steel lateral resisting system for structural support of the IHT system and other NSRST components providing assurance that functions performed by these systems and components are not adversely impacted by applicable hazards. An automobile tornado missile impact barrier is provided to protect the IAC and IHT systems providing assurance that functions performed by these systems and components are not adversely impacted.

Materials and Special Construction Techniques

The materials used for construction of the RAB superstructure are described in Table 6.4-10 and Section 6.4.2.4. There are no special construction techniques for the RAB superstructure.

Design and Analysis Procedures

The analysis and design of the RAB superstructure is performed using GT STRUDL Version 40 as the primary analysis tool.


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Building Geometry

The geometry and dimensions of the RAB superstructure are modeled using nodal coordinates. The GT STRUDL model is a representation of the primary lateral force resisting system and gravity load systems that contribute to the stability of the RAB superstructure. Primary members include main framing elements such as girders, beams, columns, and bracings that are key to the global stability of the structure. The secondary members, which do not contribute to stability of the structure, including girt systems, and construction aids are not explicitly modeled. Instead, equivalent member loads are applied to the explicitly modeled structural members. The RAB GT STRUDL model also considers the roof infill beams to capture the roof behavior accurately.

Lateral Force Resisting System

The RAB superstructure design has a lateral force resisting system comprised of steel concentrically braced frames and steel moment frames, in the east-west and north-south directions, respectively. The roof of the RAB is comprised of internal bracings to function as lateral braces for the roof and acts as a diaphragm.

Application of Loads

The RAB is analyzed for normal, severe and extreme load conditions as described in Section 6.4.2. The loadings considered include site-specific snow, wind, and seismic environmental conditions in combination with dead and live loads that exist as part of normal activities within the RAB. The code requirements of ASCE 7-16 are followed for the development of wind, snow and seismic loadings. The static loads calculated in the RAB design load calculation are applied to the analytical model based on tributary widths of the analytical members. The self-weight of the superstructure model is automatically calculated in GT STRUDL based on the densities and section properties of the analytical members.

RAB Superstructure Seismic Analysis

The RAB steel superstructure is classified as SCN3 as described in Section 6.4.1. The following design parameters are used for developing seismic demands using ASCE 7-16 procedures:

1. The safe shutdown earthquake-based response motion from the substructure is used in place of risk targeted maximum considered earthquake for analysis of the superstructure.

2. Seismic risk category is set to II, and seismic importance factor is set to 1.0.

3. The RAB is not normally occupied, thus, for RAB occupancy considerations, Chapter 15 of ASCE 7-16 is used to establish seismic design requirements treating the RAB superstructure as a “Non-building structure that are similar to buildings.”

The enveloped horizontal acceleration response spectra curves are imported into GT STRUDL for the two horizontal directions. The vertical component of the seismic effect is considered by modifying the dead load combination factors in accordance with ASCE 7-16, Section 12.4.2.2.


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Modal response spectrum analysis is performed per ASCE 7-16, Section 12.9.1 and guidance from Reference 7.8.3-8. The seismic and non-seismic loads are analyzed using the direct analysis method as set by AISC 360-16, Chapter C.

For load combinations that are exclusive of seismic loads, the RAB superstructure is analyzed utilizing GT STRUDL's built-in nonlinear (second order) analysis function using the direct analysis method from AISC 360-16, Chapter C.

For load combinations that are inclusive of seismic loads, the methodology from Reference 7.8.3-8 is used. The methodology implements the approximate second-order analysis procedure per Appendix 8 of AISC 360-16 in combination with the modal response spectrum analysis from ASCE 7-16 to meet the analysis and design requirements of AISC 360-16, Chapter C for the direct analysis method.

RAB Structural Design

RAB superstructure members are designed utilizing the load and resistance factor design methodology from AISC 360-16. In addition to the load and resistance factor design of structural members against the environmental loads, the RAB superstructure is designed to protect the NSRST IHT sodium piping and other NSRST equipment housed inside the building from automobile tornado missile impacts.

The analysis and design of the RAB steel superstructure structural elements or physical barriers subject to tornado generated missile impact effects is based on the energy balance method and procedures, to obtain an upper limit estimate of structural elasto-plastic response. The strain energy of the target at maximum response is used to balance the residual kinetic energy of the target (or target-missile combination) resulting from missile impact.

The RAB superstructure design includes built-up sections (cruciform or box sections) columns to resist the tornado missile impact in the perimeter of the building. These structural members are shown to possess adequate ductility and deflection capacity to absorb the strain energy due to impact of the external hazard tornado-generated missiles. Maximum ductility and displacements are obtained from Table NB3.1 of AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities" (Reference 7.8.3-9).

Structural Acceptance Criteria

The acceptance criteria for the RAB steel superstructure are in accordance with the limits defined in AISC 360 for the load combinations, as applicable, in Table 6.4-8.


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7.8.3.2.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the RAB superstructure is described in Section 7.8.3.2.2. The following programmatic special treatments are preliminarily identified as applicable to the RAB superstructure to meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.0.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing and Analysis Program (see Chapter 12)

**References**

7.8.3-1 International Building Code, IBC 2021

7.8.3-2 Not Used

7.8.3-3 American Concrete Institute, “Building Code Requirements for Structural Concrete and Commentary,” ACI 318-19.

7.8.3-4 GT STRUDL Version 40, November 2021, Hexagon PPM, USA.

7.8.3-5 American Society of Civil Engineers, “Seismic Analysis of Safety-Related Nuclear Structures,” ASCE 4-16.

7.8.3-6 American Society of Civil Engineers, Minimum Design Loads and Associated Criteria for Buildings and Other Structures, ASCE 7-16.

7.8.3-7 American Institute of Steel Construction, “Specification for Structural Steel Buildings,” ANSI/AISC 360-16.

7.8.3-8 R. Shankar Nair, James O. Malley and John D. Hooper, “Design of Steel Building for Earthquake and Stability by Application of ASCE 7 and AISC 360,” Engineering Journal, Third Quarter 2011.

7.8.3-9 American Institute of Steel Construction, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” ANSI/AISC N690-18.


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### 7.8.4 Nuclear Island Control Building

The NCB is composed of two main structural sections. The below-grade substructure is classified as safety-related (SR) as it provides housing for SR electrical systems and components, including the Reactor Protection System (RPS), Intermediate Sodium Pump and Primary Sodium Pump breakers; non-safety-related with special treatment (NSRST) electrical equipment; and the Remote Shutdown Complex (RSC). The above-grade superstructure, which does not house any SR equipment, is classified as non-safety-related with special treatment (NSRST), houses the main control room (MCR) and associated equipment and serves as the primary control center during normal plant operations.

The below-grade NCB substructure is designed to withstand external design-basis hazard levels (DBHLs) to provide protection of the SR and NSRST systems. The reinforced concrete NCB substructure is approximately 82-feet wide by 131-feet long and is embedded 33-feet below grade. The south half of the NCB substructure also provides foundation support to the NCB superstructure.

The above-grade NCB superstructure is a rectangular-shaped steel-framed metal building that is approximately 82-feet wide by 88-feet long and 21 feet high from the top of grade. The MCR, including the computer room and associated rooms containing support functions to the MCR, housed within the NCB is surrounded by a reinforced enclosure to provide protection from chemical, radiological, and weather-related events.

As shown on the site layout plan in Figure 1.2-1, the NCB is south of the Reactor Building. The separation distance between the Reactor Building and NCB above grade is approximately 123 feet, which meets fire separation requirements between adjacent structures in International Building Code (IBC) 2021 for non-rated exterior walls. Seismic interactions are addressed based on methodologies described in Section 6.4.1. Figure 1.1-12 through Figure 1.1-15 provide general arrangement drawings of the NCB.

#### 7.8.4.1 Nuclear Island Control Building Substructure

##### 7.8.4.1.1 Summary Description

The SR below-grade NCB substructure is divided into several compartments as shown in Figure 1.1-12 that provide housing for the RPS trains as described in Section 7.6.3, NI Uninterruptible AC Power Supply System, and NI DC Power Supply System equipment rooms and batteries discussed in Section 7.7.1, and electrical equipment associated with the Primary Sodium Pump (see Section 7.1.3) and ISP (see Section 7.1.4). In addition, the NCB substructure includes a location for the RSC, as described in Section 7.6.7.2, that is used to monitor plant safe shutdown conditions in case the MCR is no longer habitable. The main structural interior and exterior concrete walls are approximately 2-feet and 3-feet thick, respectively. The substructure roof is approximately 2-feet thick and the foundation slab is approximately 3-feet thick.


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7.8.4.1.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the NCB substructure safety design basis are described below.

Regulatory Criteria

Principal Design Criteria (PDC) are identified in Section 5.3. Compliance with PDC that are applicable to the NCB substructure is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the NCB substructure is designed to withstand the effects of natural phenomena without loss of capability to perform safety-significant functions to withstand the effects of DBHLs as described in Section 6.1.1. The design of the NCB provides protection of SSCs that perform safety-significant functions from applicable natural phenomena including tornadoes and associated missiles, external flooding, and extreme climate conditions. The NCB substructure is designed to withstand the effects of earthquakes in accordance with the methods described in Section 6.4.1. The NCB is designed, based on the appropriate combination of the effects of normal and accident conditions with the effects of natural phenomena, to maintain capability to perform safety-significant functions.

Consistent with PDC 3, *Fire protection*, the NCB substructure is designed with plant features that minimize the probability and effects of fires and explosions, including the use of noncombustible or fire-resistive construction materials. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the NCB substructure is designed in accordance with codes and standards and specific load combinations identified in Section 6.4.2 to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents without the loss of structural integrity.

Consistent with PDC 19, *Control room*, the NCB substructure includes the RSC that contains instrumentation and controls for monitoring of the plant if the MCR is evacuated as further described in Section 7.6.7.2.

Functional Requirements

PRA safety functions (PSFs) are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The NCB substructure supports the following safety-significant functions, which are designated as SR or NSRST:

External Hazard Protection for SR SSCs (SR)

The NCB substructure is designed to protect SR SSCs from the effects of design-basis hazard levels (DBHL) (Section 6.1.1) without the loss of structural integrity.


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Load Path for SR SSCs (SR)

The NCB substructure provides the load path for structural support of SR SSCs in the performance of their respective SR PSFs.

External Hazard Protection for NSRST SSCs (NSRST)

The NCB substructure is designed to protect NSRST SSCs from the effects of design-basis hazard levels (Section 6.1.1) without the loss of structural integrity.

Load Path for NSRST SSCs (NSRST)

The NCB substructure provides the load path for structural support of NSRST SSCs in the performance of their respective NSRST PSFs.

Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the NCB substructure is provided in Section 6.4.2, along with a description of the use and conformance to the guidance.

Seismic Classification and Design Codes

The seismic classification of the below grade SR NCB substructure is SCS1 and the primary design code is ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” (Reference 7.8.4-1) for reinforced concrete design, as described in Section 6.4.2.1.

Loads and Load Combinations

The NCB substructure is designed for all applicable design loads for construction, normal plant start-up, operation, shutdown, and DBHLs. The load combinations for the NCB substructure are provided in Section 6.4.2.2.

7.8.4.1.3 Structure Design and Performance Evaluation

SR Structural Characteristics that Support PSFs

The SR NCB substructure is designed in accordance with the codes and standards as identified in Table 6.4-3 to ensure that structural integrity is maintained in response to applicable DBHLs, as outlined in Section 6.1.1. The substructure design features, including the structural design of the RPS train rooms and the RSC, ensure that systems and components housed within the substructure that perform SR functions and NSRST functions are not adversely impacted by design basis hazards.

Materials and Construction Techniques

The materials used for construction of the NCB substructure are described in Section 6.4.2.4. There are no special construction techniques for the NCB substructure.


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Design and Analysis Procedures

The design and analysis procedures for the SR NCB substructure are described in Section 6.4.2.3.

Structural Acceptance Criteria

The acceptance criteria for reinforced concrete structures and components are in accordance with the limits defined in ACI 349-13 as modified by RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” Revision 3, for the load combinations, as applicable, in Table 6.4-5 for SR reinforced concrete structures. The NCB substructure meets the Factor of Safety requirements for load combinations in Table 6.4-9 for the stability analysis of SR structures.

#### 7.8.4.1.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the NCB substructure is described in Section 7.8.4.1.2. The following programmatic special treatments are preliminarily identified as applicable to the NCB substructure meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing, and Analysis Program (see Chapter 12)

### 7.8.4.2 NCB Superstructure

#### 7.8.4.2.1 Summary Description

The above-grade NSRST NCB superstructure is a one-story steel-framed structure which serves as the control center for the NI and also includes equipment for monitoring and controlling Energy Island systems and components. As shown on Figure 1.1-13, the MCR is housed within the NCB superstructure in a MCR enclosure area with associated rooms containing support functions to the MCR, including computer rooms, shift manager’s office, briefing room, restroom, kitchenette, and break room area.

The MCRE reinforced enclosure is designed in accordance with ACI 318-08 (Reference 7.8.4-5) working with ACI 349-13, Appendix F special provisions for impactive effects that provides protection from chemical, radiological, and weather-related external events. Within the MCR enclosure, the MCR and other spaces with electrical equipment, such as the computer room, have an approximate 4-feet high raised access flooring system for electrical cable distribution and equipment. Stairwells are located on both sides of the superstructure to provide access to the NCB substructure that are also designed for external hazards.


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7.8.4.2.2 Design Basis

The compliance with regulatory criteria, identification of design criteria, and codes and standards that define the NCB superstructure safety design basis are described below.

Regulatory Criteria

The PDC are identified in Section 5.3. Compliance with PDC that are applicable to the NCB superstructure is described below.

Consistent with PDC 2, *Design bases for protection against natural phenomena*, the NCB superstructure is designed in accordance with codes and standards and load combinations identified in Section 6.4.2 to withstand the effects of applicable external hazards without the loss of structural integrity providing assurance that systems and components housed within the superstructure are appropriately protected and not adversely impacted by natural phenomena.

Consistent with PDC 3, *Fire protection*, the NCB superstructure is designed with plant features that minimize the probability and effects of fires and explosions, including the use of noncombustible or fire-resistive construction materials. Fire detection and firefighting features are described in Section 7.5.2.

Consistent with PDC 4, *Environmental and dynamic effects design bases*, the NCB superstructure is designed in accordance with codes and standards and load combinations identified in Section 6.4.2 to withstand the effects of applicable dynamic effects and hazards that result from environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents.

Consistent with PDC 19, *Control room*, the NCB superstructure includes the MCR where actions are taken for safe plant operations. The MCR is located within a reinforced enclosure which provides adequate radiation protection to control room occupants as described in Section 5.3.2.10 and protection from applicable external hazards.

Functional Requirements

The PSFs are determined through the plant-specific PRA described in Section 3.1. The safety-significant PSFs are identified as SR functions or NSRST functions as defined in Section 5.2. The NCB superstructure supports the following safety-significant functions, which are designated as NSRST to indicate the safety classification of the NCB superstructure relied upon to perform the function:

External Hazard Protection for NSRST SSCs

The NCB superstructure is designed to protect NSRST SSCs from the effects of applicable external hazards as defined in Section 6.4.2 without the loss of structural integrity.

Load Path for NSRST SSCs

The NCB superstructure and the MCRE provides the load path for structural support of NSRST SSCs in the performance of their respective NSRST PSFs.


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Regulatory Guidance

NRC regulatory guidance documents are listed in Section 1.4.1. Guidance applicable to the NCB superstructure is provided in Section 6.4.2, along with a description of the use and conformance to the guidance.

Seismic Classification and Design Codes

The seismic classification of the NSRST NCB superstructure is SCN1. The primary design code for the structural design of the NCB superstructure is ANSI/AISC 360-16, “Specification for Structural Steel Buildings” (Reference 7.8.4-3) as discussed in Section 6.4.2. In addition, the MCRE reinforced enclosure encompassing the MCR and supporting rooms and areas are designed in accordance with ACI 318-08 working with ACI 349-13, Appendix F special provisions for impactive effects. The application of ACI 318-08 working with ACI 349-13, Appendix F special provisions is for protection for control room occupants, in addition of protection of NSRST functions.

Loads and Load Combinations

The NCB superstructure and foundation is designed for applicable design loads for construction, normal plant start-up, operation, shutdown, seismic interaction effects and tornado effects. The load combinations for concrete structures and steel structural elements in the NCB superstructure are provided in Table 6.4-11 and Table 6.4-12, respectively.

7.8.4.2.3 Structure Design and Performance Evaluation

NSRST Structural Characteristics that Support Risk-Significant or Defense-in-Depth Functions

The NCB superstructure is designed to not collapse under seismic or tornado loading conditions as described in Section 6.4.2 and provides structural support and appropriate levels of protection of NSRST systems and components housed within the MCR that perform safety-significant functions. The NSRST NCB superstructure is designed in accordance with the codes and standards as identified in Table 6.4-3 to ensure that structural integrity is maintained in response to applicable external hazards, as outlined in Section 6.4.2. The superstructure design features, including design features for the MCR enclosure, ensure that risk-significant and defense-in-depth functions performed by systems and components housed within the superstructure are not adversely impacted by external hazards.

Materials and Construction Techniques

The materials used for construction of the NCB superstructure are described in Section 6.4.2. There are no special construction techniques for the NCB superstructure.


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Design and Analysis Procedures

An analysis and design of the NSRST NCB superstructure is performed using the structural analysis software GT STRUDL® Version 40 (Reference 7.8.4-4) as the primary analysis tool. This analysis includes the following items:

- Structural Member Design
- Deflection Checks
- Support Reactions
- Dynamic Response Characteristics

The geometry and dimensions of the NCB superstructure are modeled using nodal coordinates and design loads calculated are applied as distributed line loads to the structural members according to their tributary widths. Where load-bearing members are not modeled (e.g., girts), the calculated reaction from such a member are applied as a load to the next member along the load path (e.g., columns).

The NCB superstructure design has a lateral force resisting system comprised of steel concentrically braced frames and steel moment frames, in the north-south and east-west directions, respectively. The NCB roof is constructed of untopped steel decking with vertical steel braced frames in the north-south direction and is idealized as a flexible diaphragm in accordance with Section 12.3.1.1(a) of ASCE 7-16. In the east-west direction ordinary moment frames are utilized. Per Section 12.8.4.2 of ASCE 7-16, application of accidental torsion is not required since the NCB structure does not possess Type 1a or Type 1b horizontal structural irregularities.

The NSRST NCB superstructure is also analyzed for seismic interaction effects to ensure that its failure does not impact the SR NCB substructure in case of a seismic event. Enveloping response spectrum acceleration curves are used as input into GT STRUDL and the seismic analysis is performed using the response spectrum analysis methodology in Section 12.9 of ASCE 7-16. The effective seismic mass is calculated in accordance with Section 12.7.2 of ASCE 7-16 and includes dead load only. Live load and roof snow load are not considered applicable in the calculation of seismic effective mass, since the NCB is a single-story structure and the flat roof snow load does not exceed 30 pounds per square foot.

Modal response spectrum analysis is performed for all modes with natural frequency less than 100 hertz (cycles per second) using the enveloped response spectra as input. Modal responses are combined using the Complete Quadratic Combination Method in accordance with methodologies described in Section 12.9.1.3 of ASCE 7-16 with a total combined modal mass participation greater than 90 percent being obtained in each orthogonal horizontal direction.

The equivalent lateral force procedure as delineated in Section 12.8 of ASCE 7-16 is used to calculate the seismic base shears for the NCB superstructure. Member forces and moments from the modal response spectra analysis are scaled by the equivalent lateral force base shear factor as required by Section 12.9.1.4.1 of ASCE 7-16.


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Horizontal seismic forces are determined based on Section 12.3.4 criteria in ASCE 7-16 and the vertical seismic load is determined in accordance with the provisions in Section 12.4 of ASCE 7-16 for input to the seismic load combinations, including the effects of other loads.

The seismic load effects are combined with the effects of other loads in accordance with the provisions of Section 2.3.6 of ASCE 7-16. In addition, seismic load effects are evaluated for both the positive and negative directions in the load combinations for strength and stability.

The NCB superstructure analysis accounts for second-order effects due to member curvature and structure sidesway in accordance with Appendix 7 and Appendix 8 of AISC 360-16. A first-order elastic analysis with amplification factors is applied to first-order forces and moments and used to obtain an estimate of the second-order forces and moments.

The NCB superstructure design for stability is governed by the lateral forces due to the seismic condition and envelope the lateral sway forces due to design wind and tornado load cases.

The calculational results from the GT STRUDL analyses are compared to the specified code criteria and the output reactions at the column supports are used in the design of the NCB substructure and foundation.

Tornado Wind-Generated Missile Impact Analysis & Design

The NSRST superstructure is designed to withstand tornado wind loads and associated tornado-generated automobile missile impact based on guidance in RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1 to ensure that its failure does not impact the SR NCB substructure in case of a tornado event. In accordance with RG 1.76, the automobile missile is considered to impact at all altitudes less than 30 feet above grade level; thus, the NCB steel superstructure roof framing that is 21 feet above grade is also evaluated for tornado missile impact effects, in addition to support columns, edge beams and vertical bracing. Therefore, multiple impact locations are postulated in order to examine the effects of tornado missile impacts on the NCB steel frame superstructure.

Analysis and design of the NCB steel superstructure primary structural elements subjected to tornado-generated missile impact effects are based on the energy balance method and procedures, to obtain an upper limit estimate of structural elasto-plastic response. The strain energy of the target at maximum response is used to balance the residual kinetic energy of the target (or target-missile combination) resulting from missile impact.

The primary components of the NCB steel superstructure are shown to possess adequate ductility and deflection capacity to absorb the strain energy due to impact of the external hazard tornado-generated missiles on the exterior wall and roof surfaces.

The MCRE reinforced enclosure is designed for tornado wind loads and associated missile effects in accordance with ACI 318-08 working with ACI 349-13, Appendix F special provisions for impactive effects.


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Structural Acceptance Criteria

The acceptance criteria for the concrete portions of the NCB superstructure are in accordance with the limits defined in ACI 318-19, “Building Code Requirements for Structural Concrete and Commentary,” (Reference 7.8.4-2) for the load combinations, as applicable, in Table 6.4-11 for NSRST concrete structures. The acceptance criteria for the NCB steel superstructure are in accordance with the limits defined in ACI 318-19 for the load combinations, as applicable, in Table 6.4-12 for NSRST steel structures.

7.8.4.2.4 Special Treatments

Special treatments are initially selected as described in Section 6.3. Preliminary selection of design and construction codes and standards applicable to the NCB superstructure is described in Section 7.8.4.2.2. The following programmatic special treatments are preliminarily identified as applicable to the NCB superstructure meet quality, design, reliability, and performance requirements, and the associated summary program descriptions are provided in Chapter 8.

- Quality Assurance Program Description
- Design Reliability Assurance Program
- Post-Construction Inspection, Testing and Analysis Program (see Chapter 12)

**References**

7.8.4-1 American Concrete Institute, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, American Concrete Institute, ACI 349-13.

7.8.4-2 American Concrete Institute, Building Code Requirements for Structural Concrete and Commentary, ACI 318-19.

7.8.4-3 American Institute of Steel Construction, Specification for Structural Steel Buildings, ANSI/AISC 360-16.

7.8.4-4 GT STRUDL Version 40, November 2021, Hexagon PPM, USA.

7.8.4-5 American Concrete Institute, Building Code Requirements for Structural Concrete and Commentary, ACI 318-08.


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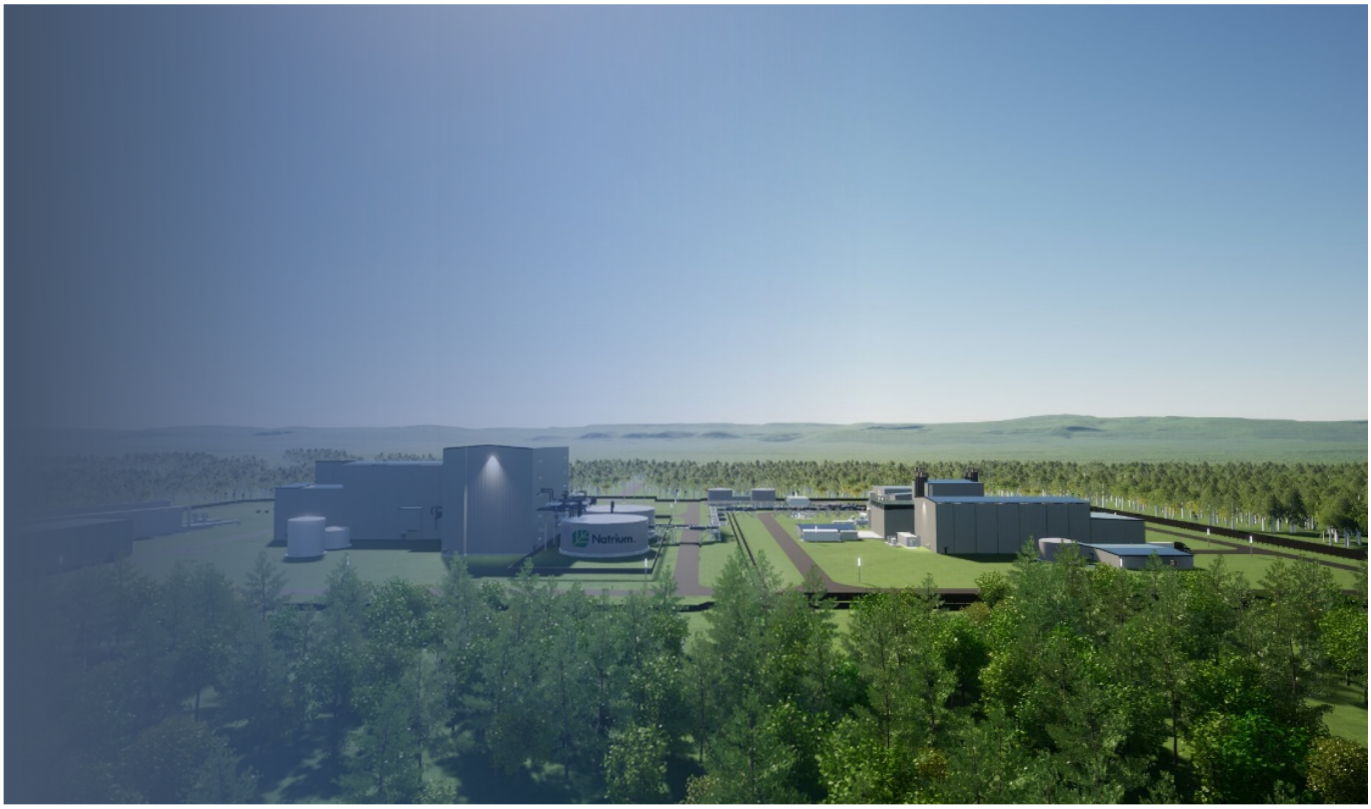

# Kemmerer Power Station Unit 1
# PSAR, Chapter 8

US SFR Owner, LLC, a wholly owned subsidiary of TerraPower, LLC

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## TABLE OF CONTENTS

**Chapter 8 Plant Programs . . . . . 8.0-1**

8.0 Plant Programs . . . . . 8.0-1

8.1 Quality Assurance . . . . . 8.1-1

8.1.1 Disposition of Limitations on NRC Approval of TerraPower QATR . . . . . 8.1-1

8.1.2 Responsibilities During Design and Construction . . . . . 8.1-3

8.1.3 Organization and Controls . . . . . 8.1-4


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## LIST OF TABLES

Table 8.0-1 Special Treatment Programs Supporting Design and Construction . . . . . 8.0-3

Table 8.0-2 Additional Special Treatment Programs for Operations . . . . . 8.0-7


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# Chapter 8 Plant Programs

## 8.0 Plant Programs

This section provides descriptions of plant programs initially considered as special treatments for safety-significant structures, systems, and components (SSCs). Programs that apply to safety-related (SR) or non-safety-related with special treatment (NSRST) SSCs are governed under the Quality Assurance Program Description (QAPD). The QAPD is described in Section 8.1.

Table 8.0-1 lists the special treatment programs initiated to support the design and construction phase. Programs discussed elsewhere in the Preliminary Safety Analysis Report (PSAR) have the respective section identified. If the program is not discussed elsewhere in the PSAR, the table provides a brief description, the objectives of the program, its SSC applicability, the applicable 10 CFR 50 regulation(s), guidance document(s), code(s), and standard(s) that are used in the program.

Table 8.0-2 lists the additional special treatment programs for the operational phase. Information for these programs will be provided at the operating license stage.

The set of programs identified provide reasonable assurance that reliability and performance targets are achieved throughout the plant lifetime and safety-significant uncertainties are effectively addressed as part of defense-in-depth.

The special treatment requirements for safety-significant SSCs are discussed in Section 6.3.

## References

8.0-1 IEEE/IEC, “International Standard - Nuclear facilities - Electrical equipment important to safety - Qualification,” IEEE/IEC 60780-323-2016.

8.0-2 IEEE/IEC, “International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification,” IEEE/IEC 60980-344-2020.

8.0-3 ASME, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” ASME QME-1-2023.

8.0-4 IEEE, “Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” IEEE 323-2003.

8.0-5 ASME, “Quality Assurance Requirements for Nuclear Facility Applications,” Requirement 11, Test Control, ASME NQA-1, 2015.

8.0-6 ASME, “2019 ASME Boiler and Pressure Vessel Code, XI, Rules for Inservice Inspection of Nuclear Reactor Facility Components, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,” ASME BPVC.XI.2-2019.


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8.0-7 U.S. NRC, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs,” SECY-95-132 (ADAMS Accession No. ML003708005)

8.0-8 U.S. NRC, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs,” SRM on SECY-95-132, June 28, 1995 (ADAMS Accession No. ML003708019)

8.0-9 ASME, “Draft - Inservice Testing Requirements for Pumps, Valves, and Dynamic Restraints at Nuclear Facilities,” Draft ASME OM-2.



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**Table 8.0-1 Special Treatment Programs Supporting Design and Construction**
**(Sheet 1 of 4)**

| Program Title, Description, and Objectives | Section | SSC Applicability | Applicable 10 CFR 50 Regulation(s) | Program Reference(s) |
|---|---|---|---|---|
| **Human Factors Engineering Program** | 11.2 | | | |
| **Comprehensive Vibration Assessment Program**<br>A Comprehensive Vibration Assessment Program (CVAP) is required for unique or first-of-its-kind designs. The results of the CVAP will be documented in a series of technical reports submitted to the NRC. These reports will be used to demonstrate compliance with regulatory requirements and to ensure safe operation of the reactor.<br>The subprograms that form the CVAP are summarized below:<br>• Vibration Stress Analysis Program<br>This program provides theoretical bases of structural integrity. It also serves as the basis for the components and specific locations to be monitored in the measurement and inspection programs. The analysis incorporates operating experience and considers the steady-state and other transient conditions.<br>• Vibration Stress Measurement Program<br>This program verifies the results of predictive analysis and addresses instrumentation needs for components without operating experience. Testing must be long enough for the slowest vibration forcing function to reach one million cycles.<br>• Inspection Program<br>This provides quantitative and qualitative verification of predictive analysis and measurement program results. It documents inspection elements (e.g., locations to be inspected that verify the vibration analysis and measurement program), methods of examinations (e.g., visual, surface). Inspection results are compared with baseline inspections for acceptability. | | SR, NSRST, and non-safety-related with no special treatment (NST) | 50.55a (b)(3)(iii)(C) | Regulatory Guide (RG) 1.20, “Comprehensive Vibration Assessment Program for Reactor internals During Preoperational and Startup Testing,” Revision 4 |


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**Table 8.0-1 Special Treatment Programs Supporting Design and Construction**
**(Sheet 2 of 4)**

| Program Title, Description, and Objectives | Section | SSC Applicability | Applicable 10 CFR 50 Regulation(s) | Program Reference(s) |
|---|---|---|---|---|
| **Equipment Qualification Program**<br>The Equipment Qualification (EQ) program provides direction on establishing and performing equipment qualification, and defines how to ensure systems, components, and designs comply with applicable technical requirements, codes, standards, and regulatory requirements. Within the EQ program are three specific qualification disciplines: Electrical Equipment Qualification, Mechanical Equipment Qualification, and Instrumentation and Controls Equipment Qualification, all of which have specific and unique regulatory requirements, codes, and standards. The EQ program includes development of testing and qualification specifications, processes, procedures, and reports. | | SR and NSRST | 50.49<br>50.34(a)(3)(i) | RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” Revision 2<br>IEEE/IEC 60780-323-2016, “IEEE/IEC International Standard - Nuclear facilities - Electrical equipment important to safety - Qualification” (Reference 8.0-1) |
| | | | | RG 1.100, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” Revision 4<br>IEEE/IEC 60980-344-2020, “IEEE/IEC International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification” (Reference 8.0-2)<br>ASME QME-1-2023, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities” (Reference 8.0-3) |
| | | | | RG 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants,” Revision 0<br>IEEE 323-2003, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations” (Reference 8.0-4) |
| **Testing Program**<br>The purpose of the Testing Program is to provide reasonable assurance that testing demonstrates that SR and NSRST SSCs perform satisfactorily in service. The Testing Program also identifies the programs and groups responsible for proof tests, preoperational tests, and operational tests needed to demonstrate compliance with the design. Testing will comply with written test procedures. | | | Appendix B<br>50.43(e)(1) | ASME NQA-1 2015, “Quality Assurance Requirements for Nuclear Facility Applications,” Requirement 11, Test Control (Reference 8.0-5) |



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**Table 8.0-1 Special Treatment Programs Supporting Design and Construction**
**(Sheet 3 of 4)**

| Program Title, Description, and Objectives | Section | SSC Applicability | Applicable 10 CFR 50 Regulation(s) | Program Reference(s) |
|---|---|---|---|---|
| **Reliability and Integrity Management Program**<br>The objective of the Reliability and Integrity Management (RIM) program is to ensure that passive components within the RIM program scope, whose failure could adversely affect plant safety and reliability, function properly throughout the life of the nuclear plant. The RIM program involves design interaction, performance monitoring, inspections, tests, maintenance, and replacements, as strategies to ensure the SSCs achieve an acceptable level of reliability to support the probabilistic risk assessment of the plant.<br>The RIM program includes the formation of the RIM expert panel (RIMEP) and a monitoring and non-destructive examination expert panel (MANDEEP). The RIMEP establishes the scope of SSCs in the program. The RIM program also establishes the development of a Degradation Mechanism Assessment, the allocation of reliability targets to the SSCs, and the identification of RIM strategies to ensure reliability targets can be met, along with various supporting functions. These supporting functions include the activities of the MANDEEP, such as establishing requirements for performance-based qualification of monitoring and non-destructive examination methods and techniques for personnel, procedures, and equipment. | | SR, NSRST, and NST | 50.34(b)(6)(iv) | RG 1.246, “Acceptability of ASME Code, Section XI, Division 2, ‘Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants,’ for Non-Light Water Reactors,” Revision 0<br>ASME BPVC-XI, Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants” (Reference 8.0-6) |
| **Design Reliability Assurance Program**<br>The purpose of the Design Reliability Assurance Program (D-RAP) is to provide the processes and programmatic controls for ensuring SSC reliability during the plant design phases. This includes assurances that:<br>• The plant is designed, constructed, and operated in a manner that is consistent with the reliabilities and capabilities needed to meet the risk targets (i.e., frequency-consequence (F-C) Target and cumulative risk targets) and defense-in-depth adequacy.<br>• SR and NSRST SSCs do not degrade to an unacceptable level of reliability, availability, or condition during plant operations.<br>• SR and NSRST SSCs function reliably when challenged.<br>The D-RAP applies to reliability assurance activities that occur before initial fuel load and does not apply to reliability assurance activities conducted during the operations phase of the plant life cycle. | | SR and NSRST | Appendix B | SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs,” Item E, Reliability Assurance Program (Reference 8.0-7)<br>SRM on SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems (RTNSS) in Passive Plant Designs” (Reference 8.0-8) |


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**Table 8.0-1 Special Treatment Programs Supporting Design and Construction**
**(Sheet 4 of 4)**

| Program Title, Description, and Objectives | Section | SSC Applicability | Applicable 10 CFR 50 Regulation(s) | Program Reference(s) |
|---|---|---|---|---|
| **Inservice Testing Program, OM Code**<br>This program provides the requirements for Inservice Testing (IST) of SR and NSRST pumps, valves, and dynamic restraints to assess their operational ability to perform their specified functions. The IST program describes baseline testing, IST, examination, and monitoring needed to provide reasonable assurance of the operational readiness of the components. | | SR and NSRST | 50.34(b)(6)(iv) | Draft ASME OM-2, “Draft - Inservice Testing Requirements for Pumps, Valves, and Dynamic Restraints at Nuclear Facilities” (Reference 8.0-9) |
| **Post-Construction Inspection, Testing and Analysis Program** | 12 | | | |
| **Quality Assurance Program Description** | 8.1 | | | |

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**Table 8.0-2 Additional Special Treatment Programs for Operations**

| Topic | Section | Applicable Regulation(s) |
|---|---|---|
| Fire Protection Program | N/A | 10 CFR 50.48(a) |
| Leakage Program | N/A | 10 CFR 50.34(f)(2)(xxvi) |
| Maintenance Rule Program | N/A | 10 CFR 50.34(b)(6)(iv)<br>10 CFR 50.65(a) and (b)<br>10 CFR 50.69 |
| Emergency Preparedness Program | 11.3 | See PSAR Section |


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## 8.1 Quality Assurance

The Quality Assurance Program Description (QAPD) for the design and construction of Kemmerer Unit 1 is prepared in accordance with the requirements of 10 CFR 50 Appendix B and American Society of Mechanical Engineers (ASME) NQA-1-2015 (Reference 8.1-1) as endorsed by Regulatory Guide 1.28, Revision 5, “Quality Assurance Program Criteria (Design and Construction).” The QAPD considers the guidance in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 17.5. The QAPD establishes quality assurance criteria for all work during design and construction of Kemmerer Unit 1. Non-safety-related (SR) work is described in Section 19 of the QAPD. Specific program controls are applied to non-SR portions of nuclear reactor projects. The specific program controls, consistent with applicable sections of the QAPD, are applied to applicable items and targeted at those characteristics or critical attributes that render the structure, system, or component (SSC) a significant contributor to plant safety. The QAPD is submitted to NRC as the TerraPower Quality Assurance Topical Report (QATR), which is maintained as a separate document. The QATR is incorporated by reference (see Table 1.4-2).

### 8.1.1 Disposition of Limitations on NRC Approval of TerraPower QATR

Conformance statements for quality assurance (QA)-related Regulatory Guides are provided in the QAPD and in Section 1.4.1. As discussed in the NRC safety evaluation for revision 12 of the QAPD (Reference 8.1-2), the NRC identified two limitations associated with the TerraPower QAPD as follows:

Limitation No. 1

“TerraPower did not include a commitment to conform with NRC’s generic letter (GL) 85-06 and Regulatory Position 3.5 in RG 1.155 in the QAPD. The NRC staff acknowledges that with the substantial differences between TerraPower’s plant design and a light water reactor design, a direct commitment to these quality guidance positions may not be practical. However, TerraPower will have to justify in its application why these two quality guidance positions are not applicable to its advanced reactor design.”

Limitation No. 2

“TerraPower stated it did not include a commitment to conform with the applicable regulatory guidance in the following RGs: 1.26, 1.29, 1.54, 1.164, 1.189, and 1.231. The NRC staff acknowledges that with the substantial differences between TerraPower’s plant design and a light water reactor design, a direct commitment to these RGs may not be appropriate at this time. However, TerraPower will have to address in its application its conformance and/or exceptions to the applicable regulatory position guidance provided in the RGs.”


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The disposition of the limitations identified in the NRC SE of the QAPD are described below:

GL 85-06 and Regulatory Guide 1.155 Revision 0, Regulatory Position 3.5

Kemmerer Unit 1 will not conform to Regulatory Guide 1.155, Revision 0, “Station Blackout,” Regulatory Position 3.5 nor Generic Letter 85-06, “Quality Assurance Guidance For ATWS Equipment That Is Not Safety-Related.” Both refer to 10 CFR 50.62 and 10 CFR 50.63. These regulations are specific to light water reactor (LWR) designs and therefore are not applicable to the design utilized for Kemmerer Unit 1.

Kemmerer Unit 1 will reduce risk associated with the loss of offsite power event using the methodologies described in Nuclear Energy Institute (NEI) 18-04 (Reference 8.1-3) as endorsed by Regulatory Guide 1.233, Revision 0, “Guidance For A Technology-Inclusive, Risk-Informed, And Performance-Based Methodology To Inform The Licensing Basis And Content Of Applications For Licenses, Certifications, And Approvals For Non-Light-Water Reactors.”

Regulatory Guide 1.26, Revision 6

Kemmerer Unit 1 does not conform to Regulatory Guide 1.26, Revision 6, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.” Kemmerer Unit 1 uses NEI 18-04 (Reference 8.1-3) as endorsed by Regulatory Guide 1.233, Revision 0, as the basis for the plant design SSC classifications. Therefore, Regulatory Guide 1.26, Revision 6, is not applicable.

Regulatory Guide 1.29, Revision 6

Kemmerer Unit 1 partially conforms to Regulatory Guide 1.29, Revision 6, “Seismic Design Classification for Nuclear Power Plants,” as shown in Section 1.4.1. The general guidance and intent of Regulatory Guide 1.29 are applicable to Kemmerer Unit 1 with the exception that the overall requirements for “seismic Category I” SSCs are instead applied to SR SSCs as identified through the use of NEI 18-04. This exception is taken based on the use of NEI 18-04, which supports a broader gradation of seismic classifications, and the application to non-LWR technology. This exception is applied to the guidance in Regulatory Guide 1.29 in the following ways:

1. Section C.1: Provisions for seismic design of seismic Category I SSCs are instead applied to SR SSCs. In addition, the explicit list of SSCs in C.1.a through C.1.h are not taken to be applicable as they are specific to LWR technology. Instead SR SSCs required to be designed to withstand the effects of the SSE and remain functional are identified through the use and application of NEI 18-04.

2. Section C.2: Provisions for seismic interaction prevention are not applied to items from C.1.a through C.1.h and “occupants of the control room” but are instead applied to SR SSCs as identified through the use and application of NEI 18-04.


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3. Section C.3: Provisions for dynamic analysis requirements at the interface of seismic Category I and nonseismic Category 1 systems are instead applicable to the interface of SR to non-safety-related with special treatment (NSRST) and non-safety-related with no special treatment (NST) systems.

4. Section C.4: Provisions for quality requirements for seismic Category I SSCs are instead applicable to SR SSCs.

Regulatory Guide 1.54, Revision 3

Regulatory Guide 1.54, Revision 3, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” is not used for Kemmerer Unit 1. No Service Level I, II, or III protective coatings are used in the design. Conformance to Regulatory Positions within Regulatory Guide 1.54 will be identified if Service Level I, II, or III protective coatings are identified as the design progresses.

Regulatory Guide 1.164, Revision 0

Kemmerer Unit 1 fully conforms to Regulatory Guide 1.164, Revision 0, “Dedication of Commercial-Grade Items for Use in Nuclear Power Plants,” as shown in Section 1.4.1. Regulatory Guide 1.164 Revision 0 is applicable to Kemmerer Unit 1 as a programmatic issue.

Regulatory Guide 1.189, Revision 5

Kemmerer Unit 1 fully conforms to Regulatory Guide 1.189, Revision 5, “Fire Protection for Nuclear Power Plants,” as shown in Section 1.4.1. The Nuclear Island Fire Protection System described in Section 7.5.2 is designed to Regulatory Guide 1.189, Revision 5, which meets the applicable regulatory requirements of 10 CFR 50.48.

Regulatory Guide 1.231, Revision 0

Kemmerer Unit 1 fully conforms to Regulatory Guide 1.231, Revision 0, “Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants,” as shown in Section 1.4.1.

### 8.1.2 Responsibilities During Design and Construction

US SFR Owner, LLC (USO), a wholly owned subsidiary of TerraPower, is responsible for the establishment and execution of quality assurance program requirements. USO accomplishes this through agreement with TerraPower for TerraPower to be responsible for all activities for Kemmerer Unit 1 using the TerraPower QAPD. TerraPower may delegate the work of establishing and executing the quality assurance program, or any parts thereof, but retains responsibility for the quality assurance program.

TerraPower has delegated major portions of the work of establishing and executing the quality assurance program to its project partners. GE-Hitachi Nuclear Energy Americas, LLC is responsible for the design of a portion of the nuclear systems and procurement of certain key components using its QA program. Bechtel Power Company is responsible for the design of


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Nuclear Island structures, Nuclear Island electrical systems, radioactive waste treatment systems, and the Energy Island and is designated as the Engineering, Procurement, and Construction company for Kemmerer Unit 1.

TerraPower has oversight of project partners through a project quality execution plan and procurement documents that flow quality and regulatory requirements down to the project partners and require flow down to the project partner's sub-contractors. Additionally, TerraPower performs audits of the project partners to verify effective implementation of the partners' quality assurance programs.

### 8.1.3 Organization and Controls

The QAPD describes the TerraPower QA organization and responsibilities during design and construction of Kemmerer Unit 1. The USO project organization and responsibilities, including those related to QA during pre-operational testing through operation, are described in Section 11.1.

TerraPower has management controls and lines of communication for QA activities distributed among the project partners to ensure all aspects of the QA program are appropriately directed.

TerraPower utilizes a quality execution plan that flows down contractual and regulatory requirements in accordance with the QAPD and its respective program as appropriate for the associate criteria to ensure that each partners' QA program meets the minimum contractual requirements based on the scope of work being executed, prior to that organization performing work on, or in support of Kemmerer Unit 1. In cases where gaps are identified, TerraPower ensures mitigating actions are in place and approved by TerraPower to ensure full contractual QA compliance in accordance with the applicable regulatory requirements and the QAPD. In cases where TerraPower determines the applicable partner organization meets or exceeds the TerraPower QAPD, the partner continues to perform work in accordance with their established processes and implementing procedures consistent with the requirements of the TerraPower QAPD. TerraPower has the responsibility and authority to review and approve direct subcontractors and suppliers during design and construction of Kemmerer Unit 1. Each partners' supply chains are required to comply with the QA requirements as identified by the quality execution plan. These requirements are handled through the procurement process.

## References

8.1-1 ASME, "Quality Assurance Requirements for Nuclear Facility Applications," NQA-1-2015

8.1-2 U.S. NRC, "Safety Evaluation Regarding the Review of TerraPower, LLC's Quality Assurance Topical Report, TP-QA-PD-0001, 'TerraPower QA Program Description,' Revision 12 (EPID No. L-2020-TOP-0043/CAC No. 000431)," January 2022 (Agencywide Document Access and Management System Accession No. ML21263A236)

8.1-3 NEI, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," NEI 18-04, Revision 1


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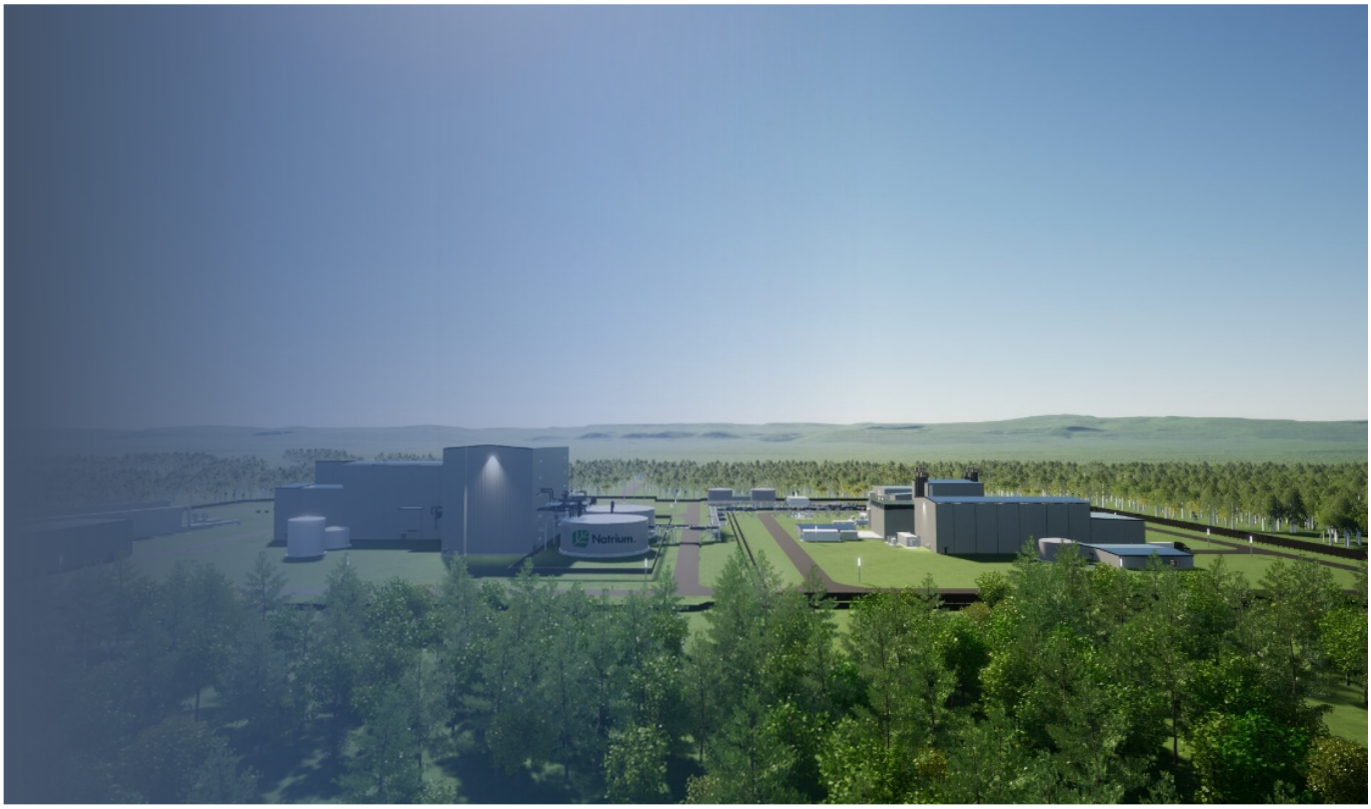

# Kemmerer Power Station Unit 1
# PSAR, Chapter 9

US SFR Owner, LLC, a wholly owned subsidiary of TerraPower, LLC

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## TABLE OF CONTENTS

**Chapter 9 Control of Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste . . . . . 9.1-1**

9.1 Liquid and Gaseous Effluents . . . . . 9.1-1

9.1.1 Source Terms . . . . . 9.1-1

9.1.2 Offsite Releases . . . . . 9.1-1

9.1.3 RWL Summary Description . . . . . 9.1-2

9.1.4 Design Basis . . . . . 9.1-5

9.2 Contamination Control . . . . . 9.2-1

9.3 Solid Radwaste Processing . . . . . 9.3-1

9.3.1 RWS Summary Description . . . . . 9.3-1

9.3.2 Design Basis . . . . . 9.3-2


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## LIST OF TABLES


Table 9.1-1 Annual Gaseous Effluent Release Activities . . . . . 9.1-7

Table 9.1-2 Liquid Source Term . . . . . 9.1-8

Table 9.1-3 Gaseous Effluent Doses to MEI . . . . . 9.1-9

Table 9.1-4 MEI External Doses with 10 CFR 50 Appendix I Criteria . . . . . 9.1-10

Table 9.1-5 Compliance of MEI to 40 CFR 190.10 Criteria . . . . . 9.1-11

Table 9.1-6 Holdup Times in Carbon Delay Beds . . . . . 9.1-12

Table 9.1-7 Codes and Standards Applicable to the RWL . . . . . 9.1-13

Table 9.1-8 RWL Interfaces. . . . . 9.1-14

Table 9.3-1 Codes and Standards Applicable to the RWS . . . . . 9.3-4



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## LIST OF FIGURES

Figure 9.1-1 RWL Simplified Diagram . . . . . 9.1-16

Figure 9.3-1 RWS System Architecture . . . . . 9.3-5





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# Chapter 9 Control of Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste

The control and management of liquid and gaseous effluents and processing of solid waste are described in Chapter 9. The Liquid Radwaste Processing System (RWL) along with liquid and gaseous effluents are described in Section 9.1. The contamination control program is described in Section 9.2. The Solid Radwaste Processing System (RWS) is described in Section 9.3. Section 9.1 and Section 9.3 provide preliminary system descriptions and address releases in compliance with 10 CFR 50.34a.

## 9.1 Liquid and Gaseous Effluents

### 9.1.1 Source Terms

The primary source of gaseous radiological effluent is from the Sodium Cover Gas System, which contains xenon, krypton, activated argon, and hydrogen. Sodium Cover Gas System effluent is processed by the Gaseous Radwaste Processing System (RWG). Sources of gaseous effluent include the RWG, Nuclear Island (NI) Heating, Ventilation, and Air Conditioning System, Reactor Air Cooling System, and tritium. Gaseous normal operating source terms are listed in Table 9.1-1.

During normal operation spent fuel assemblies are the primary source of radioisotopes in the RWL. Activity throughout the RWL is due to cesium-137 and cesium-134. All other isotopes account for 0.2% of the total activity. Liquid normal operating source terms are listed in Table 9.1-2.

### 9.1.2 Offsite Releases

#### 9.1.2.1 Liquid Effluents

Kemmerer Unit 1 is a zero radioactive liquid waste discharge plant. The RWL processes liquid radioactive waste through filtration, demineralization, and ion exchange to remove or reduce radioactivity as described in Section 9.1.3. Any remaining liquid radioactive waste is evaporated, and water vapor is released to the atmosphere by means of the NI Heating, Ventilation, and Air Conditioning System. Solids remaining after evaporation are treated as solid waste as described in Section 9.3.

The evaluation of a potential accidental release of radioactive liquid following a postulated failure of an RWL tank is provided in Section 2.5.3.2.

#### 9.1.2.2 Gaseous Effluents

The RWG collects, processes, and monitors gaseous waste streams as described in Section 7.4.1.

Airborne doses are calculated using the source terms provided in Table 9.1-1. The RWG charcoal bed delay times are listed in Table 9.1-6. Kemmerer Unit 1 is designed such that there is zero radioactive liquid waste discharge, so any dose consequences to the public from normal operation are from gaseous release.


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The radiation doses to the maximally exposed individual (MEI) at the exclusion area boundary are calculated using the GASPAR II code. The gaseous effluent doses to the MEI are tabulated in Table 9.1-3. The effluent dose objectives to members of the public are informed by the 10 Code of Federal Regulations (CFR) 50 Appendix I design objectives to maintain dose ALARA. The objectives set forth for Kemmerer Unit 1 comply with 10 CFR 20.1301 and 10 CFR 20.1302 by meeting the objectives tabulated in Table 9.1-4, and compliance of the MEI to 40 CFR 190.10 criteria is shown in Table 9.1-5. The atmospheric dispersion factor data used and distances for the MEI are described in Section 2.4.4.

### 9.1.3 RWL Summary Description

The RWL is classified as non-safety related with no special treatment and is designed to collect, segregate, process, sample, and monitor the non-sodium and non-salt liquid radioactive waste for reuse or evaporation. The failure or degradation of the RWL does not impact the execution of safety-related operations. The RWL tanks receive and store radioactive or potentially radioactive liquid waste during normal operation and during anticipated operational occurrences. Liquid waste is processed through filtration, reverse osmosis, and ion exchange. Monitoring and sampling of the liquid waste is performed prior to reuse or evaporation.

The RWL is located at grade in the Fuel Handling Building within the NI as described in Section 7.8.2 and is subject to the environmental conditions therein. The RWL also has collection components in the Fuel Auxiliary Building and Reactor Building. RWL components that may contain radioactivity are in controlled shielded access areas that maintain appropriate radiological conditions.

The RWL supports contamination control by providing a barrier to retain radionuclides within the system. The design of the system minimizes the potential for contamination and supports decommissioning, consistent with the requirements of 10 CFR 20.1406 as described in Section 9.2.

The RWL is monitored and controlled through the Auxiliary Monitoring and Control System as discussed in Section 7.6.2. The Auxiliary Monitoring and Control System provides the capability for both automatic and manual (operator-initiated) system control functions.

The manual mode is used to verify equipment or device status before plant startup, to perform necessary testing or maintenance activities, or to start or stop equipment and devices that do not require continuous control and monitoring by the plant control system.

A summary of non-safety related with no special treatment system interfaces is provided in Table 9.1-8.

RWL equipment and components are designed, constructed, and tested to the requirements set forth in the applicable codes and standards listed in Regulatory Guide 1.143, “Design Guidance for Radioactive Waste Management Structures, Systems, and Components Install in Light-Water-Cooled Nuclear Power Plants,” Revision 2.

The RWL consists of the following major components.


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Fuel Auxiliary Building RWL Tank

One atmospheric pressure tank collects sampling and chemistry sink drains, personnel decontamination laundry, decontamination hand washing, and decontamination shower drains in the Fuel Auxiliary Building (FAB). When waste is collected in the tank, the water is pumped and filtered to the waste holdup tank. The tank is 750 gallons and holds one month of expected influent. The tank is equipped with level measuring instrumentation. Any equipment leakage in the FAB RWL tank enclosure is collected in the FAB RWL sump and is routed to the waste holdup tank.

Sumps

Collection sumps and associated sump pumps are used to transfer wastewater from source locations to the RWL waste holdup tanks. Sumps are shown on Figure 9.1-1.

Oil Water Separator

One oil water separator is used to separate oil-based liquids that may enter the RWL process stream. The oil water separator is located on the Fuel Handling Building sump process line that is drained to the RWL waste holdup tank. The oil that is discharged is collected manually and sampled in the NI sampling lab. The oil is disposed of through the RWS as discussed in Section 9.3.

Filters

One carbon filter removes impurities prior to collection in the waste holdup tank. The carbon filter is located downstream of the FAB process line that drains to the waste holdup tank.

Two preconditioning filters are used to capture large particles in the process lines. One preconditioning filter is installed on the Fuel Handling Building sump process line and the other is installed downstream of the waste holdup tank pumps prior to the tubular ultrafilter skid.

One tubular ultrafilter skid is installed downstream of the waste holdup tank. It is used to remove particulate and organic materials from the process liquid.

Pressure measuring instrumentation is provided to give remote indication of pressure drops across the carbon filter, the preconditioning filters, and the tubular ultrafilter skid.

Waste Holdup Tanks

Two, 6,000-gallon, atmospheric pressure tanks collect liquid radioactive waste prior to processing. Each tank holds one week of influent. Liquid wastes that are collected into the waste holdup tanks are processed and passed through a preconditioning filter, a tubular ultrafilter, a 2-pass reverse osmosis filter, and a demineralizer before entering a monitor tank. Processing waste from the waste holdup tanks to the monitor tanks is a batch process.

Two waste holdup tanks allow for redundancy in the main receiving tank for maintenance and surge capacity. Each tank is equipped with mixing capabilities using internal sparge nozzles.


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The sources of liquid waste sent to the waste holdup tank are shown on Figure 9.1-1.

Two-Pass Reverse Osmosis

One reverse osmosis skid located downstream of the waste holdup tanks removes soluble radioactive isotopes and particulates from the process liquid. The skid is equipped with instrumentation providing remote indication of differential pressure.

Concentrate Holding Tank

One 100-gallon atmospheric pressure tank collects solids from the evaporator skid and the concentrate from the reverse osmosis skid. The tank is equipped with mixing capabilities using internal sparge nozzles, which mix the solids within the tank and prevent settling before processing. The tank is equipped with level measuring instrumentation.

Spent Resin Storage Tanks

Two atmospheric pressure tanks are used to provide collection points for spent resins and to allow for delay of short-lived radionuclides before disposal. Each tank is 1,700 gallons and holds one year of expected influent.

The RWL receives spent resins into the spent resin storage tanks from the demineralizer and the Water Pool Fuel Handling System resin tanks. Spent resin in the storage tanks is fluffed with demineralized water and flushed to the RWS dewatering skid. Liquid returned from the RWS dewatering skid is directed to the waste holdup tank as discussed in Section 9.3.

Each tank is designed as a sloped bottom tank to prevent plugging and is equipped with mixing capabilities using internal sparge nozzles to fully loosen the resin during the recirculation and flushing process.

Any leakage from the spent resin storage tanks is collected in the spent resin sump and is routed to the redundant spent resin storage tank.

Demineralizer

One mixed bed demineralizer downstream of the waste holdup tank performs ion exchange, which removes radioactive isotopes from the process liquid before entering the monitor tank. Pressure measuring instrumentation is installed to give remote indication of differential pressure across the demineralizer.

Monitor Tank

Two atmospheric pressure tanks are used to monitor the liquid processed from the waste holdup tanks prior to reuse or evaporation. Each tank is 6,000 gallons and holds one week of expected influent.


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When a monitor tank is full, it is manually aligned for discharge. The contents of the monitor tank are sampled locally and analyzed in the NI sampling lab. The liquid waste in the tank is directed to the Spent Fuel Pool for reuse, evaporated, or upon detection of high radiation reprocessed through the waste holdup tanks.

Pumps

The FAB RWL tank pump transfers liquid waste from the FAB RWL tank to the waste holdup tank.

A single waste holdup tank pump transfers liquid waste from a holdup tank to the monitor tanks. A second pump is installed for redundancy.

One concentrate slurry tank pump transfers slurry from the concentrate holding tank to the RWS.

Two slurry pumps transfer resins from the spent resin storage tank to the RWS.

RWL pumps are shut down either manually via remote operation, on low waste holdup tank level, on low monitor tank level, or upon non-diagnostic alarms from the pumps and skid-mounted equipment.

Evaporator Skid

One evaporator skid reduces the volume of the liquid in the monitor tanks. Clean water is removed as vapor and the non-volatile solids left behind are transferred to the concentrate holding tank.

Process radiation monitoring instrumentation is provided to monitor liquid for radioactivity prior to reuse, evaporation, or recycling. A description of radiation monitors is provided in Section 7.6.6.

### 9.1.4 Design Basis

The RWL uses engineering controls and procedures in compliance with 10 CFR 20.1101(b) to achieve doses to the public as low as reasonable achievable. Similarly, RWL controls and procedures, in accordance with 10 CFR 20.1406 and Regulatory Guide 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” Revision 0, minimize contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste.

Principal Design Criteria (PDC) are identified in Section 5.3. The RWL complies to the following PDC:

Consistent with PDC 60, *Control of Releases of Radioactive Materials to the Environment*, the RWL is designed to collect, segregate, process, sample, and monitor the non-sodium and non-salt liquid radioactive waste for reuse or evaporation. The RWL tanks receive and store radioactive or potentially radioactive liquid waste. Monitoring and sampling of the liquid waste is performed prior to reuse or evaporation.


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Consistent with PDC 61, *Fuel Storage and Handling and Radioactivity Control*, the RWL receives and stores radioactive liquid waste. The RWL includes process filtering systems and the storage tanks provide suitable shielding where appropriate. The spent resin storage tanks are provided with an enclosure that satisfies shielding requirements.

Consistent with PDC 63, *Monitoring Fuel and Waste Storage*, the RWL is designed to support process radiation monitoring of the monitor tank common discharge line. The RMS process radiation monitor is used to detect conditions that may result in excessive radiation levels, which would allow initiation of appropriate actions.

Regulatory Guide 1.143, Revision 2, *Design Guidance for Radioactive Waste Management Structures, Systems, and Components Installed in Light-Water-Cooled Nuclear Power Plants*

Partial Conformance. The system is designed in conformance with Section C.1 of Regulatory Guide 1.143. The methodologies described in Section 5.1 and Section 6.1.3 replace the methodologies of Regulatory Positions C.5 and C.6 respectively. The RWL meets or exceeds the guidance for RW-IIc classified systems or components. Materials used in the construction of the RWL meet the requirements of Regulatory Position C.1.1.2.

The codes and standards applicable to the RWL are listed in Table 9.1-7.

**References**


9.1-1 ANSI/ANS 55.6, *Liquid Radioactive Waste Processing System for Light Water Reactor Plants*, 2007.

9.1-2 ASME B16.5, *Pipe Flanges and Flanged Fittings*, 2017.

9.1-3 ASME B16.34, *Valves - Flanged, Threaded, and Welding End*, 2020.

9.1-4 ASME B31.3, *Process Piping*, 2020.

9.1-5 API 610, *Centrifugal Pumps for Petroleum, Petrochemical, and Natural Gas Industries*, 2021.

9.1-6 API 620, *Design and Construction of Large, Welded, Low-Pressure Storage Tank*, 2013.

9.1-7 API 650, *Welded Steel Tanks for Oil Storage*, 2020.

9.1-8 AWS D1.1, *Structural Welding Code*, 2015.

9.1-9 ASME Boiler and Pressure Vessel Code (BPVC) Section II, *Materials*, 2019.

9.1-10 ASME BPVC Section V, *Nondestructive Examination*, 2021.

9.1-11 ASME BPVC Section IX, *Welding, Brazing, and Fusing Qualifications*, 2019.

9.1-12 National Electrical Manufactures Association (NEMA) ICS-6, *Industrial Control and Systems*, 2016.



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**Table 9.1-1 Annual Gaseous Effluent Release Activities**

| | Radionuclide | Type 1 Fuel Release (Curies/ year) | Source |
|---|---|---|---|
| **Energy Island** | **H-3** | 903 | Cold Salt Tank Cover Gas |
| | | $1.59 \times 10^{-2}$ | Steam Generator |
| **Nuclear Island** | **H-3** | $3.20 \times 10^{-1}$ | Sodium-Air Heat Exchanger |
| | | $1.96 \times 10^{-1}$ | Intermediate Heat Transport System/ Nuclear Island Heating, Ventilation, and Air Conditioning System |
| | | 1.21 | Primary Heat Transport System/Nuclear Island Heating, Ventilation, and Air Conditioning System |
| | | $7.21 \times 10^{-3}$ | Primary Cover Gas |
| | | $1.89 \times 10^{-4}$ | Secondary Cover Gas |
| | | 4.91 | Gaseous Radwaste Processing System |
| | **N-16** | $3.57 \times 10^{-1}$ | Reactor Air Cooling System |
| | **Ar-41** | 78.1 | Reactor Air Cooling System |
| | **Kr-85** | 37.0 | Gaseous Radwaste Processing |
| | | 55.2 | Spent Fuel Pool/Nuclear Island Heating, Ventilation, and Air Conditioning System |
| | **Rb-88** | $6.24 \times 10^{-2}$ | Gaseous Radwaste Processing |
| | **Xe-131m** | 32.9 | Gaseous Radwaste Processing |
| | **Xe-133** | 9.67 | Gaseous Radwaste Processing |
| | **Cs-137** | $1.09 \times 10^{-4}$ | Gaseous Radwaste Processing |
| | **Ba-137m** | $1.03 \times 10^{-4}$ | Gaseous Radwaste Processing |


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**Table 9.1-2 Liquid Source Term**

| Location | Volume (gallons) | Yearly Activity (Curies)[3] | Concentration (Curies/cubic centimeter)[3] | Processes Per Year |
|---|---|---|---|---|
| Spent Fuel Pool | 574,000 | 1.09 | $5.00 \times 10^{-4}$ | N/A |
| Spent Fuel Pool Demineralizers | 750 | 6.24 | 8.79 | 4.0 |
| Spent Resin Tank[2] | 750 | 9.98 | 4.39 | 4.0 |
| Monitoring Tank[1] | 4,000 | $3.79 \times 10^{-5}$ | $3.13 \times 10^{-6}$ | 3.7 |
| Laundry Waste Tank[1] | 300 | $4.07 \times 10^{-4}$ | $4.48 \times 10^{-4}$ | 12.9 |
| Chemical Drain Tank[1] | 100 | $3.03 \times 10^{-4}$ | $1.00 \times 10^{-3}$ | 12.5 |
| Equipment Decontamination Tank[1] | 500 | $3.03 \times 10^{-4}$ | $1.00 \times 10^{-3}$ | 6.3 |
| Waste Holdup Tank[1] | 4,000 | $3.50 \times 10^{-2}$ | $5.87 \times 10^{-4}$ | 4.9 |
| Pre-Conditioning Filter | N/A | $1.75 \times 10^{-2}$ | N/A | 1.0 |
| Tubular Ultrafilter | N/A | $1.40 \times 10^{-2}$ | N/A | 1.0 |
| 2-pass Reverse Osmosis Filter[1] | 800 | $5.69 \times 10^{-4}$ | $1.88 \times 10^{-4}$ | 4.9 |
| Demineralizers | N/A | $5.60 \times 10^{-4}$ | N/A | 1.0 |
| **Totals** | **585,200** | **17.4** | **N/A** | **N/A** |

Notes:

1. This location is normally processed once it has reached 80% of the maximum capacity (volume) presented in this table. The associated activity and concentration values are based on the maximum capacity.
2. This location is processed quarterly due to radiation levels, instead of tank capacity.
3. Activity throughout the Spent Fuel Pool and RWL is attributed, as a percentage, to the following:

   Cs-137: 90.8%, Cs-134: 9.0%, and all other radioactive isotopes: <0.2%.


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**Table 9.1-3 Gaseous Effluent Doses to MEI**

| | | Dose per Unit (millirem/year)[1] | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| **Location** | | **Total Body** | **GI-Tract** | **Bone** | **Liver** | **Kidney** | **Thyroid** | **Lung** | **Skin[4]** |
| **Site exclusion area boundary[2]** | Adult | 4.70 | 4.70 | $7.96 \times 10^{-1}$ | 4.70 | 4.70 | 4.70 | 4.71 | 1.63 |
| | Teen | 4.73 | 4.73 | $7.96 \times 10^{-1}$ | 4.73 | 4.73 | 4.73 | 4.74 | 1.63 |
| | Child | 4.27 | 4.27 | $7.96 \times 10^{-1}$ | 4.27 | 4.27 | 4.27 | 4.28 | 1.63 |
| | Infant | 2.80 | 2.80 | $7.96 \times 10^{-1}$ | 2.80 | 2.80 | 2.80 | 2.80 | 1.63 |
| **Sodium Test and Fill Facility[2]** | Adult | $5.57 \times 10^{-1}$ | $5.57 \times 10^{-1}$ | $4.28 \times 10^{-1}$ | $5.57 \times 10^{-1}$ | $5.57 \times 10^{-1}$ | $5.57 \times 10^{-1}$ | $5.60 \times 10^{-1}$ | $8.75 \times 10^{-1}$ |
| | Teen | $5.58 \times 10^{-1}$ | $5.58 \times 10^{-1}$ | $4.28 \times 10^{-1}$ | $5.58 \times 10^{-1}$ | $5.58 \times 10^{-1}$ | $5.58 \times 10^{-1}$ | $5.61 \times 10^{-1}$ | $8.75 \times 10^{-1}$ |
| | Child | $5.43 \times 10^{-1}$ | $5.43 \times 10^{-1}$ | $4.28 \times 10^{-1}$ | $5.43 \times 10^{-1}$ | $5.43 \times 10^{-1}$ | $5.43 \times 10^{-1}$ | $5.46 \times 10^{-1}$ | $8.75 \times 10^{-1}$ |
| | Infant | $4.94 \times 10^{-1}$ | $4.94 \times 10^{-1}$ | $4.28 \times 10^{-1}$ | $4.94 \times 10^{-1}$ | $4.94 \times 10^{-1}$ | $4.94 \times 10^{-1}$ | $4.97 \times 10^{-1}$ | $8.75 \times 10^{-1}$ |
| **Nearest Property[3]** | Adult | 1.36 | 1.36 | $1.82 \times 10^{-2}$ | 1.36 | 1.36 | 1.36 | 1.36 | $3.75 \times 10^{-2}$ |
| | Teen | $9.00 \times 10^{-1}$ | $8.99 \times 10^{-1}$ | $1.83 \times 10^{-2}$ | $9.00 \times 10^{-1}$ | $9.00 \times 10^{-1}$ | $8.99 \times 10^{-1}$ | $8.99 \times 10^{-1}$ | $3.75 \times 10^{-2}$ |
| | Child | 1.12 | 1.12 | $1.92 \times 10^{-2}$ | 1.12 | 1.12 | 1.12 | 1.12 | $3.75 \times 10^{-2}$ |
| | Infant | $2.50 \times 10^{-1}$ | $2.50 \times 10^{-1}$ | $1.91 \times 10^{-2}$ | $2.52 \times 10^{-1}$ | $2.50 \times 10^{-1}$ | $2.50 \times 10^{-1}$ | $2.50 \times 10^{-1}$ | $3.75 \times 10^{-2}$ |

Notes:

1. Values are summed from NI and Energy Island release results.
2. Values at this location result from external (plume and ground) and inhalation pathways.
3. Values at this location result from external (plume and ground), inhalation, and ingestion (vegetable, dairy, and meat) pathways. Dairy includes both cow milk and goat milk. It is assumed that the nearest meat animal is located at the exclusion area boundary but consumed at the nearest property.
4. It is assumed that there is no skin dose from inhalation and ingestion pathways.


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**Table 9.1-4 MEI External Doses with 10 CFR 50 Appendix I Criteria**

| Type of Dose | Maximally Exposed Individuals Doses with 10 Code of Federal Regulations 50, Appendix I Criteria | |
|---|---|---|
| | Site | Limit |
| **Gamma Air (millirad)** | 1.19 | 10 |
| **Beta Air (millirad)** | 0.979 | 20 |
| **Total Body (millirem)** | 0.796 | 5 |
| **Skin (millirem)** | 1.63 | 15 |
| **Max Organ from Iodine and Particulates (millirem)** | 4.74 | 15 |

Notes:

1. Annual doses are taken from Table 9.1-3 and occur at the exclusion area boundary.
2. The max organ dose conservatively includes all isotopes and pathways.
3. Site dose is found by summing the dose from the reactor and the dose from the cold salt storage tank.


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**Table 9.1-5 Compliance of MEI to 40 CFR 190.10 Criteria**

| | Dose (millirem/year) | | | |
|---|---|---|---|---|
| | **Gaseous** | **Direct** | **Total** | **Limit** |
| **Total Body** | 4.73 | 1.00 | 5.73 | 25 |
| **Thyroid** | 4.73 | 0.00 | 4.73 | 75 |
| **Other Organ - Lung** | 4.74 | 0.00 | 4.74 | 25 |

Notes:

1. Annual doses are taken from Table 9.1-3.
2. Direct dose is assumed to be 1 millirem/year.
3. Doses occur at site boundary


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**Table 9.1-6 Holdup Times in Carbon Delay Beds**

| Element[1] | Charcoal Bed Holdup Time at 12 standard cubic feet per minute (4 beds)[4] |
|---|---|
| Ar | 1.03 days |
| Kr[2] | 5.13 days |
| Xe[3] | 102.58 days |

Notes:

1. Holdup times are applicable to all isotopes of the listed element
2. Dynamic adsorption coefficient base value for krypton is 50 cubic centimeters (standard temperature and pressure) per gram at 70 °F
3. Dynamic adsorption coefficient base value for xenon is 1000 cubic centimeters (standard temperature and pressure) per gram at 70 °F
4. The charcoal beds are operated 70 °F


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**Table 9.1-7 Codes and Standards Applicable to the RWL**

| Standard Number | Standard Version | Title |
|---|---|---|
| ANSI/ANS 55.6 | 2007 | Liquid Radioactive Waste Processing System for Light Water Reactor Plants |
| ASME B16.5 | 2017 | Pipe Flanges and Flanged Fittings |
| ASME B16.34 | 2020 | Valves - Flanged, Threaded, and Welding End |
| ASME B31.3 | 2020 | Process Piping |
| API 610 | 2021 | Centrifugal Pumps for Petroleum, Petrochemical, and Natural Gas Industries |
| API 620 | 2013 | Design and Construction of Large, Welded, Low-Pressure Storage Tank |
| API 650 | 2020 | Welded Steel Tanks for Oil Storage |
| AWS D1.1 | 2015 | Structural Welding Code |
| BPVC Section II | 2019 | Materials |
| BPVC Section V | 2021 | Nondestructive Examination |
| BPVC Section IX | 2019 | Welding, Brazing, and Fusing Qualifications |
| NEMA ICS-6 | 2016 | Industrial Control and Systems |


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**Table 9.1-8 RWL Interfaces**
**(Sheet 1 of 2)**

| System Name | Interface |
|---|---|
| Auxiliary Monitoring and Control System | The Liquid Radwaste Processing system receives control signals from the Auxiliary Monitoring and Control system. |
| Fuel Auxiliary Building | The Liquid Radwaste Processing system collects personnel and equipment decontamination, laboratory drainage from the Fuel Auxiliary Building, as well as any leakage that collects in the Fuel Auxiliary Building sump. |
| Fuel Auxiliary Building | Select Liquid Radwaste Processing equipment is located in a below-grade location in the Fuel Auxiliary Building. |
| Fuel Handling Building | The Liquid Radwaste Processing system has equipment located in a dedicated room in the Fuel Handling Building. |
| Fuel Handling Building | The Liquid Radwaste Processing system collects drainage from select floor drains and sumps in the Fuel Handling Building. |
| Water Pool Fuel Handling System | The Liquid Radwaste Processing system provides treated makeup water to the spent fuel pool. |
| Water Pool Fuel Handling System | The Liquid Radwaste Processing system receives discharge from the Water Pool Fuel Handling System system due to the humidifier pressure relief lifting during a high pressure event. |
| Water Pool Fuel Handling System | The Liquid Radwaste Processing system collects Spent Fuel Pool cooling and cleanup system leakage via a local sump. |
| Fuel Pool Purification System | The Liquid Radwaste Processing system receives discharged resin from the Fuel Pool Purification System for treatment and disposal. |
| Fuel Pool Purification System | The Liquid Radwaste Processing system receives effluent from the Fuel Pool Purification System leakage sump. |
| Nuclear Island Alternating Current Electrical Power Low Voltage System | The Liquid Radwaste Processing system receives low voltage power from the Nuclear Island Alternating Current Electrical Power Low Voltage System system. |
| Nuclear Island Heating, Ventilation, and Air Conditioning System | The Liquid Radwaste Processing system receives condensate water from Spent Fuel Pool system evaporation captured by the Nuclear Island Heating, Ventilation, and Air Conditioning System dehumidifiers. |
| Nuclear Island Heating, Ventilation, and Air Conditioning System | The Nuclear Island Heating, Ventilation, and Air Conditioning System system provides filtration for gaseous effluents produced by the Liquid Radwaste Processing system. |
| NWS | The Liquid Radwaste Processing system receives one time use demineralized water for the flushing of the Liquid Radwaste Processing spent resin storage tanks, the Liquid Radwaste Processing process radiation monitor, and the Liquid Radwaste Processing concentrate holding tank. |
| Pool Immersion Cell | The Liquid Radwaste Processing system receives liquid effluent from the Pool Immersion Cell system during a pressure transient. |


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**Table 9.1-8 RWL Interfaces**
**(Sheet 2 of 2)**

| System Name | Interface |
|---|---|
| Reactor Air Cooling System | The Liquid Radwaste Processing system receives rainwater and snowmelt that is collected in the Reactor Air Cooling System towers. |
| RMS | Liquid Radwaste Processing system effluents are monitored via the RMS. |
| Solid Radwaste Processing | The Solid Radwaste Processing system receives solid wastes from the Liquid Radwaste Processing system. |
| Solid Radwaste Processing | The Liquid Radwaste Processing system receives waste water from the Solid Radwaste Processing system following resin dewatering activities. |
| Reactor Building | Select Liquid Radwaste Processing components are located in a below-grade location in the Reactor Building. |
| Spent Fuel Pool | The Liquid Radwaste Processing system receives and treats leakage from the Spent Fuel Pool liner. |


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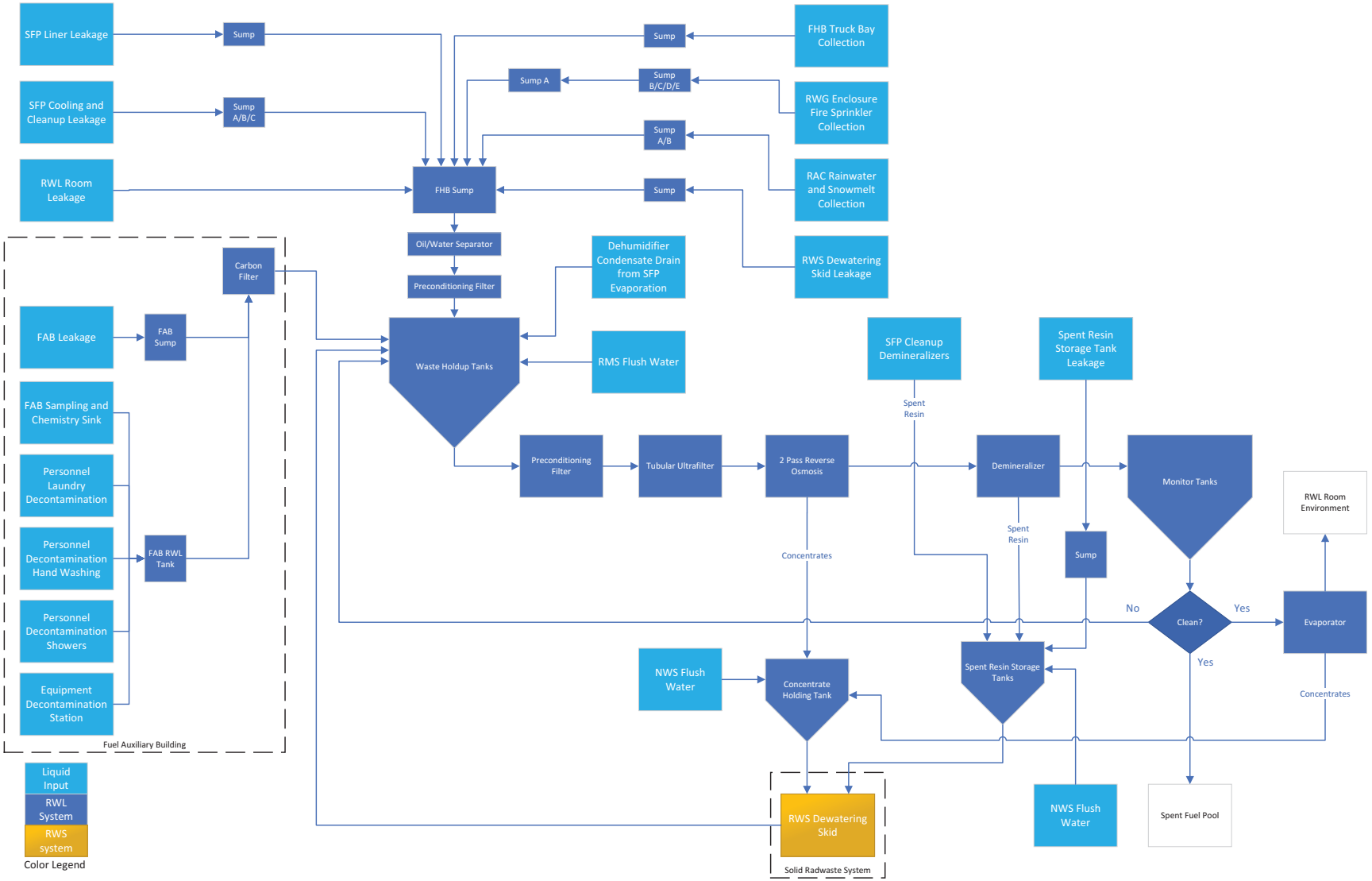
## Figure 9.1-1 RWL Simplified Diagram





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## 9.2 Contamination Control

Design features and procedures for operation minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste consistent with the requirements in 10 Code of Federal Regulations 20.1406. Structures, systems, or components with the potential to contain or handle radiological materials include design considerations to minimize contamination and control and collect spillage. The principles of contamination control applied in the design are prevention of unintended releases, early detection of unintended releases of radioactive contamination, and prompt assessment to support a timely and appropriate response.

Design features, programmatic elements, and procedures consider the guidance in Regulatory Guide 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” Revision 0, and examples of considerations and administrative controls include:

- Minimization of leaks and spills and provisions of containment
- Adequate leak detection capability
- Avoidance of the release of contamination from undetected leaks
- Measures for reducing the need to decontaminate structures, systems, or components
- Periodic review of operational practices

A description of the design features and procedures for the control of radioactive contamination for the facility, including consideration of Regulatory Guide 4.21, will be provided at the operating license stage consistent with 10 Code of Federal Regulations 50.34(b)(3).


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# 9.3 Solid Radwaste Processing

## 9.3.1 RWS Summary Description

The Solid Radwaste Processing System (RWS) for Kemmerer Unit 1 is designed to process both wet and dry solid waste resultant from plant operation during normal operation and anticipated operational occurrences. Solid wastes are processed by means of surveying, classifying, sorting, decontaminating, compacting, and dewatering for storage and eventual shipment to licensed off-site facilities in compliance with 10 CFR 61. The RWS is not intended to process large radioactive waste materials, such as core assemblies, spent fuel, or large failed equipment, nor is it intended to provide long-term storage of waste. Large failed equipment is decontaminated in-place to the extent practical and is packaged for shipment, processing, and treatment.

The RWS is located within the Fuel Handling Building and consists of three key design features: a dewatering skid, a compaction skid, and a storage area. Figure 9.3-1 provides an overview of the inputs and equipment of RWS. The storage area is dedicated to the storage of processed radwaste. The Fuel Handling Building is further described in Section 7.8.2.

Wet solid waste, dry solid waste, and miscellaneous wastes are processed by the RWS. The Liquid Radwaste Processing System resin sluicing and transfer activities are discussed in Section 9.1. Wet solid wastes are dewatered by the RWS prior to storage. Spent carbon is transferred directly to RWS waste containers to avoid mixing with resin.

Dry solid wastes generally consist of ventilation filters, rags, paper, plastic, rubber, scrap wood, glass, concrete, metal, failed equipment parts, and tools. Dry solid wastes are monitored and classified in accordance with 10 CFR 61.55, compacted and packaged prior to shipment to a licensed waste disposal facility for offsite management.

Oil sludge is collected and packaged separate from other wastes. Low-contamination wastes, such as used personnel protective equipment and maintenance parts, are collected in bags or boxes. These wastes are typically stored in 55-gallon drums or other approved containers in the low-activity storage area until transported off-site for processing and disposal.

Mixed wastes are not processed by the RWS, but the facility has the means to package and ship mixed waste material to qualified off-site processors.

The dewatering skid dries and stabilizes liquid-bearing waste to meet the requirements in 10 CFR 61 for packaging and disposal. Once a high integrity container has been filled and dewatered, it is removed from the skid and placed in the storage area to await shipment to a disposal facility. The compaction skid provides volume reduction for certain dry solid radwastes. It is designed to accommodate typical waste containers (e.g., 55-gallon drums). Exhaust from the compaction skid is locally filtered with a high efficiency particulate air filter to remove airborne contamination prior to exhaust.

Monitoring of waste activities within the RWS includes process parameters as well as local area radiation monitors. Radiation monitoring is provided by the Radiation Monitoring System as described in Section 7.6.6.


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### 9.3.2 Design Basis

The RWS collects and processes both wet solid wastes and dry solid radwastes from normal plant operation. Wet wastes are stabilized through dewatering and dry radwaste is compacted prior to storage for offsite disposal. Waste concentrations are held to the requirements of 10 CFR 61.55. Packaging is adherent to 10 CFR 71.43. Monitoring and the selection of the high integrity container ensures adherence to external radiation levels imposed by 10 CFR 71.47. Prior to shipment, it is ensured that packaging is adherent to 10 CFR 71.87 and U.S. Department of Transportation regulations. Waste is labeled in accordance with 10 CFR 20.1904 and documented in accordance with 10 CFR 20.2006 and 10 CFR 20 Appendix G.

The RWS is classified as non-safety related with no special treatment. It provides no credited safety function during a plant transient or accident, and its failure or abnormal operation does not affect the conduct of safety-related operations.

The codes and standards applicable to design are provided in Table 9.3-1.

The RWS is designed to maintain occupational dose as low as is reasonably achievable. The RWS is located in a shielded area separated from the rest of the facility. The system components are also shielded to meet the radiation zones selected for the area based on its occupancy. Local area radiation monitoring and airborne radioactive material monitoring for the compaction skid is provided.The design of the system minimizes the potential for contamination and supports decommissioning, consistent with the requirements of 10 CFR 20.1406 as described in Section 9.2.

Principal Design Criteria (PDC) are identified in Section 5.3. The RWS complies to the following PDC:

Consistent with PDC 60, *Control of Releases of Radioactive Materials to the Environment,* the RWS is designed to collect, segregate, process, sample, and monitor solid waste resulting from normal operation and anticipated operational occurrences. The RWS design provides leakage control and confinement of radionuclides against uncontrolled release.

Consistent with PDC 61, *Fuel Storage and Handling and Radioactivity Control,* the RWS receives, stores, and handles radioactive waste. The RWS design assures the performance of the solid radioactive waste confinement barrier under normal operating conditions. The RWS is designed to provide suitable shielding for radiation protection where appropriate. The RWS contains process filtering system and the RWS enclosure is constructed to satisfy shielding requirements. The RWS has processing and storage capacity to prevent limitations of plant operations (e.g., during unfavorable site environmental conditions leading to a delay in scheduled offsite shipments).

Consistent with PDC 63, *Monitoring Fuel and Waste Storage*, the RWS is provided with a radiation monitor to detect abnormalities during compaction activities. The radiation monitor is part of the Radiation Monitoring System, which is discussed in Section 7.6.6.

Regulatory Guide 1.143, Revision 2, *Design Guidance for Radioactive Waste Management Structures, Systems, and Components Installed in Light-Water-Cooled Nuclear Power Plants*


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Partial Conformance. The system is designed in conformance with Section C.3 of Regulatory Guide 1.143. The methodologies described in Section 5.1 and Section 6.1.3 replace the methodologies of Regulatory Positions C.5 and C.6 respectively. The RWS meets or exceeds the guidance for RW-IIc classified systems or components. Materials used in the construction of the RWS meet the requirements of Regulatory Position C.3.2. The RWS is not required to maintain operational functionality following a seismic event. A process control plan consistent with Nuclear Energy Institute (NEI) 07-10A, “Generic FSAR Template Guidance for Process Control Program (PCP),” to meet 10 CFR 61 will be provided at the Operating License stage.

## References


9.3-1 ANSI/ANS 55.1, *Solid Radioactive Waste Processing System for Light-Water-Cooled Reactor Plants*, 2021.

9.3-2 API 610, *Centrifugal Pumps for Petroleum, Petrochemical, and Natural Gas Industries*, 2021.

9.3-3 ASME B31.3 *Process Piping*, 2020.

9.3-4 ASME BPVC Section II, *Materials*, 2019.

9.3-5 ASME BPVC Section IX, *Welding, Brazing, and Fusing Qualifications*, 2019.

9.3-6 ASCE 7, *Minimum Design Loads and Associated Criteria for Buildings and Other Structures with Supplement 1*, 2016.




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**Table 9.3-1 Codes and Standards Applicable to the RWS**

| Item | Year/Rev | Name |
|---|---|---|
| ANSI/ANS 55.1 | 2021 | Solid Radioactive Waste Processing System for Light-Water-Cooled Reactor Plants |
| API 610 | 2021 | Centrifugal Pumps for Petroleum, Petrochemical, and Natural Gas Industries |
| ASME B31.3 | 2020 | Process Piping |
| ASME BPVC Section II | 2019 | Materials |
| ASME BPVC Section IX | 2019 | Welding, Brazing, and Fusing Qualifications |
| ASCE 7 | 2016 | Minimum Design Loads and Associated Criteria for Buildings and Other Structures with Supplement 1 |



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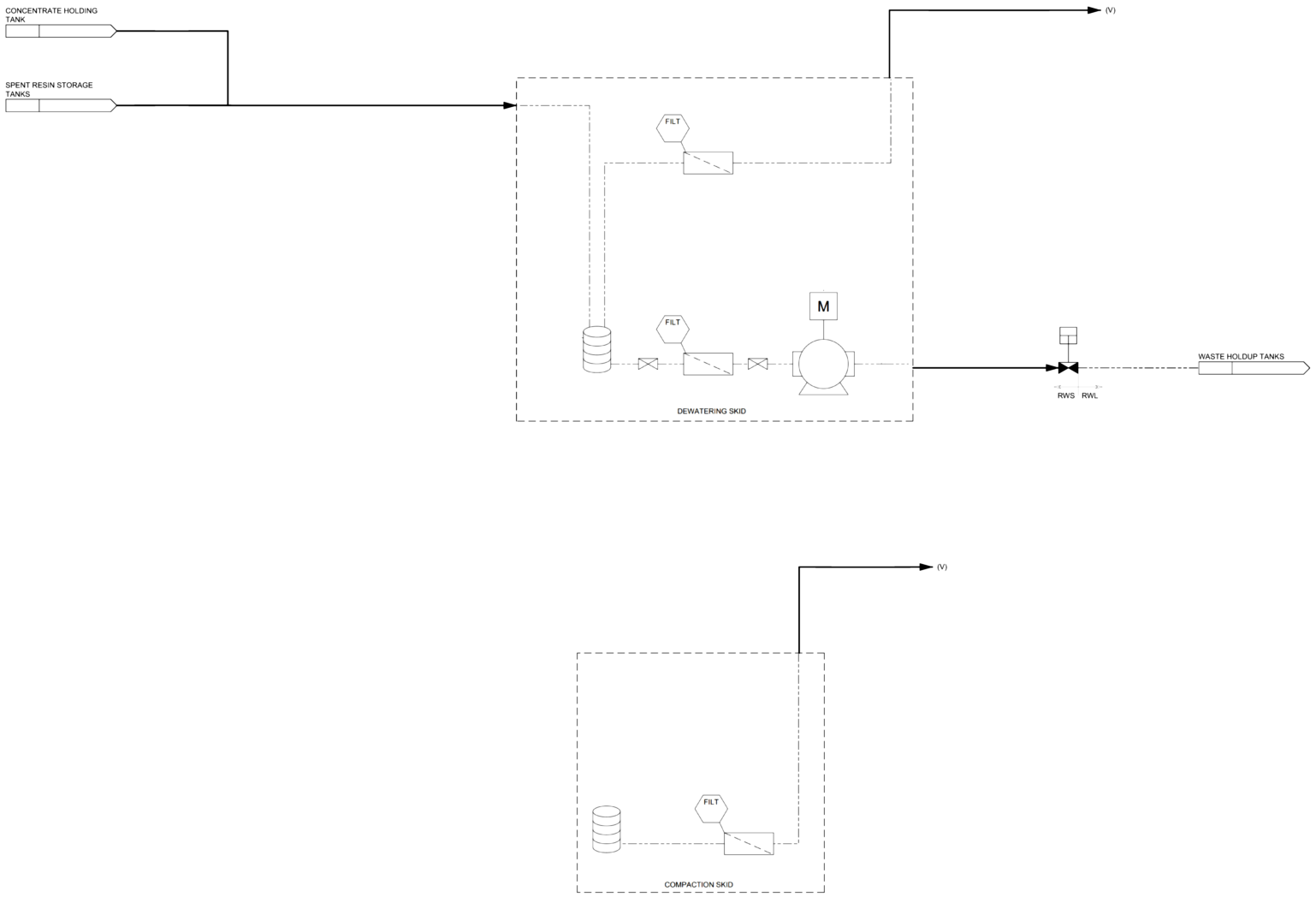
## Figure 9.3-1 RWS System Architecture





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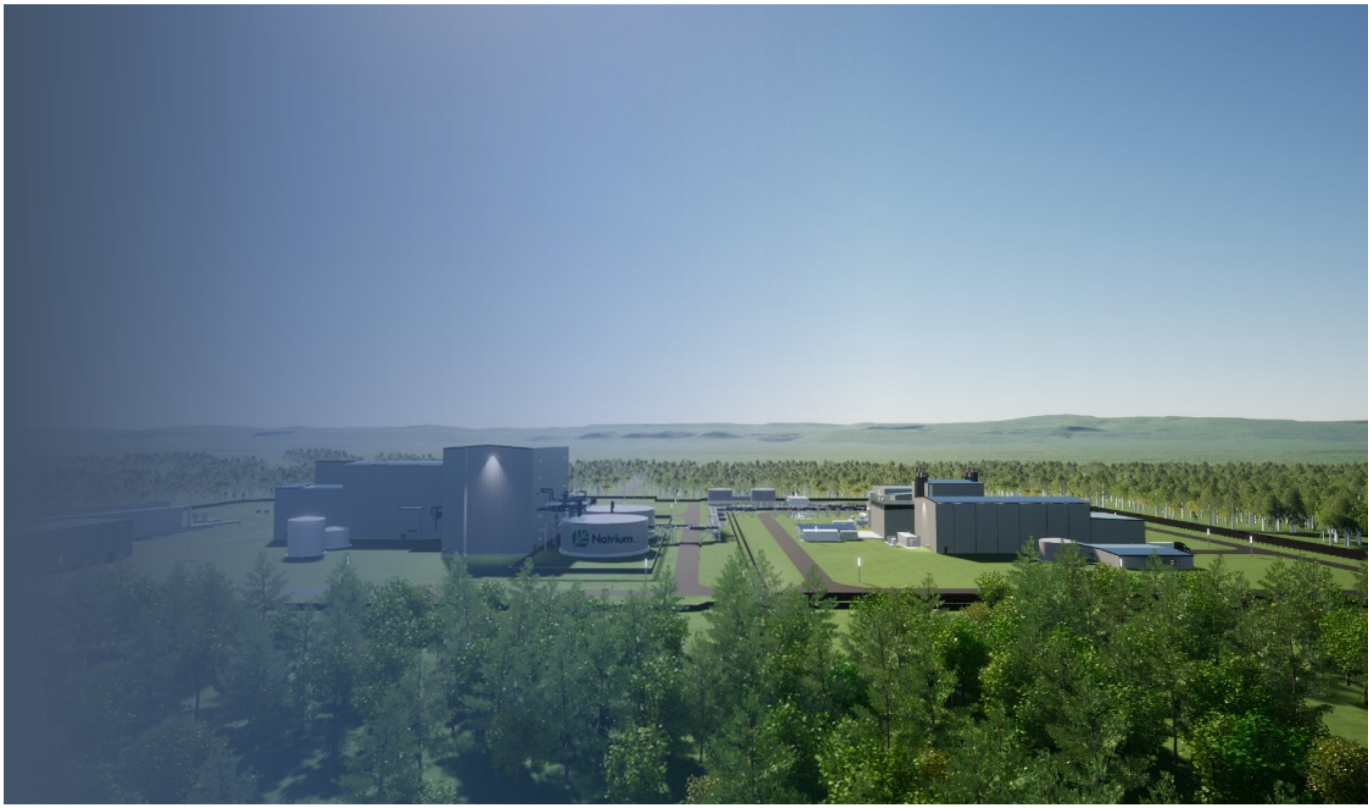

# Kemmerer Power Station Unit 1
# PSAR, Chapter 10

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## TABLE OF CONTENTS

**Chapter 10 Control of Occupational Dose. . . . . 10.1-1**

10.1 Radiation Protection Program . . . . . 10.1-1

10.1.1 Radiation Surveys and Monitoring . . . . . 10.1-2

10.1.2 Access Control and Dosimetry . . . . . 10.1-2

10.1.3 High and Very High Radiation Areas . . . . . 10.1-2

10.2 ALARA Program. . . . . 10.2-1

10.3 Design Considerations . . . . . 10.3-1

10.3.1 PDC Applicable to Control Occupational Radiation Exposures . . . . . 10.3-2

10.3.2 Major Radiation Sources . . . . . 10.3-2

10.4 Design Provisions. . . . . 10.4-1

10.4.1 Shielding. . . . . 10.4-1

10.4.2 Reduce Accumulation of Radioactive Material. . . . . 10.4-1

10.4.3 Remote Operations . . . . . 10.4-1

10.5 Preliminary Dose Assessment for Expected Occupancy . . . . . 10.5-1

10.5.1 Chief Nuclear Officer Organization. . . . . 10.5-1

10.5.2 Operations . . . . . 10.5-1

10.5.3 Maintenance. . . . . 10.5-1

10.5.4 Training . . . . . 10.5-2

10.5.5 Engineering . . . . . 10.5-2

10.5.6 Station Sciences. . . . . 10.5-2

10.5.7 Organizational Effectiveness . . . . . 10.5-2

10.5.8 Security . . . . . 10.5-2

10.5.9 Refueling . . . . . 10.5-2


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## LIST OF TABLES

Table 10.3-1 Major Radiation Sources . . . . . 10.3-3

Table 10.5-1 Routine Online Operations Dose Assessment for Expected Occupancy . . . . . 10.5-3

Table 10.5-2 Refueling Operations Dose Assessment for Expected Occupancy . . . . 10.5-4


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# Chapter 10 Control of Occupational Dose

The Radiation Protection Program (RPP) will be developed and implemented prior to receiving radioactive materials on-site to include comprehensive worker protection practices, an organizational structure to support radiation protection, training, and monitoring, and administrative programs to maintain occupational radiation exposures (ORE) as low as reasonably achievable (ALARA). In addition, considerations for maintaining ORE ALARA are implemented in the Kemmerer Unit 1 plant design features. The RPP will be developed to ensure 10 CFR 19.12 and 10 CFR 20 requirements are met. Procedures and programs that incorporate ALARA principles during plant operations, including the RPP, will be provided at the operating license (OL) stage.

## 10.1 Radiation Protection Program

The RPP will be developed as required by 10 CFR 20.1101. Policies and procedures will be enforced to keep radiation exposure of personnel within limits defined by 10 CFR 20, “Standards for Protection Against Radiation.” The RPP will be developed and implemented consistent with the following guidance:

- Regulatory Guide 8.2, Revision 1, “Administrative Practices in Radiation Surveys and Monitoring”
- Regulatory Guide 8.7, Revision 4, “Instructions for Recording and Reporting Occupational Radiation Exposure Data”
- Regulatory Guide 8.8, Revision 3, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As is Reasonably Achievable”
- Regulatory Guide 8.9, Revision 1, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program”
- Regulatory Guide 8.25, Revision 1, “Air Sampling in the Workplace”
- Regulatory Guide 8.34, Revision 1, “Monitoring Criteria and Methods To Calculate Occupational Radiation Doses”
- Regulatory Guide 8.38, Revision 1, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants”

In combination with the RPP, procedures and engineering controls will be based upon sound radiation protection principles, to assure that ORE remain ALARA. Additional details of the RPP, including a description of the equipment, methods, and procedures will be provided at the OL stage.

The RPP will ensure ORE is maintained ALARA by providing effective means to mitigate the potential for and spread of airborne radioactive material. The potential for airborne radioactive material, generated during normal operation or during maintenance activities, will be assessed within maintenance work packages, and radiological work permits as described in the RPP in accordance with Regulatory Guide 8.25, Revision 1. The RPP will describe the requirements for performing dose assessments for workers exposed to airborne radioactive material.


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All personnel are responsible for maintaining ALARA practices and have the authority to halt any operation which, in their judgment, is radiologically unsafe. Management personnel are responsible for implementing, administering, and enforcing the ALARA requirements. Training of plant personnel will be conducted to ensure the requirements of 10 CFR 19.12 are met. The extent of ALARA training provided for each person will be at least commensurate with the worker's job responsibilities and plant areas frequented.

### 10.1.1 Radiation Surveys and Monitoring

The RPP meets the requirements outlined in 10 CFR 20, Subpart F and Regulatory Guide 8.2, Revision 1. The purpose of radiation monitoring and surveys is to determine radiation levels, concentrations of radioactive materials, and potential radiological hazards that could be present in the facility, and detect releases of radioactive material from facility equipment and operations.

### 10.1.2 Access Control and Dosimetry

General access control to radiation control areas will comply with the guidance provided by Regulatory Guide 8.8, Revision 3. The RPP will establish provisions to maintain effective control over personnel accessing the radiation control areas and the movement of radiation sources to ensure ORE are ALARA and reduce the spread of radioactive contamination. Measurements of airborne radioactive material and bioassays will be used to determine that internal occupational exposures to radiation do not exceed the dose limits specified in 10 CFR 20, Subpart C, "Occupational Dose Limits." Additional details of the dosimetry and radiation exposure control for the facility, including the locations of radiological control areas, access controls, shielding, and remote handling equipment, will be provided at the OL stage.

### 10.1.3 High and Very High Radiation Areas

Access control for High Radiation Areas (HRAs) and Very High Radiation Areas (VHRAs) will conform with Regulatory Guide 8.38, Revision 1, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," to include:

- Training of workers
- Job planning
- Pre-job surveys
- Pre-job briefing
- Use and updating of radiation work permits or equivalent
- Radiation protection coverage (direct or remote continuous)
- Placement of measuring and alarming dosimeters
- Posting of areas
- Physical barriers where possible
- Locks when possible
- Temporary shielding where possible


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Generally, access to VHRAs will be forbidden unless there is a sound operational or safety reason for entering. Procedures governing access control to HRAs and VHRAs will incorporate administrative steps necessary to ensure ORE are maintained ALARA. Procedures will include the process for gaining entry to HRAs and VHRAs (e.g. issuance and control of physical keys, digital keys, badge access). For instances where procedures for activities may increase radiation to HRA or above levels, notification of personnel who have access to or are in the relevant area will be provided. This notification will be described in procedures and made using the Plant Communication System, radio, phone, or other appropriate method. In addition, areas where personnel could have normal access and are likely to see transient radiation meeting or exceeding HRA levels, whether due to normal activities or abnormal occurrences, will be equipped with radiation detection equipment capable of providing visual and audible alarms to area workers.



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## 10.2 ALARA Program

A program to ensure occupational doses and doses to members of the public are as low as reasonably achievable (ALARA) will be implemented to meet the requirements of 10 CFR 20.1101. An ALARA program will be implemented consistent with the guidance in Regulatory Guide 8.10, Revision 2, “Operating Philosophy for Maintaining Occupational and Public Radiation Exposures as Low as is Reasonably Achievable.” Additional details of the ALARA program will be provided at the operating license stage.


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## 10.3 Design Considerations

The Kemmerer Unit 1 plant design incorporates guidance from Regulatory Guide 8.8, Revision 3, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As is Reasonably Achievable” (in particular Sections C.2 and C.4) such as reviews of new designs and equipment selection, facility and equipment design features, radiation area access control, radiation shielding, process instrumentation and radiation monitoring, control of airborne contaminants, and other considerations. Kemmerer Unit 1 plant design features that are examples of conformance with Regulatory Guide 8.8 and incorporation of as low as reasonably achievable (ALARA) principles are described in Section 10.4. ALARA principles are applied throughout design and construction of Kemmerer Unit 1, including:

- Incorporation of measures for reducing the need for time spent in radiation areas
- Reducing the frequency of maintenance
- Measures to improve the accessibility to components requiring periodic maintenance or inservice inspection
- Measures to reduce the production, distribution, and retention of activated products throughout plant systems
- Measures that address conditions during all phases of plant operation, including power operations, shutdown, and refueling
- Measures for assuring that occupational radiation exposure (ORE) during decommissioning will be ALARA
- use of engineered barriers as the primary means for ensuring personnel protection from radiation
- Reviews of the design by competent radiation protection personnel
- Instructions to designers and engineers regarding ALARA design
- Experience from operating plants and past designs
- Continuing facility design reviews

Design features minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste consistent with the requirements in 10 CFR 20.1406. Structures, systems, or components (SSCs) expected to contain or handle radiological materials include design considerations that minimize contamination, including collection and control of spillage. Design features consider the guidance in Regulatory Guide 4.21, Revision 0, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning.” For example, contamination control is supported through cascade ventilation from low contamination areas to higher contamination areas. Flow created by negative pressure gradients carries contamination away from lower contamination areas and toward higher contamination areas, minimizing its spread. Contamination control is discussed further in Section 9.2.


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### 10.3.1 PDC Applicable to Control Occupational Radiation Exposures

There are no Principal Design Criteria (PDC) specific to controlling occupational radiation exposures. Consideration of the radiation protection aspects of compliance with PDC 19 for the Main Control Room is discussed in Section 7.6.7.1. Consideration of the radiation protection aspects of compliance with PDC 61 for fuel handing systems is discussed in Section 7.3.

### 10.3.2 Major Radiation Sources

Sources of radiation that contribute to ORE are those resulting from fission in the fuel (fission products and decay products) and neutron activation products, including tritium generated as a result of exposure to neutrons. Fission and activation products are located in the coolant, cover gas, and structures, and are the result of fuel defects, neutron activation of various isotopes, and corrosion and wear products. Major radiation sources are listed in Table 10.3-1.

Additional details of the radiation sources for the facility which demonstrate compliance with 10 CFR 20 will be provided at the operating license stage.


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**Table 10.3-1 Major Radiation Sources**

| Location | System | Description of Source |
|---|---|---|
| Reactor Building (RXB) | Reactor Enclosure System (RES) | Active core |
| | Primary Heat Transport System (PHT) | Activated sodium, fission and activation products |
| Reactor Auxiliary Building (RAB) | Intermediate Heat Transport System (IHT) | Activated sodium, tritium |
| | Sodium Processing System (SPS) | Activated sodium, fission and activation products, tritium |
| Fuel Handling Building (FHB) | Ex-Vessel Fuel Handling System (FHE) | Fresh and irradiated fuel assemblies, non-fuel components |
| | Water Pool Fuel Handling System (FHP) | Irradiated core assemblies |
| | Gaseous Radwaste Processing (RWG) | Activated argon, fission and activation products, tritium |
| | Solid Radwaste Processing (RWS) | Spent resins, spent filters, dry active waste |
| | Liquid Radwaste Processing (RWL) | Spent resins, water contaminated with fission and activation products |


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## 10.4 Design Provisions

The Kemmerer Unit 1 plant design incorporates features that maintain occupational radiation exposure (ORE) as low as reasonably achievable (ALARA). General features including plant layout, requirements for systems to reduce accumulation of radioactive material, and providing the capability to remotely service equipment containing substantial radiation sources are all considerations made during the design phase.

Sections 10.4.1 through 10.4.3 provide specific examples of considerations given to the design of SSCs to ensure ORE are maintained ALARA.

### 10.4.1 Shielding

The Reactor Core System (RCC) contains shield assemblies that absorb radial neutron leakage outside of the reflector assemblies to limit activation of intermediate sodium coolant and reduce the irradiation dose to permanent structures outside of the reactor core. Limiting activation of intermediate sodium coolant ensures ORE are maintained ALARA.

The Reactor Enclosure System (RES) includes features that limit the activation of intermediate sodium within the IHX. As part of the RES, the reactor head provides a thick, dense material of the head structure itself which provides bulk radiological shielding to the Head Access Area (HAA). In addition, the Reactor Fixed Internals (RFI) contain the in-vessel shield and reactor head shield plates. The in-vessel shields are designed to reduce personnel exposure and limit the activation of air within the RAC. Section 7.1.2 discusses the RES.

Shielding is provided in the Sodium Cover Gas System sampling and monitoring cabinet that pulls the cover gas sample to reduce personnel exposure in the area. In addition, a shielded pipe chase is provided for radioactive gases traveling from the SCG, Ex-Vessel Storage Tank (EVST), and tag-gas analysis exhaust cabinet. Section 7.2.3 discusses the SCG.

The Spent Fuel Pool (SFP) provides sufficient water depth to reduce radiation exposure to personnel from stored used core assemblies and core assembly transfer operations.

The tanks, filters and other equipment for the processing of gaseous, liquid, and solid radioactive waste are shielded to maintain ORE ALARA.

### 10.4.2 Reduce Accumulation of Radioactive Material

The Sodium Processing System (SPS) is designed to remove cesium and other fission products from the Primary Heat Transport System (PHT) to reduce cesium plate-out in cool areas and reduce concentrations in the cover gas that will distribute in the cover gas and gaseous radiological waste systems, each contributing to ORE. The SPS removes cesium from PHT using Reticulated Vitreous Carbon traps.

### 10.4.3 Remote Operations

The Kemmerer Unit 1 plant design utilizes a central location to remotely control fuel handling equipment as a means to reduce ORE.


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The need for personnel to enter areas with high radiation dose rates is reduced by the capability to remotely monitor and control various plant parameters.



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## 10.5 Preliminary Dose Assessment for Expected Occupancy

A preliminary dose assessment table for normal operations is presented in Table 10.5-1 and for refueling operations in Table 10.5-2. Each table contains the estimated radiation exposures to plant personnel to demonstrate the preliminary plant design complies with 10 CFR 20 and maintains occupational radiation exposure (ORE) as low as reasonably achievable (ALARA). The dose assessments are based on the expected staffing levels of Kemmerer Unit 1 and conservative target dose rates used in preliminary design for various Nuclear Island (NI) buildings. The organizations and activities performed are described in this section.

The preliminary design target dose rates are not based on expected source terms and therefore are not representative of actual dose rates that may be encountered. The initial radiation zones were developed to provide a basis for the preliminary design of the plant. For the purposes of the preliminary dose assessment, the currently defined radiation zones for various areas of the plant are used as part of the assessment. Personnel are assumed to be in the average dose rate for zones they are performing activities in. Additionally, shielding designs for various systems are not mature enough to produce actual expected radiation dose rates throughout the plant. As plant design evolves, the assessment will be revised based on calculated radiation levels considering matured shielding designs, personnel access requirements, and task analyses.

Detailed tables with dose assessment for expected occupancy, similar to those in Regulatory Guide 8.19, Revision 1, “Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-REM Estimates,” will be provided at the operating license stage.

### 10.5.1 Chief Nuclear Officer Organization

The Chief Nuclear Officer organization includes the Plant General Manager and direct reports. Members of this organization receive occupational radiation exposure (ORE) performing routine walk-downs. It also includes warehouse personnel managing shipments of radioactive material and Quality Assurance and Quality Control personnel performing inspections.

### 10.5.2 Operations

The Operations organization includes Shift Managers, Control Room Supervisors, Reactor Operators, and Non-Licensed Operators. Members of this organization receive ORE performing routine, start-up, and shutdown operations.

### 10.5.3 Maintenance

The Maintenance organization includes Instrumentation and Control (I&C), electrical, and mechanical maintenance technicians and supervisors. Members of this organization receive ORE performing routine maintenance and inspections including periodic, corrective, and predictive maintenance as well as regular performance monitoring on various structures, systems, or components (SSCs) throughout Kemmerer Unit 1. This includes online and outage inspections and maintenance for SSCs not directly part of refueling evolutions.


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### 10.5.4 Training

The Training organization includes Operations, Maintenance, and Station Sciences instructors, as well as simulator personnel. Members of this organization receive ORE performing plant walk-downs in support of training.

### 10.5.5 Engineering

The Engineering organization includes I&C engineers, reactor engineers, and supervision. Members of this organization receive ORE performing engineering walk-downs and field observations.

### 10.5.6 Station Sciences

The Station Sciences organization includes Radiation Protection (RP) and Chemistry technicians and a Safety Professional. Members of this organization receive ORE performing RP surveys, job coverage, sampling and analysis, and walk-downs. The Safety Professional provides general safety and health services.

### 10.5.7 Organizational Effectiveness

The Organizational Effectiveness organization includes Emergency Preparedness personnel. Members of this organization receive ORE performing walk-downs in support of Emergency Preparedness planning duties.

### 10.5.8 Security

The Security organization includes security officers, supervision, and a training instructor. Members of this organization receive ORE performing security patrols.

### 10.5.9 Refueling

During refueling outages, the additional personnel brought on site increase ORE for the duration of the outage. All personnel on site perform activities similar to those during routine operations.



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**Table 10.5-1 Routine Online Operations Dose Assessment for Expected Occupancy**

| Organization | Activity | Preliminary Estimated Exposure (person-rem/year) |
|---|---|---|
| Chief Nuclear Officer | General Management Duties | < 0.5 |
| | Warehouse Radioactive Material Receipt | 1.5 |
| | Quality Assurance and Quality Control Inspections | 0.5 |
| Operations | General Reactor Operations On-shift | 17.5 |
| | Clearance and Tagging | 1 |
| | Fire Marshall | < 0.5 |
| Maintenance | General Maintenance Duties | < 1 |
| | I&C | 8.5 |
| | Electrical | 1.5 |
| | Mechanical | 3 |
| Training | General Training Duties | < 0.5 |
| Engineering | General Engineering Duties (including I&C and Reactor Engineering) | 0.5 |
| Station Sciences | Safety | 0.5 |
| | RP and Chemistry Lead Technician Duties | 1 |
| | Station Sciences Technician Duties | 15 |
| Organizational Effectiveness | Emergency Preparedness Walk-downs | < 0.5 |
| Security | Security Patrols | 15 |
| | **Total** | 67.5 |


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**Table 10.5-2 Refueling Operations Dose Assessment for Expected Occupancy**

| Organization | Preliminary Estimated Exposure (person-rem/year) |
|---|---|
| Chief Nuclear Officer | 0 |
| Operations | 35 |
| Maintenance | 72 |
| Training | 2 |
| Engineering | 1.5 |
| Station Sciences | 32.5 |
| Organizational Effectiveness | 1 |
| Security | 6.5 |
| Support Staff | 47 |
| **Total** | 197.5 |


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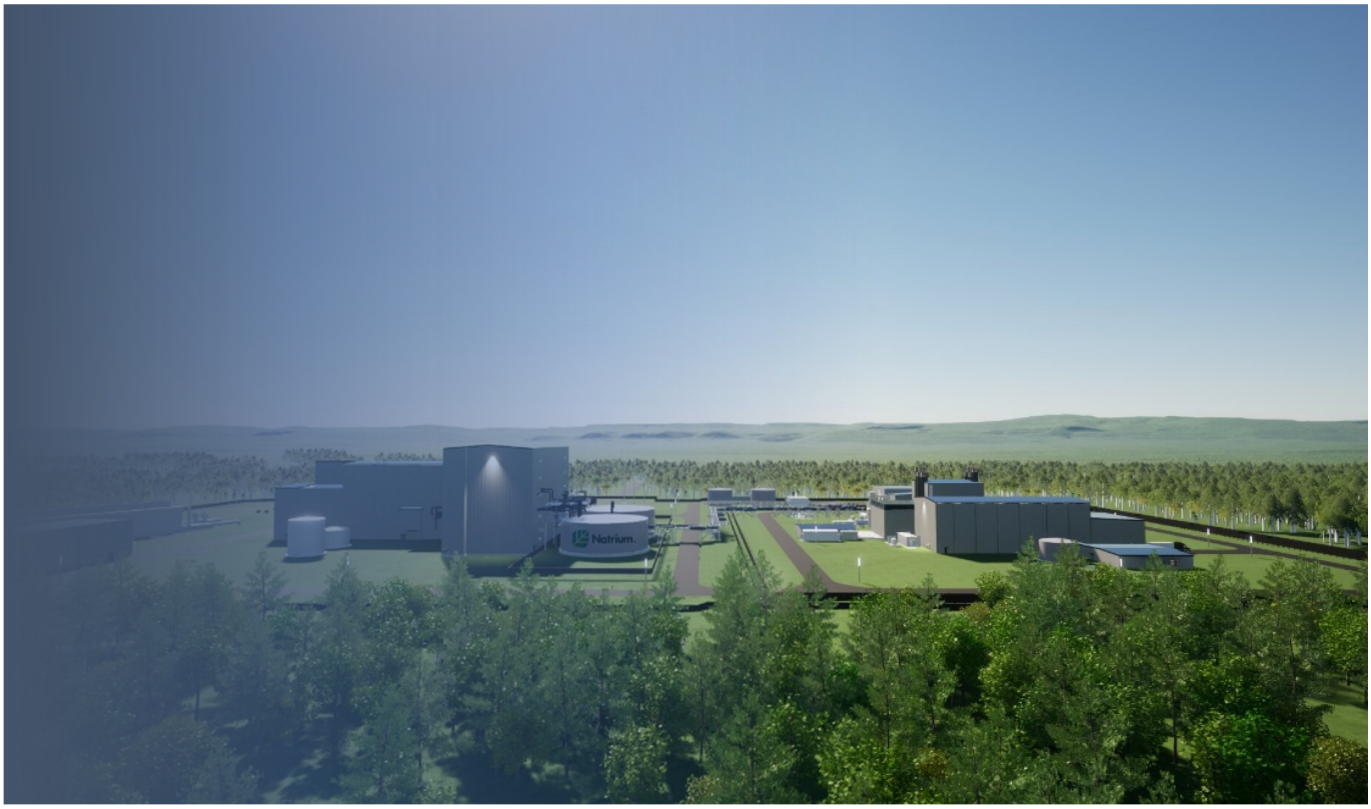

# Kemmerer Power Station Unit 1
# PSAR, Chapter 11

US SFR Owner, LLC, a wholly owned subsidiary of TerraPower, LLC

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## TABLE OF CONTENTS

**Chapter 11 Conduct of Operations . . . . . 11.1-1**

11.1 Organization . . . . . 11.1-1

11.1.1 Executive Management Organization- TerraPower . . . . . 11.1-2

11.1.2 Natrium Project Organization- TerraPower . . . . . 11.1-3

11.1.3 Construction Organization- BPC . . . . . 11.1-4

11.1.4 Operating Organization and Technical Support . . . . . 11.1-5

11.1.5 Qualification, Training, and Conduct of Operations . . . . . 11.1-8

11.2 Human Factors Engineering. . . . . 11.2-1

11.2.1 Management of the Human Factors Engineering Program . . . . . 11.2-1

11.2.2 Operating Experience Review . . . . . 11.2-14

11.2.3 Safety Function Review . . . . . 11.2-18

11.2.4 Task Analysis . . . . . 11.2-19

11.2.5 Human-System Interface Design . . . . . 11.2-21

11.2.6 Procedures, Training, and Verification and Validation . . . . . 11.2-21

11.2.7 Human Performance Monitoring . . . . . 11.2-25

11.3 Emergency Preparedness . . . . . 11.3-1

11.3.1 Bases for Emergency Preparedness Approach . . . . . 11.3-1

11.3.2 Physical Site Characteristics . . . . . 11.3-3

11.3.3 Emergency Planning Zone . . . . . 11.3-3

11.3.4 Emergency Response Organization. . . . . 11.3-4

11.3.5 Federal, State and Local Governmental Agencies . . . . . 11.3-4

11.3.6 Protective Measures. . . . . 11.3-5

11.3.7 First Aid, Decontamination, Off-site Medical Treatment, and Emergency Medical Transportation . . . . . 11.3-6

11.3.8 Training . . . . . 11.3-6


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## TABLE OF CONTENTS

11.3.9 Emergency Response Facilities and Equipment . . . . . 11.3-6

11.3.10 Performance Monitoring . . . . . 11.3-8

11.4 Aircraft Impact Assessment . . . . . 11.4-1

11.5 Technical Specifications . . . . . 11.5-1

11.5.1 Introduction . . . . . 11.5-1

11.5.2 Operating Modes . . . . . 11.5-1

11.6 Fitness-for-Duty and Security . . . . . 11.6-1

11.6.1 Fitness-for-Duty . . . . . 11.6-1

11.6.2 Security . . . . . 11.6-1


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## LIST OF TABLES

Table 11.2-1 NUREG-0711 Technical Elements Compared to Section 11.2 . . . . . 11.2-28

Table 11.2-2 Development of HFE Review Guidance for Advanced Reactors . . . . . 11.2-29

Table 11.5-1 Proposed Variables and Conditions for Technical Specifications . . . . . 11.5-3

Table 11.5-2 Operating Modes . . . . . 11.5-10


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## LIST OF FIGURES

Figure 11.1-1 Executive Management Organization . . . . . 11.1-10

Figure 11.1-2 Natrium Project Organization . . . . . 11.1-11

Figure 11.1-3 Construction Organization . . . . . 11.1-12

Figure 11.1-4 Operating Organization and Technical Support . . . . . 11.1-13


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# Chapter 11 Conduct of Operations

## 11.1 Organization

This section describes the organizational structure and key management positions in the organizations that are responsible for facility design, design review, design approval, construction management, testing, and plant operation.

TerraPower, LLC is responsible for the design and construction of Kemmerer Unit 1. TerraPower has over 15 years of experience as a nuclear innovation company, which includes designing, building, and laboratory testing of advanced nuclear technologies. TerraPower partners with GE-Hitachi Nuclear Energy Americas, LLC (GEH) as a co-developer and Bechtel Power Corporation (BPC) as an engineering and construction partner. GEH and BPC each have over 60 years of experience in the design and construction of commercial nuclear reactors. BPC has participated in the design and construction of over 150 nuclear power projects in 25 countries. TerraPower leverages the experience of its partners to ensure the safe design and construction of Kemmerer Unit 1.

The TerraPower Board of Directors has over 50 years of nuclear executive experience which it leverages to ensure a culture exists at the company where rigorous standards are maintained for nuclear safety, safety culture, and construction.

US SFR Owner, LLC (USO), a wholly owned subsidiary of TerraPower, is responsible for the safe operation of Kemmerer Unit 1 from pre-operational testing through commercial operation. Responsibilities are assigned to an Executive Management Organization, Natrium Project Organization, Construction Organization, and Operating Organization. An overview of organizational responsibilities from design to commercial operation follows.

- Design: TerraPower has overall ownership, control, oversight, and authority of the design.
- Construction: TerraPower partners with BPC to perform the construction of Kemmerer Unit 1. BPC executes this responsibility through the Construction Organization. TerraPower coordinates with, and provides oversight of, BPC through the Natrium Project Organization, which continues to be the design authority. TerraPower oversight of the quality assurance program during construction is described in Section 8.1.
- Pre-operational testing: USO conducts pre-operational testing, which includes fuel load and startup testing, through the Operating Organization. Overall technical support functions are provided through the oversight of TerraPower and USO.
- Commercial operation: USO executes responsibilities for the daily operation of Kemmerer Unit 1 through the Operating Organization and shares technical support functions with TerraPower.

Detailed descriptions of the Executive Management Organization, Natrium Project Organization, Construction Organization, and Operating Organization follows.


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### 11.1.1 Executive Management Organization- TerraPower

Executive management has the primary responsibility for executing activities and functions for Kemmerer Unit 1. Executive management establishes expectations for quality, safety, and efficiency in plant operations and support activities performed by an organization selected and trained to meet these expectations through an effective management model. The Executive Management Organization includes the President and Chief Executive Officer (CEO) of TerraPower, the TerraPower executive responsible for project execution, the TerraPower executive responsible for regulatory affairs, the TerraPower Chief Nuclear Officer (CNO), and the director responsible for quality. The roles and responsibilities of these executives are described below. The Executive Management Organization is depicted in Figure 11.1-1.

#### 11.1.1.1 President and CEO- TerraPower

The President and CEO of TerraPower has overall responsibility for functions involving design, construction, and operation. The President and CEO of TerraPower is responsible for the execution of the Quality Assurance Program Description (QAPD) and its subordinate documents and ensuring information and resources are shared between the nuclear-oriented part of the organization and the rest of the corporate organization in a manner that prioritizes nuclear safety and security.

Responsibilities for design and construction are assigned to the executive responsible for project execution. Prior to pre-operational testing, direct control of nuclear plant operation is assigned to the CNO and the CNO’s direct reports.

#### 11.1.1.2 Executive Responsible for Project Execution- TerraPower

The executive responsible for project execution is responsible for the design and construction of Kemmerer Unit 1. It is the responsibility of the executive responsible for project execution to provide guidance and direction such that activities including engineering, construction, operations, maintenance, procurement, and planning during the design and construction phase are performed in accordance with the requirements of the QAPD.

The executive responsible for project execution ensures contractor support is obtained to perform principal site-related engineering studies of meteorology, geology, seismology, hydrology, demographics, and environmental effects. The results of these studies are described in Chapter 2. The executive responsible for project execution is responsible for coordination between groups performing design and construction activities, to include coordination with the nuclear reactor vendor.

#### 11.1.1.3 Executive Responsible for Regulatory Affairs- TerraPower

The executive responsible for regulatory affairs is responsible for the licensing, fitness-for-duty (FFD), security, and records management for Kemmerer Unit 1 during design and construction. The executive responsible for regulatory affairs ensures licensing activities are performed in accordance with government regulations, that FFD and security requirements are met as described in Section 11.6, and that records management and document control program activities for TerraPower meet the requirements of the QAPD. The executive responsible for


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regulatory affairs is assisted in these activities by the director responsible for licensing, the manager responsible for cyber and physical security, and the manager responsible for records management and document control.

#### 11.1.1.4 CNO- TerraPower

The CNO has direct authority and responsibility for the management, control, and supervision of the nuclear power plant and for the execution of nuclear programs, policies, and decisions. The CNO ensures the collective experience of the senior leadership team at Kemmerer Unit 1 meets the requirements of Section 4.2 of ANSI/ANS-3.1-2014 (Reference 11.1-1), as endorsed by Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” Revision 4. The CNO is assisted in management and technical support activities by the Plant General Manager (PGM), Director of Natrium Support Services, Procurement Manager, and the Director of Nuclear Oversight.

#### 11.1.1.5 Director Responsible for Quality- TerraPower

The director responsible for quality reports to the President and CEO of TerraPower and is responsible for maintaining the QAPD and ensuring the design and construction of Kemmerer Unit 1 is performed in accordance with the QAPD. The director responsible for quality is responsible for ensuring quality audits are performed in accordance with the QAPD as well as ensuring TerraPower performs functions safely and in accordance with environmental regulations. The director responsible for quality is independent of any TerraPower project and has the authority to stop work on any project for quality, safety, or environmental concerns.

### 11.1.2 Natrium Project Organization- TerraPower

The Natrium Project Organization is responsible for the design and construction of Kemmerer Unit 1. TerraPower partners with BPC to execute this responsibility through the Construction Organization, which is described in Section 11.1.3. The Natrium Project Organization is the design authority for Kemmerer Unit 1.

The Natrium Project Organization is led by the executive responsible for project execution. The executive responsible for project execution is supported by the Deputy Project Director, Director Plant Delivery, Director Nuclear Fuel and Equipment Qualification, executive responsible for design, and Director of Safety and Engineering Analysis. The roles and responsibilities of the executive responsible for project execution are described in Section 11.1.1.2. The roles and responsibilities of individuals reporting to the executive responsible for project execution are described in Section 11.1.2.2 through Section 11.1.2.5. The Natrium Project Organization is depicted in Figure 11.1-2.

#### 11.1.2.1 Deputy Project Director- TerraPower

The Deputy Project Director is the senior project manager and is responsible for the overall planning, direction, management, procurement activities, coordination, implementation, execution, control, and completion of construction activities related to Kemmerer Unit 1. The Deputy Project Director manages the Senior Manager, Procurement; Senior Manager, Project Planning; and Manager, Subcontractor Oversight.


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#### 11.1.2.2 Director of Plant Delivery- TerraPower

The Director of Plant Delivery oversees the design of safety-significant structures, systems, and components (SSCs).

#### 11.1.2.3 Director Nuclear Fuel and Equipment Qualification- TerraPower

The Director Nuclear Fuel and Equipment Qualification oversees fuel, materials, and component specification review and approval.

#### 11.1.2.4 Executive Responsible for Design- TerraPower

The executive responsible for design is responsible for configuration management of the design; establishes and maintains design processes that ensure the design meets technical and quality standards; maintains engineering design programs; and establishes qualification standards for engineering personnel.

#### 11.1.2.5 Director of Safety and Engineering Analysis- TerraPower

The Director of Safety and Engineering Analysis oversees the development of the probabilistic risk assessment, defense-in-depth, and design basis event analyses.

The Director of Safety and Engineering Analysis is supported by the Senior Manager, Safety Analysis and Risk who manages the team performing the risk analysis activities. This team leverages the experience gained from previous TerraPower projects, including the TWR®. The team performing the risk analysis activities also benefits from the experience of GEH. The internal team is supplemented, when necessary, with external contractor personnel to ensure that the appropriate risk analysis experience is available.

### 11.1.3 Construction Organization- BPC

BPC performs the construction of Kemmerer Unit 1 and implements management systems for quality, health, safety and environmental, cost and program metrics, and monitoring across the project. Specialty contractors provide support for scopes of work that require unique capabilities or knowledge. TerraPower has oversight of the Construction Organization through a project quality execution plan as described in Section 8.1.2.

The Construction Organization is shown in Figure 11.1-3. The Project Manager is the head of the Construction Organization. The responsibilities of key management personnel are described in Section 11.1.3.1 through Section 11.1.3.3.

#### 11.1.3.1 Site Manager- BPC

The Site Manager reports to the Project Manager and is responsible for the management of the total site. The Site Manager responsibilities include environmental safety and health on-site, security, emergency response, incident reporting, quality, subcontract management, schedule progress and performance, budget control, and overall management of the project site.


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#### 11.1.3.2 Project Quality Manager- BPC

The Project Quality Manager reports directly to the Project Manager and is responsible for managing and implementing the project quality plan. The Project Quality Manager supervises a team of personnel that performs oversight of construction activities, including audits, to ensure construction is performed in accordance with the BPC Quality Assurance (QA) program.

#### 11.1.3.3 Project Field Engineer- BPC

The Project Field Engineer leads a team of field engineers who are accountable for quality, technical compliance, and quantity verification. The Project Field Engineer is the primary construction interface with the Natrium Project Organization, BPC, and GEH engineering groups.

### 11.1.4 Operating Organization and Technical Support

Before commencing pre-operational testing, the CNO establishes the organization of managers, functional managers, supervisors, and staff necessary to support required functions for the safe operation of the plant. The functions include:

- Engineering
- Quality assurance, audit, and surveillance
- Plant sciences (chemistry and radiation protection)
- Commissioning support
- Training
- Maintenance
- Operations
- Outside contractual support

USO is responsible for the safe operation of Kemmerer Unit 1 and establishes an Operating Organization prior to pre-operational testing. Personnel in the Operating Organization meet the qualification and experience requirements of ANSI/ANS 3.1-2014, as endorsed by Regulatory Guide 1.8. The PGM and Director of Operations are responsible for the Operating Organization. Prior to the commencement of pre-operational testing, the Operating Organization establishes the following:

- QA program for operational phase.
- Procedures necessary to carry out an effective QA program.
- Program for review and audit of activities affecting plant safety.
- Programs and procedures for rules of practice as described in Section 3.5 of ANSI/ANS 3.2-2012 (Reference 11.1-2) that are applicable to Kemmerer Unit 1.

Both USO and TerraPower provide technical support for the Operating Organization. TerraPower provides engineering and commissioning support, while USO is responsible for technical support functions. The organizational relationships between personnel responsible for plant operations


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and those providing technical support are shown in Figure 11.1-4. The roles and responsibilities of key individuals responsible for operating and technical support are described in Section 11.1.4.1 through Section 11.1.4.1.6.

#### 11.1.4.1 Plant General Manager- USO

The PGM reports to the CNO and oversees the daily operations of Kemmerer Unit 1. The PGM provides leadership and management and establishes goals for employee safety, reliable plant operations, and training. The PGM has overall responsibility for occupational and public radiation safety.

##### 11.1.4.1.1 Director of Operations- USO

The Director of Operations reports to the PGM and provides oversight for the on-site operations and operations support personnel.

##### 11.1.4.1.2 Director of Maintenance- USO

The Director of Maintenance reports to the PGM and is responsible for plant maintenance and work control. The Director of Maintenance provides oversight of on-site maintenance personnel.

##### 11.1.4.1.3 Station Sciences Manager- USO

The Station Sciences Manager reports to the PGM and oversees the management of the chemistry and radiation protection programs. The Station Sciences Manager provides technical guidance for development and implementation of policies and programs that aid in the safe operation of the Nuclear Island (NI) while minimizing the radiological and chemical impact on the personnel, environment, SSCs, and ensuring that regulatory requirements and industry best practices are implemented. The Functional Manager in charge of Station Sciences has the experience outlined in Sections 4.3.2 and 4.3.3 of ANSI/ANS-3.1-2014, as endorsed by Regulatory Guide 1.8.

##### 11.1.4.1.4 Director of Training- USO

The Director of Training reports to the PGM and is responsible for the qualification and continuing training of operations, maintenance, station sciences, and engineering personnel. The Director of Training is also responsible for ensuring plant personnel receive general nuclear plant training.

##### 11.1.4.1.5 Security Manager- USO

The Security Manager reports to the PGM and is responsible for the execution of the physical security plan, security training and qualification plan, access authorization and the fitness-for-duty (FFD) program. The FFD program and physical security plan are described in Section 11.6.

##### 11.1.4.1.6 Director of Organizational Effectiveness- USO

The Director of Organizational Effectiveness reports to the PGM and is responsible for emergency planning, licensing, regulatory assurance, and occupational health.


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#### 11.1.4.2 Director of Nuclear Oversight- USO

The Director of Nuclear Oversight reports to the CNO and is responsible for the overall management and coordination of independent nuclear oversight to include audits, quality control, and employee concerns. Nuclear Oversight provides integrated compliance and performance-based oversight and quality control. Independent oversight provides the CNO and senior corporate leaders, up through the Board of Directors, with an ongoing perspective of regulatory compliance at the nuclear station and in the corporate organization benchmarked to the industry, with a focus on nuclear safety, plant reliability, and emergency response effectiveness.

##### 11.1.4.2.1 QA Manager- USO

The QA Manager reports to the Director of Nuclear Oversight and is responsible for the supervision of employees who perform quality control inspections, non-destructive examination inspections, and QA audit functions at the nuclear plant. The QA Manager meets the experience outlined in Sections 4.3.5 and 4.4.11 of ANSI/ANS-3.1-2014, as endorsed by Regulatory Guide 1.8.

#### 11.1.4.3 Procurement Manager- USO

The Procurement Manager reports to the CNO and is responsible for ensuring materials and equipment are identified, specified, received, inspected, stored, and issued. The Procurement Manager has at least one year of experience working in the procurement branch of a nuclear utility or working for a nuclear supplier.

#### 11.1.4.4 Director Engineering Support Services- TerraPower

The Director Engineering Support Services reports to the CNO and is responsible for oversight of all engineering activities related to the design, operation, and maintenance of Kemmerer Unit 1 with a primary focus on the NI and interfacing systems.

##### 11.1.4.4.1 Commissioning Manager- TerraPower

The Commissioning Manager reports to the Director Engineering Support Services and is responsible for developing the organization, processes, and procedures necessary to commission the plant. The commissioning team oversees development and execution of the commissioning schedule, disposition of testing results, proper execution of system turnovers, and the retention of commissioning records.

##### 11.1.4.4.2 Plant Support Engineering- TerraPower

The functional manager for Plant Support Engineering reports to the Director Engineering Support Services and is responsible for ensuring the Plant Support Engineering team monitors and analyzes system trends and performances to ensure proper plant performance. The Plant Support Engineering team has a focus on ensuring NI and interfacing systems are performing as designed. The team provides engineering support for emergent system issues, provides input to outage scoping, and recommends design changes for plant systems.


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11.1.4.4.3 Design Support Engineering- TerraPower

The functional manager for Design Support Engineering reports to the Director Engineering Support Services and is responsible for design changes. The Design Support Engineering team ensures that NI and interfacing systems are configured, operated, and maintained in accordance with the design basis.

11.1.5 Qualification, Training, and Conduct of Operations

11.1.5.1 Qualification of Nuclear Personnel

Qualifications of managers, supervisors, operators, technicians, and other support staff meet the qualification requirements described in ANSI/ANS-3.1-2014, as endorsed by Regulatory Guide 1.8.

11.1.5.2 Training of Personnel

The operator training and requalification plan will be developed and implemented in accordance with 10 CFR 55 and 10 CFR 50.120, and USO will comply with those requirements as it pertains to power reactors. A description and plans for an operator requalification plan will be provided at the OLA stage, consistent with the requirements in 10 CFR 50.34(b)(8). The qualification process will include passing a comprehensive written exam and an operating test as required by 10 CFR 55.

11.1.5.3 Conduct of Operations

Site operations are controlled through the Main Control Room (MCR) with some activities being coordinated and issued from the online/outage control center (OCC). Maintenance activities, surveillances, and the removal or restoration of safety-significant SSCs may not commence without approval of the Control Room Supervisor in the MCR. The rules of practice for control room activities, as described by administrative procedures, incorporate the guidance of Regulatory Guide 1.114, “Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit,” Revision 3 and address the following:

- Conduct of operator at the controls
- Conduct of Control Room Supervisor in the control room
- Definition of and scope of the surveillance area
- Relief requirements for operator at the controls and the Control Room Supervisor

In accordance with 10 CFR 50.54:

- Reactivity controls are only manipulated by licensed operators and senior operators except as allowed for training under 10 CFR 55.
- Apparatuses and mechanisms other than controls which may affect reactivity or power level of the reactor shall be operated only with the knowledge and consent of the operator at the controls or the Control Room Supervisor.


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It is expected that operators and maintenance technicians operate the plant in accordance with approved procedures and adhere to technical specifications. The staffing requirements in 10 CFR 50.54(k) and (m) will be provided at the operating license stage.

## References

11.1-1 ANSI/ANS-3.1-2014, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants,” November 20, 2014.

11.1-2 ANSI/ANS-3.2-2012 (R2017), “Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plant,” April 4, 2017.



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**Figure 11.1-1 Executive Management Organization**

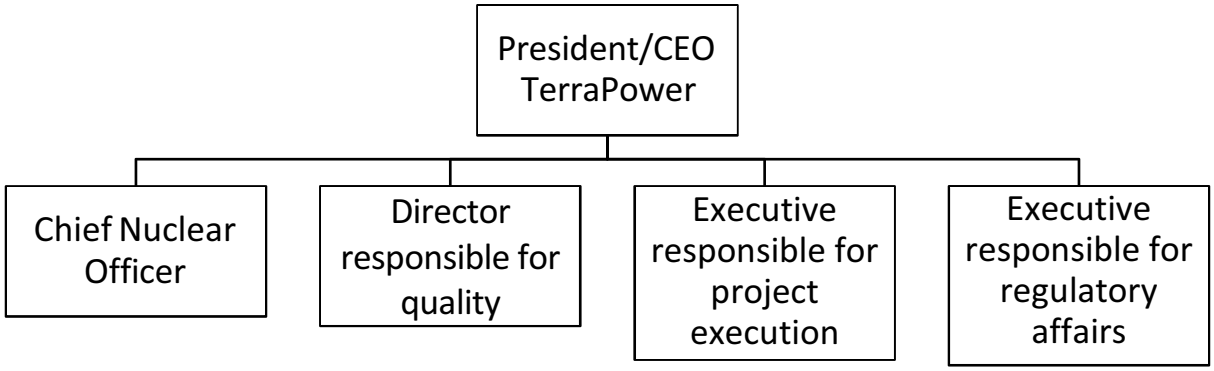



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**Figure 11.1-2 Natrium Project Organization**

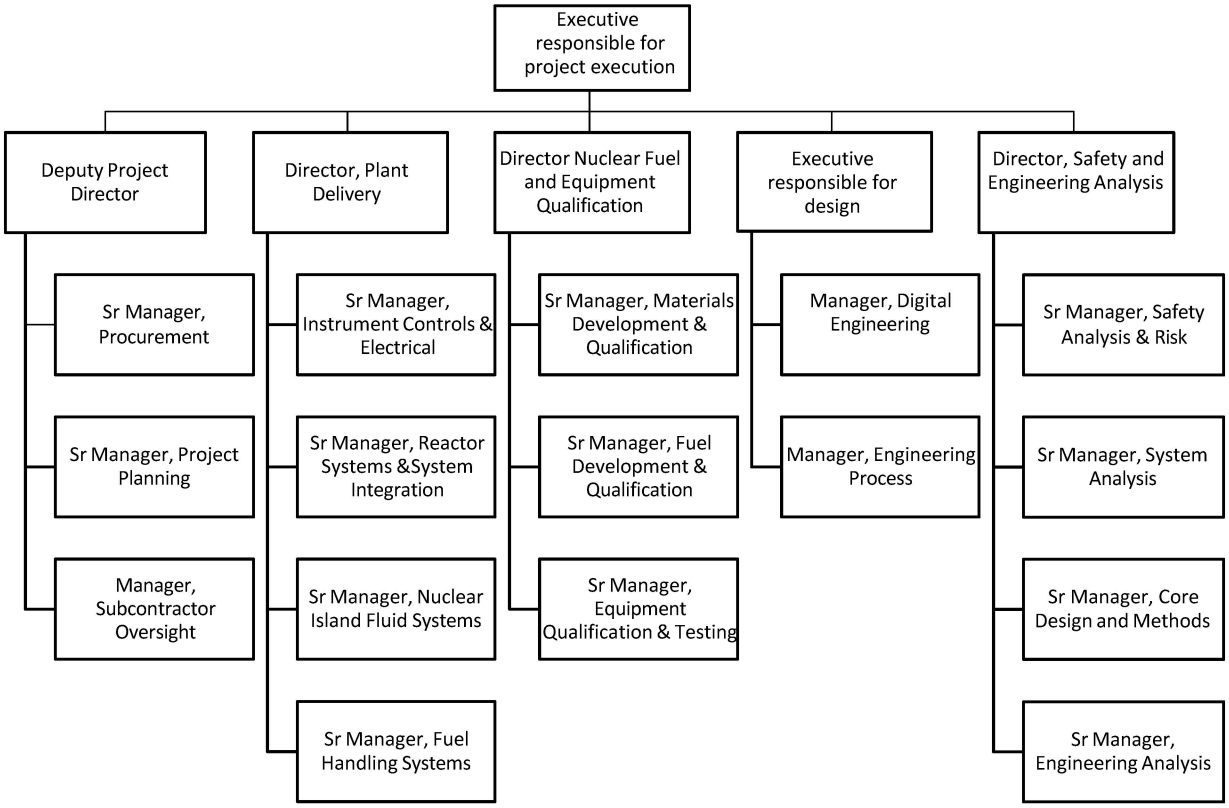



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**Figure 11.1-3 Construction Organization**

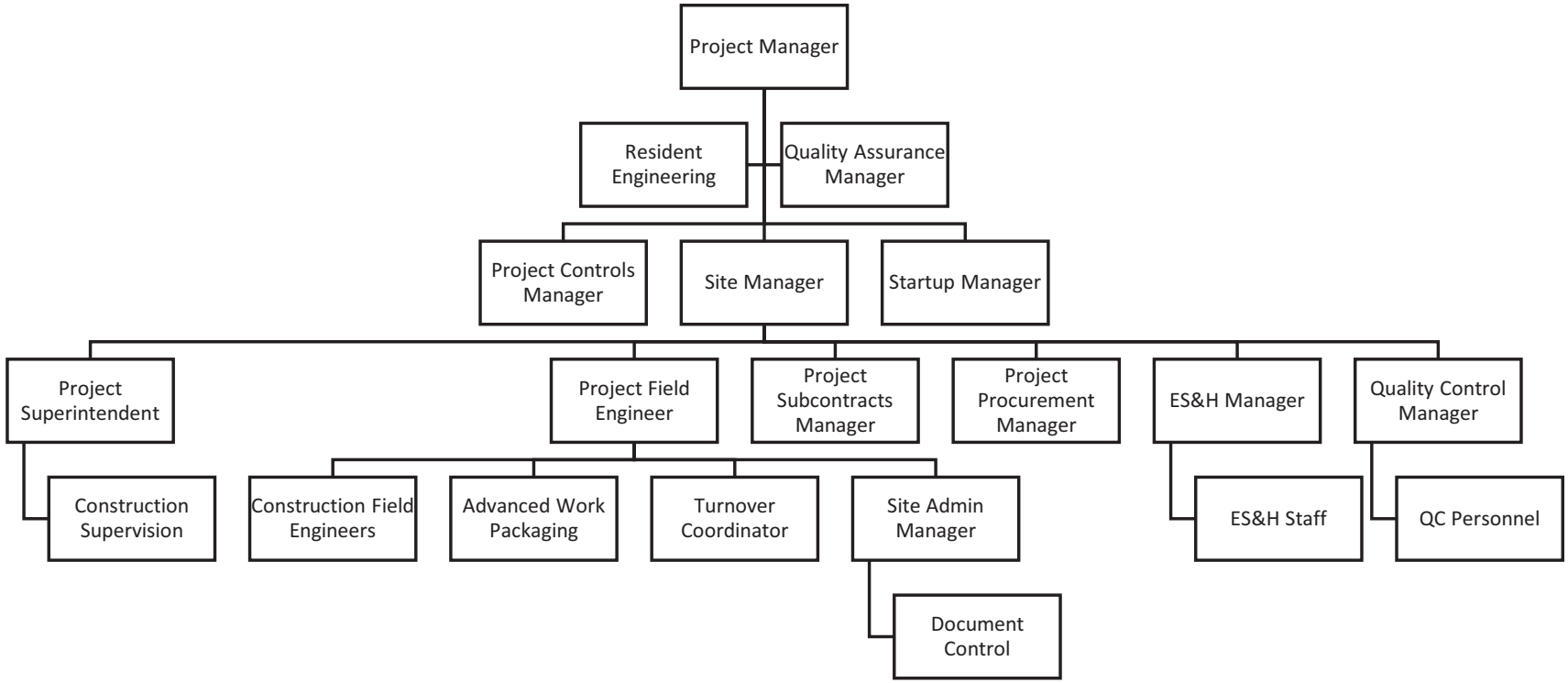




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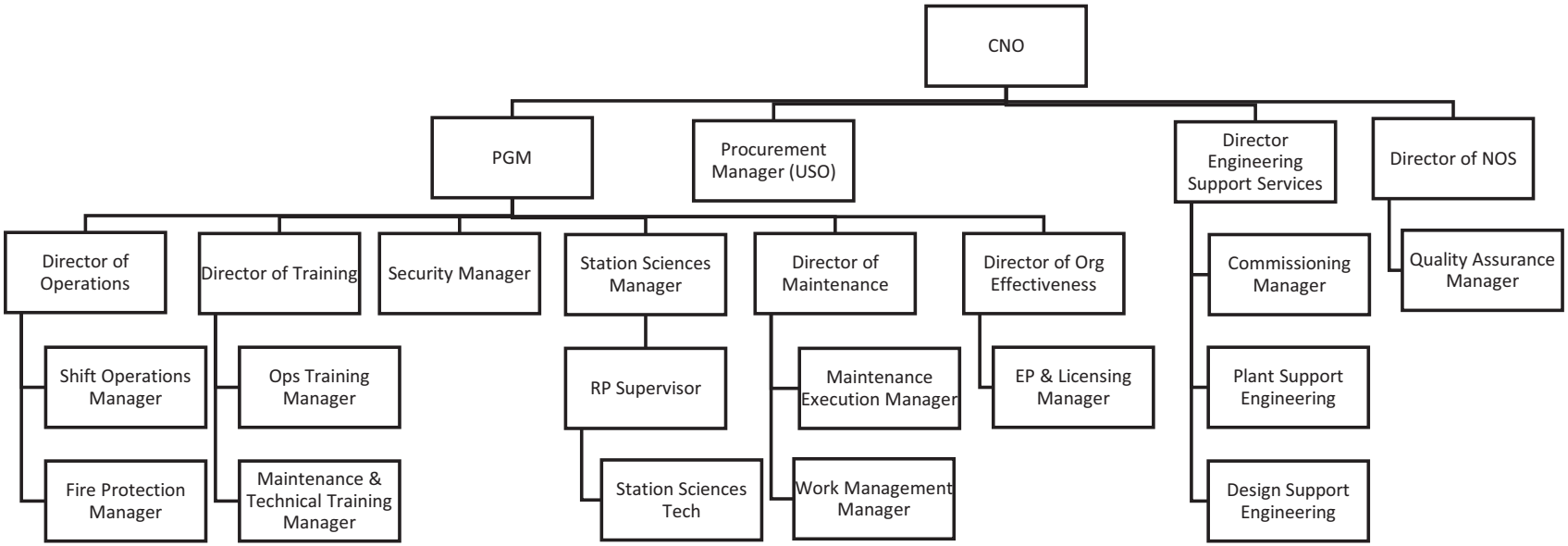
**Figure 11.1-4 Operating Organization and Technical Support**





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## 11.2 Human Factors Engineering

In accordance with 10 CFR 50.34(f)(2), Section 11.2 presents the human factors engineering (HFE) program and associated technical information for Kemmerer Unit 1. This section demonstrates the adequacy of integration of HFE requirements and analysis results into the plant design. The HFE input to the construction permit application describes the program of HFE activities and analyses informing the design of structures, systems, and components (SSCs) based on clear definition of the full set of users and a clearly defined scope of application to the Nuclear Island design, operational modes, and design stages, with focus on important human actions (HAs). The HFE content reflects the maturity of the HFE program and plant design progression appropriate to the stage of design at the time of submission. Technology is continually advancing and the complete HFE design incorporating further design development and HFE program outcomes will be provided at the operating license (OL) stage.

This CPA input accommodates humans in a modern technology-inclusive, risk-informed, and performance-based process for selection of licensing basis events (LBEs), safety classification of SSCs and associated risk-informed special treatments, and determination of defense-in-depth (DID) adequacy for non-light-water reactors. The content reflects design development as design progresses, while information provided at the OL stage will accommodate further designs and outcomes.

Section 11.2 HFE content provides a summary of the human-system Interface (HSI) design goals and bases, analyses undertaken to understand the task-based and plant-specific requirements related to task performance, the process for detailed HSI design, and activities supporting effective design implementation. The overall design and implementation process is described in detail in the Human Factors Engineering Program Plan (HFEPP), subordinate documents, and topical and technical reports. Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plant and Methodologies,” (Reference 11.2-1) is incorporated by reference into the SAR in its entirety. See Section 1.4.2 for a list of documents incorporated by reference into the SAR. The HFEPP presents the comprehensive, iterative design approach used for the development of human-centered interfaces and work environment.

Site-specific Severe Accident Management Guidelines (SAMGs) and procedures that satisfy NEI 91-04 (Reference 11.2-2) are developed and validated using guidance and emergency operating procedures (EOPs).

Section 11.2.1 provides an overview of the HFE program and outlines its activities or technical elements. The remainder of Section 11.2 provides the details, including the scope and summary of methods of the technical elements outlined in Section 11.2.1.

### 11.2.1 Management of the Human Factors Engineering Program

#### 11.2.1.1 HFE Program Goals

The high-level goal of the HFEPP is to specify a proportionate, integrated, and effective human factors (HF) program that reduces the risks and consequences influenced by human interactions. The program is developed and conducted in accordance with relevant aspects of 10 CFR 50 and Advanced Reactor Content of Application Project guidance related to organization and


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human-system considerations (Reference 11.2-3). The program also considers applicable industry guidance in NEI 18-04 (Reference 11.2-4), as endorsed by Regulatory Guide 1.233, “Guidance for a Technologically-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Application for Licenses, Certifications, and Approvals for Non-Light Water Reactors,” Revision 0, and NEI 21-07 (Reference 11.2-5). The HFEPP will be implemented via administrative procedures.

HFE program goals are the general objectives of the program, stated in human-centered terms and refined as the program develops, which are used as a basis for HFE planning, testing, and evaluation activities. HFE program goals ensure that:

- Personnel tasks are accomplished within time and performance criteria.
- HSIs, procedures, staffing and qualifications, and training variables support a high degree of operating crew situational awareness.
- Allocation of Function (AOF) accommodates human capabilities and limitations.
- Operator vigilance is ensured.
- Acceptable operator workload is met.
- Operator interfaces minimize human errors.
- Error detection and recovery capabilities are provided.

*Assumptions and Constraints* - An assumption or constraint is an aspect of the design, such as specific use of HSI technology, that is an input to the HFE program rather than the result of HFE analyses or evaluations.

*Applicable Facilities* - The HFE program addresses the main control room (MCR), Remote Shutdown Complex (RSC), Local Control Station (LCS), and Emergency Response Facility (ERF).

*Applicable HSIs, Procedures, and Training* - The applicable HSIs, procedures, and training included in the HFE program include operations, accident management, maintenance, test, inspection, and surveillance interfaces (including procedures) for Nuclear Island systems only. This includes monitoring the designs being presented by suppliers to ensure that the designs are consistent with the HFE requirements of the HFE program.

*Applicable Plant Personnel* - Plant personnel, both licensed and unlicensed, are delineated in Section 5.4 of the HFEPP. The staff members include those that perform tasks that are directly related to plant safety.

*Adopted Technology* – The proposed HSI employs digital technology to implement most of the monitoring, control, and protection functions. Standardization of hardware, software, and modularity of design is used to simplify maintenance and provide protection against obsolescence.


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The HSI activities include the development of dynamic models for evaluating the overall plant response as well as individual control systems, including operator actions. These dynamic models are used to:

- Analyze both steady state and transient behaviors.
- Confirm the design of the advanced alarm system concepts.
- Confirm the adequacy of control schemes.
- Confirm the allocation of control to a system or an operator.
- Develop and validate plant operating procedures.
- Incorporate use of simulators.

Using part-task simulation, an initial set of systems is identified through modeling, including the development of the graphical user interface (GUI). The part-task simulator is used in preliminary design and expanded to include unique design features.

As the design progresses, the part-task simulator proceeds through a series of iterative evaluations resulting in the development of a complete control room full scope simulator. Simulators are the focal point for operator evaluations and feedback checkpoints throughout the HSI design process (HFEPP Section B.3.2.2).

#### 11.2.1.2 Program Scope

The HFE scope applies to HSI components and SSCs within or that form part of facilities, systems, equipment, and components throughout the plant, including both a part-task and full-scope simulator. HSIs are defined as any region or point at which a person interacts with a system, equipment, or component. System interface means any digital and electronic Instrumentation and Control (I&C) user interfaces, as well as hardware-based user interfaces and design features on panels, equipment, and individual components. This includes HSIs within or forming part of:

- Control facilities for reactor operations (in accordance with NUREG-0700, NUREG-0711, NUREG-0800).
- Facilities for supporting response to accidents and emergencies (in accordance with Chapter 11, Section 11.3).
- Control room or stations for radioactive waste processing.
- Control room or stations supporting refueling and maintenance outage work.
- Local Control Stations (LCSs).
- Equipment-mounted and process line-mounted HSIs.
- Auxiliary and support facilities and equipment located external to the reactor building.

This includes specification to, and oversight of, HSIs that form part of SSCs supplied by external vendors to ensure that supplied design, or selection of standard equipment and components, is consistent with the HFE program requirements. The HFE program requirements will be implemented via procurement procedures.


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The HFE program also applies across the scope of users and activities that support operation, testing, inspection, surveillance, and maintenance, including but not limited to functions such as fuel handling, chemistry, radioactive waste processing, and radiation protection.

The HFE program activities consider human factors risks that could arise in phases of the plant, including:

- Design and construction
- Commissioning
- Operation
- Refueling and major refurbishment outages

The HFE program applies to the plant states including:

- Normal and abnormal operations, including radioactive waste processing
- Online and offline maintenance, testing and inspection, including refueling outages
- Postulated accident conditions and emergencies

Although the scope of the HFE program is broad and inclusive, the application of HFE to the plant operational conditions and HSI location is graded (or proportionate) as described in Section 11.2.1.7.2. The graded approach is based upon relevant approaches in NUREG-1764, “Guidance for the Review of Changes to Human Actions,” Revision 1, (Reference 11.2-6). This allows a higher level of emphasis and rigor for human interactions that are safety-significant, complex, novel, or inherently hazardous.

### 11.2.1.3 Overview of the Human Factors Engineering Program

The HFE program describes goals and scope along with items such as:

- Assumptions and constraints in conducting the program.
- Coordination of the HFE program with the overall plant design activities.
- Tools and facilities (e.g., mockups, computer simulations) used in support of the program.
- Composition, qualifications, and responsibilities of the HFE organization.
- Process and procedures followed including the process for identifying and managing technical programmatic issues.
- Documentation developed.
- Summary of how the results of the HFE analysis are incorporated into the design and operational documentation.

The HFEPP defines each of the technical elements (i.e., the specific activities that comprise the full integrated program) as outlined in the Section 11.2.1.4.


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### 11.2.1.4 Human Factors Engineering Program Technical Elements

This section and the HFEPP define each of the technical elements and the specific activities that constitute the fully integrated program. The technical elements for the HFE program are summarized below. The full description of these elements, and how they constitute a comprehensive and robust program of HFE integration across the plant design, are provided in Section 11.2.2 through Section 11.2.7.

1. Operating experience review (OER) – Identification, review, and incorporation into the design of any recommendations and learning from past events related to HFE (see Section 11.2.2). Operating experience (OE) provides information useful to the HFE program and a review of OE is part of the HFE program.

2. Functional requirements analysis (FRA) – Determination of functions required to achieve all plant goals in the plant states (see Section 11.2.3.1).

3. Allocation of Function (AOF) – Assigning the identified functions to system (technology) or human, based on respective capabilities and limitations of each (see Section 11.2.3.2).

4. Task Analysis (TA) – Identification of the tasks required to achieve the allocated functions, and decomposition into task steps to allow the identification and characterization of HSIs, personnel, locations, and support equipment (e.g., communications, lighting, personal protection) required to perform each task successfully (see Section 11.2.4).

5. Staffing Analysis – Determination of the numbers and roles of personnel required to support optimal task performance in the plant conditions, and the baseline qualifications for those personnel (see Section 11.2.3.3).

6. Treatment of Important HAs – Supporting and providing input to the safety analyses to ensure clear identification of safety-significant HAs, ensure claimed actions are achievable, and identify HSIs requiring the highest level of HFE focus and effort (see Section 11.2.4.1).

7. HSI Design – Identification and management of the set of HFE design requirements from standards, codes, and best-practice guidance, and implementation of those requirements plus results from HFE analyses into the design of HSIs, including integration of HFE specialist design support. HSI design also includes HFE testing and evaluation (T&E) activities (see Section 11.2.5). The HAs that involve interfaces with instrumentation, procedures, system controls, and communications equipment are included. It is important that the HSI is designed to minimize the likelihood of errors in comprehension and actions including a description of items such as:

    - The approach used in conducting the HSI design, including the guidance used in the design process.
    - The HSI features at a conceptual level incorporated into the design of the MCR, ERF, RSC (remote shutdown panel, remote shutdown system), and LCSs.
    - The results of early user tests and evaluations undertaken to support the HSI design.


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8. Procedure Development – Process and activities for the development of usable and validated operational documentation for plant task types (see Section 11.2.6.1). The objective of procedure development is to apply human engineering principles and guidance to develop procedures that are technically accurate, comprehensive, explicit, easy to use, and validated. To support these objectives, a procedure development methodology is developed. The procedure development methodology specifies the inputs and process for procedure development. The procedure development methodology will be implemented via administrative procedures.

9. Training Program Development – Process and activities for the development of relevant and validated training content, optimized for the plant design, operational documentation, and baseline personnel qualifications and attributes (see Section 11.2.6.2). Training of plant personnel is an essential factor in ensuring safe and reliable operation of nuclear power plants. The training program provides reasonable assurance that plant personnel have the skills, knowledge, and abilities to properly perform their roles and responsibilities adhering to the safety culture of the plant organization. The training program will be implemented via administrative procedures.

10. Verification and validation (V&V) – Detailed, staged set of activities to provide assurance of the correct and sufficient implementation of HFE requirements in the design, and the appropriate design to support required tasks (see Section 11.2.6.3). The methodology for verifying and validating that the HFE design is consistent with regulatory guidance and current accepted HF principles.

    Verification is the process of determining and documenting that an implemented design (e.g., a product, process, procedure, or method) meets its specifications. Verification answers the question: Was the design implemented appropriately?

    Validation is the process of determining and documenting that the design effectively serves the purpose for which it was intended. Validation answers the question: Was the appropriate design implemented?

    V&V, in the context of HFE, assures that the design of HSIs:

    - Are complete
    - Conform to HFE principles
    - Are operable
    - Are free of safety issues and human performance issues
    - Are correctly implemented in a final, as-built form

    HSI are the controls, displays, annunciators, procedures, data processing, and communication systems necessary to accomplish operation and maintenance tasks and actions as defined by TA, EOPs and other procedures, probabilistic risk assessment, and human reliability analysis.


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HF V&V is a verification and validation of plant HSIs and the working environment where HSIs are used. HF V&V is conducted in two major activities: HF verification (design and task support) and Integrated System Validation (ISV). HF V&V is conducted by a team of HFE engineers independent from the design development process. HF V&V is performed on a configuration-managed, baselined design and includes the following areas:

- Control area and equipment layout
- Panel and console dimensions
- Hardware-based indications, controls, alarms, and panel layout
- Software-based HSI displays, controls, alarms, and display layout

HF design verification verifies that the HSIs, as defined and baselined in the HSI inventory and characterization, are evaluated against the HFE design requirements contained in the HFEPP as well as topical and technical reports. HF Task Support Verification (TSV) verifies that the HSIs, as defined and baselined in the HSI inventory and characterization, include the necessary features (e.g., controls, information displays, and alarms) required to support tasks and that there are no unnecessary features.

HF validation HFEPP (Section 5.10) ISV is an integrated, dynamic, performance-based test activity in which participants are subjected to a set of simulated scenarios that represent a realistic, challenging, and generalizable set of conditions to ensure that the integrated HSI supports safe operation of the plant.

11. Design Implementation – Support and monitoring of the design from concept to a realized constructed plant, including integration with configuration control to ensure no loss of integrity of the HFE program goals throughout fabrication and construction (see Section 11.2.6.4) and HFEPP (Section 5.11).

12. Human Performance Monitoring – Continuous monitoring of user task performance throughout the lifetime of the plant to ensure optimum plant and organizational design and identify early trends and issues that require HFE design improvements (see Section 11.2.7).

The scope and nature of the HFE program, the HFE organization undertaking the program, and the technical elements that comprise it align with regulatory requirements and standards and guidance to provide assurance that HFE has been suitably and sufficiently integrated into the plant design.

Table 11.2-1 illustrates the alignment of the HFE section structure with NUREG-0711 and Development of HFE Review Guidance for Advanced Reactors, Technical Letter Report No. F0028-04 (Reference 11.2-7). The introductory material delineates the technical elements that are fulfilled and the section numbers listed give detailed information about those technical elements. Table 11.2-2 further defines and delineates the technical elements showing the activities accomplished within each element.


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The preliminary safety analysis report (PSAR) and HFE program technical elements inform the OL stage and the overall span of HFE activities. Although arranged somewhat differently in the PSAR section and the HFEPP, the required elements are accommodated and addressed. The elements are performed in an iterative manner with activities and outputs progressively evolving with the design and related safety analyses.

These elements inform one another. The elements also inform and are informed by plant design and safety analyses, and are aligned with design and engineering processes and requirements, best human factors practices, guidance, and regulations.

#### 11.2.1.5 Team and Organization

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference into the safety analysis report (SAR), describes the Team and Organization. See Section 1.4.2 for a list of documents incorporated by reference into the SAR.

#### 11.2.1.6 Cross-Discipline Support and Integration

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the Cross-Discipline Support and Integration.

#### 11.2.1.7 Process and Procedures

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the Process and Procedures.

##### 11.2.1.7.1 Coordination and Documentation of Activities

The HFE program is planned and conducted in accordance and alignment with overarching design and quality program processes and procedures, within accredited quality management systems. The work is performed in an integrated manner with HFE as an equal design discipline, whose cross-cutting requirements and support are embedded within other design disciplines, with activities, inputs, outputs, and dependencies coordinated through a detailed schedule and associated schedule management processes. The schedule includes activities and deliverables for the disciplines and orders them with logical connections to ensure they are completed in the required sequence.

To help ensure cross-discipline communication and coordination, activities include scheduled periodic formal design reviews conducted by representatives of each discipline. Additionally, deliverables are completed in accordance with a deliverable standard, which specifies the required content from the related disciplines, dictates the format for consistency and quality, and specifies the required discipline reviewers for each document. This includes other disciplines reviewing and incorporating outputs from HFE documentation and the HFE team reviewing and incorporating outputs from other disciplines.



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The documentation for the HFE program uses a standard design process that includes documenting internal design records to capture inputs and outputs, as well as providing the basis for formal deliverables. The information in the design records is incorporated into the design by HFE and other disciplines as appropriate and in accordance with the work breakdown structure. A full description of the management and integration of HFE activities within the project is described in the HFEPP.

Using the project processes and HFE coordination measures described in the HFEPP, the design activities related to HFE are conducted and documented such that design basis, input, and rationale for design and analysis is provided for HFE design decisions and analysis results. HFE requirements and recommendations are addressed by the requirements management process and either incorporated into the design directly or via alternate solutions agreed as acceptable by HFE, or if not implemented, tracked as an HFE issue as described in Section 11.2.1.8.

#### 11.2.1.7.2 Risk-Based Graded Approach

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the risk-based graded approach.

#### 11.2.1.7.3 Requirements Management

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the requirements management.

### 11.2.1.8 Issue Tracking

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the Human Factors Engineering Issue Tracking System (HFEITS).

### 11.2.1.9 Design Goals and Design Bases

The primary goal of HSI designs is to facilitate safe, efficient, and reliable operator performance during the phases of normal plant operation, abnormal events, and accident conditions. To achieve this goal, information displays, controls, and other interface devices in the control room and other plant areas are designed and implemented in a manner consistent with best HFE practices. Further, the following specific design bases are adopted:

- HSI design promotes efficient and reliable operation through application of automated operation capabilities.
- The layout reflects I&C restrictions.
- Systems monitoring and control capability is provided in full compliance with regulations regarding divisional separation and independence.
- The principal functions of the Safety Parameter Display System as required by Supplement 1 to NUREG-0737 are integrated into the HSI design (HFEPP Section B3 Item 7).


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- Accepted HFE principles are used for the HSI design in meeting the requirements of the HFEPP.
- The HFE Style Guide is based on NUREG-0700.

Design acceptance criteria described in the HFEPP are used to govern and direct the plant HSI design implementation. These detailed design acceptance criteria encompass the set of necessary and sufficient design implementation-related activities. These are required to maintain the implemented HSI design in compliance with accepted HFE principles and digital electronics equipment and software development methods.

As part of the detailed design phase, operator TA is performed as a basis for evaluating details of the design and specifying HSI requirements. The evaluation of the integrated control room design includes the confirmation that the MCR design features are implemented.

**Design Changes**

After the plant design is finalized, it is possible that changes to the HSI may be proposed. Plant modifications occurring prior to the OL that impact or generate high risk HAs are analyzed to determine impact on staffing and qualifications. For these plant modifications, the HFE program, in conjunction with the proposed OL holder, specifies the involvement of experienced plant personnel to provide reasonable assurance that user perspective is considered in establishing modification requirements and evaluating the design process outputs. For example, modifications and updates of the plant include consideration of:

- User understanding of how plant systems are structured and behave.
- Task demands and constraints of the existing work environment.
- Impacts on existing work processes.

The impact of these design modifications on the organization are examined by screening or analysis for their impact on implementation and goals for the construction permit holder. The staffing examination reevaluates the number and background of personnel for the full range of plant conditions and tasks including operational tasks (normal, abnormal, and emergency), plant maintenance, and plant surveillance and testing when implementing a modification to the plant base design.

11.2.1.10 Planning, Development, and Design

An integrated program plan to incorporate HFE principles and to achieve an integrated design of the I&C systems and HSI is described in the HFEPP. The HFEPP presents a comprehensive, synergistic design approach with provisions for TA and HF evaluations. Also included are formal decision analysis procedures facilitating selection of design features, which satisfy top-level requirements and goals of individual systems and the overall plant.

The HFEPP and any associated procedures provide guidance for the conduct of the HSI design development activities, including:

- Definition of the design features of the control room HSI.


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- Definition of the inventory of controls and instrumentation. This is necessary for the operators to follow the EOPs and SAMGs to complete the important operator actions described in the PRA.

#### 11.2.1.11 Design Features

The control room HSI design contains a group of design features which form the foundation for the detailed HSI design. The development of the control room HSI design features is accomplished through:

- Consideration of existing control room OE.
- Review of trends in control room designs and existing control room data presentation methods.
- Evaluation of new HSI technologies, alarm reduction, and presentation methods.
- Validation testing of a dynamic control room prototype.

The prototype is evaluated under simulated normal and abnormal reactor operating conditions by experienced nuclear plant control room operators with oversight and input from the HFE team. Following the completion of the prototype tests and results analysis, the control room HSI design features are finalized.

#### 11.2.1.12 Inventory of Controls and Displays

The emergency procedures guidelines (EPGs) and the important operator actions identified in the PRA, presented in Chapter 3, provide the bases for an analysis of the information and control capability needs of the MCR operators based upon the operation strategies. This analysis defines a minimum set of controls, displays, and alarms that allow the operators to perform the actions specified in the EOPs and the important operator actions identified in the PRA.

#### 11.2.1.13 Design Implementation Process

The process by which the detailed equipment design implementation of the HSI is completed is described in the HFEPP. This process builds upon the HSI design features described in the HFEPP and subordinate methodologies, plans, and processes. Embedded in the process are several conformance reviews in which various aspects and outputs of the process are evaluated against established design acceptance criteria.

Design implementation addresses the final as-built implementation of the HFE design requirements into the design. The following activities are conducted during design implementation:

- Confirm that the final HSIs, procedures, and training as built conform to the design resulting from the HFE design process and V&V activities.
- Perform final procedure validation on the physical plant hardware.


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- Verify aspects of the design that may not have been evaluated previously in the V&V process. This includes any hardware or software, new or modified displays that were absent from the simulator-based integrated V&V process, and any physical or environment (e.g., noise, lighting) differences between those present at the V&V process and the as-built control areas.
- Verify resolution of remaining human engineering discrepancies (HEDs) and open items from the Human Factors Engineering Issue Tracking System (HFEITS).
- Verify HFE application levels are based on the final set of inputs.

Design implementation activities are performed by the HFE team along with operations and engineering. HFE issues are resolved before declaring a system available for service.

#### 11.2.1.14 Control Room Design Features

The MCR design features are demonstrated, through broad scope control room dynamic simulation tests and evaluation, to satisfy the plant HSI design goals and design bases. Validation of the implemented MCR design includes evaluation of the design features performed as part of the design implementation process as defined by the design acceptance described in the Principal Design Criterion (PDC) and Chapter 7.

#### 11.2.1.15 Remote Shutdown Complex

The RSC provides a means to safely shut down the plant from outside the MCR. It provides control of the plant systems needed to bring the plant to hot shutdown, with the subsequent capability to attain safe shutdown if the control room becomes uninhabitable.

The RSC design is described in Section 7.6.7.2. Parameters for safe shutdown that are displayed or controlled from the MCR are also displayed on the Post-Accident Monitoring (PAM) panels in the RSC.

#### 11.2.1.16 Detailed Design of the Operator Interface System

The design features of the MCR HSI, discussed in the HFEPP and Chapters 6 and 7, provide the framework for the detailed equipment hardware and software designs developed following the design and implementation guidance and processes.

As part of the HFEPP, discussion of the HSI design and implementation plan elements and detailed acceptance criteria are specified. These criteria are used to govern and direct HSI design implementation. These detailed design acceptance criteria encompass the set of necessary and sufficient design implementation-related activities required to maintain the implemented HSI design in compliance with accepted HFE principles and accepted digital electronics equipment and software development methods.

Also, as part of the detailed design phase, operator TA is performed as a basis for evaluating details of the design implementation and that HSI requirements are specified. The evaluation of the integrated control room design includes the confirmation of the MCR design features.


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The general development of key topic plans, analyses, and evaluations are identified and delineated in this section. The HFE program goals and scope (e.g., control room(s), LCSs) are described. In addition, the process used in conducting the HFE program is described along with information on the team of HFE experts conducting the program, including items such as:

- Assumptions (e.g., staffing levels, types of I&C used in the design) used in conducting the program.
- Coordination of the HFE program with the overall plant design activities, including coordination with the plant risk analysis.
- Tools (e.g., mockups, computer simulations) used in support of the program.
- Design team organization and responsibilities.

The design activities begin with the establishment of the design team. Within the design team, the HFE group, with the support of other staff, prepares the various implementation plans required to support the HSI design activity and manages the activity through the remaining steps to the final validation of the implemented design. A composition of experienced individuals, whose collective expertise covers a broad range of disciplines relevant to the design and implementation activity, is maintained for the design team throughout the process.

The design team is comprised of at least the following areas of expertise:

- Technical Project Management
- Systems Engineering
- Nuclear Engineering
- Instrumentation & Control Engineering
- Architect Engineering
- Human Factors
- Plant Operations
- Computer Systems Engineering
- Plant Procedure Development
- Personnel Training
- System Safety Engineering
- Reliability, Availability, Maintainability, and Inspection Expertise
- Quality Assurance

A design team coordinates the design of the MCR, RSC, and local control stations. This team is made up of members from the core HFE design team and extended design team and includes involvement by USO staff, including engineering and maintenance personnel and operations staff familiar with plant normal, abnormal, and emergency operating procedures.

The duties of the design team establish and perform the activities as defined. The design team specific duties are to guide and oversee the design implementation activity and to assure that the execution and documentation of each step in the activity is carried out in accordance with the established program and procedures. The design team has the authority to ensure that all its


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areas of responsibility are accomplished and to identify problems in the implementation of the HSI design. The design team has the authority to determine where its inputs are required and to access work areas and design documentation. The design team also has the authority to control further processing, delivery, installation, or use of HSI products until the disposition of a non-conformance, deficiency, or unsatisfactory condition is achieved and hand-over to the license holder is accomplished.

The design team is responsible for:

- The development of the HFE plans and procedures.
- The oversight and review of HFE design, development, test, and evaluation activities.
- The initiation, recommendation, and provision of solutions through designated channels. for problems identified in the implementation of HFE activities.
- Verification that solutions to problems have been implemented.
- Assurance that HFE activities comply with the HFE plans and procedures.
- Ensuring that the activities of the quality plan agreed to with USO are followed.
- The methods for reviewing OE.

Independent reviews of the work of the design team are conducted in accordance with quality assurance (QA) plans to provide:

- Validation that the system and its components perform their intended functions.
- Verification that the individual steps in the process have been properly carried out.

### 11.2.2 Operating Experience Review

The HFE program includes the early review of OE to identify applicable HFE issues related to process or personnel safety that are resolved through design improvements. The issues and lessons learned from the operating experience review (OER) provide a basis for improving the plant design in a timely way (i.e., at the beginning of the design process). The OER will be implemented via administrative procedures.

OE provides information useful to the HFE program and a review of OE is part of the HFE program. The results of the review of OE applicable to the Kemmerer Unit 1 design are described, including items such as:

- The criteria used to identify the OE applicable to the design
- The OE reviewed, including recognized industry HFE issues (e.g., Three Mile Island issues, Fukushima Daiichi accident, OE reports – NUREG-1275 series)
- The process used in conducting the review (e.g., documentation reviews, interviews)
- The results of the OERs, including how the design is addressing the HF issues identified by the OERs

As part of the HFE process, OERs are conducted to identify HFE-related safety and availability issues. The OER obtains and analyzes information on the past performance of predecessor designs. In the case of a new plant such as Kemmerer Unit 1, the evolution of the design comes


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from years of nuclear experience and improvements. The issues and lessons learned from previous OE provide a basis for improving the plant design and the HSI system in a timely way (i.e., at the beginning of the design process).

History has demonstrated that valuable lessons can be learned from incidents and accidents. This was demonstrated after the accident at Three Mile Island nuclear power plant in the United States of America in 1979, when far-reaching follow-up actions were taken to minimize the risk of a recurrence and to improve the HSI and procedures for accident management. The accident at Chernobyl demonstrated that the lessons from the Three Mile Island accident had not been acted upon in the USSR: in particular, the importance of systematic evaluation OE; the need to strengthen the on-site technical and management capability, including improved operator training; and the importance of the man-machine interface (IAEA, INSAG-7 (Reference 11.2-8)).

The analysis of OE events to understand the role of HAs provides support during the design for engineering decisions regarding the HSI to enhance safety. The documentation provides a basis for design decisions, and a starting point for developing some performance indicators and an experience review system for the operating plant.

#### 11.2.2.1 Objectives and Scope

The objective of the OER is to obtain information and lessons learned from experience to support design of plant systems. OE related to the following areas is considered in the development of the design:

- Predecessor plant(s) and systems
- Experience in industries with applicable SSCs
- Applicable industry HSI design experience
- Risk-important HAs
- Specifically identified applicable industry issues
- Issues identified by predecessor or similar plant personnel
- Specifically identified positive features that support task performance

#### 11.2.2.2 Methodology

Existing and new OE is reviewed by HFE, and relevant, applicable problems, issues, and positive insights are identified and addressed throughout the design process. The OER information is made available to design engineers to support development of design features that reduce human error. Likewise, positive features of previous designs are communicated so that they are retained.

The OER process methodology establishes the process and procedures for evaluating operating, design, and construction experience, thus ensuring that the applicable important industry experiences are provided in a timely manner to those designing and constructing the plant, as required by 10 CFR 50.34(f)(3)(i).


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The methods for identifying the OE include identifying:

- OE for the selected HFE technology components from relevant predecessor plants and systems
- Risk-important HAs and recognized industry issues
- Issues identified by plant personnel

The methods for analysis and evaluation of OE include:

- Use of summarized issues from industry sources
- Development of insights from event reviews
- Development of design solutions to reduce human error

The method for keeping track of the process includes the use of the HFEITS, which permits tracking and review of the issues identified and addressed in the design.

#### 11.2.2.2.1 Predecessor Plants and Systems

Experience from the nuclear industry is considered in the plant design. The OE information is made available to design engineers to support development of design features that are expected to reduce human error. Likewise, positive features of previous designs are identified, evaluated, and retained. The baseline review record includes industry experience related to the plant and systems of Kemmerer Unit 1.

#### 11.2.2.2.2 Risk-Important Human Actions

The OER process addresses the risk-important HAs from predecessor plants and other sodium-cooled relevant facilities such as the Fast Flux Test Facility (Hanford Nuclear Reservation) including the:

- Identification of risk-important HAs in the PRAs and human reliability analyses.
- Determination if the HAs are still risk-important to the design via the design level Kemmerer Unit 1 PRA output.
- Application of HAs, identify scenarios where these actions are called for in predecessor operations.
- Integration of aspects of the predecessor design that assured HA success.
- Identification of insights related to needed improvements in human performance if errors have occurred in task execution.

The OER process identifies and documents OE related to risk-important HAs for HAs in the Kemmerer Unit 1 plant found as different from those of the predecessor facilities.

#### 11.2.2.2.3 HFE Technology

The OER associated with proposed HFE technology in the plant design is described. For example, if a computer-operated support system, computerized procedures, or advanced automation are planned, HFE issues associated with such use are described.


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#### 11.2.2.2.4 Recognized Industry Issues

The process for recognizing how industry HFE issues are addressed in the design includes categories identified in NUREG/CR-6400. The categories are:

- Unresolved safety issues/generic safety issues
- Three Mile Island, Chernobyl, Fukushima Daiichi issues
- NRC generic letters and information notices
- Reports of the NRC Office for Analysis and Evaluation of Operational Data
- Low power and shutdown operations
- OERs

#### 11.2.2.2.5 Issues identified by Plant Personnel

The OER plan includes the use of plant personnel interviews to supplement OE related to plant operations and HFE design in predecessor plants and systems. Personnel interviews include the following:

- Plant Operations
- Normal plant evolutions (e.g., startup, full power, and shutdown)
- Plant failures and accidents
- Reactor shutdown and cool-down using remote shutdown system
- HFE Design Topics
- Alarm and annunciation
- Displays
- Controls and automation
- Information processing and job aids
- Real-time communications with plant personnel and other organizations
- Procedures, training, staffing, qualifications, and job design

### 11.2.2.3 Results

The results of the OER are summarized in an OER report. The report provides the OER process description along with the review methods that were used. The results include:

- Sources of OER information
- Summaries of OER issues and improvements
- List of issues from the OER requiring special attention in the design process based on the grading process
- Information gathered from personnel interviews conducted at predecessor plants


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Implementation of the OER results into the design is managed and tracked through the assignment of, and reference to, a unique OE identification number. Communication and allocation of the results to the appropriate design team and design requirements document is managed by the HFE team.

### 11.2.3 Safety Function Review

HFE applies to the plant conditions; however, the application of HFE to the phase and location scopes is graded (or proportionate) to apply a higher level of emphasis and rigor for important human interactions that are safety critical or hazardous.

The safety functions that are performed to maintain the plant within the safety envelope are described in the PRA and Chapters 3 and 4 where design basis accidents (DBAs) are identified, analyzed, and described including items such as:

- Definition of the safety functions and the SSCs used to accomplish them.
- Identification of the most important HAs resulting from the plant risk analysis.
- AOFs to HAs, automation, or a combination of the two.
- Methodology used in making the allocations.
- Technical bases for the allocations made.
- Comparison of human versus automation allocations made on previous predecessor plants.

Nuclear safety focuses on unintended conditions or events leading to radiological releases from authorized activities. It relates mainly to intrinsic problems or hazards.

As in other industries, the design and operation of nuclear power plants aim to minimize the likelihood of accidents and avoid major human consequences if they occur. The consequences of an accident or terrorist attack are minimal compared with other commonly accepted risks. Radiological effects on people from any radioactive releases are avoided through the application of HF guidelines, principles, and methods to plant design and operation. HFE integrates outcomes of safety function analysis, risk assessment, and PRA into the design of plant SSCs, actions allocated to humans and machines, and the methodologies used to determine and implement best HFE practices. Where HAs are used to perform PRA safety functions, the description of controls and displays address human factors considerations found in OERs, TA, HSI design, and V&V.

Objectives and Scope

The important HAs are determined using both deterministic and probabilistic means including identification of the HAs that might cause or contribute to the cause of postulated initiating events (PIEs). Inclusion in the deterministic safety analysis, or other identified contribution to risk, determines the risk level in the nuclear safety category for determining the initial HFE application level applied to HFE activities. Comprehensive, systematic identification and substantiation of HAs claimed within the safety analyses, coupled with the risk-based graded approach described in Section 11.2.1.7.2, ensures that HSIs and tasks associated with important HAs are analyzed and designed with a full detailed and robust HFE effort.


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Methodology

Human reliability analysis is conducted by the risk and reliability team, and the PRA is an integral part of the development of a complete HFE and HSI design. Human reliability analysis seeks to evaluate the potential for, and mechanisms of, human error that may affect plant safety. The core HFE team presented in the HFEPP contributes to the human reliability analysis by participating in the definition of the task for the human reliability analysis. In addition, the HFE design (TAs, HSIs, procedures, and training) is an input to the definition of performance shaping factors for the human reliability analysis.

11.2.3.1 Functional Requirements Analysis

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the methodology for functional requirements analysis (FRA).

11.2.3.2 Allocation of Function

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the methodology for AOF.

11.2.3.3 Staffing and Qualification

Plant staff and their qualifications are important considerations throughout the design process and in HFE design. The planned initial staffing level is established based on experience with reference plants, staffing goals (such as optimizing the staffing levels and qualification), initial safety function allocation, TA, and regulatory staffing requirements for nuclear reactors. Staffing and qualifications plans are used to examine the Kemmerer Unit 1 assumptions during Task Analysis (TA), human reliability analysis, and HSI design.

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the methodology for the Staffing Analysis Plan.

11.2.4 Task Analysis

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the methodology for TA.

11.2.4.1 Treatment of Important Human Actions

HFE applies to the plant conditions; however, the application of HFE to the phase and location scopes is graded (or proportionate) to apply a higher level of emphasis and rigor for important human interactions that are safety critical or hazardous.


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Consideration and integration of HFE within the safety analyses, and consideration of the results and assumptions of the safety analyses within the other HFE activities, both comprise the technical element of treatment of important HAs. The set of activities supporting this HFE technical element is developed based on requirements for HFE expertise in safety analyses per IAEA SSG-51 (Reference 11.2-9).

#### 11.2.4.1.1 Objectives and Scope

HFE and review of safety analyses outcomes provides assurance that the full set of HAs that are safety-significant are explicitly identified, characterized, and substantiated and achievable within the required task performance criteria.

#### 11.2.4.1.2 Methodology

The safety functions that are performed to maintain the plant within the safety envelope described in the PRA and Chapter 3 safety analysis are identified, analyzed, and described including items such as:

- Definition of the safety functions and the SSCs used to accomplish them.
- Identification of the most important HAs resulting from the plant risk analysis.
- AOFs to HAs, automation, or a combination of the two.
- Methodology used in making the allocations.
- Technical bases for the allocations made.
- Comparison of human to automation allocations made on previous predecessor plants of similar design.

Human reliability analysis is conducted by the risk and reliability team and is an integral part of the development of a complete PRA. Human reliability analysis seeks to evaluate the potential for, and mechanisms of, human error that may affect plant safety. The core HFE team presented in the HFEPP contributes to the human reliability analysis by participating in the definition of the task for the human reliability analysis. In addition, the HFE design (TAs, HSIs, procedures, and training) is an input to the definition of performance shaping factors for the human reliability analysis.

#### 11.2.4.1.3 Results

The results of risk level determination and HFE application-level determination are captured in the Results Summary Report (RSR). The RSR is used to identify the HF activities and level of effort for each system for input into the detailed integrated schedule.

A risk-based approach to HFE design requirements, task support requirements, and testing methodologies is also applied to security considerations for the plant design. While the same general HF methodologies and tools described within this section are also applied to security, the specific details of security risk-rating, credited human actions, security success criteria, and testing scenarios are detailed in a separate document, Security Systems Conceptual Design.


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The outputs from the activities that form this technical element include:

- Design records capturing results of safety analyses reviews.
- Design records capturing HFE qualitative human error analysis and other substantiation of claimed HAs in the various safety analyses.
- HA claims database containing important HAs and critical information for each, allowing for adequate and robust considerations in the design.
- HFE Safety Analysis summary report, which summarizes the methods, results, and issues from each of the above outputs.

The results are communicated directly to the appropriate design, safety analysis, and other HFE team members performing related technical elements to ensure timely consideration of the full set of safety-significant HAs in their respective activities. The outputs from each activity and the HA claims database capture the source of the identified HAs within the safety analysis to allow full traceability from the origin, through HFE safety analysis and forward, to any design requirements or safety analysis modifications.

In addition to specific HFE safety analysis activities described in this section, the full complement of safety analysis outcomes, including the PRA and human reliability analysis, informs and acts as input to the HFE technical elements.

### 11.2.5 Human-System Interface Design

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the methodology for HSI design.

### 11.2.6 Procedures, Training, and Verification and Validation

#### 11.2.6.1 Procedures

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the development of procedures.

#### 11.2.6.2 Training and Qualification Program Development

Training and qualification of plant personnel is an important factor in ensuring safe and reliable operation of nuclear power plants. A well-developed training program and personnel qualification reflective of the plant design, and assumptions and findings of HFE analyses, provides assurance that plant personnel have the knowledge, skills, and ability to properly perform their roles and responsibilities.

Training program development is coordinated with the other elements of the HFE program, for example by using HFE TAs to conduct a systematic analysis of job and task requirements. The program development is conducted by the training team, in accordance with a plan that provides


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the methods and framework for ensuring the program meets its requirements and technical basis. The HFE team provides inputs to the training analysis activities and provides support to the qualified and experienced training team in conducting the training program activities.

11.2.6.2.1 Objectives and Scope

The aim of the training program development is to systematically incorporate information from the other HFE design tasks to support implementation of personnel training. The training program development process is intended to produce a program that:

- Identifies performance requirements of a job or duty area relating to licensed activities.
- Defines and documents the training based on a TA that provides the information to establish the skills, knowledge, and abilities to perform each task.
- Ensures that the training is designed, developed, and implemented to meet the qualification requirements.
- Ensures instructors meet and maintain documented qualification requirements, particularly in areas of subject matter expertise and instructional skills.
- Ensures that formal evaluation methods are used to confirm and document workers' qualifications.
- Implements a change management control system that systematically identifies changes to tasks and task lists for revisions of training.
- Ensures continuing training is provided as deemed necessary through training needs analysis.
- Evaluates training regularly and incorporates the results of the evaluation into a training improvement process.
- Ensures that workers records are established and maintained.
- Ensures that workers have a level of training related to nuclear safety corresponding to their position including, but not limited to, radiation safety, fire safety, onsite emergency training, and conventional health and safety.

The training and qualification program development includes the following fundamental activities:

- Analysis
- Design
- Development
- Implementation
- Evaluation

The overall scope of the resulting training program includes the following:

- Categories of personnel conducting tasks within the plant, including the full range of job roles whose actions may affect plant safety.
- The full range of plant conditions (normal operational, outage, abnormal, accident, and emergency).


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- Activities conducted throughout the plant (e.g., operations, radwaste processing, outage refueling, online and offline maintenance, testing, and inspection).
- The full range of plant functions and systems.
- The full range of relevant HSIs.

The scope of the training program development plan does not include the specific requirements for certification of plant personnel. The requirements for certification are incorporated as part of the overall program; however, they do not need to be derived through the defined development process since they are already specifically defined. The final training program specific to the operational goals of the plant and developed from the HFE and design inputs, as described in this section, are augmented by the general and specific training and certification requirements.

11.2.6.2.2 Methodology

The training and qualification program development follows the fundamentals of the systematic approach to training method. The development process as designed ensures compliance with the requirements.

The Analysis phase provides the identification of training needs, tasks, or competencies required for training and the knowledge, skills, and abilities required to perform a specified job position based on assigned tasks. Tasks that support plant functions are identified as part of detailed and basic TA, described in Section 11.2.4. Tasks are selected for training based on difficulty, importance, and frequency (DIF) analysis. Depending on the DIF ranking, a decision is made to determine if initial training and periodic retraining is needed. This evaluation of training tasks is the training equivalent to grading HAs. The results of the TA, including identification of critical steps, inform the DIF analysis and resulting rankings.

Tasks that are selected for training are then analyzed to determine the required skill, knowledge, and attributes. The knowledge, skills, and attributes necessary for each job position, including entry-level education, training, and experience, is established to support training design. Any changes to the iterative HFE or system design inputs to the analysis phase are required to be assessed for impact on the training analysis.

During the design stage, learning objectives are developed and a description of the plan for training, including purposed methods and settings, is established. Specifically, the design stages include the following activities:

- Determine the scope, purpose, and timeframe of the training.
- Determine the ideal training environment.
- Select training methods and instructional strategies in accordance with the environment.
- Determine and group the job role skill, knowledge, and attributes addressed by each training module.
- Determine the final and partial learning objectives for each training module, including defining performance statements, conditions statements and performance standards.
- Prepare the table of contents and scope for each training module. The scope includes the number and type of documents developed in the next phase.


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- Prepare master training procedures and formats to ensure consistency across the course materials.
- Prepare the training plans for each job position. Plans comprise the learning objectives, contents, learning activities, training equipment, and a list of materials needed for training, including guidance for their use.

The completion of the design stage establishes the input that is needed for the development stage.

In the development stage, detailed lesson plans and instructional materials are created, including any on-the-job training documents, and knowledge and performance assessment tests are established. The materials developed must incorporate the required features specified in the design stage. The materials are developed such that they have the following attributes:

- Course material content supports mastery of the subject learning objectives.
- Course materials are structured to provide consistent presentation.
- Course material presentation sequence supports effective learning.
- Course materials support successful presentation in the specified venue(s) the course is to be provided.
- Instructor certifications and training required to present training is specified for each course and supports successful presentation in the required venue. Instructors are trained during this phase.
- Exam question banks and examination structure and content are developed to adequately evaluate and document trainee mastery of the course and job performance objectives associated with the training.

At the end of the development stage, the training package is reviewed, piloted on trainees, and revised if necessary.

In the implementation stage, instructors prepare for and deliver the training. Trainees are tested to determine if they have mastered the objectives. The results of trainee tests are examined during the evaluation stage. The evaluation stage examines the effectiveness of the training as delivered. This appraisal is done through the review of training results, training feedback, and continual monitoring of work performance.

The training program, developed as discussed above, provides assurance that plant personnel have the capability and competence needed to perform their roles and responsibilities. Participants used for Integrated System Validation (ISV) are trained using this program and provide validation of the integrated design.

#### 11.2.6.2.3 Results

The specific program outputs include documentation defining the overall program goals and course structure, as well as the specific job role qualification and training requirements and developed course materials. In addition to the training program content itself, the results of the


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training program development are summarized in a report which documents the process and activities used in development, including any inputs used, issues identified, and recommendations made.

11.2.6.3 Verification and Validation

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes the methodology for HFE V&V.

11.2.6.4 Design Implementation

Licensing Topical Report NAT-2965, “Natrium Human Factors Engineering Program Plan and Methodologies,” (Reference 11.2-1), incorporated by reference, describes design implementation.

11.2.7 Human Performance Monitoring

The human performance monitoring (HPM) strategy links HFE methods used during the design with methods for monitoring human performance during operation. The HPM program is fully developed at the OL stage.

11.2.7.1 Objectives and Scope

The purposes of HPM are to:

- Ensure that the high safety standards established by the HFE program during the design of the plant are maintained even when changes are made to the plant.
- Detect any deterioration of task performance that may be attributable to latent or slow-developing HFE design issues.
- Provide adequate assurance that the safety bases remain valid during the operational phase of the plant.

There is no intent for the HSI designer to periodically repeat a full-integrated system validation. The strategy is to provide a monitoring plan, building upon the HFE activities during the design that is carried forward into the operational phase, using industry accepted methods. HPM incorporates the monitoring strategy into the problem identification and corrective action program, which identifies and classifies human errors, provides for evaluation of the root cause, and supports effectiveness verification and documentation of the corrective action.

The scope of the performance monitoring strategy provides assurance that:

- The HSI design is effective during:
- Normal operations
- Accidental operational occurrences
- Design basis events, including DBAs
- Significant industry events


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- Key scenarios identified by the PRA and human reliability analysis (i.e., LBEs)
- HAs, using HSI information, cues, and controls can accomplish critical tasks while maintaining margin for time and performance criteria
- Acceptable performance levels established during the HFE ISV are maintained

Changes made to the initial HSIs, procedures, and training do not have adverse effects on personnel performance (e.g., a change interferes with trained skills)

11.2.7.2 Methodology

The HPM program aligns with the overall quality program and condition reporting methods. The program includes:

- Data collection
- Importance screening
- Event analysis to determine causes
- Trend analysis
- Corrective action development

The HPM process draws upon existing information sources and programs to supplement the data collection. Data collected is anonymized and sanitized to ensure that no blame culture or personal privacy breaches develop.

The HPM strategy collects data to trend human performance, particularly seeking issues with design root causes due to non-compliance with, or inappropriate application of, HFE principles. The HPM program uses existing utility or industry programs (e.g., corrective action programs, operator training) for data collection where appropriate. The HPM program is designed to ensure that:

- HAs are monitored commensurate with their safety importance.
- Feedback of information and corrective actions are accomplished in a timely manner.
- Degradation in performance is detected and corrected before plant safety is compromised.

The HPM program maintains a database of event causes and corrective actions taken. Such data supports trending of performance anomalies.

The HPM program identifies and establishes corrective actions that reduce the potential for incident recurrence. The program systematically identifies the cause of the failure or degraded performance. The corrective actions are derived by:

- Addressing the significance of the failure through application of PRA and human reliability analysis importance measures.
- Classifying the causes and circumstances surrounding the failure or degraded human performance.


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- Illuminating the characteristics of the failure (e.g., being task specific or due to design issues).
- Determining whether the failure is isolated or has generic or common cause implications.

11.2.7.3 Results

The HPM program activities and outputs align with the overall condition reporting requirements. They include specific incident or trend analysis reports, a recommendations and action tracking database, and periodic summary reports.

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11.2-7 U.S. NRC (Nuclear Regulatory Commission), (ML21287A088), Technical Letter Report No. F0028-04, “Development of HFE Review Guidance for Advanced Reactors,” 2021.

11.2-8 IAEA (International Atomic Energy Agency), INSAG-7, “The Chernobyl Accident: Updating of INSAG-1, Safety Series No. 75-INSAG-7,” 1992.

11.2-9 IAEA (International Atomic Energy Agency), Specific Safety Guide Series No. SSG-51, “Human Factors Engineering in the Design of Nuclear Power Plants,” 2019.



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**Table 11.2-1 NUREG-0711 Technical Elements Compared to Section 11.2**

| | NUREG-0711 | Chapter 11 Section 2 | Section # |
|---|---|---|---|
| **1** | Introduction | | |
| **2** | HFE Program Management | Management of the Human Factors Engineering Program | 11.2.1 |
| **3** | OER | OER | 11.2.2 |
| **4** | Functional Requirements Analysis and Function Allocation | Safety Function Review | 11.2.3 |
| **5** | TA | Functional Requirements Analysis | 11.2.3.1 |
| **6** | Staffing and Qualifications | Allocation of Function | 11.2.3.2 |
| **7** | Treatment of Important Human Actions | Staffing and Qualification | 11.2.3.3 |
| **8** | Human-System Interface Design | TA | 11.2.4 |
| **9** | Procedure Development | Treatment of Important Human Actions | 11.2.4.1 |
| **10** | Training Program Development | Human-System Interface Design | 11.2.5 |
| **11** | Human Factors Verification and Validation | Procedures,<br>Training,<br>Verification and Validation | 11.2.6.1<br>11.2.6.2<br>11.2.6.3 |
| **12** | Design Implementation | Design Implementation | 11.2.6.4 |
| **13** | Human Performance Monitoring | Human Performance Monitoring | 11.2.7 |



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**Table 11.2-2 Development of HFE Review Guidance for Advanced Reactors (Sheet 1 of 4)**

| SECTION | CONTENT | NUREG-0711 | DEVELOPMENT OF HFE REVIEW GUIDANCE FOR ADVANCED REACTORS |
|---|---|---|---|
| 11.2.1 | Management of the Human Factors Engineering Program | Human Factors Engineering Program Plan, including:<br>• Goals, Scope, Background, Criteria for Areas of Consideration<br>• Human Factors Input, including:<br>• HFE Organization Roles and Responsibilities, Training Needs and Related Groups<br>• Methods, including intended tools and technical guides<br>• HFE Processes and Procedures<br>• Timelines, including logical links to related project activities<br>• Documentation<br>• Disposition of Human Factors Issues | *HF Planning*:<br>• Determine methods, analyses, evaluations, project interfaces, and tools<br>• Identify constraints and drivers<br>• Graded approach based on risk and complexity<br>• Organization and resources<br>• Communications<br>• Source documents<br>• Issue identification and resolution<br>• Documentation<br>• Scheduling<br>• HFE Interfaces with other groups |
| 11.2.2 | Review of OE | OER | *HF in Concept Design: OER* |
| 11.2.3 | Safety Function Review | Safety Function Review | |
| 11.2.3.1 | Functional Requirements Analysis | Functional Analysis | *HF in Concept Design*: functional analysis |
| 11.2.2.2 | Allocation of Function | Allocation of Function | *HF in Concept Design*: functional analysis (definition includes allocation of functions) |
| 11.2.4 | TA | TA, Job Design | *HF in Preliminary Design*: TA including workload and communications analysis; link analysis |


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**Table 11.2-2 Development of HFE Review Guidance for Advanced Reactors (Sheet 2 of 4)**

| SECTION | CONTENT | NUREG-0711 | DEVELOPMENT OF HFE REVIEW GUIDANCE FOR ADVANCED REACTORS |
|---|---|---|---|
| 11.2.3.3 | Staffing and Qualifications | Staffing & Minimum Shift Complement, Job Design, Shift-Work Systems | *HF Interfaces*: HF in design considers the interfaces with staffing; the information common to both HF in design and interfacing disciplines, such as staffing analysis and strategies are shared. |
| 11.2.4.1 | Treatment of Important Human Actions | Human reliability, activities with potentially hazardous human interactions | *HF in Concept Design*: identification of scenarios for analysis<br>*HF in Preliminary Design*: participation in the assessment of human actions and error consequences; assessment of the feasibility of human actions in the deterministic safety analyses<br>*HF in Detailed Design*: confirmation of the feasibility of human actions safety significant in the probabilistic and deterministic safety analyses; analyses to confirm the ability of the human to perform necessary actions |
| 11.2.5<br>11.2.1.10 | Design of the Human-system Interface<br>Design Goals and Design Bases | Design human-machine interface system; design physical working environment | |
| 11.2.5 | Human-system Interface: Design Inputs | | *HF in Concept Design*: a statement of system operational purpose and operational requirements under all anticipated conditions; development or selection of HF in design source documents; identification of SSC requirements to support necessary human actions; HFE assessment of design concepts and options<br>*HF in Preliminary Design*: document high-level HF-related requirements; input to specifications and bid evaluations; requirements derived from HF analysis results |


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**Table 11.2-2 Development of HFE Review Guidance for Advanced Reactors (Sheet 3 of 4)**

| SECTION | CONTENT | NUREG-0711 | DEVELOPMENT OF HFE REVIEW GUIDANCE FOR ADVANCED REACTORS |
|---|---|---|---|
| 11.2.5 | Human-system Interface: Detailed Design and Integration | | *HF in Detailed Design*: detailed HSI design; design integration of COTS products |
| 11.2.5 | Human-system Interface: Tests and Evaluations | | *HF in Preliminary Design*: modeling, mock-ups, or prototyping of user interfaces; evaluations. The HFE program requirements will be implemented via procurement procedures.<br>*HF in Detailed Design*: Usability testing |
| 11.2.5 | Human-system Interface: Design of Main Control Room | | Covered within all HF in design activities. |
| 11.2.6.1 | Procedure Development Program | Procedure Development | *HF in Detailed Design Stage*: HF analyses output to development of training manuals, operating procedures, and commissioning procedures |
| 11.2.6.2 | Training Program Development | Training Program Development | *HF in Detailed Design Stage*: HF analyses output to development of training manuals, operating procedures, and commissioning procedures |
| 11.2.6.3 | Human Factors Engineering Verification and Validation | Verification<br>Validation | *HF in Detailed Design Stage*:<br>Verification (carried out before the design is released for construction)<br>Validation (validation activities split between detailed design and implementation) |
| 11.2.6.4 | Design Implementation (post-construction) | Design Implementation | *HF in Design Implementation Stage*: HFE during installation and commissioning |


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**Table 11.2-2 Development of HFE Review Guidance for Advanced Reactors (Sheet 4 of 4)**

| SECTION | CONTENT | NUREG-0711 | DEVELOPMENT OF HFE REVIEW GUIDANCE FOR ADVANCED REACTORS |
|---|---|---|---|
| 11.2.7 | Human Performance Monitoring (at start of testing) | Human Performance Monitoring | N/A (scope of standard is design stages only) |



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## 11.3 Emergency Preparedness

This section provides a description of the preliminary plans for coping with emergencies. The emergency preparedness plan will be provided at the operating license stage.

A performance-based emergency preparedness plan and emergency preparedness (EP) program will be developed utilizing the requirements in 10 CFR 50.160 and the guidance in Regulatory Guide (RG) 1.242, “Performance Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities,” Revision 0.

The emergency preparedness plan and Plume Exposure Pathway (PEP) Emergency Planning Zone (EPZ) will be developed based on design and site-specific information. Postulated emergencies will be determined and used to inform the emergency response functions, emergency response organization (ERO) staffing, emergency response facilities (ERFs), and PEP EPZ size. The emergency preparedness plan will be proven acceptable by evaluating and critiquing performance in training, drills, and exercises. An initial exercise to demonstrate compliance with 10 CFR 50.160(b) will be conducted within 2 years before the issuance of the operating license.

A preliminary PEP EPZ sizing analysis as required by 10 CFR 50.33(g)(2) has been completed that supports a PEP EPZ within the Exclusion Area Boundary (EAB). The PEP EPZ sizing analysis is discussed further in Section 11.3.3.1. Based on the regulations in 10 CFR 50.160 and guidance in RG 1.242, off-site planning activities are not required for sites in which the PEP EPZ is within the EAB. As such, the emergency preparedness plan will focus primarily on on-site planning activities.

### 11.3.1 Bases for Emergency Preparedness Approach

The plant design is risk-informed and credits features that rely upon passive processes and systems. A design and site-specific probabilistic risk assessment (PRA) that considers internal and external hazards, all modes of operation, and all significant radionuclide sources will be used to formulate the emergency preparedness plan and inform the PEP EPZ size. The PRA development and methodology is discussed in Chapter 3.

10 CFR 50.160 and the guidance in RG 1.242 provide for a performance-based, technology-inclusive, risk-informed, and consequence-oriented emergency preparedness approach for small modular reactors and other new technologies, and also a scalable PEP EPZ size.

The key elements of the emergency preparedness plan to meet 10 CFR 50.160 will include:

- Maintenance of Performance – A process will be developed for maintaining and making changes to the emergency preparedness plan and implementing procedures, including methods to account for facility changes and the methods used to conduct independent reviews of the EP program. This is discussed further in Section 11.3.10.


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- Performance Objectives – A performance-based monitoring program will be developed that provides for objectives and metrics to evaluate the emergency response functions. This is discussed further in Section 11.3.10.
  - Emergency Response Performance – Capabilities and guidance will be developed to demonstrate the emergency response functions, provided below, during drills and exercises. This is discussed further in Section 11.3.10.
  - Event Classification and Mitigation – An event classification scheme will be developed that includes Emergency Classification Levels and Emergency Action Levels based on a predetermined set of facility conditions. Procedures will be developed to guide mitigation efforts. This is discussed further in Section 11.3.6.2.
  - Protective Actions – Guidance will be developed to determine, implement, and recommend appropriate protective actions. This is discussed further in Section 11.3.6.3.
  - Communications – Capabilities for 24-hour notification will be provided for initial and ongoing communication to on-site and off-site organizations, as well as a notification system for activation of the ERO. This is discussed further in Section 11.3.9.3.
  - Command and Control – An ERO with defined positions, roles, responsibilities, and authorities to maintain effective command and control of the emergency response functions will be established. This is discussed further in Section 11.3.4.2.
  - Staffing and Operations – ERO positions and staffing level will be determined based on the roles and responsibilities needed to implement the emergency response functions. This is discussed further in Section 11.3.4.1.
  - Radiological Assessment – Capabilities and guidance will be developed to assess, monitor, and report radiological conditions, protective equipment needs, core or vessel damage, and releases. This is discussed further in Section 11.3.9.2.
- Reentry – Capabilities and guidance will be developed to address and implement reentry plans to access the facility after a radiological or hostile action-based emergency.
- Critique and Corrective Actions – A methodology will be developed to critique the effectiveness of the emergency response functions after drills, exercises, and actual events. Corrective actions will be identified, implemented, and addressed by the corrective action program. This is discussed further in Section 11.3.10.

The emergency preparedness plan will address the on-site planning activities as follows:

- Public Information – Capabilities and guidance will be developed to manage and coordinate the dissemination of information to the media and to support the public information functions of the federal, state, and local authorities.
- Implementation with the Safeguards Contingency Plan – Guidance will be developed to coordinate the implementation of the emergency preparedness plan with the safeguards contingency plan, including coordination, support, and communication with the incident command post and incident commander.
- Voice and Data Communications with the NRC – Capabilities and guidance will be developed to provide emergency response status and other necessary information to the NRC. This is discussed further in Section 11.3.9.3.


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- Emergency Response Facilities – ERFs will be addressed and included in the facility design to support emergency preparedness plan implementation and execution. This is discussed further in Section 11.3.9.1.
- Site Familiarization Training – Site familiarization training will be provided to off-site response services personnel. This is discussed further in Section 11.3.8.
- Emergency Preparedness Plan Maintenance – Methods for maintaining the emergency preparedness plan, implementing procedures, and other programmatic documents up to date, including periodic reviews by the on-site and off-site organizations, will be addressed. These documents will be considered quality documents and will be addressed by the quality assurance program. Emergency preparedness plan maintenance and reviews will be conducted in accordance with 10 CFR 50.54(t) and changes to an approved emergency preparedness plan will follow the requirements in 10 CFR 50.54(q).

## 11.3.2 Physical Site Characteristics

### 11.3.2.1 Site Description

The site description is discussed in Section 1.2.

### 11.3.2.2 Hazards from Nearby Facilities

Section 2.3 identifies nearby facilities that may pose potential hazards to the site. An analysis of the potential adverse impact from any credible hazard from these facilities on implementation of emergency response plans will be addressed in the emergency preparedness plan.

## 11.3.3 Emergency Planning Zone

### 11.3.3.1 Plume Exposure Pathway Emergency Planning Zone

The guidance in RG 1.242, Appendix A, was used to determine the PEP EPZ size. Information describing the PEP EPZ methodology was submitted to the NRC as “TerraPower, LLC (TerraPower) Natrium Topical Report: Plume Exposure Pathway Emergency Planning Zone Sizing Methodology,” herein referred to as the PEP EPZ Topical Report (Reference 11.3-1).

Based on the preliminary PEP EPZ sizing analysis, “Preliminary Emergency Planning Zone Determination Analysis” (Reference 11.3-2), which is incorporated by reference into the SAR, a PEP EPZ within the EAB is supported based on meeting the criteria in 10 CFR 50.33(g)(2)(i). See Section 1.4.2 for a list of documents incorporated by reference into the SAR.

### 11.3.3.2 Ingestion Pathway

The capabilities of appropriate state and local government agencies to address the major exposure pathways associated with the ingestion of contaminated food and water to avoid exceeding ingestion protective action guideline doses will be discussed in the emergency preparedness plan. Section 11.3.5 discusses government agencies that may be called upon to support an ingestion pathway response.


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### 11.3.4 Emergency Response Organization

The ERO and staffing will be determined dependent on the functions needed to support the specific postulated emergencies. Specific roles and responsibilities for emergency response personnel will be provided in the emergency preparedness plan and implementing procedures.

#### 11.3.4.1 Facility Organization

The details of the facility organization, including reporting relationships and responsibilities for implementing and maintaining the emergency preparedness plan are described in Section 11.1. The emergency preparedness plan will identify the minimum staff required to conduct immediate emergency response, as well as the ERF ERO staffing and response times.

#### 11.3.4.2 Authorities and Responsibilities of Facility Emergency Personnel

The Shift Manager will function as the Emergency Director (ED) and will fulfill this role until duties are transitioned to an ERF ED. The health physics person on-shift will support the ED with the radiological health physics aspects of the emergency until duties are transitioned to an ERF. Staffing will be established to perform the functions in accordance with 10 CFR 50.160.

The ED will assume command and control functions following an emergency declaration and will be responsible for ensuring that the emergency response functions in 10 CFR 50.160(b)(1)(iii) are fulfilled.

Upon declaration of an emergency, designated qualified members of the staff fulfill corresponding roles and activities in responding to the emergency (e.g., health physics personnel provide radiation protection, security personnel provide security, operations personnel focus on plant operations, and engineering personnel focus on plant assessment and technical support for operations).

The ED may call in additional personnel to provide further emergency response support. The individuals called in to assist will be assigned by the ED based on the function to be performed.

### 11.3.5 Federal, State and Local Governmental Agencies

#### 11.3.5.1 Federal Agencies

The NRC acts as the primary authority for radiological events involving NRC-licensed facilities. As the primary authority, the NRC takes the technical lead for the federal government's response to the event. For incidents affecting the general public, the Federal Emergency Management Agency may assume coordination of the federal response, while the NRC would remain the primary federal authority for on-site response.


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The NRC will be notified of emergency incidents as required by the emergency preparedness plan. The methods used to notify the NRC and the information provided will be established in coordination with the NRC and will be addressed in procedures.

- Federal resources are available to support radiological emergency event response, including events that may impact the ingestion pathway response. Federal resources include the Federal Radiological Monitoring and Assessment Center and the Advisory Team for Environment, Food and Health, as well as sampling and testing laboratories.

#### 11.3.5.2 State and Local Governmental Agencies

Notification requirements, protocols, and agreements will be developed and established in coordination with the Wyoming Office of Homeland Security and local governmental agencies within the immediate vicinity of the site, such as Lincoln County and Kemmerer, Wyoming.

Applicable state and local government certified all hazards plans, including ingestion pathway capabilities, will be referenced in the emergency preparedness plan. Letters of agreement will be obtained and submitted with the emergency preparedness plan.

### 11.3.6 Protective Measures

#### 11.3.6.1 Response to Emergency Situation

The steps for taking protective measures within the site will be procedurally directed and include:

- Performing first aid.
- Moving personnel away from hazardous areas.
- Recommendations for and use of protective gear.
- Contamination control measures, including moving personnel away from contaminated areas.
- Establishing restricted areas.
- Site evacuation of non-essential personnel.
- Coordination with the fire protection and security plans.

On-site communication systems and action specific alarms (e.g., site evacuation) will be used to communicate appropriate on-site protective actions. Communication systems utilized for emergency preparedness are discussed further in Section 11.3.9.3.

#### 11.3.6.2 Emergency Classification

A design and site-specific emergency classification scheme and associated emergency action levels, including documentation of the methodology for determining the emergency action levels, will be provided in conjunction with the emergency preparedness plan.


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#### 11.3.6.3 Protective Actions

Capabilities will be provided to assess and develop protective actions for on-site personnel, as well as to recommend protective actions to off-site authorities as conditions warrant. The most likely on-site protective actions include the use of proper personal protective equipment, sheltering on-site, or site evacuation. Protective equipment will be stored and provided on-site for use to continue and expand mitigation and protective actions.

Since the preliminary sizing analysis supports PEP EPZ within the EAB, there is no requirement to develop pre-planned off-site protective actions. However, capabilities will be provided to assess on-site conditions to support recommendations for protective actions to off-site agencies.

### 11.3.7 First Aid, Decontamination, Off-site Medical Treatment, and Emergency Medical Transportation

During a declared emergency under the emergency preparedness plan, administration of first aid, off-site medical treatment, or emergency transport of personnel will be coordinated according to emergency response procedures. Any medical treatment or decontamination will be prioritized based on severity of injury and level of contamination. Provisions and agreements will be established to transport, treat, and decontaminate injured personnel at an off-site medical facility.

On-site means for decontamination and first aid for personnel, such as decontamination showers and first aid kits, will be provided within the Nuclear Island.

### 11.3.8 Training

A training program will be developed and conducted to establish and maintain the capabilities of emergency response personnel to perform their assigned functions. Personnel will receive generic emergency preparedness plan training and training specific to their assigned emergency response functions. Initial training will be provided to individuals before being assigned ERO responsibilities.The content of the training program will include the overall emergency preparedness plan and the relevant implementing procedures. Details of the training program will be provided in the emergency preparedness plan.

Off-site personnel and agencies whose assistance will be needed in response to an emergency will be provided site familiarization training.

### 11.3.9 Emergency Response Facilities and Equipment

#### 11.3.9.1 Emergency Response Facilities

A primary ERF (ERFP), located on-site, and a backup ERF (ERFB), located off-site, will be available. Either of these facilities support functions similar to those required for a Technical Support Center and Operational Support Center as described in NUREG-0696 (Reference 11.3-3) and NUREG-0654 (Reference 11.3-4). There are no differences in the functions performed by the ERFP and ERFB. The ERF staff will relieve the reactor operators of peripheral duties and communications not directly related to reactor control during emergencies,


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allowing control room personnel to focus on responding to the operational aspects of the emergency. The ERF will assist in the management and control of plant emergency response capabilities to support control room personnel, mitigate the consequences of an accident, and respond quickly to abnormal plant conditions.

During an emergency, the ERF will be the primary communications center for the plant, providing communications capabilities on-site, as well as with the NRC, other governmental agencies, and off-site stakeholders and support entities as required by the emergency preparedness plan. In this capacity, the ERFP and ERFB will both include access to technical data information and displays to assist in the detailed analysis and diagnosis of abnormal plant conditions. Additionally, methods and procedures for assessing and identifying the seriousness of the radiological consequences of emergency situations will be available.

The ERFP and ERFB will be designed to allow the ERO to perform the functions described in 10 CFR 50.160. Under normal conditions, the ERFP and ERFB will be designed to be comfortable and habitable. If the ERFP becomes inaccessible, or uninhabitable, the ERFB will be used. For events where the ERO should initially report to the ERFB (e.g., an external event or hostile action), the decision will be made by the control room ED prior to the initial activation call-out. For events that develop into an uninhabitable ERFP, the ERF ED will determine when to relocate to the ERFB. Procedures will be developed to address relocation to the ERFB facility.

#### 11.3.9.2 Radiological Assessment

##### 11.3.9.2.1 Radiological Conditions

Provisions, equipment, and procedural guidance will be provided to assess, monitor, and report radiological conditions and consequences of emergency situations in and around the facility. Capabilities to provide dose projections using real-time meteorological information will be provided.

##### 11.3.9.2.2 Core or Vessel Damage

Capabilities and procedural guidance will be provided to assess, monitor, and report the extent of any core or reactor vessel damage, or other vessel containing irradiated special nuclear material.

##### 11.3.9.2.3 Releases

Provisions, equipment, and procedural guidance will be provided to assess, monitor, and report the extent of any radiological release, including the releases of hazardous chemicals produced from licensed material.

#### 11.3.9.3 Communication Systems

Capabilities will be established and maintained to promote various forms of communication and information sharing between ERO positions, ERFs (i.e., Main Control Room, ERFP, and ERFB), and other on-site and off-site organizations that may have emergency response functions. The communication systems will use technology that reliably performs the emergency response communication functions. ERO positions will be provided computers and voice communication


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equipment needed to support their ERO functions. A system designed to collect, process, display, share, and communicate plant parameters to aid in efficient emergency response decision-making will be available for use.

Capabilities will be provided to notify and activate the ERO in a timely manner and to alert on-site personnel regarding the status of the emergency (e.g., the nature of the emergency classification, releases, location of the emergency, protective actions that are implemented on site.)

Coordination with the NRC, state, and local governmental agencies to determine the voice and data communication systems compatible with their respective systems will be conducted.

The NI communication systems are further discussed in Section 1.1.4.3.

### 11.3.10 Performance Monitoring

A performance monitoring program will be developed to demonstrate the effectiveness of the implementation of emergency preparedness plan to carry out an effective response to emergency and accident conditions. Drills and exercises will be used to demonstrate the capabilities to perform and maintain the emergency response functions listed in 10 CFR 50.160(b)(1)(iii)(A) through (H). Performance metrics and objectives for each of these emergency response functions will be developed and used for evaluation.

Drills and exercises, as well as responses to actual declared emergency events, will be critiqued to identify weaknesses or deficiencies in ERO performance and the EP program. The corrective action program will be used to evaluate, track, and correct identified EP weaknesses and deficiencies.

**References**

11.3-1 TerraPower, “TerraPower, LLC (TerraPower) Natrium Topical Report: Plume Exposure Pathway Emergency Planning Zone Sizing Methodology,” NAT-3056, November 16, 2023 (ADAMS Accession No. ML23321A036).

11.3-2 TerraPower, “Preliminary Emergency Planning Zone Determination Analysis, Technical Report,” TP-LIC-RPT-0012, March 2024

11.3-3 U.S. NRC, “Functional Criteria for Emergency Response Facilities,” NUREG-0696, February 1981 (ADAMS Accession No. ML051390358).

11.3-4 U.S. NRC, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” NUREG-0654/FEMA-REP-1, Revision 2, December 2019 (ADAMS Accession No. ML19347D139).


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## 11.4 Aircraft Impact Assessment

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# 11.5 Technical Specifications

## 11.5.1 Introduction

In accordance with the requirements of 10 CFR 50.34(a)(5), an identification and justification for the selection of those variables, conditions, or other items which are determined as the result of the preliminary safety analysis and evaluation to be the probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design, are provided in Table 11.5-1.

The full set of technical specifications that address the requirements of 10 CFR 50.36 will be submitted at the operating license (OL) stage in accordance with 10 CFR 50.34(b)(6)(vi).

The format and layout of the technical specifications are planned to be consistent with NUREG-1431, Volume 1, (Reference 11.5-1) and include:

- Use and Application
- Safety Limits
- Limiting Conditions for Operation and Surveillance Requirements
- Design Features
- Administrative Controls

## 11.5.2 Operating Modes

The operating modes for the reactor are defined in Table 11.5-2. Each operating mode is defined in terms of an inclusive combination of sodium temperature, control rod drive (CRD) status, and reactor cooling mechanism. The CRD is described in Section 7.2.5. Reactor heat removal is performed by rejecting heat from the Primary Heat Transport System (PHT) to the Intermediate Heat Transport System (IHT) and then to either the Nuclear Island Salt System (NSS) or Intermediate Air Cooling System (IAC) as described in Sections 7.1.3 and 7.1.4. Each mode is further described in Sections 11.5.2.1 through 11.5.2.4.

### 11.5.2.1 Mode-1: High Power Operation

Mode-1 is entered from Mode-2 during startup when heat removal is transferred from the IAC to the NSS via the Sodium-Salt Heat Exchangers (SHXs).

### 11.5.2.2 Mode-2: Low Power Operation

Mode-2 is entered during startup when the reactor scram breakers are closed and more than one control rod assembly (CRA) is coupled to its respective control rod drive mechanism (CRDM) driveline. Mode-2 supports control rod withdrawal to a critical state and power ascension within the heat removal capabilities of the IAC. Mode-2 can be entered during a shutdown or downpower when reactor power is lowered to within the capability of the IAC and heat removal is transferred from the NSS to the IAC via SHXs. Mode-2 is not required to be entered during a shutdown. If a reactor scram occurs prior to meeting the conditions for Mode-2, the plant can transition directly from Mode-1 to Mode-3.


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11.5.2.3 Mode-3: Shutdown

The purpose of Mode-3 is to serve as a transition state for plant startup and entry into Mode-2 or prior to entering Mode-4 during plant shutdown. The primary coolant boundary is established during startup in Mode-3. Mode-3 supports plant heatup through Primary Sodium Pump (PSP) operation. No more than one CRA is coupled to its CRDM driveline while in Mode-3, which ensures a sufficient number of CRAs are inserted to maintain the reactor shutdown. One CRA may be coupled to its CRDM driveline and manipulated to support testing.

11.5.2.4 Mode-4: Refuel

The purpose of Mode-4 is to support refueling activities. Primary sodium hot pool temperature is lowered to the refueling temperature below 400°F (204°C). No more than one CRA is coupled to its CRDM driveline, which ensures a sufficient number of CRAs are inserted to maintain the reactor shutdown. One CRA may be coupled to its CRDM driveline and manipulated to support testing.

**References**

11.5-1 U.S. NRC, NUREG-1431, “Standard Technical Specifications- Westinghouse Plants,” Volume 1, “Specifications,” Revision 5.0.

11.5-2 Technical Specification Task Force, Traveler TSTF-493, “Clarify Application of Setpoint Methodology for Limiting Safety System Settings,” Revision 4.



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**Table 11.5-1 Proposed Variables and Conditions for Technical Specifications (Sheet 1 of 7)**

| Section | Section Name | Variable or Condition | Basis |
|---|---|---|---|
| 1.0 | Use and Application<br><br>The Use and Application section of the technical specifications provides definitions; defines the plant modes used in determining Limiting Condition for Operation (LCO) applicability; explains the meaning of logical connectors; establishes the completion time convention and provides guidance for its use; and defines the proper use and application of frequency requirements. | | |
| 2.0 | Safety Limits<br><br>Safety limits (SLs) are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.<br><br>Limiting Safety System Settings (LSSSs) for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a LSSS is specified for a variable on which an SL has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded.<br><br>Proposed subjects of SLs are listed below. An administrative control technical specification will be proposed to maintain a setpoint control program consistent with Option B of TSTF-493 (Reference 11.5-2) to satisfy 10 CFR 50.36(c)(1)(ii)(A) in lieu of specifying explicit values for the LSSSs. | | |
| 2.1 | Reactor Core SLs | SLs for applicable modes of operation are established for the core to protect the integrity of the fuel assemblies. The probable subjects of the SLs are fuel cladding temperature and peak fuel temperature.<br><br>The cycle-specific parameter limits relating to the SLs are described in the Core Operating Limits Report (COLR). | Fuel cladding temperature limits protect against thermal creep and thinning of the cladding wall due to fuel-cladding chemical interactions and eutectic interactions under transient conditions. Peak fuel temperature limits protect against melting. Establishing fuel cladding temperature limits and peak fuel temperature limits protects the integrity of the fuel barrier. |


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**Table 11.5-1 Proposed Variables and Conditions for Technical Specifications (Sheet 2 of 7)**

| Section | Section Name | Variable or Condition | Basis |
|---|---|---|---|
| 2.2 | Safety Limit Violations | Direction is provided to restore compliance with the SL and shut down the reactor. | Restoring compliance with the SL and shutting down the reactor reduces the potential for further system damage and radioactive release. |
| 3.0 | Limiting Conditions for Operation and Surveillance Requirements<br><br>LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility.<br><br>The technical specifications will include the full list of LCOs that meet the intent of Criteria 1 through 4 of 10 CFR 50.36(c)(2)(ii). Below is a list of variables and conditions derived from the preliminary safety analysis and relating to structures, systems, and components that are probable subjects of LCOs.<br><br>Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within SLs, and that the LCOs will be met. The technical specifications will include the surveillance requirements. | | |



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**Table 11.5-1 Proposed Variables and Conditions for Technical Specifications (Sheet 3 of 7)**

| Section | Section Name | Variable or Condition | Basis |
|---|---|---|---|
| 3.1 | Reactivity Control Systems | The reactivity control systems are the Primary CRAs and the Secondary CRAs.<br><br>LCOs will be developed to define the following lowest functional capabilities:<br><br>• Control rods are able to be inserted via a scram.<br><br>• Control rod alignment limits are established that ensure the Specified Acceptable System Radionuclide Release Design Limits are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences.<br><br>• Control rods are capable of inserting sufficient negative reactivity at a sufficient rate to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.<br><br>The cycle-specific parameter limits relating to the reactivity control systems are described in the COLR.<br><br>The Primary CRAs and Secondary CRAs are described in Sections 7.1.1 and 7.2.5. | LCOs in this section will ensure the capability of Reactivity Control Systems to scram the reactor is met during the modes of applicability. |


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**Table 11.5-1 Proposed Variables and Conditions for Technical Specifications (Sheet 4 of 7)**

| Section | Section Name | Variable or Condition | Basis |
|---|---|---|---|
| 3.2 | Power Distribution Limits | LCOs will be developed for the power distribution limits that will define the lowest functional capabilities of maintaining fuel within design limits.<br><br>The probable subject of LCOs are hot channel factors that are identified during core design.<br><br>The cycle-specific parameter limits relating to the power distribution limits are described in the COLR. | LCOs in this section will ensure the design limits of the fuel/cladding are met during the modes of applicability. |
| 3.3 | Instrumentation | The Reactor Protection System (RPS), Nuclear Instrumentation System (XIS), and Reactor Instrumentation System (RIS) detect abnormal conditions and perform Required Safety Functions to mitigate licensing basis events (LBEs) or prevent LBEs from occurring.<br><br>The parameters, including tolerances, which instruments must be capable of detecting to identify abnormal plant conditions to ensure LSSSs are not exceeded will be determined in the setpoint control program.<br><br>LCOs will be developed for the RPS, XIS, and RIS that define the lowest functional capability of:<br>• field transmitters and process sensors<br>• components that provide signal conditioning and setpoint comparison<br>• components that perform logic functions<br>• reactor trip switchgear<br>The RPS, XIS, and RIS are described in Section 7.6. | LCOs in this section will ensure RPS, XIS, and RIS can perform the following functions during the modes of applicability:<br>• reactor scram<br>• PSP trip<br>• Intermediate Sodium Pump trip |


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**Table 11.5-1 Proposed Variables and Conditions for Technical Specifications (Sheet 5 of 7)**

| Section | Section Name | Variable or Condition | Basis |
|---|---|---|---|
| 3.4 | Primary Heat Transport System | LCOs will be developed for the PHT that define the lowest functional capability of the system and components to maintain primary sodium temperatures within design limits.<br><br>Variables initially identified to be part of LCOs include:<br><br>• primary sodium temperature<br>• sodium leakage<br>• sodium flow rate<br>• PSP coastdown time<br>• leakage into IHX<br>• PSP leakage<br><br>The PHT is described in Section 7.1.3. | LCOs in this section will ensure the PHT can perform the following functions during the modes of applicability:<br><br>• natural circulation in primary system<br>• barrier functions<br>• inputs to instrumentation systems addressed in LCO 3.3 for reactor scram<br>• PSP trip<br>• PSP coastdown |
| 3.5 | Reactor Air Cooling System | LCOs will be developed for the RAC that define the lowest functional capability of the system to remove residual heat following postulated accidents.<br><br>Variables initially identified to be part of LCOs include:<br><br>• intake/exhaust temperatures<br>• intake/exhaust flow rates<br><br>The RAC is described in Section 7.2.1. | LCOs in this section will ensure the RAC can perform the function of RAC operation during the modes of applicability. |
| 3.6 | Functional Containment | LCOs will be developed for functional containment that define the lowest functional capability of its related SSCs to limit radiological release.<br><br>Variables initially identified to be part of LCOs include system/component leakage.<br><br>Functional containment is described in Section 1.3.2.1. | LCOs in this section will ensure the functional containment can perform the function of retaining radionuclides during the modes of applicability. |


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**Table 11.5-1 Proposed Variables and Conditions for Technical Specifications (Sheet 6 of 7)**

| Section | Section Name | Variable or Condition | Basis |
|---|---|---|---|
| 3.7 | Refueling Operations | The In-Vessel Fuel Handling System (FHI), Ex-Vessel Fuel Handling System (FHE) and Water Pool Fuel Handling System (FHP) perform safety-related passive heat removal, containment barrier, and head barrier functions.<br><br>LCOs will be established for the FHI, FHE, and FHP that define the lowest functional capability of subsystems that are required for heat removal and radionuclide retention.<br><br>Variables initially identified to be part of LCOs include:<br><br>• sealing capability for subsystems performing barrier functions<br>• adequate flow paths/rates for subsystems performing passive heat removal functions<br>• heat removal capability for subsystems performing passive heat removal functions | LCOs in this section ensure the FHI, FHE, and FHP can perform the following functions during the modes of applicability:<br><br>• head barrier function<br>• passive heat removal<br>• containment barrier |
| 4.0 | Design Features are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described above. The design features to be included in the technical specifications will be provided at the OL stage.<br><br>Probable design features include:<br><br>• a description of the site location<br>• a description of the fuel assemblies that comprise the reactor core<br>• a discussion of fuel storage requirements | | |



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**Table 11.5-1 Proposed Variables and Conditions for Technical Specifications (Sheet 7 of 7)**

| Section | Section Name | Variable or Condition | Basis |
|---|---|---|---|
| 5.0 | Administrative Controls are the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The administrative controls to be included in the technical specifications will be provided at the OL stage.<br><br>Probable administrative controls include:<br><br>• delineation of responsibility<br>• description of organization<br>• unit staff qualifications<br>• requirements for procedures<br>• programs and manuals, to include a setpoint control program consistent with Option B of TSTF-493<br>• reporting requirements, to include a COLR | | |



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**Table 11.5-2 Operating Modes**

| Mode | Title | CRD Status | Reactor Cooling Mechanism | Sodium Temperature |
|---|---|---|---|---|
| 1 | High Power Operation | More than one CRA coupled to CRDM driveline | NSS (1) | N/A |
| 2 | Low Power Operation | More than one CRA coupled to CRDM driveline | IAC (2) | N/A |
| 3 | Shutdown | No more than one CRA coupled to CRDM driveline | N/A | > 400 °F (204°C) |
| 4 | Refuel | No more than one CRA coupled to CRDM driveline | N/A | ≤ 400 °F (204°C) |

(1) The transition from Mode-2 to Mode-1 will be when the transfer of IHT heat removal has been fully transferred from IAC to NSS as denoted by the IAC blower being off and the top and bottom dampers being closed and latched.

(2) The transition from Mode-1 to Mode-2 will be when the transfer of IHT heat removal has been fully transferred from NSS to IAC as denoted by the NSS isolation valves being closed.



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# 11.6 Fitness-for-Duty and Security

## 11.6.1 Fitness-for-Duty

A fitness-for-duty program consistent with NEI 06-06 (Reference 11.6-1), as endorsed by Regulatory Guide 5.84, “Fitness-for-Duty Programs at New Reactor Construction Sites,” Revision 0, will be implemented prior to the commencement of construction of safety-related and security-related structures, systems, and components. Individuals identified in 10 CFR 26.4(e) and 10 CFR 26.4(g) will be subject to a fitness-for-duty program that meets the requirements of 10 CFR Part 26, Subparts A through H, N, and O.

## 11.6.2 Security

A physical security plan, training and qualification plan, and cyber security plan that meets the requirements of 10 CFR 73.54 will be provided at the operating license stage as required by 10 CFR 50.34(c)(1), (2), (3). A safeguards contingency plan will be provided at the operating license stage as required by 10 CFR 50.34(d). The development of the security plan will consider the site characteristics described in Chapter 2 and potential vulnerabilities will be mitigated. Security measures consistent with NEI 09-01 (Reference 11.6-2) will be implemented prior to the commencement of construction of safety-related and security-related structures, systems, and components.

**References**

11.6-1 Nuclear Energy Institute, “Fitness-for-Duty Program Guidance for New Nuclear Power Plant Construction Sites,” NEI 06-06, Revision 6, April 2013.

11.6-2 Nuclear Energy Institute, “Security Measures During New Reactor Construction,” NEI 09-01, Revision 0, September 2009.


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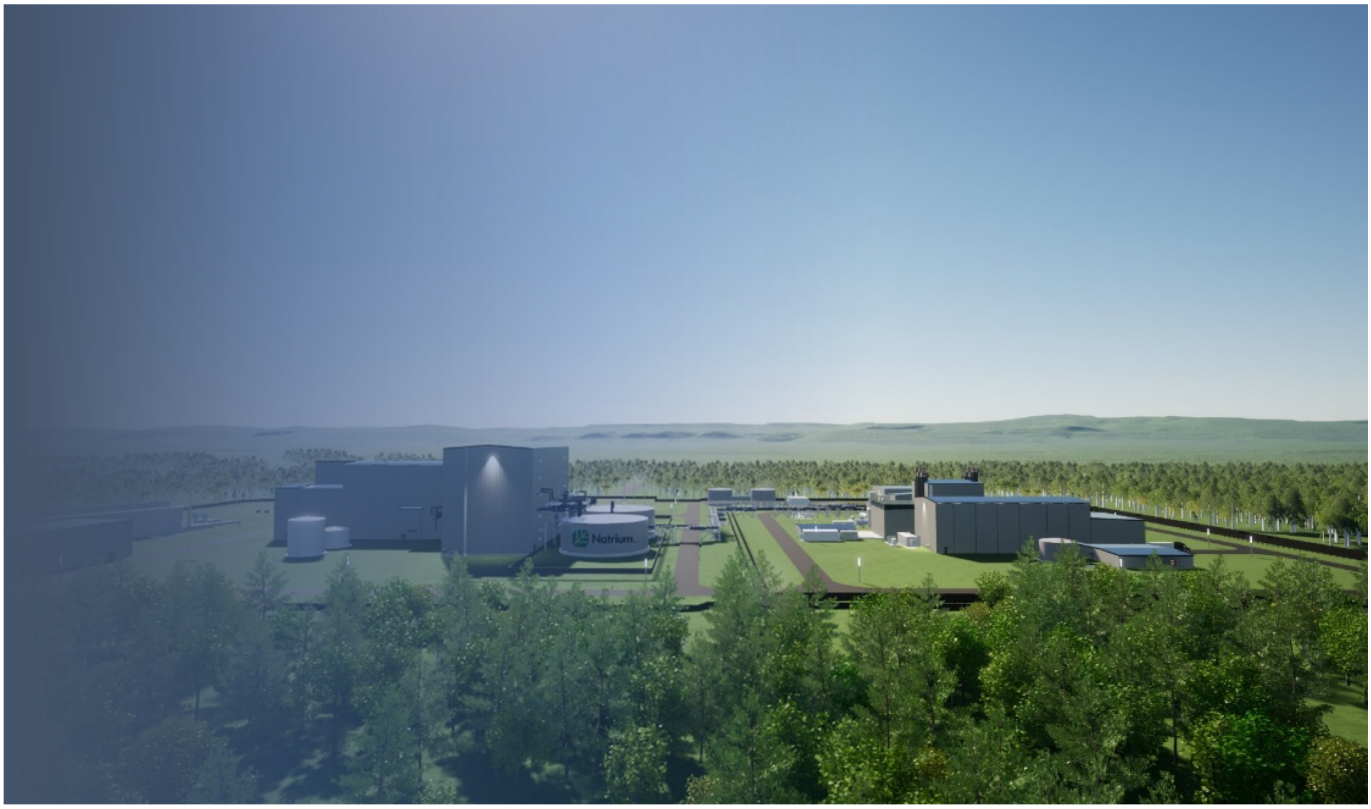

# Kemmerer Power Station Unit 1
# PSAR, Chapter 12

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## TABLE OF CONTENTS

**Chapter 12 Post-Construction Inspection, Testing, and Analysis Program . . . . . 12.1-1**

12.1 Program Summary and Objectives . . . . . 12.1-1

12.2 Program Administration . . . . . 12.2-1

12.3 Construction Testing (Stage 1). . . . . 12.3-1

12.4 PITAP Phase 1: Preoperational Test Program. . . . . 12.4-1

12.4.1 Component Testing (Stage 2) . . . . . 12.4-1

12.4.2 System Testing (Stage 3). . . . . 12.4-1

12.5 PITAP Phase 2: Initial Startup Testing (Stage 4) . . . . . 12.5-1

12.5.1 Initial Fuel Load and Reactor Physics Tests . . . . . 12.5-1

12.5.2 Low Power Testing. . . . . 12.5-1

12.5.3 Power Ascension Testing. . . . . 12.5-2

12.6 First-of-a-Kind Testing . . . . . 12.6-1


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# Chapter 12 Post-Construction Inspection, Testing, and Analysis Program

## 12.1 Program Summary and Objectives

The primary objective of the post-construction inspection, testing, and analysis program (PITAP) is to demonstrate that the plant has been constructed as designed, that the safety-significant structures, systems, and components perform consistent with the plant design, and that activities culminating in operation at full licensed power, including initial fuel load, initial criticality, and power ascension are performed in a controlled and safe manner. In addition, the program provides assurance that the operating and emergency procedures have been evaluated and demonstrated, and that the plant operations personnel are knowledgeable about the plant operating procedures and are fully prepared to operate the plant in a safe manner.

The PITAP is comprised of preoperational tests, including component, system, and pre-core load hot functional testing, and initial startup testing, including fuel load, post-core hot functional testing, initial criticality, low-power physics tests, and power-ascension tests. The PITAP establishes procedures and controls used to conduct and evaluate test results to satisfy the relevant requirements of the following regulations:

- 10 CFR 30.53, as it relates to testing radiation detection equipment and monitoring instruments.
- 10 CFR 50.34(b)(6)(iii), as it relates to providing information associated with preoperational testing and initial operations.
- Section XI of Appendix B to 10 CFR Part 50, as it relates to test programs to demonstrate that structures, systems, and components perform satisfactorily.
- A functional containment boundary leakage test program similar to Appendix J of 10 CFR Part 50, as it relates to preoperational leakage rate testing.

Preoperational testing is performed on those structures, systems, and components that are:

- Relied upon for safe shutdown and cooldown of the plant under normal conditions and for maintaining a safe condition for an extended shutdown period.
- Relied upon for safe shutdown and cooldown of the plant under postulated transient and accident conditions and for maintaining a safe condition for an extended shutdown period following such conditions.
- Relied upon for establishing conformance with safety limits or limiting conditions for operation that are included in the technical specifications.
- Identified as risk-significant in the probabilistic risk assessment.
- Used to process, store, control, or limit the release of radioactive materials.
- Relied upon to maintain their structural integrity during normal operation, anticipated transients, simulated test parameters, and design basis event conditions to avoid damage to safety-related structures, systems, and components.


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A graded approach is used to develop and implement the PITAP. The process from construction through to initial startup testing is implemented in four stages:

- Stage 1, construction testing performed by the construction organization.
- Stage 2, component testing performed by the construction start-up group.
- Stage 3, system testing performed by the construction start-up group following successful completion of the associated component testing.
- Stage 4, initial startup testing performed by licensed operators.

Stages 2 and 3 define the preoperational test program.


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## 12.2 Program Administration

For initial startup testing, Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” Revision 4, is used to inform planning and execution of the post-construction inspection, testing, and analysis program.

The design organization, part of the Natrium project organization as defined in Section 11.1, is responsible for the test objectives, prerequisites, safety precautions, and acceptance criteria for each structure, system, and component, and for providing the test abstract to the commissioning team for procedure development. The commissioning team is responsible for determining test sequence and scheduling.

The construction organization, with the construction start-up group, implements Stages 1 through 3 and is responsible for ensuring its personnel, including vendors, are qualified to perform their individual functions in support of construction and preoperational testing. The construction organization is responsible for developing all test procedures used during construction and preoperational testing.

The operations organization is responsible for creating a commissioning team led by the commissioning manager that oversees and manages the post-construction inspection, testing, and analysis program from system turnover to completion. Specifically, the commissioning team provides oversight of the construction organization during preoperational testing conducted by the construction start-up group and provides the lead for implementation and conduct of Stage 4.



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## 12.3 Construction Testing (Stage 1)

The adequacy of construction, installation, and preliminary operation of structures, systems, and components is verified by a construction test program. Construction testing is executed under the quality assurance program, as defined in Section 8.1, to ensure plant components and systems are installed to design parameters with no apparent defects. During construction testing, various electrical and mechanical tests are performed including the following:

- Cable separation, continuity, megger testing, and high potential testing
- Non-destructive testing of structural, piping, and component welding
- Hydrostatic testing of piping systems
- Hanger and pipe support inspections
- Pump and valve alignments
- Cleaning and flushing

On a system basis, completion demonstrates that the system is ready for turnover to the construction start-up group for initiation of preoperational testing. Development of the construction and installation tests is based on the engineering information provided by the Natrium project organization for the equipment and systems installed.

The system turnover process ensures that the construction start-up group and US SFR Owner, LLC are satisfied with the adequacy of the construction and installation tests performed. System turnover can only occur after Stage 1 testing is complete. System turnovers may be for full systems or sub-systems. Partial system turnovers should occur only as approved by both the construction start-up group manager and the commissioning manager. System turnover includes:

- System turnover package reviews by construction, the construction start-up group, the Natrium project organization, and the operations group.
- Walkdown and inspection of system by the Natrium project organization and the operations group.
- As-built design drawings are complete.
- Any deviations from design parameters are noted and resolved by the Natrium project organization.



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## 12.4 PITAP Phase 1: Preoperational Test Program

Preoperational testing is conducted to demonstrate the capabilities of structures, systems, and components to meet performance requirements throughout the full design operating range, as practical. This testing is completed over Stages 2 and 3. Stage 3, System Testing, includes individual system testing, followed by integrated system testing.

Preoperational testing provides reasonable assurance that systems and equipment perform in accordance with the design. Test results are analyzed to verify that systems and components have met performance requirements. Any non-conforming items are controlled per the quality assurance program described in Section 8.1.

### 12.4.1 Component Testing (Stage 2)

Component testing is the second stage of the PITAP program and the first stage performed by the construction start-up group. It includes actions such as:

- Complete system flushing and cleanliness verification
- Electrical equipment testing
- Loop checks and calibrations
- Control system checks
- Instrument calibrations
- Individual component testing (e.g., valve stroke test, pump rotation check)

Component testing provides reasonable assurance that the component is ready to move from a static to a dynamic state. Component testing is to be completed satisfactorily prior to conducting associated system testing.

### 12.4.2 System Testing (Stage 3)

System testing is performed under the conditions expected when the systems are required to function, when possible. The test abstracts generated by the Natrium project organization define the test conditions. When operating conditions cannot be attained or appropriately simulated during the test, the system is tested to the extent practicable under the given conditions with additional testing planned when operating conditions can be attained.

System testing commences the use of the test review group, whose function is to manage testing procedural changes, review test results, and interface with the Natrium project organization to ensure design requirements are met during this testing phase.


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## 12.5 PITAP Phase 2: Initial Startup Testing (Stage 4)

The objective of initial startup testing is to demonstrate that the systems perform consistent with the plant design, and to ensure that activities culminating in operation at full licensed power including initial fuel load, initial criticality, and power ascension are performed in a controlled and safe manner. The PITAP follows the requirements of 10 CFR 50, Appendix B, Section XI and is informed by RG 1.68. All work is performed by appropriately trained operations personnel utilizing approved procedures. Phase 2 testing will be provided in the OL stage in accordance with 10 CFR 50.34(b)(6)(iii). A high-level overview is provided in Sections 12.5.1 through 12.5.3.

### 12.5.1 Initial Fuel Load and Reactor Physics Tests

Initial fuel loading starts after completion of preoperational testing, system handover from construction start-up group to operations, and issuance of the plant operating license and an appropriate number of individual operators licenses. The post-core hot function testing is performed following the completion of initial fuel loading operations and prior to initial criticality. The objectives of these tests are to provide additional assurance that plant systems necessary for normal plant operation function as expected and to obtain performance data on core-related systems and components. Normal plant operating procedures, to the extent practicable, are used to bring the plant from refueling conditions through shutdown to low power operations. Testing normally proceeds directly to initial criticality and the beginning of low-power physics testing. Test hold points are established at appropriate intervals to review test results and confirm all prerequisite tests have been completed satisfactorily before moving to the next step. Testing addresses:

- Initial criticality
- Shutdown margin
- Reactivity control system performance
- Neutron monitoring instrumentation operation and calibration

### 12.5.2 Low Power Testing

After initial criticality has been achieved, a series of low-power physics tests is conducted to verify selected core design parameters. These tests serve to substantiate that the safety analysis and technical specifications assumptions and limits have been met. They also demonstrate that core characteristics are within the expected limits and provide data for benchmarking the design methodology used for predicting the core characteristics later in the life of the plant. Test hold points are established at appropriate intervals to review test results and confirm all prerequisite tests have been completed satisfactorily before moving to the next step. Testing addresses:

- Reactivity control system rod worth
- Neutron monitoring instrumentation operation and calibration
- Neutron flux distribution
- Neutron and gamma radiation surveys
- Operability of alarms and low power protective features
- Reactivity control system performance


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### 12.5.3 Power Ascension Testing

A series of power ascension tests is conducted to bring the reactor to full power. The purpose of these tests is to demonstrate that the facility operates in accordance with its design during steady-state conditions and, to the extent practicable, during anticipated transients. Test hold points are established at appropriate intervals to review test results and confirm all prerequisite tests have been completed satisfactorily before moving to the next step. Testing addresses:

- Reactivity coefficients and power to flow characteristics
- Neutron flux and power distribution
- Reactivity control system influence on power distribution and core design limits
- Reactivity control system performance
- Reactor coolant system performance
- Flow induced system vibration and thermal expansion monitoring
- Neutron and gamma radiation surveys
- Neutron monitoring instrumentation and calibration
- Operability of alarms and full power protective features
- Plant response to various anticipated operational occurrences (e.g., turbine trip, loss of normal power)


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## 12.6 First-of-a-Kind Testing

To the extent practical, first-of-a-kind (FOAK) components and equipment are tested. Special tests to develop FOAK components and equipment may be performed at the Sodium Test and Fill Facility, with operating experience from these tests contributing to future in-plant tests. Other FOAK tests not completed at the Sodium Test and Fill Facility may be completed at an owner-controlled off-site facility, at a vendor facility, or during preoperational or initial startup testing. FOAK testing is indicated on the test abstracts generated by the Natrium project organization.



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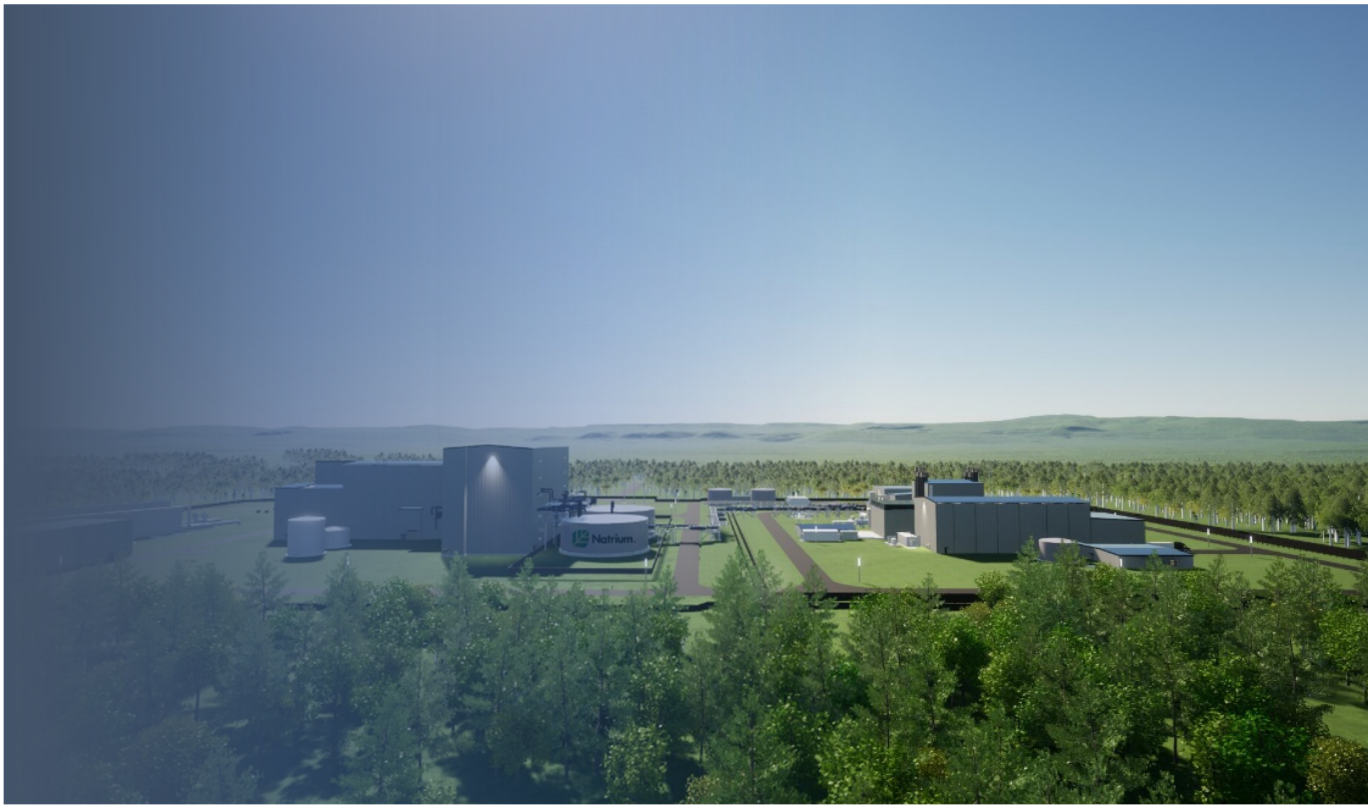

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# PSAR, Chapter 13

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## TABLE OF CONTENTS

**Chapter 13 Research and Development . . . . . 13.1-1**

13.1 Research and Development Program . . . . . 13.1-1

13.2 Structures, Systems, or Components Requiring Additional Research and Development . . . . . 13.2-1

13.2.1 Emissivity Surface Layer . . . . . 13.2-1

13.2.2 In-Vessel Fuel Handling System In-Vessel Transfer Machine Grapple Finger Operation . . . . . 13.2-1

13.2.3 Intermediate Heat Transport System Sodium – Salt Heat Exchanger Interaction. . . . . 13.2-2

13.2.4 Core Restraint System Bi-Metallic Assembly . . . . . 13.2-2


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## LIST OF TABLES

Table 13.2-1 Structures, Systems, or Components Requiring Research and Development . . . . . 13.2-3


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## LIST OF FIGURES

Figure 13.1-1 Technology Readiness Levels . . . . . 13.1-3


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# Chapter 13 Research and Development

This chapter identifies those structures, systems, or components (SSCs) that require further research and development (R&D) to confirm their design requirements. It also describes the R&D program that will be implemented consistent with the Natrium Reactor Testing Program (Reference 13.1-1).

## 13.1 Research and Development Program

The R&D program draws heavily from the United States Department of Energy, Technology Readiness Assessment / Technology Maturation Plan Process Guide (Reference 13.1-2) and the United States Government Accountability Office, GAO-20-48G (Reference 13.1-3). Both are used as guidance in developing these processes.

A technology is considered to be critical if it is new or if an existing technology is being used in a new or novel way, or is being applied in a new operating environment. Each critical technology (CT) goes through a technology readiness assessment and is assigned a technology readiness level (TRL). Technology readiness assessment is a systematic evaluation of the readiness that has been achieved by a particular technology or application of technology.

TRL is a numeric value (1-9) assigned to designate the level of maturity achieved by a particular technology. The R&D program scope includes SSCs with new or novel CT that are at TRL 4 or less at the time of construction permit application submittal that include design functions requiring additional test activities to advance the technology maturity.

A critical technology element is an individual element of an identified CT, which may be a design feature, performance characteristic, material of construction, or software for which performance needs to be adequately proven. Technology is proven based on evidence gathered through some combination of R&D, lab testing, material or other analysis, modeling, calculations, proof testing, factory testing, software verification and validation, or prototype testing. Proving the adequacy and readiness of a CT requires following a technology maturation plan. A technology maturation plan is a plan that evaluates the current state versus the required state of advancement of a particular CT and identifies activities required to advanced the technology maturity to TRL 5 or greater.

Figure 13.1-1 provides a representation of technology maturation through the various stages of R&D, verification and eventual application in an operating environment and illustrates the basis for TRL 4 or less as the basis of the R&D program.

SSCs that contain critical technology elements have been evaluated and CTs with a TRL of 4 or less were determined to need additional R&D by testing, analysis, or other means, to ensure they are functional in their design environment. R&D for non-safety-related with special treatment (NSRST) or non-safety-related with no special treatment (NST) SSCs is not required but may be included due to the nature of its function and the immature state of the technology.


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## References

13.1-1 TerraPower, LLC, “Natrium Reactor Testing Program,” NATD-EQT-RPT-0001, Revision 0.

13.1-2 U.S. Department of Energy Environmental Management, “Technology Readiness Assessment / Technology Maturation Plan Process Guide,” Mar 2008.

13.1-3 U.S. Government Accountability Office, GAO-20-48G, “Technology Readiness Assessment Guide – Best Practices for Evaluating the Readiness of Technology for Use in Acquisition Programs and Projects,” Jan 2020.



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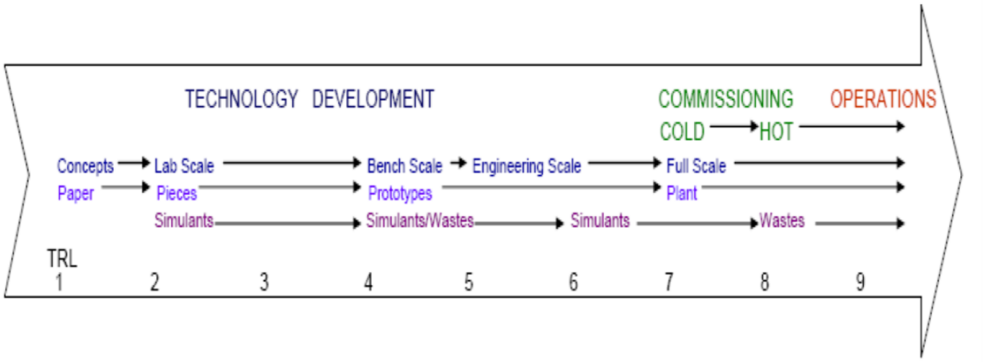
**Figure 13.1-1 Technology Readiness Levels**



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# 13.2 Structures, Systems, or Components Requiring Additional Research and Development

The research and development (R&D) activities described in this section are summarized in Table 13.2-1. The table includes those components that require additional R&D based on the technology readiness level (TRL).

## 13.2.1 Emissivity Surface Layer

The Reactor Air Cooling System provides passive safety-related cooling to remove decay heat from the Reactor Vessel (RV). Heat transfer is a key component of Reactor Air Cooling System performance, and an emissivity surface layer improves heat transfer between the RV and Guard Vessel, and between the Guard Vessel and the collector cylinder. Refer to Section 7.1.2 and Section 7.2.1 for additional details.

Reactor Air Cooling System performance analysis led to the incorporation of surface emissivity and enhanced features to the RV and Guard Vessel. RV and Guard Vessel are within the scope of the Reactor Enclosure System; however, defining the heat transfer performance for the components is the responsibility of Reactor Air Cooling System to ensure its performance is met.

Emissivity surface layers are assigned TRL 3 based on the need for surface emissivity layers to demonstrate durability to an irradiation environment. The following R&D scope remains to be completed to advance the technology maturity to TRL 5 or greater:

- Ion irradiation testing of candidate emissivity layers

Results required for TRL advancement include demonstration of durability to irradiation and potential manufacturing variance across production scale emissivity surface layers are known and documented. These activities are expected to be completed before completion of construction activities.

## 13.2.2 In-Vessel Fuel Handling System In-Vessel Transfer Machine Grapple Finger Operation

The In-Vessel Transfer Machine is a fuel handling machine that resides inside the RV. Its primary function is to manipulate core assemblies from within the RV during refueling operations. The In-Vessel Transfer Machine is equipped with a mechanical grapple assembly used to grip the handling socket at the top of a core assembly and lift it vertically out of a given location within the reactor and insert it into a different location. The In-Vessel Transfer Machine will be qualified for operation submerged in sodium. Refer to Section 7.3.3 for additional details.

In-Vessel Transfer Machine Grapple Finger operation is assigned TRL 3 based on the need for additional R&D testing in a representative sodium environment and will test the integrated performance and reliability of the grapple fingers. The interface between the grapple fingers and the core assembly handling socket is a critical interface. R&D testing of this interface, including operation in sodium to understand clearances and friction is required to advance the technology.


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The grapple finger operation will first be tested with an integrated subassembly test that will confirm the finger actuation mechanism behaves as expected in a sodium environment. The grapple finger actuation mechanism will also be tested in air with the pre-prototype test where off normal conditions including a misaligned handling socket can be simulated. Finally, grapple finger operation will be demonstrated in full scale air and sodium prototype testing. These activities are expected to be completed before completion of construction activities.

### 13.2.3 Intermediate Heat Transport System Sodium – Salt Heat Exchanger Interaction

The primary role of the Sodium-Salt Heat Exchanger is to transfer heat from the Intermediate Heat Transport System sodium to salt in the Nuclear Island Salt System where salt in the shell runs counter to sodium in the tubes. Refer to Section 7.1.4.3 for additional details.

This critical technology element is specially focused on the reaction energetics, byproducts, and evolution of any potential salt to sodium or sodium to salt leaks in the Intermediate Heat Transport System Sodium-Salt Heat Exchanger. A better understanding of the consequences of sodium and salt interacting will facilitate more appropriate decision making regarding the plant response to tube leaks in the Sodium-Salt Heat Exchanger.

Sodium to salt heat exchanger interaction is assigned TRL 5 which would put it outside of R&D scope. However, given the novelty of this critical technology element, additional R&D test activities have been identified that are not yet complete that fully justify the TRL. The following scope remains to be completed to advance the technology maturity and are expected to be completed before completion of construction activities.

- As the flow rate of sodium increases, verify that the exergonic reaction will decrease with a constant salt injection rate.
- Quantify the amount of oxide expected to develop.
- Determine the expected byproducts as a result of sodium-salt reactions.
- Quantify material corrosion.
- Develop and qualify methods of detection to inform operations and maintenance actions.

### 13.2.4 Core Restraint System Bi-Metallic Assembly

The bi-metallic assembly (BiMA) uses differential thermal expansion of materials to intentionally deflect and provide core compression force. It will be placed in specific non-fuel locations as required by the Core Restraint System. Refer to Section 7.1.1 for additional details.

The BiMA is currently evaluated at TRL 3 because the design concept has been studied in-depth but never implemented. The general science and concept of the BiMA is understood but the specific requirements have not been defined for the design. BiMAs have been simulated at the assembly level and as part of the Core Restraint System, but the specific components required for the BiMA are not defined with requirements or drawings and testing has not been performed in a relevant environment. Additional evaluation and R&D is required. These activities are expected to be completed before completion of construction activities.


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**Table 13.2-1 Structures, Systems, or Components Requiring Research and Development**

| System | Component | Critical Technology Element | Current Technology Readiness Level |
|---|---|---|---|
| Reactor Air Cooling System | Collector<br>Cylinder | Emissivity Surface Layer | 3 |
| Reactor Enclosure System | Guard Vessel<br>Reactor Vessel | Emissivity Surface Layer | 3 |
| In-Vessel Fuel Handling System | In-Vessel Transfer Machine | Grapple Finger Operation | 3 |
| Intermediate Heat Transport System | Sodium-Salt Heat Exchanger | Sodium - Salt Interaction | 5 |
| Reactor Core System | Core Restraint System | Bi-Metallic Assembly | 3 |


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