DESIGN CONTROL DOCUMENT FOR THE
US-APWR
Chapter 3
Design of Structures, Systems, Components and Equipment

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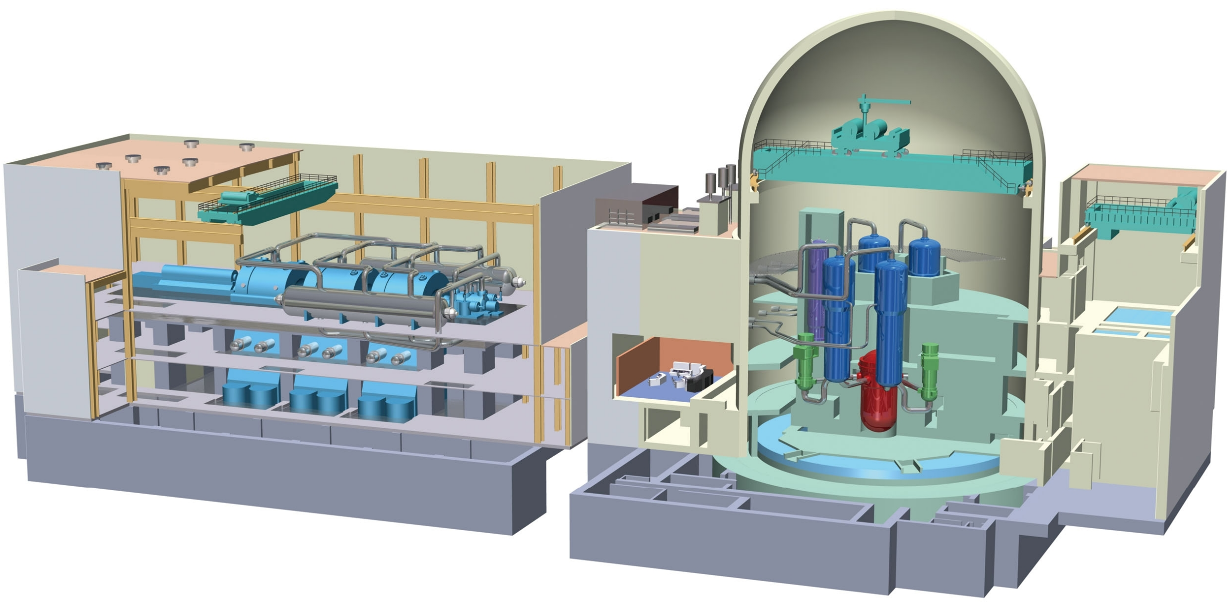

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CONTENTS

Page

3.0 DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS, AND EQUIPMENT 3.1-1

3.1 Conformance with NRC General Design Criteria 3.1-1

3.1.1 Overall Requirements 3.1-1

3.1.1.1 Criterion 1 – Quality Standards and Records 3.1-1

3.1.1.2 Criterion 2 – Design Bases for Protection Against Natural Phenomena 3.1-2

3.1.1.3 Criterion 3 – Fire Protection 3.1-3

3.1.1.4 Criterion 4 – Environmental and Dynamic Effects Design Bases 3.1-4

3.1.1.5 Criterion 5 – Sharing of Structures, Systems, and Components 3.1-4

3.1.2 Protection by Multiple Fission Product Barriers 3.1-5

3.1.2.1 Criterion 10 – Reactor Design 3.1-5

3.1.2.2 Criterion 11 – Reactor Inherent Protection 3.1-6

3.1.2.3 Criterion 12 – Suppression of Reactor Power Oscillations 3.1-6

3.1.2.4 Criterion 13 – Instrumentation and Control 3.1-7

3.1.2.5 Criterion 14 – Reactor Coolant Pressure Boundary 3.1-7

3.1.2.6 Criterion 15 – Reactor Coolant System Design 3.1-8

3.1.2.7 Criterion 16 – Containment Design 3.1-9

3.1.2.8 Criterion 17 – Electrical Power Systems 3.1-9

3.1.2.9 Criterion 18 – Inspection and Testing of Electric Power Systems 3.1-11

3.1.2.10 Criterion 19 – Control Room 3.1-12

3.1.3 Protection and Reactivity Control Systems 3.1-13

3.1.3.1 Criterion 20 – Protection System Functions 3.1-13

3.1.3.2 Criterion 21 – Protection System Reliability and Testability 3.1-14

3.1.3.3 Criterion 22 – Protection System Independence 3.1-15

3.1.3.4 Criterion 23 – Protection System Failure Modes 3.1-16

3.1.3.5 Criterion 24 – Separation of Protection and Control Systems 3.1-16

3.1.3.6 Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions 3.1-17

3.1.3.7 Criterion 26 – Reactivity Control System Redundancy and Capability 3.1-17

3.1.3.8 Criterion 27 – Combined Reactivity Control Systems Capability 3.1-18

3.1.3.9 Criterion 28 – Reactivity Limits 3.1-19

3.1.3.10 Criterion 29 – Protection against Anticipated Operational Occurrences 3.1-19

3.1.4 Fluid Systems 3.1-20

3.1.4.1 Criterion 30 – Quality of Reactor Coolant Pressure Boundary 3.1-20

3.1.4.2 Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary 3.1-20

3.1.4.3 Criterion 32 – Inspection of Reactor Coolant Pressure Boundary 3.1-21

3.1.4.4 Criterion 33 – Reactor Coolant Makeup 3.1-22

3.1.4.5 Criterion 34 – Residual Heat Removal 3.1-23

3.1.4.6 Criterion 35 – Emergency Core Cooling 3.1-24

3.1.4.7 Criterion 36 – Inspection of Emergency Core Cooling System 3.1-25

3.1.4.8 Criterion 37 – Testing of Emergency Core Cooling System 3.1-26

3.1.4.9 Criterion 38 – Containment Heat Removal 3.1-27

3.1.4.10 Criterion 39 – Inspection of Containment Heat Removal System 3.1-27

3.1.4.11 Criterion 40 – Testing of Containment Heat Removal System 3.1-28

3.1.4.12 Criterion 41 – Containment Atmosphere Cleanup 3.1-28

3.1.4.13 Criterion 42 – Inspection of Containment Atmosphere Cleanup Systems 3.1-29

3.1.4.14 Criterion 43 – Testing of Containment Atmospheric Cleanup Systems 3.1-30

3.1.4.15 Criterion 44 – Cooling Water 3.1-30

3.1.4.16 Criterion 45 – Inspection of Cooling Water System 3.1-31

3.1.4.17 Criterion 46 – Testing of Cooling Water System 3.1-31

3.1.5 Reactor Containment 3.1-32

3.1.5.1 Criterion 50 – Containment Design Basis 3.1-32

3.1.5.2 Criterion 51 – Fracture Prevention of Containment Pressure Boundary 3.1-33

3.1.5.3 Criterion 52 – Capability for Containment Leakage Rate Testing 3.1-33

3.1.5.4 Criterion 53 – Provisions for Containment Testing and Inspection 3.1-33

3.1.5.5 Criterion 54 – Piping Systems Penetrating Containment 3.1-34

3.1.5.6 Criterion 55 – Reactor Coolant Pressure Boundary Penetrating Containment 3.1-35

3.1.5.7 Criterion 56 – Primary Containment Isolation 3.1-35

3.1.5.8 Criterion 57 – Closed System Isolation Valves 3.1-36

3.1.6 Fuel and Reactivity Control 3.1-37

3.1.6.1 Criterion 60 – Control of Releases of Radioactive Materials to the Environment 3.1-37

3.1.6.2 Criterion 61 – Fuel Storage and Handling and Radioactivity Control 3.1-38

3.1.6.3 Criterion 62 – Prevention of Criticality in Fuel Storage and Handling 3.1-39

3.1.6.4 Criterion 63 – Monitoring Fuel and Waste Storage 3.1-40

3.1.6.5 Criterion 64 – Monitoring Radioactivity Releases 3.1-40

3.1.7 Combined License Information 3.1-41

3.1.8 References 3.1-41
3.2 Classification of Structures, Systems, and Components 3.2-1
3.2.1 Seismic Classification 3.2-1
3.2.1.1 Definitions 3.2-3
3.2.1.2 Classifications 3.2-5
3.2.1.3 Classification of Building Structures 3.2-5
3.2.2 System Quality Group Classification 3.2-5
3.2.2.1 Equipment Class 1 ..... 3.2-7
3.2.2.2 Equipment Class 2 ..... 3.2-8
3.2.2.3 Equipment Class 3 ..... 3.2-9
3.2.2.4 Equipment Class 4 ..... 3.2-10
3.2.2.5 Other Equipment Classes 3.2-11
3.2.2.6 Inspection Requirements 3.2-12
3.2.3 Combined License Information 3.2-13
3.2.4 References 3.2-13
3.3 Wind, Tornado and Hurricane Loadings 3.3-1
3.3.1 Wind Loadings 3.3-1
3.3.1.1 Severe Wind Velocity and Recurrence Interval 3.3-1
3.3.1.2 Determination of Applied Forces 3.3-1
3.3.2 Tornado and Hurricane Loadings 3.3-3
3.3.2.1 Applicable Design Parameters 3.3-3
3.3.2.2 Determination of Forces on Structures 3.3-4
3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado and Hurricane Loads 3.3-7
3.3.3 Combined License Information 3.3-8
3.3.4 References 3.3-8
3.4 Water Level (Flood) Design 3.4-1
3.4.1 Flood Protection 3.4-1

3.4.1.1 Flood Protection for Safety and NonSafety-Related Structures, Systems, and Components 3.4-1

3.4.1.2 Flood Protection from External Sources 3.4-2

3.4.1.3 Flood Protection from Internal Sources 3.4-5

3.4.1.4 Evaluation of External Flooding 3.4-8

3.4.1.5 Evaluation of Internal Flooding 3.4-8

3.4.2 Analysis Procedures 3.4-28

3.4.3 Combined License Information 3.4-28

3.4.4 References 3.4-29

3.5 Missile Protection 3.5-1

3.5.1 Missile Selection and Description 3.5-4

3.5.1.1 Internally Generated Missiles (Outside Containment) 3.5-4

3.5.1.2 Internally Generated Missiles (Inside Containment) 3.5-8

3.5.1.3 Turbine Missiles 3.5-10

3.5.1.4 Missiles Generated by Tornadoes and Hurricanes 3.5-11

3.5.1.5 Site Proximity Missiles (Except Aircraft) 3.5-13

3.5.1.6 Aircraft Hazards 3.5-13

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles 3.5-13

3.5.3 Barrier Design Procedures 3.5-13

3.5.3.1 Evaluation of Local Structural Effects 3.5-14

3.5.3.2 Evaluation of Overall Structural Effects 3.5-17

3.5.4 Combined License Information 3.5-18

3.5.5 References 3.5-19

3.6 Protection Against Dynamic Effects Associated with Postulated Rupture of Piping 3.6-1

3.6.1 Plant Design for Protection against Postulated Piping Failure in Fluid Systems Inside and Outside Containment 3.6-1

3.6.1.1 Design Basis 3.6-2

3.6.1.2 Safety Evaluation 3.6-6

3.6.1.3 Postulated Failures Associated with Site-Specific Piping 3.6-8

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping 3.6-8

3.6.2.1 Criteria used to Define Break and Crack Location and Configuration 3.6-9

3.6.2.2 Guard Pipe Assembly Design Criteria 3.6-16

3.6.2.3 Analytic Methods to Define Forcing Functions and Response Models 3.6-16

3.6.2.4 Dynamic Analysis Methods to Verify Integrity and Operability 3.6-19

3.6.2.5 Implementation of Criteria Dealing with Special Features 3.6-25

3.6.2.6 Outline of Pipe Break Hazard Analysis Report(s) 3.6-25

3.6.3 LBB Evaluation Procedures 3.6-26

3.6.3.1 Application of LBB Criteria 3.6-26

3.6.3.2 Design Criteria for LBB 3.6-27

3.6.3.3 Potential Failure Mechanism for Piping 3.6-28

3.6.3.4 Analytical Methods and Criteria 3.6-32

3.6.3.5 Technical Report 3.6-36

3.6.4 Combined License Information 3.6-37

3.6.5 References 3.6-38

3.7 Seismic Design 3.7-1

3.7.1 Seismic Design Parameters 3.7-1

3.7.1.1 Design Ground Motion 3.7-1

3.7.1.2 Percentage of Critical Damping Values 3.7-7

3.7.1.3 Supporting Media for Seismic Category I Structures 3.7-10

3.7.2 Seismic System Analysis 3.7-12

3.7.2.1 Seismic Analysis Methods 3.7-13

3.7.2.2 Natural Frequencies and Responses 3.7-16

3.7.2.3 Procedures Used for Analytical Modeling 3.7-17

3.7.2.4 Soil-Structure Interaction 3.7-32

3.7.2.5 Development of Floor Response Spectra 3.7-37

3.7.2.6 Three Components of Earthquake Motion 3.7-38

3.7.2.7 Combination of Modal Responses 3.7-39

3.7.2.8 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures 3.7-42

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra 3.7-44

3.7.2.10 Use of Constant Vertical Static Factors 3.7-45

3.7.2.11 Method Used to Account for Torsional Effects 3.7-45

3.7.2.12 Comparison of Responses 3.7-45

3.7.2.13 Methods for Seismic Analysis of Dams 3.7-46

3.7.2.14 Determination of Dynamic Stability of Seismic Category I Structures 3.7-46

3.7.2.15 Analysis Procedure for Damping 3.7-46

3.7.3 Seismic Subsystem Analysis 3.7-46

3.7.3.1 Seismic Analysis Methods 3.7-47

3.7.3.2 Procedures Used for Analytical Modeling 3.7-53

3.7.3.3 Analysis Procedure for Damping 3.7-54

3.7.3.4 Three Components of Earthquake Motion 3.7-54

3.7.3.5 Combination of Modal Responses 3.7-55

3.7.3.6 Use of Constant Vertical Static Factors 3.7-55

3.7.3.7 Buried Seismic Category I Piping, Conduits, and Tunnels 3.7-55

3.7.3.8 Methods for Seismic Analysis of Category I Concrete Dams 3.7-55

3.7.3.9 Methods for Seismic Analysis of Aboveground Tanks 3.7-56

3.7.4 Seismic Instrumentation 3.7-56

3.7.4.1 Comparison with Regulatory Guide 1.12 ..... 3.7-56

3.7.4.2 Location and Description of Instrumentation 3.7-58

3.7.4.3 Control Room Operator Notification 3.7-59
3.7.4.4 Comparison with Regulatory Guide 1.166 .... 3.7-60
3.7.4.5 Instrument Surveillance (Including calibration and testing) 3.7-60
3.7.4.6 Program Implementation 3.7-61
3.7.5 Combined License Information 3.7-61
3.7.6 References 3.7-64
3.8 Design of Category I Structures 3.8-1
3.8.1 Concrete Containment 3.8-1
3.8.1.1 Description of the Containment 3.8-1
3.8.1.2 Applicable Codes, Standards, and Specifications 3.8-4
3.8.1.3 Loads and Load Combinations 3.8-5
3.8.1.4 Design and Analysis Procedures 3.8-9
3.8.1.5 Structural Acceptance Criteria 3.8-17
3.8.1.6 Material, Quality Control, and Special Construction Techniques 3.8-25
3.8.1.7 Testing and Inservice Inspection Requirements 3.8-30
3.8.2 Steel Containment 3.8-33
3.8.3 Concrete and Steel Internal Structures of Concrete Containment 3.8-34
3.8.3.1 Description of the Structures Internal to Containment 3.8-34
3.8.3.2 Applicable Codes, Standards, and Specifications 3.8-38
3.8.3.3 Loads and Load Combinations 3.8-38
3.8.3.4 Design and Analysis Procedures 3.8-41
3.8.3.5 Structural Acceptance Criteria 3.8-54
3.8.3.6 Materials, Quality Control, and Special Construction Techniques 3.8-55
3.8.3.7 Testing and Inservice Inspection Requirements 3.8-55
3.8.4 Other Seismic Category I Structures 3.8-55

3.8.4.1 Description of the Structures 3.8-56
3.8.4.2 Applicable Codes, Standards, and Specifications 3.8-59
3.8.4.3 Loads and Load Combinations 3.8-61
3.8.4.4 Design and Analysis Procedures 3.8-69
3.8.4.5 Structural Acceptance Criteria 3.8-81
3.8.4.6 Materials, Quality Control, and Special Construction Techniques 3.8-82
3.8.4.7 Testing and Inservice Inspection Requirements 3.8-87
3.8.5 Foundations 3.8-88
3.8.5.1 Description of the Foundations 3.8-88
3.8.5.2 Applicable Codes, Standards and Specifications 3.8-89
3.8.5.3 Loads and Load Combinations 3.8-90
3.8.5.4 Design and Analysis Procedures 3.8-90
3.8.5.5 Structural Acceptance Criteria 3.8-95
3.8.5.6 Materials, Quality Control, and Special Construction Techniques 3.8-101
3.8.5.7 Testing and Inservice Inspection Requirements 3.8-101
3.8.6 Combined License Information 3.8-101
3.8.7 References 3.8-104
3.9 Mechanical Systems and Components 3.9-1
3.9.1 Special Topics for Mechanical Components 3.9-1
3.9.1.1 Design Transients 3.9-1
3.9.1.2 Computer Programs Used in Analyses 3.9-14
3.9.1.3 Experimental Stress Analysis 3.9-16
3.9.1.4 Considerations for the Evaluation for the Faulted Condition 3.9-16
3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment 3.9-16
3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects 3.9-16

3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment 3.9-20

3.9.2.3 Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions .. 3.9-23

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals 3.9-27

3.9.2.5 Dynamic System Analysis of the Reactor Internals under Faulted Conditions 3.9-29

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results 3.9-33

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures 3.9-34

3.9.3.1 Loading Combinations, System Operating Transients, and Stress Limits 3.9-36

3.9.3.2 Design and Installation of Pressure-Relief Devices 3.9-43

3.9.3.3 Pump and Valve Operability Assurance 3.9-45

3.9.3.4 Component Supports 3.9-48

3.9.4 Control Rod Drive Systems 3.9-58

3.9.4.1 Descriptive Information of CRDS 3.9-58

3.9.4.2 Applicable CRDS Design Specifications 3.9-63

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations .. 3.9-65

3.9.4.4 CRDS Operability Assurance Program 3.9-65

3.9.5 Reactor Pressure Vessel Internals 3.9-67

3.9.5.1 Design Arrangements 3.9-67

3.9.5.2 Loading Conditions 3.9-74

3.9.5.3 Design Bases 3.9-75

3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints 3.9-81

3.9.6.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints 3.9-85

3.9.6.2 IST Program for Pumps 3.9-86

3.9.6.3 IST Program for Valves 3.9-87

3.9.6.4 IST Program for Dynamic Restraints 3.9-96

3.9.6.5 Relief Request and Authorization to ASME OM Code 3.9-100

3.9.7 [Reserved] 3.9-101

3.9.8 [Reserved] 3.9-101

3.9.9 Combined License Information 3.9-101

3.9.10 References 3.9-101

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment 3.10-1

3.10.1 Seismic Qualification Criteria 3.10-2

3.10.1.1 Qualification Standards 3.10-2

3.10.1.2 Performance Requirements for Seismic Qualification 3.10-3

3.10.1.3 Performance Criteria 3.10-4

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation 3.10-4

3.10.2.1 Seismic Qualification of Instrumentation and Electrical Equipment 3.10-11

3.10.2.2 Seismic and Operability Qualification of Active Mechanical Equipment 3.10-12

3.10.2.3 Pump Motor and Valve Operator Qualification 3.10-16

3.10.2.4 Seismic Qualification of Other Seismic Category I Mechanical Equipment 3.10-17

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation 3.10-17

3.10.4 Test and Analyses Results and Experience Database 3.10-18

3.10.4.1 Implementation Program and Milestones 3.10-19

3.10.4.2 Experience Based Qualification 3.10-19

3.10.5 Combined License Information 3.10-20

3.10.6 References 3.10-20

3.11 Environmental Qualification of Mechanical and Electrical Equipment 3.11-1

3.11.1 Equipment Location and Environmental Conditions 3.11-5
3.11.1.1 Equipment Identification 3.11-5
3.11.1.2 Definition of Environmental Conditions 3.11-6
3.11.1.3 Equipment Operability Times (Term) 3.11-7
3.11.1.4 Standard Review Plan Evaluation 3.11-7
3.11.2 Qualification Tests and Analyses 3.11-7
3.11.2.1 Environmental Qualification of Electrical and Mechanical Equipment 3.11-8
3.11.2.2 Environmental Qualification of Mechanical Equipment 3.11-10
3.11.3 Qualification Test Results 3.11-10
3.11.4 Loss of Ventilation 3.11-11
3.11.5 Estimated Chemical and Radiation Environment 3.11-11
3.11.5.1 Chemical Environment 3.11-11
3.11.5.2 Radiation Environment 3.11-12
3.11.6 Qualification of Mechanical Equipment 3.11-12
3.11.7 Combined License Information 3.11-13
3.11.8 References 3.11-13
3.12 Piping Design Review 3.12-1
3.12.1 Introduction 3.12-1
3.12.2 Codes and Standards 3.12-1
3.12.2.1 American Society of Mechanical Engineers Boiler and Pressure Vessel Code 3.12-1
3.12.2.2 American Society of Mechanical Engineers Code Cases 3.12-2
3.12.2.3 Design Specification 3.12-2
3.12.3 Piping Analysis Methods 3.12-2
3.12.3.1 Experimental Stress Analyses 3.12-2
3.12.3.2 Modal Response Spectrum Method 3.12-2
3.12.3.3 Response Spectra Method (or Independent Support Motion Method) 3.12-5

3.12.3.4 Time-History Method 3.12-6

3.12.3.5 Inelastic Analyses Methods 3.12-6

3.12.3.6 Small-Bore Piping Method 3.12-6

3.12.3.7 Non-seismic/Seismic Interaction (II/I) 3.12-6

3.12.3.8 Seismic Category I Buried Piping 3.12-7

3.12.4 Piping Modeling Technique 3.12-7

3.12.4.1 Computer Codes 3.12-7

3.12.4.2 Dynamic Piping Model 3.12-9

3.12.4.3 Piping Benchmark Program 3.12-10

3.12.4.4 Decoupling Criteria 3.12-11

3.12.5 Piping Stress Analysis Criteria 3.12-11

3.12.5.1 Seismic Input Envelope vs. Site-Specific Spectra 3.12-11

3.12.5.2 Design Transients 3.12-12

3.12.5.3 Loadings and Load Combination 3.12-12

3.12.5.4 Damping Values 3.12-14

3.12.5.5 Combination of Modal Responses 3.12-14

3.12.5.6 High-Frequency Modes 3.12-14

3.12.5.7 Fatigue Evaluation of ASME Code Class 1 Piping 3.12-14

3.12.5.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping 3.12-14

3.12.5.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System 3.12-15

3.12.5.10 Thermal Stratification 3.12-16

3.12.5.11 Safety Relief Valve Design, Installation, and Testing 3.12-17

3.12.5.12 Functional Capability 3.12-17

3.12.5.13 Combination of Inertial and Seismic Anchor Motion Effects 3.12-18

3.12.5.14 Operating-Basis Earthquake as a Design Load 3.12-18

3.12.5.15 Welded Attachments 3.12-18
3.12.5.16 Modal Damping for Composite Structures 3.12-18
3.12.5.17 Minimum Temperature for Thermal Analyses 3.12-18
3.12.5.18 Intersystem Loss-of-Coolant Accident 3.12-18
3.12.5.19 Effects of Environment on Fatigue Design 3.12-18
3.12.6 Piping Support Design Criteria 3.12-19
3.12.6.1 Applicable Codes 3.12-19
3.12.6.2 Jurisdictional Boundaries 3.12-20
3.12.6.3 Loads and Load Combinations 3.12-20
3.12.6.4 Pipe Support Baseplate and Anchor Bolt Design 3.12-21
3.12.6.5 Use of Energy Absorber and Limit Stops 3.12-22
3.12.6.6 Use of Snubbers 3.12-22
3.12.6.7 Pipe Support Stiffness 3.12-23
3.12.6.8 Seismic Self-Weight Excitation 3.12-23
3.12.6.9 Design of Supplementary Steel 3.12-23
3.12.6.10 Consideration of Friction Forces 3.12-24
3.12.6.11 Pipe Support Gaps and Clearances 3.12-24
3.12.6.12 Instrumentation Line Support Criteria 3.12-24
3.12.6.13 Pipe Deflection Limit 3.12-25
3.12.7 Combined License Information 3.12-25
3.12.8 References 3.12-25
3.13 Threaded Fasteners (ASME Code Class 1, 2, and 3) 3.13-1
3.13.1 Design Considerations 3.13-1
3.13.1.1 Materials Selection 3.13-1
3.13.1.2 Special Materials Fabrication Processes and Special Controls 3.13-3
3.13.1.3 Fracture Toughness Requirements for Threaded Fasteners Made of Ferritic Materials 3.13-7

3.13.1.4 [Reserved] 3.13-10

3.13.1.5 Certified Material Test Reports 3.13-10

3.13.2 Inservice Inspection Requirements 3.13-11

3.13.3 Combined License Information 3.13-13

3.13.4 References 3.13-13

Appendix 3A Heating, Ventilation, and Air Conditioning Ducts and Duct Supports 3A-1

Appendix 3B Bounding Analysis Curve Development for Leak Before Break Evaluation of High-Energy Piping for US-APWR 3B-1

Appendix 3C Reactor Coolant Loop Analysis Methods 3C-1

Appendix 3D Equipment Qualification of Electrical and Mechanical Equipment 3D-1

Appendix 3E High Energy and Moderate Energy Piping in the Prestressed Concrete Containment Vessel and Reactor Building 3E-1

Appendix 3F Design of Conduit and Conduit Supports 3F-1

Appendix 3G Seismic Qualification of Cable Trays and Supports 3G-1

Appendix 3H Model Properties for Lumped Mass Stick Models of R/B-PCCV-Containment Internal Structure on a Common Basemat 3H-1

Appendix 3I In-Structure Response Spectra 3I-1

Appendix 3J Reactor, Power Source, Containment Internal and the Essential Service Water Pipe Chase Structural Design 3J-1

Appendix 3K Components Protected From Internal Flooding 3K-1

Appendix 3L Critical Sections for Reactor Building Complex 3L-1

TABLES

		Page
Table 3.2-1	Non-Safety Components Required for Normal Shutdown	3.2-16
Table 3.2-2	Classification of Mechanical and Fluid Systems, Components, and Equipment	3.2-18
Table 3.2-3	Comparison of Various Requirements to Equipment Class	3.2-72
Table 3.2-4	Seismic Classification of Buildings and Structures	3.2-73
Table 3.4-1	No crack design and crack/break postulation for moderate energy piping for flood protection design	3.4-30
Table 3.6-1	High and Moderate Energy Fluid Systems	3.6-41
Table 3.6-2	List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Internal and External Fluids	3.6-43
Table 3.7.1-1	CSDRS Horizontal Acceleration Values and Control Points	3.7-70
Table 3.7.1-2	CSDRS Vertical Acceleration Values and Control Points	3.7-71
Table 3.7.1-3	Summary of SRP 3.7 Option 1, Approach 1 Requirements Compliance	3.7-72
Table 3.7.1-4	Magnitudes and Distance Bins and Strong Motion Duration Criteria	3.7-73
Table 3.7.1-5	CEUS V/A & AD/V2 Mean Ratios ± One Standard Deviation	3.7-74
Table 3.7.1-6	Generic Soil Profile Categories	3.7-75
Table 3.7.1-7	Magnitudes, Distances, and Median Peak Accelerations	3.7-76
Table 3.7.2-1	Summary of Dynamic Analysis and Combination Techniques	3.7-77
Table 3.7.2-2	Concrete Material Constants	3.7-78
Table 3.7.2-3	Material Properties of Models Used for Seismic Response Analyses	3.7-79
Table 3.7.2-4	Fixed Base Dynamic Properties of US-APWR Category I Structures	3.7-80
Table 3.7.3-1(a)	SSE Damping Values	3.7-82
Table 3.7.3-1(b)	OBE Damping Values	3.7-82
Table 3.8.1-1	US-APWR PCCV Basic Design Specification	3.8-111

Table 3.8.1-2	PCCV Load Combinations and Load Factors	3.8-112
Table 3.8.1-3	Thermal Conditions of the R/B and PCCV	3.8-113
Table 3.8.1-4	Summary of PCCV Models and Analysis Methods	3.8-115
Table 3.8.3-1	Deleted	3.8-116
Table 3.8.3-2	Design Pressures within CIS	3.8-117
Table 3.8.3-3	Summary of CIS Models and Analysis Methods	3.8-119
Table 3.8.3-4	Summary of CIS Stiffness and Damping Values for Seismic Analysis	3.8-120
Table 3.8.3-5	Definition of Critical Section and Thicknesses for Containment Internal Structure	3.8-121
Table 3.8.3-6	Deleted	3.8-122
Table 3.8.3-7	Summary of Confirmatory Physical Test Results	3.8-123
Table 3.8.4-1	Deleted	3.8-126
Table 3.8.4-2	Thermal Conditions of the PS/Bs	3.8-127
Table 3.8.4-3	Load Combinations and Load Factors for Seismic Category I Concrete Structures	3.8-128
Table 3.8.4-4	Load Combinations and Load Factors for Seismic Category I Steel Structures	3.8-130
Table 3.8.4-5	Deleted	3.8-132
Table 3.8.4-6	R/B West Common Wall, SECTION 1, Details of Wall Reinforcement	3.8-133
Table 3.8.4-7	R/B South Interior Wall, SECTION 2, Details of Wall Reinforcement	3.8-134
Table 3.8.4-8	North Exterior Wall of Spent Fuel Pit, SECTION 3, Details of Wall Reinforcement	3.8-135
Table 3.8.4-9	R/B South Exterior Wall, SECTION 4, Details of Wall Reinforcement	3.8-136
Table 3.8.4-10	R/B Central Interior Wall, SECTION 5, Details of Wall Reinforcement	3.8-138
Table 3.8.4-11	Spent Fuel Pit Slab, AREA 3, Details of Slab Reinforcement	3.8-140

Table 3.8.4-12 Emergency Feedwater Pit Slab, AREA 4, Details of Slab Reinforcement 3.8-140

Table 3.8.4-13 Typical Reinforcement in West PS/B South Exterior Wall – SECTION 1 .... 3.8-141

Table 3.8.4-14 Typical Reinforcement in West PS/B Interior Wall – SECTION 2 .... 3.8-142

Table 3.8.4-15 Typical Reinforcement in West PS/B Floor at Elevation 3′-7″– AREA 1 .... 3.8-142

Table 3.8.4-16 Typical Reinforcement in West PS/B North Exterior Wall – SECTION 3 .... 3.8-143

Table 3.8.4-17 Typical Reinforcement in East PS/B East Exterior Wall – SECTION 1 .... 3.8-144

Table 3.8.4-18 Typical Reinforcement in East PS/B Interior Wall – SECTION 2 .... 3.8-145

Table 3.8.4-19 Typical Reinforcement in East PS/B Floor at Elevation 3’-7” – AREA 1 .... 3.8-145

Table 3.8.4-20 Typical Reinforcement in ESWPC Exterior Wall – SECTION 1 .... 3.8-146

Table 3.8.4-21 Typical Reinforcement in ESWPC Interior Wall – SECTION 2 .... 3.8-146

Table 3.8.4-22 Typical Reinforcement in ESWPC Floor at Elevation -15’-8” – AREA 1 .... 3.8-147

Table 3.8.4-23 Dynamic + Static Lateral Earth Pressures and Passive Earth Pressures 3.8-148

Table 3.8.5-1 Load Combinations and Required Minimum Factors of Safety for Stability of Seismic Category I and II Structures 3.8-149

Table 3.8.5-2 Concrete Properties 3.8-150

Table 3.8.5-3 Deleted 3.8-151

Table 3.8.5-4 Critical Sections of R/B Complex Basemat Evaluation 3.8-152

Table 3.8.5-5 Deleted 3.8-154

Table 3.8.5-6 Load Combinations and Calculated Minimum Factors of Safety for Stability of Seismic Category I and II Structures 3.8-155

Table 3.9-1 RCS Design Transients 3.9-108

Table 3.9-2 Reactor Internals Interface Load and Displacement Limits 3.9-110

Table 3.9-3 Design Loading Combinations for ASME Code, Section III, Class 1, 2, and 3 CS Systems and Components 3.9-111

Table 3.9-4 Design Loading Combinations for Supports for ASME Code, Section III, Class 1, 2, and 3 Components 3.9-112

Table 3.9-5 ASME Code, Section III, Class 1, 2, 3, CS, and Support Load Symbols and Definitions 3.9-113

Table 3.9-6 Stress Criteria for ASME Code, Section III, Class 1, Components and Supports and Class CS Core Supports......... 3.9-115

Table 3.9-7 List of Active Pumps 3.9-116

Table 3.9-8 Stress Criteria for ASME Code, Section III Class and Components and Supports 3.9-119

Table 3.9-9 CRDM Pressure Housing Loading Conditions and Load Combinations.. 3.9-120

Table 3.9-10 CRDM Pressure Housing Stress Categories and Stress Intensity Limits 3.9-121

Table 3.9-11 Core Support Structures and Threaded Structural Fasteners Loading Conditions and Load Combinations............................ 3.9-122

Table 3.9-12 Core Support Structures Stress Categories and Stress Intensity Limits 3.9-123

Table 3.9-13 Pump IST... 3.9-124

Table 3.9-14 Valve Inservice Test Requirements... 3.9-130

Table 3.9-15 Load combinations and stress limits for Secondary Core Support Structures 3.9-255

Table 3.11-1 Summary of US-APWR EQ Program Mechanical SSC Interfaces... 3.11-16

Table 3.12-1 ASME Code, Section III, Class 1, 2, 3, CS and Support Load Symbols and Definitions 3.12-29

Table 3.12-2 Loading Combinations for ASME Code, Section III, Class 1 Piping.. 3.12-30

Table 3.12-3 Loading Combinations for ASME Code, Section III, Class 2 and 3 Piping.. 3.12-32

Table 3.12-4 Loading Combinations for Piping Supports 3.12-33

CONTENTS

Page

Table 3.12-5 Piping Functional Capability – ASME Code, Section III, Class 1, 2, and 3 .... 3.12-34

Table 3.12-6 Design Transients of Reactor Coolant Piping Branch Connections 3.12-35

Table 3.12-7 Evaluation from Viewpoint of Valve Seat Leakage 3.12-36

Table 3.12-8 Evaluation from Viewpoint of Valve Gland Leakage 3.12-37

Table 3.13-1 ASME Code, Section III Criteria for Selection and Testing of Bolting Materials 3.13-15

Table 3.13-2 ASME Code, Section XI Examination Categories for Inservice Inspections of Mechanical Joints in ASME Code Class 1, 2, and 3 Systems that are Secured by Threaded Fasteners 3.13-16

FIGURES

		Page
Figure 3.6-1	Break Exclusion Region-Main Steam Pipe Room	3.6-47
Figure 3.6-2	Characteristics of Discharging Jets	3.6-48
Figure 3.6-3	Typical Rupture Restraints	3.6-49
Figure 3.6-4	LBB Evaluation Procedure	3.6-50
Figure 3.7.1-1	US–APWR Horizontal CSDRS	3.7-83
Figure 3.7.1-2	US–APWR Vertical CSDRS	3.7-84
Figure 3.7.1-3	Acceleration, Velocity, and Displacement Time History for Component H1 [NS]	3.7-85
Figure 3.7.1-4	Acceleration, Velocity, and Displacement Time History f or Component H2 [EW]	3.7-86
Figure 3.7.1-5	Acceleration, Velocity, and Displacement Time History for Component V3	.7-87
Figure 3.7.1-6a	Damped Response Spectra Plots for Northridge Mount Baldy Component H1 (180) [NS]	3.7-88
Figure 3.7.1-6b	Damped Response Spectra Plots for Northridge Mount Baldy Component H1 (180) [NS]	3.7-88
Figure 3.7.1-7a	Damped Response Spectra Plots for Northridge Mount Baldy Component H2 (090) [EW]	3.7-89
Figure 3.7.1-7b	Damped Response Spectra Plots for Northridge Mount Baldy Component H2 (090) [EW]	3.7-89
Figure 3.7.1-8a	Damped Response Spectra Plots for Mount Baldy Component V (UP)	3.7-90
Figure 3.7.1-8b	Damped Response Spectra Plots for Mount Baldy Component V (UP)	3.7-90
Figure 3.7.1-9	US-APWR Final Horizontal and Vertical Target PSDs	3.7-91
Figure 3.7.1-10	Smoothed Power Spectral Density Plots for Component H1 (180)	3.7-92
Figure 3.7.1-11	Smoothed Power Spectral Density Plots for Component H2 (090)	3.7-92
Figure 3.7.1-12	Smoothed Power Spectral Density Plots for Component V (UP)	3.7-93
Figure 3.7.1-13	Arias Intensities of the Northridge – Mount Baldy Artificial Time History Components Showing 5%-75% Duration	3.7-93

Figure 3.7.1-14 Six Generic Soil Profiles, Shear Wave Velocity $V_{(s)}$ 3.7-94

Figure 3.7.1-15 Six Generic Soil Profiles, Compression Wave Velocity (V_p)........ 3.7-95

Figure 3.7.1-16 Six Generic Soil Profiles, Shear Wave Velocity Damping (%)..... 3.7-96

Figure 3.7.2-1 Integrated R/B Complex Dynamic FE Model 3.7-97

Figure 3.7.2-2 Integrated R/B Complex Detailed FE Model............................... 3.7-98

Figure 3.7.2-3 PCCV Detailed Model... 3.7-99

Figure 3.7.2-4 CIS Detailed Model.. 3.7-100

Figure 3.7.2-5 Section View of Dynamic R/B Complex FE Model Looking East 3.7-101

Figure 3.7.2-6 Section View of Dynamic R/B Complex FE Model Looking North.. 3.7-102

Figure 3.7.2-7 PCCV Dynamic Model 3.7-103

Figure 3.7.2-8 CIS Dynamic Model – Shell Elements (Excluding RCL)............ 3.7-104

Figure 3.7.2-9 CIS Dynamic Model - Solid Elements....................................... 3.7-105

Figure 3.7.2-10 CIS Dynamic Model – Beam Elements (Excluding RCL) 3.7-106

Figure 3.7.2-11 East PS/B Dynamic Model with ESWPC 3.7-107

Figure 3.7.2-12 West PS/B Dynamic Model with ESWPC 3.7-108

Figure 3.7.2-13 A/B Dynamic Model 3.7-109

Figure 3.7.2-14 Deleted 3.7-110

Figure 3.7.2-15 Extracted Detailed FE Model of Floor Slabs............................. 3.7-111

Figure 3.7.2-16 Floor Slab Model Boundary Conditions 3.7-111

Figure 3.7.2-17 SSSI Model with Backfill and Free Field Soils (Looking East)... 3.7-112

Figure 3.8.1-1 Configuration of PCCV (Sheet 1 of 3) 3.8-156

Figure 3.8.1-2 PCCV Schematic Reinforcing and Tendons............................. 3.8-159

Figure 3.8.1-3 Cylinder Liner Anchorage System 3.8-160

Figure 3.8.1-4 Liner Anchorage System for the Upper Portion of Dome 3.8-161

Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing........... 3.8-162

Figure 3.8.1-6 Equipment Hatch 3.8-177

Figure 3.8.1-7 Personnel Airlock... 3.8-178

Figure 3.8.1-8 Containment Penetrations 3.8-179

Figure 3.8.1-9	Identification of Areas for Thermal Conditions in Table 3.8.1-3	3.8-196
Figure 3.8.1-10	Transient Conditions of PCCV Atmosphere Temperature (Accident Condition)	3.8-200
Figure 3.8.1-11	Transient Conditions of Temperature of General Sump Pool Water in the PCCV and RWSP	3.8-201
Figure 3.8.1-12	Transient Conditions of Temperature of the SG Compartment Atmosphere and Sump Pool Water (Pipe Break in the SG Compartment)	3.8-202
Figure 3.8.1-13	Transient Conditions of Temperature of the Reactor Cavity Atmosphere and Sump Pool Water (Pipe Break in the Reactor Cavity)	3.8-203
Figure 3.8.3-1	RV Support System	3.8-204
Figure 3.8.3-2	SG Support System	3.8-205
Figure 3.8.3-3	RCP Support System	3.8-209
Figure 3.8.3-4	Pressurizer Support System	3.8-211
Figure 3.8.3-5	SC Module Isometrics	3.8-212
Figure 3.8.3-6	Interior Compartments Wall Layout and Configuration	3.8-220
Figure 3.8.3-7	Typical Details of SC Modules	3.8-227
Figure 3.8.3-8	Polar Crane Supports	3.8-232
Figure 3.8.3-9	CIS Pressure Loads	3.8-235
Figure 3.8.3-10	CIS FE Model	3.8-238
Figure 3.8.3-11	Critical Sections of SC Modules	3.8-239
Figure 3.8.3-12	Structural Categories Between Elevations 3’-7” and 21’-0”	3.8-240
Figure 3.8.3-13	Structural Categories Between Elevations 21’-0” and 35’-11”	3.8-241
Figure 3.8.3-14	Structural Categories Between Elevations 37’-9” and 62’-4”	3.8-242
Figure 3.8.3-15	Structural Categories Between Elevations 62’-4” and 76’-5”	3.8-243
Figure 3.8.3-16	Structural Categories Between Elevations 76’-5” and 139’-6”	3.8-244
Figure 3.8.3-17	Structural Categories, Section A-A (Looking West)	3.8-245
Figure 3.8.3-18	Structural Categories, Section B-B (Looking North)	3.8-246
Figure 3.8.4-1	Identification of Areas for PS/B Thermal Conditions in Table 3.8.4-2	3.8-247
Figure 3.8.4-2	FE Model of R/B	3.8-250

Figure 3.8.4-3	R/B Critical Sections	3.8-252
Figure 3.8.4-4	Typical Reinforcement in R/B West Common Wall – SECTION 1	3.8-253
Figure 3.8.4-5	Typical Reinforcement in R/B South Interior Wall – SECTION 2	3.8-255
Figure 3.8.4-6	Typical Reinforcement in North Exterior Wall of Spent Fuel Pit – SECTION 3	3.8-257
Figure 3.8.4-7	Typical Reinforcement in South Exterior Wall – SECTION 4	3.8-259
Figure 3.8.4-8	Typical Reinforcement in R/B Central Interior Wall – SECTION 5	3.8-261
Figure 3.8.4-9	Typical Reinforcement in Spent Fuel Pit Slab – AREA 3	3.8-263
Figure 3.8.4-10	Typical Reinforcement in Emergency Feedwater Pit Slab – AREA 4	3.8-264
Figure 3.8.4-11	FE Model of West PS/B	3.8-265
Figure 3.8.4-12	West PS/B Wall Critical Sections (Floor Plan of B1F, EL-26’-4”)	3.8-269
Figure 3.8.4-13	West PS/B Wall and Slab Critical Sections (Floor Plan of 1F, EL 3’-7”)	3.8-270
Figure 3.8.4-14	Typical Reinforcement in West PS/B South Exterior Wall – SECTION 1 (On Column Line LR and Between Column Lines 1R & 3R)	3.8-271
Figure 3.8.4-15	Typical Reinforcement in West PS/B Interior Wall – SECTION 2 (On Column Line 7R and Between Column Lines JR & LR)	3.8-272
Figure 3.8.4-16	Typical Reinforcement in West PS/B Floor at Elevation 3′-7″ – AREA 1 (Between Column Lines KR & LR - 3R & 5R)	3.8-273
Figure 3.8.4-17	Typical Reinforcement in West PS/B North Wall – SECTION 3 (On Column Line JR and Between Column Lines 1R & 3R)	3.8-274
Figure 3.8.4-18	East PS/B Wall Critical Sections (Floor Plan of B1F, El -26’-4”)	3.8-275
Figure 3.8.4-19	East PS/B Wall and Slab Critical Sections (Floor Plan of 1F, EL 3’-7”)	3.8-276
Figure 3.8.4-20	Typical Reinforcement in East PS/B East Exterior Wall – SECTION 1 (On Column Line 20R and Between Column Lines F1R & G4R)	3.8-277
Figure 3.8.4-21	Typical Reinforcement in East PS/B Interior Wall – SECTION 2 (On Column Line K1R and Between Column Lines 18R & 20R)	3.8-278

CONTENTS

		Page
Figure 3.8.4-22	Typical Reinforcement in East PS/B Floor at Elevation 3′-7″– AREA 1 (Between Column Lines 19R & 20R – G4R & JR)	3.8-279
Figure 3.8.4-23	ESWPC Critical Sections (Floor Plan of East ESWPC, EL -15’-8”)	3.8-280
Figure 3.8.4-24	Typical Reinforcement in ESWPC Exterior Wall– SECTION 1 (On Column Line LR3 and Between Column Lines 20R & 19R8)	3.8-281
Figure 3.8.4-25	Typical Reinforcement in ESWPC Interior Wall – SECTION 2 (On Column Line 19R3 and Between Column Lines LR & LR3)	3.8-282
Figure 3.8.4-26	Typical Reinforcement in Slab at Elevation -15′-8″– AREA 1 (Between Column Lines LR & LR3 – 13R & 18R11)	3.8-283
Figure 3.8.4-27	Dynamic & Static Lateral Earth Pressure vs. Passive Earth Pressure	3.8-284
Figure 3.8.4-28	Flow Chart for Seismic Force Transfer from SSI Model to Structural Model	3.8-285
Figure 3.8.5-1	Floor Plan of 1F (El. -3’-7”)	3.8-286
Figure 3.8.5-2	Cross Section of a North-South Orientation	3.8-287
Figure 3.8.5-3	Cross Section of an East-West Orientation	3.8-288
Figure 3.8.5-4	Deleted	3.8-289
Figure 3.8.5-5	R/B, PCCV, and Containment Internal Structure Basemat	3.8-290
Figure 3.8.5-6	Global Three-Dimensional FE Model of R/B Complex Basemat at Tendon Gallery	3.8-291
Figure 3.8.5-7	Global Three-Dimensional FE Model of R/B Complex (N-S Section)	3.8-292
Figure 3.8.5-8	Global Three-Dimensional FE Model of R/B Complex (E-W Section)	3.8-293
Figure 3.8.5-9	Global Three-Dimensional FE Model of R/B Complex (West Side)	3.8-294
Figure 3.8.5-10	Global Three-Dimensional FE Model of R/B Complex Basemat (Solid Elements)	3.8-295
Figure 3.8.5-11	Reinforcing Steel of Basemat: SECTION N-S	3.8-296
Figure 3.8.5-12	Reinforcing Steel of Basemat: SECTION E-W	3.8-297

CONTENTS

		Page
Figure 3.8.5-13	Typical Reinforcement for R/B Complex around Periphery of Basemats	3.8-298
Figure 3.8.5-14	Timeline of Loading for Settlement Analyses	3.8-299
Figure 3.8.5-15	Secant Moduli for Settlement Analyses	3.8-299
Figure 3.9-1	Typical Mathematical Model of the Reactor Internals for the Seismic and LOCA Dynamic Analysis (RV, Lower Reactor Internals and CRDM)	3.9-256
Figure 3.9-2	Typical Mathematical Model for the Seismic and LOCA Dynamic Analysis (Fuel Assembly, Upper Reactor Internals and Diffuser Plate)	3.9-257
Figure 3.9-3	CRDM	3.9-258
Figure 3.9-4	Reactor Internals General Arrangement	3.9-259
Figure 3.9-5	Upper Reactor Internals Assembly	3.9-260
Figure 3.9-6	Lower Reactor Internals Assembly	3.9-261
Figure 3.9-7	Jurisdictional Boundary between Reactor Internals and RV	3.9-262
Figure 3.9-8	Reactor Internals RCS Flow and Bypass Flow Paths	3.9-263
Figure 3.9-9	Neutron Reflectors	3.9-264
Figure 3.9-10	Fasteners around the Core	3.9-265
Figure 3.11-1	Project EQ Program Implementation Framework	3.11-17

ACRONYMS AND ABBREVIATIONS

3-D	three-dimensional
A/B	auxiliary building
ac	alternating current
AC/B	access building
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ARS	acceleration response spectra
ASCE	American Society of Civil Engineers
ASD	allowable stress design
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
AWS	American Welding Society
BAC	bounding analysis curves
BE	best estimate
BTP	Branch Technical Position
CAV	cumulative absolute velocity
CCP	centrifugal charging pumps
CCW	component cooling water
CCWS	component cooling water system
CFR	Code of Federal Regulations
COL	Combined License
CQC	complete quadratic combination
CRDM	control rod drive mechanism
CRDS	control rod drive system
CS	containment spray
CSDRS	certified seismic design response spectra
CSS	containment spray system
C_v	charpy V-notch
CVCS	chemical and volume control system
CWS	circulating water system
DBA	design basis accident
DBFL	design basis flooding level

ACRONYMS AND ABBREVIATIONS (CONTINUED)

dc	direct current
DCD	Design Control Document
DIF	dynamic increase factor
DLF	dynamic load factor
DOF	degrees of freedom
DBPB	design basis pipe break
ECCS	emergency core cooling system
EFW	emergency feedwater
EFWS	emergency feedwater system
EPRI	Electrical Power Research Institute
EPS	emergency power source
EQ	environmental qualification
EQSDS	equipment qualification summary data sheet
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESQR	equipment seismic qualification report
ESWPT	essential service water pipe tunnel
ESWS	essential service water system
FE	finite element
FIRS	foundation input response spectra
FSS	fire protection water supply system
FW	feedwater
FWS	feedwater system
GA	general arrangement
GDC	General Design Criteria
GMRS	ground motion response spectra
GTG	gas turbine generator
HELB	high-energy line break
HHIS	high-head injection system
HIS	hydrogen ignition system
HMS	hydrogen monitoring system
HRC	rockwell c hardness
HSLA	high strength low alloy
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronic Engineers
ILRT	integrated leak rate test

ACRONYMS AND ABBREVIATIONS (CONTINUED)

ISI	inservice inspection
ISM	independent support motion
ISRS	in-structure response spectra
IST	inservice testing
ITAAC	inspections, tests, analyses, and acceptance criteria
ITP	initial test program
LB	lower bound
LBB	leak-before-break
LOCA	loss-of-coolant accident
LOF	left-out-force
MCR	main control room
MELB	moderate-energy line break
MFIV	main feedwater isolation valve
MOV	motor operated valve
MS	main steam
MSIV	main steam isolation valve
MSLB	main steam line break
MSS	main steam supply system
MT	magnetic particle examination method
MTC	moderator temperature coefficient
NCIG	National Construction Issues Group
NDE	nondestructive examination
NDRC	National Defense Research Council
NIST	National Institute of Standards and Technology
NPS	nominal pipe size
NRCA	non-radiological controlled area
NRC	U.S. Nuclear Regulatory Commission
OBE	operating-basis earthquake
OD	outside diameter
P&ID	piping and instrumentation diagram
PC	plant condition
PCCV	prestressed concrete containment vessel
PGA	peak ground acceleration
PIV	pressure isolation valve
PMF	probable maximum flood
PMP	probable maximum precipitation
PORV	power operated relief valve
POV	power operated valve

ACRONYMS AND ABBREVIATIONS (CONTINUED)

PS/B	power source building
PSFSV	power source fuel storage vault
PSMS	protection and safety monitoring system
PT	liquid penetrant examination method
PTFE	polytetra fluoroethylene
PWR	pressurized water reactor
QA	quality assurance
QAP	quality assurance program
R/B	reactor building
RCA	radiological controlled area
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RHRS	residual heat removal system
RIM	required input motion
RPS	reactor protection system
RRS	required response spectra
RT	reactor trip
RV	reactor vessel
RWMS	radioactive waste management systems
RWSAT	refueling water storage auxiliary tank
RWSP	refueling water storage pit
SAM	seismic anchor motion
SC	steel concrete
SCC	stress corrosion cracking
SECY	Secretary of the Commission Letter
SEI	Structural Engineering Institute
SFP	spent fuel pit
SFPCS	spent fuel pit cooling and purification system
SG	steam generator
SI	safety injection
SIP	safety injection pump
SIS	safety injection signal
SLS	safety logic system

ACRONYMS AND ABBREVIATIONS (CONTINUED)

SNL	Sandia National Laboratories
SRM	staff requirements memorandum
SRP	Standard Review Plan
SRSS	square root sum of the squares
SS	stainless steel
SSC	structure, system, and component
SSE	safe-shutdown earthquake
SSEA	safe-shutdown earthquake anchor
SSEI	safe-shutdown earthquake inertia
SSI	soil-structure interaction
T/B	turbine building
T/G	turbine generator
TRS	test response spectrum
UB	upper bound
UHS	ultimate heat sink
UHSRS	ultimate heat sink related structures
U.S.	United States
USM	uniform support motion
UT	ultrasonic examination method
UTS	ultimate tensile strength
V ac	Volts alternating current
VCT	volume control tank
V dc	Volts direct current
ZPA	zero period accelerations

3.0 DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS, AND EQUIPMENT

3.1 Conformance with NRC General Design Criteria

This section briefly discusses the extent to which the design criteria for the US-APWR safety-related structures, systems, and components (SSCs) comply with Title 10, Code of Federal Regulations (CFR), Part 50, Appendix A, General Design Criteria for Nuclear Power Plants (Reference 3.1-1). As presented in this section, each criterion is first quoted and then discussed in sufficient detail to demonstrate the compliance of the US-APWR with each criterion. For some criteria, additional information may be required for a complete discussion. In such cases, detailed evaluations of compliance with the various General Design Criteria (GDC) are incorporated in other sections (identified by reference).

3.1.1 Overall Requirements

3.1.1.1 Criterion 1 – Quality Standards and Records

> Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program 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 structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

3.1.1.1.1 Discussion

The quality assurance program description (QAPD) in Section 17.5, describes the methods for assuring quality and establishes quality assurance and administrative control requirements for SSCs important to safety per GDC 1. The Equipment Classification process will assign a Quality Assurance Classification that defines the pertinent quality requirements from full 10 CFR Appendix B, to quality controls for nonsafety-related SSCs that perform important to safety functions, to no special requirements for other nonsafety-related SSCs. SSCs are assigned to the appropriate Equipment Classifications based on their important to safety functions and, depending on these safety functions, are also classified as either safety-related or nonsafety-related. Quality assurance requirements are established and identified by the appropriate Quality Assurance Classification for safety-related and nonsafety-related SSCs based on their safety functions and the contribution to plant safety.

As described in the QAPD, 10 CFR 50, Appendix B requirements are applied to activities affecting the quality and performance of safety-related SSCs. Furthermore, the QAPD specifies selected quality controls that are applied to certain equipment and activities that are nonsafety-related, but support safe and reliable plant operations, or where other NRC guidance establishes applicable quality assurance requirements. The controls applied to nonsafety-related SSCs per the QAPD Part III are referred to as “augmented” quality assurance controls. The contribution of nonsafety-related SSCs to plant safety is determined by (1) the SSC's risk-significance as determined by the D-RAP described in Section 17.4, and (2) the reliance on the SSC to address regulatory events, which are ATWS, fire protection, and Station Blackout (SBO). The list of risk significant SSCs is provided in Table 17.4-1.

The QAP along with the quality assurance, quality engineering, and quality control programs of the major contractors and their vendors, assure that SSCs important to safety (both safety-related SSCs and certain selected nonsafety-related SSCs) are designed, procured, fabricated, inspected, erected, and tested to standards commensurate with the contribution to plant safety. This is accomplished by using recognized quality codes, standards, and design criteria that comply with the requirements of 10 CFR 50.55a (Reference 3.1-2) for safety-related SSCs, and by selecting appropriate design control measures for nonsafety-related SSCs requiring augmented quality as described in the QAPD. As necessary, additional supplemental standards, design criteria, and requirements have been developed by the Mitsubishi Heavy Industries, Ltd. and the major contractors’ engineering organizations. Appropriate records that are associated with engineering and design, procurement, fabrication, inspection, erection, and testing, document compliance with recognized codes, standards, and design criteria. These records are maintained throughout the life of the plant.

The principal design criteria, design bases, codes, and standards applicable to the facility are described in Section 3.2. Additional detail may be found in the pertinent section of the document dealing with the safety-related SSC(s) (e.g., containment as described in Subsection 3.8.1).

3.1.1.2 Criterion 2 – Design Bases for Protection Against Natural Phenomena

> Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of the capability to perform their safety functions. The design bases for these 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 which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) The importance of the safety functions to be performed.

3.1.1.2.1 Discussion

The safety-related SSCs are designed either to withstand the effects of natural phenomena without the loss of the capability to perform their safety functions, or are designed such that their response or failure will be in a safe condition. Additionally, nonsafety-related SSCs that must maintain their structural integrity to prevent unacceptable structural interaction or failure of seismic category I SSCs are designated as seismic category II and appropriately designed as described in Subsection 3.2.1. The nature and magnitude of the natural phenomena considered in the design of the plant are discussed in Chapter 2. The non-safety-related SSCs in the radwaste system are designed to be consistent with RG 1.143.

Appropriate combinations of structural loadings from normal operation, accident conditions, and natural phenomena are considered in the plant design. This chapter discusses the design of the plant relative to natural events. Seismic, quality group classifications, and other pertinent standards and information, are provided in the sections discussing individual SSCs and in Section 3.2.

3.1.1.3 Criterion 3 – Fire Protection

> Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

3.1.1.3.1 Discussion

The safety-related SSCs are designed to minimize the probability and effect of fires and explosions. Noncombustible and fire-resistant materials are used in the containment, main control room (MCR), component features of safety systems, and throughout the plant, where fire is a potential risk to safety-related systems. For example, electrical cables have a fire-retardant jacketing, and fire barriers are utilized as described in Subsection 9.5.1. Fire barriers ensure that redundant, safety-related systems and components are separated to assure that a fire in one area will not affect the redundant systems and components in an adjacent area from performing their safety functions. Equipment and facilities for fire protection, including detection, alarm, and extinguishment, are provided to protect both the plant equipment and personnel from fire, explosion, and the resultant release of toxic vapors.

Fire protection is provided by deluge systems (water spray), sprinklers, and portable extinguishers.

Firefighting systems are designed to assure that their rupture or inadvertent operation will not prevent safety-related systems from performing their design functions.

The design of the fire protection system and equipment is performed in accordance with the guidance provided in Standard Review Plan (SRP) 9.5.1 of NUREG-0800 (Reference 3.1-3) and the criteria provided in Regulatory Guide (RG) 1.189, Revision 1, Fire Protection for Nuclear Power Plants (Reference 3.1-4).

See Subsection 9.5.1, for additional information.

3.1.1.4 Criterion 4 – Environmental and Dynamic Effects Design Bases

> Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). 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.

3.1.1.4.1 Discussion

The safety-related SSCs are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the normal operation, maintenance, testing, and postulated accidents, including LOCAs. Criteria are presented in Chapter 3, and the environmental conditions are described in Section 3.11.

These SSCs are adequately 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 plant. Details of the design, environmental testing, and construction/fabrication of safety-related SSCs are provided in Chapters 3, 5, 6, 7, 8, 9, and 10. The leak-before-break (LBB) evaluation of Section 3.6 identifies the design requirements for the piping that is excluded from consideration of pipe rupture due to dynamic effects from postulated pipe failure accidents.

3.1.1.5 Criterion 5 – Sharing of Structures, Systems, and Components

> Structures, systems, and components important to safety 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, an orderly shutdown and cooldown of the remaining unit.

3.1.1.5.1 Discussion

This Design Control Document (DCD) focuses on the US-APWR as a single plant. Safety-related SSCs are not shared with other units/plants including other US-APWR unit(s).

3.1.2 Protection by Multiple Fission Product Barriers

3.1.2.1 Criterion 10 – Reactor Design

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

3.1.2.1.1 Discussion

The reactor core and associated coolant, control, and protection systems are designed to the following criteria (anticipated operational occurrences [AOOs] for events occurring one or more times in a plant lifetime, and postulated accidents/occurrences that are not expected to occur:

- No fuel damage will occur during normal core operation or operational transients or any transient conditions that may occur one or more times in a plant lifetime (AOOs). Fuel damage, as used here, is defined as penetration of the fission product barrier (i.e., the fuel rod cladding). The small number of clad defects that may occur is within the capability of the plant cleanup system and is consistent with the plant design bases (see Section 11.1). The reactor can be returned to a safe-shutdown state following an anticipated operational occurrence with only a small fraction of the fuel rods damaged, although there might be sufficient fuel damage to preclude the immediate resumption of operation.

- The core will remain intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (postulated accidents).

Reactor protection system setpoints are chosen conservatively to support the design margins. The reactor trip (RT) system is designed to actuate a RT whenever necessary to assure that the fuel design limits are not exceeded. The core design, together with the process and decay heat removal systems, provide this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow, a trip of the turbine generator, a loss of normal feedwater, and a loss of both normal and preferred power sources.

Chapter 4 discusses the design bases and design evaluation of core components including nuclear, thermal, and hydraulic design and evaluation. Details of the control and protection systems' instrumentation design and logic are discussed in Chapter 7. This information supports the accident analyses of Chapter 15, which show that the acceptable fuel design limits are not exceeded for AOOs, and adequate core cooling is available for postulated accidents/occurrences.

3.1.2.2 Criterion 11 – Reactor Inherent Protection

> The reactor core and associated coolant systems 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.

3.1.2.2.1 Discussion

The reactor core is designed to have negative reactivity feedback characteristics associated with fuel and moderator temperature. When the reactor is critical, prompt compensatory reactivity feedback effects are assured by the negative fuel temperature effect (doppler effect), which compensates for a rapid uncontrolled reactivity increase. The negative doppler coefficient is assured by the inherent design, using low enrichment Uranium fuel. For slower reactivity transients, the negative moderator temperature coefficient (MTC) provides compensatory reactivity feedback to help control such transients. In order to have a negative MTC at power-operating condition, the boron concentration in the primary system is limited using burnable absorbers. The overall core design establishes a negative MTC.

Reactivity coefficients and their effects are discussed in Chapter 4.

3.1.2.3 Criterion 12 – Suppression of Reactor Power Oscillations

> The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

3.1.2.3.1 Discussion

Total reactor power oscillations of the fundamental mode are inherently eliminated by negative Doppler and negative MTC.

Power distribution oscillations due to xenon spatial effects in the radial and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative doppler and negative MTC.

Power distribution oscillations, due to xenon spatial effects may occur in the axial first overtone mode. Assurance that fuel design limits are not exceeded by axial xenon induced power oscillations is provided by RT functions, using the measured axial flux difference as an input.

If necessary to maintain axial flux difference within the limits of the Technical Specifications/Chapter 16 (i.e., flux difference that are alarmed to the operator and are within the flux difference trip setpoints), the operator can suppress axial xenon oscillations by control rod motions, and/or temporary power reductions.

Power distribution oscillations, due to xenon spatial effects in axial modes higher than the first overtone are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

The stability of the core against xenon-induced power distribution oscillations and the functional requirements of instrumentation for monitoring are discussed in Chapter 4. Details of the instrumentation design and logic are discussed in Chapter 7.

3.1.2.4 Criterion 13 – Instrumentation and Control

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

3.1.2.4.1 Discussion

Instrumentation and controls are provided to monitor and control neutron flux, control rod position, fluid temperatures, pressures, flows, and levels, to assure that adequate plant safety can be maintained. Instrumentation is provided in the reactor coolant system (RCS), steam and power conversion system, containment, engineered safety feature (ESF) systems, radioactive waste management systems (RWMS), and other auxiliary systems. Parameters that must be provided for operator use under normal operating and accident conditions are provided in proximity to the controls so that the operator can maintain the indicated parameters within their proper ranges. Chapter 18 describes the criteria for human factors engineering for the layout, displays, and controls of such parameters.

The quantity and types of process instrumentation provided assure the safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapters 6, 7, 8, 9, 10, 11, and 12.

3.1.2.5 Criterion 14 – Reactor Coolant Pressure Boundary

> The reactor coolant pressure 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.

3.1.2.5.1 Discussion

The reactor coolant pressure boundary (RCPB) is designed, fabricated, erected, and tested in accordance with 10 CFR 50.55a (Reference 3.1-2) to provide a high degree of integrity throughout the plant life. Systems and components within the RCPB are classified as Quality Group A (Section 3.2 of this chapter). The design requirements, codes, and standards applied to this quality group help assure high integrity in keeping with the safety-related function.

The RCPB is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, with stresses within applicable limits. Consideration is given to loadings under normal operating conditions and to abnormal loadings, such as seismic loadings, as discussed in this chapter. The piping is protected from overpressure by pressure-relieving devices, as required by the American Society of Mechanical Engineers (ASME) Code, Section III (Reference 3.1-5) (see Section 5.2).

RCPB materials and fabrication techniques are such that there is a low probability of gross rupture or significant leakage. The RCS design incorporates the LBB approach, which demonstrates that the probability of a fluid system pipe rupture is extremely low under conditions consistent with the design basis for the piping, and, therefore, reduces or eliminates the need to consider the dynamic effects of pipe breaks (see Section 3.6).

The reactor vessel (RV) material, construction techniques, chemical composition of forging material, and design, limit neutron fluence as discussed in Section 5.3, and Chapter 4. The “Discussion” for Criterion 31 (Subsection 3.1.4.2.1) provides additional information on the RCPB.

Coolant chemistry is controlled to protect the RCPB’s materials of fabrication from corrosion (see Section 5.2).

The RCPB welds are accessible for inservice inspections (ISI) to assess the structural and leak-tight integrity (see Section 5.2). For the RV, a material surveillance program conforming to applicable codes is provided (see Chapter 5, Section 5.3).

Instrumentation is provided to detect significant leakage from the RCPB with indication in the MCR (see Section 5.2).

3.1.2.6 Criterion 15 – Reactor Coolant System Design

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

3.1.2.6.1 Discussion

Steady state and transient analyses are performed to assure that RCS design conditions are not exceeded during normal operation. Protection and control setpoints are based on these analyses (see Chapter 15).

Additionally, RCPB components have a large margin of safety based on the application of proven materials and design codes, the use of proven fabrication techniques, the non-destructive shop testing, and the integrated hydrostatic testing of assembled components. The RCS stress analysis including the LBB analysis are described in Sections 3.9 and 3.6 of this document.

The effect of radiation embrittlement is considered in the RV design; as surveillance samples are monitored for adherence to expected conditions throughout plant life.

Multiple spring-loaded safety relief valves are provided for the RCS. The safety relief valves and their setpoints meet the ASME criteria for over-pressure protection. Use of the ASME criteria is satisfactory, based on a long history of industrial use. Chapter 5 discusses the RCS design.

3.1.2.7 Criterion 16 – Containment Design

> The reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

3.1.2.7.1 Discussion

The prestressed concrete containment vessel (PCCV) is composed of a pre-stressed post-tensioned concrete vessel, featuring a vertical cylinder, a hemispherical dome, and a flat reinforced concrete foundation. The PCCV is surrounded by the concrete reactor building (R/B). The steel-lined PCCV completely encloses the reactor, RCS, and other related systems. The lines that penetrate the containment vessel are provided with containment isolation valves according the provisions of GDCs 54, 55, 56, and 57. The steel-lined PCCV provides an essentially leak-tight barrier and provides environmental radiation protection under all postulated accident conditions, including a LOCA. The PCCV is designed to sustain, without loss of required integrity, the effects of LOCAs up to and including the double-ended rupture of the largest pipe in the RCS or double-ended rupture of a steam or feedwater pipe. For such events, ESFs comprising the emergency core cooling system (ECCS), containment systems, and containment spray system (CSS) cool the reactor core and return the containment to near atmospheric pressure.

The containment structure and ESFs are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure and to assure the required functional capability of containing any uncontrolled release of radioactivity (including the surrounding annulus and emergency exhaust system).

See Chapters 3, 6, 9, and 15 for additional information.

3.1.2.8 Criterion 17 – Electrical Power Systems

> An onsite electric power system and an offsite electric power system shall be provided to permit the functioning of structures, systems, and components important to safety. The safety function for each system (assuming that the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions, assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights-of-way) designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time, following the loss of all onsite alternating current (ac) power supplies and the offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

3.1.2.8.1 Discussion

An onsite electric power system and an offsite electric power system are provided to permit the functioning of safety-related SSCs. As discussed in Chapter 8, each Class 1E (safety) electric power system is designed with adequate independence, capacity, redundancy, and testability to assure the functioning of the corresponding ESFs. Independence is provided by physical separation and electrical isolation of components and cables.

The onsite electric power system is supplied from two offsite transmission systems, which are physically separated. All plant loads are supplied normal power through the main transformer and four unit auxiliary transformers, or with alternate power through the reserve auxiliary transformers. The onsite electric power system consists of Class 1E (safety) electrical systems and non-safety electrical systems. The Class 1E ac power system is the power source used in (or associated with) shutting down the reactor and preventing or limiting the release of radioactive material following a design basis event. The Class 1E electrical distribution system is divided into four independent distribution power systems/trains: Train A, Train B, Train C, and Train D. Each train is fed from an independent Class 1E bus. Each train consists of: 6.9 kV, 480 V, 120 V ac, and 125 V dc distribution systems.

Each Class 1E bus is provided with two (normal and alternate) offsite preferred power sources, an emergency power source (EPS), and a diverse non-safety alternative ac power source, which are manually connected in the event of station blackout. This direct connection is performed under manual administrative controls.

The Class 1E ac power system distributes power to all safety-related loads. Also, certain selected loads that are not safety-related but are important to the plant operation are supplied with power from the alternative ac power source.

The reserve auxiliary transformer supplies normal preferred (offsite) power to the Class 1E ac system. Each reserve auxiliary transformer has the capacity to supply all connected running loads. The unit auxiliary transformers may also supply alternate preferred (offsite) power to all connected running loads. The capacity of the unit auxiliary transformer is the same as that of the reserve auxiliary transformer.

A failure of a single component will not prevent the safety-related systems from performing their function. Each of the connected preferred offsite power circuits is designed to be available in sufficient time, following a loss of all onsite power sources and the other offsite electric power circuit, to assure that the specified acceptable fuel design limits and design conditions of the RCPB are not exceeded.

Emergency onsite ac power is furnished by four EPSs. Each EPS is connected to a Class 1E bus. The ESF loads are divided between the Class 1E, 6.9 kV buses in redundant load groupings. Each EPS is capable of supplying sufficient power in sufficient time for the operation of the ESF required for the plant during a concurrent design basis accident (DBA) and a loss of offsite power. During a postulated LOCA, four EPSs start automatically. If the preferred power is available to the Class 1E bus following a LOCA, the ESF loads will be started sequentially. However, in the event that the preferred power is lost, all motor loads connected to a Class 1E bus will be shed, and the ESF equipment will be sequentially started.

The US-APWR design basis allows on-line maintenance of each of the four EPSs. The four train safety system loads are connected with each train bus. There are also some two train safety systems loads. These two train safety system loads are connected to buses that can be powered from either of two power sources. During maintenance of an EPS, the two train loads are manually switched to the alternate train feeder.

The EPSs are arranged so that a failure of a single component will not prevent the safe-shutdown of the reactor. The onsite Class 1E dc power supply consists of four independent battery systems. The failure of a single component in the dc power supply will not impair the function of the ESFs required to maintain the reactor in a safe condition.

See Chapter 8 for details and Chapter 15 for additional information.

3.1.2.9 Criterion 18 – Inspection and Testing of Electric Power Systems

> Electric power systems important to safety 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 the nuclear power unit, the offsite power system, and the onsite power system.

3.1.2.9.1 Discussion

Class 1E electric power systems are designed so that the following aspects of the system can be periodically tested:

- The operability and functional performance of the components of Class 1E electric power systems (EPSs, ESF buses, and dc system).
- The operability of these electric power systems as a whole and under conditions as close to design as practical, including the full operational sequence that actuates these systems.

The 6.9 kv and 480 V circuit breakers and the associated equipment will be tested one at a time, only while redundant equipment is operational.

The dc system is provided with detectors to indicate and alarm when there is a ground existing on any part of the system. During plant operation, normal maintenance may be performed.

Provisions for the testing of the Class 1E ac power system, the Class 1E dc power system, and the standby safety power supplies are described in Chapter 8 and in the Technical Specifications/Chapter 16.

3.1.2.10 Criterion 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, including LOCAs. 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 whole body, or its equivalent, to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

3.1.2.10.1 Discussion

A MCR is provided for the control of the US-APWR plant, from which actions can be taken to operate the plant safely under normal conditions and to maintain it in a safe manner under accident conditions, including LOCAs. Operator action outside of the MCR to mitigate the consequences of an accident is permitted. The MCR and its post-accident ventilation systems are designed to satisfy seismic category I requirements. Adequate concrete shielding and radiation protection are provided against direct gamma radiation and inhalation doses resulting from a postulated release of fission products inside the PCCV based on the assumptions contained in RG 1.183 (Reference 3.1-6). The shielding and the MCR habitability system allow access to, and occupancy of, the MCR under accident conditions without personnel receiving radiation exposures in excess of 5 rem (total effective dose equivalent) for the duration of the accident (see Chapter 6, Section 6.4, and Chapter 15). Fission product removal is provided in the MCR recirculation equipment to remove iodine and particulate matter, thereby minimizing the dose due to iodine that could result from the accident. The MCR habitability features are described in Chapter 6, Section 6.4.

In the event that the operators are forced to abandon the MCR, the remote shutdown console located outside the control room fire zone, provides safe-shutdown capability (to achieve and maintain the plant in a safe-shutdown condition or cold shutdown) through the four trains, protection and safety monitoring system (PSMS), and the safety-related human-system interface system.

See Chapter 7 for additional information.

3.1.3 Protection and Reactivity Control Systems

3.1.3.1 Criterion 20 – Protection System Functions

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

3.1.3.1.1 Discussion

The PSMS includes the reactor protection system (RPS), the engineered safety features actuation system (ESFAS), and the safety logic system (SLS). The RPS receives process signals from safety-related sensors and performs bi-stable calculations for potential RT and ESF actuation. The RPS performs two-out-of-four voting logic for like sensors coincidence to actuate trip signals to RT switchgears and actuate ESF signals to the ESFAS. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to assure that the fuel design limits are not exceeded.

RT is initiated by removing power to the control rod drive mechanisms (CRDMs) of all rod cluster control assemblies (RCCAs). This causes the rods to be inserted by gravity, thus, rapidly reducing the reactor power. The response and adequacy of the protection system has been verified by analysis of anticipated transients.

The ESFAS receives output of the ESF actuation signals from the RPS and manual system level actuation signals. The occurrence of a limiting fault (e.g., LOCA) requires a

RT plus actuation of one or more ESFs in order to prevent or mitigate damage to the core and RCS, and to assure containment integrity.

Upon approaching reactor design-set conditions, RPS is initiated automatically such that specified acceptable fuel design limits are not encroached as a result of AOOs. Upon sensing conditions of AOOs or postulated accidents, ESFAS automatically initiates operation of systems and components required to mitigate anticipated operations or postulated accident conditions.

The SLS also receives manual component level control signals. This system performs the component level control logic for safety actuators.

See Chapters 7 and 15 for additional information.

3.1.3.2 Criterion 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 the loss of the protection function and (2) removal from service of any component or channel does not result in the 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.

3.1.3.2.1 Discussion

The protection system is designed for functional reliability and inservice testability. The design employs redundant trains, measurement, and equipment diversity.

The PSMSs have the requisite redundancy to satisfy the single failure criterion during normal operation and during all planned-on-line test/maintenance configurations. Physical separation and electrical isolation are provided between redundant subsystems. All equipment within the PSMS satisfies all general requirements such as, environmental, seismic, testability, etc. The configuration of the four trains with two-out-of-four voting logic is provided from sensors to trip breakers in the RPS. The configuration of four trains with two-out-of-four voting logic is also provided in the ESFAS. In addition to train redundancy, the central processing unit and the safety bus are also redundant within each train.

To prevent a disturbance of the plant caused by the failure of the instrumentation and control system, a redundant configuration is applied to all subsystems that may directly result in spurious plant trips or a spurious system level ESF actuation. Failed components detected by the self-diagnostic features automatically switch the system to redundant stand-by components. Consequences due to transients generated either by spurious plant trip and/or by spurious ESF actuations are bounded by Chapter 15 analysis.

The safety system may be placed in bypass mode to allow for testing and maintenance while the plant is on-line. During this bypass mode, a single failure in the safety system will not result in a spurious plant trip or a system level ESF actuation. Automatic bypass management logic continuously checks for multiple bypassed conditions to assure that the minimum redundancy required by the Technical Specifications/Chapter 16 is always maintained.

This equipment includes automated testing with a high degree of coverage, and additional overlapping manual test features for those areas not covered by automated tests. Most manual tests may be conducted with the plant on-line, and with the equipment bypassed or out of service. Equipment that cannot be tested with the plant on-line can be tested during shutdown.

The instrumentation and control systems are described in Chapter 7. System operability and surveillance requirements are discussed in the Technical Specifications/Chapter 16.

3.1.3.3 Criterion 22 – Protection System Independence

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

3.1.3.3.1 Discussion

The design of the protection systems includes the consideration of natural phenomena, normal maintenance, testing, and accident conditions so that the protection functions are always available.

The protection system components are designed, arranged, and qualified for operation in the environmental conditions arising from any emergency situation for which the components are required to function.

Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a variety of postulated accidents. Diverse protection functions will automatically terminate an accident before unacceptable consequences can occur.

Sufficient redundancy and independence are designed into the protection systems to assure that no single failure or removal from service of any component or channel of a system would result in a loss of the protection function. Functional diversity and consequential location diversity are designed into the system. Automatic RTs are based upon neutron flux measurements, reactor coolant loop temperature measurements, pressurizer pressure and level measurements, reactor coolant pump flow and speed, steam generator (SG) water level, and turbine trip signal. Trips may also be initiated manually or by an ECCS actuation signal. See Chapter 7 for additional details.

High quality components, conservative design, applicable quality control, inspection, calibration, and tests are utilized to guard against common-mode failure. Qualification testing and analysis are performed on the various safety-related systems to demonstrate functional operation at normal and post-accident conditions of temperature, humidity, pressure, and radiation for specified periods, if required. Typical protection system equipment is subjected to type tests under simulated seismic conditions, using conservatively large accelerations at the applicable frequencies. The test results indicate no loss of the protection function.

Refer to the discussions in Sections 3.10 and 3.11 for further details.

3.1.3.4 Criterion 23 – Protection System Failure Modes

> The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air) or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

3.1.3.4.1 Discussion

The protection system is designed with consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. The RT functions are designed to fail to an actuated trip state (de-energize-to-trip principle) on the loss of all power, failures that are not automatically detected, or failures that are automatically detected and would prevent the proper execution of the trip function. The ESF functions are designed to fail to an un-actuated state. The un-actuated state avoids spurious plant transients and is, therefore, a safe state.

For a more detailed description of the protection system, see Chapter 7.

3.1.3.5 Criterion 24 – Separation of Protection and Control Systems

> The protection system shall be separated from the 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.

3.1.3.5.1 Discussion

Redundant divisions of the protection systems are physically and electrically isolated from the non-safety control systems. Isolation devices provide assurance that, where protection signals are used by non-safety systems, and non-safety signals are used by safety systems, credible single failures in the non-safety system will not degrade the performance of the safety system. In addition to the electrical and physical isolations, functional isolation between non-safety systems and safety systems is provided. The functional isolation is provided by priority logics in the safety systems, or by signal selector logic in the non-safety systems. The priority logics assures that safety actuation signals, both automatic and manual (system level and component level), override all control signals from the non-safety systems. Signal selection logic in the control system prevents erroneous control actions due to single sensor failures. Eliminating these erroneous control actions prevents challenges to the protection system while it is degraded due to the same sensor failure.

The adequacy of the system isolation capability has been verified by testing under conditions of postulated credible faults. The failure of any single control system component or channel, or the failure or removal from service of any single protection system component or channel, which is common to the control and protection system, leaves intact a system that satisfies the requirements of the protection system.

3.1.3.6 Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions

> The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of the control rods.

3.1.3.6.1 Discussion

The protection system is designed to limit reactivity transients so that the fuel design limits are not exceeded. Reactor shutdown by control rod insertion is completely independent of the normal control function, since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Thus, in the postulated accidental withdrawal of a control rod or control rod bank (assumed to be initiated by a control malfunction), neutron flux, temperature, pressure, level, and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

Analyses of the effects of possible malfunctions are discussed in Section 15.4. These analyses show that for postulated boron dilution during refueling, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of the dilution, terminate the source of the dilution, and initiate re-boration before the shutdown margin is lost. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

3.1.3.7 Criterion 26 – Reactivity Control System Redundancy and Capability

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity

control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that the acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

3.1.3.7.1 Discussion

Two reactivity control systems are provided. These are the RCCAs and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the control rod banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and AOOs, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in the core life is assumed in all analyses, and the most reactive RCCA is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state and can compensate for xenon burnout transients.

Details of the fabrication of the RCCAs are presented in Chapter 4, and their control system is discussed in Chapter 7. The means of controlling the boric acid concentration is described in Chapter 9. Performance analyses under accident conditions are included in Chapter 15.

3.1.3.8 Criterion 27 – Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

3.1.3.8.1 Discussion

The facility is provided with the means of making and holding the core subcritical under any anticipated condition and with appropriate margin for contingencies. These means are discussed in detail in Chapters 4, 7, and 9. The transient analysis in Chapter 15 show that the reactivity changes are controlled during AOOs and postulated accidents. The combined use of the rod cluster control system and the chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth RCCA is assumed to be stuck full out upon trip for this determination.

3.1.3.9 Criterion 28 – Reactivity Limits

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

3.1.3.9.1 Discussion

The maximum reactivity worth of the control rods and the maximum rates of reactivity insertion employing control rods and boron removal are limited by design and procedures. These values prevent any reactivity increase from rupturing the RCS boundary or disrupting the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The appropriate reactivity insertion rate for the withdrawal of RCCAs and the dilution of the boric acid in the RCS is limited by design and procedures. The RPS and ESFAS provide protection for such events as a rod ejection accident and steam line break. Protection system setpoints, control bank alignment, insertion limits, and shutdown margin requirements are contained in the Technical Specifications/Chapter 16. The reactivity insertion rates, dilution, and withdrawal limits are discussed in Chapter 4. The capability of the chemical and volume control system (CVCS) to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 9. The relationship of the reactivity insertion rates to plant safety is discussed in Section 15.4.

Core cooling capability following accidents, such as rod ejection, steam line break, etc., is assured by keeping the RCPB within faulted condition limits, as specified by applicable ASME codes. Structural deformations are also checked and limited to values that do not jeopardize the operation of needed safety-related features.

3.1.3.10 Criterion 29 – Protection against Anticipated Operational Occurrences

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

3.1.3.10.1 Discussion

The protection and reactivity control systems have an extremely high probability of performing their required safety-related functions in any AOOs. Diversity, independence, and redundancy, coupled with a QAP and analyses, support this probability. Failure modes of the system components are designed to be safe modes. A loss of power to the protection system results in a RT. The details of system design are covered in Chapters 4 and 7.

3.1.4 Fluid Systems

3.1.4.1 Criterion 30 – Quality of Reactor Coolant Pressure Boundary

> Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

3.1.4.1.1 Discussion

The RCPB components of the US-APWR are consistent with 10 CFR 50.2 and 10 CFR 50.55a (Reference 3.1-2).

Section 3.2 provides the classification of the RCPB components and attached systems and components, which are accorded all the quality measures appropriate to these classifications. The design bases and evaluations of the RCS are discussed in Chapter 5.

A number of methods are available for detecting reactor coolant leakage. The RV closure joint is provided with a temperature-monitored leak-off between double gaskets. Leakage inside the PCCV is drained to the sumps, where the level is monitored. Leakage is also detected by measuring the air-borne activity and gaseous radioactivity of the PCCV and condensate flow rate from the air coolers. Refer to Chapter 5 for a complete description of the RCPB leakage detection system. Technical Specifications/Chapter 16 limit pressure boundary, identified and unidentified leakage, and requires RCS leakage detection instrumentation to be operable in certain modes.

3.1.4.2 Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure 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 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 on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.

3.1.4.2.1 Discussion

The RCPB is designed, maintained, and tested to provide adequate assurance that the boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized throughout the life of the plant. Close control is maintained over material selection and fabrication for the RCPB to assure that the boundary behaves in a nonbrittle manner. The RCPB materials, which are exposed to the coolant, are corrosion-resistant stainless steel or nickel-based alloys. The nil-ductility transition reference temperature (RT_{NDT}) of the RV structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR 50, Appendix G (Reference 3.1-7), “Fracture Toughness Requirements.”

The following requirements are imposed on the RV, in addition to those specified by the ASME Code.

- In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens, and post irradiation testing of Charpy V-notch, tensile, and 1/2 T compact tension specimens. These programs are directed toward the evaluation of the effect of radiation on the fracture toughness of RV steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with American Society of Testing and Materials (ASTM) E-185-82 (Reference 3.1-8) Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, and the requirements of 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements (Reference 3.1-9).

- RV core region material chemistry is controlled to reduce sensitivity to embrittlement due to irradiation over the life of the plant.

The fabrication and quality control techniques used in the fabrication of the RCS are equivalent to those used for the RV. The inspections of RV, pressurizer, piping, pumps, and SGs are governed by ASME Code requirements.

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated, using methods derived from “Protection Against Non Ductile Failure” , ASME Code, Section III, Appendix G, (Reference 3.1-10). The approach specifies that the allowable stress intensity factors for all vessel-operating conditions do not exceed the reference stress intensity factor for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material reference temperatures (RT_{NDT}) due to irradiation.

Refer to Chapter 5 for additional details.

3.1.4.3 Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

> Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel.

3.1.4.3.1 Discussion

The design of the RCPB provides accessibility to the internal surfaces of the RV and most external zones of the vessel, including the nozzle to the reactor coolant piping welds, the top and bottom heads, and external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete and within the reactor coolant piping support area. The inspection capability complements the Leakage Detection System in assessing the integrity of the pressure boundary components. The RCPB will be periodically inspected under the provisions of the ASME Code, Section XI (Reference 3.1-11).

RV material surveillance samples are located within the RV. These samples are used to monitor changes in the fracture toughness properties of the RV core region, forgings, weldments, and associated heat-treated zones, which is performed in accordance with 10 CFR 50, Appendix H (Reference 3.1-9).

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in RT_{NDT} of the core region materials with irradiation will be used to confirm the allowable limits calculated for operational transients.

The design of the RCPB piping provides for accessibility of all welds requiring ISI under the provisions of the ASME Code, Section XI (Reference 3.1-11). Removable insulation is provided at all welds requiring ISI.

Refer to Section 3.9 and Chapters 5 for additional information.

3.1.4.4 Criterion 33 – Reactor Coolant Makeup

> A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components, which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

3.1.4.4.1 Discussion

The CVCS provides the normal means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the Volume Control Tank falls below a preset level. Centrifugal charging pumps (CCPs) are used as the normal means of reactor coolant makeup. The pumps are powered from the non-safety bus.

The CCPs are capable of supplying the required makeup and reactor coolant seal injection flow when power is available from the non-safety bus. Functional reliability is assured by provision of standby components assuring a safe response to probable modes of failure.

The emergency letdown system consists of two emergency letdown lines from the RCS hot legs to the refueling water storage pit (RWSP). In the event that the normal CVCS letdown and boration capability is not available, the feed and bleed emergency letdown and boration operation can be utilized to achieve a cold shutdown boration level in the reactor coolant. The emergency letdown directs reactor coolant to the RWSP. The safety injection pumps (SIPs) provide borated coolant to the RCS from the RWSP. The SIPs are powered from a safety 1E bus so they can be supplied power from either the Offsite or Onsite Electric Power Systems.

Details of the system design, including the descriptions of the effects of small piping and component ruptures, are provided in Chapters 6, 9, and 15, and details of the electric power system are included in Chapter 8.

3.1.4.5 Criterion 34 – Residual Heat Removal

> A system to remove residual heat shall be provided. 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 fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. II

3.1.4.5.1 Discussion

The containment spray/residual heat removal system (CS/RHRS), in conjunction with the steam and power conversion system, and the ECCS, are designed to transfer the fission product decay heat and other residual heat from the reactor core at a rate that keeps the fuel within acceptable limits for normal operations and accident conditions. The CS/RHRS functions when temperature and pressure are below approximately 350°F and 400 psig, respectively.

Redundancy of the CS/RHRS is provided by four independent subsystems with each subsystem containing a CS/residual heat removal (RHR) pump (located in separate compartments, with means available for draining and monitoring leakage), and a CS/RHR heat exchanger, and associated piping, cabling, and electric power sources. The CS/RHR pumps receive power from corresponding safety buses. The CS/RHRS is able to operate from either the onsite or offsite electrical power sources.

Redundancy of heat removal at temperatures above approximately 350°F is provided by the four SGs, the corresponding main steam relief valves, and the Emergency Feedwater System.

The ECCS, which includes the accumulators and high-head injection system (HHIS), cools the reactor core, provides negative reactivity, prevents fuel and cladding damage, and limits the zirconium-water reaction of the fuel cladding to a very small amount. The ECCS is designed with sufficient redundancy (four trains) to accomplish the safety-related functions. The SIPs are located in separate compartments, with means available for draining and monitoring leakage. The essential components of the ECCS receive power from corresponding safety buses and therefore are able to operate from either the onsite or offsite electrical power sources.

Details of the system design and related accident analysis are provided in Chapters 5, 6, 10, and 15. Details of the electric power systems are included in Chapter 8.

3.1.4.6 Criterion 35 – Emergency Core Cooling

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming off site power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

3.1.4.6.1 Discussion

The ECCS of the US-APWR includes the accumulator system, HHIS and emergency letdown system. The ECCS has the capability to mitigate the effects of any LOCA within the design bases. Cooling water is provided in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to assure that clad metal-water reaction is limited to less than 1%. Design provisions assure performance of the required safety functions even with a postulated single failure.

Emergency core cooling is provided even if there is a failure of any component in the system. A passive system of four accumulators, one for each RCS loop, do not require any external signals or sources of power to operate, and provide the short-term cooling requirements for breaks in the large reactor coolant pipe systems. An independent and redundant pumping system is provided by the HHIS. The HHIS consists of four independent trains, each train contains a SIP and the associated valves, and piping. One of four independent safety electrical buses is available to each SIP. The SIPs are aligned to take suction from the RWSP that is inside containment to deliver borated water to the safety injection (SI) nozzles on the RV for short-term cooling and to the hot legs and downcomer for long-term cooling. Two SI trains are capable of meeting the design cooling function for a large LOCA assuming single failure in one train with another train out of service for maintenance.

The Discussion section of GDC 33 describes the emergency letdown system, the system's flow path, and boration capability via the SIPs, which are supplied power from the safety bus from either the offsite or onsite sources, as needed.

These systems are arranged so that a single failure of any active component does not interfere with meeting the short- term cooling requirements.

Additionally, the ECCS is designed with sufficient redundancy (four trains) to accomplish its safety functions assuming a single failure of an active component or a passive component in the long term following an accident with one train out of service for maintenance.

The primary function of the ECCS is to deliver borated cooling water to the reactor core in the event of a LOCA. This limits the fuel-clad temperature; assures that the core will remain intact and in place, with its essential heat transfer geometry preserved; and prevents a return to criticality. This protection is provided for the following events:

- All pipe breaks sizes up to and including the hypothetical circumferential rupture of the largest pipe of a reactor coolant loop
- A LOCA associated with a rod ejection

The ECCS is described in Chapter 6. The LOCA, including an evaluation of consequences, is discussed in Chapter 15. Details of the electric power system are described in Chapter 8.

3.1.4.7 Criterion 36 – Inspection of Emergency Core Cooling System

> The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

3.1.4.7.1 Discussion

The ECCS is accessible for visual inspection and for nondestructive ISI, to satisfy the ASME Code, Section XI (Reference 3.1-11).

The piping, components, accumulators and the RWSP are designed to permit access for periodic inspection and testing to confirm the integrity and capability of the system.

Components outside the containment are accessible for leak-tightness inspection during operation of the reactor.

Details of the inspection program for the ECCS are discussed in Section 6.3, the Inservice Inspection Program, and the Technical Specifications/Chapter 16.

3.1.4.8 Criterion 37 – Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, 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 applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

3.1.4.8.1 Discussion

The design of the ECCS permits periodic testing of both active and passive components of the ECCS.

Preoperational performance tests of the ECCS components are performed by the manufacturer. Initial system hydrostatic and functional flow tests demonstrate structural and leak-tight integrity of components and proper functioning of the system. Thereafter, periodic tests demonstrate that components are functioning properly.

Each active component of the ECCS may be individually operated on the normal power source or transferred to standby power sources at any time during normal plant operation to demonstrate operability. The test of the SIPs, employs a minimum flow line that connects back to the RWSP that may be used during normal operation. Full flow test may be performed during plant shutdown. The pump rooms contain leakage detection, which triggers an alarms in the MCR. Remote-operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves, and pump breakers may be checked during integrated system tests performed during a planned cooldown of the RCS.

Design provisions include special instrumentation, testing, and sampling lines used to perform the tests during plant shutdown to demonstrate proper automatic operation of the ECCS. (Refer to Chapter 1, Section 1.9, for a discussion of RG 1.22 [Reference 3.1-12].) A test signal is applied to initiate automatic action, and verification is performed to confirm that the SIPs attain the required discharge head. This test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. In addition, the periodic recirculation to the RWSP can verify the ECCS delivery capability. This recirculation test includes all but the last valve, which connects to the reactor coolant piping.

The design provides the capability to initially test, to the extent practical, the full operational sequence, including transfer to alternate power sources for the ECCS to demonstrate the state of readiness and capability of the system. This functional test is performed with the pressurizer pressure setpoint below the SI signal setpoint and with the RCS initially cold and depressurized. The ECCS valving is set to initially simulate the system alignment for plant power operation.

Details of the ECCS are found in Chapter 6. Performance under accident conditions is evaluated in Chapter 15. Surveillance requirements are identified in the Technical Specifications/Chapter 16.

3.1.4.9 Criterion 38 – Containment Heat Removal

> A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.
>
> Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

3.1.4.9.1 Discussion

The CSS consists of four independent trains, each containing a CS/RHR heat exchanger, a CS/RHR pump, spray nozzles, piping and valves. The CSS maintains the containment vessel internal peak pressure below the design pressure and reduces it to approximately atmospheric pressure over time in the event of a LOCA or a main steam line break. The CS/RHR pumps suction intake is from the in-containment RWSP, which is located at the lower elevation inside containment, and provides a continuous source for the CS/RHR pumps.

The CSSs consist of four independent subsystems supplied from separate Class 1E power buses. No single failure, including loss of onsite or offsite electrical power, can cause loss of more than a quarter of the installed 200% cooling capacity. The CSS has sufficient redundancy to perform its required safety functions following an accident assuming a single failure in one train with a second train out of service for maintenance.

The CSS is discussed in Chapter 6. Electrical power systems are described in Chapter 8. A containment pressure and temperature analysis following a LOCA and steam line break is given in Chapter 6, with additional results found in Chapter 15.

3.1.4.10 Criterion 39 – Inspection of Containment Heat Removal System

> The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles and piping, to assure the integrity and capability of the system.

3.1.4.10.1 Discussion

The design of the CSS allows to the extent practical, inspection of the components to provide confirm the integrity and capability of the system. A portion of the essential equipment of the CSS is outside the containment, except for risers, distribution header piping, spray nozzles, and the RWSP. The RWSP, spray piping, and nozzles can be inspected during shutdown. Portions of the CS/RHR suction intake piping from the RWSP are embedded in concrete and are not accessible for inspection. The integrity of any inaccessible portions of piping is verified by testing as described in Subsection 3.1.4.11.1. Associated equipment outside the containment can be visually inspected.

Details of the inspection and surveillance program for the CSS are discussed in Chapter 6, the Inservice Inspection Program, as specified by the ASME Code, and the Technical Specifications/Chapter 16.

3.1.4.11 Criterion 40 – Testing of Containment Heat Removal System

> The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

3.1.4.11.1 Discussion

The CSS is designed to permit periodic testing to assure the structural and leak-tight integrity of CSS components and to assure the operability and performance of the active components of the system. All active components of the CSS, including the delivery piping up to the last powered valve before the spray nozzle, have the capability to be tested during reactor power operation. A minimum flow path return to the pump suction line is used for pump tests during normal operation. Full flow pump performance testing is conducted during plant shutdown conditions. The pump rooms contain leakage detection, which alarms in the MCR. In addition, when the plant is shut down, smoke or air can be blown through the test connections for visual verification of the flow path.

The facility design allows, under conditions as close to the design as practicable, the performance of a full operational sequence that brings this system into operation.

Detailed discussions of the testing of these systems are provided in Chapter 6, and the Technical Specifications/Chapter 16.

3.1.4.12 Criterion 41 – Containment Atmosphere Cleanup

> Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided, as necessary, to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the

containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming off site power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

3.1.4.12.1 Discussion

US-APWR fission product control systems following a DBA includes the: CSS, and Annulus Emergency Exhaust System (AEES). The CSS is a dual function ESF System that provides heat removal and fission product removal following a LOCA. The fission product in particulate form is mechanically removed by the CSS and the fission product in gaseous form (radio-iodine is the primary concern) is controlled by adjustment of the RWSP pH by use of chemicals. The CSS consists of four independent subsystems, each supplied power from separate Class 1E buses. The CSS has sufficient redundancy to perform its required safety functions following an accident assuming a single failure in one train with a second train out of service for maintenance. The AEES is designed for fission product removal by ventilation and air filtration following a DBA. The AEES prevents uncontrolled release to the environment from containment, exhaust air. The AEES is powered from the Class 1E buses so specified safety-related functions are maintained during a loss of offsite power. The system is designed to perform the safety-related functions with a single active component failure.

Hydrogen monitoring and control is provided for the unlikely occurrence of an accident that is more severe than a postulated DBA. The generation of hydrogen in the containment under these post-accident conditions has been evaluated. The hydrogen monitoring system (HMS) provides a hydrogen detector to detect hydrogen concentration in containment air extracted from containment, and provides continuous indication in the MCR. The hydrogen igniters of the hydrogen ignition system (HIS) reduce concentration of hydrogen in containment in such an environment. The HMS and HIS are supplied by the non-Class 1E P1 and P2 power system, with alternate power capability.

These systems are discussed in Chapter 6. Electrical power systems are described in Chapter 8. A containment pressure and temperature analysis following a LOCA is given in Chapter 6, with additional results found in Chapter 15. The generation of hydrogen under such post–accident conditions is described in Chapter 19.

3.1.4.13 Criterion 42 – Inspection of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

3.1.4.13.1 Discussion

The containment atmosphere cleanup, and hydrogen monitoring and control systems are designed and located so that they can be inspected periodically, as required. Portions of the essential equipment of the CSS are located outside the containment, except for risers, distribution header piping, spray nozzles, and the RWSP, which are located inside the PCCV. The AEES is located in the R/B. The hydrogen monitor is located outside containment. The hydrogen igniters are located inside the PCCV. The equipment located outside the PCCV may be inspected during normal power operation. Components of the CSS, and HIS that are located inside the PCCV, can be inspected during shutdowns. See Chapter 6 for details on these systems.

3.1.4.14 Criterion 43 – Testing of Containment Atmospheric Cleanup Systems

> The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, 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 sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

3.1.4.14.1 Discussion

The containment atmosphere cleanup, and hydrogen monitoring and control systems include the CSS, AEES, HMS and HIS. The discussion of GDC 40 demonstrates the testability, and therefore, the operability and performance of the CSS. During normal operation, two trains are placed in operation, and as such provide continuous testability. Similarly, the HMS provides MCR readout and as such provides continuous testability. The hydrogen igniters are tested by energizing the circuit during refueling outages. The AEES is periodically tested in accordance with the testing/surveillance requirements of the Technical Specifications/Chapter 16.

Chapter 6 provides system discussions, and Chapter 8 provides electrical power details. Technical Specifications/Chapter 16 provide testing/surveillance requirements.

3.1.4.15 Criterion 44 – Cooling Water

> A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be 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 assure that for onsite electric power system operation

(assuming offsite power is not available) and for off site electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

3.1.4.15.1 Discussion

The component cooling water system (CCWS) and the essential service water system (ESWS) are provided to transfer heat from plant safety-related components to the UHS. These systems are designed to transfer their respective heat loads under all anticipated normal and accident conditions. Suitable redundancy, leak detection, systems interconnection, and isolation capabilities are incorporated in the design of these systems to assure the required safety function, assuming a single failure, with either onsite or offsite power. The active components are powered from the Class 1E buses. The systems are designed to perform safety-related functions assuming a single failure in one train with another train out of service for maintenance.

Complete descriptions of the ESWS and the CCWS are provided in Chapter 9. Chapter 8 describes the electrical power system.

3.1.4.16 Criterion 45 – Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

3.1.4.16.1 Discussion

The CCWS and portions of the ESWS are capable of being monitored during normal operation. The important components of these systems are located in accessible areas. These components will have suitable inspection capability as noted in Section 9.2. The COL Applicant is to provide a design that allows for the appropriate inspections and layout features of the ESWS. The integrity of any underground piping will be demonstrated by pressure and functional tests.

These systems are discussed in Chapter 9, with operability and surveillance requirements provided in the Technical Specifications/Chapter 16.

3.1.4.17 Criterion 46 – Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and the performance of the active components of the system, 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 for reactor shutdown and for LOCA, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

3.1.4.17.1 Discussion

The CCWS and the ESWS operate continuously during normal plant operation and shutdown, under flow and pressure conditions that approximate accident conditions. These operations demonstrate the operability, performance, and structural and leak-tight integrity of all cooling water system components.

These cooling water systems are designed to include the capability for testing through the full operational sequence that brings the system into operation for reactor shutdown and for LOCAs, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources. The CCWS and the ESWS are capable of being tested during normal operation by alternating operation of the systems between the redundant trains.

For detailed descriptions of the: cooling water systems, refer to Chapter 9, and for a description of the electrical power distribution systems, refer to Chapter 8. The operability and surveillance requirements are provided in the Technical Specifications/Chapter 16.

3.1.5 Reactor Containment

3.1.5.1 Criterion 50 – Containment Design Basis

> The reactor containment structure, including access opening, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by 10 CFR 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

3.1.5.1.1 Discussion

The design of the containment structure is based on the full spectrum of containment DBAs, which include the rupture of a reactor coolant pipe in the RCS or the rupture of a main steam or feedwater line. In either case, the pipe rupture is assumed to be coupled with partial loss of the redundant safety feature systems (single failure of an active or passive component [loss of one train], and the loss of another train due to maintenance resulting in the application of minimum safety features resulting in the availability of the two remaining trains). The most limiting pressure and temperature responses are assessed to verify the adequacy of the containment structure. The maximum pressure and temperature reached for a containment DBA are presented in Chapter 6. The containment design, as discussed in Subsection 3.8.1, provides margin to the design basis limits.

See Chapters 3 and 6 for details.

3.1.5.2 Criterion 51 – Fracture Prevention of Containment Pressure Boundary

> The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

3.1.5.2.1 Discussion

The PCCV is a reinforced concrete structure with ferritic parts such as steel liner and its penetrations. Principal load-carrying components of ferritic materials exposed to the external environment are selected (as discussed in Subsection 3.8.1) so that their temperatures under normal operating and testing conditions are sufficiently above the nil ductility transition temperatures. This assures that under normal operating, maintenance, testing, and post accident conditions, ferritic materials behave in a non-brittle manner considering the associated uncertainties of material properties, stresses and size of flaws. This minimizes the probability of a rapidly propagating fracture. The preoperational testing, operational testing, surveillance program, and the QAP assure the integrity of the containment and its ability to meet all normal operational and accident requirements.

Refer to Section 3.8, Section 6.2, and Chapter 17 for details. Surveillance program requirements are discussed in the Technical Specifications/Chapter 16.

3.1.5.3 Criterion 52 – Capability for Containment Leakage Rate Testing

> The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

3.1.5.3.1 Discussion

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leak-rate tests (ILRT) during plant lifetime, in accordance with the requirements of 10 CFR 50, Appendix J (Reference 3.1-13). Details concerning the conduct of periodic ILRT are included in Chapter 6.

3.1.5.4 Criterion 53 – Provisions for Containment Testing and Inspection

> The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows.

3.1.5.4.1 Discussion

Provisions exist for conducting individual leak-rate tests on containment penetrations. Penetrations are visually inspected and pressure tested for leak-tightness at periodic intervals. Other inspections are performed, as required by 10 CFR 50 Appendix J (Reference 3.1-13) (see Chapter 6).

3.1.5.5 Criterion 54 – Piping Systems Penetrating Containment

> Piping systems penetrating the primary reactor containment shall be provided with leak detection, isolation and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

3.1.5.5.1 Discussion

The US-APWR containment isolation design satisfies the SRP 6.2.4 (Reference 3.1-14) requirements. Two barriers are provided; one inside containment and one outside containment. Usually these barriers are valves, but in some configurations, they are closed piping systems not connected to the RCS or to the containment atmosphere. Other acceptable methods on another defined basis to meet these requirements are discussed in Subsection 6.2.4, for the CS/RHRS, and the HHIS.

Piping systems penetrating the primary reactor containment are provided with containment isolation valves. Penetrations, which must be closed for containment isolation, have redundant valving and associated apparatus. Automatic isolation valves with air, or motor operators, which do not restrict normal plant operation, are periodically tested to assure operability. For those penetrations that are normally open and are required to close, these penetrations use remote operated valves for isolation that close automatically on a containment isolation signal. The containment isolation signal is generated and actuated on by the PSMS.

Secondary system piping inside the containment is considered an extension of the containment boundary, as described in Subsection 6.2.4. The isolation valve arrangements are discussed in Chapter 6.

Piping that penetrates the containment has been equipped with test connections and test vents or has other provisions to allow periodic leak-rate testing to assure that leakage is within the acceptable limit as defined by the technical specifications/Chapter 16 consistent with 10 CFR 50, Appendix J (Reference 3.1-13), as described in Chapter 6.

The fuel transfer tube is not classified as a fluid system penetration. The blind flange and the portion of the transfer tube inside the containment are an extension of the containment boundary. The blind flange isolates the transfer tube at all times, except when the reactor is shutdown for refueling. This assembly is a penetration in the same sense as are equipment hatches and personnel locks.

3.1.5.6 Criterion 55 – Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates the primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside the containment; or
2. One automatic isolation valve inside and one locked closed isolation valve outside the containment; or
3. One locked closed isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
4. One automatic isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside the containment.

Isolation valves outside the containment shall be located as close to the containment as practical and, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided, as necessary, to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication and testing additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density and use characteristics and physical characteristics of the site environs.

3.1.5.6.1 Discussion

Lines that are a part of the RCPB and penetrate the containment are provided with isolation valves meeting the intent of this criterion. Several penetrations use alternative arrangements, which satisfy containment isolation on some other defined bases. Special cases are described in Subsection 6.2.4. A simple check valve may not be used as the automatic isolation valve outside the containment.

3.1.5.7 Criterion 56 – Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates the primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside the containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside the containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside the containment.

Isolation valves outside the containment shall be located as close to the containment as practical and, upon loss of actuating power automatic isolation valves shall be designed to take the position that provides greater safety.

3.1.5.7.1 Discussion

Lines which communicate directly with the containment atmosphere and which penetrate the reactor containment are normally provided with two isolation valves in series, one inside and one outside the containment, in accordance with one of the above acceptable arrangements. Several penetrations use alternative arrangements, which satisfy containment isolation on some other defined bases. Special cases are described in Subsection 6.2.4.

Valving arrangements are combinations of locked-shut isolation valves and automatic isolation valves or remote-manual isolation valves. No simple check valves are utilized as automatic isolation valves outside the containment. Where necessary, provision for leak detection is provided for lines outside the containment.

3.1.5.8 Criterion 57 – Closed System Isolation Valves

Each line that penetrates the primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, locked closed, or capable of remote manual operation. This valve shall be outside the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

3.1.5.8.1 Discussion

Lines that penetrate the containment and that are neither part of the RCPB nor connected directly to the containment atmosphere are considered closed systems within the containment. These lines are equipped with at least one containment isolation valve of one of the following types:

- An automatic isolation valve (a simple check valve is not used as this automatic valve)
- A locked-closed valve
- A valve capable of remote manual operation

Isolation valves are located outside the containment and as close to the containment wall as practical. Valve locations are discussed in Subsection 6.2.4.

Several penetrations use alternative arrangements, which satisfy containment isolation on some other defined bases. Special cases are described in Subsection 6.2.4.

3.1.6 Fuel and Reactivity Control

3.1.6.1 Criterion 60 – Control of Releases of Radioactive Materials to the Environment

> The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for the 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.

3.1.6.1.1 Discussion

The US-APWR is designed so that releases of radioactive materials in gaseous, liquid, and solid form are minimized. Means are provided to control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including AOOs. The RWMS are designed to minimize the potential for an inadvertent release of radioactivity from the facility and to assure that the discharge of radioactive wastes is maintained as low as practicable below the regulatory limits of 10 CFR 20 (Reference 3.1-15), and below the guidelines of 10 CFR 50, Appendix I (Reference 3.1-16), during normal operation. The gaseous and liquid RWMS have adequate capacity and redundancy to meet discharge concentration limits of 10 CFR 20 (Reference 3.1-15) during periods of design-basis fuel leakage. The RWMS, the design criteria, and the amounts of estimated releases of radioactive effluents to the environment are described in Chapter 11. The radiation monitoring of discharge paths of the gaseous and liquid radwaste processing systems and isolation on high radiation is also discussed in Chapter 11. The US-APWR is designed to minimize the release of radioactive materials in accordance with RG 4.21 (Reference 3.1-18). A discussion of the design objectives and operational programs to address these radiological aspects of the system is contained in Subsection 12.3.1. System and component design features addressing RG 4.21 are summarized in Table 12.3-8.

3.1.6.2 Criterion 61 – Fuel Storage and Handling and Radioactivity Control

> The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (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 importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

3.1.6.2.1 Discussion

The spent fuel pit cooling and purification system (SFPCS), fuel handling and radioactive waste systems are designed to cool and purify spent fuel pit (SFP) water, to supply borated water, provide shielding, and assure safety under normal and postulated accident conditions. The SFPCS is a two-train system that consists of a closed circuit that includes: heat exchangers, pumps, demineralizers, and filters. The subsystem is designed to run on Class 1E power during a loss of offsite power.

The SFPCS is designed to maintain the water level of the SFP, to prevent uncovering of the stored fuel from leakage due to failure of the piping, and to assure radiation shielding. Additionally, water may be added from several other sources, if required (Subsection 9.1.3). Adequate shielding is provided for the SFP and refueling cavity as described in Chapter 12. Radiation monitoring is provided as discussed in Chapters 11 and 12.

Normal heating ventilation and air conditioning (HVAC) system for the SFP area and purification and cooling system is provided by the auxiliary building (A/B) HVAC System. Normal HVAC system for the refueling cavity area is provided by the Containment Ventilation System. These HVAC Systems are described in Chapter 9.

The SFP cooling subsystem provides cooling to remove residual heat from the fuel stored in the SFP. The SFP cooling subsystem also provides cooling to the refueling cavity and fuel handling area water within containment during refueling operations when the SFP water is in communication with these areas. The SFPCS is designed with redundancy, testability, and inspection capability. SSCs are designed and located so that appropriate periodic inspection and testing may be performed.

The design of these systems meets the requirements of Criterion 61. The following sections further discuss fuel handling and storage systems inspection and testing, decay heat removal, purification, and prevention of reduction in coolant storage inventory:

Section	Title
5.4.8	Reactor Water Cleanup
9.1.1	Fuel Storage and Handling
9.1.2	New and Spent Fuel Storage
9.1.3	Spent Fuel Pit Cooling and Purification System
9.1.4	Light Load Handling System (Related to Refueling)
9.1.5	Overheard Heavy Load Handling System
9.4.2	Spent Fuel Pit Area Ventilation System
9.4.3	Auxiliary Building Ventilation System
9.4.6	Containment Ventilation System
11.2	Liquid Waste Management System
11.4	Solid Waste Management System
12.3	Radiation Protection

3.1.6.3 Criterion 62 – Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

3.1.6.3.1 Discussion

Fuel storage and handling systems are provided to preclude accidental criticality for new and spent fuel. The restraints, interlocks, and geometrically safe physical arrangement provided for the safe handling and storage of new and spent fuel with respect to critically prevention are discussed and illustrated in Chapter 9.

As stated in Subsection 9.1.1, the new fuel storage racks, the spent fuel storage racks and containment racks are designed to have sufficient separation between adjacent fuel assemblies so the maximum k_{eff} under worst- case conditions is less than 1.0 without credit for the soluble boron, and less than 0.95 with partial credit taken for soluble boron. As also stated in Subsection 9.1.1, the new fuel storage racks are designed to have sufficient separation between adjacent fuel assemblies such that the maximum k_{eff} is less than 0.95 when flooded with unborated water, and less than 0.98 under optimum moderation conditions. New fuel storage racks, spent fuel storage racks, and containment racks are seismic category I components.

The design of the spent fuel storage rack assembly is such that it is configurationally impossible to insert the spent fuel assemblies in other than prescribed locations, without physically modifying the rack, thereby preventing any possibility of accidental criticality.

Layout of the fuel handling area is such that a spent fuel cask cannot traverse the SFP.

See Chapter 9 for details.

3.1.6.4 Criterion 63 – Monitoring Fuel and Waste Storage

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

3.1.6.4.1 Discussion

Instrumentation is provided to give indication and annunciation in the MCR of excessive temperature or low water level in the SFP. An area radiation monitor is provided in the fuel storage area for personnel protection and general surveillance. This area monitor alarms locally and in the MCR. Normally, the A/B HVAC System removes radioactivity from the atmosphere above the SFP and discharges it by way of the plant vent. The ventilation system is continuously monitored by gaseous radiation monitors. If radiation levels reach a predetermined point, an alarm is actuated in the MCR.

See Chapters 7, 9, and 12 for details.

3.1.6.5 Criterion 64 – Monitoring Radioactivity Releases

> Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents

3.1.6.5.1 Discussion

The containment atmosphere is continuously monitored during normal and transient station operations, using the containment particulate, and gaseous radiation monitors. Under accident conditions, samples of the containment atmosphere provide data on existing airborne radioactive concentrations within the containment.

Radioactivity levels contained in the facility effluent and discharge paths and in the plant environs are continuously monitored during normal and accident conditions by the plant Radiation Monitoring Systems.

Portable radiation detection instruments are provided to periodically monitor radiation levels in the R/B spaces, which contain components for recirculation of LOCA fluids and in the A/B for components that process radioactive wastes. In addition to the installed detectors, periodic plant environmental surveillance is established. Measurement capability and reporting of effluents are based on the guidelines of RG 1.183 (Reference 3.1-7) and RG 1.21 (Reference 3.1-17).

Radiation Monitoring Systems are discussed in Chapter 11, Section 11.5, and Chapter 12, Section 12.3.

3.1.7 Combined License Information

COL 3.1(1) *The COL Applicant is to provide a design that allows for the appropriate inspections and layout features of the ESWS.*

3.1.8 References

3.1-1 Domestic Licensing of Production and Utilization Facilities, General Design Criteria for Nuclear Power Plants, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix A, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-2 Codes and Standards, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50.55a, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-3 Fire Protection Program, Auxiliary Systems, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plant. NUREG-0800, SRP 9.5.1, Rev. 5, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.1-4 Fire Protection for Nuclear Power Plants. Regulatory Guide 1.189, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.1-5 Rules for Construction of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, Section III, 1992 Edition through the 1992 Addenda. American Society of Mechanical Engineers.

3.1-6 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. Regulatory Guide 1.183, U.S. Nuclear Regulatory Commission, Washington, DC, July 2000.

3.1-7 Domestic Licensing of Production and Utilization Facilities, Fracture Toughness Requirements, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix G, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-8 Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, ASTM E-185-82, American Society of Testing Materials.

3.1-9 Domestic Licensing of Production and Utilization Facilities, Reactor Vessel Material Surveillance Program Requirements, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix H, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-10 Protection Against Non Ductile Failure, ASME Code, Section III, Nuclear Power Plant Components, Appendix G.

3.1-11 Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Code, Section XI, Edition 2001, through 2003 Addenda.

3.1-12 Periodic Testing of Protection System Actuation Functions. Regulatory Guide 1.22, U.S. Nuclear Regulatory Commission, Washington, DC, February 1972.

3.1-13 Domestic Licensing of Production and Utilization Facilities, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix J, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-14 Containment Isolation System, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plant. NUREG-0800, SRP 6.2.4, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.1-15 Standards for Protection Against Radiation, Energy. Title 10, Coded of Federal Regulations, Part 20, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-16 Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, “As Low as is Reasonably Achievable” for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix I, U.S. Nuclear Regulatory Commission, Washington, DC.

3.1-17 Measuring, Evaluating, and Reporting Radioactivity in Solid Waste and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water Cooled Nuclear Power Plants. Regulatory Guide 1.21, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, June 1974.

3.1-18 Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. Regulatory Guide 4.21, Rev. 0, U.S. Nuclear Regulatory Commission, Washington, DC, June 2008.

3.2 Classification of Structures, Systems, and Components

The US-APWR SSCs are classified according to nuclear safety classification, seismic category, quality groups, quality assurance classification, and codes and standards. The US-APWR SSCs are classified as safety-related as defined in 10 CFR 50.2 (Reference 3.2-1) or nonsafety-related. The safety-related SSCs are those relied upon to remain functional during and following design basis events to assure the following:

- The integrity of the RCPB
- The capability to shut down the reactor and maintain it in a safe-shutdown condition
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1) (Reference 3.2-2) or 10 CFR 100.11 (Reference 3.2-3)

Safety-related SSCs conform to the quality assurance requirements of 10 CFR 50, Appendix B. The quality assurance program is described in Chapter 17.

This section identifies those safety-related SSCs that are designed to withstand the effects of earthquakes without the loss of capability to perform their safety functions and those fluid systems or portions thereof that are safety-related, as well as the applicable industry codes and standards for pressure-retaining components.

This section also identifies those nonsafety-related SSCs that have augmented quality assurance requirements applied commensurate with the importance of the item's contribution to plant safety.

3.2.1 Seismic Classification

GDC 2 of 10 CFR 50, Appendix A, (Reference 3.2-4) requires in part that SSCs important to safety be designed to withstand the effects of earthquakes without the loss of the capability to perform their safety functions. To meet these requirements, the guidance provided in NRC RG 1.29 (Reference 3.2-5) is used for identifying and classifying those SSCs. The earthquake against which these SSCs are designed to withstand and remain functional is defined as the safe-shutdown earthquake (SSE) in 10 CFR 50, Appendix S (Reference 3.2-6). The SSE is based upon an evaluation of the maximum earthquake potential, and is that earthquake which produces the maximum vibratory ground motion for which safety-related SSCs are designed to remain functional (see Subsection 3.7.1.1 for a discussion of the application of the SSE). Appendix S of 10 CFR 50 (Reference 3.2-6) also requires consideration of surface deformation and seismically induced floods and water waves from either a local or distant generated seismic activity and other design conditions determined in 10 CFR 100.23 (Reference 3.2-7) to assure that certain SSCs will remain functional.

The site-independent seismic design of the US-APWR sets the operating-basis earthquake (OBE) ground motion at 1/3 of the SSE as discussed in Subsection 3.7.1.1, which eliminates the requirement for performing explicit design analysis for OBE loads. In accordance with Appendix S of 10 CFR 50 (Reference 3.2-6), SSCs necessary for continued safe operation must remain functional without undue risk to the health and safety of the public and within applicable stress, strain, and deformation, during and following an OBE. The OBE is associated with plant shutdown. Table 3.2-1 is a list of nonsafety-related components required for normal plant shutdown.

GDC 1 requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. 10 CFR 50, Appendix B (Reference 3.2-8), requires that safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The Quality Assurance (QA) program, which includes augmented quality assurance controls for nonsafety-related SSCs, implements the requirements of 10 CFR 50, Appendix B (Reference 3.2-8) as well as GDC 1, as discussed in Subsection 3.1.1.1.1. These QA program requirements are applied to activities affecting safety-related functions of SSCs designated as seismic category I, commensurate with their importance to safety.

RG 1.29 (Reference 3.2-5) is used to identify and classify those SSCs (including their foundations and supports) required for safe-shutdown, that must be designed to withstand the effects of the SSE and remain functional, as seismic category I. The recommendations in RG 1.29 (Reference 3.2-5) are used for systems, other than RWMS, that contain, or may contain, radioactive material and whose postulated failure would result in potential offsite whole body (or equivalent) doses that are more than 0.5 roentgen equivalent in man (rem), and are classified as seismic category I. Compliance with RG 1.29 (Reference 3.2-5) assures that a designed-in safety margin is provided for bringing the reactor to a safe-shutdown condition, while also reducing potential offsite doses from seismic events.

Some SSCs required for operation (excluding electrical features) do not need to be designed to seismic category I requirements. Examples of these SSCs include those portions of seismic category I systems such as vent lines, drain lines, fill lines and test lines on the downstream side of isolation valves and those portions of the system not required to perform a safety function.

The SSCs that are not designated as seismic category I and are not required to remain functional following an SSE, but whose failure could reduce the functioning of any seismic category I SSCs to an unacceptable safety level are designed and constructed to maintain their structural integrity under seismic loading from the SSE. These SSCs that must maintain their structural integrity to prevent unacceptable structural interaction or failure with seismic category I SSCs are designated as seismic category II.

Seismic category I applies to both the functionality and the integrity of the SSCs. Seismic category II applies only to the integrity of SSCs. Items that are subjected to an SSE, or items that create seismically-induced flooding, are designated as seismic category II to prevent the loss of the function of any safety-related items.

US-APWR SSCs are assigned to one of three seismic categories (seismic category I, seismic category II, or NS) depending on the nuclear safety function or the particular SSC.

RG 1.151 (Reference 3.2-9) is used as guidance for the seismic design and classification of safety-related instrumentation sensing lines. The seismic classification of safety-related instrumentation sensing lines is in accordance with RG 1.151 (Reference 3.2-9), Positions C.2 and C.3. The use of this guidance assures that the instrument sensing lines used to actuate or monitor safety-related systems are appropriately classified and are capable of withstanding the effects of the SSE.

GDC 61 requires that RWMS, and other systems that may contain radioactivity, be designed to assure adequate safety under normal and postulated accident conditions. Postulated conditions considered with respect to seismic design and classification of SSCs include the loss of SSC integrity and potential radioactive releases as a result of seismic events. RG 1.143 (Reference 3.2-10) is used as guidance relative to seismic design and classification for radioactive waste management SSCs. The use of the classification information and design criteria provided in the RG 1.143 (Reference 3.2-10) assures that components and structures used in RWMS are designed, constructed, installed, and tested in a manner that protects the health and safety of the public and plant operating personnel. Compliance with GDCs 2 and 61, as they relate to designing and constructing these SSCs to withstand earthquakes, and RG 1.143 (Reference 3.2-10), for seismic design and classification, provides assurance that SSCs containing radioactivity are properly classified and radiation exposures as a result of seismic events are as low as reasonably achievable.

RG 1.189 (Reference 3.2-11) is used as guidance to establish the design requirements of fire protection systems to meet the requirements of GDC 2 as it relates to designing these SSCs to withstand earthquakes. RG 1.189 is used to identify portions of fire protection SSCs requiring some level of seismic design consideration.

RG 1.189 (Reference 3.2-11), Positions 3.2.1, 6.1.1.2, and 7.1 are used to provide guidance for the proper seismic classification of fire protection systems. The use of this guidance assures that the fire protection systems for manual firefighting in areas containing safety-related equipment, containment penetrations, and reactor coolant pump (RCP) lube oil are properly classified and analyzed for SSE loads. Compliance with the above guidance assures that the safety-related SSCs required to function during an SSE are properly classified as seismic category I, and perform their safety functions.

3.2.1.1 Definitions

3.2.1.1.1 Seismic Category I

Seismic category I applies to safety-related SSCs (including their foundations and supports) that must remain functional and/or retain their pressure integrity in the event of an SSE.

This category includes SSCs designated as seismic category I in accordance with RG 1.29 (Reference 3.2-5). These SSCs are designed to withstand the effects of the SSE and maintain their structural integrity (including pressure integrity) and their specified design functions. The new and spent fuel pit structures, including fuel racks and refueling cavity structure including the containment racks, are designated seismic category I. Equipment Class 1, 2, or 3 components are designated seismic category I.

Additionally, in accordance with RG 1.29 (Reference 3.2-5), systems, other than RWMS, that contain, or may contain, radioactive material whose postulated failure would result in potential offsite whole body doses that are more than the recommended limits, are classified as seismic category I.

Seismic category I SSCs are designed to withstand the effects of natural phenomena, including earthquakes, without jeopardizing the plant nuclear safety as discussed in Sections 3.7 and 3.10. The interaction of non-seismic category I structures with seismic category I structures is discussed in Subsection 3.7.2.8.

Seismic category I SSCs meet the QA requirements of 10 CFR 50, Appendix B (Reference 3.2-8).

3.2.1.1.2 Seismic Category II

Seismic category II applies to SSCs which perform no safety-related function, and whose continued function is not required, but whose structural failure or interaction could degrade the functioning or integrity of a seismic category I SSC to an unacceptable level, or could result in incapacitating injury to occupants of the control room. Seismic category II SSCs are designed so that the SSE could not cause unacceptable structural interaction or failure with seismic category I SSCs. For fluid systems, this requires an adequate level of pressure boundary integrity to prevent seismically-induced flooding that may cause adverse effects on safety-related SSCs.

Seismic category II SSCs are analyzed and designed for the SSE using methods appropriate to demonstrate position retention with no adverse interaction effects as specified for seismic category I SSCs. Seismic category II may be limited to the parts of SSCs where structural analyses show a credible failure or interaction due to the SSE, which in some instances is limited to their supports.

Seismic category II SSCs meet the augmented quality assurance controls for nonsafety-related SSCs as described in the QAPD Part III to such an extent as to assure that their structural failure or interaction does not degrade the functional or structural integrity of seismic category I SSCs.

3.2.1.1.3 Non-Seismic

SSCs that are not classified as seismic category I or seismic category II are classified as NS. NS SSCs have no safety-related function or nuclear safety design requirements. The NS SSCs are primarily located outside of safety-related buildings or segregated from seismic category I SSCs so that the failure of their structural integrity would not impact the seismic category I SSCs and cause adverse system interactions. If it is determined that a SSC would cause an adverse impact on a seismic category I SSC, then it is designed and/or mounted in accordance with seismic category II requirements to withstand an SSE event so that it could not fail and cause an adverse impact or interaction with the seismic category I SSC.

NS SSCs are designed and constructed to the applicable standard building code requirements, industry codes and standards, and/or manufacturer standards.

For SSCs located in the proximity of safety-related SSCs that meet seismic category II requirements by their mounting, the pre-assigned equipment class remains unchanged.

3.2.1.2 Classifications

Table 3.2-3 provides the relationship between different equipment classes, RG 1.29 (Reference 3.2-5) seismic design requirements and the associated QA requirements. Table 3.2-2 provides a list of mechanical and fluid systems, components, and equipment and their designated seismic category along with the equipment class, quality assurance classification, and design codes and standards. The equipment classification is shown on the piping and instrumentation diagrams (P&ID) included in various sections. The seismic classification is identified in Table 3.2-2 and can be determined by the equipment class on the P&IDs, or as noted in Table 3.2-2. The COL Applicant is to identify the site-specific, safety-related systems and components that are designed to withstand the effects of earthquakes without loss of capability to perform their safety function; and those site-specific, safety-related fluid systems or portions thereof; as well as the applicable industry codes and standards for pressure-retaining components.

The seismic category of electrical, mechanical, and instrumentation and control (I&C) equipment included in the Equipment Qualification Program is provided in Appendix 3D.

3.2.1.3 Classification of Building Structures

Table 3.2-4 provides the designated seismic category of building and structures (seismic category I, II, and NS). The US-APWR Nuclear Island consists of the R/B, PCCV, containment internal structure, A/B, access building (AC/B), and east and west power source buildings (PS/Bs). The US-APWR design includes the R/B, PCCV, containment internal structure, and PS/B (east and west). The turbine building (T/B) is part of the standard design but outside the Nuclear Island. Unique non-standard buildings and structures in Table 3.2-4 include the UHSRS, ESWPT, PSFSV, NS T/G Pedestal, and NS Outside Buildings. Minor NS Buildings and all structures in the plant yard are generally not listed in Table 3.2-4. Design of all plant buildings and structures are addressed where appropriate in Chapter 3 and its appendices.

3.2.2 System Quality Group Classification

GDC 1 requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. 10 CFR 50.55a (Reference 3.2-12), as they relate to safety-related SSCs, requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. RG 1.26 (Reference 3.2-13) is used to meet these requirements by classifying safety-related and certain nonsafety-related fluid systems and components and applying corresponding quality codes and standards to such systems and components.

Safety-related fluid systems may perform any of the following functions:

- Fission product containment
- Core cooling

- Reactor shutdown
- Reactivity control
- Post-accident containment heat removal
- Post-accident containment atmosphere cleanup
- Post-accident fission product removal
- Residual heat removal from the reactor and/or from the Spent Fuel Pit
- Containment of radioactive materials

Portions of fluid systems which provide cooling or heating, sealing, lubrication, fuel, motive power, isolation, flood protection, or leakage detection necessary to support the accomplishment of any of the above functions are also considered safety-related.

10 CFR 50.55a (Reference 3.2-12) identifies those ASME Code, Section III, (Reference 3.2-14), Class 1, safety-related components that are part of the RCPB. These components are designated in RG 1.26 (Reference 3.2-13) as Quality Group A. In addition, RG 1.26 (Reference 3.2-13) identifies, on a functional basis, water- and steam-containing components of those safety-related systems that are designated as Quality Groups B and C. Quality Group D applies to non-safety related water- and steam-containing components of systems that are not part of the RCPB or included in Quality Groups A, B or C, or RWMS, but are part of systems or portions of systems that contain or may contain radioactive material.

NRC Generic Letter 90-06 (Reference 3.2-15) is used as guidance for classification of power operated relief valves (PORVs), associated components, and block valves. These components are classified as safety-related for performing safety functions, such as, the mitigation of design-basis steam generator tube rupture accident, low temperature overpressure protection of the RV, and/or plant cooldown. The safety-related classification addresses the redundant and diverse control systems designed to seismic category I criteria. These PORVs and block valves are included in a quality assurance program that is in compliance with 10 CFR 50, Appendix B (Reference 3.2-8), as identified in Table 3.2-2.

The application of 10 CFR 50.55a (Reference 3.2-12) and GDC 1 provides assurance that established standard practices of proven or demonstrated effectiveness are used to achieve a high likelihood that these safety functions will be performed and that the codes and standards applied are commensurate with the importance to safety of these functions.

The scope of quality assurance controls to be applied to a specific SSC, either 10 CFR 50 Appendix B (“Q”) or augmented quality requirements (“A”), is identified in Table 3.2-2 by the “Quality Assurance Classification,” supplemented by use of “Equipment Classifications” 1 through 10 and explanatory notes. Quality assurance classification “Q” is applied to safety-related SSCs. Quality assurance classification “A” is assigned to selected nonsafety-related SSCs based on their contribution to plant safety or to meet NRC guidance that establishes applicable quality assurance requirements. The contribution of nonsafety-related SSCs to plant safety is determined by (1) the SSC's risk-significance as determined by the D-RAP and (2) the reliance on the SSC to address regulatory events, which are ATWS, fire protection and SBO. A list of risk-significant SSCs is provided in Table 17.4-1.

Safety-related and non-safety related instrumentation and electrical equipment are discussed in Chapters 7 and 8. Equipment included in the Equipment Qualification Program is identified in Appendix 3D.

The quality group classifications are based, in part, on the reliance placed on those systems that perform any of the following functions:

- Prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB
- Permit reactor shutdown and maintenance in the safe-shutdown condition
- Contain radioactive material

The US-APWR equipment classification system for safety-related SSCs follows the requirements and guidelines described above. The US-APWR SSCs are classified in Equipment Classes 1 to 10, in part, with respect to the Quality Group Classification. Table 3.2-3 correlates the US-APWR classification of SSCs with the ASME Code, Section III (Reference 3.2-14), RG 1.26 (Reference 3.2-13), NRC Quality Group classes, RG 1.29 (Reference 3.2-5) seismic category, Quality Assurance Classification including conformance with 10 CFR 50 Appendix B (Reference 3.2-8), and other applicable industry codes and standards. Table 3.2-2 provides the equipment classes, quality assurance classification, applicable codes and standards, and seismic category for the US-APWR mechanical and fluid systems, components (including pressure-retaining), and equipment. The COL Applicant is to identify the equipment class, quality assurance classification, applicable codes and standards, and seismic category of the site-specific, safety-related and non safety-related fluid systems, components (including pressure retaining), and equipment.

The identification and standards of the US-APWR equipment classifications and their definitions and criteria are described below.

3.2.2.1 Equipment Class 1

Equipment Class 1 has the highest integrity for SSCs and is equivalent to RG 1.26 (Reference 3.2-13), NRC Quality Group A and applies to the components of the RCPB as defined in 10 CFR 50.55a (Reference 3.2-12).

Equipment Class 1 SSCs are classified as safety-related, seismic category I, and the codes and standards for the NRC Quality Group A are applied. Equipment Class 1 components are designed to meet ASME Code, Section III (Reference 3.2-14), Class 1 requirements and the QA criteria of 10 CFR 50, Appendix B (Reference 3.2-8). Supports are designed and constructed to ASME Code, Section III (Reference 3.2-14), Subsection NF requirements and the same QA criteria for components applies.

3.2.2.2 Equipment Class 2

Equipment Class 2 includes equipment addressed by RG 1.26 (Reference 3.2-13), NRC Quality Group B. Equipment Class 2 applies to water- and steam-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves or non-ASME Code Section III mechanical equipment that are either (1) part of the RCPB defined in 10 CFR 50.2 (Reference 3.2-1) but excluded from the requirements of 10 CFR 50.55a (Reference 3.2-12) pursuant to paragraph c(2) of that section, or (2) not part of the RCPB but are part of the following:

- Safety-related systems or portions of systems that are designed for the following:
    - Emergency core cooling
    - Post-accident containment heat removal
    - Post-accident fission product removal
- Safety-related systems or portions of systems that are designed for the following:
    - Reactor shutdown
    - Residual heat removal
- Those portions of the steam and feedwater systems extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.
- Systems or portions of systems that are connected to the RCPB and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.

In addition to the above, the following systems and components designated NRC Quality Group B are classified in SRP 3.2.2 (Reference 3.2-16), Appendix A, as Equipment Class 2:

- Containment isolation system penetrations including associated piping and valves, and isolation barriers comprised of closed systems inside and outside containment described in Subsection 6.2.4.
- Containment isolation system instrument sensing lines penetrating containment described in “Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11) Supplement to Safety Guide 11, Backfitting Considerations”, RG 1.11

(Reference 3.2-17) that may be used to implement containment isolation design criteria for instrument lines.

- Safety-related instrument sensing lines described in RG 1.151 (Reference 3.2-9) for classifying instrument sensing lines in terms of the ASME Code, Section III (Reference 3.2-14), Class 2.

Equipment Class 2 SSCs are classified as safety-related and seismic category I. For SSCs addressed by RG 1.26, the codes and standards for NRC Quality Group B are applied. These RG 1.26 Equipment Class 2 components are designed to meet ASME Code, Section III (Reference 3.2-14), Class 2 requirements and the QA criteria of 10 CFR 50, Appendix B (Reference 3.2-8) applies. Supports are designed and constructed to ASME Code, Section III (Reference 3.2-14), Subsection NF requirements and the same QA criteria for components applies. Other non-ASME Code Section III Equipment Class 2 components are designed to meet codes and standards described in the design bases description of the applicable system, and meet the QA criteria of 10 CFR 50, Appendix B (Reference 3.2-8).

3.2.2.3 Equipment Class 3

Equipment Class 3 includes equipment addressed by RG 1.26 (Reference 3.2-13), NRC Quality Group C. Equipment Class 3 applies, in part, to water-, steam-, and radioactive-waste-containing pressure vessels; heat exchangers (other than turbines and condensers); storage tanks; piping; pumps; and valves that are not part of the RCPB or included in Quality Group B but part of the following:

- Safety-related cooling water and emergency feedwater systems or portions of those systems that are designed for the following:
    - Emergency core cooling
    - Post-accident containment heat removal
    - Post-accident containment atmosphere cleanup
- Residual heat removal from the reactor and from the spent fuel pit (including primary and secondary cooling systems), although Quality Group B includes portions of those systems that are required for their safety functions and that (i) do not operate during any mode of normal reactor operation and (ii) cannot be tested adequately
- Cooling water and seal water systems or portions of those safety-related systems that are designed for the functioning of safety-related components and systems, such as RCPs and the MCR.
- Systems or portions of systems that are connected to the RCPB and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.

- Systems, other than RWMS, not covered above, that contains or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that exceed 0.5 rem to the whole body or its equivalent to any part of the body. For those systems located in seismic category I structures, only single component failures need be assumed. No credit is taken for automatic isolation from other components in the system or treatment of released materials unless isolation or treatment capability is designed to appropriate seismic and quality group standards and can withstand loss of offsite power and a single failure of an active component.

In addition to the systems and components described above, the following systems and components are designated as Equipment Class 3:

- Emergency power sources (gas turbines)

- Equipment and floor drainage system described in Subsection 9.3.3, SRP Section 9.3.3 (Reference 3.2-18)

- Plant ventilation systems for areas such as the MCR and engineered safety feature rooms

- Safety-related instrument sensing lines described in RG 1.151 (Reference 3.2-9) for classifying instrument sensing lines in terms of the ASME Code, Section III (Reference 3.2-14), Class 3

- Ultimate heat sink (UHS) and supporting systems described in Subsection 9.2.5.

Equipment Class 3 SSCs are classified as safety-related and seismic category I. For SSCs addressed by RG 1.26, the codes and standards for NRC, Quality Group C are applied. These RG 1.26 Equipment Class 3 components are designed to meet ASME Code, Section III (Reference 3.2-14), Class 3 requirements and the QA criteria of 10 CFR 50, Appendix B (Reference 3.2-8). Supports are designed and constructed to ASME Code, Section III (Reference 3.2-14), Subsection NF requirements, and the same QA criteria for components apply to the supports. Other non-ASME Code Section III Equipment Class 3 components are designed to meet codes and standards described in the design bases description of the applicable system, and meet the QA criteria of 10 CFR 50, Appendix B (Reference 3.2-8).

3.2.2.4 Equipment Class 4

Equipment Class 4 includes selected equipment addressed by NRC, Quality Group D in accordance with RG 1.26 (Reference 3.2-13). Equipment Class 4 applies to water- and steam-containing non-safety related components that: 1) are not part of the RCPB, 2) are not included in Quality Groups B or C, or RWMS, 3) are part of systems or portions of systems that contain or may contain radioactive material, and 4) require augmented quality as defined in Section 3.1.1.1.1. Augmented quality assurance requirements for nonsafety-related SSCs, as described in the quality assurance program, are applied to Equipment Class 4 SSCs.

Equipment Class 4 SSCs are classified as nonsafety-related and NS or seismic category II. The codes and standards for NRC Quality Group D are applied as follows:

• Pressure Vessels	ASME Code, Section VIII, Division 1 (Reference 3.2-19)
• Piping	ASME B31.1 (Reference 3.2-20)
• Pumps	Manufacturers' standards
• Valves	ASME B31.1 (Reference 3.2-20)
• Atmospheric Storage Tanks	API-650 (Reference 3.2-21), AWWA D-100 (Reference 3.2-22), or ASME B96.1 (Reference 3.2-23)
• 0-15 psig Storage Tanks	API-620 (Reference 3.2-24)
• Supports	Manufacturers' standards

3.2.2.5 Other Equipment Classes

<u>Equipment Class 5</u>

Equipment Class 5 is assigned to nonsafety-related components that: 1) are listed as "risk significant" in Table 17.4-1, are designed to meet special seismic requirements such as seismic category II, or perform functions that address ATWS or station blackout, and 2) are not within the purview of Equipment Classes 4, 6, 7, and 8.

Equipment Class 5 SSCs are classified either as non-seismic (NS) or seismic category II. Augmented quality assurance requirements for nonsafety-related SSCs, as described in the quality assurance program, are applied to Equipment Class 5 SSCs. Codes and standards, as defined in the design bases, are applied to Equipment Class 5 components.

The COL Applicant is to apply DCD methods of equipment classification, quality assurance classification, and seismic categorization of risk-significant, non-safety related SSCs to site-specific, nonsafety-related SSCs based on their contribution to plant safety.

<u>Equipment Class 6</u>

Equipment Class 6 is assigned to the nonsafety-related components of the RWMS and a part of SGBDS which are outside the containment isolation valves.

The seismic category defined in RG 1.143 (Reference 3.2-10) is applied. Portions of the Equipment Class 6 SSCs on which seismic category II requirements are imposed are designed to comply with both the requirements of RG 1.143 and seismic category II. Augmented quality assurance requirements for nonsafety-related SSCs, as described in the quality assurance program, are applied to Equipment Class 6 SSCs.

The Equipment Class 6 components are designed in compliance with applicable codes and standards, and guidance provided in RG 1.143 (Reference 3.2-10).

Equipment Class 7

Equipment Class 7 is assigned to the nonsafety-related components of the fire protection system. Portions of the Equipment Class 7 SSCs on which seismic category II requirements are imposed are designed to comply with both the requirements of RG 1.189 (reference 3.2-11) and seismic category II. Augmented quality assurance requirements for nonsafety-related SSCs, as described in the quality assurance program, are applied to Equipment Class 7 SSCs.

The codes and standards applicable to fire protection systems follow the guidance of RG 1.189 (Reference 3.2-11), Section 1.7, and National Fire Protection Association 804 (Reference. 3.2-25), Section 4.6, and are applied for fire protection system design in the US-APWR buildings.

Equipment Class 8

Equipment Class 8 is assigned to non-safety related components that contain or may contain radioactive materials, that are classified as Quality Group D, are not included in Equipment Classes 1-7, and do not require augmented quality per Section 3.1.1.1.1.

Equipment Class 8 SSCs are classified as NS. The applicable codes and standards are the same as Equipment Class 4. See Equipment Class 4 codes and standards for NRC Quality Group D.

Equipment Class 9 and Equipment Class 10

Non-safety related components and structures that have impact on continuous power generation and do not fall into any one of Quality Groups A to D or Equipment Classes 1 through 8 are classified as Equipment Class 9. Nonsafety-related SSCs not in Equipment Classes 4 - 9 are classified as Equipment Class 10.

The Equipment Class 9 and Equipment Class 10 SSCs are classified NS. Codes and standards as defined in the design bases are applied to Equipment Class 9 and Equipment Class 10 components.

3.2.2.6 Inspection Requirements

Safety-related SSCs built to the requirements of the ASME Code, Section III (Reference 3.2-14), are required by 10 CFR 50.55a (Reference 3.2-12) to have in-service inspections. The ISI program is discussed in Subsection 3.9.6.

3.2.3 Combined License Information

COL 3.2(1) *Deleted*

COL 3.2(2) *Deleted*

COL 3.2(3) *Deleted*

COL 3.2(4) *The COL Applicant is to identify the site-specific, safety-related systems and components that are designed to withstand the effects of earthquakes without loss of capability to perform their safety function; and those site-specific, safety-related fluid systems or portions thereof; as well as the applicable industry codes and standards for pressure-retaining components.*

COL 3.2(5) *The COL Applicant is to identify the equipment class and seismic category of the site-specific, safety-related and non safety-related fluid systems, components (including pressure retaining), and equipment as well as the applicable industry codes and standards.*

COL 3.2(6) *The COL Applicant is to apply DCD methods of equipment classification and seismic categorization of risk-significant, non-safety related SSCs based on their safety role assumed in the PRA and treatment by the D-RAP.*

3.2.4 References

3.2-1 Definitions, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50.2, U.S. Nuclear Regulatory Commission, Washington, DC.

3.2-2 Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents—Nuclear Power Reactors, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50.34(a)(1), U.S. Nuclear Regulatory Commission, Washington, DC.

3.2-3 Determination of Exclusion area, Low Population Zone, and Population Center Distance, Reactor Site Criteria, Energy. Title 10, Code of Federal Regulations, Part 100.11, U.S. Nuclear Regulatory Commission, Washington, DC.

3.2-4 General Design Criteria for Nuclear Power Plants, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix A, U.S. Nuclear Regulatory Commission, Washington, DC.

3.2-5 Seismic Design Classification. Regulatory Guide 1.29, Rev. 4, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.2-6 Earthquake Engineering Criteria for Nuclear Power Plants, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of

Federal Regulations, Part 50, Appendix S, U.S. Nuclear Regulatory Commission, Washington, DC.

3.2-7 Geologic and Seismic Siting Criteria, Reactor Site Criteria, Energy. Title 10, Code of Federal Regulations, Part 100.23, U.S. Nuclear Regulatory Commission, Washington, DC.

3.2-8 Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50, Appendix B.

3.2-9 Instrument Sensing Lines. Regulatory Guide 1.151, U.S. Nuclear Regulatory Commission, Washington, DC, July 1983.

3.2-10 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants. Regulatory Guide 1.143, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2001.

3.2-11 Fire Protection for Nuclear Power Plants. Regulatory Guide 1.189, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.2-12 Codes and Standards, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10, Code of Federal Regulations, Part 50.55a, U.S. Nuclear Regulatory Commission, Washington, DC.

3.2-13 Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants. Regulatory Guide 1.26, Rev. 4, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

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Table 3.2-1 Non-Safety Components Required for Normal Shutdown (Sheet 1 of 2)

Systems	Components
Reactor Coolant System	Reactor Coolant System and oil lift pump
	Pressurizer heater (Control Group)
	Pressurizer spray valve
Chemical and volume control system	Charging pump
	Boric acid transfer pump
	Volume control tank
	Boric acid tank
	Hold up tank
	Regenerative heat exchanger
	Letdown heat exchanger
	Seal water heat exchanger
	Reactor coolant filter
	Seal water injection filter
	Seal water return strainer
	Charging flow control valve
	Seal water injection flow control valve
	Letdown line 1st (2nd) stop valve
	Letdown line inside prestressed concrete containment vessel
	Centrifugal charging pump inlet line volume control tank Side 1st, 2nd isolation valves
	Centrifugal charging pump inlet line boric and tank side isolation valve
	Centrifugal charging pump inlet line refueling water storage auxiliary tank side isolation valves
	Residual heat removal letdown line pressure control valve
	Seal water return line 1st, 2nd isolation valves
Primary Makeup Water System	Primary make-up water pump
	Primary make-up water storage tank
Residual Heat Removal System	CS/residual heat removal cooler outlet flow control valves
	CS/residual heat removal heat exchanger bypass flow control valves
Main Stream and Feedwater System	Main steam relief valves (Normal)
	Turbine bypass valves
	Main feedwater bypass valves
	Steam generator water filling control valves

Table 3.2-1 Non-Safety Components Required for Normal Shutdown (Sheet 2 of 2)

Systems	Components
Instrument Air System	Instrument air compressors
Secondary System	Condenser
	Condensate pump
	Deaerator
	Main feedwater pump
	Cooling towers
	Circulating water pumps
Heating, Ventilation, and Air Conditioning	Containment fan cooler unit fan
	Reactor cavity cooling fan
	Control rod drive mechanism cooling fan
	Non-Class 1E electrical room air handling unit fan
	Non-essential chiller units
	Non-essential chilled water pumps

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 1 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Primary System							
1. Reactor Systems							
Fuel assemblies	2	PCCV	N/A	Q	5	I	
Rod cluster control	2	PCCV	N/A	Q	5	I	
(Deleted)							
(Deleted)							
Upper core support assembly	3	PCCV	N/A	Q	ASME III, CS	I	
Lower core support assembly	3	PCCV	N/A	Q	ASME III, CS	I	
Guide tube assemblies	3	PCCV	N/A	Q	5	I	
Control rod drive mechanism latch housing	1	PCCV	A	Q	1	I	
Control rod drive mechanism rod travel housing	1	PCCV	A	Q	1	I	
Surveillance capsule guide basket	5	PCCV	N/A	A	5	II	Note 5.a
Internal structures	5	PCCV	N/A	A	5	II	Note 5.a
2. Reactor Coolant System							
Reactor vessel	1	PCCV	A	Q	1	I	
Reactor vessel head	1	PCCV	A	Q	1	I	
Reactor vessel insulation (shell)	5	PCCV	N/A	A	5	II	Note 5.a
Reactor vessel insulation (closure head)	5	PCCV	N/A	A	5	II	Note 5.a
Reactor coolant pump casing	1	PCCV	A	Q	1	I	
Reactor coolant pump main flange	1	PCCV	A	Q	1	I	
Reactor coolant pump thermal barrier heat exchanger	1	PCCV	A	Q	1	I	
Reactor coolant pump #1 seal housing	1	PCCV	A	Q	1	I	
Reactor coolant pump #2 seal housing	2	PCCV	B	Q	2	I	
Reactor coolant pump pressure-retaining bolting	1	PCCV	A	Q	1	I	
Reactor coolant pump insulation	5	PCCV	N/A	A	5	II	Note 5.a
Pressurizer	1	PCCV	A	Q	1	I	
Pressurizer insulation	5	PCCV	N/A	A	5	II	Note 5.a

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 2 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Pressurizer piping upstream of and including the pressurizer safety valves RCS-SRV-120,121,122,123, safety depressurization valves RCS-MOV-117A,B, and depressurization valves RCS-MOV-119	1	PCCV	A	Q	1	I	
Pressurizer piping downstream of and excluding pressurizer safety valves RCS-SRV120,121,122,123	4	PCCV	D	A	4	II	Note 5.a
Pressurizer piping downstream of and excluding safety depressurization valves RCS-MOV-117A,B	4	PCCV	D	A	4	II	Note 5.a
Pressurizer piping downstream of and excluding depressurization valves RCS-MOV-119	8	PCCV	D	N	4	NS	
Pressurizer piping upstream of and including the loop seal drain stop valves: RCS-VLV-157,158,159,160	2	PCCV	B	Q	2	I	
Pressurizer piping downstream of the loop seal drain stop valve: RCS-VLV-157,158,159,160	8	PCCV	D	N	4	NS	
Pressurizer vent piping up to RCS-VLV-153 and the valve RCS-VLV-153	2	PCCV	B	Q	2	I	
Auxiliary spray TC piping up to RCS-VLV-151 and the valve RCS-VLV-151	2	PCCV	B	Q	2	I	
Reactor vessel head vent piping upstream of and including the reactor vessel head vent valves RCS-MOV-002A, B,003A,B	1	PCCV	A	Q	1	I	
Reactor vessel head vent line piping downstream of and excluding the reactor vessel head vent valves RCS-MOV-003A,B	8	PCCV	D	N	4	NS	
Letdown line piping upstream of and including the letdown line stop valves RCS-VLV-021	1	PCCV	A	Q	1	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 3 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Reactor coolant piping drain piping upstream of and including the second drain stop valve RCS-VLV-023A, B, C, D	1	PCCV	A	Q	1	I	
Cavity / reactor coolant system water level meter piping upstream of and including the stop valves RCS-VLV-024.025	1	PCCV	A	Q	1	I	
Steam generator tube side	1	PCCV	A	Q	1	I	
Steam generator shell side	2	PCCV	B	Q	2	I	
Steam generator insulation	5	PCCV	N/A	A	5	II	Note 5.a
Pressurizer safety valves RCS-SRV-120, 121, 122, 123	1	PCCV	A	Q	1	I	
Safety depressurization valves RCS-MOV-117A, B	1	PCCV	A	Q	1	I	
Safety depressurization valve block valves RCS-MOV-116A, B	1	PCCV	A	Q	1	I	
Depressurization valves for severe accident RCS-MOV-118, 119	1	PCCV	A	Q	1	I	
Pressurizer spray valves RCS-PCV-061A,B	1	PCCV	A	Q	1	I	
Pressurizer spray block valves RCS-MOV-111A, B	1	PCCV	A	Q	1	I	
(Deleted)							
Letdown line stop valve RCS-VLV-021	1	PCCV	A	Q	1	I	
Reactor coolant piping first drain stop valvesRCS-VLV-022A, B, C, D	1	PCCV	A	Q	1	I	
Pressurizer gas phase sample line piping and valve up to and excluding valve PSS-VLV-001	2	PCCV	B	Q	2	I	
Pressurizer liquid phase sample line piping and valve up to and excluding valve PSS-VLV-004	2	PCCV	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 4 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Pressurizer spray bypass valves RCS-VLV-112A,B	1	PCCV	A	Q	1	I	
Reactor coolant piping	1	PCCV	A	Q	1	I	
Main coolant piping insulation	5	PCCV	N/A	A	5	II	Note 5.a
Pressurizer surge line piping	1	PCCV	A	Q	1	I	
Pressurizer spray line piping	1	PCCV	A	Q	1	I	
Pressurizer relief tank	4	PCCV	D	A	4	II	Note 5.a
Reactor coolant system piping and valves related to pressurizer relief tank excluding containment isolation valves, piping between the valves and upstream of VLV-135	8	PCCV	D	N	4	NS	
Following containment isolation valves and piping between valves: RCS-AOV-147, 148 RCS-AOV-132, RCS-VLV-133 RCS-AOV-138, RCS-VLV-139, 140	2	PCCV R/B	B	Q	2	I	
3. Chemical and Volume Control System							
Charging pumps	3	R/B	C	Q	3	I	
Boric acid transfer pumps	4	A/B	D	A	4	NS	Note 5.d
Boric acid evaporator feed pumps	8	A/B	D	N	4	NS	
Volume control tank	4	R/B	D	A	4	II	Note 5.a
Holdup tanks	8	A/B	D	N	4	NS	
Boric acid batching tank	10	A/B	N/A	N	5	NS	
Boric acid tanks	4	A/B	D	A	4	NS	Note 5.d
Resin fill tank	8	A/B	D	N	4	NS	
Chemical mixing tank	8	R/B	D	N	4	NS	
Reactor coolant pump purge water head tank	8	PCCV	D	N	4	NS	
Regenerative heat exchanger	3	PCCV	C	Q	3	I	
Letdown heat exchanger –tube side	3	PCCV	C	Q	3	I	
Letdown heat exchanger – component cooling water side	2	PCCV	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 5 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Seal water heat exchanger –tube side	4	R/B	D	A	4	II	Note 5.a
Seal water heat exchanger – component cooling water side	4	R/B	D	A	4	II	Note 5.a
Excess letdown heat exchanger	3	PCCV	C	Q	3	I	
Excess letdown heat exchanger – component cooling water side	2	PCCV	B	Q	2	I	
Reactor coolant filters	8	A/B	D	N	4	NS	
Seal water injection filters	4	A/B	D	A	4	NS	Note 5.d
Boric acid filter	4	A/B	D	A	4	NS	Note 5.d
Boric aid evaporator feed demineralizer filter	8	A/B	D	N	4	NS	
Mixed bed demineralizer inlet filters	8	A/B	D	N	4	NS	
Seal water return strainer	4	R/B	D	A	4	II	Note 5.a
Mixed bed demineralizers	8	A/B	D	N	4	NS	
Cation bed demineralizer	8	A/B	D	N	4	NS	
Boric acid evaporator feed demineralizer	8	A/B	D	N	4	NS	
Deborating demineralizers	8	A/B	D	N	4	NS	
Letdown orifices	3	PCCV	C	Q	3	I	
Charging pump minimum flow orifices	3	R/B	C	Q	3	I	
Boric acid transfer pump minimum flow orifices	4	A/B	D	A	4	NS	Note 5.d
Boric acid evaporator feed pumps minimum flow orifices	8	A/B	D	N	4	NS	
Charging flow control orifice	3	R/B	C	Q	3	I	
Seal water flow control orifice	3	R/B	C	Q	3	I	
Chemical mixing tank orifice	8	R/B	D	N	4	NS	
Boric acid evaporator	8	A/B	D	N	4	NS	
Boric acid blender	4	R/B	D	A	4	II	
Letdown line and valves from reactor coolant system to and including valve CVS-LCV-362 prior to regenerative heat exchanger.	1	PCCV	A	Q	1	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 6 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Letdown line piping and valves from and excluding the valve CVS-LCV-362 prior to regenerative heat exchanger to the following valves: \ Residual Heat Removal System valves (2 each) (excluding the valves) RHS-AOV-024B,C Containment isolation valve (excluding the valve) CVS-AOV-005 Letdown line relief valve CVS-SRV-002 (including the valve)	3	PCCV	C	Q	3	I	
Chemical and volume control system containment isolation valves and piping between the valves.	2	PCCV R/B	B	Q	2	I	
Excess letdown piping and valves from reactor coolant system to and including valve CVS-AOV-222 just prior to excess letdown heat exchanger.	1	PCCV	A	Q	1	I	
Excess letdown piping and valves from but excluding valve CVS-AOV-222 just prior to excess letdown heat exchanger to and excluding containment isolation valves CVS-MOV-203 and CVS-VLV-202. This includes piping related to reactor coolant pump seal water return line to the following valves. 4 valves CVS-AOV-192A,B,C,D (excluding the valves) Seal water return line relief valve CVS-SRV-201 (including the valve)	3	PCCV	C	Q	3	I	
Excess letdown heat exchanger drain piping	3	PCCV	C	Q	3	I	
Reactor coolant pump purge water piping and valves up to but excluding reactor coolant pump #2 seal housings	8	PCCV	D	N	4	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 7 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Letdown line piping and valves outside containment from and excluding the wall that separates R/B and A/B to and including the three way valve CVS-TCV-014	8	A/B	D	N	4	NS	
Letdown line piping and valves outside containment from and excluding the containment isolation valve CVS-AOV-006 to and including the wall that separates R/B and A/B	4	R/B	D	A	4	II	Note 5.a
Piping and valves contained within the demineralizer subsystem of chemical and volume control system. This subsystem branches off the letdown line between the two three way valves CVS-TCV-014 and CVS-LCV-031A. It includes reactor coolant purification filters and demineralizers, and deborating demineralizers	8	A/B	D	N	4	NS	
Letdown line piping and valves outside containment from the three way valve CVS-LCV-031A up to and excluding the wall that separates R/B and A/B	8	A/B	D	N	4	NS	
Letdown line piping and valves outside containment from and including the wall that separates R/B and A/B on the upstream valves CVS-VLV-101 to the volume control tank	4	R/B	D	A	4	II	Note 5.a
Reactor coolant pump seal water return piping and valves outside containment from and excluding containment isolation valve CVS-MOV-204 to valves CVS-VLV-213, CVS-VLV-214, and CVS-SRV-210 (excluding CVS-VLV-213, including CVS-VLV-214 and CVS-SRV-210)	4	R/B	D	A	4	II	Note 5.a

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 8 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Reactor coolant pump seal water return piping and valves from reactor coolant pump seal to and including 4 valves CVS-AOV-192A,B,C,D	2	PCCV	B	Q	2	I	
Reactor coolant pump seal water injection piping and valves excluding containment isolation valves, piping between these valves, piping downstream of CVS-VLV-180A, B, C, D (excluding valves), and seal water injection filter line valves and piping between and including CVS-VLV-168 and CVS-VLV-173	3	R/B PCCV	C	Q	3	I	
Reactor coolant pump seal water injection piping and valves downstream of including valves CVS-VLV-180A, B, C, D	1	PCCV	A	Q	1	I	
Seal water injection filter line valves and piping between and excluding CVS-VLV-168 and CVS-VLV-173	4	R/B A/B	D	A	4	NS / II	Note 5.d / Note 5.a
Charging lines from and including valves CVS-VLV-158 and CVS-AOV-159 to their penetration into the reactor coolant system	1	PCCV	A	Q	1	I	
Auxiliary spray line from and including valves CVS-AOV-155 to the penetration into the RCS	1	PCCV	A	Q	1	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 9 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Charging line and auxiliary spray line piping and valves between the following valves (excluding the valves) downstream of the regenerative heat exchanger: CVS-VLV-158 CVS-AOV-159 CVS-AOV-155 And the following valves: Containment isolation valve CVS-VLV-153 (excluding the valve)	3	PCCV	C	Q	3	I	
Charging line piping and valves from and including the volume control outlet valve CVS-LCV-031B to charging pump minimum flow orifices and following valves: CVS-VLV-213 (including valve) CVS-VLV-585 (including valve) CVS-VLV-557 (including valve) CVS-VLV-163 and 164 (excluding valves) CVS-MOV-152 (excluding valve) CVS-VLV-591 and 593 (including valves)	3	R/B	C	Q	3	I	
Volume control tank outlet line to and excluding valve CVS-LCV-031B and the line from and including CVS-VLV-568	4	R/B	D	A	4	II	Note 5.a
Volume control tank drain piping and valves up to and including CVS-VLV-636	4	R/B	D	A	4	II	Note 5.a
Volume control tank drain piping and valves from and excluding CVS-VLV-636	8	R/B	D	N	4	NS	
Chemical and volume control system make up piping and valves to and excluding valve CVS-FCV-128	4	R/B	D	A	4	II	Note 5.a

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 10 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Chemical and volume control system piping and valves on the chemical mixing tank side of and excluding the valves CVS-VLV-585 and CVS-VLV-584	8	R/B A/B	D	N	4	NS	
Chemical and volume control system piping and valves related to the chemical mixing tank and the blender	8	R/B	D	N	4	NS	
Chemical and volume control system piping and valves related to the boric acid tanks excluding valve CVS-VLV-557, up to and including valves CVS-VLV-542,547, CVS-AOV-549A,B, through boric acid transfer pump A,B, boric acid filter and including the downstream piping from and including valve CVS-VLV-525	4	A/B R/B	D	A	4	NS / II	Note 5.d / Note 5.a
Chemical and volume control system piping and valves related to the boric acid batching tank to and excluding valve CVS-VLV-525	10	A/B	N/A	N/A	5	NS	
(Deleted)							
Chemical and volume control system piping and valves related to the boric acid pump minimum flow piping up to boric acid tank, through transfer pump minimum flow orifice and valve CVS-VLV-531A,B	4	A/B	D	A	4	NS	Note 5.d
Chemical and volume control system piping and valves related to the boric acid tanks up to and including valves CVS-SRV-509A,B, CVS-VLV-511A,B, CVS-VLV-508A,B	4	A/B	D	A	4	NS	Note 5.d
Chemical and volume control system piping and valves related to the boric acid tanks excluding foregoing piping and valves	4	A/B	D	A	4	NS	Note 5.d

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 11 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Chemical and volume control system piping related to the holdup tanks within R/B including the wall that separates R/B and A/B	4	R/B	D	A	4	II	Note 5.a
Chemical and volume control system piping and valves related to the holdup tanks and the boric acid evaporator feed pumps	8	A/B	D	N	4	NS	
Chemical and volume control system piping and valves related to the boric acid evaporator and the boric acid evaporator feed demineralizer.	8	A/B	D	N	4	NS	
Chemical and volume control system piping and valves related to the primary makeup water supply isolation CVS-FCV-133A, 129, 128 and CVS-VLV-581	3	R/B	C	Q	3	I	
4. Safety Injection System							
Safety injection pumps	2	R/B	B	Q	2	I	
Safety injection piping and valves between the System penetration and including the second check valve SIS-VLV-012A, B, C, D upstream of the direct Vessel Injection penetration	1	PCCV	A	Q	1	I	
Safety injection piping and valves upstream of and excluding the second check valve SIS-VLV-012A, B, C, D upstream of the direct Vessel Injection penetration	2	PCCV, R/B	B	Q	2	I	
Hot leg injection piping downstream of and including the motor operated valves SIS-MOV-014A, B, C, D	1	PCCV	A	Q	1	I	
Hot leg injection piping upstream of but excluding the motor operated valves SIS-MOV-014A, B, C, D	2	PCCV	B	Q	2	I	
Accumulator	2	PCCV	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 12 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Accumulator piping and valves on the reactor coolant system side of and including the second check valves SIS-VLV-102A, B, C, D	1	PCCV	A	Q	1	I	
Accumulator piping and valves on the accumulator side of but excluding the second check valves SIS-VLV-102A, B, C, D	2	PCCV	B	Q	2	I	
Emergency core cooling / containment spray strainer	2	PCCV	N/A	Q	5	I	
Emergency letdown isolation valves SIS-MOV-031A, 031D, 032A, 032D and piping between valves	1	PCCV	A	Q	1	I	
Emergency let down piping from and excluding valves SIS-MOV-032A,D	2	PCCV	B	Q	2	I	
Accumulator N2 vent piping up and including valves SIS-AOV-114,SIS-SRV-116, SIS-MOV-121A,B and SIS-HCV-017	2	PCCV R/B	B	Q	2	I	
(Deleted)							
Safety injection system valve leak test piping between test line isolation valves (not including), SIS-VLV-216A,B (not including) and SIS-AOV-201B,C (not including)	8	PCCV	D	N	4	NS	
5. Residual Heat Removal System(RHRS)							
Containment Spray/Residual Heat Removal pumps	2	R/B	B	Q	2	I	
Containment spray/residual heat removal heat exchangers - tube side	2	R/B	B	Q	2	I	
Containment spray/residual heat removal heat exchangers - component cooling water side	3	R/B	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 13 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Residual heat removal suction piping and valves on the reactor coolant system side between the hot legs, up to and including the second motor operated valves RHS-MOV-002A, B, C, D	1	PCCV	A	Q	1	I	
Residual heat removal discharge piping and valves on the reactor coolant system side between the cold legs, up to and including the second check valves RHS-VLV-027A,B,C,D	1	PCCV	A	Q	1	I	
Residual heat removal system piping and valves on the residual heat removal system side from and excluding the second motor operated valves RHS-MOV-002A, B, C, D to and excluding the second check valves RHS-VLV-027A,B,C,D	2	PCCV R/B	B	Q	2	I	
Residual heat removal system piping and valves not mentioned above up to and including the valves interfacing with systems of a lower classification.	2	PCCV R/B	B	Q	2	I	
6. Emergency Feedwater System (EFWS)							
Emergency feedwater pumps	3	R/B	C	Q	3	I	
Emergency feedwater pits	3	R/B	N/A	Q	5	I	
Emergency feedwater pump turbine steam drain pots	4	R/B	D	A	4	II	Note 5.a
Emergency feedwater pump discharge piping and valves up to and excluding emergency feedwater isolation valves EFS-MOV-019A,B,C,D	3	R/B	C	Q	3	I	
Emergency feedwater pump suction piping and valves from Emergency feedwater pits	3	R/B	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 14 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Emergency feedwater system containment isolation valves EFS-MOV-101A,B,C,D, EFS-MOV-019A,B,C,D	2	R/B	B	Q	2	I	
(Deleted)							
Emergency feedwater pump miniflow and fullflow piping and valves to emergency feedwater pit	3	R/B	C	Q	3	I	
Emergency feedwater pump discharge tie line piping and valves	3	R/B	C	Q	3	I	
Emergency feedwater pit water supply piping and valves from Emergency feedwater pit up to and including EFS-VLV-001A,B	3	R/B	C	Q	3	I	
Emergency feedwater pump suction piping and valves from and including the valve EFS-MOV-006A,B	3	R/B	C	Q	3	I	
Emergency feedwater pump suction line piping from and including EFS-VLV-004	3	R/B	C	Q	3	I	
Emergency feedwater pit sampling piping up to and including EFS-VLV-041A,B	3	R/B	C	Q	3	I	
Emergency feedwater pit overflow piping	10	R/B	N/A	N	5	NS	
Emergency feedwater pit drain piping and valves up to and including EFS-VLV-032 A,B	3	R/B	C	Q	3	I	
Water supply piping and valves from emergency feedwater pits to spent fuel pit up to and including EFS-MOV-031	3	R/B	C	Q	3	I	
Emergency feedwater supply piping to spent fuel pit between and excluding EFS-VLV-031 and SFS-VLV-024	4	R/B	D	A	4	II	Note 5.a

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 15 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Turbine driven emergency feedwater pump steam supply piping and valves from and excluding EFS-MOV-101A,B,C,D to the pumps	3	R/B	C	Q	3	I	
Turbine driven emergency feedwater pump steam supply piping drain piping and valves up to and including EFS-VLV-109A,B,C,D	3	R/B	C	Q	3	I	
Turbine driven emergency feedwater pump steam supply piping warming piping and valves	3	R/B	C	Q	3	I	
Turbine driven emergency feedwater pump steam supply piping drain piping and valves up to and including EFS-VLV-117A,D, 114A,D, 111A,D, 152A, D, 154A, D, 156 A,D	3	R/B	C	Q	3	I	
Turbine driven emergency feedwater pump steam supply piping drain piping and valves downstream and excluding EFS-VLV-117A,D,114A,D, 111A,D, 152A, D, 154A, D, 156 A,D	4	R/B	D	A	4	II	Note 5.a
Turbine driven emergency feedwater pump steam exhaust piping	4	R/B	D	A	4	II	Note 5.a
Turbine driven emergency feedwater pump steam exhaust piping drain piping	4	R/B	D	A	4	II	Note 5.a
Emergency feedwater pump turbine steam drain pot drain piping and valves	4	R/B	D	A	4	II	Note 5.a
Emergency feedwater pump turbine steam drain pot cooling water supply piping and valves	5	R/B	N/A	A	5	II	Note 5.a

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 16 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
7. Condensate and Feedwater System							
The piping upstream of the main feedwater isolation valves FWS-SMV-512A, B, C, D to the first piping restraint at the interface between the R/B and T/B	3	R/B	C	Q	3	I	
Main feedwater piping and valves to the steam generators from and including the main feedwater isolation valves FWS-SMV-512A, B, C, D	2	R/B PCCV	B	Q	2	I	
Main feedwater piping upstream of the restraint at the interface between the R/B and the T/B	8	T/B	D	N	4	NS	
Emergency feedwater piping from and excluding EFS-MOV-019A,B,C,D	2	R/B	B	Q	2	I	
High pressure cleanup piping and valves in the R/B	3	R/B	C	Q	3	I	
High pressure cleanup piping and valves out of the R/B	10	T/B	N/A	N	5	NS	
Steam generator water filling piping and valves in the R/B	3	R/B	C	Q	3	I	
Steam generator water filling piping and valves out of the R/B	8	T/B	D	N	4	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 17 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
8. Main Steam Supply System (MSS)							
Main steam piping and valves including branch pipe from steam generators up to and including the following valves: 4 Nitrogen supply piping valves MSS-VLV-531A, B, C, D Main steam isolation valves MSS-SMV-515A, B, C, D Main steam bypass isolation valves MSS-HCV-565, 575, 585, 595 Main steam relief valves MSS-PCV-515, 525, 535, 545 Main steam depressurization valves MSS-MOV-508A,B,C,D Main steam safety valves MSS-SRV-509A,B,C,D, 510A,B,C,D, 511A,B,C,D, 512A,B,C,D, 513A,B,C,D, 514A,B,C,D Main steam drain isolation valves MSS-MOV-701A,B,C,D	2	R/B PCCV	B	Q	2	I	
Branch piping from the main steam piping to the turbine driven emergency feedwater system pump turbines up to and excluding EFS-MOV-101A,B,C,D	2	R/B	B	Q	2	I	
Main steam drain piping and valves located upstream of main steam isolation valves downstream and excluding the main steam drain isolation valves in the R/B.	3	R/B	C	Q	3	I	
Main steam piping downstream of main steam isolation valves and main steam bypass isolation valves up to and including the first restraint at the interface between the R/B and the T/B	3	R/B	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 18 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Main steam safety valves discharge piping in the R/B	3	R/B	C	Q	3	I	
Main steam piping downstream of the main steam relief valves and main steam depressurization valves in the R/B.	3	R/B	C	Q	3	I	
Main steam safety valves discharge piping out of the R/B	4	O/B	D	A	4	II	Note 5.a
MSS piping downstream of the main steam relief valves and main steam depressurization valves out of the R/B	4	O/B	D	A	4	II	Note 5.a
Turbine driven emergency feedwater pump steam supply piping drain piping and valves downstream and excluding EFS-VLV-109A,B,C,D	3	R/B	C	Q	3	I	
Main steam piping from the restraint at the interface between the R/B and the T/B to the turbine	4	T/B	D	A	4	NS	Note 5.d
Main steam equalization piping	8	T/B	D	N	4	NS	
Main steam drain piping and valves in the turbine building	8	T/B	D	N	4	NS	
Nitrogen supply piping and valves up to and excluding VLV-531A,B,C,D	10	PCCV	N/A	N	5	NS	
Turbine bypass valves MSS-TCV-550A,B,C,D,E,F,G,H,J,K,L,M,N,P,Q	4	T/B	D	A	4	NS	Note 5.d
9. Containment Spray System (CSS)							
Spray nozzles	2	PCCV	B	Q	2	I	
Containment spray system piping and valves	2	PCCV R/B	B	Q	2	I	
10. Post Accident pH Control System(PHS)							
NaTB baskets	2	PCCV	N/A	Q	5	I	
NaTB basket containers	2	PCCV	B	Q	2	I	
NaTB solution transfer piping	2	PCCV	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 19 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
11. Component Cooling Water System (CCWS)							
Component cooling water pumps	3	R/B	C	Q	3	I	
Component cooling water surge tanks	3	R/B	C	Q	3	I	
Component cooling water heat exchangers	3	R/B	C	Q	3	I	
Component cooling water supply/ return headers A, B ,A1 and A2 piping and valves excluding the following; Component cooling water system containment isolation valves and piping between these valves(1) Component cooling water supply/ return header A2 piping, in between but excluding the valves NCS-AOV-058A and NCS-VLV-034A(2)	3	R/B	C	Q	3	I	1. Component cooling water system containment isolation valves and piping between these valves are Equipment Class 2, Quality Group B, Seismic Category I. 2. Valves NCS-AOV-058A and NCS-VLV-034A are Equipment Class 3, Quality Group C.
Component cooling water supply/ return headers C, D, C1 and C2 piping and valves excluding the following; Component cooling water system containment isolation valves and piping between these valves(3) Component cooling water supply/ return header C2 piping, in between but excluding the valves NCS-AOV-058B and NCS-VLV-034B(4)	3	R/B	C	Q	3	I	3. Component cooling water system containment isolation valves and piping between these valves are Equipment Class 2, Quality Group B, Seismic Category I. 4. Valves NCS-AOV-058B and NCS-VLV-034B are Equipment Class 3, Quality Group C.

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 20 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Component cooling water supply/ return header A2 piping and valves between and excluding the valves NCS-AOV-058A and NCS-VLV-034A (excluding the valves), excluding the following; Component cooling water system containment isolation valves and piping between these valves(5) Component cooling water system piping and valves between these valves NCS-VLV-661A and NCS-VLV-669A (excluding the valves) Component cooling water system piping and valves between these valves NCS-VLV-601 and NCS-VLV-651 (excluding the valves)	4	R/B	D	A	4	II	Note 5.a 5. Component cooling water system containment isolation valves and piping between these valves are Equipment Class 2, Quality Group B, Seismic Category I.
Component cooling water supply/ return header C2 piping and valves between and excluding the valves NCS-AOV-058B and NCS-VLV-034B (excluding the valves), excluding the following; Component cooling water system containment isolation valves and piping between these valves(8) Component cooling water system piping and valves between these valves NCS-VLV-661B and NCS-VLV-669B (excluding the valves)	4	R/B	D	A	4	II	Note 5.a 6. Component cooling water system containment isolation valves and piping between these valves are Equipment Class 2, Quality Group B, Seismic Category I.
Component cooling water system piping and valves related to the excess letdown heat exchanger inside containment between and including the valves NCS-MOV-511,517, SRV-513	2	PCCV, R/B	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 21 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Component cooling water system piping and valves related to the letdown heat exchanger inside containment between and including the valves NCS-MOV-531,537, SRV-533	2	PCCV R/B	B	Q	2	I	
Component cooling water system piping and valves between and including the containment isolation valves NCS-MOV-402A,436A,438A and NCS-VLV-403A,437A	2	PCCV R/B	B	Q	2	I	
Component cooling water piping and valves between and including the containment isolation valves NCS-MOV-402B,436B,438B and NCS-VLV-403B,437B	2	PCCV R/B	B	Q	2	I	
Component cooling water system piping and valves related to components installed in reactor building and including the wall that separates R/B and A/B or T/B within followings: From and excluding stop valve NCS-VLV-601 up to and excluding stop valve NCS-VLV-651; From and excluding stop valve NCS-VLV-661A,B up to and excluding stop valve NCS-VLV-669A,B	4	R/B	D	A	4	II	Note 5.a
Component cooling water system piping and valves related to components installed in auxiliary building from stop valve NCS-VLV-601 up to stop valve NCS-VLV-651	8	A/B	D	N	4	NS	
Component cooling water system piping and valves related to components installed in turbine building from stop valves NCS-VLV-661A,B up to stop valves NCS-VLV-669A,B	8	T/B	D	N	4	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 22 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Component cooling water system piping and valves related to reactor coolant pumps between the containment isolation valves NCS-MOV-436A,447A (excluding) and NCS-VLV-403A,437A (excluding) and the valves NCS-SRV-406A,B,435A (including)	3	PCCV	C	Q	3	I	
Component cooling water system piping and valves related to reactor coolant pumps between the containment isolation valves NCS-MOV-436B,447B (excluding) and NCS-VLV-403B,437B (excluding) and the valves NCS-SRV-406C,D,435B (including)	3	PCCV	C	Q	3	I	
Component cooling water system Piping from component cooling water surge tank to and including the valve(NCS-SRV-003A,NCS-RCV-056A,NCS-PCV-012,NCS-VLV-047A)	3	R/B	C	Q	3	I	
Component cooling water system Piping from component cooling water surge tank to and including the valve(NCS-SRV-003B,NCS-RCV-056B,NCS-PCV-022,NCS-VLV-045B,NCS-VLV-047B)	3	R/B	C	Q	3	I	
Component cooling water surge tank surge line piping	3	R/B	C	Q	3	I	
Makeup line piping and valves from and including the valves NCS-VLV-051A,B and 054A,B up to and excluding the valves NCS-VLV-053A,B	5	R/B	N/A	A	5	II	Note 5.a
Makeup line piping and valves up to and excluding the valves NCS-VLV-063A,B	7	R/B	N/A	A	5	Note 2	Note 5.c

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 23 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Makeup line piping and valves from and including the valves NCS-VLV-053A,B and NCS-VLV-063A,B	3	R/B	C	Q	3	I	
Nitrogen gas supply line piping and valves from and including the valves NCS-VLV-041A,B up to and excluding the valves NCS-PCV-012,022 and NCS-VLV-045B	5	R/B	N/A	A	5	NS	Note 5.d
Chemical addition line piping and valves up to and excluding the valves NCS-VLV-047A,B	10	R/B	N/A	N	5	NS	
Component cooling water system piping from alternative component cooling water supply/ return headers A1 to and including the valve(NCS-MOV-321A, 323A, 325A, 326A)	3	R/B	C	Q	3	I	
Component cooling water system piping from alternative component cooling water supply/ return headers C1 to and including the valve (NCS-MOV-241, 242, 321B, 323B, 325B, 326B)	3	R/B	C	Q	3	I	
12. Spent Fuel Pit Cooling and Purification System (SFPCS)							
Spent fuel pit pumps	3	R/B	C	Q	3	I	
Spent fuel pit heat exchangers	3	R/B	C	Q	3	I	
Spent fuel pit filters	8	A/B	D	N	4	NS	
Spent fuel pit strainers	3	A/B	C	Q	5	I	
Spent fuel pit demineralizers	8	A/B	D	N	4	NS	
Spent fuel pit cooling piping and valves up to and including the following valves: Primary makeup line isolation valve SFS-MOV-028, 029 and SFS-VLV-030,031,032 Purification line isolation valves SFS-MOV-002A,B	3	R/B	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 24 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Spent fuel pit purification piping and valves inside A/B excluding the wall that separates R/B and A/B	8	A/B	D	N	4	NS	
Spent fuel pit purification piping and valves from but excluding valve SFS-MOV-002A,B and SFS-VLV-103A,B up to and including the wall that separates R/B and A/B	4	R/B	D	A	4	II	Note 5.a
Spent fuel pit purification piping and valves from and including the wall that separates R/B and A/B up to but excluding valve SFS-VLV-133A,B	4	R/B	D	A	4	II	Note 5.a
Spent fuel pit cooling piping and valves from but excluding RHS-VLV-032A,D (return line from containment spray/ residual heat removal system)	3	R/B	C	Q	3	I	
Spent fuel pit cooling piping and valves up to but excluding RHS-VLV-033A,D (towards containment spray/ residual heat removal system)	3	R/B	C	Q	3	I	
Water supply piping and valves from emergency feedwater pits	4	R/B	D	A	4	II	Note 5.a
SFP diverse makeup/spray line	10	R/B	N/A	N	5	NS	Designed to withstand the SSE
Water supply piping and valves from demineralized water storage tank up to and excluding SFS-VLV-026	10	R/B	N/A	N	5	NS	
Spent fuel pit purification line return piping to spent fuel pit from and including valves SFS-VLV-133A,B	3	R/B	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 25 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Spent fuel pit demineralizer piping and valves within the following boundaries: A,B-Main purification line-resin backwashing/primary makeup water line junction points; From and including valves SFS-VLV-141A,B; Up to and excluding valves SFS-VLV-165A,B; A,B-Demineralizer resin filling and discharge piping from and including valves SFS-VLV-152A,B up to and excluding valves SFS-VLV-153A,B; Demineralizer backwash piping from but excluding SFS-VLV-122A,B to top of demineralizer tank	8	A/B	D	N	4	NS	
13. Essential Service Water System (ESWS)							
Essential service water pumps	3	UHSRS	C	Q	3	I	
Essential service water pump discharge strainers, including essential service water supply piping and valves	3	UHSRS	C	Q	3	I	
Essential service water system strainer blowdown piping and valves and vent piping and valves	3	UHSRS	C	Q	3	I	
Essential service water supply header piping and valves	3	UHSRS ESWPT	C	Q	3	I	
Essential service water return header piping and valves	3	UHSRS ESWPT	C	Q	3	I	
Essential service water supply line piping and valves to component cooling water heat exchangers, including component cooling water heat exchanger backwash piping	3	R/B ESWPT	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 26 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Essential service water return line piping and valves from component cooling water heat exchangers	3	R/B ESWPT	C	Q	3	I	
Essential service water supply line piping and valves to essential chiller units	3	PS/B ESWPT	C	Q	3	I	
Essential service water return line piping and valves from essential chiller units	3	PS/B ESWPT	C	Q	3	I	
14. Gaseous Waste Management System (GWMS)							
Waste gas surge tanks	6	A/B	N/A	A	6	Note 1	Note 5.b
Charcoal beds	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste gas compressor packages	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste gas dryer skid	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste gas analyzer skid	6	A/B	N/A	A	6	Note 1	Note 5.b
Oxygen gas analyzer skid	6	A/B	N/A	A	6	Note 1	Note 5.b
Gaseous waste management system piping and valves in the system up to and including the first valve interfacing with a system of equipment class 8 or 9	6	A/B	N/A	A	6	Note1	Note 5.b
Piping and valves in the system up to but not including the first valve interfacing with a system of a higher classification	6	R/B A/B	N/A	A	6	Note1	Note 5.b
15. Liquid Waste Management System (LWMS)							
Containment vessel reactor coolant drain tank pumps	6	PCCV	N/A	A	6	Note 1	Note 5.b
Containment vessel sump pumps	6	PCCV	N/A	A	6	Note 1	Note 5.b
Reactor building sump pumps	6	R/B	N/A	A	6	Note 1	Note 5.b
Auxiliary building sump pumps	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste holdup tank pumps	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste monitor tank pumps	6	A/B	N/A	A	6	Note 1	Note 5.b

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 27 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Detergent drain tank pump	6	A/B	N/A	A	6	Note 1	Note 5.b
Detergent drain monitor tank pump	6	A/B	N/A	A	6	Note 1	Note 5.b
Chemical drain tank pump	6	A/B	N/A	A	6	Note 1	Note 5.b
Auxiliary building equipment drain sump pump	6	A/B	N/A	A	6	Note 1	Note 5.b
Containment vessel reactor coolant drain tank	6	PCCV	N/A	A	6	Note 1	Note 5.b
Reactor building sump tank	6	R/B	N/A	A	6	Note 1	Note 5.b
Auxiliary building sump tank	6	A/B	N/A	A	6	Note 1	Note 5.b
Auxiliary building equipment drain sump tank	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste holdup tanks	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste monitor tanks	6	A/B	N/A	A	6	Note 1	Note 5.b
Detergent drain tank	6	A/B	N/A	A	6	Note 1	Note 5.b
Detergent drain monitor tank	6	A/B	N/A	A	6	Note 1	Note 5.b
Neutralizing agent measuring tank	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste effluent inlet filters	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste demineralizers	6	A/B	N/A	A	6	Note 1	Note 5.b
Activated carbon filter	6	A/B	N/A	A	6	Note 1	Note 5.b
Waste effluent outlet strainer	6	A/B	N/A	A	6	Note 1	Note 5.b
Detergent drain strainer	6	A/B	N/A	A	6	Note 1	Note 5.b
Liquid waste management system piping and valves in the system up to and including the first valve interfacing with a system of equipment class 8, 9, or 10	6	PCCV R/B A/B	N/A	A	6	Note 1	Note 5.b
Piping and valves in the system up to but not including the first valve interfacing with a system of a higher classification	6	PCCV, R/B A/B	N/A	A	6	Note 1	Note 5.b
Liquid waste management system containment isolation valves and the piping between the valves	2	PCCV, R/B	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 28 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Piping, valves between HVAC Air cooler in C/V and the containment sump through the stand pipe	8	PCCV	D	N	4	NS	The SSE will not result in a failure and interaction of these components that adversely affect the function of containment sump level and flow monitoring system addressed by RG 1.45.
16. Solid Waste Management System							
Spent resin storage tanks	6	A/B	N/A	A	6	Note 1	Note 5.b
Solid Waste Management System piping and valves in the system up to and including the first valve interfacing with a system of equipment class 8 or 9	6	A/B	N/A	A	6	Note 1	Note 5.b
Piping and valves in the Solid Waste Management System up to but not including the first valve interfacing with a system of a higher classification	6	A/B	N/A	A	6	Note 1	Note 5.b
17. Refueling Water Storage System							
Refueling water recirculation pumps	3	R/B	C	Q	3	I	
Refueling water storage pit	2	PCCV	N/A	Q	5	I	
Refueling water storage auxiliary tank	4	O/B	D	A	4	NS	Note 5.d
Refueling water recirculation pumps discharge piping and valves in the refueling water storage system excluding piping downstream of the valve RWS-VLV-021	3	R/B	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 29 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Piping including branch piping and valves in the refueling water storage system from the refueling water storage pit up to and including the outermost containment isolation valves RWS-MOV-004,AOV-022, and valves RWS-VLV-041,042,045,061,062,075	2	PCCV R/B	B	Q	2	I	
Refueling water recirculation pump suction piping from RWS-MOV-004(excluding) and from RWS-VLV-101(including) up to pumps	3	R/B	C	Q	3	I	
Reactor cavity overflow piping to the RWSP	2	PCCV	B	Q	2	I	
Header compartment overflow piping to the RWSP	2	PCCV	B	Q	2	I	
Refueling cavity drain piping	2	PCCV	B	Q	2	I	
Debris interceptor	2	PCCV	N/A	Q	5	I	
RWSP sparger	2	PCCV	B	Q	2	I	
Piping from the refueling water storage auxiliary tank up to and excluding the valve RWS-VLV-101	4	O/B R/B	D	A	4	NS / II	Note 5.d / Note 5.a
RWSP overflow piping to C/V drain pump room	2	PCCV	B	Q	2	I	
Piping and valves in the refueling water storage system except the foregoing piping and valves	4	PCCV R/B	D	A	4	NS / II	Note 5.d / Note 5.a
18. Compressed Air and Gas System							
Nitrogen gas supply line to the accumulator for ECCS piping and valves	9	R/B, A/B, T/B, PS/B	N/A	N	5	NS	
Nitrogen gas supply system related to B - Component Cooling Water surge tank pressurization lines	5	R/B, A/B, T/B, PS/B	N/A	A	5	NS	Note 5.d
Nitrogen gas supply system other than above piping and valves	9	R/B, A/B O/B	N/A	N	5	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 30 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Hydrogen gas supply system piping and valves	9	R/B, A/B PS/B, T/B	N/A	N	5	NS	
Instrument air compressors package	9	T/B	N/A	N	5	NS	
Instrument air supply system piping and valves	9	T/B, PS/B, A/B, O/B	N/A	N	5	NS	
Instrument air supply system containment isolation valves and piping between the valves	2	PCCV	B	Q	2	I	
Service air compressors package	9	T/B	N/A	N	5	NS	
Service air supply system piping and valves in NI except the containment penetration noted below	10	PS/B, R/B, A/B, O/B	N/A	N	5	NS	
Service air supply system containment isolation valves and piping between the valves	2	PCCV	B	Q	2	I	
Service air supply system piping and valves in TI	9	T/B	N/A	N	5	NS	
19. Primary Make-Up Water System							
Primary makeup water tank	8	O/B	D	N	4	NS	
Primary makeup water pump	8	A/B	D	N	4	NS	
Primary makeup water system valves and piping from and including PWS-VLV-005	8	R/B A/B O/B	D	N	4	NS	
Primary makeup water system from deaerated supply line up to but excluding PWS-VLV-005, and upstream and downstream of PWS-VLV-060	9	R/B A/B	N/A	N	5	NS	
Primary makeup water system valves and pipings from and including the wall that separates R/B and A/B on the downstream valves PWS-VLV-027A,B up to and excluding valves RCS-VLV-136, PSS-VLV-221, CVS-FCV-128 and PWS-AOV-045A,B	4	R/B	D	A	4	II	Note 5.a

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 31 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Primary makeup water system piping and valves from and including the wall that separates R/B and A/B on the downstream valves PWS-VLV-060 up to and excluding NCS-VLV-051A,B and VWS-VLV-266A,B,C,D	5	PS/B R/B	N/A	A	5	II	Note 5.a
20. Demineralized Water System							
Demineralized water system containment isolation valves and piping between the valves	2	PCCV R/B	B	Q	2	I	
Demineralized water system piping and valves from and including the wall that separates R/B and A/B up to but excluding valve DWS-VLV-004	5	R/B	N/A	A	5	II	Note 5.a
Demineralized water system piping and valves except noted above	10	PCCV A/B	N/A	N	5	NS	
21. Auxiliary Steam Supply System							
(Deleted)							
(Deleted)							
(Deleted)							
Auxiliary steam supply system drain monitor heat exchanger	8	A/B	D	N	4	NS	
Auxiliary steam supply system drain tank	8	A/B	D	N	4	NS	
Auxiliary steam supply system drain tank pump	8	A/B	D	N	4	NS	
(Deleted)							
(Deleted)							
Auxiliary steam supply system piping and valves	8	T/B, A/B O/B, PS/B	D	N	4	NS	
22. Steam Generator Blowdown System							
System components	6	T/B, R/B A/B	N/A	A	6	Note 1	Note 5.b

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 32 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Steam generator Blowdown system piping and valves from steam generators up to and including the first containment isolation valves, on the outboard side of containment	2	PCCV R/B	B	Q	2	I	
Steam generator Blowdown system piping and valves from the outlet of the first containment isolation valves up to and including pipe anchors located in the main steam piping room wall	3	R/B	C	Q	3	I	
Steam generator Blowdown system piping and valves in the reactor building, auxiliary building, and turbine building	6	R/B, A/B, T/B	N/A	A	6	Note 1	Note 5.b
23. Fire protection water supply System							
Fire protection water supply system containment isolation valves and piping between the valves.	2	PCCV R/B	B	Q	2	I	
Fire protection water supply system piping and valves except the containment penetration noted above	7	PCCV R/B,A/B AC/B,PS/B T/B	N/A	A	7	Note 2	Note 5.c
24. Process and Post-accident Sampling System							
Sample heat exchanger –tube side	4	R/B	D	A	4	II	Note 5.a
Sample heat exchanger– component cooling water side	3	R/B	C	Q	3	I	
Accumulator sampling piping and valves from accumulator up to and including the outermost containment isolation valve	2	PCCV R/B	B	Q	2	I	
Hot leg sampling piping and valves from hot leg up to and including the outermost containment isolation valve	2	PCCV R/B	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 33 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Pressurizer liquid sampling piping and valves from hot leg up to and including the outermost containment isolation valve	2	PCCV R/B	B	Q	2	I	
Containment isolation valves PSS-MOV-071and 072 and piping between them	2	PCCV R/B	B	Q	2	I	
RHS loop sampling piping and valves up to and including the valves PSS-MOV-052A,B,C,D	2	R/B	B	Q	2	I	
Containment vessel atmosphere gas sample cooler	8	R/B	D	N	4	NS	
Containment vessel atmosphere gas sample moisture separator	8	R/B	D	N	4	NS	
Containment vessel atmosphere gas sample cooler-component cooling water side	3	R/B	C	Q	3	I	
Containment vessel atmosphere gas sampling hood	8	R/B	D	N	4	NS	
Containment vessel atmosphere gas sampling compressor	8	R/B	D	N/A	4	NS	
Containment vessel atmosphere sampling inlet, outlet valve PSS-MOV-301, 312	4	R/B	D	N/A	4	I	The SSE will not result in a failure and interaction of these components that adversely affect the function of Containment Airborne Particulate Radioactivity Monitor addressed by RG 1.45.

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 34 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Process and post-accident sampling systems piping and valves within following boundaries: From and excluding PSS-MOV-031A,B up to and including the wall that separates R/B and A/B, PSS-VLV-204, 207, 209 and 221; From and excluding PSS-MOV-052A through D up to A,B-Sample heat exchanger	4	R/B	D	A	4	II	Note 5.a
Process and post-accident sampling systems piping and valves not specifically described above (excluding PSS-MOV-301, 312)	8	R/B A/B,AC/B	D	N	4	NS	
Sample hood and sample panel	8	AC/B	D	N	4	NS	
Post accident liquid sample hood	8	R/B	D	N	4	NS	
25. Equipment and Floor drainage System							
Drain piping, valves and sump in the containment	8	PCCV	D	N/A	4	NS	The SSE will not result in a failure and interaction of these components that adversely affect the function of containment sump level and flow monitoring system addressed by RG 1.45.
Drain piping, valves in radiological controlled area	8	R/B A/B AC/B	D	N/A	4	NS	
Piping and valves in the non-radioactive sump pump discharge line between and including the wall that separates R/B and Adjusting Span and 1st check valves from the wall	4	R/B	D	A	4	II	Note 5.a

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 35 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Isolation valve located in the R/B in the non-radioactive drainage line from the A/B to the non-radioactive sump	4	R/B	D	A	4	II	Note 5.a
Drain piping from and including the wall that separates R/B and A/B up to and excluding the valve noted above	5	R/B	N/A	A	5	II	Note 5.a
Drain piping, valves, reactor building non-radioactive sump and sump pump in NRCA of reactor building except the line noted above	8	R/B	D	N	4	NS	
Drain piping, valves, turbine building sump and sump pump in turbine building	8	T/B	D	N	4	NS	
Drain piping, valves in auxiliary building and access control building, except for RCA	10	A/B,AC/B	N/A	N	5	NS	
Drain piping, valves in power source building	10	PS/B	N/A	N	5	NS	
Drain piping valves related to ESF rooms drain isolation FDS-VLV-001A,B,C,D	3	R/B	C	Q	3	I	
26. Potable and Sanitary Water System							
Potable and Sanitary Water System pipings and valves within R/B including the wall that separates R/B and other buildings	5	R/B	N/A	A	5	II	Note 5.a
Potable and Sanitary Water System components, piping and valves except noted above	10	A/B,AC/B T/B	N/A	N	5	NS	
27. Emergency Gas Turbine Auxiliary System							
Emergency Gas Turbine	3	PS/B	N/A	Q	5	I	
Fuel oil storage tanks	3	PSFSV	C	Q	3	I	
Fuel oil transfer pumps	3	PSFSV	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 36 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Fuel oil transfer pump suction lines from EPS fuel oil storage tank to EPS fuel oil transfer pumps	3	PSFSV	C	Q	3	I	
Fuel oil transfer pump suction line outlet check valves	3	PSFSV	C	Q	3	I	
Fuel oil transfer pump suction line isolation valves	3	PSFSV	C	Q	3	I	
Fuel oil day tanks	3	PS/B	C	Q	3	I	
Fuel oil transfer pump discharge lines up to EPS fuel oil day tank	3	PSFSV	C	Q	3	I	
Fuel oil transfer pump discharge line check valves	3	PSFSV	C	Q	3	I	
Fuel oil transfer pump discharge line isolation valves	3	PSFSV	C	Q	3	I	
Fuel oil day tank outlet lines up to EPS	3	PS/B	C	Q	3	I	
Fuel oil day tank outlet valves	3	PS/B	C	Q	3	I	
Air receivers	3	PS/B	C	Q	3	I	
Starting system air Compressor	5	PS/B	N/A	A	5	II	Note 5.a
Starting system air receiver inlet check valves and outlet lines up to air receivers	3	PS/B	C	Q	3	I	
Starting system air receiver relief valves	3	PS/B	C	Q	3	I	
Starting system cross-tie lines to air receivers	3	PS/B	C	Q	3	I	
Starting system air receiver outlet lines up to starting valve unit	3	PS/B	C	Q	3	I	
Starting system starting valves unit piping and valves	3	PS/B	C	Q	3	I	
Starting system starting valve unit outlet lines up to generator set enclosure	3	PS/B	C	Q	3	I	
Lube oil main oil pumps	3	PS/B	N/A	Q	5	I	
Lube oil cooler	3	PS/B	N/A	Q	5	I	
Lube oil reduction gear reservoir	3	PS/B	N/A	Q	5	I	
Main oil filters	3	PS/B	N/A	Q	5	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 37 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Main lube oil strainers	3	PS/B	N/A	Q	5	I	
Piping, fittings and valves	3	PS/B	N/A	Q	5	I	
Ventilation and cooling equipment	3	PS/B	N/A	Q	5	I	
Combustion air intake equipment and ductwork, turbine exhaust	3	PS/B	N/A	Q	5	I	
Combustion air intake and exhaust system intake silencers	3	PS/B	N/A	Q	5	I	
Combustion air intake and exhaust system turbine exhaust silencers	3	PS/B	N/A	Q	5	I	
Combustion air intake and exhaust system piping	3	PS/B	N/A	Q	5	I	
GTG Room ventilation system supply side equipment and ductwork and exhaust side equipment and ductwork	3	PS/B	N/A	Q	5	I	
Piping and valves (Safety related portion: Off skid)	3	PS/B, PSFSV	C	Q	3	I	
Piping and valves (Safety related portion: On skid)	3	PS/B	N/A	Q	5	I	
PSFSV Ventilation system containing exhaust fan, backdraft dampers, in-duct electric heater and ductwork	5	PSFSV	N/A	A	5	II	Note 5.a
28. Alternate alternating current Gas Turbine System							
Alternate alternating current Gas Turbine	5	PS/B	N/A	A	5	NS	Note 5.d
Fuel oil storage tanks	5	PSFSV	N/A	A	5	NS	Note 5.d
Fuel oil transfer pumps	5	PSFSV	N/A	A	5	NS	Note 5.d
Fuel oil day tanks	5	PS/B	N/A	A	5	NS	Note 5.d
Lube oil main oil pumps	5	PS/B	N/A	A	5	NS	Note 5.d
Lube oil cooler	5	PS/B	N/A	A	5	NS	Note 5.d
Lube oil reservoir	5	PS/B	N/A	A	5	NS	Note 5.d
Ventilation and cooling equipment	5	PS/B	N/A	A	5	NS	Note 5.d
Combustion air intake equipment and ductwork, turbine exhaust	5	PS/B	N/A	A	5	NS	Note 5.d

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 38 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
GTG Room ventilation system supply side equipment and ductwork and exhaust side equipment and ductwork	5	PS/B	N/A	A	5	NS	Note 5.d
Piping and valves (Off skid)	5	PS/B, PSFSV	N/A	A	5	NS	Note 5.d
Piping and valves (On skid)	5	PS/B	N/A	A	5	NS	Note 5.d
PSFSV Ventilation system containing exhaust fan, backdraft dampers, in-duct electric heater and ductwork	5	PSFSV	N/A	A	5	NS	Note 5.d
29. Fuel Handling and Refueling System							
Refueling machine	5	PCCV	N/A	A	5	II	Note 5.a
Fuel handling machine	5	R/B	N/A	A	5	II	Note 5.a
Spent fuel assembly handling tool	10	R/B	N/A	N	5	NS	
New fuel storage rack	3	R/B	N/A	Q	5	I	
Spent fuel storage rack	3	R/B	N/A	Q	5	I	
Fuel transfer tube	2	PCCV, R/B	B	Q	2	I	
Spent fuel pit	3	R/B	N/A	Q	5	I	
New fuel storage pit	3	R/B	N/A	Q	5	I	
Refueling canal	3	R/B	N/A	Q	5	I	
Cask pit	3	R/B	N/A	Q	5	I	
Cask washdown pit	3	R/B	N/A	Q	5	I	
Spent fuel pit gates	3	R/B	N/A	Q	5	I	
Fuel inspection pit	3	R/B	N/A	Q	5	I	
Fuel transfer system	5	PCCV, R/B	N/A	A	5	II	Note 5.a
Suspension hoist and aux. hoist on spent fuel cask handling crane	5	R/B	N/A	A	5	II	Note 5.a
New fuel elevator	5	R/B	N/A	A	5	II	Note 5.a
Containment rack	3	PCCV	N/A	Q	5	I	
New fuel assembly handling tool	10	R/B	N/A	N	5	NS	
Rod control cluster handling tool	10	R/B	N/A	N	5	NS	
Thimble plug handling tool	10	R/B	N/A	N	5	NS	
Burnable poison rod assembly handling tool	10	R/B	N/A	N	5	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 39 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Control rod drive shaft handling tool	10	PCCV	N/A	N	5	NS	
Permanent cavity seal	3	PCCV	N/A	Q	5	I	
30. Containment System							
Containment vessel	2	PCCV	N/A	Q	ASME III, MC	I	
Equipment hatch	2	PCCV	N/A	Q	ASME III, MC	I	
Personnel airlock	2	PCCV	N/A	Q	ASME III, MC	I	
31. Miscellaneous Plant Equipment							
PCCV polar crane	5	PCCV	N/A	A	5	II	The main and auxiliary hoist of the polar crane, and the main hoist of the spent fuel cask handling crane are designed in accordance with ASME NOG-1 and NUREG-0554 as Type I single-failure-proof cranes.
Spent fuel cask handling crane	5	R/B	N/A	A	5	II	
Equipment hatch hoist	5	PCCV	N/A	A	5	II	Note 5.a
Miscellaneous cranes and hoists in reactor building	5 or 10	R/B	N/A	A or N	5	II or NS	Note 5.a
Miscellaneous hoists in power source buildings	5	PS/B	N/A	A	5	II	Note 5.a
Crane for SWDS in auxiliary building	10	A/B	N/A	N	5	NS	
32. Containment Purge System							
Containment high volume purge air handling unit	10	R/B	N/A	N	5	NS	
Containment high volume purge air handling unit fan	10	R/B	N/A	N	5	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 40 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Containment high volume purge air handling unit cooling coil	10	R/B	N/A	N	5	NS	
Containment high volume purge air handling unit electric heating coil	10	R/B	N/A	N	5	NS	
Containment high volume purge exhaust filtration unit	10	A/B	N/A	N	5	NS	
Containment high volume purge exhaust filtration unit fan	10	A/B	N/A	N	5	NS	
Containment high volume purge exhaust filtration unit high-efficiency particulate air filter	10	A/B	N/A	N	5	NS	
Containment low volume purge air handling units	9	R/B	N/A	N	5	NS	
Containment low volume purge air handling unit fans	9	R/B	N/A	N	5	NS	
Containment low volume purge air handling unit cooling coils	9	R/B	N/A	N	5	NS	
Containment low volume purge air handling unit electric heating coils	9	R/B	N/A	N	5	NS	
Containment low volume purge exhaust filtration units	9	A/B	N/A	N	5	NS	
Containment low volume purge exhaust filtration unit fans	9	A/B	N/A	N	5	NS	
Containment low volume purge exhaust filtration unit high-efficiency particulate air filters	9	A/B	N/A	N	5	NS	
Containment low volume purge exhaust filtration unit charcoal adsorbers	9	A/B	N/A	N	5	NS	
Containment penetration piping	2	PCCV R/B	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 41 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Ductwork and dampers of the containment high volume purge system (supply)	5 or 10	PCCV R/B	N/A	A or N	5	II or NS	Ductwork and dampers, including supports, in areas containing safety-related equipment are seismic category II. Note 5.a
Ductwork and dampers of the containment high volume purge system (exhaust)	5 or 10	PCCV R/B A/B	N/A	A or N	5	II or NS	Ductwork and dampers, including supports, in areas containing safety-related equipment are seismic category II. Note 5.a
Ductwork and dampers of the containment low volume purge system (supply)	5 or 9	PCCV R/B	N/A	A or N	5	II or NS	Ductwork and dampers, including supports, in areas containing safety-related equipment are seismic category II. Note 5.a
Ductwork and dampers of the containment low volume purge system (exhaust)	5 or 9	PCCV R/B A/B	N/A	A or N	5	II or NS	Ductwork and dampers, including supports, in areas containing safety-related equipment are seismic category II. Note 5.a
Containment isolation valves	2	PCCV R/B	B	Q	2	I	
(Deleted)							
33. Containment Fan Cooler System							
Containment fan cooler unit fans	5	PCCV	N/A	A	5	II	Note 5.a
Containment fan cooler unit	5	PCCV	N/A	A	5	II	Note 5.a
Containment fan cooler unit cooling coils	5	PCCV	N/A	A	5	II	Note 5.a

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 42 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Ductwork	5	PCCV	N/A	A	5	II	Note 5.a
Dampers	5	PCCV	N/A	A	5	II	Note 5.a
34. Control Rod Drive Mechanism Cooling System							
Control rod drive mechanism cooling unit fans	5	PCCV	N/A	A	5	II	Note 5.a
Control rod drive mechanism cooling unit	5	PCCV	N/A	A	5	II	Note 5.a
Control rod drive mechanism cooling unit cooling coils	5	PCCV	N/A	A	5	II	Note 5.a
Dampers	5	PCCV	N/A	A	5	II	Note 5.a
Ductwork	5	PCCV	N/A	A	5	II	Note 5.a
35. Reactor Cavity Cooling System							
Reactor cavity cooling fans	5	PCCV	N/A	A	5	II	Note 5.a
Dampers	5	PCCV	N/A	A	5	II	Note 5.a
Ductwork	5	PCCV	N/A	A	5	II	Note 5.a
36. Annulus Emergency Exhaust System							
Annulus emergency exhaust filtration units	2	R/B	N/A	Q	5	I	
Annulus emergency exhaust filtration unit fans	2	R/B	N/A	Q	5	I	
Annulus emergency exhaust filtration unit high-efficiency particulate air filters	2	R/B	N/A	Q	5	I	
Dampers	2	R/B	N/A	Q	5	I	
Ductwork	2	R/B	N/A	Q	5	I	
37. MCR Heating, Ventilation, and Air Conditioning System							
Main control room air handling units	3	R/B	N/A	Q	5	I	
Main control room air handling unit fans	3	R/B	N/A	Q	5	I	
Main control room air handling unit cooling coils	3	R/B	C	Q	3	I	
Main control room air handling unit electric heating coils	3	R/B	N/A	Q	5	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 43 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Main control room toilet/kitchen exhaust fans	5	R/B	N/A	A	5	II	Note 5.a
Main control room smoke purge fan	5	R/B	N/A	A	5	II	Note 5.a
Main control room emergency filtration units	3	R/B	N/A	Q	5	I	
Main control room emergency filtration unit fans	3	R/B	N/A	Q	5	I	
Main control room emergency filtration unit electric heating coils	3	R/B	N/A	Q	5	I	
Main control room emergency filtration unit high efficiency particulate air filters	3	R/B	N/A	Q	5	I	
Main control room emergency filtration unit charcoal adsorbers	3	R/B	N/A	Q	5	I	
Ductwork and dampers excluding the following: - The smoke purge ductwork between VRS-AOD-132 and VRS-OTD-133 - The exhaust ductwork and backdraft dampers between VRS-AOD-122 and VRS-OTD-124	3	R/B	N/A	Q	5	I	
(Deleted)							
Duct heater	5	R/B	N/A	A	5	II	Note 5.a
Humidifier	5	R/B	N/A	A	5	II	Note 5.a
The exhaust ductwork and dampers VRS-OTD-123A,B between VRS-AOD-122 and VRS-OTD-124	5	R/B	N/A	A	5	II	Note 5.a
The smoke purge ductwork between VRS-AOD-132 and VRS-OTD-133	5	R/B	N/A	A	5	II	Note 5.a
38. Class 1E Electrical Room Heating, Ventilation, and Air Conditioning System							
Class 1E electrical room air handling units	3	R/B	N/A	Q	5	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 44 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Class 1E electrical room air handling unit fans	3	R/B	N/A	Q	5	I	
Class 1E electrical room air handling unit cooling coils	3	R/B	C	Q	3	I	
Class 1E electrical room air handling unit electric heating coils	3	R/B	N/A	Q	5	I	
Class 1E electrical room return air fans	3	R/B	N/A	Q	5	I	
Class 1E battery room exhaust fans	3	PS/B	N/A	Q	5	I	
Dampers	3	R/B PS/B	N/A	Q	5	I	
Ductwork	3	R/B PS/B	N/A	Q	5	I	
Duct heaters	3	R/B PS/B	N/A	Q	5	I	
(Deleted)							
39. <u>Safeguard Component Area Heating, Ventilation, and Air Conditioning System</u>							
Safeguard component area air handling units	3	R/B	N/A	Q	5	I	
Safeguard component area air handling unit fans	3	R/B	N/A	Q	5	I	
Safeguard component area air handling unit cooling coils	3	R/B	C	Q	3	I	
Safeguard component area air handling unit electric heating coils	3	R/B	N/A	Q	5	I	
Dampers	3	R/B	N/A	Q	5	I	
Ductwork	3	R/B	N/A	Q	5	I	
40. <u>Emergency Feedwater Pump Area Heating, Ventilation, and Air Conditioning System</u>							
Emergency feedwater pump area air handling units	3	R/B	N/A	Q	5	I	
Emergency feedwater pump area air handling unit fans	3	R/B	N/A	Q	5	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 45 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Emergency feedwater pump area air handling unit cooling coils	3	R/B	C	Q	3	I	
Emergency feedwater pump area air handling unit electric heating coils	3	R/B	N/A	Q	5	I	
Dampers VRS-OTD-403A, D and 404A, D	3	R/B	N/A	Q	5	I	
Dampers	5	R/B	N/A	A	5	II	Note 5.a
Ductwork	3	R/B	N/A	Q	5	I	
Ductwork	5	R/B	N/A	A	5	II	Note 5.a
41. Safety-related Component Area Heating, Ventilation, and Air Conditioning system							
Penetration area air handling units	3	R/B	N/A	Q	5	I	
(Deleted)							
Penetration area air handling unit fans	3	R/B	N/A	Q	5	I	
Penetration area air handling unit cooling coils	3	R/B	C	Q	3	I	
Penetration area air handling unit electric heating coils	3	R/B	N/A	Q	5	I	
Annulus emergency exhaust filtration unit area air handling units	3	R/B	N/A	Q	5	I	
Annulus emergency exhaust filtration unit area air handling unit fans	3	R/B	N/A	Q	5	I	
Annulus emergency exhaust filtration unit area air handling unit cooling coils	3	R/B	C	Q	3	I	
Annulus emergency exhaust filtration unit area air handling unit electric heating coils	3	R/B	N/A	Q	5	I	
Charging pump area air handling units	3	R/B	N/A	Q	5	I	
Charging pump area air handling unit fans	3	R/B	N/A	Q	5	I	
Charging pump area air handling unit cooling coils	3	R/B	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 46 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Charging pump area air handling unit electric heating coils	3	R/B	N/A	Q	5	I	
Component cooling water pump area air handling units	3	R/B	N/A	Q	5	I	
Component cooling water pump area air handling unit fans	3	R/B	N/A	Q	5	I	
Component cooling water pump area air handling unit cooling coils	3	R/B	C	Q	3	I	
Component cooling water pump area air handling unit electric heating coils	3	R/B	N/A	Q	5	I	
Essential chiller unit area air handling units	3	PS/B	N/A	Q	5	I	
Essential chiller unit area air handling unit fans	3	PS/B	N/A	Q	5	I	
Essential chiller unit area air handling unit cooling coils	3	PS/B	C	Q	3	I	
Essential chiller unit area air handling unit electric heating coils	3	PS/B	N/A	Q	5	I	
Spent fuel pit pump area air handling units	3	R/B	N/A	Q	5	I	
Spent fuel pit pump area air handling unit fans	3	R/B	N/A	Q	5	I	
Spent fuel pit pump area air handling unit cooling coils	3	R/B	C	Q	3	I	
Spent fuel pit pump area air handling unit electric heating coils	3	R/B	N/A	Q	5	I	
Ductwork and dampers	3	R/B, PS/B	N/A	Q	5	I	
42. Main Steam/Feedwater Piping Area Heating, Ventilation, and Air Conditioning System							
(Deleted)							
Main steam/feedwater piping area air handling units	9	R/B	N/A	N	5	NS	
Main steam/feedwater piping area air handling unit fans	9	R/B	N/A	N	5	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 47 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Main steam/feedwater piping area air handling unit cooling coils	9	R/B	N/A	N	5	NS	
Main steam/feedwater piping area air handling unit electric heating coils	9	R/B	N/A	N	5	NS	
Dampers	5 or 9	R/B	N/A	A or N	5	II or NS	Ductwork and dampers, including supports, in areas containing safety-related equipment are Seismic Category II. Note 5.a
Ductwork	5 or 9	R/B	N/A	A or N	5	II or NS	Ductwork and dampers, including supports, in areas containing safety-related equipment are Seismic Category II. Note 5.a
43. <u>Auxiliary Building Heating, Ventilation, and Air Conditioning System</u>							
Auxiliary building air handling units	9	A/B	N/A	N	5	NS	
Auxiliary building air handling unit fans	9	A/B	N/A	N	5	NS	
Auxiliary building air handling unit cooling coils	9	A/B	N/A	N	5	NS	
Auxiliary building air handling unit heating coils	8	A/B	D	N	4	NS	
Auxiliary building exhaust fans	9	A/B	N/A	N	5	NS	
Penetration and Safeguard Component area isolation dampers and ductwork between Penetration and Safeguard Component area isolation damper	2	R/B	N/A	Q	5	I	
Exhaust line isolation dampers	2	R/B	N/A	Q	5	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 48 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Supply ductwork and dampers of the auxiliary building HVAC system	5 or 9	R/B, PS/B A/B, AC/B	N/A	A or N	5	II or NS	Ductwork and dampers, including supports, in areas containing safety-related equipment are seismic category II. Note 5.a
Exhaust ductwork and dampers of the auxiliary building HVAC system	5 or 9	R/B, PS/B A/B, AC/B	N/A	A or N	5	II or NS	Ductwork and dampers, including supports, in areas containing safety-related equipment are seismic category II. Note 5.a
44. Non-Class 1E Electrical Room HVAC System							
Non-Class 1E electrical room air handling units	9	A/B	N/A	N	5	NS	
Non-Class 1E electrical room air handling unit fans	9	A/B	N/A	N	5	NS	
Non-Class 1E electrical room air handling unit cooling coils	9	A/B	N/A	N	5	NS	
Non-Class 1E electrical room air handling unit heating coils	8	A/B	D	N	4	NS	
Non-Class 1E electrical room Return air fans	9	A/B	N/A	N	5	NS	
Non-Class 1E battery room exhaust fans	9	A/B	N/A	N	5	NS	
Ductwork and dampers with exception of the ductwork and dampers associated with the non-class 1E battery room exhaust fans	9	A/B	N/A	N	5	NS	
Ductwork and dampers associated with the non-class 1E battery room exhaust fans	9	A/B	N/A	N	5	NS	
Duct heaters	9	A/B	N/A	N	5	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 49 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
45. Technical Support Center Heating, Ventilation, and Air Conditioning System							
Technical support center air handling unit	10	A/B	N/A	N	5	NS	
Technical support center air handling unit fan	10	A/B	N/A	N	5	NS	
Technical support center air handling unit cooling coil	10	A/B	N/A	N	5	NS	
Technical support center air handling unit electric heating coil	10	A/B	N/A	N	5	NS	
Technical support center toilet /kitchen exhaust fan	10	AC/B	N/A	N	5	NS	
Technical support center emergency filtration unit	10	A/B	N/A	N	5	NS	
Technical support center emergency filtration unit fan	10	A/B	N/A	N	5	NS	
Technical support center emergency filtration unit electric heating coil	10	A/B	N/A	N	5	NS	
Technical support center emergency filtration unit high efficiency particulate air filter	10	A/B	N/A	N	5	NS	
Technical support center emergency filtration unit charcoal adsorber	10	A/B	N/A	N	5	NS	
Dampers	10	A/B AC/B	N/A	N	5	NS	
Ductwork	10	A/B AC/B	N/A	N	5	NS	
46. Essential Chilled Water System							
Essential chiller units Evaporator side Condenser side	 3 3	 PS/B PS/B	 C C	 Q Q	 3 3	 I I	
Essential chilled water pumps	3	PS/B	C	Q	3	I	
Essential chilled water compression tanks	3	PS/B	C	Q	3	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 50 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Essential chilled water chemical feed tank	5	PS/B	N/A	A	5	II	Note 5.a
Piping and valves within the distribution loop	3	R/B PS/B	C	Q	3	I	
Essential chilled water compression tank surge line piping	3	PS/B	C	Q	3	I	
Makeup line piping and valves from and including the valves VWS-VLV-262A,B,C,D and 266A,B,C,D up to and excluding the valves VWS-VLV-258A,B,C,D	5	PS/B	N/A	A	5	II	Note 5.a
Nitrogen gas supply line piping and valves from and including the valves VWS-VLV-251A,B,C,D up to and excluding the valves VWS-VLV-252A,B,C,D	5	PS/B	N/A	A	5	II	Note 5.a
Essential chilled water chemical feed tank supply and return line piping and between and excluding the valves VWS-VLV-271A,B,C,D and VWS-VLV-274A,B,C,D	5	PS/B	N/A	A	5	II	Note 5.a
Piping from essential chilled water compression tank to and including the valves VWS-SRV-253A,B,C,D and VWS-VLV-254A,B,C,D	3	PS/B	C	Q	3	I	
47. Non-Essential Chilled Water System							
Non-essential chiller units Evaporator side Condenser side	 9 9	 A/B A/B	 N/A N/A	 N N	 5 5	 NS NS	
Non-essential chilled water pumps	9	A/B	N/A	N	5	NS	
Non-essential chilled water compression tanks	9	A/B	N/A	N	5	NS	
Non-essential chilled water system cooling towers	9	A/B	N/A	N	5	NS	Designed to withstand the SSE

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 51 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
Non-essential chilled water system condenser water pumps	9	A/B	N/A	N	5	NS	Designed to withstand the SSE
Non-essential chilled water chemical feed tank	10	A/B	N/A	N	5	NS	
Piping and valves (except portion of the containment penetration)	9	A/B PS/B T/B	N/A	N	5	NS	
Piping and valves provided in the non-essential chiller unit coolant loop excluding supply and return line for non-essential chiller units	9	A/B	N/A	N	5	NS	Designed to withstand the SSE
Piping and valves (except portion of the containment penetration)	5	PCCV R/B PS/B	N/A	A	5	II	Note 5.a
(Deleted)							
Piping and valves between and including the containment isolation valves VWS-MOV-403 and 421, MOV-422 and 407, VLV-423	2	PCCV R/B	B	Q	2	I	
Non-essential chilled water chemical feed tank supply and return line piping and valves between and excluding the valve VWS-VLV-571 and VWS-VLV-574	10	A/B	N/A	N	5	NS	
48. Containment Hydrogen Control System							
Igniters	5	PCCV	N/A	A	5	II	Note 5.a
49. Radiation monitoring system							
Piping and valves between and including the containment isolation valves	2	PCCV R/B	B	Q	2	I	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 52 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification[(5)]	Codes and Standards[(3)]	Seismic Category[(4)]	Notes
Piping and valves of the radiation monitoring system	8	PCCV R/B	D	N	4	NS	The SSE will not result in a failure and interaction of these components that adversely affect the containment radiation monitoring function (or functionality of the leakage detection systems addressed by RG 1.45).
50. Condensate Storage and Transfer System							
Condensate storage tank	8	O/B	D	N	4	NS	
The components downstream condensate storage tank	8	O/B T/B	D	N	4	NS	
51. Turbine Component Cooling Water System							
TCS components	9	T/B	N/A	N	5	NS	
52. Non-Essential Service Water System							
Non ESW components	9	T/B	N/A	N	5	NS	
53. Secondary Sampling System (SSS)							
SSS components	8	T/B	D	N	4	NS	
54. Turbine Building Area Ventilation System							
TB HVAC components	9	T/B	N/A	N	5	NS	
55. Turbine Generator							
Main turbine	8	T/B	D	N	4	NS	
Moisture separator and reheaters	8	T/B	D	N	4	NS	
Generator	9	T/B	N/A	N	5	NS	
Other system components							
Containing secondary coolant	8	T/B	D	N	4	NS	
Not containing secondary coolant	9	T/B	N/A	N	5	NS	

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment (Sheet 53 of 53)

System and Components	Equipment Class	Location	Quality Group	Quality Assurance Classification(5)	Codes and Standards(3)	Seismic Category(4)	Notes
56. Main Steam Supply System (MSS)							
Main steam supply system components from downstream of the first restraint located between the reactor building and turbine building to first isolation valves excluding the valves	4	T/B	D	A	4	NS	Note 5.d
57. Main Condenser							
Main condensers	8	T/B	D	N	4	NS	
58. Main Condenser Evacuation System (MCES)							
MCES components	8	T/B	D	N	4	NS	
59. Gland Seal System (GSS)							
GSS components	8	T/B	D	N	4	NS	
60. Circulating Water System (CWS)							
CWS components	9	O/B T/B	N/A	N	5	NS	
61. Condensate Polishing System (CPS)							
Condensate polisher	8	T/B	D	N	4	NS	
Other system components							
Containing secondary coolant	8	T/B	D	N	4	NS	
Not containing secondary coolant	10	T/B	N/A	N	5	NS	
62. Condensate and Feedwater System (CFS)							
The system components up to the first piping restraint at the interface between the reactor building and the turbine building	8	T/B	D	N	4	NS	
63. Secondary side Chemical Injection System (SCIS)							
SCIS components	9	T/B	N/A	N	5	NS	

Notes:

1. Seismic category meeting Table 2 of RG 1.143 (Reference 3.2-10) is applied, in accordance with the SSC classifications described in Section 10.4.8, 11.2, 11.3, and 11.4. Portions of the Equipment Class 6 SSCs on which seismic category II requirements are imposed are designed to comply with both the requirements of RG 1.143 and seismic category II.

2. Seismic category meeting RG 1.189 (Reference 3.2-11) is applied. Portions of the Equipment Class 7 SSCs on which seismic category II requirements are imposed are designed to comply with both the requirements of RG 1.189 and seismic category II.
3. Identification number for “Code and Standards”
   (1) ASME Code, Section III, Class 1 (Reference 3.2-14)
   (2) ASME Code, Section III, Class 2 (Reference 3.2-14)
   (3) ASME Code, Section III, Class 3 (Reference 3.2-14)
   (4) RG 1.26 (Reference 3.2-13), Table 1, Quality Standards for Class D
   (5) Codes and standards as defined in design bases
   (6) Codes and standards, and guidelines provided in Table 1 of RG 1.143 (Reference 3.2-10), for design of SSCs for Radwaste Facility
   (7) The codes and standards applicable to fire protection systems follow the guidance of RG 1.189 Section 1.7, and National Fire Protection Association 804.
4. Seismic category: The designations “I” or “II” indicate that the design requirements of Seismic Category I or II equipment are applied as described in Subsection 3.2.1 and Section 3.7, Seismic Design. Equipment that is not designated “I” or “II” is designated “NS.”
5. Quality Assurance Classification: The designation “Q” indicates that the quality assurance requirements of 10 CFR 50, Appendix B, are applied in accordance with the quality assurance program described in Chapter 17. The designation “A” indicates that augmented quality assurance requirements are applied, commensurate with the SSCs contribution to safety or credited for regulatory events for one or more of the following reasons:
   a. Nonsafety-related equipment required to be designed in accordance with special seismic design requirements, such as seismic category II requirements. See note 4.
   b. Nonsafety-related equipment required to be designed in accordance with radioactive waste management system requirements from RG 1.143 for Category RW-IIa, RW-IIb, and RW-IIc [See note 3(6)]. The radioactive waste management system components conform to Regulatory Guide 1.143 Table 1 [see note 3(6)].
   c. Nonsafety-related equipment required to be designed in accordance with fire protection requirements from 10 CFR 50.48 and RG 1.189. A quality assurance program meets the guidance of RG 1.189.
   d. Nonsafety-related equipment not otherwise identified in notes 5(a) though 5(c) and are identified as risk-significant in Table 17.4-1 or credited for regulatory events such as ATWS and SBO.

The designation “N” indicates that neither 10CFR50 Appendix B nor augmented quality standards are required.

Table 3.2-3 Comparison of Various Requirements to Equipment Class

US-APWR Equipment Class	ASME Code, Section III (Reference 3.2-14), Class	RG1.29 (Reference 3.2-5) Seismic Category	RG1.26 (Reference 3.2-13) NRC Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)
1	1	I	A	YES[2]
2	2[1]	I	B[13]	YES[2]
3	3[1]	I	C[13]	YES[2]
4	N/A[3]	NS or II	D	N/A[4]
5	N/A[5]	NS or II	N/A	N/A[4]
6	N/A[6]	N/A[7]	N/A	N/A[8]
7	N/A[9]	N/A[10]	N/A	N/A[11]
8	N/A[12]	NS	D	N/A
9	N/A	NS	N/A	N/A
10	N/A	NS	N/A	N/A

Notes:

1. Items not covered by the ASME Code are designed to other applicable codes and standards.
2. “Yes” means QA Program is required according to 10 CFR 50, Appendix B (Reference 3.2-8).
3. Refer to Subsection 3.2.2.4.
4. Augment quality assurance controls are applied.
5. Code and standard as defined in design bases are applied.
6. Code and standard meeting RG 1.143 (Reference 3.2-10) are applied.
7. Seismic category meeting RG 1.143 (Reference 3.2-10) is applied. Portions of the Equipment Class 6 SSCs on which seismic category II requirements are imposed are designed to comply with both the requirements of RG 1.143 and seismic category II.
8. A QA program meeting RG 1.143 (Reference 3.2-10) is applied as augmented quality assurance controls.
9. Code and standard meeting RG 1.189 (Reference 3.2-11) is applied.
10. Seismic category meeting RG 1.189 (Reference 3.2-11) is applied. Portions of the Equipment Class 7 SSCs on which seismic category II requirements are imposed are designed to comply with both the requirements of RG 1.189 and seismic category II.
11. A QA Program meeting RG 1.189 (Reference 3.2-11) is applied as augmented quality assurance controls.
12. Refer to Subsection 3.2.2.5 for Equipment Class 8.
13. N/A for items not covered by RG 1.26 (Reference 3.2-13) Table 1.

Table 3.2-4 Seismic Classification of Buildings and Structures[1, 6]

Structure	Acronym	Seismic Category[2]
Reactor Building[3]	R/B	I
Prestressed Concrete Containment Vessel[3]	PCCV	I
Containment Internal Structure[3]		I
Power Source Building (East and West)[3]	PS/B	I
Power Source Fuel Storage Vault	PSFSV	I
Essential Service Water Pipe Tunnel (ESWPT) (from/to UHS)[5]	ESWPT	I
UHS Related Structures[4]	UHSRS	I
A/B[3]	A/B	II
Turbine Building	T/B	II
Plant Vent Stack		II
AC/B[3]	AC/B	NS
Outside Building (e.g., maintenance facility, operations office)	O/B	NS
Turbine generator pedestal	T/G Pedestal	NS

Notes:

1. Other non-standard plant building structures, such as minor NS buildings and structures in the plant yard, are not listed in the above table and are not considered part of the US-APWR Nuclear Island.
2. Seismic category I (I)
   Seismic category II (II)
   Non-Seismic (NS)
3. US-APWR Nuclear Island
4. UHSRS include but are not limited to (1) dams, (2) ponds, or (3) cooling towers (including cooling tower enclosure, and pump house). The specific features of the UHSRS are site dependent and not part of the US-APWR standard plant. The UHSRS are seismic category I structures selected based on site-specific conditions and site-specific meteorological data.
5. The ESWPT is a site-specific structure, but the existence and functions are required by the plant standard design. The specific features of the ESWPT are site dependent and will depend on the type of UHS.
6. The common walls between Seismic Category I and Seismic Category II structures are classified as Seismic Category I.

3.3 Wind, Tornado and Hurricane Loadings

3.3.1 Wind Loadings

For US-APWR, including site-specific seismic category I and II SSCs subject to wind loads, the design basis wind loadings are determined in accordance with American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI), "Minimum Design Loads for Buildings and Other Structures", ASCE/SEI 7-05 (Reference 3.3-1). However, load combinations involving wind or tornado as given in ASCE/SEI 7-05 are not used. Instead, load combinations as defined in applicable codes and as modified by the relevant NRC RGs and SRPs are used. Load combinations, load factors, allowable stresses, and acceptance criteria for US-APWR, including site-specific seismic category I and II SSCs are discussed in Section 3.8.

Extreme winds such as hurricanes and tornadoes also have the potential to generate missiles. Missiles generated by tornadoes and extreme winds are listed in Subsection 3.5.1.4 and barrier design for missiles is discussed in Subsection 3.5.3.

3.3.1.1 Severe Wind Velocity and Recurrence Interval

The severe wind has a basic speed of 155 mph, corresponding to a 3-second gust at 33 ft above ground for exposure category C (open terrain). For all seismic category I and II SSCs, the velocity pressure associated with the basic wind speed is multiplied by an importance factor of 1.15 correlating to essential facilities in hurricane-prone regions as defined in ASCE/SEI 7-05 Tables 1-1 and 6-1. The mean recurrence interval for the basic wind speed with associated importance factor of 1.15 is 100 years, which corresponds to an annual probability of exceedance of 0.01, as discussed in commentary Subsection C6.5.5 of ASCE/SEI 7-05 (Reference 3.3-1).

The basic wind speed described above envelopes the basic speed at almost all locations in the contiguous United States (US). A basic wind speed of 155 mph for exposure category C also envelopes all locations in the contiguous US that have the more severe exposure category D (flat, unobstructed areas and water surfaces), such as potential sites near open inland waterways and the Great Lakes. This is because in the contiguous US the exposure category D is associated with regions that are not prone to hurricanes and have basic wind speeds that are much lower, typically 90 mph or less. The COL Applicant is responsible for verifying the site-specific basic wind speed is enveloped by the determinations in this section.

3.3.1.2 Determination of Applied Forces

The applied wind loads are determined by converting the basic wind speed for exposure category C into design pressures or forces using method 2 (analytical procedure) from ASCE/SEI 7-05, in accordance with NUREG 0800, SRP 3.3.1 (Reference 3.3-2). The conversion is performed by determining a velocity pressure which is transformed into an effective design pressure/force by using applicable adjustment factors (including topographic, directionality, and/or gust effect factors), and velocity pressure/force coefficients in accordance with ASCE/SEI 7-05 and SRP 3.3.1 (References 3.3-1 and 3.3-2). When determining the resulting effective design wind pressures/forces, the influences of height and location on an SSC are captured by the adjustment factors and velocity force/pressure coefficients.

For method 2 with an importance factor of 1.15 (as discussed in Subsection 3.3.1.1), equation 6-15 from Subsection 6.5.10 of ASCE/SEI 7-05 is used where the topographic and directionality factors K_{zt} and K_d are each 1.0, (in accordance with SRP 3.3.1, Reference 3.3-2) and substituting into equation 6-17 of ASCE/SEI 7-05, Subsection 6.5.12 for enclosed and partially enclosed buildings, the basic formula for effective wind velocity pressure used for building main wind-force resisting systems is:

$$p = 0.00256\, K_z V^2\, 1.15\, (GC_p \text{ +/- } GC_{pi})$$

where

p = effective wind velocity pressure, psf

K_z = velocity pressure exposure coefficient varying with height, taken from Table 6-3 of ASCE/SEI 7-05 for exposure category C; however, not less than 0.87 as recommended by SRP 3.3.1 (Reference 3.3-2)

V = basic wind speed of 155 mph per Subsection 3.3.1.1

G = gust effect factor = 0.85 or as determined per ASCE/SEI 7-05, Subsection 6.5.8 (where a combined gust effect and pressure coefficient factor is used from a figure(s) in ASCE/SEI 7-05, an individual gust effect factor is not applied)

C_p = external pressure coefficient from ASCE/SEI 7-05 Subsection 6.5.11

C_{pi} = internal pressure coefficient from ASCE/SEI 7-05 Subsection 6.5.11 where two cases shall be considered to determine the critical load requirements for the appropriate conditions:

i. a positive value of GC_{pi} applied to all internal surfaces

ii. a negative value of GC_{pi} applied to all internal surfaces

Non-building structures and components and cladding are designed using effective wind velocity force and pressure formulae from ASCE/SEI 7-05 (Reference 3.3-1), consistent with those described above.

All US-APWR and site-specific structures and components subject to wind loads are designed using the same basic wind speed defined in Subsection 3.3.1.1. For certain non-seismic, nonsafety-related structures and components, an importance factor may be used that is less than that for seismic category I and II structures. Those structures and components that are designed with a lower importance factor are investigated to assure that their failure would impact neither the function nor integrity of adjacent safety-related

SSCs, nor result in the generation of missiles having more severe effects than those discussed in Subsection 3.5.1.4. Where required by the results of the investigation, structural reinforcement and/or missile barriers are implemented so as not to jeopardize safety-related SSCs due to failure effects from wind loads.

The COL Applicant is to provide the wind load design method and importance factor for site-specific seismic category I and seismic category II buildings and structures. The COL Applicant is to also verify that the site location does not have features promoting channeling effects or buffeting in the wake of upwind obstructions that invalidate the standard plant wind load design.

3.3.2 Tornado and Hurricane Loadings

The US-APWR standard and site-specific plant is designed to protect SSCs listed in the Appendix to US NRC RG 1.117, Revision 1, dated April 1978 (Reference 3.3-3). Accordingly, all seismic category I and II SSCs subject to tornado and hurricane winds are designed to meet the acceptance criteria discussed in Section 3.8. Local damage to and/or plastic deformation of seismic category I and II structures due to the impactive loads of tornado missiles and hurricane missiles is acceptable, provided that the integrity and function of any safety-related SSCs are not adversely impacted. For this condition, where seismic category I and II structures act as missile barriers, design procedures assure that sufficient thicknesses are provided to prevent missile penetration/perforation. For concrete structures, the barrier design also assures that there is no potential for generation of secondary missiles due to spalling or scabbing effects. Subsection 3.5.3 addresses barrier design procedures for analyzing local missile impact loads.

3.3.2.1 Applicable Design Parameters

The design basis tornado parameters are for a single Rankine combined vortex tornado and are as follows.

- Maximum wind speed = 230 mph (maximum rotational + maximum translational)
- Maximum rotational speed = 184 mph
- Maximum translational speed = 46 mph
- Radius of maximum rotational wind from center of tornado, R_m = 150 ft
- Atmospheric pressure drop = 1.2 psi
- Rate of pressure change = 0.5 psi/second

The parameters listed above are based on US NRC RG 1.76, Revision 1, dated March 2007 (Reference 3.3-4). The parameters are those of a region 1 tornado as defined therein, and envelope the tornadoes of all other regions in the contiguous US. The annual probability of exceedance of the design basis tornado described above is 10^{-7} as discussed in RG 1.76 and the corresponding recurrence interval is approximately ten million years.

The design-basis hurricane wind speed is chosen from wind speed contour maps for hurricane-prone regions of the contiguous United States presented in US NRC RG 1.221 (Reference 3.3-8). The wind speed due to hurricanes that is selected for design of the standard plant is 160 mph, corresponding to a 3-second gust at 33 ft above ground for exposure category C. Exposure category C, defined in Section 6.5.6.3 of ASCE/SEI 7-05 (Reference 3.3-1), is a typical exposure category for nuclear power plants and includes flat open country, grasslands, and all water surfaces in hurricane-prone regions. The hurricane wind speed has an exceedance frequency of 10^{-7} per year which corresponds to a mean recurrence interval of ten million years as discussed in RG 1.221. The design wind speed (tornado, or hurricane) for seismic category I and II SSCs is considered in conjunction with an importance factor of 1.15 in accordance with SRP 3.3.2 (Reference 3.3-5).

The design-basis hurricane wind speeds described above envelop the design-basis hurricane wind speeds at most locations in the contiguous US. The COL Applicant is responsible for verifying that the site-specific design-basis hurricane basic wind speeds, exposure category, and resulting wind forces are enveloped by the determinations in this section.

3.3.2.2 Determination of Forces on Structures

3.3.2.2.1 Tornado and Hurricane Velocity Forces

Tornado Velocity pressures are determined by converting tornado wind speeds into effective velocity pressures in accordance with procedures accepted by SRP 3.3.2 (Reference 3.3-5). Design tornado loads are determined for enclosed and partially enclosed buildings using the analytical procedure method 2 provided in Subsection 3.3.1.2, where:

K_z is the velocity pressure exposure coefficient = 0.87

V is the maximum tornado wind speed = 230 mph

For the design basis tornado, wind speed remains constant with respect to height; therefore, no adjustment for wind speed variation with respect to height applies.

Hurricane velocity pressures are determined by converting hurricane wind speeds into effective velocity pressures in accordance with procedures accepted by SRP 3.3.2 (Reference 3.3-5). Design hurricane loads are determined for enclosed and partially enclosed buildings using the analytical procedure method 1 or method 2 provided in Subsection 3.3.1.2, where:

V is the maximum hurricane windspeed = 160 mph

For the design basis hurricane, wind pressure varies with respect to height; therefore, adjustment for wind speed variation with respect to height applies.

The design load equation in Subsection 3.3.1.2 above is for enclosed and partially enclosed buildings per ASCE/SEI 7-05, Subsection 6.5.12. ASCE/SEI 7-05

(Reference 3.3-1) Subsections 6.5.13 to 6.5.15 are used for the determination of design loads for different structure types as applicable.

3.3.2.2.2 Tornado Atmospheric Forces

The tornado atmospheric pressure loading is computed using the maximum atmospheric pressure drop defined in Subsection 3.3.2.1, and the ability of the structure to reduce atmospheric pressure change by venting.

For a structure that is enclosed (unvented structure), the atmospheric pressure outside the structure changes during the passage of a tornado, while the internal pressure remains unchanged. The resulting outward differential pressure on the roof and exterior walls are applicable for all seismic category I unvented structures including the R/B (and its annulus which houses the containment penetration areas) and the PCCV.

For a structure that is partially enclosed or vented, the atmospheric pressure change occurs over a period of time, resulting in actual pressures less than or equal to the maximum pressure drop. This is the case for the PS/Bs, A/B, T/B, and AC/B, which are designed as vented structures. Where applicable, interior walls of the PS/Bs and A/B are designed considering tornado differential atmospheric pressure loading. The design of the T/B and AC/B are discussed further in Subsection 3.3.2.3.

The COL Applicant is to note the vented and unvented requirements of this subsection to the site-specific category I buildings and structures.

3.3.2.2.3 Tornado Missile and Hurricane Missile Effects

Missiles generated by tornadoes and hurricanes are listed in Subsection 3.5.1.4 and barrier design for missiles is discussed in Subsection 3.5.3. The response of a structure or barrier to missile impact depends largely on the location of impact (e.g., midspan of a slab or near a support), on the dynamic properties of the target and missile, and on the kinetic energy of the missile. After it has been demonstrated that the missile will not penetrate the barrier, an equivalent static load concentrated at the impact area is then determined, from which the structural response, in conjunction with other design loads, is evaluated.

Overall effects of missile impact are designed for flexural, shear, and buckling effects on structural members using the equivalent static load obtained from the evaluation of structural response. The impact is assumed to be plastic, and impact forces are determined as outlined in “Impact Effect of Fragments Striking Structural Elements” (Reference 3.3-6).

3.3.2.2.4 Combined Tornado or Hurricane Effects

The loading combinations of the individual tornado loading components are in accordance with SRP 3.3.2 (Reference 3.3-5) and are supplemented with the design criteria and procedures provided in BC-TOP-3-A (Reference 3.3-7). The load combinations used to determine the combined effects of the design basis hurricane wind and missile effects are equivalent to those used for the tornado and as discussed above and in Section 3.8. The only difference is that there is no atmospheric pressure change effect for the design basis hurricane. The total tornado or hurricane load W_t, used in the load combinations discussed in Section 3.8, is determined for the combined effects using the following equations.

$$W_t = W_w$$

$$W_t = W_p$$

$$W_t = W_m$$

$$W_t = W_w + 0.5\ W_p$$

$$W_t = W_w + W_m$$

$$W_t = W_w + 0.5\ W_p + W_m$$

where

W_t = total tornado or hurricane load

W_w = load from tornado or hurricane wind effect

W_p = load from tornado atmospheric pressure change effect

W_m = load from tornado or hurricane missile impact effect

All US-APWR seismic category I structures or components subject to tornado and hurricane wind loads are designed to preclude failure effects on safety-related SSCs housed within or adjacent to them. In addition, where the trajectory of a tornado missile or hurricane missile could impact safety-related SSCs, seismic category I structures and components are designed according to procedures described in Subsection 3.5.3 for a spectrum of tornado missiles and hurricane missiles described in Subsection 3.5.1.4, in order to preclude effects caused by missiles on safety-related SSCs. Standard plant US-APWR seismic category I structures include the R/B and the PCCV which rest on the common mat.

These requirements also apply to seismic category I structures provided by the COL Applicant. Similarly, it is the responsibility of the COL Applicant to establish the methods for qualification of tornado or hurricane effects to preclude damage to safety-related SSCs. Seismic category II structures and components are required to be designed for the same tornado and hurricane wind loads as seismic category I structures, in order to preclude impact on the function and integrity of safety-related SSCs. Limited failure of seismic category II structures is acceptable provided that function and integrity of safety-related SSCs are not affected.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado and Hurricane Loads

Failure effects of structures or components not designed for tornado and hurricane loads do not jeopardize the function and integrity of safety-related SSCs.

As previously discussed, the A/B is designed as vented with respect to tornado atmospheric differential pressure loading. The nonsafety-related A/B HVAC system connects to the distribution ductwork systems in the PS/Bs and R/B, and therefore those systems could experience depressurization due to tornado loading. Consequently, these systems could experience functional failure due to the tornado depressurization loading, however their failure will not result in collapse onto safety-related SSCs contained within the PS/Bs and R/B.

The structural steel frame of the T/B is enclosed by siding. The siding fasteners are designed to retain the siding for loading caused by extreme winds with a basic wind speed of 155 mph. However, the fastener design allows for portions of the siding to be blown off in the event of a design-basis tornado or design-basis hurricane, thereby venting the T/B and reducing the effective tornado wind pressure load or hurricane wind pressure load on the building. This ensures that there is no overall failure of the T/B, due to maximum tornado wind and/or atmospheric pressure change or due to maximum hurricane wind as defined in Table 2.0-1, which could affect the ability of adjacent buildings and structures to perform their intended safety functions. Localized failures of wind girts and other exposed SSCs are permitted. However, these items are designed to remain attached to the structure. Any items (including the T/B siding) which might become dislodged and become missiles under the maximum tornado or maximum hurricane conditions do not warrant further evaluation because they are considered to be enveloped by the missiles addressed in Subsection 3.5.1.4. The use of tornado-generated missile spectrum and hurricane-generated missile spectrum described in Subsection 3.5.1.4, which is consistent with the most severe missile spectrum as identified for Region I in RG 1.76 Revision 1 and RG 1.221 and NUREG/CR-7004 (Reference 3.3-9), respectively, provides assurance that the necessary SSCs will be available to mitigate the potential effects of a tornado or hurricane on plant safety.

The AC/B is not designed for a tornado or hurricane and consequently it could potentially fail due to design basis tornado loading or design basis hurricane loading. However, since its location is sufficiently far away from seismic category I structures, and adjacent safety-related SSCs buried in the plant yard, the collapse of the AC/B would not impact any adjacent safety-related SSCs. The AC/B may also have localized failure due to tornado and hurricane loading; however, the design precludes the generation of missiles that are not bounded by Subsection 3.5.1.4. The locations of any safety-related SSCs in the plant yard adjacent to the AC/B, including those which may be field routed, are reviewed prior to installation to ensure that their distances away from the AC/B and/or burial depths are sufficient to prevent potential failure effects that could jeopardize their function and integrity. Therefore, the ability of other SSCs to perform their intended safety functions is not affected by the potential collapse or localized failure of the AC/B due to tornado and hurricane loading.

It is the responsibility of the COL Applicant to assure that site-specific structures and components not designed for tornado and hurricane loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Subsection 3.5.1.4. Where required by the results of investigations, structural reinforcement and/or missile barriers are implemented so as not to jeopardize safety-related SSCs.

3.3.3 Combined License Information

COL 3.3(1) *The COL Applicant is responsible for verifying the site-specific basic wind speed is enveloped by the determinations in this section.*

COL 3.3(2) *These requirements also apply to seismic category I structures provided by the COL Applicant. Similarly, it is the responsibility of the COL Applicant to establish the methods for qualification of tornado or hurricane effects to preclude damage to safety-related SSCs.*

COL 3.3(3) *It is the responsibility of the COL Applicant to assure that site-specific structures and components not designed for tornado and hurricane loads will not impact either the function or integrity of adjacent safety-related SSCs, or generate missiles having more severe effects than those discussed in Subsection 3.5.1.4.*

COL 3.3(4) *The COL Applicant is to provide the wind load design method and importance factor for site-specific category I and category II buildings and structures. The COL Applicant is to also verify that the site location does not have features promoting channeling effects or buffeting in the wake of upwind obstructions that invalidate the standard plant wind load design.*

COL 3.3(5) *The COL Applicant is to note the vented and unvented requirements of this subsection to the site-specific category I buildings and structures.*

COL 3.3(6) *The COL Applicant is responsible for verifying that the site specific design basis hurricane basic wind speeds, exposure category, and resulting wind forces are enveloped by the determinations in this section.*

3.3.4 References

3.3-1 Minimum Design Loads for Buildings and Other Structures, American Society of Civil Engineers/Structural Engineering Institute, ASCE/SEI 7-05, Reston, Virginia, 2006.

3.3-2 Wind Loads, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, United States Nuclear Regulatory Commission SRP 3.3.1, Rev. 3, March 2007.

3.3-3 Tornado Design Classification, United States Nuclear Regulatory Commission Regulatory Guide 1.117, Rev. 1, April 1978.

3.3-4 Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, United States Nuclear Regulatory Commission Regulatory Guide 1.76, Rev. 1, March 2007.

3.3-5 Tornado Loads, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, United States Nuclear Regulatory Commission SRP 3.3.2, Rev. 3, March 2007.

3.3-6 Williamson, R.A. and Alvy, R.R., Impact Effect of Fragments Striking Structural Elements, Holmes and Narver, Inc. Publishers, November 1973.

3.3-7 Tornado and Extreme Wind Design Criteria for Nuclear Power Plants, Bechtel Topical Report BC-TOP-3-A, Bechtel Power Corporation, San Francisco, California, Rev. 3, August 1974.

3.3-8 Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants, United States Nuclear Regulatory Commission, RG 1.221, October 2011.

3.3-9 Technical Basis for Regulatory Guidance on Design-Basis Hurricane-Borne Missile Speeds for Nuclear Power Plants, NUREG/CR-7004, U.S. Nuclear Regulatory Commission, Washington, D.C., November 2011.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

The US-APWR, including site-specific SSCs, is designed to withstand the maximum water levels caused by flooding sources that are both external and internal to the plant as listed below. Combinations of external and internal events are not considered except where noted.

The external water source events are as follows:

- Probable maximum precipitation (PMP)
- Probable maximum flood (PMF) of streams and rivers
- Probable maximum tsunami hazards
- Probable maximum surge, seiche flooding, and wave action
- Potential dam failures
- Potential cooling water canals and reservoir failures
- Ground water
- Outdoor water storage facility failures

The internal water source events are as follows:

- Earthquakes
- Pipe breaks and cracks
- Fire fighting operations
- Pump mechanical seal failures

The US-APWR structures are designed for loads due to flooding. Design loads and load combinations consider both static and dynamic load effects for internal and/or external flooding.

3.4.1.1 Flood Protection for Safety and Nonsafety-Related Structures, Systems, and Components

Seismic category I and II structures are designed to protect SSCs such that plant nuclear safety functions are not jeopardized by flooding due to the potential failure(s) of the plant SSCs or the operation of the plant fire protection system. The plant nuclear safety functions are defined as any function that is necessary to assure the following:

a. The integrity of the RCPB

b. The capability to shut down the reactor and maintain it in a safe-shutdown condition

c. The capability to prevent or mitigate the consequences of plant conditions that could result in potential offsite exposures that are comparable to the guideline exposures of 10 CFR 100, “Reactor Site Criteria” (Reference 3.4-1)

In addition, the US-APWR plant design assures control room habitability and operator access to areas requiring local actuation of equipment required to achieve or maintain the conditions described in the preceding paragraph.

The SSCs required to be protected from flooding are discussed in this section. Additional information is provided in Sections 3.2 and 3.11 of this chapter.

Safety-related SSCs are protected from flooding by external and internal sources. The US-APWR design includes the following:

- The separation of redundant trains of safety-related SSCs as addressed in Chapters 1
- Protective barriers and enclosures, where necessary, as addressed in this section
- The placement of essential SSCs above internal flood levels
- SSCs are mounted above the flood level. While safety-related SSCs that are environmentally protected in accordance with Section 3.11 are permitted below the potential flood level, no components requiring active operation to achieve their intended safety function are located below the potential flood level.

Protection from flooding of nonsafety-related SSCs is considered when the impact of the flooding on a nonsafety-related SSC could be a contributing factor to the flooding of safety-related SSCs or could result in an uncontrolled release of significant radioactivity.

3.4.1.2 Flood Protection from External Sources

The US-APWR is designed for maximum water levels caused by external flooding. The design basis for external flooding complies with 10 CFR 50, Appendix A (Reference 3.4-2), specifically General Design Criterion 2, “Design Bases for Natural Phenomena.” This compliance is accomplished by designing SSCs to withstand the effects of natural phenomena such as floods, tsunami, and seiches without the loss of capability to perform their safety functions. Additionally, the design reflects the following considerations:

- The determination of the most severe natural phenomena, which has been historically recorded, is addressed in Section 2.4.
- The effects of the most severe natural phenomena have been considered to occur during both normal and accident conditions in the plant.
- The importance of the safety functions to be performed.

If PMP were to occur, US-APWR safety-related SSCs would not be jeopardized. US-APWR seismic category I building roofs are designed as a drainage system capable of handling the PMP, including allowance for primary roof drainage issues caused by probable maximum winter precipitation. Seismic category I structures have sloped roofs designed to preclude roof ponding. This design channels rainfall expeditiously off the roof. Runoff water resulting from precipitation is directed away from the R/B, and all other US-APWR standard plant structures, by virtue of the plant site grading and drainage system. The US-APWR standard design provides a plant site grade sloped away from all seismic category I and II structures.

The COL Applicant is to address the site-specific design of plant grading and drainage. Based on the design-basis flooding level (DBFL) and the plant elevation with regard to the DBFL, (refer to Subsection 3.4.1.4 for discussion on DBFL) the safety-related SSCs are protected from flooding along with the static and dynamic forces associated with a design basis flood in accordance with the requirements of “Flood Protection For Nuclear Power Plants”, RG 1.102 (Reference 3.4-3).

The COL Applicant is to identify and design, if necessary, any site-specific flood protection measures such as levees, seawalls, floodwalls, site bulkheads, revetments, or breakwaters per the guidelines of RG 1.102 (Reference 3.4-3), or dewatering system if the plant is not built above the DBFL.

When site-specific static water pressure, corresponding to maximum flood level or maximum ground water level, is not removed by site drainage, it is considered as structural load to the foundation plate of the building. Additionally, static water pressure, which cannot be removed by site drainage, is factored into horizontal, overturning, and upward static load reactions when designing seismic category I and II structures. The total value of buoyancy is based on the water head of maximum flood level or maximum ground water level excluding the motion of waves. The horizontal, overturning and upward static reactions are based on the total value of water head including the motion of waves. This loading is addressed in the design criteria presented in Section 3.8.

Below grade, the US-APWR nuclear island and other seismic category I and II structures are primarily protected against exterior flooding and the intrusion of ground water by virtue of their thick reinforced concrete walls and base mats. As recommended by NUREG-0800, SRP 14.3.2 (Reference 3.4-4), the external walls below flood level are equal to or greater than two feet thick to protect against water seepage, and penetrations in the external walls below flood level are provided with flood protection features. Construction joints in the exterior walls and base mats are provided with water stops to prevent seepage of ground water.

The COL Applicant is to address any additional measures below grade to protect against exterior flooding and the intrusion of ground water into seismic category I buildings and structures.

Below-grade exterior wall penetrations such as for piping and conduits have been minimized to reduce the risk of in-leakage and flooding. Where below-grade piping penetrations are necessary, they are designed to preclude water intrusion. This is addressed in the design criteria presented in Section 3.8 of this chapter. Where below-grade electrical conduit penetrations are necessary, internal conduit seals are installed to preclude a fluid pathway through the conduit.

Potential external sources of flooding also include failures of the plant systems and components such as outside storage tanks and yard piping. The US-APWR design includes flooding evaluation for the failures of plant systems and components that are not protected from tornado, hurricane and other missiles, for example, the primary make-up water storage tanks, refueling water storage auxiliary tank (RWSAT), demineralized water storage tanks, and fire water storage tanks. However, the site DBFL does not result from failure of such sources. Flood protection from the failure of the plant systems such as the outside storage tanks and yard piping is achieved using dikes, levees, retention basins, component location, and/or sited grading and drainage. Dikes, levees, and retention basins are provided to retain leaks and spills due to postulated failures of tanks and vessels, when appropriate. Alternatively, external tanks and piping are located sufficiently far away so that their failure does not jeopardize safety-related equipment. This is accomplished by locating external flood sources so that any spillage or leakage is directed away from safety-related equipment by virtue of the site grading and drains, and by locating these items away from exterior doors that could act as a pathway for flood waters. In addition, buried yard piping is located either in pipe tunnels or sufficiently far away so that cracks or breaks will not result in soil erosion that undermines safety-related structures or components. Alternatively, some yard tanks and vessels are small enough in volume and protected so that they do not present a credible source of flooding. Site-specific flooding hazards from engineered features, such as from service water or circulating water piping, is to be addressed by the COL Applicant.

Above grade, all exterior doors and equipment access openings that could create a pathway for floodwater to safety-related SSCs are located above the DBFL. Moreover, the R/B and the PS/B adopt enhanced design features for beyond-design basis (BDB) flood protection to introduce lessons learned from the accident at Fukushima Dai-ichi Nuclear Power Station. For more information, refer to MUAP-13002 (Reference 3.4-8).

Access to the PCCV vertical tendon gallery inside the PCCV base mat is only through a tunnel with an access hatch located exterior to the structure at elevation 3 ft, 7 in. This hatch is above the plant DBFL and, therefore, the tendon end anchorages are not subject to exposure to flooding. Access to the PCCV horizontal tendon end anchorages is through the tendon gallery in the R/B, which runs along the tendon buttresses at the 90° and 270° azimuths of the PCCV and have access hatches located on the roof of the R/B. There are no potential sources of flooding for the R/B gallery. In addition, all tendon end anchorages are protected from the elements with grease caps, which prevent the intrusion of moisture and contaminants. Therefore, the function and integrity of the safety-related PCCV tendon end anchorages and their respective tendons are not jeopardized by any sources of internal or external flooding, or PMP.

In summary, the US-APWR seismic category I and II structures provide hardened protection as defined in RG 1.59 (Reference 3.4-5) against external flooding through such design features as sloped roofs, thick reinforced concrete with special porosity-reducing additives, waterproofing, and special sealing of joints and penetrations.

3.4.1.3 Flood Protection from Internal Sources

The US-APWR SSCs are designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. This subsection addresses the accommodations made for flooding from internal water sources, specifically from the following:

- Earthquakes
- Pipe breaks and cracks
- Fire fighting operations
- Pump mechanical seal failures

The combination of events is not considered. However, an earthquake event followed by fire fighting operations for an earthquake induced fire is considered.

Full-circumferential ruptures of non-seismic piping and failures of non-seismic equipment located in the R/B or power source buildings (PS/Bs) are considered in the evaluation of flooding caused by an earthquake. Non-seismic equipment or piping in areas outside of the area of concern is also assumed to be fully compromised, and if the discharge fluids can not be demonstrated to be excluded from the area of concern, their volume is included in the flood volume. The US-APWR is designed for maximum water levels created by internal flooding sources. The internal flood design accommodates the effects of, and is compatible with, environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including LOCAs.

Water-tight doors are used as protective barriers to prevent flood waters from spreading to adjacent divisions in various buildings and elevations. Water-tight doors have remote position indication for closure verification and are subject to periodic visual inspection and functional testing to help maintain and demonstrate meeting their design function. Any aging-related degradation or fault is identified and repaired. The COL Applicant is responsible for developing inspection and testing procedures in accordance with manufacturer recommendations so that each water-tight door remains capable of performing its intended function.

Open pits are isolated within water tight compartments using water tight doors, penetration seals, and normally closed floor drains. In this manner, flooding effects caused by open pit water sloshing are considered.

For flood events caused by the postulated failure of piping, defined in Section 3.6, the rupture of the single worst-case piping in the area of concern is assumed in the flood analysis for each area of concern. The discharge volume is calculated according to “Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors”, American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.10-1987, Section 3 (Reference 3.4-6), and is included in the pipe break and cracks flood evaluation. The structures adjacent to the postulated pipe rupture locations are also designed for the maximum associated hydrodynamic loads due to a pipe failure as discussed in Section 3.6. The loads and load combinations are addressed in detail in Section 3.8.

In the flooding effects from fire fighting operations, water discharged from only fire hose stations is assumed. The fire fighting operations are assumed to continue for a period of 2 hours from two hose stations, on the basis of the Section 3.2 of RG 1.189. The discharge rate of 125 gpm per hose station is applied, assuming the use of a hose station that can be handled by untrained persons.

Pump mechanical seal failures of concern are limited to the active pumps identified in Section 3.9. Seal failure is a low probability event based on the use of robust pump mechanical seals. Additionally, monitoring of mechanical seal water temperature, pressure, and flow rate across the pump mechanical seals provides the means of limiting the effects of pump seal failure through early detection and timely corrective action. As such, pump mechanical seal failure presents a sufficiently low probability of occurrence and flood volume that it can be credibly ignored.

The formulae and methodology of “Design Criteria for Protection against the Effects of Compartment Flooding in Light Water Reactor Plants”, ANSI/ANS-56.11-1988 (Reference 3.4-7) are used when analyzing flow rates through unusual features such as stairwells and floor/wall openings.

The areas of concern within the US-APWR are as follows:

- R/B
    - Inside the PCCV

      Systems to be protected within the PCCV are the RCS, the safety injection system (SIS), RHRS, the CSS, and the containment boundary.

      The components to be protected from flooding in the protected systems are the motor operated components, such as valves and electric/instrumentation components.

    - Outside the PCCV

      US-APWR R/B consists of a radiological controlled area (RCA) and a non-radiological controlled area (NRCA) separated physically by concrete barrier walls. These concrete barrier walls are designed to preclude flooding between the RCA and the NRCA. Piping, instrumentation, HVAC duct, conduit, and cable trays installed through a flood barrier wall are routed above the maximum flood level or provided with water-tight seals.

        - R/B RCA

          Systems to be protected in the RCA of the R/B are the SIS, the RHRS, the CSS, the containment boundary, the safeguard component area HVAC system, and the annulus air clean up system.

In the systems to be protected, the components to be protected from flooding are the motor driven pumps, the valves, and the HVAC fans and dampers, the electric panels, and the electric/instrumentation components within the relevant system.

Instrumentation for flood detection is installed in the containment annulus compartment since the compartment houses mechanical penetrations including piping systems containing water. The instrumentation is designed to alarm when the annulus compartment is flooded.

- R/B NRCA

  The NRCA of the R/B adjoins the east and west PS/Bs and the T/B, with personnel access between all three areas.

  The systems to be protected in the NRCA of the R/B are the CCWS, the emergency feedwater system (EFWS), the electrical panels, the Class 1E electric/instrumentation components, and the HVAC fans and dampers for these systems.

  Evaluation of flooding occurring in the essential service water pipe chase (ESWPC) is site-specific.

- PS/B

  The east and west PS/Bs adjoin the R/B.

  The equipment to be protected in the east and west area of PS/B is the A, B, C, and D train Essential Chiller Units, and A, B, C, and D train Class 1E GTG.

- A/B

  The A/B adjoins the R/B. There are no SSCs to be protected from flooding the A/B.

- T/B

  The T/B adjoins the NRCA of the R/B. The T/B is subject to flooding from a variety of potential sources including the circulating water, service water, condensate/feedwater, CCW, demineralized water, and fire protection systems.

  The bounding flooding source for the T/B is a break in the circulating water piping.

  A break in the circulating water system (CWS) piping would result in water flowing into the lower elevation of the T/B, elevation -18 ft, 0 in. When the flood water fills the lower level of the T/B, to prevent the CWS flood volume from affecting R/B equipment, a flood relief panel system is built into the T/B exterior walls. Passive actuation of the flood relief panels allows the CWS flood volume to drain out to the yard area. In the yard area, the flood volume is directed away from the plant structures by virtue of the site grading and yard drainage system. In addition, the water tight doors are installed in the doorways at ground level between T/B and R/B.

There is no equipment to be protected from flooding in the T/B.

Appendix 3K provides the location of components within safety-related buildings of the US-APWR, and compares the maximum internal flood elevation within the vicinity of the components. The COL Applicant is responsible for the protection from internal flooding for those site-specific SSCs that provide nuclear safety-related functions or whose postulated failure due to internal flooding could adversely affect the ability of the plant to achieve and maintain a safe shutdown condition.

3.4.1.4 Evaluation of External Flooding

The following steps outline the external flood evaluation process:

1. Identification of components required to maintain functionality during a flood event
2. Identification of flood sources and flow paths relative to the identified components
3. Risk assessment for components affected by a flood event
4. Failure mode and effects analysis for components affected by a flood event
5. Determination of appropriate actions to preclude impact to component safety functions

For purposes of the US-APWR standard plant design, the plant site DBFL caused by external source flooding is discussed in Section 2.4.

The PMP for the US-APWR standard plant design is defined in Section 2.4.

Based on the DBFL and the plant elevation with regard to the DBFL as presented in Section 2.4, the safety-related SSCs are protected from flooding along with the static and dynamic forces, if applicable, associated with a design basis flood in accordance with the requirements of RG 1.102 (Reference 3.4-3). The COL Applicant is to demonstrate the DBFL bounds their specific site, or is to identify and address applicable site conditions where static flood level exceeds the DBFL and/or generates dynamic flooding forces.

3.4.1.5 Evaluation of Internal Flooding

The following steps outline the internal flood evaluation process:

1. Identification of components required to maintain functionality during a flood event
2. Identification of flood sources and flow paths relative to the identified components
3. Risk assessment for components affected by a flood event
4. Failure mode and effects analysis for components affected by a flood event

5. Determination of appropriate actions to preclude impact to component safety functions

3.4.1.5.1 PCCV Flood Events

The interior of the US-APWR PCCV is divided into three elevations for the purposes of this discussion:

1. Elevation 25 ft, 3 in. (second floor), which separates the refueling water storage pit (RWSP) and the balance of the PCCV, with openings for the stairway access to the RWSP and alternate water flow paths
2. Elevation 50 ft, 2 in. (third floor), which corresponds with the middle doorway into each of the SG compartments (the floor corresponding to this elevation is concrete)
3. Elevation 76 ft, 5 in. (fourth floor/operating floor), which corresponds with the upper doorway into each of the SG compartments (the floor corresponding to this elevation is concrete)

The PCCV is further partitioned by the four SG compartments, the pressurizer compartment, the regenerative heat exchanger room, the letdown heat exchanger room, the excess letdown heat exchanger room, the refueling cavity, the HVAC header compartment, the C/V drain pump room, and the reactor cavity.

In order to assure the long-term cooling performance after an accident event, spray water from the CSS or spill water from a damaged component is collected in the RWSP at the bottom of the PCCV. The partitions above the RWSP are designed to permit containment spray water and spill water to flow freely through the PCCV to the RWSP.

This is accomplished through floor drains in the regenerative heat exchanger room, the letdown heat exchanger room, and the excess letdown heat exchanger room. Water flowing into each room is drained by way of the floor drain to the PCCV sump in the C/V drain pump room (elevation 3 ft, 7 in.). If the PCCV sump is full or if the floor drain is clogged, water will flow from each area/compartment through the flow paths described as follows:

- Water on the elevation 76 ft, 5 in. concrete floor flows to the C/V drain pump room through the floor drain, or alternatively, flows to the elevation 25 ft, 3 in. floor through the stairwells and equipment hatch opening.
- Water on the elevation 50 ft, 2 in. concrete floor flows to the C/V drain pump room through the floor drain, or alternatively, flows to the elevation 25 ft, 3 in. floor through the stairwells and equipment hatch opening.
- Water flowing onto the elevation 25 ft, 3 in. floor flows through floor openings in the SG compartment rooms into the header compartment, reactor cavity, and the C/V drain pump room. First, water completely fills the C/V drain pump room up to elevation 25 ft, 3 in. Then, the water level in the header compartment and reactor cavity reaches the overflow piping at elevation 20-ft, 10-in. which connects these rooms to the RWSP. The header compartment and reactor cavity are connected

through tunnels which equalizes the water level in these rooms. When the flood water in the header compartment and reactor cavity exceeds the top of the overflow piping at elevation 20-ft, 10-in., water flows into the RWSP.

- Water at the pressurizer compartment flows to the SG compartment by way of the inlet/outlet to the SG compartment.
- Water in the regenerative heat exchanger room, the letdown heat exchanger room, and the excess letdown heat exchanger room flow out their respective entrance doorways. The flow path from there is to the respective level floor drains or through the alternate paths.

Thus, flood waters in the containment reaches the RWSP, the HVAC header compartment, the C/V drain pump room, and the reactor cavity.

Inside the containment, the largest water retaining components are the refueling cavity, and the RWSP. The RWSP and the refueling cavity are robust reinforced concrete seismic category I structures with thick walls which have been designed for all applicable loads, including the potential in-containment missile loads and hydrodynamic loads. Due to their robust design, a postulated failure of these structures is not credible. Additionally, the combined fuel transfer canal/reactor cavity pit during all but refueling operations is dry. Since a LOCA represents the worst case flooding event, sloshing is not a factor in PCCV flooding.

There are piping systems inside the PCCV such as the CCW system and fire protection water supply system (fire water) that are connected to large-volume water sources. However a significant accidental release of water into the PCCV from these sources is not plausible for the following reasons:

- All CCW piping inside the PCCV is classified as seismic category I.
- Containment isolation valves outside the PCCV for the fire protection water supply system are normally closed. Therefore, there is no water released by a pipe break of the fire protection water supply system inside PCCV.
- The RCP purge water head tank and C/V reactor coolant drain tank are non-seismic components which contain water inside the PCCV. The total amount of water contained within these tanks is 106 ft^3, which is significantly less than the volume of water from a LOCA.

When water is flowing through the overflow piping and floor openings, the water level slightly exceeds the top of the overflow piping at elevation 20-ft, 3-in. and the floor level at 25-ft, 3-in. However, the components to be protected are installed at a sufficiently higher level than the height of the overflowing water.

The volume of water from a LOCA is conservatively assumed to be equal to the volume of the RCS volume, the four accumulator tank volumes, and the volume of the RWSP for a total volume of 123,000 ft^3.

The total volume of storage space below the top of the overflow piping at elevation 20 ft, 10-in. is 130,000 ft^3. All flood water is collected below the bottom-layer partition. The components to be protected are installed above the top of the overflow piping. The SSCs inside the PCCV such as emergency letdown line isolation valve, safety depressurization valve and pressurizer backup heater are not affected by flood waters.

Table 3K-1 is provided in order to list all the protective SSCs inside PCCV that require flood protection and to verify the locations of SSCs relative to the internal flood level. Table 3K-1 includes the information whether each SSC is above the maximum flood elevation inside PCCV (EL.20'-10") or not.

3.4.1.5.2 Reactor Building Flooding Events

The US-APWR R/B consists of a RCA separated physically by concrete walls and/or floors and a NRCA. These concrete walls are designed so that flooding may not cross between mutual areas by installing penetrations for pipe, ducts, and cable trays above the maximum flooding level and/or by sealing the penetrations. As recommended by SRP 14.3.2 (Reference 3.4-4), penetrations in divisional walls are at least 8 ft, 3 in. above the floor, and safety-related electrical, instrumentation, and control equipment are located sufficiently above the flood level.

Inside the R/B, the cask wash down pit, the new fuel storage pit, and the interconnected spent fuel pit, the transfer canal, the cask loading pit, and the fuel inspection pit, with the tops of these open pits all located at plant elevation 76 ft, 5 in., represent potential large volumes of water. These pits are robust seismic category I structural reinforced concrete with thick walls, which have been designed for applicable loads, including hydrodynamic loads. Due to their robust design, postulated failure of these structures is not credible. Flood water that could be displaced out of these pits due to hydrodynamic effects such as sloshing is enclosed within pit area by water tight compartmentalization.

The R/B is adjoined by the A/B, and as such, the A/B consists of a RCA and a NRCA which are physically separated. Doorways with water-tight doors are provided between the RCA of the R/B and the RCA of the A/B. Similarly, doorways with water-tight doors are provided between the NRCA of the R/B and the NRCA of the A/B, and between the NRCA of R/B and T/B.

The NRCA of the R/B adjoins the PS/B, with doorways providing potential flow paths through the areas.

3.4.1.5.2.1 Radiological Controlled Area

The RCA is arranged with each of the four safety trains separated into four quadrants around the outside of the PCCV. Physically, individual train equipment within the four quadrants is located to provide the separation between the same equipment of the other three trains within the confines of the R/B footprint. This separation minimizes the probability of an event affecting more than one of the safety trains at a given time.

All floors in the RCA of the R/B are divided into the two areas, east and west, by concrete walls and/or water-tight doors. The concrete walls are designed to prevent flood water migration from one safety train to another. This is accomplished by installing piping, electrical conduit, HVAC duct, cable trays, etc., penetrations above the maximum flood level and/or by sealing penetrations.

Two types of drain systems are provided in the RCA of the R/B outside the PCCV; an equipment drain system and a room/compartment floor drain system. The equipment drain system collects water leaking from components and routes the leakage to the R/B sump tank at elevation -26 ft, 4 in. (basement first floor). The floor drain water is collected in the R/B sump tank at elevation -26 ft, 4 in. The floor drains of safeguard component rooms are designed to be routed from their collection area to the R/B sump tank using piping independent of the other floor drains.

The floor drains of the east area are connected and ultimately go into the A-R/B sump tank at elevation -26 ft, 4 in. The floor drains of the west area are connected and ultimately go into the B-R/B sump tank at elevation -26 ft, 4 in. There is no cross-connection between east area drains and west area drains. Therefore, east and west areas are evaluated as independent areas.

Elevation -26 ft, 4 in.

The equipment to be protected in the east area of RCA at elevation -26 ft, 4 in. is the A and B train SI pumps and the A and B train CS/RHR pumps. Equipment to be protected in the west area is the C and D train SI pumps and the C and D train CS/RHR pumps.

In both the east and west area, the SI pump rooms and CS/RHR pump rooms are isolated by concrete walls and water-tight door. Moreover, floor drains of these rooms are separated by a closed valve for each train. The other equipment rooms are isolated by concrete walls, the fireproof doors and/or air-tight doors which are not water-tight. Therefore, flood water is assumed to run across the area except the SI pump rooms and CS/RHR pump rooms.

Flood Events are considered as follows;

- Earthquake

  Most of the water-containing equipment and piping in the RCA of the R/B are excluded from flooding source because they are designed to withstand Safe Shutdown Earthquake (SSE). The flood water volume is evaluated on the basis of amounts of water contained in other non-seismic equipment and/or piping.

  The amount of water released in the seismic event is 250 ft^3 in the east area and 320 ft^3 in the west area.

- High-energy line break/moderate-energy line break (HELB/MELB)

  The high energy line in the RCA of R/B consists of the charging line, letdown line, and seal water injection line of the chemical and volume control system (CVCS). All these lines are not routed within the east side of R/B. Of these lines, the line break in the charging piping in the west side of R/B RCA constitutes the most severe flooding event. The discharge water volume in this event is calculated on the assumption that the release of water from the charging piping continues without being limited by the pump capability until the letdown line is automatically isolated. The time required for automatic closure of an isolation valve after an isolation signal is conservatively estimated to be 5 minutes.

The total water volume from this HELB event in the west area is 15,000 ft^3.

Most of the water-containing moderate energy piping in the RCA of the R/B is excluded from flooding source because that piping is to be designed so that a crack is not required to be postulated in the line in accordance with the criteria described in subsection 3.6.2.1.2.2. This is attained by maintaining stress on the pipes below the threshold by means of route and support design. Portions of the water-containing moderate energy piping may be designed to be subject to crack postulation, only if it is confirmed by conservative evaluation that a crack of the parts postulated in accordance with the criteria provided in subsection 3.6.2 will not jeopardize SSCs which are required to be protected from flooding.

Refer to Table 3.4-1 for systems a part of which is designed not to crack for flood protection and systems whose piping is assumed to crack or break within the R/B or the PS/B in the internal flooding evaluation. Volume of flood water discharged from a postulated crack in the rest of the piping is conservatively estimated on the basis of the volume of water contained in the piping and any connected reservoir. The maximum water volume released in the MELB of piping routed on this elevation level is 250 ft^3 in both the east and west area.

- Fire Fighting Operations

  Flooding contribution from fire-fighting operations is based on the full operation of two hose stations for 2 hours. The flow rate from 1 hose station is 125 gpm. With two stations operating for 2 hours, the total volume of water is 4,010 ft^3.

Based on the above, the worst case flooding on the west side of the R/B is a HELB at 15,000 ft^3. On the east side of the plant, the worst case flooding is an earthquake followed by fire fighting operations due to an earthquake induced fire at 4,260 ft^3.

The square footage of floor area subject to flooding at elevation -26 ft, 4 in. is as follows:

- East side: 4,100 ft^2
- West side: 4,200 ft^2

Based on these values, the maximum water levels are as follows:

- East side: 1.04 ft above elevation -26 ft, 4 in.
- West side: 3.58 ft above elevation -26 ft, 4 in.

The SI pump and CS/RHR pump are installed in a room which prevents flow-in water by water-tight door, and floor drains of these rooms are separated by closed valve or check valve for each train. Therefore, the pumps are not flooded. Instrumentation of the SI pump and CS/RHR pump are installed above the flood water level.

Elevation 3 ft, 7 in.

Flood waters occurring above elevation -26 ft, 4 in. drain to floor elevation -26 ft, 4 in. through floor drains, stairwell, elevator shaft and/or equipment hatch. However, the evaluation above elevation -26 ft, 4 in. conservatively assumes that the water drainage will not reduce the flood water level at the floor of origin.

The equipment to be protected in the east area of RCA at elevation 3 ft, 7 in. is the A and B train CS/RHR heat exchanger (HX), the A and B train safeguard component area air handling unit, and the A train SFP pump. The equipment to be protected in the west area of RCA at elevation 3 ft, 7 in. is the C and D train CS/RHR HX, the C and D train safeguard component area air handling unit, and B train SFP pump.

The CS/RHR HX and the safeguard component area air handling unit are isolated by concrete walls and water-tight door. Moreover, floor drains of these rooms are separated from floor drains outside of these rooms and are also separated for each train. Therefore, flood water is assumed to run across the area except the CS/RHR HX and the safeguard component area air handling unit rooms.

Flood Events are considered as follows:

- Earthquake

  The flood water volume is estimated on the basis of the amount of water contained in non-seismic equipment or piping. The amount of water discharged in the seismic event is 10 ft^3 in the east area and 70 ft^3 in the west area.

- HELB/MELB

  HELB event is not a concern, because the flood water discharged from the postulated pipe break will immediately flow down to the lower floor level through floor opening.

  The maximum water volume from the MELB event is 110 ft^3 in both the east and west area.

- Fire Fighting Operations

  The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is 4,020 ft^3 in the east area and 4,080 ft^3 in the west area.

The footage of subject area and the water level are as follows:

- East side: 5,800 ft^2 area, 0.70 ft water height above elevation 3 ft, 7 in.

- West side: 5,200 ft^2 area, 0.79 ft water height above elevation 3 ft, 7 in.

CS/RHR HX and safeguard component area air handling unit are installed in the room which prevents flow-in water by water-tight door, and floor drains of these rooms are separated from floor drains outside of these rooms and are also separated for each train. Therefore, components are not flooded. The instrumentation of the CS/RHR HX and safeguard component area air handling unit are installed above the flood water level.

The height (top of concrete) of A and B train SFP pump foundations are 1.0 ft above the floor elevation 3 ft, 7 in. Therefore, the SFP pumps are not flooded.

Elevation 25 ft, 3 in.

The equipment to be protected in the east and west area of RCA elevation 25 ft, 3 in. is the containment isolation valves in piping penetration room.

Flood Events are considered as follows:

- Earthquake

  The amount of water discharged from non-seismic equipment or piping in the seismic event is 10 ft^3 in both the east and west area.

- HELB/MELB

  HELB event is not a concern, because the flood water discharged from the postulated pipe break will immediately flow to the lower floor level through floor opening.

  The maximum water volume from the MELB event is 110 ft^3 in both the east and west area.

- Fire Fighting Operations

  The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is 4,020 ft^3 in both the east and west area.

The both east and west areas are isolated by concrete walls and the fireproof doors and/or air-tight doors which are not water-tight. Therefore, flood water is assumed to run across the each area.

The footage of subject area and the water level are as follows;

- East side: 9,450 ft^2 area, 0.43 ft water height above elevation 25 ft, 3 in.

- West side: 7,200 ft^2 area, 0.56 ft water height above elevation 25 ft, 3 in.

The containment isolation valve motors are installed above the flood water level.

Elevation 50 ft, 2 in.

The equipment to be protected in the east and west area of RCA elevation 50 ft, 2 in. is annulus emergency exhaust filtration unit and junction boxes and cables in the electrical penetration rooms.

Flood Events are considered as follows;

- Earthquake

  The amount of water discharged from non-seismic equipment or piping in the seismic event is 10 ft^3 in the west area. There is no non-seismic water-containing equipment or piping in the east area on this floor.

- HELB/MELB

  HELB event is not a concern, because there is no high energy piping on or above this floor.

  There is no water-containing moderate energy-piping in which a leakage crack should be postulated on or above this floor.

- Fire Fighting Operations

  The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is 4,010 ft^3 in the east area, and 4,020 ft^3 in the west area.

The both east and west areas are isolated by concrete walls and the fireproof doors and/or air-tight doors which are not water-tight. Therefore, flood water is assumed to run across the each area.

The footage of subject area and the water level are as follows:

- East side: 7,550 ft^2 area, 0.54 ft above elevation 50 ft, 2 in.

- West side: 6,650 ft^2 area, 0.61 ft above elevation 50 ft, 2 in.

The annulus emergency exhaust filtration unit foundations (top of concrete) height is 1.0 ft above floor elevation 50 ft, 2in. As such, the annulus emergency exhaust filtration units are not flooded. The junction boxes and cables in the electrical penetration rooms is designed to be located at heights above the level of flood water.

Elevation 76 ft, 5 in.

Elevation 76 ft, 5 in. of the RCA is divided into two areas, east and west, by concrete wall and water-tight doors and the fuel handling area. The fuel handling area is isolated by installing the water-tight doors to walkway and/or doorways of stairwell to prevent flood water by sloshing of SFP spilling to other area.

The equipment to be protected from internal flooding on elevation 76 ft, 5 in. of the RCA are junction boxes and cables connected to the PCCV penetrations in the east and west electrical penetration areas.

There is no equipment to be protected in the fuel handling area.

Flood Events are considered as follows;

- Earthquake

  The amount of water discharged from non-seismic equipment or piping in the seismic event is 10 ft^3 in the west area. There is no non-seismic water-containing equipment or piping in the east area on this floor.

- HELB/MELB

  HELB event is not a concern, because there is no high energy piping on or above this floor.

  There is no water-containing moderate energy-piping in which a leakage crack should be postulated on or above this floor.

- Fire Fighting Operations

  The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is 4,010 ft^3 in the east area, and 4,020 ft^3 in the west area.

The both east and west areas are isolated by concrete walls and the fireproof doors and/or air-tight doors which are not water-tight. Therefore, flood water is assumed to run across the each area.

The footage of subject area and the water level are as follows:

- East side: 5,900 ft^2 area, 0.68 ft above elevation 76 ft, 5 in.
- West side: 4,900 ft^2 area, 0.83 ft above elevation 76 ft, 5 in.

The junction boxes and cables in the electrical penetration rooms are designed to be located at heights above the level of flood water.

3.4.1.5.2.2 NRCA

The NRCA is arranged into rooms/compartments to provide a physical separation of the water containing components from the electrical components. This separation, along with the associated physical barriers (concrete walls and floors), minimizes the probability of component leaks affecting the electrical components.

All floors in the NRCA of the R/B are divided into the two areas, east and west, by concrete walls and/or water-tight doors. The concrete walls are designed to prevent flood water migration from one safety train to another. This is accomplished by installing piping, electrical conduit, HVAC duct, cable trays, etc., penetrations above the maximum flood level and/or by sealing penetrations.

Two types of drain systems are provided in the NRCA of the R/B - an equipment drain system and a room/compartment floor drain system. The equipment drain system collects water leaking from components and routes the leakage to the non-radioactive drain sump. The floor drain water is also routed to the non-radioactive drain sump at elevation -26 ft, 4 in. The floor drains of the east areas are connected and finally go into the A-R/B non-radioactive sump. The floor drains of west areas are connected and finally go into the B-R/B non-radioactive sump. There is no cross-connection between east area drains and west area drains. Therefore, east and west areas are evaluated as independent areas.

The drains from the NRCA of A/B and the PS/B are also collected in the R/B non-radioactive sumps. The water in the R/B non-radioactive sumps is transferred to the T/B sump by sump pumps. The evaluation of flooding in the NRCA area of the R/B conservatively excludes the use of the sump pump.

The drains from the main steam (MS) / feedwater (FW) piping area is directly collected in the T/B sump. The MS/FW piping area is addressed separately below.

Elevation -26 ft, 4 in.

The systems to be protected at elevation -26 ft, 4 in. of the NRCA of the R/B are the four trains (A, B, C, and D) of the component cooling water (CCW) heat exchanger and pump and four trains (A, B ,C and D) of the emergency feedwater (EFW) pump.

The east side includes the two trains (A and B) of the CCW HX and pump rooms, and two trains (A and B) of the EFW pump room. The west side includes the two trains (C and D) of the CCW HX and pump room, and two trains (C and D) of the EFW pump room.

Equipment rooms are isolated by concrete walls and the fireproof doors which are not water-tight. Therefore, flood water is assumed to run across the area.

In addition, since the doorways to the PS/B which adjoined each east and west area of R/B are not water-tight, flood water of the NRCA of R/B is assumed to flow into the whole area of the PS/B, elevation -26 ft, 4 in.

Therefore, the subject area of east is the east side of R/B and PS/B. Similarly, the subject area of west is the west side of R/B and PS/B.

Flood events are considered as follows:

- Earthquake

  Most of the water-containing equipment and piping in the NRCA of the R/B and PS/B are excluded from flooding source because they are designed to withstand Safe Shutdown Earthquake (SSE). The flood water volume is estimated on the basis of the amount of water contained in other non-seismic equipment or piping.

  The amount of water discharged in the seismic event is 750 ft^3 in the east area and 760 ft^3 in the west area.

- HELB/MELB

  HELB event is not a concern, because there is no water-containing high energy piping which is assumed to break in the subject area.

  Most of the water-containing moderate energy piping in the NRCA of the R/B and the PS/B is excluded from flooding source because that piping is to be designed so that a crack is not required to be postulated in the line in accordance with the criteria described in subsection 3.6.2.1.2.2. This is attained by maintaining stress on the pipes below the threshold by means of route and support design. Portions of the water-containing moderate energy piping may be designed to be subject to crack postulation, only if it is confirmed by conservative evaluation that a crack of the parts postulated in accordance with the criteria provided in subsection 3.6.2 will not jeopardize SSCs which are required to be protected from flooding.

  Refer to Table 3.4-1 for systems a part of which is designed not to crack for flood protection and systems whose piping is assumed to crack or break within the R/B or the PS/B in the internal flooding evaluation. Volume of flood water discharged from a postulated crack in the rest of the piping is conservatively estimated on the basis of the volume of water contained in the piping and any connected reservoir. The maximum water volume released in the MELB of piping routed on this elevation level is 740 ft^3 in both the east and west area.

- Fire Fighting Operations

  The flooding contribution from fire fighting operations is based on the full operation of two hose stations for 2 hours. The flow rate from one hose station is 125 gpm. With two stations operating for 2 hours, the total volume of water is 4,010 ft^3.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is as follows:

- East side: 4,760 ft^3
- West side: 4,770 ft^3

The square footage of floor area subject to flooding at elevation -26 feet, 4 inches is as follows:

- East side: 11,500 ft^2
- West side: 11,100 ft^2

Based on these values, the maximum water level is as follows:

- East side: 0.42 ft above elevation -26 ft, 4 in.
- West side: 0.43 ft above elevation -26 ft, 4 in.

The pump foundations (top of concrete) height is 1.0 foot above floor elevation -26 ft, 4 in. As such, the pumps are not flooded. The instrumentation of each pump is designed to be located at heights above the level of flood water.

Elevation 3 ft, 7 in.

Flood waters occurring above elevation -26 ft, 4 in. drain to floor elevation -26 ft, 4 in. through floor drains, stairwell, elevator shaft and/or equipment hatch. However, the evaluation above elevation -26 ft, 4 in. conservatively assumes that the water drainage will not reduce the flood water level at the floor of origin.

The equipment to be protected in the east area of NRCA at elevation 3 ft, 7 in. is the A and B train Class 1E electrical panels. Similarly, the equipment to be protected in the west area is the C and D train Class 1E electrical panels. The Class 1E electrical panel rooms are isolated from corridor by concrete walls and water-tight door. There are no floor drains in the Class 1E electrical panel rooms.

Since the doorway between the corridor in the east PS/B and east area of the R/B at elevation 3 ft, 7 in. is not water-tight, flood water in the east NRCA of the R/B is assumed to flow into the east PS/B, and vice versa.

Flood events are considered as follows:

- Earthquake

  The flood water volume is estimated on the basis of the amount of water contained in non-seismic equipment or piping. The amount of water discharged in the seismic event is 10 ft^3 in both the east and west area.

- HELB/MELB

  HELB event is not a concern, because there is no water-containing high energy piping which is assumed to break in the subject area.

  The maximum water volume from the MELB event is 20 ft^3 in both the east and west area.

- Fire Fighting Operations

The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is 4,020 ft^3 in both the east and west area.

The footage of corridor area and the water level are as follows:

- East side: 4,300 ft^2 area, 0.94 ft above elevation 3 ft, 7 in.
- West side:1,500 ft^2 area, 2.68 ft above elevation 3 ft, 7 in.

Class 1E electrical panels are installed in the room which prevents flow-in water by water-tight door. Therefore, panels are not flooded.

Elevation 25 ft, 3 in.

The equipment to be protected in the NRCA portion of elevation 25 ft, 3 in. is the main control panel and Class 1E I&C panels. The main control room and Class 1E I&C rooms are isolated from corridor by concrete walls and water-tight door.

Flood events are considered as follows;

- Earthquake

  The flood water volume is estimated on the basis of the amount of water contained in non-seismic equipment or piping. The amount of water discharged in the seismic event is 10 ft^3 in both the east and west area.

- High-Energy Line Break/Moderate-Energy Line Break (HELB/MELB)

  HELB event is not a concern, because there is no water-containing high energy piping which is assumed to break in the subject area.

  The maximum water volume from the MELB event is 20 ft^3 in both the east and west area.

- Fire Fighting Operations

  The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Since the potable and sanitary water system (PSWS) piping within the MCR compartment is moderate-energy fluid system piping which is designed to be excluded from the postulation of leakage crack during normal plant operation, and is designed as seismic category II to ensure its integrity during and after the SSE, the PSWS piping within the MCR compartment is not postulated to crack or break.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is 4,020 ft^3 in both the east and west area.

The footage of corridor area and the water level are as follows;

- East side: 1,500 ft^2 area, 2.68 ft above elevation 25 ft, 3 in.
- West side: 1,700 ft^2 area, 2.37 ft above elevation 25 ft, 3 in.

The Class 1E I&C panels are installed in the room, which prevents flow-in water by the use of barriers and water-tight doors. Therefore, panels are not flooded. The MCR subject to regular access is protected from flooding by the use of barriers.

The MCR penetrations are designed to prevent water from flowing in by applying appropriate sealing features. The HVAC ducts coming from the MCR air handling units and the filter train units are routed horizontally above the postulated flooding level. The vertical HVAC ducts penetrate the MCR ceiling and are welded to embedded sleeves for penetration. The HVAC duct sections of concern and the embedded sleeves are designed to withstand the hydrostatic load of flooding. The penetrations of sanitary pipes also use the embedded sleeves (southern exterior wall of the R/B). Cables enter the MCR from beneath the raised MCR floor, and the penetrations at the control room envelope boundary may contain a liquid or clay filling and are water sealed. Therefore, flooding of the MCR through those penetrations is precluded through the use of appropriate sealing features.

Elevation 50 ft, 2 in.

The equipment to be protected in the elevation 50 ft, 2 in. of the NRCA is the MCR air handling units, Class 1E electrical room air handling units, and MCR emergency filtration units.

Flood events are considered as follows;

- Earthquake

  There is no non-seismic water-containing equipment or piping in both the east and west areas on or above this floor.

- High-Energy Line Break/Moderate-Energy Line Break (HELB/MELB)

  HELB event is not a concern, because there are no piping breaks, which are assumed to occur in the subject area.

  The maximum water volume from the MELB event is 20 ft^3 in both the east and west area.

- Fire Fighting Operations

The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the flood events described above, the worst case results are from fire fighting operations. The total volume of flood water caused by this event is 4,010 ft^3 in both the east and west area.

The footage of subject area and the water level are as follows;

- East side: 5,150 ft^2 area, 0.78 ft above elevation 50 ft, 2 in.
- West side: 5,250 ft^2 area, 0.77 ft above elevation 50 ft, 2 in.

The MCR air handling units, Class 1E electrical room air handling units, as well as the MCR emergency filter units have a steel frame base installed on the top of the concrete foundations. The additional height of this base results in a total of 1.5 feet between the floor level and the filtration units. Therefore, when considering the steel frame base units, the current design has sufficient margin (i.e., 0.72 feet above the postulated flood level) to protect against the postulated flooding.

Elevation 76 ft, 5 in.

Elevation 76 ft, 5 in. of the NRCA is divided into the MS/FW piping area and other areas by concrete walls and water-tight doors. Moreover, the MS/FW piping area is divided into the two areas, east and west, by the concrete wall.

The equipment to be protected in the MS/FW piping area is the MS isolation valve, main feedwater isolation valve (MFIV), and MS depressurization valve.

The equipment to be protected in the subject area, except the MS/FW piping area, is the instrumentation of the EFW pit, and the remote shutdown console within the remote shutdown room.

Flood events in the MS/FW piping area are considered as follows:

- Earthquake

  The amount of water discharged from non-seismic equipment or piping in the seismic event is 2,700 ft^3 in both the east and west areas.

- HELB/MELB

  In the flooding events caused by the postulated failure of piping, the high energy piping consists of main steam, feedwater, and SG blowdown piping, within the MS/FW piping area. A rupture of the feedwater piping in this area represents the worst case flooding scenario for this area. This is based on a 1.0 ft^2 break, as defined in Section 3.6, in the feedwater piping downstream of the feedwater check valve. The rupture at this point results in feedwater from the SG and from within the associated feedwater piping flow back into and flooding the compartment. In addition, the main feedwater is assumed to continue flowing out from upstream line at the maximum discharge flowrate. As a result of this scenario, the water level in the SG and the main steam line pressure would decline. As a result of low main steam line pressure ECCS actuation signal, the main feedwater regulation valve and main feedwater isolation valve are closed, which stops water discharge. The volume of water which floods the main steam/feedwater pipe/relief valve compartment, based on the maximum discharge flowrate and the time required for the valve closure, is 12,180 ft^3. The flood water occurring in the main steam/feed water piping room is drained to the T/B sump through the floor drain. Conservatively assuming that the drain line is clogged, the flood water will not be discharged by way of the floor drain.

Leakage cracks in moderate-energy piping do not result in flooding more severe than the rupture of the feedwater piping.

- Fire Fighting Operations

  The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the above, the worst case flooding in the MS/FW piping area is a piping rupture at 12,180 ft^3. The floor area of the MS/FW piping area is 2,640 ft^2; therefore the water level caused by piping rupture area is 4.62 ft above elevation 65 ft, 0 in, the bottom of the MS/FW piping area. The actuators of valve to be protected are designed to be located at heights above the level of flood water. In addition, the bottom of doorways to the MS/FW piping area is at elevation 76 ft, 5 in. This is 11 ft, 5 in. above the floor at elevation 65 ft, 0 in, and the doorways are located at a level that is higher than the level of flood water. Therefore, the flood water flow from the MS/FW piping area to the balance of the NRCA portion of the R/B is not a consideration.

Flood events in the subject area except MS/FW piping room are considered as follows;

- Earthquake

  There is no non-seismic water-containing equipment or piping in both the east and west areas on or above this floor. The EFW pit is isolated by installing the water-tight doors to doorway to prevent flood water by sloshing of EFW pit spilling to other area.

- HELB/MELB

  HELB event is not a concern, because maximum flood level within the MS/FW piping area is well below the door elevation as described above, and there are no high energy piping breaks which are assumed to occur outside of the MS/FW areas on or above this floor.

  The maximum water volume from the MELB event is 20 ft^3 in both the east and west area.

- Fire Fighting Operations

The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the flood events described above, the worst case results are from fire fighting operations. The total volume of flood water caused by this event is 4,010 ft^3 in both the east and west area.

The footage of subject area and the water level are as follows;

- East side: 3,150 ft^2 area, 1.28 ft above elevation 76 ft, 5 in.
- West side: 3,550 ft^2 area, 1.13 ft above elevation 76 ft, 5 in.

The instrumentation of the EFW pit is designed to be located at heights above the level of flood water. The remote shutdown console is installed in the remote shutdown room. There is no piping and therefore no flooding sources inside the remote shutdown room. In addition, the remote shutdown room is protected from in-flow of water from flood sources by a water-tight door.

3.4.1.5.3 R/B Flooding Events Impacting PS/B

The US-APWR PS/B includes an east and west PS/B that are adjoined by the NRCA of R/B.

The doorways provide potential flow paths from the NRCA of R/B to the PS/B. These flooding events are evaluated on a compartment basis.

All floors in the NRCA of the R/B are divided into two areas, east and west, by concrete walls and/or water-tight doors. The floor drain water is also routed to the non-radioactive drain sump at elevation -26 ft, 4 in. The floor drains of the east areas are connected and finally go into the A-R/B non-radioactive sump. The floor drains of west areas are connected and finally go into the B-R/B non-radioactive sump. There is no cross-connection between east and west area drains. Therefore, east and west areas are evaluated as independent areas.

Elevation -26 ft, 4 in.

The equipment to be protected in the east area of PS/B at elevation -26 ft, 4 in. is the A and B train Essential Chiller Units. Equipment to be protected in the west area is the C and D train Essential Chiller Units.

In both the east and west area, each room is divided by a fire rated door instead of a water-tight door. Therefore, flood water is assumed to run across the entire area. In addition, the door to the adjoined NRCA of R/B is not a water-tight door, and the flood water from NRCA R/B is assumed to run across the PS/B.

Flood Events are considered as follows;

- Earthquake

Most of the water-containing equipment and piping in the NRCA of the R/B and PS/B are excluded from flooding source because they are designed to withstand Safe Shutdown Earthquake (SSE). The flood water volume is estimated on the basis of the amount of water contained in other non-seismic equipment or piping.

The amount of water discharged in the seismic event is 750 ft^3 in the east area and 760 ft^3 in the west area.

- HELB/MELB

HELB event is not a concern, because there is no water-containing high energy piping which is assumed to break in the subject area.

Most of the water-containing moderate energy piping in the NRCA of the R/B and the PS/B is excluded from flooding source because that piping is to be designed so that a crack is not required to be postulated in the line in accordance with the criteria described in subsection 3.6.2.1.2.2. This is attained by maintaining stress on the pipes below the threshold by means of route and support design. Portions of the water-containing moderate energy piping may be designed to be subject to crack postulation, only if it is confirmed by conservative evaluation that a crack of the parts postulated in accordance with the criteria provided in subsection 3.6.2 will not jeopardize SSCs which are required to be protected from flooding.

Refer to Table 3.4-1 for systems a part of which is designed not to crack for flood protection and systems whose piping is assumed to crack or break within the R/B or the PS/B in the internal flooding evaluation. Volume of flood water discharged from a postulated crack in the rest of the piping is conservatively estimated on the basis of the volume of water contained in the piping and any connected reservoir. The maximum water volume released in the MELB of piping routed on this elevation level is 740 ft^3 in both the east and west area.

- Fire Fighting Operations

The flooding contribution from fire fighting operations is based on the full operation of two hose stations for 2 hours. The flow rate from one hose station is 125 gpm. With two stations operating for 2 hours, the total volume of water is 4,010 ft^3.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is as follows:

- East side: 4,760 ft^3
- West side: 4,770 ft^3

The square footage of floor area subject to flooding at elevation -26 feet, 4 inches is as follows:

- East side: 11,500 ft^2 (this area includes R/B NRCA floor area)
- West side: 11,100 ft^2 (this area includes R/B NRCA floor area)

Based on these values, the maximum water level is as follows:

- East side: 0.42 ft above elevation -26 ft, 4 in.
- West side: 0.43 ft above elevation -26 ft, 4 in.

The pump foundations (top of concrete) height is 1.0 foot above floor elevation -26 ft, 4 in. As such, the pumps are not flooded. The instrumentation of each pump is designed to be located at heights above the level of flood water.

Elevation 3 ft, 7 in.

The equipment to be protected in the east area of PS/B at elevation 3 ft, 7 in. is the A and B train Class 1E GTG. Similarly, the equipment to be protected in the west area is the C and D train Class 1E GTG. Since the doorway between the corridor in the east PS/B and east area of the R/B at elevation 3 ft, 7 in. is not water-tight, flood water in the east NRCA of the R/B is assumed to flow into the east PS/B, and vice versa. The Class 1E GTG rooms are isolated from corridor of R/B NRCA and corridor of east PS/B by concrete walls and water-tight door. There are no floor drains in the Class 1E GTG rooms.

Flood events are considered as follows:

- Earthquake

  The flood water volume is estimated on the basis of the amount of water contained in non-seismic equipment or piping. The amount of water discharged in the seismic event is 10 ft^3 in both the east and west area.

- HELB/MELB

  HELB event is not a concern, because there is no water-containing high energy piping which is assumed to break in the subject area.

  The maximum water volume from the MELB event is 20 ft^3 in both the east and west area.

- Fire Fighting Operations

  The total water volume from the fire fighting operation events is same as that of elevation -26 ft, 4 in.

Based on the flood events described above, the worst case results are from a combination of earthquake and fire fighting operations. The total volume of flood water caused by this combination is 4,020 ft^3 in both the east and west area.

The footage of corridor area and the water level are as follows:

- East side: 4,300 ft^2 area, 0.94 ft above elevation 3 ft, 7 in.
- West side: The entire floor of the west area of PS/B at elevation 3 ft, 7 in. consists of water tight compartments.

Class 1E GTG are installed in the room which prevents flow-in water by water-tight door.

Therefore, GTG room is not flooded.

3.4.2 Analysis Procedures

The static and dynamic effects of the design-basis flood or groundwater conditions, which are identified in Section 2.4, are applied to seismic category I structures. Section 3.8 specifies the applicable codes, standards, and specifications used in the design of seismic category I structures. The loads and load combination subsections of Section 3.8 take into consideration the static and dynamic loadings on seismic category I structures including hydrostatic loading as the result of the design-basis flood and/or ground conditions identified in Section 2.4. Section 3.8 also provides the design and analysis procedures used to transform the static and dynamic effects of the DBFL and ground water levels applied to seismic category I structures to assure their design meet the applicable acceptance criteria.

The COL Applicant is to identify any site-specific physical models used to predict prototype performance of hydraulic structures and systems involving an unusual design or configuration, or for a design or operating bases involving thermal and erosion problems.

3.4.3 Combined License Information

COL 3.4(1) *The COL Applicant is to address the site-specific design of plant grading and drainage.*

COL 3.4(2) *The COL Applicant is to demonstrate the DBFL bounds their specific site, or is to identify and address applicable site conditions where static flood level exceed the DBFL and/or generate dynamic flooding forces.*

COL 3.4(3) *Site-specific flooding hazards from engineered features, such as from cooling water system piping, is to be addressed by the COL Applicant.*

COL 3.4(4) *The COL Applicant is to address any additional measures below grade to protect against exterior flooding and the intrusion of ground water into seismic category I buildings and structures.*

COL 3.4(5) *The COL Applicant is to identify and design, if necessary, any site-specific flood protection measures such as levees, seawalls, floodwalls, site bulkheads, revetments, or breakwaters per the guidelines of RG 1.102 (Reference 3.4-3), or dewatering system if the plant is not built above the DBFL.*

COL 3.4(6) *The COL Applicant is to identify any site-specific physical models used to predict prototype performance of hydraulic structures and systems.*

COL 3.4(7) *The COL Applicant is responsible for the protection from internal flooding for those site-specific SSCs that provide nuclear safety-related functions or whose postulated failure due to internal flooding could adversely affect the ability of the plant to achieve and maintain a safe shutdown condition.*

COL 3.4(8) *The COL Applicant is responsible for developing inspection and testing procedures in accordance with manufacturer recommendations so that each water-tight door remains capable of performing its intended function.*

3.4.4 References

3.4-1 Reactor Site Criteria, Energy. Title 10 Code of Federal Regulations Part 100, U.S. Nuclear Regulatory Commission, Washington, DC.

3.4-2 General Design Criteria for Nuclear Power Plants, Domestic Licensing of Production and Utilization Facilities, Energy. Title 10 Code of Federal Regulations Part 50, Appendix A, U.S. Nuclear Regulatory Commission, Washington, DC.

3.4-3 Flood Protection for Nuclear Power Plants. Regulatory Guide 1.102, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, September 1976.

3.4-4 Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 14.3.2, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.4-5 Design Basis Floods for Nuclear Power Plants. Regulatory Guide 1.59, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, August 1977.

3.4-6 Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors. ANSI/ANS 56.10-1987, Section 3, American National Standards Institute /American Nuclear Society.

3.4-7 Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants. ANSI/ANS-56.11-1988, American National Standards Institute /American Nuclear Society.

3.4-8 US-APWR Evaluation and Design Enhancement to Incorporate Lessons Learned from TEPCO's Fukushima Dai-ichi Nuclear Power Station Accident, MUAP-13002 Rev.0, March 2013.

Table 3.4-1 No crack design and crack/break postulation for moderate energy piping for flood protection design

Systems	No Crack Design for Flood Protection(1)	Crack or Break Postulation in Flooding Evaluation(2)
Reactor Coolant System (RCS)	X	-
Chemical and Volume Control System (CVCS)	X	X
Safety Injection System (SIS)	X	-
Residual Heat Removal System (RHRS)	X	-
Emergency Feedwater System (EFWS)	X	X
Containment Spray System (CSS)	X	-
Component Cooling Water System (CCWS)	X	X
Spent Fuel Pit Cooling and Purification System (SFPCS)	X	X
Essential Service Water System (ESWS)	X	-
Liquid Waste Management System (LWMS)	X	X
Process and Post-Accident Sampling System (PSS)	-	X
Steam Generator Blowdown System (SGBDS)	X	-
Refueling Water Storage System (RWS)	X	-
Primary Makeup Water System (PMWS)	X	X
Demineralized Water System (DWS)	X	-
Fire Protection Water Supply System (FSS)	X	-
Potable and Sanitary Water System (PSWS)	X	-
Essential Chilled Water System (ECWS)	X	-
Non-Essential Chilled Water System (non-ECWS)	X	X

Notes

1. Systems a part of which is designed so that a crack is not required to be postulated in the line in accordance with the criteria described in subsection 3.6.2.1.2.2 for the purpose of flood protection.

2. Systems which include moderate energy lines that are assumed to crack or break within the R/B or the PS/B and to be flooding source in internal flooding evaluation.

3.5 Missile Protection

GDC 4 of Appendix A to 10 CFR 50 (Reference 3.5-1) requires safety-related SSCs to be protected from the effects of missiles. This includes all SSCs within containment and the containment itself. The containment is defined for the US-APWR as the PCCV. In addition, GDC 2 of Appendix A to 10 CFR 50 (Reference 3.5-1) also requires that safety-related SSCs be designed to withstand the effects of natural phenomena, which includes missiles potentially generated by tornadoes and hurricanes and similar extreme winds.

In accordance with GDC 2 and GDC 4 of 10 CFR 50, the safety-related areas of the US-APWR contain SSCs that provide the capability to safely shut down the reactor and maintain it in a safe-shutdown condition while also protecting the integrity of the RCPB and maintaining offsite radiological dose/concentration levels within the limits defined in 10 CFR 100 (Reference 3.5-2).

The SSCs to be protected from postulated missiles are identified in the Appendix of RG 1.117, Tornado Design Classification (Reference 3.5-19), and summarized by the following:

1. The RCPB.

2. Those portions of the MSS and main feedwater system up to and including the outermost isolation valves.

3. The reactor core and individual fuel assemblies at all times, including during refueling.

4. Systems or portions of systems that are required for (1) attaining safe shutdown; (2) RHR; (3) cooling the SFP; (4) mitigating the consequences of a tornado- or hurricane-caused steam line break; (5) primary makeup water system; and (6) supporting the above systems, such as essential service water, UHS, air supply, EFW, and safety-related ventilation systems.

5. The SFP, to the extent necessary to preclude significant loss of watertight integrity of the storage pit, and to prevent missiles from contacting fuel within the pit.

6. The reactivity control systems, e.g., control rod drives and boron system.

7. The MCR, including all equipment needed to maintain the MCR within safe habitability limits for personnel and safe environmental limits for tornado/hurricane-protected equipment.

8. Those portions of the gaseous waste management system whose failure due to tornado effects or hurricane effects could result in potential offsite exposures greater than the 25% of the guideline exposures of 10 CFR 100 using appropriately conservative analytical methods and assumptions.

9. Systems or portions of systems that are required for monitoring, actuating, and operating tornado/hurricane-protected portions of systems listed in items 4, 6, 7, and 13.

10. All electric and mechanical devices and circuitry between the process sensors and the input terminals of the actuator systems involved in generating signals that initiate protective actions by tornado/hurricane-protected portions of systems listed in items 4, 6, 7, and 13.

11. Those portions of the long-term ECCS that would be required to maintain the plant in a safe condition for an extended time after a LOCA.

12. PCCV and other safety-related structures, such as the R/B and PS/Bs, to the extent that they not collapse, allow perforation by missiles, or generation of secondary missiles, any of which could cause unacceptable damage to tornado/hurricane-protected items. However, the primary containment need not necessarily maintain its leaktight integrity.

13. The Class 1E electric systems, including the auxiliary systems for the onsite electric power supplies, that provide the emergency electric power needed for the functioning of plant features included in items 1 through 11 above.

14. Those portions of SSCs whose continued function is not required but whose failure could reduce to an unacceptable safety level the functional capability of any plant features included in items 1 through 13 above or could result in incapacitating injury to occupants of the MCR.

Missiles are postulated to be associated with failures of pressurized high-energy fluid system components, over-speed failures of rotating machinery (e.g., motor-driven pumps and fans), explosions within and outside the plant, falling objects, including falling objects resulting from a non-seismically designed SSC during a seismic event, and by tornados, hurricanes or transportation accidents external to the plant. This section discusses missile protection for the following sources:

- Internally generated missiles (Outside PCCV)
- Internally generated missiles (Inside PCCV)
- Turbine missiles
- Missiles generated by tornadoes and hurricanes
- Site proximity missiles (Except aircraft)
- Aircraft hazards

Missiles that could prevent SSCs from performing their intended safety functions may be statistically significant. Potential missile sources are identified and statistically evaluated in subsequent subsections using the following methodology:

1. When a potential missile source is identified, the statistical significance of missile generation is evaluated by a probability analysis. The probability of occurrence (P_1) of generating a missile by any source is not statistically significant if it is less than 10^{-7} per year.

2. When the probability of occurrence, P_1, is greater than 10^{-7} per year for any potential missile source, the probability of impact (P_2) on a significant target is also determined. When considering both the probability of missile occurrence and the probability of missile impact, the missile is not statistically significant if the product of P_1 and P_2 is less than 10^{-7} per year. If the product of P_1 and P_2 is greater than 10^{-7} per year, the probability of significant damage (P_3) is determined.

3. For those cases where the product of P_1 and P_2 is greater than 10^{-7} per year, the missiles are evaluated for the probability of significant damage (P_3) based on the size, energy, and trajectory of the postulated missile, and the proximity to any potentially impacted SSCs. Alternately, an evaluation is performed to determine if sufficient redundancy remains to achieve and maintain a safe shutdown condition. No additional missile protection is required if the evaluations determine that the ability to achieve and maintain safe shutdown is maintained. If the combined probability ($P_1 \times P_2 \times P_3$) is less than 10^{-7} per year, the potential missile is not considered statistically significant.

Therefore, factors contributing to missile protection of potentially targeted SSCs is provided by one or more of the following methods:

- Locating the system or component in a missile-proof structure
- Separating redundant systems or components for the missile path or range
- Providing local shields and barriers for systems and components
- Designing the equipment to withstand the impact of the most damaging missile
- Providing design features to prevent the generation of missiles
- Orienting missile sources to prevent missiles from striking safety-related equipment
- When necessary, missile barriers are designed in accordance with Subsection 3.5.3.

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

Section 3.2 and Section 3.11 list applicable SSCs, their location, seismic category, and quality group/equipment classifications. General arrangement drawings showing locations of the SSCs are given in Section 1.2. The following component types have the potential to produce internally-generated missiles outside the PCCV.

3.5.1.1.1 Items Containing Pressurized High-energy Fluids or Steam

Items forming a pressurized boundary of high-energy fluid or steam may be postulated to have been damaged, broken pieces could become missiles when propelled by internal pressure or jet forces.

3.5.1.1.1.1 Piping

For potential missiles originating from piping under high pressure, the probability of occurrence, P_1, is maintained less than 10^{-7} by virtue of the design in accordance with ASME Code, Section III (Reference 3.5-3), and inspection program in compliance with ASME Code, Section XI (Reference 3.5-4). If piping as evaluated in Section 3.6 were to rupture, the pipe is held in place by its supports. However, the probability of occurrence, P_1, remains less than 10^{-7} since the section remains attached to the remainder of the piping system.

Pressurized high-energy piping systems and components, if not constructed to ASME Code, Section III criteria (Reference 3.5-3), can be a significant source of missiles (that is, $P_1 > 10^{-7}$). The probability of impact, P_2, is therefore also considered. The probability of impact, P_2, is minimized by locating a potential missile source or potential target outside the zone of postulated missile strike, by the robust building walls and slabs that are designed for applicable missile strikes, or the separation of piping systems and components that are missile sources from potentially impacted SSCs. When evaluating the credibility of missile impact, the product of $P_1 \times P_2$ is less than 10^{-7} and therefore non-ASME Code piping systems and components are not credible missile sources.

For non-high energy fluid systems, the systems have insufficient stored energy to generate a missile. The probability of missile occurrence, P_1, from non-high energy fluid systems is therefore less than 10^{-7}.

3.5.1.1.1.2 Valves

In the case of postulated damage to threads, the stem (valve stem) will not eject because the backseat or valve body is larger in diameter than the stem. Therefore, the probability of missile occurrence, P_1, originating from an ejected valve stem is less than 10^{-7}.

In valves with bolted bonnets (covers), bonnets can not be perforated because the remaining bolts withstand the internal pressure even when one bolt is assumed to have been damaged. In valves of ANSI 900 Pressure Class and above, valves using bonnet types with a pressure seal will not perforate because they are pressed by a yoke or retainer ring (cover retaining ring). The valve design is in accordance with ASME Code, Section III (Reference 3.5-3), and inspected in accordance with ASME Code, Section XI (Reference 3.5-4). Therefore, the probability of missile occurrence, P_1, originating from a pressure seal-style valve bonnet is less than 10^{-7}.

In valves of threaded bonnets having canopy seals commonly used for ANSI 600 Pressure Class and below, the bonnets will not perforate due to loose threading, and have a historically low occurrence of total separation of the bonnet from the valve. Therefore, the probability of missile occurrence, P_1, originating from a threaded valve bonnet is less than 10^{-7}.

3.5.1.1.1.3 Pipe Fittings and Appurtenances

For potential missiles originating from pipe fittings and appurtenances under high pressure, the probability of occurrence, P_1, is maintained less than 10^{-7} for those components whose design is in accordance with ASME Code, Section III (Reference 3.5-3), and inspection program is in compliance with ASME Code, Section XI (Reference 3.5-4). Pipe fittings and appurtenances that may be a high probability of occurrence (that is, $P_1 > 10^{-7}$) are further qualified as statistically insignificant when the probability of missile impact on a critical component, P_2, is such that the product of $P_1 \times P_2$ is less than 10^{-7}. In addition, any postulated missile involving a pipe fitting or appurtenance is a small mass with a very small probability of significant damage (P_3). Therefore, a combined probability ($P_1 \times P_2 \times P_3$) of any postulated missile generated by a pipe fitting or appurtenance is less than 10^{-7} per year, and the potential missile is not considered statistically significant.

3.5.1.1.2 High-Speed Rotating Equipment

In general, the probability of occurrence, P_1, of safety-related rotating equipment is maintained less than 10^{-7} by virtue of the equipment design and manufacturing criteria. Justification for a low probability of occurrence, P_1, includes the fact that rotating equipment energized by ac power are governed by the frequency of the power supply. The narrow range of frequency variation for the ac power supply makes it highly unlikely that an overspeed condition of rotating equipment can occur. While it is postulated that missiles such as a fan blade may occur at rated speeds, the design of the casing prevents missile penetration. However, in the unlikely case of a high-speed rotating component penetrating the casing and P_1 is greater than 10^{-7}, the probability of impact, P_2, is also considered. P_2 is minimized by locating a potential missile source or potential target outside the zone of postulated missile strike, by the robust building walls and slabs that are designed for applicable missile strikes, or the separation of missile sources from potentially impacted SSCs. When considering both probability of occurrence and probability of impact, the product of $P_1 \times P_2$ is less than 10^{-7} and therefore high-speed rotating equipment are not credible missile sources.

Missiles are similarly not postulated from turbine-driven pumps because of the over-speed prevention system and deliberate quality assurance consideration for the inspection of materials, design, production, installation, and operation. The probability of occurrence, P_1, for turbine-driven pumps is therefore maintained less than 10^{-7}.

Missiles are also not postulated from the GTG because of the over-speed prevention system, deliberate quality assurance considerations in the inspection of materials, design, production, installation, and operation, and the casing material that prevents penetration. The probability of occurrence, P_1, for the GTG is therefore maintained less than 10^{-7}.

In the case of nonsafety-related high-speed rotating pumps, motors with thick casings are procured to prevent the probability of missile occurrence. Therefore, the probability of missile occurrence, P_1, originating from a nonsafety-related high-speed rotating pump is less than 10^{-7}.

In the case of nonsafety-related high-speed rotating fans, the probability of missile occurrence may be statistically significant ($P_1 > 10^{-7}$). When investigating these components, the probability of impact, P_2, is also evaluated to confirm that the product of $P_1 \times P_2$ is less than 10^{-7}. The probability of impact, P_2, is minimized by locating a potential missile source or potential target outside the zone of postulated missile strike, by the robust building walls and slabs that are designed for applicable missile strikes, and/or the separation of the rotating equipment that is a missile source from potentially impacted SSCs. Therefore, high-speed rotating fans are not a credible missile source since the product of $P_1 \times P_2$ is less than 10^{-7}.

Refer to Subsection 3.5.1.3 for discussion of turbine and turbine rotor missiles.

3.5.1.1.3 Gas or Pressurized Cylinder Explosion

Protective measures are taken as recommended by NUREG/CR-3551 (Reference 3.5-20), including procedures, analysis, and design details, to mitigate pressurized gas cylinders/bottles from generating or becoming a missile. Design features which resist the formation of missiles from a pressurized gas cylinder/bottle include the fabrication from rolled thick-wall steel material, and a steel collar at the neck of the bottle to protect the sensitive valve and other critical parts. In addition, the pressurized cylinders are oriented vertically with the bottle pointed towards the concrete slab roof in storage racks restrained in accordance with seismic category II requirements. Therefore, the product of the probability of occurrence, P_1, and the probability of impacting a significant target, P_2, is less than 10^{-7}.

Battery compartments are ventilated to prevent the concentration of hydrogen. The hydrogen supply system and gas bottles are installed in a compartment independent of safety-related structures, and ventilation is provided to prevent the concentration of hydrogen. The probability of occurrence, P_1, for a gas explosion in battery compartments is therefore maintained less than 10^{-7}.

3.5.1.1.4 Gravitational Missiles

The COL Applicant is to have plant procedures in place prior to fuel load that specify unsecured equipment, including portable pressurized gas cylinders, located inside or outside containment and required for maintenance or undergoing maintenance is to be removed from containment prior to operation, moved to a location where it is not a potential hazard to safety-related SSCs, or seismically restrained to prevent it from becoming a missile.

3.5.1.1.4.1 Crane Drop of Heavy Loads

As defined in ASME NOG-1 (Reference 3.5-5), a critical load is any lifted load whose uncontrolled movement or release could adversely affect any safety-related SSC when such a SSC is required for plant safety or could result in potential offsite exposure in excess of 10 CFR 100 limits.

Type I cranes are defined by ASME NOG-1 (Reference 3.5-5) as those used to handle critical loads. In accordance with ASME NOG-1, Type I cranes are designed to remain in place and are equipped with single failure-proof features to prevent load drops.

Refer to Subsection 9.1.5.1 for further discussion on the design bases for a postulated load drop by the overhead heavy load handling system.

Additionally, cranes are designed to prevent diversion and derailment. Drop prevention design is also employed based on earthquake design criteria. Therefore, load drops and derailment of cranes do not represent credible sources of missiles that would jeopardize safety-related SSCs. Therefore, the probability of occurrence, P_1, of missiles generated by a gravity load crane drop is less than 10^{-7}.

3.5.1.1.4.2 Falling Objects Resulting from Non-Seismic SSCs During Seismic Event

Seismic category II SSCs are defined as not essential for the safe shutdown of the plant, and need not remain functional during and after a SSE. However, such structures and subsystems must not fall or displace excessively where it could damage any seismic category I SSCs. Therefore, any SSCs with the potential to cause damage to safety-related SSCs are analyzed and designed using the same methods and stress limits specified for seismic category I SSCs. No non-seismic SSCs are permitted that could possibly affect the ability of the plant to achieve and maintain safe shutdown, and to maintain offsite radiological dose/concentration levels within defined limits. In addition, seismic category I SSCs are not permitted in non-seismic areas and are therefore not impacted by falling objects during a seismic event. Therefore, the product of the probability of occurrence, P_1, and the probability of significant impact, P_2, of non-seismic SSCs missiles striking seismic category I SSCs is maintained as less than 10^{-7}.

3.5.1.1.4.3 Secondary Missiles Caused by a Falling Object Striking a High-Energy System

Falling objects impacting a high-energy system or other surfaces may have the ability to generate secondary missiles. Falling objects are postulated to occur by either a crane drop of heavy load, or resulting from a non-seismic SSC during a seismic event. As described in the preceding paragraphs, these missiles sources are not credible, and no secondary missiles from these sources are capable of occurring. The probability of occurrence, P_1, is therefore inherently less than 10^{-7}.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Section 3.2 and Section 3.11 list applicable SSCs, their location, seismic category, and quality group/equipment classifications. General arrangement drawings showing locations of the SSCs are given in Section 1.2. The following component types have the potential to produce internally-generated missiles inside the PCCV.

3.5.1.2.1 Items Containing Pressurized High-energy Fluids or Steam

Subsection 3.5.1.1.1 discusses items containing pressurized high-energy fluids or steam outside containment. Conclusions relating to statistical significance of postulated missiles also apply to similar items containing pressurized high-energy fluids or steam inside containment. Pressurized items unique to inside containment are discussed as follows.

Inside the PCCV, postulated missiles originating from unique pressurized high-energy items such as the RV and associated fittings, SG, reactor coolant pump (RCP), pressurizer, and RCPB piping during normal operation are not considered credible due to ASME Code Section III (Reference 3.5-3) and Section XI (Reference 3.5-4) criteria controlling quality from production through operation, material characteristics, design strengths, and the preservice and inservice inspections. Additional assurances to prevent generation of missiles are provided by prudent operation of the system. The probability of missile occurrence from pressurized high energy items inside containment, P_1, is therefore less than 10^{-7}.

Additionally, postulated missiles in the form of a piece of the CRDM pressure housing or a control rod ejected rapidly from the core is not considered credible. In addition to a low probability of occurrence similar to the RV, the following assurances specific to the CRDMs maintain a low probability of occurrence, P_1, and low probability of impact, P_2, provided by:

- Shop hydro-testing in excess of 125% of system design pressure.
- Hydro-testing of housings to 125% of system design pressure after they are installed on the RV to the head adapters. Housings are also tested during hydro-testing of the completed RCS.
- Housings are made of materials with excellent notch toughness.

- Stress levels in the mechanism are not affected by system thermal transients at power or by thermal movement of the reactor coolant loops (RCLs).
- The welds in the pressure boundary of the CRDM satisfy ASME Code, Section III (Reference 3.5-3) requirements for design, procedure, examination, and inspection.
- A control rod ejection is considered in the safety analyses in Chapter 15, and the design transients in Subsection 3.9.1.1.

Therefore, the product of probability of missile occurrence, P_1, and probability of impact, P_2, is less than 10^{-7} for pressurized high energy items inside containment.

3.5.1.2.2 High-Speed Rotating Equipment

Subsection 3.5.1.1.2 discusses high-speed rotating equipment. Conclusions relating to statistical significance of postulated missiles also apply to similar high-speed rotating equipment inside containment. In addition, the RCP is an item unique to inside containment. The RCP and associated flywheel is designed to prevent the probability of missile occurrence by quality control, inservice inspection, and continuous monitoring for shaft vibration. The maximum allowable rotating speed in terms of the strength of the flywheel is sufficiently higher than the maximum rotating speed of the motor postulated at the plant, and the soundness of the flywheel is maintained. Therefore, the probability of missile occurrence from high-speed rotating equipment inside containment, P_1, is less than 10^{-7}.

3.5.1.2.3 Gas or Pressurized Cylinder Explosion

Conclusions relating to statistical significance of postulated missiles due to gas or pressurized cylinder explosion also apply inside containment. By an analysis similar to that in Subsection 3.5.1.1.3, it is concluded that no items have the capability of generating potential missiles related to a gas or pressurized cylinder explosion inside the containment. Therefore, the product of the probability of occurrence, P_1, and the probability of impacting a significant target, P_2, is less than 10^{-7}.

3.5.1.2.4 Gravitational Missiles

Subsection 3.5.1.1.4 discusses gravitational missiles, including crane drop of heavy loads, falling objects resulting from non-seismic SSCs during seismic event, and secondary missiles caused by a falling object striking a high-energy system. Conclusions relating to statistical significance of these postulated missiles also apply to similar potential gravitational missiles inside containment. Therefore, the probability of missile occurrence, P_1, or the product of P_1 and the probability of impacting a significant target, P_2, is less than 10^{-7}.

3.5.1.3 Turbine Missiles

The two broad categories of turbine failures are referred to as design over-speed and destructive over-speed failures. Missiles resulting from design over-speed failures are the result of brittle fracture of turbine blade wheels or portions of the turbine rotor itself. Failures of this type can occur during startup or normal operation. Missiles resulting from destructive over-speed failures would be generated if the over-speed protection system malfunctions and the turbine speed increases to a point at which the low-pressure wheels or rotor undergo ductile failure.

3.5.1.3.1 Geometry

As defined by “Protection Against Low-Trajectory Turbine Missiles”, RG 1.115, Rev. 1 (Reference 3.5-6), current evidence suggests low trajectory turbine missile strikes are concentrated within an area bounded by lines inclined at 25 degrees to the turbine wheel planes and passing through the end wheels of the low pressure stages.

The T/G is located on the plant south side of the nuclear island with its long-axis aligned in the plant north-south direction. In this orientation, the R/B, PCCV, PS/Bs, and safety-related and non-safety related SSCs located within these structures within the same unit are located such that the T/G is in a favorable orientation in accordance with NUREG-0800 Standard Review Plan Section 3.5.1.3 (Reference 3.5-7). The T/G and associated equipment with respect to essential safety-related SSCs of the standard plant design are shown in figures found in Section 1.2.

The COL applicant is to identify the site-specific systems and components to be protected and to assess the orientation of the T/G with respect to the essential site-specific SSCs using the guidance and examples in RG 1.117 (Reference 3.5-19) and use the evaluation method described in Subsection 3.5.1.3.2 to determine the probability that a turbine missile will cause unacceptable damage to an essential SSC. For multi-unit sites, the COL applicant is to also evaluate the effect of other unit(s) on this unit as specified in RG 1.115.

3.5.1.3.2 Evaluation

Protection against damage from turbine missiles to the R/B, PCCV, PS/Bs, and safety-related SSCs located within these structures is provided by the orientation of the T/G as described in Subsection 3.5.1.3.1, by the robust turbine rotors, and by the redundant and fail-safe turbine design control system as described in Section 10.2. The rotor design, material selection, preservice and inservice programs and redundant control system support a very low probability of turbine missile generation. The turbine rotor design is discussed in Subsection 10.2.3, in which material selection, fracture toughness/fracture analysis is discussed. Description of the inservice inspection and testing program that will be used to maintain an acceptably low probability of missile generation is also given in Subsection 10.2.3.

The probability of unacceptable damage resulting from turbine missiles, P_4, is expressed as the product of (a) the probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing, P_1; (b) the probability of ejected missiles perforating intervening barriers and striking safety-related SSCs, P_2; and (c) the probability of struck SSCs failing to perform their safety function, P_3.

Mathematically, $P_4 = P_1 \times P_2 \times P_3$ where RG 1.115 (Reference 3.5-6) considers an acceptable risk rate for P_4 as less than 10^{-7} per year. The determination of P_1 (probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing) is strongly influenced by the program for periodic inservice testing and inspection. Criteria as described in NUREG-0800 Standard Review Plan 3.5.1.3, Table 3.5.1.3-1 (Reference 3.5-7) correlates P_1 to operating cases necessary to obtain P_4 in an acceptable risk rate of 10^{-7} per year, where P_1 is less than $P_4 / (P_2 \times P_3)$ or 10^{-4}. The P_1 applicable to the US-APWR is described in Subsection 10.2.2. The COL Applicant is to commit to actions to maintain P_1 within this acceptable limit as outlined in RG 1.115, “Protection Against Low-Trajectory Turbine Missiles” (Reference 3.5-6) and SRP Section 3.5.1.3, “Turbine Missiles” (Reference 3.5-7). To maintain an acceptably low P_1, MUAP-07028-NP, “Probability of Missile Generation From Low Pressure Turbines” (Reference 3.5–17) and MUAP-07029-NP, “Probabilistic Evaluation of Turbine Valve Test Frequency” (Reference 3.5-18) are to be used to establish programs and criteria for preservice inspection, inservice inspection interval and turbine valve test frequency. These inspections and tests *[maintain the turbine missile generation probability, P_1, at less than 1×10^{-5} per year.]** Inservice inspection programs are to be maintained as outlined in SRP 3.5.1.3, Section II, Acceptance Criteria, Section 4 (Reference 3.5-7).

Information in this subsection that is italicized and enclosed in square brackets with an asterisk following the closing bracket is a special category of information designated by the NRC as Tier 2. Any change to this information requires prior NRC approval.*

3.5.1.4 Missiles Generated by Tornadoes and Hurricanes

The US-APWR design basis spectrum of tornado missiles conforms to the spectrum of missiles defined in Table 2 of “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants”, RG 1.76, Rev.1 (Reference 3.5-8) for a region I tornado, the most severe. The spectrum of missiles is chosen to represent: (1) a massive high-kinetic-energy missile that deforms on impact, (2) a rigid missile that tests penetration resistance, and (3) a small rigid missile of a size sufficient to pass through any opening in protective barriers.

Therefore, the spectrum of tornado missiles is as follows:

- A 4,000 pound automobile, 16.4 ft by 6.6 ft by 4.3 ft, impacting the structure at normal incidence with a horizontal velocity of 135 ft/s or a vertical velocity of 90.5 ft/s. To accommodate site-specific conditions where grades within 0.5 mile of plant structures may have elevations higher than grade at the structures, this missile is considered to potentially impact SSCs at any azimuthal direction and at any elevation up to the lowest roof level of R/B surrounding PCCV, which is 98'-5" above grade, at the maximum tornado missile velocity stated above.

- A 6.625 inch diameter by 15 ft long schedule 40 pipe, weighing 287 pounds, impacting the structure end-on at normal incidence with a horizontal velocity of 135 ft/s or a vertical velocity of 90.5 ft/s.

- A 1 inch diameter solid steel sphere assumed to impinge upon barrier openings in the most damaging direction with a horizontal velocity of 26 ft/s or a vertical velocity of 17.4 ft/s.

The US-APWR design basis spectrum of hurricane missiles conforms to the spectrum of missiles in Table 1 and Table 2 of "Design Basis Hurricane and Hurricane Missiles for Nuclear Power Plants", RG 1.221 (Reference 3.5-21) and "Technical Basis for Regulatory Guidance on Design-Basis Hurricane-Borne Missile Speeds for Nuclear Power Plants", NUREG/CR-7004 (Reference 3.5-22) with a design basis hurricane wind speed of 160 mph.

The spectrum of missiles is the following: (1) a massive high-kinetic-energy missile that deforms on impact, (2) a rigid missile that tests penetration resistance, and (3) a small rigid missile of a size sufficient to pass through any opening in protective barriers.

Therefore, the spectrum of hurricane missiles is as follows:

- A 4,000 pound automobile, 16.4 ft by 6.6 ft by 4.3 ft, impacting the structure at normal incidence with a horizontal velocity of 135 ft/s or a vertical velocity of 85 ft/s. To accommodate site-specific conditions where grades within 0.5 miles of plant structures may have elevations higher than grade at the structures, this missile is considered to potentially impact SSCs at any azimuthal direction and at any elevation up to the lowest roof level of R/B surrounding PCCV, which is 98'-5" above grade, at the maximum hurricane missile velocity stated above.

- A 6.625 inch diameter by 15 ft long schedule 40 pipe, weighing 287 pounds, impacting the structure end-on at normal incidence with a horizontal velocity of 102 ft/s or a vertical velocity of 85 ft/s.

- A 1 inch diameter solid steel sphere assumed to impinge upon barrier openings in the most damaging direction with a horizontal velocity of 89 ft/s or a vertical velocity of 85 ft/s.

Openings through the exterior walls of the seismic category I structures, and the location of equipment in the vicinity of such openings, are arranged so that a missile passing through the opening would not prevent the safe shutdown of the plant and would not result in an offsite release exceeding the limits defined in 10 CFR 100 (Reference 3.5-2). Otherwise, structural barriers are designed to resist missiles in accordance with the design procedures discussed in Subsection 3.5.3. Tornado missiles and hurricane missiles are not postulated to ricochet or strike more than once at a target location. Missile protection is provided to resist the normal component of force delivered by the missile striking in any direction. Additional tornado loading design requirements are addressed in Subsections 3.3.2 and 3.8.4. Due to the robustness of the exterior wall design, all seismic category I structures other than PCCV are designed to withstand the impact of each identified tornado missile and hurricane missile at any elevation, including the potential impact of a 4,000 pound automobile more than 30 feet above grade up to the lowest roof level of R/B surrounding PCCV, which elevation is 98'-5" above grade.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

Externally initiated missiles considered for the US-APWR standard design are based on tornado missiles and hurricane missiles as described in Subsection 3.5.1.4. As described in DCD, Section 2.2, the COL Applicant is to establish the presence of potential hazards, except aircraft, which is reviewed in Subsection 3.5.1.6, and the effects of potential accidents in the vicinity of the site. The RG followed is identified, and any deviations from this guidance or any alternative methods that are used are explained or justified. The information also describes the data collected, analyses performed, results obtained, and any previous analyses and results cited to justify any of the conclusions. Additional analyses may be required to evaluate other potential site-specific missiles.

3.5.1.6 Aircraft Hazards

The US-APWR standard plant design basis is that the plant is located such that an aircraft crash and air transportation accidents are not required to be considered as part of the design basis. It is the responsibility of the COL Applicant to verify the site interface parameters with respect to aircraft crashes and air transportation accidents as described in Section 2.2. Additional analyses may be required to evaluate potential aircraft missiles.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

Safety-related SSCs are identified in Section 3.2 and Section 3.11. Protection of these systems from external missiles is provided by the external walls and roof of the safety-related R/B and PS/Bs. The external walls and roofs are reinforced concrete. The structural design requirements for the R/B and PS/Bs are outlined in Subsection 3.8.4.

Openings through exterior walls of the seismic category I structures are evaluated as described in Subsection 3.5.1.4 to provide confidence that a missile passing through the opening would not prevent safe shutdown and would not result in an offsite release exceeding the limits defined in 10 CFR 100 (Reference 3.5-2). The COL Applicant is responsible to evaluate site-specific hazards for external events that may produce missiles more energetic than tornado missiles and hurricane missiles for the standard plant presented in Subsection 3.5.1.4, and assure that the design of seismic category I and II structures meet these loads.

3.5.3 Barrier Design Procedures

If required, components, protective shields, and missile barriers are designed to prevent damage to safety-related components by absorbing and withstanding missile impact loads. The target SSCs, shields, and barriers are evaluated for both local effects and overall structural effects due to missile impacts. The local effects in the impacted area are evaluated to predict the minimum thickness required for steel structures and for concrete structures to prevent perforation and the potential generation of secondary missiles by spalling or scabbing effects. A review of the structure for overall response is conducted to estimate forces, moments and shears induced in the barrier by the impact force of the missile.

3.5.3.1 Evaluation of Local Structural Effects

The following subsections address the design of structures to withstand and absorb missile impact loads. Formulas are provided to predict the penetration depth (x), scabbing thickness (t_s) and perforation thickness (t_p) potential created by the missile impact. Safety factors are then applied to determine required barrier thicknesses to restrict missile penetration, scabbing and/or perforation. It is assumed that the missile impacts normal to the plane of the wall on a minimum impact area and, in the case of reinforced concrete, its resistance does not credit capacity of struck reinforcing.

3.5.3.1.1 Concrete

The National Defense Research Council (NDRC) provides "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects", by R. P. Kennedy (Reference 3.5-9). Wall and roof thicknesses satisfy minimum barrier thicknesses provided in Table 1 of NUREG-0800, SRP 3.5.3 (Reference 3.5-10) to prevent local damage from tornado-generated missiles. The same design methodology as used to determine minimum wall and roof thicknesses for tornado-generated missiles is used for hurricane-generated missiles.

Modified NDRC Formula

$$x = [4\,KNW_m d\,(V/1000d)^{1.8}]^{0.5} \quad \textit{for } x/d \leq 2.0$$

$$x = KNW_m\,(V/1000d)^{1.8} + d \quad \textit{for } x/d > 2.0$$

where

x = penetration depth, inches

W_m = missile weight, pounds

d = missile diameter, inches

N = missile shape factor: N = 0.72 for flat nose body; N = 0.84 for blunt nose body; N = 1.0 for average bullet nose (spherical end); N = 1.14 for very sharp nose

V = impact velocity, ft/s

K = experimentally obtained material coefficient for penetration = $180/(f_c')^{0.5}$

f_c' = concrete compressive strength

Scabbing thickness, t_s, and perforation thickness, t_p is given by

$t_s / d = 2.12 + 1.36\ x / d$ for $0.65 \leq x / d \leq 11.67$

$t_s / d = 7.91\ (x / d) - 5.06\ (x / d)^2$ for $x / d \leq 0.65$

$t_p / d = 1.32 + 1.24\ (x / d)$ for $1.35 \leq x / d \leq 13.5$

$t_p / d = 3.19\ (x / d) - 0.718\ (x / d)^2$ for $x / d \leq 1.35$

In order to provide a sufficient safety margin, the design thickness (t_d) is 20% greater than the threshold value for the phenomenon being prevented as follows:

To prevent perforation, the design thickness, $t_d = 1.2\ t_p$

To prevent scabbing, the design thickness, $t_d = 1.2\ t_s$

3.5.3.1.2 Steel

The formula by the Ballistic Research Laboratory (BRL Formula) available in "Reactor Safeguards" (Reference 3.5-12) is utilized to determine the steel plate thickness for threshold of perforation. In addition, the results of tests by Stanford Research Institute (SRI) summarized in "US Reactor Containment Technology" (Reference 3.5-11) also establishes an equation (SRI Formula) for the steel plate thickness for threshold of perforation. The steel plate thickness for perforation threshold is to satisfy both BRL and SRI formulas.

BRL Formula

$$T_p = (E_k)^{2/3} / (672\ D)$$

where

T_p = steel plate thickness for threshold of perforation, inches

D = equivalent missile diameter, inches

E_k = missile kinetic energy, foot pounds = $M\ V^2/2$

M = mass of the missile, lb-sec^2/ft

V = impact velocity, ft/s

SRI Formula

$$E / D = (S / 46{,}500)\ [\ 16{,}000\ T^2 + 1{,}500\ (W / W_s)\ T\ ]$$

which can be re-written as:

$$T_p = \sqrt{2.906\frac{E_k}{DS} + 0.0022\left(\frac{W}{W_S}\right)^2} - 0.047\left(\frac{W}{W_S}\right)$$

where:

E = critical kinetic energy required for perforation, foot pounds

E_k = missile kinetic energy, foot pounds = $M\ V^2/2$

D = effective missile diameter, inches

L = length of projectile, inches

M = mass of the missile, lb-sec^2/ft

S = ultimate tensile strength of the target (steel plate), pounds per square inch

T = target plate thickness, inches

T_p = steel plate thickness for threshold of perforation, inches

V = impact velocity, ft/s

V_c = critical penetration velocity, ft/s

W = length of a square side between rigid supports, inches

W_s = length of a standard window, 4 inches

The ultimate tensile strength is directly reduced by the amount of bilateral tension stress already in the target. The SRI formula is valid within the following ranges of parameters, which are defined above:

$0.1 < T/D < 0.8$

$0.002 < T/L < 0.05$

$10 < L/D < 50$

$5 < W/D < 8$

$8 < W/T < 100$

$0.2 < W/L < 1.0$

$70 < V_c < 400$

For the design of steel targets, the minimum design thickness (t_d) is given below where the perforation thickness, T_p, is obtained from BRL Formula or SRI Formula as applicable:

$$t_d = 1.25\ T_p$$

3.5.3.1.3 Composite (Modular) Sections

Composite or multi-element barriers consider the residual velocity of the missile perforating the first element as the striking velocity for the next element. For steel-concrete modular sections, the outer steel plates satisfy minimum thicknesses as determined in Subsection 3.5.3.1.

The residual velocity after missile penetration of the first layer (or outer shield) is determined by the formula:

$$V_r = \sqrt{V^2 - V_B{}^2}$$

where

V_r = residual velocity after missile penetration of the first layer (or outer shield)

V = impact (or striking) velocity of the missile object

V_B = perforation velocity associated with the energy absorbed up to the threshold of perforation.

3.5.3.2 Evaluation of Overall Structural Effects

Elements required to remain elastic are evaluated to assure that the usable strength capacity exceeds the demand. For structures allowed to displace beyond yield (elasto-plastic response), an evaluation confirms that acceptable deformation limits to demonstrate ductile behavior are not exceeded by comparing computed demand ductility ratios with capacity values.

After it is determined that a missile will not penetrate the barrier, an equivalent static load concentrated at the impact area is applied in conjunction with other design loads. Refer to Subsection 3.3.2.2 for determination of tornado forces on structures, including equivalent static loads for tornado missile impact. In determining an appropriate equivalent static load for other missiles sources (as defined in Subsection 3.8.4), elasto-plastic behavior may be assumed with permissible ductility ratios as long as deflections will not result in loss of function of any safety-related system.

The flexural, shear, and buckling effects on structural members are determined using the equivalent static load obtained from the evaluation of missile impact on structural response. Stress and strain limits for the equivalent static load comply with “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)”, RG 1.142, Rev.2 (Reference 3.5-14), and “Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities”, including Supplement 2 (2004) American Institute of Steel Construction (AISC) N690-

1994 & 2004. (Reference 3.5-15). The consequences of scabbing are evaluated if the thickness is less than the minimum thickness to preclude scabbing.

The overall qualification of concrete structures is discussed in RG 1.142 (Reference 3.5-14). Regulatory position 10 of RG 1.142 indicates that evaluations completed in accordance with Appendix F of American Concrete Institute (ACI) Code ACI 349-06 (Reference 3.5-16) are acceptable, with provisions made for maximum permissible ductility ratios (μ) and dynamic increase factor (*DIF*) as stated in the RG 1.142:

1. In ACI 349-06, Section F.3.5, where flexure controls design, μ = 1.0 for the structure as a whole, except as noted in conjunction with Section F3.8 below, and μ = 3.0 for a localized area in the structure.

2. In ACI 349-06, Section F.3.7, where shear controls design, μ = 1.0 when shear is carried by concrete alone, and μ = 1.3 when shear is carried by a combination of concrete and stirrups or bent bars.

3. In ACI 349-06, Section F.3.8, where impulsive or impactive loads produce flexure in a member carrying axial compression loads, μ = 1.0 in flexure when the compressive load is greater than 0.1 f'_cA_g (where A_g is the gross area of section, in.^{2}) or one-third of that which would produce balanced conditions, whichever is smaller, or

   μ is as given in Sections F.3.3 or F.3.4 of ACI 349-06, when the compression load is less than 0.1 f'_cA_g or one-third of that which would produce balanced conditions, whichever is smaller, or

   μ varies linearly from 1.0 to that given in Sections F.3.3 or F.3.4 of ACI 349-06 for conditions between those specified above.

4. In ACI 349-06 Section F.2.1, the *DIF* is 1.0 for all materials when the dynamic load factor associated with the impactive or impulsive loading is less than 1.2.

For steel, maximum allowable ductility ratios are provided by AISC N690 including Supplement 2 (Reference 3.5-15).

3.5.4 Combined License Information

COL 3.5(1) *The COL Applicant is to have plant procedures in place prior to fuel load that specify unsecured equipment, including portable pressurized gas cylinders, located inside or outside containment and required for maintenance or undergoing maintenance is to be removed from containment prior to operation, moved to a location where it is not a potential hazard to SSCs important to safety, or seismically restrained to prevent it from becoming a missile.*

COL 3.5(2) The COL Applicant is to commit to actions to maintain P_1 within this acceptable limit as outlined in RG 1.115, "Protection Against Low-Trajectory Turbine Missiles" (Reference 3.5-6) and SRP Section 3.5.1.3, "Turbine Missiles" (Reference 3.5-7).

COL 3.5(3) As described in DCD, Section 2.2, the COL Applicant is to establish the presence of potential hazards, except aircraft, which is reviewed in Subsection 3.5.1.6, and the effects of potential accidents in the vicinity of the site.

COL 3.5(4) It is the responsibility of the COL Applicant to verify the site interface parameters with respect to aircraft crashes and air transportation accidents as described in Section 2.2.

COL 3.5(5) The COL Applicant is responsible to evaluate site-specific hazards for external events that may produce missiles more energetic than tornado missiles and hurricane missiles for the standard plant presented in Subsection 3.5.1.4, and assure that the design of seismic category I and II structures meet these loads.

COL 3.5(6) The COL applicant is to identify the site-specific systems and components to be protected and to assess the orientation of the T/G with respect to the essential site-specific SSCs using the guidance and examples in RG 1.117 and use the evaluation method described in Subsection 3.5.1.3.2 to determine the probability that a turbine missile will cause unacceptable damage to an essential SSC. For multi-unit sites, the COL applicant is to also evaluate the effect of other unit(s) on this unit as specified in RG 1.115.

3.5.5 References

3.5-1 Domestic Licensing of Production and Utilization Facilities. Title 10 Code of Federal Regulations Part 50, U.S. Nuclear Regulatory Commission, Washington, DC.

3.5-2 Reactor Site Criteria. Title 10 Code of Federal Regulations Part 100, U.S. Nuclear Regulatory Commission, Washington, DC.

3.5-3 Rules for Construction of Nuclear Facility Components, American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code Section III, 2001 Edition through the 2003 Addenda.

3.5-4 Rules for Construction of Nuclear Facility Components, American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code Section XI, 2001 Edition through the 2003 Addenda.

3.5-5 Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder), ASME NOG-1, 2004.

3.5-6 Protection Against Low-Trajectory Turbine Missiles. Regulatory Guide 1.115, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, July 1977.

3.5-7 Turbine Missiles, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Standard Review Plan, Section 3.5.1.3, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.5-8 Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants. Regulatory Guide 1.76, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.5-9 Kennedy, R.P., A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects, Nuclear Engineering and Design, Volume 37, Number 2, pp 183-202, 1976.

3.5-10 Barrier Design Procedures, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Standard Review Plan, Section 3.5.3, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

3.5-11 Cottrell, W.B. and Savolainen, A.W., U.S. Reactor Containment Technology, NSIC-5, Oak Ridge National Laboratories, Volume 1, Chapter 6, 1965.

3.5-12 Russell, C.R., Reactor Safeguards, MacMillan Publishers, New York, 1962.

3.5-13 Deleted.

3.5-14 Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments), Regulatory Guide 1.142, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, DC, November 2001.

3.5-15 Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities, including Supplement 2 (2004), ANSI/AISC N690-1994, American National Standards Institute/American Institute of Steel Construction, 1994 & 2004.

3.5-16 Code Requirements for Nuclear Safety-Related Concrete Structures (ACI 349-06) and Commentary, American Concrete Institute, 2006.

3.5-17 Probability of Missile Generation From Low Pressure Turbines, MUAP-07028-P, Rev. 2 (Proprietary) and MUAP-07028-NP, Rev. 2 (Non-Proprietary), Mitsubishi Heavy Industries, Ltd., June 2013.

3.5-18 Probabilistic Evaluation of Turbine Valve Test Frequency, MUAP-07029-P, Rev. 3 (Proprietary) and MUAP-07029-NP, Rev. 3 (Non-Proprietary), Mitsubishi Heavy Industries, Ltd., June 2013.

3.5-19 Tornado Design Classification, Regulatory Guide 1.117, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, DC, April 1978.

3.5-20 Safety Implications Associated with In-Plant Pressurized Gas Storage and Distributions Systems in Nuclear Power Plants, NUREG/CR-3551 (ORNL/NOAC-214), U.S. Regulatory Commission, Washington, DC, May 1985.

3.5-21 Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants, RG 1.221, Rev. 0, U.S. Nuclear Regulatory Commission, Washington, D.C., October 2011.

3.5-22 Technical Basis for Regulatory Guidance on Design-Basis Hurricane-Borne Missile Speeds for Nuclear Power Plants, NUREG/CR-7004, U.S. Nuclear Regulatory Commission, Washington, D.C., November 2011.

3.6 Protection Against Dynamic Effects Associated with Postulated Rupture of Piping

Appendix A of 10 CFR 50, GDC 4 (Reference 3.6-1), requires that SSCs be designed to accommodate the effects of and to be compatible with the environmental conditions associated with the normal operation, maintenance, testing, and postulated accidents, including the LOCA. This section addresses the protection against the dynamic effects of postulated pipe break accidents, specifically in the localized regions of the pipe break, including pipe whip, jet impingement, subcompartment pressurization, and fluid system decompression in the ruptured pipe. In addition, the environmental effects, spray wetting, and flooding are also addressed in this section.

The criteria used to evaluate pipe failure protection are generally consistent with the NRC guidelines including those in NUREG-0800, SRP 3.6.1, 3.6.2, and 3.6.3 (References 3.6-2, 3.6-3, and 3.6-4), and applicable Branch Technical Position (BTP) 3-4 and BTP 3-3 (References 3.6-5 and 3.6-6).

In the event of a postulated pipe failure within the plant, adequate protection is provided so that the adverse effects of the failure do not impact safety-related SSCs or equipment. Safety-related systems and components are those required to shutdown the reactor and mitigate the consequences of the postulated piping failure.

The general plant protection criteria inside and outside of the PCCV to mitigate the postulated piping failure in fluid systems are defined in Subsection 3.6.1. Subsection 3.6.1 provides design bases and addresses criteria for evaluation of essential systems. The scope of essential SSCs, addressed by this section and BTP 3-4 (Reference 3.6-5), is bounded by the definition of safety-related SSCs provided in DCD Section 3.2.

Subsection 3.6.2 defines the different types of piping failures (circumferential pipe break, longitudinal pipe break, or leakage crack) to be considered based on the energy level of the fluid system. Additionally, Subsection 3.6.2 provides criteria for determining postulated break and crack locations, including the criteria for excluding piping failures in certain portions of the piping systems. The analytical methods to evaluate forcing function, and determine and evaluate acceptability of resulting pipe responses including jet impingement, pipe motion, and restraint loads, and operability are addressed.

Subsection 3.6.3 describes the application of LBB methodology, which allows elimination of postulated pipe breaks in certain piping systems based on the system characteristics and failure mechanics-based crack growth in conjunction with leak detection capability.

3.6.1 Plant Design for Protection against Postulated Piping Failure in Fluid Systems Inside and Outside Containment

In accordance with NUREG-0800, SRP 3.6.1 (Reference 3.6-2), the plant is designed to provide protection against piping failure inside or outside the PCCV to assure that such failures would not compromise the functional capability of safety-related systems to bring the plant to a safe-shutdown condition and maintain it in that condition in the event of such failure. In order to maintain the safety of the plant when a pipe break is postulated, the following items are considered in the design of the plant including the arrangement and pipe design.

Required plant conditions following a pipe break:

- The functions of the engineered safety features and related measures required in cooling the core are not impaired.
- The reactor shutdown system[1] and its components maintain their function.
- Containment integrity is maintained.
- Radiological doses resulting from a postulated piping failure remain below the limits of 10 CFR 100 (Reference 3.6-7).
- MCR functions and habitability are maintained.

The SSCs, required to be protected from postulated piping failure in fluid systems inside and outside PCCV, are discussed in this section. Additional information is provided in Sections 3.2 and 3.11 of this chapter.

Safety-related SSCs are protected from postulated piping failure in fluid systems inside and outside PCCV. The US-APWR design includes the following design measures:

- Separation of redundant trains of safety-related SSCs addressed in Chapter 1.
- Protective barriers, restraints, and enclosures, where necessary, addressed in this section.
- Placement of essential SSCs in segregated areas, which are not subject to the effects of postulated piping failure.
- Environmental evaluation of safety-related SSCs subject to the effects of postulated piping failure addressed in Section 3.11.
- Evaluation of structural features subject to the effects of postulated piping failure addressed in Section 3.8.

3.6.1.1 Design Basis

Essential systems are evaluated for conformance to the following design bases and susceptibility to pipe failure effects.

A. The selection of the failure type is based on whether the system is high or moderate-energy during normal operating conditions of the system as defined in item B, below. High-energy fluid systems are defined to be those systems or portion of systems that, during normal plant conditions, are either in operation or are maintained pressurized under conditions where either or both of the following are met:

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1. The reactor shutdown system is designed to be equipped with the function to shift to low-temperature shutdown from high-temperature shutdown and maintains the shutdown condition by inserting negative reactivity into reactor from output operating condition to assure the reactor in less than critical.

a. Maximum operating temperature exceeds 200°F

b. Maximum operating pressure exceeds 275 psig

Moderate-energy fluid systems are defined to be those systems or portion of systems that, during normal plant conditions are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

a. Maximum operating temperature is 200°F or less.

b. Maximum operating pressure is 275 psig or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short periods in performing their system function but, for the major portion of their operational period, qualify as moderate–energy fluid systems. An operational period is considered short if the total fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2% of the total time that the system operates as a moderate-energy fluid system.

Table 3.6-1 identifies systems which contain high- and moderate-energy lines.

B. The pressure-temperature condition in the fluid when the postulated piping failure occurs determines the magnitudes of the fluid reaction forces. The technical basis used to determine the state of the fluid is as follows:

1. For piping sections that are operating at a pressure greater than atmosphere, the thermodynamic state is assumed to be the one associated with normal full power operation.

2. For the piping sections that are pressurized during normal plant operating modes other than the design power generation, such as during hot standby, start up, shut down, and refueling, the thermodynamic state is assumed as operating mode that gives the worst fluid reaction forces. For high energy lines as defined in item A above, if the piping sections are pressurized to the high energy level only for 2% of the total operating time, such piping sections are considered as moderate energy and not evaluated for high energy pipe failure.

3. High-stress-based postulated pipe rupture locations are determined based on calculated stresses due to Level A service loading (normal loads) and Level B service loading (normal plus operational transients) without contribution of seismic loads since operating basis earthquake is eliminated from the explicit design considerations. However, postulated pipe rupture locations based on fatigue effects will include seismic cyclic effects.

C. For longitudinal and circumferential pipe breaks, evaluations are performed for dynamic effects, such as pipe whip, jet impingement, and subcompartment pressurization, as well as those of environmental conditions, flooding, and spray wetting. For leakage cracks, evaluations for environmental condition, flooding and spray wetting are performed. Additionally, when LBB criteria are successfully applied, evaluation of dynamic effects is not required.

D. Circumferential and longitudinal breaks of the main steam and feedwater lines are not postulated in the break exclusion zones. However, the effects of flooding, spray wetting, and subcompartment pressurization are evaluated for a postulated 1.0 sq. ft. break for the main steam and feedwater lines at a location that has the greatest effect on essential equipment.

E. Each postulated piping failure event (pipe break or crack) is considered as a single initial event during normal plant operation. For systems not analyzed for seismic considerations, it is assumed that a SSE event will cause pressure boundary failure at any location.

F. Offsite power is assumed to be unavailable if a trip of the turbine-generator system or reactor protection system is a direct consequence of the postulated piping failure. Also, offsite power is assumed unavailable during and following seismic events.

G. A single active component failure is assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted in item H below. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power.

H. Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system (e.g., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the postulated piping failure), single active failures of components in the other train or trains of that system or other systems necessary to mitigate the consequences of the piping failure and shut down the reactor need not be assumed, provided the systems are designed to seismic category I standards, are powered from both offsite and onsite sources, and are constructed, operated, and inspected to quality assurance, testing, and ISI standards appropriate for nuclear safety systems.

I. All available systems, including those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, consideration of the postulated failure and its direct consequences, such as unit trip and loss of offsite power, are assumed together with an assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions is judged on the basis of ample time and adequate access to equipment being available for the proposed actions. For breaks in non-seismic piping systems, only seismically-qualified systems are assumed to be available to mitigate the consequences of the failure since a seismic event may have caused the pipe break.

J. Rapid motion of the pipe whip resulting from postulated pipe break is assumed to occur in the plane determined by the piping geometry. The direction of the movement is in the direction of jet reaction force.

   If a thrust force causes a whipping pipe to impact a flat surface normal to its direction of travel, the direction of the initial pipe movement is assumed to rest against that surface, without any pipe whip in other directions.

   Pipe whip restraints are provided when the whipping pipe could impair the capability of any essential system or component to perform its intended function. Pipe whip restraints are located within the length of location of plastic hinge formation when permissible. If it is not possible to locate the whip restraint within the length of plastic hinge, the consequences of pipe whipping and jet impingement effect are further investigated.

K. The pipe break reaction force and jet impingement force calculation considers the fluid internal energy considering any line restrictions (that is, flow limiter) between the pressure source and break location and the absence of energy reservoirs, as applicable.

L. Jet loads resulting from postulated pipe break in a high pressure steam system (pressure greater than 870 psia) or sub-cooled water which would flash at the break, are evaluated as follows:

   1. Any directly impacted component within 10 pipe diameters is assumed to fail, unless the components is required for safe shutdown and accident mitigation capability in which case the jet loads are computed and evaluated based on the criteria given in Subsection 3.6.2.3.

   2. Based on NUREG CR-2913 (Reference 3.6-8), components beyond 10 pipe diameter range are properly evaluated based on the computed jet impingement force.

M. By definition, a non-essential system is not required for safe shutdown of the nuclear plant following a postulated pipe rupture accident scenario and as such, it is not required to be evaluated for pipe rupture protection. However, if a non-essential system or a portion of it, which is affected by a specific piping failure event that could potentially affect an essential system or component, then it is evaluated for pipe rupture protection per the rules of this section.

N. The environmental effects of a postulated piping failure do not preclude habitability of the MCR or access to surrounding areas important to the safe control of reactor operations needed to cope with the consequences of the piping failure.

   The capability of all essential components to perform their intended safety functions is maintained.

3.6.1.2 Safety Evaluation

3.6.1.2.1 General

Safety evaluations assess safety-related systems and components for the adequacy of their intended protective actions and measures in the event of postulated pipe failures, including those required to mitigate the consequences of the failure. Protective measures include separation, barriers, and restraints as discussed below.

3.6.1.2.2 Basic Protection Measures

3.6.1.2.2.1 Separation

Separation by distance, compartments or enclosures is used as much as practicable to protect redundant safety-related systems and trains. Deliberate separation protects against the dynamic effects of postulated pipe failures of the systems and components. Redundant safety systems and components are arranged to prevent the loss of the safety function as a result of a postulated pipe failure.

A multi-step process is used to develop the placement of safety-related systems and components which consider the following means for separation.

- Wherever practical, locate safety-related systems away from high-energy piping
- Locate redundant safety systems in separate compartments
- If necessary, enclose specific components required to function as a result of a postulated pipe failure
- Design drainage routing and flood control to maintain adequate separation from equipment required to function as a result of a postulated pipe failure

Each of the four safety trains are separated into four quadrants around the outside of the PCCV. Each train is isolated by physical barriers as well as isolating the radiological control area from the non-radiological control area of the R/B. The concrete walls are designed to prevent events on one safety train from impacting another train. The segregation also includes segregation of fluid containing SSCs of a train from the electrical SSCs of the same train to the extent practical. In general, cable trays are routed at higher elevations than piping. Chases are provided between the cable trays and piping to maintain the electrical/mechanical separation if required. Physically, individual train equipment within the four quadrants is located to provide the maximum separation between the same equipment of the other three trains within the confines of the R/B footprint. This separation minimizes the probability of an event affecting more than one of the safety trains at a given time. Where components must cross between isolating barriers, the penetrations are located above flood levels to the extent possible. In addition, penetration seals maintain compartment to compartment separation.

3.6.1.2.2.2 Barriers and Shields

Where physical separation is not sufficent to protect safety-related systems and components from postulated pipe failures, structural elements such as walls, floors, columns, and foundations are designed to serve as protective barriers and shields whenever possible. Other barriers, deflectors or shields are provided where additional protection is required. The barriers, including compartments as applicable, are designed to withstand loading generated by postulated jet forces and pipe whip impact forces in combination with loadings associated with seismic category I requirements, within the respective design load limits for the structures. Refer to Subsection 3.6.2.4 for additional discussion on the design of barriers, deflectors and shields.

Portions of the containment internal structure provide a series of protective barriers. The RCLs are shielded from the containment liner by the secondary shield wall. Redundancy of each loop of the reactor coolant system is also maintained by barriers formed by the secondary shield wall, refueling cavity walls, and operating deck. The combination of physical separation and robust barriers also protects the steam and feedwater lines against possible adverse interactions with the RCL.

3.6.1.2.2.3 Piping Restraints

Piping restraints are provided for postulated pipe ruptures where unrestrained movement of either end of the ruptured pipe could adversely impact SSCs, which are required to mitigate the effects of the pipe failure. Refer to Subsection 3.6.2.4 for methods of analysis of pipe whip restraints.

3.6.1.2.3 Specific Protection Measures

3.6.1.2.3.1 Subcompartment Pressurization

Analyses of postulated pipe breaks of high-energy piping is performed to determine the subcompartment's pressurization response. Locations of postulated pipe breaks are determined in accordance with Subsection 3.6.2.1.

High-energy piping and piping evaluated for LBB requirements of Subsection 3.6.3 are identified in Appendix 3E. The subcompartments inside the PCCV are designed to accommodate the pressurization loads from the breaks in lines that are not qualified for LBB. Pressurization loads on compartments inside the PCCV enclosing high-energy piping are designed as described in Subsection 3.8.4.3.

The CVCS makeup piping is classified as high-energy due to its design pressure, but does not cause pressurization because it operates at ambient temperature.

The reactor vessel (RV) annulus (volume between the RV and biological shield wall below elevation 46 ft, 11 in.) is evaluated for asymmetric compartment pressurization, and the RV is evaluated for asymmetric pressurization. These pressures are assumed to develop based on a postulated pipe break in the largest RCS line that is not qualified for LBB.

The potential for pressurization from high-energy lines in the R/B is limited, and the localized effects are considered where applicable. The pressurization loads for the elevation 65’-0” PCCV penetration room containing the break exclusion zone are based on an assumed non-mechanistic longitudinal break with a one square foot break from either main steam or feedwater line.

3.6.1.2.3.2 MCR Habitability

MCR habitability is evaluated for adverse effects resulting from postulated pipe breaks and cracks within the R/B. Section 6.4 discusses MCR habitability, including the provision of a remote shutdown workstation. The remote shutdown workstation is not subject to postulated pipe ruptures.

There are no high-energy lines near the MCR. The closest high-energy lines to the MCR are in the main steam pipe room and are part of the break exclusion areas. The MCR is separated from the isolation valve compartment by two structural floors. The area between the two floors is used for HVAC components associated with the MCR. The floors separating the HVAC compartment room from the main steam isolation valve compartment are thick, reinforced concrete floors. Refer to Subsection 3.6.2.1 for discussion applicable to PCCV penetrations in the main steam pipe room.

The main steam pipe room is evaluated using criteria for the evaluation of a one square foot longitudinal break in a break exclusion area. This location is evaluated for the effects of flooding, spray wetting, and subcompartment pressurization resulting from a postulated one square foot break of either a main steam or feedwater line. The MCR is not affected by any of the effects of a postulated break of this piping.

3.6.1.3 Postulated Failures Associated with Site-Specific Piping

The COL Applicant is to identify the site-specific systems or components that are safety-related or required for safe shutdown that are located near high-energy or moderate-energy piping systems, and are susceptible to the consequences of these piping failures. The COL Applicant is to provide a list of site-specific high-energy and moderate-energy piping systems, which includes a description of the layout of all piping systems where physical arrangement of the piping systems provides the required protection, the design basis of structures and compartments used to protect nearby essential systems or components, or the arrangements to assure the operability of safety-related features where neither separation nor protective enclosures are practical. Additionally, the COL Applicant is to provide the failure modes and effect analyses that verifies the consequences of failures in site-specific high-energy and moderate-energy piping does not affect the ability to safely shut down the plant. The COL Applicant is to update the as-designed pipe hazards analysis report to include the impact of all site specific high and moderate piping systems.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This section describes the design bases for locating postulated breaks and cracks in high- and moderate-energy piping systems inside and outside PCCV; the methods used to define the jet thrust reaction force at the break location; the procedures used to define jet impingement loading on adjacent safety-related systems or components; and the design criteria for pipe whip restraint, jet impingement barriers and shields and guard pipes. Dynamic effects such as pipe whip and jet impingement need not be evaluated in several high energy systems, including the RCL and surge line when LBB criteria are successfully applied (see Subsection 3.6.3).

The plant meets the relevant requirements of GDC 4 as follows:

1. Criteria defining postulated pipe rupture locations and configurations inside and outside the PCCV are defined in Subsection 3.6.2.1 and are per BTP 3-4 (Reference 3.6-5).

2. Protection against postulated pipe rupture outside the PCCV is provided per BTP 3-3 (Reference 3.6-6).

3. Detailed acceptance criteria covering pipe-whip dynamic analysis, including the determination of the forcing functions of the jet thrust and the jet impingements are provided in Subsection 3.6.2.3 and per Section III of SRP 3.6.2 (Reference 3.6-3). The general bases and assumptions of the analysis are per BTP 3-4 (Reference 3.6-5).

4. The reconciliation of the as-built configuration as described by ITAAC Item 4 in Table 2.3-2 of Tier 1, Section 2.3, is provided in an as-built pipe break evaluation report prior to fuel load.

3.6.2.1 Criteria used to Define Break and Crack Location and Configuration

The following subsections establish the criteria used for selecting the locations and configuration of the postulated breaks and cracks, except for piping that satisfies the requirements for LBB described in Subsection 3.6.3.

The COL Applicant is to implement the criteria for defining break and crack locations and configurations for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to identify the postulated rupture orientation of each postulated break location for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to implement the appropriate methods to assure that as-built configuration of site-specific high-energy and moderate-energy piping systems is consistent with the design intent and provide as-built drawings showing component locations and support locations and types that confirms this consistency.

3.6.2.1.1 High-Energy Fluid Systems Piping

The designer is to identify each piping run it considers in order to postulate the break locations pursuant to Subsection 3.6.2.1.1.2. In complex systems such as those containing arrangements of headers and parallel piping running between headers, the designer is to identify and include all such piping within a designated run in order to postulate the number of breaks pursuant to these criteria.

3.6.2.1.1.1 High-Energy Fluid System Piping in PCCV Penetration Area

Breaks and cracks need not be postulated in the portions of piping from containment wall to and including the inboard or outboard isolation valves. This portion of piping meets the following criteria.

All piping in the PCCV penetration area defined above is ASME Code, Section III, Class 2 (Reference 3.6-9). For ASME Code, Section III, Class 2 piping the following design criteria are met.

1. The design criteria of the ASME Code, Section III (Reference 3.6-10), Subarticle NE-1120, is satisfied for the PCCV penetration.

2. The maximum stress ranges as calculated by the sum of Equations 9 and 10 in Paragraph NC-3653 of ASME Code, Section III (Reference 3.6-9), considering those loads and conditions thereof for which Level A and Level B stress limits have been specified in the system's design specification, does not exceed 0.8(1.8 S_h + S_A). The S_h and S_A are allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.

3. The maximum stress in this piping as calculated by Equation 9, of paragraph NC 3653 of ASME Code, Section III (Reference 3.6-9) does not exceed the smaller of 2.25 S_h or 1.8 S_y, when subjected to the combined loading of internal pressure, dead weight and postulated pipe rupture beyond this portion of piping, except that following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed, operability of the valves with such stresses is ensured in accordance with the criteria specified in SRP Section 3.9.3, the piping between the outboard isolation valve and the restraint is constructed in accordance with the Power Piping Code ANSI B31.1 and the piping should either be of seamless construction with full radiography of all circumferential welds or all longitudinal and circumferential welds should be fully radiographed.

   Primary loads include those which are deflection-limited by whip restraints.

4. The number of circumferential and longitudinal piping welds and branch connections are minimized.

5. Welded attachments, for pipe supports or other purposes, to this portion of piping are avoided. Where welded attachments are necessary, the welds are 100% volumetrically examinable and detailed stress analyses are performed to demonstrate compliance with the limits of Subsection 3.6.2.1.1.2.

6. 100% volumetric examination in accordance with IWA-2400 of ASME Code, Section XI (Reference 3.6-11) of all piping welds is performed.

7. Anchors or five way restraints do not prevent the access required to conduct inservice examination specified in ASME Code, Section XI (Reference 3.6-11). ISI

completed during each inspection interval provides examination of circumferential and longitudinal welds within the boundary of this portion of piping.

8. The length of these portions of piping is to be reduced to the minimum length practical.

Application to Main Steam Pipe Room

No breaks are postulated in the main steam piping and main feedwater piping from the PCCV penetration outboard weld to the wall of main steam pipe room (Figure 3.6-1) by applying the following actions and meeting the above eight listed criteria. However, breaks are postulated in the branch piping connected to main steam piping and main feedwater piping in accordance with subsection 3.6.2.1.1.2.

- The pipe is routed straight to lower the stresses.
- Five–way restraint (free only in axial direction) is installed in the main steam pipe room wall penetration.
- Essential equipment is protected from the environmental, flooding, and subcompartment pressurization effects of an assumed non-mechanistic longitudinal break. Each assumed non-mechanistic break has a cross sectional area of one square foot and postulated to occur at a location that has the greatest effect on essential equipment.
- The length between the outboard isolation valve and the main steam pipe room wall is to be reduced to the minimum length practical.

3.6.2.1.1.2 Postulation of Pipe Breaks in Areas Other than PCCV Penetrations

The locations for postulated breaks in high-energy piping are dependent on the classification, quality group, and design standards used for the piping system. The break locations for high-energy piping are described in the following sections. These locations are postulated based on “as-designed” analyses using the design configuration. As a result of piping reanalysis, due to differences between the design configuration and the as-built configuration, the high stress and usage factor location may be shifted. The intermediate break (if any) locations need not be changed unless one of the following conditions exists:

a. The dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraint and jet shields.

b. There is significant change in pipe design parameters such as pipe size, wall thickness, or pressure rating.

For structures that separate a high-energy line from an essential component, the separating structure is designed to withstand the consequences of the pipe break in the high-energy line, which produces the greatest effect at the structure, irrespective of the fact that the following criteria might not need such a break location to be postulated.

ASME Code, Section III, Division 1 – Class 1 Piping

Pipe breaks are postulated to occur at the following locations in piping and branch runs designed and constructed to the requirements for Class 1 piping in the ASME Code, Section III (Reference 3.6-12).

- At terminal ends of the piping, including the following:
  - The extremity of the piping connected to structures, components, or anchors that act as essentially rigid restraints to piping translation and rotational motion due to static or dynamic loading.
  - Branch intersection points are considered a terminal end for the branch line unless the following are met: the branch and the main piping systems are modeled in the same static, dynamic, and thermal analyses, and shown to have significant effect on the main line behavior (that is, the branch line can not be decoupled from the main run [see Subsection 3.12.4.4]).
  - In a piping run that is maintained pressurized during normal plant conditions for only a portion of the run, the terminal end, for the purpose of defining break locations, is the piping connection to the first normally closed valve.
- At the intermediate locations where the following conditions are satisfied:
  - Intermediate locations where the maximum stress range as calculated by Equation 10 of Paragraph NB-3653 of the ASME Code, Section III and either Equation 12 or Equation 13 of Paragraph NB-3653.6 of the ASME Code, Section III exceeds 2.4 S_m (where S_m is design stress intensity) (Reference 3.6-12).
  - Intermediate locations where the cumulative usage factor as determined by the ASME Code, exceeds 0.1.

ASME Code, Section III, Division 1 – Class 2 and Class 3 Piping

Pipe breaks are postulated to occur at the following locations in piping and branch runs designed and constructed to the requirements for Class 2 and 3 piping in the ASME Code, Section III, Division 1 (Reference 3.6-9).

- At terminal ends of the piping, using the same definition for terminal ends as for Class 1 pipe.
- At intermediate locations selected by one of the following criteria:
  - At each fitting (e.g., elbow, tee, cross, flange, and non-standard fitting), welded attachment, and valve.
  - At one location at each extreme of piping run adjacent to protective structure for piping that contains no fittings, welded attachments, or valves.

- At each location where stresses calculated by the sum of Equation 9 and 10 in NC/ND-3653 of ASME Code, Section III, exceed 0.8 times the sum of the stress limits given in NC/ND-3653 (Reference 3.6-9).

Non-ASME Class Piping

Breaks in seismically analyzed non-ASME Class piping are postulated according to the same criteria as for ASME Class 2 and 3 piping above.

Separation and interaction requirements between seismically analyzed and non-seismically analyzed piping are met as described in Subsection 3.12.3.7.

3.6.2.1.1.3 Postulation of Leakage Cracks in Areas Other than PCCV Penetrations

Leakage cracks in high energy piping are postulated as follows:

- For ASME Code, Section III (Reference 3.6-12), Division 1, Class 1 piping, at axial locations where the calculated stress range by Equation 10 in NB-3653 exceeds 1.2 S_m.
- For ASME Code, Section III (Reference 3.6-9), Division 1, Class 2 and 3 piping, at axial locations where calculated stress by the sum of Equations 9 and 10 in NC/ND-3653 exceeds 0.4 times the sum of the stress limits given in NC/ND-3653.
- For seismically analyzed non-ASME Class piping at the locations defined in the same way as ASME Code, Section III (Reference 3.6-9), Class 3 piping.
- For non-ASME Class piping, which has not been evaluated to obtain stress information, leakage cracks are postulated at axial locations that produce the most severe environmental effects.

3.6.2.1.2 Moderate-Energy Fluid System Piping Break Locations

Leakage cracks are not postulated in moderate-energy fluid system piping located in an area where a break in the high-energy fluid system is postulated, provided that such a crack does not result in environmental conditions more severe than the high-energy break. If the effects of breaks of moderate-energy fluid system piping are more severe than those of high-energy fluid system piping, then the provision of this Subsection 3.6.2.1.2.2 is applied.

Through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for about 2% of the operational period but qualify as moderate-energy fluid systems for the major operational period.

3.6.2.1.2.1 Moderate-Energy Fluid System Piping in PCCV Penetration Areas

Leakage cracks are not postulated in those portion of the piping from PCCV wall to and including the inboard and outboard isolation valves provided that the PCCV penetration meets the requirements of ASME Code, Section III (Reference 3.6-10), Subarticle

NE-1120 and the piping is designed so that the maximum stress range based on the sum of Equations (9) and (10) in Subarticle NC/ND-3653 of the ASME Code, Section III (Reference 3.6-9) does not exceed 0.4 times the sum of the stress limits given in NC/ND-3653.

3.6.2.1.2.2 Moderate-Energy Fluid System Piping in Areas Other than PCCV Penetrations

Leakage cracks are postulated in the following piping systems located adjacent to safety-related SSCs.

- For ASME Code, Section III, Class 1 piping, where the stress range calculated by Eq. (10) in NB-3653 is more than or equal to 1.2 *S(m)*
- For ASME Code, Section III (Reference 3.6-9), Class 2 and 3 and non-safety class piping, at axial locations where calculated stress by the sum of Equations 9 and 10 in NC/ND-3653 exceed 0.4 times the sum of the stress limits given in NC/ND-3653.
- For non-safety class piping, which has not been evaluated to obtain stress information, leakage cracks are postulated at axial locations that produce the most severe environmental effects.

3.6.2.1.3 Types of Break/Cracks Postulated

3.6.2.1.3.1 Circumferential Pipe Breaks

Circumferential breaks are postulated in high-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 inch at locations identified by the criteria in Subsection 3.6.2.1.1.2

No breaks are postulated in piping having a nominal diameter less than 1 inch, including instrument lines that are designed in accordance with RG 1.11 (Reference 3.6-13).

If the maximum stress range exceeds the limits specified in Subsection 3.6.2.1.1.2 and the circumferential stress range is greater than 1.5 times the axial stress range, no circumferential break is postulated; only a longitudinal break (Subsection 3.6.2.1.3.2) is postulated.

Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment. The line restrictions, flow limiters, positive pump-controlled flow and the absence of energy reservoirs may be taken into account, as applicable.

Following a circumferential break, the two ends of the broken pipe are assumed to move clear of each other unless physically limited by piping restraints, structural members, or pipe stiffness. Piping stiffness is used only when a plastic hinge is not developed in the piping. The effective cross sectional (inside diameter) flow area of the pipe is used in the jet discharge evaluation. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration and to initiate pipe movement in the direction of the jet reaction. For identifying potential targets resulting from the postulated pipe rupture, the direction of the fluid jet is based on the arrested position of the broken pipe end including variation of the broken pipe end movement during the blowdown.

3.6.2.1.3.2 Longitudinal Pipe Breaks

Longitudinal breaks are postulated in high-energy fluid system piping and branch runs in nominal pipe sizes 4 inches and larger. Longitudinal breaks are postulated in high-energy fluid system piping at locations of circumferential breaks as described in Subsection 3.6.2.1.3.1.

If the maximum stress range exceeds the limits specified in Subsection 3.6.2.1.1.2 and the axial stress range is greater than 1.5 times the circumferential stress range, no longitudinal break is postulated, only a circumferential break (Subsection 3.6.2.1.3.1) is postulated.

Longitudinal breaks need not be postulated at terminal ends.

Longitudinal breaks in the form of axial split without pipe severance are postulated in the center of the piping at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of piping configuration and produces out-of-plane bending. Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

For longitudinal breaks, the dynamic force of the fluid jet discharge is based on a circular or elliptical ($2D$ x $1/2D$) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location, where D is the effective inner diameter of the pipe. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

Piping movement is assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or pipe stiffness as demonstrated by inelastic limit analysis.

3.6.2.1.3.3 Leakage Cracks

Leakage cracks are postulated in high-energy fluid system piping at locations identified in Subsection 3.6.2.1.1.3. Leakage cracks are also postulated in moderate-energy fluid system piping at locations identified in Subsection 3.6.2.1.2.2, except where excluded by the criterion in Subsection 3.6.2.1.2.2.

Leakage cracks are not postulated in 1-inch nominal diameter and smaller piping.

Leakage cracks are postulated in those circumferential directions that result in the most severe environmental, spray wetting, and flooding consequences.

Fluid flow from leakage cracks is based on a circular orifice with a cross-sectional area equal to that of a rectangle one-half the pipe inside diameter in length and one-half the pipe wall thickness in width. The flow from the crack opening is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments based on conservatively estimated time period to effect corrective actions.

3.6.2.2 Guard Pipe Assembly Design Criteria

Piping penetrations are an integral part of the PCCV pressure boundary. The annular space of the US-APWR consists of multiple compartments encircling the PCCV. These compartments segregate the PCCV electrical and mechanical penetrations into their own isolated compartments; specifically, electrical penetration rooms and mechanical penetration rooms. By virtue of the plant configuration, as piping crosses from inside to outside the PCCV, it emerges into piping penetration compartments. These compartments are designed to address postulated piping failures and the effect there of, as such, guard pipe assemblies are not required.

3.6.2.3 Analytic Methods to Define Forcing Functions and Response Models

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. To determine the forcing function for breaks postulated based on the criteria in Subsection 3.6.2.1, the fluid conditions at the upstream source and at the break exit determine the analytical approach. For most applications, one of the following conditions exists.

- Superheated or saturated steam
- Saturated or sub-cooled water
- Cold water (non-flashing)

The analytical methods used for the calculation of the jet thrust for the above described conditions are based on SRP 3.6.2 (Reference 3.6-3) and MHI original methodologies (Reference 3.6-25) based on measurements cited in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31.

The time dependent forcing function is effected by the thrust pulse resulting from the sudden pressure drop at the initial moment of pipe rupture, the thrust transient resulting from wave propagation and reflection, and the blowdown thrust resulting from the buildup of the discharge flow rate, which may reach a steady state if there is fluid energy reservoir having sufficient capacity to develop a steady jet for a significant interval.

Alternatively, a steady state jet thrust function may be used as outlined in Subsection 3.6.2.3.1.

A rise time of one millisecond is used for the initial pulse.

The loading condition of a pipe run or branch, prior to the postulated rupture, in terms of internal pressure, temperature, and inertial effects are used in the evaluation for postulated breaks. For piping pressurized during operation at power, the initial condition is the greater of the contained energy at hot standby or at 102% power.

3.6.2.3.1 Steady State Jet Force

The steady state jet force can be represented by:

$F_j \;=\; C_T P A$ (Reference 3.6-14)

where

F_j = Jet Force

C_T = Thrust Factor

P = Pipe Internal Pressure Before Break

A = Break Plane Area

The thrust factor C_T is established as a function of fluid state as follows:

(a) Sub-Cooled Water

$C_T = 2.0\text{-}0.861\, h^{*2}$ $(0 \leq h^* \leq 0.75)$

$= 3.22 - 3.0\, h^* + 0.97\, h^{*2}$ $(0.75 \leq h^* \leq 1.0)$

where

$h^* = (h_0 - 180)/(h_{sat} - 180)$

h_0 = Sub-Cooled water enthalpy (BTU/lbm)

h_{sat} = Saturated water enthalpy at pressure P (BTU/lbm)

C_T value varies based on the pressure and enthalpy. In case of saturated water, the minimum value of 1.26 comes closer to maximum value of 2.0 as enthalpy (temperature) decreases. In case of h_0 = 180 BTU/lbm (temperature T_0 < 212°F) or lower, condition for sub-cooled, non flashing water value of 2.0 is used.

(b) Saturated water-saturated steam:

Use C_T =1.26

The above approach is a conservative method for calculating the thrust factor. Then, pipe pressure drop and diaphragm effects may be considered and the detailed analysis and experiment data may be applied.

Schematics of jets discharging from a pipe break are shown in Figure 3.6-2.

3.6.2.3.2 Time Dependent Break Forcing Function

Time dependent break forcing is applied to RCL piping from postulated pipe breaks in branch lines not included in the LBB. LBB criteria of Subsection 3.6.3 are used to demonstrate that there are no postulated pipe ruptures in the large diameter RCL piping.

A detailed description of the hydraulic transient caused by a postulated pipe rupture is developed for application to the RCL. The thrust and reactive forces resulting from a postulated branch pipe rupture connecting to the RCL are applied in calculating the hydraulic forcing functions for the intact RCL. The RCL forces result from the transient flow and pressure histories in the RCS. A two-step calculation first determines the transient pressure, mass flow rates, and thermodynamic properties as a function of time. The final step calculates the time history of forces at appropriate RCL locations (such as elbows) using the results obtained from the hydraulic analysis, and input of area and directional coordinates.

Hydraulic modeling of the entire RCS results in calculating the pressure, mass flow rate, and density. Time-dependent loads exerted by the fluid on the loops are then determined by applying the pressure, mass flow rate, density to the thrust calculation and plant layout information. Pressure and momentum flux terms become dominant for hydraulic forcing functions during a postulated LOCA. Local fluid conditions in the hydraulic model utilize the inertia and gravitational terms.

The reactor core environment is dynamically analyzed using a blowdown hydraulic analysis to determine the loop forces. A blow-down hydraulic analysis computer code is used to calculate RCS hydraulic transients from predicted flow, quality and pressure of the fluid throughout the system. Refer to Subsection 3.9.2.5 for the description of the blowdown hydraulic analysis.

The equation for determining blowdown hydraulic loads on primary loop components is:

$$F = 144A\left[(P - 14.7) + \frac{\dot{m}^2}{144\rho g A_m^2}\right]$$

where

F = Forces (lbs)

A = Area of aperture (ft^2)

P = Pressure of system (psia)

$\dot{m}$ = Mass flow rate (lbm/s)

ρ = Density (lbm/ft^3)

g = Pull force constant (gravitational speed) = 32.174 ft-lbm/lb -s^2

A_m = Mass flow area (ft^2)

The RCL is modeled similarly to the model used in blowdown analysis for the purpose of computing forcing functions. A global coordinate system is developed for the entire loop layout, where each mode is then described by blowdown hydraulic information and the streamline force node orientation in the system. The flow areas and projection coefficients are described along the three axes of the global coordinate system. Each node is described by one or two flow apertures as a separate control volume. Forces are broken down orthogonally into x, y, and z components. The summation of the total number of apertures results in orthogonal thrust forces F_x, F_y, and F_z. These thrust forces are applied as input in dynamic analyses of piping and restraints.

3.6.2.4 Dynamic Analysis Methods to Verify Integrity and Operability

Time dependent and steady state thrust reaction loads caused by saturated or superheated steam, saturated or sub-cooled water, and cold water (non-flashing) fluid from a ruptured pipe are used in the analyses of dynamic effects of pipe breaks.

3.6.2.4.1 Jet Impingement Loading on Safety-Related Components

Structural integrity of safety-related SSCs against jet impingement load caused by pipe break is evaluated based on steady state jet force from Subsection 3.6.2.3.

Jet impingement loading is a suddenly applied constant load which can have significant energy content. These loads are generally treated as statically applied loads. The Jet impingement pressure essentially has non-uniform distributions, which varies with distance from the pipe break as shown in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31. However, the maximum pressure in the non-uniform distribution is conservatively used as a uniform pressure distribution.

The MHI original methodologies (Reference 3.6-25) used to evaluate the jet effects resulting from the postulated breaks in high energy piping are based on measurements cited in References 3.6-26, 3.6-27, 3.6-28, 3.6-29, 3.6-30 and 3.6-31. Figure 3.6-2 depicts jet characteristics for the three fluid states. The short term response evaluates the jet impingement load considering a dynamic load factor of 2 and snubber supports to be active. No dynamic load factor is used and the snubbers are considered inactive for the long-term response.

3.6.2.4.1.1 Blast Wave Assessing Procedure

Computational fluid dynamic analysis confirms the generation of a blast wave from a steam pipe break. Potential effects are assessed on equipment within the US-APWR pressurizer compartment. Distance between the postulated pipe break locations and components is long enough to attenuate the effects. However, if layout in the pressurizer compartment is changed in the future, reassessment of the blast wave will be conducted.

Blast wave is not considered to occur from a sub-cooled water pipe break. This is due to having a velocity of the two-phase flow at the break point that is slower than the speed of sound in atmospheric environments.

The details of the evaluation and the design procedure are contained in section 2 of Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing a blast wave from a steam pipe break.

3.6.2.4.1.2 Jet Pressure Oscillation Assessing Procedure

Jet pressure oscillation from a steam pipe break is unlikely to occur in the US-APWR due to its high compression ratio. The jet flow expansion and Mach Disk is large. This leads to a stable flow downstream after the Mach Disk. The flow is so stable that disturbance at the impingement wall does not reach back to the Mach Disk.

When sub-cooled jet-flow impinges on the wall, pressure distributions on the wall are not of the concave type and a re-circulation vortex is not generated. This is due to having a flow velocity at the jet boundary that is lower than that of the core region.

However the jet pressure oscillation may occur, when the water flashes to steam in the later stages of a jet blowdown. Therefore, The load of jet pressure oscillation is evaluated in the later stage of blowdown process. The details of the evaluation and the design procedure are contained in section 3 of Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing a jet pressure oscillation from a steam pipe break.

3.6.2.4.1.3 Jet Reflection Assessing Procedure

When jet flow impinges on a perpendicular wall, impinged jet flow is redirected and runs along the surface of the wall. The zone of influence obtained by computational fluid dynamics is enveloped by the estimated zone of influence from MHI original methodologies (Reference 3.6-25). However, there may be a case that a jet impinges upon an oblique wall; the effect of jet reflection is considered outside of zone of influence.

The details of the evaluation and the design procedure are contained in section 4 of Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture, for details on assessing a jet reflection.

3.6.2.4.2 Dynamic Analysis for Piping Systems

3.6.2.4.2.1 RCL Piping

Appendix 3C provides analysis details for RCL piping. Loads generated by postulated breaks from branch lines are applied to determine structural response of RCL piping.

3.6.2.4.2.2 Piping Other Than RCL Piping

In evaluating the dynamic effects of breaks in high-energy-fluid system piping other than RCL piping, possible break locations and break configurations are first established based on Subsection 3.6.2.1 and the effects of pipe whipping are then evaluated based on Subsection 3.6.2.4.4.3

If the above evaluation determines that no safety-related SSCs are damaged, then dynamic analysis is not necessary. If the above evaluation determines that the structural integrity of safety-related SSCs is impaired, pipe whip restraints are incorporated in the high-energy-fluid system piping of concern and dynamic analysis is conducted for the system including the piping and the pipe whip restraints.

In general, a gap is provided between a pipe whip restraint and pipe so as not to restrict thermal movement in the pipe. In the event of a pipe-break accident, the pipe accelerates in the gap due to the jet force and collides with the pipe whip restraint. The dynamic effects of this pipe and pipe whip restraint are usually evaluated by the energy balance method.

Conservatively assuming a fixed jet force at maximum load as described in Subsection 3.6.2.3, the maximum displacement of the pipe and pipe whip restraint can be given by the following equation based on the energy balance method.

work done on system = energy absorbed by pipe + energy absorbed by restraint (1)

Generally, in Equation (1), energy absorbed by the system is conservatively ignored so that the maximum displacement of the pipe and pipe whip restraint is given by the following equation.

work done on system = energy absorbed by restraint (2)

See Subsection 3.6.2.4.4.1 for the design methodology for pipe whip restraints.

When making a more detailed evaluation to analyze the dynamic effects associated with pipe rupture events on the broken pipe, a non-linear elastic-plastic analysis is performed. In this model, restraints specifically designed to prevent pipe whip are included, i.e. pipe whip restraints. The normal supports that act during plant operational loads, including seismic events to maintain the integrity of the unbroken pipe, are not considered unless they are capable of withstanding pipe rupture loads based on a broken pipe analysis.

The five-way restraint is installed for main steam piping and feedwater piping outside of the PCCV to prevent a load from being applied to the CV isolation valve due to a postulated pipe break outside of break exclusion zone.

In other cases, the subject valve is installed sufficiently away from a postulated break location to prevent dynamic effects. Furthermore, the pipe stress in the vicinity of the valve is validated as very small by using a static force displacement methodology for the pipe displacement at the break location.

3.6.2.4.2.3 Closure of the Feedwater Check Valve

This loading has a short duration of approximately 0.5 seconds and arises from rapidly traveling pressure waves in piping systems connected to the broken piping system. The closure of the feedwater check valve due to a postulated pipe rupture upstream of the valve can increase the magnitude of these loads.

For piping systems with closing check valves, the magnitude of the loadings depends on the valve closure time, with shorter closing times generally causing higher loadings.

The maximum internal pressure and the kinetic energy of the valve disc at the time of closure are used to assure the pressure boundary integrity of the piping. The RELAP-5 code (Reference 3.6-15) can be used to calculate the pressure and kinetic energy in this loading situation.

3.6.2.4.3 Subcompartment Pressure Forces

Subcompartment pressure forces are considered in the evaluation of structures and components. The code GOTHIC (Reference 3.6-16) may be used to calculate the pressure transients in the building subcompartments. The subcompartment pressure forces are determined by integrating the pressure transient over the surface of the structure or component. Jet impingement forces, when applicable, are considered additive to subcompartment pressure forces.

3.6.2.4.4 Pipe Whip Restraints, Barriers and Shields

This section provides analytical methods used to verify integrity and operability of the safety-related SSCs needed to safely shutdown the plant, that are nearby the postulated pipe breaks. The analytical methods apply to the following structures:

- Whip restraint
- Jet impingement barriers and shields

3.6.2.4.4.1 Pipe Whip Restraints

The analytical methods for the design of the pipe whip restraints are described in this subsection.

When pipe whip restraints are required to satisfy protection requirements, the following guidelines are followed to select the type of restraint.

To satisfy varying requirements of available space, allowable building structure reaction, permissible pipe deflection, and equipment operability, the restraint may be a combination of an energy absorbing element and a restraining structure suitable for the geometry required to transfer the load from the whipping pipe to the main building structure or a relatively rigid steel frame to restrain the whipping pipe.

A typical rupture restraint is shown in Figure 3.6-3. The restraint structure is typically a structural steel frame or truss and the energy absorbing element is one of the following:

- U-Bar (One-Dimensional Restraint):
  - This is a U-shaped rod or flat plate, usually of carbon steel, looped around the pipe but not in contact with the pipe to allow unimpeded pipe movement during normal operation and a seismic event. At rupture, the pipe converges with the U-Bar(s), which absorbs the kinetic energy of the pipe by yielding plastically.
- Structural Steel (Two-Dimensional Restraint):

- This is a structural steel frame assembly enveloping the pipe but not in contact with the pipe that allows unimpeded pipe motion during normal operation and a seismic event. At rupture, pipe converges with the structural steel frame and the frame, which deflects plastically, absorbing the kinetic energy of the pipe.

Pipe whip restraints used to protect SSCs are designed as seismic category I.

Loads to be evaluated in combination with pipe break forces are Level A or B service loads and are not combined with seismic loads. Seismic loads are independently considered to confirm the structural integrity of the pipe whip restraint if the restraint becomes in contact with the pipe during the seismic event. In the evaluation of structures, loads producing primary stresses are used.

3.6.2.4.4.1.1 Location of Pipe Whip Restraints and Analytical Methods

A. To determine the pipe hinge location, the plastic moment of the pipe is determined in the following manner.

$MP = 1.1\ ZP \times SY$

where

ZP = Plastic section modulus of pipe

SY = Yield stress at pipe operating temperature

1.1 = 10% factor to account for strain hardening

Pipe whip restraints are located as close to the axis of the reaction thrust force break as practicable, but within the length of location of plastic hinge. When it is not possible to locate the whip restraint within the length of plastic hinge, the consequences of the whipping pipe and the jet impingement effect are further investigated.

B. Pipe whip restraints are installed with sufficient annular clearance between them and the process pipe. This provides sufficient clearance for insulation and thermal and seismic movement of the pipe during normal plant operation.

If restraint also functions as a seismic support, the restraint is included in the piping analysis.

C. Restraints generally must not impede the access required to carry out the ISI of pipe welds. If the position of any restraint impedes the access to the pipe welds, part of the restraint can be removed to assure the accessibility.

3.6.2.4.4.1.2 Analysis and Design of Pipe Whip Restraint

The analysis and design of the pipe whip restraints for the effects of postulated pipe rupture conform to the following criteria.

A. Pipe whip restraints are designed based on the principle of energy absorption by considering the behavior of material's elasticity/plasticity and strain hardening.

B. Coefficient of rebound 1.1 is applied to jet thrust forces.

C. Energy absorption by the broken pipe is assumed to be zero, except in the case of calculating to check the formation of a plastic hinge. The developed thrust force is assumed to be applied to move a broken pipe directly, and is not reduced by the forces required to bend the pipes.

D. In the elasticity/plasticity design, the kinetic energy of the pipe is absorbed by the restraint by yielding plastically. The strain in the restraint is limited as shown below.

$$e = 0.5\varepsilon_u$$

where

e = Allowable strain used in the design

ε_u= Ultimate homogeneous tensile strain.

3.6.2.4.4.2 Jet Impingement Barriers and Shields

Barriers or shields are provided to protect essential equipment, including instrumentation, from the effects of jet forces resulting from postulated pipe breaks. Generally, protection provided by walls, floors, and columns is sufficient to meet protection requirements. Loading combinations and design criteria for barriers and shields are described in Section 3.8. The design procedure for jet pressure oscillation is contained in section 4 of Reference 3.6-32, Evaluation of Jet Impingement Issues Associated with Postulated Pipe Rupture.

3.6.2.4.4.3 Pipe Whip Impact on Structures

The evaluation of structures that are impacted by whipping pipes is described below.

Following a postulated pipe rupture, pipe whip into surrounding structures will occur, if the pipe is not sufficiently restrained. The level of energy in the whipping pipe may be determined by calculating work quantities using simplified methods. The external work is calculated by multiplying the break force acting on the whipping pipe by the distance (from its initial position, before the pipe break, to the final position when the pipe impacts the structure) traveled by that break force.

As the impact occurs on concrete targets, the section of the pipe near the impact area is rapidly decelerated and crushed. The magnitude and the duration of the impact loading are determined by characteristics of both the whipping pipe and the concrete barrier. In the evaluation of the target, both local and overall response is considered. The evaluation procedures are as described in References 3.6-17, 3.6-18, 3.6-19, and 3.6-20. For impact into concrete, the concrete ductility ratio is calculated for this impact, and it is assumed to be within the limits specified in Subsection 3.5.3. The evaluation of the response of steel targets relies on empirical formulae established from test data. Various formulae and their range of application are described in Reference 3.6-18.

If the whipping pipe impacts another pipe, the evaluation criteria are provided below.

- A whipping pipe is not considered capable of damaging an impacted pipe of equal or greater diameter and thickness. It is considered capable of (a) rupturing impacted pipes of smaller nominal pipe sizes, and (b) developing through-wall leakage cracks in pipe of equal or greater diameter having a lesser wall thickness (Reference 3.14), except where analytical or experimental, or both, data for the expected range of impact energies demonstrate the capability to withstand the impact without rupture. Effects on environment and shutdown logics associated with the failure of the impacted pipe are considered.

3.6.2.5 Implementation of Criteria Dealing with Special Features

Special features such as pipe whip restraints, barriers, and shields are discussed in Subsection 3.6.2.4.4.

3.6.2.6 Outline of Pipe Break Hazard Analysis Report(s)

The following information provides an outline of methodology for the pipe break hazard analysis that will be completed for high and moderate energy piping systems (including the non-safety class piping) identified in Table 2.3-1 for the closure of Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) in Tier 1, Table 2.3-2 related to the pipe break hazard analysis report:

- Identification of pipe break locations in high energy piping[1]
- Identification of leakage crack locations in high and moderate energy piping
- Identification of SSCs that are safety-related or required for safe shutdown[2]
- Evaluation of consequences of pipe whip and jet impingement
- Evaluation of consequences of spray wetting, flooding, and environmental conditions including the consequences of a postulated 1.0 sq. ft. break for the main steam and feedwater lines within the break exclusion zone, at a location that has the greatest effect on essential equipment
- Design and location of protective barriers, restraints, and enclosures

Notes

1. Table 3.6-2 provides list of high energy lines for pipe break hazard analysis, including properties of internal and external fluids.
2. All the SSCs that are safety-related or required for safe shutdown in close proximity to the postulated pipe rupture will be identified.

3.6.3 LBB Evaluation Procedures

This subsection describes the design basis to eliminate the dynamic effects of pipe rupture (Subsection 3.6.2) for the selected high-energy piping systems of RCL piping, RCL branch piping, and main steam piping. GDC 4 of Appendix A to 10 CFR 50 (Reference 3.6-1) allows exclusion of dynamic effects associated with pipe rupture from the design basis, when analyses demonstrate that the probability of pipe rupture is extremely low for the applied loading resulting from normal conditions, anticipated transients and a postulated SSE. The LBB evaluation is performed in accordance with SRP 3.6.3 (Reference 3.6-4).

The LBB analysis combines normal and abnormal (including seismic) loads to determine a critical crack size for a postulated pipe break. The critical crack size is compared to the size of a leakage crack for which detection is certain. If the leakage crack size is sufficiently smaller than the critical crack size, the LBB requirements are satisfied.

The piping systems, for which the LBB criterion is not applied, are evaluated for dynamic effects of postulated pipe rupture at locations defined in Subsection 3.6.2. For piping systems for which LBB is demonstrated, the evaluation of environmental effects including spray wetting, and flooding is still performed for breaks or leakage cracks in accordance with Subsection 3.6.2.

The types of as-built materials and material specification is to be identified for base metal welds, weldments, and safe ends for piping evaluated for LBB. Additionally, information is to be provided related to as-built material and material specifications for piping including toughness (J-R curves) and tensile strength (stress-strain curves), yield and ultimate strength, welding process/methods used, provide confirmation that the actual plant-specific stress analysis based on final as-built plant piping layout and material properties and welds satisfy the bounding LBB analysis, and provide confirmation that the final bounding LBB analysis addresses all plant-specific and generic degradation mechanisms in the as-built piping systems. This issue is to be resolved in respective system ITAAC described in Tier 1.

3.6.3.1 Application of LBB Criteria

Piping systems to which LBB criteria are applied are high-energy systems with well defined loading combinations and conditions. LBB criteria are applied to the following high energy piping systems (see Appendix 3E).

- RCL Piping
- RCL branch piping with nominal diameter of 6 inches or larger, except for steam within the piping for the pressurizer safety valve and power operated relief valve
- Main Steam Pipe in PCCV

The LBB evaluations demonstrate that for piping meeting the criteria, sudden catastrophic failure of the pipe is not credible. This is demonstrated by plant design, operation history, tests, or analyses that breaks are less likely to occur under the effects of thinning by erosion or corrosion, stress corrosion cracks, water hammer, fatigue (thermal or mechanical), thermal aging, thermal stratification, creep fatigue, indirect factors, and cleavage.

3.6.3.2 Design Criteria for LBB

In order to assure successful application of LBB, the following guidelines are satisfied.

- Design Features:
  - Perform 100% preservice inspection of all welds.
  - Assure maximum stress due to steady-state vibration is well within the applicable fatigue endurance limits.
  - Use of materials of adequate toughness.
  - Leak detection systems inside the PCCV meet the requirements of RG 1.45 (Reference 3.6-21). Leak detection system is described in Subsection 5.2.5.
  - The maximum stress and the normal stress are within the (BACs) (Subsection 3.6.3.4) for an entire piping system or analyzable portion.
  - For piping systems where design drawings are used for LBB evaluation, the as-built conditions are verified to assure that the design checked for LBB is consistent with the final as-built configuration. The as-built verification includes, but is not limited to, the following:
    1. Material and material specification
    2. Pipe geometry
    3. Support locations and their characteristics
    4. Locations and weights of components such as valves
  - ISI and testing of snubbers is performed to assure low snubber failure rate (Subsection 3.12.6.6).
  - Potential degradation by erosion, erosion/corrosion and erosion cavitation is evaluated to assure low probability of pipe failure.
  - Adjacent structures and components are designed for the SSE event to assure low probability of indirect pipe failure.

Dynamic effects associated with the postulated pipe rupture need not be considered when technically justified by application of the LBB criteria. This includes the following:

- Pipe Whip and reaction forces
- Jet Impingement Loads
- Reactor cavity asymmetric pressurization transients

- Subcompartment pressurization
- Breaks associated with the transient loads in intact portion of the system, such as loads on reactor internal, or SG internal, and pump overspeed

The evaluation of flooding and environmental effects associated with leakage cracks or circumferential or longitudinal pipe breaks is not affected by application of the LBB approach.

3.6.3.3 Potential Failure Mechanism for Piping

High-energy piping is evaluated for potential failure mechanisms and other degradation sources in order to assure acceptability of the LBB criteria. Based on guidance provided by SRP 3.6.3 (Reference 3.6-4), the following pipe failure mechanisms and degradation sources could challenge the integrity of the piping:

- Water Hammer
- Creep Damage
- Wall Thinning Induced by the Effects of Erosion/Corrosion
- stress corrosion cracking (SCC)
- Fatigue
- Thermal Aging

Each failure mechanism and degradation source is evaluated below and confirmed as not credible, thereby confirming LBB eligibility.

3.6.3.3.1 Water Hammer

RCL Piping

RCL piping design features that prevent water hammer loads include maintaining temperature within specified parameters through control rod positioning and boric acid concentration, and controlling pressures within specified steady-state conditions through pressurizer heaters and pressurizer spray. Normal operating conditions for the RCL are at pressures in excess of the saturation pressure of the fluid. Water hammer does not occur under these conditions. In addition, the piping is designed for all transients identified for normal, upset, emergency and faulted service conditions as identified for Levels A, B, C and D, respectively. Therefore, water hammer is not considered a credible pipe failure mechanism within RCL piping.

RCL Branch Piping

Water hammer in RCL branch piping can occur by seal water in case of blowdown of pressurizer safety valve and relief valve lines, has been experienced in the past. Due to this reason, analysis of water hammer on these piping has been performed in order to assure integrity of piping and support design. However, LBB criteria are not applied to these piping.

As to other RCL branch piping, water hammer has been reported for ECCS piping in the past. However, in the US-APWR, operational control is applied in a way that avoids water hammer.

Water hammer is not experienced in RCL branch piping other than in these areas and the piping is designed to preclude the voiding condition according to operation at a pressure greater than the saturation pressure of the coolant. Furthermore, no valve that requires immediate action, such as pressurizer safety valve or relief valve, is present in the piping. Also, proper operating and maintenance procedures will be performed to prevent water hammer. The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer. The procedures should address the plant operating and maintenance procedures for adequate measures to avoid water hammer due to a voided line condition.

From the above reasons, water hammer is not anticipated to occur regarding RCL branch piping that LBB criteria is applied.

Main Steam Piping

Steam hammer in the main stem line is prevented by the design features included in system design. These features include prevention of slug formation by use of drain pots and proper sloping of the line. The following system design provisions address concerns regarding steam hammer within the main steam line and identify the significant dynamic loads included in the main steam piping design.

Protection against the potential occurrence of steam hammer is provided through operations and maintenance procedures that provide for slowly heating up (to avoid condensate formation from hotter steam on colder surfaces), caution against fast closing of the main steam isolation valves except when necessary, and emphasize proper draining.

A turbine trip, which initiates a rapid closure of the stop valve, is a design condition analyzed for the safety-related portion of main steam piping and associated components. This stress analyses assure that rapid valve closure does not challenge the integrity of piping. Therefore, the main steam piping is adequately designed to sustain steam hammer or similar high frequency hydrodynamic events.

3.6.3.3.2 Creep Damage

Pipe materials are selected to satisfy operational temperature limits not to exceed 700°F for ferritic steel piping and not to exceed 800°F for austenitic stainless steel piping. Therefore, the piping is designed to operate at temperatures less than that for which creep and creep-fatigue is a concern.

3.6.3.3.3 Wall Thinning Induced by the Effects of Erosion/Corrosion

RCL Piping

RCL piping utilizes austenitic stainless steel, which provides a high resistance to wall thinning induced by erosion, erosion/corrosion, and erosion/cavitation. Historically, there have been no recorded cases of erosion-induced wall thinning within the primary loops of operating plants. Since the US-APWR primary loop design maintains a flow velocity of approximately 55 feet per second and water chemistry similar to other pressurized water reactors, degradation of RCL pipe wall thicknesses by erosion, erosion/corrosion, or erosion/cavitation is not considered a credible failure mechanism for RCL piping.

RCL Branch Piping

Similar to the primary loops, RCL branch piping utilizes austenitic stainless steel highly resistant to wall thinning. In addition, no cases of erosion induced wall thinning have been reported for RCL branch piping in operating plants with similar operating parameters. Therefore, wall thinning induced by the effects of erosion is not considered a credible failure mechanism for RCL branch piping.

Main Steam Piping

Although fabricated from SA-106 Grade B carbon steel, wall thinning induced by the effects of erosion is not anticipated within main steam piping. The MSS operates with high quality steam at high temperatures, thereby circumventing the chemical and temperature initiators that erode carbon steel surfaces. Therefore, wall thinning induced by the effects of erosion is not considered a credible failure mechanism for main steam piping.

3.6.3.3.4 SCC

System operations are maintained that inhibit SCC, primarily through use of materials with low susceptibility, stress limitations, and chemistry control.

RCL Piping

The RCL piping is constructed of austenitic stainless steel materials that are proven through years of successful industry usage to resist SCC. The recommendations of RG 1.44 (Reference 3.6-22) in the use of sensitized stainless steel are applied by the US-APWR.

In addition to stress control in accordance with ASME Code, Section III, Class 1 piping (Reference 3.6-12), fluid chemistry is maintained as low- or no-oxygen environments. While RCL piping tensile stresses are within design allowables, residual tensile stresses also develop within welds. These residual stresses are self-equalizing, and industry experience with other pressurized water reactors has affirmed their acceptability.

Butting nozzles located at safe ends with a stainless steel to carbon steel interface are performed utilizing a high nickel alloy material. An alloy consisting of nickel, chromium, and iron has been selected and qualified based on the non-susceptibility to SCC.

RCL Branch Piping

Similar to the primary loops, RCL branch piping utilizes austenitic stainless steel which is highly resistant to SCC. Therefore, SCC is not considered a credible failure mechanism within both primary and branch RCL piping.

Main Steam Piping

Although fabricated from ferritic SA-106 Grade B carbon steel, SCC is not anticipated within main steam piping. While SCC in main steam piping could result from SG moisture, the secondary side utilizes an all-volatile water treatment chemistry. All volatile treatment resists causticity in the SG bulk liquid environment which resulted in chemical imbalances from SCC of SG tubing previously experienced in some operating plants. No caustic SCC on the ferritic steam lines have occurred with all volatile water treatment. Therefore, SCC is not considered a credible failure mechanism within the main steam piping.

3.6.3.3.5 Fatigue

Low-Cycle Fatigue

Class 1 piping satisfying the requirements of ASME Code, Section III (Reference 3.6-12) are designed for low-cycle fatigue postulated to occur during normal operation and anticipated transients, including thermal stratification. No significant transients that have the ability to generate low-cycle fatigue are postulated to occur during normal operation of main steam piping. Therefore, low-cycle fatigue is not a potential failure mechanism of piping.

High-Cycle Fatigue

The reactor coolant pumps generate vibrations capable of causing high-cycle fatigue within the RCL piping. Operational controls are established to minimize vibrations during hot functional testing and normal operation. In addition, pump vibrations that could lead to adverse high-cycle fatigue in piping are also monitored by alarming when the vibrations exceed acceptable limits. These precautions maintain the piping in operational ranges with very low probability of failure.

3.6.3.3.6 Thermal Aging

RCL Piping and RCL Branch Piping

Stainless casting is not used for the RCL and RCL branch piping. The fracture toughness is also not a concern, because the amount of the ferrite contents in welding material is low.

Main Steam Piping

Material selected for the MSS piping is not susceptible to dynamic strain aging effects.

3.6.3.3.7 Other Mechanisms

Failures at piping cleavages are not credible in stainless steel piping systems operated within the thermal parameters defined for the RCS. Carbon steel used for main steam piping is resistant to cleavage failure at defined operating temperatures. Material fracture toughness tests demonstrate the ability of selected piping to perform under defined operating conditions.

Other postulated pipe failure mechanisms include the effects of secondary events such as surrounding SSC failures or missile impacts.

Surrounding SSCs are designed to preclude damage to RCL piping, RCL branch piping, and main steam piping by meeting seismic category I or seismic category II requirements. Therefore, the secondary effects of surrounding SSCs will have no adverse effect on safety-related SSCs.

Damage from missiles is precluded by design or separation in accordance with Section 3.5.

3.6.3.4 Analytical Methods and Criteria

The method and criteria used for LBB analysis are consistent with the guidelines in NUREG-1061 (Reference 3.6-23) and Standard Review Plan 3.6.3, Rev. 1 (Reference 3.6-4).

LBB BACs are prepared for each applicable piping system. These curves provide the design guidelines meeting the allowable standards for stress limits and LBB acceptance criteria. The critical location having the highest stress point from piping analysis is determined and compared to the BAC. The maximum stress location must be on or below the BAC to satisfy the LBB criteria.

The bounding analysis methods are described in Appendix 3B. Preparation of BAC provides an evaluation method meeting the requirements and guidelines of the NRC documents.

Piping analysis boundary is from one terminal end or anchor to the other terminal end or anchor. Connection to a larger pipe or a component of larger diameter is generally considered a terminal end. LBB evaluation is based on the fracture mechanics of cracks and analysis of break mechanism which compares the selected leakage cracks with critical crack sizes. This analysis method is outlined below.

Crack stability is demonstrated by leak detection analysis on the assumption that postulated circumferential cracks are limited if the stresses are on or below the “LBB BAC.”

3.6.3.4.1 Leak Detection Capability

Leakage flaws are postulated for piping identified in Subsection 3.6.3.1 as following. Sizes of postulated flaws are sufficiently large so that leaks can be detected by a sufficient margin. Leak rate of 10 times the capability of the leak detector is postulated for normal operating load combinations.

Rated detection capability of the leak detector for reactor coolant in the containment is 0.5 gpm within one hour of detector response time. The methods used for the reactor coolant are the containment sump water levels, inventory balance, and the radiation in the environment of containment. The method to detect leaks from the main steam pipe in containment is the containment sump water level. Leaks can also be detected by the condensate water flow rate of containment air cooler. The containment environmental pressures and temperatures also suggest the possibility of leakage.

3.6.3.4.2 Stability and Critical Crack Sizes

The local and global break mechanisms are evaluated, as required, to provide a margin to the break size and load. Local mode of breaks deals with the behaviors of crack tips: slowdown, start, development, and instability. Mechanisms of local breaks are evaluated by using J integration method for ferritic steel pipes. Global break mode deals with the behaviors of all cross sections: initial yield, strain hardening, and plastic hinge formation. Global break mechanisms (critical loading method) are evaluated for the stainless steel pipes not containing the casting materials and shielded metal arc weld. From these evaluations, the critical crack sizes are determined, so that the cracks larger than critical crack size have unstable features of growth.

3.6.3.4.3 Allowable Standards

Crack size margin is determined by comparing the crack sizes determined above to the critical crack size. The critical crack size is determined by adding maximum individual loads by absolute summation. The margin of two applies to the leakage crack size compared to the critical crack size with the size of flaw large enough so that the leakage from the flaw during normal operation is 10 times greater than the minimum leakage the detection system is capable of sensing. The margin of 1.0 on the load is used, since the loads are added by absolute sum.

3.6.3.4.4 Bounding Analysis Methods

BACs are developed for each different combination of material type, pipe size, pressure and temperature. These curves provide “allowable maximum stress” versus “normal operating stress” to define bounding stress combinations meeting LBB standards. These curves are used to satisfy the requirements for LBB. LBB stress criteria are satisfied if the predicted maximum stress and normal operating stress pair combinations for all analyzed locations for the piping system (including all welds) are at or below the BACs.

For evaluation of the normal and maximum stresses, loads related to maximum stress calculation are added by using absolute sums. Loads are combined as shown below.

I *Pressure* I+ I *Dead Load* I + I *Thermal* (100% *power*) I + I *SSE* I

For the surge line, a second loading combination is evaluated to assess the maximum heatup/cooldown stratified condition in the line.

I *Pressure* I + I *Dead Load* I + I T*hermal* (0% *power*)* I

* Including maximum (thermal) stratification loads

Normal operating stresses are calculated according to the following combinations of loads by using arithmetical sum method at critical location.

Pressure + *Dead Load* + *Thermal* (100% *power*)

Stresses by longitudinal force and bending moment are calculated by the following formula.

$\sigma = F/A + M/Z$

where

σ = Stress

F = Longitudinal force

M = Bending moment

A = Cross-sectional area

Z = Section modulus

Bending moments for the applicable load combination are calculated by the following formula:

$$M = \sqrt{MX^2 + MY^2 + MZ^2}$$

where

M = Bending moment related to applicable load

MX = X component of bending moment

MY = Y component of bending moment

MZ = Z component of bending moment

The Y and Z axes are those perpendicular to the longitudinal direction X.

The longitudinal loads and bending moments in the normal and maximum cases are calculated by the method shown below.

Normal operating loads are calculated by the following formula.

$F = F_{DW} + F_{Th} + F_P$

$MX = (MX)_{DW} + (MX)_{Th}$

$MY = (MY)_{DW} + (MY)_{Th}$

$MZ = (MZ)_{DW} + (MZ)_{Th}$

Subscripts in the above formulae indicate the following load cases.

DW = Dead load

Th = Normal thermal expansion (100% power)*

P = Internal pressure

* Including applicable (thermal) stratification loads

Loads related to maximum stress conditions are calculated by the following formulae.

$$F = |F_{DW}| + |F_{Th}| + |F_P| + |F_{SSEINERTIA}| + |F_{SSEAM}|$$

$$MX = |(MX)_{DW}| + |(MX)_{Th}| + |(MX)_{SSEINERTIA}| + |(MX)_{SSEAM}|$$

$$MY = |(MY)_{DW}| + |(MY)_{Th}| + |(MY)_{SSEINERTIA}| + |(MY)_{SSEAM}|$$

$$MZ = |(MZ)_{DW}| + |(MZ)_{Th}| + |(MZ)_{SSEINERTIA}| + |(MZ)_{SSEAM}|$$

Subscripts in the above formulae indicate the following load cases.

Subscripts *DW, Th* and *P* indicate the same load cases as normal case.

$SSEINERTIA$ = SSE Inertia

$SSEAM$ = SSE Anchor Motion.

In order to compare the results of stress calculations with the BACs, the following procedure is used. The allowable “Maximum stress” is plotted versus “Normal stress” on the BAC for a particular piping system. LBB analysis and margin are met if the stress combinations for all analyzed locations for the piping system are on or below the BAC.

3.6.3.4.5 Develop the BACs

BACs for LBB piping systems are included in Appendix 3B.

3.6.3.4.6 Pipe, Material, and Operating Conditions

Data is summarized in Table 3B-1 in Appendix 3B.

3.6.3.4.7 Piping Physical Properties

Data is summarized in Table 3B-1 in Appendix 3B.

3.6.3.4.8 Calculation Steps

LBB Evaluation procedure is shown on the flow chart of Figure 3.6-4.

3.6.3.4.9 Evaluation of Piping System Using BAC

This information will be developed as described in Appendix 3B.

3.6.3.4.10 Bounding Analysis Results

Bounding analysis results will be documented as described in Subsection 3.6.3.4.13.

3.6.3.4.11 Differences in Inspection Criteria for Class 1 and 2 Systems

Class 1 and 2 systems are subjected to ISI requirements from ASME Code, Section XI (Reference 3.6-11). For Class 1 piping, terminal ends and dissimilar metal welds are volumetrically inspected, along with other locations, to total 25% of the welds. For Class 2 piping, the requirement is to volumetrically inspect the terminal ends and other locations to total 7.5% of the welds. These requirements were developed by ASME Code, Section XI consistent with the different safety classes of these systems.

The LBB evaluations are based on the ability to detect a potential leaking crack; not the ability to find cracks by ISI. The criteria or methods of the LBB evaluations are the same for ASME Code, Section III, Class 1 and 2 (References 3.6-12 and 3.6-9).

3.6.3.4.12 Differences in Fabrication Requirements of ASME Code, Section III Class 1 and Class 2 Piping

The significant difference among ASME Code, Section III, Class 1 and 2 seamless pipes occurs in the nondestructive examination requirements. The Class 1 seamless pipe examination requirements include an ultrasonic testing examination, whereas Class 2 does not. In addition, the Class 1 examination requirements for a circumferential butt welded joint include radiographic testing and magnetic particle or liquid penetrant examination where Class 2 does not.

For the fabrication of welds in the ASME Code, Section III, Class 1 and Class 2 pipes, there are no significant differences.

The differences in fabrication and nondestructive examination requirements do not affect the LBB analyses assumptions, criteria, or methods.

3.6.3.4.13 Documentation of LBB Evaluation

Documentation of the LBB evaluation will be developed for the as-built piping systems to which LBB criteria are applied and include the information described in subsection 3.6.3.5.

3.6.3.5 Technical Report

The following information will be developed as part of the LBB evaluation of as-built piping systems to which LBB criteria are applied.

- Representative and bounding material properties.
- Design piping isometric drawings showing location of supports and their characteristics and location and weights of components such as valves.
- Evaluation of piping system using BAC and bounding analysis results.

3.6.4 Combined License Information

COL 3.6(1) *The COL Applicant is to identify the site-specific systems or components that are safety-related or required for safe shutdown that are located near high-energy or moderate-energy piping systems, and are susceptible to the consequences of these piping failures. The COL Applicant is to provide a list of site-specific high-energy and moderate-energy piping systems, which includes a description of the layout of all piping systems where physical arrangement of the piping systems provides the required protection, the design basis of structures and compartments used to protect nearby essential systems or components, or the arrangements to assure the operability of safety-related features where neither separation nor protective enclosures are practical. Additionally, the COL Applicant is to provide the failure modes and effect analyses that verifies the consequences of failures in site-specific high-energy and moderate-energy piping does not affect the ability to safely shut down the plant. The COL Applicant is to update the as-design pipe hazards analysis report to include the impact of all site specific high and moderate piping systems.*

COL 3.6(2) *Deleted*

COL 3.6(3) *Deleted*

COL 3.6(4) *The COL Applicant is to implement the criteria for defining break and crack locations and configurations for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to identify the postulated rupture orientation of each postulated break location for site-specific high-energy and moderate-energy piping systems. The COL Applicant is to implement the appropriate methods to assure that as-built configuration of site-specific high-energy and moderate-energy piping systems is consistent with the design intent and provide as-built drawings showing component locations and support locations and types that confirms this consistency.*

COL 3.6(5) *Deleted*

COL 3.6(6) *Deleted*

COL 3.6(7) *Deleted*

COL 3.6(8) *Deleted*

COL 3.6(9) *Deleted*

COL 3.6(10) *The COL Applicant is to develop a milestone schedule for implementation of the operating and maintenance procedures for prevention of water hammer.*

3.6.5 References

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3.6-2 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. NUREG-0800, SRP 3.6.1, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

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Table 3.6-1 High and Moderate Energy Fluid Systems

Systems(3)	High-Energy(1)	Moderate-Energy(1)
Reactor Coolant System (RCS)	X	-
Chemical and Volume Control System (CVCS)	X	-
Safety Injection System (SIS)	X	-
Residual Heat Removal System (RHRS) (2)	-	X
Emergency Feedwater System (EFWS) (2)	-	X
Feedwater System (FWS)	X	-
Main Steam Supply System (MSS)	X	-
Containment Spray System (CSS)	-	X
Component Cooling Water System	-	X
Spent Fuel Pit Cooling and Purification System (SFPCS)	-	X
Essential Service Water System (ESWS)	-	X
Gaseous Waste Management System (GWMS)	-	X
Liquid Waste Management System (LWMS)	-	X
Process and Post-Accident Sampling System (PSS)	X	-
Steam Generator Blowdown System (SGBDS)	X	-
Refueling Water Storage System (RWS)	-	X
Primary Makeup Water System (PMWS)	-	X
Instrument Air System (IAS)	-	X
Fire Protection Water Supply System (FSS)	-	X
Station Service Air System (SSAS)	-	X
Essential Chilled Water System (ECWS)	-	X
Non-Essential Chilled Water System (non-ECWS)	-	X
Demineralized Water System (DWS)	-	X
Potable and Sanitary Water System (PSWS)	-	X
Compressed Gas System (CGS)	-	X
Emergency Gas Turbine Auxiliary System	-	X
Alternate Alternating Current Gas Turbine System	-	X

Notes

1. High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig.

   Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high-energy are considered as moderate-energy.

   Piping systems that exceed 200°F or 275 psig for two percent or less of the time during which the system is in operation are considered moderate-energy.

2. The RHRS and EFWS lines are classified as moderate-energy based on the 2 percent rule. These lines experience high-energy conditions for less than 2 percent of the system operation time. The portions of the RHR system from the connections to the RCS to the first closed valve in each line are high-energy.

3. Systems included on this list are high-energy or moderate-energy fluid systems located in the prestressed concrete containment vessel, reactor building or power source buildings. Systems that operate at or close to atmospheric pressure such as ventilation and gravity drains are not included.

Table 3.6-2 List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Internal and External Fluids (Sheet 1 of 4)

No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)	Pres-sure (psig)	Inside Pipe	Outside Pipe (°F , psig)
1	RCS	Primary Loop Hot Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	617	2235	Subcooled liquid	Air (120, 0)
1	RCS	Primary Loop Hot Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	617	2235	Subcooled liquid	Air (120, 0)
2	RCS	Primary Loop Crossover Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	550.6	2235	Subcooled liquid	Air (120, 0)
3	RCS	Primary Loop Cold Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316	550.6	2235	Subcooled liquid	
2	RCS	Primary Loop Crossover Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	550.6	2235	Subcooled liquid	Air (120, 0)
3	RCS	Primary Loop Cold Leg	31"ID-RCS-2501R A,B,C,D	31ID	37.12	3.06	SA182 F316LN	550.6	2235	Subcooled liquid	
4	RCS	Surge Line	16"-RCS-2501R B	16	16	1.594	SA-312 TP316	653	2235	Saturated liquid	Air (120, 0)
5	RCS	Surge Line	16"-RCS-2501R A	16	16	1.594	SA-312 TP316	449	400	Saturated liquid	Air (120, 0)
6	RCS	Residual Heat Removal System (RHRS) Hot Leg Branch Line off RCS	10"-RCS-2501R A,B,C,D, Hot Leg Side	10	10.75	1.125	SA-312 TP316	617	2235	Subcooled liquid	Air (120, 0)
7	RCS	RHRS Cold Leg Branch Line off RCS	8"- RCS -2501R A,B,C,D (COLD LEG)	8	8.625	0.906	SA-312 TP316	550.6	2235	Subcooled liquid	Air (120, 0)
8	SIS	Accumulator System	14"-RCS-2501R A,B,C,D	14	14	1.406	SA-312 TP316	550.6	2235	Subcooled liquid	Air (120, 0)
9	RCS	Pressurizer Spray Line	6"-RCS-2501R B,C	6	6.625	0.719	SA-312 TP316	550.6	2235	Subcooled liquid	Air (120, 0)
10	MSS	Main Steam Line	32"-MSS-1532N A,B,C,D	32	32	1.496	SA-106 Gr.B	535	907	Saturated steam	Air (130, 0)
11	CVS	Aux. Spray Line	3"-RCS-2501	3	3.5	0.438	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
12	CVS	Aux. Spray Line	3"-CVS-2561	3	3.5	0.438	SA-312 TP316	554.6	2366	Subcooled liquid	Air (120, 0)
13	CVS	Charging Line	4"-CVS-2501	4	4.5	0.531	SA-312 TP316	554.6	2366	Subcooled liquid	Air (120, 0)

Table 3.6-2 List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Internal and External Fluids (Sheet 2 of 4)

No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)	Pres-sure (psig)	Inside Pipe	Outside Pipe (°F , psig)
14	CVS	Charging Line	4"-CVS-2561	4	4.5	0.531	SA-312 TP316	554.6	2366	Subcooled liquid	Air (120, 0)
15	CVS	Charging Line	4"-CVS-2511 (Inside CV)	4	4.5	0.531	SA-312 TP304	130	2600	Subcooled liquid	Air (120, 0)
16	CVS	Charging Line	4"-CVS-2511 (Outside CV)	4	4.5	0.531	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
17	CVS	Charging Line	3"-CVS-2511	3	3.5	0.438	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
18	CVS	Charging Line	2"-CVS-25B1	2	-	-	-	130	2600	Subcooled liquid	Air (105, 0)
19	RCS	MCP Drain	2"-RCS-2501	2	2.375	0.344	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
20	CVS	Letdown Line	2"-RCS-2501	2	2.375	0.344	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
21	CVS	Letdown Line	3"-RCS-2501	3	3.5	0.438	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
22	CVS	Letdown Line	3"-CVS-2501	3	3.5	0.438	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
23	CVS	Letdown Line	3"-CVS-2561	3	3.5	0.438	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
24	CVS	Letdown Line	3"-CVS-0601	3	3.5	0.216	SA-312 TP304	380	350	Subcooled liquid	Air (120, 0)
25	CVS	Letdown Line	4"-CVS-0601	4	4.5	0.237	SA-312 TP304	380	350	Subcooled liquid	Air (120, 0)
26	CVS	Letdown Line	4"-CVS-06A1	4	-	-	-	200	350	Subcooled liquid	Air (105, 0)
27	SIS	Emergency Let-down Line	2"-RCS-2501	2	2.375	0.344	SA-312 TP316	621	2266	Subcooled liquid	Air (120, 0)
28	SIS	DVI Line	4"-RCS-2501	4	4.5	0.531	SA-312 TP316	554.6	2266	Subcooled liquid	Air (120, 0)
29	SIS	SI Pump Line	4"-RCS-2501	4	4.5	0.531	SA-312 TP316	621	2266	Subcooled liquid	Air (120, 0)
30	SIS	SI Pump Line	4"-SIS-2501	4	4.5	0.531	SA-312 TP316	621	2266	Subcooled liquid	Air (120, 0)
31	RCS	Pressurizer Safety Valve Line	6"-RCS-2501	6	6.625	0.719	SA-312 TP316	657	2266	Saturated steam	Air (120, 0)

Table 3.6-2 List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Internal and External Fluids (Sheet 3 of 4)

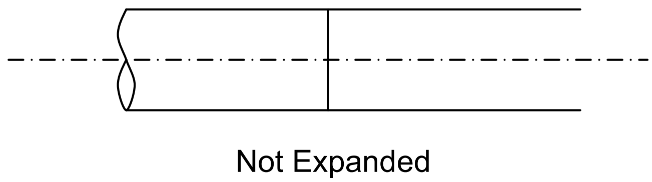
No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)	Pres-sure (psig)	Inside Pipe	Outside Pipe (°F , psig)
31	RCS	Pressurizer Safety Depressurization Valve Line	4"-RCS-2501	4	4.5	0.531	SA-312 TP316	657	2266	Saturated steam	Air (120, 0)
32	RCS	Pressurizer Safety Depressurization Valve Line	6"-RCS-2501	6	6.625	0.719	SA-312 TP316	657	2266	Saturated steam	Air (120, 0)
33	RCS	Pressurizer Safety Depressurization Valve Line	8"-RCS-2501	8	8.625	0.906	SA-312 TP316	657	2266	Saturated steam	Air (120, 0)
34	CVS	Seal Injection Line	1-1/2"-CVS-2501	1-1/2	1.9	0.281	SA-312 TP316	130	2266	Subcooled liquid	Air (120, 0)
35	CVS	Seal Injection Line	1-1/2"-CVS-2511	1-1/2	1.9	0.281	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
36	CVS	Seal Injection Line	1-1/2"-CVS-25B1	1-1/2	-	-	-	130	2600	Subcooled liquid	Air (105, 0)
37	CVS	Seal Injection Line	1"-CVS-2511	1	1.315	0.250	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
38	CVS	Seal Injection Line	2"-CVS-2511	2	2.375	0.344	SA-312 TP304	130	2600	Subcooled liquid	Air (105, 0)
39	CVS	Seal Injection Line	2"-CVS-25B1	2	-	-	-	130	2600	Subcooled liquid	Air (105, 0)
40	SIS	Accumulator Tank Drain Line	2"-SIS-06A1	2	-	-	-	300	700	Subcooled liquid	Air (120, 0)
41	SIS	Accumulator Tank Line	14"-SIS-2511	14	14	1.406	SA-312 TP304	300	2485	Subcooled liquid	Air (120, 0)
42	SIS	Accumulator Tank Line	14"-SIS-0601	14	14	0.500	SA-312 TP304	300	700	Subcooled liquid	Air (120, 0)
43	EFS	Emergency Feed-water Pump Line	3"-FWS-1522	3	3.5	0.300	SA-106 Gr.B	471	1185	Subcooled liquid	Air (130, 0)
44	EFS	Emergency Feed-water Pump Tur-bine Line	6"-EFS-1532	6	6.625	0.432	SA-106 Gr.B	539	938	Subcooled liquid	Air (130, 0)
45	EFS	Emergency Feed-water Pump Tur-bine Line	6"-MSS-1532	6	6.625	0.432	SA-106 Gr.B	539	938	Subcooled liquid	Air (130, 0)
46	FWS	Feedwater Line	18"-FWS-1805	18	18	1.375	SA-335 Gr.P22	471	1850	Subcooled liquid	Air (130, 0)

Table 3.6-2 List of High Energy Lines for Pipe Break Hazard Analysis, Including Properties of Internal and External Fluids (Sheet 4 of 4)

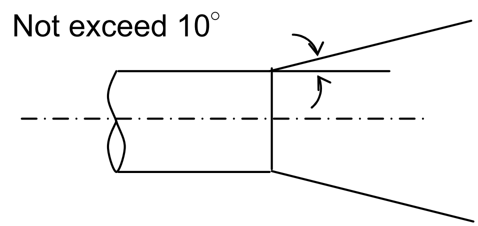
No.	System	Subsystem	Line No(s)	Nominal Diameter (Inches)	Outside Diameter (Inches)	Thickness (Inches)	Material	Temp (°F)	Pres-sure (psig)	Inside Pipe	Outside Pipe (°F , psig)
47	FWS	Feedwater Line	6"-FWS-1805	6	6.625	0.562	SA-335 Gr.P22	471	1850	Subcooled liquid	Air (130, 0)
48	FWS	Feedwater Line	16"-FWS-1525	16	16	0.844	SA-335 Gr.P22	471	1185	Subcooled liquid	Air (130, 0)
49	FWS	Feedwater Line	3"-FWS-1802	3	3.5	0.438	SA-106 Gr.B	471	1850	Subcooled liquid	Air (130, 0)
50	MSS	Main Steam Line	32"-MSS-1532	32	32	1.500	SA-106 Gr.B	539	938	Saturated steam	Air (130, 0)
51	MSS	Main Steam Line	6"-MSS-1532	6	6.625	0.432	SA-106 Gr.B	539	938	Saturated steam	Air (130, 0)
52	MSS	Main Steam Drain Line	2"-MSS-1532	2	2.375	0.218	SA-106 Gr.B	539	938	Saturated liquid	Air (130, 0)
53	MSS	Main Steam Drain Line	4"-MSS-1532	4	4.5	0.337	SA-106 Gr.B	539	938	Saturated liquid	Air (130, 0)
54	SGS	SGBD Line	3"-SGS-1532	3	3.5	0.300	SA-106 Gr.B	539	938	Saturated liquid	Air (120, 0)
55	SGS	SGBD Line	4”-SGS-1532 (Inside CV)	4	4.5	0.337	SA-106 Gr.B	539	938	Saturated liquid	Air (120, 0)
56	SGS	SGBD Line	4”-SGS-1532 (Outside CV)	4	4.5	0.337	SA-106 Gr.B	539	938	Saturated liquid	Air (105, 0)
57	SGS	SGBD Line	3/8”-SGS-2521	3/8	-	-	-	539	938	Saturated liquid	Air (120, 0)
58	SGS	SGBD Line	3/8”-SGS-25CA	3/8	-	-	-	539	938	Saturated liquid	Air (105, 0)

Security-Related Information – Withheld Under 10 CFR 2.390

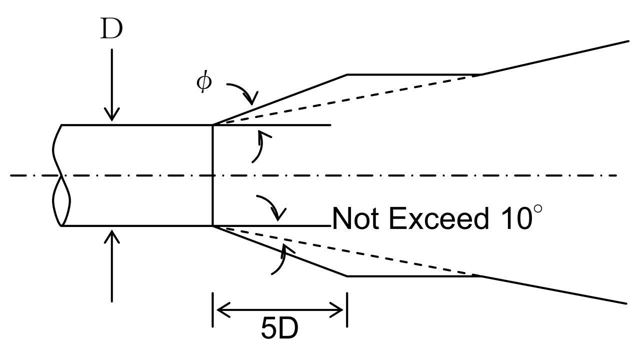
Figure 3.6-1 Break Exclusion Region-Main Steam Pipe Room



a. Sub-cooled water (non-flashing)



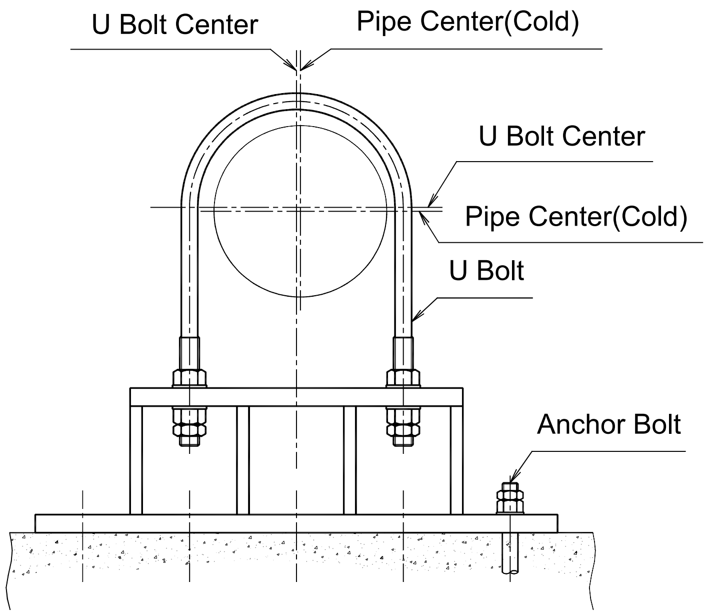
b. Sub-cooled water (flashing) and saturated water



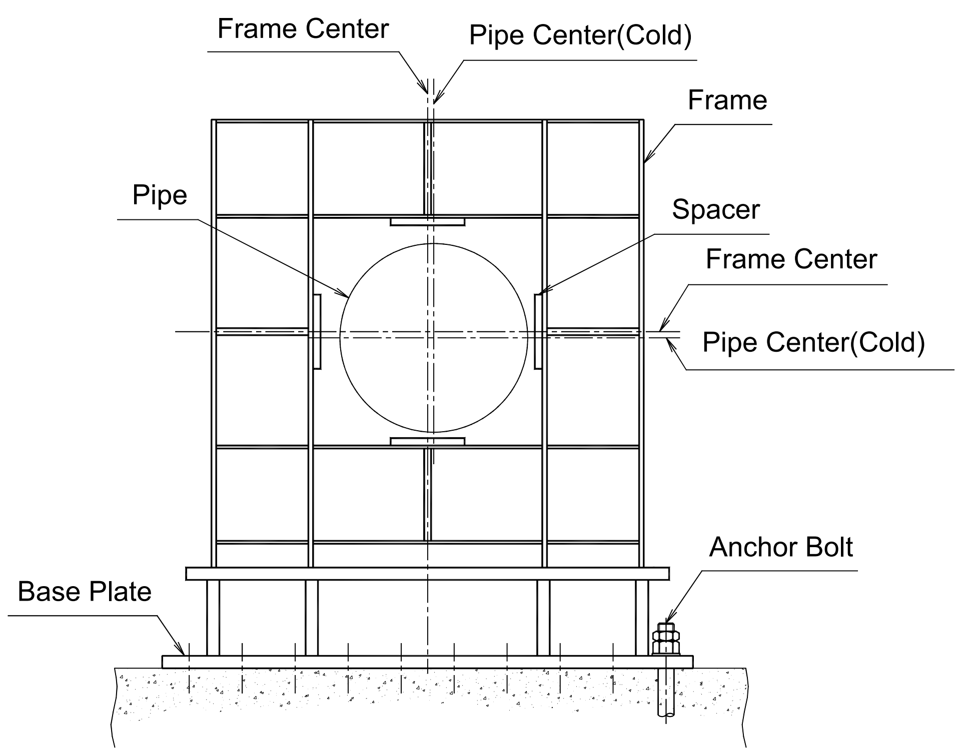
However, φ is calculated in assumption of isenthalpic expansion.

c. Saturated steam

Figure 3.6-2 Characteristics of Discharging Jets



a. U-Bar (One-Dimensional Restraint)



b. Structural Steel (Two-Dimensional Restraint)

Figure 3.6-3 Typical Rupture Restraints

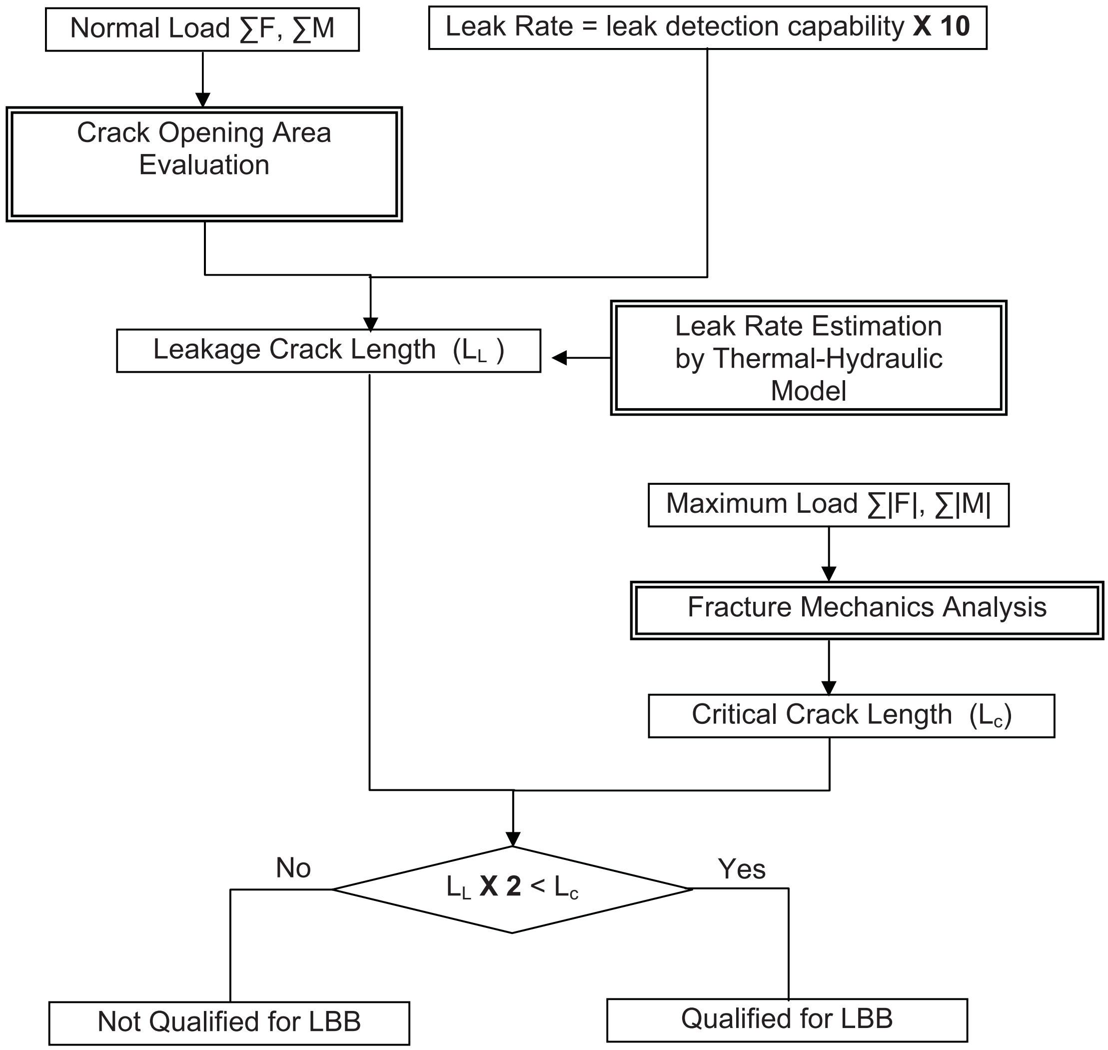


Figure 3.6-4 LBB Evaluation Procedure

3.7 Seismic Design

The SSCs of the US-APWR are designed as required by the GDC 2 of 10 CFR 50, Appendix A (Reference 3.7-1), to withstand the effects of natural phenomena, including earthquakes, without jeopardizing the plant safety. The US-APWR SSCs are assigned to one of three seismic categories (seismic category I, seismic category II, or non-seismic [NS]) depending on the nuclear safety function of the particular SSC, as discussed in Subsection 3.2.1. The US-APWR standard plant seismic design is based on the SSE and the OBE as discussed in Subsection 3.7.1.1. The OBE defines the magnitude of the ground motion that if exceeded would require that the plant be shut down.

The values of peak ground accelerations (PGAs) and the response spectra of the seismic ground motion in horizontal and vertical directions define the magnitude of the design basis earthquake. Certified seismic design response spectra (CSDRS) are used as the site-independent SSE for the seismic design of standard plant structures, and the ground motion response spectra (GMRS) define the horizontal and vertical response spectra of the site-dependent SSE design motion.

The COL Applicant is to validate the site-independent seismic design of the standard plant for the site-specific conditions, including geological, seismological, and geophysical characteristics, and to develop the site-specific GMRS and foundation input response spectra (FIRS).

The COL Applicant is responsible for the seismic design of those seismic category I and seismic category II SSCs that are not part of the US-APWR standard plant using site-specific SSE design ground motion. The response spectra of site-specific SSEs are developed following the requirements of RG 1.208 (Reference 3.7-3). The COL Applicant is to develop site specific GMRS and FIRS. The FIRS are compared to the CSDRS to assure that the US-APWR standard plant seismic design is valid for a particular site. If the FIRS are not enveloped by the CSDRS, the US-APWR standard plant seismic design is modified as part of the COLA in order to validate the US-APWR for installation at that site.

3.7.1 Seismic Design Parameters

3.7.1.1 Design Ground Motion

The Peak Ground Acceleration (PGA) of the design ground motion used for the purpose of the site-independent design of the seismic category I SSCs of the US-APWR standard plant is 0.3 g for the two horizontal directions and the vertical direction. The COL Applicant is to confirm that the site-specific PGA at the basemat level control point of the CSDRS is less than or equal to 0.3 g.

Design Ground Motion Response Spectra

Horizontal and vertical response spectra define the design seismic ground motion used for the US-APWR standard plant seismic design. The SSE, CSDRS, Site Specific GMRS, FIRS and OBE, and the spectra, which are used to characterize these earthquake motions, are discussed in the following paragraphs.

SSE

The SSE is the earthquake which produces the maximum vibratory ground motion for which certain SSCs are designed to remain functional and within applicable stress, strain, and deformation limits.

The SSCs that must remain functional are those necessary to assure the following:

1. The integrity of the RCPB.

2. The capability to shut down the reactor and maintain it in a safe-shutdown condition.

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100 (Reference 3.7-4).

The CSDRS are used as the SSE for the site-independent design of the US-APWR standard plant seismic category I and seismic category II SSCs. The major seismic category I buildings and structures of the US-APWR standard plant are the R/B, PCCV, containment internal structure, (CIS), east PS/B, west PS/B, and essential service water pipe chase (ESWPC) all on a common basemat. The common basemat also includes the seismic category II A/B. This combination of buildings on the common mat is defined as the R/B complex.

For the seismic design of seismic category I and seismic category II SSCs that are not part of the US-APWR standard plant, and for the detailed design of the US-APWR standard plant structures that are modified for the site-specific conditions , a site-specific SSE can be used. The site-specific SSE is developed following the requirements of RG 1.208 (Reference 3.7-3).

CSDRS

The CSDRS are presented as the site-independent seismic design response spectra for an approved certified design of the US-APWR standard nuclear power plant. The CSDRS are identified as an outcrop motion in the free field at the same level as the bottom of the foundation of the R/B complex.

The site-independent CSDRS that are employed for the seismic category I design of the US-APWR standard plant are shown for 2%, 3%, 5%, 7%, and 10% damping values in Figures 3.7.1-1 and 3.7.1-2 for the horizontal and vertical components, respectively. The CSDRS are derived from RG 1.60 (Reference 3.7-6) spectra by scaling the spectra contained in RG 1.60 from 1.0 g to 0.3 g zero period acceleration (ZPA) values, and by modifying the RG 1.60 control points to broaden the spectra in the higher frequency range. The RG 1.60 spectral values are based on deterministic values for western United States earthquakes. NUREG/CR-6728 (Reference 3.7-14) indicates that earthquakes in the central and eastern United States (CEUS) have more energy content in the higher frequency range than earthquakes in the western United States. Thus, the RG 1.60 (Reference 3.7-6) spectra control points have been modified by shifting the control points at 9 Hz and 33 Hz to 12 Hz and 50 Hz, respectively, for both the horizontal and the vertical spectra. Therefore, for the US-APWR CSDRS, the horizontal spectra control points are at 0.25, 2.5, 12, and 50 Hz and the vertical response spectra control points are at 0.25, 3.5, 12, and 50 Hz. The modified RG 1.60 (Reference 3.7-6) spectra used for the CSDRS are expected to envelope many sites in the central and eastern United States in order to maximize the applicability of the US-APWR standard plant design; however, it is anticipated that there are some site-specific instances, particularly on hard rock sites where high-frequency exceedances of the CSDRS may occur at close distances (≤15 km) from larger magnitude (M ~ 5) earthquake sources. In these cases, the COL Applicant may consider the seismic wave transmission incoherence of the input ground motion when performing the site-specific SSI analyses.

Consistent with RG 1.60 (Reference 3.7-6), the CSDRS representing the vertical accelerations is obtained by scaling the horizontal acceleration response spectra (ARS) by a factor of 2/3 for frequencies less than 0.25 Hz. The scaling factor that varies from 2/3 to 1.0 is applied for the frequency range between 0.25 and 3.5 Hz. The horizontal and vertical acceleration spectra are kept identical above frequency 3.5 Hz and, consequently, the vertical PGA is taken as the same as the horizontal PGA.

The US-APWR design response spectral accelerations for each of the spectral control points are presented in Tables 3.7.1-1 and 3.7.1-2. The US-APWR site-independent CSDRS as defined herein meet the requirements of 10 CFR 50, Appendix S(IV)(a)(1)(i) (Reference 3.7-7), which require that the horizontal component of the SSE ground motion in the free-field at the basemat level of the structures must be an appropriate response spectra with a PGA of at least 0.1 g.

Site-Specific GMRS

In accordance with NUREG-0800, SRP 2.5.2 (Reference 3.7-8), the site-specific GMRS, developed by the COL Applicant, define the site-specific SSE through a horizontal and vertical response spectra of the free-field motion that is specified either on the ground surface or at an outcrop (real or hypothetical) of the uppermost in-situ competent material that will exist after excavation. The competent material is defined as having a shear wave velocity of 1,000 ft/s or greater. Free-field ground motion is defined as the seismic motion of the ground that is not influenced by the presence of any basemats and structures.

Site-specific GMRS are developed at a sufficient number of frequencies (at least 25) that adequately represent the local and regional seismic hazards using the site-specific geological, seismological, and geophysical input data. A probabilistic seismic hazard analysis is performed that is based on the performance-based approach outlined in RG 1.208 (Reference 3.7-3). Horizontal GMRS are developed using a site amplification function obtained from site response analyses performed on site-specific soil profiles that include the layers of soil and rock over the generic rock conditions defined by the attenuation relationships used in the probabilistic seismic hazard analysis (PSHA). For example, attenuation relationships for the CEUS typically define generic rock as the rock with shear wave velocity exceeding 9,200 ft/s. Randomized site-specific soil profiles are used to account for the uncertainties and variations of the site soil and rock properties. The site response analysis will address probable effects of non-linearity due to strain-dependence of the subgrade materials' response. Equivalent linear methodology can be utilized with soil stiffness and damping degradation curves that represent the stiffness and damping properties of the subgrade materials as a function of strain. However, the strain-compatible soil material damping shall not exceed 15% as stipulated in SRP 3.7.1 (Reference 3.7-10).

With respect to determining the site-specific GMRS, note that Section 2.5.4 requires site-specific characterization of subsurface materials and investigation of the associated engineering properties to assure consistency with Section 3.7.2. Further, vertical GMRS are developed by combining the horizontal GMRS and the most up-to-date vertical/horizontal response spectral ratios appropriate for the site obtained from the most up-to-date attenuation relationships.

FIRS

The site-specific FIRS define the horizontal and vertical response spectra of the outcrop ground motion at the bottom elevation of the seismic category I and II basemats. Free-field outcrop spectra of site-specific horizontal ground motion are developed consistent with the horizontal GMRS using site response analyses which employ a suite of randomized soil profiles to account for uncertainties and variations in the site soil and rock properties. The profiles also include materials present above the input ground motion control point elevation in order to account for their effect on soil and rock properties.

Appendix S (IV)(a)(1)(i) of 10 CFR 50 (Reference 3.7-7) requires that the SSE ground motion in the free-field at the basemat level must be represented by an appropriate response spectra with a PGA of at least 0.1 g. This requirement is met on a site-specific basis by considering minimum horizontal response spectra that are tied to the shapes of the US-APWR CSDRS and anchored at 0.1g. Since the CSDRS are based on modified RG 1.60-spectra, this assures that there is sufficient energy content in the low-frequency range. The COL Applicant is to assure that the horizontal FIRS defining the site-specific SSE ground motion at the bottom of seismic category I or II basemats envelope the minimum response spectra required by 10 CFR 50, Appendix S (Reference 3.7-7), and the site-specific response spectra obtained from the response analysis. The same requirements apply to the vertical FIRS, which are developed from the horizontal FIRS by using vertical/horizontal response spectral ratios appropriate for the site.

The COL Applicant is to perform an analysis of the US-APWR standard plant seismic category I design to verify that the site-specific FIRS at the basemat level control point of the CSDRS are enveloped by the site-independent CSDRS. If the verification analysis proves the site-independent seismic design to be inadequate, a reanalysis of the affected SSCs is performed based on a site-specific SSE defined by the site-specific FIRS. In this case, the scoping re-design analysis may focus on affected SSCs rather than a complete analysis of all SSCs.

OBE

The OBE specifies the magnitude of ground motion that requires the shutdown of the plant operations. Appendix S of 10 CFR 50 (Reference 3.7-7) stipulates that the magnitude of an OBE can be adopted either as (A) 1/3 or less of the SSE; or (B) a value greater than 1/3 of the SSE. For Option A, the Applicant is not required to perform explicit response or design analyses. If Option B is chosen, an explicit analysis and design must be performed to demonstrate that all SSCs necessary for the continued operation without undue risk to the health and safety of the public will remain functional within applicable stress, strain, and deformation limits. For the US-APWR standard plant, the OBE is defined as 1/3 of the SSE (which is the CSDRS). Therefore, no specific analysis is required for the standard plant.

The COL Applicant is to set the value of the OBE that serves as the basis for defining the criteria for shutdown of the plant, according to the site-specific conditions. Subsection 3.7.4 describes the criteria and the seismic instrumentation used to determine whether the OBE has been exceeded.

It is recognized that during the life of the plant, the site may be subjected to seismic excitations of lower levels than the SSE. This can have an effect of reducing the “life expectancy” of those items sensitive to fatigue (i.e., piping, electrical, and mechanical equipment). Earthquake cycles are considered in the fatigue evaluation of the ASME Code, Section III, Class 1, 2, and 3. Components and Core Support Structures (Reference 3.7-11) (when required by the ASME Code) are discussed further in Sections 3.9 and 3.12, and in Section 3.10 for qualification testing of equipment. For fatigue evaluations, based on the OBE defined as less than or equal to 1/3 of the SSE, the guidance for determining the number of earthquake cycles for use in fatigue calculations is the same as the guidance provided in the U.S. NRC Staff Requirements Memorandum SECY-93-087 (Reference 3.7-12) for piping systems. The number of earthquake cycles to consider is two SSE events with 10 maximum stress cycles per event. Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than 1/3 of the maximum SSE amplitude) when derived in accordance with Institute of Electrical and Electronic Engineers (IEEE), Standard 344-2004, Appendix D (Reference 3.7-13).

Design Ground Motion Time History

A set of three statistically independent artificial ground motion time histories is generated in accordance with guidance of SRP 3.7.1 (Reference 3.7-10), Subsection 3.7.1.II.1B, Option 1 Approach 1, for use in US-APWR standard plant seismic analysis. These time histories represent ground motion for the three orthogonal directions, two horizontal ("H1" in the north-south [NS] direction, and "H2" in the east-west [EW] direction) and one vertical ("V"). Five additional sets of artificial ground motion time histories are developed as described in Section 3.8.5.5.2 to address sliding.

SRP 3.7.1 (Reference 3.7-10), Subsection 3.7.1.II SRP Acceptance Criteria 1B, Option 1 Approach 1 provides methodology used to generate a design basis time history with three components compatible with the CSDRS from seed recorded earthquake ground motions. The seed used to develop the design basis time history is a segment including the strong motion portion of the BAL (Mount Baldy, CA) recordings, i.e., the January 17th, 1994, Magnitude 6.7 Northridge Earthquake, obtained from the Pacific Earthquake Engineering Research (PEER) Center's digital ground motion library (Reference 3.7-56) recorded at the Mt. Baldy Station.

The BAL recordings of the Northridge earthquake are selected because they have the required durations and correlations (statistical independence among the three components) and because their spectral shapes, when scaled, are a reasonably good match to the CSDRS in the 2-20 Hz range for all three orthogonal components. The recorded time histories contain 4,000 digitized data points using a 0.01 second time step. The strong motion portion of the recorded time histories between t=11.0 to t=33.08 seconds, i.e., duration of 22.08 seconds, are extracted as the seeds which are developed to be compatible with the CSDRS. The digital acceleration records are linearly interpolated to obtain accelerations at every 0.005 seconds to enable the time histories to account for higher frequency content after adjustment such that their Nyquist frequency is 100 Hz.

The goal of the artificial time history development process is to produce modified time histories whose response spectra envelop the CSDRS for the US-APWR. In order to achieve this goal, the Fourier amplitudes of the seed acceleration time histories are modified to generate three new acceleration time histories. This Fourier amplitude modification process is iterated until the response spectra calculated from the modified Northridge time histories envelop the target CSDRS at damping ratios of 2%, 3%, 5%, 7%, and 10%.

Once the response spectra of the time histories envelop the CSDRS, the PSD envelope requirements are assessed. This development of PSD targets, and development of the PSD curves from the time histories, is done in conformance with guidance in NUREG/CR-5347 (Reference 3.7-59) Appendix B and SRP 3.7.1 (Reference 3.7-10) Appendix A. This process is described in more detail in MUAP-10006 (Reference 3.7-48). The target PSDs are shown in Figure 3.7.1-9. At frequencies with PSD lower than 80% of the target PSD, the Fourier amplitudes of the time histories are adjusted to satisfy the PSD requirements. Then a baseline correction is applied to the time histories.

Next, the resulting time histories are verified for their compliance with the SRP 3.7.1, Option 1, Approach 1 Acceptance Criteria. When necessary, the baseline corrected time histories are scaled to comply with the enveloping criteria for the spectra at the 2%, 3%, 5%, 7%, and 10% damping ratios, and the envelope requirements for the target power spectral density functions.

Finally, the time histories are checked for the requirements of strong motion duration, correlation coefficients, and V/A and AD/V^2 ratios, where A is the maximum ground acceleration, V is the maximum ground velocity, and D is the maximum ground displacement. The final modified Northridge time histories are the design basis time histories used as input ground motions for the SSI analyses.

The final design basis time histories are shown in Figure 3.7.1-3, Figure 3.7.1-4, and Figure 3.7.1-5. The corresponding velocity and displacement time histories have also been computed and are plotted in the same set of Figures. Each of these component time histories meets the criteria of SRP 3.7.1 Option 1, Approach 1. Compliance to these is summarized in Table 3.7.1-3.

Table 3.7.1-3 provides statistical independence values of the three components of the design basis time histories, which satisfies the pertinent SRP 3.7.1 criterion that the absolute value of correlation coefficients between the components must be less than 0.16.

As demonstrated in Table 3.7.1-3 the total durations of the design basis time histories meet the SRP guidance criteria that the durations exceed 20 seconds. The table also shows the rise time, strong motion duration, and decay time of each component. These values are computed based on the definition of strong motion duration in SRP 3.7.1, using the normalized Arias Intensity (AI). Figure 3.7.1-13 shows the normalized AI plots of cumulative energy for each component. The time history components show an initial time interval of gradual energy buildup, followed by a ramp of rapid energy accumulation, and then followed by a gradual tapering of energy accumulation. The strong motion duration should be at least six seconds according to SRP 3.7.1 and in compliance with duration criteria for earthquake magnitude and distance bins listed in Table 3.7.1-4. The strong motion durations of the design basis time history satisfy both duration criteria.

Table 3.7.1-3 also shows the V/A and AD/V^2 ratios for mean ratios ± one standard deviation for the earthquakes of magnitude bins of **M**6.5+ with distance bins from 10 to 100 km, using data provided in Table 3-6 of NUREG/CR-6728 (Reference 3.7-14). The V/A and AD/V^2 ratios of the design basis time histories are within the limits in Table 3.7.1-3.

Figure 3.7.1-6 through Figure 3.7.1-8 graphically demonstrate that the response spectra derived from the design basis time histories are developed in accordance with SRP 3.7.1 Option 1, Approach 1, for time history components 180 (H1), 090 (H2), and Vertical (UP), respectively. The response spectra of each component envelopes the CSDRS at 2%, 3%, 5%, 7%, and 10% damping values.

Figure 3.7.1-10 through Figure 3.7.12 show that the smoothed PSDs of the design basis time histories are greater than 80% of the horizontal and vertical target PSDs at all frequencies between 0.3 Hz and 50 Hz, for the three time history components.

Adequate representation of the Fourier components at low frequency is achieved by ensuring the artificial time history matches the CSDRS at all damping values and meets the PSD targets. As demonstrated above, the time histories developed from the Northridge Mt. Baldy seeds satisfy all the requirements described in the Option 1, Approach 1 of SRP 3.7.1 (Reference 3.7-10).

For site-specific design, the applicant will develop ground motion time histories that are compatible with the site-specific FIRS. The COL Applicant is to verify that the site-specific ratios V/A and AD/V^2 (*A, V, D*, are PGA, ground velocity, and ground displacement, respectively) are consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the site-specific uniform hazard response spectra. These parameters are examined to assure that they are consistent with the values determined for the low and high frequency events described in Appendix D of RG 1.208 (Reference 3.7-3).

The COL Applicant is to provide site-specific design ground motion time histories and durations of motion.

3.7.1.2 Percentage of Critical Damping Values

The dynamic FE models used for frequency domain SSI analyses described in subsection 3.7.2.4 use linear hysteric damping to account for the dissipation of energy in the subgrade materials and structural members. The hysteretic damping is proportional to the displacements of the dynamic system and is independent of frequency. The shear wave and compression wave damping coefficients (β_S and β_P) define the hysteric damping in the flowing complex formulations of the material shear modulus *G** and constrain modulus *M**:

$$G^* = G \cdot \left(1 - 2 \cdot \beta_S + 2 \cdot i \cdot \beta_S \cdot \sqrt{1 - \beta_S{}^2}\right) \qquad M^* = M \cdot \left(1 - 2 \cdot \beta_P + 2 \cdot i \cdot \beta_P \cdot \sqrt{1 - \beta_P{}^2}\right)$$

where: *G* and *M* are linear elastic shear and constrain modulii of the material and

$i = \sqrt{-1}$ is the complex number.

The strain compatible damping values assigned to the subgrade materials in the six generic profiles presented in Subsection 3.7.1.2 are well below the 15% limit set by SRP on shear wave damping and 10% limit on compression wave damping recommended by the correlation studies in Reference 3.7-62.

The same values of strain compatible shear wave damping are also used for compression wave damping in order to account, in a more realistic manner, for the dissipation of energy in the soil under the wave propagation pattern present in the SSI model. The seismic SSI analyses for horizontal and vertical seismic input motions assume that the input motions are caused by different horizontal and vertical seismic wave field excitations even though the seismic input environment is always 3-D consisting of simultaneous 3-component seismic input motions. Due to the simplified wave propagation assumption made in the SSI vertical motion analyses where the motion is applied as vertically propagating compression waves, strain iterated shear wave damping is assumed for compression wave damping to avoid unrealistic vertical motions at high frequencies.

Correlation studies of the vertical site response motions recorded in actual earthquakes with the vertical motions predicted from vertical One-Dimensional wave propagation site response analyses have been made (Reference 3.7-10).These studies conclude that, using the strain compatible soil damping values derived from the horizontal site response analyses as the damping values for vertical site response analysis, but limiting their values to no more than 10%, produces reasonably good correlation between the predicted and recorded vertical site response motions. Consistent with this conclusion, the soil damping values used for the horizontal and vertical SSI analyses are the shear strain compatible soil damping values derived from the horizontal free field site response analyses. The horizontal SSI response analyses are performed assuming vertically propagating plane shear wave field excitations. The vertical SSI response analysis is performed assuming vertically propagating plane compression wave field excitation, the shear strain compatible damping values derived from the horizontal site response analyses are used but with their values limited to no more than 10%, as recommended by the correlation studies reported in Reference 3.7-62.

The site response analyses use very low values for the material damping of hard base rock in order to model the low dissipation of energy in the deep hard rock strata. In order to improve the numerical stability of SSI results, the damping of the base rock material when included in the site profile is set to a low nominal value of 0.1%. This modification does not affect the SSI response because, unlike in the site response analyses, the thickness of the modeled hard rock strata in the SSI site model has a finite thickness so the use of higher damping values realistically projects the actual dissipation of energy in the base rock.

In the dynamic FE models used for frequency domain SSI analyses, the damping values listed in Table 3.7.2-3 are assigned both to the shear wave and compression wave damping coefficients representing the dissipation of energy in the different types of structural members. The damping values assigned to the structural model are consistent with the critical damping values specified in RG 1.61 (Reference 3.7-15) for elastic modal dynamic seismic analysis where energy dissipation is accounted for by frequency dependent viscous damping proportional to the velocity of the dynamic system. The implemented modeling approach results in the same amplitude of peak resonance responses for structures with viscous damping and hysteretic damping with less dissipation of energy occurring at other frequencies for the models with frequency independent hysteretic damping.

Two levels of stiffness and damping are developed and assigned to structural models used for seismic response analyses in order to capture structural stiffness and damping variations caused by concrete cracking: (1) full stiffness (uncracked concrete) corresponding to low stress levels; and (2) reduced stiffness (cracked concrete) corresponding to high stress levels.

In accordance with RG 1.61 (Reference 3.7-15) guidance and associated stress levels and industry standards, OBE structural damping values are used with the full stiffness (uncracked concrete) and SSE structural damping values are used with the reduced stiffness (cracked concrete). CIS and PCCV stiffness and damping are based on loading conditions as described in Section 3.7.2.3.5.

OBE structural damping values shown in Table 3.7.3-1(b) are for reinforced concrete, prestressed concrete and steel concrete modules assigned to the full stiffness (uncracked) model to calculate the effects from lesser dissipation of energy in the structures when they are subjected to low stress levels. SSE structural damping values shown in Table 3.7.3-1(a) are for reinforced concrete, prestressed concrete and steel concrete modules assigned to the reduced stiffness (cracked concrete) model to calculate the effects of greater dissipation of energy in the structures when they are subjected to higher stress levels.

The COL Applicant is to review the resulting level of seismic response and determine appropriate damping values for the site-specific calculations of ISRS that serve as input for the seismic analysis of seismic category I and seismic category II subsystems.

The damping coefficient values in Table 3.7.2-3 are assigned to the models used for the response spectra analyses to quantify the dissipation energy associated with the two bounding levels of stiffness. Unlike in the frequency domain SSI analyses where different values can be assigned to different finite elements, in the response spectra analysis and modal superposition time history analysis only a single value of critical damping is used to represent the dissipation of energy in the whole dynamic system.

The damping values for response spectra and modal superposition time history analyses of systems that include two or more substructures, such as a concrete and steel composite structure, may also be obtained using the strain energy method. This is the same as the stiffness weighted composite modal damping method as provided in to SRP 3.7.2 (Reference 3.7-16).

The stiffness weighted modal damping ratio h_j of the j^{th} mode is obtained from the following equation:

$$h_j = \frac{\vec{\phi}_j^{\,T}\left[\overline{K}\right]\vec{\phi}_j}{\vec{\phi}_j^{\,T}\left[K\right]\vec{\phi}_j}$$

where

$[K]$	=	the stiffness matrix of the combined soil-structure system
$\vec{\phi}_j$	=	the j^{th} normalized mode shape vector
$\left[\overline{K}\right] = \sum [k_i] \cdot \xi_i$	=	the modified stiffness matrix constructed from the products of the element stiffness matrices $[k_i]$ and the applicable damping ratio ξ_i

3.7.1.3 Supporting Media for Seismic Category I Structures

A range of soil parameters of the basemat supporting media are considered in the seismic design of seismic category I building structures for the US-APWR standard plant. The R/B complex is approximately 336 ft 4 in. in the north-south (NS) direction and 409 ft 8 in. in the east-west (EW) direction. The total footprint area is 127,016 ft^2. The nominal bottom elevation is -39 ft 8 in. Embedment depth is approximately 42 feet from grade which is at 2 ft 7 in. The basemat is nominally 13 ft 4 in. thick, however it is 30 ft 6 in. thick under the PCCV and 43 ft 3 in. thick under the CIS. See the figures in Section 1.2 for detailed elevation and plan views of the structure.

The minimum allowable static bearing capacity for the R/B complex is 15 ksf. The minimum allowable dynamic bearing capacity for the R/B complex is 35 ksf. These values are developed in Subsection 3.8.5.4.1. The dynamic bearing loads for seismic category I structure basemats are dependent upon the magnitude of the seismic loads that can be obtained from a site-specific seismic analysis that considers FIRS. The COL Applicant is to determine the allowable static and dynamic bearing capacities based on site conditions, including the properties of fill concrete placed to provide a level surface for the bottom of foundation elevations,and to evaluate the bearing loads to these capacities. A minimum factor of safety of 2.5 is suggested for the ultimate bearing capacity versus the allowable static bearing capacity; however, a different value may be justified based on site-specific geotechnical conditions. A minimum factor of safety of 2 is suggested for the ultimate bearing capacity versus the allowable dynamic bearing capacity; however, a different value may be justified based on site-specific geotechnical conditions.

To select the soil profiles to use for design and analysis of the US-APWR, a database of soil profiles and depths to basement was evaluated as described in MUAP-10006. (Reference 3.7-48). Six small strain profiles were selected for the development of strain compatible properties. These six profiles include soft and hard soil profiles (nominal shear wave velocity of 270 m/s and 560 m/s respectively) and soft and firm rock profiles (nominal shear wave velocity of 900 m/s and 2,032 m/s, respectively). The development of softer soil strain compatible profiles considers additional soil removal if necessary to maintain a minimum strain compatible shear wave velocity of at least 800 ft/s near the plant grade surface.

There are two 270 m/s profiles where the top 68 ft of soil is replaced. These two profiles representing layers of dense cohesionless soil and/or over-consolidated stiff clay are considered with depths of 200 ft and 500 ft above rock foundations consisting of sedimentary or weathered rock section overlying Precambrian basement material.

The third soil profile considered is a 500 ft thick layer of stiff 560 m/s soil representative of glacial till sites consisting of highly consolidated mixtures of fine and course grained soils over 1000 ft deep strata of sedimentary or weathered rock resting on the rock basement.

Two soft rock profiles (900 m/s) are considered with depths of 100 ft and 200 ft.

The sixth profile is a firm rock profile with a nominal shear wave velocity of 2,032 m/s and depth of 100 ft is selected to represent a residual soil (saprolite) over weathered rock and underlain by hard rock. This profile is intended to reflect hard rock foundation depths after removal of the soft surficial residual soils.

The six generic layered profiles reflect range of realistic site conditions and provide a wide range of SSI responses to ensure the broad applicability of the design for the CEUS. The final soil profile categories are summarized in Table 3.7.1-6.

The small strain shear wave velocity (V_s) and compression wave velocity (V_p) are plotted in Figure 3.7.1-14 and Figure 3.7.1-15. The shear strain damping is plotted in Figure 3.7.1-16. The nomenclature for the final soil profiles gives both the shear wave velocity and the depth to bedrock. For example, soil profile 560-500 designates the soil with shear wave velocity of 560 m/s with a depth of 500 ft.

The generic profiles are representative of saturated soil properties and a water table located at the plant grade elevation. Generic soil 270-200, 270-500 and 560-500 profiles representative of unsaturated soil properties were developed and analyzed in Technical Report MUAP-11007 (Reference 3.7-52). MUAP-11007 concluded that the use of saturated soil profiles as a site independent analysis parameter results in a standard plant design that envelops the seismic demands at a majority of candidate sites within the CEUS.

The generic backfill properties used in the SSI and SSSI analyses for the standard plant are discussed in Subsection 3.7.2.4.

The site-specific SSI analyses will use site-specific input soil/rock properties that are compatible to the site-specific ground motion compatible to site-specific FIRS discussed in Subsection 3.7.1.1. The primary non-linear material behavior of the soil must be considered and may be approximated by using equivalent linear material properties that are compatible to the free-field strains generated by the site-specific design ground motion. The strain-compatible soil properties are obtained from a 1-dimensional wave propagation analysis by using equivalent-linear methodology and site-specific soil stiffness and damping degradation curves.

The site-independent SSI analyses include the subgrade as horizontally infinite layers resting on the surface of an elasto-viscous half-space, representing the stiffness, material, and damping of geological hard rock stratum. The soil material damping values used in conjunction with the shear and compression wave profiles in the SSI analysis models are identical. The seismic models used in the SSI analyses are discussed further in Section 3.7.2.3. The site-independent SSI analyses are discussed further in Section 3.7.2.4, as well as the suggested methodologies for analyzing the effect of site-specific conditions on the SSI response.

Site response analyses using the equivalent linear Random Vibration Theory (RVT) approach described in MUAP-10006 (Reference 3.7-48) are performed to develop the CSDRS strain-compatible soil properties used as input for the SSI analyses. The site response analyses to develop the strain-compatible properties use the point-source model to generate both the input horizontal and vertical motions. A Magnitude **M**7.5 earthquake is used since its broad spectral shape is consistent with that of the CSDRS. A smaller magnitude would result in higher short period motions and higher strains. Distances to the **M**7.5 control earthquake are adjusted such that the median spectrum as full column outcrop spectrum at foundation level computed for each profile approaches, but does not exceed, the horizontal and vertical CSDRS. The distances and median estimates of the horizontal and vertical peak accelerations are listed in Table 3.7.1-7 and the median spectrum computed for each profile is compared to the CSDRS as described in MUAP-10006 (Reference 3.7-48).

3.7.2 Seismic System Analysis

Seismic system analysis is discussed in the following Subsections, 3.7.2.1 through 3.7.2.15. Following the guidance of the acceptance criteria in section II.3(a) of SRP 3.7.2 (Reference 3.7-16), two categories of seismic category I SSCs are defined: (1) seismic systems that include major seismic category I buildings and structures that are analyzed in conjunction with their basemats and supporting media (subgrade); and (2) seismic subsystems that include other seismic category I SSCs that are not analyzed in conjunction with basemats and subgrade. The details of the seismic system analysis is provided in Technical Report MUAP-10006 (Reference 3.7-48).

All standard plant seismic category I structures are part of the R/B complex. The R/B complex consists of the R/B, PCCV, CIS, East PS/B, West PS/B, ESWPC, and the seismic category II A/B all on a common basemat.

The T/B, consisting of the Turbine Building and the Electrical Room on their common basemat, has been classified as a seismic category II structure, and is in close proximity to the R/B complex. The T/B is analyzed in the same manner as the R/B complex as described in MUAP-11002 (Reference 3.7-61). The effects of structure-soil-structure interaction (SSSI) between the T/B and the R/B have been considered and shown to have a negligible effect on SSI response of the R/B complex and have been included in the development of the design ISRS. SSSI effects and how the effects were considered are discussed in more detail in Section 3.7.2.8.

The seismic responses of the major seismic category I and seismic category II structures are obtained from frequency domain time history analysis of seismic models considering a frequency-dependent SSI system. These site-independent analyses are performed with the set of generic layered soil profiles described in Section 3.7.1.3, which represent a wide range of site conditions. Subsections 3.7.2.1 and 3.7.2.3, respectively, describe the analysis and modeling methods used for the seismic analyses, and Subsection 3.7.2.2 discusses the natural frequencies and results obtained from the seismic analyses. To address effects of concrete cracking on the standard seismic design, seismic responses obtained from SSI analyses of models with two bounding levels of stiffness and damping are considered as discussed in Subsection 3.7.2.3.5. The dynamic analyses considered the torsional, rocking, and translational responses of the structures and their foundations. The effects are discussed in Sections 3.7.2.3.

The hydrodynamic effects considered are discussed in Section 3.7.2.3.

The standard design of R/B Complex superstructures is based on SSE load demands obtained from Response Spectra Analyses (RSA) of fixed base models of PCCV, CIS and R/B-A/B-PS/B's integrated shear wall structure. ISRS/ARS obtained from the results of site-independent SSI analyses serve as input for the RSA. Subsection 3.7.2.1 describes the methodology used for RSA. The methodology used for combination of modal responses is described in subsection 3.7.2.7. In Subsection 3.7.2.12, the responses obtained from the time history SSI analyses of R/B Complex dynamic model are compared with the responses obtained from RSA analyses of the major Category I structures to demonstrate the adequacy of the seismic design of the major structural members.

Subsection 3.7.3 discusses the seismic analyses applicable to seismic category I civil structure subsystems housed within or supported by the major seismic category I structures. Seismic and dynamic qualification of mechanical and electrical equipment and subsystems performed by testing is discussed in Section 3.10 and Appendix 3D. Mechanical subsystems include mechanical equipment, piping, vessels, tanks, heat exchangers, valves, and instrumentation tubing and tubing supports. The seismic analysis of mechanical subsystems is addressed in Sections 3.9 and 3.12. The mass inertia properties of the major civil structural, mechanical, and all other seismic subsystems are addressed in the seismic system analyses, as explained further in Subsection 3.7.2.3.

3.7.2.1 Seismic Analysis Methods

The methods used for the seismic analysis of the US-APWR seismic category I systems conform to the requirements of SRP Subsections 3.7.1 (Reference 3.7-10) and 3.7.2 (Reference 3.7-16). Table 3.7.2-1, as updated by the COL Applicant to include site-specific seismic category I structures, presents a summary of dynamic analysis and combination techniques including types of models and computer programs used, seismic analysis methods, and method of combination for the three directional components for the seismic analysis of the US-APWR standard plant seismic category I buildings and structures.

Seismic Response SSE Analysis Methodology

The seismic design of US-APWR standard plant is based on responses obtained from time history analyses performed using the SASSI computer program (Reference 3.7-17). The program uses the substructuring method to account for the interaction of the structure with the subgrade consisting of horizontally infinite layers overlaying a uniform half-space. For that purpose, the near field zone of the SSI system is partitioned in two substructures, the building superstructure and the basement minus the excavated soil. The dynamic properties are represented using the following complex frequency dependent stiffness matrix, $C(\omega)$, formulation:

$$C(\omega) = K - 2 \cdot i \cdot D \cdot K - \omega^2 M$$

where, ω is frequency of vibration $i = \sqrt{-1}$ is the complex number and K, M and D are stiffness, mass and linear hysteric damping matrices, respectively. The global complex stiffness matrix are assembled from the element complex stiffness matrices that are developed using FE technique.

The seismic response of the near field zone is obtained from the solution for the following complex matrix equation of motion in frequency domain:

$$\begin{bmatrix} C_{SS}(\omega) & C_{SI}(\omega) \\ C_{IS}(\omega) & \left(C_{II}(\omega) - C_{FF}(\omega) + X_{FF}(\omega)\right) \end{bmatrix} \cdot \begin{bmatrix} u_S(\omega) \\ u'_F(\omega) \end{bmatrix} = \begin{bmatrix} 0 \\ X_{FF}(\omega) \cdot u'_F(\omega) \end{bmatrix}$$

The subscripts S, I and F in the above matrix equations refer to the degrees of freedom associated with the building, basement and excavated soil. $u(\omega)$ are the vectors of complex nodal point displacements of the structure.

The vector $u'_F(\omega)$ of free field displacements at the interaction nodes is obtained from the solution of the site response problem. The impedance matrix $X_{FF}(\omega)$ representing the dynamic stiffness of the foundation at the interaction with the subgrade is calculated from the impedance analysis. Two methods are used for computation of the impedance matrix of embedded foundation SSI models:

a. The Direct Flexible Volume Method in which all the nodes of the excavated volume FE are specified as interaction nodes; and

b. The Modified Subtraction Method in which the solution is simplified by specifying the interaction nodes at the periphery of the excavated volume.

The seismic response analyses of R/B complex are performed on embedded models using the Modified Subtraction Method. The seismic response analysis for the Turbine Building (T/B) described in Subsection 3.7.2.8 is performed using the explicit Flexible Volume Method. The modified subtraction method is a simplified approach of representing the continuity between the free field zone and the near field zone. Based on comparisons with solutions obtained from the Flexible Volume Method, the results of the study demonstrate that the Modified Subtraction Method provides a realistic and reasonably accurate representation of the R/B Complex seismic response.

The structural analysis provides solution of the near field zone response by solving the equation of motion mentions above for selected frequencies of analysis. The solution is then interpolated for the range of frequencies of interest. The Fast Fourier Transformation (FFT) and inverse FFT technique are used to transform the input motion and the nodal responses of the system between frequency and time domains. Subsection 3.7.2.4 describes the development of within (inlayer) acceleration time histories used as input control motion in the SSI analyses of embedded foundations.

The number of FFT points is set to 8,192 (or 2^{13}) for the R/B complex model. This number is acceptable since the input excitation duration is about 22 seconds (4,417 time steps of 0.005 seconds) in addition to a quiet zone for free vibration of 20 seconds. This quiet zone ensures that the structure will be at rest after the entire 42 second duration.

The three components of the earthquake are applied to the SSI model separately and the solutions are superimposed to provide the solution for combined S- and P-wave excitations to all nodes. The vertically propagating S-waves represent the two horizontal components of the design earthquake motion H1 and H2 that are applied in NS and EW direction, respectively. The vertical component of the design earthquake (V) is represented by vertically propagating P-waves. Seismic input motions are considered in the SSI analyses. The SSI analyses use within motions at the bottom of the R/B complex as control motions. Refer to Section 3.7.2.4 for details.

Cut-off Frequency of the Analyses

The cut-off frequency is the highest frequency used in the dynamic analysis of the soil structure system. It sets an upper limit on the number of frequencies to be analyzed, and controls the maximum allowable element size. The maximum frequency of analysis is determined from the wave passage frequency (f_{pass}) of the soil layer and soil element size. The wave passage frequency is the maximum wave frequency that the soil layer can accurately transmit. It is determined using the Equation below (Reference 3.7-17):

$$f_{pass} = \frac{V_s}{5 \cdot d}$$

where V_s is the shear wave velocity of the soil and d is either the thickness of the soil layer or the maximum size of the FE mesh of the structural model at the SSI interface or the excavated soil volume mesh size.

Based on the calculated wave passage frequencies for each generic soil profile, the cut-off frequencies in the analyses are set to 40 Hz for 270-200 and 270-500 soil profiles, and 50 Hz for 560-500, 900-100, 900-200 and 2032-100 soil profiles.

Based on the maximum frequencies and intervals of frequency points, for SSI analysis, a total of 132 frequencies are analyzed for soil profiles 270-200 and 270-500, and a total of 152 frequencies are analyzed for soil profiles 560-500, 900-100, 900-200 and 2032-100.

3.7.2.2 Natural Frequencies and Responses

Table 3.7.2-4 presents a summary of the fixed base dynamic properties of R/B Complex Category I structures that are obtained from the ANSYS modal analysis. These are correlated with ACS SASSI analysis of R/B Complex Dynamic FE Model used for the site-independent SSI analyses presented in subsection 3.7.2. The natural frequencies, periods and effective masses of the dominant fixed base modes of vibration are provided for the R/B, the PCCV, the CIS and the east and west PS/Bs. Part 2 of Technical Report MUAP-10006 (Reference 3.7-48) provides comparisons and plots of the dominant mode shapes for each of the R/B Complex structures.

The seismic design of the US-APWR standard plant is based on responses obtained from a set of twelve (12) SSI analyses performed for the six generic site profiles of dynamic soil/rock properties presented in subsection 3.7.1.3 and the two levels of structural stiffness and damping properties (i.e. cracked and uncracked conditions) are described in subsection 3.7.2.3.

Amplitudes of the acceleration transfer functions are calculated from each SSI analysis for the response of the R/B Complex at the center of containment at the bottom of the foundation. The SSI responses for the generic site conditions are identified by comparing these transfer functions peak frequencies as described in Section 03.4.1.1 of Technical Report MUAP-10006 (Reference 3.7-48). These comparisons show that the site-independent SSI analyses provide a range of SSI responses of the R/B complex that envelope the possible responses of the building at a number of candidate sites.

The analyses of the generic soil profiles 270-200, 270-500, 560-500 provide seismic responses that are dominated by SSI effects and the dynamic characteristics of the subgrade. The responses obtained from the SSI analyses of these generic soil profiles define the standard design ISRS up to a frequency of approximately 5 Hz. The seismic responses obtained from the analyses performed for the rock profiles 900-200, 900-100 and 2032-100 are dominated by the dynamic properties of the structures. The responses for these generic rock sites define the design basis ISRS at higher frequencies.

Table 03.4.3-1 through Table 03.4.3-6 of Technical Report MUAP-10006 (Reference 3.7-48) provide weighted average floor accelerations for the R/B Complex structures that are calculated from the results of the site-independent SSI analyses. These weighted average accelerations are the envelope of the results obtained from the SSI analyses for the six generic site profiles of the model with full (uncracked concrete) stiffness properties and the model with reduced (cracked concrete) stiffness properties. Figure 03.4.3-1 through Figure 03.4.3-12 of Technical Report MUAP-10006 (Reference 3.7-48) show shear force diagrams calculated using the weighted average floor accelerations in the two horizontal directions. Table 03.4.3-7 of Technical Report MUAP-10006 (Reference 3.7-48) provides a comparison of the base reaction results calculated from the twelve different site-independent SSI analyses. The comparison shows that maximum base shears are from the SSI analysis of full stiffness model for hard rock site 2032-100. The maximum vertical base reaction is from the SSI analyses of reduced stiffness model for 900-200 generic rock profile.

The site independent SSI analyses of R/B Complex also provide results for the maximum displacements relative to the free field motion and the building basemat. These maximum relative displacements are calculated by following the methodology described in Section 03.3.8 of Technical Report MUAP-10006 (Reference 3.7-48). The maximum displacements due to three directions of the input motion are combined using the SRSS method. Tables 03.4.4-1 and 03.4.4-2 of Technical Report MUAP-10006 (Reference 3.7-48) present the envelope of the results from the site-independent SSI analyses for maximum displacements for different locations at the PCCV - R/B and PCCV - CIS interfaces. The adequacy of the 4 inch gaps between the PCCV - R/B and PCCV - CIS are evaluated based on the largest coupled seismic displacement conservatively calculated. The maximum relative seismic displacements of 3.2 and 1.6 inches are obtained for the gaps at the R/B-PCCV and PCCV-CIS interfaces, respectively. This results in clearances of 0.8 and 2.4 inches respectively. Therefore the gaps of 4 inches are adequate.

Subsection 3.7.2.5 discusses development of ISRS based on the results of the site-independent seismic analyses for the US-APWR standard plant.

3.7.2.3 Procedures Used for Analytical Modeling

3.7.2.3.1 General Discussion of Analytical Models

The procedures used for development of analytical models for seismic analysis are consistent with the procedures and guidelines of SRP 3.7.2, Section II.3 (Reference 3.7-16). Structural element mass and stiffness characteristics, as well as load and tributary masses, and damping characteristics, are incorporated into the models.

The Dynamic FE model of the R/B complex is developed and validated using ANSYS (Reference 3.7-21) and then translated into SASSI (Reference 3.7-17) format. The dynamic model is a simplified, coarsely meshed model that is validated against a more refined, detailed model. The translation process is described in the following steps:

Step 1: <u>Develop the R/B Complex Dynamic FE Model</u>

ANSYS Workbench and ANSYS Parametric Design Language (APDL) are used to develop an integrated 3-D FE model that includes the R/B, PCCV, CIS, A/B, East and West PS/Bs, and ESWPC coupled with the model representing the dynamic properties of the RCL. The numbering of the nodes is adjusted following the guidelines of the SASSI manual in order to optimize the computational effort.

Step 2: <u>Validate the R/B Complex Dynamic FE Model to ensure that the model adequately captures the dynamic behavior of the structures</u>

The Dynamic FE model is separated into six parts for the purpose of validation: R/B-FH/A, PCCV, CIS (with RCL), A/B, East PS/B, and West PS/B. The ESWPC is split and included in the R/B-FH/A, East PS/B, and West PS/B models. Static, modal, harmonic response, and stiffness analyses using ANSYS solvers are performed on each of the six parts of the dynamic model by establishing fixed boundary conditions at the base of each structure. An identical set of fixed base analyses are also performed on detailed FE models of each structure. The results obtained from the dynamic and detailed models are compared to demonstrate the ability of the less refined dynamic models to adequately capture the dynamic behavior of the corresponding detailed models. After all six parts are validated independently; the same process is used to validate the combined Dynamic FE model.

Step 3: Translate the Dynamic FE Model into SASSI format and verify the accuracy of the translation

The translator built into the SASSI code serves as the platform for the translation of the Dynamic FE model from ANSYS to SASSI house module format. In order to validate the translation of the model, a validation SSI analysis is performed on the SASSI Dynamic FE model resting on a very stiff elastic half space. The dynamic properties of the model, revealed by the resulting ATFs at selected locations, are compared to the fixed base dynamic properties and responses obtained from ANSYS modal analyses to ensure the translation is completed correctly.

The R/B complex Dynamic FE model consists of beam, shell, solid, and spring elements. The use of finite elements provides an accurate representation of the dynamic properties of the structures and the foundation that enables an accurate modeling of dynamic interaction with the flexible foundation and the surrounding soil. Shell elements are used to model the reinforced concrete shear walls and slabs. Three-dimensional (3-D) beam elements model the reinforced concrete or steel columns and beams. Solid elements are used to model the basemat foundation and the massive structural members of the CIS. Spring and beam elements are used to model the supports and connection of the RCL model with the CIS mesh. The finite element types used in the ANSYS model are compatible with the SASSI built in converter.

The Dynamic FE model is presented in Figure 3.7.2-1. This model has a total of 33,564 nodes, 47,580 elements and has an average mesh size of approximately 9 ft. The Dynamic FE model is based on the Detailed FE model presented in Figure 3.7.2-2. The Detailed FE model has a total of 62,252 nodes, 74,961 elements, with an average mesh size of approximately 5 ft. Figure 3.7.2-3 and Figure 3.7.2-4 present the detailed PCCV and CIS finite element models, respectively.

The development of the model ensures that the connection between two different element types is such that an adequate transfer of forces and/or moments from one structural component to the other is enabled. The nodes of the solid elements have only three translational degrees of freedom and can therefore not transfer the moments from shell or beam elements. In order to enable the transfer of bending moments from the walls modeled by shell elements to the basemat and massive concrete sections of the CIS modeled by solid elements, the shell elements are extended into or overlaid on the solid elements. A special layer of transitional rigid shell elements is created between the CIS reactor cavity top flange solid elements and the adjacent surrounding SC walls.

In addition, each node of the SASSI shell elements has five degrees of freedom that enable beam elements to transfer forces and bending moments to shell elements but not torsional moments. Therefore, massless beam elements are generated on the surface of the shell or solid elements to provide adequate transfer of moments from beams in all three rotational degrees of freedom. For beams or columns connecting to slabs or walls in the R/B model, the effect of adding torsional stiffness to the slab and wall shell elements is evaluated and the impact on the results is found to be negligible.

Refer to MUAP-10006 (Reference 3.7-48) for additional discussion on the development of the Dynamic FE model.

When the subsystem analysis is performed, reduced degrees of freedom (DOF) can be used to represent the dynamic behavior at locations needed for equipment qualification, provided that they can provide an adequate and conservative prediction of the response of the equipment.

The seismic analyses of the US-APWR standard plant are performed on three-dimensional seismic models representing seismic category I and seismic category II structures. The basic dimensions of these buildings and structures as considered in the seismic analyses are presented in the general arrangement drawings in Section 1.2. The 3-D FE models have an adequate number of discrete mass DOF to capture the global and local translational, rocking, and torsional responses of the structures. Torsional and rocking/swaying effects are also captured at the basemat/subgrade interface by taking into account SSI, including effects related to the flexibility of the basemat foundation.

It is the responsibility of the COL Applicant to develop analytical models appropriate for the seismic analysis of buildings and structures that are designed on a site-specific basis including, but not limited to, the following:

- PSFSVs (seismic category I)
- ESWPT (seismic category I)
- UHSRS (seismic category I)

3.7.2.3.2 R/B ComplexDynamic Finite Element Model

Technical Report MUAP-10006 (Reference 3.7-47) presents a detailed discussion of the approach taken for development and validation of the R/B complex FE model.

The R/B complex Dynamic FE model is an integrated 3-D model of the R/B, PCCV, CIS, East and West PS/B, A/B, and ESWPC structures sharing common shear walls and resting on top of a common 13'-4" to 43'-3" thick basemat. Figure 3.7.2-1 shows an overview of the R/B complex model, while Figure 3.7.2-5 and Figure 3.7.2-6 reveal the interior structures with section views. Figure 3.7.2-7 through Figure 3.7.2-13 show the PCCV, CIS shell elements (excluding the RCL), CIS solid elements, CIS beam elements (excluding RCL), East PS/B with ESWPC, West PS/B with ESWPC, and A/B as individual structures, respectively. The global origin is located at the center of the PCCV and top of the basement with the X axis pointing north, Y axis pointing west, and Z axis pointing upward. Once the model is translated into SASSI format, the global coordinate system is rotated 180 degrees about the Z axis so that the X axis is pointing south and the Y axis pointing east. Typical element size in the basemat and the slabs is approximately 9 ft. The element mesh used in the dynamic model is selected to provide sufficient modeling to capture the dynamic properties of the structure. The validation discussed in Subsection 3.7.2.3.10 show that no further refinement of the Dynamic FE model is necessary.

The R/B complex Dynamic FE model is developed incrementally using ANSYS in the following seven steps:

Step 1. R/B complex structures geometry is created in a manner that allows control of the model FE mesh.

Step 2. Attributes are assigned and additional masses are applied on each of the structures.

Step 3. Mesh controls are set and the model, excluding the RCL, is meshed.

Step 4. Modifications are implemented as needed to make the model more consistent with the Detailed FE model.

Step 5. The nodes are renumbered sequentially in the order of their X, Y, and Z coordinates as recommended by the SASSI Manual in order to enhance computational efficiency.

Step 6. The model used for ANSYS analyses of RCL is translated into a format that can be translated into SASSI.

Step 7. RCL structure is added to the FE model of R/B complex structures and proper connections are implemented to represent the physical supports attached to the CIS.

For simplicity without compromising accuracy, slab elevations in the Dynamic FE model are slightly shifted upward or downward such that the middle planes of nearby upper or lower slabs fall into a major common horizontal plane. Also, only large openings in slabs and walls are included in the model.

Special attention is given in the Dynamic FE model as to how wall and slab shell elements are connected to the basement/mat solid elements. Wall shell elements are extended into the basemat solid elements to ensure a proper transfer of bending moment between them. Likewise, slab shell elements joining the basement walls are extended one element to overlay the basement top surface. These connecting elements are also assigned a zero density not to increase the overall mass of the basemat.

Also for simplicity, only the main steel frame in the Fuel Handling crane support system, including a simplified rail truss girder, and the main steel framing in the CIS are modeled in the Dynamic FE model. The steel sections are modeled as beam elements to share nodes with the concrete shell elements representing the Fuel Handling exterior walls and slab. Thus, the composite behavior of the crane support system would not be fully represented in the FE model without further adjustment. The adjustment made is that all steel sections are assigned an increased moment of inertia in their strong axis to account for their composite behavior. An adjusted moment of inertia is also assigned to the embedded sections of the crane support steel columns between elevation 65'-0" and 76'-5", encased in 7'-8" by 4'-0" concrete columns.

The thickness of the PCCV is also simplified for ease of modeling. For the Dynamic FE model, only the large equipment hatch is modeled and the elements modeling the buttresses on the East and West sides of the structure are not offset with respect to adjacent elements. Also, the personnel airlocks as well as the Main Steam and Feed Water penetrations are not modeled in the Dynamic FE model. The stiffness and weight of the PCCV Dynamic model are not adjusted to account for the openings since they were found to have a negligible impact on the overall response of the structure. Figure 3.7.2-7 shows the PCCV Dynamic model.

To account for the effects of dynamic coupling with the building structures, the PCCV polar crane and the R/B fuel handling crane are incorporated into the standard plant design by using typical mass and stiffness properties anticipated for the cranes in the R/B complex dynamic and detailed structural FE models. The cranes are modeled in their parked positions as occupied during normal plant operations. The parked position for the polar crane is parallel to the centerline of the PCCV running between azimuth 0° and azimuth 180° with the hoist trolley located over the roof slab above the pressurizer. The fuel handling crane is stored above the truck access area on the east side of the fuel handling area when not in service. The building models include the crane's lifted mass, mass of the trolley, crane bridge girders, and end trucks. The building models include the stiffness of the crane bridge girders, end trucks, and the local stiffness of the supporting structural steel at the end truck locations. This is a generic crane design intended solely to be used for seismic analyses. Therefore, the polar crane is modeled to approximate the design weight.

The requirements of NOG-1 (Reference 3.7-22) require that the crane design analyses be performed by coupling the crane models with the building models. The PCCV polar crane and R/B fuel handling crane are procured on a site-specific basis. It is the responsibility of the COL Applicant to confirm the masses and frequencies of the PCCV polar crane and fuel handling crane and to determine if coupled site-specific analyses are required. If found that this is required, the site-specific seismic analysis of the US-APWR standard plant must be performed on models that incorporate the PCCV polar crane and the fuel handling crane, as appropriate in the site-specific SSI analyses and site-specific crane analyses.

The CIS portion of the Dynamic FE model, excluding the Reactor Coolant Loop (RCL), contains approximately 4,631 elements and 3,876 nodes with nominal mesh size of 7.2 ft in the vertical direction and 9 ft in the horizontal direction. It consists of a combination of shell, solid and beam elements. The solid elements shown in Figure 3.7.2-1 make up the CIS base which starts at elevation 2'-7" and the reactor support which extends up to elevation 35'-10.87". The shell elements make up the remaining walls and slabs of the structure and begin at the same elevation as the CIS solid elements, but extend to the top of the pressurizer compartment at elevation 139'-6". The beam elements shown in Figure 3.7.2-10 represent the steel frames and the supports for the RCL components.

The lumped mass stick model used for dynamic analyses of the RCL includes several parts representing the dynamic properties of the Nuclear Steam Supply System (NSSS) components and the main coolant piping. Appendix 3C discusses the RCL model.

The model of the RCL and major pipe components used for seismic analyses of the NSSS are translated into elements acceptable to SASSI format and then coupled with the dynamic CIS model. The translation included changes of ANSYS modeling features such as pipe element types, rigid links and constraint equations that can be supported by the SASSI translator. The pipe elements are replaced by 3-D beam elements with stiffness values equivalent to those of the straight and curved pipe sections. The rigid links and constraint equations are replaced by rigid beams. The coupling of the RCL to the CIS is accomplished such that there are no local effects from the CIS imparted upon the RCL. The validation of the model in Subsection 3.7.2.3.10 demonstrates that these modifications do not affect the overall stiffness of the model and thus the dynamic response of the RCL components.

3.7.2.3.3 Not Used

3.7.2.3.4 Subsystem Coupling Requirements

For purposes of modeling the R/B-PCCV-containment internal structure on their common basemat, large seismic subsystems contained within these structures are evaluated against the mass and frequency ratio criteria given in SRP 3.7.2, Section II.3(b) (Reference 3.7-16), as follows:

- If $R_m < 0.01$, decoupling can be done for any R_f
- If $0.01 \leq R_m$ and ≤ 0.1, decoupling can be done if $0.8 \geq R_f \geq 1.25$
- If $R_m > 0.1$, a subsystem model should be included in the primary system model

where

R_m = (total mass of supported subsystem)/(total mass of supporting system)

R_f = (fundamental frequency of supported system)/(dominant frequency of support motion)

If these criteria require the subsystem to be coupled with the primary seismic model, both the stiffness and the mass of the subsystem are included in the overall model to assure the accuracy of the calculated frequencies. This is the approach used for integrating the RCL seismic subsystem with the R/B complex dynamic FE model discussed in Technical Report MUAP-10006 (Reference 3.7-48). To account for the effects of dynamic coupling of the containment internal structure with the equipment and the piping, the dynamic FE model of the R/B complex also includes a lumped mass stick model of the RCL representing the stiffness and mass inertia properties of the major equipment and piping located in the PCCV. Spring elements are used to model the stiffness of the supports of the components and piping. The lumped mass stick model of the RCL and major piping components used for seismic analyses of nuclear steam supply system are translated into an acceptable ACS SASSI format and then coupled with the dynamic containment internal structure model.

When it has been determined through investigation of the above criteria that a subsystem is not required to be coupled with the primary seismic model, then the subsystem is assumed absolutely rigid and only its mass is included at appropriate node points of the global seismic model.

3.7.2.3.5 Section and Material Properties

The values of the modulus of elasticity and Poisson's ratio (ν) for concrete and steel used in the dynamic models are discussed below. The values are for materials at or near ambient temperatures.

a. Concrete

The concrete modulus of elasticity E_c, and shear modulus G_c corresponding to the compressive strengths of normal weight concrete used in the R/B, PCCV, and containment internal structure and their common basemat are summarized in Table 3.7.2-2 and are computed as follows:

E_c (psi)= 57,000 $\sqrt{f'_c}$

G_c (psi)= $E_c / 2\,(1 + \nu_c)$

where

f'_c = specified 28-day compressive strength of concrete (psi)

ν_c = 0.17 (Poisson's ratio for concrete)

b. Steel

The properties of ferritic structural steel and non-prestressed reinforcement: Young's modulus of elasticity E_s and Poisson's ratio for steel ν_s are as follows:

E_s = 29,000 ksi and ν_s = 0.3

Effects of Concrete Cracking on Reinforced Concrete Structures

Reinforced concrete structures include the R/B, East and West PS/Bs, A/B and ESWPC. In accordance with ASCE 4-98 (Reference 3.7-9), Section 3.1.3, traditional reinforced concrete members and elements are to be modeled as either cracked or uncracked sections. For the uncracked sections/elements, the stiffness is directly obtained from the concrete linear elastic properties and the section or element geometric dimensions. For the cracked concrete, a reduction to the uncracked concrete stiffness included. A 50% reduced value of the concrete modulus of elasticity is used in linear elastic analysis to address the effects of concrete cracking on the seismic response.

The design of the reinforced concrete structures is based on the ultimate capacity of the reinforced concrete sections. Therefore, the design of reinforced concrete members addresses code stress limits corresponding to reduced cracked concrete stiffness properties and higher SSE material damping levels as discussed in Section 1.2 of RG 1.61 (Reference 3.7-15). However, there is a possibility that the response of the structure under lower stress levels at certain frequency ranges will be higher than the response corresponding to the higher stress state under cracked conditions. In order to ensure that the structural integrity and functionality of the components and the equipment is not compromised under seismic loading conditions, the development of ISRS and seismic loads and displacement also considers the responses of the reinforced concrete structure with full (uncracked concrete) stiffness properties and lower OBE damping levels.

The seismic response analyses of reinforced concrete structures consider two stiffness and damping values in order to address the possible variations in the extent of concrete cracking:

1. Full stiffness representing low stress levels corresponding to uncracked concrete properties where the stiffness of the members are represented by gross cross sectional properties.

2. Reduced stiffness representing higher stress levels resulting in cracking of the concrete where the stiffness of the members are reduced in accordance with guidelines provided in Table 3-1 of ASCE/SEI 43-05. The stiffness of the composite members made of reinforced concrete and steel beams, such as the walls and the roof of Fuel Handling Area (FH/A), are also reduced accordingly to represent 50% reduction in stiffness in the reinforced concrete part of the composite sections.

The structural material damping values used for these two different stress levels are OBE damping of 4% for the full (uncracked concrete) stiffness condition and SSE damping of 7% for the reduced (cracked concrete) stiffness condition, are obtained from RG 1.61 (Reference 3.7-15) and are shownin Table 3.7.2-3.

Effects of Concrete Cracking on the CIS

The CIS is comprised of different types of structural members including composite SC walls, massive reinforced concrete sections, and reinforced concrete slabs. The members can experience varying levels of stress resulting in different patterns of concrete cracking under the different loading conditions that can occur. Depending on the plant conditions, the CIS members can be subjected to design seismic loads in combination with normal operating or design basis accidental thermal loads resulting in different levels of stiffness reduction due to concrete cracking. Table 3.8.3-4 shows the summary of CIS stiffness and damping considered during seismic analysis. The CIS members are classified in six categories, two stiffness levels corresponding to:

1. Loading Condition A: (SSE Seismic, plus operating temperatures): conditions characterized with insignificant reduction of stiffness and concrete cracking; and;

2. Loading Condition B: (SSE Seismic, plus accident temperatures): conditions characterized with significant reduction of stiffness due to cracking of the concrete under high design basis accidental thermal loads and SSE seismic.

Different material damping values are assigned to the different members depending on the level of stresses and corresponding concrete cracking.

Effects of Concrete Cracking on the PCCV

Similar to the CIS, the level of stress in the PCCV during a seismic design event depends on the plant conditions. The design of the PCCV structure is based on the premise that during normal operating conditions the pre-stressed concrete cross sections remain in compression. During the normal operating conditions, the earthquake design loads can cause only limited cracking having insignificant effect on the overall stiffness of the PCCV. Accordingly, the dissipation of energy due to material damping of the PCCV structure under normal operating conditions is low. The accident loading conditions include high temperatures and pressure loads in the reactor containment that can generate high stresses in the pre-stressed concrete accompanied with cracking that can result in a reduction of the global stiffness of the PCCV structure and higher dissipation of energy due to the material damping. The stress evaluations provided in Appendix 2-A of MUAP-10006 (Reference 3.7-48), indicate that the reduction of the overall stiffness of PCCV structure under seismic design loads in combination with accident loads can be up to 50%.

Two stiffness levels are considered for the seismic response analyses of PCCV:

1. Normal operating conditions corresponding to insignificant concrete cracking and full (uncracked concrete) stiffness of the pre-stressed concrete structure, using 3% damping, and;

2. Accident conditions when the high thermal and pressure loads generate high stresses that can result in significant cracking of the pre-stressed concrete and a 50% reduction of the stiffness, using 5% damping.

The structural material damping values used for these two different stiffness and stress levels are also provided in Table 3.7.2-3.

3.7.2.3.6 Modeling of Mass

The mass included in the R/B complex Dynamic FE model includes contributions from the structural mass in addition to that of equipment, dead loads, and live loads.

Generally, the structural mass is assigned as a density to the finite elements based on the material properties of the components of the structures. The density is then increased to account for equipment, live, snow and other applicable loads. A mass equivalent to 25% of floor design live load and 75% of roof design snow load, as applicable, is included in the model in accordance with SRP 3.7.2 Acceptance Criteria II.1.D (Reference 3.7-17). Each load is applied over a particular area and the density of the elements in that area is increased such that the total increase in mass matches the mass of the applied loads.

Equipment load also includes a 50 psf dead load to account for miscellaneous pipe, minor equipment, and raceway loads applied on slabs in the R/B complex model, with the exception of a few locations where a heavier pipe load is used instead (e.g., main steam and feedwater pipe).

The above process is not applicable for the NSSS and major pipe that constitutes the RCL. The RCL dynamic mass is included directly in the RCL model.

The mass is applied to the Dynamic FE model in two steps. First, a mass density equal to the sum of the structural self-weight and pipe load is calculated and assigned to each of the shell elements modeling the R/B complex slabs. Where mass is carried by grating not explicitly modeled, the total mass supported is evenly distributed on the supporting walls and slabs.The remaining loads are applied as either additional mass densities on slab shell elements or concentrated lumped masses on wall and slab key points.

The density and thickness of the elements are further modified to account for stiffness reductions due to minor openings and cracking, but it is done in such a way as to not change the mass of the elements. Refer to Subsection 3.7.2.3.2 for further discussion.

The PCCV Polar Crane and Fuel Handling cranes are modeled in their respective parked locations with trolley masses and lifted load masses included.

The mass used for the New Fuel Storage Pit (NFSP) and Spent Fuel Pit (SFP) includes the mass of the fuel and the fuel storage racks contained within the pits. This is accomplished by adding the masses as lumped masses to the concrete slabs of the pits (pools). The dynamic characteristics of the racks are not modeled or coupled with the structure. Liquid masses contained in the SFP, Emergency Feed Water Pits (EFWP), and Refueling Water Storage Pit (RWSP) are modeled as directional masses using mass elements rigidly attached to walls and slabs. The sloshing effects are not considered in the model since the effects are negligible.

3.7.2.3.7 Adjustment of Stiffness and Mass Properties

The coarse mesh of the dynamic FE model has limited resolution for modeling of openings in the walls. The elastic modulus and thickness of shell elements are adjusted to accurately model the reduction of shear stiffness of the wall due to openings. The density of shell elements is also adjusted to accurately represent the mass of the wall accounting for openings and the adjusted wall thickness.

Finite Element analyses are performed using ANSYS to obtain the stiffness reduction factors needed to adjust the material properties and account for the reduced stiffness of the shear wall openings. The correction factors are obtained by comparing the results from the static analyses of two detailed solid FE models. Model A represents the actual geometry of the wall with openings, and Model B represents the wall without openings. Unit displacements are applied at the top of each model in both the in-plane and the out-of-plane directions, to generate the reactions at the bottom, which can then be used to calculate the in-plane and out-of-plane wall stiffness. The ratio between the reaction obtained from Model A and Model B is used to determine out-of-plane stiffness reduction factors (m) and the in-plane stiffness reduction factor (n) that are then used to determine the adjusted elastic modulus (E_o), thickness (t_o), and density (y_o) of the wall. Further details on the development and implementation of stiffness reduction of elements in the FE model are described in Technical Report MUAP-10006 (Reference 3.7-48).

3.7.2.3.8 Stiffness of Steel Reinforced Concrete Beams and Columns

In the fuel handling area (FH/A), the crane supporting steel columns and girts are continuously anchored to the exterior concrete walls with headed steel studs. The steel roof beams are also continuously anchored to the concrete roof slabs.

Based on AISC 360-05 Commentary (Reference 3.7-57), 75% of the composite transformed moment of inertia is used in calculating the effective moment of inertia of the composite section (I_{eff}):

$$I_{eff} = 0.75 \cdot I_{tr}$$ for a fully composite member

$$I_{eff} = I_x + \sqrt{Q_n/C_f} \cdot (I_{tr} - I_x)$$ for a partially composite member, with $Q_n/C_f \geq 0.25$

Where: Q_n = shear capacity of the studs between the points of inflection (zero: moment)

C_f = smaller of steel yield force or concrete ultimate compressive force

I_x = moment of inertia of the steel column or beam

I_{tr}= composite transformed moment of inertia, calculated as follows

$$I_{tr} = I_x + \left(t_c + t_d + \frac{d}{2} - y_{bar}\right)^2 \cdot A_s + \frac{b_{eff} \cdot t_c^3}{12} + \left(y_{bar} - \frac{t_c}{2}\right)^2 \cdot b_{eff} \cdot t_c$$

Where: t_c = slab or wall thickness

t_d = steel deck thickness, if any

d = depth of the steel member

y_{bar} = centroidal distance of the transformed section, measured from the top of concrete:

$$y_{bar} = \frac{0.5 \cdot b_{eff} \cdot t_c^2 + A_s(t_c + t_d + 0.5 \cdot d)}{b_{eff} \cdot t_c + A_s}$$

A_s = area of the steel member

b_{eff}= effective width of concrete, after transforming to steel = b_e/n

b_e = effective width of concrete, before transforming to steel

n = modular ratio = E_s/E_c

E_s = Young's Modulus for steel

E_c = Young's Modulus for concrete

In order to incorporate the composite stiffness of the steel beams and the reinforced concrete slabs the moments of inertia of the beams are increased. This modeling approach provides an accurate representation of the actual out-of-plane bending stiffness of the composite concrete-steel cross-sections which is validated through comparison of responses obtained from the detailed and dynamic models.

In the above, the effective width of concrete (before transformation to steel section) is based on AISC 360-05, Section I3 as shown below.

In the Dynamic FE model, beam and shell elements are used to represent the individual members. The locations of the centerlines of the beam and shell elements are coincident so the effective bending stiffness of the section (EI) in the Dynamic FE model is the sum of the individual moments of inertia:

$$EI = \frac{t_c^3 \cdot b_e}{12} \cdot E_c + I_x + E_s$$

Therefore, the moment of inertia of the beam element that results in bending stiffness of the section (EI) that is equivalent to the stiffness of the actual composite section is calculated as follows:

$$I_s = I_x \cdot \alpha = I_{eff} - \frac{t_c^3 \cdot b_e}{12} \cdot \frac{E_c}{E_s}$$

Where $\alpha = I_s / I_x$ is a factor used to adjust the bending moment of inertia of the beam element in the FE model in order to simulate the actual composite stiffness of the reinforced concrete-steel beam cross sections. The effect of concrete cracking on composite members is a 50% reduction in stiffness as described in Section 3.7.2.3 and shown in Table 3.7.2-3.

3.7.2.3.9 Dynamic Properties of R/B Slabs and SC Modules

3.7.2.3.9.1 Dynamic Properties of R/B Slabs

The development of the Dynamic FE model requires simplifications of the model geometry in order to produce a regular FE mesh and to minimize the size of the model to be suitable for SSI analyses using SASSI. These simplifications in modeling the building geometry affect the spans of some of the floor slabs in the R/B model and their local out-of-plane response. The stiffness and mass properties of these flexible slabs are adjusted to model the actual mass and stiffness properties of the slab.

The dynamic stiffness properties of the slabs at each of the major floor elevations are obtained by isolating each elevation. Figure 3.7.2-15 shows an FE model of a R/B floor slab that is extracted from the Detailed FE model. Boundary conditions are established as shown in Figure 3.7.2-16 at the upper and lower border of the model to restrain horizontal displacements of the walls and accurately model the bending stiffness at the wall/slab interfaces. Figure 3.7.2-16 is meant to show representative boundary conditions, not the exact support conditions of all the individual slabs, which may be supported on three or four sides with walls. The horizontal and vertical displacements of the slab at the junctions of the slab with the supporting walls are also restrained in order to eliminate the effects of the axial stiffness of the walls on the modal analyses results and to ignore the slab horizontal modes as well. Where the slab is supported by columns, the vertical displacement is constrained.

Modal analysis using ANSYS is performed on the isolated elevations of Detailed FE model and Dynamic FE model to obtain the dynamic properties. If the frequency of the first dominant mode of the slab obtained from Detailed FE model with full (uncracked concrete) stiffness properties is greater than 70 Hz, the slab is considered rigid. There is no need to adjust the stiffness of the shell elements modeling rigid slabs.See Subsection 3.7.2.3.10 for additional discussion about the 70 Hz cutoff frequency.

For the flexible slabs with frequency below 70 Hz, the stiffness is adjusted as needed by tuning the modulus of elasticity of the slab shell elements in the Dynamic FE model to match the frequency obtained from the Detailed FE model. The difference in first dominant frequency of vibration of the slabs obtained from the modal analyses of the two FE models is minimized through an iterative process. This process is iterated until the difference in dominant frequencies for slabs at a given elevation are at a minimum. The largest difference in slab frequency after the completion of the above process is 6%. See MUAP-10006 (Reference 3.7-48) for additional discussion about the development and validation of the dynamic model.

3.7.2.3.9.2 Dynamic Properties of SC Modules

Simplifications in the geometry of the dynamic containment internal structure model are made to produce a coarser FE mesh in order to be suitable for SSI analyses using ACS SASSI. Stiffness and mass properties of elements modeling some of the SC walls of the containment internal structure are adjusted in order to calibrate the dynamic response of the simplified dynamic FE model to match the actual response of the containment internal structure as represented in the detailed FE model. The adjustments of the unit density and the elastic moduli of the shell elements are introduced to capture the actual distribution of mass and stiffness. The calibration of the model properties is performed based on the results of a 1g static analysis, and then verified using the results of modal and time history analyses.

3.7.2.3.10 Validation of the Seismic Models

The development of the R/B complex Dynamic FE Model is based on a number of adjustments in geometry and load configurations in order to minimize the size of the model and make it suitable for SSI analysis using SASSI. The validation ensures that these modeling adjustments do not affect the ability of the Dynamic FE Model to accurately represent the dynamic response of the R/B complex structures as described by SRP Sections 3.7.2.II.1 and 3.7.2.II.3 (Reference 3.7-16) and by ISG-01, Section 3.1 (Reference 3.7-54).

The R/B complex Dynamic FE Model is divided into six parts: R/B-FH/A-ESWPC, CIS coupled with RCL, PCCV, East PS/B, West PS/B, and A/B. The integrated model is divided into the individual components such that each structure is independent of the others. Common walls in the integrated model are included in each individual model for the purpose of validation. A series of fixed base analyses are performed on the six separate models using ANSYS and the results are compared to the ones obtained from corresponding analyses on the Detailed FE models of the R/B-FH/A-ESWPC, CIS, PCCV, East PS/B, West PS/B, and A/B structures. Once validation of the individual models is complete, confirmatory validation analyses on the integrated R/B complex model are performed.

The validation of the Dynamic FE Model of the R/B complex, with the exception of the CIS, that are carried out on models with full (uncracked concrete) stiffness are also valid for the models with reduced (cracked concrete) stiffness. The result of the global stiffness reduction is manifested by a shift of the response of the structure to lower frequencies. Hence, a 50% stiffness reduction corresponds to a shift of frequencies by $\sqrt{2} = 1.4$ times. Therefore, the dynamic validation analyses consider responses for frequencies up to 1.4X50 = 70 Hz and higher in order to ensure that the model with reduced (cracked concrete) stiffness properties can also meet the requirement of ISG-1, Section 3.1 (Reference 3.7-54) to accurately capture responses with frequencies up to 50 Hz.

Due to the complexity of the CIS, different stiffness and damping values are assigned to different types of structural components for the two bounding stiffness and damping conditions. As shown in Table 3.8.3-4, the reduction of stiffness applied to the CIS to account for cracking of the concrete of SC modules, reinforced concrete slabs and massive concrete portions is not uniform. Therefore, unlike the other structures, two sets of validation analyses are performed for the CIS to ensure the adequacy of the CIS Dynamic FE Model with full (uncracked concrete) stiffness and reduced (cracked concrete) stiffness.

The FE analysis computer program ANSYS (Reference 3.7-21) serves as the platform for three different types of analyses performed to validate the dynamic properties of the R/B complex Dynamic FE Model.

Sets of static analyses are performed on both the Dynamic FE Models and Detailed FE Models by applying 1-g quasi-static acceleration on the models with fixed boundary conditions established at the bottom of the model to calculate nodal displacements and reaction forces. The reaction force results are compared to ensure that the mass assigned to the Dynamic FE Model and Detailed FE Model are similar. For the R/B, the masses assigned to each major floor elevation are also compared in order to check the correlation of the mass distribution in the two models. The global distribution of mass and stiffness of the structure is checked further by comparison of the deflection results from the 1-g static analyses of the two FE models.

The comparison of the results for deflection along the corners of the structures under 1-g quasi-static acceleration in the two horizontal and the vertical directions, respectively, are used to determine if the number of discrete mass degrees of freedom are sufficient to capture accurately the dynamic response of the structures. The check is performed to ensure that the deflection shapes calculated from the analyses of the Dynamic FE Model correlate well with those obtained from the analyses of the Detailed FE Models. The

Dynamic FE Model is considered to adequately represent the stiffness and mass distribution if the differences in the displacements results obtained from 1-g static analyses of the Dynamic FE Model and Detailed FE Model are small.

Modal analyses are performed using ANSYS on the Dynamic FE Models and the Detailed FE Models of R/B, CIS, PCCV, East and West PS/Bs, and A/B with fixed conditions at the bottom of the models. The analyses provide the fixed base dynamic properties of the models, such as the natural frequencies, mode shapes, modal mass participation and the total effective mass (mobilized mass) of all of the extracted natural modes of vibration of the structures.

In order to depict the global dynamic response of the structures and determine the dominant frequencies of vibration, the results of the modal analyses of the Dynamic FE Model and Detailed FE Model, the cumulative mass versus frequency are plotted together and compared. The Dynamic FE Model is considered to have sufficient accuracy if the cumulative mass versus frequency plots are consistent with those obtained from the Detailed FE Model. Additionally, the individual models are analyzed for frequencies and mode shapes up to 100 Hz and the modal data for each direction are extracted. A comparative analysis of the modes between individual buildings of the Detailed FE Model dynamics and the Dynamic FE Model dynamics is performed. Parameters and discussion are provided that demonstrate that the models are dynamically equivalent.

After the models are developed the mass statistics are extracted and presented. The mode shapes were normalized to mass. Thus there was no direction given to force the maximum displacement of a shape to be positive. Consequently, some plots will show that a mode shape of the Detailed FE Model will appear as a mirror image, i.e., reversal of sign, of the Dynamic FE Model. There is no impact on the results since the signs of the participation factors will also be reversed.

In addition to the 1g static and modal analyses performed above which only provide a global comparison between the Dynamic FE Model and the Detailed FE Model, a series of full harmonic analyses are performed in ANSYS on the Dynamic FE Models and the Detailed FE Models of R/B, CIS, PCCV, East and West PS/Bs, and A/B with fixed condition at the bottom of the models. The harmonic analysis calculates the response of the structure to cyclic loads over a frequency range. To model the fundamental concept of Acceleration Transfer Function (ATF) in SASSI which directly relates the input motion to the structural response, a 1-g ground (global) acceleration is applied in each of the three orthogonal directions, respectively, from which the ATFs at selected locations are derived based on the displacement response at the specified range of frequencies. A constant damping ratio of 5% is applied in all the harmonic analyses.

3.7.2.3.10.1 Validation Method

The validation results presented in Section 02.5 of MUAP-10006 (Reference 3.7-48) conclude that the dynamic model and detailed model represent approximately the same mass and stiffness distribution, the same dynamic properties in terms of the fundamental frequencies and associated mode shapes and modal masses, and comparable ATFs at various locations. Therefore the R/B complex Dynamic FE Model adequately represents the building mass stiffness and dynamic properties for soil-structure interaction analysis.

3.7.2.4 Soil-Structure Interaction

The seismic design of US-APWR standard plant is based on responses obtained from the site-independent SSI analysis of the R/B complex structures. Subsection 3.7.2.2 presents the results and describes the SSI responses captured by the site-independent SSI analyses.

The SSI analysis consider models with two levels of structural stiffness (cracked anduncracked).See Section 3.7.2.3.5 for discussion of the cracked and uncracked modeling approach.

The SSI analysis is performed for the six generic layered soil profiles developed in Section 3.7.1.3: 270-200, 270-500, 560-500, 900-100, 900-200 and 2032-100.

A total of twelve cases combining two stiffness levels and six soil profiles are performed in the SSI analysis that as described in subsection 3.7.2.2, envelope the seismic responses of the US-APWR standard plant structures at wide range of candidate sites.

3.7.2.4.1 Dynamic Soil Properties

Section 3.7.1.3 describes the development of six generic soil profiles.The site models in the SASSI analyses use infinite horizontal layers (referred to as fixed layers whose depths vary with the soil profiles) to represent the approximately 1000 feet of the top soils. An additional 10 layers, referred to as variable layer, represents a half space of visco-elastic medium. For the same soil profile, the total thickness of variable depth layer varies with the frequency analyzed and is determined as $1.5V_S/f$, where V_S is shear wave velocity of the half space and f, in Hz, is the frequency of analysis.

The site-independent SSI analyses are performed on embedded models with near field soil solid elements connecting the FE model of the building basement with the free field zone. These near field solid elements represent the dynamic properties of the soil backfilled around the building basement after the construction of the plant. Table 03.3.1-10 through Table 03.3.1-15 of MUAP-10006 (Reference 3.7-48) present the properties assigned to the near field soil elements representative of strain compatible properties of typical granular backfill materials. In order to cover a wide range of soil-structure frequencies, a backfill with relatively soft properties is used in conjunction with the generic soil sites, 270-200, 270-500 and 560-500. A backfill with relatively stiff properties is used in conjunction with the generic rock sites, 900-100, 900-200, and 2032-100. Additional detail regarding the soil profiles is provided in MUAP-10006 (Reference 3.7-48).

The first six inches of natural soil located beneath the R/B and the T/B is required to satisfy the requirements for subgrade materials to achieve a kinetic friction coefficient of 0.5 or higher. If the kinetic friction coefficient is less than 0.5 then the natural soil shall be replaced by compacted granular backfill to provide a kinetic friction coefficient of 0.5 or higher.

3.7.2.4.2 Input Control Ground Motions

Section 3.7.1.1 provides a set of three acceleration time histories (H1, H2, and V). In the SSI and SSSI analyses, the H1, H2, and V acceleration time histories are used to derive the input control motions in Standard Plant North-South (NS), East-West (EW) and Vertical direction, respectively.

CSDRS and the CSDRS compatible time histories define the standard design ground motion as a free field outcrop motion at the bottom elevation of the R/B complex foundation basemat. A set of linear elastic site response analyses are performed on the generic strain compatible shear wave profiles presented in Figure 3.7.1-14 and compression wave profiles presented in Figure 3.7.1-15 to convert the outcrop motion time histories to within (inlayer) acceleration time histories used in the SSI analyses. Section 03.3.2 of MUAP-10006 (Reference 3.7-48) provides details about these site response analyses and the resulting within motions.

A total of six sets of within motions are generated for the six soil profiles, one set for each profile. Each set of within motion includes two horizontal and one vertical motion.

3.7.2.4.3 Structure-Soil-Structure Interaction (SSSI Model)

See MUAP-10006 (Reference 3.7-48) for discussion about the SSSI analysis. The results also indicate that the presence of the R/B complex produces a noticeable effect on the seismic response of the T/B. SSSI effects tend to increase the seismic design forces in the NS direction, result in seismic design forces that are generally the same or slightly higher in the EW direction, and increase the seismic design forces in the vertical direction. See MUAP-11002 (Reference 3.7-61) for additional discussion of the SSSI analysis. Seismic Load demands for the structural design of the Seismic Category II T/B envelope both the SSI and the SSSI analyses results.

Based on the findings, it can be concluded that the R/B complex will not be affected by SSSI effects from the Access Building or Tank House. Unlike the T/B, with size and weight comparable to the R/B complex, these two buildings are too small and light to have any significant effect on the response of the much heavier and larger R/B complex.

The SSSI effects on ISRS are conservatively considered in the standard design by enveloping all twenty cases, i.e., the twelve cases for SSI and eight cases for SSSI.

3.7.2.4.4 Summary of the Site Independent SSI Analysis of US-APWR Standard Plant

The seismic analyses of the R/B complex structures considers the following effects:

- Concrete cracking and associated stiffness variation through the consideration of two bounding stiffness levels for the structures;
- Flexibility of the foundation and basement by using FE models;
- Layering of the subgrade by using layered generic soil profiles;
- Embedment by directly analyzing the structures as embedded structures;
- SSSI effects by performing the seismic coupling analysis of the structure soil structure system as described in Subsection 3.7.2.8.

The analyses are performed using SASSI. Therefore, the frequency dependent impedance of foundation soils is considered as well. The CSDRS is identified as an outcrop motion in the free field at the R/B complex foundation level. The corresponding within (inlayer) motions at the foundation level are used as input control motion for the analyses. A set of three statistically independent artificial time histories representing the input ground motion for the three orthogonal directions is used to derive the within motions.

The results of the SSI analyses are used for development of the following seismic basis parameters for the structural design of the R/B complex:

- The building analysis and design are conducted in accordance with the procedures and guidelines provided in Subsection 3.8.4.4.
- ISRS that are the input for design and seismic evaluation of SSCs and equipment in the R/B complex. The ISRS are developed by enveloping and broadening the results of the SSI and SSSI analyses.
- Maximum relative displacements that are used as input for evaluation of the adequacy of the gaps between the CIS and PCCV, PCCV and R/B.
- The R/B complex SSI analyses also provide artificial time histories of the seismict response of the R/B complex structure at each nodal point. These are used as input for the evaluation of overturning and bearing pressure of the R/B complex.

These results are presented in Part 03 of MUAP-10006 (Reference 3.7-48).

The SSI analyses are performed for the generic soil profiles that consider full saturated soil conditions. MUAP-11007 (Reference 3.7-52) presents the evaluation of the significance of water table effects for seismic standard plant design basis.

3.7.2.4.5 Requirements for Site-Specific SSI Analysis of US-APWR Standard Plant and Site-Specific Structures

The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B complex utilizing a computer program such as ACS SASSI (Reference 3.7-17) which contains time history input incoherence function capability. The SSI analysis using SASSI is required in order to confirm that site-specific effects are enveloped by the standard design.

SSI effects are also considered by the COL Applicant in site-specific seismic design of any seismic category I and II structures that are not included in the US-APWR standard plant. The site-specific SSI analysis is performed for buildings and structures including, but not limited to, to the following:

- Seismic category I ESWPT
- Seismic category I PSFSV
- Seismic category I UHSRS

It is the responsibility of the COL Applicant to address the potential SSSI effect of the R/B complex and T/B on the site specific seismic category I structures.

The site-specific seismic response analysis of the R/B complex and the site-specific Category I structures addresses factors that affect the response of the combined soil-structure dynamic system that include, but are not limited to, the following:

- Properties and layering of the soil, including fill concrete and backfill modeled depending on its horizontal extent
- Depth of the water table, including seasonal variations when appropriate
- Basemat embedment
- Flexibility of the basemat
- Presence of nearby structures

Appropriate modeling techniques capable of capturing the various site-specific SSI effects are used for the analysis. The dynamic properties are assigned to the models for site-specific Category I structures corresponding to the level of stresses generated.

The input used for the site-specific analysis must be derived from geotechnical and seismological investigations of the site. The input control motion derived from the site-specific FIRS is applied in the SASSI analyses at the bottom-of-foundation control point. Site-specific SSI analyses account for the uncertainties and variations of the subgrade properties by using at least three sets of site profiles that represent the best estimate, lower bound, and upper bound (BE, LB, and UB, respectively) soil and rock properties. If sufficient and adequate soil investigation data are available, the LB and UB soil properties are established to cover the mean plus or minus one standard deviation for every layer. In accordance with the specific guidelines for SSI analysis contained in Section II.4 of SRP 3.7.2 (Reference 3.7-16), the LB and UB values for initial soil shear modulii (G_S) are established as follows:

$$G_s^{(LB)} = \frac{G_s^{(BE)}}{(1+C_v)} \quad \text{and} \quad G_s^{(UB)} = G_s^{(BE)} \quad (1+C_v)$$

For well investigated sites, the C_V should be no less than 0.5. For sites that are not well investigated, the C_V for shear modulus shall be at least 1.0.

The site-specific SSI analysis must use stiffness and damping properties of the subgrade materials that are compatible with the strains generated by the site-specific design earthquake (SSE or/and OBE). However, soil material damping shall not exceed 15% as stipulated in SRP 3.7.1 (Reference 3.7-10). The COL Applicant is to evaluate the strain-dependent variation of the material dynamic properties for site materials. If the strains in the subgrade media are less than 2%, the strain-compatible properties can be obtained from equivalent linear site-response analyses using soil degradation curves. Degradation curves that are published in literature can be used after demonstrating their applicability for the specific site conditions. The strain-compatible soil profiles for the site-specific verification SSI analyses of the major seismic category I structures can be obtained from the results of the site response analyses that are performed to calculate site-amplification factors for the development of GMRS, as described in Subsection 3.7.1.1.

To assure the proper comparability, the site-specific verification SSI analyses must use the same verified and validated models of the R/B complex as those used for the US-APWR standard plant design.

The ISRS at major floor and equipment locations and soil pressures on the basement exterior walls that are obtained from all considered soil cases are enveloped and broadened in the site-independent analysis. The plots, tables, and digitized data are then documented for review and comparison with the corresponding results from site-specific analyses. The COL Applicant is to verify that the results of the site-specific SSI analysis for the broadened ISRS are enveloped by the US-APWR standard design. This is accomplished by comparing site specific ISRS results for all locations provided in Appendix 3B of MUAP-10006 (Reference 3.7-48) and ensuring the site-independent results in MUAP-10006 bound the site-specific results.

Simplified SSI modeling approaches, such as a lumped parameter model, can be employed for the site-specific seismic response analyses of seismic category I and II buildings and structures that are not part of the US-APWR standard design if it is demonstrated that for the specific site conditions the following applies:

- The basemats are much stiffer than the supporting subgrade
- The SSI impedance functions remain relatively constant in the range of frequencies important for the design
- The consideration of basemat embedment yields conservative results

In accordance with SRP 3.7.2 (Reference 3.7-16), Section II.4, fixed base response analysis can be performed if the basemats are supported by subgrades having a shear wave velocity of 8,000 ft/s or higher, under the entire surface of the foundation.

3.7.2.5 Development of Floor Response Spectra

The SASSI analyses provide results for the response of the R/B complex due to the three directional design input ground motion for both the cracked and uncracked R/B complex models for each of the generic soil profiles. ISRS are generated for various areas of the R/B complex in accordance with RG 1.122 (Reference 3.7-26) to serve as the seismic design basis for the design of pipe and equipment. The ISRS may be developed from the SASSI ARS data for any node location or damping values, or for variable damping where permitted by ASME Code Case N411-1, as discussed in RG 1.61 (Reference 3.7-15). At selected node locations, ARS in the three orthogonal directions are calculated for each of the three orthogonal directions of the input ground motion from time histories generated by SASSI. The ARS are calculated at 301 frequency points equally distributed on the logarithmic scale at the range of frequency from 0.1 Hz to 100 Hz. The ARS for particular damping value obtained for the three directions of the input ground motion are then combined using the Square Root Sum of the Squares (SRSS) method as follows:

$$ARS_X = \sqrt{ARS_{XX}^2 + ARS_{YX}^2 + ARS_{ZX}^2}$$

$$ARS_Y = \sqrt{ARS_{XY}^2 + ARS_{YY}^2 + ARS_{ZY}^2}$$

$$ARS_Z = \sqrt{ARS_{XZ}^2 + ARS_{YZ}^2 + ARS_{ZZ}^2}$$

where:

- $ARS_{(m)(n)}$ are the SASSI ARS results for the response in "n" direction due to earthquake in "m" direction;
- ARS_X, ARS_Y, and ARS_Z are the combined ARS of the structural response in NS (x), EW (y), and vertical (z) direction, respectively.

Once the results of each of the generic soil cases are combined through SRSS at the nodes, the results are grouped for the nodes within the footprint/support of the equipment or floor areas for which the ISRS is developed.

The grouped nodal results are then enveloped for each of the soil cases and both structural stiffness levels. Enveloping the responses at the grouped nodes is to provide an ISRS for the equipment design and qualification that considers the potential non-uniform input at their support locations including the rocking and torsional effects. The spectra from each analysis (SSI and SSSI) are enveloped. In order to incorporate the effects of SSSI with the adjacent T/B in the R/B Complex design ISRS, the results obtained from the site-independent SSI of R/B Complex FE model are enveloped with the SSSI analyses of the combined model of R/B Complex and T/B presented in subsection 3.7.2.8. The resulting spectra are broadened by 15% in spectral frequency to account for uncertainties in the analysis parameters.

Further, when the ISRS are used for equipment qualification,the valleys between adjacent peaks in the enveloped ISRS are filled to capture potential frequency shifts within the range of the SSI and SSSI responses obtained from the generic soil profiles. To fill in the valleys in the ISRS, the lower peak is extended diagonally until it intersects with the side slope of the adjacent higher peak.

To generate additional ISRS at other damping values as necessary for design of SSCs, the same process described above is repeated.

In the case where seismic qualification by testing is performed in accordance with IEEE Std 344-2004 (Reference 3.7-13), test response spectra which replicate the OBE response spectra are not required since the OBE condition is no longer used as a design basis. The US-APWR program for seismic and dynamic qualification of mechanical and electrical equipment is discussed in Section 3.10.

No safety-related systems and components are present in non-seismic category I building structures, such as the AC/B, A/B and T/B. The design, installation, and mounting of non safety-related systems and components in these buildings are based on the applicable site-specific building codes and standards.

3.7.2.6 Three Components of Earthquake Motion

As previously discussed in Subsection 3.7.1.1, the seismic analyses of the major seismic category I structures are based on one set of three mutually orthogonal artificial time histories, with each of the three directional components being statistically independent of the other two. The acceleration time histories of the horizontal H1 and H2 components of the earthquake are applied in N-S direction and E-W directions respectively. The acceleration time history V is applied in the vertical direction.

The three components of the earthquake are applied on the seismic model separately in ACS SASSI (Reference 3.7-17) for obtaining the maximum accelerations of the response in the three orthogonal directions. The maximum responses of interest of SSCs obtained from the responses of each of the three components of motion are then combined using SRSS in accordance with RG 1.92, Rev.2 (Reference 3.7-27). The combined maximum accelerations, obtained through the process described previously in Subsection 3.7.2, are then used as basis for development of the SSE loads used for the design of structural members, components and connections of US-APWR standard plant. These SSE design loads are applied as static loads on the detailed FE model in conjunction with other design loads and load combinations.

The development of the ISRS uses the SRSS method to combine the responses from the three components of the earthquake motion.

Although the above approach has been used for seismic analysis of the major seismic category I structures, seismic responses of other seismic systems and subsystems due to the three components of earthquake motion can be combined using any one of the following methods in accordance with RG 1.92, Rev.2 (Reference 3.7-27):

i. The peak responses due to the three earthquake components from the response spectra and equivalent static analyses are combined using the SRSS method.

ii. The peak responses due to the three earthquake components are combined directly, using the Newmark combination method that assumes that when the peak response from one component occurs, the responses from the other two components are 40% of the peak (100%-40%-40% method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus) are considered.

iii. The time-history of the responses from the three earthquake components that are applied simultaneously can be combined algebraically at each time step to obtain the combined response time-history. The design seismic loads are selected from the maximum values or the most critical combination of values extracted from the time history results representing the responses directly related to the design of the particular element considering sign reversals, such as the relevant internal forces or stresses in the element.

3.7.2.7 Combination of Modal Responses

As previously discussed, the seismic responses of the seismic category I building models are obtained using three-dimensional SSI models with the program ACS SASSI (Reference 3.7-17). ACS SASSI utilizes time history analysis in the frequency domain in which the equations of motion are solved using a global complex matrix that is assembled from the complex matrices for the soil and structural elements. Therefore modal combination is not utilized.

When the modal superposition time history analyses or response spectra analyses are used for seismic design of other seismic category I and seismic category II systems and subsystems, it may not be practical to capture higher frequency modes that are not excited by the input motion. In modal superposition, only modes with frequencies less than the frequencies defining the cutoff or ZPA response participate in the modal solution. The modal contribution of the residual rigid response for modes with frequencies greater than the cutoff or ZPA frequency is accounted for by using the missing mass method. As permitted in Section 1.4.1 of RG 1.92 (Reference 3.7-27), the missing mass contribution, scaled to the instantaneous input acceleration, is treated as an additional mode in the algebraic summation of modal responses at each time step. The missing mass contribution is considered for all DOF. When using the Lindley-Yow method in response spectra analyses, the missing mass may be captured using the Static ZPA method as described in Section 1.4.2 of RG 1.92, Rev. 2 (Reference 3.7-27).

When the response spectra method of analysis is used (see Subsection 3.7.3.1 for a discussion of response spectra methods of analysis), modal responses have been combined by one of the RG 1.92, Rev.2 (Reference 3.7-27), methods, or by the 10% grouping method described below. In some applications, the more conservative modal combination methods contained in Rev.1 of RG 1.92 (Reference 3.7-28) are also used, as permitted in Revision 2 of RG 1.92 (Reference 3.7-27).

For the grouping method, the total unidirectional seismic response for subsystems is obtained by combining the individual modal responses using the SRSS method for frequencies spaced more than 10%.

For subsystems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen so that the differences between the frequencies of the first mode and the last mode in the group do not exceed 10% of the lower frequency.

The combined total response for systems having such closely spaced modal frequencies is obtained by adding to the SRSS of all modes the product of the responses of the modes in each group of closely spaced modes.

This can be represented mathematically as follows:

$$R^2 = \sum_{k=1}^{N} R_k{}^2 + \sum_{q=1}^{P} \sum_{l=i}^{j} \sum_{m=i}^{j} \left| R_{lq} \cdot R_{mq} \right| \qquad l \neq m$$

where

R = total unidirectional response

R_k = the peak value of the response due to the k^{th} mode

R_{lq}, R_{mq} = are the modal responses, R_l and R_m within the q^{th} group

N = total number of modes considered

P = number of groups of closely spaced modes

i = lowest modal number associated with group j of closely spaced modes

j = highest modal number associated with group j of closely spaced modes

Alternatively, a more conservative ten percent grouping method can be used in the seismic response spectra analyses. The groups of closely spaced modes are chosen so that the difference between two frequencies (the first and last mode in a group) is no greater than 10%. Therefore,

$$R^2 = \sum_{k=1}^{N} R_k{}^2 + 2\sum \left| R_i R_j \right| \qquad i \neq j$$

The second summation is to be done on all i and j modes whose frequencies are closely spaced to each other.

All terms for the modal combination remain the same as defined above.

The 10% grouping method is more conservative than the grouping method because the same mode can appear in more than one group. The 10% grouping method is used for piping as described in Subsection 3.12.3.2.4.

For the seismic response spectra analysis, the ZPA cut-off frequency is 50 Hz. High frequency or rigid modes must be considered using the static ZPA method, the left-out force method as described in Subsection 3.7.2.7 below, or the Kennedy Missing Mass method contained in Revision 2 of RG 1.92 (Reference 3.7-27).

3.7.2.7.1 Left-Out-Force Method (or Missing Mass Correction for High Frequency Modes)

The left-out-force method is based on the Left-Out-Force Theorem. This theorem states that for every time history load, there is a frequency, f_r, called the “rigid mode cutoff frequency” above which the response in modes with natural frequencies above f_r will very closely resemble the applied load at each instant of time. These modes are called “rigid modes.” The formulation follows and is based on the method used in the computer program PIPESTRESS (Reference 3.7-29). The left-out-force method is not used for seismic analysis of the major seismic category I structures; however, it may be used for other seismic category I and II systems and subsystems.

The left-out-force vector for time history analyses, { *Fr* }, is calculated based on lower modes:

$$\{ Fr \} = [1 - \sum M e_j e_j^T] f(t)$$

where

f (t) = the applied load vector

M = the mass matrix

e_j = the eigenvector

Note that ∑ only represents the flexible modes, not including the rigid modes.

In the response spectra analysis, the total inertia force contribution of higher modes can be interpreted as:

$$\{ Fr \} = A_m [M] [\{ r \} - \sum P_j e_j]$$

where

A_m = the maximum spectral acceleration beyond the flexible modes

[M] = the mass matrix

{ r } = the influence vector or displacement vector due to unit displacement

P_j = participation factor, where

$$P_j = e_j^T [M] \{ r \}, \quad \{ Fr \} = A_m [M] \{ r \} [1 - \sum M e_j e_j^T]$$

In the response spectra analysis, the low frequency modes are combined by one of the modal combination methods in accordance with RG 1.92, Rev.2 (Reference 3.7-27) as discussed above. For each support level, there is a pseudo-load vector or left-out-force vector in the X, Y, and Z directions.

These left-out-force vectors are used to generate left-out-force solutions which are multiplied by a scalar amplitude equal to a magnification factor specified by the user. As an alternative the acceleration associated with a cutoff frequency can be used instead of the ZPA provided the number of modes chosen is such that the results of the analysis are within 10 percent of the results of an analysis that considers the additional number of modes. This factor is usually the ZPA of the response spectra for the corresponding direction. The resultant low frequency responses are combined by the SRSS with the high frequency responses (rigid modes results).

3.7.2.8 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

The locations of all major buildings within the power block are shown on the general arrangement drawings in Section 1.2.

Seismic category II structures have been analyzed for the same seismic loads and using the same seismic analysis methods described for seismic category I SSCs in Subsection 3.7.2.1 to verify that they will not collapse or adversely interfere with the standard plant seismic category I R/B complex or adversely affect the MCR occupants. Seismic category II is defined in Section 3.2. By definition, seismic category II structures are designed to retain their position to the extent necessary to assure that they will not impact the function or integrity of seismic category I SSCs.

NS structures have been located such that, in case of their collapse or failure, they do not have the potential to impact seismic category I SSCs, either directly or indirectly.

Maximum lateral earth pressure due to the backfill, surcharge due to live load or adjacent basemat bearing pressures, groundwater, and other such static-load effects on below-grade exterior walls are discussed in Section 3.8. The design of below grade exterior walls for US-APWR seismic category I structures takes into account any dynamic increases of these loads due to a seismic event. This is accomplished through the use of conservative maximum static and dynamic lateral pressure distribution profiles developed using analysis methods provided in Section 3.5.3 of ASCE 4-98 (Reference 3.7-9) and as discussed in Subsection 3.8.4.

The COL Applicant is to assure that the design or location of any site-specific seismic category I SSCs, for example pipe tunnels or duct banks, will not expose those SSCs to possible impact due to the failure or collapse of non-seismic category I structures, or with any other SSCs that could potentially impact, such as heavy haul route loads, transmission towers, non safety-related storage tanks, etc. Alternately, site-specific seismic category I SSCs may be designed for impact loads due to postulated failure of the non-seismic category I SSCs.

Following is a discussion of major structures in the power block area with respect to potential interaction with seismic category I structures.

3.7.2.8.1 AC/B

The AC/B is designed as a NS structure on reinforced concrete foundation located approximately 16 inches from the west side of the A/B (seismic category II). If the AC/B were to fail or collapse, it could impact the A/B which is a seismic category II structure located on the R/B complex common basemat. The AC/B is smaller, shorter, and much less massive than the reinforced concrete A/B. In the unlikely event of impact, there would not be sufficient kinetic energy transfer to cause the A/B to displace beyond acceptable limits. Specifically, the A/B would not displace enough to impact the R/B or PS/Bs.

The design philosophy of the AC/B is stated as follows.

- The seismic design is in accordance with the International Building Code (Reference 3.7-30) with an Importance Factor of 1.0.

- The structure is designed in accordance with applicable building codes.

3.7.2.8.2 T/B

The T/B is structurally designed as seismic category II, such that its integrity will not be impacted by a design basis seismic event; that is the T/B will not fail or collapse due to seismic loading. The T/B is located on the south sides of the R/B complex and is separated from these structures by approximately 20 feet (see Figures in Section 1.2 for details). This is sufficient distance to preclude interaction due to seismic motion of either structure. SSSI interaction is discussed in Section 3.7.2.4 and sliding interaction is discussed in Section 3.8.5.

The T/B is a reinforced concrete structure below grade and a braced steel frame structure above grade. The design philosophy of the T/B is stated as follows.

- The reinforced concrete structure is designed in accordance with the ACI 349-06 code (Reference 3.7-31), and the braced steel frame structure is designed in accordance with the AISC N690 code (Reference 3.7-32).

- The design of the T/B is based on static and dynamic analyses utilizing three dimensional FE models.

- Although the T/B is a seismic Category II structure, the T/B is designed and analyzed as a seismic category I structure. This is described in MUAP-11002 (Reference 3.7-61).

3.7.2.8.3 A/B

The A/B contains the US-APWR standard plant radioactive waste processing facility. This facility is designated as Classification RW-IIa in accordance with RG 1.143, the criteria in Sections 5.1 and 5.2 of (Reference 3.7-19). However, the A/B is designed as seismic category II. The seismic, severe wind, tornado, hurricane, and flood design requirements for seismic category II are more stringent than those of Classification RW-IIa as outlined in RG 1.143 (Reference 3.7-19). The A/B is located on a common basemat with the R/B, PCCV, CIS, East and West PS/B, and ESWPC. The A/B is situated on the west side of the R/B, and has the west PS/B on its south side and the AC/B on its west side.

The majority of the A/B is a reinforced concrete structure with one floor level below grade and three stories above grade. The design philosophy of the A/B is stated as follows.

- The reinforced concrete structure is designed in accordance with the ACI 349-06 code (Reference 3.7-31), and the steel beams supporting some floor slabs are designed in accordance with the AISC N690 code (Reference 3.7-32).

- The A/B is designed as a seismic category I structure and analyzed as part of the R/B complex.

3.7.2.8.4 R/B and PCCV

The R/B and PCCV are seismic category I structures within the R/B complex. The modeling of the R/B complex is described in Technical Report MUAP-10006 (Reference 3.7-48). The R/B rests on a common basemat with and envelopes the PCCV up to the R/B roof, which varies in elevation as shown on the general arrangement drawings in Section 1.2. However, to preclude seismic and structural interaction above the common basemat, the R/B is separated from the PCCV with a 4 in. minimum gap at all above-basemat locations. The gap has been sized to prevent contact between the R/B and PCCV super-structures even if the maximum translational and rotational displacements due to a seismic loading (and other loading) were to occur. The gap size has been determined by considering, at all potential interaction locations, the absolute summation of the deflection associated with each super-structure, obtained from the time history analysis results for those structures.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

To account for variations in the structural frequencies due to the uncertainties in parameters, such as material and mass properties of the structures, damping values, soil properties, SSI analysis techniques, and the seismic modeling methods, the ISRS are developed from six SSI soil profiles representing a range of soft soil to hard rock conditions and two structural stiffnesses representing cracked and uncracked conditions values. These 12 cases and 8 additional cases from SSSI analysis are enveloped and then broadened by ±15% as described in Section 3.7.2.5. Developing enveloping ISRS using this range of parameters and the CSDRS as an input motion creates a design envelop that will encompass most variations in site-specific conditions.

3.7.2.10 Use of Constant Vertical Static Factors

The plant design does not utilize constant vertical static factors in the seismic design. The vertical component of the seismic motion is obtained using one of the analysis methods described in Subsection 3.7.2.1. The vertical component is combined with the horizontal components of the seismic motion as described in Subsection 3.7.2.6.

3.7.2.11 Method Used to Account for Torsional Effects

Inertial torsional effects are inherently considered in the seismic analysis using a 3D FE model.The site-independent SSI analyses are performed using FE models described in Section 3.7.2.3 that represent the general layout of the building and explicity account for eccentricities between the center of mass and center of rigidities.

The structural members of category I and II buildings are designed for two types of torsional effects: (1) torsional responses captured in the seismic response analysis; and (2) accidental torsion. The accidental torsion considers torsional effects that are not captured in the seismic response analyses such as torsion that is due to incoherency (spatial variation) of the input ground motion, non vertically propagating incident waves, and/or accidental eccentricities. The accidental torsional effect is included in accordance with SRP 3.7.2 Section II (Reference 3.7-16) in the design of all seismic category I and II structures by use of the following process:

- The accidental torsional moments are computed by determining an additional building torsion equal to story shear force with a moment arm of +/- 5% of the plan dimension of the floor perpendicular to the direction of the applied motion. This computation is performed for both horizontal directions.

- The accidental torsional moments are assumed to act in the same direction on each structure unless otherwise demonstrated in the seismic analysis. Both positive and negative accidental torsional moments are considered in the design of building structures in order to capture worst case effects.

- The accidental torsional moment is combined with the inertial torsional moment. This is computed conservatively so that the combined torsional moment is additive for each floor elevation. The combined torsional moment is distributed to the resisting structural elements in proportion to their relative stiffnesses.

3.7.2.12 Comparison of Responses

The R/B complex is analyzed using time history analysis methods.

As described in Subsection 3.7.1.1, the time history analyses are based on design ground motion time histories which have been developed from seed recorded time histories and meet the requirements of “Acceptance Criteria, Design of Time History Option 1: Single Set of Time Histories, Approach 1”, NUREG-0800, SRP 3.7.1, Section II (Reference 3.7-10). Since only a time history analysis method is used, comparison of the responses between the response spectrum method and a time history analysis method, as per SRP Section 3.7.2.II.12 (Reference 3.7-16), is not applicable.

3.7.2.13 Methods for Seismic Analysis of Dams

The US-APWR standard plant design does not include dams. It is the responsibility of the COL Applicant to perform any site-specific seismic analysis for dams that may be required.

3.7.2.14 Determination of Dynamic Stability of Seismic Category I Structures

Dynamic stability of the R/B complex is determined in Section 3.8.5. The dynamic FE model described in Section 3.7.2.3 is used to calculate overturning, flotation and dynamic bearing pressure. The R/B complex and T/B will slide during a large earthquake. A non-linear analysis utilizing five separate acceleration time histories and the dynamic FE model is used for the sliding analysis described in Section 3.8.5.5.

The US-APWR standard plant design is based on the assumption, as discussed in Chapter 2, that there is no potential for liquefaction of the supporting media. In order to verify the dynamic stability of US-APWR standard plant and site-specific seismic category I structures, site-specific investigations are performed of the supporting media as described in Subsection 2.5.4.8 to verify that there is no potential for liquefaction. The site-specific factor of safety against liquefaction is determined to confirm the dynamic stability of seismic category I structures for the US-APWR standard design with respect to liquefaction.

3.7.2.15 Analysis Procedure for Damping

The analysis procedure of damping in the various elements of the soil-structure system model has been discussed in Subsections 3.7.1.2, 3.7.2.3, and 3.7.2.4.

3.7.3 Seismic Subsystem Analysis

This section addresses seismic analysis of civil structure-related seismic category I subsystems, which are analyzed in accordance with NUREG-0800, SRP 3.7.3 (Reference 3.7-35). The civil structure-related subsystems are accounted for in the global seismic models of the seismic category I building structures described in Subsection 3.7.2.3 by considering the mass and mass distribution of the subsystems in the models. However, seismic analysis of the subsystems are generally performed separately because the subsystems do not contribute to the building stiffness and because the seismic responses of the buildings (ISRS as discussed in Subsection 3.7.2.5) serve as the seismic design input motion for the subsystems. SSCs that are seismically analyzed as civil structure-related subsystems include:

- Structures such as miscellaneous steel platforms, stairs, and walkways.
- Structures such as reinforced masonry block walls and enclosures.
- HVAC ducts and duct supports. The design of HVAC ducts and duct supports is addressed further in Appendix 3A.
- Conduits and conduit supports. The design of conduits and conduit supports is addressed further in Appendix 3F.
- Cable trays and tray supports. The seismic qualification of cable trays and tray supports is addressed in Appendix 3G.
- Pipe racks and pipe support framing. These structures may also be analyzed as part of mechanical piping subsystems as discussed in Section 3.12.
- Pipe whip restraints. See Section 3.6 and Appendix 3B for a discussion of the design of pipe whip restraints for dynamic loads due to pipe rupture and Appendix 3E for discussion of high energy piping design.
- Equipment cabinet structural framing and/or mounting.

In addition to the above, civil structure-related subsystems also include those seismic category I and II SSCs such as pipe tunnels, conduit tunnels, dams, dikes, aboveground tanks, and the like, which are exterior to the R/B, PCCV, PS/Bs, and the ESWPT.

Each non-category I system and component is designed to be isolated from any seismic category I systems and components by either a constraint or barrier, or is remotely located with regard to the seismic category I systems and components. If it is not feasible or practical to isolate the seismic category I systems and components, adjacent non-category I systems and components are analyzed for the same seismic input motion that is applicable to the seismic category I systems and components. In this case, the analysis demonstrates position retention of the non-category I subsystems and components, with no adverse interaction effects on seismic category I SSCs. For non-category I systems and components attached to seismic category I systems and components, the dynamic effects of the non-category I subsystems and components are simulated in the modeling of the seismic category I systems and components. The attached non-category I systems and components, up to the first anchor beyond the interface, are designed in such a manner that during an earthquake of SSE intensity, the structural integrity and safety functions of the seismic category I systems and components are not jeopardized.

Seismic and dynamic qualification of mechanical and electrical equipment and subsystems performed by testing is discussed in Section 3.10 and Appendix 3D. Mechanical subsystems include mechanical equipment, piping, vessels, tanks, heat exchangers, valves, and instrumentation tubing and tubing supports. The seismic analysis of mechanical subsystems is addressed in Sections 3.9 and 3.12. The RCL analysis is discussed in Appendix 3C.

A list of seismic category I mechanical and fluid systems, components, and equipment is given in Table 3.2-2. Seismic analysis of civil structural items related to those subsystems is discussed in this subsection.

3.7.3.1 Seismic Analysis Methods

Modal response spectra analysis, time history analysis, or equivalent static load analysis methods may be used for seismic analysis of seismic category I subsystems. The methods are the same as those discussed in Subsection 3.7.2.1 and conform to the requirements of SRP 3.7.1 and SRP 3.7.2 (References 3.7-10 and 3.7-16).

Time history analysis of seismic systems is discussed in Subsection 3.7.2. The time-history seismic analysis of a subsystem can be performed by simultaneously applying the displacements and rotations at the interface point(s) between the subsystem and the system. These displacements and rotations are the results obtained from a model of a larger subsystem or a system that includes a simplified representation of the subsystem.

The choice of applied seismic analysis method depends on the desired level of precision and the level of complexity of the particular subsystem being designed. The equivalent static load method of analysis is predominantly used for civil structure-related seismic subsystems and is generally the preferred method because it is relatively simple and at least as conservative as the other more detailed methods. For example, the equivalent static load analysis method is generally used for miscellaneous steel platforms, stairs, and walkways, reinforced masonry block walls and enclosures, HVAC ducts and duct supports, electrical tray and tray supports, and conduits and conduit supports.

The time history or response spectra generated at the support locations of the subsystem are utilized as the input motion for performing the seismic dynamic analysis of the subsystem. However, where these input motions are not readily available, the input motions generated at the closest distances away from the structural support location can be adapted for use. The structural linkage (i.e., intervening structural element) between these two locations, and the additional amplification of the response due to the presence of the intervening structural element are considered in the analysis. For cases where the intervening structure is rigid (i.e., frequency > 50 Hz), the transformation effect due to the rigid body motion of the intervening structure can be taken into account by linear interpolation of the ISRS at the reference locations adjacent to the structural supported locations of the subsystem. Alternatively, the effect can be represented by adding a rigid link in the subsystem model from the reference location associated with the input motion to the support of subsystem location.

For places where the intervening structural element is flexible (i.e., frequency < 50 Hz), the seismic dynamic analysis of the subsystem model can be expanded to include the mass and stiffness of the flexible intervening structural element to analyze the subsystem response. Alternatively, the subsystem seismic input amplified time history and, if necessary, additional ISRS at the subsystem support locations can be generated by using a detailed de-coupled model of the flexible intervening structure provided the applicable de-coupling criteria of SRP 3.7.2 Acceptance Criteria 3B (Reference 3.7-35) or Section 4153.2 of NOG1-2004 (Reference 3.7-22) for cranes are met for the subsystem. When time histories of in-structure motions from dynamic analysis of the supporting soil-structure system are used, frequency content of the time histories is varied to be consistent with the broadening of ISRS. An acceptable method to vary the frequency content of the in-structure accelerations time history for the best estimate soil properties is by expanding and shrinking the time history within 1/(1 ± 0.15) so as to change the frequency content within ± 15%.

Torsional effects due to the significant effect of eccentric masses connected to a subsystem are included in the subsystem analysis. For rigid components (i.e., those with natural frequencies greater than the ZPA cutoff frequency of 50 Hz), the lumped mass is modeled at the center of gravity of the component with a rigid link to the appropriate point in the subsystem. For flexible components having a frequency less than the ZPA, the subsystem model is expanded to include an appropriate model of the component.

Regardless of the method chosen, to avoid resonance, the fundamental frequencies of components and equipment are preferably selected to be less than one half or more than twice the dominant frequencies of the support structure. If this is not practical, equipment and components with fundamental frequencies within this range are designed for any associated resonance effects in conjunction with all other applicable loads.

The equivalent static load method of analysis and the various modal response spectra analysis methods are described in the following subsections.

3.7.3.1.1 Equivalent Static Load Method of Analysis

The equivalent static load method involves the use of equivalent horizontal and vertical static forces applied at the center of gravity of various masses. The equivalent force at a mass location is computed as the product of the mass and the seismic acceleration value applicable to that mass location. Loads, stresses, or deflections obtained using the equivalent static load methods are adjusted to account for the relative motion between points of support when significant.

3.7.3.1.2 Single DOF, Single Mode Dominant or Rigid Structures and Components

For rigid structures and components, single DOF structures and components, or for cases where the response is such that the response of the system is single mode dominant, the following procedures may be used:

- For rigid SSCs (fundamental frequency greater than 50 Hz), an equivalent seismic load is defined for the direction of excitation as the product of the component mass and the ZPA value obtained from the applicable ISRS.

- A rigid component (fundamental frequency greater than 50 Hz), whose support can be adequately represented by a flexible spring, can be modeled as a single DOF model in the direction of excitation (horizontal or vertical directions). The equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value corresponding to the natural frequency of the supported component from the applicable ISRS. If the frequency of the supported component is not determined, the peak acceleration from the applicable ISRS of the supported component is used. Supported components which have been determined to have natural frequencies less than the frequency corresponding to the peak floor acceleration (i.e., whose natural frequencies are to the left of spectra peak on an acceleration versus the frequency spectra plot) also utilize the peak acceleration .

- If the structure, equipment, or component has a distributed mass whose dynamic response is single mode dominant, the equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value at the component natural frequency from the applicable ISRS times a factor of 1.5, with exceptions noted as follows. A factor of less than 1.5 may be used if justified, such as using a factor of 1.0 when the component natural frequency is in the rigid range (greater than 50 Hz), such that no dynamic amplification will occur. A factor of 1.0 is used for structures or equipment that can be represented as simply supported, fixed-simply supported, or fixed-fixed beams as discussed in References 3.7-36 and 3.7-37. In accordance with ASCE 4-98, Subsection 3.2.5.2 (Reference 3.7-9), for cantilever beams with uniform mass distribution, the equivalent-static-load base shear is determined using the peak acceleration, and the base moment is determined using the peak acceleration times a factor of 1.1. If the frequency of a structure, equipment, or component is not determined, the peak acceleration from the applicable ISRS times a factor of 1.5 is used, unless a lower factor is applicable as discussed herein or otherwise justified. Any structures, equipment, or components which have been determined to have natural frequencies less than the frequency corresponding to the peak floor acceleration (i.e., whose natural frequencies are to the left of spectra peak on an acceleration versus the frequency spectra plot) also utilize the peak acceleration times a factor of 1.5 unless a lower factor is applicable as discussed herein or as otherwise justified.

3.7.3.1.3 Multiple DOF Response

This procedure applies to piping, instrumentation tubing, conduit, cable trays, HVAC, and other structural subsystems consisting of multiple spans. The equivalent static load method of analysis can be used for the design of piping systems, and the instrumentation and supports that have significant responses at several vibrational frequencies. In this case, a static load factor of 1.5 is applied to the peak accelerations of the applicable ISRS, unless a lower value is justified. For runs with axial supports, the acceleration value of the mass of piping in its axial direction may be limited to 1.0 times its calculated spectral acceleration value. The spectral acceleration value is based on the frequency of the piping system along the axial direction. The relative motion between support points is also considered.

3.7.3.1.4 Modal Response Spectra Analysis

The methods of modal response spectra analysis that may be utilized for the design of seismic category I and II SSCs are the envelope broadened response spectra method, the peak shifting method, the uniform support motion method and the independent support motion method, described in the following subsections.

3.7.3.1.5 Envelope Broadened Response Spectra Method

The envelope broadened response spectra method is based on the utilization of the ISRS. The envelope broadened response spectra method is discussed in Subsection 3.7.2.5.

3.7.3.1.6 Seismic Response Spectra Peak Shifting

The peak shifting method may be used in place of the broadened spectra method. It determines the natural frequencies $(f_e)_n$ of the system to be qualified in the broadened range of the maximum spectra acceleration peak. If no equipment or piping system natural frequencies exists in the ±15% interval associated with the maximum spectra acceleration peak, then the interval associated with the next highest spectra acceleration peak is selected and used in the following procedure.

Consider all N natural frequencies in the interval:

$$f_j - 0.15f_j \le (f_e)_n \le f_j + 0.15f_j$$

where

f_j = the frequency of maximum acceleration in the envelope spectra

n = 1 to N

The system is evaluated by performing N+3 separate analyses using the envelope un-broadened ISRS and the envelope un-broadened spectra modified by shifting the frequencies associated with each of the spectral values by a factor of +0.15; -0.15; and

$$\frac{(f_e)_n - f_j}{f_j}$$

where

n = 1 to N

The results of these separate seismic analyses are then enveloped to obtain the final result desired (e.g., stress, support loads, acceleration) at any given point in the system. If three different ISRS curves are used to define the response in the two horizontal and the vertical directions, then the shifting of the spectral values, as defined above, may be applied independently to these three response spectra curves.

3.7.3.1.7 Multiple Support Response Spectra Input Methods

The uniform support motion method and the independent support motion methods use multiple-input response spectra which account for the phasing and interdependence characteristics of the various support points. These methods are based on the guidelines provided by the "Pressure Vessel Research Committee Technical Committee on Piping Systems" (Reference 3.7-38) and have been most often applied to plant piping subsystems but are also applicable to other subsystems with multiple support points.

3.7.3.1.7.1 Uniform Support Motion Method

For analyzing plant SSCs supported at multiple locations within a single structure, a uniform response spectrum is defined that envelopes all of the individual response spectra at the various support locations. The uniform response spectrum is applied at all support locations to calculate the maximum inertial responses of the plant SSCs. This is referred to as the uniform support motion method. Modal combinations for this method including missing mass computations must be performed in accordance with RG 1.92, Rev. 2 (Reference 3.7-27). The analysis of seismic anchor motions (i.e., maximum relative support displacement), is performed as a static analysis with all dynamic supports active and the results of this analysis are combined with the piping system seismic inertia analysis results by absolute summation. The seismic response spectrum, which envelopes the supports, is used in place of the spectra at each support in the envelope uniform response spectra. The contribution from the seismic anchor motion of the support points is assumed to be in phase and is added algebraically as follows:

$$q_i = d_j \Sigma P_{ij}$$

where

q_i = combined displacement response in the normal coordinate for mode i

d_j = maximum value of d_{ij}

P_{ij} = participation factor for mode i associated with support j

Σ = summation for support points from j = 1 to N

N = total number of support points

The enveloped response spectra are developed as the seismic input in three perpendicular directions of the coordinate system to include the spectra at all floor elevations of the attachment points and the piping module or equipment, if applicable. The mode shapes and frequencies below the cut-off frequency are calculated in the response spectra analysis. The modal participation factors in each direction of the earthquake motion and the spectral accelerations for each significant mode are calculated. Based on the calculated mode shapes, participation factors, and spectral accelerations of individual modes, the modal inertia response forces, moments, displacements, and accelerations are calculated. For a given direction, these modal inertia responses are combined based on the consideration of closely spaced modes and high frequency modes to obtain the resultant forces, moments, displacements, accelerations, and support loads. The total seismic responses are combined by the SRSS method for all three earthquake directions.

3.7.3.1.7.2 Independent Support Motion Method

When there is more than one supporting structure, the independent support motion method for seismic response spectra may be used.

Each support group is considered to be in a random-phase relationship to the other support groups. The responses caused by each support group are combined by the absolute sum method. The analysis of piping systems for multiply supported piping with independent inputs will be consistent with the recommendation provided in Section 2.4 of NUREG-1061, Volume 4 (Reference 3.7-46), which describes independent support motion (ISM) methodology, sequence of combination, and high frequency modes. If the ISM method is utilized, the criteria presented in NUREG-1061 related to the ISM method are required to be followed according to SRP subsection 3.7.2.II, item 9 (Reference 3.7-16) as provided under SRP Acceptance Criteria. The displacement response in the modal coordinate becomes:

$$q_i = \Sigma P_{ij} d_{ij}$$

A support group is defined by supports that have the same time-history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.

3.7.3.1.7.3 Analysis of Seismic Subsystems versus Qualification by Testing

For the purpose of seismic and dynamic qualification of civil structure-related SSCs by *analysis* using the methods described above in this section, the rigid range is defined as having a natural frequency greater than 50 Hz. This is consistent with the CSDRS defined in Subsection 3.7.1.1. However, for the purpose of *testing* equipment that is not sensitive to response levels caused by high frequency ground motions, rigid is defined as equipment with a natural frequency greater than 33 Hz. If the equipment to be tested is sensitive to the response caused by high frequency ground motions, then rigid is defined as equipment having a natural frequency greater than 50 Hz. This approach is further clarified in the following paragraphs.

Historically, there have been occurrences of ground motions which have caused an exceedance of a plant's design spectra in the high frequency range, where high frequency is defined as 10 Hz or greater. Based on this nuclear plant operating experience, the high frequency response motion exceedances were found to be non-damaging to passive civil structure-related components such as those addressed in the section above, which are typically qualified by analysis. However, nuclear industry experience has found that certain SSCs, in particular components such as relays and other electrical and instrumentation and control devices whose output signals could be affected by high frequency excitation, are potentially sensitive to high frequency motion and can be damaged by high frequency exceedances of the design spectra. A test program is established to identify, evaluate, and qualify or eliminate such SSCs that are potentially sensitive to high frequency exceedances. The US-APWR seismic and dynamic equipment qualification test program for active components including valves, piping, and other plant SSCs is in accordance with IEEE Std 344-2004 (Reference 3.7-13) and is addressed in Section 3.10.

3.7.3.2 Procedures Used for Analytical Modeling

Seismic subsystems are defined as those systems that are not analyzed in conjunction with basemats and subgrade, as previously discussed in Subsection 3.7.2. The procedures used for analytical modeling of subsystems include the use of mathematical computer models comprised of nodes and elements used to represent connections and members. Depending on the complexity of the subsystem, the models may be lumped mass stick models or FE models. The models contain sufficient detail and DOFs to represent the structural and seismic response of the subsystem, and are incorporated into the overall building model when required by the coupling criteria discussed in Subsection 3.7.2.3.4. Depending on the complexity of the seismic subsystem, structure, or component being analyzed, detailed member design may be performed by hand calculations using the results of the overall building structural and seismic analyses. Alternatively, the computer model may be sufficiently detailed to be used for the design calculation of the individual members. In all cases, the computer programs used for analytical modeling of subsystems are verified and validated in accordance with ANSI/ASME NQA-1-2004 (Reference 3.7-23) requirements.

3.7.3.3 Analysis Procedure for Damping

Energy dissipation within a structural system is represented by equivalent viscous dampers in the mathematical model. The damping coefficients used are based on the material, load conditions, and type of construction used in the structural system. The SSE damping values to be used in the dynamic analysis for various seismic category I and II subsystems and their related supports are shown in Table 3.7.3-1(a). The damping values are based on RG 1.61 (Reference 3.7-15). The damping value of conduit, empty cable trays, and their related supports is similar to that of a bolted structure, namely 7% of critical. The damping value of filled cable trays and supports increases with increased cable fill and level of seismic excitation. The use of higher damping values for cable trays with flexible support systems (e.g., rod-hung trapeze systems, strut-hung trapeze systems, and strut-type cantilever and braced cantilever support systems) is permissible, subject to obtaining NRC review for acceptance on a case-by-case basis.

For subsystems that are composed of different material types, the composite modal damping approach with either the weighted mass or stiffness method is used to determine the composite modal damping value. Alternately, the minimum damping value may be used for these systems.

Piping systems are analyzed for SSE using 4% damping. Alternatively, frequency-dependent damping values may be utilized as noted and described in Tables 3.7.3-1(a) and 3.7.3-1(b). The seismic analysis of piping and other mechanical subsystems is addressed in further detail in Sections 3.9 and 3.12.

For subsystems analyzed with the time history direct integration method, Rayleigh damping is used. The Rayleigh damping matrix of the system [C] proportional to the stiffness matrix [K] and mass matrix [M] is obtained as [C] = α [M] + β [K]. In order to model the dissipation of energy in the dynamic system in a conservative manner, the values of the coefficients α and β are adjusted to assure that the damping of the system in a selected range of dominant frequencies remains below the target values of critical damping ratios ξ_i. The selected damping ratio is in accordance with the requirements of RG 1.61. The dominant frequency range is selected considering the natural frequencies of the system being analyzed and the frequency content of the input seismic excitation.

3.7.3.4 Three Components of Earthquake Motion

For seismic category I subsystems, the three components of earthquake motion are considered in the same manner as described in Subsection 3.7.2.6.

Two horizontal components and one vertical component of seismic response spectra are employed as input to a modal response spectra analysis. The spectra are associated with the SSE. In the response spectra and equivalent static analyses, the effects of the three components of earthquake motion are combined using one of the following methods:

- The peak responses due to the three earthquake components from the response spectra analyses are combined using the SRSS method.

- The peak responses due to the three earthquake components are combined directly, using the assumption that when the peak response from one component occurs, the responses from the other two components are 40% of the peak (100%-40%-40% method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus), are considered. This method is not used for piping systems.

3.7.3.5 Combination of Modal Responses

Where seismic subsystems are analyzed by the equivalent static load method of analysis, a combination of modal responses is not applicable. For this method of analysis, static load factors are applied to acceleration values, which are taken from the appropriate ISRS discussed in Subsection 3.7.2.5. The static load factors are chosen using the guideline of Reference 3.7-9 to be sufficiently conservative to capture multi-modal response effects.

For the response spectra method of analysis, the combination of modal responses is performed in the same manner as described in Subsection 3.7.2.7.

3.7.3.6 Use of Constant Vertical Static Factors

As discussed in Subsection 3.7.2.10, the plant design does not utilize constant vertical static factors in the seismic design.

3.7.3.7 Buried Seismic Category I Piping, Conduits, and Tunnels

Buried seismic category I piping, conduits, and tunnels are not present in the US-APWR standard plant design. Physical space is reserved and planned to provide a site-specific seismic category I ESWPT which connects the east and west ends of the ESWPC to the site specific UHS structures. A representative anticipated configuration of the ESWPT is shown on the general arrangement drawings in Section 1.2.

To design and qualify the site-specific safety-related SSCs mounted or housed within the tunnel, the following requirements apply to the site-specific design of the ESWPT as described in Subsection 3.7.2.8:

- ISRS are required. To generate the ISRS on the tunnel walls, basemat and roof, a SASSI program (Reference 3.7-17) SSI analysis is required if soil supported. The SASSI analysis is required to be documented and comply with the same general requirements described for the standard plant design.

3.7.3.8 Methods for Seismic Analysis of Category I Concrete Dams

The US-APWR standard plant design does not include dams. It is the responsibility of the COL Applicant to perform any site-specific seismic analysis for dams that may be required.

3.7.3.9 Methods for Seismic Analysis of Aboveground Tanks

It is the responsibility of the COL Applicant to design seismic category I below- or above-ground liquid-retaining metal tanks such that they are enclosed by a tornado/hurricane missile protecting concrete vault or wall, in order to confine the emergency gas turbine fuel supply.

The other seismic category I liquid-retaining vessels utilized in the design are reinforced concrete vessels whose walls and floors form part of the building structural framework, including the following:

- Spent fuel pit, located in the R/B with top of vessel at level 4F
- Refueling cavity, located in PCCV with top of vessel at level 4F
- Fuel transfer canal, which connects the spent fuel pit and refueling cavity

- Cask washdown pit located in the R/B with top of vessel at level 4F
- Cask loading pit and fuel inspection pit located in the R/B and connected to the spent fuel pit with a canal, with tops of vessels at level 4F
- New fuel storage pit located in the R/B with top of vessel at level 4F
- Refueling water storage pit, located in PCCV below level 2F

Hydrodynamic loads on these liquid-retaining vessels are determined using methods that conform to the provisions of Subsection II.14 of SRP 3.7.3 (Reference 3.7-35) and guidance of ASCE 4-98, Subsection 3.5.4 (Reference 3.7-9). The horizontal response analysis considers both the impulsive mode (in which a portion of the water moves in unison with the tank wall) and the horizontal convective mode (water motion associated with wave oscillation). The seismic analysis of convective hydrodynamic effects also considers the maximum wave oscillation with respect to the potential of creating flooding, which is discussed in Section 3.4.

3.7.4 Seismic Instrumentation

The proposed seismic instrumentation program for the US-APWR is in accordance with NUREG-0800, SRP 3.7.4 (Reference 3.7-39) and all aspects of 10 CFR 50, Appendix S (Reference 3.7-7), which requires that “suitable instrumentation must be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake.” Appendix S of 10 CFR 50 (Reference 3.7-7) also requires a shutdown of the plant if vibratory ground motion exceeding that of the OBE ground motion occurs, or significant plant damage occurs.

3.7.4.1 Comparison with Regulatory Guide 1.12

The proposed seismic instrumentation program is generally in accordance with RG 1.12 and RG 1.166 (References 3.7-40, 3.7-41), and consistent with the methodology used for seismic analysis that is discussed in Subsection 3.7.2. The seismic design of US-APWR standard plant is based on site-independent seismic response analysis of basemats resting on generic supporting media that are subjected to the CSDRS input control motion. The site-independent OBE is defined as 1/3 of the CSDRS presented in Subsection 3.7.1.1. Verification of the site-independent standard design is performed during seismic analyses that consider site-specific conditions, such as soil layering, basemat embedment, water table depth etc. The FIRS, which are developed consistent with the site-specific GMRS define the site-specific control design motion.

The criteria that define the vibratory motion that requires the shutdown of the US-APWR plant are based on the site-specific OBE. The 5% damping FIRS associated with the site-specific OBE must be enveloped by 1/3 of the 5% damping CSDRS. The conditions that require a shutdown of the US-APWR plant are defined by the site-specific OBE at the free-field instrumentation located at grade in the plant yard, unless otherwise justified by the COL Applicant. Unless site-specific OBE is set at 1/3 of the site-specific SSE or lower, these spectra shall be obtained from analysis using as input the site-specific OBE ground motion and properties of the supporting media that are strain-compatible to the site-specific OBE ground motion. When the site-specific OBE is equal or lower than 1/3 of site-specific SSE, the spectra scaled from the 5% damping site-specific SSE response spectra may be used directly for OBE exceedance checks. An OBE exceedance check is performed in accordance with Section 4 of RG 1.166 (Reference 3.7-41) using both a response spectrum check and a cumulative absolute velocity (CAV) check. The comparison evaluation is to be performed within 4 hours of the earthquake using data obtained from the three components of the earthquake motion as defined by the three orthogonal axes of the standard plant (two horizontal and one vertical) on the uncorrected earthquake records. The evaluation is also to include a check on the operability of the seismic instrumentation as mandated by Section 4.3 of RG 1.166 (Reference 3.7-41).

The locations of seismic monitors for the US-APWR standard plant are provided in Subsection 3.7.4.2. The COL Applicant shall provide free-field seismic instrumentation in the vicinity of the power block area at surface grade, which shall be used for shutdown determination, unless otherwise justified. Any such justification shall be based on conditions and requirements specific to the site, and shall include justification for evaluation of OBE exceedance using only measurements from instrumentation installed on the buildings and the structures of the US-APWR standard plant.

The calculation of the CAV is performed in the manner provided in Electric Power Research Institute (EPRI) Report TR-100082 (Reference 3.7-42). As stated in RG 1.166 (Reference 3.7-41), the range of the spectral velocity limit should be 1.0 to 2.0 Hz which is different than that recommended by EPRI. In accordance with RG 1.166 (Reference 3.7-41), for each component of the free-field ground motion, the CAV should be calculated as follows: (1) the absolute acceleration (g units) time-history is divided into 1-second intervals, (2) each 1-second interval that has at least 1 exceedance of 0.025 g is integrated over time, (3) all the integrated values are summed together to arrive at the CAV. The approaches in EPRI Report NP-5930 (Reference 3.7-43) and EPRI Report TR-100082 (Reference 3.7-42) provide additional guidance on determining the CAV.

The site-specific OBE is exceeded and plant shutdown is required in accordance with the criteria of RG 1.166 (Reference 3.7-41), if the first of the following three conditions in combination with either the second or third conditions are met:

1. Any calculation of CAV described above yields a value that is greater than 0.16 g-second.

2. 5% damping ARS generated by free-field ground motion ARS are higher than 0.2 g at frequencies between 2 and 10 Hz, or higher than the site-specific OBE ARS between 2 and 10 Hz, whichever is greater.

3. 5% damping velocity response spectra generated by free-field ground motion are higher than 6 in./sec at frequencies between 1 and 2 Hz, or higher than the site-specific OBE velocity response spectra between 1 and 2 Hz, whichever is greater.

If free-field instrumentation is not used, the criteria of RG 1.166 Appendix A are used for OBE exceedance checks, it is assumed that the checks of CAV and free-field ground spectra are exceeded, and shutdown of the plant is required if the 5% damping spectra are exceeded at any of the in-structure instrumentation.

Additionally, low-level seismic effects would be included in the design of certain equipment potentially sensitive to a number of such events, based on a percentage of the responses calculated for the SSE.

3.7.4.2 Location and Description of Instrumentation

Consistent with the guidance of RG 1.12 (Reference 3.7-40), the seismic instrumentation for the US-APWR standard plant is solid-state multi-channel digital instrumentation with computerized recording and playback capability that allows the processing of data at the plant site within 4 hours of a seismic or other dynamic event.

The US-APWR triaxial time-history accelerograph consists of a centralized digital time history analyzer/recorder with multi-channel capability, which is located in a panel in a room adjacent to the plant MCR, and triaxial acceleration sensors that are provided at the following plant locations:

- On the PCCV basemat, located in the R/B on the B1F level at elevation -23 ft, 4 in.
- On level 2F of PCCV at elevation 25 ft, 3 in., located in the southwest quadrant outside the steam generator and reactor coolant compartment.
- On level 4F of PCCV operating deck slab at elevation 76 ft, 5 in., located in the southwest quadrant outside the steam generator and reactor coolant compartment underneath the access stairs adjacent to the west PCCV buttress.
- On the basemat of the east PS/B on the B1F level at elevation -23 ft, 4 in., in the non-safety related turbine generator anteroom.
- On level 1F of the east PS/B at elevation 3 ft, 7 in., in the non-safety related turbine generator control room.
- Unless otherwise justified by the COL Applicant based on site-specific conditions, at a surface grade location in the vicinity of the power block area, sufficiently far away from structures in order to appropriately measure free-field ground motion.

The locations listed above correlate to structural elements in the structures which have been modeled as mass points in the dynamic analysis so that the measured motion can be directly compared to the design spectra. The instrumentation mounted at the locations listed above is not mounted on equipment, piping, supports, or secondary structural frame members. These locations have been reviewed in accordance with RG 8.8 (Reference 3.7-44) and determined to be consistent with maintaining dose rates as low as practical and maintaining occupational radiation exposures as low as is reasonably achievable for access and maintenance of the instrumentation.

A time-history analyzer/recorder is provided which has the capability to provide pre-event recording time of 3 seconds minimum and post-event recording time of 5 seconds minimum, and to record at least 25 minutes of sensed motion. The recorder portion of the time-history analyzer is to have the capability of a sample rate of at least 200 samples per second in each of the three orthogonal directions of the plant, a bandwidth of 0.20 Hz to 100 Hz, and a dynamic range of 1,000:1 zero to peak. The triaxial acceleration sensors are to have the same dynamic range as the time-history analyzer recorder and a frequency range of 0.20 Hz to 100 Hz. The triggers of the tri-axial acceleration sensor units are to be capable of being set within the range of 0.001g to 0.02g. Batteries are provided with enough capacity for a minimum of 25 minutes of system operation at any time over a 24-hour period, without recharging, in combination with a battery charger whose line power is connected to an uninterruptible power supply.

The seismic instrumentation serves no safety-related function and, therefore, has no nuclear safety design requirements. However, its design and location are in accordance with RG 1.12 (Reference 3.7-40), which requires that the seismic instrumentation:

- will not be affected by the failure of adjacent SSCs during an earthquake;
- will operate during all modes of plant operation, including periods of plant shutdown; and
- is protected as much as practical against accidental impacts.

As required by RG 1.12 (Reference 3.7-40), the seismic instrumentation is rigidly mounted and oriented so that the horizontal components are parallel to the horizontal axes of the standard plant used in the seismic analyses. These features of the seismic monitoring instrumentation are obtained by qualifying the equipment to IEEE Std 344-2004 (Reference 3.7-13); the seismic qualification program is discussed in Section 3.10.

3.7.4.3 Control Room Operator Notification

The US-APWR standard plant is designed such that triggering of the instrumentation described above is annunciated in the MCR of the plant. For sites which will have more than one US-APWR unit, only one unit is required to have seismic instrumentation, provided that the anticipated seismic response at each of the units is considered essentially the same and provided that annunciation is provided at all unit MCRs. The COL Applicant is to determine from the site-specific geological and seismological conditions if multiple US-APWR units at a site will have essentially the same seismic response, and based on that determination, choose if more than one unit is provided with seismic instrumentation at a multiple-unit site.

3.7.4.4 Comparison with Regulatory Guide 1.166

As previously discussed in Subsection 3.7.4.1, the seismic instrumentation and OBE exceedance checks meet the intent of RG 1.166 (Reference 3.7-41). In the case that the COL Applicant provides acceptable justification for not utilizing free-field instrumentation, the OBE exceedance checks can be performed using only uncorrected earthquake data for the three orthogonal plant directions (two horizontal and one vertical) obtained from seismic instrumentation installed at five plant locations (two basemat locations and three upper level locations as described in Subsection 3.7.4.2). It should be noted that the use of five instrument locations is more conservative than the interim OBE exceedance guidelines given in Appendix A of RG 1.166 (Reference 3.7-41), which allow basemat-level only instrumental checks.

The seismic instrumentation program must be in accordance with the guidelines of RG 1.166 (Reference 3.7-41) and EPRI NP-6695 (Reference 3.7-45) which are summarized as follows:

- Assure that a file containing information on all seismic instrumentation is maintained at the plant in accordance with regulatory position C1.1 of RG 1.166 (Reference 3.7-41).
- Implement planning for post-earthquake walkdown inspections by pre-selecting equipment and structures for inspections and pre-determining the content of the baseline inspections.
- Implement guidelines for actions to be performed immediately after an earthquake, including a check of the neutron flux monitoring sensors as part of the specific MCR board checks.
- Assure proper evaluation of ground motion records.
- Assure that after an earthquake at the plant site, an operability check is performed on the seismic instrumentation.
- If a shutdown is required, assure that the pre-shutdown inspections, including a check of the containment isolation system, are performed.

3.7.4.5 Instrument Surveillance (Including calibration and testing)

The seismic instrumentation is in accordance with the type and location requirements discussed in Subsection 3.7.4.2 and RG 1.12 (Reference 3.7-40). The instrumentation requires minimal maintenance and in-service inspection, as well as minimal time and numbers of personnel to conduct installation and maintenance. The seismic monitoring instrumentation is configured such that testing or maintenance can be performed on a single channel without affecting the functioning of other channels.

A seismic monitoring system preoperational test is outlined in Chapter 14.

As required by RG 1.12 (Reference 3.7-40), instrumentation systems are to be given channel checks every 2 weeks for the first 3 months of service after startup. Failures of devices normally occur during initial operation. After the initial 3-month period and 3 consecutive successful checks, monthly channel checks are sufficient. The monthly channel check is to include checking the batteries. The channel functional test should be performed every 6 months. Channel calibration should be performed during each refueling outage at a minimum.

3.7.4.6 Program Implementation

The COL Applicant is to identify the implementation milestone for the seismic instrumentation implementation program based on the discussion in Subsections 3.7.4.1 through 3.7.4.5.

3.7.5 Combined License Information

COL 3.7(1) *The COL Applicant is to confirm that the site-specific PGA at the basemat level control point of the CSDRS is less than or equal to 0.3 g.*

COL 3.7(2) *The COL Applicant is to perform an analysis of the US-APWR standard plant seismic category I design to verify that the site-specific FIRS at the basemat level control point of the CSDRS are enveloped by the site-independent CSDRS.*

COL 3.7(3) *It is the responsibility of the COL Applicant to develop analytical models appropriate for the seismic analysis of buildings and structures that are designed on a site-specific basis including, but not limited to, the following:*

- *PSFSVs (seismic category I)*
- *ESWPT (seismic category I)*
- *UHSRS (seismic category I)*

COL 3.7(4) *The COL Applicant is to review the resulting level of seismic response and determine appropriate damping values for the site-specific calculations of ISRS that serve as input for the seismic analysis of seismic category I and seismic category II subsystems.*

COL 3.7(5) *The COL Applicant is to assure that the horizontal FIRS defining the site-specific SSE ground motion at the bottom of seismic category I or II basemats envelope the minimum response spectra required by 10 CFR 50, Appendix S, and the site-specific response spectra obtained from the response analysis.*

COL 3.7(6) *The COL Applicant is to develop site-specific GMRS and FIRS. The FIRS are compared to the CSDRS to assure that the US-APWR standard plant seismic design is valid for a particular site. If the FIRS are not enveloped by the CSDRS, the US-APWR standard plant seismic design is modified as part of the COLA in order to validate the US-APWR for installation at that site.*

COL 3.7(7) *The COL Applicant is to determine the allowable static and dynamic bearing capacities based on site conditions, including the properties of fill concrete placed to provide a level surface for the bottom of foundation elevations, and to evaluate the bearing loads to these capacities.*

COL 3.7(8) *The COL Applicant is to evaluate the strain-dependent variation of the material dynamic properties for site materials.*

COL 3.7(9)	*The COL Applicant is to assure that the design or location of any site-specific safety-related SSCs, for example pipe tunnels or duct banks, will not expose those SSCs to possible impact due to the failure or collapse of non-seismic category I structures, or with any other SSCs that could potentially impact, such as heavy haul route loads, transmission towers, non safety-related storage tanks, etc.*
COL 3.7(10)	*It is the responsibility of the COL Applicant to address the potential SSSI effect of the R/B complex and T/B on the site specific seismic category I structures.*
COL 3.7(11)	*It is the responsibility of the COL Applicant to confirm the masses and frequencies of the PCCV polar crane and fuel handling crane and to determine if coupled site-specific analyses are required.*
COL 3.7(12)	*It is the responsibility of the COL Applicant to design seismic category I below- or above-ground liquid-retaining metal tanks such that they are enclosed by a tornado/hurricane missile protecting concrete vault or wall, in order to confine the emergency gas turbine fuel supply.*
COL 3.7(13)	*The COL Applicant is to set the value of the OBE that serves as the basis for defining the criteria for shutdown of the plant, according to the site specific conditions.*
COL 3.7(14)	*The COL Applicant is to determine from the site-specific geological and seismological conditions if multiple US-APWR units at a site will have essentially the same seismic response, and based on that determination, choose if more than one unit is provided with seismic instrumentation at a multiple-unit site.*
COL 3.7(15)	*Deleted*
COL 3.7(16)	*The COL Applicant shall provide free-field seismic instrumentation in the vicinity of the power block area at surface grade which shall be used for shutdown determination, unless otherwise justified. Any such justification shall be based on conditions and requirements specific to the site, and shall include justification for evaluation of OBE exceedance using only measurements from instrumentation installed on the buildings and the structures of the US-APWR standard plant.*
COL 3.7(17)	*Deleted*
COL 3.7(18)	*Deleted*
COL 3.7(19)	*The COL Applicant is to identify the implementation milestone for the seismic instrumentation implementation program based on the discussion in Subsections 3.7.4.1 through 3.7.4.5.*

COL 3.7(20) *The COL Applicant is to validate the site-independent seismic design of the standard plant for site-specific conditions, including geological, seismological, and geophysical characteristics, and to develop the site-specific GMRS.*

COL 3.7(21) *The COL Applicant is responsible for the seismic design of those seismic category I and seismic category II SSCs that are not part of the US-APWR standard plant using site-specific SSE design ground motion.*

COL 3.7(22) *The COL Applicant may consider the seismic wave transmission incoherence of the input ground motion when performing the site-specific SSI analyses.*

COL 3.7(23) *The COL Applicant is to verify that the results of the site-specific SSI analysis for the broadened ISRS are enveloped by the US-APWR standard design.*

COL 3.7(24) *The COL Applicant is to verify that the site-specific ratios V/A and AD/V^2 (A, V, D, are PGA, ground velocity, and ground displacement, respectively) are consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the site-specific uniform hazard response spectra.*

COL 3.7(25) *The COL Applicant referencing the US-APWR standard design is required to perform a site-specific SSI analysis for the R/B complex, utilizing a SASSI program such as ACS SASSI (Reference 3.7-17) which contains time history input incoherence function capability. The SSI analysis using SASSI is required in order to confirm that site-specific effects are enveloped by the standard design.*

COL 3.7(26) *SSI effects are also considered by the COL Applicant in site-specific seismic design of any seismic category I and II structures that are not included in the US-APWR standard plant. The site-specific SSI analysis is performed for buildings and structures including, but not limited to, to the following:*

- *Seismic category I ESWPT*
- *Seismic category I PSFSV*
- *Seismic category I UHSRS*

COL 3.7(27) *It is the responsibility of the COL Applicant to perform any site-specific seismic analysis for dams that may be required.*

COL 3.7(28) *Deleted.*

COL 3.7(29) *Table 3.7.2-1, as updated by the COL Applicant to include site-specific seismic category I structures, presents a summary of dynamic analysis and combination techniques including types of models and computer programs used, seismic analysis methods, and method of combination for the three directional components for the seismic analysis of the US-APWR standard plant seismic category I buildings and structures.*

COL 3.7(30) *The COL Applicant is to provide site-specific design ground motion time histories and durations of motion.*

3.7.6 References

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3.7-4 Reactor Site Criteria, Energy. Title 10 Code of Federal Regulations Part 100, U.S. Nuclear Regulatory Commission, Washington, DC.

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3.7-20 Deleted.

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Table 3.7.1-1 CSDRS Horizontal Acceleration Values and Control Points

Control Point (Hz)	Acceleration (g)
2% Damping	
A (50)	0.3
B (12)	1.06
C (2.5)	1.28
D (0.25)	0.17
E (0.1)	0.028
3% Damping	
A (50)	0.3
B (12)	0.92
C (2.5)	1.10
D (0.25)	0.154
E (0.1)	0.0251
5% Damping	
A (50)	0.3
B (12)	0.78
C (2.5)	0.94
D (0.25)	0.14
E (0.1)	0.0226
7% Damping	
A (50)	0.3
B (12)	0.68
C (2.5)	0.82
D (0.25)	0.13
E (0.1)	0.021
10% Damping	
A (50)	0.3
B (12)	0.57
C (2.5)	0.68
D (0.25)	0.12
E (0.1)	0.019

Notes:

1. 0.3 g PGA
2. Based on RG 1.60, Rev. 1 (Reference 3.7-6) amplification factors
3. For Control Points D & E, acceleration is computed as follows:

Acceleration = $(\varpi^2 D / 386.4 \text{ in/sec}^2) \times F_A \times 0.3$

ϖ = 2π x frequency (rad/sec)

D = Displacement (in)

F_A = Amplification Factor from Regulatory Guide 1.60

Table 3.7.1-2 CSDRS Vertical Acceleration Values and Control Points

Control Point (Hz)	Acceleration (g)
2% Damping	
A (50)	0.3
B (12)	1.06
C (3.5)	1.22
D (0.25)	0.12
E (0.1)	0.018
3% Damping	
A (50)	0.3
B (12)	0.92
C (3.5)	1.05
D (0.25)	0.106
E (0.1)	0.0164
5% Damping	
A (50)	0.3
B (12)	0.78
C (3.5)	0.89
D (0.25)	0.094
E (0.1)	0.015
7% Damping	
A (50)	0.3
B (12)	0.68
C (3.5)	0.78
D (0.25)	0.086
E (0.1)	0.014
10% Damping	
A (50)	0.3
B (12)	0.57
C (3.5)	0.65
D (0.25)	0.08
E (0.1)	0.012

Notes:

1. 0.3 g PGA
2. Based on RG 1.60, Rev. 1 (Reference 3.7-6) amplification factors
3. For Control Points D & E, acceleration is computed as follows:

Acceleration = $(\varpi^2 D / 386.4 \text{ in/sec}^2) \times F_A \times 0.3$

ϖ = 2π x frequency (rad/sec)

D = Displacement (in)

F_A = Amplification Factor from Regulatory Guide 1.60

Table 3.7.1-3 Summary of SRP 3.7 Option 1, Approach 1 Requirements Compliance

Requirement		H2(90)	H1(180)	V(UP)
Time Histories Requirements				
Total duration in seconds (if ≥ 20 seconds OK)		22.08	22.08	22.08
Rise time in seconds: Arias intensity 5% (if 1 second or longer OK)		2.815	3.031	1.337
Strong motion duration in seconds: Arias intensity between 5% and 75% (minimum 6 seconds and satisfying NUREG/CR-6728 criteria)(1)		9.543	7.868	10.35
Decay time in seconds: Arias intensity between 75% and 100% (if 5 seconds or longer OK)		9.722	11.181	10.393
Statistical independence (if absolute value ≤ 0.16 OK)		-0.0179	-0.0179	
		-0.0552		-0.0552
			-0.0696	-0.0696
V/A (if 7.51 ≤ V/A ≤ 66.40 OK)(2)		53.179	66.355	42.661
AD/V^2 (if 1.86 ≤ AD/V^2 ≤ 16.79 OK)(2)		4.306	2.997	5.766
Response Spectra Requirements				
SRP 3.7.1 Option 1, Approach 1				
Number of points with acceleration ratio < 1 (if ≤ 5 OK)	2%	2	5	5
	3%	0	0	1
	5%	0	0	0
	7%	0	0	0
	10%	4	0	2
Number of points with acceleration ratio < 0.9 (if 0 OK)	All	0	0	0
Power Spectral Density Function Requirements				
Number of points below 80% of target between 0.3 and 50 hz (if 0 OK)		0	0	0

(1) Refer to Table 3.7.1-4.

(2) Refer to Table 3.7.1-5.

Table 3.7.1-4 Magnitudes and Distance Bins and Strong Motion Duration Criteria (NUREG/CR-6728, Table 3-2, Reference 3.7-14)

M	R (Km)	Duration	
		Rock	Soil
6.5 (6 – 7)	10–50	**3.1**–7.0	3.6–8.2
	50–100	5.1–11.6	5.7–12.8
	100–200	8.1–18.3	8.7–19.5
7.5 (7+)	10–50	6.6–14.0	7.2–16.1
	50–100	8.7–19.5	12.2–27.5
	100–200	11.7–26.3	16.2–**36.5**

Table 3.7.1-5 CEUS V/A & AD/V2 Mean Ratios ± One Standard Deviation

Distance Bin	M	V / A (cm/sec/g), σ_{ln}(1)	AD / V^2, σ(1)	V/A / $e^{(\sigma_{ln})}$ (in/sec/g) (2)	V/A * $e^{(\sigma_{ln})}$ (in/sec/g) (2)	AD/V^2/ $e^{(\sigma_{ln})}$	AD/V^2 * $e^{(\sigma_{ln})}$
10-50, Rock	6.32	31.75, 0.51	6.58, 0.70	**7.51**	20.82	3.27	13.25
10-50, Soil	6.41	51.74, 0.35	3.49, 0.47	14.35	28.91	2.18	5.58
50-100, Rock	6.38	32.59, 0.33	4.66, 0.52	9.22	17.85	2.77	7.84
50-100, Soil	6.57	56.04, 0.36	3.01, 0.48	15.39	31.62	**1.86**	4.86
10-50, Rock	7.38	58.24, 0.72	7.78, 0.63	11.16	47.11	4.14	14.61
10-50, Soil	7.47	128.74, 0.27	3.57, 0.35	38.69	**66.4**	2.52	5.07
50-100, Rock	7.49	50.29, 0.56	10.60, 0.46	11.31	34.66	6.69	**16.79**

(1) See NUREG/CR-6728, Table 3-6, Reference 3.7-14.

(2) Units are changed to facilitate comparison to time history results.

Table 3.7.1-6 Generic Soil Profile Categories

Category (Initial V_S [in top 30m])	Depth to Rock* (ft) for each Category (ft)
270 m/s	200 500
560 m/s	500
900 m/s	100 200
2,032 m/s	100

* For soil and soft rock profiles 270 m/sec and 560 m/sec, underlying baserock conditions reflect soft rock with a shear wave velocity of 1 km/sec. For firm rock profiles 900 m/sec and 2,032 m/sec, underlying baserock conditions reflect hard rock with a shear wave velocity of 2.83 km/sec.

Table 3.7.1-7 Magnitudes, Distances, and Median Peak Accelerations

Profile	Magnitude	Distance(km)	PGAH(g)	PGAV(g)
270-200	7.5	68.0	0.268	0.117
270-500	7.5	62.0	0.232	0.124
560-500	7.5	59.5	0.259	0.130
900-100	7.5	68.0	0.198	0.078
900-200	7.5	65.0	0.204	0.087
2032-100	7.5	52.0	0.193	0.089

Table 3.7.2-1 Summary of Dynamic Analysis and Combination Techniques

Summary of Dynamic Analyses & Combination Techniques				
Model	**Analysis Method**	**Program**	**Three Components Combination (for purposes of dynamic analysis)**	**Modal Combination**
Three-dimensional R/B complex SSI Model [1]	Time History Analysis in Frequency Domain using sub-structuring technique	ACS SASSI	SRSS	N/A
Three-dimensional R/B complex FE Model [2]	1g Static Analysis & Time History Analysis in Time Domain	ANSYS	N/A[2]	N/A
Three-dimensional T/B SSI Model[3]	Time History Analysis in Frequency Domain using sub-structuring technique	ACS SASSI	SRSS	N/A
Three-dimensional T/B FE models[3]	1g Static Analysis & Time History Analysis in Time Domain	ANSYS	N/A	N/A

Notes:

1. The three-dimensional R/B complex SSI model is addressed in Technical Report MUAP-10006 (Reference 3.7-48).
2. The FE models for the R/B complex is used for validation of the dynamic FE seismic models and for static analysis for design of structural members and components as addressed in Section 3.8.
3. The three-dimensional T/B model is addressed in Technical Report MUAP-11002 (Reference 3.7-61).

Table 3.7.2-2 Concrete Material Constants

	Modulus of Elasticity (Young's Modulus) E_c (ksi)	Shear Modulus G_c (ksi)	Poisson's Ratio ν	Remark
PCCV	4,769	2,040	0.17	f'_c = 7,000 psi
R/B	4,031	1,723	0.17	f'_c = 5,000 psi
Containment Internal Structure	3,605	1,540	0.17	f'_c = 4,000 psi

Table 3.7.2-3 Material Properties of Models Used for Seismic Response Analyses

Stiffness Level	Structural Component	Stiffness	Damping
Full (Uncracked) Stiffness	SC module (CIS)	Loading Condition A in Table 3.8.3-4	
	Pre-stressed (PCCV)	100%	3%
	Reinforced Concrete	100%	4%
	Composite (FH/A)	See note (1)	4% concrete 3% steel
	Steel	100%	3%
	RCL	100%	3%
	Massive concrete	100%	4%
Reduced (Cracked) Stiffness	SC module (CIS)	Loading Condition B in Table 3.8.3-4	
	Pre-stressed (PCCV)	50%	5%
	Reinforced Concrete	50%	7%
	Composite (FH/A)	See note (2)	7% concrete 4% steel
	Steel	100%	4%
	RCL	100%	3%
	Massive concrete	100%	4%

(1) See equations in Section 02.4.1.1.6 of MUAP-10006

(2) See equations in Section 02.4.1.1.6 of MUAP-10006, use E = 50% E_c

Table 3.7.2-4 Fixed Base Dynamic Properties of US-APWR Category I Structures (Sheet 1 of 2)

Structure	Direction	Fixed Base Modal Properties	
		Frequency (Hz)	Effective Mass (kip sec^2/ft)
PCCV	NS	4.3	2,042
		12.6	233.4
	EW	4.2	1,535
		12.4	423.9
	Vertical	12.1	2,797
		20.4	258.7
CIS	NS	7.6	80.37
		8.3	570.8
		12.0	338.6
		20.0	132.5
	EW	6.3	1,344
		12.4	423.9
		24.8	171.1
	Vertical	12.1	2,796
		15.4	187.0
		21.0	144.3
R/B	NS	4.9	2,996
		6.0	600.1
		9.5	661.7
		10.9	1,478
		13.8	718.6
	EW	6.1	4,070
		7.5	176.5
		10.4	718.6
		15.6	135.6
	Vertical	11.3	445.0
		12.1	2,797
		13.1	878.1
		16.5	134.7
		18.9	147.0

Table 3.7.2-4 Fixed Base Dynamic Properties of US-APWR Category I Structures (Sheet 2 of 2)

Structure	Direction	Fixed Base Modal Properties	
		Frequency (Hz)	Effective Mass (kip sec^2/ft)
East PS/B	NS	6.4	3,636
		10.9	1,478
		13.2	174.2
	EW	7.1	1,549
		14.3	460.7
		15.6	135.6
	Vertical	12.7	3,645
		16.5	134.7
		21.0	144.3
West PS/B	NS	8.8	302.7
		10.9	1,477
		18.7	112.4
	EW	7.1	1,549
		13.2	128.9
	Vertical	12.7	3,654
		15.1	123.7
		20.6	153.9

Table 3.7.3-1(a) SSE Damping Values

Welded and friction-bolted steel structures and equipment (%)	4
Bearing bolted structures and equipment (%)	7
Prestressed concrete structures (%)	5
Reinforced concrete structures (%)	7(4)
Steel-Concrete Modules (%)	5(4)
Piping systems(1)	4
Full cable trays & related supports (%)	10(2)
Empty cable trays and related supports (%)	7
Full Conduits & related supports (%)	7
Empty conduits & related supports (%)	5
HVAC pocket lock ductwork (%)	10
HVAC companion angle ductwork (%)	7
HVAC welded ductwork (%)	4
Cabinets and panels for electrical equipment (%)	3
Equipment such as welded instrument racks and tanks (impulsive mode) (%)	3(3)
Motors, fans, housings, pressure vessels, heat exchangers, pumps, valve bodies (%)	3

Table 3.7.3-1(b) OBE Damping Values

Welded and friction-bolted steel structures and equipment (%)	3
Bearing bolted structures and equipment (%)	5
Prestressed concrete structures (%)	3
Reinforced concrete structures (%)	4
Steel Concrete Modules (%)	4
Piping systems(1)	3
Full cable trays & related supports (%)	7(2)
Empty cable trays and related supports (%)	5
Full conduits & related supports (%)	5
Empty conduits & related supports (%)	3
HVAC pocket lock ductwork (%)	7
HVAC companion angle ductwork (%)	5
HVAC welded ductwork (%)	3
Cabinets and panels for electrical equipment (%)	2
Equipment such as welded instrument racks and tanks (impulsive mode)(%)	2(3)
Motors, fans, housings, pressure vessels, heat exchangers, pumps, valve bodies (%)	2

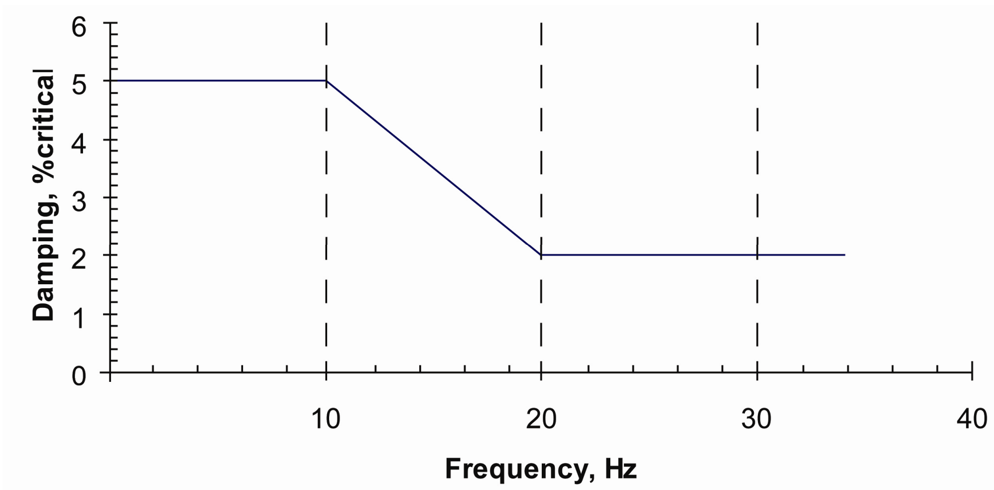
Notes for Tables 3.7.3-1(a) and 3.7.3-1(b):

1. As an alternative for response spectrum analyses using an envelope of the SSE or OBE response spectra at all support points (uniform support motion), frequency-dependent damping values shown in the graph below may be used, subject to the following restrictions:

   - Frequency-dependent damping should be used completely and consistently, if at all. Damping values for equipment other than piping are to be consistent with the values in the above table and RG 1.61 (Reference 3.7-15).

   - Use of the specified damping values is limited only to response spectral analyses. Acceptance of the use of the specified damping values with other types of dynamic analyses (e.g., time-history analyses or independent support motion method) requires further justification.

- When used for reconciliation or support optimization of existing designs, the effects of increased motion on existing clearances and online mounted equipment should be checked.
- Frequency-dependent damping is not appropriate for analyzing the dynamic response of piping systems using supports designed to dissipate energy by yielding.
- Frequency-dependent damping is not applicable to piping in which stress corrosion cracking has occurred, unless a case-specific evaluation is provided and reviewed, and found acceptable by the NRC staff.



2. The use of higher damping values for cable trays with flexible support systems (e.g., rod-hung trapeze systems, strut-hung trapeze systems, and strut-type cantilever and braced cantilever support systems) is permissible, subject to obtaining NRC review for acceptance on a case-by-case basis.
3. Use 0.5% damping for sloshing mode for tanks
4. Refer to Table 3.8.3-4 for appropriate damping values of the containment internal structure

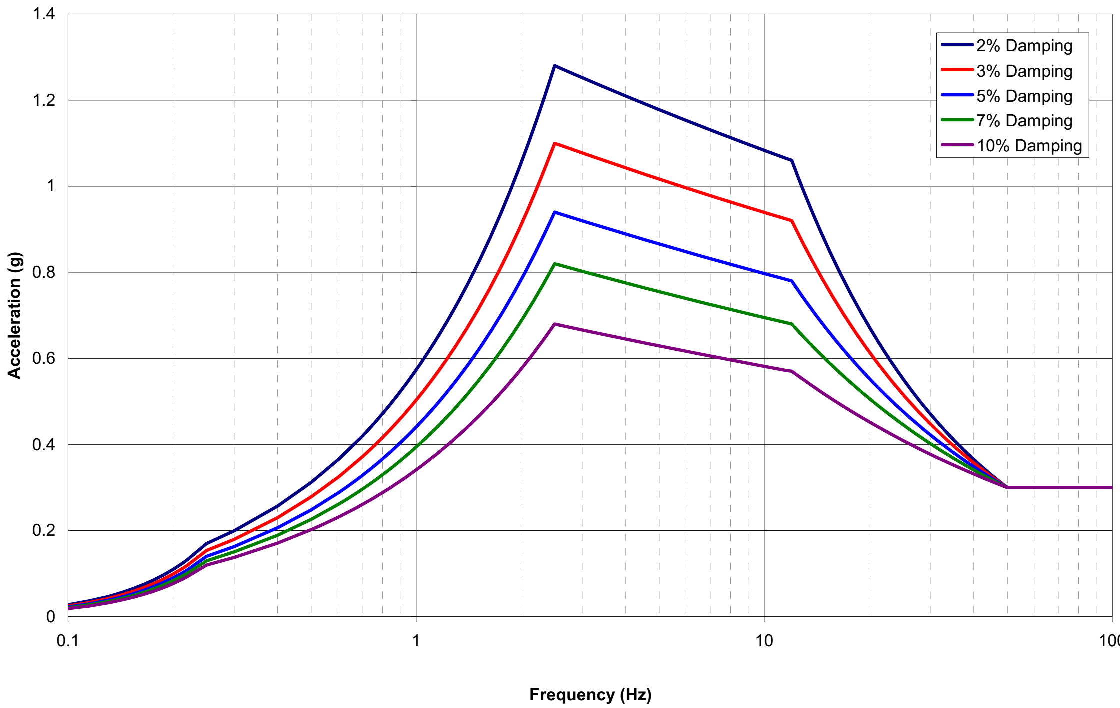


Figure 3.7.1-1 US–APWR Horizontal CSDRS

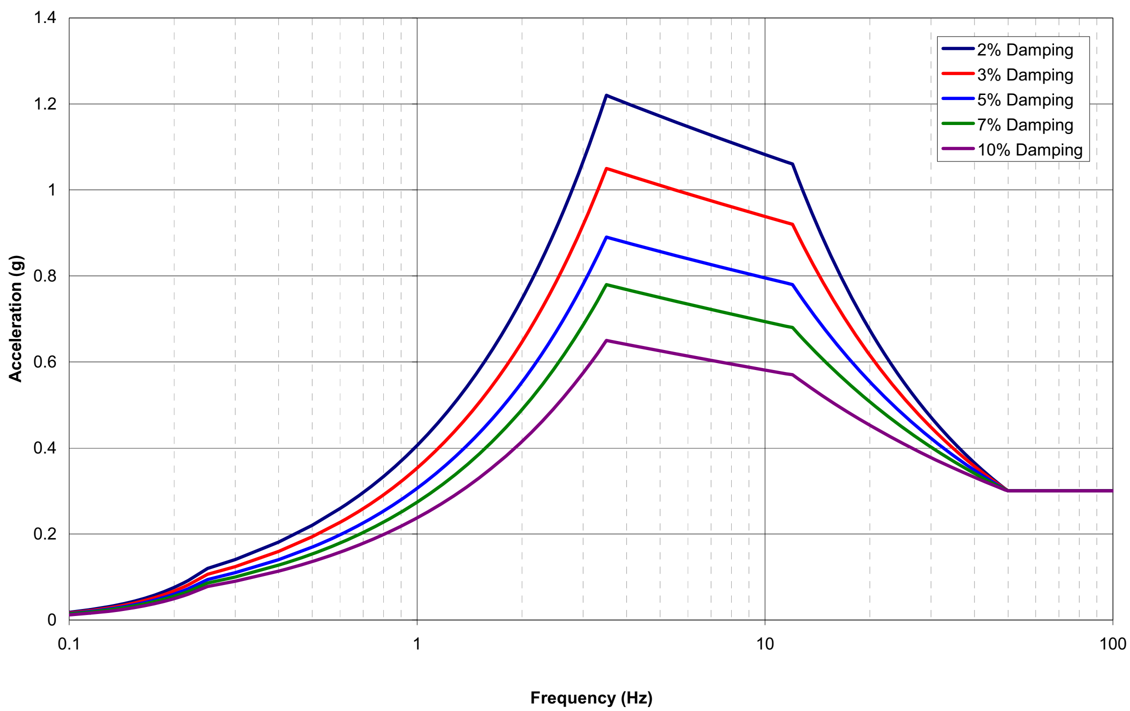


Figure 3.7.1-2 US–APWR Vertical CSDRS

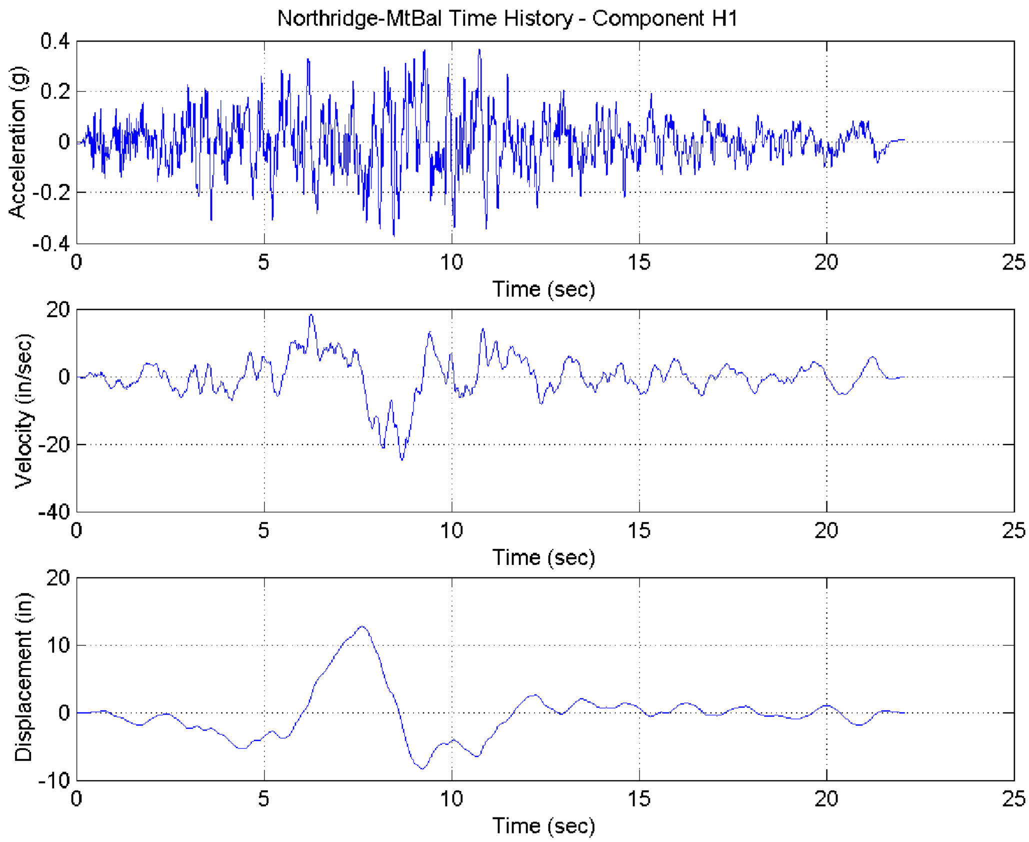


Figure 3.7.1-3 Acceleration, Velocity, and Displacement Time History for Component H1 [NS]

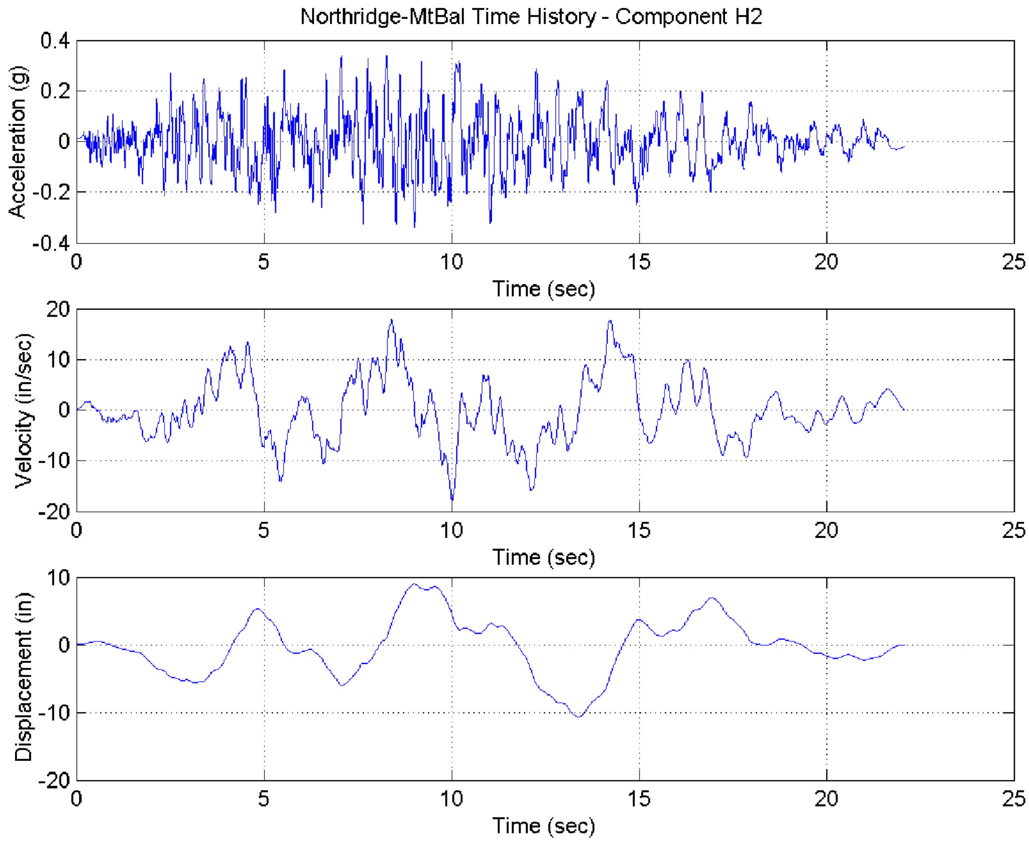


Figure 3.7.1-4 Acceleration, Velocity, and Displacement Time History for Component H2 [EW]

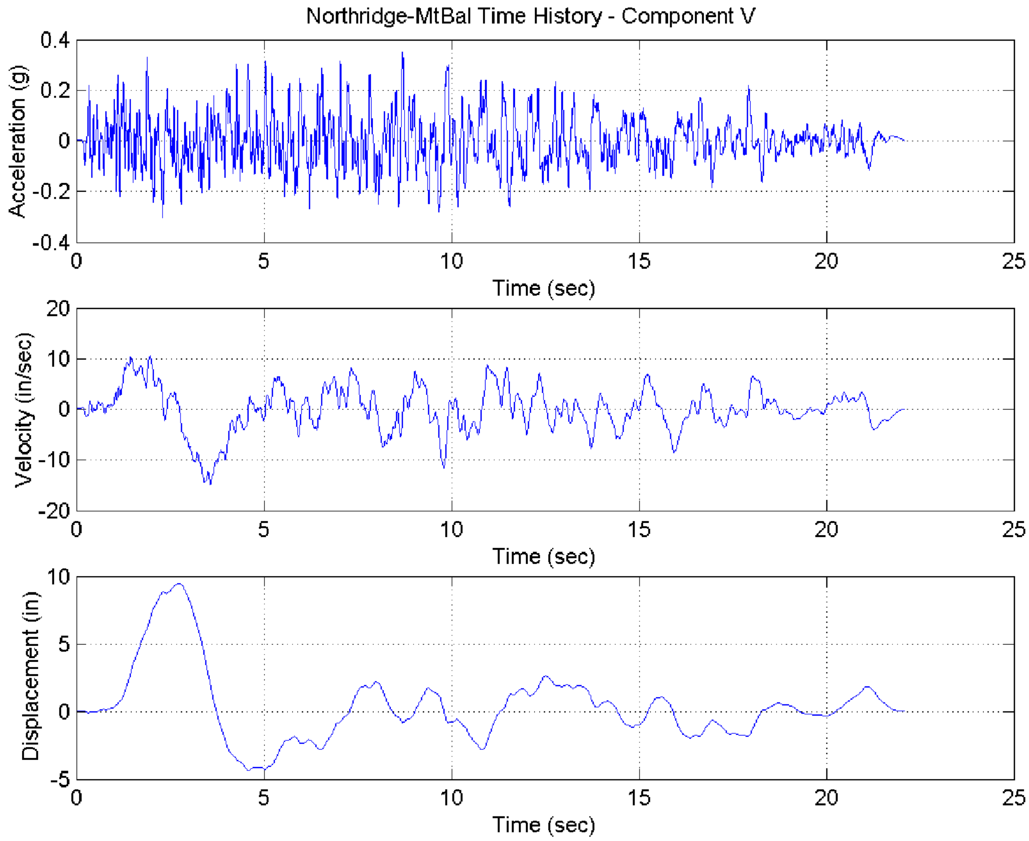


Figure 3.7.1-5 Acceleration, Velocity, and Displacement Time History for Component V

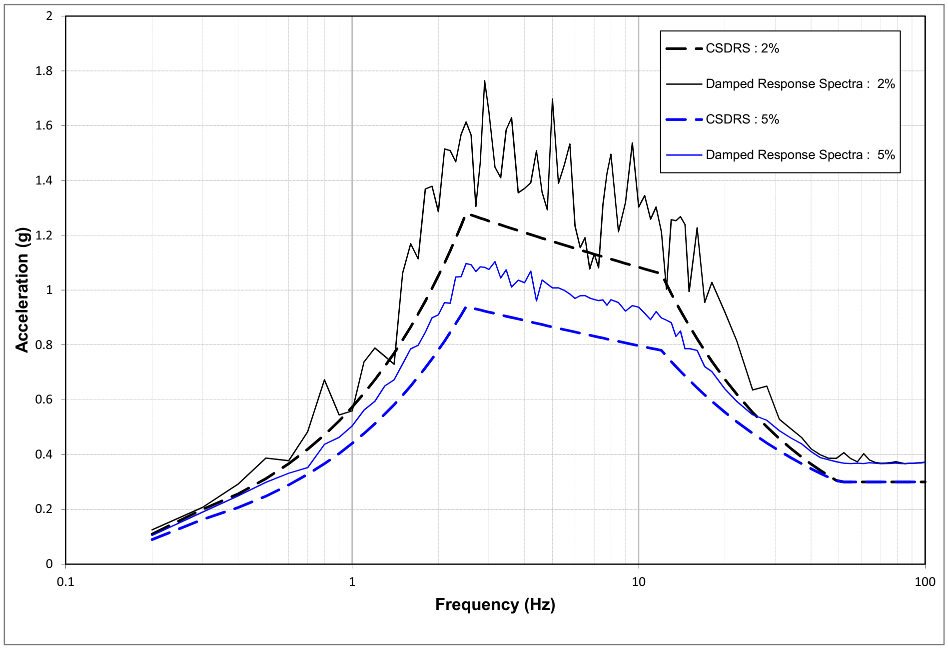


Figure 3.7.1-6a Damped Response Spectra Plots for Northridge Mount Baldy Component H1 (180) [NS]

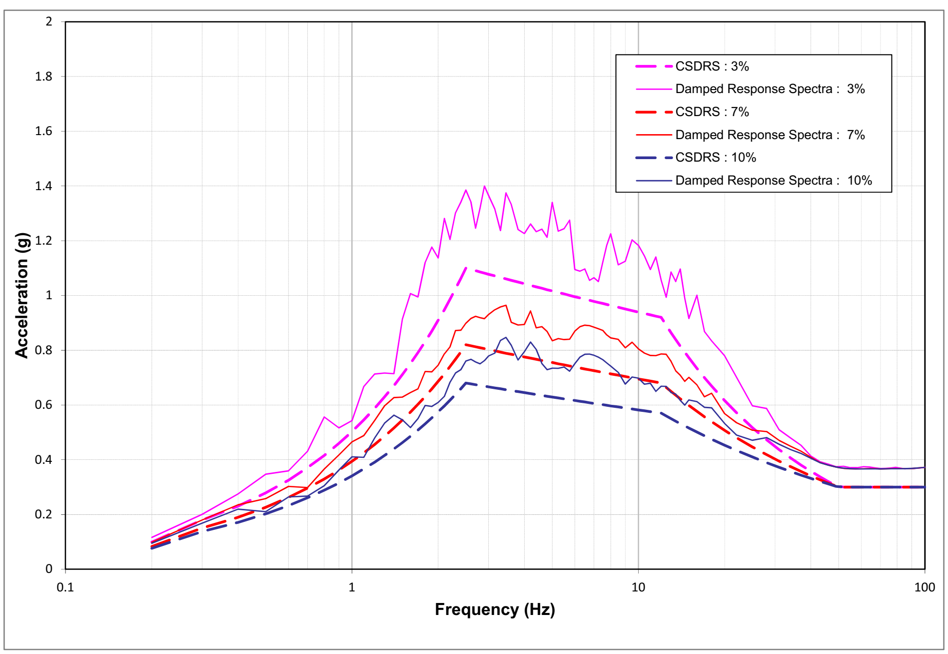


Figure 3.7.1-6b Damped Response Spectra Plots for Northridge Mount Baldy Component H1 (180) [NS]

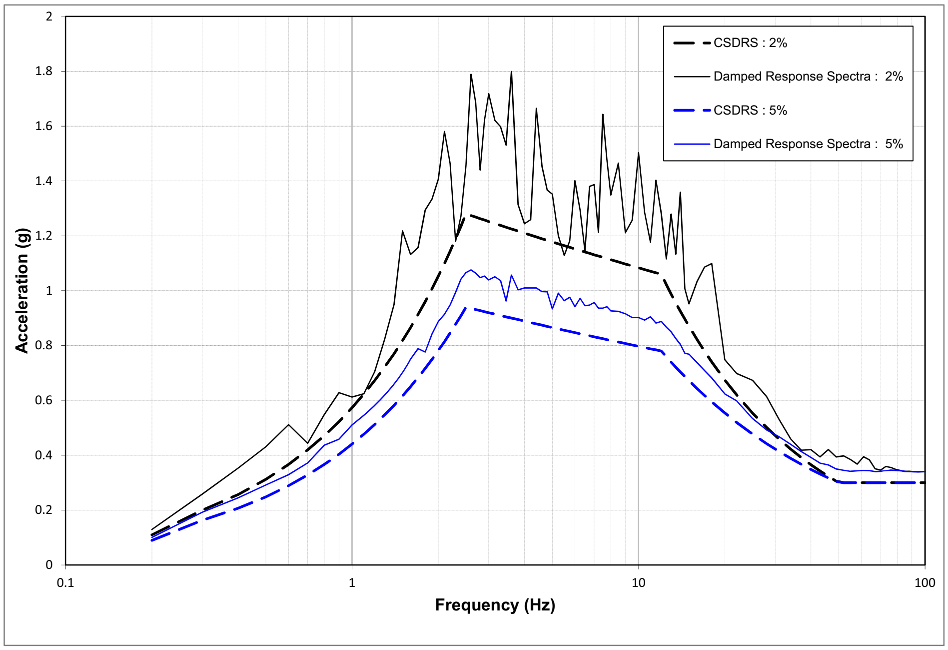


Figure 3.7.1-7a Damped Response Spectra Plots for Northridge Mount Baldy Component H2 (090) [EW]

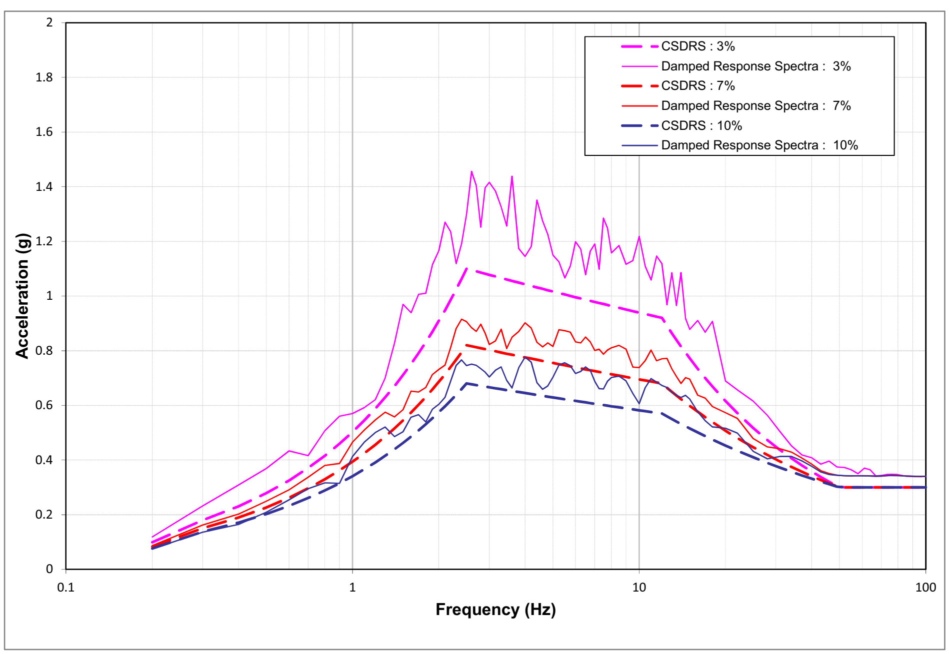


Figure 3.7.1-7b Damped Response Spectra Plots for Northridge Mount Baldy Component H2 (090) [EW]

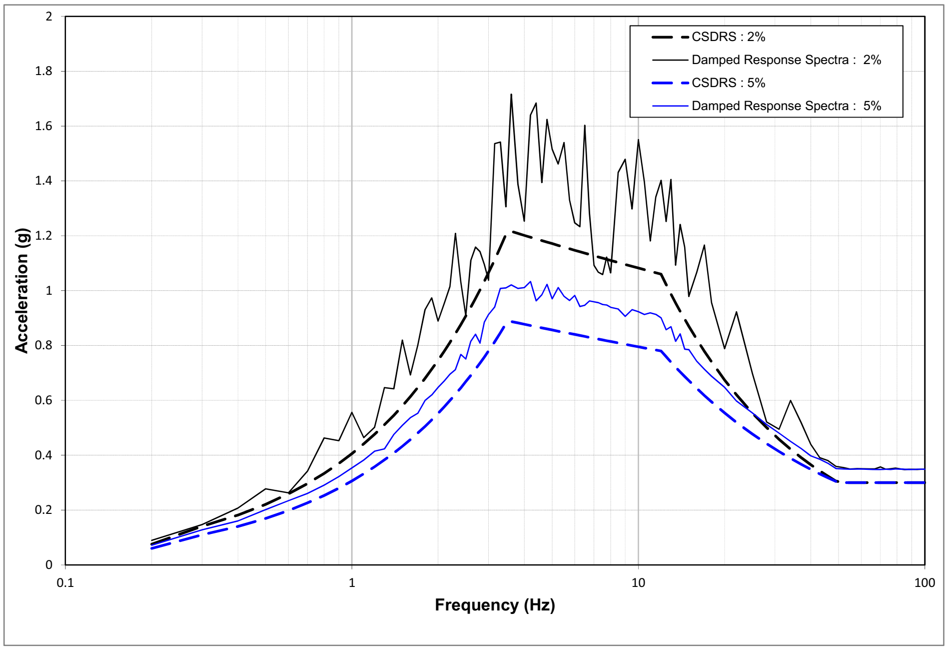


Figure 3.7.1-8a Damped Response Spectra Plots for Mount Baldy Component V (UP)

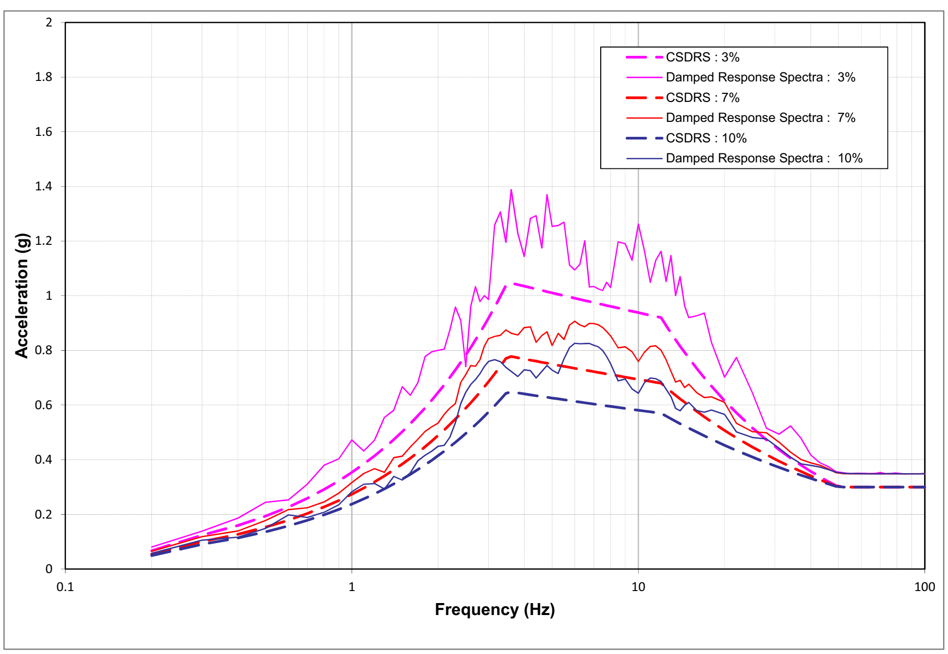
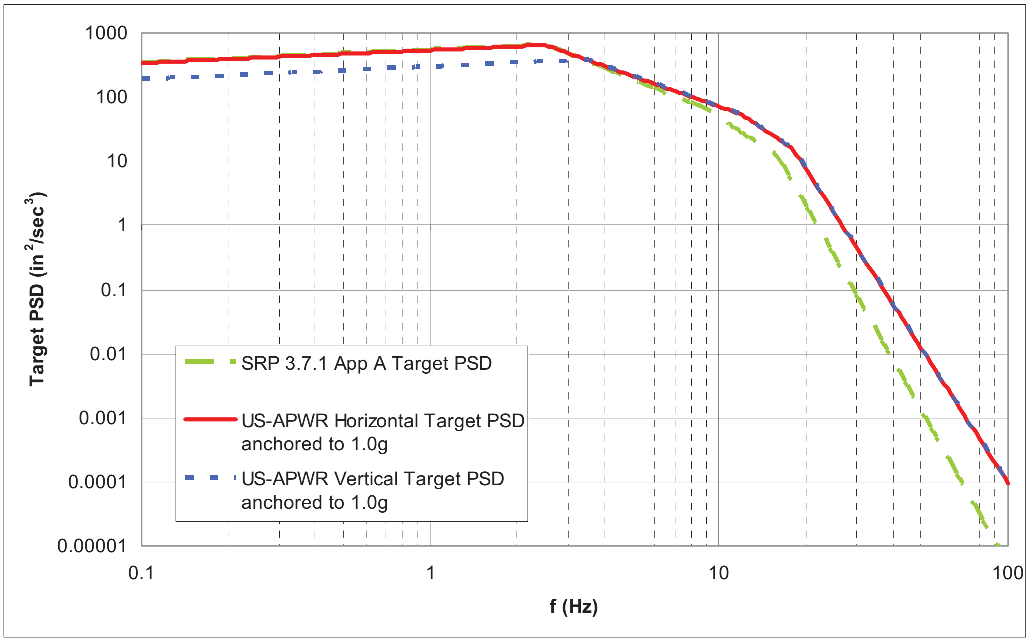


Figure 3.7.1-8b Damped Response Spectra Plots for Mount Baldy Component V (UP)



The final horizontal target PSD anchored to 1.0g is as follows:

$$S_{0H}(f) = \begin{cases} 650(f/2.5)^{0.2} \text{ for } f < 2.5\text{Hz} \\ 650(2.5/f)^{1.6} \text{ for } 2.5\text{Hz} \le f < 12\text{Hz} \\ 52.9(12.0/f)^{3} \text{ for } 12\text{Hz} \le f < 18\text{Hz} \\ 15.7(18/f)^{7} \text{ for } 18\text{Hz} \le f \end{cases} \quad (\text{in}^2/\text{sec}^3)$$

The final vertical target PSD anchored to 1.0g is as follows:

$$S_{0V}(f) = \begin{cases} 380(f/3.5)^{0.2} \text{ for } f < 3.5\text{Hz} \\ 380(3.5/f)^{1.6} \text{ for } 3.5\text{Hz} \le f < 12\text{Hz} \\ 52.9(12.0/f)^{3} \text{ for } 12\text{Hz} \le f < 18\text{Hz} \\ 15.7(18/f)^{7} \text{ for } 18\text{Hz} \le f \end{cases} \quad (\text{in}^2/\text{sec}^3)$$

Figure 3.7.1-9 US-APWR Final Horizontal and Vertical Target PSDs

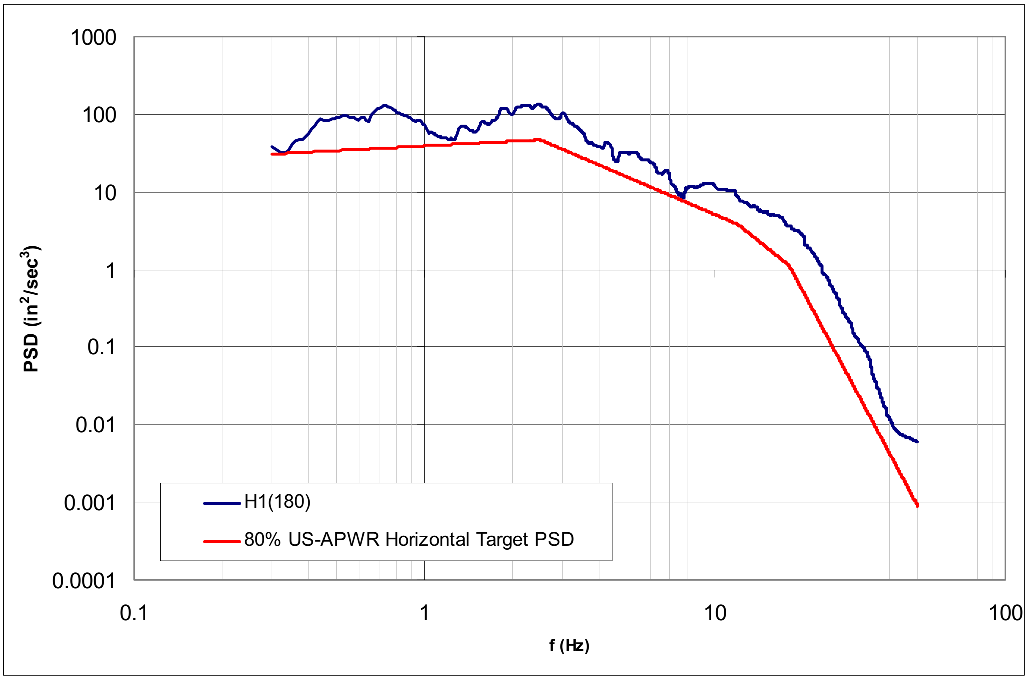


Figure 3.7.1-10 Smoothed Power Spectral Density Plots for Component H1 (180)

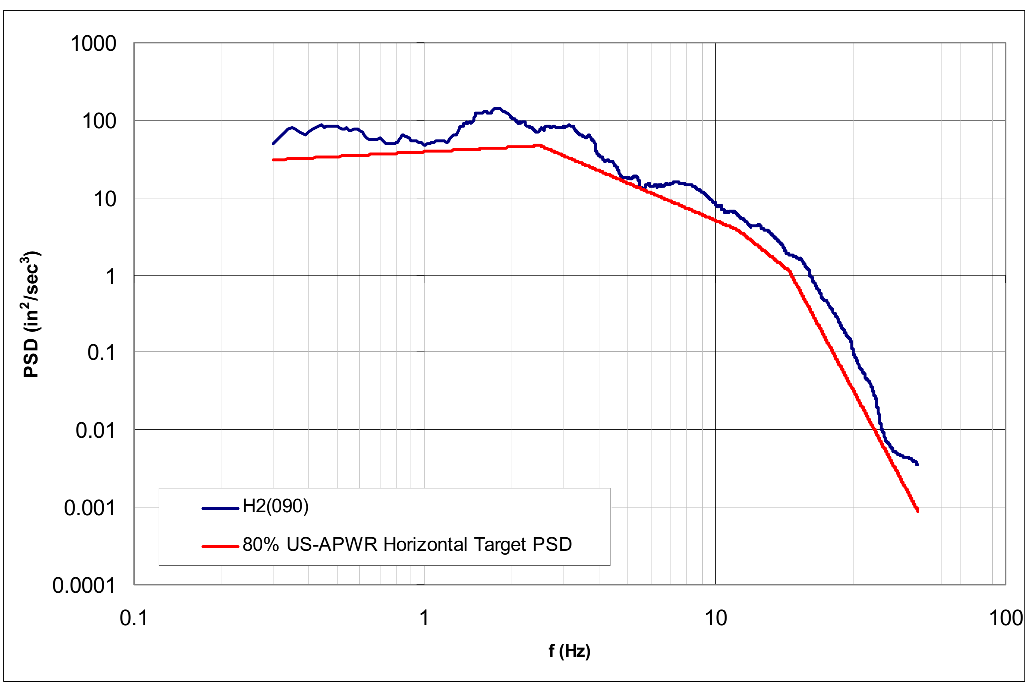


Figure 3.7.1-11 Smoothed Power Spectral Density Plots for Component H2 (090)

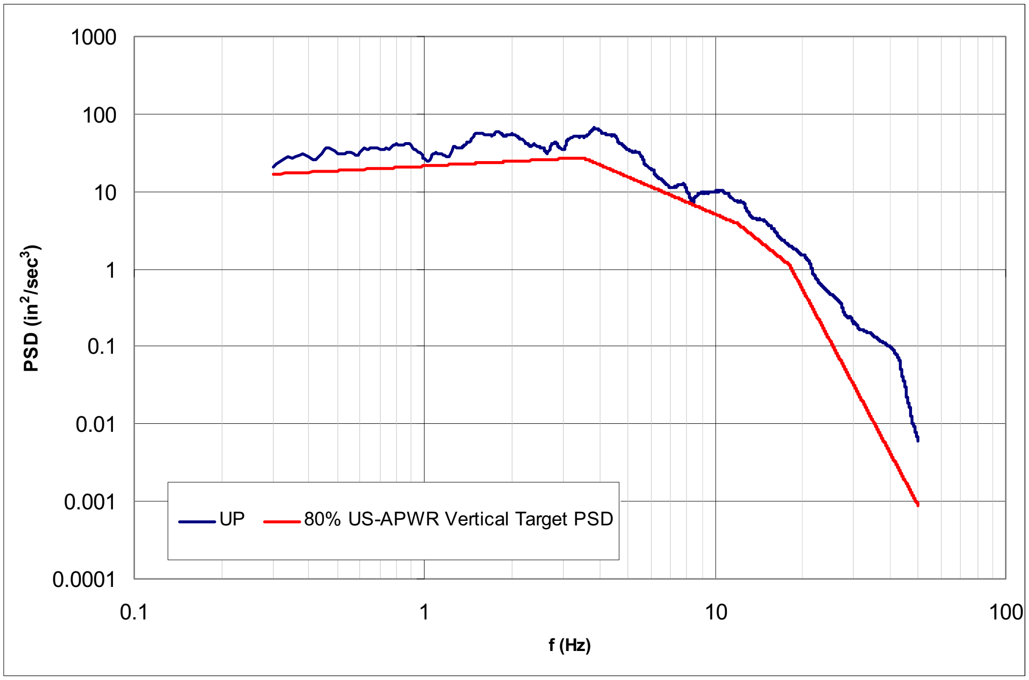


Figure 3.7.1-12 Smoothed Power Spectral Density Plots for Component V (UP)

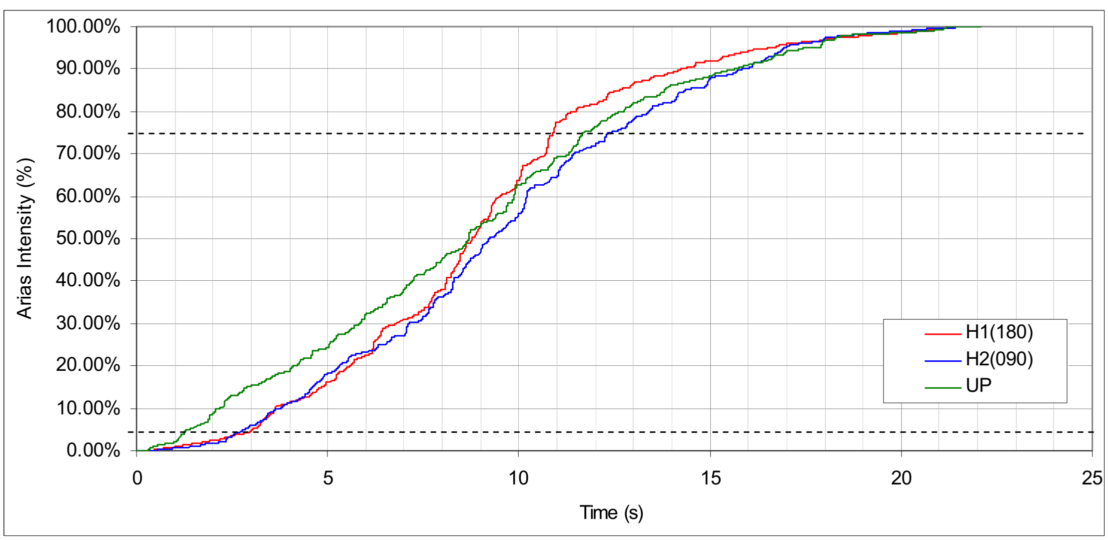


Figure 3.7.1-13 Arias Intensities of the Northridge – Mount Baldy Artificial Time History Components Showing 5%-75% Duration

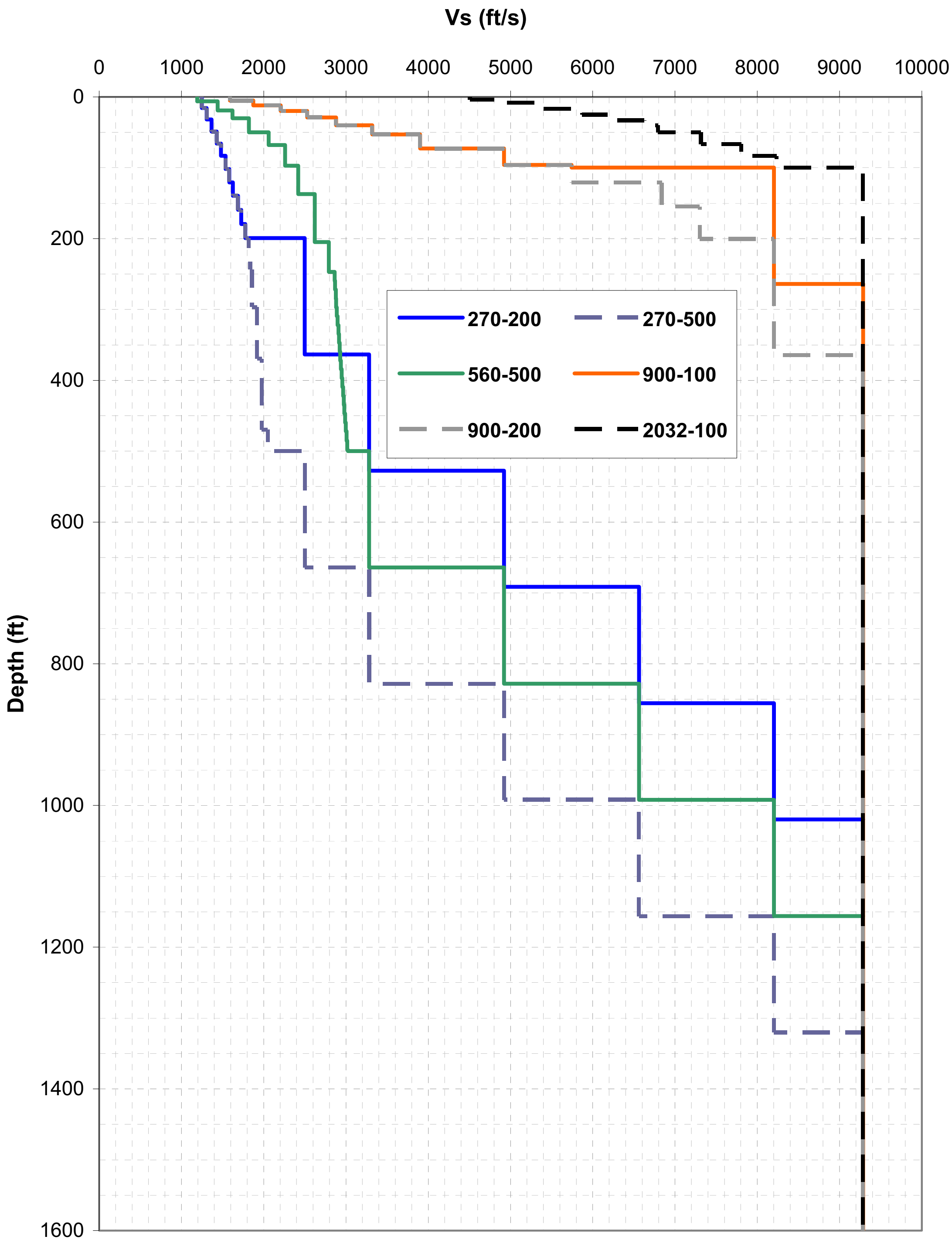


Figure 3.7.1-14 Six Generic Soil Profiles, Shear Wave Velocity $V_{(s)}$

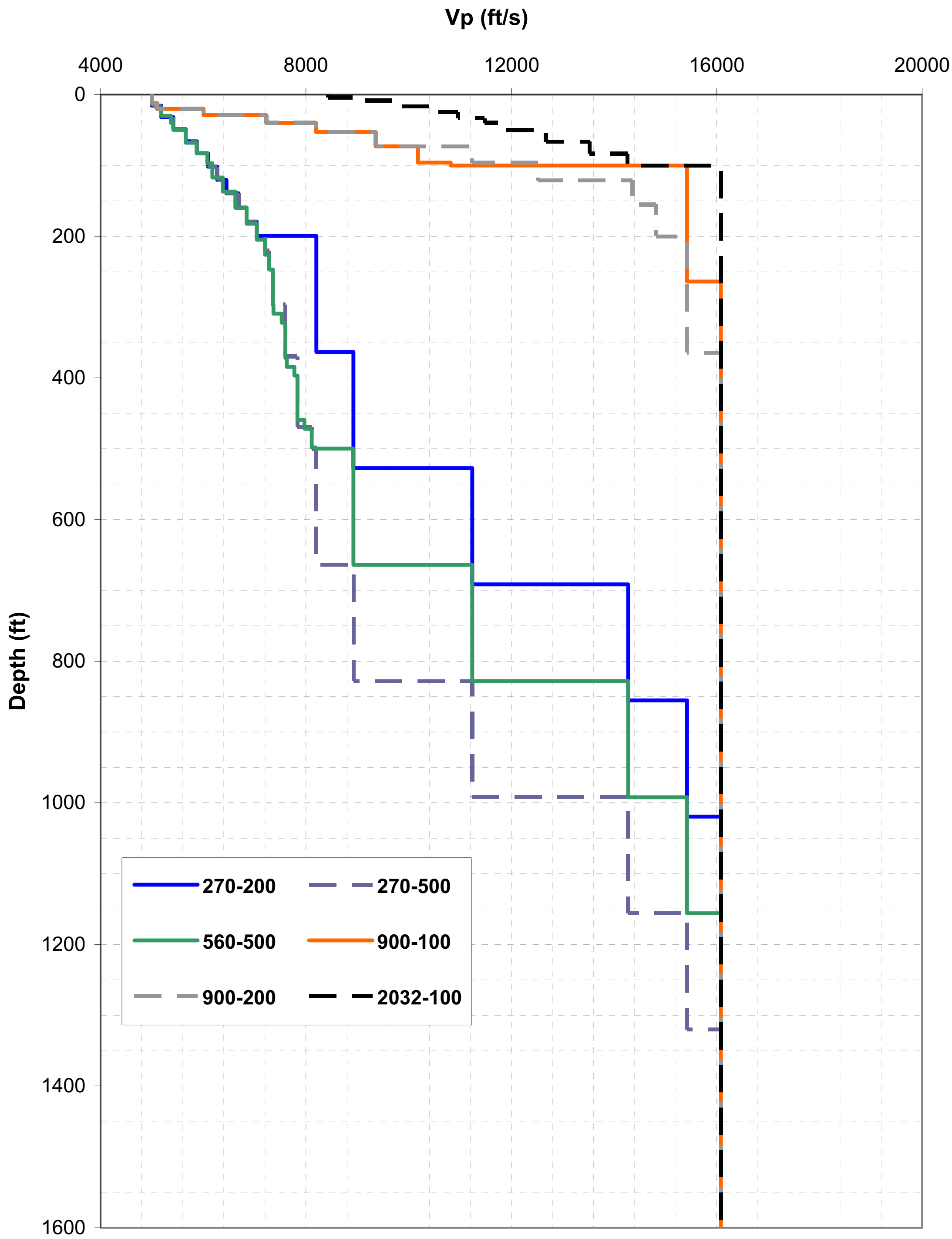


Figure 3.7.1-15 Six Generic Soil Profiles, Compression Wave Velocity (V_p)

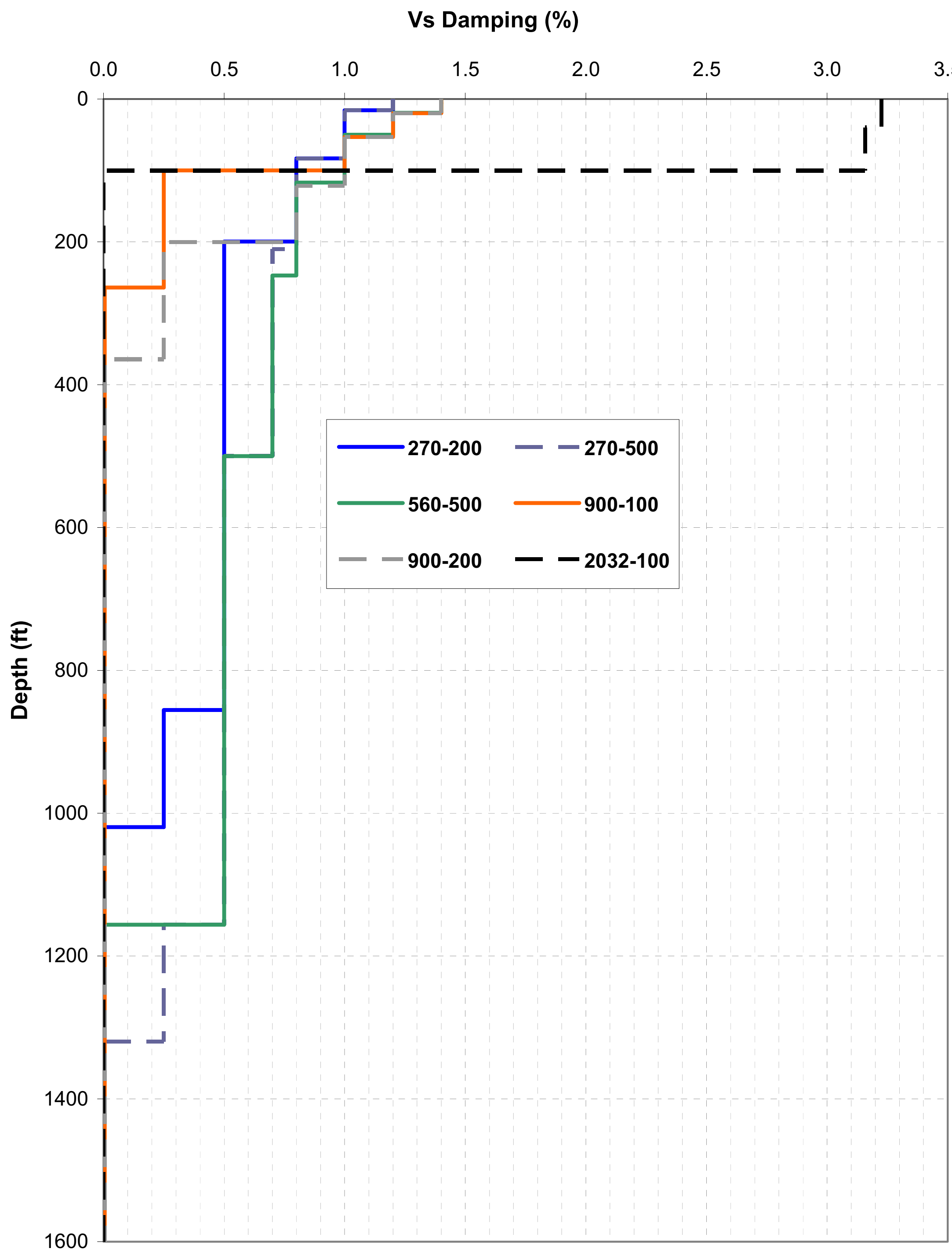


Figure 3.7.1-16 Six Generic Soil Profiles, Shear Wave Velocity Damping (%)

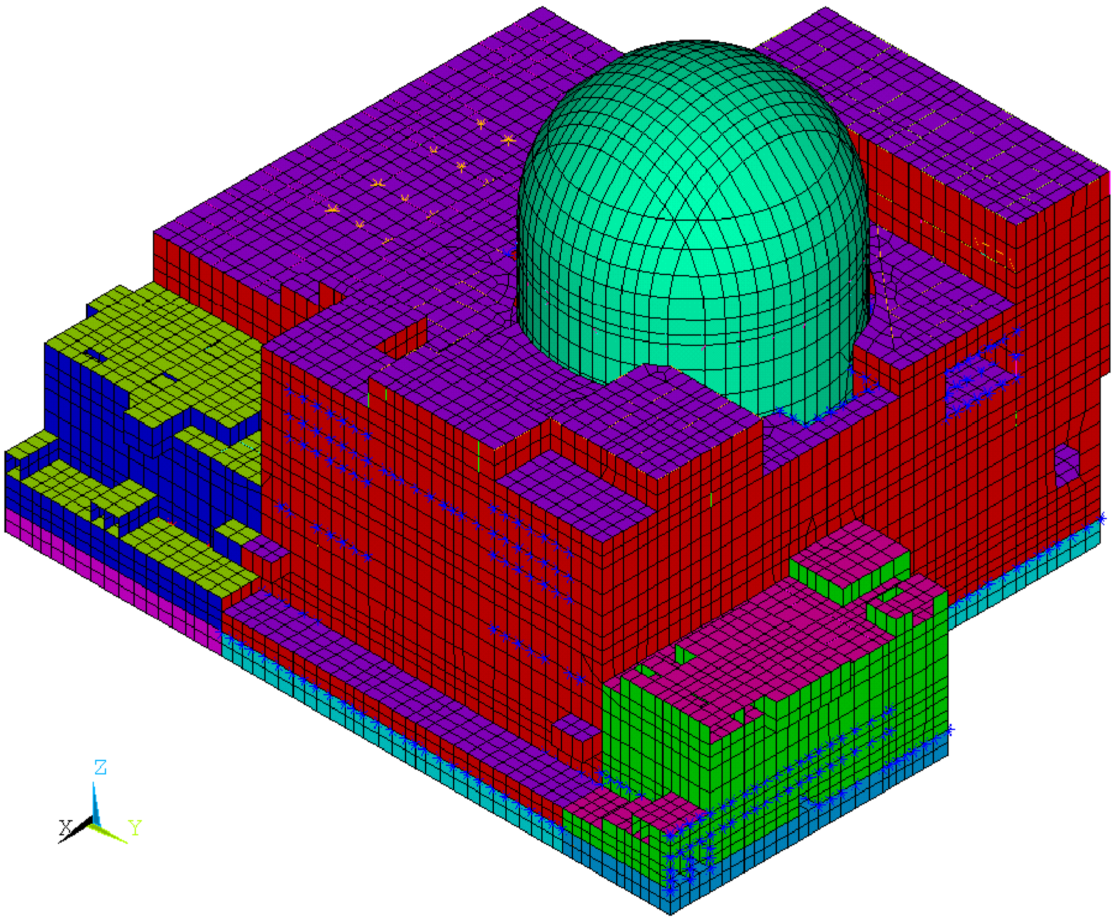


Figure 3.7.2-1 Integrated R/B Complex Dynamic FE Model

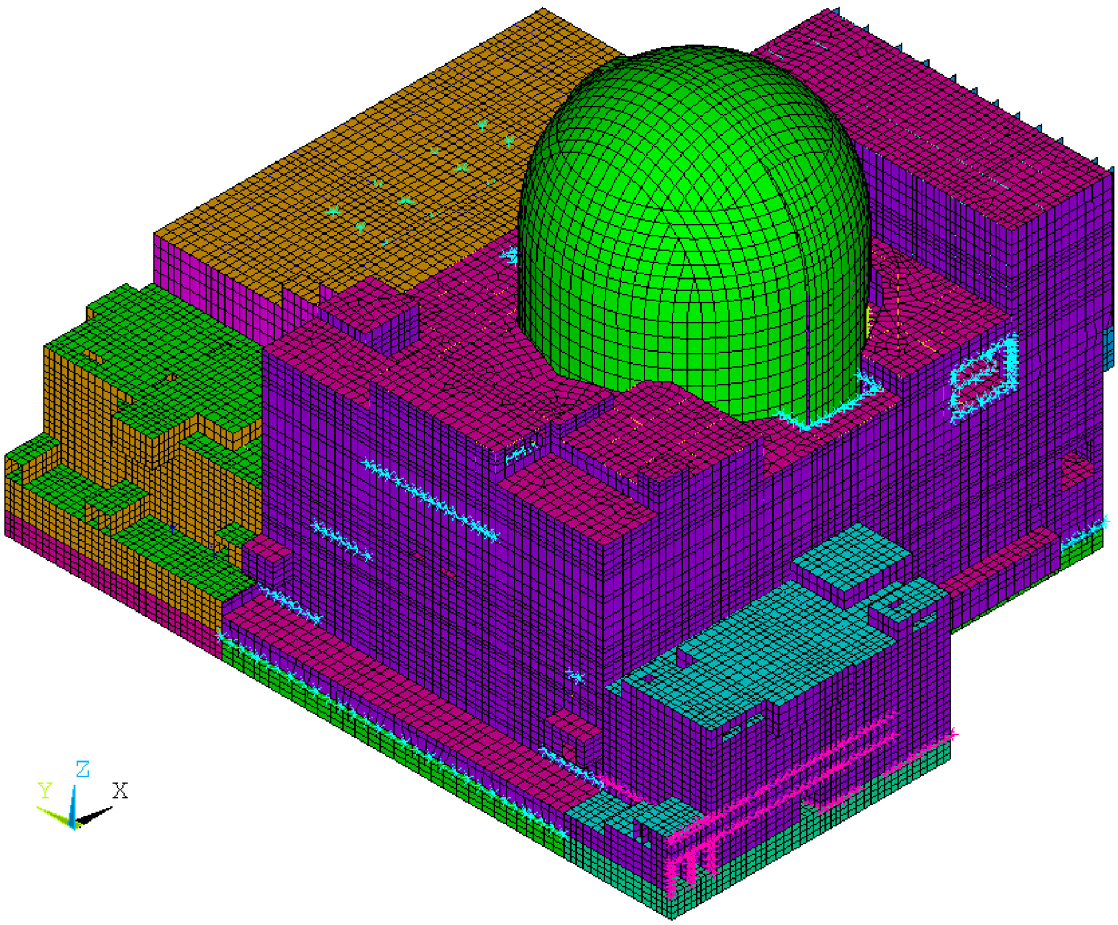


Figure 3.7.2-2 Integrated R/B Complex Detailed FE Model

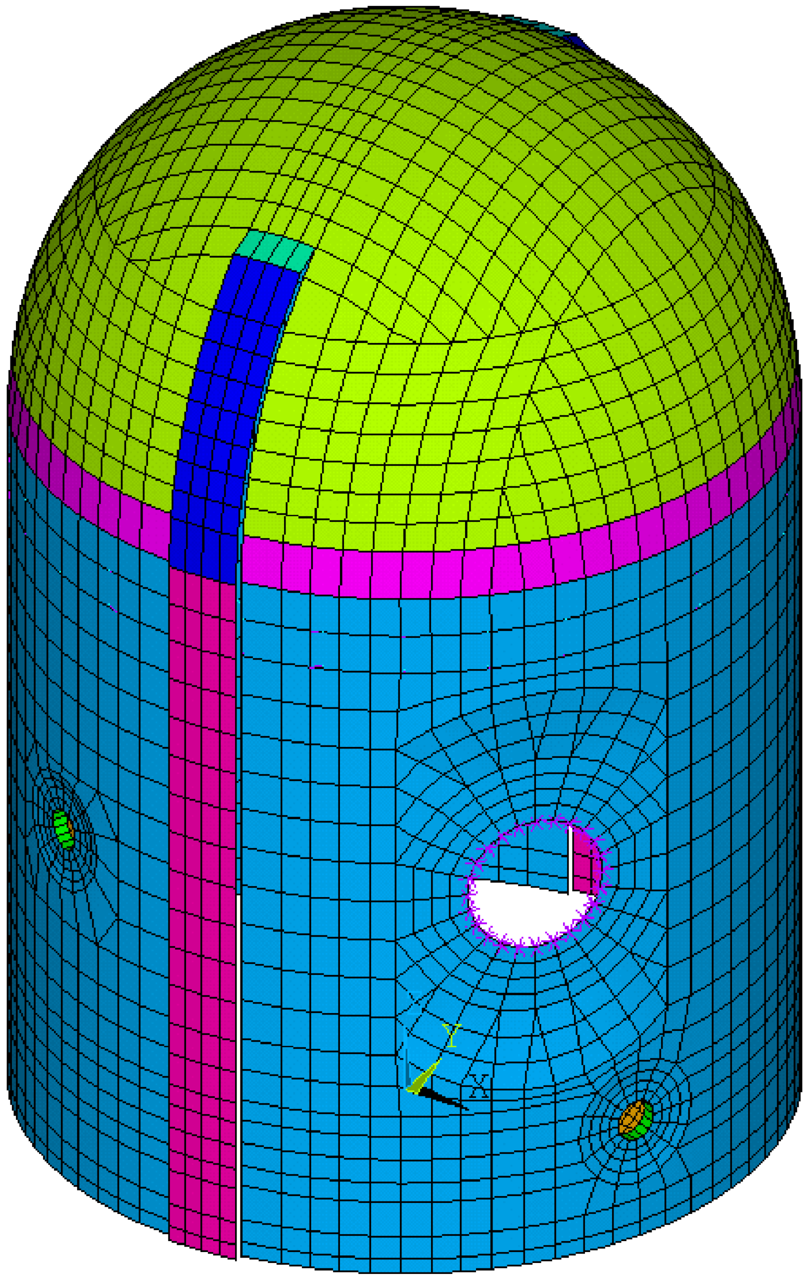


Figure 3.7.2-3 PCCV Detailed Model

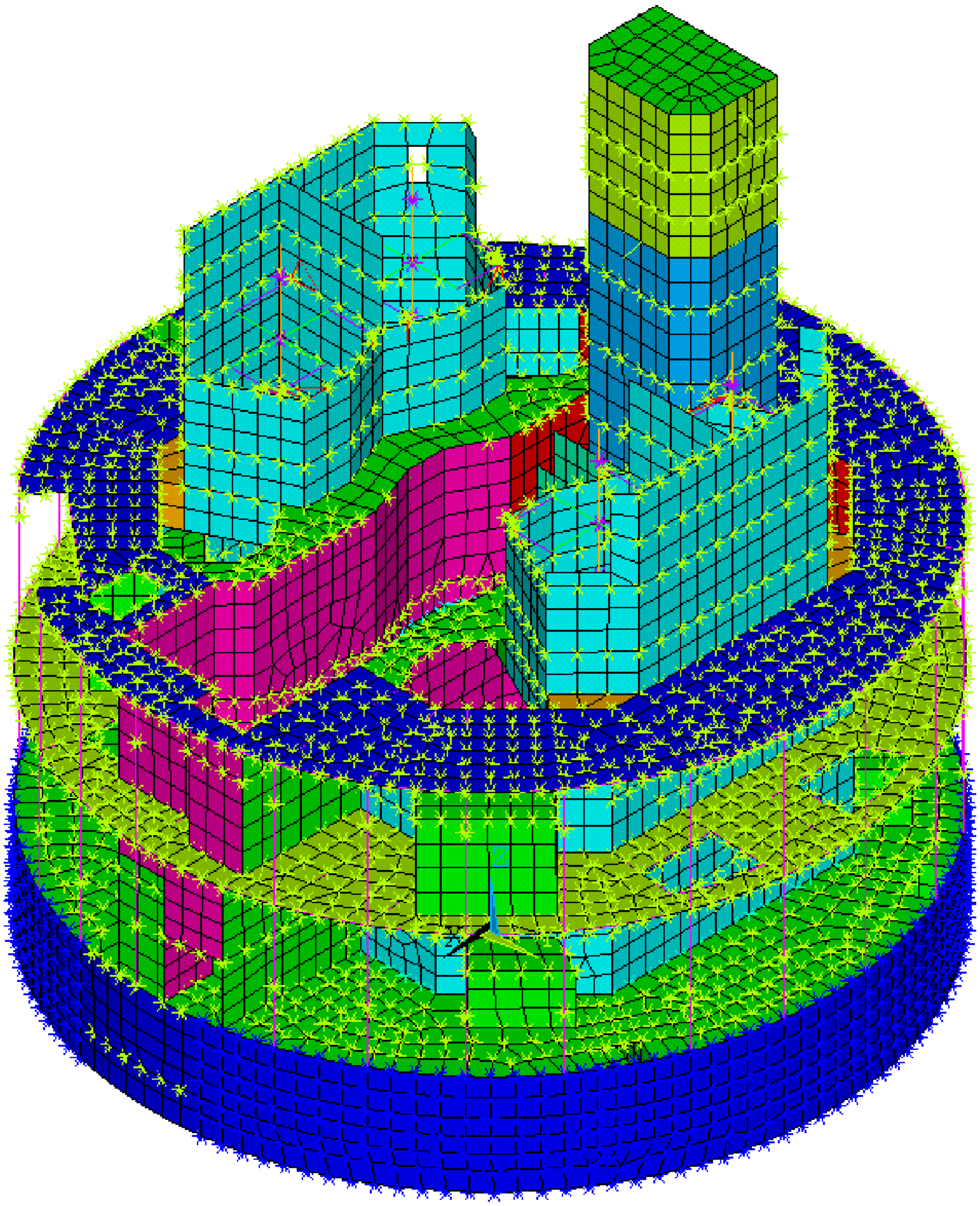


Figure 3.7.2-4 CIS Detailed Model
(includes the CIS and the RCL models)

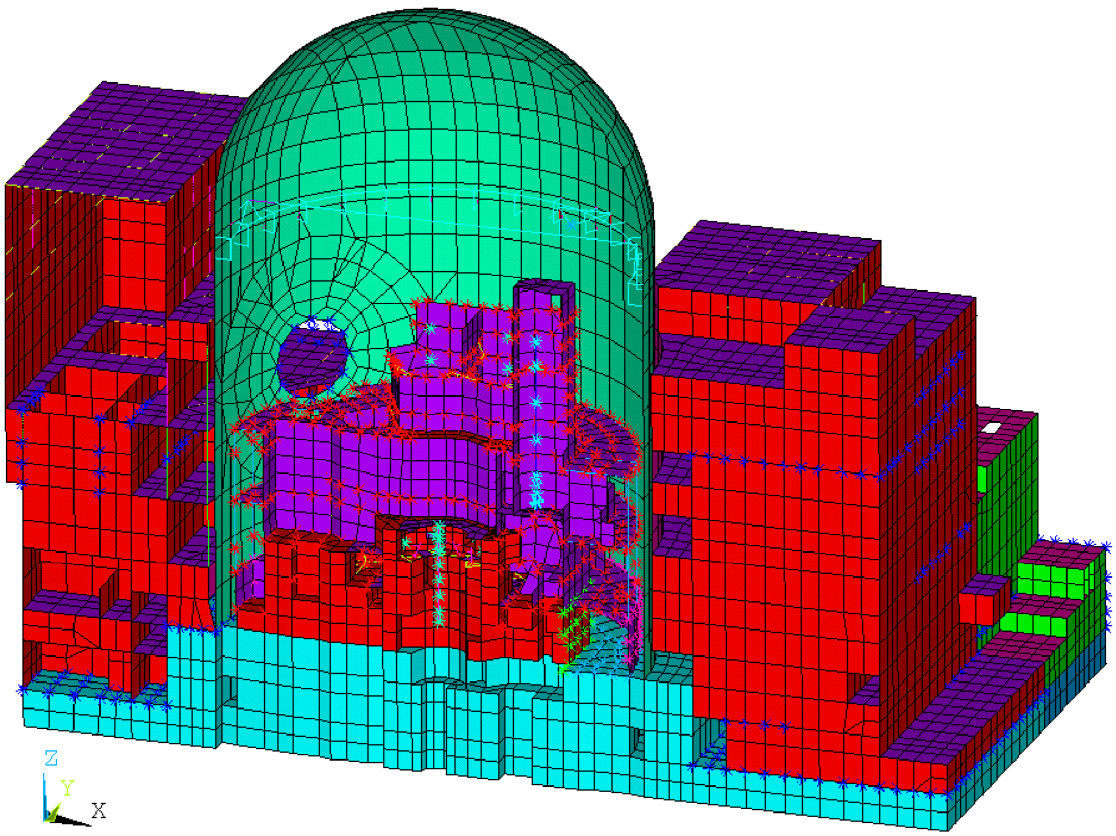


Figure 3.7.2-5 Section View of Dynamic R/B Complex FE Model Looking East

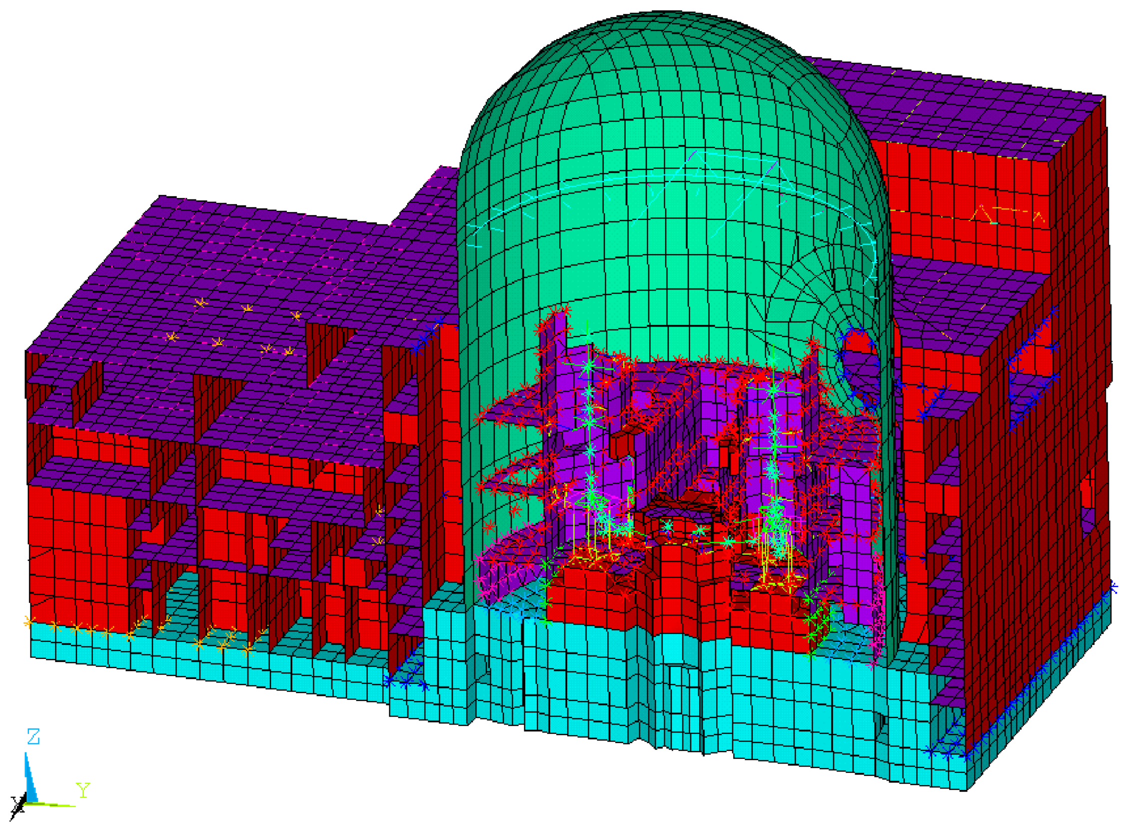


Figure 3.7.2-6 Section View of Dynamic R/B Complex FE Model Looking North

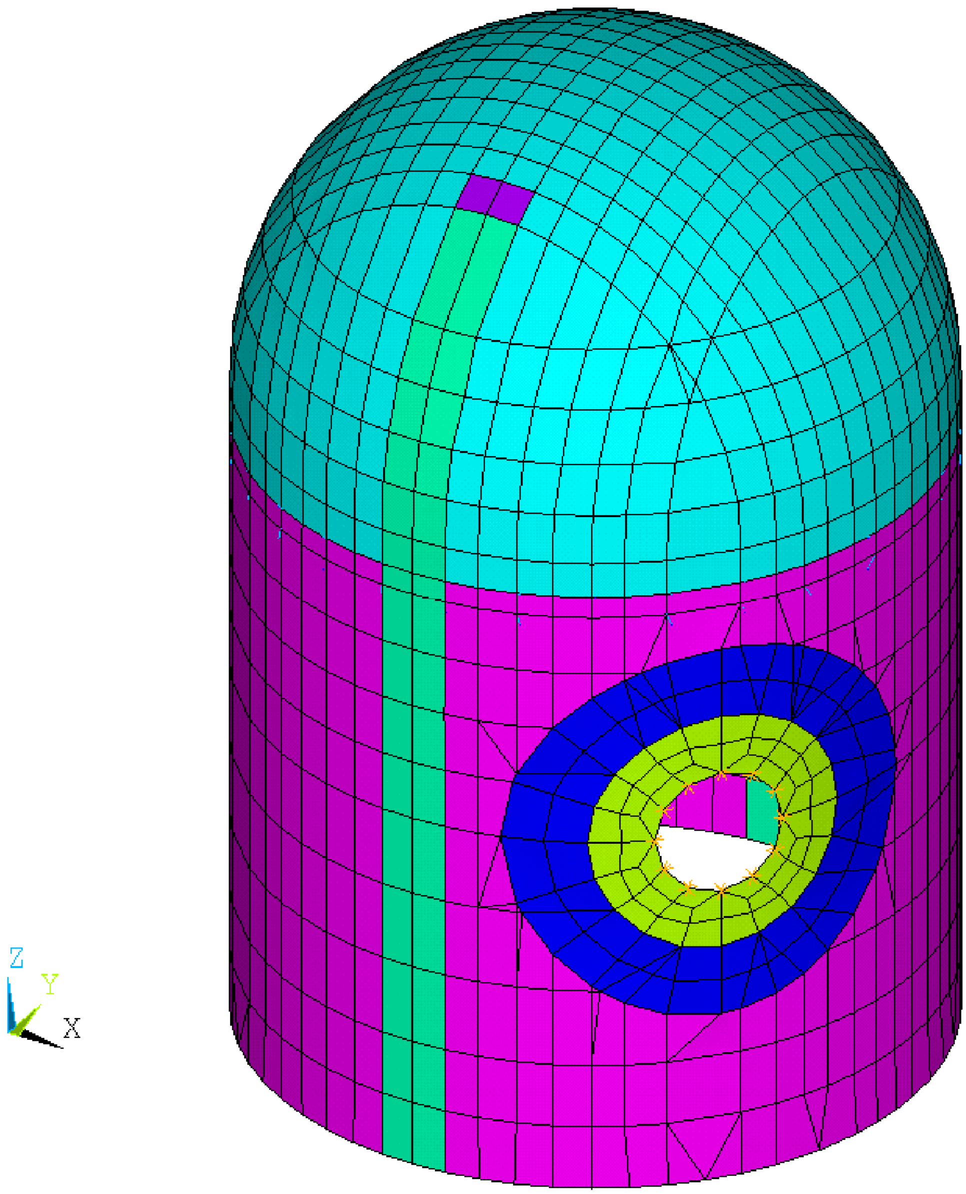


Figure 3.7.2-7 PCCV Dynamic Model

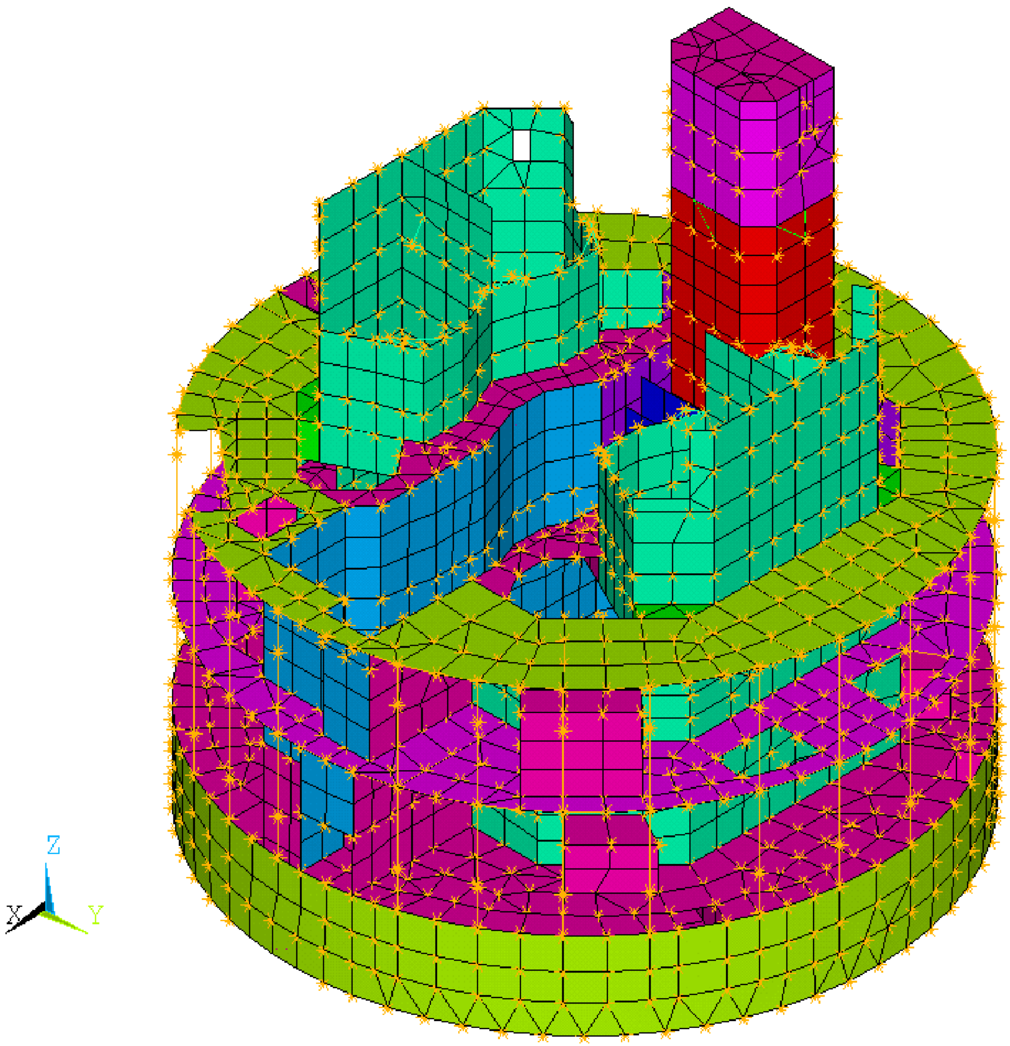


Figure 3.7.2-8 CIS Dynamic Model – Shell Elements (Excluding RCL)

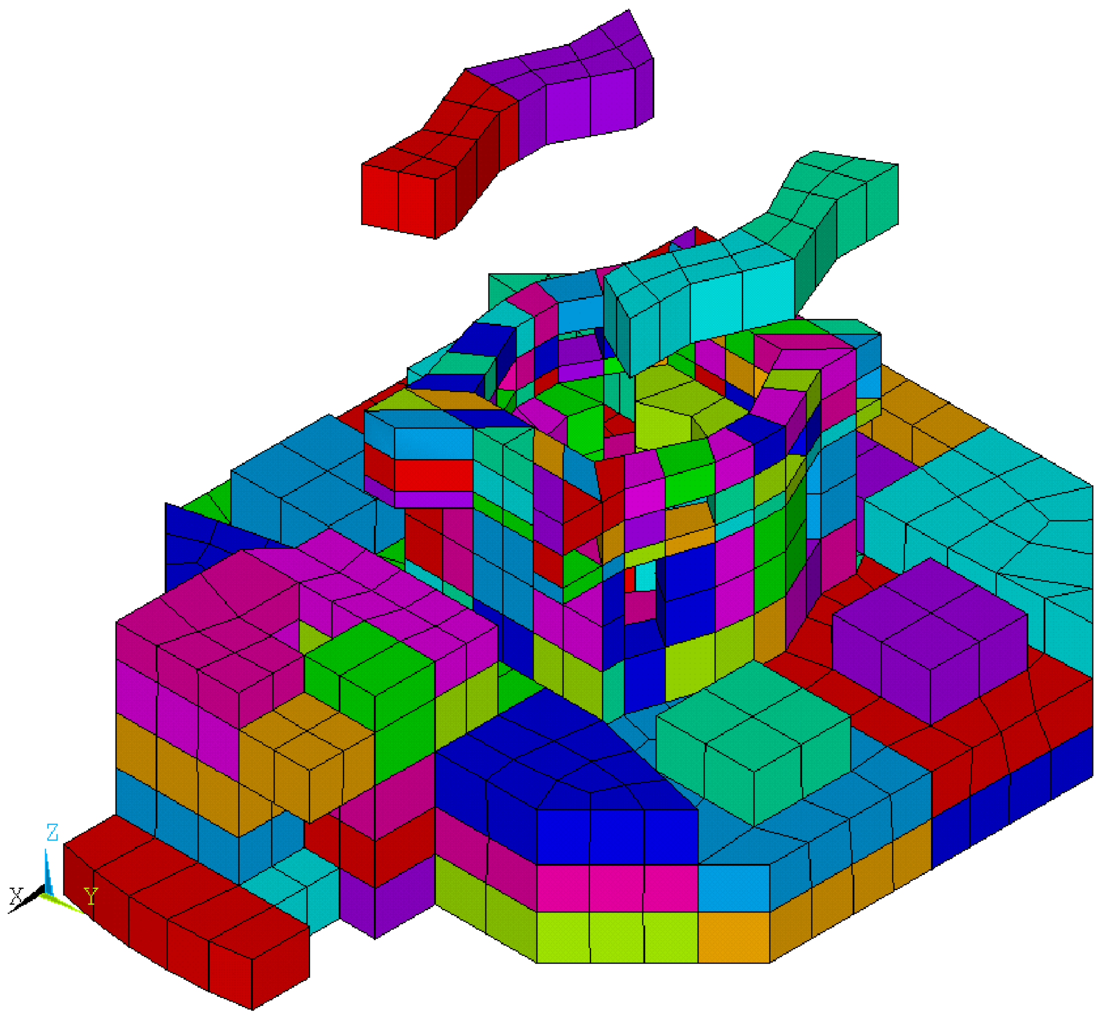


Figure 3.7.2-9 CIS Dynamic Model - Solid Elements

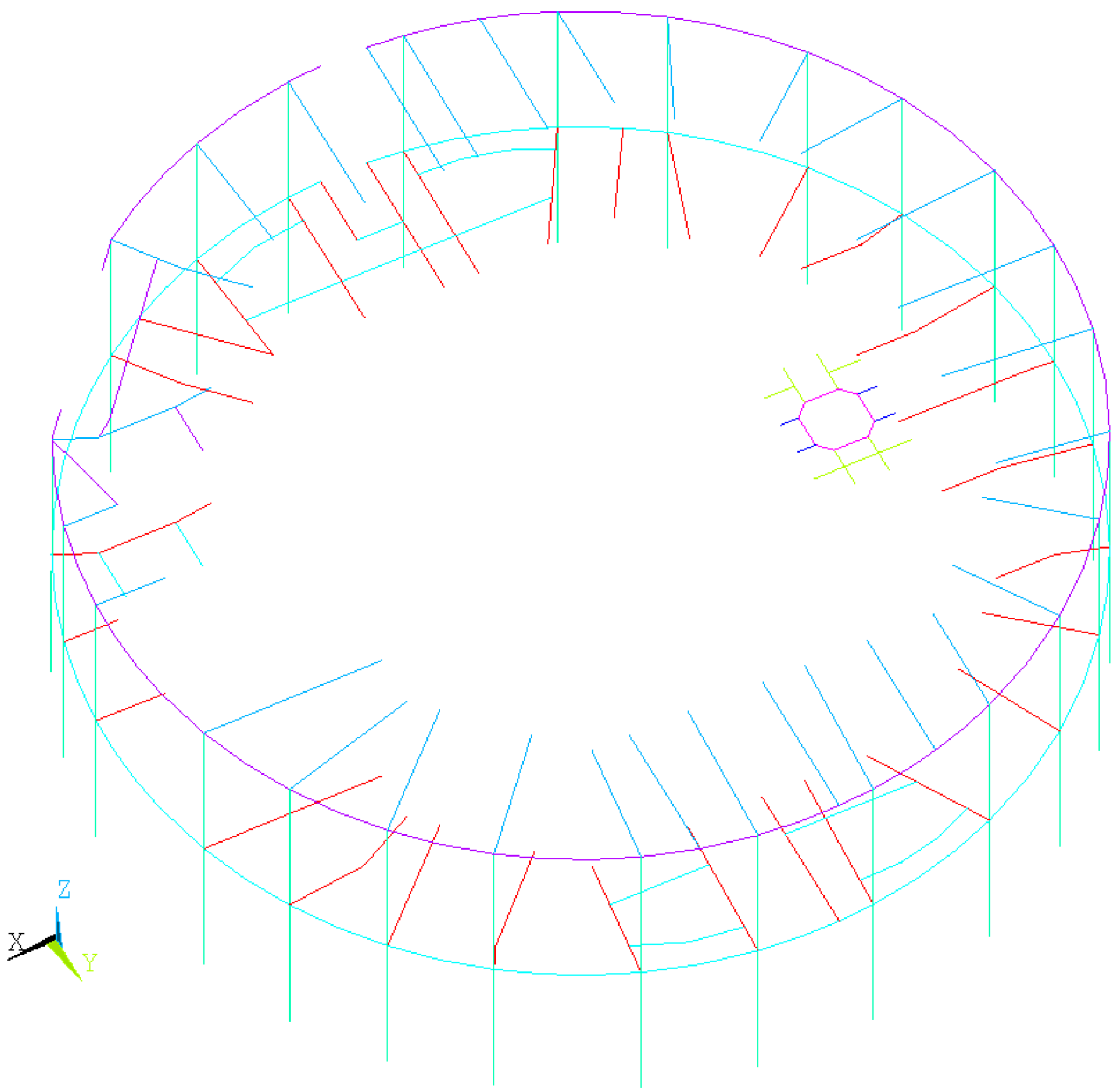


Figure 3.7.2-10 CIS Dynamic Model – Beam Elements (Excluding RCL)

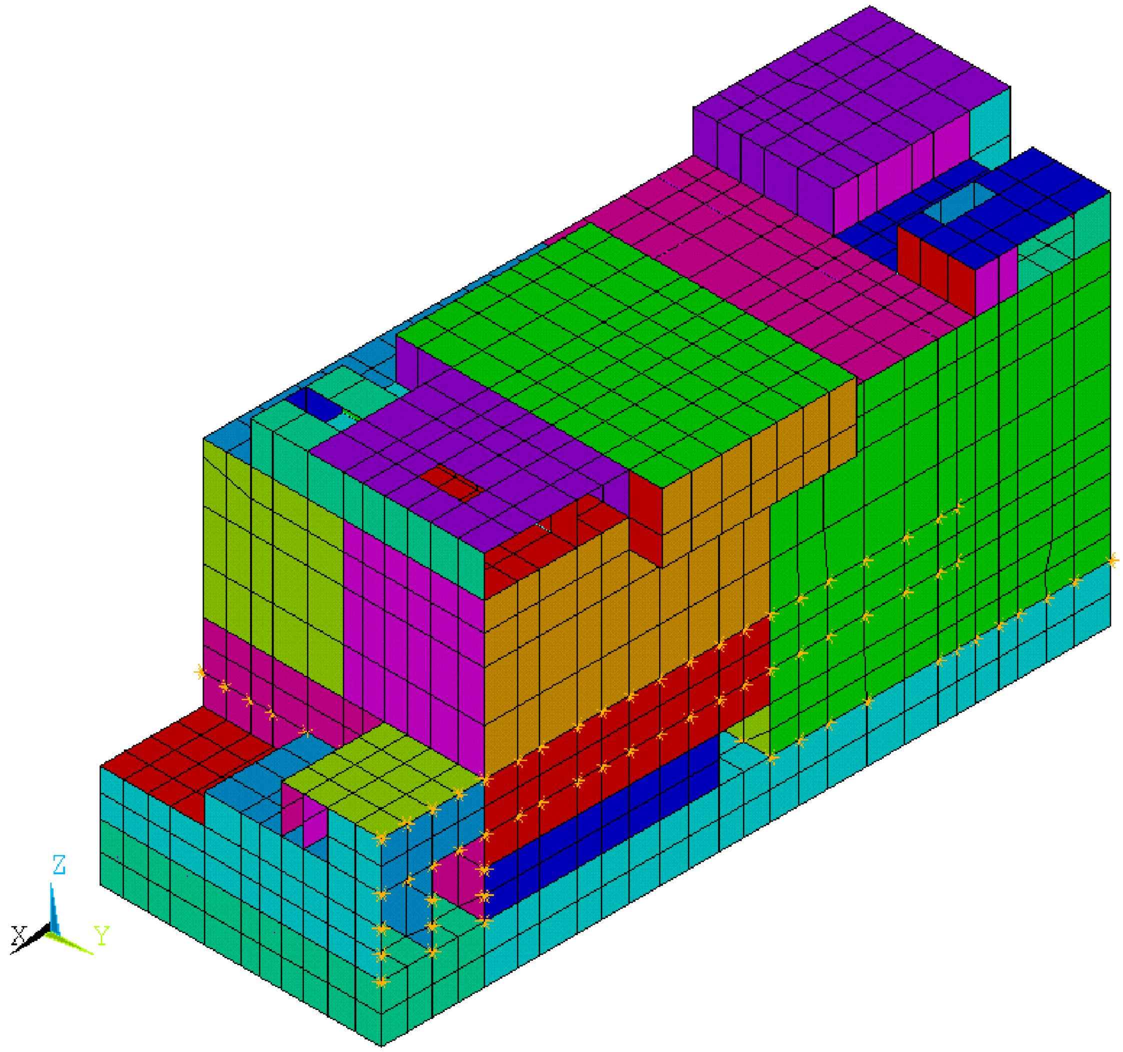


Figure 3.7.2-11 East PS/B Dynamic Model with ESWPC

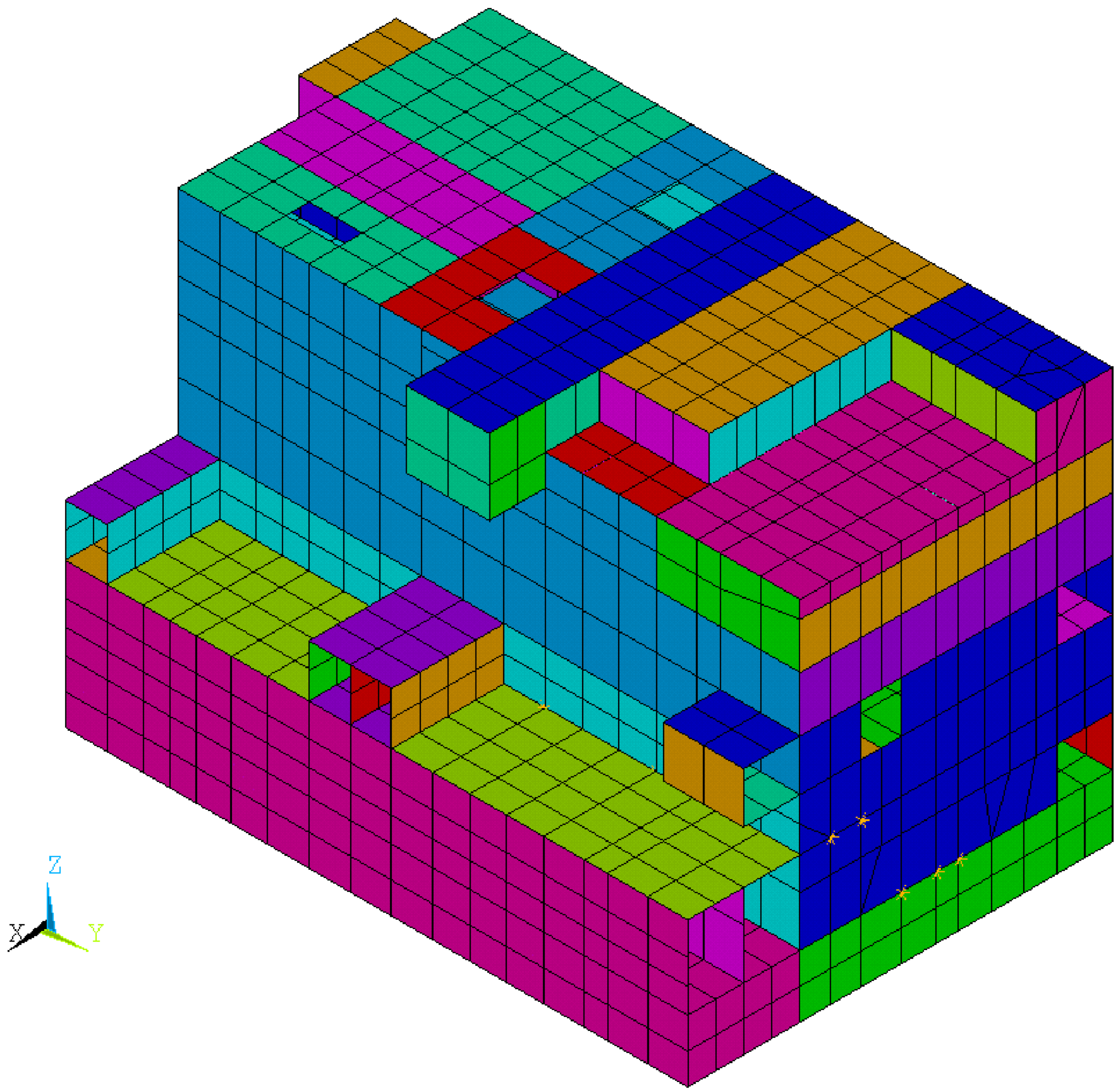


Figure 3.7.2-12 West PS/B Dynamic Model with ESWPC

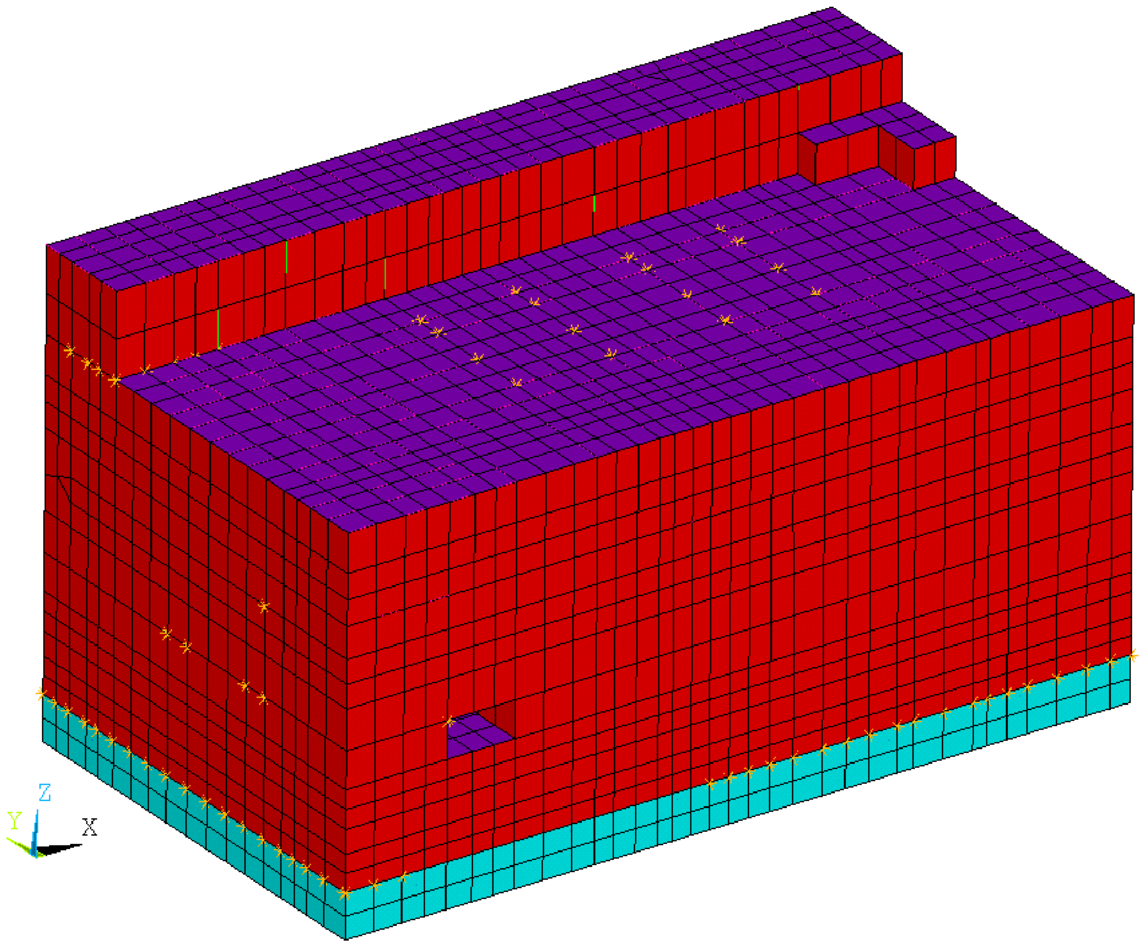


Figure 3.7.2-13 A/B Dynamic Model

Figure 3.7.2-14 Deleted

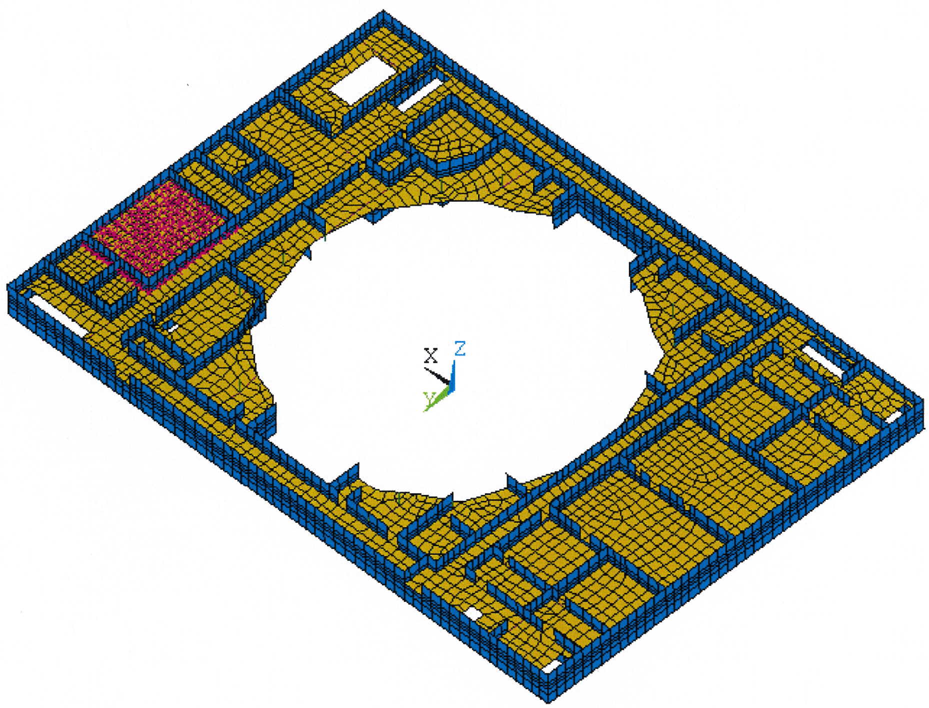


Figure 3.7.2-15 Extracted Detailed FE Model of Floor Slabs

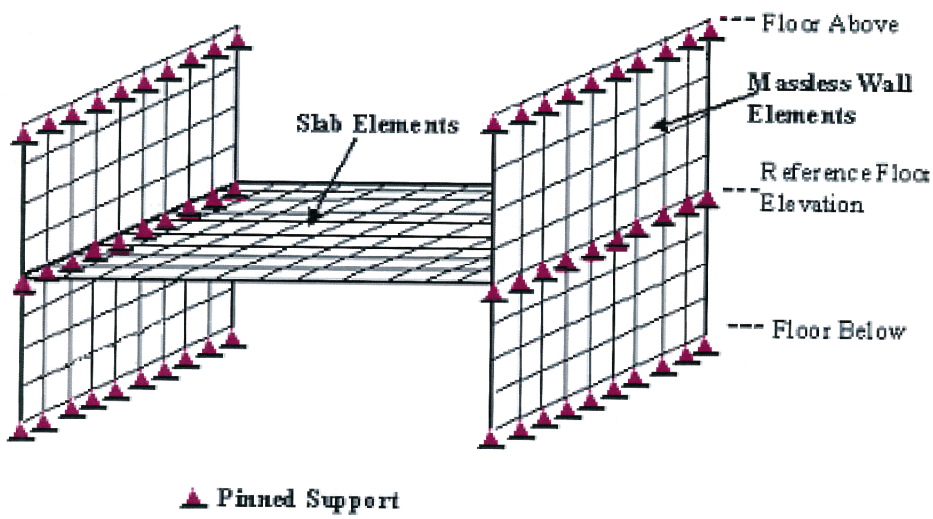


Figure 3.7.2-16 Floor Slab Model Boundary Conditions

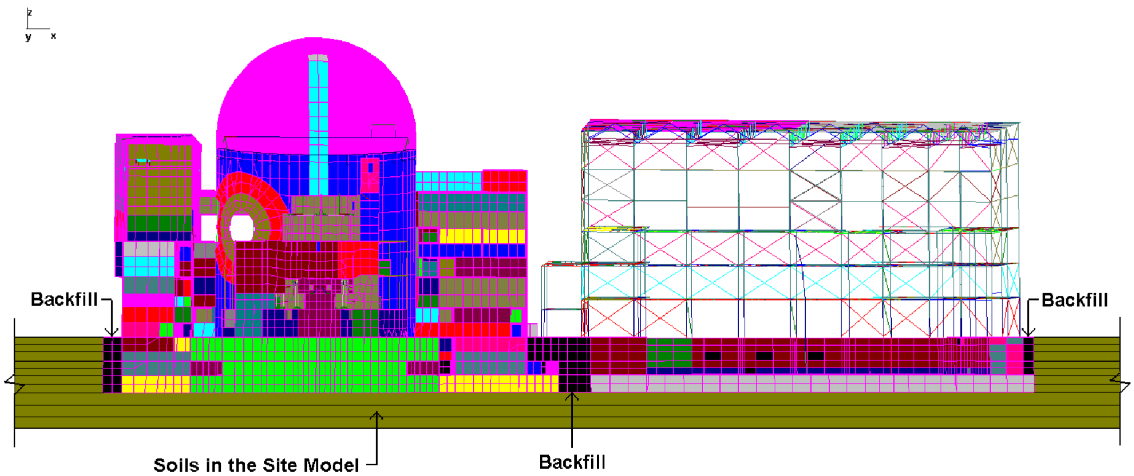


Figure 3.7.2-17 SSSI Model with Backfill and Free Field Soils (Looking East)

3.8 Design of Category I Structures

3.8.1 Concrete Containment

3.8.1.1 Description of the Containment

3.8.1.1.1 General Arrangement

The general arrangement (GA) drawings in Chapter 1 show the overall layout of the US-APWR PCCV including the vessel general outline, floor plans, and elevations of the overall structure. The geometric shape of the PCCV is a vertically oriented cylinder topped by a hemispherical dome with no ring girder at the dome/cylinder interface. The GA drawings reflect major equipment locations, including the nuclear steam supply system, and overall contents such as the RWSP, reactor cavity, refueling cavity, refueling canal, operating deck, polar crane, and major piping, mechanical, and electrical penetrations. Locations of other features are also shown including the containment internal structure, buttresses, equipment hatch, personnel airlocks, basemat, and tendon gallery.

The PCCV is anchored to a common basemat, which is described in more detail in Subsection 3.8.5, that it shares with the R/B complex.

The PCCV has an inside diameter of 149 ft, 2 in. and an inside height of 226 ft, 5 in. The thickness is 4 ft, 4 in. for the cylinder and 3 ft, 8 in. for the dome. Areas around the large openings are thickened to provide additional strength and provide space for the prestressing tendons that are deflected around the openings. The materials used to construct the PCCV are discussed in Subsection 3.8.1.6.

The PCCV consists of a prestressed concrete shell containing unbonded tendons and reinforcement steel. Prestressing is obtained through post-tensioning – a method of prestressing in which tendons are tensioned after concrete has hardened. Reinforcing steel is provided overall in the cylinder and dome. Additional reinforcement is provided at discontinuities, such as the cylinder-basemat interface, around penetrations and openings, at buttresses, and at other areas.

The PCCV is a concrete containment vessel with a metallic liner and is designed to the requirements of ASME Section III Division 2. Metallic penetrations through the PCCV are necessary for maintenance, access and process functions but because they penetrate the PCCV they are also required to perform containment pressure boundary and containment functions. A summary of the types of penetrations through the PCCV is provided as follows:

Equipment Hatch

Personnel Airlocks (2)

Fuel Transfer Tube

Large and Small Diameter Piping

Electrical Penetrations

The concrete shell inner surface is lined with a minimum 1/4-in. carbon steel plate that is anchored to the concrete shell and dome to provide the required pressure boundary leak tightness. Areas around penetrations, support brackets, inner walls, and heavy components bases have thickened steel liner plates. The other items integrally welded to the liner form part of the overall pressure boundary, including but not limited to, the equipment hatch at elevation 86 ft, 3 in. , an airlock at elevation 28 ft, 10 in. and a personnel airlock at elevation 80 ft, 2 in., various piping and electrical penetrations, and miscellaneous supports that are embedded in the concrete shell such as the polar crane brackets. The liner plate system is not designed or considered as a structural member in providing for the overall PCCV load resistance. The liner plate system is attached to the PCCV shell with an anchorage system that is depicted on Figure 3.8.1-2. In the cylinder portion of the PCCV, the liner is anchored with WT5x11s running vertically at a pitch of 1.6° (approximately 25 in. spacing along the inside face of the PCCV shell), and stiffened with 1/2 in. by 6 in. rib plates running horizontally in the hoop direction. In the dome portion of the PCCV, except the lowest panel portion where the cylinder liner anchorage system is also adopted, the upper portion of dome liner is anchored with 3/8 in. by 6 in. rib plates (spaced at approximately 32-1/4 in. maximum) which are oriented in a radial pattern originating at the dome apex. The rib plates are stiffened with 5 in. by 3 in. by 1/4 in. angles running horizontally in the hoop direction, spaced at 33 in. maximum. Where acceptable based on the results of design analyses performed for the liner-and-anchorage system (discussed in Subsection 3.8.1.4), the liner anchors are connected to the liner using discontinuous welds such as stitched fillet welds.

Figure 3.8.1-1 provides the overall dimensions of PCCV and Figure 3.8.1-5 provides GA of prestressing tendons and conventional reinforcement of the PCCV shell. Figure 3.8.1-3 and 3.8.1-4 also show the liner anchorage system arrangement.

3.8.1.1.2 Equipment Hatch

Figure 3.8.1-6 provides the equipment hatch general layout. The hatch is located at centerline elevation 86 ft, 3 in., azimuth 40 degrees, and is a 27 ft, 11 in. diameter spherical dish with a convex profile projecting into the PCCV volume. The containment internal pressure places the hatch head into compression against a double-sealed seat on the frame. The space between the two seals is capable of pressure testing for leakage across either seal.

A lifting rig with an electrically powered hoist is provided to disengage, raise, and store the hatch in a secure position above the opening during outages. When required to seal the opening, the hatch is lowered back by hoist, repositioned, refastened, and pressure tested for leaks. The hoist and lifting rig are the only components necessary to open or close the equipment hatch. The hoist is ac-powered by offsite power sources and the onsite alternate ACs (AACs).

3.8.1.1.3 Personnel Airlocks

Figure 3.8.1-7 provides the general layout for the two personnel airlocks. The lower airlock at centerline elevation 28 ft, 10 in. is located at azimuth 24 degrees, and upper airlock at centerline elevation 80 ft, 2 in. is located at azimuth 120 degrees. The airlock inside diameter is 8 ft, 6-3/8 in.

3.8.1.1.4 Mechanical Penetrations

Several typical PCCV penetrations are shown in Figure 3.8.1-8.

Figure 3.8.1-8, Sheet 13, shows typical details for the main steam penetrations. An anchor flange disc is embedded along the outer surface of the PCCV wall, with 12 triangular gussets at equal spacing connecting the flange disc and a 60 in. Outside Diameter (OD) cylindrical pipe sleeve, which is capped with a flexible boot outside the PCCV. A similar gusset configuration exists at the PCCV inner wall surface connecting the pipe sleeve to the thickened steel liner. The sleeve extends inside containment and is welded to the flued head. The distance from the inner surface of the containment to the flued head is 3 ft, 9-1/4 in. for Loops B (P510) and C (P511), and 4 ft, 3-1/4 in. for Loops A (P509) and D (P512). The 32 in. OD main steam pipe passes through the sleeve opening.

Figure 3.8.1-8, Sheet 14, shows typical details for the startup feedwater penetration. An anchor flange disc is embedded along the outer surface of the PCCV wall, with eight triangular gussets at equal spacing connecting the flange disc and 30 in. OD cylindrical pipe sleeve which is capped with a flexible boot outside the PCCV. A similar gusset configuration exists at the PCCV inner surface connecting the pipe sleeve to the thickened steel liner. The sleeve extends inside containment and is welded to the flued head. The distance from the inner surface of the containment to the flued head is 3 ft, 7-1/4 in. for Loops B (P502) and C (P503), and 3 ft, 9-1/4 in. for Loops A (P501) and D (P504). The 16 in. OD feedwater supply pipe passes through the sleeve opening. The 4 in. SG blowdown pipe (Figure 3.8.1-8 Sheet 15) passes through a 14 in. OD pipe sleeve that is anchored in the PCCV wall with four rectangular gussets embedded approximately midway in the wall. The sleeve extends inside containment and is welded to the flued head. The distance from the inner surface of the containment to the flued head is 1 ft, 10-5/8 in. for all loops. The SG blowdown pipe sleeve is capped with a flexible boot outside the PCCV.

The fuel transfer tube penetrates the PCCV wall near azimuth 0 degrees, connecting the refueling canal in the R/B with the refueling cavity in the interior of the PCCV. The fuel transfer tube penetration is sealed with the PCCV wall similar to other mechanical penetrations. The containment boundary is a double-gasketed blind flange at the refueling cavity end. The expansion bellows are independent of the containment boundary; however, they maintain water seals by accommodating differential movement of the structures. The fuel transfer tube penetration is shown on Sheet 17 of Figure 3.8.1-8.

In accordance with the Regulatory Position Section C.IV.1 “Combined License Application Acceptance Review Checklist” of RG 1.206 (Reference 3.8-1), the US-APWR PCCV is also equipped with dedicated PCCV penetrations, equivalent in size to a single 3 ft diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. These penetrations are shown on Sheet 6 of Figure 3.8.1-8.

Figure 3.8.1-8, Sheets 1 through 5, 7, 8, 11, and 16 show other typical mechanical penetration details. Figure 3.8.1-8, Sheet 17, provides the penetration detail of the refueling canal.

3.8.1.1.5 Electrical Penetrations

Figure 3.8.1-8, Sheet 9, 10, and 12, shows a typical electrical penetration detail.

3.8.1.1.6 Prestressing Configuration

Horizontal hoop tendons are used in the cylinder and the lower part of the dome. The horizontal tendons wrap around the entire circumference, and are anchored at two vertical buttresses 180 degrees apart. The anchors for the horizontal tendons are staggered such that adjacent tendons are anchored on opposite buttresses. The horizontal tendons anchored at the two vertical buttresses are accessed for servicing through vertical chases provided in the R/B at each buttress.

The inverted U tendons run vertically up the cylinder, over the dome in a non-radial mesh pattern, and down to the tendon gallery on the opposite side. These inverted U tendons, approximately configured in the form of an inverted “U,” are anchored at each end in a tendon gallery. The circular tendon gallery allows for servicing and installation of the inverted U tendons and is located entirely within the reinforced concrete basemat. The tendon gallery is accessed through a hallway, which passes horizontally through the basemat to the exterior plant yard.

Typical PCCV structural details are given in Figures 3.8.1-1 and 3.8.1-2. Design details include tendon and tendon anchorage, typical liner and liner anchorage, typical conventional reinforcing (non-prestressed) layouts, anchorage of the PCCV shell to the basemat, polar crane bracket, tendon buttress, structural reinforcing, and tendon spacing at openings. Table 3.8.1-1 presents basic design data for the PCCV that functions as the primary containment for the US-APWR.

3.8.1.2 Applicable Codes, Standards, and Specifications

The following industry codes, standards and specifications are applicable for the design, construction, materials, testing and inspections of the PCCV.

Rules for Construction of Nuclear Facility Components, Division 2, Concrete Containments, Section III, American Society of Mechanical Engineers, 2001 Edition through 2003 Addenda [hereafter referred to as ASME Code]. (Reference 3.8-2).

> Note: Articles CC-1000 through CC-6000 of Section III, Division 2 are acceptable for the scope, material, design, construction, examination, and testing of concrete containments of nuclear power plants subject to the regulatory positions provided by RG 1.136 (Reference 3.8-3).

Rules for Inservice Inspection of Nuclear Power Plant Components, Section XI, American Society of Mechanical Engineers, 2001 Edition through 2003 Addenda (Reference 3.8-4).

Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments, RG 1.136, U.S. Nuclear Regulatory Commission, Washington, DC, Revision 3, March 2007 (Reference 3.8-3).

Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments, RG 1.35, U.S. Nuclear Regulatory Commission, Washington, DC, Revision 3, July 1990 (Reference 3.8-5).

Determining Prestressing Forces for Inspection of Prestressed Concrete Containments, RG 1.35.1 U.S. Nuclear Regulatory Commission, Washington, DC, July 1990 (Reference 3.8-6).

Concrete Containment, NUREG-0800 SRP Section 3.8.1, U.S. Nuclear Regulatory Commission, Washington, DC, March, 2007 (Reference 3.8-7).

3.8.1.3 Loads and Load Combinations

The PCCV is designed for the loads and load combinations defined in the ASME Code, Section III (Reference 3.8-2), in Article CC-3200 "Load Criteria" and Table CC-3230-1 "Load Combinations and Load Factors," except as noted in RG 1.136 (Reference 3.8-3) Regulatory Position 5:

- The post LOCA flooding combined with the OBE set at one-third or less of the plant SSE is eliminated, since the load combination is less severe than the post-LOCA flooding combined with a SSE.

- ASME Code, Section III, Subarticle CC-3720 is satisfied by addressing an accident that releases hydrogen generated from 100% fuel clad-coolant reaction accompanied by hydrogen burning, including the effects of temperature and prestress. See Subsection 3.8.1.3.2.2 for further discussion of this design condition.

Load combinations and factors based on ASME Table CC-3230-1 are presented in Table 3.8.1-2. Load combinations involving severe wind, hurricane and tornado have been determined to be less severe than other cases through comparison calculations to the design-basis earthquake loads and, therefore, load combinations involving severe wind, hurricane and tornado are not used in the full detailed design analyses of the overall PCCV structure and its liner.

3.8.1.3.1 Loads

The following is a brief description of loads unique to the PCCV and liner used in Table 3.8.1-2 for design and analysis. Subsection 3.8.4.3 gives definitions and descriptions of other loads based on the ACI 349-06 (Reference 3.8-8) and AISC N690-1994, including Supplement 2 (Reference 3.8-9), which are consistent with the ASME Code, Section III.

- **Prestress Load**

  For purposes of the US-APWR PCCV design, prestress is defined as the load on the PCCV dome and cylinder walls that, when applied by mechanical force from tendons after the concrete has hardened, results in the introduction of internal stresses to reduce potential tensile stresses in concrete resulting from other loads. The initial prestress governs the cylinder wall and dome thickness. It is not governed by radiation shielding. The minimum prestress level including all losses after design life applied to the PCCV is 1.20 times the design pressure.

- **Design-Basis Accident Pressure (P_a) and Test Pressure (P_t)**

The DBA pressure is 68 psig. The DBA pressure is increased for structural design purposes using load factors as shown in ASME Code, Section III, Table CC-3230-1, depending on the particular load combination considered.

The structural integrity test pressure P_t is 1.15 times the design pressure (P_t = 78.2 psig).

External or internal events such as containment spray actuation may induce a negative pressure on the PCCV. See Chapter 6 for further discussion. Therefore, the PCCV is designed for a negative pressure of 3.9 psig as a separate event.

With respect to accident pressure loads, 10 CFR 50.44 (Reference 3.8-10) requires that an analysis be performed that demonstrates that the containment structural integrity is maintained under loads resulting from combustible gases generated from metal-water reaction of the fuel cladding. In determining loads from combustible gases, the US-APWR design follows the guidance of RG 1.7 (Reference 3.8-11), in determining and analyzing the design accident pressure loads.

- **Thermal Loads (T_o) and Accident Thermal Loads (T_a)**

The normal operating environment inside and outside the PCCV is specified in Table 3.8.1-3 and Figures 3.8.1-9 through 3.8.1-13. Normal thermal loads for the exterior walls and roofs are addressed in the design of the PCCV. For the effects of transient loads such as T_a, the overall behavior of the PCCV is first determined. A portion of the PCCV shell can then be analyzed for local effects using the results obtained from the global analysis as boundary conditions, for example at penetrations and/or at its anchorages to the basemat.

During normal operation, a linear temperature gradient develops across the PCCV wall thickness. After a LOCA, however, the sudden increase in temperature in the liner and adjacent concrete produces a nonlinear transient temperature gradient. The temperature versus time is considered when combining with accident pressure in the specified load combinations, and worst case temperature gradients within the volume of the PCCV are used in the thermal analyses as discussed in Subsection 3.8.1.4. The calculated thermal gradients are developed in a manner consistent with the methodology of ACI 349-06 (Reference 3.8-8) Appendix E and its corresponding commentary.

- **Earthquake Loads (E_{ss})**

For the PCCV, earthquake loads E_{ss} and the seismic analysis are discussed and summarized in Section 3.7. There are two horizontal and one vertical earthquake components that require combination as discussed in Subsection 3.7.2.6. Earthquake loads are applied to the three dimensional structural FE model.

3.8.1.3.2 Other Loads

Loads other than those discussed in the previous subsection, such as crane or other attachment loads, hydrodynamic, jet impingement or pipe impact loads resulting from HELB, and flooding have also been investigated in the overall design but also in particular for local effects. Construction loads on the liner are of particular concern and are included in the discussion in Subsection 3.8.1.3.

3.8.1.3.2.1 OBE-Induced Stress Cycles

As recommended in Section II.3.C of NUREG-0800, SRP 3.8.1 (Reference 3.8-7), OBE-induced stress cycles are considered in the design of the liner adjacent to crane brackets. In determining the number of earthquake cycles for use in design, the guidance of NRC Staff Requirements Memorandum SECY-93-087 (Reference 3.8-12) is used. The number of earthquake cycles used is two SSE events with 10 maximum stress cycles per event or equivalent.

3.8.1.3.2.2 Hydrogen Burn

Containment integrity is maintained by applying Subarticle CC-3720 of the ASME Code, Section III (Reference 3.8-2), to an accident (exclusive of seismic or DBA) condition that releases hydrogen generated from 100% metal-water reaction of the fuel cladding accompanied by hydrogen burning. Under these conditions, the loadings do not produce strains in the PCCV liner in excess of the limits established in Subarticle CC-3720 of the ASME Code, Section III (Reference 3.8-2).

For the factored load design associated with the prestressed concrete wall:

$$D + P_g1 + [P_g2 \text{ or } P_g3\ ]$$

where

D = Dead load

P_g1 = Pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction = 46.7 psia

P_g2 = Pressure resulting from uncontrolled hydrogen burning (if applicable) = 127 psia

P_g3 = Pressure resulting from post-accident inerting assuming carbon dioxide is the inerting agent (Not applicable to US-APWR)

The factored load design of the US-APWR PCCV complies with the guidance of RG 1.136 (Reference 3.8-3). MHI Technical Report MUAP-10018 “US-APWR Containment Performance for Pressure Loads” (Reference 3.8-55) documents the methodology used to determine the pressure effects of an accident that releases hydrogen generated from 100% fuel clad metal-water reaction and uncontrolled hydrogen burning on the PCCV. The maximum pressure considered in the analysis in MUAP-10018 (Reference 3.8-55) is $P_g1 + P_g2$ = 173.7 psia = 159 psig. The analysis also includes effects of dead load D.

3.8.1.3.3 Load Combinations

Load combinations and applicable load factors are presented in Table 3.8.1-2, for which the containment structure is designed.

3.8.1.3.4 Liner Plate Loads and Load Combinations

Liner plate strains are evaluated for the same loads and load combinations as those used to design the PCCV shell, which are presented in Table 3.8.1-2, except that all load factors for the liner plate are 1.0 in accordance with Subarticle CC-3720 of the ASME Code, Section III (Reference 3.8-2). In general, load cases that are shown to be less severe than other cases do not receive a full design analysis.

Liner plate stresses are evaluated for the construction load category and for the mechanical loads applied to attachments on the liner plate. During construction, the liner plate functions as the inner concrete form and as such it is subject to pressure from concrete placement as a primary load. This pressure can be treated as a hydraulic load with a maximum pressure determined as follows: the head height is the sum of the placement rate plus one foot for vibration plus one foot for miscellaneous factors. After the concrete sets, this load on the liner is no longer a real mechanical load; therefore, it is not combined with other primary loads.

A condition of the liner which is considered in the design occurs after the postulated DBA, when the pressure has decreased and the temperature is high in the liner, but has not yet significantly increased in the concrete shell. This condition produces large loads in the liner due to the concrete anchorage restraining expansion of the liner steel.

Accident pressure has little effect on the liner plate since it is backed by the concrete shell which is constructed against it.

Other loads and effects for the liner, penetrations, brackets, and attachments are considered. Local thickening of the liner is provided as necessary at penetration assemblies. The liner analysis considers deviations in the liner geometry due to fabrication and erection tolerances, including secondary stresses caused by service and factored loads to the displaced shape of the liner caused during construction as discussed above. Stresses imposed by mechanical load of concrete are not included since those stresses do not pose real loads once the concrete has hardened.

The effects of anchors, embedments, or other attachment details not attached to the steel liner or a load carrying steel element that provide anchorage into the PCCV from the external surface, are considered for their effect on the PCCV. The liner is not considered as a structural member when determining overall PCCV integrity, however where necessary the liner may be considered to satisfy the requirement of 0.0020 times the gross cross-sectional area for reinforcement in each direction on the inside face of the PCCV to resist effects of shrinkage, temperature, and membrane tension.

3.8.1.4 Design and Analysis Procedures

Design and analysis procedures for structural portions of the PCCV, and specified allowable limits for stresses and strains as discussed in Subsection 3.8.1.5, are in accordance with Article CC-3000 of the ASME Code, Section III (Reference 3.8-2). The design and analysis procedures for the PCCV, including the steel liner, are according to those stipulated in Article CC-3300 of the ASME Code, Section III (Reference 3.8-2) and RG 1.136 (Reference 3.8-3). ASME Code, Section III, Article CC-3100 applies to the design of the "Concrete Containment" and the "Metallic Liner." ASME Code, Section III (Reference 3.8-2), covers both the "Service Load Category" and the "Factored Load Category." Loads are classified as "Primary" or "Secondary" in accordance with definitions provided by the ASME Code, Section III (Reference 3.8-2).

The PCCV analysis methods are summarized in Table 3.8.1-4. For the US-APWR, the PCCV analysis assumes a fixed base condition. The basemat design is further described in Subsection 3.8.5, the R/B in Subsection 3.8.4, and the containment internal structure in Subsection 3.8.3. The SSI design and analysis approach is discussed further in Subsection 3.7.2.4.

The detailed PCCV analyses use general purpose global FE models. The global FE model addresses discontinuities and openings in the PCCV structure, such as the cylinder-basemat interface, cylinder-dome springline, buttress-wall interface, equipment hatch, and personnel airlock openings. Changes in material properties, changes in physical dimensions such as thicknesses, and changes in boundary or support conditions between elements are accounted for in the models. The FEs used have membrane, bending, and tangential and radial shear capability.

Computer code development, verification, validation, configuration control, and error reporting and resolution are in accordance with the Quality Assurance requirements of Chapter 17.

3.8.1.4.1 Analyses for Design Conditions

3.8.1.4.1.1 Analytical Methods

The PCCV structure is analyzed by the use of the linear elastic FE computer program ANSYS (Reference 3.8-14). The PCCV is isolated from other structures for the analysis of shell and dome stresses, however, it is supported on and anchored to a common basemat with those structures. The PCCV structure is idealized for analysis and modeled with ANSYS as a structure consisting of isoparametric membrane-bending plate elements.

The three-dimensional global FE analysis model as represented in Figures 3.8.5-5 through 3.8.5-10 includes the overall PCCV structure, as well as the R/B, east PS/B, west PS/B, A/B, containment internal structure and the common basemat to which all these structures are supported. The FEs used for the PCCV analyses have membrane, bending, tangential, and radial shear capability. The model accounts for effective prestress equivalent to the variation of tendon friction due to losses or changing geometry, for example the inverted U-shape tendons' transition from cylinder to dome. In developing the model, the mesh size is chosen to comply with the following basic considerations and empirical checks.

- When considering areas, such as the main steam penetration, concentrated load, or reaction areas, the critical location for shear is generally one-half the thickness away from the opening edge and, the element size should account for this fact.
- The mesh discretization is chosen to assure adequate representation of the controlling stresses for key elements of the design such as for the general shell, the basemat, the discontinuities at cylinder base and the intersection with the dome, the large openings, buttresses, high energy piping penetrations, and pipe whip restraint locations, where required.

The behavior of the PCCV model overall is typically axisymmetric, particularly under dead and pressure loads. The non-axisymmetric effects of such loads including but not limited to severe wind, tornadoes, hurricane, earthquake, and pipe rupture are taken into account in the FE analysis as required by SRP 3.8.1, Section II.4.B (Reference 3.8-7).

In designing the PCCV superstructure, the square root of the sum of the squares (SRSS) method based on elastic analyses is used to evaluate the seismic load for the three components of the earthquake. The design forces due to the seismic load obtained by the SRSS method are beyond those obtained from inelastic analysis, at the PCCV shell/mat interface. The associated redistribution effects are found to be insignificant.

Stress analyses of the FE models are performed considering the following loads defined in accordance with ASME Code, Section III, Article CC-3000 (Reference 3.8-2):

- Dead load
- Live load (including polar crane loads as applicable)
- Prestressing load
- Internal pressure
- Seismic load
- Wind load
- Thermal load

With regard to thermal load, in order to consider thermal effects in the global FE model due to expansion of the liner, the liner plate loading is taken into account without explicitly modeling the stiffness of the liner.

Prestressing force is calculated considering the losses due to slip at anchorage, elastic shortening, creep of concrete, shrinkage of concrete, stress relaxation and tendon friction.

Large openings are modeled in the three-dimensional global FE analysis model described above. The design of the large openings for the one equipment hatch and the two personnel airlocks use the results of FE analyses using this global FE model. In accordance with ASME Code, Section III, Subarticle CC-3544 “Curved Tendons” and Subarticle CC-3340(a), the global model accounts for all forces imposed by tendons curved around the opening, such as effective prestress equivalent to the variation of prestress forces due to friction and other loses. The global FE model has membrane, bending and tangential, and peripheral and radial shear capability.

The PCCV buttresses are modeled in the three-dimensional global FE analysis model described above so that the discontinuity effects of the normal shell and the thickened buttress can be evaluated in the design. Local effects are also considered using the test results documented in Testing of Large Prestressing Tendon End Anchor Anchorage Regions, by T.E. Johnson (Reference 3.8-15).

Seismic Analysis of the PCCV for Structural Design

Response spectrum analysis with the Lindley-Yow Method, described in NRC RG 1.92 (Reference 3.8-75), is selected for the seismic analysis of the PCCV. The Lindley-Yow method divides the total seismic response into two components: periodic response with the ground motion (“out-of-phase” response) and rigid response with the ground motion (“in-phase” response). Response spectrum analysis is performed for the periodic response with modified response spectrum in accordance to Section 1.3.2 of RG 1.92 (Reference 3.8-75). The complete quadratic combination (CQC) method is used to combine modal responses. Static ZPA Method, which considers missing mass response as described in Section 1.4 of RG 1.92 (Reference 3.8-75), is used for the rigid response. The complete (periodic plus rigid) response spectrum analysis solution is calculated in accordance to Section 1.5.2 of NRC RG 1.92 (Reference 3.8-75). The maximum earthquake-induced response is combined by SRSS combination of the maximum representative responses from the three earthquake components (one vertical and two horizontal earthquake components) calculated separately in accordance with Section 2.1 of NRC RG 1.92 (Reference 3.8-75).

In order to perform the response spectrum analysis, response spectra need to be created using in-structure response spectra (ISRS) from the soil structure interaction (SSI) analyses. SSI analyses include 6 soil profiles and cracked and uncracked conditions. For a more detailed explanation of the SSI analyses, refer to Section 3.7 of the DCD.

For the response spectrum for each profile, the ISRS at 5 nodes (4 nodes around the circumference and one node in the center) at the bottom of the PCCV at elevation 3 ft, 7 in. are enveloped. After the response spectra for individual soil profiles with cracked and uncracked condition (the total number of 12 response spectra for 6 soil profiles and cracked and uncracked conditions) are created, they are enveloped and 15% broadened (referred to as “the enveloped soil case”) to represent all soil conditions with design conservatism.

A constant modal damping ratio of 3% and uncracked concrete stiffness is considered in the response spectrum analysis. It is considered conservative for structural design of prestressed concrete structures, for which RG 1.61 (Reference 3.8-64) permits use of 5% SSE and 3% OBE damping. In order to justify the seismic force transfer from the dynamic model to structural model for structural member design, the seismic shear determined from the response spectrum analysis is compared to the seismic shear diagrams provided in Section 03.4.3 of Technical Report MUAP-10006 (Reference 3.8-85).

For the design of the PCCV structural members, the response spectrum for the enveloped soil case combined with other loads was used.

3.8.1.4.1.2 Thermal Analyses

The PCCV atmosphere and open-air or indoor atmosphere are subject to a steady temperature condition during normal operation. The steady temperature conditions result in a linear temperature distribution within the PCCV shell. Normal operating temperature gradients are based on values provided in Table 3.8.1-3, Thermal Conditions of the R/B and PCCV.

The PCCV is subject to a rapid temperature transient in the event of a LOCA. Figure 3.8.1-10, Transient Conditions of PCCV Atmosphere Temperature (Accident Condition), illustrates the timeline of the interior temperature used in the thermal analyses under accident conditions. LOCA affects the liner more significantly initially and the concrete is more time-dependent as the temperature transient moves through the wall thickness. The temperature transients result in a nonlinear temperature distribution within the PCCV shell and the resulting profiles are calculated in a uni-dimensional heat flow analysis. This uni-dimensional heat flow is normalized and the average temperature distribution and equivalent linear gradients are created and applied to the FE model of the PCCV.

3.8.1.4.1.3 Variation of Physical Material Properties

In the design analysis of the PCCV, the physical properties of materials are based on the values specified in applicable codes and standards. The design analysis takes into account the minimum/maximum values permitted by the codes and standards as appropriate to capture worst case analysis scenarios.

3.8.1.4.2 Design Methods

The design of the PCCV structure is based on the membrane forces, shear forces and bending moments resulting from the loads and load combinations defined in Subsection 3.8.1.3. The membrane forces, shear forces and bending moments in selected sections are obtained from the linear FE analysis.

3.8.1.4.2.1 Concrete Cracking Considerations

As discussed in SRP 3.8.1 (Reference 3.8-7) Section II.4.D, concrete cracking can affect the stiffness of the PCCV and cause shifting of the natural frequency, thereby affecting the response/loads used to design the PCCV. Accordingly, the analysis used to calculate the dynamic response of the PCCV resulting from dynamic loads such as earthquake and hydrodynamic loads considers the potential effects of concrete cracking where significant. The addition of stiffness to the concrete sections due to the presence of the liner is not considered in the analysis and design of the PCCV concrete shell.

The concrete and reinforcement stresses are calculated considering the extent of concrete cracking at these sections. The following are assumptions for calculations:

- The concrete is isotropic and linear elastic but with zero tensile strength
- The thermal forces and moments are reduced according to the concrete cracking depth
- The redistribution of section forces and moments that occurs due to concrete cracking is taken into account

The depths of cracks were determined using an iterative process that initially determines the total load applied to an uncracked section. A crack depth is then postulated on the tensile face, the neutral axis is shifted, and the redistributed forces and moments are recalculated. This process is repeated with the depth of the crack being increased until force equilibrium is obtained. This iterative process allows the location and depth of the crack to be analytically evaluated and also establishes a deterministic approach to obtain the force and moment reduction within the concrete section.

For thermal loads, the effects of concrete cracking are considered in developing the internal forces and moments in the section. For these loads, concrete cracking relieves the thermal stress, as well as redistributes the internal forces and moments on the sections from those obtained from a linear analysis.

Thermal loading, particularly the accident temperature applied to the PCCV steel liner, generates a force on the surface of the PCCV wall and creates a moment relative to the neutral axis. This thermal moment is generally capable of cracking the concrete and loads are redistributed to reinforcing steel with the development of a couple and more specifically reduced by a change in the distance of the crack to the shifted neutral axis. The shifting of the neutral axis results in the redistribution of the forces and moments within a local area.

Primary loads and combined primary and secondary thermal loads are considered both individually and in combination. Both conditions are considered because the location of the tensile face can be different for each loading condition and is influenced by the geometry of the structure.

The PCCV shell is evaluated for a condition in which the liner is heated as a result of a LOCA while the concrete maintains a normal operating temperature gradient. The difference in temperature induces a compressive stress and strain in the liner plate. This condition is defined as the liner plate spike load.

3.8.1.4.3 Ultimate Capacity of the PCCV

The US-APWR ultimate pressure capacity analyses are based on detailed 3-D FE modeling, advanced material constitutive relations including material degradation with temperature, and an assessment of uncertainties within a probabilistic framework.

Fragility for over-pressurization is a function of temperature because of thermal induced stresses and material property degradation at elevated temperatures. Thus, the fragility analyses are conducted for three different thermal conditions, 1) normal operating steady-state conditions, 2) a long term accident condition, and 3) a hydrogen burning condition.

Analyses indicate that the ultimate capacity is limited by liner tearing, which first initiates at the transition to the thickened concrete section for the equipment hatch under both normal operating and long-term accident conditions. The expected or median pressure to initiate tearing is found to be 223.6 psig or 3.29 times the design pressure (Pd) of 68 psig for the steady state thermal conditions associated with a long term accident condition. The expected medial pressure to initiate tearing is found to be 230.0 psig, or 3.38*P_d, for the steady state thermal conditions associated with normal operating conditions. This limitation in pressure capacity due to liner tearing for the long-term accident condition is consistent with the ¼ scale PCCV tests performed at Sandia National Laboratories (SNL). The 95% confidence value for liner tearing is determined to be 184.9 psig, or 2.72*P_d for normal operating conditions and 176 psig, or 2.59*P_d, under long term accident conditions.

The limiting mechanism for pressure capacity at the hydrogen burning condition is determined to be buckling and subsequent tearing of the equipment hatch cover. The analyses indicate buckling develops in the equipment hatch cover creating a plastic hinge and ultimately tearing at the outer periphery of cover. The median value for pressure capacity due to failure of the equipment hatch cover under hydrogen burning conditions is 220.9 psig (3.25*P_d) with a 95% confidence value of 163.3 psig (2.40*P_d). The median capacity due to liner tearing for the hydrogen burning case is found to be 238.5 psig or 3.51*P_d. This pressure is higher than that at normal operating conditions, which is attributed to the compressive stress induced into the liner due to the locally higher temperatures of the liner relative to the concrete.

A lower pressure of 171 psig (2.51*P_d) is found for the 95% confidence value for rebar failure around the equipment hatch with the median pressure capacity being 237.4 psig (3.49*P_d). This ultimate capacity develops in the local reinforcement on the outside surface of the PCCV around the equipment hatch. The local equipment hatch model indicates that local liner capacity around the equipment hatch is enveloped by the PCCV liner capacity, thus it is concluded that the PCCV liner capacity would be the limiting pressure capacity near the 95% value (176 psig) determined for that postulated condition.

The 95% high confidence value for pressure capacity due to liner tearing, away from openings, under hydrogen burning conditions is lower than that for normal operating conditions reflecting the additional uncertainty for the severe accident conditions and effects of high temperatures. For ultimate capacity based on rebar and tendon rupture, the median pressure capacity for long term design accident conditions is found to be 243.6 psig or 3.58*Pd. It is also determined that the ultimate capacity is not limited by the concrete strength. These results are again consistent with the SNL test for the ¼ scale PCCV model. These analyses also indicate that the ultimate capacity does not strongly depend on temperature. The median ultimate capacity at normal operating temperature is determined to be 3.65*Pd and the median ultimate capacity under hydrogen burning conditions is 3.60*Pd.

The fragility analyses and detailed description of the methodologies are summarized in MUAP-10018 (Reference 3.8-55).

3.8.1.4.4 Liner System Design and Analysis Procedures

The design and analysis procedures for the liner as well as its liner anchors and all penetration assemblies, brackets, and attachments that could affect leak-tightness, are in accordance with the ASME Code, Section III, Division 2 requirements given in Articles CC-3600 and CC-3700 (Reference 3.8-2).

The liner design and analysis procedure do not take credit for the liner as contributing to the strength of the PCCV shell. Instead, the liner design and analysis procedures assure that the liner is designed to accommodate the strains induced by the deformation of the PCCV shell to which the liner is anchored without loss of liner leak-tight integrity under the loads and load combination discussed in Subsection 3.8.1.3. This is in accordance with ASME Code, Section III, Division 2 (Reference 3.8-2), Subarticles CC-3121 and CC-3122. The liner is also designed and analyzed for the loads imposed during construction, such as concrete placement form pressure defined in Subsection 3.8.1.3.4, and mechanical loads applied to attachments on the liner plate. In particular, local and overall dome stability for concrete placement is verified.

The liner plate analysis also considers potential deviations in geometry due to fabrication and erection tolerances. The design strains, stresses, and forces in the liner and its anchors are within allowable limits defined by the ASME Code, Section III, Article CC-3700 (Reference 3.8-2) and discussed further in Subsection 3.8.1.5.

The stiffness of the liner plate is not included in the FE model of the PCCV shell. For evaluation of the PCCV concrete and reinforcement, no credit is taken for strength contribution of the liner to the PCCV shell. The results of the analyses are evaluated utilizing a post-processor that considers concrete cracking and strain compatibility among concrete, liner, tendons, and reinforcement on the section subject to primary and secondary loads. To evaluate the liner strains for secondary loads such as thermal, the post-processor evaluation considers strain compatibility among the concrete, tendons, reinforcement, and liner plate, including liner thermal expansion effects. When considering effects of thermal gradients in the PCCV wall from accident conditions on the liner, reduction in thermal-gradient-induced thermal moments due to concrete cracking is considered.

Liner Anchorage

The liner anchorage system is designed so that its mechanical behavior is reasonably predictable for all design-basis loadings. The design and analysis procedures for the liner anchorage system conform to the ASME Code, Section III, Division 2 requirements given in Subarticles CC-3630 and CC-3730, and are similar to the anchorage system analysis approaches illustrated in BC-TOP-1 (Reference 3.8-17). Liner anchors are analyzed considering elastic behavior of the liner plate and non-linear behavior of liner anchors. In the cylinder, the liner is treated as a one-way strip in the hoop direction (analyzed uniaxially) with multiple continuous spans across liner anchors. Liner anchor spring behavior for the analysis of the liner-anchorage system are based on test results obtained from liner anchorage shear tests.

A non-linear analysis is performed to determine the maximum displacement in the liner anchor and the resultant stresses and strains in the liner. The analysis assumes that a buckle in the liner could occur at any location.

Two analyses are performed. In the first analysis, the buckle is assumed to occur in the membrane region and the thickness of the plate is ¼ inch on either side of the buckled panel. In the second analysis, the buckled panel is assumed to occur in the ¼ inch panel when it connects to a thickened panel at penetrations. The second analysis takes into account the larger forces imposed by the thicker panels and also incorporates the restraint provided by the penetrations. In both analyses, the restraint provided by the buckled panel is conservatively neglected.

Penetration Assemblies and Openings

For the penetration assemblies and openings, the US-APWR follows the ASME Code, Section III, Division 2 (Reference 3.8-2), requirements given in Subarticles CC-3640 and CC-3740. Penetration assemblies and openings, such as personnel airlocks, equipment hatch, and the fuel transfer tube assembly are analyzed using the same techniques and procedures as defined in Division 1 of ASME Code, Section III (Reference 3.8-2), where these components are not backed by concrete. The analysis considers the concrete confinement for the embedded portions of penetration sleeves as required by ASME Code, Section III.

For brackets and attachments that form part of the liner system, the design and analysis procedures conform to the ASME Code, Section III requirements given in Subarticle CC-3650 (Reference 3.8-2).

3.8.1.4.5 Design and Analysis Procedures for Impactive and Impulsive Loading

The methods of analysis for impactive and impulsive loading used on the PCCV and its liner are either an energy balance technique or a non-linear dynamic analysis with a forcing function that represents the impulsive and/or impactive loading condition. The empirical missile penetration formulas used are described in Section 3.5. For the PCCV and its liner, missile penetration is limited to well below 75% of the total section thickness while at the same time ensuring that the overall structural integrity of the section is not compromised. The PCCV nominal thickness dimensions of 4 ft, 4 in. for the cylinder and 3 ft, 8 in. minimum for the dome exceeds the required 16 in. thickness for Region 1 tornado missiles and for hurricane missiles with minimum concrete strength of 7,000 psi. Based on the robust nature of the PCCV, externally generated design-basis missiles including tornado missiles, as discussed in Section 3.5, do not challenge the PCCV cylinder or dome. The SG and pressurizer compartments protect the liner from direct missile impact. In other areas of the PCCV where a high-energy piping missile potential is not discounted due to the LBB analysis discussed in Subsection 3.6.3, missile shielding in accordance with Section 3.5 is utilized inside the PCCV to prevent missile impact on the liner.

3.8.1.4.6 Design Report

A Design Report of the PCCV is provided separately from the DCD. The Design Report has sufficient detail to show that the applicable stress limitations are satisfied when components are subjected to the design loading conditions.

3.8.1.5 Structural Acceptance Criteria

The PCCV, including its liner, is designed considering the loads and load combinations discussed in Subsection 3.8.1.3, and meets the structural acceptance criteria discussed in this subsection. The US-APWR PCCV structural acceptance criteria are based on the allowable stress and strain requirements given in Article CC-3400 of the ASME Code, Section III (Reference 3.8-2), and Article CC-3700 for the liner. In accordance with those requirements, the PCCV structure is designed to remain elastic under service load conditions and below the range of general yield under load conditions involving factored primary loads. In limited instances when load conditions involve primary plus secondary factored loads, a general yield state may occur only for some secondary components as permitted by Subarticle CC-3110, and not with respect to radial shear stress; however, reinforcement and concrete strains are maintained within allowable limits given in Subarticle CC-3420. The allowable stresses and strains are summarized in the following paragraphs where the major components of the PCCV and its liner are discussed with respect to factored loads and then service loads.

3.8.1.5.1 Acceptance Criteria for Factored Load Conditions

Factored loads include loads encountered infrequently, such as severe environmental, extreme environmental, and abnormal loads.

3.8.1.5.1.1 Concrete

The US-APWR design follows the requirements of Subarticle CC-3421.1 and Table CC-3421-1 of the ASME Code Section III (Reference 3.8-2), which define the allowable concrete stresses for membrane and membrane plus bending. The allowable stresses therein are defined for both primary and primary-plus-secondary factored loads. Primary and secondary forces are defined in Subarticle CC-3136 of the ASME Code, Section III (Reference 3.8-2). Primary forces are the result of items such as actual loads, whereas secondary forces result from conditions caused by internal self-constraint and are self-limiting. The forces which result from thermal strain of the concrete wall are an example of secondary forces.

As stated in Subarticle CC-3421.2 of ASME Code, Section III (Reference 3.8-2), concrete tensile strength is not relied upon to resist membrane and flexural tension forces.

<u>**Concrete in General Shear**</u>

The US-APWR complies with ASME Code, Section III (Reference 3.8-2) requirements for qualification of concrete shear. Shear capacity is defined using two components. One component is that carried by the concrete defined as V_c, and the other, if required, is that carried by the reinforcing steel V_s. The total shear capacity of the concrete, provided by the sum of the two, is greater than the applied shear load. In the ASME Code, Section III

(Reference 3.8-2), the concrete capacities are defined in Subarticle CC-3420 "Allowable Stress for Factored Loads." The steel reinforcement capacities for factored load design are defined in Subarticle CC-3521 "Design of Shear Reinforcement."

Radial Shear

The radial shear provisions for the US-APWR are in accordance with the ASME Code, Section III (Reference 3.8-2), as stated in Subarticles CC-3421.4.2 "Prestressed Concrete" and CC-3521.2 "Radial Shear."

Tangential Shear

The allowable tangential shear stress in concrete is defined in Subarticle CC-3421.5.2, which defines concrete tangential shear strength based on providing a minimum amount of prestress as described in Subarticle CC-3521.1.2. ASME Code, Section III (Reference 3.8-2), Subarticle CC-3521.1.2 requires that: "(a) A sufficient amount of prestress shall be provided so that N_h and N_m are negative (compression) or zero. Thermal membrane forces shall be included in N_h and N_m for the calculation of effective prestress" and "(b) No additional reinforcement is required for tangential shear forces if $V_u \leq 0.85\ V_c$ where V_c is calculated according to Subarticle CC-3421.5.2." Item (c) of Subarticle CC-3521.1.2 further states "When the section under consideration does not meet the requirements of either Subarticle CC-3521.1.2 (a) or (b), additional reinforcement shall be provided according to requirements of Subarticle CC-3521.1.1." The requirements of Subarticle CC-3521.1.1 include provisions for inclined reinforcement. Because it is highly undesirable from a construction standpoint to provide inclined shear reinforcing, the PCCV shell is designed such that any tangential shear reinforcement provided is orthogonal only (hoop/meridional), and the amount of prestress used in the design is increased as necessary to preclude the use of inclined shear reinforcement.

For purposes of tangential shear reinforcement design per Subarticle CC-3521, the membrane forces N_h and N_m include thermal, pressure, prestress and dead loads but do not include earthquake, severe wind, tornado, or hurricane loads. The lateral membrane loads from earthquake or wind are defined in N_{hl} and N_{ml} and the lateral tangential shear force is defined in V_u.

For the structural portions of the PCCV, the specified allowable limits for stresses and strains are in accordance with Article CC-3400 of the ASME Code, Section III (Reference 3.8-2), with additional limits provided by RG 1.136, Regulatory Position 5.C. (Reference 3.8-3).

Peripheral Shear

This type of shear loading (also known as punching shear) is applicable to penetration areas and items such as the crane brackets. The PCCV complies with shear allowable relative to the concrete shear capacity as given in Subarticle CC-3421.6, and Subarticle CC-3521.3 for reinforcement shear capacity (Reference 3.8-2).

Torsional Shear

This type of loading can occur at piping penetrations due to applied piping loads. The torsional shear allowable relative to concrete is given in Subarticle CC-3421.7 (Reference 3.8-2). At penetrations and similar situations, when the applied shear exceeds that determined in Subarticle CC-3421.7, shear reinforcement is provided in accordance with Subarticle CC-3521.4 (Reference 3.8-2).

3.8.1.5.1.2 Prestressing System

Tendons

The allowable for factored loads is 90% of yield as stated in Subarticle CC-3423 (Reference 3.8-2).

End Anchor

The US-APWR is in accordance with ASME Code, Section III, Subarticle CC-3431.1 for concrete compression allowable under the tendon bearing plates. The anchorage components meet the requirements Subarticles CC-2430, CC-2450 and CC-2460 (Reference 3.8-2).

Prestressing Losses

The losses considered in the tendons are based on the items defined in ASME Code, Section III, Subarticle CC-3542 (Reference 3.8-2) including:

1. Slip at anchorage
2. Elastic shortening of concrete
3. Creep of concrete
4. Shrinkage of concrete
5. Stress relaxation
6. Friction loss due to intended or unintended curvature in the tendons

In addition, “Determining Prestressing Forces for Inspection of Prestressed Concrete Containments,” RG 1.35.1 (Reference 3.8-6) is used as guidance for determination of prestressing losses. Prestressing losses are computed on the basis of a 60 year design life.

3.8.1.5.1.3 Reinforcement Steel

Tension

In accordance with ASME Code, Section III, Subarticle CC-3422.1 (Reference 3.8-2), the design yield strength is limited to 60,000 psi and the allowable stress for load resisting purposes does not exceed $0.9f_y$. Under combined primary and secondary forces, the tensile strain in reinforcement may exceed $0.9\epsilon_y$.

In limited cases such as at the edge of large openings, a limited amount of yielding is permitted in accordance with the provisions in Subarticle CC-3422.1.

Compression

In accordance with ASME Code, Section III, Subarticle CC-3422.2 (Reference 3.8-2), the allowable stress does not exceed $0.9f_y$. In limited situations where the concrete is required to strain during development of design concrete capacity, the reinforcement is allowed to strain beyond the point of yield.

General Shear

See discussion in Subsection 3.8.1.5 for qualification of general shear capacity with factored loads.

Radial Shear

The radial shear provisions are in accordance with the ASME Code, Section III as stated in Subarticles CC-3421.4.2 “Prestressed Concrete” and CC-3521.2 “Radial Shear” (Reference 3.8-2).

Tangential Shear

Orthogonal tangential shear reinforcement is provided in accordance with the allowable stresses and the formulas in Subarticle CC-3521.1.1 (Reference 3.8-2).

Peripheral Shear

This type of shear loading is applicable to penetration areas and items such as the crane brackets. The allowable stresses used in the US-APWR design relative to the concrete shear capacity are as per ASME Code, Section III, Subarticle CC-3421.6 (Reference 3.8-2). Shear reinforcement is provided in accordance with Subarticle CC-3521.3, when applied shear exceeds that determined in Subarticle CC-3421.6.

Torsional Shear

Torsional shear reinforcement is provided in accordance with Subarticle CC-3521.4, when nominal torsional shear stresses exceed the allowable concrete torsional shear stresses determined in accordance with Subarticle CC-3421.7.

Radial Tension

Radial tension, as addressed in Subarticle CC-3545, exists in the through thickness direction in the outer portion of the cylinder wall and dome. Radial reinforcement is provided to resist the loads from this effect assuming no concrete tensile capability, even though this is not a Code requirement. Provision of this reinforcement also allows for an increase in compression stress allowable as stated in Note (2) in ASME Code, Section III, Table CC-3431-1 (Reference 3.8-2).

End Anchor Region

End anchor region requirements are stated in ASME Code, Section III, Subarticle CC-3543 (Reference 3.8-2). This section allows either calculations or testing for the determination of the required reinforcement.

3.8.1.5.2 Acceptance Criteria for Service Load Conditions

Service loads are any loads encountered during construction and in the normal operation of a nuclear power plant. Included in such loads are any anticipated transient or test loads during normal and emergency startup and shutdown of the nuclear steam supply, safety, and auxiliary systems. Also included in this category are those severe environmental loads which may be anticipated during the life of the facility.

The straight line theory of stress and strain is used based on assumptions specified in ASME Code, Section III, Subarticle CC-3511.2 (Reference 3.8-2).

3.8.1.5.2.1 Concrete

Membrane Compression, Tension and Bending

Subarticle CC-3431.1 and Table CC-3431-1 of the ASME Code, Section III (Reference 3.8-2) define the allowable concrete stresses for both membrane and membrane plus bending. The allowable stresses are defined for both primary and primary-plus-secondary service loads. Primary and secondary forces are defined in Subarticle CC-3136 of the ASME Code, Section III (Reference 3.8-2).

Table CC-3431-1 notes (2) and (3) state that if radial tension reinforcement is used in the cylinder wall and/or dome, the compression stress under initial prestress condition may be increased to $0.40f'_c$ and the normal allowable increased to $0.35f'_c$. For the PCCV, radial tension reinforcing is provided.

Concrete tensile strength is not relied upon to resist membrane and flexural tension forces in compliance with Subarticle CC-3431.2 of the ASME Code, Section III (Reference 3.8-2).

General Shear Capacity

The US-APWR complies with the ASME Code, Section III (Reference 3.8-2) requirement for qualification of concrete shear. Shear capacity is usually defined using two components. One component is that carried by the concrete defined as V_c, and the other, if required, is that carried by the reinforcing steel V_s. The total shear capacity being the sum of the two must be greater than the applied shear load. Concrete capacities are defined in Subarticle CC-3431.3, and steel reinforcement capacities are defined in Subarticle CC-3522 of the ASME Code, Section III (Reference 3.8-2).

Radial Shear

The radial shear provisions in the ASME Code, Section III are stated in Subarticle CC-3431.3 “Shear, Torsion, and Bearing” and Subarticle CC-3522 “Service Load Design” (Reference 3.8-2).

<u>**Tangential Shear**</u>

The US-APWR design is in accordance with ASME Code, Section III, Subarticle CC-3431.3 (Reference 3.8-2). Since only wind load generates tangential shear in the service load category, it should have no impact on the design.

<u>**Peripheral Shear**</u>

The US-APWR design complies with allowable stresses as identified in ASME Code, Section III, Subarticle CC-3431.3 (Reference 3.8-2).

<u>**Torsional Shear**</u>

The US-APWR design complies with allowable stresses as identified in ASME Code, Section III, Subarticle CC-3431.3 (Reference 3.8-2).

<u>**Radial Tension**</u>

Radial reinforcement is provided to resist the loads from this effect assuming no concrete tensile capability, even though this is not a Code requirement.

3.8.1.5.2.2 Prestressing System

<u>**Tendon**</u>

The allowable tendon stresses are defined in the ASME Code, Section III, Subarticle CC-3433. The tendons are allowed to be temporarily tensioned up to 0.80f_{pu} or 0.94f_{py}, whichever is less. The tension stress at the anchor point after seating is then allowed to be 0.73f_{pu}. The calculated average tension stress over the length of tendon (effective prestress after anchoring) is not to exceed 0.70f_{pu}.

<u>**End Anchor**</u>

ASME Code, Section III, Subarticle CC-3431.1 specifies concrete compression allowable under the tendon bearing plates. The anchorage components meet the requirements of Subarticles CC-2430, CC-2450 and CC-2460 (Reference 3.8-2).

<u>**Losses**</u>

The losses considered in the tendons are based on the items defined in ASME Code, Section III (Reference 3.8-2), Subarticle CC-3542. In addition, RG 1.35.1 (Reference 3.8-6) is used as guidance in the determination of prestressing losses. Prestressing losses are computed on the basis of 60-year design life.

3.8.1.5.2.3 Reinforcing Steel Systems

Tension

In accordance with ASME Code, Section III, Subarticle CC-3432.1 (Reference 3.8-2), the average tensile stress is limited to $0.5f_y$; however, provisions are included for increases under certain conditions.

Compression

In accordance with ASME Code, Section III, Subarticle CC-3432.2 (Reference 3.8-2), the compressive stress is limited to $0.5f_y$; however, provisions are included for increases under certain conditions.

General Shear

See discussion in Subsection 3.8.1.5 for qualification of general shear capacity with service loads.

Radial Shear

The radial shear provisions for the US-APWR are in accordance with the ASME Code, Section III, Subarticle CC-3431.3 "Shear, Torsion, and Bearing" and Subarticle CC-3522 "Service Load Design" (Reference 3.8-2).

Tangential Shear

The US-APWR design is in accordance with ASME Code, Section III, Subarticle CC-3522 (Reference 3.8-2). Since only wind generates tangential shear in the service load category, wind does not govern the design.

Peripheral Shear

The US-APWR design complies with allowable stresses as identified in ASME Code, Section III, Subarticle CC-3522 (Reference 3.8-2).

Torsional Shear

The US-APWR design complies with allowable stresses as identified in ASME Code, Section III, Subarticle CC-3522 (Reference 3.8-2).

Radial Tension

Radial tension, as addressed in ASME Code, Section III, Subarticle CC-3545 (Reference 3.8-2), exists in the through thickness direction in the outer portion of the cylinder wall and dome. Radial reinforcing is provided to resist the loads from this effect assuming no concrete tensile capability. Provision of this reinforcement also allows for an increase in compression stress allowable as stated in Note (2) in ASME Code, Section III, Table CC-3431-1 (Reference 3.8-2).

End Anchor

End anchor region requirements are stated in ASME Code, Section III, Subarticle CC-3543 (Reference 3.8-2). This section allows either calculations or testing for the determination of the required reinforcement.

3.8.1.5.3 Acceptance Criteria with respect to Concrete Temperatures

The US-APWR complies with ASME Code, Section III, Subarticle CC-3440 (Reference 3.8-2), which states temperature limits for the concrete temperature for normal and accident conditions, as follows.

a. For normal operation or any other long-term period, the temperatures are not to exceed 150° F except for local areas, such as around a penetration, which are allowed to have increased temperatures not to exceed 200°F.

b. For accident or any other short-term period, the temperatures are not to exceed 350°F for the interior surface. However, local areas are allowed to reach 650°F from steam or water jets in the event of a pipe rupture.

c. There are provisions to exceed these limits provided the design accounts for reduction in concrete strength and it can be proven there will not be concrete deterioration.

3.8.1.5.4 Acceptance Criteria for Impactive and Impulsive Loading

Yield strain and displacement values are permitted to exceed general stress and strain limits due to impactive and impulsive loading. In the case of impulse loads the usable ductility is 33% of the failure value and for impact effects the usable ductility is 67% of the failure value in accordance with ASME Code, Section III, Subarticle CC-3920 (Reference 3.8-2) for the design of the PCCV. Examples of impactive and impulsive loading include loading due to high-energy piping line breaks, localized yielding due to jet impingement and whip restraint loads, and external and internal missile loading. The design of containment internal structure is addressed in Subsection 3.8.3. General design for missiles is addressed in Section 3.5. A detailed discussion of those piping systems that exhibit LBB performance is given in Section 3.6.

3.8.1.5.5 Acceptance Criteria for Liner System

Liner Plate

The acceptance criteria for the PCCV liner plate are the stress and strain limits specified in the ASME Code, Section III (Reference 3.8-2), Table CC-3720-1, when considering the load combinations stated in Table CC-3230-1 with load factors of 1.0.

Liner Anchors

The acceptance criteria for the liner anchors are the force and displacement allowable values given in ASME Code, Section III (Reference 3.8-2), Table CC-3730-1. The allowable displacements used are based on percentages of ultimate break displacement values obtained from shear load and pull-out testing of the liner anchorage system.

Penetration Assemblies

The acceptance criteria are the design allowables given in ASME Code, Section III (Reference 3.8-2), Subarticles CC-3740 and CC-3820.

In accordance with Subarticle CC-3740(b), the design allowables for penetration nozzles are the same as used for ASME Code, Section III, Division 1, where a nozzle is defined as that part of the penetration assembly not backed by concrete.

In accordance with Subarticle CC-3740(c), the design allowables for the liner in the vicinity of the penetration are the same as those given in the AISC Code for resisting mechanical loads in the service load category. For factored load categories, the allowables are increased by a factor of 1.5, except for impulse loads and impact effects.

In accordance with Subarticle CC-3740(d), the portion of the penetration sleeves backed by concrete is designed to meet the acceptance criteria described above for the liner plate and anchors. Additionally, consistent with requirements in Subarticle CC-3820, to verify acceptability, the structural capacities of penetration assemblies that are designed for pipe loads are compared against either (a) the ultimate moment, axial, torque, and shear loadings that the piping is capable of producing, or (b) penetration loads based on a dynamic analysis considering pipe rupture thrust as a function of time. In the case of (b), penetration designs are later verified using results of piping analysis to assure the load used in the design is not exceeded.

Typically for the US-APWR, in order to preclude pipe rupture effects, flued heads are used for high-energy piping if large applied pipe rupture design loads are anticipated. Detailed discussion on this topic is provided in Section 3.6.

Brackets and Attachments

The allowables given in the ASME Code, Section III (Reference 3.8-2), Subarticles CC-3650 and CC-3750 are used as the acceptance criteria for brackets and attachments to the liner.

The US-APWR design avoids the use of brackets and similar items that transmit loads to the liner in the through-thickness direction. As much as practical in the design of attachments that have structural components carrying major loads, for example the upper plates of crane brackets, such a structural component of the attachment is made continuous through the liner. When through-thickness liner loads cannot be avoided and the liner is 1 in. or more in thickness, then the special welding and material requirements of Subarticle CC-4543.6 are applied. In addition to the requirements given in Subarticle CC-4543.6 (a) through (d), ultrasonic examinations are required prior to fabrication to preclude the existence of laminations in the installed material.

3.8.1.6 Material, Quality Control, and Special Construction Techniques

The major materials that are used for the design of the PCCV are defined herein. It is the responsibility of the COL Applicant to assure that any material changes based on site-specific material selection for construction of the PCCV meet the requirements specified in ASME Code, Section III (Reference 3.8-2), Article CC-2000, and supplementary requirements of RG 1.136 (Reference 3.8-3) as well as SRP 3.8.1 (Reference 3.8-7).

Quality control programs are in accordance with applicable portions of Articles CC-4000 and CC-5000 of the ASME Code, Section III (Reference 3.8-2). Additional quality assurance requirements are also implemented as provided by RG 1.136 (Reference 3.8-3). Chapter 17 provides additional discussion of the QAP.

The information listed below is specifically for the PCCV and does not preclude the selection of site-specific material provided that they are rectified with the standard design and meet the ASME Code, Section III (Reference 3.8-2), SRP 3.8.1 (Reference 3.8-7), and RG 1.136 (Reference 3.8-3) requirements.

Concrete

The concrete constituents and concrete mix design comply with the requirements of Article CC-2200 of the ASME Code, Section III (Reference 3.8-2).

Cement used in the concrete conforms to the requirements of ASTM C 150, Specification for Portland Cement, Type I, Type II, Type IV, Type V, or ASTM C 595, Specification for Blended Hydraulic Cements, Type IP, Type IP (MS), or Type (MH).

Aggregates used in the concrete conform to the requirements of ASTM C 33, Specification for Concrete Aggregates (Reference 3.8-44).

Mixing water used in the concrete conforms to the requirements of Subarticle CC-2223 of the ASME Code, Section III (Reference 3.8-2).

Admixtures include air-entraining admixtures, chemical admixtures, and mineral admixtures. The admixtures, except mineral admixtures, are stored in a liquid state. Air-entraining admixtures conform to the requirements of ASTM C 260, Air-Entraining Admixtures for Concrete (Reference 3.8-76).

Mineral admixtures conform to the requirements of ASTM C 618, Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete (Reference 3.8-77). Chemical admixtures conform to the requirements of ASTM C 494, Chemical Admixtures for Concrete (Reference 3.8-78).

Compressive Strength

The concrete design compressive strength for the PCCV is f'_c = 7,000 psi

The concrete design compressive strength for the basemat is f'_c = 5,000 psi

As previously discussed in Subsection 3.8.1.5, concrete is not allowed to rely on tensile strength to resist flexural and membrane tension except where permitted in ASME Code, Section III (Reference 3.8-2) allowable shear provisions. The concrete creep for the 60 year design life is 400μ in/in; for purposes of design, it is considered that 2/3 of this occurs in the first year after completion of prestressing. The concrete shrinkage for the 60 year design life is 150μ in/in; for purposes of design, it is considered that 2/3 of this shrinkage occurs in the first year after completion of concrete placement. The concrete specification defines the concrete constituents such as aggregates, cement, water, and admixtures that constitute the mix design, cement grout, and production testing requirements. The materials comply with the requirements of Article CC-2200 of the ASME Code, Section III (Reference 3.8-2).

Additionally, it is the responsibility of the COL Applicant to determine the site-specific aggressivity of the ground water/soil and accommodate this parameter into the concrete mix design as well as into the site-specific structural surveillance program. As required by SRP 3.8.1 (Reference 3.8-7), for plants with nonaggressive ground water/soil (i.e., pH is greater than 5.5, chlorides are less than 500 ppm, and sulfates are less than 1,500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and basemats is to (1) examine the exposed portions of below-grade concrete for signs of degradation, when excavated for any reason; and (2) conduct periodic site monitoring of ground water chemistry, to confirm that the ground water remains nonaggressive. For plants with aggressive ground water/soil (i.e., exceeding any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible, below-grade concrete for signs of degradation.

Liner Plate System

Liner Plate

The steel liner plate is designed as SA-516 Grade 60, 1/4 in. minimum thickness.

Where thickened for embedded plates, attachment bracket locations, openings, penetrations, and other such applications, the steel liner plate is SA-516 Grade 70. Grade 60 is used where justified in the design with respect to acceptance criteria previously discussed in Subsection 3.8.1.5.

The ASME Code, Section III (Reference 3.8-2) does not specifically require a corrosion allowance for the liner, and none is provided. The design of the PCCV is sufficient to prevent significant corrosion by protecting the liner against a corrosive environment. A suitable protective coating such as an epoxy coating is applied where necessary for corrosion protection, where suitability implies that the coating is DBA/LOCA-certified, resistant to break-down due to radiation exposure, and easily decontaminated. Further, corrosion allowance is accounted for in the design by demonstrating sufficient margin on the thickness to accommodate a small amount of corrosion that may occur over the 60-year design life.

The Liner Plate System specification complies with Article CC-2500 of the ASME Code, Section III (Reference 3.8-2). Fracture toughness requirements for the liner plate material are in accordance with Subarticle CC-2520 (Reference 3.8-2).

The PCCV is a prestressed concrete containment structure designed to the requirements of ASME Section III Division 2. The portion of the metallic penetrations that are in contact with the concrete shall be designed in accordance with the ASME Section III Division 2, Subarticle CC-3740. The areas of the metallic penetrations, e.g., piping nozzles, equipment hatch, personnel airlocks, and fuel transfer tubes, that form part of the pressure boundary, or are appurtenances attached to the load bearing sleeves/nozzles, shall be designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III Division 1 Subarticle NE (Class MC components) provided as Reference 3.8-48. The jurisdictional boundary between the two codes shall be at the location where the sleeve is not directly supported by the concrete.

Liner Anchor System

The liner anchors that are tees, angles, flat bars, and miscellaneous shapes are SA-36 structural steel.

Penetration Assemblies

Penetration assembly thickened plates are SA-516 Grade 70. Grade 60 may be used in some places where justified in the design. Penetration pipe sleeves/nozzles are SA-333 Grade 6, SA-516 Grade 70 or SA-312 TP304. Flat head and collar material used at small-bore pipe penetrations (less than 3 in. nominal diameter) is SA-516, or any material listed in Appendix I of the ASME Code, Section III (Reference 3.8-2), which is compatible with the penetration nozzle and piping in terms of weldability.

Brackets and Attachments

Brackets and Attachments are SA-36 structural steel.

Miscellaneous

The use of leak chases, although not an ASME Code requirement, is employed on the US-APWR in locations where the liner plate pressure boundary welds are not accessible after completion of construction. Leak chase material is SA-36 structural steel or any other acceptable material in Mandatory Appendix I of the ASME Code, Section III (Reference 3.8-2).

Prestressing System

The material chosen for the design of the tendons meets the requirements of Article CC-2400 of the ASME Code, Section III (Reference 3.8-2). The prestressing system is designed as a strand system, however, the system material may be switched to a wire system at the choice of the COL Applicant. If this is done, the COL Applicant is to adjust the US-APWR standard plant tendon system design and details on a site-specific basis. The ultimate capacity of an individual tendon as designed is 2.9 million pounds; however, it may be supplied within a plus or minus 5% tolerance, which is accounted for in the prestressing and overall design.

All tendons are unbonded (ungrouted) and have the capability to be detensioned and retensioned to a higher value and have a wire or strand removed after detensioning during a tendon surveillance operation.

Tendon Material

A strand system is utilized for the US-APWR standard plant design based on the following description of material requirements:

- The strand systems are fabricated from ASTM A416, Grade 270 #15, 0.6 in. diameter strands. The strands are of the low relaxation type. The relaxation losses are documented by a minimum of 3 manufacturer's tests performed as required by ASME Code, Section III (Reference 3.8-2), Subarticle CC-2424 and under the conditions as specified by ASTM A416.

If a wire system is selected, the design is reviewed and prestressing system details adjusted to accommodate the following wire system material requirements:

- Wire systems are fabricated from ASTM A421, Type BA, 1/4 in. diameter solid wire. The wire is of the low relaxation type. The relaxation losses are documented by a minimum of three manufacturer's tests performed as required by ASME Code, Section III (Reference 3.8-2), Subarticle CC-2424 and under the conditions as described in ASTM A421 supplementary requirements for low-relaxation wire.

For either tendon system, the relaxation losses are not more than 2.5% when initially loaded to 70% of the minimum breaking strength or not more than 3.5% when loaded to 80% of specified minimum breaking strength of the strand after 1,000 hours of testing. The temperature of the test specimens are maintained at 68° ± 3.5°F. As recommended by RG 1.35.1 (Reference 3.8-6), there are to be a sufficient number of data points in each of the three tests to extrapolate the data to the 60 year design life of the PCCV at a sustained temperature of 90°F. The extrapolation is performed using regression analysis.

For both systems, as recommended in RG 1.35.1 (Reference 3.8-6) Section 2.3, the design provides allowances to accommodate breakage (during construction) of individual wires or strands in the tendons, on both an overall as well as a localized basis.

<u>Anchorage Components</u>

The tendon end anchorage material selected in the design meets the requirements of ASME Code, Section III (Reference 3.8-2), Subarticle CC-2430. Additional material requirements per RG 1.136 (Reference 3.8-3) follow.

- The specification that defines the material and special material testing requirements for the Prestressing System complies with Article CC-2400 of the ASME Code, Section III (Reference 3.8-2) for items where applicable.

In addition to the requirements of ASME Code, Section III (Reference 3.8-2), Subarticle CC-2433.2.3, "Acceptance Standards," the following guidance per RG 1.136 (Reference 3.8-3) is used:

- The maximum hardness for material of anchor head assemblies and wedge blocks are not to exceed that of Rockwell C40. To maintain uniformity in hardness, the tolerance on a designated hardness number does not exceed ± 2.

In addition to the requirements in ASME Code, Section III (Reference 3.8-2), Subarticle CC-2434, "Wedges and Anchor Nuts," the following guidance is used to protect prestressing materials from low-temperature effects:

- Materials for all load-bearing components of prestressing systems should be selected so that they can withstand the anticipated low-temperature effects without a loss in their ductility. Methods and procedures similar to those used for materials of liners in Subarticle CC-2520, “Fracture Toughness Requirements for Materials,” are acceptable for qualifying the materials. Additionally, suitable tests should be conducted to demonstrate that with the maximum allowable flaw size (cracked button heads, wedges, and anchor nuts); the specific components exhibit the required strength and ductility under the lowest anticipated temperatures.

In addition to the requirements of ASME Code, Section III (Reference 3.8-2), Subarticle CC-2463.1, “Static Tensile Test,” the following guidance is used: Any system of prestressing should be subjected to a sufficient number of tests to establish its adequacy. Justification that a sufficient number of tests have been performed, as well as a description of the test program, should be available for NRC review.

Nonload-Carrying and Accessory Materials

Tendon duct, channel, trumpet, and transition cone material meets the requirements of ASME Code, Section III (Reference 3.8-2), Subarticle CC-2440. Corrosion prevention coatings are required for unbonded tendons and are in accordance with Subarticle CC-2442.

Reinforcing Steel Systems

The material meets the requirements of Article CC-2300 of the ASME Code, Section III (Reference 3.8-2).

Splicing material also meets the requirements of Article CC-2300 of the ASME Code, Section III (Reference 3.8-2).

All material for the reinforcing steel system including bars and splices conforms to Article CC-2300 of the ASME Code, Section III (Reference 3.8-2).

3.8.1.7 Testing and Inservice Inspection Requirements

Structural integrity testing of the PCCV is performed in accordance with Article CC-6000 of the ASME Code, Section III (Reference 3.8-2), RG 1.35 (Reference 3.8-5), and RG 1.35.1 (Reference 3.8-6). The testing meets the same requirements for ILRT and Containment Leakage Testing as given in RG 1.206 Subsection C.I.6.2.6 (Reference 3.8-1).

Preoperational structural testing is performed for the overall PCCV, equipment hatch and personnel airlocks in accordance with Article CC-6000 of the ASME Code, Section III (Reference 3.8-2).

It is the responsibility of the COL Applicant to establish programs for testing and ISI of the PCCV, including periodic inservice surveillance and inspection of the PCCV liner and prestressing tendons in accordance with ASME Code Section XI, Subsection IWL (Reference 3.8-4).

Chapter 6 defines the ILRT requirements for the overall PCCV in addition to ILRT requirements for the penetrations and openings and containment isolation valves. The ILRT program meets the requirements of 10 CFR 50, Appendix J (Reference 3.8-18). Chapter 6 discusses the test and instrument plan, frequency of measurements, structural response predictions, and any other necessary requirements in accordance with Article CC-6000 of the ASME Code, Section III (Reference 3.8-2).

Specific structural requirements for the Structural Integrity Test of the PCCV are summarized as follows:

Displacement Measurements

Displacement measurements of the PCCV as defined in ASME Code, Section III (Reference 3.8-2) Subarticle CC-6360 meet the following provisions.

- Radial displacements of the cylinder are measured at a minimum of five approximately equally spaced elevations located at 20%, 40%, 60%, 80%, and 100% of the distance between the base and the spring line. These measurements are made at a minimum of four approximately equally spaced azimuths. Measurement of the total displacement may be made between diametrically opposite locations on the PCCV wall. The radial displacement may be assumed to be equal to one-half of the measured change in diameter.

- Radial displacements of the PCCV wall adjacent to the largest opening, are measured at a minimum of 12 points, four equally spaced on each of three concentric circles. The diameter of the inner circle is just large enough to permit measurements to be made on the concrete rather than on the steel sleeve; the middle approximately 1.75 times the diameter of the opening; and the outer approximately 2.5 times the diameter of the opening. For hatch designs with thickened wall sections, the concentric circle at 1.75 times the diameter is relocated at the wall thickness discontinuity and the remaining circle is relocated approximately two wall thicknesses outside the discontinuity. The increase in diameter of the opening is measured in the horizontal and vertical directions. If other openings require structural verification as determined by the designer, displacement measurements are made in the same manner as stipulated for the largest opening.

- Vertical displacement of the top of the cylinder relative to the base is measured at a minimum of four approximately equally spaced azimuths.

- Vertical displacements of the dome of the PCCV are measured at a point near the apex and two other approximately equally spaced intermediate points between the apex and the spring line on at least one azimuth.

Concrete Crack Observations

At a minimum the following areas are observed based on the techniques defined in Subarticles CC-6225 and CC-6350 of the ASME Code, Section III (Reference 3.8-2) at these locations:

- The top or bottom of the equipment hatch opening at the edge of the opening.
- The top or bottom of the equipment hatch opening where the thickened area meets the normal shell.
- The center elevation of the equipment hatch opening where the thickened area meets the normal shell, on both sides.
- The cylinder where it intersects the basemat with the longer direction being 3 times the wall thickness.
- The cylinder midheight where it intersects the vertical buttress.
- At a typical cylinder midheight location away from buttresses and openings.
- The cylinder dome intersection.
- In the dome at about 45 degrees from the springline, where there are two overlapping sets of tendons (i.e., one vertical dome and one hoop dome).
- In the dome at about 45 degrees from the springline, where there are three overlapping sets of tendons (i.e., two vertical dome and one hoop dome).
- At the dome apex.

In general surveillances are scheduled after the structural integrity test starting at 1, 3, and 5 years and every 5 years thereafter. There is some flexibility to this as stated in ASME Code, Section XI, Subarticle IWL-2400 (Reference 3.8-4).

<u>Sample Selection</u>

ASME Code, Section XI (Reference 3.8-4) requires that measurements and sampling be performed on randomly selected tendons. The PCCV tendons are detensionable and are in compliance with this requirement.

Tendons are to be placed in groups with similar characteristics. For the US-APWR, the two basic groups are the inverted U-tendons and hoop tendons, which consist primarily of cylinder hoop tendons and also a smaller number of dome hoop tendons. The minimum requirements for sample selection are as discussed below. RG 1.35 (Reference 3.8-5) requires a 4% sample with a minimum of four tendons per group and ASME Code, Section XI (Reference 3.8-4) agrees, but there is some relaxation after 10 years. The 4% sample is taken as four inverted U tendons and six horizontal hoop tendons. The six hoop tendons are divided into four cylinder and two dome hoop tendons. This amount also satisfies the minimum number. Two dome hoop tendons are provided in the design as sample tendons since the dome hoop tendons are of a slightly different geometry/configuration than the cylinder hoop tendons. Sample selection for testing of tendons is made on a random basis. Thus, the design permits two tendons from each group to be detensioned for strand removal. Also, the design considers that two additional strands are detensioned during construction due to other reasons.

Acceptance Standards

The acceptance standards for both the RG and ASME Code, Section XI (Reference 3.8-4) are similar and both are satisfied. RG 1.35.1 (Reference 3.8-6) gives guidance on how to determine the tendon prestress loss curve as a function of time. The prestress loss curve is determined based on regression analysis. For the US-APWR PCCV, the curve is for 60 years. A correction is allowed to account for initial installation force variation and elastic losses resulting from when in the prestressing sequence the tendon is tensioned. The acceptance criteria listed below are for values after these corrections have been applied, except the last five items, which apply regardless of corrections.

- The average lift-off of each group is equal to or above the minimum required prestress. For the PCCV, this would mean the average of all four inverted U tendons in the group and then the six hoop tendons in that group. The groups have different force time loss curves.
- For each tendon the measured lift-off value is equal to or above the predicted value at that surveillance time on the curve.
- An extrapolation of the average previous surveillance and the average current surveillance shows that the next surveillance has forces that are above or equal to the next surveillance for each group.
- The elongation during re-stressing does vary by more than 10% from the initial installation value.
- The test results for the removed strand or wire meet the applicable ASTM requirements for yield strength, ultimate strength and elongation.
- The corrosion protection material is in accordance with the applicable standards.
- The tendon anchorage areas do not show evidence of active corrosion and steel items do not show cracking or other deterioration.
- The concrete in the anchor head area does not show unacceptable cracking or any other deterioration.
- There is no evidence of free water in the prestressing system.

Additional Required Actions and Responsibilities

If any of the conditions listed above are not satisfied, an investigation and additional action must be taken for the required items, which are listed in the two applicable referenced documents.

3.8.2 Steel Containment

The US-APWR does not utilize a steel containment. Portions of the US-APWR design which fall under Division 1 of the ASME Code, Section III (Reference 3.8-2), which are pressure-retaining but not backed by concrete, have been discussed previously in Section 3.8.1.

3.8.3 Concrete and Steel Internal Structures of Concrete Containment

Concrete and steel structures internal to the PCCV, but not part of the containment pressure boundary, provide support of the RCS components and related piping systems and equipment. The containment internal structure is the primary support structure that provides compartmentalization and radiation shielding within the PCCV. The major structures internal to containment include:

- Reactor support system
- SG support system
- RCP support system
- Pressurizer support system
- Primary shield wall as part of containment internal structure
- Secondary shield walls as part of containment internal structure
- Reactor cavity and refueling cavity as part of containment internal structure
- Other structures internal to containment include additional supports, RWSP, the operating floor, intermediate floors and platforms, and polar crane supporting elements

These structures internal to containment are capable of resisting the design loads and load combinations to which they may be subjected. The containment internal structure mitigates the consequences of an accident by protecting the containment and other engineered safety features from the effects induced by an accident, such as jet impingement forces and whipping pipes.

3.8.3.1 Description of the Structures Internal to Containment

3.8.3.1.1 Reactor Vessel Support System

The RV support system consists of eight steel support pads which are integrated with the inlet and outlet nozzle forgings. The support pads are placed on support brackets, which are supported by an embedded steel structure on the primary shield wall elevation 35 ft, 10.87 in. The support system is designed for operating and accident load cases caused by seismic and postulated pipe rupture, including LOCAs. The supports are formed by sliding surfaces between the shim plates and support pads to allow radial thermal growth of the RCS and RV. The vessel position is maintained unchanged by controlling the horizontal load through the support brackets and the base plate. Figure 3.8.3-1 provides the detail of the RV supports and relationship with the primary shield wall.

3.8.3.1.2 Steam Generator Support System

The SG support system consists of an upper shell support structure at centerline elevation 96 ft, 7 in., an intermediate shell support structure at centerline elevation 75 ft, 5 in., and a lower shell support structure at centerline elevation 45 ft, 7.64 in.

The upper and intermediate shell supports are lateral restraints utilizing snubbers attached to structural steel brackets, while the lower support structure is constructed entirely of structural steel and provides both vertical and lateral support. All support systems are designed considering thermal expansion of piping. The support system also restrains horizontal movement of the SG in the event of earthquake or other DBAs.

Four columns transfer the vertical loads of the SG to the reinforced concrete slab at elevation 25 ft, 3 in. The upper and lower ends of the columns are pin-jointed to permit movement of the SGs caused by thermal expansion of piping. Figure 3.8.3-2 depicts the SG support system.

3.8.3.1.3 Reactor Coolant Pump Support System

Each RCP support system consists of a lateral support structure, and three support columns.

The lateral support structure at centerline elevation 42 ft, 7.3 in. is constructed entirely of structural steel. Both support structures are designed considering thermal expansion of piping. The support structure also restrains horizontal movement of the RCPs in the event of an earthquake or other DBAs.

The three support columns carry the vertical loads of the RCP from the reinforced concrete slab at elevation 25 ft, 3 in. The upper and lower ends of the supports are pin-jointed to permit movement of the pumps caused by thermal expansion of piping. Figure 3.8.3-3 depicts the RCP support system.

3.8.3.1.4 Pressurizer Support System

The pressurizer is supported by an upper support structure and a lower support skirt. The upper support structure constructed of four structural steel struts at centerline elevation 110 ft, 9 in. does not restrain movement by thermal expansion, but restrains horizontal movements in the event of design-basis earthquakes or accidents. The lower support structure supports the vertical load through a continuous structural steel skirt welded to the bottom of the pressurizer supported at elevation 59 ft, 1 in. Figure 3.8.3-4 depicts the pressurizer support system.

3.8.3.1.5 Primary Shield Wall

The RV is located at the center of the PCCV. Primary shield walls form the perimeter around the RV, which also serve to support the RV at elevation 35 ft, 10.87 in. The top of primary shield wall elevation is 46 ft, 11in. The general arrangement drawings in Chapter 1 show the location and configuration. Isometrics of the primary shield walls are shown in Figure 3.8.3-5.

The primary shield wall and other walls inside containment are fabricated as steel-concrete (SC) module walls. The modules are formed using permanently placed carbon steel faceplates and web-plates with a nominal thickness of 1/2 in. The faceplates, connected by tie bars, fabricated from carbon steel plate, or by carbon steel web-plates, also function as formwork for concrete placed in the interior. The primary purpose of the tie bars and web-plates is to stiffen and hold together the faceplates during handling, erection, and concrete placement, and to provide out-of-plane shear strength. The nominal pitch of the tie bars is 24 in. for the SC module walls. The primary functions of the web-plates are to mitigate faceplate stress concentration, maintain the SC module configuration, and stiffen corners of faceplates. Shear studs are welded to the steel faceplates to provide shear transfer between the steel plates and the concrete. Where SC modules intersect, web-plates are installed in-line with faceplates to maintain continuity across the point of intersection. The nominal pitch of studs is 8 in. to 12 in. in both directions. Faceplates are welded to adjacent plates with full penetration welds so that the weld is at least as strong as the plate. The SC module walls are welded at the base to a continuous embedded plate in the basemat. After erection, concrete is placed between the faceplates.

3.8.3.1.6 Secondary Shield Walls

The secondary shield walls surround the primary loops from the SG compartments. SC modules also form supports for intermediate floors and operating floors. The secondary shield walls are a series of walls that enclose the SGs and the pressurizer. Each of the four secondary shield wall compartments provides supports and houses a SG and RCL piping. The GA drawings in Chapter 1 show the location and configuration. Isometrics of secondary shield walls are shown in Figure 3.8.3-5.

3.8.3.1.7 Refueling Cavity

The cavity space directly above the RV and between SC module walls to the north is referred to as the refueling cavity. The refueling cavity connects to the fuel transfer tubes that penetrate the north end of PCCV. The floor of the refueling cavity varies in elevation from 19 ft, 4 in. to 46 ft, 11 in. The top of the refueling cavity is 76 ft, 5 in. Additionally, containment racks are installed in the refueling cavity to temporarily store new or irradiated fuel assemblies. A more detailed description of the containment racks is provided in Section 9.1.

The walls of the refueling cavity are formed by SC modules, which are lined with stainless steel over the 1/2-in. thick carbon steel plates, referred to as “clad steel.” The ceiling and floor slabs are also lined with clad steel.

3.8.3.1.8 RWSP

The RWSP is located at the lowest part of the PCCV. The RWSP is formed by wall of SC modules using clad steel. A floor at elevation 3 ft, 7 in. is formed of clad steel in a layer of concrete that covers the containment liner and basemat. The ceiling is similarly lined with stainless steel. Subsection 6.2.1.1 provides a description of the RWSP layout and design features.

3.8.3.1.9 Interior Compartments

The containment internal structure includes several subcompartments designed to provide containment, radiation shielding, and protection of safety-related components. These compartments are formed by the secondary shield walls surrounding the primary loops from the SGs. They also protect the containment from postulated pipe ruptures inside the containment. These SC wall modules also form supports for intermediate floors and the operating deck at elevation 76 ft, 5 in. The walls are designed for load cases including earthquake and DBAs.

Subcompartments and/or rooms comprising the containment internal structure are summarized as follows:

- reactor cavity EL. -9 ft, 2 in.
- containment drain sump room EL. 9 ft, 6 in.
- letdown heat exchanger room EL. 25 ft, 3 in.
- regenerative heat exchanger room EL. 50 ft, 2 in.
- excess letdown heat exchanger room EL. 50 ft, 2 in.

Labyrinths are provided beside the shield wall openings at several elevations for radiation protection, which consist of SC modules and reinforced concrete walls, floors, and ceilings.

Reinforced concrete slabs are used for the floor above the RWSP at elevation 25 ft, 3 in., the intermediate floor at elevation 50 ft, 2 in., and the operating floor at elevation 76 ft, 5 in. The floors are shown on the GA drawings in Chapter 1. The floor at elevation 25 ft, 3 in. is supported by the primary shield wall, the secondary shield wall, and the RWSP. The floors at elevations 50 ft, 2 in. and 76 ft, 5 in. are supported by the secondary shield wall and the structural steel framing (beams and columns) arranged between the secondary shield wall and the PCCV. The floors consist of reinforced concrete slabs, placed on steel beams and deck plate.

3.8.3.1.10 SC Modules

Figure 3.8.3-5 provides isometric views of the SC modules.

The module framework, consisting of the steel faceplates prior to concrete placement, is positioned on the supporting reinforced concrete basemat. The SC modules are anchored to the basemat through reinforcement doweled with the slab. Seaming of adjacent plates is accomplished using full penetration welding that maintains full design strength of the plate units. The interior of the modular unit is filled with concrete to complete the installation process. Figure 3.8.3-6 depicts the containment internal structure compartment wall layout and configuration. Figure 3.8.3-7 provides typical details for the SC module construction including connection details and anchorage connection details to the reinforced concrete basemat.

3.8.3.1.11 Polar Crane Supports

An internal polar crane is supported by the PCCV. A continuous crane girder transfers the polar crane loads to the PCCV wall. Refer to Subsection 3.8.4.3 for loads applicable to the polar crane supports. Figure 3.8.3-8 depicts the polar crane supports layout and construction.

3.8.3.1.12 Structural Steel Framing

Structural steel framing within the interior of PCCV is primarily for support of floor slab, equipment, distribution systems such as piping, valves, and cable trays, and access platforms. Service platforms and secondary intermediate floors consist of steel grating or checkered plate supported by structural steel framing. All structural steel members are capable of resisting the loads and load combinations to which they may be subjected.

3.8.3.2 Applicable Codes, Standards, and Specifications

Refer to Subsection 3.8.4.2 for industry standards applicable to the design and construction of seismic category I structures inside containment. Other codes, standards and specifications applicable to materials, testing and inspections are identified in Subsections 3.8.4.6 and 3.8.4.7.

3.8.3.3 Loads and Load Combinations

Typical loads and load combinations are detailed in Subsection 3.8.4.3. Load combinations to be utilized for the design of the containment internal structure include hydrostatic, pressure, and thermal loads as summarized below. Hydrostatic loads reflect the water inventory and its location during various plant conditions.

Seismic category I concrete structures are designed for impulsive and impactive loads in accordance with the ACI 349-06 Code (Reference 3.8-8), with exceptions given in RG 1.142 (Reference 3.8-19). Impactive and impulsive loads must be considered concurrent with seismic and other loads (i.e., dead and live load) in determining the required load resistance of structural elements.

Subcompartment pressure loads are the result of postulated high-energy pipe ruptures. In determining an appropriate equivalent static load for Y_r, Y_j, and Y_m, elasto-plastic behavior is acceptable with appropriate ductility ratios, provided excessive deflections do not result in loss of function of any safety-related system.

3.8.3.3.1 Floor Loads Inside Containment

The following are the minimum values for live loads used in load combinations involving non-seismic loads. Live loads for the seismic analysis are defined in Subsection 3.8.4.3.

Containment operating deck	950 lb/ft^2 (during maintenance and refueling outages)
	200 lb/ft^2 (during normal operation)

Maintenance and service platforms	The load is calculated for individual locations based on the functional requirements and service equipment
All other floors (ground floor and elevated floors, including stairs and walkways)	200 lb/ft^2 (For non-seismic load combinations and for global seismic analysis, this load may be reduced if the equivalent live load on the floor is more than 50 lb/ft^2. The sum of the live load and equivalent live load need not exceed 250 lb/ft^2)

In design reconciliation analysis, if actual loads are determined to be lower than the above loads, the actual loads may be used for reconciliation. Floor live loads for design are not reduced below 100 lb/ft^2.

3.8.3.3.2 Liquid Loads (F)

The vertical and lateral pressures of liquids inside containment are treated as dead loads. Structures supporting fluid loads during normal operation and accident conditions are designed for the hydrostatic as well as hydrodynamic loads.

Hydrostatic loads are based on the tank or flooded volume. The water inventory is considered to be in any one of the following locations with other areas being dry.

RWSP	Water in the RWSP. Normal water level is elevation 20 ft, 2 in.
Refueling Cavity	Water in the refueling cavity during refueling operations. Normal water level during refueling is elevation 75 ft, 3 in.

The overall seismic analyses and ISRS considers the water to be in the RWSP, which is its normal location. Water inventory at any one of these locations is also considered as a normal operating liquid load. In the event of a SSE, the containment internal structure is designed with the water inventory in any one of the above locations.

The RWSP design also considers the hydrodynamic response of the refueling water under seismic excitation. The manner in which the impulsive and convective response components are considered is discussed in Subsection 3.8.3.4.2.

3.8.3.3.3 Accident Pressure Load (P_a)

Accident pressure loads within or across a compartment and/or building are considered in the design. Differential pressure is generated by postulated pipe rupture and includes the dynamic effects due to pressure time-history. The containment internal structure subcompartments are designed to the pressures shown in Table 3.8.3-2 and identified on Figure 3.8.3-9. These pressures are combined by SRSS with SSE loads, including sloshing loads, or by using more conservative combinations. The water inventory is assumed to be in the RWSP. Steel floors with grating need not be designed for differential pressure.

3.8.3.3.4 Operating Thermal Loads (T_o)

The normal operating environment inside and outside the PCCV is specified in Table 3.8.1-3. Under the normal operating condition, the primary shield wall, and the secondary shield wall (in the proximity of the main steam and feedwater pipes) experience temperature rises, including temperature distribution through the wall thicknesses. The loads resulting from these thermal gradients provided in Table 3.8.1-3 are combined with other loads for the containment internal structure as specified in the load combinations in Table 3.8.4-3.

3.8.3.3.5 Accident Thermal Load (T_a)

Thermal loads due to temperature gradients caused by the postulated pipe breaks are considered in the design. The temperature gradients are calculated using the temperatures corresponding to LOCA and main steam line break (MSLB) and are presented in Table 3.8.1-3. Local areas are designed for the elevated temperature effects and the loads resulting from the postulated accidents.

Temperatures during an accident do not exceed 450°F at the surface. However, local areas are allowed to reach 650°F from steam or water jets in the event of a pipe failure in accordance with Section E.4.2 of ACI 349-06 Appendix E. Although the 450°F accident temperatures exceed the 350°F surface temperature limit of ACI 349-06 Section E.4.2, the accident temperatures do not reduce the design strengths of the CIS SC modules. This assessment is described in Section 9.0 of Technical Report MUAP-11019 and Appendix B, Section 10 of MUAP-11013. For the reinforced concrete slabs in the CIS, temperatures during an accident do not exceed 350°F at the surface.

3.8.3.3.6 Accident Thermal Pipe Reaction (R_a)

Pipe and equipment reactions under thermal conditions are generated by the postulated pipe break and includes R_o (see Subsection 3.8.4.3).

3.8.3.3.7 Reaction Due to Pipe Ruptures (Y_r)

The load on a structure generated by the reaction of a ruptured high-energy pipe during the postulated event includes an appropriate dynamic load factor. The time dependent nature of the load and the ability of the structure to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of Y_r.

3.8.3.3.8 Jet Impingement (Y_j)

The load on a structure generated by the jet impingement from a ruptured high-energy pipe during a postulated event includes an appropriate dynamic load factor. The time-dependent nature of the load and the ability of the structure to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of Y_j. The dynamic load factor is calculated using a long duration step function for the load. The target resistance is idealized as bilinear elasto-perfectly plastic.

3.8.3.3.9 Impact of Ruptured Pipe (Y_m)

The load on a structure or a pipe restraint resulting from the impact of a ruptured high-energy pipe during the postulated event includes an appropriate dynamic load factor. The type of impact (i.e., plastic, elastic), together with the ability of the structure to deform beyond yield are considered in establishing the structural capacity necessary to resist the impact.

3.8.3.4 Design and Analysis Procedures

The CIS is a complex structure that includes several different structure categories. As discussed in previous sections, a significant portion of the CIS consists of SC walls, including the primary shield walls, the secondary shield walls, the walls of the refueling cavity, and the walls of the RWSP.

As presented in Technical Report MUAP-11005 (Reference 3.8-63), experimental investigations have been conducted in the past to evaluate the behavior of the SC walls with geometries representative of those in the US-APWR CIS, as follows:

- 1/10th scale cyclic pushover test of a complete CIS
- 1/6th scale cyclic pushover test of the primary shield structure
- Component in-plane shear tests of SC walls with flanges
- Component tests of SC wall panels without flanges subjected to combined axial compression and cyclic in-plane shear
- Component out-of-plane shear tests of SC beams
- Component axial compression test of SC stub columns
- Component tests on the effects of thermal gradients on cracking and mechanical behavior of SC walls

Technical Report MUAP-11005 Appendices A, B, C, and D explain the correlation of the SC wall geometries considered in these tests to the SC walls in the US-APWR CIS. In addition, the technical report describes the key results of these tests that demonstrates the performance of SC walls under the design loading conditions for the CIS, including seismic and thermal loading.

The experimental results presented in Technical Report MUAP-11005 (Reference 3.8-63) also demonstrate the similarity of SC wall behavior to that of standard reinforced concrete walls. SC walls are similar to reinforced concrete walls, as they both consist of thick concrete sections that are reinforced by steel. In SC walls, the concrete section is reinforced with steel faceplates that are anchored to the concrete using shear studs and connected to each other using tie bars. In reinforced concrete walls, the concrete section is reinforced with orthogonal grids of steel rebars that are embedded within the concrete.

In several aspects of structural behavior, such as axial tension, compression, flexure, and out-of-plane shear, the behavior of SC walls is very similar to that of reinforced concrete walls. In other aspects (e.g., in-plane shear and thermal effects), the general behavior is similar to that of reinforced concrete, but there are some differences that must be addressed in the design of the SC walls.

The design of the CIS SC walls is based on ACI 349-06 (Reference 3.8-8) code provisions. The overall approach for confirming the applicability of the ACI 349-06 (Reference 3.8-8) code equations, evaluating the results of the small-scale (1/10th and 1/6th scale) tests, and developing SC wall section details that prevent SC-specific limit states not specifically addressed in the ACI 349-06 code were evaluated as described in Technical Report MUAP-11013 (Reference 3.8-68).

The results of the 1/6th scale and 1/10th scale test results have been evaluated and analyzed to confirm the performance of containment internal structures constructed with SC modules under seismic loading up to SSE and beyond SSE loading levels. Additionally, the component-level tests were also evaluated using benchmarked nonlinear analysis methods to confirm that the behavior of SC walls is appropriately addressed using ACI 349-06 provisions. The benchmarked nonlinear analysis of these test results confirm the behavior of the SC modules, but is not a basis of design. The analysis is summarized in MUAP-11013 Appendix A.

To further confirm the applicability of the ACI 349-06 design provisions for the US-APWR specific SC module design details, a series of confirmatory physical tests were conducted, as summarized in Table 3.8.3-7. The results of these tests demonstrate behavior of the SC walls and confirm the conservatism of the ACI 349-06 design strength equations. The US-APWR confirmatory testing program is further summarized in Technical Report MUAP-11013 Appendix B.

The analysis and design procedures for the CIS are organized into three sets of criteria, as follows:

Stiffness and Damping: The stiffness and damping terms used for analysis of the CIS are defined for six structure categories and two basic loading conditions, as described in Table 3.8.3-4 and following in this section. This is also summarized in detail in Technical Report MUAP-11018 (Reference 3.8-70).

SC Wall Design Criteria: The design criteria for the US-APWR SC walls address the SC specific design issues and limit states observed in the experimental database, and present the detailing approaches required to prevent these limit states from governing the design. The design criteria also addresses the applicability of the ACI 349-06 code provisions for each loading condition, including axial tension, axial compression, out-of-plane flexure, out-of-plane shear, in-plane shear, design for combined forces, and accident thermal considerations. Based upon observation of behavior in the experimental research, conservative forms of the ACI 349-06 (Reference 3.8-8) code provisions are identified as required. The key aspects of these design procedures are summarized in Subsection 3.8.3.4.5, and in greater detail in Technical Report MUAP-11019 (Reference 3.8-71).

SC Wall Connection Design and Detailing: The design criteria for SC wall connection design and detailing addresses design procedures for all anchorages and connections in the CIS involving SC walls. The criteria includes two connection design philosophies that are intended to ensure sufficient strength and ductility of the SC wall connections. These include the full-strength design philosophy, which designs the connection to develop the expected strength of the weaker of the connected parts, and the overstrength design philosophy, which provides significant overstrength (e.g., 200%) with respect to the design demands on the connection. The full-strength design philosophy is intended to be used for all SC wall connections in the US-APWR CIS. The overstrength design philosophy is to be utilized only in limited circumstances where a full strength connection cannot be provided. The design criteria are in accordance with ACI 349-06 provisions for anchorages and connections. In addition, the criteria require that the SC wall anchorage connection to the basemat (e.g., welding faceplates and studs to baseplate and couplers to baseplate) be designed per the provisions of both ACI 349-06 and ASME Section III Division 2 because this connection is at a jurisdictional boundary with the containment pressure boundary. Three connections are designed as representative using the full strength design approach, as summarized in Subsection 3.8.3.5.2. The SC wall connection design and detailing criteria are summarized in further detail in Technical Report MUAP-11020 (Reference 3.8-72).

<u>Summary of Stiffness and Damping for Analysis</u>:

The containment internal structure is unique among the R/B complex structures in that it is comprised of a number of different structural types. The structural types include composite SC walls of varying thickness, massive reinforced concrete sections, and reinforced concrete slabs. These structures experience varying levels of stress and resultant concrete cracking under the seismic and accident thermal loading applied to the containment internal structure. Each structural type exhibits unique stiffness and damping characteristics before and after cracking. Thus, it is not appropriate to apply a uniform stiffness reduction to the entire containment internal structure for the SSI analyses of the R/B complex. Each structural component is assigned stiffness and damping values appropriate for its structural type and estimated cracking levels. This assignment is simplified by grouping structural components into six structural categories with common behavior. Stiffness and damping values are then defined for each category under two basic loading conditions that encompass the full range of stresses and resultant cracking anticipated for the containment internal structure seismic response.

The six structural categories defined for stiffness and damping characterization are described below and summarized in Table 3.8.3-4. As discussed in Technical Report MUAP-11018 (Reference 3.8-70), the values are derived from supporting experimental data for the SC modules and from industry standards for reinforced concrete structures. Plan and elevation views illustrating the use of each of the six structural categories are presented in Figures 3.8.3-12 through 3.8.3-18.

Overall thicknesses of the single-celled SC walls vary from 36” to 67”, while the multi-celled primary shielding SC walls have overall thickness in excess of 9’-11”. The range of experimental data establishing the composite stiffness characteristics of SC walls is applicable to sections with overall thickness less than or equal to 56” and steel plate reinforcement ratio (ρ) greater than 1.5%.

$$\rho = 2 \cdot t_p / T > 0.015$$

Where

t_p = faceplate thickness,

T = overall wall thickness

The SC walls are separated into three categories, as follows:

CIS Category 1: SC Walls with thickness less than or equal to 56 in. These SC walls have material and geometric parameters that are within the range of the experimental database. This category includes the majority of the secondary shielding walls in the containment internal structure. The most common SC wall is 48 in. thick with 0.5 in. thick steel faceplates.

CIS Category 2: SC Walls with thickness greater than 56 in. This category includes a relatively small portion of the containment internal structure SC walls with thicknesses ranging from 58.5 in. to 67 in.

CIS Category 3: Primary Shield Walls. The primary shield walls below elevation 35'-11" range in thickness from 9'-11" to 15'-4". They have a multi-cellular arrangement comprised of two steel faceplates, a mid-thickness steel plate, and numerous transverse web plates.

Non-SC structural components of the CIS are separated into three additional categories, as follows:

CIS Category 4: Reinforced concrete slabs. Standard reinforced concrete floor slabs are used at various elevations throughout the containment internal structure.

CIS Category 5: Massive reinforced concrete. This category includes the thick reinforced concrete blocks at the base of the containment internal structure that support the steam generators and reactor coolant pumps. These blocks are nominally 8 to 32 feet deep and are anchored to the basemat of the reactor building complex with steel reinforcement.

CIS Category 6: Steel structures with nonstructural concrete infill. These structures consist of steel plates or steel shape grillages with nonstructural concrete provided for shielding purposes.

Discussion of Basic Loading Conditions for Consideration of Concrete Cracking:

The design loading conditions for the CIS are condensed to two basic loading conditions that are evaluated to assess the range of concrete cracking. This is discussed in further detail in MUAP-11018 Section 3.1 (Reference 3.8-70). These two basic loading conditions consider the load cases with the most significant potential to cause cracking, (i.e., safe-shutdown earthquake and accident thermal loads).

Condition A: Seismic + Operating Thermal. The normal operating thermal loading involves ambient temperatures of 105°F to 120°F, which are not anticipated to cause cracking that would significantly reduce the stiffness of the SC modules or any of the reinforced concrete structures. The operating temperature of the reactor cavity is 150°F, such that a linear temperature distribution is postulated through the nominally 10-ft thickness of the primary shielding walls, varying from 150°F at the interior face to 105-120°F at the exterior face. As discussed in Technical Report MUAP-11018, Appendix F (Reference 3.8-70), this shallow linear gradient is not anticipated to cause significant cracking of the primary shielding walls. Thus, the stiffness for Condition A is estimated by evaluating stresses resulting from the seismic loading condition only.

Condition B: Seismic + Accident Thermal. The accident thermal conditions postulated involve initial temperatures of 450 to 550°F as shown in Figure 3.8.1-12 and Figure 3.8.3-13 in the pipe-rupture side, with an immediate increase of temperature 270 to 300°F in Containment Vessel and RWSP as shown in Figure 3.8.1-11. The more detail of compartments for accident thermal conditions are summarized in Technical Report MUAP-11018 (Reference 3.8-70). Within approximately 10,000 seconds (2.8 hours), the temperatures on each face equilibrate to 300°F, which sets up parabolic (U-shaped) temperature distributions through the thickness of the SC walls.

This distribution will cause through-thickness cracks in the SC walls. These cracks will reduce the in-plane shear stiffness, cause overall thermal deformations and out-of-plane flexural cracking at restraints.

Estimated Stiffness for Each Category and Loading Condition:

The following is a summary of the estimated stiffness for each CIS structural category and loading condition. The stiffness terms summarized below are utilized in the two SSI analysis models involving upper and lower bound stiffness terms (as discussed in Section 3.7.2,) and in the two more detailed CIS structural design models with stiffnesses corresponding to Conditions A and B. Further discussion of the structural design analysis models is given in Section 3.8.3.4.1. Further detail on the basis for the CIS Condition A and B stiffness terms is provided in Technical Report MUAP-11018 (Reference 3.8-70).

Category 1, Condition A: An assessment of the maximum seismic in-plane shear demands in each SC wall of the containment internal structures indicated that these demands were generally lower than the cracking threshold for in-plane shear. Thus, the best estimate in-plane shear stiffness for Condition A is that of the uncracked composite section (i.e., $G_cA_c + G_sA_s$).

where

G_c = shear modulus of concrete

A_c = area of concrete per unit length

G_s = shear modulus of steel

A_s = area of steel per unit length

Note that the cracking threshold for SC walls was assumed at a concrete stress of $2\sqrt{f'_c}$. Typically the cracking threshold for concrete is related to concrete stress of $4\sqrt{f'_c}$, but the limit for SC walls is reduced to account for shrinkage and other effects, as described in Technical Report MUAP-11018, Section 4.1.2 (Reference 3.8-70). In addition, the uncracked stiffness estimated for this condition takes into account the recommendation to increase calculated secant stiffness values by a factor of 1.25 to obtain effective in-plane shear stiffness values appropriate for use in an equivalent linear elastic model as described in Technical Report MUAP-11018, Section 4.1.4 (Reference 3.8-70). Note that the effective stiffness values resulting from calculation of 1.25 times the secant stiffness are not to exceed the initial uncracked stiffness.

As discussed in Technical Report MUAP-11018 Appendix E, (Reference 3.8-70), experimental data indicates there is little to no uncracked out-of-plane flexural stiffness manifest in SC walls. This is due to effects of shrinkage cracking and partial composite action resulting from the discrete nature of the shear connectors (studs) between the face plates and the concrete core. Instead, the stiffness ($E_c I_{ct}$) associated with the cracked-transformed section is exhibited very early during the application of out-of-plane moments to SC walls.

where

E_c = modulus of elasticity of concrete

I_{ct} = cracked-transformed moment of inertia of concrete

Category 1, Condition B: The through-thickness temperature gradient resulting from the accident thermal loading can cause significant cracking that reduces the in-plane shear stiffness of the SC walls. An empirical relationship providing a best-estimate of secant in-plane shear stiffness of cracked SC walls is as follows, and as described in Technical Report MUAP-11018, Appendix C (Reference 3.8-70):

$$K_{cr} = 0.5(\bar{\rho}^{-0.42})G_s A_s$$

where

$$\bar{\rho} = \frac{A_s F_y}{\sqrt{f'_c}A_c}$$

G_s = shear modulus of steel

A_s = area of steel per unit length

F_y = yield strength of steel plates

f'_c = specified compressive strength of concrete

A_c = area of concrete core per unit length

Category 2, Condition A: Stress evaluation indicates these thick walls remain uncracked for Condition A. Thus, uncracked stiffness values of the concrete section shall be used; i.e., G_cA_c for in-plane shear and E_cI_c for out-of-plane flexure.

where

G_c = shear modulus of concrete

A_c = area of concrete per unit length

E_c = modulus of elasticity of concrete

I_c = moment of inertia of concrete

Category 2, Condition B: Stiffness of these walls shall account for cracking due to accidental thermal loading. Stiffness values of $0.5G_cA_c$ and $0.5E_cI_c$ are assigned per the recommendations for cracked concrete walls as shown in ASCE 43-05 (Reference 3.8-60).

Category 3, Condition A: The linear temperature gradient through the primary shield walls for normal operating conditions is not anticipated to cause significant cracking, and seismic demands on these walls are relatively limited in comparison to wall strength capabilities. Thus the primary shield wall stiffness shall be modeled as that of uncracked concrete (G_cA_c and E_cI_c). No credit is taken for the stiffness of the steel plates.

Category 3, Condition B: The accident thermal loading conditions are anticipated to cause only localized cracking in the thick primary shielding walls, which are largely enclosed by the mass concrete (Category 5) at the base of the containment internal structures. Thus, the stiffness for this condition is the same as that assigned for Condition A (uncracked).

Category 4, Condition A: In-plane shear stiffness of the reinforced concrete slabs shall be that of the gross concrete section (G_cA_c, in accordance with ASCE 43-05 (Reference 3.8-60)). Out-of-plane flexural stiffness is equal to that of the gross concrete section (E_cI_c), as seismic-induced moments in the slabs are shown generally to be less than cracking moments (M_{cr}):

$$M_{cr} = f_r \cdot S$$

where

S = gross section modulus

f_r = modulus of rupture of concrete

Category 4, Condition B: In-plane shear stiffness of the reinforced concrete slabs for this condition shall also be that of the gross concrete section (G_cA_c). Out-of-plane flexural stiffness is taken as $0.5E_cI_c$, as described by ASCE 43-05 (Reference 3.8-60).

Category 5 (both conditions): No significant cracking is anticipated in the massive reinforced concrete at the base of the structure as a result of either seismic or accident thermal loading. Thus, the stiffness is taken to be equal to that of uncracked concrete for both the A and B loading conditions.

Category 6 (both conditions): The stiffness of in-fill concrete provided for shielding purposes is not modeled for the A and B loading conditions; only the mass of these sections is included. For the pressurizer support platform, which is comprised of a grillage of steel shapes with in-fill concrete, only the stiffness of the steel members is modeled.

Damping values are assigned to each structural category based on the estimated level of cracking. A damping value of 4% is assigned to composite SC walls with uncracked conditions (Condition A), and 5% when significant cracking is anticipated (Condition B). This is based on the results of the 1/10th scale test discussed in Technical Report MUAP-10002 (Reference 3.8-80). For walls and slabs modeled as reinforced concrete structures, 4% damping is specified in RG 1.61 (Reference 3.8-64) for the limited levels of cracking associated with the OBE, while 7% damping is specified for cracked response exhibited during SSE loading. The massive concrete in the containment internal structures (Category 5) is not expected to exhibit significant cracking, such that 4% damping is considered appropriate in all cases. It is noted that the structural steel members within the CIS are very limited in scope relative to the mass and stiffness of the SC and RC members in the CIS. Recognizing that the amplified seismic response of the containment internal structure is dominated by the response of the SC walls, constant damping ratios of 4% for Condition A and 5% for Condition B are conservatively used for the seismic response analyses (See Table 3.8.3-4).

3.8.3.4.1 SC Module Stress Analyses

As discussed in Technical Report MUAP-11013 Section 3.2 (Reference 3.8-68), the design forces and moments for each member of the containment internal structure are calculated using two detailed 3-D FE models with stiffness and damping corresponding to loading Conditions A and B. Table 3.8.3-3 summarizes the analysis methods and objectives for the FE analyses performed for structural design. The geometry and element mesh of the detailed FE models are shown in Figure 3.8.3-10. Table 3.8.3-3 summarizes the objectives, analysis methods, and boundary conditions for the FE analyses performed with the detailed 3-D models.

As shown in Figure 3.8.3-10, the Category 1 and 2 SC modules are simulated within the detailed FE model using three-dimensional shell elements. The Category 3 (primary shield) SC modules are modeled using three-dimensional solid elements. Equivalent elastic stiffness constants are computed for each of the SC walls, as well as the RC slabs, to achieve the stiffness terms identified for Conditions A and B summarized above in Subsection 3.8.3.4 and in Technical Report MUAP-11018 (Reference 3.8-70). The method of calculating the equivalent elastic constants is explained in Section 8.0 of Technical Report MUAP-11018.

To generate the SSE load cases for structural design, response spectrum analysis is performed on each of the two detailed FE models (Condition A and Condition B). The inputs to both of these response spectrum analyses are the broadened, enveloped ARS generated at the base of the CIS by the SSI analyses. Likewise, each of the other design load cases (such as dead load, live load, and fluid load) are also run on each of the two detailed FE models, and combined with the corresponding SSE load case according to the applicable design loading combinations summarized in Table 3.8.4-3. This results in two sets of design loading combinations for the CIS; one set generated with the Condition A stiffness and a second set generated with the Condition B stiffness. The complete set of load combination results is then considered in the verification of the structure for the applied loads.

3.8.3.4.2 Hydrodynamic Analyses

The vertical and lateral pressures of liquids inside containment are treated as dead loads. Structures supporting fluid loads during normal operation and accident conditions are designed for the hydrostatic as well as hydrodynamic loads as discussed in Subsection 3.7.3.9. The hydrodynamic analyses take into account the flexibility of walls in considering fluid-structure interaction. Sloshing height, however, is calculated using a conservative simplified assumption of a rigid tank shell in accordance with guidance provided in ASCE 4-98 (Reference 3.8-34), Subsection 3.5.4.3.

3.8.3.4.3 Thermal Analyses

The RWSP water and containment operating atmosphere temperatures are considered stable. The operating thermal load for each concrete member is calculated as the average and gradient based on this condition. The stress analysis is carried out by inputting these loads into a 3-D FE model of the containment internal structures and the R/B basemat. For thermal effects on dynamic response, see the discussion of stiffness reductions due to thermal loading in Subsection 3.8.3.4.

The RWSP water and containment atmosphere are subject to temperature transients in the event of a LOCA as described in Subsection 3.8.3.3. The accident temperature transients result in a nonlinear temperature distribution within the members. Temperatures within the concrete members are calculated in a unidimensional heat flow analysis. The accident thermal load (average and equivalent linear gradients) is calculated from this analysis, at selected times during the transient.

The stress analysis for accident thermal loading is carried out by inputting the accident thermal load into a three dimensional FE model of the CIS and the portion of the R/B basemat to which the CIS is connected. Inclusion of the basemat in the model is necessary to obtain realistic restraint of the structure walls at the basemat connection. The SC walls and reinforced concrete slabs in the containment internal structures are assigned the reduced stiffness values resulting from thermally induced cracking, as identified for Condition "B" in Technical Report MUAP-11018 (Reference 3.8-70) and in Subsection 3.8.3.4 above. The moments and forces induced in the modeled structure are then included in the ACI 349-06 (Reference 3.8-8) design load combinations that involve accident thermal loading.

Thermal transients for the DBAs are described in Section 6.3.

3.8.3.4.4 Design Procedures

The reinforced concrete members of the containment internal structure are designed by the strength method, as specified in the ACI 349-06 (Reference 3.8-8).

The primary and secondary shield walls, RWSP, refueling cavity, and other structural walls are designed using SC modules. SC modules are designed using the methodology of reinforced concrete structures in accordance with ACI 349-06 (Reference 3.8-8), as supplemented in Technical Reports MUAP-11019 (Reference 3.8-71) and MUAP-11020 (Reference 3.8-72).

The concrete floor slabs and massive concrete sections near the base of the CIS are designed as reinforced concrete structures in accordance with ACI 349-06 (Reference 3.8-8). The floor slabs at elevation 76 ft, 5 in. (Operating floor) and elevation 50 ft, 2 in. are supported by structural steel framing.

Methods of analysis used are based on accepted principles of structural mechanics and are consistent with the geometry and boundary conditions of the structures.

The safe shutdown earthquake loads are determined from the results of seismic response analysis described in Section 3.7.

The determination of pressure and temperature loads due to pipe breaks is described in Subsections 3.6.1 and 6.2.1.2. Subcompartments inside containment containing high energy piping are designed for pressurization loads of 2 to 39 psi.

Determination of RCL support loads is described in Subsection 3.9.3. Design of the RCL supports is in accordance with ASME Code, Section III, Division 1, Subsection NF (Reference 3.8-2) as described in Subsections 3.9.3.

Computer codes used are general purpose codes. The code development, verification, validation, configuration control, and error reporting and resolution are according to the Quality Assurance requirements of Chapter 17.

3.8.3.4.5 SC Modules Design and Analysis

The SC modules are designed for dead, live, operating and accident thermal, accident pressure, and safe shutdown earthquake loads. The RWSP walls are also designed for the hydrostatic head due to the water in the pit and the hydrodynamic pressure effects of the water due to the safe shutdown earthquake loads. The walls of the refueling cavity are also designed for the hydrostatic head due to the water in the refueling cavity during refueling operations.

Figure 3.8.3-7 shows the typical design details of the SC modules, typical anchorages of the SC modules to the reinforced concrete basemat, connections between adjacent walls, and connections between reinforced concrete slabs and SC walls. SC modules are designed using the methodology of reinforced concrete structures in accordance with ACI 349-06 (Reference 3.8-8), as supplemented in Technical Reports MUAP-11019 (Reference 3.8-71) and MUAP-11020 (Reference 3.8-72). The faceplates are considered as the reinforcing steel, bonded to the concrete by headed studs.

The procedures of Technical Report MUAP-11019 (Reference 3.8-71) are used to design the SC walls for the design loading conditions. The procedures of Technical Report

MUAP-11020 (Reference 3.8-72) are used to design connections involving SC walls, such as SC wall-to-wall connections, reinforced concrete slab-to-SC wall connections, and SC wall basemat anchorage connections. The SC wall anchorage connection to the basemat is evaluated in accordance with the applicable requirements of both ACI 349-06 and ASME Section III, Division 2, since the connection crosses a code jurisdictional boundary as shown in Figure 3.8.3-7 sheet 5. The application of both codes is required because the steel baseplate and rebar anchors in this connection serve both as part of the force transfer mechanism between the SC faceplates and the reinforced concrete basemat, and as part of the containment pressure boundary liner and liner anchorage. The applicable code requirements are detailed in Technical Report MUAP-11020 Section 7.1. It is further noted that the applicable ACI or ASME load combinations are used to evaluate the corresponding requirements from each code. The application of these loading combinations to the basemat anchorage calculation is detailed in the CIS basic design calculations.

3.8.3.4.5.1 Design for Axial Loads and Bending

Design for axial loads (tension and compression) and out-of-plane bending is in accordance with the methodology of ACI 349-06 (Reference 3.8-8) Chapters 10 and 14, as supplemented by Sections 3, 4, and 5 of Technical Report MUAP-11019 (Reference 3.8-71). The design of the SC module faceplate reinforcement for combined axial loading, out-of-plane bending, and in-plane shear is performed as described in Technical Report MUAP-11019, Chapter 8.0, (Reference 3.8-71).

The design approach is based on SC module experimental research, in which the behavior of SC walls subjected to axial compression and out-of-plane flexural loading is similar to that of reinforced concrete walls subjected to these loads, provided that SC-specific limit states such as faceplate local buckling and interfacial shear failure are prevented. The observations and results of experimental research on SC wall out-of-plane flexure and axial compression behavior are summarized in Technical Report MUAP-11005, Appendices B and D, respectively (Reference 3.8-63). The manner in which the SC walls are detailed to prevent SC-specific limit states is presented in Technical Report MUAP-11019, Chapter 2 (Reference 3.8-71).

3.8.3.4.5.2 Design for In-Plane Shear

Design for in-plane shear is in accordance with the methodology of ACI 349-06 (Reference 3.8-8) Chapters 11 and 21, as supplemented by Section 7 of Technical Report MUAP-11019 (Reference 3.8-71). The steel faceplates are treated as reinforcement for the concrete which satisfy the provisions of Section 21.7 of ACI 349-06 (Reference 3.8-8).

The design approach is based on SC module experimental research in which the in-plane shear behavior of the infill concrete and longitudinal (faceplate) reinforcement was observed to be similar to that of reinforced concrete shear walls. The observations and results of experimental research on SC wall in-plane shear behavior are summarized in Technical Report MUAP-11005, Appendix C (Reference 3.8-63). The steel plate acts as shear reinforcement in each of two orthogonal directions, similar to the grids of longitudinal reinforcement provided in standard reinforced concrete shear walls.

However, as discussed in Technical Report MUAP-11019, Section 7 (Reference 3.8-71), the ACI 349-06 (Reference 3.8-8) code design strength for in-plane shear is conservatively modified by neglecting the initial concrete contribution before cracking. The concrete contribution to in-plane shear strength after cracking is included directly in the A_sf_y term.

3.8.3.4.5.3 Design for Out-of-Plane Shear

Design for out-of-plane shear is in accordance with the methodology of ACI 349-06 (Reference 3.8-8) Chapter 11, as supplemented by Section 6 of Technical Report MUAP-11019 (Reference 3.8-71).

The design approach is based on SC module experimental research in which the out-of-plane shear behavior of the infill concrete and transverse (tie bar) reinforcement was observed to be similar to that of reinforced concrete members. The observations and results of experimental research on SC wall out-of-plane shear behavior are summarized in Technical Report MUAP-11005, Appendix B (Reference 3.8-63). As discussed in Section 6 of Technical Report MUAP-11019, (Reference 3.8-71) the concrete contribution to out-of-plane shear strength is reduced to account for size effects. In addition, the concrete contribution to out-of-plane shear strength is ignored for load cases involving seismic loading.

3.8.3.4.5.4 Evaluation for Thermal Loads

The forces and moments induced in the SC walls due to restraint of thermal growth are included in the design load combinations in accordance with ACI 349-06 (Reference 3.8-8). As discussed in Section 9 of Technical Report MUAP-11019 (Reference 3.8-71), empirical data derived from experiments demonstrates that design basis accident thermal conditions cause no significant reduction in SC wall design strength. Thus, SC walls are evaluated and designed to resist combined design basis accident mechanical and thermal loads consistent with provisions of ACI 349-06 (Reference 3.8-8), as supplemented by Technical Report MUAP-11019 (Reference 3.8-71).

3.8.3.4.5.5 Design of Tie Bars

During SC module transportation and erection, the tie bars welded between the steel faceplates maintain the module configuration and separation between the faceplates, and act as "form ties" between the faceplates when concrete is being placed. The tie bars are fabricated from steel plates as shown in Technical Report MUAP-11019, Section 2.8 (Reference 3.8-71) and assembled in the manner discussed in Technical Report MUAP-12006 Section 3.0 (Reference 3.8-79). Welding between the tie bars and the faceplates is in accordance with American Welding Society (AWS D1.1) requirements. After the concrete has cured, the tie bars provide out-of-plane shear reinforcement similar to the transverse stirrups or ties provided in reinforced concrete members. The tie bars are designed as out-of-plane shear reinforcement according to the requirements of ACI 349-06, Section 11.5 (Reference 3.8-8), as supplemented by Sections 2.6 and 6.0 of Technical Report MUAP-11019 (Reference 3.8-71). The tie bar spacing is selected to meet the shear reinforcement spacing limits of ACI 349-06, Section 11.5.5 (Reference 3.8-8). The tie bar size is selected to ensure the development of ductile flexural yielding in the SC wall connection regions prior to concrete shear failure under out-of-plane loading, as discussed in Technical Report MUAP-11020, Sections 3.1 and 3.2, (Reference 3.8-72). The tie bar size and spacing selected for the connection regions is then used conservatively throughout the expanse of the SC walls for fabrication simplicity. Finally, the selected tie bar size and spacing is confirmed to maintain structural integrity of the SC walls by preventing section delamination or splitting failure, as discussed in Technical Report MUAP-11019, Section 2.7 (Reference 3.8-71).

3.8.3.4.5.6 Design of Shear Studs

The SC modules are designed as reinforced concrete elements, with the faceplates serving as reinforcing steel. Since the faceplates do not have deformation patterns typical of reinforcing steel, shear studs are provided to transfer the forces between the concrete and the steel faceplates. The shear studs are designed according to Appendix D of ACI 349-06 (Reference 3.8-8), as supplemented by Sections 2.1 through 2.5 of Technical Report MUAP-11019 (Reference 3.8-71). As discussed in Technical Report MUAP-11019, Section 2.2 (Reference 3.8-71), the shear stud spacing is selected so that the shear stud spacing to faceplate thickness ratio, or faceplate slenderness ratio, is less than or equal to 20. This is to prevent faceplate local buckling under applied compression, based on the behavior observed in experimental research. This research is summarized in Technical Report MUAP-11005, Appendix C (Reference 3.8-63). As discussed in Technical Report MUAP-11019, Section 2.3, the design shear strength of the studs is determined in accordance with ACI 349-06 Appendix D Section D.4.5 (Reference 3.8-8). Using these provisions, the shear studs are sized to prevent interfacial shear failure of the cross section under out-of-plane loading, as discussed in Technical Report MUAP-11019, Section 2.5. Finally, as discussed in Technical Report MUAP-11019, Section 2.4, the shear studs are confirmed to provide faceplate development lengths comparable to those of standard reinforcing bars typically used in reinforced concrete nuclear structures.

3.8.3.4.6 Floor Slabs

The reinforced concrete floor slabs are analyzed and designed according to ACI 349-06 (Reference 3.8-8) considering the same design loading conditions as for the SC modules. The floor design does not rely on composite action with supporting structural steel beams.

3.8.3.4.7 Structural Steel Design and Analysis

Structural steel framing within the interior of the PCCV is primarily for support of floor slabs, equipment, distribution systems, and access platforms. Design and analysis procedures, including assumptions on boundary conditions and expected behavior under loads, are in accordance with the allowable stress design (ASD) method in AISC N690 (Reference 3.8-9). Analysis methods are generally simple calculations using seismic loads obtained from Section 3.7 methodologies in load combinations. Frame connections are detailed for simply-supported beams unless otherwise analyzed and detailed.

3.8.3.4.8 RCL Supports

The RCL piping and support system is analyzed for the dynamic effects of a SSE. A coupled model of the containment internal structure and the RCS is dynamically evaluated using a time-history integration method of analysis. Appendix 3C provides additional information regarding the qualification of RCL supports.

3.8.3.5 Structural Acceptance Criteria

Structural acceptance criteria is reflected in Table 3.8.4-3 for concrete structures and Table 3.8.4-4 for steel structures, and are in accordance with ACI 349-06 (Reference 3.8-8) and AISC N690 (Reference 3.8-9), except as provided in the table notes.

3.8.3.5.1 Design Report

A Design Report of the containment internal structure is provided separately from the DCD. The Design Report has sufficient detail to show that the applicable stress limitations are satisfied when components are subjected to the design loading conditions.

Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Section 3.7 and 3.8 provided the following acceptance criteria are met.

- The structural design meets the acceptance criteria specified in Section 3.8.
- The ISRS meet the acceptance criteria specified in Subsection 3.7.2.5.

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design. The results of the evaluation are documented in an as-built summary report.

3.8.3.5.2 Design Summary of Representative Elements

This subsection summarizes the design of the following representative elements:

- Wall 1 - North-east wall of refueling cavity (4 ft, 8 in. thick)
- Wall 2 - North-west wall of secondary shield (4 ft, 0 in. thick)
- Wall 3 - North-east wall of RWSP (3 ft, 3 in. thick)
- Connection 1 - SC Wall Basemat Anchorage
- Connection 2 - SC Wall to SC Wall T Connection
- Connection 3 - Reinforced Concrete Slab to SC Wall Connection

Representative elements are selected to illustrate SC wall and connection designs for the CIS. The details and locations of the six representative elements are defined in Table 3.8.3-5. Locations are shown in Figure 3.8.3-7 Sheet 2 for connections and Figure 3.8.3-11 for walls. The structural configuration and typical details are shown in Figures 3.8.3-5, 3.8.3-6, 3.8.3-7, and 3.8.3-10. The structural analyses described in Subsection 3.8.3.4 are summarized in Table 3.8.3-4. The design procedures are described in Subsection 3.8.3.4.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

Subsection 3.8.4.6 contains information pertaining to the materials, quality control programs, and any special construction techniques utilized in the construction of seismic category I structures for the US-APWR.

3.8.3.6.1 Special Construction Techniques

Special module construction techniques, in addition to the methodology described in Subsection 3.8.3.1, is provided as necessary in Technical Report MUAP-12006, “Steel Concrete (SC) Wall Fabrication, Construction and Inspection" (Reference 3.8-79). The COL Applicant is to provide detailed construction and inspection plans and documents in accordance with MUAP-12006.

3.8.3.7 Testing and Inservice Inspection Requirements

Monitoring of seismic category I structures is performed in accordance with the requirements of NUMARC 93-01 (Reference 3.8-28) and 10 CFR 50.65 (Reference 3.8-29) as detailed in RG 1.160 (Reference 3.8-30), specifically Section 1.5 of RG 1.160. Subsection 3.8.4.7 describes the applicable testing and ISI requirements.

3.8.3.7.1 Construction Inspection

Inspection relating to the construction of seismic category I SSCs is in accordance with the codes applicable to the construction activities and/or materials. In addition, weld acceptance is performed in accordance with the National Construction Issues Group (NCIG), Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, NCIG-01 (Reference 3.8-31).

3.8.4 Other Seismic Category I Structures

Other seismic category I structures include those standard plant buildings which house safety-related systems and components, except the PCCV (Subsection 3.8.1) and CIS (Subsection 3.8.3). Distribution subsystems are also included in this discussion, such as safety-related HVAC ducts, conduits, cable trays, and their respective seismic category I supports.

US-APWR standard plant seismic category I structures and subsystems are designed for a SSE which is equivalent to the CSDRS defined in Subsection 3.7.1.1. Major US-APWR standard plant seismic category I structures with seismic designs based on the CSDRS are identified as:

- R/B
- East and west PS/Bs
- ESWPC

Discussion of design methodology, applicable loads, load combinations and acceptance criteria within this subsection is applicable for the R/B structures, the east and west PS/Bs, and the ESWPC, which are part of the US-APWR standard plant.

The COL Applicant is responsible for the seismic design of those seismic category I and seismic category II SSCs not seismically designed as part of the US-APWR standard plant, including the following seismic category I structures:

- ESWPT
- UHSRS
- PSFSVs

Note that the system descriptions of PSFSVs and ESWPT are within the scope of the US-APWR standard plant design.

Non-standard seismic category I SSCs are site-specific, and are designed for the site specific or more conservative SSE based on the ground motion response spectra.

3.8.4.1 Description of the Structures

The US-APWR R/B complex consists of the R/B, PCCV, CIS, A/B, east PS/B, west PS/B, and ESWPC supported on a common reinforced concrete basemat. The R/B, east PS/B, west PS/B, and A/B are combined structures that share structural shear walls. The PCCV and CIS are independent structures that share the common basemat with the other structures. The ESWPC located at the south side of the R/B complex shares, as a common wall, a portion of the southern wall of the R/B, east PS/B and west PS/B below grade. The R/B complex superstructure is separated from the T/B by approximately 13 ft. 2 in. at the closest interface point. The R/B complex basemat, discussed in Subsection 3.8.5, is horizontally separated from the T/B basemat by approximately 20 ft, 6 in. The AC/B and tank house are located adjacent to the A/B, with a 16 in. gap in between.

3.8.4.1.1 R/B

The R/B has five main floors. In plan, the R/B surrounds the PCCV and containment internal structure, and is founded with those structures on a common basemat. The outer perimeter of the R/B is basically rectangular, and is constructed of reinforced concrete walls, floors, and roofs. In cross-section, the height of the R/B varies from elevation 101 ft, 0 in. to 157 ft, 6 in., and the PCCV extends above the R/B to elevation 232 ft, 0 in.

The R/B consists of the following areas, defined by their functions.

- Safety system pumps and heat exchangers area
- Fuel handling area
- Main steam and feed water area
- Safety-related electrical area

The PCCV is discussed in detail in Subsection 3.8.1. The PCCV includes the containment internal structure comprising the primary shield wall and interior compartmentalization which are discussed in Subsection 3.8.3. Outside the PCCV and part of the R/B is the annulus. The annulus, which consists of concrete walled areas around the PCCV, serves a secondary containment function, and is made up of all areas with containment penetrations. It is maintained at a slightly negative pressure to control release of any radioactive materials to the environment.

The safety system pumps and heat exchanger areas are located at the lowest level of the R/B to secure the required net positive suction head. The safety system heat exchangers are located on the upper floor.

The fuel handling area is located on the plant northern side of the R/B at the same level as the containment vessel operating floor, and houses the following equipment and facilities:

Design and analysis of the spent fuel pit, the spent fuel racks, and the fuel handling system is in accordance with Appendix D of NUREG-0800, SRP 3.8.4 (Reference 3.8-40). Additional general information is provided by ANSI/ANS-57.7 (Reference 3.8-33). Subsection 9.1.2 describes the design basis and layout of the spent fuel pit, the spent fuel racks, and the fuel handling system.

- Fuel handling machine
- Cask pit with the spent fuel cask handling crane
- New fuel pit
- Cask washdown pit
- Spent fuel pit and storage racks

The main steam and feed water area is located on the plant southern side of the R/B, between the PCCV and the turbine building (T/B). The piping rooms are located on the top floor of this area where they pass between the PCCV and T/B.

The safety-related electrical area has two floors located on the plant southern side of the R/B and under the main steam and feed water area. This is a non-radioactive zone and is completely separated from the radioactive zones of the R/B. This area houses the following safety-related facilities.

- main control room (MCR)
- Switchgear and load center
- Instrumentation and control cabinet room

Four redundant safety systems containing radioactive material are located in each zone of the four quadrants surrounding the containment structure. Each of the quadrant areas is separated by physical barriers to assure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high energy pipe break events.

Non-radioactive safety systems such as the ESWS, CCWS and electrical system, etc., are located in the plant southern area of the R/B. This area is also separated into four divisions by a physical barrier to assure that the functions of the safety-related systems are maintained in the event of postulated incidents such as fires, floods, and high energy line break events.

3.8.4.1.2 PS/Bs

The east and west PS/Bs are arranged adjacent to the R/B; one to the east and one to the west. These buildings share a common basemat with the other structures of the R/B complex.

Each building contains two emergency power sources and one alternate power source which are separated from each other by a physical barrier. In addition, the safety-related chillers are also located in these buildings.

Details of the design and analysis of the east and west PS/Bs are provided in Subsection 3.8.4.4.

3.8.4.1.3 ESWPT, UHSRS, PSFSVs, and Other Site-Specific Structures

The ESWPT is a seismic category I structure constructed of reinforced concrete. The ESWPT is terminated at the south east and the south west corners of the R/B Complex, interfacing to the ESWPC in the R/B Complex. The other termination point is the UHSRS at the source of the ESWS. (The UHSRS consist of a cooling tower enclosure, ESWS pump houses, and the UHS basin.) The PSFSVs are structures which house the safety-related and non-safety related fuel oil tanks.

The design and analysis of the ESWPT, UHSRS, PSFSVs, and other site-specific structures are to be provided by the COL Applicant based on site-specific conditions.

3.8.4.1.4 Heating, Ventilating and Air Conditioning Ducts and Duct Supports

Seismic category I HVAC ducts and duct supports are routed as necessary to supply safety-related functions of air distribution. Appendix 3A describes the qualification of HVAC ducts and duct supports.

3.8.4.1.5 Conduits and Conduit Supports

Seismic category I conduits and conduit supports are routed as necessary to support safety-related Class-1E cable. The conduit consists of a metal wall of minimum thickness as specified, and is assembled using standard industry fittings and clips. Appendix 3F describes the qualification of conduits and conduit supports.

3.8.4.1.6 Cable Trays and Cable Tray Supports

Seismic category I cable trays and cable tray supports are routed as necessary to support safety-related Class-1E cable. Cable trays are manufactured using thin-gauge steel channels, and supports are constructed using cold formed or rolled steel shapes. Appendix 3G describes the qualification of cable trays and cable tray supports.

3.8.4.1.7 ESWPC

The ESWPC is arranged adjacent to the R/B and shares, as a common wall, a portion of the subgrade south wall of the R/B, east and west PS/Bs, and shares a common basemat. The ESWPC contains portions of the piping from the essential service water system, which provides service water for the component cooling water heat exchangers and essential chiller units.

3.8.4.2 Applicable Codes, Standards, and Specifications

The following industry standards are applicable for the design and construction of seismic category I structures and subsystems. Other codes, standards and specifications applicable to materials, testing and inspections are provided in Subsections 3.8.4.6 and 3.8.4.7.

- Code Requirements for Nuclear Safety-Related Concrete Structures (ACI 349-06) and Commentary, American Concrete Institute, 2006 (Reference 3.8-8).
- ACI 350.3-01, Seismic Design of Liquid-Containing Concrete Structures and Commentary, American Concrete Institute, 2001 (Reference 3.8–73).
- ANSI/AISC N690-1994, Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities, including Supplement 2 (2004), American National Standards Institute/American Institute of Steel Construction, 1994 & 2004 (Reference 3.8-9).
- ANSI/ANS-57.7 Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type), American National Standards Institute/American Nuclear Society, 1997 (Reference 3.8-33).
- ASCE 7-05, Minimum Design Loads for Buildings and Other Structures, including Supplement No. 1, American Society of Civil Engineers, 2005 (Reference 3.8-35).
- ASCE 37-02, Design Loads on Structures During Construction, American Society of Civil Engineers, 2002 (Reference 3.8-36).
- ASME BPVC-III, Rules for Construction of Nuclear Facility Components - Section III Division 1 - Subsection NF - Supports, American Society of Mechanical Engineers, 2001 Edition through the 2003 Addenda (Reference 3.8-2).
- Rules for Construction of Nuclear Facility Components, Division 2, Concrete Containments. Section III, American Society of Mechanical Engineers, 2001 Edition through the 2003 Addenda (Reference 3.8-2).

- ASME NQA-2-1983, Quality Assurance Requirements for Nuclear Power Plants, with ASME NQA-2a-1985, Addenda to ASME NQA-2-1983, American Society of Mechanical Engineers (Reference 3.8-37).

- Specification for the Design of Cold-Formed Steel Members. 1996 Edition and Supplement No 1, American Iron and Steel Institute, July 30, 1999 (Reference 3.8-38).

- NUREG-0800 SRP 3.8.4, Other Seismic Category 1 Structures, U.S. Nuclear Regulatory Commission, March 2007 (Reference 3.8–40).

- DC/COL-ISG-7, Interim Staff Guidance on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures, U.S. Nuclear Regulatory Commission (Reference 3.8–74).

- ACI 304R, Guide for Measuring, Mixing, Transporting, and Placing Concrete, American Concrete Institute, 2000 (Reference 3.8-39).

- ACI 224R, Control of Cracking in Concrete Structures, American Concrete Institute, 2001 (Reference 3.8-54).

- RG 1.61, Damping Values for Seismic Design of Nuclear Power Plants, U.S. Nuclear Regulatory Commission, March 2007 (Reference 3.8–64).

- RG 1.69, Concrete Radiation Shields for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, December 1973 (Reference 3.8-20).

- RG 1.91, Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants, U.S. Nuclear Regulatory Commission, February 1978 (Reference 3.8-49).

- RG 1.92, Combining Modal Responses and Spatial Components in Seismic Response Analysis, U.S. Nuclear Regulatory Commission, July 2006 (Reference 3.8-75).

- RG 1.115, Protection Against Low-Trajectory Turbine Missiles, U.S. Nuclear Regulatory Commission, July 1977 (Reference 3.8-50).

- RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants, U.S. Nuclear Regulatory Commission, March 1978 (Reference 3.8-47).

- RG 1.142, Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments), U.S. Nuclear Regulatory Commission, November 2001 (Reference 3.8-19).

- RG 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants, U.S. Nuclear Regulatory Commission, November 2001 (Reference 3.8-51).

- RG 1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, U.S. Nuclear Regulatory Commission, March 1997 (Reference 3.8-30).
- RG 1.199, Anchoring Components and Structural Supports in Concrete, U.S. Nuclear Regulatory Commission, November 2003 (Reference 3.8-41).

Appendix 3A, Section 3A.2, lists the applicable codes, standards and specifications for HVAC ducts and duct supports. Appendix 3F, Section 3F.2, lists the applicable codes, standards and specifications for conduit and conduit supports. Appendix 3G, Section 3G.2, lists the applicable codes, standards and specifications for cable trays and cable tray supports.

3.8.4.3 Loads and Load Combinations

Loads considered in the design are listed below. Not all loads listed are necessarily applicable to all structures and their elements. The loads for which each structure is designed are dependent on the applicable conditions.

The COL Applicant is to identify any applicable externally generated loads. Such site-specific loads include those induced by floods, potential non-terrorism related aircraft crashes, explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations. Loads that are due to malevolent vehicle assault, aircraft impact, and accidental explosion are taken as W_t in load combination 5 in accordance with RG 1.142 (Reference 3.8-19), Regulatory Position 7. Externally generated loads are not normally postulated to occur simultaneously with abnormal plant loads; however, the applicable loads and the related load combinations are determined on a case-by-case basis.

3.8.4.3.1 Dead Loads (D)

Dead loads are taken as the weight of all permanent construction/installations including fixed equipment and tanks. Uniform and/or concentrated dead loads are generally utilized for design of individual members. Equivalent dead loads are used during global analyses as conservative uniform load allowances of minor equipment and distribution systems, including small bore piping.

3.8.4.3.1.1 Dead Loads (Uniform and/or Concentrated)

Dead loads include the weight of structures such as slabs, roofs, decking, framing (beams, columns, bracing, and walls), and the weight of permanently attached major equipment, tanks, machinery, cranes, elevators, etc. The deadweight of equipment is based on its bounding operating condition including the weight of fluids. In addition, permanently attached non-structural elements such as siding, partitions, and insulation are included. Dead loads of cranes and elevators do not include the rated capacity lift or impact.

3.8.4.3.1.2 Equivalent Dead Load (Uniform)

Equivalent dead load includes the weight of minor equipment not specifically included in the dead load defined in Subsection 3.8.4.3 and the weight of piping, cables and cable trays, ducts, and their supports. It also includes fluid contained within the piping and minor equipment under operating conditions. Floors are checked for the actual equipment loads. To account for permanently attached small equipment, piping, ductwork and cable trays, a minimum equivalent dead load of 50 lb/ft^2 is applied. Where piping, ductwork, or cable trays are supported from platforms or walkway beams, actual loads may be determined and used in lieu of a conservative loading.

For floors with a significant number of small pieces of equipment (e.g., electrical cabinet rooms), the equivalent dead load is determined by dividing the total equipment weight by the floor area that effectively supports the equipment within the room, plus an additional 50 lb/ft^2.

3.8.4.3.2 Liquid Loads (F)

The vertical and lateral pressures of liquids are treated as dead loads except for external pressures due to ground water which are treated as live loads. The effects of buoyancy and flooding on SSCs are considered, where applicable. Structures supporting fluid loads during normal operation and accident conditions are designed for the hydrostatic as well as hydrodynamic loads. Impulsive and convective hydrodynamic loads due to seismic events are determined as discussed in Subsection 3.7.3.9, and included in the earthquake load as described in Subsection 3.8.4.3. For the purposes of evaluating flotation in Subsection 3.8.5.5, F_b is the buoyant force of the design-basis flood or high ground water table, whichever is greater.

3.8.4.3.3 Earth Pressure (H)

A static earth pressure acting on the structures during normal operation, considered as fully saturated to account for ground and flood water levels, is included in the analysis as *H*. The dynamic soil pressure, induced during an SSE event, is considered as an earthquake load E_{ss}. The analysis methods for static, dynamic, and passive earth pressure, including treatment of groundwater, are presented in Subsection 3.8.4.4.

3.8.4.3.4 Live Loads (L)

Live load is the load imposed by the use and occupancy of the building/structure. Live loads include floor area loads, laydown loads, fuel transfer casks, equipment handling loads, trucks, railroad vehicles, and similar items. The floor area live load need not be applied on areas occupied by equipment whose weight is specifically included in the dead load. Live load is applicable on floors under equipment where access is provided; for instance, the floor under an elevated tank supported on legs.

The following live load items are considered in design.

3.8.4.3.4.1 Building Floor Loads

Floor live loads account for heavily loaded areas for component laydown, such as the fuel cask loading dock and the containment refueling floor. The design live loads reflect the temporary location of major pieces of equipment, their safe load path during movement/ relocation, and their foot-print loads or equivalent uniformly distributed loads.

In addition, the following minimum values for live loads are used in load combinations involving non-seismic loads. Live loads for the seismic analysis are defined in Subsection 3.8.4.3.

Containment operating deck	950 lb/ft^2 (during maintenance and refueling outages)
	200 lb/ft^2 (during normal operation)
Offices	50 lb/ft^2
Assembly and locker rooms	100 lb/ft^2
Laboratories and laundry Rooms	100 lb/ft^2
Stairs and walkways	100 lb/ft^2 (or a moving concentrated load of 1,000 pounds)
Structural platforms & gratings	100 lb/ft^2
	(However, grating areas of concrete floors are designed for the same live load as the adjacent concrete floor)
Maintenance and service platforms	Load is calculated for individual locations based on the functional requirements and service equipment
All other floors (ground floor and elevated floors)	200 lb/ft^2
Rail road support structures:	Based on AREMA Manual
Truck support structures:	HS-20 loading per AASHTO standards

In design reconciliation analysis if actual loads are established to be lower than the above loads, the actual loads may be used for reconciliation. Floor live loads for design are not reduced below 100 lb/ft^2, except for offices which are maintained as 50 lb/ft^2 minimum.

3.8.4.3.4.2 Roof Snow Loads and Roof Live Loads

The roof is designed for uniform snow live load as specified in Chapter 2. Normal winter precipitation roof loads are added to all other live loads that may be expected to be present at the time to determine the design live load on the roof, and include appropriate load factors in applicable loading combinations. The extreme winter precipitation roof load is included as live load in extreme loading combinations using the applicable load factor. Other extreme environmental loads, e.g., seismic, tornado, and hurricane loads are not considered as occurring simultaneously. Slope roof snow loads, partially loaded, unbalanced roof snow loads, and drifts (including sliding snow) on lower roofs, as applicable, are determined in accordance with ASCE 7-05 (Reference 3.8-35).

The roof is designed for minimum 50 psf normal winter precipitation roof load and extreme winter precipitation load of 75 psf snow load for Seismic Category I structures for standard design. Consistent with DC/COL-ISG-7, this load represents the 100-year snowpack maximum snow weight, including the contributing portion of either extreme frozen winter precipitation event or extreme liquid winter precipitation event. The roof design accommodates a minimum roof live load of 40 psf to account for loads produced by workers, equipment, and materials. Roof live load is not added to roof snow load when evaluating the design load combinations.

3.8.4.3.4.3 Roof Rain Loads

Roof rain load is accounted for in accordance with Chapter 8 of ASCE 7-05 (Reference 3.8-35), and applied as applicable in load combinations. Roof rain load is included in live load in applicable load combinations, including additive effects with roof snow load as identified in Section 7.10 of ASCE 7-05. Subsection 3.4.1.2 provides additional discussion of design features to limit ponding of rain on the roofs of plant buildings.

3.8.4.3.4.4 Concentrated Loads for the Design of Local Members

Concentrated load on beams and girders (in load combinations that do not include seismic load)	5,000 lbs to be applied as to maximize moment or shear. This load is not carried to columns. It is not applied in office or access control areas [1]
Concentrated load on slabs (to be considered with dead load only)	5,000 lbs to be so applied as to maximize moment or shear. This load is not cumulative and is not carried to columns. It is not applied in office or access control areas[1]

[1] Area where no heavy equipment is located or transported.

In the design reconciliation analysis, if actual loads are established to be lower than the above loads, the actual loads may be used for reconciliation.

3.8.4.3.4.5 Temporary Exterior Wall Surcharge

The most critical of either a minimum surcharge of 450 psf (attributed to wheel loading converted to equivalent uniform load) or a railroad surcharge is applied. The surcharge is applied at plant grade adjacent to below-grade walls when such loading may be present.

3.8.4.3.4.6 Construction Loads

In the load combination for the construction case, the live load is defined as the additional construction loads produced by cranes, trucks, or any type of vehicle with its pick-up load, as required by construction. ASCE 37-02 (Reference 3.8-36) provides additional guidance. For steel beams supporting concrete floors, the weight of the concrete plus 100 lb/ft^2 uniform load or 5,000 pounds concentrated load, distributed near points of maximum shear and moment, are applied. A one third increase in allowable stress is permitted in this case.

Metal decking and precast concrete panels used as formwork for concrete floors are designed for the wet weight of the concrete plus a construction live load of 20 lb/ft^2 uniform or 150 pound concentrated. The deflection for these items used as a form is limited to the lesser of 0.75 in. or the span length (in inches) divided by 180. For relatively high construction loads, temporary supports may be used to prop floor beams without increasing their size.

3.8.4.3.4.7 Crane Loads

Crane and equipment supplier's information are used to determine wheel loads, equipment loads, weights of moving parts, and reactions of clamps (if any). Construction loads are considered where applicable.

Impact allowance for traveling crane supports and runway horizontal forces are in accordance with AISC N690 (Reference 3.8-9) for seismic category I and II structures, unless the crane manufacturer's design specifies higher impact loads. The vertical live load is increased by 25% to account for vertical impact of cab-operated traveling cranes and 10% of pendant-operated traveling cranes. A lateral force, equal to 20% of the lifted load and crane trolley are applied at the top and perpendicular to the crane rails. A longitudinal force equal to 10% of the maximum wheel load is applied at the top of the rails. Crane runways are also designed for crane stop forces.

Crane lift loads are not combined with wind loads. During construction; however, wind effects on the crane are considered. For load combinations, including SSE, all cranes in seismic category I areas are considered with a "most probable lift load" or heaviest load to be lifted over seismic category I SSCs/fuel, whichever is greater. Impact and seismic forces are not applied simultaneously.

3.8.4.3.4.8 Elevator Loads

Impact allowance for supports of elevators is 100%, applied to design capacity and weight of car plus appurtenances, or as specified by the equipment supplier.

3.8.4.3.4.9 Equipment Laydown and Major Maintenance

Floors are designed for planned refueling and maintenance activities as defined on equipment laydown drawings. Plans are developed for major equipment removal (such as SGs) and laydown. Temporary supports can be included in these plans provided such supports are easy to install and the installation of such supports is described in the plans.

3.8.4.3.5 Wind Load

3.8.4.3.5.1 Severe Wind (W)

The severe wind is determined as discussed in Subsection 3.3.1 for values specified in Chapter 2. Wind loads are not combined with seismic loads.

3.8.4.3.5.2 Tornado or Hurricane Load (Wt)

The design for tornado or hurricane loads is in accordance with Subsection 3.3.2 for values specified in Chapter 2. In addition, extreme winds such as hurricanes and tornadoes have the potential to generate missiles. Missiles generated by tornadoes and hurricanes are listed in Subsection 3.5.1.4 and barrier design for missiles is discussed in Subsection 3.5.3. These subsections describe the determination of tornado or hurricane loads applicable to the protection of safety-related equipment.

3.8.4.3.6 Seismic Loads

3.8.4.3.6.1 Operating Basis (E_{ob})

For seismic category I SSCs whose design is site-specific, that is, not included in the seismic design of the US-APWR standard plant, OBE loading has to be considered only if the site specific value for OBE response spectra acceleration is set higher than 1/3 of the site-specific SSE response spectra acceleration.

3.8.4.3.6.2 Safe Shutdown (E_{ss})

E_{ss} is defined as the loads generated by the SSE specified for the plant, including the associated hydrodynamic loads and dynamic incremental soil pressure (based on three-dimensional SSI analysis results). Earthquake loads (E_{ss}), are derived for evaluation of seismic category I structures using ground motion accelerations in accordance with Section 3.7.

Seismic dynamic analyses of the buildings consider the dead load and the equivalent dead loads as the accelerated mass. In addition to the dead load, 25% of the floor live load during normal operation and 75% of the roof snow load, whichever is applicable, is also considered as accelerated mass in the seismic models.

For the local design of members loaded individually, such as the floors and beams, seismic member forces include the vertical response due to masses equal to 50% of the specified floor live loads instead of 25% of floor live load, as follows:

$$a_v(0.5L)$$

where

a_v = Vertical seismic acceleration obtained from the seismic dynamic analysis results

L = Floor live load per Subsection 3.8.4.3.4

In locations where live loads are expected to always be present, the percentage of live load acting as accelerated mass is increased up to 100% of the live load for the affected members.

For the seismic load combination, the containment operating deck is designed for a live load of 200 lb/ft^2 which is appropriate for plant operating conditions, and 25% of this live load is included as mass in the seismic analyses. The mass of equipment and distributed system are included in both the dead and seismic loads.

3.8.4.3.7 Normal Operating Loads

3.8.4.3.7.1 Operating Thermal Loads (T_o)

The normal operating environment inside and outside the R/B is specified in Table 3.8.1-3. Thermal Conditions of the PS/Bs are provided in Table 3.8.4-2 and Figure 3.8.4-1. Normal thermal loads for the exterior walls and roofs are caused by positive and negative temperature variations through the concrete wall. The thermal gradient is also applied to the portion of the R/B at the outer face of the PCCV buttress shaft.

The COL Applicant is to specify normal operating thermal loads for site-specific structures, as applicable.

3.8.4.3.7.2 Operating Pipe Reactions (R_o)

Pipe and equipment reactions during normal operation or shutdown conditions are based on the most critical transient or steady state condition.

3.8.4.3.8 Effects of Pipe Rupture (Y) and other Accidents (P_a, T_a, R_a)

3.8.4.3.8.1 Accident Pressure Load (P_a)

Accident pressure loads are considered within or across a compartment and/or building due to a differential pressure generated by postulated pipe rupture. Dynamic effects due to pressure time-history are also included in the design.

3.8.4.3.8.2 Accident Thermal Loads (T_a)

Thermal loads due to temperature gradients caused by the postulated pipe breaks are considered in the design. The temperature gradients are calculated using the temperatures, corresponding to LOCA and MSLB, and are presented in Table 3.8.1-3. Local areas are designed for the elevated temperature effects and the loads resulting from the postulated accidents.

3.8.4.3.8.3 Accident Thermal Pipe Reaction (R_a)

Pipe and equipment reactions under thermal conditions are generated by the postulated pipe break, including (R_o).

3.8.4.3.8.4 Reaction Due to Pipe Ruptures (Y_r)

Pipe breaks within the R/B are postulated in accordance with the requirements of the SRP 3.6.2 (Reference 3.8-86) and 3.6.3 (Reference3.8-87).

The load on a structure generated by the reaction of a ruptured high-energy pipe during the postulated event is included using an appropriate dynamic load factor. The time dependent nature of the load and the ability of the structure to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of (Y_r).

3.8.4.3.8.5 Jet Impingement (Y_j)

Load on structure generated by the jet impingement from a ruptured high-energy pipe during the postulated event is included using an appropriate dynamic load factor. The time-dependent nature of the load and the ability of the structure to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of Y_j. The dynamic load factor is calculated using a long duration step function for the load. The target resistance is idealized as bilinear elasto-perfectly plastic.

The structural evaluation considers a double-ended break and a longitudinal break (equal to the pipe cross-sectional area) for calculating the jet impingement load from the main steam and feedwater lines. This evaluation is applicable to the floor at elevation 65 ft, 0 in. of the main steam isolation valve (MSIV) subcompartment in the R/B break exclusion zone. The design pressure for LOCA and MSLB is considered for 100% power operation.

3.8.4.3.8.6 Impact of Ruptured Pipe (Y_m)

The load resulting from the impact of a ruptured high-energy pipe on a structure or a pipe restraint during the postulated event includes an appropriate dynamic load factor. The type of impact (i.e., plastic, elastic), together with the ability of the structure to deform beyond yield are considered in establishing the structural capacity necessary to resist the impact.

3.8.4.3.9 Load Combinations

Concrete structures are designed in accordance with ACI 349-06 (Reference 3.8-8) and the provisions of RG 1.142 (Reference 3.8-19) where applicable, with the load combinations and load factors provided in Table 3.8.4-3.

Steel structures are designed using the allowable strength design method in accordance with AISC N690 (Reference 3.8-9) for the load combinations and allowable strength factors provided in Table 3.8.4-4.

3.8.4.4 Design and Analysis Procedures

The following discussion describes the design and analysis procedures used for seismic category I structures in accordance with ACI 349-06 (Reference 3.8-8), with supplemental guidance by RG 1.142 (Reference 3.8-19) for concrete structures, and AISC N690 (Reference 3.8-9) for steel structures. This subsection also discusses items such as general assumptions on boundary conditions, expected behavior under loads, methods by which loads and forces are transmitted to supports and ultimately the structure basemat, and computer programs used.

A Design Report prepared in accordance with guidance from Appendix C to SRP 3.8.4 (Reference 3.8-40) provides design and construction information more specific than contained within this DCD. The Design Report information quantitatively represents the actual design computations and the final design results. In addition, the Design Report provides criteria for reconciliation between design and as-built conditions.

3.8.4.4.1 R/B

The R/B includes the MCR and the fuel storage area, and is a reinforced concrete structure consisting of vertical shear/bearing walls and horizontal slabs. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs.

The fuel handling area is a reinforced concrete structure supported by structural steel framing. The new fuel is stored in racks in a dry, unlined pit. The spent fuel pit is lined with stainless steel and is normally flooded to an elevation 1 ft, 2 in. below the operating floor deck. Subsection 9.1.2 describes the design bases and layout of the fuel storage area.

The design and analysis procedures for the R/B, other than the PCCV and containment internal structure, including assumptions on boundary conditions and expected behavior under loads, are in accordance with ACI 349-06 (Reference 3.8-8) for concrete structures, with AISC N690 (Reference 3.8-9) for steel structures, and with American Iron and Steel Institute (AISI) specification for cold formed steel structures (Reference 3.8-38).

The design considers normal loads (including construction, dead, live, and thermal), and the SSE. These loads are applied to the linear elastic FE model. The design of the R/B complex is performed considering a fixed-base condition at the top of the basemat. Loads and load combinations are given in Subsection 3.8.4.3.

The design of the R/B’s flexible shear walls and floor slabs, like that of the main steam piping room with many openings, takes into account the out-of-plane bending and shear loads, such as live load, dead load, and seismic load. Also, the walls and slabs of the spent fuel pit and the emergency feedwater pit are designed to resist the out-of-plane bending and shear loads, created by hydrostatic and hydrodynamic pressure.

The R/B is analyzed using a three-dimensional FE model with the ANSYS computer codes (Reference 3.8-14). The FE model is shown in Figure 3.8.4-2.

The basemat design is described in Subsection 3.8.5.

Seismic Analysis of the R/B, East and West PS/Bs and ESWPC for Structural Design

Response spectrum analysis with the Lindley-Yow Method, described in NRC RG 1.92 (Reference 3.8-75),was selected for developing the seismic loads for design of the R/B, east and west PS/Bs, and the ESWPC. The Lindley-Yow method divides the total seismic response into two components: 1) periodic response with the ground motion ("out-of-phase" response) and 2) rigid response with the ground motion ("in-phase" response). Response spectrum analysis is performed for the periodic response with modified response spectrum in accordance to Section 1.3.2 of RG 1.92 (Reference 3.8-75). The Complete Quadratic Combination (CQC) method is used to combine modal responses. Static ZPA Method, which considers missing mass response as described in Section 1.4 of RG 1.92 (Reference 3.8-75), is used for the rigid response. The complete (periodic plus rigid) response spectrum analysis solution is calculated in accordance to Section 1.5.2 of NRC RG 1.92 (Reference 3.8-75). The maximum earthquake-induced response is combined by SRSS combination of the maximum representative responses from the three earthquake components (one vertical and two horizontal earthquake components) calculated separately in accordance with Section 2.1 of NRC RG 1.92 (Reference 3.8-75).

Response Spectrum Analysis Methodology for the R/B Complex

Seismic load for design is based on the SSI analysis results from 6 soil profiles using cracked and uncracked concrete conditions. A design approach, which satisfies the required seismic design loads determined from the 12 SSI analyses (6 soil profiles using cracked and uncracked concrete conditions) follows.

In order to validate the seismic load transfer from the SSI analysis results, structural responses (story shears) from SSI analyses equivalent static loads (ESL) and the response spectrum analyses (RSA) were compared to address SRP 3.7.2 II 3 E. Story shears were computed from basic SSI analysis results in a consistent manner with the two-step process used for member strength design.

Seismic Load Transfer from SSI Analysis Results to Structural Model

In order to perform the response spectrum analysis, acceleration response spectra are created using data from the soil structure interaction (SSI) analyses (Figure 3.8.4-28). SSI analyses include 6 soil cases with cracked and uncracked concrete conditions.

The ARS at the 4 corner nodes at the top of basemat were enveloped and broadened including all soils as well as cracked and uncracked concrete structures to create ISRS. These ISRS were then used as the input for the fixed base model of the R/B Complex to develop the loads on the structure for Step-1 below. The design of the R/B Complex structural members uses these loads as follows:

Step-1: Perform member strength check using loads derived from RSA with the enveloped and broadened soil case combined with other load cases. Because of the enveloping and broadening this method contains conservatisms and some structural members did not pass this screening.

Step-2: For those members not passing the Step-1 screen, a refined member strength check was performed using loads derived from the RSA of individual soil profiles unbroadened ARS combined with other load cases.

For Step-2 design, the governing soil cases were selected from the 12 SSI analyses. They were selected based on the: (1) comparison of ARS; and (2) comparison of equivalent static load profile.

Only the uncracked concrete condition is compared because the final selection of the governing soil profiles are based on the comparison of shear profiles determined from the 12 SSI analyses.

The 2032-100, 900-100, and 900-200 soil profiles for the uncracked model demonstrated that in combination they were the governing ARS compared to the remaining three soil profiles with the uncracked model and all soil profiles with the cracked model. In addition to the ARS comparison, the equivalent static load (ESL) calculated from the SSI time history analysis results are compared to select the two governing soil profiles for the Step-2 RSA as shown in Figure 3.8.4-28.

This confirmed the 2032-100 and 900-100 soil profiles were governing.

Validation of Response Spectrum Analyses for the R/B Complex

In order to validate the seismic force transfer from the SSI analysis results using a dynamic FE model to the more refined FE model for structural member design, the equivalent static load (ESL) calculated from the SSI analysis results was compared to the story shear and the vertical force (member design force) determined from the RSA results at each elevation. For appropriate validation, the story shear and the vertical force from the RSA must be greater than or approximately equal to the ESLs from SSI analyses.

3.8.4.4.1.1 Structural Design of Critical Sections

This subsection summarizes the structural design of representative seismic category I structural elements in the R/B. These structural elements are listed below and the corresponding location numbers are shown on Figure 3.8.4-3.

SECTION 1 West common wall between R/B and A/B, elevation -26 ft, 4 in. to elevation 101 ft, 0 in. This wall illustrates a common wall resisting seismic loads carried by both buildings.

SECTION 2 South interior wall of R/B, elevation -26 ft, 4 in. to elevation 101 ft, 0 in.

SECTION 3 The north exterior wall of spent fuel pit, elevation 30 ft, 1 in. to elevation 76 ft, 5 in. The wall is subjected to temperature gradients, seismic, hydrostatic and hydrodynamic loads.

AREA 3 The slab of spent fuel pit at elevation 30 ft, 1 in. The slab is subjected to temperature gradients, seismic, hydrostatic and hydrodynamic loads.

SECTION 4 South exterior wall of R/B, elevation -26 ft, 4 in. to elevation 115 ft, 6 in. This exterior wall is subjected to typical loads such as temperature gradients, seismic, hydrodynamic pressure, tornado missile, and hurricane missile.

AREA 4 The slab of emergency feedwater pit at elevation 76 ft, 5 in. The slab is a unique area encompassing the water storage pit. The slab is subjected to hydrostatic, hydrodynamic, and seismic loads.

SECTION 5 Central interior wall of R/B, elevation -26 ft, 4 in. to elevation 115 ft, 6 in.

3.8.4.4.1.2 Shear Walls

Structural Description

Shear walls in the R/B vary in thickness, configuration, aspect ratio, and amount of reinforcement. The stress levels in shear walls depend on these parameters and the seismic acceleration level. The walls are monolithically cast with the concrete floor slabs. The in-plane behavior of these shear walls, including the large openings, is adequately represented in the analytical models for the global seismic response. The shear walls are used as the primary system for resisting lateral loads, such as earthquakes.

Design Approach

The R/B shear walls are designed to withstand the loads specified in Subsection 3.8.4.3. Dead, live, thermal, seismic, and other normal operating condition loads are considered in the shear wall design.

West Common Wall

The west reinforced concrete wall is a common wall between R/B and A/B, which extends from elevation -26 ft, 4 in. to the roof at elevation 101 ft, 0 in. The wall is designed as a Category I structure. The wall segments are typically 28 in. to 40 in. thick. The wall is designed for the applicable loads including dead load, live load, and seismic loads. As shown in Figure 3.8.4-4, the wall is divided in 5 segments for design purposes. Table 3.8.4-6 presents the typical details of the reinforcement for each SECTION 1 wall zone. Figure 3.8.4-4 shows the typical reinforcement for the west common wall at SECTION 1.

South Interior Wall

The south interior reinforced concrete wall extends from elevation -26 ft, 4 in. to the roof at elevation 101 ft, 0 in. The wall segments are typically 40 in. to 52 in. thick. The wall is designed for the applicable loads including dead load, live load, and seismic loads. As shown in Figure 3.8.4-5, the wall is divided in 6 segments for design purposes. Table 3.8.4-7 presents the typical details of the reinforcement for each SECTION 2 wall zone. Figure 3.8.4-5 shows the typical reinforcement for the south interior wall at SECTION 2.

North Exterior Wall of Spent Fuel Pit

The north exterior reinforced concrete wall of the spent fuel pit extends from elevation 30 ft, 1 in. to the roof at elevation 76 ft, 5 in. The wall segments are typically 93 in. to 152 in. thick. The wall is designed for the applicable loads including dead load, live load, hydrostatic and hydrodynamic loads, seismic loads, spent fuel rack reaction loads, and thermal loads. As shown in Figure 3.8.4-6, the wall is divided in 3 segments for design purposes. Table 3.8.4-8 presents the typical details of the reinforcement for each SECTION 3 wall zone. Figure 3.8.4-6 shows the typical reinforcement for the north exterior wall at SECTION 3.

South Exterior Wall

The south exterior reinforced concrete wall extends from elevation -26 ft, 4 in. to the roof at elevation 115 ft, 6 in. The wall segments are typically 60 in. thick. The wall is designed for the applicable loads including dead load, live load, hydrostatic and hydrodynamic loads (for Emergency Feedwater Pit wall), seismic loads, thermal loads, tornado loads and hurricane loads. As shown in Figure 3.8.4-7, the wall is divided in 6 segments for design purposes. Table 3.8.4-9 presents the typical details of the reinforcement for each SECTION 4 wall zone. Figure 3.8.4-7 shows the typical reinforcement for the south exterior wall at SECTION 4.

Central Interior Wall

The central interior reinforced concrete wall extends from elevation -26 ft, 4 in. to the roof at elevation 115 ft, 6 in. The wall segments are typically 40 in. thick. As shown in Figure 3.8.4-8, the wall is divided into 6 segments for design purposes. Table 3.8.4-10 presents the typical details of the reinforcement for each SECTION 5 wall zone.

3.8.4.4.1.3 Floor and Roof

Design Approach

The concrete slab and the steel reinforcement of the composite section are evaluated for normal and extreme environmental conditions. The slab concrete and the reinforcement are designed to meet the requirements of the ACI 349-06 Code (Reference 3.8-8). The slab design considers the in-plane and out-of-plane seismic forces. The global in-plane and out-of-plane forces are obtained from the three-dimensional FE model of the R/B complex.

Spent Fuel Pit Slab at Elevation 30 ft, 1 in., AREA 3

This concrete slab is designed for the applicable loads including dead load, live load, hydrostatic and hydrodynamic loads, seismic loads, spent fuel rack reaction loads, and thermal loads. The concrete slab is 126 in. thick. Table 3.8.4-11 presents the typical details of the reinforcement for AREA 3. Figure 3.8.4-9 shows the typical reinforcement at AREA 3.

Emergency Feedwater Pit Slab at Elevation 76 ft, 5 in., AREA 4

This concrete slab is designed for the applicable loads including dead load, live load, hydrostatic and hydrodynamic loads, and seismic loads. The concrete slab is 52 in. thick.

Table 3.8.4-12 presents the typical details of the reinforcement for AREA 4. Figure 3.8.4-10 shows the typical reinforcement at AREA 4.

3.8.4.4.1.4 Below Grade Exterior Walls

Exterior concrete walls below grade of seismic category I structures are designed using load combinations accounting for static lateral earth pressure (including soil surcharges) and dynamic lateral earth pressure, including effects of the water table. Load combinations are presented in Table 3.8.4-3.

The lateral earth pressure distribution profiles on below-grade exterior walls are developed in accordance with Acceptance Criterion II.4.H of SRP 3.8.4 (Reference 3.8-40) by evaluating: (1) lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with Section 3.5.3.2 of ASCE 4 (Reference 3.8-34); and (2) lateral earth pressure equal to the passive earth pressure.

The envelope of the two pressure profiles is applied to the detailed model as equivalent static loads for purposes of design, in combination with other applicable loads. When computing lateral earth pressure due to static plus dynamic pressure, the water table is considered to be at plant grade to maximize the load on the walls. When computing the lateral earth pressure due to passive earth pressure, the water table is considered to be at the bottom of the basemat level in order to maximize the passive resistance for conservative design of the walls.

<u>Hydrostatic Pressure</u>

The hydrostatic pressure on the wall at depth of z from the ground surface is calculated as:

$$P_{hydro} = \gamma_{water} \cdot z$$

where γ_{water} = 62.4 pcf is the unit weight of water.

<u>Static Lateral Earth Pressure</u>

The static lateral earth pressures on the exterior walls consist of static at-rest earth pressure, and pressure due to surcharge. The total static lateral earth pressure at depth of z from the ground surface is calculated as:

$$P_{static} = K_0 \cdot \gamma_{eff} \cdot z + K_0 \cdot P_{surcharge}$$

where $P_{surcharge}$ = 450 psf is the surcharge load on the ground surface and K_0= 0.5 is the at-rest coefficient of lateral soil pressure.

Dynamic Lateral Earth Pressure due to Horizontal Motion

The horizontal earthquake excitation induced lateral pressure, denoted as P_{sh}, is calculated by interpolating and applying Wood's solution included in Figure 3.5-1 of ASCE 4 (Reference 3.8-34) for the following soil and seismic parameters:

Poisson's ratio ν = 0.4 (conservative value for granular soil)

Coefficient C_ν as a function of Poisson's ratio = 1.04

Soil unit weight: saturated, γ_{sat} = 130 pcf

Wall height H = 42.25 ft

Horizontal seismic coefficient in g's α_h = 0.5

The saturated unit weight of the backfill soil is used assuming that the pore water will move together (in-phase) with the soil during earthquake shaking, and the inertial force is proportional to the total weight of the embedment soil. This assumption is conservative since it does not consider the dissipation of energy due to the viscous flow of the ground water in the soil skeleton. The SSI and SSSI analyses of the R/B complex result in a maximum average horizontal acceleration of approximately 0.5g along the embedded perimeter of its exterior walls. Therefore, a value of 0.5 is used for the horizontal seismic coefficient.

Therefore, the total static and dynamic lateral earth pressure on the below-grade exterior walls is calculated as follows:

$$P_s = P_{sh} + P_{static} + P_{hydro}$$

Passive Earth Pressure

The passive earth pressure, assuming Rankine's theory, has the expression:

$$P_p = K_p \cdot \gamma_{unsat} \cdot z + K_p \cdot P_{surcharge}$$

where γ_{unsat} is taken as in-situ unit weight (125 pcf), and K_p = 3.69 is the passive earth pressure coefficient calculated assuming an internal friction angle of 35°.

Table 3.8.4-23 presents the numeric values of "Dynamic + Static Pressure" and "Total Passive Pressure." Figure 3.8.4-27 presents the "Dynamic + Static Pressure" and "Total Passive Pressure" profiles.

The total passive earth pressure is greater than the dynamic plus static lateral earth pressure below elevation -3.0 ft, while the dynamic plus static lateral earth pressure is the controlling pressure profile above elevation -3.0 ft. Therefore, a conservative envelope of the two pressure profiles is applied to the detailed model to design the exterior walls below-grade, in combination with other applicable loads.

Passive earth pressures on the R/B complex exterior walls at the interfaces with the AC/B and tank house structure are increased from those presented in Table 3.8.4-23 and Figure 3.8.4-27 and are calculated considering potential sliding effects. The passive pressures calculated in this manner are based on the Rankine earth pressure theory, modified to account for presence of a rigid structure within the passive soil wedge. The following conservative assumptions are used:

The effect of out-of-phase motion is considered by adding horizontal inertia forces induced by sliding in the adjacent buildings (AC/B and tank house), acting in the direction of increasing passive reaction.

Lateral earth pressure on the south side of the R/B complex is affected by relative sliding between the R/B complex and the T/B. The envelope of the passive earth pressures and those pressures induced by sliding is used for the design of the exterior below grade walls on the south side of the R/B complex (and on the north side of the T/B).

The COL Applicant is to verify that lateral earth pressures used in the standard plant design envelope site-specific lateral earth pressures. The COL Applicant will satisfy the earth pressure enveloping criteria if the site-specific earth pressure demands on the basement exterior walls are enveloped by the standard design earth pressure loads. Since the walls of the standard design structures below grade are designed for the R/B complex sliding in to the adjacent soil, the pressures for the site specific structure will be below the design values if the soil weight and friction angles are below the standard plant design values used in design. Thus the maximum unit weight for the side soils is 125 pounds per cubic foot and the friction angle must be at or below 35 degrees. Further, the passive soil wedge must not intersect with the cut line of the excavation.

If the passive soil wedge intersects with the cut line of the excavation, then passive pressures must be reconciled with the passive pressure shown in the Standard Plant design calculations using the properties of the native soils and the backfill.

The site independent standard design has a calculated value for sliding of 0.75 in. in any direction. This is the site specific design value for attachments when the standard plant design envelopes the site-specific design according to the ARS comparative criteria described in Subsection 3.7.2.4.5. If the COL applicant can demonstrate that the standard plant does not slide in any direction then the above requirement can be relaxed. This can be demonstrated by using conventional quasi-static analysis techniques that ignore the effect of passive pressure resisting sliding and by satisfying the factor of safety provided in the SRP 3.8.5.

If any of the comparisons are not satisfactory, the COL Applicant shall re-analyze the plant for the site specific conditions following the methodology presented in the Standard Plant design basis calculations.

3.8.4.4.2 East and West PS/Bs

The east and west PS/Bs provide two emergency power sources, and are reinforced concrete structures consisting of vertical shear/bearing walls and horizontal slabs. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs.

The design and analysis procedures for the PS/Bs, as described above for the R/B including assumptions on boundary conditions and expected behavior under loads (see Subsection 3.8.4.4), are in accordance with ACI 349-06 (Reference 3.8-8) for concrete structures, with AISC N690 (Reference 3.8-9) for steel structures, and AISI specification for cold formed steel structures Reference 3.8-38).

The design considers normal loads including construction, dead, live, thermal, and the SSE. Loads and load combinations are provided in Subsection 3.8.4.3.

The PS/Bs are analyzed using a three-dimensional FE model with the ANSYS computer codes (Reference 3.8-14). The FE model is shown in Figure 3.8.4-11. The basemat design is described in Subsection 3.8.5.

3.8.4.4.2.1 Structural Design of Critical Sections

This subsection summarizes the structural design of representative seismic category I structural elements in the PS/Bs. These structural elements listed below are for the west and east PS/Bs. Locations within the west PS/B are shown with corresponding sections and area on Figure 3.8.4-12 and Figure 3.8.4-13. Locations within the east PS/B are shown with corresponding sections and area on Figure 3.8.4-18 and Figure 3.8.4-19.

WEST PS/B:

Section 1 South exterior wall of West PS/B, elevation -26 ft, 4 in. to elevation 49 ft, 0 in.

Section 2 Interior wall of the West PS/B, elevation -26 ft, 4 in. to elevation 3 ft, 7 in.

Section 3 North exterior wall of the West PS/B, elevation -26 ft, 4 in. to elevation 49 ft, 0 in. This wall is a common wall with the A/B.

Area 1 The slab of PS/B at elevation 3 ft, 7 in.

EAST PS/B:

Section 1 East exterior wall of East PS/B, elevation -26 ft, 4 in. to elevation 39 ft, 6 in.

Section 2 Interior wall of the East PS/B, elevation -26 ft, 4 in. to elevation 3 ft, 7 in.

Area 1 The slab of the East PS/B, elevation 3 ft, 7 in.

3.8.4.4.2.2 Shear Walls

Structural Description

All exterior walls are shear walls, however internal shear walls exist only in the north-south axis. The stress levels in shear walls depend on thickness, configuration, aspect ratio, amount of reinforcement and the seismic acceleration level. The walls are monolithically cast with the concrete floor slabs. The in-plane behavior of these shear walls, including the large openings, is adequately represented in the analytical models for the global seismic response. The shear walls are used as the primary system for resisting the lateral loads, such as earthquakes.

Design Approach

The PS/B shear walls are designed to withstand the loads specified in Subsection 3.8.4.3. Dead, live, thermal, and other normal operating condition loads are considered in the shear wall design.

South Exterior Wall

The south exterior reinforced concrete wall extends from the top of the basemat area at elevation -26 ft, 4 in. to the roof at elevation 39 ft, 6 in. The walls are typically 40 in. thick. The wall is designed for the applicable loads including dead load, live load, seismic loads, thermal loads, and tornado or hurricane loads. As shown in Figure 3.8.4-14, the wall is divided into four sections, each for design purposes. Table 3.8.4-13 presents the typical details of the reinforcement for the SECTION 1 wall zone. Figure 3.8.4-14 shows the typical reinforcement of the south exterior wall at SECTION 1.

West Interior Wall

The west interior reinforced concrete wall extends from the top of the basemat area at elevation -26 ft, 4 in. to the slab at elevation 3 ft, 7 in. The walls are 20 in. thick. The wall is designed for the applicable loads including dead load, live load, seismic loads, and thermal loads. Table 3.8.4-14 presents the typical details of the reinforcement for SECTION 2 wall zone 1, which is applicable for all interior walls. Figure 3.8.4-15 shows the typical reinforcement for the interior wall at SECTION 2.

3.8.4.4.2.3 Floor

Design Approach

The concrete slab and the steel reinforcement of the composite section are evaluated for normal and extreme environmental conditions. The slab concrete and the reinforcement are designed to meet the requirements of American Concrete Institute standard ACI 349-06 (Reference 3.8-8). The slab design considers the in-plane and out-of-plane seismic forces. The global in-plane and out-of-plane forces are obtained from the 3-D FE model of the PS/B.

Slab at Elevation 3 ft, 7 in., AREA 1

The concrete slab is designed for the applicable loads including dead load, live load, seismic loads, and thermal loads. The concrete slab is 32 in. thick. Table 3.8.4-15 presents the typical details of the reinforcement for AREA 1. Figure 3.8.4-16 shows the typical reinforcement at AREA 1.

3.8.4.4.3 ESWPC

The ESWPC contains portions of the piping from the ESWS, which provides service water for the component cooling water heat exchangers and essential chiller units. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs.

The design and analysis procedures for the ESWPC, including assumptions on boundary conditions and expected behavior under loads, are in accordance with ACI 349-06

(Reference 3.8-8) for concrete structures. The design considers normal loads including construction, dead, live, thermal, and the SSE. Loads and load combinations are provided in Subsection 3.8.4.3.

The ESWPC is analyzed using a three-dimensional FE model with the ANSYS computer codes (Reference 3.8-14). The FE model is shown in Figures 3.8.4-2 and 3.8.4-11.

3.8.4.4.3.1 Structural Design of Critical Sections

This subsection summarizes the structural design of the representative seismic category I structural elements of the ESWPC. These structural elements, listed below, are subjected to large stress demands, and are considered to be the best representation of the structural design. The following locations within the eastern half of the ESWPC are shown with corresponding critical wall sections and slab area in Figure 3.8.4-23.

Section 1 South exterior wall of the ESWPC, elevation -26ft, 4in. to elevation 1ft, 7in.

Section 2 Exterior transverse wall of the ESWPC, elevation -9ft, 8in. to elevation 1ft, 7in.

Area 1 The slab of the ESWPC, elevation -15ft, 8in.

3.8.4.4.3.2 Shear Walls

Structural Description

The stress levels in shear walls depend on thickness, configuration, aspect ratio, amount of reinforcement and seismic acceleration level. The walls are cast with the concrete floor slabs; reinforcing steel bars are adequately developed between structural elements. The in-plane behavior of these shear walls is adequately represented in the analytical model for the global seismic response. The shear walls are used as the primary system for resisting the lateral loads, such as earthquakes.

Design Approach

The ESWPC shear walls are designed to withstand the loads specified in Subsection 3.8.4.3. Dead, live, thermal, seismic, and other normal operating condition loads are considered in the shear wall design.

South Exterior Wall

The south exterior reinforced concrete wall extends from the top of the basemat area at elevation -26 ft, 4 in. to the slab at elevation 1 ft, 7 in. The wall is 36 in. thick. The wall is designed for the applicable loads including dead load, live load, seismic loads, thermal loads, and tornado or hurricane loads. As shown in Figure 3.8.4-24, the wall is divided into 3 zones for design purposes. Table 3.8.4-20 presents the typical details of the reinforcement for the south exterior wall. Figure 3.8.4-24 shows the typical reinforcement of the south exterior wall (SECTION 1).

Exterior Transverse Wall

The typical exterior transverse reinforced concrete wall extends from the top of the second floor at elevation -9 ft, 8 in. to the slab at elevation 1 ft, 7 in. The wall is 24 in. thick. The wall is designed for the applicable loads including dead load, live load, seismic loads, and thermal loads. Table 3.8.4-21 presents the typical details of the reinforcement for SECTION 2. Figure 3.8.4-25 shows the typical reinforcement of SECTION 2.

3.8.4.4.3.3 Floor

<u>Design Approach</u>

The concrete slab and the steel reinforcement are designed to withstand the loads specified in Subsection 3.8.4.3. Dead, live, thermal, seismic, and other normal operating condition loads are considered in the slab design. The slab concrete and the reinforcement are designed to meet the requirements of ACI 349-06 (Reference 3.8-8). The slab design considers the in-plane and out-of-plane seismic forces. The global in-plane and out-of-plane forces are obtained from the 3-D FE model of the ESWPC.

<u>Slab at Elevation -15 ft, 8 in., AREA 1</u>

The concrete slab is designed for the applicable loads including dead load, live load, seismic loads, and thermal loads. The concrete slab is 24 in. thick. Table 3.8.4-22 presents the typical details of the reinforcement for AREA 1. Figure 3.8.4-26 shows the typical reinforcement at AREA 1.

3.8.4.4.4 Other Seismic Category I Structures

The design and analysis procedures for other seismic category I concrete structures are in accordance with ACI 349-06 (Reference 3.8-8). The design and analysis procedures for seismic category I steel structures are in accordance with AISC N690 (Reference 3.8-9).

Seismic category I structures are modeled globally using applicable loads, including equivalent dead and live loads, in load combinations that include design-basis earthquake accelerations as described in Section 3.7. Computer modeling utilizes three-dimensional FE models to globally analyze the beams, columns, slabs, and shear walls. Individual structural members are further analyzed for localized loading as described in specific load cases.

Concrete components such as walls, slabs, and basemats are evaluated for the effects of frame interaction when the flexural moment from seismic loads is a large percentage of the flexural capacity. When at least two-thirds of the flexural capacity of a component is from seismic loads alone, the component is designed as a frame to assure design capacity even under a seismic margin earthquake equal to 150% of the SSE, in accordance with RG 1.142 (Reference 3.8-19), Regulatory Position 3.

Concrete members that are subject to torsion and combined shear and torsion are evaluated to the standards of Section 11.6 of ACI 349-06 (Reference 3.8-8).

Exterior concrete walls below grade and basemat of seismic category I structures are designed using load combinations accounting for sub-grade loads including static and dynamic lateral earth pressure, soil surcharges, and effects of maximum water table.

Structural steel framing in seismic category I structures is primarily for the support of distribution systems, access platforms, and other plant appurtenances. Steel members are sized and detailed based on maximum stresses and reactions determined through conservative manual calculations and computer models based on pinned-end connections, including slotted hole clip angle connections, to relieve thermal expansion forces where appropriate, unless detailed to develop end moments in accordance with AISC N690 (Reference 3.8-9). The design of the support anchorage to the concrete structure is in accordance with ACI 349-06, Appendix D (Reference 3.8-8), and RG 1.199 (Reference 3.8-41).

The design and analysis procedures for seismic category I distribution systems, such as HVAC ducts, conduits, and cable trays including their respective seismic category I supports, are in accordance with AISC N690 (Reference 3.8-9) and AISI Specification for Design of Cold-Formed Steel Members (Reference 3.8-38). The following appendices provide additional discussion of the design and analysis of these subsystems.

- Appendix 3A Heating, Ventilation, and Air Conditioning Ducts and Duct Supports
- Appendix 3F Design of Conduits and Conduit Supports
- Appendix 3G Seismic Qualification of Cable Trays and Supports

The COL Applicant is to provide design and analysis procedures for the ESWPT, UHSRS, and PSFSVs.

3.8.4.4.5 Seismic Category II Structures

Seismic category II structures need not remain functional during and after an SSE. However, such structures must not fall or displace to the point they could damage seismic category I SSCs.

Seismic category II structures and subsystems are analyzed and designed using the same methods and stress limits specified for seismic category I structures and subsystems, and the same load combinations and stress coefficients given in Table 3.8.4-4.

3.8.4.5 Structural Acceptance Criteria

Structural acceptance criteria are listed in Table 3.8.4-3 for concrete structures and in Table 3.8.4-4 for steel structures, and are in accordance with ACI 349-06 (Reference 3.8-8) and AISC N690 (Reference 3.8-9), except as provided in the table notes.

The deflection of the structural members is limited to the maximum values as specified in ACI 349-06 (Reference 3.8-8) and AISC N690 (Reference 3.8-9), as applicable.

Subsection 3.8.5.5 identifies acceptance criteria applicable to additional basemat load combinations.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

The following information pertains to the materials, quality control programs, and any special construction techniques utilized in the construction of the seismic category I structures for the US-APWR.

3.8.4.6.1 Materials

The major materials of construction in seismic category I structures are concrete, grout, steel reinforcement bars, splices of steel reinforcing bars, structural steel, and anchors.

3.8.4.6.1.1 Concrete

Concrete which has a compressive strength of 5000 psi is utilized in standard plant seismic category I structures other than PCCV, upper part of the tendon gallery in the basemat and the containment internal structure (CIS). Concrete utilized in the PCCV and upper part of the tendon gallery in the basemat has a compressive strength of f'_c = 7,000 psi at a test age of 56 days and is subject to the PCCV material requirements in Subsection 3.8.1.6, including the requirements of ASME III, Division 2 (Reference 3.8-2). Concrete utilized in the CIS has a compressive strength of F'c=4,000. The COL Applicant is to specify concrete strength utilized in non-standard plant seismic category I structures. A test age of 28 days is used for normal concrete. Batching and placement of concrete is performed in accordance with ACI 349-06 (Reference 3.8-8), ACI 304R (Reference 3.8-39), and ASTM C 94 (Reference 3.8-42). During construction, volume changes in mass concrete are controlled where necessary by applying measures and provisions outlined in ACI 207.2R (Reference 3.8-52) and ACI 207.4R (Reference 3.8-53).

Portland cement is used in the concrete conforms to ASTM C 150, Type II (Reference 3.8-43) standards. The confirmation of the chemical composition of the cement properties is validated by certified copies of test reports showing the chemical composition of each Portland cement shipment.

Aggregates used in the concrete conform to ASTM C 33 (Reference 3.8-44). Aggregate and source acceptance is based on documented test results for each source and random sampling of shipments based on MIL-STD-1916 (Reference 3.8-45).

Water and ice used in the concrete conform to the requirements of ACI 349-06 (Reference 3.8-8).

Admixtures include an air entraining admixture, pozzolans, and a water reducing admixture. The admixtures, except the pozzolans, are stored in a liquid state. For certain concrete placement operations, self-consolidating concrete is used to minimize the potential for voids in areas of high congestion or limited access. Self-consolidating concrete is able to flow under its own weight and improves fluidity while resisting segregation. It is able to completely fill the formwork, even in the presence of dense reinforcement, without the need of vibration, while maintaining homogeneity. Concrete material and installation practices for self-consolidating concrete are in accordance with ACI 237R (Reference 3.8-81).

Admixtures and concrete mix conform to the following requirements:

Pozzolans	ASTM C 618
Sampling and Testing of Pozzolans	ASTM C 311
Air Entraining Admixtures	ASTM C 260
Water Reducing Admixtures	ASTM C 494 or ASTM C 1017
Concrete Mix	ACI 211.1 and ASTM C 94 (Reference 3.8-45)
Concrete Mix Testing	ASTM C 172, ASTM C 192, and ASTM C 39
Minimum Number of Strength Tests	ACI 349-06 (Reference 3.8-8)

Samples for strength tests of concrete should be taken at least once per day for each class of concrete placed or at least once for each 100 cubic yards of concrete placed. When the standard deviation for 30 consecutive tests of a given class is less than 600 psi, the amount of concrete placed between tests may be increased by 50 cubic yards for each 100 psi the standard deviation is below 600 psi, except that the minimum testing rate should not be less than one test for each shift when the concrete is placed on more than one shift per day or not less than one test for each 200 cubic yards of concrete placed. The test frequency should revert to once for each 100 cubic yards placed if the data for any 30 consecutive tests indicate a higher standard deviation than the value controlling the decreased test frequency.

3.8.4.6.1.2 Grout

Grout is used to transfer load from machinery, equipment, and column bases to their foundations, and to anchor the reinforcing bars, dowels, and anchor rods into hardened concrete. Grout generally consists of Portland cement, sand, water, and admixtures. Epoxy grout is only used in areas where radiation levels and temperature levels are compatible with epoxy use.

Portland cement used in the concrete conforms to ASTM C 150, Type II (Reference 3.8-43). Sand must be clean with gradation and fineness in accordance with ASTM C33 (Reference 3.8-44). Water and ice used in the grout conforms to the requirements of ACI 349-06 (Reference 3.8-8). Water-reducing and/or retarding admixtures conform to ASTM C494.

3.8.4.6.1.3 Steel for Concrete Reinforcement

Steel bars for concrete reinforcement are deformed bars conforming to ASTM A 615, Grade 60, or ASTM A 706, Grade 60 (minimum yield strength of 60,000 psi). For each heat (batch) of reinforcing steel bars, certified mill test reports are provided. Additionally, for each 50 tons/bar size/heat, a minimum of one tensile test is performed.

Coated reinforcing steel is not used. Placement of concrete reinforcement is in accordance with ACI 349-06 (Reference 3.8-8), Sections 7.5 and 7.6.

3.8.4.6.1.4 Splices

Reinforcement splices comply with ACI 349-06, Chapter 12 (Reference 3.8-8). All bars are sheared or cut to the correct length shown on the bar bending schedules from continuous rolled bar stock. In general, all splices are made with a wire-tied lap of length in accordance with ACI 408R. Mechanical splices used are in conformance with ACI 439.3R. Mechanical splices develop 125% of the specified yield strength of the spliced bar. Welding of reinforcing steel, other than in the PCCV, is performed in accordance with American Welding Society (AWS) D1.4 (Reference 3.8-46).

3.8.4.6.1.5 Structural Steel

Structural steel used in other seismic category I structures conform to the following standards:

Standard	Description
ASTM A 1	Standard Specification for Carbon Steel Tee Rails
ASTM A 3	Standard Specification for Steel Joint Bars, Low, Medium, and High Carbon (Non-Heat Treated)
ASTM A 36	Standard Specification for Carbon Structural Steel
ASTM A 49	Standard Specification for Heat Treated Carbon Steel Joint Bars, Microalloyed Joint Bars, and Forged Carbon Steel Compromise Joint Bars
ASTM A 53	Standard Specification for Pipe, Steel, Black and Hot-Dipped, Zinc-Coated, Welded and Seamless
ASTM A 90	Standard Test Method for Weight (Mass) of Coating on Iron or Steel Articles with Zinc or Zinc-Alloy Coatings
ASTM A 108	Standard Specification for Steel Bars, Carbon, and Alloy Cold-Finished
ASTM A 123	Standard Specification for Zinc (Hot-Dip Galvanized) Coatings on Iron and Steel Products
ASTM A 143	Standard Practice for Safeguarding Against Embrittlement of Hot-Dip Galvanized Structural Steel Products and Procedure for Detecting Embrittlement

Standard	Description
ASTM A 153	Standard Specification for Zinc Coating (Hot-Dip) on Iron and Steel Hardware
ASTM A 240	Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels and General Applications
ASTM A 307	Standard Specification for Carbon Steel Bolts and Studs, 60,000 psi Tensile Strength
ASTM A 325	Standard Specification for Structural Bolts, Steel, Heat Treated, 120/105 ksi Minimum Tensile strength
ASTM A 354	Standard Specification for Quenched and Tempered Alloy Steel Bolts, Studs, and Other Externally Threaded Fasteners
ASTM A 449	Standard Specification for Quenched and Tempered Steel Bolts and Studs
ASTM A 490	Standard Specification for Structural Bolts, Alloy Steel, Heat Treated 150 ksi Minimum Tensile Strength
ASTM A 500	Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes
ASTM A 501	Standard Specification for Hot-Formed Welded and Seamless Carbon Steel Structural Tubing
ASTM A 563	Standard Specification for Carbon and Alloy Steel Nuts
ASTM A 572	Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel
ASTM A 588	Standard Specification for High-Strength Low-Alloy Structural Steel with 50 ksi Minimum Yield Point to 4-in Thick
ASTM A 615	Standard Specification for Deformed and Plain Carbon-Steel Bars for Concrete Reinforcement
ASTM A 653	Standard Specification for Steel Sheet, Zinc-Coated (Galvanized) or Zinc-Iron Alloy-Coated (Galvannealed) by the Hot-Dip Process
ASTM A 668	Standard Specification for Steel Forgings, Carbon and Alloy, for General Industrial Use
ASTM A 706	Standard Specification for Low-Alloy Steel Deformed and Plain Bars for Concrete Reinforcement
ASTM A 759	Standard Specification for Carbon Steel Crane Rails

Standard	Description
ASTM A 786	Standard Specification for Hot-Rolled Carbon, Low-Alloy, High-Strength Low-Alloy, and Alloy Steel Floor Plates
ASTM A 924	Standard Specification for General Requirements for Steel Sheet, Metallic-Coated by the Hot-Dip Process
ASTM A 992	Standard Specification for Structural Steel Shapes for Use in Building Framing
ASTM A 1011	Standard Specification for Steel, Sheet and Strip, Hot-Rolled, Carbon, Structural, High-Strength Low-Alloy and High-Strength Low-Alloy with Improved Formability
ASTM F 436	Standard Specification for Hardened Steel Washers
ASTM F 959	Standard Specification for Compressible-Washer-Type Direct Tension Indicators for Use with Structural Fasteners
ASTM F 1554	Standard Specification for Anchor Bolts, Steel, 36, 55, and 105-ksi Yield Strength
ASTM F 1852	Standard Specification for “Twist Off” Type Tension Control Structural Bolt/Nut/Washer Assemblies, Steel, Heat Treated, 120/105 ksi Minimum Tensile Strength

3.8.4.6.1.6 Anchors

Anchoring components and structural supports in concrete conform to following industry standards, RG 1.142 (Reference 3.8-19), and RG 1.199 (Reference 3.8-41). Expansion anchor bolts, where used, are as supplied by the manufacturer in accordance with their specifications.

ASTM A 193	Standard Specification for Alloy-Steel and Stainless Steel Bolting Materials for High Temperature Service
ASTM A 194	Standard Specification for Carbon and Alloy Steel Nuts for Bolts for High Pressure or High Temperature Service, or Both
ASTM A 307	Standard Specification for Carbon Steel Bolts and Studs, 60,000 psi Tensile Strength
ASTM A 325	Standard Specification for Structural Bolts, Steel, Heat Treated, 120/105 ksi Minimum Tensile Strength

3.8.4.6.1.7 Masonry Walls

There are no safety-related reinforced masonry walls in seismic category I structures. A non-safety related masonry wall exists in the spray pump room located at the lowest level of the R/B, which is not subjected to pressure loads and is restrained against seismic accelerations to preclude damage to safety-related SSCs.

3.8.4.6.2 Quality Control

Chapter 17 details the quality assurance program for the US-APWR.

3.8.4.6.3 Special Construction Techniques

Standard provisions of ACI are to be applied where necessary to address issues related to the use of massive concrete pours. As stated in Subsection 3.8.4.6.1.1, volume changes in mass concrete are controlled where necessary by applying measures and provisions outlined in ACI 207.2R (Reference 3.8-52) and ACI 207.4R (Reference 3.8-53). The following summarizes the construction techniques commonly associated, either singularly or in combination, with massive concrete pours such as basemats:

- Use of supplementary cementitious materials as a replacement for a portion of the Portland Cement.
- Limit the amount of Portland Cement used in Concrete mixtures through specification of 56-day compressive strengths.
- Protection of concrete surfaces from early heat loss.
- Limit the size of concrete placement.
- Use a checkerboard pattern of concrete placement in a single lift. To avoid a weak horizontal shear plane, a double lift placement of concrete, in general, is avoided. However, when it is absolutely needed to have two lifts, adequate design considerations and also, in general, shear stirrups are provided.
- Schedule concrete placements for the most advantageous day and time to control temperature rise in the concrete.
- Post-cooling can be performed by cooling the freshly placed concrete with running chilled water lines in the concrete.

3.8.4.7 Testing and Inservice Inspection Requirements

Seismic category I structures, except the PCCV, are monitored in accordance with paragraph (a)(2) of 10 CFR 50.65 (Reference 3.8-29), provided there is not significant degradation of the structure. Condition monitoring, is similar to that performed as part of the inservice inspection activities required by the ASME codes, is applied to these structures. The condition of all structures is assessed periodically. The appropriate frequency of the assessments is commensurate with the safety significance of the structure and its condition.

The COL Applicant is to establish a site-specific program for monitoring and maintenance of seismic category I structures in accordance with the requirements of NUMARC 93-01 (Reference 3.8-28) and 10 CFR 50.65 (Reference 3.8-29) as detailed in RG 1.160

(Reference 3.8-30). For seismic category I structures, monitoring is to include base settlements and differential displacements.

For water control structures, ISI programs are acceptable if in accordance with RG 1.127 (Reference 3.8-47). Water control structures covered by this program include concrete structures, embankment structures, spillway structures, outlet works, reservoirs, cooling water channels, canals and intake and discharge structures, and safety and performance instrumentation.

For seismic category I structures, it is important to accommodate ISI of representative locations. Monitoring and maintaining the condition of other seismic category I structures are essential for plant safety. Any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high-radiation areas) to accommodate ISI of other seismic category I structures are to be provided on a case-by-case basis.

For plants with nonaggressive ground water/soil (i.e., pH greater than 5.5, chlorides less than 500 ppm, and sulfates less than 1,500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and foundations is to (1) examine the exposed portions of the below-grade concrete, when excavated for any reason, for signs of degradation; and (2) conduct periodic site monitoring of ground water chemistry, to confirm that the ground water remains nonaggressive.

For plants with aggressive ground water/soil (i.e., it exceeds any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible, below-grade concrete for signs of degradation.

3.8.4.7.1 Construction Inspection

Inspections relating to the construction of seismic category I and II SSCs are conducted in accordance with the codes applicable to the construction activities and/or materials. In addition, weld acceptance is performed in accordance with the NCIG, Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, NCIG-01, Revision 3 (Reference 3.8-31).

3.8.5 Foundations

3.8.5.1 Description of the Foundations

The R/B, PCCV, east PS/B, west PS/B, ESWPC, A/B and containment internal structure are supported on a common basemat. Adjacent building basemats for the AC/B and the tank house are structurally separated by a 16 in. gap at and below grade. The T/B basemat is located approximately 20 ft, 6 in. away from the R/B complex structure.

The R/B complex basemat is located at a depth below the zone of maximum frost penetration, taken as 4 ft below grade. The COL Applicant is to determine if the site-specific zone of maximum frost penetration extends below the depth of the basemats for the standard plant, and to pour a mud mat under any basemat above the frost line so that the bottom of mud mat is below the maximum frost penetration level.

3.8.5.1.1 Reactor Building Complex

The R/B, PCCV, east PS/B, west PS/B, ESWPC, A/B and CIS are built on a common basemat and separated from adjacent AC/B and T/B. The basemat of the R/B complex is essentially a rectangular shaped reinforced concrete mat. The length of the basemat in the north-south direction is 334 ft, 7 in., and in the east-west direction at its greatest point is 413 ft-0 in., as shown in Figure 3J-1. The central region of the basemat with a diameter of approximately 187 ft supports the PCCV and CIS with a thickness of approximately 41 ft, 7 in. The peripheral portion, which supports the east PS/B, west PS/B, ESWPC and A/B is 13 ft, 4 in.

The basemat includes hollow portions such as the tendon gallery, tendon gallery access tunnel, and other portions such as in-core chase and CV recirculation sump. Since the vertical tendons are anchored at the roof of the tendon gallery, the upper part of the tendon gallery is important from the structural point of view.

The basemat reinforcement consists of a top horizontal layer of reinforcement, a bottom horizontal layer of reinforcement, and vertical shear reinforcement. The bottom layer of reinforcement is arranged in a rectangular grid. The top layer of reinforcement is arranged in a rectangular grid at the center of the PCCV and radiates outward in a polar pattern in order to avoid interference with PCCV reinforcement. The top and bottom reinforcement at the upper portion of the tendon gallery is in a polar pattern.

Outlines of the R/B, PCCV and CIS including the basemat are provided in Figures 3.8.5-1 through 3.8.5-3.

3.8.5.1.2 Deleted

3.8.5.1.3 Site Specific Structures

Other non-standard seismic category I plant buildings and structures of the US-APWR are designed by the COL Applicant based on site-specific subgrade conditions.

3.8.5.2 Applicable Codes, Standards and Specifications

The following industry codes, standards and specifications are applicable for the design, construction, materials, testing and inspections of the R/B complex basemat. Pressure retention requirements of the vessel are in accordance with the guidance from SRP 3.8.1. (Reference 3.8-7).

- Rules for Construction of Nuclear Facility Components, Division 2, Concrete Containments, Section III, American Society of Mechanical Engineers, 2001 Edition through the 2003 Addenda (hereafter referred to as ASME Code). (Reference 3.8-2).

  Note: Articles CC-1000 through CC-6000 of Section III, Division 2 are acceptable for the scope, material, design, construction, examination, and testing of concrete containments of nuclear power plants subject to the regulatory positions provided by RG 1.136 (Reference 3.8-3).

The basemat is considered an integral structure with the PCCV.

3.8.5.3 Loads and Load Combinations

Loads and load combinations are discussed in detail in Subsections 3.8.1.3 and 3.8.4.3. The containment design pressure P_d of 68 psi is included as an accident pressure in these load cases. Other load combinations applicable to the design of the basemat include acceptance criteria for overturning, sliding, and flotation as detailed in Table 3.8.5-1. The reinforced concrete basemat for the R/B complex is designed in accordance with ASME Code Section III, Division 2, Subsection CC (Reference 3.8-2).

3.8.5.4 Design and Analysis Procedures

Based on the premise that seismic category I buildings basemats are not supported on bedrock, a computer analysis of the SSI is performed for static and dynamic loads. Subsection 3.7.2 provides further information.

The seismic category I structures are concrete, shear-wall structures consisting of vertical shear/bearing walls and horizontal floor slabs designed to SSE accelerations as discussed in Section 3.7. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs. The walls then transmit the loads to the basemat. The walls also provide stiffness to the basemat and distribute the loads between them.

The applicable codes and standards for the design of the reinforced concrete basemat for R/B complex are discussed in Subsection 3.8.5.2. Other seismic category I basemats of reinforced concrete are designed in accordance with ACI 349-06 (Reference 3.8-8) and the provisions of RG 1.142 (Reference 3.8-19) where applicable. Table 3.8.5-2 identifies the material properties of concrete.

3.8.5.4.1 Properties of Subgrade

For the purposes of the US-APWR standard design, the SSI effects are captured using a representative suite of soil profiles and depths to baserock material with frequency dependent properties. The standard plant SSI and SSSI analyses also consider backfill properties as described in Subsection 3.7.2.4.1. Section 3.7.2.4 provides further discussion relating to SSI and the selection of subgrade types.

The soil profiles, due to the frequency-dependency of multiple soil layers, are variable opposed to fixed values. Documenting of typical (generic) subgrade conditions is not applicable.

A set of six (6) generic layered profiles are considered for SSI analyses, which strain-compatible properties, shear wave and compression wave velocities (Vs and Vp) and corresponding hysteretic damping values provide a wide variation of properties that addresses soil properties. The development of frequency dependent properties used in the seismic analyses is described further in Section 3.7.2.4.

The minimum allowable subgrade bearing capacity of 15,000 psf represents the maximum bearing pressures resulting from static load cases for the R/B complex common basemat, while the minimum allowable dynamic soil bearing capacity of 35,000 psf represents the maximum bearing pressure resulting from Normal plus SSE loads. These bearing pressures envelope the foundation bearing pressures for all other standard plant building structures.

3.8.5.4.2 Analyses for Basemat Loads during Operation

The major seismic category I structures basemat analyses use 3-D ANSYS FE models of the major seismic category I structures, which are described in Subsection 3.7.2.3. Non-linear contact elements are used in the FE model to determine the interaction of the R/B complex basemat with the overlying structures and with the soil subgrade. The model is capable of determining the degree of uplift of the basemat from the soil subgrade in non-linear analyses.

The three-dimensional FE model of the basemat includes the structures above the basemat and their effect on the distribution of loads on the basemat. The combined global FE model of the R/B, PCCV, A/B, PS/Bs, and ESWPC including basemat, is presented on Figures 3.8.5-5 through 3.8.5-10.

The analysis considers normal and extreme environmental loads and containment pressure loads. The normal loads include dead loads and live loads. Extreme environmental loads include the SSE.

The dead loads and the SSE loads are applied as equivalent static accelerations to the nodes of the FE model. The live loads are applied to the surface of elements as static pressure. The SSE loads are applied as equivalent static loads. For the structural design of the R/B complex basemat concrete and reinforcement, the three directions of the earthquake loading are combined using the Newmark 100-40-40 method.

The results of the linear analysis are combined with the non-linear analyses to form the governing load combinations. The results from these analyses include the forces, shears, and moments in the basemat; the bearing pressures under the basemat; and the area of the basemat that is uplifted. Minimum area of steel reinforcement is calculated from the section forces for the governing load combinations.

The required reinforcement for the R/B complex basemat is determined by considering the governing load from the combined linear and non-linear analyses.

3.8.5.4.2.1 Global Three-Dimensional FE Modeling of Basemat

The stress conditions of the basemat for the R/B complex are generated by numerous types of loads from the superstructure. The modeling of the basemat therefore involves evaluating the interaction between the basemat and the superstructures to determine the stress conditions at the interface. The global FE model is analyzed utilizing the FE computer program ANSYS (Reference 3.8-14).

The upper portion of tendon gallery is conservatively modeled using a concrete strength of 5,000 psi to simplify design while providing for the potential in variation of construction joints.

The R/B complex basemat is simulated with solid elements (ANSYS SOLID45 elements) that are defined by eight nodes having three degrees of freedom at each node; translations in the nodal x, y, and z directions. The R/B complex basemat is divided into six layers in the vertical dimension for areas away from the PCCV. The area below the PCCV is divided into 12 layers in the vertical direction. The portion modeled to simulate the reactor cavity is divided into ten layers in the vertical direction.

The R/B complex basemat is modeled with element divisions in the horizontal direction in a rectangular grid pattern, in areas away from the PCCV. The element divisions in the horizontal direction within the PCCV boundaries, mainly between the primary shield wall and the secondary shield wall, are mostly in a rectangular grid pattern. The element divisions between the secondary shield wall and the PCCV exterior wall are generally in a polar grid pattern as shown in Figure 3.8.5-5.

3.8.5.4.3 Boundary Conditions of Basemat

The basemat subgrade is included in the FE models used for structural design by meshing a sufficiently large volume of soil/rock below and around the basemat. For seismic load cases, the stiffness of the backfill soil is only activated along the face of the R/B complex basemat in the opposite direction of the applied earthquake load. The activation of soil stiffness allows the backfill soil to contribute compression to the exterior walls of the R/B complex basemat during the earthquake load condition. Dynamic lateral soil pressure at applicable locations was superimposed on seismic loads to account for soil–to–structure interactions. For basemat analysis, the equivalent static seismic accelerations are linearly reduced such that for each soil profile, the maximum shear produced by an earthquake in a given direction is 10% greater than the corresponding maximum shear values produced in the SSI analysis. The backfill and side soil are modeled from elevation 2’-7” to elevation -42’-8”. Backfill soil ranges from approximately 11’ to 14’ in width while the side soil extends from the edge of the backfill soil to the extremities of the subgrade soil. The backfill properties used for the standard design are discussed in Subsection 3.7.2.4.1. To increase computational efficiency for the non-linear analyses, the soil subgrade portion of the FE model is condensed into a super element. For all linear analyses, soil layers are modeled explicitly. The properties of the subgrade layers used in the FE model of the subgrade are established based on several profiles selected from the generic layered soil profiles described in Technical Report MUAP-10006 (Reference 3.8-85) to cover a large range of soil/rock conditions at representative nuclear power plant sites within the central and eastern US.

3.8.5.4.4 Analyses of Settlement

Maximum values of total settlement, differential settlement, and tilt are calculated for design of the standard plant structures. These quantities are calculated at the end of construction and at the end of plant operating life. All settlement and tilt values calculated for the standard plant are less than the maximum allowable values presented in Table 2.0-1 of DCD Chapter 2.

Differential settlements within the same structure are defined as the maximum difference (in the vertical direction) between settlements of any two points of the basemat. The tilt induced by differential settlement, used in calculating gap closure, is a rigid body rotation conservatively calculated as the maximum differential settlement within the same structure divided by the distance between the points on the basemat where this differential settlement occurs. Differential settlements between adjacent structures are defined as the maximum difference between settlements of any two neighboring points on the basemats, each of them on one of the adjacent structures. Differential settlements between adjacent structures are important for key connections between buildings and commodities and their supports and tunnels.

The modeling and analysis procedures for settlement account for the flexibility of structures and subgrade. Settlements are calculated by 3-D FE analysis using ANSYS for short term (resulting from dead loads introduced during plant construction) and long term static loads (acting over the operating life of plant). ANSYS FE models of both the R/B complex and the T/B are placed on a layered subgrade modeled by solid elements. The weight of the AC/B is also included in the model for the settlement analysis. The volume of subgrade included in the analysis (3000 ft by 2400 ft in a horizontal plane, and 960 ft in depth) is chosen to be sufficient to avoid the effects of boundary conditions on the resulting settlements. Two sets of 3-D settlement analyses are performed; one for a predominantly sand site and the other for a predominantly clay site. The results of the settlement analyses indicate that soil sites composed predominantly of clay layers have the maximum total and differential settlements. The deformability properties of the subgrade layers are established to simulate immediate and time dependent deformability of natural soil materials using soil deformation properties similar to profile 270-500 (see Table 3.7.1-6), which is the most deformable subgrade profile considered for the standard plant. The subgrade layers placed 500 ft or deeper below the plant grade were assigned rock properties equivalent to the corresponding layers described in profile 270-500. The settlements obtained for the 270-500 profile envelope results for all other design-basis profiles listed in Table 3.7.1-6.

The timeline of loading considered in the standard plant settlement analyses is illustrated in Figure 3.8.5-14. Subgrade settlements consist of immediate settlements that occur at load application and are elastic-plastic, and time-dependent settlements that develop in time under constant load (viscous deformations, primary consolidation settlements). All immediate settlements and most of the time-dependent settlements will occur by the time of completion of construction. To capture the relatively complex nonlinear and time-dependent behavior with a linear elastic numerical model, the soil deformation moduli used in the model are calculated as equivalent elastic secant moduli at two significant points in time: end of construction and end of plant life. The secant moduli are calculated based on primary consolidation theory and viscous deformation analysis. These secant moduli are determined in an iterative process from the condition that the average settlements of each structure at end of life and end of construction obtained from the linear analyses are approximately equal to the corresponding settlements that account for time dependent deformability and are produced after a time T_C (for the construction phase) and after a time T_L (for the entire life of the plant). As illustrated symbolically in Figure 3.8.5-15, total deformations at end of construction at every location in each structure and the subgrade, δ_{EoC}, are calculated using secant moduli at end of construction, E_{EoC}, and loading during the construction phase. Similarly, the total deformations at end of life at every location, δ_{EoL}, are calculated in a separate 3-D FE analysis using secant moduli at end of life, E_{EoL}, and loading during the plant operational life. The deformations produced during the operation life of the plant are obtained as the difference: $\delta_{EoL} - \delta_{EoC}$.

The loads considered in the settlement analyses are as follows:

- Dead Loads (D), introduced during construction and present throughout the operating life of the plant, are assumed to increase linearly from zero at the beginning of the construction period to their nominal value at end of construction.

- Live Loads (L), assumed to act with 25% of their maximum intensity considered for structural design (i.e., long term values), during the operational life of the plant.

- Weight of the backfill (B) placed around the structures and acting during the operational life of the plant.

- Heave produced by stress reduction due to excavation that reduces settlements for clay soils (materials with large time-dependent deformations), and is accounted for by calculating an equivalent reduction in loads, where heave is applicable.

- Groundwater level and the resulting buoyant loads on the structure tend to reduce settlement. For the purpose of settlement calculations, the groundwater level has been conservatively assumed to be below the basemat elevation.

Early stages of basemat construction are most vulnerable to differential loading and deformations. The construction of the basemat is anticipated to be a continuous concrete placement. The differential settlement is susceptible immediately following the concrete placement when the ratio of the slab depth to length is very small. Measures to prevent settlement are implemented by dewatering the excavation pit and maintaining it dry during basemat placement, curing, and construction of exterior walls.

In the event of suspended or sequenced construction, the basemat may remain unstiffened by the lack of shear walls for extended periods. Differential stresses in the basemat are also possible based on construction sequence, such as tension maximized on the top of the basemat due to the placement of foundation walls along the edge without additional mass and shear walls in the center of the basemat. The design of the basemat is sufficiently reinforced to control both compressive and tensile stresses until such time as the concrete placement of basemat walls and containment internal structure are completed. Therefore, the potential for differential settlement is controlled during alternative construction scenarios, until the basemat is stiffened by transverse shear walls.

Actual total and differential settlements are dependent on site-specific conditions (e.g., soil variability, construction sequence and schedule (including basemat stiffening), loading conditions, excavation plans, and dewatering plans). The COL Applicant is to perform settlement analysis for the specific site, the in-situ soil properties, and for the specific construction schedule to verify that the site-specific total and differential settlements, and tilt, are bounded by the settlements and tilt in Table 2.0-1 of Chapter 2. If the site-specific settlements and tilt are bounded by the values in Table 2.0-1, detailed site-specific stress and gap closure verifications are not required with regard to settlement and tilt effects.

3.8.5.4.5 Verification of Critical Sections

The basemat is designed to meet the acceptance criteria presented in Subsection 3.8.5.5. For the R/B complex basemat, Table 3.8.5-4 provides critical section thickness and reinforcement steel to concrete ratio used in the evaluation. Figures 3.8.5-11 and 3.8.5-12 show the basemat reinforcement arrangement of SECTION N-S and SECTION E-W, respectively. The basemat reinforcement arrangement in typical peripheral areas is detailed on Figure 3.8.5-13.

3.8.5.4.6 Design Report

A Design Report prepared in accordance with guidance from Appendix C to SRP 3.8.4 (Reference 3.8-40) provides design and construction information more specific than that contained within this DCD. The Design Report information quantitatively presents the actual design computations and the final design results. In addition, the Design Report provides criteria for reconciliation between design and as-built conditions.

3.8.5.5 Structural Acceptance Criteria

Structural acceptance criteria are discussed in detail in Subsections 3.8.1.5 and 3.8.4.5. The design soil conditions are as provided in Section 2.5 and Subsection 3.7.1.3. The COL Applicant is to ensure that the design parameters listed in Chapter 2, Table 2.0-1, envelope the site-specific conditions.

Seismic category I and II structures are evaluated against acceptance criteria with respect to overturning, sliding, and flotation stability. The load combinations applicable to the stability evaluations are specified in Table 3.8.5-1. For each of the specified load combinations, the acceptance criterion for the overturning, sliding, and flotation stability evaluations is the minimum factor of safety identified in Table 3.8.5-1. The design methodology and requirements for calculating the factors of safety are described further in Subsections 3.8.5.5 below. The minimum calculated factor of safety for each load combination considered in the stability evaluations is presented in Table 3.8.5-6. Site-specific stability evaluations are required to be performed by the COL Applicant for standard plant seismic category I and II structures to confirm the minimum required values in Table 3.8.5-1, unless the COL Applicant can demonstrate that the site specific conditions for evaluating stability are enveloped by the standard plant design. The COL Applicant is to also provide the factors of safety for site-specific seismic category I structures in Table 3.8.5-6 based on the methodology and acceptance criteria presented in Subsection 3.8.5.5.

3.8.5.5.1 Overturning Acceptance Criteria

The factor of safety against overturning is identified as the ratio of the moment resisting overturning (M_r) divided by the overturning moment (M_o). Therefore,

FS_o = [M_r / M_o] , not less than FS_{ot} as determined from Table 3.8.5-1.

where

FS_o = Structure factor of safety against overturning by the maximum design basis severe wind, tornado, hurricane, or earthquake load.

M_r = Resisting moment provided by the dead load of the structure, minus the buoyant force created by the design ground water table.

Passive earth pressure is not considered for overturning stability.

M_o = Overturning moment caused by the maximum design basis severe wind, tornado, hurricane or earthquake load.

The calculated minimum factors of safety presented in Table 3.8.5-6 show that the SSE load combination governs over wind and tornado load combinations for evaluating overturning stability. The standard plant SSE overturning stability evaluations are performed using the dynamic FE models and the seismic driving forces/moments obtained from the site independent SSI analyses. The SSI analyses are conducted separately for each earthquake direction. The earthquake responses from the separate SSI analyses are then applied simultaneously to evaluate overturning stability. The SSE overturning stability analyses, which are based on loads/masses extracted from the SSI analyses, include 25% of the live load in calculating both the resisting moment and the overturning moment. The presence of live loads has insignificant effect on the calculated overturning factor of safety because live loads make up an insignificant portion of the mass considered. Further, even though live loads have a stabilizing effect by increasing the overturning resisting moment, they increase the overturning moment. Therefore, the effects of live loads on overturning stability are insignificant.

Unbalanced lateral earth pressures are included in the analyses. This means that the overturning stability analysis considers the contribution of static soil pressure (at-rest lateral earth pressure), lateral earth pressure due to surcharge of 450 psf, and dynamic (Wood's) pressure acting in the same direction as the horizontal inertia forces on the below-grade walls and basemat, but conservatively considers only static at-rest pressure in resisting overturning loads. This is conservative because any passive reaction forces acting on the side walls and basemat below grade will reduce the global overturning effects during the stability analysis. Soil pressures acting on the below grade side walls and basemat are considered for the strength design as discussed in Subsection 3.8.4.4. The effects of basemat uplift are included at every time step by determining the reduction in contact area due to the time varying vertical force (up or down) and moments. The overturning safety factors are calculated at each time step of the design earthquake excitation as the ratio between the resisting moments and the driving/overturning moments. The minimum value of the safety factor during the total duration of the earthquake for any of the design soil conditions is reported in Table 3.8.5-6.

3.8.5.5.2 Sliding Acceptance Criteria

The factor of safety against sliding caused by wind, tornado or hurricane is identified by the ratio:

FS_{sw} = [F_s] / F_h , not less than FS_{sl} as determined from Table 3.8.5-1,

where

FS_{sw} = Structure factor of safety against sliding caused by severe wind, tornado or hurricane

F_s = Shear (or sliding) resistance along bottom of structure basemat. No credit is taken for side wall friction or passive soil pressure in calculating the factor of safety against sliding in standard plant building structures.

F_h = Lateral force due to active soil pressure, including surcharge, and tornado, hurricane or severe wind load, as applicable

The factor of safety against sliding caused by earthquake is identified by the ratio:

FS_{se} = [F_s] / [F_d + F_h], not less than FS_{sl} as determined from Table 3.8.5-1, unless resulting sliding displacements are evaluated for design acceptability.

where

FS_{se} = Structure factor of safety against sliding caused by earthquake

F_s = Shear (or sliding) resistance along bottom of structure basemat. No credit is taken for side wall friction or passive soil pressure in calculating the factor of safety against sliding in standard plant building structures.

F_d = Dynamic lateral force, including dynamic earth pressures caused by seismic loads

F_h = Other lateral forces concurrent with seismic loads

The factor of safety against sliding caused by earthquake, FS_{SE}, was calculated as shown above using a linear time history approach. This pseudo-static FS_{SE} resulted less than 1.1 during short time intervals. It was therefore decided to perform seismic sliding evaluations for the R/B complex and the T/B structures using nonlinear time history analysis that is more realistic than the pseudo-static approach.

The maximum expected seismic induced sliding for the R/B complex and the T/B resulted 0.75 in and 0.2 in, respectively [see Section 6 of Technical Report MUAP-12002 (Reference 3.8-82). The design of all aspects related to interaction between adjacent structures and components (namely: structural gaps, structural connections, such as buried tunnels and other umbilicals, buried commodities) will accommodate the displacements corresponding to the maximum expected sliding, and therefore, safety and functionality of the plant is not affected by seismic induced sliding. The nonlinear sliding analysis method and the results are documented in MUAP-12002 (Reference 3.8-82). The main features of the methodology are summarized as follows:

- Three-dimensional nonlinear time history sliding analyses are performed, including seismic acceleration input in two orthogonal horizontal directions and in the vertical direction, and also rocking.

- The nonlinear sliding analyses are performed with the 3-D FE model used for SSI analyses that accurately represents the dynamic characteristics of the structure.

- Sliding analyses are performed for all six generic layered subgrade profiles that envelope the range of soil and rock properties at the sites considered for the US-APWR standard plant. Both cracked and uncracked concrete section properties are considered for the structures analyzed. For the T/B, it was demonstrated that the uncracked section clearly dominates sliding (Section 5.3.2.3 of MUAP-12002, Reference 3.8-82). Therefore the T/B with cracked section was not analyzed for all cases.

- Five sets of acceleration time histories are used for each subgrade profile and each set of concrete section properties. The acceleration time histories for nonlinear sliding analysis are developed to be compatible with the CSDRS at 5% damping and in compliance with SRP 3.7.1, Acceptance Criteria II.1.B, Option 2, following the Criteria in Section II.1.B, Option 1, Approach 1, and Option 1, Aproach 2, Paragraph (a).

- The acceleration input motion applied at the foundation of the 3-D FE model is developed from the response of linear SSI analyses that assume perfect bonding between structure and subgrade. It is demonstrated that this decoupled analysis does not affect the results in terms of sliding and does not produce under-conservative results as compared to a fully coupled analysis.

- All input motions to sliding analyses were amplified by a factor of 1.1. This is a conservative amplification factor as discussed in Section 6.3 of Technical Report MUAP-12002 (Reference 3.8-82).

- Sliding and uplift between structure and subgrade are simulated by the mathematical model by using no-tension contact elements with Coulomb friction at the basemat-subgrade interface. The friction coefficient at the interface is 0.5. This is the value of the kinetic friction coefficient determined based on the results of a large number of laboratory and large scale tests available in literature to conservatively envelope all types of subgrade materials considered for the US-APWR standard plant design. This kinetic friction coefficient is conservatively used throughout the nonlinear sliding analyses for both sliding and non-sliding phases.

- The nonlinear sliding analyses are based on loads used in the SSI analyses and therefore include 25% of live loads. As demonstrated in Appendix B of Technical

Report MUAP-12002 (Reference 3.8-82), the effects of live loads on sliding analysis results are insignificant.

- Buoyant forces corresponding to a maximum groundwater level at one foot below plant grade are conservatively considered in the sliding analysis.

- No credit is taken for side wall friction or passive soil resistance.

- The results are processed in terms of the absolute maximum sliding in each run. The resulting sliding displacement is calculated separately for soil profiles (270-500, 270-200 and 560-500) and for rock profiles (900-200, 900-100 and 2032-100). The sliding for each type of subgrade (soil or rock) is the envelope of two results: maximum value in each sample - from all acceleration time histories, and the maximum expected value with probability of 2.5% of being exceeded. The net sliding values are once more enveloped over both subgrade types to obtain the maximum expected sliding for the Standard Plant, as discussed in Section 6.3 of Technical Report MUAP-12002 (Reference 3.8-82).

Nonlinear sliding analysis was performed with the FE model used in SSI analyses (Dynamic FE model), for both the R/B complex and the T/B. The Dynamic FE model was modified for sliding (and termed “FE model”) as described in Technical Report MUAP-12002. Two lump mass stick models were developed for the R/B complex (one for the cracked section and one for the uncracked section properties) and used for screening the most representative cases to be analyzed with the FE model. These lump mass stick models were also used for a series of sensitivity analyses described in Appendix B of the Technical Report. All sliding analyses for the T/B were performed using the FE model and therefore no lump mass stick model was developed for this structure.

The development and calibration of the lump mass stick model for the R/B complex are presented in Appendix A of Technical Report MUAP-12002. The lump mass stick model was calibrated based on the FE model by matching the dynamic properties, and was subsequently fine tuned and validated by comparing the calculated maximum sliding obtained with the LMSM with the corresponding values calculated with the FE model. Further verifications were performed on (1) comparison of overall seismic demands, and (2) comparisons of base reactions between the lump mass stick model with fixed base and the FE model used in the SSI analyses. These lump mass stick models are used for screening and are validated only for sliding analyses. The lump mass stick models are not appropriate for inferring any structural responses other than seismic induced sliding.

The standard plant non-linear sliding stability calculations use a friction coefficient of 0.5, which is a kinetic coefficient of friction. To ensure an adequate friction coefficient is achieved, the following requirements apply to the subgrade conditions at the plant site:

- A minimum 35° internal friction angle is required for natural (in-situ) or engineered granular soil materials

- Fine-grained materials i.e., silts and clays classified as ML, CL, MH, CH in the Unified Soil Classification System immediately below the basemat will be replaced by granular backfill having a minimum 35° internal friction angle. The backfill will be specified to be 4 to 6 inches thick with a maximum of 1 foot. This fill will be topped with a 3 to 4 inch mud mat. This layer must be made of well graded clean sand and/or gravel with at most 5% fines. The degree of compaction is determined to ensure that the dynamic properties of the backfill are similar to that of the native soil. Backfill placed below the R/B Complex and the T/B basemats shall be compacted to a dry density of at least 95% of the maximum dry density obtained from ASTM D1557 (Reference 3.8-84), to within 3 percent of its optimum moisture content. At least one field density test shall be performed for every 200 cubic yards of backfill placed.

The COL applicant shall verify that: (1) the degree of compaction for the backfill placed beneath foundation has to be analyzed by field density tests only, since shear wave velocity or SPT measurements cannot be performed for such a thin layer of soil and (2) the friction resistance requirement, specified as a friction angle of at least 35°, is met.

These requirements apply to backfill placed under the mats of the R/B Complex and the T/B to ensure that the dynamic properties are similar to the native soil.

- At basemat or mud mat interfaces with rock, the rock surface must be cleaned, with fissures and fractures filled in, as specified in a construction specification.
- The interface between the basemat concrete and the top surface of the mud mat must be clean and free of laitance. When a coefficient of friction > 0.6 is used in calculating sliding resistance F_s, roughening of mud mat is required per criteria given in Section 11.7.9 of ACI 349-06 (Reference 3.8-8). If a coefficient of friction ≤ 0.6 is used by the COL Applicant in a pseudo-static sliding stability analysis, roughening of mud mat is not required.

Unless the COL Applicant can demonstrate by means of pseudo-static analysis that seismic induced sliding does not occur and that a safety factor against sliding ≥ 1.1 is achieved, site-specific seismic sliding stability analyses is to be performed using the seismic sliding stability analysis methodology described in Technical Report MUAP-12002 (Reference 3.8-82). If non-linear sliding analysis is performed, the COL Applicant is to demonstrate that resulting sliding is ≤ 0.75 in. for the R/B complex and ≤ 0.20 for the T/B.

3.8.5.5.3 Flotation Acceptance Criteria

The factor of safety against flotation is identified as the ratio of the total dead load of the structure including basemat (D_r) divided by the buoyant force (F_b). Therefore,

$FS_f = D_r / F_b$, not less than FS_{fl} as determined from Table 3.8.5-1.

where

FS_f = Structure factor of safety against flotation by the maximum design basis flood or ground water table.

D_r = Total dead load of the structure including basemat.

F_b = Buoyant force caused by the design basis flood or high ground water table, whichever is greater.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Subsection 3.8.4.6 describes the materials, quality control, and special construction techniques applicable to seismic category I basemats, including water control structures and below-grade concrete walls and basemat. Subsection 3.8.1.7 provides testing and surveillance requirements relating to the R/B complex basemat.

3.8.5.7 Testing and Inservice Inspection Requirements

Subsection 3.8.4.7 identifies the testing and inservice surveillances applicable to seismic category I basemats, including water control structures and below-grade concrete walls and basemats. Subsection 3.8.1.7 also identifies testing and surveillance requirements relating to concrete crack observations of the R/B complex basemat. Monitoring and maintenance of seismic category I basemats is performed in accordance with RG 1.160 (Reference 3.8-30) to ensure that design basis assumptions and margins are not unacceptably degraded.

3.8.5.7.1 Construction Inspection

Inspection relating to the construction of seismic category I structures is in accordance with the codes applicable to the construction activities and/or materials. Subsection 3.8.4.7 contains a discussion of construction inspection requirements.

3.8.6 Combined License Information

COL 3.8(1) *Deleted*

COL 3.8(2) *Deleted*

COL 3.8(3) *It is the responsibility of the COL Applicant to assure that any material changes based on site-specific material selection for construction of the PCCV meet the requirements specified in ASME Code, Section III, Article CC-2000 of the code and supplementary requirements of RG 1.136 as well as SRP 3.8.1.*

COL 3.8(4) *Deleted*

COL 3.8(5) *Deleted*

COL 3.8(6) *Deleted*

COL 3.8(7) *It is the responsibility of the COL Applicant to determine the site-specific aggressivity of the ground water/soil and accommodate this parameter into the concrete mix design as well as into the site-specific structural surveillance program.*

COL 3.8(8) *Deleted*

COL 3.8(9) *Deleted*

COL 3.8(10) *The prestressing system is designed as a strand system, however the system material may be switched to a wire system at the choice of the COL Applicant. If this is done, the COL Applicant is to adjust the US-APWR standard plant tendon system design and details on a site-specific basis.*

COL 3.8(11) *Deleted*

COL 3.8(12) *Deleted*

COL 3.8(13) *Deleted*

COL 3.8(14) *It is the responsibility of the COL Applicant to establish programs for testing and ISI of the PCCV, including periodic inservice surveillance and inspection of the PCCV liner and prestressing tendons in accordance with ASME Code Section XI, Subsection IWL.*

COL 3.8(15) *The COL Applicant is responsible for the seismic design of those seismic category I and seismic category II SSCs not seismically designed as part of the US-APWR standard plant, including the following seismic category I structures:*

- *ESWPT*
- *UHSRS*
- *PSFSVs*

COL 3.8(16) *Deleted*

COL 3.8(17) *Deleted*

COL 3.8(18) *Deleted*

COL 3.8(19) *The design and analysis of the ESWPT, UHSRS, PSFSVs, and other site-specific structures are to be provided by the COL Applicant based on site-specific seismic criteria.*

COL 3.8(20) *The COL Applicant is to identify any applicable externally generated loads. Such site-specific loads include those induced by floods, potential non-terrorism related aircraft crashes, explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.*

COL 3.8(21) *Deleted*

COL 3.8(22)	*The COL Applicant is to establish a site-specific program for monitoring and maintenance of seismic category I structures in accordance with the requirements of NUMARC 93-01 (Reference 3.8-28) and 10 CFR 50.65 (Reference 3.8-29) as detailed in RG 1.160 (Reference 3.8-30). For seismic category I structures, monitoring is to include base settlements and differential displacements.*
COL 3.8(23)	*The COL Applicant is to determine if the site-specific zone of maximum frost penetration extends below the depth of the basemats for the standard plant, and to pour a mud mat under any basemat above the frost line so that the bottom of mud mat is below the maximum frost penetration level.*
COL 3.8(24)	*Other non-standard seismic category I buildings and structures of the US-APWR are designed by the COL Applicant based on site-specific subgrade conditions.*
COL 3.8(25)	*The COL Applicant is to ensure that the design parameters listed in Chapter 2, Table 2.0-1, envelope the site-specific conditions.*
COL 3.8(26)	*Actual total and differential settlements are dependent on site-specific conditions (e.g., soil variability, construction sequence and schedule (including basemat stiffening), loading conditions, excavation plans, and dewatering plans). The COL Applicant is to perform settlement analysis for the specific site, the in-situ soil properties, and for the specific construction schedule to verify that the site-specific total and differential settlements, and tilt, are bounded by the settlements and tilt in Table 2.0-1 of Chapter 2. If the site-specific settlements and tilt are bounded by the values in Table 2.0-1, detailed site-specific stress and gap closure verifications are not required with regard to settlement and tilt effects.*
COL 3.8(27)	*The COL Applicant is to specify normal operating thermal loads for site-specific structures, as applicable.*
COL 3.8(28)	*The COL Applicant is to specify concrete strength utilized in non-standard plant seismic category I structures.*
COL 3.8(29)	*The COL Applicant is to provide design and analysis procedures for the ESWPT, UHSRS, and PSFSVs.*
COL 3.8(30)	*When a coefficient of friction > 0.6 is used in calculating sliding resistance F_s, roughening of mud mat is required per criteria given in Section 11.7.9 of ACI 349-06 (Reference 3.8-8). If a coefficient of friction $\leq$ 0.6 is used by the COL Applicant in a pseudo-static sliding stability analysis, roughening of mud mat is not required.*

COL 3.8(31) *Site-specific stability evaluations are required to be performed by the COL Applicant for standard plant seismic category I and II structures to confirm the minimum required values in Table 3.8.5-1, unless the COL Applicant can demonstrate that the site-specific conditions for evaluating stability are enveloped by the standard plant design. The COL Applicant is to also provide the factors of safety for site-specific seismic category I structures in Table 3.8.5-6 based on the methodology and acceptance criteria presented in Subsection 3.8.5.5.*

COL 3.8(32) *Unless the COL Applicant can demonstrate by means of pseudo-static analysis that seismic induced sliding does not occur and that a safety factor against sliding $\geq$ 1.1 is achieved, site-specific seismic sliding stability analyses is to be performed using the seismic sliding stability analysis methodology described in Technical Report MUAP-12002 (Reference 3.8-82). If non-linear sliding analysis is performed, the COL Applicant is to demonstrate that resulting sliding is $\leq$ 0.75 in. for the R/B complex and $\leq$ 0.20 for the T/B.*

COL 3.8(33) *The COL applicant is to provide detailed construction and inspection plans and documents in accordance with MUAP-12006.*

COL 3.8(34) *The COL Applicant is to verify that lateral earth pressures used in the standard plant design envelope site-specific lateral earth pressures.*

COL 3.8(35) *The COL applicant shall verify that: (1) the degree of compaction for the backfill placed beneath foundation has to be analyzed by field density tests only, since shear wave velocity or SPT measurements cannot be performed for such a thin layer of soil and (2) the friction resistance requirement, specified as a friction angle of at least 35°, is met.*

3.8.7 References

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Table 3.8.1-1 US-APWR PCCV Basic Design Specification

	US-APWR	Remarks
Design Condition		
Design Pressure (P_d)	68 psig	
Test Pressure (P_t)	78.2 psig	
Design External Pressure (P)	3.9 psig	
Design External Pressure (p)	5 psid	For Equipment Hatch and Personnel Airlock Component Design
Design Accident Temperature	300°F	PCCV
Dimension		
Inner Diameter	149 ft - 2 in.	
Inner Height	226 ft - 5 in.	
Wall Thickness (Cylinder)	4 ft - 4 in.	
Wall Thickness (Dome)	3 ft - 8 in.	
Liner Thickness	0.25 in.	
Large Opening		
Equipment Hatch	ID 27 ft - 11 in.	One Set
Personnel Air Lock	ID 8 ft - 6 3/8 in.	Two Sets
Free Volume	2.80×10^6 ft^3	
Design Leakage Rate	0.1% mass/24 hours	
Design Life	60 years	
Material		
Concrete Design Strength	7000 psi	PCCV
	5000 psi	Basemat
Reinforcement	ASTM A615 or ASTM A706	
Liner Plate	SA-516 Gr. 60 or SA-516 Gr. 70	
Tendon Specification		
PS System	strand or wire	
Tendon Capacity	2.9×10^6 lb +/- 5%	
Strands	ASTM A416 Grade 270 #15 (Low Relaxation)	
Number of Strands per Tendon	49	
Number of Cylinder Hoop Tendons	94	1 ft – 6 in. Pitch
Number of Dome Hoop Tendons	18	2.5° Radial Pitch
Number of Inverted U-shape Tendons	90	2° Radial Pitch

Table 3.8.1-2 PCCV Load Combinations and Load Factors

Category	D	$L^{(1)}$	F	P_t	G	P_a	T_t	T_o	T_a	E_o	E_{ss}	W	W_t	R_o	R_a	R_r	P_v	H_a
Service																		
Test	1.0	1.0	1.0	1.0			1.0											
Construction	1.0	1.0	1.0					1.0				1.0						
Normal	1.0	1.0	1.0		1.0			1.0						1.0			1.0	
Factored																		
Severe Environmental	1.0	1.3	1.0		1.0			1.0		1.5				1.0			1.0	
	1.0	1.3	1.0		1.0			1.0				1.5		1.0			1.0	
Extreme Environmental	1.0	1.0	1.0		1.0			1.0			1.0			1.0			1.0	
	1.0	1.0	1.0		1.0			1.0					1.0	1.0			1.0	
Abnormal	1.0	1.0	1.0		1.0	1.5			1.0						1.0			
	1.0	1.0	1.0		1.0	1.0			1.0						1.25			
	1.0	1.0	1.0		1.25	1.25			1.0						1.0			
Abnormal/ Severe Environmental	1.0	1.0	1.0		1.0	1.25			1.0	1.25					1.0			
	1.0	1.0	1.0		1.0	1.25			1.0			1.25			1.0			
	1.0	1.0	1.0		1.0			1.0		1.0								1.0
	1.0	1.0	1.0		1.0			1.0				1.0						1.0
Abnormal/ Extreme Environmental	1.0	1.0	1.0		1.0	1.0			1.0		1.0				1.0	1.0		

NOTE:

1. Includes all temporary construction loading during and after construction of containment.

Table 3.8.1-3 Thermal Conditions of the R/B and PCCV (Sheet 1 of 2)

Area		Normal Operation, T_o (°F)		Accident Condition T_a (°F)	
(See Figure 3.8.1-9 for Identification of Location)		Winter	Summer	Pipe Break in Reactor Cavity (Winter, Summer)	Pipe Break in SG Compartment (Winter, Summer)
1 Annulus / Safeguard Component Room		50	105	50,130	50,130
2 CCW Pump/Heat Exchanger Room		50	105	50, 130	50, 130
3 M/D, T/D Emergency Feed Water Pump Rooms		50	105	equal to temperature during normal operation	equal to temperature during normal operation
4 Class 1E Electrical Room		50	95	equal to temperature during normal operation	equal to temperature during normal operation
4’ Class 1E UPS Room		50	95	equal to temperature during normal operation	equal to temperature during normal operation
5 Buttress Shaft		-40	115	equal to temperature during normal operation	equal to temperature during normal operation
6 Main Control Room		73	78	equal to temperature during normal operation	equal to temperature during normal operation
6’ Remote Shutdown Console Room		73	78	equal to temperature during normal operation	equal to temperature during normal operation
7 [Deleted]					
8 Class 1E I&C Room		68	79	equal to temperature during normal operation	equal to temperature during normal operation
9 [Deleted]					
10 MS/FW Piping Room		50	130	equal to temperature during normal operation	equal to temperature during normal operation
11 Safety HVAC Equipment Room		50	105	50, 130	50, 130
12 Spent Fuel Pit Water	Normal operation	120		equal to temperature during normal operation	equal to temperature during normal operation
	single failure	140		-	-
13 MG Set Room		50	95	equal to temperature during normal operation	equal to temperature during normal operation
14 Emergency Feed Water Pit Water		50	105	equal to temperature during normal operation	equal to temperature during normal operation
15 Control Rod Drive Mechanism Cabinet Room		50	95	equal to temperature during normal operation	equal to temperature during normal operation
16 Fuel Handling Area		50	105	equal to temperature during normal operation	equal to temperature during normal operation
17 CCW Surge Tank Area		50	105	50, 130	50, 130
18 R/B Atmosphere (except 1-17)		50	105	equal to temperature during normal operation	equal to temperature during normal operation

Table 3.8.1-3 Thermal Conditions of the R/B and PCCV (Sheet 2 of 2)

Area (See Figure 3.8.1-9 for Identification of Location)	Normal Operation, T_o (°F)		Accident Condition T_a (°F)	
	Winter	Summer	Pipe Break in Reactor Cavity (Winter, Summer)	Pipe Break in SG Compartment (Winter, Summer)
19 PCCV Atmosphere	105	120	Figure 3.8.1-10	Figure 3.8.1-10
20 SG Compartment Atmosphere	105	120	Figure 3.8.1-12	Figure 3.8.1-12
21 [Deleted]				
22 Reactor Cavity Atmosphere (upper)(2)	150		Figure 3.8.1-13(4)	Figure 3.8.1-12(4)
23 Reactor Cavity Atmosphere (lower)(3)	105	120	- (See No. 26)	- (See No. 26)
24 PCCV Sump Pool Water (except SG Compartment Sump, Reactor Cavity Sump and RWSP)	-		Figure 3.8.1-12	Figure 3.8.1-12
25 SG Compartment Sump Water(7)	-		Figure 3.8.1-12	Figure 3.8.1-12
26 Reactor Cavity Sump Water	-		Figure 3.8.1-13	Figure 3.8.1-13
27 RWSP Water(6)	105	120	Figure 3.8.1-11	Figure 3.8.1-11
28 C/V Sump Pump Area	105	120	- (See No. 24)	- (See No. 24)
29 Outdoor Air Temperature	-40	115	equal to temperature during normal operation	equal to temperature during normal operation
30 Basemat Side Temperature	calculated by the linear interpolation between earth temperature and outdoor air temperature			
31 Earth Temperature	35	80	equal to temperature during normal operation	equal to temperature during normal operation
32 Essential Service Water Pipe Chase	-4	140	equal to temperature during normal operation	equal to temperature during normal operation

NOTES:

1. [Deleted]
2. EL. 7'-3" to 46'-11" (atmosphere around RV)
3. Below EL. 7'-3" (atmosphere under RV)
4. Below EL. 21'-3": The temperature of "26 Reactor cavity sump water" shall be applied. (EL. 21'-3" is the maximum water level in a LOCA)
5. [Deleted]
6. The water level of the RWSP is EL. 20'-2" in a normal operation mode and EL. 7'-7" in a recirculation mode.
7. The temperature conditions of "25 SG compartment sump water" shall be applied from EL 25'-3" to EL 25'-9" in SG compartment from EL 15'-10" to EL 21'-3" in header compartment.

Table 3.8.1-4 Summary of PCCV Models and Analysis Methods

Model	Analysis Method	Program	Purpose
FE shell	Static linear and response spectrum	ANSYS	To calculate PCCV shell stress including the buttresses and vicinity of the large openings such as the equipment hatch and personnel airlocks To calculate local shell stress in vicinity of main steam pipes and feedwater pipes
FE shell	Static linear	ANSYS	To calculate local shell stress in PCCV liner plate
FE solid (basemat)	Static linear using non-linear contact elements	ANSYS	To calculate PCCV basemat stress and strain. Refer to Subsection 3.8.5.4 for further description of the basemat model.

Table 3.8.3-1 Deleted

Table 3.8.3-2 Design Pressures within CIS (Sheet 1 of 2)

Compartment No.	Compartment	Design Pressure psi
SG1	SG Compartment (25'-3" - 36'-5")	18
SG2	SG Compartment (36'-5" - 46'-5.08")	18
SG3	SG Compartment (46'-5.08" - 55'-1")	13
SG4	SG Compartment (55'-1" - 73'-1")	7
SG5	SG Compartment (73'-1" - 83'-9")	8
SG6	SG Compartment (83'-9" - 95'-1")	34
SG7	SG Compartment (95'-1" - 112'-0")	10
Pzr1	Pressurizer Surge Line Compartment (25'-3" - 58'-5")	2
Pzr2	Pressurizer Compartment (58'-5" - 76'-1")	14
Pzr3	Pressurizer Compartment (76'-1" - 89'-9")	
Pzr4	Pressurizer Compartment (89'-9" - 116'-8")	
Pzr5	Pressurizer Compartment (116'-8" - 127'-10")	
Pzr6	Pressurizer Compartment (127'-10" - 137'-8")	
V1	Inspection Gallery	39
V2	RV Annulus	14
V3	NIS Storage Box	18
V4	Lower Reactor Cavity	

Table 3.8.3-2 Design Pressures within CIS
(Sheet 2 of 2)

Compartment No.	Compartment	Design Pressure psi
RHx1	Regenerative Heat Exchanger Room	7
RHx2	Regenerative Heat Exchanger Valve Room	15
LHx1	Letdown Heat Exchanger Room	8

Table 3.8.3-3 Summary of CIS Models and Analysis Methods

Computer Program and Model	Analysis Method	Purpose	Concrete Stiffness[1]
Three Dimensional ANSYS FE of CIS fixed at elevation 3 ft, 7 in.	- Static Analysis for Mechanical Loads - Dynamic Analysis (Response Spectrum Analysis) for Seismic Loads	To obtain member forces for seismic and mechanical loads	Condition A (Operating) Condition B (Accident)
Three Dimensional ANSYS FE of CIS and R/B basemat	Static Analysis	To obtain member forces for thermal load	Condition A (Operating) Condition B (Accident)

Note:

1. See Table 3.8.3-4 for description of stiffness conditions.

Table 3.8.3-4 Summary of CIS Stiffness and Damping Values for Seismic Analysis

Structural Category	Description	Loading Condition A ($E_{ss} + T_o$)			Loading Condition B ($E_{ss} + T_a$)		
		Shear Stiffness	**Flexural Stiffness**	**Damping**	**Shear Stiffness**	**Flexural Stiffness**	**Damping**
1	SC Walls, T ≤ 56"	Uncracked $G_cA_c + G_sA_s$	Cracked-Transformed E_cI_{ct}	4%	Fully Cracked $0.5(\bar{\rho}^{-0.42})A_sG_s$	Cracked-Transformed E_cI_{ct}	5%
2	SC Walls with T > 56"	Uncracked G_cA_c	Uncracked E_cI_c	4%	Cracked $0.5G_cA_c$	Cracked $0.5E_cI_c$	7%
3	Primary Shielding	Uncracked G_cA_c	Uncracked E_cI_c	4%	Uncracked G_cA_c	Uncracked E_cI_c	4%
4	Reinforced Concrete Slabs	Uncracked G_cA_c	Uncracked E_cI_c	4%	Uncracked G_cA_c	Cracked $0.5E_cI_c$	7%
5	Massive Reinforced Concrete Sections	Uncracked G_cA_c	Uncracked E_cI_c	4%	Uncracked G_cA_c	Uncracked E_cI_c	4%
6	Steel structure with non-structural concrete fill	No Concrete Stiffness or Damping Applied					

Note: The damping values provided are those considered for SSI and SSSI analysis. Constant damping values are considered for seismic design analysis as described in Subsection 3.8.3.4.

Table 3.8.3-5 Definition of Critical Section and Thicknesses for Containment Internal Structure(1)

Wall Identifier	Applicable Wall Location	Applicable Elevation Range	Member Thickness(2)	Thickness of Face Plates Provided
Wall 1	Northeast wall of Refueling Cavity	Elevation 46'-11" to 76'-5"	4'-8" SC Wall with 0.5-in. thick steel plate on inside and outside of wall	0.5 in.
Wall 2	Northwest Wall of Secondary Shield	Elevation 50'-2" to 76'-5"	4'-0" SC Wall with 0.5-in. thick steel plate on inside and outside of wall	0.5 in.
Wall 3	Northeast Wall of RWSP	Elevation 1'-11" to 25'-3"	3'-3" SC Wall with 0.5-in. thick steel plate on inside and outside of wall	0.5 in.
Connection Identifier	**Applicable Connection Location**	**First Connected Member**	**Second Connected Member**	**Connection Design Methodology(3)**
Connection 1	SC Wall Basemat Anchorage	3'-3" SC Wall at Outside Face of RWSP	Basemat	Full Strength
Connection 2	SC Wall to SC Wall T Connection	4'-0" SC Wall between SG	4'-0" SC Wall at Outside Face of SG Compartments	Full Strength
Connection 3	Reinforced Concrete Slab to SC Wall Connection	Compartments 3'-4" Reinforced Concrete Slab at Top-of-Concrete Elevation 25'-3"	4'-0" SC Wall at Outside Face of SG Compartments	Full Strength

NOTES:

1. The applicable locations of each section are identified in Figure 3.8.3-7 (Sht 2) and Figure 3.8.3-11.
2. The member thickness includes the steel face plates.
3. Connection Design Methodology refers to the Full Strength and Overstrength design approaches defined in Technical Report MUAP-11020 (Reference 3.8-72).

Table 3.8.3-6 Deleted

Table 3.8.3-7 Summary of Confirmatory Physical Test Results (Sheet 1 of 3)

Test Series	Acceptance Criteria	Summary of Test Results
1.0 - Pushout Test	The push-out tests are to experimentally confirm that the shear strength of steel headed shear studs used in US-APWR SC walls can be calculated conservatively using MUAP-11019, Equation 2.3-1. As explained in MUAP-11019, this equation is based on ACI 349-06 Appendix D.6.1 Equation D-18 and the applicable resistance factor of 0.75 from ACI 349-06 Appendix D.4.5.	ACI 349-06 previsions are conservative when compared to test results; SP1.1 : Tie bar oriented parallel to the force direction, Acceptance Ratio = 1.34 SP1.2 : Tie bar oriented perpendicular to the force direction, Acceptance Ratio = 1.84
2.1 - Scaled Out-of-Plane Shear Tests	Out of Plane (OOP) Shear Scaled Tests is to experimentally confirm that the out-of-plane shear strength of USAPWR SC walls with their specific rectangular tie bar details can be predicted conservatively using ACI 349-06 code equations, modified by technical report MUAP-11019, Section 6.2.	ACI 349-06 provisions modified by MUAP-11019 are conservative when compared to test results; SP2.1.1 : Tie bars oriented perpendicular to the specimen length, a/d = 2.0: Acceptance Ratio = 1.52 SP2.1.2 : Tie bars oriented parallel to the specimen length, a/d = 2.0: Acceptance Ratio = 1.42 SP2.1.3 : Tie bars oriented perpendicular to the specimen length, a/d = 3.0: Acceptance Ratio = 1.26 SP2.1.4 : Tie bars oriented parallel to the specimen length, a/d = 3.0: Acceptance Ratio = 1.23
2.2 - Full Scale Out-of-Plane Shear Tests	Full Scale Out-of-Plane (Monotonic Loading) Shear Tests is to experimentally confirm that the out-of-plane strength of US-APWR SC walls with their specific rectangular tie bar detail designs is governed by flexural yielding rather than brittle shear behavior for shear span ratios greater than or equal to 2. Flexural strength of SC walls is provided by technical report MUAP-11019 Section 5.3.	The Test Series 2.2 specimens have failed in flexure, confirming the objective of the test series. Additionally MUAP-11019 provisions are conservative when compared to test results; SP2.2.1 : Tie bars oriented parallel to the specimen length, a/d = 2.0: Acceptance Ratio = 1.18 SP2.2.2 : Tie bars oriented parallel to the specimen length, a/d = 2.0: Acceptance Ratio = 1.24

Table 3.8.3-7 Summary of Confirmatory Physical Test Results (Sheet 2 of 3)

Test Series	Acceptance Criteria	Summary of Test Results
3.0 - Accident Thermal + Out-of-Plane Shear Tests	Accident Thermal + Out-of-Plane Shear Tests is to experimentally confirm that the flexural strength and out-of-plane shear strength of the typical US-APWR SC walls subjected to nonlinear thermal gradients resulting in concrete cracking can be estimated conservatively using the MUAP-11019 equations. This is discussed in MUAP-11019 Section 9.0 and is based on Section 5.3 for flexural strength and Section 6.2 for out-of-plane shear strength.	The results of the Accident Thermal + Out-of-Plane Test Series 3.0 indicate that both of the specimens exceeded the total shear strength calculated using the corresponding equations provided in MUAP 11019. SP3.1 : Tie bars oriented parallel to the specimen length (full scale), a/d = 2.0: Acceptance Ratio = 1.25 SP3.2 : Tie bars oriented parallel to the specimen length, (scaled), a/d = 2.0: Acceptance Ratio = 1.18
4.0 - Cyclic Joint Shear Test	Cyclic Joint Shear Test is to experimentally confirm that the joint shear strength of SC wall-to-wall joints of US-APWR SC walls with their specific rectangular tie bar details can be predicted using an equation given in ACI 349-06 Section 21.5.3.	ACI 349-06 provisions are conservative when compared to test results; SP4.1 : Tie bars oriented perpendicular to the specimen length, a/d = 2.0: Acceptance Ratio = 1.06
5.1 - Direct Shear Test of Anchorage Rebar- Coupler System	Direct Shear Test of Basemat Anchorage Rebar-Coupler System is to experimentally confirm that the direct shear strength of the #18 rebar-coupler system can be calculated conservatively using ACI 349-06 Appendix D.6.1 Equation D-19.	ACI 349-06 provisions are conservative when compared to TS 5.1 test results and no specimen failure occur for both specimens; SP5.1.1 : #18 single anchor specimen (full scale): Acceptance Ratio = 1.27 SP5.1.2 : #11 double anchor specimen (scaled): Acceptance Ratio = 1.35

Table 3.8.3-7 Summary of Confirmatory Physical Test Results (Sheet 3 of 3)

Test Series	Acceptance Criteria	Summary of Test Results
5.2 - Transverse Shear Test of SC Wall-to-Basemat Anchorage	Transverse Shear Test of SC Wall-to-Basemat Anchorage is to experimentally confirm that the transverse shear strength of the SC Wall-to-Basemat Anchorage is governed by the inelastic behavior and yielding of the SC wall rather than the failure of the basemat anchorage. This is achieved by demonstration of the “full strength” connection requirements defined in MUAP-11020.	The specimen developed the expected transverse shear capacity of the SC wall system. The SC wall showed inelastic behavior and yielding of the faceplate at the base of the wall when it reached final failure. The Basemat anchors demonstrated elastic behavior throughout the whole process of the test.
6. 0 - Cyclic In-Plane Shear Test of SC Anchorage	The objective of Test Series 6 – Cyclic In-Plane Shear Test of SC Anchorage is to experimentally confirm that the in-plane shear behavior of the SC Wall-to-Basemat Anchorage is governed by the inelastic behavior and yielding of the SC wall rather than the failure of the basemat anchorage. This is achieved by demonstration of the “full strength” connection requirement defined in MUAP-11020.	The specimen developed the expected lateral load capacity of the SC Wall but it could not sustain it with increasing cycles of inelastic deformation. The specimen demonstrated that the SC wall anchorage system remained in the elastic range of the response up to the peak load. The SC wall underwent significant yielding and inelastic strains at the base of the SC wall. However, this good energy dissipating behavior of the SC wall was truncated abruptly by the fracture of the weld between the steel faceplate and the steel baseplate.

Note:

1. SP: Specimen
2. Acceptance Ratio is conservatively obtained by comparing specimen nominal strength (i.e., without resistance factor) and test results (specimen nominal strength / test result), so it is acceptable that the ratio is greater than one (1).

Table 3.8.4-1 Deleted

Table 3.8.4-2 Thermal Conditions of the PS/Bs

Area (See Figure 3.8.4-1 for Identification of location)	Normal operation °F (°C)		Accident condition °F (°C)	
	Winter	Summer	Winter	Summer
1. Essential Chiller Unit Area	50 (10)	105 (40.6)	equal to temperature during normal operation	equal to temperature during normal operation
2. GTG Auxiliary Component Room	50 (10)	105 (40.6)	equal to temperature during normal operation	equal to temperature during normal operation
3. Class 1E Battery Room	65 (18.3)	77 (25)	equal to temperature during normal operation	equal to temperature during normal operation
4. AAC Power Source Starter Battery Room	65 (18.3)	77 (25)	50 (10)	105 (40.6)
5. Class 1E Battery Charger Room	50 (10)	95 (35)	equal to temperature during normal operation	equal to temperature during normal operation
6. Spare Battery Charger Room	50 (10)	95 (35)	equal to temperature during normal operation	equal to temperature during normal operation
7. AAC Selector Circuit Panel Room	50 (10)	95 (35)	50 (10)	105 (40.6)
8. Class 1E GTG Room	50 (10)	105 (40.6)	50 (10)	120 (48.9)
9. ACC GTG Room	50 (10)	105 (40.6)	50 (10)	120 (48.9)
10. Tray Space	50 (10)	105 (40.6)	equal to temperature during normal operation	equal to temperature during normal operation
11. Class 1E MOV Inverter Room	50 (10)	95 (35)	equal to temperature during normal operation	equal to temperature during normal operation
12. Essential Service Water Pipe Chase	-4 (-20)	140 (60)	equal to temperature during normal operation	equal to temperature during normal operation
13. Outdoor Air Temperature	-40 (-40)	115 (46.1)	equal to temperature during normal operation	equal to temperature during normal operation
14. Basemat Side Temperature	calculated by the linear interpolation between earth temperature and outdoor air temperature			
15. Earth Temperature	35 (1.7)	80 (26.7)	equal to temperature during normal operation	equal to temperature during normal operation
16. PS/B Atmosphere (except 1 to 12)	50 (10)	105 (40.6)	equal to temperature during normal operation	equal to temperature during normal operation

Table 3.8.4-3 Load Combinations and Load Factors for Seismic Category I Concrete Structures

LOAD COMBINATIONS AND FACTORS(1)(2)												
ACI 349-06 Load Combination:		**1**	**2**	**3**	**4**	**5(7)**	**6(6)**	**7(6)(7)**	**8(6)(7)**	**9**	**10**	**11**
Load Type												
Dead	D	1.4	1.4	1.4	1.0	1.0	1.0	1.0	1.0	1.05	1.05	1.05
Liquid	F	1.4	1.4	1.4	1.0	1.0	1.0	1.0	1.0	1.05	1.05	1.05
Live	L	1.7	1.7	1.7	1.0	1.0	1.0	1.0	1.0	1.3	1.3	1.3
Earth	H	1.7	1.7	1.7	1.0	1.0	1.0	1.0	1.0	1.3	1.3	1.3
Design pressure	P_d											
Normal pipe reactions	R_o	1.7	1.7	1.7	1.0	1.0				1.3	1.3	1.3
Normal thermal	T_o				1.0	1.0				1.2(5)	1.2(5)	1.2(5)
Severe wind	W			1.7(9)								1.3(9)
OBE	E_{ob}		1.7(3)					1.15(3)			1.3(3)	
SSE	E_{ss}				1.0(4)				1.0(4)			
Tornado or Hurricane	W_t					1.0(10)						
Accident pressure	P_a						1.4(5)	1.15	1.0			
Accident thermal	T_a						1.0	1.0	1.0			
Accident thermal pipe reactions	R_a						1.0	1.0	1.0			
Pipe rupture reactions	Y_r							1.0	1.0			
Jet impingement	Y_j							1.0	1.0			
Pipe Impact	Y_m							1.0	1.0			
Crane Load	C_{cr}	1.4			1.0(11)		1.0					
Acceptance Criteria(8)		U	U	U	U	U	U	U	U	U	U	U

Notes:
1. Design per ACI 349-06 (Reference 3.8-8), Appendix C, for all load combinations.
2. Where any load reduces the effects of other loads, the corresponding coefficient for that load is taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise the coefficient is taken as zero.
3. OBE loading is applicable for site-specific seismic category I SSCs, only if the value of site-specific OBE is set higher than 1/3 of the site-specific SSE.
4. SSE includes all seismic related hydrodynamic loads and percentage of live loads.
5. Load factor adjusted in accordance with RG 1.142, Regulatory Position 6 (Reference 3.8-19).
6. The maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m including an appropriate dynamic load factor are used, unless an appropriate time history analysis is performed to justify otherwise.
7. Satisfy the load combination first without W_t, Y_r, Y_j, and Y_m. When considering concentrated loads, exceedances of local strengths and stresses may be considered in analyses for impactive or impulsive effects in accordance with ACI 349-06 (Reference 3.8-8), Appendix F, except as noted in RG 1.142 Regulatory Positions 10 and 11.

8. The required strength U shall be equal to or greater than the strength required to resist the factored loads and/or related internal moments and forces, for each of the load combinations shown in this table.
9. Severe wind loads are per Subsection 3.3.1.
10. Extreme wind loads including tornado and hurricane loads. Velocity pressure loads, atmospheric pressure loads (tornado only) and the missile loads due to tornadoes or hurricanes are combined as described in Subsection 3.3.2. Tornado-generated missiles and hurricane-generated missiles are given in Subsection 3.5.1.4.
11. The crane load may be omitted if probability analysis demonstrates that the simultaneous occurrence of an SSE (Design Basis Event) with crane usage is not credible per Section C.2.9 of ACI 349-06 (Reference 3.8-8).

Table 3.8.4-4 Load Combinations and Load Factors for Seismic Category I Steel Structures

ALLOWABLE STRESS DESIGN (ASD) LOAD COMBINATIONS AND APPLICABLE STRESS LIMIT COEFFICIENTS													
AISC N690 Load Combination: (6)		**1**	**2**	**3**(9)	**4**(9)	**5**(9)	**6**(9)	**7**	**8**	**9**(4)	**9a**(4)(10)	**10**(4)(5)	**11**(4)(5)
Load Type													
Dead	***D***	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Live	***L***	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Normal pipe reactions	R_o		1.0			1.0	1.0	1.0	1.0				
Normal thermal	T_o		1.0			1.0	1.0	1.0	1.0				
Severe wind	***W***			1.0(13)		1.0(13)							
OBE	E_{ob}				1.0		1.0					1.0	
SSE	E_{ss}								1.0				1.0
Tornado or Hurricane	W_t							1.0(14)					
Accident pressure	P_a									1.0		1.0	1.0
Accident thermal	T_a									1.0	1.0	1.0	1.0
Accident thermal pipe reactions	R_a									1.0	1.0	1.0	1.0
Pipe rupture reactions	Y_r											1.0	1.0
Jet impingement	Y_j											1.0	1.0
Pipe Impact	Y_m											1.0	1.0
Stress Limit Coefficient (1)(2)(8)(12)		1.0(3)	1.0(3)	1.0(3)	1.0(3)	1.0(3)	1.0(3)	1.6(7)(11)	1.6(7)(11)	1.6(7)(11)	1.6(7)(11)	1.6(7)(11)	1.7(7)(11)

Notes:

1. Coefficients are applicable to primary stress limits given in ANSI/AISC N690-1994 Sections Q1.5.1, Q1.5.2, Q1.5.3, Q1.5.4, Q1.5.5, Q1.6, Q1.10, and Q1.11. Calculated stresses shall not exceed allowable stresses for each of the load combinations shown in this table.
2. In no instance shall the allowable stress exceed $0.7F_u$ in axial tension nor $0.7F_u$ times the ratio *Z/S* for tension plus bending.
3. For primary plus secondary stress, the allowable limits are increased by a factor of 1.5.
4. The maximum values of P_a, T_a, R_a, Y_j, Y_r, and Y_m, including an appropriate dynamic load factor, is used in load combinations 9 through 11, unless an appropriate time history analysis is performed to justify otherwise.
5. In combining loads from a postulated high-energy pipe break accident and a seismic event, the SRSS may be used, provided that the responses are calculated on a linear basis.
6. All load combinations is checked for a no-live-load condition.
7. In load combinations 7 through 11, the stress limit coefficient in shear shall not exceed 1.4 in members and bolts.
8. Secondary stresses which are used to limit primary stresses are treated as primary stresses.
9. Consideration is also given to snow and other loads as defined in ASCE 7.
10. This load combination is to be used when the global (non-transient) sustained effects of T_a are considered.
11. The stress limit coefficient where axial compression exceeds 20% of normal allowable, is 1.5 for load combinations 7, 8, 9, 9a, and 10, and 1.6 for load combination 11. For load combinations 7 through 11 the allowable stress shall not exceed 1.0 F_y.
12. Load combinations and stress limit coefficients are applicable for AISI design of cold-formed steel structural members used in subsystem supports. Allowable strengths per AISI may be increased by the stress limit coefficients shown, subject to the limits noted in this table. The allowable strength shall equal or exceed the required strength calculated, in accordance with AISI, for each of the load combinations shown in this table.

13. Severe wind loads are per Subsection 3.3.1.
14. Extreme wind loads including tornado and hurricane loads. Velocity pressure loads, atmospheric pressure loads (tornado only), and missile loads due to tornadoes or hurricanes are combined as described in Subsection 3.3.2. Tornado-generated missiles and hurricane-generated missiles are given in Subsection 3.5.1.4.

Table 3.8.4-5 Deleted

Table 3.8.4-6 R/B West Common Wall, SECTION 1, Details of Wall Reinforcement

	Provided Reinforcement		
	Vertical	Horizontal	Shear
WALL ZONE 1 (Concrete Thickness 40 in.) El -26'-4" → El 3'-7"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	Not Req'd
Outside Face	#11@12"+#11@12" (0.650)	#11@12"+#11@12" (0.650)	
Inside Face	#11@12"+#10@12" (0.590)	#11@12"+#10@12" (0.590)	
WALL ZONE 2 (Concrete Thickness 40 in.) El 3'-7" → El 25'-3"			
Load Combination	1.0D+1.0F+1.0L+1.0H +1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H +1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Outside Face	#11@12"+#11@12" (0.650)	#11@12"+#11@12" (0.650)	
Inside Face	#11@12" +#10@12" (0.590)	#11@12" +#10@12" (0.590)	
WALL ZONE 3 (Concrete Thickness 40 in.) El 25'-3" → El 50'-2"			
Load Combination	1.0D+1.0F+1.0L+1.0H +1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H +1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Outside Face	#11@12"+#11@12" (0.650)	#11@12"+#11@12" (0.650)	
Inside Face	#11@12" +#10@12" (0.590)	#11@12" +#10@12" (0.590)	
WALL ZONE 4 (Concrete Thickness 32 in.) El 50'-2" → El 76'-5"			
Load Combination	1.0D+1.0F+1.0L+1.0H +1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H +1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Outside Face	#11@12" +#9@12" (0.667)	#11@6" (0.813)	
Inside Face	#11@12" (0.406)	#11@12" (0.406)	
WALL ZONE 5 (Concrete Thickness 28 in.) El 76'-5" → El 101'-0"			
Load Combination	1.0D+1.0F+1.0L+1.0H +1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+1. 0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Outside Face	#11@12" +#9@12" (0.762)	#11@6" (0.929)	
Inside Face	#11@12" (0.464)	#11@12" (0.464)	

Notes:
1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.

Table 3.8.4-7 R/B South Interior Wall, SECTION 2, Details of Wall Reinforcement

	Provided Reinforcement		
	Vertical	Horizontal	Shear
WALL ZONE 1 (Concrete Thickness 44 in.) El -26'-4" → El 3'-7"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#6@12" (Vert. and Horiz.)[3]
Each Face	#11@6"+#11@12" (0.886)	#11@6"+#11@12" (0.886)	
WALL ZONE 2 (Concrete Thickness 52 in.) El 3'-7" → El 25'-3"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Each Face	#11@6" +#11@12" (0.750)	#11@6" +#11@12" (0.750)	
WALL ZONE 3 (Concrete Thickness 40 in.) El 25'-3" → El 50'-2"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Each Face	#11@6" +#9@12" (0.858)	#11@6" +#9@12" (0.858)	
WALL ZONE 4 (Concrete Thickness 40 in.) El 50'-2" → El 76'-5"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss} +T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss} +T_a	Not Req'd
Each Face	#11@12" +#9@12" (0.533)	#11@12" +#9@12" (0.533)	
WALL ZONE 5 (Concrete Thickness 40 in.) El 76'-5" → El 86'-4"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	Not Req'd
Each Face	#11@12" +#9@12" (0.533)	#11@12"+#9@12" (0.533)	
WALL ZONE 6 (Concrete Thickness 40 in.) El 86'-4" → El 101'-0"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	Not Req'd
Each Face	#11@12" +#9@12" (0.533)	#11@12"+#9@12" (0.533)	

Notes:
1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.

Table 3.8.4-8 North Exterior Wall of Spent Fuel Pit, SECTION 3, Details of Wall Reinforcement

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
WALL ZONE 1 (Concrete Thickness 93 in.) El 30'-1'' → El 50'-2''			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	Not Req'd
Inside Face	#14@12" +#14@12" (0.403)	#14@12" +#14@12" (0.403)	
Outside Face	#14@6" +#14@6" (0.806)	#14@6" +#14@6" (0.806)	
WALL ZONE 2 (Concrete Thickness 93 in.) El 50'-2'' → El 65'-0''			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	Not Req'd
Inside Face	#14@12" +#14@12" (0.403)	#14@12" +#14@12" (0.403)	
Outside Face	#14@6" +#14@6" (0.806)	#14@6" +#14@6" (0.806)	
WALL ZONE 3 (Concrete Thickness 152 in.) El 65'-0'' → El 76'-5''			
Load Combination	1.0D+1.0F+1.0L+1.0H +1.0E_{ss} +T_a	1.0D+1.0F+1.0L+1.0H +1.0E_{ss} +T_a	Not Req'd
Inside Face	#14@12" +#14@12" (0.247)	#14@12" +#14@12" (0.247)	
Outside Face	#14@6" +#14@12" (0.370)	#14@6" +#14@6"+#14@12" (0.617)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.

Table 3.8.4-9 R/B South Exterior Wall, SECTION 4, Details of Wall Reinforcement (Sheet 1 of 2)

	Provided Reinforcement		
	Vertical	Horizontal	Shear
WALL ZONE 1 (Concrete Thickness 60 in.) El -26'-4" → El 3'-7"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	#5@12" (Vert. and Horiz.)[3]
Inside Face	#11@6"+#11@12" (0.650)	#11@12"+#11@12" (0.433)	
Outside Face	#11@6"+#11@6" (0.867)	#11@6"+#11@6" (0.867)	
WALL ZONE 2 (Concrete Thickness 60 in.) El 3'-7" → El 25'-3"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	#5@12" (Vert. and Horiz.)[3]
Inside Face	#11@6" +#11@12" (0.650)	#11@12" +#11@12" (0.433)	
Outside Face	#11@6" +#11@6" (0.867)	#11@6" +#11@6" (0.867)	
WALL ZONE 3 (Concrete Thickness 60 in.) El 25'-3" → El 50'-2"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	#5@12" (Vert. and Horiz.)[3]
Inside Face	#11@12" +#11@12" (0.433)	#11@12" +#11@12" (0.433)	
Outside Face	#11@6" +#11@6" (0.867)	#11@6" +#11@12" (0.650)	
WALL ZONE 4 (Concrete Thickness 60 in.) El 50'-2" → El 76'-5"			
Load Combination	1.0D+1.0F+1.0L+1.0H +$1.0E_{ss}$ +T_a	1.0D+1.0F+1.0L+1.0H +$1.0E_{ss}$ +T_a	#5@12" (Vert. and Horiz.)[3]
Inside Face	#11@12" +#11@12" (0.433)	#11@12" +#11@12" (0.433)	
Outside Face	#11@6" +#11@6" (0.867)	#11@6" +#11@12" (0.650)	
WALL ZONE 5 (Concrete Thickness 60 in.) El 76'-5" → El 101'-0"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	#5@6" (Vert. and Horiz.)[3]
Inside Face	#11@12" +#11@12" (0.433)	#11@6" +#11@12" (0.650)	
Outside Face	#11@6" +#11@12" (0.650)	#11@6"+#11@12" (0.650)	

Table 3.8.4-9 R/B South Exterior Wall, SECTION 4, Details of Wall Reinforcement (Sheet 2 of 2)

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
WALL ZONE 6 (Concrete Thickness 60 in.) El 101'-0" → El 115'-6"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	1.0D+1.0F+1.0L+1.0H+ $1.0E_{ss}+T_a$	#5@6" (Vert. and Horiz.)[3]
Inside Face	#11@12" +#11@12" (0.433)	#11@6" +#11@12" (0.650)	
Outside Face	#11@6" +#11@12" (0.650)	#11@6"+#11@6" (0.867)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.

Table 3.8.4-10 R/B Central Interior Wall, SECTION 5, Details of Wall Reinforcement

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
WALL ZONE 1 (Concrete Thickness 40 in.) El -26'-4" → El 3'-7"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Each Face	#11@6"+#11@12" (0.975)	#11@6"+#11@12" (0.975)	
WALL ZONE 2 (Concrete Thickness 40 in.) El 3'-7" → El 25'-3"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Each Face	#11@6"+#11@12" (0.975)	#11@6"+#11@12" (0.975)	
WALL ZONE 3 (Concrete Thickness 40 in.) El 25'-3" → El 50'-2"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Each Face	#11@6"+#11@12" (0.975)	#11@6"+#11@12" (0.975)	
WALL ZONE 4 (Concrete Thickness 40 in.) El 50'-2" → El 76'-5"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Each Face	#11@6"+#11@12" (0.975)	#11@6"+#11@12" (0.975)	
WALL ZONE 5 (Concrete Thickness 40 in.) El 76'-5" → El 101'-0"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Each Face	#11@6"+#7@12" (0.775)	#11@6"+#7@12" (0.775)	
WALL ZONE 6 (Concrete Thickness 40 in.) El 101'-0" → El 115'-6"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#5@12" (Vert. and Horiz.)[3]
Each Face	#11@6"+#7@12" (0.775)	#11@6"+#7@12" (0.775)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.

Table 3.8.4-11 Spent Fuel Pit Slab, AREA 3, Details of Slab Reinforcement

	Provided Reinforcement		
	NS-Dir.	**EW-Dir.**	**Shear**
AREA 3 (Concrete Thickness 126 in.) El 30'-1"			
Load Combination	1.0D+1.0F+1.0L+1.0H+1.0E $_{ss}$+T_a	1.0D+1.0F+1.0L+1.0H+1.0E $_{ss}$+T_a	Not Req'd
Top & Bottom	#11@12"+#11@12"+ #11@12" (0.310)	#11@12"+#11@12"+ #11@12" (0.310)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.

Table 3.8.4-12 Emergency Feedwater Pit Slab, AREA 4, Details of Slab Reinforcement

	Provided Reinforcement		
	NS-Dir.	**EW-Dir.**	**Shear**
AREA 4 (Concrete Thickness 52 in.) El 76'-5"			
Load Combination	1.0D+1.0F+1.0L+1.0H+1.0E$_s$ $_s$+T_a	1.0D+1.0F+1.0L+1.0H+1.0E$_s$ $_s$+T_a	#6@12" (Each Way)[3]
Top & Bottom	#14@12"+#14@12" (0.721)	#14@12"+#14@12" (0.721)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this slab.

Table 3.8.4-13 Typical Reinforcement in West PS/B South Exterior Wall – SECTION 1 (On Column Line LR and Between Column Lines 1R & 3R)

	Provided Reinforcement		
	Vertical	Horizontal	Shear
WALL ZONE 1 (Concrete Thickness 40 in.) El -26'-4" → El -14'-2"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#4@12" (Vert. and Horiz.)[3]
Outside Face	#11@6" (0.650)	#9@6" (0.417)	
Inside Face	#11@6" (0.650)	#11@12" (0.325)	
WALL ZONE 2 (Concrete Thickness 40 in.) El -14'-2" → El 3'-7"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#4@12" (Vert. and Horiz.)[3]
Outside Face	#11@6" (0.650)	#11@6" (0.650)	
Inside Face	#11@6 (0.650)	#11@6" (0.650)	
WALL ZONE 3 (Concrete Thickness 40 in.) El 3'-7" → El 24'-2"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	#4@12" (Vert. and Horiz.)[3]
Outside Face	#11@6" (0.650)	#11@6" (0.650)	
Inside Face	#11@6" (0.650)	#11@6" (0.650)	
WALL ZONE 4 (Concrete Thickness 40 in.) El 24'-2" → El 39'-6"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	Not Req'd
Outside Face	#9@6" (0.417)	#11@6" (0.650)	
Inside Face	#9@6" (0.417)	#9@6" (0.417)	
WALL ZONE 5 (Concrete Thickness 21 in.) El 39'-6" → El 49'-0"			
Load Combination	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	1.0D+1.0F+1.0L+1.0H+ 1.0E_{ss}+T_a	Not Req'd
Outside Face	#9@6" (0.794)	#9@6" (0.794)	
Inside Face	#9@6" (0.794)	#9@12" (0.397)	

Notes:
1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.

Table 3.8.4-14 Typical Reinforcement in West PS/B Interior Wall – SECTION 2 (On Column Line 7R and Between Column Lines JR & LR)

	Provided Reinforcement		
	Vertical	Horizontal	Shear
WALL ZONE 1 (Concrete Thickness 20 in.) El -26’-4” → El 3’-7”			
Load Combination	1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+ T_a	1.0D+1.0F+1.0L+1.0H+1.0E_{ss} +T_a	#4@9" (Vert. and Horiz.)[3]
Each Face	#9@6” (0.833)	#10@6” (1.058)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.

Table 3.8.4-15 Typical Reinforcement in West PS/B Floor at Elevation 3′-7″– AREA 1 (Between Column Lines 3R & 5R - KR & LR)

	Provided Reinforcement		
	Top	Bottom	Shear
AREA 1 (Concrete Thickness 32 in.) El 3’-7”			
Load Combination	1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+ T_a	1.0D+1.0F+1.0L+1.0H+1.0E_{ss} +T_a	#3@12" (Each Way)[3]
Each Direction	#11@12” (0.406)	#11@12” (0.406)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this slab.

Table 3.8.4-16 Typical Reinforcement in West PS/B North Exterior Wall – SECTION 3 (On Column Line JR and Between Column Lines 1R & 3R)

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
WALL ZONE 1 (Concrete Thickness 50 in.) El -26'-4" → El -14'-2"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	#4@12" (Vert. and Horiz.)[3]
Outside Face	#11@6" (0.520)	#11@6" (0.520)	
Inside Face	#11@6" (0.520)	#11@6" (0.520)	
WALL ZONE 2 (Concrete Thickness 50 in.) El -14'-2" → El 3'-7"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	Not Req'd
Outside Face	#11@6" (0.520)	#11@6" (0.520)	
Inside Face	#11@6" (0.520)	#11@6" (0.520)	
WALL ZONE 3 (Concrete Thickness 50 in.) El 3'-7" → El 24'-2"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	Not Req'd
Outside Face	#11@6" (0.520)	#11@6" (0.520)	
Inside Face	#11@6" (0.520)	#11@6" (0.520)	
WALL ZONE 4 (Concrete Thickness 40 in.) El 24'-2" → El 39'-6"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	#4@12" (Vert. and Horiz.)[3]
Outside Face	#9@6" (0.417)	#9@6" (0.417)	
Inside Face	#9@6" (0.417)	#9@6 (0.417)	
WALL ZONE 5 (Concrete Thickness 40 in.) El 39'-6"→ El 49'-0"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	Not Req'd
Outside Face	#9@6" (0.417)	#9@6" (0.417)	
Inside Face	#9@6" (0.417)	#9@6" (0.417)	

Notes:
1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.

Table 3.8.4-17 Typical Reinforcement in East PS/B East Exterior Wall – SECTION 1 (On Column Line 20R and Between Column Lines F1R & G4R)

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
WALL ZONE 1 (Concrete Thickness 40 in.) El -26'-4" → El -14'-2"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	#4@9" (Vert. and Horiz.)[4]
Outside Face	#9@6" (0.417)	#9@6" (0.417)	
Inside Face	#9@6" (0.417)	#9@6" (0.417)	
WALL ZONE 2 (Concrete Thickness 40 in.) El -14'-2" → El 3'-7"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	#4@12" (Vert. and Horiz.)[3]
Outside Face	#9@6" (0.417)	#9@6" (0.417)	
Inside Face	#9@6” (0.417)	#9@6" (0.417)	
WALL ZONE 3 (Concrete Thickness 40 in.) El 3'-7" → El 26'-11"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	Not Req'd
Outside Face	#9@6" (0.417)	#11@6" (0.650)	
Inside Face	#9@6" (0.417)	#11@12" (0.325)	
WALL ZONE 4 (Concrete Thickness 40 in.) El 26'-11" → El 39'-6"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	Not Req'd
Outside Face	#9@6" (0.417)	#11@6" (0.650)	
Inside Face	#9@6" (0.417)	#11@12” (0.325)	

Notes:
1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.
4. Shear reinforcement steel is required in all areas of this wall zone.

Table 3.8.4-18 Typical Reinforcement in East PS/B Interior Wall – SECTION 2 (On Column Line K1R and Between Column Lines 18R & 20R)

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
WALL ZONE 1 (Concrete Thickness 40 in.) El -26'-4" → El 3'-7"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	Not Req'd
Each Face	#11@6" (0.650)	#11@6" (0.650)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.

Table 3.8.4-19 Typical Reinforcement in East PS/B Floor at Elevation 3'-7" – AREA 1 (Between Column Lines 19R & 20R – G4R & JR)

	Provided Reinforcement		
	Top	**Bottom**	**Shear**
AREA 1 (Concrete Thickness 32 in.) El 3'-7"			
Load Combination	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	$1.0D+1.0F+1.0L+1.0H+1.0E_{ss}+T_a$	#5@6" (Each Way)[3]
Each Direction	#9@12" (0.260)	#9@12" (0.260)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this slab.

Table 3.8.4-20 Typical Reinforcement in ESWPC Exterior Wall – SECTION 1

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
WALL ZONE 1 (Concrete Thickness 36 in.) El -26'-4" → El -15'-8"			
Outside Face	#11@6" (0.722)	#11@6" (0.722)	#7@6" (Vert. and Horiz.)[4]
Inside Face	#11@6" (0.722)	#11@6" (0.722)	
WALL ZONE 2 (Concrete Thickness 36 in.) El -15'-8" → El -9'-8"			
Outside Face	#11@6" (0.722)	#11@6" (0.722)	#7@12" (Vert. and Horiz.)[4]
Inside Face	#11@6" (0.722)	#11@6" (0.722)	
WALL ZONE 3 (Concrete Thickness 36 in.) El -9'-8" → El 1'-7"			
Outside Face	#9@6" (0.463)	#9@6" (0.463)	Not Req'd
Inside Face	#9@6" (0.463)	#9@6" (0.463)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Not used.
4. Shear reinforcement steel is required in all areas of this wall zone.

Table 3.8.4-21 Typical Reinforcement in ESWPC Interior Wall – SECTION 2

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
WALL ZONE 1 (Concrete Thickness 24 in.) El -9'-8" → El 1'-7"			
Outside Face	#11@6" (1.083)	#11@6" (1.083)	#6@12" (Vert. and Horiz.)[3]
Inside Face	#11@6" (1.083)	#11@6" (1.083)	

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Shear reinforcing steel is required in localized areas of this wall zone.

Table 3.8.4-22 Typical Reinforcement in ESWPC Floor at Elevation -15’-8” – AREA 1

	Provided Reinforcement		
	Vertical	**Horizontal**	**Shear**
AREA 1 (Concrete Thickness 24 in.) El -15'-8"			
Each Direction	#11@6" (1.083)	#11@6" (1.083)	#7@6” (Each Way)[4]

Notes:

1. ( ) Indicates reinforcement to concrete ratio in terms of a percentage.
2. Load combination indicated includes all permutations of this load case combination.
3. Not used.
4. Shear reinforcement steel is required in all areas of this floor zone.

Table 3.8.4-23 Dynamic + Static Lateral Earth Pressures and Passive Earth Pressures

Elevation (ft)	Depth (ft)	Dynamic + Static Pressure (ksf)	Total Passive Pressure (ksf)
2.58	0.00	3.25	1.66
-3.00	5.58	4.19	4.23
-5.87	8.45	4.53	5.56
-8.58	11.16	4.79	6.81
-14.32	16.90	5.17	9.46
-15.58	18.16	5.24	10.04
-20.96	23.54	5.48	12.52
-22.77	25.35	5.55	13.35
-26.34	28.92	5.64	15.00
-33.01	35.59	5.63	18.08
-39.67	42.25	5.17	21.15

Table 3.8.5-1 Load Combinations and Required Minimum Factors of Safety for Stability of Seismic Category I and II Structures

Load Combination	Overturning (FS_{ot})	Sliding (FS_{sl})	Flotation (FS_{fl})
$D + H + W$	1.5	1.5	N/A
$D + H + E_{ss}$	1.1	1.1	N/A
$D + H + W_t$	1.1	1.1 (See note 1)	N/A
$D + F_b$	N/A	N/A	1.1

Note 1: Psuedo-static analyses result in a factor of safety less than 1.1 during short time intervals. More realistic non-linear sliding analyses are performed to evaluate sliding during a design-basis earthquake. All input motions in the non-linear sliding analyses are conservatively amplified by 1.1. The maximum sliding displacements are included in the design of structures, systems, components and equipment, where applicable.

Table 3.8.5-2 Concrete Properties

Part		Compressive Strength f'c	Modulus of Elasticity Ec(1)	Poisson's Ratio υ	Thermal Expansion Coefficient α	Unit Weight γ
PCCV		7,000 psi	4,769 ksi	0.17	5.5×10^{-6}/°F	150lb/ft^3
Containment Internal Structure		4,000 psi	3,605 ksi	0.17	5.5×10^{-6}/°F	150lb/ft^3
R/B		5,000 psi	4,031 ksi	0.17	5.5×10^{-6}/°F	150lb/ft^3
Basemat	Peripheral	5,000 psi	4,031 ksi	0.17	5.5×10^{-6}/°F	150lb/ft^3
	Upper part of Tendon Gallery	7,000 psi	4,769 ksi	0.17	5.5×10^{-6}/°F	150lb/ft^3

Note :

1. $Ec=57{,}000(f'c)^{1/2}$ psi (ACI 349-06, 8.5.1)

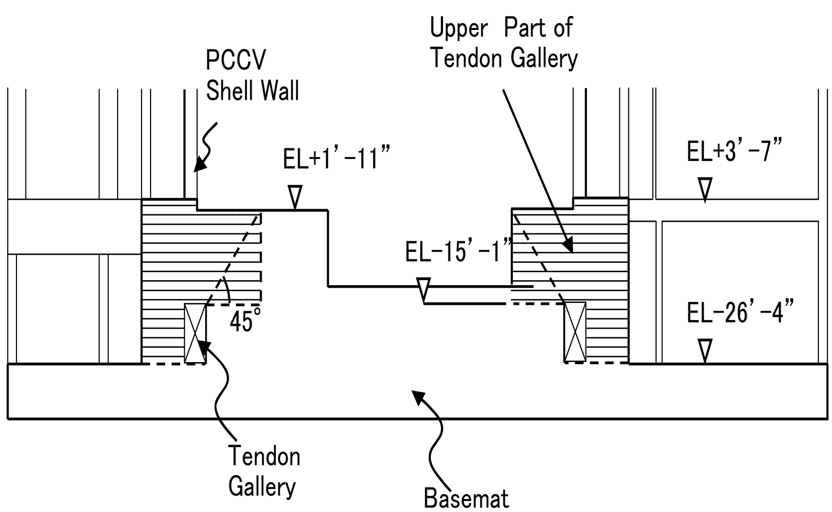


Table 3.8.5-3 Deleted

Table 3.8.5-4 Critical Sections of R/B Complex Basemat Evaluation (Sheet 1 of 2)

Location	Thickness (in)	Primary Reinforcement					Shear Tie		Control Load Case[7]
		Position	Direction 1*		Direction 2*				
			Arrange-ment	Ratio (%)	Arrange-ment	Ratio (%)	Arrange-ment	Ratio (%)	
Upper Part of Tendon Gallery[1]	204 (17')	Top	4-#18@2° +4-#18@2° +2-#18@2°	0.62	1-#14@12" +2-#18@12"	0.42	2-#11/2°@ 12"	0.82	Construction
		Bottom	2-#18@2° +2-#18@2°	0.25	3-#14@12"	0.28			
Lower Part of Tendon Gallery[2]	160 (13'-4")	Top	3-#14@2° +3-#14@2°	0.27	2-#14@12"	0.23	2-#11/2°@ 24"	0.41	Abnormal / Extreme
		Bottom	3-#18@12"	0.63	1-#14@12" +3-#18@12"	0.74			
Lower Part of Cavity[3]	326 (27'-2")	Top	2-#14@12" +3/4" LP	0.35	2-#14@12" +3/4" LP	0.35	#10@24"× 24"	0.22	Test
		Bottom	3-#18@12"	0.31	3-#18@12"	0.31			
Inside Secondary Shield Wall[4]	499 (41'-7")	Top	1-#14@12" +2-#18@12" +1/4" LP	0.22	1-#14@12" +2-#18@12" +1/4" LP	0.22	#10@24"× 24"	0.22	Test
		Bottom	3-#18@12"	0.20	3-#18@12"	0.20			
Outside Secondary Shield Wall[5]	499 (41'-7")	Top	4-#18@2° +4-#18@2° +2-#18@2°	0.22	3-#18@12"	0.20	#10@12"× 24"	0.44	Construction
		Bottom	3-#18@12"	0.20	1-#14@12" +3-#18@12"	0.24			
Outside Secondary Shield Wall[5a]	499 (41'-7")	Top	4-#18@2° +4-#18@2° +2-#18@2°	0.22	1-#14@12" 2-#18@12" +1/4" LP	0.22	#10@12"× 24"	0.44	Test
		Bottom	3-#18@12"	0.20	1-#14@12" +3-#18@12"	0.24			
Peripheral Areas[6]	160 (13'-4")	Top	2-#14@12"	0.23	2-#14@12"	0.23	#10@24"× 24"	0.22	Abnormal / Extreme
		Bottom	2-#14@12"	0.23	2-#14@12"	0.23			
Peripheral Areas[6a]	160 (13'-4")	Top	2-#14@12"	0.23	2-#14@12"	0.23	#10@12"× 12"	0.88	Abnormal / Extreme
		Bottom	3-#18@12"	0.63	2-#14@12"	0.23			
Peripheral Areas[6b]	160 (13'-4")	Top	2-#14@12"	0.23	2-#14@12"	0.23	#10@24"× 24"	0.22	Construction
		Bottom	2-#14@12"	0.23	2-#14@12"	0.23			

Note : 1. Upper Part of Tendon Gallery Direction 1: Radial, Direction 2: Circumferential

Table 3.8.5-4 Critical Sections of R/B Complex Basemat Evaluation (Sheet 2 of 2)

2. Lower Part of Tendon Gallery	Direction 1: Top: Radial, Bottom: N-S + Circumferential Direction 2: Top: Circumferential, Bottom: E-W + Circumferential
3. Lower Part of Cavity	Direction 1: N-S, Direction 2: E-W
4. Inside Secondary Shield Wall	Direction 1: N-S, Direction 2: E-W
5. 5a Outside Secondary Shield Wall	Direction 1: Top: Radial, Bottom: N-S + Circumferential Direction 2: Top: Circumferential, Bottom: E-W + Circumferential
6. 6a, 6b Peripheral Areas	Direction 1: N-S, Direction 2: E-W

7. For the controlling load cases of locations 1 through 6b, see DCD Table 3.8.1-2.

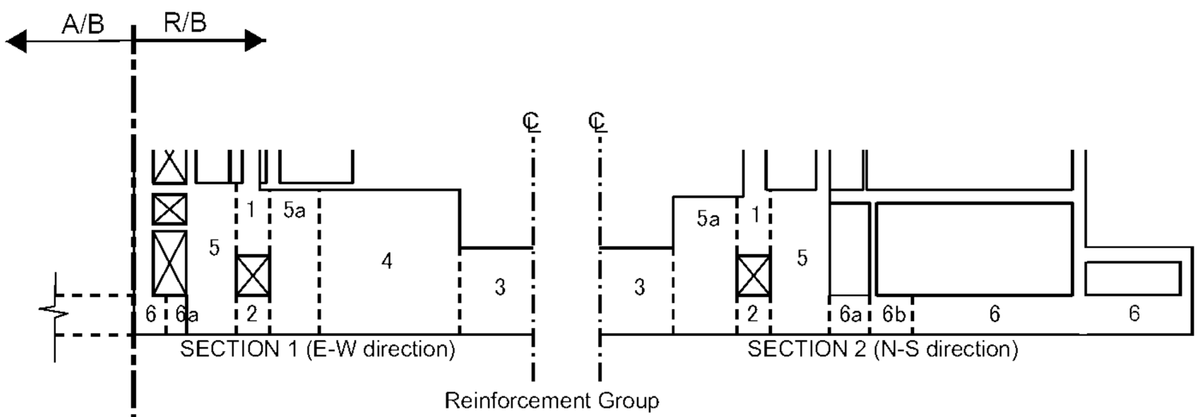


Table 3.8.5-5 Deleted

Table 3.8.5-6 Load Combinations and Calculated[1] Minimum Factors of Safety for Stability of Seismic Category I and II Structures

Building/ Structure	Load Combination	Overturning (FS_{ot})	Sliding (FS_{sl})	*Flotation* (FS_{fl})
R/B complex	$D + H + W$	>10	>10	N/A
	$D + H + E_{ss}$	1.2	See Note 2.	N/A
	$D + H + W_t$	>10	>10	N/A
	$D + F_b$	N/A	N/A	3.8
T/B	$D + H + W$	>10	>10	N/A
	$D + H + E_{ss}$	1.2	See Note 3.	N/A
	$D + H + W_t$	>5	>5	N/A
	$D + F_b$	N/A	N/A	1.9

Note 1: Factors of safety reported in this table may show values which have been conservatively rounded down from calculated values.

Note 2: Sliding analyses documented in Technical Report MUAP-12002 (Reference 3.8-82) have determined that sliding occurs. The maximum 0.75 in. sliding displacement, which has been conservatively rounded up, is utilized in conjunction with other structural displacements for the design of attached structures, piping and/or equipment, and evaluation of gaps between structures.

Note 3: Sliding analyses documented in Technical Report MUAP-12002 (Reference 3.8-82) have determined that sliding displacements occurs. The maximum 0.20 in. sliding displacement, which has been conservatively rounded up, is utilized in conjunction with other structural displacements for the design of attached structures, piping and/or equipment, and gaps between structures.

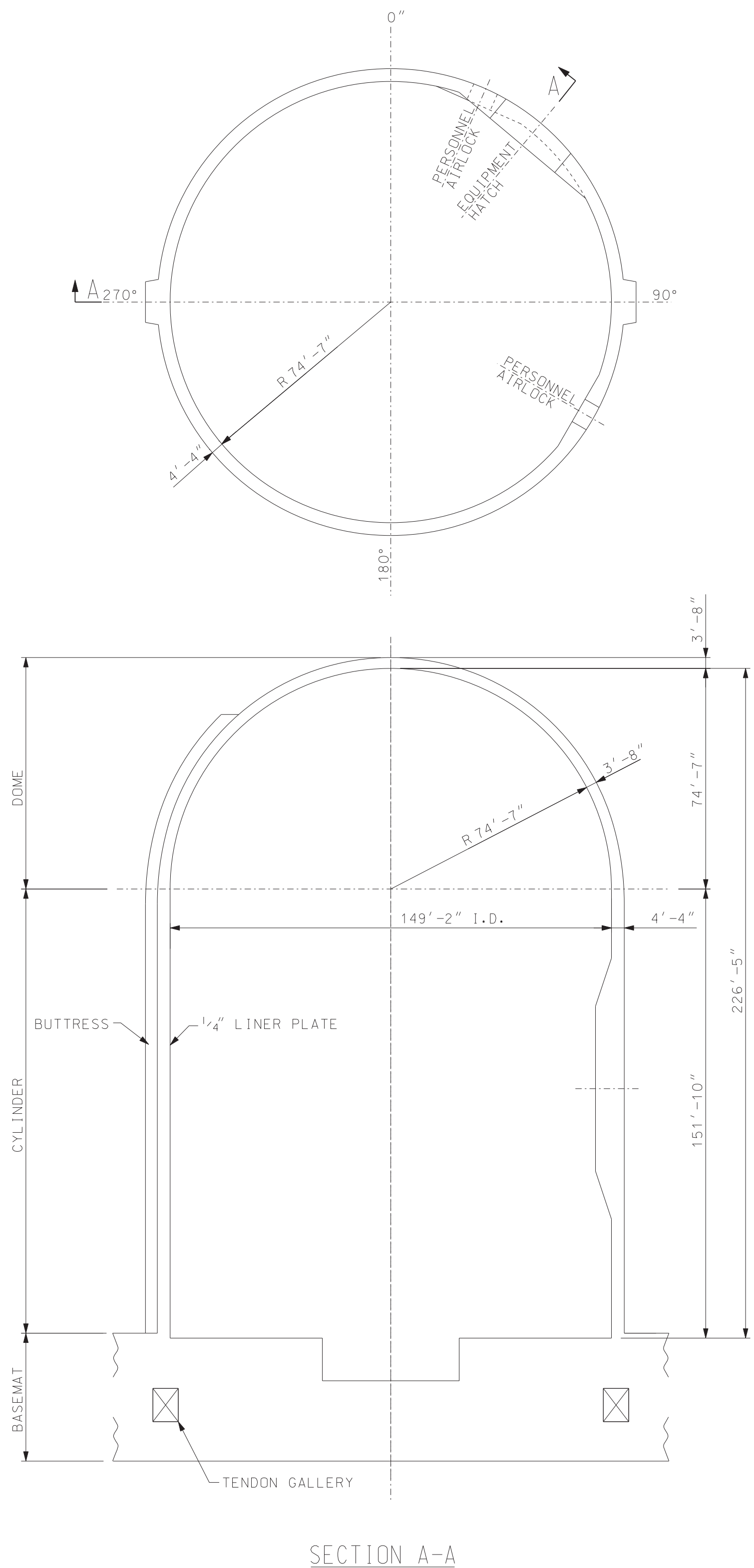


Figure 3.8.1-1 Configuration of PCCV (Sheet 1 of 3)

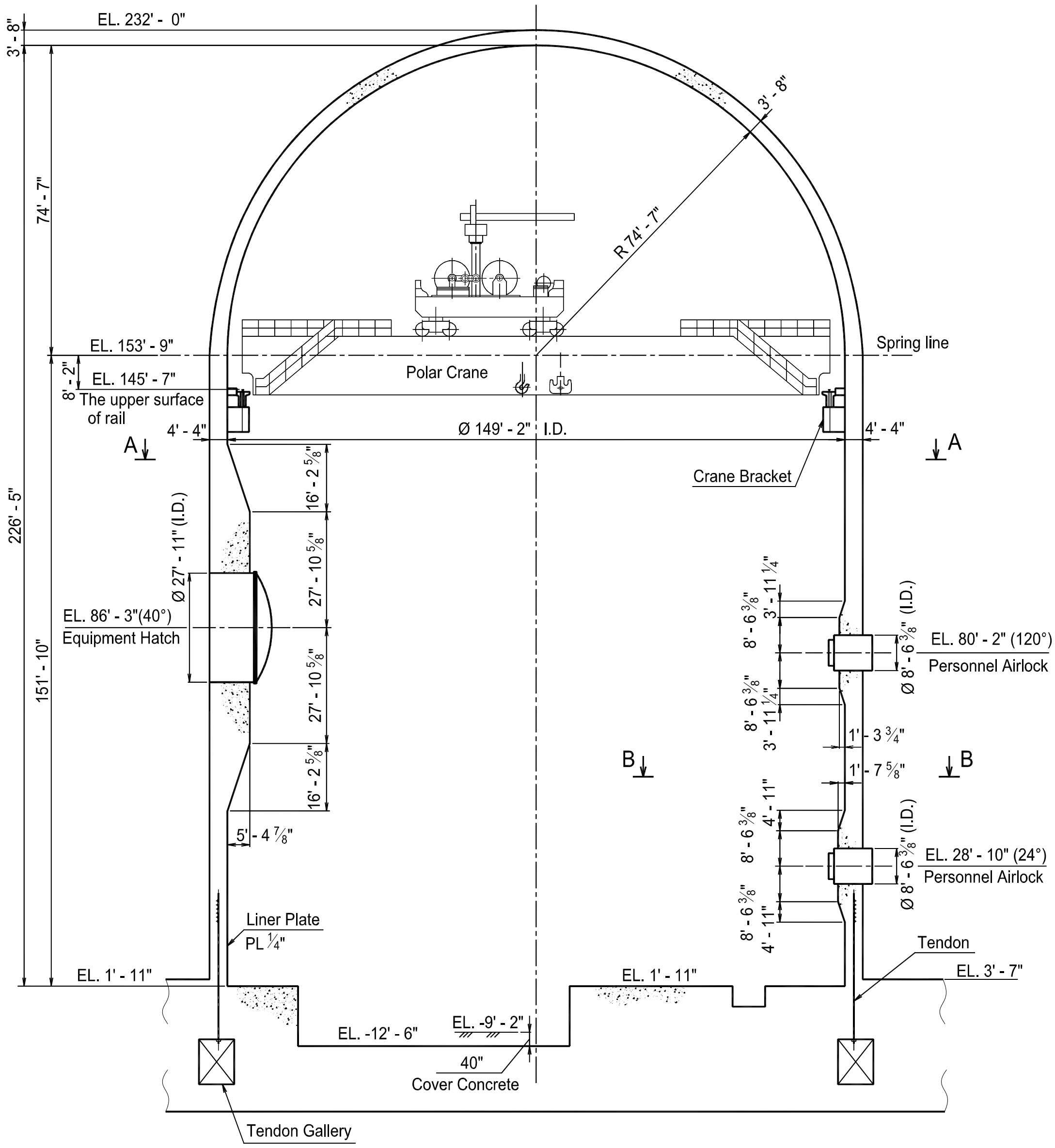


Figure 3.8.1-1 Configuration of PCCV (Sheet 2 of 3)

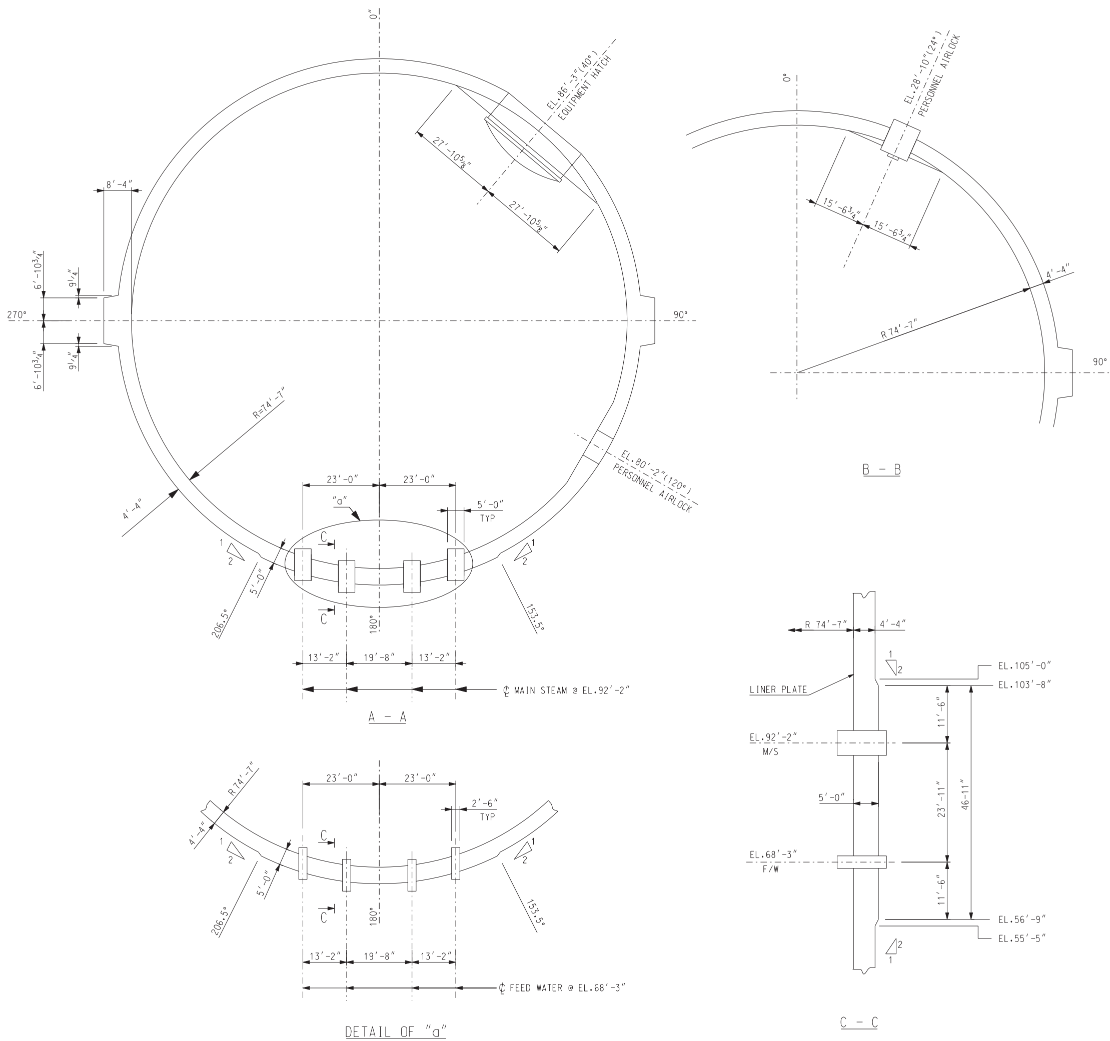


Figure 3.8.1-1 Configuration of PCCV (Sheet 3 of 3)

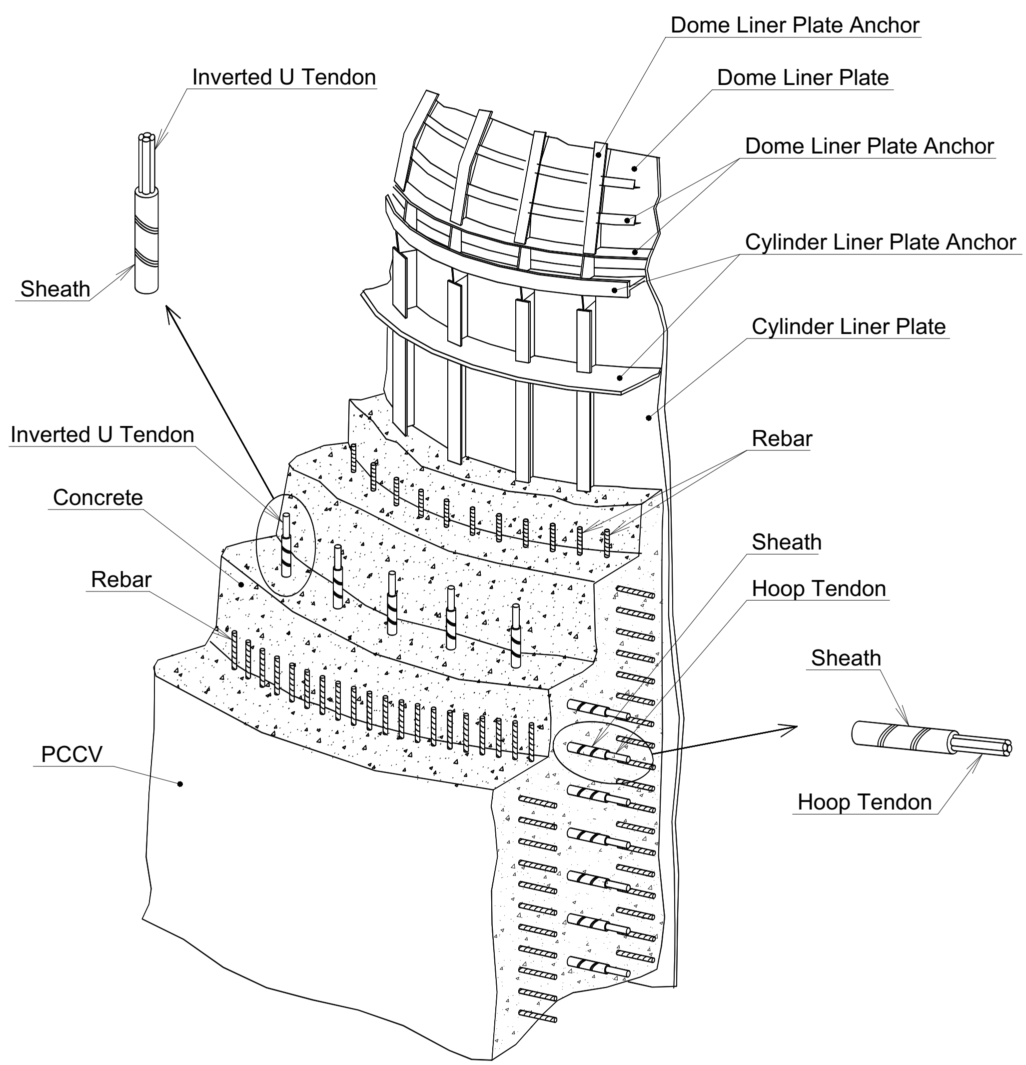


Figure 3.8.1-2 PCCV Schematic Reinforcing and Tendons

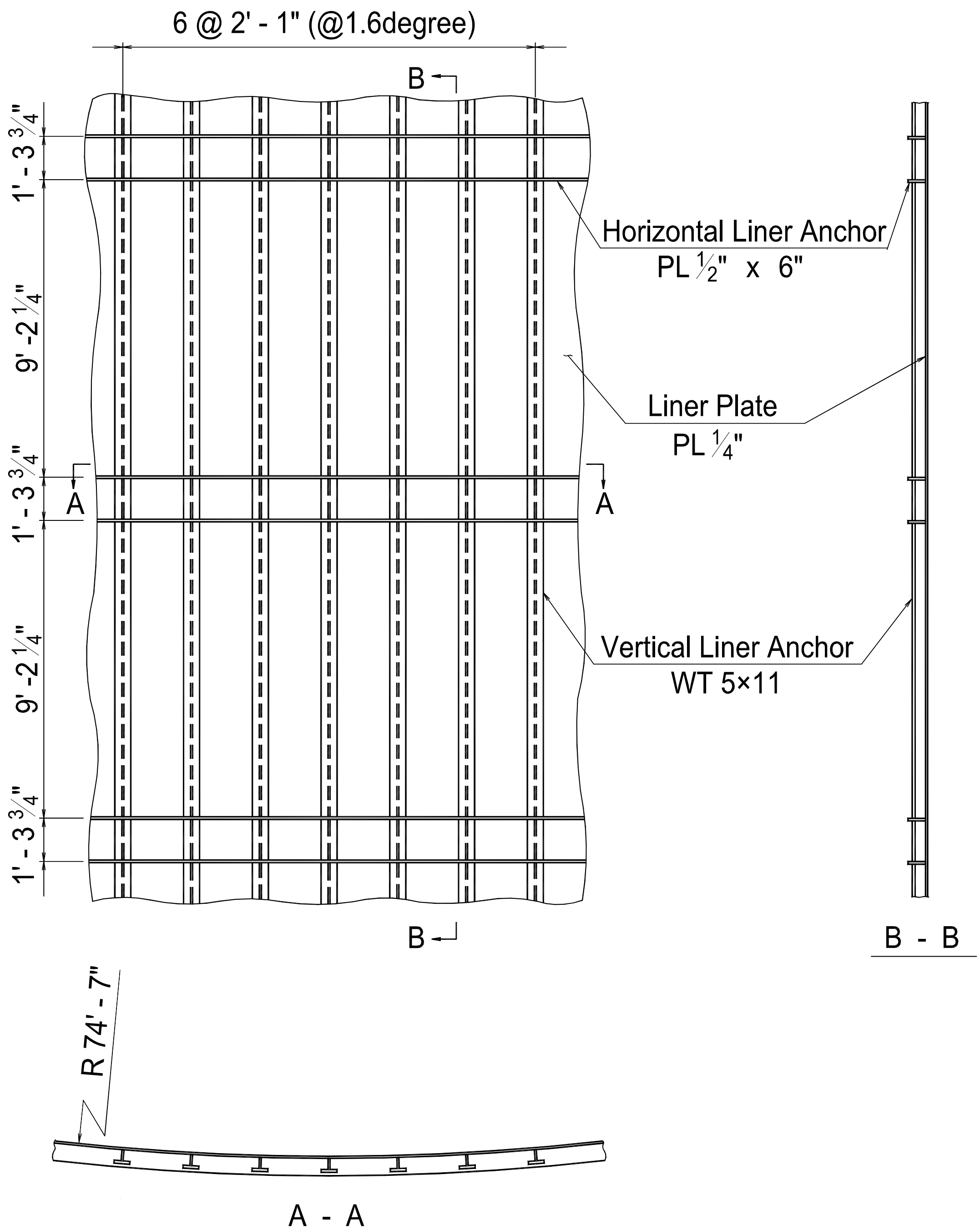


Figure 3.8.1-3 Cylinder Liner Anchorage System

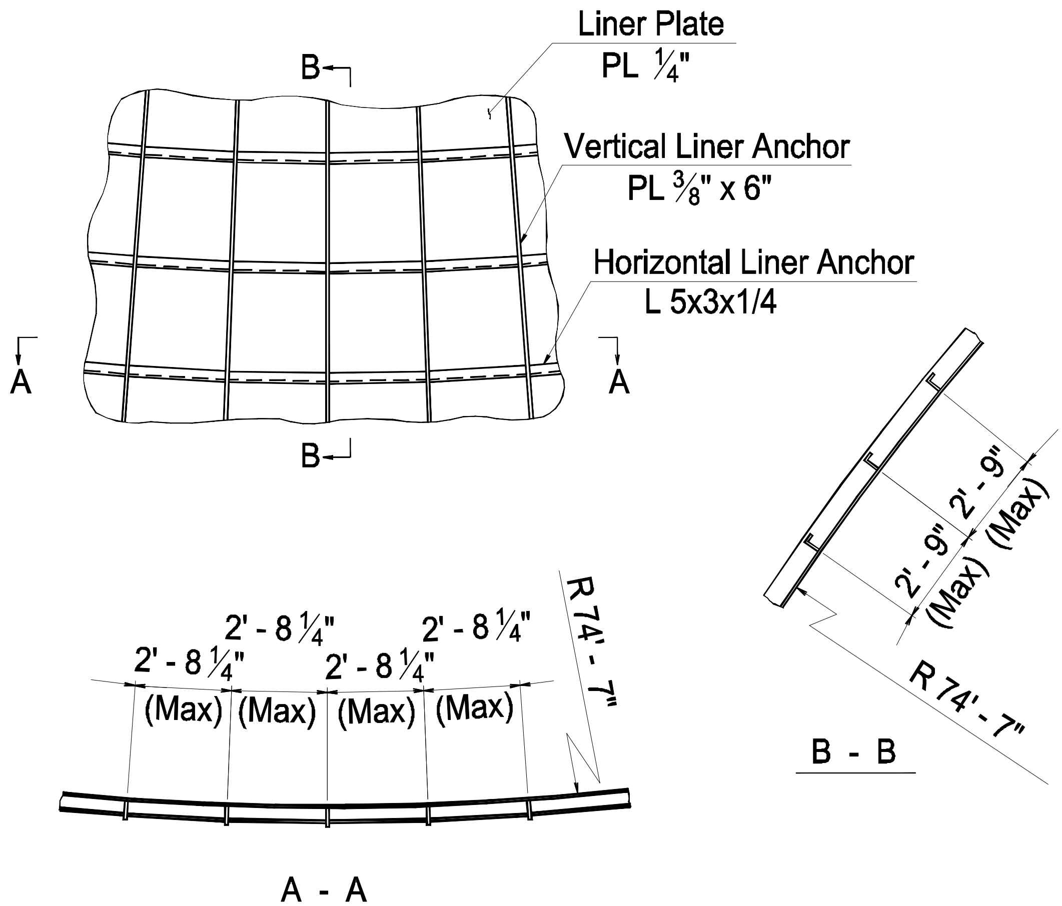


Figure 3.8.1-4 Liner Anchorage System for the Upper Portion of Dome

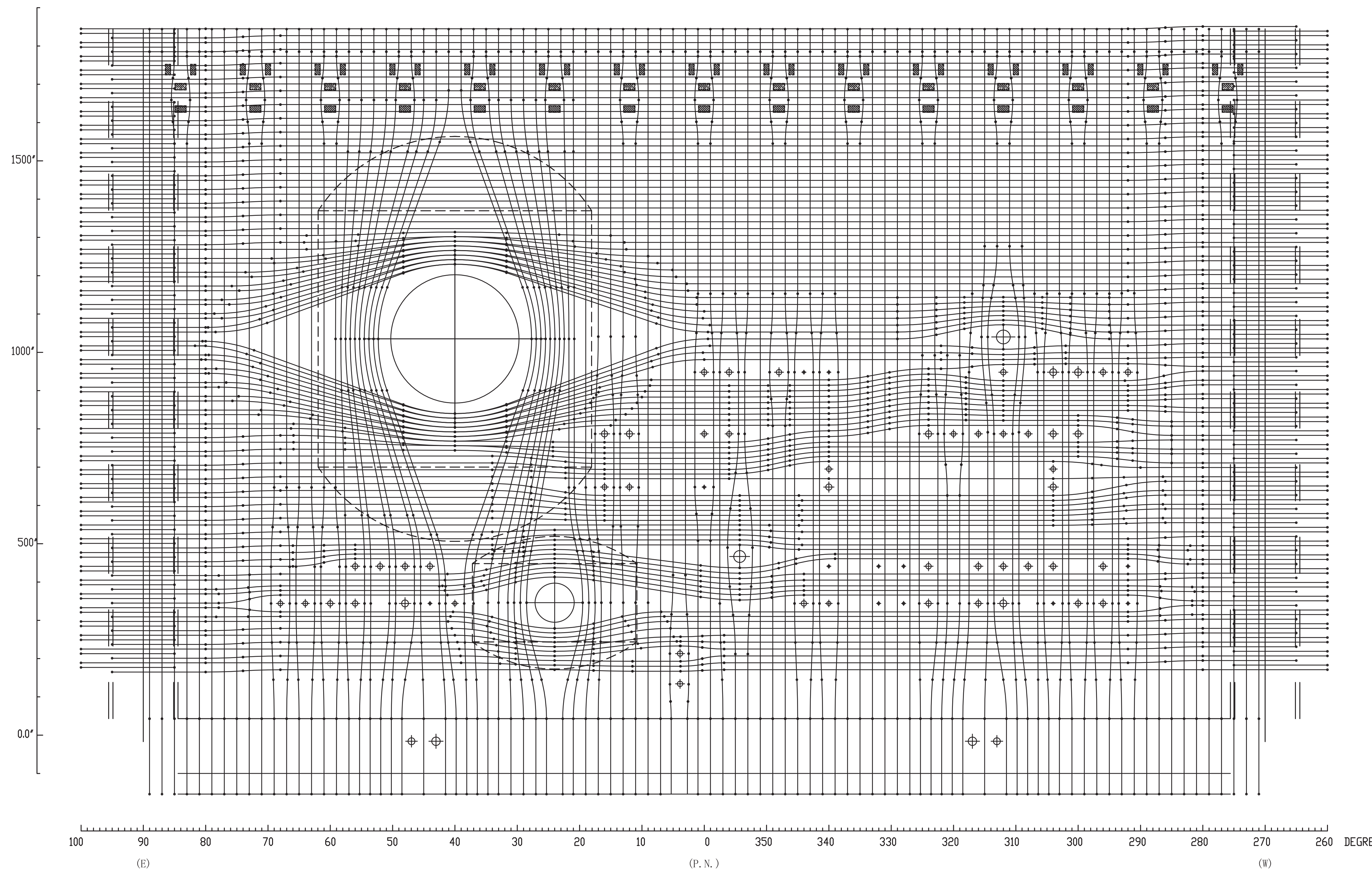


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 1 of 15)

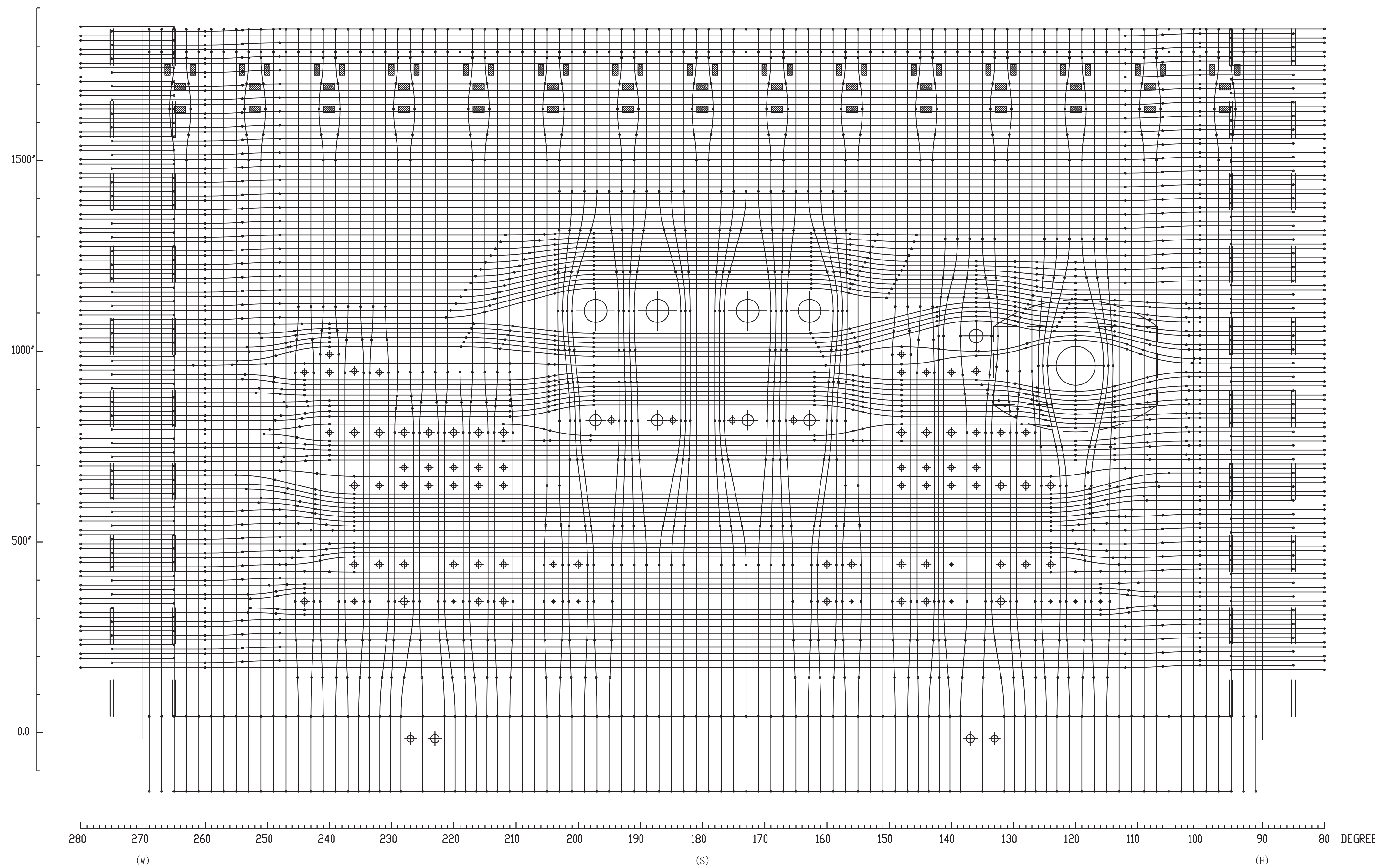


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 2 of 15)

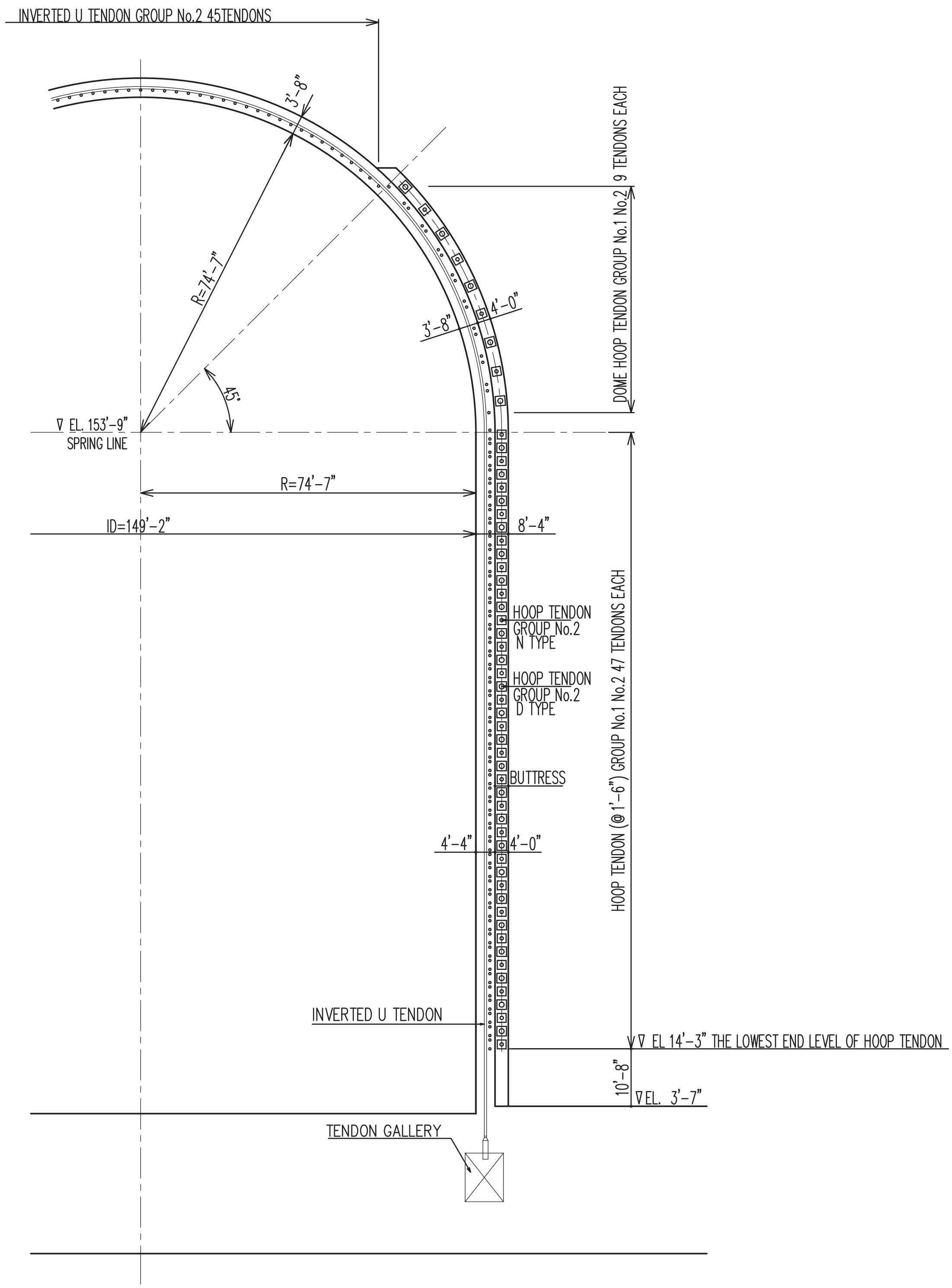


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing
(Sheet 3 of 15)

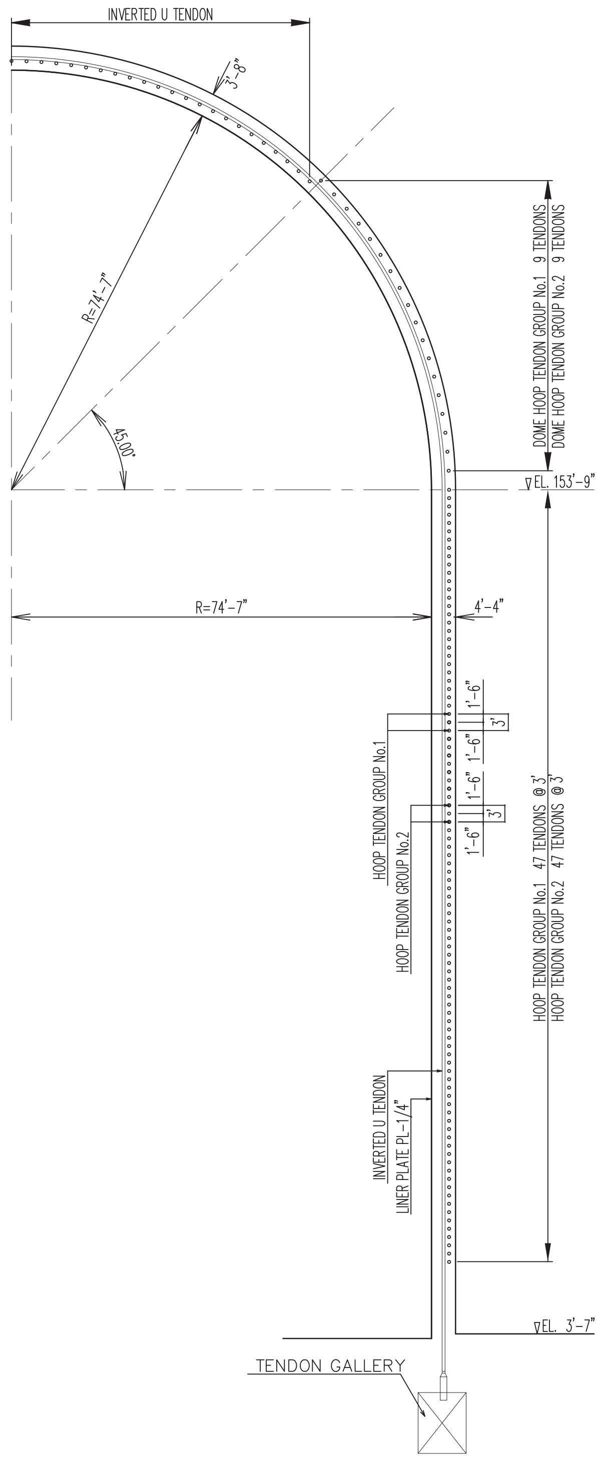
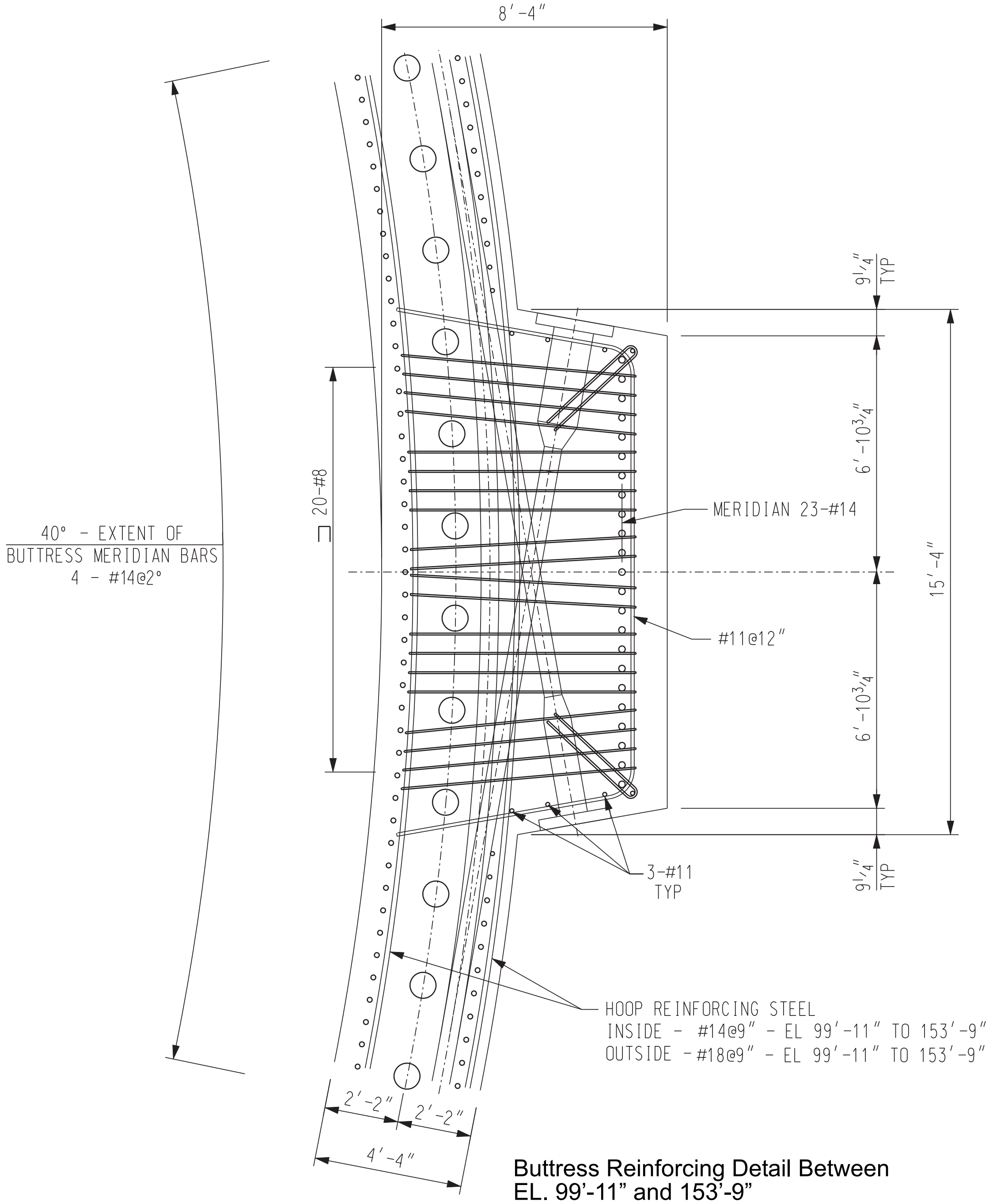


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 4 of 15)



Buttress Reinforcing Detail Between EL. 99'-11" and 153'-9"

Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 5 of 15)

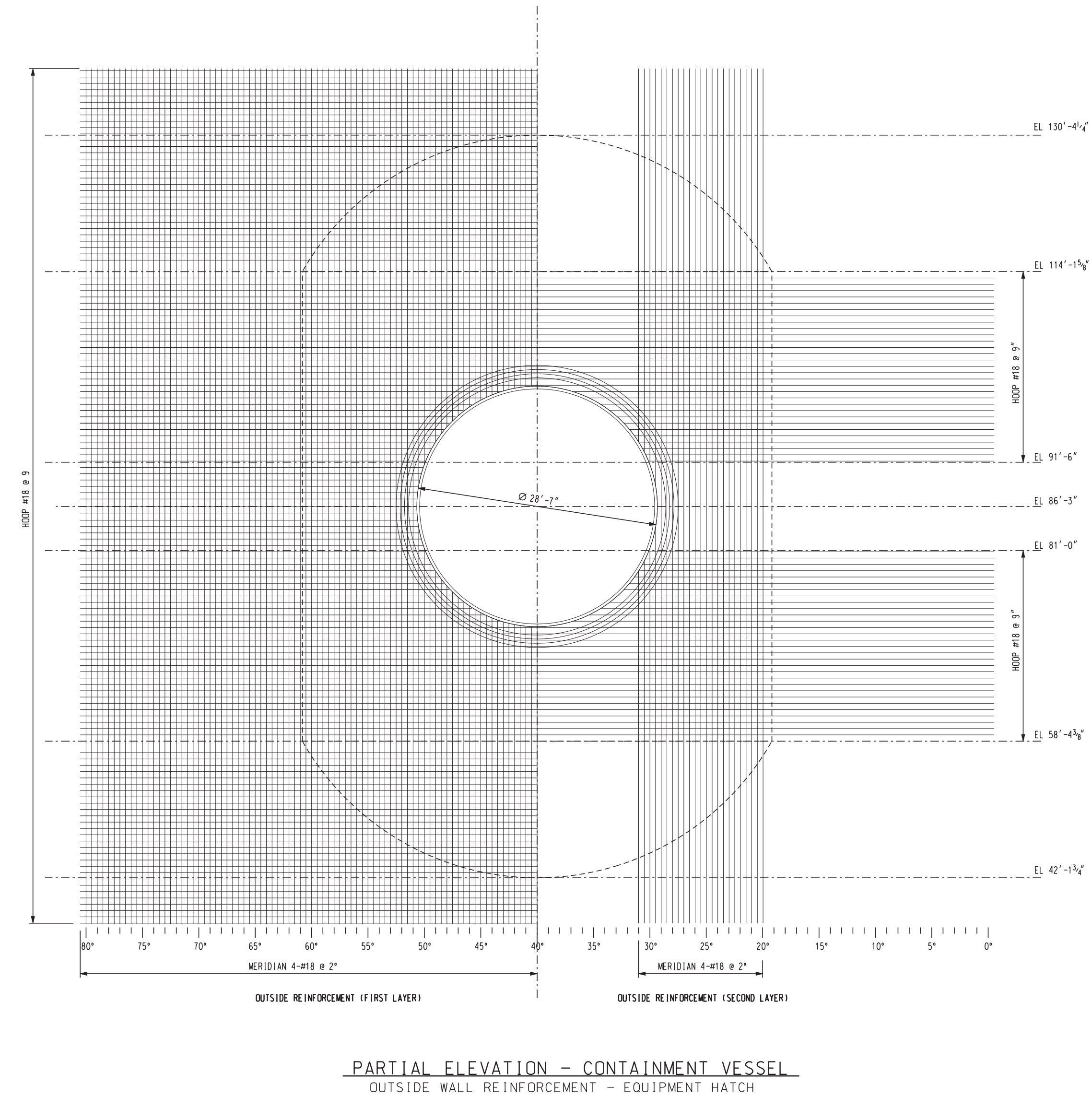


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 6 of 15)

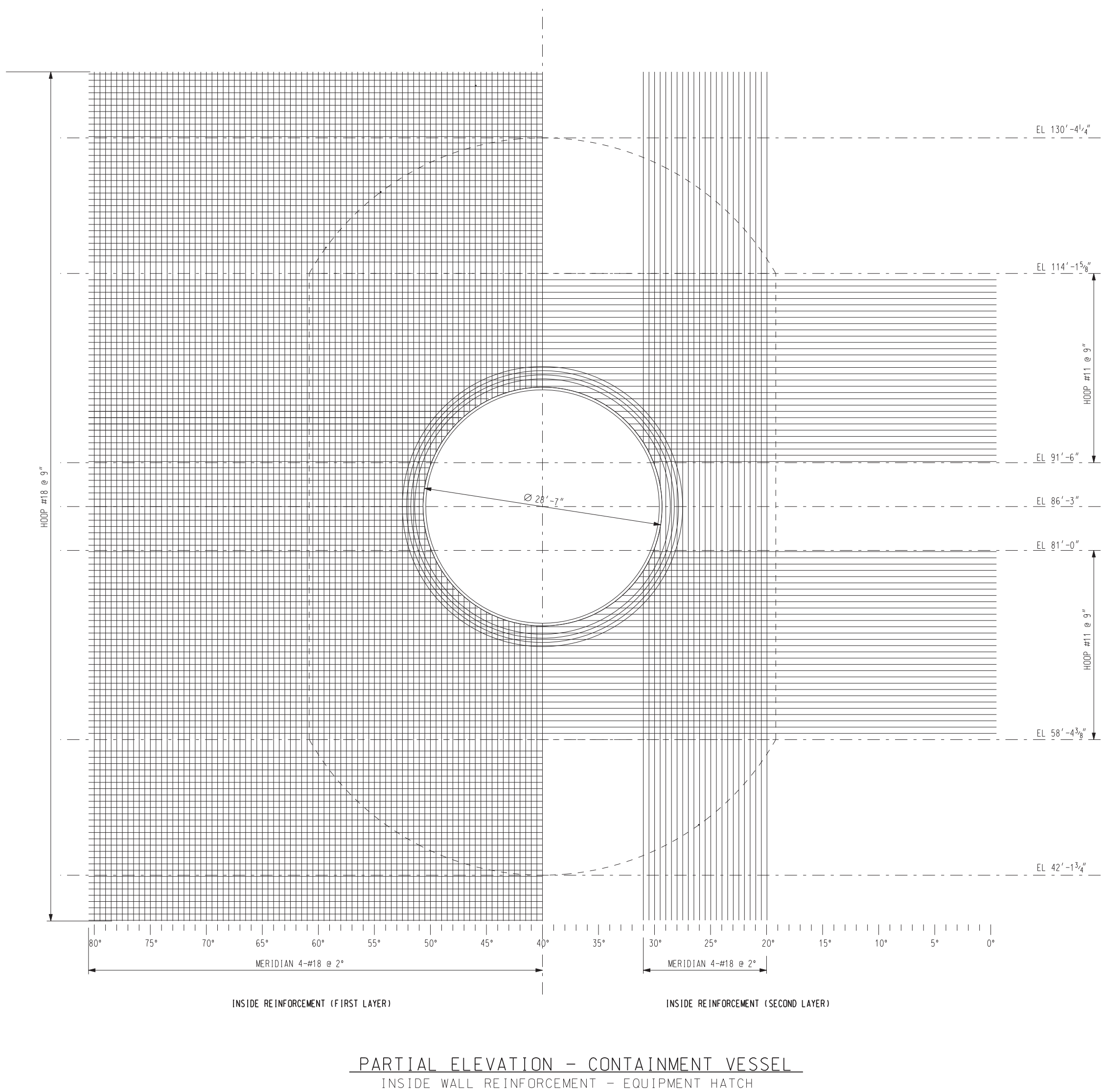
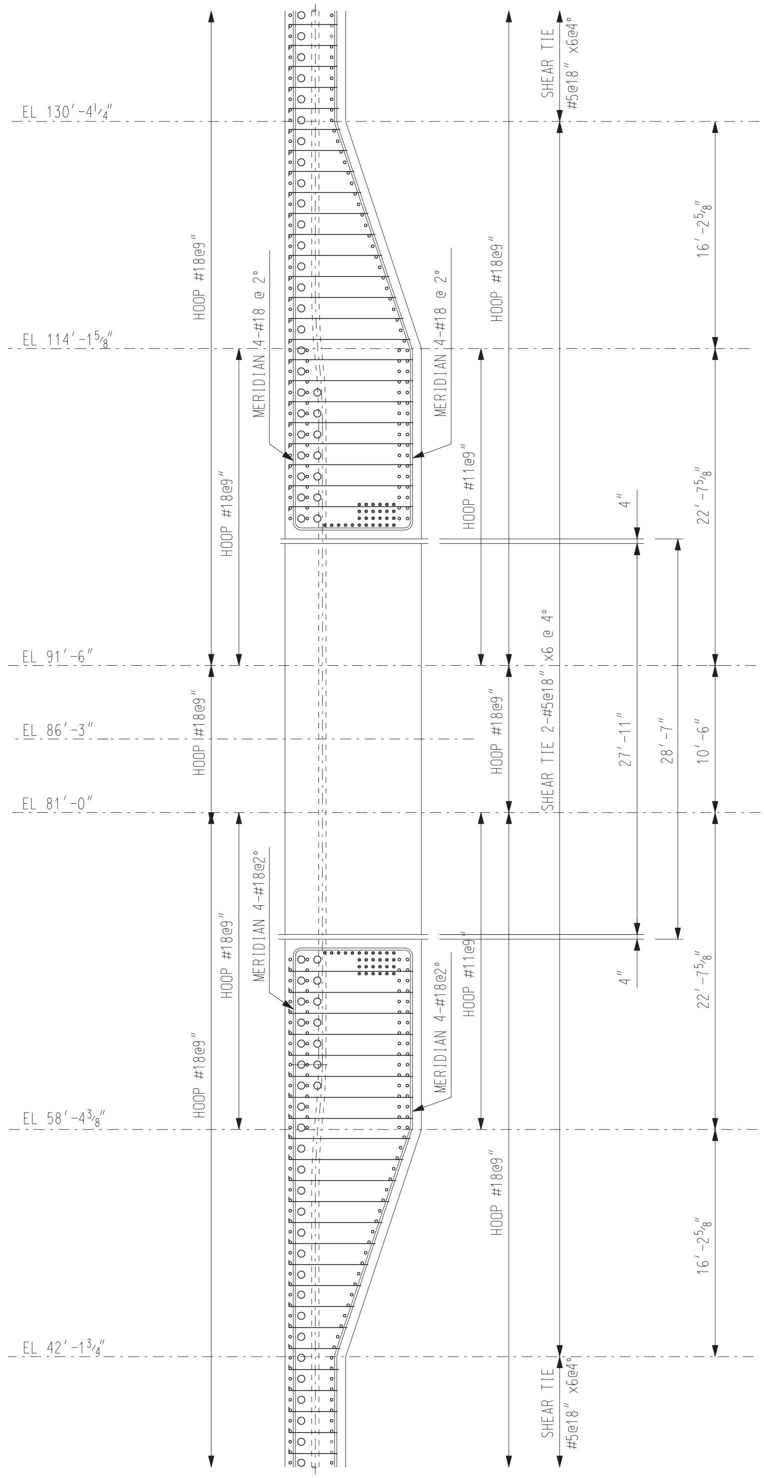


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 7 of 15)



E/H Section

Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 8 of 15)

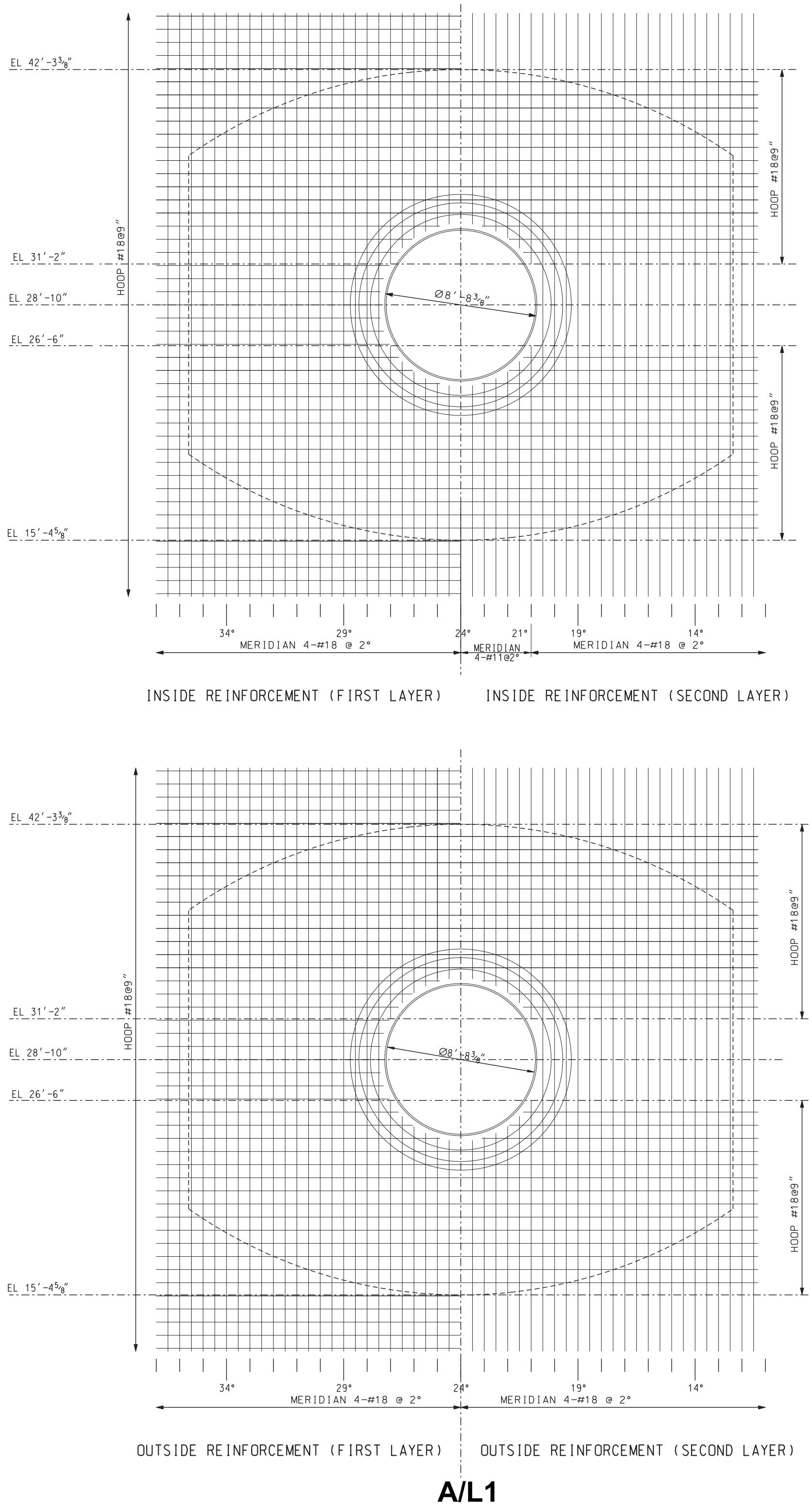
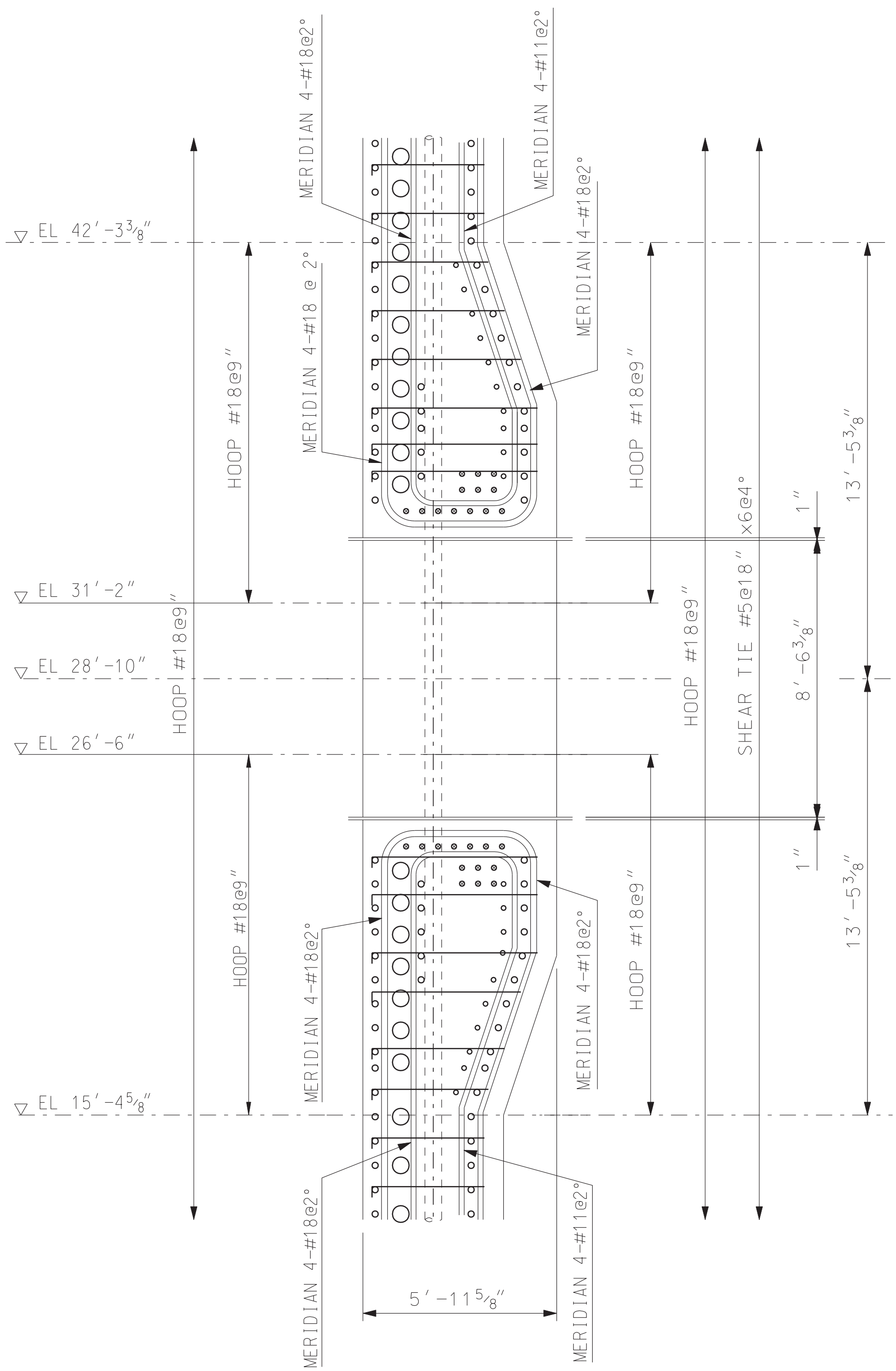
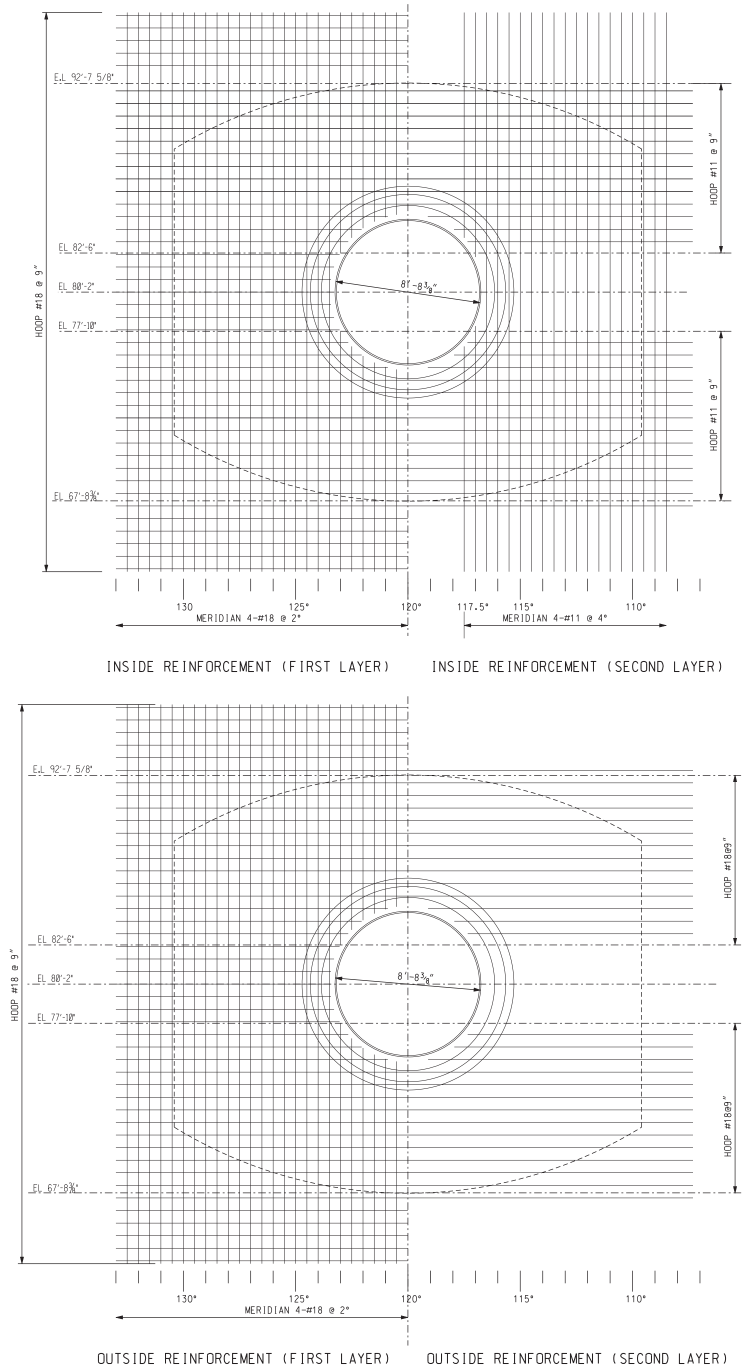


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 9 of 15)



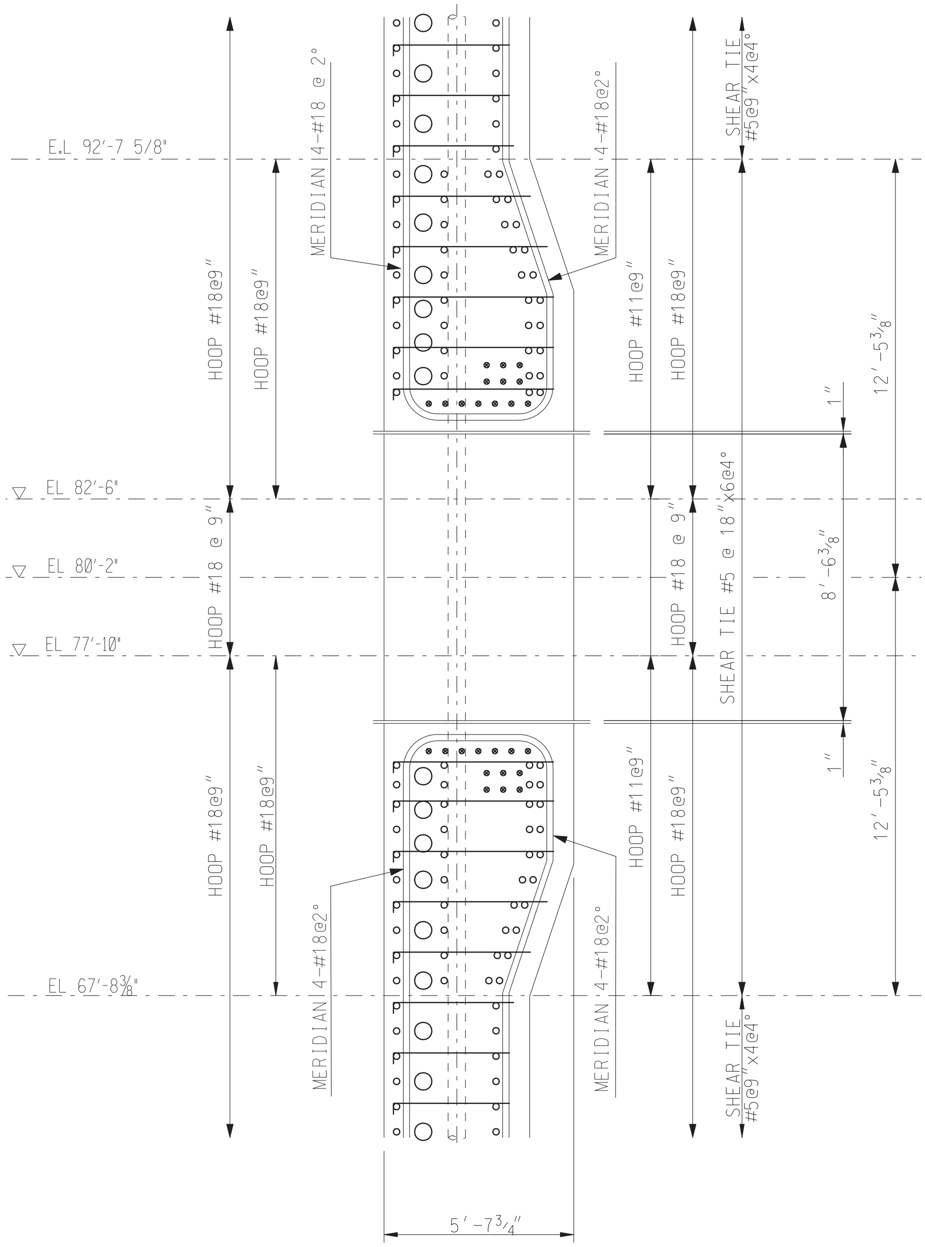
A/L1 Key Section

Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 10 of 15)



A/L2

Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 11 of 15)



A/L2 Key Section

Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 12 of 15)

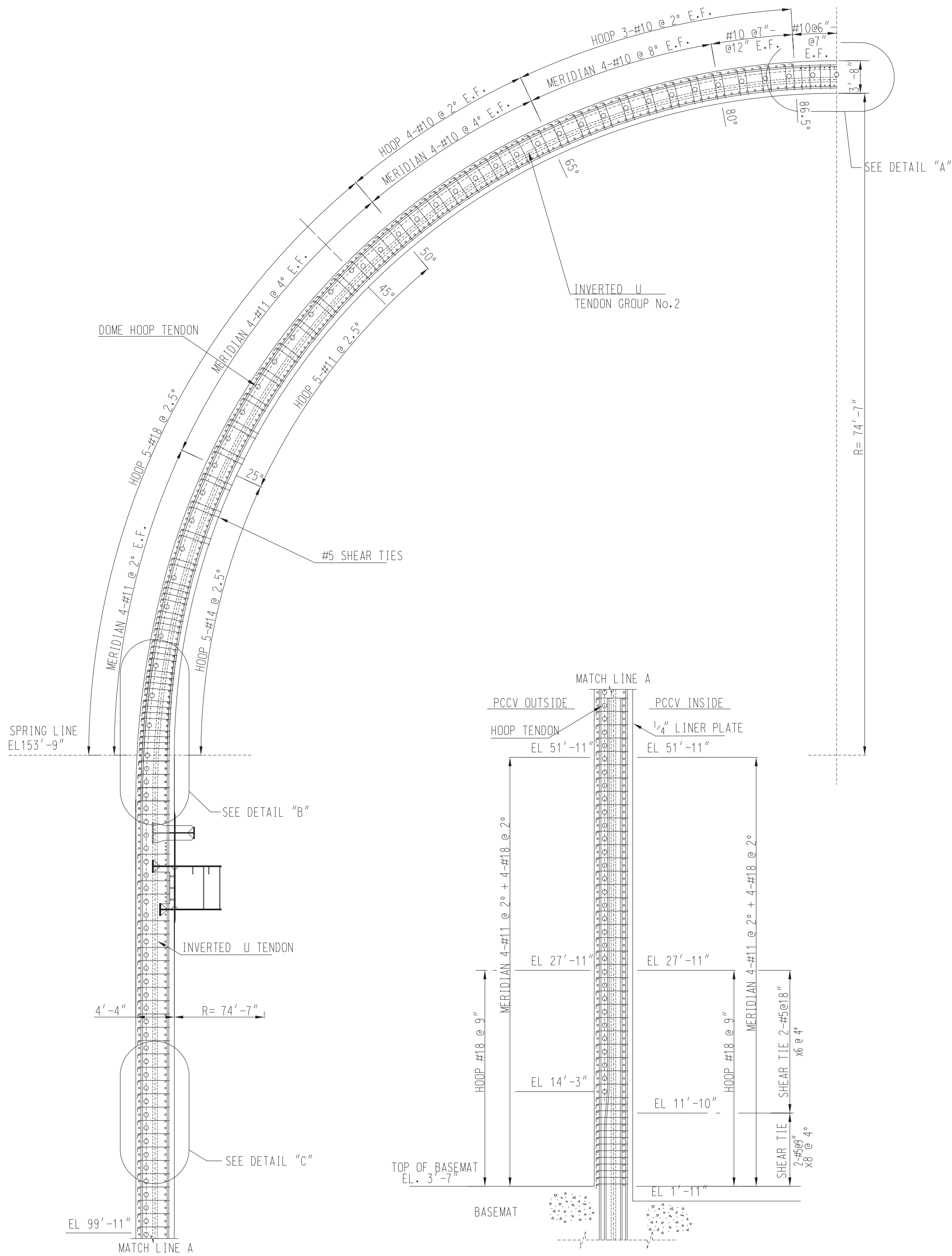


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 13 of 15)

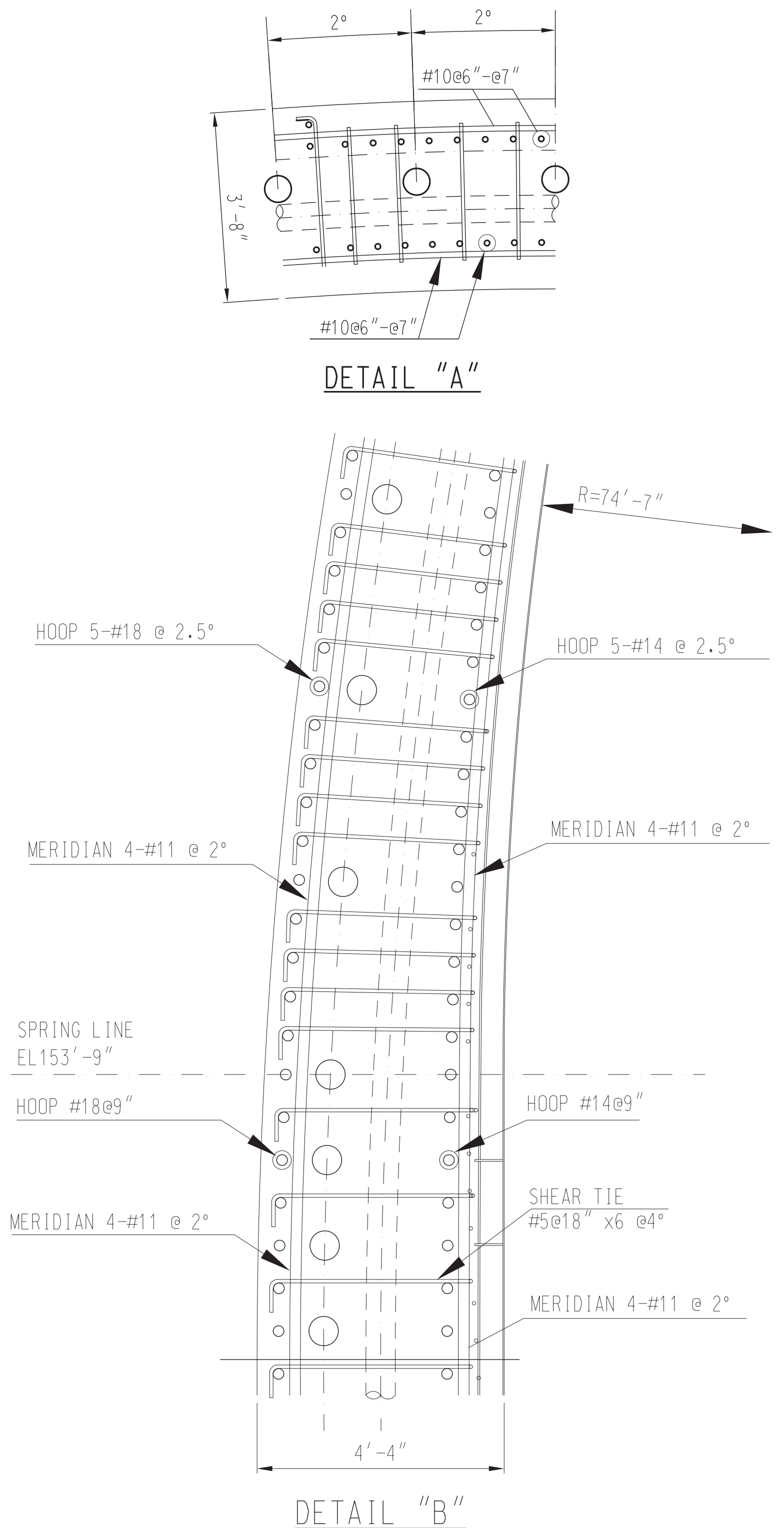


Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 14 of 15)

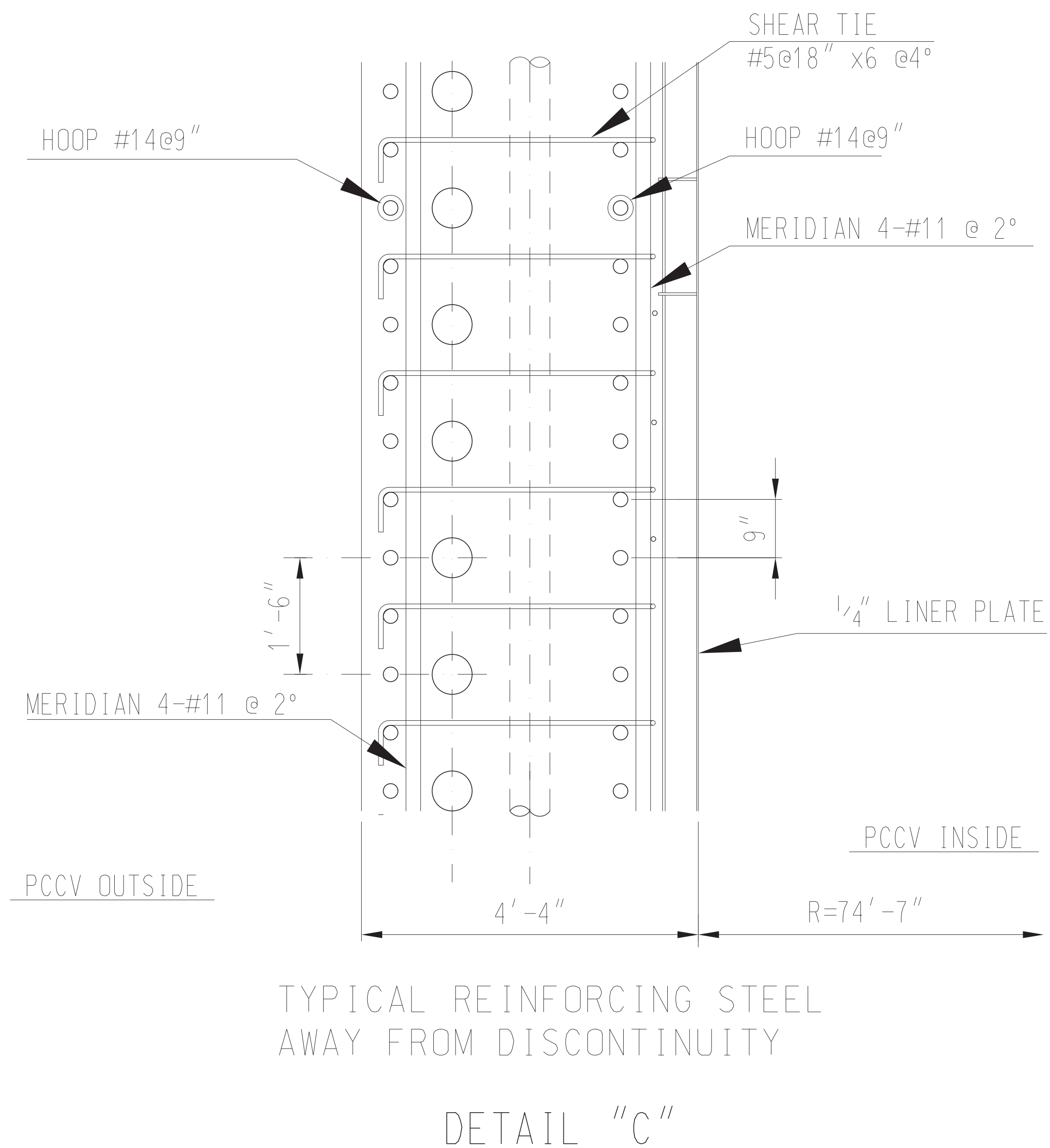


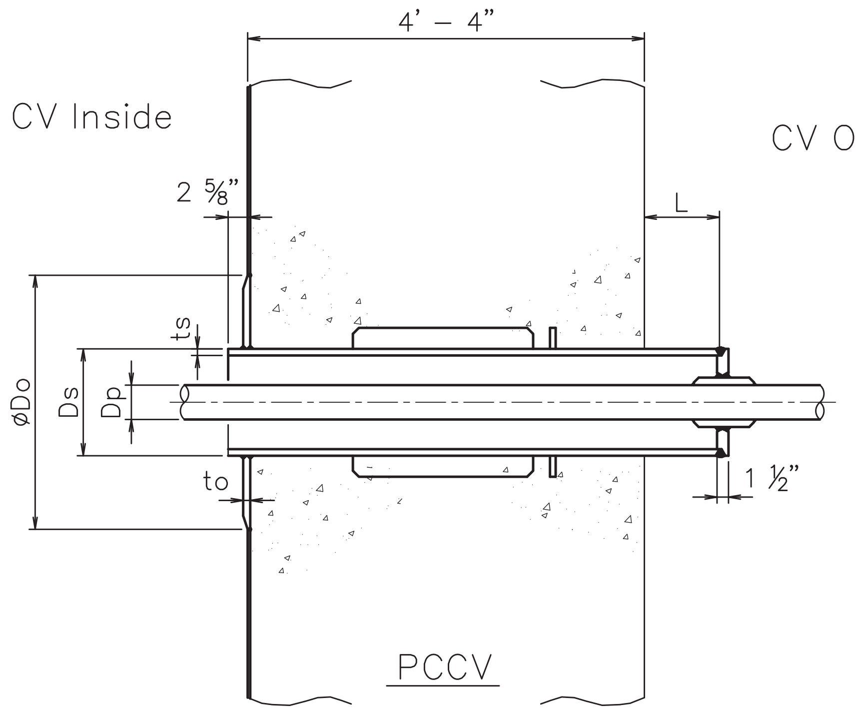
Figure 3.8.1-5 PCCV General Arrangement, Tendons and Reinforcing (Sheet 15 of 15)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.1-6 Equipment Hatch

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.1-7 Personnel Airlock

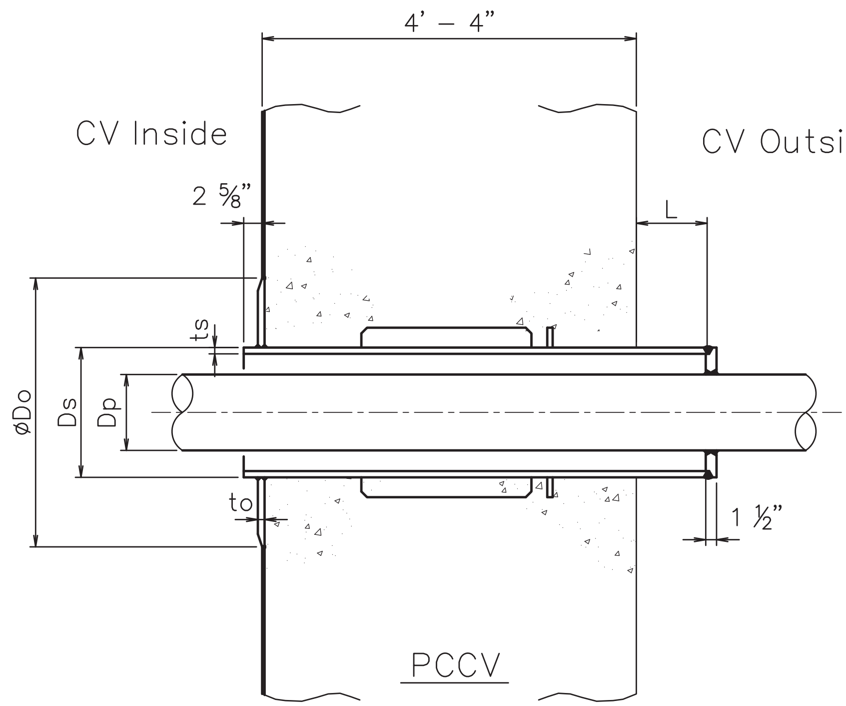


TYPE-1

Ds	ts	Dp	L	Do	to	SLEEVE NO.
6"	1/2"	3/4"	7"	11 5/8"	1/2"	P220,P222,P231,P270,P416,P417
		1"	7"			P236,P247,P265,P266
		2"	7"			P207,P230,P245,P253,P284
10"	3/8"	3"	7"	1'-5 1/4"	1/2"	P205,P248,P260,P283
14"	5/8"	4"	7"	1'-8 3/4"	1/2"	P162,P210,P227,P233,P235,P258
			7"			P274

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 1 of 17)

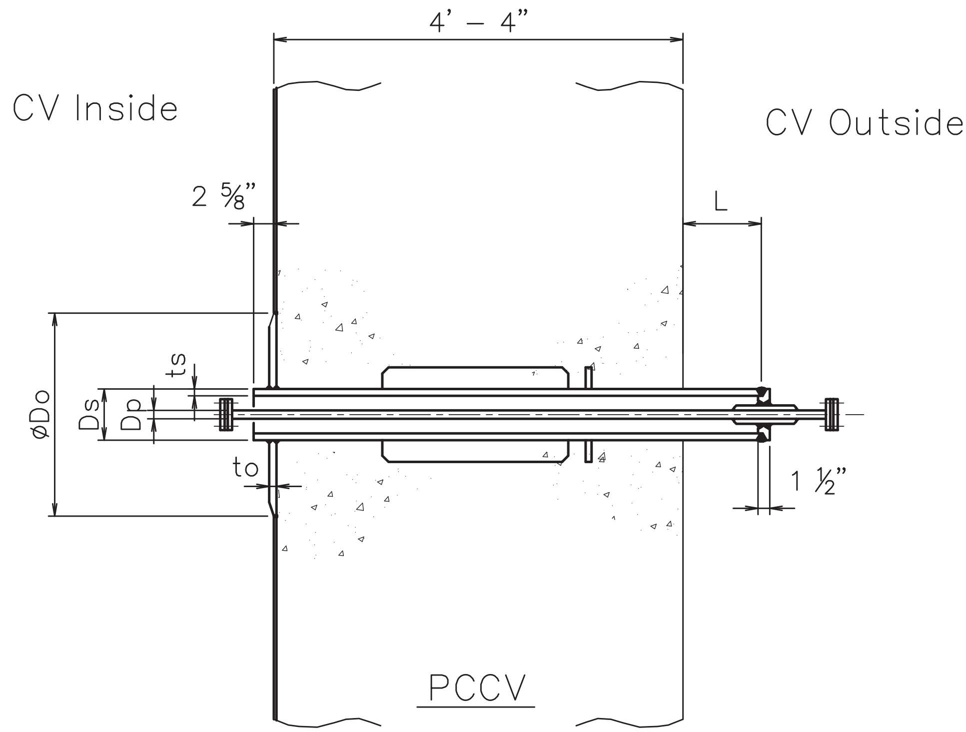


TYPE-2

Ds	ts	Dp	L	Do	to	SLEEVE NO.
14"	1"	6"	7"	1'-8 3/4"	1/2"	P161,P238
		8"	7"			P214,P224,P232,P234,P249,P250
						P251,P252,P261,P271,P401,P410
			23"			P212,P225,P259,P272
18"	1"	10"	7"	2'-1 3/8"	1/2"	P408,P409
			23"			P209,P226,P257,P273

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 2 of 17)

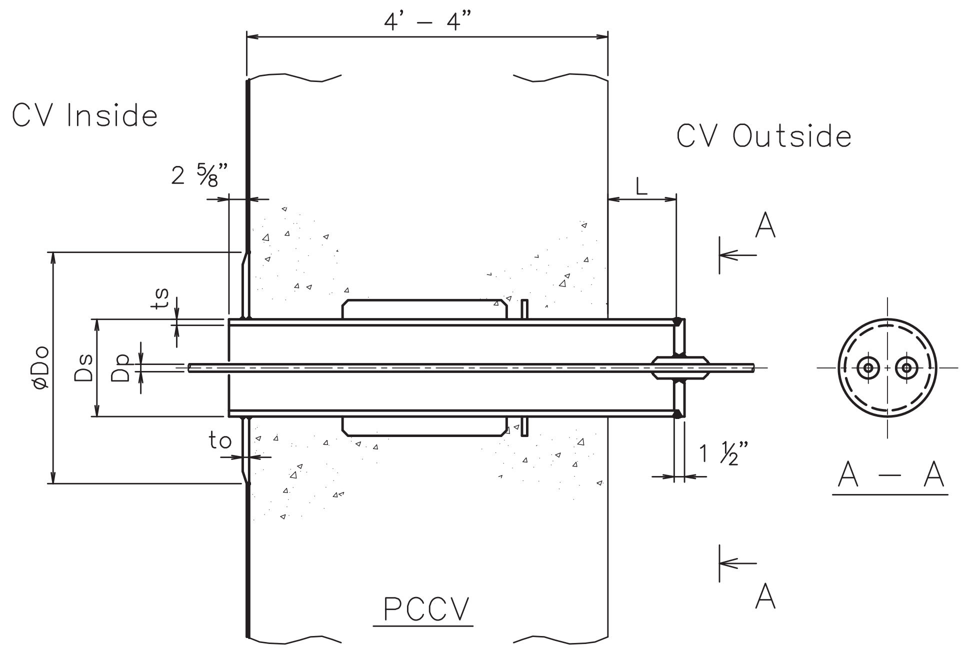


TYPE-3

Ds	ts	Dp	L	Do	to	SLEEVE NO.
6"	1/2"	3/4"	7"	11 5/8"	1/2"	P223

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 3 of 17)

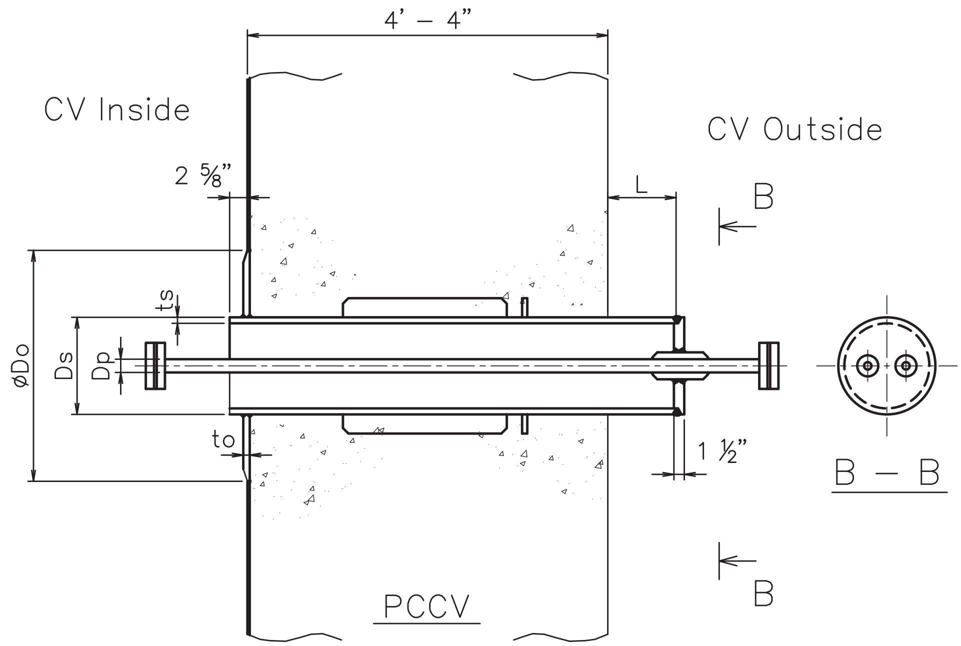


TYPE-4

Ds	ts	Dp	L	Do	to	SLEEVE NO.
14"	5/8"	3/4"	7"	1'-8 3/4"	1/2"	P262, P405
			15"			P237,P239,P276
			23"			P267,P269

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 4 of 17)

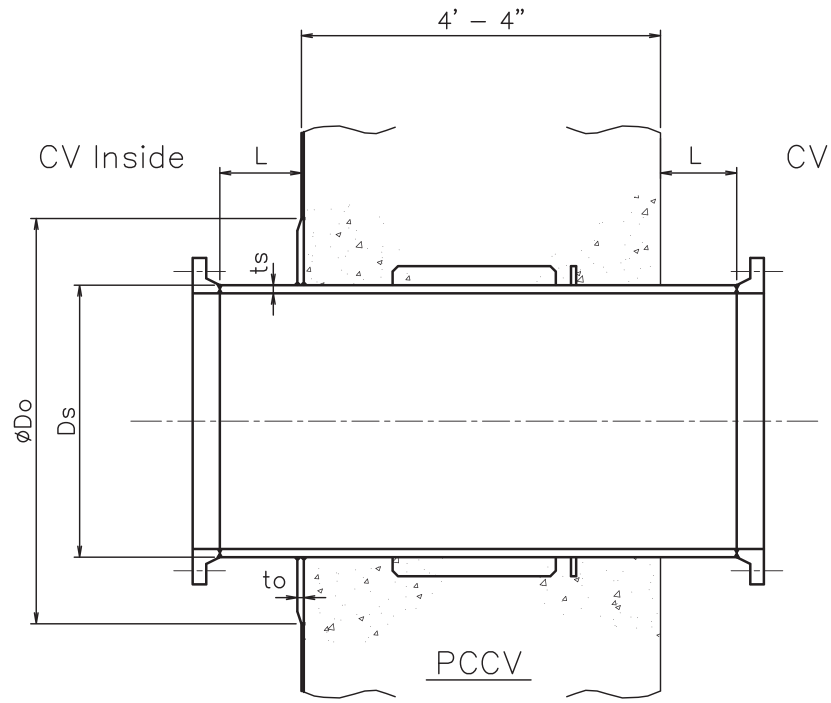


TYPE-5

Ds	ts	Dp	L	Do	to	SLEEVE NO.
14"	5/8"	1 1/2"	7"	1'-8 3/4"	1/2"	P418

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 5 of 17)

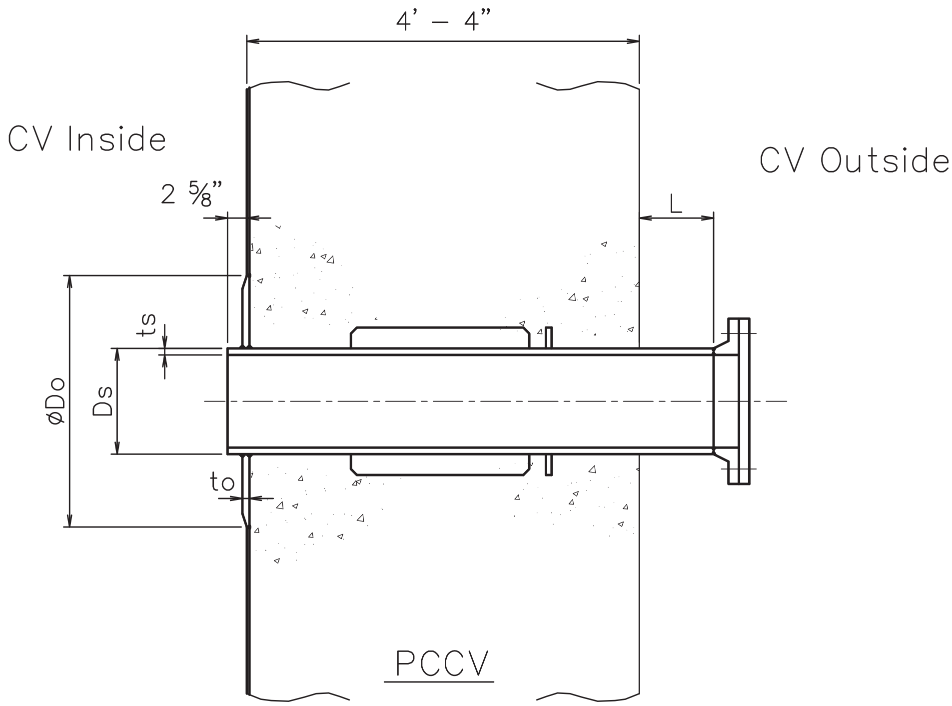


TYPE-8

Ds	ts	L	Do	to	SLEEVE NO.
36"	5/8"	15"	3'-7 3/8"	1/2"	P451,P452

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 6 of 17)

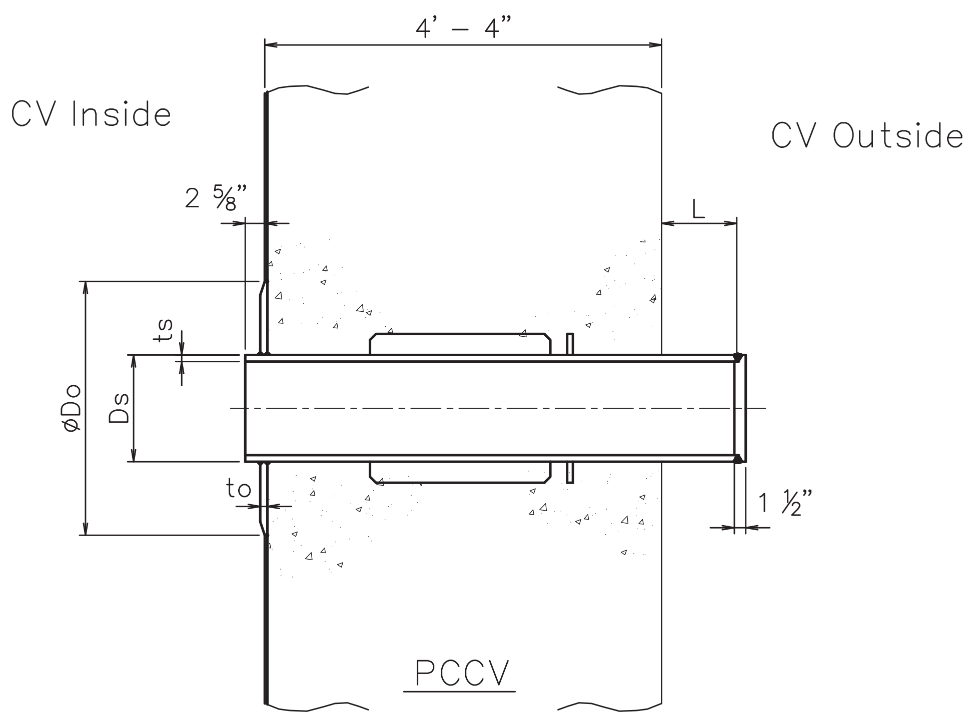


TYPE-9

Ds	ts	L	Do	to	SLEEVE NO.
6"	1/2"	7"	11 5/8"	1/2"	P301
12"	5/8"	7"	1'-7 1/4"	1/2"	P216,P218
14"	1"	7"	1'-8 3/4"	1/2"	P419,P420

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 7 of 17)

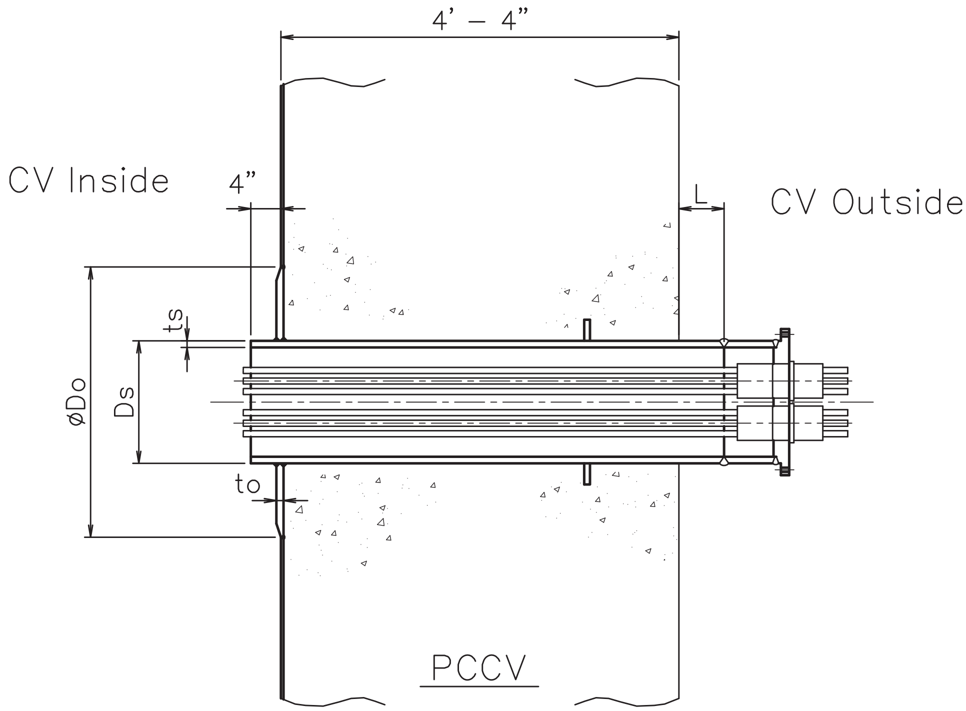


TYPE-10

Ds	ts	L	Do	to	SLEEVE NO.
14"	1"	24"	1'-8 3/4"	1/2"	P208,P213,P215,P246,P254,P268
					P275,P285,P406,P407

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 8 of 17)

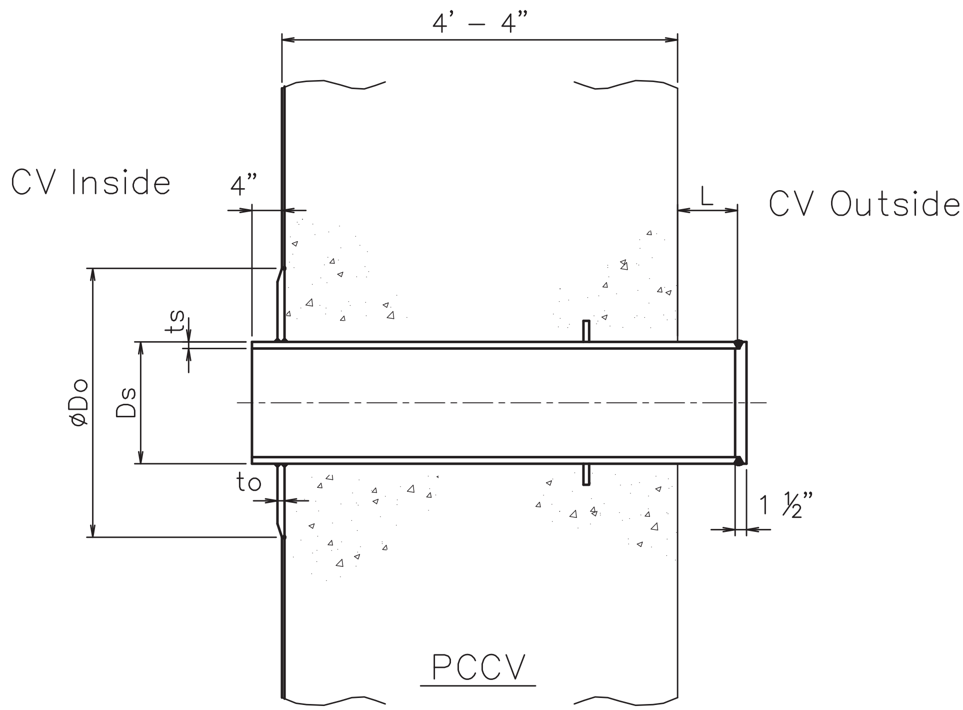


TYPE-11

Ds	ts	L	Do	to	SLEEVE NO.
12"	5/8"	6"	1'-7 1/4"	1/2"	E606,E607,E608,E609,E610,E612,E613,E615
					E616,E620,E621,E623,E624,E626,E627,E629
					E630,E632,E633,E635,E654,E655,E656,E657
					E663,E664,E665,E667,E703,E704,E710,E711
		8"			E701,E702,E709,E712
16"	3/4"	6"	1'-11 3/8"	1/2"	E602,E604,E611,E614,E617,E622,E625,E628
					E631,E636,E651,E661,E666,E668
		8"			E634,E637,E650,E653

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 9 of 17)



TYPE-12

Ds	ts	L	Do	to	SLEEVE NO.
12"	5/8"	6"	1'-7 1/4"	1/2"	E603,E605,E639,E652
		8"			E662
16"	3/4"	6"	1'-11 3/8"	1/2"	E601,E638,E658

All dimensions shown above are nominal dimensions.

Figure 3.8.1-8 Containment Penetrations (Sheet 10 of 17)

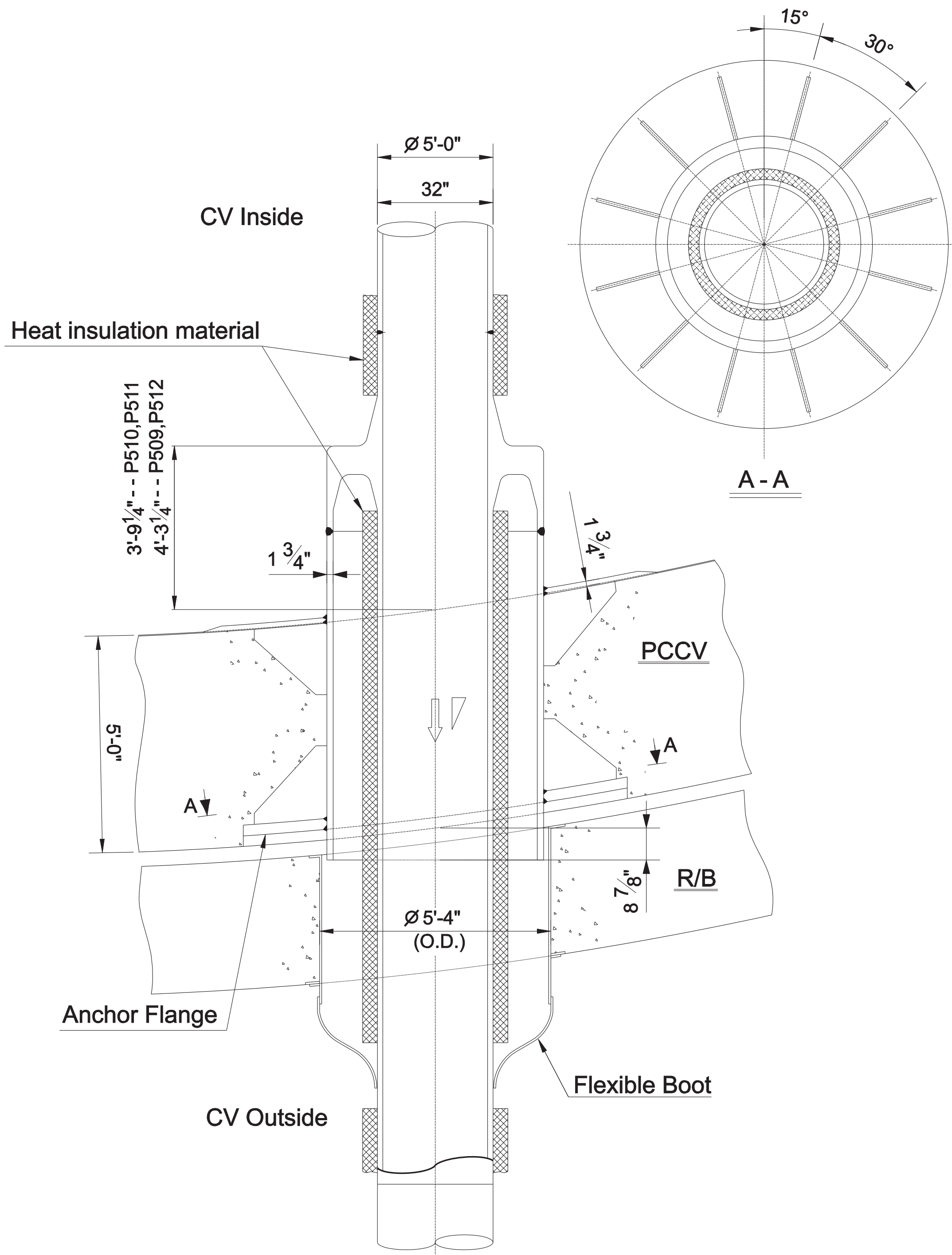
Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.1-8 Containment Penetrations (Sheet 11 of 17)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Typical Electrical Penetration Detail

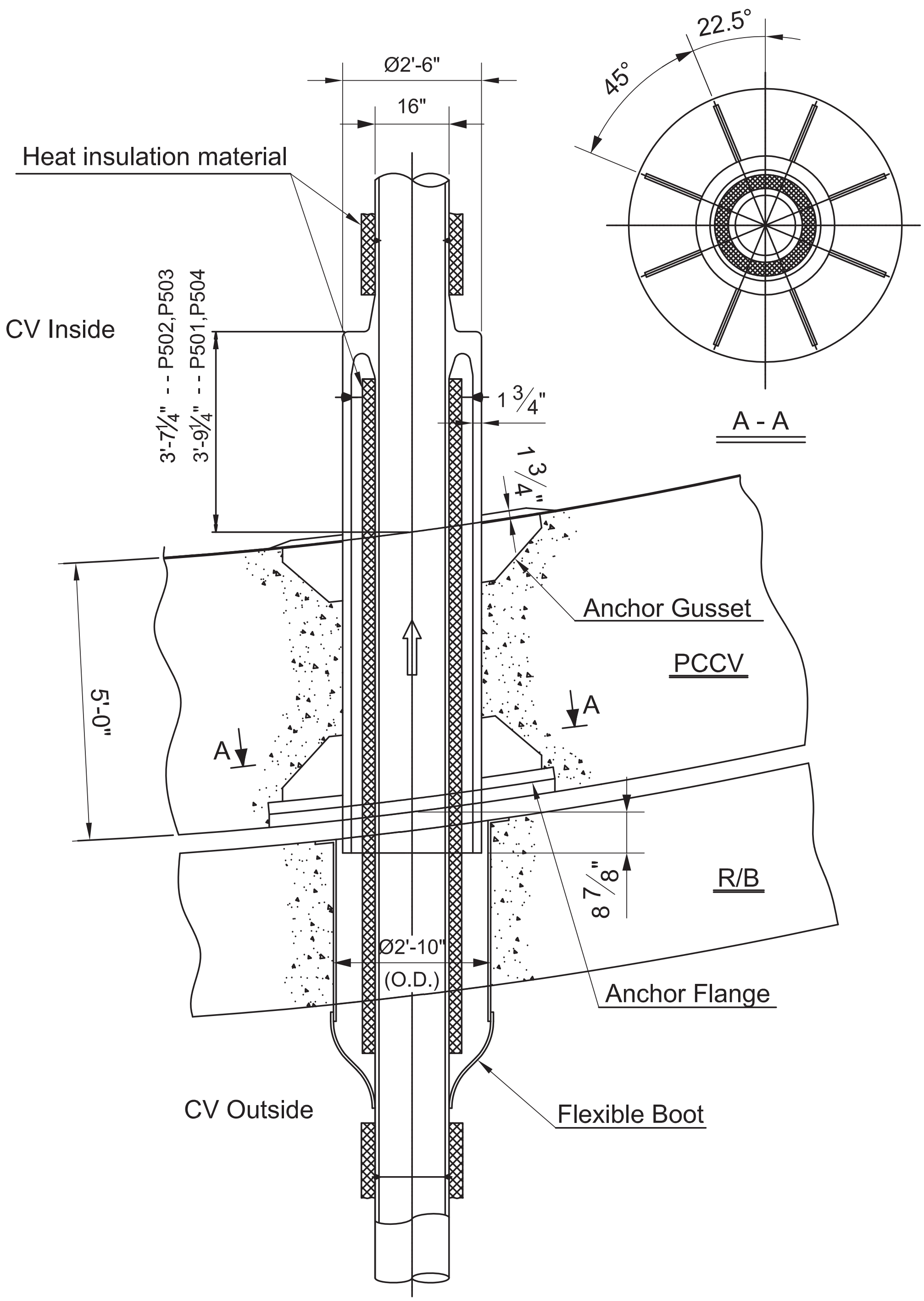
Figure 3.8.1-8 Containment Penetrations (Sheet 12 of 17)



P509 ~ P512
(Main Steam)

All dimensions shown above are nominal dimensions.

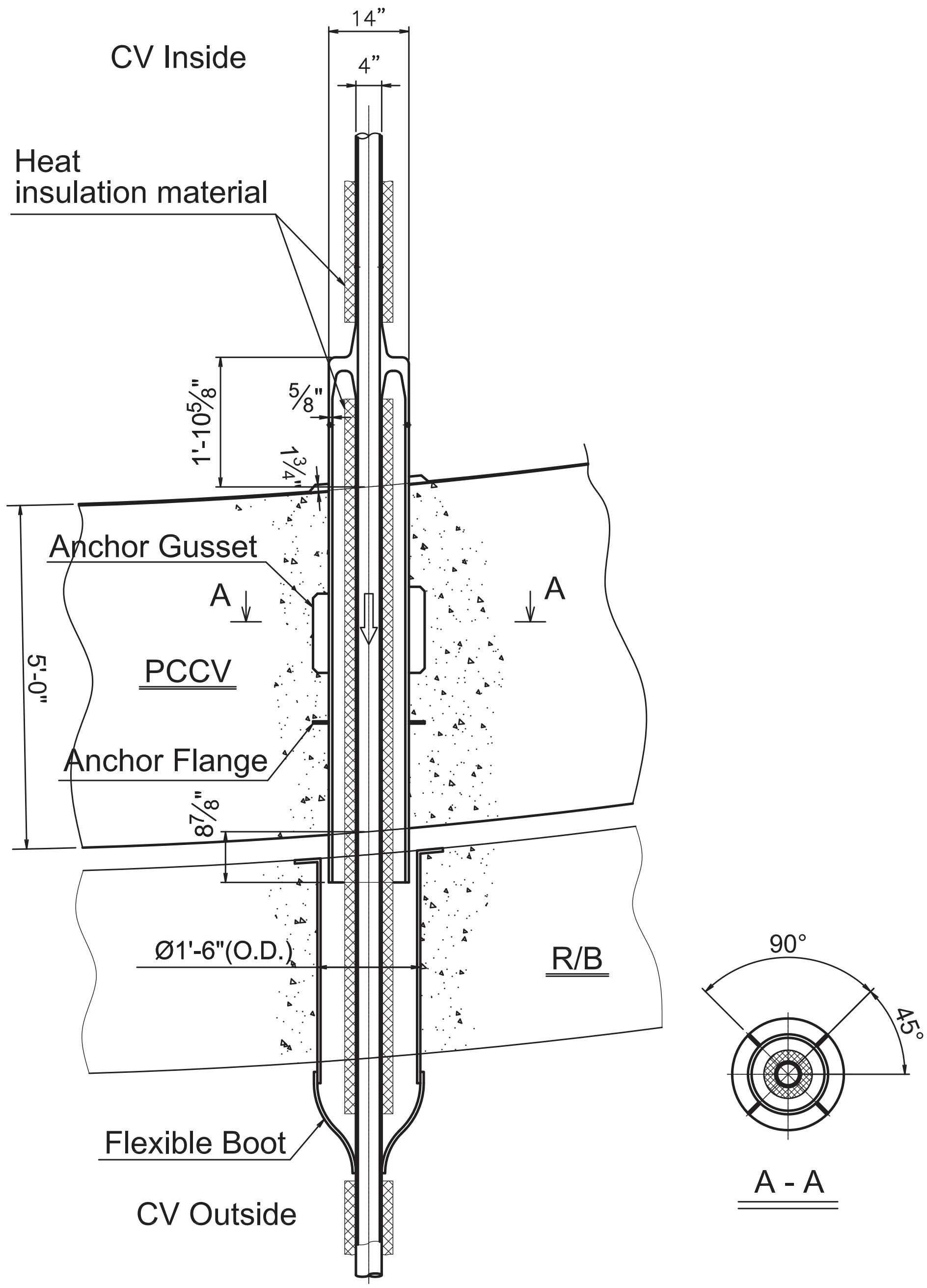
Figure 3.8.1-8 Containment Penetrations (Sheet 13 of 17)



P501 – P504
(Feedwater)

All dimensions shown above are nominal dimensions

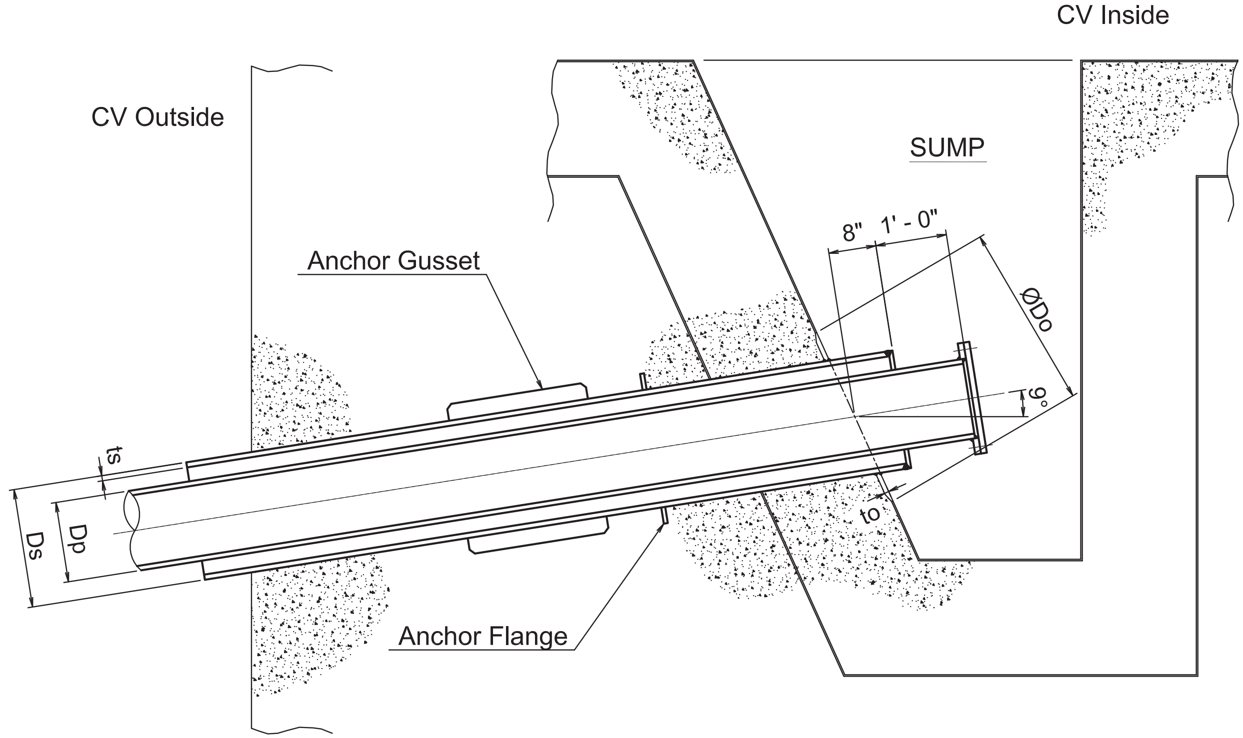
Figure 3.8.1-8 Containment Penetrations (Sheet 14 of 17)



P505~P508
(SG Blowdown)

All dimensions shown above are nominal dimensions

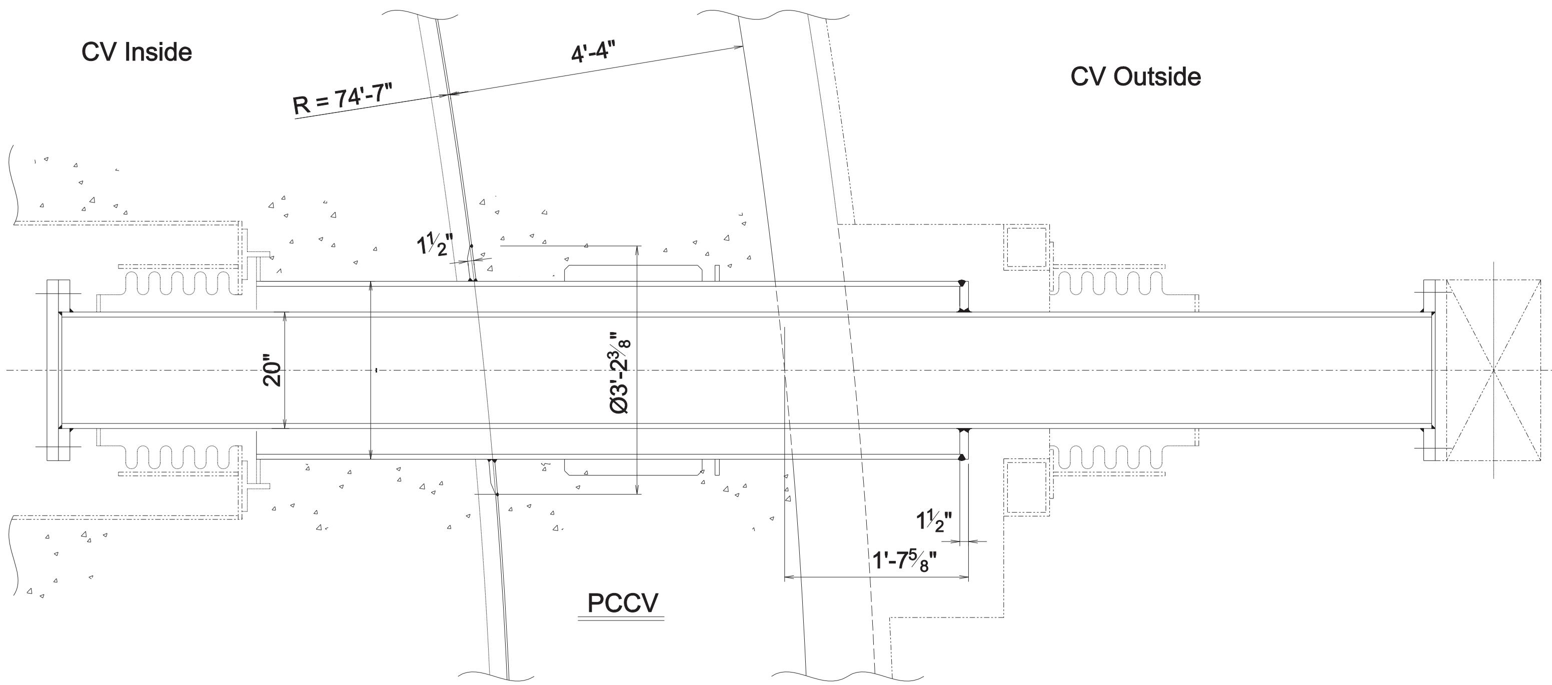
Figure 3.8.1-8 Containment Penetrations (Sheet 15 of 17)



P151~P158
(RWSP Sump)

Ds	ts	Dp	Do	to	SLEEVE NO.
18”	1/2”	10”	2’-1 3/8”	1/2”	P152,P153,P156,P157
22”	1/2”	14”	2’-5 3/8”	1/2”	P151,P154,P155,P158

Figure 3.8.1-8 Containment Penetrations (Sheet 16 of 17)



P200
(Fuel Transfer Tube)

All dimensions shown above are nominal dimensions

Figure 3.8.1-8 Containment Penetrations (Sheet 17 of 17)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.1-9 Identification of Areas for Thermal Conditions in Table 3.8.1-3 (Sheet 1 of 4)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

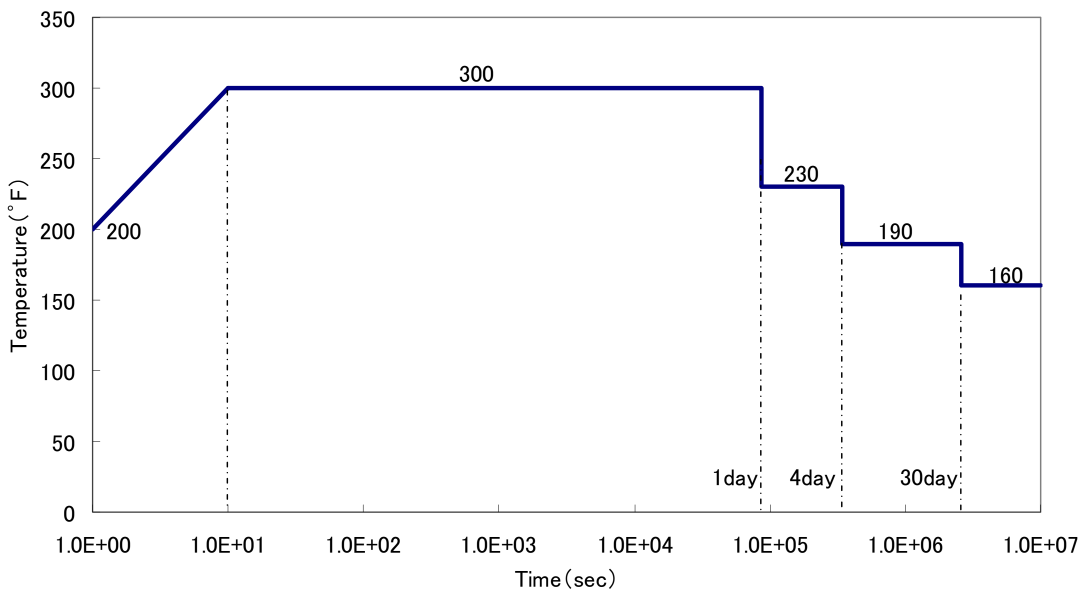
Figure 3.8.1-9 Identification of Areas for Thermal Conditions in Table 3.8.1-3 (Sheet 2 of 4)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.1-9 Identification of Areas for Thermal Conditions in Table 3.8.1-3 (Sheet 3 of 4)

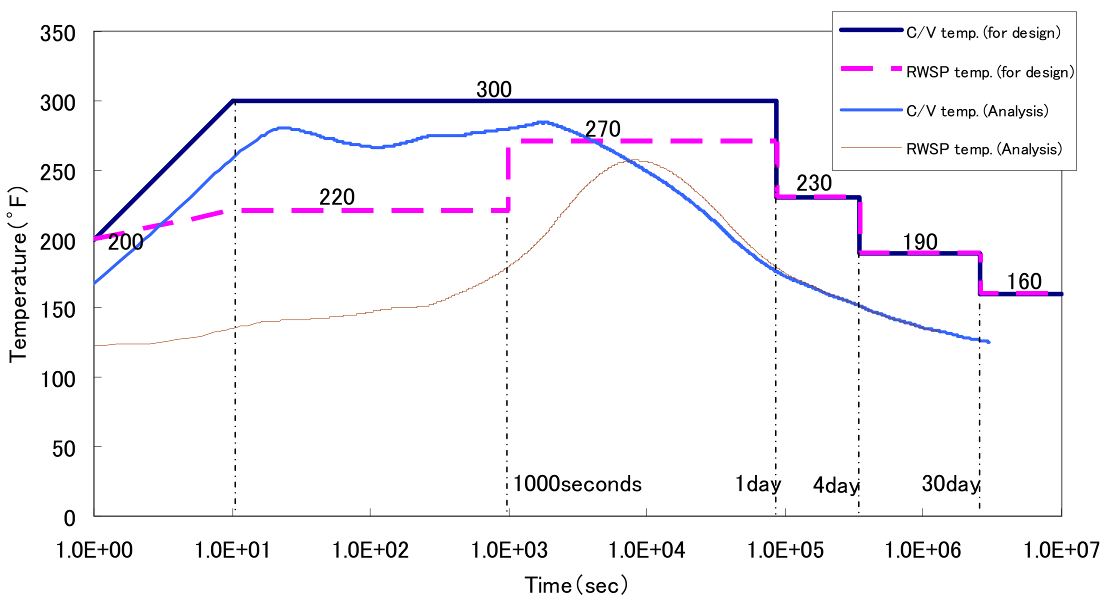
Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.1-9 Identification of Areas for Thermal Conditions in Table 3.8.1-3 (Sheet 4 of 4)



Note: In the temperature distribution analyses, temperature during normal operation is used as initial temperature.

Figure 3.8.1-10 Transient Conditions of PCCV Atmosphere Temperature (Accident Condition)



Note: In the temperature distribution analyses, temperature during normal operation is used as initial temperature.

Figure 3.8.1-11 Transient Conditions of Temperature of General Sump Pool Water in the PCCV and RWSP

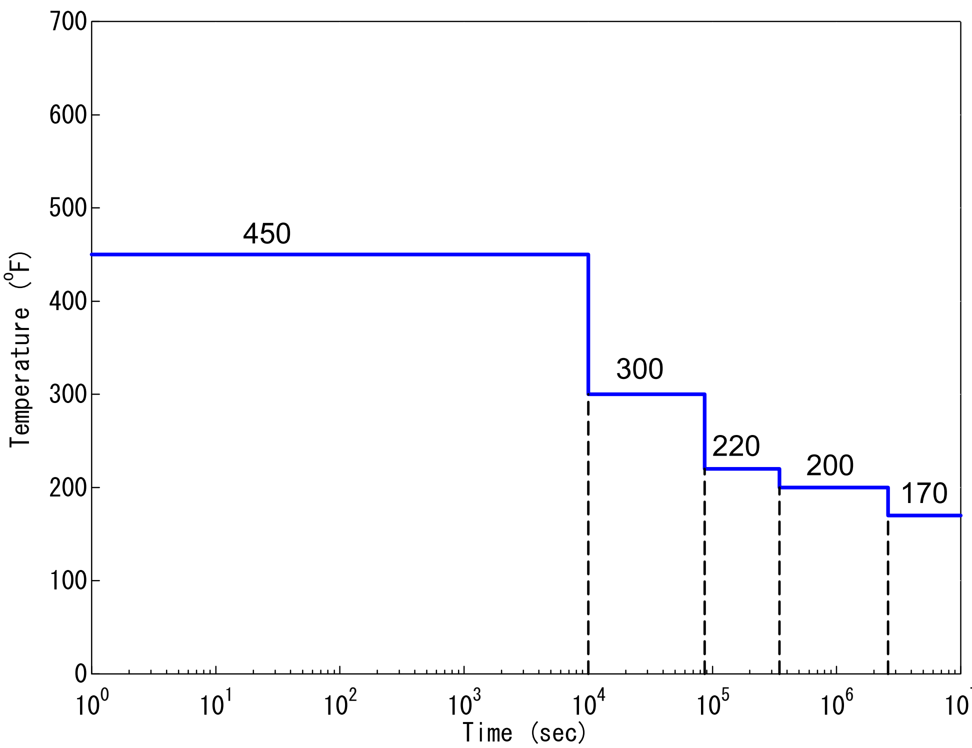


Figure 3.8.1-12 Transient Conditions of Temperature of the SG Compartment Atmosphere and Sump Pool Water (Pipe Break in the SG Compartment)

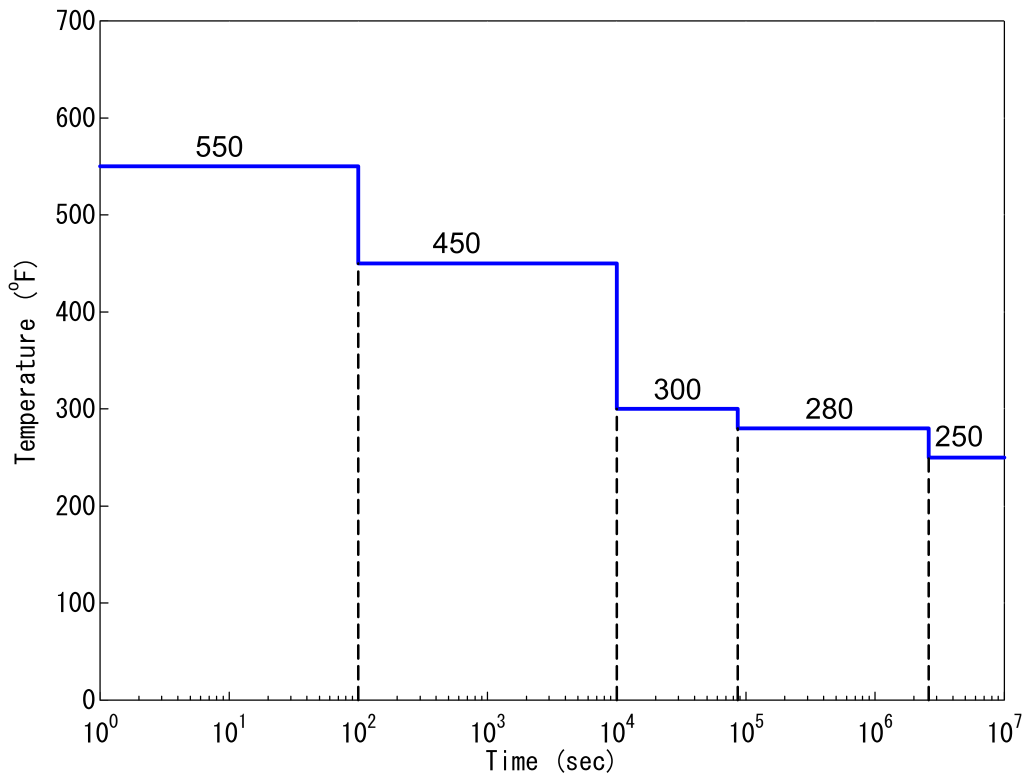


Figure 3.8.1-13 Transient Conditions of Temperature of the Reactor Cavity Atmosphere and Sump Pool Water (Pipe Break in the Reactor Cavity)

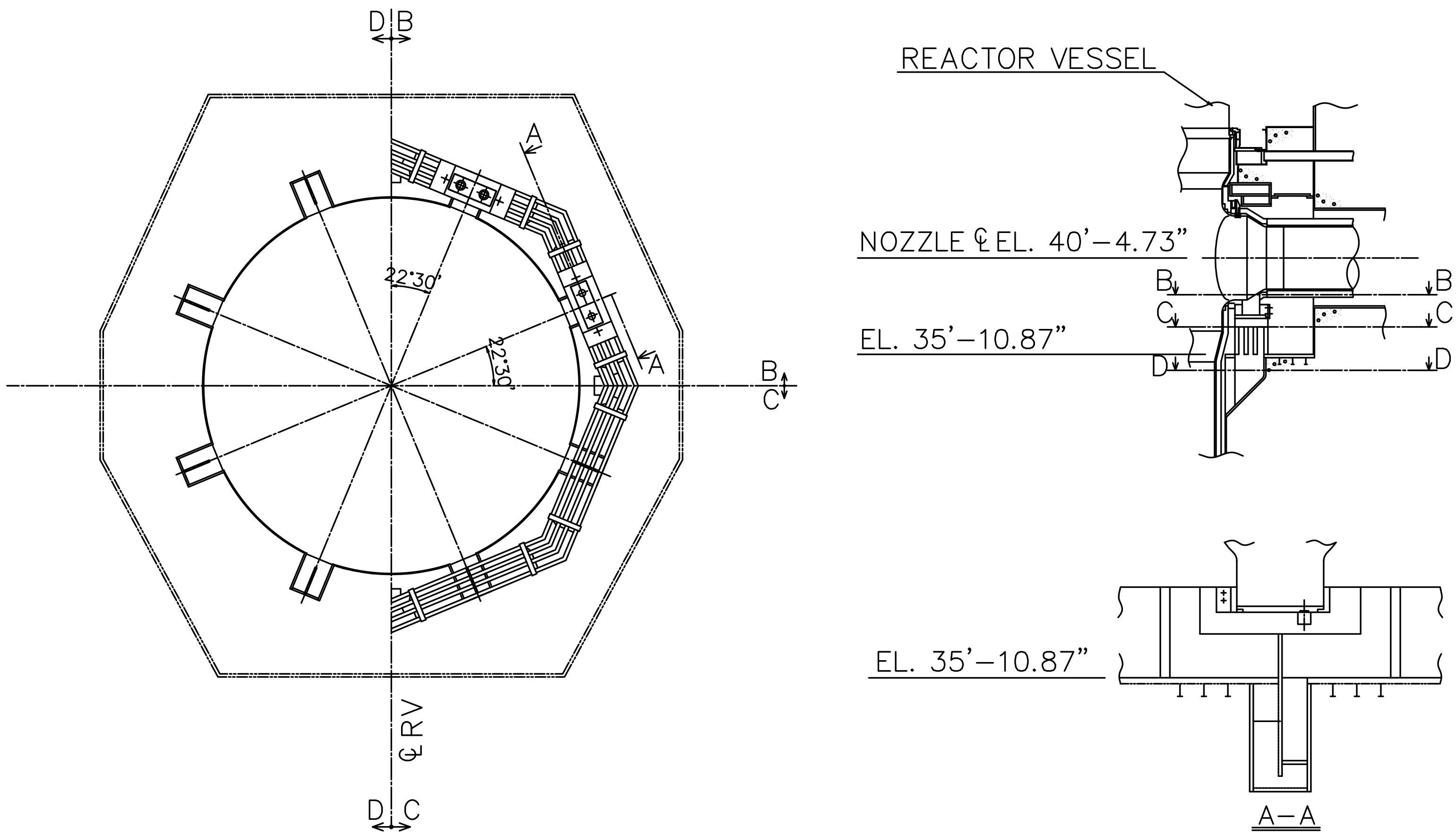


Figure 3.8.3-1 RV Support System

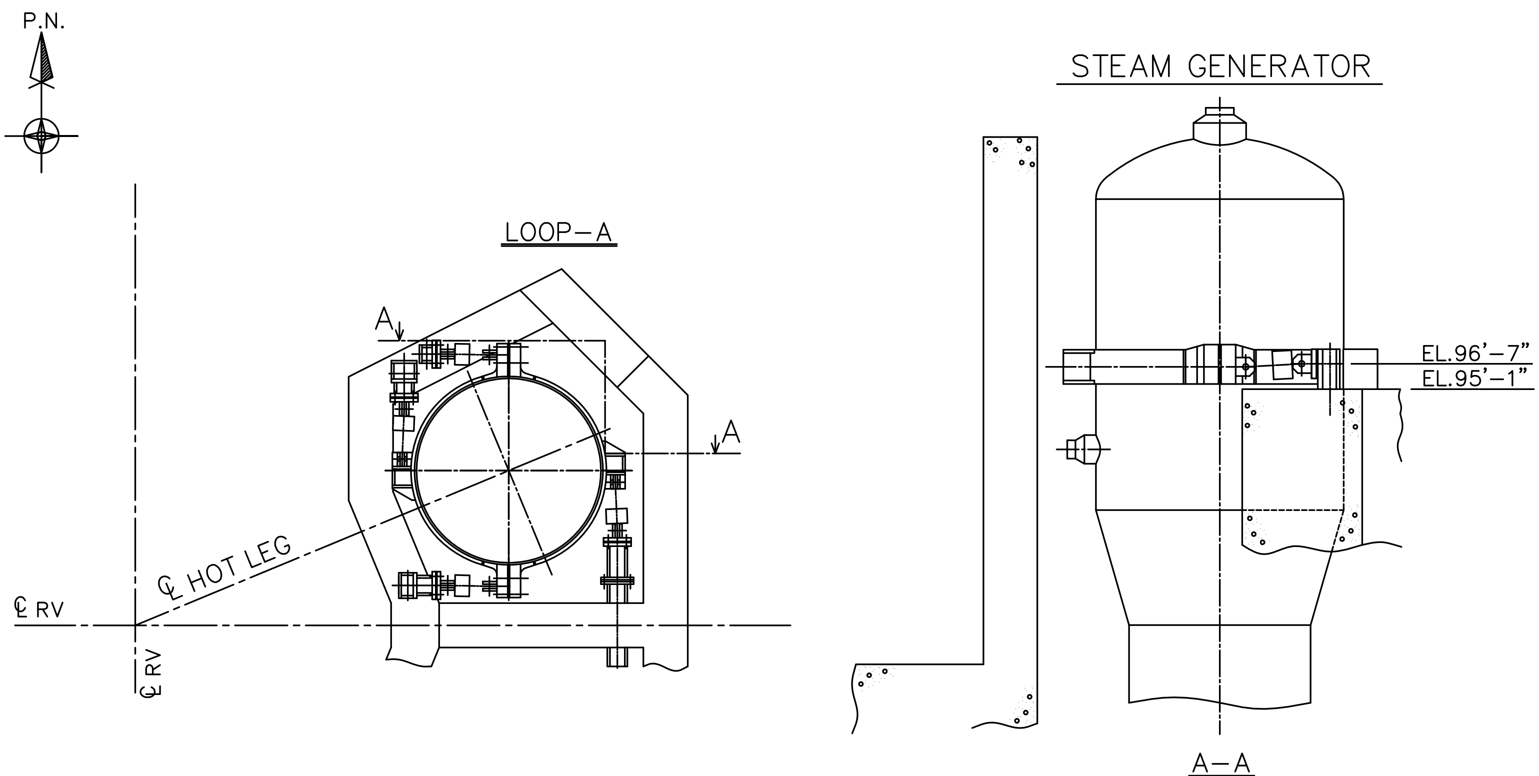


Figure 3.8.3-2 SG Support System (Sheet 1 of 4)

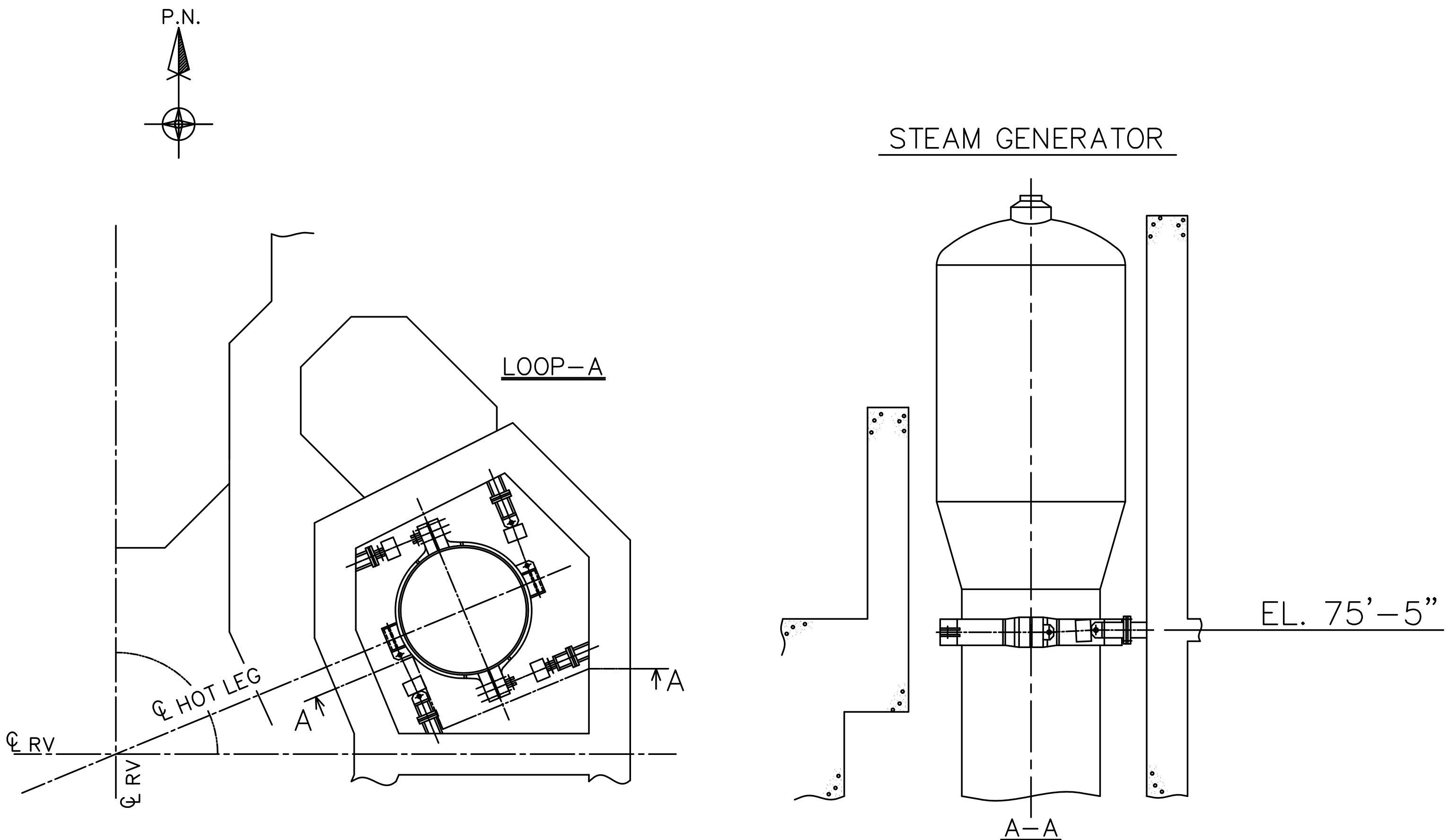


Figure 3.8.3-2 SG Support System (Sheet 2 of 4)

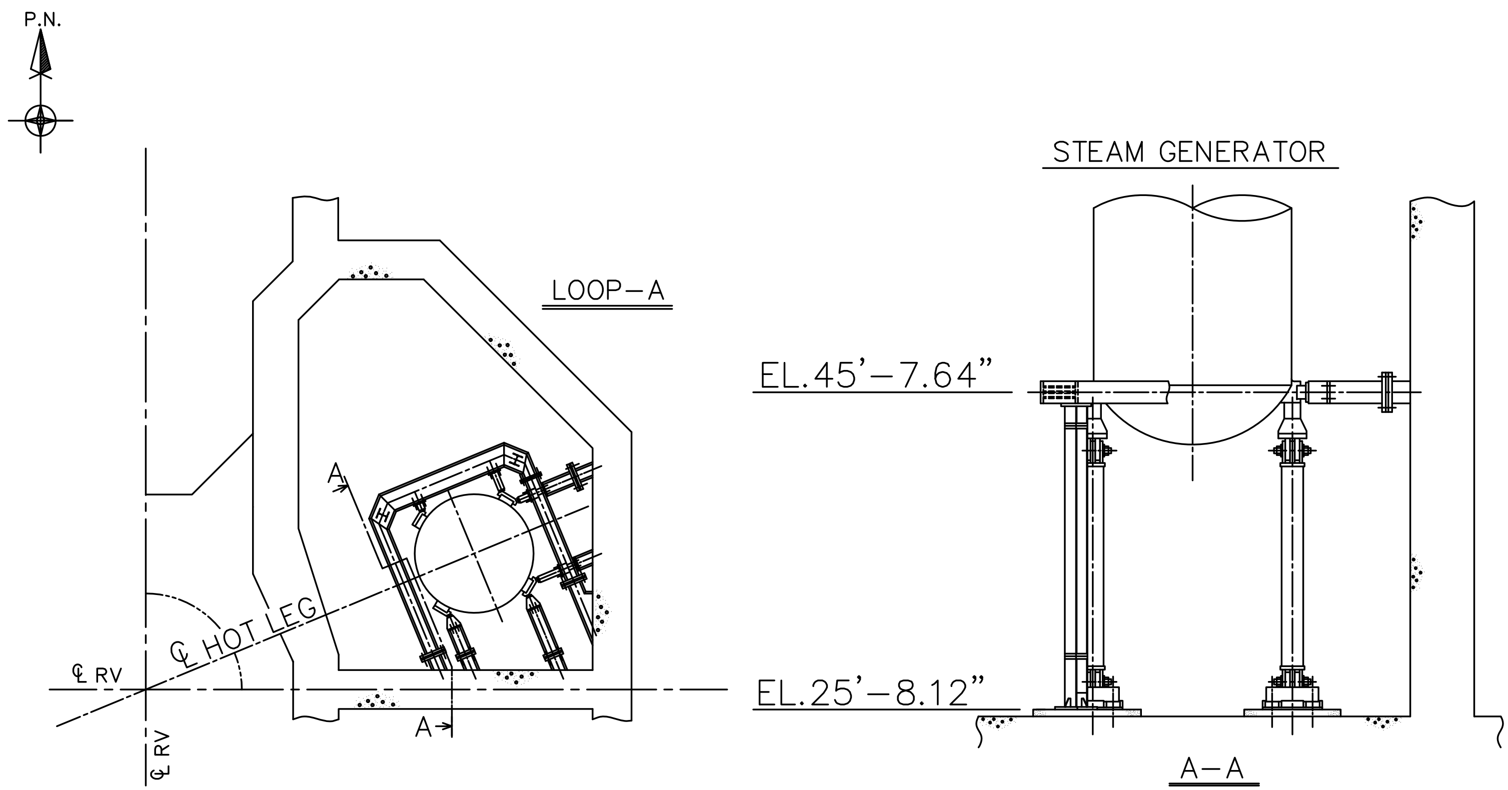


Figure 3.8.3-2 SG Support System (Sheet 3 of 4)

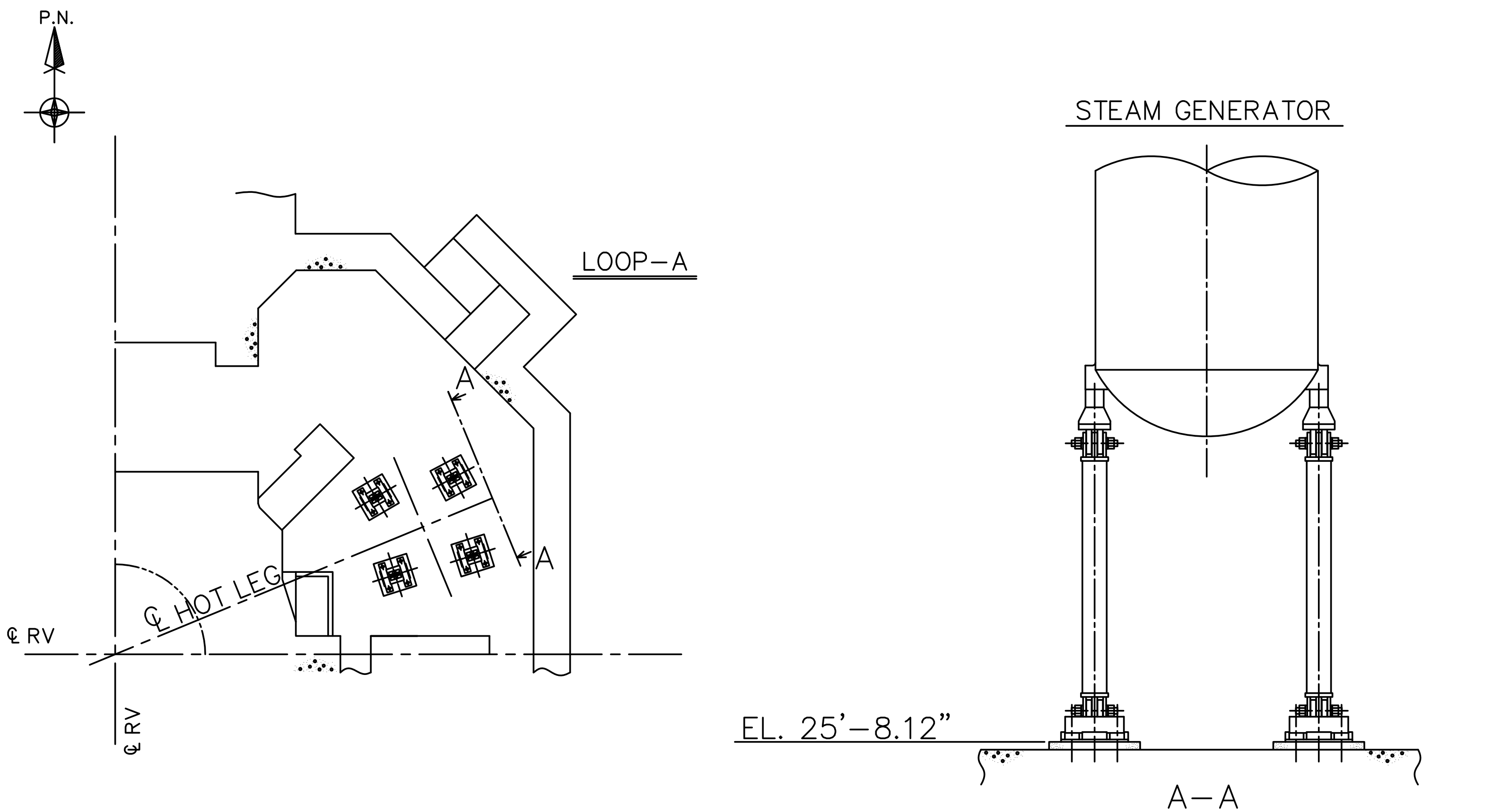


Figure 3.8.3-2 SG Support System (Sheet 4 of 4)

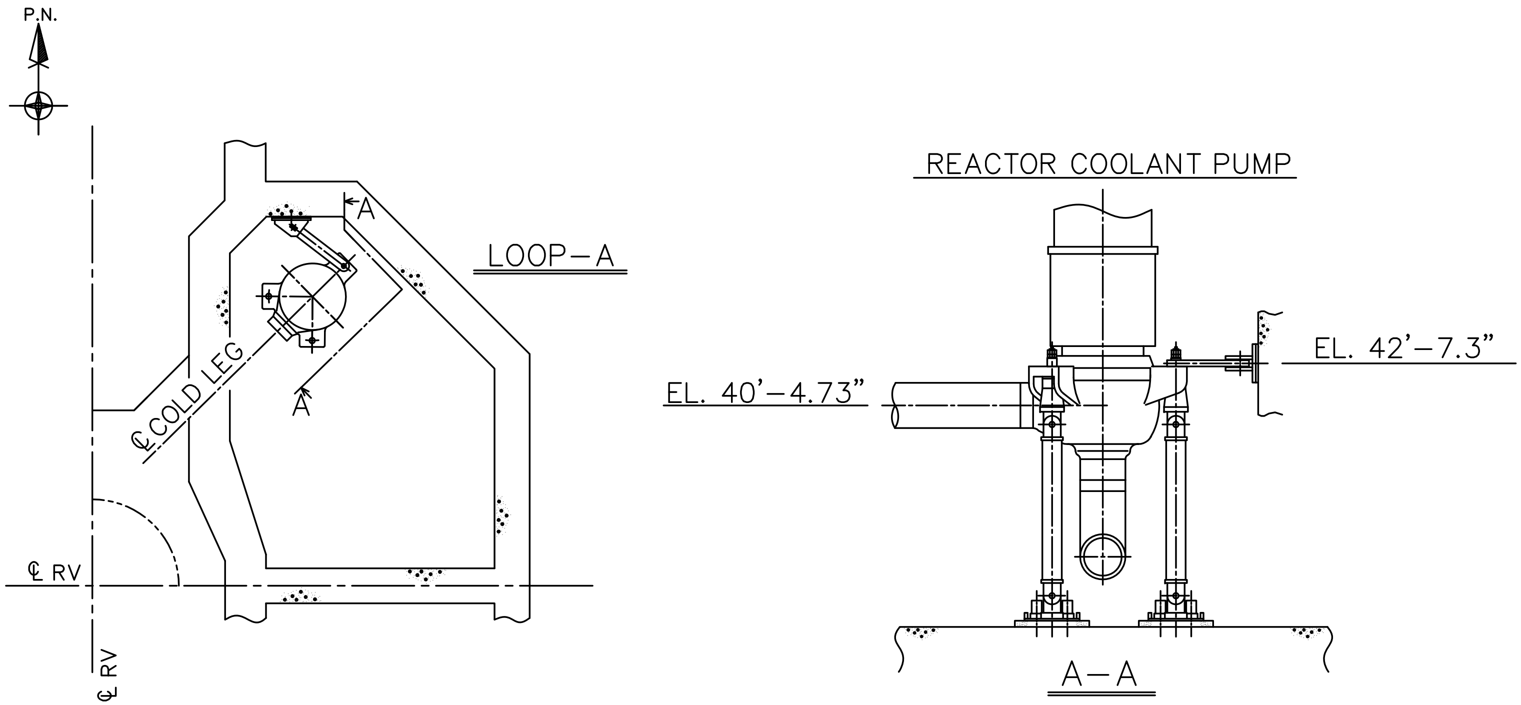


Figure 3.8.3-3 RCP Support System (Sheet 1 of 2)

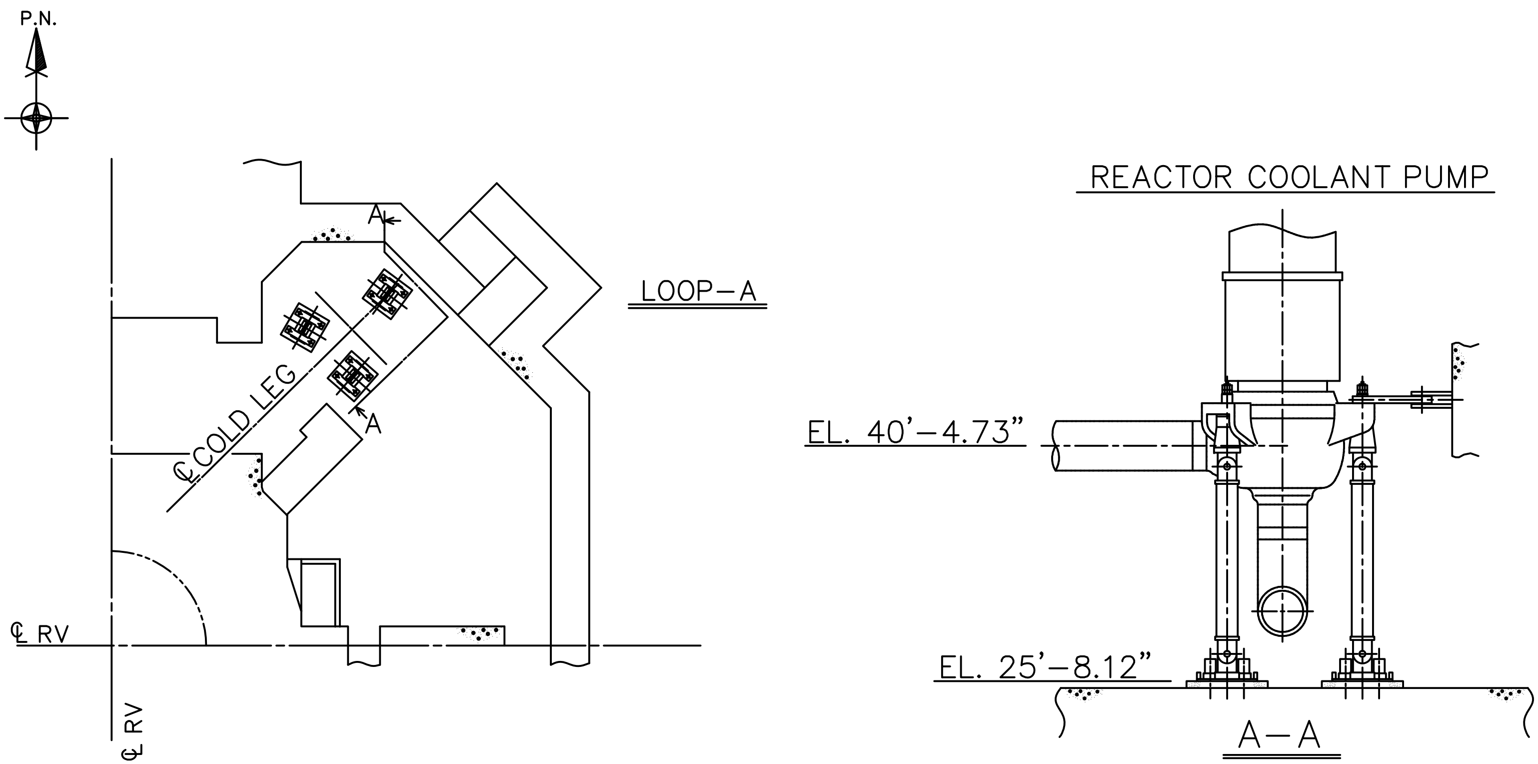


Figure 3.8.3-3 RCP Support System (Sheet 2 of 2)

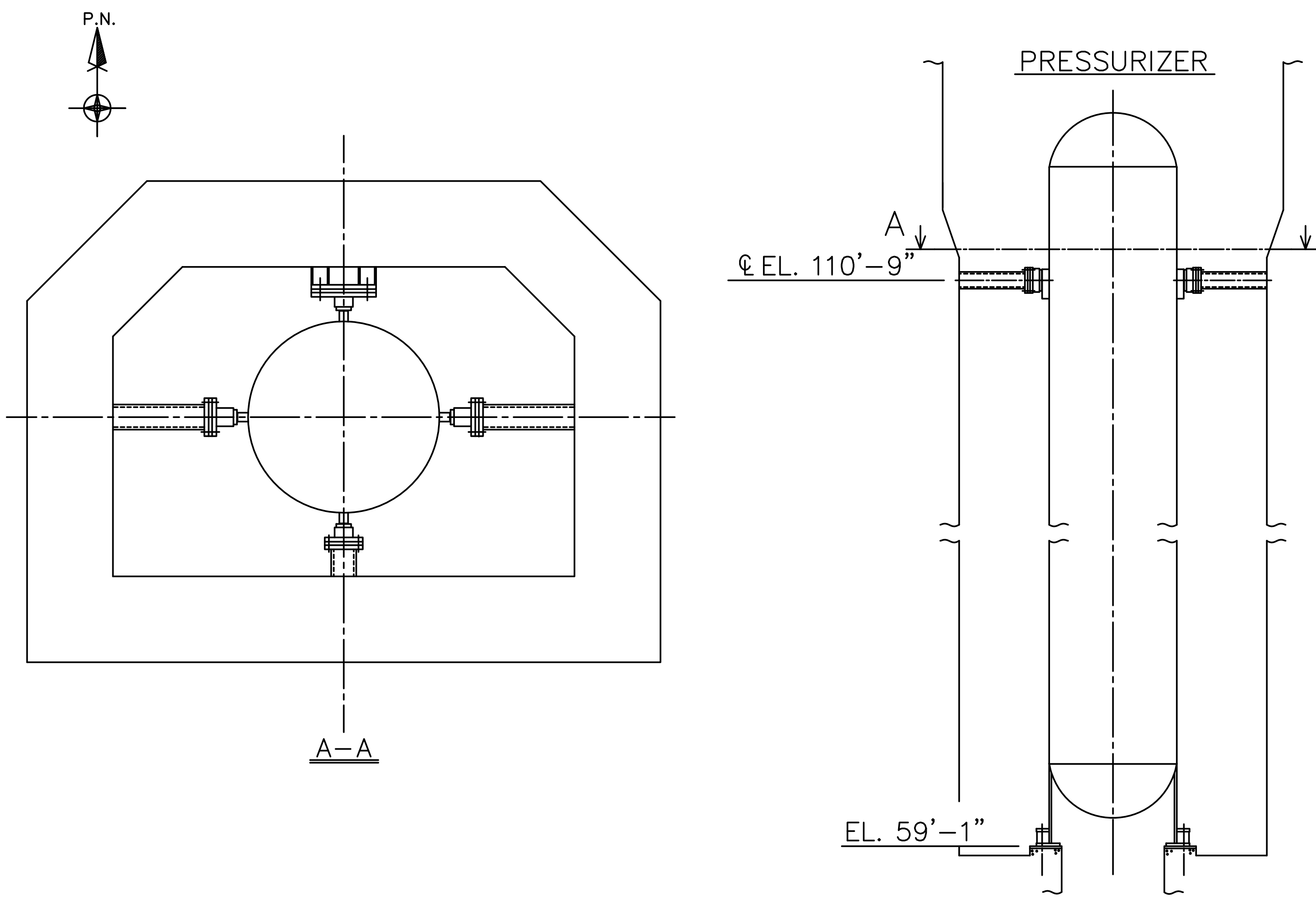
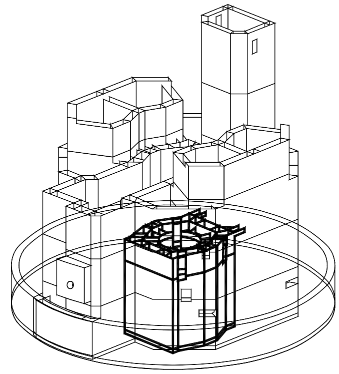


Figure 3.8.3-4 Pressurizer Support System



Key plan

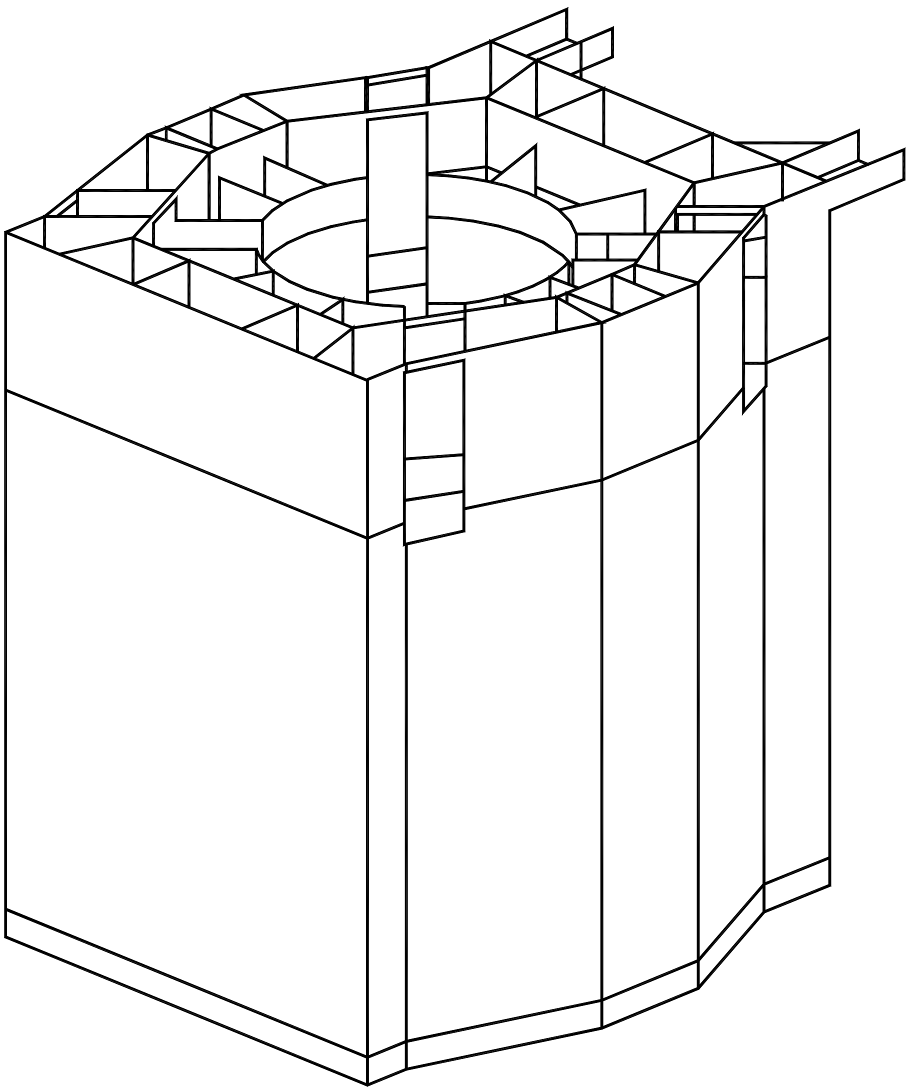


Figure 3.8.3-5 SC Module Isometrics (Sheet 1 of 8)

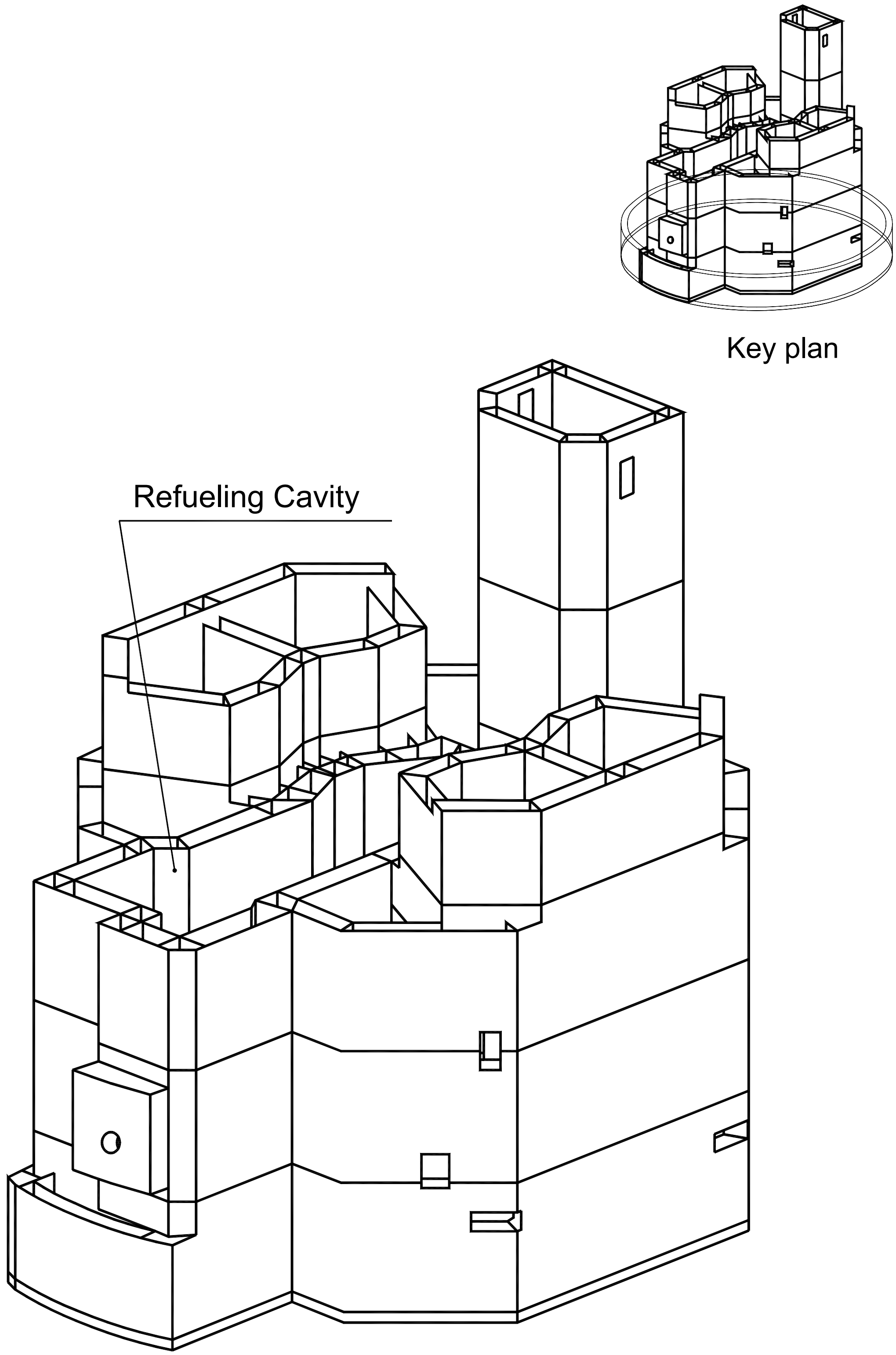
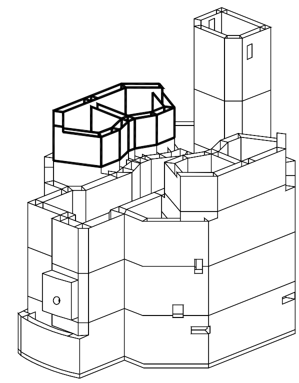
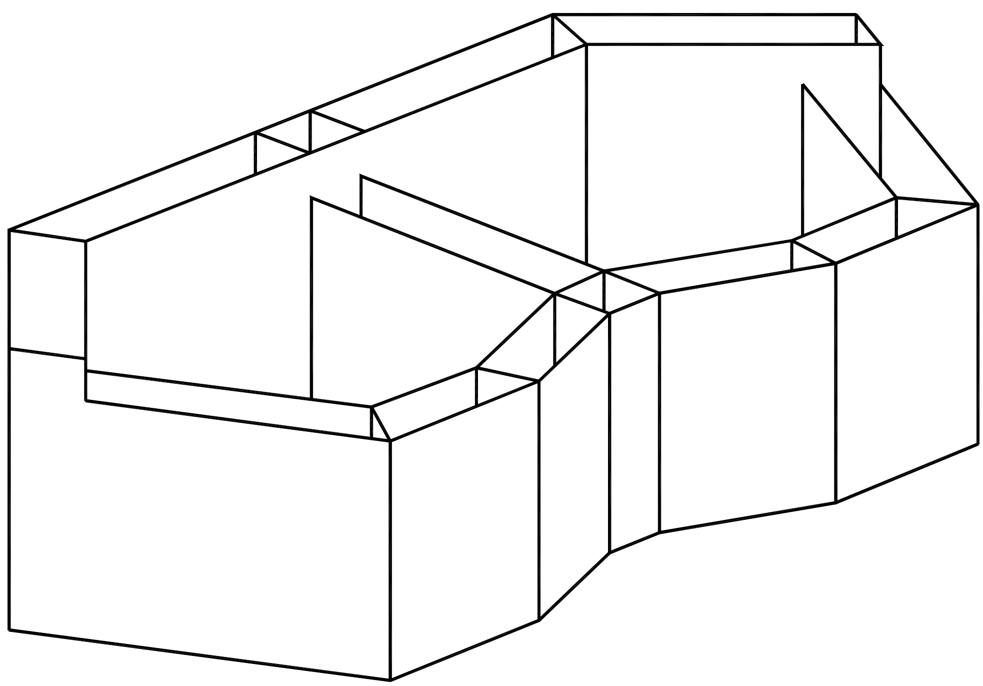


Figure 3.8.3-5 SC Module Isometrics (Sheet 2 of 8)

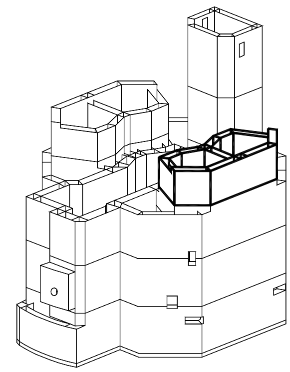


Key Plan

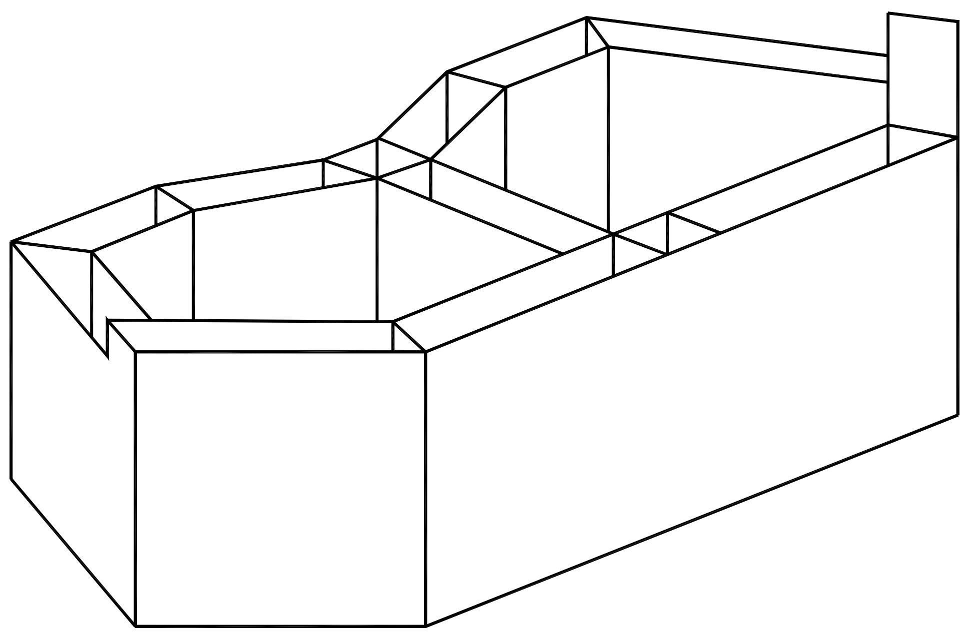


Part of Secondary Shield Walls

Figure 3.8.3-5 SC Module Isometrics (Sheet 3 of 8)

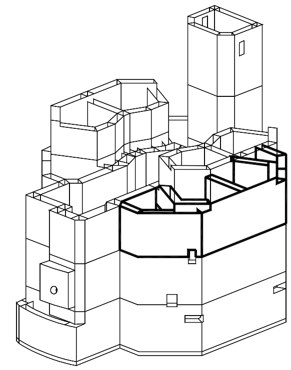


Key Plan

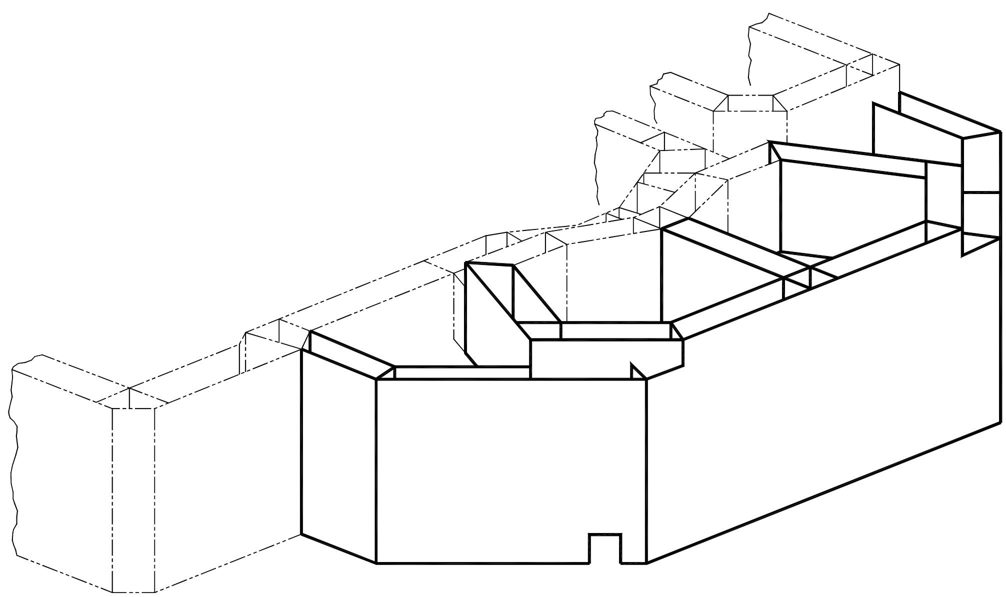


Part of Secondary Shield Walls

Figure 3.8.3-5 SC Module Isometrics (Sheet 4 of 8)

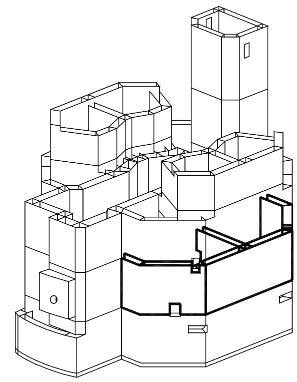


Key plan

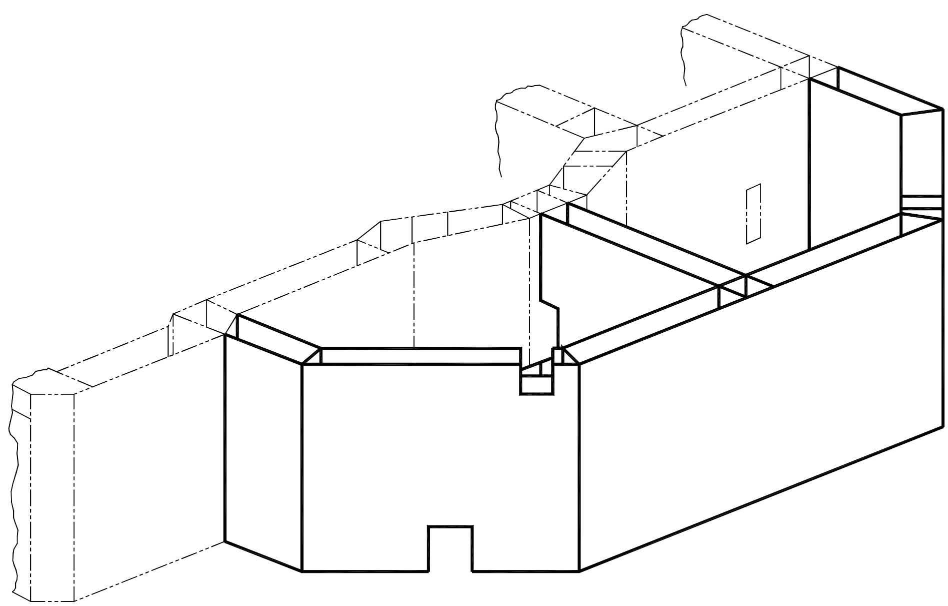


Part of Secondary Shield Walls

Figure 3.8.3-5 SC Module Isometrics (Sheet 5 of 8)

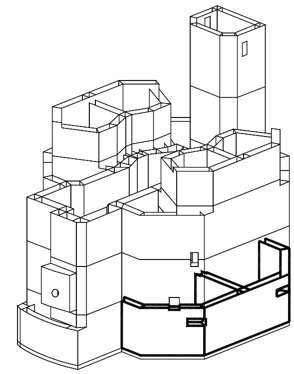


Key plan

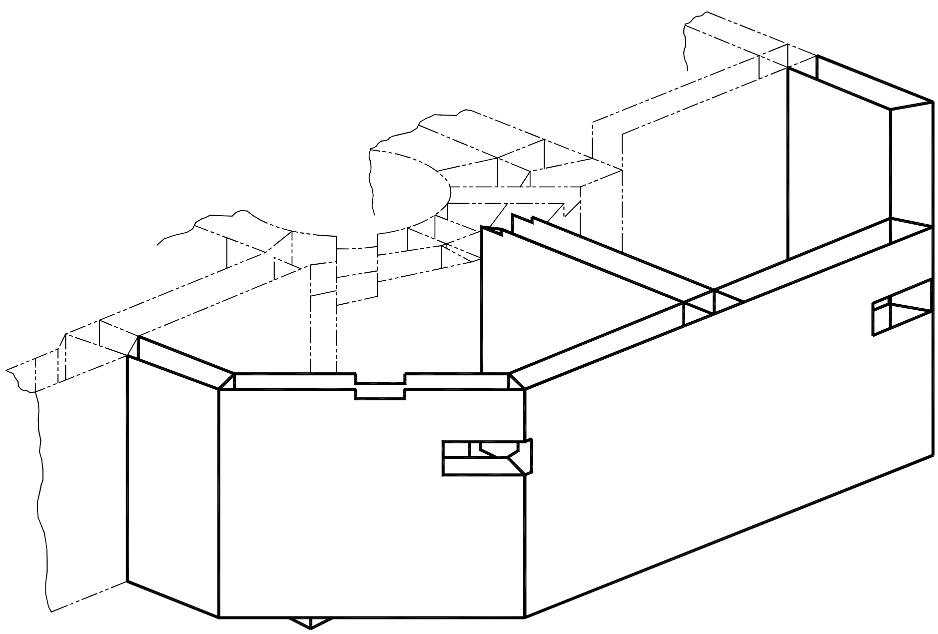


Part of Secondary Shield Walls

Figure 3.8.3-5 SC Module Isometrics (Sheet 6 of 8)

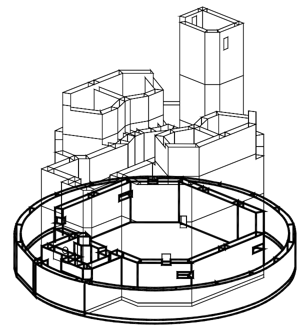


Key plan

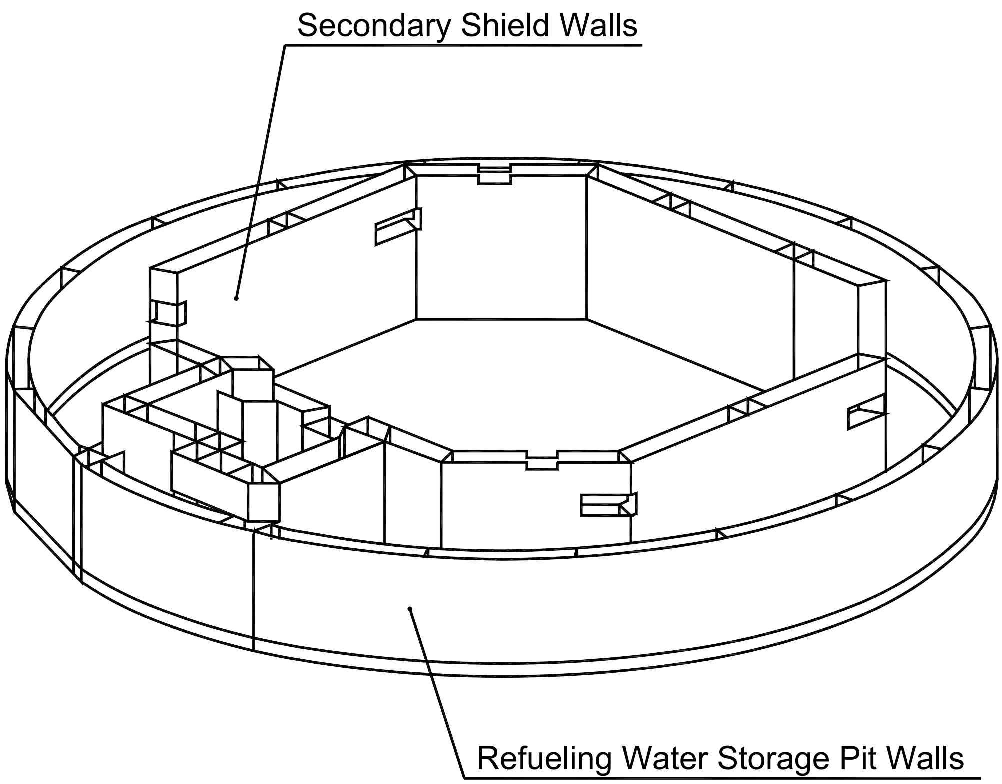


Part of Secondary Shield Walls

Figure 3.8.3-5 SC Module Isometrics (Sheet 7 of 8)



Key plan



Part of Secondary Shield Walls

Figure 3.8.3-5 SC Module Isometrics (Sheet 8 of 8)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.3-6 Interior Compartments Wall Layout and Configuration (Sheet 1 of 7)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.3-6 Interior Compartments Wall Layout and Configuration (Sheet 2 of 7)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.3-6 Interior Compartments Wall Layout and Configuration (Sheet 3 of 7)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.3-6 Interior Compartments Wall Layout and Configuration (Sheet 4 of 7)

Security-Related Information – Withheld Under 10 CFR 2.390(d)(1)

Figure 3.8.3-6 Interior Compartments Wall Layout and Configuration (Sheet 5 of 7)

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-6 Interior Compartments Wall Layout and Configuration (Sheet 6 of 7)

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-6 Interior Compartments Wall Layout and Configuration (Sheet 7 of 7)

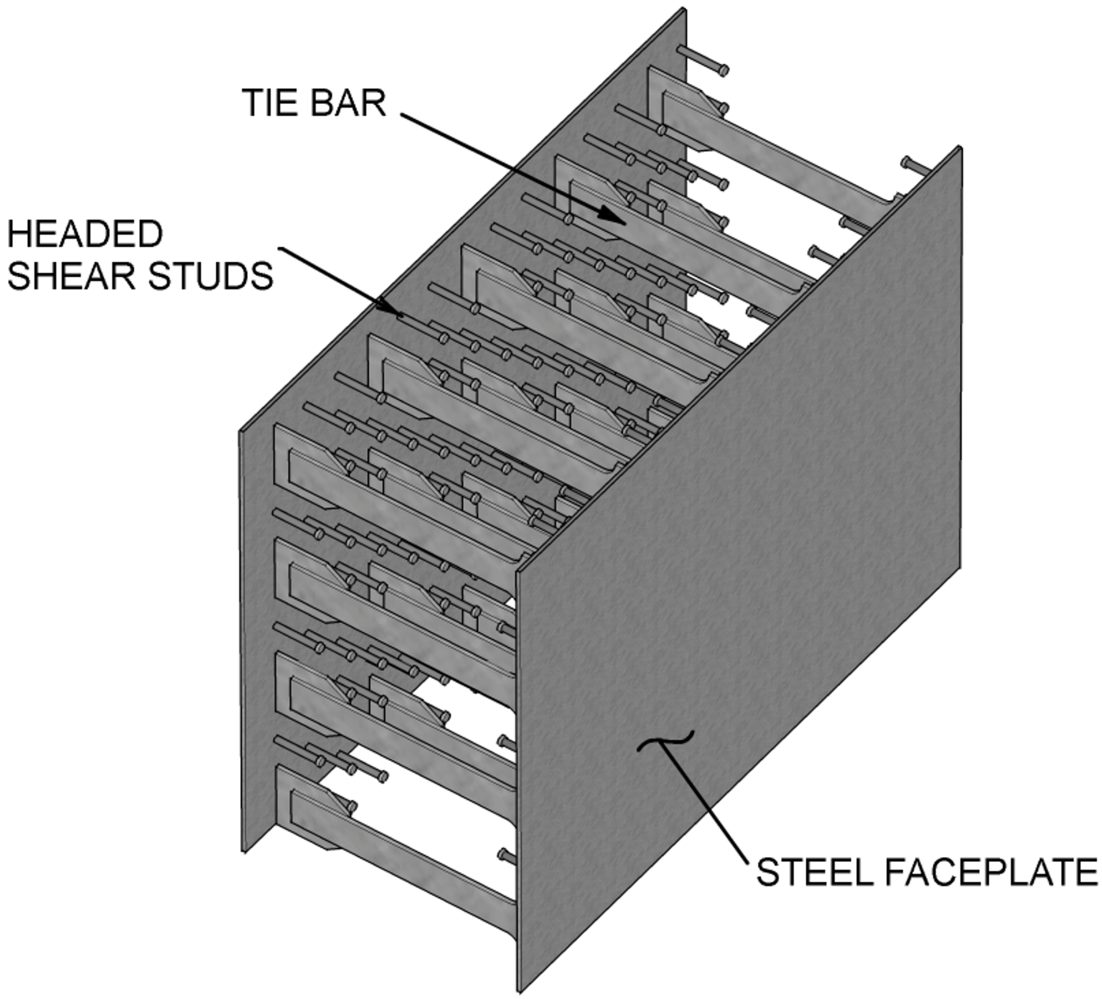
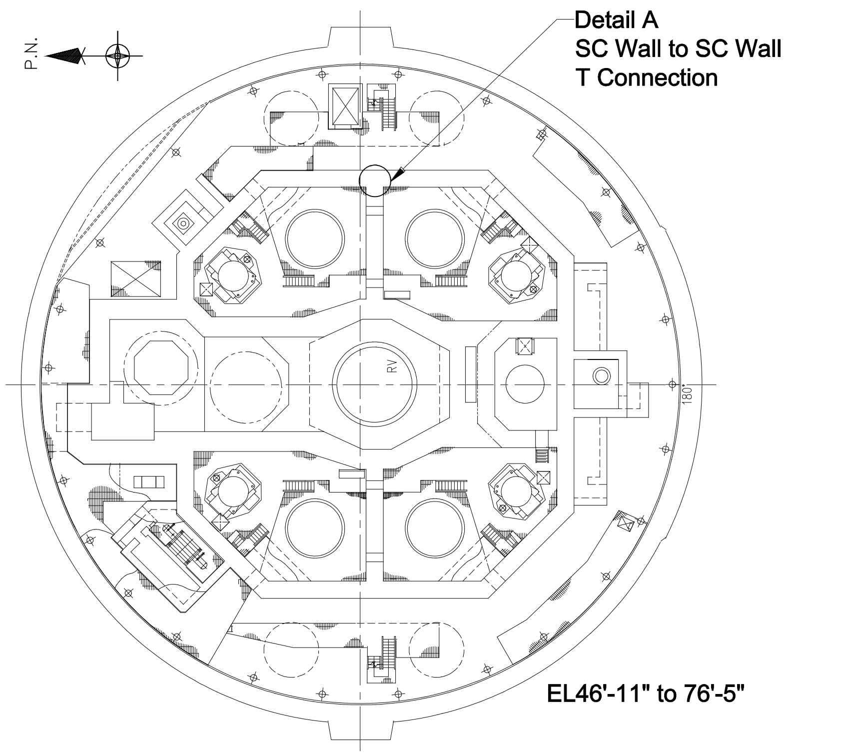


Figure 3.8.3-7 Typical Details of SC Modules (Sheet 1 of 5)



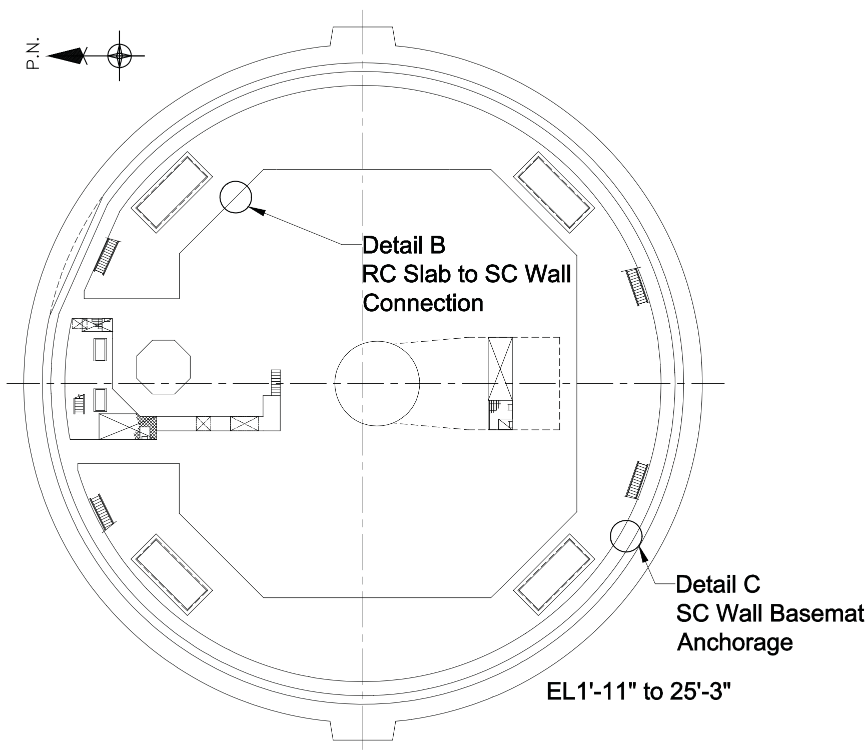


Figure 3.8.3-7 Typical Details of SC Modules (Sheet 2 of 5)

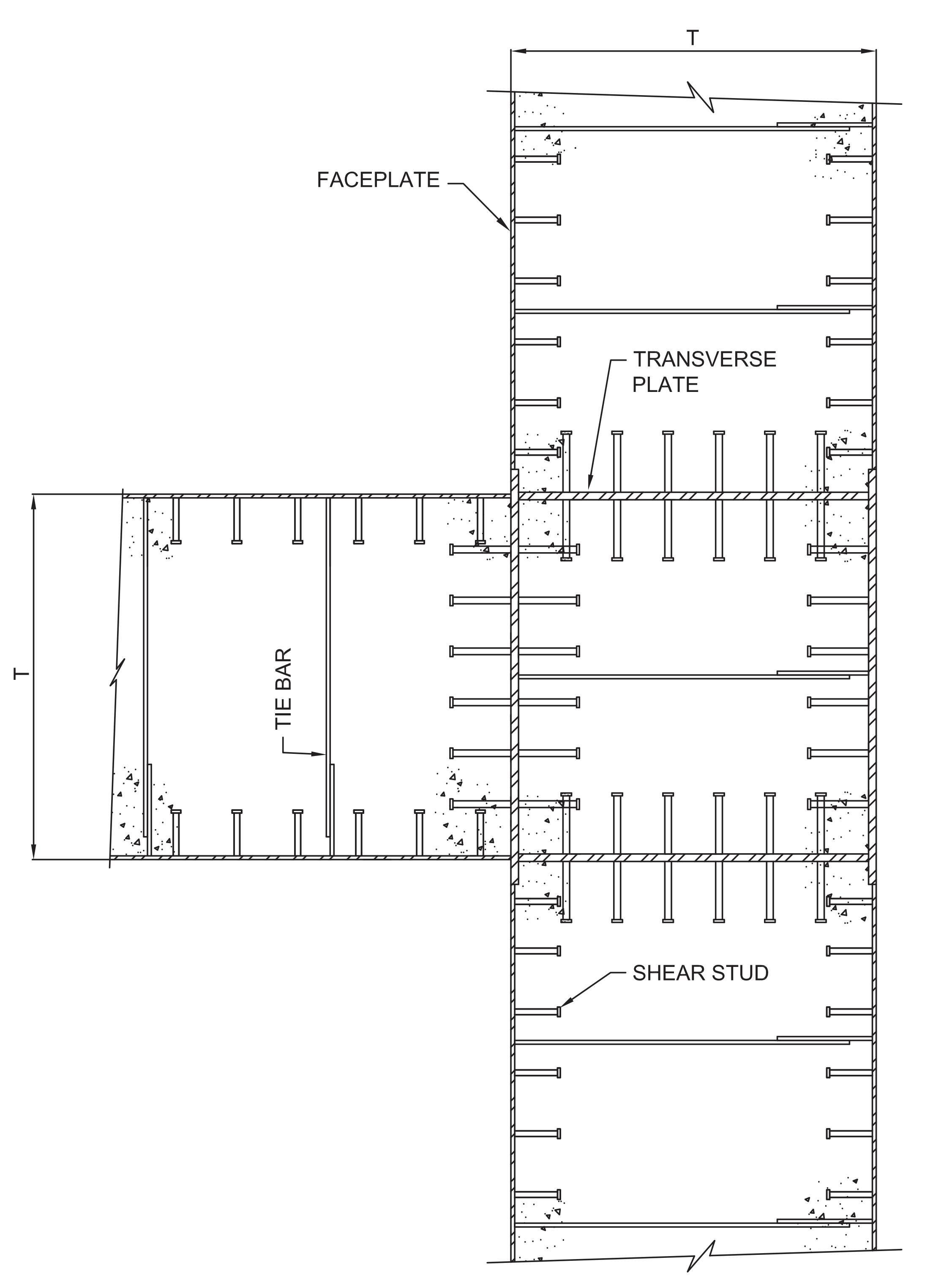


Figure 3.8.3-7 Typical Details of SC Modules (Sheet 3 of 5)

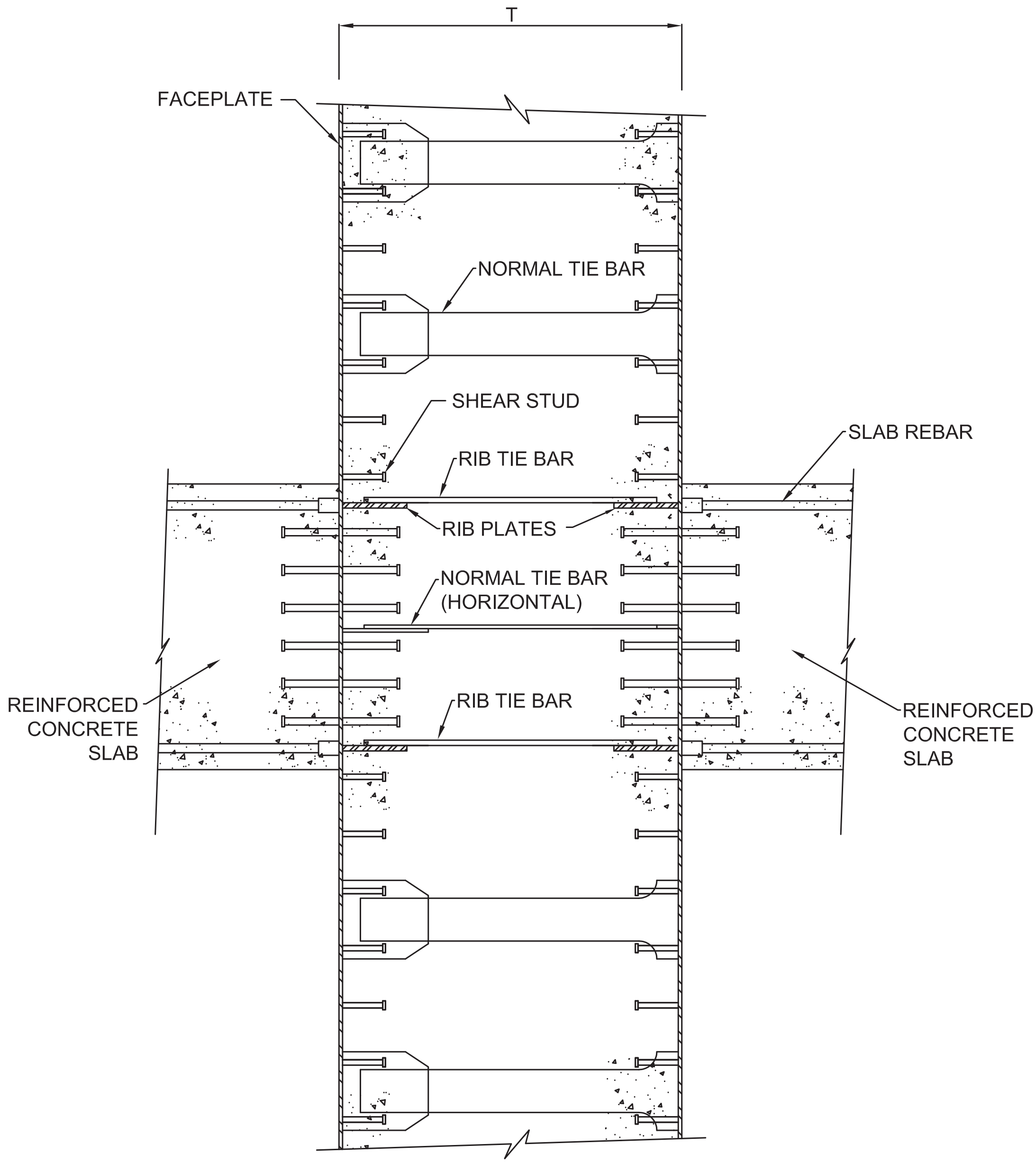


Figure 3.8.3-7 Typical Details of SC Modules (Sheet 4 of 5)

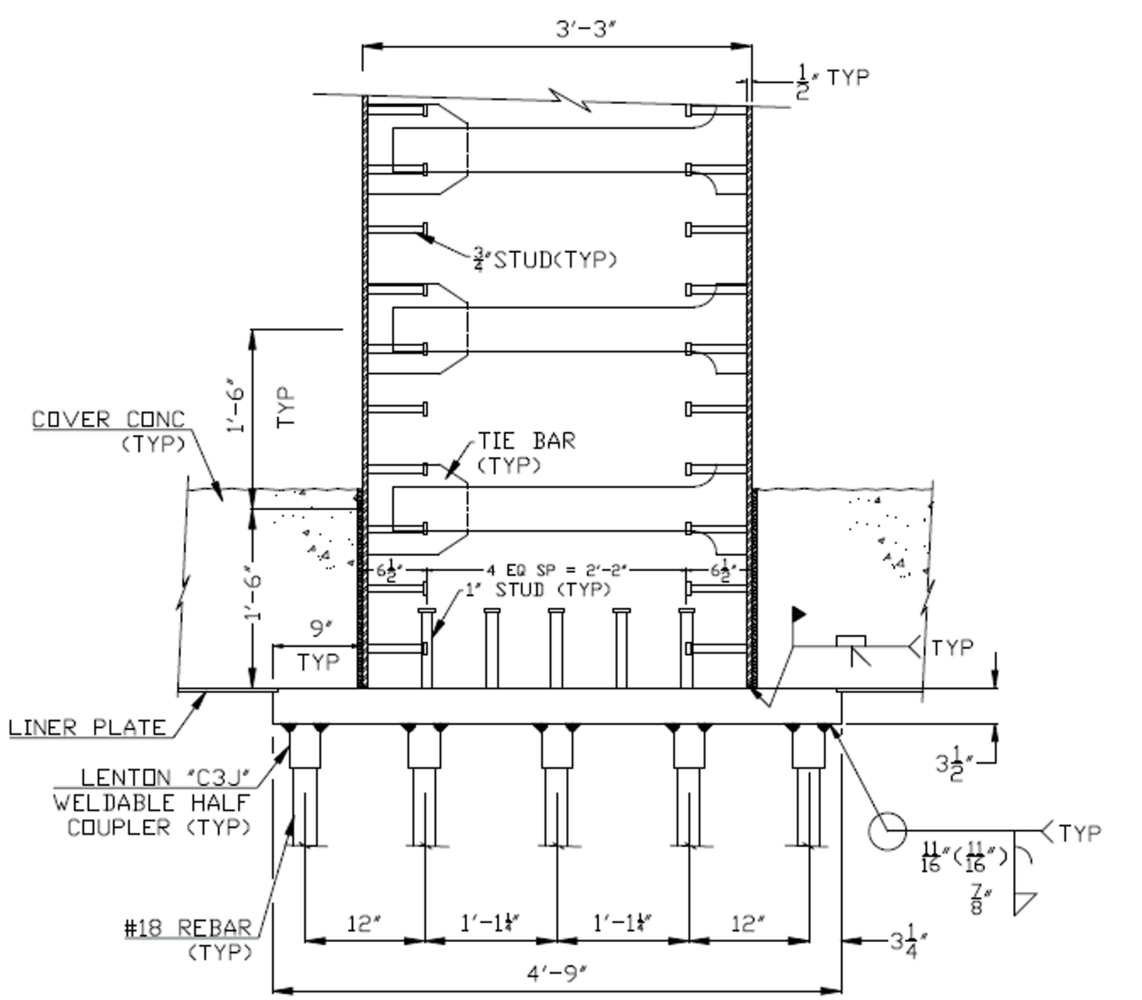
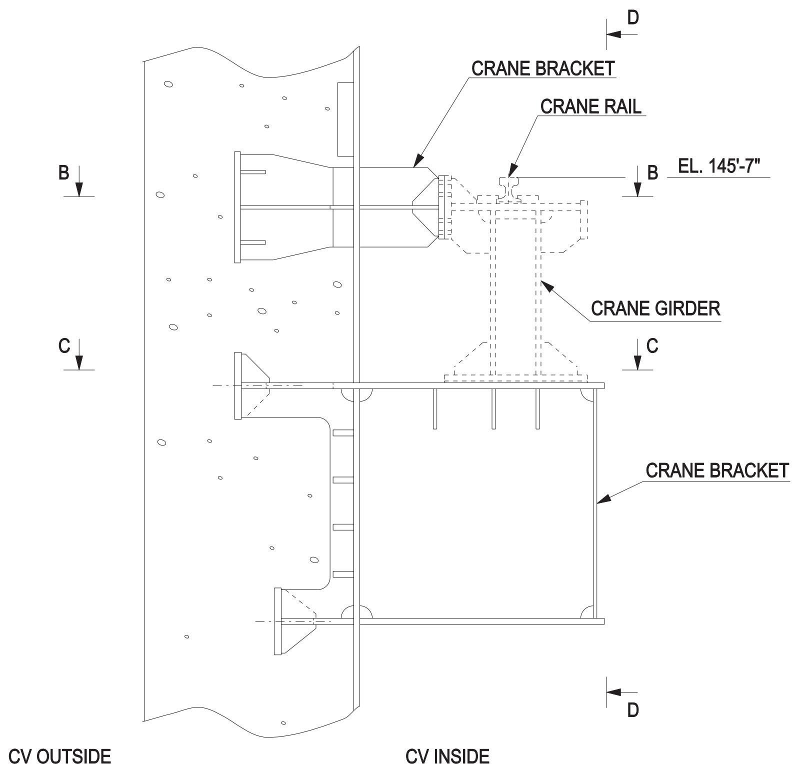
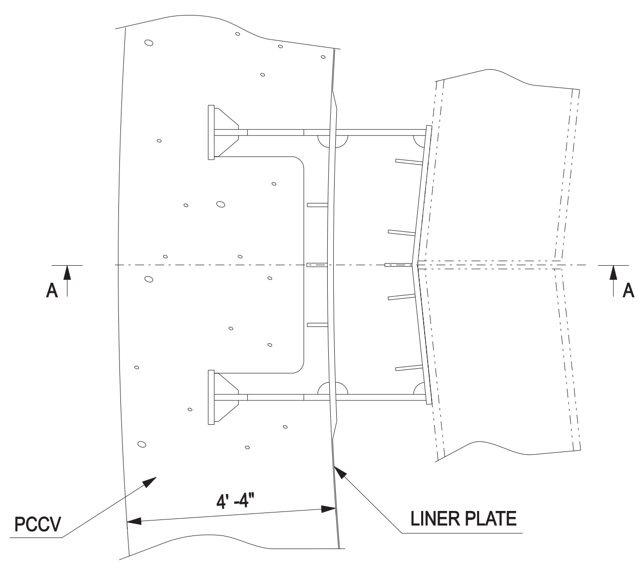


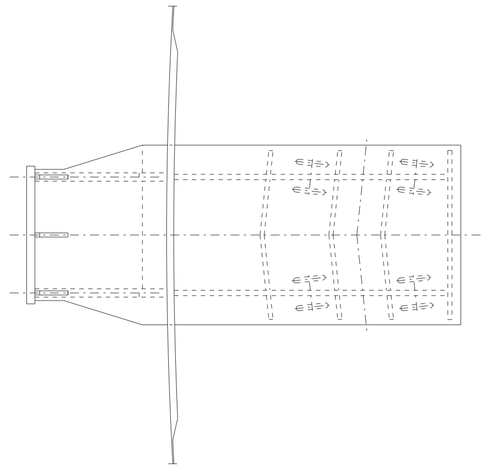
Figure 3.8.3-7 Typical Details of SC Modules (Sheet 5 of 5)



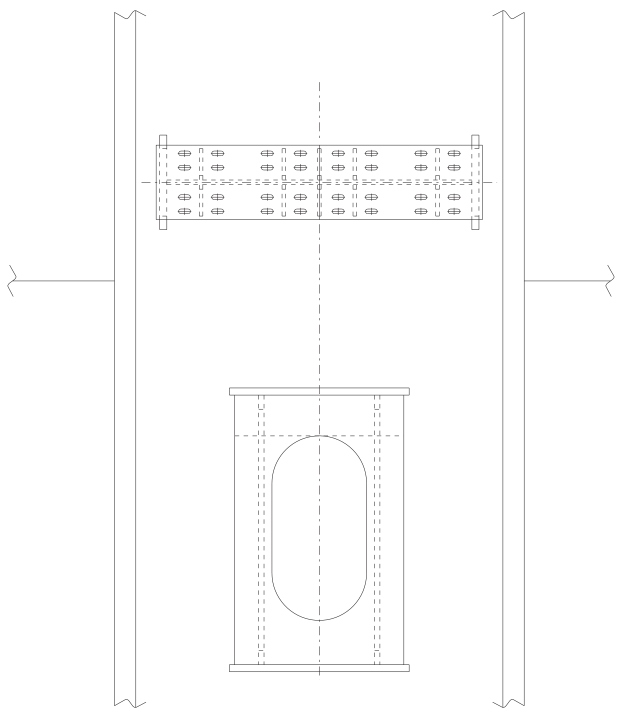


B - B

Figure 3.8.3-8 Polar Crane Supports (Sheet 1 of 3)



C - C



D - D

Figure 3.8.3-8 Polar Crane Supports (Sheet 2 of 3)

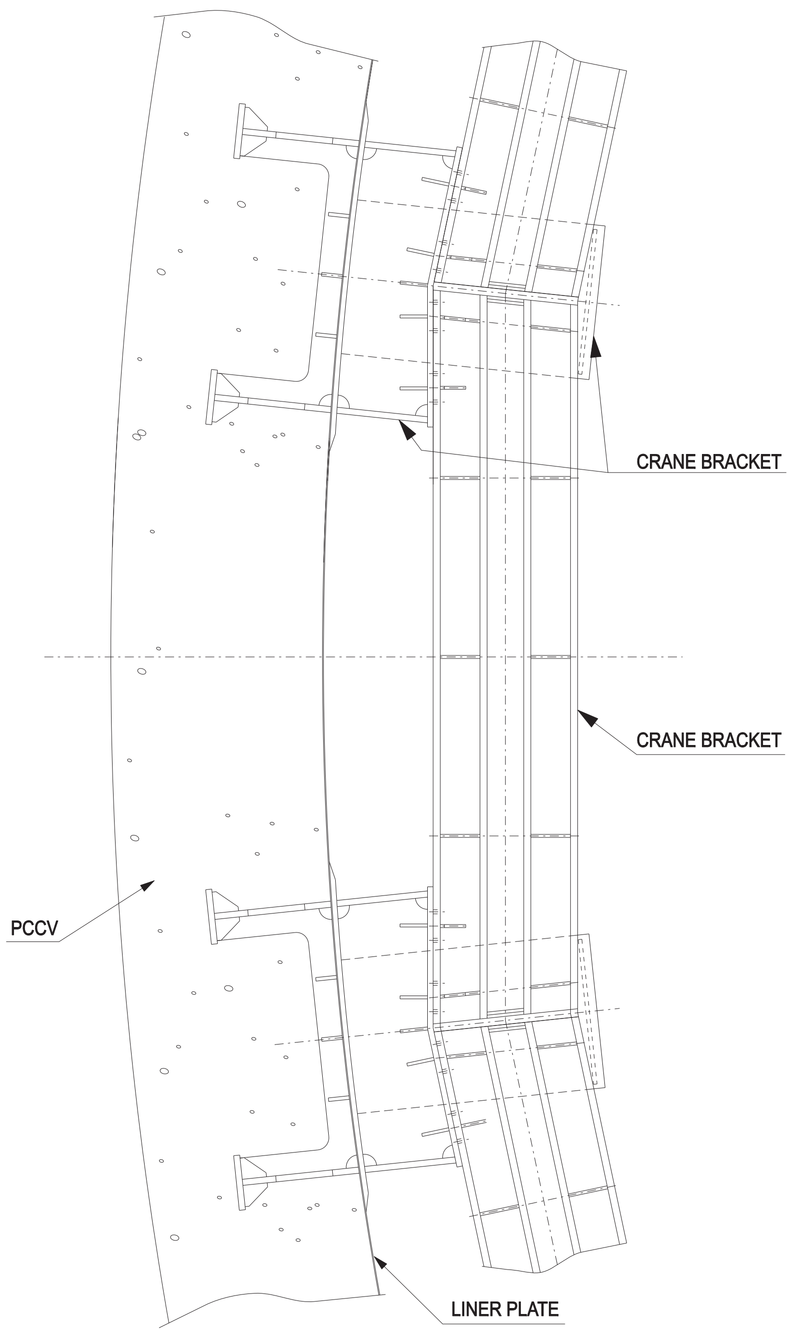
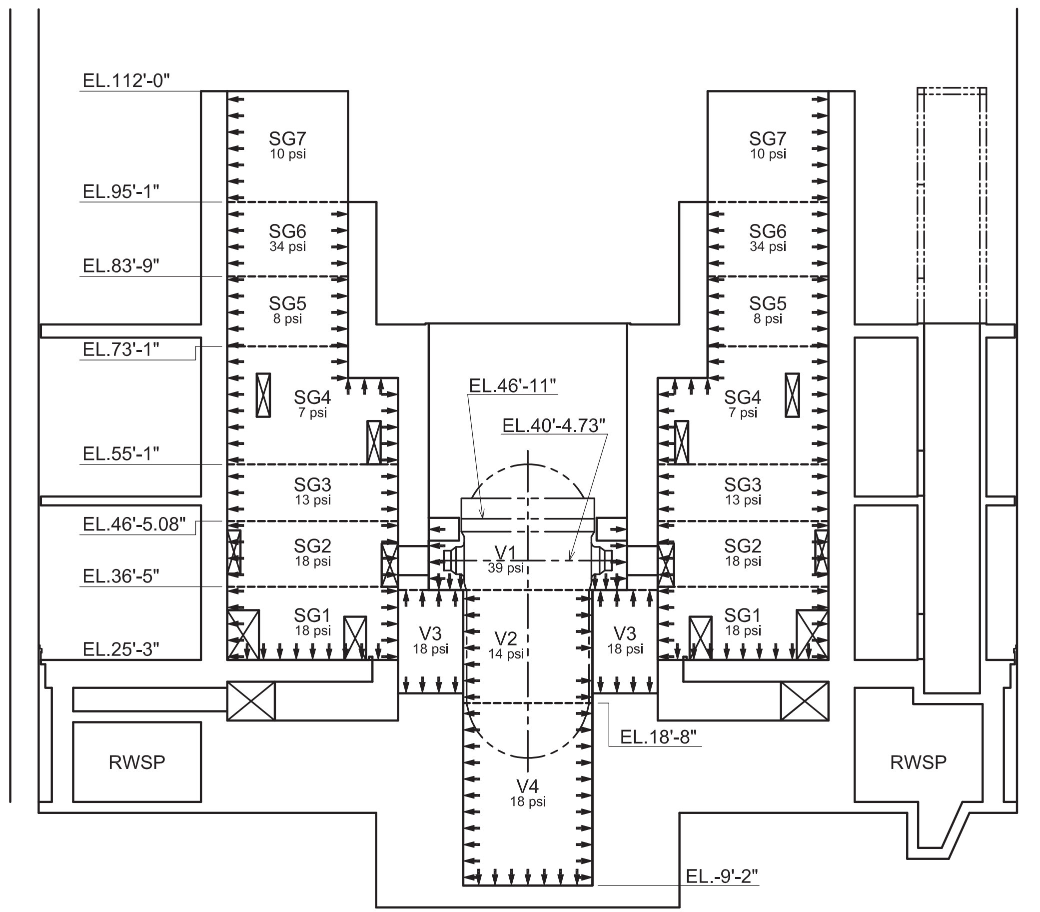
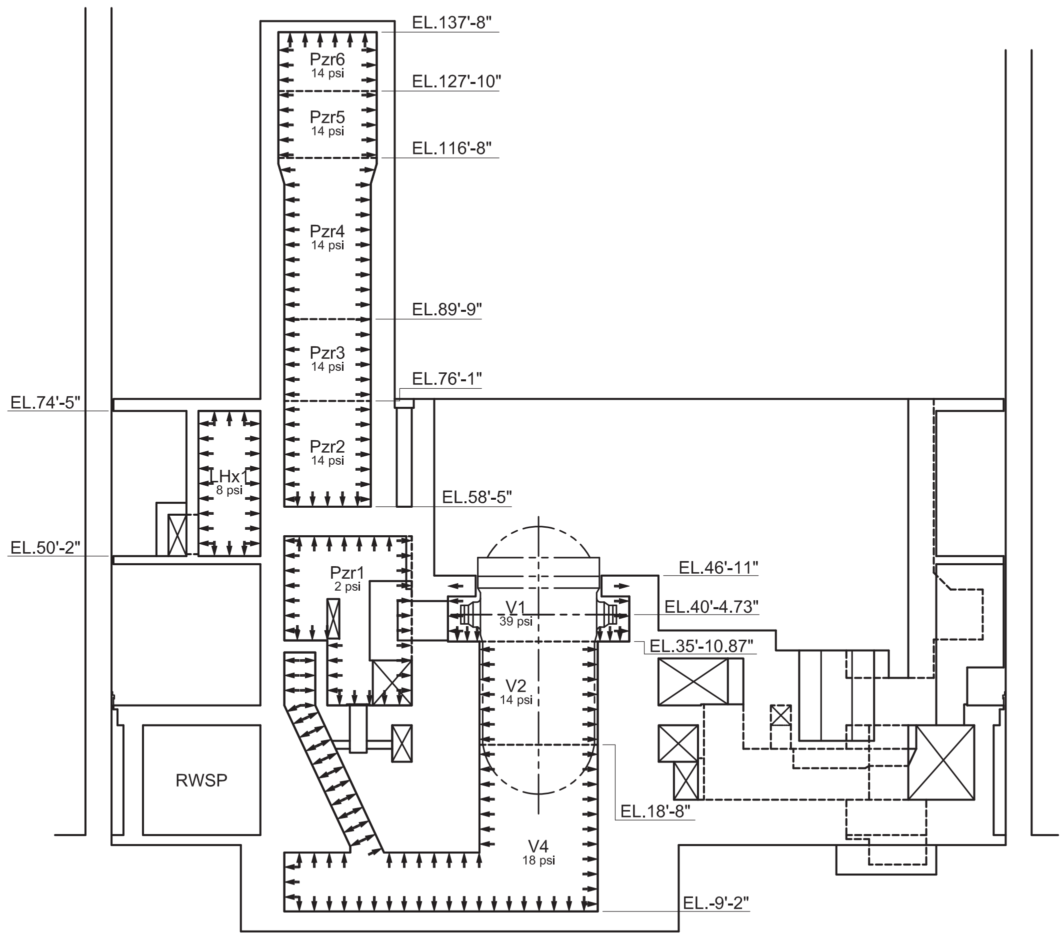


Figure 3.8.3-8 Polar Crane Supports (Sheet 3 of 3)



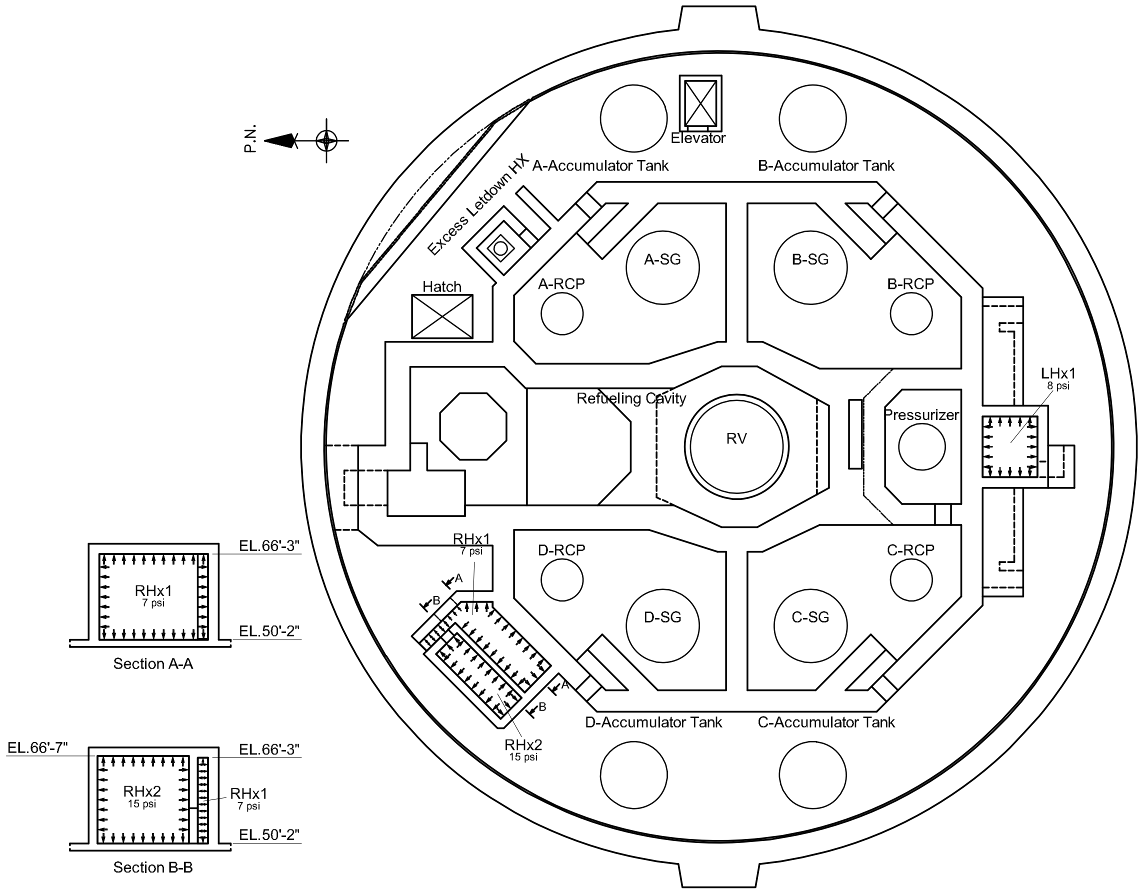
East-West Section

Figure 3.8.3-9 CIS Pressure Loads
(Sheet 1 of 3)



North-South Section

Figure 3.8.3-9 CIS Pressure Loads (Sheet 2 of 3)



Plan EL.50'-2"

Figure 3.8.3-9 CIS Pressure Loads (Sheet 3 of 3)

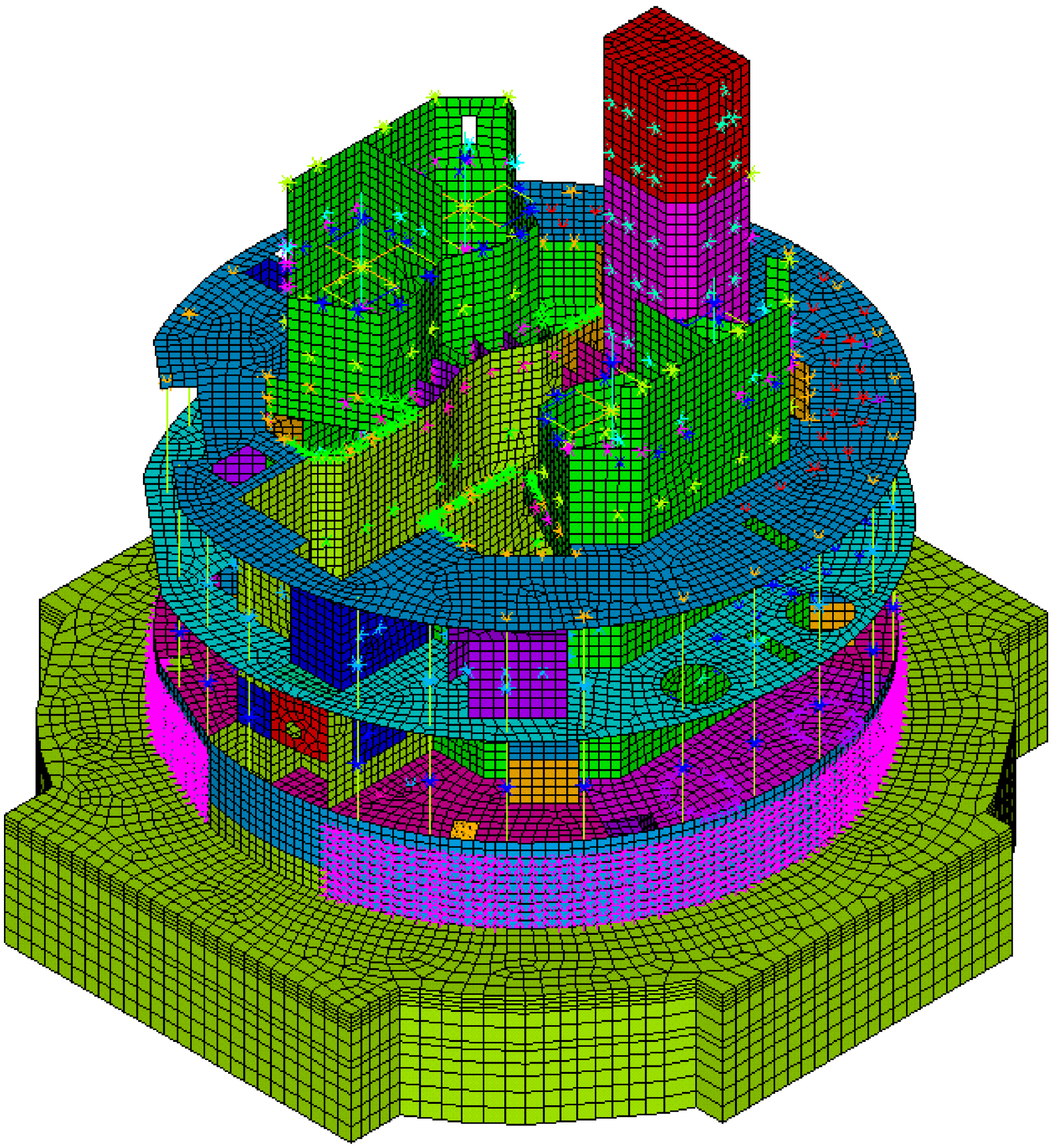


Figure 3.8.3-10 CIS FE Model

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-11 Critical Sections of SC Modules

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-12 Structural Categories Between Elevations 3'-7" and 21'-0"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-13 Structural Categories Between Elevations 21'-0" and 35'-11"

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-14 Structural Categories Between Elevations 37’-9” and 62’-4”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-15 Structural Categories Between Elevations 62’-4” and 76’-5”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-16 Structural Categories Between Elevations 76’-5” and 139’-6”

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-17 Structural Categories, Section A-A (Looking West)

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.3-18 Structural Categories, Section B-B (Looking North)

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.4-1 (Sheet 1 of 3)
Identification of Areas for PS/B Thermal Conditions in Table 3.8.4-2

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.4-1 (Sheet 2 of 3)
Identification of Areas for PS/B Thermal Conditions in Table 3.8.4-2

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.4-1 (Sheet 3 of 3)
Identification of Areas for PS/B Thermal Conditions in Table 3.8.4-2

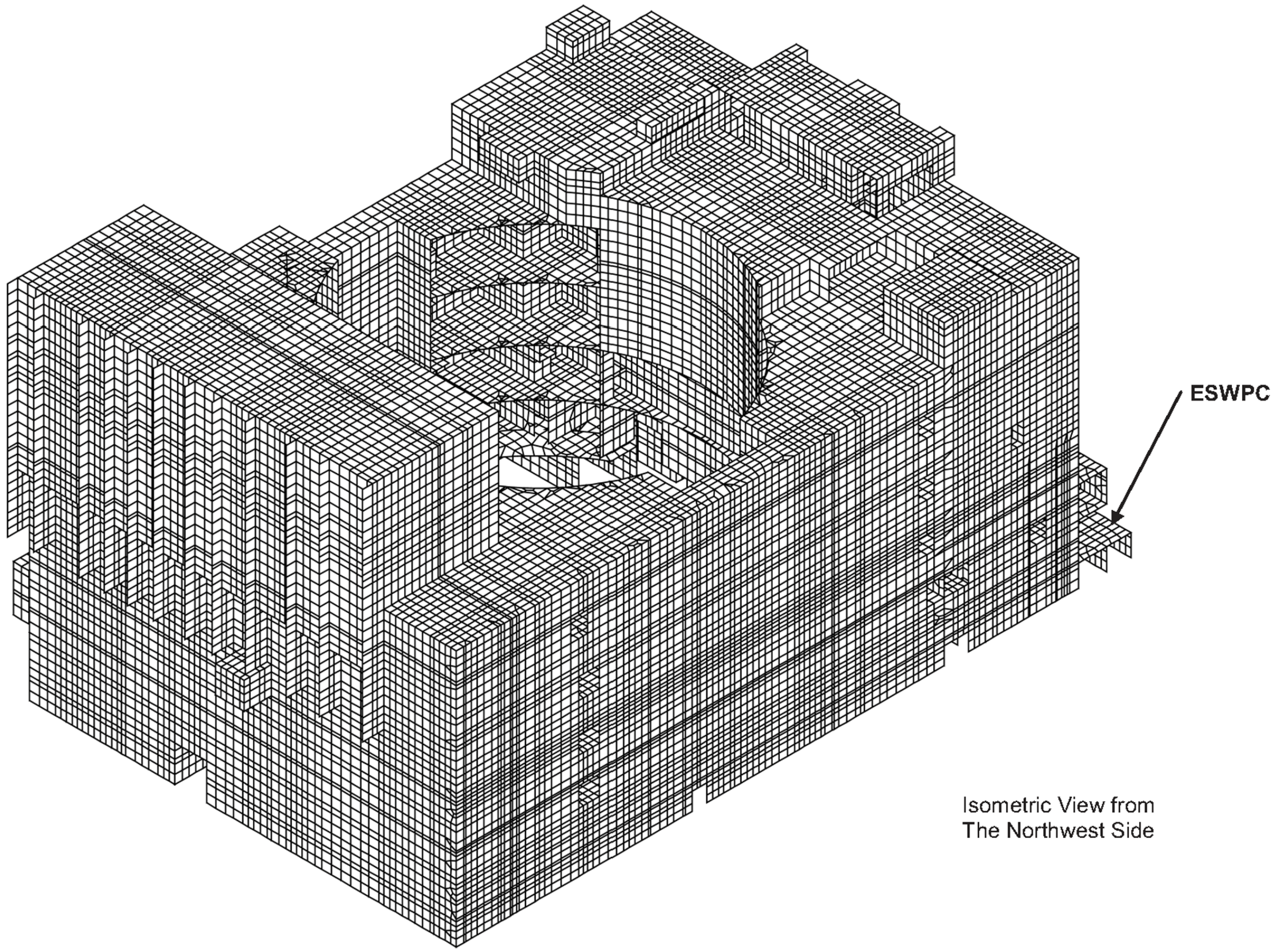


Figure 3.8.4-2 FE Model of R/B (Sheet 1 of 2)

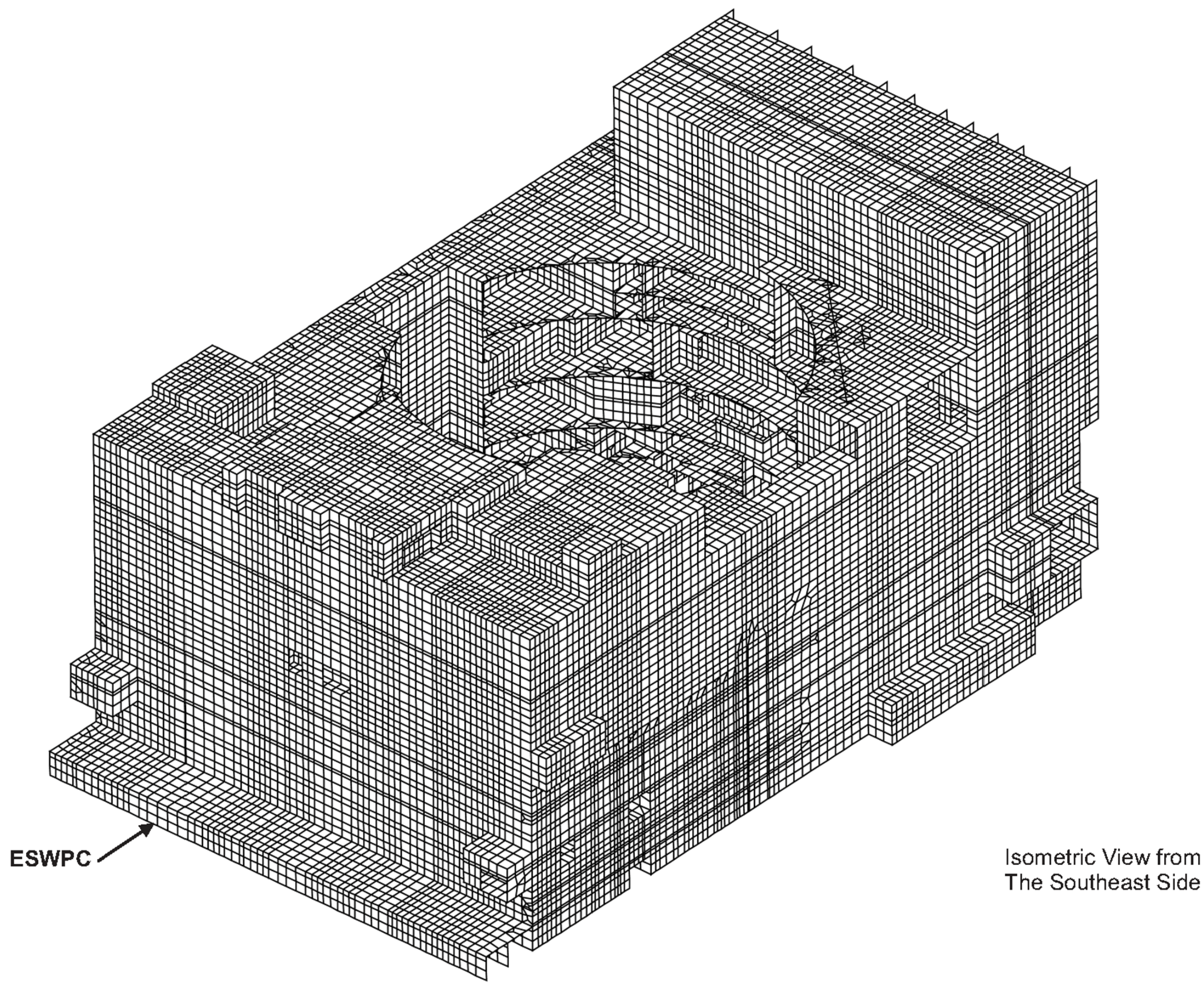
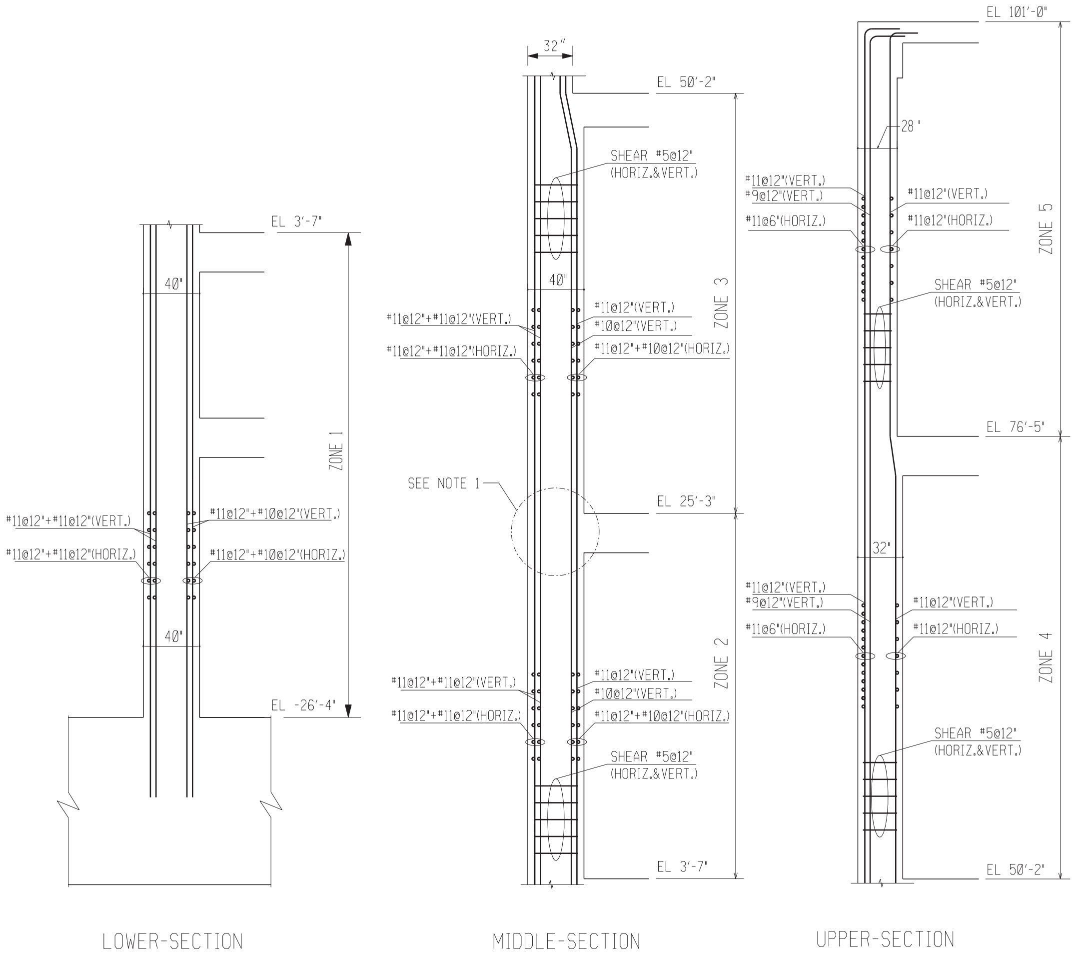


Figure 3.8.4-2 FE Model of R/B (Sheet 2 of 2)

Security-Related Information – Withheld Under 10 CFR 2.390

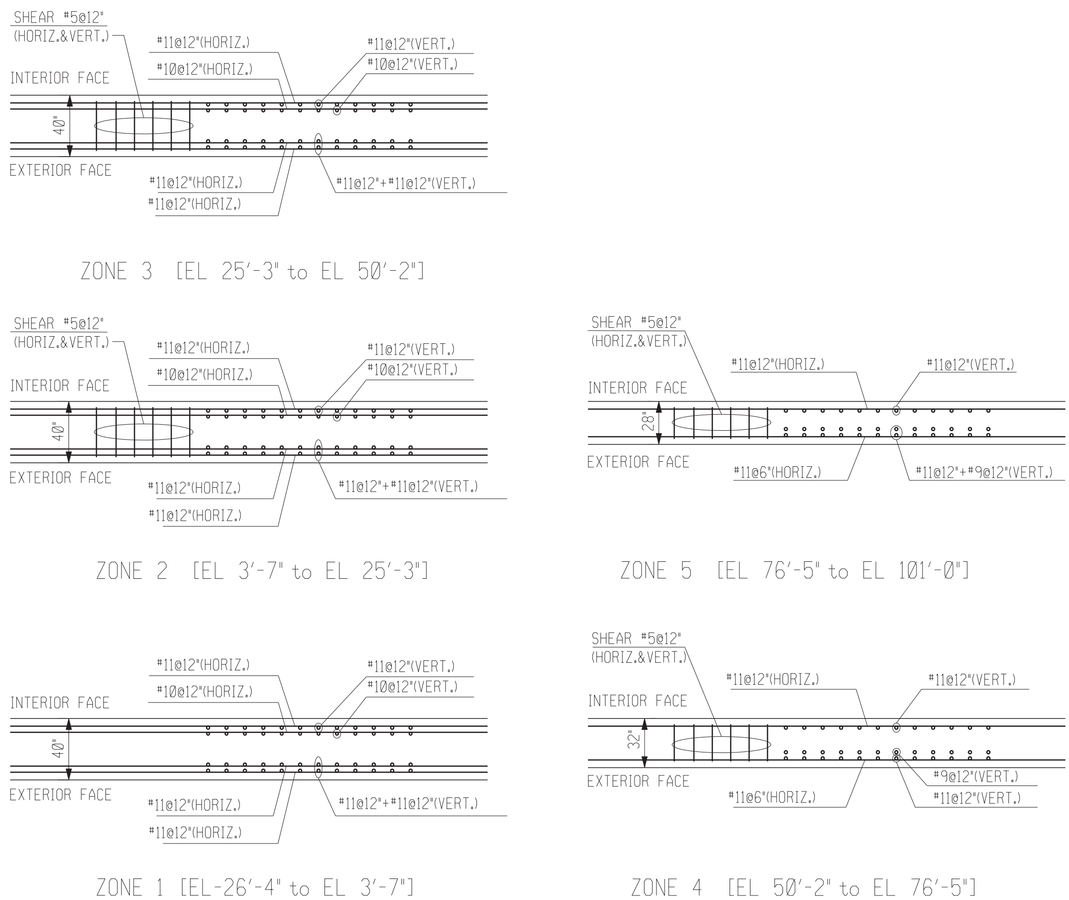
Figure 3.8.4-3 R/B Critical Sections



Vertical Cross Section

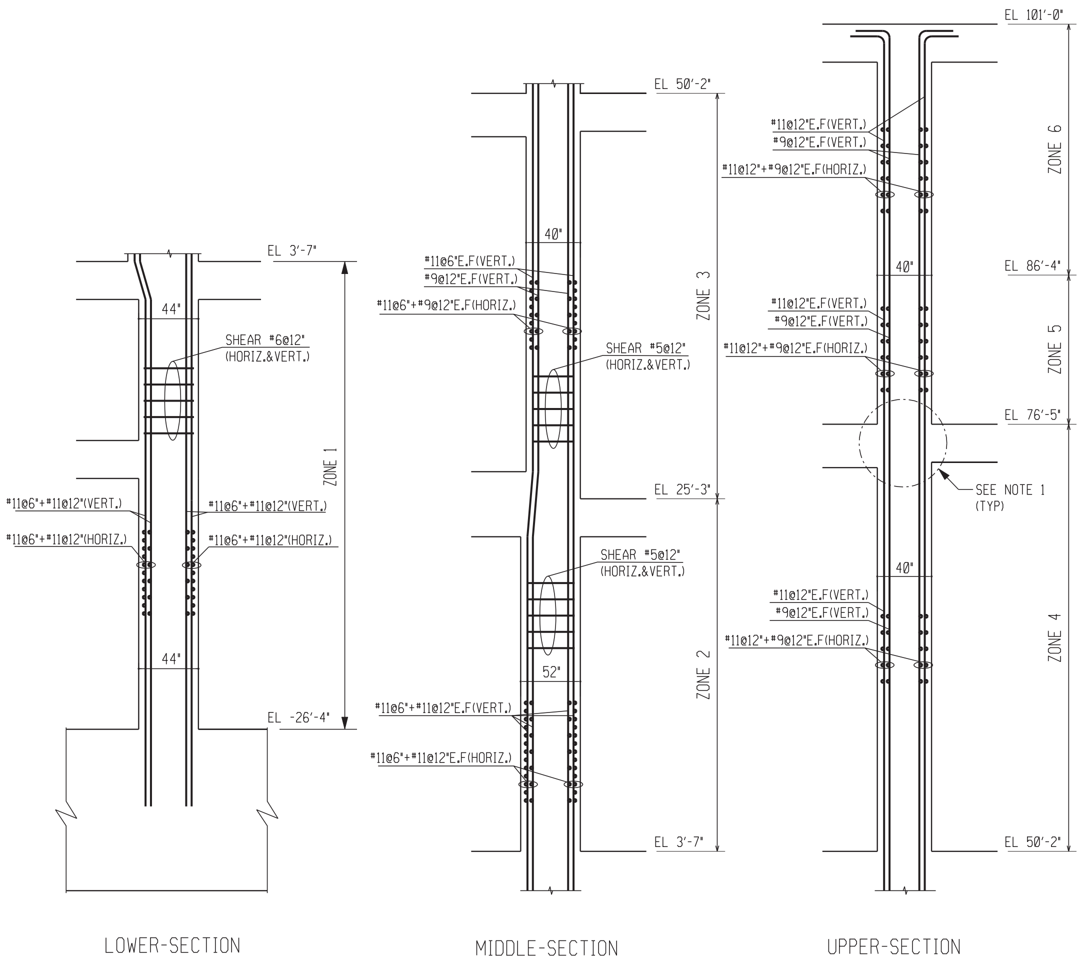
Notes:
1. Connection detail is provided in Figure 3L-45 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-4 Typical Reinforcement in R/B West Common Wall – SECTION 1 (Sheet 1 of 2)



Horizontal Cross Section

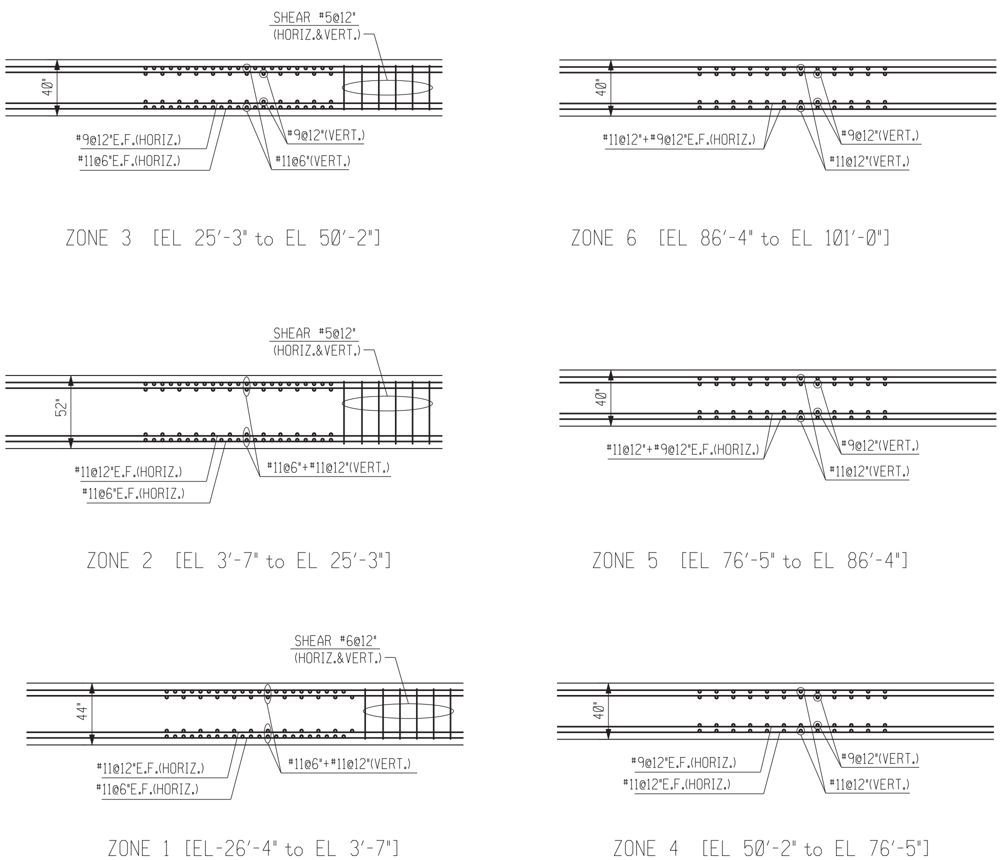
Figure 3.8.4-4 Typical Reinforcement in R/B West Common Wall – SECTION 1 (Sheet 2 of 2)



Vertical Cross Section

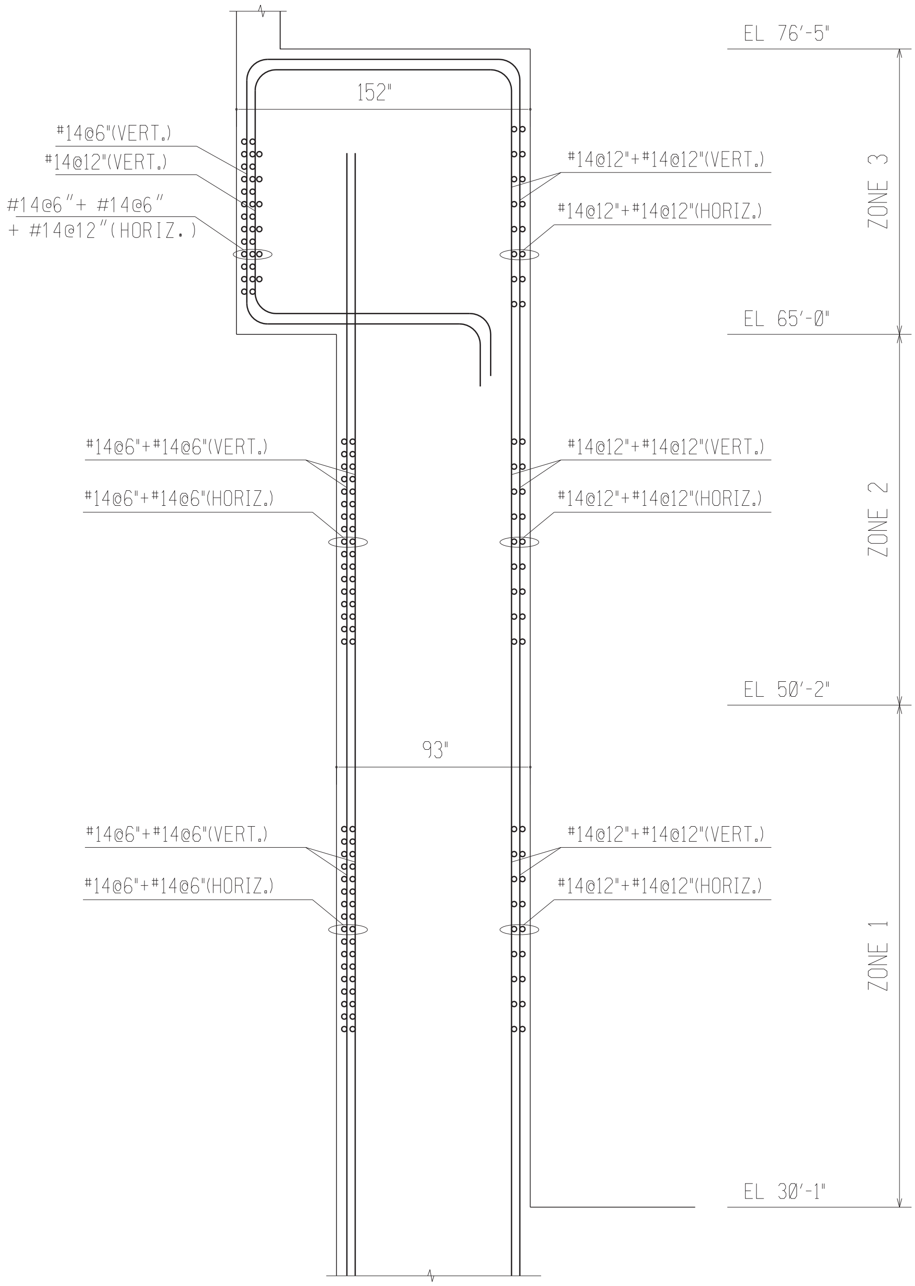
Notes:
1. Connection detail is provided in Figure 3L-47 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-5 Typical Reinforcement in R/B South Interior Wall – SECTION 2 (Sheet 1 of 2)



Horizontal Cross Section

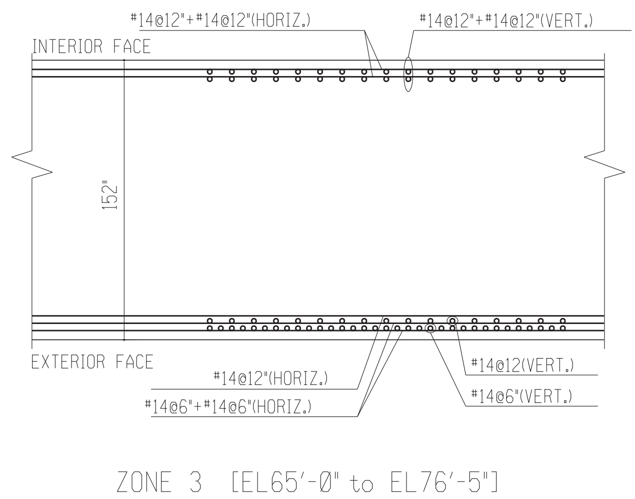
Figure 3.8.4-5 Typical Reinforcement in R/B South Interior Wall – SECTION 2 (Sheet 2 of 2)

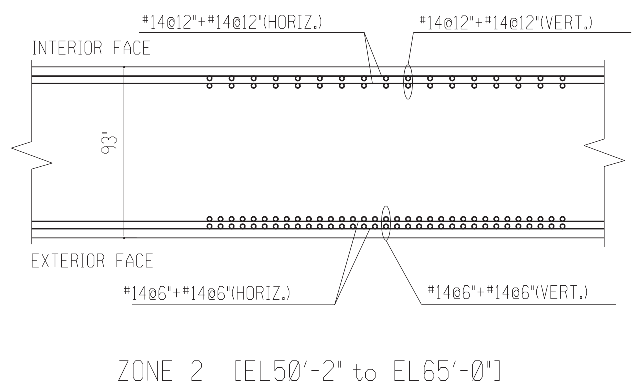


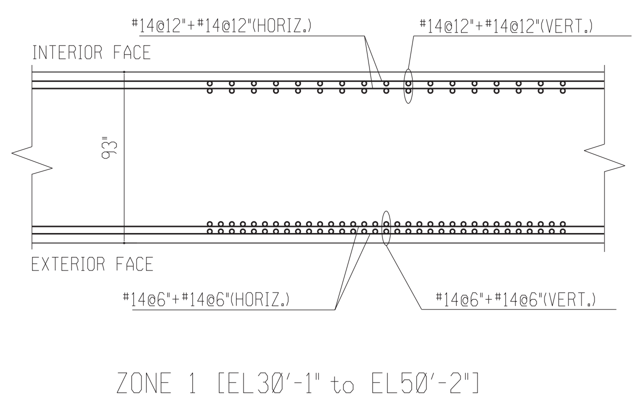
Vertical Cross Section

Notes:
1. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-6 Typical Reinforcement in North Exterior Wall of Spent Fuel Pit – SECTION 3 (Sheet 1 of 2)

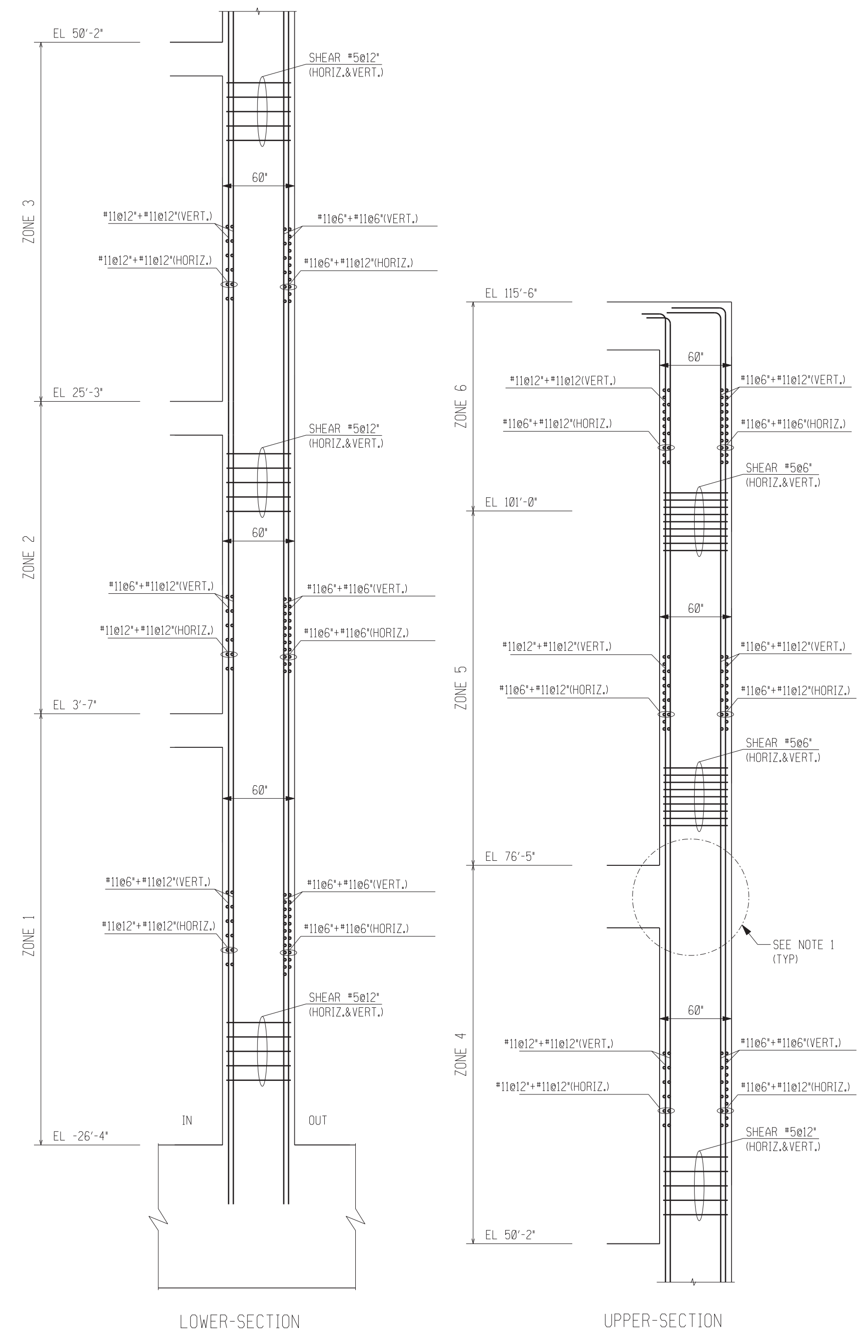






Horizontal Cross Section

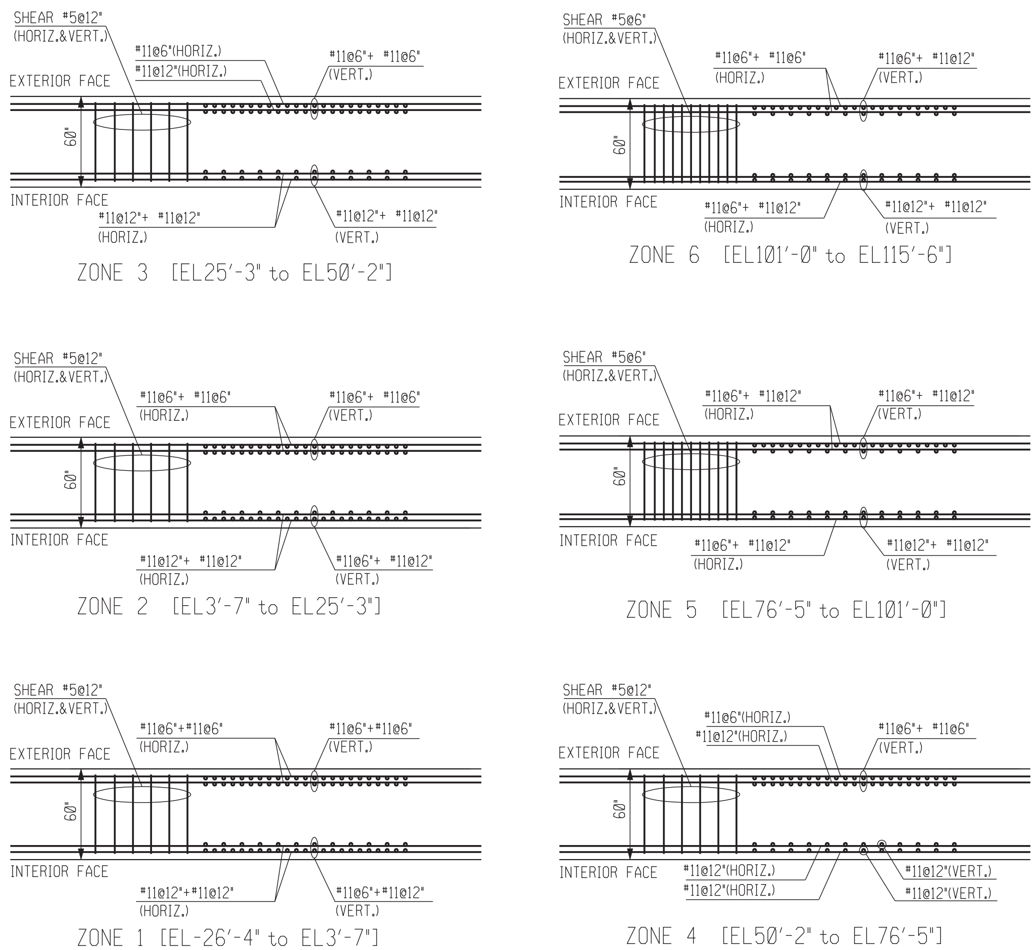
Figure 3.8.4-6 Typical Reinforcement in North Exterior Wall of Spent Fuel Pit – SECTION 3 (Sheet 2 of 2)



Vertical Cross Section

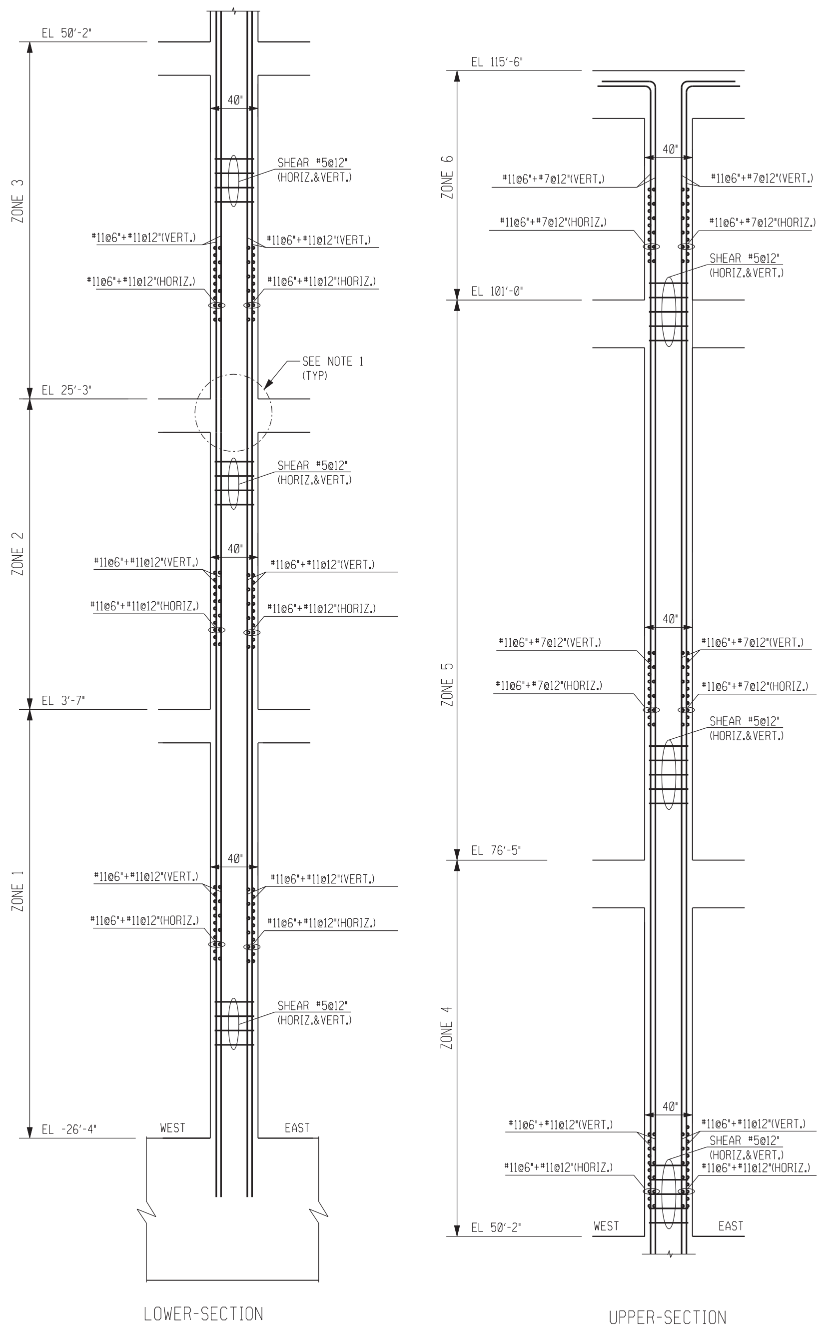
Notes:
1. Connection detail is provided in Figure 3L-45 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-7 Typical Reinforcement in South Exterior Wall – SECTION 4 (Sheet 1 of 2)



<u>**Horizontal Cross Section**</u>

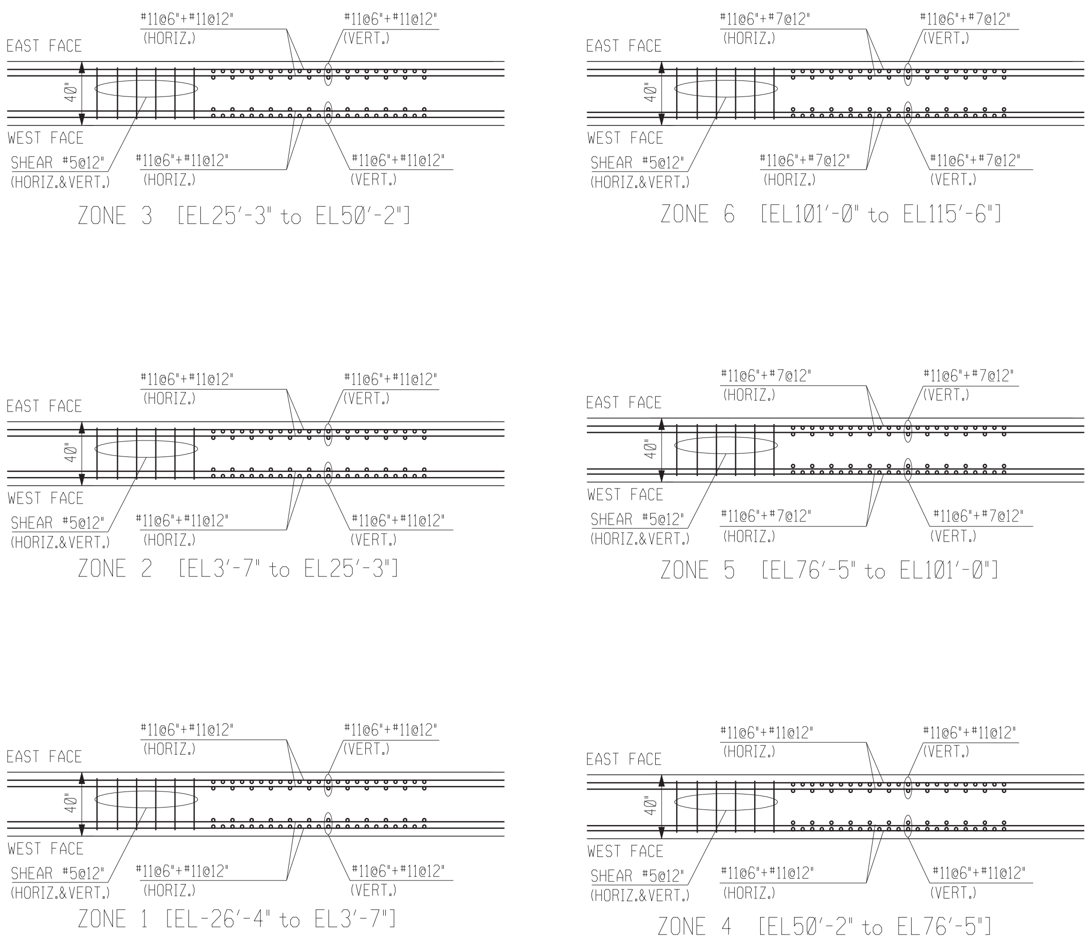
Figure 3.8.4-7 Typical Reinforcement in South Exterior Wall – SECTION 4 (Sheet 2 of 2)



Vertical Cross Section

Notes:
1. Connection detail is provided in Figure 3L-47 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-8 Typical Reinforcement in R/B Central Interior Wall – SECTION 5 (Sheet 1 of 2)



Horizontal Cross Section

Figure 3.8.4-8 Typical Reinforcement in R/B Central Interior Wall – SECTION 5 (Sheet 2 of 2)

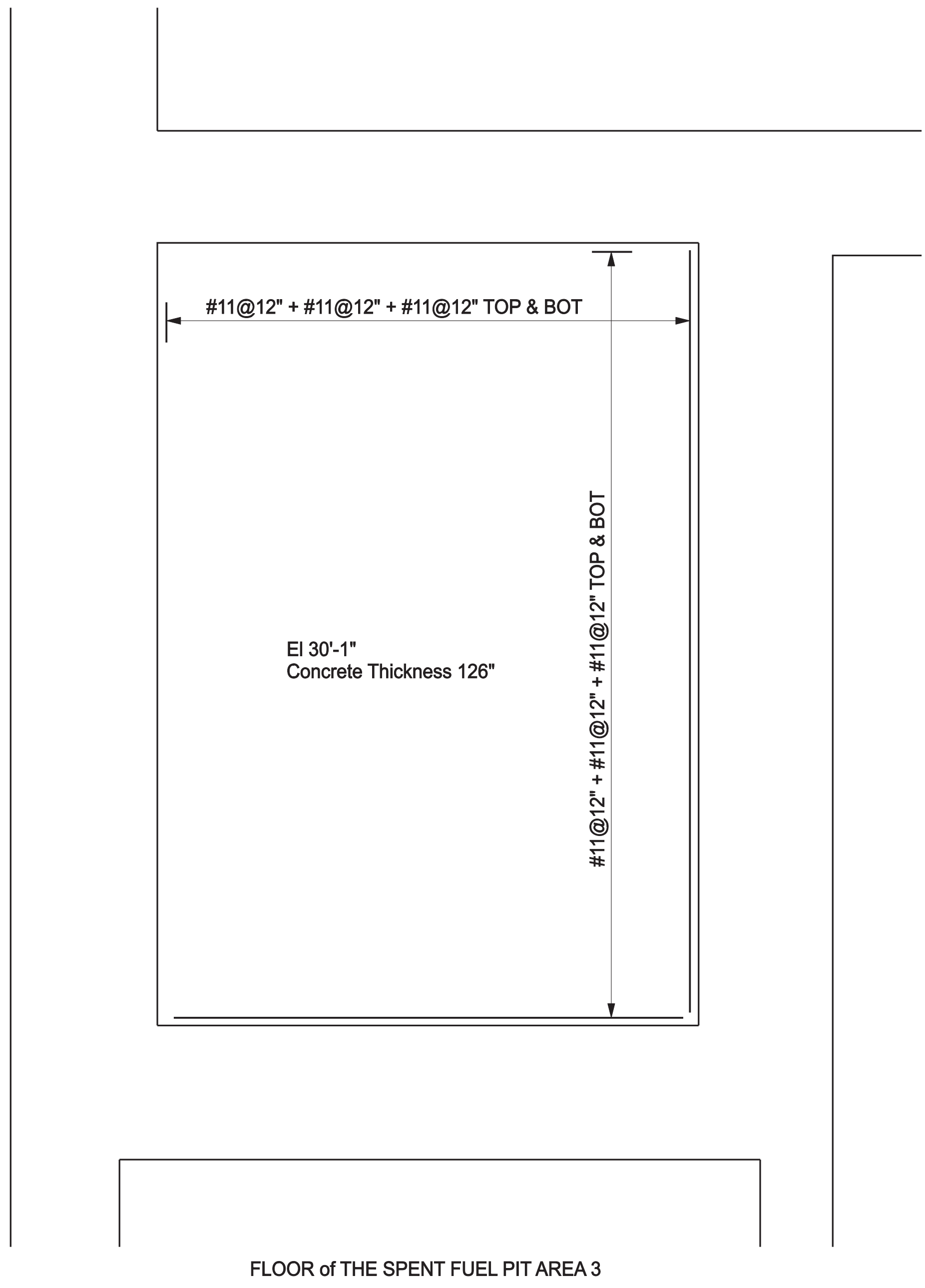


Figure 3.8.4-9 Typical Reinforcement in Spent Fuel Pit Slab – AREA 3

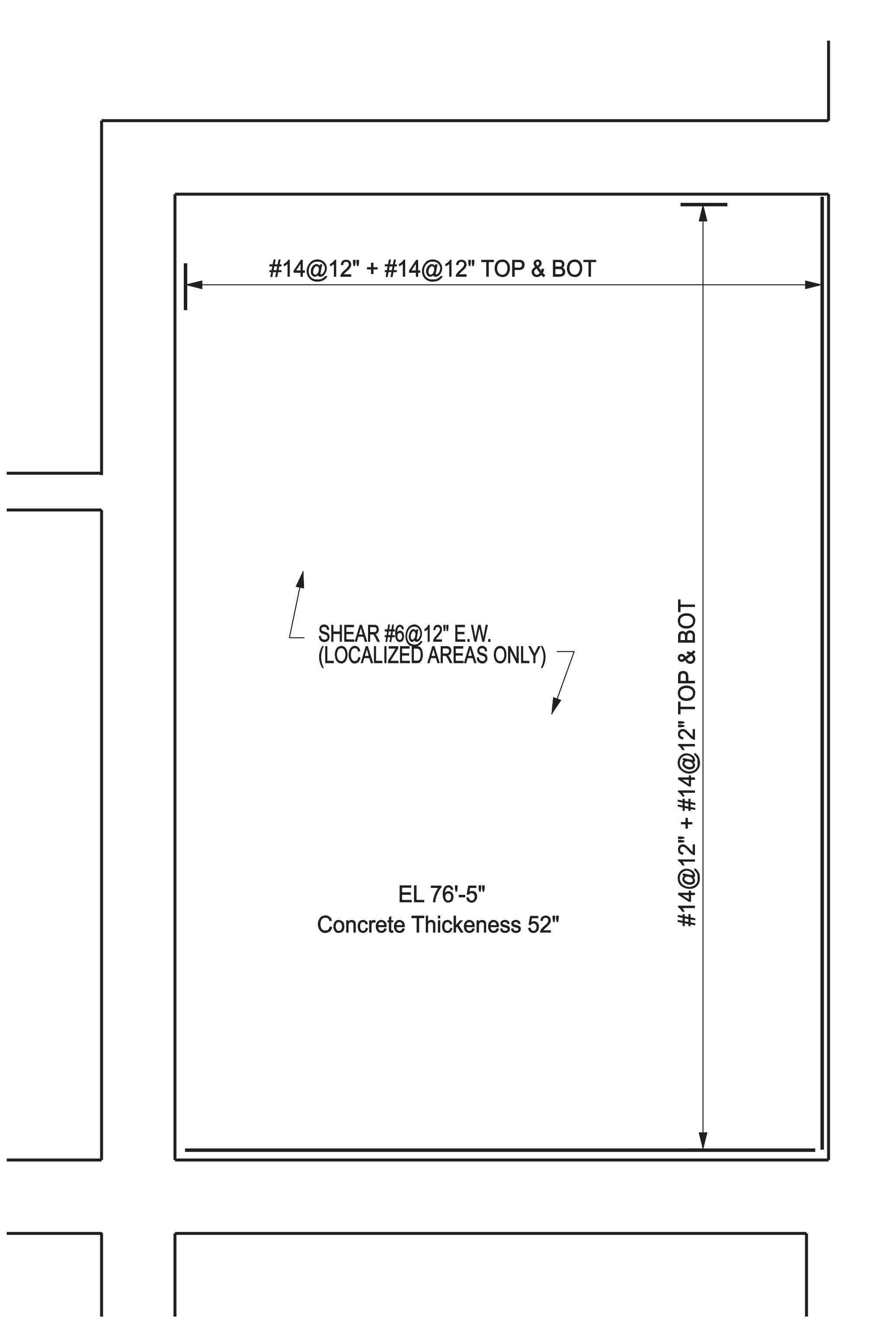


Figure 3.8.4-10 Typical Reinforcement in Emergency Feedwater Pit Slab – AREA 4

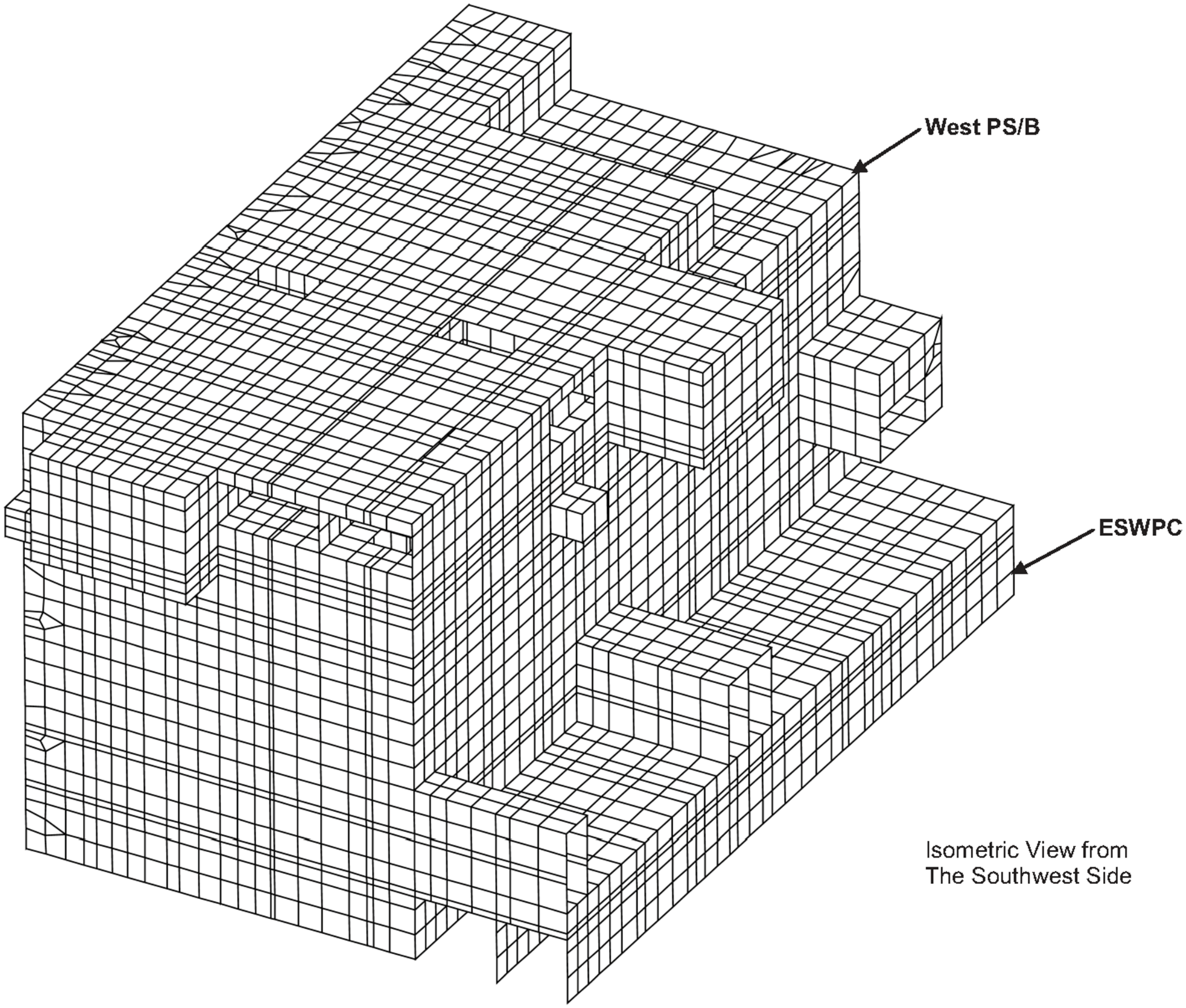


Figure 3.8.4-11 FE Model of West PS/B (Sheet 1 of 4)

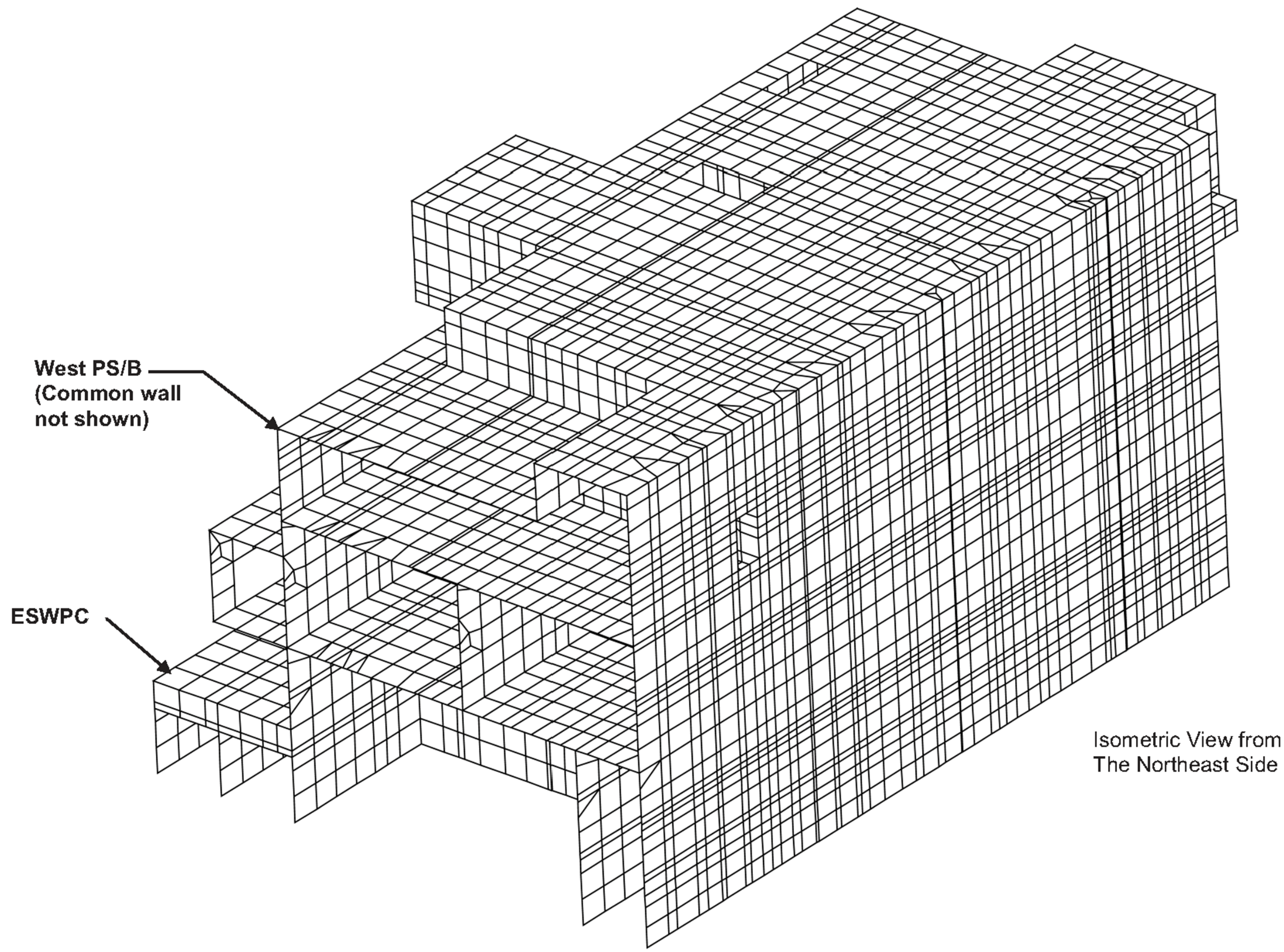


Figure 3.8.4-11 FE Model of West PS/B (Sheet 2 of 4)

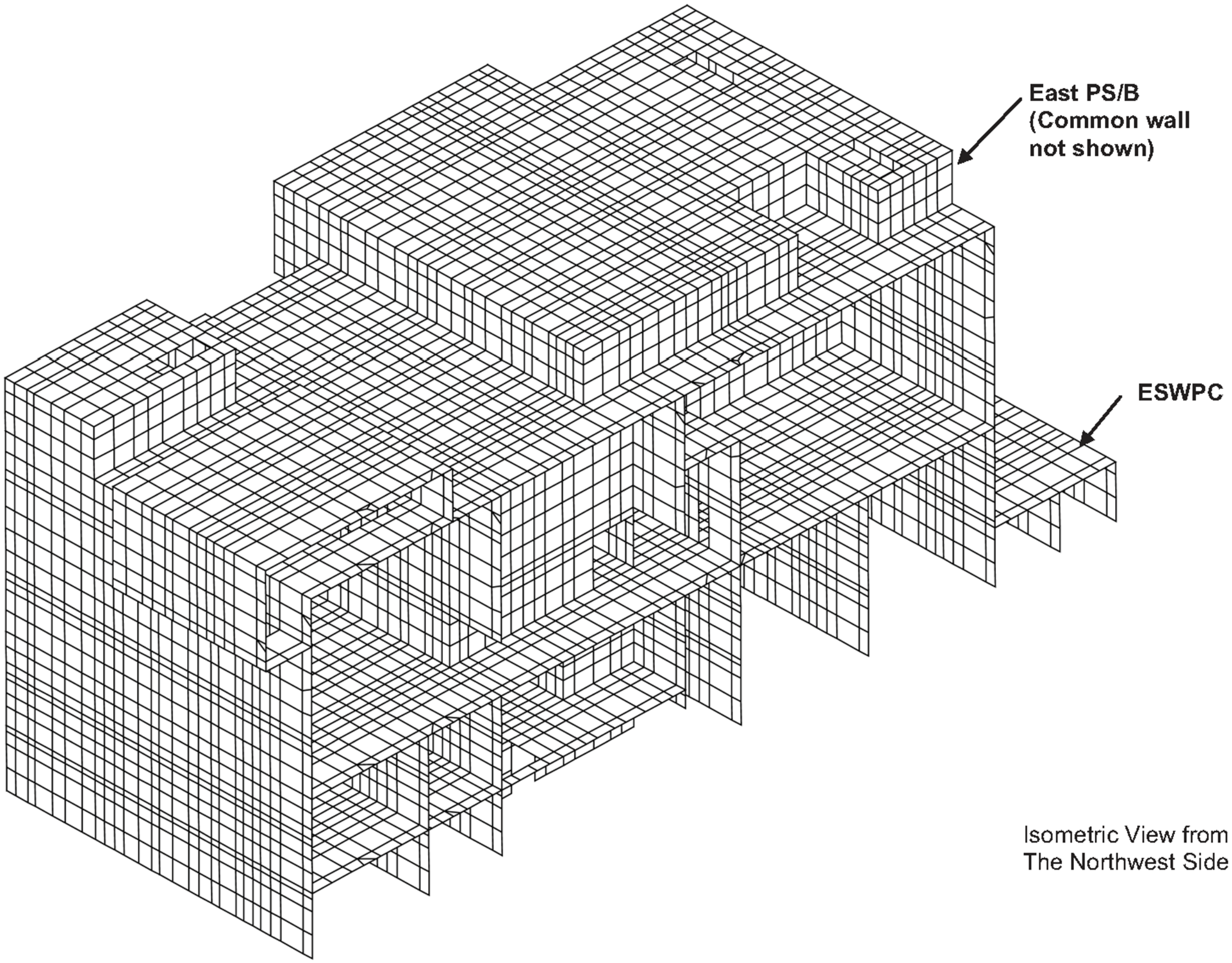


Figure 3.8.4-11 FE Model of West PS/B (Sheet 3 of 4)

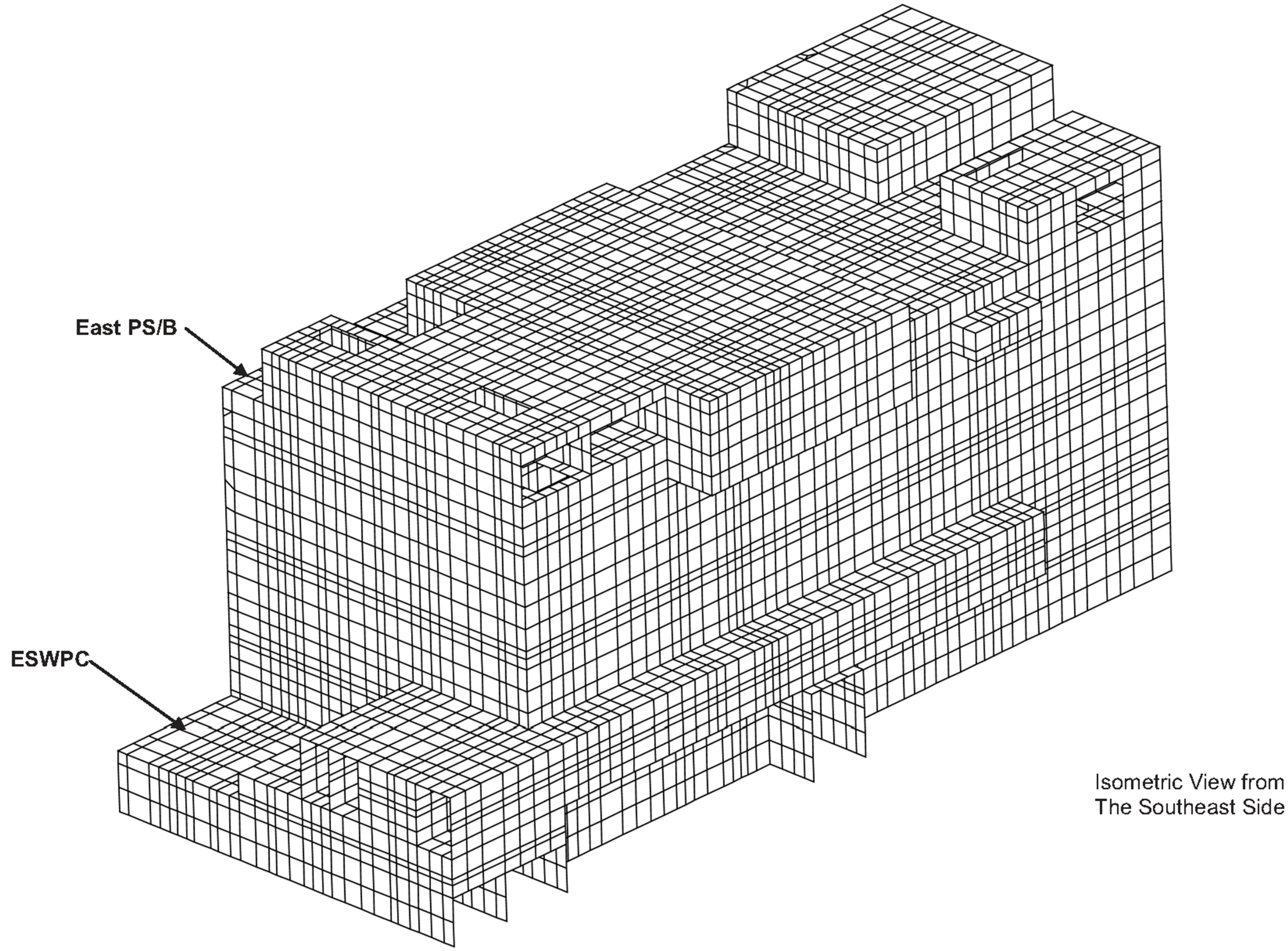


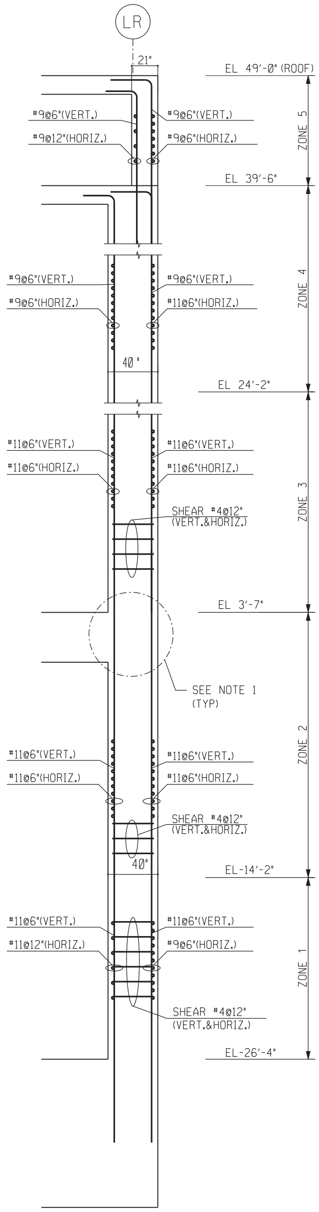
Figure 3.8.4-11 FE Model of West PS/B (Sheet 4 of 4)

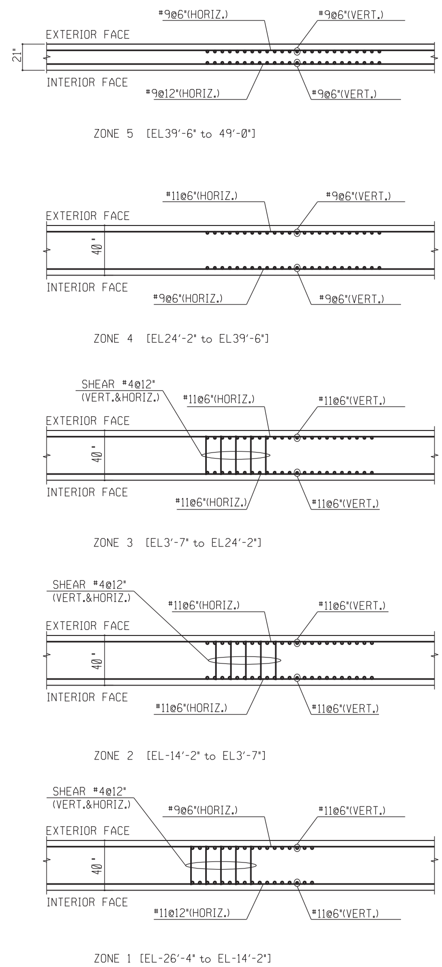
Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.4-12 West PS/B Wall Critical Sections (Floor Plan of B1F, EL-26'-4")

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.4-13 West PS/B Wall and Slab Critical Sections (Floor Plan of 1F, EL 3'-7")



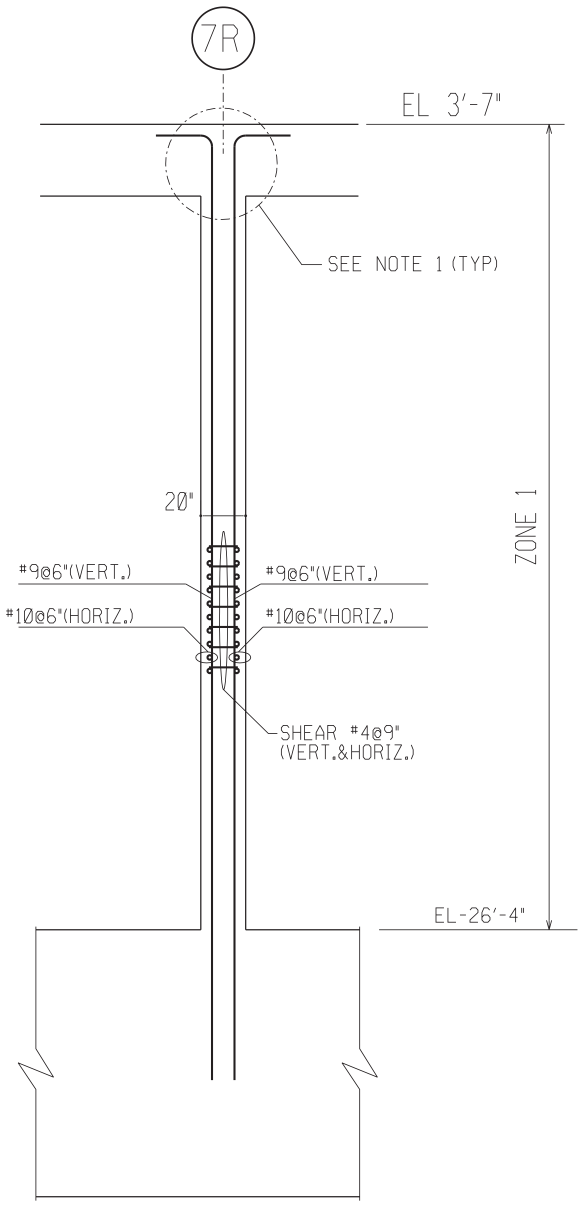


Horizontal Cross Section

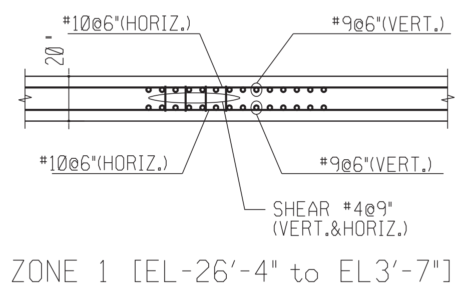
Vertical Cross Section

Notes:
1. Connection detail is provided in Figure 3L-45 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-14 Typical Reinforcement in West PS/B South Exterior Wall – SECTION 1 (On Column Line LR and Between Column Lines 1R & 3R)



Vertical Cross Section



Horizontal Cross Section

Notes:
1. Connection detail is provided in Figure 3L-48 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-15 Typical Reinforcement in West PS/B Interior Wall – SECTION 2 (On Column Line 7R and Between Column Lines JR & LR)

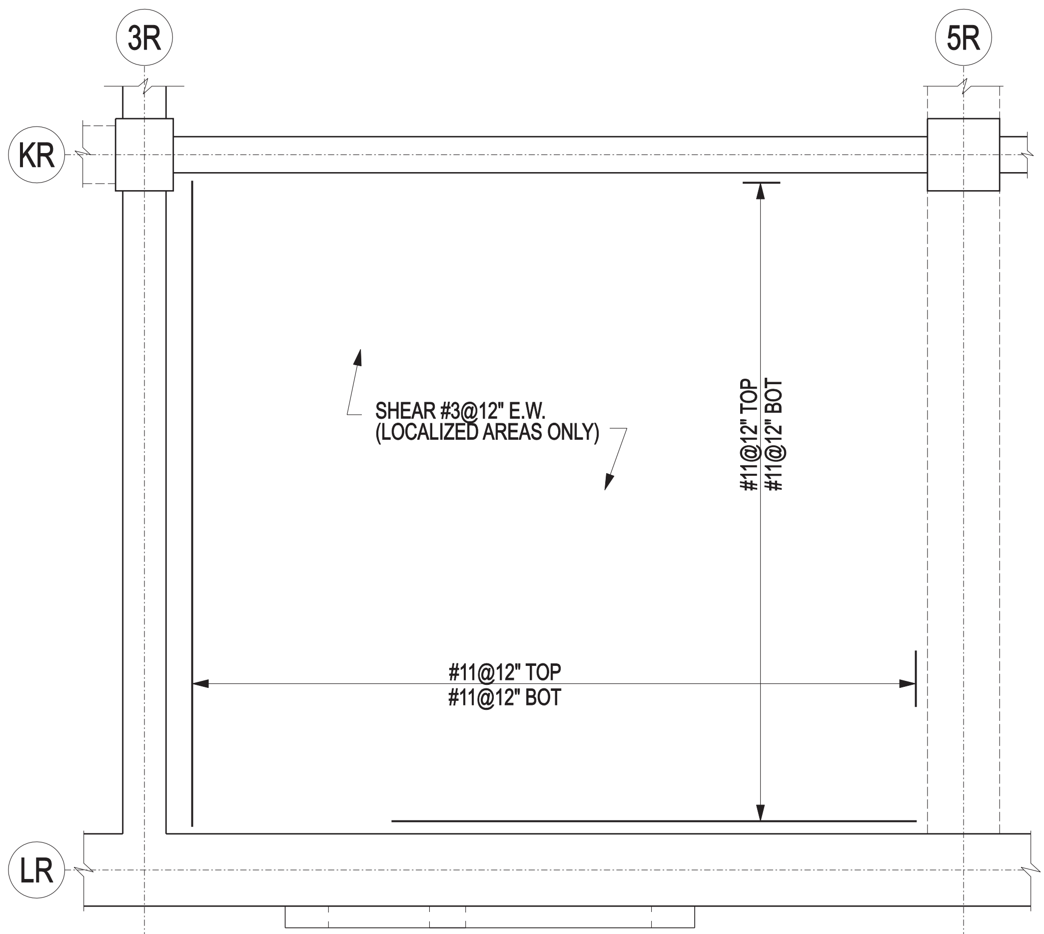
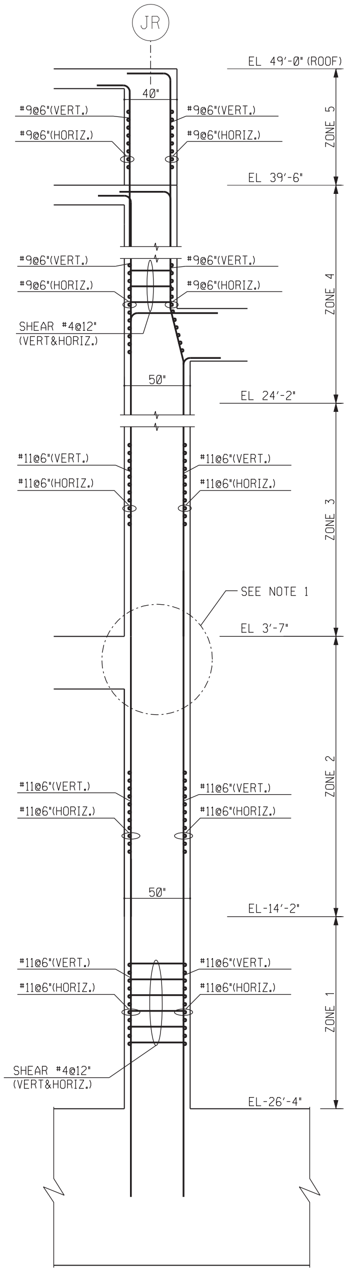
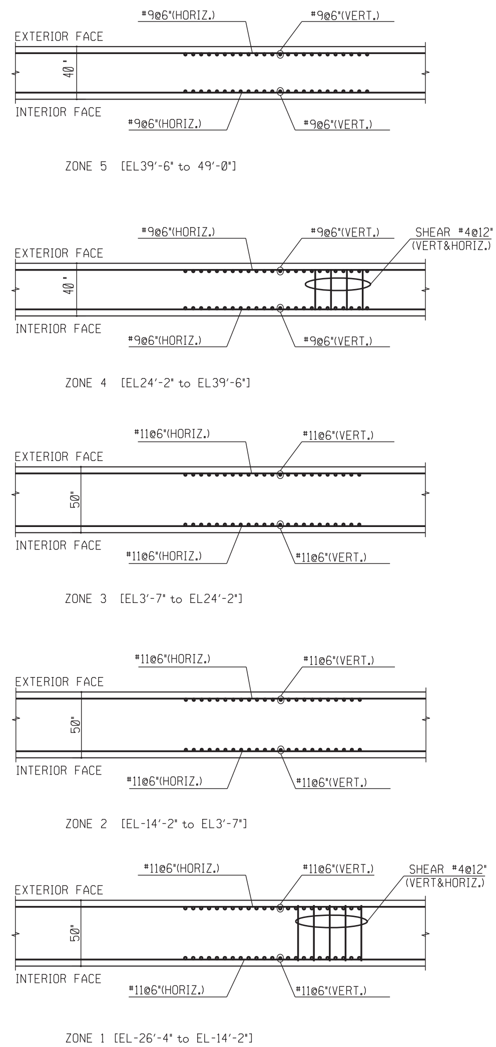


Figure 3.8.4-16 Typical Reinforcement in West PS/B Floor at Elevation 3′-7″– AREA 1 (Between Column Lines KR & LR - 3R & 5R)



Vertical Cross Section



Horizontal Cross Section

Notes:
1. Connection detail is provided in Figure 3L-45 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

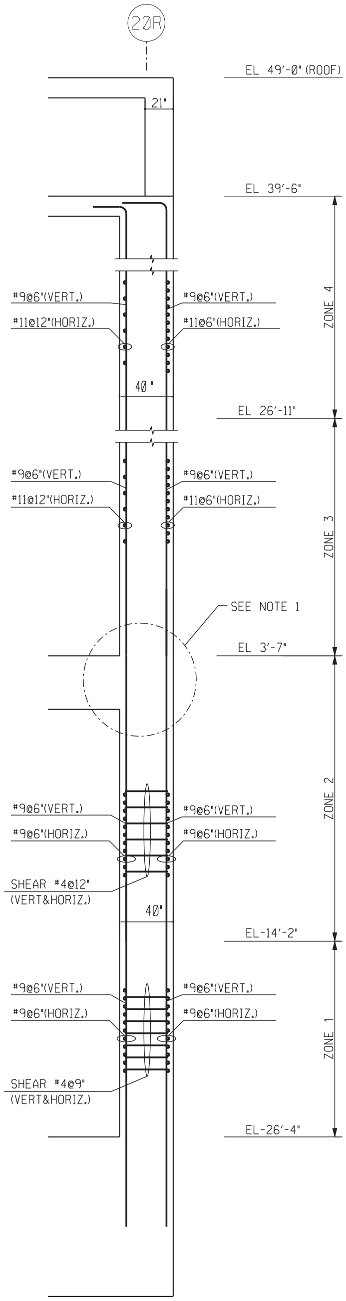
Figure 3.8.4-17 Typical Reinforcement in West PS/B North Wall – SECTION 3 (On Column Line JR and Between Column Lines 1R & 3R)

Security-Related Information – Withheld Under 10 CFR 2.390

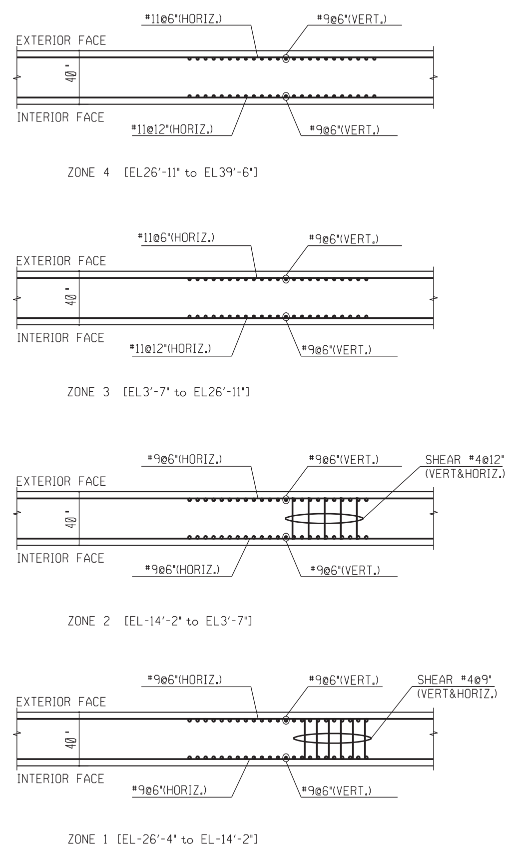
Figure 3.8.4-18 East PS/B Wall Critical Sections
(Floor Plan of B1F, El -26'-4")

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.4-19 East PS/B Wall and Slab Critical Sections (Floor Plan of 1F, EL 3'-7")



Vertical Cross Section

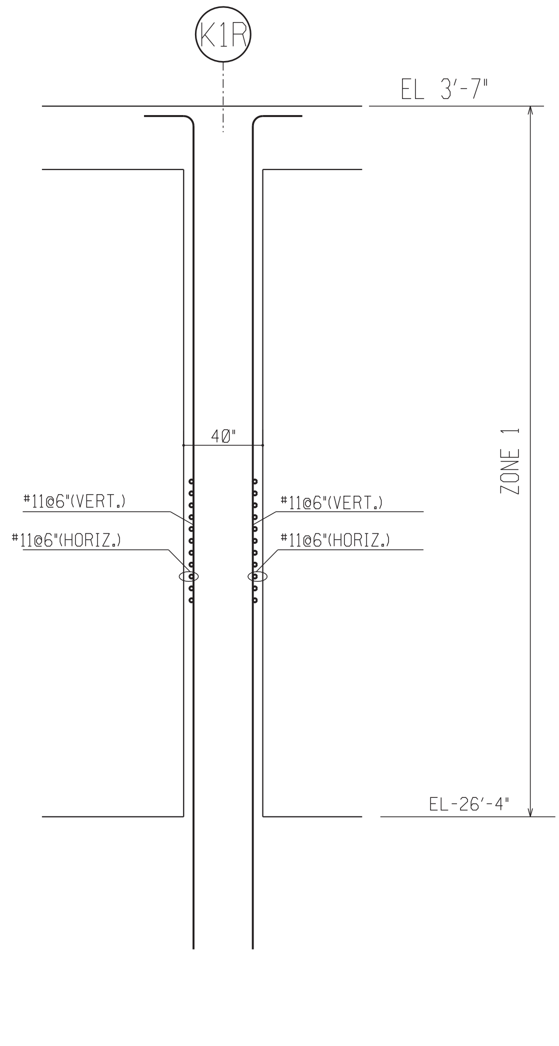


Horizontal Cross Section

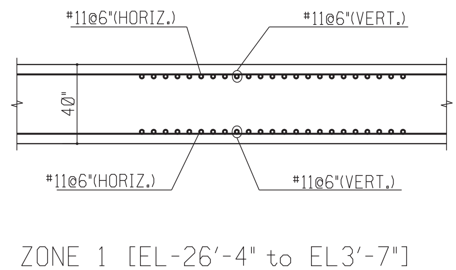
Notes:

1. Connection detail is provided in Figure 3L-45 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-20 Typical Reinforcement in East PS/B East Exterior Wall – SECTION 1 (On Column Line 20R and Between Column Lines F1R & G4R)



Vertical Cross Section



Horizontal Cross Section

Note: Final reinforcement configuration details may vary and are subject to changes for fabrication and or constructability.

Figure 3.8.4-21 Typical Reinforcement in East PS/B Interior Wall – SECTION 2 (On Column Line K1R and Between Column Lines 18R & 20R)

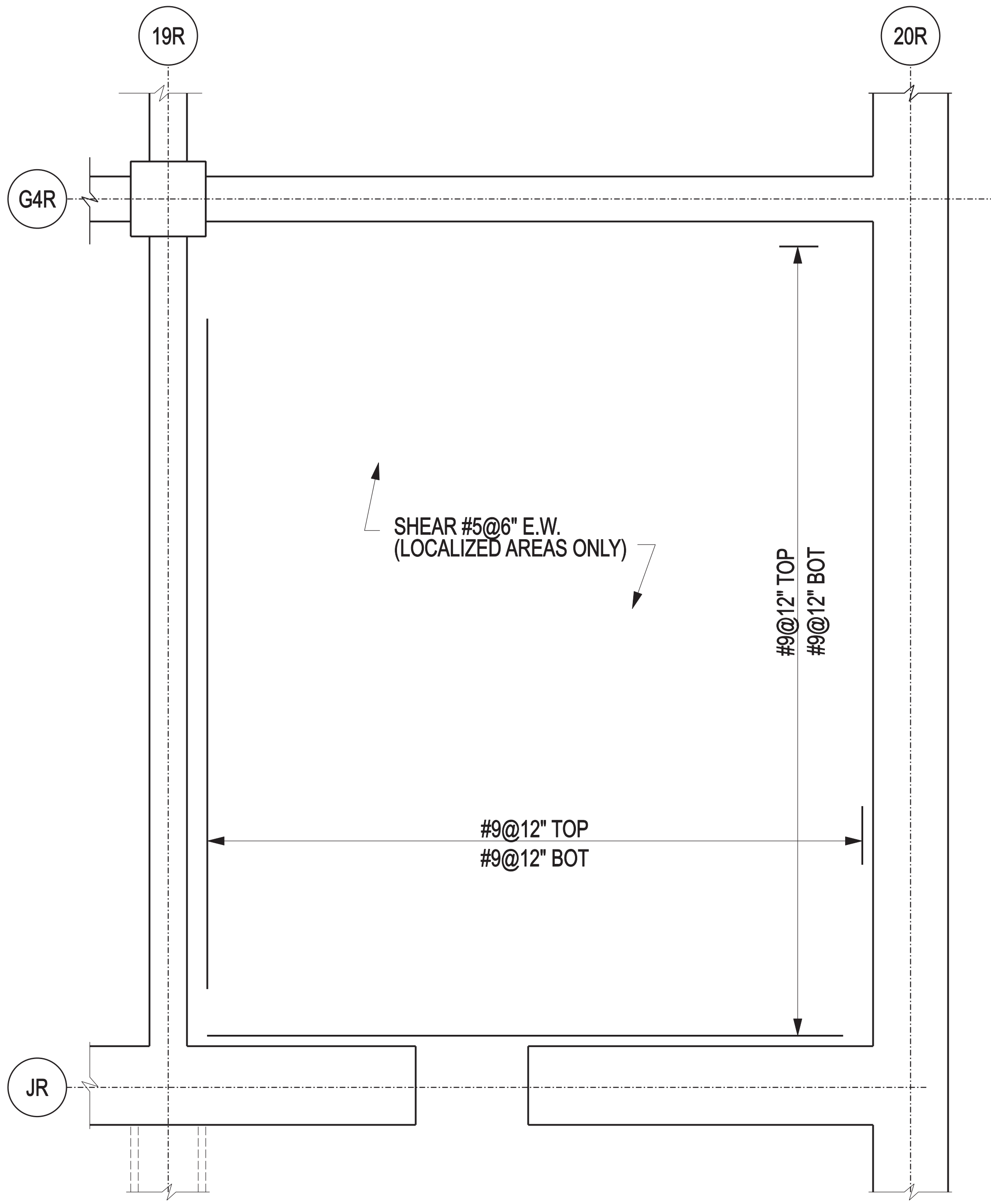


Figure 3.8.4-22 Typical Reinforcement in East PS/B Floor at Elevation 3′-7″– AREA 1 (Between Column Lines 19R & 20R – G4R & JR)

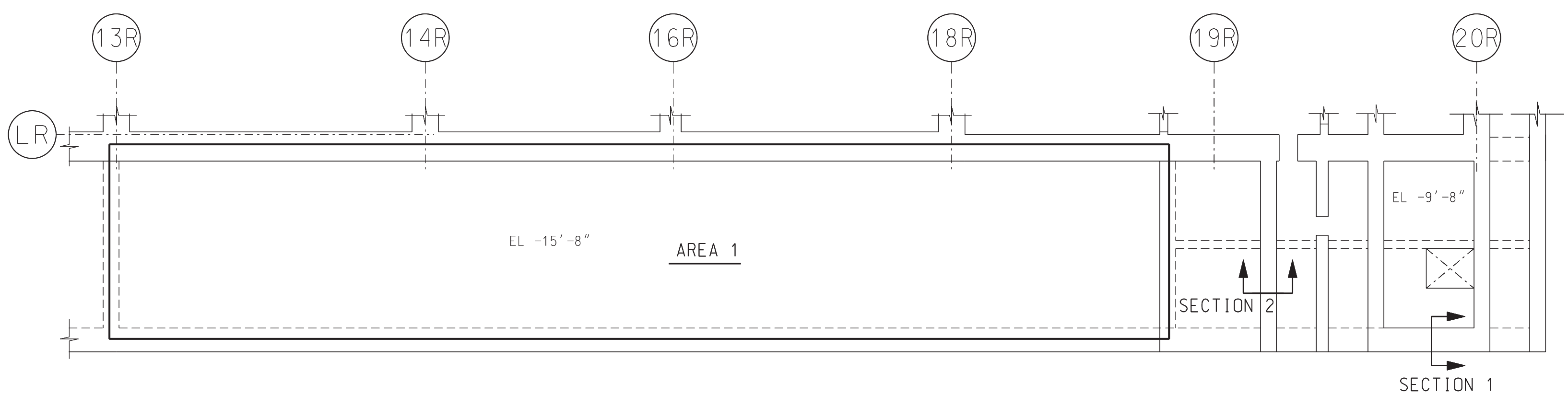
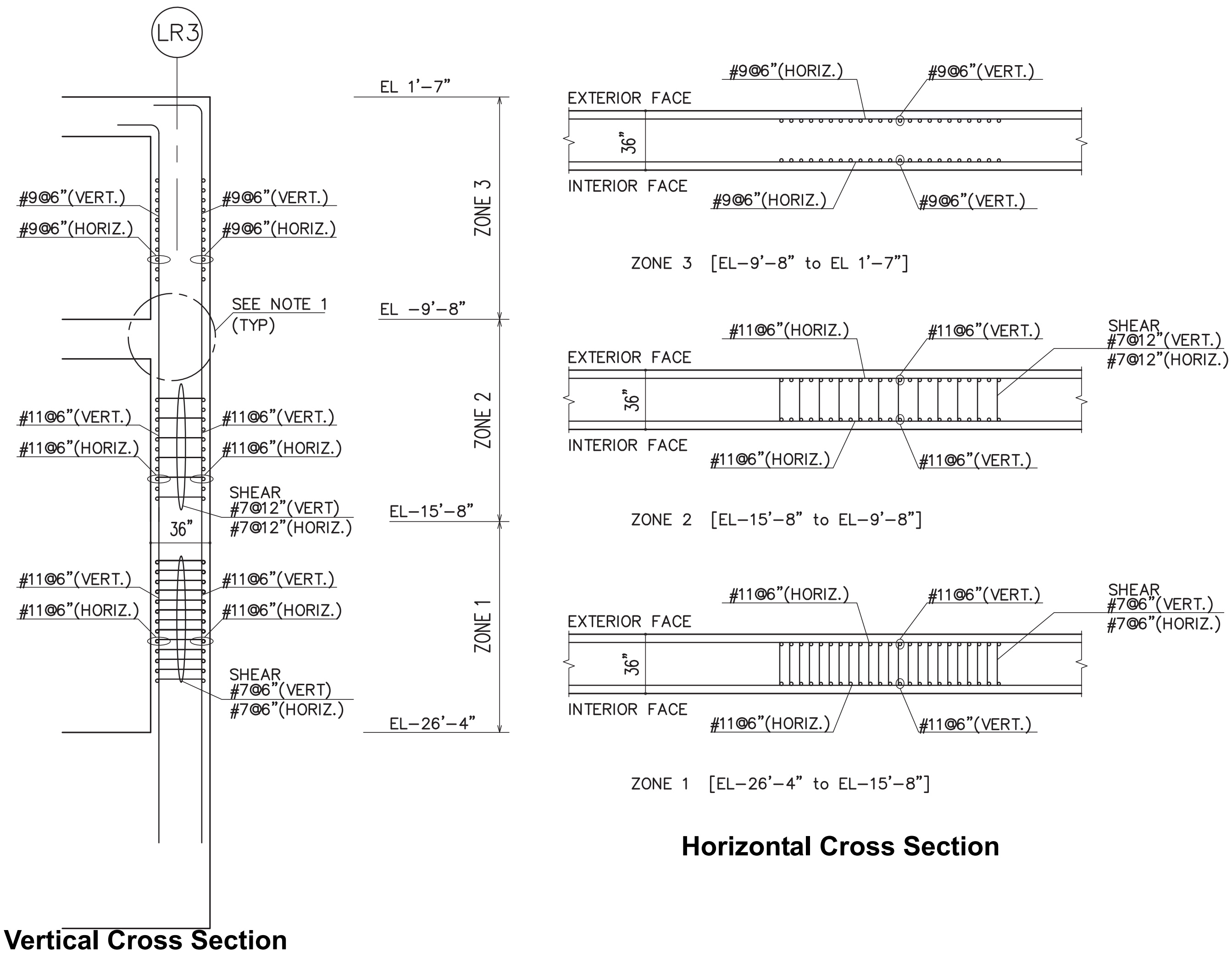
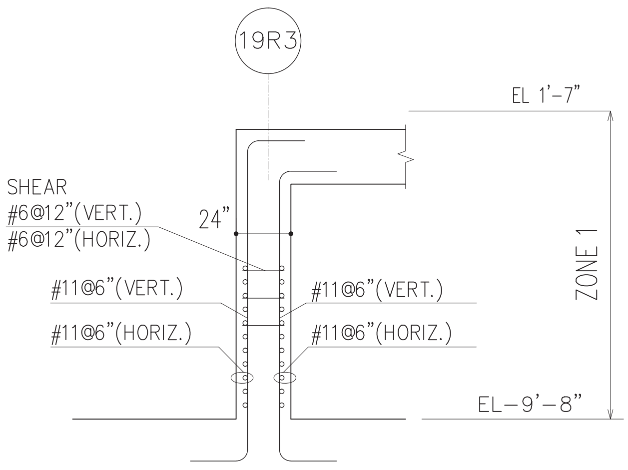


Figure 3.8.4-23 ESWPC Critical Sections (Floor Plan of East ESWPC, EL -15'-8")

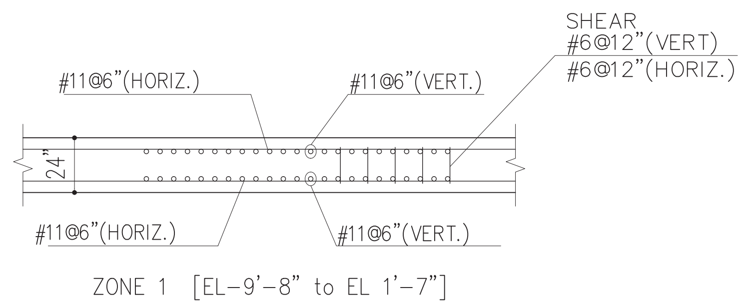


Notes:
1. Connection detail is provided in Figure 3L-45 in Appendix 3L.
2. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-24 Typical Reinforcement in ESWPC Exterior Wall– SECTION 1 (On Column Line LR3 and Between Column Lines 20R & 19R8)



Vertical Cross Section



Horizontal Cross Section

Notes:
1. Final reinforcement configuration details may vary and are subject to changes necessary for fabrication and/or constructability.

Figure 3.8.4-25 Typical Reinforcement in ESWPC Interior Wall – SECTION 2 (On Column Line 19R3 and Between Column Lines LR & LR3)

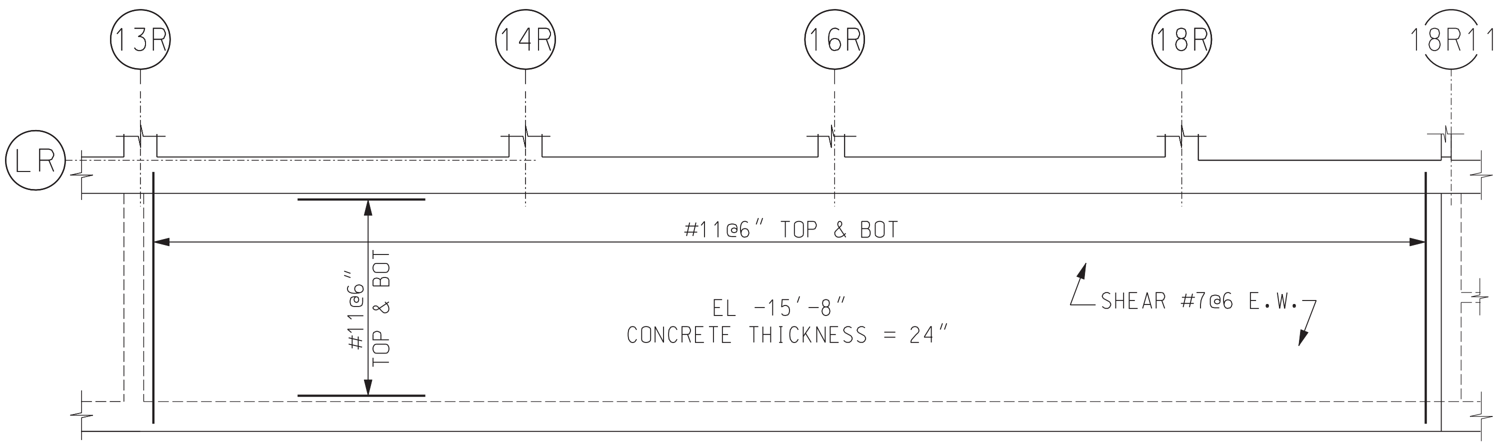


Figure 3.8.4-26 Typical Reinforcement in Slab at Elevation -15′-8″– AREA 1 (Between Column Lines LR & LR3 – 13R & 18R11)

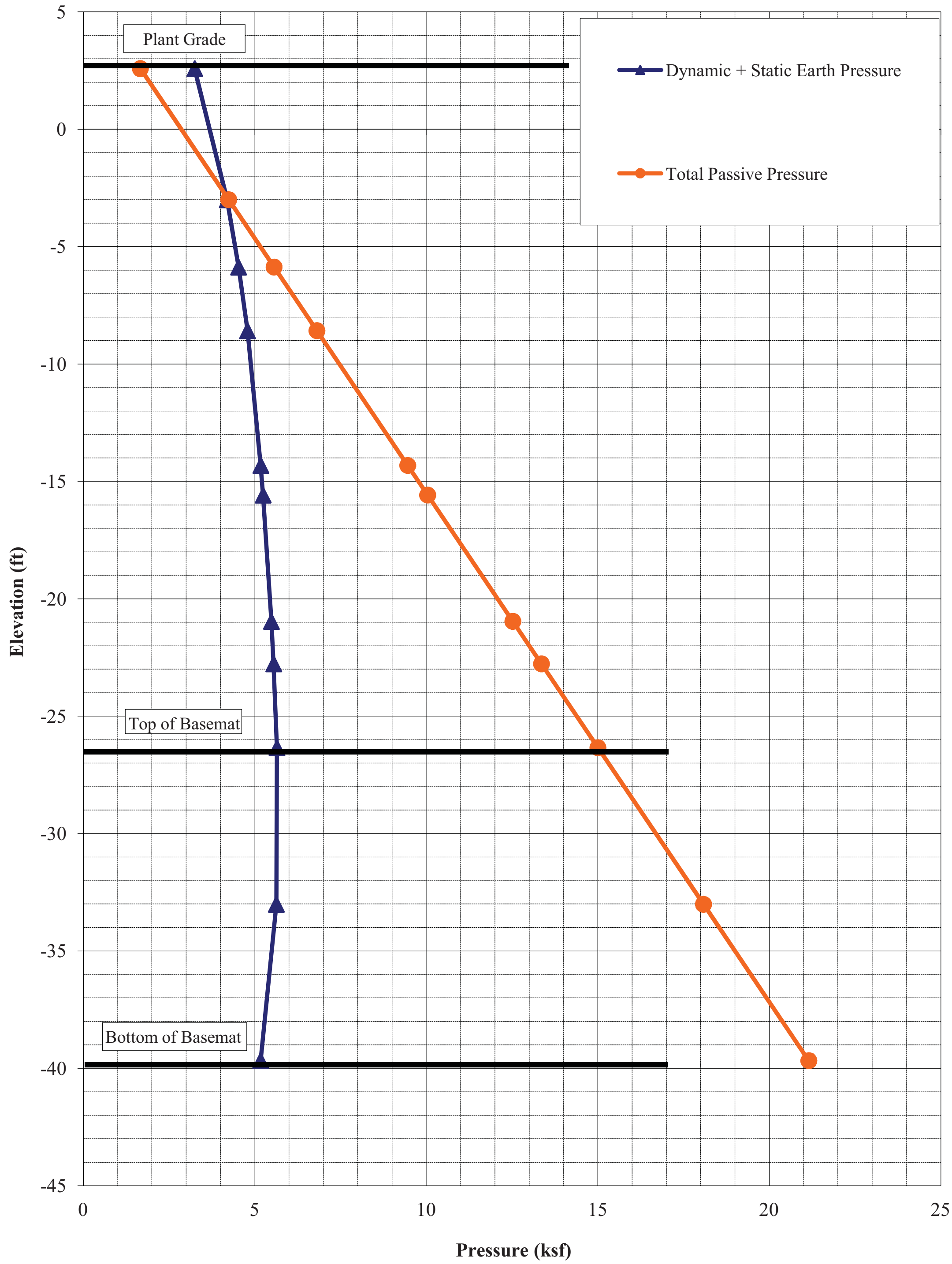


Figure 3.8.4-27 Dynamic & Static Lateral Earth Pressure vs. Passive Earth Pressure

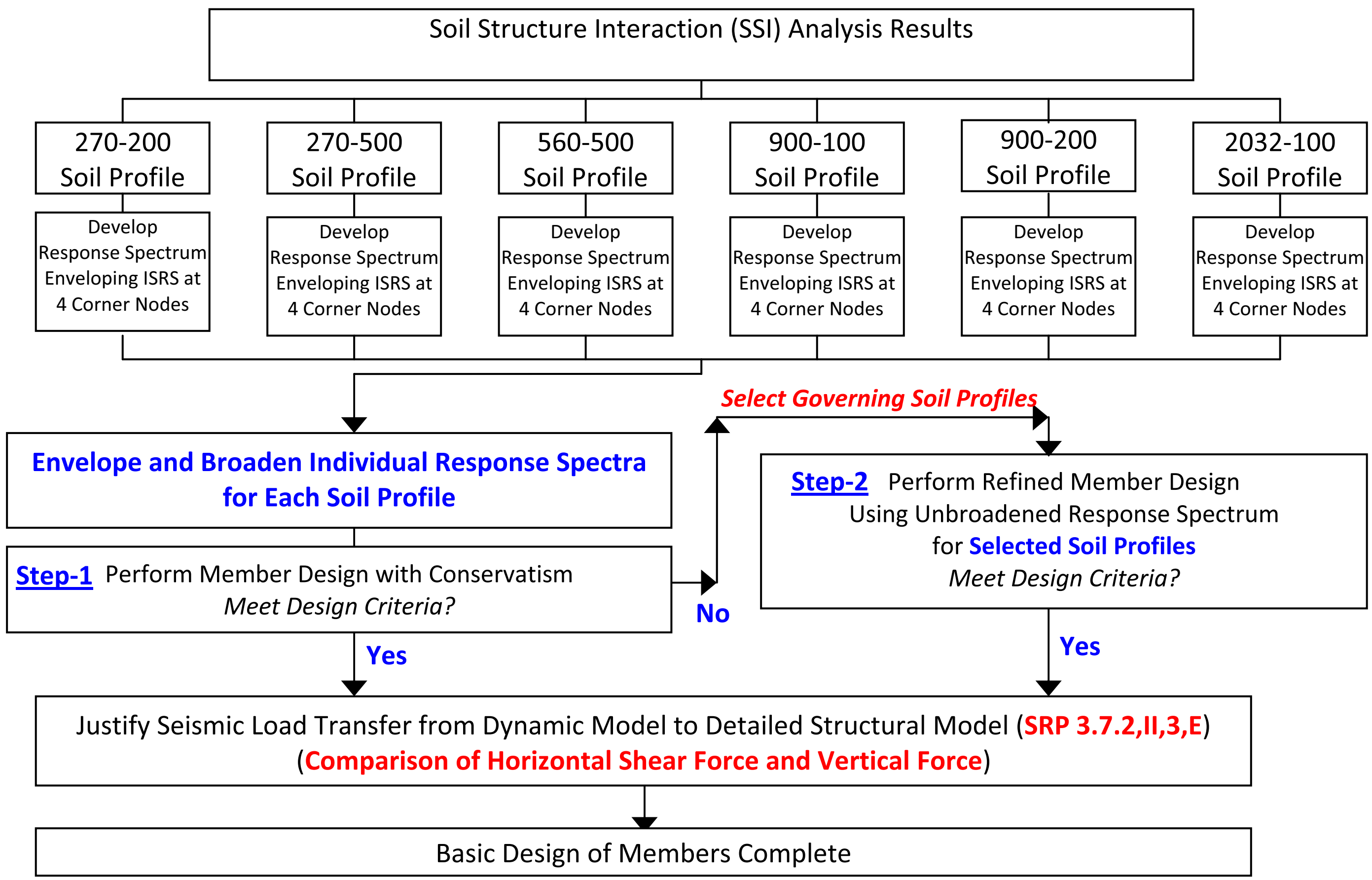


Figure 3.8.4-28 Flow Chart for Seismic Force Transfer from SSI Model to Structural Model

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.5-1 Floor Plan of 1F (El. -3'-7")

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.5-2 Cross Section of a North-South Orientation

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 3.8.5-3 Cross Section of an East-West Orientation

Figure 3.8.5-4 Deleted

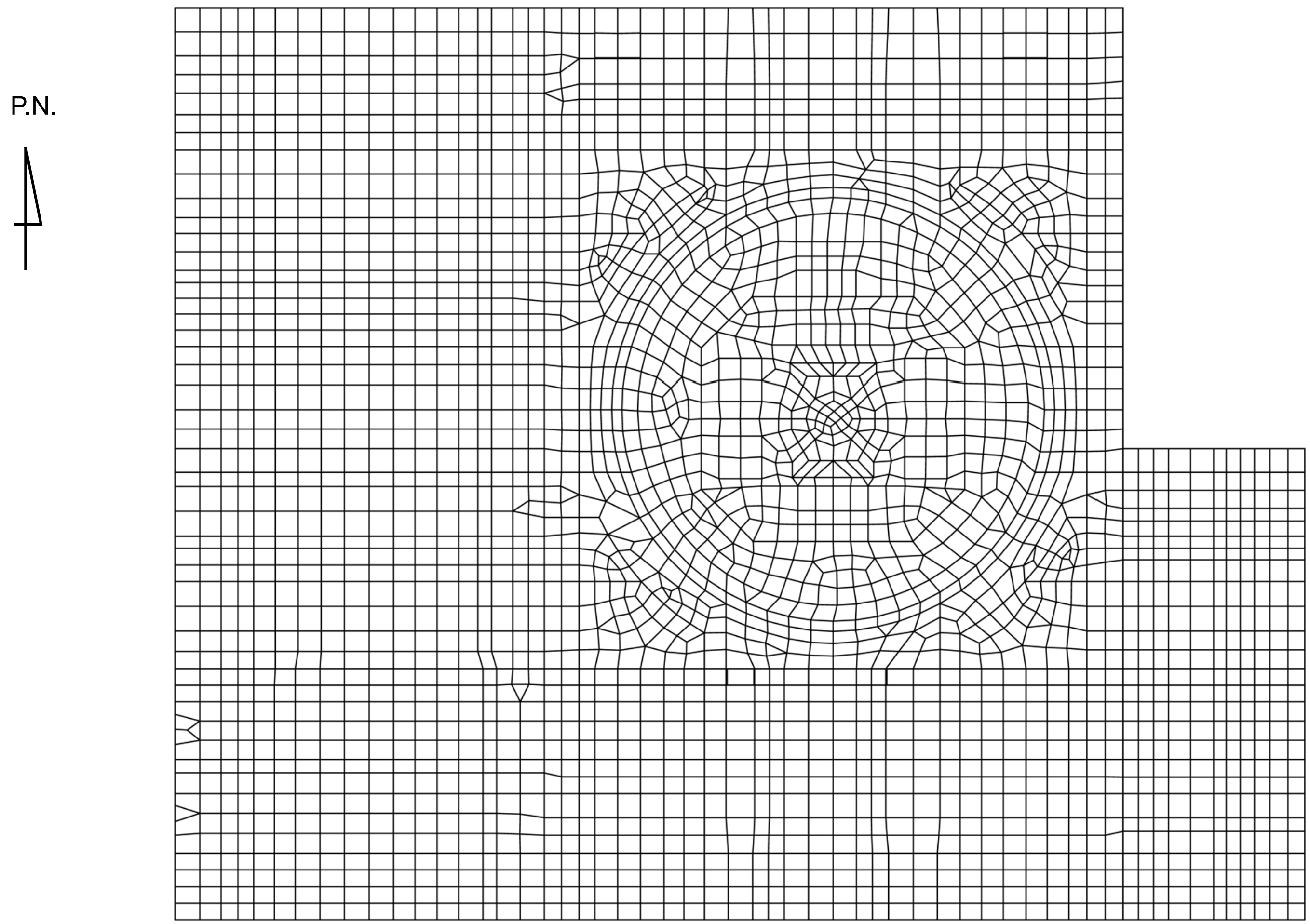


Figure 3.8.5-5 R/B, PCCV, and Containment Internal Structure Basemat

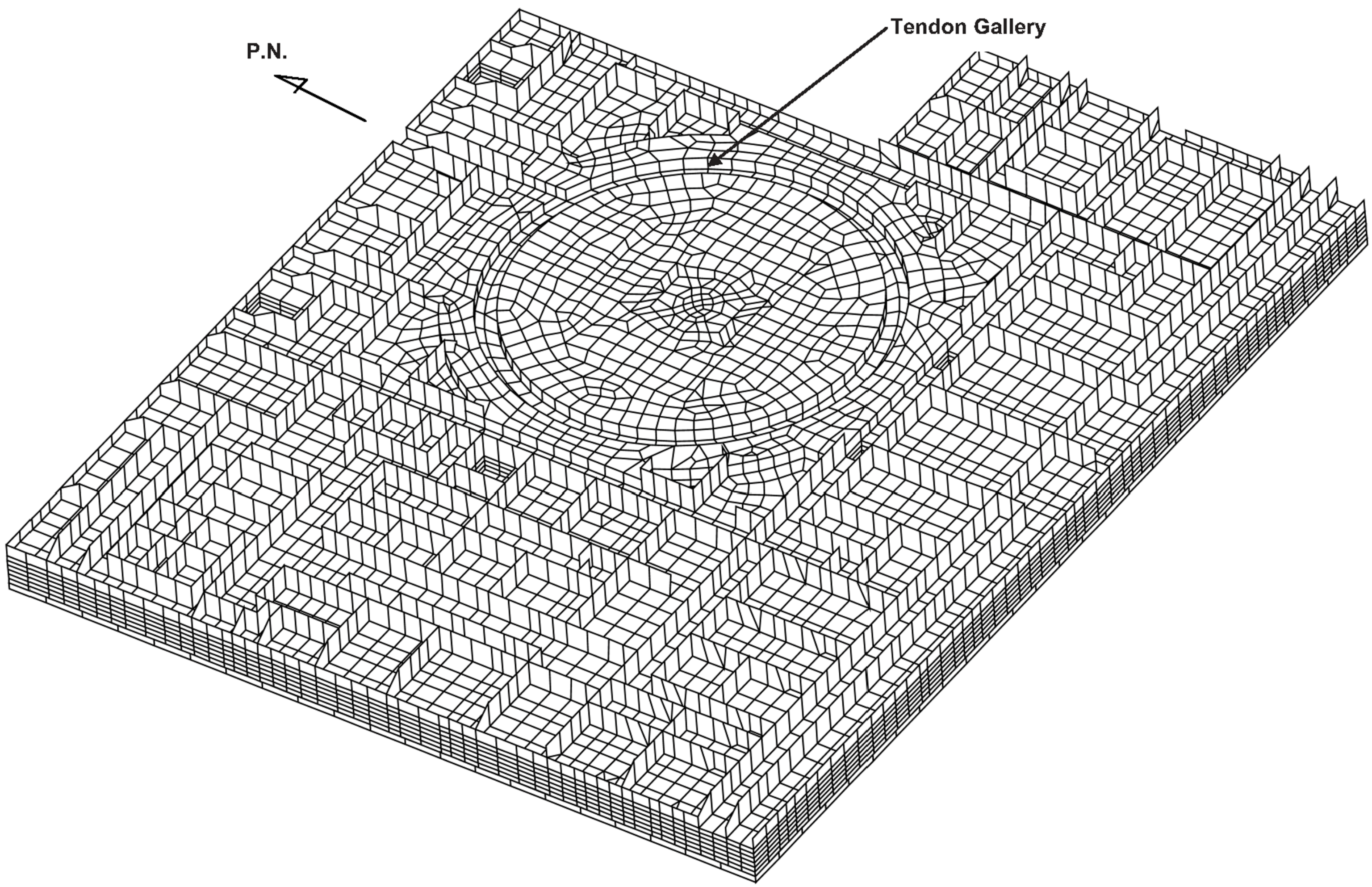


Figure 3.8.5-6 Global Three-Dimensional FE Model of R/B Complex Basemat at Tendon Gallery

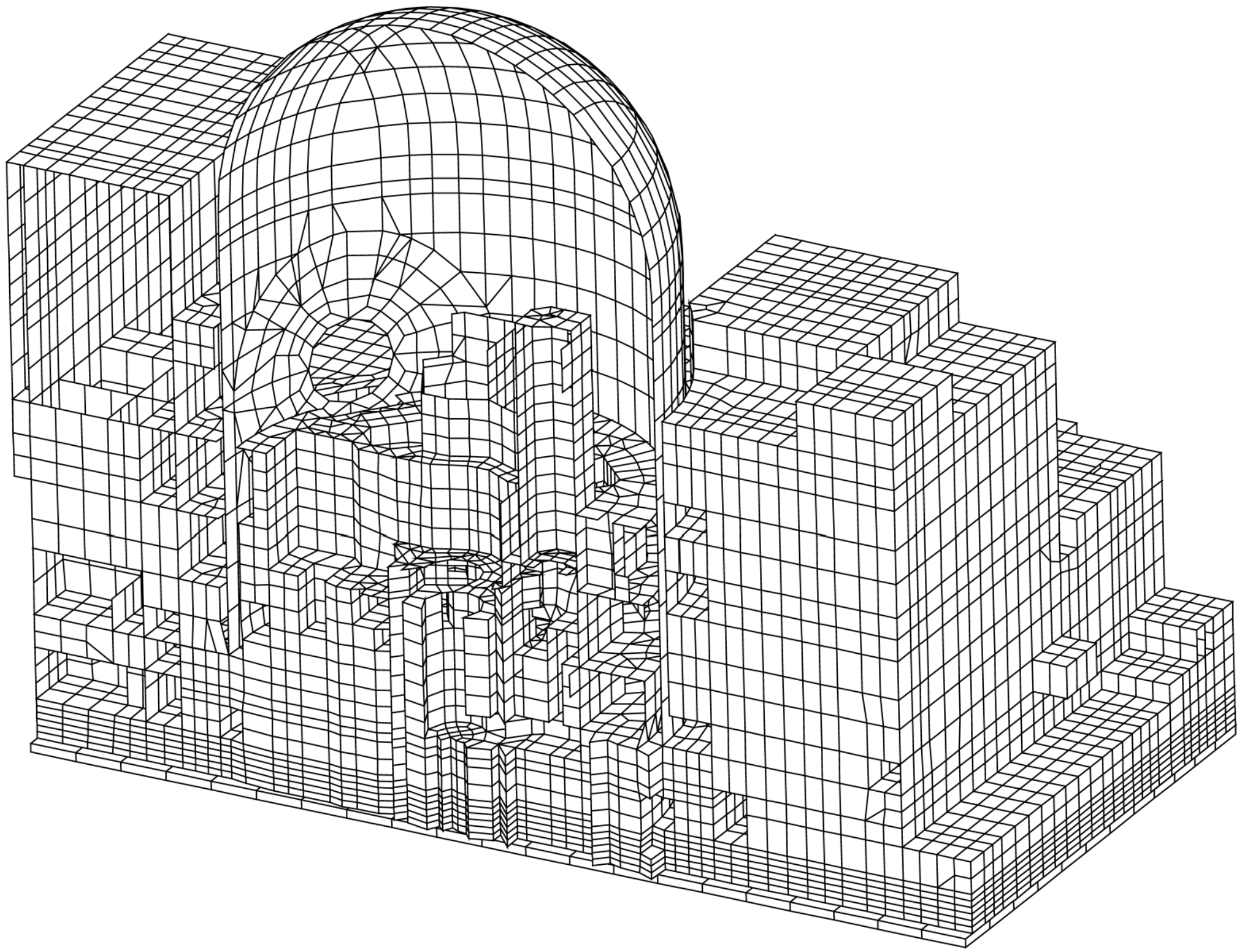


Figure 3.8.5-7 Global Three-Dimensional FE Model of R/B Complex (N-S Section)

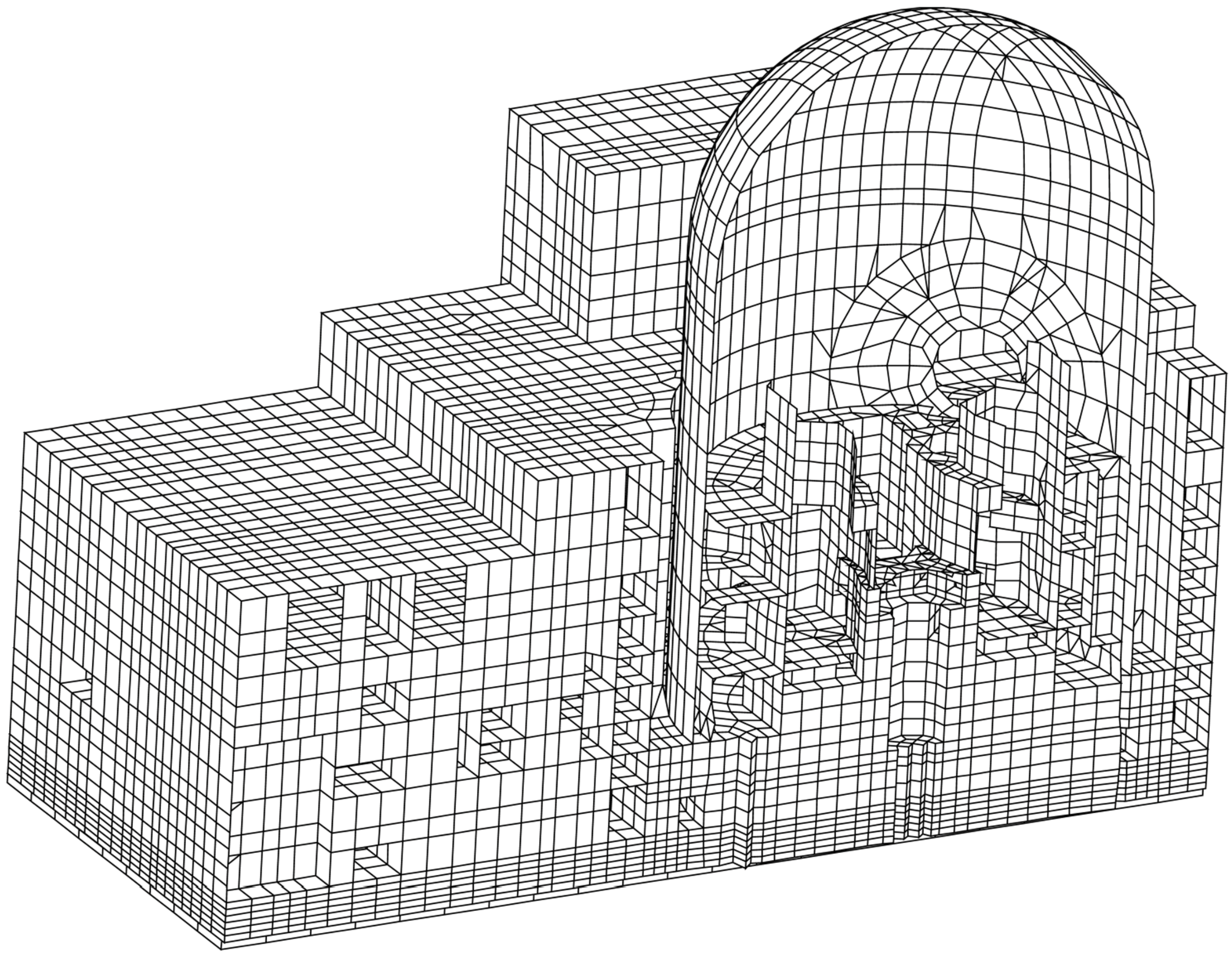


Figure 3.8.5-8 Global Three-Dimensional FE Model of R/B Complex (E-W Section)

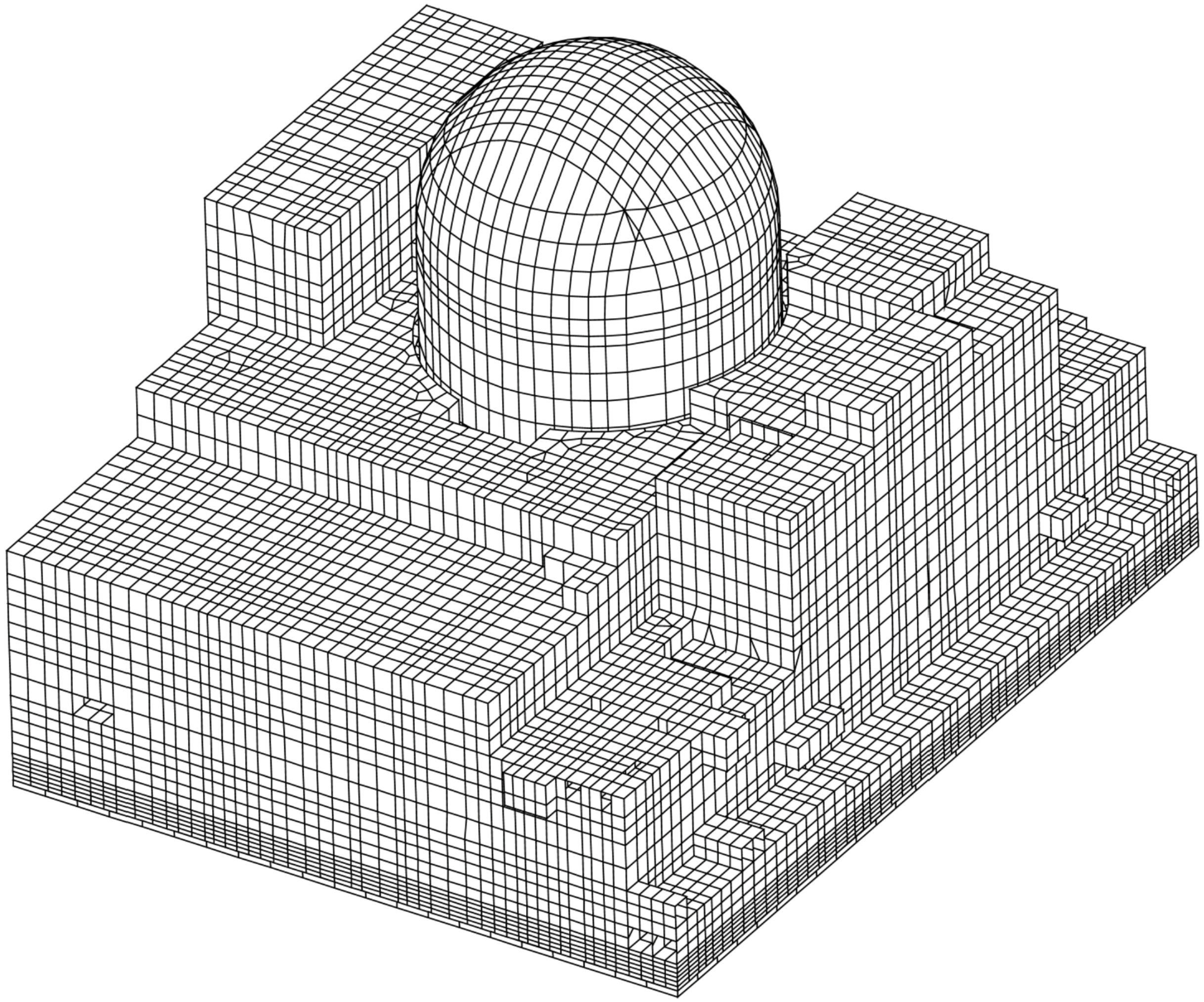


Figure 3.8.5-9 Global Three-Dimensional FE Model of R/B Complex (West Side)

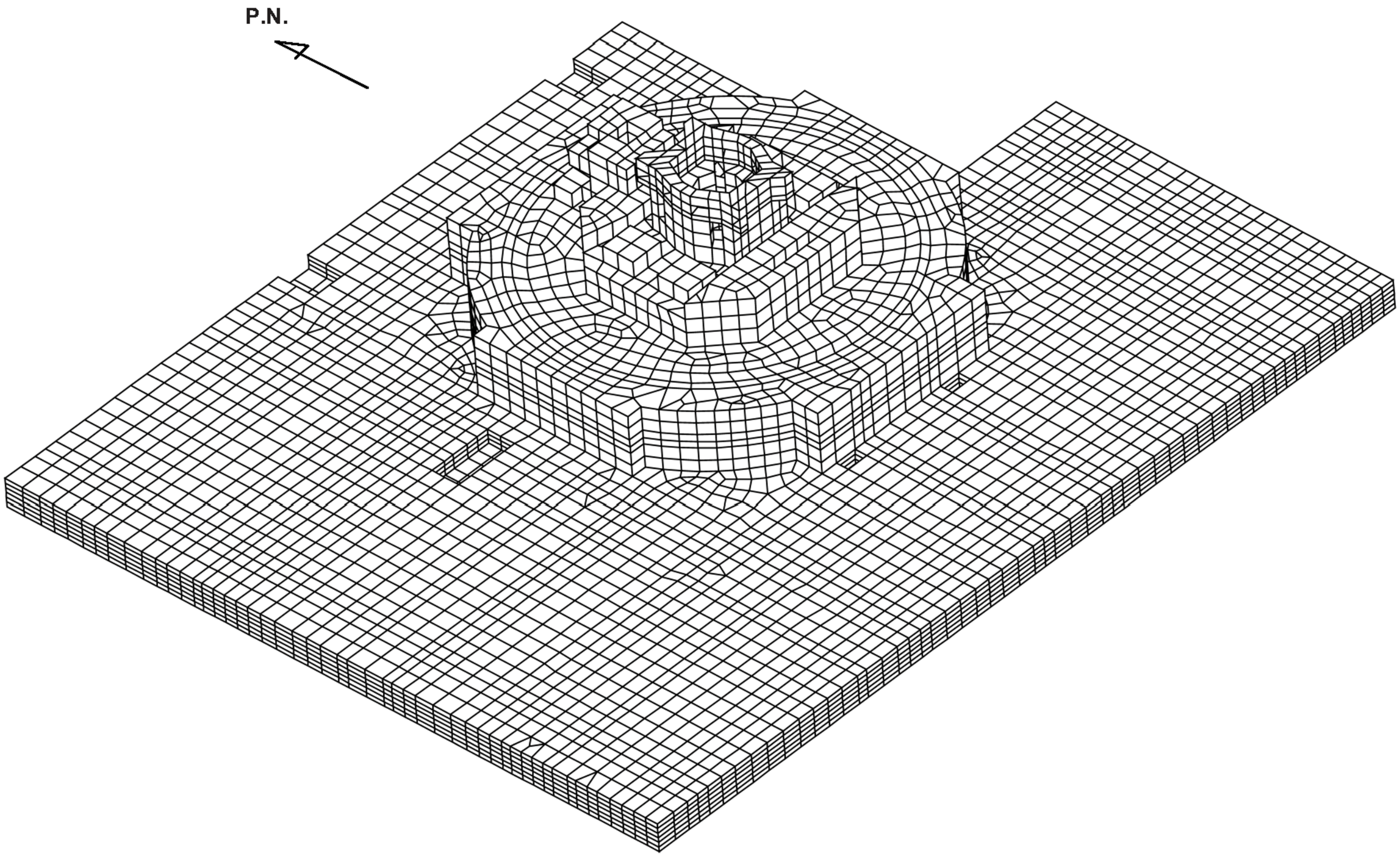


Figure 3.8.5-10 Global Three-Dimensional FE Model of R/B Complex Basemat (Solid Elements)

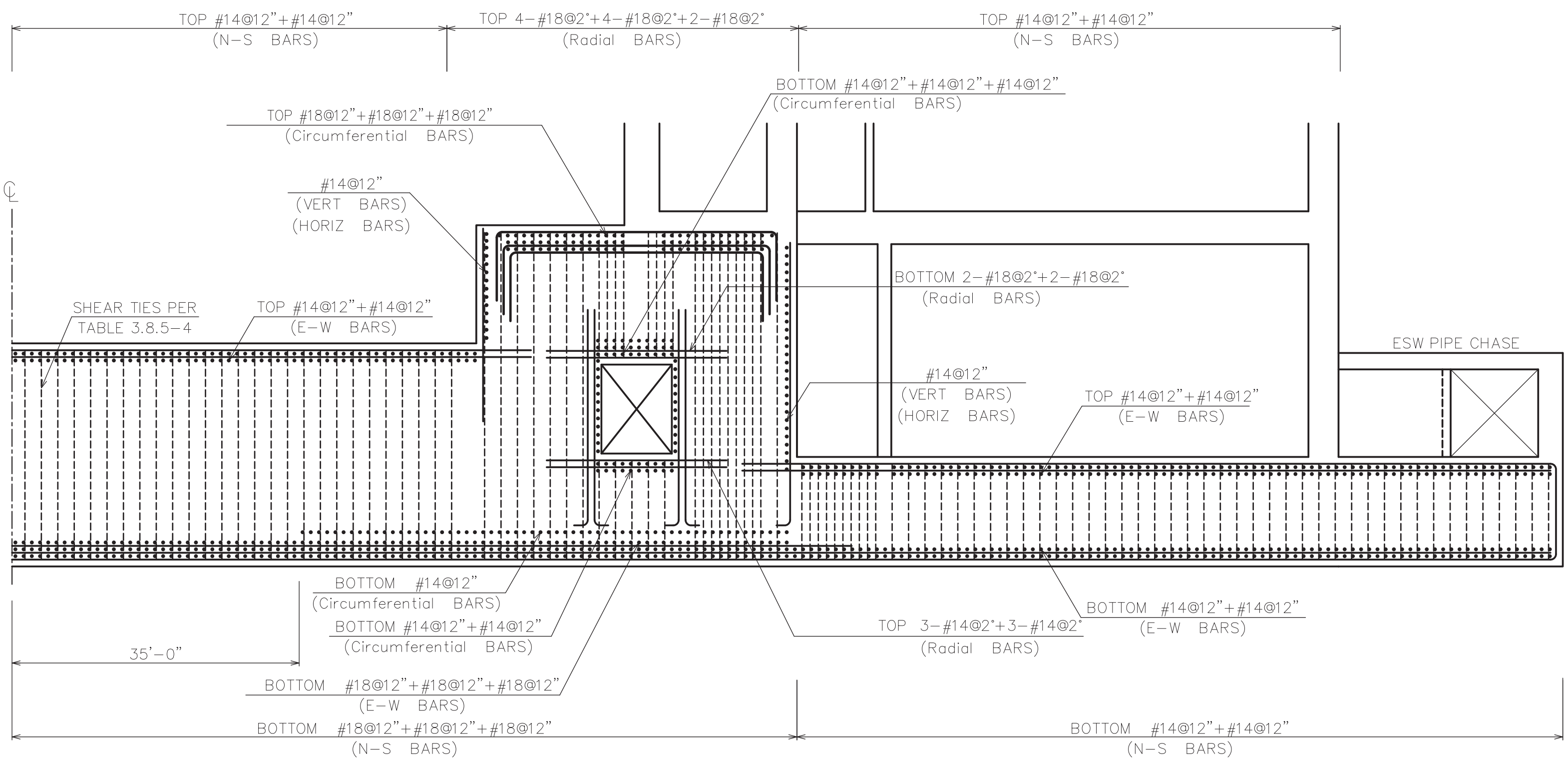


Figure 3.8.5-11 Reinforcing Steel of Basemat: SECTION N-S

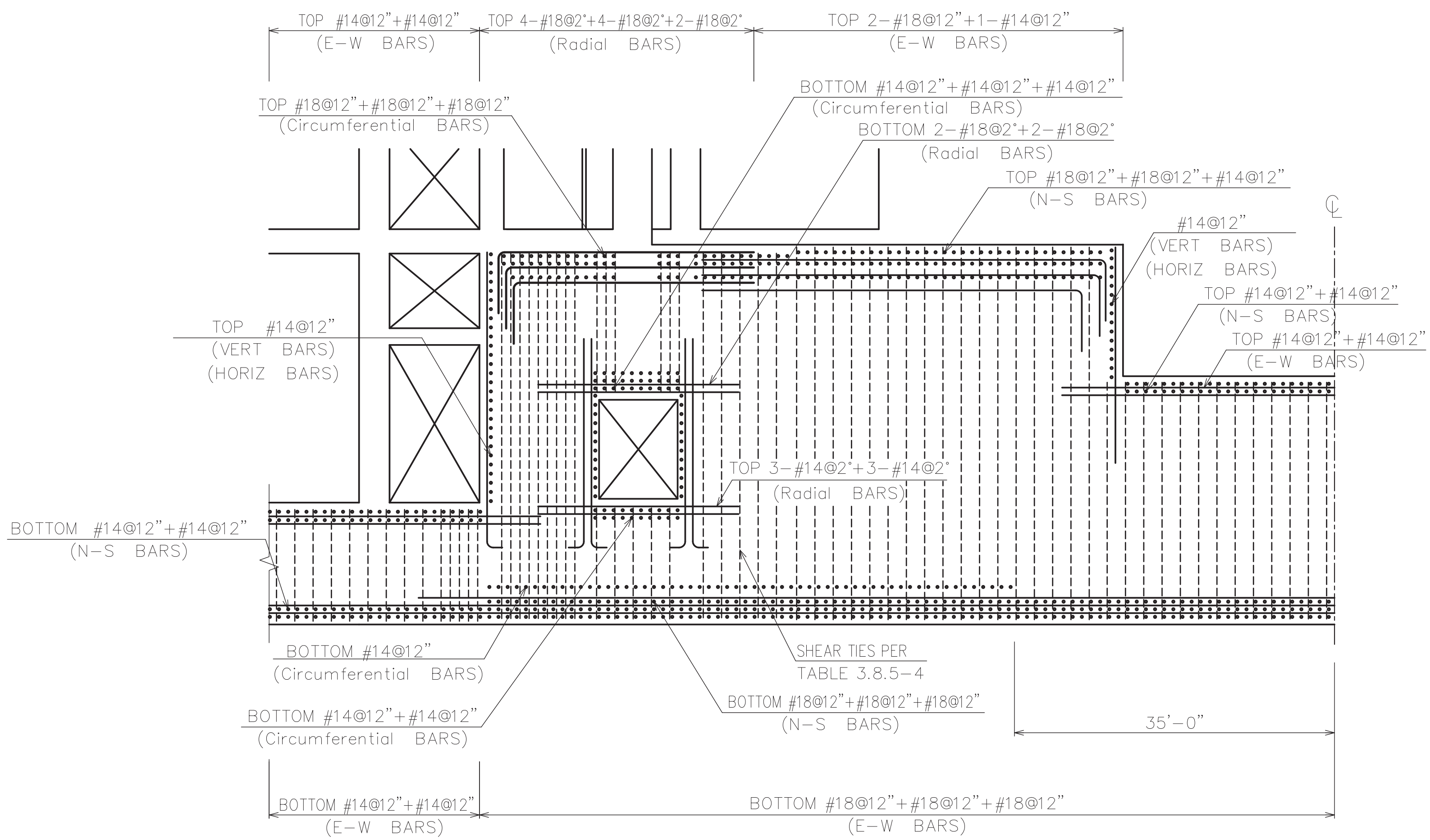


Figure 3.8.5-12 Reinforcing Steel of Basemat: SECTION E-W

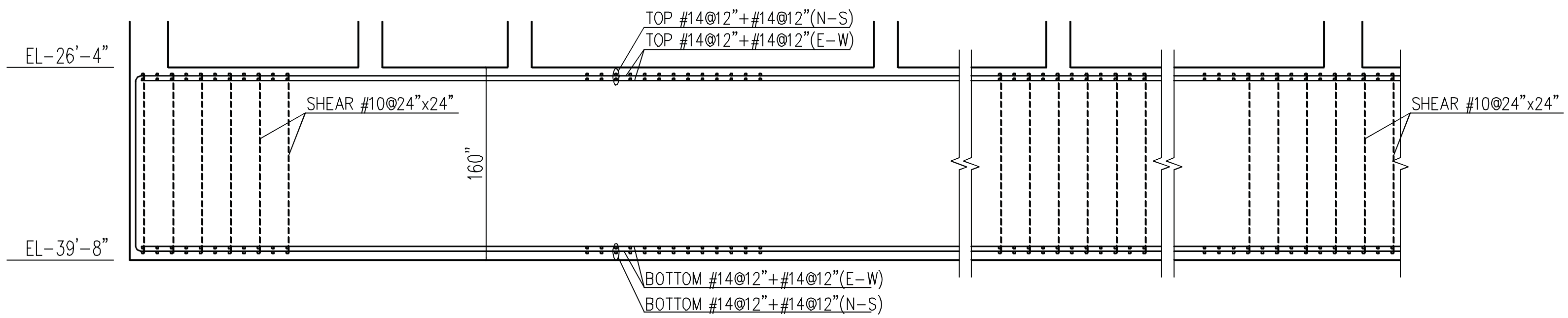


Figure 3.8.5-13 Typical Reinforcement for R/B Complex around Periphery of Basemats

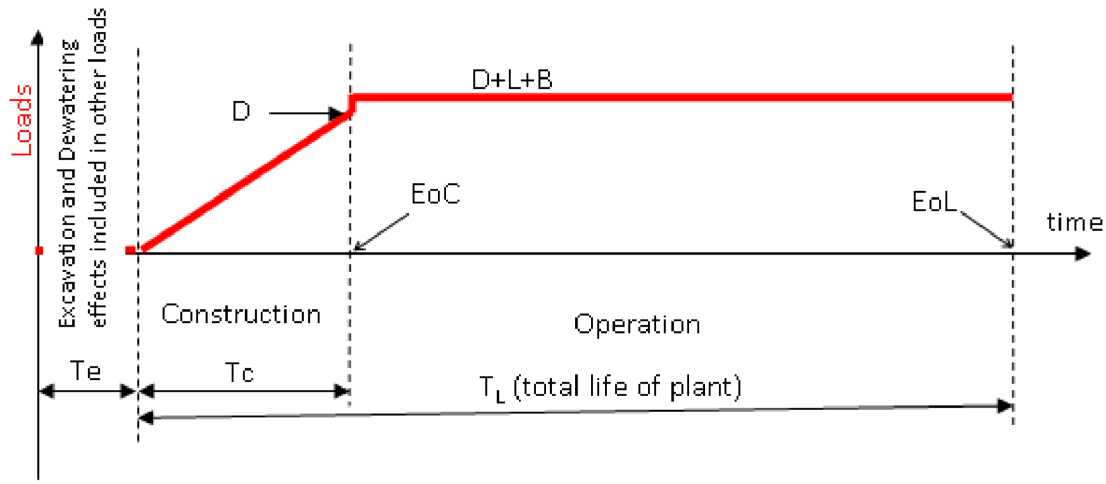


Figure 3.8.5-14 Timeline of Loading for Settlement Analyses

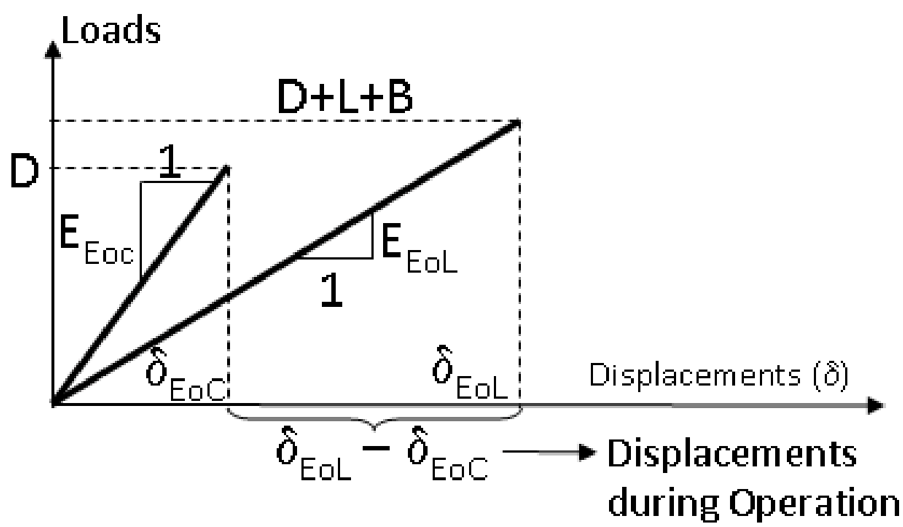


Figure 3.8.5-15 Secant Moduli for Settlement Analyses

Note: See Subsection 3.8.5.4 for definition of terms used in the figure.