3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

TABLE OF CONTENTS

Section	Title	Page
3.1	CONFORMANCE WITH NUCLEAR REGULATORY COMMISSION (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	3.1-3
3.1.1.5	Criterion 5 - Sharing of Structures, Systems, and Components	3.1-4
3.1.2	PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS	3.1-4
3.1.2.1	Criterion 10 - Reactor Design	3.1-4
3.1.2.2	Criterion 11 - Reactor Inherent Protection	3.1-5
3.1.2.3	Criterion 12 - Suppression of Reactor Power Oscillations	3.1-5
3.1.2.4	Criterion 13 - Instrumentation and Control	3.1-6
3.1.2.5	Criterion 14 - Reactor Coolant Pressure Boundary	3.1-6
3.1.2.6	Criterion 15 - Reactor Coolant System Design	3.1-7
3.1.2.7	Criterion 16 - Containment Design	3.1-8
3.1.2.8	Criterion 17 - Electric Power Systems	3.1-8
3.1.2.9	Criterion 18 - Inspection and Testing of Electric Power System	3.1-10
3.1.2.10	Criterion 19 Control Room	3.1-10
3.1.3	PROTECTION AND REACTIVITY CONTROL SYSTEMS	3.1-11
3.1.3.1	Criterion 20 - Protection System Functions	3.1-11
3.1.3.2	Criterion 21 - Protection Systems Reliability and Testability	3.1-12
3.1.3.3	Criterion 22 - Protection System Independence	3.1-12
3.1.3.4	Criterion 23 - Protection System Failure Modes	3.1-13
3.1.3.5	Criterion 24 - Separation of Protection and Control Systems	3.1-13
3.1.3.6	Criterion 25 - Protection System Requirements For Reactivity Control Malfunctions	3.1-14
3.1.3.7	Criterion 26 - Reactivity Control System Redundancy and Capability	3.1-14
3.1.3.8	Criterion 27 - Combined Reactivity Control Systems Capability	3.1-15
3.1.3.9	Criterion 28 - Reactivity Limits	3.1-15
3.1.3.10	Criterion 29 - Protection Against Anticipated Operational Occurrences	3.1-16
3.1.4	FLUID SYSTEMS	3.1-16
3.1.4.1	Criterion 30 - Quality of Reactor Coolant Pressure Boundary	3.1-16
3.1.4.2	Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary	3.1-17
3.1.4.3	Criterion 32 - Inspection of Reactor Coolant Pressure Boundary	3.1-18
3.1.4.4	Criterion 33 - Reactor Coolant Makeup	3.1-18
3.1.4.5	Criterion 34 - Residual Heat Removal	3.1-19
3.1.4.6	Criterion 35 - Emergency Core Cooling	3.1-20
3.1.4.7	Criterion 36 - Inspection of Emergency Core Cooling System	3.1-20
3.1.4.8	Criterion 37 - Testing of Emergency Core Cooling System	3.1-20
3.1.4.9	Criterion 38 - Containment Heat Removal	3.1-21

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.1.4.10	Criterion 39 - Inspection of Containment Heat Removal System	3.1-21
3.1.4.11	Criterion 40 - Testing of Containment Heat Removal System	3.1-22
3.1.4.12	Criterion 41 - Containment Atmosphere Cleanup	3.1-22
3.1.4.13	Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems	3.1-23
3.1.4.14	Criterion 43 - Testing of Containment Atmosphere Cleanup Systems	3.1-23
3.1.4.15	Criterion 44 - Cooling Water	3.1-24
3.1.4.16	Criterion 45 - Inspection of Cooling Water System	3.1-24
3.1.4.17	Criterion 46 - Testing of Cooling Water System	3.1-25
3.1.5	REACTOR CONTAINMENT	3.1-25
3.1.5.1	Criterion 50 - Containment Design Basis	3.1-25
3.1.5.2	Criterion 51 - Fracture Prevention of Containment Pressure Boundary	3.1-26
3.1.5.3	Criterion 52 - Capability For Containment Leakage Rate Testing	3.1-26
3.1.5.4	Criterion 53 - Provisions for Containment Testing and Inspection	3.1-26
3.1.5.5	Criterion 54 - Piping Systems Penetrating Containment	3.1-27
3.1.5.6	Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment	3.1-27
3.1.5.7	Criterion 56 - Primary Containment Isolation	3.1-28
3.1.5.8	Criterion 57 - Closed System Isolation Valves	3.1-29
3.1.6	FUEL AND RADIOACTIVITY CONTROL	3.1-29
3.1.6.1	Criterion 60 - Control of Releases of Radioactive Materials to the Environment	3.1-29
3.1.6.2	Criterion 61 - Fuel Storage and Handling and Radioactivity Control	3.1-30
3.1.6.3	Criterion 62 - Prevention of Criticality in Fuel Storage and Handling	3.1-31
3.1.6.4	Criterion 63 - Monitoring Fuel and Waste Storage	3.1-31
3.1.6.5	Criterion 64 - Monitoring Radioactivity Releases	3.1-32
	REFERENCES	3.1-32
3.2	CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS	3.2-1
3.2.1	SEISMIC CLASSIFICATION	3.2-1
3.2.1.1	Seismic Category I	3.2-1
3.2.1.1.1	Seismic Category I Structures	3.2-1
3.2.1.1.2	Seismic Category I Mechanical Systems and Components	3.2-2
3.2.1.1.3	Seismic Category I Electrical Systems and Components	3.2-3
3.2.1.1.4	Structures and Systems of Mixed Category	3.2-4
3.2.1.2	Seismic Category II	3.2-4
3.2.2	SYSTEM QUALITY GROUP CLASSIFICATION	3.2-6
3.2.3	PLANT DRAWINGS	3.2-9
3.2.3.1	Flow Diagrams	3.2-9
3.2.3.2	One Line Diagrams	3.2-10
3.2.3.3	Radiation Zone and Shield Thickness Drawings	3.2-11
	REFERENCES	3.2-12

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.3	WIND AND TORNADO LOADINGS	3.3-1
3.3.1	WIND LOADINGS	3.3-1
3.3.1.1	Design Wind Velocity	3.3-1
3.3.1.2	Determination of Applied Forces	3.3-1
3.3.2	TORNADO LOADINGS	3.3-2
3.3.2.1	Applicable Design Parameters	3.3-2
3.3.2.2	Tornado Protection Design Features	3.3-2
3.3.2.3	Method of Analysis	3.3-4
3.3.2.4	Design Evaluation	3.3-5
	REFERENCES	3.3-7
3.4	WATER LEVEL (FLOOD) DESIGN	3.4-1
3.4.1	FLOOD PROTECTION	3.4-1
3.4.2	ANALYSIS PROCEDURES	3.4-2
3.4.3	FLOODING FROM TANK RUPTURE	3.4-2
3.5	MISSILE PROTECTION	3.5-1
3.5.1	MISSILE SELECTION AND DESCRIPTION	3.5-1
3.5.1.1	Internally Generated Missiles (Outside Containment)	3.5-1
3.5.1.1.1	Missile Selection	3.5-1
3.5.1.1.2	Missile Protection	3.5-1
3.5.1.1.3	Design Adequacy of Equipment	3.5-2
3.5.1.2	Internally Generated Missiles (Inside Containment)	3.5-3
3.5.1.2.1	Missile Selection	3.5-3
3.5.1.2.2	Missile Description	3.5-6
3.5.1.2.3	Missile Protection	3.5-7
3.5.1.2.4	The Control Rod Drive Shaft Missile Shield	3.5-7
3.5.1.2.5	Secondary Missiles	3.5-7
3.5.1.3	Turbine Missiles	3.5-7
3.5.1.3.1	Turbine Placement and Orientation	3.5-7
3.5.1.3.2	Missile Identification and Characteristics	3.5-8
3.5.1.3.3	Strike Probability Analysis	3.5-8
3.5.1.3.4	Turbine Overspeed Protection	3.5-9
3.5.1.3.5	Turbine Valve Testing	3.5-10
3.5.1.3.6	Turbine Characteristics	3.5-10
3.5.1.4	Missiles Generated by Natural Phenomena	3.5-10
3.5.1.5	Missiles Generated by Events Near the Site	3.5-12
3.5.1.6	Aircraft Hazards	3.5-12
3.5.2	SYSTEMS TO BE PROTECTED	3.5-12
3.5.3	BARRIER DESIGN PROCEDURES	3.5-12
3.5.3.1	Local Effects	3.5-12
3.5.3.2	Overall Barrier Response	3.5-13

TABLE OF CONTENTS (Continued)

Section	Title	Page
	REFERENCES	3.5-14
APP. 3.5A	THEORY OF MISSILE ENERGY	3.5A-1
APP. 3.5B	TURBINE MISSILE STRIKE PROBABILITY DISTRIBUTIONS	3.5B-1
3.6N	PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING	3.6N-1
3.6B	PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING	3.6B-1
3.6B.1	POSTULATED PIPING FAILURES IN FLUID SYSTEMS	3.6B-1
3.6B.1.1	Design Basis	3.6B-1
3.6B.1.2	Description	3.6B-2
3.6B.1.2.1	Protection Criteria	3.6B-2
3.6B.1.2.2	LOCA Break Propagation Criteria	3.6B-4
3.6B.1.2.3	Environmental Analysis to Determine HELB Temperature Transients for Purposes of Equipment Qualification (Outside Containment)	3.6B-5
3.6B.1.3	Safety Evaluation	3.6B-12
3.6B.2	DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING (INSIDE AND OUTSIDE CONTAINMENT)	3.6B-14
3.6B.2.1	Criteria Used To Define Break And Crack Location And Configuration	3.6B-14
3.6B.2.1.1	Reactor Coolant System (RCS) Main Loop Piping	3.6B-14
3.6B.2.1.2	High Energy Piping Other Than The RCS Main Loop	3.6B-15
3.6B.2.1.3	Type of Breaks Postulated in Fluid System Piping Other Than the RCS Main Loop	3.6B-18
3.6B.2.1.4	Moderate Energy Piping	3.6B-19
3.6B.2.1.5	Definition of Plant Operating Conditions	3.6B-20
3.6B.2.2	Analytical Methods to Define Forcing Functions and Response Models	3.6B-20
3.6B.2.2.1	Reactor Coolant System Main Loop Piping	3.6B-20
3.6B.2.2.2	High-Energy Piping Other Than RCS Main Loop	3.6B-22
3.6B.2.3	Dynamic Analysis Methods to Verify Integrity and Operability	3.6B-25
3.6B.2.3.1	Reactor Coolant System Main Loop	3.6B-25
3.6B.2.3.2	High-Energy Piping Other than the RCS Main Loop	3.6B-25
3.6B.2.3.3	Pipe Whip Restraint Design Criteria	3.6B-33
3.6B.2.4	Guard Pipe Assembly Design Criteria	3.6B-36
3.6B.2.5	Material to be Submitted for the Operating License Review	3.6B-36
3.6B.2.5.1	Reactor Coolant System Main Loop Piping	3.6B-37
3.6B.2.5.2	High-Energy Piping Other Than RCS Main Loop	3.6B-37
3.6B.2.5.3	Moderate Energy Piping	3.6B-51
	REFERENCES	3.6B-52

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.7N	SEISMIC DESIGN	3.7N-1
3.7N.1	SEISMIC INPUT	3.7N-1
3.7N.1.1	Design Response Spectra	3.7N-1
3.7N.1.2	Design Time History	3.7N-1
3.7N.1.3	Critical Damping Values	3.7N-2
3.7N.1.4	Supporting Media for Seismic Category I Structures	3.7N-3
3.7N.2	SEISMIC SYSTEM ANALYSIS	3.7N-3
3.7N.2.1	Seismic Analysis Methods	3.7N-3
3.7N.2.1.1	Dynamic Analysis - Mathematical Model	3.7N-3
3.7N.2.1.2	Modal Analysis	3.7N-5
3.7N.2.1.3	Response Spectrum Analysis	3.7N-8
3.7N.2.1.4	Integration of Modal Equations	3.7N-9
3.7N.2.1.5	Integration of Coupled Equations of Motion	3.7N-10
3.7N.2.1.6	Static Analysis - Rigid and Limited Flexible Equipment	3.7N-12
3.7N.2.2	Natural Frequencies and Response Loads	3.7N-12
3.7N.2.3	Procedures Used for Modeling	3.7N-12
3.7N.2.4	Soil/Structure Interaction	3.7N-13
3.7N.2.5	Development of Floor Response Spectra	3.7N-13
3.7N.2.6	Three Components of Earthquake Motion	3.7N-13
3.7N.2.7	Combination of Modal Responses	3.7N-14
3.7N.2.8	Interaction of Non-Category I Structures with Seismic Category I Structures	3.7N-15
3.7N.2.9	Effects of Parameter Variations on Floor Response Spectra	3.7N-15
3.7N.2.10	Use of Constant Vertical Static Factors	3.7N-16
3.7N.2.11	Methods Used to Account for Torsional Effects	3.7N-16
3.7N.2.12	Comparison of Responses	3.7N-16
3.7N.2.13	Methods for Seismic Analysis of Dams	3.7N-16
3.7N.2.14	Determination of Seismic Category I Structure Overturning Moments	3.7N-16
3.7N.2.15	Analysis Procedure for Damping	3.7N-16
3.7N.3	SEISMIC SUBSYSTEM ANALYSIS	3.7N-16
3.7N.3.1	Seismic Analysis Methods	3.7N-16
3.7N.3.2	Determination of Number of Earthquake Cycles	3.7N-16
3.7N.3.3	Procedures Used for Modeling	3.7N-17
3.7N.3.4	Basis for Selection of Frequencies	3.7N-17
3.7N.3.5	Use of Equivalent Static Load Method of Analysis	3.7N-18
3.7N.3.6	Three Components for Earthquake Motion	3.7N-18
3.7N.3.7	Combination of Modal Responses	3.7N-18
3.7N.3.8	Analytical Procedures for Piping	3.7N-18
3.7N.3.9	Multiply Supported Equipment Components with Distinct Inputs	3.7N-18
3.7N.3.10	Use of Constant Vertical Static Factors	3.7N-19
3.7N.3.11	Torsional Effects of Eccentric Masses	3.7N-19
3.7N.3.12	Buried Seismic Category I Piping Systems and Tunnels	3.7N-19
3.7N.3.13	Interaction of Other Piping with Seismic Category I Piping	3.7N-19
3.7N.3.14	Seismic Analyses for Reactor Internals	3.7N-19

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.7N.3.15	Analysis Procedure for Damping	3.7N-21
3.7N.4	SEISMIC INSTRUMENTATION	3.7N-21
	REFERENCES	3.7N-21
3.7B	SEISMIC DESIGN	3.7B-1
3.7B.1	SEISMIC INPUT	3.7B-1
3.7B.1.1	Design Response Spectra	3.7B-1
3.7B.1.2	Design Time History	3.7B-1
3.7B.1.3	Critical Damping Values	3.7B-3
3.7B.1.4	Supporting Media for Seismic Category I Structures	3.7B-4
3.7B.2	SEISMIC SYSTEM ANALYSIS	3.7B-4
3.7B.2.1	Seismic Analysis Methods	3.7B-4
3.7B.2.1.1	Idealization of Seismic Category I Structures, Systems, and Components	3.7B-5
3.7B.2.1.2	Analytical Approach	3.7B-6
3.7B.2.1.3	Testing and Analysis for Equipment	3.7B-8
3.7B.2.1.4	Differential Seismic Movement of Interconnected Components	3.7B-13
3.7B.2.1.5	Stress and Deformation Criteria	3.7B-13
3.7B.2.1.6	Descriptions of Mathematical Models	3.7B-14
3.7B.2.2	Natural Frequencies and Response Loads	3.7B-16
3.7B.2.3	Procedure Used for Modeling	3.7B-16
3.7B.2.4	Soil-Structure Interaction	3.7B-18
3.7B.2.4.1	Evaluation of Soil-Structure Interaction Parameters	3.7B-19
3.7B.2.4.2	Effect of Embedment on Foundation Rigidities	3.7B-23
3.7B.2.5	Development of Floor Response Spectra	3.7B-25
3.7B.2.6	Three Components of Earthquake Motion	3.7B-27
3.7B.2.7	Combination of Modal Responses	3.7B-27
3.7B.2.8	Interaction of Non-Category I Structures with Seismic Category I Structures	3.7B-27
3.7B.2.9	Effects of Parameter Variations on Floor Response Spectra	3.7B-28
3.7B.2.10	Use of Constant Vertical Static Factors	3.7B-29
3.7B.2.11	Method Used to Account for Torsional Effects	3.7B-29
3.7B.2.12	Comparison of Responses	3.7B-30
3.7B.2.13	Methods for Seismic Analysis of Dams	3.7B-30
3.7B.2.14	Determination of Seismic Category I Structure Overturning Moments	3.7B-31
3.7B.2.15	Analysis Procedure for Damping	3.7B-33
3.7B.3	SEISMIC SUBSYSTEM ANALYSIS	3.7B-34
3.7B.3.1	Seismic Analysis Methods	3.7B-34
3.7B.3.2	Determination of Number of Earthquake Cycles	3.7B-35
3.7B.3.3	Procedure Used for Modeling	3.7B-35
3.7B.3.4	Basis for Selection of Frequencies	3.7B-36
3.7B.3.5	Use of Equivalent Static Load Method of Analysis	3.7B-36
3.7B.3.6	Three Components of Earthquake Motion	3.7B-37
3.7B.3.7	Combination of Modal Responses	3.7B-37

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.7B.3.8	Analytical Procedures for Piping	3.7B-37
3.7B.3.8.1	Design Criteria	3.7B-37
3.7B.3.8.2	Basis for Computing Combined Responses	3.7B-38
3.7B.3.8.3	Amplified Seismic Responses	3.7B-38
3.7B.3.9	Multiple Supported Equipment Components with Distinct Inputs	3.7B-39
3.7B.3.10	Use of Constant Vertical Static Factors	3.7B-39
3.7B.3.11	Torsional Effects of Eccentric Masses	3.7B-39
3.7B.3.12	Buried Seismic Category I Piping Systems and Tunnels	3.7B-40
3.7B.3.12.1	Stresses Caused by Free-Field Seismic Wave Propagation	3.7B-40
3.7B.3.12.2	Stresses Caused by Differential Displacements Between Soil and Structure	3.7B-41
3.7B.3.13	Interaction of Other Piping with Seismic Category I Piping	3.7B-42
3.7B.3.13.1	Seismic Category I Piping with Connecting Non-Category I Piping	3.7B-42
3.7B.3.13.2	Seismic Category I Piping with Adjacent Non-Category I Piping	3.7B-42
3.7B.3.14	Seismic Analyses for Reactor Internals	3.7B-43
3.7B.3.15	Analysis Procedure for Damping	3.7B-43
3.7B.3.16	Analysis Procedure for High Density Spent Fuel Storage Racks	3.7B-43
3.7B.4	SEISMIC INSTRUMENTATION	3.7B-44
3.7B.4.1	Comparison with Regulatory Guide 1.12	3.7B-44
3.7B.4.2	Location and Description of Instrumentation	3.7B-45
3.7B.4.3	Control Room Operator Notification	3.7B-45
3.7B.4.4	Comparison of Measured and Predicted Responses	3.7B-45
	REFERENCES	3.7B-45
APP. 3.7B(A)	COMPUTER PROGRAMS USED IN DYNAMIC AND STATIC ANALYSES	3.7B(A)-1
3.7B(A).1	INTRODUCTION	3.7B(A)-1
3.7B(A).2	COMPUTER PROGRAM (QUAKE)	3.7B(A)-1
3.7B(A).3	FINITE ELEMENT COMPUTER PROGRAM (NASTRAN)	3.7B(A)-6
3.7B(A).4	COMPUTER PROGRAM (CSMP)	3.7B(A)-6
3.7B(A).5	FINITE ELEMENT COMPUTER PROGRAM (STARDYNE)	3.7B(A)-6
3.7B(A).6	FINITE ELEMENT COMPUTER PROGRAM (ICES STRUDL-II)	3.7B(A)-7
3.7B(A).7	STRUCTURAL ENGINEERING SYSTEM SOLVER (STRESS) COMPUTER PROGRAM	3.7B(A)-7
3.7B(A).8	COMPUTER PROGRAM FOR THE ANALYSIS OF SHELLS OF REVOLUTION (KALNINS)	3.7B(A)-7
3.7B(A).9	COMPUTER PROGRAM (SCONV)	3.7B(A)-8
3.7B(A).10	STORING OF DESIGN RESPONSE SPECTRUM ON THE COMPUTER (SPECTRA)	3.7B(A)-10
3.7B(A).11	FLOOR RESPONSE SPECTRA PROGRAM (TIME)	3.7B(A)-10
3.7B(A).12	COMPUTER PROGRAM (ADLPIPE)	3.7B(A)-10
3.7B(A).12.1	Description of the Analytical Techniques	3.7B(A)-11
3.7B(A).13	FINITE ELEMENT COMPUTER PROGRAM (ANSYS)	3.7B(A)-11
3.7B(A).14	COMPUTER PROGRAM (SUPERPIPE)	3.7B(A)-12

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.7B(A).15	STRUDL-SW (STRUCTURAL DESIGN LANGUAGE)	3.7B(A)-12
3.7B(A).16	SHELL-1 (THIN SHELL OF REVOLUTION UNDER ARBITRARY LOADING)	3.7B(A)-12
3.7B(A).17	TIMHIS6 (TIME HISTORY ANALYSIS)	3.7B(A)-13
3.7B(A).18	SBMMI (SINGLE BARRIER MASS MISSILE IMPACT)	3.7B(A)-13
3.7B(A).19	CLASSI (CONTINUUM LINEAR ANALYSIS OF SOIL STRUCTURE INTERACTION)	3.7B(A)-13
3.7B(A).20	RESPEC (RESPONSE SPECTRA GENERATOR)	3.7B(A)-14
3.7B(A).21	LS-DYNA	3.7B(A)-14
3.7B(A).22	MR216 (DYNARACK)	3.7B(A)-14
3.7B(A).23	OTHER COMPUTER PROGRAMS	3.7B(A)-15
	REFERENCES	3.7B(A)-15
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
3.8.1.1.1	General Description	3.8-1
3.8.1.1.2	Dimensions of Containment	3.8-1
3.8.1.1.3	Containment Function	3.8-1
3.8.1.1.4	Arrangement of Main Reinforcing Steel	3.8-1
3.8.1.1.5	Steel Liner	3.8-2
3.8.1.1.6	Containment Penetrations and Attachments	3.8-2
3.8.1.1.7	Drawings	3.8-4
3.8.1.2	Applicable Codes, Standards, and Specifications	3.8-4
3.8.1.2.1	Basic Code	3.8-4
3.8.1.2.2	ACI Committee 349	3.8-6
3.8.1.2.3	Additional Specifications and Standards	3.8-6
3.8.1.2.4	Specifications and Standards Not Referred to in ASME-ACI 359	3.8-9
3.8.1.2.5	Summary of Principal Plant Specifications	3.8-9
3.8.1.2.6	Applicable NRC Regulatory Guides	3.8-16
3.8.1.3	Loads and Load Combinations	3.8-16
3.8.1.3.1	Loads	3.8-16
3.8.1.3.2	Load Combinations	3.8-18
3.8.1.4	Design and Analysis Procedures	3.8-20
3.8.1.4.1	General Analysis of Entire Containment Structure	3.8-20
3.8.1.4.2	Analysis at Major Penetrations	3.8-22
3.8.1.4.3	General Design Criteria and Procedures	3.8-22
3.8.1.4.4	Design of Reinforced Concrete at Penetrations	3.8-23
3.8.1.4.5	Analysis and Design for Shear Effects	3.8-24
3.8.1.4.6	Analysis and Design of Liner and Anchorage	3.8-25
3.8.1.4.7	Design of ASME B&PV Code, Section III, Division 1, Class MC Components	3.8-27
3.8.1.5	Structural Acceptance Criteria	3.8-27
3.8.1.5.1	Reinforcing Steel - Allowable Stresses and Strains	3.8-27

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.8.1.5.2	Concrete - Allowable Stresses and Strains	3.8-27
3.8.1.5.3	Liners, Anchors, and Attachments - Allowable Stresses and Strains	3.8-29
3.8.1.5.4	Effect of Two- and Three-Dimensional Stress Strain Fields on the Behavior of the Structure	3.8-31
3.8.1.5.5	Effects of Repeated Reactor Shutdowns and Startup During the Life of the Plant	3.8-31
3.8.1.5.6	Connections and Joints	3.8-32
3.8.1.5.7	Conditions at End of Service Life of Structure	3.8-32
3.8.1.6	Materials, Quality Control, and Special Construction Techniques	3.8-32
3.8.1.6.1	Concrete	3.8-32
3.8.1.6.2	Reinforcing Steel	3.8-36
3.8.1.6.3	Mechanical Butt Splices (Cadwelds)	3.8-37
3.8.1.6.4	Welding to Reinforcing Bars	3.8-37
3.8.1.6.5	Liner and Attachments	3.8-37
3.8.1.6.6	Personnel Airlock, Equipment Hatch, and Emergency Airlock	3.8-39
3.8.1.6.7	Quality Assurance Program	3.8-40
3.8.1.7	Testing and Inservice Inspection Requirements	3.8-40
3.8.1.7.1	Structural Acceptance Test of Containment	3.8-40
3.8.1.7.2	Initial Leakage Rate Tests	3.8-41
3.8.1.7.3	Inservice Inspection Program	3.8-41
3.8.2	STEEL CONTAINMENT	3.8-41
3.8.2.1	Description of the Containment	3.8-42
3.8.2.2	Applicable Codes, Standards, and Specifications	3.8-42
3.8.2.2.1	Basic Code	3.8-42
3.8.2.2.2	Other Applicable Codes, Specifications, and Standards	3.8-42
3.8.2.2.3	Applicable NRC Regulatory Guide	3.8-42
3.8.2.3	Loads and Load Combinations	3.8-42
3.8.2.4	Design and Analysis Procedures	3.8-42
3.8.2.5	Structural Acceptance Criteria	3.8-43
3.8.2.5.1	General Criteria	3.8-43
3.8.2.5.2	Design Load Combination Stress Limits	3.8-43
3.8.2.6	Materials, Quality Control, and Special Construction Techniques	3.8-45
3.8.2.7	Testing and Inservice Inspection Requirements	3.8-45
3.8.3	CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS	3.8-46
3.8.3.1	Description of the Internal Structures	3.8-46
3.8.3.1.1	General Description	3.8-46
3.8.3.1.2	Polar Crane and Telescopic Jib Crane	3.8-47
3.8.3.1.3	Supports for Reactor Pressure Vessel, Steam Generator, Reactor Coolant Pump, Pressurizer, and Loop Piping	3.8-48
3.8.3.1.4	Drawings	3.8-48
3.8.3.2	Applicable Codes, Standards, and Specifications	3.8-48
3.8.3.2.1	Basic Codes	3.8-48
3.8.3.2.2	Supplementary Codes	3.8-49
3.8.3.2.3	Additional Specifications and Standards	3.8-49

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.8.3.2.4	Summary of Principal Plant Specifications	3.8-50
3.8.3.2.5	Applicable NRC Regulatory Guides	3.8-51
3.8.3.3	Loads and Load Combinations	3.8-51
3.8.3.3.1	Loads	3.8-51
3.8.3.3.2	Load Combinations and Acceptance Criteria for Internal Concrete Structures of the Containment	3.8-53
3.8.3.3.3	Load Combinations and Acceptance Criteria for Internal Steel Structures of the Containment	3.8-54
3.8.3.3.4	Variable Loads	3.8-55
3.8.3.3.5	Interrelated Loads	3.8-55
3.8.3.3.6	Live Loads	3.8-56
3.8.3.3.7	Load Combinations for Localized Areas	3.8-56
3.8.3.3.8	Time-Dependent Loads	3.8-56
3.8.3.4	Design and Analysis Procedures	3.8-56
3.8.3.4.1	General Analysis of Internal Structure	3.8-56
3.8.3.4.2	Additional Analysis at Local Areas	3.8-57
3.8.3.4.3	Interaction with Nuclear Steam Supply System Equipment Supports	3.8-57
3.8.3.4.4	Lateral Load Transfer at Foundation Mat	3.8-57
3.8.3.4.5	Design Variables	3.8-58
3.8.3.4.6	Temperature Effects Analysis	3.8-58
3.8.3.4.7	Reinforced Concrete Design	3.8-58
3.8.3.4.8	Evaluation of Radiation-Generated Heat Effects	3.8-58
3.8.3.4.9	Structural Steel Design	3.8-59
3.8.3.4.10	Polar Crane and Telescopic Jib Crane	3.8-59
3.8.3.4.11	Structural Response of Missile Barriers	3.8-59
3.8.3.4.12	Supports for Reactor Coolant System Components	3.8-59
3.8.3.5	Structural Acceptance Criteria	3.8-59
3.8.3.5.1	Functional Criteria	3.8-59
3.8.3.5.2	Reinforced Concrete Acceptance Criteria	3.8-59
3.8.3.5.3	Thermal Gradient Effects	3.8-60
3.8.3.5.4	Structural Steel Allowable Stresses and Strains	3.8-60
3.8.3.5.5	Earthquake Response of Interior Structure Related to the Requirements of Attached Equipment	3.8-60
3.8.3.5.6	Shear Response of Internal Structures	3.8-61
3.8.3.5.7	Missile Load and Pipe Break Criteria at Local Areas	3.8-61
3.8.3.5.8	Criteria for Reactor Coolant System Supports	3.8-61
3.8.3.6	Materials, Quality Control, and Special Construction Techniques	3.8-61
3.8.3.6.1	Concrete	3.8-61
3.8.3.6.2	Reinforcing Steel	3.8-62
3.8.3.6.3	Mechanical Butt Splices (Cadwelds)	3.8-62
3.8.3.6.4	Structural and Miscellaneous Steel	3.8-62
3.8.3.6.5	Effect of Radiation on Structural Materials	3.8-63
3.8.3.6.6	Quality Assurance Program	3.8-63
3.8.3.7	Testing and Inservice Inspection Requirements	3.8-63
3.8.3.7.1	Stainless Steel Liner	3.8-63

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.8.3.7.2	Testing of Internal Structure	3.8-63
3.8.3.7.3	Testing of Polar Crane	3.8-64
3.8.3.7.4	Inservice Inspection Requirements	3.8-64
3.8.3.7.5	Tests and Inspections for Reactor Coolant System Supports	3.8-64
3.8.4	OTHER SEISMIC CATEGORY I STRUCTURES	3.8-64
3.8.4.1	Description of the Structures	3.8-64
3.8.4.1.1	Safeguards Building	3.8-64
3.8.4.1.2	Auxiliary Building/Electrical and Control Building	3.8-64
3.8.4.1.3	Fuel Building	3.8-65
3.8.4.1.4	Service Water Intake Structure	3.8-65
3.8.4.1.5	Drawings	3.8-66
3.8.4.1.6	Outdoor Seismic Category I Tanks (Refueling Water Storage, Condensate Storage, and Reactor Makeup Water Storage)	3.8-66
3.8.4.1.7	Concrete Barriers Provided for Category I Structures	3.8-67
3.8.4.2	Applicable Codes, Standards, and Specifications	3.8-67
3.8.4.3	Loads and Load Combinations	3.8-67
3.8.4.3.1	Loads	3.8-67
3.8.4.3.2	Load Combinations and Acceptance Criteria for Other Seismic Category I Concrete Structures	3.8-69
3.8.4.3.3	Load Combinations and Acceptance Criteria for other Seismic Category I Steel Structures	3.8-71
3.8.4.3.4	Variable Loads	3.8-72
3.8.4.3.5	Time Dependent Loads	3.8-73
3.8.4.4	Design and Analysis Procedures	3.8-73
3.8.4.5	Structural Acceptance Criteria	3.8-73
3.8.4.5.1	General Criteria	3.8-73
3.8.4.5.2	Reinforced Concrete Acceptance Criteria	3.8-73
3.8.4.5.3	Missile Load and Pipe Break Criteria at Local Areas	3.8-74
3.8.4.5.4	Bracket or Corbel Criteria at Local Areas	3.8-74
3.8.4.6	Materials, Quality Control, and Special Construction Techniques	3.8-74
3.8.4.7	Testing and Inservice Inspection Requirements	3.8-74
3.8.5	FOUNDATIONS	3.8-74
3.8.5.1	Description of the Foundations	3.8-74
3.8.5.1.1	Foundation Mat and Supports for Containment and Internal Structure	3.8-74
3.8.5.1.2	Foundation For Other Seismic Category I Structures	3.8-75
3.8.5.1.3	Relationship Between Various Seismic Category I Structures	3.8-75
3.8.5.1.4	Underlying Foundation Material	3.8-76
3.8.5.1.5	Effects of Ground Water	3.8-76
3.8.5.1.6	Cathodic Protection	3.8-76
3.8.5.1.7	Drawings	3.8-76
3.8.5.2	Applicable Codes, Standards, and Specifications	3.8-76
3.8.5.3	Loads and Load Combinations	3.8-76
3.8.5.3.1	Foundation Design Loads and Combinations	3.8-76
3.8.5.3.2	Various Effects at Foundations	3.8-77
3.8.5.3.3	Loads Transferred from Supported Components to Foundations	3.8-77

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.8.5.4	Design and Analysis Procedures	3.8-77
3.8.5.4.1	Foundation and Supports for Containment and Internal Structure	3.8-77
3.8.5.4.2	Foundations for Other Seismic Category I Structures	3.8-78
3.8.5.4.3	Descriptions of Computer Programs	3.8-78
3.8.5.4.4	Computer Program Check	3.8-78
3.8.5.4.5	Design Variables	3.8-78
3.8.5.4.6	Lateral Load Transfer to Soil	3.8-79
3.8.5.4.7	Effects of Settlement	3.8-79
3.8.5.5	Structural Acceptance Criteria	3.8-79
3.8.5.5.1	Allowable Stresses and Strains	3.8-79
3.8.5.5.2	Foundation Settlements	3.8-79
3.8.5.5.3	Factors of Safety Against Overturning, Sliding, and Floatation	3.8-79
3.8.5.5.4	Effects of Rock Structure Interaction	3.8-80
3.8.5.6	Materials, Quality Control, and Special Construction Techniques	3.8-80
3.8.5.7	Testing and Inservice Inspection Requirements	3.8-80
	REFERENCES	3.8-80
APP. 3.8A	VISUAL WELD ACCEPTANCE CRITERIA FOR STRUCTURAL WELDING AT NUCLEAR POWER PLANTS	3.8A-1
3.9N	MECHANICAL SYSTEMS AND COMPONENTS	3.9N-1
3.9N.1	SPECIAL TOPICS FOR MECHANICAL COMPONENTS	3.9N-1
3.9N.1.1	Design Transients	3.9N-1
3.9N.1.2	Computer Programs Used in Analyses	3.9N-15
3.9N.1.3	Experimental Stress Analysis	3.9N-16
3.9N.1.4	Considerations for the Evaluation of the Faulted Condition	3.9N-16
3.9N.1.4.1	Loading Conditions	3.9N-16
3.9N.1.4.2	Analysis of the Reactor Coolant Loop and Supports	3.9N-16
3.9N.1.4.3	Reactor Coolant Loop Models and Methods	3.9N-18
3.9N.1.4.4	Primary Component Supports Models and Methods	3.9N-25
3.9N.1.4.5	Analysis of Primary Components	3.9N-27
3.9N.1.4.6	Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss of Coolant Accident	3.9N-29
3.9N.1.4.7	Stress Criteria for Class 1 Components and Component Supports	3.9N-32
3.9N.1.4.8	Analytical Methods for RCS Class 1 Branch Lines	3.9N-33
3.9N.2	DYNAMIC TESTING AND ANALYSIS	3.9N-35
3.9N.2.1	Preoperational Vibration and Dynamic Effects Testing on Piping	3.9N-35
3.9N.2.2	Seismic Qualification Testing of Safety-Related Mechanical Equipment	3.9N-35
3.9N.2.3	Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady State Conditions	3.9N-36
3.9N.2.4	Preoperational Flow-Induced Vibration Testing of Reactor Internals	3.9N-37
3.9N.2.5	Dynamic System Analysis of the Reactor Internals Under Faulted Conditions	3.9N-39

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.9N.2.6	Correlations of Reactor Internals Vibration Tests with the Analytical Results	3.9N-44
3.9N.3	ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS AND CORE SUPPORT STRUCTURES	3.9N-44
3.9N.3.1	Loading Combinations Design Transients, and Stress Limits	3.9N-45
3.9N.3.1.1	Design Loading Combinations	3.9N-45
3.9N.3.1.2	Design Stress Limits	3.9N-46
3.9N.3.2	Pump and Valve Operability Assurance	3.9N-47
3.9N.3.3	Design and Installation Details in Mounting of Pressure Relief Devices (Pressurizer Safety and Relief System)	3.9N-51
3.9N.3.4	Component Supports	3.9N-53
3.9N.4	CONTROL ROD DRIVE SYSTEM (CRDS)	3.9N-54
3.9N.4.1	Descriptive Information of CRDS	3.9N-54
3.9N.4.2	Applicable CRDS Design Specifications	3.9N-58
3.9N.4.3	Design Loads, Stress Limits, and Allowable Deformations	3.9N-59
3.9N.4.3.1	Pressure Vessel	3.9N-59
3.9N.4.3.2	Drive Rod Assembly	3.9N-60
3.9N.4.3.3	Latch Assembly and Coil Stack Assembly	3.9N-60
3.9N.4.3.4	Evaluation of Control Rod Drive Mechanisms and Supports	3.9N-62
3.9N.4.4	CRDMS Performance Assurance Program	3.9N-62
3.9N.5	REACTOR VESSEL INTERNALS	3.9N-64
3.9N.5.1	Design Arrangements	3.9N-64
3.9N.5.2	Design Loading Conditions	3.9N-67
3.9N.5.3	Design Loading Categories	3.9N-68
3.9N.5.4	Design Bases	3.9N-69
3.9N.6	INSERVICE TESTING OF PUMPS AND VALVES	3.9N-70
	REFERENCES	3.9N-70
3.9B	MECHANICAL SYSTEMS AND COMPONENTS	3.9B-1
3.9B.1	SPECIAL TOPICS FOR MECHANICAL COMPONENTS	3.9B-1
3.9B.1.1	Plant Conditions & Component Operating Conditions	3.9B-1
3.9B.1.1.1	Plant Conditions	3.9B-1
3.9B.1.1.2	Component Operating Conditions	3.9B-3
3.9B.1.2	Computer Programs Used in Analyses	3.9B-4
3.9B.1.2.3	Computer Program (SUPERPIPE)	3.9B-6
3.9B.1.2.4	LEAP, (ME101) Linear Elastic Analysis of Piping	3.9B-7
3.9B.1.2.5	LSAPS (ME214) Local Stress Analysis for Piping Systems	3.9B-8
3.9B.1.2.6	SAPCAS (ME 215) Stress Analysis for Pipe Component and Pipe Support Using Finite Element Method - Unit 2 only	3.9B-9
3.9B.1.3	Experimental Stress Analysis	3.9B-9
3.9B.1.4	Considerations for the Evaluation of the Faulted Condition	3.9B-9
3.9B.2	DYNAMIC TESTING AND ANALYSIS	3.9B-11
3.9B.2.1	Preoperational Vibration and Dynamic Effects Testing on Piping	3.9B-11
3.9B.2.1.1	Thermal Expansion Test	3.9B-11

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.9B.2.1.2	Dynamic Transient Response Testing	3.9B-12
3.9B.2.1.3	Steady State Vibrations Tests	3.9B-12
3.9B.2.1.4	Test Performance and Acceptance	3.9B-14
3.9B.2.2	Seismic Qualification Testing of Safety-Related Mechanical Equipment	3.9B-14
3.9B.2.3	Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions	3.9B-16
3.9B.2.4	Preoperational Flow-Induced Vibration Testing of Reactor Internals	3.9B-16
3.9B.2.5	Dynamic System Analysis of the Reactor Internals Under Faulted Condition	3.9B-16
3.9B.2.6	Correlations of Reactor Internals Vibration Tests with the Analytical Results	3.9B-16
3.9B.3	ASME CODE CLASS 2 AND 3 COMPONENTS AND COMPONENT SUPPORTS	3.9B-16
3.9B.3.1	Loading Combinations, Design Transients, and Stress Limits	3.9B-17
3.9B.3.1.1	Design Loading Combinations	3.9B-17
3.9B.3.2	Pump and Valve Operability Assurance	3.9B-18
3.9B.3.3	Design and Installation of Pressure Relief Devices	3.9B-22
3.9B.3.4	Component Supports	3.9B-24
3.9B.3.4.1	Nuclear piping (ASME Class 1)	3.9B-24
3.9B.3.4.2	Nuclear Piping (ASME Class 2 and 3)	3.9B-24
3.9B.3.4.3	Instrument Impulse Tubing Supports for ASME III Class 2 and Class 3 Safety Related Applications	3.9B-26
3.9B.4	CONTROL ROD DRIVE SYSTEM (CRDS)	3.9B-27
3.9B.5	REACTOR VESSEL INTERNALS	3.9B-27
3.9B.6	INSERVICE TESTING OF PUMPS & VALVES	3.9B-27
3.9B.6.1	Inservice Testing of Pumps	3.9B-27
3.9B.6.1.1	Scope	3.9B-27
3.9B.6.1.2	Test Program	3.9B-27
3.9B.6.1.3	Test Frequencies and Durations	3.9B-27
3.9B.6.1.4	Methods of Measurement	3.9B-28
3.9B.6.2	Inservice Testing of Valves	3.9B-28
3.9B.6.2.1	Scope	3.9B-28
3.9B.6.2.2	Valve Test List	3.9B-28
3.9B.6.2.3	Test Procedures	3.9B-28
	REFERENCES	3.9B-28
3.10N	SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT	3.10N-1
3.10N.1	SEISMIC QUALIFICATION CRITERIA	3.10N-1
3.10N.1.1	Qualification Standards	3.10N-1
3.10N.1.2	Performance Requirements for Seismic Qualification	3.10N-1
3.10N.1.3	Acceptance Criteria	3.10N-1
3.10N.2	METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION	3.10N-2

TABLE OF CONTENTS (Continued)

Section	Title	Page
3.10N.2.1	Seismic Qualification by Type Test	3.10N-2
3.10N.2.2	Seismic Qualification by Analysis	3.10N-3
3.10N.3	METHOD AND PROCEDURES FOR QUALIFYING SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION	3.10N-3
3.10N.4	OPERATING LICENSE REVIEW	3.10N-3
	REFERENCES	3.10N-4
3.10B	SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT	3.10B-1
3.10B.1	SEISMIC QUALIFICATION CRITERIA	3.10B-1
3.10B.2	METHODS AND PROCEDURES FOR QUALIFYING INSTRUMENTATION AND ELECTRICAL EQUIPMENT	3.10B-3
3.10B.3	METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT	3.10B-3
3.10B.4	OPERATING LICENSE REVIEW	3.10B-5
	REFERENCES	3.10B-5
3.11N	ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT	3.11N-1
3.11N.1	EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS	3.11N-1
3.11N.2	QUALIFICATION TESTS AND ANALYSIS	3.11N-1
3.11N.3	QUALIFICATION TEST RESULTS	3.11N-2
3.11N.4	LOSS OF VENTILATION	3.11N-2
3.11N.5	ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT	3.11N-2
	REFERENCES	3.11N-3
3.11B	ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT	3.11B-1
3.11B.1	EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS	3.11B-1
3.11B.2	QUALIFICATION TESTS AND ANALYSES	3.11B-4
3.11B.3	QUALIFICATION TEST RESULTS	3.11B-7
3.11B.4	LOSS OF VENTILATION	3.11B-7
3.11B.4.1	Environmental Design Basis	3.11B-7
3.11B.4.2	Limiting Environmental Conditions	3.11B-8
3.11B.4.3	Testing of Control and Electrical Equipment	3.11B-9
3.11B.5	ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT	3.11B-9
3.11B.5.1	Chemical Environment	3.11B-9
3.11B.5.2	Radiation Environment	3.11B-9
	REFERENCES	3.11B-10

TABLE OF CONTENTS (Continued)

Section	Title	Page
3A	ENVIRONMENTAL QUALIFICATION REPORT	3A-1
3A.1	INTRODUCTION	3A-1
3A.2	CPNPP QUALIFICATION PROGRAM	3A-1
3A.2.1	NSSS QUALIFICATION PROGRAM	3A-2
3A.2.2	BOP QUALIFICATION PROGRAM	3A-2
3A.2.3	CPNPP QUALIFICATION REVIEW PROGRAM	3A-2
3A.2.4	CPNPP QUALIFICATION MAINTENANCE PROGRAM	3A-3
3A.3	CPNPP COMPARISON TO NUREG-0588	3A-4
3B	COMPUTER PROGRAMS USED IN THE DYNAMIC AND STATIC ANALYSIS OF ASME CODE CLASS 2 AND 3 PIPING SYSTEMS, INCLUDING SUPPORTS FOR ASME CODE CLASS 1, 2, AND 3 PIPING	3B-1

LIST OF TABLES

Number	Title
3.2-1	QUALITY STANDARDS
3.2-2	EQUIPMENT CODE AND CLASSIFICATION LIST FLUID SYSTEM COMPONENTS
3.2-3	PLANT DRAWING VERSUS FIGURE CROSS REFERENCE
3.2-4	FIGURE v/s PLANT DRAWING CROSS REFERENCE
3.3-1	DELETED
3.4-1	EFFECT OF LIQUID STORAGE TANKS FAILURE ON THE SAFE SHUTDOWN OF THE PLANT
3.5-1	INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT
3.5-2A	SUMMARY OF THE UNIT 1 CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS
3.5-2B	SUMMARY OF THE UNIT 2 CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS
3.5-3	VALVE MISSILE CHARACTERISTICS
3.5-4	PIPING TEMPERATURE ELEMENT ASSEMBLY MISSILE CHARACTERISTICS
3.5-5	CHARACTERISTICS OF OTHER MISSILES POSTULATED WITHIN REACTOR CONTAINMENT
3.5-6	INTERNALLY GENERATED MISSILES (INSIDE CONTAINMENT)
3.5-7	HIGH TRAJECTORY MISSILE TARGETS AND TARGET AREAS
3.5-8	TORNADO-GENERATED MISSILES (HORIZONTAL MODE)
3.5-9	DELETED
3.5-10	DELETED
3.6B-1	HIGH ENERGY LINE LIST
3.6B-2	POSTULATED BREAK LOCATIONS FOR THE LOCA ANALYSIS OF THE PRIMARY COOLANT LOOP
3.6B-3	DELETED

LIST OF TABLES (Continued)

Number	Title
3.6B-4A	MASS/ENERGY RELEASES FULL POWER (INCLUDING POWER UNCERTAINTY) - 1.0 FT2 MAIN STEAM LINE BREAK (Unit 1)
3.6B-4B	MASS/ENERGY RELEASES FULL POWER (INCLUDING POWER UNCERTAINTY) - 1.0 FT^2 MAIN STEAM LINE BREAK (Unit 2)
3.6B-5	EQUIPMENT AFFECTED BY MSLB SUPERHEAT CONDITION
3.6B-6	MASS AND ENERGY RELEASE RATES 1 FT2 BREAK - FEED WATER LINE (Unit 1 and Unit 2 with original steam generators)
3.6B-7	DELETED
3.6B-8	DELETED
3.7N-1	DAMPING VALUES USED FOR SEISMIC SYSTEMS ANALYSIS FOR WESTINGHOUSE SUPPLIED EQUIPMENT
3.7B-1	DAMPING VALUES
3.7B-2	METHODS OF SEISMIC ANALYSIS USED FOR SEISMIC CATEGORY I STRUCTURES, SYSTEMS, AND COMPONENTS
3.7B-3	DYNAMIC FOUNDATION DESIGN PARAMETERS
3.7B-3A	SOIL PROPERTIES FOR SOIL STRUCTURE INTERACTION ANALYSIS OF FUEL BUILDING
3.7B-4	SEISMIC CATEGORY I STRUCTURES FOUNDATION CHARACTERISTICS
3.7B-5	SPRING CONSTANTS FOR RIGID CIRCULAR FOOTINGS RESTING ON ELASTIC HALF-SPACE
3.7B-6	SPRING CONSTANTS FOR RIGID RECTANGULAR FOOTINGS RESTING ON ELASTIC HALF-SPACE
3.7B-7	PROPERTIES OF MASS-FOUNDATION SYSTEM THEORY OF RIGID CIRCULAR FOOTINGS ON ELASTIC HALF-SPACE
3.7B-8	COEFFICIENTS FOR ESTIMATING EFFECTIVE MASS AND MASS MOMENT OF INERTIA OBTAINED FROM THEORY OF ELASTIC HALF-SPACE
3.7B-9	CONTAINMENT AND INTERNAL STRUCTURES MASS POINT NODAL COORDINATES

LIST OF TABLES (Continued)

Number	Title
3.7B-10	SAFEGUARDS BUILDING MASS POINT NODAL COORDINATES
3.7B-11	ELECTRICAL AND AUXILIARY BUILDINGS MASS POINT NODAL COORDINATES
3.7B-12	FUEL BUILDING MASS POINT NODAL COORDINATES
3.7B-12A	FUEL BUILDING RE-ANALYSIS MODEL NODAL COORDINATES
3.7B-13	SERVICE WATER INTAKE STRUCTURE MASS POINT NODAL COORDINATES
3.7B-14	CONTAINMENT AND INTERNAL STRUCTURES MASS POINT MASS DATA
3.7B-15	SAFEGUARDS BUILDING MASS POINT MASS DATA
3.7B-16	ELECTRICAL AND AUXILIARY BUILDINGS MASS POINT MASS DATA
3.7B-17	FUEL BUILDING MASS POINT MASS DATA
3.7B-17A	FUEL BUILDING RE-ANALYSIS MASS POINT DATA
3.7B-18	SERVICE WATER INTAKE STRUCTURE MASS POINT MASS DATA
3.7B-19	CONTAINMENT AND INTERNAL STRUCTURES DEGREES OF FREEDOM
3.7B-20	SAFEGUARDS BUILDING DEGREES OF FREEDOM
3.7B-21	ELECTRICAL AND AUXILIARY BUILDINGS DEGREES OF FREEDOM
3.7B-22	FUEL BUILDING DEGREES OF FREEDOM
3.7B-23	SERVICE WATER INTAKE STRUCTURE DEGREES OF FREEDOM
3.7B-24	CONTAINMENT AND INTERNAL STRUCTURES FOUNDATION SPRING CONSTANTS
3.7B-25	SAFEGUARDS BUILDING FOUNDATION SPRING CONSTANTS
3.7B-26	ELECTRICAL BUILDING FOUNDATION SPRING CONSTANTS
3.7B-27	ELECTRICAL BUILDING FOUNDATION SPRING CONSTANTS
3.7B-28	FUEL BUILDING FOUNDATION SPRING CONSTANTS

LIST OF TABLES (Continued)

Number	Title
3.7B-29	SERVICE WATER INTAKE STRUCTURE FOUNDATION SPRING CONSTANTS
3.7B-30	CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, LOWER BOUND, SSE
3.7B-31	CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, BEST ESTIMATE, SSE
3.7B-32	CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, UPPER BOUND, SSE
3.7B-33	CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, LOWER BOUND, SSE
3.7B-34	CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, BEST ESTIMATE, SSE
3.7B-35	CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, UPPER BOUND, SSE
3.7B-36	SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE
3.7B-37	SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE
3.7B-38	SERVICE WATER INTAKE STRUCTURE MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE
3.7B-39	SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS UPPER BOUND, SSE
3.7B-40	ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE
3.7B-41	ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE
3.7B-42	ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE
3.7B-43	FUEL BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE

LIST OF TABLES (Continued)

Number	Title
3.7B-44	FUEL BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS UPPER BOUND, SSE
3.7B-45	SERVICE WATER INTAKE STRUCTURE MODAL FREQUENCIES AND PARTICIPATION FACTORS UPPER BOUND, SSE
3.7B-46	CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS
3.7B-47	SAFEGUARDS BUILDING SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS
3.7B-48	ELECTRICAL AND AUXILIARY BUILDING SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS
3.7B-49	FUEL BUILDING SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS
3.7B-50	SERVICE WATER INTAKE STRUCTURE SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS UPPER BOUND SOIL CASE
3.7B-51	ELECTRICAL AND AUXILIARY BUILDINGS PEAK ACCELERATIONS FOR SSE FROM TIME-HISTORY ANALYSIS
3.7B-52	FUEL BUILDING PEAK ACCELERATIONS FOR SSE FROM TIME-HISTORY ANALYSIS
3.7B(A)-1	LIST OF COMPUTER PROGRAMS USED
3.8-1	SUMMARY TABLE FOR CATEGORY ‘I’ STRUCTURES
3.9N-1	SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS
3.9N-1A	COMPONENT CYCLIC OR TRANSIENT LIMITS
3.9N-2	LOADING COMBINATIONS FOR ASME CLASS 1 COMPONENTS AND COMPONENT SUPPORTS (EXCLUDING PIPE SUPPORTS)
3.9N-3	ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS
3.9N-4	DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND COMPONENTS AND COMPONENT SUPPORTS (EXCLUDING PIPING AND PIPE SUPPORTS)

LIST OF TABLES (Continued)

Number	Title
3.9N-5	STRESS CRITERIA FOR SAFETY RELATED ASME CLASS 2 AND CLASS 3 TANKS
3.9N-6	STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS
3.9N-7	DESIGN CRITERIA FOR ACTIVE PUMPS
3.9N-8	STRESS CRITERIA FOR SAFETY RELATED ASME CODE CLASS 2 AND CLASS 3 VALVES
3.9N-9	ACTIVE PUMPS
3.9N-10	UNIT 1 & 2 ACTIVE VALVES
3.9N-11	MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES
3.9N-12	MAXIMUM REACTOR VESSEL DISPLACEMENTS AT REACTOR VESSEL CENTERLINE
3.9N-13	MAXIMUM REACTOR VESSEL SUPPORT LOADS FOR POSTULATED PIPE RUPTURE CONDITIONS
3.9N-14A	STEAM GENERATOR LOWER SUPPORT MEMBER STRESSES FOR UNIT 1
3.9N-14B	STEAM GENERATOR LOWER SUPPORT MEMBER STRESSES FOR UNIT 2
3.9N-15A	STEAM GENERATOR UPPER SUPPORT MEMBER STRESSES FOR UNIT 1
3.9N-15B	STEAM GENERATOR UPPER SUPPORT MEMBER STRESSES FOR UNIT 2
3.9N-16A	REACTOR COOLANT PUMP SUPPORT MEMBER STRESSES FOR UNIT 1
3.9N-16B	REACTOR COOLANT PUMP SUPPORT MEMBER STRESSES FOR UNIT 2
3.9N-17	PRESSURIZER UPPER LATERAL SUPPORT MEMBER STRESSES
3.9N-18	PRIMARY PIPE RESTRAINT LOADS AND STRESSES
3.9N-19A	REACTOR VESSEL SUPPORT LOADS AND STRESSES FOR UNIT 1 SUPPORT LOADS
3.9N-19B	REACTOR VESSEL SUPPORT LOADS AND STRESSES FOR UNIT 2 SUPPORT LOADS

LIST OF TABLES (Continued)

Number	Title
3.9N-20	CRDM HEAD ADAPTOR BENDING MOMENTS
3.9N-21A	MAXIMUM STRESSES IN THE REACTOR COOLANT LOOP PIPING FOR UNIT 1
3.9N-21B	MAXIMUM STRESSES IN THE REACTOR COOLANT LOOP PIPING FOR UNIT 2
3.9N-22	ADDITIONAL LOADING COMBINATIONS FOR SAFETY AND POWER RELIEF VALVE LINES
3.9B-1A	DESIGN LOAD COMBINATIONS FOR ASME CODE CLASS 2 AND CLASS 3 COMPONENTS (EXCLUSIVE OF PIPING AND PIPING SUPPORTS)
3.9B-1B	LOADING COMBINATION AND STRESS LIMITS FOR ASME III CLASS 2 & 3 PIPING
3.9B-1C	LOADING COMBINATIONS ASME SECTION III CLASS 1, 2, AND 3 PIPING SUPPORTS
3.9B-1D	ALLOWABLE STRESSES ASME B&PV CODE SECTION III CLASS 1 PIPING SUPPORTS
3.9B-1E	ALLOWABLE STRESSES ASME B&PV CODE SECTION III CLASS 2 AND 3 PIPING SUPPORTS
3.9B-1F	USES OF CODE CASE N-318
3.9B-2	STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 VESSELS
3.9B-3	STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS
3.9B-4	DESIGN CRITERIA FOR ACTIVE PUMPS
3.9B-5	STRESS CRITERIA FOR SAFETY RELATED ASME CODE CLASS 2 AND 3 VALVES
3.9B-6	NOT USED
3.9B-7	NOT USED
3.9B-8	ACTIVE PUMPS

LIST OF TABLES (Continued)

Number	Title
3.9B-9	BENCHMARK CALCULATIONS FOR ADLPIPE COMPUTER PROGRAM
3.9B-10	ACTIVE VALVES
3.9B-11	LOCATION OF SNUBBERS IN THE CONTAINMENT, SAFEGUARDS, AUXILIARY AND FUEL BUILDINGS
3.9B-12	RATED LOAD OF MECHANICAL SNUBBERS
3.10N-1	SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY
3.10B-1	DELETED
3.11N-1	DELETED
3.11N-2	DELETED
3.11N-3	DELETED
3.11B-1	DELETED
3.11B-2	INTENTIONALLY BLANK
3.11B-3	DELETED
3.11B-4	INTENTIONALLY BLANK
3.11B-5	INTENTIONALLY BLANK
3A.3-1	CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT
3A.3-2	CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT
3A.4-1	DELETED
3B.1-1	COMPARISON OF NATURAL FREQUENCIES FOR FIGURE 3B.1-1
3B.1-2	COMPARISON OF INTERNAL MEMBER LOADS AND NODAL DEFLECTIONS COMBINED RESULTS FOR MODES 1-5 FOR FIGURE 1.0
3B.1-3	COMPARISON OF NATURAL FREQUENCIES FOR FIGURE 3B.1-2

LIST OF TABLES (Continued)

Number	Title
3B.1-4	COMPARISON OF INTERNAL MEMBER LOADS AND NODAL DEFLECTIONS COMBINED RESULTS FOR MODES 1-5 FOR FIGURE 3B.1-2
3B.1-5	COMPARISON OF SUPPORT LOADS BETWEEN NUPIPE II AND NUPIPE-SW FOR FIGURE 3B.1-3
3B.1-6	COMPARISON OF SUPPORT LOAD FOR FIGURE 3B.1-4
3B.1-7	COMPARISON OF SUPPORT REACTIONS DUE TO THERMAL EXPANSION FOR FIGURE 3B.1-5
3B.1-8	COMPARISON OF DEFLECTIONS DUE TO THERMAL EXPANSION FOR FIGURE 3B.1-5
3B.1-9	COMPARISON OF STRESS DUE TO THERMAL EXPANSION FOR FIGURE 3B.1-5
3B.1-10	COMPARISON OF INTERNAL FORCES AND MOMENTS DUE TO THERMAL EXPANSION FOR FIGURE 3B.1-5
3B.3-1	COMPARISON OF BSPLT COMPUTER PROGRAM RESULTS WITH HAND CALCULATION RESULTS
3B.5-1	COMPARISON OF PITRUST WITH FRANKLIN INSTITUTE PROGRAM, CYLNOZ, AND HAND CALCULATION
3B.5-2	COMPARISON OF PITRUST WITH REFERENCE 2 RESULTS
3B.6-1	COMPARISON OF PILUG COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS
3B.7-1	COMPARISON OF PITRIFE COMPUTER PROGRAM OUTPUT WITH STUDL-II OUTPUT
3B.7-2	COMPARISON OF PITRIFE COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS
3B.8-1	NODAL FORCE COMPARISON
3B.9-1	INPUT DATA FOR WATHAM
3B.9-2	COMPARISON OF NODAL FORCE CALCULATION AT TIME = 2.34 sec
3B.10-1	INPUT DATA FOR WATSLUG

LIST OF TABLES (Continued)

Number	Title
3B.10-2	INPUT DATA FOR WATSLUG
3B.11-1	ELBOW PROGRAM – ELBOW PROPERTIES USED FOR VERIFICATION PROBLEMS
3B.11-2	ELBOW PROGRAM – CASE 1 RESULTS
3B.11-3	ELBOW PROGRAM – CASE 2 RESULTS
3B.11-4	ELBOW PROGRAM – CASE 3 RESULTS
3B.13-1	COMPARISON OF EIGENVALUES FROM THEORETICAL RESULTS, STRUDL RESULTS, AND GT-STRUDL RESULTS (PROBLEM NO. 1)
3B.13-2	COMPARISON OF EIGENVECTORS FROM STRUDL AND GT-STRUDL (PROBLEM NO. 1)
3B.13-3	THE MEMBER FORCES FROM STRUDL AND GT-STRUDL COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS (PROBLEM NO. 2)
3B.13-4	THE JOINT LOADS (AT SUPPORTS) FROM STRUDL AND GT-STRUDL COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS (PROBLEM NO. 2)
3B.13-5	COMPARISON OF ELEMENT (RANDOMLY SELECTED) STRESSES (PROBLEM NO. 3)
3B.13-6	COMPARISON OF RESULTANT (RANDOMLY SELECTED) JOINT DISP. SUPPORTS (GLOBAL) (PROBLEM NO. 3)
3B.13-7	COMPARISON OF MEMBERS (RANDOMLY SELECTED) FORCES AND MOMENTS (PROBLEM NO.4)
3B.13-8	COMPARISON OF RESULTANT JOINT LOADS - SUPPORTS (PROBLEM NO. 4) (GLOBAL)
3B.13-9	COMPARISON OF EIGENVALUES, FREQUENCIES, AND PERIODS (PROBLEM NO. 5)
3B.13-10	COMPARISON OF EIGENVECTORS FOR FEW RANDOMLY SELECTED MODES AND JOINTS (PROBLEM NO. 5)
3B.13-11	COMPARISON OF JOINT DISPLACEMENTS AT THE FREE JOINTS FOR RANDOMLY SELECTED JOINTS (GLOBAL) (PROBLEM NO. 5)

LIST OF TABLES (Continued)

Number	Title
3B.14-1	COMPARISON OF SANDUL LOAD COMBINATIONS WITH HAND COMPUTATION
3B.14-2	COMPARISON OF MEMBER FORCES BETWEEN SANDUL AND STRUDL-SW
3B.14-3	COMPARISON OF STRESSES AND WELD SIZE BETWEEN SANDUL AND HAND COMPUTATION
3B.17-1	APE PROGRAM VERIFICATION PROBLEM COMPARISON OF DRILLED-IN ANCHOR ROAD

LIST OF FIGURES

Number	Title
3.2-1	Mechanical Symbols and Notes (M1-0200)
3.3-1	Typical Exterior Wall Opening for Tornado Dampers
3.3-2	Typical Roof Opening, Missile Barrier
3.3-3	Typical Roof Opening, Missile Barrier with Protective Grating
3.5-1	Stationary Storage Scheme Roll-Away Missile Shield
3.5-2	Roll-Away Missile Shield
3.5-3	Turbine Missile Strike Zone, Plan View
3.5-4	Deleted
3.5-5	Deleted
3.5-6	Deleted
3.5-7	Deleted
3.5-8	Deleted
3.5-9	Deleted
3.5-10	Deleted
3.5-11	Deleted
3.5A-1	Deleted
3.5B-1	Deleted
3.5B-2	Deleted
3.6B-1	Typical U-Bar Restraint
3.6B-2	Typical Pipe Whip Restraint - Crushable Bumper Type
3.6B-3	Typical Pipe Whip Restraint - Hard Type
3.6B-4	Typical Pipe Whip Restraint - Impact Absorbing Type
3.6B-5	Typical Moment Restraint and Break Exclusion Area Boundary

LIST OF FIGURES (Continued)

Number	Title
3.6B-6	Typical Pipe Whip Analysis Mathematical Model
3.6B-7	Piping Stress-Strain Characteristics
3.6B-8	Restraint Impact
3.6B-9	Location of Postulated Breaks in Main Reactor Coolant Loop
3.6B-10	Loss of Reactor Coolant Accident Boundary Limits
3.6B-11	Unit 1 Main Steam System: Loop 1 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-12	Unit 1 Main Steam System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-13	Unit 1 Main Steam System: Loop 3 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-14	Unit 1 Main Steam System: Loop 4 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-15	Unit 1 Main Steam System: Loop 1 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-16	Unit 1 Main Steam System: Loop 2 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-17	Unit 1 Main Steam System: Loop 3 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-18	Unit 1 Main Steam System: Loop 4 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-19	Unit 1 Feedwater System: Loop 1 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-20	Unit 1 Feedwater System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-21	Unit 1 Feedwater System: 1-3 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-22	Unit 1 Feedwater System: Loop 4 Inside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-23-1	Unit 1 Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-23-2	Unit 1 Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-23-3	Unit 1 Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-24-1	Unit 1 Feedwater System: Feedwater Bypass Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-24-2	Unit 1 Feedwater System: Feedwater Bypass Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-24-3	Unit 1 Feedwater System: Feedwater Bypass Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-24-4	Unit 1 Feedwater System: Feedwater Bypass Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-25-1	Unit 1 Main Steam System: M.S. Blowdown Outside Containment, Stress Node, Break Point and Resistant Locations
3.6B-25-2	Unit 1 Main Steam System: M.S. Blowdown Outside Containment, Stress Node, Break Point and Resistant Locations
3.6B-25-3	Unit 1 Main Steam System: M.S. Blowdown Outside Containment, Stress Node, Break Point and Resistant Locations
3.6B-25-4	Unit 1 Main Steam System: M.S. Blowdown Outside Containment, Stress Node, Break Point and Resistant Locations
3.6B-26	Auxiliary Feedwater System: Stress Node, Break Point and Restraint Locations
3.6B-27	Unit 1 Auxiliary Feedwater System: Loop 1 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-28	Unit 1 Auxiliary Feedwater System: Loop 2 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-29	Unit 1 Auxiliary Feedwater System: Loop 3 Outside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-30	Unit 1 Auxiliary Feedwater System: Loop 4 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-31-1	Unit 1 Auxiliary Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-31-2	Unit 1 Auxiliary Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-32	Unit 1 Auxiliary Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-33	Auxiliary Feedwater System: Stress Node, Break Point and Restraint Locations
3.6B-34	Unit 1 Feedwater System: Loop 1 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-35	Unit 1 Feedwater System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-36	Unit 1 Feedwater System: Loop 3 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-37	Unit 1 Feedwater System: Loop 4 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-38	Unit 1 Steam Generator Blowdown System: Loop 1 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-39-1	Unit 1 Steam Generator Blowdown System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-39-2	Unit 1 Steam Generator Blowdown System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-40	Unit 1 Steam Generator Blowdown System: Loop 3 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-41	Unit 1 Steam Generator Blowdown System: Loop 4 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-42	Unit 1 Steam Generator Blowdown System: Outside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-43	Unit 1 Steam Generator Blowdown System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-44	Unit 1 Steam Generator Blowdown System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-45	Unit 1 Steam Generator Blowdown System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-46	Steam Generator Blowdown System: Stress Node, Break Point and Restraint Locations
3.6B-47	Steam Generator Blowdown System: Stress Node, Break Point and Restraint Locations
3.6B-48-1	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-48-2	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-49	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-50-1	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-50-2	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-51-1	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-51-2	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-52-1	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-52-2	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-53	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-54	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-55	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-56	Unit 1 Safety Injection System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-57-1	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-57-2	Unit 1 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-58-1	Unit 1 Auxiliary Steam System: Stress Node, Break Point and Restraint Locations
3.6B-58-2	Unit 1 Auxiliary Steam System: Stress Node, Break Point and Restraint Locations
3.6B-58-3	Unit 1 Auxiliary Steam System: Stress Node, Break Point and Restraint Locations
3.6B-59-1	Unit 1 Auxiliary Steam System: Stress Node, Break Point and Restraint Locations
3.6B-59-2	Unit 1 Auxiliary Steam System: Stress Node, Break Point and Restraint Locations
3.6B-60	Unit 1 Auxiliary Steam System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-61-1	Unit 1 Auxiliary Steam System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-61-2	Unit 1 Auxiliary Steam System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-62	Unit 1 Auxiliary Steam System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-63	Unit 1 Auxiliary Steam System: Auxiliary Building, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-64-1	Unit 1 RHR System: Stress Node, Break Point and Restraint Locations
3.6B-64-2	Unit 1 RHR System: Stress Node, Break Point and Restraint Locations
3.6B-65	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-66-1	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-66-2	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-67-1	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-67-2	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-67-3	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-68-1	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-68-2	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-69	Unit 1 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-70	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-71	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-72	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-73-1	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-73-2	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-74	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-75	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-76	CVCS System: Stress Node, Break Point and Restraint Locations
3.6B-77	CVCS System: Stress Node, Break Point and Restraint Locations
3.6B-78	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-79	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-80	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-81	Unit 1 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-82-1	Unit 1 CVCS System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-82-2	Unit 1 CVCS System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-82-3	Unit 1 CVCS System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-83	Unit 1 CVCS System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-84	Unit 1 CVCS System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-85	Unit 1 CVCS System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-86	Unit 1 CVCS System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-87-1	Unit 1 CVCS System: Auxiliary Building, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-87-2	Unit 1 CVCS System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-88-1	Unit 1 CVCS System: Safeguards Building, Stress Node, Break Point and Restraint Locations
3.6B-88-2	Unit 1 CVCS System: Safeguards Building, Stress Node, Break Point and Restraint Locations
3.6B-89	Reactor Coolant Loop #1, Stress Node, Break Point and Restraint Locations
3.6B-90	Reactor Coolant Loop #2, Stress Node, Break Point and Restraint Locations
3.6B-91	Reactor Coolant Loop #3, Stress Node, Break Point and Restraint Locations
3.6B-92	Reactor Coolant Loop #4, Stress Node, Break Point and Restraint Locations
3.6B-93-1	Deleted
3.6B-93-2	Deleted
3.6B-94	Deleted
3.6B-95	Deleted
3.6B-96A	Circumferential Pipe Break with Full Separation Jet Core
3.6B-96B	Longitudinal Pipe Break Jet Core
3.6B-96C	Jet Core Region Geometry for a Circumferential Pipe Break with Full Seperation
3.6B-96D	Effect of Irreversible Losses on Jet Subcooling
3.6B-96E	Effect of Irreversible Losses of Asymptotic Area Ratio
3.6B-97	Not Used
3.6B-98	Not Used
3.6B-99	Not Used

LIST OF FIGURES (Continued)

Number	Title
3.6B-100	Unit 2 Main Steam System: Loop 1 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-101	Unit 2 Main Steam System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-102	Unit 2 Main Steam System: Loop 3 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-103	Unit 2 Main Steam System: Loop 4 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-104	Unit 2 Main Steam System: Loop 1 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-105	Unit 2 Main Steam System: Loop 2 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-106	Unit 2 Main Steam System: Loop 3 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-107	Unit 2 Main Steam System: Loop 4 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-108	Unit 2 Feedwater System: Loop 1 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-109	Unit 2 Feedwater System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-110	Unit 2 Feedwater System: Loop 3 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-111	Unit 2 Feedwater System: Loop 4 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-112	Unit 2 Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-113	Unit 2 Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-114	Unit 2 Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-115	Unit 2 Feedwater System: Feedwater Bypass Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-116	Unit 2 Feedwater System: Feedwater Bypass Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-117	Unit 2 Feedwater System: Feedwater Bypass Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-118	Unit 2 Feedwater System: Feedwater Bypass Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-119	Unit 2 Main Steam System: Main Steam Blowdown Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-120	Unit 2 Main Steam System: Main Steam Blowdown Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-121	Unit 2 Main Steam System: Main Steam Blowdown Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-122	Unit 2 Main Steam System: Main Steam Blowdown Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-123	Not Used
3.6B-124	Unit 2 Auxiliary Feedwater System: Loop 1 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-125	Unit 2 Auxiliary Feedwater System: Loop 2 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-126	Unit 2 Auxiliary Feedwater System: Loop 3 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-127	Unit 2 Auxiliary Feedwater System: Loop 4 Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-128	Unit 2 Auxiliary Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-129	Unit 2 Auxiliary Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-130	Unit 2 Auxiliary Feedwater System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-131	Not Used
3.6B-132	Unit 2 Feedwater System: Loop 1 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-133	Unit 2 Feedwater System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-134	Unit 2 Feedwater System: Loop 3 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-135	Unit 2 Feedwater System: Loop 4 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-136	Unit 2 Steam Generator Blowdown System: Loop 1 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-137	Unit 2 Steam Generator Blowdown System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-138	Unit 2 Steam Generator Blowdown System: Loop 2 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-139	Unit 2 Steam Generator Blowdown System: Loop 3 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-140	Unit 2 Steam Generator Blowdown System: Loop 4 Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-141	Unit 2 Steam Generator Blowdown System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-142	Unit 2 Steam Generator Blowdown System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-143	Unit 2 Steam Generator Blowdown System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-144	Unit 2 Steam Generator Blowdown System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-145	Not Used

LIST OF FIGURES (Continued)

Number	Title
3.6B-146	Not Used
3.6B-147	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-148	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-149	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-150	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and restraint Locations
3.6B-151	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-152	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-153	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-154	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-155	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-156	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-157	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-158	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-159	Unit 2 Safety Injection System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-160	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-161	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-162	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-163	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-164	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-165	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-166	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-167	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-168	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-169	Unit 2 Safety Injection System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-170	Not Used
3.6B-171	Not Used
3.6B-172	Unit 2 RHR System: Stress Node, Break Point and Restraint Location
3.6B-173	Unit 2 RHR System: Stress Node, Break Point and Restraint Location
3.6B-174	Not Used
3.6B-175	Unit 2 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-176	Unit 2 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-177	Unit 2 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-178	Unit 2 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-179	Unit 2 Reactor Coolant System: Inside Containment, Stress Node, Break and Restraint Locations
3.6B-180	Unit 2 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-181	Unit 2 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-182	Unit 2 Reactor Coolant System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-183	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-184	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-185	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-186	Not Used
3.6B-187	Not Used
3.6B-188	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-189	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-190	Not Used
3.6B-191	Not Used
3.6B-192	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-193	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations

LIST OF FIGURES (Continued)

Number	Title
3.6B-194	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-195	Unit 2 CVCS System: Inside Containment, Stress Node, Break Point and Restraint Locations
3.6B-196	Unit 2 CVCS System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-197	Unit 2 CVCS System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-198	Unit 2 CVCS System: Outside Containment, Stress Node, Break Point and Restraint Locations
3.6B-199	Unit 2 CVCS System: Auxiliary Building, Stress Node, Break Point and Restaint Locations
3.6B-200	Unit 2 CVCS System: Auxiliary Building, Stress Node, Break Point and Restaint Locations
3.6B-201	Unit 2 CVCS System: Auxiliary Building, Stress Node, Break Point and Retraint Locations
3.6B-202	Unit 2 CVCS System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-203	Unit 2 CVCS System: Auxiliary Building, Stress Node, Break Point and Restraint Locations
3.6B-204	Unit 2 CVCS System: Safeguards Building, Stress Node, Break Point and Restraint Locations
3.6B-205	Unit 2 CVCS System: Safeguards Building, Stress Node, Break Point and Restraint Locations
3.6B-206	Unit 2 CVCS System: Safeguards Building, Stress Node, Break Point and Restraint Locations
3.6B-207	Safeguards Building Main Steam and Feedwater Pipe Break Model Nodal Boundaries (4 Sheets)
3.6B-208	Safeguards Building Main Steam and Feedwater Lines Arrangement (3 Sheets)

LIST OF FIGURES (Continued)

Number	Title
3.7N-1	Multi-Degree of Freedom System
3.7N-2	Horizontal Design Response Spectra - Scaled to 0.4g Horizontal Ground Acceleration
3.7N-3	Vertical Design Response Spectra - Scaled to 0.4g Horizontal Ground Acceleration
3.7N-4	Typical Reactor Internals Horizontal Seismic Model (Response Spectrum Analysis Method)
3.7N-5	Typical Reactor Internals Vertical Seismic Structural Model (Response Spectrum Analysis Method)
3.7B-1	Design Response Spectra for Horizontal Safe Shutdown Earthquake
3.7B-2	Horizontal Response Spectra, Safe Shutdown Earthquake, 2 Percent Damping
3.7B-3	Horizontal Response Spectra, Safe Shutdown Earthquake, 5 Percent Damping
3.7B-3A	Fuel Building Re-Analysis, East-West Response Spectra, Safe Shutdown Earthquake, 5 Percent Damping
3.7B-3B	Fuel Building Re-Analysis, North-South Response Spectra, Safe Shutdown Earthquake, 5 Percent Damping
3.7B-4	Horizontal Response Spectra, Safe Shutdown Earthquake, 7 Percent Damping
3.7B-4A	Fuel Building Re-Analysis, East West Response Spectra, Safe Shutdown Earthquake, 7 Percent Damping
3.7B-4B	Fuel Building Re-Analysis, North South Response Spectra, Safe Shutdown Earthquake, 7 Percent Damping
3.7B-5	Horizontal Response Spectra, Safe Shutdown Earthquake, 10 Percent Damping
3.7B-5A	Fuel Building Re-Analysis, East-West Response Spectra, Safe Shutdown Earthquake, 10 Percent Damping
3.7B-5B	Fuel Building Re-Analysis, North-South Response Spectra, Safe Shutdown Earthquake, 10 Percent Damping

LIST OF FIGURES (Continued)

Number	Title
3.7B-6	Horizontal Response Spectra, Safe Shutdown Earthquake, 15 Percent Damping
3.7B-7	Design Response Spectra for Vertical Safe Shutdown Earthquake
3.7B-8	Vertical Response Spectra, Safe Shutdown Earthquake, 2 Percent Damping
3.7B-9	Vertical Response Spectra, Safe Shutdown Earthquake, 5 Percent Damping
3.7B-9A	Fuel Building Re-Analysis, Vertical Response Spectra, Safe Shutdown Earthquake, 5 Percent Damping
3.7B-10	Vertical Response Spectra, Safe Shutdown Earthquake, 7 Percent Damping
3.7B-10A	Fuel Building Re-Analysis, Vertical Response Spectra, Safe Shutdown Earthquake, 7 Percent Damping
3.7B-11	Vertical Response Spectra, Safe Shutdown Earthquake, 10 Percent Damping
3.7B-11A	Fuel Building Re-Analysis, Vertical Response Spectra, Safe Shutdown Earthquake, 10 Percent Damping
3.7B-12	Vertical Response Spectra, Safe Shutdown Earthquake, 15 Percent Damping
3.7B-13	Deleted
3.7B-14	Horizontal Artificial Acceleration Time History
3.7B-14A	Fuel Building Re-Analysis, East-West Artificial Acceleration Time History
3.7B-14B	Fuel Building Re-Analysis, North-South Artificial Acceleration Time History
3.7B-15	Deleted
thru	
3.7B-18	
3.7B-19	Vertical Artificial Acceleration Time History

LIST OF FIGURES (Continued)

Number	Title
3.7B-19A	Fuel Building Re-Analysis, Vertical Artificial Acceleration Time History
3.7B-20	Fuel Building Re-Analysis, East-West Power Spectral Density
3.7B-21	Fuel Building Re-Analysis, North-South Power Spectral Density
3.7B-22	Deleted
3.7B-23	Plan View, Containment and Internal Structures
3.7B-24	Elevation View, Containment and Internal Structures
3.7B-25	Plan View, Safeguards Building
3.7B-26	Elevation View, Safeguards Building
3.7B-27	Elevation View, Safeguards Building
3.7B-28	Plan View, Electrical and Auxiliary Buildings
3.7B-29	Elevation View, Electrical and Auxiliary Buildings
3.7B-30	Plan View, Fuel Building
3.7B-31	Elevation View, Fuel Building
3.7B-32	Plan View, Service Water Intake Structure
3.7B-33	Elevation View, Service Water Intake Structure
3.7B-34	Dynamic Model, Containment and Internal Structures
3.7B-35	Dynamic Model, Safeguards Building
3.7B-36	Dynamic Model, Electrical and Auxiliary Buildings
3.7B-37	Dynamic Model, Fuel Building
3.7B-37A	Dynamic Model, Fuel Building Re-Analysis
3.7B-38	Dynamic Model, Service Water Intake Structure
3.7B-39	Coefficients βx, βz and $\beta \psi$ for Rectangular Footings
3.7B-40	Determination of Overturning Moment

LIST OF FIGURES (Continued)

Number	Title
3.7B-41	Reactor Building Internal Structure, Refined
3.7B-42	Containment Building, Refined Response Spectra
3.7B-43	Containment Building, Refined Response Spectra
3.7B-44	Auxiliary Building, Refined Response Spectra
3.7B-45	Auxiliary Building, Refined Response Spectra
3.7B-46	Electrical Building, Refined Response Spectra
3.7B-47	Electrical Building, Refined Response Spectra
3.7B-48	Safeguards Building, Refined Response Spectra
3.7B-49	Safeguards Building, Refined Response Spectra
3.7B-50	Fuel Building, Refined Response Spectra
3.7B-50A	Fuel Building, Refined Response Spectra
3.7B-50AA	Fuel Building Refined Response Spectra
3.7B-50AB	Fuel Building Refined Response Spectra
3.7B-50B	Service Water Intake Structure, Refined Response Spectra
3.7B-50C	Service Water Intake Structure, Refined Response Spectra
3.7B-51	Instructure Response Spectra, Steam Generator Support, X-Earthquake, SSE, 2 Percent Damping
3.7B-52	Instructure Response Spectra, Steam Generator Support, Y-Earthquake, SSE, 2 Percent Damping
3.7B-53	Instructure Response Spectra, Steam Generator Support, Z-Earthquake, SSE, 2 Percent Damping
3.7B-54	Seismic Instrumentation Schematic Diagram
3.7B-55	Simplified Dynamic Analysis for Piping Systems
3.7B(A)-1	Maximum Amplification Curve

LIST OF FIGURES (Continued)

Number	Title
3.8-1	Structural Plant Arrangement Plan
3.8-2	Structural Plant Arrangement Section A-A
3.8-3	Structural Plant Arrangement Section B-B
3.8-4	Structural Plant Arrangement Section C-C
3.8-5	Containment Liner Details
3.8-6	Containment Liner Details
3.8-7	Containment Typical Penetration Details
3.8-8	Containment Typical Penetration Details
3.8-9	Containment Arrangement of Air Locks & Equipment Hatch
3.8-10	Containment Typical Wall Reinforcement
3.8-11	Containment Typical Dome Reinforcement
3.8-11A	Typical Wall Shear Ties Near Spring Line - El. 1000'-6"
3.8-12	Containment Typical Mat Reinforcement
3.8-13	Containment Typical Reinforcement at Penetrations
3.8-14	Containment Typical Reinforcement at Airlock & Equipment Hatch
3.8-15	Containment Internal Structure
3.8-16	Service Water Intake Structure
3.8-17	Reactor Supports, Reactor Coolant Pump & Steam Generator Vertical Supports
3.8-18	Reactor Coolant Pump Tie Rod Support & Steam Generator Lower Lateral Support
3.8-19	Unit 1 Containment Structural Integrity Test, Predicted Responses
3.8-19A	Unit 2 Containment Structural Integrity Test, Predicted Responses
3.8-20	General Arrangement, Personnel Air Lock

LIST OF FIGURES (Continued)

Number	Title
3.8-21	Personnel Air Lock, Gasket Testing Arrangement Section
3.8-22	Flow Diagram - Airlocks (M1-0245) Flow Diagram - Personnel Airlock (M2-0245)
3.8-23	Flow Diagram - Emergency Airlock (M2-0245-A)
3.8-24	Location of Mis-Oriented Supports
3.8-25	Unit 2 Reinforced Concrete Reactor Pressure Vessel Supports
3.8A-3-1	Figure 3-1
3.8A-3-2	Figure 3-2
3.8A-3-3	Figure 3-3
3.8A-3-4	Figure 3-4
3.9N-1	Reactor Coolant Loop Supports System, Dynamic Structural Model
3.9N-2	Reactor Pressure Vessel Lumped Mass Model
3.9N-3	Through-Wall Thermal Gradients
3.9N-4	Westinghouse Pump Bearing Vibration Limits Compared to Industry Standards
3.9N-4A	Control Volume for Momentum Equation
3.9N-5A	Full Length Control Rod Drive Mechanism (Unit 1)
3.9N-5B	Full Length Control Rod Drive Mechanism (Unit 2)
3.9N-6A	Full Length Control Rod Drive Mechanism Schematic (Unit 1)
3.9N-6B	Full Length Control Rod Drive Mechanism Schematic (Unit 2)
3.9N-7	Nominal Latch Clearance at Minimum and Maximum Temperature
3.9N-8	Control Rod Drive Mechanism Latch Clearance Thermal Effect
3.9N-9	Lower Core Support Assembly (Core Barrel Assembly)
3.9N-10	Upper Core Support Structure

LIST OF FIGURES (Continued)

Number	Title
3.9N-11	Plan View of Upper Core Support Structure
3.9N-12	Reactor Internals Model for DARI Variables
3.9N-13	Reactor Internals Mathematical Model for WOSTAS Variables
3.9B-1	Typical Force-Time Plot for One Leg of Piping
3.9B-2	Normalized Reference Curve
3.9B-3	Peak Forces Versus Volume for Various Leg Lengths
3.9B-4	Valve Flow Area Correction Coefficient
3.9B-5	Transient Initiation Time
3.9B-6	Positive Force Duration
3.9B-7	Negative Force Duration
3.9B-8	Typical Force, Time Traces for a Three Leg Piping System

3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 CONFORMANCE WITH NUCLEAR REGULATORY COMMISSION (NRC) GENERAL DESIGN CRITERIA

This section briefly discusses the extent to which the design criteria for the plant structures, systems, and components important to safety comply with Title 10, Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix A, General Design Criteria (GDC) for Nuclear Power Plants. Each criterion is first quoted directly from 10 CFR Part 50 and then discussed. In the discussion of each criterion, it is shown how the CPNPP design meets the criterion, and the sections of the FSAR, where more detailed information is presented to demonstrate compliance with the criterion, are referenced.

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, 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 [1]."

Discussion

Structures are identified and classified in accordance with the requirement that they be designed to withstand the effects of earthquakes, as delineated in Section 3.2.

The systems and components of the facility are classified according to their importance in the prevention and mitigation of accidents. Reactor components use the classification system developed by American National Standards Institute (ANSI) N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. Classifications and any deviations are described in Section 3.2. Each component is given a safety class designation.

The codes, standards, and quality control applicable to each type of component are identified in pertinent equipment specifications. Where applicable, design and fabrication are in accordance with the codes specified in 10 CFR Part 50, Section 55a. See Section 5.2.1.1. Alternative requirements, as provided by ASME Code Cases, are utilized at CPNPP in accordance with 10 CFR Part 50, Section 55a(a)(3). By reference to ASME Section III requirements in the procurement specifications, the use code cases by mechanical equipment suppliers requires mutual consent of the Owner or his agent and the manufacturer. The ASME Code Cases which are used for design and erection at CPNPP are identified in the appropriate mechanical design and erection specifications or the Brown & Root QA Manual; conditionally-approved Code Cases will show justification for their use, as required by NRC, in these documents. This application of ASME Code Cases is documented on the ASME Data Report Forms. For further discussion, see

the text concerning Regulatory Guide 1.85 in Section 1A(N). The quality assurance (QA) program that is implemented to ensure that safety-related systems, components, and structures perform as intended is discussed in Chapter 17. The QA program conforms to the intent of 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. Those items for which the requirements of 10 CFR Part 50, Appendix B, are met are listed in the list of quality assured items in Appendix 17A.

Chapter 14 describes initial tests to ensure performance of installed equipment commensurate with the importance of the safety function.

The component safety classifications are also shown on the flow diagrams presented with their respective sections. Records for the design, fabrication, erection, and testing of safety-related systems, components, and structures are maintained as described in Chapter 17.

Westinghouse, in accordance with the Westinghouse Nuclear Energy System Division's (WNESD's) Quality Assurance Plan, maintains (either in its possession or under its control) a complete set of records of the design, fabrication, construction, and testing of safety components which it supplies.

The Westinghouse QA program conforms to the requirements of 10 CFR Part 50, Appendix B, and is discussed in Westinghouse Commercial Atomic Power (WCAP) 8370, Revision 7A.

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 loss of 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 in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, (3) the importance of the safety functions to be performed [1]."

Discussion

The natural phenomena and their magnitude are selected in accordance with their probability of occurrence at the Comanche Peak site. The design is based on the most severe of the natural phenomena recorded for the site, with an appropriate margin to account for uncertainties in the historical data. The natural phenomena postulated in the design are presented in Sections 2.3, 2.4, 2.5, and 3.3. The design criteria for the structures, systems, and components affected by each natural phenomenon are presented in Sections 3.2, 3.3, 3.4, 3.5, 3.7, and 3.8.

Combinations of natural phenomena and plant originated accidents considered in the design are identified in Sections 3.8, 3.9, and 3.10. The importance of the safety functions is identified with the classification system developed by the American Nuclear Society (ANS) and is generally in accordance with ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. This identification and any deviations are included 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. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of the structures, systems, and components [1]."

Discussion

Comanche Peak Nuclear Power Plant is designed to minimize the probability and effect of fires and explosions. Noncombustible and fire resistant materials are used in the Containment Building, in the Control Room, and wherever practical throughout the station. Atmospheric conditions inside the Containment Building and Control Room are not of an explosive nature. Alarm and detection devices are located throughout the station and connected to the annunciation system in the Control Room.

To protect both plant equipment and personnel from fire normal fire protection is provided by water and halon fire suppression systems, hose stations, and portable fire extinguishers. The fire fighting systems are designed to ensure that rupture or inadvertent operation does not significantly impair systems important to safety.

In the design and installation of fire protection equipment, the requirements of the National Fire Protection Association (NFPA), American Nuclear Insurers (ANI), and applicable codes and regulations have been followed. The applicable codes, standards, and regulations used in the design of the Fire Protection System and plant equipment are further discussed in Section 9.5.1 and the Fire Protection Report.

Finally, fire protection and detection system reliability is ensured by periodic tests and inspections.

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. 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, the 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 basis conditions fluid system piping.[1]"

Discussion

The station's structures, systems, and components important to safety 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 loss-of-coolant accident (LOCA). Environmental conditions are described in Section 3.11.

These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that can result from equipment failures and from events and conditions outside the nuclear power unit.

Details of the design, environmental testing, and construction of these systems, structures, and components are included in Chapters 3, 5, 6, 7, 8, 9 and 10. Evaluation of the performance of safety features is contained in Chapter 15.

The leak before break methodology demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated pipe ruptures in the primary coolant loop piping and 10 inch and larger reactor coolant loop branch lines, as discussed in Sections 3.6B.2.5.1 and 3.6B.2.5. Implementation of this technology eliminates the need for respectively pipe whip restraints and jet impingement barriers. Containment design, emergency core cooling and environmental qualification requirements are not influenced by this modification.

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

"Structures, systems, and components important to safety shall not be shared between 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 [1]."

Discussion

The facilities which have shared systems or components are tabulated in Section 1.2 along with references to the appropriate sections containing specific design details. Construction of the second unit will not impair the safety functions of any shared system or component.

The sharing of systems or components does not impair system capability to perform safety functions, e.g., the orderly shutdown and cooldown of one unit in the event of an accident in the other unit.

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 [1]."

Discussion

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to do the following:

1. To preclude significant fuel damage during normal core operation and operational transients (Condition I)[a] or during transient conditions arising from occurrences of moderate frequency (Condition II).[a]

2. To ensure return of the reactor to a safe state following infrequent faults (Condition III)[a] with only a small fraction of fuel rods damaged, although sufficient fuel damage can occur to preclude resumption of operation without considerable outage time.

3. To ensure that the core is intact, with acceptable heat transfer geometry, following transients arising from occurrences of limiting faults (Condition IV).[a]

Chapter 4 discusses the design bases and design evaluation of reactor components, including the fuel, reactor vessel internals, and reactivity control systems. 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 Condition I and II 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 [1]."

Discussion

When the reactor is critical, prompt compensatory reactivity feedback effects are ensured by the negative fuel temperature effect (Doppler effect) and by the operational limit on moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is ensured by the inherent design using low-enrichment fuel; the limit on moderator temperature coefficient of reactivity is ensured by administratively controlling the dissolved absorber concentration or by burnable poison.

These reactivity coefficients are discussed in Section 4.3.

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 [1]."

(a.) Defined by ANSI N18.2-1973

Discussion

Power oscillations of the fundamental mode are inherently eliminated by the negative power coefficient of reactivity.

In addition, oscillations, due to xenon spatial effects, in the radial, diametral, and azimuthal overtone modes are heavily damped because of the inherent design and because of the negative power coefficient of reactivity.

Also, oscillations, caused by xenon spatial effects, in the axial first overtone mode may occur; however, assurance that fuel design limits are not exceeded by xenon axial oscillations is provided by reactor trip functions using the measured axial power imbalance as an input.

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

Xenon stability control is discussed in Section 4.3.

3.1.2.4 Criterion 13 - Instrumentation and Control

"Instrumentation and control 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 [1]."

Discussion

To ensure adequate safety, instrumentation and control systems are provided to monitor and control significant variables over their anticipated range for all conditions in the Reactor Core, Reactor Coolant System (RCS), steam and power conversion system, radioactive waste systems, engineered safety features (ESF) systems, and the Containment Building. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the Control Room in close proximity to the controls which maintain the indicated parameters in the proper range.

The quantity and types of process instrumentation provided ensures 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 [1]."

Discussion

The RCS pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, and to maintain the stresses within applicable stress limits (Section 5.2.). Also, reactor coolant pressure boundary (RCPB) materials and selection and fabrication techniques ensure a low probability of gross rupture of significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture, which are discussed in Sections 3.6 and 3.7.

The leak before break methodology demonstrates that the probability of rupturing primary coolant piping is extremely low under design basis conditions. It has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated ruptures in the primary coolant loop piping, as discussed in Section 3.6B.2.5.1. Implementation of this technology eliminates the need for primary coolant loop piping whip restraints and jet impingement barriers. Containment design, emergency core cooling and environmental qualification requirements are not influenced by this modification.

The system is protected from overpressure by means of pressure-relieving devices as required by applicable codes.

In conclusion, the RCS boundary has provisions for inspection, testing, and surveillance of critical areas to assess the structural and leaktight integrity (Section 5.2). For the reactor vessel, a material surveillance program conforming to applicable codes is provided (Section 5.3).

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 [1].”

Discussion

The design pressure and temperature for each component in the RCS and associated auxiliary, control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

In addition, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and of integrated hydrostatic testing of assembled components.

Chapter 5 discusses the RCS design.

3.1.2.7 Criterion 16 - Containment Design

“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 [1].”

Discussion

A steel-lined, reinforced concrete containment structure encloses the entire RCS and is designed to withstand the pressures and temperatures resulting from a spectrum of postulated LOCAs and secondary system breaks.

The Emergency Core Cooling System cools the reactor core and limits the release of radioactive materials to the environment.

Next, to ensure its integrity, the Containment Spray System is incorporated in the containment design.

The Containment Spray System is designed to function after a LOCA to reduce the pressure inside the containment to near atmospheric pressure and to remove fission product activity from the containment atmosphere.

To sum up, the Containment structure and ESF systems are designed to safely sustain internal and external environmental conditions that may reasonably be expected to occur during the life of the plant, including both short- and long-term effects following a LOCA (See Sections 6.2, 6.5, 15.6, and 3.8.1).

3.1.2.8 Criterion 17 - Electric Power Systems

“An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming 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 sources, including the batteries, and the onsite electrical 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 a loss of all onsite alternating current power supplies and the other offsite electrical 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 loss-of-coolant accident 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 electrical power supplies [1]."

Discussion

To ensure the functioning of safety-related systems, an onsite electrical power system and an offsite electrical power system are provided and are designed with adequate independence, capacity, redundancy, and testability.

Independence is provided for both systems by physical separation of components and cables to minimize vulnerability of redundant systems to single credible accidents. For details of separation, see Chapter 8.

Offsite power is provided to the safety-related systems through the startup transformers. The startup transformers are independent of each other and are each capable of supplying the Class 1E systems of each unit. One transformer is connected to the 345 kilovolts (kV) Switchyard and the other to the 138 kV system, so that a single component failure will not prevent the safety-related systems from performing their function (Section 8.2).

The onsite AC source of electrical power consists of four diesel generators, one connected and used exclusively for Unit 1 train A, one connected and used exclusively for Unit 1 train B, one connected and used exclusively for Unit 2 train A, and one connected and used exclusively for Unit 2 train B. One diesel generator is capable of supplying sufficient power for the operation of the minimum safety features required for the unit during a postulated LOCA. However, during a postulated LOCA, both diesel generators start automatically, connect to the two safety features buses of the unit, and sequentially start the safety-related loads.

The two safety features buses and their associated diesel generators are so arranged that a failure of a single component will not prevent the power supply systems from performing their function (Section 8.3).

In conclusion, for each unit, two DC battery systems are provided in physically separated rooms; each one is adequate to supply the DC control power required for the engineered safety features. Failure of a single component in this system does not impair control of the minimum safety features required to maintain each unit in a safe condition. See Section 8.3 for failure mode analysis.

3.1.2.9 Criterion 18 - Inspection and Testing of Electric Power System

"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 operational 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 [1]."

Discussion

All Class 1E power systems are designed to permit periodic inspection and testing. The testing procedure simulates a loss of bus voltage to start a diesel generator and connect it to the bus. Full load testing of the diesel generator is performed by manually synchronizing the generator to the normal power supply. Under conditions as close to design as practical, these tests, therefore, prove the capability of the power supply system to assess the continuity of the systems and the conditions of components.

Inspections and tests which constitute compliance with this criterion are included in Section 8.3.

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 loss-of-coolant accidents. 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 [1]."

Discussion

Safe occupancy of the Control Room during abnormal conditions is provided for in the design. The Control Room is in a seismic Category I structure (See Figures 1.2-33 and 13.5-1). Adequate shielding is provided for the Control Room in the event of a design basis accident (DBA), as detailed in Sections 12.1 and 15.6. The Control Room Air-Conditioning System features redundant equipment, and radiation and smoke detectors, with appropriate alarms and interlocks. Provisions are made for Control Room air to be recirculated through high-efficiency particulate air (HEPA) and charcoal filters, as discussed in Sections 6.4, 6.5 and 9.4.

The Control Room is continuously occupied by qualified operating personnel under all operating and accident conditions.

In the unlikely event that access to the Control Room is restricted, the hot shutdown panel, local control stations, or manual operation of critical components can be used to effect hot shutdown from outside the Control Room for an extended period.

Before evacuation takes place the reactor can be manually tripped and neutron level and control rod position can be verified. By use of appropriate procedures and equipment, the unit can also be brought to cold shutdown conditions. (See Section 7.4.)

3.1.3 PROTECTION AND REACTIVITY CONTROL SYSTEMS

3.1.3.1 Criterion 20 - Protection System Functions

"The protection systems shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, and 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 [1]."

Discussion

A fully automatic protection system with appropriate redundant channels is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of Institute of Electrical and Electronic Engineers (IEEE) 279-1971 and IEEE 379-1972. The Reactor Protection System automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Set points are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the control rod drive mechanisms (CRDM) of all full-length rod cluster control assemblies (RCCA). This causes the rods to insert by the force of gravity, rapidly reducing the reactor power output. The response and adequacy of the protection system has been verified by analysis of anticipated transients.

Refer to Chapter 7, Instrumentation and Controls, for additional information regarding actuating devices to the protection system.

The ESF Actuation System automatically initiates emergency core cooling and other safeguards functions by sensing accident conditions using redundant analog channels measuring diverse variables. In addition, manual actuation of safeguards can be performed where ample time is available for operator action. In either case, the ESF Actuation System automatically trips the reactor on manual or automatic safety injection signal generation.

3.1.3.2 Criterion 21 - Protection Systems Reliability and Testability

"The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred [1]."

Discussion

The protection system is designed for high functional reliability and inservice testability. It is designed with sufficient redundancy and independence to permit acceptable reliability of operation. System design also includes the capability to perform periodic testing of channels independently while the reactor is in operation.

Compliance with this criterion is discussed in detail in Sections 7.2.2.2.3 and 7.3.2.2.5.

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 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 [1]."

Discussion

Protection system components are designed and arranged so that there is no loss of safety function in the environment accompanying any emergency situation in which the components are required to function. Various means are used to accomplish this. Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a wide variety of postulated accidents. Also, diverse protection functions will automatically terminate an accident before intolerable consequences can occur.

Automatic reactor trips are based upon neutron flux measurements, reactor coolant loop (RCL) temperature measurements, pressurizer pressure and level measurements, and reactor coolant pump power underfrequency and undervoltage measurements. Trips can also be initiated manually or by safety injection signal (Section 7.2).

With regard to the ESF Actuation System, a LOCA safety injection signal can be obtained manually or by automatic initiation from two diverse sets of signals:

1. Low pressurizer pressure

2. High containment pressure

For a steam line break accident, diversity of safety injection signal for automatic actuation is provided by:

1. Low compensated steamline pressure

2. High containment pressure (for a steam line break inside the Containment Building)

For information regarding the independence of the protection system logic, see Section 7.3.

High quality components, conservative design, applicable quality control, inspection, calibration, and tests are used to guard against common mode failure. Qualification testing is performed on

the various safety systems to demonstrate functional operation at normal and post-accident conditions of temperature, humidity, pressure, chemistry, and radiation for specified periods, if required. See Section 3.11 for further details. Typical protection system equipment is subjected to tests under simulated seismic conditions using conservatively large accelerations and applicable frequencies. The test results indicate no loss of the protection function. See Section 3.10 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 [1]."

Discussion

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip channel is designed on the de-energize-to-trip principle so that loss of power, disconnection, open-channel faults, and the majority of internal channel short-circuit faults cause the channel to go into its tripped mode.

The protection system is discussed in Sections 7.2 and 7.3.

3.1.3.5 Criterion 24 - Separation of Protection and Control Systems

"The protection system shall be separated from control systems to the extent that failure of a 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 [1]."

Discussion

The protection system is separate and distinct from the control systems. Control systems may be dependent on the protection system because control signals are derived from protection system measurements where applicable. These signals are transferred to the control system by isolation amplifiers, which are classified as protection components. The adequacy of system isolation has been verified by testing under conditions of postulated credible faults. The failure of a 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. See Chapter 7 for details. Distinction between channel and train is made in this discussion. The removal of a train from service is allowed only during testing of the train.

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 control rods [1]."

Discussion

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by full-length 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. In the postulated accidental withdrawal (assumed to be initiated by a control malfunction), flux, temperature, pressure, level, and flow signals are independently generated. Any of these signals (trip demands) can operate the breakers to trip the reactor.

Analyses of the effects of possible malfunctions are discussed in Chapter 15. These analyses show that for postulated dilution during refueling, cold shutdown, hot standby, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of dilution, and to initiate reboration before the shutdown margin is lost. The analyses also 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 acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions [1]."

Discussion

Two reactivity control methods are provided. These are rod control cluster assemblies (RCCA) 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 full-length 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 full-length control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

The boron system maintains the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCA are presented in Chapter 4; the operation is discussed in Chapter 7. The means of controlling the boric acid concentration are 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 [1]."

Discussion

The facility is provided with means of making the core subcritical and maintaining it at that level under any anticipated conditions and with an appropriate margin for contingencies. These means are discussed in detail in Chapters 4 and 9. Combined use of the rod cluster control assemblies and the chemical shim permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. Upon trip for this determination, the single highest worth control cluster is assumed to be stuck full-out upon trip.

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 [1]."

Discussion

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values that prevent rupture of the RCS boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The maximum positive reactivity insertion rates for the withdrawal of RCCA and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCA and of the Chemical and Volume Control System (CVCS). Technical specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in Section 4.3. The capability of the CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 15.

Assurance of core cooling capability following Condition IV accidents, such as rod ejections, steam line break, and similar accidents, is given by keeping the RCPB stresses within faulted condition limits as specified by applicable ASME codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of necessary safety 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 [1]."

Discussion

The protection and reactivity control systems are designed to ensure extremely high probability of performing their required safety functions in any anticipated operational occurrences. Likely failure modes of system components are designed to be safe modes. Equipment used in these systems is designed, constructed, operated, and maintained with a high level of reliability. In addition, loss of power to the protection system results in a reactor trip. Details of system design are covered in Chapter 7; also refer to the discussions of GDC 20 through 25.

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 [1]."

Discussion

RCPB components are designed, fabricated, inspected, and tested in conformance with ASME Boiler and Pressure Vessel (B&PV) Code, Section III. All components are classified according to ANSI N18.2-1973 except as noted in Sections 3.2 and 5.2 and are accorded the quality measures appropriate to the classification. The design bases and evaluations of RCPB components are discussed in Chapter 5.

Leakage from components is detected by an increase in the amount of makeup of water required to maintain a normal level in the pressurizer.

The reactor vessel closure joint is provided with a temperature-monitored leakoff between double gaskets. Leakage into the containment is drained to the Containment Building sump.

Leakage is also detected by measuring both the airborne activity and the rate at which condensate is drained from the Containment Building recirculation units. Monitoring the inventory of reactor coolant in the system at the pressurizer, volume control tank, and coolant drain collection tanks makes available an accurate indication of integrated leakage. The RCPB leakage detection system is discussed in Section 5.2.5.

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 [1]."

Discussion

Close control is maintained over material selection and fabrication for the RCS to ensure that the boundary behaves in a nonbrittle manner. Those RCS materials which are exposed to the coolant are corrosion resistant, stainless steel, or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR Part 50, Appendix G.

As part of the reactor vessel specification, certain requirements which are not specified by the applicable ASME codes are performed, as follows:

1. Ultrasonic Testing

   In addition to code requirements, the performance of a 100 percent volumetric ultrasonic test of reactor vessel plate for shear wave and a post-hydro-test ultrasonic map of all welds in the pressure vessel are required. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the codes is also required to preclude interpretation problems during inservice inspection.

2. Radiation Surveillance Program

   In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens. These programs evaluate the effect of radiation on the fracture toughness of reactor vessel steels based on the measured change in reference transition temperature and fracture mechanics measurements. These measurements are performed in accordance with American Society Testing Material (ASTM) E 185-1982, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, and the requirements of 10 CFR Part 50, Appendix H.

3. Material chemistry (copper, nickel, phosphorous, sulfur, and vanadium) of the reactor vessel core region is controlled to reduce sensitivity to embrittlement, which is caused by 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 reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generator are governed by the requirements of ASME Codes (see Chapter 5.0).

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

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 leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel [1]."

Discussion

The design of the RCPB provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel (including the nozzle to reactor coolant piping welds and the top and bottom heads), and external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing pressure boundary component integrity. The RCPB is periodically inspected under the provisions of ASME B&PV Code, Section XI.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates forging, weldments, and associated heat-treated zones are performed in accordance with 10 CFR Part 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements. Samples of reactor vessel plate materials are retained and cataloged in case future engineering development shows the need for further testing.

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 are used to confirm the allowable limits calculated for all operational transients.

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 reactor operation [1]."

Discussion

The CVCS provides a 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 the preset level. The high pressure centrifugal charging pumps provided are capable of supplying the required makeup and reactor coolant seal injection flow when power is available from either onsite or offsite electric power systems. These pumps also serve as high head safety injection pumps. Functional reliability is ensured by the provision of standby components ensuring a safe response to probable modes of failure. Details of system design are included in Sections 6.3 and 9.3; 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 [1]."

Discussion

The Residual Heat Removal (RHR) System in conjunction with the steam and power conversion system is designed to transfer the fission production decay heat and other residual heat from the reactor core within acceptable limits. The Auxiliary Feedwater System provides backup for the steam and power conversion system in this function. The Auxiliary Feedwater System is described in Section 10.4.9.

The crossover from the steam and power conversion system to the RHR System occurs at approximately 350 degrees Fahrenheit (°F) and 425 pounds per square inch gauge (psig).

Suitable redundancy at temperatures below approximately 350°F is accomplished with the two RHR pumps (located in separate compartments with means available for draining and monitoring of leakage), the two heat exchangers, and the associated piping, cabling, and electric power sources. The RHR System is capable of operating on either onsite or offsite electrical power systems.

Suitable redundancy at temperatures above approximately 350°F is provided by the four steam generators and associated piping systems.

Details of system design are in Section 5.4.7.

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 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 [1]."

Discussion

An Emergency Core Cooling System (ECCS) is provided to cope with any LOCA in the plant design basis. Abundant cooling water is available in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to ensure that clad metal-water reaction is limited to less than 1 percent. Adequate design provisions are made to ensure performance of the required safety functions even with a single failure.

Details of the capability of the systems are included in Section 6.3. An evaluation of the adequacy of the system functions is included in Chapter 15. Performance evaluations are conducted in accordance with 10 CFR Part 50, Section 50.46, and Appendix K.

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 [1]."

Discussion

Design provisions facilitate access to the critical parts of the injection nozzles, pipes, and valves for visual inspection and for nondestructive inspection where such techniques are desirable and appropriate. The design is in accordance with the requirements of ASME B&PV Code, Section XI, Subarticle IWA-1500.

The components outside the Containment Building are accessible for leaktightness inspection during operation of the reactor.

Details of the inspection program of the ECCS are discussed in Section 6.3 and in the technical specifications.

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 leaktight 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 [1]."

Discussion

Each active component of the ECCS can be individually actuated on the normal power source and can be transferred to the emergency power source at any time during appropriate plant periodic tests.

Tests can also be performed during shutdown to demonstrate proper automatic operation of the ECCS.

The details of the ECCS testing program are included in Section 6.3.

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 loss-of-coolant accident 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 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 [1]."

Discussion

Two separate containment spray recirculation trains described in Section 6.2.2, each with 100 percent capacity, remove heat from the Containment Building following certain design basis accidents. Each train contains two separate pumps, one heat exchanger, and seven spray headers, and each system is fed from its individual electrical Class 1E bus. Each Class 1E bus is connected to a separate offsite power source and is also connected to its individual onsite power source as described in Section 8.3. Containment isolation valves separate all components from the containment penetrations.

Finally, the containment heat removal system is designed to ensure that the failure of any single active component, assuming the availability of either onsite or offsite power exclusively, does not prevent the system from accomplishing its planned safety function.

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.[1]"

Discussion

The containment heat removal system design permits appropriate periodic inspection of the components as described in Section 6.2.2. The essential equipment of the Containment Spray System is outside the Containment Building, except for risers, distribution header piping, spray nozzles, and the containment sump.

The containment sump, spray piping, and nozzles can be inspected during shutdowns. Those portions of the containment spray suction piping between the sump and the Refueling Water Storage Tank as well as associated equipment outside the Containment Building are accessible for inspection.

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 leaktight 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 [1]."

Discussion

System piping, valves, pumps, heat exchangers, and other components of the containment heat removal system are arranged so that each component can be tested periodically for operability, including transfer to emergency power sources. The delivery capability of the Containment Spray System is tested periodically, to the extent practical, up to the last isolation valve before the spray nozzles. The delivery capability of the spray nozzles is tested periodically by blowing low pressure air through the nozzles and verifying the flow. The Containment Spray Systems are tested for operational sequence as close to the design as practical (see Section 6.2.2).

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 electrical power system operation (assuming offsite 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 [1]."

Discussion

The Containment Spray System includes a chemical additive subsystem using a basic sodium hydroxide solution to enhance post-accident fission product removal efficiency as described in Section 6.5.2. The unit is equipped with two independent spray systems supplied from separate buses, as described in Chapter 8, and either system alone can provide the iodine removal capacity for which credit is taken as described in Section 15.6.

Post-accident combustible gas control is ensured by a Hydrogen Purge System described in Section 6.2.5.

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 [1]."

Discussion

The containment atmosphere cleanup systems are designed and located so that they can be inspected periodically, as detailed in Section 6.5.

3.1.4.14 Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight 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 [1]."

Discussion

The containment atmosphere cleanup system can be tested as follows:

The operation of the spray pumps can be tested by recirculation through a test line to the Refueling Water Storage Tank (RWST). The system valves can be operated through their full travel, and the system can be checked for leaktightness during testing. See Section 6.5 for details. Power transfer is described in Chapter 8.

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 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 [1]."

Discussion

The cooling water system for safety-related functions consists of the Station Service Water System (SSWS) and the Component Cooling Water System (CCWS).

The SSWS removes heat from the component cooling heat exchangers.

The CCWS is a closed system. It is designed to remove residual heat from the RCS, cool the letdown flow to the CVCS, cool safety-feature heat loads, and dissipate rejected heat from various plant components. The ultimate heat sink used to dissipate rejected heat from the reactor facility during normal and emergency shutdown conditions is the Safe Shutdown Impoundment (SSI).

Both the SWS and CCWS have two flow loops with redundant pumps, heat exchangers, and piping arrangements. The system is designed to meet the required safety function so that no single failure impairs cooling of essential equipment. See Section 9.2 for details.

The ultimate heat sink has the capability to ensure either the simultaneous shutdown and cooldown of both units or the shutdown and cooldown of one unit simultaneously with the dissipation of post- accident heat from the other unit. Both systems are operable from either the offsite power system or the onsite diesel generators.

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 [1]."

Discussion

The essential components are located in accessible areas and are interchanged periodically to enable inspection to be performed. Appropriate design and layout features to allow periodic visual inspection are also incorporated. The system (the Station Service Water System and the Component Cooling Water System), therefore, can be inspected completely to find and correct incipient malfunctions. See Section 9.2 for details.

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 leaktight 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 loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources [1]."

Discussion

The design provides for testing of operating system components, which are interchanged periodically, for operability and functional performance.

Redundancy and isolation are provided to allow periodic pressure and functional testing of the system as a whole, including the functional sequence that initiates system operation. At any time during reactor operation, each component of the system can be mechanically connected to the preferred power source to demonstrate operability.

3.1.5 REACTOR CONTAINMENT

3.1.5.1 Criterion 50 - Containment Design Basis

"The reactor containment structure, including access openings, 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 loss-of-coolant accident. 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 energy from metal water and other chemical reactions that may result from degraded 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 [1]."

Discussion

The Containment Building is designed to withstand the pressures and temperatures resulting from a spectrum of LOCAs and secondary system ruptures.

Vital Containment Building subcompartments, such as the steam generator compartment, the reactor cavity, and the pressurizer compartment, are designed to withstand, with a safe margin, peak differential pressures resulting from postulated hot-leg, cold-leg, and pressurizer line breaks.

The assumptions used, other details of the containment pressure temperature transient analysis, and the subcompartmental differential pressure analysis are presented in Section 6.2.1. The structural details are described in Section 3.8. See Section 6.2.2 for containment heat removal; Section 3.8 for access openings; and Sections 3.8, 6.2.4, and 8.3 for penetrations.

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 [1]."

Discussion

The containment liner material meets the requirements of the Proposed Standard Code for Concrete Reactor Vessels and Containment (April 1973) developed by the joint American Concrete Institute (ACI)-ASME Technical Committee on Concrete Pressure Components for Nuclear Service, which is made up of ACI Committee 359 and the ASME B&PV Code, Section III, Division 2, Subgroup on Concrete Components.

Piping penetrating the containment and forming part of the containment pressure boundary meets the requirements of the ASME B&PV Code, Section III, Division 1, particularly paragraph NE 1131 and Subarticle NE 2300.

These requirements ensure conformance to GDC 51.

For additional details, refer to Section 3.8.

NOTE: Inspection of attachment welds during hydrostatic testing, as required by the ASME B&PV Code was not performed on all type MV containment penetrations for Unit 1.

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 [1]."

Discussion

The containment is designed and constructed, and the necessary equipment is provided, to permit periodic integrated leakage-rate tests during plant lifetime, in accordance with reduced pressure-test program requirements of 10 CFR 50, Appendix J. For additional details, see Section 6.2.6 and the technical specifications.

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 leaktightness of penetrations which have resilient seals and expansion bellows [1]"

Discussion

In accordance with the requirements stated in Section 6.2.6, the design of the containment permits the periodic inspection, surveillance, and testing of the leaktightness of the containment and its penetrations which have resilient seals and expansion bellows.

3.1.5.5 Criterion 54 - Piping Systems Penetrating Containment

"Piping systems penetrating 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 [1]."

Discussion

Piping systems penetrating the primary reactor containment are provided with containment isolation valves.

Fittings are provided to permit periodic leakage-rate testing of isolation valves to ensure that leakage is within the acceptable limit. (see the technical specifications).

Penetrations that must be closed for containment isolation have redundant valving and associated apparatus as described in Section 6.2.4. Each valve is tested periodically during normal operation or during shutdown conditions to ensure its operability when needed.

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 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 containment; or

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

"Isolation valves outside 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, use characteristics, and physical characteristics of the site environs [1]."

Discussion

Each line that is part of the RCPB and penetrates the containment is provided with isolation valves meeting this criterion. For a detailed description of the Containment Isolation System, see Section 6.2.4. Instrument lines are designed in accordance with the requirements of NRC Regulatory Guide 1.11, Instrument Lines Penetrating Primary Reactor Containment.

3.1.5.7 Criterion 56 - Primary Containment Isolation

"Each line that connects directly to the containment atmosphere and penetrates 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 containment; or

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

"Isolation valves outside 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 [1]."

Discussion

Each line that connects directly to the containment atmosphere and penetrates the containment is provided with redundant containment isolation barriers, except where it is demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable. The failure of one barrier does not result in the loss of isolation capability. Details are provided in Section 6.2.4 and on the corresponding figures listed in Table 6.2.4-1.

The isolation system for each line is designed to fail in a safe mode by designing the air-operated valves and solenoid-operated valves to fail in the direction of greatest safety.

Motor-operated valves, on the other hand, fail in the mode in which they are when failure occurs. Different power sources for each valve in series ensures that isolation is not defeated by a single failure.

3.1.5.8 Criterion 57 - Closed System Isolation Valves

"Each line that penetrates 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, or 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 [1]."

Discussion

Each line that penetrates containment and is not connected directly to the containment atmosphere or is not part of the RCPB has at least one isolation valve located outside the Containment Building near the penetration. Details are provided in Section 6.2.4, and on the corresponding figures listed in Table 6.2.4-1.

3.1.6 FUEL AND RADIOACTIVITY 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 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 [1]."

Discussion

Waste handling systems are incorporated in the facility design for processing and/or retention of radioactive wastes for normal operation and anticipated operational occurrences. Controls and monitoring are provided to ensure that releases during normal operation do not exceed a few percent of the limits of 10 CFR Part 20 and yield offsite doses within the numerical guides for design objectives and limiting conditions of operation set forth in 10 CFR Part 50, Appendix I.

Section 9.4 describes the Primary Plant Ventilation System and non-ESF exhaust units which satisfy GDC-60.

Chapter 11 describes the radioactive waste processing systems' design criteria, holdup capacities, and estimated releases of radioactive effluents to the environment. Compliance with 10 CFR Part 50, Appendix I, is described in Appendix 11A.

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 (1) to assure adequate safety under normal and postulated accident conditions. These systems shall be designed 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 [1]."

Discussion

The spent fuel pool, Spent Fuel Pool Cooling and Cleanup System, fuel storage and handling system, radioactive waste processing systems, and other systems that contain radioactivity are designed as follows to ensure adequate safety under normal and postulated accident conditions:

1. Components are designed and located so that appropriate periodic inspection and testing can be performed.

2. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Section 12.3.

3. Individual components that contain significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems. Radioactive waste management is discussed in detail in Chapter 11.

4. The Spent Fuel Pool Cooling and Cleanup System provides cooling to remove residual decay heat from the fuel stored in the spent fuel pool and is designed with redundancy and testability to ensure continued heat removal. A purification loop is provided to remove fission product activity. The Spent Fuel Pool Cooling and Cleanup System is described in Section 9.1.

5. The spent fuel pool is designed so that no postulated accident can cause excessive loss-of-coolant inventory.

6. The Primary Plant Ventilation System is designed to filter the exhaust from the fuel storage and handling, radioactive waste and other systems which may contain radioactivity. The non-ESF exhaust units which satisfy GDC-61 are described in Section 9.4.

The piping connected to the fuel pool is designed so that a significant loss of fuel pool water does not occur because of a pipe rupture. Level instrumentation indicates a reduction in fuel pool water level, and redundant sources of fuel pool water are available.

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 [1].”

Discussion

Criticality in new fuel and spent fuel storage areas is prevented by both physical separation of fuel assemblies and the presence of borated water in the spent fuel storage pool. For the high density Region I racks in Spent Fuel Pool 1 (SFP1) and Spent Fuel Pool 2 (SFP2), the space between storage positions within the rack module and between individual rack modules is blocked to prevent insertion of fuel assemblies. With respect to criticality, there are no administrative restrictions on placement of spent fuel within the high density Region I racks in SFP1 and SFP2. For the high density Region II racks in SFP1 and SFP2, there is no space between storage positions within the rack module and the space between individual rack modules is blocked to prevent insertion of fuel assemblies. Placement of fuel in the high density Region II racks in SFP1 and SFP2 is administratively controlled based on initial enrichment and minimum burnup requirements. Placement of fuel in the spent fuel pool outside of a rack module, such as a gap between the rack module and pool wall or within the oversized inspection cells, is administratively controlled. A discussion of the SFP Region I and Region II oversized inspection cells is contained in the Technical Specification Bases 3.7.17. For high density Region I racks (any storage configuration) and high density Region II racks (1 out of 4 storage

configuration and 2 out of 4 storage configuration), fuel assembly spacing is such that subcriticality is ensured ($k_{eff} \leq 0.95$) even if assemblies are immersed in unborated water. For high density Region II racks (3 out of 4 storage configuration and 4 out of 4 storage configuration) fuel assembly spacing is such that subcriticality is ensured ($k_{eff} \leq 0.95$ taking credit for soluble boron and $k_{eff} < 1.0$ assuming unborated water). Criticality prevention and criticality considerations are discussed in Sections 9.1 and 4.3, respectively. CPNPP complies with 10CFR50.68(b) in lieu of maintaining a system capable of detecting a criticality event as described in 10 CFR 70.24.

The containment refueling cavity of each unit has additional interim storage space for one (1) low density rack. For the low density rack in the refueling cavity, the space between storage positions within the rack module is blocked to prevent insertion of fuel assemblies. There are no administrative restrictions on placement of spent fuel within the low density rack. Fuel assembly spacing is such that subcriticality is ensured ($k_{eff} \leq 0.95$) even if assemblies are immersed in unborated water.

3.1.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

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

Discussion

Monitoring systems are provided to alarm on excessive temperature or low water level in the spent fuel pool so that appropriate safety actions can be initiated by operator action.

Radiation monitors and alarms are also provided to warn personnel of impending excessive levels of radiation or airborne activity. The Area Radiation Monitoring System is described in Section 12.3.4, and the Process Radiation Monitoring System is detailed in Section 11.5.

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 loss-of-coolant accidents 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 [1]."

Discussion

The Containment Building atmosphere is monitored continuously during normal and transient operations, using particulate, iodine, and gaseous monitors. Under post-accident conditions, samples of the containment atmosphere provide data of existing airborne radioactive concentrations within the Containment Building. Radioactivity levels contained in the facility effluent discharge paths and in the plant environs are continuously monitored during normal and accident conditions by the Process Radiation Monitoring System described in Section 11.5. Provisions for monitoring the plant areas for radioactivity are included in the Area Radiation Monitoring System described in Section 12.3.4. In addition to the installed detectors, periodic surveys are conducted using portable equipment as discussed in Section 12.5.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 SEISMIC CLASSIFICATION

3.2.1.1 Seismic Category I

The plant structures, Reactor Coolant System, engineered safety features, and safety-related systems and components are identified and classified in accordance with the seismic requirements of General Design Criterion 2 of Appendix A to 10 CFR Part 50, General Design Criteria for Nuclear Power Plants [5]. NRC Regulatory Guide 1.29 [2] designates those structures, systems, and components which must remain functional during the safe shutdown earthquake (SSE) as seismic Category I items.

Specifically, all seismic Category I structures, systems, and components must withstand the effects of the SSE and ensure the following:

1. The integrity of the reactor coolant pressure boundary

2. The capability to shut down the reactor and maintain it in a safe shutdown condition, or

3. The capability to prevent or mitigate the consequences of incidents which can result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100

The following structures, systems, and components of the Comanche Peak Nuclear Power Plant (CPNPP), including their foundations and supports, are designated as seismic Category I and are designed to withstand the effects of the SSE.

The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 are applied to all activities affecting the safety-related functions of these structures, systems, and components as shown in Table 17A-1.

3.2.1.1.1 Seismic Category I Structures

1. Containment Buildings including internal structures

2. Safeguards Buildings including diesel generator room and emergency switchgear room

3. Auxiliary Building

4. Electrical and Control Building

5. Fuel Building

6. Service Water Intake Structure

7. Safe Shutdown Impoundment Dam

8. Refueling Water Storage Tanks and Associated Piping Tunnels

9. Reactor Makeup Water Storage Tanks and Associated Piping Tunnels

10. Condensate Storage Tanks and Associated Piping Tunnels

3.2.1.1.2 Seismic Category I Mechanical Systems and Components

All, or portions, of the following mechanical systems or components are seismic Category I as described in Appendix 17A and Table 17A-1:

17A SYSTEM

NO.	SYSTEM
1.	Reactor Coolant System (RCS)
2.	Chemical and Volume Control System (CVCS)
3.	Boron Thermal Regeneration Subsystem (BTRS)
4.	Safety Injection System (SIS)
5.	Residual Heat Removal (RHR) System
6.	Boron Recycle System (BRS)
7.	Containment Spray System (CSS)
8.	Containment Isolation System
9.	Combustible Gas Control System
10.	Component Cooling Water System (CCWS)
11.	Station Service Water System (SSWS)
12.	Main Steam, Reheat and Steam Dump System
13.	Auxiliary Feedwater System
14.	Steam Generator Feedwater System
15.	Diesel Generator, Fuel Oil, and Auxiliary Systems
16.	Spent Fuel Pool Cooling and Cleanup System
17.	Liquid Waste Processing System (LWPS)
18.	Gaseous Waste Processing System (GWPS)
20.	Demineralized Water Makeup System
21.	Vents and Drains System
22.	Containment Ventilation Systems

17A SYSTEM

NO.	SYSTEM
23.	Control Room Air-Conditioning System
24.	Safeguards Building HVAC System
25.	Fuel Building Ventilation System
26.	Diesel Generator Building Ventilation System
27.	Uncontrolled Access Area Ventilation System
28.	Primary Plant Ventilation System
29.	Auxiliary Building HVAC System
30.	Service Water Intake Structure Ventilation System
31.	Chilled Water Systems
32.	Process Sampling System
32 a.	Post Accident Sample System
33.	Fuel Handling Equipment
34.	Containment Equipment
35.	Miscellaneous Handling Equipment
39.	Fire Protection System
40.	Plant Gas System
42.	Tornado Venting Components
43.	Compressed Air Systems
45.	Potable and Sanitary Water System
49.	Pipe Whip Restraints
51.	Uninterruptible Power Supply (UPS) Area Air-Conditioning System

3.2.1.1.3 Seismic Category I Electrical Systems and Components

All, or portions, of the following electrical systems or components are seismic Category I as described in Appendix 17A and Table 17A-1:

17A SYSTEM

NO.	SYSTEM
37.	Electrical Equipment
38.	Radiation Monitoring System
41.	Instrumentation and Control Equipment

3.2.1.1.4 Structures and Systems of Mixed Category

None of the plant structures are classified as partially seismic Category I; however, certain structural items within seismic Category I structures are classified as seismic Category II or non-seismic as appropriate. See Table 17A-1, item 36, for typical structural classifications. The boundaries of seismic Category I portions of systems are shown on the piping and instrumentation diagrams in appropriate sections of the FSAR. Seismic category II piping segments located inside a non-seismic building are described in Section 3.2.2.d.

3.2.1.2 Seismic Category II

Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any seismic Category I system or component required to satisfy the requirements of C.1.a through C.1.q of Regulatory Guide 1.29 to an unacceptable safety level or could result in incapacitating injury to occupants of the control room (See Figure 13.5-1) are designated seismic Category II and are designed and constructed so that the SSE would not cause such failure. The pertinent quality assurance requirements of Appendix B to 10CFR50 are applied to all activities affecting the seismic requirements of those systems and components.

Insulation (for in-Containment piping) and/or some non-nuclear-safety portions of the fluid systems listed in 3.2.1.1.2 are designated seismic Category II in Table 17A-1. NNS portions of the HVAC systems listed in 3.2.1.1.2 are also designated seismic Category II in Table 17A-1.

In addition, all, or portions, of the following mechanical systems or components are seismic Category II as described in Appendix 17A and Table 17A-1:

17A SYSTEM

NO.	SYSTEM
6.	Boron Recycle System (BRS)
9.	Combustible Gas Control System
17.	Liquid Waste Processing System (LWPS)
18.	Gaseous Waste Processing System (GWPS)
19.	Solid Waste Management System (SWMS)

17A SYSTEM

NO.	SYSTEM
25.	Fuel Building Ventilation System
28.	Primary Plant Ventilation System
29	Auxiliary Building HVAC System
32.	Process Sampling System
32.a	Post Accident Sample System
32.b	Secondary Sampling System
33.	Fuel Handling Equipment
34.	Containment Building Miscellaneous Equipment
35.	Miscellaneous Handling Equipment
39.	Fire Protection System
42.	Tornado Venting Components
43.	Compressed Air Systems
46.	Condensate System
47.	Auxiliary Steam System
48.	Steam Generator Blowdown & Cleanup System
49.	Pipe Whip Restraints
50.	Meteorological Instrumentation
51.	Uninterruptible Power Supply (UPS) And Distribution Rooms System
52.	Turbine Plant Cooling Water System
53.	Condensate Polishing System
54.	Condenser Vacuum and Waterbox Priming System
55.	Heater Drains System
56.	Chemical Feed System
58.	Chlorination Systems

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

1. General Classification

The system quality group classification applies to fluid systems which are directly depended on to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary or which are relied on to permit reactor shutdown and to maintain the reactor in the safe shutdown condition. These fluid systems are also depended on to contain radioactive material.

Fluid system components important to safety are classified in accordance with the ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants [4] classification except as described below. This classification system is compatible with requirements of NRC Regulatory Guide 1.26 [1] and is submitted as an alternate acceptable method of meeting the intent of NRC Regulatory Guide 1.26. Quality standards generally applicable to the safety classifications are given in Table 3.2-1. Clarification of the application of quality standards on a system basis is provided in Appendix 17A.

2. Safety Class Definitions

Fluid system components are classified as Safety Class 1, Safety Class 2, Safety Class 3, and non-nuclear-safety (NNS) in accordance with their importance to nuclear safety. This importance, as established by the assigned safety class, shall be applied in the design, materials, manufacture or fabrication, assembly, erection, construction, and operation. A single system may have components in more than one safety class.

The definitions of safety classes listed apply to fluid pressure boundary components and the reactor Containment. Supports that have a nuclear safety function shall be the same safety class as the components that they support. Selection of loading combinations and design methods for supports is the responsibility of the designer.

a. Safety Class 1

Safety Class 1 applies to components whose failure could cause a Condition III or Condition IV loss of reactor coolant as defined in ANSI N18.2 [4]. Condition III occurrences include incidents any one of which may occur during the lifetime of a particular plant. Condition IV occurrences are faults that are not expected to occur but are postulated because their consequences would include the potential for release of significant amounts of radioactive material. Condition IV faults are the most drastic which must be designed against and thus represent the limiting design case. All components located within the reactor coolant pressure boundary (as defined by 10CFR50.2) are Safety Class 1 as required by 10CFR50.55a with the exception of the 3/4” instrument lines and other 3/4” piping connected to the pressurizer above the normal water level and the pressurizer relief and safety valve discharge line. See Section 5.2.1.1 for additional information.

b. Safety Class 2

Safety Class 2 applies to the reactor containment and to the following components:

1. Components of the reactor coolant pressure boundary not in Safety Class 1

2. Components of safety systems that are necessary to remove heat directly from the reactor or reactor containment, to circulate reactor coolant of any safety system purpose, to control within the reactor containment radioactivity released, or to control hydrogen in the reactor containment. A safety system (in this context) is any system that is necessary to shut down the reactor, cool the core or cool another safety system or the reactor containment (after an accident), or it is any system that contains, controls, or reduces radioactivity released in an accident. Only those portions of a system that are designed primarily to accomplish one of those functions, or the failure of which could prevent accomplishing one of those functions, are included.

c. Safety Class 3

Safety Class 3 is applied to those components not in Safety Class 1 or Safety Class 2:

1. The failure of which would result in release to the environment of radioactive gases normally required to be held for decay, or that are necessary to:

   (a) Provide or support any safety system function

   (b) Control outside the reactor containment airborne radioactivity released in an accident, or

   (c) Remove decay heat from spent fuel

d. Non-Nuclear-Safety

Non-nuclear-safety (NNS) applies to portions of the nuclear power plant not covered by Safety Classes 1, 2, or 3 that can influence safe, normal operation or that may contain radioactive fluids. Design of non-nuclear-safety components shall be to applicable industry codes and standards.

The piping Class G designation is used to identify those non-nuclear safety related (NNS) piping and plumbing lines which are not located in Seismic Category I structures.

The piping Class 5 designation is used to identify those non-nuclear safety (NNS) piping and plumbing lines which are located in Seismic Category I structures. Class 5 piping is designed as seismic Category II or non-seismic. Based on

specific routing, all non-seismic Class 5 lines larger than 2" (larger than 4" for air filled copper tubing) are evaluated for their capability to reduce the functioning of Seismic Category I systems and components as defined in position C.1.a through C.1.q of Regulatory Guide 1.29 to an unacceptable level as the result of an SSE and are seismically supported where required. In some special cases as noted in Table 17A-1, Class 5 lines 2" and smaller are designated as NNS, seismic Category II and seismically supported. As such, all activities affecting the design and construction of Seismic Category II systems are subject to the pertinent quality assurance requirements of Appendix B to 10CFR50.

High energy piping segments, which are listed in FSAR Section 3.7B.2.8 as located in the Turbine Building, are designated class 5 piping and classified as seismic Category II. These piping segments are seismically analyzed for break postulation and seismic qualification. Additional analyses are performed to demonstrate that unacceptable interactions of these piping segments with non-Category I structures/components will not occur during a seismic event.

Class 5 lines, which are determined as not reducing the functioning of the systems and components described above to an unacceptable degree, and Class G lines are fabricated and installed in accordance with applicable industry codes and standards.

All, or portions of the systems or components that are Seismic Category II are described in Appendix 17A and Table 17A-1.

3. Radioactive Waste Management System (RWMS)

The RWMS designation is used to identify the boundaries of the radioactive waste management system on applicable CPNPP flow diagrams, as the RWMS does not match the functional system boundaries. The RWMS is considered to begin at the interface valve(s) in each line from other systems provided for collecting wastes that may contain radioactive materials and to terminate at the point of controlled discharge to the environment, at the point of recycle back to storage for reuse in the reactor, or at the point of storage of packaged solid wastes prior to shipment offsite to a licensed burial ground. The RWMS portion of the steam generator blowdown system begins at, but does not include, the outermost containment isolation valve on the blowdown line and terminates at the point of controlled discharge to the environment, at the point of the interface with other liquid waste systems, or at the point of recycle back to the secondary system.

The radioactive waste management system (RWMS) includes portions of the following systems and system which interface with them:

LWPS	Liquid Waste Processing System
GWPS	Gaseous Waste Processing System
SWMS	Solid Waste Management System
CVCS	Chemical and Volume Control System

BRS	Boron Recycle System
SGBS	Steam Generator Blowdown System
SFPCCS	Spent Fuel Pool Cooling and Cleanup System

The piping and components comprising the RWMS are classified seismic category I, II or NONE, and ANS Safety Class 2, 3 or NNS based on their functional requirements as defined in Chapter 11 and as shown in Appendix 17A, Table 17A-1.

3.2.3 PLANT DRAWINGS

Tables 3.2-3 and 3.2-4 correlate FSAR figure numbers to plant drawings. In the electronic FSAR, figures are not located after the tables and can only be accessed/viewed by the use of links. These links are located 1) in the List of Figures portion for each chapter Table of Contents, 2) in FSAR text/tables which reference the specific figure and, 3) in Table 3.2-4 (only for FSAR figures that are also Station drawings (i.e., “E” and “M” series drawings)). FSAR Tables 3.2-3 and 3.2-4 provide a cross reference of station drawing numbers to FSAR figure numbers (only for station drawings that are also FSAR figures.) Links to FSAR figures that have multiple sheets will take you to the first sheet. Once you are linked to the first sheet you can step through the remaining sheets of the figure.

3.2.3.1 Flow Diagrams

The safety classes and their interfaces are depicted in the flow diagrams. Coordinates are provided to all piping and instrument drawings (P&ID’s) for locating equipment and interconnections from figure to figure. For meaning of mechanical symbols and abbreviations, see Figure 3.2-1.

Table 3.2-3 include listing of Unit 1 and Common flow diagrams and of Unit 2 flow diagrams and their respective FSAR Figure Numbers. These drawings provide the details required by Regulatory Guide 1.70, Revision 2, for their respective system sections and for FSAR Sections 3.9 (active pumps and valves), 6.2.4 (containment isolation), Section 6.4 (control room habitability) and for Chapter 7 (instrumentation & controls) which refer to them. The flow diagrams also provide the correlation for ANSI Safety Class 1, 2, and 3 and for NNS fluid system components in Table 17A-1.

Table 3.2-4 provides a numerical listing of FSAR Figure Numbers and their respective flow diagrams.

The flow diagrams contain the following significant information for their respective system:

1. All major components, piping, points where connecting systems and subsystems tie together, and instrumentation associated with the system and subsystems.

2. Safety Classes and their interfaces.

3. Tag numbers for equipment referenced in text and tables throughout the FSAR.

4. The extent of systems located in Containment and the extent of system located Seismic Category I buildings.

5. Other details required by RG 1.70, Revision 2.

The flow diagrams contain the following information which is not significant:

1. Line numbers, manual valve/damper numbers (not referenced in text or tables), and air flow rates on HVAC drawings (except for fan capacity and control room ESF filtration air flows (Figure 9.4-1, Sheets 1 & 2)).

2. Drawing notes not related to system operation or function and not related to RG 1.70, Revision 2 content requirements.

3. Reducer sizes and/or their use attached directly to equipment.

4. Other trivial details which satisfy regulatory requirements and commitments, have no potential safety impact, and was not the basis for the NRC safety review as documented in the SER/SSERs.

3.2.3.2 One Line Diagrams

The safety classification of electrical systems is depicted on plant one line diagrams. Table 3.2-3 include listing of Unit 1 and common one line diagrams and of Unit 2 one line diagrams and their respective FSAR figure numbers.

One line diagrams provide the details required by Regulatory Guide 1.70, Revision 2, for AC and DC power distribution systems.

The one line diagrams contain the following significant information for AC and DC power distribution system.

1. Power supply feeders.

2. Loads supplied from each bus.

3. Interconnections between buses, buses and loads, and buses and supplies.

4. Interconnection between safety-related and non safety-related buses.

5. Power distribution equipment capacities.

6. Safety related equipment identification.

7. Power distribution equipment and buses shared between units.

8. Power distribution circuit protective devices.

9. Other details required by RG 1.70, Revision 2.

The one line diagrams contain the following information which is not significant for the purposes of FSAR figures.

1. Cable numbers.

2. Circuit breaker/fuse numbers

3. Protective device, circuit breaker/fuse/relay, manufacturer type and part numbers.

4. Bus compartment and circuit numbers.

5. Control circuit details showing control and indicating devices, e.g., control switches, elector switches, alarms, status lights, and indicating lights.

6. Schematic and wiring drawing references.

7. Notes not related to power distribution system operation/function and not related to RG 1.70 content requirements.

8. Other trivial details which satisfy regulatory requirements and commitments, have no potential safety impact, and was not the basis for the NRC safety review as documented in the SER/SSERs.

3.2.3.3 Radiation Zone and Shield Thickness Drawings

These drawings depict the radiation protection design features for the entire plant. The information depicted in these drawings provides the details required by Regulatory Guide 1.70, Revision 2 with respect to radiation protection design features. These figures are used to trigger engineering ALARA evaluations of proposed plant changes and thus help to ensure that collective station radiation exposure as well as individual radiation exposure remains ALARA throughout the life of the plant.

Table 3.2-3 includes a listing of Unit 1 and Common radiation zone and shield thickness drawings and of Unit 2 radiation zone and shield thickness drawings and their respective FSAR figure numbers.

Table 3.2-4 provides a numerical listing of FSAR Figure Numbers and their respective radiation zone and shield thickness drawings.

The radiation zone and shield thickness drawings contain the following significant information to describe the CPNPP radiation protection design features.

1. Room and Area Design Radiation Zones

2. Shield Wall Thickness

3. Room and Equipment Layouts

4. Room and Area Access Control Features

5. Area and Process Radiation Monitor Locations

REFERENCES

1. NRC Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Revision 3, February 1976, U.S. Nuclear Regulatory Commission.

2. NRC Regulatory Guide 1.29, Seismic Design Classification, Revision 2, February 1976, U.S. Nuclear Regulatory Commission.

3. ASME B&PV Code, Section III, Nuclear Power Plant Components, American Society of Mechanical Engineers, 1974.

4. ANSI N18.2, Nuclear Safety Criteria For The Design of Stationary Pressurized Water Reactor Plants, 1973.

5. Code of Federal Regulations, 10 CFR Part 50, Appendix A, General Design Criteria.

6. ASME B&PV Code, Section VIII, Pressure Vessels, Division 1 and Division 2, July 1974.

7. ANSI B31.1-1973, Power Piping, American National Standards Institute, 1973.

8. API Standard 650, including Revision 1, Welded Steel Tanks for Oil Storage, February 1984.

9. AWWA D-100, Welded Steel Tanks for Water Storage, 1984.

TABLE 3.2-1
QUALITY STANDARDS

(Sheet 1 of 2)

Components	Safety Class 1 (Note 1)	Safety Class 2 (Notes 2 and 5)	Safety Class 3 (Notes 2 and 5)	NNS Class (Notes 3 and 4)
Pressure Vessels	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 1, Components NB-3300	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 2, Components NC-3300, NC-3200	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 3, Components ND-3300 and ND-3800	ASME B&PV Code, Section VIII, Division 1
Atmospheric Storage Tanks	N/A	N/A	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 3, Components Article ND-3800	ASME B&PV Code, Section VIII, Division 1; API 620; or AWWA D-100.
0-15 PSIG Tanks	N/A	N/A	N/A	API Standard API-620 or 650
Steel-lined Concrete Storage Tanks	N/A	ACI-318	ACI-318	N/A
Supports	ASME B&PV Code Section III, Nuclear Power Plant Components, Class 1, Components Article NF	ASME B&PV Code Section III, Nuclear Power Plant Components, Class 2, Article NF	ASME B&PV CODE Section III, Nuclear Power Plant Components, Class 3, Article NF, ANSI B31.1	ANSI B31.
Piping	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 1, Components NB-3600	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 2, Components NC-3600	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 3, Components ND-3600	ANSI B31.1, Power Piping

TABLE 3.2-1
QUALITY STANDARDS

(Sheet 2 of 2)

Components	Safety Class 1 (Note 1)	Safety Class 2 (Notes 2 and 5)	Safety Class 3 (Notes 2 and 5)	NNS Class (Notes 3 and 4)
Pumps	ASME B&P V Code, Section III, Nuclear Power Plant Components, Class 1, Components NB-3400	ASME B&PV Code, Section III, Nuclear Power Plant Components Class 2, Components NC-3400	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 3, Components ND-3400	Mfrs. Standards
Valves	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 1, Components NB-3500	ASME B&PV Code, Section III, Nuclear Power Plant Components Class 2, Components NC-3500	ASME B&PV Code, Section III, Nuclear Power Plant Components, Class 3, Components ND-3500	ANSI B31.1

Notes:

1. Pressure vessels which are part of the reactor coolant pressure boundary meet the requirements of 1971 Version of the ASME B&PV Code, with application of all Addenda through and including the Summer 1972 Addenda. Pumps, valves and piping which are part of the reactor coolant pressure boundary meet the requirements of the 1971 Version, with application of all Addenda through and including the Winter 1972 Addenda. Later Code revisions may be used optionally in accordance with requirements of 10 CFR 50.55a.

2. See Reference 3. Code Class 2 and 3 components purchased prior to the effective date of Reference 3 meet the requirements of the Code in effect at the time of purchase (earliest Code specified was the Code in effect on May 1972). Later Code versions, including Code Cases, may be used optionally in accordance with ASME Code implementation requirements.

3. See Reference 6. Later Code versions may be used optionally.

4. See Reference 7, 8, 9. Later Standard versions may be used optionally.

5. See Section 3.8 and Table 17A-1 (for clarification and exceptions).

TABLE 3.2-2
EQUIPMENT CODE AND CLASSIFICATION LIST FLUID SYSTEM COMPONENTS

This table has been superseded by Table 17A-1 in FSAR Appendix 17A. All information formerly provided in this table is provided there.

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 1 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
a) ONE LINE DIAGRAM						
E1-0001	CP-32	106				8.3-1
E1-0001-A	CP-18	103	E2-0001-A	CP-4	103	8.3.13
E1-0002	CP-26	107				8.3.2
E1-0002-A	CP-25	106	E2-0002-A	CP-21	104	8.3-2
E1-0002-B	CP-15	106	E2-0002-B	CP-8	104	8.3-2
E1-0003	CP-30	107	E2-0003	CP-27	107	8.3-5
E1-0003-A	CP-32	107	E2-0003-A	CP-24	107	8.3-5
E1-0003-B	CP-14	107				8.3-5
E1-0004	CP-40	107	E2-0004	CP-30	107	8.3-6
E1-0004-A	CP-31	107	E2-0004-A	CP-27	107	8.3-6
E1-0005	CP-27	106	E2-0005	CP-15	102	8.3-8
E1-0005-A	CP-23	105	E2-0005-A	CP-17	104	8.3-8
E1-0006	CP-15	96	E2-0006	CP-11	107	8.3-7
E1-0006-A	CP-27	105	E2-0006-A	CP-15	107	8.3-7
E1-0007	CP-35	107	E2-0007	CP-27	107	8.3-9
E1-0007-A	CP-35	102	E2-0007-A	CP-27	107	8.3-9
E1-0007-B	CP-37	107	E2-0007-B	CP-21	107	8.3-9
E1-0007-C	CP-42	107	E2-0007-C	CP-29	107	8.3-9
E1-0009	CP-27	107	E2-0009	CP-17	107	8.3-10
E1-0009-A	CP-24	104	E2-0009-A	CP-17	104	8.3.10
E1-0009-B	CP-8	101	E2-0009-B	CP-13	101	8.3-10
E1-0010	CP-44	105				8.3-11
E1-0010-A	CP-39	101				8.3-11
E1-0010-B	CP-42	101				8.3-11
E1-0014	CP-31	106	E2-0014	CP-15	101	8.3-12
E1-0014-A	CP-21	101	E2-0014-A	CP-13	106	8.3-12
E1-0014-B	CP-9	97				8.3-12
E1-0014-C	CP-9	101				8.3-12

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 2 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
E1-0018	CP-24	107	E2-0018	CP-19	107	8.3-15
E1-0018-A	CP-22	107	E2-0018-A	CP-13	105	8.3-15
E1-0018-B	CP-16	104	E2-0018-B	CP-12	106	8.3-15
E1-0018-C	CP-15	107	E2-0018-C	CP-13	107	8.3-15
E1-0018-D	CP-26	105	E2-0018-D	CP-18	105	8.3-15
E1-0018-E	CP-18	102	E2-0018-E	CP-11	101	8.3-15
E1-0018-F	CP-19	105	E2-0018-F	CP-14	105	8.3-15
E1-0018-G	CP-17	102	E2-0018-G	CP-10	104	8.3-15
E1-0018-H	CP-9	99	E2-0018-H	CP-8	99	8.3-15
E1-0018-J	CP-3	96	E2-0018-J	CP-4	96	8.3-15A
E1-0018-01	CP-49	107				8.3-15A
E1-0018-01A	CP-16	99				8.3-15A
E1-0018-01B	CP-31	103				8.3-15A
E1-0018-01C	CP-17	100				8.3-15A
E1-0018-01D	CP-30	99				8.3-15A
E1-0018-01E	CP-33	107				8.3-15A
E1-0018-01F	CP-12	104	E2-0018-01	CP-12	101	8.3-15A
E1-0018-01G	CP-13	99	E2-0018-01A	CP-10	100	8.3-15A
E1-0018-01H	CP-16	101	E2-0018-01B	CP-12	100	8.3-15A
E1-0018-02	CP-18	98	E2-0018-02	CP-9	98	8.3-15B
E1-0018-02A	CP-15	100	E2-0018-02A	CP-7	100	8.3-15B
E1-0018-02B	CP-4	96	E2-0018-02B	CP-3	96	8.3-15B
E1-0018-02C	CP-16	104	E2-0018-02C	CP-9	104	8.3-15B
E1-0018-02D	CP-14	104	E2-0018-02D	CP-10	104	8.3-15B
E1-0018-03	CP-25	105	E2-0018-03	CP-14	101	8.3-15C
E1-0019	CP-27	102	E2-0019	CP-18	102	8.3-14A
E1-0019-A	CP-16	107	E2-0019-A	CP-11	107	8.3-14A
E1-0020	CP-20	98	E2-0020	CP-11	100	8.3-14
E1-0020-A	CP-14	98	E2-0020-A	CP-6	98	8.3-14

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 3 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
E1-0020-B	CP-18	102	E2-0020-B	CP-6	101	8.3-14
E1-0020-C	CP-8	102	E2-0020-C	CP-8	101	8.3-14
E1-0020-D	CP-18	101	E2-0020-D	CP-9	101	8.3-14
E1-0020-E	CP-11	101	E2-0020-E	CP-5	101	8.3-14
E1-0020-F	CP-17	102	E2-0020-F	CP-10	104	8.3-14
E1-0020-G	CP-9	107	E2-0020-G	CP-7	101	8.3-14
E1-0020-H	CP-17	106	E2-0020-H	CP-13	100	8.3-14
E1-0020-J	CP-11	106	E2-0020-J	CP-5	101	8.3-14
E1-0020-K	CP-24	102				8.3-14
E1-0020-L	CP-23	103				8.3-14
E1-0022-02	CP-1	100				8.3-3
E1-0022-04	CP-1	100	E2-0022-04	CP-5	101	8.3-18
E1-0022-05	CP-2	102				8.3-4
E1-0024-03	CP-16	103	E2-0024-03	CP-7	98	8.3-15
E1-0024-03A	CP-9	100	E2-0024-03A	CP-3	96	8.3-15
b) FLOW DIAGRAM						
M1-0200	CP-27	107				3.2-1
M1-0202	CP-33	98	M2-0202	CP-22	99	10.3-1
M1-0202-01	CP-21	102	M2-0202-01	CP-16	103	10.3-1
M1-0202-01A	CP-23	105	M2-0202-01A	CP-13	102	10.3-1
M1-0202-01B	CP-12	102	M2-0202-01B	CP-8	102	10.3-1
M1-0202-02	CP-21	102	M2-0202-02	CP-16	101	10.3-1
M1-0202-03	CP-2	97	M2-0202-03	CP-2	97	10.3-1
M1-0203	CP-29	101	M2-0203	CP-14	107	10.4-9
M1-0203-01	CP-25	102	M2-0203-01	CP-15	99	10.4-9
M1-0203-01A	CP-15	102	M2-0203-01A	CP-8	96	10.4-9
M1-0204	CP-39	105	M2-0204	CP-29	105	10.4-8
M1-0204-02	CP-36	103	M2-0204-02	CP-31	104	10.4-8
M1-0204-03	CP-20	103	M2-0204-03	CP-14	104	10.4-8

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 4 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0205	CP-20	102	M2-0205	CP-13	103	10.4-13
M1-0205-A	CP-21	106	M2-0205-A	CP-14	105	10.4-13
M1-0206	CP-21	107	M2-0206	CP-15	107	10.4-11
M1-0206-01	CP-19	107	M2-0206-01	CP-14	107	10.4-11
M1-0206-02	CP-22	107	M2-0206-02	CP-12	107	10.4-11
M1-0207	CP-28	107	M2-0207	CP-19	107	10.4-14
M1-0207-A	CP-14	106	M2-0207-A	CP-13	105	10.4-14
M1-0207-B	CP-6	104	M2-0207-B	CP-3	105	10.4-14
M1-0208	CP-13	103	M2-0208	CP-10	103	10.4-14
M1-0209	CP-33	103	M2-0209	CP-19	103	10.4-14
M1-0209-A	CP-26	103	M2-0209-A	CP-18	103	10.4-14
M1-0210	CP-31	107	M2-0210	CP-32	107	10.4-5
M1-0210-A	CP-13	104	M2-0210-A	CP-8	104	10.4-5
M1-0210-B	CP-17	106	M2-0210-B	CP-16	102	10.4-5
			M2-0210-C	CP-9	105	10.4-5
M1-0210-01	CP-39	106				10.4-5
M1-0211	CP-42	103	M2-0211	CP-29	103	10.4-3
M1-0211-01	CP-22	103	M2-0211-01	CP-14	107	10.4-3
M1-0211-02	CP-7	103				10.4-3
M1-0212	CP-32	105	M2-0212	CP-18	101	10.4-15
M1-0212-A	CP-28	105	M2-0212-A	CP-22	103	10.4-15
M1-0212-B	CP-37	105	M2-0212-B	CP-16	104	10.4-15
M1-0213	CP-23	101				10.4-16
M1-0213-A	CP-11	105				10.4-16
M1-0213-B	CP-12	101				10.4-16
M1-0213-C	CP-8	95				10.4-16
M1-0213-01	CP-20	99				10.4-16
M1-0214	CP-24	100	M2-0214	CP-13	98	10.4-17
M1-0214-A	CP-23	104	M2-0214-A	CP-22	99	10.4-17

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 5 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0214-B	CP-15	102	M2-0214-B	CP-14	102	10.4-17
M1-0214-01	CP-12	101	M2-0214-C	CP-14	101	10.4-17
M1-0215	CP-18	97	M2-0215	CP-9	97	9.5-57
M1-0215-A	CP-13	97	M2-0215-A	CP-6	97	9.5-57
M1-0215-B	CP-12	95	M2-0215-B	CP-8	95	9.5-56
M1-0215-C	CP-13	95	M2-0215-C	CP-8	95	9.5-56
M1-0215-D	CP-25	106	M2-0215-D	CP-17	106	9.5-55
M1-0215-E	CP-27	106	M2-0215-E	CP-20	107	9.5-55
M1-0215-F	CP-8	106	M2-0215-F	CP-9	106	9.5-52
M1-0215-G	CP-9	106	M2-0215-G	CP-12	106	9.5-52
M1-0215-H	CP-10	98	M2-0215-H	CP-9	95	9.5-54
M1-0215-J	CP-10	98	M2-0215-J	CP-9	95	9.5-54
M1-0216	CP-45	101	M2-0216	CP-21	101	9.3-1
M1-0216-A	CP-41	105	M2-0216-A	CP-10	100	9.3-1
M1-0216-B	CP-21	102	M2-0216-B	CP-20	102	9.3-1
M1-0216-C	CP-11	103				9.3-1
M1-0216-01	CP-28	106				9.3-1
M1-0217	CP-19	100	M2-0217	CP-11	96	9.3-2
M1-0217-A	CP-19	107	M2-0217-A	CP-5	95	9.3-2
M1-0218	CP-14	94	M2-0218	CP-8	94	9.3-1
M1-0218-01	CP-20	102	M2-0218-01	CP-19	101	9.3-1
M1-0218-01A	CP-20	101				9.3-1
M1-0218-02	CP-17	102	M2-0218-02	CP-12	99	9.3-1
			M2-0218-04	CP-10	99	9.3-1
			M2-0218-04A	CP-8	98	9.3-1
M1-0219	CP-21	107	M2-0219	CP-15	103	9.3-1
M1-0219-01	CP-15	94				9.3-1
M1-0219-02	CP-20	104				9.3-1
M1-0219-02A	CP-15	104				9.3-1

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 6 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0219-03	CP-25	101				9.3-1
M1-0220	CP-52	105				9.3-1
M1-0220-01	CP-33	103	M2-0220-01	CP-12	102	9.3-1
M1-0220-01A	CP-28	107	M2-0220-01A	CP-9	101	9.3-1
			M2-0220-02	CP-24	105	9.3-1
M1-0222	CP-23	105	M2-0222	CP-21	103	10.4-20
M1-0222-A	CP-14	105				10.4-20
M1-0222-B	CP-16	103				10.4-20
M1-0222-C	CP-22	103				10.4-20
			M2-0222-01	CP-22	107	10.4-20
M1-0223	CP-18	101	M2-0223	CP-13	101	10.4-4
M1-0223-A	CP-9	107	M2-0223-A	CP-8	105	10.4-4
M1-0224	CP-17	107				9.2-4
M1-0224-A	CP-7	96				9.2-4
M1-0224-01	CP-21	97				9.2-4A
M1-0224-01A	CP-13	97				9.2-4A
M1-0224-01B	CP-18	107				9.2-4A
M1-0224-01C	CP-20	106				9.2-4A
M1-0224-01D	CP-17	106				9.2-4A
M1-0224-01E	CP-15	106				9.2-4A
M1-0224-01F	CP-7	105				9.2-4A
M1-0224-01G	CP-8	107				9.2-4A
M1-0224-01H	CP-10	106				9.2-4A
M1-0224-01J	CP-4	97				9.2-4A
M1-0224-01K	CP-2	94				9.2-4A
M1-0224-01L	CP-7	100				9.2-4A
M1-0224-01M	CP-7	103				9.2-4A
M1-0224-01N	CP-3	105				9.2-4A
M1-0225	CP-23	103				9.5-43

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 7 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0225-01	CP-13	105	M2-0225-01	CP-16	107	9.5-44
M1-0225-02	CP-23	102				9.5-45
M1-0225-02A	CP-15	104				9.5-45
M1-0225-03	CP-18	104	M2-0225-03	CP-12	103	9.5-46
M1-0225-03A	CP-8	97	M2-0225-03A	CP-5	94	9.5-46
M1-0225-04	CP-12	97				9.5-47
M1-0225-04A	CP-15	102				9.5-47
M1-0225-05	CP-12	94	M2-0225-05	CP-6	96	9.5-48
M1-0225-06	CP-34	107				9.5-61
MX-0225-07	CP-16	102				9.5-62
MX-0225-08	CP-11	101				9.5-62
MX-0225-09	CP-11	97				9.5.62
M1-0226	CP-25	107				9.2-7
M1-0227	CP-60	107				9.2-6
M1-0227-01	CP-15	105				9.2-6
M1-0227-03	CP-7	106				9.2-6
M1-0228	CP-32	101	M2-0228	CP-17	101	9.3-4
M1-0228-A	CP-22	102	M2-0228-A	CP-15	102	9.3-4
M1-0228-B	CP-14	100	M2-0228-B	CP-11	100	9.3-4
M1-0228-C	CP-10	99	M2-0228-C	CP-9	98	9.3-4
M1-0228-01	CP-20	100	M2-0228-01	CP-10	100	9.3-4
M1-0228-02	CP-7	95				9.3-4
M1-0229	CP-23	105	M2-0229	CP-19	100	9.2-3
M1-0229-A	CP-21	100	M2-0229-A	CP-14	105	9.2-3
M1-0229-B	CP-25	100	M2-0229-B	CP-15	99	9.2-3
M1-0230	CP-26	98	M2-0230	CP-19	103	9.2-3
M1-0230-A	CP-22	106	M2-0230-A	CP-6	95	9.2-3
M1-0230-B	CP-23	107				9.2-3
M1-0230-C	CP-8	103				9.2-3

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 8 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0231	CP-24	98	M2-0231	CP-16	102	9.2-3
M1-0231-A	CP-14	106	M2-0231-A	CP-15	98	9.2-3
M1-0232	CP-31	105	M2-0232	CP-22	105	6.2.2-1
M1-0232-A	CP-25	107	M2-0232-A	CP-20	107	6.2.2-1
M1-0233	CP-44	107	M2-0233	CP-20	105	9.2-1
M1-0233-A	CP-18	98	M2-0233-A	CP-8	102	9.2-1
M1-0234	CP-28	106	M2-0234	CP-11	105	9.2-1
M1-0235	CP-23	106	M2-0235	CP-17	107	9.1-13
M1-0235-01	CP-19	99				9.1-13
M1-0235-02	CP-25	106				9.1-13
M1-0236	CP-21	106	M2-0236	CP-20	103	9.3-6
M1-0236-A	CP-21	106	M2-0236-A	CP-15	103	9.3-6
M1-0236-B	CP-3	103	M2-0236-B	CP-4	106	9.3-6
M1-0236-01	CP-17	101				9.3-7
M1-0236-01A	CP-13	103				9.3-7
M1-0236-02	CP-17	97				9.3-7
M1-0236-02A	CP-13	104				9.3-7
M1-0236-03	CP-15	103	M2-0236-03	CP-16	103	9.3-7
M1-0236-04	CP-15	106				9.3-7
M1-0237	CP-49	105	M2-0237	CP-19	105	9.3-8
M1-0237-A	CP-14	95	M2-0237-A	CP-9	97	9.3-8
M1-0237-01	CP-16	101	M2-0237-01	CP-9	103	9.3-9
M1-0237-01A	CP-11	102				9.3-9
M1-0238	CP-22	107	M2-0238	CP-14	103	9.3-5
M1-0238-A	CP-12	101	M2-0238-A	CP-12	102	9.3-5
M1-0239	CP-27	101	M2-0239	CP-18	107	10.4-10
M1-0239-01	CP-16	98	M2-0239-01	CP-5	96	10.4-10
M1-0240	CP-16	107				9.2-1
M1-0240-01	CP-13	105				10.4-6

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 9 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0241	CP-32	106	M2-0241	CP-22	107	9.2-5
M1-0241-A	CP-31	105				9.2-5
M1-0241-01	CP-28	107				9.2-5
M1-0242	CP-32	105	M2-0242	CP-17	102	9.2-5
M1-0242-A	CP-19	104				9.2-5
M1-0242-B	CP-14	99				9.2-5
M1-0243	CP-26	99	M2-0243	CP-9	94	10.4-18
M1-0243-A	CP-26	101	M2-0243-A	CP-11	105	10.4-18
M1-0243-B	CP-11	99				10.4-18
M1-0243-02	CP-23	102	M2-0243-02	CP-14	103	10.4-19
M1-0244	CP-15	99	M2-0244	CP-12	99	10.4-7
M1-0244-A	CP-8	99	M2-0244-A	CP-9	99	10.4-7
M1-0244-B	CP-11	104	M2-0244-B	CP-9	100	10.4-7
M1-0244-01	CP-24	98	M2-0244-01	CP-24	98	10.4-7
M1-0244-01A	CP-14	103	M2-0244-01A	CP-15	103	10.4-7
M1-0245	CP-11	105	M2-0245	CP-11	105	3.8-22
			M2-0245-A	CP-5	99	3.8-23
M1-0250	CP-34	106	M2-0250	CP-18	104	5.1-1
M1-0251	CP-33	106	M2-0251	CP-21	103	5.1-1
M1-0253	CP-21	98	M2-0253	CP-7	102	9.3-10
M1-0253-A	CP-10	98	M2-0253-A	CP-16	103	9.3-10
M1-0254	CP-25	102	M2-0254	CP-26	105	9.3-10
M1-0255	CP-29	107	M2-0255	CP-14	107	9.3-10
M1-0255-01	CP-27	107	M2-0255-01	CP-7	96	9.3-10
M1-0255-02	CP-14	99	M2-0255-02	CP-15	107	9.3-10
M1-0256	CP-11	97				9.3-10
M1-0256-A	CP-18	102	M2-0256-A	CP-14	102	9.3-10
			M2-0256-B	CP-6	96	9.3-10
M1-0257	CP-32	107				9.3-10

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 10 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0258	CP-31	106				9.3-11
M1-0259	CP-23	99				9.3-11
M1-0259-A	CP-8	103				9.3-11
M1-0259-01	CP-10	94				9.3-11
M1-0260	CP-37	106	M2-0260	CP-21	103	5.4-6
M1-0261	CP-23	107	M2-0261	CP-16	107	6.3-1
M1-0262	CP-27	107	M2-0262	CP-21	107	6.3-1
M1-0263	CP-17	107	M2-0263	CP-17	107	6.3-1
M1-0263-A	CP-20	107	M2-0263-A	CP-7	95	6.3-1
M1-0263-B	CP-14	103	M2-0263-B	CP-13	103	6.3-1
			M2-0263-C	CP-7	105	6.3-1
M1-0264	CP-19	101	M2-0264	CP-11	101	11.2-2
M1-0265	CP-31	101				11.2-3
M1-0265-01	CP-11	98				11.2-3
M1-0266	CP-43	104				11.2-4
M1-0266-A	CP-18	107				11.2-4
M1-0266-B	CP-5	104				11.2-4
M1-0266-01	CP-12	98				11.2-4
M1-0267	CP-29	99				11.2-5
M1-0268	CP-26	106				11.2.6
M1-0268-01	CP-16	94				11.2-7
M1-0268-02	CP-23	98				11.2-8
M1-0269	CP-24	105				11.3-1
M1-0269-A	CP-19	107				11.3-1
M1-0269-B	CP-21	107				11.3-1
M1-0269-C	CP-1	107				11.3-1
M1-0269-D	CP-2	107				11.3-1
M1-0269-01	CP-22	105				11.3-1
M1-0270	CP-20	105				11.3-1

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 11 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0270-A	CP-6	94				11.3-1
M1-0280	CP-27	107				9.2-15
M1-0280-A	CP-15	100				9.2-15
M1-0281	CP-17	105				9.2-16
M1-0281-A	CP-14	96				9.2-16
M1-0281-B	CP-10	99				9.2-16
M1-0300	CP-12	107	M2-0300	CP-6	100	9.4-5
M1-0300-A	CP-10	105	M2-0300-A	CP-11	101	9.4-5
M1-0301	CP-21	98	M2-0301	CP-13	103	9.4-6
M1-0301-A	CP-17	106	M2-0301-A	CP-15	106	9.4-6
M1-0302	CP-23	107	M2-0302	CP-13	101	9.4-4
M1-0302-A	CP-13	101	M2-0302-A	CP-4	100	9.4-4
M1-0302-B	CP-17	101	M2-0302-B	CP-11	101	9.4-4
M1-0302-C	CP-8	100	M2-0302-C	CP-7	100	9.4-4
M1-0303	CP-18	100				9.4-2
M1-0303-A	CP-8	100				9.4-2
M1-0303-B	CP-15	100				9.4-2
M1-0303-C	CP-16	106				9.4-2
M1-0303-01	CP-21	104				9.4-2
M1-0304	CP-34	101				9.4-1
M1-0304-A	CP-14	99				9.4-1
M1-0304-B	CP-19	101				9.4-1
M1-0304-C	CP-4	102				9.4-1
M1-0304-D	CP-4	101				9.4-1
M1-0304-01	CP-14	105				9.4-14
M1-0304-01A	CP-8	105				9.4-14
M1-0304-01B	CP-13	105				9.4-14
M1-0305	CP-19	100				9.4-8
M1-0305-A	CP-10	100				9.4-8

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 12 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-0306	CP-30	104	M2-0306	CP-23	100	9.4-3
M1-306-A	CP-4	102				9.4-3
M1-0307	CP-45	104	M2-0307	CP-14	96	9.4-11
M1-0307-A	CP-8	102	M2-0307-A	CP-27	105	9.4-11
M1-0307-B	CP-8	105				9.4-11
M1-0307-C	CP-4	94				9.4-11
M1-0308	CP-12	101				9.4-1
M1-0308-A	CP-5	101				9.4-1
M1-0309	CP-30	101				9.4-9
M1-0309-A	CP-20	101				9.4-9
M1-0309-B	CP-11	101				9.4-9
M1-0311	CP-30	101	M2-0311	CP-19	105	9.4-12
M1-0311-A	CP-11	98	M2-0311-A	CP-6	100	9.4.12
M1-0311-B	CP-9	98	M2-0311-B	CP-7	100	9.4-12
M1-0312	CP-17	100				9.4-7
M1-0313	CP-21	104				9.4-15
c) RADIATION ZONES & SHIELD THICKNESS						
M1-3500-001	CP-01	100				12.3-4
M1-3500-002	CP-01	100				12.3-5
M1-3500-003	CP-01	100				12.3-5.1
M1-3500-004	CP-02	102				12.3-6
M1-3500-005	CP-01	100				12.3-7
M1-3500-006	CP-01	100				12.3-8
M1-3500-007	CP-01	100				12.3-9
M1-3500-008	CP-01	100				12.3-10
M1-3500-009	CP-01	100				12.3-11
M1-3500-010	CP-02	100				12.3-12
M1-3500-011	CP-01	100				12.3-13
M1-3500-012	CP-01	100				12.3-14

TABLE 3.2-3
PLANT DRAWING VERSUS F'IGURE CROSS REFERENCE

(Sheet 13 of 13)

U 1 & X Drawing	Rev.	AM	U2 Drawing	Rev.	AM	FSAR Figure No.
M1-3500-013	CP-01	100				12.3-14.1
M1-3500-014	CP-01	100				12.3-15
M1-3500-015	CP-01	100				12.3-16
M1-3500-016	CP-01	100				12.3-17
M1-3500-017	CP-01	100				12.3-18
M1-3500-018	CP-01	100				12.3-19
M1-3500-019	CP-01	100				12.3-20
M1-3500-020	CP-01	100				12.3-21.1
M1-3500-021	CP-01	100				12.3-21.2
M1-3500-022	CP-01	100				12.3-23
M1-3500-023	CP-02	100				12.3-21
			M2-3500-001	CP-01	100	12.3-23.2
			M2-3500-002	CP-01	100	12.3-23.3
			M2-3500-003	CP-02	102	12.3-23.4
			M2-3500-004	CP-01	100	12.3-23.5
			M2-3500-005	CP-01	100	12.3-23.6
			M2-3500-006	CP-01	100	12.3-23.7
			M2-3500-007	CP-01	100	12.3-23.8
			M2-3500-008	CP-01	100	12.3-22.1
			M2-3500-009	CP-01	100	12.3-22.2
			M2-3500-010	CP-01	100	12.3-23.1
			M2-3500-011	CP-01	100	12.3-22

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 1 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
3.2-1	M1-0200	
3.8-22	M1-0245	M2-0245
3.8-23		M2-0245-A
5.1-1	M1-0250	M2-0250
5.1-1	M1-0251	M2-0251
5.4-6	M1-0260	M2-0260
6.2.2-1	M1-0232	M2-0232
6.2.2-1	M1-0232-A	M2-0232-A
6.3-1	M1-0261	M2-0261
6.3-1	M1-0262	M2-0262
6.3-1	M1-0263	M2-0263
6.3-1	M1-0263-A	M2-0263-A
6.3-1	M1-0263-B	M2-0263-B
6.3-1		M2-0263-C
8.3-1	E1-0001	
8.3-2	E1-0002	
8.3-2	E1-0002-A	E2-0002-A
8.3-2	E1-0002-B	E2-0002-B
8.3-3	E1-0022-02	
8.3-4	E1-0022-05	
8.3-5	E1-0003-A	E2-0003-A
8.3-5	E1-0003	E2-0003
8.3-5	E1-0003-B	

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 2 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
8.3.6	E1-0004	E2-0004
8.3-6	E1-0004-A	E2-0004-A
8.3-7	E1-0006	E2-0006
8.3-7	E1-0006-A	E2-0006-A
8.3-8	E1-0005	E2-0005
8.3-8	E1-0005-A	E2-0005-A
8.3-9	E1-0007	E2-0007
8.3-9	E1-0007-A	E2-0007-A
8.3-9	E1-0007-B	E2-0007-B
8.3-9	E1-0007-C	E2-0007-C
8.3-10	E1-0009	E2-0009
8.3-10	E1-0009-A	E2-0009-A
8.3-10	E1-0009-B	E2-0009-B
8.3-11	E1-0010	
8.3-11	E1-0010-A	
8.3-11	E1-0010-B	
8.3-12	E1-0014	E2-0014
8.3-12	E1-0014-A	E2-0014-A
8.3-12	E1-0014-B	
8.3-12	E1-0014-C	
8.3-13	E1-0001-A	E2-0001-A
8.3-14	E1-0020	E2-0020
8.3-14	E1-0020-A	E2-0020-A

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 3 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
8.3-14	E1-0020-B	E2-0020-B
8.3-14	E1-0020-C	E2-0020-C
8.3-14	E1-0020-D	E2-0020-D
8.3-14	E1-0020-E	E2-0020-E
8.3-14	E1-0020-F	E2-0020-F
8.3-14	E1-0020-G	E2-0020-G
8.3-14	E1-0020-H	E2-0020-H
8.3-14	E1-0020-J	E2-0020-J
8.3-14	E1-0020-K	
8.3-14	E1-0020-L	
8.3-14A	E1-0019	E2-0019
8.3-14A	E1-0019-A	E2-0019-A
8.3-15	E1-0018	E2-0018
8.3-15	E1-0018-A	E2-0018-A
8.3-15	E1-0018-B	E2-0018-B
8.3-15	E1-0018-C	E2-0018-C
8.3-15	E1-0018-D	E2-0018-D
8.3-15	E1-0018-E	E2-0018-E
8.3-15	E1-0018-F	E2-0018-F
8.3-15	E1-0018-G	E2-0018-G
8.3-15	E1-0018-H	E2-0018-H
8.3-15	E1-0024-03	E2-0024-03
8.3-15	E1-0024-03A	E2-0024-03A

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 4 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
8.3-15A	E1-0018-J	E2-0018-J
8.3-15A	E1-0018-01	
8.3-15A	E1-0018-01A	
8.3-15A	E1-0018-01B	
8.3-15A	E1-0018-01C	
8.3-15A	E1-0018-01D	
8.3-15A	E1-0018-01E	
8.3-15A	E1-0018-01F	E2-0018-01
8.3-15A	E1-0018-01G	E2-0018-01A
8.3-15A	E1-0018-01H	E2-0018-01B
8.3-15B	E1-0018-02	E2-0018-02
8.3-15B	E1-0018-2A	E2-0018-02A
8.3-15B	E1-0018-2B	E2-0018-02B
8.3-15B	E1-0018-2C	E2-0018-02C
8.3-15B	E1-0018-2D	E2-0018-02D
8.3-15C	E1-0018-03	E2-0018-03
8.3-18	E1-0022-04	E2-0022-04
9.1-13	M1-0235	M2-0235
9.1-13	M1-0235-01	
9.1-13	M1-0235-02	
9.2-1	M1-0233	M2-0233
9.2-1	M1-0233-A	M2-0233-A
9.2-1	M1-0234	M2-0234

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 5 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
9.2-1	M1-0240	
9.2-3	M1-0229	M2-0229
9.2-3	M1-0229-A	M2-0229-A
9.2-3	M1-0229-B	M2-0229-B
9.2-3	M1-0230	M2-0230
9.2-3	M1-0230-A	M2-0230-A
9.2-3	M1-0230-B	
9.2-3	M1-0230-C	
9.2-3	M1-0231	M2-0231
9.2-3	M1-0231-A	
9.2-4	M1-0224	
9.2-4	M1-0224-A	
9.2-4A	M1-0224-01	
9.2-4A	M1-0224-01A	
9.2-4A	M1-0224-01B	
9.2-4A	M1-0224-01C	
9.2-4A	M1-0224-01D	
9.2-4A	M1-0224-01E	
9.2-4A	M1-0224-01F	
9.2-4A	M1-0224-01G	
9.2-4A	M1-0224-01H	
9.2-4A	M1-0224-01J	
9.2-4A	M1-0224-01K	

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 6 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
9.2-4A	M1-0224-01L	
9.2-4A	M1-0224-01M	
9.2-4A	M1-0224-01N	
9.2-5	M1-0241	M2-0241
9.2-5	M1-0241-A	
9.2-5	M1-0241-01	
9.2-5	M1-0242	M2-0242
9.2-5	M1-0242-A	
9.2-5	M1-0242-B	
9.2-6	M1-0227	
9.2-6	M1-0227-01	
9.2-6	M1-0227-03	
9.2-7	M1-0226	
9.2-15	M1-0280	
9.2-15	M1-0280-A	
9.2-16	M1-0281	
9.2-16	M1-0281-A	
9.2-16	M1-0281-B	
9.3-1	M1-0216	M2-0216
9.3-1	M1-0216-A	M2-0216-A
9.3-1	M1-0216-B	M2-0216-B
9.3-1	M1-0216-C	
9.3-1	M1-0216-01	

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 7 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
9.3-1	M1-0218	M2-0218
9.3-1	M1-0218-01	M2-0218-01
9.3-1	M1-0218-01A	
9.3-1	M1-0218-02	M2-0218-02
9.3-1		M2-0218-04
9.3-1		M2-0218-04A
9.3-1	M1-0219	M2-0219
9.3-1	M1-0219-01	
9.3-1	M1-0219-02	
9.3-1	M1-0219-02A	
9.3-1	M1-0219-03	
9.3-1	M1-0220	
9.3-1	M1-0220-01	M2-0220-01
9.3-1	M1-0220-01A	M2-0220-01A
9.3-1		M2-0220-02
9.3-2	M1-0217	M2-0217
9.3-2	M1-0217-A	M2-0217-A
9.3-4	M1-0228	M2-0228
9.3-4	M1-0228-A	M2-0228-A
9.3-4	M1-0228-B	M2-0228-B
9.3-4	M1-0228-C	M2-0228-C
9.3-4	M1-0228-01	M2-0228-01
9.3-4	M1-0228-02	

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 8 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
9.3-5	M1-0238	M2-0238
9.3-5	M1-0238-A	M2-0238-A
9.3-6	M1-0236	M2-0236
9.3-6	M1-0236-A	M2-0236-A
9.3-6	M1-0236-B	M2-0236-B
9.3-7	M1-0236-01	
9.3-7	M1-0236-01A	
9.3-7	M1-0236-02	
9.3-7	M1-0236-02A	
9.3-7	M1-0236-03	M2-0236-03
9.3-7	M1-0236-04	
9.3-8	M1-0237	M2-0237
9.3-8	M1-0237-A	M2-0237-A
9.3-9	M1-0237-01	M2-0237-01
9.3-9	M1-0237-01A	
9.3-10	M1-0253	M2-0253
9.3-10	M1-0253-A	M2-0253-A
9.3-10	M1-0254	M2-0254
9.3-10	M1-0255	M2-0255
9.3-10	M1-0255-01	M2-0255-01
9.3-10	M1-0255-02	M2-0255-02
9.3-10	M1-0256	
9.3-10	M1-0256-A	M2-0256-A

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 9 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
		M2-0256-B
9.3-10	M1-0257	
9.3-11	M1-0258	
9.3-11	M1-0259	
9.3-11	M1-0259-A	
9.3-11	M1-0259-01	
9.4-1	M1-0304	
9.4-1	M1-0304-A	
9.4-1	M1-0304-B	
9.4-1	M1-0304-C	
9.4-1	M1-0304-D	
9.4-1	M1-0308	
9.4-1	M1-0308-A	
9.4-2	M1-0303	
9.4-2	M1-0303-A	
9.4-2	M1-0303-B	
9.4-2	M1-0303-C	
9.4-2	M1-0303-01	
9.4-3	M1-0306	M2-0306
9.4-3	M1-0306-A	
9.4-4	M1-0302	M2-0302
9.4-4	M1-0302-A	M2-0302-A
9.4-4	M1-0302-B	M2-0302-B

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 10 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
9.4-4	M1-0302-C	M2-0302-C
9.4-5	M1-0300	M2-0300
9.4-5	M1-0300-A	M2-0300-A
9.4-6	M1-0301	M2-0301
9.4-6	M1-0301-A	M2-0301-A
9.4-7	M1-0312	
9.4-8	M1-0305	
9.4-8	M1-0305-A	
9.4-9	M1-0309	
9.4-9	M1-0309-A	
9.4-9	M1-0309-B	
9.4-11	M1-0307	M2-0307
9.4-11	M1-0307-A	M2-0307-A
9.4-11	M1-0307-B	
9.4-11	M1-0307-C	
9.4-12	M1-0311	M2-0311
9.4-12	M1-0311-A	M2-0311-A
9.4-12	M1-0311-B	M2-0311-B
9.4-14	M1-0304-01	
9.4-14	M1-0304-01A	
9.4-14	M1-0304-01B	
9.4-15	M1-0313	
9.5-43	M1-0225	

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 11 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
9.5-44	M1-0225-01	M2-0225-01
9.5-45	M1-0225-02	
9.5-45	M1-0225-02A	
9.5-46	M1-0225-03	M2-0225-03
9.5-46	M1-0225-03A	M2-0225-03A
9.5-47	M1-0225-04	
9.5-47	M1-0225-04A	
9.5-48	M1-0225-05	M2-0225-05
9.5-52	M1-0215-F	M2-0215-F
9.5-52	M1-0215-G	M2-0215-G
9.5-54	M1-0215-H	M2-0215-H
9.5-54	M1-0215-J	M2-0215-J
9.5-55	M1-0215-D	M2-0215-D
9.5-55	M1-0215-E	M2-0215-E
9.5-56	M1-0215-B	M2-0215-B
9.5-56	M1-0215-C	M2-0215-C
9.5-57	M1-0215	M2-0215
9.5-57	M1-0215-A	M2-0215-A
9.5-61	M1-0225-06	
9.5-62	MX-0225-07	
9.5-62	MX-0225-08	
9.5.62	MX-0225-09	
10.3-1	M1-0202	M2-0202

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 12 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
10.3-1	M1-0202-01	M2-0202-01
10.3-1	M1-0202-01A	M2-0202-01A
10.3-1	M1-0202-01B	M2-0202-01B
10.3-1	M1-0202-02	M2-0202-02
10.3-1	M1-0202-3	M2-0202-03
10.4-3	M1-0211	M2-0211
10.4-3	M1-0211-01	M2-0211-01
10.4.3	M1-0211-02	
10.4-4	M1-0223	M2-0223
10.4-4	M1-0223-A	M2-0223-A
10.4-5	M1-0210	M2-0210
10.4-5	M1-0210-A	M2-0210-A
10.4-5	M1-0210-B	M2-0210-B
10.4-5		M2-0210-C
10.4-5	M1-0210-01	
10.4-6	M1-0240-01	
10.4-7	M1-0244	M2-0244
10.4-7	M1-0244-A	M2-0244-A
10.4-7	M1-0244-B	M2-0244-B
10.4-7	M1-0244-01	M2-0244-01
10.4-7	M1-0244-01A	M2-0244-01A
10.4-8	M1-0204	M2-0204
10.4-8	M1-0204-02	M2-0204-02

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 13 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
10.4-8	M1-0204-03	M2-0204-03
10.4-9	M1-0203	M2-0203
10.4-9	M1-0203-01	M2-0203-01
10.4-9	M1-0203-01A	M2-0203-01A
10.4-10	M1-0239	M2-0239
10.4-10	M1-0239-01	M2-0239-01
10.4-11	M1-0206	M2-0206
10.4-11	M1-0206-01	M2-0206-01
10.4-11	M1-0206-02	M2-0206-02
10.4-13	M1-0205	M2-0205
10.4-13	M1-0205-A	M2-0205-A
10.4-14	M1-0207	M2-0207
10.4-14	M1-0207-A	M2-0207-A
10.4-14	M1-0207-B	M2-0207-B
10.4-14	M1-0208	M2-0208
10.4-14	M1-0209	M2-0209
10.4-14	M1-0209-A	M2-0209-A
10.4-15	M1-0212	M2-0212
10.4-15	M1-0212-A	M2-0212-A
10.4-15	M1-0212-B	M2-0212-B
10.4-16	M1-0213	
10.4-16	M1-0213-A	
10.4-16	M1-0213-B	

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 14 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
10.4-16	M1-0213-C	
10.4-16	M1-0213-01	
10.4-17	M1-0214	M2-0214
10.4-17	M1-0214-A	M2-0214-A
10.4-17	M1-0214-B	M2-0214-B
10.4-17	M1-0214-01	M2-0214-C
10.4-18	M1-0243	M2-0243
10.4-18	M1-0243-A	M2-0243-A
10.4-18	M1-0243-B	
10.4-19	M1-0243-02	M2-0243-02
10.4-20	M1-0222	M2-0222
10.4-20	M1-222-A	
10.4-20	M1-0222-B	
10.4-20	M1-0222-C	
10.4-20		M2-0222-01
11.2-2	M1-0264	M2-0264
11.2-3	M1-0265	
11.2-3	M1-0265-01	
11.2-4	M1-0266	
11.2-4	M1-0266-A	
11.2-4	M1-0266-01	
11.2-5	M1-0267	
11.2-6	M1-0268	

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 15 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
11.2-7	M1-0268-01	
11.2-8	M1-0268-02	
11.2-9	M1-0266-B	
11.3-1	M1-0269	
11.3-1	M1-0269-A	
11.3-1	M1-0269-B	
11.3-1	M1-0269-C	
11.3-1	M1-0269-D	
11.3-1	M1-0269-01	
11.3-1	M1-0270	
11.3-1	M1-0270-A	
12.3-4	M1-3500-001	
12.3-5	M1-3500-002	
12.2-5.1	M1-3500-003	
12.3-6	M1-3500-004	
12.3-7	M1-3500-005	
12.3-8	M1-3500-006	
12.3-9	M1-3500-007	
12.3-10	M1-3500-008	
12.3-11	M1-3500-009	
12.3-12	M1-3500-010	
12.3-13	M1-3500-011	
12.3-14	M1-3500-012	

TABLE 3.2-4
FIGURE v/s PLANT DRAWING CROSS REFERENCE

(Sheet 16 of 16)

FSAR Figure No.	U1&X Drawing	U2 Drawing
12.3-14.1	M1-3500-013	
12.3-15	M1-3500-014	
12.3-16	M1-3500-015	
12.3-17	M1-3500-016	
12.3-18	M1-3500-017	
12.3-19	M1-3500-018	
12.3-20	M1-3500-019	
12.3-21	M1-3500-023	
12.3-21.1	M1-3500-020	
12.3-21.2	M1-3500-021	
12.3-22		M2-3500-011
12.3-22.1		M2-3500-008
12.3-22.2		M2-3500-009
12.3-23	M1-3500-022	
12.3-23.1		M2-3500-010
12.3-23.2		M2-3500-001
12.3-23.3		M2-3500-002
12.3-23.4		M2-3500-003
12.3-23.5		M2-3500-004
12.3-23.6		M2-3500-005
12.3-23.7		M2-3500-006
12.3-23.8		M2-3500-007

3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

3.3.1.1 Design Wind Velocity

Seismic Category I structures are designed for a basic maximum wind velocity of 80 mph at 30 ft above ground based on a 100-year period of recurrence.

The design wind velocity is selected on the basis of U.S. Weather Bureau data for the general area, as indicated in Figure 1(b) of ASCE Paper No. 3269, “Wind Forces on Structures”.

Wind velocity distribution for various height zones above the ground is in accordance with Table 1 of ASCE Paper No. 3269, “Wind Forces on Structures”, and is as follows:

Height Above Ground (ft)	Wind Velocity (mph)
0 to 50	80
50 to 150	95
150 to 400	110
400 to 700	120

A gust factor of 1.1 is applied to the above wind velocities for all seismic Category 1 structures except the yard tanks for which a gust factor of 1.3 is used. The adequacy of a 1.1 gust factor is confirmed by calculations in accordance with Reference 6.

3.3.1.2 Determination of Applied Forces

The dynamic wind pressure q is defined as $q = 0.00256V^2$, where q is in psf and V, the wind velocity (including the gust factor), is in mph. ASCE Paper No. 3269 is used as a guide in determining wind pressure distribution on the structures. For the containment structure and cylindrical tanks, the coefficients are in accordance with Table 4(f) of ASCE paper 3269. For rectangular structures, the shape coefficient is 1.3. The total wind pressure load on the structure is 1.3q; of the 1.3q, 0.8q is applied as positive pressure on the windward walls, and 0.5q is applied as negative pressure on the leeward walls. Concurrently, the end walls receive 0.8q negative pressure, and the roof receives 0.5q uplift.

Any wind shielding effect that one structure can provide to another is neglected.

Wind loads are applied to the structures as static loads. Only dead loads are considered to resist uplift caused by wind loads. Horizontal wind loads are distributed by the walls to roof and floor diaphragms, and they are distributed from these to the foundations by the lateral load-carrying elements of the structural system.

Load combinations that include wind are described in Section 3.8.

3.3.2 TORNADO LOADINGS

All essential systems and components are protected from tornado induced loads in order to maintain the following:

1. The ability to achieve and maintain a safe shutdown in both units.

2. The integrity of the reactor-coolant pressure boundary.

3. The ability to prevent failures that could lead to radioactive releases resulting in offsite exposures greater than 25 percent of the guideline exposures of 10 CFR Part 100.

These systems and components are located within Seismic Category I buildings as listed in Section 3.2.1.1. The arrangement of these structures can be found on the plot plan, Fig. 1.2-1.

Seismic Category I buildings are vented to the atmosphere in the event of a tornado. These buildings are designed to withstand the loadings due to wind, depressurization and repressurization, and tornado generated missiles. These buildings are also designed to provide protection from the tornado and its effects to Seismic Category I systems and components within these structures.

Tornado impingement is not considered coincident with other low probability events such as a Safe Shutdown Earthquake (SSE), loss of coolant accident (LOCA), or a fire. Because of the potential switchyard damage, a trip of the turbine-generators and loss of offsite power are assumed to result from the design basis tornado.

3.3.2.1 Applicable Design Parameters

The design basis tornado has a peripheral tangential velocity of 300 mph. and a translational velocity of 60 mph.

The design basis tornado drops the local atmospheric pressure 3 psi at a constant rate of 1.0 psi/sec. The pressure is then held at -3 psig for 10 seconds at which time repressurization occurs at a rate of 1.0 psi/sec., for a total duration of 16 seconds. The number 10 seconds for the time held at the negative pressure is chosen for analysis purposes to ensure that complete depressurization is considered.

The spectrum and pertinent characteristics of tornado generated missiles are described in Section 3.5 on Table 3.5-8. Missile loads are considered to act simultaneously with wind and pressure loads.

3.3.2.2 Tornado Protection Design Features

Venting from the building interior compartments to the exterior is provided by blow-out door F-4EX, tornado pressure relief dampers, and tornado pressure relief blowout panels.

1. Blow-out door F-4EX

   The Fuel Building is primarily vented by an opening covered by blow-out door F-4EX. This door is designed to blowout when subjected to tornado induced loading.

2. Tornado Pressure Relief Dampers in Exterior Walls

All exterior tornado dampers are specified to remain closed during winds with speeds up to 119 mph. Tornado dampers in exterior walls are specified to open in either direction when the differential pressure across the damper reaches 0.25 psi in either direction. Concrete missile shields, as shown on Figure 3.3-1, protect the dampers and the building interior from tornado generated missiles.

3. Tornado Pressure Relief Blowout Panels

Airtight blowout panels are used in the venting of the Auxiliary Building, the Control Room and the Safeguards Building. All blowout panels are specified to remain airtight during winds with speeds up to 119 mph. Blowout panels are specified to open when the differential pressure across the panel reaches 0.25 psi. Where a missile can impact a blowout panel, a Seismic Category I building is protected by reinforced concrete missile resisting walls and roof so arranged as to stop a missile as shown in Figure 3.3-2 and Figure 3.3-3. Where a missile is stopped by a concrete barrier and could then enter by gravity into a Category I building protective grating is provided as shown in Figure 3.3-3.

The Diesel Generator Building is primarily vented through the Diesel Generator air intakes.

Venting between interior compartments that are separate fire areas is provided by fire rated architectural door openings, tornado pressure relief dampers (in series with fire dampers), and fire rated HVAC air transfer grilles.

1. Fire Rated Architectural Door Openings

Door E-3AX is specified to blowout of the E&C building into the turbine building. The door is designed to blowout when subjected to tornado induced loading.

Several standard hollow metal doors are held open with fusible link arm-holders designed to close the door during a fire. Wire-mesh doors are also provided for radiation protection (access control) as required. Where other considerations, such as HVAC integrity, require that a hollow metal door be closed, and venting is still required the door is modified in such a way as to allow it to release during a tornado. These doors are also restrained during release to prevent destruction of the door. Fire rated doors are also provided in the same doorway and are also held open with fusible link arm-holders designed to close the door during a fire.

2. Tornado Pressure Relief Dampers in Interior Fire Walls

These dampers are normally closed, where required to provide a ventilation or other environmental boundary, and are specified to open in either direction when the differential pressure across the damper reaches 3" wg. except for selected dampers located in the Safeguards Buildings which have been modified by permanently opening a portion of the modules of each damper to provide an airflow path through areas served by the Electrical Area Ventilation System to the Main Steam and Feedwater Area Ventilation System air inlet duct in the Mechanical Equipment Room (1(2)-104).

The tornado dampers located in the UPS HVAC chase are also normally closed and open in either direction when the differential pressure reaches 7.5" wg.

3. Fire Rated Air Transfer Grilles

Several fire-rated HVAC air transfer grilles are considered for tornado venting. These vents are not attached to ductwork which would potentially block the vent. These vents are normally open.

Venting between interior compartments that are not separate fire areas is provided by access gratings, excess area in wall penetrations, tornado pressure relief dampers and architectural door openings.

1. Tornado Pressure Relief Dampers in Interior Walls

These dampers operate in the same manner as the interior dampers in the fire-rated openings.

2. Architectural Door Openings

Several wire-mesh doors are provided for tornado venting.

Several hollow-metal doors are provided with air transfer louvers. In cases where doors are required by other considerations to be in closed positions, some doors are designed to blow open when subjected to tornado induced loading.

Since the tornado venting characteristics of the buildings are dependent on the positions of architectural doors, these positions must be considered. Applicable plant design documents show all doors that have a required open position for purposes of tornado venting. Seismic Category I structures are designed to withstand the impact loads from the spectrum of tornado generated missiles as shown in Table 3.5-8. Structures are of sufficient thickness to prevent perforation or the generation of secondary missiles by tornado generated missiles to which they can be exposed.

Non-Seismic Category I structures or structural components whose failure under tornado loading could possibly impair the function of Seismic Category I structures, equipment, systems or components are designed to withstand the effects of the tornado loading to the extent that the functional requirements of Seismic Category I structures, equipment, systems and components are not impaired.

3.3.2.3 Method of Analysis

The dynamic wind pressure, q, is calculated from the following equation:

$$q = 0.00256\ V^2$$

where:

q = wind pressure [psf]

v = wind velocity [mph]

The tornado wind velocity is not assumed to vary with height above the ground. The tornado wind pressure is applied as a static load. The pressure distribution and shape coefficients for tornado loadings are the same as the coefficients used for wind, as described in Subsection 3.3.1.2. For all Seismic Category I structures a 360 mph wind velocity is applied uniformly. A gust factor of 1.0 shall be used.

For Seismic Category I buildings other than the Containment the effects of venting are considered in determining the maximum pressure differential across building wall, floor and roof systems. Calculations of differential pressure loads are made utilizing an improved version of the COMPARE - Mod 1 computer code. The subcompartments are represented as volumes that are connected by pressure-dependent variable area junctions or vents. The volume thermodynamics are based on a homogeneous gas, assumed to undergo a quasi-static process. Flow between volumes is based on a one dimensional solution of the momentum equation that includes an accounting for the effects of inertia. Air is assumed to be an ideal gas. The relative humidity is assumed to be zero for conservatism in terms of pressure. For analysis purposes the turbine building is considered to exist at atmospheric environmental conditions. This is justified by the nature of the building design. No flow is considered available through HVAC ductwork due to the potential for damage to it.

Volumes in the flow model represent separate rooms, groups of rooms, or sections of rooms so that the maximum attainable differential pressure across any structure inside a volume is analytically shown to be 0.25 psi.

Tornado missile impact velocities and barrier requirements are presented in Section 3.5. The procedures used for transforming tornado-generated missile loadings, which are considered impactive dynamic loads, into effective loads are described in Section 3.5. Load combinations that include tornados are described on Section 3.8. A load factor of 1.0 is used for tornado loads in the load combination equations. The load factor is justified by the low probability of occurrence at a specific point, by the short duration of associated loading, and by the conservative nature of the maximum tornado wind velocity selected. Non-Seismic Category I equipment that is not in a Seismic category I building and hence can possibly become a tornado generated missile is evaluated as a potential missile. Because the damage potential of such a missile is not greater than the effect of the spectrum of tornado-generated missiles listed in Table 3.5-8 the equipment is not protected from tornado loads.

3.3.2.4 Design Evaluation

The various combinations of tornado effects which the structures will withstand are as follows:

$W_t = W_w$

$W_t = W_p$

$W_t = W_m$

$W_t = W_w + 0.5\ W_p$

$W_t = W_w + W_m$

$W_t = W_w + 0.5\ W_p + W_m$

where:

W_t = total tornado load

W_w = tornado wind load

W_p = tornado differential pressure load

W_m = equivalent tornado missile load

The local design of the exterior walls and roofs of each Seismic Category I structure provides each element with the capability to withstand the most adverse of the six load combinations in each direction. All interior structural components in Seismic Category I structures are verified to withstand the pressure load only. This is justified by the low air velocities induced by the relatively large flow areas inside the rooms. These interior structures include non-load bearing partitions such as fire-walls, windows, and accoustic ceilings which could potentially become missiles if damaged. The same loading combinations are considered to act on all doors that are not specified as a pressure relief door or not required to be open and which could potentially become a missile if damaged. The structural capabilities of hollow metal doors are verified by the ASTM test method.

Since the interior of the building depressurizes, the electrical and I&C cabinets are subject to pressure loadings in the same manner as the buildings. Available vents are measured and pressure loadings are determined in the same manners as the buildings. The pressure loadings are transformed into member stresses and displacements by a stress analysis. All cabinets containing equipment required to achieve and maintain a safe shutdown during and after a tornado, and to operate tornado protection devices are verified to maintain the structural integrity and the operability of the equipment when subjected to tornado pressure loads.

The Safe Shutdown Impoundment (SSI) Dam has outer surfaces or shells of limestone rock with a minimum thickness of 10 feet. This minimum thickness occurs only at the crest of the dam; the rock thickness increases rapidly below the crest to a maximum of more than 100 ft. (each surface) at the base of the embankment. The penetration of any tornado missile from the spectrum presented in Table 3.5-8 into such a surface is far less than the minimum thickness of the limestone. However, even if total penetration of the longest missile of the spectrum considered can somehow occur, and even if the direction of penetration is parallel to and located at the SSI water level, a limestone rock and filter thickness that exceeds the total missile length by 56 percent is present on either dam surface. The longest missile is considered to be a 35-ft. (10.68-m) long utility pole.

Squaw Creek Dam is not a Category I structure and the effect of a tornado missile is not pertinent.

REFERENCES

1. American Society of Civil Engineers, Wind Forces on Structures, Transactions of the American Society of Civil Engineers, Volume 126, Part II, Paper No. 3269, 1961.

2. R. G. Gido, G. J. E. Willcutt, Jr., J. L. Lunsford, and J. S. Gilbert, COMPARE-MOD 1 Code Addendum, NUREG/CR-1185, Addendum 1 (LA-7199-MS, Addendum), June 1980, and Attachment September, 1980.

3. K. C. Mehata, et al. The Kalamazoo Tornado of May 13, 1980. Committee on Natural Disasters, Commission on Sociotechnial Systems. National Research Council, National Academy Press 1981.

4. Amercian Society for Testing and Materials. Standard Test Method for Structural Performance of Exterior Windows, Curtain-Walls, and Doors by Uniform Static Air Pressure Difference, ANSI/ASTM E 330-79.

5. American Society of Civil Engineers, Nuclear Power Plant Tornado Design Considerations Journal of the Power Division, Proc. ASCE, Vol. 97, No. P02.

6. American Nuclear Society, Building Code Requirements for Minimum Design Loads in Buildings and Other Structures, ANSI A58.1-1972.

TABLE 3.3-1
HAS BEEN DELETED.

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD PROTECTION

Seismic Category I equipment, systems, and components in the Service Water Intake Structure are located above the probable maximum flood (PMF) level and do not require protection from flood. The PMF is considered in the design of the submerged pump shafts.

The following seismic Category I systems and components are located below the PMF level of 789.7 ft:

Safeguards Building:

1. Containment Spray System - containment spray pumps

2. Residual Heat Removal System - RHR pumps

3. Safety Injection System - SIS pumps

4. Vents and Drains System - floor drain sumps and pumps

Grade elevation for the structure in which these seismic Category I system components are located is 810.0 ft. No doors or entries to these structures are located below 810.0 ft. There is, then, no possibility of flooding these components from the PMF condition.

Water stops are used in all construction joints below grade even though the ground water elevation as outlined in Section 2.5.4.6 is no higher than 775.0 ft. Sump pumps are provided to handle the minute amounts of water which could leak into the structures.

Since there is no possibility of flooding any of these seismic Category I components as a result of PMF or ground flooding, no specific flood protection of the above equipment is required.

Electrical and Control Building:

1. Safety Chilled Water System - Chillers and pumps

2. UPS and Distribution Room Air Conditioning System - Air Conditioning Units

Grade elevation for the structure in which these seismic Category I system components are located is 810.0 feet. Doors to this structure are located above PMF level with the exception of three doors into the Turbine Building at Elevation 778 feet (see Figures 1.2-31, 22, and 27).

Flooding of the Turbine Building due to PMF is prevented because the Circulating Water and its interconnecting system are closed systems (See Section 10.4). In addition, if Squaw Creek Reservoir elevation rises above 778 feet, the operators will verify that any Circulating Water System equipment which is to be opened for maintenance will be isolated from Squaw Creek Reservoir by valves and/or stop-gates. Due to the access opening in the operate deck of the circulating water discharge structure, the stop gates isolate the CW Tunnels for lake levels below 778 feet only. Any open CW system at elevation below the PMF level of 789.7 feet must be closed or isolated in order to isolate the lake levels above 778 from the open pathway that could

lead to flooding of the Turbine Building and lower level of the Electrical & Control Building on Elevation 778.

Therefore, it is not necessary to bring the reactor to a cold shutdown for flood conditions.

3.4.2 ANALYSIS PROCEDURES

Grade elevation of safety-related structures, except the Service Water Intake Structure, is 810.0 ft. The PMF level is 789.7 ft. Thus no direct static or dynamic effects of the design basis flood conditions are considered for these structures. However, the static hydrostatic pressure that results from the ground water elevation of 775.0 ft. is considered in the design of structures located below this elevation. Hydrostatic pressure p is determined from the following equation:

$$p = \gamma h$$

where

γ = the unit weight of water

h = the depth below the groundwater elevation

The Service Water Intake Structure is protected from the effects of wind-wave activity on the Squaw Creek Reservoir by the Safe Shutdown Impoundment Dam, but it is subject to wind-wave activity on the Safe Shutdown Impoundment. A 40-mph wind is assumed coincidental with the PMF (789.7 ft). The wind-wave activity at the Service Water Intake is negligible; therefore, there are no significant dynamic forces to be considered. The maximum water level used in design, as a hydrostatic load, takes into consideration the maximum runup elevation caused by a 40-mph wind coincident with the PMF. An 80-mph sustained wind has been superimposed on a normal water level of 775 ft which results in a lower combined setup and runup elevation than the still water elevation that occurs during the PMF. Therefore, this is not a critical design condition with regard to maximum water level at the Service Water Intake Structure.

The Service Water Intake Structure is not subject to flood currents; therefore, dynamic water force as a result of flood currents is not a design load.

3.4.3 FLOODING FROM TANK RUPTURE

Table 3.4-1 lists all the non-seismic Category I liquid storage tanks larger than 1500 gallons located in the Containment, Safeguards and Auxiliary buildings. The effects of flooding on shutdown capability of the plant were analyzed assuming that the release of tank contents is instantaneous and the sump pumps will not be operational. In all cases, tank rupture results in local flooding only, with no effect on plant safety or the plant safe shutdown capability. Tanks less than or equal to 1500 gallons are bounded by the room design described in Section 9.3.3.3.

TABLE 3.4-1
EFFECT OF LIQUID STORAGE TANKS FAILURE ON THE SAFE SHUTDOWN OF THE PLANT

(Sheet 1 of 3)

ITEM	TANK	SYSTEM	CAPACITY	LOCATION BLDG.	EL	POINT OF COLLECTION OF SPILLS	EFFECTS OF FLOODING SHUTDOWN CAPABILITY OF THE PLANT	COMMENTS
1	Pressurizer relief tank	RC	1800 ft^3	Containment	822'-9"	Sump No. 1 via floor drains	None	
2	DELETED							
3	DELETED							
4	DELETED							
5	DELETED							
6	Laundry and hot shower tank	WP	10,000 gal.	Auxiliary	790'-6"	Sump No. 7 via floor drains	None	
7	Spent resin storage tank	WP	4,100 gal.	Auxiliary	810'-6"	Floor drain tank No. 1 via floor drains	None	
8	Waste holdup tank	WP	10,000 gal.	Auxiliary	790'-6"	Waste holdup tank compartment	None	The tank is located in a watertight compartment which drains to sump No. 7 through normally locked-closed valves
9	DELETED							
10	Recycle holdup tanks	BR	112,000 gal.	Auxiliary	790'-6"	Recycle holdup tanks compartment	None	The tank is located in a watertight compartment which drains to sump No.8 through normally locked-closed valves. Backflow thru floor drains was evaluated.
11	Steam generator blowdown spent resin storage tank	S.G. blow-and cleanup	3700 gal	Auxiliary	790'-6"	Steam generator blowdown tank compartment and AB Sump No. 1 via drains	None	The tank is located in a watertight compartment, the spill will be confined to the tank compartment and AB Sump No. 1
12	DELETED							

TABLE 3.4-1
EFFECT OF LIQUID STORAGE TANKS FAILURE ON THE SAFE SHUTDOWN OF THE PLANT

(Sheet 2 of 3)

ITEM	TANK	SYSTEM	CAPACITY	LOCATION BLDG.	EL	POINT OF COLLECTION OF SPILLS	EFFECTS OF FLOODING SHUTDOWN CAPABILITY OF THE PLANT	COMMENTS
13	Laundry holdup and monitor tank	WP	5,000 gal.	Auxiliary	790'-6"	Sump No. 7 via floor drains	None	
14	Waste evaporator condensate tank	WP	5,000 gal	Auxiliary	790'-6"	Waste evaporator condensate tank compartment and AB Sump No. 6	None	The spill will be confined to the tank compartment and sump
15	Floor drains Waste Monitor tanks	WP	5,000 gal.	Auxiliary	790'-6"	Waste monitor tanks compartment and AB Sump No. 5	None	The spill will be primarily confined to the tank compartment and sump
16	DELETED							
17	DELETED							
18	DELETED							
19	Laundry water head tank	WP	5,000 gal.	Auxiliary	852'-6"	Floor drain tank No. 1 via floor drains	None	
20	Floor drain tanks	WP	10,000 gal.	Safeguard	773'-0"	Floor drain tank compartment	None	The tank is located in a watertight compartment which drains to sump No.1 via through normally locked-closed valves
			30,000 gal.	Auxiliary	790'-6"	Floor drain tank compartment and AB Sump No. 4	None	The contents of the tank will be retained partially inside the compartment, the balance will be disposed via the floor drain system
21	DELETED							
22	Component cooling water drain tank	CCW	2,300 gal.	Safeguards	773'-0"	Component cooling water drain tank compartment and SB Sumps Nos. 1 and 3	None	The spill will be contained inside the tank compartment and sumps

TABLE 3.4-1
EFFECT OF LIQUID STORAGE TANKS FAILURE ON THE SAFE SHUTDOWN OF THE PLANT

(Sheet 3 of 3)

ITEM	TANK	SYSTEM	CAPACITY	LOCATION BLDG.	EL	POINT OF COLLECTION OF SPILLS	EFFECTS OF FLOODING SHUTDOWN CAPABILITY OF THE PLANT	COMMENTS
23	DELETED							
24	DELETED							
25	DELETED							
26	Hot phase separator	CP	11,420 gal.	Fuel	860'-0"	FB Sump No. 2 via floor drains	None	
27	Waste conditioning tank	WP	2,000 gal.	Fuel	841'-0"	FB Sump No. 2 via floor drains	None	
28	Tanks ≤ 1,500 gallons	Various	≤ 1,500 gal	Various	Various	Local room and floor drains	None	See Section 9.3.3.3

3.5 MISSILE PROTECTION

3.5.1 MISSILE SELECTION AND DESCRIPTION

The basic approach to ensure missile protection of systems and components both inside and outside of Containment involves:

1. Examination of systems in order to identify and classify potential missiles

2. Postulation of the generation of potential missiles and the provision of protection against them

3. Insurance of the design adequacy of equipment against the generation of missiles.

3.5.1.1 Internally Generated Missiles (Outside Containment)

The principal design bases are that missiles generated outside of Containment but internal to the plantsite do not cause damage to a system or component, (which is provided for either continued safe operation or shutdown during operating conditions, operational transients, and postulated accident conditions) which are associated with the effects of missile formation. The seismic category and quality group classifications for these systems are identified in Section 3.2.

Equipment outside Containment (see Section 3.5.2) has been evaluated for potential missile sources. The information in this section concerning potential missile sources and systems which require protection from internally generated missiles outside Containment is provided as a result of that evaluation.

The specific systems which require protection from missiles are a function of the missile source analysis and are listed in Table 3.5-1. Systems outside Containment which are required for safe shutdown of the reactor are protected regardless of the missile source.

3.5.1.1.1 Missile Selection

In order to conservatively establish the protection and redundancy requirements for safety-related equipment, missile generation is postulated to occur from high energy systems and rotating equipment. Only one source of a primary missiles is postulated for each event.

3.5.1.1.2 Missile Protection

To prevent a total loss of safety-related system integrity due to internally generated missiles, general design provides two redundant trains for each safety-related system. These trains are separated by missile barriers so that one missile, if it is generated in a system, cannot damage a redundant train of any system.

In addition, systems and components which are identified as potential missile sources are, wherever possible, arranged and oriented so that the tangent structure or component is capable of withstanding the impact, and the design criteria are not violated.

In order to determine the effects of a postulated missile, safety-related equipment is considered and protection is only provided for essential safety-related equipment (as defined in 3.6B.1.2.1) on a case by case basis.

In general, safety-related pumps, tanks, heat exchangers, and other large components are placed in separate compartments surrounded by missile barriers. Piping is run in common pipe chases, and valves are located in common valve rooms. For safety-related equipment, the pipe chases and valve rooms for different trains in general are isolated from each other so that a missile in one pipe chase or valve room cannot affect any equipment in the other chase or room.

Information on postulated turbine missiles is provided in Section 3.5.1.3.

3.5.1.1.3 Design Adequacy of Equipment

Equipment within Westinghouse scope has been reviewed for design adequacy against missile generation.

1. Valves

   Valves in high-pressure systems have been reviewed. As a result of this review, it is concluded that there are no credible sources of missiles associated with valves because there is no single failure associated with any potential valve parts that can result in the generation of a missile. Therefore, there are no missiles associated with valves within Westinghouse scope outside the Containment.

2. Pumps

   Pumps located (within Westinghouse scope) outside the Containment have been evaluated for missiles associated with overspeed failure. The maximum no-load speed of these pumps is equivalent to the operating speed of their motors. Consequently, no pipe break or single failure in the suction line can increase pump speed over that of the no-load condition. Furthermore, there are no pipe break plus single failure combinations which can result in a significant increase in pump suction or discharge head. Therefore, no overspeed is expected, and missiles associated with pumps within Westinghouse scope outside Containment are not credible.

3. Generator Flywheel

   The fabrication specifications of the motor generator set flywheels control the material, and are in accordance with ASTM A 533, 7D, Grade B, Class I, with inspections in accordance with MIL-I-45208A; flame cutting and machining operations are governed to prevent flaws in the material. Nondestructive testing for nil ductility (ASTM E208) and Charpy V-notch (ASTM A 593), as well as ultrasonic testing (ASTM A 578 and A 579) and magnetic particle testing (ASME B&PV Code, Section III, Paragraph NB 2545) are performed on each flywheel material lot. In addition to these requirements, stress calculations are performed in conformance with guidelines of the ASME B&PV Code, Section III, Appendix A, to show that the combined primary stresses caused by centrifugal forces and the shaft interference fit do not exceed one-third of the yield strength at normal operating speeds (1800 rpm) and do not exceed two-thirds of the yield strength at 25 percent overspeed. However, no overspeed is expected because the

flywheel weighs approximately 1300 lb and has dimensions of 35.26 in. in diameter by 4.76 in. in width. The flywheel mounted on the generator shaft, which is directly coupled to the motor shaft, is driven by a 200 hp, 1800 rpm induction motor. The torque developed by the motor is insufficient for overspeed. Therefore, there are no credible missiles from the motor generator sets.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The principal design bases are that missiles generated within the Containment do not cause loss of function of any structure, system or component whose failure could lead to offsite radiological consequences or which is required for safe plant shutdown to a cold condition assuming an additional single failure. The seismic category and quality group classifications for nuclear steam supply systems and components are in Section 3.2.

Equipment (outside of and within Westinghouse scope) inside Containment (see Section 3.5.2) has been evaluated for potential missile sources. As a result of this review, the following information concerning potential missile sources and systems which require protection from internally generated missiles inside Containment is provided:

3.5.1.2.1 Missile Selection

1. Equipment Within Westinghouse Scope

The possibility of a catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casings, and piping that can lead to generation of missiles is not considered credible. Massive and rapid failure of these components is not credible because of the material characteristics; inspection; quality control during fabrication, erection, and operation; conservative design; and prudent operation, as applied to the particular component. The reactor coolant pump flywheel is not considered a source of missiles for the reasons discussed in Section 5.4.1. Nuts and bolts are of negligible concern because of their small amount of stored elastic energy.

NSSS components which are considered to have a potential for Missile Generation inside the reactor containment are as follows:

a. control rod drive mechanism (CRDM) housing plug, drive shaft, and the drive shaft and drive mechanism latched together

b. Certain valves

c. Temperatures and pressure sensor assemblies

d. Pressurizer heaters

Gross failure of a CRDM housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

a. Full-length CRDMs are shop tested at 3107 psig (Unit 1) or 3450 psig and 4105 psig, respectively (Unit 2).

b. Unit 1

The mechanism housings are inspected and leak tested as they are installed on the reactor vessel to the head adapters.

Unit 2

The mechanism housings are individually hydrotested to 3107 psig as they are installed on the reactor vessel to the head adapters, and checked during the hydrotest of the completed RCS.

c. Stress levels in the mechanism are not affected by system transients at power, or by thermal movement of the coolant loops.

d. The mechanism housings are made of Type F304LN (Unit 1) or Type 304 (Unit 2) stainless steel. This material exhibits excellent notch toughness at all temperatures encountered.

However, it is postulated that the top plug on the CRDM becomes loose and it is forced upward by the water jet (Unit 2). The following sequence of events is assumed: the drive shaft and control rod cluster are forced out of the core by the differential pressure of 2500 psi across the drive shaft. The drive shaft and control rod cluster, latched together, are assumed to be fully inserted when the accident starts. After approximately 12 ft of travel, the rod cluster control spider hits the underside of the upper support plate. Upon impact, the flexure arms in the coupling, which join the drive shaft and control cluster, fracture and completely free the drive shaft from the control rod cluster. The control cluster is completely stopped by the upper support plate; however, the drive shaft continues to be accelerated upward to hit the missile shield provided. A description of the missile shield is provided in Section 3.5.1.2.4.

a. Valves

Valve stems are not considered credible sources of missiles.

All the isolation valves installed in the RCS have stems with a back seat. This effectively eliminates the possibility of ejecting valve stems even if the stem threads fail. Stems are threaded into a yoke and are locked in place with a locking bar. The actuator assembly around the yoke consists of other structural components as the frame which surrounds the spring adjustment and spring. Failure of the stem below the threaded region would be blocked by these components which are directly above the stem. The stem cannot be ejected sideways due to the fact it would be guided by the bonnet, packing and/or the trim depending on the location of the failure. In addition, the valves have a backseat or stop which will provide another mechanism against the stem being ejected. The backseat or stop would prevent the lower portion of the stem from exiting the valve [12].

Additional interference is encountered with air- and motor-operated valves.

Valves with a nominal diameter larger than 2 in. have been designed against bonnet body connection failure and subsequent bonnet ejection by the following means:

1. By using the design practice of the ASME B&PV Code, Section VIII, which limits the allowable stress of bolting material to less than 20 percent of its yield strength

2. By using the design practice of the ASME B&PV Code, Section VIII for flange design

3. By controlling the load during the bonnet body connection stud tightening process

The pressure-containing parts are designed in accordance with Class 1, 2, or 3 requirements established by the ASME B&PV Code, Section III.

The proper stud-torquing procedures and the use of torque wrench, with indication of the applied torque, restrict the stress of the studs to the allowable limits established in the ASME B&PV Code, Section III. This stress level is far below the material yield. The complete valves are hydrotested in accordance with the ASME B&PV Code, Section III. The stainless steel bodies and bonnets are volumeterically tested and surface-tested to verify soundness.

Valves with a nominal diameter of 2 in. or smaller are forged and have a screwed bonnet with a canopy seal. The canopy seal is the pressure boundary, while the bonnet threads are designed to withstand the hydrostatic and force. The pressure-containing parts are designed in conformance with the criteria established by the ASME B&PV Code, Section III.

While valve missiles are not generally postulated (the reasons are as previously mentioned), the valves in the region where the pressurizer extends above the operating deck are exceptions. Valves in this region are the pressurizer safety valves, the motor-operated isolation valves in the relief line, the air-operated relief valves, and the air-operated spray valves. Although failure of these valves is also considered not credible, failure of the valve bonnet body bolts is nevertheless postulated, and provisions are made to assure integrity of the Containment liner from the resultant bonnet missile.

To the extent practical, all valves are also oriented so that a missile will strike a barrier.

All BOP valves in high energy systems located inside the Containment Building are evaluated.

b. Thermowells

The only credible source of jet-propelled missiles from the reactor coolant piping and piping systems connected to the RCS is that represented by the temperature and pressure sensor assemblies. The resistance temperature sensor assemblies

can be of two types: with well and without well. Two rupture locations have been postulated: around the weld (or thread) between the temperature element assembly and the boss for the without well element, and the weld (or thread) between the well and boss for the with well element.

A temperature sensor is installed on the reactor coolant pumps close to the radial bearing assembly. A hole is drilled in the gasket and sealed on the internal end of the steel plate. In evaluating missile potential, it is assumed that this plate can break and the pipe plug on the external end of the hole can become a missile.

In addition, it is assumed that the welding between the instrumentation well and the pressurizer wall can fail and that the well and sensor assembly can become a jet-propelled missile.

c. Pressurizer Heaters

Finally, it is assumed that the pressurizer heaters can become loose and become jet-propelled missiles.

2. Equipment Outside of Westinghouse Scope

In order to conservatively establish the protection and redundancy requirements for safety-related equipment, credible missile generation from the following sources are evaluated:

a. Valves in high energy systems;

b. Temperature and Sensor Assemblies in high energy systems;

c. Vessels and Compressed Gas Cylinders in high energy systems; and

d. Dynamic Equipment.

3.5.1.2.2 Missile Description

The CRDM Missiles are summarized in Table 3.5-2.

The missile characteristics of the valve bonnets in the region where the pressurizer extends above the operating deck are given in Table 3.5-3 while the missile characteristics of the piping temperature sensor assemblies are given in Table 3.5-4. A 10-degree expansion, half angle water jet is assumed. The missile characteristics of the piping pressure element assemblies are less severe than those of Table 3.5-4.

The missile characteristics of the reactor coolant pump temperature sensor, the instrumentation well of the pressurizer, and the pressurizer heaters are given in Table 3.5-5. A 10-degree expansion, half-angle water jet is assumed.

The missile characteristics of equipment that are in Westinghouse scope are given in Table 3.5-6.

3.5.1.2.3 Missile Protection

Internal missiles which can be generated from pressure-containing components that are part of the RCS or main steam and feedwater systems are considered in the design of the Containment. The entire RCS and parts of the main steam and feedwater system are surrounded by the secondary shield wall and their components, arranged so that a missile generated from one component does not damage any other component whose failure can lead to an offsite radiation dose in excess of that prescribed in 10 CFR Part 100. Equipment that is not in the Westinghouse scope is either surrounded by concrete barriers, provided with redundant counterparts or designed to adequately resist missile generation or external penetration.

Missile barriers and missile resistant structures are of sufficient thickness and strength to prevent emission of secondary missiles when struck by a primary missile.

3.5.1.2.4 The Control Rod Drive Shaft Missile Shield

Unit 1 Missile Shield

Damage to equipment inside the Containment or to the Containment itself caused by missiles generated by the failure of a CRDM housing (as described in Section 3.5.1.2.1) is prevented by an integral steel missile shield. This missile shield is circular with an approximate outside diameter of 134 inches, constructed of an array of W6 x 16 carbon steel beams welded to the top of a 1.5 inch thick carbon steel plate and built to seismic Category I requirements. The missile shield is part of the CRDM cooling plenum assembly and is approximately 5 feet above the top of the CRDM pressure housings. The missile shield's array of W6 x 16 carbon steel beams are bolted to the plenum's carbon steel support column assembilies. These support columns are bolted to plenum's upper ring girder and lower ring girder. The plenum is mounted to the head assembly and is positioned above the seismic support platform and the CRDM and DRPI cable connections. The plenum assembly is attached to the head lift rig via the lift rig extension sections. The plenum missile shield assembly remains installed of the reactor vessel head assembly during refueling.

Unit 2 Missile Shield

Damage to equipment inside the Containment or to the Containment itself caused by missiles generated by the failure of a CRDM housing (as described in Section 3.5.1.2.1) is prevented by a roll-away steel missile shield. (See Figure 3.5-1) This missile shield is approximately 16 ft. square and constructed of 2-inch-thick carbon steel plate and built to seismic Category I requirements (no function after the LOCA is required). The shield which is located over the refueling cavity and is approximately 2 feet above the top of the CRDM housings is mounted on the manipulator rails for ease of removal during refueling. (See Figure 3.5-2.)

The CRDM cooling fans, along with cable trays containing cables which are connected to the CRDMs, are mounted on the missile shield. Neither the fans nor the cables are nuclear-safety-related and therefore do no have to be protected from damage resulting from missile impact on the shield.

During refueling, the cables are removed from the cable trays and the shield is rolled to the far end of the manipulator rail where it is stored throughout refueling. The fans are not removed

from the shield during refueling. When refueling is completed, the shield is positioned over the reactor and the cables are reinstalled in the cable trays.

3.5.1.2.5 Secondary Missiles

As stated in Section 3.5.1 the basic approach to ensure missile protection of safety systems and equipment inside the Containment Building has been the assurance of the design adequacy of equipment against the generation of missiles rather than to try to contain the missile. The precautions taken to prevent missile generation are as described in Section 3.5. In spite of the design precautions, the generation of potential missiles is postulated and the provision for protection against them is described in Section 3.5. It has not been postulated that secondary missiles are generated by impingement of these missiles.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Turbine Placement and Orientation

The turbine placement and orientation of the twin unit are indicated in the plant layout drawing, Figure 3.5-3. The orientation of the turbine axis is radial to the containment. There are no essential systems or structures located inside the low trajectory missile zones defined in NRC Regulatory Guide 1.115; therefore only high trajectory missiles need be considered.

3.5.1.3.2 Missile Identification and Characteristics

Each of the two CPNPP low pressure turbines is double flow Siemens Westinghouse 1800 rev/min steam turbine-generator with 46-inch last stage blades designed for Light Water Reactor (LWR) applications. Each rotor is made from a stepped shaft with a total of 8 shrunk-on blade disks arranged in symmetrical groups of four disks per flow.

Reference [15] provides a description of the weight, dimensions and location of each disk relative to the turbine center. In addition, this report describes the manufacturing process and design features of the turbine disks used to reduce or eliminate the potential for stress corrosion cracking (SCC) of the disks which could lead to a burst of the turbine disks.

As discussed in Reference [15], the generation of external turbine missiles by virtue of turbine casing penetration does not occur until speeds exceed 145% of rated speed for all four disks of the low pressure turbines. This speed is based upon design values using deterministic missile energy analysis techniques. The calculated speeds at which final ductile burst of disks would occur exceed 160% of rated speed for all four disks.

As presented in Reference [15], the probability of generating external turbine missiles resulting from a hypothetical LP turbine disk failure which could adversely affect safety related SSCs is less than the NRC threshold of 1E-07 per year for favorably oriented turbines. As can be seen in Figure 3.5-3, the turbines are oriented to prevent low-trajectory missiles from impacting safety related SSCs. The missile deflection angles postulated for a last stage disk burst is based upon 25 degrees. [9]

FSAR Section 3.5.1.3.3 describes the probabilistic analysis used in evaluating the generation and associated threat of external turbine missiles for the low pressure turbines.

The largest potential missile from a hypothetical LP turbine disk failure is a missile of the last stage disk. The missile deflection angles postulated for the last stage disk were to 25 degrees. [9]

The mathematical model for selecting the missile size and its sector angle assumes maximization of the translational energy is described in Appendix 3.5A. [8]

3.5.1.3.3 Strike Probability Analysis

The missile probability analysis used for this turbine is based upon methodologies and principles previously accepted by the NRC at plants such as Grand Gulf, Connecticut Yankee, Limerick as well as the original Comanche Peak turbine design. Results are based on theoretical design specifications for disc material properties rather than the actual material properties. [17]

The analysis for this turbine design is presented and summarized in Reference [15]. The analysis is based upon previously accepted external missile probabilities based upon the design of the turbine, turbine inspection interal, overspeed protection test interval and vulnerability of plant SSCs relative to the orientation of the turbine.

The overall probability, P4, of generating external turbine missiles which could pose a threat to safety related SSCs is calculated as follows:

P4 = P1 * P2 * P3

where:

P1 = Probability of occurrence of turbine missile per turbine year

P2 = Strike probability

P3 = Probability of penetration and damage due to the strike

For this turbine design, the following probabilistic values are used.

P1 = 1.56E-4 at 100,000 hours (inspection interval) for a 26 weeks turbine valve test frequency [15]

Since the turbines are oriented favorably to the plant (that is, the safety related structures housing safety related equipment are located outside the postulated low trajectory missile strike zone – see Figure 3.5-3), P2*P3 probability is assumed to equal 1E-3 per year. This value is consistent with credit given by the NRC for favorably oriented turbines [16].

With the use of P2*P3 as assumed above, the focus of the probability analysis shifts to the prevention of an external turbine missile. Since P1 at 100,000 hours inspection interval is less than the NRC limit of 11.42E-4 when adjusted to 100,000 hours inspection interval versus 1E-4 per year per Table 1 of Reference [16], the overall probability, P4, is less than the overall NRC limit of 1E-7 per year, thereby ensuring an acceptable risk rate for the loss of an essential system from a single event. Hence this turbine design satisfies regulatory requirements and is thereby acceptable.

3.5.1.3.4 Turbine Overspeed Protection

For a detailed description of the operational mechanism for the overspeed protection components of the turbine, refer to Section 10.2.2. The components reliability analysis has been based on the following assumptions:

1. Failure rates for the turbine valves and controls derived primarily from actual operating experience and calculated with a statistical confidence level of 95 percent

2. The turbine valves are tested every 26 weeks. The overspeed protection system is automatically tested periodically during each shift and is verified operational every 2 weeks. The turbine trip block is tested every 2 weeks.

3. An average of one true load rejection per year

4. Consideration of the turbine extraction of common mode failure possibilities in addition to random failures

Thus, the overspeed failure probability has been estimated to be very small (1.37×10^{-5} per turbine-year for a four flow turbine generator) and 26 week turbine valve test interval [15], and subsequently a high degree of component reliability exists.

3.5.1.3.5 Turbine Valve Testing

Periodic testing of the stop and control valves is necessary to insure reliability and continuity of service; therefore, valve testing for all nuclear units is recommended with a manual test or with the aid of an automatic turbine tester (ATT) every 26 weeks.

For high reliability of the electrohydraulic control (EHC), one primary AC power source for valve testing is provided from safeguard bus; two separate DC sources for control circuits are provided from one +24 VDC system with common neutral. For more details of inservice inspection, refer to Section 10.2.3.6.

3.5.1.3.6 Turbine Characteristics

The turbine is a multicasing, tandem compound, four flow, reaction type, 1800-rpm unit with 46-in. last stage blades. For more detailed characteristics, refer to Section 10.2.2.

For major components, rotor materials and their properties, steam environment, and other properties related to the turbine disk integrity, refer to Section 10.2.3.

Turbine operational and transient characteristics are analytically described in Section 10.2.1 (Design Bases).

3.5.1.4 Missiles Generated by Natural Phenomena

The seismic Category I structures are designed for the effects of tornado-generated missiles. The missile characteristics are shown in Table 3.5-8.

Missiles, generated by floods, are not a design consideration as the PMF is lower than grade elevation. The Service Water Intake Structure is designed for tornado-generated missiles, among other loads. This provides adequate protection for any low velocity waterborne object.

The safety-related systems/components and non-safety related 1E components not contained within reinforced concrete buildings or structures are:

1. Service Water traveling screens

2. Diesel fuel oil storage tanks

3. Instruments located in the turbine building

4. Feedwater control and bypass valve appurtenances

5. Auxiliary Feedwater Pump (AFWPT) Exhaust Stack

The safety related Service Water traveling screens are intended for long term protection of the Service Water System from accumulation of debris (ie. CCW Heat Exchanger tube plates) and for short term protection against floating debris, small enough to pass through the trash racks positioned ahead of the screens, but large enough to potentially damage the Service Water Pumps.

In the event that the exposed parts of both screens, are struck by missiles, the only mechanical consequence is that the screens could lose their ability to rotate. The screens are positioned in channels which prevent the screen from collapsing following a postulated missile strike on the non-protected portion.

Local essential lighting for the Service Water System is located within the reinforced concrete service water structure but adjacent to a door not designed to resist damage from a tornado missile. The essential lighting fixtures, fixture primary supports, lamps and bulbs are classified non-nuclear safety grade. Essential lighting systems are discussed in Section 9.5.3.2.

The diesel fuel oil storage tanks and associated piping are located underground adjacent to the diesel building and are protected by a 1 foot 9 inch thick reinforced concrete slab at grade level. Manholes for the tanks are protected by steel plate covers with a minimum thickness of 1 3/4 inches.

Instruments that actuate anticipatory reactor trip on turbine trip are located in the turbine building as shown on Figure 7.1-3, sheet 16. They are included in the design as good engineering practice. The instruments are procured as Class 1E but are not Nuclear Safety Related and are not protected from tornado missiles because their function is anticipatory, not essential. The anticipatory reactor trip on turbine trip is discussed in Section 7.2.1.1.2 item 6.

The electrical appurtances on the Feedwater control and bypass valves, and the cables and junction boxes serving them, are non-safety related, but designated as Class 1E. Credit is taken for these valves in the analysis of the Feedwater Line Breaks and Main Steam Line Breaks inside Containment as non-safety related backup to the Nuclear Safety Related Feedwater Isolation Valves. This equipment is not protected against tornado missiles because these pipe rupture events are not postulated to occur simultaneous with a tornado.

The Turbine Driven Auxiliary Feedwater (TDAFW) pump is required to be operational for response to an occurrence of a station blackout [i.e. SBO = loss of offsite power (LOOP) + loss of onsite AC power (LOSP)]. As identified in FSAR 15.2.5.1, the turbine exhausts the secondary steam to the atmosphere. The occurrence of a tornado missile striking the exhaust pipe of the Auxiliary Feedwater Pump Turbine (AFWPT) with a large missile concurrent with SBO is considered highly improbable based on the location of the exhaust pipe relative to the elevations of source missiles.

The AFW system is designed with sufficient redundancy and diversity to ensure the plant can be safely shutdown and maintained in a safe condition. The motor driven pumps and the TDAFW pump provide the required redundancy for DBAs such as a main feedwater line break or diversity for events such as SBO. The AFWPT exhaust pipe is designed to withstand the effects of a DBT wind load.

The impact of tornado-generated missiles upon the AFWPT exhaust stack has been evaluated for the most critical conditions and the scenario. It was found in that analysis that the exhaust stack would not crimp shut in a manner that would adversely impact plant SSCs or shutdown capability. The cross-sectional fracture of the pipe is anticipated to occur during a missile strike, allowing steam to exhaust from the line. Other scenarios of missile strikes, such as vertical missiles acting on the pipe are considered extremely unlikely in causing blockage of steam. Therefore, no significant release of steam inside the building would be likely.

The exposed section of the Main Steam line from the AFWPT above the Safeguards Building Roof, if struck by a tornado missile, would result in no adverse condition to the plant because the safe shutdown function is met by two 100% motor driven AFW pumps during a LOOP event. The analysis for the MS system did not require the qualification of the exhaust pipe for such interaction and did not consider the effects of such a tornado missile strike concurrent with an SBO when the AFWPT would be required due to the highly improbable combination of loss of offsite power with loss of onsite AC power concurrent with a tornado and associated missiles.

All other safety related equipment is located within reinforced concrete structures.

The Refueling Water Storage Tank vents described in Section 6.3.3.7 and the Condensate Storage tank vents described in Section 9.2.6 are classified as non-Nuclear Safety related and, in accordance with ANSI N18.2a, are designed to break away if struck by a tornado missile. The exterior walls and roofs of all building housing safety-related equipment are of reinforced concrete and are designed to protect safety-related equipment against damage from the effects, including tornado missile effects, of the design basis tornado of Region I.

The buildings and structures housing safety-related equipment are shown on FSAR Plot Plan Figure 1.2-1.

Barriers are located at all exterior wall and roof openings of buildings or structures housing safety related equipment to prevent tornado missiles from damaging the contained safety related equipment.

The types of barriers used are:

1. Door openings - All safety related equipment is protected against damage from tornado missiles by missile resistant steel doors or steel plates completely covering the opening.

2. Ventilation Pressure Barriers - Where openings are provided for blow-out panels, tornado dampers, or air intake louvers, all safety related equipment is protected against damage from tornado missiles either by exterior missile resistant reinforced concrete barriers completely surrounding the opening or by missile resistant concrete barriers inside the building. In cases where a postulated missile is stopped by an interior concrete barrier and could then fall into the structure, a grating is provided.

3. Hatch Openings - All safety related equipment is protected against damage from tornado missiles by missile resistant steel plates completely covering the roof openings. Manhole covers on top of all outdoor seismic Category I tanks are steel plate with a thickness of no less than 1 3/4 inches.

   Tornado missiles are postulated to impact normal to the plant structure with maximum mass and velocity as defined in Table 3.5-8 of the FSAR.

3.5.1.5 Missiles Generated by Events Near the Site

As described in Section 2.2, potential missiles generated by events near the site will not endanger any safety-related facilities.

3.5.1.6 Aircraft Hazards

As discussed in Section 2.2, aircraft missiles are not considered a design basis.

3.5.2 SYSTEMS TO BE PROTECTED

All plant structures, systems, and components whose failure can lead to offsite radiological consequences or which are required to shut down the reactor and maintain it in a safe condition, assuming an additional single failure, are listed in Section 3.2. In general, reinforce concrete floors and internal compartment walls are used as barriers for protection of all essential components and systems against postulated missiles. For arrangement of floors and internal walls, see Figures 3.8-1 through 3.8-4. For general arrangement, see Figures 1.2-1 through 1.2-46.

3.5.3 BARRIER DESIGN PROCEDURES

Only one missile is postulated at a time and is assumed to strike the barrier end-on.

The missiles have two effects on structures, walls, or any other barrier: local effects and overall response. The local effects include penetration, perforation, and spalling or scabbing. The overall response includes the flexural and shear effects.

Reinforced concrete external roofs and walls of seismic Category I structures form barriers against tornado-generated missiles. Buried underground, safety-related duct runs and piping are also protected.

3.5.3.1 Local Effects

The estimate of missile penetration (D) in concrete barriers is based on the Modified Petry Formula [1]. Sufficient thickness of concrete is provided to prevent perforation, and when

required, to prevent backface spalling or scabbing. To prevent perforation, a concrete thickness of at least twice the penetration thickness determined for an infinitely thick slab is provided. The thickness (T) of the concrete barrier, to prevent backface spalling or secondary missiles which are injurious to safety-related equipment, is determined by the following equation [1]:

$$T = 2.4D$$

D is the maximum penetration in an infinitely thick slab. A minimum concrete barrier thickness of 30 inches for exterior walls and 21 inches for roofs is provided for protection from tornado-generated missiles. See Section 3.8.4.1.7 for a discussion of the properties of the concrete and typical reinforcement details.

Determination of penetration by missiles into steel plates is in accordance with the Stanford Research Institute Formula [2], or the Ballistic Research Lab Formula [10].

3.5.3.2 Overall Barrier Response

The overall response of structural barriers to missile impact depends upon the available ductility. This ductility is a function of the controlling nature of the structural behavior. Thus, for concrete beams, walls, and slabs, where flexure controls design, the permissible ductility ratio is taken as 0.05/(p - p'), p being the ratio of tension reinforcement and p', the ratio of compression reinforcement. However, the ductility ratio does not exceed 10.0 for beams, walls, and slabs. For beam, walls, and slabs where shear may control design, the ductility ratio is 1.0 when shear is carried by concrete alone, and 1.3 when shear is carried by concrete and stirrups or bent bars. Flexural strength is determined based on ultimate strength theory with the limitations on ductility as follows:

1. For beam-column members where the compressive load does not exceed either 0.1 f'C Ag or one-third of that which would produce balanced conditions, whichever is smaller, the allowable ductility ratio is 10.

2. For beam-column members where the design is controlled by compression, the allowable ductility is 1.3.

3. For members which are between these two values, ductility shall be taken as decreasing linearly from 10 to 1.3.

For steel barriers, the ductility ratio does not exceed 10.0 for flexure, compression or shear. For steel columns proportioned to preclude elastic buckling, (l/r equal to or less than 20) the limiting ductility ratio is 1.3. For steel columns not proportioned to preclude elastic buckling (l/r greater than 20), the limiting ductility ratio is 1.0, where l is the effective length of the member and r is the least radius of gyration. For steel tension members, the limiting ductility ratio is taken as 0.5 Eu/Ey, where Eu is the ultimate strain and Ey is the yield strain.

The design load, caused by the missile, is based on the absorption of the kinetic energy of the missile by the target at its maximum Deflection [4]. It is limited by the yielding, buckling, crushing, or local failure of the missile.

REFERENCES

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TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 1 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Containment Spray System				
Containment spray pumps	Safeguards building, floor elevation 773 ft 0 in	Section 6.2 Figure 1.2-10 Figure 1.2-16	Valve bonnet or part from rotating equipment	Two pumps for each train are located in an enclosed cubicle Pumps cannot be damaged by failed components from other systems.
Spray additive tank	Safeguards building, elevation 790 ft 6 in.	Sections 6.2.2, 6.5 Figure 1.2-10 Figure 1.2-16	Valve bonnet or part from rotating equipment	Tank is located in isolated compartment.
Containment spray heat exchangers	Safeguards building, floor elevation 790 ft 6 in.	Section 6.2.2 Figure 1.2-10 Figure 1.2-16	Missile from valve in adjacent valve room	Heat exchanger is separated from valve room by wall.
Piping and valves	Safeguards building	Section 6.2.2	Valve bonnet from other pipes in pipe chase	Redundant train
Chemical eductor	Safeguards building, elevation 786 ft 7 in.	Section 6.5	Valve bonnet or parts from rotating equipment	Redundant train
Containment sumps valve isolation tanks	Safeguards building, elevation 790 ft 6 in.	Section 6.2 Figure 1.2-10 Figure 1.2-16	None postulated	Tanks for redundant trains are separated from each other by concrete barriers.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 2 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Service Water System				
Pumps	Pump house	Section 9.2 Figure 1.2-45	Missile from service water pump	Redundant and separated pumps
Piping and valves	Pump house, auxiliary and safe guards buildings	Section 9.2	Valve missile from other piping in pipe chase	Redundant and separated trains
Spent Fuel Pool Cooling System (not needed for reactor shutdown)				
Spent fuel cooling pumps	Fuel building, floor elevation 810 ft 6 in.	Section 9.1 Figure 1.2-38	Missile from spent fuel cooling pump	Heat exchanger and pump of one train share a common cubicle which is separated from the other train by a wall.
Spent fuel cooling heat exchangers	Fuel building, floor elevation 810 ft 6 in.	Section 9.1 Figure 1.2-38	Missile from spent fuel cooling pump	Only one train is needed to cool both pools except during abnormal design conditions
Piping and valves	fuel building	Section 9.1	Missile from pump	Pumps are enclosed in concrete cubicle with piping associated only with that train.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 3 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Component Cooling Water System				
Pumps	Auxiliary building, floor elevation 810 ft 6 in.	Section 9.2 Figure 1.2-32	Missile from other CCWS pump	Pumps from redundant trains in separate cubicles are separated by corridors. Pump from same train of different units is separated by wall.
Heat exchangers	Auxiliary building, floor elevation 790 ft 6 in.	Section 9.2 Figure 1.2-31	None postulated	Four heat exchangers (two per unit) are in a common compartment. No pumps or high pressure valves are in area.
Surge tank		Figure 1.2-35		
Piping and valves	Auxiliary building	Section 9.2	Valve bonnet	Redundant and separated trains
Auxiliary Feedwater System				
Condensate storage tank	Yard	Section 10.4.9 Figure 1.2-1	None postulated	Tank is shielded against tornado-generated missiles.
Pumps	Safeguards building, floor elevation 790 ft 6 in.	Section 10.4.9 Figure 1.2-10 Figure 1.2-16	Missiles from other auxiliary feedwater pumps	Each pump is enclosed in a separate cubicle.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 4 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Piping and valves	Yard and safeguards building	Section 10.4.9	Missiles from other auxiliary feedwater pumps	Each pump is enclosed in a separate cubical with its train associated piping.
Main Steam Supply System				
Piping and valves	Safeguards building, floor elevation 873 ft 6 in.	Section 10.3 Figure 1.2-14 Figure 1.2-20	Valve bonnets from safety and relief valves	Each steam line is located in separate compartment. Each compartment is divided to separate relief and safety valves from isolation valve.
Power-operated relief valves	Safeguards building, floor elevation 885 ft	Section 10.3 Figure 1.2-14 Figure 1.2-20	Valve bonnets from safety of same steam line	Safety valves are pointed away from relief valve and each other.
Feedwater System				
Piping and valves	Safeguards building	Section 10.4.7.2	Valve bonnets from isolation valves	Each line is in own separate cubicle. Valve bonnet from one isolation valve cannot damage any other feedline.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 5 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Safety Injection System				
Refueling water storage tank	Yard	Section 6.3 Figure 1.2-1	No postulated internal source	Tank is shielded against tornado-generated missiles.
Piping and valves	Safeguards building and yard	Section 6.3	Valve parts from other systems	Redundant trains
High-head SIS pumps	Safeguards building, floor elevation 773 ft 0 in	Section 6.3 Figure 1.2-10 Figure 1.2-16	No postulated internal source	Each safety injection pump is in its own cubicle on opposite sides of the safeguards building
Boron injection tank	Safeguards building, floor elevation 810 ft 6 in.	Section 6.3.2	N/A	Tank has been deleted from the CPNPP design.
BIT recirculation pump	Safeguards building, floor elevation 810 ft 6 in.	Section 6.3.2	N/A	Pump has been deleted from the CPNPP design.
Boron injection surge tank	Safeguards building, elevation 852 ft 6 in.	Section 6.3.2	N/A	Surge tank has been deleted from the CPNPP design.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 6 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
CVCS and Liquid Poison System				
Centrifugal charging pumps	Auxiliary building, floor elevation 810 ft 6 in.	Section 9.3.4 Figure 1.2-32	No postulated internal missile per discussion on Westinghouse-supplied pumps	Charging pumps are in separate cubicles with walls between them.
Boric acid tanks	Auxiliary building, floor elevation 810 ft 6 in.	Section 9.3.4 Figure 1.2-32	No postulated missile	Two tanks per unit are located in common cubicle with no missile sources in or near cubicle.
Piping and valves	Auxiliary and safeguards parts buildings	Section 9.3.4	Missile from valve or pump shutdown.	Redundant trains for parts of system are needed for safe
Residual Heat Removal System				
Containment sumps valve isolation tanks	Safeguards building floor elevation 790 ft 6 in.	Section 5.4.7 Figure 1.2-10 Figure 1.2-16	Part from containment spray valve	Spray valve located in separate tank on opposite side of column and redundant trains.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 7 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Pump	Safeguards building, floor elevation 773 ft 0 in.	Section 5.4.7 Figure 1.2-10 Figure 1.2-16	Missile from containment spray pump in next room	Each RHR pump is in a separate cubicle with walls separating pumps from each other and from containment spray pumps.
Heat exchanger	Safeguards building, floor elevation 90 ft 6 in.	Section 5.4.7 Figure 1.2-10 Figure 1.2-16	Missile from valve in adjacent valve room	RHR heat exchangers are separated from the valve room by a wall. Two redundant trains are provided.
Hydrogen Purge System				
Filters (charcoal, HEPA, roughing)	Auxiliary building, elevation 873 ft 6 in.	Section 6.2.5 Figure 1.2-35	No postulated missiles from other internal sources other than fan in same train	Trains A and B are isolated.
Fans	Auxiliary building, elevation 873 ft 6 in.	Section 6.2.5 Figure 1.2-35	No postulated missiles from external sources	Trains A and B are isolated.
Piping and valves	Auxiliary building	Section 6.2.5	No postulated missile	None provided; there are no high-pressure piping or pumps nearby.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 8 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Containment Isolation System				
Discussed in Section 6.2.4				
Control Room HVAC				
Air conditioning units (roughing filter, fan, heating and chilled water cooling coils)	Control building, elevation 854 ft 4 in.	Section 9.4.1 Figure 1.2-34	Fan wheel from fan in same train	Fan housing and separation wall between trains; one train can be damaged but other is protected
Emergency filtering unit (roughing, charcoal and HEPA filter and fans)	Control building, elevation 854 ft 4 in.	Section 9.4.1 Figure 1.2-34	Fan wheel from fan in same train	Fan housing and separation wall between trains; one train can be damaged but other is protected.
Piping, valves, and dampers	Control building, elevation 854 ft 4 in.	Section 9.4.1	Fan wheel from fan in same train	Fan housing and separation wall between trains; one train can be damaged but other is protected.
Fans	Control building, elevation 854 ft 4 in.	Section 9.4.1 Figure 1.2-34	Wheel in fan	Fan housing and separation wall between trains; one train can be damaged but other is protected.
Safeguards building emergency closed loop air cooling system (cooling coils, fans)	Inside of various pump rooms	Section 9.4	Missile from pump which the unit cools	Separate cooling units for redundant pumps.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 9 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Fuel Building Ventilation System (For Fuel Accident Only)				
Filters (roughing, charcoal, HEPA)	Auxiliary building, elevation 873 ft 6 in.	Section 9.4.2 Figure 1.2-35	Fan blade	Complete redundancy and separation of trains
Fans (exhaust)	Auxiliary building, elevation 873 ft 6 in.	Section 9.4.2 Figure 1.2-35	Fan blade	Complete redundancy and separation of trains
Ducts (exhaust)	Fuel and auxiliary buildings	Section 9.4.2	Fan blade	Complete redundancy and separation of trains
Standby Power System				
Diesel generators	Diesel generator building, floor elevation 810 ft 6 in.	Section 8.3 Figure 1.2-11 Figure 1.2-17	Piston from one diesel generator	Each diesel generator is located in separate room with missile barrier between generators.
Diesel generator supportive subsystems (including lubrication, intercooling, and jacket water systems)	Diesel generator building, floor elevation 810 ft 6 in.	Section 9.5.5 Figure 1.2-11 Figure 1.2-17	Piston from one diesel generator	Supportive subsystems are in room with generator they service; redundant subsystem are for each generator.

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 10 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Diesel Generator Starting System				
Air compressors	Diesel generator, floor elevation 844'-0"	Section 9.5.6 Figure 1.2-12 Figure 1.2-18	Piston from diesel generator	Separate starting system for each diesel in same room as diesel.
Air receivers	Diesel generator floor elevation 844'-0"	Section 9.5.6 Figure 1.2-12 Figure 1.2-18	Piston from diesel generator	Separate starting system for each diesel in same room as diesel.
Piping and valves	Diesel generator floor elevation	Section 9.5.6	Piston from diesel generator	Separate starting system for each diesel in same room as diesel.
Emergency Diesel Fuel Oil Storage and Transfer System				
Transfer pumps	Diesel generator building, floor elevation 844’-0”	Section 9.5.4 Figure 1.2-12 Figure 1.2-18	Diesel generator piston	Located in pit in room of diesel generator that it services; each generator has separate pumps.
Storage tanks	Buried in yard	Section 9.5.4 Figure 1.2-1	None postulated	Tank is buried in ground. Piping is separated and buried underground.
Day tanks	Diesel generator building, floor elevation 844'-0"	Section 9.5.4 Figure 1.2-12 Figure 1.2-18	Diesel generator piston	Separate tank for each diesel generator

TABLE 3.5-1
INTERNALLY GENERATED MISSILES OUTSIDE CONTAINMENT

(Sheet 11 of 11)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Diesel fuel filter	Diesel generator building, floor elevation	Section 9.5.4	Diesel generator piston	Separate filter for each unit
Piping and valves (required for continued operation of diesel generators)	Yard and diesel generator building	Section 9.5.4	Diesel generator piston (only for piping in building)	Interior piping and valves are confined to the cubical containing the diesel which they serve.

TABLE 3.5-2A
SUMMARY OF THE UNIT 1 CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

	Calculation Data			
Postulated Missiles	Missile Weight (lb)	Impact Velocity (ft/sec)	Kinetic Energy (ft-lb)	Assumptions
Drive shaft	133	140	40478	------

TABLE 3.5-2B
SUMMARY OF THE UNIT 2 CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

Typical Examples of Postulated Missiles	Calculation Data				Assumptions
	Missile Weight (lb)	Impact Velocity (ft/sec)	Kinetic Energy (ft-lb)	Penetration (in.)	
CRDM housing plug becomes loose and accelerated by the	11	380	24700	$\leq$1.69	Plug is water jet
Drive Shaft (Drive Rod)	120	173	55800	$\leq$1.69	------
CRDM and drive shaft latched together	1500	25.7	15400	$\leq$1.69	------

TABLE 3.5-3
VALVE MISSILE CHARACTERISTICS

Missile Description	Weight (lb)	Flow Discharge Area (in^2)	Thrust Area (in^2)	To Impact Area (in^2)	Impact Area Ratio (psi)	Velocity (ft/sec)
Safety relief valve bonnet (3 in. x 6 in. or 6 in. x 6 in.)	350	2.86	80	24	15.6	110
3-in. motor-operated isolation valve bonnet, plus motor and stem (3 in.)	400	5.5	113	28	14.1	135
2-in. air-operated relief valve bonnet, plus stem	75	1.8	20	20	3.75	115
3-in. air-operated spray valve bonnet, plus stem	120	5.5	50	50	2.4	190
4-in. air-operated spray valve	200	9.3	50	50	4	190

TABLE 3.5-4
PIPING TEMPERATURE ELEMENT ASSEMBLY MISSILE CHARACTERISTICS

1. For a Tear Around the Weld Between the Boss and the Pipe

	Without Well	With Well
Flow discharge area, in^2	0.11	0.60
Thrust area, in^2	7.1	9.6
Missile weight, lb	11.0	15.2
Area of impact, in^2	3.14	3.14
Missile Weight		
Impact Area, psi	3.5	4.84
Velocity, ft/sec	20	120

2. For a Tear at the Junction Between the Temperature Element Assembly and the Boss for the Without Well Element and at the Junction Between the Boss and the Well for the With Well Element

	Without Well	With Well
Flow discharge area, in^2	0.11	0.60
Thrust area, in^2	3.14	3.14
Missile weight, lb	4.0	6.1
Area of impact, in^2	3.14	3.14
Missile Weight		
Impact Area, psi	1.27	1.94
Velocity, ft/sec	75	120

TABLE 3.5-5
CHARACTERISTICS OF OTHER MISSILES POSTULATED WITHIN REACTOR CONTAINMENT

	Reactor Coolant Pump Temperature Element	Instrument Wall of Pressurizer	Pressurizer Heaters
Weight, lb	0.25	55	15
Discharge area, in^2	0.50	0.442	0.80
Thrust area, in^2	0.50	1.35	2.4
Impact area, in^2	0.50	1.35	2.4
Missile Weight			
Impact area, psi	0.5	4.1	6.25
Velocity, ft/sec	260	100	55

TABLE 3.5-6
INTERNALLY GENERATED MISSILES (INSIDE CONTAINMENT)

(Sheet 1 of 2)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Control Rod Drive Mechanism (CRDM)	Containment Building	Section 5.3 Figure 1.2-12	Unit 1 Missile from CRDM drive shaft Unit 2 Missile from CRDM housing plug, drive shaft and drive mechanism latched together	Unit 1 Integral steel missile shield above CDRM housing Unit 2 Roll away steel missile shield above CRDM housing
Valves (Pressurizer) – pressurizer safety valves - motor operated isolation valves in the relief line - air operated relief valves - air operated spray valves	Containment Building	Section 5. 4	failure of valve body bonnet bolts	Valves orientated so that a missile will strike a missile barrier
Pressurizer Heaters	Containment building El. 853'-6"	Section 5.4 Figure 1.2-13	pressurizer heaters become loose	located in a concrete cubicle
Pressurizer	Containment Building El. 853'-6"	Section 5.4 Figure 1.2-13	Instrumentation well	located in a concrete cubicle

TABLE 3.5-6
INTERNALLY GENERATED MISSILES (INSIDE CONTAINMENT)

(Sheet 2 of 2)

System and Component	Location	Section for Description and Reference Drawings	Identification of Missiles	Missile Protection Provided
Reactor Coolant Pump (RCP)	Containment Building El. 832'-6"	Section 5.4 Figure 1.2-13	RCP temperature sensor pipe plug	oriented to strike a concrete barrier
Reactor Coolant System (RCS) Piping	Containment Building	Section 5.4	Temperature sensors	RCS is surrounded by the secondary shield wall and components are arranged so that a missile does not damage a component whose failure can lead to an offsite radiation dose in excess of that prescribed in 10 CFR 100

TABLE 3.5-7
HIGH TRAJECTORY MISSILE TARGETS AND TARGET AREAS

Target Building	Target Components	Target Building Area (Ft^2)	Target Components Area (Ft^2)
Unit 1 Safeguard Building	Main steam piping including main steam isolation valves, safety relief valves and main steam moments restraints	4,030	732
Unit 1 Diesel Generator Building	Diesel generator exhaust piping	4,160	712
Unit 1 Reactor Containment	Steam generators, reactor coolant pumps, reactor vessel and portion on reactor coolant piping, main steam piping	14,314	1,146
Auxiliary Building	Component Cooling water surge tanks, chilled water surge tanks	22,080	552
Fuel Building	Spent fuel pool	15,857	1,987
Unit 2 Reactor Containment	(Same as in Unit 1)	14,314	1,146
Unit 2 Safeguard Building	(Same as in Unit 1)	4,030	732
Unit 2 Diesel Generator Building	(Same as in Unit 1)	4,160	712
Electrical and Control Building	Entire building area included for conservatism	16,355	16,355
Service Water Intake Structure	Service Water System Components (Pumps and Piping)	4,686	562

TABLE 3.5-8
TORNADO-GENERATED MISSILES (HORIZONTAL MODE)

Missile		Mass (kg)	Dimensions (m)	Velocity (m/sec)	Kinetic Energy (kg-m)
A.	Wood plank	52	.092 x .289 x 3.66	83	18,300
B.	6-in. Schedule 40 pipe	130	.168D x 4.58	52	17,900
C	1-in. steel rod	4	.0254D x .915	51	530
D.	Utility pole	510	.343D x 10.68	55	78,600
E.	12-in. Schedule 40 pipe	340	.32D x 4.58	47	38,300
F.	Automobile	1810	5 x 2 x 1.3	59	321,000

Note:

Vertical velocities of 70 percent of the postulated horizontal velocities are used except for Missile C, which is used to test barrier openings. Missile C is assumed at the same speed in all directions. Kinetic energy for vertical impact is reduced according to missile velocity.

TABLE 3.5-9
TABLE 3.5-9 HAS BEEN DELETED

TABLE 3.5-10
TABLE 3.5-10 HAS BEEN DELETED

APPENDIX 3.5A - THEORY OF MISSILE ENERGY [8]

3.5A HAS BEEN DELETED

APPENDIX 3.5B - TURBINE MISSILE STRIKE PROBABILITY DISTRIBUTIONS [8]

3.5B HAS BEEN DELETED

3.6N PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6N HAS BEEN DELETED

3.6B PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

This section describes the design bases and protective measures used to ensure that all essential structures, systems and components required for a safe shutdown and to maintain the reactor in a cold shutdown condition are adequately protected from the dynamic effects associated with postulated pipe ruptures located inside and outside of the containment. The criteria for protection against pipe ruptures conforms to the requirements of Appendix A of 10 CFR 50 General Design Criterion 1, 2, 4 and 14. The pipe rupture protection criteria also conforms to the guidelines of NRC Regulatory Guide 1.46 [1], Standard Review Plans 3.6.1 and 3.6.2 [2], and Branch Technical Positions APCSB 3-1 [3] and MEB 3-1 [2].

3.6B.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS

This section sets forth the design bases, description, and safety evaluation for protection and determination of the dynamic effects associated with the postulated rupture of piping in fluid systems both inside and outside containment.

3.6B.1.1 Design Basis

The following design bases are utilized in determining the consequences of pipe failures on essential systems or components important to plant safety or shutdown (see Section 3.6B.1.2) which are located in the vicinity of high or moderate energy piping:

1. Piping systems, valves and components required to achieve a safe shutdown will be protected.

2. Required redundancy will be maintained in the protection system (IEEE Standard 279), Class 1E electric systems (IEEE Standard 308), ESF equipment, cable penetrations and their interconnecting cables.

3. HVAC equipment required for safe shutdown will be protected. Portions of Primary Plant Ventilation and associated chilled water are credited after 72 hours to mitigate the long term effects of HELBs outside containment. This is sufficient time to effect required repairs.

4. Instrumentation required for post accident monitoring will be protected as described in Section 7.5.1.3.1.3.

5. Containment leaktightness will be maintained.

6. LOCA breaks will not propagate to steam and/or feed water line breaks and vice versa.

7. A non-LOCA break will not be allowed to propagate into a LOCA.

8. LOCA break propagation to the unaffected Reactor Coolant System loops will be prevented.

9. LOCA propagation in the affected loop is permitted, but is limited as discussed in Section 3.6B.1.2.2.

10. In-core instrumentation lines will be protected.

The criteria for determining the location of pipe breaks is given in section 3.6B.2.1. A discussion of the effects of pipe failure on each system and typical piping runs with the location of failure points shown on drawings is given in subsection 3.6B.2.5 for high energy systems.

A list of high energy lines that are considered for pipe rupture analysis is included as Table 3.6B-1.

All large bore High Energy Lines in safety related structures are Seismic Category I and II. Small bore High Energy Lines in safety related structures are Seismic Category I or II except for the main steam line drip pot drain lines in the Safeguards Building, Main Steam and Feedwater Penetration Area. See Table 3.6B-1 for applicable line numbers.

3.6B.1.2 Description

Essential systems are defined as those systems that are needed to shut down the reactor and mitigate the consequences of the pipe break for a given postulated piping break.

3.6B.1.2.1 Protection Criteria

Depending upon the type and location of the postulated pipe break, certain safety equipment may not be classified as essential for the particular event. Some safety equipment will be essential for almost all cases. This category includes service water to the ultimate heat sink and the pressurizer level instrumentation. The containment integrity and leak tightness will be maintained for any LOCA break. The effects of a postulated piping failure, including environmental conditions resulting from the escape of contained fluids, will not preclude habitability of the control room or access to surrounding areas important to the safe control of reactor operations needed to cope with the consequences of the piping failure. Accordingly, protection from the effects of pipe rupture will be provided for only that safety-related equipment considered as essential on a case-by-case basis.

The systems or portions of systems and equipment for which protection against postulated pipe failures is required are identified below. However, in general, protection from pipe failure need not be provided if any of the following conditions exists:

1. The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or is restrained from whipping.

2. Following a single break, the unrestrained pipe movement of either end of the ruptured pipe about a plastic hinge formed at the location determined by calculation cannot impact any structure, system, or component important to safety.

3. The internal energy level associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

The following systems or portions of these systems are required to mitigate the consequences of a postulated pipe failure:

1. Auxiliary Feedwater System
2. Chemical and Volume Control System
3. Feedwater System
4. Main Steam System
5. Reactor Coolant System
6. Residual Heat Removal System
7. Safety Injection System
8. Safety Chilled Water System
9. Hydrogen and Nitrogen System
10. Containment Spray System
11. Diesel Generator System
12. Component Cooling Water System
13. Service Water System
14. HVAC Systems required for the operation of other systems
15. Containment Isolation System
16. Control Room Ventilation System
17. Class 1E electrical systems, ac and dc (including switchgear, cables, batteries and distribution systems)

The instrumentation required to mitigate the consequences of a postulated pipe break and/or for accident monitoring is discussed in Sections 7.4, 7.5 and 7.6.12.

The results of the environmental analysis, which discuss the effects of pressure, temperature, humidity, and flooding on safety related equipment, are provided in Sections 3.11 and 3.6B.2.5. The only high-energy lines located near the Control Room are the main steam and feedwater lines which are located outside the Control Room adjacent to the north and south walls. Accordingly, protection is provided as described in Section 3.6B.2.5 so that habitability of the Control Room is not jeopardized by postulated pipe failures.

3.6B.1.2.2 LOCA Break Propagation Criteria

A loss of reactor coolant accident is assumed to occur for a branch line break down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3.6B-10) on outgoing lines and down to and including the second check valve (Case III in Figure 3.6B-10) on incoming lines normally with flow. A pipe break beyond the restraint or second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line closes.

Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3.6B-10) a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the Reactor Coolant System are defined as “large” for the purpose of these criteria if they have an inside diameter greater than 4 inches. Rupture of these lines results in a rapid blowdown from the Reactor Coolant System and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

Branch lines connected to the Reactor Coolant System are defined as “small” if they have an inside diameter equal to or less than 4 inches. This size is such that Emergency Core Cooling System analyses using realistic assumptions show that no clad damage is expected for a break area of up to 12.5 square inches.

In order to assure the design function of essential systems in the event of a LOCA, break propagation within the affected loop shall be limited as follows:

1. Reactor Coolant Loop Piping

   Propagation of damage resulting from rupture of the main reactor coolant loop is permitted to occur but must not exceed the design basis for the containment, environmental qualification of equipment and Emergency Core Cooling System performance.

   The application of Leak-Before-Break technology to CPNPP Unit 1 and 2, discussed in Section 3.6B.2.1.1, has shown that the dynamic effects of main reactor coolant loop rupture, such as loop hydraulic forces, reactor internals reaction loads and primary equipment support loads, can be excluded from the design basis, as allowed by General Design Criterion 4 (Section 3.1.1.4).

2. Large Branch Lines

   In the event of a rupture of a large branch line resulting in a LOCA, propagation of the break in the affected loop must not exceed 20 percent of the flow area of the line which initially ruptured.

3. Small Branch Lines

In the event of a rupture of a small branch line resulting in a LOCA:

a. Break propagation must be limited to the affected leg of the affected loop.

b. Propagation of the break in the affected leg must be limited to a total break area of 12.5 square inches. However, no propagation is permitted when the initiating small break is the high head safety injection line.

c. Propagation of the break to a high head safety injection line connected to the affected leg is prevented if the line break results in a loss of core cooling capability due to a spilling injection line.

3.6B.1.2.3 Environmental Analysis to Determine HELB Temperature Transients for Purposes of Equipment Qualification (Outside Containment)

1. Design Bases

Environmental analyses are performed to model the thermal effects of HELBs on safety related areas and equipment outside containment that are needed for safe shutdown of the plant. This information is used to determine that the plant can be safely shutdown following a high energy line break.

The environmental effects of HELBs are analyzed for those systems where breaks or cracks are required to be postulated that have the potential of releasing steam or flashing water. These systems include Main Steam, Feedwater, Chemical and Volume Control, Steam Generator Blowdown and Auxiliary Steam.

2. Analytical Techniques

The COMPARE/MOD1A computer code is used to model the effects of HELBs. COMPARE is a thermal-hydraulic modeling code that is used for subcompartment transient analyses. It models the physical plant arrangement as a series of volumes and junctions. The volumes are made up of rooms, parts of rooms or groups of rooms that are combined together to form one volume. The junctions are naturally occurring flow paths based on the geometry of the plant.

The COMPARE code has the ability to calculate the transient conditions in up to 100 volumes that are connected by up to 200 junctions. Each volume is assumed to contain a homogeneous mixture of air, steam and water. The volumes may contain heat conduction elements. Mass and energy addition (blowdown) may also be added to a volume.

Mass and energy accounting is performed on each volume for each discrete time step. This then describes the thermodynamic conditions, which are assumed to be constant, from which the new junction flows are calculated. Several options are available to calculate the junction flows. These are a homogeneous equilibrium gas mixture, a two-phase Moody critical flow model or an inertial flow model based on the solution of the

one-dimensional momentum equation with critical flow based on one of the first two methods listed. The third is the option used for the C.P.S.E.S.

Doors whose angular positions are a function of pressure, spring forces, gravity forces or inertia forces can be represented to model the vent flow area.

Heat sinks can be modeled into the volumes and the associated parameters can include density, specific heat, initial conditions, boundary conditions, changes in materials and changes in heat transfer coefficients.

3. Plant Models

Because of the locations of the postulated high energy line breaks outside containment, several detailed computer models of the plant are developed. These models are used to calculate the effects of breaks occurring in the main steam and feedwater penetration areas, the safeguards building, the auxiliary building and the electrical and controls building. The models are based on the physical arrangement of the as-built plant and include door positions, tornado dampers, blowout panels, blowout doors, equipment removal gratings, ventilation transfer grills and physically isolated areas. The door positions are in accordance with those specified by the Tornado Analysis.

For each of the plant models many different kinds of input information is calculated. Some of the calculated input parameters include the room volumes corrected by subtracting the volume occupied by large piping and major equipment, the specific surface areas of the walls, floor and ceiling for input as a heat sink and junction inertial values (L/A) specific to each of the flow paths of the as-built plant. Also the exact free area of the tornado dampers, blowout panels and blowout doors are used, along with the geometry of each to accurately model the swing of each as a function of time and differential pressure.

Sensitivity analyses are performed by changing several input parameters, one at a time, to determine the adequacy of the parameters used in the analysis.

A survey is made of all the compartments in the plant that contain HELBs and the governing worst case break for each of the compartments is determined. Using the plant models mentioned above, each of the individual breaks that is the governing case for its compartment is then analyzed to determine the plant environment.

Using the results of the above analyses, a study is then performed to determine equipment affected by the environment. The affected equipment is then evaluated, on a case by case basis, to determine the equipment required for safe shutdown.

4. The governing results for each room and piece of equipment in terms of temperature, pressures and relative humidities are used as input to Section 3.11.

This input is then used to equate whether the existing demonstrated (maximum) qualification parameters envelope the short duration HELB parameters for temperature, pressure and relative humidity. Where the HELB parameters are not enveloped by the existing environmental qualification parameters, an Engineering Analysis is performed. This analysis is performed in order to justify continued operability during (where required)

and after the HELB event, with no appreciable degradation in performance or qualified life of the affected safety or non-safety related electrical equipment.

5. Break Exclusion Areas

For the main steam piping, the break exclusion region extends to the moment restraint downstream of the main steam isolation valve. The restraint is located next to the main steam isolation valve (at the outside wall of the Safeguards Building). It is designed to withstand the loadings resulting from a postulated piping failure beyond the break exclusion region so that neither isolation valve operability nor the leaktight integrity of the containment will be impaired.

For the main feedwater piping, break exclusion region extends to the moment restraint upstream of the main feedwater control valve. The moment restraint is designed to withstand the loadings resulting from a postulated piping failure beyond the break exclusion region. In addition to the main steam and main feedwater piping, about 10 ft of main feedwater bypass piping connected to each of the four main feedwater piping is also designated as a break exclusion region (Unit 2 only).

The piping in the break exclusion regions is designed to meet the requirements of the ASME B & PV Code, Section III, Subarticle NE-1120 and the additional design requirements specified in Branch Technical Position MEB 3-1.

The combination low stress and fatigue, in combination with augmented inservice inspection, precludes the possibility of pipe breaks or cracks in the break exclusion regions. However, for the purpose of determining environmental qualification requirements for safety related equipment in the break exclusion region, a non-mechanistic one square foot crack is postulated.

Method of Analysis

1. Introduction

A subcompartment multinode pressurization analysis has been performed using the COMPARE - MOD 1A computer code which is described in Reference [27]. The analysis results provide information on environmental effects of cracks in the 32-inch main steam pipe or the 18-inch main feedwater pipe within their respective compartments in the Safeguards Building. In addition, flooding due to the above has also been evaluated.

In accordance with the criteria stated in Section 3.6B.2.1.2, no specific pipe breaks are postulated in the above lines within the Safeguards Building. However, to provide an additional level of assurance of safety-related equipment operability in these compartments, the building structure and safety related equipment were reviewed for the environmental conditions (pressure, temperature and flooding), that would result from a non-mechanistic crack, equal in area to one square foot of either a main steam line or feedwater line. The mass and energy release data assumes that the non-mechanistic crack is independent of the location of the MSIV; that is, the open MSIV on the line with the postulated crack is not required to close on steam line isolation to limit the mass and energy release. This conservative assumption is the only failure of equipment outside the main steam and feedwater compartments or qualified equipment inside the compartment

for this scenario. Thus, no single failure or loss of offsite power need be assumed. In addition, the scenario need not assume that the event causes a loss of offsite power since FSAR Section 8.2.2 provides the results of analyses which show that loss of offsite power due to a reactor trip/turbine trip from 100% power is not probable. It is assumed that jets are not generated and hence jet loads are not considered.

Peak compartment temperatures were found to occur following the onset of a postulated MSLB using blowdown data that includes superheated steam effects. This analysis is detailed in FSAR Section 3.6B.2.5.2 and Tables 3.6B-4A (Unit 1), 3.6B-4B (Unit 2) and 3.6B-5.

The method used for the pressurization analysis of the structural compartments in the Safeguards Building is similar to the method used in the subcompartment analyses of the containment structure (Section 6.2.1.2) except that: (1) no asymmetric loading has been analyzed and (2) the type and size of break is limited to a one square foot non-mechanistic crack.

Analyses have been performed for each unit.

2. Compartment Nodalization

The affected compartments in the Safeguards Building containing the high energy lines are represented in a nodalization scheme by a series of volume nodes. The compartment boundaries are shown in Figure 3.6B-207. The line cracks were postulated in room numbers 108A through 108H for main steam and in room numbers 100A through 100H for feedwater, respectively.

In addition, flooding level calculations were based on a feedwater line crack in room numbers 100A, 100B, 100E and 100F.

The choice of nodal subdivisions was governed by the actual physical subdivision of the structure (natural boundaries) which form flow restrictions.

3. Vent Openings and Flow Obstructions

In determining the maximum levels of pressure and temperature resulting from the postulated cracks, a conservative approach was taken in that no credit was taken for utilizing the various building structures and equipment as heat sinks that would contribute to lowering the calculated pressure and temperature.

Venting between the affected compartments is obtained by the following:

a. Floor openings in main steam compartments nearest the Containment Building (room numbers 108A, 108B, 108C, 108D).

b. The seven-feet by three-feet openings in walls of all the compartments, each opening being located seven-feet above the compartment floor.

c. Pipe sleeve openings.

The total area considered for venting from the main steam and feedwater compartments consists of the following:

a. Pipe sleeve openings through the west wall of the Safeguards Building from room numbers 108E, 108F, 108G, 108H, 100E, 100F, 100G and 100H.

b. Pipe sleeve openings for the safety and relief valve vent stack piping through the roof of room numbers 108A, 108B, 108C and 108D.

Seals for these pipe sleeves are provided for weather protection. They have little strength and will "blow out" at a very low differential pressure. For the purpose of analysis, the seals are discounted. The vent openings are modeled as flow paths (or junctions) between each compartment.

For the purpose of calculating the temperature and pressure inside the main steam and feedwater compartments, flow obstruction is assumed to occur as a result of the failure of HVAC ducts passing between the various main steam and feedwater compartments from the compartments to outside areas. These ducts are assumed to fail in such a way as to block any flow that normally would pass through them. Also, the equipment hatches located on the roof of the main steam isolation valve compartments and the seals provided in the gaps between the Safeguards and Containment Buildings are assumed to remain in place.

Structural Design Margins

The main steam line and feedwater line compartments in the Safeguards Building are designed to withstand the environmental effects (pressure, temperature, humidity and flooding) of a non-mechanistic crack in a main steam or feedwater line with a flow area of one square foot. These effects are not combined with seismic effects and are not factored (the load factor is taken as unity). In addition it is assumed that jets are not generated and hence jet loads are not considered.

Mass and Energy Release Rates

1. Main Steam Line Break

The mass and energy release rates were obtained from Westinghouse. Data for the limiting case (1 ft^2 steam line split at 90% including power uncertainty) is shown in Table 3.6B-4A (Unit 1) and Table 3.6B-4B (Unit 2).

2. Main Feedwater Line Break

The worst case crack in the main feedwater line occurs downstream of the check valve. For Unit 1, the mass and energy release rates were developed using RELAP5/MOD 3 [36].

For Unit 2, the mass and energy release rates were developed based on methodology consistent with ANSI/ANS 58.2, Appendix E (see Table 3.6B-6).

Design Data

The main steam and feedwater isolation valve compartments are located on the two top floors of the Safeguard Building which is situated between the Containment Building and the Electrical and Control Building. Figure 3.6B-207 provides plan and elevation drawings of the Safeguard Building.

The main steam and feedwater piping in this area consists of straight piping runs approximately 40-feet long, extending from the containment penetrations, through the compartments to moment limiting restraints mounted in the Safeguards Building wall through which these lines exit to the outside. The compartments contain the main steam line isolation valves, main steam safety valves, main steam power-operated relief valves, main steam isolation bypass valves, steam supply valves to the auxiliary feedwater turbine pump, feedwater isolation valves, feedwater isolation bypass valves and auxiliary feedwater containment isolation valves. The piping arrangement is shown in Figure 3.6B-208.

Also in the subcompartments are various instruments and controls, cables and branch piping lines for the main steam drain and secondary sampling system.

Environmental Effects

The ability to safely shutdown the plant will not be affected by the release of steam and water in the main steam and feedwater compartments. Plant design allows for loss of one main steam and feedwater loop and associated equipment without jeopardizing the safe shutdown of the plant. The following discussion addresses design, operation and consequences of failure of equipment located inside the main steam and feedwater compartments (See Table 3.6B-5 for detail of equipment qualification for MSLB).

1. Main Steam Isolation Valves (MSIVs) and Bypass Valves

   The MSIVs and their electrical supplies are designed to operate at the maximum environmental conditions resulting from the postulated main steam or feedwater pipe crack. These valves except for the valve in the compartment where the crack is located, are required to close following a main steam line or feedwater line crack in order to limit the blowdown through the crack to one steam generator. The signal to close is derived from steam line pressure transducers located on the main steam line inside the compartment. These instruments are qualified for the above environment.

   The MSIV bypass valves are not required for safe-shutdown. These valves are locked closed manual valves and no failure could cause them to open.

2. Power-Operated Main Steam Relief Valves (PORV's)

   These valves are air-to-open, fail closed. The pressure transducers which provide the signal to open these valves must not fail in a mode which could cause the PORV on an intact steam generator to inadvertently open. The PORVs for the steam generator associated with the crack may fail in either the open or closed position without increasing the severity of the event beyond the present analysis or decreasing the ability to mitigate the event and safely shutdown the unit. The PORVs and pressure transducers have a

demonstrated operability beyond that required for their safety function (See Table 3.6B-5).

3. Auxiliary Feedwater Turbine Steam Supply Valves

The failure mode of the valve in the compartment with the crack due to the accident environment does not prevent safe shutdown of the unit. The valve in the compartments without the crack have a demonstrated operability beyond that required for its safety function.

4. Auxiliary Feedwater Containment Isolation Valves

These valves are motor operated, fail-as-is and are normally open and de-energized. Consequently, the accident environment cannot cause these valves to close. It is not postulated that there is a requirement to close the valve supplying auxiliary feedwater to the faulted steam generator because the auxiliary feedwater pump and flow control valves which are isolated from the break environment would be available to perform this function. The valves have a demonstrated operability beyond that required for their safety function.

5. Auxiliary Feedwater Flow Indicators

These instruments indicate flow in the control room for accident monitoring, but have no control function. Although other indication could perform this function (eg. steam generator level) these instruments must not fail in a manner that could lead operators to defeat or fail to accomplish a required safety function. They have a demonstrated operability beyond that required for the safe plant shutdown.

Flooding Evaluation

The worst flooding will occur as a result of a feedwater line break in any one of the feedwater piping compartments 100A through 100D. The water spills over an opening 7' high and 3' wide into compartments 100E through 100H. The maximum flood height in the break compartment is 9.56'. The maximum flood level in the spill over compartment is 8.06'. The structural integrity of the building will not be impaired as a result of these flood levels concurrent with the maximum pressure developed.

Chronology of Events

During the following events, the maximum temperature occurs inside the compartments during the main steam line crack and the maximum pressure occurs during the feedwater line crack.

1. Main Steam Line Crack

Maximum temperature inside the compartments containing essential equipment is reached within 155 seconds into the event. The time sequence of events for the MSLB with superheat is described in Table 3.6B-4A (Unit 1) and Table 3.6B-4B (Unit 2).

2. Feedwater Line Crack

   Low water level in the affected steam generator causes reactor trip and auxiliary feedwater pump start. Feedwater pumps will continue to deliver feedwater to the crack for approximately two minutes when low water level in the condenser hot wells causes the condensate pumps to trip. Low suction pressure will trip the main feed pumps and consequently pumped flow to the crack will be terminated. At that time steam from all steam generators and from the auxiliary feedwater being added to the faulted loop steam generator will be emitted from the crack. Main steam isolation occurs on low steam pressure or on operator action. Operator action terminates auxiliary feedwater addition to the faulted steam generator. The steam leakage phase of the feedwater line crack accident will increase the environmental temperature to a level which is less than that reached during the main steam line crack accident. The maximum pressure occurs during the water leakage phase of the feedwater line crack accident.

3.6B.1.3 Safety Evaluation

The results of the pipe whip and jet impingement analyses, which verify that the consequences of failures of high energy lines do not affect the ability to safely shut down the plant, are provided in the corresponding FSAR sections for the systems listed in Section 3.6B.1.2.1. Incorporated in the analyses are the results of the failure modes and effects analysis, which are described in the safety evaluation section for each system.

Specific design features utilized for the protection of essential systems are identified in Section 3.6B.2.5. Protection for essential structures and components to ensure that their minimum required function can be accomplished following a postulated pipe rupture is provided by one or more of the following methods:

1. Separation and remote location of piping from essential structures and components. Separation is achieved by physical plant arrangements that provided sufficient distances between essential structures, components and piping such that the effects of postulated pipe ruptures cannot impair the structural integrity or operability of the essential structure or component.

2. Barriers, shields and enclosures. Barriers, shields or enclosures are provided for piping or, alternatively, protection of components within structures or compartments is provided. These elements are designed to withstand and contain the effects of the postulated pipe ruptures.

3. Pipe whip restraints. Where physical separation, shields or enclosures are not feasible, protection of essential systems and components is attained by the use of pipe whip restraints and barriers. Alternatively, essential systems and components may be designed to withstand the effects of the postulated pipe rupture.

The design criteria for the restraints are given in Section 3.6B.2.3.

No protection or restraints are provided when it is determined that the piping failure will not cause unacceptable damage to essential systems.

Each longitudinal or circumferential break in high-energy fluid system piping is considered separately as a single postulated initial event occurring during normal plant conditions.

In analyzing the effects of postulated piping failures, the following assumptions are considered with regard to the operability of systems and components:

1. 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.

2. A single active failure is assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted in (3) 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.

3. 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, (i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure), single failures in the other train or trains of that system are not assumed provided the following conditions are met:

   a. The system is designed to seismic Category I Standards.

   b. The system is powered from both offsite and onsite sources.

   c. The system is constructed, operated, and inspected to quality assurance, testing, and in-service inspection standards appropriate for nuclear safety systems.

4. All available systems, including those actuated by operator actions, are employed to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account is taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, and of the assumed single active failure and its direct consequences.

5. Operator action credited to mitigate the consequences of the postulated pipe failure is analyzed for each specific event. In general, no operator action is considered to mitigate the consequences of a postulated pipe failure for 30 minutes.

   However, if detection and termination of blowdown can be accomplished in the control room, then an operator action of 10 minutes is credited.

6. A whipping pipe has insufficient energy to cause loss of structural (pressure retaining) integrity in target piping systems of equal or larger nominal pipe size and wall thickness and has sufficient energy to cause through-wall leakage cracks in target piping systems of equal or larger nominal pipe size but with less wall thickness. Any whipping pipe has sufficient energy to cause loss of function of active components such as valves, pumps and motors.

7. Unless it is determined by appropriate calculations or tests, a jet emanating from either a circumferential or a longitudinal break of a high energy system has sufficient energy to

cause loss of structural integrity in piping systems of smaller diameter, through wall leakage cracks in larger pipes with lesser wall thickness, and loss of function in active components such as valves, pumps and motors. Check valves that have equal or larger nominal pipe size and wall thickness than the pipe containing the break shall remain operable.

8. Credit is taken, where possible for automatic valves which fail in the safe position.

9. Essential components which are qualified or can be demonstrated to operate in an adverse environment due to a pipe failure are not protected.

The protective measures utilized, as described in this section, will not prevent the access required to conduct the inservice examinations specified in the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, “Rules for Inspection and Testing of Components in Light-Water Cooled Plants.”

All essential systems, components, piping, component enclosures, protective structural barriers and structures supporting pipe whip restraints are designed to seismic Category I requirements, with the following exceptions. The moderate energy spray shields have structural supports that are designed to Seismic Category I, however, the shielding is constructed of a plastic type material and as such has no seismic pedigree. Architectural features (e.g. doors), tornado dampers, HELB mitigation dampers, ductwork, penetration seals, and backwater valves of the floor drain system are non-seismic category I. (Although some backwater valves are credited to operate for the mitigation of postulated piping failures, a seismic event is not assumed in the analysis of these failures.) Primary plant ventilation (Section 9.4.3) supply (Figure 9.4-9, Sh. 3) and associated chilled water (Section 9.4E) are non-seismic category I. Also see Section 7.6.12 for detection and mitigation equipment exceptions.

3.6B.2 DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING (INSIDE AND OUTSIDE CONTAINMENT)

The design bases for locating postulated breaks and cracks in piping inside and outside of the Containment, the procedure used to define the jet thrust reaction at the break or crack location and the jet impingement loading on adjacent structures, equipment, systems, and components are described as follows.

3.6B.2.1 Criteria Used To Define Break And Crack Location And Configuration

This section provides criteria for the location and configuration of postulated pipe breaks in high energy piping systems.

3.6B.2.1.1 Reactor Coolant System (RCS) Main Loop Piping

The generic Leak-Before-Break technology described in NUREG-1061 Volume 3 [18] has been applied to the CPNPP Units 1 and 2 RCS main loop piping.

This application of Leak-Before-Break methodology, allowed under the modified GDC-4, is discussed in reference [19].

The analyses show that the probability of RCS main loop piping breaks is extremely low, thus the dynamic effects of these breaks are not considered in the design basis of CPNPP. CPNPP leak-before-break was developed from detailed analysis of the following factors [19]:

1. The loads, material properties, transients, and geometry for CPNPP Units 1 and 2 RCS primary loop are enveloped by the Westinghouse generic leak-before-break analyses.

2. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.

3. Water hammer is precluded in the RCS primary loop piping because of system design, testing, and operational considerations.

4. The effects of low and high cycle fatigue on the integrity of the primary piping are negligible.

5. Ample margin exists between the leak rate of the reference flaw and the criteria of Reg. Guide 1.45. (See FSAR Sections 1A(B) and 5.2.5).

6. Ample margin exists between the reference flaw chosen for leak detectability and the “critical” flaw.

7. Ample margin exists in the material properties used to demonstrate end-of-life (relative to aging) stability of the reference flaw.

The reference flaw will be stable throughout reactor life because of the ample margins in e, f, and g, above and will leak at a detectable rate which will assure a safe plant shutdown.

Although the dynamic effects of the RCS main loop piping breaks are not considered in the design basis, GDC 4 (FSAR Section 3.1.1.4) requires that primary coolant piping breaks be included in the design of the reactor containment, the ECCS, and in environmental qualification. In order to comply with this requirement, main loop breaks are postulated as discussed in Sections 6.2, 6.3 and 3.11 respectively.

The new Leak Before Break (LBB) leak rate calculation methodology for Alloy 82/182 welds with Structural Weld Overlay (SWOL), has been approved by the Nuclear Regulatory Commission for use in Pressurizer Surge Line (PSL) LBB analysis at the Waterford nuclear power plant. This methodology is applicable for CPNPP and has been utilized for the LBB analysis for Alloy 82/182 welds with SWOL for the pressurize surge line nozzle.

3.6B.2.1.2 High Energy Piping Other Than The RCS Main Loop

High energy fluid systems are those systems that during normal plant conditions are either in operation or maintained pressurized under conditions where either or both of the following are met:

Maximum operating temperature exceeds 200°F

Maximum operating pressure exceeds 275 psig.

It is noted that systems which operate within the pressure and temperature conditions specified for high energy fluid systems for less than two percent (2%) of the time are considered moderate energy. Breaks for these systems are postulated as described in Section 3.6B.2.1.4. A list of high energy lines is given in Table 3.6B-1.

Design basis break locations and types are postulated in accordance with NRC Regulatory Guide 1.46 and Branch Technical Position MEB 3-1. Where required, postulated pipe breaks are selected as described below and analyzed to demonstrate the capability for a safe shutdown of the plant.

Application of leak before break methodology justifies the elimination of postulated pipe ruptures in 10 inch and larger RCS branch lines. This analysis demonstrates the piping integrity and serves as the basis for excluding from consideration the dynamic effects associated with postulated pipe ruptures.

1. ASME Section III, Code Class 1 Piping (Excluding Primary Reactor Coolant Loop Piping)

   For high energy piping in the RC, CVC, RHR, and SI systems, the 1977 Edition up to and including Summer 1979 Addenda of ASME B&PV Code Section III, Code Class 1 will be used to postulate pipe breaks. For all other high energy piping the 1974 Edition up to and including Winter 1975 Addenda is applicable. The pipe breaks are postulated to occur at terminal ends and at all intermediate locations in the piping system where none of the following criteria is met:

   a. The primary plus secondary stress intensity range (equation (10) of ASME III Subparagraph NB-3653) derived on an elastically calculated basis under loadings associated with the OBE and normal and upset plant conditions does not exceed 2.4 S_m or

   b. The primary plus secondary stress intensity range derived on an elastically calculated basis under loadings associated with the OBE and normal and upset plant conditions exceeds 2.4 S_m but is less than 3.0 S_m, and the cumulative usage factor is less than 0.1 or

   c. The primary plus secondary stress intensity range derived on an elastically calculated basis under loadings associated with the OBE and normal and upset plant conditions exceeds 3.0 S_m but the stress ranges computed by Equations (12) and (13) of Subparagraph NB-3653 of ASME III are less than 2.4 S_m and the cumulative usage factor is less than 0.1.

2. ASME Section III, Code Class 2 and 3 Piping

   ASME B&PV Code, Section III, Class 2 and Class 3 piping breaks are postulated to occur at terminal ends and intermediate locations in each piping run or branch run. Breaks at intermediate locations are selected by either of the following criteria:

   a. At each location where the stresses associated with normal and upset plant conditions and an OBE event, calculated by the sum of Equations (9) and (10) of paragraph NC-3652 of the ASME B&PV Code, Section III, exceed 0.8(1.2 S_h+S_A).

b. Certain pipe runs contain Class 2 piping extensions to Class 1 lines up to the first anchor point beyond the Class 1/Class 2 boundary. For these pipe runs a break is postulated at the Class 2 terminal end. Breaks are not postulated in the Class 2 portion if the stresses in the Class 1 portion, calculated using Equation (10) of NB-3653, are above 2.4 S_m and the sum of the stresses in the Class 2 portion calculated using Equations (9) and (10) of NC- 3652, are below 0.8 (1.2 S_h + S_A).

3. Non-Nuclear Piping

Breaks are postulated to occur in non-nuclear piping systems at the locations as specified for ASME Section III, Class 2 and 3 piping in accordance with Section B criteria above when a seismic stress analysis is performed.

Breaks in non-nuclear piping systems are postulated at terminal ends and at intermediate points, such as fittings (elbows, tees, reducers, etc.), welded attachments and valves in each run or branch run where a stress analysis is not performed.

4. Fluid System Piping Between Containment Isolation Valves

There are no ASME Section III, Class I piping penetrations. The piping between containment Isolation valves is ASME B&PV Code, Section III, Class 2 piping.

Breaks are not postulated in the break exclusion areas of high energy fluid system piping from the inside containment process pipe-to-penetration weld to and including the outside containment isolation valve moment restraint forging as shown on Figures 3.6B-15, 16, 17, 18, 23, 25, 26, 27, 28, 29, 30, 42, 82, 85, 87 and 88. Moment restraints/forgings are provided to protect the containment isolation valves, process pipes and penetrations within the break exclusion areas. A typical moment restraint/forging, including the boundaries of the break exclusion area, is shown in Figure 3.6B-5.

The following stress limits of Branch Technical Positions APCSB 3-1 [3] and MEB 3-1 [2] for ASME B&PV Code, Section III, Class 2 piping are met within the break exclusion areas:

a. The maximum stress ranges as calculated by the sum of Equations (9) and (10), paragraph NC-3652 of the ASME B&PV Code, Section III, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, and thermal expansion), and an OBE event do not exceed 0.8 ($1.2S_h$+ S_A).

b. The maximum stress, as calculated by Equation (9), paragraph NC-3652 of the ASME B&PV Code, Section III, under the loadings resulting from a postulated piping failure beyond the break exclusion area, does not exceed 1.8 S_h. The moment restraint forging is permitted higher stresses outboard of the restrained locations provided a plastic hinge is not formed.

Welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses or tests are performed to demonstrate compliance with the limits defined in 4.a and 4.b above. In

addition, the number of circumferential and longitudinal piping welds and branch connections is minimized. Guard pipes are not used.

The length of these portions of piping is reduced to the minimum length practical. The design of pipe anchors or restraints (e.g., connections to Containment penetrations and pipe whip restraints) does not require welding directly to the outer surface of the piping (e.g., flued-integrally forged pipe fittings are used). If welded attachments are required, such welds are 100-percent volumetrically examinable in service, and a detailed stress analysis shall be performed to demonstrate compliance with the limits defined in 4.a and 4.b above.

3.6B.2.1.3 Type of Breaks Postulated in Fluid System Piping Other Than the RCS Main Loop

Circumferential and longitudinal breaks are postulated in piping systems at locations previously discussed. The following are definitions and criteria used to determine the type of break to be considered:

1. Circumferential Pipe Breaks

Circumferential breaks are defined as a full cross section area break, with at least one diameter lateral displacement of the ruptured pipe. Circumferential breaks are postulated in piping with nominal size greater than 1", unless the break separation is physically limited by piping restraints, by structural members, or by piping stiffness as demonstrated by inelastic limit analysis. Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces at the break location are based on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. These forces separate the piping axially causing the ruptured pipes to move in the direction of the thrust force. The ruptured pipe is assumed to whip in the plane defined by the piping geometry and configuration causing the pipe to move in the direction of the jet reaction.

Only circumferential breaks are postulated at terminal points, whether or not the piping is seamless. For a terminal point at an equipment or vessel nozzle, a circumferential break is postulated at the pipe-to-nozzle weld. If the terminal point is at a normally closed valve pressurized on one side, the circumferential break is at the pipe-to-valve weld on the high energy side. For a terminal point where a fitting is installed, a circumferential pipe break is postulated only at terminal-to-fitting weld. The pipe-to-fitting weld (the second weld on fitting) may be used as an intermediate pipe break if required by stress level.

2. Longitudinal Pipe Breaks

Longitudinal breaks are assumed to result in an axial split of the pipe without severance. Splits are oriented (but not concurrently) at two diametrically opposed points on the piping circumference so that a jet reaction causing out-of-plane bending of the piping configuration results. Alternatively, a single split may be assumed at the section of highest tensile stress as determined by an elastic stress analysis.

Longitudinal breaks in high energy fluid system piping are postulated in nominal pipe sizes 4-inches and larger. Longitudinal breaks are not postulated at terminal ends. The provision for longitudinal breaks at terminal points applies not only to anchored points, but

also to terminal points at branch connections and at the point immediately outboard of the break exclusion area. Longitudinal breaks are also not postulated at intermediate locations where the criterion for the minimum number of breaks as described in Sections 3.6B.2.1.2.1 and 3.6B.2.1.2.1 is satisfied.

If a detailed stress analysis or test is performed the results may be used to predict the most probable rupture location(s) for the longitudinal breaks. Without the benefit of the stress analysis report, longitudinal breaks will be postulated to occur on each side of a fitting.

Only one type of break need be postulated at a location based on the stresses as identified by a stress analysis. If the primary plus secondary stress in the axial direction is found to be at least 1.5 times that in the circumferential direction for the most severe normal plus upset loading combination, then only a circumferential break need be postulated. Conversely, if the primary plus secondary stress in the circumferential direction is found to be at least 1.5 times that in the axial direction for the most severe loading combination noted above, then only a longitudinal break need be postulated.

The dynamic force of the fluid jet discharge is based on a circular 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.

Piping movement is assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis.

3.6B.2.1.4 Moderate Energy Piping

Moderate-energy fluid systems are the fluid systems that during normal plant conditions are either in normal operation or maintained pressurized under conditions where both of the following are met:

Maximum operating temperature is 200°F or less

Maximum operating pressure is 275 psig or less

Through-wall leakage cracks are postulated for moderate energy piping systems inside and outside the containment in accordance with the following criteria:

1. Through wall leakage cracks are postulated either at any location in moderate energy piping larger than 1 inch, where essential equipment may be affected by the environmental effects from fluid spray and flooding or only where the stresses associated with normal and upset plant conditions and an OBE event as calculated by the sum of equations (9) and (10) of paragraph NC-3652 of the ASME B&PV Code, Section III exceed .4 ($1.2\ S_h + S_a$).

2. Fluid flow from a crack is based on a circular opening of area equal to that of a rectangle which is one-half pipe internal diameter in length and one-half pipe wall thickness in width.

3. Cracks instead of breaks are postulated in the piping of those fluid systems that qualify as high energy fluid systems for only short operational periods, but qualify as moderate energy fluid systems for the major operational period.

   An operational period is considered short if the fraction of time that the system operates within the pressure-temperature conditions specified for high energy fluid systems is less than two percent (2%) of the time that the system operates as a moderate energy fluid system.

   Where through wall leakage cracks are postulated in one of two or more redundant trains of a dual-purpose moderate-energy essential system (i.e., one train required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure), single failures of components in the other train or trains of that system are not assumed provided the system is designed to seismic Category I standards, is powered from both offsite and onsite power sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems.

3.6B.2.1.5 Definition of Plant Operating Conditions

Plant operating conditions which are used in calculating the stresses for ASME Section III piping to establish pipe break locations are as follows:

1. Normal Plant Conditions

   Plant operating conditions during reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

2. Upset Plant Conditions

   Plant operating conditions during system transients that may occur with moderate frequency during plant service life and are anticipated operational occurrences, but not during system testing.

3.6B.2.2 Analytical Methods to Define Forcing Functions and Response Models

The analytical methods used to define the forcing functions and the response models are described in the following sections.

3.6B.2.2.1 Reactor Coolant System Main Loop Piping

In order to determine the thrust and reactive force loads to be applied to the reactor coolant loops during the postulated loss of coolant accident (LOCA), it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the ruptured and intact reactor coolant loops as a result of a postulated LOCA. These forces result from the transient flow and pressure histories in the RCS. The calculation is performed in two steps. The

first step is to calculate the transient pressure, mass flow rates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis along with input of areas and direction coordinates and calculates the time history of forces at appropriate locations in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the entire RCS. Key parameters calculated by the hydraulic model are pressure, mass flow rate, and fluid density. These are input to the thrust calculation, with appropriate plant layout information, to determine the time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated LOCA, the pressure and momentum flux terms are dominant. The inertia and gravitational terms are taken into account in evaluation of the local fluid conditions in the hydraulic mode.

The blowdown hydraulic analysis is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid throughout the reactor system. The SATAN-IV code [11] was developed with a capability to provide this information.

1. Unit 1

   The MULTIFLEX 3.0 [42] computer code was developed with this capability, which is an enhancement and extension of MULTIFLEX 1.0 [37] NRC reviewed and approved computer code developed for the space-time dependent analysis of nuclear power plants. The MULTIFLEX 3.0 features which differ from MULTIFLEX 1.0 are primarily related to vessel forces. The loop forcing functions do not differ significantly from those generated using the NRC approved MULTIFLEX 1.0 model. MULTIFLEX 3.0 has been accepted by the NRC for several other applications [38, 39, 40 and 41] and also has been extensively used for the LOCA analyses of reactor internals components of numerous other 2, 3, and 4 loop nuclear power plants.

   MULTIFLEX is a digital computer program for calculation of pressure, velocity and force transients in reactor primary coolant systems during the subcooled, transition and the early saturation portion of blowdown caused by a LOCA. During this phase of the accident, large amplitude rarefaction waves are propagated through the system with the velocity of sound causing large differences in local pressures. As local pressures drop below saturation, causing the formation of steam, the amplitudes and velocities of these waves drastically decrease. Therefore, the largest forces across the loop piping due to wave propagation occur during the subcooled portions of the blowdown transient. MULTIFLEX includes mechanical structure models and their interaction with the thermal-hydraulic system, although these features are only involved in the vessel and steam generator modeling.

2. Unit 2

The SATAN-IV code performs a comprehensive space time-dependent analysis of a LOCA and is designed to treat all phases of the blowdown. The stages are:

1. A subcooled stage where the rapidly changing pressure gradients in the subcooled fluid exerts an influence upon the RCS internals and support structures

2. A two phase depressurization stage

3. The saturated stage.

The code employs a one dimensional analysis in which the entire RCS is divided into control volumes. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. Pump characteristics, pump coastdown and cavitation, core and steam generator heat transfer, including the W-3 departure from nucleate boiling (DNB) correlation, and reactor kinetics are incorporated in the code.

The THRUST (Unit 1) or STHRUST (Unit 2) computer programs [12] were developed to compute the transient (blowdown) hydraulic loads resulting from a LOCA.

The blowdown hydraulic loads on primary loop components are computed from the equation

$$F = 144A\left[\frac{(P - 14.7) + (m^2)}{144\rho\ g\ A^2 m}\right]$$

which includes both the static and dynamic effects. The symbols and units are:

F = force (lb_f)

A = aperture area (ft^2)

P = system pressure (psig)

m = mass flow rate (lbm/sec)

ρ = density (lb_m/ft^3)

g = gravitational constant (32.174 ft-$lb_m/lb_f sec^{2)}$

A_m = mass flow area (ft^2)

In the model to compute forcing functions the reactor coolant loop system is represented by a similar model as employed in the blowdown analysis. The entire loop layout is described in a global coordinate system. Each node is fully described by blowdown hydraulic information and the orientation of the streamlines of the force nodes in the system. These include flow area and projection coefficients along the three axes of the global coordinate system. Each node is modeled as a separate control volume, with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y, and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, total y force, and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis described in Section 3.9N.14.

3.6B.2.2.2 High-Energy Piping Other Than RCS Main Loop

The time dependent function representing the thrust forces caused by the jet flow from a postulated pipe break or crack includes the combined effects of the thrust impulse 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 buildup of the discharge flow rate which may reach steady state if there is a fluid energy reservoir having sufficient capacity to develop a steady jet for a significant interval. Alternatively, in a simplified method, the jet thrust force is represented by a steady state function. This function, representing the force, would have a magnitude not less than:

$$F_{ss} = C_t P A$$

where:

F_{ss} = steady state thrust force (lb_f)

P = system pressure prior to pipe break (lb_f/in^2)

A = pipe break area (in^2)

C_t = steady state thrust coefficient

The steady state thrust coefficient C_t is dependent on the fluid state and the frictional loss terms. The value of steady state thrust coefficient and the time to reach steady state flow conditions are calculated from references [15], [16] and [22].

The rigorous time dependent blowdown forces resulting from a postulated pipe rupture are determined using the RELAP-5 computer code [6 or 34] except where STEHAM [28] is used as part of the alternative dynamic methodology for postulated main steam line breaks described later in this section. RELAP-5 is a thermal/hydraulic program commonly used in the nuclear industry to evaluate the behavior of water cooled reactor systems during postulated accidents such as pipe ruptures. The program is acceptable (see Reference [7]) as a means of determining the hydraulic forcing function at the pipe break. CALPLOTF-III [20] or REPIPE [29] or other post processor programs to the RELAP program are used to develop the break force time-history plots.

The RELAP-5 program solves the transient energy, momentum, and fluid state equations to determine the system flow, pressure, and thermodynamic conditions. The break force is computed using the one-dimensional momentum equation and the appropriate density, internal energy, and pressure values. The rupture load is the summation of the pressure, momentum, and change in momentum terms at the time interval in question.

RELAP-5 has the capability of solving the fluid state equation for subcooled water, flashing water, two-phase steam/water mixtures, and superheated steam. The ASME steam tables [9] have been incorporated into RELAP-5 so that the fluid state properties are accurately determined. RELAP-5 has a provision for modeling components such as valves, check valves, pumps, heat exchangers, and reactors along with the associated piping.

Transients can be initiated by the control card added to the program which is used to describe leaks (pipe breaks), valves opening and closing, check valve pressure drop-flow-characteristics, pump coastdowns, and so forth.

The flow system is described as a series of volumes connected by flow paths or junctions. RELAP-5 requires input data that completely describe the thermodynamic conditions and physical data of the system being analyzed. Pressure, temperature, and flow conditions along with physical dimensions, flow areas, friction characteristics must all be specified as initial conditions. The break area can be reduced by an analytically or experimentally derived discharge coefficient. However, in lieu of such data it is conservatively assumed that the discharge coefficient is 1.0 for both longitudinal and circumferential breaks. In a similar manner, the break area is assumed to open within one millisecond (0.001 second).

Rigorous time dependent blowdown forces and pipe segment forces resulting from the postulated breaks were also determined by using the STEHAM computer program. The analysis is based on the method of characteristics with finite difference approximations both in space and in time. STEHAM calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changing of piping components, such as the stop valve and the safety/relief valve which are used to represent pipe break. Flow characteristics of piping components are mathematically formulated as boundary condition and frictional effects are appropriately taken into consideration in this computer program. Output from STEHAM consists of time-histories of flow pressures, flow densities, flow velocities, inertia, and momentum functions. CHPLOT [32] program is used to plot the force time-history data developed by STEHAM. These data are used as input to the LIMITA3 structural computer program.

The piping dynamic responses resulting from a postulated pipe rupture are determined using the PIPERUP [13], SHPLAST 2267 [24], ABAQUS [21], ANSYS [30 or 35] or LIMITA3 [31] computer codes. The programs are adaptations of the finite element method to the requirements of pipe rupture analyses. They perform a dynamic, nonlinear, elastic-plastic analysis of piping systems subjected to time-history forcing functions. These forces result from fluid jet thrust at the location of a postulated longitudinal or circumferential rupture of high energy piping and ensuing acoustic disturbances within the piping.

The piping is mathematically modeled in the PIPERUP, SHPLAST 2267, ABAQUS, ANSYS or LIMITA3 program as an assembly of weightless structural members connecting discrete nodal points. A typical pipe whip mathematical model is shown in Figure 3.6B-6. Weight of the system, including distributed weight of the piping and concentrated weights (e.g., valves), is lumped at selected mass points (lumped parameter analysis model). Nodal points are placed in such a manner as to isolate particular types of piping elements such as straight runs of pipe, valves, elbows, etc. for which force-deformation characteristics may be determined. Nodal points are also placed at all discontinuities such as piping restraints, branch lines, and changes in cross-section. Piping restraints are modeled with an initial gap and in PIPERUP with a bilinear stiffness curve, or, in SHPLAST 2267, ABAQUS, ANSYS and LIMITA3 with multilinear stiffness curve. A typical piping stress-strain curve is shown in Figure 3.6B-7. The first stiffness represents linear elastic behavior and the second stiffness models linear strain hardening behavior. All five programs utilize a direct step-by-step integration method to determine the time history response of the ruptured piping system. A typical restraint impact curve is shown in Figure 3.6B-8. An incremental procedure is used to account for the nonlinear deformation and elastic-plastic effect of the pipe and restraints.

3.6B.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6B.2.3.1 Reactor Coolant System Main Loop

The leak-before-break technology has been applied to CPNPP Units 1 and 2 to exclude from the design basis the dynamic effects of postulated ruptures in the RCS main loop piping. This applies, in particular, to jet impingement loads on components and supports.

Jet loads from large branch nozzle breaks are addressed in Section 3.6B.2.3.2.

3.6B.2.3.2 High-Energy Piping Other than the RCS Main Loop

Pipe beaks are postulated in high-energy piping in accordance with the criteria in Section 3.6B.2.1.2. The analyses for determining the dynamic effects of pipe break are as follows:

1. Jet Impingement

   A circumferential or longitudinal break in a high energy line results in a jet of fluid emanating from the break point. For subcooled high energy lines where the fluid temperature is less than its saturation temperature at the surrounding environmental pressure, the discharge jet is characterized by a nearly constant diameter jet approximately equal to the break diameter. Since the fluid temperature is below saturation it will not flash but instead will form an incompressible fluid jet.

   In general, most of the high energy line breaks result in a two-phase choked (critical) flow at the break exit plane. Fluid pressure at the exit plane is in general at some pressure greater than ambient. As the fluid leaves the pipe break area, it expands as the jet pressure decreases from the higher exit (break) plane pressure to the atmospheric pressure surrounding the jet.

   A jet discharging from a saturated steam line will accelerate and expand due to the pressure differential, and it will partially condense to a low-moisture wet steam with the liquid phase in the form of dispersed, entrained water droplets. A jet discharging from a subcooled or saturated hot water line (greater than 212°F) will flash to a low quality wet steam. The flashing will cause the jet diameter to expand very rapidly.

   ANSI/ANS-58.2-1988 [22] provides an acceptable basis (including conservative analytical models) for the evaluation of jet impingement loads. The CPNPP jet impingement methodologies and models are consistent with ANSI/ANS 58.2-1988, as briefly described in the following sections:

   a. Jet Category and Geometry

      The area of the break is assumed to be equal to the flow area of the ruptured pipe. All the high energy line break jets can be summarized into the following three categories:

      1. Category I Jets - Non-Expanding Jets

For the liquid jets whose temperature is below the saturation temperature at ambient pressure, the initial free expansion does not occur. Incompressible liquid jets are assumed to travel with no increase in jet area. However, for target identification a conservative zone of influence of two diameters is utilized. The pressure is assumed to be uniform throughout the jet area.

2. Category II Jets - Steam and Flashing Water Jets which meet the Criteria of NUREG/CR-2913[25]:

The high energy two-phase jet is a complicated multidimensional flow phenomena. The high pressure and high temperature fluid that exits the break expands with supersonic velocities downstream of the break. Upon encountering a target (or obstacle) a shock wave forms in the flow field, and it is the thermodynamic properties downstream of this shock that determine the pressure field and load on the target. A multidimensional analysis, such as demonstrated in NUREG/CR-2913 [25], more realistically evaluates the thermodynamic properties of these jets. The jet cone shape for these Category II jets is a function of the break initial conditions (i.e., temperature and pressure). These jet shapes are determined by using the NUREG/CR-2913 contour curves and are used to establish the jet zone of influence for target identification. The NUREG/CR-2913 model provides a method for calculating target loads for initial pipe rupture fluid conditions of pressure between 60 and 170 bars (870 psia -2466 psia) and with subcooling of 0°C (0°F) to 70°C (126°F). Note that, for Category II jets, the jet shape from the break exit to the jet core region is not defined in NUREG/CR-2913 [25] and should be computed from ANSI/ANS-58.2-1988 [22] (see next section).

3. Category III Jets - All other steam and flashing water jets:

Category III jets are assumed to expand as a three region cone defined in Figure 3.6B-96A for circumferential breaks and Figure 3.6B-96B for longitudinal breaks [22].

(a) Circumferential Break with Full Separation:

Jet Region 1 ($L < L_c$). Region 1 includes a cone-shaped region containing the jet core and the remainder of the jet. This geometry is shown in Figure 3.6B-96C.

The jet core length is related to the jet subcooling at the jet break plane and has been correlated using the following expression

$$L_c/D_e = 0.26 \quad (\Delta T_{sub})^{1/2} + 0.5 \qquad (1)$$

where:

L_c = core length

D_e = pipe inside diameter

T_{sub} = jet subcooling at stagnation conditions in °F at the break plane

Figure 3.6B-96D can be used to relate jet stagnation subcooling at the break plane to stagnation conditions in the vessel supplying the jet flow, accounting for irreversible losses in the blowdown line.

In Region 1, for $0 < L < L_c$, the jet core diameter, D_c, is given by

$$D_c/D_e = (C^*_{Te})^{1/2} (1-L/L_c) \quad (2)$$

where:

C^*_{Te} = 2.0 for $\Delta T_{sub} > 0$

1.26 for $\Delta T_{sub} = 0$

L = distance from break plane to target

Jet Region 2 ($L_c < L < L_a$). In Region 2, the jet expands to its asymptotic area which can be calculated as:

$$A_a/A_e = G_e^2/(g_c \rho P_{ma} C_T P_o) \quad (3)$$

where:

A_a = jet area at the asymptotic plane

A_e = break plane area

C_T = steady-state thrust coefficient (as defined in ANSI/ANS-58.2 [22])

G_e = mass flow rate per unit area from the break plane

g_c = gravitational constant

P_o = initial total (stagnation) pressure in the vessel

ρ_{ma} = asymptotic plane density. If two-phase, density will be given by

ρ_{ma} = $1/\lfloor x/\rho_g + (1-x)/\rho_f \rfloor$

x = mixture vapor mass fraction i.e. quality, at the asymptotic plane pressure, Pa, and stagnation enthalpy

ρ_f = saturated liquid density at the asymptotic plane pressure

ρ_g = saturated vapor density at the asymptotic plane pressure

[Note: Figure 3.6B-96E may be used in place of Equation (3)]

The jet pressure at the asymptotic plane, Pa can be expressed as the following expression

$$(Pa)/P_{amb} = 1 - 0.5\ (1 - 2P_{amb}/P_o) f(h_o) \qquad (4)$$

where:

P_{amb} = ambient pressure

Pa = asymptotic plane static pressure

$$f(h_o) = \begin{cases} [0.1 + (h_o - h_f)/h_{fg}]^{1/2} & \text{for } (h_o - h_f)/h_{fg} > -0.1 \\ 0 & \text{for } (h_o - h_f)/h_{fg} < -0.1 \end{cases}$$

h_o = stagnation enthalpy in the vessel(a)

h_f, h_{fg} = saturated liquid enthalpy and heat of vaporization in the vessel

---

a) h_o in the vessel and at the break plane are assumed to be equal.

The distance from the break plane to the asymptotic plane is defined by:

$$L_a/D_e = 1/2\lfloor (A_a/A_e)^{1/2} - 1 \rfloor \qquad (5)$$

The jet area at any location from the break plane to the asymptotic plane (Regions 1 and 2) may be calculated from the following relationship:

$$A_j/A_{je} = \lfloor 1 + L/L_a\ (A_a/A_{je} - 1) \rfloor, \qquad (6)$$

where:

A_j = jet area

A_{je} = jet area at break plane

Jet Region 3 ($L \geq L_a$). In Region 3, the jet area is given by

$$A_j A_a = (1 + (2(L - L_a)/D_a)(\tan 10°))^2 \quad (7)$$

where:

D_a = jet diameter at the asymptotic plane

(b) Longitudinal Break

The jet shape for longitudinal breaks, as shown in Figure 3.6B-96B shall be assumed to be the same as the circumferential break defined in 3.6B.2.3.2.1.a.3.a. A jet diameter for a circular break of the same area may be used and the jet direction taken to be perpendicular to the axis of the pipe.

2. Effective Target Distance

a. Category II Jets

For steam and flashing water jets within the limits of NUREG/CR-2913 [i.e., stagnation pressure from 60 bars (870 psia) to 170 bars (2466 psia) and subcooling of 0°C (0°F) to 70°C (126°F)] the effective target distance is taken as ten (10) times the inside diameter of the ruptured pipes [25].

b. Category I & III Jets

For all other high energy line break jets, jets are assumed to travel until impact with a target or a barrier.

3. Jet Force and Pressure

a. Category I Jets

The generalized momentum equation that describes the jet blowdown force, F_b, is;

$$F_b = G^2{}_e A_e/\rho_e g_c + A_e(P_e - P_{amb}) \quad (8)$$

where:

P_e = fluid pressure at the break flow area

ρ_e = fluid density at the break flow area

And for calculating target loads a conservative quasi-steady-state jet force, F_j, is used:

$$F_j = A_e (C_T P_o - P_{amb}) - C_T P_o A_e \quad (9)$$

However, the above equation for F_j is modified as follows:

1. For jets where $\Delta T_{sub} > 0$, C_T is multiplied by $(2.0/C_{Te})$ in region 1, where $C_{Te} = C_T$ based on FL/D=O and the break plane stagnation condition.

2. Unless otherwise justified, F_j is not to be less than the initial jet force based on equation (8).

For liquid jets whose temperature is below the saturation temperature at ambient pressure, a uniform pressure over the jet cross-section can be assumed, which is consistent with the jet area and the total jet force as defined by equation (8) or (9).

b. Category II Jets

In NUREG/CR-2913 [25], an advanced two-dimensional computer code (CSQ) was used to develop a two-phase jet load data base. The results showed that the jet force/pressure can be expressed as functions of four independent variables:

$F_j = F_j(r/D_e, L/D_e, P_o, T_o, \text{or } X_o)$ and

$P_j = P_j(r/D_e, L/D_e, P_o, T_o, \text{or } X_o)$

where:

r = radial distance from the jet center line

T_o = stagnation temperature in the vessel

X_o = stagnation quality in the vessel

P_j = radial pressure distribution on and an area perpendicular to the jet centerline

The values of F_j and P_j corresponding to some specific conditions of r/D_e, L/D_e, P_o, T_o or X_o can be found in NUREG/CR-2913 [25].

c. Category III Jets

The total jet force defined by Equations (8) and (9) can also be applied to Category III Jets. However, it should be recognized that typical Category III Jets (except low pressure jets which are discussed later) are under expanded, which means that the jet pressure at the break plane is greater than the ambient pressure. The jet expands rapidly as it leaves the break and comes to equilibrium with the surrounding pressure. Therefore, the jet impingement pressure decreases in a radial direction, away from the jet centerline. According to ANSI/ ANS-58.2-1988, the radial pressure distributions in the different jet regions are given below.

1. For Region 1, defined as $0 < L < L_c$, the jet pressure in the core and outside the core is given by

   Jet Core; for $0 < r < D_c/2$,

$$P_j = P_{oe} = \frac{F_j}{A_{je}} \quad = \text{stagnation pressure at the break} \tag{11}$$

   Outside Core; for $D_c/2 < r < D_j/2$,

$$\frac{P_j}{P_{oc}} = \left(\frac{D_j - 2r}{D_j - D_c}\right)\left[1 - \frac{2[D_j^2 + D_j D_c + D_c^2 - 3D_c^2\ C^*_{Te}\ ]}{(D_j^2 - D_c^2\ )}\right] \tag{12a}$$

$$\text{if}\left[(D_j/D_e)^2 + 2(D_j\ D_c/D_e^2\ ) + 3(D_c/D_e)^2\right] > 6C^*_{T_e} \text{ then}$$
$$\frac{P_j}{P_{oe}} = \left(\frac{D_j - 2r}{D_j - D_c}\right)^2\left[\frac{6(D_e^2\ C^*_{T_e} - D_c{}^2)}{(D_j - D_c)(D_j + 3\ D_c)}\right] \tag{12b}$$

2. For Region 2, defined as $L_c < L < L_a$, the jet pressure is given by

$$\frac{P_j}{P_{jc}} = \left(1 - \frac{2r}{D_j}\right)\left[1 - 2\left(\frac{2r}{D_j}\right)\left[1 - 3C_{Te}\left(\frac{D_e}{D_j}\right)^2\left(\frac{P_{oe}}{P_{jc}}\right)\right]\right] \tag{13}$$

   where

$$\frac{P_{jc}}{P_{oe}} = \left[F_c - 3C_{Te}\left(\frac{D_e}{D_a}\right)^2\right]\frac{L_a(L - L_c)}{L(L_a - L_c)} \quad \begin{array}{l}\text{= Jet centerline}\\ \text{pressure for}\\ (L_c<L<L_a)\end{array} \tag{14}$$

   F_c = 1.0 for $(D_j/D_e)^2 < 6C_{Te}^*$ at $L=L_c$

F_c = 6 CTe* $(D_e/D_j)^2$ for $(D_j/D_e)^2 > 6$ C_{Te}* at $L=L_c$

C_{Te} = Steady-state thrust coefficient defined at break plane conditions

3. For Region 3, defined as $L \geq L_a$, the jet pressure is given by

$$\frac{P_j}{P_{jc}} = \frac{D_j - 2r}{D_j} \quad (15)$$

where

Pjc = $3 F_j/A_j$, jet centerline pressure for $(L > L_a)$ (16)

It should be noted that the above pressure distribution can only be applied to large targets on the jet center line and underexpanded jets. For a small target which is not on the jet center line, judgement should be made to decide if the impingement load can be computed using the jet radial pressure distribution. This is particularly true at the outer edge of the jet where the jet flow may be away from the jet center line and not parallel to it. Additionally, if the stagnation pressure at the break flow area is sufficiently close to the ambient pressure (low pressure jets), the core length, L_c, calculated by equation (1) may be greater than the distance to the asymptotic surface La, calculated by equation (5). For this bounding case, the core length may be set to zero (L_c =0) and the jet pressure distribution assumed to be uniform over the jet cross section and equal to F_j/A_j.

4. Jet Impingement Force

The jet impingement force which is applied to a given target is a function of the fraction of the jet which is intercepted by the target. If the entire jet is intercepted, then the entire jet force is applied to the target.

$$F_{jt} = F_j \quad (17)$$

If the target intercepts a fraction of the jet, but not the entire jet, the jet pressure distribution over the target must be integrated to obtain the jet force.

$$F_{jt} = \int P_j \, dA_t \quad (18)$$

where

P_j = radial jet pressure distribution at the impingement plane, as described in 3.6B.2.3.2.3 above.

A_t = Target area

The impingement load may be estimated from the jet axial force and an approximate correction factor,

$$F_{imp} = K\varphi\ F_{jt}(DLF) \quad (19)$$

where

F_{imp} = impingement force on the target, as a function of time

$K\varphi$ = the shape factor, a measure of the target's potential for changing the momentum of the jet, as described in Appendix D of Reference 22

DLF = Dynamic Load Factor [26]

5. Pipe Whip Dynamic Analysis Criteria

An analysis of the pipe run or branch is performed for each longitudinal and/or circumferential postulated rupture at the break locations determined in accordance with the criteria of Section 3.6B.2.1.2. The loading condition of a pipe run or branch prior to postulated rupture, in terms of internal pressure, temperature and stress state, is assumed to be the condition associated with the normal plant operating condition.

For a circumferential rupture, pipe whip dynamic analyses are only performed for that end (or ends) of the pipe or branch that is connected to a contained fluid energy reservoir having sufficient capacity to develop a jet stream. Dynamic analytical methods used for calculating the piping and piping/restraint system response to the jet thrust developed after a postulated rupture adequately account for the effects of the following:

a. Translational masses (and rotational masses for major components) and stiffness properties of the piping system, restraint system, major components and support walls.

b. Transient forcing function(s) acting on the piping system and jet thrusts on affected structures.

c. Elastic and inelastic deformation of piping and/or restraint.

A 10 percent increase of minimum specified design yield strength (S_y) is used to account for strain rate effects in inelastic nonlinear analyses.

3.6B.2.3.3 Pipe Whip Restraint Design Criteria

1. Design Bases

Pipe whip restraints function primarily as a load carrying member for the low probability occurrence of a pipe break. The restraints are designed for one time use only and function to control the movement of the ruptured pipe. The design basis for the pipe

break event is that of a faulted condition. The restraints and the structure which supports them are analyzed accordingly.

2. Functional Requirements

High-energy pipe whip restraints are designed to ensure that the pipe whip will be eliminated or minimized. On the other hand, the restraints are designed to permit the predicted thermal and seismic movements of the pipes.

3. Design Parameters

After the pipe restraint locations are identified, the following design parameters are determined:

a. Jet thrust force

b. Pipe seismic displacements

c. Pipe thermal displacements

d. Pipe insulation thickness

e. Minimum allowable tolerance between restraint and pipe insulation.

f. Maximum allowable pipe movement.

The jet thrust force and maximum allowable pipe movement are used in the analysis process. Tolerance, insulation, and seismic and thermal movements are used in determining the minimum gap between the restraint and pipe surfaces.

The pipe whip restraints used for the CPNPP project are U-bar, crushable pipe, crushable pad (honeycomb), and elastic hard restraints as shown in Figures 3.6B-1 through 3.6B-4. Existing pipe supports may also be considered as additional hardware capable of providing protection when their use can be analytically justified.

4. Analysis and Design

The maximum allowable design limits for the restraints are as follows:

a. The permanent strain in the metallic ductile materials is limited to fifty percent of the minimum ultimate uniform strain (strain corresponding to the maximum stress point on the appropriate engineering stress-strain curve) based on restraint material tests and/or ASME Code [4].

b. The design limit for crushable pipe restraints longer than three outside pipe diameters is 50 percent of energy absorbing capacity (crushed to 70 percent of inside diameter of pipe).

c. The permissible diameter deformations limit for crushable pipe restraints shorter than two outside pipe diameters is the lesser of D/2 or ASTM where;

D = initial outside diameter of the crushable pipe

ASTM = maximum flattening limit for A106 Gr. B carbon steel pipe as prescribed by ASTM A 530-76.

d. If a crushable material, such as honeycomb, is used, the allowable capacity of crushable material shall be limited to 80% of its rated energy dissipating capacity as determined by dynamic testing, at loading rates within + 50% of the specified design loading rate. The rated energy dissipating capacity shall be taken as not greater than the area under the load-deflection curve. The portion of the curve in which the value of load vs. deflection has departed from the essentially horizontal portion shall not be used.

Typical characteristics of pipe whip restraint components are as follows:

a. Energy absorption members of yielding type restraints are those which, under the influence of the whipping pipe, absorb energy by significant plastic deformation. U-bars, crushable pipe, and crushable pad assemblies, designed with materials having high ultimate strain and relatively high energy absorption capacity, are used in CPNPP.

b. Elastic pipe whip restraints are also used. They are essentially plane frames or space frames. Upon rupture, the energy of the whipping pipe is completely absorbed by the steel restraint structure. Movement of the ruptured pipe is controlled by permitting deformation of the elements of the structural restraints within the elastic range of the material. Existing pipe supports activated during pipe rupture are governed by the criteria applicable to their faulted design rating or other acceptable limits which assures protection is provided such that the effects of the break are mitigated.

c. Restraint connecting members are components which form a direct link between the plastic yielding restraint members and the structure (e.g., clevises, brackets, and pins of a U-bar restraint).

d. Restraint connecting member structural attachments are fasteners which provide the means of securing the restraint connecting members of the structure (e.g., weld attachments and bolts).

e. Structural and civil components are steel and concrete structures which ultimately carry the restraint load (e.g., walls, frames, columns, and beams). A dynamic load factor is considered in the design of these structural and civil components. This factor depends on the natural frequency of these components and the restraint force time history. It is evaluated for each postulated break affecting a specific structural or civil component.

f. The design of the pipe whip restraint is for one time usage.

In designing a restraint, the following three loading conditions are considered:

a. An in-plane (the plane of the U-bar) impact loading at an angle up to 45 degrees from the axis of the U-bar.

b. An out-of-plane (the plane of the U-bar) impact loading at an angle up to 10 degrees from the axis of the U-bar.

c. An impact loading at an angle up to 10 degrees from the normal to the crushable pad/pipe axis.

The materials used in restraint design are selected to ensure ductile behavior. The U-bars are made of SA 479 type 304 annealed stainless steel. The crushable pipes are made of SA 106 Grade B carbon steel. The crushable pads are metallurgically bonded type 304 stainless steel sheets. The elastic restraints are made of ASTM A-588 Grade 50 steel. The other restraint components, such as pins, bolts, and anchors, are designed to remain within their elastic limits.

3.6B.2.4 Guard Pipe Assembly Design Criteria

The CPNPP containment is of single barrier design. Guard pipes are not used in the penetration design.

3.6B.2.5 Material to be Submitted for the Operating License Review

This section presents a summary of the dynamic analyses applicable to high-energy piping systems resulting from postulated pipe breaks. The following information is provided for the various high energy systems in the subparagraphs of this Section:

1. Implementation of the stress criteria as outlined in Section 3.6B.2.1

2. The type, number, and location of postulated breaks on which the dynamic analyses are based.

3. The number and locations of pipe whip restraints required to protect essential systems.

4. The results of the jet thrust, impingement functions, and pipe break analysis as described in Section 3.6B.2.2 and 3.6B.2.3 consistent with Reference [5], where the stress intensity ranges and/or usage factors exceed the criteria of 2.4 Sm and 0.2, respectively.

5. The design adequacy of essential systems and components to ensure that their design-intended functions will not be impaired to an unacceptable level of integrity or operability as the result of high-energy pipe breaks.

6. Description of protective enclosures provided to protect safety-related equipment from the effects of a possible rupture in a high energy fluid piping system, including openings in these enclosures.

The implementation of criteria for inservice inspection is discussed in Section 6.6.8.

3.6B.2.5.1 Reactor Coolant System Main Loop Piping

Table 3.6B-2 and Figure 3.6B-9 identify the RCS main loop break locations. The eight main loop piping and three large branch nozzle break locations (breaks 1 to 11 in Table 3.6B-2 and Figure 3.6B-9) are included in the CPNPP design basis for containment design, ECCS and environmental qualification requirements. These eleven postulated breaks are not part of the design basis for dynamic effects, as discussed in Sections 3.6B.2.1.1 and 3.6B.2.1.2. The two remaining branch nozzle breaks (breaks 12 and 13 in Table 3.6B-2 and Figure 3.6B-9) are included in the design basis for dynamic effects.

Design loading combinations and applicable criteria for ASME Class 1 components and supports are provided in Section 3.9N. The forces associated with rupture of the branch nozzles (12 and 13, Table 3.6B-2) to the reactor coolant loop piping systems are considered in combination with normal operating loads and earthquake loads for the reactor coolant loop design in order to assure continued integrity of vital components and engineered safety features. Pipe rupture loads include not only the jet thrust forces acting on the piping but also jet impingement loads on the primary equipment supports.

Barriers and layout are used to provide protection from pipe whip, blowdown jet and reactive forces. Some of the barriers utilized for protection against pipe whip are as follows. The steam generator compartment walls serve as a barrier between the reactor coolant loops and the containment liner. In addition, the refueling cavity walls, various structural beams, the operating floor, and the steam generator compartment walls enclosed each reactor coolant loop into a separate compartment, thereby preventing an accident, which may occur in any loop, from affecting another loop or the containment liner. The portion of the steam and feedwater lines within the containment have been routed behind barriers which separate these lines from all reactor coolant piping. The barriers described above will withstand loadings caused by jet forces and pipe whip impact forces.

Other than for the Emergency Core Cooling System lines, which must circulate cooling water to the vessel, the engineered safety features are located outside of the steam generator compartment walls. The Emergency Core Cooling System lines which penetrate the steam generator compartment walls are routed around and outside the walls to penetrate the walls in the vicinity of the loop to which they are attached.

3.6B.2.5.2 High-Energy Piping Other Than RCS Main Loop

In this section, a summary is presented giving the results of the detailed stress analysis and, describing methods of protection employed to protect essential equipment against the effects of pipe breaks for the high energy systems outlined in Section 3.6B.1.2.1.

1. Main Steam System

   a. General Description

      The main steam piping inside containment is carbon steel ASME SA-155, Grade KCF 70 material designed in accordance with the ASME Code, Section III, Class 2 criteria. The main steam system inside containment consists of four 32 inch OD (1.25 inch minimum wall thickness) lines running from each steam generator to the containment penetrations.

The main steam piping outside containment from the containment penetrations to the main steam isolation valve moment restraints is the same as the main steam piping inside containment. The piping from the MSIV moment restraints to the high pressure turbine is carbon steel ASME SA-155 Grade KC 70 material designed in accordance with ANSI B31.1 as non-nuclear class piping. These lines are 34 inch OD (1.25 inch minimum wall thickness). The piping connected to the main steam drip pots are of carbon steel material with portions designed in accordance with ANSI B31.1 and ASME Code, Section III, Class 2 criteria. The Class 2 portion of the system consists of ASME SA-333 Grade 6, two inch schedule 80 pipe and the non-nuclear portion of the system consists of ASME SA-106 Grade B, two inch schedule 80 pipe. The location and configuration of the main steam lines with respect to structures, equipment, and other piping are shown on Figures 1.2-8, 1.2-14 and 1.2-25. The criteria described above and as follows is applicable for both Units 1 and 2.

b. Pipe Whip Analysis

Isometrics of the main steam lines inside Containment indicating the location of the highest stress node points, postulated breakpoints, and restraints are provided in Figures 3.6B-11 through 3.6B-14 and 3.6B-100 through 3.6B-103. The systems and equipment necessary to mitigate the consequences of a main steam line break are described in Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at inter-mediate locations in accordance with the criteria out-lined in Section 3.6B.2.1. The steam generator nozzles and the flued heads at the containment penetrations are considered terminal ends. A circumferential break is postulated to occur at any one of these points. Restraints are provided on each line to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

Isometrics of the main steam lines outside containment indicating the location of postulated breakpoints and restraints are provided in Figures 3.6B-15 through 3.6B-18, and 3.6B-104 through 3.6B-107. Since these lines consist of non-nuclear piping, pipe breaks are postulated at each fitting, valve, or welded attachment. Since these lines are greater than 4 inches in diameter, circumferential or longitudinal breaks are postulated. Pipe whip restraints are provided as necessary that are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects. A portion of the main steam lines between the containment penetrations and isolation valves, are designed in accordance with the criteria in Section 3.6B.2.1.2. Therefore, circumferential pipe breaks are not postulated in the regions between the penetrations and the moment restraints after the isolation valves. However, a 1.0 ft^2 crack is postulated and evaluated for environmental effects, in accordance with the criteria in BTP ASB 3-1. As shown on Figures 3.6B-25 and 3.6B-119 through 3.6B-122, pipe breaks are postulated in the main steam blowdown lines outside containment. Since these lines consist of non-nuclear piping, break points are postulated at terminal ends and at each intermediate fitting, valve or welded attachment with the exception of the four break exclusion areas as shown.

Review of the piping layout and plant arrangement showed that breaks in portions of the main steam lines could adversely impact the safeguards, switchgear, electrical and controls and turbine buildings. Accordingly, these lines are restrained as necessary to protect the structures. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects on the structures. Since there are no essential systems or components in the turbine building, protection is primarily provided to protect the structures noted above from a main steam pipe break.

c. Jet Impingement Analysis

The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

In piping tunnels that contain break exclusion regions of main steam lines, the safety-related equipment in these tunnels are designed to withstand the environmental effects of a non-mechanistic crack with a flow area of 1.0 ft^2. Jets are not generated, and hence jet Loads are not considered in break exclusion regions (superpipe areas).

A small quantity of essential cables is located in the superpipe areas; however, these cables service equipment that because of their function, are appropriately located in these areas. Other essential cables are not located in and do not transit through these areas. The low concentrations of the essential cables located in these areas does not violate the NRC staff's pipe break separation goal for break exclusion areas.

d. Environmental Analysis

The safety-related systems required to mitigate the consequences of a main steam line break inside Containment are designed to perform their safety function under the environmental conditions resulting from a LOCA or MSLB as discussed in Section 3.11.

The penetration area is isolated from the rest of the Safeguards Building by the seismic gap seal between the Safeguards Building and the containment wall, fast closing isolation dampers in the interconnecting HVAC ducts and the pressure resisting watertight doors. These features preclude this crack from producing any detrimental environmental effects on the rest of the plant.

Any one of the main steam lines is postulated to develop a 1 ft^2 crack in the penetration area outside Containment. This crack is postulated to be the most severe break for all rooms, based on the impact to equipment in rooms outside the

break area, the amount of energy released, and the generation of the highest peak. The postulation of this crack is in accordance with Branch Technical Position ASB 3-1.

The only equipment that could be required to operate and that could be in the room where the break occurs are the ventilation dampers. These dampers close on pressure differential; pressure helps keep these dampers closed. The temperature spike will not degrade the ability of these dampers to close and remain closed.

Mass and energy release data were calculated for 100% including power uncertainty, 90%, 80%, 70% and 50% power. Full power including power uncertainty with the 1.0 ft^2 break provided the worst environmental results, i.e., the highest peak temperature. Table 3.6B-4A (Unit 1) and Table 3.6B-4B (Unit 2) provide the mass and energy release data. Mass/energy releases were calculated using a modified version of the LOFTRAN code [33], and it and the related blowdown analysis are plant-specific to CPNPP. LOFTRAN was modified to model the heat transfer which may occur in the uncovered portion of the tube bundle of a steam generator and to calculate the resulting superheated steam mass and energy release. Conservative assumptions were made in order to lead to early tube bundle uncovery, and as a result the earliest superheat initiation time. Primary system temperatures were maximized to increase the primary-to-secondary heat transfer, promoting earlier tube bundle uncovery and maximizing the superheat steam enthalpy following tube bundle uncovery. Table 3.6B-4A and Table 3.6B-4B also provides the time sequence of events based on a 1.0 ft^2 break and an initial power level of 90% including power uncertainty. The environmental analysis used to determine the temperature transient due to this break is described in Section 3.6B.1.2.3.

An Equipment Qualification review of Class 1E equipment in the affected compartment has been performed and Class 1E items have been determined to be qualified for the environment. The Class 1E cables located in the areas affected by the crack are those cables required to support the Class 1E equipment in the same areas. All of these cables are qualified for the LOCA/MSLB inside containment. As such, these cables will remain fully operational throughout this event. Cable separation in these areas meets the requirements of Regulatory Guide 1.75. See Table 3.6B-5 for the remaining safety-related equipment located in the area affected by the steam line break with superheat.

2. Feedwater System

a. General Description

The main feedwater lines inside containment are carbon steel ASME SA-333, Grade 6 material designed in accordance with the ASME Code, Section III, Class 2 criteria. Due to the Unit RSG installation the 18” x 16” reducing elbow material at the RCS nozzle inlet is now SA-508, Grade 2, Class 1. Each of the lines consists of an 18 inch schedule 80 seamless pipe running from the Containment penetration to each steam generator. The main feedwater piping

outside Containment from the containment penetration to the feedwater containment isolation valve is the same as the feedwater piping inside Containment. The piping from the feedwater containment isolation valve to the feedwater control valve moment restraint is also the same as the feedwater piping inside containment, except that these lines are 18 inch schedule 140 seamless pipe. The main feedwater lines from the feedwater control valve moment restraint connect to the main feedwater header combining into one main feedwater line which originates from the feedwater heater in the turbine building. The feedwater lines to the main feedwater header and the feedwater control valve by-pass lines are carbon steel ASME SA-106, Grade B material designed in accordance with ANSI B31.1 as non-nuclear class piping. These lines consist of 18 inch schedule 140 and eight inch schedule 120 pipe. The main feedwater header and feedwater line in the turbine building are carbon steel ASTM A155 Gr KC 60 material designed in accordance with ANSI B31.1 as non-nuclear class piping. These lines are 30 inch OD (2.125 inch minimum wall thickness) pipe. The location and configuration of the feedwater lines with respect to structures, equipment, and other piping are shown in Figures 1.2-8, 1.2-13 and 1.2-25.

The criteria described above and as follows is applicable for both Units 1 and 2.

b. Pipe Whip Analysis

Isometrics of the main feedwater lines inside Containment indicating the location of the highest stress node points, postulated breakpoints, and restraints are provided in Figures 3.6B-19 through 3.6B-22 and 3.6B-108 through 3.6B-111. The systems and equipment necessary to mitigate the consequences of a main feedwater line break are described in Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. The flued heads at the containment penetrations and the steam generator nozzles are considered terminal ends. A circumferential break is postulated to occur at any one of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects. Existing pipe supports were also activated in the pipe whip analysis as additional existing hardware capable of mitigating the effects of the break.

Isometrics of the main feedwater lines outside containment, indicating the location of postulated breakpoints and restraints, are provided in Figures 3.6B-23, 3.6B-24 and 3.6B-112 through 3.6B-118. Since these lines consist largely of non-nuclear piping, pipe breaks are postulated at each fitting, valve, or welded attachment. For all lines, circumferential or longitudinal breaks are postulated. Pipe whip restraints are provided to prevent unacceptable damage to essential components and building structures. The pipe whip restraints are designed to prevent plastic hinge formation to preclude adverse pipe whip effects. A portion of the main feedwater lines between the containment penetrations and the feedwater control valves, is designed in accordance with the criteria in Section 3.6B.2.1.2. Therefore, pipe breaks are not postulated in the regions between the penetrations and the moment restraints after the control valves. However, a one square foot

crack is postulated and evaluated for environmental effects in accordance with the criteria in BTP ASB 3-1.

Review of the piping layout and plant arrangement showed that breaks in portions of the main feedwater lines could adversely impact the safeguards, switchgear, electrical and controls and turbine buildings. Accordingly, these lines are restrained as necessary to protect the structures. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects on the structures. Since there are no essential systems or components in the turbine building, protection is primarily provided to protect the structures noted above from a feedwater line break.

c. Jet Impingement Analysis

The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

Jet Loads are not considered in break exclusion regions (superpipe areas). A small quantity of essential cables is located in the superpipe areas; however, these cables service equipment that because of their function, are appropriately located in these areas. Other essential cables are not located in and do not transit through these areas. The low concentrations of the essential cables located in these areas does not violate the NRC staff's pipe break separation goal for break exclusion areas.

d. Environmental Analysis

The safety-related systems required to mitigate the consequences of a main feedwater line break inside Containment are designed to perform their safety function under the environmental conditions resulting from a LOCA or MSLB as discussed in Section 3.11.

Any one of the main feedwater lines is postulated to develop a 1 ft^2 crack in the penetration area outside Containment. The postulation of this crack is in accordance with Branch Technical Position ASB 3-1.

The penetration area is isolated from the rest of the safeguards building by the seismic gap seal between the safeguards building and the containment wall, fast closing isolation dampers in the interconnecting HVAC ducts and the pressure resisting watertight doors. These features preclude this crack from producing any detrimental environmental effects on the rest of the plant. High energy flooding is evaluated in the same manner as moderate energy flooding per Section 3.6B.2.5.3.

3. Auxiliary Feedwater System

a. General Description

The Auxiliary Feedwater System lines are carbon steel, ASME SA 106, Grade B and SA 333, Grade 6 material designed in accordance with ASME Code Section III, Class 2 or 3 criteria as applicable. The suction lines of the auxiliary feedwater pumps consist of 10, 8 and 6 inch category 152 piping which is schedule 40, with design pressure of 150 psig. These lines run from the condensate storage tank and service water piping to the suction of each auxiliary feedwater pump. The discharge lines of the auxiliary feedwater pumps consist of 6, 4 and 3 inch category 2002 (category 2003, directly after the isolation valve) piping which is schedule 160 up to 3 inches, and schedule 120 for sizes above 3 inches. The design pressure is 1800 psig. The piping that connects to the main feedwater piping after the last auxiliary feedwater isolation valves is category 1303 which is schedule 80, with a design pressure of 1200 psig to match the main feedwater piping. The discharge lines of the auxiliary feedwater pumps connect to the main feedwater lines. The turbine driven auxiliary feedwater pump discharge piping up to isolation check valves AF-078, AF-106, AF-086, and AF-096 is classified as moderate energy piping since operation as a high energy fluid system is for a limited period as prescribed by 3.6B.2.1.4. The location and configuration of the auxiliary feedwater lines with respect to structures, equipment and other piping are shown in Figure 1.2-10. The criteria described above and as follows are applicable for both Units 1 and 2.

b. Pipe Whip Analysis

Isometrics of the Auxiliary Feedwater System lines indicating the location of the highest stress node points, postulated breakpoints and restraints are provided in Figures 3.6B-26 through 3.6B-33 and 3.6B-124 through 3.6B-130. The systems and equipment necessary to mitigate the consequences of a Auxiliary Feedwater System line break are described in Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. Intermediate breaks are postulated as shown. A circumferential break is postulated to occur at each of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

c. Jet Impingement Analysis

The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

d. Environmental Analysis

The environmental conditions resulting from a break in an Auxiliary Feedwater System line do not affect the operation of safety-related systems required to mitigate the consequences of the accident.

The Auxiliary Feedwater System is high energy only on the basis of the system pressure. During the AFW System operation, for breaks upstream of isolating check valves 1AF-093, 1AF-101, 1AF-083, 1AF-075, 2AF-093, 2AF-101, 2AF-083 and 2AF-075, cold water jets are developed. For breaks downstream of these check valves there exist jets that can produce a steam atmosphere in the feedwater tunnel area. These high temperature jets originate from feedwater during plant warm-up (up to 250°F maximum) (Unit 2 only). The resultant pressure and temperature from these breaks are less severe than those analyzed for this area due to the non-mechanistic one square foot crack of the feedwater piping.

High energy flooding is evaluated in the same manner as moderate energy flooding per Section 3.6B.2.5.3.

4. Auxiliary Steam System

a. General Description

The Auxiliary Steam System Piping is comprised of non-nuclear safety related carbon steel, ASME SA-106 Grade B material. The auxiliary steam system supplies steam to both Units 1 and 2 components.

The steam supply piping to the floor drain waste evaporator package, WPS waste evaporator package, BRS recycle evaporator package and CVCS boric acid batching tank, all located in the Auxiliary Building, are 10, 6, 4 and 2 inch, schedule 40 category 152 Class 5 piping with a design pressure of 150 psig. These lines run from the auxiliary steam header of the components described above and are seismically supported. The return lines of the above components are 2, 1-1/2 and 3/4 inch schedule 40, category 152 Class 5 piping with a design pressure of 150 psig. The return lines are connected to the auxiliary steam drain tank. The location and configuration of the auxiliary steam lines with respect to structures, equipment and other piping are shown in Figures 1.2-31 and 1.2-32.

b. Pipe Whip Analysis

Isometric drawings of the Auxiliary Steam System Piping indicating the locations of the highest stresses, postulated breakpoints and restraints are provided in Figures 3.6B-58 through 3.6B-63. The systems and equipment necessary to mitigate the consequences of an Auxiliary Steam System line break are described in Section 3.6B.1. Break locations were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. A circumferential break is postulated to occur at each of these points. Pipe whip restraints are provided as required to prevent impact on

essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

c. Jet Impingement Analysis

The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

d. Environmental Analysis

The environmental conditions resulting from a break in an Auxiliary Steam System line do not affect the operation of safety-related systems required to mitigate the consequences of the accident.

The most significant environmental conditions generated by breaks in the auxiliary steam system are the various elevated temperatures. The pressure transients are very slight and do not pose a threat to the safe operation or the structure of the plant. The temperature transients caused by any of the auxiliary system breaks do not affect the safe shutdown of the plant because: 1) equipment required for safe shutdown is located in an area not affected by the break; 2) the equipment is qualified to parameters higher than those experienced during the break; 3) the equipment fails in a safe position; or 4) the equipment has been analyzed to show that the short-duration increased temperatures will not cause appreciable degradation in performance or qualified life.

Blowdown from any of the postulated auxiliary steam system line breaks is terminated automatically. The transients were analyzed for the full period of the blowdown.

Details of the instrumentation required to mitigate the blowdown is in Section 7.6.12.

5. Steam Generator Blowdown Cleanup System

a. General Description

The Steam Generator Blowdown Cleanup System Piping is comprised of non-nuclear safety related, carbon steel ASME SA-106 Grade B material, or non-nuclear safety related stainless steel ASTM A-312 TP 304 or 316 material.

The blowdown lines from each steam generator up to the isolation valve moment restraints are 2", 3" or 4" safety class 2, Schedule 80, category 1303 piping with a design pressure of 1200 psig. The piping connecting the 8-inch header of the steam generator blowdown heat exchanger to the pressure reducing

valve PV-5180 is schedule 80, category 1302, Class 5, with a design pressure of 1200 psig. The portion of the blowdown piping on the Switchgear Building roof is category 1302G, and not seismically supported. The piping from valve PV-5180 up to valve SB-170 is 8-inch, category 302, Class 5, schedule 40, with a design pressure of 450 psig. The piping from valve SB-170 to the filters and demineralizers is 6-inch, category 301, Class 5, schedule 40S with a design pressure of 370 psig. The piping that connects the discharge of relief valve SB020 to the condenser is 4 and 8 inch, category 302, Class 5, Schedule 40, with a design pressure of 450 psig. The piping downstream of the filters is moderate energy piping. The location of the steam generator blowdown cleanup system with respect to structures and equipment are shown in Figures 1.2-31 and 1.2-35.

b. Pipe Whip Analysis

Isometrics of the Steam Generator Blowdown Cleanup System Piping indicating the locations of the highest stresses postulated breakpoints and restraints are provided in Figures 3.6B-38 through 3.6B-45 and 3.6B-136 through 3.6B-144. The systems and equipment necessary to mitigate the consequences of a Steam Generator Blowdown Cleanup System line break are described in Section 3.6B.1. Break locations were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1 except for category 1302G piping, where breaks are postulated at terminal ends and at each intermediate fitting, valve or welded attachment. A circumferential break is postulated to occur at each of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

c. Jet Impingement Analysis

The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

d. Environmental Analysis

The environmental conditions resulting from a break in a Steam Generator Blowdown System line do not affect the operation of safety-related systems required to mitigate the consequences of the accident.

All equipment required for mitigating the consequences of the accident are located in areas that are isolated from the pipe break event by the automatic closure of HVAC dampers. The leakage that occurs prior to the closure of the dampers has been analyzed and the effects of temperature, pressure and

humidity intrusion evaluated. A total of 20 dampers, 2 in each HVAC duct, are installed and designed to close in a short time after the pipe break occurrence (less than 5 seconds). The dampers are qualified for the transient environmental conditions of the break.

Blowdown from any of the postulated blowdown line breaks is terminated automatically. The transients were analyzed for the full period of the blowdown.

Details of the instrumentation required to mitigate the blowdown is in Section 7.6.12.

High energy flooding is evaluated in the same manner as moderate energy flooding per Section 3.6B.2.5.3.

6. Chemical & Volume Control System

a. General Description

The Chemical and Volume Control System lines are of stainless steel ASME SA-312, TP 304 material or stainless steel ASME SA-312 TP 304 material or stainless steel ASME SA-376 TP 304 material designed in accordance with ASME Code Section III, Class 1, 2 or 3 criteria as applicable. The letdown line from the cold leg of the RCS to the letdown orifices consists of 3 and 2 in category 2501 schedule 160 piping with a design pressure of 2485 psig. The letdown orifices on the low pressure letdown valve PCV-131 is 2 and 3 inch category 601, schedule 40S piping with a design pressure of 700 psig. The letdown line from the low pressure letdown valve to the demineralizers and RHT of the BRS is 3 inch category 301, Schedule 40S, with a design pressure of 370 psig. The suction lines of the charging pumps consist of 1, 2, 3, 4 and 6 inch category 151, schedule 40S piping with a design pressure of 150 psig. These lines run from the volume control tank, CVCS boric acid tank and chemical mixing tank to the suction of the charging pumps. The discharge lines of the charging pumps consist of 4, 3 and 2 inch category 2501 schedule 160 piping, with a design pressure of 2485 psig. These lines run from the charging pumps to the cold leg of the RCS and the RCP seals. The location and configuration of the CVCS lines with respect to structures and equipment are shown in Figures 1.2-11, 1.2-12, 1.2-17, 1.2-18 and 1.2-32.

b. Pipe Whip Analysis

Isometrics of the CVS system piping indicating location of the highest stresses, postulated breakpoints and re-straints are provided in Figures 3.6B-70 through 3.6B-88 and 3.6B-183 through 3.6B-206. The systems and equipment necessary to mitigate the consequences of a CVC system line break are described in Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. A circumferential break is postulated to occur at each of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

c. Jet Impingement Analysis

The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shields or barriers to protect the target.

d. Environmental Analysis

The environmental conditions resulting from a break in an CVC System line do not affect the operation of safety-related systems required to mitigate the consequences of the accident.

The most significant environmental conditions generated by breaks in the CVC system are the various elevated temperatures. The pressure transients are small and do not pose a threat to the safe operation or the structure of the plant. The temperature transients caused by any of the CVCS system breaks do not affect the safe shutdown of the plant because: 1) equipment required for safe shutdown is located in an area not affected by the break; 2) the equipment is qualified to parameters higher than those experienced during the break; 3) the equipment fails in a safe position; or 4) the equipment has been analyzed to show that the short-duration increased temperatures will not cause appreciable degradation in performance or qualified life.

Details of the instrumentation required to mitigate the blowdown is in Section 7.6.12.

High energy flooding is evaluated in the same manner as moderate energy flooding per Section 3.6B.2.5.3.

7. Residual Heat Removal System

a. General Description

The Residual Heat Removal System piping from the RCS hot leg of loop 1 and 4 to the second isolation valve are of stainless steel, ASME SA-376 TP 316 material designed in accordance with ASME Code Section III, Class 1 criteria. These lines consist of 12 inch category 2501, schedule 140 piping with a design pressure of 2485 psig. The piping from the second isolation valve to the containment penetrations is of stainless steel ASME SA-312 TP 304 material designed in accordance with ASME Code section III, Class 2 criteria. These lines are 12 inch category 601, schedule 40S piping with a design pressure of 600 psig. The piping from the containment penetration to the suction of the RHR pumps is of stainless steel ASME SA-358, Class 1. These lines are 16 inch (0.5 inch minimum wall thickness) category 601 with a design pressure of 600 psig. The discharge lines of the Residual Heat Removal pumps are of stainless steel, ASME SA-312

TP 304 material designed in accordance with ASME Code Section III, Class 2 criteria. These lines consist of 8, 10 and 12 inch category 601, schedule 40S piping with a design pressure of 600 psig. The discharge lines of the residual heat removal pumps connect to safety injection system cold leg injection headers. The location of the Residual Heat Removal System piping with respect to structures and equipment is shown in Figures 1.2-10 and 1.2-16.

b. Pipe Whip Analysis

Isometric drawings of the RHR System piping indicating the locations of the highest stresses postulated breakpoints and restraints are provided in Figures 3.6B-64, 3.6B-172 and 3.6B-173. The systems and equipment necessary to mitigate the consequences of a RHR system line break are described in Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. A circumferential break is postulated to occur at each of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

c. Jet Impingement Analysis

The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components (as defined in Section 3.6B.1.2.1), associated supports and building structures.

In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

d. Environmental Analysis

The RHR System is considered a moderate energy system outside containment in accordance with Branch Technical Position MEB 3-1.

Moderate energy flooding is evaluated in accordance with Section 3.6B.2.5.3.

8. Safety Injection System

a. General Description

The Safety Injection System Piping from the discharge header of the charging pumps to the Reactor Coolant System loop cold legs are stainless steel, ASME SA-376 TP 304 or TP 316 material, designed in accordance with ASME Code Section III, Class 1 or 2 criteria, as applicable. These lines consist of 1, 1-1/2, 2, 3, 4 and 6 inch category 2501, schedule 160 piping with a design pressure of 2485 psig. The piping from the accumulators to the Reactor Coolant System loop cold legs is stainless steel, ASME SA-376 TP 304 or TP 316 material, designed in accordance with ASME Code Section III, Class 1 or 2

criteria, as applicable. These lines are 10 inch category 2501 schedule 140 piping with a design pressure of 2485 psig.

The safety injection pump discharge piping up to the isolation valves outside the containment are stainless steel ASME SA-312 TP 304 or TP 316 material, designed in accordance with ASME Code Section III, Class 2 criteria. These lines consist of 3 and 4 inch category 1501, Schedule 80S piping with a design pressure of 1860 psig. The safety injection pump discharge piping from the isolation valves outside the containment to the Reactor Coolant System hot legs and Safety Injection System cold legs is stainless steel ASME SA-376 TP 304 or TP 316 material, designed in accordance with ASME Code Section III, Class 1 or 2 criteria as applicable. These lines consist of 3/4, 2, 4, 6, 8 and 10 inch category 2501, schedule 160 or 140 piping, as applicable, with a design pressure of 2485 psig. The safety injection pump suction piping is of stainless steel, ASME SA-312, TP 304 or TP 316 material, designed in accordance with ASME Code Section III, Class 2 criteria. These lines consist of 6 and 8 inch category 151 schedule 40S piping with a design pressure of 150 psig. The location of the Safety Injection System piping with respect to structures and equipment are shown in Figures 1.2-10, 1.2-11, 1.2-16 and 1.2-17.

b. Pipe Whip Analysis

Isometric drawings of the Safety Injection System lines indicating the locations of the highest stresses postulated breakpoints and restraints are provided in Figures 3.6B-48 through 3.6B-57 and 3.6B-147 through 3.6B-169. The systems and equipment necessary to mitigate the consequences of a Safety Injection System line break are described in Section 3.6B.1. Breakpoints were postulated at the terminal ends of the piping run and at intermediate locations in accordance with the criteria outlined in Section 3.6B.2.1. A circumferential break is postulated to occur at each of these points. Pipe whip restraints are provided as required to prevent impact on essential components. The pipe whip restraints are designed to prevent plastic hinge formation and thereby preclude adverse pipe whip effects.

c. Jet Impingement Analysis

The jet impingement analysis for this system is performed to determine the effects of jet impingement loading on essential components, associated supports and building structures.

In those cases where the analysis shows that the component or structure is not capable of withstanding the load then protection is required. Protection consists of either relocating the target, or installing jet shield or barriers to protect the target.

d. Environmental Analysis

The environmental conditions resulting from a break in an SI System line do not affect the operation of safety-related systems required to mitigate the consequences of the accident.

Those portions of the Safety Injection System which are considered high energy, outside the containment, are based on the system pressure. The piping is considered as containing cold water so a break in a line cannot generate a steam atmosphere.

High energy flooding is evaluated in the same manner as moderate energy flooding per Section 3.6B.2.5.3.

3.6B.2.5.3 Moderate Energy Piping

In evaluating the effects of a moderate energy system piping failure, the postulated failure is a crack which results in neither pipe whip nor jet impingement but rather in spraying water streams. As such, the consequences are of an environmental/flooding nature. The effects of cracks, as postulated in Section 3.6B.2.1.4, are evaluated for all essential equipment on a case by case basis.

If it is determined that an essential component is not qualified or cannot be demonstrated to operate under the adverse conditions caused by the crack then the essential component is protected. Protection is accomplished either by relocating the component, installing a barrier or curb or by designing a shield.

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TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 1 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
SYSTEM AUXILIARY FEEDWATER (AF)					
4AF-1-49-2002-3	SB	X		MOTOR DRIVEN AF PUMP TRAIN A (DISCHARGE)	10.4-11
4AF-1-36-2002-3	SB	X		AF PUMP TRAIN A DISCH. (TEST LINE INCLUDED)	10.4-11
3AF-1-38-2002-3	SB	X		AF PUMP TRAIN A MINIMUM FLOW RECIRC. LINE	10.4-11
2AF-1-65-2002-3	SB	X		AF PUMP TRAIN A MINIMUM FLOW RECIRC. LINE	10.4-11
3AF-1-28-2002-3 (FROM 2X3 REDUCER TO BREAKDOWN ORIFICE CP1-AFORBO-01 ONLY)	SB	X		AF PUMP TRAIN A MINIMUM FLOW RECIRC.	10.4-11
6AF-1-10-2002-3	SB	X		AF TRAIN A HEADER TO SG-01 AND SG-02	10.4-11
4AF-1-50-2002-3 (THRU VALVE 1AF-074 TO 4X3 REDUCER)	SB	X		AF TRAIN A TO SG-01	
3AF-1-67-2002-3	SB	X		AF TRAIN A TO SG-01	10.4-11
3AF-1-75-2002-3	SB	X		AF TRAIN A TO SG-01	10.4-11
4AF-1-97-2002-3 (FROM 4X3 REDUCER THRU VALVE 1-AF-0121 AND CP1-AFORFR-08 TO VALVE 1-AF-0075)	SB	X		AF TRAIN A TO SG-01	
4AF-1-105-2002-3		X		AF TRAIN A TO SG-01	10.4-114
4AF-1-52-2003-2	RB	X		AF TRAIN A TO SG-01 (NOTE 1)	10.4-11
4AF-1-17-1303-2	SB	X		AF TRAIN A TO SG-01 (NOTE 1)	10.4-11
4AF-1-104-2002-3	SB	X		TURBINE DRIVEN AF PUMP TO SG-01	10.4-11
4AF-1-21-2003-3	SB	X		TURBINE DRIVEN AF PUMP TO SG-01 (NOTE 1)	10.4-11

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 2 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
4AF-1-61-2002-3	SB	X		AF TRAIN A TO SG-02	10.4-11
3AF-1-69-2002-3	SB	X		AF TRAIN A TO SG-02	10.4-11
3AF-1-77-2002-3	SB	X		AF TRAIN A TO SG-02	10.4-11
4AF-1-99-2002-3 (THRU VALVE 1AF-123 AND ORIFICE CP1-AFORFR-06 TO VALVE 1AF-083)	SB	X		AF TRAIN A TO SG-02	10.4-11
4AF-1-107-2002-3	SB	X		AF TRAIN A TO SG-02	10.4-11
4AF-1-53-2003-2	SB	X		AF TRAIN A TO SG-02 (NOTE 1)	10.4-11
4AF-1-18-1303-2	SB	X		AF TRAIN A TO SG-02 (NOTE 1)	10.4-11
4AF-1-22-2003-2	SB	X		TURBINE DRIVEN AF PUMP TO SG-02 (NOTE 1)	10.4-11
4AF-1-106-2002-3	SB	X		TURBINE DRIVEN AF PUMP TO SG-02	10.4-11
4AF-1-42-2002-3	SB	X		MOTOR DRIVEN AF PUMP, TRAIN B DISCHARGE TO SG-03 AND SG-04	10.4-11
6AF-1-09-2002-3	SB	X		MOTOR DRIVEN AF PUMP, TRAIN B DISCHARGE TO SG-03 AND SG-04	10.4-11
3AF-1-31-2002-3	SB	X		AF PUMP TRAIN B MINIMUM FLOW RECIRC LINE	10.4-11
2AF-1-66-2002-3	SB	X		AF PUMP TRAIN B MINIMUM FLOW RECIRC LINE	10.4-11
3AF-1-30-2002-3 (FROM 2X3 REDUCER TO BREAK DOWN ORIFICE CP1-AFORBO-02 ONLY)	SB	X		AF PUMP TRAIN B MINIMUM FLOW RECIRC LINE	10.4-11
6AF-1-33-2002-3	SB	X		TEST LINE, TRAIN B	10.4-11
4AF-1-62-2002-3	SB	X		TRAIN B TO SG-03	10.4-11
3AF-1-71-2002-3	SB	X		AF TRAIN B TO SG-03	10.4-11

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 3 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
3AF-1-79-2002-3	SB	X		AF TRAIN B TO SG-03	10.4-11
4AF-1-101-2002-2 (FROM 4X3 REDUCER THRU VALVE 1AF-125 AND ORIFICE CP1-AFORFR-04 TO VALVE 1AF-093	SB	X		AF TRAIN B TO SG-03	10.4-11
4AF-1-109-2002-3	SB	X		AF TRAIN B TO SG-3	10.4-11
4AF-1-54-2003-2	SB	X		AF TRAIN B TO SG-3 (NOTE 1)	10.4-11
4AF-1-019-1303-2	SB	X		AF TRAIN B TO SG-3 (NOTE 1)	10.4-11
4AF-1-23-2003-2	SB	X		TURBINE DRIVEN AF PUMP TO SG-3 (NOTE 1)	10.4-11
4AF-1-108-2002-3	SB	X		TURBINE DRIVEN AF PUMP TO SG-03	10.4-11
4AF-1-51-2002-3	SB	X		AF TRAIN B TO SG-04	10.4-11
3AF-1-73-2002-3	SB	X		AF TRAIN B TO SG-04	10.4-11
3AF-1-81-2002-3	SB	X		AF TRAIN B TO SG-04	10.4-11
4AF-1-103-2002-3 (FROM 4X3 REDUCER THRU VALVE 1AF-127 AND ORIFICE CP1-AFORFR-10 TO VALVE 1AF-101)	SB	X		AF TRAIN B TO SG-04	10.4-11
4AF-1-111-2002-3	SB	X		AF TRAIN B TO SG-04 (NOTE 1)	10.4-11
4AF-1-55-2003-2	SB	X		AF TRAIN B TO SG-04 (NOTE 1)	10.4-11
4AF-1-20-1303-2	SB	X		AF TRAIN B TO SG-04 (NOTE 1)	10.4-11
4AF-1-24-2003-2	SB	X		TURBINE DRIVEN AF PUMP TO SG-04 (NOTE 1)	10.4-11
4AF-1-110-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-04	10.4-11

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 4 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
8AF-1-11-2002-3	SB	X		TURBINE DRIVEN AF FEED PUMP TO ALL STEAM GENERATORS (NOTE 2)	10.4-11
3AF-1-68-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-01 (NOTE 2)	10.4-11
3AF-1-76-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-01 (NOTE 2)	10.4-11
4AF-1-96-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-01 (NOTE 2)	10.4-11
3AF-1-70-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-02 (NOTE 2)	10.4-11
3AF-1-78-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-02 (NOTE 2)	10.4-11
4AF-1-98-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-02 (NOTE 2)	10.4-11
3AF-1-72-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-03 (NOTE 2)	10.4-11
3AF-1-80-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-03 (NOTE 2)	10.4-11
4AF-1-100-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-03 (NOTE 2)	10.4-11
3AF-1-74-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-04 (NOTE 2)	10.4-11
6AF-1-25-2002-3	SB	X		TURBINE DRIVEN AF FEED PUMP TO ALL STEAM GENERATORS (NOTE 2)	10.4-11
3AF-1-26-2002-3	SB	X		TURBINE DRIVEN PUMP MIN. FLOW RECIRC LINE (NOTE 2)	10.4-11
3AF-1-82-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-04 (NOTE 2)	10.4-11
4AF-1-102-2002-3	SB	X		TURBINE DRIVEN PUMP TO SG-04 (NOTE 2)	10.4-11
6AF-1-29-2002-3	SB	X		TURBINE DRIVEN AUX. FEEDWATER PUMP TEST LINE (NOTE 2)	10.4-11

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 5 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
6AF-1-88-2002-3	SB	X		MOTOR DRIVEN PUMPS (TRAIN A + B CROSS-TIE LINE EXCLUDING THE SEGMENT BETWEEN VALVES 1AF-090 AND 1AF-091	10.4-11
SYSTEM CHEMICAL AND VOLUME CONTROL (CS)					
3CS-1-235-2501R-1	RB		X	RCS LETDOWN LINE	9.3-10
3CS-1-001-2501R-2	RB		X	LETDOWN LINE PRIMARY SIDE H. EX. INLET	9.3-10
3CS-1-002-2501R-2	X		X	LETDOWN LINE PRIMARY SIDE H. EX OUTLET	9.3-10
2CS-1-005-2501R-2	RB		X	LETDOWN ORIFICE INLET	9.3-10
2CS-1-008-2501R-2	RB		X	LETDOWN ORIFICE DISCHARGE	9.3-10
2CS-1-004-2501R-2	RB		X	LETDOWN ORIFICE INLET	9.3-10
2CS-1-007-2501R-2	RB		X	LETDOWN ORIFICE DISCHARGE	9.3-10
2CS-1-003-2501R-2	RB		X	LETDOWN ORIFICE	9.3-10
2CS-1-006-2501R-2	RB		X	LETDOW ORIFICE DISCHARGE	9.3-10
2CS-1-009-601R-2	RB		X	LETDOWN ORIFICE DISCHARGE	9.3-10
2CS-1-010-601R-2	RB		X	LETDOWN ORIFICE DISCHARGE	9.3-10
2CS-1-011-601R-2	RB		X	LETDOWN ORIFICE DISCHARGE	9.3-10
3CS-1-012-601R-2	SB RB		X	LETDOWN LINE (NOTE 1)	9.3-10
2CS-1-546-601R-2	RB		X	LETDOWN LINE RELIEF VALVE LINE	9.3-10
3CS-1-013-601R-2	SB		X	CONNECTION TO LETDOWN REHEAT H EX	9.3-10

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 6 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
4CS-1-549-601R-2	SB		X	LETDOWN REHEAT HX INLET	9.3-10
4CS-1-550-601R-2	SB		X	LETDOWN REHEAT HX OUTLET	9.3-10
3CS-1-014-601R-2	SB		X	LETDOWN REHEAT HX OUTLET	9.3-10
3CS-1-354-601R-2	SB		X	VALVES TCV-381B BY PASS LINE	9.3-10
2CS-1-348-601R-2	SB		X	RHRS CLEAN UP CROSS CONNECTION	9.3-10
3CS-1-015-601R-2	SB	X		LETDOWN LINE	9.3-10
2CS-1-017-601R-2	AB	X		CHARGING DISCHARGE FROM TBX-CSAPPD-01	9.3-10
4CS-1-076-2501R-2	AB	X		CHARGING PUMP DISCHARGE FROM TBX-CSAPCH-01	9.3-10
4CS-1-344-2501R-2	AB	X		CHARGING PUMP DISCHARGE FROM TBX-CSAPCH-01	9.3-10
4CS-1-085-2501R-2	AB	X		CHARGING PUMP DISCHARGE FROM TBX-CSAPCH-02	9.3-10
4CS-1-346-2501R-2	AB	X		CHARGING PUMP DISCHARGE FROM TBX-CSAPCH-02	9.3-10
4CS-1-110-2501R-2	AB	X		CHARGING PUMP DISCHARGE HEADER	9.3-10
3CS-1-075-2501R-2	AB	X		CHARGING PUMP DISCHARGE HEADER	9.3-10
3CS-1-343-2501R-2	AB	X		CENTRIFUGAL CHARGING PUMP TRAIN A, BY PASS	9.3-10
2CS-1-345-2501R-2	AB	X		CENTRIGUGAL CHARGING PUMP TRAIN B, BY PASS	9.3-10
2CS-1-483-2501R-2	AB	X		CENTRIFUGAL CHARGING PUMP TRAIN A/B CROSS CONNECTIONS (NOTE 20)	9.3-10
2CS-1-482-2501R-2	AB	X		FCV-0121 BYPASS (NOTE 19)	9.3-10
2CS-1-338-2501R-2	AB	X		POS. DISPL. PUMP BYPASS VALVE LINE (NOTE 4)	9.3-10

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 7 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
3CS-1-074-2501R-2	AB SB	X		CHARGING LINE	9.3-10
2CS-1-349-2501R-2	AB	X		FLOW CONTROL VALVE BY PASS LINE	9.3-10
3CS-1-077-2501R-2	RB SB	X		CHARGING LINE (REGEN H. EX. INLET) (NOTE 1)	9.3-10
3CS-1-547-2501R-2	RB		X	CHARGING LINE (REGEN. H. EX. OUTLET)	9.3-10
3CS-1-079-2501R-2	RB		X	CHARGING LINE	9.3-10
3CS-1-081-2501R-2	RB		X	CHARGING LINE	9.3-10
3CS-1-078-2501R-2	RB		X	ALTERNATE CHARGING LINE (NOTE 7)	9.3-10
3CS-1-0802501R-2	RB	X		ALTERNATE CHARGING LINE (NOTE 8)	9.3-10
2CS-1-111-2501R-2	RB		X	PRESSURIZER AUX SPRAY LINE	9.3-10
2CS-1-112-2501R-2	RB	X		PRESSURIZER AUX SPRAY LINE (NOTES 4, 6)	9.3-10
2CS-1-087-2501R-2	AB SB	X		RCP SEAL WATER INJECTION	9.3-10
2CS-1-088-2501R-2	AB	X		SEAL WATER INJECTION FILTER	9.3-10
2CS-1-089-2501R-2	AB	X		SEAL WATER INJECTION FILTER OUTLET	9.3-10
2CS-1-090-2501R-2	AB SB	X		SEAL INJECTION SUPPLY HEADER	9.3-10
2CS-1-092-2501R-2	SB	X		SEAL INJECTION LINE RCP, LOOP 1 (NOTE 1)	9.3-10
2CS-1-093-2501R-2	SB	X		SEAL INJECTION LINE RCP, LOOP 1	9.3-10
2CS-1-102-2501R-2	RB	X		SEAL INJECTION LINE RCP, LOOP 1	9.3-10

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 8 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
2CS-1-103-2501R-1	RB	X		SEAL INJECTION LINE RCP, LOOP 1	9.3-10
		(A JET)	(B JET)		
1 1/2CS-1-248-2501R-1	RB	X	X	SEAL INJECTION RCP, LOOP 1 (NOTE 5)	9.3-10
2CS-1-095-2501R-2	RB	X		SEAL INJECTION LINE RCP, LOOP 2 (NOTE 1)	9.3-10
2CS-1-104-2501R-2	RB	X		SEAL INJECTION RCP, LOOP 2	9.3-10
2CS-1-105-2501R-2	RB	X		SEAL INJECTION RCP, LOOP 2	9.3-10
		(A JET)	(B JET)		
1 1/2CS-1-249-2501R-2	RB	X	X	SEAL INJECTION RCP, LOOP 2 (NOTE 5)	9.3-10
2CS-1-096-2501R-2	SB	X		SEAL INJECTION RCP, LOOP 3	9.3-10
2CS-1-098-2501R-2	RB SB	X		SEAL INJECTION RCP, LOOP 3 (NOTE 1)	9.3-10
2CS-1-106-2501R-2	RB	X		SEAL INJECTION RCP, LOOP 3	9.3-10
2CS-1-107-2501R-1	RB	X		SEAL INJECTION RCP, LOOP 3	9.3-10
		(A JET)	(B JET)		
1 1/2 RCS-1-243-2501R-2	RB	X	X	SEAL INJECTION RCP LOOP 3 (NOTE 5)	9.3-10
2CS-1-099-2501R-2	SB	X		SEAL INJECTION LINE RCP, LOOP 4	9.3-10
2CS-1-101-2501R-2	SB	X		SEAL INJECTION RCP LOOP 4 (NOTE 1)	9.3-10
2CS-1-108-2501R-1	RB	X		SEAL INJECTION RCP, LOOP 4	9.3-10
2CS-1-109-2501R-1	RB	X		SEAL INJECTION RCP, LOOP 4	9.3-10

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 9 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21)		REMARKS	FLOW DIAGRAM FIGURE NO.
		SUB COOLED (A JET)	2 PHASE FLOW (B JET)		
1 1/2CS-244-2501R-1	RB	X	X	SEAL INJECTION RCP LOOP 4 (NOTE 5)	9.3-10
2CS-X-008-152-5	AB		X	BORIC ACID BATCHING TANK	9.3-10
2CS-1-165-2501R-2	RB	X		EXCESS LETDOWN HX OUTLET	9.3-10
3CS-1-628-2501R-2	AB	X		CHARGING PUMP A BYPASS	9.3-10
3CS-1-901-2501R-2	SB	X		LETDOWN LINE	9.3-10
3CS-1-902-2501R-2	SB		X	TCV BYPASS	9.3-10
3CS-1-903-2501R-2	SB	X		LETDOWN LINE	9.3-10
3CS-1-904-601R-2	SB		X	LETDOWN LINE	9.3-10
3CS-1-906-601R-2	SB		X	LETDOWN LINE	9.3-10
3CS-1-907-601R-2	SB		X	LETDOWN LINE	9.3-10
3CS-1-908-2501R-2	SB		X	LETDOWN LINE	9.3-10
3CS-1-910-2501R-2	SB		X	LETDOWN REHEAT HX INLET	9.3-10
3CS-1-911-601R-2	SB		X	LETDOWN REHEAT HX INLET	9.3-10
3CS-1-912-2501R-2	SB		X	LETDOWN REHEAT HX OUTLET	9.3-10
3CS-1-913-601R	SB		X	LETDOWN REHEAT HX OUTLET	9.3-10
3CS-1-922-601R-2	SB	X		LETDOWN REHEAT HX OUTLET	9.3-10
3CS-2-921-601R-2 (UNIT 2)	SB	X		LETDOWN REHEAT HX OUTLET	9.3-10
2CS-X-09-152-5	AB		X	AUX. STEAM TO BORIC ACID BATCHING TANK	9.3-10

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 10 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
2CS-1-579-2501R-2	RB		X	LETDOWN HX INLET	9.3-10
2CS-1-945-2501R-2	AB		X	CENTRIFUGAL CHARGING PUMP MINIMUM FLOW BYPASS (NOTE 17)	9.3-10
2CS-2-946-2501R-2 (UNIT 2)	AB	X		CENTRIFUGAL CHARGING PUMP MINIMUM FLOW BYPASS (NOTE 17)	9.3-10
2CS-1-946-2501R-2	AB	X		CENTRIFUGAL CHARGING PUMP MINIMUM FLOW BYPASS (NOTE 18)	9.3-10
2CS-2-947-25014-2 (UNIT 2)	AB	X		CENTRIFUGAL CHARGING PUMP MINIMUM FLOW BYPASS (NOTE 18)	9.3-10
SYSTEM FEEDWATER SYSTEM (FW)					
3CS-1-629-2501R-2	AB	X		CHARGING PUMP B BYPASS	9.3-10
3CS-1-960-601R-2	RB		X	LETDOWN LINE RV LINE	9.3-10
18FW-1-13-2002-5	SB (Roof)		X	SG NUMBER 3 FEED LINE	10.4-9
18FW-1-34-2003-2 AND 2" DR. CONN. TO 1FW-230	SB		X	SG NUMBER 3 FEED LINE (NOTE 1)	10.4-0
18FW-1-17-1303-2 AND 2" CONN TO VALVES 1FW-111 (DRAIN) 1FW-127 (VENT)	RB, SB SB RB	X		SG NUMBER 3 FEED LINE (NOTE 1)	10.4-9
16FW-1-73-1303-2	RB		X	SG NUMBER 3	10.4-9
8FW-1-54-2002-5	SB (Roof)		X	SG NUMBER 3 FEED FLOW CONTROL VALVE BYPASS CONNECTION	10.4-9
8FW-1-42-2002-5 AND 2" CONN TO VENT VALVE 1FW-067	SB (Roof)		X	CONTROL VALVE BYPASS	10.4-9

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 11 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
3FW-1-105-2003-2	SB		X	SG NUMBER 3 FWIV BYPASS (NOTE 1)	10.4-9
3FW-1-114-1303-2	SB		X	(NOTE 1)	
6FW-2-93-203-2	SB		X	SG NUMBER 3 BYPASS LINE (NOTES 1 and 25)	10.4-9
6FW-1-97-1303-2 AND 2" LINES TO VALVES 1FW-247 (VENT) 1FW-264 (DRAIN) 1FW-265 (DRAIN)	RB, SB SB RB RB		X	SG NUMBER 3 PREHEATER BYPASS LINE (NOTE 26)	10.4-9
1 1/2 FW-1-30-1303-2	SB		X	CONNECTION FROM PREHEATER BYPASS (NOTES 1 and 26)	10.4-9
6FW-2-101-1303-2	RB		X	FW SPLIT FLOW BYPASS FOR SG NUMBER 3 (NOTE 25)	10.4-9
18FW-1-14-2002-5	SB (Roof)		X	SG NUMBER 2 FEED LINE	10.4-9
18FW-1-35-2003-2 AND 2" DR. CONN. TO IFW-228	SB		X	SG NUMBER 2 FEED LINE	10.4-9
18FW-1-18-1303-2 AND 2" CONNS TO VALVES 1FW-114 (DRAIN) 1FW-129 (VENT)	SB, RB SB RB		X	SG NUMBER 2 FEED LINE	10.4-9
16FW-1-74-1303-2	RB		X	SG NUMBER 2 FEED LINE	10.4-9
8FW-1-55-2002-5	SB (Roof)	X		SG NUMBER 2 FEED FLOW CONTROL VALVE BYPASS	10.4-9
8FW-1-43-2002-5 AND 2" CONN TO VENT VALVE 1FW-073	SB (Roof)		X	SG NUMBER 2 FEED FLOW CONTROL VALVE BYPASS	10.4-9
3FW-1-104-2003-2	SB		X	SG NUMBER 2 FWIV BYPASS (NOTE 2)	10.4-9
2FW-1-113-1303-2	SB		X	SG NUMBER FWIV BYPASS (NOTE 2)	10.4-9

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 12 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
6FW-2-92-2003-2	SB		X	SG NUMBER 2 PREHEATER BYPASS LINE (NOTES 1 and 25)	10.4-9
6FW-1-96-1303-2 AND 2" LINES TO	RB, SB		X	SG NUMBER 2 PREHEATER BYPASS LINE	10.4-9
1FW-246 (VENT)	SB			(NOTES 1 and 26)	10.4-9
1FW-261 (DRAIN)	RB				10.4-9
1FW-262 (DRAIN)	RB				10.4-9
1 1/2FW-1-31-1303-2	SB		X	CONNECTION FROM PREHEATER BYPASS LINE (NOTES 1 and 26)	10.4-9
6FW-2-100-1303-2	RB		X	FW SPLIT FLOW BYPASS FOR SG NUMBER 2 (NOTE 25)	10.4-9
18FW-1-15-2002-5	SB (Roof)		X	SG NUMBER 1 FEED LINE	10.4-9
18FW-1-36-2003-2 AND 2" DR CONN TO 1FW-220	SB		X	SG NUMBER 1 FEED LINE (NOTE 1)	10.4-9
18FW-1-19-1303-2 AND 2" CONN TO VALVES	RB, SB		X	SG NUMBER 1 FEED LINE (NOTE 1)	10.4-9
1FW-117 (VENT)	SB				
16FW-1-75-1303-2	RB		X	SG NUMBER 1 FEED LINE	10.4-9
8FW-1-56-2002-5	SB (Roof)		X	SG NUMBER 1 FEED FLOW CONTROL VALVE BYPASS CONNECTION	10.4-9
8FW-1-44-2002-5 AND 2" CONN TO VENT	SB (Roof)		X	SG NUMBER 1 FEED FLOW CONTROL VALVE BYPASS	10.4-9
VALVE 1FW-079	SB			CONNECTION	
3FW-1-103-2003-2	SB		X	SG NUMBER 1 FWIV BYPASS (NOTE 1)	10.4-9
3FW-1-112-1303-2	SB		X	SG NUMBER 1 FWIV BYPASS (NOTE 1)	10.4-9
6FW-2-91-2003-2	SB		X	SG NUMBER 1 PREHEATER BYPASS LINE (NOTES 1 and 25)	10.4-0

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 13 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
6FW-1-95-1303-2 AND 2" LINES TO VALVES 1FW-245 (VENT) 1FW-258 (DRAIN) 1FW-259 (DRAIN)	RB SB SB RB RB		XX	SG NUMBER PREHEATER BYPASS LINE (NOTES 1 and 25)	10.4-9
1 1/2FW-1-32-1303-2	SB		X	CONNECTION FROM CHEM PREHEATER BYPASS LINE (NOTES 1 and 26)	10.4-9
6FW-2-99-1303-2	RB		X	SPLIT FLOW BYPASS LINE FOR SG NUMBER 1 (NOTE 25)	10.4-9
18FW-1-16-2002-5	SB (Roof)		X	SG NUMBER 4 FEED LINE	10.4-9
18FW-1-37-2003-2 AND 2" DR CONN TO 1FW-226	SB		X	SG NUMBER 4 FEED LINE (NOTE 1)	10.4-9
18FW-1-20-1303-2 AND 2" CONN TO VALVES 1FW-120 (VENT)	RB, SB SB		X	SG NUMBER FEED LINE (NOTE 1)	10.4-9
16FW-1-76-1303-2	RB		X	SG NUMBER 4 FEED LINE	10.4-9
8FW-1-57-2002-5	SB (Roof)		X	SG NUMBER 4 FEED FLOW CONTROL VALVE BYPASS	10.4-9
8FW-1-45-2002-5 AND 2" DR. CONN TO VALVE 1FW-185	SB (Roof) SB		X	SG NUMBER 4 FEED FLOW CONTROL VALVE BYPASS	10.4-9
3FW-1-106-2003-2	SB		X	SG NUMBER 4 FWIV BYPASS (NOTE 1)	10.4-9
3FW-1-111-1303-2 BYPASS (NOTE 1)	SB		X	SG NUMBER 4 FWIV	10.4-9
6FW-2-94-2003-2 LINE (NOTE 1)	SB		X	SG NUMBER 4 PREHEATER BYPASS (NOTE 25)	10.4-9

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 14 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
6FW-2-98-1303-2 AND 2" LINES TO VALVES 1FW-248 (VENT) 1FW-255 (DRAIN) 1FW-256 (DRAIN)	RB, SB SB RB RB		X	(NOTES 1 and 26)	10.4-9
1 1/2FW-1-33-1303-2	SB		X	CONNECTION FROM PREHEATER BYPASS LINE (NOTES 1 and 26)	10.4-9
6FW-2-102-1303-2	RB		X	FW SPLIT FLOW BYPASS FOR SG NUMBER 4 (Unit 25)	10.4-9
30FW-1-12-2002-G	TB		X	FEEDWATER HEATERS OUTLET HEADER	10.4-9
30FW-1-011-2002G	TB		X	FEEDWATER HEADERS OUTLET	10.4-9
2FW-2-905-1303-2	RB		X	CHEMICAL FEED SG NUMBER 3	10.4-9
2FW-2-906-1303-2	RB		X	CHEMICAL FEED SG NUMBER 4	10.4-9
2FW-2-907-1303-2	RB		X	CHEMICAL FEED SG NUMBER 2	10.4-9
2FW-2-900-1303-2	RB		X	CHEMICAL FEED SG NUMBER 2	10.4-9
2FW-2-901-1303-2	RB		X	CHEMICAL FEED SG NUMBER 4	10.4-9
2FW-2-902-1303-2	RB		X	CHEMICAL FEED SG NUMBER 1	10.4-9
2FW-2-903-1303-2	RB		X	CHEMICAL FEED SG NUMBER 1	10.4-9
2FW-2-904-1303-2	RB		X	CHEMICAL FEED SG NUMBER 3	10.4-9
SYSTEM MAIN STEAM (MS)					
32MS-1-01-1303-2 AND 2" V. CON TO VALVE 1MS-615	RB SB		X	MAIN STEAM PIPES & INCLUDING SSV CONNECTIONS (NOTE 1)	10.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 15 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
32MS-1-02-1303-2 AND 2" V. CON TO VALVE 1MS-614	RB SB		X	MAIN STEAM PIPES & INCLUDING SSV CONNECTIONS (NOTE 1)	10.3-1
32MS-1-03-1303-2 AND 2" V. CON TO VALVE 1MS-616	RB SB		X	MAIN STEAM PIPES & INCLUDING SSG CONNECTIONS (NOTE 1)	10.3-1
32MS-1-04-1303-2 AND 2" V CON TO VALVE 1MS-617	RB SB		X	MAIN STEAM PIPES & INCLUDING SSV CONNECTIONS (NOTE 1)	10.3-1
2MS-1-05-1303-2	SB		X	MAIN STEAM LINE DRAIN POT DRAIN LINES (NOTE 1)	10.3-1
2MS-1-166-1302-5	SB		X	MAIN STEAM LINE DRAIN POT DRAIN LINES (NOTE 1) (NOTE 24)	10.3-1
2MS-1-10-1303-2 POT DRAIN LINES (NOTE 1)	SB		X	MAIN STEAM LINE DRAIN	10.3-1
2MS-1-164-1302-5	SB		X	MAIN STEAM LINE DRAIN POT DRAIN LINES (NOTE 24)	10.3-1
2MS-1-15-1303-2	SB		X	MAIN STEAM LINE DRAIN POT DRAIN LINES (NOTE 1)	10.3-1
2MS-1-162-1302-5	SB		X	MAIN STEAM LINE DRAIN POT DRAIN LINES (NOTE 24)	10.3-1
2MS-1-20-1303-2	SB		X	MAIN STEAM LINE DRAIN POT DRAIN LINES (NOTE 1)	10.3-1
2MS-1-168-1302-5	SB		X	MAIN STEAM LINE DRAIN POT DRAIN LINES (NOTE 24)	10.3-1
34MS-1-07-1302-5 34MS-1-07-1302-G	SB (Roof) TB		X	MAIN STEAM PIPES DOWN STREAM OF MSIV'S	10.3
34MS-1-12-1302-5 34MS-1-12-1302-G	SB (Roof) TB		X	MAIN STEAM PIPES DOWNSTREAM OF MSIVs	10.3-1
34MS-1-17-1302-5 34MS-1-17-1302-G	SB (Roof) TB		X	MAIN STEAM PIPES DOWNSTREAM OF MSIVs	10.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 16 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
34MS-1-22-1302-5 34MS-1-22-1302-G	SB (Roof) TB		X	MAIN STEAM PIPES DOWNSTREAM OF MSIVs	10.3-1
4MS-1-25-1303-2	SB		X	MS TO AF PUMP TURBINE (NOTE 1)	10.3-1
4MS-1-26-1303-2	SB		X	MS TO AF PUMP TURBINE (NOTE 1)	10.3-1
8MS-1-223-1303-2	SB		X	ATMOSPHERIC STEAM DUMPS (NOTE 1)	10.3-1
8MS-1-240-1303-2	SB		X	ATMOSPHERIC STEAM DUMPS (NOTE 1)	10.3-1
8MS-1-257-1303-2	SB		X	ATMOSPHERIC STEAM DUMPS (NOTE 1)	10.3-1
8MS-1-274-1303-2	SB		X	ATMOSPHERIC STEAM DUMPS (NOTE 1)	10.3-1
2MS-1-206-1303-2	RB		X	SG NUMBER 1 BLOWDOWN SYSTEM	10.3-1
3MS-1-74-1303-2	RB SB		X	SG NUMBER 1 BLOWDOWN SYSTEM (NOTE 1)	10.3-1
2MS-1-203-1303-2	RB		X	SG NUMBER 1 BLOWDOWN SYSTEM	10.3-1
2MS-1-341-1303-2	RB		X	SG NUMBER 1 BLOWDOWN SYSTEM	10.3-1
3MS-1-342-1303-2	RB		X	SG NUMBER 1 BLOWDOWN SYSTEM	10.3-1
3MS-78-1302-5	SB		X	SG NUMBER 1 BLOWDOWN SYSTEM	10.3-1
2MS-199-1303-2	RB		X	SG NUMBER 2 BLOWDOWN SYSTEM	10.3-1
3MS-1-75-1303-2	RB SB		X	SG NUMBER 2 BLOWDOWN SYSTEM (NOTE 1)	10.3-1
4MS-1-151-1303-2	RB		X	SG NUMBER 2 BLOWDOWN SYSTEM	10.3-1
2MS-1-195-1303-2	RB		X	SG NUMBER 2 BLOWDOWN SYSTEM	10.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 17 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
2MS-1-339-1303-2	RB		X	SG NUMBER 2 BLOWDOWN SYSTEM	10.3-1
4MS-1-340-1303-2	RB		X	SG NUMBER 2 BLOWDOWN SYSTEM	10.3-1
3MS-1-79-1302-5	SB		X	SG NUMBER 2 BLOWDOWN SYSTEM	10.3-1
2MS-1-210-1303-2	RB		X	SG NUMBER 3 BLOWDOWN SYSTEM	10.3-1
4MS-1-150-1303-2	RB		X	SG NUMBER 3 BLOWDOWN SYSTEM	10.3-1
3MS-1-76-1303-2	RB SB		X	SG NUMBER 3 BLOWDOWN SYSTEM (NOTE 1)	10.3-1
3MS-1-80-1302-5	SB		X	SG NUMBER 3 BLOWDOWN SYSTEM	10.3-1
2MS-1-213-1303-2	RB		X	SG NUMBER 3 BLOWDOWN SYSTEM	10.3-1
2MS-1-343-1303-2	RB		X	SG NUMBER 3 BLOWDOWN SYSTEM	10.3-1
4MS-1-344-1303-2	RB		X	SG NUMBER 3 BLOWDOWN SYSTEM	10.3-1
2MS-1-218-1303-2	RB		X	SG NUMBER 4 BLOWDOWN SYSTEM	10.3-1
3MS-1-73-1303-2	RB SB		X	SG NUMBER 4 BLOWDOWN SYSTEM (NOTE 1)	10.3-1
3MS-1-77-1302-5	SB		X	SG NUMBER 4 BLOWDOWN SYSTEM	10.3-1
2MS-1-221-1303-2	RB		X	SG NUMBER 4 BLOWDOWN SYSTEM	10.3-1
2-MS-1-346-1303-2	RB		X	SG NUMBER 4 BLOWDOWN SYSTEM	10.3-1
3-MS-1-345-1303-2	RB		X	SG NUMBER 4 BLOWDOWN SYSTEM	10.3-1
2MS-1-163-1302-5	SB		X	MAIN STEAM DRAIN POT DR. LINES, DOWN STREAM OF FLOW CONTROL ORIFICES (Note 24)	10.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 18 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
2MS-1-165-1302-5	SB		X	MAIN STEAM DRAIN POT DR. LINES, DOWN STREAM OF FLOW CONTROL ORIFICES (Note 24)	10.3-1
2MS-1-167-1302-5	SB		X	MAIN STEAM DRAIN POT DR. LINES, DOWN STREAM OF FLOW CONTROL ORIFICES (Note 24)	10.3-1
2MS-1-169-1302-5	SB		X	MAIN STEAM DRAIN POT DR. LINES, DOWN STREAM OF FLOW CONTROL ORIFICES (Note 24)	10.3-1
4MS-1-27-1302-3	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
4MS-1-28-1302-3	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
4MS-1-29-1302-3	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
8MS-1-415-152-3	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
16MS-1-416-152-3	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
10MS-1-425-152-3	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
1 1/2MS-1-419-152-3	SB	X		(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
1 1/2MS-1-420-152-3	SB	X		(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
4-MS-1-437-152-3	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
2-MS-1-438-152-3	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
2-MS-1-439-152-5	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
2-MS-1-440-152-5	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
2-MS-1-441-151-5	SB		X	(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
6-MS-1-442-152-5	SB		X	(NOTE 3) MS TO AF PMP TURBINE	10.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 19 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
4MS-1-914-152-3 4MS-2-907-152-3 (U2)	SB	X		(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
2MS-1-915-152-3 2MS-2-908-152-3 (U2)	SB	X		(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
2MS-916-152-3	SB	X		(NOTE 3) MS TO AF PUMP TURBINE	10.3-1
2MS-1-918-1303-2	RB		X	CHEM. FD TO BLOWDOWN LINE	10.3-1
2MS-1-919-1303-2	RB		X	CHEM. FD TO BLOWDOWN LINE	10.3-1
2MS-1-920-1303-2	RB		X	CHEM. FD TO BLOWDOWN LINE	10.3-1
2MS-1-921-1303-2	RB		X	CHEM. TD TO BLOWDOWN LINE	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-03 TO SAFETY RELIEF VALVE 1MS-0093 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-03 TO SAFETY RELIEF VALVE 1MS-0094 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-03 TO SAFETY RELIEF VALVE 1MS-0095	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-03 TO SAFETY RELIEF VALVE 1MS-0096 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-03 TO SAFETY RELIEF VALVE 1MS-0097 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-02 TO SAFETY RELIEF VALVE 1MS-58 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-02 TO SAFETY RELIEF VALVE 1MS-59 (NOTE 1)	10.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 20 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-02 TO SAFETY RELIEF VALVE 1MS-60 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-02 TO SAFETY RELIEF VALVE 1MS-61 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-02 TO SAFETY RELIEF VALVE 1MS-62 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-02 TO SAFETY RELIEF VALVE 1MS-21 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-01 TO SAFETY RELIEF VALVE 1MS-22 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-01 TO SAFETY RELIEF VALVE 1MS-23 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-01 TO SAFETY RELIEF VALVE 1MS-24 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-01 TO SAFETY RELIEF VALVE 1MS-25 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-04 TO SAFETY RELIEF VALVE 1MS-129 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-04 TO SAFETY RELIEF VALVE 1MS-130	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-04 TO SAFETY RELIEF VALVE 1MS-131 (NOTE 1)	10.3-1
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-04 TO SAFETY RELIEF VALVE 1MS-132 (NOTE 1)	10.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 21 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
6-2003-2	SB		X	6 INCH LINE FROM 32-MS-1-04 TO SAFETY RELIEF VALVE 1MS-133 (NOTE 1)	10.3-1
SYSTEM REACTOR COOLANT (RC)					
29RC-1-001-2501R-1	RB		X	HOT LEG	5.1-1
31RC-1-002-2501R-1	RB		X	CROSS OVER	5.1-1
27.5RC-1-003-2501R-1	RB		X	COLD LEG	5.1-1
12RC-1-007-2501R-1	RB		X	RHR TRAIN A CONNECTION	5.1-1
6RC-1-008-2501R-1	RB		X	SIS INJ. PUMP CONNECTION	5.1-1
2RC-1-015-2501R-1	RB		X	TO WPS (L) VIA RCDT	5.1-1
4RC-1-018-2501R-1	RB		X	TO PRESSURIZER SPRAY LINE	5.1-1
3RC-1-019-2501R-1	RB		X	ALTERNATE CH. LINE	5.1-1
10RC-1-021-2501R-1	RB		X	SIS ACCUMULATOR & RHR INJECTION	5.1-1
1 1/2 RC-1-020-2501-1	RB		X	SIS BIT CONN. RC LOOP #01	5.1-1
29RC-1-058-2501R-1	RB		X	HOT LEG	5.1-1
31RC-1-059-2501R-1	RB		X	CROSS OVER	5.1-1
27.5RC-1-060-2501R-1	RB		X	COLD LEG	5.1-1
12RC-1-069-2501R-1	RB		X	RHR TRAIN B CONN.	5.1-1
6RC-1-070-2501R-1	RB		X	SIS-INJ. PUMP CONN.	5.1-1
14RC-1-135-25O1R-1	RB		X	PRESSURIZER SURGE LINE	5.1-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 22 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
4RC-1-075-2501R-1	RB		X	TO PRESSURIZER SPRAY LINE	5.1-1
3RC-1-076-2501R-1	RB		X	CHARGING LINE	5.1-1
10RC-1-078-2501R-1	RB		X	SIS ACCUMULATOR RHRS INJ	5.1-1
1 1/2RC-1-079-2501R-1	RB		X	SIS BIT CONN.	5.1-1
2RC-1-072-2501R-1	RB		X	TO WPS (L) VIA RDT	5.1-1
29RC-1-040-2501R-1	RB		X	HOT LEG	5.1-1
31RC-1-041-2501R-1	RB		X	CROSS OVER	5.1-1
27.5RC-1-042-2501R-1	RB		X	COLD LEG	5.1-1
6RC-1-029-2501R-1	RB		X	SIS INJ. PUMP CONN.	5.1-1
2RC-1-035-2501R-1	RB		X	TO WPS (L) VIA RCDT	5.1-1
10RC-1-037-2501R-1	RB		X	SIS ACCUMULATOR & RHRS INJ.	5.1-1
1 1/2RC-1-039-2501R-1	RB		X	SIS BIT CONN. RC LOOP #02	
29RC-1-023-2501R-1	RB		X	HOT LEG	5.1-1
31RC-1-024-2501R-1	RB		X	CROSS OVER	5.1-1
27.5RC-1-025-2501R-1	RB		X	COLD LEG	5.1-1
3RC-1-052-2501R-1	RB		X	RCS LETDOWN LINE	5.1-1
2RC-053-2501R-1	RB		X	TO WPS (L) VIA RCDT	5.1-1
6RC-1-046-2501R-1	RB		X	SIS INJ. PUMP CONN.	5.1-1
1 1/2RC-1-057-2501R-1	RB		X	SIS BIT CONN.	5.1-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 23 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
10RC-1-055-2501R-1	RB		X	SIS ACCUMULATOR & RHRS INJ.	5.1-1
4RC-1-087-2501R-1	RB		X	PRESSURIZER SPRAY LINE FROM RC LOOP 1	5.1-1
4RC-1-088-2501R-1	RB		X	PRESSURIZER SPRAY LINE FROM RC LOOP 4	5.1-1
6RC-1-147-2501R-1	RB		X	PRESSURIZER SPRAY LINE	5.1-1
4RC-1-091-2501R-1	RB		X	PRESSURIZER SPRAY LINE	5.1-1
2RC-1-132-2501R-1	RB		X	AUX. SPRAY LINE FROM CVCS	5.1-1
6RC-1-096-2501R-1	RB		X	PRESSURIZER SAFETY LINES	5.1-1
6RC-1-098-2501R-1	RB		X	PRESSURIZER SAFETY LINES	5.1-1
6RC-1-100-2501R-1	RB		X	PRESSURIZER SAFETY LINES	5.1-1
6RC-1-108-2502R-1	RB		X	PRESSURIZER RELIEF LINE	5.1-1
3RC-1-146-2501R-1	RB		X	RELIEF VALVE INLET	5.1-1
3RC-1-111-2501R-1	RB		X	RELIEF VALVE INLET	5.1-1
6RC-1-161-2501R-1	RB		X	PRESSURIZER SPRAY	5.1-1
4RC-1-162-2501R-1	RB		X	PRESSURIZER SPRAY	5.1-1
6RC-1-163-2501R-1	RB		X	PRESSURIZER SPRAY	5.1-1
4RC-1-164-2501R-1	RB		X	PRESSURIZER SPRAY	5.1-1
SYSTEM RESIDUAL HEAT REMOVAL (RHR)					
12-RH-1-001-2501R-1	RB	X		FROM RCS HL LOOP 4 (NOTE 16)	5.4-6
12-RH-1-002-2501R-1	RB	X		FROM RCS HL LOOP 4 (NOTE 16)	5.4-6

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 24 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
SYSTEM AUXILIARY STEAM (SA)					
6SA-X-18-152-5	AB EC TB		X	BORON RECYCLE EVAPORATOR AND BORIC ACID BATCHING TANK SUPPLY HEADER	10.4-16
2SA-X-63-152-5	AB		X	BORIC ACID BATCHING TANK STEAM SUPPLY	10.4-16
2SA-X-54-152-5	AB		X	DEMIN. WATER “FLUSHING CONN.” (NOTE 9)	10.4-16
4SA-X-57-152-5	AB		X	BRS RECYCLE EVAPORATOR SUPPLY	10.4-16
4SA-X-58-152-5	AB		X	BRS RECYCLE EVAPORATOR FEED PREHEATER STEAM	10.4-16
2SA-X-59-152-5	AB		X	BRS RECYCLE EVAPORATOR FEED PREHEATER STEAM	10.4-16
1-1/2SA-X-84-152-5	AB		X	BRS RECYCLE EVAPORATOR CONDENSATE	10.4-16
1-1/2SA-X-85-152-5	AB		X	BORON RECYCLE EVAPORATOR CONDENSATE TRIP BYPASS	10.4-16
2SA-X-48-152-5	AB		X	BRS RECYCLE EVAPORATOR CONDENSATE	10.4-16
8SA-X-115-152-5	AB		X	AUXILIARY STEAM CONDENSATE COOLER INLET HEADER	10.4-16
2SA-X-93-152-5	AB		X	BORIC ACID BATCHING TANK CONDENSATE	10.4-16
2SA-X-120-152-5	AB		X	BORIC ACID BATCHING TANK CONDENSATE TRIP BYPASS	10.4-16
10SA-X-19-152-5	AB EC TB		X	FLOOR DRAIN WASTE EVAPORATOR AND WPS WASTE EVAPORATOR STEAM SUPPLY HEADER	10.4-16

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 25 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
2SA-X-38-152-5	AB		X	DEMIN. WATER FLUSHING CONN. (NOTE 9)	10.4-16
6SA-X-39-152-5	AB		X	FLOOR DRAIN WASTE EVAPORATOR STEAM SUPPLY	10.4-16
4SA-X-40-152-5	AB		X	FLOOR DRAIN WASTE EVAPORATOR STEAM SUPPLY	10.4-16
2SA-X-42-152-5	AB		X	FLOOR DRAIN WASTE EVAPORATOR FEED PREHEATER STEAM SUPPLY	10.4-16
1-1/2SA-X-65-152-5	AB		X	FLOOR DRAIN WASTE EVAPORATOR CONDENSATE	10.4-16
1-1/2SA-X-66-152-5	AB		X	FLOOR DRAIN WASTE EVAPORATOR CONDENSATE STRN. BYPASS	10.4-16
2SA-X-52-152-5	AB		X	FLOOR DRAIN WASTE EVAP. CONDENSATE	10.4-16
6SA-X-46-152-5	AB		X	WPS WASTE EVAPORATOR STEAM SUPPLY	10.4-16
2SA-X-53-152-5	AB		X	DEMIN. WATER FLUSHING CONN. (NOTE 9)	10.4-16
4SA-X-47-152-5	AB		X	WPS WASTE EVAPORATOR STEAM SUPPLY	10.4-16
4SA-X-41-152-5	AB		X	WPS WASTE EVAPORATOR FEED PREHEATER STEAM SUPPLY	10.4-16
2SA-X-49-152-5	AB	X		WPS WASTE EVAPORATOR FEED PREHEATER STEAM SUPPLY	10.4-16
1-1/2SA-X-74-152-5	AB		X	WPS WASTE EVAPORATOR CONDENSATE	10.4-16
1-1/2SA-X-75-152-5	AB		X	WPS WASTE EVAP. CONDENSATE STRN. BYPASS	10.4-16
2SA-X-50-152-5	AB		X	WPS WASTE EVAP CONDENSATE	10.4-16
2SA-X-129-152-5	EC TB		X	DRAIN POT X-SA-7 TO MAIN CONDENSER	10.4-16

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 26 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
2SA-X-130-152-5	EC		X	DRAIN POT X-SA-7 TO MAIN CONDENSER	10.4-16
6SA-X-906-152-5	AB		X	BRS RECYCLE EVAPORATOR SUPPLY	10.4-16
SYSTEM STEAM GENERATOR BLOWDOWN (SB)					
3SB-1-01-1302-5	SB		X	STEAM GENERATOR BLOWDOWN SG. 01	10.4-10
3SB-1-03-1302	SB		X	STEAM GENERATOR BLOWDOWN, SG. 02	10.4-10
3SB-1-09-1302-5	SB		X	STEAM GENERATOR BLOWDOWN SG. 03	10.4-10
3SB-1-18-1302-5	SB		X	STEAM GENERATOR BLOWDOWN, SG. 04	10.4-10
8SB-1-60-1302-5	SB TB EC		X	STEAM GENERATOR BLOWDOWN, HEADER TO SG. BLOWDOEN HT. EXCHANGE CP1-SBAHSB-01	10.4-10
2SB-1-72-1302-5	SB		X	N2 GAS SUPPLY CONNECTION (NOTE 10)	10.4-10
8SB-1-25-1302-5	EC		X	SG. HT. EXCH. BYPASS	10.4-10
8SB-1-48-1302-5	EC	X		SG. HT. EXCH. DISCHARGE	10.4-10
3SB-1-109-1302-5	EC	X		SG. HT. EXCH. DISCHARGE	10.4-10
3SB-1-110-1302-5	EC	X		SG. HT. EXCH. DISCHARGE	10.4-10
8SB-1-37-302-5	EC AB	X		SG. HT. EXCH. DISCHARGE	10.4-10
3SB-1-40-302-5	EC	X		RELIEF VALVE CONNECTION	10.4-10
8SB-1-68-301-5	AB	X		SG. BLOWDOWN FILTER HEADER	10.4-10
6SB-1-101-301-5	AB	X		SG. BLOWDOWN FILTER CP1-SBFLSB-01 INLET CONNECTION	10.4-10

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 27 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
6SB-1-52-301-5	AB	X		FILTER BYPASS (NOTE 11)	10.4-10
6SB-1-24-301-5	AB	X		SG. BLOWDOWN FILTER CP1-SBFLSB-02 INLET CONNECTION	10.4-10
8SB-1-30-302-5	EC		X	SG. BLOWDOWN HX COND. OUTLET	10.4-10
SYSTEM SAFETY INJECTION SYSTEM (SI)					
10SI-1-119-2501R-2	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 1	6.3-1
10SI-1-103-2501R-2	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 1 (NOTE 12)	6.3-1
10SI-1-179-2501R-2	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 1 (NOTE 12)	6.3-1
2SI-1-123-601R-2	RB	X		TO WPS VIA RCDT PUMPS (NOTE 15)	6.3-1
2" LEVEL TRANSMITTER PIPING	RB	X		LEVEL TRANSMITTER 1-LT-950 AND 1-LT-951 ON ACCUMULATOR INJ. TANK 1 (NOTE 13)	6.3-1
10SI-1-120-601R-2	RB	X		ACCUMULATOR INJ. TANK 2 TO RCS COLD LEG LOOP 2	6.3-1
10SI-1-104-2501R-2	RB	X		ACCUMULATOR INJ. TANK 2 TO RCS COLD LEG LOOP 2 (NOTE 12)	6.3-1
10SI-1-180-2501R-1	RB	X		ACCUMULATOR INJ. TANK 2 TO RCS COLD LEG LOOP 2 (NOTE 12)	6.3-1
2SI-1-124-601R-2	RB	X		TO WPS VIA RCDT PUMPS (NOTE 15)	6.3-1
2" LEVEL TRANSMITTER PIPING	RB	X		LEVEL TRANSMITTER 1-LT-952 AND 1-LT-953 ON ACCUMULATOR INJ. TANK 2 (NOTE 13)	6.3-1
10SI-1-121-601R-2	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 3	6.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 28 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
10SI-1-105-2501R-2	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 3 (NOTE 12)	6.3-1
10SI-1-181-2501R-1	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 3 (NOTE 12)	6.3-1
2SI-1-125-601R-2	RB	X		TO WPS VIA RCDT PUMPS (NOTE 15)	6.3-1
2" LEVEL TRANSMITTER PIPING	RB	X		LEVEL TRANSMITTER 1-LT-954 AND 1-LT-955 ON ACCUMULATOR INJ. TANK 3 (NOTE 13)	6.3-1
10SI-1-122-601R-2	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 4	6.3-1
10SI-1-106-2501R-2	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 4 (NOTE 12)	6.3-1
10SI-1-182-2501R-1	RB	X		ACCUMULATOR INJ. TO RCS COLD LEG LOOP 4 (NOTE 12)	6.3-1
2SI-1-126-601R-2	RB	X		TO WPS VIA RCDT PUMPS (NOTE 15)	6.3-1
2" LEVEL TRANSMITTER	RB	X		LEVEL TRANSMITTER 1-LT-956 AND 1-LT-957 ON ACCUMULATOR INJ. TANK 4 (NOTE 13)	6.3-3
2SI-1-063-2501R-1	TB	X		SI TO CL LOOP (NOTE 14)	6.3-1
6SI-1-089-2501R-1	RB	X		SI/RHR TO CL LOOP 1 (NOTE 14)	6.3-1
2SI-1-065-2501R-1	RB	X		SI TO CL LOOP 2	6.3-1
6SI-1-328-2501R-1	RB	X		SI/RHR TO CL LOOP 2 (NOTE 14)	6.3-1
8SI-1-090-2501R-1	RB	X		SI/RHR TO CL LOOP 2 (NOTE 14)	6.3-1
2SI-1-067-2501R-1	RB	X		SI TO CL LOOP 3 (NOTE 14)	6.3-1

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 29 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.
6SI-1-330-2501R-1	RB	X		SI/RHR TO CL LOOP 3 (NOTE 14)	6.3-1
8SI-1-091-2501R-1	RB	X		SI/RHR TO CL LOOP 3 (NOTE 14)	6.3-1
2SI-1-069-2501R-1	RB	X		SI TO CL LOOP 4 (NOTE 14)	6.3-1
6SI-1-092-2501R-1	RB	X		SI/RHR TO CL LOOP 4 (NOTE 14)	6.3-1
4SI-1-001-2501R-1	AB	X		ECCS FROM CHG PUMPs	6.3-1
4SI-1-008-2501R-2	SB	X		ECCS FROM CHG PUMPs	6.3-1

NOTES TO THE HIGH ENERGY LINE LIST

LEGEND:

SB - SAFEGUARDS BUILDING

EC - ELECTRICAL AND CONTROLS BUILDING

RB - CONTAINMENT BUILDING

AB - AUXILIARY BUILDING

TB - TURBINE BUILDING

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 30 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.

1. This piping is within a Break Exclusion Area. This includes the piping outside containment from the containment penetration to the containment isolation valve and including the applicable moment restraint(s).

2. These Steam Turbine Driven AF Pump discharge lines are classified as moderate energy piping since operation as a high energy fluid system is for a limited period as prescribed by 3.6B.2.1.4. Breaks in these lines are not postulated because they are not operational during normal plant conditions and because of insufficient stored energy due to back leakage of any of the following flow check valves:

   1AF-078 1AF-098 1AF-086 1AF-106

   This leakage will place the lines under pressure back to valve 1AF-038 in line 8-AF-1-011-2002-3. This latter valve is assumed to hold.

3. These MS lines are classified as moderate energy piping since operation as a high energy fluid system is for a limited period as prescribed by 3.6B.2.1.4.

4. Not used.

5. This line is downstream of the last check valve (closest to the RCP). Therefore the back flow jet from the downstream side (B Jet) of any break (i.e. flow of RC water from the pump body) will be at 2192 psig & 559^{o}F. The flow from this source will be 2-phase flow.

6. This line (2-CS-1-112-2501R-1) is between a check valve (1CS-8377) and a flow control valve (1-8145), both normally closed during power operation. Pressurization of this line is assumed due to possible valve leakage. Temperature in this line, however, approaches ambient. Therefore, a break of this line will result in a subcooled (solid) jet of water.

7. This line (3-CS-1-078-2501R-2) is high energy up to normally closed valve 1-8147.

8 This line (3-CS-1-080-2501R-1) is between check valves 1-8379A and 1-8379B. Pressurization of this line is assumed due to possible valve leakage.

9. This flushing connection will be subject to SA operating line pressure and temperature of 50 psig and 297^{o}F during normal operation of the SA System up to the check valve.

10. Valve 1NG-026 (see 2323-M1-0243 Sheet A, Zone E-4) is closed and sealed during normal plant operation at power. Therefore, a break in the N2 gas supply line (2-SB-1-072-1302-5) would result in a high energy jet consisting of two phase back flow from the Steam Blowdown System header.

11. This line is considered to be High Energy up to normally closed valve 1SB-054.

12. Emergency supply of borated water from Accumulator Injection Tanks 1 through 4 to RCS Cold Leg Loops 1 through 4. It is assumed that check valves 1-8956 A,B,C, and D hold and 1-8948 A,B,C, and D leak.

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 31 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.

13. This line connects to nitrogen overpressure space in the accumulator tank.

14. To SIS Cold Legs Loops 1 through 4, it is assumed that check valves 1SI-8819 A,B,C, and D and 1SI-8818 A,B,C, and C, all hold.

15. These lines are assumed to be high energy only up to normally closed valves 1SI-8955 A<B<C< and D.

16. For normally closed isolation valves a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the first valve.

17. This line (2CS-1-945-2501R-2) is high energy up to normally closed valve 1-8511A.

18. This line (2CS-1-946-2501R-2) is high energy up to normally closed valve 1-8511B.

19. This line (2CS-1-482-2501R-2) is high energy up to normally closed valve 1-CS-8387A.

20. This line (2CS-1-483-2501R-2) is high energy up to normally closed valve 1-CS-08387B.

21. The categorization of the break jets (subcooled liquid, two phase flow) is arrived at by consideration of contributing fluid inventory upstream and downstream of the break (i.e. piping, pressure vessels, heat exchangers, pressurizer, steam generator). The plant operating conditions for the jet selection are defined in Section 6.2.2 of ANS 58.2 (1988 final draft issue) as stated below:

    (a) For those portions of piping systems which are normally pressurized during operation, the thermodynamic state in the pipe and the associated reservoirs shall be that corresponding to 100 percent power.

    (b) For those portions of high energy piping systems which are normally pressurized only during plant conditions other that 100 percent power, the thermodynamic state and associated operating conditions shall be determined using the most severe mode. The most severe jet was selected for the purposes of calculating fluid reaction forces and plant environmental conditions.

22. Vent and drain lines (1 1/2 inches and below) branching off high energy lines are also considered high energy lines up to and including the closed valve.

23. Unit 1 lines listed; Unit 2 lines similar unless otherwise noted.

24. This high energy line is located is safety related area; however, no detailed seismic design was performed. All essential equipment required for mitigation are seismically designed and/or supported.

25. Unit 2 only - Line eliminated on Unit 1 with RSG installation.

TABLE 3.6B-1
HIGH ENERGY LINE LIST (See Notes 22 and 23)

(Sheet 32 of 32)

LINE NUMBER	BUILDING	CONDITION OF JET (SEE NOTE 21) SUB COOLED	CONDITION OF JET (SEE NOTE 21) 2 PHASE FLOW	REMARKS	FLOW DIAGRAM FIGURE NO.

26. Preheater Bypass Line Unit 2 only. For Unit 1, line partially removed from connection to 18” main feedwater line to connection with 4” auxiliary feedwater line. Remaining portion of line is for auxiliary feedwater flow only.

TABLE 3.6B-2
POSTULATED BREAK LOCATIONS FOR THE LOCA ANALYSIS OF THE PRIMARY COOLANT LOOP[a]

Location of Postulated Rupture [b]	Type	Break Opening Area[c]
1. Reactor vessel inlet nozzle	Guillotine	Effective cross-sectional flow area of the loop pipe
2. Reactor vessel outlet nozzle	Guillotine	Effective cross-sectional flow area of the loop pipe
3. Steam generator inlet nozzle	Guillotine	Cross-sectional flow area of the loop pipe
4. Steam generator outlet nozzle	Guillotine	Cross-sectional flow area of the loop pipe
5. Reactor coolant pump inlet nozzle	Guillotine	Cross-sectional flow area of the loop pipe
6. Reactor coolant pump outlet nozzle	Guillotine	Cross-sectional flow area of the loop pipe
7. 50° elbow on the intrados	Longitudinal	Cross-sectional flow area of the loop pipe
8. Loop closure weld in crossover leg	Guillotine	Cross-sectional flow area of the loop pipe
9. Residual heat removal (RHR) line/primary coolant loop connection	Guillotine (viewed from the RHR Line)	Cross-sectional flow area of the RHR line
10. Accumulator (ACC) line/primary Coolant loop connection	Guillotine (viewed from the ACC line)	Cross-sectional flow area of the ACC line
11. Pressurizer surge (PS) line/primary coolant loop connection	Guillotine (viewed from the PS line)	Cross-sectional flow area of the PS line
12. Six inch (6") safety injection (SI) line primary coolant loop connection	Guillotine (Viewed from the SI line)	Cross-Sectional flow area of the SI line
13. Four inch (4") Pressurizer Spray line (Spray Line) primary coolant loop	Guillotine (Viewed from the Spray Line)	Cross-Sectional Flow area of the Spray line

a) Rupture locations 1 through 11 are postulated for containment design, ECCS and Environmental Qualification. These breaks are not postulated for dynamic effects (Sections 3.6B.2.1.1 and 3.6B.2.1.2)

b) Refer to Figure 3.6B-9 for location of postulated breaks in reactor coolant loop.

c) Less break opening area will be used if justified by analysis, experiments, or considerations of physical restraints such as concrete walls or structural steel.

TABLE 3.6B-3
THIS TABLE HAS BEEN DELETED

TABLE 3.6B-4A
MASS/ENERGY RELEASES 90% POWER (INCLUDING POWER UNCERTAINTY) - MODERATOR DENSITY COEFFICIENT = 0.38 Δk/gm/cc FOR 1.0 FT^2 MAIN STEAM LINE BREAK (UNIT 1)
(Sheet 1 of 2)

TIME (sec)	FLOW (lbm/sec)	ENTHALPY (Btu/lbm)
0.0	0	0
0.2	2196.38	1188.307
6.0	2009.046	1191.674
12.0	1898.802	1193.597
18.0	1836.559	1194.632
24.0	1975.015	1193.251
29.6	2180.119	1189.614
30.0	2179.208	1189.639
36.0	2072.859	1191.627
42.0	2002.969	1192.88
48.0	1927.116	1194.198
54.0	1852.632	1195.434
60.0	1781.938	1196.541
66.0	1715.045	1197.543
72.0	1651.719	1198.437
78.0	1592.69	1199.241
84.0	1539.075	1199.937
90.0	1490.214	1200.549
96.0	1442.744	1201.115
102.0	1396.781	1201.633
108.0	1357.275	1202.05
114.0	1324.872	1202.372
120.0	1300.142	1202.604
126.0	1283.228	1202.756
132.0	1272.949	1202.846
138.0	1267.352	1202.894

TABLE 3.6B-4A
MASS/ENERGY RELEASES 90% POWER (INCLUDING POWER UNCERTAINTY) - MODERATOR DENSITY COEFFICIENT = 0.38 Δk/gm/cc FOR 1.0 FT^2 MAIN STEAM LINE BREAK (UNIT 1)
(Sheet 2 of 2)

TIME (sec)	FLOW (lbm/sec)	ENTHALPY (Btu/lbm)
144.0	1264.349	1202.919
150.0	1263.092	1202.93
162.0	1262.974	1202.931
174.0	1262.364	1202.936
186.0	1255.459	1203.546
198.0	1243.922	1204.636
210.0	1228.394	1206.321
218.0	1209.594	1209.435
226.0	1185.714	1212.072
234.0	1155.766	1214.435
242.0	1119.07	1216.633
250.0	1078.283	1218.95
258.0	1036.129	1221.4
266.0	993.7287	1223.977
274.0	949.1581	1226.713
282.0	900.4384	1229.79
290.0	846.1362	1233.41
298.0	785.2863	1237.589
306.0	358.9854	1266.399
314.0	165.2518	1277.871
322.0	96.78773	1281.54
330.0	86.57766	1281.399
1800.0	99.79793	1275.065
1815.0	0	0

TABLE 3.6B-4A
MASS/ENERGY RELEASES 90% POWER (INCLUDING POWER UNCERTAINTY) - MODERATOR DENSITY COEFFICIENT = 0.38 Δk/gm/cc FOR 1.0 FT^2 MAIN STEAM LINE BREAK (UNIT 1)
(Sheet 3 of 3)

Reactor Trip on Overpower N-16 Setpoint at 22.5 sec.

Feedwater Isolation at 32.2 sec.

Safety Injection on Low Pressurizer Pressure at 53.3 sec.

Low SG Water Level Setpoint reached at 71.5 sec.

Tube Bundle Uncovery at 182.2 sec.

Steam Line Isolation at 297.8 sec.

AFW Isolation assumed at 138.3 sec.

TABLE 3.6B-4B
MASS/ENERGY RELEASES 90% POWER (INCLUDING POWER UNCERTAINTY) MODERATOR DENSITY COEFFICIENT = 0.38 Dk/gm/cc FOR 1.0 FT^2 MAIN STEAM LINE BREAK (UNIT 2)
(Sheet 1 of 3)

TIME (sec)	FLOW (lbm/sec)	ENTHALPY (Btu/lbm)
0.0	0	0
0.2	2166.222	1188.911
6.0	1965.62	1192.472
12.0	1853.074	1194.388
18.0	1795.158	1195.318
24.0	2103.635	1190.959
29.6	2167.3	1189.858
30.0	2101.651	1191.124
36.0	2011.238	1192.735
42.0	1933.979	1194.085
48.0	1853.376	1195.426
54.0	1777.098	1196.618
60.0	1705.335	1197.686
66.0	1638.036	1198.628
72.0	1576.008	1199.464
78.0	1519.357	1200.193
84.0	1468.629	1200.814
90.0	1425.646	1201.31
96.0	1390.138	1201.703
102.0	1358.259	1202.038
108.0	1331.529	1202.306
114.0	1310.011	1202.512
120.0	1293.449	1202.665
126.0	1281.434	1202.773
132.0	1273.271	1202.844
138.0	1268.199	1202.887

TABLE 3.6B-4B
MASS/ENERGY RELEASES 90% POWER (INCLUDING POWER UNCERTAINTY) MODERATOR DENSITY COEFFICIENT = 0.38 Dk/gm/cc FOR 1.0 FT^2 MAIN STEAM LINE BREAK (UNIT 2)
(Sheet 2 of 3)

144.0	1265.266	1202.912	
150.0	1264.22	1202.921	
162.0	1262.706	1203.782	
174.0	1258.14	1204.527	
186.0	1240.455	1208.688	
198.0	1208.474	1212.962	
210.0	1158.5	1218.177	
218.0	1110.042	1222.49	
226.0	1043.983	1228.022	
234.0	955.1543	1235.031	
242.0	843.4539	1243.472	
250.0	543.1462	1260.558	
258.0	200.8643	1278.106	
266.0	104.4548	1283.656	
274.0	87.92005	1283.82	
282.0	92.18698	1282.319	
290.0	97.37733	1281.058	
298.0	99.62243	1280.323	
306.0	99.99.89142	1279.861	
314.0	99.76444	1279.483	
322.0	99.6874	1279.141	
330.0	99.67809	1278.837	
1800.0	99.80119	1268.596	
1815.0	0	0	

TABLE 3.6B-4B
MASS/ENERGY RELEASES 90% POWER (INCLUDING POWER UNCERTAINTY) MODERATOR DENSITY COEFFICIENT = 0.38 Dk/gm/cc FOR 1.0 FT^2 MAIN STEAM LINE BREAK (UNIT 2)
(Sheet 3 of 3)

Reactor Trip on High Neutron Flux Setpoint at 19.8 sec.

Feedwater Isolation at 31.1 sec.

Safety Injection on Low Pressurizer Pressure at 55.4 sec.

Low SG Water Level Setpoint reached at 177.6 sec.

Tube Bundle Uncovery at 152.0 sec.

Steam Line Isolation at 247.1 sec.

AFW Isolation assumed at 140.4 sec

TABLE 3.6B-5
EQUIPMENT AFFECTED BY MSLB SUPERHEAT CONDITION

(Sheet 1 of 4)

EQUIPMENT	REQUIRED OPERABILITY TIME	POSTULATED TEMPERATURE PEAK	DEMONSTRATED TEMPERATURE PEAK (3)	DEMONSTRATED OPERABILITY TIME	TIME MARGIN	NOTES
MSIV	Not required (in room with break)	<334°F	355°F	N/A	N/A	(1)
	72 hours (not in room with break)	<334°F	355°F	30 days	27 days	(2 and 5)
MSIV Bypass (MSIVBP)	Not required	<334°F	N/A	N/A	N/A	(13)
Main Steam Drain Pot Isolation Valves (MSDPIV)	Not required (in room with break)	<334°F	346°F	N/A	N/A	(1)
	30 minutes (not in room with break)	<334°F	346°F	100 days	>99 days	(4)
Feedwater Isolation Valves (FIVs)	72 hours	<334°F	390°F	30 days	27 days	(5 and 14)
Feedwater Bypass Valves (FIBVS)	30 minutes	<334°F	346°F	100 days	>99 days	(4)
Feedwater Sample Isolation Valves (FSIVs)	30 minutes	<334°F	346°F	100 days	>99 days	(4)
Turbine Driven Auxiliary Feedwater Pump Turbine Steam Supply Valve (TDAFPTSUP)	Not required (in room with break)	<334°F	345°F	N/A	N/A	(1)
	72 hours (not in room with break)	<334°F	345°F	30 days	27 days	(5)
Main Steam Power Operated Relief Valves (MSPORVs)	Not required (in room with break)	<334°F	334°F	N/A	N/A	(1)
	72 hours (not in room with break)	<334°F	334°F	>30 days	>27 days	(5)

TABLE 3.6B-5
EQUIPMENT AFFECTED BY MSLB SUPERHEAT CONDITION

(Sheet 2 of 4)

EQUIPMENT	REQUIRED OPERABILITY TIME	POSTULATED TEMPERATURE PEAK	DEMONSTRATED TEMPERATURE PEAK (3)	DEMONSTRATED OPERABILITY TIME	TIME MARGIN	NOTES
Barton Steam Line Pressure Transmitters	72 hours	<334°F	400°F	100 days	97 days	(5, 6 and 12)
Rosemount Steam Line Pressure Transmitters	Not required (in room with break)	N/A	N/A	N/A	N/A	(7)
	72 hours (not in room with break)	<334°F	350°F	100 days	97 days	(5 and 15)
Watertight Doors	N/A	N/A	N/A	N/A	N/A	(8)
HVAC Isolation Dampers	72 hours	<334°F	336°F	100 days	>97 days	(5 and 9)
Main Steam Safety Valve (MSSVs)	Not required (in room with break)	<334°F	600°F	N/A	N/A	(1)
	72 hours (not in room with break)	<334°F	600°F	100 days	97 days	(5 and 11)
Accessory Limit Switches for MSIV and MSIVBP (on intact steam generators)	72 hours	<334°F	380°F	100 days	97 days	(5 and 12)
Accessory Limit Switches for MSDPIV, TDAFPTSUP, SPORV, FIV and FIBV (on intact steam generators)	72 hours	<334°F	372°F	100 days	97 days	(5 and 12)
Accessory Limit Switches for FSIVs and FIBV	Not required	<334°F	N/A	N/A	N/A	(16)
Auxiliary Feedwater Flow Transmitters (on intact steam generators)	72 hours	<334°F	411°F	100 days	97 days	(5 and 12)

TABLE 3.6B-5
EQUIPMENT AFFECTED BY MSLB SUPERHEAT CONDITION

(Sheet 3 of 4)

EQUIPMENT	REQUIRED OPERABILITY TIME	POSTULATED TEMPERATURE PEAK	DEMONSTRATED TEMPERATURE PEAK (3)	DEMONSTRATED OPERABILITY TIME	TIME MARGIN	NOTES
Main Steam Safety Valve Position Indicator (on intact steam generators)	72 hours	<334°F	510°F	100 days	97 days	(5 and 12)
Auxiliary Feedwater Containment Isolation MOVs	Not Required	<334°F	385°F	100 days	100 days	(10)

Notes:

(1) These valves may fail in either the open or closed position without increasing the severity of the event beyond the present analysis or decreasing the ability to mitigate the event and safely shut down the unit.

(2) These valves will operate to close within 30 minutes. Beyond this time (for up to 72 hours while proceeding to cold shutdown) these valves must remain closed. Since the Class 1E solenoid valves of the MSIV operator are energized to close and are only qualified for 90 minutes, the non Class 1E solenoid on each MSIV (which prevent reopening of the valve should the Class 1E solenoids fail) are qualified for greater than 72 hours to ensure that the MSIV remains closed. Should offsite power not be available, guidance is provided in the applicable emergency operating procedure for early restoration of power to the battery charger for the batteries which power these non-1E solenoids. See the Table 17A-1 for the MSIV Train C solenoids.

(3) For no equipment where operability is required does the postulated accident profile exceed the demonstrated qualification profile.

(4) These valves operate to close within 30 minutes. Operability is not required beyond then for the valves will remain in a safe position.

(5) 72 hour operability time is based on achieving cold shutdown within 72 hours such that the operability of these items would no longer be required.

(6) The low steamline pressure signal or high negative pressure rate signal is provided by these protection channel transmitters. Thirty minute operability time is based on providing steam line isolation signal and providing required instrumentation to identify and isolate the faulted steam line. Operability after 30 minutes is based on control of cooldown rates.

7) This transmitter provides an analog signal for the operation of the Main Steam PORV associated with the break. This transmitter may fail without increasing the severity of the event beyond the present analysis or decreasing the ability to mitigate the event and safely shutdown the unit.

(8) These are mechanical devices that do not have an active function. The materials have been analyzed to show that the doors will not lose their leak tightness.

TABLE 3.6B-5
EQUIPMENT AFFECTED BY MSLB SUPERHEAT CONDITION

(Sheet 4 of 4)

EQUIPMENT	REQUIRED OPERABILITY TIME	POSTULATED TEMPERATURE PEAK	DEMONSTRATED TEMPERATURE PEAK (3)	DEMONSTRATED OPERABILITY TIME	TIME MARGIN	NOTES

(9) The dampers are mechanical devices. They are mechanically closed by the pressure buildup from the break and are latched in the closed position (as well as being held closed by compartment pressure). Failure of the environmentally sensitive components due to this event will not prevent closure of the damper nor cause the damper to reopen. Therefore, the required operability is early in the event, but the dampers remain continuously operable in spite of the postulated environment. The data shown is for damper gasket material which ensures damper leak tightness.

(10) This equipment is not required to isolate auxiliary feedwater supply to the faulted steam generator. Isolation is achieved by valves not exposed to the high energy environment.

(11) The Main Steam safety valve is required to operate to maintain hot standby conditions. No Class 1E cables/components are associated with the Main Steam safety valve's function.

(12) This equipment is required to indicate the status of valve positions or process variables for accident monitoring.

(13) These valves are locked closed manual valves and no failure could cause them to open.

(14) These valves will isolate to close within 30 minutes. Beyond this time (for up to 72 hours while proceeding to cold shutdown) these valves must remain fully closed. Since the Class 1E solenoid valves of the FIV are energized to close and are only qualified for 80 minutes, the valve is maintained closed by automatic de-energization of the electric pump which provides hydraulic fluid for opening.

(15) These transmitters provide analog signals for the control operations of the main steam PORV's. Although they are not required to function, they must not fail in a manner which would open the PORVs.

(16) Valve Position Indication of these normally closed valves is not essential for cracks in break exclusion piping.

TABLE 3.6B-6
MASS AND ENERGY RELEASE RATES 1 FT^2 BREAK - FEED WATER LINE
(Unit 1 and Unit 2 with original steam generators)

TIME (SEC.)	FLOW (LB/SEC.)	ENTHALPY FLOW (BTU/SEC)
0.0	6200	$2.64x10^6$
8.281	5100	$2.437x10^6$
108.6	0	0

Note: For Unit 1 with replacement steam generators, mass and energy release data were generated for each loop utilizing RELAP5/MOD3.

TABLE 3.6B-7
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TABLE 3.6B-8
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3.7N SEISMIC DESIGN

In addition to the steady state loads imposed on the system under normal operating conditions, the design of equipment and equipment supports requires that conditions also be given to abnormal loading conditions such as earthquakes. Seismic loadings are considered for earthquakes of two magnitudes: Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE). The SSE is defined as the maximum vibratory ground motion at the plant site that can reasonably be predicted from geologic and seismic evidence. The OBE is that earthquake which, considering the local geology and seismology, can be reasonably expected to occur during the plant life.

For the OBE loading condition, the Nuclear Steam Supply System is designed to be capable of continued safe operation. The design for the SSE is intended to assure:

1. That the integrity of the reactor coolant pressure boundary is not comprised;

2. That the capability to shutdown the reactor and maintain it in a safe condition is not comprised; and

3. That the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10CFR100 is not comprised.

It is necessary to ensure that required critical structures and components do not loose their capability to perform their safety function. Not all critical components have the same functional safety requirements. For example, a safety injection pump must retain its capability to function normally during the SSE. Therefore, the deformation in the pump must be restricted to appropriate limits in order to assure its ability to function. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain their contents and allow fluid flow.

The seismic requirements for safety-related instrumentation and electrical equipment are covered in Section 3.10. The safety class definitions, classification lists, operating condition categories and the methods used for seismic qualification of mechanical equipment are given in Section 3.2.

3.7N.1 SEISMIC INPUT

3.7N.1.1 Design Response Spectra

Refer to Section 3.7B.1.1.

3.7N.1.2 Design Time History

Refer to Section 3.7B.1.2.

3.7N.1.3 Critical Damping Values

The damping values given in Table 3.7N-1 are used for the systems analysis of Westinghouse equipment and the original Westinghouse analyses. These are consistent with the damping values recommended in Regulatory Guide 1.61 except in the case of the primary coolant loop system components and large piping (excluding reactor pressure vessel internals) for which the damping values of 2 percent and 4 percent are used as established in testing programs reported in Reference [1]. As an alternative and as defined in Table 3.7N-1, piping systems analyzed by the response spectrum method may use ASME Code Case N-411 damping values. Qualification analyses of the primary coolant loop equipment (steam generators; reactor pressure vessel and reactor coolant pumps) incorporate equipment damping values as defined in Table 3.7N-1. ASME Code Case N-411 damping values are used for the reconciliation analyses of all NSSS scope piping stress analysis packages, including the Primary Coolant Loop piping for Unit 2. The time history method of seismic analysis and composite modal damping are used as discussed in Section 3.9N.1.4.3 for the reactor coolant loops for Unit 1. ASME Code Case N-397 is not used for NSSS scope piping analysis but any subsequent usage will be detailed in the FSAR and in compliance with reference [8].

The damping values for control rod drive mechanisms (CRDM's) and the fuel assemblies of the Nuclear Steam Supply System, when used in seismic system analysis, are in conformance with the values for welded and/or bolted steel structures (as appropriate) listed in Regulatory Guide 1.61.

Tests on fuel assembly bundles justified conservative component damping values of 7 percent for OBE and 10 percent for SSE to be used in the fuel assembly component qualification. Documentation of the fuel assembly tests is found in Reference [2].

The damping values used in component analysis of CRDM's and their seismic supports were developed by testing programs performed by Westinghouse. These tests were performed during the design of the CRDM support; the support was designed so that the damping in Table 3.7N-1 could be conservatively used in the seismic analysis. The CRDM support system is designed with plates at the top of the mechanism and gaps between mechanisms. These are encircled by a box section frame which is attached by tie rods to the refueling cavity wall. The test conducted was on a full size CRDM complete with rod position indicator coils, attachment to a simulated vessel head, and variable gap between the top of the pressure housing support plate and a rigid bumper representing the support. The internal pressure of the CRDM was 2250 pounds per square inch (psi) and the temperature on the outside of the pressure housing was 400°F.

The program consisted of transient vibration tests in which the CRDM was deflected a specified initial amount and suddenly released. A logarithmic decrement analysis of the decaying transient provides the effective damping of the assembly. The effect on damping of variations in the drive shaft axial position, upper seismic support clearance and initial deflection amplitude was investigated.

The upper support clearance had the largest effect on the CRDM damping with the damping increasing with increasing clearance. With an upper clearance of 0.06 inches, the measured damping was approximately 8 percent. The clearances in a typical upper seismic CRDM support is a minimum of 0.10 inches. The increasing damping with increasing clearances trend from the test results indicated that the damping would be greater than 8 percent for both the OBE and the SSE based on a comparison between typical deflections during these seismic events to the initial

deflections of the mechanisms in the test. Component damping values of 5 percent are therefore conservative for both the OBE and the SSE. The CRDM component damping applies to both Units 1 and 2 since the upper support clearances are the same and the L106A and L106C models are structurally similar.

These damping values are used and applied to CRDM component analyses by response spectra and time history techniques.

3.7N.1.4 Supporting Media for Seismic Category I Structures

Refer to Section 3.7B.1.4.

3.7N.2 SEISMIC SYSTEM ANALYSIS

This section describes the methods of seismic analysis performed for safety-related components and systems within Westinghouse's scope.

3.7N.2.1 Seismic Analysis Methods

Those components and systems that must remain functional in the event of the SSE (Seismic Category I) are identified by applying the criteria of Section 3.2.1. This equipment is classified into three types according to its dynamic characteristics. The analysis methods used for this equipment also depended on these classifications.

The first type is flexible equipment. This equipment is characterized by several modes in the frequency range that could produce amplification of the base input motion. In addition, more than one mode could be subjected to the peak response. Because of these reasons, dynamic analyses were performed for these components using modal analysis techniques with either the response spectrum method or integration of the uncoupled modal equations of motion, or by direct integration of the coupled differential equation of motion. Details of the methods used for these analyses are described in Section 3.7N.2.1.1 through 3.7N.2.1.5.

The second classification is rigid equipment. This equipment has a fundamental natural frequency that is sufficiently high (greater than 33 Hz) so that it is not excited by input motion at the base. Such equipment is particularly suitable for static analysis as described in Section 3.7N.2.1.6.

Finally, the third type of equipment is classified as limited flexible, with only one predominate mode in the frequency range subject to possible amplification of the input motion. The fundamental mode of this type of equipment is basically a translational bending mode at a frequency less than 33 Hz. The second mode is usually a rocking mode with a frequency greater than 33 Hz. Because of the simple response characteristics of the equipment, dynamic analysis techniques that account for multiple mode effects and closely spaced modes are not required. Therefore, this equipment was evaluated using static analysis methods as described in Section 3.7N.2.1.6.

3.7N.2.1.1 Dynamic Analysis - Mathematical Model

The first step in any dynamic analysis is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dashpots suitable for

mathematical analysis. The essence of this step is to select a model so that the displacements obtained will be a good representation of the motion of the structure or component. Stated differently, the true inertia forces should not be altered so as to appreciably affect the internal stresses in the structure of component. Some typical modeling techniques are presented in Reference [3].

Equations of Motion

Consider the multi-degree of freedom system shown in Figure 3.7N-1. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r \ddot{y}_r + \sum^{i} c_{ri} \dot{u}_i + \sum^{i} k_{ri} u_i = 0 \qquad (3.7N\text{-}1)$$

where

m_r = the value of the mass or mass moment of rotational inertia at mass point r

$\ddot{y}_r$ = absolute translational or angular acceleration of mass point r

c_{ri} = damping coefficient - external force or moment required at mass point r to produce a unit translational or angular velocity at mass point i, maintaining zero translational or angular velocity at all other mass points. Force or moment is positive in the direction of positive translational or angular velocity

$\dot{u}_i$ = translational or angular velocity of mass point i relative to the base

k_{ri} = stiffness coefficient - the external force (moment) required at mass point r to produce a unit deflection (rotation) at mass point i, maintaining zero displacement (rotation) at all other mass points

Force (moment) is positive in the direction of the displacement (rotation)

u_i = displacement (rotation) of mass point i relative to the base

As an example, note that Figure 3.7N-1 does not attempt to show all of the springs (and none of the dashpots) which are presented in Equation (3.7N-1).

Since:

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s \qquad (3.7N\text{-}2)$$

where

$\ddot{y}_s$ = absolute translational (angular) acceleration of the base

$\ddot{u}_r$ = translational (angular) acceleration of mass point r relative to the base

Equation (3.7N-1) can be written as:

$$m_r\ddot{u}_r + \sum^{i} c_{ri}\dot{u}_i + \sum^{i} k_{ri}u_i = -m_r\ddot{y}_s \qquad (3.7N\text{-}3)$$

For a single degree of freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is:

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad (3.7N\text{-}4)$$

3.7N.2.1.2 Modal Analysis

Natural Frequencies and Mode Shapes

The first step in the modal analysis method is to establish the normal modes, which were determined by eigen solution of Equation (3.7N-3). The right hand side and the damping term are set equal to zero for this purpose as illustrated in Reference [4] (pages 83 through 111). Thus, Equation (3.7N-3) becomes:

$$m_r\ddot{u}_r + \sum^{i} k_{ri}u_i = 0 \qquad (3.7N\text{-}5)$$

The equation given for each mass point r in Equation (3.7N-5) can be written as a system of equations in matrix form as:

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = 0 \qquad (3.7N\text{-}6)$$

where

[M] = mass and rotational inertia matrix

{Δ} = column matrix of the general displacement and rotation at each mass point relative to the base

[K] = square stiffness matrix

$\{\ddot{\Delta}\}$ = column matrix of general translational and angular accelerations at each mass point relative to the base, $d^2\{\Delta\}/dt^2$

Harmonic motion is assumed and the $\{\Delta\}$ is expressed as:

$$\{\Delta\} = \{\delta\}\sin_{\omega}t \qquad (3.7N\text{-}7)$$

where

$\{\delta\}$ = column matrix of the spatial displacement and rotation at each mass point relative to the base

ω = natural frequency of harmonic motion in radians per second

The displacement function and its second derivative are substituted into Equation (3.7N-6) and yield:

$$[K]\{\delta\} = \omega^2[M]\{\delta\} \qquad (3.7N\text{-}8)$$

The determinant [K] - ω^2 [M] is set equal to zero and is then solved for the natural frequencies. The associated mode shapes are then obtained from Equation (3.7N-8). This yields n natural frequencies and mode shapes where n equals the number of dynamic degrees of freedom of the system. The mode shapes are all orthogonal to each other and are sometimes referred to as normal mode vibrations. For a single degree of freedom system, the stiffness matrix and mass matrix are single terms and the determinant $[K]^{-2}$ [M] when set equal to zero yields simply:

$$k - \omega^2 m = 0$$

or:

$$\omega = \sqrt{\frac{k}{m}} \qquad (3.7N\text{-}9)$$

where ω is the natural angular frequency in radians per second.

The natural frequency in cycles per second is therefore:

$$f = \frac{1}{2\pi}\sqrt{\frac{k}{m}} \qquad (3.7N\text{-}10)$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation (3.7N-8).

Modal Equations

The response of a structure or component is always some combination of its normal modes. Good accuracy can usually be obtained by using only the first few modes of vibration. In the normal mode method, the mode shapes are used as principal coordinates to reduce the equations of motion to a set of uncoupled differential equations that describe the motion of each mode n. These equations may be written as (Reference [4], pages 116 through 125):

$$\ddot{A}_n + 2_{\omega n} p_n \dot{A}_n = -\Gamma n \ddot{y}_s \qquad (3.7N\text{-}11)$$

where the modal displacement or rotation, A_n, is related to the displacement or rotation of mass point r in mode n, u_{rn}, by the equation:

$$u_{rn} = A_n \phi_{rn} \qquad (3.7N\text{-}12)$$

where

ϕ_{rn} = the modal displacement for mode n at mass point r

ω_n = natural frequency of mode n in radians per second

p_n = critical damping ratio of mode n

Γn = modal participation factor of mode n given by:

$$\Gamma n = \frac{\sum^{n} m_r \varphi'_{rn}}{\sum^{n} m^2_{r\varphi} rn} \qquad (3.7N\text{-}13)$$

where

φ'_{rn} = value of φ_{rn} in the direction of the earthquake

The essence of the modal analysis lies in the fact that Equation (3.7N-11) is analogous to the equation of motion for a single degree of freedom system that will be developed from Equation (3.7N-4). Dividing Equation (3.7N-4) by m gives:

$$\ddot{u} + \frac{c}{m}\dot{u} + \frac{k}{m}u = -\ddot{y}_s \qquad (3.7N\text{-}14)$$

The critical damping ratio of a single degree of freedom system, p, is defined by the equation:

$$p \equiv \frac{c}{c_c} \qquad (3.7N\text{-}15)$$

where the critical damping coefficient is given by the expression:

$$c_c = 2m\omega \qquad (3.7N\text{-}16)$$

Substituting Equation (3.7N-16) into Equation (3.7N-15) and solving for c/m gives:

$$\frac{c}{m} = 2\omega\rho \qquad (3.7N\text{-}17)$$

Substituting this expression and the expression for k/m given by Equation (3.7N-9) into Equation (3.7N-14) gives:

$$\ddot{u} + 2\omega\rho\dot{u} + \omega^2 u = -\ddot{y}_s \qquad (3.7N\text{-}18)$$

Note the similarity of Equations (3.7N-11) and (3.7N-18). Thus each mode may be analyzed as though it were a single degree of freedom system and all modes are independent of each other. By this method a fraction of critical damping, i.e., c/c_c, may be assigned to each mode and it is not necessary to identify or evaluate individual damping coefficients, i.e., c. However, assigning only a single damping ratio to each mode has a drawback. Normally, there are two ways used to overcome this limitation when considering a slightly damped structure (e.g., steel supported by a massive moderately damped structure (e.g., concrete)).

The first method is to develop and analyze separate mathematical models for both structures using their respective damping values. The massive moderately damped support structure is analyzed first. The calculated response at the support points for the slightly damped structures are used as forcing functions for their subsequent detailed analysis. The second method is to inspect the mode shapes to determine which modes correspond to the slightly damped structure and then use the damping associated with the structure having predominant motion.

3.7N.2.1.3 Response Spectrum Analysis

The response spectrum is a plot showing the variation in the maximum response (Reference [5], pages 24 through 51) (displacement, velocity, and acceleration) of a single degree of freedom system versus its natural frequency of vibration when subjected to a time history motion of its base. Examples of response spectra are shown in Figures 3.7N-2 and 3.7N-3.

The response spectrum concept can be best explained by outlining the steps involved in developing a spectrum curve. Determination of a single point on the curve requires that the response (displacement, velocity, and acceleration) of a single degree of freedom system with a given damping and natural frequency is calculated for a given base motion. The variations in

response are established and the maximum absolute value of each is plotted as an ordinate with the natural frequency used as the abscissa. The process is repeated for other assumed values of frequency is sufficient detail to establish the complete curve. Other curves corresponding to different fractions of critical damping are obtained in a similar fashion. Thus, the determination of each point of the curve requires a complete dynamic response analysis, and the determination of a complete spectrum may involve hundreds of such analyses. However, once a response spectrum plot is generated for the particular base motion, it may be used to analyze each structure and component with that base motion. The spectral acceleration, velocity, and displacement are related by the equation:

$$S_{a_n} = \omega_n S_{v_n} = \omega_n^2 S_{d_n} \qquad (3.7N\text{-}19)$$

There are two types of response spectra that must be considered. If a given building is shown to be rigid and to have a hard foundation, the ground response spectrum or ground time history is used. It is referred to as a ground response spectrum. If the building is flexible and/or has a soft foundation, the ground response spectrum is modified to include these effects. The response spectrum at various support points must be developed. These are called floor response spectra. The specific response spectra used are discussed in Sections 3.7B.1 and 3.7B.2.5.

3.7N.2.1.4 Integration of Modal Equations

This method can be separated into the following two basic parts:

1. Integration procedure for the uncoupled modal Equation (3.7N-11) to obtain the modal displacements and accelerations as a function of time.

2. Using these modal displacements and accelerations to obtain the total displacements, accelerations, forces, and stresses.

Integration Procedure

Integration of these uncoupled modal equations is done by step-by-step numerical integration. The step-by-step numerical integration procedure consists of selecting a suitable time interval, Δt, and calculating modal acceleration, $\ddot{A}_n$, modal velocity, $\dot{A}_n$, and modal displacement, A_n, at discrete time stations Δt apart, starting at t = 0 and continuing through the range of interest for a given time history of base acceleration.

Total Displacements, Accelerations, Forces and Stresses

From the modal displacements and accelerations, the total displacements, accelerations, forces and stresses can be determined as follows:

1. Displacement of mass point r in mode n as a function of time is given by Equation (3.7N-12) as:

$$u_{rn} = An\varphi rn \qquad (3.7N\text{-}20)$$

with the corresponding acceleration of mass point r in mode n as:

$$\ddot{u}_{rn} = \dot{A}_n \varphi_{rn} \qquad (3.7N\text{-}21)$$

2. The displacement and acceleration values obtained for the various modes are superimposed algebraically to give the total displacement and acceleration at each time interval.

3. The total acceleration at each time interval is multiplied by the mass to give an equivalent static force. Stresses are calculated by applying these forces to the model or from the deflections at each time interval.

3.7N.2.1.5 Integration of Coupled Equations of Motion

The dynamic transient analysis is a time history solution of the response of a given structure to know forces and/or displacement forcing functions. The structure may include linear or nonlinear elements, gaps, interfaces, plastic elements, and viscous and Coulomb dampers. Nodal displacements, nodal forces, pressure, and/or temperatures may be considered as forcing functions. Nodal displacements and elemental stresses for the complete structure are calculated as functions of time.

The basic equations for the dynamic analysis are as follows:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} \qquad (3.7N\text{-}22)$$

where the terms are as defined earlier and {F(t)} may include the effects of applied displacements, forces, pressures, temperatures, or nonlinear effects such as plasticity and dynamic elements with gaps. Options of translational accelerations input to a structural system and the inclusion of static deformation and/or preload may be considered in the nonlinear dynamic transient analysis. The option of translational input such as uniform base motion to a structural system is considered by introducing an inertia force term of $-[M]\{\ddot{z}\}$ to the right hand side of the basic Equation (3.7N-22), i.e.,

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{f\} - [M]\{\ddot{z}\} \qquad (3.7N\text{-}23)$$

The vector $\{\ddot{z}\}$ is defined by its components $\ddot{z}_i$ where i refers to each degree of freedom of the system. $\ddot{z}_i$ is equal to a1, a2, or a3 if the i-th degree of freedom is aligned with the direction of the system translational acceleration a1, a2, or a3, respectively. $\ddot{z}_i$ = 0 if the i-th degree of freedom is not aligned with any direction of the system translational acceleration. Typical application of this option is a structural system subjected to a seismic excitation of a given ground acceleration record. The displacement {x} obtained from the solution of Equation (3.7N-23) is the displacement relative to the ground.

The option of the inclusion of initial static deformation or preload in a nonlinear transient dynamic structural analysis is considered by solving the static problem prior to the dynamic analysis. At each stage of integration in transient analysis, the portion of internal forces due to static deformation is always balanced by the portion of the forces which are statically applied. Hence, only the portion of the forces which deviate from the static loads will produce dynamic effects. The output of this analysis is the total result due to static and dynamic applied loads.

One available method for the numerical integration of Equations (3.7N- 22) and (3.7N-23) is the Newmark Beta integration scheme preposed by Chan, Cox, and Benfield (Reference [6]). In this integration scheme, Equations (3.7N-22) and (3.7N-23) are replaced by:

$$\begin{aligned}\frac{1}{(\Delta t)^2}[M]\{x_{n+2}-2x_{n+1}+x_n\}+\frac{1}{2(\Delta t)}\{x_{n+2}-x_n\}[C]\\ +[K]\{\beta x_{n+2}+(1-2\beta)x_{n+1}+\beta x_n\}\\ =\{\beta F_{n+2}+(1-2\beta)F_{n+1}+\beta F_n\}\end{aligned} \qquad \text{(3.7N-24)}$$

where

n, n+1, n+2 = past, present, and future (updated) values of the variables

β = parameter to be selected on the basis of numerical stability and accuracy

F = the total right hand side of the equation of motion (Equation (3.7N-22) or (3.7N-23))

$\Delta t = t_{n+2} - t_{n+1} = t_{n+1} - t_n$

The value of β is chosen equal to 1/3 in order to provide a margin of numerical stability for nonlinear problems. Since the numerical stability of Equation (3.7N-24) is mostly determined by the left hand side terms of that equation, the right hand side terms were replaced by F_{n+2}. Furthermore, since the time increment may vary between two successive time substeps, Equation (3.7N-24) may be modified as follows:

$$\begin{aligned}\frac{2}{(\Delta t+\Delta t_1)}[M]\left\{\frac{x_{n+2}-x_{n+1}}{\Delta t}-\frac{x_{n+1}-x_n}{\Delta t_1}\right\}+\frac{1}{(\Delta t+\Delta t_1)}[C]\{x_{n+2}-x_n\}\\ +\frac{1}{3}[K]\{x_{n+2}+x_{n+1}+x_n\}=\{F_{n+2}\}\end{aligned} \qquad \text{(3.7N-25)}$$

By factoring x_{n+2}, x_{n+1}, and x, and rearranging terms, Equation (3.7N-26) is obtained as follows:

$$\begin{aligned}\{C_5[M]+C_3[C]+(1/3)[K]\}\{x_{n+2}\}=\{F_{n+2}\}\\ +\{C_7[M]-(1/3)[K]\}\{x_{n+1}\}\\ +\{-C_2[M]+C_3[C]-(1/3)K\}\{x_n\}\end{aligned} \qquad \text{(3.7N-26)}$$

Where

$$C_2 = \frac{2}{\Delta t_1(\Delta t + \Delta t_1)}$$

$$C_3 = \frac{1}{\Delta t + \Delta t_1}$$

$$C_5 = \frac{2}{\Delta t_1(\Delta t + \Delta t_1)}$$

$$C_7 = C_2 + C_5$$

The above set of simultaneous linear equations is solved to obtain the present values of nodal displacements x_t in terms of the previous (known) values of the nodal displacements. Since [M], [C], and [K] are included in the equation, they can also be time or displacement dependent.

3.7N.2.1.6 Static Analysis - Rigid and Limited Flexible Equipment

Rigid equipment and limited flexible equipment as defined in Section 3.7N.2.1 are generally analyzed using the static analysis method. This technique involves the multiplication of the total weight of the equipment or component member by a specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient was established on the basis of the excitation level that the component was expected to experience in the plant.

For rigid equipment, the seismic acceleration coefficients were compared with the high frequency (greater than 33 Hz) acceleration levels for the applicable response spectra developed for the plant to confirm the design analysis. The seismic acceleration coefficients for limited flexible equipment are compared with the acceleration levels from the applicable response spectra at the calculated fundamental natural frequency of the component. If the design seismic acceleration coefficients for either rigid or limited flexible equipment are exceeded by the actual plant acceleration levels, the design analyses is performed again at the actual level to confirm the equipment adequacy.

3.7N.2.2 Natural Frequencies and Response Loads

Refer to Section 3.7B.2.2.

3.7N.2.3 Procedures Used for Modeling

Procedures used for modeling are discussed in Section 3.7N.2.1.1.

The mathematical model used for the dynamic analyses of the Reactor Coolant System is shown in Figure 3.9N-1. Figure 3.9N-2 shows the mathematical model used for the reactor pressure vessel.

3.7N.2.4 Soil/Structure Interaction

Refer to Section 3.7B.2.4.

3.7N.2.5 Development of Floor Response Spectra

Refer to Section 3.7B.2.5.

3.7N.2.6 Three Components of Earthquake Motion

The seismic design of the piping and equipment includes the effect of the seismic response of the supports, equipment, structures and components. The system and equipment response is determined using three earthquake components, two horizontal and one vertical. The design ground response spectra, specified in Section 3.7B.1, are the bases for generating these three input components. Floor response spectra are generated for two perpendicular horizontal directions, (i.e., N-S, E-W) and the vertical direction. System and equipment analysis is performed with these input components applied in the N-S, E-W and vertical directions. The damping values used in the analysis are those given in Table 3.7N-1.

In computing the system and equipment response by response spectrum modal analysis the methods of Section 3.7N.2.7 are used to combine all significant modal responses to obtain the combined unidirectional responses.

The combined total response is then calculated using the square root of the sum of the squares formula applied to the resultant unidirectional responses. For instance, for each item of interest such as displacement, force, stresses, etc., the total response is obtained by applying the above described method. The mathematical expression for this method (with R as the item of interest) is:

$$R_C = \left[\sum_{T=1}^{3} R_T^{\,2} \right]^{1/2} \qquad (3.7N\text{-}27)$$

where

$$R_T = \left[\sum_{i=1}^{N} R_{Ti}^{\,2} \right]^{1/2} \qquad (3.7N\text{-}28)$$

where

R_C = total combined response at a point

R_T = value of combined response of direction T

R_{Ti} = absolute value of response for direction T, mode i

N = total number of modes considered

The subscripts can be reversed without changing the results of the combination.

Again, for the case of closely spaced modes, R_T in Equation (3.7N-28) shall be replaced with R_T as given by Equation (3.7N-29) in Section 3.7N.2.7.

3.7N.2.7 Combination of Modal Responses

The total unidirectional seismic response is obtained by combining the individual modal responses utilizing the square root of the sum of the squares method. For systems 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 such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10 percent of the lower frequency. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root of the sum of the squares of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor. This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{j=1}^{S} \sum_{K=M_j}^{N_j - 1} \sum_{\ell = K+1}^{N_j} R_K R_\ell \varepsilon K\ell \qquad \text{(3.7N-29)}$$

where

R_T = total unidirectional response

R_I = absolute value of response of mode i

N = total number of modes considered

S = number of groups of closely spaced modes

M_j = lowest modal number associated with group j of closely spaced modes

N_j = highest modal number associated with group j of closely spaced modes

$\varepsilon K\ell$ = coupled factor with

$$\varepsilon K\ell = \left\{ 1 + \left(\frac{\omega_K^i - \omega_\ell^i}{\beta_K^i \omega_K + \beta_\ell^i \omega_\ell} \right)^2 \right\}^{-1} \qquad \text{(3.7N-30)}$$

and

$$\omega_{K'} = \omega_K [1 - (\beta K')^2]^{1/2} \qquad \text{(3.7N-31)}$$

$$\beta' K = \beta K + \frac{2}{\omega_K t_d} \qquad \text{(3.7N-32)}$$

where

ω_K = frequency of closely spaced mode K

βK = fraction of critical damping in closely spaced mode K

t_d = duration of the earthquake

An example of this equation applied to a system can be supplied with the following considerations. Assume that the predominant contributing modes have frequencies as given below:

Mode	1	2	3	4	5	6	7	8
Frequency	5.0	8.0	8.3	8.6	11.0	15.5	16.0	20

There are two groups of closely spaced modes, namely with modes {2, 3, 4} and {6, 7}. Therefore:

S = 2 number of groups of closely spaced modes

M_1 = 2 lowest modal number associated with group 1

N_1 = 4 highest modal number associated with group 1

M_2 = 6 lowest modal number associated with group 2

N_2 = 7 highest modal number associated with group 2

N = 8 total number of modes considered

The total response for this system is, as derived form the expansion of Equation (3.7N-29):

$$R_T^2 = [R_1^2 + R_2^2 + R_3^2 + \ldots + R_8^2] + 2\,R_2\,R_3\,\varepsilon_{23} + 2\,R_2\,R_4\,\varepsilon_{24} + 2\,R_3\,R_4\,\varepsilon_{34} + 2\,R_6\,R_7\,\varepsilon_{67} \quad \text{(3.7N-33)}$$

3.7N.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

Refer to Section 3.7B.2.8.

3.7N.2.9 Effects of Parameter Variations on Floor Response Spectra

Refer to Section 3.7B.2.9.

3.7N.2.10 Use of Constant Vertical Static Factors

Constant vertical static factors are not used as the vertical floor response load for the seismic design of safety classed systems and components within Westinghouse's scope of responsibility. All such systems and components are analyzed in the vertical direction.

3.7N.2.11 Methods Used to Account for Torsional Effects

Refer to Section 3.7B.2.11.

3.7N.2.12 Comparison of Responses

Refer to Section 3.7B.2.12.

3.7N.2.13 Methods for Seismic Analysis of Dams

Refer to Section 3.7B.2.13.

3.7N.2.14 Determination of Seismic Category I Structure Overturning Moments

Refer to Section 3.7B.2.14.

3.7N.2.15 Analysis Procedure for Damping

In instances under the standard scope of Westinghouse supply and analysis, either the lowest damping value associated with the elements of the system are used for all modes, or an equivalent modal damping value is determined by testing programs such as was done for the reactor coolant loop (Reference [1]).

3.7N.3 SEISMIC SUBSYSTEM ANALYSIS

This section describes the seismic analysis performed on subsystems within Westinghouse's scope of responsibility.

3.7N.3.1 Seismic Analysis Methods

Seismic analysis methods for subsystems within Westinghouse's scope of responsibility are given in Section 3.7N.2.1.

3.7N.3.2 Determination of Number of Earthquake Cycles

For each OBE the system and component will have a maximum response corresponding to the maximum induced stresses. The effect of these maximum stresses for the total number of OBE's must be evaluated to assure resistance to cyclic loading.

The OBE is conservatively assumed to occur 20 times over the life of the plant. The number of maximum stress cycles for each occurrence depends on the system and component damping values, complexity of the system and component, duration and frequency contents of the input earthquake. A precise determination of the number of maximum stress cycles can only be made using time history analysis for each item which is not feasible. Instead, a time history study has

been conducted to arrive at a realistic number of maximum stress cycles for all Westinghouse systems and components.

To determine the conservative equivalent number of cycles of maximum stress associated with each occurrence, an evaluation was performed considering both equipment and its supporting building structure as single degree of freedom systems. The natural frequencies of the building and the equipment are conservatively chosen to coincide. The damping in the equipment and building are equivalent to the damping values in Table 3.7N-1.

The results of this study indicate that the total number of maximum stress cycles in the equipment having peak acceleration above 90 percent of the maximum absolute acceleration did not exceed eight cycles.

If the equipment was assumed to be rigid in a flexible building, the number of cycles exceeding 90 percent of the maximum stress was not greater than three cycles.

This study was conservative since it was performed with single degree of freedom models which tend to produce a more uniform and unattenuated response than a complex interacted system. The conclusions indicate that 10 maximum stress cycles for flexible equipment (natural frequencies less than 33 Hz) and 5 maximum stress cycles for rigid equipment (natural frequencies greater than 33 Hz) for each of 20 OBE occurrences should be used for fatigue evaluation of Westinghouse systems and components.

3.7N.3.3 Procedures Used for Modeling

Refer to Section 3.7N.2.3 for modeling procedures for subsystems in Westinghouse's scope of responsibility.

3.7N.3.4 Basis for Selection of Frequencies

The analysis of equipment subjected to seismic loading involves several basic steps, the first of which is the establishment of the intensity of the seismic loading. Considering that the seismic input originates at the point of support, the response of the equipment and its associated supports based upon the mass and stiffness characteristics of the system, will determine the seismic accelerations which the equipment must withstand.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

1. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low period region of the floor response spectra.

2. If the equipment is very flexible relative to the structure, the internal distortion of the structure is unimportant and the equipment behaves as though supported on the ground.

3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

Also, as noted in Section 3.7N.3.2, rigid equipment/support systems have natural frequencies greater than 33 Hz.

3.7N.3.5 Use of Equivalent Static Load Method of Analysis

The equivalent static load or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as single degree of freedom systems are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom system which may be in the resonance region of the amplified response spectra curves are increased by 50 percent to account conservatively for the increased modal participation.

3.7N.3.6 Three Components for Earthquake Motion

Methods used to account for three components of earthquake motion for subsystems in Westinghouse's scope of responsibility are given in Section 3.7N.2.6.

3.7N.3.7 Combination of Modal Responses

Methods used to combine modal responses for subsystems in Westinghouse's scope of responsibility are given in Section 3.7N.2.7.

3.7N.3.8 Analytical Procedures for Piping

The Class 1 piping systems are analyzed to the rules of the American Society of Mechanical Engineers (ASME) Code, Section III, NB-3650. When response spectrum methods are used to evaluate piping systems supported at different elevations, the following procedures are used. The effect of differential seismic movement of piping supports is included in the piping analysis according to the rules of the ASME Code, Section III, NB-3653. According to ASME definitions, these displacements cause secondary stresses in the piping system. The response quantity of interest induced by differential seismic motion of the support is computed statically and addresses the relative phase of the seismic motion at each support.

In the response spectrum dynamic analysis for evaluation of piping systems supported at different elevations, the envelope floor response spectrum corresponding to the support locations is used. Westinghouse does not have in their scope of analysis any piping systems interconnected between buildings.

3.7N.3.9 Multiply Supported Equipment Components with Distinct Inputs

When response spectrum methods are used to evaluate Reactor Coolant System primary components interconnected between floors, the procedures in the following paragraphs are used. There are no components in Westinghouse scope of analysis which are connected between buildings. The primary components of the Reactor Coolant System are supported at no more than two floor elevations.

A dynamic response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis is the most severe floor response spectra.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis. The results of the building analysis are reviewed on a mode-by-mode basis to determine the differential motion in each mode. Per ASME Code rules, the stress caused by differential seismic motion is clearly secondary for piping (NB-3650) and component supports (NF-3231). For components, the differential motion is evaluated as a free end displacement, since, per NB-3213.19, examples of a free end displacement are motions "that would occur because of relative thermal expansion of piping, equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping." The effect of the differential motion is to impose a rotation on the component from the building. This motion, then, being a free end displacement and being similar to thermal expansion loads, will cause stresses which will be evaluated with ASME Code methods including the rules of NB-3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7N.3.10 Use of Constant Vertical Static Factors

Constant vertical load factors are not used as the vertical floor response load for the seismic design of safety-related components and equipment within Westinghouse's scope of responsibility.

3.7N.3.11 Torsional Effects of Eccentric Masses

The effect of eccentric masses, such as valves and valve operators, is considered in the seismic piping analyses. These eccentric masses are modeled in the system analysis and the torsional effects caused by them are evaluated and included in the total system response. The total response must meet the limits of the criteria applicable to the safety class of the piping.

3.7N.3.12 Buried Seismic Category I Piping Systems and Tunnels

Refer to Section 3.7B.3.12.

3.7N.3.13 Interaction of Other Piping with Seismic Category I Piping

Refer to Section 3.7B.3.13.

3.7N.3.14 Seismic Analyses for Reactor Internals

The seismic analysis of the reactor internals is conducted in accordance with the guidelines specified in Regulatory Guide 1.92. The seismic analysis determines the response of the reactor internals to Operational Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) vertical and horizontal seismic shock components.

The input response spectra used in the reactor internals analysis are based on the accelerations at the reactor vessel supports taken either from the plant specific spectra or from Westinghouse generic E-spec spectra which envelop the applicable plant specific spectra. The horizontal and vertical seismic analysis use the modal response spectrum method and the WECAN general purpose finite element program to determine the internals response. The method used to obtain the combined response of the modal spectral responses is square-root-of-the-sum-of-the squares (SRSS).

The effect of closely spaced modes is considered using the Ten Percent Method (Regulatory Guide 1.92, Paragraph 1.2.2); however, the effect has been shown to be insignificant. The maximum, or total, seismic response value of the reactor internals is obtained by taking the SRSS of the maximum values of the co-directional responses due to the three components of earthquake motion. In general, this combination is made in the Stress Analysis section of the particular structural component.

The horizontal and vertical seismic models contain 118 and 23 active dynamic degrees of freedom, respectively. Results from the modal analysis of the horizontal and vertical systems indicates, in general, 12 and 3 modes present with frequencies less than 33Hz.

In developing the seismic model of the reactor vessel and internals, a systematic approach was used to ensure that basic fundamental frequencies, i.e., both component and system frequencies, are described and inherent in the mathematical models. The approach used to verify the mathematical modeling of reactor vessel and internals was to compare and require that the system frequencies and mode shapes, from the mathematical models, be in agreement with plant test and scale model test data.

In determining the seismic response of the reactor system, due to the excitation of unidirectional shock spectrum, only those modes contributing to the first 80-90 percent of total system mass were considered in the solution.

Hydrodynamic mass effects, for horizontal and vertical directions, was included in the reactor vessel-internals system models. The numerical values for the various hydrodynamic mass effects within the reactor system were based on scale model and plant tests and applicable analytical expressions; e.g., Fritz, Fritz & Kiss, etc.

The effect of significant nonlinearities in the reactor system, i.e., gaps between reactor vessel and internals, on the seismic response is considered in the system analysis. The nonlinearities due to the gaps are included by determining an effective stiffness at the gap location. The validity of this approach has been investigated and found to be conservative for the frequency response range of the reactor internals.

The structural damping values used in the system seismic analysis are in accordance with Regulatory Guide 1.61 (i.e. 2 and 4 percent for OBE and SSE respectively.)

In addition, the stiffness of the primary piping and the stiffness of reactor vessel supports are considered in the analysis. Coupling effects between the horizontal and vertical directions are small and are not included in the analysis. Diagrams of the internals model are shown in Figures 3.7N-4 and 3.7N-5 for the horizontal and vertical analysis respectively.

3.7N.3.15 Analysis Procedure for Damping

Analysis procedures for damping for subsystems in Westinghouse's scope of responsibility are given in Section 3.7N.2.15.

3.7N.4 SEISMIC INSTRUMENTATION

Refer to Section 3.7B.4.

REFERENCES

1. "Damping values of Nuclear Plant Components," WCAP-7921-AR, May 1974.

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4. Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill, New York, 1964.

5. Thomas, T. H., et al., "Nuclear Reactors and Earthquakes," TID- 7024, U. S. Atomic Energy Commission, Washington, D. C., August 1963.

6. Chan, S. P., Cox, H. L. and Benfield, W. A., "Transient Analysis of Forced Vibration of Complex Structural-Mechanical Systems," J. Royal Aeronautical Society, July 1962.

7. "Safety Analysis of the 8-Gird 17 x 17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP-8236, Addendum 1 (Proprietary), March 1974 and WCAP-8288, Addendum 1 (Non-Proprietary), April 1974.

8. U.S. Nuclear Regulatory Commission Evaluation of Request for Use of ASME Code Cases N-397 and N-411, March 13, 1985.

TABLE 3.7N-1
DAMPING VALUES USED FOR SEISMIC SYSTEMS ANALYSIS FOR
WESTINGHOUSE SUPPLIED EQUIPMENT

Item	Damping (Percent of Critical)	
	Upset Conditions (OBE)	Faulted Condition (SSE, DBA)
Primary coolant loop system components and large piping[a],[b]	2	4
Small piping[b]	1	2
Welded steel structures	2	4
Bolted and/or riveted steel structures	4	7

a) Applicable to 12 inch or larger diameter piping.

b) For piping analyzed by the response spectrum method, ASME Code Case N-411 damping values may also be used in lieu ofthe damping provided in this table.

3.7B SEISMIC DESIGN

3.7B.1 SEISMIC INPUT

3.7B.1.1 Design Response Spectra

Design response spectra for both horizontal and vertical ground motion for the SSE are shown in Figures 3.7B-1 and 3.7B-7, respectively. Response spectra for 2, 5, 7, 10, and 15 percent of critical damping are provided for both the horizontal and vertical motions and are scaled to the maximum ground accelerations of 0.12g and 0.08g selected for the SSE. For the OBE, a scaling factor of 0.5 is applied to the SSE design spectra.

The response spectra are constructed on the basis of the recommendations of Newmark, Blume, and Kapur [14] and conform to the requirements of NRC Regulatory Guide 1.60, Revision 1, with the exception of the 33 Hz to 50 Hz frequency range. In that range, the vertical response spectrum of NRC Regulatory Guide 1.60, Revision 1, differs from the vertical response spectrum of Reference [14]. The effects of this deviation on the results of the analyses of structures and systems are negligible because they only affect the modes which have low amplification. Similarly, the method recommended in Reference [14] for the construction of vertical response spectra leads to a slight deviation from NRC Regulatory Guide 1.60, Revision 1 recommendations for accelerations corresponding to 3.5 Hz. The magnitudes of these differences are negligible.

The response spectra indicate the estimated response of a structure subject to significant nearby earthquake ground motion. The spectra are presented over a range of frequencies corresponding to natural frequencies of structural elements, and they represent the maximum amplitude of motion in structural elements for typical degrees of structural damping. Because the design response spectra have been developed from a large number of real records, following the procedures recommended by Newmark, the effect of strong motion duration and distance of focal depth are included [29].

There are, of course, general associations between duration of strong motion and the size of an event. Longer durations of strong motion are expected with greater-sized earthquakes. Higher frequency accelerations are attenuated with greater distance from the epicenter of the earthquake. These conditions are inherent in the strong motion records which are the source of Newmark's work. In no case are the amplification factors less than one.

3.7B.1.2 Design Time History

One horizontal and one vertical SSE artificial time history have been developed for the design response spectra requirements presented in this section and Section 3.7B.1.1.

As an alternative to a site-dependent analysis, these artificial time history records are suitable for use as base excitations for the dynamic structural analysis.

The mathematical procedures used to generate these artificial time history records can be briefly summarized as follows:

1. The spectral characteristics of the selected site SSE design response spectra are extracted to construct a frequency response function with proper phase factor modification.

2. A fast Fourier transform of the frequency response function is performed to obtain a filter impulse response function.

3. The filter impulse response function is then integrated with a set of pseudorandom numbers to obtain an artificial time history record.

4. A comparison is made between the response spectrum derived from the artificial time history and the site SSE design response spectrum. Any unacceptable deviations are corrected by adding a series of sinusoidal impulses with proper amplitude and phase angles until the desirable fit is achieved.

5. The artificial time history records meet the minimum acceptance criteria given by Table 3.7.1-1 in Section 3.7.1 of the Standard Review Plan.

The response spectra derived from the horizontal artificial time history record and the selected site SSE design response spectra are presented in Figures 3.7B-2 through 3.7B-6, for five structural damping values. The corresponding artificial time history is presented in Figure 3.7B-14. The response spectra from the vertical artificial time history record and the SSE design response spectra are presented on Figures 3.7B-8 through 3.7B-12, and the corresponding artificial time history is presented on Figure 3.7B-19.

Time history durations of approximately 10 sec have been found necessary to allow the modifications of the time histories to match response spectra values at periods of three to four sec. A 10 sec record allows two to three cycles for modification by sinusoidal impulses. A record length of 10.24 sec is obtained because the fast Fourier transform used for this purpose operates on sets of numbers which are as powers to time; i.e., 1024 is equal to two raised to the tenth power.

The artificial time history records are generated at 0.01 sec equal time intervals with a time duration of 10.24 sec. They are in the digitized form of 1024 acceleration values.

New ground motion time history accelerations were generated for the re-analysis of the Fuel Building due to the addition of high density spent fuel storage racks. Three statistically independent acceleration time histories were developed, two in the horizontal direction used as East-West and North-South ground motions and one for the Vertical direction. An iterative method was used to compute ground motion time history accelerations using the design response spectra in Figures 3.7B-3 and 3.7B-9 as target spectra. In each iteration the response spectrum was recomputed from the generated time history accelerations and compared to the target response spectra. Iterations were repeated until a satisfactory match was achieved. The resulting artificial ground acceleration time history motions are 20 seconds long and the time size is 0.005 seconds.

The response spectra resulting from the horizontal artificial time history records and the selected site SSE design response spectra are presented in Figures 3.7B-3A and 3.7B-3B for 5% damping, Figures 3.7B-4A and 3.7B-4B for 7% damping, and Figures 3.7B-5A and 3.7B-5B for 10% damping. The corresponding artificial acceleration time histories are presented in

Figures 3.7B-14A and 3.7B-14B. The response spectra from the vertical acceleration time history record and the SSE design response spectra are presented in Figures 3.7B-9A, 3.7B-10A, and 3.7B-11A for 5%, 7%, and 10% damping, respectively. The corresponding artificial acceleration time history is shown in Figure 3.7B-19A. The artificial acceleration time histories are normalized to 1.0g and 0.67g for the horizontal and vertical directions, respectively. Scaling factors of 0.12 and 0.06 were applied to represent the design maximum ground accelerations described in Section 3.7B.1.1 (i.e., 0.12 times 1.0g and 0.67g results in maximum ground accelerations of 0.12g and 0.08g horizontal and vertical, respectively, for the SSE and 0.06 times 1.0g and 0.67g results in maximum ground accelerations of 0.06g and 0.04g horizontal and vertical, respectively, for the OBE).

Three checks were made to show the validity of the new time history accelerations:

1. Statistical independence of the three artificial acceleration time histories was verified via the normalized correlation coefficients recommended by Clough and Penzien [52].

2. The power spectral density of the generated acceleration time history motions were computed and shown to meet the criteria of Appendix A of Standard Review Plan Section 3.7.1 [46].

   The computed power spectral density function was smoothed in accordance with Standard Review Plan Section 3.7.1 by averaging the computed values in the frequency band of ±20% around the frequency point.

   The computed smoothed power spectral density functions are shown in Figures 3.7B-20 and 3.7B-21 and exceed 80% of the target PSD throughout the frequency range from 0.3 Hz to 24 Hz.

3. Response spectra for each ground motion acceleration were computed at damping levels of 5, 7, and 10%. Response spectra computations were carried out using the method recommended by Nigam and Jennings [48] at 400 equally spaced intervals from 0.8 Hz to 40 Hz.

   The computed spectra envelop the target spectra in the entire frequency range from 0.8 Hz to 33 Hz with only a few minor exceptions. In very few cases the computed spectra dip below the target values. However, these points are acceptable based on the provisions of Standard Review Plan Section 3.7.1. None of the points are below the prescribed allowable of 10 percent below the design response spectra and the number of points below the design response spectra is less than the prescribed maximum of five.

3.7B.1.3 Critical Damping Values

The specific percentages of critical damping values used for Category I structures, systems, and components are based on the materials, stress levels, and type of connections of the particular structure or component. They are determined in accordance with the recommendations of NRC Regulatory Guide 1.61 and Reference [14]. For piping systems analyzed by the response spectrum method, ASME Code Case N-411 damping values may also be used in lieu of the damping values in Regulatory Guide 1.61.

Structure and component damping values used in the response spectrum and time history analyses are given in Table 3.7B-1. With the exception of piping stress analysis problem number 1-081, all non-NSSS piping stress analysis of safety related piping at CPNPP utilizes Code Case N-411. This includes all analysis of new piping and support optimization work, as well as all pipe stress reconciliation work. Damping factors associated with foundation springs are discussed in Section 3.7B.2.4. Damping values used for qualifying Westinghouse piping and equipment are shown in Section 3.7N.

3.7B.1.4 Supporting Media for Seismic Category I Structures

All seismic Category I structures are founded on the firm, unweathered Glen Rose Limestone which constitutes the principal bedrock formation on the site.

Below the Glen Rose unit lies the Twin Mountains Formation, which forms a gradational contact with the Glen Rose unit and is composed principally of sandstone, limestone, and clay stone. The portion of the Glen Rose unit which provides the founding material for the Category I structures consists of argillaceous limestone with lenses and zones of calcereous clay stone. Approximately 150 to 160 ft of this formation is present beneath the lowermost foundation. The upper portion of the Glen Rose unit consists of weathered rock and a soil cover of a few feet. Prevailing rock characteristics are presented in Table 3.7B-3.

The soil cover and the upper 40 ft (approximately) of the Glen Rose Limestone are totally removed by foundation excavation. Thus, all of the moderately-to-severely weathered rock present at the site is removed.

With the exception of the Service Water Intake Structure, no structural backfill is used under or against Category I structures.

More detailed description of the site geology, the subsurface conditions, and the engineering properties of site materials are included in Section 2.5.

Foundation elevations, depths of embedment, total structural heights, and foundation plan dimensions for the Category I structures are presented in Table 3.7B-4.

3.7B.2 SEISMIC SYSTEM ANALYSIS

3.7B.2.1 Seismic Analysis Methods

Methods of seismic analysis used for seismic Category I structures, systems, and components, as well as applicable stress and deformation criteria, and mathematical models, are described in this section.

Seismic analysis of seismic Category I structures, systems, and components is performed by the use of the response spectrum or the time history concept of analysis, or both [28], [30], [35]. The use of the response spectrum concept provides a convenient procedure for seismic analysis. Spectrum analysis uses the natural frequencies, mode shapes, and appropriate modal dampings as a fraction of critical damping, and is an approximate method for determining the seismic response of linear elastic multidegree-of-freedom systems with lumped masses and elastic properties in discrete parts.

In a time history analysis, there are two basic ways of using the time history for linear elastic systems, namely, by a modal analysis time history, which uses the same free vibration characteristics and damping factors as the spectrum analysis, or by solving a system of coupled differential equations of motion by direct numerical integration. In the latter case, the numerical integration using a suitable technique must be performed simultaneously for all of the coupled equations. This procedure is cumbersome, requiring a large amount of computations, and is susceptible to computational difficulties. For example, it is difficult to know how small the time intervals should be to avoid mathematical instability. Furthermore, there is no really satisfactory way to determine all of the damping coefficients in these coupled differential equations of motion. Because of these difficulties, the modal method of analysis is used. Only in the case of nonlinear behavior when structures, systems, and components cannot be regarded as linear elastic, such as springs with nonlinear restoring-force functions and nonlinear elastic properties of materials, is the method of direct numerical integration of coupled differential equations of motion used.

Where the aforementioned methods do not provide reliable results, or where analysis appears impractical, dynamic testing of equipment is performed to ensure functional integrity.

The methods used for seismic analysis of particular seismic Category I structures, systems, and components are summarized in Table 3.7B-2.

It should be noted that the modal analysis time history method is used to generate responses at selected locations, such as the ones required for the development of instructure response spectra. Responses at selected locations resulting from both response spectrum concept and time history are compared. Static loads resulting from a dynamic analysis are used in the design of some structural components such as foundation mats, floors, and shear walls [34].

3.7B.2.1.1 Idealization of Seismic Category I Structures, Systems, and Components

A most important part of seismic analysis is devising a mathematical model that satisfactorily represents the dynamic behavior of a seismic Category I structure, system, or component. The modeling technique used results in mathematical models composed of a network of lumped masses and elastic properties in discrete parts. Normally, characteristic points or nodes are selected so that they coincide with concentrations of mass, e.g., at floors, changes of cross sectional area, or at locations which are important for stiffness. The characteristic points for lumping of the masses of an axisymmetric shell-type structure are selected at the centroids of horizontal cross-sections through individual components of the structure. These centroids lie on the vertical centerline of the structure. Each mass has six degrees of freedom, namely, three translations in the three principal orthogonal directions and three rotations about the three principal orthogonal axes. In general, responses associated with all of these degrees of freedom can be coupled and excited by each direction of seismic motion. Bending and shearing effects are considered in determining the discrete rigidities between the lumped masses.

For all seismic Category I structures except the Service Water Intake Structure, Seismic Category I Tanks, and re-analysis of the Fuel Building for addition of high density spent fuel storage racks, finite element techniques that simulate floor slabs and shear wall assemblies are used to generate the reduced stiffness matrix associated with the number of dynamic degrees of freedom required for the dynamic analysis. The mathematical model for which this reduced matrix is generated consists of lumped masses, viscous dashpots, and elastic properties in discrete parts. For the Service Water Intake Structure, Seismic Category I Tanks, and re-analysis of the Fuel Building, stiffness properties are calculated for the structural elements between

lumped mass elevations using standard structural techniques. The mathematical models representing the seismic Category I structures and the method chosen for the selection of the number of masses are described in Subsection 3.7B.2.1.6.

For ease of computation, the mathematical model is reduced to contain as few dynamic degrees of freedom as feasible so that it can be analyzed successfully by means of algorithms adopted for today's high-speed digital computers.

Foundation structure interaction is represented by decoupled springs, dashpots, and effective masses generally associated with the six degrees of freedom in a global orthogonal system. The methods used to determine the foundation parameters related to torsion, rocking, and translation are described in Subsection 3.7B.2.4.

The re-analysis of the Fuel Building for added mass due to the addition of high density spent fuel storage racks uses the computer code CLASSI to perform the soil structure interaction analysis. The soil was modeled as a horizontally layered viscoelastic system overlying a half-space. The method used to perform this analysis is described in Subsection 3.7B.2.4.

3.7B.2.1.2 Analytical Approach

In order to analyze the response of a linear elastic lumped mass system, the natural frequencies and corresponding mode shapes are first determined. This determination is accomplished by extracting the eigenvalues and associated eigenvectors from a homogeneous system of equations which result from undamped free vibration and are comprised of stiffness or flexibility and mass matrices developed from the mathematical model. These free vibration characteristics are calculated by using any one of the suitable algorithms coded into the computer programs, such as the diagonalization method originated by Jacobi, Householder's tridiagonalization method combined with the Sturm sequence method, and methods such as those used in computer programs presented in Section 3.7B(A). After establishing the free vibration characteristics, such as natural frequencies and associated mode shapes, the next step consists of response computations obtained by using the response spectrum approach or time history analysis or both [28], [30], [31], [35], [38].

1. Response Spectrum Analysis

   The response spectrum analysis is performed using various computer programs consisting of different subroutines developed by Luminant Power or it's engineering services contractor as described in Section 3.7B(A).

   The analysis of the structures founded on bedrock uses spectral values from the free-field horizontal and vertical ground response spectra developed for this site. Spectral values associated with modal dampings and natural frequencies are obtained for each mode. Then the maximum absolute accelerations, inertia forces, shears, moments, and relative displacements are obtained in each mode. The maximum modal responses of all the modes are combined by the square root of the sum of the squares (SRSS), by absolute sum, and by combinations thereof, as discussed in Subsection 3.7B.2.7.

   A separate analysis is made on the model representing the structure founded on bedrock for each of the three orthogonal principal directions of input ground excitations.

Vertical and two horizontal ground excitations are assumed to act simultaneously. Hence, the combined effects of earthquakes on structures, components, or elements are computed by taking the SRSS of the particular effects at any particular point, caused by each of the three components of earthquake motion (two horizontal motions at right angles and one vertical motion).

In the case of shell structures when shell theory is used, maximum stress resultants (membrane shears, moments, and forces), as well as unit stresses and displacements, are obtained. This is accomplished by applying distributed inertia forces and using a suitable computer program.

The total overturning moment at the base of a structure is obtained. The maximum dynamic foundation pressure is evaluated to ensure that it is within permissible limits.

The analysis is performed for both the SSE and OBE unless it is apparent that one of these controls the design.

2. Time History Analysis and Instructure Response Spectra

After the mathematical models of structures are analyzed for their characteristics of free vibration, the time history responses at selected mass points are obtained using the artificial time history ground motion [30], [31], [38]. Derivation of the appropriate time history ground motion is discussed in Subsection 3.7B.1.2. Once the time history response of a selected mass point is generated, the next step is to subject a single-degree-of-freedom system, with the natural frequency range of interest and various damping ratios, to this time history motion. A spectrum response curve is obtained by plotting the maximum acceleration response as ordinates and the corresponding natural periods of the single-degree-of-freedom system as abscissa. The enveloping technique used for the construction of instructure response spectra consists of enveloping the maximum peaks. Since the frequencies of the structures can only be computed approximately because of the linear and nonlinear deformability, the energy dissipation, variation in elastic properties of both structure and foundation, and the idealization of structure with lumped masses and elastic properties in discrete parts, parametric studies are performed in order to take into account these effects for the construction of instructure response spectra. These effects result in the shifting of the resonance peaks of the instructure response spectra. The peaks are widened by at least ±10 percent of the resonance frequencies to account for these effects. The widening exceeds ±10 percent if the parametric studies indicate that such widening is necessary to achieve conservative results. The ground design response spectra and design time history are discussed in Section 3.7B.1.1 and 3.7B.1.2 respectively. Parametric studies and spectra widening are discussed in Subsection 3.7B.2.9.

The Fuel Building was re-analyzed to determine the effects of added mass due to the addition of high density spent fuel storage racks. These analyses were based on best estimate soil properties and the resulting spectrum peak responses were widened by at least ± 15% in accordance with RG 1.122. Best estimate soil properties were used in the re-rack analysis in lieu of parametric variations involving upper and lower bound soil properties. Results of previous studies show that variations in soil properties and building structure stiffness have minimal effect on dynamic response (less than 7%).

Results of the Fuel Building re-analysis demonstrate that the original instructure response spectra largely envelope instructure response spectra considering the effects of high density spent fuel storage. Where the original instructure response spectra were not enveloped, new design instructure response spectra was defined as the envelope of the two as discussed in Section 3.7B.2.5.

The preceding analyses are accomplished by using suitable computer programs as presented in Section 3.7B(A) and in accordance with References [30], [31], [36], and [38].

3.7B.2.1.3 Testing and Analysis for Equipment

Seismic Category I equipment, equipment supports, and components are designed to ensure functional operability during and after an earthquake of magnitude up to and including the SSE (refer to Section 3.2 for the list of seismic Category I mechanical and electrical equipment). The capability of all seismic Category I electrical and mechanical equipment and equipment supports to satisfy this requirement is verified by testing or analysis, or both.

The seismic qualification of seismic Category I electrical equipment and equipment supports is in accordance with IEEE 344-1975. Seismic Category I mechanical equipment and equipment supports are qualified in accordance with the ASME B&PV Code, Section III, Division 1, and the applicable NRC regulatory guides.

When dynamic analysis or testing is employed, the equipment specification or other controlled means furnishes the vendor with appropriate instructure response spectra or acceleration requirements for both SSE and OBE with instructions on their use. In the case where a supporting structure is required to be modeled as an integral part of the equipment, the vendor is furnished with the mathematical model representing the structure consisting of lumped masses and elastic properties in discrete parts. The lumped masses, flexibility or stiffness characteristics, or both, can be presented in matrix form. Any additional information such as the floor motion time history is furnished only upon request.

Where applicable, the response spectrum technique is employed in the dynamic analysis of seismic Category I equipment and components. The time history modal analysis of seismic Category I structures, as previously explained, generates time histories at various support elevations and instructure response spectra for use in analysis of systems and equipment.

In general, at each level of the supporting structure where vital items are located, two horizontal and one vertical response spectra corresponding to coupled translational motions of the supporting structure in the three orthogonal principal directions and the coupled rotational motions of the structure about the three axes for each direction of ground excitation are developed. The instructure response spectra are smoothed so that the response curve is an upper bound envelope of all the acceleration points. Parametric studies and spectra widening are discussed in Subsection 3.7B.2.9.

Generally, multimass presentation of seismic Category I equipment and components, except piping, is used in accordance with the lumped parameter modeling techniques and normal mode theory, as described in the references listed in Section 3.7B. Piping systems are analyzed as described in Subsection 3.7B.3.8.

Simplified analytical models such as one or two degree-of-freedom models are used where they provide a suitable representation of the systems.

Equipment whose lowest dominant natural frequency is 33 Hz or higher is considered rigid. In this case the acceleration of the equipment is assumed to be the same as the zero period acceleration of the appropriate response spectrum.

Where dynamic testing is used to ensure functional integrity, test performance data and results reflect the following:

1. Performance data for equipment which, under the specified conditions, has been subjected to equal or greater dynamic loads than those to be experienced under the specified seismic conditions

2. Test data from previously tested comparable equipment which, under similar conditions, has been subjected to equal or greater dynamic loads than those specified

Mechanical and electrical equipment and components which are required to maintain functional integrity and operability during and after a postulated seismic event are described in Subsection 3.9B.2 and Section 3.10B. Actual testing of such equipment and components to demonstrate seismic adequacy is performed by the suppliers using a shake table, test bed, or other appropriate device.

1. Seismic testing for equipment operability conforms to the following:

    a. A test required to confirm the functional operability of seismic Category I electrical and mechanical equipment and instrumentation during and after an earthquake of magnitude up to and including the SSE is performed. Analysis without testing may be performed only if structural integrity alone can ensure the design intended function. When a complete seismic testing is impracticable, a combination of test and analysis is performed.

    b. The characteristics of the required input motion are specified by one of the following:

        1. Response spectrum

        2. Time history

        Such characteristics, as derived from the structures or systems seismic analysis, are representative of the input motion at the equipment mounting locations.

    c. Where practicable, equipment which is required to function during and/or after an earthquake is tested in the operational condition. Operability is verified during and/or after the testing.

    d. The actual input motion is characterized in the same manner as the required input motion and the conservatism in amplitude and frequency content is demonstrated. The frequency spectrum covers the range from 1 through 33 Hz.

e. Seismic excitations generally have a broad frequency content. Therefore, random vibration input motion is generally used. Single frequency input, such as sine beats, can be applicable provided one of the following conditions is met:

1. The characteristics of the required input motion indicate that the motion is dominated by one frequency, i.e., by structural filtering effects.

2. The anticipated response of the equipment is adequately represented by one mode.

3. The input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra envelope the corresponding response spectra of the individual modes.

f. To account for the simultaneous occurrence of random earthquake motions, multiaxis testing is performed. Test input motions are applied simultaneously in the vertical and each of the horizontal principal directions. Single-axis testing is accepted in accordance with the provisions of IEEE 344-1975, i.e., where justified by demonstrating that earthquake motions at the mounting locations are amplified in one direction, or that the equipment is constrained to respond in one direction, or that coupling between the responses is negligible. In the case of single-frequency input, the time phasing of the inputs in the vertical and horizontal directions is such that a purely rectilinear resultant input is avoided.

The acceptable alternate to this procedure is to test the equipment with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase.

In addition, it is required that the test be repeated with the equipment rotated 90 degrees horizontally.

g. The fixture design meets the following requirements:

1. Simulates the actual service mounting

2. Causes no dynamic coupling to the test item

h. The in situ application of vibratory devices to superimpose the seismic vibratory loadings on the complex active device to provide data for operability testing is acceptable when application is justifiable.

i. The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, and similar items on a prototype basis.

j. Where a random vibration input motion is used, the amplitudes of the excitations are controlled to provide a test response spectrum which meets or exceeds the applicable required response spectra.

2. Seismic design adequacy of supports conforms to the following:

   a. Analyses or tests are performed for all supports of electrical and mechanical equipment and instrumentation to ensure their structural capability to withstand seismic excitation.

      In accordance with the intent of IEEE 344-1975, if supports are similar and justified as such, or if based on engineering investigation, the worst case support is chosen from a group of supports to be qualified and is justified, only one of the similar supports or the worst case support requires a complete dynamic seismic analysis or full scale test, or a combination of both.

      Justification of this procedure is based upon comparison analysis or past experience that the supports to be qualified are similar or that the worst case has been chosen. Upon such justification and dynamic analysis or full scale testing, or a combination of both, of the similar or worst case support, the group of supports being investigated is accepted as seismically qualified.

   b. The analytical results include the following:

      1. The required input motions to the mounted equipment are obtained and characterized in the manner stated in Item 1.b of this subsection.

      2. The combined stresses of the support structures are within the limits of the ASME B&PV Code, Section III, Division 1, Subsection NF, Component Support Structures, or other comparable stress limits.

   c. Supports are tested with equipment installed. If the equipment is inoperative during the support test, the response at the equipment mounting locations is monitored and characterized in the manner stated in Item 1.b. In such a case, equipment is tested separately and the actual input to the equipment is made more conservative in amplitude and frequency content than the monitored response.

   d. The requirements of Items 1.b, 1.d, 1.e, 1.f, and 1.g are applicable when tests are conducted on the equipment supports.

Equipment, when under test, is mounted in a manner that simulates the intended service mounting in its operating condition.

When sinusoidal beat input motion is applicable as per Items 1.e.1), 1.e.2), and 1.e.3), testing is normally performed in two phases as follows:

1. Phase 1 consists of a low-amplitude continuous sweep frequency search, with a sinusoidal steady-state input of at least 0.2g over a minimum frequency range of 1.0 to 33 Hz in order to determine potential resonant regions.

2. Phase 2 consists of sinusoidal beat testing at resonant frequencies determined in Phase 1 with amplitudes correlated to the maximum acceleration of the equipment support. In addition to sinusoidal beat testing at the known resonant frequencies, the test

is also performed at a number of preselected frequencies in the 1.0 to 33 Hz band. This ensures that any shifts in resonant frequencies caused by nonlinear effects are adequately evaluated.

A sinusoidal beat consists of 10 cycles at the frequency being tested with amplitudes of each cycle varying as a sine function from zero to a maximum at the fifth cycle and then decreasing to zero at the tenth cycle. The maximum sine beat amplitude is made to correspond to the maximum zero period SSE acceleration of the structure at the appropriate floor.

A test at any frequency consists of five beats with a pause between beats of approximately two sec or more, as required to ensure no superposition of motion.

Sinusoidal steadystate testing at resonant frequencies rather than sinusoidal beat testing is performed at the option of the equipment supplier.

When sinusoidal beat testing or sinusoidal steady-state testing is used, the equipment is subjected to the input motion for at least the entire duration of an earthquake of 30 sec.

All tests are performed independently for each of the two horizontal directions and the vertical direction. Three-axes simultaneous testing is also required. Satisfactory alternatives as recommended in Section 6.6.6 of IEEE 344-1975, Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, are allowed where three-axis testing is impractical.

Equipment when under test is required to demonstrate a capability to perform its intended function, and sufficient monitoring equipment is used to evaluate performance during test.

Other test procedures proposed by equipment suppliers equivalent to the methods described previously are reviewed by Luminant Power or its engineering services contractor, and used for seismic qualification of equipment when formally approved by Luminant Power or its engineering services contractor.

The equipment supplier is required to document his previous experience in analysis or testing of seismic Category I equipment similar in nature to that being provided.

The equipment supplier is required to furnish documentation to demonstrate that the equipment meets the seismic design criteria established in the specification in both performance and structural integrity.

If proof of performance is obtained by analytical means, the equipment supplier is required to present documentation in a step-by-step form which is readily auditable by persons skilled in such analyses.

If proof of performance is obtained by testing, the test data include detailed information on the following:

1. Equipment identification

2. Test facility (including location)

3. Test equipment

4. Test method

5. Test data (including unsuccessful test of components and subsequent remedial measures)

6. Results and conclusions

7. Attestation

3.7B.2.1.4 Differential Seismic Movement of Interconnected Components

The seismic analysis of seismic Category I subsystems and equipment subjected to differential support motion is performed in three parts using lumped mass mathematical models as follows:

1. Modal response spectrum analysis is performed for all three principal orthogonal directions of support motion for each direction of ground excitation using appropriate instructure response spectra constructed on the basis of superimposing the spectra for all support points and enveloping them as stated in Subsection 3.7B.2.5. The analysis of these subsystems or components follows the same considerations as those described in Subsection 3.7B.2.1 for seismic Category I structures. The vertical analysis is combined with both horizontals, according to the statement in Subsection 3.7B.2.1.2, to produce basic dynamic loading conditions.

2. The same multimass lumped parameter model is subjected to a stress analysis due to differential displacements of the support points. The displacements used are consistent with the directions of structural excitation being considered in the spectrum analysis. This results in basic differential displacement loading conditions.

3. The results obtained from the spectrum analysis and differential displacement analysis are then combined directly. The effects of these loading conditions on the components and the supporting structures are determined.

3.7B.2.1.5 Stress and Deformation Criteria

The ground design response spectra and design time history are discussed in Section 3.7B.1.1 and 3.7B.1.2 respectively.

Primary steady-state stresses including the effects of the normal operating loads plus the OBE loads are maintained well within the elastic limit of the material affected.

For systems and equipment, self-limiting secondary stresses may exceed allowable primary stress to the extent permitted by the appropriate codes. For the OBE, the equipment function is performed without permanent deformation.

Primary steady-state stresses, including the effects of the normal operating loads plus the SSE loads, are limited to prevent loss of function of the equipment. For the purpose of calculation, the no-loss-of-function stresses are limited to 90 percent of the yield strength of the material, except when valid plastic analysis demonstrates structural integrity or when the stress limits are

specifically controlled by an applicable code or standard as committed to by CPNPP for example, see Section 3.9B. Local, self-limiting, secondary stresses may exceed yield stress levels to the extent set forth in the appropriate design standards and codes.

Deformations resulting from the combined influence of normal operating loads and the loads from the SSE are investigated to verify that they do not impair the functional performance required for a safe and orderly shutdown of the plant.

For fatigue analysis required by some codes, the number of expected earthquakes, the duration of strong motion vibration, and the number of cycles the equipment or system is exposed to are evaluated for the OBE, and specified as 600 load cycles.

Because the earthquake transients are part of mechanical loading conditions, the origin of their determination is separate and distinct from those transients which result from fluid pressure and temperature. Therefore, the superposition of the earthquake cycles on the fluid pressure and thermal transients is not considered.

3.7B.2.1.6 Descriptions of Mathematical Models

The mathematical models used for the dynamic analysis of the major structures are described as follows:

1. Model for Containment and Internal Structures

   The mathematical model for the Containment and Internal Structures consists of lumped masses, elastic properties, and viscous dashpots in discrete parts. The number of masses and associated degrees of freedom is kept to a minimum to reduce unnecessary complexity.

   Plan and elevation views of the Containment and Internal Structures are shown on Figures 3.7B-23 and 3.7B-24. The corresponding mathematical models are shown on Figure 3.7B-34. The coordinates of the lumped masses are listed in Table 3.7B-9. The values of the masses and mass moments of inertia associated with the lumped masses of the structures and foundation are presented in Table 3.7B-14. The degrees of freedom assigned to the lumped masses of the structures are identified in Table 3.7B-19.

   The dome and the cylinder of the Containment Structure are simulated by five masses. The Internal Structure, made up of concrete walls, columns, slabs, and beams, is represented by three masses lumped at floor levels. The mat is simulated by a single mass. Each mass is assumed to have six dynamic degrees of freedom, namely, three translations in the three orthogonal principal directions and three rotations about these three axes.

   Two parametric analyses are performed for the Containment Structure where the full thickness (uncracked) and the half thickness (cracked) of the cylindrical wall are considered to be effective.

   Foundation-structure interaction, such as the foundation spring constants, damping ratios associated with each foundation spring, and effective masses of the foundation material, is based on the theory of a circular base resting on elastic half space as described in

Subsection 3.7B.2.4 [1], [2], [23], [32], [37]. These values are determined for all six degrees of freedom of the global orthogonal system. The effect of the embedment is also evaluated. Estimated values, upper bound values, and lower bound values of foundation spring constants used in the parametric analyses described in Subsection 3.7B.2.4 are presented in Table 3.7B-24.

The stiffness or the flexibility matrices of the Containment Structure and Internal Structure are generated using suitable computer programs based on finite element techniques. The Containment Structure wall and dome and Internal Structure walls, columns, and floors are modeled with finite elements. The stiffness matrix which corresponds to the finite element model is then reduced to the number of dynamic degrees of freedom required for the dynamic analysis [3], [23], [30], [38].

Past experience shows that the mathematical model described previously conservatively predicts the dynamic behavior of the actual structure subjected to seismic disturbances.

2. Models for the Safeguards Building, Electrical and Auxiliary Buildings, Fuel Building, Service Water Intake Structure and Category I Tanks.

The mathematical models for these structures are comprised of lumped masses, elastic properties, and dashpots in discrete parts.

The locations of the mass points are chosen at floor levels and points considered of critical interest, such as equipment support levels. Because the structures cannot be considered symmetrical and the torsional modes of vibration can be excited by ground motions, each mass is assumed to have six degrees of freedom, namely, three translations in the orthogonal principal directions and three rotations about the three principal axes which account for the torsional and rocking modes of vibration.

Plan and elevation views of the Safeguards Building, Electrical and Auxiliary Buildings, Fuel Building, and Service Water Intake Structure are shown on Figures 3.7B-25 through 3.7B-33. The corresponding mathematical models are shown on Figures 3.7B-35 through 3.7B-38. The coordinates of the lumped masses are listed in Tables 3.7B-10 through 3.7B-13. The values of the masses and mass moments of inertia associated with the lumped masses of the structure and foundation are presented in Tables 3.7B-15 through 3.7B-18. The degrees of freedom assigned to the lumped masses of the structures are identified in Tables 3.7B-20 through 3.7B-23.

The Fuel Building was re-analyzed to determine the effects of added mass due to the addition of high density spent fuel storage racks. The refined lumped mass model incorporates a new node at the bottom of the spent fuel pools (elevation 819.56 ft) to allow finer distribution of the building mass and more accurate determination of the input motion to the new fuel racks. With the additional node at the 819.56 ft elevation, the fluid mass is distributed between the 819.56 ft, 838.75 ft, and 860 ft elevation. A portion of the fluid is modeled as a sloshing mass at the 806 ft elevation in accordance with TID-7024. As a further refinement, at each floor, four (4) nodes are added at extreme locations and connected rigidly with the floor center of mass. The model incorporates rigid links between the center of rigidity and center of mass at each floor elevation to account for the effects of torsion. The math model used for the Fuel Building re-analysis is shown in

Figure 3.7B-37A. The coordinates of the lumped masses are shown in Table 3.7B-12A and the values of the masses are presented in Table 3.7B-17A.

Foundation spring constants associated with three orthogonal principal translations and two rocking motions about the two horizontal orthogonal axes are determined on the basis of a rectangular or circular base resting on an elastic half space [1], [2], [23], [37]. Torsional foundation spring constants, damping ratios, and effective masses and rotary inertias for foundation below the vibrating mat associated with the foundation springs are determined on the basis of the equivalent radius for the rectangular base of dimensions 2c by 2d using the theory of the elastic half space for a circular footing according to Subsection 3.7B.2.4 [1], [2], [23], [32], [37]. The effects of the embedment of the structures are evaluated and taken into consideration in the analysis. Values of foundation spring constants used in the parametric analyses described in Subsection 3.7B.2.4 are presented in Tables 3.7B-25 through 3.7B-29.

The Fuel Building re-analysis was based on soil-structure interaction analysis using the computer program CLASSI. Soil properties and impedances used in the soil-structure interaction analysis are discussed in Subsection 3.7B.2.4. The effects of embedment are evaluated and taken into consideration in the analysis.

The stiffness matrices of the buildings are generated using suitable computer programs based on finite element techniques or on stiffness properties calculated for the structural elements between lumped mass elevations. For unsymmetric structures the stiffness matrices include the effects of torsional rigidities of shear wall assemblies between floors. The stiffness matrices obtained for finite element models are reduced to conform to the number of degrees-of-freedom of the dynamic models which are used in the dynamic analysis [3], [23], [30], [38].

3.7B.2.2 Natural Frequencies and Response Loads

The natural frequencies and modal participation factors for all modes resulting from the parametric analyses of representative seismic Category I structures are presented in Tables 3.7B-30 through 3.7B-45. Response loads for these structures obtained by the square root of the sum of the squares method (SRSS) are summarized in Tables 3.7B-46 through 3.7B-50 in the form of modal accelerations.

For comparison, envelope values of time history analysis results for the Electrical and Auxiliary Buildings and for the Fuel Building are presented in Tables 3.7B-51 and 3.7B-52.

Seismic loads used for the design of seismic Category I structures are obtained by multiplying the response accelerations with the appropriate masses.

Response spectra at all floors are developed for all seismic Category I structures as indicated in Subsection 3.7B.2.5.

3.7B.2.3 Procedure Used for Modeling

The structures and their contents possess mass which contributes to the inertia loading of the structure. The complexity of the spatial distribution makes it necessary to concentrate the mass at characteristic points or nodes. These points are selected so that they coincide with

concentrations of mass, e.g., at the floors, or with locations which are important for stiffness. In some instances, the nodes are selected at intermediate points of structures and equipment that can be regarded as being of uniform construction. This discretization into characteristic points permits a more accurate prediction of the dynamic behavior of actual structures and equipment.

At each node, the structure or system is given six degrees of freedom (three translation components and three rotation components).

No simplifications aimed at reducing the total number of degrees-of-freedom considered in the analysis are made. All six degrees-of-freedom of each node are treated as generalized displacements for all seismic Category I structures.

The idealization of the mass is performed on the basis of relative displacements. If the horizontal cross-section of the structural component, for example, does not deform significantly, and the contents undergo essentially the same displacement as the structure, all mass in a given place can be represented by a point mass placed at the centroid.

It is not feasible to formulate a mathematical model which would include, in addition to the primary structure, all of the equipment, piping systems, and other lightweight structures. These subsystems are therefore uncoupled from the primary structures and are analyzed by the response spectrum approach procedure. In order to use the spectrum analysis for secondary systems, floor response spectra are developed as described in Subsection 3.7B.2.5.

The criteria employed for system/subsystem decoupling are consistent with the provisions of USNRC Standard Review Plan, Subsection 3.7.2, June 1975. They are based on the mass ratio, R_m of the supported subsystem mass to the corresponding support mass, and the frequency ratio, R_f of the supported subsystem fundamental frequency to the corresponding supporting system dominant frequency such that:

1. If $R_m < 0.01$, decoupling can be done for any R_f

2. If $0.01 < R_m < 0.1$, decoupling can be done if $R_f < 0.8$ or $R_f > 1.25$

3. If $R_m > 0.1$, an approximate model of the subsystem should be included in the primary system model.

where:

$$R_m = \frac{\text{Total mass of supported subsystem}}{\text{Mass of supporting structure}}$$

$$R_f = \frac{\text{Fundamental frequency of the supported system}}{\text{Frequency of the dominant support motion}}$$

The floor response spectra are generated using the mathematical models which consist of the lumped masses computed from tributary structure dead loads, a portion of live loads, and fixed equipment loads. In some cases, the uncoupled mathematical models, with lumped masses

representing the equipment, include the effective masses and flexibility of the supporting structure.

3.7B.2.4 Soil-Structure Interaction

The mathematical model for performing the dynamic analysis of seismic Category I structures supported on the ground is comprised of lumped masses and elastic properties in discrete parts. Because these structures are founded on sound bedrock (Glen Rose Limestone) with shear wave velocities of 5500 to 6000 ft/sec, the foundation-structure interaction is evaluated using the conventional elastic half-space theory in accordance with References [1], [2], [23], [32], and [37]. The justification for the use of this theory is based on the fact that sound bedrock is much closer to being a truly elastic material than any other common foundation material. Using the half-space theory, foundation spring constants with associated effective masses of the rock and damping ratios caused by radiation damping are determined.

Principles of radiation damping apply more correctly for this case than for foundations on other materials. The radiation damping is associated with the energy which is carried away from the immediate vicinity of the foundation by stress waves. In addition to the radiation damping, the internal damping is also determined. The internal damping within the foundation material arises primarily because of a nonlinear effect known as hysteresis; i.e., the stress-strain relationship during unloading is not the same as that during loading. The combined effects of radiation and internal damping are evaluated in accordance with References [32] and [37]. However, damping values used in the analysis are limited to a maximum of 10 percent associated with horizontal and vertical translational foundation springs and five percent for rocking and torsional foundation springs.

Because of the possible variations in measuring the properties of the foundation material, parametric studies are performed on the basis of the best estimate as well as the upper and lower bound estimates of the foundation spring constants with the exception of the Fuel Building and Service Water Intake Structure where only lower bound and upper bound values are used. The range of foundation spring constants considered in the parametric studies accounts for the variations in the properties of the foundation material encountered at the site, as well as the anticipated variations in measuring these properties. These variations are considered to be 10 percent in measuring the mass density and 15 percent in measuring the shear-wave velocity.

The Fuel Building was re-analyzed to determine the effects of added mass due to high density spent fuel storage. The re-analysis incorporates soil-structure interaction using the computer code CLASSI. In the SSI analyses, the soil was modeled as a horizontally layered viscoelastic system overlying a half-space. The solution utilizes soil impedances derived from elastic half-space theory and based on best estimate soil properties.

CLASSI requires as input the modal properties of the building (mode shapes, frequencies, and modal damping ratios) with fixed base. Modal damping ratios for the fixed base building are in accordance with Table 3.7B-1. Modes with frequencies higher than 35 Hz do not need to be considered, however, the number of modes used in the CLASSI analyses account for at least 90% of the total mass.

3.7B.2.4.1 Evaluation of Soil-Structure Interaction Parameters

The foundation-structure interaction in the mathematical models represented by foundation spring constants, associated damping factors, and effective foundation material masses and rotary inertias (in general, corresponding to all six degrees-of-freedom) is evaluated from the theory of a rigid base resting on an elastic half space in accordance with References [1], [2], [23], [32], and [37]. The solution of this problem was originally given by Boussinesq [8].

Based on the preceding theory, the foundation spring constants are obtained from Tables 3.7B-5 and 3.7B-6 in conjunction with Figure 3.7B-39 for rigid circular and rectangular bases resting on the surface of an elastic half-space. An approximate evaluation is made by using an equivalent circular base having the same second moment about its centroidal axis for the contact area. On this basis, the equivalent radius for the rectangular base of dimensions 2c by 2d is determined by the following equation:

$$r_0 = \sqrt[4]{\frac{8dc(c^2+d^2)}{3\pi}} \qquad (3.7B\ 1)$$

The determination of radiation damping (see Table 3.7B-7), occasionally called geometric damping, caused by foundation structure interaction is also based on vibrations of a rigid circular base on the elastic half-space.

The internal damping caused by energy loss during stress reversals is evaluated on an empirical basis.

Combined effects of radiation and internal damping are taken into account by direct addition of the two values of damping. For rotational motions, the radiation damping is low, and the internal damping can be a significant part of the total damping. However, for translational motions, radiation damping is much greater than internal damping.

There are practical reasons why the damping shown by the theory may not be realistic in actual mass foundation systems. Therefore, damping values that are used in the dynamic analysis are limited to the maximum values set forth in Subsection 3.7B.2.4.

The effective masses and rotary inertias of the foundation material are estimated on the basis of the theory for the elastic half-space.

Effective mass and mass moment of inertia for foundation material below a vibrating footing are estimated from the following formulas:

$$M_{eff} = \propto \bullet \rho \bullet r_0^{\,3} \qquad (3.7B\text{-}2)$$

and

$$I_{eff} = \beta \bullet \rho \bullet r_0^{\,5} \qquad (3.7B\text{-}3)$$

where

M_{eff}	denotes the effective mass of the foundation material.
I_{eff}	is the effective rotary inertia of the foundation material.
ρ	is the mass density of the foundation material.
$\propto$ and β	are the coefficients given by theory and shown in Table 3.7B-8 [2].

For rectangular footings of dimensions 2c by 2d, the equivalent radii are obtained for torsion from Equation 3.7B-1 and for translation and rocking from the following equations:

For translation:

$$r_0 \sqrt{\frac{4cd}{\pi}} \qquad (3.7B\text{-}4)$$

For rocking:

$$r_0 = \sqrt[4]{\frac{16cd^3}{3\pi}} \qquad (3.7B\text{-}5)$$

The values of effective foundation masses and mass moments of inertia determined for representative seismic Category I structures are presented in Tables 3.7B-14 through 3.7B-18. They are calculated based upon the assumption that the centers of gravity of the effective soil masses are located on the top of the soil at the centroid of the soil mat contact areas. These values are then added to the lumped mass points representing the building foundations.

Foundation embedment depths, including a description of the foundation medium and its properties, are given in Subsection 3.7B.1.4.

The effect of embedment on foundation rigidities is also determined approximately from the theory of elastic half-space. For example, the torsional rigidity and the rigidity in horizontal translation are determined on the basis of the assumption that the vertical contact area, engaging the elastic half-space, is an imaginary area of rectangular shape whose vertical dimension is twice that of the actual contact area of embedment. The rigidities obtained by the foregoing procedure are divided in half to account for the actual vertical contact area of embedment.

The equations used for evaluating translational and rotational spring constants for embedment are presented in Subsection 3.7B.2.4.2.

The additional nomenclature used in Tables 3.7B-5, 3.7B-6, 3.7B-7, and 3.7B-8 is as follows:

E and G are the modulus of elasticity and the shear modulus of the foundation material, respectively. Nu is the Poisson's ratio of the foundation material.

The shear modulus is obtained from the field-measured, shear-wave velocity C and mass density of the foundation material from the equation

$$G = C_s^2 \cdot \rho \qquad (3.7B\text{-}6)$$

The methods of determining the shear-wave velocity are discussed in Section 2.5.4.4. Strains developed by these methods are in the order of 10^{-5} to 10^{-3} percent. Based on the shear modulus of the competent rock on which the structure is founded (8×10^5 lb/in^2), the estimated strains caused by a seismic event are of the same order of magnitude. In this range of strains, the shear modulus of rock is relatively independent of the strain levels.

The analysis of all CPNPP seismic Category I structures is based upon a rigid mat approach, and the spring constants of the foundation material are determined by means of the formulas listed in Table 3.7B-5 and Table 3.7B-6 and by means of the coefficients in Figure 3.7B-39.

The CLASSI analyses used for the Fuel Building re-analysis were carried out based on a surface founded structure, rigid foundation, layered soil over a half-space, and input motions applies at the free field surface level.

CLASSI is based on a specialized form of substructuring which uses the finite element method to perform the detailed analysis of the superstructure and uses the continuum mechanics method to calculate the interaction of the foundation with the soil medium and with incident seismic waves. These substructuring procedures are made possible by balancing the forces and moments at the foundation, which serves as the common ground for both the superstructure and the soil medium.

The CLASSI substructure approach divides the SSI problem into the following three steps:

1. Determination of the foundation scattering matrices.

2. Determination of the frequency-dependent impedance functions.

3. Analysis of the coupled soil-structure system, using results from steps 1 and 2 and the dynamic properties of the structure.

In the first step, CLASSI evaluates the harmonic response of the rigid, massless foundation bonded to the soil and subjected to a given incident seismic wave in the absence of the superstructure. The free field motion is then used in conjunction with the complex, frequency-dependent scattering matrix in order to determine the foundation input motion. The foundation input motion corresponds to the response of the rigid, massless foundation to the seismic environment described by the free-field in the absence of the superstructure. The response of the rigid massless foundation to the seismic excitation can be described by the six component vector:

$$U^*_o = (\Delta^*_X, \Delta^*_Y, \Delta^*_Z, \theta^*_X, \theta^*_Y, \theta^*_Z)^T$$

In which $\Delta^*_X, \Delta^*_Y, \Delta^*_Z$ represents the translational components of the response and $\theta^*_X, \theta^*_Y, \theta^*_Z$ represent the rotational components of the response.

The foundation input motion U^*_o is related to the free-field ground motion by means of the complex-valued, frequency dependent scattering matrix [S(w)]:

$U^*_o = [S(w)]\ f(w)$

Where the vector f(w) is the complex Fourier transform of the free-field ground motion. At a given frequency, w, each complex number in f(w) corresponds to the amplitude and phase of a wave component of the free-field motion. Each column of the scattering matrix [S(w)] represents the response of a massless rigid foundation to a given incident wave of unit amplitude. The matrix product [S(w)] f(w) is therefore the response of the rigid massless foundation to a particular free-field motion.

For a surface-founded rigid foundation subjected to vertically propagating shear or compressional waves, the response of the foundation includes only translational components with amplitudes equal to those of the free-field motion on the ground surface. However, if the foundation is embedded, a horizontal component of the control motion consisting of vertically propagating shear waves produces both a horizontal translation and a rocking motion of the massless foundation. This is primarily due to the scattering waves from the soil-foundation interface and the kinematic constraints imposed on the soil by the rigid foundation.

Input for the first step in the SSI analysis consists of the properties of the underlying soil layers and the frequency points for Green's function. Soil properties used in the SSI analysis are best estimate and are shown in Table 3.7B-3A.

In the second step, the foundation impedances corresponding to a rigid foundation on a uniform or layered viscoelastic media are developed. The foundation impedances are complex-valued, frequency-dependent functions which relate the dynamic forces that the foundation exerts on the soil to the resulting soil displacements, i.e.:

$F_S(w) = [K(w)]\ U_S$

Where $F_S(w)$ represents the generalized forces, [K(w)] is the complex impedance matrix, and U_S represents the generalized displacements. The real part of the complex impedance matrix represents the stiffness of the soil and the imaginary part represents the energy dissipation of the soil, including both radiation and material damping.

In the third step, analysis of the coupled soil-structure system is carried out in the frequency domain. The impedances and scattering matrices are used to solve the equations of the coupled soil-structure system. For this step, the dynamic characteristics of the structure are used to reduce the effects of the superstructure to six dynamic inertial parameters for each mode and a rigid body mass matrix of the structure about a reference point on the foundation where the SSI response is determined. Once the motion of the foundation has been obtained, the time history response at any level of the structure is computed using Fourier transform techniques.

The Fuel Building has an irregular basemat which exhibits a range of embedment from five (5) to thirty 30) feet. Soil properties were adjusted to account for embedment as described in Subsection 3.7B.2.4.2. Vertical and horizontal time histories were generated by vertically propagating waves; compressional (P) waves for the vertical earthquake and shear waves for the

horizontal earthquake. Translational acceleration time histories were calculated for the extreme location nodes at each floor.

3.7B.2.4.2 Effect of Embedment on Foundation Rigidities

The effect of the embedment on foundation rigidities is based on the conservative assumption that there is no friction between the vertical contact surfaces of the structures and the foundation medium. Therefore, the effect of the embedment on vertical translational rigidity and torsional rigidity, in the case of circular mats and axisymmetric shells of revolution structures, is neglected. The effect of the embedment on rocking rigidities, about the two horizontal axes located at the base, is normally small and is also neglected, unless a significant contribution to the total rocking rigidities due to this cause is expected to play an important role in the analysis and design.

The effect of the embedment on foundation torsional rigidity due to the two opposite, vertical contact surfaces of a structure having a rectangular shape is obtained, according to References [23] and [33], from the following formula:

$$k_\theta = \frac{G}{1-\nu}\beta\psi 4hd^2 \qquad (3.7B\text{-}13)$$

where

h	=	the depth of the embedment
2d	=	the length of the contact surface
G	=	the rigidity modulus
ν	=	the Poisson's ratio
$\beta\Psi$	=	the value given in Figure 3.7B-39 for various values of d/h.

For example, for h = 20 ft and d = 100 ft, d/h = 5; this corresponds to $\beta\Psi$ which equals 0.9 in Figure 3.7B-39. Then, the foundation torsional rigidity constant is:

$$k_e = \frac{G}{1-\nu}0.9x4x20x100^2 = 720{,}000\frac{G}{1-\nu}$$

This torsional rigidity is added to the torsional rigidity associated with the other two opposite, vertical surfaces and the torsional rigidity associated with the horizontal contact surface obtained in accordance with Subsection 3.7B.2.4.

The effect of the embedment on horizontal translational rigidity is given by the following formula:

$$k_x = \frac{G}{1-\nu}\beta_z\sqrt{hd} \qquad (3.7B\text{-}14)$$

where β_z is the value given in Figure 3.7B-39 for various values of d/h.

Taking the same example as for torsional rigidity, the following horizontal translational rigidity constant is obtained for the value of β_z = 2.5 resulting from Figure 3.7B-39:

$$k_\Psi = \frac{G}{1-\nu}(2.5\sqrt{20x100}) = 112\frac{G}{1-\nu}$$

This horizontal translational rigidity is added to the horizontal translational rigidity corresponding to the horizontal contact surface obtained as presented in Subsection 3.7B.2.4.

A matter of practical interest is that the rigidities in torsion and horizontal translation due to the effects of embedment, obtained on the basis of the elastic half-space theory, are essentially in agreement with the rigidities obtained using References [39] and [40] when the sloping planes of the effective foundation zone are assumed to slope at an angle of 30 degrees. It should be noted that these values are also consistent for practical purposes with the values obtained from Reference [7].

Where the effect of embedment on rocking rigidities plays an important role in the analysis and design of seismic Category I structures, equipment, systems, and components, the rocking rigidity constants for embedment are based on the conservative assumption that the side pressure distribution is uniform due to horizontal load. In this case, the horizontal translational spring constant k_x is obtained from the formula presented in Table 3.7B-6. Then, the rocking rigidity constant due to two opposite vertical contact surfaces of the embedment is obtained using Equation 3.7B-9 of Subsection 3.7B.2.4 as follows:

$$k_\Psi = \frac{1}{12}\frac{G}{1-\nu}\beta_z h^2\sqrt{2hd} \qquad (3.7B\text{-}15)$$

where β_z is a value given in Figure 3.7B-39 for various values of 2d/h.

For example, the rocking rigidity constant corresponding to a length of embedment 2d = 200 ft and a depth of embedment h = 20 ft (β_z = 2.85) is:

$$k_\Psi = \frac{1}{12}\frac{G}{1-\nu}(2.85x400\sqrt{4{,}000}) = 6000\frac{G}{1-\nu}$$

It should be noted that this value can be insignificant compared to the rocking rigidity constant associated with the horizontal contact surface of the base with dimensions 2c by 2d. For example, assume that the width of the base mat is 2c = 100 ft (c/d = 0.5); then, the rocking rigidity constant corresponding to the horizontal contact surface is:

$$k_{\Psi b} = \frac{G}{1-\nu}(0.42x8x100x50^2) = 840{,}000\frac{G}{1-\nu}$$

This value is much larger than the value of rocking rigidity constant obtained for the effect of embedment. Therefore, in this case, the effect of the embedment on rocking rigidity is neglected.

In reality, the rocking rigidity constant for embedment is higher in value than the one obtained here. Perhaps a more realistic value can be obtained by assuming that the vertical contact surface of the embedment with the depth h has a mirror image surface with the depth of 2h. Then half of the value for rocking rigidity constant based on the elastic half space theory seems to be more appropriate when the ratio of the actual depth of embedment to the length of embedment is less than unity. For example, using this approach, the following value for rocking rigidity constant for embedment is obtained:

$$k_{\Psi} = \frac{G}{1-\nu} 0.4 \cdot 8 \cdot 100 \cdot 20^2 x \frac{1}{2} = 64{,}000 \frac{G}{1-\nu}$$

Incidentally, this value and the values obtained for the ratios of the depth to the length of embedment less than one are in close agreement with the values obtained on the basis of the approach to the problem for cohesive soils as presented in References [39] and [40]. These values also compare well for practical purposes with the ones obtained using formulation presented in Reference [7].

For the dynamic analysis of seismic Category I structures which have relatively shallow depths of embedment (such as the Safeguards, Electrical and Auxiliary, and Fuel buildings), the effect of embedment on rotational foundation rigidities (torsion and rocking) is negligible. The Service Water Intake Structure (SWIS), which has a greater depth of embedment, is analyzed by including the effects of embedment according to procedures recommended in References 42, 43, 44 and 45. The Service Water Intake Structure is basically socketed into rock with soil backfill on three sides above the top of rock. The embedment effects were calculated for all rock and then for all soil above the founding levels and an average effect representative of the actual soil/rock profile was selected. Although the Fuel Building has a relatively shallow embedment, the affects of embedment were considered in the re-analysis for addition of high density spent fuel storage racks. The procedures recommended in References 42, 43, 44, and 45 were used to evaluate the effects of embedment for the re-analysis.

3.7B.2.5 Development of Floor Response Spectra

The methods of seismic analysis are covered in Subsection 3.7B.2.1. The response spectrum method for the development of instructure response spectra is not used.

Instructure response spectra at selected locations of interest are developed on the basis of computed responses to an artificial time history input of ground motion. The time history of the simulated earthquake ground motion is developed to be compatible to the given ground response spectra. Having established the time history of the ground motion, the lumped mass mathematical models of seismic Category I structures are analyzed and time histories at desired masses lumped at floor levels or any other location of interest are generated. Once the time history of the floor motion is obtained, the next step consists of subjecting a single degree-of-freedom system with the natural frequency range of interest and various damping ratios to the floor time history motion. The maximum acceleration responses obtained are then plotted as ordinates and the corresponding natural periods of the single oscillators are plotted as abscissa. The envelope of maximum peaks is used for the construction of instructure response spectra.

In constructing instructure response spectra, uncertainties inherent to the analysis, such as the material properties of the foundation material and the structures, damping values, soil structure interaction, approximations in the modeling techniques, and computation of structure natural frequencies, are accounted for by parametric variations incorporated into the analysis and by broadening of the peaks of the resulting envelope response spectra as described in Subsection 3.7B.2.9.

The procedure of parametric variations consists of evaluating and using in the dynamic analysis lower bound, best estimate, and upper bound values for the foundation spring constants in the case of all seismic Category I structures with the exception of the Fuel Building and the Service Water Intake Structure where only lower bound and upper bound values are used. In addition, the analysis of the Containment Building is performed for each set of foundation spring constants by considering a cracked and an uncracked Containment wall.

Responses including translational and rotational effects are calculated for each structure at each center of mass, and transferred to the points farthest from the center of mass. Responses in each direction from each of the three directions of earthquake input are calculated separately, and resulting response spectra are combined by the square root of the sum of the squares (SRSS) method to calculate the total response.

Typical response spectra for the Category I Buildings are presented in Figures 3.7B-41 through 3.7B-50C. Curves Ax, Ay, and Az represent the spectra in the X, Y, and Z directions for the combined effect of the three simultaneous earthquakes. The coupling effects of the nonsymmetric structure are included. These design spectra were generated and peak broadened by computer and are therefore labelled refined. Some other design spectra were generated by computer but not peak broadened by computer and therefore have extra conservatism due to the hand smoothing technique.

The Fuel Building was re-analyzed to determine the effects of added mass due to the addition of high density spent fuel storage racks. The acceleration response spectra of the CLASSI output time histories were calculated using the computer program RESPEC. The program uses the procedure recommended by Nigam and Jennings [48]. The model for the Fuel Building re-analysis incorporated node points at extreme locations at each floor level. At each extreme floor location, the generated spectra from the soil structure interaction analysis for each response direction were combined by the square root of the sum of the squares (SRSS) method. This process combined the “X” direction response due to the “X” direction input motion along with the “X” direction responses from the “Y” and “Z” direction input motions, and was repeated for the “Y” and “Z” direction responses. This method is acceptable because of the statistical independence of the ground motion time history accelerations.

Best estimate soil properties were used in the re-analysis in lieu of parametric variations involving upper and lower bound soil properties. The resulting instructure response spectra was combined with the original instructure response spectra to define new enveloped design response spectra for the Fuel Building. The resulting response spectra is dominated by the original analysis as shown in Figures 3.7B-50AA and 3.7B-50AB. The response spectra shown in Figure 3.7B-50A completely envelopes the re-rack spectra and is not affected.

For certain special subsystems such as the RCL subsystem, response spectra at the exact locations of the subsystems considered (e.g., at the steam generator support or the reactor nozzle) are developed as follows: Floor time histories for the three translational and three

rotational degrees-of-freedom and for each earthquake excitation (SSE and OBE) are derived at the nodes corresponding to the floors which contain the selected locations. Response spectra are developed at these nodes by subjecting a single-degree-of-freedom system with the natural frequency range of interest and various damping ratios to the floor time history motions obtained. The response spectra at the selected points are then developed by rigid body transformations.

Figures 3.7B-51, 3.7B-52, and 3.7B-53 represent the response spectra of translational accelerations in three orthogonal directions at the location of the outermost support of the steam generator for two percent equipment damping and for SSE excitations in X, Y, and Z directions, respectively.

3.7B.2.6 Three Components of Earthquake Motion

The three orthogonal components of the design earthquake motion are assumed to act simultaneously. The combined responses (shears, moments, deflections, and so forth) of structures, components, and elements to the simultaneous application of the two horizontal and one vertical ground excitations are obtained by means of the SRSS method because it is considered unlikely that the peak values of the responses from ground excitations in different directions can coincide. This procedure is in conformance with the recommendations of NRC Regulatory Guide 1.92.

3.7B.2.7 Combination of Modal Responses

When the response spectrum concept of analysis is used, only the maximum modal responses are known and the phasing of modes cannot be determined as in the time history analysis. Therefore, the total response at a point in the multi-degree-of-freedom system can only be approximated. The maximum modal responses are normally combined by SRSS, by absolute sum, or by combinations thereof.

The method of combining maximum modal responses is not straightforward. When frequencies of the modes are closely spaced (differences of 10 percent in frequency of the lower mode), the absolute sum procedure of combining the responses in these modes is used.

When the absolute sum procedure for combining some of the modal responses is used, the total maximum response is obtained by treating the responses resulting from the absolute sum as pseudomodal responses and combining them with all other modal responses in an SRSS manner. This procedure conforms to the recommendations of NRC Regulatory Guide 1.92. When additional conservatism is desired, the total maximum response is obtained by adding the values of the responses resulting from the absolute sum to the SRSS value of the rest of the modal responses.

3.7B.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

A number of structures such as the Turbine Building, the Switchgear Buildings, the Circulating Water Intake and Discharge Structures, the Maintenance Building, and the Administration Building are designated as non-Category I.

The only non-Category I structures which are adjacent to any seismic Category I structure are the Turbine Building and the Switchgear Buildings. These structures do not share a common mat with the adjacent seismic Category I structure, and all structures are founded on firm rock.

Therefore, there is no possible interaction of non-Category I structures with seismic Category I structures resulting from seismic motion. Sufficient space is provided between the Turbine and Switchgear Buildings and the adjacent seismic Category I structure so as to prevent contact because of deformations occurring in the structures during a seismic event.

The possibility of structural failure during a seismic event is considered for the Turbine Building. Structural failure in the direction of the adjacent seismic Category I structure is prevented by the bearing of the mezzanine and operating floor slabs on the concrete turbine generator pedestal. The Switchgear Buildings are design to withstand a seismic event equal to the SSE.

The seismic Category II high energy piping segments which are located inside the Turbine Building, and attached to a seismic Category I structure are shown by analysis to remain undamaged by non-Category I structures and components during a seismic event. The piping segments involved are:

- portion of the Steam Generator Blowdown line (8-SB-2-060-1302-5)

- portion of the Heater Drain line (8-HD-2-069-302-5)

Non-Category I equipment and components located in seismic Category I buildings are investigated by analysis or testing, or both, to ensure that under the prescribed earthquake loading, structural integrity is maintained, or the non-Category I equipment and components do not adversely affect the integrity or operability, or both, of any designated seismic Category I structure, equipment, or component to the extent that these seismic Category I items cannot perform their required functions.

3.7B.2.9 Effects of Parameter Variations on Floor Response Spectra

The instructure response spectra are developed using the time history of the instructure motion resulting from the ground motion time history. The instructure response is evaluated by performing time history modal analysis on a lumped-mass mathematical model which simulates the structure and foundation-structure interaction. Because of the uncertainties associated with the energy dissipation, the variation in elastic properties of both structure and foundation, the idealization of structure with lumped masses and elastic properties in discrete parts, and the frequency content and amplitude modulation of simulated ground motion, the free vibration characteristics and the response of structures can only be approximately computed. Parametric studies and conservative assumptions are made to take into account these uncertainties in construction of instructure response spectra.

The amplifications at resonance peaks of the instructure response spectra are generally not sensitive to frequency-shifting because the eigenvalues and eigenvectors of the structure do not change appreciably with small to moderate changes of mass or flexibility and the amplification region of the free field is very wide at dominant structure modes.

Damping factors play an important role in determining amplification of the structure; the smaller the damping, the greater the amplification. Therefore, the damping values are determined conservatively as discussed and presented in Subsections 3.7B.1.3, 3.7B.2.4, and 3.7B.2.15. The conservatism of the amplification is also reflected in the chosen artificial ground motion.

The enveloping technique used for the construction of instructure response spectra consists of enveloping the maximum peaks. Since the frequencies of the structures can only be computed approximately because of the linear and nonlinear deformability, the energy dissipation, variation in elastic properties of both structure and foundation, and the idealization of structure with lumped masses and elastic properties in discrete parts, parametric studies are performed in order to take into account these effects for the construction of instructure response spectra. These effects result in the shifting of the resonance peaks of the instructure response spectra. The peaks are widened by at least ±10 percent of the resonance frequencies to account for these effects. The widening exceeds ±10 percent if the parametric studies indicate that such widening is necessary to achieve conservative results. The ground design response spectra and design time history are discussed in Section 3.7B.1.1 and 3.7B.1.2 respectively.

The Fuel Building was re-analyzed to determine the effects of added mass due to the addition of high density spent fuel storage racks. These analyses were based on best estimate soil properties and the resulting spectrum peak responses were widened by at least ± 15% in accordance with RG 1.122. Best estimate soil properties were used in the re-rack analysis in lieu of parametric variations involving upper and lower bound soil properties. Results of previous studies show that variations in soil properties and building structure stiffness have minimal effect on dynamic response (less than 7%). Widening the peaks of instructure spectra based on best estimate soil properties by ± 15% provides additional margin in the design.

The preceding analysis are accomplished by using suitable computer programs as presented in Section 3.7B(A) and in accordance with References [30], [31], [36], and [38].

3.7B.2.10 Use of Constant Vertical Static Factors

Constant static factors such as vertical response loads for the seismic design of seismic Category I structures, systems, and components are not used. Instead, multimass dynamic analysis for both horizontal and vertical directions of excitation is performed as described in Subsection 3.7B.2.1.

3.7B.2.11 Method Used to Account for Torsional Effects

The methods of seismic analysis of structures are presented in Subsection 3.7B.2.1. The torsional effects on structures depend on the geometric configuration of the structure, the elastic properties of the material of the structure, and the foundation structure interaction.

Torsional effects in unsymmetric structures are induced primarily because the centroids of masses which simulate floor slabs and portions of the shear walls between the adjacent floors do not coincide with the shear centers of the shear wall assemblies.

A finite element computer program is used for the seismic Category I structures except the Service Water Intake Structure, seismic Category I Tanks, and re-analysis of the Fuel Building for the addition of high density spent fuel storage racks. The structure walls and floors are modeled with finite elements. The stiffness matrix corresponding to the finite element model is reduced to the number of dynamic degrees-of-freedom required for the dynamic analysis in accordance with References [3], [23], and [38]. The approach used in the seismic analysis accounts for all torsional effects. For the Service Water Intake Structure, seismic Category I tanks, and re-analysis of the Fuel Building, stiffness properties are calculated for the structural elements between lumped mass elevations using standard structural techniques. Static factors are not

used nor are any other approximate methods used in lieu of a combined vertical, horizontal, and torsional system dynamic analysis to account for torsional accelerations in the seismic design of seismic Category I structures.

In the design of the Seismic Category I structures, shears resulting from torsion were computed using the larger of either the actual computed eccentricity at each floor elevation or 5% of the maximum building dimension (normal to the direction of excitation).

For seismic Category I structures, the mathematical model is normally composed of masses lumped at floor levels. Each mass is assumed to have six degrees-of-freedom; namely, three translations in the three orthogonal directions and three rotations about these axes which account for the rotational modes of vibration. Torsional rigidities of shear wall assemblies between the floors are determined and taken into account in the analysis. Torsional spring constants of the foundation, associated damping ratios, and effective inertia of the foundation are determined on the basis presented in Subsection 3.7B.2.4. The model used for the re-analysis of the Fuel Building incorporates offsets between the center of mass and the center of rigidity to account for torsional effects.

3.7B.2.12 Comparison of Responses

The results of the response spectrum analysis are used for the design of all seismic Category I structures. The procedure for combining modal responses is presented in Subsection 3.7B.2.7.

The time history method of analysis is used to generate time histories of the instructure support motion at selected critical locations of interest. The maximum responses attained from the time history analysis are compared with the ones resulting from the spectrum analysis. This comparison is used to check both analyses and to verify the conservatism of the procedure for combining modal responses in the spectrum analysis. Responses at selected points in the seismic Category I structures resulting from both response spectrum concept and the time history analysis are listed in Tables 3.7B-46 through 3.7B-52.

3.7B.2.13 Methods for Seismic Analysis of Dams

The Safe Shutdown Impoundment (SSI) Dam impounds water which serves as the ultimate heat sink for the power plant.

The SSI Dam is analyzed using a two-dimensional finite element model in which the dam is assumed to behave elastically and the elastic continuum is modeled as an assemblage of discrete elements connected at a set of common nodal points [25]. The elastic properties are selected based on predictions of the strain distribution in the dam during an earthquake.

Hydro-dynamic forces are included, using one-dimensional elements to model the water [26]. The effect of dam-foundation interaction is studied by extending two-dimensional elements into the foundation. Vertical as well as horizontal ground motion is considered.

The finite element representation yields the stiffness and mass matrices for the structure. The resulting eigenvalue problem for undamped free vibration is solved using conventional eigenvalue techniques [27] to give the natural frequencies and mode shapes for the dam. The mode shapes and spectral response curves are used in a conventional modal superposition method of analysis in which damping coefficients are introduced into the various modes.

It should be noted that the main dam is not considered as a seismic Category I structure. Therefore, no seismic analysis of the main dam is performed.

3.7B.2.14 Determination of Seismic Category I Structure Overturning Moments

A description of the dynamic methods and procedures used to determine seismic Category I structure overturning moments is provided in this subsection. A description of the procedures used to account for foundation reactions and vertical earthquake effects is also included.

All dynamic analyses performed for seismic Category I structures are based on the assumption that the structures can be simulated by mathematical models corresponding to linear elastic lumped systems.

A coefficient of stability C_{st} and an eccentricity e shown on Figure 3.7B-40, which govern the evaluation of the dynamic contribution to the foundation pressure beneath the base, are determined.

The degree of stability is dependent on the coefficient of stability C_{st}, which is defined as the ratio of the resisting moment to the overturning moment taken about the extreme point A of the foundation (see Figure 3.7B-40) as expressed in the following equation:

$$C_{st} = \frac{Wa}{M_o} \qquad (3.7B\text{-}22)$$

where

- W = the weight of the entire structure
- a = the horizontal distance from the center of gravity of the entire structure to the point of rotation designated as A on Figure 3.7B-40
- M_o = the total overturning moment about point A induced by seismic disturbances.

The overturning moment in each significant structure mode is given as:

$$M_i = \sum_{n=1}^{N} M_{i,n} + \sum_{n=1}^{N} H_{i,n}\bar{Y}_n + \sum_{n=1}^{N} V_{i,n}\bar{X}_n \qquad (3.7B\text{-}23)$$

where

- N = the total number of nodes in the dynamic model of the structure
- $M_{i,n}$ = the modal inertia moment at each node
- $H_{i,n}$ = the horizontal modal inertia force at each node

$V_{i,n}$ = the vertical modal inertia force at each node

$\bar{X}_n$ = the horizontal distances from the lines of action of the vertical inertia forces to the point of rotation A

$\bar{Y}_n$ = the vertical distances from the lines of action of the horizontal inertia forces to the point of rotation A

The total overturning moment M_o in the stability Equation 3.7B-22 is obtained from the combined effect of all dominant structure modes. The modal overturning moments for each direction of ground excitation are combined, as discussed in Subsection 3.7B.2.7. The combined overturning moments resulting from the vertical and the two horizontal ground motions are calculated by the SRSS combination. In addition, the overturning moment resulting from the vertical ground motion is combined with the larger of the two overturning moments resulting from the two horizontal ground motions by absolute sum combination. Both results are compared and the stability of the structures is checked based upon the most critical of the two cases. For a conservative evaluation of stability against overturning, the vertical seismic forces acting on the seismic Category I systems are considered to act in an upward direction and thus combined with the lateral forces.

An alternate method for calculating the total overturning moment M_o is as follows:

Horizontal and vertical inertia forces are obtained by multiplying the building mass at each story level by the corresponding building acceleration. The building acceleration used considers the three directions of earthquake motion, (i.e., the square root of the sum of the squares of the maximum representative accelerations of the codirectional accelerations caused by each of the three components of earthquake motion). The overturning moment due to horizontal inertia forces is determined by multiplying the individual horizontal inertia force by the vertical distance from it's center of mass to the point of rotation. Similarly, the overturning moment due to vertical inertia is found by multiplying the individual vertical inertia force by the horizontal distance from its center of mass to the point of rotation. The total overturning for the structure, in a particular direction, is found by summing the individual overturning moments due to the horizontal inertia force in the corresponding direction and the overturning moment due to the vertical inertia force.

The criteria used to select an acceptable structural configuration are as follows:

1. The stability coefficient C_{st} is not less than 1.10 for the SSE.

2. Maximum static and dynamic foundation pressures remain within the allowable ultimate capacity.

To calculate dynamic foundation pressures, the resultant reaction R and the eccentricity e shown on Figure 3.7B-40 are first determined. The resultant reaction is a function of the weight of the entire structure and the total vertical seismic load. The total vertical seismic load is determined by the combined effects of all dominant structure modes induced by horizontal and vertical seismic excitations; i.e., the resultant modal vertical earthquake loads for each direction of ground motion and the simultaneous effect of three earthquakes are combined as discussed in Subsection 3.7B.2.7.

3.7B.2.15 Analysis Procedure for Damping

Equivalent modal damping is evaluated using the concept of weighted modal damping [13]. This is an approximate rule for determining modal damping by weighing the damping associated with the individual components according to the energy stored in each component. Concrete structures, steel structures and systems, and foundation materials have inherently different damping properties; the effective damping in any vibration mode of the total system depends on the degree of participation of these components in the modal response.

The weighted modal damping is evaluated as follows:

$$D_n = \frac{\Sigma E_{nc} D_c}{\Sigma E_{nc}} \qquad (3.7B\text{-}24)$$

where

D_n = the weighted damping expressed as a fraction of critical damping in mode n

D_c = the damping expressed as a fraction of critical damping for component c

E_{nc} = the modal energy stored in component c

If a dynamic analysis is performed in a coupled system of equations of motion using a direct integration technique, a damping matrix [C] is computed and used in the dynamic solution. The damping matrix is expressed as a function of critical modal damping [23], [28], [37], [38] in the following equation in matrix form:

$$[C] = 2[m][\Phi][M]^{-1}[\omega]\,[\beta]\,[\Phi]^T[m] \qquad (3.7B\text{-}25)$$

where

$[m]$ = mass matrix

$[\Phi]$ = mode shape matrix

$[\Phi]^T$ = transpose of mode shape matrix

$[M] = [\Phi]^T[m]\,[\Phi]$ = generalized mass matrix

$[\omega]$ = diagonal matrix of natural circular frequencies

$[\beta]$ = diagonal matrix of modal damping expressed in fractions of critical damping

Equation 3.7B-25 can be used in the solution of the coupled differential equations of motion presented in matrix form as follows:

$$[m] \{\ddot{u}\} + [c] \{\dot{u}\} + [k] \{u\} = \{f(t)\} \quad (3.7B\text{-}26)$$

where

[k]	=	stiffness matrix
$\{\ddot{u}\}, \{\dot{u}\}, \{u\}$	=	column matrices of relative accelerations, velocities, and displacements, respectively
{f (t)}	=	column matrix of time dependent forcing functions

3.7B.3 SEISMIC SUBSYSTEM ANALYSIS

3.7B.3.1 Seismic Analysis Methods

The seismic Category I subsystems which comprise all seismic Category I mechanical and electrical systems and components and their supports, are analyzed by the response spectrum method, or the equivalent static load method. The seismic excitation used for the analysis of subsystems is obtained from the appropriate floor response spectra as described in Section 3.7B.2.

Subsystems and equipment are generally analyzed by the response spectrum method which consists of the following steps:

1. The system is modeled using a multi-degree-of-freedom representation. The size of the model is reviewed to assure that an adequate and sufficient number of masses or degrees of freedom are used to compute the response of the system. A sufficient number is considered adequate provided additional degrees of freedom do not result in more than a 10% increase in response, or the number of degrees of freedom equals or exceeds twice the number of modes with frequencies less than 33 Hz, or based upon engineering judgement that the number of degrees of freedom chosen will meet or exceed the intent of the prior two guidelines. The mathematical model consists of a network of elastic springs and lumped masses which are selected to coincide with actual mass concentrations, or to provide adequate representation of mass distribution. The degrees of freedom assigned to each mass of the model correspond to translations along three orthogonal axes, and rotations about these axes, completely defining the motion of the mass point in space. At points of support the total number of degrees of freedom is reduced by the number of constraints at these points. Reduced degrees of freedom are acceptable provided the analysis adequately and conservatively predicts the dynamic response of the equipment.

2. A dynamic analysis is performed to determine the natural frequencies and mode shapes of the mathematical model.

3. The participation factors for each direction of support motion are calculated.

4. For each significant mode a spectral response value corresponding to the modal natural frequency is determined from the applicable floor response spectra.

5. The modal responses consisting of modal absolute accelerations, modal relative displacements, and modal inertia loads are calculated for a sufficient number of modes. The results are combined by the SRSS method as described in paragraph 3.7B.3.7 or other methods in conformance with USNRC Regulatory Guide 1.92.

The number of modes chosen is considered adequate provided the inclusion of additional modes does not result in more than a 10% increase in responses, or based upon evaluation of the dynamic participation factors to assure that all significant modes have been included.

An analytical technique, developed in accordance with NUREG/CR-1161 (Reference 41), is used for piping systems to account for the modal contribution above the cutoff frequency.

Structurally simple equipment and systems, which can be represented either by a single degree-of-freedom model or a simple mathematical model, and equipment and subsystems which have been found to have no natural frequencies below 33 Hz are generally analyzed by the equivalent static load method as described in Section 3.7B.3.5.

The seismic response loads obtained by either the modal response analysis or equivalent static load method are combined with all other external loads such as operating loads, hydrodynamic loads, and piping interaction loads for design purposes. Non-linear responses of subsystems are considered on an individual basis where such phenomena are identified as existing, and are accounted for by analysis. Such an analysis was performed to account for the predetermined support clearance tolerances of the Service Water Intake Structure pumps.

For further details on seismic analysis methods, see Section 3.7B.2.1.

3.7B.3.2 Determination of Number of Earthquake Cycles

The number of maximum amplitude loading cycles is specified for seismic Category I structures, systems, and components as a minimum of 600 loading cycles for the OBE, and 120 loading cycles for the SSE.

For ASME Code Class 2 and 3 piping systems including supports for ASME Code Class 1, 2, and 3 piping a minimum of 50 loading cycles for the OBE and 10 loading cycles for the SSE is specified.

3.7B.3.3 Procedure Used for Modeling

The dynamic analysis of any complex system requires the discretization of its mass and elastic properties. This is accomplished by concentrating the mass of the system at distinct characteristic points or nodes, and interconnecting them by a network of elastic springs representing the stiffness properties of the systems, which are computed either by hand calculations for simple systems, or by finite element methods for more complex systems. Nodes are located at all mass concentrations and at additional points within the system, selected in such a way as to provide an adequate representation of the mass distribution of the system. At each node, degrees of freedom corresponding to translations along three orthogonal axes, and rotations about these axes are assigned. The number of degrees of freedom is reduced by the

number of constraints, where applicable. For equipment qualification, reduced degrees of freedom are acceptable provided the analysis adequately and conservatively predicts the response of the equipment.

For additional description of modeling procedures see Sections 3.7B.2.3 and 3.7B.3.1.

3.7B.3.4 Basis for Selection of Frequencies

In general, subsystems can be analyzed using time history or model response spectrum methods, or both.

Mathematical models representing subsystems are subjected to their support motions, which reflect the seismic environment of the free- field and structural amplifications. Therefore, when these support motions are used as input to the dynamic system, each mode responds according to the amplification which has been predetermined in the time history analysis of the supporting structure.

Elimination of resonance condition is considered good practice in the design of subsystems. The resonance peaks are readily identified from the appropriate response spectra. Elimination of resonance is the principal aim of the design. To eliminate this resonance condition, the fundamental frequency of seismic Category I subsystems can be selected to be greater than twice or less than one-half the dominant frequency of the supporting system or some modification of the dominant natural frequencies can be achieved by providing stiffer or more flexible supports and smaller or bigger mass characteristics of the subsystem. When this becomes impossible or impractical, the subsystem is analyzed and designed for amplified responses, including the resonance condition.

3.7B.3.5 Use of Equivalent Static Load Method of Analysis

Where a subsystem can be adequately and realistically represented as a one-degree-of-freedom system, and no determination of natural frequency is made, the response of the subsystem is assumed to be the peak acceleration of the appropriate floor response spectra curves at the appropriate value of damping.

For a subsystem which can be adequately and realistically represented as a simple model, similar to the guidelines of NRC Regulatory Guide 1.100, Rev. 1, and produce conservative analysis results, and no determination of natural frequencies is made, the response of the subsystem is assumed to be the peak of the appropriate floor response spectra at the appropriate value of damping multiplied by a factor of 1.5. A factor less than 1.5, but not less than 1.0 may be used, provided conservative results are obtained and proper justification provided.

Equipment having a minimum natural frequency equal to or greater than 33 Hz is also sometimes designed by the equivalent static load method, in which case the applied seismic loads correspond to accelerations equal to at least the zero-period accelerations of the appropriate floor response spectra.

3.7B.3.6 Three Components of Earthquake Motion

The combined effect of the three components of earthquake motion on seismic Category I subsystems is determined by the SRSS method as described in Subsection 3.7B.2.6.

3.7B.3.7 Combination of Modal Responses

When the response spectrum concept of analysis is used, only the maximum modal responses are known, and the phasing of modes cannot be determined. Therefore, the total response at a point in the multi-degree-of freedom system can only be approximated. The maximum modal responses are combined by the methods of NRC Regulatory Guide 1.92, Revision 1. For equipment and subsystem analyses, the methods presented in the Regulatory Guide paragraphs 1.1, 1.2.1, 1.2.2, or 1.2.3 are acceptable methods for vendor qualification.

3.7B.3.8 Analytical Procedures for Piping

3.7B.3.8.1 Design Criteria

Piping design criteria for Code Class 1 piping are in accordance with NB-3000 of the ASME B&PV Code, Section III. For Code Class 2 and Code Class 3 piping, see Section 3.9B.3.

Piping is anchored so that the total movements caused by relative building motion plus thermal growth do not overstress the system.

Critical areas of valve and piping inside the Containment are affected by relative motion between the Containment Building and the internal structure. Similar criteria are followed in these areas, especially at elevations where relative movements between Containment wall and internal structure are greater.

Piping is analyzed as an elastic system subject to thermal loadings and given displacements at anchor points.

Two analyses are made to determine the following:

1. Stresses imposed by thermal movements between equipment and anchors and by anchor movements between structures

2. Dynamic stresses imposed by seismic loading as a result of relative motion of buildings

Each piping system is idealized as a mathematical model consisting of lumped masses connected by elastic members. In order to adequately represent the dynamic and elastic characteristics of the piping system, lumped masses are located at carefully selected points. Sufficient mass points are located to ensure that all modes with frequencies less than 33 Hz are considered in the analysis. The number of degrees of freedom is verified to be equal to or greater than twice the number of modes with frequencies less than 33 Hz. In the modeling of the piping system, valves, reducers, tee and branch connections attached to the pipe are included. The location, type and stiffness of supports provided are reviewed and included in the analysis. Class 2 and 3 piping systems, whose nominal pipe diameter is 2 inches or less and whose operating temperature is less than 200°F may be analyzed by using section 3.7B.3.5 for seismic inertial effect.

Anchors with all six degrees restrained have thermal movement included in the analysis (i.e., anchors at equipment nozzles, containment penetrations, or embedded pipes).

There are three (3) categories of displacement for each direction of earthquake. Two of these categories represent rigid body motion of the structure, motions that are common to all points on the structure. The third category represents deformation of the structure, that is relative displacements of points on the structure.

When all of the points of fixity are located on a single structure, the rigid body motions of the structure, translation and rotation, do not result in relative motion of the points of fixity. Since the third category of displacement, deformation of the structure, represents a small portion of the total displacement profile, the effects of this displacement on the points of fixity are neglected.

For piping passing between buildings or equipment mounted on individual structures or foundations (such as big tanks), the relative displacement of support points located in different structures is considered in piping stress analysis.

Maximum relative displacements in two horizontal and the vertical direction between piping supports and anchor points between buildings are used as equivalent static displacement boundary conditions in order to calculate the secondary stresses of the piping system. Relative seismic displacements used are obtained from a dynamic analysis of the structures, and are always considered to be out-of-phase between different buildings and the equipment if applicable to obtain the most conservative piping responses.

3.7B.3.8.2 Basis for Computing Combined Responses

For the seismic design of piping, the horizontal and vertical loadings are obtained from the instructure response spectra that have been generated for the appropriate structures and elevations as outlined in Subsection 3.7B.2.1.2, and References [30], [31], and [36].

The combined effect of the three components of earthquake motion on the seismic design of piping is determined by the SRSS method (section 3.7B.2.6). The maximum modal responses are combined by the methods of NRC Regulatory Guide 1.92, Revision 1. The methods presented in Regulatory Guide paragraphs 1.1, 1.2.1, 1.2.2 or 1.2.3 are acceptable methods for vendor qualification.

Restraints are designed for loadings that are obtained from the piping analysis.

3.7B.3.8.3 Amplified Seismic Responses

For the seismic design of piping, input loading is obtained from the vertical and two horizontal modal response spectra curves for the appropriate damping of the building and/or structure.

Where a piping system is subjected to more than one amplified response spectrum, such as support points located in different structures or different elevations of the same structure, either of the following two methods shall be used, to generate loads at the pipe supports of the piping system.

1. Envelope all the amplified response spectra and apply to the piping system

2. Utilize independent support motion by applying the applicable amplified response spectra to each pipe support of the piping system.

3.7B.3.9 Multiple Supported Equipment Components with Distinct Inputs

The seismic analysis of multiply supported seismic Category I subsystems and equipment subjected to differential support motion within a building or between two buildings is performed in three parts, using lumped mass mathematical models, as follows:

1. Modal response spectrum analysis is performed for all three principal orthogonal directions of support motion for each direction of ground excitation using appropriate in-structure response spectra, constructed on the basis of superimposing the spectra for all support points and enveloping them as stated in Subsection 3.7B.2.5. The vertical analysis is combined with both horizontals as described in Subsection 3.7B.2.1.2, Item 1.

2. The same multimass lumped parameter model is subjected to a static analysis for the differential displacements of the support points. The displacements used are consistent with the directions of structural excitation considered in the spectrum analysis. This results in basic differential displacement loading conditions.

3. The results obtained from the spectrum analysis and differential displacement analysis are then combined absolutely. The effects of these loading conditions on the components and the supporting structures are determined.

3.7B.3.10 Use of Constant Vertical Static Factors

Constant static factors are used in some cases for the design of seismic Category I subsystems and equipment. The criteria for using this method are presented in Subsection 3.7B.3.5.

3.7B.3.11 Torsional Effects of Eccentric Masses

The criteria used to account for the torsional effects of valves and other eccentric masses (e.g., valve operators) in the seismic piping analyses are as follows:

1. When valves and other eccentric masses are considered rigid, the mass of the operator and valve body or other eccentric mass will be located at their respective center of gravity. The eccentric components (i.e., yoke, valve body) will be modeled as rigid members.

2. When valves and other eccentric masses are not considered rigid, the dynamic models are simulated by the lumped masses in discrete locations (i.e., center of gravity of valve body and valve operator), coupled by elastic members with properties of the eccentric components.

3.7B.3.12 Buried Seismic Category I Piping Systems and Tunnels

For seismic Category I piping systems outside the Containment structure, including those placed in underground concrete conduits but excluding those directly buried underground, the same design criteria and analytical procedures described in Subsection 3.7B.3.8 are used to ascertain

that allowable piping and structural stresses are not exceeded at Containment penetrations and at entry points into other structures.

Some seismic Category I piping systems are comprised of segments which are completely buried underground and which interface with the Auxiliary Building or the Service Water Intake Structure, or with other seismic Category I structures. Other seismic Category I piping segments are enclosed in concrete conduits which are buried underground and are connected to the conduit walls by appropriate restraints and supports.

All seismic Category I buried piping and concrete conduits are encased in a lean concrete fill or located in compacted backfill with a density sufficient to ensure that the backfill does not lose its integrity as a result of liquefaction during an SSE. If required, the effects of small settlements of structures on adjacent piping are reduced by providing flexible joints, split sleeves, and similar devices. Consolidation of the backfill is expected to be negligible under the pipe and conduit weights. Shearing distortions assumed for the design of the piping and conduits are based on consideration of the elastic properties of the compacted backfill or concrete fill, as well as those of the surrounding natural ground.

The following procedures are considered in the design of seismic Category I buried piping and concrete conduits.

3.7B.3.12.1 Stresses Caused by Free-Field Seismic Wave Propagation

The basic concept governing the response of those portions of buried underground pipes and concrete conduits away from the interaction effects of any external support except the surrounding soil is that, compared to the pipes and conduits, the soil is stiff; therefore, the earthquake deformation of the soil is imposed on the pipes and conduits which must conform to this deformation. Thus, the complex dynamic analysis of a discrete system involving soil-pipe or soil-conduit interaction that may be required for relatively soft soils is not considered. The analysis of seismic Category I tunnels is consistent with the recommendations of both References 15 and 29. The procedures outlined in paragraph 10.6 of Reference 15 are employed in analyzing buried piping.

1. The maximum bending stress resulting from shear wave is obtained by assuming that the shear wave travels parallel to the axis of the pipe or conduit. The curvature of the pipe or conduit is then determined as the ratio of the maximum ground acceleration perpendicular to the axis of the pipe or conduit and the shear wave velocity squared [15]. The product of the curvature and the flexural rigidity of the pipe or conduit yields the bending moment from which the unit stresses are obtained.

2. The maximum axial stress resulting from compressional wave is obtained by assuming that the compressional wave propagates in the soil along the pipe axis. The axial strain is then determined as the ratio of the maximum ground velocity in the direction of the axis of the pipe or conduit and the compressional wave velocity [15], [16], [17], [21], [22]. The maximum axial stress is obtained as a product of the axial strain and the modulus of elasticity of the pipe material. Slippage occurs if the total induced axial force over a straight pipe exceeds the friction force between the soil and the pipe surface.

   Then, the maximum axial stress in a buried pipe or concrete conduit under free-field conditions corresponds to the limiting frictional force.

3. The maximum axial stress caused by a shear wave occurs when the shear wave propagates at the incident angle of 45 degrees from the longitudinal axis of the pipe or concrete conduit [18], [21], [22]. This subjects different parts of the pipe or concrete conduit to out-of-phase displacements and results in a compressional wave propagating along the axis of the pipe or concrete conduit. The maximum induced axial stress is computed as the product of the strain, which can be obtained in accordance with References [18], [21], and [22] and the modulus of elasticity of the pipe material.

4. The maximum combined axial stress caused by stretching and bending strains induced by a single shear wave is computed following the studies concerning subways [18], [21]. It should be noted that it is impossible to have a single shear wave that simultaneously induces the maximum stresses described in Subsection 3.7B.3.12.1, Items 1 and 3. The axial effect of the shear wave propagation assumes its maximum value at the incident angle of 45 degrees measured from the longitudinal axis of the pipe or conduit, while the maximum bending effect value is at the incident angle of shear wave propagation of 0 degrees that is measured in the same way. The maximum strain is obtained by maximizing the expression of the sum of the two effects as presented in References [18], [21], and [22], the results of which are extended to buried pipe applications as in Reference [19].

3.7B.3.12.2 Stresses Caused by Differential Displacements Between Soil and Structure

As a result of soil-structure interaction, differential displacements during seismic disturbance are usually experienced between the structure and the soil at the entry points of buried pipes. The maximum horizontal and vertical differential displacements are obtained by performing the seismic spectrum analysis of each seismic Category I structure. These displacements are used in obtaining additional stresses in buried pipes. For pipes extending from one structure to another, an out-of-phase assumption is made to account for the possible phase differences of the seismic ground waves.

1. Bending and shearing stresses caused by differential displacements perpendicular to the pipe axis are obtained from the studies concerning elastic pile theory involving coefficients of subgrade reaction [4], [5], [6], [23]. When the soil surrounding the pipe can be assumed to be a homogeneous isotropic medium, solutions for beams on an elastic foundation such as the ones presented in Reference [24] are used.

2. The maximum axial stresses resulting from differential displacement along the pipe or conduit axis are computed from the consideration of load transfer from the pipe or conduit to the surrounding soil by friction to accommodate axial differential displacement at the location where the pipe is entering a structure, as well as from the elastic deformation of the soil at the other end of the pipe or conduit. However, a conservative estimate of this maximum axial stress can be obtained as the product of the axial displacement, the coefficient of horizontal subgrade reaction, and the ratio of the moduli of elasticity of the pipe or conduit material and the soil. The procedure is based on the assumption that the strain in the pipe or conduit is the same as that of the surrounding soil.

If the computed combined stresses, which include stresses resulting from earthquake, internal pressure, thermal expansion, and other operating loads, exceed the allowable limits at the penetrations, one or more of the following devices are used to relieve the stresses caused by the differential displacements:

1. The portions of the pipe at the entry points are protected from soil pressure by providing a concentric split sleeve.

2. The stresses resulting from differential displacements are reduced by replacing the compacted backfill soil or concrete fill around the pipe near the penetrations by another softer soil material.

3.7B.3.13 Interaction of Other Piping with Seismic Category I Piping

3.7B.3.13.1 Seismic Category I Piping with Connecting Non-Category I Piping

Interaction of seismic Category I piping with non-Category I piping connected to it is considered in the following two respects:

1. The loads transmitted under seismic excitation between the two systems locally at the point of their connection

2. The effect of the non-Category I system on the dynamic characteristics and the seismic response of the seismic Category I system

Consideration of both effects is achieved by incorporating into the analysis of the seismic Category I system a length of pipe that represents the actual dynamic behavior of the complete run of the non-Category I system. The length considered extends, but is not limited to, the first anchor point beyond the point of change from seismic Category I to non-Category I. Whenever possible, an anchor is located at the intersection of the seismic Category I piping with the non-Category I piping. In cases where location of the anchor or restraint is not possible at the category change, it is placed on the non-Category pipe, and that portion of the line up to the anchor or restraint is analyzed according to seismic Category I criteria. In either case, the non-Category I piping is always isolated from the Category I piping by anchors or seismic restraints.

3.7B.3.13.2 Seismic Category I Piping with Adjacent Non-Category I Piping

Non-Category I Piping Systems whose failure is not acceptable, adjacent to Seismic Category I piping, are analyzed by the nomograph method, other simplified dynamic analysis methods, or enveloping methods that will limit stress levels to assure structural integrity and prevent any unacceptable physical interaction with adjacent seismic Category I piping and components. The nomograph method provides seismic restraint spacing based on the natural frequency of the supported piping. This support spacing assures that the first natural frequency of the non-Category I piping is beyond that value which is twice the resonant frequency.

3.7B.3.14 Seismic Analyses for Reactor Internals

Seismic analyses for the reactor internals are presented in Section 3.7N.

3.7B.3.15 Analysis Procedure for Damping

Damping values expressed as percents of critical damping are determined for the type of material and fabrication of subsystems in accordance with the recommendations of NRC Regulatory Guide 1.61.

Typical damping values are presented in Table 3.7B-1. For the analysis of multidegree-of-freedom systems, equivalent modal dampings are determined according to the concept of weighted modal damping as described in Subsection 3.7B.2.15 and in Reference [13].

3.7B.3.16 Analysis Procedure for High Density Spent Fuel Storage Racks

The analysis undertaken to confirm the structural integrity of the high density spent fuel storage racks was performed in compliance with the USNRC Standard Review Plan [46, 49, 50, and 51] and the OT Position Paper [47]. The primary objective of the analysis is to determine the nature of the dynamic response of the SFP rack configuration to plant-specific seismic excitation and to establish the factor of safety. Both single rack and Whole-Pool-Multi-Rack (WPMR) analyses were performed. In the WPMR analysis, the entire assemblage of rack modules was modeled in one comprehensive simulation.

The response of a freestanding rack module to seismic inputs is highly nonlinear and involves a complex combination of motions (sliding, rocking, twisting, and turning), resulting in impacts and friction effects. Some of the unique attributes of the rack dynamic behavior include a large fraction of the total structural mass in a confined rattling motion, friction support of rack pedestals against lateral motion, and large fluid coupling effects due to deep submergence and motion of closely spaced adjacent structures.

Linear methods, such as modal analysis and response spectrum techniques, cannot accurately simulate the structural response of such a highly nonlinear structure to seismic excitation. Therefore, evaluation of the high density spent fuel storage racks is based on direct integration of the nonlinear equations of motion with the pool slab acceleration time-histories applied as the forcing functions.

The three-dimensional single rack dynamic model used in the rack analyses has the capability to effect momentum transfers which occur due to rattling of fuel assemblies inside storage cells and the capability to simulate lift-off and subsequent impact of support pedestals with the pool liner (or bearing pad). The contribution of the water mass in the interstitial spaces around and between the rack modules and around fuel assemblies was also included. This fluid mass significantly alters the dynamic behavior of fuel and rack masses during a seismic event.

Despite the versatility of the three dimensional seismic model, the accuracy of the single rack simulations has been suspect due to one key element; hydrodynamic participation of water around the racks. During dynamic rack motion, hydraulic energy is either drawn from or added to the moving rack, modifying its submerged motion in a significant manner. Therefore, the dynamics of one rack affects the motion of all others in the pool.

For closely spaced racks, demonstration of kinematic compliance was verified by including all modules in one comprehensive simulation using a WPMR model. In the WPMR analysis, all rack modules were modeled simultaneously and the coupling effect due to this multi-body motion was included. The worst case loads and stresses that result from either the three-dimensional single rack dynamic model and the WPMR model were used to determine the structural adequacy of the racks.

3.7B.4 SEISMIC INSTRUMENTATION

Seismic instrumentation is provided within the plant so that in case of an earthquake, sufficient data is generated to permit verification of the dynamic analysis of the plant and evaluation of the safety of continued operation.

3.7B.4.1 Comparison with Regulatory Guide 1.12

The seismic monitoring system meets the intent of RG 1.12, Rev. 1, with respect to the ability to determining exceedance of the OBE in a timely manner. Seismic instrumentation includes only the free-field triaxial accelerometer installed in the Yard. For the purpose of determining whether or not the OBE has been exceeded, ANSI/ANS-2.10-2003, “Criteria for the Handling and Initial Evaluation of Records from Nuclear Power Plant Seismic Instrumentation,” uses the free-field accelerometer data as input to address 10CFR100 Appendix A requirements in paragraph (V)(a)(2). Exceedance above the OBE ground motion level will require structural response analyses to be performed (ANSI/ANS-2.23-2002, “Nuclear Power Plant Response to an Earthquake”). The seismic monitoring system used to record a seismic event is in accordance with the requirements described in ANSI/ANS-2.2-2002, “Earthquake Instrumentation Criteria for Nuclear Power Plants.” The seismic monitoring system has the following components:

1. A triaxial accelerometer located in the free-field. The function of the triaxial accelerometer is to provide the acceleration time-history responses where the affects associated with surface features, buildings, and components will be insignificant.

2. A digital recorder that continuously monitors the free-field accelerometers with a solid state archival system. The function of the recorder is to provide a recording of the seismic event for post-event evaluation. The recorder will capture the 3-seconds preceding the exceedance of the seismic trigger threshold, and will continue to record data for a minimum of 5-seconds beyond the last recorded exceedance of the seismic trigger threshold.

3. A controller that will evaluate the recorded data in a timely manner. The function of the controller is to determine whether the OBE has been exceeded.

4. An uninterrupted power supply battery backup. The function of the battery backup is to ensure a minimum of 25-minutes of recorded data will be collected.

5. The free-field accelerometers will also function as the seismic switch. The recorder will continuously monitor the free-field accelerometers. The seismic trigger is a threshold ground acceleration value designed to initiate the event recording process at a value significantly below that of the OBE.

A schematic diagram of the seismic monitoring system is presented on Figure 3.7B-54. The free-field triaxial accelerometer described previously is provided for CPNPP Unit 1 as allowed by Section 4.5 of ANSI/ANS-2.2-2002. The seismic monitoring system conforms to the requirements of ANSI/ANS-2.2-2002, as described above, including calibration and channel checks.

3.7B.4.2 Location and Description of Instrumentation

The recorder, controller, computer screen display, printer, and uninterruptible power system are located in an instrument rack in the Control Room. In case of any seismic activity of sufficient intensity to activate the seismic monitoring system, the Control Room operator is alerted by means of the seismic annunciation system, which consists of visual and audible alarms. The first notification will be that an event is in progress by the exceedance of the seismic trigger threshold and activation of the recorder. The second notification will be provided by the controller to indicate that the OBE has been exceeded and that shut down is required. A redundant set of notification indications is provided on the front of the seismic monitoring system in the form of light-emitting diodes (LEDs).

3.7B.4.3 Control Room Operator Notification

This equipment computes and displays the free-field response spectra automatically on the computer screen following the event and prints a hardcopy on the system's printer. No Operator action is required to compute and print the response spectra. The seismic monitoring instrumentation facilitates the Operators ability to rapidly determine whether or not the OBE has been exceeded. In the event that the system fails to provide annunciation to one or both Control Rooms, local LED indication is provided on the face of the controller. The Control Room Staff will be alerted to the seismic event through perception of the building motions and external news sources. In addition, the Control Room Operator is provided various screen displays that present the results of the analysis of the ground motion data. Therefore, failure to fully annunciate will not prevent the Control Room Staff from being alerted to the situation. The operator is required to request maintenance and engineering support to evaluate the validity of any alarm or indication of a seismic event. In response to a seismic event, the operator is required to perform focused walkdowns to assess plant damage and the availability of safe shutdown equipment. The operator is required to shutdown both units if the Seismic Monitoring System determines that the OBE has been exceeded. In the event that the System is not available or functioning, the operator is required to request engineering to determine whether or not the OBE was exceeded.

3.7B.4.4 Comparison of Measured and Predicted Responses

Recorded actual time histories from a significant seismic event that has occurred at the site are used to determine whether the OBE has been exceeded. The acceleration response spectrum from the actual time histories will be compared to the design spectra in the FSAR Figures 3.7B-1 and 3.7B-6. Due to large conservatism in the generation of in-structure responses, a detailed comparison of measured to predicted responses is not required.

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TABLE 3.7B-1
DAMPING VALUES

Item, Equipment, or Structure	Damping, Percent Critical[a] OBE	SSE
Equipment	2	3
Welded steel structures	2	4
Bolted steel structures	4	7
Reinforced concrete structures	4	7
Piping [b]	variable	

a) NRC Regulatory Guide 1.61

b) Damping used in piping analysis is in accordance with the recommendations of NRC Regulatory Guide 1.61 or as specified in Code Case N-411 of the ASME Boiler and Pressure Vessel Code.

TABLE 3.7B-2
METHODS OF SEISMIC ANALYSIS USED FOR SEISMIC CATEGORY I
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure System, or Component	Equivalent Static Load	Response Spectrum Analysis	Modal Analysis Time History	Remarks
1. Structures				
Containment and internal structures	X	X	X	
Safeguards Building	X	X	X	
Electrical and Auxiliary Building	X	X	X	
Fuel Building	X	X	X	
Condensate storage tank	X	X	X	
RWST	X	X	X	
Reactor makeup Water storage tank	X	X	X	
Service Water Intake Structure	X	X	X	
2. Mechanical Components and Systems				
Westinghouse equipment				See Section 3.7N
High density spent fuel storage racks				See Section 3.7B.3.16
Other equipment	X	X		
3. Electrical Components and Systems				
Westinghouse equipment				See Section 3.7N
Other equipment	X	X		

TABLE 3.7B-3
DYNAMIC FOUNDATION DESIGN PARAMETERS

Foundation Material	Compressional Wave Velocity, V_c (ft/sec)	Shear Wave Velocity (ft/sec)	In-Situ Wet Density[a] (lb/ft^3)	Poisson's Ratio (ν)	Shear Modulus G or Modulus of Rigidity (lb/in^2)
Weathered rock	2,600	1,000	144	0.35	4.5×10^4
Moderately weathered to fresh rock (Glen Rose limestone above approximate elevation 770 ft)	9,500 to 11,000	5,500 to 6,000	150	0.30	8.0×10^5
Underlying massive fresh rock (Glen Rose limestone below approximate elevation 770 ft)	11,000 to 12,500	6,000 to 6,500	155	0.27	1.2×10^6
Underlying Twin Mountains rock	7,000 to 8,000	3,200	135	0.32	3.0×10^5

Young's modulus, $E = 2(1 + \nu) G$

$$E = pV_c^2 \frac{(1+\nu)(1-2\nu)}{(1-\nu)}$$

a) Mass density = $p = \frac{\text{in-situ density}}{32.2}$

TABLE 3.7B-3A
SOIL PROPERTIES FOR SOIL STRUCTURE INTERACTION ANALYSIS OF FUEL BUILDING

Layer No.	Elevation From	Elevation To	Thickness (ft)	Unit Weight (K/ft^3)	Mass Density ($K\text{-}s^2/ft^4$)	Shear Modulus (Ksf)	Shear Wave Velocity (ft/s)	Poisson's Ratio
A	796 ft	787 ft	9	0.150	0.00466	93,600	4482	0.35
B	787 ft	771 ft	16	0.145	0.00450	33,120	2713	0.35
C	771 ft	610 ft	161	0.155	0.00481	144,000	5472	0.35
D	610 ft	450 ft	160	0.135	0.00419	40,320	3102	0.35
Halfspace	450 ft	-200 ft	α	0.150	0.00466	93,600	4482	0.35
E								

TABLE 3.7B-4
SEISMIC CATEGORY I STRUCTURES FOUNDATION CHARACTERISTICS

Structure	Bottom of Mat Elevation	Plan Dimensions	Depth of Embedment	Structural Height
Containment Building	769 ft 2 in.	73 ft 6 in. radius	40 ft 10 in.	Top of dome elevation, 1070 ft 6 in.
Internal structure	793 ft 6 in.	72 ft 0 in. radius	-	Top floor elevation, 905 ft 9 in.
Safeguards Building	767 ft 4 in.	62 ft 6 in. by 98 ft 0 in.	42 ft 8 in.	Top of roof elevation, 918 ft 4 in.
Electrical Building	772 ft 10 in.	165 ft 6 in. by 101 ft 0 in.	37 ft 2 in.	Top of roof elevation, 873 ft 4 in.
Auxiliary Building	784 ft 10 in.	136 ft 0 in. by 192 ft 6 in.	25 ft 2 in.	Top of roof elevation, 900 ft 3 in.
Fuel Building	780 ft 6 in.	143 ft 6 in. by 137 ft 5 in.	29 ft 6 in.	Top of roof elevation, 918 ft 0 in.
Service Water Intake Structure	749 ft 0 in.	106 ft 6 in. by 74 ft 9 in.	Approx. 41 ft (varies)	Top of roof elevation, 838 ft 0 in.
Condensate storage tank	805 ft 6 in.	26 ft 6 in. radius	Approx. 4 ft 6 in.	Top of roof elevation, 860 ft 0 in.
RWST	805 ft 6 in.	26 ft 6 in. radius	Approx. 4 ft 6 in.	Top of roof elevation 860 ft 0 in.
Reactor makeup water storage tank	806 ft 6 in.	16 ft 6 in. radius	Approx. 3 ft 6 in.	Top of roof elevation, 846 ft 0 in.

TABLE 3.7B-5
SPRING CONSTANTS FOR RIGID CIRCULAR FOOTINGS RESTING ON ELASTIC HALF-SPACE

Motion	Spring Constant
Vertical	$k_z = \frac{4GR_o}{(1-\nu)}$
Horizontal	$k_x = \frac{32(1-\nu)GR_o}{7-8\nu}$
Rocking	$k_\psi = \frac{8GR_o^3}{3(1-\nu)}$
Torsion	$k_\theta = \frac{16GR_o^3}{3}$

Note:

$$G = \frac{E}{2(1+\nu)}$$

TABLE 3.7B-6
SPRING CONSTANTS FOR RIGID RECTANGULAR FOOTINGS RESTING ON ELASTIC HALF-SPACE

Motion	Spring Constant
Vertical	$k_z = \frac{G}{1-\nu}\beta_z\sqrt{4cd}$
Horizontal	$k_x = 4(1+\nu)G\beta_x\sqrt{cd}$
Rocking	$k_\psi = \frac{G}{1-\nu}\beta_\psi 8cd^2$

Note:

Values for $B_{z,}B_{x,}$ and B_ψ are given on Figure 3.7B-39 for various values of d/c.

TABLE 3.7B-7
PROPERTIES OF MASS-FOUNDATION SYSTEM THEORY OF RIGID CIRCULAR FOOTINGS ON ELASTIC HALF-SPACE

Mode of Vibration	Mass or Inertia Patio	Damping Ratio $D = C/C_c$
Vertical translation	$B_z = \frac{(1-\nu)m}{4pr_o^3}$	$D_z = \frac{0.425}{\sqrt{B_z}}$
Horizontal translation	$B_x = \frac{(7-8\nu)m}{32(1-\nu)pr_o^3}$	$D_x = \frac{0.288}{\sqrt{B_x}}$
Rocking	$B_\psi = \frac{3(1-\nu)I_\psi}{8pr_o^5}$	$D_\psi = \frac{0.15}{(1+B_\psi)\sqrt{B_\psi}}$
Torsion (about vertical axis)	$B_\theta = \frac{I_\theta}{pr_o^5}$	$D_\theta = \frac{0.5}{1+2B_\theta}$

Note:

- m = mass of the mat
- I_ψ = moment of inertia about horizontal axis located at base
- I_θ = moment of inertia about vertical axis
- p = mass density of foundation medium
- ν = Poisson's Ratio of foundation medium
- C = coefficient of viscous damping
- C_c = coefficient of viscous critical damping

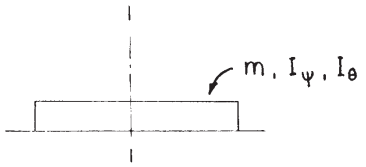

TABLE 3.7B-8
COEFFICIENTS FOR ESTIMATING EFFECTIVE MASS AND MASS MOMENT OF INERTIA OBTAINED FROM THEORY OF ELASTIC HALF-SPACE

Mode of Vibration	Coefficient		
	$\nu = 0$	$\nu = 0.25$	$\nu = 0.5$
Vertical translation	$\alpha = 0.5$	$\alpha = 1.0$	$\alpha = 2.0$
Horizontal translation	$\alpha = 0.2$	$\alpha = 0.2$	$\alpha = 0.1$
Rocking	$\beta = 0.4$	Not computed	
Torsion (about vertical axis)	$\beta = 0.3$	$\beta = 0.3$	$\beta = 0.3$

TABLE 3.7B-9
CONTAINMENT AND INTERNAL STRUCTURES MASS POINT NODAL COORDINATES

Mass Point	X (ft)	Y (ft)	Z (ft)
1	0.00	246.60	0.00
2	0.00	197.00	0.00
3	0.00	145.00	0.00
4	0.00	87.75	0.00
5	0.00	29.25	0.00
6	9.45	89.83	4.44
7	-0.28	55.94	8.27
8	-5.87	26.98	2.42
9	2.40	- 7.75	0.38
Foundation spring	0.00	-18.85	0.00

Notes:

a. The origin of coordinates is at elevation 805 ft 6 in.

b. For orientation of coordinate axes, see Figure 3.7B-23

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-34.

TABLE 3.7B-10
SAFEGUARDS BUILDING MASS POINT NODAL COORDINATES

Mass Point	X (ft)	Y (ft)	Z (ft)
1	48.35	893.49	35.64
2	52.66	872.40	70.80
3	54.11	851.80	82.06
4	58.60	831.34	90.09
5	61.01	809.47	103.87
6	44.15	789.68	69.87
7	31.69	772.53	49.74
Foundation spring		(See Table 3.7B-25.)	

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-25.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-35.

TABLE 3.7B-11
ELECTRICAL AND AUXILIARY BUILDINGS MASS POINT NODAL COORDINATES

Mass Point	X (ft)	Y (ft)	Z (ft)
1	59.31	871.29	-1.12
2	55.47	852.50	-0.18
3	46.31	829.55	0.15
4	53.78	805.63	0.75
5	57.20	776.79	-0.14
6	179.37	893.85	-0.22
7	166.49	872.32	3.47
8	172.92	849.55	0.40
9	174.87	831.15	-2.09
10	178.11	811.46	1.58
11	162.22	791.36	-3.30
Foundation spring		(See Tables 3.7B-26 and 3.7B-27.)	

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-28

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-36.

TABLE 3.7B-12
FUEL BUILDING MASS POINT NODAL COORDINATES

Mass Point	X (ft)	Y (ft)	Z (ft)
1	68.33	916.17	0 .00
2	68.21	894.27	-46.68
3	54.86	859.10	-41.12
4	55.91	840.06	-41.15
5	54.24	824.69	-40.52
6	65.72	894.67	48.60
7	60.53	859.37	44.54
8	52.83	840.41	38.59
9	52.07	824.65	40.55
10	51.09	812.53	6.47
Foundation spring	65.40	805.50	-2.17

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-30.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-37.

TABLE 3.7B-12A
FUEL BUILDING RE-ANALYSIS MODEL NODAL COORDINATES

(Sheet 1 of 2)

Mass Points

Mass Point	X (ft)	Y (ft)	Z (ft)
10	67.46	805.33	-0.59
20	48.60	819.56	0.20
1	50.34	838.75	-0.64
2	56.21	860.00	3.82
3	67.74	899.50	1.61
4	69.37	918.25	0.64
2030 Spent Fuel/Racks	40.13	827.06	0.00
220 Sloshing	40.13	860.00	0.00
109	35.13	805.33	0.00
107	51.68	810.50	0.00
108	35.13	810.50	0.00
208	35.13	819.56	0.00
209	51.68	819.56	-1.28
19	51.68	838.75	-1.28
18	45.80	838.75	-0.35
28	45.80	860.00	-0.35
29	67.18	860.00	0.00
39	67.18	899.50	0.00
38	68.88	899.50	0.00
48	68.88	918.25	0.00
210	56.21	860.00	0.00
2010	48.60	819.56	0.00
2020	48.60	827.06	0.00
1002	-18.08	805.33	-0.59
1003	153.00	805.33	-0.59
1004	67.46	805.33	-0.59

TABLE 3.7B-12A
FUEL BUILDING RE-ANALYSIS MODEL NODAL COORDINATES

(Sheet 2 of 2)

	Mass Points		
Mass Point	X (ft)	Y (ft)	Z (ft)
1005	67.46	805.33	-98.61
2002	-40.55	819.56	0.20
2003	137.75	819.56	0.20
2004	48.60	819.56	97.83
2005	48.60	819.56	-97.43
102	-37.07	838.75	-0.64
103	137.75	838.75	-0.64
104	50.34	838.75	97.43
105	50.34	838.75	-98.71
202	-25.33	860.00	3.82
203	137.75	860.00	3.82
204	56.21	860.00	105.07
205	56.21	860.00	-97.43
302	-2.27	899.50	1.61
303	137.75	899.50	1.61
304	67.74	899.50	100.65
305	67.74	899.50	-97.43
402	0.00	918.25	0.64
403	138.74	918.25	0.64
404	69.37	918.25	39.28
405	69.37	918.25	-38.00

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate axes and structure dynamic model showing the mass points, see Figure 3.7B-37A.

TABLE 3.7B-13
SERVICE WATER INTAKE STRUCTURE MASS POINT NODAL COORDINATES

Mass Point	X (ft)	Y (ft)	Z (ft)
1	29.65	0	54.86
2	21.91	20	60.81
25	21.92	41	55.56
3	19.96	62	53.36

Notes:

a. The origin of coordinates is at elevation 776 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-32

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-38.

TABLE 3.7B-14
CONTAINMENT AND INTERNAL STRUCTURES MASS POINT MASS DATA

Mass Point	Mass (kip-sec^2-ft)			Mass Moment of Inertia (kip-sec^2-ft)		
	Mx	My	Mz	Ix	Iy	Iz
1	173	173	173	186,400	340,500	186,400
2	375	375	375	966,000	1,734,000	966,000
3	542	555	542	1,395,000	2,510,000	1,395,000
4	538	538	538	1,463,000	2,618,000	1,463,000
5	538	538	538	1,463,000	2,618,000	1,463,000
6	440	440	440	599,800	982,500	429,600
7	708	531	710	816,400	1,112,000	623,600
8	602	640	602	440,900	984,800	588,200
9	1821	1959	1821	2,033,000	3,888,000	2,097,000
9A	322	2146	322	3,864,000	2,898,000	3,864,000

Notes:

a. Mass points 1 through 9 are for the structures. Mass point 9A is for the foundation material.

b. The origin of coordinates is at elevation 805 ft 6 in. For orientation of coordinate axes, see Figure 3.7B-23.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-34.

TABLE 3.7B-15
SAFEGUARDS BUILDING MASS POINT MASS DATA

Mass Point	Mass (kip-sec^2-ft)			Mass Moment of Inertia (kip-sec^2-ft)		
	Mx	My	Mz	Ix	Iy	Iz
1	171	171	171	138,564	218,891	132,264
2	323	323	323	719,229	1,104,267	399,863
3	393	393	393	1,014,757	1,560,267	565,170
4	479	479	479	1,353,810	2,020,784	698,972
5	569	569	569	1,728,231	2,663,410	957,458
6	470	470	470	844,598	1,154,650	336,715
7	254	254	254	216,360	303,136	94,036
5A	106	703	106	356,800	727,400	1,679,000
6A	65	432	65	1,190,000	607,900	464,400
7A	64	430	64	489,500	237,200	160,700

Notes:

a. Mass points 1 through 7 are for the structure. Mass points 5A, 6A and 7A are for the foundation material.

b. The origin of coordinates is at elevation 0 ft 0 in. For orientation of coordinate axes, see Figure 3.7B-25.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-35.

TABLE 3.7B-16
ELECTRICAL AND AUXILIARY BUILDINGS MASS POINT MASS DATA

Mass Point	Mass (kip-sec^2-ft)			Mass Moment of Inertia (kip-sec^2-ft)		
	Mx	My	Mz	Ix	Iy	Iz
1	191	191	191	352,578	621,762	273,288
2	371	371	371	1,029,267	1,540,460	525,576
3	335	335	335	935,606	1,329,372	413,593
4	404	404	404	1,232,237	1,789,889	585,062
5	570	570	570	1,475,049	2,636,970	1,184,607
6	373	373	373	1,170,341	1,756,542	611,452
7	530	530	530	1,543,640	2,372,764	859,910
8	860	860	860	2,390,334	3,484,584	1,151,946
9	710	710	710	2,110,568	3,065,440	989,810
10	663	663	663	2,193,210	2,996,694	859,967
11	1022	1022	1022	2,877,228	5,305,467	2,585,200
5A	310	2065	310	7,211,000	3,355,000	2,046,000
11A	623	4153	623	18,830,000	9,913,000	8,046,000

Notes:

a. Mass points 1 through 11 are for the structures. Mass points 5A and 11A are for the foundation material.

b. The origin of coordinates is at elevation 0 ft 0 in. For orientation of coordinate axes, see Figure 3.7B-28.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-36.

TABLE 3.7B-17
FUEL BUILDING MASS POINT MASS DATA

Mass Point	Mass (kip-sec^2-ft)			Mass Moment of Inertia (kip-sec^2-ft)		
	Mx	My	Mz	Ix	Iy	Iz
1	156	156	156	97,540	370,520	274,990
2	144	144	144	62,892	329,880	282,850
3	234	234	234	148,060	548,670	432,090
4	304	304	304	149,840	563,140	425,110
5	209	209	209	107,960	429,550	328,460
6	167	167	167	73,818	392,660	338,010
7	261	261	261	140,530	562,750	455,640
8	342	342	342	191,730	623,910	445,600
9	188	188	188	105,060	358,800	259,940
10	1434	1434	1434	3,032,500	4,287,500	2,147,000
10A	445	2574	445	7,134,826	4,677,522	5,362,199

Notes:

a. Mass points 1 through 10 are for the structure. Mass point 10A is for the foundation material.

b. The origin of coordinates is at elevation 0 ft 0 in. For orientation of coordinate axes, see Figure 3.7B-30.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-37.

TABLE 3.7B-17A
FUEL BUILDING RE-ANALYSIS MASS POINT DATA

Mass Point	Mass (kip-sec^2-ft)			Mass Moment of Inertia (kip-sec^2-ft)		
	Mx	My	Mz	Ix	Iy	Iz
10	1124	1124	1124	1,996,564	1,535,468	3,351,743
20	701	701	701	1,497,815	2,356,604	893,774
1	907	907	907	1,854,838	3,049,616	1,226,955
2	577	577	577	1,325,568	2,288,451	1,030,240
3	407	407	407	981,658	1,700,226	756,403
4	211	211	211	132,334	510,951	380,890
2030 Spent Fuel/Racks	66.8	66.8	66.8	178,310	186,076	10,271
220 Sloshing	45.6	45.6	35.8	125,015	127,022	10,306

TABLE 3.7B-18
SERVICE WATER INTAKE STRUCTURE MASS POINT MASS DATA

Mass Point	Mass (kip-sec^2-ft)			Mass Moment of Inertia (kip-sec^2-ft)		
	Mx	My	Mz	Ix	Iy	Iz
1	736	1446	736	1.964×10^6	1.308×10^6	5.79×10^5
2	280	280	280	3.59×10^5	4.17×10^5	1.08×10^5
25	98	98	98	1.57×10^5	1.83×10^5	$.335 \times 10^5$
3	89	89	89	0.99×10^5	1.21×10^5	$.25 \times 10^5$

Notes:

a. Mass 1 includes the equivalent soil mass.

b. The origin of coordinates is at elevation 776 ft, 0 in. For orientation of coordinate axes, see Figure 3.7B-32.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-38.

TABLE 3.7B-19
CONTAINMENT AND INTERNAL STRUCTURES DEGREES OF FREEDOM

(Sheet 1 of 2)

Mass Point	Degree of Freedom	Direction of Motion	
1	1	Translation	X
1	2	Translation	Y
1	3	Translation	Z
1	4	Rotation	θ_x
1	5	Rotation	θ_y
1	6	Rotation	θ_z
2	7	Translation	X
2	8	Translation	Y
2	9	Translation	Z
2	10	Rotation	θ_x
2	11	Rotation	θ_y
2	12	Rotation	θ_z
3	13	Translation	X
3	14	Translation	Y
3	15	Translation	Z
3	16	Rotation	θ_x
3	17	Rotation	θ_y
3	18	Rotation	θ_z
4	19	Translation	X
4	20	Translation	Y
4	21	Translation	Z
4	22	Rotation	θ_x
4	23	Rotation	θ_y
4	24	Rotation	θ_z
5	25	Translation	X
5	26	Translation	Y
5	27	Translation	Z
5	28	Rotation	θ_x
5	29	Rotation	θ_y
5	30	Rotation	θ_z
6	31	Translation	X

TABLE 3.7B-19
CONTAINMENT AND INTERNAL STRUCTURES DEGREES OF FREEDOM

(Sheet 2 of 2)

Mass Point	Degree of Freedom	Direction of Motion	
6	32	Translation	Y
6	33	Translation	Z
6	34	Rotation	θ_x
6	35	Rotation	θ_y
6	36	Rotation	θ_z
7	37	Translation	X
7	38	Translation	Y
7	39	Translation	Z
7	40	Rotation	θ_x
7	41	Rotation	θ_y
7	42	Rotation	θ_z
8	43	Translation	X
8	44	Translation	Y
8	45	Translation	Z
8	46	Rotation	θ_x
8	47	Rotation	θ_y
8	48	Rotation	θ_z
9	49	Translation	X
9	50	Translation	Y
9	51	Translation	Z
9	52	Rotation	θ_x
9	53	Rotation	θ_y
9	54	Rotation	θ_z

Notes:

a. The origin of coordinates is at elevation 805 ft 6 in.

b. For orientation of coordinate axes, see Figure 3.7B-23.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-34.

TABLE 3.7B-20
SAFEGUARDS BUILDING DEGREES OF FREEDOM

(Sheet 1 of 2)

Mass Point	Degree of Freedom	Direction of Motion	
1	1	Translation	X
1	2	Translation	Y
1	3	Translation	Z
1	4	Rotation	θ_x
1	5	Rotation	θ_y
1	6	Rotation	θ_z
2	7	Translation	X
2	8	Translation	Y
2	9	Translation	Z
2	10	Rotation	θ_x
2	11	Rotation	θ_y
2	12	Rotation	θ_z
3	13	Translation	X
3	14	Translation	Y
3	15	Translation	Z
3	16	Rotation	θ_x
3	17	Rotation	θ_y
3	18	Rotation	θ_z
4	19	Translation	X
4	20	Translation	Y
4	21	Translation	Z
4	22	Rotation	θ_x
4	23	Rotation	θ_y
4	24	Rotation	θ_z

TABLE 3.7B-20
SAFEGUARDS BUILDING DEGREES OF FREEDOM

(Sheet 2 of 2)

Mass Point	Degree of Freedom	Direction of Motion	
5	25	Translation	X
5	26	Translation	Y
5	27	Translation	Z
5	28	Rotation	θ_x
5	29	Rotation	θ_y
5	30	Rotation	θ_z
6	31	Translation	X
6	32	Translation	Y
6	33	Translation	Z
6	34	Rotation	θ_x
6	35	Rotation	θ_y
6	36	Rotation	θ_z
7	37	Translation	X
7	38	Translation	Y
7	39	Translation	Z
7	40	Rotation	θ_x
7	41	Rotation	θ_y
7	42	Rotation	θ_z

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of ccordinate axes, see Figure 3.7B-25.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-35.

TABLE 3.7B-21
ELECTRICAL AND AUXILIARY BUILDINGS DEGREES OF FREEDOM

(Sheet 1 of 3)

Mass Point	Degree of Freedom	Direction of Motion	
1	1	Translation	X
1	2	Translation	Y
1	3	Translation	Z
1	4	Rotation	θ_x
1	5	Rotation	θ_y
1	6	Rotation	θ_z
2	7	Translation	X
2	8	Translation	Y
2	9	Translation	Z
2	10	Rotation	θ_x
2	11	Rotation	θ_y
2	12	Rotation	θ_z
3	13	Translation	X
3	14	Translation	Y
3	15	Translation	Z
3	16	Rotation	θ_x
3	17	Rotation	θ_y
3	18	Rotation	θ_z
4	19	Translation	X
4	20	Translation	Y
4	21	Translation	Z
4	22	Rotation	θ_x
4	23	Rotation	θ_y
4	24	Rotation	θ_z

TABLE 3.7B-21
ELECTRICAL AND AUXILIARY BUILDINGS DEGREES OF FREEDOM

(Sheet 2 of 3)

Mass Point	Degree of Freedom	Direction of Motion	
5	25	Translation	X
5	26	Translation	Y
5	27	Translation	Z
5	28	Rotation	θ_x
5	29	Rotation	θ_y
5	30	Rotation	θ_z
6	31	Translation	X
6	32	Translation	Y
6	33	Translation	Z
6	34	Rotation	θ_x
6	35	Rotation	θ_y
6	36	Rotation	θ_z
7	37	Translation	X
7	38	Translation	Y
7	39	Translation	Z
7	40	Rotation	θ_x
7	41	Rotation	θ_y
7	42	Rotation	θ_z
8	43	Translation	X
8	44	Translation	Y
8	45	Translation	Z
8	46	Rotation	θ_x
8	47	Rotation	θ_y
8	48	Rotation	θ_z

TABLE 3.7B-21
ELECTRICAL AND AUXILIARY BUILDINGS DEGREES OF FREEDOM

(Sheet 3 of 3)

Mass Point	Degree of Freedom	Direction of Motion	
9	49	Translation	X
9	50	Translation	Y
9	51	Translation	Z
9	52	Rotation	θ_x
9	53	Rotation	θ_y
9	54	Rotation	θ_z
10	55	Translation	X
10	56	Translation	Y
10	57	Translation	Z
10	58	Rotation	θ_x
10	59	Rotation	θy
10	60	Rotation	θ_z
11	61	Translation	X
11	62	Translation	Y
11	63	Translation	Z
11	64	Rotation	θ_x
11	65	Rotation	θ_y
11	66	Rotation	θ_z

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-28.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-36.

TABLE 3.7B-22
FUEL BUILDING DEGREES OF FREEDOM

(Sheet 1 of 3)

Mass Point	Degree of Freedom	Direction of Motion	
1	1	Translation	X
1	2	Translation	Y
1	3	Translation	Z
1	4	Rotation	θ_x
1	5	Rotation	θ_y
1	6	Rotation	θ_z
2	7	Translation	X
2	8	Translation	Y
2	9	Translation	Z
2	10	Rotation	θ_x
2	11	Rotation	θ_y
2	12	Rotation	θ_z
3	13	Translation	X
3	14	Translation	Y
3	15	Translation	Z
3	16	Rotation	θ_x
3	17	Rotation	θ_y
3	18	Rotation	θ_z
4	19	Translation	X
4	20	Translation	Y
4	21	Translation	Z
4	22	Rotation	θ_x

TABLE 3.7B-22
FUEL BUILDING DEGREES OF FREEDOM

(Sheet 2 of 3)

Mass Point	Degree of Freedom	Direction of Motion	
4	23	Rotation	θ_y
4	24	Rotation	θ_z
5	25	Translation	X
5	26	Translation	Y
5	27	Translation	Z
5	28	Rotation	θ_x
5	29	Rotation	θ_y
5	30	Rotation	θ_z
6	31	Translation	X
6	32	Translation	Y
6	33	Translation	Z
6	34	Rotation	θ_x
6	35	Rotation	θ_y
6	36	Rotation	θ_z
7	37	Translation	X
7	38	Translation	Y
7	39	Translation	Z
7	40	Rotation	θ_x
7	41	Rotation	θ_y
7	42	Rotation	θ_z
8	43	Translation	X
8	44	Translation	Y

TABLE 3.7B-22
FUEL BUILDING DEGREES OF FREEDOM

(Sheet 3 of 3)

Mass Point	Degree of Freedom	Direction of Motion	
8	45	Translation	Z
8	46	Rotation	θ_x
8	47	Rotation	θ_y
8	48	Rotation	θ_z
9	49	Translation	X
9	50	Translation	Y
9	51	Translation	Z
9	52	Rotation	θ_x
9	53	Rotation	θ_y
9	54	Rotation	θ_z
10	55	Translation	X
10	56	Translation	Y
10	57	Translation	Z
10	58	Rotation	θ_x
10	59	Rotation	θ_y
10	60	Rotation	θ_z

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-30.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-37.

TABLE 3.7B-23
SERVICE WATER INTAKE STRUCTURE DEGREES OF FREEDOM

Mass Point	Degree of Freedom	Direction of Motion	
1	1	Translation	X
1	2	Translation	Y
1	3	Translation	Z
1	4	Rotation	θ_x
1	5	Rotation	θ_y
1	6	Rotation	θ_z
2	7	Translation	X
2	8	Translation	Y
2	9	Translation	Z
2	10	Rotation	θ_x
2	11	Rotation	θ_y
2	12	Rotation	θ_z
25	13	Translation	X
25	14	Translation	Y
25	15	Translation	Z
25	16	Rotation	θ_x
25	17	Rotation	θ_y
25	18	Rotation	θ_z
3	19	Translation	X
3	20	Translation	Y
3	21	Translation	Z
3	22	Rotation	θ_x
3	23	Rotation	θ_y
3	24	Rotation	θ_z

Notes:

a. The origin of coordinates is at elevation 776 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-32.

c. For structure dynamic model showing the mass point numbers, see Figure 3.7B-38.

TABLE 3.7B-24
CONTAINMENT AND INTERNAL STRUCTURES FOUNDATION SPRING CONSTANTS

Direction	Lower Pound	Best Estimate	Upper Pound
Translation along X axis (kip/ft)	0.116×10^{8}	0.465×10^{8}	0.930×10^{8}
Translation along Y axis (kip/ft)	0.125×10^{8}	0.499×10^{8}	0.998×10^{8}
Translation along Z axis (kip/ft)	0.116×10^{8}	0.465×10^{8}	0.930×10^{8}
Rotation along X axis (kip-ft/rad)	0.390×10^{11}	0.156×10^{12}	0.312×10^{12}
Rotation along Y axis (kip-ft/rad)	0.600×10^{11}	0.240×10^{12}	0.480×10^{12}
Rotation along Z axis (kip-ft/rad)	0.390×10^{11}	0.156×10^{12}	0.312×10^{12}

Notes:

a. The origin of coordinates is at elevation 805 ft 6 in.

b. For orientation of coordinate axes, see Figure 3.7B-23.

TABLE 3.7B-25
SAFEGUARDS BUILDING FOUNDATION SPRING CONSTANTS

(Sheet 1 of 3)

Mass No.	Spring Location X (ft)	Spring Location Y (ft)	Spring Location Z (ft)	Lower Bound Foundation Spring Constants: Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
5	92.22	806.50	149.62	5565×10^{7}	7033×10^{7}	$.5565 \times 10^{7}$	$.2970 \times 10^{10}$	$.1047 \times 10^{11}$	$.1026 \times 10^{11}$
5	58.50	801.125	127.50	4338×10^{6}	0.0	$.5245 \times 10^{6}$	$.2020 \times 10^{8}$	$.2329 \times 10^{9}$	$.1672 \times 10^{8}$
5	60.75	801.125	125.625	4775×10^{6}	0.0	$.3760 \times 10^{6}$	$.1445 \times 10^{8}$	$.1466 \times 10^{9}$	$.4595 \times 10^{7}$
5	53.93	785.00	90.71	$.4730 \times 10^{7}$	$.6554 \times 10^{7}$	$.4730 \times 10^{7}$	$.9880 \times 10^{10}$	$.1067 \times 10^{11}$	$.4655 \times 10^{10}$
6	31.25	780.44	0.00	$.3415 \times 10^{6}$	0.0	$.4348 \times 10^{6}$	$.1088 \times 10^{8}$	$.1648 \times 10^{9}$	$.8470 \times 10^{7}$
6	62.50	780.625	47.625	$.5650 \times 10^{6}$	0.0	$.4475 \times 10^{6}$	$.1141 \times 10^{8}$	$.4273 \times 10^{9}$	$.1442 \times 10^{8}$
6	31.25	780.44	95.25	$.3415 \times 10^{6}$	0.0	$.4348 \times 10^{6}$	$.1088 \times 10^{8}$	$.1648 \times 10^{9}$	$.8470 \times 10^{7}$
6	58.50	790.375	127.50	$.4338 \times 10^{6}$	0.0	$.5245 \times 10^{6}$	$.2020 \times 10^{8}$	$.2329 \times 10^{9}$	$.1672 \times 10^{8}$
6	60.75	790.375	125.625	$.4775 \times 10^{6}$	0.0	$.3760 \times 10^{6}$	$.1445 \times 10^{8}$	$.1466 \times 10^{9}$	$.4595 \times 10^{7}$
7	30.50	767.50	47.625	$.4723 \times 10^{7}$	$.5708 \times 10^{7}$	$.4723 \times 10^{7}$	$.4315 \times 10^{10}$	$.5035 \times 10^{10}$	$.1770 \times 10^{10}$
7	31.25	771.81	0.00	$.3415 \times 10^{6}$	0.0	$.4348 \times 10^{6}$	$.1088 \times 10^{6}$	$.1648 \times 10^{9}$	$.8470 \times 10^{7}$
7	62.50	771.875	47.625	$.5650 \times 10^{6}$	0.0	$.4475 \times 10^{6}$	$.1141 \times 10^{8}$	$.4273 \times 10^{9}$	$.1442 \times 10^{8}$
7	31.25	771.81	95.25	$.3415 \times 10^{6}$	0.0	$.4348 \times 10^{6}$	$.1088 \times 10^{8}$	$.1648 \times 10^{9}$	$.8470 \times 10^{7}$

TABLE 3.7B-25
SAFEGUARDS BUILDING FOUNDATION SPRING CONSTANTS

(Sheet 2 of 3)

Mass No.	Spring Location X (ft)	Spring Location Y (ft)	Spring Location Z (ft)	Best Estimate Foundation Spring Constants: Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
5	92.22	806.50	149.62	$.2226 \times 10^{8}$	$.2813 \times 10^{8}$	$.2226 \times 10^{8}$	$.1188 \times 10^{11}$	$.4187 \times 10^{11}$	$.4102 \times 10^{11}$
5	58.50	801.125	127.50	$.1735 \times 10^{7}$	0.0	$.2098 \times 10^{7}$	$.8080 \times 10^{8}$	$.9315 \times 10^{9}$	$.6687 \times 10^{8}$
5	60.75	801.125	125.625	$.1910 \times 10^{7}$	0.0	$.1504 \times 10^{7}$	$.5780 \times 10^{8}$	$.5865 \times 10^{9}$	$.1838 \times 10^{8}$
6	53.93	785.00	90.71	$.1892 \times 10^{8}$	$.2578 \times 10^{8}$	$.1892 \times 10^{8}$	$.3952 \times 10^{11}$	$.4267 \times 10^{11}$	$.1862 \times 10^{11}$
6	31.25	780.44	0.0	$.1366 \times 10^{7}$	0.0	$.1739 \times 10^{7}$	$.4351 \times 10^{8}$	$.6590 \times 10^{9}$	$.3388 \times 10^{8}$
6	62.50	780.625	47.625	$.2260 \times 10^{7}$	0.0	$.1790 \times 10^{7}$	$.4565 \times 10^{8}$	$.1709 \times 10^{10}$	$.5768 \times 10^{8}$
6	31.25	780.44	95.25	$.1366 \times 10^{7}$	0.0	$.1739 \times 10^{7}$	$.4351 \times 10^{8}$	$.6590 \times 10^{9}$	$.3388 \times 10^{8}$
6	58.50	790.375	127.50	$.1735 \times 10^{7}$	0.0	$.2098 \times 10^{7}$	$.8080 \times 10^{8}$	$.9315 \times 10^{9}$	$.6687 \times 10^{8}$
6	60.75	790.375	125.625	$.1910 \times 10^{7}$	0.0	$.1504 \times 10^{7}$	$.5780 \times 10^{8}$	$.5865 \times 10^{9}$	$.1838 \times 10^{8}$
7	30.50	767.50	47.625	$.1889 \times 10^{8}$	$.2283 \times 10^{8}$	$.1889 \times 10^{8}$	$.1726 \times 10^{11}$	$.2014 \times 10^{11}$	$.7079 \times 10^{10}$
7	31.25	771.81	0.0	$.1366 \times 10^{7}$	0.0	$.1739 \times 10^{7}$	$.4351 \times 10^{8}$	$.6590 \times 10^{9}$	$.3388 \times 10^{8}$
7	62.50	771.875	47.625	$.2260 \times 10^{7}$	0.0	$.1790 \times 10^{7}$	$.4565 \times 10^{8}$	$.1709 \times 10^{10}$	$.5768 \times 10^{8}$
7	31.25	771.81	95.25	$.1366 \times 10^{7}$	0.0	$.1739 \times 10^{7}$	$.4351 \times 10^{8}$	$.6590 \times 10^{9}$	$.3388 \times 10^{8}$

TABLE 3.7B-25
SAFEGUARDS BUILDING FOUNDATION SPRING CONSTANTS

(Sheet 3 of 3)

Mass No.	Spring Location X (ft)	Spring Location Y (ft)	Spring Location Z (ft)	Upper Bound Foundation Spring Constants: Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
5	92.22	806.50	149.62	$.4452 \times 10^{8}$	$.5626 \times 10^{8}$	$.4452 \times 10^{8}$	$.2376 \times 10^{11}$	$.8374 \times 10^{11}$	$.8204 \times 10^{11}$
5	58.50	801.125	127.5	$.3470 \times 10^{7}$	0.0	$.4196 \times 10^{7}$	$.1616 \times 10^{9}$	$.1863 \times 10^{10}$	$.1337 \times 10^{9}$
5	60.75	801.125	125.625	$.3820 \times 10^{7}$	0.0	$.3008 \times 10^{7}$	$.1156 \times 10^{9}$	$.1173 \times 10^{10}$	$.3676 \times 10^{8}$
6	53.93	785.00	90.71	$.3784 \times 10^{8}$	$.5156 \times 10^{8}$	$.3784 \times 10^{8}$	$.7904 \times 10^{11}$	$.8534 \times 10^{11}$	$.3724 \times 10^{11}$
6	31.25	780.44	0.0	$.2732 \times 10^{7}$	0.0	$.3478 \times 10^{7}$	$.8702 \times 10^{8}$	$.1318 \times 10^{10}$	$.6776 \times 10^{8}$
6	62.50	780.625	47.625	$.4520 \times 10^{7}$	0.0	$.3580 \times 10^{7}$	$.9130 \times 10^{8}$	$.3418 \times 10^{10}$	$.1154 \times 10^{9}$
6	31.25	780.44	95.25	$.2732 \times 10^{7}$	0.0	$.3478 \times 10^{7}$	$.8702 \times 10^{8}$	$.1318 \times 10^{10}$	$.6776 \times 10^{8}$
6	58.50	790.375	127.50	$.3470 \times 10^{7}$	0.0	$.4196 \times 10^{7}$	$.1616 \times 10^{9}$	$.1863 \times 10^{10}$	$.1337 \times 10^{9}$
6	60.75	790.375	125.625	$.3820 \times 10^{7}$	0.0	$.3008 \times 10^{7}$	$.1156 \times 10^{9}$	$.1173 \times 10^{10}$	$.3676 \times 10^{8}$
7	30.50	767.50	47.625	$.3778 \times 10^{8}$	$.4566 \times 10^{8}$	$.3778 \times 10^{8}$	$.3452 \times 10^{11}$	$.4028 \times 10^{11}$	$.1416 \times 10^{11}$
7	31.25	771.81	0.0	$.2732 \times 10^{7}$	0.0	$.3478 \times 10^{7}$	$.8702 \times 10^{8}$	$.1318 \times 10^{10}$	$.6776 \times 10^{8}$
7	62.50	771.875	47.625	$.4520 \times 10^{7}$	0.0	$.3580 \times 10^{7}$	$.9130 \times 10^{8}$	$.3418 \times 10^{10}$	$.1154 \times 10^{9}$
7	31.25	771.81	95.25	$.2732 \times 10^{7}$	0.0	$.3478 \times 10^{7}$	$.8702 \times 10^{8}$	$.1318 \times 10^{10}$	$.6776 \times 10^{8}$

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate area see Figure 3.7B-25.

TABLE 3.7B-26
ELECTRICAL BUILDING FOUNDATION SPRING CONSTANTS

(Sheet 1 of 2)

Lower Bound Foundation Spring Constants

Mass No.	Spring Location X (ft)	Y (ft)	Z (ft)	Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
5	98.75	778.50	0.0	0.1587×10^7	0.0	0.1195×10^7	0.0	0.3626×10^{10}	0.0
5	48.75	772.25	0.0	0.7970×10^7	$.9632 \times 10^7$	0.7970×10^7	2.1985×10^{10}	2.4830×10^{10}	0.8027×10^{10}
5	48.75	787.875	-82.75	0.1225×10^7	0.0	0.1615×10^7	0.0	0.1345×10^{10}	0.0
5	48.75	787.875	82.75	0.1225×10^7	0.0	0.1615×10^7	0.0	0.1345×10^{10}	0.0

Best Estimate Foundation Spring Constants

Mass No.	Spring Location X (ft)	Y (ft)	Z (ft)	Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
5	98.75	778.50	0.0	0.6355×10^7	0.0	0.4782×10^7	0.0	1.4506×10^{10}	0.0
5	48.75	772.25	0.0	3.1880×10^7	3.8530×10^7	3.1880×10^7	8.7940×10^{10}	9.9320×10^{10}	3.2110×10^{10}
5	48.75	787.875	-82.75	0.4900×10^7	0.0	0.6460×10^7	0.0	0.5382×10^{10}	0.0
5	48.75	787.875	82.75	0.4900×10^7	0.0	0.6460×10^7	0.0	0.5382×10^{10}	0.0

TABLE 3.7B-26
ELECTRICAL BUILDING FOUNDATION SPRING CONSTANTS

(Sheet 2 of 2)

Mass No.	Spring Location X (ft)	Spring Location Y (ft)	Spring Location Z (ft)	Upper Bound Foundation Spring Constants: Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
5	98.75	778.50	0.0	1.2700×10^{7}	0.0	0.9564×10^{7}	0.0	2.9012×10^{10}	0.0
5	48.75	772.25	0.0	6.3760×10^{7}	7.7060×10^{7}	6.3760×10^{7}	1.7588×10^{11}	19.8640×10^{10}	6.4220×10^{10}
5	48.75	787.875	-82.75	0.9800×10^{7}	0.0	1.2920×10^{7}	0.0	1.0764×10^{10}	0.0
5	48.75	787.875	82.75	0.9800×10^{7}	0.0	1.2920×10^{7}	0.0	1.0764×10^{10}	0.0

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate area see Figure 3.7B-28.

TABLE 3.7B-27
ELECTRICAL BUILDING FOUNDATION SPRING CONSTANTS

(Sheet 1 of 2)

Lower Bound Foundation Spring Constants

Mass No.	Spring Location X (ft)	Y (ft)	Z (ft)	Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
11	167.25	784.75	0.0	1.0060×10^{7}	1.2160×10^{7}	1.0060×10^{10}	3.7550×10^{10}	4.6800×10^{10}	1.9015×10^{10}
11	235.75	795.00	0.0	0.2117×10^{7}	0.0	0.1650×10^{7}	0.0	0.6535×10^{10}	0.0

Best Estimate Foundation Spring Constants

Mass No.	Spring Location X (ft)	Y (ft)	Z (ft)	Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
11	167.25	784.75	0.0	4.0240×10^{7}	4.8640×10^{7}	4.0240×10^{7}	15.0200×10^{10}	1.8720×10^{10}	7.6060×10^{10}
11	235.75	795.00	0.0	0.8466×10^{7}	0.0	0.6604×10^{7}	0.0	2.6139×10^{10}	0.0

TABLE 3.7B-27
ELECTRICAL BUILDING FOUNDATION SPRING CONSTANTS

(Sheet 2 of 2)

Mass No.	Spring Location X (ft)	Spring Location Y (ft)	Spring Location Z (ft)	Upper Bound Foundation Spring Constants: Translation Along X Axis (kip/ft)	Translation Along Y Axis (kip/ft)	Translation Along Z Axis (kip/ft)	Rotation About X Axis (kip-ft/rad)	Rotation About Y Axis (kip-ft/rad)	Rotation About Z Axis (kip-ft/rad)
11	167.25	784.75	0.0	8.0480×10^{7}	9.7280×10^{7}	8.0480×10^{7}	3.0040×10^{11}	3.7440×10^{11}	1.5212×10^{11}
11	235.75	795.00	0.0	1.6940×10^{7}	0.0	1.3200×10^{7}	0.0	5.2280×10^{10}	0.0

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate area see Figure 3.7B-28.

TABLE 3.7B-28
FUEL BUILDING FOUNDATION SPRING CONSTANTS

Direction	Upper Bound	Lower Bound
Translation along X axis (kip/ft)	3.9913×10^{7}	1.99565×10^{7}
Translation along Y axis (kip/ft)	5.4593×10^{7}	2.72965×10^{7}
Translation along Z axis (kip/ft)	3.9913×10^{7}	1.99565×10^{7}
Rotation along X axis (kip-ft/rad)	2.4524×10^{11}	1.22620×10^{11}
Rotation along Y axis (kip-ft/rad)	1.3182×10^{11}	6.59100×10^{10}
Rotation along Z axis (kip-ft/rad)	2.1453×10^{11}	1.07265×10^{11}

Notes:

a. The origin of coordinates is at elevation 0 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-30.

TABLE 3.7B-29
SERVICE WATER INTAKE STRUCTURE FOUNDATION SPRING CONSTANTS

Direction	Upper Bound	Lower Bound
Translation along X axis (kip/ft)	5.21×10^{7}	2.60×10^{7}
Translation along Y axis (kip/ft)	4.14×10^{7}	2.07×10^{7}
Translation along Z axis (kip/ft)	5.21×10^{7}	2.60×10^{7}
Rotation along X axis (kip-ft/rad)	1.28×10^{11}	0.64×10^{11}
Rotation along Y axis (kip-ft/rad)	1.94×10^{11}	0.97×10^{11}
Rotation along Z axis (kip-ft/rad)	7.72×10^{10}	3.86×10^{10}

Notes:

a. The origin of coordinates is at elevation 776 ft 0 in.

b. For orientation of coordinate axes, see Figure 3.7B-32.

c. Spring constants include embedment effects.

TABLE 3.7B-30
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, LOWER BOUND, SSE

(Sheet 1 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	2.574	4.03	-0.74	52.06
2	2.577	51.87	-0.71	-4.03
3	5.664	0.16	-6.38	35.01
4	5.965	-25.94	63.51	1.65
5	6.068	28.56	60.37	3.26
6	6.369	-9.68	-2.02	7.42
7	8.575	-1.41	-0.18	24.63
8	9.182	25.84	-1.14	18.95
9	9.256	-20.83	0.36	20.85
10	12.254	5.70	1.37	22.12
11	12.422	22.37	2.52	-6.72
12	13.000	8.31	-15.72	1.95
13	15.056	-0.51	-0.75	1.57
14	15.633	0.15	0.96	-3.05
15	16.338	1.72	0.79	0.89
16	17.835	0.11	0.22	6.31
17	18.685	-3.98	-2.30	-0.51
18	20.265	-1.05	-0.38	1.04
19	20.523	-1.20	-0.61	-3.00
20	20.944	3.31	1.35	3.18
21	21.087	-3.46	-1.27	3.81
22	21.629	-0.76	0.97	-0.43
23	22.904	0.57	-0.85	0.47
24	23.099	-2.14	1.81	0.18
25	24.515	0.05	-0.01	-0.32
26	25.593	0.30	0.77	-0.03
27	26.181	0.17	-0.08	2.66
28	26.345	-2.90	0.13	0.19

TABLE 3.7B-30
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, LOWER BOUND, SSE

(Sheet 2 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
29	28.101	1.12	-1.84	-0.35
30	28.280	-0.07	0.03	-0.79
31	28.455	-0.81	-0.76	-0.07
32	29.221	0.08	-0.01	-0.25
33	29.602	0.22	0.88	0.23
34	32.176	0.01	0.05	-0.73
35	32.491	0.18	-0.13	1.60
36	35.039	-1.35	-0.06	0.06
37	37.042	0.11	0.02	-0.14
38	38.465	-0.02	-0.64	-0.02
39	38.730	0.05	0.14	-0.19
40	39.202	-0.21	0.14	-0.06
41	39.440	-0.05	0.11	0.11
42	40.519	-0.00	-0.00	0.04
43	41.729	0.06	0.02	-0.00
44	44.048	0.01	-0.27	0.04
45	49.042	0.03	-0.38	-0.00
46	52.598	0.05	-0.00	0.11
47	52.664	-0.11	0.02	0.05
48	53.887	0.02	-0.34	-0.00
49	56.683	-0.01	0.00	-0.06
50	56.705	0.07	-0.03	-0.02
51	59.069	-0.00	0.18	-0.00
52	69.662	-0.03	-0.07	-0.02
53	73.651	-0.00	-0.04	0.04
54	81.116	0.01	-0.06	-0.00

TABLE 3.7B-31
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, BEST ESTIMATE, SSE

(Sheet 1 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	3.289	3.26	-0.26	44.06
2	3.290	43.86	-0.26	-3.25
3	6.317	0.91	-1.21	34.84
4	7.006	38.38	-3.67	-0.62
5	7.462	-2.89	0.23	2.68
6	9.019	-2.22	-0.51	13.02
7	9.672	2.44	72.63	0.71
8	10.238	7.31	-0.75	23.27
9	10.251	22.35	-1.39	-6.96
10	14.686	12.12	40.48	6.38
11	15.224	-1.46	-8.75	28.55
12	16.676	29.58	-21.94	-0.57
13	18.522	1.30	2.13	9.06
14	18.765	4.09	7.17	-2.95
15	19.173	1.01	-0.91	-5.77
16	19.583	2.12	1.70	10.20
17	20.124	7.92	1.55	1.13
18	20.632	1.16	0.89	2.04
19	20.961	-0.74	-0.40	2.82
20	22.412	8.65	12.58	1.78
21	23.357	-0.53	-0.60	22.20
22	23.837	-18.69	0.95	-0.24
23	24.885	3.23	4.20	-9.53
24	26.140	-3.54	-6.77	-0.04
25	26.241	0.13	2.19	0.56
26	27.501	-9.33	-3.83	9.80
27	27.615	9.06	0.97	12.21
28	28.316	0.26	-0.32	-1.45

TABLE 3.7B-31
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, BEST ESTIMATE, SSE

(Sheet 2 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
29	28.682	-0.75	-6.47	1.05
30	28.817	4.15	0.74	8.58
31	29.251	-15.59	4.42	2.89
32	30.440	1.34	0.02	-7.45
33	31.276	0.40	-4.13	-3.58
34	33.304	-0.64	-0.11	3.35
35	33.645	1.25	-0.55	10.75
36	35.739	8.76	0.16	-0.24
37	37.197	-1.09	-0.09	1.34
38	38.610	-0.27	-3.03	0.09
39	39.486	1.74	-0.48	0.27
40	39.981	-0.55	0.74	0.65
41	40.158	-0.03	0.00	0.76
42	41.576	-0.00	-0.20	-1.11
43	42.263	0.03	-0.13	-0.09
44	44.258	0.03	-1.20	0.30
45	49.154	0.16	-1.67	-0.04
46	52.958	0.32	-0.01	0.60
47	53.039	-0.60	-0.00	0.33
48	54.006	0.20	-1.50	-0.01
49	56.811	0.10	-0.05	0.38
50	56.850	-0.41	0.17	0.11
51	59.109	-0.06	0.81	-0.00
52	69.706	-0.18	-0.33	-0.12
53	73.707	-0.04	-0.20	0.21
54	81.140	-0.08	0.25	0.04

TABLE 3.7B-32
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, UPPER BOUND, SSE

(Sheet 1 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	3.455	2.93	-0.12	41.23
2	3.456	41.09	-0.07	42.93
3	6.523	0.87	-0.88	34.22
4	7.339	37.32	-2.83	-0.47
5	7.642	-2.27	0.15	1.47
6	9.138	-2.39	-0.12	12.39
7	10.372	6.98	0.57	31.44
8	10.376	20.94	1.26	-6.79
9	10.628	0.31	51.84	0.13
10	15.806	4.48	6.72	23.53
11	16.363	18.71	25.50	4.80
12	18.818	17.93	-18.08	0.96
13	18.974	-1.22	1.00	11.52
14	19.477	7.39	-27.49	-0.32
15	19.678	-1.54	-1.86	3.60
16	20.271	0.68	3.17	-4.27
17	20.537	3.97	0.59	0.31
18	20.662	-0.42	-1.11	-2.25
19	21.491	0.17	0.23	-0.92
20	24.138	3.19	19.73	0.94
21	24.737	-0.03	-0.39	15.31
22	24.812	13.43	-1.62	-0.01
23	25.476	1.02	-2.86	-0.33
24	27.638	0.19	0.17	6.03
25	27.704	4.30	-1.46	-0.32
26	28.566	1.66	0.77	1.02
27	28.570	1.35	1.09	-1.05
28	29.988	0.58	-1.43	14.24

TABLE 3.7B-32
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS UNCRACKED MODEL, UPPER BOUND, SSE

(Sheet 2 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
29	30.654	0.61	-0.29	2.82
30	32.704	18.02	5.99	-0.60
31	34.011	-3.42	21.82	0.40
32	35.289	-0.95	0.58	0.50
33	36.745	1.85	0.70	-0.66
34	36.745	0.59	-0.71	-3.04
35	37.206	-1.42	-0.14	4.46
36	39.727	-13.63	2.10	-3.79
37	40.369	0.07	-0.27	-3.39
38	40.558	-0.17	9.57	17.27
39	40.909	-2.65	-7.10	24.17
40	41.954	24.56	1.65	1.44
41	43.860	3.08	-7.11	-2.29
42	48.424	1.94	2.43	0.11
43	48.600	0.00	1.49	-2.03
44	50.527	-0.24	5.49	0.00
45	55.746	0.71	2.44	0.06
46	55.809	0.05	0.35	-0.70
47	56.278	-0.04	5.92	-0.01
48	60.814	-0.40	-6.43	-0.03
49	62.444	-0.09	-0.10	1.27
50	68.033	0.40	-1.90	1.53
51	70.793	1.03	-2.65	-1.23
52	76.486	2.61	0.50	0.04
53	80.807	-0.89	1.14	-2.28
54	83.946	1.21	-1.30	-1.20

TABLE 3.7B-33
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, LOWER BOUND, SSE

(Sheet 1 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	2.124	3.81	-0.45	47.31
2	2.125	47.21	-0.42	-3.81
3	5.013	3.06	0.15	-6.51
4	5.260	2.96	-6.09	42.73
5	5.512	44.36	-23.07	-3.45
6	5.671	13.36	80.54	2.59
7	7.117	0.29	-0.34	21.11
8	7.223	15.39	-0.69	-0.05
9	8.365	-0.64	-0.06	5.46
10	9.746	4.51	28.16	2.62
11	10.913	3.71	-2.37	35.41
12	11.138	35.00	-3.61	-3.83
13	13.008	-0.33	0.87	-5.29
14	13.257	4.86	0.08	-1.39
15	13.356	-2.59	0.22	-2.25
16	14.355	-0.34	0.02	-4.69
17	14.555	-6.71	1.00	0.63
18	16.117	-0.55	-0.70	3.48
19	16.806	1.99	0.56	0.76
20	17.325	-0.64	-0.35	-2.76
21	17.445	-0.78	-0.14	0.80
22	17.614	-0.41	0.14	-0.08
23	17.850	-0.78	1.76	0.02
24	19.269	-0.53	-0.16	5.78
25	19.593	3.84	1.34	1.05
26	20.024	-0.82	-0.33	1.09
27	20.219	-3.05	-0.88	-3.49
28	20.319	-2.35	-0.50	3.82

TABLE 3.7B-33
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, LOWER BOUND, SSE

(Sheet 2 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
29	20.689	3.59	1.73	3.23
30	21.174	3.12	1.47	-3.60
31	22.172	0.21	0.43	0.42
32	23.212	-3.89	1.88	-0.17
33	23.301	-0.87	0.31	1.06
34	26.483	-0.06	0.09	0.12
35	27.299	0.09	0.01	-0.56
36	27.692	0.06	-1.38	-0.01
37	28.018	-0.02	0.08	-0.54
38	28.661	0.03	-0.05	-0.14
39	29.212	-0.34	1.88	0.34
40	32.234	-0.16	0.17	-1.50
41	34.961	-1.23	-0.07	0.05
42	37.119	0.35	-0.02	0.07
43	37.208	-0.10	0.02	0.15
44	37.857	0.08	-0.24	-0.23
45	37.953	-0.01	-0.51	0.12
46	39.945	-0.16	0.02	-0.04
47	40.061	-0.05	0.04	0.12
48	41.664	-0.00	0.27	0.01
49	41.796	-0.00	-0.06	0.02
50	44.157	-0.00	-0.31	0.07
51	49.352	0.06	-0.48	-0.02
52	69.633	-0.03	-0.07	-0.02
53	73.620	-0.00	-0.04	0.04
54	81.101	0.01	-0.06	-0.00

TABLE 3.7B-34
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, BEST ESTIMATE, SSE

(Sheet 1 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	2.441	3.26	-0.13	41.00
2	2.441	40.95	-0.12	-3.26
3	5.399	0.38	0.00	-0.67
4	6.243	1.21	-1.37	39.72
5	6.856	45.55	-4.27	-1.14
6	7.320	0.25	0.65	18.99
7	7.387	6.37	9.39	-0.00
8	7.461	0.83	56.16	0.13
9	8.990	-1.79	-0.14	10.23
10	13.254	4.48	4.18	18.19
11	13.278	16.68	14.63	-3.70
12	13.655	3.14	60.53	5.18
13	14.451	1.38	2.58	-7.09
14	14.458	6.27	-2.44	1.10
15	15.104	-0.00	-0.35	-1.80
16	15.293	1.10	-4.02	26.09
17	16.428	27.84	-14.08	-1.04
18	17.528	-0.64	-0.17	-8.01
19	17.600	-3.79	0.14	0.86
20	18.040	6.14	-12.29	0.01
21	19.073	1.99	2.09	-1.81
22	19.257	-0.66	-1.95	1.61
23	19.435	-1.94	0.33	3.40
24	19.831	-1.01	-0.81	4.08
25	19.901	2.63	0.63	0.45
26	20.155	-1.55	-1.28	1.97
27	21.618	0.53	0.57	-2.22
28	22.379	5.07	12.10	1.29

TABLE 3.7B-34
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, BEST ESTIMATE, SSE

(Sheet 2 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
29	24.264	10.65	7.58	-0.37
30	24.635	1.05	-0.06	-12.99
31	24.980	-0.77	0.31	5.72
32	25.651	1.92	-0.71	26.71
33	26.519	-7.36	-0.06	-2.31
34	26.613	27.71	0.28	-1.48
35	28.514	-1.03	7.28	-0.34
36	28.521	-1.11	6.05	0.56
37	38.126	-0.65	1.02	-8.52
38	31.134	2.92	-5.71	-4.38
39	32.955	-0.14	0.75	-9.49
40	33.507	-2.06	0.41	-3.68
41	35.547	-7.46	-0.31	0.39
42	37.687	3.03	0.32	0.52
43	37.824	0.39	0.87	-1.05
44	38.107	1.06	-2.53	-0.54
45	38.725	-0.22	-0.34	1.87
46	40.028	-1.30	0.18	-0.38
47	40.334	-0.55	0.36	1.21
48	41.703	0.05	-1.30	-0.12
49	42.475	-0.70	0.25	-0.33
50	44.475	-0.10	-1.36	0.68
51	49.594	0.41	-2.17	-0.12
52	69.670	-0.17	-0.31	-0.11
53	73.669	-0.04	-0.19	0.20
54	81.123	-0.08	0.25	0.04

TABLE 3.7B-35
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, UPPER BOUND, SSE

(Sheet 1 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	2.502	2.95	-0.05	39.49
2	2.502	39.46	-0.03	-2.95
3	5.459	0.16	-0.00	-0.27
4	6.501	0.99	-0.94	37.09
5	7.229	43.28	-2.52	-0.73
6	7.352	0.23	0.11	19.62
7	7.438	7.95	-2.92	-0.15
8	7.712	0.94	46.33	0.09
9	9.131	-2.13	-0.08	11.00
10	13.439	1.56	0.12	13.85
11	13.446	13.03	0.95	-1.40
12	14.541	-0.11	-0.44	-3.93
13	14.543	3.92	0.04	-0.02
14	15.329	0.13	0.32	-0.08
15	15.808	-5.32	10.81	22.44
16	16.235	16.43	32.13	-6.41
17	17.394	-11.81	14.47	0.11
18	17.620	1.46	-0.15	-8.70
19	17.638	6.53	-1.40	-1.47
20	19.330	12.88	-26.39	0.10
21	19.484	-0.86	2.15	2.37
22	19.549	-4.04	15.52	-0.06
23	19.707	-1.64	-2.61	3.76
24	19.942	-0.02	0.98	-1.69
25	19.950	1.41	0.84	0.16
26	20.398	0.77	3.02	-1.34
27	21.904	0.09	0.13	-0.53
28	24.254	-1.26	-20.44	-0.72

TABLE 3.7B-35
CONTAINMENT AND INTERNAL STRUCTURES MODAL FREQUENCIES AND PARTICIPATION FACTORS CRACKED MODEL, UPPER BOUND, SSE

(Sheet 2 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
29	25.308	0.07	-0.01	-0.68
30	26.257	1.23	4.68	-0.45
31	26.554	-0.27	0.00	0.52
32	26.666	0.52	1.03	0.48
33	28.548	0.02	0.01	-0.38
34	29.675	0.83	-1.09	19.54
35	32.420	22.28	3.06	-0.86
36	34.720	-2.14	18.34	0.06
37	35.223	0.53	3.22	-1.50
38	35.377	0.88	-7.93	-1.77
39	37.005	-2.25	0.34	6.43
40	38.961	-15.73	3.07	-5.54
41	39.470	0.74	6.96	15.07
42	39.648	-1.35	-4.78	-15.72
43	39.689	-7.42	3.51	-6.53
44	39.952	-9.23	-6.92	20.32
45	41.471	20.52	-0.00	2.78
46	42.092	4.75	5.26	-2.40
47	44.362	2.01	-8.31	-2.04
48	53.338	1.16	12.28	-0.14
49	59.789	-0.07	-0.06	1.44
50	65.451	-0.72	1.23	-3.18
51	68.199	2.87	-2.84	-1.29
52	73.305	-2.08	-0.75	-0.67
53	78.154	0.24	-0.94	2.32
54	82.853	0.96	-1.36	-0.74

TABLE 3.7B-36
SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE

(Sheet 1 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	4.768	31.13	-2.71	-16.53
2	6.696	19.08	20.59	30.70
3	9.581	-3.96	56.19	-15.67
4	10.594	2.74	-12.57	17.20
5	13.228	27.50	-12.15	-16.98
6	15.762	19.26	15.99	18.70
7	18.075	11.04	-1.47	-7.10
8	18.526	1.83	-5.70	15.37
9	23.246	6.57	-1.94	3.00
10	24.216	-6.99	-3.07	13.24
11	24.669	11.63	2.24	5.36
12	26.456	-6.46	5.25	0.72
13	27.883	-1.63	1.26	-0.27
14	30.224	0.44	2.80	4.56
15	32.850	2.16	1.01	-2.59
16	33.823	1.12	1.41	-0.58
17	34.718	0.98	-1.98	-0.44
18	37.609	-0.47	-0.35	0.17
19	40.132	-1.58	-0.12	0.71
20	41.165	0.23	0.16	-1.07
21	41.993	-0.69	-0.21	0.06

TABLE 3.7B-36
SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE

(Sheet 2 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
22	44.005	0.17	-0.11	-0.08
23	44.050	0.66	-0.51	-0.19
24	45.674	0.01	0.26	-0.38
25	48.003	-0.43	0.21	-0.52
26	48.878	0.94	0.08	-0.30
27	51.920	-0.02	1.05	0.05
28	52.807	-0.04	1.09	0.12
29	55.348	0.50	0.20	0.04
30	59.281	0.08	-0.68	0.01
31	60.856	-0.12	0.22	0.00
32	63.802	0.14	0.03	0.00
33	65.907	0.07	-0.05	0.00
34	68.573	-0.00	-0.46	-0.01
35	74.424	0.02	0.11	0.01
36	77.527	-0.03	0.01	0.01
37	86.736	-0.02	-0.02	-0.00
38	91.109	-0.00	0.32	0.00
39	97.830	0.00	-0.08	-0.00
40	06.868	0.00	-0.00	-0.00
41	117.617	-0.00	-0.18	-0.00
42	137.447	-0.00	-0.06	-0.00

TABLE 3.7B-37
SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE

(Sheet 1 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	7.082	33.05	-1.63	-12.02
2	8.662	11.20	8.72	32.89
3	14.099	9.35	12.96	6.86
4	14.916	2.79	48.62	-12.85
5	18.540	-19.12	24.17	5.70
6	21.540	0.35	4.63	17.32
7	22.725	14.22	10.02	3.09
8	27.266	-1.76	13.72	3.47
9	29.366	3.64	6.73	13.11
10	30.466	-4.91	-19.58	9.45
11	31.796	-0.85	4.76	-3.19
12	34.137	17.68	-8.78	1.58
13	36.238	-11.75	-4.07	-3.65
14	38.271	-3.70	-4.86	-16.85
15	39.362	-1.43	8.94	-11.32
16	41.648	7.15	-3.70	-3.46
17	42.641	-11.17	-4.21	-11.94
18	43.723	6.58	1.62	1.03
19	44.481	-13.22	0.19	13.95
20	46.026	1.95	-0.06	-5.03
21	46.838	5.60	0.03	0.70

TABLE 3.7B-37
SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE

(Sheet 2 of 2)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
22	49.260	1.64	0.71	2.24
23	49.949	2.50	-0.37	0.41
24	51.666	-5.20	-0.80	2.40
25	52.897	-3.03	0.72	-0.38
26	54.688	-1.40	6.02	0.11
27	56.262	3.68	1.99	-2.70
28	58.654	3.27	0.12	0.80
29	60.134	-0.29	-0.41	-2.25
30	60.400	-1.14	2.44	-0.24
31	63.725	0.48	-2.07	0.22
32	64.124	-0.82	-1.25	0.05
33	69.447	-0.33	-1.10	-0.07
34	69.987	0.41	-1.75	-0.05
35	75.545	0.11	0.78	0.06
36	77.734	0.18	0.02	-0.06
37	87.496	0.10	0.24	0.01
38	92.059	-0.02	1.35	0.03
39	97.915	-0.02	0.40	0.02
40	106.878	-0.00	0.00	0.00
41	118.232	-0.02	-0.81	-0.04
42	137.557	-0.01	-0.27	-0.01

TABLE 3.7B-38
SERVICE WATER INTAKE STRUCTURE MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	11.313	-21.293	-1.791	0.213
2	14.626	-.5070001	3.043	-25.66
3	16.241	-1.043	47.581	1.991
4	17.707	-3.707	0.337	2.373
5	23.975	-28.005	-0.264	0.01
6	24.767	-0.155	0.998	-22.391
7	29.819	-0.363	0.288	13.161
8	33.756	-3.922	-2.451	-0.89
9	34.974	-9.597	0.324	-1.413
10	35.444	-3.913	-1.371	1.243
11	41.936	-1.891	0.861	8.776
12	43.327	-3.692	2.796	-2.289
13	46.842	-4.94	-0.888	0.158
14	49.277	-1.391	-0.141	-3.584
15	52.073	-1.658	-0.116	0.915
16	54.557	-0.759	0.269	-0.205
17	57.317	-1.173	0.076	0.382
18	59.172	-0.04	-0.063	-0.677

TABLE 3.7B-39
SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS UPPER BOUND, SSE

(Sheet 1 of 2)

Mode	Frequency (Hz)	Participation Factors		
		X-Earthquake	Y-Earthquake	Z-Earthquake
1	7.882	33.22	-1.11	-8.20
2	9.169	6.52	6.53	33.16
3	15.142	11.04	5.77	6.88
4	16.759	4.77	42.63	-10.11
5	20.776	15.97	-25.10	-5.36
6	22.414	3.32	3.36	14.15
7	25.931	-10.50	14.08	0.25
8	30.545	-2.03	8.49	11.39
9	31.660	-2.44	6.69	-6.53
10	35.141	-4.87	-22.00	3.52
11	35.591	4.09	5.04	1.63
12	37.550	-10.23	17.18	-0.27
13	39.709	8.97	4.45	-6.01
14	40.757	4.73	9.78	6.53
15	43.854	1.43	6.41	1.62
16	47.031	-15.06	8.15	-8.62
17	48.584	5.14	7.19	-3.43
18	50.038	2.51	4.55	4.31
19	52.349	4.75	-0.86	14.50
20	52.780	-3.39	-4.50	2.12
21	53.786	-4.84	2.33	-15.83

TABLE 3.7B-39
SAFEGUARDS BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS UPPER BOUND, SSE

(Sheet 2 of 2)

Mode	Frequency (Hz)	Participation Factors		
		X-Earthquake	Y-Earthquake	Z-Earthquake
22	55.488	-0.78	-8.67	-15.75
23	58.298	-7.77	-7.76	1.99
24	59.792	18.14	-1.69	-1.57
25	61.324	-4.96	-4.87	9.78
26	61.574	-0.36	-4.31	-8.73
27	63.237	-5.66	2.00	2.20
28	64.686	-12.00	0.35	5.60
29	66.737	1.46	-0.69	-2.61
30	68.534	-0.13	-7.23	-0.10
31	70.712	6.23	1.88	-0.84
32	71.898	2.00	-2.34	3.37
33	73.001	-0.38	-0.57	-5.12
34	75.630	3.23	-2.04	0.43
35	77.047	-0.22	2.50	0.06
36	78.285	-1.04	-0.60	0.35
37	88.803	-0.26	-0.96	-0.00
38	93.0569	-0.05	2.82	0.09
39	98.114	0.07	-1.14	-0.08
40	106.897	0.00	0.00	-0.00
41	119.263	0.04	1.84	0.09
42	137.744	0.02	0.63	0.03

TABLE 3.7B-40
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE

(Sheet 1 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	4.309	-3.45	3.42	53.63
2	4.615	57.29	-22.12	3.79
3	6.360	21.59	103.60	-2.22
4	6.937	-1.20	-5.31	-8.91
5	8.086	39.73	-17.46	2.02
6	8.431	1.44	-2.44	-55.24
7	9.539	13.16	-24.79	0.46
8	12.920	-28.79	-1.82	0.85
9	14.204	0.00	0.22	-0.37
10	15.470	0.60	-0.47	10.77
11	15.919	-2.65	0.19	-16.86
12	18.125	14.90	0.46	-10.31
13	18.212	8.14	0.68	15.11
14	19.347	4.92	-1.56	2.34
15	19.983	7.87	-2.15	-2.12
16	21.023	-9.06	-0.81	-0.48
17	22.377	-0.58	0.54	-9.27
18	22.525	2.64	5.71	0.92
19	22.940	5.84	1.64	-1.20
20	24.146	-0.42	0.16	-4.27
21	24.566	5.43	0.59	-0.27
22	26.007	-7.92	2.39	0.97
23	26.276	1.45	-0.42	4.49

TABLE 3.7B-40
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE

(Sheet 2 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
24	27.308	0.79	-0.34	2.68
25	27.842	0.24	-0.38	-2.45
26	29.259	3.32	1.52	0.47
27	30.170	0.51	0.13	-5.83
28	32.358	-0.07	-0.05	2.35
29	32.848	-0.63	-0.26	-0.58
30	34.566	0.09	-0.06	0.01
31	36.090	-0.06	0.02	-0.74
32	37.811	0.59	-0.04	-0.08
33	38.080	-1.36	0.07	0.18
34	39.431	0.28	-0.00	0.91
35	41.089	-0.37	-0.00	-0.71
36	41.336	-0.48	-0.00	0.58
37	43.906	-0.11	-0.02	-0.24
38	44.187	0.59	0.05	-0.10
39	44.989	0.03	0.35	0.01
40	47.079	-0.13	0.05	0.11
41	48.701	-0.01	0.01	-0.23
42	48.954	0.02	-0.16	0.02
43	50.440	0.06	-0.01	0.11
44	51.466	0.00	0.08	0.02
45	51.779	-0.01	0.03	0.07

TABLE 3.7B-40
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS LOWER BOUND, SSE

(Sheet 3 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
46	52.841	0.03	0.00	0.08
47	56.324	-0.02	-0.00	-0.05
48	58.716	0.01	-0.28	-0.00
49	60.178	0.00	0.00	0.02
50	64.398	-0.00	0.00	0.02
51	65.300	0.01	-0.05	0.00
52	69.281	0.03	0.00	-0.00
53	69.974	0.01	0.14	0.00
54	78.774	0.00	-0.02	0.01
55	85.197	-0.00	0.04	0.00
56	85.517	-0.01	0.00	-0.00
57	91.319	-0.00	-0.05	0.00
58	93.209	0.00	-0.01	-0.00
59	94.674	-0.00	0.02	-0.00
60	95.244	-0.00	0.00	0.00
61	100.430	0.00	0.00	0.00
62	101.711	0.00	0.00	0.00
63	106.190	-0.00	0.01	-0.00
64	110.577	-0.00	0.00	0.00
65	113.414	-0.00	0.00	0.00
66	113.800	-0.00	0.00	-0.00

TABLE 3.7B-41
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE

(Sheet 1 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	6.070	61.27	-3.63	-5.41
2	6.478	5.97	-0.17	61.42
3	8.708	1.74	-1.17	-6.78
4	11.194	7.76	94.79	0.01
5	12.456	23.34	-21.79	-3.03
6	12.680	-0.92	3.06	-30.31
7	14.694	11.01	36.54	0.23
8	15.613	18.85	-22.12	-0.42
9	16.403	-3.29	1.61	-6.98
10	18.671	1.94	4.63	-4.39
11	19.891	-8.71	6.27	1.07
12	20.783	0.77	0.23	18.60
13	22.109	1.39	-1.90	-11.75
14	22.413	18.45	-9.59	0.41
15	22.897	3.26	-0.36	-0.18
16	23.177	-10.82	8.85	-0.01
17	24.474	5.16	2.41	-0.70
18	24.639	13.75	22.96	-0.85
19	26.736	0.49	-0.43	-23.66
20	27.260	0.66	-0.13	5.27
21	28.337	5.02	-1.32	-3.81
22	29.000	-11.27	6.30	-9.05
23	29.040	9.16	-7.05	-7.30

TABLE 3.7B-41
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE

(Sheet 2 of 3)

Mode	Frequency (Hz)	Participation Factors		
		X-Earthquake	Y-Earthquake	Z-Earthquake
24	30.076	-0.26	0.53	10.86
25	31.354	-5.93	-8.24	0.39
26	32.998	-19.66	0.25	-0.16
27	34.047	1.39	0.01	5.08
28	36.599	2.64	0.72	-1.51
29	36.969	0.19	-0.10	18.44
30	38.039	-0.80	0.20	6.29
31	38.630	-1.42	-0.54	0.75
32	39.407	20.56	0.04	-1.62
33	39.851	5.70	-0.05	2.97
34	41.640	-5.31	-0.00	-6.36
35	41.904	11.92	0.07	-0.41
36	43.084	0.94	-0.03	-15.90
37	44.515	-3.09	-0.19	1.41
38	44.777	0.26	-0.01	7.39
39	45.109	0.28	1.58	0.08
40	47.194	-0.30	0.24	0.68
41	48.735	0.03	-0.07	1.53
42	49.017	0.21	-0.69	0.05
43	50.510	-0.29	0.09	-0.37
44	51.523	0.00	0.33	-0.07
45	51.799	-0.11	0.14	0.51

TABLE 3.7B-41
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE

(Sheet 3 of 3)

Mode	Frequency (Hz)	Participation Factors		
		X-Earthquake	Y-Earthquake	Z-Earthquake
46	52.891	0.15	0.01	0.78
47	56.336	0.10	0.03	0.26
48	58.819	0.06	-1.25	-0.00
49	60.184	-0.00	0.00	0.15
50	64.433	-0.01	0.02	0.12
51	65.364	-0.09	0.26	-0.01
52	69.300	-0.15	-0.03	0.03
53	70.023	0.07	0.61	0.03
54	78.789	0.00	-0.10	0.06
55	85.204	-0.00	0.19	0.00
56	88.522	-0.06	0.00	-0.00
57	91.331	0.00	0.21	-0.00
58	93.212	0.00	-0.05	-0.01
59	94.681	0.01	-0.10	0.00
60	95.247	0.01	-0.00	0.02
61	100.430	0.00	0.03	0.00
62	101.713	0.03	0.03	0.00
63	106.195	-0.00	0.05	-0.00
64	110.577	0.00	-0.00	-0.00
65	113.415	-0.01	0.02	-0.01
66	113.801	0.00	0.01	0.00

TABLE 3.7B-42
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE

(Sheet 1 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	6.385	60.53	-1.01	-3.62
2	7.114	4.19	-0.43	62.20
3	9.070	2.05	-0.53	-8.38
4	13.825	-9.86	76.26	4.55
5	13.862	23.69	-36.36	-1.37
6	15.231	-1.87	14.58	-22.50
7	16.628	-0.79	4.18	-2.39
8	16.874	5.39	37.73	0.82
9	18.410	10.03	-24.13	-0.12
10	19.974	-8.55	-8.20	-0.70
11	20.459	4.72	2.23	0.68
12	22.594	0.01	-1.99	-17.19
13	23.065	8.36	-6.25	-0.92
14	23.357	0.46	-0.98	2.19
15	24.502	15.40	-14.13	0.63
16	25.142	2.31	-3.73	-0.41
17	26.162	-7.27	-14.28	-0.10
18	27.292	5.37	41.10	-1.38
19	28.190	1.18	1.95	-2.21
20	29.172	0.67	-2.32	-16.59
21	31.297	4.76	-8.08	-0.08
22	31.631	5.33	-4.43	-10.71
23	31.910	4.53	-2.47	13.98

TABLE 3.7B-42
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE

(Sheet 2 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
24	32.948	-1.93	1.19	1.02
25	34.683	6.60	-21.69	0.23
26	35.956	-18.66	6.63	0.05
27	36.731	-1.47	0.24	-0.20
28	38.904	2.52	-0.44	12.43
29	39.227	2.06	2.76	1.06
30	40.951	-6.96	0.06	6.93
31	41.102	-8.02	0.12	-3.23
32	41.557	-15.43	1.27	1.09
33	42.411	0.37	-0.09	12.35
34	44.175	-0.76	-0.06	-1.61
35	45.267	2.33	-3.39	0.65
36	45.969	13.10	1.58	-3.62
37	46.418	6.57	0.43	22.52
38	47.822	15.22	-0.18	-6.82
39	48.952	-0.67	0.19	7.62
40	49.135	-0.37	1.40	-3.33
41	50.695	1.98	-0.25	6.80
42	51.594	0.08	0.67	2.69
43	51.855	-0.19	0.39	1.39
44	52.892	2.02	0.04	-1.84
45	54.565	26.38	0.00	0.13

TABLE 3.7B-42
ELECTRICAL AND AUXILIARY BUILDINGS MODAL FREQUENCIES AND PARTICIPATION FACTORS BEST ESTIMATE, SSE

(Sheet 3 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
46	56.360	0.24	0.08	2.47
47	57.441	-0.09	-0.01	22.34
48	58.991	-0.07	-2.82	-0.05
49	60.220	-0.04	-0.00	2.16
50	64.486	-0.07	0.05	0.48
51	65.468	-0.42	0.65	-0.05
52	69.329	-0.39	-0.06	0.08
53	70.094	-0.19	-1.30	-0.07
54	78.811	-0.01	0.21	-0.14
55	85.214	0.00	0.40	0.00
56	88.530	-0.14	0.02	-0.01
57	91.345	-0.01	-0.43	0.00
58	93.218	0.02	-0.10	-0.04
59	94.693	0.04	-0.22	0.01
60	95.254	-0.04	0.00	0.06
61	100.431	0.01	0.06	0.00
62	101.717	0.07	0.06	0.00
63	106.197	0.01	-0.11	0.00
64	110.579	0.00	-0.00	-0.01
65	113.418	-0.01	0.05	-0.02
66	113.801	0.00	-0.03	0.00

TABLE 3.7B-43
FUEL BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS
LOWER BOUND, SSE

(Sheet 1 of 3)

Mode	Frequency (Hz)	Participation Factors		
		X-Earthquake	Y-Earthquake	Z-Earthquake
1	8.014	46.79	12.94	4.23
2	8.395	-2.53	-12.17	49.62
3	9.260	-4.86	0.61	-6.76
4	10.372	-11.69	74.34	10.36
5	14.203	38.39	5.27	0.12
6	14.551	-0.07	-6.66	33.83
7	18.252	2.96	7.07	-2.86
8	18.578	-3.47	-2.13	-5.19
9	19.580	0.60	3.02	5.48
10	29.180	0.74	-4.43	-0.05
11	31.674	-2.72	-3.18	2.29
12	32.240	1.03	1.39	4.78
13	33.654	0.46	0.26	-0.26
14	34.636	-1.93	0.37	-1.12
15	36.847	-1.54	0.97	1.78
16	37.046	-3.90	0.55	-0.13
17	38.128	0.89	-0.25	2.12
18	40.202	-0.03	-0.36	-1.56
19	42.014	0.23	0.16	-0.22
20	43.311	-0.16	-0.75	0.48
21	46.039	0.51	0.14	0.37

TABLE 3.7B-43
FUEL BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS
LOWER BOUND, SSE

(Sheet 2 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
22	46.562	0.33	-0.02	-0.30
23	51.890	-0.18	0.29	-0.10
24	52.565	0.04	-0.06	0.00
25	52.759	0.12	0.17	-0.21
26	55.343	0.04	0.00	0.04
27	56.413	-0.19	0.90	-0.20
28	57.616	-0.30	0.40	0.30
29	59.398	0.09	-0.29	-0.43
30	62.495	-0.51	-0.08	-0.42
31	63.225	-0.30	-0.34	0.30
32	65.080	-0.15	-0.62	0.08
33	68.111	0.02	-0.23	-0.01
34	71.964	0.17	-0.00	-0.08
35	73.321	-0.20	0.09	-0.10
36	74.423	-0.20	0.07	0.04
37	78.191	-0.00	-0.05	-0.01
38	38.532	0.01	-0.02	-0.12
39	87.897	0.01	0.01	0.02
40	101.105	-0.01	-0.05	0.01
41	102.827	-0.02	-0.00	-0.12
42	105.013	0.01	0.01	-0.09

TABLE 3.7B-43
FUEL BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS
LOWER BOUND, SSE

(Sheet 3 of 3)

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
43	106.400	-0.10	-0.05	0.01
44	107.610	-0.00	-0.02	-0.07
45	109.281	-0.11	0.00	-0.00
46	111.256	-0.01	0.00	-0.02
47	113.420	-0.02	-0.02	-2.02
48	114.764	0.02	0.00	-0.00
49	115.340	0.02	0.04	-0.00
50	118.027	0.04	-0.07	-0.00
51	122.716	-0.00	0.00	-0.00
52	133.602	0.00	0.00	0.00
53	137.483	0.00	0.00	-0.00
54	139.213	-0.00	0.04	-0.00
55	172.564	0.00	0.00	0.00
56	185.389	-0.00	-0.00	-0.00
57	189.382	-0.00	-0.03	0.00
58	197.406	-0.00	0.01	-0.00
59	200.907	-0.00	-0.00	0.00
60	269.036	0.00	-0.00	0.00

TABLE 3.7B-44
FUEL BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS
UPPER BOUND, SSE

(Sheet 1 of 3)

Mode	Frequency (Hz)	Participation Factors		
		X-Earthquake	Y-Earthquake	Z-Earthquake
1	9.482	41.46	5.61	0.94
2	10.341	0.04	-6.13	46.06
3	11.725	1.12	-0.99	7.91
4	14.219	-2.81	75.50	6.00
5	17.529	42.37	-3.00	2.61
6	18.283	-6.85	-3.18	32.46
7	20.578	-3.12	2.56	-20.52
8	22.398	12.35	4.80	1.49
9	22.826	2.48	6.12	-1.61
10	29.562	-1.88	8.96	0.10
11	33.147	3.61	3.48	-9.13
12	33.712	1.19	1.15	4.34
13	33.822	-7.08	-6.22	-3.23
14	35.564	1.41	-1.64	1.25
15	37.509	-0.92	-1.50	1.10
16	38.558	-3.80	1.66	5.06
17	39.105	7.30	-2.01	2.85
18	40.455	-0.16	-0.97	-4.51
19	42.050	-0.72	-0.30	0.58
20	43.400	-0.29	-1.73	1.25
21	46.100	1.21	0.37	0.97
22	46.616	0.79	0.05	-0.72
23	51.913	0.41	-0.58	0.22

TABLE 3.7B-44
FUEL BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS
UPPER BOUND, SSE

(Sheet 2 of 3)

Mode	Frequency (Hz)	Participation Factors		
		X-Earthquake	Y-Earthquake	Z-Earthquake
24	52.598	-0.07	0.15	-0.04
25	52.781	0.28	0.35	-0.01
26	55.346	0.09	0.01	0.10
27	56.566	0.40	-1.85	0.47
28	57.734	-0.64	0.85	0.61
29	59.521	0.24	-0.65	-0.96
30	62.721	-1.03	-0.10	-1.01
31	63.381	-0.76	-0.72	0.56
32	65.320	-0.36	-1.32	0.19
33	68.151	-0.06	0.52	0.02
34	72.008	0.28	0.00	-0.16
35	73.365	0.40	-0.19	0.22
36	74.483	-0.43	0.17	0.08
37	78.203	-0.01	-0.11	-0.02
38	83.553	0.03	-0.05	-0.24
39	87.899	-0.03	-0.02	-0.05
40	101.110	-0.03	-0.11	0.03
41	102.847	0.05	0.01	0.25
42	105.024	0.02	0.02	-0.19
43	106.426	0.21	0.12	-0.03
44	107.621	0.00	0.04	0.15
45	109.302	-0.24	0.01	-0.01

TABLE 3.7B-44
FUEL BUILDING MODAL FREQUENCIES AND PARTICIPATION FACTORS
UPPER BOUND, SSE

(Sheet 3 of 3)

Mode	Frequency (Hz)	Participation Factors		
		X-Earthquake	Y-Earthquake	Z-Earthquake
46	111.259	-0.04	0.00	-0.05
47	113.426	-0.05	-0.05	-0.04
48	114.772	0.05	0.00	-0.01
49	115.350	0.04	0.09	-0.01
50	118.048	0.08	-0.15	0.00
51	122.723	0.00	0.03	-0.00
52	133.608	0.00	0.00	0.00
53	137.490	0.00	0.00	-0.00
54	139.228	0.01	-0.09	0.00
55	172.565	0.00	0.00	0.00
56	185.393	-0.00	-0.00	-0.00
57	189.393	-0.00	-0.06	0.00
58	197.410	-0.00	0.03	-0.00
59	200.912	0.00	0.00	-0.00
60	269.038	0.00	-0.00	0.00

TABLE 3.7B-45
SERVICE WATER INTAKE STRUCTURE MODAL FREQUENCIES AND
PARTICIPATION FACTORS UPPER BOUND, SSE

		Participation Factors		
Mode	Frequency (Hz)	X-Earthquake	Y-Earthquake	Z-Earthquake
1	11.808	-19.204	-0.995	0.186
2	16.07	-0.466	1.042	-23.067
3	18.161	-2.899	-0.288	2.952
4	22.329	-0.714	47.003	1.004
5	27.753	-23.194	-0.41	-0.318
6	31.033	-0.3	-1.656	21.343
7	35.528	-3.489	0.61	-5.757
8	35.606	-4.602	2.56	6.371
9	39.008	-10.54	1.926	-1.433
10	41.556	-8.903	-7.544	-0.517
11	47.207	-10.72	3.002	6.139
12	47.285	-3.796	1.4	-14.066
13	51.626	-12.732	-0.8	0.785
14	52.885	-1.152	-0.052	-11.36
15	55.16	-0.838	-0.038	0.066
16	59.357	-0.354	0.197	2.053
17	61.053	-3.046	0.465	-0.466
18	66.59	-1.022	0.073	0.668

TABLE 3.7B-46
CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 1 of 2)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
1	0.67	0.11	0.09
2	0.11	0.57	0.03
3	0.09	0.04	0.68
4	0.11×10^{-2}	0.13×10^{-2}	0.36×10^{-2}
5	0.13×10^{-2}	0.03×10^{-2}	0.14×10^{-2}
6	0.39×10^{-2}	0.45×10^{-2}	0.12×10^{-2}
7	0.51	0.07	0.06
8	0.09	0.44	0.05
9	0.06	0.02	0.51
10	0.08×10^{-2}	0.05×10^{-2}	0.33×10^{-2}
11	0.13×10^{-2}	0.03×10^{-2}	0.12×10^{-2}
12	0.32×10^{-2}	0.11×10^{-2}	0.08×10^{-2}
13	0.39	0.07	0.05
14	0.08	0.40	0.05
15	0.05	0.02	0.40
16	0.06×10^{-2}	0.04×10^{-2}	0.29×10^{-2}
17	0.12×10^{-2}	0.03×10^{-2}	0.11×10^{-2}
18	0.28×10^{-2}	0.08×10^{-2}	0.06×10^{-2}
19	0.29	0.09	0.06
20	0.07	0.32	0.03
21	0.06	0.03	0.31
22	0.04×10^{-2}	0.03×10^{-2}	0.22×10^{-2}
23	0.10×10^{-2}	0.02×10^{-2}	0.08×10^{-2}
24	0.21×10^{-2}	0.10×10^{-2}	0.04×10^{-2}
25	0.17	0.08	0.04
26	0.05	0.26	0.02
27	0.04	0.03	0.18
28	0.02×10^{-2}	0.05×10^{-2}	0.15×10^{-2}
29	0.06×10^{-2}	0.01×10^{-2}	0.07×10^{-2}
30	0.15×10^{-2}	0.08×10^{-2}	0.03×10^{-2}

TABLE 3.7B-46
CONTAINMENT AND INTERNAL STRUCTURES SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 2 of 2)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
31	0.41	0.11	0.04
32	0.11	0.27	0.05
33	0.04	0.07	0.42
34	0.09×10^{-2}	0.12×10^{-2}	0.34×10^{-2}
35	0.09×10^{-2}	0.04×10^{-2}	0.34×10^{-2}
36	0.42×10^{-2}	0.44×10^{-2}	0.14×10^{-2}
37	0.26	0.10	0.03
38	0.06	0.25	0.03
39	0.03	0.06	0.25
40	0.07×10^{-2}	0.08×10^{-2}	0.27×10^{-2}
41	0.08×10^{-2}	0.05×10^{-2}	0.20×10^{-2}
42	0.27×10^{-2}	0.19×10^{-2}	0.09×10^{-2}
43	0.20	0.10	0.02
44	0.04	0.24	0.03
45	0.03	0.05	0.17
46	0.06×10^{-2}	0.06×10^{-2}	0.20×10^{-2}
47	0.06×10^{-2}	0.04×10^{-2}	0.14×10^{-2}
48	0.22×10^{-2}	0.10×10^{-2}	0.06×10^{-2}
49	0.14	0.04	0.02
50	0.04	0.23	0.02
51	0.02	0.02	0.14
52	0.02×10^{-2}	0.02×10^{-2}	0.13×10^{-2}
53	0.04×10^{-2}	0.01×10^{-2}	0.03×10^{-2}
54	0.13×10^{-2}	0.03×10^{-2}	0.02×10^{-2}

a) Degrees of freedom corresponding to translations in the x, y, and z directions, and rotations about x, y, and z axes are identified in Table 3.7B-19.

b) Tabulated accelerations are in units of (g) for translational degrees of freedom, and (g/ft) for rotational degrees of freedom.

TABLE 3.7B-47
SAFEGUARDS BUILDING SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 1 of 2)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
1	0.38	0.13	0.23
2	0.09	0.30	0.13
3	0.22	0.22	0.37
4	0.20×10^{-2}	0.30×10^{-2}	0.18×10^{-2}
5	0.28×10^{-2}	0.27×10^{-2}	0.37×10^{-2}
6	0.27×10^{-2}	0.32×10^{-2}	0.19×10^{-2}
7	0.28	0.08	0.17
8	0.05	0.22	0.09
9	0.18	0.19	0.28
10	0.19×10^{-2}	0.25×10^{-2}	0.16×10^{-2}
11	0.20×10^{-2}	0.23×10^{-2}	0.19×10^{-2}
12	0.25×10^{-2}	0.25×10^{-2}	0.18×10^{-2}
13	0.21	0.08	0.14
14	0.04	0.20	0.07
15	0.14	0.16	0.21
16	0.18×10^{-2}	0.20×10^{-2}	0.16×10^{-2}
17	0.18×10^{-2}	0.20×10^{-2}	0.17×10^{-2}
18	0.24×10^{-2}	0.20×10^{-2}	0.17×10^{-2}
19	0.15	0.09	0.10
20	0.04	0.18	0.07
21	0.10	0.11	0.13
22	0.17×10^{-2}	0.17×10^{-2}	0.14×10^{-2}
23	0.14×10^{-2}	0.14×10^{-2}	0.13×10^{-2}

TABLE 3.7B-47
SAFEGUARDS BUILDING SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 2 of 2)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
24	0.21×10^{-2}	0.14×10^{-2}	0.15×10^{-2}
25	0.12	0.05	0.07
26	0.05	0.15	0.06
27	0.06	0.07	0.12
28	0.16×10^{-2}	0.13×10^{-2}	0.14×10^{-2}
29	0.08×10^{-2}	0.09×10^{-2}	0.09×10^{-2}
30	0.20×10^{-2}	0.13×10^{-2}	0.14×10^{-2}
31	0.12	0.04	0.05
32	0.03	0.17	0.07
33	0.05	0.04	0.12
34	0.15×10^{-2}	0.11×10^{-2}	0.13×10^{-2}
35	0.08×10^{-2}	0.06×10^{-2}	0.10×10^{-2}
36	0.19×10^{-2}	0.11×10^{-2}	0.12×10^{-2}
37	0.12	0.05	0.05
38	0.04	0.18	0.08
39	0.05	0.04	0.12
40	0.14×10^{-2}	0.10×10^{-2}	0.13×10^{-2}
41	0.21×10^{-2}	0.05×10^{-2}	0.10×10^{-2}
42	0.18×10^{-2}	0.09×10^{-2}	0.12×10^{-2}

a) Degrees of freedom corresponding to translations in the x, y, and z directions, and rotations about x, y, and z axes are identified in Table 3.7B-20.

b) Tabulated accelerations are in units of (g) for translational degrees of freedom, and (g/ft) for rotational degrees of freedom.

TABLE 3.7B-48
ELECTRICAL AND AUXILIARY BUILDING SRSS ACCELERATIONS FOR SSE
FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 1 of 3)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
1	0.33	0.15	0.05
2	0.12	0.26	0.02
3	0.05	0.03	0.30
4	0.05×10^{-2}	0.05×10^{-2}	0.24×10^{-2}
5	0.05×10^{-2}	0.03×10^{-2}	0.10×10^{-2}
6	0.23×10^{-2}	0.28×10^{-2}	0.04×10^{-2}
7	0.28	0.14	0.04
8	0.12	0.24	0.01
9	0.04	0.03	0.28
10	0.05×10^{-2}	0.05×10^{-2}	0.23×10^{-2}
11	0.05×10^{-2}	0.03×10^{-2}	0.07×10^{-2}
12	0.22×10^{-2}	0.27×10^{-2}	0.03×10^{-2}
13	0.20	0.11	0.03
14	0.11	0.20	0.01
15	0.03	0.03	0.24
16	0.04×10^{-2}	0.04×10^{-2}	0.23×10^{-2}
17	0.04×10^{-2}	0.03×10^{-2}	0.06×10^{-2}
18	0.19×10^{-2}	0.24×10^{-2}	0.03×10^{-2}
19	0.14	0.08	0.02
20	0.11	0.19	0.01
21	0.03	0.02	0.17
22	0.03×10^{-2}	0.03×10^{-2}	0.22×10^{-2}
23	0.04×10^{-2}	0.02×10^{-2}	0.06×10^{-2}
24	0.18×10^{-2}	0.24×10^{-2}	0.02×10^{-2}

TABLE 3.7B-48
ELECTRICAL AND AUXILIARY BUILDING SRSS ACCELERATIONS FOR SSE
FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 2 of 3)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
25	0.12	0.04	0.01
26	0.10	0.18	0.01
27	0.02	0.01	0.12
28	0.03×10^{-2}	0.02×10^{-2}	0.21×10^{-2}
29	0.03×10^{-2}	0.01×10^{-2}	0.05×10^{-2}
30	0.17×10^{-2}	0.25×10^{-2}	0.02×10^{-2}
31	0.46	0.22	0.06
32	0.12	0.35	0.02
33	0.06	0.04	0.41
34	0.04×10^{-2}	0.12×10^{-2}	0.27×10^{-2}
35	0.06×10^{-2}	0.05×10^{-2}	0.14×10^{-2}
36	0.29×10^{-2}	0.20×10^{-2}	0.04×10^{-2}
37	0.34	0.17	0.04
38	0.09	0.32	0.02
39	0.05	0.03	0.35
40	0.04×10^{-2}	0.11×10^{-2}	0.26×10^{-2}
41	0.04×10^{-2}	0.04×10^{-2}	0.11×10^{-2}
42	0.28×10^{-2}	0.17×10^{-2}	0.04×10^{-2}
43	0.27	0.14	0.04
44	0.10	0.29	0.02
45	0.04	0.03	0.29
46	0.03×10^{-2}	0.10×10^{-2}	0.26×10^{-2}
47	0.03×10^{-2}	0.03×10^{-2}	0.09×10^{-2}
48	0.27×10^{-2}	0.15×10^{-2}	0.03×10^{-2}

TABLE 3.7B-48
ELECTRICAL AND AUXILIARY BUILDING SRSS ACCELERATIONS FOR SSE
FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 3 of 3)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
49	0.20	0.11	0.03
50	0.10	0.27	0.02
51	0.03	0.04	0.23
52	0.03×10^{-2}	0.08×10^{-2}	0.25×10^{-2}
53	0.03×10^{-2}	0.03×10^{-2}	0.07×10^{-2}
54	0.25×10^{-2}	0.12×10^{-2}	0.03×10^{-2}
55	0.14	0.08	0.03
56	0.10	0.24	0.01
57	0.02	0.03	0.16
58	0.02×10^{-2}	0.05×10^{-2}	0.24×10^{-2}
59	0.03×10^{-2}	0.02×10^{-2}	0.05×10^{-2}
60	0.23×10^{-2}	0.11×10^{-2}	0.02×10^{-2}
61	0.12	0.05	0.02
62	0.07	0.23	0.01
63	0.02	0.01	0.12
64	0.02×10^{-2}	0.03×10^{-2}	0.23×10^{-2}
65	0.02×10^{-2}	0.02×10^{-2}	0.04×10^{-2}
66	0.23×10^{-2}	0.11×10^{-2}	0.02×10^{-2}

a) Degrees of freedom corresponding to translations in the x, y, and z directions, and rotations about x, y, and z axes are identified in Table 3.7B-21.

b) Tabulated accelerations are in units of (g) for translational degrees of freedom, and (g/ft) for rotational degrees of freedom.

TABLE 3.7B-49
FUEL BUILDING SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 1 of 2)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
1	0.50	0.18	0.07
2	0.05	0.25	0.04
3	0.05	0.13	0.48
4	0.07×10^{-2}	0.06×10^{-2}	0.25×10^{-2}
5	0.16×10^{-2}	0.07×10^{-2}	0.27×10^{-2}
6	0.26×10^{-2}	0.09×10^{-2}	0.05×10^{-2}
7	0.40	0.17	0.16
8	0.05	0.23	0.12
9	0.03	0.11	0.37
10	0.05×10^{-2}	0.09×10^{-2}	0.24×10^{-2}
11	0.14×10^{-2}	0.06×10^{-2}	0.20×10^{-2}
12	0.25×10^{-2}	0.08×10^{-2}	0.04×10^{-2}
13	0.19	0.08	0.07
14	0.04	0.21	0.08
15	0.02	0.08	0.22
16	0.03×10^{-2}	0.05×10^{-2}	0.20×10^{-2}
17	0.08×10^{-2}	0.04×10^{-2}	0.10×10^{-2}
18	0.19×10^{-2}	0.06×10^{-2}	0.04×10^{-2}
19	0.16	0.06	0.06
20	0.04	0.20	0.08
21	0.02	0.07	0.19
22	0.02×10^{-2}	0.04×10^{-2}	0.17×10^{-2}
23	0.07×10^{-2}	0.04×10^{-2}	0.10×10^{-2}
24	0.16×10^{-2}	0.06×10^{-2}	0.03×10^{-2}
25	0.15	0.06	0.05
26	0.04	0.19	0.07
27	0.02	0.06	0.16
28	0.02×10^{-2}	0.04×10^{-2}	0.16×10^{-2}
29	0.07×10^{-2}	0.04×10^{-2}	0.10×10^{-2}
30	0.14×10^{-2}	0.06×10^{-2}	0.03×10^{-2}
31	0.46	0.17	0.11
32	0.04	0.25	0.13
33	0.04	0.11	0.37
34	0.06×10^{-2}	0.07×10^{-2}	0.25×10^{-2}

TABLE 3.7B-49
FUEL BUILDING SRSS ACCELERATIONS FOR SSE FROM RESPONSE SPECTRUM ANALYSIS

(Sheet 2 of 2)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
35	0.12×10^{-2}	0.06×10^{-2}	0.21×10^{-2}
36	0.25×10^{-2}	0.09×10^{-2}	0.05×10^{-2}
37	0.23	0.08	0.04
38	0.04	0.22	0.10
39	0.02	0.08	0.23
40	0.04×10^{-2}	0.05×10^{-2}	0.22×10^{-2}
41	0.08×10^{-2}	0.04×10^{-2}	0.10×10^{-2}
42	0.20×10^{-2}	0.06×10^{-2}	0.03×10^{-2}
43	0.20	0.07	0.03
44	0.04	0.21	0.09
45	0.02	0.07	0.19
46	0.04×10^{-2}	0.05×10^{-2}	0.19×10^{-2}
47	0.07×10^{-2}	0.04×10^{-2}	0.10×10^{-2}
48	0.17×10^{-2}	0.06×10^{-2}	0.03×10^{-2}
49	0.17	0.06	0.03
50	0.04	0.20	0.08
51	0.02	0.06	0.17
52	0.02×10^{-2}	0.04×10^{-2}	0.17×10^{-2}
53	0.07×10^{-2}	0.04×10^{-2}	0.10×10^{-2}
54	0.15×10^{-2}	0.06×10^{-2}	0.03×10^{-2}
55	0.14	0.05	0.01
56	0.04	0.19	0.04
57	0.02	0.05	0.14
58	0.02×10^{-2}	0.04×10^{-2}	0.15×10^{-2}
59	0.06×10^{-2}	0.04×10^{-2}	0.09×10^{-2}
60	0.12×10^{-2}	0.06×10^{-2}	0.03×10^{-2}

a) Degrees of freedom corresponding to translations in the x, y, and z directions, and rotations about x, y, and z axes are identified in Table 3.7B-22.

b) Tabulated accelerations are in units of (g) for translational degrees of freedom, and (g/ft) for rotational degrees of freedom.

TABLE 3.7B-50
SERVICE WATER INTAKE STRUCTURE SRSS ACCELERATIONS FOR SSE
FROM RESPONSE SPECTRUM ANALYSIS UPPER BOUND SOIL CASE

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
1	2.19	0.41	0.75
2	0.52	4.16	0.32
3	0.75	0.25	2.01
4	0.0136	0.0050	0.0317
5	0.0105	0.0019	0.0049
6	0.0285	0.0127	0.0187
7	4.74	0.33	0.85
8	0.32	5.18	0.36
9	0.68	0.41	4.26
10	0.0127	0.0095	0.0447
11	0.0577	0.0120	0.0237
12	0.0588	0.0282	0.0340
13	7.98	1.14	2.28
14	2.09	6.58	1.25
15	1.08	0.60	7.29
16	0.0302	0.0128	0.0969
17	0.0370	0.0033	0.0316
18	0.1745	0.0455	0.0748
19	12.00	1.03	1.20
20	3.05	7.40	2.04
21	1.03	0.71	10.25
22	0.0440	0.0165	0.1271
23	0.0776	0.0126	0.0472
24	0.2329	0.0571	0.0973

a) Degrees of freedom are identified in Table 3.7B-23.

b) Tabulated accelerations are in units of ft/sec./sec. for transitional degrees of freedom and radians/sec./sec. for rotational degrees of freedom.

TABLE 3.7B-51
ELECTRICAL AND AUXILIARY BUILDINGS PEAK ACCELERATIONS FOR SSE
FROM TIME-HISTORY ANALYSIS

(Sheet 1 of 3)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
1	0.34	0.15	0.02
2	0.12	0.26	0.01
3	0.02	0.02	0.32
4	0.01×10^{-2}	0.03×10^{-2}	0.23×10^{-2}
5	0.02×10^{-2}	0.02×10^{-2}	0.06×10^{-2}
6	0.19×10^{-2}	0.21×10^{-2}	0.01×10^{-2}
7	0.29	0.13	0.02
8	0.12	0.26	0.01
9	0.02	0.02	0.28
10	0.01×10^{-2}	0.03×10^{-2}	0.23×10^{-2}
11	0.01×10^{-2}	0.02×10^{-2}	0.05×10^{-2}
12	0.19×10^{-2}	0.21×10^{-2}	0.01×10^{-2}
13	0.24	0.10	0.01
14	0.12	0.26	0.01
15	0.02	0.02	0.24
16	0.01×10^{-2}	0.03×10^{-2}	0.23×10^{-2}
17	0.01×10^{-2}	0.01×10^{-2}	0.05×10^{-2}
18	0.18×10^{-2}	0.21×10^{-2}	0.01×10^{-2}
19	0.20	0.08	0.01
20	0.11	0.25	0.01
21	0.01	0.01	0.22
22	0.01×10^{-2}	0.02×10^{-2}	0.22×10^{-2}
23	0.01×10^{-2}	0.01×10^{-2}	0.04×10^{-2}
24	0.16×10^{-2}	0.23×10^{-2}	0.01×10^{-2}

TABLE 3.7B-51
ELECTRICAL AND AUXILIARY BUILDINGS PEAK ACCELERATIONS FOR SSE
FROM TIME-HISTORY ANALYSIS

(Sheet 2 of 3)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
25	0.15	0.04	0.04
26	0.10	0.23	0.01
27	0.00	0.00	0.17
28	0.01×10^{-2}	0.01×10^{-2}	0.21×10^{-2}
29	0.00×10^{-2}	0.01×10^{-2}	0.03×10^{-2}
30	0.15×10^{-2}	0.24×10^{-2}	0.01×10^{-2}
31	0.52	0.22	0.03
32	0.13	0.34	0.01
33	0.04	0.02	0.43
34	0.02×10^{-2}	0.06×10^{-2}	0.27×10^{-2}
35	0.02×10^{-2}	0.02×10^{-2}	0.12×10^{-2}
36	0.29×10^{-2}	0.15×10^{-2}	0.02×10^{-2}
37	0.37	0.17	0.02
38	0.10	0.34	0.02
39	0.03	0.02	0.37
40	0.02×10^{-2}	0.06×10^{-2}	0.26×10^{-2}
41	0.02×10^{-2}	0.02×10^{-2}	0.10×10^{-2}
42	0.28×10^{-2}	0.15×10^{-2}	0.02×10^{-2}
43	0.29	0.14	0.02
44	0.11	0.32	0.01
45	0.02	0.02	0.31
46	0.02×10^{-2}	0.05×10^{-2}	0.26×10^{-2}
47	0.03×10^{-2}	0.02×10^{-2}	0.08×10^{-2}
48	0.27×10^{-2}	0.14×10^{-2}	0.01×10^{-2}

TABLE 3.7B-51
ELECTRICAL AND AUXILIARY BUILDINGS PEAK ACCELERATIONS FOR SSE
FROM TIME-HISTORY ANALYSIS

(Sheet 3 of 3)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
49	0.24	0.11	0.01
50	0.11	0.30	0.01
51	0.02	0.02	0.28
52	0.02×10^{-2}	0.04×10^{-2}	0.25×10^{-2}
53	0.01×10^{-2}	0.02×10^{-2}	0.06×10^{-2}
54	0.26×10^{-2}	0.13×10^{-2}	0.01×10^{-2}
55	0.19	0.08	0.01
56	0.12	0.26	0.01
57	0.01	0.01	0.20
58	0.01×10^{-2}	0.03×10^{-2}	0.24×10^{-2}
59	0.01×10^{-2}	0.01×10^{-2}	0.04×10^{-2}
60	0.17×10^{-2}	0.11×10^{-2}	0.01×10^{-2}
61	0.16	0.05	0.01
62	0.08	0.25	0.01
63	0.00	0.01	0.18
64	0.01×10^{-2}	0.02×10^{-2}	0.23×10^{-2}
65	0.01×10^{-2}	0.01×10^{-2}	0.03×10^{-2}
66	0.23×10^{-2}	0.10×10^{-2}	0.01×10^{-2}

a) Degrees of freedom corresponding to translations in the x, y, and z directions, and rotations about x, y, and z axes are identified in Table 3.7B-21.

b) Tabulated accelerations are in units of (g) for translational degrees of freedom, and (g/ft) for rotational degrees of freedom.

TABLE 3.7B-52
FUEL BUILDING PEAK ACCELERATIONS FOR SSE FROM TIME-HISTORY ANALYSIS

(Sheet 1 of 2)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
1	0.55	0.16	0.02
2	0.04	0.27	0.03
3	0.01	0.11	0.51
4	0.01×10^{-2}	0.05×10^{-2}	0.24×10^{-2}
5	0.08×10^{-2}	0.02×10^{-2}	0.12×10^{-2}
6	0.25×10^{-2}	0.08×10^{-2}	0.01×10^{-2}
7	0.44	0.14	0.07
8	0.03	0.26	0.08
9	0.01	0.09	0.42
10	0.01×10^{-2}	0.06×10^{-2}	0.23×10^{-2}
11	0.08×10^{-2}	0.02×10^{-2}	0.10×10^{-2}
12	0.24×10^{-2}	0.07×10^{-2}	0.01×10^{-2}
13	0.23	0.06	0.03
14	0.03	0.24	0.07
15	0.01	0.05	0.27
16	0.01×10^{-2}	0.04×10^{-2}	0.20×10^{-2}
17	0.05×10^{-2}	0.01×10^{-2}	0.06×10^{-2}
18	0.18×10^{-2}	0.07×10^{-2}	0.02×10^{-2}
19	0.21	0.05	0.03
20	0.03	0.23	0.07
21	0.01	0.04	0.22
22	0.01×10^{-2}	0.04×10^{-2}	0.17×10^{-2}
23	0.05×10^{-2}	0.01×10^{-2}	0.06×10^{-2}
24	0.16×10^{-2}	0.06×10^{-2}	0.01×10^{-2}
25	0.19	0.04	0.03
26	0.03	0.23	0.06
27	0.01	0.04	0.19
28	0.01×10^{-2}	0.04×10^{-2}	0.16×10^{-2}
29	0.05×10^{-2}	0.01×10^{-2}	0.07×10^{-2}
30	0.14×10^{-2}	0.06×10^{-2}	0.01×10^{-2}
31	0.49	0.14	0.06
32	0.04	0.26	0.12
33	0.01	0.09	0.42
34	0.01×10^{-2}	0.06×10^{-2}	0.25×10^{-2}

TABLE 3.7B-52
FUEL BUILDING PEAK ACCELERATIONS FOR SSE FROM TIME-HISTORY ANALYSIS

(Sheet 2 of 2)

Degree of Freedom[a]	X-Earthquake[b]	Y-Earthquake[b]	Z-Earthquake[b]
35	0.06×10^{-2}	0.01×10^{-2}	0.10×10^{-2}
36	0.25×10^{-2}	0.07×10^{-2}	0.01×10^{-2}
37	0.24	0.07	0.04
38	0.03	0.25	0.09
39	0.01	0.05	0.27
40	0.01×10^{-2}	0.05×10^{-2}	0.21×10^{-2}
41	0.06×10^{-2}	0.01×10^{-2}	0.06×10^{-2}
42	0.19×10^{-2}	0.06×10^{-2}	0.01×10^{-2}
43	0.23	0.05	0.03
44	0.04	0.25	0.07
45	0.01	0.05	0.22
46	0.01×10^{-2}	0.04×10^{-2}	0.19×10^{-2}
47	0.05×10^{-2}	0.01×10^{-2}	0.06×10^{-2}
48	0.17×10^{-2}	0.06×10^{-2}	0.01×10^{-2}
49	0.21	0.04	0.03
50	0.04	0.24	0.07
51	0.01	0.04	0.19
52	0.01×10^{-2}	0.04×10^{-2}	0.17×10^{-2}
53	0.04×10^{-2}	0.01×10^{-2}	0.07×10^{-2}
54	0.14×10^{-2}	0.06×10^{-2}	0.01×10^{-2}
55	0.19	0.04	0.01
56	0.03	0.23	0.03
57	0.01	0.03	0.18
58	0.01×10^{-2}	0.04×10^{-2}	0.15×10^{-2}
59	0.04×10^{-2}	0.01×10^{-2}	0.07×10^{-2}
60	0.12×10^{-2}	0.06×10^{-2}	0.01×10^{-2}

a) Degrees of freedom corresponding to translations in the x, y, and z directions, and rotations about x, y, and z axes are identified in Table 3.7B-22.

b) Tabulated accelerations are in units of (g) for translational degrees of freedom, and (g/ft) for rotational degrees of freedom.

APPENDIX 3.7B(A) - COMPUTER PROGRAMS USED IN DYNAMIC AND STATIC ANALYSES

3.7B(A).1 INTRODUCTION

Computer programs used in the dynamic and static analyses of seismic Category I structures, systems, and equipment are described herein and listed in Table 3.7B(A)-1. Program version, software or operating system, computer hardware, and status of recognition in the public domain are also specified in Table 3.7B(A)-1. The recognized computer programs in the public domain presented in Section 3.7B(A) have sufficient history of use to justify their applicability and validity without further verification. Other computer programs presented in this section were developed by Gibbs & Hill. These programs are occasionally linked and modified by G&H to suit a particular need for the solution of a problem. They are made operational on the CDC 6600 system. The applicability and validity of these programs are demonstrated by comparing the results obtained from each computer program with the results derived from a similar program available in the public domain. Verification of these programs is completed and maintained at G&H. See also Section 3.9B.1.2

Additional computer programs used in the analysis of ASME Code Class 2 and Class 3 piping systems, including supports for ASME Code Class 1, 2, and 3 piping are provided in Appendix 3B.

3.7B(A).2 COMPUTER PROGRAM (QUAKE)

The QUAKE program performs the dynamic analysis of a lumped mass system. The input includes mass data, structural stiffness, foundation spring constants, structure and foundation damping values, and other data related to the dynamic analysis model. The output provides modal shears and moments of statically determinate structures and combines them by the square root of the sum of the squares of modal values, by absolute sum, and by algebraic sum. QUAKE is comprised of selected subroutines presented in Reference [1]; the program is verified by using the STARDYNE computer program described in Subsection 3.7B(A).5 and by hand calculations. The QUAKE program has the following capabilities:

1. Extracting the eigenvalues and the corresponding eigenvectors from the following equation as described in Reference [35] of Section 3.7B:

$$\left[\frac{1}{\omega^2} [I] - [\delta][m] \right] \{\varnothing\} = 0 \qquad \text{(3.7B(A)-1)}$$

where

[I] = the unit matrix

ω = the natural circular frequency

[m] = the mass matrix

$[\delta]$ = the flexibility matrix

$\{\varnothing\}$ = the column matrix of eigenvectors

Equation 3.7B(A)-1 can be written in terms of the stiffness matrix [k] as follows:

$$[[k] - \omega^2 [m]] \{\varnothing\} = 0 \qquad (3.7B(A)\text{-}2)$$

To extract eigenvalues and eigenvectors, Equation 3.7B(A)-1 is converted by using the following substitution:

$$[m] = [U]^T [U] \qquad (3.7B(A)\text{-}3)$$

In general, matrix [U] consists of only diagonal and upper diagonal elements.

Substitution of [m] from Equation 3.7B(A)-3 and $\{\varnothing\}$ from Equation $\{\varnothing\} = [U]^{-1} \{\Phi\}$ into Equation 3.7B(A)-2 yields the following:

$$[[k] [U]^{-1} - \omega^2 [U]^T [U] [U]^{-1}] \{\Phi\} = 0 \qquad (3.7B(A)\text{-}4)$$

Premultiplication of Equation 3.7B(A)-4 by $\frac{[u][k]^{-1}}{w^2}$ yields the equation:

$$[{}_{\omega}1_2[I] - [U][\delta][U]^T] \{\Phi\} = 0 \qquad (3.7B(A)\text{-}5)$$

The matrix product $[U] [\delta] [U]^T$ is a symmetric matrix. The eigenvalues and eigenvectors are extracted from Equation 3.7B(A)-5 using a subroutine called EIGEN [1]. This subroutine uses an algorithm known as the Jacobi Diagonalization method.

The mode shapes $[\varnothing]$ as corresponding eigenvectors of Equation 3.7B(A)1 are obtained from the following equation:

$$[\varnothing] = [U]^{-1} [\Phi] \qquad (3.7B(A)\text{-}6)$$

2. Computing participation factors Γ by using a matrix manipulation in the following manner:

$$\{\Gamma\}_j = [M]^{-1} [\varnothing]^T [m] \{D\}_j \qquad (3.7B(A)\text{-}7)$$

where

$$[M] = [\varnothing]^T [m] [\varnothing] \qquad (3.7B(A)\text{-}8)$$

$[\varnothing]$ = the matrix of mode shapes

$[\varnothing]^T$ = the transpose of the mode shape matrix

$[M]$ = the diagonal generalized mass matrix

$\{\Gamma\}_j$ = the column matrix of participation factors for seismic motion in the jth direction

$[D]_j$ = the column matrix governed by the seismic motion in the jth direction

The normalization and orthogonality conditions are represented by matrix Equation 3.7B(A)-8.

3. Computing equivalent modal damping (or composite modal damping) according to the energy stored in each component and in each vibration mode.

   Each component, such as concrete structures, steel structures and systems, and foundation materials, can have different damping properties. The effective damping in any vibration mode of the total system depends upon the degree of participation of these materials in the modal response.

The stiffness matrix [k] of the entire system is obtained by the summation of the stiffness matrices [k] for each component r that has inherently different damping properties as expressed by the following equation:

$$[k] = \sum_{r=1}^{N} [k]r \qquad (3.7B(A)\text{-}9)$$

Where

N = total number of components with different damping properties

The corresponding flexibility matrix is obtained by inverting the stiffness matrix as follows:

$$[\delta] = [k]^{-1} \qquad (3.7B(A)\text{-}10)$$

The fictitious force matrix [f] is obtained as a product of the stiffness matrix and the mode shape matrix $[\Theta]^r$ as follows:

$$[f] = [k] [\varnothing] \qquad (3.7B(A)\text{-}11)$$

The mode shape matrix $[\Theta]_r$ for each component r is given by the following equation:

$$[\varnothing]_r = [k]_r^{-1} [f] \qquad (3.7B(A)\text{-}12)$$

The associated modal energy component matrix, which comprises diagonal terms only, is given by the following equation:

$$[E]_r = \frac{1}{2}[\varnothing]_r^T [k]_r [\varnothing]_r \qquad (3.7B(A)\text{-}13)$$

The total energy for the entire system is given by the following equation:

$$[E] = \tfrac{1}{2}[M] [\omega^2] \qquad (3.7B(A)\text{-}14)$$

The fraction of modal energy components is given by the following equation:

$$[MEC] = [E]^{-1} [E]_r \qquad (3.7B(A)\text{-}15)$$

so that $$\sum_{r=1}^{N} [E]_r = [E] \qquad (3.7B(A)\text{-}16)$$

and $$\sum_{r=1}^{N} [MEC]_r = [1] \qquad (3.7B(A)\text{-}17)$$

The column matrix of weighted modal damping ratios is found by using the following equation:

$$[D] = [E]^{-1} \sum_{r=1}^{N} [E]_r \quad \{D\}_r \qquad (3.7B(A)\text{-}18)$$

For a system comprised of the foundation and structure components, the modal energy matrix for the foundation is first generated by means of Equation 3.7B(A)-13. The modal energy matrix for the structure is obtained by using Equations 3.7B(A)- 16. The column matrix of weighted modal damping ratios for the entire system is obtained as indicated in Equation 3.7B(A)-18.

4. Computing absolute modal accelerations, relative displacements, and inertia loads using the spectrum approach; also, the modal responses are combined as described in Section 3.7B.2.7.

The matrix of maximum modal absolute accelerations [W] is obtained from the following equation in matrix form:

$$[\ddot{W}] = [\varnothing][Sa][r] \qquad (3.7B(A)\text{-}19)$$

where

[Sa] = the diagonal matrix of maximum modal spectral accelerations

[Γ] = the diagonal matrix of modal participation factors

The matrix of maximum modal relative displacements [d] is computed by the following equation:

$$[d] = [\ddot{W}][{}_{\omega}1_2] \qquad (3.7B(A)\text{-}20)$$

where

$[{}_{\omega}1_2]$ = the diagonal matrix of eigenvalues, if the flexibility matrix is used in the equation for the extraction of eigenvalues and eigenvectors

The matrix of maximum modal inertia forces [F] is the product of mass matrix [m] and maximum modal acceleration matrix $[\ddot{W}]$.

$$[F] = [m][\ddot{W}] \qquad (3.7B(A)\text{-}21)$$

The maximum modal inertia forces [F] are combined by the square root of the sum of the squares, by absolute sum, and by algebraic sum, and in accordance with the provisions of the NRC Regulatory Guide 1.92.

5. Calculating the modal shears and moments of structures and combining them by the square root of the sum of the squares (SRSS) of modal values, by absolute sum, and by algebraic sum, and in accordance with the provisions of the NRC Regulatory Guide 1.92.

The matrix of modal shears and bending moments [Q] is computed from the following equation:

$$[Q]^T = [F]^{T\,[J]} \qquad (3.7B(A)\text{-}22)$$

where

$[F]^T$ = the transpose of the matrix of inertia forces and inertia moments

[J] = the summation matrix

3.7B(A).3 FINITE ELEMENT COMPUTER PROGRAM (NASTRAN)

This is a large computer program for dynamic and static analysis of linear elastic systems by the finite element approach. The program can handle large problems with many degrees-of-freedom. The only limitations on problem size are those imposed by practical considerations of running time and by the ultimate capacity of auxiliary storage devices. The program contains various types of finite elements. A manual of this program is given in Reference [2]. The official documentation consists of manuals presented in References [3], [4], [5], and [6]. These documents are available from COSMIC, Barrow Hall, University of Georgia, Athens, Georgia 30601.

3.7B(A).4 COMPUTER PROGRAM (CSMP)

The CSMP program is an IBM system/360 program for the simulation of continuous systems [7]. It provides an application-oriented input language that accepts problems expressed in the form of either an analog block diagram or a system of ordinary differential equations. Data input and output are facilitated by means of application oriented-control statements.

The manual presented in Reference [7] contains a general description of the program, detailed programming information, and a description of the inputs and outputs.

Several different types of routines are available to perform the integration operation. They include both fixed integration step-size routines and variable step-size routines. Five fixed step-size routines are available: fixed Runge-Kutta, Simpson's, trapezoidal, rectangular, and second order Adams. Two variable step-size routines are available: fifth-order Milne predictor-corrector and fourth-order Runge-Kutta.

3.7B(A).5 FINITE ELEMENT COMPUTER PROGRAM (STARDYNE)

The STARDYNE program consists of a series of compatible digital computer programs designed to analyze linear elastic structural models. It encompasses the full range of static and dynamic analyses.

This program, whose user's manual is presented in Reference [8], is used to evaluate a wide variety of static and dynamic problems. It contains various types of finite elements.

The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, applied loads or prescribed displacements, or both.

Dynamic response analysis is performed for a wide range of loading conditions using the normal mode technique.

Transient, steady-state harmonic, random, and shock spectra excitation type inputs are applied. Different plotting capabilities are available.

3.7B(A).6 FINITE ELEMENT COMPUTER PROGRAM (ICES STRUDL-II)

The ICES STRUDL-II program was developed at the Massachusetts Institute of Technology. It is a part of the Integrated Civil Engineering System (ICES), which is a project of cooperation between government, industry, and university groups interested in the development of a large-scale, computer-based system [9]. The program has been updated and debugged by McDonnell-Douglas Automation Company [10].

This program provides the design engineer with an integrated set of facilities for the selection of structural members; these facilities are structured so that knowledgeable users can easily modify and expand the current capabilities. A user can add his own selection procedures and design codes, as well as change information such as assumed values for optional design data.

The system is organized for the easy addition of new codes and procedures.

This program contains various types of finite elements, such as triangular, rectangular, and quadrilateral plate elements representing or bending behavior, or both three dimensional solid elements, and beam elements.

3.7B(A).7 STRUCTURAL ENGINEERING SYSTEM SOLVER (STRESS) COMPUTER PROGRAM

The STRESS program is designed specifically for the solution of structural engineering problems as presented in Reference [11]. It uses a problem-oriented input language that enables the structural engineer to communicate with the computer even without previous programming experience.

The program can analyze structures with prismatic members in two or three dimensions, with either pinned or rigid joints, and subjected to concentrated or distributed loads, displacements, or temperature effects.

3.7B(A).8 COMPUTER PROGRAM FOR THE ANALYSIS OF SHELLS OF REVOLUTION (KALNINS)

This program calculates the stresses and displacements in thin-walled elastic, orthotropic shells of revolution when subjected to symmetrical and nonsymmetrical loads by means of the method of analysis presented in Reference [12]. The program is applicable to rotationally symmetric shells to which any number of axisymmetric branches are attached. The only restriction on the shell is that shell geometry, boundaries, and material properties are symmetric about one axis; that is, each section of the shell perpendicular to the axis must be circular. The geometry and elastic parameters, however, can vary in an arbitrary manner along the meridian of the shell. The circumferential loads are expressed in terms of Fourier harmonics.

This program uses a numerical method which combines the direct integration and the finite difference techniques for the solution of general shell equations.

3.7B(A).9 COMPUTER PROGRAM (SCONV)

The SCONV program uses convolution integration to solve a general dynamic problem for a linear elastic system with the support time history excitation RS(t) where R is the scaling factor and S(t) is the time-dependent support acceleration. For any linear elastic system, the governing equation for motions in matrix form is as shown in References [30] and [38] of Section 3.7B:

$$[m]\{\ddot{u}\} + [c]\{\dot{u}\} + [k]\{u\} = -[m]\{D\}R\ddot{S}(t) \qquad \text{3.7B(A)-23}$$

can be decoupled into a system of second order differential equations as follows:

$$\{\ddot{q}\} + 2[\beta][\omega]\{\dot{q}\} + \lfloor\omega^2\rfloor\{q\} = -\{\gamma\}R\ddot{S}(t) \qquad \text{(3.7B(A)-24)}$$

where

[q] = the normal coordinate, i.e., {u} = [ϕ] {q}

and [β], = the matrices of coefficients of critical damping,

[ω],{γ}, circular frequencies, participation factors, direction

[D] and [ϕ] vector, and eigenvectors, respectively

The general solution of Equation 3.7B(A)-24 for any system is as follows:

$$q_n(t) = C_n(t) - \frac{\gamma n}{\omega_n^2} A_n(t) \qquad \text{(3.7B(A)-25)}$$

where

$$C_n(t) = e^{-\beta_n\omega_n t}\left(\frac{\dot{q}_0 + \beta_n\omega_n q_0}{\bar{\omega}_n}\sin\bar{\omega}_n t + q_0\cos\bar{\omega}_n t\right)$$

$$A_n(t) = \int_0^t \frac{\omega_n^2}{\bar{\omega}_n} e^{-\beta_n\omega_n(t-\tau)} R\ddot{S}(\tau)\sin[\bar{\omega}_n(t-\tau)]d\tau \qquad \text{(3.7B(A)-26)}$$

$$\text{and } \bar{\omega}_n = \sqrt{\omega_n^2\ 1-\beta_n^2}$$

$q_n(t)$ is the relative displacement of the nth mode at time t, and q_o and q_o are the initial velocity and displacement.

By differentiating Equation 3.7B(A)-25 once and twice, the relative velocity and acceleration are obtained.

$$\dot{q}_n(t) = \beta_n\omega_n C_n(t) + D_n(t) + \frac{\gamma_n}{\omega_n^2}(\beta_n\omega_n A_n(t) - \beta_n(t)) \qquad \text{(3.7B(A)-27)}$$

$$\ddot{q}_n(t) = -(\omega_n^{-2} - \beta_n^2\ \omega_n^2\ )C_n(t) - 2\beta_n\omega_n D_n(t) + \frac{\gamma_n}{\omega_n^2}[(\omega_n^2\ - 2\beta_n^2\ \omega_n^2\ )A_n(t) + 2\beta_n\omega_n B_n(t)] - \gamma_n R\ddot{S}(t) \qquad \text{(3.7B(A)-28)}$$

where

$$B_n(t) = \int_0^t \omega_n^2\ R\ddot{S}(\tau)e^{-\beta_n\omega_n(t-\tau)}\cos[\bar{\omega}_n(t-\tau)]d\tau$$

and

$$D_n(t) = e^{-\beta_n\omega_n t}[(\dot{q}_0 + \beta_n\omega_n q_0)\cos\bar{\omega}_n t - q_0\bar{\omega}_n\ \sin\bar{\omega}_n t] \qquad \text{(3.7B(A)-29)}$$

The solution procedure is based on the step-by-step method. The total time duration is divided into small time intervals or steps. Displacement and velocity from the previous step are used as initial conditions for the next step. At each step, the relative displacement, velocity, and acceleration are calculated. The absolute acceleration is then combined by modal superposition as follows:

$$[\ddot{W}] = [\varphi]\{\ddot{q}\} + \{D\}R\ddot{S}(t) \qquad \text{(3.7B(A)-30)}$$

where

$\{\ddot{W}\}$ = the column matrix of absolute accelerations

The numerical method used for integration of Equations 3.7B(A)-26 and 3.7B(A)-29 is Simpson's rule.

Different output options are available. The user can ask for any or all of the following output

1. Time history of the absolute acceleration of any mass

2. Maximum absolute acceleration of all masses

3. Maximum load effects, such as shears, moments, and similar load effects at any mass point of the system

4. All load effects of the system corresponding to the inertia forces at any time instant

This program was verified by using the IBM program, CSMP, described in Section 3.7B(A).4.

3.7B(A).10 STORING OF DESIGN RESPONSE SPECTRUM ON THE COMPUTER (SPECTRA)

Dynamic analysis of structures by the spectrum approach necessitates reading the spectral acceleration values at discrete pairs of frequency and damping values. Design response spectra provided are usually in log-log scale and are for a few damping values only (10 percent, 7 percent, 5 percent, 2 percent, and 0.5 percent, normally). Calculated modal damping can fall in between these values or beyond; if so, it becomes necessary to interpolate or extrapolate. All this can be done visually, but the values thus read off the graphs are susceptible to errors. To achieve satisfactory accuracy, to avoid the tedium of visual reading, and to expedite the computer solution of the problem at hand, a subroutine is developed to simulate the reading process by certain measured parameters. A single call of this subroutine yields the spectral accelerations corresponding to a set of specified pairs of frequencies and the associated modal damping values.

Essentially, the method involves finding a polynomial expression for the maximum amplification as a function of damping. This is done by using the least square method, described in References [13], [14], [15], and [16], to fit a curve of a frequency control point that goes through the available points determined by amplification factors and damping ratios as shown in Figure 3.7B(A)-1. The maximum amplification occurs over a range of frequencies, beyond which the amplification is found to decay linearly in the log field. The decaying segments may be subsequently followed by other segments of different slopes. The points of discontinuities of the spectral curves, consisting of segments, are assumed to correspond to the same frequencies, irrespective of damping. The slopes of the decaying segments are, however, allowed to vary, i.e., the spectral curves need not be parallel to each other for each damping. The slopes of these segments for various damping values are again described by a least square polynomial fit as a function of damping. With this amount of information it is possible to find out the spectral accelerations corresponding to any frequency and the associated damping.

3.7B(A).11 FLOOR RESPONSE SPECTRA PROGRAM (TIME)

This program, developed by G&H, is used to obtain floor response spectra for equipment from the floor time histories of the building. Response spectra for different equipment dampings can be obtained; the output data are used in plotting response spectra curves.

TIME is used to produce response spectra for equipment mounted on flexible, i.e., low, natural-frequency floors.

The program uses an exact solution for the second order differential equation, assuming the forcing function (i.e., floor time history) to be linear between each time step; this solution is based on the formulations given in the dynamic analysis section of Reference [17].

3.7B(A).12 COMPUTER PROGRAM (ADLPIPE)

This computer program provides analyses of Class 1, 2 and 3 piping systems, which are classified as, Seismic Category I, for piping flexibility due to thermal expansion, as well as static

and dynamic analyses of elastic multibranch piping systems with flexible or rigid anchors and intermediate restraints. The piping system can contain a number of sections between network nodal points comprised of straight or curved members. A network nodal point is a branching point of pipes, an anchor, or any point at which motion is prescribed, or a point of lumped mass. Input preparation is described in Reference [19].

This program computes at each point within the piping system, the forces, moments, and displacements resulting from thermal expansion, dead loads, static loads, and displacements applied in the directions associated with all six degrees-of-freedom of the global orthogonal system and earthquake disturbances.

The normal mode technique in conjunction with the three-dimensional response spectra is used for obtaining seismic response. The resultant internal forces and moments are computed from the SRSS of their modal values. Algorithms used in this program for the extraction of eigenvalues and associated eigenvectors are the Jacobi rotation scheme [20] and the Givens-Householder scheme with modification as suggested in Reference [18]. The program is based on the systematic use of transfer matrices.

The program computes stresses within piping systems in accordance with ANSI B31.1-1973 and the ASME B&PV Code, Section III, Nuclear Power Plant Components. The combination of load conditions to compute usage factors in accordance with the latter code is included in this version of the program.

Typical benchmark calculations for the verification of the ADLPIPE program are presented in Reference [21].

3.7B(A).12.1 Description of the Analytical Techniques

The following documents describe the mathematical techniques that are used:

1. Generalized Piping System Response to Ground Shock Spectra by I.W. Dingwell, Arthur D. Little, Inc.

2. A Method of Computing Stress Range and Fatigue Damage in a Nuclear Piping System by W.B. Wright and E.C. Rodabaugh, Nuclear Engineering and Design, vol. 22, no. 2, October 1972, pp. 318- 325.

3. Method of Calculating Static and Dynamic Moments for Stress Evaluation at Tees and Branches, Arthur D. Little, Inc., May 1973.

4. Method of Calculating Thermal Stress Range for dT1, dT2, dTa, and dTb Terms, Arthur D. Little, Inc., May 1973.

5. Mathematical Analysis and Logical Procedure by I.W. Dingwell and R.T. Bradshaw, Arthur D. Little, Inc., 1970.

3.7B(A).13 FINITE ELEMENT COMPUTER PROGRAM (ANSYS)

This is a general purpose three-dimensional computer program for the solution of a large class of problems in engineering. This computer program, identified by the acronym ANSYS for

Engineering Analysis System [22], provides a flexible framework for implementation of the finite element analysis technology. The program is capable of taking practically any type of loading conditions and boundary conditions and has the capabilities for static and dynamic, elastic and plastic, fluid flow, and transient heat transfer analyses.

The matrix displacement method of analysis, based on finite element idealization, is used throughout. The library of finite elements available numbers more than twenty for static and dynamic analyses.

These elements are of various types, e.g., frames, plane stress and axisymmetric triangles, three-dimensional solids, springs, masses, dampers, plates, axisymmetric shells, general shells, and friction interface elements.

The program uses a direct solution, developed by the matrix displacement method, for the system of simultaneous linear equations.

Plotting subroutines are available.

3.7B(A).14 COMPUTER PROGRAM (SUPERPIPE)

See Section 3.9B.1.2.3.

3.7B(A).15 STRUDL-SW (STRUCTURAL DESIGN LANGUAGE)

The STURDL-SW computer program [24] is a general finite element program which uses the stiffness analysis method to analyze a wide range of structural problems. It handles two- and three-dimensional trusses and frames, having linear elastic members with either static or dynamic loadings.

STRUDL-SW has been documented by bench marking procedures against the GTSTRUDL computer code. GTSTURDL is a recognized program in the public domain.

3.7B(A).16 SHELL-1 (THIN SHELL OF REVOLUTION UNDER ARBITRARY LOADING)

The SHELL-1 computer program [25] is a general shell program. This program is based upon the general numerical procedure proposed by B. Budiansky and P. P. Radowski, and Greenbaum to analyze a shell of revolution subjected to arbitrary loadings. The program is applicable only for axisymmetric structures such as containment shells; however, the program has the capacity of taking non-symmetric loads and boundary conditions through the technique of Fourier harmonic decomposition.

This program uses a finite difference stress analysis solution. It can be used to determine the forces, moments, shears, displacements, rotations, and stresses in a shell of revolution subject to arbitrary loads expanded in a Fourier series of up to 150 terms. Single layer shells with up to 30 simply connected branches may be analyzed. Poisson’s ratios may change at discontinuity points, and Young’s modulus and the thermal coefficient of expansion may be different at each point. The allowed types of loading include elastic restraints, pressures in three orthogonal directions, temperature changes which may have a gradient through the shell thickness, and simplified input for weight of the shell or earthquake forces.

This computer program has been qualified by comparing it's results against those of hand calculations.

3.7B(A).17 TIMHIS6 (TIME HISTORY ANALYSIS)

The TIMHIS6 Program [26] computes time history response and amplified response spectra (ARS) at any mass location of a lumped mass system due to a synthetic earthquake input. The responses are computed by integration of the modal equations of the system by exact methods. The program's main application is the generation of ARS.

This program has been documented by bench marking procedures against the STARDYNE computer program. STARDYNE is a recognized program in the public domain.

3.7B(A).18 SBMMI (SINGLE BARRIER MASS MISSILE IMPACT)

THE SBMMI computer program [27] computes the elastic-plastic structural response of a barrier due to the following loads: (a) static loads; (b) suddenly applied constant dynamic loads which remain permanently on the structure; (c) suddenly applied constant dynamic loads representing missile impact with a finite force and specific momentum; and (d) suddenly applied dynamic loads of zero time duration and specific momentum representing missile impact. The barrier is modelled as a single barrier mass and a non-linear spring, with the above loads applied. The equation of motion is integrated in time assuming constant acceleration in each time step.

This computer program has been qualified by comparing it's results against those of hand calculations.

3.7B(A).19 CLASSI (CONTINUUM LINEAR ANALYSIS OF SOIL STRUCTURE INTERACTION)

The program CLASSI is a computer code generated to calculate the three-dimensional dynamic response of structures including soil-structure interaction effects. CLASSI is based on a specialized form of substructuring which uses the finite element method to perform the detailed analysis of the superstructure and uses the continuum mechanics method to calculate the interaction of the foundation with the soil medium and with incident seismic waves. These substructuring procedures are made possible by balancing the forces and moments at the foundation, which serves as the common ground for both the superstructure and the soil medium.

The capabilities of CLASSI in modeling systems which combine the soil, the foundation, and the structure are:

1. The soil profile may consist of multiple layers lying over a homogenous halfspace. The layers may have different shear moduli, Poisson's ratio, density, depth, and material damping characteristics.

2. The foundation can be of arbitrary geometry but must be surface mounted. The foundation can be discretized into regular (square or rectangular) segments. The stress distribution between the foundation and the soil depends on the degree of discretization of the foundation since constant stress is assumed for each segment.

3. The structural model can be developed using any standard finite element program. The fixed base structural modal properties obtained from the finite element program are subsequently used as input to CLASSI. In this way,the structure can have any degree of complexity.

The CLASSI substructure approach divides the SSI problem into the following three steps.

1. Determination of the foundation scattering matrices.

2. Determination of the frequency-dependent impedance functions.

3. Analysis of the coupled soil-structure system, using results from steps 1 and 2 and the dynamic properties of the structure.

In the first step, CLASSI evaluates the harmonic response of the rigid, massless foundation bonded to the soil and subjected to a given incident seismic wave in the absence of the superstructure. The free field motion is then used in conjunction with the complex, frequency-dependent scattering matrix in order to determine the foundation input motion.

In the second step, the foundation impedances corresponding to a rigid foundation on a uniform or layered viscoelastic media are developed.

In the third step, analysis of the coupled soil-structure system is carried out by CLASSI in the frequency domain. Time history of responses are obtained by inverse Fourier transform techniques.

3.7B(A).20 RESPEC (RESPONSE SPECTRA GENERATOR)

The computer program RESPEC computes the response spectra of acceleration time histories digitized at equal intervals. Spectral accelerations are computed for frequencies and damping ratios selected by the user. Each spectral value is obtained by computing the maximum response of a damped single-degree-of-freedom oscillator subjected to the specified acceleration time history. The response is computed by explicit integration of the differential equations of motion, assuming “at rest” initial conditions. The program uses the numerical method set forth by Nigam and Jennings in Earthquake Engineering Research Laboratory report “Diginal Calculation of Response Spectra from Strong-Motion Earthquake Records.” The spectral velocity and spectral displacement may also be calculated.

3.7B(A).21 LS-DYNA

The computer program LS-DYNA (developed by Lawrency Livermore Laboratories) is used to perform an elasto-plastic finite element analysis of a spent fuel handling accident. The program simulates the transient collision event with full consideration of plastic, large deformation, wave propagation, and elastic/plastic buckling modes.

3.7B(A).22 MR216 (DYNARACK)

The computer program DYNARACK is used to perform non-linear, direct integration finite element analyses of the Whole Pool Multi-Rack (WPMR) configuration for the spent fuel storage racks. The program uses a step-by-step solution in time employing a central difference algorithm

to evolve to a converged solution. Using the structural model for every set of 22-DOF rack models that comprise a Whole Pool Multi-Rack simulation, equations of motion corresponding to each degree-of-freedom are obtained using Lagrange's Formulation of the dynamic equations of motion. The system kinetic evergy includes contributions from solid structures and from trapped and surrounding fluid. The final system of equations obtained have the matrix form:

$$[M] \{d^2 q / dt^2 \} = \{Q\} + \{G\}$$

where:

- [M] = total mass matrix (including structural and fluid mass contributions). The size of this matrix will be (22 x NOR) x (22 x NOR). NOR = number of racks in the spent fuel pool.
- {q} = the nodal displacement vector relative to the pool slab displacement.
- {G} = a vector dependent on the given ground acceleration.
- {Q} = a vector dependent on the spring forces (linear and nonlinear) and the coupling between degrees-of-freedom.

The equation can be rewritten as:

$$\{d^2 q / dt^2\} = [M]^{-1} \{Q\} + [M]^{-1} \{G\}$$

This equation set is mass uncoupled, displacement coupled at each instant in time. The numerical solution uses a central difference scheme build into the proprietary computer program. Results are archived at appropriate time intervals for permanent record and for performing subsequent post-processing for structural integrity evaluation.

This computer program has the capability to execute concurrent sliding, rocking, bending, twisting, and other motion forms compatible with the freestanding rack design. The program has the capability to effect momentum transfers which occur due to rattling of fuel assemblies inside storage cells and the capability to simulate lift-off and subsequent impact of support pedestals.

3.7B(A).23 OTHER COMPUTER PROGRAMS

There are other small programs not described above. These programs are developed for the purpose of bulk data manipulation to replace regular manual calculations. From time to time, additional small programs of this kind are developed to suit particular needs. The verification of these programs is generally accomplished by sample calculations.

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19. I.W. Dingwell of Arthur D. Little, Inc., April 1977, ADLPIPE, Static and Dynamic Pipe Design and Pipe Stress Analysis, Input Preparation Manual, Cambridge, Massachusetts.

20. Greenstadt, J., 1959, The Determination of the Characteristic Roots of a Matrix by the Jacobi Method, Chapter 7, in Mathematical Methods for Digital Computers, John Wiley, New York.

21. Documentation of ADLPIPE for Static and Dynamic Loads and Stress Evaluation, September 1973, Arthur D. Little, Inc., Cambridge, Massachusetts.

22. ANSYS, Engineering Analysis System, December 1, 1971, User's Manual, John A. Swanson, Ph.D., Swanson Analysis System, Inc., 870 Pine View Drive, Elizabeth, Pennsylvania 15037.

23. SUPERPIPE, Static and Dynamic Loads and Stress Evaluation of Piping Systems, Program Version Number 15c, Impell Corp., San Francisco, Calif.

24. STRUDL-SW, "Structural Design Language," Stone and Webster Engineering Corporation, April 1980.

25. SHELL-1, "Thin Shell of Revolution Under Arbitrary Loading," Stone and Webster Engineering Corporation, September 1971.

26. TIMHIS6, "Time History Analysis," Stone and Webster Engineering Corporation, March 1978.

27. SBMMI, "Single Barriers Mass Missile Impacts," Stone and Webster Engineering Corporation, January 1981.

28. CLASSI (Continuum Linear Analysis of Soil Structure Interaction), "A Computer Program for Soil-Structure Interaction Using a Substructuring Technique," Version v.0.0, Impell Corporation.

29. RESPEC, "A Computer Program for the Generation of Response Spectra," Version v.6/10/75, Impell Corporation.

30. LS-DYNA, Version 950, Liversomre Software Technology Corporation, 1999.

31. MR216, "Multi-Rack Transient Analysis Code", Holtec International, v 2.0 Code.

TABLE 3.7B(A)-1
LIST OF COMPUTER PROGRAMS USED

(Sheet 1 of 2)

Computer Program Version	Software or Operating System	Computer Hardware	Note
Program Number 1130-CM-02X	IBM Application Program Scientific Subroutine Package	IBM 1130 and IBM 370	a
Program Number 360A-CX-16X	IBM Application Program, System/360, Continuous System Modeling Program (CSMP)	IBM 360	
Version 3 April, 1974	MRI/STARDYNE	CDC 6600	a
December 1971	ANSYS	CDC 6600 Series and IBM 360/65	a
May 1974	McAuto/ICES-STRUDL ICES-STRUDL-II	IBM 370	
Version II Program Number 1130-EC-03X	IBM Application Program Structural Engineering System Solver (STRESS)	IBM 1130 and IBM 370	a
February 1977	ADLPIPE	IBM 1130, IBM 360/370, and CDC 6600	a
Program Number 2D17	ADLPIPE	CYBER 175/176	a
October 1971	NASTRAN	CDC 6600	a
1974	QUAKE	CDC 6600	b
April 1971	KALNINS	CDC 6600	a
Program Number 15c	SUPERPIPE	CYBER 170 CYBER 175 CYBER 176	a

TABLE 3.7B(A)-1
LIST OF COMPUTER PROGRAMS USED

(Sheet 2 of 2)

Computer Program Version	Software or Operating System	Computer Hardware	Note
April 1973	SCONV	CDC 6600	b
1974	SPECTRA	CDC 6600	b
May 1974	TIME	CDC 6600	b
V03/L03	STRUDL-SW	IBM 3081	c
V01/L07	SHELL-1	IBM 3081	c
V00/L02	TIMHIS6	IBM 3081	c
V00/L00	SBMMI	IBM 3081	c
4.2A	ANSYS	IBM 3081	a
5.4	ANSYS	Compac/DEC Alphaserver 1200	a
12.0	ANSYS	Personal Computer	e
12.1	ANSYS	Personal Computer	e
13.0	ANSYS	Personal Computer	e
14.0	ANSYS	Personal Computer	e
0.0	CLASSI	CYBER	d
June 1975	RESPEC	CYBER	d
v2.0	DYNARACK	Personal Computer	e
950	LS-DYNA	Personal Computer	a

Notes:

a. Recognized computer program in public domain.

b. Computer program developed, verified, and documented by G&H.

c. Computer program developed, verified and documented by SWEC.

d. Computer program verified and documented by Impell.

e. Computer program verified and documented by Holtec International.

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 Description

The Reactor Containment structure is a fully continuous, steel-lined, reinforced concrete structure. It consists of a vertical cylinder and a hemispherical dome and is supported on an essentially flat foundation mat with a reactor cavity pit projection. The Containment superstructure is independent of the adjacent interior and exterior structures. Sufficient space is provided between the Containment and the adjacent structures to prevent contact under all combinations of loadings.

3.8.1.1.2 Dimensions of Containment

The dimensions of the Containment are as follows:

1. Inside diameter (ID): 135 ft 0 in.

2. Height of cylinder (top of foundation mat to dome spring line): 195 ft 0 in.

3. Inside radius of hemispherical dome: 67 ft 6 in.

4. Thickness of cylindrical walls: 4 ft 6 in.

5. Thickness of dome: 2 ft 6 in.

6. Foundation mat thickness: 12 ft 0 in.

7. Top of the foundation mat: approximately 4 ft 6 in. below grade

3.8.1.1.3 Containment Function

The Containment structure is designed to serve the following functions:

1. Provide vapor containment and limit the leakage from the Containment following an accident within the Containment

2. Isolate the RCS from postulated extreme environmental conditions, including tornadoes and external missiles

3. Biological shielding

3.8.1.1.4 Arrangement of Main Reinforcing Steel

The principal reinforcement used in the mat, walls, and dome are No. 18 bars, made continuous at splices by the use of Cadweld mechanical connectors produced by the Erico Corporation.

The reinforcing steel pattern in the cylinder wall consists of vertical bars at each face, horizontal hoop bars at each face, and 45-degree diagonal bars, in each direction, near the outside face.

The foundation mat is reinforced with top and bottom layers of bars placed as shown on Figure 3.8-12.

The dome reinforcement consists of top and bottom meridional layers of rebars, extending from the vertical bars of the cylindrical wall and top and bottom layers of circumferential hoop bars, as shown on Figure 3.8-11.

The meridional reinforcement terminated in the dome is anchored by the use of a positive mechanical anchor, such as a bearing plate cadwelded to the end of the bar, and satisfies the other anchorage requirements in accordance with CC-3531.1.2 of the ASME-ACI 359 document.

At penetration openings, reinforcing steel is generally curved around the openings where practical, and supplemental bars are provided around the opening as required. At large major penetrations such as the personnel lock and the equipment hatch some of the wall reinforcement is terminated at the opening by cadwelding steel plates on the end of the bar. Additional reinforcing is provided around these openings to carry stress concentrations and redistributions at these discontinuities. For details, see Figures 3.8-13 and 3.8-14.

3.8.1.1.5 Steel Liner

The entire inside face of the Containment (mat, walls, and dome) is lined with a continuous welded steel liner plate, attached with anchors to the reinforced concrete, to ensure a high degree of leaktightness. The thickness of the liner in the wall is 3/8 in. and in the dome is 1/2 in.; a 1/4 in. thick plate is used on top of the foundation mat and covered with a layer of concrete. Local thickened liner plate sections are provided at penetrations, at major pipe and duct support attachments and at crane and rotating platform girder brackets, and at the bottom of the cylindrical wall's steel liner. Overlay plates and/or structural shapes may be used on the interior side of the liner for support of minor pipes and ducts, conduits, cable trays, and equipment.

Leak-chase channels are provided at liner seams which, after construction, are inaccessible for other means of leaktightness examination. For typical liner details, see Figures 3.8-5 and 3.8-6.

3.8.1.1.6 Containment Penetrations and Attachments

Access to the Containment structure is provided by a personnel airlock, an emergency airlock, and an equipment hatch. Containment airlocks are tested in accordance with 10CFR Part 50 Appendix J, Option B. A constant pressure of P_a is used to pressurize the volume between the airlock seals.

1. Personnel Airlock

   The personnel airlock is an approximately 9 ft inside diameter electro-hydraulically operated double-door assembly. Each door is hinged and gasketed, with leakage test taps aligned to the annulus between the gasket sealing surfaces. Both doors are interlocked so that if one door is open, the other cannot be activated. Both doors are also furnished with hydraulic actuated as well as manual pressure equalizing valves which can

be operated by persons leaving or entering the personnel hatch. Plan and elevation drawings of the personnel airlock are shown in Figures 3.8-20 and 3.8-21. The configuration of the airlock seal test provisions is shown in Figure 3.8-21. Figure 3.8-22 identifies all personnel airlock mechanical and electrical penetrations. The personnel airlock has provisions for test pressurization at a pressure of P_a of the space between the two grooves at both ends of the airlock as well as provisions for pressurization at a pressure of P_a of the volume between the airlock doors. The doors are designed to maintain their functional capability during testing with no additional requirements for blocking beyond normal locking procedure.

2. Emergency Airlock

The emergency airlock is an approximately 5-ft 9-in. inside diameter manually operated double-door assembly, with 2-ft 6-in. diameter doors. Both doors of the emergency airlock are furnished with manually operated pressure-equalizing connection and valves which are interlocked with the door operating mechanism and serve to equalize differential pressure across locked doors. The reactor building to airlock door (interior) requires installation of strongbacks for the performance of the overall leakage check. Other operating features are similar to those of the personnel airlock described previously. Figure 3.8-23 identifies all emergency airlock mechanical and electrical penetrations.

3. Equipment Hatch

The equipment hatch is a 16-ft 0-in. ID single closure penetration. The bolted hatch cover is mounted on the inside of the Containment, and is double-gasketed with a leakage test tap between the gaskets. The hatch cover is provided with a hoist for handling. For details of the airlocks and equipment hatch, see Figure 3.8-9.

4. Pipe Penetrations

Other smaller penetrations through the Containment include the main steam and feedwater lines, hot and cold pipes, the fuel transfer tube, and electrical conductors. All penetration sleeves are welded to the liner and anchored into the reinforced concrete Containment wall. For typical details, see Figures 3.8-7 and 3.8-8.

5. Fuel Transfer Tube Penetration

A fuel transfer tube penetration is provided for fuel transfer between the refueling canal in the Containment structure and the spent fuel pools in the Fuel Building. The penetration consists of a 20 in. stainless steel pipe inside a carbon steel sleeve. The inner pipe acts as the transfer tube; the outer tube is welded to the Containment liner. Bellows expansion joints are provided to permit differential movements. The fuel transfer tube is equipped with a bolted blind flange with double O-ring seals inside the containment.

6. Electrical Penetrations

Header plate type penetrations are used for electrical conductors passing through the Containment. The penetration header plate with double O-ring gaskets is bolted to a

weld neck flange which is welded to a steel penetration sleeve. The steel penetration sleeves are welded to the Containment vessel liner. See Section 8.3 for additional information.

7. Liner Attachments

Major pipe and duct supports, and crane and Containment access rotating platform girder support brackets are welded to a thickened section of the liner plate, and anchored into the reinforced concrete Containment wall, as shown on Figure 3.8-6. Overlay plates and/or structural shapes may be used from the interior side of the liner, for support of minor pipes and ducts, conduits, cable trays and equipment.

8. Containment Alternate Access for the Steam Generator and Reactor Pressure Vessel Head Replacement (Unit 1)

The Steam Generator (SG) and Reactor Vessel Head (RVH) Replacement Project created and restored a construction alternate access in the Containment Building (Containment Alternate Access) in accordance with administrative procedures and the design control program. The alternate access was used to facilitate the movement of original and replacement SGs and RVH out of and into the Containment Building. In accordance with the ASME Section XI repair/replacement program, the alternate access was restored consistent with the original containment specifications with any exceptions reconciled to the original specification.

Codes and Specifications

Restoration of the Containment Alternate Access was performed as a repair/replacement activity in accordance with the requirements of ASME Section XI, 1998 edition, 1999 and 2000 addenda.

The basic code for the restored Containment Building structure are appropriate portions of the Proposed Standard Code for Concrete Reactor Vessels and Containments (April 1973); ASME-ACI 359. The restored structure meets all applicable design loads and load combinations required by ASME-ACI 359.

Concrete placement, curing, and repair were in accordance with ACI 301-05. Concrete mix proportioning was per ACI 211.1-91 (reapproved 2002).

Project specification for restoration of the Containment Alternate Access address:

- Reinforcing steel procurement, testing and placement
- Cadweld® reinforcing steel splices procurement, testing and installation
- Concrete mix design, testing and placement
- Structural steel and materials procurement

Liner Restoration

The cut section of the Containment Building liner plate was rewelded to the liner plate with a full penetration weld. The new liner plate seam welds were examined using NDE methods specified within CC-5520. Liner weld was leak tested by vacuum box test method to satisfy leaktightness requirements of NRC Regulatory Guide 1.19.

Replacement material was purchased for Nelson studs in accordance with the requirements of the original plant specification for the Unit 1 liner plate.

Reinforcing Steel Restoration

The reinforcing steel bars cut during the creation of the Containment Alternate Access were reinstalled using Cadweld® splices or welding, as required. Reinforcing steel bars that were damaged during the creation of the access were repaired in accordance with AWS D1.4-98 or were replaced with reinforcing steel procured in accordance with the project specification. New No. 6 and No. 18 reinforcing steel used for the Containment Building wall restoration conform to ASTM A615 Grade 60 or ASTM A706 and meet or exceed the additional physical and chemical composition requirements described in UFSAR Section 3.8.1.6.2 for the Containment Building structure existing reinforcing steel.

Concrete Restoration

The concrete removed from within the Containment Alternate Access was restored with fresh concrete with a specified 28-day compressive strength of 4000 psi. Fresh concrete was qualified, tested, mixed, and placed in accordance with the project specification.

3.8.1.1.7 Drawings

For various Containment structure details described in Subsection 3.8.1.1, see Figures 3.8-1 through 3.8-15.

3.8.1.2 Applicable Codes, Standards, and Specifications

3.8.1.2.1 Basic Code

The basic code used for the materials, design, fabrication, construction, examination, testing, and surveillance of the Containment are the appropriate portions of the Proposed Standard Code for Concrete Reactor Vessels and Containments (April 1973), which was issued for trial use and comments. This basic code was developed by the joint ACI-ASME Technical Committee on Concrete Pressure Components for Nuclear Service, which is made up of ACI Committee 359 and the ASME B&PV Code, Section III, Division 2, Subgroup on Concrete Components. The specific portions of this document that apply (except where otherwise specifically indicated in this FSAR) are as follows:

1. Subsection CA (General Requirements)

   Article CA-4000 Quality Assurance

2. Subsection CC (Concrete Containment)

Article CC-1000	Introduction
Article CC-2000	Materials (except CC-2232.4, CC-2240, and CC-2400, for Preplaced Aggregate Concrete, Grout, and Materials for Prestressing Systems; and except portions of CC-2231.2 for testing for: coefficient of thermal expansion - CRD-C-39 thermal conductivity - CRD-C-44 creep - ASTM C-512 shrinkage coefficient - ASTM C-157 aggregates for radiation - shielding concrete - ASTM C-637)
Article CC-3000	Design (except CC-3830 for Transitions from Concrete Containment to Steel Containment Vessels and Prestressed Concrete sections)
Article CC-4000	Fabrication, Construction, and Installation (except CC-4230 for Preplaced Aggregate Concrete, and CC-4400 for Fabrication and Installation of Prestressing Systems)
Article CC-5000	Construction, Testing and Examination (except CC-5235 for Preplaced Aggregate Concrete and CC-5400, for Examination of Prestressing Systems)
Article CC-7000	Concrete Containment Structures Protection Against Overpressure
Article CC-9000	Inservice Surveillance (except CC-9230 for Structural Integrity of Prestressed Concrete Containments)
Appendix I	Tables of Prestressing and Liner Materials (except that materials for prestressing do not apply)
Appendix III	Glossary of Terms and Symbols
Appendix VI	Porosity Charts
Appendix IX	Nondestructive Examination Methods
Appendix D	Nonmandatory Preheat Procedures

The procedures used in the design of this facility are consistent with the requirements contained in the applicable portions of the ASME-ACI 359 document as described in the following paragraphs.

The code (hereinafter referred to as the ASME-ACI 359 document) is used as the basic code for this facility because it is by far the most complete, comprehensive document available for the design, construction, inspection and testing of a concrete containment structure. Another advantage is that this document is virtually complete, in itself, without reference to portions of other codes or standards. Reference to other documents such as ASTM specifications or ACI standards are made only where the total document is applicable.

The previously listed portions of the ASME-ACI document that apply are all of the applicable requirements of Subsection CC (which comprises the technical requirements for a concrete containment), the applicable appendices, and some aspects of the Quality Assurance Article (CA-4000). The exceptions taken concern items which are not applicable (preplaced aggregate, grout, and prestress systems). Also, in answer to AEC (presently NRC) Question 3.6 of Amendment No. 1, conservative modifications are made regarding some load combinations and allowable stresses. (See Subsections 3.8.1.2.1, 3.8.1.2.5, 3.8.1.3.1, 3.8.1.3.2, 3.8.1.4.5, 3.8.1.5.1, 3.8.1.5.2, and 3.8.3.2.4.)

Requirements for punching shear are in accordance with Section 11.10.3 of the 1971 edition of the ACI 318 Code.

Specific references to the articles in Subsection CA, General Requirements, which are of a legal nature rather than a technical nature (Articles CA-1000, CA-2000, CA-3000, CA-4000, CA-5000, and CA-8000) have been omitted. These articles include requirements for such items as code jurisdiction, effective dates of code edition and addenda, certificates of authorization, responsibility of parties, stamping of containment, inspector's certification, authorized inspection agency, and so forth. These legal requirements are not applicable to the Comanche Peak Nuclear Power Plant (CPNPP) since the Code edition in force for this project is the trial use and comments issue.

3.8.1.2.2 ACI Committee 349

Since the ASME-ACI 359 document used as the basic code is co-sponsored by the ACI, it supersedes the ACI Committee 349 document, Criteria For Reinforced Concrete Nuclear Power Containment Structures; therefore, the ACI 349 criteria, published in the ACI Journal, January 1972, is not used as the basic containment criteria. However, the following sections of the ACI 349 criteria complement the requirements of the ASME-ACI 359 document and are used as a reference in the design of this facility:

1. Section 2.2.1 — Concerning strain limitations on self-limiting-type bending moments (used in conjunction with CC-3110(b) of the ASME-ACI 359 document)
2. Appendix C — method of calculating stresses and strains when a thermal gradient is combined with other loads.
3. The ASME-ACI 359 document does not provide guidance in determining thermal stresses. Therefore, the guidance provided in ACI 349-76, "Code Requirements for Nuclear Safety Related Concrete Structures", Appendix A, "Thermal Considerations" is used.

3.8.1.2.3 Additional Specifications and Standards

The following is a list of specifications and standards that are referred to in the applicable portions of the ASME-ACI 359 document described in Subsection 3.8.1.2.1 and which are applicable to this facility.

1. Liner, Penetrations, Containment Vessel Metal Components, and Attachments:

ASME B&PV Code, Section III, Division I, Subsection NE, 1971 through and including the 1973 Summer addenda (for the electrical penetration sleeves, fuel transfer tube penetration sleeve, emergency and personnel air locks, and equipment hatch) and 1974 through and including the Summer 1976 addenda (for process piping penetrations subjected to pressure-induced stresses and unsupported by concrete for load-carrying purposes).

ASME B&PV Code, Section V, 1974 (for liner radiographic examinations and electrical penetration ultrasonic examinations)

ASME B&PV Code, Section IX, 1971 through and including the Summer 1973 addenda (for welding qualifications)

ASME B&PV Code, Section II, 1971 through and including the Summer 1973 addenda, Part A (ferrous materials) and Part C (welding rods, electrodes, and filler materials)

AISC Specification for the Design, Fabrication and Erection of Structural Steel For Buildings, 1969 including Supplement Numbers 1, 2, and 3 hereafter referred to as AISC Specification.

Except: when supported by an engineering analysis, connections using A325 or A490 high strength bolts need not be pretensioned to the values required by AISC Specification, Table 1.23.5 (for steel brackets and attachments)

ASTM A 20-72a	Specification for General Requirements for Delivery of Steel Plates for Pressure Vessels
SA 370-74	Specification for Methods and Definitions for Mechanical Testing of Steel Products
ASTM A 578-71b	Specification for Straight-Beam Ultrasonic Examination of Plain and Clad Steel Plates for Special Applications
SA 537-74	Specification for Carbon Manganese Silicon Steel Plates, Heat Treated for Pressure Vessels
SA 333-74	Specification for Seamless and Welded Steel Pipe for Low-Temperature Service
SA 182-74	Specification for Forged or Rolled Alloy Steel Pipe Flanges, Forged Fittings, and Valve and Parts for High-Temperature Service
SA 350-74	Specification for Forged or Rolled Carbon and Alloy Steel Flanges, Forged Fittings, and Valves and Parts for Low-Temperature Service
SA-320-74	Specification for Alloys Steel Bolting Materials for Low-Temperature Service
SA-105-74	Specifications for Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for High-Temperature Service

SA-354-74	Specification for Quenched and Tempered Alloy Steel Bolts and Studs with Suitable Nuts
ASTM A 36-74	Specification for Structural Steel (for miscellaneous attachments)
ASTM A 108-73	Specification for Steel Bars, Carbon, Cold-Finished, Standard Quality
SA 516-74	Specification for Carbon Steel Plates for Pressure Vessel for Moderate and Lower Temperature Services

2. Reinforcing Steel

ASTM A 615-72	Specification for Deformed and Plain Billet Steel Bars for Concrete Reinforcement
ASME SFA 5.1	Specification for Mild Steel Covered Arc-Welding Electrodes

3. Concrete

ASTM C 150-74	Specification for Portland Cement
ASTM C 33-74	Specification for Concrete Aggregates
ASTM C 131-69	Test for Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine
ASTM C 142-71	Test for Clay Lumps and Friable Particles in Aggregates
ASTM C 117-69	Test for Materials Finer Than No. 200 Sieve in Mineral Aggregates by Washing
ASTM C 87-69	Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
ASTM C 40-73	Test for Organic Impurities in Sands for Concrete
ASTM C 289-71	Test for Potential Reactivity of Aggregates (Chemical Method)
ASTM C 136-71	Test for Sieve or Screen Analysis of Fine and Coarse Aggregates
ASTM C 88-73	Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C 127-73	Test for Specific Gravity and Absorption of Coarse Aggregate
ASTM C 295-65	Recommended Practice for Petrographic Examination of Aggregates for Concrete
ASTM D 512-67	Tests for Chloride Ion in Water and Waste Water

ASTM C 151-74a	Test for Autoclave Expansion of Portland Cement
ASTM C 191-74	Test for Time of Setting of Hydraulic Cement by Vicat Needle
ASTM C 260-74	Specification for Air-Entraining Admixtures for Concrete
ASTM C 494-71	Specification for Chemical Admixtures for Concrete
ACI 211.1-74	Recommended Practice for Selecting Proportions for Normal and Heavy-Weight Concrete
ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ASTM C 143-74	Test for Slump of Portland Cement Concrete
ASTM C 172-71	Sampling Fresh Concrete
ASTM C 192-69	Making and Curing Concrete Test Specimens in the Laboratory
ASTM C 31-69	Making and Curing Concrete Test Specimens in the Field
ASTM C 39-72	Test for Compressive Strength of Cylindrical Concrete Specimens
ASTM C 109-73	Test for Compressive Strength of Hydraulic Cement Mortars (Using 2-in. (50 mm) cube specimens)
ASTM C 231-74	Test for Air Content of Freshly Mixed Concrete by the Pressure Method
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ASTM C 78-64	Test for Flexural Strength of Concrete
ASTM C 496-71	Test for Splitting Tensile Strength of Cylindrical Concrete Specimens
ASTM C 469-65	Test for Static Modulus of Elasticity and Poisson’s Ratio of Concrete in Compression
ASTM C 642-69T	Test for Specific Gravity, Absorption, and Voids in Hardened Concrete
ASTM C 94-74	Specification for Ready-Mixed Concrete
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 305-72	Recommended Practice for Hot Weather Concreting
ACI 306-66	Recommended Practice for Cold Weather Concreting

4. Testing and Surveillance

10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors (2/5/73) ANSI N45.4-1972 Leakage-Rate Testing of Containment Structures for Nuclear Reactors

3.8.1.2.4 Specifications and Standards Not Referred to in ASME-ACI 359

AWS D12.1-61 Recommended Practices for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction

AWS D1.4-1998 Structural Welding Code (Unit 1)

3.8.1.2.5 Summary of Principal Plant Specifications

1. The principal plant specifications for the steel-lined, reinforced concrete containment are the fabrication and construction specifications for the following:

   a. Liner (including penetrations and attachments)

   b. Reinforcing steel

   c. Splicing of reinforcing steel

   d. Concrete

2. The applicable portions of the ASME-ACI 359 document, as described in Subsection 3.8.1.2.1, are included in the plant construction specifications in regard to materials, construction techniques, fabrication, welding, examination, testing, and so forth. The following are the principal appropriate portions of the ASME-ACI 359 document which are incorporated into the plant specifications:

   a. Liner Specifications

CC-2500	Materials for Liners
CC-2520	Special Materials Testing
CC-2530	Examination and Repair of Liner Materials
CC-2540	Marking of Liner Materials
CC-2600	Welding Materials
CC-2612	Weld Metal Tests NB-2432 may be used in lieu of CC-2612.2 for chemical analysis of filler metal or weld deposit.
CC-2620	Stud Welding Materials Except, 1100 Aluminum may be used as a flux for stud welding without a chemical analysis of each batch as stipulated by CC-2623.2.
CC-2630	Identification of Welding Materials

CC-2700	Materials Manufacturers' Quality Assurance Programs
CC-3840	Design of Welded Construction
CC-4120	Certification of Materials and Fabrication by Component Manufacturer and/or Installer and/or Constructor
CC-4520	Forming, Fitting, and Aligning
	Except for the following: When qualifying the procedure for the forming and bending process, the Charpy V-notch impact test temperatures, of the specimens used to establish a transition curve, may be conducted at a minimum of five different temperatures distributed throughout the transition region, in lieu of CC-4521.3.2(e) requirement for conducting tests at each temperature increment of 10°F.
	When thermal cutting is performed to prepare weld joints or edges, to remove attachments or defective material or any other purpose, preheating may be in accordance with the applicable section of the ASME III B&PV Code in lieu CC-4521.1.1.
	Engineers may approve on a case by case basis other stud welding equipment in addition to the requirements of CC-4543.5(a).
	Engineers may review post-weld heat treatment records in lieu of CC-4552.2.2 requirement of review by inspector.
CC-4530	Welding Qualifications
CC-4540	Rules Governing Making, Examining, and Repairing Welds

Material for non-pressure parts or pads which are permanently attached by welding to liners may use the following acceptance criteria in lieu of CC-4543.1:

The allowable loadings for attachment materials may be determined by the principles of fracture mechanics based on calculated normal tensile stresses in a defect free stress model at the surface nearest the location of an assumed defect or charpy impact testing may be waived in accordance with the provisions of the 1986 edition of ASME III, Division 2, Subsection CC4543.1.

CC-4550	Heat Treatment
CC-5520	Required Examination of Welds

The criteria for extent and frequency of radiographic examination may be as follows in lieu of CC-5521.1.1.(a) (b) (c) (d) (e) and (f):

1. For each welder and welding position (flat, vertical, horizontal and overhead), the first 10 feet of weld shall be examined radiographically. If this radiograph meets the acceptance standards that 10 feet of weld shall be accepted.

2. For the first three shell rings only, for each welder and welding position (flat, vertical, horizontal and overhead) the first 10 feet of weld shall be performed on a representative mock-up. These mock-ups shall simulate as close as practicable the actual conditions that the welder will experience during [Aduction. These welds shall be 100 percent radiographically examined. Should a question of interpretation arise as to the acceptance of the weld in accordance with the radiographic acceptance standard, a cross-sectional coupon can be cut from the weld to visually verify or refute the film interpretation.

   Further, all production welds on the first three shell rings are to be 100 percent examined by magnetic particle inspection and vacuum box tested, in lieu of radiographic examination.

3. If the radiography in “1” above does not meet the acceptance standards, the portions of the 10 foot increment of weld which do not meet the performance requirements shall be repaired and re-radiographed.

4. Welders who have satisfactorily welded the first 10 feet of weld as described above shall have one 12-inch-long radiograph made of each subsequent 50 feet of weld or fraction thereof which he produces. If the first radiograph in each 50 foot increment meets the acceptance standards, the welder shall be permitted to continue welding the next 50-foot increment of production weld. A minimum of 2 percent of all liner seam welds shall be examined by radiography.

5. If the 12-inch radiograph in the 50-foot-long increment of weld does not meet the acceptance standards, two 12-inch films shall be taken at least 1 ft removed on each side from the original spot within the 50-foot-long increment. If these radiographs meet the radiographic acceptance standards, the 50 feet of weld represented shall be accepted. The defective areas shown in the first radiograph shall be repaired and re-radiographed.

6. If either of the second radiographs does not meet the acceptance standards, the entire weld test unit is unacceptable. The remaining portion of the 50-foot increment of this weld shall be radiographed. The portions of the 50-foot increment of weld which do not meet the acceptance standards are to be repaired and re-radiographed.

The criteria for magnetic-particle (MT or MPE) or liquid-penetrant (PT or LPE) examination in lieu of radiographic examination may be used as follows in lieu of CC-5521.2.1:

Where radiographic examination of liner seam welds is not feasible or where the weld is located in areas which will not be accessible after construction, the entire length of weld shall be examined by the magnetic-particle or liquid-penetrant method. Where magnetic-particle or liquid-penetrant inspection discloses welding that does not meet the magnetic-particle or liquid-penetrant acceptance standards, additional testing shall be performed to the same extent as required for radiography in CC-5520. Unacceptable indications of the weld shall be eliminated or repaired by welding as required.

In addition to the required radiography of CC-5520, all seam welds at abrupt changes in liner configuration (i.e., cylinder to sphere) shall be examined by the magnetic particle method for their entire length.

The following criteria may be used in lieu of CC-5523.1 for magnetic-particle (MT or MPE) or liquid-penetrant (PT or LPE) examination of full penetration attachment welds to containment liner inserts for welding and examinations performed prior to and including June, 1989:

Full penetration attachment welds shall be examined by examining a portion of the welds using the magnetic-particle (MT or MPE) or liquid penetrant (PT or LPE) method with a linear indication of greater than 1/2" being unacceptable. The accessible portion of the welds below the springline (El. 1000'-6") will be inspected to establish the overall acceptability of the welds inspected in this manner.

This exception does not include the insert plates for the polar crane and the rotating platform. Modifications/additions to full penetration attachment welds to the containment liner performed after June, 1989, will use the CC-5523.1 criteria.

CC-5530 Acceptance Standards

The acceptance standards for the liner butt welds examined by radiography may be in accordance with ASME B&PV Code Section VIII, Division 1, paragraph UW-51 and AEC Regulatory Guide 1.19 in lieu of CC-5532 Radiography Acceptance Standards.

Non-full penetration attachment welds to the reactor containment liner may use the following acceptance standards:

The fillet welds of the attachment to the reactor containment liner shall be examined by either the magnetic particle method or liquid penetrant method. The acceptance standards shall be in accordance with the following criteria.

For attachment welds, fillet weld size specified on the drawings are the minimum size required, except as permitted below, along the full length of the weld joint. Additional welding is acceptable provided it does not distort the items being joined together.

A fillet weld in any single continuous weld may be less than the fillet weld dimension by not more than 1/16 inch provided that the undersize portion of the weld does not exceed 10 percent of the length of weld.

Liquid Penetrant Acceptance Standard

1. Linear indications in which the length is more than three times the width.

2. Round indications are indications which are circular or elliptical with the length less than three times the width.

3. Only indications with major dimensions greater than 1/16 inch are considered relevant.

4. Unless otherwise specified, the following relevant indications are unacceptable.

   (a) Any cracks and linear indications.

   (b) Rounded indications with dimensions greater than 3/16 inch.

   (c) Four or more rounded indications aligned and separated by 1/16 inch or less, edge to edge.

   (d) Ten or more rounded indications in any 6 square inch of surface, the major dimension of this area not exceeding 6 inches, with the area taken in the most unfavorable location relative to the indications being evaluated.

Magnetic Particle Acceptance Standards

1. Only indications with major dimensions greater than 1/16 inch are considered relevant.

2. Unless otherwise specified, the following relevant indications are unacceptable:

   (a) Any cracks or linear indications.

   (b) Rounded indications with dimensions greater than 3/16 inch.

   (c) Four or more rounded indications aligned and separated by 1/16 inch or less, edge to edge.

   (d) Ten or more rounded indications in any 6 square inch of surface, the major dimension of this area not exceeding 6 inches, with the area taken in the most unfavorable location relative to the indications being evaluated.

CC-5540 Examination of Stud Welds

(For protective coatings on liner and other steel, see Subsection 3.8.1.6.5, Item 2.g.)

b. Reinforcing Steel Specification

CC-2300	Materials for Reinforcing Systems
CC-2320	Material Identification
CC-2330	Special Materials Testing
CC-2700	Materials Manufacturers' Quality Assurance Programs
CC-3430	Concrete Temperatures
CC-3533	Reinforcing Steel Cover and Spacing Requirements
CC-4120	Certification of Materials and Fabrication by Component Manufacturer and/or Installer and/or Constructor
CC-4320	Bending of Reinforcing Bars
CC-4340	Placing Reinforcement
CC-4350	Spacing of Reinforcement
CC-4360	Surface Condition

c. Specification for Mechanical Butt Splicing (Cadwelds) of Reinforcing Steel

CC-2700	Materials Manufacturers' Quality Assurance Programs
CC-4120	Certification of Materials and Fabrication by Component Manufacturer and/or Installer and/or Constructor
CC-4333	Mechanical Butt Splices Using Sleeve and Filler Material
CC-4335	Mechanical Joints
CC-5320	Examination of Sleeve with Filler Metal Connection

d. Concrete Specification

CC-2220	Materials for Concrete
CC-2230	Concrete Mix Design
CC-2250	Marking and Identification of Concrete Materials
CC-2700	Materials Manufacturers' Quality Assurance Programs
CC-4220	Batching, Mixing, and Transporting
CC-4225	Depositing
CC-4240	Curing

CC-4250	Formwork and Construction Joints
CC-4260	Cold and Hot Weather Conditions
CC-4270	Repairs to Concrete
CC-5220	Concrete Materials (testing and examination of)
CC-5230	Concrete (testing and examination of)

3.8.1.2.6 Applicable NRC Regulatory Guides

The following NRC (formerly AEC) Regulatory Guides are applicable to this Containment and are complied with:

NRC Regulatory Guide 1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1, 1-2-73, of former Safety Guide 10)
NRC Regulatory Guide 1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1, 12-28-72 of former Safety Guide 15)
NRC Regulatory Guide 1.18	Structural Acceptance Test for Concrete Primary Reactor Containments (Revision 1, 12-28-72 of former Safety Guide 18)
NRC Regulatory Guide 1.19	Nondestructive Examination of Primary Containment Liner Welds (Revision 1, 8-11-72 of former Safety Guide 19)
NRC Regulatory Guide 1.28	Quality Assurance Program Requirements (Design and Construction) (6-7-72 of former Safety Guide 28)
NRC Regulatory Guide 1.29	Seismic Design Classification (Revision 2, 2-76 of former Safety Guide 29)
NRC Regulatory Guide 1.55	Concrete Placement in Category I Structures (6-73)

3.8.1.3 Loads and Load Combinations

3.8.1.3.1 Loads

The following loads are considered in the design of the steel-lined, reinforced concrete Containment structure (essentially in accordance with the ASME-ACI 359 document):

1. D = dead load of the Containment, and all superimposed permanent loads

2. L = live loads, comprising conventional floor and roof live loads, movable equipment loads, cables, and lateral soil pressure

3. Pa = Containment pressure load due to the DBA, at 50 psig

4. T = thermal effect

   a. To = thermal loads during normal operating conditions, including liner expansion and temperature gradients in the wall

      1. Normal operating temperature range inside the Containment is 50°F to 120°F.

      2. Ambient temperature range at the outside face of the Containment wall is 20°F to 110°F.

   b. Ta = added thermal loads (over and above operating thermal loads), exerted by the liner, which may occur during an accident and which correspond to the factored accident pressure (i.e., 1.0 Pa, 1.25 Pa, or 1.5 Pa); the accident temperature causes an almost instantaneous increase in the liner temperature, with little initial effect on the temperature of the relatively thick concrete wall. This sudden increase in liner temperature creates compressive stresses and strains in the liner which thrusts against the reinforced concrete section, having an effect on the reinforcing steel similar to an added internal pressure.

      1. The design temperature corresponding to an accident pressure of 50 psig is 280°F.

      2. The design temperature corresponding to an accident pressure of 1.25 x 50 psig is 295°F.

      3. The design temperature corresponding to an accident pressure of 1.5 x 50 psig is 305°F.

   c. Tt = Thermal loads during pressure test, including liner expansion and a temperature gradient in the wall; a maximum gradient of 40°F is assumed during test.

5. Seismic loads representing two magnitudes of earthquake are considered, as follows:

   a. E' = SSE

   b. E = 1/2 of SSE=OBE

   The vertical and horizontal earthquake accelerations are assumed to act simultaneously.

   The earthquake forces acting on the Containment structure are taken from the results of dynamic analyses, based on seismic input described in Section 3.7.

6. W = design wind load (See Section 3.3.)

7. Wt = tornado load including wind, differential pressure and missiles (See Sections 3.3 and 3.5.)

8. Ro = piping loads acting on the Containment during operating conditions

9. Ra = piping loads acting on the Containment, due to increased temperature resulting from the design accident

10. Yr = equivalent static load on structure or penetration generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

11. Pv = negative internal pressure during operation; maximum Pv equals 5 psig.

12. Yj = equivalent static jet impingement load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

13. Ym = equivalent static missile impact load on a structure generated by or during the postulated break, as from pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

14. Ha = represents the load on the Containment resulting from post-LOCA internal flooding.

3.8.1.3.2 Load Combinations

The design of the reinforced concrete Containment structure incorporates the service load combination requirements and the factored load combination requirements, as follows (in accordance with the ASME-ACI 359 document):

1. Service Load Combinations

   a. Construction Category

      1.0 D + 1.0 L + 1.0 To

   b. Test Category

      1.0 D + 1.0 L + 1.15 Pa + 1.0 Tt

   c. Normal Category

      1. 1.0 D + 1.0 L + 1.0 To + 1.0 E + 1.0 Ro + 1.0 Pv

      2. 1.0 D + 1.0 L + 1.0 To + 1.0 W + 1.0 Ro + 1.0 Pv

2. Factored Load Combinations

   a. Abnormal Category

      1.0 D + 1.0 L + 1.5 Pa + 1.0 (To + Ta) + 1.0 Ra

b. Extreme Environmental Category

1. 1.0 D + 1.0 L + 1.0 To + 1.0 Wt + 1.0 Ro + 1.0 Pv

2. 1.0 D + 1.0 L + 1.0 To + 1.0 E' + 1.0 Ro + 1.0 Pv

c. Abnormal Severe Environmental Category

1. 1.0 D + 1.0 L + 1.25 Pa + 1.0 (To + Ta) + 1.25 E + 1.0 Ra + 1.0 (Yr+Yj+Ym)

2. 1.0 D + 1.0 L + 1.25 Pa + 1.0 (To + Ta) + 1.25 W + 1.0 Ra + 1.0 (Yr+Yj+Ym)

3. 1.0 D + 1.0 L + 1.0 To + 1.0 E + 1.0 Ha

4. 1.0 D + 1.0 L + 1.0 To + 1.0 W + 1.0 Ha

d. Abnormal-Extreme Environmental Category

1.0 D + 1.0 L + 1.0 Pa + 1.0 (To + Ta) + E' + 1.0 Ra + 1.0 (Yr+Yj+Ym)

3. Variable and Interrelated Loads

For loads which may vary, the values (within the possible range) which produce the most critical combination of loading are used in design. For loads which are interrelated as a function of time, the maximum values of these effects do not necessarily occur simultaneously. Recognition is given to the time increments associated with such conditions.

4. Allowable Stresses

The allowable stresses associated with the service load combinations and the factored load combinations are given in Subsection 3.8.1.5.

5. Load Combinations for Localized Areas

The design load combinations used to examine the effects on localized areas such as penetrations, shell discontinuities, crane girder brackets, and local areas of high thermal gradients are the same load combinations used for the general Containment structure, as previously described.

6. Time-Dependent Loads

Time-dependent loads such as the effects of creep, shrinkage, and other related effects are ignored in the design of the reinforced concrete Containment structure. (See Subsection 3.8.1.4.1, Subsection 2).

7. Explanation of the Use of an Ultimate Strength Approach With a Load Factor of 1.0

Factored load combinations that include extreme environmental effects (SSE or tornado effects) incorporate a load factor of 1.0 using an ultimate strength approach with stresses within the range of general yield. This approach is justified based on the fact that the extreme environmental effects that are considered are of an upper bound conservative magnitude and have an extremely low probability of occurrence. In addition, the maximum SSE is assumed to occur concurrently with the DBA, an extremely unlikely occurrence. Additional margin of safety is provided by the fact that, under these factored load combinations, the average stress in the reinforcing steel is limited to 90 percent of yield, rather than full yield.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General Analysis of Entire Containment Structure

The Containment structure, including the foundation mat, is analyzed and designed for all load combinations as described in Subsection 3.8.1.3. Table 3.8-1 shows critical loading combinations, type of stress and computed and corresponding allowable stresses at key locations in the Containment structure.

1. Foundation Mat Analysis

The Containment structure foundation mat is analyzed by a finite element method of analysis using the ANSYS computer program. (See Appendix 3.7A for a description of the program.) The program uses the stiffness method of structural analysis and contains the various types of finite elements, i.e., triangular, rectangular, and quadrilateral plate elements representing membrane or bending behavior, or both, and beam elements. The model used for the foundation mat analysis includes the mat and the Containment cylindrical wall to a height of approximately 76 ft above the mat. This height of Containment wall is sufficient to represent the effect of the Containment wall stiffness on the mat behavior under the various loading conditions. The rock beneath the foundation mat is represented by appropriate linear springs. The rock mat contact area under various loading combinations is discussed in Subsection 3.8.5.4.1, Item 1.

The input to this program consists of the geometry of the structure, the material properties, the appropriate boundary conditions, and the loadings. The boundary conditions at the cut Containment wall section (approximately 76 ft above the mat) are represented by the equivalent load reactions at this point for each type of loading (dead load, pressure, thermal gradient, seismic, and so forth). The loads acting on the mat from the concrete internal structure are represented in the model as equivalent pressures at the interface surface of the mat.

The output of this program contains the displacements, rotations, forces, shears, moments, and stresses throughout the structure. This output is used for the design of the foundation mat. In regard to the Containment wall design, the output from this analysis is used only to check the design of the wall at the junction with the mat. The Containment superstructure design (walls and dome) is based on a supplementary analysis as described in Subsection 3.8.1.4.1, Item 2.

2. Containment Superstructure Analysis

The Containment shell is analyzed using the SHELL-1 computer program. (See Appendix 3.7A for a description of the program.) This program uses a numerical method which combines the direct integration and the finite difference techniques for solving general shell equations. The model consists of the hemispherical dome and the cylindrical wall down to the top of the mat. The input to this model consists of the geometry of the structure, the material properties, and the loadings. Fixed boundary conditions are used at the junction of the bottom of the wall and the top of the mat. The results of the mat analysis are used to verify the design at the bottom of the wall.

Creep and shrinkage of concrete are important considerations in the analysis and design of a prestressed concrete containment. However, for the nonprestressed reinforced concrete containment being used on this project, the effects of creep and shrinkage are not significant and can be safely ignored. Shrinkage in a reinforced concrete containment results in meridional and radial displacements which are the opposite of the displacements caused by the principal loadings, internal pressure, and temperature. Since it is not additive to these major loads, it can be ignored. The amount of cracking in a reinforced concrete containment is dependent on the tensile stresses in the shell. For load combinations which do not include internal pressure, the containment is assumed to be completely uncracked. For load combinations which include internal pressures, the analysis uses a variably cracked model, in which the concrete is completely cracked in the membrane regions, with the stiffness of the concrete ignored in both directions and only the properties of the reinforcing steel considered. The stiffness is increased in the vertical direction of the cylinder due to the decrease in the net tensile force when the dead weight of the containment is included. Also the stiffness is adjusted at the discontinuities to account for the increased stiffness when compression and/or high moment exist.

The overall Containment shell is analyzed, neglecting the presence of the penetrations. The analysis of the portions of the containment shell where the stress pattern is influenced by the major penetrations (airlocks and equipment hatch) is performed as described in Subsection 3.8.1.4.2.

Below the dome spring line, radial shear forces occur in the containment wall as a result of the design loadings, due to the discontinuity between the wall and the dome. In the original design, reinforcement was provided for the Unit 2 containment wall between elevations 993'-1 1/2" and 996'-9 1/2" to conservatively resist maximum envelope shear forces, for the most critical loading combinations using the model which gave the maximum forces. No attempt was made to reduce the reinforcement by investigating the applicability of the model used to the critical loading condition.

Some of the shear reinforcement indicated on the construction drawings for the Unit 2 containment wall at the elevations indicated above were inadvertently omitted. When this omission was discovered, after the concrete had been poured, additional shear reinforcement was provided to that called for on the construction drawings between the construction joint at elevation 997'-6" and the dome spring line.

Due to the effects of the high vertical and circumferential tensile membrane forces in the wall resulting from the critical LOCA loading combination, the following model was used to determine shear forces and to verify the as-built shear reinforcement: stiffness of the concrete was ignored in the stiffness calculations and the stiffness of the reinforcing steel only was used in both horizontal and vertical directions. All shear load in the subject area

was assumed to be resisted by the shear tie reinforcing only and no reliance was placed on the shear capacity of the concrete. The as-built condition was found to be more than adequate. Though ignored, the concrete would resist some of the shear load; in fact, the total shear load would be resisted by the concrete together with the as-built reinforcing steel shear ties. All code allowables were met for the model assumptions applicable to this structural modification of Unit 2.

3. Check of Model Validity and Analysis Results

For methods of checking the validity of the model and the results of the analysis, see Subsection 3.8.3.4.1.

3.8.1.4.2 Analysis at Major Penetrations

The effect of the major penetrations (airlocks and equipment hatch) on the Containment wall is analyzed using a finite element model. The computer program used ANSYS (See Appendix 3.7A for description.) For the purpose of this analysis, a segment of the Containment wall containing the penetration is isolated and analyzed for the same loading conditions as those for which the entire Containment shell is analyzed. The boundary of the segment is approximately three times the penetration diameter from the center of the penetrations, except at the boundary with the mat which is approximately two times the penetration diameter from the center of the penetration. The finite element model of the segment consists of solid finite elements or plate elements, or both connected at their nodes. The boundary conditions applied to this model are obtained from the analysis of the entire Containment shell, as described in Subsection 3.8.1.4.1, Item 2. The model considers the various degrees of cracking as described in Subsection 3.8.1.4.1, Item 2. The program used in this analysis considers, in addition to the strains in the plane of the wall, strains in the orthogonal direction.

The output of this analysis includes the displacements, rotations, forces, shears, and moments which are used for the design of the reinforced concrete ring girder around the penetration.

3.8.1.4.3 General Design Criteria and Procedures

The General Design Criteria (GDC) are in accordance with CC-3000 of the ASME-ACI 359 document, except as otherwise specifically indicated.

1. Concrete Tensile Strength

Concrete tensile strength is not relied upon to resist flexural and membrane tension.

2. Interaction of Liner Plate and Reinforced Concrete

The SHELL-1 analysis considers the interaction of the liner plate and the concrete structure under all conditions of loading. However, if the action of the liner results in lower concrete or reinforcement stresses, the presence of the liner is disregarded for that particular case.

In considering the interaction between the liner, when subjected to the hot accident temperature, and the reinforcing steel within the cracked concrete section, strain compatibility between the liner and the reinforcing steel is considered. In the equations of

strain compatibility for the design of the reinforcing steel for this hot liner condition, the effect of the concrete is ignored, since it is assumed to be cracked and incapable of carrying any of the tensile loads.

The interaction of the liner plate is only used to increase the effective internal pressure during an accident case. It is not used to reduce the internal pressure acting on the concrete.

3. Evaluation of Effect of Variations in Assumptions and Materials

The fact that reinforced concrete is not a homogeneous material is accounted for in the design; stiffness properties are altered where the section is assumed to crack. Properties of materials are known with sufficient accuracy, and assumptions made are sufficiently conservative so that other variations need not be considered.

4. Temperature Effects

The temperature gradient through the containment wall during operation is essentially linear and is a function of the internal operating temperature and the average external ambient temperature. The accident temperature primarily affects the liner, rather than the concrete and reinforcing steel, due to insulating properties of the concrete. By the time the temperature of the concrete within the interior of the concrete shell begins to rise significantly, the internal accident pressure within the Containment has fallen off to a point below the peak values. Therefore, it is not necessary to consider peak accident temperature in the concrete coincident with peak pressures in the Containment. (The thrust caused by the instantaneously hot liner against the reinforced concrete wall is considered simultaneously with the peak pressure.) Also, temperature stresses of the reinforcing steel in the Containment shell caused by the maximum thermal gradient do not significantly influence the capacity of the structure to resist membrane forces. Temperature gradients induce stresses in the structure which are internal in nature, causing tension on one face and compression on the other face; the resultant membrane force is zero. If loading combinations concurrent with these temperature gradient effects cause local stresses in the horizontal and vertical bars of one face to reach the yield point, any further load is transferred to the unyielding elements on the other face of the wall. Because of the self-limiting nature of stresses resulting from a thermal gradient, the reinforcing bars across a horizontal or vertical section have a magnitude of final membrane load resistance essentially equal to that which would be carried if temperature gradient effects were neglected. This design approach is basically in accordance with CC-3110 of the ASME-ACI 359 document. However, for factored load combinations which include a thermal gradient, the maximum strain in the reinforcing steel is limited to approximately 1.5 times the yield strain (in accordance with ACI 349, Section 2.2.1). The total reinforcement across any section for any factored load combination has an average tensile stress not more than 0.9 times the yield stress.

3.8.1.4.4 Design of Reinforced Concrete at Penetrations

1. Major Penetrations (Airlocks and Equipment Hatch)

The Containment wall at the major penetrations is designed as a ring beam around the openings and is thickened around these penetrations. The results of the analyses

performed at these openings (see Subsection 3.8.1.4.2) are used to design the ring beams. The ring beams are designed to resist biaxial bending moments, axial tension, torsion, and biaxial shear, resulting from all the load combinations listed in Subsection 3.8.1.3.2. At the openings, some of the typical wall reinforcement close to the outside of the opening is curved around the openings. The remainder of the typical wall reinforcement is terminated at the opening by cadwelding steel plates on the end of the bars.

Additional reinforcement is provided around the opening, principally in circumferential and radial directions relative to the centerline of the opening, to limit stresses to the allowable values (see Subsection 3.8.1.5). For the typical arrangement of reinforcing steel at these major openings, see Figure 3.8-14.

2. Smaller Penetrations (for Pipe Lines)

Penetration sleeves are anchored into the reinforced concrete Containment wall by means of steel lugs welded to the sleeve. The anchors and local wall reinforcement are designed to resist the torsion, bending, and shear that the pipe is capable of exerting on the penetration, as limited by the full plastic strength of the pipe and an axial load based on the maximum possible pipe jet reaction. All possible combinations of loadings are considered to act simultaneously. The typical wall reinforcement is curved around these penetrations and kept continuous wherever possible. Additional local reinforcement is provided around the opening, as required to resist the loads imposed by the pipe, as described in the preceding paragraphs. This additional reinforcement is provided in vertical, horizontal, and radial directions, relative to the centerline of the opening. For typical arrangement of reinforcing steel at these penetrations, see Figure 3.8-13.

3.8.1.4.5 Analysis and Design for Shear Effects

1. Tangential Shear

Tangential shear is due principally to horizontal seismic motion. The maximum intensity of concrete tangential shear stress, Vu (in psi), is obtained from the results of the analysis in accordance with Subsection 3.8.1.4.1, Item 2, for all load combinations.

The factored load design for tangential shear is in accordance with CC-3411.5 of the ASME-ACI 359 document, except that the maximum allowable tangential shear stress carried by the concrete, Vc, does not exceed 60 psi. In regard to this section of the ASME-ACI 359 document, this Containment complies with the requirements CC-3411.5 (a) through (d), and it does not support principal equipment laterally; therefore, the maximum allowable tangential shear stress which the concrete can be assumed to safely resist is 60 psi.

Where the maximum tangential shear stress, Vu, exceeds the concrete allowable shear, Vc, the excess (Vu-Vc) is resisted by inclined reinforcement placed near the outside of the wall at 45 degrees in each direction. Design of the reinforcing steel for tangential shear is in accordance with CC-3521.1.1 of the ASME-ACI 359 document for factored load design. In calculating the stresses in the reinforcing steel caused by tangential shear, compatibility of strains between the inclined steel and the vertical and horizontal steel is considered in accordance with M.J. Holley's Provision of Required Seismic

Resistance, included in Seismic Design for Nuclear Power Plants by the MIT Press. For arrangement of reinforcing steel, see Figure 3.8-10. (Service load combinations are checked in accordance with CC-3420 and CC-3522 of the ASME-ACI 359 document.)

2. Radial Shear

The maximum radial shear occurs at the junction of the bottom of the Containment wall and the top of the foundation mat, under the pressure loading. The values of radial shear are obtained from the results of the analysis described in Subsection 3.8.1.4.1, Item 2, for all load combinations. Design for radial shear is in accordance with CC-3411, CC-3421, and CC-3521 of the ASME-ACI 359 document. Radial shear loads are resisted by radial bars inclined at 45 degrees with the horizontal and extending between the vertical bars near the inside surface and the outside surface of the cylinder wall. Above the mat, where the radial shear is maximum, plate bars 4 in by 1 in are welded to the vertical reinforcing steel. See Figure 3.8 10 for arrangement of radial shear reinforcement.

3.8.1.4.6 Analysis and Design of Liner and Anchorage

The liner for the cylindrical walls is 3/8 in-thick steel plate and for the dome 1/2 in-thick steel plate, each anchored into the concrete with 5/8 in. by 6 3/8 in. long, headed, welded studs, Type H4, as produced by the Nelson Stud Welding Co., or engineer-approved equal. Studs are spaced to satisfy the criteria described in Subsection 3.8.1.5.3. The approximate spacing of the anchor studs in the cylindrical wall and dome is 12 in. each way. The wall and dome liner serve as the inside formwork for placing of concrete. The liner on top of the mat is 1/4-in. thick. This bottom liner is installed after foundation mat construction by welding at seams to structural members that have been embedded in the top of the mat. These embedded structural anchors are approximately 8 to 10 ft apart. The liner on top of the mat is covered with approximately 30 in. of concrete. The vertical wall liner is anchored at the foundation mat; this end anchor is designed to resist the maximum compression and tension to which the liner plate is subjected. See Figure 3.8-5 for liner anchorage details.

The analysis and design of the liner, anchors, and attachments are in accordance with CC-3120, CC-3600, CC-3700, and CC-3800 of the ASME-ACI 359 document.

The liner and anchors are designed to withstand the effects of all load combinations as described in Subsection 3.8.1.3.2, using load factors equal to 1.0.

The stability of the liner is ensured by anchorage to the reinforced concrete. The anchorage system prevents distortions sufficient to impair leaktightness. The liner plate anchorage system is designed to withstand without rupture all in-plane (shear) loads or deformations exerted by the liner plate and also to resist all loads applied normal to the liner surface. The anchors are designed to elastically carry the forces resulting from the various load combinations, or to have sufficient ductility to relieve the forces, or to bring necessary additional anchors into action without rupture of the liner or anchor. The anchorage is designed so that if any one anchor fails, successive failure of adjacent anchors does not occur in the manner of a chain reaction. See ACI 349, Section 2.6.5.5.

In general the maximum load affecting the design of the liner and anchors, is that caused by the maximum temperature rise due to an accident. This temperature increase causes the liner, which is restrained by the reinforced concrete wall, to be stressed in compression. The

compressive stress is calculated by equating the strains between the liner and the reinforced wall. The resulting stresses and strains in the liner are less than allowable as stated in Subsection 3.8.1.5.3. The maximum load in an anchor stud is an in-plane shear load which can occur if the plate on one side of an anchor bows inward in a flexural mode, causing a reduction of membrane compression on one side of the anchor. This inward bowing of the plate can be caused by initial construction deformity, variation of liner plate curvature, loss of an anchor, and similar occurrences.

The resulting unbalanced plate stress imparts a shear load and corresponding displacement on the adjacent anchor, and, to a lesser degree, on each successive anchor further away from the bowed-in plate. The shear load versus displacement of the anchor stud is based on test data developed by the Nelson Stud Welding Co. The analysis to determine the load and displacement on each stud is performed by making a series of successive approximations using the test curves for anchor shear load versus displacement. The analysis is based on maintaining equilibrium of loads in the plate and anchors for any free body cut through a section of plate and compatibility between the strain in the plate and displacement of the studs. The resulting maximum loads and displacements in the studs are less than allowable as stated in Subsection 3.8.1.5.3.

The anchor design and analysis consider the effects of the following (in accordance with CC-3800 of the ASME-ACI 359 document):

1. Variation of liner plate curvature: the anchors are designed for possible inward bowing of the plate, as described previously.

2. Variation in liner plate thickness due to rolling tolerances: the range of maximum and minimum plate thickness is assumed in the design of the liner plate and anchors.

3. Variation of liner plate yield strength: the anchors are analyzed assuming the liner remains elastic under all conditions, i.e., the liner strains are converted to stress using Hooke's Law with the modulus of elasticity and Poisson's Ratio below yield. See CC-3630 of the ASME-ACI 359 document.

4. Liner plate seam offset: stresses due to maximum allowable seam offset, as stated in the construction specification, are considered in design.

5. Variation in anchor spacing: maximum range of anchor spacing, as allowed in the construction specification, are considered in design.

6. Variation in anchor stiffness due to a variation of the concrete modulus: this variation is considered by modifying the load versus displacement test data for the stud anchors and considering in the analysis a range of minimum and maximum possible values of concrete moduli.

7. Local concrete crushing in the anchor zone: such crushing is reflected in the anchor stud test data.

The liner plate is thickened at penetrations in accordance with the requirements of Subsection NE of the ASME B&PV Code, Section III.

3.8.1.4.7 Design of ASME B&PV Code, Section III, Division 1, Class MC Components

For the design of ASME B&PV Code, Section III, Division 1, Class MC Steel Components, such as the airlocks, equipment hatch, and portions of penetration sleeves subject to pressure induced stresses, see Subsection 3.8.2.

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Reinforcing Steel - Allowable Stresses and Strains

1. Based on factored load combinations, as described in Subsection 3.8.1.3.2, Item 2, the allowable average tensile stress in reinforcing steel at any section in the Containment is 90 percent of the yield stress (0.90 fy) in accordance with CC-3412 of the ASME-ACI 359 document. When considering load combinations which include a self-limiting thermal gradient, the maximum allowable strain in the reinforcing steel in one face may reach 1.5 times the yield strain, provided that the average stress in the reinforcing steel across the entire section does not exceed 0.9 fy. For additional discussion on this criterion, see Subsection 3.8.1.4.3, Item 4.

2. Based on service load combinations, as described in Subsection 3.8.1.3.2, Item 1, the allowable tensile stress in reinforcing steel is 50 percent of the yield stress (0.50 fy). However, this value is increased by 33 1/3 percent when considering load combinations which include any one (or more) of the following temporary loads:

    a. Temporary pressure loads during the test condition

    b. Temperature effects

This criterion is in accordance with CC-3422 of the ASME-ACI 359 document, except that the allowable stress may not be increased 33 1/3 percent for earthquake or wind loads.

3.8.1.5.2 Concrete - Allowable Stresses and Strains

1. Based on factored load combinations, as described in Subsection 3.8.1.3.2, Item 2, the allowable concrete stresses are in accordance with CC-3411 of the ASME-ACI 359 document, as follows:

    a. Membrane compression = 0.60 f'c

    b. Membrane plus bending compression = 0.75 f'c

    c. Local compression at discontinuities and at the inside face due to temperature gradients from accident conditions = 0.90 f'c.

    d. Concrete tensile strength is not relied upon to resist flexural and membrane tension.

    e. Radial shear is in accordance with CC-3411.4.1 of the ASME-ACI 359 document. (Where the calculated shear is greater than the allowable concrete shear, steel

shear reinforcement is provided in accordance with CC-3521 of the ASME-ACI 359 document.)

f. Tangential shear is in accordance with CC-3411.5.1 of the ASME-ACI 359, as follows, except that the maximum allowable tangential shear stress carried by the concrete, Vc, does not exceed 60 psi.

1. Provisions (a) and (d) of CC-3411.5.1 are satisfied, and this Containment does not support principal equipment laterally; therefore, the allowable tangential shear stress that can be resisted by the concrete, Vc, equals 60 psi.

2. Where the maximum tangential shear stress, Vu, exceeds the concrete allowable shear, Vc, inclined reinforcement is provided to resist the excess (Vu - Vc).

3. Design of the reinforcing steel for tangential shear is in accordance with CC-3521.1.1 of the ASME-ACI 359 document.

g. Requirements for punching shear are in accordance with Section 11.10.3 of the 1971 edition of the ACI 318 Code.

2. Based on service load combinations as described in Subsection 3.8.1.3.2, Item 1, the allowable concrete stresses are in accordance with CC-3421 of the ASME-ACI 359 document, as follows:

a. Membrane compression = 0.30 f'c

b. Membrane plus bending compression = 0.45 f'c

Note: Allowable stresses indicated in Subsection 3.8.1.5.2, Items 2.a and 2.b are increased by 33 1/3 percent when considering load combinations which include thermal. (The allowable increase when considering wind or earthquake, as permitted by the ASME-ACI 359 document, is not applied.)

c. Local compression at discontinuities and in the vicinity of liner anchors = 0.60 f'c

d. Concrete tensile strength is not relied upon to resist flexural and membrane tension.

e. Allowable stress in shear is 50 percent of the values indicated for factored loads, except that 67 percent of the factored load stresses are allowed for load combinations which include the pressure loads during the test condition. Additional requirements are in accordance with ASME-ACI 359 document, CC-3421.3 and CC-3522, except that the tangential shear stress carried by the concrete, Vc, does not exceed 40 psi.

3.8.1.5.3 Liners, Anchors, and Attachments - Allowable Stresses and Strains

1. The allowable stresses and strains in the liner plate are in accordance with Table CC-3700 1 of the ASME-ACI 359 document, as follows:

   a. Considering Calculated Membrane Stresses and Strains Only

      1. Construction category: tensile or compressive stress = 2/3 x yield stress

      2. Test category: compressive strain = .002 in./in.; tensile strain = .001 in./in.

      3. Normal category: same as test category

      4. Severe environmental category: same as test category

      5. Extreme environmental category: same as test category

      6. Abnormal category: compressive strain = .005 in./in.; tensile strain = .003 in./in.

      7. Abnormal-severe environmental category: same as abnormal category

      8. Abnormal-extreme environmental category: same as abnormal category

   b. Considering Combined Membrane and Bending Stresses and Strains

      1. Construction category: tensile and compressive stress = 2/3 x yield stress

      2. Test category: compressive strain = .004 in./in.; tensile strain = .002 in./in.

      3. Normal category: same as test category

      4. Severe environmental category: same as test category

      5. Extreme environmental category: same as test category

      6. Abnormal category: compressive strain = .014 in./in.; tensile strain = .010 in./in.

      7. Abnormal-severe environmental category: same as abnormal category

      8. Abnormal-extreme environmental category: same as abnormal category

   c. The load categories indicated previously include loads as defined in Subsection 3.8.1.3.2, except that load factors for all load cases are equal to 1.0.

2. The allowable forces and displacements of the liner anchors are in accordance with Table CC-3700-2 of the ASME-ACI 359 document, as follows:

   a. Considering Mechanical Loads Only

      1. Test category: applied load equals the lesser of 0.67 x yield load or 0.33 x ultimate load

      2. Normal category: same as test category

      3. Severe environmental category: same as test category

      4. Extreme environmental category: same as test category

      5. Abnormal category: applied load equals the lesser of 0.9 x yield load or 0.50 x ultimate load

      6. Abnormal-severe environmental category: same as abnormal category

      7. Abnormal-extreme environmental category: same as abnormal category

   b. Considering Displacement Limited Loads

      1. Test category: displacement equals 0.25 x ultimate displacement

      2. Normal category: same as test category

      3. Severe environmental category: same as test category

      4. Extreme environmental category: same as test category

      5. Abnormal category: displacement equals 0.50 x ultimate displacement

      6. Abnormal-severe environmental category: same as abnormal category

      7. Abnormal-extreme environmental category: same as abnormal category

   c. The load categories previously indicated include loads as defined in Subsection 3.8.1.3.2, except that load factors for all cases are equal to 1.0. Mechanical loads are defined as those which are not self-limiting or self-relieving with load application. Displacement limited loads are those resulting from constraint of the structure or constraint of adjacent material and are self-limiting or self-relieving. The yield and ultimate capacity of the liner stud anchors are based on the results of tests performed by the Nelson Stud Welding Company.

3. Allowable stresses and strains in penetration assemblies are in accordance with CC-3740 of the ASME-ACI 359 document. The design allowables for the penetration nozzles are the same as those used for metal containments (ASME B&PV Code, Section III, Division I), as discussed in Subsection 3.8.2. For additional criteria for ASME B&PV

Code, Section III, Class MC steel components such as the airlocks and the equipment hatch, see Subsection 3.8.2.

4. The design allowables for overlay plates, brackets, and attachments are the same as those given in AISC Specification for resisting mechanical loads in the construction, test, and normal categories. For all other categories indicated in Subsection 3.8.1.3.2, the allowable can be increased by a factor of 1.5.

   For overlay plates, brackets, and attachments which resist external mechanical loads and are not continuous through the liner plate, the allowable strength of the liner plate in the through-the-thickness direction will be taken as one-half of that in the transverse direction.

5. The liner is investigated for fatigue using the methods and limits established by ASME B&PV Code, Section III, Division I, Subsection NE.

3.8.1.5.4 Effect of Two- and Three-Dimensional Stress Strain Fields on the Behavior of the Structure

1. Liner

   The computed stresses and strains in the liner consider the effect of the two dimensional stress-strain field by the use of Poisson's Ratio in stress and strain determination. The liner anchor design considers biaxial liner loading by not relying on liner plate yielding to limit the forces applied to the anchor; liner strains are converted to stress and membrane forces assuming the material remains elastic. Because of the use of this conservative design approach, biaxial yield test values are not required.

2. Major Penetrations

   The analysis performed at the major penetrations, as described in Subsection 3.8.1.4.2, considers the three dimensional stress-strain field. Allowable stresses are the same as indicated in Subsections 3.8.1.5.1, 3.8.1.5.2, and 3.8.1.5.3.

3. Reinforced Concrete Section

   Based on the conservative assumption of fully cracked concrete, in both directions, used in the design of the reinforcing steel, the effect of a two- or three-dimensional stress-strain field is not a consideration in the design of the typical reinforced concrete section in the wall and dome of the Containment. However, the effect of the two-dimensional-stress strain field in the liner, in calculating the hot liner thrust against the reinforced concrete section, is considered.

3.8.1.5.5 Effects of Repeated Reactor Shutdowns and Startup During the Life of the Plant

1. The number of assumed reactor shutdowns and startups during the life of the plant is assumed to be 200 cycles over a period of 40 years.

2. The cycled stresses and strains in the reinforced concrete sections caused by reactor shutdowns and startups are minor compared to the stresses caused by the critical design

loading based on the abnormal (accident pressure and temperature) and extreme environmental (SSE) conditions. Therefore, the cycled stresses and strains caused by reactor shutdowns and startups do not degrade the margin of safety in the reinforced concrete.

3. The effect of cycled stresses and strains in the liner is considered by performing a fatigue analysis, in accordance with Subsection 3.8.1.5.3, Item 5, which includes the reactor shutdown and startup cycles.

3.8.1.5.6 Connections and Joints

The connections and joints of the various elements of the Containment, such as the crane girder bracket, liner end anchorage at bottom of wall, and similar elements, are designed using the same appropriate stress and strain allowables described elsewhere in Subsection 3.8.1.5.

3.8.1.5.7 Conditions at End of Service Life of Structure

Since a conventionally reinforced concrete containment is essentially a passive structure, as compared to a prestressed structure which relies on active prestress forces to meet its design function requirements, the margins of safety against all loading conditions are essentially the same throughout the life of this structure.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

3.8.1.6.1 Concrete

1. Materials

a. Cement

Cement is in conformance with the requirements of ASTM C-150-74, Specification for Portland Cement, Type II.

b. Aggregates

Aggregates are in conformance with the requirements of ASTM C 33-74, Specification for Concrete Aggregates, with the following additional requirements:

1. Gradations 357 or 467 are not furnished as one graded aggregate, but are obtained by combining at least two separate gradation sizes.

2. The potential reactivity of the aggregate is established by the methods described in the Appendix to ASTM C 33-74.

3. Aggregate shapes and sizes are in accordance with CC-2222.1.1 of the ASME-ACI 359 document.

c. Mixing Water

Mixing water is clean and free from injurious amounts of oils, acids, alkalis, salts, and organic materials, or other substances which could be deleterious to concrete or steel. Quality control tests are in accordance with the requirements of CC-2223 of the ASME-ACI 359 document.

d. Admixtures

1. Air-entraining admixtures conform to the requirements of ASTM C 260-74, Specification for Air-Entraining Admixtures for Concrete.

2. Chemical admixtures conform to the requirements of ASTM C 494-71, Specification for Chemical Admixtures for Concrete

2. Concrete Strength

Concrete has a minimum compressive strength of 4000 psi, in 28 days, when tested in accordance with ASTM C 39-72.

3. Other Concrete Properties

The following concrete properties are determined in accordance with the noted ASTM standards:

ASTM C 78-64 for flexural strength

ASTM C 496-71 for splitting tensile strength

ASTM C 469-65 for static modulus of elasticity

ASTM C 642-69T for specific gravity of concrete

4. Selection of Concrete Mix Proportions

Concrete mix proportions are established on the basis of laboratory trial batches, in accordance with the requirements of CC-2232 of the ASME-ACI 359 document. The following industry standards are referred to in this document:

ACI 211.1-74	Recommended Practice for Selecting Proportions for Normal and Heavy Weight Concrete
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ASTM C 39-72	Test for Compressive Strength of Cylindrical Concrete Specimens
ASTM C 192-69	Making and Curing Concrete Test Specimens in the Laboratory

5. Construction of Concrete

Concrete construction, including the stockpiling, batching, mixing, conveying, depositing, consolidation, curing, and construction joint preparation, is in accordance with CC-4200 of the ASME-ACI 359 document. The following industry standards are referred to in this document:

ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ASTM C 94-74	Specification for Ready-Mixed Concrete
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 305-72	Recommended Practice for Hot Weather Concreting
ACI 306-66	Recommended Practice for Cold Weather Concreting

6. Examination, Testing and Other Quality Control Procedures for Concrete

a. Quality Assurance

The QA procedures are in accordance with the requirements, in general, as described throughout the ASME-ACI 359 document.

The quality control (QC) for concrete begins with the selection and testing of the ingredients of the mix and extends through proportioning, batching, mixing, transporting, placing, and curing.

b. Testing and Examination of Concrete Ingredients

1. Cement

QC testing is in accordance with ASTM C 109-73, Test for Compressive Strength of Hydraulic Cement Mortars. Other QA requirements, including testing frequency, are in accordance with CC-5221 of the ASME-ACI 359 document.

2. Aggregates

QC testing is in accordance with CC-5224 of the ASME-ACI 359 document with the exception that the frequency of tests in accordance with ASTM C289 can be one test for each two month period provided low alkali cement is used. The following industry standards are called for in testing of aggregates to ascertain conformance to ASTM C 33-74, Specification for Concrete Aggregates:

ASTM C 131-69	Test for Resistance to Abrasion of 11 Size Coarse Aggregate by Use of the Los Angeles Machine

ASTM C 142-71	Test for Clay Lumps and Friable Particles in Aggregates
ASTM C 117-69	Test for Materials Finer Than No. 200 Sieve in Mineral Aggregates by Washing
ASTM C 87-69	Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
ASTM C 40-73	Test for Organic Impurities in Sands for Concrete
ASTM C 289-71	Test for Potential Reactivity of Aggregates (Chemical Method)
ASTM C 136-71	Test for Sieve or Screen Analysis of Fine and Coarse Aggregates
ASTM C 88-73	Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C 127-73	Test for Specific Gravity and Absorption of Coarse Aggregate
ASTM C 295-65	Recommended Practice for Petrographic Examination of Aggregates for Concrete

3. Mixing Water

Quality control testing requirements, including testing frequency, are in accordance with CC-5225 of the ASME-ACI 359 document.

c. Testing and Examination of Concrete

Concrete is tested in accordance with CC-5230 of the ASME-ACI 359 document. The following is a summary of the major testing requirements, as stated in this document:

1. Slump Test

Testing is performed in accordance with ASTM C 172-71, Sampling Fresh Concrete, and ASTM C 143-74, Test for Slump of Portland Cement Concrete, CC-5232 of the ASME-ACI 359 document.

2. Air Content

Testing is performed in accordance with ASTM C 231-74, Test for Air Content of Freshly Mixed Concrete by the Pressure Method, CC-5233 of the ASME-ACI 359 document.

3. Mechanical Properties

Testing is performed in accordance with ASTM C 39-72, Test for Compressive Strength of Cylindrical Concrete Specimens, and ASTM C 31-69, Making and Curing Concrete Test Cylindrical Specimens in the Field. The samples for strength tests are taken in accordance with ASTM C 172-71, Sampling Fresh Concrete. Testing frequency, and the acceptance criteria, are in accordance with CC-5234.2 of the ASME-ACI 359 document.

d. Certification, Marking, and Identification of Materials

Certified materials test reports are prepared in accordance with the requirements of CC-2130 of the ASME-ACI 359 document. Marking and identification of materials are in accordance with CC-2250 of the ASME-ACI 359 document.

e. Certification and Tests and Examinations

Certification of tests and examinations is provided in accordance with the requirements of CC-4120 of the ASME-ACI 359 document.

3.8.1.6.2 Reinforcing Steel

1. Material Specification

Reinforcing steel conforms to the requirements of ASTM A 615-72 Grade 60 or ASTM A706 (Unit 1).

2. Physical Properties

The specified minimum yield strength is 60,000 psi, and the specified minimum ultimate strength is 90,000 psi. The minimum elongation is 7 percent in 8 inches.

3. Chemical Properties

Some arc-welding is performed on the reinforcing steel, such as the welded shear bars at the Containment wall (See Figure 3.8-10) and restored reinforcing steel for the Containment Alternate Access (Unit 1); the reinforcing steel which is welded has a limited chemistry as defined in CC-2333 of the ASME-ACI 359 document, or carbon equivalency determined and reinforcing steel welded in accordance with the requirements of AWS D1.4-1998 (Unit 1).

4. Fabrication and Installation of Reinforcing Steel

Fabrication and installation of reinforcing steel are in accordance with CC-4300 of the ASME-ACI 359 document.

5. Examination and Testing of Reinforcing Steel

Special materials testing is performed in accordance with CC-2330 of the ASME-ACI 359 document. This section requires that one full diameter tensile test bar from each bar size be tested for each 50 tons, or fraction thereof, of reinforcing bars produced from each heat of steel. Bend tests also are performed in accordance with CC-2332 of the ASME-ACI 359 document. This section requires that, for No. 14 and No. 18 bars, bend testing of bars be conducted at the rate of one test bar for each bar size from each heat. NRC Regulatory Guide 1.15 is also complied with.

6. Certification, Marking, and Identification of Materials

Certified materials test reports are furnished in accordance with the requirements of CC-2130 of the ASME-ACI 359 document. Marking and identification of reinforcing steel are in accordance with the requirements of CC-2320 and CC-4122 of the ASME-ACI 359 document.

7. Certification of Tests and Examinations

Certification of tests and examinations is provided in accordance with CC-4120 of the ASME-ACI 359 document.

3.8.1.6.3 Mechanical Butt Splices (Cadwelds)

No. 14 and No. 18 reinforcing bars are spliced by use of Cadweld connections, as described in CC-4333 of the ASME-ACI 359 document. Such splices develop the tensile limits shown in Table CC-4330 of that document. Crew qualification is in accordance with CC-5321. Nondestructive testing is in accordance with CC-5322. Tensile testing and test frequency conform to the applicable requirements CC-5323 and CC-5324. The procedure for substandard test results conforms to CC-5325. Certified mill test reports are furnished in accordance with CC-2131. Marking and identification conform to CC-4122. Tests and examinations are certified in accordance with CC-4120. The requirements of the NRC Regulatory Guide 1.10 are also complied with.

3.8.1.6.4 Welding to Reinforcing Bars

Limited welding to reinforcing bars, such as required for welding shear bars to vertical reinforcing at the base of the Containment wall and restoration of the Containment Alternate Access (Unit 1), conforms to the requirements of CC-4334 of the ASME-ACI 359 document or chemistry composition determined to ensure proper welding (Unit 1).

3.8.1.6.5 Liner and Attachments

1. Materials

a. Liner, including thickened plates at penetrations and crane girder brackets, is in accordance with SA 537-74 Class 2.

b. Penetration sleeves are in accordance with SA 333-74 Grade 6 (seamless), SA 537-74 Class 2, or SA 516-74 Grade 70.

c. Stud anchors are in accordance with ASTM A 108-73.

d. Embedded steel members in mat are in accordance with ASTM A 36-74.

e. Penetration caps are in accordance with SA 105-74, SA 350-74 Grade LF1 or LF2, SA 516-74 Grade 60, 65, or 70 or SA 333-74 Grade 6.

f. Forgings, including penetration forgings, are in accordance with SA 350-74 Grade LF1 or LF2 or SA 182-74 Type F316.

g. Penetration anchorage studs and lugs are in accordance with 108-73 Grades 1015, 1016, or 1018 or ASTM A36-74 or SA 516-74 Grades 60, 65, or 70 or SA 537-74 Class 2.

h. Welding materials are in accordance with CC-2600 of the ASME-ACI 359 document, with the exceptions as specified in Section 3.8.1.2.5.2.a.

2. Special Materials Testing and Examination

a. Notch Toughness Testing

Notch toughness testing is performed on liner materials in accordance with the requirements of CC-2520 of the ASME-ACI 359 document.

b. Ultrasonic Testing

Liner plate materials which must transmit orthogonal loads in the through-the-thickness direction, such as the thickened liner plate at the crane girder brackets, are examined by the straight-beam ultrasonic method in accordance with ASTM A 578-71b. This examination, the acceptance standards, and related procedures are in accordance with CC-2530 of the ASME-ACI 359 document.

Liner plate in the area of overlay plates and/or structural attachments are not ultrasonic tested. The liner plate design, discussed in Subsection 3.8.1.5.3.4, will consider through-the-thickness properties.

c. Fabrication, Installation, and Welding of Liner

The following is a list of the major portions of CC-4500 of the ASME-ACI 359 document, covering the requirements for fabrication, installation, and welding of the liner, with the exceptions as specified in Section 3.8.1.2.5.2.a.

CC-4520	Forming, Fitting, and Aligning Exemptions and clarifications are discussed in Subsection 3.8.1.2.5.2.a.
CC-4522	Forming Tolerances
CC-4530	Welding Qualifications

CC-4540	Rules Governing Making, Examining, and Repairing Welds Clarifications are described in 3.8.1.2.5.2.a
CC-4550	Heat Treatment Clarifications are described in 3.8.1.2.5.2.a

d. Examination of Liners and Attachments

Examination of liners and attachments are in accordance with CC-5500 of the ASME-ACI 359 document as described in 3.8.1.2.5.2.a. Leak chase systems are installed at inaccessible welds. NRC Regulatory Guide 1.19 is also complied with.

e. Certification, Marking, and Identification of Materials

Certified materials test reports are prepared in accordance with the requirements of CC-2130 of the ASME-ACI 359 document. Marking and identification of liner materials are in accordance with the requirements of CC-2541 of the ASME-ACI 359 document.

f. Certification of Tests and Examinations

Certification of tests and examinations are provided in accordance with the requirements of CC-4120 of the ASME-ACI 359 document.

g. Protective Coatings

Suitable protective coatings are applied to the interior surfaces of the Containment liner as described in 6.1B.2.

3.8.1.6.6 Personnel Airlock, Equipment Hatch, and Emergency Airlock

1. Materials

See Subsection 3.8.2.6.

2. Code Requirements

The personnel airlock, equipment hatch, and emergency airlock satisfy all the requirements (materials, fabricating, welding, examination, testing, and other requirements) of the ASME B&PV Code, Section III, Division 1, Subsection NE, Class MC components. (See Subsection 3.8.2.) The personnel and emergency airlocks meet the requirements to obtain an N-stamp.

3. Outage Equipment Hatch

During Mode 5, 6 or defueled, an outage equipment hatch may be installed to allow access to containment. The outage equipment hatch is designed to provide a fission

product barrier in the event of a design basis Fuel Handling accident, and provides the same level of protection as the permanent inner equipment hatch in Mode 5, 6 or defueled. The outage equipment hatch will not replace the permanent inner equipment hatch during reduced inventory operation.

3.8.1.6.7 Quality Assurance Program

The documentation and maintenance of a QA program in the construction of the Containment are in accordance with CC-2700 of the ASME-ACI 359 document and in accordance with Chapter 17 of this FSAR. NRC Regulatory Guide No. 1.28 is also complied with.

3.8.1.7 Testing and Inservice Inspection Requirements

3.8.1.7.1 Structural Acceptance Test of Containment

The completed Containment structure is subjected to an acceptance test by which the internal pressure is increased from atmospheric pressure to a value of 1.15 times the Containment design pressure (1.15 x 50 psig = 57.5 psig) in five or more approximately equal pressure increments. The Containment is depressurized in the same number of increments. The purpose of the structural acceptance test is to demonstrate that the structure responds satisfactorily to the required internal pressure loads by making measurements of deflections and deformations under load to provide correlation with the theoretically predicted response.

Since this Containment is not a prototype, the measurements are limited to gross deformations and crack mapping. Strain measurements are not taken. The prototype containment on which strain measurements are correlated with deflection measurements is Consolidated Edison's Indian Point No. 2, NRC Docket No. 50247-47.

To the extent feasible, the test is conducted during a period of stable ambient temperature, atmospheric pressure, and humidity. Inside and outside of the Containment, atmospheric temperature, pressure, and humidity are monitored continuously during each test. The test will not be conducted under extreme weather conditions such as snow, heavy rain, or strong wind.

Deflection measurements are recorded at atmospheric pressure and at each level of the pressurization and depressurization cycle. At each level, the pressure is held constant for at least one hour before the deflections are recorded. Radial displacements of the containment cylinder are measured at five approximately equally spaced elevations between the base slab and dome springline and at the dome springline. These measurements are made along four azimuths spaced approximately equally around the containment. Radial deflections of the containment wall adjacent to the largest opening (the equipment hatch) are measured at 12 points as described in paragraph CC-6232(b) of the ASME B&PV Code, Section III, Division 2 (1980 Edition). The increase in diameter of the opening is measured in two mutually perpendicular directions. Vertical displacements of the cylinder at the top relative to the base are measured along four azimuths as described above. Vertical deflections of the containment dome at the apex and at two other equally spaced intermediate points, between a point near the apex and the springline, are measured along one azimuth. A listing of the numerical values, in the form of acceptance criteria for the measurements that are taken during the structural acceptance test, are evaluated in accordance with the ASME B&PV Code referenced below.

Crack patterns that exceed 0.01 inches in width before, during or after the test are mapped at locations described in paragraph CC-6233 of the ASME B&PV Code, Section III, Division 2 (1980 Edition).

The analysis of data and preparation of a test results report will be in accordance with paragraph CC-6260 of the ASME B&PV Code, Section III, Division 2 (1980 Edition).

All aspects of the structural integrity test, including the acceptance criteria are in accordance with paragraph CC-6000 of the ASME B&PV Code, Section III, Division 2, 1980 Edition with Summer 1980 Addenda (except that paragraph CC-6212.1, CC-6234 and CC-6236 do not apply since the CPNPP Containment is not a prototype containment).

3.8.1.7.2 Initial Leakage Rate Tests

Initial Containment leakage testing was in accordance with all the requirements of 10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors (2/5/73).

A preoperational Type A integrated leakage-rate test was performed on both units at the calculated peak Containment internal pressure. In addition, a reduced pressure test of not less than 50% of the peak pressure was performed for Unit 1. Type B tests of components and Type C tests of Containment isolation valves were performed in accordance with 10 CFR Part 50, Appendix J. For calculated peak containment internal pressure, see Section 6.2.1.

The maximum allowable leakage-rate (La as defined in 10 CFR Part 50, Appendix J), related to the maximum Containment leakage under design basis pressurization accident conditions, is 0.10 percent of the weight of contained air, at the calculated peak Containment internal pressure per 24 hour period.

For a discussion of the test objectives and the acceptance criteria, see the Technical Specifications for Containment Tests.

Initial test methods were in accordance with ANSI N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors (03/26/73), with the exceptions of the calculation of leakage rates using the mass-point method of ANSI/ANS 56.8-1987 and the isolation of penetrations, as stated in Table 14.2-2 (Sheet 59).

3.8.1.7.3 Inservice Inspection Program

The inservice inspection program conforms to the requirements of 10 CFR Part 50, Appendix J, Option B. Periodic testing, test objectives, and acceptance criteria are in accordance with 10CFR50 Appendix J, Option B.

3.8.2 STEEL CONTAINMENT

This section, as outlined in the NRC Regulatory Guide 1.70, Rev. 2., Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, regarding a steel containment, is not applicable to the Comanche Peak Containment structure because a steel-lined reinforced concrete containment is being used, as described in Subsection 3.8.1. Certain steel components in the Containment system are classified in accordance with ASME B&PV Code, Section III, as

Class MC components, as described in Subsection 3.8.1. These are the personnel airlock, the equipment hatch, the emergency airlock, and other penetrations subject to pressure-induced stresses.

This section addresses itself to the requirements of these ASME B&PV Code, Section III, Class MC steel components.

3.8.2.1 Description of the Containment

For the description of the components (airlocks, equipment hatch, and other penetrations), see Subsection 3.8.1.1.6. The Containment itself is a steel-lined reinforced concrete containment.

3.8.2.2 Applicable Codes, Standards, and Specifications

3.8.2.2.1 Basic Code

The basic code for these steel components is the ASME B&PV Code, Section III, Division 1, Subsection NE, for Class MC components. This code is applicable for all the requirements of the components, including materials, design, fabrication, examination, testing, and so forth, except the equipment hatch and the sleeves for the electrical penetrations, fuel transfer tube, and process piping penetrations are not code stamped and are pressure tested as part of the reinforced concrete containment structure as discussed in Section 3.8.1.7. The 1971 edition of the code, through and including the summer 1973 addenda, is used for the electrical penetration sleeves, fuel transfer tube penetration sleeve, emergency and personnel airlocks, and equipment hatch. The 1974 edition through and including the Summer 1976 Addenda is used for the process piping penetrations.

3.8.2.2.2 Other Applicable Codes, Specifications, and Standards

See Section 3.8.1.2.3 and 3.8.2.6.6 of this FSAR.

3.8.2.2.3 Applicable NRC Regulatory Guide

NRC Regulatory Guide 1.57 Design Limits and loading Combinations for Metal Primary Reactor Containment System Components (6-73) (Applicable only to appropriate containment components constructed in accordance with Subsection NE of the ASME B&PV Code, Section III, Class MC.)

3.8.2.3 Loads and Load Combinations

The applicable loads stated in Subsection 3.8.1.3 are considered in the design of the ASME B&PV Code, Section III, Class MC steel components. For the various load combinations used in the design of Class M components and the related allowable stresses and strains, see Subsection 3.8.2.5.

3.8.2.4 Design and Analysis Procedures

The design and analysis of the Class MC components are in accordance with all the requirements of Subsection NE of the ASME B&PV Code, Section III, including the applicable

portions of Appendix A. Analysis procedures also use published formulas such as Roark's Formulas for Stress and Strain and Timoshenko's Theory of Plates and Shells.

3.8.2.5 Structural Acceptance Criteria

3.8.2.5.1 General Criteria

The design is such that all the stress and strain limits, as defined in NE-3000 of the ASME B&PV Code, Section III, are satisfied for pressure loads in combination with all mechanical loads and thermal loads, as discussed in Subsection 3.8.2.5.2.

3.8.2.5.2 Design Load Combination Stress Limits

1. The ASME B&PV Code, Section III, design criteria for Class MC components are based on establishing stress and strain limits which vary depending on the following factors:

   a. Types of Stress

      As defined by ASME B & PV Code Section III:

      P_m - General Primary Membrane Stress

      P_l - Local Primary Membrane Stress

      P_b - Primary Bending Stress

      P_e - Secondary Expansion Stress

      Q - Secondary Membrane plus Bending Stress

      F - Peak Stress

      S_y - Yield Strength

      S_u - Tensile Strength

      S_m - Design Stress Intensity Value

   b. Types of Loads

      The loads considered in the load combinations in 3.8.2.5.2.2 are as defined in Section 3.8.1.3.1 and as follows:

      P_e - Design External Pressure

      T_e - Thermal loads under thermal conditions during event causing external pressure

      R_e - Pipe reaction under thermal conditions during event causing external pressure

2. Load Combinations and Acceptance Criteria

LOAD COMBINATION	ACCEPTANCE CRITERIA
$D+L+1.15\ P_a+T_t$	$P_m < 0.9\ S_y$
	P_l or $P_l + P_b \leq 1.25\ S_y$
	$P_l+P_b+P_e+Q \leq 3\ S_m$
	$P_l+P_b+P_e+Q+F$ Fatigue Analysis
$D+L+T_o+R_o$	$P_m \leq S_m$
$D+L+T_o+R_o+E$	P_l or $P_l+Pb \leq 1.5\ S_m$
	$P_l+P_b+P_e+Q \leq 3\ S_m$
	$P_l+P_b+P+Q+F$ Fatigue Analysis
$D+L+T_a+R_a+P_a+E$	$P_m \leq S_m$
$D+L+T_e+R_e+P_e+E$	P_l or $P_l + P_b \leq 1.5\ S_m$
$D+L+T_a+R_a+P_a+E$	$P_m \leq$ Larger of $1.2\ S_m$ or S_y
$D+L+T_e+R_e+P_e+E$	P_l or $P_l + P_b \leq$ Larger of
	$1.8\ S_m$ or $1.5\ S_y$
$D+L+T_a+R_a+P_a+Y_r+Y_j+Y_m+E$	$P_m \leq 0.85$ times smaller
	of $0.7\ S_u$
	or $\left(S_y + \frac{S_u - S_y}{3}\right)$
	P_l or $P_l+P_b \leq (1.5)(0.85)$
	times smaller of $0.7\ S_u$ or
	$\left(S_y + \frac{S_u - S_y}{3}\right)$

3. Compressive Stresses

In areas of compressive stress, buckling criteria are considered in accordance with the applicable sections of NE-3000 of the ASME B&PV Code, Section III.

4. Fatigue Analysis

   The requirements for an analysis for cyclic operation is investigated in accordance with NE-3131 (d) and the referenced portions therein.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The Containment is a steel-lined reinforced concrete structure. The airlocks, equipment hatch, and other penetrations are fabricated from the following materials in accordance with Section III of ASME-ACI 359, Proposed Standard Code for Concrete Reactor Vessels and Containments (April 1973):

1. Plate is in accordance with ASME SA-537 Class 2 OR SA-616 Grade 70, or equivalent.

2. Penetrations are in accordance with ASME SA-333 Grade 6, SA-537 Class 2, or SA-516 Grade 70, or equivalent.

   Penetration caps are in accordance with ASME SA-105, SA-350 Grade LF1 or LF2, SA-516 Grade 60,65 or 70 or SA-333 Grade 6, or equivalent.

3. Forgings including penetration forgings are in accordance with ASME SA-350 Grade LF1 or LF2 or SA-182 Type F316, or equivalent.

4. Bolting is in accordance with ASME SA-320, or equivalent.

   The materials meet all the requirements in the ASME B&PV Code, Section III, Division 1, Subsection NE, for Class MC components, including the Charpy impact test requirements.

   The QA program for fabrication and erection is in accordance with the requirements of Subsection NE of the ASME B&PV Code, Section III, Division 1.

   The full penetration butt weld between the sleeve and weld neck flange of the electrical penetration assemblies may be examined by the ultrasonic examination method plus the magnetic particle or liquid penetrant method per the special exceptions of NE-5231.1 in lieu of the radiographic examination method of NE-5211.

3.8.2.7 Testing and Inservice Inspection Requirements

Testing of Class MC components is described in Section 6.2.6. Overall Containment testing is in accordance with Subsection 3.8.1.7.

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

3.8.3.1 Description of the Internal Structures

3.8.3.1.1 General Description

The Containment internal structures are primarily of reinforced concrete and consist of the following major elements:

1. Primary Shield Wall (Reactor Cavity)

   The primary shield wall, a heavily reinforced concrete cylinder, is situated at the approximate center of the Containment vessel, and extends up from the interior base slab to surround the reactor vessel. This reactor cavity structure provides support for the reactor vessel. The vessel supports consist of support pads and shoes which are mounted on support members within the concrete cavity structure. During normal operation, the primary shield wall provides biological shielding for maintenance inspection. Under seismic loading, this structure serves to provide seismic shear resistance and stiffens the Containment internal structure.

2. Primary Loop Compartment Walls (Steam Generator Compartment)

   The compartments are formed by the secondary shielding walls on the exterior and by the reactor and refueling canal walls on the interior. These walls extend from the interior base slab up to the operating floor. The compartment houses the steam generator, reactor coolant pumps, and the RCLs. The compartment walls provide radiation shielding, isolation of the RCS, and lateral restraint for the steam generator, pump, and pressurizer.

3. Operating Floor

   The operating floor is supported by the primary loop compartment walls and concrete columns adjacent to the containment shell which extend down to the interior base slab. The operating floor provides a working and access floor during refueling, maintenance, and repair operations. Vent areas are provided where required.

4. Refueling Cavity

   The refueling cavity provides shielded access for transport of spent fuel and new fuel between the reactor vessel and the fuel transfer penetration. It also provides shielding storage space for the reactor vessel internals during refueling or maintenance. The cavity is lined with stainless steel.

5. Interior Base Slab

   The interior base slab is placed on top of the foundation mat liner plate. This slab provides lateral and flexural restraint at the base of the primary loop compartment walls and the primary shield wall. The slab ties the primary loop compartment walls to the primary shield walls and provides a diaphragm for seismic shear distribution at the bottom

of the internal structure. It also protects the foundation mat liner from any missiles generated in the primary loop compartments and from the effects of accident temperatures.

6. Missile Shields

The primary loop compartment walls and the operating deck provide missile protection for the RCS from potential missiles outside the primary loop compartment. Conversely, the walls and interior base slab provide missile protection to the Containment and to the safeguard and auxiliary systems located outside the primary loop compartment from postulated missiles originating inside the primary loop compartments. The missile-shielding function of the primary loop compartment walls is supplemented by a control rod drive missile shield positioned over the reactor vessel head, which is designed to contain any postulated ejected control rods from the reactor vessel. The Unit 1 missile shield is removed with the reactor vessel head assembly and the Unit 2 missile shield is removed during refueling operations.

7. Intermediate Floors

Intermediate floors are provided at several elevations, including a principal floor at elevation 860 ft for miscellaneous equipment supports, access, maintenance, and similar items.

8. Removable Slabs and Walls

Removable slabs and walls are provided, where required, for maintenance access, e.g., the missile shield over the reactor and in the walls around the regenerative heat exchanger and reactor coolant drain pumps.

3.8.3.1.2 Polar Crane and Telescopic Jib Crane

A Nuclear Safety Related polar crane is provided inside the Containment, on a circular steel runway girder which is supported by brackets from the cylindrical Containment wall. The polar crane will remain stable and not become derailed when subjected to the specified load combinations (See FSAR Sections 3.8.3.3.3 and 9.1.4).

The primary load path from the crane wheels to the runway girders is maintained within allowable stress limits.

A telescopic jib crane is provided inside Containment. It is supported by a structural steel support mounted on east-west divider wall between S.G. compartments at elevation 905'-9" (see Figures 1.2-8 and 1.2-15). Two such supports are provided-one each between S.G. compartments 1 & 3 and 2 & 4. The telescopic jib crane can be located as required at either of these two supports. This crane will only be used to handle loads during plant shutdown (e.g., MODES 5, 6, and defueled). The telescopic jib crane and it's support are Non-Nuclear Safety Related, Seismic Category II.

Both the Polar Crane and Telescopic Jib Crane operating at the same time constitutes Multi-crane Operation which requires special administrative controls to reduce the likelihood of

unplanned crane interactions. These administrative controls include a qualified multi-crane coordinator located in a position of good visibility to observe both cranes.

A Containment Anti-Collision Control System is also installed on the telescopic jib crane to reduce the likelihood of unplanned crane interactions with the polar crane bridge. The function of the Programmable Logic Controller (PLC) and associated components is to keep the jib crane boom below the Polar Crane bridge. In addition, an administratively controlled key, which is normally placed in the Polar Crane allows full operation from the polar crane radio controller. In this mode without the key, the PLC restricts the height of the Jib Crane Boom. When the key is placed in the Jib Crane controls, the boom height limit is bypassed; however the polar crane bridge cannot be controlled by the radio controls. Operation of the trolley and hoists from the polar crane radio controls and full operation from the polar crane cab is still available under the administrative controls for multi-crane operation.

The PLC is removed from containment during power operation to protect it from radiation.

3.8.3.1.3 Supports for Reactor Pressure Vessel, Steam Generator, Reactor Coolant Pump, Pressurizer, and Loop Piping

Descriptions of the supports for the reactor pressure vessel, steam generator, reactor coolant pump, pressurizer, and loop piping are presented in Section 5.4.14.

3.8.3.1.4 Drawings

For various details of internal structures, see Figure 3.8-15.

3.8.3.2 Applicable Codes, Standards, and Specifications

3.8.3.2.1 Basic Codes

Except where specifically noted otherwise, the basic codes used for the materials, design, construction, and so forth of the Containment internal structures are the applicable sections of the following:

1. Reinforced concrete is in accordance with ACI 318-71 Building Code Requirements for Reinforced Concrete.

2. Structural steel is in accordance with AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings (1969) including Supplement Numbers 1, 2, and 3 hereafter referred to as AISC Specification. Except:

   a. When supported by an engineering analysis, connections using A325 or A490 high strength bolts need not be pretensioned to the values required by AISC Specification, Table 1.23.5.

   b. Welded construction is in accordance with “Structural Welding Code, AWS D1.1-75” (except paragraph 5.23.2.4) instead of “AWS D1.1 - Rev. 1-73” as required by AISC, section 1.17.

Welder qualification for partial groove and fillet welds on tubular joints (T, K and Y connections) is in accordance with paragraph 5.23.2.4 and Table 10.5 Columns 9, 10 and 11 “Minimum Welder Qualifications for all Positions” of “Structural Welding Code, AWS D1.1-90”.

c. For prequalified joint details and base metals, “AWSD1.1 - Rev. 1-73” and later editions may be used instead of only “AWSD1.1-Rev. 1-73” as required by AISC, section 1.17.

Visual inspection of structural welds will be in accordance with Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants (VWAC) Revision 2 dated May 7, 1985. (SEE Appendix 3.8A)

The acceptance criteria contained in VWAC Revision 2 are applicable to all structural steel welds at CPNPP other than ASME-class structural welds.

3.8.3.2.2 Supplementary Codes

Where specifically referred to in this section on internal structures, certain portions of the ASME-ACI 359 document apply, as described in Subsection 3.8.1.2.1.

ACI 349-76, Appendix A, is used to determine stresses resulting from thermal gradients.

3.8.3.2.3 Additional Specifications and Standards

The following is a list of additional specifications and standards which are applicable to the internal structures:

1. Concrete

(Same as Subsection 3.8.1.2.3, Item 3.)

2. Steel

ASTM A 36-74	Specification for Structural Steel
ASTM A 307-74	Specification for Carbon Steel Externally and Internally Threaded Standard Fasteners
ASTM A 325-74	Specification for High-Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers
ASTM A 240-74a	Specification for Heat Resisting Chromium and Chromium Nickel Stainless Steel Plate, Sheet, and Strip for Fusion Welded Unfired Pressure Vessels
ASTM A 370-74	Methods and Definitions for Mechanical Testing of Steel Products
ASTM A 578-71b	Specification for Straight-Beam Ultrasonic Examination of Plain and Clad Steel Plates for Special Applications
ASTM A 6-74	Specification for General Requirements for Delivery of Rolled Steel Plates, Shapes, Sheet Piling, and Bars for Structural Use
RCRBSJ - 1974	Specification for Structural Joints Using ASTM A325 or A490 Bolts, AISC 1974 Edition. RCRBSJ - Research Council on Rivited and Bolted Structural Joints.

3. Reinforcing Steel

ASTM A 615-72	Specification for Deformed and Plain Billet Steel Bars for Concrete Reinforcement (Grade 60)

4. Polar Crane

Crane Manufacturer's Association of America, Inc. (CMAA) Specification No. 70

3.8.3.2.4 Summary of Principal Plant Specifications

The principal plant specifications for the internal structures are primarily those involving reinforced concrete construction. The applicable portions of the ASME-ACI 359 document, as described in Subsection 3.8.1.2.1, are included in the plant construction specifications in regard to materials, construction techniques, examination, quality control, and so forth. The following principal portions of the ASME-ACI 359 document are incorporated into the plant specifications:

1. Reinforcing Steel Specification Same as Subsection 3.8.1.2.5, Item 2.b.

2. Concrete Specifications

Same as Subsection 3.8.1.2.5, Item 2.d.

3. Cadweld Splices

Generally, large-size bars requiring Cadweld splices are not used in the internal concrete structures. Where such splices are used, the requirements indicated in Subsection 3.8.1.2.5, Item 2.c are complied with.

4. Structural Steel

The plant specification for structural steel includes the requirements in AISC Specification.

3.8.3.2.5 Applicable NRC Regulatory Guides

The following NRC Regulatory Guides are applicable to the internal structures and are complied with:

NRC Regulatory Guide 1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1, 12-28- 72 of former Safety Guide 15)
NRC Regulatory Guide 1.28	Quality Assurance Program Requirements (Design and Construction) (6-7-72 of former Safety Guide 28)

3.8.3.3 Loads and Load Combinations

3.8.3.3.1 Loads

The loads and load combinations for supports which are supplied by Westinghouse are provided in Section 3.9N.1.4. The following loads are considered in the design of the internal structures of the Containment:

1. Normal Loads

Normal loads are those loads which are encountered during normal plant operation and shutdown. They include the following:

a. D = dead loads, including any permanent equipment loads, and their related moments and forces

b. L = live loads, including any movable equipment loads and other loads which vary in intensity and occurrence such as soil and hydrostatic pressures, pressure differences caused by variation in heating, cooling, and their related moments and forces

c. To = thermal effects and loads during normal operating or shutdown conditions based on the most critical transient or steady-state condition

d. Ro = pipe reactions during normal operating or shutdown conditions based on the most critical transient or steady-state condition

2. Severe Environmental Loads

Severe environmental loads are those loads that could be encountered infrequently during the plant life. This category includes the following:

Feqo = loads generated by 1/2 the SSE

= OBE

3. Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but highly improbable. They include the following:

Feqs = loads generated by the SSE

4. Abnormal Loads

Abnormal loads are loads generated by a postulated high energy pipe break accident within the Containment or compartment thereof. This category includes the following:

a. Pa = pressure equivalent static load within or across a compartment generated by the postulated break, including an appropriate dynamic factor to account for the dynamic nature of the load

b. Ta = thermal loads under thermal conditions generated by the postulated break, including To

c. Ra = pipe reactions under thermal conditions generated by the postulated break, including Ro

d. Yr = equivalent static load on the structure generated by the reaction on the broken high energy pipe during the postulated break, including an appropriate dynamic factor to account for the dynamic nature of the load

e. Yj = jet impingement equivalent static load on the structure generated by the postulated break, including an appropriate dynamic factor to account for the dynamic nature of the load

f. Ym = missile impact equivalent static load on the structure generated by or during the postulated break, such as pipe whipping, including an appropriate dynamic factor to account for the dynamic nature of the load

In determining an appropriate equivalent static load for Yr, Yj, and Ym, elastoplastic behavior is assumed with appropriate ductility ratios as long as excessive deflections do not result in loss of function. For concrete structures, the ductility ratios are described in Section 3.5.3.2.

5. Other Definitions

a. For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in AISC Specification.

A 33-percent increase in allowable stresses for concrete and steel because of seismic or wind loadings is not permitted.

b. For concrete structures, U is the section strength required to resist design loads and is based on methods described in ACI 318-71.

c. For structural steel, Y is the section strength required to resist design loads and is based on plastic design methods described in Part 2 of AISC Specification.

3.8.3.3.2 Load Combinations and Acceptance Criteria for Internal Concrete Structures of the Containment

1. Load Combinations for Service Load Conditions

a. U = 1.4 D + 1.7 L

b. U = 1.4 D + 1.7 L + 1.9 Feqo

If thermal stresses due to To and Ro are present the following combinations also apply:

c. U = .75 (1.4 D + 1.7 L + 1.7 To + 1.7 Ro)

d. U = .75 (1.4 D + 1.7 L + 1.9 Feqo + 1.7 To + 1.7 Ro)

L is considered for its full value or its complete absence.

2. Load Combinations for Factored Load Conditions

For conditions that represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, respectively, the following load combinations are satisfied:

a. U = D + L + To + Ro + Feqs

b. U = D + L + Ta + Ra + 1.5 Pa

c. U = D + L + Ta + Ra + 1.25 Pa + 1.0 (Yr + Yj + Ym) + 1.25 Feqo

d. U = D + L + Ta + Ra + 1.0 Pa + 1.0 (Yr + Yj + Ym) + 1.0 Feqs

In combinations b, c, and d, the maximum values of Pa, Ta, Ra, Yj, Yr, and Ym, including an appropriate dynamic factor, are used unless a time history analysis is performed to justify otherwise. Combinations c and d and the corresponding structural acceptance criteria shall be first satisfied without Yr, Yj, and Ym. When considering these loads, local

section strength capacities may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

L is considered for its full value or its complete absence.

3.8.3.3.3 Load Combinations and Acceptance Criteria for Internal Steel Structures of the Containment

1. Load Combinations for Service Load Conditions

Either the elastic working stress design methods of Part 1 of AISC Specification or the plastic design methods of Part 2 of AISC Specification are used.

a. When the elastic working stress design methods are used, the following apply:

1. S = D + L

2. S = D + L + Feqo

When thermal stresses caused by To and Ro are present the following combinations are satisfied:

3. 1.5 S = D + L + To + Ro

4. 1.5 S = D + L + To + Ro + Feqo

L is considered for its full value or its complete absence.

b. When plastic design methods are used, the following apply:

1. Y = 1.7D + 1.7L

2. Y = 1.7D + 1.7L + 1.7 Feqo

When thermal stresses caused by To and Ro are present, the following combinations are satisfied:

3. Y = 1.3 D + 1.3 L + 1.3 To + 1.3 Ro

4. Y = 1.3 D + 1.3 L + 1.3 To + 1.3 Ro + 1.3 Feqo

L is considered for its full value or its complete absence.

2. Load Combinations for Factored Load Conditions

a. If elastic working stress design methods are used, the following load combinations are satisfied:

1. 1.6 S = D + L + To + Ro + Feqs

2. 1.6 S = D + L + Ta + Ra + Pa

3. 1.6 S = D + L + Ta + Ra + Pa + 1.0 (Yj + Yr + Ym) + Feqo

4. 1.7 S = D + L + Ta + Ra + Pa + 1.0 (Yj + Yr + Ym) + Feqs

b. If plastic design methods are used, the following load combinations are satisfied:

1. .90 Y = D + L + To + Ro + Feqs

2. .90 Y = D + L + Ta + Ra + 1.5 Pa

3. .90 Y = D + L + Ta + Ra + 1.25 Pa
+ 1.0 (Yj + Yr + Ym) + 1.25 Feqo

4. .90 Y = D + L + Ta + Ra + 1.0 Pa
+ 1.0 (Yj + Yr + Ym) + 1.0 Feqs

In these combinations, thermal loads are neglected when they are secondary and self-limiting in nature and when the material is ductile.

In combinations shown in Items 2.a.2), 3), and 4), and in Items 2.b.2) 3), and 4), the maximum values of Pa, Ta, Ra, Yj, Yr, and Ym, including an appropriate dynamic factor, are used unless a time history analysis is performed to justify otherwise.

In determining the equivalent static load for the differential pressure Pa, the impulsive nature of the load is taken into account by considering the time history of the applied pressure and the natural frequencies of the structures to which the pressure is applied (including the secondary shield walls and operating and intermediate floors). The steel is designed so that the maximum stress for any load combination, which includes differential pressure, is less than the yield stress, thus assuring elastic behavior.

For combinations shown in Subsection 3.8.3.3.3, Items 2.a.3) and 4), and in Subsection 3.8.3.3.3, Items 2.b.3) and 4), local stresses caused by the concentrated loads Yr, Yj, and Ym may exceed the allowables when there is no loss of function of any safety-related system. Furthermore, in computing the required section strength, the plastic section modulus of steel shapes is used.

3.8.3.3.4 Variable Loads

For loads which vary, the values (within the possible range) which produce the most critical combination of loading are used in design.

3.8.3.3.5 Interrelated Loads

For loads which are interrelated as a function of time, such as accident-induced pressure and jet and thermal effects, the maximum values of these effects do not necessarily occur

simultaneously. Recognition is given to the time increments associated with these postulated failure conditions.

3.8.3.3.6 Live Loads

The live loads used in design for each category of loading are the live loads consistent with the conditions assumed for that particular category of loading.

3.8.3.3.7 Load Combinations for Localized Areas

The design load combinations which are used to examine the effects on localized areas, such as loads transferred from support structures, are the same load combinations used for the general internal structure, as described above.

3.8.3.3.8 Time-Dependent Loads

Where significant, time-dependent loads such as the effects of creep, shrinkage, and other related effects are included with dead load effects as described in Section 9.3.7 of ACI 318-71.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 General Analysis of Internal Structure

The internal structure is analyzed and designed for all load combinations as described in Subsection 3.8.3.3. Table 3.8-1 shows critical loading combinations, type of stress and computed and corresponding allowable stresses at key location in the Containment internal structures.

1. General Analysis Procedure

   The Containment internal structure is analyzed by hand calculations and finite element methods of analysis using the ANSYS computer program (see Appendix 3.7B (A) for description of program), unless noted otherwise. The Unit 1 Steam Generator compartments are analyzed, using hand calculations and the STRUDL computer program (see Appendix 3.7B(A) for description of program.) The ANSYS program uses the stiffness method of structural analysis and contains plate elements and beam elements to represent membrane and bending behavior. The models used for the analysis included the interior base slab and the internal structure components described in Subsection 3.8.3.1, such as the primary shield wall, steam generator compartments, refueling canal walls, and intermediate floors.

   The input to these analyses consist of the geometry of the structure, the material properties, the appropriate boundary conditions, and the loadings.

   All loading combinations specified in Section 3.8.3.3.2 have been addressed in these analyses. Equivalent static loads were developed to include appropriate dynamic load factors to account for the dynamic nature of these loads (e.g., subcompartment pressurization due to postulated high-energy line pipe break accidents). Thermal effects and loadings during normal operating (To) and postulated high-energy line pipe break conditions were addressed.

The output of these analyses contain the displacements, rotations, forces, shears, moments, and stresses throughout the structure.

This output is used for the general design of the internal structure.

2. Geometry Check of Model

The finite element model of the structure consists of numerous nodes and elements. During the preparation of the finite element models, it is necessary to identify and eliminate any input errors in the coordinates of the nodes and the connectivity of the elements to the nodes. Computerized plotting is used in preparing plans, sections, and isometric views of the finite element model from which input errors are readily detected.

3. Check of Validity of Computer Program

The validity of the computer program used in the structural analysis is verified before the program is used. Appendix 3.7B(A) lists computer programs used in this project.

4. Check of Analysis Results

Results of the finite element analyses are checked before they are used for design. The checks include the following:

a. Force and moment equilibrium check.

b. Global equilibrium check.

3.8.3.4.2 Additional Analysis at Local Areas

Analysis is performed in local critical areas, such as wall openings, or areas subject to localized accident loads such as jet forces or missile forces.

3.8.3.4.3 Interaction with Nuclear Steam Supply System Equipment Supports

The seismic dynamic analysis of the major NSSS equipment supports (reactor vessel, steam generators, reactor coolant pumps, and pressurizer) takes into consideration the interaction between the equipment and supports and the internal structure. The internal structures supporting the equipment are designed for the resulting seismic loads transmitted by the equipment supports. Loads transmitted from the equipment supports to the internal structures as a result of a pipe rupture accident are also considered in the design.

3.8.3.4.4 Lateral Load Transfer at Foundation Mat

Lateral loads such as seismic forces or LOCA forces are transmitted down to the foundation mat, primarily by means of shear wall action, through the primary and secondary shield walls. At the base of these walls, these lateral loads are transferred to the interior base slab by shear keys. This interior base slab acts as a diaphragm, transferring the lateral load to the key formed by the reactor cavity pit, and from there to the foundation mat.

3.8.3.4.5 Design Variables

The general analysis performed according to Subsection 3.8.3.4.1 assumes a linear elastic response with uncracked concrete section properties. The local analysis of critical areas, described in Subsection 3.8.3.4.2, considers, for loads where significant cracking is probable, the effects of cracked section properties on the analysis. Properties of materials are known with sufficient accuracy, and assumptions made are sufficiently conservative, so that other variables need not be considered in design.

3.8.3.4.6 Temperature Effects Analysis

When subjected to service load combinations described in Subsection 3.8.3.3.2, stresses caused by thermal gradients are calculated in accordance with ACI 349-76, Appendix A and conform to allowable stresses as described in Subsection 3.8.3.5. For factored load combinations described in Subsection 3.8.3.3.2, subsection 2, thermal gradient effects are determined by analysis as recommended by ACI 349-76, Appendix A, the maximum strain in the reinforcing steel does not exceed 1.5 times the yield strain, as described in Subsection 3.8.1.4.3, Item 4.

3.8.3.4.7 Reinforced Concrete Design

1. Geometry of Reinforcing Steel

   In general, all walls and slabs are reinforced in two orthogonal directions at each face. Shear reinforcement is provided in accordance with ACI 318-71. Beams and girders are conventionally reinforced using top and bottom longitudinal bars and vertical stirrups. Most of the bars are No. 11 size or smaller, permitting the use of lapped splices. In some areas where large size No. 14 or No. 18 bars are required, Cadweld splices are used.

2. Proportioning of Reinforcing Steel

   The results of the analysis under all load combinations include moments, axial forces, and shears at each section of the walls, slabs, and beams. Sufficient reinforcing steel is provided to resist the moments, axial forces, and shears as required to satisfy the requirements of ACI 318-71.

3. Bond and Anchorage Requirements of Reinforcing Steel

   Chapter 12 of ACI 318-71, Development of Reinforcement, is complied with in determining bond and anchorage requirements or they are based on available test data.

3.8.3.4.8 Evaluation of Radiation-Generated Heat Effects

Concrete temperatures do not exceed the values indicated in the ASME- ACI 359 document, CC-3430(a) for long-term loading and CC-3430(b) for accident or other short-term loading. If required, insulation or cooling systems, or both, are provided to limit the temperature of the concrete to an acceptable level.

3.8.3.4.9 Structural Steel Design

The methods of designing structural steel components are in accordance with the AISC Specification, including design for bending moments, tension, compression, connections, buckling criteria, and so forth.

3.8.3.4.10 Polar Crane and Telescopic Jib Crane

The polar crane is designed in accordance with the criteria in CMAA Specification No. 70.

The telescopic jib crane is designed in accordance with requirements of API Spec. 2C, ASME NOG-1 and ASME NUM-1.

3.8.3.4.11 Structural Response of Missile Barriers

The methods for predicting overall structural response of the missile barriers and other seismic Category I structures, subject to missile impact, are described in Section 3.5.3.

3.8.3.4.12 Supports for Reactor Coolant System Components

The models and methods of analysis for the RCS component supports are presented in Section 3.9N.1.4.4.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Functional Criteria

The Containment internal structure is designed to satisfy the following functional requirements:

1. To support (under all load conditions) and provide access to the NSSS and auxiliary and safeguard systems located within the Containment

2. To provide shielded access for spent fuel and new fuel between the reactor vessel and the fuel transfer penetration

3. To provide isolation of the Containment liner from postulated missile effects resulting from the loads associated with rupture of the RCS

4. To provide isolation of the RCS from the effects of postulated failures in systems and components located inside the Containment, but outside the primary loop compartments

5. To provide biological shielding

3.8.3.5.2 Reinforced Concrete Acceptance Criteria

The design of the reinforced concrete structures is based on an ultimate strength concept in accordance with the requirements of ACI 318-71. In this ultimate strength method of design, the factor of safety is applied to the individual loads (as load factors) rather than to the allowable stresses. The ultimate load on the reinforced concrete member is based on the factored load combinations described in Subsection 3.8.3.3.2 (for both service load combinations and factored

load combinations). The ultimate capacity of the reinforced concrete member, as defined in ACI 318-71 is based on the ultimate concrete strength (in compression or shear) and the yield strength of the reinforcing steel. The ultimate capacity of the reinforced concrete member is equal to, or greater than, the required capacity based on the factored load combinations. A further factor of safety, provided by ACI 318-71 or is that the calculated ultimate capacity of the member is reduced by a capacity reduction factor, as follows:

1. 0 = 0.90 for flexure, with or without axial tension

2. 0 = 0.90 for axial tension

3. 0 = 0.85 for shear and torsion

4. 0 = 0.75 for spirally reinforced concrete compression members

5. 0 = 0.70 for other members in compression

The magnitude of the load factors applied to each type of load varies depending on the nature of the applied load, the frequency or probability of occurrence, the accuracy in predicting the magnitude of load, and the degree of conservatism in establishing the unfactored load.

3.8.3.5.3 Thermal Gradient Effects

For factored load combinations, as described in Subsection 3.8.3.3.2, Item 2, which include thermal gradient effects, the permissible maximum strain in the reinforcing steel does not exceed 1.5 times the strain corresponding to the yield stress, as described in Subsection 3.8.3.4.6.

3.8.3.5.4 Structural Steel Allowable Stresses and Strains

1. Service Load Conditions

   For service load combinations (see Subsection 3.8.3.3.3, Item 1) the allowable steel stresses, fs, are equal to the normal allowable stresses as stated in AISC Specification, except as noted for thermal stresses in Subsection 3.8.3.3.3, Item 1.

2. Factored Load Conditions

   For factored load combinations, as indicated in Subsection 3.8.3.3.3, Item 2, the allowable steel stresses are equal to 1.60 times the normal allowable stresses, with a maximum of 0.9 x Fy in tension, compression, or bending and a maximum of 0.50 Fy in shear.

3.8.3.5.5 Earthquake Response of Interior Structure Related to the Requirements of Attached Equipment

The seismic dynamic analysis of the NSSS equipment considers the interaction between the equipment and supports and the internal structure. Deformations of the structure and equipment are checked and, when required, are limited to ensure that no loss of function of any component can occur.

3.8.3.5.6 Shear Response of Internal Structures

The concrete shear capacity, including shear reinforcement where required, is in accordance with the requirements of ACI 318-71.

3.8.3.5.7 Missile Load and Pipe Break Criteria at Local Areas

When subjected to impact loads by missiles and forces caused by a pipe rupture, localized yielding is permitted when it is demonstrated that the deflections or deformations of the structures and supports are within the ductility limits (Section 3.5.3.2) necessary to ensure that functional requirements are not impaired.

3.8.3.5.8 Criteria for Reactor Coolant System Supports

The stress criteria for the RCS supports are presented in Section 3.9N.1.4.8.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Concrete

1. Materials

   a. Cement is in conformance with the requirements of ASTM C 150-74, Specification for Portland Cement, Type II.

   b. Aggregates are in conformance with the requirements of ASTM C 33-74, Specification for Concrete Aggregates.

   c. Mixing water is potable or nonpotable, but is clean and free from injurious amounts of oils, acids, alkalis, salts, and organic materials or other substances which are deleterious to concrete or steel. Tests are in accordance with the requirements of CC-2223 of the ASME-ACI 359 document.

   d. Air-entraining admixtures conform to the requirements of ASTM C 260-74, Specification for Air-Entraining Admixtures for Concrete.

   e. Chemical admixtures conform to the requirements of ASTM C 494-71, Specification for Chemical Admixtures for Concrete.

2. Concrete Strength

   Concrete, when tested in accordance with ASTM C39-72, has a minimum compressive strength of 4000 psi in 28 days.

3. Other Concrete Properties

   See Subsection 3.8.1.6.1, Item 3.

4. Selection of Concrete Mix Proportions

See Subsection 3.8.1.6.1, Item 4.

5. Construction of Concrete

See Subsection 3.8.1.6.1, Item 5.

6. Examination, Testing, and Other Quality Control Procedures for Concrete

See Subsection 3.8.1.6.1, Item 6.

3.8.3.6.2 Reinforcing Steel

See Subsection 3.8.1.6.2.

3.8.3.6.3 Mechanical Butt Splices (Cadwelds)

Refer to Subsection 3.8.1.6.3 and to Appendix 1A(B).

3.8.3.6.4 Structural and Miscellaneous Steel

1. Materials

Listed below are specifications for structural and miscellaneous steel generally used. Other ASTM, conforming materials may be used as specified by project specifications/ design drawings.

ASTM A 36-74,	Specification for Structural Steel
ASTM A 537-74a,	Specification for Pressure Vessel Plates, Heat Treated, Carbon-Manganese-Silicon
ASTM A 307-74,	Specification for Carbon Steel Externally and Internally Threaded Standard Fasteners
ASTM A 325-74,	Specification for High-Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers
ASTM A 540-70,	Specification for Alloy Steel Bolting Materials for Special Applications
ASTM A 240-74a,	Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion Welded Unfired Pressure Vessels-Type 304 L

2. Fabrication and Erection of Structural Steel

Fabrication and erection of structural and miscellaneous steel are in accordance with the requirements of the AISC Specification.

3. Examination and Testing of Structural Steel

Structural and miscellaneous steel are examined and tested in accordance with the AISC Specification and the material specifications.

4. Certification, Marking, and Identification of Materials

All materials used will be provided with one of the following certifications which will become a part of the permanent project records:

a. Certified mill test reports giving chemical and physical properties,

b. Certified Materials Test Reports (independent party testing),

c. Certificates of Compliance which specify material specification, grade, class, and heat treatment as applicable.

Marking and identification are in accordance with the material specifications.

5. Certification of Tests and Examinations

Certification of all necessary tests and examinations are obtained from the mill, fabricator, or erector, as required.

3.8.3.6.5 Effect of Radiation on Structural Materials

The effects of radiation for the life of the plant do not degrade the structural integrity of the internal structures.

3.8.3.6.6 Quality Assurance Program

The documentation and maintenance of a QA program in the construction of the internal structures are in accordance with Chapter 17. NRC Regulatory Guide 1.28 is also complied with.

3.8.3.7 Testing and Inservice Inspection Requirements

3.8.3.7.1 Stainless Steel Liner

The seam welds of the stainless steel liner of the fuel transfer canal are vacuum box tested for leaktightness prior to filling with water. In addition, the stainless steel liner is provided with a leak chase system behind the liner, which is subject to periodic inservice monitoring for leaks.

3.8.3.7.2 Testing of Internal Structure

Except for the pneumatic pressure test of the entire Containment interior, as described in Subsection 3.8.1.7, there is no testing of the interior structures, such as differential pressure testing of individual compartments.

3.8.3.7.3 Testing of Polar Crane

The polar crane is tested to 125 percent of the rated capacity, in accordance with the requirements of ANSI B 30.2.0 (referenced in CMAA Specification No. 70).

3.8.3.7.4 Inservice Inspection Requirements

There is no inservice inspection program for the internal structures.

3.8.3.7.5 Tests and Inspections for Reactor Coolant System Supports

The tests and inspections for RCS supports are presented in Section 5.4.14.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

Other seismic Category I structures are as listed in Section 3.2.1.1. The design bases for these structures for normal operating conditions are governed by the applicable codes and standards. These structures are also designed so that there will be no loss of function caused by a rupture of any single pipe and a SSE if that function is related to public safety. Table 3.8-1 shows critical loading combinations, type of stress and computed and corresponding allowable stresses at key locations in other Seismic Category I structures.

3.8.4.1 Description of the Structures

3.8.4.1.1 Safeguards Building

The Safeguards Building is a multistory, reinforced concrete structure. There are two stories below grade and four stories above grade. Floor systems, columns, interior walls, and exterior walls are of reinforced concrete and are designed to support the vertical loads. In addition, the reinforced concrete floor systems and walls are designed to function as diaphragms and shear walls to resist the horizontal loads.

Adequate space is provided between the Safeguards Building and adjacent structures to avoid the contact of structures during SSE-induced motions. Piping and other systems crossing from the Safeguards and Electrical Buildings to other buildings or structures are designed with sufficient flexibility to maintain their function within applicable code allowable stress limits during the SSE.

The floors below grade house the safety injection pumps, RHR pumps and coolers, Containment spray pumps and coolers, and the auxiliary feedwater pumps. Floors above grade house the emergency diesel generators, electrical switchgear, motor control centers, and CRDM controls. Equipment locations are shown on Figures 1.2-1 through 1.2-11.

3.8.4.1.2 Auxiliary Building/Electrical and Control Building

The Auxiliary Building is a multi-story, reinforced concrete structure with one story below grade and four stories above grade. Floor systems, columns, and walls are of reinforced concrete and function to support vertical and horizontal loads in the same manner as those of the Safeguards Building. As with the Safeguards Building, space is provided to avoid the contact of structures, and systems are designed to accommodate the differential movements of adjacent buildings.

The Electrical and Control Building has five internal masonry block walls at Elevation 830'-0". Masonry block walls within Seismic Category I structures shall be designed to the load combinations and acceptance criteria specified in the NRC Structural Engineering Branch Interim Criteria for Masonry Walls. Upon evaluation, four of the walls were found to be acceptable based on no adverse seismic interaction with safety-related equipment; the fifth was found to be acceptable upon modification to prevent seismic interaction with ductwork. An additional masonry block wall is located in the Service Water Intake Structure; see Section 3.8.4.1.4. General information on masonry walls and removable precast block walls are contained within Table 17A-1.

The Auxiliary Building/Electrical and Control Building houses the Control Room, battery room, compressors, ventilating equipment, waste treatment equipment, and other fluid auxiliary systems. Equipment locations are shown on Figures 1.2-1 through 1.2-11.

3.8.4.1.3 Fuel Building

The Fuel Building is a reinforced concrete structure whose principal function is to house the new fuel storage area and the two spent fuel storage pools.

Both new and spent fuel bundles are stored in stainless steel racks which are located in the spent fuel pools which are filled with borated water. The spent fuel pools have thick concrete floors and walls and are lined with stainless steel plates for leaktightness.

The Fuel Building has an overhead electric crane capable of handling the fuel shipping cask. The crane is located so that it does not pass over either of the spent fuel pools. In addition, interlocks are provided to prevent movement of the shipping cask over the new fuel storage area. A fuel handling crane is mounted on the operating floor to transport new and spent fuel assemblies.

For the Fuel Building arrangement and equipment locations, see Figures 1.2-1 through 1.2-11.

3.8.4.1.4 Service Water Intake Structure

The Service Water Intake Structure is a reinforced concrete building located at the SSI. It houses the service water pumps and fire pumps and is equipped with trash racks, travelling screens, and screen wash pumps.

The base mat, walls, and floors are of reinforced concrete. Floor systems and walls are designed to support both vertical and horizontal loads, functioning as diaphragms and shear walls for horizontal loads.

Seismic loads used in the design of the Service Water Intake Structure include the hydrodynamic loads caused by seismic effects from the water contained in the SSI. The safety equipment contained within the building is suitably protected from tornado wind loads and tornado-generated missiles.

The Service Water Intake Structure has one internal masonry block wall. Conduits and an MCC are located in its vicinity, but the conduits and MCC were evaluated and found to provide no safety-related function. Therefore, the wall construction was found to be acceptable. General information on masonry walls and removable precast block walls is contained within Table 17A-1.

3.8.4.1.5 Drawings

For various details of other seismic Category I structures, see Figures 3.8-1 through 3.8-4 and Figure 3.8-16.

3.8.4.1.6 Outdoor Seismic Category I Tanks (Refueling Water Storage, Condensate Storage, and Reactor Makeup Water Storage)

The outdoor seismic Category I tanks are reinforced concrete structures, cylindrical in shape, with stainless steel liners to provide leaktightness and prevent absorption of radioactive material by the concrete (Refueling Water Storage Tank (RWST) only).

REFUELING WATER STORAGE AND CONDENSATE STORAGE TANKS:

Outside diameter of wall	50'-0"
Outside diameter of mat	53'-0"
Concrete wall thickness	2'-6"
Concrete mat thickness	5'-0"
Concrete roof thickness	1'-9"
Total height	54'-6"

REACTOR MAKE UP WATER STORAGE TANK

Outside diameter of wall	30'-0"
Outside diameter of mat	33'-0"
Concrete wall thickness	2'-6"
Concrete mat thickness	4'-0"
Concrete roof thickness	1'-9"
Total height	39'-6"

The tanks are designed to withstand all credible loadings and to maintain their integrity during operation. These loadings include both normal operating loads such as structure weight, hydrostatic pressure of the contained fluid, live loads on the roof, thermal loads and environmental loads such as the 1/2 SSE, SSE, normal wind and tornados (wind, differential pressure and missiles), and hydrodynamic forces caused by seismic effects on the contained fluid in accordance with methods as shown in Reference 21.

The load combinations given in Subsection 3.8.4.3 are used for the design of the structures, using design methods and strength requirements in accordance with ACI 318-71. Flexural

tensile cracking is permitted but is controlled by reinforcing steel. A minimum of 0.25 percent reinforcing steel is provided in the tank walls in both directions, vertical and hoop.

The structural analysis of the tanks is performed using the finite difference method. The SHELL-1 computer program is used for the analysis of axisymmetric static loads and nonaxisymmetric loads caused by wind, tornados, and earthquakes. This program is discussed in Section 3.7B(A). The motion of the contained fluid during an earthquake is accounted for by incorporating fluid masses into the mathematical model of the tank. The nonaxisymmetrical cases of wind, tornado, and seismic load (individually applied) are superimposed on the axisymmetric load cases to produce the highest design values, without regard to the direction of the nonaxisymmetric loads.

3.8.4.1.7 Concrete Barriers Provided for Category I Structures

In general, reinforced concrete external roofs and walls of seismic Category I structures form barriers against tornado generated missiles.

The compressive strength of the concrete used in the Category I structures which must resist the effects of postulated tornado winds and missiles, f'c is 4000 psi at 28 days.

Reinforcement steel is provided at each face in both directions with typical wall reinforcement #8 @ 8 each face each way (p = 0.004 E.F., E.W.) and typical roof reinforcement #10 @ 10 each face, each way (p = 0.006 E.F., E.W.).

3.8.4.2 Applicable Codes, Standards, and Specifications

Unless otherwise indicated, the design and construction of other seismic Category I structures are based upon the appropriate sections of the applicable codes, standards, specifications, recommendations, and references listed in Subsection 3.8.3.2. For anchorage requirements of reinforcing steel see Section 3.8.3.4.7(3).

3.8.4.3 Loads and Load Combinations

3.8.4.3.1 Loads

The following loads are considered in the design of other seismic Category I structures:

1. Normal Loads

   Normal loads are those which are encountered during normal plant operation and shutdown. They include the following:

   a. D = dead loads and their related moments and forces, including any permanent equipment loads

b. L = live loads and their related moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence such as soil and hydrostatic pressures and pressure differences caused by variation in heating and cooling and outside atmospheric changes. Hydrostatic pressures are included in this category. Hydrostatic loads from the water contained in the spent fuel pools and transfer canal may be treated as dead load (D) since they are maintained filled and the water level is maintained within specified tolerances. Ground water level is discussed in Subsection 3.8.5.1.5

c. To = thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition

d. Ro = pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition

2. Severe Environmental Loads

Severe environmental loads are those loads that could be encountered infrequently during the plant life. They include the following:

a. Feqo = loads generated by half the SSE = OBE

b. W = loads generated by the design wind specified for the plant

3. Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but highly improbable. They include the following:

a. Feqs = loads generated by the SSE

b. Wt = loads generated by the design tornado specified for the plant: the loads include those caused by the tornado wind pressure, tornado-created differential pressures, and tornado-generated missiles.

c. Fd = Impact effects and loads resulting from a fuel handling accident in the Spent Fuel Pool.

4. Abnormal Loads

Abnormal loads are those loads generated by a postulated high-energy pipe break accident within a building or compartment thereof, or both. They include the following:

a. Pa = pressure-equivalent static load within or across a compartment or building, or both, generated by the postulated break; it includes an appropriate dynamic factor to account for the dynamic nature of the load

b. Ta = thermal loads under thermal conditions generated by the postulated break and including To

c. Ra = pipe reactions under thermal conditions generated by the postulated break and including Ro

d. Yr = equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break; it includes an appropriate dynamic factor to account for the dynamic nature of the load.

e. Yj = jet impingement equivalent static load on a structure generated by the postulated break; it includes an appropriate dynamic factor to account for the dynamic nature of the load.

f. Ym = Missile impact equivalent static load on a structure generated by or during the postulated break, e.g., pipe whipping; it includes an appropriate dynamic factor to account for the dynamic nature of the load

For evaluation of the Fuel Building Spent Fuel Pool walls and slabs the effects of gamma heating shall be included along with To and Ta.

In determining an appropriate equivalent static load for Yr, Yj, and Ym, elastoplastic behavior is assumed with appropriate ductility ratios as long as excessive deflections do not result in loss of function.

5. Other Definitions

a. For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of AISC Specification.

The 33-percent increase in allowable stresses for concrete and steel because of seismic or wind loadings is not permitted.

b. For concrete structures, U is the section strength required to resist design loads based on methods described in ACI 318-71.

c. For structural steel, Y is the section strength required to resist design loads based on plastic design methods described in Part 2 of the AISC Specification.

3.8.4.3.2 Load Combinations and Acceptance Criteria for Other Seismic Category I Concrete Structures

1. Load Combinations for Service Load Conditions

a. U = 1.4 D + 1.7 L

b. U = 1.4 D + 1.7 L + 1.9 Feqo

c. U = 1.4 D + 1.7 L + 1.7 W

If thermal stresses due to To and Ro are present, the following also apply:

d. U = .75 (1.4 D + 1.7 L + 1.7 To + 1.7 Ro)

e. U = .75 (1.4 D + 1.7 L + 1.9 Feqo + 1.7 To + 1.7 Ro)

f. U = .75 (1.4 D + 1.7 L + 1.7 W + 1.7 To + 1.7 Ro)

L is considered for its full value or for its complete absence, and the following combinations are also satisfied:

g. U = 1.2 D + 1.9 Feqo

h. U = 1.2 D + 1.7 W

2. Load Combinations for Factored Load Conditions

For these conditions, which represent extreme environmental, abnormal environmental, abnormal/severe environmental, and abnormal/extreme environmental conditions, respectively, the following load combinations are satisfied:

a. U = D + L + To + Ro + Feqs

b. U = D + L + To + Ro + Wt

c. U = D + L + Ta + Ra + 1.5 Pa

d. U = D + L + Ta + Ra + 1.25 Pa
+ 1.0 (Yr + Yj + Ym) + 1.25 Feqo

e. U = D + L + Ta + Ra + 1.0 Pa
+ 1.0 (Yr + Yj + Ym) + 1.0 Feqs

See Section 3.6B.1.2.3 “Structural Design Margins” for additional requirements in the MS/FW penetration areas inside the Safeguards Building.

For the evaluation of local bearing effects in the spent fuel pools resulting from postulated fuel handling accidents, the following load combination shall be used:

f. U^a = D + L + To + Fd

In combinations shown in Items c, d, and e, the maximum values of Pa, Ta, Ra, Yj, Yr, and Ym, including an appropriate dynamic factor, are used, unless a time-history analysis is performed to justify otherwise. Combinations (b), (d), and (e) and the corresponding structural acceptance criteria should be satisfied first without the tornado missile load in (b) and without Yr, Ym, Yj, in

a. U is determined per the tri-axial compressive strength methodology described in NUREG/CR-6608

(d) and (e). For combinations shown in Items b, d and e, local stresses caused by the concentrated loads Wt, Yr, Yj, and Ym may exceed the allowables when there is no loss of function of any safety-related system.

L is considered for its full value or for its absence.

3.8.4.3.3 Load Combinations and Acceptance Criteria for other Seismic Category I Steel Structures

1. Load Combinations for Service Load Conditions

The AISC Specification is used for either the elastic working stress design methods of Part 1 or the plastic design methods of Part 2.

a. If the elastic working stress design methods are used, the following combinations apply:

1. S = D + L

2. S = D + L + Feqo

3. S = D + L + W

If thermal stresses due to To and Ro are present, the following combinations are satisfied:

4. 1.5 S = D + L + To + Ro

5. 1.5 S = D + L + To + Ro + Feqo

6. 1.5 S = D + L + To + Ro + W

L is considered for its full value or for its absence.

b. If plastic design methods are used, the following apply:

1. Y = 1.7 D + 1.7 L

2. Y = 1.7 D + 1.7 L + 1.7 Feqo

3. Y = 1.7 D + 1.7 L + 1.7 W

If thermal stresses due to To and Ro are present, the following also apply:

1a. Y = 1.3 D + 1.3 L + 1.3 To + 1.3 Ro

2a. Y = 1.3 D + 1.3 L + 1.3 Feqo + 1.3 To + 1.3 Ro

3a. Y = 1.3 D + 1.3 L + 1.3 W + 1.3 To + 1.3 Ro

L is considered for its full value or for its complete absence.

2. Load Combinations for Factored Load Conditions

a. If elastic working stress design methods are used, the following load combinations are satisfied:

1. 1.6 S = D + L + To + Ro + Feqs

2. 1.6 S = D + L + To + Ro + Wt

3. 1.6 S = D + L + Ta + Ra + Pa

4. 1.6 S = D + L + Ta + Ra +Pa
+ 1.0 (Yj + Yr + Ym) + Feqo

5. 1.7 S = D + L + Ta + Ra + Pa
+ 1.0 (Yj + Yr + Ym) + Feqs

b. If plastic design methods are used, the following load combinations are satisfied:

1. .90 Y = D + L + To + Ro + Feqs

2. .90 Y = D + L + To + Ro + Wt

3. .90 Y = D + L + Ta + Ra + 1.5 Pa

4. .90 Y = D + L + Ta + Ra + 1.25 Pa
+ 1.0 (Yj + Yr + Ym) + 1.25 Feqo

5. .90 Y = D + L + Ta + Ra + 1.0 Pa
+ 1.0 (Yj + Yr + Ym) + 1.0 Feqs

In these combinations, thermal loads are neglected when they are secondary and self-limiting in nature and when the material is ductile.

See Section 3.6B.1.2.3 “Structural Design Margins” for additional requirements in the MS/FW penetration areas inside the Safeguards Building.

In combinations shown in Items 2.a.3), 4), and 5), and in Items 2.b.3), 4), and 5) the maximum values of Pa, Ta, Ra, Yj, Yr, and Ym, including an appropriate dynamic factor, are used, unless a time history analysis is performed to justify otherwise. For combinations shown in Items 2.b.2), 4), and 5), local stresses caused by concentrated loads Yr, Yj, Ym, and Wt may exceed the allowables when there is no loss of function of any safety-related system. Furthermore, in computing the required section strength, the plastic section modulus of steel shapes is used.

3.8.4.3.4 Variable Loads

For loads which may vary, the values (within the possible range) which produce the most critical combination of loading are used in design.

3.8.4.3.5 Time Dependent Loads

Time dependent loads, such as the effects of creep, shrinkage, and other related effects, are included with dead load effects, as described in Section 9.3.7 of ACI 318-71 building code, where such loads affect design.

3.8.4.4 Design and Analysis Procedures

The static analysis of the other Seismic Category I structures is performed utilizing a combination of hand calculations and computerized analysis to determine the distribution of member forces and moments. These analyses use conventional methods of frame, shell, or shear wall analysis depending on the type of structural element under consideration.

Base mats of the other Seismic Category I structures are analyzed utilizing either computerized finite element analyses or hand calculations to determine the distribution of forces and moments within the mats. The mat analyses account for the stiffness of the supporting rock and do not consider any resistance to uplift from the rock (i.e., the interface between the mat and rock can not transfer any tension forces). A range of rock stiffness is considered in the analysis to account for uncertainties in the determination of the rock properties.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 General Criteria

The structural design criteria for other seismic Category I structures are established to provide structures that withstand all normal operating loads and also the extreme environmental loads of an earthquake or a tornado. The structures are designed to meet the requirements of the codes and standards listed in Subsection 3.8.3.2. The design of reinforced concrete structures conforms to the requirements of ACI 318-71, and structural steel design is in accord with AISC Specification.

The structural systems of these other seismic Category I buildings, except the outdoor seismic Category I tanks, consist of reinforced concrete floor slabs, beams, girders, walls, columns, and raft type foundation mats. Floor systems and walls are designed for vertical and lateral loads. Seismic, tornado, and other lateral loads applied to the total structure are resisted by the diaphragm action of the floors and shear walls and are transmitted to the foundation mats. Columns, because of their relatively low lateral stiffness, are not assumed to participate in resisting these lateral loads.

The seismic Category I structures other than the Containment are separated from each other and from the Containment to prevent contact caused by earthquake-induced motion. The widths of separations are determined from the deflections calculated in the dynamic analyses of the various structures.

The criteria for the outdoor seismic Category I tanks are described in Subsection 3.8.4.1.6.

3.8.4.5.2 Reinforced Concrete Acceptance Criteria

The design of the reinforced concrete structures is based on an ultimate strength concept, in accordance with the requirements of ACI 318-71. In this ultimate strength method of design, as

compared to previous working stress methods of design, the factor of safety is applied to the individual loads (as load factors) rather than to the allowable stresses. The ultimate load on the reinforced concrete member is based on the factored load combinations, as described in Subsection 3.8.4.3.2, Item 2. The ultimate capacity of the reinforced concrete member, as defined in ACI 318-71, is based on the ultimate concrete strength (in compression or shear) and on the yield strength of the reinforcing steel. The ultimate capacity of the reinforced concrete member is equal to or greater than the capacity required, based on the factored load combinations. A further factor of safety is provided by ACI 318-71 in that the calculated ultimate capacity of the member is reduced by a capacity reduction factor, as indicated in Subsection 3.8.3.5.2.

The magnitude of the load factors applied to each type of load varies, depending on the factors discussed in Subsection 3.8.3.5.2.

3.8.4.5.3 Missile Load and Pipe Break Criteria at Local Areas

For local areas subjected to loads, such as missiles, and to forces caused by pipe rupture, localized yielding is permitted when the deflections or deformations of the structures and supports are within the (ductility) limits (Section 3.5.3.2) necessary to ensure that functional requirements are not impaired.

3.8.4.5.4 Bracket or Corbel Criteria at Local Areas

The fuel building crane corbel supports at elevation 831 are designed in accordance with the PCI Design Handbook Second Edition 1978.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

The materials and QC procedures used in the construction of other seismic Category I structures are as discussed in Subsection 3.8.3.6. No special construction techniques are required for these structures.

3.8.4.7 Testing and Inservice Inspection Requirements

With the exception of the stainless steel liners for the spent fuel pool and the outdoor seismic Category I tanks, no special testing of the completed structure or inservice inspection is required for seismic Category I structures other than the Containment. The seam welds of the stainless steel liners are tested for leaktightness prior to filling with water by the vacuum box method. In addition, the spent fuel pools are provided with a leak-chase system behind their liners which is subject to periodic inservice monitoring for leaks.

3.8.5 FOUNDATIONS

3.8.5.1 Description of the Foundations

3.8.5.1.1 Foundation Mat and Supports for Containment and Internal Structure

The foundation mat for the Containment and internal structure is described in Subsection 3.8.1.1.1. It is essentially a flat mat, approximately 12 ft thick, with a reactor cavity pit projection. The foundation mat is covered with a steel liner that is an integral part of the

Containment liner system, which is then covered with a 30-in.- thick interior base slab, as described in Subsection 3.8.3.1.1. The arrangement of reinforcing steel at the junction of the bottom of the Containment wall and the top of the foundation mat is shown on Figure 3.8-10. A discussion of the lateral load transfer to the mat from the vertical members of the internal structure is provided in Subsection 3.8.3.4.4. The geometry of the reinforcing steel in the mat is described in Subsection 3.8.1.1.4. The principal reinforcing in the mat consists of No. 18 bars, cadwelded at splices.

The steel supports for the steam generators, the reactor pressure vessel, the reactor coolant pump, and the pressurizer are discussed in Section 5.4.14 and shown on Figures 5.5-1 to 5.5-5. Embedded steel, suitably anchored, is provided in the concrete internal structure to support the NSSS equipment supports. Typical details, illustrating the manner in which the embedded steel used to support NSSS equipment is anchored to the internal structures, are shown on Figures 3.8-17 and 3.8-18.

Reinforced concrete supports are approximately equally spaced around the periphery of the reactor vessel. Initial drawings for the Unit 2 reactor pressure vessel support were prepared with the supports oriented as a mirror image of the Unit 1 supports. The reactor vessel supplied for Unit 2, however, is a duplicate for the Unit 1 reactor vessel. As a result, the Unit 2 reactor vessel supports were misoriented by about 45 degrees from their proper location. See Figure 3.8-24. Since concrete had been poured to elevation 819'-0", vertical holes were drilled, anchor bolts installed in the holes and then grouted as a substitute for the anchor bolts with anchor plates, which had been improperly located. In place of the #18 horizontal rebars used in the wrong location, #11 rebars of equal total area were installed and grouted in drilled holes as shown in Figure 3.8-25. The procedure and materials used in the installation of the rebars in drilled holes in concrete supports which were then grouted is similar to the procedures used in the drilling and grouting described in Reference [23]. Design and analysis procedures for these supports is discussed in Section 3.8.5.4.1.

3.8.5.1.2 Foundation For Other Seismic Category I Structures

The foundations for other seismic Category I structures are generally flat slabs that vary in thickness from approximately 3 to 6 ft, depending on the requirements for shear and bending moment. Reinforcing steel is provided in two mutually perpendicular directions, at both the top and bottom faces. Shear reinforcement is provided where required by the provisions of ACI 318-71. The principal reinforcing in the foundations for other seismic Category I structures consists of No. 11 bars, lap spliced in accordance with ACI 318-71. For transfer of base shear, see Subsection 3.8.5.3.3.

3.8.5.1.3 Relationship Between Various Seismic Category I Structures

For the physical relationship between the various foundations, see Figures 3.8-1 through 3.8-4. Sufficient space, not less than one inch, is provided between foundations to prevent contact during SSE. However, during construction, it was observed that some debris had fallen into the gap between foundation mats, which could not be removed. Since all the foundations are on bed rock and are very stiff, it was determined that the presence of debris in the foundation gaps will not result in significant load transfer from one building to another such that the seismic response is altered during SSE.

3.8.5.1.4 Underlying Foundation Material

All seismic Category I foundations are founded on competent rock, as described in Sections 2.5 and 3.7.

3.8.5.1.5 Effects of Ground Water

Although some of the Category I foundations are below the PMF level of the reservoir (elevation 789.7 ft), ground water is not expected to reach higher than elevation 775.0 ft because of the impermeable nature of the rock. See Section 2.4.13.1.3 for a discussion of the level of permeability of the Glen Rose Formation, and Section 2.4.13.1.5 for a discussion of the onsite water table.

Elevation 775.0 ft. is only slightly above the deepest of the seismic Category I foundations (Safeguards Building, floor elevation 773'-0). In the event of in-leakage into the building, the small amount will be collected by the floor drains and routed to the floor drain sumps. Each of the two floor drain sumps is equipped with duplex sump pumps cable of pumping 50 gpm of fluid per pump. Therefore waterproofing is not required. Also refer to Section 9.3.3.

3.8.5.1.6 Cathodic Protection

Substrata and water resistivity tests have been performed, and where resistivity is less than approximately 5000 ohms cm, a suitable cathodic protection system is provided to protect buried and submerged metallic structures such as piping, fuel oil tanks, and intake structures.

3.8.5.1.7 Drawings

For various details of foundations, see Figures 3.8-1 through 3.8-4 and Figure 3.8-12.

3.8.5.2 Applicable Codes, Standards, and Specifications

The applicable codes, standards, and specifications for foundations and concrete supports are listed in the following applicable sections:

1. For Containment foundation and internal structure, see Subsections 3.8.1.2 and 3.8.3.2, respectively.

2. For other seismic Category I structures foundations, see Subsection 3.8.4.2.

3. For the additional standard ACI 436, see Subsection 3.8.5.4.2, Item 2.

3.8.5.3 Loads and Load Combinations

3.8.5.3.1 Foundation Design Loads and Combinations

The loads and load combinations used in the design of the foundations are the same as the loads and load combinations used in the design of the superstructures. Following is a list of the referenced sections that apply:

1. For Containment foundation and internal structure, see Subsections 3.8.1.3 and 3.8.3.3.

2. For other seismic Category I structures' foundations, see Subsection 3.8.4.3.

3.8.5.3.2 Various Effects at Foundations

1. The analysis performed for each structure considers the effect of torsional moments.

2. The factor of safety against overturning, for each structure, is greater than 1.1 when considering any load combinations (see Subsection 3.8.5.5.3).

3. The base shear is transferred from the foundation to rock through bond and surface friction and is not a problem for structures founded on competent rock, such as that which exists at the Comanche Peak site.

3.8.5.3.3 Loads Transferred from Supported Components to Foundations

The load combinations considered in the determination of the total loads transferred from supported components, such as the NSSS equipment, to the foundations are the same load combinations as those described in Subsection 3.8.5.3.1. For additional discussion, see Subsection 3.8.3.4.3.

3.8.5.4 Design and Analysis Procedures

3.8.5.4.1 Foundation and Supports for Containment and Internal Structure

The analysis of the foundation mat for the Containment and internal structure is described in Subsection 3.8.1.4.1, Item 1. The output of this analysis includes the displacements, rotations, forces, shears, moments, and stresses which are used for the design of the foundation mat.

1. Determination of Rock Contact Area Under the Foundation

   For load combinations which include the overturning effects of earthquake or tornado forces, some lift-off may occur, resulting in a condition where only a portion of the foundation is in contact with the rock. The area of contact between the bottom surface of the foundation and the rock is dependent on the resultant forces applied to the entire structure, based on the load combination being considered. The rock reaction is simulated by attaching appropriate springs to the nodes on the foundation mat that are within the area of contact. The predicted amount of foundation contact area affects the results of the analysis described in Subsection 3.8.1.4.1, Item 1.

   The following procedures are used to determine the contact area. Before proceeding with the analysis of the structure under a combined loading, which includes earthquake loads or tornado loads, an area of rock contact is postulated. This postulated contact area is determined by first assuming that the structure is rigid with respect to the foundation springs. Then, by considering equilibrium of the applied force and the rock-spring reactions, the postulated contact area is determined by trial and error through a systematic search process. After the analysis based on the postulated contact area is performed, the resulting contact area is checked. If the postulated and resulting contact areas are significantly different, a new postulated contact area is determined based on

analysis results, and the structure is reanalyzed. The checking cycle is terminated when the postulated contact area and the resulting contact area of the same analysis converge within a tolerable limit (approximately 5 percent difference).

2. NSSS Equipment Concrete Supports

The concrete supports for the NSSS equipment are designed for all the loading combinations (listed in Subsection 3.8.3.3) which include seismic and blowdown effects. A discussion of the blowdown effects as a result of a LOCA is contained in Sections 3.6 and 3.9. The dynamic analysis under seismic loading is described in Section 3.7.

Shear resistance capability through the use of rebar reinforcement (described in Section 3.8.5.1.1) is assumed to be developed by means of shear friction. The shear friction design procedure is described in ACI 318-77 [Reference 24]. Effectiveness of the grout used in the installation of the rebars in drilled holes in concrete supports is confirmed by Reference [23].

3.8.5.4.2 Foundations for Other Seismic Category I Structures

1. The analysis of foundations for other seismic Category I structures is described in Subsection 3.8.4.4. The output of the analysis includes the displacements, rotations, forces, shears, moments, and stresses in the foundation mat which are used for the design of the mat.

The determination of the rock contact area under the foundation when the structure is subjected to lateral forces, such as earthquake or tornado, is as described in Subsection 3.8.5.4.1, Item 1.

2. For the other seismic Category I structures that are not analyzed by using computer programs, such as the Fuel Building and the Service Water Intake Structure (see Subsection 3.8.4.4), the foundations are designed using techniques similar to those described in the report of ACI Committee 436, Suggested Design Procedures for Combined Footings and Mats, ACI Manual of Concrete Practice, Part 2, 1968.

3.8.5.4.3 Descriptions of Computer Programs

For the descriptions of the computer programs used in the referenced Subsections of 3.8.5, see Appendix 3.7B(A).

3.8.5.4.4 Computer Program Check

For the validity of the computer programs, the analytical models, and the results of the analyses, see Subsection 3.8.3.4.1.

3.8.5.4.5 Design Variables

For a general discussion of design variables, see Subsection 3.8.3.4.5. In addition, a minimum and maximum value of rock springs are considered for foundation design to account for a realistic range of probable values. Upper and lower bound estimates of the rock spring values are based on possible variations in measuring the material proportion of the rock. The variations

estimated on the basis of past experience are five percent in the mass density of the rock and 15 percent in the measurement of shear wave velocity.

3.8.5.4.6 Lateral Load Transfer to Soil

All seismic Category I structures are founded on bedrock. The transfer of lateral loads from the bottom of the foundation to the rock is primarily through bond and surface friction between concrete and rock. In addition, for the Containment foundation mat, the reactor cavity pit provides keying action.

3.8.5.4.7 Effects of Settlement

The nature of the bedrock on which all the seismic Category I structures are founded is such that both the gross and differential settlements are insignificant.

3.8.5.5 Structural Acceptance Criteria

3.8.5.5.1 Allowable Stresses and Strains

The allowable stresses and strains for the foundations are essentially the same as the allowables for the superstructures which are supported thereon, considering the same load combinations. The applicable Subsections are 3.8.1.5 and 3.8.3.5 (for foundation for Containment and internal structure) and 3.8.4.5 (for foundation for other seismic Category I structures).

3.8.5.5.2 Foundation Settlements

Considering the nature of the rock on which all seismic Category I structures are founded, gross and differential settlements are insignificant.

3.8.5.5.3 Factors of Safety Against Overturning, Sliding, and Floatation

The factors of safety against overturning, sliding, and floatation, when considering all load combinations including the SSE or tornado loading, are as follows:

	Minimum Factors of Safety		
Load Combination	Overturning	Sliding	Floatation
D + H + Feqo	1.5	1.5	-
D + H + W	1.5	1.5	-
D + H + Feqs	1.1	1.1	-
D + H + W_t	1.1	1.1	-
D + F'	-	-	1.5

D, Feqo, Feqs, W, and W_t are loads defined in Subsection 3.8.4.3. H is lateral earth pressure load. F' is the buoyant force due to the PMF. For earthquake loads Feqo and Feqs, three components of earthquakes are combined as described in Section 3.7B.

3.8.5.5.4 Effects of Rock Structure Interaction

The rock structure interaction effects are considered in the seismic dynamic analysis discussed in Section 3.7B. In regard to the static analysis of the foundations, the interaction between the foundation and the rock is reflected in the use of appropriate rock springs. The maximum pressure of the foundation on the rock is determined for each load combination, and is within the allowable limits as defined in Section 2.5.4.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The materials, quality control, and special construction techniques for the foundations are essentially the same as those required for the structures which are supported thereon. The following sections apply:

1. For Containment foundation Subsection 3.8.1.6

2. For other seismic Category I structures foundations Subsection 3.8.4.6

The QC procedures for the concrete, including the type of tests required, the frequency and location of sampling, and the test requirements are described in the referenced sections cited in Section 3.8.1.6.1.

3.8.5.7 Testing and Inservice Inspection Requirements

1. The structural acceptance test of the Containment, as described in Subsection 3.8.1.7.1, subjects the Containment foundation to a test pressure load which demonstrates that the foundation responds satisfactorily to the maximum internal pressure.

2. Leak chase channels for leaktightness examination are provided at seams in the Containment mat liner that are inaccessible for other means of inspection.

3. Other than a visual inspection after construction, no other special testing or inservice inspection is required for seismic Category I foundations.

REFERENCES

1. ASME-ACI 359 document, Proposed Standard Code for Concrete Reactor Vessels And Containments, issued for trial use and comments, April 1973.

2. ACI Committee 349, Criteria For Reinforced Concrete Nuclear Power Containment Structures, ACI Journal, January 1972.

3. 10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, February 5, 1973.

4. ANSI N45.4-1972, Leakage-Rate Testing of Containment Structures For Nuclear Reactors, March 16, 1972.

5. NRC Regulatory Guide 1.10, Mechanical (Caldweld) Splices in Reinforcing Bars of Category I Concrete Structures, Revision I, January 2, 1973, U.S. Nuclear Regulatory Commission.

6. NRC Regulatory Guide 1.15, Testing of Reinforcing Bars for Category I Concrete Structures, Revision 1, December 28, 1972, U.S. Nuclear Regulatory Commission.

7. NRC Regulatory Guide 1.18, Structural Acceptance Test for Concrete Primary Reactor Containments, Revision 1, December 28, 1972, U.S. Nuclear Regulatory Commission.

8. NRC Regulatory Guide 1.19, Nondestructive Examination of Primary Containment Liner Welds, Revision 1, August 11, 1972, U.S. Nuclear Regulatory Commission.

9. NRC Regulatory Guide 1.28, Quality Assurance Program Requirements (Design and Construction), June 7, 1972, U.S. Nuclear Regulatory Commission.

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14. Roark, R.J., Formulas For Stress and Strain, McGraw-Hill, 4th ed., 1965.

15. Timoshenko, S., Theory of Plates and Shells, McGraw-Hill, 1940.

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17. ACI 349-76, Code Requirements for Nuclear Safety Related Concrete Structures, Appendix A, 1976.

18. PCI Design Handbook, precast and prestressed concrete Second Edition 1978.

19. AWS D1.1-75 Structural Welding Code requirements for Structural Welded Construction.

20. AWS D1.1- Rev. 1-73 and later edition for prequalified joint details and base materials.

21. U. S. Atomic Energy Comm., “Nuclear Reactors and Earthquakes”, TID-7024, Office of Technical Service, Wash., D. C. 1963 p. 183-195 and 367-390.

22. AWS D1.1-90 “Structural Welding Code” for Welder Qualifications of partial penetration groove and fillet welds on tubular T, K and Y connections.

23. Report of pull-out tests and grouped rebars, Robert W. Hunt Company, December 1, 1997.

24. ACI 318-77, Building Code Requirements for Reinforced Concrete, 1977.

25. 10CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Option B, Performance Based Requirements, effective October 26, 1995.

TABLE 3.8-1
SUMMARY TABLE FOR CATEGORY 'I' STRUCTURES

(Sheet 1 of 3)

TYPE OF STRUCTURE	KEY LOCATIONS[a]	CRITICAL LOADING COMBINATION	TYPE OF STRESS	ACTUAL STRESS (KSi)	ALLOW STRESS (KSi)
CONTAINMENT BLDG.	Bottom of Containment Wall @ EL.805'-6"	D+L+1.25 Pa+To+Ta+1.25E+ Ra+Yr+Yj+Ym	Tension	49.3	54.0
			Shear	13.6	50.0[b]
	Top of Containment Wall @ EL.1000'-6"	D+L+1.25 Pa+To+Ta+1.25E+ Ra+Yr+Yj+Ym	Tension	54.0	54.0
	Inside Face of Equipment Hatch at = 225° for Unit 1, 315° for Unit 2	D+L+1.5 Pa+To+Ta+Ra	Tension	54.0	54.0
REACTOR BLDG. (Unit 1)	Steam Generator Compartment #3. Northeast Wall EL 897	D+L+To+Ra+Pa+(Yr+Yj+Ym) + Feqs	Axial Tension and Flexure	54.0	54.0
			Transverse Shear	0.202	0.203
REACTOR BLDG. (Unit 2)	Steam Generator Compartment #1.West Cavity Wall EL 863	D+L+Ta+Ra+Pa+(Yr+Yj+Ym) + Feqs	Axial Tension and Flexure	54.0	54.0
			Transverse Shear	0.087	0.101[c]
	Steam Generator Upper Lateral Support Beam (Struct. Stl.)	D+L+Ta+Ra+Pa+(Yr+Yj+Ym) + Feqs	Axial Compression & Flexure Shear	Unit 1: 25.1 8.43	Unit 1: 45.0 25.0
				Unit 2: 20.6 6.72	Unit 2: 45.0 20.0

TABLE 3.8-1
SUMMARY TABLE FOR CATEGORY 'I' STRUCTURES

(Sheet 2 of 3)

TYPE OF STRUCTURE	KEY LOCATIONS[a]	CRITICAL LOADING COMBINATION	TYPE OF STRESS	ACTUAL STRESS (KSi)	ALLOW STRESS (KSi)
AUXILIARY BLDG.	5-A Wall East of F-A Above EL 790'-6"	0.75[1.4D+1.7L+1.9Feqo+ 1.7To+1.7Ro]	Flexure Shear[d]	54 25.5	54 51
FUEL BLDG.	C-F Wall Above EL 860'-0" South of col. Line 1-F	D+L+To+Ro+W_t	Flexure Shear[d]	37.8 37.2	54 51
	Bottom of A-F Wall Between 4-F and 2-F	D+L+Ta+Ra+Pa+ (Yr+Yj+Ym) + Feqs	Flexure Shear	35.3 0.063	54 0.183[c]
SAFEGUARDS BLDG.	4-S Wall Below EL 810'-6" for Unit 1, 13-S Wall Below EL 810'-6" for Unit 2	0.75[1.4D+1.7L+1.9Feqo+ 1.7To+1.7Ro]	Flexure Shear[d]	48.6(U1) 38.9(U2) 20.6(U1) 29.2(U2)	54 51
	8-S Wall EL 810'-6" for Unit 1, 9-S Wall Between EL 773'-0" and 790'-6" for Unit 2	0.75[1.4D+1.7L+1.9Feqo+ 1.7To + 1.7Ro] 1.4D+1.7L+1.9 Feqo	Flexure Shear[d]	54 (U1) 44.3(U2) 39.0(U1) 41.8(U2)	54 51
SERVICE WATER INTAKE STRUCTURE	Bottom of North Exterior Wall EL 755'-0"	1.4D+1.7L+1.9Feqo	Flexure	53.2	54.0

TABLE 3.8-1
SUMMARY TABLE FOR CATEGORY 'I' STRUCTURES

(Sheet 3 of 3)

TYPE OF STRUCTURE	KEY LOCATIONS[a]	CRITICAL LOADING COMBINATION	TYPE OF STRESS	ACTUAL STRESS (KSi)	ALLOW STRESS (KSi)
	Bottom of West Exterior Wall EL 755'-0"			53.7	
REFUELING WATER STOR. TANKS & CONDENSATE STOR. TANKS	Base of the Circumferential Exterior Wall	1.4D+1.7L+1.7+Hydro+ 1.9Feqo	Flexure Shear	52.4 .122	54 .126[c]
REACTOR MAKE-UP WATER STOR. TANK	Base of the Circumferential Exterior Wall	0.75[1.4D+1.7L+1.9Feqo+ 1.7To+1.7Ro]	Flexure Shear	50.0 .068	54 .090[c]

Notes:

a) Unit 2 stesses are the same as Unit 1 stresses except as noted.

b) 1" x 4" Bar Stock

c) Concrete Shear Stress

d) Stress shown is for shear reinforcing steel.

VISUAL WELD ACCEPTANCE CRITERIA

FOR STRUCTURAL WELDING

AT NUCLEAR POWER PLANTS

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TABLE OF CONTENTS

Page

Preface iii

1.0 INTRODUCTION 3.8A-1

2.0 ACCEPTANCE CRITERIA 3.8A-4

2.1 Introduction 3.8A-5
2.2 Scope 3.8A-5
2.3 Materials 3.8A-6
2.4 Measurement Units 3.8A-6
2.5 Acceptance Criteria 3.8A-7

3.0 INSPECTION GUIDELINES 3.8A-10

3.1 The inspector 3.8A-11
3.2 General Inspection Requirements 3.8A-11
3.3 Inspection Philosophy 3.8A-11
3.4 Inspection Principles 3.8A-12
3.5 Weld Attribute Descriptions, Acceptance Criteria and Inspection Guidelines 3.8-A-13
3.5.1 Weld Cracks 3.8A-13
3.5.2 Fillet Weld Size 3.8A-13
3.5.3 Incomplete Fusion 3.8A-14
3.5.4 Weld Overlap 3.8A-14
3.5.5 Underfilled Craters 3.8A-15
3.5.6 Weld Profiles 3.8A-15
3.5.7 Undercut 3.8A-16
3.5.8 Surface Porosity 3.8A-18
3.5.9 Weld Length and Location 3.8A-18
3.5.10 Arc Strikes 3.8A-19
3.5.11 Surface Slag and Weld Spatter 3.8A-20
3.6 Significant Measurement Units 3.8A-20
3.7 Equivalency 3.8A-20

4.0 COMMENTARY 3.8A-22

4.1 Criteria Evaluation 3.8A-23
4.1.1 Weld Cracks 3.8A-23
4.1.2 Fillet Weld Size 3.8A-23

TABLE OF CONTENTS (continued)

Page

4.1.3 Incomplete Fusion 3.8A-24
4.1.4 Weld Overlap 3.8A-25
4.1.5 Underfilled Craters 3.8A-25
4.1.6 Weld Profiles 3.8A-26
4.1.7 Undercut 3.8A-26
4.1.8 Surface Porosity 3.8A-29
4.1.9 Weld Length and Location 3.8A-29
4.1.10 Arc Strikes 3.8A-30
4.1.11 Surface Slag and Weld Spatter 3.8A-31
4.2 Cumulative Effects 3.8A-31
4.3 Sample Calculations 3.8A-34
4.3.1 For the combined Effect of Underlength and Undersize 3.8A-34
4.3.2 For the Effect of Undercut in 5/8 Inch Material 3.8A-35
4.3.3 For the Effect of Undersize in a 3/16 Inch Fillet Weld 3.8A-35
4.3.4 For the Effect of Weld Underlength 3.8A-36

PREFACE

This document has been prepared and issued under the auspices of the NUCLEAR CONSTRUCTION ISSUES GROUP (NCIG). NCIG was formed by several utilities for the purpose of developing a common approach to the resolution of issues at nuclear power plant construction sites. The first issue considered by NCIG was in the area of acceptance criteria for and inspection of welds in certain supports and structures. This document contains NCIG's resolutions of that issue.

SECTION 1.0

INTRODUCTION

1.0 INTRODUCTION

1.1 This document provides Acceptance Criteria for visual inspection of structural welds in nuclear power plants. The development of such acceptance criteria by the owner and the Engineer[a] falls within the provisions of the AISC Specification[b] and AWS D1.1[c]. This provision is clarified in the 1985 edition of AWS D1.1. A new paragraph 1.1.1.1 has been added which states:

> "1.1.1.1 The fundamental premise of the Code is to provide general stipulations adequate to cover any situation. Acceptance criteria for production welds different from those specified in the Code may be used for a particular application provided they are suitably documented by the proposer and approved by the Engineer. These alternate acceptance criteria can be based upon evaluation of suitability for service using past experience, experimental evidence or engineering analysis considering material type, service load effects, and environmental factors."

In addition, the commentary for this new paragraph reads:

> "C1.1.1.1 The workmanship criteria provided in Section 3 of the Code are based upon knowledgeable judgement of what is achievable by a qualified welder. The criteria in Section 3 should not be considered as a boundary of suitability for service. Suitability for service analysis would lead to widely varying workmanship criteria unsuitable for a standard code. Furthermore, in some cases, the criteria would be more liberal than what is desirable and producible by a qualified welder. In general, the appropriate quality acceptance criteria and whether or not a deviation is harmful to the end use of the product should be the Engineer's decision. When modifications are approved, evaluation of suitability for service using modern fracture mechanics techniques, a history of satisfactory service, or experimental evidence is recognized as a suitable basis for alternate acceptance criteria for welds."

1.2 Engineer's Responsibility: The Engineer, as used in this document, is the individual or the organization designated by the owner as being responsible for the design of the structure being welded or inspected. The owner may fulfill the role of the Engineer for the design and inspection of any structure, in accordance with the provisions of the owner's Quality Assurance Program.

(a). See paragraph 1.2 for a definition of the Engineer, as used in this document.

(b). American Institute of Steel Construction, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings."

(c). American Welding Society, "Structural Welding Code - Steel, D1.1."

Before the provisions of this document are used for structures which have already been designed, the Engineer shall review this document to assure applicability to the structures for which it will be used. Before application, the Engineer shall determine that the provisions of this document are consistent with the engineering considerations used for the original design.

The Engineer shall specify the structures to which these Acceptance Criteria will be applied.

It is not the intent of this document to preclude the development and use of other visual weld acceptance criteria. The Engineer may specify alternative acceptance criteria; this may be done based on the provisions of paragraph 1.1.1.1 of the 1985 edition of AWS D1.1, with documented justification and appropriate project approvals.

1.3 After the Engineer accepts this document for use, it may be used to supplement specifications for design and construction as appropriate.

1.4 The application and distribution of these Acceptance Criteria and Inspection Guidelines shall be controlled in accordance with applicable document control procedures.

1.5 The workmanship provisions of AWS D1.1 are not modified by the Acceptance Criteria presented in this document and are to be implemented by the fabricator or erector during production.

1.6 Section 2.0 of this document contains a Scope statement, specifies applicable materials, weld Acceptance Criteria, and measurement units; Section 3.0 provides Inspection Guidelines for use in applying these Acceptance Criteria; Section 4.0 contains engineering evaluations for Acceptance Criteria.

SECTION 2.0

ACCEPTANCE CRITERIA

2.0 ACCEPTANCE CRITERIA

2.1 Introduction

2.1.1 This Section contains Acceptance Criteria for use when inspecting welds in nuclear power plant structures and supports that have been designed and fabricated to the requirements of the AISC Specification and AWS D1.1.

2.1.2 The provisions of this document do not apply to ASME Code[(d)] stamped work.

2.2 Scope

2.2.1 This document is intended to be used with design and construction specifications for nuclear power plants where the structures have already been designed, fabricated, and/or erected, and for new work.

2.2.2 This document is applicable to welded construction for safety and non-safety related structures, including seismically loaded structures, where fatigue[(e)] is not the governing design consideration. Examples of typical structures to which these criteria apply include, but are not necessarily limited to, steel components such as:

2.2.2.1 Main building framing members and connecting members;

2.2.2.2 Supports for equipment, components and piping[(f)], cable trays and conduit, and HVAC ducts;

2.2.2.3 Miscellaneous steel including bracing and stiffeners; embedments; stairways and handrails, doors and door frames, windows and window frames, gratings, covers, etc.

(d). American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Nuclear Components.

(e). For the purposes of this document, structures whose design is governed by fatigue are those structures for which an analysis is required for cyclic service and whose endurance limit must be considered in the design. At a nuclear power plant, most supports and pipe hangers, which might be designed considering seismic loads, are not governed by fatigue design limits.

(f). Excluding component supports stamped in accordance with the ASME Code Section III, Subsection NF.

2.2.3 This document and its provisions may be used for the inspection and acceptance of welds in other structures and systems not specifically addressed in 2.2.2, provided the provisions of Sections 1.2 and 2.3 are met.

2.3 Materials

2.3.1 This document is applicable to the inspection of the following structural steels:

Material Number 1, Groups Number 1 and 2, as listed in Table C2 of AWS B2.1. (These materials include those structural steels also listed in Table 4.1.1 of AWS D1.1 as Groups I and II; P Number 1, Groups 1 and 2, as listed in Table QW-422 of Section IX of the ASME Code; and S Number 1, Groups 1 and 2 as listed in Table 1 of ASME Code Case N-71-12.)

2.4 Measurement Units

Table 2-1 identifies the smallest measurement units the Inspector will use when inspecting the listed weld attributes. When measuring and recording dimensions of the weld attributes listed in Table 2-1, these dimensions shall be rounded off to the nearest significant unit.

Table 2-1

Weld Attribute	Reference Section	Smallest Measurement Unit (Significant Unit - Inches)
Fillet Weld Size	2.5.2.2	1/16
Incomplete Fusion	2.5.2.3	1/8
Weld Overlap	2.5.2.4	1/8
Undercut Depth	2.5.2.7	1/32
Surface Porosity	2.5.2.8	1/16
Weld Length	2.5.2.9	1/8 or 1/4
Weld Location	2.5.2.9	1
Slag	2.5.2.11	1/4

2.5 Acceptance Criteria

2.5.1 These Acceptance Criteria are to be used for the acceptance inspection of welds in the uncoated condition. These criteria may also be used for subsequent inspections after the welds have been coated, with the concurrence of the Engineer.

2.5.2 A weld shall be acceptable by visual inspection, subject to the following:

2.5.2.1 Weld Cracks: The weld shall have no cracks.

2.5.2.2 Fillet Weld Size: A fillet weld shall be permitted to be less than the size specified by 1/16 inch for 1/4 the length of the weld. Oversized fillet welds shall be acceptable if the oversized weld does not interfere with mating parts.

2.5.2.3 Incomplete Fusion: In fillet welds, incomplete fusion of 3/8 inch in any 4 inch segment, and 1/4 inch in welds less than 4 inches long, is acceptable. For groove welds, incomplete fusion is not acceptable. For fillet and groove welds, rounded end conditions that occur in welding (starts and stops) shall not be considered indications of incomplete fusion and are irrelevant.

2.5.2.4 Weld Overlap: Overlap is acceptable provided the criteria for weld size and fusion can be satisfied. When fusion in the overlap length cannot be verified, an overlap length of 3/8 inch in any 4 inch segment, and 1/4 inch in welds less than 4 inches long, is acceptable.

2.5.2.5 Underfilled Craters: Underfilled craters shall be acceptable provided the criteria for weld size are met. Craters which occur outside the specified weld length are irrelevant provided there are no cracks.

2.5.2.6 Weld Profiles

2.5.2.6.1 The faces of fillet welds may be convex, flat, or concave, provided the criteria for weld size are met.

2.5.2.6.2 The faces of groove welds may be flat or convex.

2.5.2.6.3 Convexity of fillet and groove welds are not criteria for acceptance and need not be measured.

2.5.2.6.4 The thickness of groove welds is permitted to be a maximum of 1/32 inch less than the thinner member being joined.

2.5.2.7 Undercut

2.5.2.7.1 For material 3/8 inch and less nominal thickness, undercut depth of 1/32 inch on one side for the full length of the weld, or 1/32 inch on one side for 1/2 the length of the weld and 1/16 inch for 1/4 the length of the weld on the same side of the member, is acceptable. For members welded on both sides where undercut exists in the same plane of a member, the cumulative lengths of undercut shall be limited to the lengths of undercut allowed on one side. Melt-through that results in a hole in the base metal is unacceptable.

2.5.2.7.2 For materials greater than 3/8 inch nominal thickness, undercut depth of 1/32 inch for the full length of the weld and 1/16 inch for 1/4 the length of the weld on both sides of the member is acceptable. When either welds or undercut exist only on one side of the member or are not in the same plane, the allowable undercut depth of 1/32 inch may be increased to 1/16 inch for the full length of the weld.

2.5.2.8 Surface Porosity: Only surface porosity whose major surface dimension exceeds 1/16 inch shall be considered relevant. Fillet and groove welds which contain surface porosity shall be considered unacceptable if:

2.5.2.8.1 The sum of diameters of random porosity exceeds 3/8 inch in any linear inch of weld or 3/4 inch in any 12 inches of weld; or

2.5.2.8.2 Four or more pores are aligned and the pores are separated by 1/16 inch or less, edge to edge.

2.5.2.9 Weld Length and Location: The length and location of welds shall be as specified on the detail drawing, except that weld lengths may be longer than specified. For weld lengths less than 3 inches, the permissible underlength is 1/8 inch and for welds 3 inches and longer the permissible underlength is 1/4 inch. Intermittent welds shall be spaced within 1 inch of the specified location.

2.5.2.10 Arc Strikes: Arc strikes and associated blemishes are acceptable provided no cracking is visually detected.

2.5.2.11 Surface Slag and Weld Spatter: Slag whose major surface dimension is 1/8 inch or less is irrelevant. Isolated surface slag that remains after weld cleaning and which does not exceed 1/4 inch in its major surface dimension, is acceptable. (Slag is considered to be isolated when it does not occur more frequently than once per weld or more than once in a 3 inch weld segment.) Spatter remaining after the cleaning operation is acceptable.

SECTION 3.0

INSPECTION GUIDELINES

3.0 INSPECTION GUIDELINES

These Inspection Guidelines are to be used for visual inspection of structural welds made in accordance with the provisions of AWS D1.1. These guidelines provide background information and instructions to assist the Inspector in evaluating weld attributes. Measuring techniques and guidance on the accuracy, frequency, and locations for measuring welds are discussed. It is important for the Inspector to understand weld size tolerances and significant measurement units in order to preclude rejection of adequate welds.

Effective implementation of the Acceptance Criteria requires that all Inspectors evaluating any weld use the same inspection techniques, acceptance criteria, and measurement accuracy. The Inspector's duties, inspection philosophy, and inspection guidelines are presented with this goal in mind.

3.1 The Inspector

3.1.1 As used herein, the Inspector is that person performing acceptance inspections of completed welds.

3.1.2 The Inspector using these Acceptance Criteria is to perform inspections in accordance with the guidelines provided in this document. Section 6 of AWS D1.1 addresses some inspection provisions for structural welding, but they are applicable to the in-process welding control of the project.

3.2 General Inspection Requirements

3.2.1 Weld inspection and tests are performed as necessary prior to assembly, during assembly, during welding, and after welding to ensure that materials and workmanship meet the requirements of contract documents. It is the responsibility of the contractor to assure that the workmanship standards and in-process controls of AWS D1.1 are met as appropriate.

3.2.2 In nuclear construction, in-process workmanship may be monitored and audited by the Quality Control group as required by the constructor's Quality Assurance Program. The Acceptance Criteria of this document are to be used to determine the adequacy of completed structural welds.

3.2.3 The inspection of structural welds to the Acceptance Criteria of this document shall be documented in accordance with project requirements. The marking of welds during in-process work is not required as part of the acceptance inspections addressed by this document.

3.3 Inspection Philosophy

3.3.1 The workmanship provisions of Section 3 of AWS D1.1 are necessary for in-process quality control purposes. The determination of whether corrective action is necessary to assure good workmanship during

production shall be made in accordance with the in-process welding control of the project.

3.3.2 Inspections subsequent to the acceptance inspection are intended to evaluate and verify that previous inspections were competently performed and to demonstrate compliance with design requirements. They are not intended to upgrade or downgrade the level of workmanship or impose more stringent criteria or examination methods.

3.4 Inspection Principles

The following identifies general inspection principles that will enable Inspectors to consistently inspect welds to verify compliance with the Acceptance Criteria.

3.4.1 Each weld attribute shall be inspected and evaluated independently. It is not necessary to consider cumulative effects.

3.4.2 Acceptance inspections shall be performed promptly after welding has been completed so that deficiencies, if any, may be identified and resolved in a timely manner.

3.4.3 Visual inspection of welds is normally performed on the as-welded surface after the weld cleaning operation.

3.4.4 Measuring devices should be graduated in increments compatible with the applicable significant unit, e.g., 1/16 inch increments, rather than decimals. When taking or recording measurements, measurements should be rounded off to the nearest significant unit.

3.4.5 Lighting, natural or artificial, shall be of sufficient intensity and placement to illuminate the area being examined. Lighting shall be considered adequate when, for example, the Inspector can resolve a black line 1/32 inch wide or less on an 18 percent neutral gray card placed on the surface to be inspected. Backlighting is not required. Visual inspection may be aided by using a flashlight.

3.4.6 Applicable drawings for the structure shall be reviewed prior to inspection, and referred to during inspection, as necessary.

3.4.7 These Acceptance Criteria are to be used for the acceptance inspection of welds in the uncoated condition. These criteria may also be used for subsequent inspection after the welds have been coated, with the concurrence of the Engineer. Subsequent inspections related to suspected weldment cracking may require the removal of the coating or the use of appropriate magnetic particle inspection.

3.5 Weld Attribute Descriptions, Acceptance Criteria and Inspection Guidelines

This Section includes a definition or description of each weld attribute to be inspected, identifies the weld Acceptance Criteria to be applied, and provides specific inspection guidelines.

3.5.1 Weld Cracks

3.5.1.1 Description of Weld Cracks: Weld cracks are discontinuities characterized by a sharp tip and high ratio of length and depth to opening displacement.

3.5.1.2 Acceptance Criteria: The weld shall have no cracks.

3.5.1.3 Inspection Guidelines: Weld cracks shall be identified by visually examining the weld and heat affected zone. Visually detected cracks shall be identified for repair. The use of magnification devices for detection of cracks by visual examination is not required. However, additional lighting, such as a flashlight, or magnifiers may be appropriate for further investigation of suspected cracks.

3.5.2 Fillet Weld Size

3.5.2.1 Description of Fillet Weld Undersize: Fillet weld undersize occurs when the amount of weld metal deposited results in a size that is significantly less than specified.

The undersize condition may occur in the legs or the throat of fillet welds, as shown in Figure 3-1.

3.5.2.2 Acceptance Criteria: A fillet weld shall be permitted to be less than the size specified by 1/16 inch for 1/4 the length of the weld. Oversized fillet welds shall be acceptable if the oversized weld does not interfere with mating parts.

3.5.2.3 Inspection Guidelines: The Inspector shall measure weld size using appropriate measuring devices. Potentially undersized areas are the most appropriate locations for measuring weld size. Such areas can usually be determined by viewing the weld length. Areas of the weld that appear to be significantly undersized should be examined based on the requirements of the Acceptance Criteria.

Scales or fillet weld gages may be used for measuring the size of welds. Measurement increments of 1/16 inch are appropriate for these gages. The Inspector has the responsibility for assuring that the weld meets the Acceptance Criteria for size, but

continuous measurement of size over the full length of the weld is not mandatory and weld size measurements should be rounded off to the nearest 1/16 inch.

3.5.3 Incomplete Fusion

3.5.3.1 Description of Incompletion Fusion: Incomplete fusion is a condition in which coalescence of weld metal and base metal (or previous passes of filler metal) did not occur.

3.5.3.2 Acceptance Criteria: In fillet welds, incomplete fusion of 3/8 inch in any 4 inch segment, and 1/4 inch in welds less than 4 inches long, is acceptable. For groove welds, incomplete fusion is not acceptable. For fillet and groove welds, rounded end conditions that occur in welding (starts and stops) shall not be considered indications of incomplete fusion and are irrelevant.

3.5.3.3 Inspection Guidelines: Inspection for incomplete fusion shall be performed by visually examining the weld. The wetting and flow of weld metal at the fusion line is the best indication of fusion. Incomplete fusion in fillet welds which does not exceed 3/8 inch in any 4 inch segment of weld or 1/4 inch in welds less than 4 inches long is acceptable. Visually detected incomplete fusion in groove welds is not acceptable.

Indications of incomplete fusion at the start and stop of a weld which are observed only at the weld root are not indicative of the fusion in the main fun of the weld. (See Figure 3-2.) This apparent incomplete fusion is the result of transient conditions and should be considered acceptable. These brief transients affect only a very short portion of the weld and are not a cause for rejection because the main run of the weld is not affected. Measurements of incomplete fusion length should be rounded off to the nearest 1/8 inch.

3.5.4 Weld Overlap

3.5.4.1 Description of Weld Overlap: Weld overlap is the protrusion or rollover of weld metal over the adjacent base metal at the toe of the weld.

Fillet and Groove Weld overlap is shown in Figure 3-3.

3.5.4.2 Acceptance Criteria: Overlap is acceptable provided the criteria for weld size and fusion can be satisfied. When fusion in the overlap length cannot be verified, an overlap length of 3/8 inch in

any 4 inch segment, and 1/4 inch in welds less than 4 inches long, is acceptable.

3.5.4.3 Inspection Guidelines: The area of any overlap should be inspected to assure weld size and fusion requirements are met. Generally, in an overlap condition, if incomplete fusion occurs, it will be short in length. The wetting and flow of weld metal at the fusion line is the best indication of fusion. Portions of individual weld ripples or weave patterns which overlap are not a concern when there is fusion on each side. Where fusion cannot be verified, the unverified length should not exceed 3/8 inch in any 4 inch segment or 1/4 inch in welds less than 4 inches long. Overlap in excess of 3/8 inch (or 1/4 inch) length is acceptable when both fusion and weld size can be verified. Measurements of overlap length should be rounded off to the nearest 1/8 inch.

3.5.5 Underfilled Craters

3.5.5.1 Description of Underfilled Craters: A crater is a depression at the termination of a weld bead. Underfilled craters are small areas in which the nominal weld size has not been achieved.

3.5.5.2 Acceptance Criteria: Underfilled craters shall be acceptable provided the criteria for weld size are met. Craters which occur outside the specified weld length are irrelevant provided there are no cracks.

3.5.5.3 Inspection Guidelines: The length of the weld should be visually examined to locate the craters. Craters are to be evaluated using the acceptance criteria for undersized welds. Craters should be visually inspected for cracks.

3.5.6 Weld Profiles

3.5.6.1 Descriptions Applicable to Weld Profiles:

Description of Convexity and Concavity:

Convexity is the maximum distance from the face of a convex fillet weld perpendicular to a line joining the weld toes. Concavity is the maximum distance from the face of a concave fillet weld perpendicular to a line joining the weld toes.

Description of Groove Weld Reinforcement:

Groove weld reinforcement is the weld metal added in excess of the quantity needed to fill the weld joint. It increases the joint thickness.

3.5.6.2 Acceptance Criteria:

3.5.6.2.1 The faces of fillet welds may be convex, flat, or concave provided the criteria for weld size are met.

3.5.6.2.2 The faces of groove welds may be flat or convex.

3.5.6.2.3 Convexity of fillet and groove welds are not criteria for acceptance and need not be measured.

3.5.6.2.4 The thickness of groove welds is permitted to be a maximum of 1/32 inch less than the thinner member being joined.

3.5.6.3 Inspection Guidelines: Fillet weld convexity is controlled during the welding process and need not be measured as part of the acceptance inspection. Concavity is acceptable provided the requirements for weld size are met.

Groove weld reinforcement is controlled during the welding process and need not be measured as part of the acceptance inspection.

When evaluating groove weld thickness, the 1/32 inch undersize allowance is a maximum value. It is equivalent to rounding off to the nearest 1/16 inch; that is, if the thickness of a groove weld is 1/32 inch or less than the specified thickness, it is acceptable; if it is more than 1/32 inch less than the specified thickness, it is to be rejected.

3.5.7 Undercut

3.5.7.1 Description of Undercut: Undercut appears as a groove melted into the base metal adjacent to the toe or root of a weld and is an area left unfilled by weld metal. Undercut may occur in any length from a fraction of an inch to the full length of the weld.

Fillet and groove weld undercut is shown in Figure 3-4.

3.5.7.2 Acceptance Criteria:

3.5.7.2.1 For material 3/8 inch and less nominal thickness, undercut depth of 1/32 inch on one side for the full length of the weld, or 1/32 inch on one side for 1/2 the length of the weld and 1/16 inch for 1/4 the length of the weld on the same side of the member, is acceptable. For members welded on both sides where undercut exists in the same plane of a

member, the cumulative lengths of undercut shall be limited to the lengths of undercut allowed on one side. Melt-through that results in a hole in the base metal is unacceptable.

3.5.7.2.2 For materials greater than 3/8 inch nominal thickness, undercut depth of 1/32 inch for the full length of the weld and 1/16 inch for 1/4 the length of the weld on both sides of the member is acceptable. When either welds or undercut exist only on one side of the member or are not in the same plane, the allowable undercut depth of 1/32 inch may be increased to 1/16 inch for the full length of the weld.

3.5.7.3 Inspection Guidelines: The criteria for determining the acceptance of undercut are depth and length. The method to be used for determining acceptable depth is primarily visual. Potentially rejectable areas may be evaluated using a scale, comparative sample, or suitable gage to determine the depth of undercut. Melt-through is unacceptable.

These Acceptance Criteria allow undercut for the full length of the weld on one side of a member. However if undercut exists for the full length, it may be indicative of a welding process control problem, and should be brought to the attention of appropriate personnel.

The acceptance criteria refer to situations where undercut may exist on both sides of the member and in the same plane. Sketch A of Figure 3-4 illustrates undercut on only one side of a member. Sketch B shows undercut on both sides of a member and in the same plane. Sketch C is an example of undercut on both sides of a member but not in the same plane. In Sketch B, the total length of undercut is determined by adding together the length of undercut on each side of the member; for material 3/8 inch and less nominal thickness the total length of undercut should not exceed the length of undercut that would be allowed if there was undercut on only one side of the member. In Sketch C, the undercut on one side is not in the same plane as the undercut on the opposite side of the member and therefore, the length of undercut at each location is to be evaluated independently.

Undercut depth is to be estimated to the nearest 1/32 inch and undercut length is to be estimated to determine compliance with the length of undercut criteria. There are significant margins in these criteria so that it is not necessary to sum isolated intermittent undercut (i.e., 1/4 inch here, 5/8 inch there) to develop the total length of undercut.

3.5.8 Surface Porosity

3.5.8.1 Description of Surface Porosity: Porosity consists of discontinuities formed by gas entrapment during solidification.

Porosity appears on the surface of welds as circular voids. The voids are caused by gas in the molten weld material which collects in discrete bubbles. The bubbles, moving toward the weld surface while it is still molten, form voids which are trapped as the metal solidifies. Gas bubbles merge and varying void sizes result. This type of porosity is frequently referred to as “pin holes” when the porosity extends to the weld surface.

3.5.8.2 Acceptance Criteria: Only surface porosity whose major surface dimension exceeds 1/16 inch shall be considered relevant. Fillet and groove welds which contain surface porosity shall be considered unacceptable if:

3.5.8.2.1 The sum of diameters of random porosity exceeds 3/8 inch in any linear inch of weld or 3/4 inch in any 12 inches of weld; or

3.5.8.2.2 Four or more pores are aligned and the pores are separated by 1/16 inch or less, edge to edge.

3.5.8.3 Inspection Guidelines: Porosity voids are generally less than 1/16 inch in diameter. Thus, relatively few are relevant. Very seldom should it be necessary to use any measuring device, but when there is concert, a 1/16 inch wire and a tape or scale are adequate.

Estimating the number of random voids which appear to be over 1/16 inch in diameter within one inch and twelve inch spans will usually provide adequate information for weld acceptance.

When evaluating potentially rejectable aligned porosity, a 1/16 inch diameter wire can be used to measure both the size of the pores and the edge to edge spacing of the pores. If the size of a pore is 1/16 inch or less it is irrelevant and need not be counted; if the spacing between any two relevant pores exceeds 1/16 inch, those pores are irrelevant for the purpose of evaluating aligned porosity.

3.5.9 Weld Length and Location

3.5.9.1 Description of Weld Length and Location: Weld length and location are an expression of the design and detail drawing

requirements specifying the length of welds and the physical location of welds on the structure.

3.5.9.2 Acceptance Criteria: The length and location of welds shall be as specified on the detail drawing, except that weld lengths may be longer than specified. For weld lengths less than 3 inches, the permissible underlength is 1/8 inch and for welds 3 inches and longer the permissible underlength is 1/4 inch. Intermittent welds shall be spaced within 1 inch of the specified location.

3.5.9.3 Inspection Guidelines: Welds should be visually inspected to determine that they are in the locations shown on the design drawings. Weld length should be measured to the tolerances designated in the drawings. When the design drawings do not provide tolerances for weld length and location, an underlength of 1/8 inch is acceptable for welds less than 3 inches and 1/4 inch for welds 3 inches or longer. Welds may be longer than specified.

Welds are usually designated as full length of the member or continuous. When the length of a weld is specified, it usually applies to intermittent welding, such as, “3 inches long at 12 inches center-to-center,” or “6 inches long at 12 inches center-to-center.” The total sum of the lengths of the intermittent welds is more important than the actual length of the individual intermittent welds. Either way, a 1/8 inch or 1/4 inch tolerance, as applicable to segment length is acceptable if no other tolerance is given.

Intermittent welds may be located within 1 inch of the designated location.

Measurements shall be rounded off to the nearest 1/16 inch for size; to the nearest 1/8 or 1/4 inch for length; and to the nearest inch for location of intermittent welds.

3.5.10 Arc Strikes

3.5.10.1 Description of Arc Strikes: An arc strike is an area of base metal struck by inadvertent arcing from the welding electrode, which may cause a small amount of base metal melting in the impingement region. Arc strikes sometimes cause a discoloration of the base metal.

3.5.10.2 Acceptance Criteria: Arc strikes and associated blemishes are acceptable provided no cracking is visually detected.

3.5.10.3 Inspection Guidelines: No special inspection is required for detection of arc strikes. Arc strikes are usually found near the weld and the Inspector should review them while performing

inspections for other weld attributes. Arc strikes and the associated blemishes should be visually examined for cracks.

3.5.11 Surface Slag and Weld Spatter

3.5.11.1 Description of Surface Slag and Weld Spatter: Surface slag is the nonmetallic residue remaining from the reactions occurring in the arc and weld pool during welding. Spatter consists of metal particles expelled during fusion welding that do not form a part of the weld. Slag and spatter are removed during the cleaning operation by brushing, grinding, chipping, or other suitable means.

3.5.11.2 Acceptance Criteria: Slag whose major surface dimension is 1/8 inch or less is irrelevant. Isolated surface slag that remains after weld cleaning and which does not exceed 1/4 inch in its major surface dimension, is acceptable. (Slag is considered to be isolated when it does not occur more frequently than once per weld or more than once in a 3 inch weld segment.) Spatter remaining after the cleaning operations is acceptable.

3.5.11.3 Inspection Guidelines: Inspection for surface slag and weld spatter should be performed by visually examining the length of the weld and adjacent base metal areas. Any areas not meeting the acceptance criteria should be cleaned again and re-examined. (Cleaning required for subsequent processing, such as painting or NDE, is not addressed here.)

3.6 Significant Measurement Units

3.6.1 Measurement units are identified in Table 2-1.

3.6.2 It is not appropriate to mix or interchange fractional and decimal dimensions or tolerances. For example, undercut must not be measured in terms of hundredths or thousandths of an inch. When acceptance limits are given in fractions of an inch, all measuring tools must also be graduated in fractions of an inch. Decimal gages are only to be used when dimensions or tolerances are explicitly specified in decimals.

3.7 Equivalency

These Acceptance Criteria have been developed based on the effect that discontinuities may have on the reduction of cross-sectional area. The Acceptance Criteria for some discontinuities (such as weld size, incomplete fusion, weld overlap, undercut and surface porosity) provide limits based on both depth or surface dimensions and length. It is intended that the Engineer may provide for acceptance of welds on an equivalency basis; for example, in material 3/8 inch or less, in nominal thickness, undercut depth that exceeds 1/32 inch may

be acceptable up to 1/16, if the length of undercut is proportionately less. Equivalency is meant as a means for the Engineer to evaluate a weld condition documented by the Inspector as not meeting the acceptance criteria.

The Inspector must base his acceptance of welds on the criteria specified and provided by the Engineer.

SECTION 4.0

COMMENTARY

4.0 COMMENTARY

4.1 Criteria Evaluation

This Section contains engineering evaluations of the Acceptance Criteria given in Section 2.0. In general, it is important to recognize that the engineering design of structures and weld connections is based on conservatively estimated loads. The allowable stresses for materials given in Codes are also conservative. Code allowable stresses have been chosen by applying a "safety factor" or "design factor" to compensate for design and analysis techniques and assumptions, fabrication tolerances, inspection techniques, variables in material properties, and other unknowns. For these reasons, there is no need to modify allowable stresses or design techniques when implementing the Acceptance Criteria of this document.

Because fatigue is not a concern for the structures addressed in this document, the evaluation of the Acceptance Criteria is primarily based on an allowable reduction of area consistent with construction Codes and Standards. Welding Research Council Bulletin 222, "The Significance of Weld Discontinuities" reports that geometrical discontinuities are influential in static behavior only to the extent they reduce cross-sectional area.

4.1.1 Weld Cracks

4.1.1.1 Acceptance Criteria: The weld shall have no cracks.

4.1.1.2 Discussion: Visually detected cracks in welds are unacceptable

4.1.2 Fillet Weld Size

4.1.2.1 Acceptance Criteria: A fillet weld shall be permitted to be less than the size specified by 1/16 inch for 1/4 the length of the weld. Oversized fillet welds shall be acceptable if the oversized weld does not interfere with mating parts.

4.1.2.2 Discussion: Undersized fillet welds could cause increased stress in the throat of the weld and a subsequent reduction of the load carrying capacity of the weld. Therefore it is appropriate to review the increase in stress due to the allowable reduction of weld size.

Evaluation of the Acceptance Criteria for a 3/16 inch fillet weld results in a potential reduction in shear area of about 8%. (See paragraph 4.3 for a sample calculation of the effect on shear area due to undersized welds.)

This evaluation is based on the assumption that both legs are reduced equally, whereas the condition generally found is that

one leg is reduced and the other is not. In addition, no consideration is given for either convexity or penetration, both of which may increase the weld shear area. Also, no credit is taken for actual weld metal strength, which is generally about 10% greater than specified.

4.1.3 Incomplete Fusion

4.1.3.1 Acceptance Criteria: In fillet welds, incomplete fusion of 3/8 inch in any 4 inch segment, and 1/4 inch in welds less than 4 inches long, is acceptable. For groove welds, incomplete fusion is not acceptable. For fillet and groove welds, rounded end conditions that occur in welding (starts and stops) shall not be considered indications of incomplete fusion and are irrelevant.

4.1.3.2 Discussion: Incomplete fusion describes a weld zone region within which the desired coalescence of base metal and weld metal did not take place due to the absence of complete melting. An important aspect of structural welding is the welding procedure and the performance test which all welders must execute satisfactorily. These are the industry accepted primary safeguards for prevention of unacceptable incomplete fusion.

The use of an acceptance criterion which allows an incomplete fusion length of 3/8 inch in any 4 inch segment is equivalent to a maximum potential reduction of load carrying area of less than 10% (9.4%). Furthermore, the 9.4% calculated reduction in the strength of any 4 inch segment of a fillet weld that contains a 3/8 inch length of incomplete fusion (where fusion cannot be readily verified) is based on the assumption of no weld fusion for the width of the indication and the full depth of the weld. This is considered to be an extremely conservative assumption because the condition is local and usually affects only a minor portion of one weld bead; that is, the condition does not affect the full depth of the weld. Such indications (i.e., short lengths of visually detected incomplete fusion), when further investigated, generally exhibit a high percentage of complete fusion.

Indications of incomplete fusion at the starts and stops of welds are usually observed only at the weld root and are not indicative of the fusion in the main run of the weld, provided fusion is evident at the weld toes. (See Figure 3-2.) This apparent incomplete fusion is the result of transient conditions and is acceptable. These brief transients affect only a very short portion of the weld at the start and stop, and are not a cause for rejection because the main run of the weld is not affected.

4.1.4 Weld Overlap

4.1.4.1 Acceptance Criteria: Overlap is acceptable provided the criteria for weld size and fusion can be satisfied. When fusion in the overlap length cannot be verified, an overlap length of 3/8 inch in any 4 inch segment, and 1/4 inch in welds less than 4 inches long, is acceptable.

4.1.4.2 Discussion: Weld overlap or rollover may occur intermittently or as a continuous condition. Short lengths of localized or intermittent overlap at weld ripples, weld weaves or at bead starts is not a concern when the adjacent portions of a weld are fused to the base material. Fusion to the base metal is a function of the arc energy and the heat content of the molten weld pool. Neither of these factors change rapidly enough to adversely affect fusion in a short length. A short length of overlap is evidence of a localized variation in welding technique that alters the weld surface appearance, but not the underlying weld deposit.

Overlap of 3/8 inch in any 4 inch segment results in a maximum potential reduction of load carrying capacity of the weld of less than 10% (9.4%). The basis and assumptions for this calculated reduction of 9.4% due to overlap are the same as those for incomplete fusion. (See paragraph 4.1.3.2.) Thus, the evaluation of the Acceptance Criteria for overlap is consistent with the evaluation for incomplete fusion. This is desirable, because at times there is difficulty in distinguishing between overlap and incomplete fusion. With equivalent acceptance criteria, any difference is immaterial.

4.1.5 Underfilled Craters

4.1.5.1 Acceptance Criteria: Underfilled craters shall be acceptable provided the criteria for weld size are met. Craters which occur outside the specified weld length are irrelevant provided there are no cracks.

4.1.5.2 Discussion: Weld craters are slightly underfilled areas sometimes found at weld stops. A crater may constitute a short length of weld undersize. The justification for undersized fillet welds is given in Section 4.1.2.

4.1.6 Weld Profiles

4.1.6.1 Acceptance Criteria:

4.1.6.1.1 The faces of fillet welds may be convex, flat, or concave provided the criteria for weld size are met.

4.1.6.1.2 The faces of groove welds may be flat or convex.

4.1.6.1.3 Convexity of fillet and groove welds are not criteria for acceptance need not be measured.

4.1.6.1.4 The thickness of groove welds is permitted to be a maximum of 1/32 inch less than the thinner member being joined.

4.1.6.2 Discussion: Fillet weld convexity and groove weld reinforcement do not adversely affect weldments for these structural applications, but removal is not necessarily beneficial to suitability for service, and could create other problems.

The undersize allowances of up to 1/32 inch for groove welds is consistent with the requirements of AWS D1.1, paragraph 3.6.3 for flush butt joints and is equivalent to rounding off to the nearest 1/16 inch.

4.1.7 Undercut

4.1.7.1 Acceptance Criteria:

4.1.7.1.1 For material 3/8 inch and less nominal thickness, undercut depth of 1/32 inch on one side for the full length of the weld, or 1/32 inch on one side for 1/2 the length of the weld and 1/16 inch for 1/4 the length of the weld on the same side of the member, is acceptable. For members welded on both sides where undercut exists in the same plane of a member, the cumulative lengths of undercut shall be limited to the lengths of undercut allowed on one side. Melt-through that results in a hole in the base metal is unacceptable.

4.1.7.1.2 For materials greater than 3/8 inch nominal thickness, undercut depth of 1/32 inch for the full length of the weld and 1/16 inch for 1/4 the length of the weld on both sides of the member is acceptable. When either welds or undercut exist only on one side of the member or are not in the same plane, the allowable

undercut depth of 1/32 inch may be increases to 1/16 inch for the full length of the weld.

4.1.7.2 Discussion: Weld undercut is caused by melting away of base metal at the toe of a fillet weld or groove weld. Thus, undercut is best characterized as a broad shallow discontinuity rather than a crack-like defect. Undercut is a base metal condition rather than weld metal condition.

For some structures, undercut limitations have been based upon concern for fatigue loading and other factors which by scope definition are not a concern for the service conditions being covered by these Acceptance Criteria. In some cases, because of plant conditions, the Engineer may specify the alternative criteria.

Undercut develops when base metal is melted and flows under the effects of gravity and/or surface tension, without being replaced by weld filler metal. Melting and flow cause the defect to have a shallow, broad contour, as limited by the molten metal surface tension and viscosity. The broad shallow nature of undercut will permit stress redistribution in that area.

Because fatigue is not a concern for the structures addressed in this document, undercut allowance is properly based on potential reduction of area. There is significant technical support for this approach in addressing the acceptability of undercut. Principal reference in this regard is Welding Research Council Bulletin 222, “The Significance of Weld Discontinuities.” Bulletin 222 reports that geometrical discontinuities are influential in static behavior only to the extent that they reduce the cross-sectional area.

The NCIG undercut criteria were evaluated considering the effect of undercut on the net load carrying base metal area. Table 4-1 shows the effect of the undercut allowed by the Acceptance Criteria of this document on area reduction for several thicknesses of material. (See Section 4.3 for a sample calculation for the effect of undercut.)

Figure 8.15.1.5 of AWS D1.1 specifies permissible undercut values for various thicknesses of materials and principal tensile stress conditions. Table 4-1A provides a comparison of the percent reduction in area allowed based on the NCIG Acceptance Criteria and the requirements of AWS D1.1 for various material thicknesses.

TABLE 4-1

EFFECT OF UNDERCUT ON REDUCTION OF CROSS SECTION

Material Thickness (Inches)	Allowable Undercut (Depth and Length	Maximum Area Change (%)
3/16	1/32 for L (one side) or 1/32 for L/2 + 1/16 for L/4 (one side)	16.7
1/4	(same as for 3/16")	12.5
5/16	(same as for 3/16")	10.0
3/18	(same as for 3/16")	8.3
7/16	1/16 for L/4 + 1/32 for 3L/4 (both sides)	17.9
5/8	(same as for 7/16")	12.5

L is the full weld length.

TABLE 4-1A

COMPARISON OF THE EFFECT OF UNDERCUT ON PERCENT REDUCTION OF AREA

Material Thickness	NCIG	AWS D1.1		
		No Calculated Stress	Stress Parallel to Undercut	Stress Perpendicular to Undercut
3/16"	16.7	24.9	12.4	12.4
1/4"	12.5	28.0	18.7	9.3
5/16"	10.0	28.0	22.4	7.5
3/8"	8.3	27.8	18.7	6.2
7/16"	17.9	23.0	16.0	5.3
5/8"	12.5	20.0	11.2	3.7

From the above comparison, the reduction of area based on the NCIG acceptance criteria for undercut is comparable to the reduction of area permitted by AWS D1.1.

4.1.8 Surface Porosity

4.1.8.1 Acceptance Criteria: Only surface porosity whose major surface dimension exceeds 1/16 inch shall be considered relevant. Fillet and groove welds which contain surface porosity shall be considered unacceptable if:

4.1.8.1.1 The sum of diameters of random porosity exceeds 3/8 inch in any linear inch of weld or 3/4 inch in any 12 inches of weld; or

4.1.8.1.2 Four or more pores are aligned and the pores are separated by 1/16 inch or less, edge to edge.

4.1.8.2 Discussion: Steel weld metal porosity is usually caused by improper cleaning of the base metal or contamination of the weld metal, and additionally, by inadequate shielding of the welding arc. Porosity is created when the molten steel solidifies before all the gas has issued from the liquid weld metal.

Numerous publications indicate that the presence of porosity in amounts of 5% to 7% has an insignificant influence on weld strength in non-fatigue stress applications, and that 10% porosity does not diminish weld metal strength below the minimum specified values. If the porosity is assumed to extend through the full depth of the weld, the 3/4 inch in 12 inches represents 6% of a weld, therefore there is a significant margin before porosity becomes a concern. Since this is surface porosity, the effect is really insignificant.

The conclusion that porosity less than 1/16 inch in diameter is irrelevant is consistent with the provisions of ASME Section III, Subsection NF, paragraph NF-5360(a); the criteria for aligned porosity is consistent with paragraph NF 5360(a)(4).

4.1.9 Weld Length and Location

4.1.9.1 Acceptance Criteria: The length and location of welds shall be as specified on the detail drawing, except that weld lengths may be longer than specified. For weld lengths less than 3 inches, the permissible underlength is 1/8 inch and for welds 3 inches and longer the permissible underlength is 1/4 inch. Intermittent welds shall be spaced within 1 inch of the specified location.

4.1.9.2 Discussion: Welds have been rejected when their length is slightly more or less than specified on construction drawings. The purpose of verifying that welds are of proper length and location is to assure that the welds are of proper length and

location is to assure that the welds are capable of transmitting loads through the structure in accordance with the design.

Welds are most often designated on drawings as being the full length of the member or continuous around the member. Specific weld lengths are designated for intermittent welding. The total sum of the lengths of the intermittent welds is more important than the actual length oft he individual intermittent welds. Intermittent welds that are longer than specified increase total weld length, reduce stress, and are acceptable.

For the minimum allowable intermittent fillet weld length of 1-1/2 inch, the 1/8 inch permitted underlength results in an area reduction of about 8%. A similar reduction, about 8% occurs for the 1/4 inch permitted underlength applied to a 3 inch weld. This effect decreases with increasing weld length, being less than 5% for welds over 5 inches in length. This reduction of area is consistent with the reductions of area allowed for other weld attributes. (See Section 4.3 for a sample calculation of the effect of weld underlength.)

In general, welds should be located as shown on drawings. However, for welds of specific length, such as intermittent welds, a linear offset in location by as much as 1 inch is not considered to be significant because the load will still be properly transmitted through the structure.

4.1.10 Arc Strikes

4.1.10.1 Acceptance Criteria: Arc strikes and associated blemishes are acceptable provided no cracking is visually detected.

4.1.10.2 Discussion: Arc strikes are caused by inadvertent arcing between the welding electrode and the base metal, causing a small amount of base metal melting in the arc impingement region. More than ordinary attention is sometimes given arc strike areas because of potential hardness and the possibility of cracking.

For the structural steels to which this document applies, the maximum hardness of the arc strike region is limited by the relatively low carbon equivalents of the materials. Recently published work[(g)] has shown that cracking does not occur in arc strikes on materials of this class, and concludes that since

(g). Van Malssen, S.H., "The Effects of Arc Strikes on Steels Used in Nuclear Construction," Welding Journal, July 1984.

uncorrected arc strikes do not affect the strength of the material, they should not be of concern when the design is not governed by fatigue. The article states one of the problems suspected to be associated with arc strikes is the possibility of cracking occurring as a result of the rapid heating and cooling of the material. The results documented in the article demonstrate that the suspected cracking does not occur.

The article also discusses potential encroachment on required thickness by arc strikes and concludes that such encroachment on the required thickness by arc strikes is unlikely.

4.1.11 Surface Slag and Weld Spatter

4.1.11.1 Acceptance Criteria: Slag whose major surface dimension is 1/8 inch or less is irrelevant. Isolated surface slag that remains after weld cleaning and which does not exceed 1/4 inch in its major dimension, is acceptable. (Slag is considered to be isolated when it does not occur more frequently than once per weld or more than once in a 3 inch weld segment.) Spatter remaining after the cleaning operation is acceptable.

4.1.11.2 Discussion: Sometimes removal of all slag from completed welds requires extraordinary effort since occasional tightly adhering slag in small areas requires the use of pneumatic chisels or grinders to completely remove the slag. This usually results in a less presentable surface and removal may cause more harm than good. It is good practice to remove surface slag by wire brushing or chipping hammers, but there are limits as to what can be accomplished with such tools. It is not considered a necessity to remove small tightly adherent areas of slag since it has no effect on the weld. The amount of slag permitted by the Acceptance Criteria should not mask any significant condition of concern.

The basic purpose for removing weld spatter is to clean the area so it can be properly inspected and painted. Tightly adhering weld spatter is not harmful to the weld or base metal.

4.2 Cumulative Effects

4.2.1 Eleven attributes are considered in this document. The cumulative effect that deficiencies in any one of these attributes may, or may not have, on any other attribute is summarized in Table 4-2. It is concluded that it is not necessary to consider cumulative effects when evaluating welds in these

types of structures and therefore each weld attribute should be inspected and evaluated independently.

4.2.2 The justification for not considering the cumulative effects of any of the potentially negative combinations that are possible, follows:

4.2.2.1 Both base metal and weld metal strength are significantly higher than minimum specified values.

4.2.2.2 Small size welds are rarely loaded to allowable stress limits.

4.2.2.3 Convexity and reinforcement do not reduce the load carrying capacity of the weld.

4.2.2.4 These structures are designed using conservative load definitions, load combinations, analytical techniques, and design methods.

4.2.2.5 Designers round-up to the next larger fillet weld size in 1/16 in. increments.

4.2.2.6 Member sizes and material thicknesses are selected by the designer based on manufacturer's standards or material availability. As a result, these structures are usually fabricated using material that exceeds the size or thickness needed to satisfy design requirements for area or load carrying capability.

4.2.3 Weld Underlength and Fillet Weld Undersize: It is possible that these attributes could combine to decrease the load capacity of a weld. However, underlength is a consideration that is generally applicable only to intermittent welds and such welds are not common in these types of structures.

TABLE 4-2

COMMENTS APPLICABLE TO CUMULATIVE EFFECTS

Weld Attributes	Comment
Weld Cracks	Not allowed, therefore not cumulative.
Fillet Weld Undersize	Could combine with weld underlength (refer to Section 4.2.3 for Discussion).
Incomplete Fustion	Local condition which does not produce a general reduction in strength.
Weld Overlap	Local condition which does not produce a general reduction in strength.
Underfilled Craters	Weld size restrictions must be met; therefore, not cumulative.
Weld Profiles:	
Convexity and Groove Weld Reinforcement	Do not result in a reduction in weld or material strength or area.
Concavity	Weld size restrictions must be met; therefore not cumulative.
Undercut	This is a base metal condition which does not affect the weld.
Surface Porosity	Porosity does not adversely affect weld strength and therefore is not cumulative.
Weld Length	Could combine with fillet weld undersize (refer to Section 4.2.3 for Discussion).
Arc Strikes	Could result in minor reduction in base metal area, but is not a cumulative effect with other base metal considerations.
Surface Slag and Weld Spatter	Do not reduce weld strength or area.

On a worst case basis, if a fillet weld specified to be 3/16 inch by 3 inches long was undersized and underlength but met the Acceptance Criteria for both of these attributes, the total reduction in throat area compared to the throat area of the weld as specified, would be approximately 16 percent, which is reasonable on the basis of the conservatisms stated in 4.2.2. (See Section 4.3 for a sample calculation of percent reduction in throat area.)

4.2.4 Fillet Weld Undersize and Undercut: Fillet weld undersize is a condition which may affect the load carrying capability of the weld and undercut is a condition in the base material adjacent to the weld. These two effects do not occur at the same point and are not cumulative. The failure plane for a structure with fillet welds is through the weld. This failure plane is not in line with nor in close proximity to any plane affected by undercut. Thus the two attributes cannot combine nor bear upon one another.

4.2.5 Incomplete Fusion or Weld Overlap and Other Weld Attributes: The 9.4% calculated reduction in the strength of a 4 inch segment of a fillet weld that contains a 3/8 inch length of incomplete fusion or weld overlap (where fusion cannot be readily verified) is based on the assumption of no weld fusion for the width of the indication and the full depth of the weld. This is considered to be an extremely conservative assumption because the condition is local and usually affects only a minor portion of one weld bead; that is, the condition does not affect the full depth of the weld. Such indications (i.e., short lengths of visually detected incomplete fusion or weld overlap), when further investigated, generally exhibit complete fusion. Because the assumed condition is extremely rare, it is not necessary to consider the cumulative effect of either incomplete fusion or weld overlap with any other deficiency (such as weld underlength or weld undersize) when inspecting welds in these types of structures.

4.2.6 Other Weld Attributes: The other attributes are either not related or are clearly independent of each other.

4.3 Sample Calculations

4.3.1 Sample Calculation for the Combined Effect of Weld Underlength and Undersize

For a 3/16” X 3” long fillet weld that meets NCIG criteria for length and size:

Theoretical throat area:

= (S) (L) (.707)

= (3/16) (3) (0.707) = 0.398 $in.^2$

Actual throat area:

= (S) (L’) (.707) - (s) (L’) (%L’) (.707)

= (3/16) (2-3/4) (.707) -
(1/16) (2-3/4) (0.25) (.707)

= 0.365 - 0.030

$= 0.335 \text{ in.}^2$

$$\text{Area reduction} = \frac{0.398 - 0.335}{0.398}$$

$= 0.158$ or approximately 16%(h)

Where:

S = Specified Weld Size
L = Specified Weld Length
L^1 = Permissible Weld Length = L-1/4”
s = Permissible Undersize = 1/16”
$\%L^1$ = % of Weld Length that may be Undersize = 25%

4.3.2 Sample Calculation for the Effect of Undercut in 5/8 Inch Material

$$\text{Area Reduction} = \frac{(2)\,(u)\,(L/4) + (2)\,(u')\,(3L/4)}{(t)\,(L)}$$

$$= \frac{(2)\,(1/16)\,(L/4) + (2)\,(1/32)\,(3L/4)}{(5/16)\,(L)}$$

$$= \frac{0.031 + 0.047}{0.625}$$

$= 0.125$ or 12.5%

Where:

t = Base Metal Thickness

L = Specified Weld Length

u = Permissible undercut depth for L/4

u’ = Permissible undercut depth for 3L/4

4.3.3 Sample Calculation for the Effect of Undersize in a 3/16 Inch Fillet Weld

$$\text{Area Reduction} = \frac{(s)\,(L/4)\,(.707)}{(S)\,(L)\,(.707)}$$

(h). A calculation to determine the reduction in weld metal area along the heat affected zone (weld leg) would obviously yield an identical result. This same observation can be made for the results given in 4.3.3 and 4.3.4.

$$= \frac{(1/16)\ (L/4)\ (.707)}{(3/16)\ (L)\ (.707)}$$

$$= \frac{(0.0156)}{(0.1875)}$$

= 0.083 or 8.3%

Where:

s = Permissible Undersize

S = Specified Weld Size

L = Specified Weld Length

4.3.4 Sample Calculations for the Effect of Weld Underlength

For an intermittent fillet weld specified to be 1-1/2 inches long that meets the NCIG criteria for length:

Theoretical throat area:

= (S) (L) (.707)

= (S) (1.5) (.707)

= 1.061 (S) $in.^2$

Actual throat area:

= (S) (L') (.707)

= (S) (1.5 - 0.125) (.707)

= 0.972 (S) $in.^2$

$$\text{Area reduction} = \frac{1.061 - 0.972}{1.061}$$

= 0.083 or 8.3%

Where:

S = Specified Weld Size

L = Specified Weld Length

L' = Permissible Weld Length

For an intermittent weld specified to be 5 inches long that meets the NCIG criteria for length:

Theoretical throat area:

= (S) (5) (.707)

= 3.535 (S) $in.^2$

Actual throat area:

= (S) (5 - 0.25) (.707)

= 3.358 (S) $in.^2$

Area reduction = $\frac{3.535 - 3.358}{3.535}$

= 0.050 or 5%

3.9N MECHANICAL SYSTEMS AND COMPONENTS

3.9N.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9N.1.1 Design Transients

The following five operating conditions as defined in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are considered in the design of the Reactor Coolant System (RCS), RCS component supports, and reactor internals.

1. Normal conditions

   Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted or testing conditions.

2. Upset conditions (incidents of moderate frequency)

   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition shall be included in the design specifications.

3. Emergency conditions (infrequent incidents)

   Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not cause more than 25 stress cycles having an Sa value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code, Section III.

4. Faulted conditions (limiting faults)

   Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

5. Testing conditions

   Testing conditions are those pressure overload tests including hydrostatic tests, pneumatic tests, and leak tests specified. Other types of tests shall be classified under normal, upset, emergency or faulted conditions.

To provide the necessary high degree of integrity for the equipment in the RCS the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgement and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cycle behavior. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The following design conditions are given in the equipment specification for RCS components.

The design transients and the number of cycles of each that is normally used for fatigue evaluations are shown in Table 3.9N-1. In accordance with the ASME Code, Section III, emergency and faulted conditions are not included in fatigue evaluations.

The Components Cyclic or Transient Limits Program provides controls to track the FSAR Table 3.9N-1A, cyclic and transient occurrences to ensure that components are maintained within the design limits.

Normal Conditions

The following primary system transients are considered normal conditions:

1. Heatup and cooldown at 100°F per hour.
2. Unit loading and unloading at 5 percent of full power per minute.
3. Step load increase and decrease of 10 percent of full power.
4. Large step load decrease with steam dump.
5. Steady state fluctuations.
6. Feedwater cycling at hot shutdown.
7. Loop out of service.
8. Unit loading and unloading between 0 and 15 percent of full power.
9. Boron concentration equalization.
10. Refueling.

1. Heatup and cooldown at 100°F per hour

   The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F per hour. (These operations

can take place at lower rates approaching the minimum of 0°F per hour. The expected normal rates are 50°F per hour).

For these cases, the heatup occurs from ambient (assumed to be 120°F) to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

a. Material ductility considerations which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature, which are below the design rate of 100°F per hour.

b. Slower initial heatup rates when using pump energy only.

c. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

The number of such complete heatup and cooldown operations is specified as 200 each, which corresponds to five such occurrences per year for the 40 year plant design life.

2. Unit loading and unloading at 5 percent of full power per minute

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5 percent per minute between 15 percent load and full load. This load swing is the maximum possible, consistent with operation under automatic reactor control. The reactor temperature will vary with load as prescribed by the Reactor Control System. The number of loading and unloading operations is defined as 13,200. One loading operation per day yields 14,600 such operations during the 40 year design life of the plant. By assuming a 90 percent availability factor, this number is reduced to 13,200.

3. Step load increase and decrease of 10 percent of full power

The ±10 percent step change in load demand is a transient which is assumed to be a change in turbine control valve opening due to disturbances in the electrical network into which the plant output is tied. The Reactor Control System is designed to restore plant equilibrium without reactor trip following a ±10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 percent and 100 percent full load, the power range for automatic reactor control. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power

mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2000 times or 50 per year for the 40 year plant design life.

4. Large step load decrease with steam dump

This transient applies to a step decrease in turbine load from full power, of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator safety valves. Thus, since the Comanche Peak Nuclear Power Plant (CPNPP) is designed to accept a step decrease of 50 percent from full power the Steam Dump System provides the heat sink to accept 40 percent of the turbine load. The remaining 10 percent of the total step change is assumed by the Reactor Control System (control rods). If a Steam Dump System was not provided to cope with this transient, there would be such a strong mismatch between what the turbine is asking for and what the reactor is delivering that a reactor trip and lifting of steam generator safety valves would occur.

The number of occurrences of this transient is specified at 200 times or 5 per year for the 40 year plant design life.

5. Steady state fluctuations

It is assumed that the reactor coolant temperature and pressure at any point in the system vary around the nominal (steady state) values. For design purposes two cases are considered:

a. Initial fluctuations

These are due to control rod cycling during the first 20 full power months of reactor operation. Temperature is assumed to vary ±3°F and pressure by

±25 pounds per square inch (psi), once during each 2 minute period. The total number of occurrences is limited to 1.5×10^5. These fluctuations are assumed to occur consecutively, and not simultaneously with the random fluctuations.

b. Random fluctuations

Temperature is assumed to vary by ±0.5°F and pressure by ±6 psi, once every 6 minutes. With a 6 minute period, the total number of occurrences during the plant design life does not exceed 3.0×10^6.

6. Feedwater cycling at hot shutdown

These transients can occur when the plant is at no-load conditions, during which intermittent feeding of 32°F feedwater into the steam generators is assumed. Due to fluctuations arising from this mode of operation, the reactor coolant average temperature decreases to a lower value and then immediately begins to return to normal no-load temperature. This transient is assumed to occur 2000 times over the life of the plant.

7. Loop out of service

The plant may be operated at a reduced power level with a single loop out of service for limited periods of time. This is accomplished by reducing power level and tripping a single reactor coolant pump.

It is assumed that this transient occurs twice per year or 80 times in the life of the plant. Conservatively, it is assumed that all 80 occurrences can occur in the same loop. In other words it must be assumed that the whole RCS is subjected to 80 transients while each loop is also subjected to 80 inactive loop transients.

When an inactive loop is brought back into service, the power level is reduced to approximately 10 percent and the pump is started. It is assumed that an inactive loop is inadvertently started up at maximum allowable power level 10 times over the life of the plant. (This transient is covered under Upset Conditions.) Thus, the normal startup of an inactive loop is assumed to occur 70 times during the life of the plant.

8. Unit loading and unloading between 0 and 15 percent of full power

The unit loading and unloading cases between 0 and 15 percent power are represented by continuous and uniform ramp power changes, requiring 30 minutes for loading and 5 minutes for unloading. During loading, reactor coolant temperatures are increased from the no-load value to the normal load program temperatures at the 15 percent power level. The reverse temperature change occurs during unloading.

Prior to loading, it is assumed that the plant is at hot shutdown conditions, with 32°F feedwater cycling. During the 2 hour period following the beginning of loading, the feedwater temperature increases from 32°F to 300°F due to steam dump and turbine startup heat input to the feedwater. Subsequent to unloading, feedwater heating is terminated, steam dump is reduced to residual heat removal requirements, and feedwater temperature decays from 300°F to 32°F.

The number of these loading and unloading transients is assumed to be 500 each during the 40 year plant design life, which is equivalent to about one occurrence per month.

9. Boron concentration equalization

Following any large change in boron concentration in the RCS, spray is initiated in order to equalize concentration between the loops and the pressurizer. This can be done by manually operating the pressurizer backup heaters, thus causing a pressure increase, which will initiate spray at a compensated pressurizer pressure of approximately 2275 psia. The proportional sprays return the pressure to 2250 psia and maintain this pressure by matching the heat input from the backup heater until the concentration is equalized. For design purposes, it is assumed that this operation is performed once after each load change in the design load follow cycle. With two changes per day and a 90 percent plant availability factor over the 40 year design life, the total number of occurrences is 26,400.

10. Refueling

At the end of plant cooldown the fluid in the RCS is at 140°F. At this time the vessel head is removed and the refueling canal is filled. This is done by pumping water from the refueling water storage tank, which is outside and conservatively assumed to be at 32°F, into the loops by means of the residual heat removal pumps. It is conservatively assumed that the cold water is replaced with the colder water within 10 minutes.

This operation is assumed to occur twice per year or 80 times over the life of the plant.

Upset Conditions

The following primary system transients are considered upset conditions.

1. Loss of load (without immediate reactor trip).

2. Loss of power.

3. Partial loss of flow.

4. Reactor trip from full power.

5. Inadvertent Reactor Coolant System depressurization.

6. Inadvertent startup of an inactive loop.

7. Control rod drop.

8. Inadvertent Emergency Core Cooling System actuation.

9. Operating Basis Earthquake.

10. Excessive Feedwater Flow

1. Loss of Load (without immediate reactor trip)

   This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the Reactor Protection System (RPS). Since redundant means of tripping the reactor are provided as a part of the RPS, transients of this nature are not expected, but are included to ensure a conservative design.

   The number of occurrences of this transient is specified at 80 times or 2 times per year for the 40 year plant design life.

2. Loss of power

   This transient applies to a blackout situation involving the loss of outside electrical power to the station, assumed to be operating initially at 100 percent power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are de-energized and, following coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater, assumed to be at 32°F, from the Auxiliary Feedwater System operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

   The number of occurrences of this transient is specified at 40 times or 1 per year for the 40 year plant design life.

3. Partial loss of flow

   This transient applies to a partial loss of flow from full power, in which a reactor coolant pump is tripped out of service as the result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the Steam Dump System and flow reversal in the affected loop. The flow reversal causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooler water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

   The number of occurrences of this transient is specified at 80 times or 2 times per year for the 40 year plant design life.

4. Reactor trip from full power

   A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued

supply of feedwater and controlled dumping of steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core.

Various moderator cooldown transients associated with reactor trips can occur as a result of excessive feed or steam dump after trip or large load increase. For design purposes, reactor trip is assumed to occur a total of 400 times or 10 times per year over the life of the plant. The various types of trips and the number of occurrences for each are as follows:

a. Reactor trip with no inadvertent cooldown - 230 occurrences.

b. Reactor trip with cooldown but no safety injection - 160 occurrences.

c. Reactor trip with cooldown actuating safety injection - 10 occurrences.

5. Inadvertent Reactor Coolant System depressurization

Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

a. Actuation of a single pressurizer safety valve.

b. Inadvertent opening of one pressurizer power operated relief valve due either to equipment malfunction or operator error.

c. Malfunction of a single pressurizer pressure controller causing one power operated relief valve and two pressurizer spray valves to open.

d. Inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error.

e. Inadvertent auxiliary spray.

Of these events, the pressurizer safety valve actuation causes the most severe transients, and is used as an “umbrella” case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

When a pressurizer safety valve opens, and remains open, the system rapidly depressurizes, the reactor trips, and the Emergency Core Cooling System (ECCS) is actuated. Also, the passive accumulators of the ECCS are actuated when pressure decreases by approximately 1600 psi, about 12 minutes after the depressurization begins. The depressurization and cooldown are eventually terminated by operator action. All of these effects are completed within approximately 18 minutes. It is conservatively assumed that none of the pressurizer heaters are energized.

With pressure constant and safety injection in operation, boiloff of hot leg liquid through the pressurizer and open safety valve will continue.

For design purposes this transient is assumed to occur 20 times during the 40 year design life of the plant.

6. Inadvertent startup of an inactive loop

This transient can occur when a loop is out of service. With the plant operating at maximum allowable level the reactor coolant pump in the inactive loop is started as a result of operator error. Reactor trip occurs on high nuclear flux. This transient is assumed to occur 10 times during the life of the plant.

7. Control rod drop

This transient occurs if a bank of control rods drop into the fully inserted position due to a single component failure. The reactor is tripped on low pressurizer pressure. It is assumed that this transient occurs 80 times over the life of the plant.

8. Inadvertent Emergency Core Cooling System Actuation

A spurious safety injection signal results in an immediate reactor trip followed by actuation of the high head centrifugal charging pumps. These pumps deliver cold water from Refueling Water Storage Tank to the RCS cold legs. The initial portion of this transient is similar to the reactor trip from full power with no cooldown. Controlled steam dump and feedwater flow after trip removes core residual heat. Reactor coolant temperature and pressure decreases as the control rods move into the core.

Later in the transient, the injected water causes the RCS pressure to increase to the pressurizer power operated relief valve setpoint and the primary and secondary temperatures to decrease gradually. The transient continues until the operator stops the charging pumps. It is assumed that the plant is then returned to no-load conditions, with pressure and temperature changes controlled within normal limits.

For design purposes this transient is assumed to occur 60 times over the life of the plant.

9. Operating Basis Earthquake

The mechanical stresses resulting from the Operating Basis Earthquake are considered on a component basis. Fatigue analysis, where required by the codes, if performed by the supplier as part of the stress anal report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis. The number of occurrences for fatigue evaluation is assumed to be 20 earthquakes at 10 cycles each (200 cycles total).

10. Excessive Feedwater Flow

The pressure and temperature variations associated with this transient are considered in connection with analyzing the primary and secondary sides of the Steam Generator, the Reactor Coolant System, and the Pressurizer.

This transient is conservatively defined to cover the occurrence of several events of the same general nature. These include:

- Inadvertent opening of a feedwater control valve
- Turbine overspeed (110 percent) with an open feedwater control valve.
- Small steam break with an open feedwater control valve.

The excessive feedwater flow transient results from inadvertent opening of a feedwater control valve when the plant is at hot shutdown and the steam generator is in the no-load condition. The feedwater, condensate, and heater drain systems are in operation. The stem of a feedwater control valve has been assumed to fail, with the valve immediately reaching the full-open position. The feedwater flow to the affected loop is assumed to stop from essentially zero flow to the value determined by the steam flow is assumed to remain at zero, and the temperature of the feedwater entering the steam generator is conservatively assumed to be 32°F. Feedwater flow is isolated on a safety injection signal actuated by low steam line pressure or low pressurizer pressure signal.

Auxiliary feedwater flow, initiated by the safety injection signal, is assumed to continue with all pumps discharging into the affected steam generator. It is also assumed, for conservatism in the secondary side analysis, that auxiliary feedwater flows to the steam generators not affected by the malfunctioned valve, in the so-called “unfailed loops.” Plant conditions stabilize at the values reached in 600 seconds, at which time auxiliary feedwater flow is terminated. The plant is then either taken to cold shutdown, or returned to the no-load condition at a normal heatup rate with the auxiliary feedwater system under manual control.

For design purposes, this transient is assumed to occur 30 times during the life of the plant.

Emergency Conditions

The following primary system transients are considered emergency conditions:

1. Small loss of coolant accident.
2. Small steam break.
3. Complete loss of flow.

1. Small loss of coolant accident

For design transient purposes the small loss of coolant accident is defined as a break equivalent to the severance of a 1 inch inside diameter branch connection. (Breaks smaller than 0.375 inch inside diameter can be handled by the normal makeup system and produce no significant fluid systems transients.) Breaks which are much larger than 1 inch will cause accumulator injection soon after the accident and are regarded as faulted conditions. For design purposes it is assumed that this transient occurs five times during the life of the plant. It should be assumed that the ECCS is actuated immediately

after the break occurs and subsequently delivers water at a minimum temperature of 32°F to the RCS.

2. Small steam break

   For design transient purposes, a small steam break is defined as a break equivalent in effect to a steam safety valve opening and remaining open. This transient is assumed to occur five times during the life of the plant. The following conservative assumptions are used in defining the transients:

   a. The reactor is initially in a hot, zero-power condition.

   b. The small steam break results in immediate reactor trip and ECCS actuation.

   c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

   d. The ECCS operates at a design capacity and repressurizes the RCS within a relatively short time.

3. Complete loss of flow

   This accident involves a complete loss of flow from full power resulting from simultaneous loss of power to all reactor coolant pumps. The consequences of this incident are a reactor trip and turbine trip on undervoltage followed by automatic opening of the Steam Dump System. For design purposes this transient is assumed to occur five times during the plant lifetime.

Faulted Conditions

The following primary system transients are considered faulted conditions. Each of the following accidents should be evaluated for one occurrence:

1. Reactor coolant pipe break (large loss of coolant accident).

2. Large steam line break.

3. Feedwater line break.

4. Reactor coolant pump locked rotor.

5. Control rod ejection.

6. Steam generator tube rupture.

7. Safe Shutdown Earthquake.

1. Reactor coolant pipe break (large loss of coolant accident)

   Following rupture of a reactor coolant pipe resulting in a large loss of coolant, the primary system pressure decreases causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will still be at or near the operating temperature by the end of blowdown. It is conservatively assumed that the ECCS is actuated to introduce water at a minimum temperature of 32°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

2. Large steam line break

   This transient is based on the complete severance of the largest steam line. The following conservative assumptions were made:

   a. The reactor is initially in a hot, zero-power condition.

   b. The steam line break results in immediate reactor trip and ECCS actuation.

   c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

   d. The ECCS operates at design capacity and repressurizes the RCS within a relatively short time.

   The above conditions result in the most severe temperature and pressure variations which the primary system will encounter during a steam break accident.

3. Feedwater line break

   This accident involves a double ended rupture of the main feed-water piping from full power, resulting in the rapid blowdown of one steam generator and the termination of main feedwater flow to the others. The blowdown is completed in approximately 27 seconds. Conditions were conservatively chosen to give the most severe primary side and secondary side transients. All auxiliary feedwater flow exits at the break. The incident is terminated when the operator manually realigns the Auxiliary Feedwater System to isolate the break and to deliver auxiliary feedwater to the intact steam generators.

4. Reactor coolant pump locked rotor

   This accident is based on the instantaneous seizure of a reactor coolant pump with the plant operating at full power. The locked rotor can occur in any loop. Reactor trip occurs almost immediately, as the result of low coolant flow in the affected loop.

5. Control rod ejection

   This accident is based on the single most reactive control rod being instantaneously ejected from the core. This reactivity insertion in a particular region of the core causes a severe pressure increase in the RCS such that the pressurizer safety valves will lift and

also causes a more severe temperature transient in the loop associated with the affected region than in the other loops. For conservatism the analysis is based on the reactivity insertion and does not include the mitigating effects (on the pressure transient) of coolant blowdown through the hole in the vessel head vacated by the ejected rod.

6. Steam generator tube rupture

   This accident postulates the double ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reaction coolant pressure. Reactor trip will occur due to the resulting safety injection signal. In addition, safety injection actuation automatically isolates the feedwater lines, by tripping all feedwater pumps and closing the feedwater isolation valves. When this accident occurs, some of the reactor coolant blows down into the affected steam generator causing the shell side level to rise. The primary system pressure is reduced below the secondary safety valve setting. Subsequent recovery procedures call for isolation of the steam line leading from the affected steam generator. This accident will result in a transient which is no more severe than that associated with a reactor trip from full power. Therefore it requires no special treatment insofar as fatigue evaluation is concerned, and no specific number of occurrences is postulated.

7. Safe Shutdown Earthquake

   The mechanical dynamic or static equivalent loads due to the vibratory motion of the Safe Shutdown Earthquake are considered on a component basis.

Test Conditions

The following primary system transients under test conditions are discussed:

1. Turbine roll test.
2. Primary side hydrostatic test.
3. Secondary side hydrostatic test.
4. Primary side leakage test.
5. Secondary side leakage test.
6. Tube leakage test.

Pressure tests are performed in accordance with either the: ASME Code, Section III, Article NB-6000, 1980 edition and Addenda through Summer 1981 or ASME Code, Section XI, as described in FSAR Section 5.2.4.7.

1. Turbine roll test

   This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power will be used to heat the reactor coolant to operating temperature (no-load conditions) and the steam generated will be used to

perform a turbine roll test. However, the plant cooldown during this test will exceed the 100°F per hour design rate.

The number of such test cycles is specified at 20 times, to be performed at the beginning of plant operating life prior to irradiation. This transient occurs before plant startup and the number of cycles is therefore independent of other operating transients.

2. Primary side hydrostatic test

The pressure tests include both stop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3107 psig (1.25 times design pressure). In this test, the RCS is pressurized to 3107 psig coincident with steam generator secondary side pressure of 0 psig. The RCS is designed for 10 cycles of these hydrostatic tests, which are performed prior to plant startup. The number of cycles is independent of other operating transients.

Additional hydrostatic tests will be performed to meet the inservice inspection requirements of the ASME Code, Section XI, subarticle IS5-20 (Winter 1972 Addenda). A total of four such tests is expected. The increase in the fatigue usage factor caused by these tests is easily covered by the conservative number (200) of primary side leakage tests that are considered for design.

3. Secondary side hyrostatic test

The secondary side of the steam generator is pressurized to 1.25 design pressure with a minimum water temperature of 120°F coincident with the primary side at 0 psig.

For design purposes it is assumed that the steam generator will experience 10 cycles of this test.

These tests may be performed either prior to plant startup, or subsequently following shutdown for major repairs or both. The number of cycles is therefore independent of other operating transients.

4. Primary side leakage test

For design purposes the following is considered:

1) Subsequent to each time the primary system has been opened, a leakage test will be performed, 2) During this test the primary system pressure is, for design purposes, raised to 2500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks. 3) It is assumed that 200 cycles of this test will occur during the 40 year life of the plant. In actual practice the system leakage test is performed in accordance with ASME Section XI following each refueling outage.

5. Secondary side leakage test

During the life of the plant it may be necessary to check the secondary side of the steam generator (particularly, the manway closure) for leakage. For design purposes it is assumed that the steam generator secondary side is pressurized to just below its design pressure, to prevent the safety valves from lifting. In order not to exceed a secondary side to primary side pressure differential of 670 psi, the primary side must also be pressurized. The primary system must be above the minimum temperature imposed by reactor vessel material ductility requirements. It is assumed that this test is performed 80 times during the 40 year life of the plant.

6. Tube leakage test

During the life of the plant it may be necessary to check the steam generator for tube leakage and tube-to-tubesheet leakage. This is done by visual inspection of the underside (channel head side) of the tubesheet for water leakage, with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdowns.

For these tests the secondary side of the steam generator is pressurized with water, initially at a relatively low pressure, and the primary system remains depressurized. The underside of the tubesheet is examined visually for leaks. If any are observed, the secondary side is then depressurized and repairs made by tube plugging. The secondary side is then repressurized (to a higher pressure) and the underside of the tubesheet is again checked for leaks. This process is repeated until all the leaks are repaired. The maximum (final) secondary side test pressure reached is 840 psig.

The total number of tube leakage test cycles is defined as 800 during the 40 year life of the plant. Following is a breakdown of the anticipated number of occurrences at each secondary side test pressure:

Test Pressure (psig)	Occurrences
200	400
400	200
600	120
840	80

Both the primary and secondary sides of the steam generators will be at ambient temperatures during these tests.

3.9N.1.2 Computer Programs Used in Analyses

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment. These are described and verified in Reference [1] and [13]. The description and verification procedure for STRUDL and NASTRAN is presented in reference [16] and [17].

1. WESTDYN - static and dynamic analysis of piping systems.

2. FIXFM - time history response of three dimensional structures; also called FIXFM3; now a subroutine combined into WESTDYN.

3. WESDYN-2 - piping system stress analysis from time history displacement data; also called WESTDYN; now a subroutine combined into WESTDYN.

4. STHRUST - hydraulic loads on loop components from blowdown information.

5. WECAN - finite element structural analysis, includes dynamic analysis of piping systems.

6. DARI-WOSTAS - Dynamic transient response analysis of reactor vessel internals.

7. STRUDL - linear elastic structural analysis.

8. NASTRAN - finite element structural analysis.

3.9N.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Category I systems or components. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests as discussed in Section 3.9N.2.

3.9N.1.4 Considerations for the Evaluation of the Faulted Condition

3.9N.1.4.1 Loading Conditions

The structural stress analyses performed on the RCS consider the loadings specified as shown in Table 3.9N-2. These loads result from thermal expansion, pressure, weight, Operating Basis Earthquake (OBE), Safe Shutdown Earthquake (SSE), design basis loss of coolant accident, and plant operational thermal and pressure transients.

3.9N.1.4.2 Analysis of the Reactor Coolant Loop and Supports

The reactor coolant loop piping is evaluated in accordance with the criteria of ASME III, NB-3650 and Appendix F. The loads included in the evaluation result from the SSE, deadweight, pressure, and LOCA loadings (loop hydraulic forces, asymmetric subcompartment pressurization forces, and reactor vessel motion including uplift and lateral displacement). The results of the stress analysis of the reactor coolant loop piping are given in Table 3.9N-21.

The loads used in the analysis of the reactor coolant loop/supports System are described in detail below.

Pressure

Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations in accordance with the ASME Code.

The term operating pressure is used in connection with determination of the system deflections and support forces. The steady state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the reactor coolant loop model at change in direction or flow area.

Weight

A dead weight analysis is performed to meet ASME Code requirements by applying a 1.0 g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

Seismic for Unit 1

For the complete elimination of all the snubbers from the upper supports of the steam generators on the reactor coolant loops, the piping stress analysis of the reactor coolant loops for seismic loading is done using the time history method of seismic analysis. The response spectrum method of seismic analysis had been used in the piping stress analysis done previous to the elimination of the snubbers. The time history method of seismic analysis is now used in the piping stress analysis done for the elimination of the snubbers. Because the structural model includes the reactor coolant loops coupled to the internal concrete structure and the foundation spring, the time history accelerations at the foundation spring are applied to the model as the input seismic loading.

Two types of seismic moments are required in the calculation of piping stresses according to the equations found in Section NB-3650 of the ASME Code. The first type, for equations 9 and 13, includes the effects from just inertia. The second type, for equations 10 and 11, includes the effects from both inertia and anchor motions. The moments resulting from the time history method of seismic analysis inherently include the effects from both inertia and anchor motions. Therefore, these moments are used appropriately in equations 10 and 11, but they are also used conservatively in equations 9 and 13.

Seismic for Unit 2

The forcing functions for the reactor coolant loop seismic piping analyses are derived from dynamic response analyses of the Containment Building subjected to seismic ground motion. Input is in the form of floor response spectrum curves at various elevations within the Containment Building.

For the OBE and SSE seismic analyses Code Case N-411 damping is used in analysis of the reactor coolant loop/supports system.

In the response spectrum method of analysis, the total response loading obtained from the seismic analysis consists of two parts: the inertia response loading of the piping system and the differential anchor movements loading. Two sets of seismic moments are required to perform an ASME Code analysis. The first set includes only the moments resulting from inertia effects and these moments are used in the resultant moment (Mi) value for equations 9 and 13 of NB-3650. The second set includes the moments resulting from seismic anchor motions and are used in equations 10 and 11 of NB-3650. Differential anchor movement is discussed in Section 3.7N.

Loss of Coolant Accident

Blowdown loads are developed in the broken and unbroken reactor coolant loops as a result of transient flow and pressure fluctuations following a postulated pipe break to the large branch nozzles of the reactor coolant loop (breaks 12 and 13, Table 3.6B-2). Structural consideration of dynamic effects of postulated pipe breaks requires postulation of a finite number of break locations. Postulated pipe break locations are given in Section 3.6B.

Broken loop time history dynamic analysis is performed for these postulated break cases. Hydraulic models are used to generate time- dependent hydraulic forcing functions used in the analysis of the reactor coolant loop for each break case. For a further description of the hydraulic forcing functions, refer to Section 3.6B.2.2.1.

Transients

The ASME Code requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are tabulated in Section 3.9N.1.1.

The vertical thermal growth of the reactor pressure vessel nozzle centerlines is considered in the thermal analysis to account for equipment nozzle displacement as an external movement.

The hot moduli of elasticity E, the coefficient of thermal expansion at the metal temperature, the external movements transmitted to the piping due to vessel growth, and the temperature rise above the ambient temperature T, define the required input data to perform the flexibility analysis for thermal expansion.

To provide the necessary high degree of integrity for the RCS, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operation conditions.

3.9N.1.4.3 Reactor Coolant Loop Models and Methods

The analytical methods used in obtaining the solution consists of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectra method for seismic dynamic analysis for Unit 2, the time history method of seismic analysis for Unit 1, and time history integration method for the loss of coolant accident dynamic analysis.

The integrated reactor coolant loop/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, the stiffnesses of auxiliary line piping which affect the system and the stiffness of piping restraints. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

Static

The reactor coolant loop/supports system model, constructed for the WESTDYN computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 3.9N-1 shows an isometric line schematic of this mathematical model. The

steam generator and reactor coolant pump vertical and lateral support members are described in Section 5.4.14.

The spatial geometric description of the reactor coolant loop model is based upon the reactor coolant loop piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus of elasticity E, the coefficient of thermal expansion, the average temperature change from ambient temperature T, and the weight per unit length are specified for each element. The primary equipment supports are represented by stiffness matrics which define restraint characteristics of the supports. Due to the symmetry of the static loadings, the reactor pressure vessel centerline is represented by a fixed boundary in the system mathematical model. The horizontal thermal growth of the reactor pressure vessel (RPV) is included in the loop model by modeling the RPV as an equivalent pipe from the RPV centerline to the RPV nozzle. The vertical thermal growth at the RPV nozzle location includes the thermal growth of the RPV support structure and RPV nozzle and is applied to the loop model as an external displacement.

The model is made-up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this matter, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the net-work point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for deadweight, thermal, and general pressure loading conditions are obtained by using the WESTDYN-7 computer program. The derivation of the hydraulic loads for the loss coolant accident analysis of the loop is covered in Section 3.6N.2.

Seismic for Unit 1

For the complete elimination of all the snubbers from the upper supports of the steam generators on the reactor coolant loops, the piping stress analysis of the reactor coolant loops for seismic loading is done using the time history method of seismic analysis. More specifically, the modal superposition method of time history analysis for nonlinear structures in the WECAN computer program is used. The analysis accounts for the replacement steam generator in addition to the elimination of the snubbers.

The reactor pressure vessel; the hot leg piping, replacement steam generator, crossover leg piping, reactor coolant pump, and cold leg piping for each of the four reactor coolant loops; the internal structure, containment building, and foundation spring, as shown in Figure 3.7B-34; and the supports connecting the reactor coolant loops to the internal structure are all included in the structural model created for WECAN. The stiffness and mass characteristics of each of these items are accounted for in the structural model. Pipe, mass, stiffness matrix, and mass elements are used in the modeling of the equipment; pipe, elbow, mass, stiffness matrix, and mass

elements are used in the modeling of the piping; stiffness matrix and mass elements are used in the modeling of the building and foundation spring; and stiffness matrix and nonlinear impact elements are used in the modeling of the supports.

The reactor pressure vessel is represented by nine discrete masses distributed along the vertical centerline of its shell and the vertical centerline of its core internals, as shown in Figure 3.9N-2. The replacement steam generator is represented by eight discrete masses distributed along the vertical centerline of its shell. The reactor coolant pump is represented by two discrete masses, one located at the intersection of the axial centerlines of the suction and discharge nozzles and the other located at the center of gravity of the motor.

The reactor pressure vessel supports, the steam generator columns, and the reactor coolant pump columns are active for all operating conditions of the plant. they are represented by stiffness matrices in the model. The horizontal component of the steam generator lower support, the steam generator upper support, and the reactor coolant pump tie rods are inactive for heatup, cooldown, and normal operating conditions of the plant, but they become active for seismic loading. They are represented by stiffness matrices and/or nonlinear impact elements in the model. Seismic loading is assumed to occur at the full power condition.

The modeling of the internal structure, containment building, and foundation spring is discussed in Section 3.7B.2.1.6. Two sets of stiffness properties for the containment building, uncracked and cracked, and three sets of stiffness values for the foundation spring, best-estimate, lower-bound, and upper-bound, are discussed. Six different models result from the various combinations of these stiffness characteristics. A separate time history analysis is made for each of the six models, and then the results for the six models are enveloped.

The time history accelerations for the safe shutdown earthquake are presented in Section 3.7B.1.2. The time history accelerations are applied to the model at the foundation spring as the input seismic loading. The horizontal time history acceleration shown in Figure 3.7B-14 is applied in the north-south and east-west directions. The vertical time history acceleration shown in Figure 3.7B-19 is applied in the vertical direction. A separate time history analysis is made for each of the three directions of shock, north-south, east-west, and vertical, and then the results for the three directions are combined by the square-root-sum-of-the-squares method. More specifically, to calculate the total seismic result for an item such as a piping load, support load, displacement, or acceleration, the maximum value of the item over time is determined for each of the three directions, and then the three maximum values are combined by the square-root-sum-of-the-squares method.

Analysis is also done for the operational basis earthquake. The input accelerations for the operational basis earthquake are one half the input accelerations for the safe shutdown earthquake, as noted in Section 3.7B.1.1.

The uncertainties in modeling discussed in Section 3.7B.2.9 are addressed by varying the time increment of the input time history acceleration. Three cases are considered. In the first case, the time increment is used as is. In the second case, the time increment is decreased by 10% to raise the effective frequency content of the input time history acceleration by about 10%. In the third case, the time increment is increased by 10% to lower the effective frequency content by about 10%. A separate time history analysis is made for each of the three cases, and then the results for the three cases are enveloped.

Composite modal damping is used. The damping values presented in Sections 3.7B.1.3 and 3.7B.2.4 and Table 3.7B-1 are used, with 2% of critical damping being used for the piping for the operational basis earthquake, and 3% for the safe shutdown earthquake.

The modal superposition method of time history analysis is used. The modal analysis of the model is done with all the nonlinear impact elements inactive, but then the modal results are corrected to account for the activity of the nonlinear impact elements in the modal superposition analysis.

Seismic for Unit 2

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. All of the piping loops are included in the system model. The effect of the equipment motion on the reactor coolant/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The steam generator is typically represented by three discrete masses. The lower mass is located at the intersection of the centerlines of the inlet and outlet nozzles of the steam generator. The middle mass is located at the steam generator upper support elevation, and the third mass is located at the top of the steam generator.

The reactor coolant pump is typically represented by a two discrete mass model. The lower mass is located at the intersection of the centerlines of the pump suction and discharge nozzles. The upper mass is located near the center of gravity of the motor.

The reactor vessel and core internals are typically represented by approximately 10 discrete masses. The masses are lumped at various locations along the length of the vessel and along the length of the representation of the core internals, as shown in Figure 3.9N-2.

The component upper and lower lateral supports are inactive during plant heatup, cooldown and normal plant operating conditions. However, these restraints become active when the plant is at power and under the rapid motions of the reactor coolant loop components that occur from the dynamic loadings, and are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The response spectra method employs the lumped mass technique, linear elastic properties, and the principal of modal superposition. The floor response spectra are applied along both horizontal axes and the vertical axis simultaneously.

From the mathematical description of the system, the overall stiffness matrix [K] is developed from the individual element stiffness matrices using the transfer matrix method. After deleting the rows and columns representing rigid restraints, the stiffness matrix is revised to obtain a reduced stiffness matrix [K_R] associated with mass degree of freedom only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes are determined. The modal participation factor matrix is computed and combined with the appropriate response spectra value to give the modal amplitude for each mode. The total modal amplitude is obtained by taking the square root of the sum of the squares of the contributions for each direction.

The modal amplitudes are then converted to displacements in the global coordinate system and applied to the corresponding mass point. From these data the forces, moments, deflections, rotations, support reactions and piping stresses are calculated for all significant modes.

The total seismic response is computed by combining the contributions of the significant modes by the square root of the sum of the squares method and accounting for closely spaced modes as described in Section 3.7N.

Loss of Coolant Accident

The mathematical model used in the static analyses is modified for the loss of coolant accident analyses to represent the severance of the reactor coolant loop piping nozzles at the postulated break location. Modifications include addition of the mass characteristic of the piping and equipment.

The time-history hydraulic forces at the node points are combined to obtain the forces and moments acting at the corresponding structural lumped mass node points.

The dynamic structural solution for the full power loss of coolant accident and steam line break is obtained by using a modified-predictor-corrector-integration technique and normal mode theory.

When elements of the system can be represented as single acting members (tension or compression members), they are considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiplied nonlinear elements can be applied at the same node, if necessary.

The time-history solution is performed in subprogram FIXFM3. The input to this subprogram consists of the natural frequencies, normal modes, applied forces and nonlinear elements. The natural frequencies and normal modes for the modified reactor coolant loop dynamic model are determined with the WESTDYN program. To properly simulate the release of the strain energy in the pipe, the internal forces in the system at the postulated break location due to the initial steady state hydraulic forces, thermal forces, and weight forces are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the solution is only for the transient problem (the dynamic response of the system from the static equilibrium position). The time-history displacement solution of all dynamic degrees of freedom is obtained using subprogram FIXFM and employing 4 percent critical damping.

The loss of coolant accident displacements of the reactor vessel are applied in time-history form as input to the dynamic analysis of the reactor coolant loop. The loss of coolant accident analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions, cavity pressure loads, and loop mechanical loads. The reactor vessel analysis is described in Section 3.9N.1.4.6.

The main loop piping breaks and, therefore, the vessel cavity pressurization effects are not part of the design basis (because they are excluded as dynamic effects section 3.1.1.4). However, they are included as a conservative estimate of vessel motion due to LOCA. If required, vessel motion for the postulated branch nozzle breaks can be used in place of the conservative main loop piping breaks discussed in this section.

The resultant asymmetric external pressure loads on the RCP and steam generator resulting from a postulated pipe rupture and pressure buildup in the loop compartments are applied to the same integrated RCL/supports system model used to compute loadings on the components, component supports, and RCL piping as discussed above. The response of the entire system is obtained for the various external pressure loading cases from which the internal member forces and piping stresses are calculated. For each pipe break case considered, the equipment support loads and piping stresses resulting from the external pressure loading are added to the support loads and piping stresses calculated using the loop LOCA hydraulic forces and RPV motion.

The break locations considered for subcompartment pressurization are those postulated for the RCL LOCA analysis as discussed in Section 3.6N and WCAP-8172 (Reference [1] of Section 3.6N). The asymmetric subcompartment pressure loads are provided to Westinghouse. The analysis to determine these loads is discussed in Section 6.2.

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping.

The time-history displacements of the FIXFM3 program are used as input to program WESDYN to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation the displacements are treated as imposed deflections on the reactor coolant loop masses.

Transient

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the Code into three parts, a uniform, a linear, and a nonlinear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall and the nonlinear portion causes a skin stress.

The transients as defined in Section 3.9N.1.1 are used to define the fluctuations in plant parameters. A one dimensional finite difference heat conduction program is used to solve the thermal transient problem. The pipe is represented by at least 50 elements through the thickness of the pipe. The convective heat transfer coefficient employed in this program represents the time varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. An arbitrary temperature distribution across the wall is shown in Figure 3.9N-3.

The average through-wall temperature, T_A, is calculated by integrating the temperature distribution across the wall. This integration is performed for all time steps so that T_A is determined as a function of time.

$$T_A(t) = \frac{1}{H}\int_0^H T(X, t)\ dX$$

The range of temperature between the largest and smallest value of T_A is used in the flexibility analysis to generate the moment loadings caused by the associated temperature changes.

The thermal moment about the mid-thickness of the wall caused by the temperature distribution through the wall is equal to:

$$M = E\alpha\int_0^H \left(X - \frac{H}{2}\right) T(X, t)dX$$

The equivalent thermal moment produced by the linear thermal gradient as shown in Figure 3.9N-3 about the mid-wall thickness is equal to:

$$M_L = E\alpha_{12}\frac{\Delta T_1}{}H^2$$

Equating M_L and M, the solution for T_1 as a function of time is:

$$\Delta T_1(t) = \frac{12}{M^2}\int_0^H \left(X - \frac{H}{2}\right) T(X, t)dX$$

The maximum nonlinear thermal gradient, T_2, will occur on the inside surface and can be determined as the difference between the actual metal temperature on this surface and half of the average linear thermal gradient plus the average temperature.

$$\Delta T_{21}(t) = \left|T(0, t) - T_A(t)\right| - \frac{\left|\Delta T_1(t)\right|}{2}$$

Load Set Generation

A load set is defined as a set of pressure loads, moment loads, and through-wall thermal effects at a given location and time in each transient. The method of load set generation is based on Reference [2]. The through-wall thermal effects are functions of time and can be sub-divided into four parts:

1. Average temperature (T_A) is the average temperature through-wall of the pipe which contributes to general expansion loads.

2. Radial linear thermal gradient which contributes to the through-wall bending moment (ΔT_1).

3. Radial nonlinear thermal gradient (ΔT_2) which contributes to a peak stress associated with shearing of the surface.

4. Discontinuity temperature (T_A - T_B) represents the difference in average temperature at the cross sections on each side of a discontinuity.

Each transient is described by at least two load sets representing the maximum and minimum stress state during each transient. The construction of the load sets is accomplished by combining the following to yield the maximum (minimum) stress state during each transient.

1. ΔT_1

2. ΔT_2

3. $\alpha_A T_A - \alpha_B T_B$

4. Moment loads due to T_A.

5. Pressure loads.

This procedure produces at least twice as many load sets as transients for each point.

As a result of the normal mode spectral technique employed in the seismic analysis the, load components cannot be given signed values. Seismic loads are considered in the fatique analysis as having either a positive or negative sign, thus ensuring the most conservative combination of seismic loads are used in the stress evaluation.

For all possible load set combinations, the primary plus secondary and peak stress intensities, fatigue reduction factors (Ke) and cumulative usage factors, U, are calculated. The WESTDYN program is used to perform this analysis in accordance with the ASME Code, Section III, Subsection NB-3650. Alternatively, detailed finite element stress analyses may be used to determine primary plus secondary and peak stress intensities, for the load set combinations. Since it is impossible to predict the order of occurrence of the transients over a 40 year life it is assumed that the transients can occur in any sequence. This is a very conservative assumption.

The combination of load sets yielding the highest alternating stress intensity range is used to calculate the incremental usage factor. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9N.1.4.4 Primary Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped parameter, multimass structural models. The equipment support structure models are dual purpose since they are required: 1) to quantitatively represent the elastic restraints which the supports impose upon the loop, and 2) to evaluate the individual support member stresses due to the forces imposed upon the supports by the loop.

A description of the supports is found in Section 5.4.14. Detailed models are developed using beam elements and plate elements, where applicable.

The reactor vessel supports are modeled using the NASTRAN computer program. Structure geometry, topology and member properties are used in the modeling. NASTRAN is a general purpose finite element computer program. Its capability, broad in scope, permits the solution of

large, complicated structural problems. Static, dynamic, modal, plastic and thermal analyses are possible. NASTRAN employs the matrix displacement method for each finite element in the idealized structure. A “wave front” direct-solution technique is employed to give accurate results in a minimum of computer time. The analysis solution output includes geometry plots, nodal displacements, element stresses and nodal forces. Steam generator and reactor coolant pump supports are modeled as linear or non-linear springs.

The steam generator lower beam was modeled using the STRUDL Program. STRUDL, part of the integrated civil engineering computer system (ICES), is widely used for the analysis and design of structures. It is applicable to linear elastic two and three-dimensional frame or truss structures, employs the stiffness formulation, and is valid only for small displacements.

Structure geometry, topology, element orientation and cross-section properties are described in free format. Member and support joint releases, such as pins and rollers, are specified. Otherwise, six restraint components are assumed at each end of each member and at each support joint.

The STRUDL system performs structural stability and equilibrium checks during the solution process and prints error messages if these conditions are violated. Type, location, and magnitude of applied loads or displacements are specified for any number of loading conditions.

One important feature of STRUDL is that any desired changes, deletions, or additions can be made to the structural model during the solution process. This produces results for a number of structure configurations, each with any number of loading conditions.

Printed output content, specified by input commands, includes member forces and distortions, joint displacements, support joint reactions, and member stresses.

The steam generator upper lateral supports are modeled/analyzed using the STRUDL computer program. The following discussion about snubbers applies only for Unit 2. The snubbers have been eliminated for Unit 1. The average snubber stiffness is combined with the stiffness of the entire upper support and used in the seismic analysis. Non-linear tension and compression spring rates are used in the LOCA model. The spring rates are 5400 k/in compression and 4300 k/in tension. Entrapped air is bled from the system during installation and therefore has no effect on the snubber operation. The characteristics of the fluid over a temperature range of 50°F to 500°F were considered, and it was determined the temperature of the fluid has no effect on snubber operation over its operating temperature range (<300°F). Load conditions/transients analyzed include normal operating and seismic (OBE), and SSE seismic combined with LOCA. The load ratios of actual load/rated capacity are: normal & OBE, 0.460; SSE seismic combined with LOCA, 0.265.

For each operating condition, the loads (obtained from the RCL analysis) acting on the support structures are appropriately combined. The main loop piping breaks are not part of the design basis (excluded as dynamic effects, Section 3.1.1.4). Reactor coolant loop normal and upset condition thermal expansion loads are treated as primary loadings for the primary component supports. The adequacy of each member of the steam generator supports, reactor coolant pump supports, and piping restraints is verified by solving the ASME III Subsection NF stress and interaction equations. The adequacy of the RPV support structure is verified using the NASTRAN computer program and comparing the resultant stresses to the criteria given in ASME III Subsection NF.

Tables 3.9N-14 through 3.9N-17 present maximum stresses in each member of the steam generator, reactor coolant pump, and pressurizer support structures expressed as a percentage of maximum permissible values for all operating condition loadings. The loads on the reactor vessel supports and the resulting stresses are shown in Table 3.9N-19. The above loads and stresses include the effects of loads resulting from asymmetric subcompartment pressurization.

3.9N.1.4.5 Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop includes the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is evaluated for the loading combinations outlined in Table 3.9N-2. The equipment is analyzed for: 1) the normal loads of deadweight, pressure and thermal, 2) mechanical transients of OBE, SSE, and pipe ruptures, including the effects of asymmetric subcompartment pressurization and 3) pressure and temperature transients outlined in Section 3.9N.1.1.

The results of the reactor coolant loop analysis are used to determine the loads acting on the nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an “umbrella” load basis. That is, on the basis of previous plant analyses a set of loads are determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviations where the actual load is larger than the umbrella load will be handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generator and reactor coolant pump are performed using 2 percent damping for the OBE and 4 percent damping for the SSE. The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

Reactor coolant pressure boundary components are further qualified to ensure against unstable crack growth under faulted conditions by performing detailed fracture analyses of the critical areas of this boundary. Actuation of the ECCS produces relatively high thermal stresses in the system. Regions of the pressure boundary which come into contact with ECCS water are given primary consideration. These regions include the reactor vessel beltline region, the reactor vessel inlet nozzles, and the safety injection nozzles in the piping system.

Two methods of analysis are used to evaluate thermal effects in the regions of interest. The first method is linear elastic fracture mechanics (LEFM). The LEFM approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Constantly, a basic assumption employed in LEFM is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

The elastic stress field at the crack tip in any cracked body can be described by a single parameter designated as the stress intensity factor, K. The magnitude of the stress intensity

factor K is a function of the geometry of the body containing the crack, the size and location of the crack, and the magnitude and distribution of the stress.

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses perpendicular to the major plane of the crack) the stress intensity factor is designated as K_I and the critical stress intensity factor is designated K_{IC}. Commonly called the fracture toughness, K_{IC} is an inherent material property which is a function of temperature and strain rate. Any combination of applied load, structural configuration, crack geometry and size which yields a stress intensity factor K_{IC} for the material will result in crack instability.

The criterion of the applicability of LEFM is based on plasticity considerations at the postulated crack tip. Strict applicability (as defined by ASTM) of LEFM to large structures where plane strain conditions prevail requires that the plastic zone developed at the tip of the crack does not exceed 2.25 percent of the crack depth. In the present analysis, the plastic zone at the tip of the postulated crack can reach 20 percent of the crack depth. However, LEFM has been successfully used quite often to provide conservative brittle fracture prevention evaluations, even in cases where strict applicability of the theory is not permitted due to excessive plasticity. Recently, experimental results from Heavy Section Steel Technology (HSST) Program intermediate pressure vessel tests have shown that LEFM can be applied conservatively as long as the pressure component of the stress does not exceed the yield strength of the material. The addition of the thermal stresses, calculated elastically, which results in total stresses in excess of the yield strength does not affect the conservatism of the results, provided that these thermal stresses are included in the evaluation of the stress intensity factors. Therefore, for faulted condition analyses, LEFM is considered applicable for the evaluation of the vessel inlet nozzle and beltline region.

In addition, it has been well established that the crack propagation of existing flaws in a structure subjected to cyclic loading can be defined in terms of fracture mechanics parameters. Thus, the principles of LEFM are also applicable to fatigue growth of a postulated flaw at the vessel inlet nozzle and beltline region.

For the safety injection and charging line nozzles, which are fabricated from Type 304 stainless steel, LEFM is not applicable because of extreme ductility of the material. For these nozzles, the thermal effects are evaluated using the principles of Miner's hypothesis of linear cumulative damage in conjunction with fatigue data from constant stress or strain fatigue tests. The cumulative usage fatigue defined as the sum of the ratios of the number of cycles of each transient (n) to the allowable number of cycles for the stress range associated with the transient (N) must not exceed 1.0.

An example of a faulted condition evaluation carried out according to the procedure discussed above in given in Reference [3]. This report discusses the evaluation procedure in detail as applied to a severe faulted condition (a postulated loss of coolant accident), and concludes that the integrity of the reactor coolant pressure boundary would be maintained in the event of such an accident.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of the ASME Code, Section III. These valves are identified in Section 3.9N.3.2.

Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the line connect to the primary system piping are orificed to a 3/8 inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.

3.9N.1.4.6 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss of Coolant Accident

1. Introduction

This section presents the method of computing the reactor pressure vessel response to a postulated loss of coolant accident (LOCA). The dynamic analysis of the reactor vessel was performed prior to the leak-before-break update to GDC-4 and therefore included the effects of main loop piping breaks and reactor vessel cavity pressurization. These effects, no longer required for the dynamic analysis, are conservative loadings for the postulated LOCA analysis of the reactor vessel and internals discussed in this section and identified in Table 3.6B-2. The structural analysis considers simultaneous application of the time-history loads on the reactor vessel resulting from the reactor coolant loop mechanical loads, internal hydraulic pressure transients, and reactor cavity pressurization (for postulated breaks in the reactor coolant pipe at the vessel nozzles). The vessel is restrained by reactor vessel support pads and shoes beneath four of the reactor vessel nozzles and the reactor coolant loops with the primary supports of the steam generators and the reactor coolant pumps.

Pipe displacement restraints installed in the primary shield wall limit the break opening area of the vessel nozzle pipe breaks to less than 144 square inches. This break area was determined to be an upper bound by using worst case vessel and pipe relative motions based on similar plant analyses. Detailed studies have shown that pipe breaks at the hot or cold leg reactor vessel nozzles, even with a limited break area, would give the highest reactor vessel support loads and the highest vessel displacements, primarily due to the influence of reactor cavity pressurization. The dynamic analysis of the reactor vessel was performed prior to the leak-before-break update to GDC-4 and therefore included the effects of main loop piping breaks and reactor vessel cavity pressurization. These effects, no longer required for the dynamic analysis, are conservative loadings for the postulated LOCA analysis of the reactor vessel and internals discussed in this section and identified in Table 3.6B-2. By considering these breaks, the most severe reactor vessel support loads are determined. For completeness, an additional break outside the shield wall, for which there is no cavity pressurization, was also analyzed, specifically, the pump outlet nozzle pipe break.

2. Interface Information

Asymmetric reactor cavity pressurization loads were provided to Westinghouse by Gibbs and Hill, Inc.

All other input information was developed within Westinghouse. This information includes: reactor internals properties, loop mechanical loads and loop stiffness, internal hydraulic pressure transients, and reactor support stiffnesses. These inputs allowed formulation of the mathematical models and performance of the analyses, as will be described.

3. Loading Conditions

The dynamic analysis of the reactor vessel was performed prior to the leak-before-break update to GDC-4 and therefore included the effects of main loop piping breaks and reactor vessel cavity pressurization. These effects, no longer required for the dynamic analysis, are conservative loadings for the postulated LOCA analysis of the reactor vessel and internals discussed in this section and identified in Table 3.6B-2.

Following a postulated pipe rupture at the reactor vessel nozzle, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of three phenomena: (1) reactor coolant loop mechanical loads, (2) reactor cavity pressurization forces and (3) reactor internal hydraulic forces.

The reactor coolant loop mechanical forces are derived from the elastic analysis of the loop piping for the postulated break. This analysis is described in Section 3.9N.1.4.3. The loop mechanical forces which are released at the broken nozzle are applied to the vessel in the RPV blowdown analysis.

Reactor cavity pressurization forces arise for the pipe breaks at the vessel nozzles from the steam and water which is released into the reactor cavity through the annulus around the broken pipe. The reactor cavity is pressurized asymmetrically with higher pressure on the side of the broken pipe resulting in horizontal forces applied to the reactor vessel. Smaller vertical forces arising from pressure on the bottom of the vessel and the vessel flanges are also applied to the reactor vessel. The cavity pressure analysis is described in Section 6.2.

The internals reaction forces develop from asymmetric pressure distributions inside the reactor vessel. For a vessel inlet nozzle break and pump outlet nozzle break, the depressurization wave path is through the broken loop inlet nozzle and into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, down and around the downcomer annulus and up through the fuel. In the case of an RPV outlet nozzle break the wave passes through the RPV outlet nozzle and directly into the upper internals region, depressurizes the core, and enters the downcomer annulus from the bottom of the vessel. Thus, for an outlet nozzle break, the downcomer annulus is depressurized with much smaller differences in pressure horizontally across the core barrel than for the inlet break. For both the inlet and outlet nozzle breaks, the depressurization waves continue their propagation by reflection and translation through the reactor vessel fluid but the initial depressurization wave has the greatest effect on the loads.

The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion is coupled with the pressure transients. This phenomena has been referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP- 8708[8].

4. Reactor Vessel and Internals Modeling

The reactor vessel is restrained by two mechanisms: (1) the four attached reactor coolant loops with the steam generator and reactor coolant pump primary supports and (2) four reactor vessel supports, two beneath reactor vessel inlet nozzles and two beneath reactor vessel outlet nozzles. The reactor vessel supports are described in Section 5.4.14 and are shown in Figures 5.4-12, and 3.8-17. The support shoe provides restraint in the horizontal directions and for downward reactor vessel motion.

The reactor vessel model consists of two non-linear elastic models connected at a common node. One model represents the dynamic vertical characteristics of the vessel and its internals, and the other model represents the translational and rotational characteristics of the structure. These two models are combined in the DARI-WOSTAS code[1] to represent motion of the reactor vessel and its internals in the plane of the vessel centerline and the broken pipe centerline.

The model for horizontal motion is shown in Figure 3.9N-12. Each node has one translational and one rotational degree of freedom in the vertical plane containing the centerline of the nozzle attached to the broken pipe and the centerline of the vessel. A combination of beam elements and concentrated masses are used to represent the components including the vessel, core barrel, neutron panels, fuel assemblies, and upper support columns. Connections between the various components are either pin-pin rigid links, translational impact springs with damping or rotational springs.

The model for vertical motion is shown in Figure 3.9N-13. Each mass node has one translational degree of freedom. The structure is represented by concentrated masses, springs, dampers, gaps, and frictional elements. The model includes the core barrel, lower support columns, bottom nozzles, fuel rods, top nozzles, upper support structure, and reactor vessel.

The horizontal and vertical models are coupled at the elevation of the primary nozzle centerlines. Node 1 of the horizontal model is coupled with node 2 of the vertical model at the reactor vessel nozzle elevation. This coupled node has external restraints characterized by a 3 x 3 matrix which represents the reactor coolant loop stiffness characteristics, by linear horizontal springs which describe the tangential resistance of the supports, and by individual non-linear vertical stiffness elements which provide downward restraint only. The supports as represented in the horizontal and vertical models (Figures 3.9N-12 and 3.9N-13) are not indicative of the complexity of the support system used in the analysis. The individual supports are located at the actual support pad locations and accurately represent the independent non-linear behavior of each support.

5. Analytical Methods

The time-history effects of the cavity pressurization loads, internals loads and loops mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports.

Also, the reactor vessel displacements are applied as a time-history input to the dynamic reactor coolant loop blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown loads and reactor vessel displacements. Thus, the effect of vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

6. Results of the Analysis

As described, the reactor vessel and internals were analyzed for three postulated break locations. Table 3.9N-12 summarizes the displacements and rotations of and about a point representing the intersection of the centerline of the nozzle attached to the leg in which the break was postulated to occur and the vertical centerline of the reactor vessel. Positive vertical displacement is up and positive horizontal displacement is away from and along the centerline of the vessel nozzle in the loop in which the break was postulated to occur. These displacements were calculated using an assumed break opening area for the postulated pipe ruptures at the vessel nozzles of 144 in^2 and a double-ended rupture at the pump outlet nozzle. These areas are estimated prior to performing the analysis. Following the reactor coolant system structural analysis, the relative motions of the broken pipe ends are obtained from the reactor coolant loop blowdown analysis. The actual break opening area is then verified to be less than the estimated area used in the analysis and assures that the analysis is conservative.

The maximum loads induced in the vessel supports due to the postulated pipe break are given in Table 3.9N-13. These loads are per vessel support and are applied at the vessel nozzle pad. It is conservatively assumed that the maximum horizontal and vertical loads occur simultaneously and on the same support, even though the time-history results show that these loads occur neither simultaneously nor on the same support. The largest vertical loads are produced on the support opposite the broken nozzle. The largest horizontal loads are produced on the supports which are perpendicular to the broken nozzle horizontal centerline. Peak loads are based on a dynamic analysis of breaks 12 and 13 as defined in Table 3.6B-2 and Figure 3.6B-9. Analysis results are detailed in the Westinghouse Topical Report WCAP-12106 [15].

3.9N.1.4.7 Stress Criteria for Class 1 Components and Component Supports

All Class 1 components and supports are designed and analyzed for the design, normal, upset, and emergency conditions to the rules and requirements of the ASME Code, Section III. The design analysis or test methods and associated stress or load allowable limits that will be used to evaluation of faulted condition are those that are defined in Appendix F of the ASME code with the following supplementary option:

1. Elastic/inelastic system analysis and component/test load method

The test load method given in F 1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

The test load option was used to qualify the reactor pressure vessel nozzle support pads. To duplicate the loads that act on the pads during faulted conditions, the tests, which

utilized a one-eighth linear scale model, were performed by applying a unidirectional static load to the nozzle pad. The load on the nozzle pad was reacted by a support shoe which was mounted to the test fixture.

This modeling and application of load thus allows the maximum load capacity of the support pads to be accurately established. The test load, LT, was then determined by multiplying the maximum collapse load by sixty-four (ratio of prototype area to model area) and including temperature effects in accordance with the rules of the ASME Code, Section III.

The loads on the reactor vessel support pads, as calculated in the system analysis for faulted conditions are limited to the value of .80 LT. The tests performed and the limits established for test load method ensure that the experimentally obtained value for LT is accurate and that the support pad design is adequate for its intended function.

Loading combinations and allowable stresses for ASME Code, Section III, Class 1 components and supports are given in Table 3.9N-2 and 3.9N-3. For component support bolting materials, the stress limits and increase factors applicable for the faulted condition are provided in the discussion of Regulatory Guide 1.124 in FSAR Appendix 1A(N). For faulted condition evaluations, the effects of the SSE and loss of coolant accident (LOCA) are combined using the square root of the sum of the squares method. Information supporting this methodology is provided in References [18], [19], [20] and [21].

For faulted condition analysis of Class 1 branch piping attached to the reactor coolant loop, Equation (9) of ASME III Subsection NB-3652 is applied with a stress limit of 3.0 Sm. This criterion provides sufficient assurance that the piping will not collapse or experience cross distortion such that the function of the system would be impaired. The basis for this position is described in the Westinghouse response to NRC Question 110.34 on the RESAR-414 application (Docket No. STN 50-572), which subsequently received a Preliminary Design Approval (PDA) in November 1978.

3.9N.1.4.8 Analytical Methods for RCS Class 1 Branch Lines

The analytical methods used to obtain the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectrum method for seismic dynamic analysis, and dynamic structural analysis for the effect of a reactor coolant loop pipe break.

The integrated Class 1 piping/supports system model is the basic system model used to compute loadings on components, component and piping supports, and piping. The system models include the stiffness and mass characteristics of the Class 1 piping components, the reactor coolant loop, and the stiffness of supports which affect the system response. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

Static

The Class 1 piping system models are constructed for the WESTDYN computer program, which numerically describes the physical system. A network model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic

properties of the section are used to define the characteristic stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system. The support loads are also computed by multiplying the stiffness matrix by the displacement vector at the support point.

Seismic

The models used in the static analyses are modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping systems is accomplished by locating the total mass at points in the system which will appropriately represent the response of the distributed system. Effects of the primary equipment motion, that is, reactor vessel, steam generator, reactor coolant pump, and pressurizer, on the Class 1 piping system are obtained by modeling the mass and the stiffness characteristics of the primary equipment and loop piping in the overall system model.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Shock suppressors which resist rapid motions are also included in the analysis. The solution for the seismic disturbance employs the response spectra method. This method employs the lumped mass technique, linear elastic properties, and the principle of model superposition.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system and the response from differential anchor motions. The stresses resulting from the anchor motions are considered to be secondary and, therefore, are included in the fatigue evaluation.

Loss of Coolant Accident

The mathematical models used in the seismic analyses of the Class 1 lines are also used for RCL pipe break effect analysis. To obtain the proper dynamic solution both for lines attached to the unbroken loops and lines attached to the broken loop, the time history deflections from the analysis of the reactor coolant loop are applied at branch nozzle connections.

Fatigue

A thermal transient heat transfer analysis is performed for each different piping component on all the Class 1 branch lines. The normal, upset, and test condition transients identified in Section 3.9.1.1 are considered in the fatigue evaluation.

The thermal quantities T_1, T_2 and a_aT_a, $-a_bT_b$ are calculated on a time history basis, using a one-dimensional finite difference heat transfer computer program. Stresses due to these quantities were calculated for each time increment using the methods of NB-3650 of ASME III.

For each thermal transient, two loadsets are defined, representing the maximum and minimum stress states for that transient.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus insuring the most conservative combinations of seismic loads are used in the stress evaluation.

The WESTDYN computer program is used to calculate the primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors and cumulative usage factors for all possible load set combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.

The combination of load sets yielding the highest alternating stress intensity range is determined and the incremental usage factor calculated. Likewise, the next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9N.2 DYNAMIC TESTING AND ANALYSIS

3.9N.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

This section is discussed in 3.9B.2.1.

3.9N.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

The operability of Category I mechanical equipment must be demonstrated if the equipment is determined to be active, i.e., mechanical operation is relied on to perform a safety function. The operability of active Class 2 and 3 pumps, active Class 1, 2, or 3 valves, and their respective drives, operators and vital auxiliary equipment will be shown by satisfying the criteria given in Section 3.9N.3.2. Other active mechanical equipment will be shown operable by either testing, analysis or a combination of testing and analysis. The operability programs implemented on this other active equipment will be similar to the program described in Section 3.9N.3.2 for pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10N for electrical equipment will be used to demonstrate operability if the component is mechanically or structurally complex such that its response cannot be adequately predicted analytically. Analysis may be used if the equipment is amenable to modeling and dynamic analysis.

Inactive Seismic Category I equipment will be shown to have structural integrity during all plant conditions in one of the following manners: 1) by analysis satisfying the stress criteria applicable to the particular piece of equipment, or 2) by test showing that the equipment retains its structural integrity under the simulated test environment.

A list of Seismic Category I equipment and the method of qualification used is provided in Table 3.2-2.

3.9N.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady State Conditions

The vibration characteristics and behavior due to flow induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations cannot be practically performed independently of the dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicate that the most important forcing functions are flow turbulence, and pump related excitation. The relevance of such excitations depends on many factors such as type and location of component and flow conditions.

The effects of these forcing functions have been studied from test runs on models, prototype plants and in component tests, References [4], [5], and [6].

The Indian Point No. 2 plant has been established as the prototype for a four-loop plant internals verification program and was fully instrumented and tested during hot functional testing. In addition, the Trojan plant and the Sequoyah No. 1 plant have provided prototype data applicable to CPNPP, Reference [5], [6], [7] and [14].

The CPNPP is similar to Indian Point No. 2; the only significant differences are the modifications resulting from the use of 17 x 17 fuel, replacement of the annular thermal shield with neutron shielding panels, and the change to the UHI-style inverted top hat support structure configuration. These differences are addressed below.

1. 17 x 17 fuel

   The only structural changes in the internals resulting from the design change from the 15 x 15 to the 17 x 17 fuel assembly are the guide tube and control rod drive line. The new 17 x 17 guide tubes are stronger and more rigid, hence they are less susceptible to flow induced vibration. The fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation of internals vibration is expected from the vibration with the 15 x 15 fuel assemblies.

2. Neutron shielding pads lower internals

   The primary cause of core barrel excitation is flow turbulence, which is not affected by the upper internals [5]. The vibration levels due to core barrel excitation for Trojan and CPNPP both having neutron shielding pads are expected to be similar. The coolant inlet temperature of CPNPP is slightly higher than Trojan 1 and the flowrate is slightly higher.

Scale model tests show that core barrel vibration varies as velocity raised to a small power [4]. The difference in fluid density and flowrate result in an approximately 6% higher core barrel vibration for CPNPP than for Trojan 1. However, scale model test results [4] and results from Trojan [7] show that core barrel vibration of plants with neutron shielding pads is significantly less than that of plants with thermal shields. This information and the fact that low core barrel stresses with large safety margins were measured at Indian Point No. 2 (thermal shield configuration) lead to the conclusion that stresses approximately equal to or less than those of Indian Point No. 2 result on the CPNPP internals with the attendant large safety margins.

3. UHI-style inverted top hat upper support configuration

The components of the upper internals are excited by turbulent forces due to axial and cross flows in the upper plenum and by pump related excitations [5], [7]. Sequoyah and CPNPP have the same basic upper internals configuration, therefore, the general vibration behavior is not changed. Comanche Peak upper internals adequacy has been determined from data from the instrumented plant test at Sequoyah 1, scale model tests and numerous operation plants. The results of testing at Sequoyah 1, [14] showed that the components are excited by flow induced and pump related excitations. Analyses of the data indicate that the instrumented components have adequate factors of safety, that random flow-induced responses are adequately predicted by scale models and that the margins are higher with the core in place than during hot functional testing.

In addition, the Comanche Peak upper internals configuration was tested in a scale model using the same modeling techniques as for the scale model tests of the UHI configuration. The responses of the Comanche Peak upper internals have been calculated using the Sequoyah 1 and scale model information. The results show adequate factors of safety for all components.

The original test and analysis of the four-loop configuration is augmented by References [4], [5], [6], [7] and [14] to cover the effects of successive hardwater modifications.

3.9N.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because of CPNPP reactor internals design configuration is well characterized, as was discussed in Section 3.9N.2.3, it is not considered necessary to conduct instrumented tests of the CPNPP hardware. The requirements of Regulatory Guide 1.20 will be met by conducting the confirmatory preoperational testing examination for integrity per Section D, of Regulatory Guide 1.20, “Regulations for Reactor Internals Similar to the Prototype Design.” This examination will include some 35 points with special emphasis on the following areas.

1. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place.

2. The lateral, vertical and torsional restraints provided within the vessel.

3. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

4. Those others locations on the reactor internal components which are similar to those which were examined on the prototype Indian Point No. 2 and on Trojan 1 and Sequoyah 1.

5. The inside of the vessel will be inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts or foreign material are in evidence.

A particularly close inspection will be made on the following items or areas using a 5X or 10X magnifying glass where applicable.

1. Lower internals

   a. Upper barrel to flange girth weld.

   b. Upper barrel to lower barrel girth weld.

   c. Upper core plate aligning pin. Examine bearing surfaces for any shadow marks, burnishing, buffing or scoring. Inspect welds for integrity.

   d. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

   e. Baffle assembly locking devices. Check for lockweld integrity.

   f. Lower barrel to core support girth weld.

   g. Neutron shield panel screw locking devices and dowel pin welds. Examine the interface surfaces for evidence of tightness and for lockweld integrity.

   h. Radial support key welds.

   i. Insert screw locking devices. Examine soundness of lockwelds.

   j. Core support columns and instrumentation guides. Check joints for tightness and soundness of the locking devices.

   k. Secondary core support assembly weld integrity.

   l. Lower radial support keys and inserts. Examine for shadow marks, burnishing, buffing or scoring. Check the integrity of the lockwelds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing or shadow marks which would indicate pressure loading and relative motion between the two parts. Some scoring of engaging surfaces is also possible and acceptable.

   m. Check gaps between baffle plate joints.

2. Upper internals

   a. Thermocouple conduits, clamps and couplings.

   b. Guide tube, support column, orifice plate, and thermocouple assembly locking devices.

   c. Support column and thermocouple conduit assembly clamp welds.

   d. Upper core plate alignment inserts. Examine for shadow marks, burnishing, buffing or scoring. Check the locking devices for integrity of lockwelds.

   e. Thermocouple conduit gusset and clamp welds (where applicable).

   f. Thermocouple conduit end-plugs. Check for tightness.

   g. Guide tube enclosure welds, tube-transition plate welds and card welds.

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals will be subjected to a total operating time at greater than normal full-flow conditions (four pumps operating) of at least 240 hours. This provides a cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition there will be some operating time with only one, two and three pumps operating.

Pre- and post-hot functional inspection results serve to confirm predictions that the internals are well behaved. When no signs of abnormal wear, no harmful vibrations are detected and no apparent structural changes take place, the four-loop core support structures are considered to be structurally adequate and sound for operation.

3.9N.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

Analysis of the reactor internals for blowdown loads resulting from a loss of coolant accident is based on the time-history response of the internals to simultaneously applied blowdown forcing functions.

The reactor internals hydraulic pressure transients were calculated including the assumptions that the structural motion is coupled with the pressure transients. This is referred to as fluid-structure-interaction. The hydraulic analysis considers the fluid structure interaction of the core barrel by accounting for the deflections of the constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-9735 [27] for (Unit 1) and WCAP-8708 [8] (Unit 2).

Unit 1

The horizontal and vertical forces exerted on reactor internals and the core, following a LOCA, are computed by employing the MULTIFLEX 3.0 [27], which is an enhancement and extension of the MULTIFLEX 1.0 [22], NRC reviewed and approved computer code developed for the space-time dependent analysis of nuclear power plants. MULTIFLEX 3.0 has been accepted by the NRC for several other applications [23, 24, 25 and 26] and also has been extensively used for the LOCA analyses of reactor internals components of numerous other 2, 3, and 4 loop nuclear power plants.

Unit 2

The horizontal and vertical models are coupled at the elevation of the primary nozzle centerlines. Node 1 of the horizontal model is coupled with node 2 of the vertical model at the reactor vessel nozzle elevation. This coupled node has external restraints characterized by a 3 x 3 matrix which represents the reactor coolant loop stiffness characteristics, by linear horizontal springs which describe the tangential resistance of the supports, and by individual non-linear vertical stiffness elements which provide downward restraint only. The supports are represented in the horizontal and vertical models (Figures 3.9N-12 and 3.9N-13) are not indicative of the complexity of the support system used in the analysis. The individual supports are located at the actual support pad locations and accurately represent the independent non-linear behavior of each support.

System blowdown analysis of the reactor internals carried out using DARI-WOSTAS computer code (Reference 10) is a non-linear elastic analysis, wherein the nonlinearities are due to system mechanical gaps. ASME Code Section III, Subsection NG does allow the system analyses to be performed on elastic basis.

In the mathematical models of the Blowdown System Analysis, the values used for structural damping are within the Regulatory guide 1.61 (i.e., 4 percent of critical damping) value. For impact damping a coefficient of restitution C_R value of about 55 percent was selected to consider impact losses in the analysis. This C_R value is considered to be conservative based on some published results. The gaps considered in the analysis are the physical gaps based on nominal dimensions obtained from assembly drawings.

In addition, because of the complexity of the system and the components, it is necessary to use finite element stress analysis codes to provide more detailed information at various points.

A blowdown digital computer program [8] which was developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in Pressurized Water Reactor coolant systems during a loss of coolant accident is applied to the subcooled, transition, and saturated two-phase blowdown regimes. This is in contrast to programs such as WHAM [9] which are applicable only to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place. This blowdown code is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy, are solved numerically using a fixed mesh in both space and time.

Although spatially one dimension conservation laws are employed, the code can be applied to describe three dimensional system geometries by use of the equivalent piping networks. Such

piping networks may contain any number of pipes of channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, etc., are considered.

The blowdown code evaluates the pressure and velocity transients for a maximum of 2400 locations throughout the system. Theses pressure and velocity transients are stored as a permanent tape file and are made available to the program FORCE 2 (Reference 8) which utilizes a detailed geometric description in evaluating the loadings on the reactor internals.

Each reactor component for which FORCE 2 (Reference 8) calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element.

2. Flow stagnation on, and unrecovered orifice losses across the element.

3. Friction losses along the element.

Input to the code, in addition to the blowdown pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

The mechanical analysis [10] has been performed using conservative assumptions. Some of the more significant assumptions are:

1. The mechanical and hydraulic analyses have considered the effect of hydroelasticity.

2. The reactor internals are represented by a multi-mass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. The modeling is conducted in such as way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs.

The model described is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the vertical direction. This model is conservative in the sense that further mass-spring resolution of the system would lead to further attenuation of the shock effects obtained with the present model.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loadings on the components. A one millisecond severance time is taken as the limiting case.

In the case of the hot leg break, the vertical hydraulic forces produce an initial upward lift of the core. A rarefraction wave propagates through the reactor hot leg nozzle into the interior of the upper core barrel. Since the wave has not reached the flow annulus on the outside of the barrel, the upper barrel is subjected to an impulsive compressive wave. Thus, dynamic instability (buckling) or large deflections of the upper core barrel, or both, is a possible response of the

barrel during hot leg break results in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of the cold leg break, a rarefraction wave propagates along a reactor inlet pipe, arriving first at the core barrel at the inlet nozzle of the broken loop. The upper barrel is then subjected to a non-axisymmetric expansion radial impulse which changes at the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel. After the cold leg break, the initial steady state hydraulic lift forces (upward) decreases rapidly (within a few milliseconds) and the increase in the downward direction. These cause the reactor core and the lower support structure to move initially downward.

If the simultaneous seismic event with the intensity of the SSE is postulated with the loss of coolant accident, the imposed loading on the internals component may be additive in certain cases and therefore the combined loading must be considered. In general, however, the loading imposed by the earthquake is small compared to the blowdown loading.

The summary of the mechanical analysis follows:

Vertical Excitation Model for Blowdown

For the vertical excitation, the reactor internals are represented by a multi-mass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. Also incorporated in the multi-mass system is a representation of the motion of the fuel elements relative to the fuel assembly grids. The fuel elements in the fuel assemblies are kept in position by friction forces originating from the preloaded fuel assembly grid fingers. Coulomb type friction is assumed in the event that sliding between the rods and the grid fingers occurs. In order to obtain an accurate simulation of the reactor internals response, the effects of the internal damping, clearances between various internals, snubbing action caused by solid impact, Coulomb friction induced by fuel rod motion relative to the grids, and preloads in hold down springs have been incorporated in the analytical model. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs.

The appropriate dynamic differential equations for the multi-mass model describing the aforementioned phenomena are formulated and the results obtained using a digital computer program which computes the response of the multi-mass model when excited by a set of time dependent forcing functions. The appropriate forcing functions are applied simultaneously and independently to each of the masses in the system. The results from the program give the forces, displacements and deflections as functions of time for all the reactor internals components (lumped masses). Reactor internals response to both hot and cold leg pipe ruptures were analyzed.

Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

Core Barrel - For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse.

The barrel is then analyzed for dynamic buckling using the following conservative assumptions:

1. The effect of the fluid environment is neglected.

2. The shell is treated as simply supported.

During cold leg blowdown, the upper barrel is subjected to a non-axisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold leg blowdown is performed as follows:

1. The core barrel is treated as a simply supported cylindrical shell of constant thickness between the upper flange weldment and the lower support plate weldment. No credit is taken for the supports at the barrel midspan offered by the outlet nozzles. This assumption leads to conservative deflection estimates of the upper core barrel.

2. The core barrel is analyzed as a shell with two variable sections to model the support flange and core barrel.

3. The barrel with the core and thermal shielding pads is analyzed as a beam fixed at the top and elastically supported at the lower radial support and the dynamic response is obtained.

Guide Tubes - The guide tubes in closest proximity to the outlet nozzle of the ruptured loop are the most severely loaded during a blowdown. The transverse guide tube forces decrease with increasing distance from the ruptured nozzle location.

All of the guide tubes are designed to maintain the function of the control rods for a break size of 144 $in.^2$ and smaller. No credit for the function of the control rods is assumed for break size areas above 144 $in.^2$. However, the design of the guide tube will permit control rod operation in all but four control rod positions, which is sufficient to maintain the core in a subcritical configuration, for break sizes up to a double-ended hot leg break. This double-ended hot leg break imposes the limiting lateral guide tube loading.

Upper Support Columns - Upper support columns located close to the broken nozzle hot leg break will be subjected to transverse loads due to cross flow. The loads applied to the columns are computed with a method similar to the one used for the guide tubes, i.e., by taking into consideration the increase in flow across the column during the accident. The columns are studied as beams with variable section and the resulting stresses are obtained using the reduced section modulus and appropriate stress risers for the various section.

The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, and between fuel assemblies and baffle plates are considered in the analysis. Linear analysis will not provide information about the impact forces generated when components impinge each other, but can, and is, applied prior to gap closure. Reference [10] provides further details of the blowdown method used in the analysis of the reactor internals.

The stresses due to the Safe Shutdown Earthquake (vertical and horizontal components) are combined with the blowdown stresses in order to obtain the largest principal stress and deflection.

All reactor internals components were found to be within acceptable stress and deflection limits for both hot leg and cold leg loss of coolant accidents occurring simultaneously with the Safe Shutdown Earthquake.

The results obtained from the linear analysis indicate that during blowdown, the relative displacement between the components will close the gaps and consequently the structures will impinge on each other, making the linear analysis unrealistic and forcing the application of nonlinear methods to study the problem. Although linear analysis will not provide information about the impact forces generated when components impinge on each other, it can, and is, applied prior to gap closure. The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, between fuel assemblies and baffle plates, and between the control rods and their guide paths were considered in the analysis. Both static and dynamic stress intensities are within acceptable limits. In addition, the cumulative fatigue usage factor is also within the allowable usage factor of unity.

The stresses due to the Safe Shutdown Earthquake (vertical and horizontal components) were combined in the most unfavorable manner with the blowdown stresses in order to obtain the largest principal stress and deflection.

The results indicate that the maximum deflections and stress in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimental to assure control rod insertion. For the guide tubes deflected above the no loss of function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no loss of function limit.

3.9N.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

As stated in Section 3.9N.2.3, it is not considered necessary to conduct instrumented tests of the CPNPP reactor vessel internals. Adequacy of these internals will be verified by use of the Sequoyah and Trojan results. References [5] and [6] describe predicted vibration behavior based on studies performed prior to the plant tests. These studies, which utilize analytical models, scale model test results, component tests, and results of previous plant tests, are used to characterize the forcing functions and establish component structural characteristics so that the flow induced vibratory behavior and response levels for CPNPP are estimated. These estimates are then compared to values deduced from plant test data obtained from the Sequoyah and Trojan internals vibration measurements programs.

3.9N.3 ASME CODE CLASS 1, 2 AND 3 COMPONENTS, COMPONENT SUPPORTS AND CORE SUPPORT STRUCTURES

The ASME Code Class components are built to accepted ASME Code, Section III requirements. For Code Class 1 components, very stringent requirements are imposed and are met. For Code Class 2 and 3 components, the requirements are less stringent but adequate, in accordance with the lower classification.

See Section 3.9N.1 for more detailed discussions on ASME Code Class 1 components.

3.9N.3.1 Loading Combinations Design Transients, and Stress Limits

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

3.9N.3.1.1 Design Loading Combinations

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3.9N-4. The design loading combinations are categorized with respect to normal, upset, emergency, and faulted conditions. Loads for each plant operating condition are combined by the absolute sum method. Stress limits for each of the loading combinations are component oriented and are presented in Tables 3.9N-5, 3.9N-6, 3.9N-7, and 3.9N-8 for tanks, inactive[(a)] pumps, active pumps, and valves, respectively. Active[(b)] pumps and valves are discussed in Section 3.9N.3.2. The component supports are designed in accordance with the ASME Code, Section III, Subsection NF. Details regarding the application and usages of Code Case N-318 are as listed in Table 3.9B-1F.

For core support structures, design loading conditions are given in Section 4.2.2. Loading conditions are discussed in Section 4.2.3.

In general, for reactor internals components and for core support structures the criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and stability of the parts in addition to a stress criterion to assure integrity of the components.

For the loss of coolant plus the Safe Shutdown Earthquake condition, deflections of critical internal structures are limited. In a hypothesized downward vertical displacement of the

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a. Inactive components are those whose operability are not relied upon to perform a safety function during the transients or events considered in the respective operating condition category.

b. Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories.

internals, energy absorbing devices limit the displacement after contacting the vessel bottom head, ensuring that the geometry of the core remains intact.

The following mechanical functional performance criteria apply:

1. Following the design basis accident, the functional criterion to be met for the reactor internals is that the plant shall be shutdown and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This criterion implies that the deformation of critical components must be kept sufficiently small to allow core cooling.

2. For large breaks, the reduction in water density reduces the reactivity of the core, thereby shutting down the core whether the rods are tripped or not. The subsequent refilling of the core by the Emergency Core Cooling System uses borated water to maintain the core in a subcritical state. Therefore, the main requirement is to assure effectiveness of the Emergency Core Cooling System. Insertion of the control rods, although not needed, gives further assurance of ability to shut the plant down and keep it in a safe shutdown condition.

3. The inward upper barrel deflections are controlled to ensure no contacting of the nearest rod cluster control guide tube. The outward upper barrel deflections are controlled in order to maintain an adequate annulus for the coolant between the vessel inner diameter and core barrel outer diameter.

4. The rod cluster control guide tube deflections are limited to ensure operability of the control rods.

5. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited.

Methods of analysis of testing for core support structures are discussed in Sections 3.9N.1.3, 3.9N.1.4.1, 3.9N.2.3, 3.9N.2.5, and 3.9N.2.6. Stress limits and deformation criteria are discussed in Section 4.2.1.

3.9N.3.1.2 Design Stress Limits

Design stress limits established for the components are sufficiently low to assure that violation of the pressure retaining boundary will not occur. Limits, for each of the loading combinations, are component oriented and are presented in Tables 3.9N-5 through 3.9N-8.

Specifically, maximum calculated support stresses for each of the loading combinations, described in Table 3.9N-4, are limited by the stress criteria presented in Table 3.9N-7. These criteria assure that stresses in the pump supports remain essentially elastic; thus, a pump/motor misalignment will not occur. The pump support stress criteria, along with the balance of the operability program described in Section 3.9N.3.2, demonstrates that pump operability would not be impaired during or following any of the loading conditions postulated to occur.

In addition, design procedures for Level C and D stress limits, as delinated in Section III of the ASME Code, provide adequate assurance that structural discontiuities in piping, tanks and vessels will retain their specified geometric configuration during the improbable emergency and faulted condition events. These procedures provide adequate margins to assure the primary

pressure boundary of components and the functions of component supports. Conservative stress indices and intensification factors based upon analytical and experimental results as specified in the Code are used for analysis in the areas of structural discontiuity of piping, tanks and vessels. The use of these proven procedures and conformance with ASME III requirements provides an acceptable basis to assure the functional capability of these components.

3.9N.3.2 Pump and Valve Operability Assurance

Mechanical equipment classified as safety-related must be shown capable of performing its function during the life of the plant under postulated plant conditions. Equipment with faulted condition function requirements include active pumps and valves in fluid systems important to safety. Seismic analysis is presented in Section 3.7N and covers all safety-related mechanical equipment. A list of all active pumps supplied by Westinghouse is presented in Table 3.9N-9. As can be seen on this table, CPNPP has no Class 1 active pumps. Active valves supplied by Westinghouse are listed in Table 3.9N-10.

The methods used to demonstrate the operability of active pumps and valves are described in this section. Equipment specifications include requirements for operability under the specified plant conditions and define appropriate acceptance criteria to ensure that the program requirements defined below are satisfied.

All active pumps are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shops tests include: 1) hydrostatic tests of pressure-retaining parts to 150 percent times the design pressure times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature, 2) seal leakage tests at the same pressure used in the hydrostatic tests, 3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements and other pump/motor parameters. Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined, by the manufacturer, based on the bearing material, clearances, oil type, and rotational speed. These limits are approved by Westinghouse. A comparison of the Westinghouse specified bearing vibration limits to industry standards is presented in Figure 3.9N-4. Both limits are below appropriate limits specified to the manufacturer for the design of each active pump. After the pump is installed in the plant, it undergoes the cold hydro tests, hot functional tests, and the required periodic inservice inspection and operation. These tests demonstrate that the pump will function as required during all normal operating conditions for the design life of the plant.

In addition to these tests, the safety-related active pumps are qualified for operability during SSE condition by assuring that 1) the pump will not be damaged during the seismic event, and 2) the pump will continued operating despite the SSE loads.

The pump manufacturer is required to show that the pump will operate normally (i.e., perform their safety function) when subjected to the maximum seismic accelerations and maximum faulted nozzle loads. Maximum nozzle loads imposed by the piping systems and the seismic accelerations imposed due to building locations and/or piping system design are confirmed in this process to be less than the maximum nozzle loads and seismic loads used for component design. Thus, active pumps are qualified for loads which are at least as severe as the maximum loads which are expected to occur as a result of faulted condition loadings.

It is required that test or dynamic analysis be used to show the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, is considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 2.1 g in two orthogonal horizontal directions and 2.1 g vertical acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) will be of short duration. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads will be limited to the material elastic limit, as indicated in Table 3.9N-7. Stress limits which are applied to active pumps, in Table 3.9N-7, are only nominally higher than Level B stress limits and less than Level C, as delineated in Section III of the ASME Code. This assures that the pumps will not experience permanent deformation or otherwise be damaged during the short duration of the faulted condition event.

The average membrane stresses (σ_m) for the faulted condition loads are maintained at 1.2S, or approximately 0.75 σ_y (σ_y = yield stress) and the maximum stress in local fibers (σ_m + bending stress σ_b) are limited to 1.8S, or approximately 1.1 σ_y. In addition, the pump casing stresses caused by the maximum seismic nozzle loads are limited to stresses outlined in Table 3.9N-7. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9N-7 as allowables assures that critical parts of the pump will not be damaged during the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic event.

If the natural frequency is found to be below 33 Hz, an analysis will be performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations will be determined using the same conservatisms contained in the 2.1 g horizontal and 2.1 g vertical accelerations used for “rigid” structures. The static analysis will be performed using the adjusted accelerations; the stress limits stated in Table 3.9N-7 must still be satisfied.

The second criterion necessary to assure operability is that the pump will function throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely sized, i.e., with no rotation. Typically, the rotor can be seized 5 full seconds before a circuit-breaker, to prevent damage to the motor, shuts down the pump. However, the high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event, will prevent the rotor from losing its function. In actuality, the seismic loadings will cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump will not shutdown during the SSE and will operate at the design speed despite the SSE loads.

To complete the seismic qualification procedures, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment which is identified to be vital to the operation of the pump or pump motor, and which is not qualified for operation during the pump analysis or motor qualifications, is also separately qualified for operation at the

accelerations it would see at its mounting. The pump motor and vital auxiliary equipment is qualified by meeting the requirements of IEEE Standard 344-1975. If the testing option is chosen, sine-beat testing for electrical equipment will be justified by satisfying one or more of the following requirements to demonstrate that multi-frequency response is negligible to the sinebeat input of sufficient magnitude to conservatively account for this effect.

1. The equipment response is basically due to one mode.

2. The sine-beat response spectra envelopes the floor response spectra in the region of significant response.

3. The floor response spectra consists of one dominant mode and has a narrow peak at this frequency.

The degree of coupling in the equipment will, in general, determine if a single or multi-axis test is required. Multi-axis testing will be required if there is considerable cross coupling. If coupling is very light, then single axis testing is justified. Or, if the degree of coupling can be determined, then single axis testing can be used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

From the previous arguments, the safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings, and therefore, will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

The functional ability of active pumps after a faulted condition is assured since only normal operating loads and steady state nozzle loads will then exist. Since it is demonstrated that the pumps are not damaged during the faulted condition, the post-faulted condition operating loads will be identical to the normal plant operating loads. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions are limited by the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

Safety-related active valves must perform their mechanical motion in times of an accident. Assurance is supplied that these valves will operate during a seismic event. Qualification tests accompanied by analyses are conducted for all active valves.

The safety-related valves are subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, functional tests to verify that the valve will open and close within the specified time limits when subjected to the design differential pressure, operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging radiation, accident environment simulation, etc.) according to IEEE Standard 382. Cold hydro qualification tests, hot functional qualification tests, periodic inservice inspections, and periodic inservice operation are performed in-situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant. The valves are designed using either stress analyses or the pressure containing minimum wall thickness requirements. On all active valves, an analysis of the extended structure is also performed for static equivalent seismic SSE loads applied at the

center of gravity of the extended structure. Stress limits imposed on the non-ASME Code extended structures of active valves assure that the extended structures do not experience permanent deformation or damage and that the functional capability of the valve is not impaired.

Such analyses verify that active valves will perform their safety function when subjected to the most severe loads which would be imposed by the SSE coincident with the maximum faulted plant condition nozzle loads. The maximum nozzle loads imposed by the piping systems and the seismic accelerations imposed due to building locations and/or piping system design are confirmed in this process to be less than the maximum nozzle loads and seismic loads used for component design. Thus, active valves are qualified for loads which are at least as severe as the maximum loads which are expected to occur as a result of faulted condition loadings. Stress limits for active valves are presented in Tables 3.9N-3 (Class 1) and 3.9N-8 (Class 2 & 3). The maximum stress limits allowed in these analyses show structural integrity and are the limits recommended by the ASME for the particular ASME class of valve analyzed.

In addition to these tests and analyses, representative valves of each design type will be tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits. The testing procedures are described below.

The valve will be mounted in a manner which will conservatively represent typical valve installations. In general, active valves are supported by the pipe attached to the valve. The valve will include the operator and all accessories normally attached to the valve in service. The operability of the valve during SSE is demonstrated by satisfying the following criteria:

1. All the active valves are designed to have a first natural frequency which is greater than 33 Hz. This is shown by suitable test or analysis.

2. The extended structure of the valve system will be statically deflected an amount equal to that determined by an analysis as representing SSE accelerations applied at the center of gravity of the operator along the direction of the weakest axis of the yoke. The design pressure of the valve will be simultaneously applied to the valve during the static deflection tests.

3. The valve will then be operated while in the deflected position. The valve must perform its safety-related function within the specified operating time limits.

4. Motor operators, pilot solenoid valves and limit switches necessary for operation will be qualified as operable during SSE by appropriate IEEE Seismic Qualification Standards, prior to their installation on the valve.

The accelerations which are used for the static valve qualification are equivalent, as justified by analysis, to the simultaneous application of not less than 2.1 g in two orthogonal horizontal directions and not less than 2.1 g in the vertical direction. Accelerations imparted to the valve assemblies by the piping are not greater than the accelerations used for the static valve qualification.

If the frequency of the valve is less than 33 Hz, a dynamic analysis of the valve will be performed to determine the equivalent acceleration which will be applied during the static test. The analysis will provide the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted

accelerations will be determined using the same conservatisms contained in the 2.1 g horizontal and 2.1 g vertical accelerations used for “rigid” valves. The adjusted accelerations will then be used in the static analysis and the valve operability will be assured by the methods outlined in steps (2) to (4) above using the modified acceleration input.

The above testing program applies to valves with extended structures. The testing will be conducted on a representative number of valves. Valves from each of the primary safety-related design types will be tested. Valve sizes which cover the range of sizes in service will be qualified by the tests and the results will be used to qualify all valves within the intermediate range of sizes.

Valves which are safety-related but can be classified as not having an extended structure, such as check valves and safety-relief valves, will be considered separately.

The check valves are characteristically simple in design and their operation will not be affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact and there are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads due to maximum seismic excitation will not affect the functional ability of the valve since the valve designed to be isolated from the casing wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any casing disortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard design or analysis methods, the ability of the valve to operate is assured by the design features. In addition to these design considerations, the valve will also undergo 1) in-shop hydrostatic tests, 2) in-shop seat leakage test, and 3) periodic in-situ valve exercising and inspection to assure the functional ability of the valve.

The pressurizer safety valves are qualified by the following procedures. These valves are also subjected to tests and analysis similar to check valves: stress and deformation analyses for SSE loads, in-shop hydrostatic and seat leakage tests, and periodic in situ valve inspection. In addition to these tests, a static load equivalent to the SSE is applied at the top of the bonnet and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assures its overpressurization safety capabilities during a seismic event.

Using the methods described, all the safety-related valves in the systems are qualified for operability during a seismic event. These methods conservatively simulate the seismic event and ensure that the active valves will perform their safety-related function when necessary.

3.9N.3.3 Design and Installation Details in Mounting of Pressure Relief Devices (Pressurizer Safety and Relief System)

The pressurizer safety and relief valve discharge piping systems provide overpressure protection for the RCS. The three spring-loaded safety valves, located on top of the pressurizer, are designed to prevent system pressure from exceeding design pressure by more than 10 percent. The two power-operated relief valves, also located on top of the pressurizer, are designed to prevent system pressure from exceeding the normal operating pressure by more than 100 psi. A water seal is maintained upstream of each valve to minimize leakage. Condensate accumulation on the inlet side of each valve prevents any leakage of hydrogen gas or steam through the valves. The valve outlet side is sloped to prevent the formulation of additional water pockets.

If the pressure exceeds the set point and the valves open, the water slug from the loop seal discharges. The water slug, driven by high system pressure, generates transient thrust forces at each location where a change in flow direction or area occurs. The valve discharge conditions considered in the analysis of the PSARV piping systems are as follows: 1) the three safety valves are assumed to open simultaneously while the relief valves remain closed, and 2) the two relief valves open simultaneously while the safety valves are closed.

For each pressurizer safety and relief piping system, an analytical hydraulic mode is developed. The piping from the pressurizer nozzle to the relief tank nozzle is modeled as a series of control volumes and flow paths. The pressurizer is modeled as a reservoir which contains steam at constant pressure (2500 psia for safety system and 2350 psia for relief system) and at constant temperature of 680°F. The pressurizer relief tank is modeled as a sink which contains steam and water mixture.

Fluid acceleration inside the pipe generates reaction forces on all segments of the line which are bounded at either end by an elbow or bend. Reactor forces resulting from fluid pressure and momentum variations are calculated. These forces can be expressed in terms of the fluid properties available for the transient hydraulic analysis. The momentum equation can be expressed in vector form as:

$$\Sigma \vec{F}_{CV} = \frac{1}{g_C}\frac{\partial}{\partial t}\int_v \vec{\gamma} u dv + \frac{1}{g_C}\int_A \gamma \vec{u}(u \cdot \vec{n} dA)$$

From which the total force on the pipe can be derived:

$$\Sigma F_{pipe} = F_1 - F_S - F_2 - f_{s1} r_1 \frac{(1-\cos\alpha_1)}{\sin\alpha_1} - f_{s2} r_2 \frac{(1-\cos\alpha_2)}{\sin\alpha_2}$$

$$= \frac{r_1}{g_C}\frac{(1-\cos\alpha_1)}{\sin\alpha_1}\frac{\partial W}{\partial t}\bigg|_{Bend1} + \frac{r_2}{g_C}\frac{(1-\cos\alpha_2)}{\sin\alpha_2}\frac{\partial W}{\partial t}\bigg|_{Bend2}$$

$$+ \frac{1}{g_C}\int_{pipe}^{straight}\frac{\partial W}{\partial t} d\ell$$

Where $W = \frac{\gamma \vec{u}}{g_C}$ weight density. g_C = gravitational constant

The terms are indicated in Figure 3.9N-4a.

Unbalanced forces are calculated for each straight segment of pipe from the pressurizer to the relief tank. The hydraulic analysis includes the effects of water slug discharge and two phase flow. The time histories of these forces are used for the subsequent structural analysis of the pressurizer safety and relief lines.

The structural model used in the seismic analysis of the safety and relief lines is modified for the valve thrust analysis to represent the safety and relief valve discharge. The time-history

hydraulic forces are applied to the piping system lump mass points. The dynamic solution for the valve thrust is obtained by using a modified predictor-corrector-integration technique and normal mode theory.

The time-history solution is performed in subprogram FIXFM. The input to this subprogram consists of the natural frequencies and normal modes, applied forces, and nonlinear elements. The natural frequencies and normal modes for the modified pressurizer safety and relief line dynamic model are determined with the WESTDYN program. The support loads are computed by multiplying the support stiffness matrix and the displacement vector at each support point. The time- history displacements of the FIXFM subprogram are used as input to the WESDYN2 subprogram to determine the internal forces, deflections, and stresses at each end of the piping elements.

The loading combinations considered in the analysis of the PSARV piping are given in Table 3.9N.2. Additional load combinations for consideration of valve thrust are indicated in Table 3.9N-22, as well as the applicable stress criteria.

3.9N.3.4 Component Supports

See Section 3.9N.1 for ASME Code Class 1 component supports.

Class 2 and 3 supports are designed as follows:

Class 2 and 3 component supports for the NSSS are generally of linear or plate and shell type; however, standard component supports may be used. The mandatory date for compliance with ASME B&PV Code, Section III, Subsection NF is July 1, 1974. Therefore, the standard component supports purchased prior to July 1, 1974 satisfy the requirements for the linear type presented in this Section. Those procured after July 1, 1974 comply with the requirements of Subsection NF.

1. Linear

   a. Normal - The allowable stresses of Appendix XVII of ASME Section III, as referenced in Subsection NF, are employed for normal condition limits.

   b. Upset - Stress limits for upset conditions are the same as normal condition stress limits. This is consistent with Subsection NF of ASME Section III (see NF-3320).

   c. Emergency - For emergency conditions, the allowable stresses or load ratings are 33 percent higher than those specified for normal conditions. This is consistent with Subsection NF of ASME Section III in which (see NF-3231) limits for emergency conditions are 33 percent greater than the normal condition limits.

   d. Faulted - Section 3.9N.1 specifies limits which assure that no large plastic deformations will occur (Stress <1.2 Sy). If any inelastic behavior is considered in the design, detailed justification will be provided for this limit. Otherwise the supports for active components will be designed so that stresses are less than or equal to S_y. Thus the operability of active components will not be endangered by the supports during faulted conditions.

2. Plates and shells

   a. Normal - Normal condition limits are those specified in Subsection NF of ASME Section III (see NF-3320).

   b. Upset - Limits for upset conditions equal normal condition limits and are consistent with Subsection NF of ASME Section III (see NF-3320).

   c. Emergency - For emergency conditions, the allowable stresses or load ratings are 20 percent higher than those specified for normal conditions.

   d. Faulted - Same as faulted limits for linear supports.

For active Class 2 or 3 pumps, support adequacy will be proven by satisfying the criteria in Section 3.9N.3.2. The requirements consist of both stress analysis and an evaluation of pump/motor support misalignment.

Active valves are, in general, supported only by the pipe attached to the valve. Exterior supports on the valve are generally not used.

3.9N.4 CONTROL ROD DRIVE SYSTEM (CRDS)

3.9N.4.1 Descriptive Information of CRDS

Full Length Control Rod Drive Mechanism

Control rod drive mechanisms are located on the dome of the reactor vessel. They are coupled to rod control clusters which have absorber material over the entire length of the control rods and derive their name from this feature. The full length control rod drive mechanism is shown in Figure 3.9N-5 and schematically in Figure 3.9N-6.

The primary function of the full length control rod drive mechanism is to insert or withdraw rod cluster control assemblies within the core to control average core temperature and to shutdown the reactor.

The full length control rod drive mechanism is a magnetically operated jack. A magnetic jack is an arrangement of three electromagnets which are energized in a controlled sequence by a power cycler to inset or withdraw rod cluster control assemblies in the reactor core in discrete steps. Rapid insertion of the rod cluster control assemblies occurs when electrical power is interrupted.

The control rod drive mechanism consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, latch assembly, and the drive rod assembly.

1. Unit 1

   For Unit 1, the pressure vessel is a one-piece housing which is connected to the reactor vessel had adapter by a full penetration weld. The lower portion of the pressure vessel contains the latch assembly. The upper portion of the pressure vessel provides space for

the drive rod during its upward movement as the control rods are withdrawn from the core.

Unit 2

For Unit 2, the pressure vessel includes a latch housing and a rod travel housing which are connected by a threaded, seal welded, maintenance joint which facilitates replacement of the latch assembly. The closure at the top of the rod travel housing is a threaded plug for pressure integrity with a canopy seal weld. Mechanical canopy seal clamp assemblies may be used to contain or prevent leaks in the canopy seal.

The latch housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the core.

2. The coil stack assembly includes the coil housings, and electrical conduit and connector, and three operating coils: 1) the stationary gripper coil, 2) the movable gripper coil, and 3) the lift coil.

   The coil stack assembly is a separate unit which is installed on the drive mechanism by sliding it over the outside of the latch housing/pressure vessel. It rests on the base of the latch housing/pressure vessel without mechanical attachment.

   Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

3. The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches: 1) the movable gripper latches and 2) the stationary gripper latches.

   The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in 5/8 inch steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next 5/8 inch step.

4. The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

   The drive rod has 5/8 inch grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and provides the means for coupling to the rod cluster control assembly.

   The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the rod cluster control assembly and permits remote disconnection of the drive rod.

The control rod drive mechanism is a trip design. Tripping can occur during any part of the power cycler sequencing if electrical power to the coils is interrupted.

Unit 1

The Unit 1 control rod drive mechanism is welded with a full penetration weld to an adaptor on top of the reactor vessel and is coupled to the rod cluster control assembly directly below.

Unit 2

The Unit 2 control rod drive mechanism is threaded and seal welded on an adaptor on top of the reactor vessel and is coupled to the rod cluster control assembly directly below.

The mechanism is capable of raising or lowering a 360 pound load, (which includes the drive rod weight) at a rate of 45 inches/minute. Withdrawal of the rod cluster control assembly is accomplished by magnetic forces while insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2500 psia. The three operating coils are designed to operate at 392°F with forced air cooling required to maintain that temperature.

The full length control rod drive mechanism shown schematically in Figure 3.9N-6 withdraws and inserts a rod cluster control assembly as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing/pressure vessel. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

During plant operation the stationary gripper coil of the drive mechanism holds the rod cluster control assembly in a static position until a stepping sequence is initiated at which time the movable gripper coil and lift coil is energized sequentially.

Rod Cluster Control Assembly Withdrawal

The rod cluster control assembly is withdrawn by repetition of the following sequence of events (refer to Figure 3.9N-6).

1. Movable gripper coil (B) - ON

   The latch locking plunger raises and swings the movable gripper latches into the drive rod assembly groove. A 0.047 inch axial clearance exists between the latch teeth and the drive rod.

2. Stationary gripper coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 0.047 inch until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

3. Lift coil (C) - ON

The 5⁄8 inch gap between the movable gripper pole and the lift pole closes and the drive rod assembly raises one step length (5/8 inch).

4. Stationary gripper coil (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing and the stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 0.047 inch. The 0.047 inch vertical drive rod assembly movement transfer the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

5. Movable gripper coil (B) - OFF

   The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

6. Lift coil (C) - OFF

   The gap between the movable gripper pole and lift pole opens. The movable gripper latches drop 5⁄8 inch to a position adjacent to a drive rod assembly groove.

7. Repeat step 1

   The sequence described above (items 1 through 6) is termed as one step or one cycle. The rod cluster control assembly moves 5⁄8 inch for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute and the drive rod assembly (which has a 5⁄8 inch groove pitch) is raised 72 grooves per minute. The rod cluster control assembly is thus withdrawn at a rate up to 45 inches per minute.

Rod Cluster Control Assembly Insertion

The sequence for rod cluster control assembly insertion is similar to that for control rod withdrawal, except the timing of lift coil (C) ON and OFF is changed to permit lowering the control assembly.

1. Lift coil (C) - ON

   The 5/8 inch gap between the movable gripper and lift pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

2. Movable gripper coil (B) - ON

   The latch locking plunger raises and swings the movable gripper latches into a drive rod assembly groove. A 0.047 inch axial clearance exists between the latch teeth and the drive rod assembly.

3. Stationary gripper coil (A) - OFF

The force of gravity, acting upon the drive rod assembly and attached rod cluster control assembly, causes the stationary gripper latches and plunger to move downward 0.047 inch until the load of the drive rod assembly and attached rod cluster control assembly is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

4. Lift coil (C) - OFF

   The force of gravity and spring force separates the movable gripper pole from pole from the lift pole and the drive rod assembly and attached rod cluster control drop down 5⁄8 inch.

5. Stationary gripper (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 0.047 inch. The 0.047 inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

6. Movable gripper coil (B) - OFF

   The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

7. Repeat step 1

   The sequence is repeated, as for rod cluster control assembly withdrawal, up to 72 times per minute which gives a insertion rate of 45 inches per minute.

Holding and Tripping of the Control Rods

During most of the plant operating time, the control rod drive mechanisms hold the rod cluster control assemblies withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil (A), is energized on each mechanism. The drive rod assembly and attached rod cluster control assemblies hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the rod cluster control assembly plus the stationary gripper return spring is sufficient to move latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the weight stationary gripper return spring and weight acting upon the latches. After the rod cluster control assembly is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9N.4.2 Applicable CRDS Design Specifications

For those components in the CRDS comprising portions of the reactor coolant pressure boundary, conformance with the General Design Criteria and 10CFR50, Section 50.55a is discussed in Sections 3.1 and 5.2. Conformance with pertaining regulatory guides is discussed in Sections 4.5 and 5.2.3 and Appendix 1A(N).

Design Bases

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

Design Stresses

The CRDS is designed to withstand stresses originating from various operating conditions as summarized in Table 3.9N-1.

Allowable stresses: For normal operating conditions Section III of the ASME Code is used. All pressure boundary components are analyzed as Class 1 components.

Dynamic analysis: The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the CRDS.

Control Rod Drive Mechanisms

The control rod drive mechanisms (CRDM's) pressure housings are Class 1 components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

A dynamic seismic analysis is required on the CRDM's when a seismic disturbance has been postulated to confirm the ability of the pressure housing to meet ASME Code, Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

Full Length Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the full length CRDM's are:

1. 5/8 inch step.

2. 144 inch travel.

3. 360 pound maximum load.

4. Step in or out at 45 inches/minute (72 steps/minute).

5. Electrical power interruption shall initiate release of drive rod assembly.

6. Trip delay time of less than 150 milliseconds - Free fall of drive rod assembly shall begin less than 150 milliseconds after power interruption no matter what holding or stepping action is being executed with any load and coolant temperature of 100°F to 550°F.

7. 40 year design life with normal refurbishment.

3.9N.4.3 Design Loads, Stress Limits, and Allowable Deformations

3.9N.4.3.1 Pressure Vessel

The pressure retaining components are analyzed for loads corresponding to normal, upset, emergency and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control rod trip (equivalent static load).

2. Differential pressure.

3. Spring preloads.

4. Coolant flow forces (static).

5. Temperature gradients.

6. Differences in thermal expansion

   a. Due to temperature differences.

   b. Due to expansion of different materials.

7. Interference between components.

8. Vibration (mechanically or hydraulically induced).

9. All operational transients listed in Table 3.9N-1.

10. Pump overspeed.

11. Seismic loads (Operational Basis Earthquake and Safe Shutdown Earthquake).

12. Blowdown forces (due to cold and hot leg break).

The main objective of the analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak

stresses will not reach unacceptable values, but also to limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors.

3.9N.4.3.2 Drive Rod Assembly

All postulated failures of the full length drive rod assemblies either by fracture or uncoupling lead to are reduction in reactivity. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with and is guided by the rod cluster control assembly. This always results in reactivity decrease for full length control rods.

3.9N.4.3.3 Latch Assembly and Coil Stack Assembly

Results of Dimensional and Tolerance Analysis

With respect to the CRDM system as a whole, critical clearances are present in the following areas:

1. Latch assembly (diametral clearances).

2. Latch arm-drive rod clearances.

3. Coil stack assembly-thermal clearances.

4. Coil fit in coil housing.

The following write-up defines clearances that are designed to provide reliable operation in the CRDM in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

Latch Assembly - Thermal Clearances

The magnetic jack has several clearances where parts made of Type 410 stainless steel fit over parts made from Type F304LN (Unit 1) or Type 304 (Unit 2) stainless steel. Differential thermal expansion is therefore important. Minimum clearances of these parts at 68°F is 0.011 inches. At the maximum design temperature of 650°F minimum clearances is 0.0045 inches and at the maximum expected operating temperatures of 550°F is 0.0057 inches.

Latch Arm - Drive Rod Clearances

The CRDM incorporates a load transfer action. The movable or stationary gripper latch are not under load during engagement, as previously explained, due to load transfer action.

Figure 3.9N-7 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9N-8 shows clearance variations over the design temperature range.

Coil Stack Assembly - Thermal Clearances

The assembly clearances of the coil stack assembly over the latch housing/pressure vessel was selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F the inside diameter of the coil stack is 7.308/7.298 inches. The outside diameter of the latch housing/pressure vessel is 7.260/7.270 inches.

Thermal expansion of the mechanism due to operating temperature of the CRDM result in minimum inside diameter of the coil stack being 7.310 inches at 222°F and the maximum latch housing/pressure vessel diameter being 7.302 inches at 532°F.

Under the extreme tolerance conditions listed above it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

Four coil stack assemblies were removed for four hot CRDMs mounted on 11.035 inch centers on a 550°F test loop, allowed to cool, and then placed without incident as a test to prove the preceding.

Coil Fit in Core Housing

CRDM and coil housing clearances are selected so that coil heat up results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot CRDM.

3.9N.4.3.4 Evaluation of Control Rod Drive Mechanisms and Supports

The control rod drive mechanisms (CRDMs) and CRDM support structure are evaluated for the loading combinations outlined in Table 3.9(N)-2.

Unit 1: A detailed finite element model of the CRDMs and CRDM supports is constructed using the WECAN (seismic analysis) or STRUDL (LOCA) analysis) computer programs. The models consist of beam, pipe, and spring elements. For the seismic analysis, nonlinearities in the structure are represented. The time history is derived from an envelope of the seismic response spectra at the reactor vessel head center of gravity and the tie rod containment building elevation. Maximum forces and moments in the CRDMs and support structure are then determined. For the LOCA analysis, the structural model is linearized and the reactor vessel head center of gravity LOCA response spectra is applied to determine the maximum forces and moments in the structure.

Unit 2: A detailed finite element model of the CRDMs and CRDM supports is constructed using the WECAN computer program with beam, pipe, and spring elements. For the LOCA analysis, nonlinearities in the structure are represented. These include RPI plate impact, tie rods, and lifting leg clevis/RPV head interface. The time history motion of the reactor vessel head, obtained from the RPV analysis described in 3.9(N).1.4.6, is input to the dynamic model. Maximum forces and moments in the CRDMs and support structure are then determined. For the seismic analysis, the structural model is linearized and the floor response spectra corresponding to the CRDM tie rod elevation is applied to determine the maximum forces and moments in the structure.

The bending moments calculated for the CRDMs for the various loading conditions are compared with maximum allowable moments determined from a detailed finite element stress evaluation of the CRDMs. Adequacy of the CRDM support structure is verified by comparing the calculated stresses to the criteria given in ASME III, Subsection NF.

The highest loads occur at the head adaptor, the location where the mechanisms penetrate the vessel head. The bending moments at this location are presented in Table 3.9N-20 for the longest and shortest CRDM.

3.9N.4.4 CRDMS Performance Assurance Program

Evaluation of Material's Adequacy

The ability of the pressure housing components to perform throughout the design lifetime as defined in the equipment specification is confirmed by the stress analysis report required by the ASME Code, Section III.

Internal components subject to wear will withstand a minimum of 3,000,000 steps without refurbishment as confirmed by life tests (Reference [11]). Latch assembly inspection is recommended after 2.5×10^6 steps have been accumulated on a single CRDM.

To confirm the mechanical adequacy of the fuel assembly, the CRDM, and full length rod cluster control assembly, functional test programs have been conducted on a full scale 12 foot control rod. The 12 foot prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the full length CRDMs meet the design requirements of 2.4 seconds from start of rod cluster control assembly motion to dashpot entry. This trip time requirements will be confirmed for each CRDM prior to initial reactor operation and at periodic intervals after initial reactor operation as required by the proposed Technical Specifications.

There are no significant differences between the prototype CRDMs and the production units. Design materials, tolerances and fabrication techniques (Section 4.2.3.3.2) are the same.

These tests have been reported in Reference [11].

In addition, dynamic testing programs have been conducted by Westinghouse and Westinghouse Licensees to demonstrate that control rod scram time is not adversely affected by postulated seismic events. Acceptable scram performance is assured by also including the effects of the allowable displacements of the driveline components in the evaluation of the test results.

It is expected that all CRDMs will meet specified operating requirements for the duration of plant life with normal refurbishment. However, a Technical Specification pertaining to an inoperable

rod cluster control assembly has been set. Latch assembly inspection is recommended after 2.5×10^6 steps have been accumulated on a single CRDM.

If a rod cluster control assembly cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one rod cluster control assembly can be tolerated. More than one inoperable rod cluster control assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable rod cluster control assemblies has been limited to one as discussed in the Technical Specifications.

In order to demonstrate proper operation of the CRDM and to ensure acceptable core power distributions during rod cluster control assembly partial-movement checks are performed on the rod cluster control assemblies (refer to the Technical Specifications). In addition, periodic drop tests of the full length rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical rod cluster control assembly ejection. During these tests the acceptable drop time of each assembly is not greater than 2.4 seconds, at full flow and operating temperature, from the beginning of motion to dashpot entry.

Actual experience in operating Westinghouse plants indicates excellent performances of CRDMs.

All units are production tested prior to shipment to confirm ability of the CRDM to meet design specification-operation requirements.

Each production full length CRDM undergoes a production tests as listed below:

Test	Acceptance Criteria
Cold (ambient) hydrostatic	ASME Section III
Confirm step length and load transfer (stationary gripper movable gripper or movable gripper to stationary gripper)	Step Length 5/8 ± 0.015 inches axial movement to Load Transfer 0.047 inches nominal axial movement
Cold (ambient) performance Test at design load – 45 5 full travel excursions	Operating Speed 45 inches/minute Trip Delay Free fall of drive rod to begin within 150 milliseconds

3.9N.5 REACTOR VESSEL INTERNALS

3.9N.5.1 Design Arrangements

The reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and guides for the incore instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the described inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

Lower Core Support Structures

Bottom Mounted Instrumentation wear reduction sleeves are installed in the reactor vessel lower internals instrumentation columns of Unit 1. The wear reduction sleeves are completely contained in the instrumentation columns with the flange resting on the top of the lower core plate. The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.9N-9. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support which is welded to the core barrel. All the major material for this structure is Type 304 stainless steel. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel in order to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of Type 304 stainless steel and are approximately 48 inches wide by 148 inches long by 2.8 inches thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides in which material surveillance samples can be inserted and irradiated during reactor operation are attached to the pads. The samples are held in the guides by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given in Reference [12].

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads and earthquake acceleration are carried by the lower core plate partially into the

lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall, and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

The main radial support system of the lower end of the core barrel is accomplished by “key” and “keyway” joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel clevis block is welded to the vessel inner diameter. Another Inconel insert block is bolted to each of these blocks and has a “keyway” geometry. Opposite each of these is a “key” which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal structures are within the ASME Code, Section III limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy absorbing devices limit the displacement after contacting the vessel bottom head. The load is then transferred through the energy absorbing devices of the internals to the vessel.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Assuming a downward vertical displacement the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

Upper Core Support Assembly

The upper core support assembly, shown in Figures 3.9N-10 and 3.9N-11 consists of the top support plate, assembly, and the upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the top support plate assembly and the upper core plate and are fastened at top and bottom to these plates. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies, sheath and guide the control rod drive shafts and control rods. They are fastened to the top support plate and are restrained by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support plate and guide tube.

The upper core support assembly is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which is turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90 degrees from each other. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main intenals, the slots in the plant engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core

support structure, the upper core support assembly, the fuel assemblies and control rods are thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the top support plate assembly and then the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom (Figure 7.7-9 shows the basic flux-mapping system).

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the training ends of the thimbles (at the seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence. These are the only conditions which affect the incore instrumentation support structure.

3.9N.5.2 Design Loading Conditions

The design loading conditions that provide the basis for the design of the reactor internals are:

1. Fuel assembly weight.
2. Fuel assembly spring forces
3. Internals weight.
4. Control rod trip (equivalent static load).
5. Differential pressure.
6. Spring preloads.
7. Coolant flow forces (static).
8. Temperature gradients.
9. Differences in thermal expansion
    a. Due to temperature differences.
    b. Due to expansion of different materials.
10. Interface between components.
11. Vibration (mechanically or hydraulically induced).
12. One or more loops out of service
13. All operational transients listed in Table 3.9N-1.
14. Pump overspeed.
15. Seismic loads (Operation Basis Earthquake and Safe Shutdown Earthquake).
16. Blowdown forces (due to cold and hot leg break).

The main objective of the design analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also to limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Both low and high cycle fatigue stresses are considered when the allowable amplitude of oscillation is established. Dynamic analysis on the reactor internals are provided in Section 3.9N.2.

As part of the evaluation of design loading conditions, extensive testing and inspections are performed from the initial selection of raw materials up to and including component installation

and plant operation. Among these tests and inspections are those performed during component fabrication, plant construction, startup and checkout, and during plant operation.

3.9N.5.3 Design Loading Categories

The combination of design loadings fit into the normal, upset, emergency or faulted conditions as defined in the ASME Code, Section III.

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions as summarized in Table 3.9N-1.

The scope of the stress analysis problem is very large requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For the loss of coolant accident plus the Safe Shutdown Earthquake condition, the deflection criteria of critical internal structures are the limiting values given in Table 3.9N-11. The corresponding no loss of function limits are included in Table 3.9N-11 for comparison purposes with the allowed criteria.

The criteria for the core drop accident is based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately one half inch. Additional displacement of approximately 3/4 inch would occur due to strain of the energy absorbing devices of the secondary core support; thus the total drop distance is about 1-1/4 inch which is insufficient to permit the trips of the rod cluster control assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are 4 supports in each reactor. This device limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless steel, loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15 percent, after which a positive stop is provided to ensure support.

3.9N.5.4 Design Bases

The design bases for the mechanical design of the reactor vessel internals components are as follows:

1. The reactor internals in conjunction with the fuel assemblies shall direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

2. In addition to neutron shielding provided by the reactor coolant, a separate neutron pad assembly is provided to limit the exposure of the pressure vessel in order to maintain the required ductility of the material for all modes of operation.

3. Provisions shall be made for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance programs.

4. The core internals are designed to withstand mechanical loads arising from Operating Basis Earthquake, Safe Shutdown Earthquake and pipe ruptures and meet the requirement of Item 5 below.

5. The reactor shall have mechanical provisions which are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

6. Following the design basis accident, the plant shall be capable of being shutdown and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in Table 3.9N-11. To ensure no column loading or rod cluster control guide tubes, the upper core plate deflection is limited to not exceed the value shown in Table 3.9N-11.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9N.2.

The basis for the design stress and deflection criteria is identified below:

Allowable Stresses

For normal operating conditions Section III Subsection NG of the ASME Code is used as a basis for evaluating acceptability of calculated stresses. Both static and alternating stress intensities are considered.

It should be noted that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the

Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the design basis accident used for the core support structures are based on the 1974 Edition of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions.

In summary, the design and construction of the CPNPP core support structures conforms to the requirements of the Subsection NG of Section III of the ASME code, except that a) the internals are not “stamped” and b) a specific stress report is note required.

3.9N.6 INSERVICE TESTING OF PUMPS AND VALVES

Refer to Section 3.9B.6.

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TABLE 3.9N-1
SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

(Sheet 1 of 3)

Normal Conditions		Occurrences
1.	Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200 (each)
2.	Unit loading and unloading at 5% of full power/min	13,200 (each)
3.	Step load increase and decrease of 10% of full power	2,000 (each)
4.	Large step load decrease with steam dump	200
5.	Steady state fluctuations	
	a. Initial fluctuations	1.5×10^5
	b. Random fluctuations	3.0×10^6
6.	Feedwater cycling at hot shutdown	2000
7.	Loop out of service	
	a. Normal loop shutdown	80
	b. Normal loop startup	70
8.	Unit loading and unloading between 0 to 15% of full power	500 (each)
9.	Boron concentration equalization	26,400
10.	Refueling	80

Upset Conditions		Occurrences
1.	Loss of load, without immediate reactor trip	80
2.	Loss of power (blackout with natural circulation in the Reactor Coolant System)	40
3.	Partial loss of flow (loss of one pump)	80

TABLE 3.9N-1
SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

(Sheet 2 of 3)

4.	Reactor trip from full power	
	a. Without cooldown	230
	b. With cooldown, without safety injection	160
	c. With cooldown and safety injection	10
5.	Inadvertent reactor coolant depressurization	20
6.	Inadvertent startup of an inactive loop	10
7.	Control rod drop	80
8.	Inadvertent Emergency Core Cooling System actuation	60
9.	Operating Basis Earthquake (20 earthquakes of 10 cycles each)	200 cycles
10.	Excessive Feedwater Flow	30

Emergency Conditions[(a)]		Occurrences
1.	Small loss of coolant accident	5
2.	Small steam break	5
3.	Complete loss of flow	5

Faulted Conditions[(a)]		Occurrences
1.	Main reactor coolant pipe break (dynamic effects, resulting from large loss of coolant accident at the RHR, surge and accumulator nozzles, Section 3.6)	1
2.	Large steam break	1
3.	Feedwater line break	1
4.	Reactor coolant pump locked rotor	1
5.	Control rod ejection	1

TABLE 3.9N-1
SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

(Sheet 3 of 3)

6.	Steam generator tube rupture	(included under upset conditions, reactor trip from full power with safety injection)
7.	Safe Shutdown Earthquake	1

Test Conditions		Occurrences
1.	Turbine roll test	20
2.	Primary side hydrostatic test	10
3.	Secondary side hydrostatic test	10
4.	Primary side leak test	200
5.	Secondary side leak test	80
6.	Tube leakage test	800

a) In accordance with the ASME Nuclear Power Plant Components Code, emergency and faulted conditions are not included in fatigue evaluation.

TABLE 3.9N-1A
COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle -T_{avg} from ≤ 200°F to ≥ 500°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor Trip	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray actuation cycles.	Spray water temperature differential > 320°F, but ≤ 625°F
	200 leak tests.	Pressurized to ≥ 2485 psig.
	10 hydrostatic pressure tests.	Pressurized to ≥ 3107 psig.
Secondary Coolant System	1 steam line break.	Break in a > 6-inch steam line.
	10 hydrostatic pressure tests.	Pressurized to ≥ 1481 psig.

TABLE 3.9N-2
LOADING COMBINATIONS FOR ASME CLASS 1 COMPONENTS AND COMPONENT SUPPORTS (EXCLUDING PIPE SUPPORTS)

Condition Classification	Loading Combination
Design	Design Pressure, Design Temperature, Deadweight, Operating Basis Earthquake[a]
Normal	Normal Condition Transient, Deadweight
Upset	Upset Condition Transients, Deadweight, Operating Basis Earthquake
Emergency	Emergency Condition Transients, Deadweight
Faulted	Faulted Condition Transients, Deadweight, Safe Shutdown Earthquake or (Safe Shutdown Earthquake and Pipe Rupture Loads)

a) The Operating Basis Earthquake is not considered a design condition for NSSS Class 1 valves and piping. The primary stresses are calculated for the Operating Basis Earthquake loads and compared with the upset condition allowable stresses.

TABLE 3.9N-3
ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS[a]

(Sheet 1 of 2)

Operating Condition Classification	Vessels/Tanks	Piping	Pumps	Valves	Component Supports
Normal	ASME Section III	ASME Section III	ASME Section III	ASME Section III	ASME Section III Subsection NF
Upset	ASME Section III	ASME Section III	ASME Section III	ASME Section III	ASME Section III Subsection NF
Emergency	ASME Section III	ASME Section III	ASME Section III	ASME Section III	ASME Section III Subsection NF
Faulted	ASME Section III see Section 3.9N.1.4	ASME Section III see Section 3.9N.1.4	ASME Section III See Section 3.9N.1.4 (No active class 1 pump used)	See Note 1	ASME Section III Subsection NF See Section 3.9N.1.4

P_e, P_m, P_b, Q_t, C_p, S_n & S_m as defined by Section II of the ASME Code

TABLE 3.9N-3
ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS[a]

(Sheet 2 of 2)

NOTE 1 TO TABLE 3.9N-3

CLASS 1 VALVE FAULTED CONDITION CRITERIA

	ACTIVE		INACTIVE
a)	Calculate Pm from para. NB3545.1 with Internal Pressure Ps = 1.25Ps Pm ≤ 1.5Sm	a)	Calculate Pm from para. NB3545.1 with Internal Pressure Ps = 1.50Ps Pm ≤ 2.4 Sm or 0.7Su
b)	Calculate Sn from para. NB3545.2 with Cp = 1.5 Ps = 1.25Ps Qt2 = 0 Ped = 1.3X value of Ped from equations of 3545.2(b) (1) Sn ≤ 3Sm	b)	Calculate Sn from para. NB3545.2 with Cp = 1.5 Ps = 1.50Ps Qt2 = 0 Ped = 1.3X value of Ped from equations of NB3545.2(b) (1) Sn ≤ 3Sm

a) A test of the components may be performed in lieu of analysis.

TABLE 3.9N-4
DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND COMPONENTS AND COMPONENT SUPPORTS (EXCLUDING PIPING AND PIPE SUPPORTS)[a]

Condition Classification	Loading Combination
Design and Normal	Design pressure design temperature,[b] deadweight, nozzle loads[c]
Upset	Upset condition pressure, upset condition metal temperature,[b] deadweight, OBE, nozzle loads[c]
Emergency	Emergency condition pressure, emergency condition metal temperature,[b] deadweight, nozzle loads[c]
Faulted	Faulted condition pressure, faulted condition metal temperature,[b] deadweight, SSE, nozzle loads[c]

a) For loading combinations and stress limits for ASME Class 2 and 3 piping, refer to Table 3.9B-1B and Table 3.9B-1C for Class 2 and 3 supports.

b) Temperature is used to determine allowable stress only.

c) Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration.

TABLE 3.9N-5
STRESS CRITERIA FOR SAFETY RELATED ASME CLASS 2 AND CLASS 3 TANKS

Condition	Stress Limits
Design and Normal	$\sigma_m \leq 1.0\ S$ $(\sigma_m \text{ or } \sigma_L) +$ $\sigma_b \leq 1.5\ S$
Upset	$\sigma_m \leq 1.1\ S$ $(\sigma_m \text{ or } \sigma_L) +$ $\sigma_b \leq 1.65\ S$
Emergency	$\sigma_m \leq 1.5\ S$ $(\sigma_m \text{ or } \sigma_L) +$ $\sigma_b \leq 1.80\ S$
Faulted	$\sigma_m \leq 2.0\ S$ $(\sigma_m \text{ or } \sigma_L) +$ $\sigma_b \leq 2.4\ S$

TABLE 3.9N-6
STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS

Condition	Stress Limits	P_{max}[(a)]
Design and Normal	The pump shall conform to the requirements of ASME Section III, NC-3400 (or ND-3400)	
Upset	$\sigma_m \leq 1.1\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65\ S$	1.1
Emergency	$\sigma_m \leq 1.5\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80\ S$	1.2
Faulted	$\sigma_m \leq 2.0\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4\ S$	1.5

a) The maximum pressure shall not exceed the tabulated factors listed under P_{max} times the design pressure.

TABLE 3.9N-7
DESIGN CRITERIA FOR ACTIVE PUMPS

Condition	Design Criteria
Normal	ASME Section III Subsection NC-3400 and ND-3400
Upset	$\sigma_m \leq 1.0\ S$ $\sigma_m + \sigma_b \leq 1.5\ S$
Emergency	$\sigma_m \leq 1.2\ S$ $\sigma_m + \sigma_b \leq 1.65\ S$
Faulted	$\sigma_m \leq 1.2\ S$ $\sigma_m + \sigma_b \leq 1.8\ S$

TABLE 3.9N-8
STRESS CRITERIA FOR SAFETY RELATED ASME CODE CLASS 2 AND CLASS 3 VALVES

Condition	Stress Limits (Notes 1-5)	P_{max} (Note 6)
Design and Normal	Valve bodies shall conform to the requirements of ASME Section III, NC-3500 (or ND-3500)	
Upset	$\sigma_m \leq 1.1\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65\ S$	1.1
Emergency	$\sigma_m \leq 1.5\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80\ S$	1.2
Faulted	$\sigma_m \leq 2.0\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4\ S$	1.5

Notes:

1. Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation: 1) section modulus and area of a plane, normal to the flow, through the region of valve body crotch is at least 10% greater than the piping connected (or joined) to the valve body inlet and outlet nozzles; and, 2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connecting piping material. If the valve body material allowable stress is less than of the connected piping, the valve section modulus and area as calculated in (1) above shall be multiplied by the ratio (S_{pipe}/S_{valve}). If unable to comply with this requirement, the design by analysis procedure of NB-3545.2 is an acceptable alternate method.

2. Casting quality factor of 1.0 shall be used.

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

4. Design requirements listed in this Table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

5. These rules do not apply to Class 2 and 3 safety relief valves. Safety relief valves will be designed in accordance with ASME Section III requirements.

6. The maximum pressure resulting from upset, emergency or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in Table 3.9N-4 are considered to be satisfied.

TABLE 3.9N-9
ACTIVE PUMPS

Pump	Item No.	System	ANS Safety Class	Normal Mode	Post LOCA Mode	Basis
Centrifugal charging pump no. 1	APCH	CVCS	2a	ON/OFF	ON	High head safety injection
Centrifugal Charging pump no. 2	APCH	CVCS	2a	ON/OFF	ON	High head safety injection
Boric acid transfer pumps	APBA	CVCS	2b	ON/OFF	ON/OFF	Required for emergency boration safe shutdown
Residual heat removal pump no. 1 and 2	APRH	RHRS	2a	OFF	ON	Required for safety injection and plant cooldown
Safety injection pump no. 1 and 2	APSI	SIS	2a	OFF	ON	Required for safety injection

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 1 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
HV-3607	RCS	Solenoid	1"	Globe/2	Closed	RCS high point vents
HV-3608	RCS	Solenoid	1"	Globe/2	Closed	RCS high point vents
HV-3609	RCS	Solenoid	1"	Globe/2	Closed	Pressurizer high point vents
HV-3610	RCS	Solenoid	1"	Globe/2	Closed	Pressurizer high point vents
PCV-455A	RCS	Nitrogen	3"	Globe/1	Closed	Primary Pressure Control after SGTR only. (Open/Close)
PCV-456	RCS	Nitrogen	3"	Globe/1	Closed	Primary Pressure Control after SGTR only. (Open/Close)
8000A/B	RCS	Motor	3"	Gate/1	Open/Closed	Unblock Pressurizer PORV after SGTR (Open); Block Stuck Open Pressurizer PORV after SGTR (Close)
8010A/B/C	RCS	Self-actuated	6"	Safety/1	Closed	Pressurizer code safety valves
8026	RCS	Air	1"	Diaphragm/2	Open if sample is being taken or PRT is being vented	Containment isolation
8027	RCS	Air	1"	Diaphragm/2	Open if sample is being taken or PRT is being vented	Containment isolation
8046	RCS	Self-actuated	3"	Check/2	N/A	Containment isolation

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 2 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
8047	RCS	Air	3"	Diaphragm/2	Open if PRT spray is actuated	Containment isolation
RC-036	RCS	Self-actuated	3/4"	Relief/2	Closed	Pressure relief, Containment isolation
XCS-0037 (Note 1)	CVCS	Self-actuated	3/4"	Check/3	Closed	Pump Protection
XCS-0039 (Note 1)	CVCS	Self-actuated	3/4"	Check/3	Closed	Pump Protection
XCS-0041 (Note 1)	CVCS	Self-actuated	3/4"	Check/3	Closed	Pump Protection
XCS-0044 (Note 1)	CVCS	Self-actuated	3/4"	Check/3	Closed	Pump Protection
8100	CVCS	Motor	2"	Globe/2	Open	Containment isolation
8104	CVCS	Motor	2"	Globe/2	Closed	Required for safe shutdown redundancy
8105	CVCS	Motor	3"	Gate/2	Open	ECCS operation and containment isolation
8106	CVCS	Motor	3"	Gate/2	Open	ECCS operation
8110	CVCS	Motor	2"	Globe/2	Open	ECCS operation
8111	CVCS	Motor	2"	Globe/2	Open	ECCS operation
8112	CVCS	Motor	2"	Globe/2	Open	Containment isolation
8145	CVCS	Air	2"	Globe/1	Closed	RCS pressure boundary
8152	CVCS	Air	3"	Globe/2	Open	Containment isolation
8153	CVCS	Air	1"	Globe/1	Closed	RCS pressure boundary
8154	CVCS	Air	1"	Globe/1	Closed	RCS pressure boundary

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 3 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
1-8121	CVCS	Self-actuated	2”	Relief/2	Closed	Pressure relief during LOCA
2CS-8000	CVCS	Self-actuated	1”	Relief/2	Closed	Pressure relief during LOCA
8160	CVCS	Air	3"	Globe/2	Open	Containment isolation
CS-8180	CVCS	Self-actuated	3/4"	Check/2	N/A	Pressure relief, Containment isolation
8202A/B (Note 1)	CVCS	Solenoid	1"	Globe/2	N/A	ECCS operation
8210A/B (Note 1)	CVCS	Solenoid	1"	Globe/2	N/A	ECCS operation
HV-8220 (Note 1)	CVCS	Air	1"	Ball/2	Open	ECCS operation
HV-8221 (Note 1)	CVCS	Air	1"	Ball/2	Open	ECCS operation
CS-8350A/B/C/D	CVCS	Self-actuated	2"	Check/1	N/A	RCS pressure boundary
8351A/B/C/D	CVCS	Motor	2"	Globe/2	Open	Containment isolation
CS-8367A/B/C/D	CVCS	Self-actuated	2"	Check/1	N/A	RCS pressure boundary
CS-8368A/B/C/D	CVCS	Self-actuated	2"	Check/2	N/A	Containment isolation
CS-8377	CVCS	Self-actuated	2"	Check/1	N/A	RCS pressure boundary
8378A/B	CVCS	Self-actuated	3"	Check/1	N/A	RCS pressure boundary, Boration
8379A/B	CVCS	Self-actuated	3"	Check/1	N/A	RCS pressure boundary, Boration
8381	CVCS	Self-actuated	3"	Check/2	N/A	Containment isolation, Boration
CS-8442	CVCS	Self-actuated	2"	Check/2	N/A	Required for safe shutdown redundancy
CS-8473	CVCS	Self-actuated	2"	Check/3	N/A	Boric acid flowpath

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 4 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
CS-8480A/B (Note 1)	CVCS	Self-actuated	2"	Check/2	N/A	Minimum flow path, ECCS flowpath
8481A/B	CVCS	Self-actuated	4"	Check/2	N/A	ECCS operation, Boration
CS-8487	CVCS	Self-actuated	2"	Check/3	N/A	Boric acid flowpath
8497	CVCS	Self-actuated	3"	Check/2	N/A	ECCS flowpath boundary
8510A/B	CVCS	Self-actuated	1-1/2"	Relief/2	Closed	Pump protection x 2"
8511A/B	CVCS	Motor	2"	Globe/2	Closed	Pump protection
8512A/B	CVCS	Motor	2"	Globe/2	Open	ECCS operation
8546	CVCS	Self-actuated	8"	Check/2	N/A	ECCS operation, Boration
LCV-112B/C	CVCS	Motor	4"	Gate/2	Open	ECCS operation
LCV-112D/E	CVCS	Motor	8"	Gate/2	Closed	ECCS operation
LCV-459	CVCS	Air	3"	Globe/1	Open	RCS pressure boundary (Remote manual isolation)
LCV-460	CVCS	Air	3"	Globe/1	Open	RCS pressure boundary (Remote manual isolation)
8701A/B	RHRS	Motor	12"	Gate/1	Closed	RCS pressuure boundary, Containment isolation Open for normal plant cooldown
8702A/B	RHRS	Motor	12"	Gate/1	Closed	RCS pressure boundary, Open for normal plant cooldown
8716A/B	RHRS	Motor	10"	Gate/2	Open	ECCS operation

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 5 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
RH-0033	RHRS	Self	3/4" x 1"	Relief/2	Closed	Thermal Relief for Bonnet of 8716A
RH-0034	RHRS	Self	3/4" x 1"	Relief/2	Closed	Thermal Relief for Bonnet of 8716B
8730A/B	RHRS	Self-actuated	10"	Check/2	N/A	ECCS operation and normal cooldown
FCV-610	RHRS	Motor	3"	Globe/2	Open/Closed	RHR pump miniflow provides pump protection
FCV-611	RHRS	Motor	3"	Globe/2	Open/Closed	Control RHRS miniflow, provides pump protection
SI-166 (Note 1)	SIS	Self-actuated	3/4	Check/3	Closed	Nitrogen Accumulator isolation
SI-167 (Note 1)	SIS	Self-actuated	3/4	Check/3	Closed	Nitrogen Accumulator isolation
SI-168 (Note 1)	SIS	Self-actuated	3/4	Check/3	Closed	Nitrogen Accumulator isolation
SI-169 (Note 1)	SIS	Self-actuated	3/4	Check/3	Closed	Nitrogen Accumulator isolation
8800A/B	SIS	Air	3"	Globe/2	Closed	Closes on "S" signal if Open
8801A/B	SIS	Motor	4"	Gate/2	Closed	ECCS flowpath, Containment isolation
8802A/B	SIS	Motor	4"	Gate/2	Closed	ECCS flowpath cold leg to hot leg switchover sequence, Containment isolation
8804A/B	SIS	Motor	8"	Gate/2	Closed	ECCS flowpath, recirculation
8806	SIS	Motor	8"	Gate/2	Open	ECCS flowpath, recirculation boundary
8807A/B	SIS	Motor	6"	Gate/2	Closed	ECCS recirculation, operation, safety injection charging pump cross connection

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 6 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
8809A/B	SIS	Motor	10"	Gate/2	Open	ECCS and plant cooldown flowpath, Containment isolation
8811A/B	SIS	Motor	14"	Gate/2	Closed	ECCS flowpath from Containment sump, Containment isolation
SI-0182	SIS	Self-actuated	3/4"	Relief/2	Closed	Thermal relief, Containment isolation
SI-0183	SIS	Self-actuated	3/4"	Relief/2	Closed	Thermal relief, Containment isolation
8812A/B	SIS	Motor	14"	Gate/2	Open	ECCS operation
8813	SIS	Motor	2"	Globe/2	Open	Safety injection pump miniflow and RWST isolation during recirculation
8814A/B	SIS	Motor	1-1/2"	Globe/2	Open	Safety injection pump miniflow and RWST isolation during recirculation
8815	SIS	Self-actuated	3"	Check/1	N/A	ECCS flowpath and RCS pressure boundary, Containment isolation
8818A/B/C/D	SIS	Self-actuated	6"	Check/1	N/A	ECCS flowpath and RCS pressure boundary, Containment isolation
SI-8819A/B/C/D	SIS	Self-actuated	2"	Check/1	N/A	ECCS flowpath and RCS pressure boundary, Containment isolation
8821A/B	SIS	Motor	4"	Gate/2	Open	Cold leg to hot leg recirculation switchover during ECCS sequence
8823	SIS	Air	3/4"	Globe/2	Closed	Containment isolation

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 7 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
8824	SIS	Air	3/4"	Globe/2	Closed	Containment isolation
8825	SIS	Air	3/4"	Globe/2	Closed	Containment isolation
8835	SIS	Motor	4"	Gate/2	Open	Cold leg to hot leg recirculation switchover during ECCS sequence, Containment isolation
8840	SIS	Motor	10"	Gate/2	Closed	ECCS flowpath, hot leg recirculation, Containment isolation
8841A/B	SIS	Self-actuated	6"	Check/1	N/A	ECCS flowpath, RCS pressure boundary, Containment isolation
8843	SIS	Air	3/4"	Globe/2	Closed	Containment isolation, Must Close if Open on Phase A signal
8871	SIS	Air	3/4"	Globe/2	Closed	Containment isolation, Closes on Phase A signal if Open
8880	SIS	Air	1"	Globe/2	Open	Containment isolation, Closes on Phase A signal if Open
8881	SIS	Air	3/4"	Globe/2	Closed	Containment isolation
8888	SIS	Air	3/4"	Globe/2	Closed	Containment isolation
8890A/B	SIS	Air	3/4"	Globe/2	Closed	Containment isolation
SI-8900A/B/C/D	SIS	Self-actuated	1-1/2"	Check/1	N/A	ECCS operation, RCS pressure boundary

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 8 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
SI-8905A/B/C/D	SIS	Self-actuated	2"	Check/1	N/A	ECCS operation, RCS pressure boundary, Containment isolation
SI-8919A/B	SIS	Self-actuated	1-1/2"	Check/2	Closed	Recirculation flow path
8922A/B	SIS	Self-actuated	4"	Check/2	N/A	ECCS flowpath
8923A/B	SIS	Motor	6"	Gate/2	Open	Provide train isolation in case of passive failure
8924	SIS	Motor	6"	Gate/2	Open	Provides redundant train isolation of safety injection charging pump cross connection (passive failure protection)
8926	SIS	Self-actuated	8"	Check/2	N/A	ECCS flowpath
8948A/B/C/D	SIS	Self-actuated	10"	Check/1	N/A	ECCS operation, RCS pressure boundary
8949A/B/C/D	SIS	Self-actuated	6"	Check/1	N/A	ECCS operation, RCS pressure boundary
8956A/B/C/D	SIS	Self-actuated	10"	Check/1	N/A	ECCS operation, RCS pressure boundary
8958A/B	SIS	Self-actuated	14"	Check/2	N/A	ECCS flowpath
8964	SIS	Air	3/4"	Globe/2	Closed	Containment isolation, Closes on Phase A signal if Open
SI-8968	SIS	Self Actuated	1"	Check/2	N/A	Containment isolation
8969A/B	SIS	Self-actuated	8"	Check/2	N/A	ECCS operation
1SI-8972 (2SI-8983) (Note 1)	SIS	Self-actuated	3/4"	Relief/2	Closed	Pressure relief, Containment isolation

TABLE 3.9N-10
UNIT 1 & 2 ACTIVE VALVES (NOTE 2, 3, 4)

(Sheet 9 of 9)

Valve Location Number	System	Actuated By	Size	Type/ANS Safety Class	Normal Position	Basis
7126	LWPS	Air	3/4"	Diaphragm/2	Open	Containment isolation
7136	LWPS	Air	3"	Diaphragm/2	Open	Containment isolation
7150	LWPS	Air	3/4"	Diaphragm/2	Open/Closed	Containment isolation
WP-7176 (Note 1)	LWPS	Self-actuated	3/4"	Relief/2	Closed	Pressure relief, Containment isolation
WP-7177 (Note 1)	LWPS	Self-actuated	3/4"	Relief/2	Closed	Pressure relief, Containment isolation
LCV-1003	LWPS	Air	3"	Globe/2	Open	Containment isolation

Notes:

1) Valves procured under BOP criteria. See Section 3.9B.

2) Unit 1 and Unit 2 valve tag numbers are the same except for the prefix, as follows:
Examples:
1-8027 is Unit 1
2-8027 is Unit 2
8027 indicates both Units
XCS-0037 indicates a common valve which serves both Units.

3) See Section 3.9B, Table 3.9B-10 for Balance-of-Plant Active Valve List.

4) See Section 6.2.4 for containment isolation details.

TABLE 3.9N-11
MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable Deflections (in)	No-Loss-of Function Deflections (in)
Upper Barrel		
Radial Inward	4.1	8.2
Radial outward	1.0	1.0
Upper package	0.10	0.15
Rod cluster guide tubes	1.00	1.75

TABLE 3.9N-12
MAXIMUM REACTOR VESSEL DISPLACEMENTS AT REACTOR VESSEL CENTERLINE[a]

	Maximum Horizontal Displacement (inches)	Maximum Vertical Displacement (inches)	Maximum Rotation (radians)
144 Square Inch	+0.116	+0.095	+0.00057
RPV Inlet	0.0	-0.087	-0.00066
144 Square Inch	0.0	+0.0014	0.0
RPV Outlet	-0.0081	-0.046	-0.0029
Double Ended	+0.074	+0.0029	+0.0031
Pump Outlet	-0.00064	-0.06	-0.00047
6" Safety Injection	+0.02365	-0.01194	+0.00011
Line Break	-0.02539	-0.02229	-0.00013
4" Spray Line	0.01986	-0.01468	0.0001
Break	-0.02255	-0.02182	-0.000084

a) Although the main loop piping breaks are not part of the design basis (excluded as dynamic effects, Section 3.1.1.4), they are conservatively included in the design basis for Reactor Vessel Displacements. The load and stress evaluation tables for the primary component supports are based on the results obtained from the effects of the 4 and 6 inch auxiliary nozzle breaks (numbers 12 and 13 provided on Table 3.6B-2 and Figure 3.6B.9).

TABLE 3.9N-13
MAXIMUM REACTOR VESSEL SUPPORT LOADS FOR POSTULATED PIPE RUPTURE CONDITIONS

LOCA Maximum Vertical Load Per Support Including Deadweight	LOCA Maximum Horizontal Load Per Support
(Kips)	(Kips)
1113	606

TABLE 3.9N-14A
STEAM GENERATOR LOWER SUPPORT MEMBER STRESSES FOR UNIT 1

Member Stresses
(Percent of Allowable/Loading Conditions)
Units (kips)

(Sheet 1 of 2)

	Normal		Upset		Faulted		
	Load[a]	Percent Stressed	Load[a]	Percent Stressed	Load[a]	Break	Percent Stressed
LS-1 (Bumper)	176	7.4	525	22.1	1205	envelope	44.7
LS-2 (Beam)	0	0	379	80.0[d]	691	envelope	86.0[d]
LS-3 (Bumper)	11	0.5	301	12.7	547	envelope	20.3
Column 1	0	0.0	+332	27.9	+542	envelope	26.3
	-255	17.2	-616	41.5	-1597	envelope	94.7
Column 2	0	0.0	+129	10.8	+102	envelope	5.0
	-332	22.4	-686	46.2	-1207	envelope	71.7

TABLE 3.9N-14A
STEAM GENERATOR LOWER SUPPORT MEMBER STRESSES FOR UNIT 1

Member Stresses
(Percent of Allowable/Loading Conditions)
Units (kips)

(Sheet 2 of 2)

	Normal		Upset		Faulted		
	Load[(a)]	Percent Stressed	Load[(a)]	Percent Stressed	Load[(a)]	Break	Percent Stressed
Column 3	0	0.0	+171	14.4	+145	envelope	7.0
	-338	22.8	-746	50.3	-1251	envelope	74.3
Column 4	0	0.0	+253	21.3	+393	envelope	19.1
	-267	18.0	-567	38.2	-1223	envelope	72.7

Notes:

a) (+) = tensions and (-) = compressions for column loadings.

b) The load table does not reflect the dynamic effects of breaks 1 through 11 (Table 3.6B-2 and Figure 3.6B-9).

c) The bumpers, beam, and columns are located as shown in Figure 304 of revision 5 of volume 2 of Westinghouse report no. WCAP-10197.

d) These percentages are conservative for the loads reported, as noted in Table 6-1 of revision 5 of volume 2 of Westinghouse report no. WCAP-10197. The 80.0 percent is actually based on a load of 623 kips, and the 86.0 percent is actually based on a load of 1048 kips.

TABLE 3.9N-14B
STEAM GENERATOR LOWER SUPPORT MEMBER STRESSES FOR UNIT 2

Member Stresses
(Percent of Allowable/Loading Conditions)
Units (kips)

(Sheet 1 of 2)

	Normal		Upset		Faulted		
	Load[(a)]	Percent Stressed	Load[(a)]	Percent Stressed	Load[(a)]	Break	Percent Stressed
LS-1 (Bumper)	+192	8.1	+423	17.8	+1311	SI	48.7
LS-2 (Beam)	0	--	+379	38.0	+625	SPRAY	61.3
	--	--	--	--	--		--
LS-3 (Bumper)	+16	0.7	+395	16.6	+925	FW	34.3
Column 1	+7	0.6	+194	16.3	+347	SPRAY	16.8
	-234	15.8	-421	28.4	-898	MS	53.4
Column 2	0	0.0	0	0.0	+171	MS	8.3
	-327	22.0	-512	34.5	-1016	MS	59.7

TABLE 3.9N-14B
STEAM GENERATOR LOWER SUPPORT MEMBER STRESSES FOR UNIT 2

Member Stresses
(Percent of Allowable/Loading Conditions)
Units (kips)

(Sheet 2 of 2)

	Normal		Upset		Faulted		
	Load[(a)]	Percent Stressed	Load[(a)]	Percent Stressed	Load[(a)]	Break	Percent Stressed
Column 3	0	0.0	0	0.0	+247	SPRAY	12.0
	-322	21.7	-501	33.8	-1004	MS	59.7
Column 4	0	0.0	+171	14.4	+310	MS	15.0
	-244	16.4	-417	28.1	-908	MS	54.0

Notes:

a) (+) = tensions and (-) = compressions for column loadings.

b) The load table does not reflect the dynamic effects of breaks 1 through 11 (Table 3.6B-2 and Figure 3.6B-9).

TABLE 3.9N-15A
STEAM GENERATOR UPPER SUPPORT MEMBER STRESSES FOR UNIT 1

Member Stresses
(Percent of Allowable/Loading Condition)
Units (kips)

	Normal		Upset		Faulted		
	Load	Percent Stressed	Load	Percent Stressed	Load	Break	Percent Stressed
US-1 (Snubbers)[b]							
US-2 (Bumper)	0	0.0	785	22.9	1445	envelope	37.2
US-3 (Bumper)	0	0.0	833	24.3	1726	envelope	44.4
US-4 (Beam)	0	0.0	727	21.2	1030	envelope	26.5

Notes:

a) The bupers and beam are located as shown in Figure 3-5 of revision 5 of volume 2 of Westinghouse Report no. WCAP-10197.

b) The snubbers have been eliminated for Unit 1.

c) The load table does not reflect the dynamic effects of breaks 1 through 11 (Table 3.6B-2 and Figure 3.6B-9).

TABLE 3.9N-15B
STEAM GENERATOR UPPER SUPPORT MEMBER STRESSES FOR UNIT 2

Member Stresses
(Percent of Allowable/Loading Condition)
Units (kips)

	Normal		Upset		Faulted		
	Load	Percent Stressed	Load	Percent Stressed	Load	Break	Percent Stressed
US-1 (Snubbers)[b]	0	0.0	+131 -623	14.6 69.2	+238 -1170	MS	11.9 58.5
US-2 (Bumper)	0	0.0	511	14.9	1043	MS	26.9
US-3 (Bumper)	0	0.0	618	18.0	1065	MS	27.4
US-4 (Beam)	0	(a)	622	(a)	1035	FW	(a)

Notes:

a) Scope of evaluation by others.

b) The snubbers were rated at 450 kips for normal and OBE seismic operation, and at 1000 kips for the faulted condition, per snubber in accordance with the snubber stress report.[1]

c) The load table does not reflect the dynamic effects of breaks 1 through 11 (Table 3.6B-2 and Figure 3.6B-9).

[1] Paul-Monroe Hydraulics Inc., Report A-690623, Revision O, entitled “Multiplant II 100 kip Snubber Stress Report”.

TABLE 3.9N-16A
REACTOR COOLANT PUMP SUPPORT MEMBER STRESSES FOR UNIT 1

Member Stresses
(Percent of Allowable/Loading Condition)
Units (kips)

	Normal		Upset		Faulted		
	Load[(a)]	Percent Stressed	Load[(a)]	Percent Stressed	Load[(a)]	Break	Percent Stressed
Tie Rod-A	0	0.0	255	22.7	406	envelope	25.8
Tie Rod-B	264	23.5	424	37.8	1356	envelope	86.3
Tie Rod-C	161	14.3	234	20.8	847	envelope	53.9
Column-1	+0	0.0	+69	5.8	+169	envelope	8.2
	-258	16.8	-407	26.4	-502	envelope	28.3
Column-2	+96	8.1	+331	27.8	+450	envelope	21.8
	-25	1.6	-260	16.9	-383	envelope	21.6
Column-3	+0	0.0	+224	18.8	+299	envelope	14.5
	-136	8.8	-370	24.0	-547	envelope	30.8

Notes:

a) (+) = tensions and (-) = compression for column loadings.
b) The load tables does not reflect the dynamic effects of breaks 1 through 11 (Table 3.6B-2 and Figure 3.6B-9)
c) The tie rods and columns are located as shown in figure 3-7 revision 5 of volume 2 of Westinghouse Report no. WCAP-101097.

TABLE 3.9N-16B
REACTOR COOLANT PUMP SUPPORT MEMBER STRESSES FOR UNIT 2

Member Stresses
(Percent of Allowable/Loading Condition)
Units (kips)

	Normal		Upset		Faulted		
	Load[(a)]	Percent Stressed	Load[(a)]	Percent Stressed	Load[(a)]	Break	Percent Stressed
Tie Rod-A	0	0.0	151	13.4	334	SPRAY	21.3
Tie Rod-B	270	24.0	434	38.6	1376	SPRAY	87.5
Tie Rod-C	170	15.1	334	29.7	879	SPRAY	55.9
Column-1	+0	0.0	+204	17.1	+437	SPRAY	21.2
	-275	17.9	-560	36.4	-788	SPRAY	45.2
Column-2	+115	9.7	+404	33.9	+617	SPRAY	29.9
	-19	1.2	-308	20.0	-567	FW	32.5
Column-3	+0	0.0	+282	23.7	+479	SPRAY	23.2
	-142	9.2	-427	27.7	-677	SPRAY	38.8

Notes:

a) (+) = tensions and (-) = compression for column loadings.

b) The load tables does not reflect the dynamic effects of breaks 1 through 11 (Table 3.6B-2 and Figure 3.6B-9)

TABLE 3.9N-17
PRESSURIZER UPPER LATERAL SUPPORT MEMBER STRESSES

(Sheet 1 of 2)

Member Stresses
(Percent of Allowable/Loading Condition)
Units (kips)

Member	Normal		Upset		Faulted[(a)]	
	Load	Percent Stressed	Load	Percent Stressed	Load	Percent Stressed
S-1 (Strut)	N = -7	2.4	N = -119	46.1	N = -226	70.5
	V = ±3	2.4	V = ±45	46.1	V = ±96	70.5
S-2 (Strut)	N = -4	1.8	N = -75	32.2	N = -159	43.7
	V = ±2	1.8	V = ±32	32.2	V = ±60	43.7
S-3 (Strut)	N = -7	1.3	N = -119	26.2	N = -226	39.3
	V = ±2	1.3	V = ±17	26.2	V = ±36	39.3
S-4 (Strut)	N = -4	1.2	N = - 75	23.2	N = -159	31.8
	V = ±1	1.2	V = ±13	23.2	V = ±24	31.8

TABLE 3.9N-17
PRESSURIZER UPPER LATERAL SUPPORT MEMBER STRESSES

(Sheet 2 of 2)

PRESSURIZER LOWER SUPPORT LOADS

LOADING CONDITIONS	AXIAL LOAD (kips)	SHEAR LOAD (kips)	OVERTURNING MOMENT (in.-kips)	ANCHOR BOLT Stress Ratio
Normal	- 385	± 29	±2320	3.2
Upset	- 439	±102	±11070	11.1
Faulted	+ 626	±195	±18880	12.4

a) A pressurizer safety and relief line break is controlling

TABLE 3.9N-18
PRIMARY PIPE RESTRAINT LOADS AND STRESSES

Restraints	Loading Condition	Stresses	
		Load (kips)	(Pct of Allowable)
Crossover leg bumper, SG side	Upset Faulted	(a)	
Crossover leg bumper, RCP side	Upset Faulted	(a)	
Crossover leg vertical run	Upset Faulted	(a)	
Steam generator inlet	Upset Faulted	(a)	
Primary shield wall (hot leg)	Faulted	(a)	
Primary shield wall (cold leg)	Faulted	(a)	

a) The postulated break dynamic effects associated with the primary pipe whip restraints are no longer required to be considered as justified by leak-before-break application to the primary coolant loop piping.

TABLE 3.9N-19A
REACTOR VESSEL SUPPORT LOADS AND STRESSES FOR UNIT 1
SUPPORT LOADS[a]

(Sheet 1 of 2)

Loading Condition	Load (kips)	
	Vertical	Tangential
	Hot Leg / Cold Leg	Hot Leg / Cold Leg
Pressure + Weight	586 / 582	7 / 12
Normal Thermal	442 / 204	28 / 18
Upset	497 / 247	46 / 13
Over-temperature Thermal		
Faulted	800 / 443	126 / 148
Over-temperature Thermal		
OBE	407 / 306	283 / 412
SSE	582 / 428	435 / 655
LOCA	1098 / 1132	314 / 461
Normal[b]	1028 / 786	35 / 6
Upset[c]	1435 / 1092	318 / 418
Faulted[d]	1285	897

TABLE 3.9N-19A
REACTOR VESSEL SUPPORT LOADS AND STRESSES FOR UNIT 1
SUPPORT LOADS[a]

(Sheet 2 of 2)

REACTOR VESSEL SUPPORT STRESSES

Loading Condition	Stress Ratio (pct of Allowable)
Normal	28.0
Upset	57.6
Faulted	64.6

a) The load table does not reflect the dynamic effects of the breaks 1 through 11 (Table 3.6B-2 and Figure 3.6B-9).

b) Pressure + Weight + Normal Thermal.

c) Pressure + Weight + Upset Thermal + OBE.

d) Higher of (1) Pressure + Weight + $(SSE^2 + LOCA^2)^{1/2}$ or (2) Pressure + Weight + Faulted Thermal.

TABLE 3.9N-19B
REACTOR VESSEL SUPPORT LOADS AND STRESSES FOR UNIT 2
SUPPORT LOADS[a]

(Sheet 1 of 2)

	Load (kips)	
Loading Condition	Vertical	Tangential
Dead Weight	540	41
Thermal	465	41
O. T. Thermal	338	114
Pressure	2	10
OBE	404	405
SSE	742	648
LOCA-1[b],[c]	1113	370
LOCA-2[b],[d]	590	606
Normal [e]	1345	165
Upset [f]	1749	570
Faulted-1 [g]	1480	756
Faulted-2 [g]	1285	897

TABLE 3.9N-19B
REACTOR VESSEL SUPPORT LOADS AND STRESSES FOR UNIT 2
SUPPORT LOADS[a]

(Sheet 2 of 2)

REACTOR VESSEL SUPPORT STRESSES

Loading Condition	Actual Stress (ksi)	Allowable Stress (ksi)	Stress Ratio (Pct of Allowable)
Normal	P_M = 6.71 $P_M + P_B$ = 6.87	S_m = 23.3 1.5SM = 34.95	28.8 19.7
Upset	P_M = 13.7 $P_M + P_B$ = 17.1	S_m = 23.3 1.5 S_m = 34.95	58.6 48.8
Faulted[g]	P_M = 43.5 $P_M + P_B$ = 48.8	0.70 S_u = 49.0 1.05 S_u = 73.	88.9 66.5

a) The load table does not reflect the dynamic effects of the breaks 1 through 11 (Table 3.6B-2 and Figure 3.6B-9).

b) Includes dead weight

c) LOCA-1: Maximum Vertical Load with corresponding Tangential Load

d) LOCA-2: Maximum Tangential Load with corresponding Vertical Load

e) Dead weight + Thermal + O.T. Thermal + Pressure

f) Normal + OBE

g) Deadweight + Pressure + $(SSE^2 + LOCA^2)^{1/2}$

TABLE 3.9N-20
CRDM HEAD ADAPTOR BENDING MOMENTS

	LOCA (in-kip)	COMBINATION OF SSE AND LOCA (in-kip)	% OF ALLOWABLE
Unit 1			
Longest CRDM	32	109	45%
Shortest CRDM	86	141	58.8%
Unit 2			
Longest CRDM	101.30	108.67	45.3%
Shortest CRDM	143.125	166.69	69.5%

TABLE 3.9N-21A
MAXIMUM STRESSES IN THE REACTOR COOLANT LOOP PIPING FOR UNIT 1

	Hot Leg		Crossover Leg		Cold Leg	
Evaluation	Maximum	Allowable[a]	Maximum	Allowable[a]	Maximum	Allowable[a]
Eq 9 design stress (ksi) (DW, P, OBE)	25.4	28.35	24.4	28.35	22.2	28.35
Eq 9 faulted stress (ksi) (DW, P, SSE, LOCA)[b]	53.5	56.7	30.7	56.7	26.1	56.7
Eq 12 stress (ksi)	31.3	56.7	6.5	56.7	20.2	56.7
Eq 13 stress (ksi)	56.8	57.46[c]	55.6	56.7	57.1	57.46[c]
Fatigue usage factor	0.85	1.0	.012	1.0	0.22	1.0

Notes:

[a]: Allowable stress based on Material SA-351 CF8A @ 650°F

[b]: Reactor Coolant Loop piping stresses in this table are based on postulated breaks to the 6" SI, 4" Spray, SG Feedwater, SG Auxiliary Feedwater and Main Steam Line.

[c]: Allowable stress based on Material SA-351 CF8A @ 618°F

TABLE 3.9N-21B
MAXIMUM STRESSES IN THE REACTOR COOLANT LOOP PIPING FOR UNIT 2

	Hot Leg		Crossover Leg		Cold Leg	
Evaluation	Maximum	Allowable[a]	Maximum	Allowable[a]	Maximum	Allowable[a]
Eq 9 design stress (ksi) (DW, P, OBE)	20.7	28.35	22.7	28.35	22.3	28.35
Eq 9 faulted stress (ksi) (DW, P, SSE, LOCA)[b]	26.4	56.7	29.8	56.7	28.6	56.7
Eq 12 stress (ksi)	31.4	56.7	6.6	56.7	20.2	56.7
Eq 13 stress (ksi)	57.45	57.46[c]	55.6	56.7	57.1	57.46[c]
Fatigue usage factor	.85	1.0	.10	1.0	.20	1.0

Notes:

[a]: Allowable stress based on Material SA-351 CF8A @ 650°F

[b]: Reactor Coolant Loop piping stresses in this table are based on postulated breaks to the 6" SI, 4" Spray, SG Feedwater, SG Auxiliary Feedwater and Main Steam Line.

[c]: Allowable stress based on Material SA-351 CF8A @ 618°F

TABLE 3.9N-22
ADDITIONAL LOADING COMBINATIONS FOR SAFETY AND POWER RELIEF VALVE LINES

Condition Classification	Loading Combination	Applicable Stress Criteria
Design	Design Pressure Deadweight Valve Thrust	Design
Normal and Upset (Fatigue)	Normal Transient Upset Transients Deadweight Valve Thrust	Levels A and B (Fatigure Evaluation)
Faulted	Pressure Deadweight Valve Thrust	Level D

3.9B MECHANICAL SYSTEMS AND COMPONENTS

The text marked “later” will be included in future amendments after the required analysis has been completed in a timely fashion according to licensing requirements.

3.9B.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9B.1.1 Plant Conditions & Component Operating Conditions

3.9B.1.1.1 Plant Conditions

Plant conditions categorized as Normal, Upset, Emergency, Faulted and Test are defined in the ASME B&PV Code Section III, paragraph NB 3113, and are discussed below.

The following five plant conditions as defined in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are considered in the design of mechanical systems and system supports.

1. Normal conditions

   Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted or testing conditions.

2. Upset conditions (incidents of moderate frequency)

   Any deviations from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage.

3. Emergency conditions (infrequent incidents)

   Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage. The conditions have a low probability of occurrence.

4. Faulted conditions (limiting faults)

   Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

5. Testing conditions

   Testing conditions are those pressure tests including hydrostatic tests, pneumatic tests, and leak tests specified. Other type of tests shall be classified under normal, upset, emergency or faulted conditions.

Normal Conditions

The following primary system transients are considered normal conditions:

1. Heatup and cooldown at 100°F per hour.
2. Unit loading and unloading at 5 percent of full power per minute.
3. Step load increase and decrease of 10 percent of full power.
4. Large step load decrease with steam dump.
5. Steady state fluctuations.
6. Feedwater cycling at hot shutdown.
7. Loop out of service.
8. Unit loading and unloading between 0 and 15 percent of full power.
9. Boron concentration equalization.
10. Refueling.

Upset Conditions

The following primary system transients are considered upset conditions:

1. Loss of load (without immediate reactor trip).
2. Loss of power.
3. Partial loss of flow.
4. Reactor trip from full power.
5. Inadvertent Reactor Coolant System depressurization.
6. Inadvertent startup of an inactive loop.
7. Control rod drop.

8. Inadvertent Emergency Core Cooling System actuation.

9. Operating Basis Earthquake.

Emergency Conditions

The following primary system transients are considered emergency conditions:

1. Small loss of coolant accident.

2. Small steam break.

3. Complete loss of flow.

4. High Energy Line Break (except as noted in items 1, 2, and 3 in the Faulted Condition)

Faulted Conditions

The following primary system transients are considered faulted conditions. Each of the following accidents should be evaluated for one occurrence:

1. Reactor coolant pipe break (large loss of coolant accident).

2. Large steam line break.

3. Feedwater line break.

4. Reactor coolant pump locked rotor.

5. Control rod ejection.

6. Steam generator tube rupture.

7. Safe Shutdown Earthquake.

3.9B.1.1.2 Component Operating Conditions

Component operating conditions are categorized under the same categories as plant conditions (Normal, Upset, Emergency, Faulted, and Test). These component operating conditions do not necessarily correspond to the plant conditions. Combinations of loadings associated with various plant conditions are categorized and specified as one or more component operating conditions.

All combinations of loads imposed on active components, which are relied upon to operate normally in the course of accomplishing a safety function under all plant conditions, are specified as normal component operating conditions and as design conditions for ASME B&PV Code, Section III design of the component.

For inactive components which accomplish a safety-related function by virtue of their pressure retaining integrity and are not required to perform a mechanical motion, some loading

combinations are to be specified as upset, emergency, or faulted component operating conditions in addition to normal conditions.

3.9B.1.2 Computer Programs Used in Analyses

Computer programs that can be used in the dynamic and static analyses of seismic Category I piping systems are described herein. Developed by G&H and others, these programs are also occasionally linked and modified to suit the need for a solution to a particular dynamic problem. See also Section 3.7B(A). Additional computer programs used in the analysis of ASME Code Class 2 and Class 3 piping systems, including supports for ASME Code Class 1, 2, and 3 piping, are provided in Appendix 3B.

1. Finite Element Computer Program (ANSYS)

   This is a general purpose computer program for the solution of a large class of problems in engineering. This computer program, identified by the acronym ANSYS for engineering analysis system [1], provides a flexible framework for implementation of the finite element analysis technology. The program has the capabilities for static and dynamic, elastic and plastic, fluid flow, and transient heat transfer analyses.

   The matrix displacement method of analysis, based on finite element idealization, is used throughout. A library of more than 40 finite elements is available for static and dynamic analyses. There are various types of these elements, e.g., plane stress, axisymmetric triangles, three dimensional solids, springs, masses, dampers, plates, axisymmetric shells, general shells, and friction interface elements.

   The program uses a direct solution, developed by the matrix displacement method, for the system of simultaneous linear equations.

   Plotting subroutines are available.

2. Computer Program (ADLPIPE)

   The ADLPIPE [2] computer program performs static and dynamic analyses of complex nuclear safety class 1, 2 & 3 piping systems. All safety-related piping systems that require analysis are analyzed using the ADLPIPE program. The input data is preprocessed and plots are made for input and model evaluation. The output automatically includes a stress analysis in accordance with requirements of ANSI B31.1 (1967), ANSI B31.1 (1973), and ASME B&PV Code, Section III, Classes 1, 2, and 3.

   The analysis in ASME B&PV Code, Section III, Class 1 includes fatigue usage and simplified elastic-plastic analyses. All forces, moments, deflections, and a summary stress report are included in the output. Additionally, the program has orthographic, isometric, and stereoscopic plotting capabilities to aid in checking input and interpreting computed results.

   The static loads on the piping systems are thermal internal pressure deadweight, static equivalent of seismic forces, externally applied forces and moments, and wind effects.

The dynamic seismic loads are computed using normal mode theory and seismic response spectra, or time history forcing functions.

Normal mode technique in conjunction with the three-dimensional response spectra is used for obtaining seismic response. The resultant internal forces and moments are computed from the SRSS of their modal values and for closely spaced modes, ADLPIPE has an option to take absolute sum of forces and moments. Algorithms used in this program for the extraction of eigenvalues and associated eigenvectors are the Jacobi rotation scheme and the Givens-Householder scheme with modification [3]. The program is based on the systematic use of transfer matrices.

One of the techniques used is documentation by benchmark calculations. This type of documentation transcends the complex mathematics, programming, and computer systems that are involved in program solution and allows direct comparison of computed results. The benchmark calculation is the principal form of documentation of ADLPIPE, and a number of these benchmarks are presented in part in this section.

Two auxiliary methods of problem solution evaluation are also included in ADLPIPE. The first is a detailed input error check routine, and the second is an intermediate printout of mathematical manipulations and calculations. There are three types of documentation of ADLPIPE. The first is the multitude of hand checks that are made during the development and change of the program. The second is by the many user groups, who have their own method of evaluation and documentation, both analytical and experimental; these groups have contributed immeasurably to the current state of ADLPIPE reliability.

The third type is the documentation and internal checks that Arthur D. Little, Inc., has generated. This type of documentation and internal checks is in four forms as follows: 52 common errors are checked for and automatically reported; all internal program data can be printed during problem solution; sample problems (benchmarks) are compared to other solutions; there is a description of the mathematical techniques that are used.

a. Input Check

   An automatic message for 52 different types of input error is provided [4].

b. Intermediate Data

   1. Force vectors are printed prior to inversion of the stiffness matrix.

   2. Deflection vector is printed after stiffness matrix inversion.

   3. Member data is printed out after input is read.

   4. Contracted stiffness matrix is printed prior to inversion.

   5. Eigenvectors or dynamical matrix are printed after eigenvalue routine.

   6. Eigenvalues of dynamical matrix are printed after eigenvalue routine.

7. Dynamical matrix is printed after formation from stiffness matrix.

8. Flexibility matrix is printed after inversion of stiffness matrix.

9. Reduced stiffness matrix and mass vector are printed after reduction of stiffness matrix to order of dynamical matrix.

10. Flags, properties, stress coefficients, and moments for each member are provided.

11. Modal effective mass for dynamic model/solution evaluation are provided.

c. Typical Benchmark Calculations

Table 3.9B-9 defines and references eight benchmark calculations typical of the verification that has been done with ADLPIPE. The first four are illustrated in Appendix A of Reference [4], where each problem is briefly defined and solutions from other sources are compared to the ADLPIPE solution.

d. Description of the Analytical Techniques

The following documents describe the mathematical techniques that are used:

1. Generalized Piping System Response to Ground Shock Spectra by I.W. Dingwell, Arthur D. Little, Inc.

2. A Method of Computing Stress Range and Fatigue Damage in a Nuclear Piping System by W.B. Wright and E.C. Rodabaugh, Nuclear Engineering and Design, Volume 22, 1972.

3. Method of Calculating Static and Dynamic Moments for Stress Evaluation at Tees and Branches, Arthur D. Little, Inc., May 1973.

4. Method of Calculating Thermal Stress Range for T1, T2, Ta, and Tb Terms, Arthur D. Little, Inc., May 1973.

5. Mathematical Analysis and Logical Procedure, I.W. Dingwell and R.T. Bradshaw, Arthur D. Little, Inc., 1970.

3.9B.1.2.3 Computer Program (SUPERPIPE)

SUPERPIPE is a general purpose piping program which performs comprehensive structural analyses of linear elastic piping systems for dead weight, thermal expansion, seismic spectra or time history, arbitrary force time history and other loading conditions. Analyses are performed to ASME requirements for Class 1, 2 and 3 systems. See Reference [23] of Section 3.7B(A).

A piping system is idealized as a mathematical model consisting of lumped masses connected by massless elastic members. The location of lumped masses is chosen to accurately represent the dynamic characteristics of the system for a dynamic analysis, and to adequately represent the weight distribution of the system for dead load analysis. Static or dynamic equilibrium

equations are formulated using the direct stiffness method, in which element stiffness matrices are formed according to virtual work principles and assembled to form a global stiffness matrix for the system, relating external forces and moment to joint displacements and rotations. Appropriate stiffness modifications for curved components are included. Diagonal mass and damping matrices are assumed.

Static equilibrium equations are solved using Gaussian reduction techniques on the global stiffness matrix. For dynamic problems, the equilibrium equations may be solved using either step-by-step direct integration of the coupled equations of motion, or by first calculating natural frequencies and mode shapes and transforming the system into a set of uncoupled equations of motion. For seismic analysis of piping systems the latter approach is typically used in the dynamic analysis technique known as the response spectrum mode super-position method. In this technique, the earthquake excitation is characterized by acceleration response spectra, and the total response of the system is evaluated as a combination of the individual responses of the significant natural modes of vibration of the system. Natural frequencies and mode shapes are calculated using the determinant search technique. The method of combination of model responses can be selected from any one of those specified in Regulatory Guide 1.92. Earthquakes acting in all three directions simultaneously may be computed.

SUPERPIPE has been verified for a comprehensive set of sample problems. This has included bench marking against the ASME Sample Problems 1 and 6 contained in ASME publication “Pressure Vessel and Piping 1972, Computer Program Verification”, and against a Class 1 sample problem contained in ASME publication “Sample Analysis of a Piping System, Class 1 Nuclear”, 1972. Extensive bench marking has also been performed against the programs, PISOL1A and PISOL3A which are well recognized and utilized throughout the industry. Additionally, the program has been bench marked against the programs such as HUPIPE, ADLPIPE, PIPESD and EDSGAP. SUPERPIPE has been used on a number of domestic and foreign nuclear plants.

3.9B.1.2.4 LEAP, (ME101) Linear Elastic Analysis of Piping

Author: Bechtel Corporation, San Francisco, California

Source: Bechtel Corporation, San Francisco, California

Description

ME101 is a finite element computer program which performs linear elastic analysis of piping systems, transient thermal analysis, local stress analysis, and response spectra merging.

The input data format is specifically designed for pipe stress analysis. ME101 performs a thorough check of the input prior to performing analysis. The program rearranges the geometry automatically to optimize the finite element model. The coordinate and keyword data can be specified in English, Metric or S1 units.

ME101 performs stataic and dynamic analysis of piping systems in accordance with Section III of the ASME Boiler and Pressure Vessel Code, ANSI B31.1 Power Piping Code and ANSI B31.3 Chemical Plant and Petroleum Refinery Piping Code. The flexibility factors, stress intensification factors (SIF) and stress indices (B1,B2) are incorporated in the stress evaluations.

Static analysis capabilities include thermal expansion, deadweight, seismic and uniformly distributed loads, and externally applied forces, moments, displacements and rotations. For dead weight and thermal expansion analysis, supports with gaps can be considered.

Response spectrum analysis is based upon standard modal superposition techniques. The input excitation may be in the form of single or multiple seismic response spectra or time dependent loading functions. In the response spectrum analysis, the user may request modal combinations by various alternate methods typical of the industry. ME101 can further consider differential damping according to NRC RG 1.61 and PVRC Code Case N-411 within a single response spectrum analysis.

ME101 uses an out-of-core solution and active column techniques for both static and dynamic analyses. It has no limit to the number of equations or bandwidth.

ME101 considers zero period accelerations with or without missing mass corrections in the seismic response spectrum anayses. In the time history analysis (modal or direct integration), the excitation may be in the form of nodal forces, support displacements or support accelerations. ME101 can also compute the response to dynamic loads using a direct integration time history method considering nonlinear kinematic hardening supports (such as the Bechtel energy absorber) with or without gaps. Transient analyses and response spectrum analyses of this nonlinear type support are also available.

ME101 compares stresses vs. allowables according to ASME/ANSI code equations. ME101 has load combination capabilities that allow the results of several load cases to be combined according to certain algebraic rules to form additional load cases. The additional load cases resulting from the combination may be utilized in stress comparisons or restraint load summaries. ME101 also provides a customized report writer that is designed to create report or table used in equipment nozzle evaluation, piping support/hanger guidance, valve acceleration summary, penetration load summary, special piping component qualification, pipe movement summary etc. This report is construed, directly from data stored in the internal ME101 Data Base (“MASTER” file). This report enables user to manually input the allowable side by side with the extracted value, or allows the user to uniquely define the interaction equations. The report writer module can also screen or filter results according to allowable criteria.

ME101 can generate isometric plots of the piping configuration with optional node numbering. It can also generate plots for deflections and mode shapes.

The program is validated utilizing fifty-nine validation problems, including the eleven NUREG/CR 1677 benchmark problems. Validation is documented in the ME101 program validation manual.

3.9B.1.2.5 LSAPS (ME214) Local Stress Analysis for Piping Systems

Author: Bechtel Corporation, San Francisco, California

Source: Bechtel Corporation, San Francisco, California

ME214 is an interactive computer program developed to perform local stress analysis due to integral welded attachments and non-integral welded attachments such as line bearing and point contact loads, restraint of local pressure and thermal expansion of the pipe. The program is capable of calculating local pipe stresses based on Welding Research Council Bulletin No. 107

and ASME Code Case N-318 and N-392. The program also calculates local pipe stresses due to non-integral welded attachments using analytical solutions as provided in Comanche Peak Design criteria documents.

The program is developed in accordance with ASME Section III Subsection NC and ND. The loading conditions and allowables are in accordance with Subsection NC and ND piping stress requirements. Code case allowables are used if the local pipe stresses are calculated based on the methodology provided in the code cases.

The program has been verified using Bechtel standard computer program ME210 and hand calculation.

Validation is documented in the program validation manual.

3.9B.1.2.6 SAPCAS (ME 215) (ME101FE) Stress Analysis for Pipe Component and Pipe Support Using Finite Element Method

Author: Bechtel Corporation, San Francisco, California

Source: Bechtel Corporation, San Francisco, California

ME 215 (ME101FE) is a special purpose finite element computer program for computing membrane and membrane-plus-bending stress intensities at pipes, pads, attachments, and welds. The piping component can be a circular run pipe, elbow, or square tubular steel. The attachment can be circular pipe, rectangular tube, or rectangular solid Lug. The program is capable of computing local stress on pipe components and supports such as stanchion, trunnion, lug, etc. This program is more refined and will provide a realistic solution for local piping stress as compared to, conservative ME214. The program is capable of calculating local pipe stresses where the results may exceed the limitation of the welding research council (WRC) Bulletin No. 107, ASME code cases, and/or other design documents such as CPNPP 2EP-5.12 and 2EP-5.13. The program is developed based on well established SAP computer program.

The program has been design verified using ANSYS Version 4.4 and ASME Bench Mark Problems. Additional Comparisons have been made with conservative ME214 results and a recent PVRC Technical Paper by D. C. Foster.

3.9B.1.3 Experimental Stress Analysis

Experimental stress analysis methods will not be used in the design of code or non-code components for the faulted plant condition.

3.9B.1.4 Considerations for the Evaluation of the Faulted Condition

1. ASME Code Class 2 and 3 Components

   The ASME Code Class 2 and 3 components are constructed in accordance with the ASME B&PV Code, Section III, Subsections NA, NC, and ND, respectively. The methods for the dynamic analysis for the Class 2 and 3 components are defined in Section 3.9B.

In the event that the design stress limits permitted in the design criteria exceed the yield strength of the material, analysis is performed in accordance with the ASME B&PV Code, Section III, Article NB-3000. The following types of analyses are used:

a. Elastic analysis for the design of Class 2 and 3 components in accordance with subparagraph NB 3227.6 of ASME B&PV Code, Section III.

b. Limit analysis is done at a specific location when the limits on local membrane stress intensity and primary bending stress intensity do not satisfy the limits of ASME B&PV Code, Section III, subparagraph NB 3228.2.

2. Bolts

For the Code Class 2 and 3 components not in Westinghouse's scope, the following are the criteria for the design of bolting:

a. Bolts for Component Supports (Other than Pipe Supports)

Normal and Upset Conditions - The design limits are as per ASME B&PV Code, Section III, Article XVII-2460 and NRC approved ASME Code Cases as indicated in the specification.

Emergency Condition - The bolt allowables are 1.33 times the allowable used in the normal and upset conditions.

Faulted Condition - The bolt allowables for normal and upset conditions are increased by a factor which is the lower of 1.2 (Sy/Ft) or 0.7 (Su/Ft). However, the total allowable stress shall not exceed 0.9 Sy in any case.

Sy = Yield stress

Ft = Allowable tensile stress

Su = Ultimate tensile stress

b. Bolts for Pipe Support Joints

The design limits for bolted pipe support joints are in accordance with the ASME B&PV Code, Section III, 1983 Edition including the Summer 1983 Addenda, paragraphs NF 3225 and NF 3324.6.

c. Bolts for Flange Joints

The design limits for bolted flange joints connecting pipe to pipe and/or pipe to equipment are in accordance with the ASME B&PV Code, Section III, 1983 Edition, subparagraphs NC/ND 3658.2 and NC/ND 3658.3.

d. Bolts and Components

Bolts for components do not exceed the Code allowable stress limits.

3. Non-ASME Code Class Components

The supplier of the mechanical components not covered by the ASME B&PV Code must demonstrate by experimental testing or calculation that the design stress limits (with use of applicable codes and standards) are sufficiently below the elastic limits to ensure operability.

3.9B.2 DYNAMIC TESTING AND ANALYSIS

3.9B.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

The purpose of preoperational vibration and dynamic effects tests of the piping systems and their supports, as described below, is to confirm that these piping systems, restraints, components and supports have been adequately designed to withstand flow induced dynamic loadings under steady state and operational transient conditions anticipated during the life of the plant, and that normal thermal motion is not restrained.

3.9B.2.1.1 Thermal Expansion Test

Thermal expansion (hot deflections) test will be conducted on the following piping systems:

Reactor Coolant System

Main Steam

Steam Supply to Auxiliary Feedwater Pump Turbine

Main Feedwater

Pressurizer Relief Line

RHR in Shutdown Cooling Mode

Auxiliary Steam (within seismic Category I Structure)

Steam Generator Blowdown

Safety Injection System (those line adjoining RCS which experience temp. > 200°F)

Auxiliary Feedwater

CVCS (Charging line from Regen. Hx to RCS, Letdown Line from RCS to Letdown Hx)

During the thermal expansion test, pipe deflections will be measured or observed at various locations based on the location of snubber, hangers, and expected large displacement. One complete thermal cycle, (i.e., cold position to hot position to cold position) will be monitored. For most systems, the thermal expansion will be monitored at cold conditions and at normal operating temperature. Intermediate temperatures are generally not practical due to the short time during which the normal operating temperature is reached. For the Reactor Coolant System

and the Main Steam System, measurements will be made at cold, 250°F, 350°F, 450°F and normal operating temperatures.

Acceptance criteria for the thermal expansion test verify that the piping system is free to expand thermally (i.e., piping does not bind or lock at spring hangers and snubbers nor interfere with structures or other piping), and to confirm that piping displacements do not exceed design limits, as described by ASME Section III, (i.e., the induced stresses do not exceed the sum of the basic material allowable stress at design temperature and the allowable stress range for expansion stresses).

3.9B.2.1.2 Dynamic Transient Response Testing

Instrumented dynamic transient response testing augmented by visual observation for the specification flow modes analyzed will be performed on the following piping systems:

	System	Transient Tested
1.	Main Steam System	Main Turbine Trip at 100% steam flow
2.	Safety Injection System	Safety Injection Pump Trip During Injection
3.	Main Feedwater System	Main Feedwater Pump Trip
4.	Auxiliary Feedwater System	Auxiliary Feedwater Pump (including Steam Supply Line) Trip
5.	Pressurizer Relief Valve	Pressurizer Relief Valve Discharge Piping of Reactor Actuation Coolant System
6.	Service Water System	Service Water Pump Trip & Restart

During the dynamic transient response testing for the specified piping systems, the piping will be instrumented to measure transient loads at selected locations and visually observed at other selected locations to ensure that severe vibrations do not exist. The instrument data will be compared with the analytically predicted values.

Instrumented measurements will also be conducted as needed for other systems and flow modes. The acceptance criteria for dynamic transients is based on the allowable design stress limits for occasional loads, as described by ASME Section III, such that the induced stress does not exceed allowables at design temperature as specified in ASME Section III Subsection NB, NC or ND, as applicable, Paragraph 3650.

If evidence of excessive piping motion is noted, the line(s) will be reviewed to determine what remedial action, if any, is necessary. Repetition of these transients may be required to make the determination of acceptability.

3.9B.2.1.3 Steady State Vibrations Tests

Steady State vibration tests will be conducted on (1) ASME Code Class 1, 2, 3 systems (2) other high-energy piping systems inside Seismic Category I Structures, (3) high-energy portions of

systems whose failure could reduce the functioning of any Seismic Category I Plant Feature to an unacceptable level.

The steady state tests are performed for the systems listed below at normal operating conditions consistent with the requirements of Chapter 14.2.

Reactor Coolant System

Residual Heat Removal System

Safety Injection System

Chemical and Volume Control System

Boron Thermal Regeneration System

Boron Recycle System

Containment Spray System (Minimum flow and test lines only)

Liquid Waste Processing System

Diesel Generator Fuel Oil System

Service Water System

Auxiliary Feedwater System

Spent Fuel Cooling and Cleanup System

Main Feedwater System

Main Steam Reheat and Steam Dump System

Component Cooling Water System

Condensate System

Steam Generator Blowdown System

Vents and Drain System

Chilled Water System

Demineralized and Reactor Makeup Water System

Process Sampling System

Auxiliary Steam System

During normal operating conditions, qualified engineers will observe the lines where accessible, to determine the acceptability of the steady state vibrations. One of the following resolutions will be applied if observed vibrations are excessive: (1) the piping will be monitored by suitable instrumentation at or near the locations where vibrations appear to be excessive to demonstrate that the vibration level does not cause ASME Code stress and fatigue allowables to be exceeded, (2) the cause of the vibration will be eliminated, or (3) the pipe routing and/or the support system will be modified to reduce the vibrations to an acceptable level.

3.9B.2.1.4 Test Performance and Acceptance

Where practical, the above tests will be performed during the preoperational testing phase of the startup program. Those tests that cannot be performed as part of the preoperational testing phase due to required plant conditions will be performed as part of the initial startup and power escalation phase.

For those piping systems in which corrective action is required, the vibration and/or thermal expansion observation shall be repeated to determine that the thermal movements and/or vibration have been reduced to satisfy the acceptance criteria .

Detailed test procedures will be prepared and approved as described by Chapter 14.2.

Design personnel will review the test procedures and acceptance criteria for technical adequacy and evaluate test results. The test results will be compared with the predicted results and acceptance criteria to determine acceptability.

3.9B.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

All safety-related mechanical equipment which is required to retain structural integrity or structural integrity and operability during and after a postulated earthquake is subject to seismic qualification.

Active pumps and valves which must perform a mechanical motion during the course of accomplishing a system safety function include the active ASME Boiler and Pressure Vessel (B&PV) Code Class 2 and Class 3 pumps, Code Class 1, Class 2, and Class 3 valves, and their respective drives, operators, and vital auxiliary equipment. This equipment is qualified by testing, or analysis, or both, in accordance with the criteria given in Subsection 3.9B.3.2 and the recommendations of NRC Regulatory Guide 1.48 and as described in the following paragraphs.

Analysis without testing is accepted if it can be conservatively demonstrated that structural integrity alone can ensure operability of the seismic Category I equipment. When a complete seismic test is impracticable, combinations of testing and analysis are performed.

Seismic qualification by analysis is applicable to mechanical equipment which has relatively simple configurations and which can be modeled accurately. When analytical modeling is used, the equipment is modeled as a network of lumped masses and elastic springs in discrete parts. The response spectrum method is applied to calculate stresses and deformations resulting from the base excitations characterized by the required OBE and SSE in-structure floor response spectra of the seismic Category I buildings, and for the seismic analysis and testing of all seismic Category I subsystems and equipment located in the seismic Category I buildings as described in Section 3.7B.

The calculated seismic stresses are combined with the design load and thermal stresses for the various plant conditions defined in the ASME B&PV Code, Section III. It is ascertained that for each condition the resulting stresses are within the limits specified by the code and the recommendations of NRC Regulatory Guide 1.48.

When structural integrity alone cannot ensure operability for mechanically or structurally complex equipment not amenable to modeling and dynamic analysis, structural integrity and operability during and after a postulated earthquake are ensured by testing. This method consists of mounting the equipment to be qualified on a shake table, which is vibrated in such a way as to equal or exceed the required OBE and SSE in-structure floor response spectra applicable at the equipment locations in the seismic Category I buildings. A minimum of five OBE tests and one SSE test are performed. Equipment is tested in its operational condition; and, when possible, during the tests, operating, thermal and seismic loads are applied simultaneously. Operability is verified both during and after the tests.

Multi-directional seismic loading effects and dynamic coupling of the equipment are considered through the use of multi-axis testing techniques as recommended by IEEE 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, and described in Section 3.7B.2.1.3. When dynamic analysis is used for seismic qualification, the dynamic coupling effect is considered by modeling the equipment with masses with a sufficient number of degrees of freedom and elastic properties representing the multi-directional stiffness of its various interconnecting parts. Multi-directional seismic loading is accomplished by performing the response spectrum analysis for each of the three orthogonal directions of earthquake excitation and combining the results by the square root of the sum of the squares (SRSS) technique.

A general classification of safety-related mechanical equipment and applicable quality standards is given in Table 3.2-1. Fluid system components and the applicable codes are classified in Table 17A-1.

A detailed description of seismic analysis and testing procedures is given in Section 3.7B.2.1.3.

The seismic qualification program for safety-related mechanical equipment is conducted by the equipment vendor. The procedures proposed for qualification and the results obtained by analysis or testing, or both, are submitted to the Applicant or his Agent for review.

All supports of seismic Category I mechanical equipment are seismically qualified to ensure their structural capability to withstand seismic excitation. The seismic qualification is accomplished by analysis, testing, or a combination of both for a particular support or a support representative of a group of supports.

If supports are similar and justified as such or if the worst case support determined by consideration of dynamic response (stiffness, structural strength, supported load) is chosen from a group of supports to be qualified and justified as such, only one of the similar supports or the worst case support requires a complete dynamic seismic analysis or a full-scale test, or a combination of both.

Justification of this procedure is based upon a simplified comparison analysis or by past experience indicating that the supports to be qualified are similar or that the worst case has been chosen. Upon such justification and dynamic analysis, or full scale testing, or a combination of

both of the similar or worst case support, the group of supports being investigated is accepted as seismically qualified.

The criteria governing the analysis or testing methods for seismic qualification of supports are presented in Section 3.7B.2.1.3

3.9B.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

See Section 3.9N.2.3

3.9B.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

See Section 3.9N.2.4

3.9B.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Condition

See Section 3.9N.2.5

3.9B.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

See Section 3.9N.2.6

3.9B.3 ASME CODE CLASS 2 AND 3 COMPONENTS AND COMPONENT SUPPORTS

The ASME Code class components are designed and fabricated in accordance with ASME B&PV Code, Section III requirements.

Note: Inspection of Attachment welds during Hydrostatic Testing, as required by the ASME B&PV Code, was not performed on all type MV containment penetrations for Unit 1.

The qualification of ASME Section III Code Class 2 and 3 piping systems and their supports are in accordance with:

ASME Boiler and Pressure Vessel Code, Section III, Division 1 Nuclear Power Plant Components, 1974 Edition including the Summer 1974 Addenda Subsection NC and ND, and 1974 Edition including the Winter 1974 Addenda Subsection NF.

In addition to the above, as permitted by paragraph NA-1140 of the 1974 Edition of the Code, specific paragraphs in more recent editions and addenda of the ASME Code have been invoked.

Specific ASME Code Edition, Addenda and Cases utilized for the qualification of ASME Section III Code Class 2 and 3 piping systems and supports are documented in design, purchase and construction specifications. The use of Code Case N-397 is not utilized for non-NSSS piping stress analysis for Unit 1 but any subsequent usage will be detailed in the FSAR and in compliance with reference [9]. Details regarding the application and usages of N-318 are as listed in Table 3.9B-1F.

3.9B.3.1 Loading Combinations, Design Transients, and Stress Limits

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems for ASME B&PV Code Class 2 and Class 3 components are presented in the sections which describe the systems.

3.9B.3.1.1 Design Loading Combinations

Design loading combinations and stress limits for ASME B&PV components, piping and piping supports are provided in the following tables:

	Loading Combinations	Stress Limits
Components	3.9B-1A	Note 1
Piping	3.9B-1B	3.9B-1B
Piping Supports	3.9B-1C	3.9B-1D 3.9B-1E

Note 1 Stress limits for each of the loading combinations are presented in Tables 3.9B-2, 3.9B-3, 3.9B-4, and 3.9B-5 for vessels, inactive[(a)] pumps, active pumps, and valves respectively.

The design loading combinations are categorized with respect to normal, upset, emergency, and faulted plant conditions. (Refer to Section 3.9B.1.1 for definition of plant conditions). Peak dynamic responses from loadings shown in Tables 3.9B-1A and 3.9B-1B are combined using the Square Root of the Sums of the Squares (SRSS) technique. This method of combining dynamic responses is consistent with the position outlined in NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1 dated May, 1980. Active[(b)] pumps and valves are discussed in Subsection 3.9B.3.2. Table 3.9B-8 lists all non-NSSS active pumps by system. Table 3.9B-10 lists all non-NSSS active valves including their design parameter and safety function.

For the design of supports for Class 2 and 3 active pump and valves, load combinations are also given in the component design specification. Stress limits conform to the requirements of Regulatory Guide 1.48 and Subsection NF of the ASME B&PV Code, Section III. In addition, component supports are included in the analysis of the component to ensure that deformations do not preclude operability. Operability requirements are presented in Section 3.9B.3.2.

a. Inactive components are those whose operability is not relied upon to perform a safety function while being subjected to the loading combinations associated with the respective plant operating condition categories.

b. Active components are those whose operability is relied upon to perform a safety function while being subjected to the loading combinations associated with the respective plant operating condition categories.

Systems which are required to operate during and after a postulated plant accident condition comply with the functional capability requirements delineated in References [5], [6], [7], and [8] in addition to the ASME Code requirements.

This requirement will ensure that the piping system will maintain its capability to deliver the rated flow and retain its dimensional stability under events specified above.

References [7] and [8] provides an alternative functional capability evaluation for stainless steel elbows and bends.

3.9B.3.2 Pump and Valve Operability Assurance

A list that identifies all active Code Class 2 and 3 pumps and valves is presented in Tables 3.9.B-8 and 3.9B-10.

Design specifications for active pumps and valves include the requirements for operability under the specified plant conditions. The design specifications define the design loads and the corresponding stress limits as discussed in Section 3.9B.3.1.1, relevant environmental conditions and operability requirements.

Active pumps and valves are qualified for operability in accordance with the requirements of Regulatory Guide 1.48. Pump and valve supports are designed in accordance with ASME B&PV Code Section III, Subsection NF. The following qualification methodology is used in the acceptance review of a vendor's operability program to ensure active valve or pump operability.

All active pumps are qualified for operability by first being subjected to rigid tests both prior to and after installation in the plant. The in-shop tests include the following:

1. Hydrostatic tests of pressure retaining parts to 150 percent times the design pressure.

2. Performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, net positive suction head (NPSH), and other pump/motor parameters.

Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined by the manufacturer based on the bearing material, clearances, oil type, and rotational speed.

In addition to these tests, the safety-related active pumps are qualified for operability during an SSE condition by ensuring that the pump is not damaged during the seismic event and that the pump continues operating after being subjected to the SSE loads.

The pump manufacturer is required to show that the pump operates normally when subjected to the maximum seismic accelerations and maximum nozzle loads associated with the plant faulted condition. It is required that a test or a dynamic analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. When having a natural frequency above 33 Hz, the pump is considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with conservative SSE accelerations. The deflections determined from the static shaft analysis are compared to the allowable rotor

clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) is of short duration. To avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to below the material elastic limit(s), as indicated in Table 3.9B-4. The average membrane stresses (σm) for the faulted condition loads are maintained at 1.0 S, and the maximum stress in local fibers (σm + bending stress σb) is limited to 1.5 S. In addition, the pump casing stresses caused by the maximum seismic nozzle loads are limited to stresses outlined in Table 3.9B-4. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to ensure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9B-4 as allowables can ensure that critical parts of the pump are not damaged during the faulted condition and that the reliability of the pump for postfaulted condition operation is not impaired by the seismic event.

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations are determined using the same conservatisms contained in the accelerations used for rigid structures. The static analysis is performed using the adjusted accelerations; the stress limits stated in Table 3.9B-4 must still be satisfied.

The second criterion necessary to ensure operability is that the pump functions throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized; i.e., there is no rotation. Typically, the rotor can be seized five full seconds before a circuit breaker opens to prevent damage to the motor. However, the high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event prevents the rotor from losing its function. In actuality, the seismic loadings cause only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed. Therefore, the pump does not shut down during the SSE and it operates at the design speed despite the SSE loads.

To complete the seismic qualification procedures, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment which is identified to be vital to the operation of the pump or pump motor and which is not qualified for operation during the pump analysis or motor qualifications is also separately qualified for operation at the accelerations it will be subjected to where it is mounted. The pump motor and vital auxiliary equipment are qualified by meeting the requirements of IEEE 344-1975. If the testing option is chosen, sine-beat testing for electrical equipment will be justified by satisfying one or more of the following requirements to demonstrate that multi-frequency response is negligible or the sine-beat input is of sufficient magnitude to conservatively account for this effect.

1. The equipment response is basically due to one mode.

2. The sine-beat response spectra envelop the floor response spectra in the region of significant response.

3. The floor response spectra consist of one dominant mode and have a narrow peak at this frequency.

The degree of coupling in the equipment does, in general, determine if a single or multi-axis test is required. Multi-axis testing is required if there is considerable cross-coupling. If coupling is very light, then single-axis testing is justified; or, if the degree of coupling can be determined, single-axis testing can be used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

From the previous arguments, the safety-related pump/motor assemblies are not damaged and continue operating under SSE loadings, and therefore, they perform their intended functions.

These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate, and ensure, the seismic operability of the active pumps.

The functional capability of active pumps after a faulted condition is ensured because only normal operating loads and steady-state nozzle loads then exist. Because the pumps are not damaged during the faulted condition, the postfaulted condition operating loads are identical to the normal plant operating loads. This is ensured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and postfaulted conditions be limited to the magnitudes of the normal condition nozzle loads. The postfaulted condition capability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

Safety-related active valves must perform their mechanical motion during the course of accomplishing a system safety function. When appropriate, assurance is supplied that these valves operate during a seismic event. When full scale tests are not performed, tests accompanied by analyses are conducted for these active valves.

The safety-related valves are subjected to a series of stringent tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME B&PV Code, Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, functional tests to verify that the valve opens and closes within the specified time limits when subjected to the design differential pressure, operability qualification of motor operators for the environmental conditions over the installed life (i.e., aging, radiation, accident environment simulation, and so forth) according to procedures specified in IEEE 382. Cold hydro qualification tests, hot functional qualification tests, periodic inservice inspections, and periodic inservice operation are performed in situ to verify and ensure the functional capability of the valve. These tests guarantee reliability of the valve for the design life of the plant. The valves are designed using either stress analyses or the pressure containing minimum wall thickness requirements. On active valves not subject to dynamic testing for SSE conditions, an analysis of the extended structure is also performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure; the maximum stress limits allowed in these analyses show structural integrity and are the limits recommended by ASME for the particular ASME class of valve analyzed.

In addition to these tests and analyses, representative valves of each design type are tested for verification of operability during a simulated seismic event by demonstrating operational capabilities within the specified limits. The testing procedures are described as follows:

The valve is mounted in a manner which conservatively represents typical valve installations. The valve includes the operator and all accessories normally attached to the valve in service. The operability of the valve during SSE is demonstrated by satisfying the following criteria:

1. Active valves are typically designed to have a first natural frequency which is greater than 33 Hz. This can be shown by a suitable test or analysis.

2. The extended structure of the valve system is statically deflected an amount equal to that determined by an analysis as representing SSE accelerations applied at the center of gravity of the operator alone in the direction of the weakest axis of the yoke. The design pressure of the valve is simultaneously applied to the valve during the static deflection tests.

3. The valve is then operated while in the deflected position. The valve must perform its safety-related function within the specified operating time limits.

4. Motor operators, pilot solenoid valves, and limit switches necessary for operation are qualified as operable during SSE by appropriate IEEE seismic qualification standards, prior to their installation on the valve.

5. Operability may be demonstrated on a shaketable, with the valve body at design pressure, and temperature, in lieu of the static deflection tests.

6. The accelerations which are used for the valve qualification are equivalent, as justified by analysis, to (3.0g) in two orthogonal horizontal directions and (2.0g) in a vertical direction. The analyst will verify that valve seismic accelerations do not exceed the design g valves above.

7. If the frequency of the valve is less than 33 Hz, a dynamic analysis of the valve is performed to determine the equivalent deflection which is applied during the static test. The analysis provides the amplification of the input acceleration with consideration of the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations are determined using the same conservatisms contained in the (3.0g) horizontal and (2.0g) vertical accelerations used for rigid valves. The adjusted accelerations are then used in the static analysis, and valve operability is ensured by the methods previously outlined in steps 2 through 4, using the modified acceleration input.

This testing program applies to valves with powered operators which have extended structures. The testing is conducted on a representative number of valves. Valves from each of the primary safety-related design types are tested. Valve sizes which cover the range of sized in service are qualified by the test; the results are used to qualify all valves within the intermediate range of sizes.

Valves which are safety-related but can be classified as not having an extended structure, such as check valves and safety relief valves, are considered separately.

The check valves are characteristically simple in design and their operation is not affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact and there are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads caused by maximum seismic excitation do not affect the functional capability of the valve since the valve disc is designed to be isolated from the casing wall. The clearance supplied by the design around the disc prevents the disc from becoming bound or restricted because of any casing distortions caused by nozzle loads.

Therefore, the design of the valves is such that once the structural integrity of the valve is ensured, using standard design or analysis methods, the capability of the valve to operate is ensured by the design features. In addition to these design considerations, the valve also undergoes in-shop hydrostatic tests, and in-shop seat leakage test, and periodic in situ valve exercising and inspection to ensure the functional capability of the valve.

Manual valves which are Active are those for which credit is taken for post-accident operator action to perform a Nuclear Safety Function. The utilization of a manual Active valve is such that these valves are not required to perform a mechanical motion while being subjected to seismic loads. Therefore, a demonstration of operability of the valve during SSE is not required. Once the structural integrity of the valve is ensured, by compliance with stress limits using standard design or analysis methods, the capability of the valve to operate is assured by the design features. In addition to these design considerations, the valve also undergoes in-shop hydrostatic tests, in-shop seat leakage tests, and periodic in-situ valve exercising and inspection to assure the functional capability of the valve.

The functional capability of the feedwater check valves to operate following a postulated feedwater line break are also demonstrated.

Feedwater check valve maximum disc impact velocity and differential pressure across the valve disc are determined. A stress analysis of the valve considers the impact and the seismic inertia loads, which demonstrates the valve design for the postulated break.

Using the methods described, all the safety-related active valves in the systems are qualified for operability during a seismic/dynamic event. These methods conservatively simulate the seismic/dynamic event and ensure that the active valves perform their safety-related function when necessary.

3.9B.3.3 Design and Installation of Pressure Relief Devices

The piping systems are designed to accommodate the effects of weight, dynamic blowdown thrust loads, and bending torsional and axial loads. Deflections resulting from these loading conditions are calculated in the stress analysis. Nozzle size and corresponding wall thickness are selected to ensure that the requirements of the design pressure and temperature are met and that the allowable stress limits are not exceeded.

The relief/safety valve stations in seismic Category I piping systems are shown on the piping and instrumentation diagrams in appropriate sections of the FSAR.

1. Installation Criteria

   The relief/safety valve nozzles, the exhaust stacks, and the piping on which the valves are installed are designed to withstand the thrusts that are imposed on the piping system when the relief/safety valve opens.

   The requirements set herein are in addition to all requirements of the applicable specifications, codes, and standards.

Piping systems subject to relief/safety valve reaction forces are provided with shock absorbing devices if necessary. This does not interfere with normal or intended thermal movement of the piping system.

The discharge piping is suspended, supported, restrained, guided, or anchored to avoid interference with structural members or equipment as well as to accommodate insulation limitations.

2. Design Criteria

a. Open Systems

For relief valves which are vented to the atmosphere and mounted in seismic Category I piping systems, Equation (9) of paragraph NC 3652, of the ASME B&PV Code, Section III, is used with the MB term to include the reaction force moment. In addition, the criteria of ASME Code, Section III, Appendix O are met.

Total resultant forces on the outlet elbow of pressure relieving devices are determined as follows:

$$\text{Total resultant forces} = \frac{WV}{g} + P\,A = Fv + Fp$$

where: W = lb/sec

A = area of pipe, in^2

V = velocity, ft/sec

P = pressure, psig

g = 32.2 ft/sec^2

Fv = thrust due to velocity, lb

Fp = thrust due to pressure, lb

b. Closed Systems

A closed relief system may be either a system in which fluid discharges into a closed vessel or an open discharge system with a long discharge pipe. Of particular concern in closed relief systems are the large forces that may occur on piping that contains water seals (slug flow), two-phase flow, or if there is a water column in the discharge piping.

To establish the forcing functions necessary to perform a structural analysis of the piping, thermal/hydrodynamic models of the piping system are constructed. These models consist of one-dimensional representation of the piping system divided into reservoirs, pumps, valves, lengths of piped segments, branch connections, and other special piping components. Effects such as flow

restrictions and frictional resistance are considered. The time dependent pressure, temperature, density, velocity, and momentum are computed. Unbalanced segment forces are then obtained as a function of time.

The forcing functions are them applied to the piping structural model and system responses are determined by performing a time-history dynamic analysis.

As an alternative to using dynamic analysis to generate fluid transient forcing functions, conservative hand calculations may be performed to develop bounding pipe segment forces. These forces then may be analyzed statically, with a Dynamic Load Factor (DLF) or 2.0, or dynamically to obtain piping structural responses.

3.9B.3.4 Component Supports

3.9B.3.4.1 Nuclear piping (ASME Class 1)

Class 1 pipe support are designed in accordance with FSAR Table 3.9B-1C and 3.9B-1D.

3.9B.3.4.2 Nuclear Piping (ASME Class 2 and 3)

Plant conditions and load combinations for Class 2 and Class 3 components are shown in Table 3.9B-1A. Safety-related component supports are designated with the same safety class as their respective components, and are subject to the same plant conditions and loading combinations.

Class 2 and 3 supports are designed as follows:

1. Component Standard Supports

Normal Condition	Load Rating = T.L. x $1.0(S \text{ or } F_{All})/S_u$
Upset Condition	Load Rating = T.L. x $1.0(S \text{ or } F_{All})/S_u$
Emergency Condition	Load Rating = T.L. x $1.33\ (F_{All}/S_u)$
Emergency Condition (Plate and Shell Type)	Load Rating = T.L. x $1.2\ (S/S_u)$

where

T.L. = support test load equal to less than the load under which the component support fails to perform its specified support function

F_{All} = allowable value for the type of stress in XVII-1100 of the ASME III Code

S = allowable stress value at the design temperature (NF- 3112.2) from the applicable table of Appendix I of the ASME III Code

S_u = specified minimum ultimate tensile strength of the material used in the support as given in the applicable table of Appendix I of the ASME III Code

Faulted Condition - Analyzed in accordance with Tables 3.9B-1D and 3.9B-1E.

2. Linear Supports

a. Normal

The allowable stresses of Appendix XVII, as referenced in subsection NF, of the ASME B&PV Code, Section III, are used for normal condition limits.

b. Upset

Stress limits for upset conditions are the same as normal condition stress limits. This is consistent with subsection NF of the ASME B&PV Code, Section III (see Subarticle NF-3230).

c. Emergency

For emergency conditions, the allowable stresses or load ratings are 33 percent higher than those specified for normal conditions. This is consistent with subsection NF of ASME B&PV Code, Section III (see Subarticle NF-3231.lb), in which limits for emergency conditions are 33 percent greater than the normal condition limits.

d. Faulted

The allowable stresses of NF-3231.1c subsection NF, of the ASME Code Section III are used for the faulted condition.

3. Plate and Shell Supports

a. Normal

Normal condition limits are those specified in subsection NF of the ASME B&PV Code, Section III (see Subarticle NF- 3320).

b. Upset

Limits for upset conditions equal normal condition limits and are consistent with subsection NF of the ASME B&PV Code, Section III (see Subarticle NF-3320).

c. Emergency

For emergency conditions, the allowable stresses or load ratings are 20 percent higher than those specified for normal conditions.

d. Faulted

Limits for faulted conditions are the same as those for linear supports.

For active Class 2 and Class 3 pumps, support adequacy is proven by satisfying the criteria in Subsection 3.9B.3.2. In general, active valves are supported only by the pipe attached to the valve. Exterior supports on the valve are generally not used.

4. Mechanical Snubbers

Mechanical snubbers are utilized in the Containment Building, Safeguards Building, Auxiliary Building and Fuel Building as indicated on Table 3.9B-11, and are used for both shock and/or vibration. The rated load of the various models used is provided in Table 3.9-12.

The snubber design specification establishes the design, performance, fabrication, inspection, testing and installation requirements for these mechanical snubbers. It is the intent of the specification to provide nuclear quality equipment and be in compliance with ASME Boiler and Pressure Code Section III, including Subsection NF.

Performance testing consists of calculating the breakaway friction force, lost motion, and an acceleration/load test. In addition, the snubbers are tested for operation for the faulted load.

The average of the minimum tension and compression spring constants is used as input to the program. Load conditions for snubber design is SSE seismic for all safety-related systems. When applicable, loads due to relief valve discharge, turbine trip and valve closure are included in the analysis. This information is available via computer printout upon completion of the code analysis.

3.9B.3.4.3 Instrument Impulse Tubing Supports for ASME III Class 2 and Class 3 Safety Related Applications

1. All instrument impulse tubing, valves and fittings connecting instruments to ASME III Class 2 and Class 3 piping root valves will be seismically supported.

2. All instrument impulse tubing, valves and fittings connecting nuclear safety related instruments to Non-ASME piping or ducting in seismic category I structures will be seismically supported.

3. The support design will consider gravity, seismic and thermal loading combinations and will be consistent with the design requirements of AISC.

4. Any welding of the support system will be per AWS specifications consistent with that performed on other nuclear safety related, Non-ASME supports.

5. Subsection NF supports will be employed on the ASME III main line piping including the instrument root valve.

6. The material used for fabrication of the tubing supports will be purchased with certificates of compliance to applicable ASTM standards.

7. Other seismically designed support systems such as cable tray supports, pipe supports or conduit supports may be used for tubing if paragraph c above is met. Any necessary re-analysis will be performed to justify the additional loads.

8. AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings" Nov. 1, 1978 including Supplement 1 March 11, 1986 may be utilized in the analysis of structural tubing.

9. The tubing stress analysis will conform to the loading combinations and stress limits of ASME Section III class 2 and 3 equations 8 thru 11.

3.9B.4 CONTROL ROD DRIVE SYSTEM (CRDS)

Refer to Section 3.9N.4

3.9B.5 REACTOR VESSEL INTERNALS

Refer to Section 3.9N.5

3.9B.6 INSERVICE TESTING OF PUMPS & VALVES

3.9B.6.1 Inservice Testing of Pumps

3.9B.6.1.1 Scope

Inservice testing of pumps shall be in accordance with the Inservice Testing Plan.

3.9B.6.1.2 Test Program

Establishment of reference values and a periodic testing schedule shall be in accordance with the Inservice Testing Plan.

3.9B.6.1.3 Test Frequencies and Durations

Test frequencies and durations shall be in accordance with the Inservice Testing Plan.

3.9B.6.1.4 Methods of Measurement

Methods of measurements shall be in accordance with the Inservice Testing Plan.

3.9B.6.2 Inservice Testing of Valves

3.9B.6.2.1 Scope

Inservice testing of valves shall be in accordance with the Inservice Testing Plan.

3.9B.6.2.2 Valve Test List

The scope of the valves tested and format for the valve tables are provided in the Inservice Testing Plan.

3.9B.6.2.3 Test Procedures

Valve test provisions in Technical Specifications shall meet any additional conditions of the Inservice Testing Plan.

REFERENCES

1. Swanson, John A., Engineering Analysis System, User's Manual (ANSYS), Swanson Analysis System, Inc., Elizabeth, Pa.

2. Dignwell, I.W., Static, Thermal, Dynamic Pipe Stress Analysis, Input Preparation IBM 1130 for Gibbs & Hill, Inc., (ADLPIPE), Arthur D. Little, Inc., Cambridge, Mass., Sept. 1972.

3. Greenstadt, J., "The Determination of the Characteristic Roots of a Matrix by Jacobi Method," Mathematical Methods for Digital Computers, John Wiley & Sons, Inc., New York, N.Y., 1959.

4. Documentation of ADLPIPE for Static and Dynamic Loads and Stress Evaluation, Arthur D. Little, Inc., Cambridge, Mass., Sept., 1973.

5. NEDO-21985, Functional Capability Criteria for Essential Mark II Piping, September 1978, prepared by Battelle Columbus Laboratories for General Electric Company.

6. NRC Evaluation of Topical Report, Piping Functional Capability Criteria, May 1980.

7. Stress Criteria for Demonstrating Functional Capability of ASME Class 2 and 3 Stainless Steel Elbows, Westinghouse Letter Number NS-LT-9447 TBX/TCS-4705, September 1981.

8. TUGCO Letter to NRC Proposing Stress Limits, TXX-3414, October 6, 1981.

9. U. S. Nuclear Regulatory Commission Evaluation of Request for Use of ASME Code Cases N-397 and N-411, March 13, 1985.

TABLE 3.9B-1A
DESIGN LOAD COMBINATIONS FOR ASME CODE CLASS 2 AND CLASS 3 COMPONENTS (EXCLUSIVE OF PIPING AND PIPING SUPPORTS)

Plant Conditions(1)	Loading Combinations
Normal	Design pressure + design temperature(2) + deadweight + thermal expansion + sustained loads + nozzle loads(3) + occasional loads, as applicable.
Upset	Design pressure + deadweight + thermal expansion + sustained loads + occasional loads + 0.5 SSE + transients + nozzle loads(3) + dynamic upset events(5)
Emergency	Design pressure + deadweight + sustained loads + thermal expansion + nozzle loads(3) + occasional load, as applicable
Faulted(4)	Design pressure + deadweight + sustained loads + SSE + pipe rupture and/or impingement effects where applicable + nozzle loads(3) + thermal expansion + occasional load + dynamic faulted events(5), as applicable.

Notes:

1. For active pumps the plant conditions listed are all considered normal component operating conditions and are specified as design conditions. For valves and inactive components, the plant conditions listed may be specified as component normal, upset, emergency or faulted conditions, depending on the function to be performed by the component.

2. Design temperature is used to determine allowable stress only.

3. Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration.

4. For the faulted condition. For the purpose of analysis, a pipe rupture shall be considered as required by section 3.6 of the FSAR for any pipe connected to the component. The results of the analysis shall demonstrate that the effects of a pipe rupture, if any, will not cause any other pipe connected to the component to rupture, or cause the component to move from its foundation.

5. Other dynamic events include relief valve discharge/waterhammer, etc.

TABLE 3.9B-1B
LOADING COMBINATION AND STRESS LIMITS FOR ASME III CLASS 2 & 3 PIPING

(Sheet 1 of 3)

Plant Condition	Loading Combination	Load Definition	Allowable Stresses (Paragraphs NC and ND-3650)	
Normal/ Testing	Design pressure deadweight	Sustained	S_h (eq 8)	$S_h + S_A$ (eq 11)
	Temperature (6) Thermal Anchor Movements Containment Displacements (10)	Thermal expansion Anchor Movements	S_A (eq 10)	
Upset	Design press. (3) Deadweight	Sustained	S_h (eq 8)	$S_h + S_h$ (eq 11) (5)
	1/2 SSE inertia 1/2 SSE anchor movement (2) Dynamic Upset Events (4)	Occasional	1.2 S_h (eq 9)	
	Temperature (6) Thermal Anchor Movements 1/2 SSE Anchor Movement (2)	Thermal expansion Anchor movements	S_A (eq 10)	
Emergency	Design pressure (3) Deadweight Jet impingement loads (7) Dynamic Emergency Events Pipe Impact Loads (7)	Sustained Occasional	1.8 S_h (eq 9)	

TABLE 3.9B-1B
LOADING COMBINATION AND STRESS LIMITS FOR ASME III CLASS 2 & 3 PIPING

(Sheet 2 of 3)

Plant Condition	Loading Combination	Load Definition	Allowable Stresses (Paragraphs NC and ND-3650)	
Faulted	Design pressure (3) Deadweight	Sustained		$S_h + S_A$ (8) (eq 11)
	SSE Inertia Jet impingement loads (7) Dynamic Faulted Events Pipe Impact Loads (7) LOCA	Occasional	2.4 S_h (eq 9)	
	Temperature (12) Thermal Anchor Movements Containment Displacements (11) SSE Anchor Movements	Thermal expansion Anchor Movements	S_A (eq 10) (8)	

Notes:

1. Based on ASME Code Case 1606-1 “Stress Criteria Section III, Class 2 & 3 piping subject to upset, emergency and faulted operating conditions.”
2. Anchor movement included in Equation (10) if omitted from Equation (9).
3. Design pressure is used since peak pressure and earthquake are not taken to be acting concurrently.
4. Dynamic upset events include: Relief Valve Actuation Steam/Water Hammer, etc.

TABLE 3.9B-1B
LOADING COMBINATION AND STRESS LIMITS FOR ASME III CLASS 2 & 3 PIPING

(Sheet 3 of 3)

Plant Condition	Loading Combination	Load Definition	Allowable Stresses (Paragraphs NC and ND-3650)

5. Does not include occasional loads. Equation (11) = Equation (8) + Equation (10).
6. Temperatures used correspond to Normal/Upset system operating conditions.
7. For essential piping only, as identified in Damage Study Analyses performed in accordance with FSAR Section 3.6B.
8. Applicable to those piping systems whose normal function is to prevent or mitigate the consequences of events associated with a plant faulted condition (i.e., containment spray and safety injection systems).
9. Deleted.
10. Containment displacements due to Preoperational Containment Structural Integrity Pressure Test and subsequent Containment Integrated Leak Rate Pressure Test.
11. Containment displacements due to pressure and temperature response of the containment during a faulted plant condition (LOCA).
12. Temperatures used to correspond to faulted plant conditions.

TABLE 3.9B-1C
LOADING COMBINATIONS ASME SECTION III CLASS 1, 2, AND 3 PIPING SUPPORTS

SYSTEM CONDITION	LOADING COMBINATION
Testing	Deadweight, Thermal, Containment Pressurization
Normal	Thermal Expansion Deadweight Equipment and Insulation Weights
Upset	Thermal Expansion Deadweight Equipment and Insulation Weights 1/2 SSE (Inertia and Anchor Movements) Plant Upset Dynamic Events (2)
Emergency	Thermal Expansion Deadweight Equipment and Insulation Weights Pipe Impact Loads (1) (2) Jet Impingement Loads (1) (2) Plant Emergency Dynamic Events (2)
Faulted	Thermal Expansion Deadweight Equipment and Insulation Weights 1.0 SSE (Inertia and Anchor Movements) Pipe Impact Loads (1) (2) Jet Impingement Loads (1) (2) Plant Faulted Dynamic Events (2) Containment Anchor Movements

Notes:

1. Not to be considered simultaneously

2. When applicable

TABLE 3.9B-1D
ALLOWABLE STRESSES ASME B&PV CODE SECTION III CLASS 1 PIPING SUPPORTS

		Allowable Stresses and Component Classification			
		Component Non-Standard Supports		Component Standard Supports	
Type Analysis	System Condition	Plates & Shell	Linear	Plate & Shell	Linear
	Design Normal Upset	Fig. NF-3221.1	NF-3231.1(a)	Fig. NF-3221.1	NF-3231.1(a)
Elastic	Emergency	Fig. NF-3221.1	NF-3231.1(b)	Fig. NF-3221.1	NF-3231.1(b)
	Faulted (4)	Appendix F (NA-2140)	NF-3231.1(c)	Appendix F (NA-2140)	NF-3231.1(c)
Load Rating	Normal Upset Emergency	NF-3262.2	NF-3262.3	NF-3262.4	NF-3262.4
	Faulted (4)	Note (2)	Note (2)	Note (2)	Note (2)

Notes:

1. Limit Analyses per Appendix XVII, Article 4000 may be used for elastic analysis of linear supports.
2. Guidelines not available. Faulted condition analysis to be performed with elastic analysis.
3. Loading combinations provided in Table 3.9B-1C.
4. Piping supports in systems whose normal function is to prevent or mitigate the consequences of events associated with a plant faulted condition (i.e., containment spray and safety injection systems) are designed within the limits specified for emergency condition.

TABLE 3.9B-1E
ALLOWABLE STRESSES ASME B&PV CODE SECTION III CLASS 2 AND 3 PIPING SUPPORTS

		Allowable Stresses and Component Classification			
		Component Non-Standard Supports		Component Standard Supports	
Type Analysis	System Condition	Plates & Shell	Linear	Plate & Shell	Linear
	Design Normal Upset	NF-3321.1 NF-3321.2(a) NF-3321.2(b)	NF-3231.1(a)	NF-3321.1 NF-3321.2(a) NF-3321.2(b)	NF-3231.1(a)
Elastic	Emergency	NF-3321.2(c)	NF-3231.1(b)	NF-3321.2(c)	NF-3231.1(b)
	Faulted (4)	NF-3321.2(d)	NF-3231.1(c)	NF-3321.2(d)	NF-3231.1(c)
Load Rating	Normal Upset Emergency	NF-3262.2	NF-3262.3	NF-3262.4	NF-3262.4
	Faulted (4)	Note (2)	Note (2)	Note (2)	Note (2)

Notes:

1. Limit Analyses per Appendix XVII, Article 4000 may be used for elastic analysis of linear supports.
2. Guidelines not available. Faulted condition analysis to be performed with elastic analysis.
3. Loading combinations provided in Table 3.9B-1C.
4. Piping supports in systems whose normal function is to prevent or mitigate the consequences of events associated with a plant faulted condition (i.e., containment spray and safety injection systems) are designed within the limits specified for emergency condition.

TABLE 3.9B-1F
USES OF CODE CASE N-318

(Sheet 1 of 3)

PIPE SUPPORT MARK NUMBER	TYPE OF WELD	NUMBER OF SIDES WELDED
CC-1-034-016-C82R	FILLET	THREE
CC-1-106-001-A43R	FILLET	THREE
CC-1-234-019-C53R	FILLET	TWO
CH-1-269-705-A43R	FILLET	TWO
CT-1-049-432-C82S	FILLET	TWO
CT-1-135-417-C72R	FILLET	TWO
CT-1-077-407-C72R	FILLET	TWO
CT-1-075-403-C72S	FILLET	TWO
CC-1-008-001-A43R	FILLET	THREE
CT-1-013-402-C52S	FILLET	THREE
SI-1-196-004-S32R	FILLET	THREE
CC-X-039-001-F43S	FILLET	THREE
CC-X-043-700-F43S	FILLET	THREE
CC-X-019-009-A43R	PARTIAL PENETRATION	THREE
CC-X-910-708-E23R	FILLET	TWO
CT-1-075-409-C62S	FILLET	TWO
VA-X-002-711-A75R	FILLET	THREE
CC-1-031-010-S43S	FILLET	FOUR (ALL ROUND)
CC-1-040-019-E33S	FILLET	THREE
CC-1-123-001-A43R	FILLET	THREE
FW-1-017-709-C72K	FILLET	THREE
SI-1-031-709-A32R	FILLET	THREE
CC-1-195-006-C42R	FILLET	THREE
SW-1-129-736-A43R	FILLET	THREE
SB-1-037-008-A45R	FILLET	THREE
AF-1-102-107-S33R	FILLET	THREE
FW-1-017-717-C52S	FILLET	THREE

TABLE 3.9B-1F
USES OF CODE CASE N-318

(Sheet 2 of 3)

PIPE SUPPORT MARK NUMBER	TYPE OF WELD	NUMBER OF SIDES WELDED
CT-1-013-405-C82S	FILLET	THREE
CC-1-050-701-A43R	PARTIAL PENETRATION	THREE
CT-1-074-411-C82R	FILLET	TWO
CT-1-039-443-C42K	FILLET	TWO
FW-2-018-441-C72R	FULL PENETRATION	TWO
AF-2-006-402-S33R	FILLET	THREE
CT-2-046-015-C92K	FILLET	THREE
CT-2-013-432-C52S	FILLET	THREE
CT-2-013-403-C72S	FILLET	THREE
CT-2-013-442-C82S	FILLET	THREE
CC-2-195-408-C52R	FILLET	TWO
CC-2-077-403-S43S	FILLET	TWO
SI-2-095-403-S42R	FILLET	TWO
H-FW-2-SB-023-003-5	FILLET	TWO
H-PS-2-RB-007-011-2	FILLET	TWO
H-MS-2-RB-011-005-2	FILLET	TWO
SI-2-051-415-C42K	FILLET	TWO
SI-2-051-421-C42R	FILLET	THREE
SI-2-095-415-C42R	FILLET	THREE
SI-2-087-416-C42K	FILLET	TWO
SI-2-306-404-C42R	FILLET	TWO
SI-2-306-416-C42R	FILLET	TWO
SI-2-306-420-C42R	FILLET	THREE
CS-2-001-454-C42S	FILLET	TWO
CS-2-079-408-C42S	FILLET	TWO
CS-2-079-409-C42S	FILLET	TWO

TABLE 3.9B-1F
USES OF CODE CASE N-318

(Sheet 3 of 3)

PIPE SUPPORT MARK NUMBER	TYPE OF WELD	NUMBER OF SIDES WELDED
CS-2-RB-022-703-2	FILLET	TWO
CS-2-079-431-C42K	PARTIAL PENETRATION	TWO
CS-2-RB-003-709-2	FILLET	TWO
PS-2-RB-024-003-2	FILLET	TWO
H-CH-2-SB-051-007-3	FILLET	THREE
H-CH-2-SB-051-008-3	FILLET	THREE
H-CH-2-SB-041-017-E	FILLET	THREE
H-CH-2-SB-042-010-3	FILLET	THREE
H-CH-2-SB-045-009-3	FILLET	THREE
H-CH-2-SB-039-001-3	FILLET	THREE
H-CH-2-SB-043-011-3	FILLET	TWO
H-CH-2-SB-044-006-3	FILLET	TWO
H-CH-2-SB-044-007-3	FILLET	TWO
H-CH-2-SB-046-003-3	FILLET	TWO
H-CH-2-SB-046-009-3	FILLET	TWO
H-PS-2-RB-015-007-2	FILLET	TWO
H-CH-2-SB-043-012-3	FILLET	TWO
H-CH-2-SB-044-003-3	FILLET	TWO
H-CH-2-SB-046-007-3	FILLET	TWO
H-PS-2-RB-010-003-2	FILLET	TWO
H-PS-2-RB-010-014-2	FILLET	TWO
H-PS-2-RB-010-016-2	FILLET	TWO
RC-2-097-403-C86S	FILLET	TWO
RC-2-115-417-C76S	FILLET	TWO
RC-2-115-419-C66K	FILLET	THREE
H-RC-2-SB-007-001-3	FILLET	TWO
H-RC-2-AB-001-016-3	FILLET	TWO

TABLE 3.9B-2
STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 VESSELS

Plant Condition[a]	Stress Limits
Normal	The vessel shall conform to the requirements of ASME B&PV Code, Section VIII, Division 1.
Upset	$\sigma_m \leq 1.1s \geq \frac{\sigma_m + \sigma_b}{1.5}$
Emergency	$\sigma_m \leq 1.1s \geq \frac{\sigma_m + \sigma_b}{1.5}$
Faulted	$\sigma_m \leq 1.5s \geq \frac{\sigma_m + \sigma_b}{1.5}$

a) See Note 1 on Table 3.9B-1A.

TABLE 3.9B-3
STRESS CRITERIA FOR ASME CODE CLASS 2 AND CLASS 3 INACTIVE PUMPS

Plant Condition[a]	Stress Limits
Normal	The pump shall conform to the requirement of ASME B&PV Code, Section III, NC-3400 or ND-3400.
Upset	$\sigma_m < 1.1\ S$
	$\sigma_m + \sigma_b < 1.65\ S$
Emergency	$\sigma_m < 1.1\ S$
	$\sigma_m + \sigma_b < 1.65\ S$
Faulted	$\sigma_m < 1.2\ S$
	$\sigma_m + \sigma_b < 1.8\ S$

a) See note 1 on Table 3.9B-1A.

TABLE 3.9B-4
DESIGN CRITERIA FOR ACTIVE PUMPS

Plant Condition[a]	Design Criteria
Normal	ASME B&PV Code, Section III, NC-3400 and ND-3400
Upset	$\sigma_m < 1.0\ S$
	$\sigma_m + \sigma_b < 1.5\ S$
Emergency	$\sigma_m < 1.0\ S$
	$\sigma_m + \sigma_b < 1.5\ S$
Faulted	$\sigma_m < 1.0\ S$
	$\sigma_m + \sigma_b < 1.5\ S$

a) See Note 1 on Table 3.9B-1A.

TABLE 3.9B-5
STRESS CRITERIA FOR SAFETY RELATED ASME CODE CLASS 2 AND 3 VALVES

Condition	Stress Limits (Notes 1-5)	P_{max}(Note 6)
Design & Normal	Valve bodies shall conform to the requirements of ASME Section III, NC-3500 (or ND-3500)	
Upset	$\sigma_m \leq 1.1\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b < 1.65\ S$	1.1
Emergency	$\sigma_m \leq 1.5\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80\ S$	1.2
Faulted	$\sigma_m \leq 2.0\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4\ S$	1.5

Notes:

1. Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied by calculation: 1) section modulus and area of a plane, normal to the flow, through the region of valve body crotch is at least 10% greater than the piping connected (or joined) to the valve body inlet and outlet nozzles; and, 2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material.

   If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above shall be multipled by the ratio (S_{pipe}/S_{valve}). If unable to comply with this requirement, the design by analysis procedure of NB-3545.2 is an acceptable alternate method.

2. Casting quality factor of 1.0 shall be used.

3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.

5. These rules do not apply to Class 2 and 3 safety relief valves. Safety relief valves will be designed in accordance with ASME Section III requirements.

6. The maximum pressure resulting from upset, emergency or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature.

TABLE 3.9B-6
THIS TABLE IS INTENTIONALLY NOT USED

TABLE 3.9B-7
THIS TABLE IS INTENTIONALLY NOT USED

TABLE 3.9B-8
ACTIVE PUMPS

(Sheet 1 of 2)

PUMP	TAG NUMBER	SYSTEM	CLASS	NORMAL MODE	ACCIDENT MODE	ACTIVE FUNCTION
Auxiliary Feedwater (Motor Driven)	CP1&CP2-AFAPMD-01, 02	AF	3	ON/OFF	ON	Required for Removing Reactor Decay Heat (Safe Hot Shutdown)
Auxiliary Feedwater (Turbine Driven)	CP1&CP2-AFAPTD-01	AF	3	OFF	ON	Required for Removing Reactor Decay Heat (Safe Hot Shutdown)
Service Water	CP1&CP2-SWAPSW-01, 02	SW	3	ON/OFF	ON	Required for Transferring Heat from Primary Plant Safeguards Components to the Ultimate Heat Sink
Component Cooling Water	CP1&CP2-CCAPCC-01, 02	CC	3	ON/OFF	ON	Required for Transferring Heat from Components to the Service Water System
Containment Spray	CP1&CP2-CTAPCS-01, 02, 03, 04	CT	2	OFF	ON	Required for Containment Heat Removal
Spent Fuel Pool	CPX-SFAPSF-01, 02	SF	3	ON/OFF	ON	Required for Cooling of Spent Fuel
Reactor Makeup Water	CP1&CP2-DDAPRM-01	DD	3	ON/OFF	ON/OFF	Required to Provide Seismic Category I Makeup for Spent Fuel Pool, Component Cooling Water, and Safety Chilled Water

TABLE 3.9B-8
ACTIVE PUMPS

(Sheet 2 of 2)

PUMP	TAG NUMBER	SYSTEM	CLASS	NORMAL MODE	ACCIDENT MODE	ACTIVE FUNCTION
Reactor Makeup Water	CPX-DDAPRM-01	DD	3	ON/OFF	ON/OFF	Required to Provide Seismic Category I Makeup for Spent Fuel Pool, Component Cooling Water, and Safety Chilled Water
Safety Chilled Water	CP1&CP2-CHAPCP-05, 06	CH	3	ON/OFF	ON	Required to Transfer the Heat Rejected by Engineering Safety Feature Pump Motors and Electrical Switchgear to CCW
Diesel Generator Fuel Oil Transfer Pump	CP1&CP2-DOAPFT-01, 02, 03, 04	DO	3	OFF	ON/OFF	Required to Transfer Fuel Oil to Day Tank from the Storage Tank
Safeguard Bldg Floor Drain Sump Pumps	CP1&CP2-WPAPSS-01, 02, 03, 04	VD	3	ON/OFF	ON/OFF	Required to Detect and Mitigate passive failures in the ECCS and Containment Spray System post-LOCA and to prevent Flooding of Safety Related Equipment

TABLE 3.9B-9
BENCHMARK CALCULATIONS FOR ADLPIPE COMPUTER PROGRAM

Type of Analysis	Checks	Reference
Thermal and deadweight combined	Forces, moments, and deflections through the system	Pressure Vessel and Piping/1972 Computer Program Verification ASME Chapter 6, p. 1
Dynamic	Natural frequencies of a three-dimensional structure; mode shapes are checked (not published)	Pressure Vessel and Piping/1972 Computer Program Verification ASME Chapter 6, p. 1
Stress and usage factor	Stress range calculation and fatigue usage factor	Sample Analysis of a Piping System-Class 1 Nuclear, ASME
Thermal and deadweight (separate)	Forces, defelctions, stresses in accordance with ANSI B31.1	Stress in Three Dimensional Pipe Bends by W. Hovgaard Trans. ASME, Volume 57, 1935 pp. 401-456
Thermal	Thermal stress in accordance with ANSI B31.1, anchor reactions	Design of Piping System, M. W. Kellogg Company, p. 47
Dynamic	Natural frequencies, mode shapes, and response spectra, deflections and moments	Shock and Vibration by Young, ASME
Stress	All stress coefficients (product of stress indexes and geometry) used in Section III Class 1 Piping Analysis, that is specified in footnotes to stress indexes	Hand calculations
Stress	All stress components and their sum on selected piping	Hand calculations

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 1 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
HV-2333A	MS	Globe/Gas-Hydraulic	32	2	Auto Trip	Open	Steam Line Isolation, Containment Isolation
HV-2333B	MS	Globe/Handwheel	4	2	Local Manual	Closed	Steam Line Isolation, Containment Isolation
HV-2409	MS	Globe/Air	2	2	Auto Trip	Open	Steam Line Isolation, Containment Isolation
HV-2452-1	MS	Globe/Air	4	2	Auto Trip/ Remote Manual	Closed	Turbine Driven AFW Pump Steam Supply, Containment Isolation
PV-2325	MS	Globe/Air	8	2	Remote Manual	Closed	Containment Isolation, Cooldown
HV-2334A	MS	Globe/Gas-Hydraulic	32	2	Auto Trip	Open	Steam Line Isolation, Containment Isolation
HV-2334B	MS	Globe/Handwheel	4	2	Local Manual	Closed	Steam Line Isolation, Containment Isolation
HV-2410	MS	Globe/Air	2	2	Auto Trip	Open	Steam Line Isolation, Containment Isolation
PV-2326	MS	Globe/Air	8	2	Remote Manual	Closed	Containment Isolation, Cooldown
HV-2335A	MS	Globe/Gas-Hydraulic	32	2	Auto Trip	Open	Steam Line Isolation, Containment Isolation
HV-2335B	MS	Globe/Handwheel	4	2	Local Manual	Closed	Steam Line Isolation, Containment Isolation
HV-2411	MS	Globe/Air	2	2	Auto Trip	Open	Steam Line Isolation, Containment Isolation
MS-0101	MS	Gate/Handwheel	4	2	Local Manual	Open	TDAFWP Steam Supply, Containment Isolation
MS-0128	MS	Gate/Handwheel	4	2	Local Manual	Open	TDAFWP Steam Supply Containment Isolation
PV-2327	MS	Globe/Air	8	2	Remote Manual	Closed	Containment Isolation, Cooldown
HV-2336A	MS	Globe/Gas-Hydraulic	32	2	Auto Trip	Open	Steam Line Isolation, Containment Isolation
HV-2336B	MS	Globe/Handwheel	4	2	Local Manual	Closed	Steam Line Isolation, Containment Isolation
HV-2412	MS	Globe/Air	2	2	Auto Trip	Open	Steam Line Isolation, Containment Isolation
HV-2452-2	MS	Globe/Air	4	2	Auto Trip/ Remote Manual	Closed	Turbine Driven AFW Pump Steam Supply, Containment Isolation
PV-2328	MS	Globe/Air	8	2	Remote Manual	Closed	Containment Isolation Cooldown
HV-2397	MS	Globe/Air	3	2	Auto Trip	Open	Auxiliary Feedwater System Actuation, Containment Isolation & Steam Generator Isolation

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 2 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
HV-2398	MS	Globe/Air	3	2	Auto Trip	Open	Auxiliary Feedwater System Actuation, Containment Isolation & Steam Generator Isolation
HV-2399	MS	Globe/Air	3	2	Auto Trip	Open	Auxiliary Feedwater System Actuation, Containment Isolation & Steam Generator Isolation
HV-2400	MS	Globe/Air	3	2	Auto Trip	Open	Auxiliary Feedwater System Actuation, Containment Isolation & Steam Generator Isolation
HV-2401A/B	MS	Globe/Air	3/4	2	Auto Trip	Open	Auxiliary Feedwater System Actuation
HV-2402A/B	MS	Globe/Air	3/4	2	Auto Trip	Open	Auxiliary Feedwater System Actuation
HV-2403A/B	MS	Globe/Air	3/4	2	Auto Trip	Open	Auxiliary Feedwater System Actuation
HV-2404A/B	MS	Globe/Air	3/4	2	Auto Trip	Open	Auxiliary Feedwater System Actuation
HV-2405	MS	Globe/Air	3/4	2	Auto Trip	Open	Containment Isolation
HV-2406	MS	Globe/Air	3/4	2	Auto Trip	Open	Containment Isolation
HV-2407	MS	Globe/Air	3/4	2	Auto Trip	Open	Containment Isolation
HV-2408	MS	Globe/Air	3/4	2	Auto Trip	Open	Containment Isolation
MS-0021	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, Decay Heat Removal, CODE Safety Valve
MS-0022	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation CODE Safety Valve
MS-0023	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0024	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0025	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0026	MS	Gate/Gear/Extension Stem	8	2	Local Manual	Open	Atmospheric Relief Valve Isolation
MS-0058	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, Decay Heat Removal, CODE Safety Valve
MS-0059	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Value

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 3 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
MS-0060	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0061	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0062	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0063	MS	Gate/Gear/Extension Stem	8	2	Local Manual	Open	Atmospheric Relief Valve Isolation
MS-0093	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, Decay Heat Removal, CODE Safety Valve
MS-0094	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0095	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0096	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0097	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0098	MS	Gate/Gear/Extension Stem	8	2	Local Manual	Open	Atmospheric Relief Valve Isolation
MS-0129	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, Decay Heat Removal, CODE Safety Valve
MS-0130	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0131	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0132	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0133	MS	Safety	6	2	Self-Actuated	Closed	Steam Line Isolation, CODE Safety Valve
MS-0134	MS	Gate/Gear/Extension Stem	8	2	Local Manual	Open	Atmospheric Relief Valve Isolation
MS-0142	MS	Check	4	3	Self-Actuated	Closed	Turbine Driven AFW Pump Steam Supply Path and Boundary
MS-0143	MS	Check	4	3	Self-Actuated	Closed	Turbine Driven AFW Pump Steam Supply Path and Boundary

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 4 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
1MS-0680 (2MS-0664)	MS	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply To S.G. PORV After Loss of Instrument Air
1MS-0681 (2MS-0665)	MS	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply To S.G. PORV After Loss of Instrument Air
1MS-0682 (2MS-0666)	MS	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply To S.G. PORV After Loss of Instrument Air
1MS-0683 (2MS-0665)	MS	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply To S.G. PORV After Loss of Instrument Air
1MS-0684 (2MS-0668)	MS	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply To S.G. PORV After Loss of Instrument Air
1MS-0685 (2MS-0667)	MS	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply To S.G. PORV After Loss of Instrument Air
1MS-0686 (2MS-0670)	MS	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply To S.G. PORV After Loss of Instrument Air
1MS-0687 (2MS-0669)	MS	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply To S.G. PORV After Loss of Instrument Air
HV-2397A	MS	Globe/Air	3	2	Auto Trip	Open	Steam Generator Isolation
HV-2398A	MS	Globe/Air	3	2	Auto Trip	Open	Steam Generator Isolation
HV-2399A	MS	Globe/Air	3	2	Auto Trip	Open	Steam Generator Isolation
HV-2400A	MS	Globe/Air	3	2	Auto Trip	Open	Steam Generator Isolation
HV-4075B/C	FP	Gate/Motor	4	2	Auto Trip	Closed	Containment Isolation
HV-2134	FW	Gate/Gas-Hydraulic	18	2	Auto Trip	Open	Feedwater Isolation, Containment Isolation
HV-2135	FW	Gate/Gas-Hydraulic	18	2	Auto Trip	Open	Feedwater Isolation, Containment Isolation
HV-2136	FW	Gate/Gas-Hydraulic	18	2	Auto Trip	Open	Feedwater Isolation Containment Isolation
HV-2137	FW	Gate/Gas-Hydraulic	18	2	Auto Trip	Open	Feedwater Isolation, Containment Isolation
HV-2185	FW	Globe/Air	3	2	Auto Close	Closed	Feedwater Isolation, Containment Isolation

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 5 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
HV-2186	FW	Globe/Air	3	2	Auto Close	Closed	Feedwater Isolation, Containment Isolation
HV-2187	FW	Globe/Air	3	2	Auto Close	Closed	Feedwater Isolation Containment Isolation
HV-2188	FW	Globe/Air	3	2	Auto Close	Closed	Feedwater Isolation, Containment Isolation
2-FV-2193	FW	Globe/Air	6	2	Auto Close	Closed	Feedwater Isolation, Containment Isolation
2-FV-2194	FW	Globe/Air	6	2	Auto Close	Closed	Feedwater Isolation, Containment Isolation
2-FV-2195	FW	Globe/Air	6	2	Auto Close	Closed	Feedwater Isolation, Containment Isolation
2-FV-2196	FW	Globe/Air	6	2	Auto Close	Closed	Feedwater Isolation, Containment Isolation
FW-0070	FW	Check	18	2	Self-Actuated	Open	Feedwater Isolation
FW-0076	FW	Check	18	2	Self-Actuated	Open	Feedwater Isolation
FW-0082	FW	Check	18	2	Self-Actuated	Open	Feedwater Isolation
FW-0088	FW	Check	18	2	Self-Actuated	Open	Feedwater Isolation
FW-0195	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
FW-0196	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
FW-0197	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
FW-0198	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
1-FW-0199	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
1-FW-0200	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
1-FW-0201	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
1-FW-0202	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
2-FW-0199	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
2-FW-0200	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
2-FW-0201	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below
2-FW-0202	FW	Check	6	2	Self-Actuated	Open/Closed	See Note 9, below

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 6 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
2-FW-0191	FW	Check	6	2	Self-Actuated	Closed	AFW Flow Path Boundary
2-FW-0192	FW	Check	6	2	Self-Actuated	Closed	AFW Flow Path Boundary
2-FW-0193	FW	Check	6	2	Self-Actuated	Closed	AFW Flow Path Boundary
2-FW-0194	FW	Check	6	2	Self-Actuated	Closed	AFW Flow Path Boundary
HV-2491A/B	AF	Gate/Motor	4	2	Remote Manual	Open	Containment Isolation
2-FV-2181	FW	Butterfly/Air	6	2	Auto Trip	Open	AFW Flow Path Boundary
2-FV-2182	FW	Butterfly/Air	6	2	Auto Trip	Open	AFW Flow Path Boundary
2-FV-2183	FW	Butterfly/Air	6	2	Auto Trip	Open	AFW Flow Path Boundary
2-FV-2184	FW	Butterfly/Air	6	2	Auto Trip	Open	AFW Flow Path Boundary
HV-2492A/B	AF	Gate Motor	4	2	Remote Manual	Open	Containment Isolation
HV-2493A/B	AF	Gate Motor	4	2	Remote Manual	Open	Containment Isolation
HV-2494A/B	AF	Gate Motor	4	2	Remote Manual	Open	Containment Isolation
HV-2480	AF	Gate Motor	6	3	Remote Manual	Closed	SW Flow Path to Suction
HV-2481	AF	Gate Motor	6	3	Remote Manual	Closed	SW Flow Path to Suction
HV-2482	AF	Gate Motor	8	3	Remote Manual	Closed	SW Flow Path to Suction
FV-2456	AF	Globe/Air	2	3	Auto Trip	Open	Recirculation Flow Path/ Closes to ensure adequate flow for accident mitigation
FV-2457	AF	Globe/Air	2	3	Auto Trip	Open	Recirculation Flow Path/ Closes to ensure adequate flow for accident mitigation
AF-0009	AF	Check	3	3	Self-Actuated	Open	Condensate Storage Tank Safety Class Isolation
AF-0350	AF	Check	2	3	Self-Actuated	Open	Condensate Storage Tank Safety Class Isolation
AF-0351	AF	Check	3	3	Self-Actuated	Closed	Condensate Storage Tank Safety Class Isolation
AF-0014	AF	Check	6	3	Self-Actuated	Closed	Suction from Condensate Storage Tank
AF-0024	AF	Check	6	3	Self-Actuated	Closed	Suction from Condensate Storage Tank

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 7 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
AF-0032	AF	Check	8	3	Self-Actuated	Closed	Suction from Condensate Storage Tank
AF-0065	AF	Check	6	3	Self-Actuated	Closed	Discharge Flow Path
AF-0051	AF	Check	6	3	Self-Actuated	Closed	Discharge Flow Path
AF-0038	AF	Check	8	3	Self-Actuated	Closed	Discharge Flow Path
AF-0093	AF	Check	4	3	Self-Actuated	Closed	Discharge Flow Path
AF-0098	AF	Check	4	3	Self-Actuated	Closed	Discharge Flow Path
AF-0083	AF	Check	4	3	Self-Actuated	Closed	Discharge Flow Path
AF-0086	AF	Check	4	3	Self-Actuated	Closed	Discharge Flow Path
AF-0075	AF	Check	4	3	Self-Actuated	Closed	Discharge Flow Path
AF-0078	AF	Check	4	3	Self-Actuated	Closed	Discharge Flow Path
AF-0101	AF	Check	4	3	Self-Actuated	Closed	Discharge Flow Path
AF-0106	AF	Check	4	3	Self-Actuated	Closed	Discharge Flow Path
1AF-0215 (2AF-0291)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0216 (2AF-0236)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0217 (2AF-0237)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0218 (2AF-0238)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0219 (2AF-0239)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0220 (2AF-0240)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
AF-0221	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 8 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
AF-0222	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0223 (2AF-0224)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0224 (2AF-0223)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0226 (2AF-0227)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0227 (2AF-0226)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
AF-0228	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
AF-0229	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0230 (2AF-0231)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
1AF-0231 (2AF-0230)	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Aux. F. W. Reg Valve After Loss of Instrument Air
AF-0232	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to AFW Turbine Isolation Valve After Loss of Instrument Air
AF-0233	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to AFW Turbine Isolation Valve After Loss of Instrument Air
AF-0234	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to AFW Turbine Isolation Valve After Loss of Instrument Air
AF-0235	AF	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to AFW Turbine Isolation Valve after Loss of Instrument Air
PV-2453A/B	AF	Globe/Air/Handwheel	3	3	Remote Manual/ Local Manual	Open	AFW Flow Path, Steam Generator Isolation, Manually Modulated for Flow Control

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 9 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
PV-2454A/B	AF	Globe/Air/Handwheel	3	3	Remote Manual/ Local Manual	Open	AFW Flow Path, Steam Generator Isolation, Manually Modulated for Flow Control
HV-2459	AF	Globe/Air/Handwheel	3	3	Remote Manual/ Local Manual	Open	AFW Flow Path, Steam Generator Isolation, Manually Modulated for Flow Control
HV-2460	AF	Globe/Air/Handwheel	3	3	Remote Manual/ Local Manual	Open	AFW Flow Path, Steam Generator Isolation, Manually Modulated for Flow Control
HV-2461	AF	Globe/Air/Handwheel	3	3	Remote Manual/ Local Manual	Open	AFW Flow Path, Steam Generator Isolation, Manually Modulated for Flow Control
HV-2462	AF	Globe/Air/Handwheel	3	3	Remote Manual/ Local Manual	Open	AFW Flow Path, Steam Generator Isolation, Manually Modulated for Flow Control
HV-2484	AF	Butterfly/Motor	12	3	Auto Trip	Open	Condensate Storage Tank Isolation
HV-2485	AF	Butterfly/Motor	12	3	Auto Trip	Open	Condensate Storage Tank Isolation
AF-0167	AF	Check	8	3	Self-Actuated	Closed	Pump Recirculation Flow Path
HV-4777	CT	Gate/Motor	16	2	Auto Trip	Closed	Discharge Path Flow, Containment Isolation
CT-0145	CT	Check	16	2	Self-Actuated	Closed	Discharge Path Flow, Containment Isolation
HV-4776	CT	Gate/Motor	16	2	Auto Trip	Closed	Discharge Path Flow, Containment Isolation
CT-0142	CT	Check	16	2	Self-Actuated	Closed	Discharge Path Flow, Containment Isolation
HV-4782	CT	Gate/Motor	16	2	Remote Manual	Closed	Recirculation Flow Path, Containment Isolation
HV-4783	CT	Gate/Motor	16	2	Remote Manual	Closed	Recirculation Flow Path, Containment Isolation
CT-0309	CT	Relief	3/4	2	Self-Actuated	Closed	Thermal Relief, Containment Isolation
CT-0310	CT	Relief	3/4	2	Self-Actuated	Closed	Thermal Relief, Containment Isolation
CT-0047	CT	Check	4	2	Self-Actuated	Closed	Pump Recirculation Flow Path
CT-0048	CT	Check	4	2	Self-Actuated	Closed	Pump Recirculation Flow Path
CT-0063	CT	Check	4	2	Self-Actuated	Closed	Pump Recirculation Flow Path
CT-0064	CT	Check	4	2	Self-Actuated	Closed	Pump Recirculation Flow Path

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 10 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
HV-4758	CT	Butterfly/Motor	16	2	Remote Manual	Open	CT Flow Path from RWST, Train A, Recirculation Boundary
HV-4759	CT	Butterfly/Motor	16	2	Remote Manual	Open	CT Flow Path from RWST, Train B, Recirculation Boundary
LV-4755	CT	Diaphragm/Motor	3	3	Auto Trip	Closed	Chemical Additive Flow to Suction Train B
LV-4754	CT	Diaphragm/Motor	3	3	Auto Trip	Closed	Chemical Additive Flow to Suction Train A
CT-0072	CT	Check	2	2	Self-Actuated	Closed	Chemical Additive Flowpath Train A
CT-0082	CT	Check	2	2	Self-Actuated	Closed	Chemical Additive Flowpath Train A
CT-0020	CT	Check	2	2	Self-Actuated	Closed	Chemical Additive Flowpath Train B
CT-0031	CT	Check	2	2	Self-Actuated	Closed	Chemical Additive Flowpath Train B
CT-0065	CT	Check	10	2	Self-Actuated	Closed	Containment Spray Discharge Flow, Train A
CT-0094	CT	Check	10	2	Self-Actuated	Closed	Containment Spray Discharge Flow, Train A
CT-0013	CT	Check	10	2	Self-Actuated	Closed	Containment Spray Pump Discharge Flow, Train B
CT-0042	CT	Check	10	2	Self-Actuated	Closed	Containment Spray Pump Discharge Flow, Train B
CT-0025	CT	Check	16	2	Self-Actuated	Closed	Containment Spray Pump Suction Flow, RWST Isolation, Train B
CT-0148	CT	Check	16	2	Self-Actuated	Closed	Containment Spray Pump Suction Flow Path, Train B
CT-0077	CT	Check	16	2	Self-Actuated	Closed	Containment Spray Pump Suction Flow, RWST Isolation, Train A
CT-0149	CT	Check	16	2	Self-Actuated	Closed	Containment Spray Pump Suction Flow Path, Train A
FV-4772-1	CT	Globe/Motor	4	2	Auto Trip	Open	Pump Recirculation Flow Path
FV-4772-2	CT	Globe/Motor	4	2	Auto Trip	Open	Pump Recirculation Flow Path
FV-4773-1	CT	Globe/Motor	4	2	Auto Trip	Open	Pump Recirculation Flow Path
FV-4773-2	CT	Globe/Motor	4	2	Auto Trip	Open	Pump Recirculation Flow Path
CP1/CP2-CTVBCA-01	CT	Vacuum Bkr	2	3	Self-Actuated	Closed	Vacuum relief (CAT)

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 11 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
CP1/CP2-CTVBCA-02	CT	Vacuum Bkr	2	3	Self-Actuated	Closed	Vacuum relief (CAT)
HV-4710	CC	Globe/Air	4	2	Auto Trip	Open	Containment Isolation
HV-4711	CC	Globe/Air	4	2	Auto Trip	Open	Containment Isolation
HV-4725	CC	Globe/Air	2	2	Auto Trip	Open	Containment Isolation
HV-4726	CC	Globe/Air	2	2	Auto Trip	Open	Containment Isolation
1CC-1067 (2CC-1090)	CC	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief, Containment Isolation
CC-0611	CC	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief During LOCA
CC-0618	CC	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief During LOCA
HV-4708	CC	Gate/Motor	8	2	Auto Trip	Open	Containment Isolation
HV-4701	CC	Gate/Motor	8	2	Auto Trip	Open	Containment Isolation
CC-0629	CC	Check	2	2	Self-Actuated	Closed	Containment Isolation
HV-4700	CC	Gate/Motor	8	2	Auto Trip	Open	Containment Isolation
CC-0713	CC	Check	8	2	Self-Actuated	Open	Containment Isolation
HV-4699	CC	Gate/Motor	8	2	Auto Trip	Open	CCW System Isolation
HV-4709	CC	Gate/Motor	4	2	Auto Trip	Open	Containment Isolation
HV-4696	CC	Gate/Motor	4	2	Auto Trip	Open	Containment Isolation
CC-0831	CC	Check	1	2	Self-Actuated	Closed	Containment Isolation
HV-4514	CC	Butterfly/Motor	24	3	Auto Trip	Open	CCW Loop Isolation, Train A
HV-4515	CC	Butterfly/Motor	24	3	Auto Trip	Open	CCW Loop Isolation, Train B
HV-4572	CC	Butterfly/Motor	18	3	Auto Trip	Closed	CCW Flow Path RHR Loop, Train A
CC-0109	CC	Butterfly/Handwheel	18	3	Local Manual	Closed	CCW Flow Path RHR Loop, Train A
HV-4573	CC	Butterfly/Motor	18	3	Auto Trip	Closed	CCW Flow Path RHR Loop, Train B
CC-0157	CC	Butterfly/Handwheel	18	3	Local Manual	Closed	CCW Flow Path RHR Loop, Train B

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 12 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
HV-4574	CC	Butterfly/Motor	18	3	Auto Trip	Closed	CCW Flow Path Containment Spray Loop, Train A
HV-4575	CC	Butterfly/Motor	18	3	Auto Trip	Closed	CCW Flow Path Containment Spray Loop, Train B
HV-4512	CC	Butterfly/Motor	24	3	Auto Trip	Open	CCW Loop Isolation, Train A
HV-4513	CC	Butterfly/Motor	24	3	Auto Trip	Open	CCW Loop Isolation, Train B
HV-4631A	CC	Globe/Air	2	3	Auto Trip	Open	Primary Sampling System Loop Isolation
HV-4631B	CC	Globe/Air	2	3	Auto Trip	Open	Primary Sampling System Loop Isolation
FV-4536	CC	Butterfly/Air	10	3	Auto Trip	Open	Recirculation Loop Isolation
FV-4537	CC	Butterfly/Air	10	3	Auto Trip	Open	Recirculation Loop Isolation
HV-4524	CC	Butterfly/Motor	24	3	Auto Trip	Open	CCW Non-safeguard Loop Isolation
HV-4525	CC	Butterfly/Motor	24	3	Auto Trip	Open	CCW Non-safeguard Loop Isolation
HV-4526	CC	Butterfly/Motor	24	3	Auto Trip	Open	CCW Non-safeguard Loop Isolation
HV-4527	CC	Butterfly/Motor	24	3	Auto Trip	Open	CCW Non-safeguard Loop Isolation
LV-4500	CC	Globe/Air/Handwheel	3	3	Note 8	Closed	Emergency Makeup Water Path
LV-4500-01	CC	Globe/Air	3	3	Note 8	Open/Closed	Emergency Makeup Water Path
LV-4501	CC	Globe/Air/Handwheel	3	3	Note 8	Closed	Emergency Makeup Water Path
CC-0003	CC	Check	3	3	Self-Actuated	Closed	Emergency Makeup Water Path
CC-0004	CC	Check	3	3	Self-Actuated	Open	Emergency Makeup Water Path Boundary
CC-0031	CC	Check	24	3	Self-Actuated	Open	Discharge Flow Path
CC-0061	CC	Check	24	3	Self-Actuated	Open	Discharge Flow Path
FV-4650A	CC	Butterfly/Air	10	3	Auto Trip	Open	Ventilation Chiller Isolation
FV-4650B	CC	Butterfly/Air	10	3	Auto Trip	Open	Ventilation Chiller Isolation
X-PV-3583	CC	Plug/Motor	3	3	Auto Pressure Control	Open	Control Room A/C Condensor CCW Regulating Valve
X-PV-3584	CC	Plug/Motor	3	3	Auto Pressure Control	Open	Control Room A/C Condensor CCW Regulating Valve

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 13 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
X-PV-3585	CC	Plug/Motor	3	3	Auto Pressure Control	Open	Control Room A/C Condensor CCW Regulating Valve
X-PV-3586	CC	Plug/Motor	3	3	Auto Pressure Control	Open	Control Room A/C Condensor CCW Regulating Valve
PV-4552	CC	Ball/Air/Handwheel	3	3	Auto/Local Manual	Open	Safety Chiller Condenser CCW regulating valve
PV-4553	CC	Ball/Air/Handwheel	3	3	Auto/Local Manual	Open	Safety Chiller Condenser CCW regulating valve
X-PCV-H116A	CC	Plug/Air	1	3	Auto Pressure Control	Open	UPS A/C Condenser CCW Regulating Valve
X-PCV-H116B	CC	Plug/Air	1	3	Auto Pressure Control	Open	UPS A/C Condenser CCW Regulating Valve
CC-0646	CC	Check	2	3	Self-Actuated	Open	RCP Thermal Barrier Hx Back Flow Prevention
CC-0657	CC	Check	2	3	Self-Actuated	Open	RCP Thermal Barrier Hx Back Flow Prevention
CC-0687	CC	Check	2	3	Self-Actuated	Open	RCP Thermal Barrier Hx Back Flow Prevention
CC-0694	CC	Check	2	3	Self-Actuated	Open	RCP Thermal Barrier Hx Back Flow Prevention
1CC-1075 (2CC-0371)	CC	Check	2	3	Self-Actuated	Open	RCP Thermal Barrier Hx Back Flow Prevention
1CC-1076 (2CC-0372)	CC	Check	2	3	Self-Actuated	Open	RCP Thermal Barrier Hx Back Flow Prevention
1CC-1077 (2CC-0373)	CC	Check	2	3	Self-Actuated	Open	RCP Thermal Barrier Hx Back Flow Prevention
1CC-1078 (2CC-0374)	CC	Check	2	3	Self-Actuated	Open	RCP Thermal Barrier Hx Back Flow Prevention
1CC-1079 (2CC-1091)	CC	Check	1/2	3	Self-Actuated	Closed	Inst. Air Safety Class Isolation
1CC-1080 (2CC-1092)	CC	Check	1/2	3	Self-Actuated	Open	Inst. Air Safety Class Isolation
1CC-1081 (2CC-1093)	CC	Check	1/2	3	Self-Actuated	Open	Inst. Air Safety Class Isolation
1CC-1082 (2CC-1094)	CC	Check	1/2	3	Self-Actuated	Open	Inst. Air Safety Class Isolation
HV-4286	SW	Butterfly/Motor	24	3	Auto Trip	Open	SW Flow Path, Train A
HV-4287	SW	Butterfly/Motor	24	3	Auto Trip	Open	SW Flow Path, Train B
HV-4393	SW	Butterfly/Motor	10	3	Auto Trip	Open	SW Flow Path, Train A
HV-4394	SW	Butterfly/Motor	10	3	Auto Trip	Open	SW Flow Path, Train B
HV-4395	SW	Butterfly/Motor	10	3	Remote Manual	Closed	Alternate AFW Flow Path, Train A

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 14 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
HV-4396	SW	Butterfly/Motor	10	3	Remote Manual	Closed	Alternate AFW Flow Path, Train B
SW-0373	SW	Check	24	3	Self-Actuated	Open	Discharge Flow Path
SW-0374	SW	Check	24	3	Self-Actuated	Open	Discharge Flow Path
CP1(CP2)-SWVAVB-01	SW	Vacuum Relief	2	3	Self-Actuated	Closed	Mitigates Water Hammer
CP1(CP2)-SWVAVB-02	SW	Vacuum Relief	2	3	Self-Actuated	Closed	Mitigates Water Hammer
CP1(CP2)-SWVAVB-03	SW	Vacuum Relief	1	3	Self-Actuated	Closed	Mitigates Water Hammer
CP1(CP2)-SWVAVB-04	SW	Vacuum Relief	1	3	Self-Actuated	Closed	Mitigates Water Hammer
XSF-0003	SF	Check	10	3	Self-Actuated	Open	Discharge Flow Path
XSF-0004	SF	Check	10	3	Self-Actuated	Open	Discharge Flow Path
XSF-0160	SF	Check	3	3	Self-Actuated	Closed	Makeup Path
XSF-0161	SF	Diaphram/Handwheel	3	3	Local Manual	Closed	Spent Fuel Pool Makeup
XSF-0179	SF	Diaphram/Handwheel	3	3	Local Manual	Closed	Spent Fuel Pool Makeup
XSF-0180	SF	Check	3	3	Self-Actuated	Closed	Makeup Path
SI-0166	SI	Check	3/4	3	Self-Actuated	Closed/Open	Close on Non-Safety Line Break to maintain Accumulator nitrogen pressure.
SI-0167	SI	Check	3/4	3	Self-Actuated	Closed/Open	Close on Non-Safety Line Break to maintain Accumulator nitrogen pressure.
SI-0168	SI	Check	3/4	3	Self-Actuated	Closed/Open	Close on Non-Safety Line Break to maintain Accumulator nitrogen pressure.
SI-0169	SI	Check	3/4	3	Self-Actuated	Closed/Open	Close on Non-Safety Line Break to maintain Accumulator nitrogen pressure.
HV-5365	DD	Globe/Air	3	2	Auto Trip	Closed	Containment Isolation
HV-5366	DD	Globe/Air	3	2	Auto Trip	Closed	Containment Isolation
DD-0006	DD	Check	3	3	Self-Actuated	Open	Safety Class Isolation
1DD-0064 (2DD-0008)	DD	Check	2	3	Self-Actuated	Open	Safety Class Isolation

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 15 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
1DD-0065 (2DD-0002)	DD	Check	3	3	Self-Actuated	Open	Safety Class Isolation
1DD-0066 (2DD-0009)	DD	Check	2	3	Self-Actuated	Open	Safety Class Isolation
DD-0016	DD	Check	2	3	Self-Actuated	Open	Recirculation Flow Path
DD-0018	DD	Check	3	3	Self-Actuated	Open	Discharge Flow Path
1DD-0020	DD	Ball/Handwheel	3	3	Local Manual	Open	Safety Class Isolation
XDD-0103	DD	Ball/Handwheel	3	3	Local Manual	Open	Safety Class Isolation
DD-0430	DD	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief, Containment Isolation
XDD-0044	DD	Check	2	3	Self-Actuated	Open	Recirculation Flow Path
XDD-0048	DD	Check	3	3	Self-Actuated	Closed	Discharge Flow Path
DO-0005	DO	Check	2	3	Self-Actuated	Closed	Discharge Flow Path
DO-0004	DO	Check	2	3	Self-Actuated	Closed	Discharge Flow Path
DO-0016	DO	Check	2	3	Self-Actuated	Closed	Discharge Flow Path
DO-0017	DO	Check	2	3	Self-Actuated	Closed	Discharge Flow Path
DO-0049	DO	Check	2	3	Self-Actuated	Closed	Fuel Oil Flow Path
1DO-0050 (2DO-0052)	DO	Check	2	3	Self-Actuated	Closed	Fuel Oil Flow Path
DO-0111	DO	Relief	1-1/2	3	Self Actuated	Closed	Recirculation Flow Path (miniflow)
DO-0187	DO	Relief	1-1/2	3	Self Actuated	Closed	Recirculation Flow Path (miniflow)
DO-0211	DO	Relief	1-1/2	3	Self Actuated	Closed	Recirculation Flow Path (miniflow)
DO-0287	DO	Relief	1-1/2	3	Self Actuated	Closed	Recirculation Flow Path (miniflow)
HV-3487	CA	Globe/Air	3	2	Auto Trip	Open	Containment Isolation
CI-0030	CA	Check	3	2	Self-Actuated	Open	Containment Isolation
1CI-0644 (Note 4)	CI	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Control Room Damper After A Non-Safety Air Line Break.

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 16 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
1CI-0645 (Note 4)	CI	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Control Room Damper After A Non-Safety Air Line Break.
1CI-0646 (Note 4)	CI	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Control Room Damper After A Non-Safety Air Line Break.
1CI-0647 (Note 4)	CI	Check	1/2	3	Self-Actuated	Closed/Open	Prevent Air Loss From Accumulator Air Supply to Control Room Damper After A Non-Safety Air Line Break.
HV-4168	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4169	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4170	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4167	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4166	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
1DO-0062	DG	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary
1DO-0063	DG	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary
1DO-0064	DG	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary
1DO-0065	DG	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary
2DO-0074	DG	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary
2DO-0075	DG	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary
2DO-0076	DG	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary
2DO-0077	DG	Check	1 1/2	3	Self-Actuated	Closed	Starting Air Pressure Boundary
DO-0104	DG	Check	1	3	Self-Activated	Open	Jacket Water Flow Path Boundary
DO-0204	DG	Check	1	3	Salf-Activated	Open	Jacket Water Flow Path Boundary
DO-0107	DG	3-Way	8	3	Self-Activated	Open	Jacket Water Temperature Control

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 17 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
DO-0207	DG	3-Way	8	3	Self-Activated	Open	Jacket Water Temperature Control
DO-0157	DG	Check	6	3	Self-Activated	Closed	Lube Oil Flow Path
DO-0257	DG	Check	6	3	Self-Activated	Closed	Lube Oil Flow Path
DO-0158	DG	Check	6	3	Self-Activated	Closed	Lube Oil Flow Path Boundary
DO-0258	DG	Check	6	3	Self-Activated	Closed	Lube Oil Flow Path Boundary
SV-3421-1E	DG	3-way/Solenoid	1/2	3	Auto Trip	Open	Starting Air Pressure Boundary
SV-3421-1F	DG	3-way/Solenoid	1/2	3	Auto Trip	Open	Starting Air Pressure Boundary
SV-3422-1E	DG	3-way/Solenoid	1/2	3	Auto Trip	Open	Starting Air Pressure Boundary
SV-3422-1F	DG	3-way/Solenoid	1/2	3	Auto Trip	Open	Starting Air Pressure Boundary
HV-4176	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4165	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4175	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4171	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
PS-0500	PS	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief, Containment Isolation
PS-0501	PS	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief, Containment Isolation
PS-0502	PS	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief Containment Isolation
PS-0503	PS	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief, Containment Isolation
HV-4172	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4173	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4174	PS	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-4178	PS	Angle/Air	3/4	2	Auto Trip	Closed	RHR Loop to Primary Sampling System
HV-4179	PS	Angle/Air	3/4	2	Auto Trip	Closed	RHR Loop to Primary Sampling System
HV-4182	PAS	Angle/Air	3/4	2	Auto Trip	Closed	ECCS Operation

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 18 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
CA-0016	CA	Check	3	2	Self-Actuated	Closed	Containment Isolation
HV-3486	CA	Globe/Air	3	2	Auto Trip	Closed	Containment Isolation
HV-6084	CH	Gate/Motor	6	2	Auto Trip	Open	Containment Isolation
CH-0024	CH	Check	6	2	Self-Actuated	Open	Containment Isolation
HV-6082	CH	Gate/Motor	6	2	Auto Trip	Open	Containment Isolation
HV-6083	CH	Gate/Motor	6	2	Auto Trip	Open	Containment Isolation
1CH-0271 (2CH-0281)	CH	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief, Containment Isolation
1CH-0272 (2CH-0282)	CH	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief, Containment Isolation
HV-6720	CH	Globe/Air	1	3	Remote Manual	Closed	Emergency Makeup Water Path
CH-0300	CH	Check	1	3	Self-Actuated	Closed	Emergency Makeup Water Path
CH-0301	CH	Check	1	3	Self-Actuated	Open	Emergency Makeup Water Path Boundary
CH-0302	CH	Globe/Handwheel	1	3	Local Manual	Closed	Safety Chilled Water Surge Tank Makeup
CH-0305	CH	Globe/Handwheel	1	3	Local Manual	Closed	Safety Chilled Water Surge Tank Makeup
HV-5542	VA	Butterfly/Motor	12	2	Auto Trip	Closed	Containment Isolation
HV-5543	VA	Butterfly/Motor	12	2	Auto Trip	Closed	Containment Isolation
HV-5563	VA	Butterfly/Motor	12	2	Auto Trip	Closed	Containment Isolation
HV-5540	VA	Butterfly/Motor	12	2	Auto Trip	Closed	Containment Isolation
HV-5541	VA	Butterfly/Motor	12	2	Auto Trip	Closed	Containment Isolation
HV-5562	VA	Butterfly/Motor	12	2	Auto Trip	Closed	Containment Isolation
HV-5556	PAS	Globe/Solenoid	1	2	Auto Trip	Closed	Containment Isolation
HV-5557	PAS	Globe/Solenoid	1	2	Auto Trip	Closed	Containment Isolation
HV-5544	RM	Globe/Solenoid	1	2	Auto Trip	Open	Containment Isolation
HV-5545	RM	Globe/Solenoid	1	2	Auto Trip	Open	Containment Isolation

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 19 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
HV-5558	PAS	Globe/Solenoid	1	2	Auto Trip	Closed	Containment Isolation
HV-5559	PAS	Globe/Solenoid	1	2	Auto Trip	Closed	Containment Isolation
HV-5560	PAS	Globe/Solenoid	1	2	Auto Trip	Closed	Containment Isolation
HV-5561	PAS	Globe/Solenoid	1	2	Auto Trip	Closed	Containment Isolation
HV-5546	RM	Globe/Solenoid	1	2	Auto Trip	Open	Containment Isolation
HV-5547	RM	Globe/Solenoid	1	2	Auto Trip	Open	Containment Isolation
HV-5548	VA	Butterfly/Air	18	2	Auto Trip	Closed	Containment Isolation
HV-5549	VA	Butterfly/Air	18	2	Auto Trip	Closed	Containment Isolation
HV-5157	VD	Diaphragm/Air	4	2	Auto Trip	Open	Containment Isolation
HV-5158	VD	Diaphragm/Air	4	2	Auto Trip	Open	Containment Isolation
VD-0003	VD	Check	2	3	Self-Actuated	Closed	Discharge Flow Path
VD-0004	VD	Check	2	3	Self-Actuated	Closed	Discharge Flow Path
VD-0011	VD	Check	2	3	Self-Actuated	Closed	Discharge Flow Path
VD-0012	VD	Check	2	3	Self-Actuated	Closed	Discharge Flow Path
HV-7311	WP	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
HV-7312	WP	Angle/Air	3/4	2	Auto Trip	Closed	Containment Isolation
1VD-0907 (2VD-0896)	VD	Relief	3/4	2	Self-Actuated	Closed	Pressure Relief, Containment Isolation
1BS-0064 (Note 6)	BS	Relief	1/4	2 (Note 7)	Self-Actuated	Closed	Pressure Relief, during LOCA
1BS-0065 (Note 6)	BS	Relief	1/4	2 (Note 7)	Self-Actuated	Closed	Pressure Relief, during LOCA
1BS-0066 (Note 6)	BS	Relief	1/4	2 (Note 7)	Self-Actuated	Closed	Pressure Relief, during LOCA
1BS-0067 (Note 6)	BS	Relief	1/4	2 (Note 7)	Self-Actuated	Closed	Pressure Relief, during LOCA
1BS-0068 (Note 6)	BS	Relief	1/4	2 (Note 7)	Self-Actuated	Closed	Pressure Relief, during LOCA
1BS-0069 (Note 6)	BS	Relief	1/4	2 (Note 7)	Self-Actuated	Closed	Pressure Relief, during LOCA

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 20 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)

SYMBOL	SYSTEM
AF	AF Auxiliary Feedwater
BS	Buildings and Structures (Containment Pressure Airlock)
CA	Compressed Air-Service Air
CC	Component Cooling Water
CH	Chilled Water
CI	Compressed Air - Instrument Air
CS	Chemical and Volume Control System
CT	Containment Spray
DD	Demineralized and Reactor Makeup
DG	Diesel Generator and Auxiliary Systems
SYMBOL	SYSTEM
DO	Diesel Generator Fuel Oil
FP	Fire Protection
FW	Steam Generator Feedwater
MS	Main Steam, Reheat and Dump
PAS	Post Accident Sampling
PS	Process Sampling Primary Plant
RHR	Residual Heat Removal
SF	Spent Fuel Pool Cooling and Clean-Up
SW	Service Water
VA	Heating and Ventilation

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 21 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)
VD	Vents and Drains						
WP	Waste Processing						
RM	Radiation Monitoring						

GENERAL

RMWST	Reactor makeup water storage tank
RWST	Refueling Water Storage Tank
ATM	Atmospheric
N/A	Not Applicable

TABLE 3.9B-10
ACTIVE VALVES

(Sheet 22 of 22)

Valve Identification or Location No. (Note 1)	System	Valve Type And Actuator	Size In.	ANS Safety Class	Method of Actuation	Normal Position	Function (Note 5)

Notes:

1) See Section 3.9N, Table 3.9N-10 for NSSS Active Valve List.

2) Unit 1 and Unit 2 Tag Numbers are generally the same except for the prefix. Any Unit 2 difference will be identified in a future amendment.

   Examples:

   1-HV-2333A is Unit 1
   2-HV-2333A is Unit 2
   HV-2333A indicates both Units
   CP1-CTVBCA-01 is Unit 1
   CP2-CTVBVCA-01 is Unit 2
   CP1/CP2-CTVBCA-01 indicates both Units
   X-PV-3533 indicates a common valve which serves both Units
   1AF-230 indicates Unit 1
   (2AF-231) indicates equivalent Unit 2 valves

3) Deleted.

4) Common valve; function is XCI.

5) See Section 6.2.4 for containment isolation details.

   Note: Steamline Isolation and Feedwater Isolation indicates valves which also function as Containment Isolation valves.

6) This equipment is unit specific. There is no corresponding equipment for the opposite unit.

7) These valves are not ASME Code and are not included in the Inservice Test Program. However, they are included in another program to assure their operability (as allowed by NUREG-1482, Appendix A, Position 11). Also, these valves are not included in the containment isolation tables of Section 6.2.4. These valves are within a closed system and satisfy the requirements of NUREG-0800, Section 6.2.4.11.6 paragraph o.

8) Valves LV-4500 and LV-4501 are fail closed valves to prevent overfill of the CCW surge tank on loss of power or air. Their nuclear safety function is to be manually opened via their hand wheels to provide seismic Category I makeup via LV-4500-1 which is a fail open valve. LV-4500-1 may be either open or closed in normal operation.

9) The feedwater check valves in the auxiliary feedwater lines to the steam generator open to provide the AFW flowpath during design basis events. These valves must close in the event of a line break upstream of the outboard check valve in order to terminate a Condition III loss of feedwater event. In addition, these valves also must close post-LOCA to protect the cold containment penetration during intermittent operation of AFW.

TABLE 3.9B-11
LOCATION OF SNUBBERS IN THE CONTAINMENT, SAFEGUARDS,
AUXILIARY AND FUEL BUILDINGS

System	Containment Building	Safeguards Building	Auxiliary Building	Fuel Building
Main Steam	X	X		
Feedwater	X	X		
Auxiliary Feedwater		X		
Safety Injection	X	X		
Component Cooling Water	X	X	X	X
Residual Heat Removal	X	X		
Containment Spray	X	X		
Service Water		X	X	
Reactor Coolant System	X			
Boron Recycle System	(Only Rigid Restraints Planned)			
Diesel Oil System	(Only Rigid Restraints Planned)			
Chemical and Volume Control	X	X	X	
Spent Fuel Pool Cooling and Cleanup System	X	X		X
Boron Thermal Regeneration Sub-system	(Only Rigid Restraints Planned)			
Demineralized Water Makeup System	(Only Rigid Restraints Planned)			

TABLE 3.9B-12
RATED LOAD OF MECHANICAL SNUBBERS

The snubbers are manufactured by Pacific Scientific and are rated as follows:

Model	Rated Loads (lbs)
PSA-1/2	650
PSA-1/4	350
PSA-100	120,000
PSA-35	50,000
PSA-10	15,000
PSA-3	6,000
PSA-1	1,500

3.10N SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that instrumentation and electrical equipment classified as seismic Category I is capable of performing designated safety related functions in the event of an earthquake. The information presented includes identification of the Category I instrumentation and electrical equipment that are within the scope of the Westinghouse Nuclear Steam Supply System (NSSS), the qualification criteria employed and for each item of equipment; the designated safety related functional requirements, definition of the applicable seismic environment and documentation of the qualification process employed to demonstrate the required seismic capability.

3.10N.1 SEISMIC QUALIFICATION CRITERIA

3.10N.1.1 Qualification Standards

The methods of meeting the general requirements for seismic qualification of Category I instrumentation and electrical equipment as described by General Design Criteria (GDC) 1, 2, and 23 are described in Section 3.1. The general methods of implementing the requirements of Appendix B to 10CFR Part 50 are described in Chapter 17.

The Commission's recommendations concerning the methods to be employed for seismic qualification of electrical equipment are contained in Regulatory Guide 1.100, which endorses IEEE-344-1975. Westinghouse will meet this standard, as modified by Reg. Guide 1.100, by either type test, analysis, or an appropriate combination of these methods. Westinghouse will meet this commitment employing the methodology described in Reference [1], and the qualification program plans and procedures described in Reference [2].

3.10N.1.2 Performance Requirements for Seismic Qualification

Reference 2 contains an equipment qualification data package (EQDP) for every item of instrumentation and electrical equipment classified as seismic Category I within the Westinghouse NSSS scope of supply. Table 3.10N.1 identifies the Category I equipment supplied by Westinghouse for this application and references the applicable EQDP contained in Supplement 1. Each EQDP in Supplement 1 contains a section entitled "Performance Specifications". This specification establishes the safety related functional requirements of the equipment to be demonstrated during and after a seismic event. The required response spectrum (RRS) employed by Westinghouse for generic seismic qualification is also identified in the specification, as applicable. The spectra employed envelopes the plant specific spectra defined in Section 3.7.

3.10N.1.3 Acceptance Criteria

Seismic qualification must demonstrate that Category I instrumentation and electrical equipment is capable of performing designated safety related functions during and after an earthquake of magnitude up to and including the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) without the initiation of undesired spurious actuation which might result in consequences adverse to safety. The qualification must also demonstrate the structural integrity of mechanical supports and structures at the OBE level. Some permanent mechanical

deformation of supports and structures is acceptable at the SSE level providing that the ability to perform the designated safety related functions is not impaired.

3.10N.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

In accordance with IEEE 344-1975, seismic qualification of safety related electrical equipment is demonstrated by either type testing, analysis or a combination of these methods. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors including; practicability, complexity of equipment, economics, availability of previous seismic qualification to earlier standards, etc. The qualification method employed for a particular item of equipment is identified in the individual Equipment Qualification Data Packages (EQDP's) of Reference 2.

All NSSS safety-related electrical equipment located in a mild environment area will be seismically qualified as described in FSAR Section 3.10N, except that the additional requirements imposed by IEEE Standard 323-1974 do not apply. The procurement documents will specify that the effects of aging on seismic qualification be assessed and if there are aging effects, require pre-aging or analysis of aging effects as part of the seismic qualification.

3.10N.2.1 Seismic Qualification by Type Test

From 1969 to mid-1974 Westinghouse seismic test procedures employed single axis sine beat inputs in accordance with IEEE 344-1971 to seismically qualify equipment. The input form selected by Westinghouse was chosen following an investigation of building responses to seismic events as reported in Reference 3. In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program (Reference 4). This retesting was performed at the request of the NRC staff on agreed selected items of equipment employing multi-frequency, multi-axis test inputs (Reference 5) to demonstrate the conservatism of the original sine-beat test method with respect to the modified methods of testing for complex equipment recommended by IEEE 344-1975.

The original single axis sine beat testing and the additional retesting completed under the Supplemental Test Program has been the subject of generic review by the Staff. For equipment which has been previously qualified by the single axis sine beat method and included in the NRC seismic audit and, where required by the Staff, the Supplemental Qualification Program (Reference 4), no additional qualification testing is required to demonstrate acceptability to IEEE 344-1975 provided that:

1. The Westinghouse aging evaluation program for aging effects on complex electronic equipmet located outside containment demonstrates there are no deleterious aging phenomena. In the event that the aging evaluation program identifies materials that are marginal, either the materials will be replaced or the projected qualified life will be adjusted.

2. Any changes made to the equipment due to 1. above or due to design modifications do not significantly affect the seismic characteristics of the equipment.

3. The previously employed test inputs can be shown to be conservative with respect to applicable plant specific response spectra.

This equipment is identified in Reference 1, Table 7.1 and the test results in the applicable EQDP's of Reference 2.

For equipment tests after July, 1974 (i.e. new designs, equipment not previously qualified or previously qualified that does not meet 1, 2 and 3 above) seismic qualification by test is performed in accordance with IEEE 344-1975. Where testing is utilized, multi-frequency multi-axis inputs are developed by the general procedures outlined in Reference 5. The test results contained in the individual EQDP's of Reference 2 demonstrate that the measured Test Response Spectrum envelopes the applicable Required Response Spectrum (RRS) defined for generic testing as specified in Section 1 of the EQDP (Reference 2). Qualification for plant specific use is established by verification that the generic RRS specified by Westinghouse envelopes the applicable plant specific response spectrum. Alternative test methods, such as single frequency, single axis inputs, are used in selected cases as permitted by IEEE 344-1975 and Regulatory Guide 1.100.

3.10N.2.2 Seismic Qualification by Analysis

The structural integrity of safety related motors (Table 3.10N.1 EQDP- AE-2 and 3) is demonstrated by a static seismic analysis in accordance with IEEE 344-1975 with justification. Should analysis fail to show the resonant frequency to be significantly greater than 33 Hz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads and stresses under various combinations of seismic, gravitational and operational loads. The worst case (maximum) values calculated are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas; 1) maximum rotor deflection, 2) maximum shaft stresses, 3) maximum bearing load and shaft slope at the bearings, 4) maximum stresses in the stator core welds, 5) maximum stresses in the stator core to frame welds, 6) maximum stresses in the motor mounting bolts and, 7) maximum stresses in the motor feet.

The analytical models employed and the results of the analysis are described in Section 4 of the applicable EQDP's (Reference 2).

3.10N.3 METHOD AND PROCEDURES FOR QUALIFYING SUPPORTS OF ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Where supports for the electrical equipment and instrumentation are within the Westinghouse NSSS scope of supply, the seismic qualification tests and/or analysis are conducted including the supplied supports. The EQDP's contained in Reference 2 identify the equipment mounting employed for qualification purposes and establish interface requirements for the equipment to ensure subsequent in-plant installation does not prejudice the qualification established by Westinghouse.

3.10N.4 OPERATING LICENSE REVIEW

The results of tests and analyses that ensure that the criteria established in Section 3.10N.1 have been satisfied employing the qualification methods described in Sections 3.10N.2 and 3.10N.3 are included in the individual EQDP's contained in Reference 2 as they become available.

REFERENCES

1. Butterworth, G. and Miller, R. B., “Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment,” WCAP-8587, Revision 6, March 1983.

2. “Equipment Qualification Data Packages,” Supplement 1 to WCAP- 8587.

3. Morrone, A., “Seismic Vibration Testing with Sine Beats,” WCAP- 7558, October 1971.

4. NS-CE-692, Letter dated July 10, 1975 from C. Eicheldinger (Westinghouse) to D. B. Vasello (NRC).

5. Jarecki, S. J., “General Method of Developing Multi-frequency Biaxial Test Inputs for Bistables,” WCAP-8624 (Proprietary) September 1975 and WCAP-8695 (Non-Proprietary) August 1975.

TABLE 3.10N-1
SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY

EQUIPMENT	EQDP[a]
SAFETY RELATED VALVE ELECTRIC MOTOR OPERATORS	EQDP-HE-1 AND 4
SAFETY RELATED PILOT SOLENOID VALVES	EQDP-HE-2
SAFETY RELATED EXTERNALLY MOUNTED LIMIT SWITCHES	EQDP-HE-3
RCS SAFETY VALVE LIMIT SWITCHES	EQDP-HE-7
CONAX CONNECTORS	EQDP-HE-8
SOLENOID VALVES - RCS VENT	EQDP-HE-10
LARGE PUMP MOTORS (OUTSIDE CONTAINMENT)	EQDP-AE-2
CANNED PUMP MOTORS (OUTSIDE CONTAINMENT)	EQDP-AE-3
PRESSURE TRANSMITTERS	EQDP-ESE-1 AND 2
DIFFERENTIAL PRESSURE TRANSMITTERS	EQDP-ESE-3 AND 4
RESISTANCE TEMPERATURE DETECTORS	EQDP-ESE-6 AND 7
NUCLEAR INSTRUMENTATION SYSTEM (NIS)	EQDP-ESE-10
PROCESS PROTECTION SETS	EQDP-ESE-13
INDICATORS, POST-ACCIDENT MONITORING	EQDP-ESE-14
RECORDERS, POST-ACCIDENT MONITORING	EQDP-ESE-15
SOLID-STATE PROTECTION SYSTEM & SAFEGUARD TEST CABINET (2 TRAIN)	EQDP-ESE-16
REACTOR TRIP SWITCHGEAR	EQDP-ESE-20
CONTAINMENT PRESSURE SENSOR	EQDP-ESE-21
EXCORE NEUTRON DETECTORS (POWER RANGE)	EQDP-ESE-22
NITROGEN-16 DETECTOR	EQDP-ESE-27
DIFFERENTIAL PRESSURE INDICATING SWITCHES	EQDP-ESE-40
GROUP B	
INCORE THERMOCOUPLE CONNECTORS	EQDP-ESE-43
THERMOCOUPLE JUNCTION BOX	EQDP-ESE-44
EXCORE NEUTRON DETECTORS (SOURCE RANGE), PRE-AMPLIFIERS AND FLUX DOUBLING CIRCUIT	EQDP-ESE-47

a) Equipment Qualification Data Package

3.10B SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10B.1 SEISMIC QUALIFICATION CRITERIA

The seismic Category I instrumentation and electrical equipment which require seismic qualification are as follows:

1. 6900-V switchgear (nuclear-safety-related)
2. 6900-V to 480-V transformers (associated with nuclear-safety-related buses)
3. 480-V switchgear (nuclear-safety-related)
4. 480-V motor control and motor control centers (nuclear-safety- related)
5. 125-V station batteries and racks
6. 480-VAC to 125-VDC battery chargers
7. 125-VDC panels and switchboards (nuclear-safety-related)
8. 125-VDC to 120-VAC inverters (nuclear-safety-related instrument buses)
9. Nuclear-safety-related instrument bus panels
10. Containment penetration assemblies
11. Not Used
12. Diesel generator and accessories
13. Diesel generator control panels
14. Relay boards and racks (nuclear-safety-related)
15. Instrument racks (nuclear-safety-related)
16. Hot shutdown panel (used in event of Control Room evacuation)
17. Main Control boards
18. Vertical panels (including solid state sequencer and isolation equipment)
19. Wire and cable raceway system (nuclear-safety-related)
20. Electrical supports (nuclear-safety-related)
21. Containment particulate, iodine, and gas radiation monitors

22. Motors (nuclear-safety-related)

23. 120/208 VAC power distribution and lighting panels

24. Miscellaneous 3-phase 480-108/120v and 1-phase 480-120v transformers

25. Control Room ventilation radiation monitors

26. Accident Monitoring System including sensors (see Section 7.5)

27. Electronic Transmitters-pressure and differential pressure (nuclear-safety-related).

28. Analog Control Cabinets (Train A and Train B)

29. Resistance Temperature Detectors (nuclear-safety-related)

30. Level switches (nuclear-safety-related)

31. Pressure switches (nuclear-safety-related)

32. Differential pressure switches (nuclear-safety-related)

The criteria for establishing seismic design adequacy, by testing or analysis, are described in Section 3.7B.2.

Seismic Category I instrumentation and electrical equipment including standby power systems are designed to maintain structural integrity and functional operability during and after an earthquake of magnitude up to and including the SSE. Seismic design of the Reactor Protection System and ESF circuits is discussed in Section 3.10N.

Horizontal and vertical ground accelerations during an SSE are used to formulate floor response spectra at each equipment location as described in Section 3.7B.2. Equipment and system specifications include the appropriate response spectra used for the analysis or testing of the equipment. The seismic qualification program for electrical equipment is as described in Section 3.7B.2.1.3. Analysis is used for the qualification of equipment of relatively simple geometry which can be modeled accurately.

Suppliers are required to furnish documentation of actual test results or detailed computations, if an analytical approach is used, to substantiate the capability of the equipment to perform its intended function under the specified conditions.

Seismic Category I instrumentation and electrical equipment are seismically qualified in accordance with the procedures and documentation requirements specified in IEEE 344-1975," Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." For equipment in mild environment areas, the procurement documents will require that the effects of aging on seismic qualification be assessed and, if there are aging effects, will require pre-aging or analysis of aging effects as part of the seismic qualification requirements.

3.10B.2 METHODS AND PROCEDURES FOR QUALIFYING INSTRUMENTATION AND ELECTRICAL EQUIPMENT

All seismic Category I instrumentation and electrical equipment are qualified by analysis or testing, or both. Seismic analysis without testing is used if it is demonstrated that the performance of the equipment subjected to earthquake motion can be conservatively predicted and if the functional operability of the instrumentation or equipment is assured by its structural integrity alone.

The seismic analysis methods (response spectrum, time history) and the testing procedures used are described in Section 3.7B.2.1.

Where analysis is used for qualifying the seismic Category I instrumentation and electrical equipment, it is required that the maximum stresses and deformations in the equipment including the effects of the normal operating loads plus the SSE be limited to prevent loss of function of the equipment. To ensure structural integrity and functional operability of the equipment after several occurrences of the OBE, it is required that the maximum stresses in the equipment, including the effects of the normal operating loads plus the OBE, be maintained within the normal allowable material working stress limits set forth in the appropriate design standards and codes, and that the equipment operate and maintain structural integrity without permanent deformation.

When testing is used for the seismic qualification of seismic Category I instrumentation and electrical equipment, tests at the SSE level are performed to qualify the equipment for operation or structural integrity, or both, during and after the SSE. Each test at the SSE level is preceded by a minimum of five tests at the OBE level to verify the structural and functional integrity of the equipment after several occurrences of the OBE.

3.10B.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Supports of instrumentation and electrical equipment such as battery racks, instrument racks, control consoles, cabinets, and panels are analyzed or tested, or both, by their suppliers in accordance with the methods and procedures described in Section 3.7B.2.1.3.

Such supports are generally required to have overall natural frequencies greater than 33 Hz. Where this requirement cannot be met, suppliers are required to qualify their products by performing full dynamic analysis or testing, or both, to demonstrate their structural integrity during and after the SSE, and to generate response spectra or derive maximum amplification factors at all equipment and instrument mounting locations. The equipment and instruments to be mounted on these supports are then analyzed or tested by their suppliers on the basis of the response spectra or maximum accelerations furnished by support manufacturers.

The supplier accounts for possible amplification through equipment supports by analysis or testing, as described in Section 3.7B.2. Verification is by documentation based on either actual tests or analytical methods.

Documentation pertaining to the seismic qualification of all seismic Category I electrical equipment, instrumentation, and supports is reviewed for compliance with the requirements set forth in Section 3.7B.2, NRC regulatory guides, and applicable codes and standards.

The following is a description of the analysis and testing procedures used for qualification of Seismic Category I cable trays:

The types of cable trays used are: ladder type with or without tray covers and solid bottom type with or without covers. Trays are supported on structural steel frames anchored to the ceilings, walls, or floors, and braced transversely and longitudinally. The configuration of the tray and support system consists of tray segments with bolted splice connections, tray attachment to the respective supports by bolted connections and support attachment to the concrete super-structures generally by bolted connections (or by welded connections if embedded plates are available).

Under normal conditions of loading and support spacing, trays and their supports may have natural frequencies which are lower than 33 Hz; therefore, they are required to sustain amplified accelerations, and are designed for the peak values of the floor response spectra at the applicable support elevation, unless a frequency analysis is performed to justify the use of lower acceleration values.

For the qualification of trays, maximum equivalent static loads for SSE are determined from the peaks of floor response spectra curves corresponding to 7% damping, unless a frequency analysis is performed to justify the use of lower acceleration values and are based on the full mass of the tray (weight of the tray plus cable). Suppliers are required to determine by static testing on representative trays of each type being used, the strength of the trays in each of the three principal directions of load application, and to compare the equivalent static loads in each direction with the respective strengths of the trays. The results are combined using an interaction formula to account for the simultaneous application of dead load and earthquake excitation in the three principal directions.

Supports for the Seismic Category I electrical cable tray and conduit system are designed to resist the gravity and seismic loads imposed on them by the cable trays and/or conduits. The design of the supports is generally based on the peak values of the applicable floor response spectra, unless a frequency analysis is performed to justify the use of lower acceleration values. Floor response spectra corresponding to 7% and 4% dampings for the SSE and OBE, respectively, are used for the design of cable tray/support systems, consistent with Regulatory Guide 1.61 for bolted steel structures. For the design of conduit supports, floor response spectra corresponding to 3% and 2% dampings for the SSE and OBE, respectively are used. For Thermolag upgrade to safety related conduits, damping values of 4% and 7% for the OBE and SSE are used, also consistent with Regulatory Guide 1.61 for bolted structures.

Supports for cable trays are made of relatively light weight structural steel shapes, welded together such as to provide a structural element to support the cable tray raceway. Supports for conduits are made of structural steel or unistrut members. The cable tray or conduit raceway support systems are adequately braced in the transverse and longitudinal directions to account for the loadings in these directions.

The design of the cable tray supports is based on elastic working stress design methods. Upon evaluation, there were found no conditions under SSE loading combinations that would exceed the yield strength of the materials used in the supports except as permitted by Reference [1].

The factor of safety on expansion anchors used for all Seismic Category I cable tray supports (Hilti Kwik-Bolt) has been established to be a minimum of 4 for both OBE and SSE loading

conditions. Steel structures subject to this loading condition are designed based on elastic working stress design methods, as noted above, and meet the requirements of AISC working stress allowables. The expansion anchors which are used with these steel structures have been designed in accordance with the manufacturer's recommendation of a minimum factor of safety of 4 for the maximum working load conditions (OBE) and maximum faulted (extreme) load conditions (SSE) for Seismic Category I cable tray hangers.

The factor of safety on expansion anchors used for Seismic Category II cable tray supports when subjected to full SSE loading conditions is a minimum of 3. The steel structures which are subjected to this loading condition when designed using working stress design methods are allowed to be stressed to a level of 1.6 times the stress of OBE loading conditions. This increased allowable in stress levels is partly attributed to the high improbability of the occurrence of such SSEloading combinations. This same rationale is also applicable to the associated factor of safety of 3 for expansion anchors. Since the loading condition in question is one of extreme environmental conditions and highly improbable, the lower factor of safety is applicable. Also, Seismic Category II hangers are not required to be functional for a safe shutdown of the plant. During a SSE event, their major function is not to adversely affect other components which are required for safe shutdown of the plant. In addition to manufacturer's recommendations, technical information supporting this decision is found in Appendix B of Reference [2] as well as Reference [3].

3.10B.4 OPERATING LICENSE REVIEW

The available seismic qualification documentation for seismic Category I instrumentation and electrical equipment is retained in the plant records.

REFERENCES

1. AISC Specification For the Design, Fabrication and Erection of Structural Steel for Buildings, 1969 including Supplement Numbers 1, 2, and 3.

2. ACI Committee 349, Criteria for Reinforced Concrete Nuclear Power Containment Structures, ACI Journal, January 1972.

3. NRC IE Information Notice No. 79-14, Safety Classification of Electrical Cable Support Systems, June 11, 1979.

TABLE 3.10B-1
DELETED

3.11N ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

The mechanical and electrical portions of the Engineered Safety Features and the Reactor Protection Systems are designed to ensure acceptable performance in all environments anticipated under normal, test, and design basis accident conditions. This section presents information on the design basis and qualification verifications for mechanical and electrical equipment in the Engineered Safety Features and the Reactor Protection System that are within the scope of the Westinghouse Nuclear Steam Supply System (NSSS). Also, this section refers to equipment located in a potentially harsh environment, unless otherwise noted. Section 3.7N presents the seismic design requirements and Section 3.10N presents the seismic qualification of electrical equipment.

3.11N.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

A list of Class 1E equipment located in potentially harsh environments and their postulated environmental extremes are reflected in controlled documents, (e.g., EEQSPs, EQML). The definition of potentially harsh and mild environments at CPNPP is provided in Section 3.11B.1.

The safety-related mechanical equipment at CPNPP has been designed to withstand environmental effects as required by GDC-4 and Appendix B of 10 CFR Part 50. These requirements are satisfied through the design, specification, procurement, and quality assurance procedures at CPNPP as supplemented by the pump and valve operability programs and the CPNPP maintenance and surveillance programs.

The energy and environmental variations to which the Westinghouse supplied protection system equipment will be qualified are defined in the applicable equipment qualification data package (Supplement 1 to WCAP-8587).

The CPNPP POST-LOCA long-term operability duration is 100 days, per TXX-88742, letter from W. G. Counsil to USNRC [4].

3.11N.2 QUALIFICATION TESTS AND ANALYSIS

For Westinghouse NSSS Class 1E equipment located in a potentially harsh environment, Westinghouse meets the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” including IEEE Standard 323a-1975, the Nuclear Power Engineering Committee (NPEC) Position Statement of July 24, 1975, by an appropriate combination of any or all of the following: type testing, operating experience, qualification by analysis, and ongoing qualification. The Westinghouse approach to satisfying IEEE-Standard 323-1974 is documented in WCAP-8587 [1] which has been accepted by the NRC Staff. Controlled documents (e.g., EEQSPs, EQML) identify the Westinghouse supplied Class 1E equipment located in a potentially harsh environment and the corresponding Equipment Qualification Data Packages.

For NSSS Class 1E equipment located in a mild environment area, CPNPP demonstrates qualification through procurement documents, which include the most stringent requirements for the equipment’s end use. The maintenance and surveillance programs in conjunction with a trending program provide the assurance that equipment which meets the requirements of the procurement documents is maintained throughout the equipment’s installed life (see Appendix 3A Section 2.4).

All seismic Category I Class 1E equipment located in a mild environment area is seismically qualified to IEEE Standard 344-1975 and Regulatory Guide 1.100 using the methods, procedures, and documentation described in FSAR Sections 3.7 and 3.10N.

The overall Class 1E Westinghouse equipment qualification program includes generic environmental conditions, e.g., temperature, pressure, relative humidity, chemistry, radiation, which are established for the various pieces of Westinghouse supplied Class 1E equipment. The conditions vary according to location of the equipment. The generic environmental conditions for which the equipment is qualified are reported in the specific Equipment Qualification Data Package and the Environmental Equipment Qualification Summary Packages. The postulated environmental extremes used for equipment qualification are provided in the EEQSPs.

The peak calculated containment vapor temperature for the design basis MSLB can be found in Table 6.2.1-2A. Component thermal analysis is required only if the environmental qualification test conditions are less than those calculated from the containment pressure temperature transient response analysis. Transmitters and valves/valve operators inside containment and electrical cables are qualified to the required parameter by testing or by a combination of testing/ analysis in accordance with IEEE 323-1974 (and the daughter standards such as IEEE 383-1974 for the electrical cables) if these are required to operate for design basis MSLB.

How the requirements of the General Design Criteria (GDC) 1, 4, 23, and 50 are met is addressed in Section 3.1. Specific information concerning how GDC 1 and 4 are met is reported in the applicable Equipment Qualification Data Packages [1, 3]. Specific information concerning how GDC 23 is met can be found in Section 7.2.2.2. Specific information concerning how GDC 50 is met is provided in Section 6.2. Specific information concerning how Appendix B of 10 CFR Part 50 is met is located in Chapter 17. Regulatory Guides 1.30, 1.40, 1.73 and 1.89 are addressed in Appendix 1A(N).

3.11N.3 QUALIFICATION TEST RESULTS

Qualification test results are provided in the Equipment Qualification Data Packages and in the Environmental Equipment Qualification Summary Packages.

3.11N.4 LOSS OF VENTILATION

Refer to Section 3.11B.4

3.11N.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

Chemical environments in the primary containment as result of an accident condition are shown in the EEQSPs and discussed in Section 3.11B.5.1. The postulated integrated radiation environments for normal and accident conditions are shown in the EEQSPs and discussed in Section 3.11B.5.2.

REFERENCES

1. Jordan, W. G., Lorentz, D. G. and Miller, R. B., “Methodology For Qualifying Westinghouse PWR-SD Supplied NSSS Safety Related Electrical Equipment”, WCAP-8587, Revision 1, September, 1977.

2. Wilson, J. F., “Electric Hydrogen Recombiner for PWR Containments,” WCAP-7709L, Supplements 1 through 7 (Proprietary) and WCAP-7820, Supplements 1 through 7 (Non-Proprietary), 1971 through 1977.

3. “Equipment Qualification Data Packages”, Supplement to WCAP-8587, Revision 1, April 1978.

4. TXX-88742, Letter from W. G. Counsil to USNRC dated October 24, 1988.

TABLE 3.11N-1

Table 3.11N-1 is deleted and the specific environmental details are reflected in control documents (EEQSPs, EQML, DBDs).

TABLE 3.11N-2

Table 3.11N-2 is deleted and the specific environmental details are reflected in control documents (EEQSPs, EQML, DBDs).

TABLE 3.11N-3

Table 3.11N-3 is deleted and the specific environmental details are reflected in control documents (EEQSPs, EQML, DBDs).

3.11B ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

To ensure proper performance during normal, abnormal, and design basis accident conditions, instrumentation and electrical components of the Engineered Safety Features (ESF) and other safety-related systems are designed for the environmental conditions and bases described in Section 3.11B.1. Specific information is also included in Sections 3.11B.2 and 3.11B.5 to demonstrate that the safety-related components have the capability to function as required in the combined temperature, pressure, humidity, chemistry, and radiation dose of the post-accident environment. In addition, an evaluation is made in Section 3.11B.4 of the environmental effects that would follow the loss of plant ventilation system used for cooling Class 1E electrical equipment. This Section refers to equipment located in a potentially harsh environment, unless otherwise noted.

Certain Class 1E equipment is furnished with the same train environmental support system (See Sections 9.4.5, 9.4C.1, 9.4C.3, 9.4C.4 and 9.4F). Redundant environmental support systems are not required for any Class 1E equipment because Class 1E equipment is already redundant. Redundant environmental support systems are provided where both trains of Class 1E equipment are located and emergency ventilation is required (See Sections 9.4.1, 9.4B and 9.4C.8). This type of design enables the safety related environmental systems to sustain a single active component failure without total loss of function (See Section 9.4).

The accident (limiting) environmental conditions are discussed in Section 3.11B.4.

Section 3.7B presents the seismic design requirements, and Section 3.10B presents the seismic qualification of electrical equipment.

LOCAs and steam line breaks are design basis accidents that will cause enveloping environmental changes within the Containment. Pressure and temperature transients for these design basis accidents are presented in Section 6.2. Loss of ventilation and High Energy Line Breaks (HELBs) outside containment are postulated events that can cause environmental changes in areas outside the containment. The environmental analysis performed to determine HELB environmental consequences for purposes of qualifying equipment located outside containment is described in Section 3.6B.1.2.3. Loss of ventilation is discussed in Section 3.11B.4 below.

In addition to HELBs as described in Section 3.6B, non-mechanistic cracks in break exclusion piping (i.e., superpipe) in the main steam and feedwater penetration area outside Containment are postulated and evaluated for environmental effects. (See Section 3.6B.2.5.2 for details).

3.11B.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Safety-related equipment and components are designed to function during and subsequent to the DBA and are located within the Containment as well as in other seismic Category I structures (Safeguards buildings, Auxiliary Building, Fuel Building, Electrical and Control Building and Service Water Intake Structure).

The environmental qualification program at CPNPP includes all Class 1E equipment located in potentially harsh environment areas. Class 1E equipment located in a mild environment area is included in the maintenance and surveillance programs discussed in Appendix 3A, Section 2.4. For CPNPP, mild environment areas are defined as areas, outside the containment, that are not

potentially harsh due to a design basis event. In addition, a mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

A potentially harsh environment is defined as an environment where safety-related equipment would experience, due to the direct effects of a design basis event (Loss of Coolant Accident, Main Steam Line Break, High Energy Line Break, Superpipe Cracks) any of the following parameters:

1. An ambient pressure increase greater than two pounds per square inch (2psi) above atmospheric, or

2. An ambient temperature increase greater than five degrees centigrade (5°C or 9°F) above the postulated maximum temperature based on normal and anticipated operational occurrences, or

3. A total integrated radiation exposure dose of $1x10^4$ rads Gamma.

NOTE: For postulated radiation environments between $1x10^3$ and $1x10^4$ rads Gamma, electronic equipment including semiconductors are evaluated to demonstrate qualification to the postulated radiation dose.

4. A relative humidity value of 100%

5. Flooding

   Class 1E equipment subject to flooding due to a design basis accident (as described above) is qualified to operate for the flooding environment. If the equipment cannot be qualified, an evaluation is performed to determine if it is required to operate for mitigation of the accident. If the equipment is not required to operate, an evaluation is performed to demonstrate that failure of the equipment is not detrimental to the safe shutdown of the plant.

   For Class 1E equipment which experiences flooding due to a Moderate Energy Line Crack, the impact of that flooding is addressed by the Systems Interaction Program as described in Section 3.6 of the FSAR. Wherever reasonable, CPNPP has been designed to locate Class 1E equipment such that the equipment does not experience flooding.

   As part of this process, all safety and non-safety-related electrical equipment, instrumentation and control circuits, and components located inside containment below elevation 816'-10", which may become submerged as a result of LOCA, have been identified and evaluated. The analysis concluded that:

   a. There is no safety significance from the failure of this electrical equipment (e.g., spurious actuation or loss of actuation function) as a result of flooding.

   b. There will be no degradation of Class 1E electrical power sources serving this electrical equipment as a result of such submergence.

c. The design changes for the required safety-related equipment, conduit and/or cable routing subject to submergence have been identified to resolve any unacceptable environmental interaction with required safety-related equipment.

A list of Class 1E equipment located in potentially harsh environments and their postulated environmental extremes are provided in Controlled Documents (e.g., EEQSPs, EQML). Environmental qualification documentation for Class 1E equipment is documented in Environmental Equipment Qualification Summary Packages (EEQSPs) and is retained in the plant records.

The ESF and other safety-related equipment which must remain operable during and after the DBA are further discussed in appropriate chapters of the FSAR as follows:

1. ESF and other Safety-Related Equipment Located Inside Containment

   a. Mechanical equipment (which contains Class 1E electrical components) in Chapter 6.

   b. Class 1E equipment in Chapter 8.

   c. Instrumentation and controls in Chapter 7.

2. ESF and other Safety-Related Equipment Located Outside Containment

   a. Mechanical equipment (which contains Class 1E electrical components) is described in Chapters 6, 9 and 10.

   b. Instrumentation and controls are described in Chapter 7.

   c. Class 1E equipment is described in Chapter 8.

The safety-related mechanical equipment at CPNPP has been designed to withstand environmental effects as required by GDC-4 and Appendix B of 10 CFR Part 50. These requirements are satisfied through the design, specification, procurement, and quality assurance procedures used at CPNPP as supplemented by the pump and valve operability programs and the CPNPP maintenance and surveillance programs.

3.11B.2 QUALIFICATION TESTS AND ANALYSES

Qualification tests and analyses are performed on all ESF equipment and components located in a potentially harsh environment as necessary to ensure their availability during and after a DBA. These tests consist of simulation of actual physical conditions on the equipment or a prototype on a generic basis, or analysis, or a combination of prototype tests and analysis as applicable. Qualification testing is performed under simulated conditions of temperature, pressure, relative humidity, chemistry, and radiation dose in excess of those expected for post-DBA conditions. The testing period is sufficient to ensure the capability to function during and after a DBA.

In order to provide assurance that all ESF and other Safety Related equipment has the capability to meet environmental conditions as required the appropriate quality assurance programs are established and implemented.

Class 1E instrumentation and electrical equipment is capable of operating in the worst expected environmental conditions as required for each component and its location. The Class 1E electrical equipment is specified for manufacture in accordance with the criteria listed in Section 8.3. All Class 1E equipment located in a potentially harsh environment will be qualified per IEEE 323-1974 [11] and other applicable standards per Section 8.3 and below.

IEEE-323-1974 requirements include the need to establish the qualified life of Class 1E equipment. This requirement has, in many cases, represented a state-of-the-art challenge in assessing the longevity of equipment under normal and accident environments. Many of the equipment qualification test reports have identified relatively short lives of certain components. In some cases, these estimated, seemingly short, qualified lives may be the result of testing or accelerated aging limitations, rather than due to intrinsic equipment limitations. Luminant Power, as the first operating license applicant with an IEEE-323-1974 commitment has been confronted with an enormous developmental program in meeting these new requirements.

The CPNPP POST-LOCA long-term operability duration is 100 days, per TXX-88742, letter from W. G. Counsil to USNRC [20].

Consistent with Luminant Power's commitment to meet the requirements of IEEE-323-1974 subject to state-of-the-art limitations, Luminant Power will modify any of the existing qualified lives as additional information and better testing and analytical techniques are developed. Therefore, the existing qualified lives currently listed in the Environmental Equipment Qualification Summary Packages should be regarded as decision points with regard to an ongoing aging evaluation, rather than a fixed component replacement interval. Changes from those qualified lives currently indicated will be documented by revisions to the appropriate summary packages.

Instrumentation and control equipment mounted in the Control Room area is not subject to damaging vibration in either normal or postaccident modes, since no rotating equipment or fluid system components which could induce vibration are in proximity to the Control Room areas.

For Class 1E instrumentation and control equipment, including sensors, mounted in seismic Category I areas outside of the Control Room, the following design practices ensure that non-seismic vibration does not degrade the equipment:

1. Pressure and differential pressure transmitters and switches are mounted "off line," on secure and rigid (fn>33 hertz) instrument racks and supports. The instruments are supported independently of their connection to the process.

2. In-line temperature switches (thermostats) are not used.

3. In-line flow meters (such as target meters, rotameters, turbine meters, or paddle switches) are not used.

4. For equipment such as RTDs, level switches and valve limit switches, which are subject to non-seismic vibration, the qualification reports account for expected vibration, including the OBE tests required by IEEE 344-1975.

In addition to these design features, assurance that damaging vibration effects do not occur in service are provided by the preoperational tests and inspections as described in Section 14.2 as well as by the periodic on-line testing performed in accordance with Technical Specifications.

Electrical equipment and components located in potentially harsh environments outside of the Containment are qualified to meet the plant environmental conditions as listed in the EEQSPs.

Qualification tests for all safety related electric valve operators outside as well as inside containment are designed in accordance with References [8, 11, 13 and 15] to demonstrate their capability to function during and after accident environmental conditions. In addition, safety related electric valve operators installed inside the containment are also qualified in accordance with R.G. 1.73 [6].

The Class 1E instrumentation and control equipment in a potentially harsh environment have been qualified in accordance with IEEE 344-1975 and IEEE 323-1974 criteria. Designated non-Class 1E Accident Monitoring equipment located in a potentially harsh environment (see FSAR Section 7.5) have also been environmentally qualified.

Because the Containment has no Class 1E motors which are required to operate on a continuous basis during or following a DBA, NRC Regulatory Guide 1.40 [3] is not applicable to CPNPP.

Safety related equipment located inside the Containment that must remain operable during a DBA are identified in controlled documents. As indicated in the list referenced previously, the only motors in the Containment required to operate in the event of a DBA are those motor-operated isolation valves which operate at the onset of a DBA.

The peak calculated containment vapor temperature for the design basis MSLB can be found in Table 6.2.1-2A. Component thermal analysis is required only if the environmental qualification test conditions are less than those calculated from the containment pressure-temperature transient response analysis. Transmitters and valves/valve operators inside containment, containment electrical penetrations/header plate assemblies and electrical cables are qualified to the required parameter by testing or by a combination of testing/analysis in accordance with IEEE 323-1974 (and the daughter standards such as IEEE 317-1976 for the penetration assembly and IEEE 383-1974 for the electrical cables) if the devices are required to operate for design basis MSLB.

The Class 1E motors located outside the Containment in a potentially harsh environment are type-tested in accordance with the intent of IEEE 334-1974 [12].

Class 1E splices and cable connectors inside containment are environmentally qualified to meet the requirements of IEEE 323-1974 and IEEE 383-1974. The splices are listed and their use justified in Section 8.3 and Appendix 8A, “Analysis to Justify Cable Splices in Raceway.” There are no class 1E termination cabinets or terminal blocks inside containment other than class 1E terminal blocks which are part of class 1E equipment (such as Limitorque) which are qualified to IEEE-323-1974 requirements.

The initial preoperational test procedures incorporate the recommendations of NRC Regulatory Guide 1.41 [4] for verifying proper load group assignment.

Provisions are integrated into the design criteria for required test circuitry, undervoltage-sensing relays, and simulated accident test signals to permit testing as outlined in Regulatory Guide 1.41.

Subsequent to the initial service date, the station standard maintenance and test procedures require that any major modification or repair to the onsite power systems or redundant load groups necessitates retesting of these systems to verify independence among redundant systems.

The design basis for the maintenance, testing, and replacement of plant batteries is in compliance with the recommendations set forth in IEEE 450-1995 [17], as discussed in Section 8.3.2.

Inspection of the batteries are described in Section 8.3.

It is not proposed to test equipment of a passive nature, such as primary and secondary shielding.

Qualification tests for electric penetration assemblies are based on References [10 and 11].

Qualification tests for electric cables are based on References [11 and 16]. Test temperature profiles used for environmental qualification and included in the applicable EEQSP.

For Class 1E equipment located in a mild environment area, CPNPP will demonstrate qualification through procurement documents, which include the most stringent requirements for the equipment's end use. The maintenance and surveillance programs in conjunction with a trending program provide the assurance that equipment which meets the requirements of the procurement documents is maintained throughout the equipment's installed life (see Appendix 3A, Section 2.4).

All seismic Category I Class 1E equipment located in a mild environment will be seismically qualified to IEEE Standard 344-1975 and Regulatory Guide 1.100 [19] using the methods, procedures, and documentation described in FSAR Sections 3.7 and 3.10B.

How the requirements of GDC 1, 4, 23 and 50 are met is addressed in Section 3.1. Specific information concerning how GDC 1 and 4 are met is reported in Appendix A of Reference [18]. Specific information concerning how GDC 23 is met can be found in Sections 7.2 and 7.3.

The quality assurance program to be applied to the design, fabrication and testing of all BOP safety related equipment conform to the requirements of 10CFR part 50 Appendix B. The QA program is described in Chapter 17.

Specific information concerning how Appendix B of 10 CFR Part 50 is met is discussed in Chapter 17. Level of compliance with NRC Regulatory Guides 1.30, 1.40, 1.63, 1.73, 1.89, 1.100 and 1.131 is addressed in Appendix 1A(B).

3.11B.3 QUALIFICATION TEST RESULTS

All safety-related equipment and components in a potentially harsh environment are demonstrated to perform their designed safety function under all normal, abnormal and accident conditions, by appropriate testing and analyses.

The detailed qualification information and test results are documented in the Environmental Equipment Qualification Summary Packages.

3.11B.4 LOSS OF VENTILATION

3.11B.4.1 Environmental Design Basis

The plant design features ensure that room ambient temperatures do not exceed the maximum operational temperature limit for instrumentation and electrical equipment. The design conditions for normal plant operation are given in Table 9.4-2 for each room/area.

1. Control Room Ventilation System

   The Control Room air-conditioning system is designed to maintain its ambient temperature at 75°F (±5°F) and 35-50 percent relative humidity. The system is of seismic Category I design and includes the following design features:

   a. Sufficient redundancy in equipment and power supplies is provided to enable the system to sustain a single active component failure without loss of safety function.

   b. Redundant fans are connected to separate Class 1E buses as described in Section 8.3.

   c. Instrumentation and controls which incorporate audible and visual alarms enable the operator to continuously monitor system performance and alert him in the event of system malfunction.

   d. Failure modes for isolation valves and dampers (as described in Section 9.4.1) are set so that their failure does not render the system inoperable.

2. Primary Plant Ventilation

   a. Areas with Emergency Fan Coil Units.

      Certain areas with safety related equipment are provided with Class 1E emergency fan coil units which are powered from the same train as the equipment being cooled.

   b. Areas without Emergency Fan Coil Units.

Class 1E equipment in these areas are qualified for the temperature resulting from loss of ventilation (following a LOCA with Loss of Offsite Power).

3. Ventilation Systems of Other Areas

   Other ventilation systems which serve areas containing safety-related equipment are described in Section 9.4.

3.11B.4.2 Limiting Environmental Conditions

1. UPPER LIMITS

The normal upper limit environmental conditions for the Control Room are as follows:

Temperature, °F	80
Relative humidity %	50
Chemistry	N/A
Radiation environment	Refer to plant drawings and calculations

All other areas upper limit environmental conditions for safety related equipment are shown in plant drawings and calculations.

2. LOWER LIMITS

The lower limit operating environment that Class 1E equipment is expected to operate in is specified in Table 9.4-2 and augmented by the following:

a. The lower limit operating temperature:
40°F for all areas outside containment except the Control Room and Battery Room;
70°F for the Control Room and Battery Room;
50°F for all areas inside containment.

The IE Devices outside of a permanent building or in an unheated enclosure are qualified for the Outdoor Design Conditions, protected or analyzed as acceptable as is. The Extreme Outdoor Temperatures have also been analyzed to have no effect on these components.

b. The lower limit operating relative humidity:
uncontrolled

c. The lower limit operating pressure:
atmospheric for all areas except inside containment;
-0.5 psig inside containment.

3.11B.4.3 Testing of Control and Electrical Equipment

Factory testing of protective system equipment is performed as stated in Section 3.11B.2. Periodic testing is provided in accordance with IEEE 338, “Criteria for the Periodic Testing of Nuclear Power Generation Station Protection Systems” [14], and IEEE 279, “Criteria for Protection Systems” [8] as described in Chapters 7 and 8.

The previously stated criteria for onsite testing are applied to all components of protection systems and are performed under ambient environmental conditions.

Overall testing activities are described in quality assurance programs identified in Chapter 17.

3.11B.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

3.11B.5.1 Chemical Environment

The majority of pressure boundary components are constructed principally of stainless steel. Containment structures are principally constructed of concrete and carbon steel (galvanized or coated). These materials are compatible with, and do not suffer significant degradation in the containment spray environment. Other pressure boundary component (i.e. gaskets) materials are also selected for their compatibility with the spray environment.

Details concerning the chemicals used in and the chemistry of the containment spray system is presented in Sections 6.2.2 and 6.5.2. Accident chemistry qualification of safety-related electrical and instrumentation components in a potentially harsh environment (i.e. inside containment) is shown in the EEQSPs.

The atmosphere inside the containment building after a LOCA consists of steam-air-water spray mixture. The water spray solution is borated water with a maximum of 2600 ppm boron buffered with a sodium hydroxide solution (i.e. 30% wt. NaOH in water) to a final spray solution pH between 8.25 to 10.5.

3.11B.5.2 Radiation Environment

The radiation environment for normal operation and for the DBA for each safety-related component has been determined. The accident source terms are based on the TID-14844 [18] release model and, accordingly, are consistent with NRC Regulatory Guide 1.4 [1].

The postulated integrated normal/accident dose is given in the EEQSPs.

REFERENCES

1. NRC Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Revision 2, 6/74.

2. NRC Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment, 8/11/72.

3. NRC Regulatory Guide 1.40, Qualification Tests of Continuous- Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants, 3/16/73.

4. NRC Regulatory Guide 1.41, Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments, 3/16/73.

5. NRC Regulatory Guide 1.63, Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants, Revision 2, 7/78.

6. NRC Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants, 1/74.

7. NRC Regulatory Guide 1.89, Qualification of Class 1E Equipment for Nuclear Power Plants, 11/74.

8. IEEE 279-1971, Criteria for Protection Systems for Nuclear Power Generating Stations, sponsored by the IEEE Joint Committee on Nuclear Power Standards.

9. IEEE 308-1971, Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.

10. IEEE 317-1976, Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations.

11. IEEE 323-1974, Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

12. IEEE 334-1974, Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations.

13. IEEE 336-1971, Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations.

14. IEEE 338-1971, Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems, prepared by Subcommittee 3 of the IEEE Joint Committee on Nuclear Power Standards, Revision No. 15, May 15, 1971.

15. IEEE 382-1972, IEEE Trial-Use Guide for Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations.

16. IEEE 383-1974, Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations.

17. IEEE 450-1995, IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead Acid Batteries for Stationary Applications.

18. TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, by J.J. DiNunno, F.D. Anderson, R.E. Bauer, and R.L. Waterfield, Division of Licensing and Regulation, United States Nuclear Regulatory Commission.

19. NRC Regulatory Guide 1.100, Seismic Qualification of Electric Equipment for Nuclear Power Plants, 8/77.

20. TXX-88742, Letter from W. G. Counsil to USNRC dated October 24, 1988.

TABLE 3.11B-1
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TABLE 3.11B-2
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TABLE 3.11B-3
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TABLE 3.11B-4
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3A ENVIRONMENTAL QUALIFICATION REPORT

3A.1 INTRODUCTION

This Environmental Qualification Report (EQR) supports the operating license application for Comanche Peak Nuclear Power Plant (CPNPP). This report provides design information in sufficient detail to allow a definitive evaluation of the equipment environmental qualification program for CPNPP.

The CPNPP environmental qualification program is described in Section 3A.2 and a detailed comparison to NUREG-0588 is provided in Section 3A.3. A summary of Class 1E equipment types located in a potentially harsh environment and their postulated environmental extremes are provided in controlled documents (e.g., EEQSPs, EQML).

The basic scope of the program includes all Class 1E equipment at CPNPP located in a potentially harsh environment. The definition of a potentially harsh environment at CPNPP is provided in Section 3.11B.1. Detailed system and equipment descriptions are included in the appropriate sections of the FSAR; in particular, those systems required for Hot Standby and Cold Shutdown are described in FSAR Section 7.4. Section 7.1 identifies all safety-related systems.

The basic scope of the CPNPP environmental qualification program which will qualify equipment to the standards of IEEE 344-1975 and IEEE 323- 1974, includes all safety related Class 1E equipment located in a potentially harsh environment, designated Accident Monitoring equipment (see FSAR Section 7.5), and any other electrical equipment which CPNPP has committed to qualify in accordance with these criteria. Certain non-safety related equipment, if subjected to adverse environmental conditions (such as high energy line breaks), could impact safety analyses and the adequacy of the protection functions performed for by safety grade equipment. Such equipment (e.g., Automatic rod control system, the pressurizer PORV control system, main feedwater control system and the steam generator PORV control system) may be required to be qualified to operate in this environment or be located in areas not subject to those environments. All safety-related systems are identified in FSAR Section 7.1. Detailed system and equipment descriptions are included in the appropriate sections of the FSAR; in particular, the equipment required for Hot Standby and Cold Shutdown is described in FSAR Section 7.4. A list of major equipment which is classified as IEEE Class 1E is provided in FSAR Table 17A-1.

A review has been conducted of the CPNPP Emergency Operating procedures. The review identified all display instrumentation referenced by the emergency operating procedures and listed the safety classification for these instruments. This review verified that the monitoring instrumentation as described in FSAR Section 7.5 is adequate for the execution of the emergency operating procedures.

Modifications to the CPNPP design due to the incorporation of TMI lessons learned have been included into the Environmental Qualification Program, as applicable.

3A.2 CPNPP QUALIFICATION PROGRAM

The equipment qualification program for CPNPP can be discussed as two separate programs. The first program covers the Balance of Plant (BOP) equipment and the second program covers the Nuclear Steam Supply System (NSSS) equipment. Of course, many aspects of these two programs are similar and will be discussed only once under the general category of the CPNPP

qualification program. Where it is better to present the information separately, the two programs shall be referred to as the NSSS qualification program and the BOP qualification program. The final review of all equipment qualification programs is referred to as the CPNPP qualification review program and is described in Section 3A.2.3 below.

3A.2.1 NSSS QUALIFICATION PROGRAM

The NSSS qualification program covers the Class 1E Electrical Equipment located in a potentially harsh environment supplied to CPNPP by the Westinghouse Electric Corporation under the CPNPP NSSS contract. This program is basically described in WCAP-8587.

3A.2.2 BOP QUALIFICATION PROGRAM

The BOP qualification program covers all Class 1E Electrical Equipment located in a potentially harsh environment not included in the NSSS qualification program.

The BOP and NSSS qualification programs are comprised of approximately 70 individual qualification packages. In each of the individual packages, the functional requirements and environmental conditions for the Class 1E electrical equipment are defined. The equipment vendor used the functional requirements and environmental conditions specified in procurement documents to develop the individual qualification program for his equipment. The vendor's program is reconciled in the individual qualification packages.

Some individual programs were "generic" programs. These programs cover a range of environmental conditions and functional requirements and were not conducted specifically for CPNPP. For these generic programs, the vendor usually provides the generic report along with documentation to show that the generic report adequately envelopes the CPNPP requirements and that the report applies to the equipment supplied to CPNPP. This documentation is submitted to Luminant Power for review, comment and acceptance.

Many individual qualification programs were conducted specifically for CPNPP. For these programs, a qualification program is developed by the vendor. These plans are submitted to Luminant Power for review, comment and acceptance. Using these plans as a basis, the testing and/or analysis is conducted and a final report is submitted for acceptance. These programs should normally result in an accepted qualification plan and an accepted qualification report. In some cases, the qualification report will contain all the required information in the plan and thus become a combined plan/report. A combined plan/report will usually supercede all previous plans.

3A.2.3 CPNPP QUALIFICATION REVIEW PROGRAM

The CPNPP qualification review program provides a final review of the BOP and NSSS programs in order to accept these programs for CPNPP. The review process consists of an overall review of the programs as well as a specific review of the environmental qualification for each type of Class 1E electrical equipment located in a potentially harsh environment.

The overall reviews of the programs are documented in Section 3A.3 where each program (BOP and NSSS) is compared to NUREG-0588. Each item by item comparison describes how the respective program meets each requirement of NUREG-0588.

All environmental qualification data are contained in specific Environmental Equipment Qualification Summary Packages (EEQSPs) which are maintained in the Environmental Qualification Files. These auditable files are updated and kept current as equipment is qualification tested, replaced or modified. Each EEQSP is reviewed and approved by Luminant Power.

3A.2.4 CPNPP QUALIFICATION MAINTENANCE PROGRAM

The Qualification Maintenance Program at CPNPP is designed to be integrated into the existing maintenance program while assuring that all qualified life contingencies and qualification requirements are fulfilled. The Qualification Maintenance Program consists of maintenance activities which in many cases, occur normally at a power plant, but that are formalized in the case of Class 1E electrical equipment located in a potentially harsh environment.

All maintenance activities that are required by the equipment qualification reports for equipment located in a potentially harsh environment shall be performed within the time intervals specified. In addition, this equipment shall be replaced before their qualified lives expire. As an alternative to replacement, it may be possible to extend qualified lives through additional testing or analysis. Any requalification of equipment will be controlled and documented in a process similar to the original qualification effort.

A maintenance and surveillance program shall be implemented to identify and prevent significant age related degradation in electrical and mechanical equipment. Provisions for preventing or detecting age related degradation in safety grade equipment are specified and include the following:

1. Development of a maintenance program to maintain safety related equipment at the quality required for it to perform its intended function.

2. Evaluation of causes of equipment failures and review of experience with failed equipment and similar components to determine whether a replacement component of the same type can be expected to perform its function reliably.

3. Inspection and performance testing to ensure an appropriate quality level.

4. Development of preventive maintenance schedules that describe the frequency and type of maintenance to be performed. A preliminary schedule should be developed early in plant life and it should be refined and changed as experience is gained with the equipment.

5. Surveillance testing related to the results of reliability analyses, frequency and type of service, or age of the item or system, as appropriate.

Whenever possible, exact replacement spare parts and replacements shall be specified for Class 1E equipment. An exact replacement part shall be considered to be one that has been evaluated by engineering to confirm that the form, fit and functional characteristics satisfy the original design basis. Parts with expired shelf lives will not be used on Class 1E equipment at CPNPP.

The beginning of life date for Class 1E electrical equipment located in a potentially harsh environment at CPNPP shall be considered to be the date of the initial fuel load. For equipment with short qualified lives, an engineering analysis shall be performed to determine if a portion of a qualified life has been used before the initial fuel load date. If this is determined, the initial qualified life of this equipment shall be adjusted to account for this age related degradation or the equipment shall be refurbished.

3A.3 CPNPP COMPARISON TO NUREG-0588

The CPNPP environmental qualification program has been reviewed using the criteria provided in NUREG-0588. The results of this review are provided in Tables 3A.3-1 and 3A.3-2. These tables describe the BOP and NSSS qualification programs, respectively, relative to each criteria in NUREG-0588. As defined in Section 3A.1, the scope of this program does not include equipment which is located in a mild environment area.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 1 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
1.	ESTABLISHMENT OF THE QUALIFICATION PARAMETER FOR DESIGN BASIS EVENTS	1.	
1.1)	Temperature and Pressure Conditions Inside Containment - Loss-of-Coolant Accident (LOCA)	1.1	
(1)	The time-dependent temperature and pressure, established for the design of the containment structure and found acceptable by the staff, may be used for environmental qualification of equipment.	(1)	A spectrum of high energy line breaks in the primary and secondary systems was considered in developing the maximum containment and temperature envelopes. The temperature and pressure envelopes which were used for the qualification equipment are provided in design documents. The development of the individual temperature and pressure profiles for containment (for specific accident scenarios) is discussed in FSAR Section 6.2.
(2)	Acceptable methods for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified are summarized below. Acceptable methods for calculating mass and energy release rates are summarized in Appendix A.	(2)	See (1) above.
Pressurized Water Reactors (PWRs)			
Dry Containment - Calculate LOCA containment environment using CONTEMPT-LT or equivalent industry codes. Additional guidance is provided in Standard Review Plan (SRP) Section 6.2.1.1.A, NUREG-75/087.			

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 2 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
Ice Condenser Containment - Calculate LOCA containment environment using LOTIC or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.A, NUREG-75/087.			
Boiling Water Reactors (BWRs)			
Mark I, II and III Containment - Calculate environment using methods of GESSAR Appendix 3B or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.C, NUREG-75/087.			
(3)	In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser types of plants, the generic envelope shown in Appendix C may be used for qualification testing.	(3)	Not applicable
(4)	The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4)	See (1) above
1.2	Temperature and Pressure Conditions Inside Containment-Main Steam Line Break (MSLB)	1.2	
(1)	The environmental parameters used for equipment qualification should be calculated with a plant-specific model reviewed and approved by the staff.	(1)	See 1.1(1) above.
(2)	Models that are acceptable for calculating containment parameters are listed in Section 1.1(2).	(2)	See 1.1(1) above.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 3 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(3)	In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser plants, the generic envelope shown in Appendix C may be used.	(3)	Not applicable
(4)	The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4)	See 1.1(1) above.
(5)	Where qualifications has been completed but only LOCA conditions were considered, it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:	(5)	The methods of 1.1(1) were used in determining the specified environment
(a)	Calculate the peak temperature envelope from an MSLB using a model based on the staff's approved assumptions defined in Section 1.1(2)		
(b)	Show that the peak surface temperature of the component to be qualified does not exceed the LOCA qualification temperature by the method discussed in item 2 of Appendix B.		

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 4 of 21)

NUREG-0588 Category I Position	CPNPP STATUS
(c) If the calculated surface temperature exceeds the qualification temperature, the staff requires that (i) requalification testing be performed with appropriate margins, or (ii) qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature. For plants that are currently being reviewed, or will be submitted for an operating license review within six months from issue date of this report, compliance with items (i) or (ii) above may represent a substantial impact. For those plants, the staff will consider additional information submitted by the applicant to demonstrate that the equipment can maintain its functional operability if its surface temperature rises to the value calculated.	
1.3 Effects of Chemical Spray The effects of caustic spray should be addressed for the equipment qualification. The concentration of cuastics used for qualification should be equivalent to or more severe than those used in the plant containment spray system. If the chemical composition of the caustic spray can be affected by equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Section 6.5.2 (NUREG-75/087), paragraph II, item (e) for caustic spray solution guidelines.	1.3 The effect of caustic spray was addressed for the qualification of Class 1E equipment inside containment which is required to function during or after an accident that can initiate containment spray. See FSAR Section 3.11B.5.1.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 5 of 21)

NUREG-0588 Category I Position	CPNPP STATUS
1.4 Radiation Conditions Inside and Outside Containment The radiation environment for qualification of equipment should be based on the normally expected radiation environment over the equipment qualified life, plus that associated with the most severe design basis accident (DBA) during or following which that equipment must remain functional. It should be assumed that the DBA related environmental conditions occur at the end of the equipment qualified life. The sample calculations in Appendix D and the following positions provide an acceptable approach for establishing radiation limits for Qualification. Additional radiation margins identified in Section 6.3.1.5 of IEEE Std. 323-1974 for qualification type testing are not required if these methods are used.	1.4 The specified radiation environment includes the expected normal dose for the qualified life of the equipment and the most severe DBA dose following which that equipment is required to remain functional. The accident dose is assumed to occur at the end of the qualified life of the equipment.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 6 of 21)

NUREG-0588 Category I Position	CPNPP STATUS
(1) The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products. For all other non-LOCA design basis accident conditions, a source term involving an instantaneous release from the fuel to the atmosphere of 10 percent of the noble gases (except Kr-85 for which a release of 30 percent should be assumed) and 10 percent of the iodines is acceptable.	(1) The same conservative source terms, namely 100 percent of noble gases, and 50 percent of the iodines are assumed to be instantaneously released from the fuel to the containment atmosphere for the design basis LOCA. The 1 percent of the remaining fission products are assumed to be released from the fuel and carried with the primary coolant to the containment sump, as given in Appendix D NUREG-0588. These assumptions are used for all DDA's for which core damage must be assumed, assumed, except fuel handling accidents. For fuel handling accidents, the guidance of Regulatory Guide 1.25 is followed.
(2) The calculation of the radiation environment associated with design basis accidents should take into account the time-dependent transport of released fission products within various regions of containment and auxiliary structures	(2) The time dependent transport of released fission products has been considered in calculation of the radiation environment associated with DBAs
(3) The initial distribution of activity within the containment should be based on a mechanistically rational assumption. Hence, for compartmented containments, such as in a BWR, a large portion of the source should be assumed to be initially contained in the drywell. The assumption of uniform distribution of activity throughout the containment at time zero is not appropriate.	(3) Both models of assuming uniform distribution in the entire containment free volume and uniform distribution in the containment volume excluding the relatively isolated compartments at time zero after LOCA are analyzed. The latter generates more conservative doses and the results are used as the required radiation environment.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 7 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(4)	Effects of ESF systems, such as containment sprays and containment ventilation and filtration systems, which act to remove airborne activity and redistribute activity within containment, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Section 15.6.5 (NUREG-75/087 and the related sections referenced in the Appendices to that section.	(4)	A two-region spray model is utilized to calculate the airborne iodine activity in containment atmosphere. The assumption and limits of spray removal credit are consistent with Appendix D of NUREG-0588. Effects of ventilation and filtration have not been considered.
(5)	Natural deposition (i.e., plate-out) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters. The assumption of 50 percent instantaneous plate-out of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow or washoff by the containment spray may be assumed if such effects can be justified and quantified by analysis or experiment.	(5)	Plateout of airborne iodines on containment heat-sink surfaces is modeled as first-order rate removal process using a calculated plate-out removal constant. Once plated out, the iodines remain as the surface source, reduced only by the radioactive decay process.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 8 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(6)	For unshielded equipment located in the containment, the gamma dose and dose rate should be equal to the dose and dose rate at the centerpoint of the containment plus the contribution from location dependent sources such as the sump water and plate-out, unless it can be shown by analyses that location and shielding of the equipment reduces the dose and dose rate.	(6)	Gamma doses and dose rates have been calculated for the center point of the containment, at the surface of containment sump water and as a function of distance from the plateout sources. However, in shielded compartments or other areas where the dose rate is reduced due to location of shielding effects, computations have been performed for specific configurations to determine the post-accident integrated dose. The sources used in these computations include the airborne source, plateout source and sump water source in the containment sump as appropriate in addition to the recirculated emergency core cooling system.
(7)	For unshielded equipment, the beta doses at the surface of the equipment should be the sum of the airborne and plate-out sources. The airborne beta dose should be taken as the beta dose calculated for a point at the containment center.	(7)	The beta doses have been calculated at containment center point, sump water surface and as a function of distance from the plateout source. The doses from airborne, plateout source, and/or sump source are summed. The airborne beta dose at containment center is calculated using an infinite medium model. The semi-finite medium approximation is used for the airborne beta dose at sump water surface. For a dose point close to the containment wall. The airborne beta source includes a semi-infinite medium plus an infinite slab source of finite thickness.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 9 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(8)	Shielded components need be qualified only to the gamma radiation levels required, provided an analysis or test shows that the sensitive portions of the component or equipment are not exposed to beta radiation or that the effects of beta radiation heating and ionization have no deleterious effects on component performance.	(8)	Equipment is qualified to the gamma and beta dose specified unless a specific test or analysis is provided to show that the beta dose has no deleterious effects on the equipment performance.
(9)	Cables arranged in cable trays in the containment should be assumed to be exposed to half the beta radiation dose calculated for a point at the center of the containment plus the gamma ray dose calculated in accordance with Section 1.4(6). This reduction in beta dose is allowed because of the localized shielding by other cables plus the cable tray itself.	(9)	A semi-infinite medium has been used to calculate the airborne beta dose where justified.
(10)	Paints and coatings should be assumed to be exposed to both beta and gamma rays in assessing their resistance to radiation. Plate-out activity should be assumed to remain on the equipment surface unless the effects of the removal mechanisms, such as spray wash-off or steam condensate flow, can be justified and quantified by analysis or experiment.	(10)	Failure of the paint or coating on class 1E equipment will not effect the ability of class 1E equipment to perform its required function and the CPNPP design is such that the failure of these coating's will not effect the operation of safety systems inside containment; therefore, no additional special calculations were performed relative to the coatings on class 1E equipment.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 10 of 21)

NUREG-0588 Category I Position	CPNPP STATUS
(11) Components of the emergency core cooling system (ECCS located outside containment (e.g., pumps, valves, seals and electrical equipment) should be qualified to withstand the radiation equivalent to that penetrating the containment, plus the exposure from the sump fluid using assumptions consistent with the Appendix K to 10 CFR Part 50.	(11) All safety related Class 1E electrical equipment, including ECCS Components, located in a harsh environment outside containment are qualified to a total radiation level which included radiation penetrating the containment wall or streaming through containment openings, from sump fluid contained in recirculation piping outside containment and from 40 year normal power operation.
(12) Equipment that may be exposed to radiation doses below 10^4 rads should not be considered to be exempt from radiation qualification, unless analysis supported by test data is provided to verify that these levels will not degrade the operability of the equipment below acceptable values.	(12) For each piece of Class 1E equipment regardless of environment, the specified radiation dose is addressed.
(13) The staff will accept a given component to be qualified provided it can be shown that the component has been qualified to integrated beta and gamma doses which are equal to or higher than those levels resulting from an analysis similar in nature and scope to that included in Appendix D (which uses the source term given in item (1) above), and that the component incorporates appropriate factors pertinent to the plant design and operating characteristics, as given in these general guidelines.	(13) Class 1E equipment for CPNPP is qualified to integrated radiation levels greater than or equal to the levels resulting from analyses similar to Appendix D

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 11 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(14)	When a conservative analysis has not been provided by the applicant for staff review, the staff will use the radiation environment guidelines contained in Appendix D, suitably corrected for the differences in reactor power level, type, containment size, and other appropriate factors	(14)	A conservative analysis similar to Appendix D has been performed to determine the required radiation environment.
1.5	Environmental Conditions for Outside Containment	1.5	
(1)	Equipment located outside containment that could be pipe breaks should be qualified to the conditions resulting from the accident for the duration required. The techniques to calculate the environmental parameters described in Sections 1.1 through 1.4 above should be applied.	(1)	HELB's outside containment are described in FSAR Section resulting form the HELB's will be incorporated into the specified environment for the Class 1E equipment per Section 5.
(2)	Equipment located in general plant areas outside containment where equipment is not subjected to a design basis accident environment should be qualified to the normal and abnormal range of environmental conditions postulated to occur at the equipment location.	(2)	Class 1E equipment which is not subject to a design basis environment is qualified to the normal and abnormal range of environmental conditions postulated for that equipment.
(3)	Equipment not served by Class 1E environmental support systems, or served by Class 1E support systems that may be secured during plant operation or shutdown, should be qualified to the limiting environmental conditions that are postulated for that location, assuming a loss of the environmental support system.	(3)	Class 1E equipment which is supported by non-Class 1E environmental support systems (or secured Class 1E systems) is qualified to the limiting environmental conditions that are postulated assuming the loss of those support systems (provided that the function or integrity of this Class 1E equipment is required under these conditions).

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 12 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
2.	QUALIFICATION METHODS	2.	
2.1	Selection of Methods	2.1	
(1)	Qualification methods should conform to the requirements defined in IEEE Std. 323-1974.	(1)	The qualification methods used conform to IEEE Standard 323-1974.
(2)	The choice of the methods selected is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident Environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless (a) testing of the component is impractical due to size limitations, and (b) partial type test data is provided to support the analytical assumptions and conclusions reached.	(2)	Equipment subjected to an accident environment Is qualified using test data or at least partial type test data to support the analytical assumptions and conclusions.
(3)	The environmental qualification of equipment exposed to DBA environments should conform to the following positions. The basis should be provided for the time interval required for operability of this equipment. The operability and failure criteria should be specified and the safety margins defined.	(3)	The time interval required for operability was based on the operability assumptions used in the various applicable accident analysis. The operability requirements and safety margins are provided in various locations including the appropriate sections of the FSAR, purchase specifications, and qualification documentation. CPNPP uses failure to meet operability requirements as its failure criteria, unless specific failure criteria is provided in the qualification documentation.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 13 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(a)	Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident.	(a)	Class 1E equipment is qualified to function in the environment resulting from the accident it is designed to mitigate.
(b)	Any equipment (safety-related or non-safety-related) that need not function in order to mitigate any accident but that must not fail in a manner detrimental to plant safety should be qualified by test to demonstrate its capability to withstand any accident environment for the time during which it must not fail.	(b)	Class 1E equipment that need not function but must not fail during an accident is qualified not to fail due to the environment conditions resulting from that accident, per paragraph (1) above.
(c)	Equipment that need not function in order to mitigate any accident and whose failure in any mode in any accident environment is not detrimental to plant safety need only be qualified for its non-accident service environment. Although actual type testing is preferred, other methods when justified may be found acceptable. The bases should be provided for concluding that such equipment is not required to function in order to mitigate any accident, and that its failure in any mode in any accident environment is not detrimental to plant safety.	(c)	Class 1E equipment whose function is only required in non-accident environment conditions and whose failure in any mode in an accident environment is not detrimental to plant safety is only analyzed to its non-accident environments.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 14 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(4)	For environment qualification of equipment subject to events other than a DBA, which result in abnormal environmental conditions, actual type testing is preferred. However, analysis or operating history, or any applicable combination thereof, coupled with partial type test data may be found acceptable, subject to the applicability and detail of information provided.	(4)	For Class 1E equipment which is not subjected to environmental conditions resulting directly from a DBA, type testing is normally used to demonstrate qualification. When alternate or combination methods are used, the methods are justified.
2.2	Qualification by Test	2.2	
(1)	The failure criteria should be established prior to Testing.	(1)	Failure criteria may be specifically specified or implied as not meeting the acceptance criteria.
(2)	Test results should demonstrate that the equipment can perform its required function for all service conditions postulated (with margin) during its installed life.	(2)	Test results demonstrate that the equipment can perform its required function during its qualified life and during all required service conditions.
(3)	The items described in Section 6.3 of IEEE Std. 323-1974 supplemented by items (4) through (12) below constitute acceptable test procedures.	(3)	The guidelines of Section 6.3 of IEEE Std. 323-1974 were used in establishing test procedures.
(4)	When establishing the simulated environmental profile for qualifying equipment located inside containment, it is preferred that a single profile be used that envelopes the environmental conditions resulting from any design basis event during any mode of plant operation (e.g., a profile that envelopes the conditions produced by the main steamline break and loss-of-coolant accidents).	(4)	A single composite profile was used for the simulation of DBA (LOCA/ MSLB) environmental conditions with adequate margin.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 15 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(5)	Equipment should be located above flood level or protected against submergence by locating the equipment in qualified watertight enclosures. Where equipment is located in watertight enclosures, qualification by test or analysis should be used to demonstrate the adequacy of such protection. Where equipment could be submerged, it should be identified and demonstrated to be qualified by test for the duration required.	(5)	See FSAR Section 3.11B.1.
(6)	The temperature to which equipment is qualified, when exposed to the simulated accident environment, should be defined by thermocouple readings on or as close as practical to the surface of the component being qualified.	(6)	During the simulation of DBA conditions, temperature reading were taken to ensure as best as possible that the environment in the vicinity of the test component is the same as the environment required by the test envelope.
(7)	Performance characteristics of equipment should be verified before, after, and periodically during testing throughout its range of required operability.	(7)	Functional testing is performed at various points to verify the required operability of the equipment.
(8)	Caustic spray should be incorporated during simulated event testing at the maximum pressure and at the temperature conditions that would occur when the onsite spray systems actuate.	(8)	Caustic spray is incorporated in the testing during the portion of the accident envelope that best simulates DBA conditions.
(9)	The operability status of equipment should be monitored continuously during testing. For long-term testing, however monitoring at discrete intervals should be justified if used.	(9)	The operability status of equipment is monitored as necessary to verify the required operability of the equipment as defined by its Class 1E function.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 16 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(10)	Expected extremes in power supply voltage range and frequency should be applied during simulated event environmental testing.	(10)	Characteristics of the equipment under voltage and frequency extremes is demonstrated during the performance test of the equipment and not necessarily during the simulated event test. These characteristics are re-examined during simulated event testing if there is reason to believe or test evidence that the ability of the equipment to perform under these extremes are changed.
(11)	Dust environments should be addressed when establishing qualification service conditions.	(11)	The dust environment at CPNPP was considered in the drafting of the specifications of Class 1E equipment for CPNPP. The dust condition at CPNPP by design are not considered a significant factor in the qualification of the CPNPP Class 1E equipment.
(12)	Cobalt-60 is an acceptable gamma radiation source for environmental qualification.	(12)	Cobalt-60 and Cesium 137 are used as the radiation sources for environmental qualification.
2.3	Test Sequence	2.3	
(1)	The test sequence should conform fully to the guidelines established in Section 6.3.2 of IEEE Std. 323-1974. The test procedures should insure that the same piece of equipment is used throughout the test sequence, and that the test simulates as closely as practicable the postulated accident environment.	(1)	The test sequence provided in Section 6.3.2 of IEEE Std. 323-1974 was followed. The testing was performed on the same piece of equipment (with the exception that some performance tests may have been run on identical or essentially similar equipment as allowed by Section 6.3.2(3). Where other test sequences are used, suitable justification is provided.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 17 of 21)

NUREG-0588 Category I Position	CPNPP STATUS
2.4 Other Qualification Methods	2.4
Qualification by analysis operating experience implemented, as described in IEEE Std. 323-1974 and other ancillary standards, may be found acceptable. The adequacy of these methods will be evaluated on the basis of the quality and detail of the information submitted in support of the assumptions made and the specific function and location of the equipment. These methods are most suitable for equipment where testing is precluded by physical size of the equipment being qualified. It is required that, when these methods are employed, some partical type tests on vital components of the equipment be provided in support of these methods.	Then qualification is by methods other than type test, adequate support information have been provided to find these methods acceptable. On some equipment, some partial type tests of vital components have been performed to support the overall qualification methods.
3. MARGINS	3.
(1) Quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. These margins should be applied in addition to any margins (conservatism) applied during the derivation of the specified plant parameters.	(1) Adequate margin has been applied to the design parameters in the simulation of postulated DBA conditions during type tests.
(2) In lieu of other proposed margins that may be found acceptable, the suggested values indicated in IEEE Std. 323-1974, Section 6.3.1.5, should be used as a guide. (Note exceptions stated in Section 1.4.)	(2) The margins suggested in Sections 6.3.1.5 of IEEE Std. 323-1974 have been met or exceeded (except as allowed by Section 1.4).

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 18 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(3)	When the qualification envelope in Appendix C is used, the only required margins are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for uncertainties such as production errors and errors associated with defining satisfactory performance (e.g., when only a small number of units are tested).	(3)	Not applicable to CPNPP
(4)	Some equipment may be required by the design to only perform its safety function within a short time period into the event (i.e., within seconds or minutes), and, once its function is complete, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not be required to perform a safety function but must not fail within a short time period into the event, and subsequent failures are also shown not to be detrimental to plant safety. Equipment in these categories is required to remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident. For all other equipment (e.g., post-accident monitoring, recombiners, etc.), the 10 percent time margin identified in Section 6.3.1.5 of IEEE Std. 323-1974 may be used.	(4)	No BOP equipment falls into this category.
4.	<u>AGING</u>	4.	
(1)	Aging effects on all equipment, regardless of its location in the plant, should be considered and included in the qualification program.	(1)	Aging effects were considered for all Class 1E equipment located in a harsh environment at CPNPP. All qualification programs for CPNPP Class 1E equipment address aging.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 19 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(2)	The degrading influences discussed in Sections 6.3.3, 6.3.4 and 6.3.5 of IEEE Std. 323-1974 and the electrical and mechanical stresses associated with cyclic operation of equipment should be considered and included as part of the aging programs.	(2)	Aging effects (including cyclic operation, integrated radiation dose, and vibrations) were included in the aging programs where these degrading influences could effect the required operability of the equipment.
(3)	Synergistic effects should be considered in the accelerated aging programs. Investigation should be performed to assure that no known synergistic effects have been identified on materials that are included in the equipment being qualified. Where synergistic effects have been identified, they should be accounted for in the qualification programs. Refer to NUREG/CR-0276 (SAND 78-0799) and NUREG/CR-0401 (SAND 78-1452), “Qualification Testing Evaluation Quarterly Reports,” for additional information.	(3)	Synergistic effects have been considered in the review of the qualification programs and the aging performed. No known synergistic effects have been defined that could have a significant impact on the qualification of any of the equipment qualified for CPNPP.
(4)	The Arrhenius methodology is considered an acceptable method of addressing accelerated aging. Other aging methods that can be supported by type tests will be evaluated on a case-by-case basis.	(4)	Arrhenius methodology was used in addressing thermal aging.
(5)	Known material phase changes and reactions should be defined to insure that no known changes occur within the extrapolation limits.	(5)	The accelerated aging temperatures and extrapolation limits used in the qualification programs were reviewed. No known material phase changes of concern could be identified for the programs accepted for CPNPP.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 20 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
(6)	The aging acceleration rate used during qualification testing and the basis upon which the rate was established should be described and justified.	(6)	The methodologies for the accelerated aging rates complies with (4) above.
(7)	Periodic surveillance testing under normal service conditions is not considered an acceptable method for on-going qualification, unless the plant design includes provisions for subjecting the equipment to the limiting service environment conditions (specified in Section 3(7) of IEEE Std. 279-1971) during such testing	(7)	Periodic surveillance testing is not used as a method of on-going qualification.
(8)	Effects of relative humidity need not be considered in the aging of electrical cable insulation.	(8)	_____
(9)	The qualified life of the equipment (and/or component as applicable) and the basis for its selection should be defined	(9)	The qualified life for all Class 1E equipment at CPNPP is provided and its basis is defined.
(10)	Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and in other methods that may be reasonably assumed, coupled with good engineering judgement.	(10)	The qualified life is normally based on the accelerated aging (and other aging) performed during testing but may have as a partial basis other factors such as, operating history, stress test performed, analysis of failure modes and aging mechanisms, etc, - all applied with good engineering judgement.

TABLE 3A.3-1
CPNPP COMPLIANCE WITH NUREG-0588 FOR BOP CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 21 of 21)

NUREG-0588 Category I Position		CPNPP STATUS	
5.	QUALIFICATION DOCUMENTATION	5.	
(1)	The staff endorses the requirements stated in IEEE Std. 323-1974 that, “The qualification documentation shall verify that each type of electrical equipment is qualified for its application and meets its specified performance requirements. The basis of qualification shall be explained to show the relationship of all facets of proof needed to support adequacy of the complete equipment. Data used to demonstrate the qualification of the equipment shall be pertinent to the application and organized in an auditable form.”	(1)	Proper qualification documentation is provided for all CPNPP Class 1E equipment located in a harsh environment
(2)	The guidelines for documentation in IEEE Std. 323-1974 when fully implemented are acceptable. The documentation should include sufficient information to address the required information identified in Appendix E. A certificate of conformance by itself is not acceptable unless it is accompanied by test data and information on the qualification program.	(2)	The CPNPP documentation complies with these requirements.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 1 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
1.	ESTABLISHMENT OF THE QUALIFICATION PARAMETER FOR DESIGN BASIS EVENTS	1.	See Table 3A.3-1 for all information on this portion of NUREG-0588.
1.1)	Temperature and Pressure Conditions Inside Containment - Loss-of-Coolant Accident (LOCA)	1.1	See Table 3A.3-1.
(1)	The time-dependent temperature and pressure, established for the design of the containment structure and found acceptable by the staff, may be used for environmental qualification of equipment.	(1)	See Table 3A.3-1.
(2)	Acceptable methods for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified are summarized below. Acceptable methods for calculating mass and energy release rates are summarized in Appendix A.	(2)	See Table 3A.3-1.

Pressurized Water Reactors (PWRs)

Dry Containment - Calculate LOCA containment environment using CONTEMPT-LT or equivalent industry codes. Additional guidance is provided in Standard Review Plan (SRP) Section 6.2.1.1.A, NUREG-75/087.

Ice Condenser Containment - Calculate LOCA containment environment using LOTIC or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.B, NUREG-75/087.

Boiling Water Reactors (BWRs)

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 2 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
Mark I, II and III Containment - Calculate LOCA environment using methods of GESSAR Appendix 3B or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.C, NUREG-75/087.			
(3)	In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser types of plants, the generic envelope shown in Appendix C may be used for qualification testing.	(3)	See Table 3A.3-1.
(4)	The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4)	See Table 3A.3-1.
1.2	Temperature and Pressure Conditions Inside Containment-Main Steam Line Break (MSLB)	1.2	
(1)	The environmental parameters used for equipment qualification should be calculated with a plant-specific model reviewed and approved by the staff.	(1)	See Table 3A.3-1.
(2)	Models that are acceptable for calculating containment parameters are listed in Section 1.1(2).	(2)	See Table 3A.3-1.
(3)	In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser plants, the generic envelope shown in Appendix C may be used.	(3)	See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 3 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(4)	The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4)	See Table 3A.3-1.
(5)	Where qualifications has been completed but only LOCA conditions were considered, it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:	(5)	See Table 3A.3-1.
(a)	Calculate the peak temperature envelope from an MSLB using a model based on the staff's approved assumptions defined in Section 1.1(2).		
(b)	Show that the peak surface temperature of the component to be qualified does not exceed the LOCA qualification temperature by the method discussed in item 2 of Appendix B.		

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 4 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
(c) If the calculated surface temperature exceeds the qualification temperature, the staff requires that (i) requalification testing be performed with appropriate margins, or (ii) qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature. For plants that are currently being reviewed, or will be submitted for an operating license review within six months from issue date of this report, compliance with items (i) or (ii) above may represent a substantial impact. For those plants, the staff will consider additional information submitted by the applicant to demonstrate that the equipment can maintain its functional operability if its surface temperature rises to the value calculated.	

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 5 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
1.3 Effects of Chemical Spray The effects of caustic spray should be addressed for the equipment qualification. The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system. If the chemical composition of the caustic spray can be affected by equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Section 6.5.2 (NUREG-75/087), paragraph II, item (e) for caustic spray solution guidelines.	1.3 The maximum concentration of boron employed for containment spray is 2600 ppm and the maximum permitted pH of the final spray solution is 10.5. For qualification testing Westinghouse specifies a chemical spray of 2500 ppm boron buffered with 0.9 dissolved sodium hydroxide to maintain a pH of approximately 10.7, starting at time zero and terminating after 24 hours. This spray concentration results in an increase in alkalinity of at least 10% compared to the maximum concentration defined by the specification and significantly exceeds the range of sump pH values permitted long-term by the technical specification. See also Table 3A.3-1.
1.4 Radiation Conditions Inside and Outside Containment The radiation environment for qualification of equipment should be based on the normally expected radiation environment over the equipment qualified life, plus that associated with the most severe design basis accident (DBA) during or following which that equipment must remain functional. It should be assumed that the DBA related environmental conditions occur at the end of the equipment qualified life.	1.4 See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 6 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
	The sample calculations in Appendix D and the following positions provide an acceptable approach for establishing radiation limits for qualification. Additional radiation margins identified in Section 6.3.1.5 of IEEE Std. 323-1974 for qualification type testing are not required if these methods are used.		
(1)	The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products. For all other non-LOCA design basis accident conditions, a source term involving an instantaneous release from the fuel to the atmosphere of 10 percent of the noble gases (except Kr-85 for which a release of 30 percent should be assumed) and 10 percent of the iodines is acceptable.	(1)	See Table 3A.3-1.
(2)	The calculation of the radiation environment associated with design basis accidents should take into account the time-dependent transport of released fission products within various regions of containment and auxiliary structures.	(2)	See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 7 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(3)	The initial distribution of activity within the containment should be based on a mechanistically rational assumption. Hence, for compartmented containments, such as in a BWR, a large portion of the source should be assumed to be initially contained in the drywell. The assumption of uniform distribution of activity throughout the containment at time zero is not appropriate.	(3)	See Table 3A.3-1.
(4)	Effects of ESF systems, such as containment sprays and containment ventilation and filtration systems, which act to remove airborne activity and redistribute activity within containment, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Section 15.6.5 (NUREG-75/087) and the related sections referenced in the Appendices to that section.	(4)	See Table 3A.3-1.
(5)	Natural deposition (i.e., plate-out) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters. The assumption of 50 percent instantaneous plate-out of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow or washoff by the containment spray may be assumed if such effects can be justified and quantified by analysis or experiment.	(5)	See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 8 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(6)	For unshielded equipment located in the containment, the gamma dose and dose rate should be equal to the dose and dose rate at the centerpoint of the containment plus the contribution from location dependent sources such as the sump water and plate-out, unless it can be shown by analyses that location and shielding of the equipment reduces the dose and dose rate.	(6)	See Table 3A.3-1.
(7)	For unshielded equipment, the beta doses at the surface of the equipment should be the sum of the airborne and plate-out sources. The airborne beta dose should be taken as the beta dose calculated for a point at the containment center.	(7)	See Table 3A.3-1.
(8)	Shielded components need be qualified only to the gamma radiation levels required, provided an analysis or test shows that the sensitive portions of the component or equipment are not exposed to beta radiation or that the effects of beta radiation heating and ionization have no deleterious effects on component performance.	(8)	See Table 3A.3-1.
(9)	Cables arranged in cable trays in the containment should be assumed to be exposed to half the beta radiation dose calculated for a point at the center of the containment plus the gamma ray dose calculated in accordance with Section 1.4(6). This reduction in beta dose is allowed because of the localized shielding by other cables plus the cable tray itself.	(9)	See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 9 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(10)	Paints and coatings should be assumed to be exposed to both beta and gamma rays in assessing their resistance to radiation. Plate-out activity should be assumed to remain on the equipment surface unless the effects of the removal mechanisms, such as spray wash-off or steam condensate flow, can be justified and quantified by analysis or experiment.	(10)	See Table 3A.3-1.
(11)	Components of the emergency core cooling system (ECCS) located outside containment (e.g., pumps, valves, seals and electrical equipment) should be qualified to withstand the radiation equivalent to that penetrating the containment, plus the exposure from the sump fluid using assumptions consistent with the Appendix K to 10 CFR Part 50.	(11)	See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 10 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
(12) Equipment that may be exposed to radiation doses below 10^4 rads should not be considered to be exempt from radiation qualification, unless analysis supported by test data is provided to verify that these levels will not degrade the operability of the equipment below acceptable values.	(12) For safety-related electrical equipment that is not required to operate in a high energy line break (HELB) environment and for which the anticipated qualified life integrated radiation dose is 10^4 Rads or less, Westinghouse did not include a radiation aging simulation as part of any qualification testing. For such equipment and components, a supplement to WCAP-8587 demonstrates, based on available test information on materials and components, that up to approximately 10^5 Rads there is no detectable effect on the structural characteristics of materials and components that would affect the capability of equipment to perform during a seismic event. See also Table 3A.3-1.
(13) The staff will accept a given component to be qualified provided it can be shown that the component has been qualified to integrated beta and gamma doses which are equal to or higher than those levels resulting from an analysis similar in nature and scope to that included in Appendix D (which uses the source term given in item (1) above), and that the component incorporates appropriate factors pertinent to the plant design and operating characteristics, as given in these general guidelines.	(13) See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 11 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(14)	When a conservative analysis has not been provided by the applicant for staff review, the staff will use the radiation environment guidelines contained in Appendix D, suitably corrected for the differences in reactor power level, type, containment size, and other appropriate factors	(14)	See Table 3A.3-1.
1.5	Environmental Conditions for Outside Containment	1.5	
(1)	Equipment located outside containment that could be subjected to high-energy pipe breaks should be qualified to the conditions resulting from the accident for the duration required. The techniques to calculate the environmental parameters described in Sections 1.1 through 1.4 above should be applied.	(1)	See Table 3A.3-1.
(2)	Equipment located in general plant areas outside containment where equipment is not subjected to a design basis accident environment should be qualified to the normal and abnormal range of environmental conditions postulated to occur at the equipment location.	(2)	See Table 3A.3-1.
(3)	Equipment not served by Class 1E environmental support systems, or served by Class 1E support systems that may be secured during plant operation or shutdown, should be qualified to the limiting environmental conditions that are postulated for that location, assuming a loss of the environmental support system.	(3)	See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 12 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
2.	QUALIFICATION METHODS	2.	
2.1	Selection of Methods	2.1	
(1)	Qualification methods should conform to the requirements defined in IEEE Std. 323-1974.	(1)	The methodology employed by Westinghouse to qualify safety-related electrical equipment is described in WCAP-8587 and conforms to the requirements of IEEE Std. 323-1974.
(2)	The choice of the methods selected is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless (a) testing of the component is impractical due to size limitations, and (b) partial type test data is provided to support the analytical assumptions and conclusions reached.	(2)	Westinghouse qualifies equipment that is required to perform a safety function in a high energy line break environment by test.
(3)	The environmental qualification of equipment exposed to DBA environments should conform to the following positions. The bases should be provided for the time interval required for operability of this equipment. The operability and failure criteria should be specified and the safety margins defined.	(3)	The required and demonstrated duration of the safety function of equipment subject to high energy line break conditions is defined for equipment supplied by Westinghouse. The specified required duration is selected to envelope the range of operability assumptions contained in the Accident Analyses reported in the FSAR.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 13 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
			The primary purpose of equipment qualification is to reduce the potential for common-mode failures due to postulated environmental conditions. A test unit will therefore be considered to have failed the test if the functional requirements identified cannot be met, unless an investigation can establish that the failure mechanism is not of common-mode origin or that plant specific analyses can demonstrate that the reduced capability is acceptable. Margins are discussed under 3 below.
(a)	Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident.	(a)	When Westinghouse employs testing to qualify electrical equipment, that must function in order to mitigate any accident, the acceptance criterion for the test is that the safety related function must be demonstrated for the specified duration while the equipment is exposed to the simulated environmental conditions resulting from the accident.
(b)	Any equipment (safety-related or non-safety-related) that need not function in order to mitigate any accident but that must not fail in a manner detrimental to plant safety should be qualified by test to demonstrate its capability to withstand any accident environment for the time during which it must not fail.	(b)	See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 14 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(c)	Equipment that need not function in order to mitigate any accident and whose failure in any mode in any accident environment is not detrimental to plant safety need only be qualified for its non-accident service environment. Although actual type testing is preferred, other methods when justified may be found acceptable. The bases should be provided for concluding that such equipment is not required to function in order to mitigate any accident, and that its failure in any mode in any accident environment is not detrimental to plant safety.	(c)	Where Westinghouse supplies an item of safety-related electrical equipment that is located in an area where it can experience the environment resulting from a high energy line break, but is not required to perform any safety-function, Westinghouse has verified that any consequential failure of such equipment, due to the adverse environment, does not prejudice the safety related functions of other equipment claimed in the accident analysis.
(4)	For environment qualification of equipment subject to events other than a DBA, which result in abnormal environmental conditions, actual type testing is preferred. However, analysis or operating history, or any applicable combination thereof, coupled with partial type test data may be found acceptable, subject to the applicability and detail of information provided.	(4)	See Table 3A.3-1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 15 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
2.2	Qualification by Test	2.2	
(1)	The failure criteria should be established prior to Testing.	(1)	In Supplement 1 to WCAP-8587 Westinghouse has identified, for each item of safety-related equipment, the safety functions to be performed for all normal, abnormal or accident conditions during or after which the equipment is required to provide a protective function. As stated, a test unit will be considered to have failed the test if the safety related functional requirements cannot be met, unless an investigation can establish that the failure mechanism is not of common-mode origin or that plant specific analyses can demonstrate that the reduced capability is acceptable.
(2)	Test results should demonstrate that the equipment can perform its required function for all service conditions postulated (with margin) during its installed life.	(2)	Westinghouse qualification programs are designed to demonstrate that the equipment can perform its required safety function(s) for all postulated service conditions.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 16 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
			Via the treatment of aging, as described in the response to position 4. below, a generic qualified life is established by Westinghouse for each item of equipment. This generic qualified life may be extended on a plant specific basis by employing less conservative plant specific assumptions concerning the plant normal operating environmental conditions or by employing less conservative aging assumptions (i.e., less conservative activation energy for equipment). The qualified life established for the equipment on a specific plant will ultimately define the permitted installed life of the equipment. The subject of margin is discussed in the response to position 3. below.
(3)	The items described in Section 6.3 of IEEE Std. 323-1974 supplemented by items (4) through (12) below constitute acceptable guidelines for establishing test procedures.	(3)	When testing is the selected methodology for qualifying equipment, Westinghouse has established the test program in conformance with Section 6.3 of IEEE 323-1974.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 17 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
(4) When establishing the simulated environmental profile for qualifying equipment located inside containment, it is preferred that a single profile be used that envelopes the environmental conditions resulting from any design basis event during any mode of plant operation (e.g., a profile that envelopes the conditions produced by the main steamline break and loss-of-coolant accidents).	(4) Westinghouse prefers to use a single profile, enveloping both MSLB and LOCA, for qualification of equipment located inside containment which is required to perform a safety function to mitigate both High Energy Line Breaks (HELB). This approach is optimum in terms of schedule, manpower and materials. However, there is no technical justification for making this a requirement for all equipment inside containment. The exceptions to the use of a single qualification envelope for LOCA and MSLB are identified. In general, these occur when: - A component is only claimed to mitigate against one of the HELB's. In such a case, qualification has been completed to conditions enveloping the possible consequences inside containment from the single HELB and Westinghouse has verified that failure of the component in any other more limiting HELB environment will not prejudice any safety-related function. - The test conditions are found to be unacceptably conservative.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 18 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(5)	Equipment should be located above flood level or protected against submergence by locating the equipment in qualified watertight enclosures. Where equipment is located in watertight enclosures, qualification by test or analysis should be used to demonstrate the adequacy of such protection. Where equipment could be submerged, it should be identified and demonstrated to be qualified by test for the duration required.	(5)	See Table 3A.3-1.
(6)	The temperature to which equipment is qualified, when exposed to the simulated accident environment, should be defined by thermocouple readings on or as close as practical to the surface of the component being qualified.	(6)	In performing qualification tests for high energy line break environments, Westinghouse requires that the external environment temperature be measured as close to the test unit surface as practicable.
(7)	Performance characteristics of equipment should be verified before, after, and periodically during testing throughout its range of required operability.	(7)	Where the safety-related function of the equipment requires operation during the HELB, Westinghouse verifies the equipment performance before, during and after the simulated event and verifies that the safety-related function is demonstrated for the specified required duration of the function.
(8)	Caustic spray should be incorporated during simulated event testing at the maximum pressure and at the temperature conditions that would occur when the onsite spray systems actuate.	(8)	The response to item 1.3(1) is applicable for equipment located inside containment and qualified by test to operate in a high energy line break environment.
(9)	The operability status of equipment should be monitored continuously during testing. For long-term testing, however, monitoring at discrete intervals should be justified if used.	(9)	The response to item 2.2(7) is applicable.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 19 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(10)	Expected extremes in power supply voltage range and frequency should be applied during simulated event environmental testing.	(10)	Westinghouse does not require that power supply voltage and frequency be varied during the HELB simulation. Most of the Class 1E equipment to be qualified in Westinghouse scope is suppled by a guaranteed stabilized power supply. As a consequence, the range of electrical parameters employed is extremely small and variations within the permitted range are considered insignificant. Exceptions to this generic position are those items of equipment which could experience a significant change in power supply voltage and frequency at the time they are called upon to perform the specified safety function, and a justification is provided for the adequacy of the qualification of these items to HELB conditions.
(11)	Dust environments should be addressed when establishing qualification service conditions.	(11)	See Table 3A.3-1.
(12)	Cobalt-60 is an acceptable gamma radiation source for environmental qualification.	(12)	Westinghouse employs Cobalt-60 sources to simulate the effects of gamma and, in some cases, beta radiation for equipment qualified by test to operate in a HELB environment.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 20 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
2.3 Test Sequence	2.3
(1) The test sequence should conform fully to the guidelines established in Section 6.3.2 of IEEE Std. 323-1974. The test procedures should insure that the same piece of equipment is used throughout the test sequence, and that the test simulates as closely as practicable the postulated accident environment.	(1) Section 6.3.2 of IEEE 323-1974 neither mandates a single unique test sequence or requires that the same piece of equipment be used through the test sequence: - the standard identifies a test sequence that is thought to be the most conservative for most equipment, however, alternative sequences are clearly permitted with justification. - Section 6.3.2(3) specifically permits the performance test at extremes of the normal ambient to be performed on another, essentially similar, piece of equipment. The following basic test sequences are employed by Westinghouse: (a.) Equipment required to operate in a HELB environment The preferred test sequence is that recommended by IEEE 323-1974: - All production units were subjected to a calibration and/or verification test at ambient conditions. This test included verification of all safety-related functions. - No specific abnormal tests were completed since the HELB environment envelopes the abnormal condition with ample margin.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 21 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
	- The selected test unit(s) were aged employing the methodology described in Subprogram A of Appendix B to WCAP-8587. - The same production unit was tested to verify equipment capability during a simulated seismic event. - The same production unit was tested under applicable simulated HELB and post-HELB conditions. Exceptions to the above generic position are identified and justified. (b.) <u>Equipment not required to operated in a HELB environment</u> The following test sequence is employed: - Electronic production units were, in general subject to a burn-in period. - All production units were subjected to a calibration and/or verification test at ambient conditions. This test included verification of all safety-related functions. - A performance test was completed on a sample production unit at the specified abnormal environmental extremes. - A similar production unit was tested to verify the capability of the equipment during and after a simulated seismic event.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 22 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
	As indicated by this test sequence Westinghouse performs seismic testing on new equipment. Aging is addressed by a generic test program on a representative sample of aged components as described in Subprogram C of Appendix B to WCAP-8587. Westinghouse has deliberately selected this approach due to the advantages stated in Section 6 of Appendix B of WCAP-8587. The test sequence to be employed for component test program requires seismic testing of aged components, in conformance with IEEE 323-1974. The objective of the component test program is to demonstrate that there is no in-service aging mechanism capable of reducing electrical component capability to perform during a seismic event. Thus, success of this program verifies the validity of completing seismic tests on un-aged equipment. Exceptions to the above generic position are identified and justified. The subject of margin is discussed in the response to Position 3 above.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 23 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
2.4	Other Qualification Methods Qualification by analysis or operating experience implemented, as described in IEEE Std. 323-1974 and other ancillary standards, may be found acceptable. The adequacy of these methods will be evaluated on the basis of the quality and detail of the information submitted in support of the assumptions made and the specific function and location of the equipment. These methods are most suitable for equipment where testing is precluded by physical size of the equipment being qualified. It is required that, when these methods are employed, some partial type tests on vital components of the equipment be provided in support of these methods.	2.4 See Table 3A.3-1.	
3.	MARGINS	3.	
(1)	Quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. These margins should be applied in addition to any margins (conservatism) applied during the derivation of the specified plant parameters.	(1)	In general, Westinghouse has applied margin with respect to the design postulated accident conditions defined for each item of equipment, in Section 1 of the corresponding EQDP.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 24 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(2)	In lieu of other proposed margins that may be found acceptable, the suggested values indicated in IEEE Std. 323-1974, Section 6.3.1.5, should be used as a guide. (Note exceptions stated in Section 1.4.)	(2)	Westinghouse has applied specific margin to design parameters, in deriving type test parameters, as described in Section 7.1 of WCAP-8587. This method of applying margin is in accordance with Section 6.3.1.5 of IEEE 323-1974, which recognizes increasing test duration as methods of incorporating margin in the test plan.
(3)	When the qualification envelope in Appendix C is used, the only required margins are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for uncertainties such as production errors and errors associated with defining satisfactory performance (e.g., when only a small number of units are tested).	(3)	Westinghouse does not employ the envelope identified in Appendix C for qualification purposes.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 25 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
(4) Some equipment may be required by the design to only perform its safety function within a short time period into the event (i.e., within seconds or minutes), and, once its function is complete, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not be required to perform a safety function but must not fail within a short time period into the event, and subsequent failures are also shown not to be detrimental to plant safety. Equipment in these categories is required to remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis. For all other equipment (e.g., post-accident monitoring, recombiners, etc.), the 10 percent time margin identified in Section 6.3.1.5 of IEEE Std. 323-1974 may be used.	(4) In general, equipment required to operate in a high energy line break environment is qualified to perform its safety function over a considerable period of time (weeks/ months). The exceptions to this generic position are identified and justified. In qualifying equipment, Westinghouse did not always include any systematic margin on the specified duration of the safety function. Margin was included in qualification testing, as described in the response to position 3.(2). In addition to those areas where margin has been specifically included, Westinghouse has employed a sequence of testing that adds additional conservatism to the qualification program. Some of the areas where margin is usually implicit in the test sequence is as follows: - The full radiation dose, simulating the effects of in-service and high energy line break (HELB) applications, is applied in a single step prior to seismic and HELB test simulations. - When the aging is applied, the sequence in which the postulated aging mechanisms are simulated is designed to ensure a conservative simulation as discussed in the response to Staff position 4.1.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 26 of 31)

NUREG-0588 Category I Position		CPNPP STATUS
		- The seismic event simulation applies significant mechanical stress to the equipment prior to the HELB simulation. - The single envelope normally employed for HELB simulation, not only encompasses the effects of LOCA and MSLB accidents, but a whole spectrum of break sizes and locations within these accident definitions. As a consequence, the envelope employed invariably contains significant margin with respect to the transient for any single break size and location. - The single HELB simulation normally employed combines the high irradiation dose associated with the LOCA with the high temperature associated with the MSLB.
4.	AGING	4.
(1)	Aging effects on all equipment, regardless of its location in the plant, should be considered and included in the qualification program.	(1) Westinghouse considers and includes in the equipment qualification programs, the effects of aging, as applicable, irrespective of the location of the equipment in the plant.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 27 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(2)	The degrading influences discussed in Sections 6.3.3, 6.3.4 and 6.3.5 of IEEE Std. 323-1974 and the electrical and mechanical stresses associated with cyclic operation of equipment should be considered and included as part of the aging programs.	(2)	Appendix B to WCAP-8587 describes the aging mechanisms to be considered and the methodology to be employed in simulating the aging effects. The mechanisms to be considered in the Westinghouse program include the degrading influences discussed in Sections 6.3.3, 6.3.4 and 6.3.5 of IEEE 323-1974 including electrical and mechanical stresses associated with cyclic operation of equipment. The order in which these aging mechanisms are applied has been selected so that no aging mechanism, when simulated, reduces the impact of any previously applied mechanisms. On the contrary, the order selected tends to enhance the effects of any simulated aging mechanism as described in paragraph 40 of Appendix B to WCAP-8587.
(3)	Synergistic effects should be considered in the accelerated aging programs. Investigation should be performed to assure that no known synergistic effects have been identified on materials that are included in the equipment being qualified. Where synergistic effects have been identified, they should be accounted for in the qualification programs. Refer to (SAND 78-0799) and (SAND 78-1452), “Qualification Testing Evaluation Quarterly Reports,” for additional information.	(3)	Westinghouse has not identified any synergistic effects involving the materials and components comprising the equipment to be qualified in this program.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 28 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(4)	The Arrhenius methodology is considered an acceptable method of addressing accelerated aging. Other aging methods that can be supported by type tests will be evaluated on a case-by-case basis.	(4)	The Arrhenius equation is only employed by Westinghouse in calculating appropriate temperature and duration parameters to accelerate the effects of thermal aging.
(5)	Known material phase changes and reactions should be defined to insure that no known changes occur within the extrapolation limits.	(5)	Only when such changes are in a non-conservative direction would such mechanisms be a concern. Westinghouse will identify any know phase change or reaction which reduces the effects produced by any accelerated aging process and thereby makes extrapolating the results non-conservative. At this time no such mechanisms are known.
(6)	The aging acceleration rate used during qualification testing and the basis upon which the rate was established should be described and justified.	(6)	A supplement to WCAP-8587 justifies the acceleration parameters and rates employed for the aging program described in Appendix B to WCAP-8587.
(7)	Periodic surveillance testing under normal service conditions is not considered an acceptable method for on-going qualification, unless the plant design includes provisions for subjecting the equipment to the limiting service environment conditions (specified in Section 3(7) of IEEE Std. 279-1971) during such testing.	(7)	Westinghouse does not employ periodic surveillance testing or any form of on-going qualification in the program described in WCAP-8587.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 29 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
(8)	Effects of relative humidity need not be considered in the aging of electrical cable insulation.	(8)	For equipment subjected to High Energy Line Break (HELB) environments, the aging effects due to humidity during normal operation are judged to be insignificant compared to the effects of the high temperature steam accident simulation and therefore no additional humidity aging simulation is required. For equipment not subjected to HELB environments, the use of materials and components know to be significantly affected by humidity has been avoided.
(9)	The qualified life of the equipment (and/or component as applicable) and the basis for its selection should be defined	(9)	On completing the qualification program for a particular item of equipment, Westinghouse completed the corresponding Equipment Qualification Data Package (EQDP). The final version of the EQDP identifies the demonstrated qualified life and justifies the value selected based on the aging mechanisms that have been simulated.
(10)	Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and in other methods that may be reasonably assumed, coupled with good engineering judgement.	(10)	Westinghouse conforms to this position in establishing the demonstrated qualified life of WRD supplied equipment.

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 30 of 31)

NUREG-0588 Category I Position		CPNPP STATUS	
5.	QUALIFICATION DOCUMENTATION	5.	
(1)	The staff endorses the requirements stated in IEEE Std. 323-1974 that, "The qualification documentation shall verify that each type of electrical equipment is qualified for its application and meets its specified performance requirements. The basis of qualification shall be explained to show the relationship of all facets of proof needed to support adequacy of the complete equipment. Data used to demonstrate the qualification of the equipment shall be pertinent to the application and organized in an auditable form."	(1)	See Table 3A.3-1. In support of this effort Westinghouse has supplied: - identification of the limits to any environmental qualification - copies of all referenced Westinghouse topical reports - a summary of the Westinghouse review to establish the adequacy of its qualification programs with respect to NUREG-0588 In addition, Westinghouse will maintain the available raw test data which supports the referenced qualification tests on equipment subject to high energy line break and available information concerning the performance testing of production units.
(2)	The guidelines for documentation in IEEE Std. 323-1974 when fully implemented are acceptable. The documentation should include sufficient information to address the required information identified in Appendix E. A certificate of conformance by itself is not acceptable unless it is accompanied by test data and information on the qualification program.	(2)	The Westinghouse qualification test reports meet the requirements of Section 5 to IEEE 323-1974 by providing the following essential information as a minimum: - safety related functional requirements to be demonstrated - range of applicable environmental parameters to be considered - identification of the test unit - description of the test facility and monitoring instrumentation

TABLE 3A.3-2
CPNPP COMPLIANCE WITH NUREG-0588 FOR NSSS CLASS 1E ELECTRICAL EQUIPMENT

(Sheet 31 of 31)

NUREG-0588 Category I Position	CPNPP STATUS
	- description of test unit mounting and interfaces - summary of the test procedures - summary of the test results Westinghouse maintains the supporting raw test data, detailed procedures employed etc. on file and available for audit. Significant exceptions to this generic position are identified and justified.

TABLE 3A.4-1
IS DELETED AND THE SPECIFIC ENVIRONMENTAL DETAILS ARE
REFLECTED IN CONTROL DOCUMENTS (EEQSPs, EQML, DBDs)

APPENDIX 3B

COMPUTER PROGRAMS USED IN THE SYNAMIC AND STATIC ANALYSIS OF ASME CODE CLASS 2 AND 3 PIPING SYSTEMS INCLUDING SUPPORTS FOR ASME CODE CLASS 1, 2 AND 3 PIPING

INTRODUCTION

The following computer programs are used for the analysis of ASME Code Class 2 and 3 piping systems, including supports for ASME Code Class 1, 2, and 3 piping:

1. NUPIPE-SW
2. BAP
3. BSPLT
4. STARDYNE
5. PITRUST
6. PILUG
7. PITRIFE
8. STEHAM
9. WATHAM
10. WATSLUG
11. ELBOW
12. PSPECTRA
13. STRUDL-SW
14. STRUDAT AND SANDUL
15. BASEPLATE-II
16. BIP
17. APE
18. CHPLOT
19. RELAP5
20. NUDL
21. CCW-318
22. CCW-392
23. LOTUS
24. ANSYS
25. PSAP
26. REPIPE
27. SCAP
28. ETA
29. HYDTRAN
30. ME101, LEAP
31. ME150, FAPPS
32. ME035, BASEPLATE
33. ME153, MAPPS
34. ME149, SIGNIT
35. ME148, CAPPS
36. ME214, LSAPS

37. *WESTDYN and Associated Pre- and Post-processors
38. *PIPSAN
39. *WESPLAT
40. *LOCAL
41. *PCLUG
42. *Stress Calculation Spreadsheets
43. *WECAN
44. ME215 (ME101FE), SAPCAS
45. NE820, HSTA

For each computer program there is a brief description of the program's theoretical basis, the assumptions, and the references used in the program, the extent of its application, and a summary of manual or comparison qualification.

* Unit 2 analyses only

TABLE OF CONTENTS (Continued)

Section	Title	Page
3B.1	NUPIPE-SW	3B-1
3B.2	BAP	3B-2
3B.3	BSPLT	3B-2
3B.4	STARDYNE	3B-2
3B.5	PITRUST	3B-3
3B.6	PILUG	3B-3
3B.7	PITRIFE	3B-4
3B.8	STEHAM	3B-4
3B.9	WATHAM	3B-5
3B.10	WATSLUG	3B-6
3B.11	ELBOW	3B-6
3B.12	PSPECTRA	3B-8
3B.13	STRUDL	3B-8
3B.14	STRUDAT AND SANDUL	3B-12
3B.15	CDC - BASEPLATE-II	3B-12
3B.16	BIP	3B-13
3B.17	APE	3B-13
3B.18	CHPLOT	3B-14
3B.19	RELAP5	3B-14
3B.20	NUDL	3B-15
3B.21	CCW-318	3B-15
3B.22	CCW-392	3B-15
3B.23	LOTUS	3B-15
3B.24	ANSYS	3B-16
3B.25	PSAP	3B-16
3B.26	REPIPE	3B-16
3B.27	SCAP	3B-16
3B.28	ETA	3B-17
3B.29	HYDTRAN	3B-17
3B.30	LEAP	3B-17
3B.31	FAPPS	3B-18
3B.32	BASEPLATE	3B-18
3B.33	MAPPS	3B-19
3B.34	SIGNIT	3B-20
3B.35	CAPPS	3B-20
3B.36	LSAPS	3B-20

3B.37	*WESTDYN and Associated Pre- and Post-Processors	3B-21
3B.38	*PIPSAN	3B-21
3B.39	*WESPLAT	3B-21
3B.40	LOCAL	3B-22
3B.41	PCLUG	3B-22
3B.42	ME215 (ME101FE), SAPCAS	3B-22
3B.43	*WECAN	3B-23
3B.44	*ME215, SAPCAS	3B-23
3B.45	NE820, HSTA	3B-24
REFERENCES		3B-25

* Unit 2 analyses only

LIST OF TABLES

Table No.	Title
3B.1-1	Comparison of Natural Frequencies for Figure 3B.1-1
3B.1-2	Comparison of Internal Member Loads and Nodal Deflections, Combined Results for Modes 1 thru 5 for Figure 1.0
3B.1-3	Comparison of Natural Frequencies for Figure 3B.1-2
3B.1-4	Comparison of Internal Member Loads and Nodal Deflections, Combined Results for Modes 1 thru 5 for Figure 3B.1-2
3B.1-5	Comparison of Support Loads Between NUPIPE II and NUPIE-SW for Figure 3B.1.3
3B.1-6	Comparison of Support Load for Figure 3B.1-4
3B.1-7	Comparison of Support Reactions Due to Thermal Expansion for Figure 3B.1-5
3B.1-8	Comparison of Deflections Due to Thermal Expansion for Figure 3B.1-5
3B.1-9	Comparison of Stress Due to Thermal Expansion for Figure 3B.1-5
3B.1-10	Comparison of Internal Forces and Moments Due to Thermal Expansion for Figure 3B.1-5
3B.3-1	Comparison of BSPLT Computer Program Results With Hand Calculation Results
3B.5-1	Comparison of PITRUST with Franklin Institute Program, CYLNOZ, and Hand Calculation
3B.5-2	Comparison of PITRUST With Reference 2 Results
3B.6-1	Comparison of PILUG Computer Program Output With Hand Calculations (2 Sheets)
3B.7-1	Comparison of PITRIFE Computer Program Output With STRUDL-II Output
3B.7-2	Comparison of PITRIFE Computer Program Output With Hand Calculations
3B.8-1	Nodal Force Comparison
3B.9-1	Input Data for WATHAM
3B.9-2	Comparison of Nodal Force Calculation at Time = 2.34 Seconds

LIST OF TABLES

Table No.	Title
3B.10-1	Input Data for WATSLUG
3B.10-2	Input Data for WATSLUG
3B.11-1	ELBOW Program - ELBOW Properties Used for Verification Purposes
3B.11-2	ELBOW Program - Case 1 Results
3B.11-3	ELBOW Program - Case 2 Results
3B.11-4	ELBOW Program - Case 3 Results
3B.13-1	Comparison of Eigenvalues from Theoretical Results, STRUDL Results, and GT-STRUDL Results (Problem No. 1)
3B.13-2	Comparison of Eigenvectors from STRUDL and GT-STRUDL (Problem No. 1) (2 Sheets)
3B.13-3	The Member Forces from STRUDL and GT-STRUDL Computer Runs for Different Loading Conditions (Problem No. 2) (2 Sheets)
3B.13-4	The Joint Loads (at Supports) from STRUDL and GT-STRUDL Computer Runs for Different Loading Conditions (Problem No. 2)
3B.13-5	Comparison of Element (Randomly Selected) Stresses (Problem No. 3) (2 Sheets)
3B.13-6	Comparison of Resultant (Randomly Selected) Joint Displacement Supports (Global) (Problem No. 3)
3B.13-7	Comparison of Members (Randomly Selected) Forces and Moments (Problem No. 4)
3B.13-8	Comparison of Resultant Joint Loads - Supports (Problem No. 4) (Global)
3B.13-9	Comparison of Eigenvalues, Frequencies and Periods (Problem No. 5)
3B.13-10	Comparison of Eigenvectors for Few Randomly Selected Modes and Joints (Problem No. 5)
3B.13-11	Comparison of Joint Displacements at the Free Joints for Randomly Selected Joints (Global) (Problem No. 5)
3B.14-1	Comparison of SANDUL Load Combinations With Hand Computation
3B.14-2	Comparison of Member Forces Between SANDUL and STRUDL-SW

LIST OF TABLES

Table No.	Title
3B.14-3	Comparison of Stresses and Weld Size Between SANDUL and Hand Computation
3B.17-1	APE Program Verification Problem Comparison of Drilled-In Anchor Load (3 Sheets)

LIST OF FIGURES

Figure No.	Title
3B.1-1	Mathematical Model for Response Spectra Seismic Verification
3B.1-2	Mathematical Model for Response Spectra Seismic Verification
3B.1-3	Mathematical Model for Missing Mass Verification
3B.1-4	Mathematical Model for Missing Mass Verification
3B.1-5	Mathematical Model for Thermal Analysis
3B.8-1	Sudden Discharge of a Gas from a Pipeline Through a Nozzle (Case A)
3B.8-2	Sudden Discharge of a Gas from a Pipeline Through a Nozzle (Case B)
3B.8-3	Comparison of Pressure Response at the Closed End
3B.8-4	Comparison of Pressure Response at the Open End
3B.8-5	Comparison of Pressure Response by STEHAM and Experiment
3B.9-1	Hydraulic Network for Verification Program
3B.9-2	Hydraulic Network for WATHAM Verification
3B.9-3	Head-Versus-Tim Plot for Junction J
3B.9-4	Head-Versus-Tim Plot at Valve
3B.10-1	WATSLUG Model of EPRI Sample Problem
3B.10-2	NUPIPE-SW Model of EPRI Sample Problem
3B.10-3	Comparison of Segment 2 Forcing Function
3B.10-4	Comparison of Segment 3 Forcing Function
3B.10-5	Comparison of Segment 2 Support Reaction
3B.10-6	Comparison of Segment 3 Support Reaction
3B.12-1	PSPECTRA - Absolute Summation of ARS Curves
3B.12-2	PSPECTRA - Required Response Spectrum Generation
3B.13-1	Finite Element Model of the Foundation Mat for a Portion of the Off-Gas Building

LIST OF FIGURES

Figure No.	Title
3B.13-2	Positive Sign Convention for Results of Plate Bending Element
3B.13-3	Model - Suspended Ceiling
3B.14-1	STRUDL Input ASME Anchor

APPENDIX 3B

3B.1 NUPIPE - SW

General Description

The NUPIPE-SW piping program performs a linear elastic analysis of three dimensional piping systems subjected to thermal, static, and dynamic loads. It utilizes the finite element method of analysis.

NUPIPE-SW handles all loading conditions required for complete nuclear piping analyses. A given piping configuration may be analyzed successively for a number of static and dynamic load conditions in a single computer run. Separate load cases, such as thermal expansion and anchor displacements may be combined to form additional analysis cases. The piping dead load analysis considers both distributed weight properties of the piping and any added concentrated weights.

A lumped mass model of the system is used for all dynamic analysis; both translational and rotational degrees of freedom may be considered. Location of lumped masses and degrees of freedom at each mass point are pre-selected by the analyst. The program automatically computes values of translational lumped masses.

Program input consists basically of program control, piping configuration description, and load specification information. As part of the program input process a “missing mass” corrective feature is utilized in the response spectra analysis. This provides an approximation to the response of modes above the specified cutoff frequency in calculating the support loads. The output for each loading condition analyzed consists of support reactions, internal forces and moments, deflections, rotations, and member stresses. Output from seismic analysis includes system normal mode information. Several reports may be generated based on report specification. These reports include pipe stress summaries, pipe support tabulations and piping isometric plots.

The NUPIPE-SW program performs analysis in accordance with ASME Section III, Nuclear Power Plant Components (Code). Features ensuring code conformance include use of accepted analysis methods, incorporation of specified stress intensification factors, stress indices and flexibility factors, proper combination of moment resultants, and provision to (automatically) generate results of combined loading cases. A program option is available to specify among Class 1 analysis in accordance with NB-3600 of the Code, Analysis per ANSI B31.1.0 power piping code and combined Class 1 and Class 2 analysis per Articles NB-3600 and NC-3600 of the Code.

Program Verification

The NUPIPE-SW program has been verified against the NRC Benchmark problems for response spectrum seismic analysis. The model shown in Figure 3B.1-1 is a three dimensional piping system consisting of straight and curved elements starting and terminating at anchors. The seismic analysis used a standard SRSS combination. The second model shown in Figure 3B.1-2 is a multi branched piping system which resembles a table. For the seismic analysis a 10 percent grouping method utilizing a standard SRSS combination is selected. The results of

comparisons from both programs are presented in Tables 3B.1-1 through 3B.1-4. The models used are presented in Figures 3B.1-1 and 3B.1-2.

The NUPIPE-SW missing mass option has been verified against NUPIPE-II. The mathematical model in Figure 3B.1-3 has a total of 87 translational dynamic degrees of freedom. The NUPIPE-SW and NUPIPE-II cutoff mode is mode 19 with a frequency of 44 hertz. The ANCHOR stiffnesses input are the same for the NUPIPE-SW and NUPIPE-II analyses. The results of comparison are in Table 3B.1-5 with the model appearing in Figure 3B.1-3.

The additional model used in the missing mass verification is shown in Figure 3B.1-4. It is a free-ended piping system with three equally spaced lumped mass points. The three lumped mass points have a translational dynamic degree of freedom in the X-direction only. This results in a maximum of three modes being generated. Several computer runs were made using NUPIPE-SW. The first computer run had the missing mass option with a cutoff mode set at 2. The second run and no missing mass option and a cutoff mode equal to 3. The results of comparison are in Table 3B.1-6.

The NUPIPE-SW program has been verified with ADLPIPE (A.D. Little Corp.) for thermal analysis The results from both programs are presented in Tables 3B.1-7 through 3B.1-10. The model used for this comparison is shown in Figure 3B.1-5 with the operating conditions indicated on the figure.

3B.2 BAP

The Baseplate Analysis Processor (BAP) computer program is a preprocessor/postprocessor that works in conjunction with the program ANSYS. The purpose of BAP is to generate the ANSYS input necessary for the static, non-linear analysis of baseplates subjected to out-of-plane loads, to distribute in-plane loads to the anchor bolts assuming an infinitely rigid baseplate, and to post process the ANSYS results into a report-style format.

BAP has been documented by benchmarking procedures against the Baseplate Investigation Processor computer code, which is a recognized program in the public domain.

3B.3 BSPLT

BSPLT was developed by SWEC and is a fully documented computer program. BSPLT is used for the qualification of baseplates with four anchor bolts. This program computes bolt tension, shear, tension-shear interaction factors, and plate-bending stresses for various load conditions based on loads applied at the surface of the plate.

The BSPLT computer program has been verified by a comparison with a hand calculation. The sample problem was a 19-in. by 21-in. four-bolt baseplate with a plate thickness of 1 in. and a bolt size of 1 in. The results are summarized in Table 3B.3-1. The computer program results are within acceptable limits.

3B.4 STARDYNE

The STARDYNE Structural Analysis System, written by Mechanics Research, Inc., of Los Angeles, California, is a fully warranted and documented computer program available at Control Data Corporation.

The MRI STARDYNE Systems consists of a series of compatible, digital computer programs designed to analyze linear and non-linear elastic structural methods. The system encompasses the full range of static and dynamic analyses.

The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, nodal-applied loads, and prescribed displacements.

Using the normal mode technique, linear dynamic response analyses can be performed for a wide range of loading conditions, including transient, steady-state harmonic, random, and shock spectra excitation types. Dynamic response results can be presented as structural deformations, internal member loads and stresses, and statistical data.

The non-linear dynamic analysis program is integrated in the rest of the STARDYNE system. The equations of motion for the linear portion of the structural model are generated and modified to account for the non-linear springs. The resulting non-linear equations of motion are directly integrated, using either the Newmark or Wilson implicit integration operators. The user may enter sets of structural loadings, which vary with time, and specify time points at which the program is to output the structural response.

This computer program is considered verified by constant use and by the vendor's original documentation and qualification.

3B.5 PITRUST

PITRUST is a program to calculate local stresses in the pipe caused by cylindrical welded attachments under external loadings. This program uses the Bijlaard method to calculate local stresses in the pipe wall caused by cylindrical welded attachments under external loadings, including pressure, dead load, thermal load, and combinations of maximum dynamic loads.[1]

PITRUST has been verified by comparing its solution of a test problem to the solution of the same problem by an independently written piping local stress program, CYLNOZ, in the public domain. The CYLNOZ piping local stress program was written by Franklin Institute (Philadelphia, PA) and is presently used by engineering companies. The test program is of a 72.375-in. outside diameter by 0.375-in. thick run pipe, reacting under an external loading condition of 1,000 lb. force (normal and shear) and 1,000 in.-lb. bending and torsional moments transmitted by a 16-in. outside diameter nozzle. A comparison of results is tabulated in Table 3B.5-1. PITRUST also has been verified by comparing its solution of the test problem to the experimental results.[2] A comparison of these results is tabulated in Table 3B.5-2.

3B.6 PILUG

PILUG is a program to calculate local stresses in the pipe wall caused by rectangular welded attachments under external loadings. This program uses the Bijlaard method to calculate local stresses in pipe wall caused by rectangular welded attachments under external loadings, including pressure, dead load, thermal load, and combinations of maximum dynamic loads.[3]

PILUG has been verified by comparing its solution of test problem to results obtained by hand calculations using the formulations of Reference 3. A comparison of results is tabulated in Table 3B.6-1.

3B.7 PITRIFE

General Description

PITRIFE (SWEC 1982) is a computer program for calculating the local discontinuity stresses in a pipe at the intersection with a circular trunnion due to loads applied to the trunnion. It is a post-processor program that uses the results of a finite element model of two intersecting cylinders. Based upon the stresses calculated with the finite element model, nondimensional stress coefficients are computed for a size-on-size pipe-trunnion configuration for three different values of average pipe radius to wall thickness (R/t = 5, 10, 20). Additionally, non-dimensional stress coefficients are computed for a trunnion radius equal to 0.707 times the pipe radius (0.707 size-on-size) for the three values of R/t. To facilitate the determination of non-dimensional stress coefficients for other values of R/t, a rotated parabola curve that fits the three R/t data points is generated for both the size-on-size and the 0.707 size-on-size data. The PITRIFE program reconstructs these curves and uses them to interpolate and extrapolate for stress coefficients for different values of R/t. The finite element models are analyzed using the STRUDL-II computer program (ICES) SWEC 1977.

Program Verification

The PITRIFE computer program has been verified by demonstrating that the maximum stress intensities as given by PITRIFE equal the values given by the finite element analysis for specific size-on-size and 0.707 size-on-size models. A comparison of these results is tabulated in Table 3B.7-1. The program was verified for other ratios of trunnion to pipe radius by demonstrating that the stress coefficients and maximum stress intensities derived by hand calculation equal the coefficients used in the program to calculate maximum stress intensity. A comparison of these results is given Table 3B.7-2.

3B.8 STEHAM

General Description

STEHAM is a computer program which is used to determine the steam hammer transients of piping systems. This program uses the method of characteristics with finite difference approximations both in space and in time[4,5,6]. It calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changes of piping components, such as the stop valve and the safety relief valve. Flow characteristics of piping components are mathematically formulated as boundary conditions in the program. These components include the flow control valve, the stop valve, the safety relief valve, the steam manifold, and the steam reservoir. Frictional effects are taken into consideration.

This program accepts the following as input:

a. The flow network representation of the piping system.

b. The initial flow conditions along the piping system.
c. Time-dependent flow characteristics of piping components.

Output consists of time-histories of flow pressures, flow densities, flow velocities, inertia, and momentum functions.

Program Verification

STEHAM is verified by comparing its solutions of a test problem (Figures 3B.8-1 and 3B.8-2) to the results of the same problem obtained by an independent analytical approach, as well as an experimental measurement, as published in Reference 7 and 8. A comparison of results for time-history pressure responses is plotted on Figures 3B.8-3, 3B.8-4, and 3B.8-5. The forcing functions developed for nodal points of the piping system calculated from the relation $F = (p + PV^2/g) A - p_a A$ also have been checked by hand calculation as tabulated in Table 3B.8-1.

3B.9 WATHAM

General Description

WATHAM is a computer program which is used to determine the flow-induced forcing functions acting on piping systems due to water hammer. These forcing functions may then be used as input to a structural dynamic analysis, such as a NUPIPE program run.

WATHAM is applicable to a water hammer problem or, more generally, any unsteady, incompressible fluid flow. These events may be caused by normal or abnormal operational changes of piping components, such as the start up and trip of pumps or the rapid opening and closing of valves.

The analysis is based upon the method of characteristics with finite-difference approximations, both in time and space for the solution of one-dimensional liquid flows. Influences of piping components, including flow valves, pipe connection, reservoirs, and pumps have been considered in the analysis.

WATHAM input requires the geometry of the piping system, pipe properties, water properties, operational characteristics of pump and valve, flow frictional coefficients, and the initial water flow conditions. The output provides the time-history functions of piezometric heads, velocities, and nodal forces for all nodes and the inertial unbalanced force for each segment. It also gives the maximum value of all the preceding functions and their occurring time in the process of flow-transient.

Program Verification

Figure 3B.9-1 depicts a flow network with nine pipes, its geometrical properties, and steady-state flow conditions. The flow transient mode analyzed is the sudden closure of a valve at the down steam end. Figure 3.9B-2 shows the hydraulic network for WATHAM. Table 3B.9-1 illustrates the input data needed for WATHAM run. Figures 3B.9-3 and 3B.9-4 show a comparison of head-time curves[9,10] with WATHAM. Table 3B.9-2 presents the comparison of nodal forces between hand calculation and WATHAM computation.

In general, WATHAM results are in agreement with Streeter's results[9]. The small discrepancy is attributed to the modeling of reservoir boundary condition. In WATHAM, the energy equation between the reservoir is used, rather than assuming that the head of pipe entrance is the same as that of the reservoir.

3B.10 WATSLUG

General Description

The purpose of WATSLUG is to determine forcing functions on piping systems during water slug discharge events for subsequent input to piping dynamic analysis.

The analysis is based upon rigid body motion of the generally sub-cooled water slug and ideal gas representations of the steam or air using rigid column theory to facilitate tracking the several water-steam or water-air interfaces. The driving force is the steam pressure between the valve and the slug, less friction and other losses, and back pressure. Density changes due to possible local flashing of the water slug are considered. Having recourse to the control volume theory, the subsequent segment-forced calculation is carried out.

The input consists of complete piping system geometry, pipe dimensions (Table 3B.10-1), valve flow characteristics, valve opening time, detail upstream steam conditions, and initial downstream steam or air conditions (Table 3B.10-2), while the output contains forcing functions for each piping segment based upon flow velocities, pressures, and densities during the water slug discharge event. Forces were written on tape for direct input to NUPIPE-SW.

Program Verification

The WATSLUG model of the test problem is diagrammed on Figure 3B.10-1, while the NUPIPE-SW model is diagrammed on Figure 3B.10-2. WATSLUG is verified by comparing the solution of this test problem to the results for the same problem obtained by an independent analytical approach (RELAP5/MOD 1), as shown on Figures 3B.10-3 and 3B.10-4, and the comparison of predicted-versus-measured support reactions. NUPIPE-SW generated support reactions due to WATSLUG forcing functions were compared with experimental measurements from a test run of this problem, EPRI Test 980 (RELAP/MOD 1) shown on Figures 3B.10-5 and 3B.10-6.

The WATSLUG forcing functions and the resultant NUPIPE-SW support reactions compare favorably with the RELAP/MOD 1 predicted forcing functions and the EPRI-measured support reactions, respectively.

3B.11 ELBOW

General Description

ELBOW calculates the circumferential and longitudinal stresses on the inside and outside surfaces of an elbow subjected to internal pressure, in-plane bending, out-of-plane bending, torsion, and linear temperature gradient through the wall. Stress indexes and flexibility factors for the elbows are also calculated. Results can be used directly for design and analysis of elbows in accordance with Article NB-3600 of ASME Section III.

The solution method uses Table NB-3685.1-1 relative to internal pressure and Table NB-3685.1-2, with modifications as indicated by Dodge and Moore (1972) relative to moment loadings and flexibility factors.

The complete analysis of Rodabough and George (1957), based on the minimum potential energy method, was written in terms of infinite series. A modified version of the analytical method by Rodabough and George (1957), which considered both in-plane and out-of-plan bending as well as the influence of internal pressure, was selected by ORNL as the most appropriate basis in determining the stresses and flexibility for elbows. The analysis is a generation of the work done by Von Karman. The modification to the analysis method of Rodabough and George (1957) include a generalization of the "correction for transverse compression" recommended by Gross. The ORNL computer program ORNL-ELBOW was written by Dodge and Moore (1973) to implement this analysis procedure.

The SWEC computer program ELBOW uses the same theoretical considerations to obtain the flexibility factor and detailed stresses in the elbows.

When this program is used, it is considered to be a detailed analysis. For elbows free from local discontinuities, ELBOW solutions are the detailed solutions to Equation 10, Table NB-3653, including the consideration of C_1 and C_2, but without $|\alpha_a T_a - \alpha_b T_b|$ term. The solutions are also the detailed solutions to Equation 11 if $|\Delta T_2|$ term is negligible.

The program does not take into account the effects of discontinuities on the elbows. The influence length of a concentrated force or moment in a shells structure is about 2.5 $\sqrt{rt}$, where r is the radius of curvature of the shell surface; or for a pipe, r is the mean radius and t is the thickness of the pipe. For the portion of elbow at a distance of 2.5 $\sqrt{rt}$ away from local discontinuities, detailed stresses can be obtained by using this program.

In general, for an elbow welded to tangent pipe of the same thickness, the effects of straight tangent pipe on the elbow can be neglected (Table NB-3683.2). However, if two elbows are welded together or joined by a piece of straight pipe that is less than one pipe diameter in length, some intensification effects may have to be considered.

This program is an efficient and easy-to-use program for determining stresses, stress indexes, and flexibility factors for elbows. Comparison with experimental results indicates that the results accurately represent the maximum stresses which occur at the center of the bend. Since end effects are not included in the analytical solution on which this program is based, the calculated stresses and flexibility of the elbow may be larger than the actual values.

Required input data include elbow dimensions, material properties, applied moments and forces, internal pressure, and linear temperature gradients.

Output includes stress indexes, flexibility factor for the elbow, and circumferential and longitudinal stresses on the inside and outside surfaces at specific locations.

Program Verification

Sample problems were selected for solution by ELBOW, and these results were compared with those obtained from hand calculations. The following cases were selected for purposes of verification:

Case 1 - Elbow is subjected to internal pressures. Results are given in Table 3B.11-2.

Case 2 - Elbow is subjected to a linear temperature gradient through the pipe wall. Results are given in Table 3B.11-3.

Case 3 - Elbow is subjected to combined loadings at one end. Results are given in Table 3B.11-4.

Elbow properties used for the analyses are given in Table 3B.11-1.

3B.12 PSPECTRA

General Description

PSPECTRA (ME-164) is a data-generating program written and fully documented by SWEC for in-house use. It is used to combine amplified response spectra of seismic and other dynamic events. The methods of spectrum combination include absolute summation, square root of the sum of the squares (SRSS), and maximum value enveloping. PSPECTRA is also used to generate required response spectra which are in accordance with Regulatory Guide 1.122, Revision 1. This involves spreading the peak accelerations and sloping the sides parallel to the original peaks of the input amplified response spectra. The output curves can be generated in terms of accelerations (g's) and either period (sec) or frequency (Hz).

Program Verification

A comparison of a generated response spectrum versus the two input response spectra that were combined by absolute summation is provided on Figure 3B.12-1. Figure 3B.12-2 provides a generated required response spectrum with spread peaks and parallel sloped sides superimposed on the input amplified response spectrum (ARS). The ARS is generated by the time-history method. These figures demonstrate the function and adequacy of the program.

3B.13 STRUDL (STRUCTURAL DESIGN LANGUAGE)

General Description

The STRUDL computer code used within SWEC was developed from Version 2, Modification 2 (June 1972) of the Integrated Civil Engineering Systems (ICES) STRUDL II program which was design and formulated by the Department of Civil Engineering at the Massachusetts Institute of Technology. STRUDL II is a recognized program in the public domain. The software system is IBM-MVS Release 3.8. The hardware configuration is IBM-3033.

The finite element method provides for the solution of a wide range of solid mechanics problems. Its implementation within the context of the STRUDL analysis facilities expands these problems

for the treatment of plane stress, plane strain, plate bending, shallow shell, and three-dimensional stress analysis problems.

The three-dimensional finite element capability of STRUDL is used to analyze the drywell at the region of the equipment hatch and personnel door assembly and other regions of interest.

Seismic Category I structures are analyzed for seismic effect using the dynamic analysis capability of STRUDL. The analysis yields frequencies of vibration, mode shapes, displacements, velocities, accelerations, and forces.

Program Verification

comparisons of results for five test problems performed by both STRUDL and GT-STRUDL are provided herein. GT-STRUDL is a recognized program in the public domain, developed by the GT-ICES Systems Laboratory, School of Civil Engineering, Georgia Institute of Technology, Atlanta, Georgia. In all cases, there is excellent agreement of results between STRUDL and GT-STRUDL.

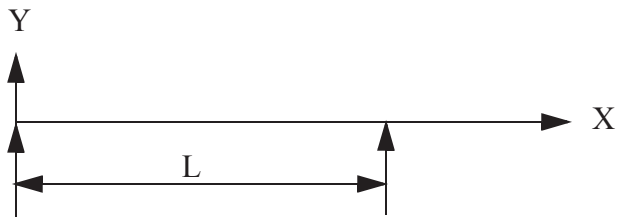
COMPARISON OF STRUDL VERSUS GT-STRUDL RESULTS
FOR DYNAMIC ANALYSIS CAPABILITY OF STRUDL

Problem No. 1

Find the natural frequencies F (I) of vibration for an I-beam with simply supported end, vibrating in the plane of its web.

The pertinent parameters of the beam are as follows:

Length	=	30 ft.
Modulus of Elasticity	=	30×10^6 psi
Moment of Inertia	=	3021 $in.^4$
Weight per foot	=	100 lb.



The theoretical results can be verified from Vibration Problems in Engineering, Fourth Edition, S. Timoshenko, D. W. Young, and W. Weaver, page 423, Problem 1.

Results: Natural Frequency F(I)

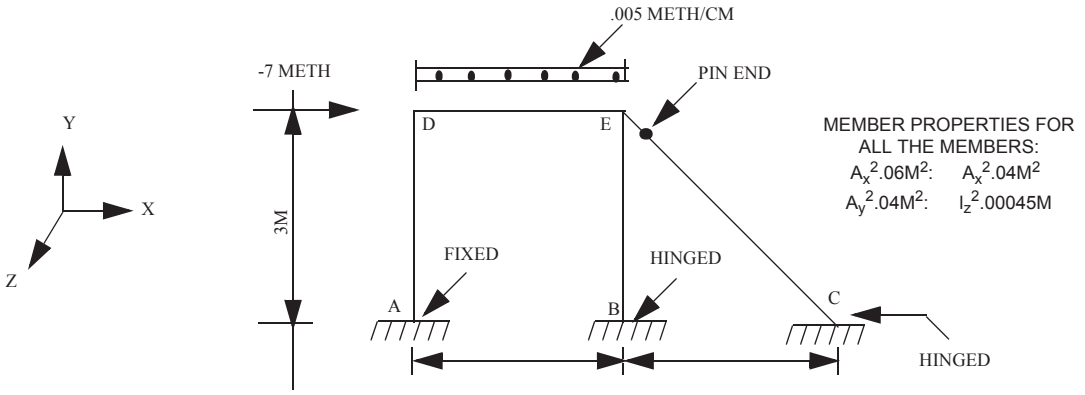
where:

I = mode number
= $24.8\ (I)^2$ cycles/sec
= $155.82\ (I)^2$ rad/sec

The comparison of results (i.e., eigenvalues and eigenvectors) of the theoretical values, STRUDL and GT-STRUDL, is tabulated in Tables 3B.13-1 and 3B.13-2. The eigenvalues for STRUDL and GT-STRUDL agree with each other (Table 3B.13-1). The eigenvectors for STRUDL and GT-STRUDL agree with each other (Table 3B.13-2).

COMPARISON OF STRUDL VERSUS GT-STRUDL RESULTS
FOR STATIC ANALYSIS CAPABILITY OF STRUDL

Problem No. 2



The frame as shown on the sketch is tested for the loads as shown on the sketch. Also, the frame was tested for joint displacement of joints A and B in the Y direction and also the joint displacement of joint A in the X direction. The member forces and the joint forces of the STRUDL run agreed with the GT-STRUDL run. The comparison of the results is tabulated in Table 3B.13-3 and 3B.13-4.

Loading Condition 1

Member DE force Y uniform W-0.005 Metn/cm

Joint D load force X-0.7 Metn

Loading Condition 2

Joint A displaced Y -0.8 cm

Joint B displaced Y -0.3 cm

Loading Condition 3

Joint A displaced X -0.2 cm

Problem No. 3

A foundation mat was analyzed using the finite element capability of STRUDL for a variety of loading combinations. A comparison check is performed for a loading condition which combines the self weight of the substructure and superstructure, dead load of 2.5 ft. of soil above the mat, and east-west tornado loading, by using the finite element capability of GT-STRUDL, a computer program in the public domain. A finite element model is provided in Figure 3B.13-1. Sign convention details are provided in Figure 3B.13-2. Refer to Table 3B.13-5 and 3B.13-6 for comparison between the results obtained from STRUDL and GT-STRUDL.

Problem No. 4

A comparison check is performed for suspended ceiling design using the static analysis capability of STRUDL versus GT-STRUDL. A model is provided in Figure 3B.13-3. The loading condition accounts for the dead loads of ceiling. Refer to Tables 3B.13-7 and 3B.13-8 for comparison between the results obtained from STRUDL and GT-STRUDL.

Problem No. 5

A comparison check is performed for suspended ceiling design using the dynamic analysis capability of STRUDL versus GT-STRUDL. A model is provided in Figure 3B.13-3. The loading condition accounts for the dynamic seismic loads resulting from ceiling dead load. Refer to Tables 3B.13-9, 3B.13-10, and 3B.13-11 for comparison between the results obtained from STRUDL and GT-STRUDL.

3B.14 STRUDAT AND SANDUL

STRUDAT and SANDUL were developed by SWEC and are fully qualified and documented. The computer programs perform ASME III, Appendix XVII and AISC code checks and size fillet welds for pipe support frames and anchors for various loading conditions and combinations.

STRUDAT is a post-processor for STRUDL-SW and translates STRUDL-SW output for applied unit loads into a protected disk file, subsequently accessed by SANDUL. This access is fully traceable.

SANDUL combines the load components for each applied load into the appropriate load conditions, then applies the appropriate signs to each applied load for each loading condition. This load condition matrix is multiplied by the unit load member force and unit load reaction matrixes from STRUDAT to generate the computed member forces and reactions. This is based on the principle of superposition. These member forces are used to calculate the required fillet weld sizes and member stresses. Allowable stresses are also computed for each member. The output provides a ratio of the member stresses and required weld size for each member and weld, respectively. In addition, a summary of the support reactions is provided in the output.

The STRUDAT and SANDUL computer programs have been verified by comparison with STRUDL-SW and hand computations. These results are presented in Table 3B.14-1 through 3B.14-3. The sample model is shown on Figure 3B.14-1. This comparison provides results within an acceptable range.

3B.15 CDC - BASEPLATE II

General Description

BASEPLATE II is a combination of a pre and post-processor to the STARDYNE program for the purpose of analyzing flexible baseplates on a geometrically non-linear foundation. The program employs an automatic mesh generation technique with the user in control of mesh size and element configuration.

Input includes plate geometry, nonstandard and standard (library) attachments, anchor locations, anchor and concrete stiffnesses, material properties, anchor allowables, and up to 50 loading conditions.

Output consists of a printer plot of the baseplate showing attachment and anchor locations, plate deformations and principal stresses, anchor bolt tension, and resultant shear load for each anchor, together with calculated tension/shear interaction and factor of safety.

Program Verification

BASEPLATE II is verified and qualified through Control Data Corporation quality assurance programs, which are periodically evaluated by SWEC and which have been found to be satisfactory.

3B.16 BIP

General Description

BIP (Baseplate Information Processor) is a set of two programs, BIP1 and BIP2, which are pre and post-processors to ANSYS. These programs reduce the time required for baseplate analysis. Baseplates are analyzed for in-plane and out-of-plane loads that are transferred through the attachments. The baseplate may be treated as infinitely rigid for in-plant loads, resulting in a statically determinate solution for anchor bolt shear loads. Out-of-plane loads are analyzed by ANSYS to account for plate flexibility as well as gaps between the baseplate and concrete and interference fit with anchors.

Input to BIP includes plate and attachment geometry, anchor locations, anchor and concrete stiffness values, anchor and concrete gaps, material properties, anchor allowables with reduction factors, and up to 10 loading conditions.

Output consists of an input echo, a printer plot of the baseplate showing attachment and anchor locations, resultant shear at each anchor together with reduced tension allowables based on tension-shear interaction, plate deformations and stresses, and reactions (including bolt pullout loads).

Program Verification

The BIP program is a publicly available program and is verified and qualified through Boeing Computer Services Quality Assurance Programs. These are periodically evaluated by SWEC and have been found to be satisfactory.

3B.17 APE

General Description

Computer program APE (Anchor Plate Evaluation) calculates the shear and tension loads for each anchor of a group of drilled-in anchors of a baseplate subjected to in-plane and out-of-plane loads. A reduced tension allowable, based on the calculated shear load and tension shear interaction, is also calculated.

Program input includes plate and attachment geometry, anchor bolt locations and allowables, and applied loads.

Output consists of anchor bolt pattern center of gravity and polar moment inertia and resultant shear and allowable tension load for each anchor. Load factors for out-of-plane loading, and anchor tension loads, including load factors, for each anchor.

Program Verification

Verification of the APE program (Version 01, Level 00) was performed by comparing results of APE analyses with similar results obtained from Boeing Computer Services Program BIP (refer to Section 3B.16). Comparison of results from APE (Version 01, Level 00) and Boeing Computer Services Program BIP are shown in Table 3B.17-1. APE results are shown to be conservative or comparable with respect to BIP. All results are based on loadings at the critical anchor.

3B.18 CHPLOT

CHPLOT is a program which will plot any number of data values (variables) versus time. Although the plot input data file can be in the form of card data, the more appropriate application of this program is to be used in conjunction with a program that creates a plot data file (on disk or tape) having the format required for input to this program.

Plots are available in two sizes: one with axes of 5 in. (Ordinate) by 8 in. (Absicissa) which fits the standard 8 1/2 in. by 11-in. page, and the other is 8 in. by 12 in. for fitting an 11-in. by 15-in. page. Plots are normally one data value versus time per graph, although up to 14 data values (plots) can be plotted on one graph.

3B.19 RELAP5

General Description

RELAP5 is a public-domain computer program which was written by Idaho National Engineering Laboratory and is available from several sources including Control Data Corporation (CDC). The program may be used to evaluate the behavior of a system subjected to postulated transients such as loss of coolant from large or small pipe breaks, pump failures, etc., or may be used to evaluate the thermal-hydraulic response of fluid transients in piping systems.

RELAP5 uses a five equation two-phase flow hydrodynamic model consisting of two phase continuity equations, two-phase momentum equations and an overall energy equation augmented by the requirement that one of the phases is assumed saturated. In this model, only two interphase constitutive relations are required, those for interphase drag and interphase mass exchange. Models are included for abrupt area changes, choking, mass transfer interphase drag, wall friction and branching.

For hydrodynamics, the space approximation uses a staggered mesh where integral forms of the continuity and energy equations are approximated over control volumes and line integral forms of the momentum equations are applied from the midpoint of one control volume to the midpoint of the adjoining control volume. Hydrodynamic equations are advanced in time using a semi-implicit, linearized method. Head conduction is approximated by finite differences and advanced by the Crank-Nicolson scheme. A modified Runge-Kutta technique for stiff equations is used to

solve the reactor kinetics equations. The interaction among hydrodynamics, heat conduction, trips, reactor kinetics and the control system is explicit.

The program requires numerical input data that completely describe the initial fluid conditions and geometry of the system being analyzed. The input data include physical characteristics such as fluid volume geometry and pump characteristics, range of time step size, output variables, output frequency and trips. The piping network is described to RELAP5 as a set of connected control volumes. RELAP5 computes fluid properties within these volumes and in the connecting flow junctions at discrete time points (steps).

In general the program provides, as output, variables necessary to describe the transient state of the system being analyzed. These results include, as applicable, time varying pressure, momentum flux and energy states throughout a fluid system containing water, steam, and/or a two-phase mixture.

Program Verification

RELAP5 is a recognized program in the public domain.

3B.20 NUDL (ME-268)

This computer program is a pipe support load combination generator that serves as an aid in the design of pipe supports. The program reads the pipe support loads from a file created by NUPIPE-SW, combines them according to a set of specified load combination equations, and outputs them to the printer in tabular format and to a disk file in STRUDL command format.

This computer program has been qualified by SWEC in accordance with SWEC Engineering Assurance Procedures.

3B.21 CCW-318

This program's purpose is to evaluate the design of rectangular cross section welded attachments on ASME Code Class 2 and 3 and ANSI B31.1 piping using Code Case N-318-1, -2, -3.

This program has been qualified by comparing the results against those of hand calculations.

3B.22 CCW-392

This program's purpose is to evaluate the design of hollow circular cross section integral welded attachments to ASME Code Class 2 and 3 and ANSI B-31.1 piping using Code Case N-392.

This program has been qualified by comparing the results against those of hand calculations.

3B.23 LOTUS

ME-277 is a collection of worksheets which have been converted to LOTUS 1-2-3 templates closely replicating the printed forms, allowing the assistance of an IBM PC in performing these tasks.

The ME-277 templates cover calculations on weld stresses, local and bearing stresses, code checks, and other stress calculations for pipe support evaluation. These evaluations are primarily to confirm the Subsection NF or AISC code requirements.

This program has been qualified by comparing the results against those of hand calculations.

3B.24 ANSYS (ST-384)

ANSYS is a general purpose finite element analysis program. It can be used to evaluate problems which are not solvable through standard project methods. Among the available options are static and dynamic finite element analysis for both linear and nonlinear behavior. Various types of elements are available for optimal solution to the particular problem.

This program has been qualified by SWEC in accordance with SWEC Engineering Assurance Procedures.

3B.25 PSAP (HY-066)

This program performs a comprehensive hydraulic analysis and design of a network piping system. It provides energy gradients, pressures, and flow rate distribution for the entire system under steady state flow balance condition.

This program has been qualified by SWEC in accordance with SWEC Engineering Assurance Procedures.

3B.26 REPIPE (ME-279)

This program computes the loading time-history on a piping network based upon results from computer program RELAP5 hydrodynamic analysis of the contained fluid. The time-varying pressure, momentum flus, and energy states throughout a fluid system containing water, steam, and/or a two-phase mixture from RELAP5 output are used as an input to the REPIPE program to produce force time-histories for input to the piping stress analysis program.

This program has been qualified by SWEC in accordance with SWEC Engineering Assurance Procedures.

3B.27 SCAP

The SCAP Computer Program calculates the resulting forces and pipe local stresses at a stiff clamp U-bolt pipe support caused by all possible load conditions. The methodology used is based on CPPP-7, Rev. 4, Attach. 4-6C, D & E.

The SCAP program is applicable only to the analysis of a stiff clamp attached to a single strut or snubber.

The SCAP program computes the resulting forces and local stresses in the pipe. The local pipe stresses are checked against the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Class 2 and 3 piping allowables, and ANSI B31.1 code allowables for different load conditions.

This program has been qualified by comparing the results against those of hand calculations.

3B.28 ETA

This program's purpose is to evaluate the design of hollow circular cross section integral welded attachments to ASME Code Class 2 and 3 and ANSI B31.1 piping elbows using Code Class N-392 and SWEC analyzed stress intensification factors, per CPPP-7, Rev. 4, Attach. 4-6H.

This program has been qualified by comparing the results against those of hand calculations.

3B.29 HYDTRAN (HY-001)

This program simulates the hydraulic transient response in a piping system resulting from pump startup, pump shutdown, pump trip, and/or valve operation and was developed for use with a generalized complex hydraulic system. It is capable of analyzing various combinations of hydraulic components normally encountered in power plant systems. The method of characteristics is used for the solution of these transients which produces pressures, flows, and flow-induced forcing functions which may be input to a dynamic analysis such as a NUPIPE program run.

This program has been qualified by SWEC in accordance with SWEC Engineering Assurance Procedures.

3B.30 LEAP, (ME-101) LINEAR ELASTIC ANALYSIS OF PIPING

Author: Bechtel Corporation, San Francisco, California
Source: Bechtel Corporation, San Francisco, California

Description

ME101 is a finite element computer program which performs linear elastic analysis of piping systems, transient thermal analysis, local stress analysis, and response spectra merging.

The input data format is specifically designed for pipe stress analysis. ME-101 performs a thorough check of the input prior to performing analysis. The program rearranges the geometry automatically to optimize the finite element model. The coordinate and keyword data can be specified in English, Metric or S1 units. ME101 performs static and dynamic analysis of piping systems in accordance with Section III of the ASME Boiler and Pressure Vessel Code, ANSI B31.1 Power Piping Code and ANSI B31.3 Chemical Plant and Petroleum Refinery Piping Code. The flexibility factors, stress intensification factors (SIF) and stress indices (B1, B2) are incorporated in the stress evaluations.

Static analysis capabilities include thermal expansion, deadweight, seismic and uniformly distributed loads, and externally applied forces, moments, displacements and rotations. For dead weight and thermal expansion analysis, supports with gaps can be considered.

Response spectrum analysis is based upon standard modal superposition techniques. The input excitation may be in the form of a single or multiple seismic response spectra or time dependent loading functions. In the response spectrum analysis, the user may request modal combinations by various alternate methods typical of the industry. ME101 can further consider differential damping according to NRC RG 1.61 and PVRC Code Case N-411 within a single response spectrum analysis.

ME101 uses an out-of-core solution and active column techniques for both static and dynamic analyses. It has no limit to the number of equations or bandwidth.

ME101 considers zero period accelerations with or without missing mass corrections in the seismic response spectrum analyses. In the time history analysis (modal or direct integration), the excitation may be in the form of nodal forces, support displacements or support accelerations. ME101 can also compute the response to dynamic loads using a direct integration time history method considering nonlinear kinematic hardening supports (such as the Bechtel energy absorber) with or without gaps. Transient analyses and response spectrum analyses of this nonlinear type support are also available. ME101 compares stresses vs. allowables according to ASME/ANSI code equations. ME101 has load combination capabilities that allow the results of several load cases to be combined according to certain algebraic rules to form additional load cases. The additional load cases resulting from the combination may be utilized in stress comparisons or restraint load summaries. ME101 also provides a customized report writer that is designed to create report or table use in equipment nozzle evaluation, piping support/hanger guidance, valve acceleration summary, penetration load summary etc. This report is construed directly from data stored in the internal ME101 Data Base ("MASTER" file). This report enables user to manually input the allowable side by side with the extracted value, or allows the user to uniquely define the interaction equations. The report writer module can also screen or filter results according to allowable criteria.

ME101 can generate isometric plots of the piping configuration with optional node numbering. It can also generate plots for deflections and mode shapes.

The program is validated utilizing fifty-nine validation problems, including the eleven NUREG/CR 1677 benchmark problems. Validation is documented in the ME101 program validation manual.

3B.31 FAPPS (ME150), FRAME ANALYSIS PROGRAM FOR PIPE SUPPORTS

Author: Bechtel Corporation, San Francisco, California
Source: Bechtel Corporation, San Francisco, California

FAPPS is an interactive program specifically developed for the analysis of pipe support frames, associated welds, baseplates, embedments and local effects such as punching shear, web crippling and local flange bending. It performs static linear analyses of elastic structures. The program performs code checks for AISC, ASME III subsection NF and AIJ codes for "normal," "upset," "emergency" and "faulted" load conditions. The program addresses recent NRC and industry issues. It provides margin factors for all analyzed components.

Validation is documented in the program validation manual.

3B.32 BASEPLATE (ME035), NONLINEAR BASE PLATE ANALYSIS PROGRAM

Author: Bechtel Corporation, San Francisco, California
Source: Bechtel Corporation, San Francisco, California

ME035 is a finite-element program for the evaluation of baseplates. It has the capability of analyzing flexible baseplates on a geometrically nonlinear foundation. It has a library of standard attachments and accepts nonstandard and multiple attachments. It accommodates welded and bolted conditions of the baseplate.

Validation is documented in the program validation manual.

3B.33 MAPPS (ME153), MISCELLANEOUS APPLICATION PROGRAM FOR PIPE SUPPORTS

Author: Bechtel Corporation, San Francisco, California
Source: Bechtel Corporation, San Francisco, California

MAPPS is an interactive program that performs various pipe support related analyses. A description of the several analytical modules of MAPPS is provided below:

a. UNIFORM WELD ANALYSIS:

This module presents a method of determining weld size for the connecting structural members based on an approach described in Design of Welded Structures and Design of Weldments, both by O.W. Blodgett. It accepts five different types of shapes and analyzes from two to sixteen weld configurations. The program computes weld properties, minimum weld requirements, and adjusts weld effective throats to account for skewed angles. It also computes weld stresses and margin factors.

b. NONUNIFORM WELD ANALYSIS:

This module provides analysis of any two dimensional, nonuniform weld pattern with variable effective throat. It permits the user to model the weld as a series of joints and elements for welds without a constant effective throat over the entire weld configuration. Actual “as-built,” skewed T-joint and pipe-to-pipe connections are examples where effective throat over the entire weld length is not always achieved. The program gives the user the choice of either algebraic or absolute combinations of stress components and computes the combined stresses at every joint of the weld.

c. CLIP ANGLE ANALYSIS:

This module analyses stresses in clip angles used in pin-ended connections. The program also analyzes welds between clip angles and framed members and clip angles and supporting members. The program analyzes clip angles for three forces and for axial moment. It also permits variations of the orientation angle between the framed member and supporting structure.

d. BETA ANGLE CHECK:

The program determines “beta” angles used in programs like STRUDL and FAPPS. Beta angles are measured in a plane normal to the longitudinal axis of a structural member and serve to define the orientation of the member’s cross-section.

e. LOCAL EFFECT EVALUATION:

The program analyzes W-beam connections for local effects due to local flange bending and web crippling. It also performs a punching shear and web crippling analysis for tube and pipe members.

Validation is documented in the program validation manual.

3B.34 SIGNIT (ME149), SIGNATURE READY ENGINEERED CALCULATION SYSTEM FOR PIPE SUPPORTS

Author: Bechtel Corporation, San Francisco, California
Source: Bechtel Corporation, San Francisco, California

SIGNIT provides a system for generating “signature-ready” calculations for pipe supports that integrates all necessary program outputs together, develops the necessary portions of the calculations such as objective, methodology, conclusions, summary of results, check list, assumptions, sources of data and references and organizes the calculation package.

Validation is documented in the program validation manual.

3B.35 CAPPS (ME148), COMANCHE PEAK APPLICATIONS PROGRAMS FOR PIPE SUPPORTS

CAPPS is an interactive program which serves as a special application program for analysis of various pipe support components to their specific requirements. The various modules that cover these special application are listed below. This program has been specifically developed to address unique requirements applicable to pipe support design at Comanche Peak. The program provides to the user instructions regarding key points to be considered in the design and key data to be used in the form of a technical advisor. It calculates allowable stresses for plates and welds for all load conditions and all pipe support class in various buildings. It also provides footprint loads in project specific form to be utilized for evaluation of building structures. The various modules which are included in this program are:

Friction clamp Anchor, Intermediate Plate, Closed End Tubing, Richmond Inserts, Nuts on tubing, Flare bevel weld, Pipe weld to pipe/elbow and Strut/Snubber Spring/U-bolt/U-strap.

The program has been verified using hand calculations, validation is documented in the program validation manual.

3B.36 LSAPS (ME214) LOCAL STRESS ANALYSIS FOR PIPING SYSTEMS

Author: Bechtel Corporation, San Francisco, California
Source: Bechtel Corporation, San Francisco, California

ME214 is an interactive computer program developed to perform local stress analysis due to integral welded attachments and non-integral welded attachments such as line bearing and point contact loads, restraint of local pressure and thermal expansion of the pipe. The program is capable of calculation local pipe stresses based on Welding Research Council Bulletin No. 107 and ASME Code Case N-318 and N-392. The program also calculates local pipe stresses due to non-integral welded attachments using analytical solutions as provided in Comanche Peak design criteria documents.

The program is developed in accordance with ASME Section III Subsection NC and ND. The loading conditions and allowables are in accordance with Subsection NC and ND piping stress

requirements. Code case allowables are used if the local pipe stresses are calculated based on the methodology provided in the code cases.

3B.37 WESTDYN AND ASSOCIATED PRE- AND POST-PROCESSORS

Author: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania
Source: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania

WESTDYN is used for the structural analysis of piping system in Unit 2 only. WESTDYN, (described in WCAP-8252), calculates displacement, forces, and stresses distributions in three-dimensional piping models subjected to static and dynamic loads.

The dynamic analysis incudes seismic or hydrodynamic response spectra and time-history dynamic analysis. The time-history dynamic analysis incudes options for nonwear support, support gaps, and unidirectional single-acting restraints.

The following standard post-processing programs are provided as a part of the WESTDYN system: SELCOMB, STRSCHK, WESFOR, VALVE and CONFOR. In addition the following time history pre- and post-processing programs are provided: MERGE, HYDFO, SUPFOR, FORCE and DEFLOUT.

The WESTDYN post-processors provide for the stress analysis of ASME Class 1, 2, 3, or ANSI B31.1 code. The Post-processors are also used for generating support load summary sheets, and equipment and component qualification input data.

This family of programs has been verified in accordance with the Westinghouse QA program as specified in WCAP-9565.

3B.38 PIPSAN

Author: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania
Source: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania

PIPSAN (PIPe Support ANalysis) is an interactive program utilizing finite beam elements to model, analyze, and evaluate three-dimensional linear elastic structures in Unit 2 only. The required input consists of model geometry; connectivity of members; cross-sectional and material properties; external restraints; and location, direction, and magnitude of applied loads. Loading conditions (i.e., normal, upset, emergency or faulted) are required for stress evaluation.

This program has been verified in accordance with the Westinghouse QA program as specified in WCAP-9565.

3B.39 WESPLAT

Author: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania
Source: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania

WESPHALT is used for the analysis of any rectangular plate attached by anchors to a rigid foundation in Unit 2 only. WESPLAT utilizes a non-linear plate bending finite element approach.

The baseplate is modeled using an assembly of plate elements at node points. The plate is divided into elements dependent upon the user-specified problem.

This program has been verified in accordance with the Westinghouse QA program as specified in WCAP-9565.

3B.40 LOCAL

Author: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania
Source: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania

Program LOCAL is a PC based computer program which evaluates pipe primary and secondary local stresses that are due to support attachments in Unit 2 only. It also computes fatigue local stress effects due to upset and thermal loading for Class 1 pipe support attachments. The types of attachments considered include U-bolts, anchors, super stiff clamps, straps, lugs, trunnions, pads, and structural beams in contact with the pipe. The analysis method is in accordance with the procedures outlined in

- WRC Bulletin #107
- ASME B&PV Code, Section III, 1977 Edition through Summer, 1979 Addenda.
- Code Case N-391, 11/28/83
- Code Case N-392, 11/28/83
- Code Case N-122, 1/21/82

Program verification has been performed in accordance with the Westinghouse QA program as specified in WCAP-9565.

3B.41 PCLUG

Author: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania
Source: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania

PCLUG is a PC based program used to evaluate secondary stresses due to integral structural attachments to straight piping in Unit 2 only. The evaluations are made as per the ASME Code Case N-318-2, for Class 2 or 3 piping.

Program verification has been performed in accordance with the Westinghouse QA program as specified in WCAP-9565.

3B.42 STRESS CALCULATION SPREADSHEETS

Author: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania
Source: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania

Lotus 123 release 2.2 with the IMPRESS add-in is used to perform several repetitive stress calculations in Unit 2 only. Use of a spreadsheet assures consistency of calculations and permits engineers to focus on the engineering process vs. hand calculator arithmetic. IMPRESS add-in prints the resulting calculations in a neat, legible form.

The following spreadsheets are in use:

1. Swing Angle - calculates swing angles, binding angles, component extension/retraction (as applicable) for springes, snubbers, and struts.

2. Tube Steel Local Stress - analyzes acceptability of local stress on tube steel connections. This is actually a family of four spreadsheets. One is simply a menu allowing the operator to select one of the three analysis spreadsheets. Each spreadsheet addresses one of the following weld configurations: two-sided weld normal to main member, two-sided weld parallel to main member, and three/four sided welds. TUBSTL01, 02, and 03.WK1.

3. Tube Steel Welds - analyzes forces on specific weld configurations for a tube steel-to-tube steel weld, tube steel-to-base plate weld, and finally, flexibility was included to allow analysis of tube steel-to-rear bracket welds.
WELDTS01.WK1.

Additional spreadsheets will be developed as the need arises based on the approach stated above and the QA requirements as stated below.

The spreadsheet quality control complies with the NATD Quality Assurance Program, WCAP-9565, which defines the requirements for the development, verification, configuration management, and error reporting of software related nuclear Basic Components.

3B.43 WECAN

Author: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania
Source: Westinghouse Electrical Corporation, Pittsburgh, Pennsylvania

WECAN is a general purpose finite element program with the capability to solve a wide variety of structural analysis problems in Unit 2 only. These problems can be one, two, or three-dimensional in nature. Capabilities include:

- Static elastic and inelastic analysis
- Steady state and transient heat conduction
- Steady state hydraulic analysis
- Full and reduced modal analysis
- Harmonic response analysis
- Dynamic transient analysis
- Linear buckling analysis

Bench mark problem solutions for the verification of WECAN are contained in WCAP-8929. WECAN verified is performed in accordance with the Westinghouse QA program as specified in WCAP-9565.

3B.44 ME215 (ME101FE), SAPCAS

Author: Bechtel Corporation, San Francisco, California
Source: Bechtel Corporation, San Francisco, California

ME215 (ME101FE) is a special purpose finite element computer program for computing membrane and membrane-plus-bending stress intensities at pipes, pads, attachments, and welds. The piping components can be a circular run pipe, rectangular tube, or rectangular solid lug. The program is capable of computing local stress on pipe components and supports such as stanchion, trunnion, lug, etc. This program is more refined and will provide a realistic solution for local pipe stress as compared to, conservative ME214. The program is capable of calculating local pipe stresses where the results may exceed the limitation of the welding research council (WRC) Bulletin No. 107, ASME code cases, and/or other design documents such as CPNPP 2EP-5.12 and 2EP-5.13. The program is developed based on well established SAP computer program.

3B.45 HSTA (NE820) HYDRAULIC SYSTEM TRANSIENT ANALYSIS

Author: Bechtel Software, Inc., San Francisco, California
Source: Bechtel Software, Inc., San Francisco, California

Description

The Hydraulic System Transient Analysis (HSTA) is a PC based computer code that analyzes flow transients in liquid piping systems. The program generates piping segment time history forcing functions, as water hammer events, to be input into piping stress dynamic analysis codes such as ME101 for overall piping and support stress evaluations. The HSTA program solves one-dimensional unsteady flow problems utilizing the mass and momentum conservation equations (along the pipe axis), to obtain pressure and velocity as functions of time and distance. The solution is based on the finite difference solution of the method of characteristics (MOC), where the differential equations of flow are linearized to analyze transients resulting from valve opening and closure, pump startup or seizure, rapid decompression, column separation, and filling of a voided line. The code can also be applied to steam or gaseous systems where no significant changes to fluid density occurs during the transients.

HSTA input includes the piping modeled as a network in the form of links connected to each other through junctions. Hydraulic devices are located at these junctions and treated as boundary conditions. Piping frictional losses are also considered and the code enables modeling changes in elevation. The output features has the capability of providing snapshots of head or velocities at all nodes of selected links viewed on computer monitor. Other features including echoing of input date, printouts of pressure and velocities, and plotting of time-history pipe segment forces in a format compatible for input to ME101 piping stress code. A Windows compatible post-processor program (PLOTHG2) is used for plotting the results that are computed by HSTA as provided in the Code's user manual. The HSTA Code Validation & Verification process, that included the control and execution of a Quality Software application used by Engineering for Piping Stress and Support Analysis, was performed in accordance with the requirements of Station Procedures. Once the HSTA program is executed, the output file "input.o" is generated. There are additional files geneated depending on the options selected in the input file. These output files are identified by various file extensions as follows:

File Extension	Type
Input .o	Normal Output File
.b	Bubble Size file
.m	ME101 - format force time history file

.t	Columnar - format force time history file
.h	Head plot file
.v	Velocity plot file
.f	Pipe-run force plot file
.a	Surge Vessel plot file
.r	Restart file
.s	Save file

Another executable file is used to run the plotting program by placing the file in the same directory where the input files are located.

Furthermore, the Validation Report by Bechtel Software addresses limitations and comparisons regarding the validation between the program output results and either measured or independently calculated data for valve closure, branching, pump failure, open surge vessel (with one-way check valve), air tank (for suppression of pressure surges), positive displacement pump, liquid (water) column separation in a horizontal line or a siphon system, and a line filling case.

The HSTA program is compatible with other industry recognized codes utilized in CPNPP piping analysis, namely, “WATHAM” and “HYDTRAN”. Nonetheless, for a given analytical model previously analyzed by another equivalent code and is to be executed under the HSTA Code, it is the responsibility oft he user to ensure acceptable comparisons are obtained and consistency in the input/output results is maintained and in accordance with the applicable design criteria. This is typical practice for the execution of a compatible code performing similar functions when the code of the analysis of record is no longer available. Also, to ensure the safety of the program and related data files, other areas that are also checked is the virus detection and protection software included in the booting process activated whenever the computer is turned on. Any error reporting and/or corrective action are addressed. Any interference between such software and the Bechtel computer applications would be detected during the validation procedures described above.

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TABLE 3B.1-1
COMPARISON OF NATURAL FREQUENCIES FOR FIGURE 3B.1-1

	Frequency (Hz)	
Mode	NUPIPE-SW	NRC
1	28.510	28.53
2	55.698	55.77
3	81.411	81.50
4	141.618	141.7
5	162.633	162.8

TABLE 3B.1-2
COMPARISON OF INTERNAL MEMBER LOADS AND NODAL DEFLECTIONS COMBINED RESULTS FOR MODES 1-5 FOR FIGURE 1.0

	Member	Forces			Moments			Deflections		
Source	Ends	Fx	Fy	Fz	Mx	My	Mz	Dx	Dy	Dz
NUPIPE-SW	1	5.	18.	36.	53.	269.	116.	.000	.000	.000
	2	5.	18.	36.	53.	106.	39.	.002	.000	.005
NRC	1	5.	18.	36.	52.	269.	116.	.00000	.00000	.00000
	2	5.	18.	36.	52.	105.	39.	.00200	.00000	.00480
NUPIPE-SW	3	5.	7.	8.	53.	34.	29.	.006	.000	.015
	4	6.	7.	8.	12.	69.	24.	.007	.001	.017
NRC	3	5.	7.	8.	52.	34.	29.	.00580	.00000	.0146
	4	6.	7.	8.	12.	69.	24.	.00740	.00060	.017400
NUPIPE-SW	4	9.	9.	11.	12.	69.	24.	.007	.001	.017
	5	12.	4.	11.	35.	45.	22.	.008	.002	.014
NRC	4	9.	9.	11.	12.	69.	24.	.00740	.00060	.01740
	5	12.	4.	11.	35.	45.	22.	.00780	.00160	.01420
NUPIPE-SW	7	28.	24.	4.	35.	21.	41.	.008	.003	.007
	8	25.	27.	4.	35.	18.	73.	.006	.002	.002
NRC	7	28.	24.	4.	35.	20.	41.	.00780	.00249	.00680
	8	25.	27.	4.	35.	18.	73.	.00580	.00199	.00168
NUPIPE-SW	8	26.	30.	6.	35.	18.	73.	.006	.002	.002
	9	24.	31.	6.	9.	45.	115.	.002	.001	.000
NRC	8	26.	30.	6.	36.	18.	73.	.00580	.00199	.00168
	9	24.	31.	6.	9.	45.	115.	.00220	.00074	.00000
NUPIPE-SW	10	24.	7.	35.	9.	156.	54.	.001	.000	.000
	11	24.	7.	35.	9.	207.	65.	.000	.000	.000
NRC	10	24.	7.	35.	9.	156.	54.	.00062	.00019	.00000
	11	24.	7.	35.	9.	206.	65.	.00000	.00000	.00000

TABLE 3B.1-3
COMPARISON OF NATURAL FREQUENCIES FOR FIGURE 3B.1-2

	Frequency (Hz)	
Mode	NUPIPE-SW	NRC
1	8.711	8.712
2	8.805	8.806
3	17.507	17.51
4	40.364	40.37
5	41.624	41.63

TABLE 3B.1-4
COMPARISON OF INTERNAL MEMBER LOADS AND NODAL DEFLECTIONS COMBINED RESULTS FOR MODES 1-5 FOR FIGURE 3B.1-2

	Member	Forces			Moments			Deflections		
Source	Ends	Fx	Fy	Fz	Mx	My	Mz	Dx	Dy	Dz
NUPIPE-SW	15	767.	109.	108.	0.	428.	436.	.000	.000	.000
	1	757.	109.	108.	0.	23.	20.	.230	.001	.225
NRC	15	766.	109.	108.	0.	428.	436.	.00000	.00000	.00000
	1	766.	109.	108.	0.	23.	20.	.23000	.00127	.22500
NUPIPE-SW	1	766.	78.	77.	0.	23.	20.	.230	.001	.225
	7	766.	78.	77.	0.	337.	337.	.462	.003	.447
NRC	1	766.	78.	77.	0.	23.	20.	.23000	.00127	.22500
	7	766.	78.	77.	0.	337.	337.	.46180	.00253	.44620
NUPIPE-SW	7	7.	469.	13.	0.	6.	337.	.462	.003	.447
	6	7.	469.	13.	0.	3.	0.	.462	.001	.447
NRC	7	7.	469.	13.	0.	6.	337.	.46180	.00253	.44620
	6	7.	469.	13.	0.	3.	0.	.46180	.00097	.44620
NUPIPE-SW	5	19.	297.	13.	0.	6.	337.	.462	.003	.447
	14	19.	297.	13.	0.	4.	134.	.462	.003	.447
NRC	5	19.	297.	13.	0.	6.	337.	.46180	.00252	.44620
	14	19.	297.	13.	0.	4.	124.	.46180	.00332	.44630
NUPIPE-SW	14	7.	297.	1.	0.	4.	124.	.462	.003	.447
	13	7.	297.	1.	0.	4.	124.	.462	.003	.447
NRC	14	7.	297.	1.	0.	4.	124.	.46180	.00332	.44630
	13	7.	297.	1.	0.	4.	124.	.46180	.00332	.44630

TABLE 3B.1-5
COMPARISON OF SUPPORT LOADS BETWEEN NUPIPE II AND NUPIPE-SW
FOR FIGURE 3B.1-3

Program	Node	FX	FY	FZ	MX	MY	MZ
NUPIPE II	5520	556.	437.	961.	1102.	2202.	530.
NUPIPE SW		560.	436.	966.	1097.	2189.	533.
NUPIPE II	158		1361.				
NUPIPE SW			1344.				
NUPIPE II	159		1380.				
NUPIPE SW			1384.				
NUPIPE II	178	2019.	1060.	966.	7853.	21812.	3588.
NUPIPE SW		2020.	1059.	965.	7838.	21794.	3581.

TABLE 3B.1-6
COMPARISON OF SUPPORT LOAD FOR FIGURE 3B.1-4

	Mode	Node Pt.	FX(lb)	MX(ft-lb)
NUPIPE-SW with missing mass cutoff mode = 2.0	1,2, PSEUDO MODE	5	2072	42114
NUPIPE-SW with no missing mass cutoff mode = 3.0	1-3 SRSS	5	2072	42114

TABLE 3B.1-7
COMPARISON OF SUPPORT REACTIONS DUE TO THERMAL EXPANSION
FOR FIGURE 3B.1-5

Node	Program	Force (lb)			Moments (ft-lb)		
		FX	FY	FZ	MX	MY	MZ
1	NUPIPE-SW	-3275	-2934	-1214	-9307	2858	22302
1	ADLPIPE	-3274	-2934	-1214	-9307	2858	22302
11	NUPIPE-SW	3275	2934	1214	15281	-14855	-9415
11	ADLPIPE	3274	2934	1214	15281	-14855	-9415

TABLE 3B.1-8
COMPARISON OF DEFLECTIONS DUE TO THERMAL EXPANSION FOR
FIGURE 3B.1-5

Node	Program	Deflection (in.)		
		DX	DY	DZ
2	NUPIPE-SW	-0.074	0.135	-0.032
2	ADLPIPE	-0.074	0.135	-0.032
5	NUPIPE-SW	-0.132	0.259	-0.093
5	ADLPIPE	-0.132	0.259	-0.093
7	NUPIPE-SW	0.014	0.084	-0.157
7	ALDPIPE	0.014	0.084	-0.157

TABLE 3B.1-9
COMPARISON OF STRESS DUE TO THERMAL EXPANSION FOR
FIGURE 3B.1-5

Node	NUPIPE-SW (psi)	ADLPIPE (psi)
1	6411	6410
2	1158	1158
3	6622	6619
5	6938	6936
7	4908	4906
9	2725	2724
11	6138	6137

TABLE 3B.1-10
COMPARISON OF INTERNAL FORCES AND MOMENTS DUE TO THERMAL EXPANSION FOR FIGURE 3B.1-5

Node	Program	Force (lb)			Moments (ft-lb)		
		FX	FY	FZ	MX	MY	MZ
2	NUPIPE-SW	-3275	-2934	-1214	-2024	2858	2655
2	ADLPIPE	-3274	-2934	-1214	-2024	2858	2655
5	NUPIPE-SW	-3275	-2934	-1214	5258	1340	-13324
5	ADLPIPE	-3274	-2934	-1214	5258	1340	-13324
7	NUPIPE-SW	-3275	-2934	-1214	5258	-6549	5748
7	ALDPIPE	-3274	-2934	-1214	5258	-6549	5748

TABLE 3B.3-1
COMPARISON OF BSPLT COMPUTER PROGRAM RESULTS WITH HAND CALCULATION RESULTS

Input Data

Loads

F_X = 4201 lb

F_Y = 3785 lb

F_Z = 0

M_X = 0

M_Y = 0

M_Z = 7857 in.-lb

Output Data	Hand Calculation	BSPLT
Bolt Tension	2188 lb	2175 lb
Bolt Shear	946 lb	946 lb
Tension-Shear		
Interaction Factor	0.66	0.65
Plate Bending Stress	8637 psi	8534 psi

TABLE 3B.5-1
COMPARISON OF PITRUST WITH FRANKLIN INSTITUTE PROGRAM, CYLNOZ, AND HAND CALCULATION

Source of Stress	Stress (psi) Franklin Institute Corrected Values	Output From PITRUST	Hand Calculation
Circumferential			
P (normal)	395	399	399.99
P (bending)	1,875	1,833	1,877.3
M_c (normal)	35.85	35.57	36.06
M_c (bending)	364.7	366.6	354.3
M_L (normal)	79.05	79.66	79.54
M_L (bending)	90.52	80.57	79.42
Axial			
P (normal)	813	812	814.8
P (bending)	812.3	827	810.6
M_c (normal)	91.79	105	95.45
M_c (bending)	158.8	158.8	160
M_L (normal)	37.06	37	37.12
M_L (bending)	117.9	105	103.85
Shear Stress by M_T	6.63	6.63	6.63
Shear Stress by V_c	106.1	106.1	106.1
Shear Stress by V_L	106.1	106.1	106.1

TABLE 3B.5-2
COMPARISON OF PITRUST WITH REFERENCE 2 RESULTS

Location and Cause	PITRUST Results (psi)	Experimental Results (psi)
Element A Longitudinal Moment, M_L		
Circumferential Stress	20,438.9	20,000
Axial Stress	26,292.6	25,000
Element B Longitudinal Moment, M_C		
Circumferential Stress	22,016.2	24,000
Axial Stress	13,105.8	13,000

TABLE 3B.6-1
COMPARISON OF PILUG COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS

(Sheet 1 of 2)

Test Problem: Run Pipe Outside Diameter = 17 in.

Run Pipe Thickness = 0.812 in.

Axial Length of LUG = 12 in.

Width of LUG along Circumference = 3 in.

Loads: P = 3300 lb

V_c = 1788 lb

V_L = 2478 lb

M_c = 81834 in.-lb

M_L = 103320 in.-lb

M_T = 76284 in.-lb

Stress in Circumferential Direction (psi)

		Stress from		
Figure[a]	B	Hand Calculation	Computer Output	Remarks
3C	0.5485	387	330	Membrane stress due to P
1C	0.326	2,165	2,160	Bending stress due to P
3A	0.294	671	629	Membrane stress due to M_c
1A	0.388	18,976	19,904	Bending stress due to M_c
3B	0.467	3,014	2,961	Membrane stress due to M_L
1B	0.416	6,143	5,969	Bending stress due to M_L

TABLE 3B.6-1
COMPARISON OF PILUG COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS

(Sheet 2 of 2)

Figure[a]	B	Stress from Hand Calculation	Stress from Computer Output	Remarks
Stress in Axial Direction (psi)				
4C	0.4447	683	690	Membrane stress due to P
2C	0.4632	773	792	Bending stress due to P
4A	0.294	1,897	1,864	Membrane stress due to M_c
2A	0.550	6,357	5,942	Bending stress due to M_c
4B	0.467	2,365	2,328	Membrane stress due to M_L
2B	0.582	4,989.7	4,842	Bending stress due to M_L
Shear Stress (psi)				
---	---	1,304.8	1,304.8	Shear stress due to M_T
---	---	-366.99	-366.99	Shear stress due to V_L
---	---	127.15	127.16	Shear stress due to V_c

a) Local Stress in Spherical and Cylindrical Shells due to External Loading. Welding Research Council Bulletin, WRC-107, 1965

TABLE 3B.7-1
COMPARISON OF PITRIFE COMPUTER PROGRAM OUTPUT WITH STUDL-II OUTPUT

Test Problem	Size-On-Size	0.707 Size-on-Size
Average Pipe Radius (in.)	3.00	3.00
Average Trunnion Radius (in.)	3.00	2.12
Pipe Wall Thickness (in.)	0.30	0.30
Trunnion Wall Thickness (in.)	0.30	0.21

SIZE-ON-SIZE
MAXIMUM STRESS INTENSITY, psi (α = 30°)

Load	PITRIFE Output	STRUDL-II Output
FX = 10,000 lb	5,763	5,768
FY = 10,000 lb	7,844	7,846
FZ = 10,000 lb	6,507	6,506
MX = 10,000 in.-lb	1,329	1,329
MY = 10,000 in.-lb	1,688	1,687
MY = 10,000 in.-lb	4,068	4,066

0.707 SIZE-ON-SIZE
MAXIMUM STRESS INTENSITY, psi (α = 30°)

Load	PITRIFE Output	STRUDL-II Output
FX = 10,000 lb	13,471	13,458
FY = 10,000 lb	9,616	9,611
FZ = 10,000 lb	20,105	20,030
MX = 10,000 in.-lb	4,371	4,368
MY = 10,000 in.-lb	2,467	2,467
MZ = 10,000 in.-lb	6,178	6,176

TABLE 3B.7-2
COMPARISON OF PITRIFE COMPUTER PROGRAM OUTPUT WITH HAND CALCULATIONS

Test Problem

Average Pipe Radius	= 1.5 in.
Average Trunnion Radius	= 1.35 in.
Trunnion Wall Thickness	= 0.27 in.
Pipe Wall Thickness	= 0.30 in.

LOADS FOR EACH LOAD TYPE COMBINED (DL, OBEI, THER, OCCU, ETC)

FX = FY = FZ = 10,000 lb

MX = MY = MZ = 10,000 in.-lb

MNS = 200 psi

Internal Pressure = 100 psi

STRESS COEFFICIENTS - 0.9 SIZE-ON-SIZE - FX LOADING (α = 30°C)

Stress Type	Coefficient by Hand Calculation	Coefficient From PITRIFE
Longitudinal - Inside Fiber	-1.2652	-1.2652
Circumferential - Inside Fiber	-0.2764	-0.2764
Shear - Inside Fiber	0.2041	0.2041
Longitudinal - Outside Fiber	0.7454	0.7454
Circumferential - Outside Fiber	1.3509	1.3509
Shear - Outside Fiber	0.2041	0.2041

MAXIMUM STRESS INTENSITY - 0.9 SIZE-ON-SIZE (α = 30°C)

	Maximum Stress Intensity, psi	
Load Condition	Hand Calculation	PITRIFE
P + DL + MNS_1	28,181	28,182
P + DL + SRSS (OBEI, OCCU) + MNS_2	73,220	73,220
P + DL + OBEA + THER + MNS_3	88,216	88,216
P + DL + OCCE + MNS_4	59,853	59,853
P + DL + SRSS (SSEI, OCCF) + MNS_5	73,220	73,220

TABLE 3B.8-1
NODAL FORCE COMPARISON

Diameter D = 0.25 ft

Area A = $\pi D^2/_4$ = 0.0490874 ft^2

p = pressure lb/ft^2

= density lb/ft^3

V = velocity ft/sec

g = gravitational constant 32.2 ft/sec^2

p_a = ambient pressure (14.7 x 144 lb/ft^2)

at time t = 0.00650 sec

Node No.	Pressure (psia)	Velocity (fps)	Density (lb/ft^3)	Force	
				STEHAM (lb)	Hand Calculation (lb)
1	42.523	0.0	0.23954	186.57	196.67
5	42.785	5.7843	0.24076	198.43	198.53
10	44.231	31.219	0.24647	209.00	209.11
15	47.003	78.172	0.25737	230.62	230.73
20	50.214	129.89	0.26979	257.84	257.97
25	52.095	159.43	0.27697	274.93	275.06
30	52.209	161.97	0.27742	276.09	276.23
35	52.168	162.21	0.27731	275.83	275.97

TABLE 3B.9-1
INPUT DATA FOR WATHAM

Pipe No.	Total Length (ft)	Inside Diameter (ft)	Friction Factor	No. of Nodes	Nodal Span (ft)	Thickness (in.)	Velocity (fps)
1	2,000	3.0	0.03	7	333.33	0.30824	4.24413
2	3,000	2.5	0.28	9	375	0.44	2.92132
3	2,000	2.0	0.024	6	400	0.50026	4.98473
4	1,800	1.5	0.02	7	300	0.11108	3.59336
5	1,500	1.5	0.022[a]	5	375	0.264	4.52142
6	1,600	1.5	0.025	6	320	0.13796	2.29183
7	2,200	2.5	0.04	8	314.29	0.21534	3.65878
8	1,500	2.0	0.03	6	300	0.14811	3.83245
9	2,000	3.0	0.024	7	333.33	0.30824	4.24413

a) Friction factor in Pipe 5, 0.022, differs slightly from that of hand calculation, 0.020.

Note:

The initial heads of all nodes are calculated by using the Darcy-Weisbach equation.

TABLE 3B.9-2
COMPARISON OF NODAL FORCE CALCULATION AT TIME = 2.34 sec

Pipe No.	Node No.	Force (kip)	
		WATHAM	Hand Calculation
1	1	276.34	276.48
1	2	300.46	300.62
1	3	317.78	317.94
1	4	329.59	329.76
1	5	341.39	341.56
1	6	355.31	355.49
1	7	369.52	369.71

Nodal force calculation is based on the following equation:

$$F = A\left(\rho H + \frac{\rho V^2}{g}\right)$$

where:

F = nodal force, lb

ρ = density, lb/cu ft

H = nodal head, ft

g = 32.2 ft/sec^2

V = nodal velocity, fps

A = pipe area, sq ft

TABLE 3B.10-1
INPUT DATA FOR WATSLUG

Cutoff Mode	Cutoff Frequency	Time Step	Integration Time	Damping Ratio
53	433 Hz	0.0009 sec	0.5 sec	10 percent

Pipe Section	Total Length (ft)	Outside Diameter (in.)	Thickness (in.)	Weight (lb/ft)
1	4.73	8.625	0.906	74.71
2	12.31	6.625	0.864	53.16
3	12.43	6.625	0.28	18.97
4	69.0	12.75	0.688	88.60
5	1.1	12.75	1.5	-----
6	1.0	8.625	0.322	28.55
7	0.83	6.625	0.432	28.57

$E_{hot} = E_{cold}$ = Young's Modulus of pipe = 28.3×10^6 psi

TABLE 3B.10-2
INPUT DATA FOR WATSLUG

Pipe No.	Total Length (ft)	Inside Diameter (ft)	Friction Factor
1	16.125	0.408	0.015
2	12.563	0.5054	0.015
3	63.562	0.948	0.013

Valve Characteristrics

Orifice Area (ft^2)	Opening Time (sec)	Discharge Coefficient	Flow Rate (lbm/sec)
0.0253	0.015	0.805	120.83

Upstream Steam Conditions

Pressure (psia)	Temperature	Density lbm/ft^3	Pressure Rise Rate psi/sec
2690	679°F (1139°R)	8.862	-40*

Downstream Gas Conditions

Pressure (psia)	Temperature	Density lbm/ft^3
15	80°F (540°R)	0.09975

Waterslug Weight = 69.8 lb

Notes:

* Pressure is decreasing after valve opens

TABLE 3B.11-1
ELBOW PROGRAM – ELBOW PROPERTIES USED FOR VERIFICATION PROBLEMS

Outside diameter	30.0 in.
Minimum wall thichness	0.5239 in.
Bend radius	44.214 in.
Pipe radius	14.735 in.
Young's modulus	28.3×10^{6} psi
Poisson's ratio	0.3
Coefficient of thermal expansion	9.11×10^{-6} in./in. °F

TABLE 3B.11-2
ELBOW PROGRAM – CASE 1 RESULTS

Internal Pressure Equals 413.58 psi

	Circumferential Stresses, psi		Longitudinal Stresses, psi		Stress Intensities, psi	
	Inside	Outside	Inside	Outside	Inside	Outside
ELBOW Program	11,676	11,676	5,714	5,714	12,090	11,676
Hand Calculation[(a)]	11,676	11,676	5,714	5,714	12,090	11,676

Note:

a) Hand calculation is based on Article NB-3685.1 of ASME Section III, 1974.

TABLE 3B.11-3
ELBOW PROGRAM – CASE 2 RESULTS

Liner Temperature Gradient Through Wall Equals 100ºF

	Circumferential Stresses, psi		Longitudinal Stresses, psi		Stress Intensities, psi	
	Inside	Outside	Inside	Outside	Inside	Outside
ELBOW Program	18,415	-18,415	18,415	-18,415	18,415	18,415
Hand Calculation[(a)]	18,415	-18,415	18,415	-18,415	18,415	18,415

Note:

a) Hand calculation of Timoshenko and Goodier 1970.

TABLE 3B.11-4
ELBOW PROGRAM – CASE 3 RESULTS

Combined loadings as follows:

1. Internal pressure equals 413.58 psi

2. Linear Temperature gradient equals 100°F

3. Axial Force equal to 60,000 lb

4. Torsional moment equal to 3,500,000

	Circumferential Stresses, psi		Longitudinal Stresses, psi		Stress Intensities, psi	
	Inside	Outside	Inside	Outside	Inside	Outside
ELBOW Program	30,091	-6,739	24,129	-12,701	32,165	14,362
Hand Calculation*	30,091	-6,739	24,129	-12,701	32,165	14,362

TABLE 3B.13-1
COMPARISON OF EIGENVALUES FROM THEORETICAL RESULTS, STRUDL RESULTS, AND GT-STRUDL RESULTS (PROBLEM NO. 1)

"I" Mode No.	Theoretical Results: Frequency cycles/sec	STRUDL Results: Frequency cycles/sec	GT-STRUDL Results: Frequency cycles/sec
1	24.8	24.84	24.84
2	99.2	99.33	99.33
3	223.2	223.37	223.38
4	396.8	396.39	396.40

Note:

For comparison purposes, the results of four modes have been tabulated.

TABLE 3B.13-2
COMPARISON OF EIGENVECTORS FROM STRUDL AND GT-STRUDL
(PROBLEM NO. 1)

(Sheet 1 of 2)

		Y-Displacement	
Mode	Joint	STRUDL	GT-STRUDL
1	1	0.0	0.0
	2	0.309	0.309
	3	0.588	0.588
	4	0.809	0.809
	5	0.951	0.951
	6	1.000	1.000
	7	0.951	0.951
	8	0.809	0.809
	9	0.588	0.588
	10	0.309	0.309
	11	0.0	0.0
2	1	0.000	0.000
	2	0.618	0.618
	3	1.000	1.000
	4	1.000	1.000
	5	0.618	0.618
	6	0.000	0.000
	7	-0.618	-0.618
	8	-1.000	-1.000
	9	-1.000	-1.000
	10	-0.618	-0.618
	11	0.000	0.000

TABLE 3B.13-2
COMPARISON OF EIGENVECTORS FROM STRUDL AND GT-STRUDL
(PROBLEM NO. 1)

(Sheet 2 of 2)

		Y-Displacement	
Mode	Joint	STRUDL	GT-STRUDL
3	1	0.0	0.0
	2	-0.809	-0.809
	3	-0.951	-0.951
	4	-0.309	-0.309
	5	0.588	0.588
	6	1.000	1.000
	7	0.588	0.588
	8	-0.309	-0.309
	9	-0.951	-0.951
	10	-0.809	-0.809
	11	0.0	0.0
4	1	0.0	0.0
	2	1.0	1.0
	3	0.618	0.618
	4	-0.618	-0.618
	5	-1.0	-1.0
	6	0.000	0.000
	7	-1.0	-1.0
	8	0.618	0.618
	9	-0.618	-0.618
	10	-1.0	-1.0
	11	0.0	0.0

Note:

For comparison purposes, the results of four modes have been tabulated.

TABLE 3B.13-3
THE MEMBER FORCES FROM STRUDL AND GT-STRUDL COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS
(PROBLEM NO. 2)

(Sheet 1 of 2)

			STRUDL			GT-STRUDL		
Member	Loading Condition	Joint	Axial	Shear Y	Mom Z	Axial	Shear Y	Mom Z
AD	1	A	1652.25	-237.33	-7958.75	1652.25	-237.33	-7958.75
		D	-1652.25	237.33	-20071.84	-1652.25	237.33	-20071.86
	2	A	-583.75	-675.01	-104030.94	-583.75	-675.01	-104031.4
		D	583.75	675.01	24305.86	583.75	675.01	24305.9
	3	A	-694.20	1480.59	107811.12	-694.20	1480.59	170811.6
		D	694.20	-1480.59	67060.81	694.20	-1480.59	67061.1
DE	1	D	1780.56	1652.25	20071.84	1780.56	1652.24	20071.8
		E	-1780.56	1654.68	-20215.77	-1780.56	1654.69	-20215.9
	2	D	675.01	-583.75	-24305.86	675.01	-583.75	-24305.9
		E	-675.01	583.75	-44640.70	-675.01	583.75	-44640.8
	3	D	-1480.59	-694.20	-67060.81	-1480.59	-694.20	-67061.1
		E	1480.59	694.20	-14930.68	1480.59	694.20	-14930.7

TABLE 3B.13-3
THE MEMBER FORCES FROM STRUDL AND GT-STRUDL COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS
(PROBLEM NO. 2)

(Sheet 2 of 2)

			STRUDL			GT-STRUDL		
Member	Loading Condition	Joint	Axial	Shear Y	Mom Z	Axial	Shear Y	Mom Z
BE	1	B	45.28	171.16	0.00	45.29	171.16	0.00
		E	-45.28	-171.16	20215.80	-45.29	-171.16	20216.93
	2	B	286.70	377.96	-0.01	286.70	377.96	0.00
		E	-286.70	-377.96	44640.70	-286.70	-377.96	44640.84
	3	B	2301.20	126.41	0.00	2301.20	126.41	0.00
		E	-2301.20	-126.41	-14930.65	-2301.20	-126.41	14930.73
EC	1	E	2276.03	0.0	0.0	2276.04	0.0	0.0
		C	-2276.03	0.0	0.0	-2276.04	0.0	0.0
	2	E	420.09	0.0	0.0	420.09	0.0	0.0
		C	-420.09	0.0	0.0	-420.09	0.0	0.0
	3	E	-2272.64	0.0	0.0	-2272.65	0.0	0.0
		C	2272.64	0.0	0.0	2272.65	0.0	0.0

TABLE 3B.13-4
THE JOINT LOADS (AT SUPPORTS) FROM STRUDL AND GT-STRUDL COMPUTER RUNS FOR DIFFERENT LOADING CONDITIONS (PROBLEM NO. 2)

		STRUDL			GT-STRUDL		
Joint	Loading	X Force	Y Force	Z Mom	X Force	Y Force	Z Mom
A	1	237.33	1652.25	-7958.75	237.33	1652.25	-7958.75
	2	675.01	-583.75	-104030.94	675.01	-583.75	-104031.48
	3	-1480.59	-694.20	107811.12	-1480.59	-694.20	107811.62
B	1	-171.16	45.28	0.00	-171.16	45.29	0.00
	2	-377.96	286.70	-0.01	-377.96	286.70	0.00
	3	-126.41	2301.20	0.00	-126.41	2301.20	0.00
C	1	-1609.40	1609.40	0.00	-1609.40	1609.40	0.00
	2	-297.05	297.05	0.00	-297.05	297.05	0.00
	3	1607.00	-1607.00	0.00	1607.00	-1607.00	0.00

TABLE 3B.13-5
COMPARISON OF ELEMENT (RANDOMLY SELECTED) STRESSES (PROBLEM NO. 3)

(Sheet 1 of 3)

		Mxx		Myy		Mxy		Vxx		Vyy	
Element	Node	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1	-12.546	-12.543	-78.334	-78.354	19.545	19.547	13.417	13.412	3.617	3.618
	2	-4.098	-4.109	-18.611	-18.642	19.107	19.109	13.417	13.412	-8.769	-8.756
	10	-13.852	-13.831	-69.887	-69.860	46.293	46.298	-3.549	-3.538	-8.769	-8.756
	9	-8.936	-8.952	-56.769	-56.764	46.730	46.735	-3.549	-3.538	-3.617	3.618
6	6	-2.145	-2.146	-16.671	-16.678	13.396	13.400	-0.595	-0.597	-0.158	-0.158
	7	-2.876	-2.879	-19.390	-19.409	10.630	10.635	-0.595	-0.597	0.916	0.924
	15	-3.956	-3.940	-11.933	-11.916	9.555	9.558	0.694	0.701	0.916	0.924
	14	-6.974	-6.986	-12.939	-12.935	12.320	12.323	0.694	0.701	-0.158	-0.158
10	11	-10.414	-10.417	-4.335	-4.336	65.119	65.122	1.027	1.026	-13.922	-13.924
	12	-15.984	-15.983	7.190	7.180	13.283	13.285	1.027	1.026	-9.953	-9.948
	20	-74.387	-74.369	-3.683	-3.672	22.101	22.102	5.789	5.798	-9.953	-9.948
	19	-99.338	-99.307	-12.307	-12.310	73.937	73.939	5.789	5.798	-13.922	-13.924

TABLE 3B.13-5
COMPARISON OF ELEMENT (RANDOMLY SELECTED) STRESSES (PROBLEM NO. 3)

(Sheet 2 of 3)

		Mxx		Myy		Mxy		Vxx		Vyy	
Element	Node	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
16	18	3.765	3.747	26.738	26.721	94.743	94.475	-15.849	-15.844	0.332	0.335
	19	-56.181	-56.179	-5.239	-5.250	71.624	71.624	-15.849	-15.844	-9.083	-9.078
	27	-117.077	-117.062	-7.560	-7.547	49.705	49.704	-27.147	-27.139	-9.083	-9.078
	26	1.424	1.396	31.393	31.403	72.554	72.555	-27.147	-27.139	0.332	0.335
22	25	-2.301	-2.323	56.058	56.053	74.477	74.481	-3.966	-3.962	-1.428	-1.429
	26	0.446	0.451	33.162	33.148	71.634	71.637	-3.966	-3.962	-2.312	-2.305
	34	11.675	-11.652	29.188	29.207	49.249	49.251	-5.178	-5.163	-2.312	-2.305
	33	-1.561	-1.594	45.381	45.381	52.093	52.094	-5.178	-5.178	-1.428	-1.429
45	51	-149.275	-149.281	-2.222	-2.223	29.027	29.028	6.244	6.243	0.660	0.658
	52	-113.469	-113.472	-1.815	-1.825	26.628	26.629	6.244	6.243	-1.319	-1.314
	60	-124.368	-124.354	-0.097	-0.085	23.243	23.244	3.869	3.877	-1.319	-1.314
	59	-144.766	-144.787	-2.139	-2.142	25.642	25.643	3.869	3.877	0.660	0.658
52	59	-144.756	-144.773	-2.047	-2.055	24.031	24.030	3.874	3.879	2.303	2.305
	60	-124.346	-124.339	0.0146	0.0123	22.659	22.658	3.874	3.879	-2.621	-2.620

TABLE 3B.13-5
COMPARISON OF ELEMENT (RANDOMLY SELECTED) STRESSES (PROBLEM NO. 3)

(Sheet 3 of 3)

		Mxx		Myy		Mxy		Vxx		Vyy	
Element	Node	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
	68	-143.165	-143.158	0.591	0.595	24.111	24.111	-2.034	-2.030	-2.621	-2.620
	67	-129.535	-129.548	-1.240	-1.237	25.483	25.482	-2.034	-2.030	2.303	2.305
60	68	-130.197	-130.198	2.811	2.810	22.296	22.298	22.805	22.805	-4.389	-4.388
	69	-20.373	-20.368	25.257	25.246	11.012	11.013	22.805	22.805	5.429	5.433
	77	-19.106	-19.097	61.775	61.791	34.229	34.230	34.586	34.591	5.429	5.433
	76	-157.545	-157.547	-0.386	-0.386	45.512	45.512	34.586	34.591	-4.389	-4.388
68	77	-24.513	-24.517	54.034	54.028	35.968	35.968	14.908	14.911	0.413	0.414
	78	11.657	11.666	104.330	104.326	39.457	39.457	14.908	14.911	1.380	1.384
	86	21.811	21.826	103.782	103.797	82.856	82.858	16.069	16.074	1.380	1.384
	85	-24.572	-24.578	56.964	56.973	79.367	79.369	16.069	16.074	0.413	0.414
80	91	-36.410	-36.423	2.194	2.186	30.708	30.708	1.185	1.190	7.587	7.593
	92	35.756	-35.747	8.412	8.143	65.545	65.546	1.185	1.190	6.573	6.572
	100	-7.093	-7.089	8.341	8.342	69.355	69.356	0.424	0.424	6.573	6.572
	99	-4.321	-4.324	3.110	3.110	34.518	34.518	0.424	0.424	7.587	7.593

TABLE 3B.13-6
COMPARISON OF RESULTANT (RANDOMLY SELECTED) JOINT DISP. SUPPORTS (GLOBAL) (PROBLEM NO. 3)

	Z Displacement		X Rotation		Y Rotation	
Joint	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	-0.0447149	-0.0447119	0.0004818	0.0004817	0.0001927	0.0001925
5	-0.0484141	-0.0484130	0.0003599	0.0003598	0.0001907	0.0001908
15	-0.0483233	-0.0483236	0.0003430	0.0003429	0.0002060	0.0002060
20	-0.0427901	-0.0427890	0.0003791	0.0003790	0.0002582	0.0002580
30	-0.0435345	-0.0435547	0.0002832	0.0002832	0.0002928	0.0002920
40	-0.0442976	-0.0442990	0.0002317	0.0002317	0.0002164	0.0002163
70	-0.0354678	-0.354688	0.0002588	0.0002586	0.0004367	0.0004369
100	-0.0259728	-0.0259738	0.0001780	0.0001779	0.0008314	0.0008313
112	-0.0426962	-0.0426994	0.0001656	0.0001656	0.0008937	0.0008930

TABLE 3B.13-7
COMPARISON OF MEMBERS (RANDOMLY SELECTED) FORCES AND MOMENTS (PROBLEM NO.4)

		Axial		Shear Y		Shear Z		Torsional Moment		Bending Y Moment		Bending Z Moment	
Member	Joint	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1	0.0	0.0	0.0	0.0	0.527	0.527	0.0	0.0	0.0	0.0	0.0	0.0.
	2	0.0	0.0	0.0	0.0	-0.371	-0.371	0.0	0.0	-21.567	-21.568	0.0	0.0
4	4	0.0	0.0	0.0	0.0	-0.713	-0.713	0.0	0.0	5.158	5.158	0.0	0.0
	5	0.0	0.0	0.0	0.0	0.869	0.869	0.0	0.0	32.818	32.818	0.0	0.0
10	10	0.0	0.0	0.0	0.0	0.226	0.226	0.0	0.0	5.627	5.627	0.0	0.0
	11	0.0	0.0	0.0	0.0	-0.070	-0.070	0.0	0.0	-12.738	-12.738	0.0	0.0
20	3	0.0	0.0	0.0	0.0	0.158	0.158	0.0	0.0	0.0	0.0	0.0	0.0
	23	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-5.670	-5.670	0.0	0.0
25	14	0.0	0.0	0.0	0.0	0.117	0.117	0.0	0.0	1.818	1.818	0.0	0.0
	17	0.0	0.0	0.0	0.0	-0.093	-0.093	0.0	0.0	-4.338	-4.338	0.0	0.0
30	36	0.0	0.0	0.0	0.0	-0.093	-0.093	0.0	0.0	4.338	4.338	0.0	0.0
	39	0.0	0.0	0.0	0.0	0.117	0.117	0.0	0.0	-1.818	-1.818	0.0	0.0
50	12	0.0	0.0	0.0	0.0	0.426	0.426	0.0	0.0	0.0	0.0	0.0	0.0
	54	0.0	0.0	0.0	0.0	0.000	0.000	0.0	0.0	-15.336	-15.336	0.0	0.0
60	47	0.0	0.0	0.0	0.0	-0.069	-0.069	0.0	0.0	12.621	12.621	0.0	0.0
	48	0.0	0.0	0.0	0.0	0.225	0.225	0.0	0.0	-5.569	-5.569	0.0	0.0

TABLE 3B.13-8
COMPARISON OF RESULTANT JOINT LOADS - SUPPORTS (PROBLEM NO. 4) (GLOBAL)

	X Force		Y Force		Z Force		X Mom		Y Mom		Z Mom	
Joint	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	0.0	0.0	0.0	0.0	0.953	0.95	30.0	0.0	0.0	0.0	0.0	0.0
5	0.0	0.0	0.0	0.0	1.885	1.885	0.0	0.0	0.0	0.0	0.0	0.0
9	0.0	0.0	0.0	0.0	1.536	1.536	0.0	0.0	0.0	0.0	0.0	0.0
12	0.0	0.0	0.0	0.0	0.769	0.769	0.0	0.0	0.0	0.0	0.0	0.0
41	0.0	0.0	0.0	0.0	0.953	0.953	0.0	0.0	0.0	0.0	0.0	0.0
45	0.0	0.0	0.0	0.0	1.885	1.885	0.0	0.0	0.0	0.0	0.0	0.0
49	0.0	0.0	0.0	0.0	1.536	1.536	0.0	0.0	0.0	0.0	0.0	0.0
52	0.0	0.0	0.0	0.0	0.769	0.769	0.0	0.0	0.0	0.0	0.0	0.0

TABLE 3B.13-9
COMPARISON OF EIGENVALUES, FREQUENCIES, AND PERIODS (PROBLEM NO. 5)

(Sheet 1 of 2)

	Eigenvalue		Frequency (Cycles/Time Unit)		Period (Time Unit/Cycle)	
MODE	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1.791123D 03	1.791171D+03	6.735703D 00	6.735791D+00	1.484626D-01	1.484607D-01
2	1.892565D 03	1.892596D+03	6.923816D 00	6.923873D+00	1.444290D-01	1.444278D-01
3	1.925683D 03	1.925715D+03	6.934134D 00	6.984191D+00	1.431817D-01	1.431805D-01
4	1.935117D 03	1.938148D+03	7.006645D 00	7.006702D+00	1.427217D-01	1.427205D-01
5	1.947453D 03	1.947485D+03	7.023501D 00	7.023558D+00	1.423791D-01	1.423780D-01
6	1.949031D 03	1.949063D+03	7.026403D 00	7.026403D+00	1.423215D-01	1.423203D-01
7	1.949454D 03	1.949486D+03	7.027108D 00	7.027166D+00	1.423060D-01	1.423049D-01
8	1.949909D 03	1.949940D+03	7.027927D 00	7.027984D+00	1.422895D-01	1.422883D-01
9	2.894955D 03	2.895024D+03	8.563297D 00	8.563400D+00	1.167775D-01	1.167760D-01
10	2.935117D 03	2.935188D+03	8.622493D 00	8.622597D+00	1.159757D-01	1.159743D-01
11	2.935117D 03	2.935188D+03	8.622493S 00	8.622597D+00	1.159757D-01	1.159743D-01
12	3.566743D 03	3.566836D+03	9.505086D 00	9.505210D+00	1.052068D-01	1.052055D-01
13	3.941438D 03	3.941542D+03	9.991886D 00	9.992018D+00	1.000812D-01	1.000799D-01
14	8.732367D 03	8.732600D+03	1.487257D 01	1.487277D+00	6.723786D-02	6.723696D-92
15	1.345767D 04	1.345793D+04	1.846330D 01	1.846330D+01	5.416204D-02	5.416151D-02

TABLE 3B.13-9
COMPARISON OF EIGENVALUES, FREQUENCIES, AND PERIODS (PROBLEM NO. 5)

(Sheet 2 of 2)

	Eigenvalue		Frequency (Cycles/Time Unit)		Period (Time Unit/Cycle)	
MODE	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
16	1.346419D 04	1.346445D+04	1.846759D 01	1.846777D+01	5.414892D-02	5.414839D-02
17	1.346597D 04	1.346623D+04	1.846881D 01	1.846899D+01	5.414534D-02	5.414481D-02
18	2.573237D 04	2.573314D+04	2.553054D 01	2.553092D+01	3.916877D-02	3.916819D-02
19	2.650393D 04	2.650463D+04	2.591047D 01	2.591081D+01	3.859444D-02	3.859393D-02
20	2.745771D 04	2.745854D+04	2.637256D 01	2.637296D+01	3.791820D-02	3.791763D-02
21	3.177105D 04	3.177167D+04	2.836847D 01	2.836875D+01	3.525040D-02	3.525006D-02
22	3.248699D 04	3.248762D+04	2.868632D 01	2.868660D+01	3.485952D-02	3.485948D-02
23	3.525845D 04	3.525916D+04	2.988489D 01	2.988520D+01	3.346172D-02	3.346138D-02
24	3.953962D 04	3.954066D+04	3.164728D 01	3.164769D+01	3.159829D-02	3.159788D-02
25	5.170876D 04	5.171001D+04	3.619113D 01	3.619157D+01	2.763108D-02	2.763074D-02
26	5.592949D 04	5.593058D+04	3.763922D 01	3.763958D+01	2.656804D-02	2.656778D-02
27	5.592949D 04	5.593058D+04	3.763922D 01	3.763958D+01	2.656804D-02	2.656778D-02
28	5.959983D 04	5.960165D+04	3.885462D 01	3.885521D+01	2.573697D-02	2.573657D-02
29	6.538255D 04	6.538428D+04	4.069594D 01	4.069648D+01	2.457248D-02	2.457215D-02
30	8.206529D 04	8.206758D+04	4.559318D 01	4.559382D+01	2.193311D-02	2.193280D-02

TABLE 3B.13-10
COMPARISON OF EIGENVECTORS FOR FEW RANDOMLY SELECTED MODES AND JOINTS (PROBLEM NO. 5)

Eigenvectors (Global)

		X-Displacement		Z-Displacement	
Mode	Joint	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
1	1	0.0	0.0	0.0	0.0
	2	0.721	-0.721	0.000	0.000
	13	0.721	-0.721	0.000	0.000
	23	1.00	-1.00	0.000	0.000
	43	0.999	-0.999	0.000	0.000
6	1	0.0	0.0	0.0	0.0
	2	0.000	0.000	-0.0009	-0.0009
	13	0.000	0.000	-0.0014	-0.0014
	23	0.000	0.000	-0.567	-0.567
	43	0.000	0.000	-0.0005	-0.0005
3	1	0.0	0.0	0.0	0.0
	2	0.000	0.000	0.0095	0.0095
	13	0.000	0.000	0.0153	0.0153
	23	0.000	0.000	1.000	1.000
	43	0.000	0.000	0.013	0.013

Y-displacements are approximately 0.000 for all three modes.

TABLE 3B.13-11
COMPARISON OF JOINT DISPLACEMENTS AT THE FREE JOINTS FOR RANDOMLY SELECTED JOINTS (GLOBAL)
(PROBLEM NO. 5)

Joint	Response Type	X Displacement		Y Displacement		Z Displacement	
		STRUDL	GT-STRUDL	STRUDL	GT-STRUDL	STRUDL	GT-STRUDL
7	RMS	0.000	0.000	0.000	0.000	0.00099	0.00099
	ABS SUM	0.000	0.000	0.000	0.000	0.00213	0.00212
	CMS	0.000	0.000	0.000	0.000	0.00117	0.00116
15	RMS	0.000	0.000	0.000	0.000	0.0105	0.0104
	ABS SUM	0.000	0.000	0.000	0.000	0.0105	0.0104
	CSM	0.000	0.000	0.000	0.000	0.0105	0.0104
22	RMS	0.000	0.000	0.000	0.000	0.0424	0.0424
	ABS SUM	0.000	0.000	0.000	0.000	0.0449	0.0449
	CSM	0.000	0.000	0.000	0.000	0.0424	0.0424
36	RMS	0.000	0.000	0.000	0.000	0.0287	0.0288
	ABS SUM	0.000	0.000	0.000	0.000	0.0287	0.02920
	CSM	0.000	0.000	0.000	0.000	0.0287	0.02920

RMS = Root Mean Square

ABS SUM = Absolute Sum

CSM = Closely Spaced Mode

TABLE 3B.14-1
COMPARISON OF SANDUL LOAD COMBINATIONS WITH HAND COMPUTATION

Load Condition 3

Load	Hand	SANDUL
FX	1424	1424
	-2102	-2102
FY	0	0
	-834	-834
FZ	978	978
	-652	-652
MX	2758	2758
	0	0
MY	3830	3830
	-5542	-5542
MZ	216	216
	-114	-114

TABLE 3B.14-2
COMPARISON OF MEMBER FORCES BETWEEN SANDUL AND STRUDL-SW

Load Condition 3
Loading Combination 49
Member 4, Joint 5

Member Force	STRUDL-SW	SANDUL
Axial (lb)	600	600
Shear Y-direction (lb)	298	298
Shear Z-direction (lb)	535	535
Torsion (in.-lb)	-3707	-3707
Bending about Y axis (in.-lb)	5872	5872
Bending about Z axis (in.-lb)	-2025	-2025

TABLE 3B.14-3
COMPARISON OF STRESSES AND WELD SIZE BETWEEN SANDUL AND HAND COMPUTATION

Same case as Table 3B.14-2

Item	Hand	SANDUL
Axial Stress (psi)	34	34
Normal Stress (psi)	483	483
Allowable Normal Stress (psi)	27,720	27,620
Shear Stress (psi)	147	147
Allowable Shear Stress (psi)	22,260	22,399
Fillet Weld Size (in.)	0.013	0.013

TABLE 3B.17-1
APE PROGRAM VERIFICATION PROBLEM COMPARISON OF DRILLED-IN ANCHOR ROAD

(Sheet 1 of 3)

			BIP (ST 361)		APE (ST 378, V01L00)	
Attachment	Attachment Location	Loading	Anchor Shear	Anchor Tension	Anchor Shear	Anchor Tension
TS 4 x 4	Corner of Surface-Mounted Plate	F_z = 3,000 lb	0	2,269.36	0	2,296.87
TS 4 x 4	Corner of Surface-Mounted Plate	M_x = 12,000 in.-lb	0	1,256.61	0	2,625.00
TS 4 x 4	Corner of Surface-Mounted Plate	M_x = 8,000 in.-lb M_y = 8,000 in.-lb	0	1,204.13	0	3,500.00
L 4 x 4	Corner of Surface-Mounted Plate	F_z = 3,000 lb	0	2,116.39	0	2,268.52
L 4 x 4	Corner of Surface-Mounted Plate	M_x = 12,000 in.-lb	0	1,418.15	0	2,333.33
L 4 x 4	Corner of Surface-Mounted Plate	M_x = 8,000 in.-lb M_y = 8,000 in.-lb	0	1,506.74	0	3,111.11
TS 4 x 4	Center of Embedded Plate	F_z = 20,000 lb	0	12,742.60	0	12,752.10
TS 4 x 4	Center of Embedded Plate	M_x = 50,000 in.-lb	0	4,285.70	0	5,934.86

TABLE 3B.17-1
APE PROGRAM VERIFICATION PROBLEM COMPARISON OF DRILLED-IN ANCHOR ROAD

(Sheet 2 of 3)

			BIP (ST 361)		APE (ST 378, V01L00)	
Attachment	Attachment Location	Loading	Anchor Shear	Anchor Tension	Anchor Shear	Anchor Tension
TS 4 x 4	Center of Embedded Plate	M_x = 25,000 in.-lb M_y = 25,000 in.-lb	0	3,756.76	0	5,934.86
TS 4 x 4	Corner of Embedded Plate	F_z = 20,000 lb	0	17,591.00	0	17,607.22
TS 4 x 4	Corner of Embedded Plate	M_x = 50,000 lb	0	9,735.84	0	10,937.50
TS 4 x 4	Corner of Embedded Plate	M_x = 25,000 in.-lb M_y = 25,000 in.-lb	0	5,575.90	0	10,937.50
L 4 x 4	Center of Embedded Plate	F_z = 20,000 lb	0	14,585.30	0	15,940.12
L 4 x 4	Center of Embedded Plate	M_x = 50,000 in.-lb	0	4,959.99	0	5,934.86
L 4 x 4	Center of Embedded Plate	M_x = 25,000 in.-lb M_y = 25,000 in.-lb	0	4,344.17	0	5,934.86
L 4 x 4	Corner of Embedded Plate	F_z = 20,000 lb	0	18,596.50	0	22,009.02

TABLE 3B.17-1
APE PROGRAM VERIFICATION PROBLEM COMPARISON OF DRILLED-IN ANCHOR ROAD

(Sheet 3 of 3)

			BIP (ST 361)		APE (ST 378, V01L00)	
Attachment	Attachment Location	Loading	Anchor Shear	Anchor Tension	Anchor Shear	Anchor Tension
L 4 x 4	Corner of Embedded Plate	M_x = 50,000 in.-lb	0	10,559.60	0	10,937.50
L 4 x 4	Corner of Embedded Plate	M_x = 25,000 in.-lb M_y = 25,000 in.-lb	0	8,648.15	0	10,937.50
C 4 x 5.4	Corner of Embedded Plate	$F_x = F_y$ = 2,800 lb F_z = 1,970 lb M_x = 30,430 in.-lb M_y = 27,390 in.-lb M_z = 0	1,763.35	11,067	1,763.35	32,351.37
TS 4 x 4	Along Edge of Surface-Mounted Plate	F_z = 3,000 lb	0	1,237.02	0	1,458.33
TS 4 x 4	Along Edge of Surface-Mounted Plate	M_x = 12,000 in.-lb	0	786.89	0	1,250.00
TS 4 x 4	Along Edge of Surface-Mounted Plate	M_x = 8,000 in.-lb M_y = 8,000 in.-lb	0	856.54	0	1,833.33

4.0 REACTOR

TABLE OF CONTENTS

Section	Title	Page
4.1	SUMMARY DESCRIPTION	4.1-1
4.2	FUEL SYSTEM DESIGN	4.2-1
4.2.1	DESIGN BASES	4.2-2
4.2.1.1	Cladding	4.2-2
4.2.1.2	Fuel Material	4.2-3
4.2.1.3	Fuel Rod Performance	4.2-3
4.2.1.4	Spacer Grids	4.2-4
4.2.1.5	Fuel Assembly	4.2-5
4.2.1.6	Incore Control Components	4.2-6
4.2.1.7	Surveillance Program	4.2-8
4.2.2	DESIGN DESCRIPTION	4.2-8
4.2.2.1	Fuel Rods	4.2-9
4.2.2.2	Fuel Assembly Structure	4.2-11
4.2.2.2.1	Bottom Nozzle	4.2-11
4.2.2.2.2	Top Nozzle	4.2-12
4.2.2.2.3	Guide and Instrument Thimbles	4.2-12
4.2.2.2.4	Grid Assemblies	4.2-12
4.2.2.3	Incore Control Components	4.2-14
4.2.2.3.1	Full Length Rod Cluster Control Assembly	4.2-15
4.2.2.3.2	Burnable Poison Assembly	4.2-16
4.2.2.3.3	Neutron Source Assembly	4.2-16
4.2.2.3.4	Thimble Plug Assembly	4.2-17
4.2.3	DESIGN EVALUATION	4.2-18
4.2.3.1	Cladding	4.2-18
4.2.3.2	Fuel Materials Consideration	4.2-20
4.2.3.3	Fuel Rod Performance	4.2-21
4.2.3.4	Spacer Grids	4.2-24
4.2.3.5	Fuel Assembly	4.2-25
4.2.3.5.1	Stresses and Deflections	4.2-25
4.2.3.5.2	Dimensional Stability	4.2-25
4.2.3.6	Incore Control Components	4.2-26
4.2.4	TESTING AND INSPECTION PLAN	4.2-31
4.2.4.1	Quality Assurance Program	4.2-31
4.2.4.2	Quality Control	4.2-31
4.2.4.3	Incore Control Component Testing and Inspection	4.2-33
4.2.4.4	Tests and Inspections by Others	4.2-34
4.2.4.5	In-Service Surveillance	4.2-34
4.2.4.6	Onsite Inspection	4.2-34
4.2.4.7	Post-Irradiation Surveillance	4.2-35
	REFERENCES	4.2-35

TABLE OF CONTENTS (Continued)

Section	Title	Page
4.3	NUCLEAR DESIGN	4.3-1
4.3.1	DESIGN BASES	4.3-1
4.3.1.1	Fuel Burnup	4.3-2
4.3.1.2	Negative Reactivity Feedbacks (Reactivity Coefficient)	4.3-2
4.3.1.3	Control of Power Distribution	4.3-3
4.3.1.4	Maximum Controlled Reactivity Insertion Rate	4.3-3
4.3.1.5	Shutdown Margins	4.3-4
4.3.1.6	Stability	4.3-5
4.3.1.7	Anticipated Transients Without Trip	4.3-6
4.3.2	DESCRIPTION	4.3-6
4.3.2.1	Nuclear Design Description	4.3-6
4.3.2.2	Power Distributions	4.3-7
4.3.2.2.1	Definitions	4.3-7
4.3.2.2.2	Radial Power Distributions	4.3-9
4.3.2.2.3	Assembly Power Distributions	4.3-10
4.3.2.2.4	Axial Power Distributions	4.3-10
4.3.2.2.5	Local Power Peaking	4.3-11
4.3.2.2.6	Limiting Power Distributions	4.3-11
4.3.2.2.7	Experimental Verification of Power Distribution Analysis	4.3-14
4.3.2.2.8	Testing and Operations Support	4.3-15
4.3.2.2.9	Monitoring Instrumentation	4.3-15
4.3.2.3	Reactivity Coefficients	4.3-16
4.3.2.3.1	Fuel Temperature (Doppler) Coefficient	4.3-16
4.3.2.3.2	Moderator Coefficients	4.3-17
4.3.2.3.3	Power Coefficient	4.3-18
4.3.2.3.4	Comparison of Calculated and Experimental Reactivity Coefficients	4.3-18
4.3.2.3.5	Reactivity Coefficients Used in Transient Analysis	4.3-19
4.3.2.4	Control Requirements	4.3-19
4.3.2.4.1	Doppler	4.3-20
4.3.2.4.2	Variable Average Moderator Temperature	4.3-20
4.3.2.4.3	Redistribution	4.3-20
4.3.2.4.4	Void Content	4.3-20
4.3.2.4.5	Rod Insertion Allowance	4.3-20
4.3.2.4.6	Burnup	4.3-20
4.3.2.4.7	Xenon and Samarium Poisoning	4.3-21
4.3.2.4.8	pH Effects	4.3-21
4.3.2.4.9	Experimental Confirmation	4.3-21
4.3.2.4.10	Control	4.3-21
4.3.2.4.11	Chemical Poison	4.3-21
4.3.2.4.12	Rod Cluster Control Assemblies	4.3-21
4.3.2.4.13	Reactor Coolant Temperature	4.3-22
4.3.2.4.14	Burnable Absorber Rods	4.3-23
4.3.2.4.15	Peak Xenon Startup	4.3-23
4.3.2.4.16	Load Follow Control and Xenon Control	4.3-23

TABLE OF CONTENTS (Continued)

Section	Title	Page
4.3.2.4.17	Burnup	4.3-23
4.3.2.5	Control Rod Patterns and Reactivity Worth	4.3-23
4.3.2.6	Criticality of the Reactor During Refueling and Criticality of Fuel Assemblies	4.3-25
4.3.2.7	Stability	4.3-27
4.3.2.7.1	Introduction	4.3-27
4.3.2.7.2	Stability Index	4.3-27
4.3.2.7.3	Prediction of the Core Stability	4.3-28
4.3.2.7.4	Stability Measurements	4.3-28
4.3.2.7.5	Comparison of Calculations with Measurements	4.3-29
4.3.2.7.6	Stability Control and Protection	4.3-30
4.3.2.8	Vessel Irradiation	4.3-31
4.3.3	ANALYTICAL METHODS	4.3-31
	REFERENCES	4.3-32
4.4	THERMAL AND HYDRAULIC DESIGN	4.4-1
4.4.1	DESIGN BASES	4.4-1
4.4.1.1	Departure from Nucleate Boiling Design Basis	4.4-1
4.4.1.2	Fuel Temperature Design Basis	4.4-2
4.4.1.3	Core Flow Design Basis	4.4-2
4.4.1.4	Hydrodynamic Stability Design Basis	4.4-3
4.4.1.5	Other Considerations	4.4-3
4.4.2	DESCRIPTION OF THERMAL AND HYDRAULIC DESIGN OF THE REACTOR CORE	4.4-3
4.4.2.1	Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology	4.4-3
4.4.2.1.1	DNB Technology	4.4-3
4.4.2.1.2	Definition of DNBR	
4.4.2.1.3	Mixing Technology	
4.4.2.1.4	Hot Channel Factors	4.4-3
4.4.2.1.5	Effects of Rod Bow on DNBR	4.4-5
4.4.2.2	Linear Heat Generation Rate	
4.4.2.3	Void Fraction Distribution	
4.4.2.4	Normalized Core Flow and Enthalpy Rise Distributions	
4.4.2.5	Core Pressure Drops and Hydraulic Loads	4.4-5
4.4.2.5.1	Core Pressure Drops	4.4-5
4.4.2.5.2	Hydraulic Loads	4.4-5
4.4.2.6	Correlation and Physical Data	
4.4.2.6.1	Surface Heat Transfer Coefficients	
4.4.2.6.2	Total Core and Vessel Pressure Drop	
4.4.2.6.3	Void Fraction Correlation	
4.4.2.7	Thermal Effects of Operation Transients	4.4-6
4.4.2.8	Uncertainties in Estimates	4.4-6
4.4.2.8.1	Uncertainties in Fuel and Clad Temperatures	4.4-6

TABLE OF CONTENTS (Continued)

Section	Title	Page
4.4.2.8.2	Uncertainties in Pressure Drops	4.4-6
4.4.2.8.3	Uncertainties Due to Inlet Flow Maldistribution	4.4-7
4.4.2.8.4	Uncertainty in DNB Correlations	4.4-7
4.4.2.8.5	Uncertainties in DNBR Calculations	4.4-7
4.4.2.8.6	Uncertainties in Flow Rates	4.4-7
4.4.2.8.7	Uncertainties in Hydraulic Loads	4.4-7
4.4.2.9	Flux Tilt Considerations	4.4-7
4.4.2.10	Fuel and Cladding Temperatures	4.4-8
4.4.2.9.1	UO_2 Thermal Conductivity	
4.4.2.9.2	Radial Power Distribution in UO_2 Fuel Rods	
4.4.2.9.3	Gap Conductance	
4.4.2.9.4	Surface Heat Transfer Coefficients	
4.4.2.10.5	Fuel Clad Temperatures	4.4-9
4.4.2.10.6	Treatment of Peaking Factors	4.4-9
4.4.3	DESCRIPTION OF THE THERMAL AND HYDRAULIC DESIGN OF THE REACTOR COOLANT SYSTEM	4.4-9
4.4.3.1	Plant Configuration Data	4.4-9
4.4.3.2	Operating Restrictions on Pumps	4.4-9
4.4.3.3	Power-Flow Operating Map (BWR)	4.4-10
4.4.3.4	Temperature-Power Operating Map	4.4-10
4.4.3.5	Load Following Characteristics	4.4-10
4.4.3.6	Thermal and Hydraulic Characteristics Summary Table	
4.4.4	EVALUATION	4.4-10
4.4.4.1	Critical Heat Flux	4.4-10
4.4.4.2	Core Hydraulics	4.4-10
4.4.4.2.1	Flow Paths Considered in Core Pressure Drop and Thermal Design	4.4-10
4.4.4.2.2	Inlet Flow Distributions	4.4-11
4.4.4.2.3	Empirical Friction Factor Correlations	
4.4.4.3	Influence of Power Distribution	4.4-11
4.4.4.3.1	Nuclear Enthalpy Rise Hot Channel Factor, F^N_{DH}	4.4-11
4.4.4.3.2	Axial Heat Flux Distributions	4.4-12
4.4.4.4	Core Thermal Response	4.4-12
4.4.4.5	Analytical Techniques	4.4-12
4.4.4.5.1	Core Analysis	
4.4.4.5.2	Steady State Analysis	
4.4.4.5.3	Experimental Verification	
4.4.4.5.4	Transient Analysis	
4.4.4.6	Hydrodynamic and Flow Power Coupled Instability	4.4-12
4.4.4.7	Fuel Rod Behavior Effects from Coolant Flow Blockage	4.4-14
4.4.5	TESTING AND VERIFICATION	4.4-15
4.4.5.1	Tests Prior to Initial Criticality	4.4-15
4.4.5.2	Initial Power and Plant Operation	4.4-15
4.4.5.3	Component and Fuel Inspections	4.4-15
4.4.6	INSTRUMENTATION REQUIREMENTS	4.4-15
4.4.6.1	Incore Instrumentation	4.4-15

TABLE OF CONTENTS (Continued)

Section	Title	Page
4.4.6.2	Overtemperature and Overpower N-16 Instrumentation	4.4-16
4.4.6.3	Instrumentation to Limit Maximum Power Output	4.4-16
4.4.6.4	Loose Parts Monitoring System	4.4-16
	REFERENCES	4.4-17
4.5	REACTOR MATERIALS	4.5-1
4.5.1	CONTROL ROD SYSTEM STRUCTURAL MATERIALS	4.5-1
4.5.1.1	Materials Specifications	4.5-1
4.5.1.2	Fabrication and Processing of Austenitic Stainless Steel Components	4.5-2
4.5.1.3	Contamination Protection and Cleaning of Austenitic Stainless Steel	4.5-2
4.5.2	REACTOR INTERNALS MATERIALS	4.5-2
4.5.2.1	Materials Specifications	4.5-2
4.5.2.2	Controls on Welding	4.5-2
4.5.2.3	Nondestructive Examination of Wrought Seamless Tubular Products and Fittings	4.5-2
4.5.2.4	Fabrication and Processing of Austenitic Stainless Steel Components	4.5-2
4.5.2.5	Contamination Protection and Cleaning of Austenitic Stainless Steel	4.5-3
4.6	FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS	4.6-1
4.6.1	INFORMATION FOR CONTROL ROD DRIVE SYSTEM (CRDS)	4.6-1
4.6.2	EVALUATION OF THE CRDS	4.6-1
4.6.3	TESTING AND VERIFICATION OF THE CRDS	4.6-1
4.6.4	INFORMATION FOR COMBINED PERFORMANCE OF REACTIVITY SYSTEMS	4.6-1
4.6.5	EVALUATION OF COMBINED PERFORMANCE	4.6-2
4.6.6	STARTUP REPORTS	4.6-2
	REFERENCES	4.6-3

LIST OF TABLES

Number	Title
4.1-1	REACTOR DESIGN COMPARISON TABLE
4.1-2	ANALYTICAL TECHNIQUES IN CORE DESIGN
4.1-3	DESIGN LOADING CONDITIONS FOR REACTOR CORE COMPONENTS
4.3-1	DELETED
4.3-2A	DELETED
4.3-2B	DELETED
4.3-3	TYPICAL REACTIVITY PARAMETERS FOR ROD CLUSTER CONTROL ASSEMBLIES
4.3-4	DELETED
4.3-5	AXIAL STABILITY INDEX PRESSURIZED WATER REACTOR CORE WITH A 12 FOOT HEIGHT
4.3-6	TYPICAL NEUTRON FLUX LEVELS (n/cm^2-sec) AT FULL POWER
4.3-7	DELETED
4.3-8	DELETED
4.3-9	DELETED
4.3-10	DELETED
4.3-11	DELETED
4.4-1	THERMAL AND HYDRAULIC COMPARISON TABLE
4.4-1B	DELETED
4.4-2	DELETED
4.4-3	DELETED
4.4-4	DELETED

LIST OF FIGURES

Number	Title
4.2-1	Typical Fuel Assembly Cross Section 17 x 17
4.2-2	Typical Fuel Assembly Outline 17 x 17
4.2-3	Typical Fuel Rod Schematic
4.2-4 thru 4.2-7	Deleted
4.2-8	Full Length Rod Cluster Control and Drive Rod Assembly With Interfacing Components
4.2-9	Typical Full Length Rod Cluster Control Assembly Outline
4.2-10	Full Length Absorber Rod - All Ag-In-Cd Design
4.2-11	Deleted
4.2-12	Typical Burnable Poison Assembly
4.2-13	Deleted
4.2-14	Deleted
4.2-15	Typical Secondary Source Assembly
4.2-16	Typical Thimble Plug Assembly
4.3-1	Deleted
4.3-2	Typical Production and Consumption of Higher Isotopes
4.3-3	Typical Boron Concentration Versus First Cycle Burnup With And Without Burnable Absorber Rods
4.3-4	Typical Burnable Poison Rod Arrangement Within An Assembly
4.3-5	Typical Burnable Absorber Loading Pattern
4.3-6	Typical Normalized Power Density Distribution Near Beginning of Life, Unrodded Core, Hot Full Power, No Xenon

LIST OF FIGURES (Continued)

Number	Title
4.3-7	Typical Normalized Power Density Distribution Near Beginning of Life, Unrodded Core, Hot Full Power, Equilibrium Xenon
4.3-8	Typical Normalized Power Density Distribution Near Beginning of Life, Group D 35%, Hot Full Power, Equilibrium Xenon
4.3-9	Typical Normalized Power Density Distribution Near Middle of Life, Unrodded Core, Hot Full Power, Equilibrium Xenon
4.3-10	Typical Normalized Power Density Distribution Near End of Life, Unrodded Core, Hot Full Power, Equilibrium Xenon
4.3-11	Typical Normalized Power Density Distribution Near End of Life, Group D 35%, Hot Full Power, Equilibrium Xenon
4.3-12	Typical Rodwise Power Distribution in a Typical Assembly Near Beginning of Life, Hot Full Power, Equilibrium Xenon, Unrodded Core
4.3-13	Typical Rodwise Power Distribution in a Typical Assembly Near End of Life, Hot Full Power, Equilibrium Xenon, Unrodded Core
4.3-14	Typical Axial Power Shapes Occurring At Beginning of Life
4.3-15	Typical Axial Power Shapes Occurring at Middle of Life
4.3-16	Typical Axial Power Shapes Occurring at End of Life
4.3-17	Comparison of Assembly Axial Power Distribution with Core Average Axial Distribution, D Bank Slightly Inserted
4.3-18 thru 4.3-23	Deleted
4.3-24	Typical Comparison Between Calculated and Measured Relative Fuel Assembly Power Distribution
4.3-25	Comparison of Calculated and Measured Axial Shape
4.3-26	Measured Values of FQ for Full Power Rod Configurations
4.3-27	Doppler Temperature Coefficient at BOL and EOL

LIST OF FIGURES (Continued)

Number	Title
4.3-28	Typical Doppler - Only Power Coefficient - BOL, EOL
4.3-29	Typical Doppler - Only Power Defect - BOL, EOL
4.3-30	Typical Moderator Temperature Coefficient - BOL, No Rods
4.3-31	Typical Moderator Temperature Coefficient - EOL
4.3-32	Typical Moderator Temperature Coefficient as a Function of Boron Concentration - BOL, No Rods
4.3-33	Typical Hot Full Power Temperature Coefficient for the Critical Boron Concentration
4.3-34	Typical Total Power Coefficient - BOL, EOL
4.3-35	Typical Total Power Defect BOL, EOL
4.3-36	Rod Cluster Control Assembly Pattern
4.3-37	Typical Accidental Simultaneous Withdrawal of Two Control Banks EOL, HZP Banks C and B Moving in the Same Plane
4.3-38	Typical Trip Curve
4.3-39	Typical Normalized Rod Worth versus Percent Insertion all Rods But One
4.3-40	Axial Offset Vs. Time PWR Core with a 12-Ft Height and 121 Assemblies
4.3-41	XY Xenon Test Thermocouple Response Quadrant Tilt Difference Vs. Time
4.3.42	Deleted
thru	
4.3-45	
4.4-1	Deleted
4.4-2	Normalized Radial Flow and Enthalpy Distribution at 4 - Ft Elevation Typical of Cycle 1
4.4-3	Normalized Radial Flow and Enthalpy Distribution at 8 - Ft Elevation Typical of Cycle 1

LIST OF FIGURES (Continued)

Number	Title
4.4-4	Normalized Radial Flow and Enthalpy Distribution at 12 Ft Elevation Typical of Cycle 1
4.4-5 thru 4.4-19	Deleted
4.4-20	Distribution of Incore Instrumentation
4.4-21	Reactor Coolant System Temperature Percent Power Map

4.1 SUMMARY DESCRIPTION

This chapter describes: 1) the mechanical components of the reactor and reactor core including the fuel rods and fuel assemblies, 2) the nuclear design, and 3) the thermal-hydraulic design. The initial reactor core is composed of an array of fuel assemblies that are identical in mechanical design but different in fuel enrichment. Within each fuel assembly all rods are of the same enrichment. Three different enrichments are employed in the first core: 2.10 (region 1), 2.60 (region 2), and 3.10 (region 3) weight percents. It was required that the initial core loading maximum enrichment not exceed 3.2 weight percent U235. For subsequent reloads, the target maximum enrichment is up to 5.0 weight percent. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment not to exceed 5.0 weight percent U235.

A fuel assembly is composed of 264 fuel rods in a 17x17 square array. The center position in the fuel assembly is reserved for incore instrumentation. The remaining 24 positions in the fuel assembly have guide thimbles which are joined to the top and bottom nozzles of the fuel assembly and serve to support the fuel grids. A fuel assembly may have limited substitution of zirconium alloy or stainless steel filler rods in place of fuel rods, in accordance with NRC approved applications of fuel rod configurations. The fuel grids consist of an egg crate arrangement of interlocked straps that maintain lateral spacing between the rods. The grid straps have spring fingers and dimples which grip and support the fuel rods. The middle grids also have coolant-mixing vanes. The flow mixer grid straps contain only support dimples and coolant mixing vanes.

The reactor core is comprised of an array of fuel assemblies which have different fuel enrichments. Fuel cycle times of 6 months to 24 months are expected and may be employed with the core described herein.

Commencing with Unit 1 Cycle 12 and Unit 2 Cycle 10, the Westinghouse fuel assemblies are of the VANTAGE + design with Intermediate Flow Mixing grids (IFMs) and include a thin zirconium oxide coating applied to the bottom seven inches of each fuel rod to provide additional debris fretting protection in this area. All fuel assemblies are equipped with the “Small Hole” debris filtering bottom nozzle, an alternate protective grid (P-grid), and long solid end plugs on the fuel rods.

The IFM grids are designed to improve CPNPP fuel performance in the following areas:

- enhanced DNB margin as a result of increased flow mixing;
- mitigate Crud Induced Power Shift (CIPS); and
- stabilize fuel assemblies to decrease susceptibility to Incomplete Rod Insertion (IRI) events.

Commencing with Unit 2 Cycle 13 and Unit 1 Cycle 16, the Westinghouse fuel assemblies utilize the Standardized Debris Filter Bottom Nozzle (SDFBN) and the Robust Protective Grid (RPG). These features are discussed in detail in Section 4.2.2.

The Westinghouse fuel assemblies consist of 17x17 fuel rods with 0.360" nominal outer diameter fuel rods. The Westinghouse assemblies typically have 2.60 w/o enriched U235 axial blankets in the top and bottom six inch zones of each fuel rod.

Wet annular burnable absorber (WABA) rodlets consisting of B4C - Al2O3 pellets can be used in selected fresh Westinghouse fuel for the reload core design to shape the power distribution and achieve a desirable moderator temperature coefficient. Additionally, selected fresh Westinghouse fuel will contain Integral Fuel-Burnable Absorber (IFBA) pellets, which also act as a burnable absorber. These fuel pellets are coated with a thin layer of ZrB2. Both the WABA and IFBA rods used typically contain 120 inches of absorber material, axially centered in the active fuel region, but the length can vary depending on the specific core design requirements.

The core is cooled and moderated by light water at a pressure of 2250 pounds per square inch absolute (psia) in the Reactor Coolant System. The moderator coolant contains boron as a neutron poison. The concentration of boron in the coolant is varied as required to control relatively slow reactivity changes including the effects of fuel burnup. Additional boron, in the form of burnable poison rods, is employed in the core to compensate for excess reactivity, control of radical flux distributions, and to maintain the required moderator temperature coefficient.

The fuel rods are supported at intervals along their length by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. The fuel rods consist of slightly enriched uranium dioxide ceramic cylindrical pellets contained in slightly cold worked Zircaloy-4 or ZIRLO® High Performance Fuel Cladding Material which is plugged and seal welded at the ends to encapsulate the fuel. All fuel rods are pressurized with helium during fabrication.

The center position in the assembly is reserved for the incore instrumentation, while the remaining 24 positions in the array are equipped with guide thimbles joined to the grids and the top and bottom nozzles. Depending upon the position of the assembly in the core, the guide thimbles are used as core locations for rod cluster control assemblies (RCCAs), neutron source assemblies, and burnable poison assemblies. Otherwise, the guide thimbles are fitted with plugging devices to limit bypass flow.

The bottom nozzle is a box-like structure which serves as a bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly.

The top nozzle assembly functions as the upper structural element of the fuel assembly in addition to providing a partial protective housing for the RCCA or other components.

The RCCAs each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies contain full length absorber material to control the reactivity of the core under operating conditions, to control axial power distribution and to provide a mechanism for reactor shutdown.

The nuclear design analyses and evaluation establish physical locations for control rods and burnable poisons and physical parameters such as fuel enrichments and boron concentration in the coolant. The nuclear design evaluation established that the reactor core has inherent characteristics which together with corrective actions of the reactor control, and protective

systems provide adequate reactivity control even if the highest reactivity worth RCCA is stuck in the fully withdrawn position.

The design also provides for inherent stability against diametral and azimuthal power oscillations and for control of induced axial power oscillation through the use of the control rods.

The thermal-hydraulic design analyses and evaluation establish coolant flow parameters which assure that adequate heat transfer is provided between the fuel clad and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution and mixing. The fuel assembly spacer grid design induces additional flow mixing between the various flow channels within a fuel assembly as well as between adjacent assemblies.

Instrumentation is provided in and out of the core to monitor the nuclear, thermal-hydraulic, and mechanical performance of the reactor and to provide inputs to automatic control functions. Table 4.1-1 presents a comparison of the principal nuclear, thermal-hydraulic, and mechanical design parameters of the Comanche Peak units.

The analysis techniques employed in the core design are addressed in Table 4.1-2. The loading conditions considered in general for the core internals and components are addressed in Section 4.2 and Table 4.1-3.

TABLE 4.1-1
REACTOR DESIGN COMPARISION TABLE

(Sheet 1 of 3)

Design Parameters[a]		Luminant Pre-Uprate Design	Luminant Uprate Design
1.	Reactor core heat output (MW_t)[b]	3458	3612
2.	Reactor core heat output (10^6) BTU/hr)[b]	11,799	12,325
3.	Heat generated in fuel (%)	97.4	97.4
4.	System pressure, nominal (psia)	2250	2250
5.	Minimum DNBR for design transients		
	Typical flow channel		
	WRB-1	1.23	1.23
	WRB-2	1.23	1.23
	Thimble (cold wall) flow channel		
	WRB-1	1.23	1.23
	WRB-2	1.22	1.22
6.	DNB correlation[c]	WRB-2/WRB-1 W-3	WRB-2/WRB-1 W-3
Coolant flow(e)			
7.	Total vessel flow-rate (10^6 lb_m/hr) (based on thermal design flow - TDF, including bypass)	142.04	142.30
8.	Effective flow-rate for heat transfer (10^6 lb_m/hr) (based on TDF, excluding bypass)[d]	133.80	134.04
9.	Effective flow area for heat transfer (ft^2)	54.13	54.13
10.	Average mass velocity along fuel rods (ft/sec) (based on TDF, excluding bypass)[d]	14.84	14.84
11.	Average mass velocity (10^6 lb_m/hr-ft^2) (based on TDF, excluding bypass)[d]	2.472	2.476
Coolant temperature			
12.	Nominal inlet (^oF)	559.2	558.0
13.	Average rise in vessel (^oF)	60.0	62.4
14.	Average rise in core (^oF)	63.3	65.8
15.	Average in core (^oF)	592.6	592.8
16.	Average in vessel (^oF)	589.2	589.2

TABLE 4.1-1
REACTOR DESIGN COMPARISION TABLE

(Sheet 2 of 3)

Design Parameters[a]		Luminant Pre-Uprate Design	Luminant Uprate Design
Heat transfer			
17.	Active heat transfer surface area (ft^2)[e]	57,505	57,505
18.	Average heat flux (BTU/hr-ft^2)[e]	199,900	208,802
19.	Maximum heat flux for normal operation (BTU/hr-ft^2)[e][f]	483,758	522,005
20.	Average linear power (kW/ft)[e]	5.520	5.766
21.	Peak linear power for normal operation (kW/ft)[e][f]	13.36	14.41
22.	Peak linear power resulting from overpower transients/ operator errors, assuming a maximum overpower of 118.5% (kW/ft)[e]	$\leq$22.4	$\leq$22.4
23.	Heat flux hot channel factor (FQ)	2.42	2.50
24.	Peak fuel central temperature for prevention of centerline melt (°F)	4700	4700
25.	Design	RCC canless 17 x 17	RCC canless 17 x 17
26.	Number of fuel assemblies	193	193
27.	UO_2 rods per assembly	264	264
28.	Rod pitch (in.)	0.496	0.496
29.	Overall dimensions (in.)	8.246 x 8.426	8.246 x 8.426
30.	Fuel weight, as UO_2 (lb)	205,352	205,352
31.	Zircaloy/ZIRLO weight (lb) (active core)	50,759	50,759
32.	Number of grids per assembly	2 - R type 6 - Z type 3 - IFM 1 - Alt P-Grid	2 - R type 6 - Z type 3 - IFM 1 - Alt P-Grid or RPG
Fuel rods			
33.	Number	50,952	50,952

TABLE 4.1-1
REACTOR DESIGN COMPARISION TABLE

(Sheet 3 of 3)

Design Parameters[a]		Luminant Pre-Uprate Design	Luminant Uprate Design
34.	Outside diameter (in.)	0.360	0.360
35.	Diametral gap (in.)	0.0062	0.0062
36.	Clad thickness (in.)	0.0225	0.0225
37.	Clad material	ZIRLO	ZIRLO
Fuel pellets			
38.	Material	UO_2 sintered	UO_2 sintered
39.	Density (% of theoretical)	95	95
40.	Diameter (in.)	0.3088	0.3088
41.	Length (in.) (standard pellet/blanket pellet)	0.370/0.500	0.370/0.500
Rod cluster control assemblies			
42.	Neutron absorber	Ag-In-Cd	Ag-In-Cd
43.	Cladding material	Type 304 SS cold-worked	Type 304 SS cold-worked
44.	Clad thickness (in.)	0.0185	0.0185
45.	Number of clusters	53	53
46.	Number of absorber rods per cluster	24	24
47.	Core height, active fuel (in.)[e]	143.7	143.7

a) Unit 1 flow rates are used for comparison. The lower Unit 1 flow rates are more limiting with respect to DNB. Also, these flows are to be used for both units for uprate conditions. Unit 2 coolant temperatures are used for comparison. The higher Unit 2 temperatures are more limiting with respect to DNB.

b) The proposed power level of 3612 MW_t has been used for thermal-hydraulic design analyses.

c) See paragraph 4.4.2.1.1 for the use of the W-3 correlation.

d) Based on thimble plugs inserted.

e) Based on densified active fuel length.

f) Based on maximum FQ of 2.42 for pre-uprate design and 2.50 for uprate design.

TABLE 4.1-2
ANALYTICAL TECHNIQUES IN CORE DESIGN

(Sheet 1 of 2)

Analysis	Technique	Computer Code Referenced	Section
Mechanical design of core internals loads, deflections, and stress analysis	Static and dynamic modeling structural analysis code, and others	Blowdown code, FORCE, finite element	3.7.2.1 3.9.2 3.9.3
Fuel rod design Full performance characteristics (temperature, internal pressure, clad stress, etc.)	Semi-empirical thermal model of fuel rod with consideration of fuel density changes, heat transfer, fission gas release, etc.	Westinghouse fuel rod design model	4.2.1.1 4.2.3.2 4.2.3.3 4.3.3.1 4.4.2.8
Nuclear design Cross-sections and group constants	Microscopic data; macroscopic constants for homogenized core regions	Modified ENDF/B library LEOPARD/CINDER type or PHOENIX-P	4.3.3.2
	Group constants for control rods with selfshielding	HAMMER-AIM or PHOENIX-P	4.3.3.2
X-Y and X-Y-Z power distributions, fuel depletion, critical boron concentrations, X-Y and X-Y-Z xenon distributions, reactivity coefficients	2-group diffusion theory	TURTLE (2-D) or ANC (2-D or 3-D)	4.3.3.3
Axial power distributions, control rod worths, and axial xenon distribution	1-D, 2-group diffusion theory	PANDA	4.3.3.3
Fuel rod power	Integral transport theory	LASER	4.3.3.1
Effective resonance temperature	Monte Carlo weighting function	REPAD	4.3.3.1

TABLE 4.1-2
ANALYTICAL TECHNIQUES IN CORE DESIGN

(Sheet 2 of 2)

Analysis	Technique	Computer Code Referenced	Section
Criticality of reactor and fuel assemblies	2-D, 2-group diffusion theory	APX system of codes, KENO-IV, SCALE 5.1, PARAGON	4.3.2.6
Vessel irradiation	Multigroup spatial dependent transport theory	DOT	4.3.2.8
Thermal-hydraulic design Steady state	Sub-channel analysis of local fluid conditions in rod bundles, including inertial and cross-flow resistance terms; solution progresses from core-wide to hot assembly to hot channel.	VIPRE-01	4.4.4.5.1
Transient departure from nucleate boiling	Sub-channel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservative equations; solution progresses from core-wide to hot assembly to hot channel.	VIPRE-01	4.4.5.4

TABLE 4.1-3
DESIGN LOADING CONDITIONS FOR REACTOR CORE COMPONENTS

1. Fuel assembly weight and core component weights (BAs, sources, plugging devices)
2. Fuel assembly spring forces and core component spring forces
3. Internals weight
4. Control rod trip (equivalent static load)
5. Differential pressure
6. Spring preloads
7. Coolant flow forces (static)
8. Temperature gradients
9. Differences in thermal expansion
    a. Due to temperature differences
    b. Due to expansion of different materials
10. Interference between components
11. Vibration (mechanically or hydraulically induced)
12. One or more loops out of service
13. All operational transients listed in table 3.9.N.1-1
14. Pump overspeed
15. Seismic loads (operating basis earthquake and safe shutdown earthquake)
16. Blowdown forces (due to cold and hot leg break)

4.2 FUEL SYSTEM DESIGN

The plant conditions for design are divided into four categories in accordance with their anticipated frequency of occurrence and risk to the public: Condition I - Normal Operation; Condition II - Incidents of Moderate Frequency; Condition III - Infrequent Incidents; Condition IV - Limiting Faults. The bases and description of plant operation and events involving each condition are given in the accident analysis, Chapter 15.

The reactor is designed so that its components meet the following performance and safety criteria:

1. The mechanical design of the reactor core components and their physical arrangement, together with corrective actions of the reactor control, protection and emergency cooling systems (when applicable) assure that:

   a. Fuel damage[a] is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant cleanup system and are consistent with plant design bases.

   b. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged[a] although sufficient fuel damage might occur to preclude immediate resumption of operation.

   c. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

2. The fuel assemblies are designed to withstand, without exceeding the criteria of Section 4.2.1.5, loads induced during shipping, handling and core loading.

3. The fuel assemblies are designed to accept control rod insertions in order to provide the required reactivity control for power operations and reactivity shutdown conditions.

4. All fuel assemblies have provisions for the insertion of incore instrumentation necessary for plant operation.

5. The reactor internals in conjunction with the fuel assemblies and incore control components direct reactor coolant through the core. This achieves acceptable flow distribution and restricts bypass flow so that the heat transfer performance requirements can be met for all modes of operation.

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a. Fuel damage as used here is defined as penetration of the fission barrier (i.e., the fuel rod clad).

4.2.1 DESIGN BASES

The fuel rod and fuel assembly design bases are established to satisfy the general performance and safety criteria presented in Section 4.2.

The detailed fuel rod design establishes such parameters as pellet size and density, clad-pellet diametral gap, gas plenum size, and helium pre-pressurization level. The design also considers effects such as fuel density changes, fission gas release, clad creep, and other physical properties which vary with burnup. The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive internal rod gas pressures due to fission gas releases, and excessive cladding stresses and strains. This is achieved by designing the fuel rods so that the conservative design basis in the following subsections are satisfied during Condition I and Condition II events over the fuel lifetime. For each design basis, the performance of the limiting fuel rod must not exceed the limits specified by the design basis.

Structural integrity of the fuel assembly structure is assured by setting limits on stresses and deformations due to various loads and by determining that the assembly structure does not interfere with the functioning of other components. Two types of loads are considered.

1. Normal and abnormal loads which are defined for Conditions I and II.

2. Abnormal loads which are defined for Conditions III and IV.

The design bases for the incore control components are described in Section 4.2.1.6.

4.2.1.1 Cladding

1. The desired fuel rod clad is a material which has a superior combination of neutron economy (low absorption cross section), high strength to resist deformation due to differential pressures and mechanical interaction between fuel and clad, high corrosion resistance to coolant, fuel and fission products, and high reliability. Zircaloy-4 and ZIRLO have this desired combination of clad properties. There is considerable PWR operating experience on the capability of Zirconium alloys as a clad material. Information of the material chemical and mechanical properties of the cladding is given in Reference 2 and Reference 9 with due consideration of temperature and irradiation effects.

   Other fuel rod criteria are specified in Reference 32.

2. Stress-strain limits

   a. Clad stress

      The von Mises criterion is used to calculate the effective stresses. The cladding stresses under Condition I and II events are less than the Zircaloy 0.2% offset yield stress with due consideration of temperature and irradiation effects. While the clad has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design basis.

   b. Clad tensile strain

The total tensile creep strain is less than 1 percent from the un-irradiated condition. The elastic tensile strain during a transient is less than 1 percent from the pre-transient value. This limit is consistent with proven practice.

3. Vibration and fatigue

a. Strain fatigue

The cumulative strain fatigue cycles are less than the design strain fatigue life. This basis is consistent with proven practice (Ref. [1]).

b. Vibration

Potential fretting wear due to vibration is prevented by design of the fuel assembly grid springs and dimples assuring that the stress-strain limits are not exceeded during design life. Fretting of the clad surface can occur due to flow-induced vibration between the fuel rods and fuel assembly grid springs. Vibration and fretting forces vary during the fuel life due to clad diameter creepdown combined with grid spring relaxation.

4.2.1.2 Fuel Material

1. Thermal-physical properties

Fuel pellet temperatures - The center temperature of the hottest pellet is to be below the melting temperature of the UO_2 (melting point of 5080ºF [3] unirradiated and decreasing by 58ºF per 10,000 MWD/MTU). While a limited amount of center melting can be tolerated, the design conservatively precludes center melting. A calculated fuel centerline temperature of 4700ºF has been selected as an overpower limit to assure no fuel melting. This provides sufficient margin for uncertainties as described in Section 4.4.2.9.

The normal design density of the fuel is 95 percent of theoretical.

2. Fuel densification and fission product swelling

The design bases and models used for fuel densification and swelling are provided in Reference 5.

3. Chemical properties

The mechanical and chemical characteristics of the cladding is discussed in Reference 9.

4.2.1.3 Fuel Rod Performance

1. Fuel rod models

The basic fuel rod models and the ability to predict operating characteristics are given in References 5 and 9.

2. Mechanical design limits

Cladding collapse shall be precluded during the fuel rod design lifetime. The models described in References 6 and 29 are used for this evaluation.

The rod internal gas pressure shall remain below the value which causes the fuel-clad diametral gap to increase due to outward cladding creep during steady state operation. Rod pressure is also limited such that extensive departure from nucleate boiling (DNB) propagation shall not occur during normal operation and accident events.

The fuel rod internal pressure design criteria will permit the internal pressure to exceed the reactor coolant system pressure. Consideration has been given to the effects of a postulated propagation process wherein fuel rods are simultaneously in DNB and exceed system pressure, causing subsequent ballooning and touching of adjacent rods resulting in more rods reaching DNB and failing.

Even with a very conservative assessment of this process, the number of fuel rods which could fail in Condition III and IV events increased by only a small percentage. The dose consequences of the accidents evaluated in Chapter 15 would remain essentially unchanged considering this increase.

4.2.1.4 Spacer Grids

1. Mechanical limits and materials properties

The grid component strength criteria are based on experimental tests. The limit is established at 0.9 Pc, where Pc is the experimental collapse load. This limit is sufficient to assure that under worst-case combined seismic and blowdown loads the core will maintain a geometry amenable to cooling.

2. Vibration and fatigue

The grids shall provide sufficient fuel rod support to limit fuel rod vibration and maintain clad fretting wear to within acceptable limits (defined in Section 4.2.1.1).

4.2.1.5 Fuel Assembly

1. Structural design

As previously discussed in Section 4.2.1, the structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various operational and accident loads.

These limits are applied to the design and evaluation of the top and bottom nozzles, guide thimbles, grids, and the thimble joints.

The design bases for evaluating the structural integrity of the fuel assemblies are:

For the normal operating and upset conditions, the fuel assembly component structural design criteria are established for the two primary material categories, namely austenitic

steels and Zircaloy/ZIRLO. The stress categories and strength theory presented in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, are used as a general guide. The maximum shear-theory (Tresca criterion) for combined stresses is used to determine the stress intensities for the austenitic steel components. The stress intensity is defined as the numerically largest difference between the various principal stresses in a three dimensional field. The allowable stress intensity value for austenitic steels, such as nickel-chromium-iron alloys, is given by the lowest of the following:

a. One-third of the specified minimum tensile strength or 2/3 of the specified minimum yield strength at room temperature;

b. One-third of the tensile strength or 90 percent of the yield strength at temperature but not to exceed 2/3 of the specified minimum yield strength at room temperature.

The stress limits for the austenitic steel components are given below. All stress nomenclature is per the ASME Boiler and Pressure Vessel Code, Section III.

Stress Intensity Limits

Categories	Limit
General Primary Membrane Stress Intensity	S_m
Local Primary Membrane Stress Intensity	1.5 S_m
Primary Membrane plus Bending Stress Intensity	1.5 S_m
Total Primary plus Secondary Stress Intensity	3.0 S_m

The Zircaloy-4 or ZIRLO structural components which consist of guide thimbles, spacer grids, Intermediate Flow Mixing (IFM) grids and fuel tubes are in turn subdivided into two categories because of material differences and functional requirements. The fuel tube and grid design criteria is covered separately in Section 4.2.1.1 and 4.2.1.4, respectively. The maximum shear theory is used to evaluate the guide thimble design. For conservative purposes, the Zircaloy/ZIRLO unirradiated properties are used to define the stress limits.

c. Abnormal loads during Conditions III or IV.

Westinghouse (W) fuel assemblies (except W Optimized Fuel Assemblies): worst case represented by combined seismic and blowdown loads.

W Optimized Fuel Assemblies: worst case represented by seismic loads, or blowdown loads during a LOCA event.

1. Deflections or failures of components cannot interfere with the reactor shutdown or emergency cooling of the fuel rods.

2. The fuel assembly structural component stresses under faulted conditions are evaluated using primarily the methods outlined in Appendix F of the ASME Boiler and Pressure Vessel Code, Section III. Since the current analytical methods utilize elastic analysis, the stress allowables are defined as the smaller value of 2.4 S_m or 0.70 S_u for primary membrane and 3.6 S_m or 1.05 S_u for primary membrane plus primary bending, where Su is the material ultimate tinsels strength. For the austenitic steel fuel assembly components, the stress intensity is defined in accordance with the rules described in the previous section for normal operating conditions. For the Zircaloy-4 or ZIRLO components the stress intensity limits are set at 2/3 of the material yield strength, S_y, at reactor operating temperature. This results in Zircaloy stress limits being the smaller of 1.6 S_y or 0.70 S_u for primary membrane and 2.4 S_y or 1.05 S_u for primary membrane plus bending. For conservative purposes, the Zircaloy/ZIRLO unirradiated properties are used to define the stress limits.

   Typical material and chemical properties of the fuel assembly components are given in References [2] and [9] for Zircaloy-4 and ZIRLO, respectively.

2. Thermal-hydraulic design

   This topic is covered in Section 4.4.

4.2.1.6 Incore Control Components

The core components consist of the rod cluster control assemblies (RCCAs), the primary and secondary source assemblies, the thimble plug assemblies and burnable absorber assemblies. A description of these components is provided in subsection 4.2.2.

1. Mechanical Design Basis

   a. Control Rod Assemblies

      The mechanical design bases for the control rods are consistent with the loading conditions under Article NB-3000 of the ASME Boiler and Pressure Vessel Code, Section III.

      - External pressure equal to the Reactor Coolant System operating pressure with appropriate allowance for overpressure transients.
      - Wear allowance equivalent to 1000 reactor trips.
      - Bending of the rod due to a misalignment in the guide tube.
      - Forces imposed on the rods during rod drop.
      - Loads imposed by the accelerations of the control rod drive mechanism.
      - Radiation exposure during maximum core life.

- Temperature effects from room to operating conditions.

b. Burnable Absorber, Thimble Plug and Source Rod Assemblies

The burnable absorber assemblies, thimble plug assemblies and source assemblies are static core components.

The mechanical design of these components satisfy the following:

1. Accommodate the differential thermal expansion between the fuel assembly and the core internals,

2. Maintain positive contact with the fuel assembly and the core internals.

3. Limits the flow through each occupied thimble to acceptable design values.

2. Thermal-Physical Properties of the Absorber Material

The absorber material for the RCCAs is Ag-In-Cd with nominal proportions silver 80%, Indium, 15% and Cadmium 5%.

The burnable absorber material is non-structural with nominal lengths between 120 and 142 inches depending on the cycle specific core design. The structural elements of the burnable poison rod are designed to maintain the absorber geometry even if the absorber material is fractured. The rods are designed so that the absorber material is below its softening temperature. In addition, the structural elements are designed to prevent excessive slumping.

3. Compatibility of the Absorber and Cladding Materials

The control rod and source rod cladding is cold drawn type 304 stainless steel tubing. Extensive in-reactor experience and available quantitative information show that reaction rates between 304 stainless steel and water, or other in-core metals are negligible at operational temperatures (Reference [2]).

4.2.1.7 Surveillance Program

A testing and fuel surveillance operational experience program has been and is being conducted to verify the adequacy of the fuel performance and design bases. For W-supplied fuel, the program for standard fuel is discussed in Section 4.2.4.5. Reference [7] provides a description of the tests performed and a summary of the results for the W fuel. Fuel surveillance and testing results, as they become available, are used to improve fuel rod design and manufacturing processes and assure that the design bases and safety criteria are satisfied.

4.2.2 DESIGN DESCRIPTION

Commencing with Unit 1 Cycle 12 and Unit 2 Cycle 10, the Westinghouse fuel assembly design is the VANTAGE + fuel design. The fuel assemblies each contain 264 fuel rods of 0.360" nominal outer diameter, twenty-four guide thimble tubes, and one instrumentation tube in a 17x17 array

supported by eight spacer grids, three Intermediate Flow Mixer (IFM) grids and one debris filtering protective grid (P-grid) in the fuel assembly structure. The instrumentation tube is located in the center position of the fuel assembly and provides a channel for insertion of an incore neutron detector, if the fuel assembly is located in an instrumented core position. The guide thimble tubes provide channels for insertion of either a rod cluster control assembly (RCCA), a neutron source assembly (primary or secondary), a burnable poison assembly (WABA) or a plugging device, depending on the position of the particular fuel assembly in the core. The assembly mid-span grids and the IFM grids are made of ZIRLO and the top and bottom grids and the P-grid are made of Inconel. Each fuel assembly contains a reconstitutable top nozzle (RTN) that has a unique serial number engraved on the nozzle for assembly ID. Each fuel assembly also contains a debris filter bottom nozzle (DFBN). The DFBN, P-grid, debris mitigation long bottom end-plugs on the fuel rods and a protective oxide coating applied to approximately the bottom 7" of each fuel rod provides the maximum protection for debris-induced fretting. The zirconium oxide pre-coating provides a hardened surface on the fuel cladding making the clad more resistant to debris-induced fretting during the first few months of operation before the oxide coating develops naturally. Since zirconium oxide forms naturally on the surface of the fuel rod clad during normal fuel operations, the pre-coating of the clad does not create any new potential adverse impacts to safety analyses or fuel operations. The central fuel pellets are 5 w/o enriched U^{235}, dished and chamfered on the ends. Typically, a nominal 6” of pellets at the top and bottom of the fuel rods are axial blanket pellets which are 2.6 w/o enriched U^{235}. The central fuel pellets may be coated with zirconium di-boride (ZrB_2) as a burnable absorber. These pellets are normally referred to as Integral Fuel Burnable Absorber (IFBA) pellets and can have varying loading of boron to support the fuel management design of the core. The IFBA fuel rods are typically pressurized to 100 psig and utilize annular blanket pellets to allow for the extra gas production associated with the helium release from the ZrB_2. Non-IFBA fuel rods are typically pressurized to 275 psig. The annular blankets and reduced diameter plenum springs (variable pitch plenum spring) also provide a larger internal plenum to mitigate the effects of the increased gas production to ensure the fuel rods meet their design criterion for rod internal pressure. All of the fuel pellets are sintered to a nominal 95.5% of theoretical density. Figure 4.2-1 shows a cross section of a typical fuel assembly array, and Figure 4.2-2 shows a typical fuel assembly full length view. Figure 4.2-3 shows a typical fuel rod configuration.

If a fuel rod becomes damaged during operations, such as debris fretting failures, the rod can be removed and a stainless steel filler rod can be inserted to allow continued use of the fuel assembly in subsequent cycles.

The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles. All fuel assemblies in the core are functionally identical.

Each fuel assembly is installed vertically in the reactor vessel and stands upright on the lower core plate, which is fitted with alignment pins to locate and orient the assembly. After all fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears downward against the fuel assembly top nozzle via the holddown springs to hold the fuel assemblies in place.

Improper orientation of fuel assemblies within the core is prevented by the use of an indexing hole in one corner of the top nozzle top plate. The assembly is oriented with respect to the

handling tool and the core by means of a pin which is inserted into this indexing hole. Additionally, visual confirmation of orientation is provided by an engraved identification number on the opposite corner clamp.

Several optional design enhancements have been made to the W optimized fuel design being used at Comanche Peak. These enhancements include the following:

- An alternate protective grid is included at the bottom of the assembly, to provide an additional debris barrier, thereby improving fuel reliability. The protective grid will also provide additional grid/rod fretting resistance by supporting the bottom of the fuel rod. Beginning with Unit 2 Cycle 13 and Unit 1 Cycle 16, the Westinghouse fuel assemblies utilize the Robust Protective Grid (RPG). This feature is discussed in detail in Section 4.2.2.2.4.

- A lengthened, debris-mitigating fuel rod bottom end plug, longer than prior design, is used which extends up through the protective grid, limiting the span for fretting to the solid end plug.

- A lengthened external grip top end plug is also used for reconstitution capability.

- A smaller gap between the rods and the bottom nozzle exists as a result of the longer end plug while maintaining the fuel stack at the same elevation.

- The mid-grids and IFM grids are fabricated from the ZIRLO alloy for improved corrosion performance margin to high burnup.

- The top and bottom nozzles and top grid assembly sleeve are fabricated of a low cobalt Type 304 Stainless Steel for dose reduction.

- Extended burnup bottom grid design.

- An oxide coating at the bottom 7 inches of the fuel rod for increased resistance to debris in this area.

4.2.2.1 Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in Zircaloy-4 or ZIRLO tubing which is plugged and seal welded at the ends to encapsulate the fuel (the reconstituted rods are made of stainless steel). A schematic of a typical fuel rod is shown in Figure 4.2-3. The fuel pellets are right circular cylinders consisting of slightly enriched uranium dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly to allow greater axial expansion at the center of the pellets.

To avoid overstressing of the clad or seal welds, void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the clad and the fuel, and fuel density changes during irradiation.

Shifting of the fuel within the clad during handling or shipping prior to core loading is prevented by a stainless steel or Inconel helical spring which bears on top of the fuel. At assembly the pellets are stacked in the clad to the required fuel height, the spring is then inserted into the top end of

the fuel tube and the end plugs pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process in order to minimize compressive clad stresses and prevent clad flattening due to coolant operating pressures.

The fuel rods are presently being pre-pressurized and designed so that: 1) the internal gas pressure mechanical design limit given in Section 4.2.1.3 is not exceeded, 2) the cladding stress-strain limits (Section 4.2.1.1) are not exceeded for Condition I and II events, and 3) clad flattening will not occur during the fuel core life.

Integral Fuel Burnable Absorber (IFBA)

The IFBA coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin zirconium diboride (ZrB_2) coating on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of IFBA rods within an assembly may vary depending on specific application. The ends of the enriched coated pellets and enriched uncoated pellets are dished to allow for greater axial expansion at the pellet centerline and to increase the void volume for fission gas release. Analysis of IFBA rods includes any geometry changes necessary to model the presence of burnable absorber, and conservatively models the gas release from the coating.

Variation of IFBA loading is allowed according to the needs presented for power distribution control in designing the core each cycle. Enriched IFBA, containing a greater presence of the neutron absorbing isotope B-10, can be specified in terms of multiples of the natural boron IFBA rod. IFBA loadings of 1.5X or 2.0X have been applied in various regions, for example. This is not to preclude other loadings (1.25X, 1.6X) according to need. Also, variations in the length of the span of the fuel stack that include IFBA pellets is at the discretion of the core designer, according to requirements imposed. The acceptability of a proposed IFBA configuration is dependent on meeting fuel rod design criteria. The principle criteria which come to bear are the fuel rod internal pressure criterion and the clad stress criterion.

The rod internal pressure design criterion is impacted by the IFBA loading because some additional amount of helium is produced from the “burnout” of the ZrB_2 pellet coating over and above the inventory of fission product gases. Different IFBA loadings will produce varying increases of rod internal pressure. To offset this extra helium within the rod, the initial helium prepressurization of the IFBA rod (100 psig backfill pressure) is reduced as compared to the non-IFBA rod (275 psig backfill pressure). Also, for greater IFBA loadings, further pressure reduction measures are employed, such as providing additional void volume in the rod by inserting annular axial blankets. The annular axial blanket is formed by inserting pellets with an annulus (and void of approximately 25% of a solid pellet) for several inches in the top and bottom ends of the fuel stack. Regardless of the region specific IFBA load, continued compliance to the rod internal pressure design criterion is assured by reload analysis and evaluation.

With respect to the clad stress design criterion, generic analyses and significant design experience have demonstrated that changes in boron loadings and pre-pressurization levels consistent with current W non-IFBA and IFBA designs do not significantly impact the ability of the fuel to satisfy clad stress limits. Furthermore, for Comanche Peak, assuming 2X IFBA design (with annular axial blankets), the significant design parameters which can impact clad stress, including overall IFBA pellet diameter and rod internal pressure (which impacts the rate of clad creepdown onto the pellet surface), have been shown to satisfy all design criteria.

Axial Blankets

The axial blankets are a section of fuel pellets at each end of the fuel rod pellet stack. Axial blankets reduce neutron leakage and improve fuel utilization (thus reducing the fuel costs). Neutrons born near the top or bottom of the assembly have a high probability of migrating into the surrounding water/core support structure without causing fission. By implementing axial blankets, a larger proportion of enriched fuel is located in higher flux regions of the core, thus reducing the probability of neutron leakage. The axial blankets utilize chamfered pellets which are physically different (length) than the other fuel pellets to help prevent accidental mixing during manufacturing.

4.2.2.2 Fuel Assembly Structure

The fuel assembly structure consists of a bottom nozzle, top nozzle, guide thimbles and grids. A typical fuel assembly structure is shown in Figure 4.2-2.

4.2.2.2.1 Bottom Nozzle

The bottom nozzle serves as a bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The square nozzle is fabricated from Type 304 stainless steel and consists of a structural support with flow channels and four angle legs with bearing plates. The legs form a plenum for the inlet coolant flow to the fuel assembly. The structural support also prevents accidental downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel assembly guide tubes by locked screws which penetrate through the nozzle and mate with a threaded plug in each guide tube.

Coolant flow through the fuel assembly is directed from the plenum in the bottom nozzle upward through the penetrations in the plate to the channels between the fuel rods.

Axial loads (holddown) imposed on the fuel assembly and the weight of the fuel assembly are transmitted through the bottom nozzle to the lower core plate. Indexing and positioning of the fuel assembly is controlled by alignment holes in two diagonally opposite bearing plates which mate with locating pins in the lower core plate. Any lateral loads on the fuel assembly are transmitted to the lower core plate through the locating pins.

The bottom nozzle design has a reconstitution feature which allows the nozzle to be easily removed. A locking cup is used to lock the guide thimble screw in place at assembly or during reconstitution and nozzle reattachment if required.

The Comanche Peak fuel also has a debris-filter bottom nozzle (DFBN) feature. The DFBN is a revised version of the 17x17 nozzle design. The revised design includes an improved pattern of flow holes which:

1. Reduces the passage of debris into the fuel assembly,

2. Maintains the structural integrity of the nozzle design,

3. Maintains the hydraulic performance of the design.

Westinghouse has developed the Standardized Debris Filter Bottom Nozzle (SDFBN) for Westinghouse 17x17 fuel which is designed to have a loss coefficient that is the same independent of supplier. The SDFBN has eliminated the side skirt communication flow holes as a means of improving the debris mitigation performance of the bottom nozzle. This nozzle has been evaluated and meets all of the applicable mechanical design criteria. In addition, there is no adverse affect on the thermal hydraulic performance of the SDFBN either with respect to the pressure drop or with respect to DNB. The SDFBN was implemented at Comanche Peak beginning with Unit 2 Cycle 13 and Unit 1 Cycle 16.

4.2.2.2.2 Top Nozzle

The top nozzle assembly functions as the upper structural element of the fuel assembly in addition to providing a partial protective housing for the rod cluster control assembly or other components. It consists of an adapter plate, enclosure, top plate, and pads. The assembly has holddown springs mounted on the top nozzle. The springs are made of Inconel-718 and spring screws are made of Inconel-600 or Inconel-718, whereas other components are made of Type 304 stainless steel.

The square adapter plate is provided with penetrations to permit the flow of coolant upward through the top nozzle. Other round holes are provided to accept thimble tubes. The ligaments in the plate cover the tops of the fuel rods and prevent their upward ejection from the fuel assembly. The enclosure is a box-like structure which sets the distance between the adapter plate and the top plate. The top plate has a large square hole in the center to permit access for the control rods and the control rod spiders. Holddown springs are mounted on the top plate and are fastened in place by bolts and clamps located at two diagonally opposite corners. On the other two corners integral pads are positioned which contain alignment holes for locating the upper end of the fuel assembly. The top nozzle is designed to allow reconstitution of the assembly.

The Westinghouse Integral Nozzle (WIN) top nozzle design, first introduced in Unit 2 Cycle 12 and Unit 1 Cycle 15, is a direct replacement for the removable top nozzle design. The WIN design eliminates the spring screw for attachment of the hold-down springs, which are now assembled into the nozzle pad and pinned in place. The flow plate, thermal characteristics, and method of attachment of the nozzle are all unchanged from the removable top nozzle design.

4.2.2.2.3 Guide and Instrument Thimbles

The guide thimbles are structural members which also provide channels for the neutron absorber rods, burnable poison rods, neutron source or thimble plug assemblies. Each thimble is fabricated from Zircaloy-4 or ZIRLO tubing having two different diameters. The tube diameter at the top section provides the annular area necessary to permit rapid control rod insertion during a reactor trip. Holes are provided on the thimble tube above the dashpot to reduce the rod drop time. The lower portion of the guide thimble is swaged to a smaller diameter to reduce diametral clearances and produce a dashpot action near the end of the control rod travel during normal trip operation. The dashpot is closed at the bottom by means of an end plug which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation. The top end of the guide thimble is fastened to a sleeve by a mechanism which fits into the top nozzle adapter plate. The lower end of the guide thimble is fitted with an end plug which is then fastened into the bottom nozzle by a locked thimble screw. Each mid-grid is fastened to the guide thimble assemblies to create an integrated structure. The described methods of grid fastening are

standard and have been used successfully since the introduction of Zircaloy guide thimbles in 1969.

The central instrumentation thimble of each fuel assembly is constrained by seating in counterbores in each nozzle. This tube is a constant diameter and guides the incore neutron detectors.

4.2.2.2.4 Grid Assemblies

The fuel rods are supported at intervals along their length by grid assemblies which maintain the lateral spacing between the rods. Each fuel rod is supported within each grid by the combination of support dimples and springs. The grid assembly consists of individual slotted straps interlocked in an “egg-crate” arrangement. The straps contain spring fingers, support dimples and mixing vanes. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

For the W optimized fuel, the grid material is Inconel-718 or ZIRLO. The magnitude of the grid-restraining force on the fuel rod is set high enough to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods. The chemical composition of the IFM and mid-grids fabricated with ZIRLO alloy is similar to Zircaloy-4 except for a slight reduction in the content of tin (Sn) and iron (Fe) and the elimination of chromium (Cr). The ZIRLO alloy also contains a nominal amount of niobium (Nb). These changes, although small, are responsible for the improved corrosion resistance of ZIRLO compared to Zircaloy-4. The ZIRLO mid-grids and IFM grids have shorter sleeves.

Three types of grid assemblies are used in the W optimized fuel assembly. Six ZIRLO grids, with mixing vanes projecting from the edges of the straps into the coolant stream, are used in the high heat flux elevations of the fuel assemblies to promote mixing of the coolant. Two grids, one at each end of the assembly, do not contain mixing vanes on the internal straps. The top and bottom non-mixing vane grids are Inconel. Three ZIRLO IFM grids promote flow mixing and are non-structural components. The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past adjacent surfaces during handling or loading and unloading of the core. The Inconel and ZIRLO grids have been sized appropriately to assure sufficient structural strength for the assembly.

The IFM grids are located in the three uppermost spans between the ZIRLO mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the IFM grid can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids are not intended to be structural members. The outer strap configuration was designed similar to current fuel designs to preclude grid hang-up and damage during fuel handling. Additionally, the grid outer dimensions are smaller which further minimizes the potential for damage and reduces calculated forces during seismic/LOCA events. A coolable

geometry is, therefore, assured of the IFM grid elevation, as well as at the structural grid elevation.

The protective grid is made of Inconel 718. The protective grid is not intended to be a structural member.

Later, the “alternate” protective grid as used at Comanche Peak was developed to address plant specific concerns of Regulatory Guide 1.82, which requires that no passageway in the fuel assembly be less than the size of the mesh in the containment sump filter. The alternate protective grid design meets the regulatory guide by using an inner strap that is 0.1 inch shorter than the previous protective grid. The 0.1 inch is taken off the bottom of the inner straps and has the effect of raising the protective grid an additional 0.1 inch off the bottom nozzle.

Westinghouse has developed the Robust Protective Grid (RPG) as a result of observed failures in the field as noted in Post Irradiation Exams (PIE) performed at several different plants. It was determined that observed failures were the result of two primary issues; 1) fatigue failure within the protective grid itself at the top of the end strap and 2) stress corrosion cracking (SCC) primarily within the rod support dimples. The RPG implemented design changes such as increasing the maximum nominal height of the grid, increasing the ligament length and the radii of the ligament cutouts, and the use of four additional spacers or inserts to help strengthen the grid. The nominal height of the grid was increased to allow “V-notch” window cutouts to be added to help minimum flow-induced vibration caused by vortex shedding at the trailing edge of the inner grid straps. The design changes incorporated into the RPG design helped address the issues of fatigue failures and failures due to SCC. It was determined that the above changes do not impact the thermal hydraulic performance of the RPG as there is no change to the loss coefficient. In addition, the RPG retains the original protective grid function as a debris mitigation feature. The RPG was implemented at Comanche Peak beginning with Unit 2 Cycle 13 and Unit 1 Cycle 16.

4.2.2.3 Incore Control Components

Reactivity control is provided by neutron absorbing rods and a soluble chemical neutron absorber (boric acid). The boric acid concentration is varied to control long term reactivity changes such as:

1. Fuel depletion and fission product buildup.

2. Cold to hot, zero power reactivity change.

3. Reactivity change produced by intermediate term fission products such as xenon and samarium.

4. Burnable poison depletion.

Chemical and volume control is discussed in Chapter 9.

The rod cluster control assemblies provide reactivity control for:

1. Shutdown

2. Reactivity changes due to coolant temperature changes in the power range.

3. Reactivity changes associated with the power coefficient or reactivity.

4. Reactivity changes due to void formation.

Burnable absorbers may also be used for reactivity control. As the boron concentrations in the reactor coolant system is increased, the moderator temperature coefficient becomes more positive. The use of a soluble absorber alone could result in an unacceptably positive modertor temperature coefficient at beginning-of-life. Therefore, burnable absorbers may be used to reduce the soluble boron concentration sufficiently to ensure that the moderator temperature coefficient is acceptable for power operating conditions.

The most effective reactivity control components are the full length rod cluster control assemblies and their corresponding control rod drive mechanisms (CRDMs) which are the only moving parts in the reactor. Figure 4.2-8 identifies the full length rod cluster control and CRDM assembly, in addition to the arrangement of these components in the reactor relative to the interfacing fuel assembly and guide tubes. In the following paragraphs, each reactivity control component is described in detail. The CRDM assembly is described in Section 3.9N.4.

The neutron source assemblies provide a means of monitoring the core during periods of low neutron activity. The thimble plug assemblies limit bypass flow through those fuel assembly thimbles which do not contain control rods, burnable poison rods, or neutron source rods.

4.2.2.3.1 Full Length Rod Cluster Control Assembly

The full length rod cluster control assemblies are divided into two categories: control and shutdown. The control groups compensate for reactivity changes due to variations in operating conditions of the reactor, i.e., power and temperature variations. Two nuclear design criteria have been employed for selection of the control group. First the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that the power capability is met. The control and shutdown group provides adequate shutdown margin.

The rod cluster control assembly comprises a group of individual neutron absorber rods fastened at the top end to a common spider assembly, as illustrated in Figure 4.2-9.

The absorber material used in the control rod is silver-indium-cadium alloy. The absorber materials are essentially “black” to thermal neutrons and have sufficient additional resonance absorption to significantly increase their worth. The absorber is in the form of rods which are sealed in cold-worked, chromium plated, stainless steel tubes to prevent the absorber materials from coming in direct contact with the coolant (Figure 4.2-10). Sufficient diametral and end clearance is provided to accommodate relative thermal expansions and material swelling, as shown in Section 4.2.3.6.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles.

The material used in the absorber rod end plugs is Type 308 stainless steel. The design stresses used for the Type 308 material are the same as those defined in the ASME Code, Section III, for Type 304 stainless steel. At room temperature the yield and ultimate stresses per ASTM-580 are exactly the same for the two alloys. In view of the similarity of the alloy composition, the temperature dependence of strength for the two materials is also assumed to be the same.

The allowable stresses used as a function of temperature are listed in Table 1-1.2 of Section III of the ASME Boiler and Pressure Vessel Code. The fatigue strength for the Type 308 material is based on the S-N curve for austenitic stainless steels in Figure 1-9.2 of Section III.

The spider assembly is in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive rod assembly are machined into the upper end of the hub. A coil spring inside the spider body absorbs the impact energy at the end of a trip insertion. The radial vanes are joined to the hub by tack weld and braze and the fingers are joined to the vanes by brazing. A centerpost which holds the spring and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from Types 304 and 308 stainless steel except for the retainer which is fabricated of 17-4 pH material and the springs which are Inconel-718 alloy.

The absorber rods are fastened securely to the spider to assure trouble free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are tack welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

The overall length is such that when the assembly is withdrawn through its full travel the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble.

4.2.2.3.2 Burnable Poison Assembly

The poison rods in each fuel assembly are grouped and attached together at the top end of the rods to a hold down assembly by a flat perforated retaining plate which fits within the fuel assembly top nozzle and rests on the adaptor plate. The retaining plate and the poison rods are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals assembly is lowered into the reactor. This arrangement ensures that the poison rods cannot be ejected from the core by flow forces. Each rod is permanently attached to the base plate by a nut which is secured into place.

Wet annular burnable absorber (WABA) rodlets consisting of $B_4C - Al_2O_3$ pellets are used in selected fresh Westinghouse fuel depending upon the fuel management requirements for the cycle to shape the power distribution and achieve a desirable moderator temperature coefficient. Additionally, selected fresh Westinghouse fuel will contain Integral Fuel-Burnable Absorber (IFBA) pellets, which also act as a burnable absorber. These fuel pellets are coated with a thin layer of ZrB_2. The WABA and IFBA rods typically contain 120 inches of absorber material, axially

centered in the active fuel region, but the length may vary depending on the fuel management requirements of the cycle. Each burnable poison assembly consists of burnable poison rods attached to a hold down assembly. A typical burnable poison assembly is shown in Figure 4.2-12. When needed for nuclear considerations, burnable poison assemblies are inserted into selected thimbles within fuel assemblies.

The poison rods consist of absorber material contained within Zircaloy tubular cladding which is plugged and seal welded at the ends to encapsulate the absorber.

The absorber provides sufficient boron content to meet the criteria discussed in Section 4.3.1.

4.2.2.3.3 Neutron Source Assembly

The purpose of the neutron source assembly is to provide a base neutron level to ensure that the detectors are operational and responding to core multiplication neutrons. Since there is very little neutron activity during loading, refueling, shutdown, and approach to criticality, a neutron source is placed in the reactor to provide a positive neutron count of at least 2 counts per second on the source range detectors attributable to core neutrons. The detectors, called source range detectors, are used primarily when the core is subcritical and during special subcritical modes of operations.

The source assembly also permits detection of changes in the core multiplication factor during core loading refueling, and approach to criticality. This can be done since the multiplication factor is related to an inverse function of the detector count rate. Therefore a change in the multiplication factor can be detected during addition of fuel assemblies while loading the core, a change in control rod positions, and changes in boron concentration.

Both primary and secondary neutron source rods can be used. A primary source rod, containing a radioactive material, spontaneously emits neutrons during initial core loading and reactor startup. After the primary source rod decays beyond the desired neutron flux level, neutrons are then supplied by the secondary source rod. The secondary source rod contains a stable material, which must be activated by neutron bombardment during reactor operation. The activation results in the subsequent release of neutrons. This becomes a source of neutrons during periods of low neutron flux, such as during refueling and subsequent startups.

The reactor cores employ a sufficient number of secondary source assemblies (typically two) to ensure that the required base neutron level is provided.

Each secondary source assembly contains a symmetrical grouping of secondary source rods. Locations not filled with a source rod contain a thimble plug. A typical source assembly is shown in Figure 4.2-15.

Neutron source assemblies are employed at opposite sides of the core. The assemblies are inserted into the rod cluster control guide thimbles in fuel assemblies at selected unrodded locations.

As shown in Figure 4.2-15, the source assemblies contain a holddown assembly similar to that of the burnable poison assembly.

The primary and secondary source rods utilize the same cladding material as the absorber control rods. The secondary source rods contain Sb-Be pellets stacked to a height of approximately 88 inches. The primary source rods contain capsules of californium (Pu-Be possible alternate) source material and alumina spacer pellets to position the source material within the cladding. The rods in each assembly are permanently fastened at the top end to a holddown assembly.

The other structural members are constructed of Type 304 stainless steel except for the springs. The springs exposed to the reactor coolant are Inconel 718.

4.2.2.3.4 Thimble Plug Assembly

In order to limit bypass flow through the rod cluster control guide thimbles in fuel assemblies which do not contain either control rods, source rods, or burnable poison rods, the fuel assemblies are fitted with thimble plug assemblies at those locations.

A typical thimble plug assembly is shown in Figure 4.2-16 and consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The 24 short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow.

Each thimble plug is permanently attached to the base plate by a nut which is secured to the threaded end of the plug. Similar short rods are also used on the source assemblies and burnable poison assemblies to plug the ends of all vacant fuel assembly guide thimbles. At installation in core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adaptor plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place.

All components in the thimble plug assembly, except for the springs, are constructed from Type 304 stainless steel. The springs are Inconel-718.

4.2.3 DESIGN EVALUATION

The fuel assemblies, fuel rods and incore control components are designed to satisfy the performance and safety criteria of Section 4.2, the mechanical design bases of Section 4.2.1, and other interfacing nuclear and thermal-hydraulic design bases specified in Sections 4.3 and 4.4. Effects of Conditions II, III, IV or anticipated transients without trip on fuel integrity are presented in Chapter 15 or the supporting topical reports.

A representative Fuel System Design Evaluation for fuel and components supplied by Westinghouse is provided in this section.

The initial step in fuel rod design evaluation for a region of fuel is to determine the limiting rod(s). Limiting rods are defined as those rod(s) whose predicted performance provides the minimum margin to each of the design criteria. For a number of design criteria the limiting rod is the lead burnup rod of a fuel region. In other instances it may be the maximum power or the minimum burnup rod. For the most part, no single rod will be limiting with respect to all design criteria.

After identifying the limiting rod(s), a worst-case evaluation is made which utilizes the limiting rod design basis power history and considers the effects of model uncertainties and dimensional variations. Furthermore, to verify adherence to the design criteria, the conservative case

evaluation also considers the effects of postulated transient power increases which are achievable during operation consistent with Conditions I and II. These transient power increases can affect both rod average and local power levels. The analytical methods used in the evaluation result in performance parameters which demonstrate the fuel rod behavior. Examples of parameters considered include rod internal pressure, fuel temperature, clad stress, and clad strain. In fuel rod design analyses these performance parameters provide the basis for comparison between expected fuel rod behavior and the corresponding design criteria limits.

Fuel rod and assembly models used for the various evaluations are documented and maintained under an appropriate control system. Materials properties used in the design evaluations are given in Reference [2] and [9].

4.2.3.1 Cladding

1. Vibration and wear

Fuel rod vibrations are flow induced. The effect of the vibration on the fuel assembly and individual fuel rods is minimal. The cyclic stress range associated with deflections of such small magnitude is insignificant and has no effect on the structural integrity of the fuel rod.

The reaction force on the grid supports due to rod vibration motions is also small and is much less than the spring preload. Firm fuel clad spring contact is maintained. No significant wear of the clad or grid supports is expected during the life of the fuel assembly based on out-of-pile flow tests performance of similarly designed fuel in operating reactors, and design analysis.

Clad fretting and fuel vibration have been experimentally investigated as shown in Reference 20 for the Westinghouse optimized fuel.

2. Fuel rod internal pressure and cladding stresses

The burnup dependent fission gas release model (Reference 5) is used in determining the internal gas pressures as a function of irradiation time. The plenum height of the fuel rod has been designed to ensure that the maximum internal pressure of the fuel rod will not exceed the value which would cause the fuel-clad diametral gap to increase during steady state operation.

The clad stresses at a constant local fuel rod power are low. Compressive stresses are created by the pressure differential between the coolant pressure and the rod internal gas pressure. Because of the pre-pressurization with helium, the volume average effective stresses are always less than approximately 10,000 psi at the pressurization level used in this fuel rod design. Stresses due to the temperature gradient are not included in this average effective stress because thermal stresses are, in general, negative at the clad inside diameter and positive at the clad outside diameter and their contribution to the clad volume average stress is small. Furthermore, the thermal stress decreases with time during steady state operation due to stress relaxation. The stress due to pressure differential is highest in the minimum power rod at the beginning-of-life due to low internal gas pressure and the thermal stress is highest in the maximum power rod due to steep

temperature gradient.

Tensile stresses could be created once the clad has come in contact with the pellet. These stresses would be induced by the fuel pellet swelling during irradiation. Fuel swelling can result in small clad strains (< 1 percent) for expected discharge burnups but the associated clad stresses are very low because of clad creep (thermal and irradiation-induced creep). Furthermore, the 1 percent strain criterion is extremely conservative for fuel-swelling driven clad strain because the strain rate associated with solid fission products swelling is very slow.

3. Materials and chemical evaluation

Zircaloy-4 and ZIRLO clad have a high corrosion resistance to the coolant, fuel and fission products. As shown in References [1] and [9], there is considerable PWR operating experience on the capability of Zircaloy-4 and ZIRLO as clad material. Controls on fuel fabrication specify maximum moisture levels to preclude clad hydriding.

Metallographic examination of irradiated commercial fuel rods has shown occurrences of fuel/clad chemical interaction. Reaction layers of < 1 mil in thickness have been observed between fuel and clad at limited points around the circumference. Metallographic data indicates that this interface layer remains very thin even at high burnup. Thus, there is no indication of propagation of the layer and eventual clad penetration.

4. Fretting

Cladding fretting has been experimentally investigated as shown in Reference [9 and 20]. No significant fretting of the cladding is expected during the life of the fuel assembly.

5. Stress Corrosion

Stress corrosion cracking is another postulated phenomenon related to fuel/clad chemical interaction. Out of pile tests have shown that in the presence of high clad tensile stresses, large concentrations of iodine can chemically attack the Zircaloy-4 and ZIRLO tubing and can lead to eventual clad cracking. Extensive post irradation examination has produced no in pile evidence that this mechanism is operative in commercial fuel.

6. Cycling and Fatigue

A comprehensive review of the available strain fatigue models was conducted by Westinghouse as early as 1968. This review include the Langer-O'Donnell model (Ref. [13], the Yao-Munse model and the Manson-Halford model. Upon completion of this review and using the results of the Westinghouse experimental programs, it was concluded that the approach defined by Langer-O'Donnell would be retained and the emperical factors of their correlation bound the results of the Westinghouse testing program.

7. Rod bowing

Reference [12] presents the NRC approved model used for the evaluation of fuel rod bowing. The effects of rod bowing on DNBR are described in Section 4.4.2.1.3.

8. Consequences of power-coolant mismatch

This subject is discussed in Chapter 15.

9. Irradiation Stability of the Cladding

As shown in Reference [1], there is considerable PWR operating experience on the capability of Zircaloy as a cladding material. Extensive experience with irradiated Zircaloy-4 is summarized in Reference [2], and in the Appendices A-E in Reference [9] for ZIRLO.

10. Creep collapse and creepdown

This subject and the associated irradiation stability of cladding have been evaluated using the models described in Reference [6] and [29].

It has been established that the design basis of no clad collapse during planned core life can be satisfied by limiting fuel densification and by having a sufficiently high initial internal rod pressure.

4.2.3.2 Fuel Materials Consideration

Sintered, high density uranium dioxide fuel reacts only slightly with the clad, at core operating temperatures and pressures. In the event of clad defects, the high resistance of uranium dioxide to attack by water protects against fuel deterioration although limited fuel erosion can occur. As has been shown by operating experience and extensive experimental work, the thermal design parameters conservatively account for changes in the thermal performance of the fuel elements due to pellet fracture which may occur during power operation. The consequences of defects in the clad are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. Observations have shown that fuel pellets can densify under irradiation to a density higher than the manufactured values. Fuel densification and subsequent settling of the fuel pellets can result in local and distributed gaps in the fuel rods. Fuel densification has been minimized by improvements in the fuel manufacturing process and by specifying a nominal 95 percent initial fuel density.

The evaluation of fuel densification effects and their consideration in fuel design are described in References [4], [5] and [10]. The treatment of fuel swelling and fission gas release are described in References [5] and [10].

Waterlogging considerations on fuel behavior are discussed in Section 4.2.3.3.

4.2.3.3 Fuel Rod Performance

In calculating the steady state performance of a nuclear fuel rod, the following interacting factors are considered:

1. Clad creep and elastic deflection.

2. Pellet density changes, thermal expansion, gas release, and thermal properties as a function of temperature and fuel burnup.

3. Internal pressure as a function of fission gas release, rod geometry, and temperature distribution.

These effects are evaluated using a fuel rod design model [5].

The model modifications for time dependent fuel densification are given in References [4] and [5]. With these interacting factors considered, the model determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and clad temperatures, and clad deflections are calculated. The fuel rod is divided into several axial sections and radially into a number of annular zones. Fuel density changes are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure.

The initial rod internal pressure is selected to delay fuel/clad mechanical interaction and to avoid the potential for flattened rod formation. It is limited, however, by the design criteria for the rod internal pressure, Section 4.2.1.3.

The gap conductance between the pellet surface and the clad inner diameter is calculated as a function of the composition, temperature, and pressure of the gas mixture, and the gap size of contact pressure between clad and pellet. After computing the fuel temperature for each pellet annular zone, the fractional fission gas release is assessed using an empirical model derived from experimental data [5]. The total amount of gas released is based on the average fractional release within each axial and radial zone and the gas generation rate which in turn is a function of burnup. Finally, the gas released is summed over all zones and the pressure is calculated.

The code shows good agreement in fit for a variety of published and proprietary data on fission gas release, fuel temperatures and clad deflections (Refs. [5] and [10]). Included in this spectrum are variations in power, time, fuel density, and geometry.

1. Fuel-cladding mechanical interaction

   One factor in fuel element duty is potential mechanical interaction of fuel and clad. This fuel/clad interaction produces cyclic stresses and strains in the clad, and these in turn consume clad fatigue life. The reduction of fuel/clad interaction is therefore a goal of design. In order to achieve this goal and to enhance the cyclic operational capability of the fuel rod, the technology for using pre-pressurized fuel rods in Westinghouse PWRs has been developed.

   Initially the gap between the fuel and clad is sufficient to prevent hard contact between the two. However, during power operation a gradual compressive creep of the clad onto the fuel pellet occurs due to the external pressure exerted on the rod by the coolant. Clad compressive creep eventually results in the fuel/clad contact. During this period of fuel/clad contact, changes in power level could result in changes in clad stresses and strains. By using pre-pressurized fuel rods to partially offset the effect of the coolant external pressure, the rate of clad creep toward the surface of the fuel is reduced. Fuel rod pre-pressurization delays the time at which fuel/clad interaction and contact occur and hence significantly reduces the number and extent of cyclic stresses and strains

experienced by the clad both before and after fuel/clad contact. These factors result in an increase in the fatigue life margin of the clad and lead to greater clad reliability. If gaps should form in the fuel stacks, clad flattening will be prevented by the rod pre-pressurization so that the flattening time will be greater than the fuel core life.

A two dimensional (r, θ) finite element model has been established to investigate the effects of radial pellet cracks on stress concentrations in the clad. Stress concentration, herein, is defined as the difference between the maximum clad stress in the θ direction and the mean clad stress. The first case has the fuel and clad in mechanical equilibrium and as a result the stress in the clad is close to zero. In subsequent cases the pellet power is increased in steps and the resultant fuel thermal expansion imposes tensile stress in the clad. In addition to uniform clad stresses, stress concentrations develop in the clad adjacent to radial cracks in the pellet. These radial cracks have a tendency to open during a power increase but the frictional forces between fuel and clad oppose the opening of these cracks and result in localized increases in clad stress. As the power is further increased, large tensile stresses exceed the ultimate tensile strength of UO_2 and additional cracks in the fuel are created which limits the magnitude of the stress concentration in the clad.

As part of the standard fuel rod design analysis, the maximum stress concentration evaluated from finite element calculations is added to the volume averaged effective stress in the clad as determined from one dimensional stress/strain calculations. The resultant clad stress is then compared to the temperature dependent Zircaloy/ZIRLO yield stress in order to assure that the stress/strain criteria are satisfied.

Transient Evaluation Method

Pellet thermal expansion due to power increases is considered the only mechanism by which significant stresses and strains can be imposed on the clad. Power increases in commercial reactors can result from fuel shuffling (e.g., Region 3 positioned near the center of the core for Cycle 2 operation after operating near the periphery during Cycle 1), reactor power escalation following extended reduced power operation, and full length control rod movement. In the mechanical design model, lead rods are depleted using best estimate power histories as determined by core physics calculations. During the depletion, the amount of diametral gap closure is evaluated based upon the pellet expansion-cracking model, clad creep model, and fuel swelling model. At various times during depletion, the power is increased locally on the rod to the burnup dependent attainable power density as determined by core physics calculations. The radial, tangential and axial clad stresses resulting from the power increase are combined into a volume average effective clad stress.

2. Irradiation experience

Westinghouse fuel operational experience is presented in Reference [1].

3. Fuel and cladding temperature

The methods used for evaluation of fuel rod temperatures are presented in Section 4.4.2.11.

4. Waterlogging

Local cladding deformations typical of water-logging[b] bursts have never been observed in commercial Westinghouse fuel. Experience has shown that the small number of rods which have acquired clad defects, regardless of primary mechanism, remain intact and do not progressively distort or restrict coolant flow. In fact such small defects are normally observed through reductions in coolant activity to be progressively closed upon further operation due to the buildup of zirconium oxide and other substances. Secondary failures which have been observed in defected rods are attributed to hydrogen embrittlement of the cladding. Post-irradiation examinations point to the hydriding failure mechanism rather than a waterlogging mechanism; the secondary failures occur as axial cracks in the cladding and are similar regardless of the primary failure mechanism. Such cracks do not result in flow blockage. Hence, the presence of such fuel, the quantity of which must be maintained below Technical Specification limits, does not in any way exacerbate the effects of any postulated transients.

Zircaloy clad fuel rods which have failed due to waterlogging (Refs. [14] and [15] indicate that very rapid power transients are required for fuel failure. Normal operational transients are limited to about 40 cal/gm-min. (peak rod), while the Spert tests [14] indicate that 120 cal/gm to 150 cal/gm is required to rupture the clad even with very short transients (5.5 msec period).

Local cladding deformations typical of waterlogging bursts have never been observed in commercial Westinghouse fuel. Secondary failures which have been observed in defected rods are attributed to hydrogen embrittlement of the cladding. Post-irradiation examinations point to the hydriding failure mechanism rather than a waterlogging mechanism; the secondary failures occur as axial cracks in the cladding and are similar regardless of the primary failure mechanism. Such cracks, of course, do not result in flow blockage. Therefore, waterlogging is not considered to be a concern and operation with PCI defects poses no problem as long as the coolant activity remains below established levels.

5. Potentially damaging temperature effects during transients

The fuel rod experiences many operational transients (intentional maneuvers) during its residence in the core. A number of thermal effects must be considered when analyzing the fuel rod performance.

The clad can be in contact with the fuel pellet at some time in the fuel lifetime. Clad-pellet interaction occurs if the fuel pellet temperature is increased after the clad is in contact with the pellet. Clad-pellet interaction is discussed in Section 4.2.3.3.

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b. Waterlogging damage of a previously defected fuel rod has occasionally been postulated as a mechanism for subsequent rupture of the cladding. Such damage has been postulated as a consequence of a power increase on a rod after water has entered such a rod through a clad defect of appropriate size. Rupture is postulated upon power increase if the rod internal pressure increase is excessive due to insufficient venting of water to the reactor coolant.

The potential effects of operation with waterlogged fuel are discussed in Section 4.2.3.3 which concluded that waterlogging is not a concern during operational transients.

Clad flattening, as shown in Reference [6], has been observed in some operating power reactors. Thermal expansion (axial) of the fuel rod stack against a flattened section of clad could cause failure of the clad. This is no longer a concern because clad flattening is precluded during the fuel residence in the core (see Section 4.2.3.1).

Potential differential thermal expansion between the fuel rods and the guide thimbles during a transient is considered in the design. Excessive bowing of the fuel rods is precluded because the grid assemblies allow axial movement of the fuel rods relative to the grids. Specifically thermal expansion of the fuel rods is considered in the grid design so that axial loads imposed on the fuel rods during a thermal transient will not result in excessively bowed fuel rods.

6. Fuel element burnout and potential energy release

As discussed in Section 4.4.2.2, the core is protected from DNB over the full range of possible operating conditions. In the extremely unlikely event that DNB should occur, the clad temperature will rise due to the steam blanketing at the rod surface and the consequent degradation in heat transfer. During this time there is a potential for chemical reaction between the cladding and the coolant. However, because of the relatively good film boiling heat transfer following DNB, the energy release resulting from this reaction is insignificant compared to the power produced by the fuel.

7. Coolant flow blockage effects on fuel rods

This evaluation is presented in Section 4.4.4.7.

4.2.3.4 Spacer Grids

The coolant flow channels are established and maintained by the structure composed of grids and guide thimbles. The lateral spacing between fuel rods is provided and controlled by the support dimples of adjacent grid cells. Contact of the fuel rods on the dimples is maintained through the clamping force of the grid springs. Lateral motion of the fuel rods is opposed by the spring force and the internal moments generated between the spring and the support dimples. Grid testing is discussed in References 16 and 20.

As shown in References 16, 20, and 21, grid crushing tests and seismic and LOCA evaluations show that the grids will maintain a geometry that is capable of being cooled under the worst-case accident Condition IV event.

4.2.3.5 Fuel Assembly

4.2.3.5.1 Stresses and Deflections

The fuel assembly component stress levels are limited by the design. For example, stresses in the fuel rod due to thermal expansion and Zircaloy/ZIRLO irradiation growth are limited by the relative motion of the rod as it slips over the grid spring and dimple surfaces. Clearances between the fuel rod ends and nozzles are provided so that Zircaloy/ZIRLO irradiation growth

does not result in rod end interferences. Stresses in the fuel assembly caused by tripping of the rod cluster control assembly have little influence on fatigue because of the small number of events during the life of an assembly. Assembly components and prototype fuel assemblies made from production parts have been subjected to structural tests to verify that the design bases requirements are met.

The fuel assembly design loads for shipping have been established at 4 g's axially and 6 g's laterally and transverse for the Westinghouse optimized fuel. Accelerometers are permanently placed into the shipping cask to monitor and detect fuel assembly accelerations that would exceed the criteria. Past history and experience has indicated that loads which exceed the allowable limits rarely occur. Exceeding the limits requires reinspection of the fuel assembly for damage. Tests on various fuel assembly components such as the grid assembly, sleeves, inserts and structure joints have been performed to assure that the shipping design limits do not result in impairment of fuel assembly function. Seismic analysis of the fuel assembly is presented in Reference [16].

4.2.3.5.2 Dimensional Stability

A prototype fuel assembly has been subjected to column loads in excess of those expected in normal service and faulted conditions [16, 20].

No interference with control rod insertion into thimble tubes will occur during a postulated loss of coolant accident transient due to fuel rod swelling, thermal expansion, or bowing. In the early phase of the transient following the coolant break, the high axial loads, which could be generated by the difference in thermal expansion between fuel clad and thimbles, are relieved by slippage of the fuel rods through the grids. The relatively low drag force restraint on the fuel rods will induce only minor thermal bowing, which is insufficient to close the fuel rod-to-thimble tube gap.

References [16] and [20] shows that the fuel assemblies will maintain a geometry amenable to cooling during a combined seismic and double ended LOCA.

4.2.3.6 Incore Control Components

The components are analyzed for loads corresponding to normal, upset, emergency and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control rod trip (equivalent static load).

2. Differential pressure.

3. Spring preloads.

4. Coolant flow forces (static).

5. Temperature gradients.

6. Differences in thermal expansion

   a. Due to temperature differences.

   b. Due to expansion of different materials.

7. Interference between components.

8. Vibration (mechanically or hydraulically induced).

9. Operational transients.

10. Pump overspeed.

11. Seismic loads (Operating Basis Earthquake and Safe Shutdown Earthquake).

12. Blowdown forces (due to cold or hot leg break).

13. Material swelling and gas generation pressure.

The main objective of the analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors.

Sufficient diametral and end clearances have been provided in the neutron absorber, burnable poison, and source rods to accommodate the relative thermal expansions and material swelling between the enclosed material and the surrounding clad and end plugs. There is no bending or warping induced in the rods although the clearance offered by the guide thimble would permit a postulated warpage to occur if there were no restraint on the rods. Bending, therefore, is not considered in the analysis of the rods.

Experience with incore control components is discussed in Reference [1]. Materials data and evaluations are given in References [2] and [23].

1. Mechanical strength evaluation

   The design of incore component rods provides a sufficient cold void volume within the burnable poison and source rods to limit the internal pressures to a value which satisfies the criteria in Section 4.2.1.6. A gas plenum at the top of source and burnable poison rods provides void volume for the pressure buildup. This is not a concern for the all Ag-In-Cd control rod designs since no gases are generated by this absorber material. The void volume for the helium in the burnable poison rods is also obtained through the use of glass in tubular form which provides a central void along the length of the rods. Helium gas is not released by the Ag-In-Cd neutron absorber rod material. The internal pressure of source rods continues to increase from ambient until end of life. The stress

analysis of reactivity component rods assumes 100 percent gas release to the rod void volume and satisfies the criteria in Section 4.2.1.6.

Based on available data for properties of the borosilicate glass and on nuclear and thermal calculations for the standard burnable poison rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube could occur but would continue only until the glass came in contact with the inner liner. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping and to collapse locally before rupture of the exterior cladding if unexpected large volume changes due to swelling or cracking should occur. The top of the inner liner is open to allow communication to the central void by helium which diffuses out of the glass.

The designs of the Wet Annular Burnable Absorber (WABA) provide a sufficient cold void volume to accommodate the internal pressure increase during operation. An annular void volume is provided between the two tubes at the top and along the length of the rods. The stress analysis of the WABA rods assumes a 30% maximum gas release (Reference 19). During normal transient and accident conditions, the void volume limits the internal pressures to values which satisfy the subsection 4.2.1.6.

Analysis on the full length rod cluster control spider indicates the spider is structurally adequate to withstand the various operating loads including the higher loads which occur during the drive mechanism stepping action and rod drop. Experimental verification of the spider structural capability has been completed.

2. Thermal evaluation

The radial and axial temperature profiles have been determined by considering gap conductance, thermal expansion, and neutron or gamma heating of the contained material as well as gamma heating of the clad. The maximum neutron absorber material temperature was found to be less than 850ºF for the Ag-In-Cd which occur axially at only the highest flux region. The maximum standard poison assembly borosilicate glass temperature was calculated to be about 1200ºF and less than 1200ºF for WABA's. This takes place following the initial rise to power. The glass temperature then decreases rapidly for the following reasons: 1) reduction in power generation due to B10 depletion; 2) better gap conductance as the helium produced diffuses to the gap; and 3) external gap reduction (standard poison assembly only) due to boronsilicate glass creep. Rod, guide thimble, and dashpot flow analysis indicates that the flow is sufficient to prevent coolant boiling and maintain clad temperatures at which the clad material has adequate strength to resist coolant operating pressures and rod internal pressures.

3. Irradiation and chemical evaluation

The materials selected are considered to be the best available from the standpoint of resistance to irradiation damage and compatibility to the reactor environment. The materials selected partially dictate the reactor environment (e.g., C1- control in the coolant). The irradiation stability of the absorber material is discussed in Reference [2 and 19]. Irradiation produces no deleterious effects on Ag-In-Cd neutron absorbers.

At high fluences the austenitic materials increase in strength with a corresponding decreased ductility (as measured by tensile tests) but energy absorption (as measured by impact tests) remain quite high. Corrosion of the materials exposed to the coolant is quite low and proper chemical control in the coolant will prevent the occurrence of stress corrosion. All of the austenitic stainless steel base materials used are processed and fabricated by furnace brazing, the procedure used requires that the pieces be rapidly cooled so that the time-at-temperature is minimized. The time that is spent by the control rod spiders in the sensitization range, 800-1500ºF, during fabrication is controlled to preclude sensitization. The 17-4 pH parts are all aged at the highest standard aging temperature of 1100ºF to avoid stress corrosion problems exhibited by aging at lower temperatures.

Based on the preceding considerations, it is judged that the potential for interference with rod cluster control movement due to possible corrosion phenomena is very low. Additional information on irradiation and chemical effects is given in Reference [2].

4. Failure evaluations

Analysis of the full length rod cluster control assemblies show that if the drive mechanism housing ruptures, the rod cluster control assembly will be ejected from the core by the pressure across the drive rod assembly. The ejection is also predicted on the failure of the drive mechanism to retain the drive rod/rod cluster control assembly position.

It should be emphasized that a drive mechanism housing rupture will cause the ejection of only one rod cluster control assembly with the other assemblies remaining in the core. Analysis shows that a pressure drop in excess of 4000 psi must occur across a two-fingered vane to break the vane/spider body joint causing ejection of two neutron absorber rods from the core. Since the greatest pressure drop in the system is only 2250 psi, a pressure drop in excess of 4000 psi is not credible. Thus, the ejection of the neutron absorber rods is not possible.

Ejection of a burnable poison or thimble plug assembly is conceivable based on the postulation that the hold down bar fails and that the base plate and burnable poison rods are severely deformed. In the unlikely event that failure of the hold down bar occurs, the upward displacement of the burnable poison assembly only permits the base plate to contact the upper core plate. Since this displacement is small, the major portion of the borosilicate glass tubing remains positioned within the core. In the case of the thimble plug assembly, the thimble plugs will partially remain in the fuel assembly guide thimbles thus maintaining a majority of the desired flow impedance. Further displacement or complete ejection would necessitate the square base plate and burnable poison rods be forced, thus plastically deformed, to fit up through a smaller diameter hole. It is expected that this condition requires a substantially higher force or pressure drop than that of the hold down bar failure.

The mechanical design of the reactivity control components provides for the protection of the active elements to prevent the loss of control capability and functional failure of critical components. The components have been reviewed for potential failure and consequences of a functional failure of critical parts. The results of the review are summarized below.

Full Length Rod Cluster Control Assembly

a. The basic absorbing materials are sealed from contact with the primary coolant and the fuel assembly and guidance surfaces by a high quality chromium plated stainless steel clad. Potential loss of absorber mass or reduction in reactivity control material due to mechanical or chemical erosion or wear is, therefore, reliably prevented.

b. The silver-indium-cadmium absorber material is relatively inert and would still remain remote from high coolant velocity regions for Conditions I, II and III. For Condition IV situations, refer to Reference 30 for assessment of control rod survivability.

c. The individually clad absorber rods are doubly secured to the retaining spider vane by a threaded joint and a welded lock pin. This joint has been qualified by functional testing and actual service in operating plants.

   It should also be noted that in several instances of control rod jamming caused by foreign particles, the individual rods at the site of the jam have borne the full capacity of the control rod drive mechanism and higher impact loads to dislodge the jam without failure. The conclusion to be drawn from this experience is that this joint is extremely insensitive to potential mechanical damage. A failure of the joint would result in the insertion of the individual rod into the core.

d. The spider finger braze joint by which the individual rods are fastened to the vanes has also experienced the service described above and been subjected to the same jam freeing procedures also without failure. A failure of this joint would also result in insertion of the individual rod into the core.

e. The radial vanes are attached to the spider body by a welded and brazed joint. The joints are designed to a theoretical strength in excess of that of the components joined.

   It is a feature of the design that the guidance of the rod cluster control is accomplished by the inner fingers of these vanes. They are, therefore, the most susceptible to mechanical damage. Since these vanes carry two rods, failure of the vane-to-hub joint does not prevent the free insertion of the rod pair. Neither does such a failure interfere with the continuous free operation of the drive line.

   Failure of the vane-to-hub joint of a single rod vane could potentially result in failure of the separated vane and rod to insert. This could occur only at withdrawal elevations where the spider is above the continuous guidance section of the guide tube (in the upper internals). A rotation of the disconnected vane could cause it to hang on one of the guide cards in the intermediate guide tube. Such an occurrence would be evident from the failure of the rod cluster control to insert below a certain elevation but with free motion above this point.

   This possibility is considered extremely remote because the single rod vanes are subjected to only vertical loads and very light lateral reactions from the rods. The consequences of such a failure are not considered critical since only one drive line

of the reactivity control system would be involved. This condition is readily observed and can be corrected at shutdown.

f. The spider hub being of single unit cylindrical construction is very rugged and of extremely low potential for damage. Rod cluster control assembly spider tests have verified the structural adequacy of the design including the vane-to-hub joint and spring pack load deflection tests (see Section 4.2.4.3). It is difficult to postulate condition to cause failure. Should some unforeseen event cause fracture of the hub above the vanes, the lower portion with the vanes and rods attached would insert by gravity into the core causing a reactivity decrease. The rod could then not be removed by the drive line. Fracture below the vanes cannot be postulated since all loads, including scram impact, are taken above the vane elevation.

g. The guide thimbles of the fuel assemblies provide a clear channel for insertion of the rod cluster control rods. Fuel rod failure due to postulated control rod contact is prevented by providing this physical barrier between the fuel rod and the intended insertion channel. Distortion of the fuel rods by bending cannot apply sufficient force to damage or significantly distort the guide thimble. Fuel rod distortion by swelling, though precluded by design, would be terminated by fracture before contact with the guide thimble occurs. If such were not the case, it would be expected that a force reaction at the point of contact would cause a slight deflection of the guide thimble. The radius of curvature of the deflected shape of the guide thimbles would be sufficiently large to have a negligible influence on rod cluster control insertion.

PWR operating experience has shown that fretting wear has occurred between control rods and guide thimble tubes. A report describing extensive post irradiation examinations of control rod guide thimble tubes in both reactor site and in hot cells [24], along with additional Westinghouse data [25], shows that the guide thimble wear is not a safety concern in Westinghouse facilities.

Burnable Poison Assemblies

The burnable poison assemblies are static temporary reactivity control elements. The axial position is assured by the hold down assembly which bears against the upper core plate. Their lateral position is maintained by the guide thimbles of the fuel assemblies.

The individual rods are shouldered against the underside of the retainer plate and securely fastened at the top by a threaded nut which is then locked in place. The square dimension of the retainer plate is larger than the diameter of the flow holes through the core plate. Failure of the hold down bar or spring pack, therefore, does not result in ejection of the burnable poison rods from the core.

The only incident that could potentially result in ejection of the burnable poison rods is a multiple fracture of the retainer plate. This is not considered credible because of the light loads borne by this component. During normal operation the loads borne by the plate are distributed at the points of attachment. Even a multiple fracture of the retainer plate would result in jamming of the plate

segments against the upper core plate, again preventing ejection. Excessive reactivity increase due to burnable poison ejection is, therefore, prevented.

4.2.4 TESTING AND INSPECTION PLAN

4.2.4.1 Quality Assurance Program

The Quality Assurance Program Plan of the Westinghouse Nuclear Fuel Division, as summarized in References [17] and [22], has been developed to serve the division in planning and monitoring its activities for the design and manufacture of nuclear fuel assemblies and associated components.

This program provides for control over all activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage, and transportation. This program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

4.2.4.2 Quality Control

Quality control philosophy is generally based on the following inspections being performed to a 95 percent confidence that at least 95 percent of the product meets specification, unless otherwise noted.

1. Fuel system components and parts

   The characteristics inspected depend upon the component parts and includes dimensional, visual, check audits of test reports, material certification and nondestructive examination such as X-ray and ultrasonic.

   All material used in this core is accepted and released by Quality Control.

2. Pellets

   Inspection is performed for dimensional characteristics such as diameter, density, length and squareness of ends. Additional visual inspections are performed for cracks, chips, missing pellet surfaces and surface conditions according to approved standards.

   Pellet densities are determined and compared to specification requirements. Chemical analyses are taken on a specified sample basis throughout pellet production.

3. Rod inspection

   Fuel rod, control rodlet, burnable poison and source rod inspection consists of the following nondestructive examination techniques and methods, as applicable:

   a. Leak testing

   b. Enclosure welds

c. Dimensional Inspections

d. Plenum dimensions

e. Pellet-to-pellet gaps

f. 100 percent of the fuel rods are active gamma scanned to verify enrichment control prior to acceptance for assembly loading.

g. Traceability

4. Assemblies

Each fuel, control rod, burnable poison and source rod assembly is inspected for drawing and/or specification requirements. Other incore control component inspection and specification requirements are given in Section 4.2.4.3.

5. Other inspections

The following inspections are performed as part of the routine inspection operation:

a. Tool and gage inspection and control. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and conditions of tools.

b. Audits are performed of inspection activities and records to assure that prescribed methods are followed and that records are correct and properly maintained.

c. Surveillance inspection where appropriate, and audits of outside contractors are performed to ensure conformance with specified requirements.

6. Process control

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The UO_2 powder is kept in sealed containers which are fully identified by descriptive tagging and preselected color coding or, for the closed system, the material is monitored by a computer data management information system. For the sealed container system, a Westinghouse identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by analysis.

Finished pellets are placed on trays and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by Quality Control. Physical barriers prevent mixing of pellets of different nominal densities and enrichments in this storage area. Unused powder and substandard pellets are returned to storage in identified containers.

Pellets are loaded into fuel cladding tubes on isolated production lines. Each production line contains only rods of one type at any one time. A serialized traceability code is placed on each fuel rod which identifies the contract and enrichment. The end plugs are inserted and welded to seal the tube. The fuel tube remains coded, and traceability identified. The traceability code provide a cross reference of the fuel contained in the fuel rods.

At the time of installation into an assembly, a matrix is generated to identify each rod in its position within a given assembly. After the fuel rods are installed, all fuel rods in an assembly are verified to carry the correct identification character describing the fuel enrichment for the core region being fabricated. The top nozzle is inscribed with a permanent identification number providing traceability to the fuel contained in the assembly.

Similar traceability is provided for burnable poison, source rods and control rodlets as required.

4.2.4.3 Incore Control Component Testing and Inspection

Tests and inspections are performed on each reactivity control component to verify the mechanical characteristics. In the case of the full length rod cluster control assembly, prototype testing has been conducted and both manufacturing test/inspections and functional testing at the plant site are performed.

During the component manufacturing phase, the following requirements apply to the reactivity control components to assure the proper functioning during reactor operation:

1. All materials are procured to specifications to attain the desired standard of quality.

2. The spiders are load-tested to insure that the design load requirements are met.

3. All rods are checked for integrity by the methods described in Section 4.2.4.2.3.

4. To assure proper fitup with the fuel assembly, the rod cluster control, burnable poison and source assemblies are installed in the fuel assembly without restriction or binding in the dry condition. Also a straightness of 0.01 in/ft is required on the entire inserted length of each rod assembly.

The full length rod cluster control assemblies are functionally tested, following each refueling outage prior to criticality to demonstrate reliable operation of the assemblies. Each assembly is operated (and tripped) at full flow/hot conditions.

In order to demonstrate continuous free movement of the full length rod cluster control assemblies, and to ensure acceptable core power distributions during operations, partial movement checks are performed on every full length rod cluster control assembly every 31 days during reactor critical operation. In addition, periodic drop tests of the full length rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical rod cluster control assembly ejection.

If a rod cluster control assembly cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus inability to move one rod cluster control assembly can be tolerated. More than one inoperable rod cluster control assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable rod cluster control assemblies has been limited to one.

4.2.4.4 Tests and Inspections by Others

If any tests and inspections are to be performed on behalf of Westinghouse, the fuel supplier will review and approve the quality control procedures, inspection plans, etc. to be utilized to ensure that they are equivalent to the description provided above and are performed properly to meet all fuel vendor's requirements.

4.2.4.5 In-Service Surveillance

Westinghouse conducts a program to examine detailed aspects of their fuel assembly performance. The Westinghouse program is described in Reference [7]. This document is periodically updated in order to provide recent results of operating experience with Westinghouse fuel and incore control components.

4.2.4.6 Onsite Inspection

Detailed written procedures are used by the station staff for the post shipment inspection of all new fuel and associated components such as control rods, plugs, and inserts. Fuel handling procedures specify the sequence in which handling and inspection takes place.

Loaded fuel containers, when received onsite, are externally inspected to ensure that labels and markings are intact and seals are unbroken. After the containers are opened, the shock indicators attached to the suspended internals are inspected to determine if movement during transit exceeded design limitations.

Following removal of the fuel assembly from the container in accordance with detailed procedures, the fuel assembly polyethylene wrapper is examined for evidence of damage. The polyethylene wrapper is then removed and a visual inspection of the entire bundle is performed.

Control rod assemblies are typically shipped in fuel assemblies and are inspected during fuel receipt operations. The control rod assembly is withdrawn from the fuel assembly to ensure free and unrestricted movement. The exposed section is then visibly inspected for mechanical integrity, replaced in the fuel assembly and stored with the fuel assembly.

4.2.4.7 Post-Irradiation Surveillance

To assure continued fuel integrity, a post-irradiation fuel surveillance program is conducted via visual examination of a sample of post-irradiated fuel assemblies through the use of visual aids, such as binoculars and/or underwater television cameras. During visual inspection, indications of fretting, crud, fuel assembly damage and anomalies are observed. If significant anomalies are encountered that would affect fuel performance, a more detailed examination of those fuel assemblies will be performed.

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4.3 NUCLEAR DESIGN

4.3.1 DESIGN BASES

This section describes the design bases and functional requirements used in the nuclear design of the fuel and reactivity control system and relates these design bases to the General Design Criteria (GDC) in 10CFR50 Appendix A. Where appropriate, supplemental criteria such as the Final Acceptance Criteria for Emergency Core Cooling Systems are discussed. Before discussing the nuclear design bases, it is appropriate to briefly review the four major categories ascribed to conditions of plant operation.

The full spectrum of plant conditions is divided into four categories, in accordance with the anticipated frequency of occurrence and risk to the public:

1. Condition I - Normal Operation.

2. Condition II - Incidents of Moderate Frequency.

3. Condition III - Infrequent Faults.

4. Condition IV - Limiting Faults.

In general the Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Condition II incidents are accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action. Fuel damage (fuel damage as used here is defined as penetration of the fission product barrier i.e. the fuel rod clad) is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the Chemical and Volume Control System (CVCS) and are consistent with the plant design basis.

Condition III incidents shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude immediate resumption of operation. The release of radioactive material due to Condition III incidents should not be sufficient to interrupt or restrict public use of these areas beyond the exclusion radius. Furthermore, a Condition III incident shall not, by itself generate a Condition IV fault or result in a consequential loss of function of the reactor coolant or reactor containment barriers.

Condition IV occurrences are faults that are not expected to occur but are defined as limiting faults which must be designed against. Condition IV faults shall not cause a release of radioactive material that results in exceeding the limits of 10CFR100.

The core design power distribution limits related to fuel integrity are met for Condition I occurrences through conservative design and maintained by the action of the control system. The requirements for Condition II occurrences are met by providing an adequate protection system which monitors reactor parameters. The control and protection systems are described in Chapter 7 and the consequences of Condition II, III and IV occurrences are given in Chapter 15.

4.3.1.1 Fuel Burnup

Basis

The fuel rod design basis is described in Section 4.2. The nuclear design basis is to install sufficient reactivity in the fuel to attain the design region discharge burnup. The above along with the design basis in Section 4.3.1.3 satisfies GDC-10.

Discussion

Fuel burnup is a measure of fuel depletion which represents the integrated energy output of the fuel (MWD/MTU) and is a convenient means for quantifying fuel exposure criteria.

The core design lifetime or design discharge burnup is achieved by installing sufficient initial excess reactivity in each fuel region and by following a fuel replacement program (such as that described in Section 4.3.2) that meets all safety-related criteria in each cycle of operation.

Initial excess reactivity installed in the fuel, although not a design basis, must be sufficient to maintain core criticality at full power operating conditions throughout cycle life with equilibrium xenon, samarium, and other fission products present. The end of design cycle life is defined to occur when the chemical shim concentration is essentially zero.

A limitation on initial installed excess reactivity is not required other than as is quantified in terms of other design bases such as core negative reactivity feedback and shutdown margin discussed below.

4.3.1.2 Negative Reactivity Feedbacks (Reactivity Coefficient)

Basis

The fuel temperature coefficient will be negative and the moderator temperature coefficient of reactivity will be non-positive for full power operating conditions, thereby providing negative reactivity feedback characteristics. The design basis meets GDC-11.

Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption effects (Doppler) associated with changing fuel temperature and the spectrum effect resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium ensures that the Doppler coefficient of reactivity is negative. This coefficient provides the most rapid reactivity compensation. The core is also designed to have an overall negative moderator temperature coefficient of reactivity at most operating conditions so that average coolant temperature or void content provides another, slower compensatory effect. Full power operation is permitted only in a range of overall non-positive moderator temperature coefficient. The non-positive moderator temperature coefficient can be achieved through use of fixed burnable absorber rods, and/or integral fuel burnable absorber (IFBA), and/or control rods by limiting the reactivity held down by soluble boron.

Burnable absorber content (quantity and distribution) is not stated as a design basis other than as it relates to accomplishment of a non-positive moderator temperature coefficient at power operating conditions discussed above.

4.3.1.3 Control of Power Distribution

Basis

The nuclear design basis is that:

1. The fuel will not be operated at greater than 14.51 kW/ft under normal operating conditions, including an allowance of 0.6% for calorimetric uncertainty.

2. With at least a 95 percent confidence level, there at least a 95% probability that, under abnormal conditions including the maximum overpower condition, the fuel peak power will not cause melting as defined in Section 4.4.1.2.

3. With at least a 95 percent confidence level, there is at least a 95% probability that the fuel will not operate with a power distribution that violates the departure from nucleate boiling (DNB) design basis discussed in Section 4.4.1 under Condition I and II events including the maximum overpower condition.

4. Fuel management will be such as to produce rod powers and burnups consistent with the assumptions in the fuel rod mechanical integrity analysis of Section 4.2.

The above basis meets GDC-10.

Discussion

Calculation of extreme power shapes which affect fuel design limits is performed with approved methods. The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state.

Even though there is good agreement between measured peak power calculations and measurements, a nuclear uncertainty margin (see Section 4.3.2.2.7) is applied to calculated peak local power. Such a margin is provided both for the analysis for normal operating states and for anticipated transients.

4.3.1.4 Maximum Controlled Reactivity Insertion Rate

Basis

The maximum reactivity insertion rate due to withdrawal of rod cluster control assemblies at power or by boron dilution is limited. A maximum reactivity change rate for accidental withdrawal of control banks is set such that peak heat generation rate and DNBR do not exceed the maximum allowable at overpower conditions. This satisfies GDC-25.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited so as to preclude rupture of the coolant pressure boundary or

disruption of the core internals to a degree which would impair core cooling capacity due to a rod withdrawal or ejection accident (see Chapter 15).

Following any Condition IV event (rod ejection, steam line break, etc.) the reactor can be brought to the shutdown condition and the core will maintain acceptable heat transfer geometry. This satisfies GDC-28.

Discussion

Reactivity addition associated with an accidental withdrawal of a control bank (or banks) is limited by the maximum rod speed (or travel rate) and by the worth of the bank(s). The maximum control rod speed is 45 inches per minute.

The reactivity change rates are conservatively calculated assuming unfavorable axial power and xenon distributions. The peak xenon burnout rate is significantly lower than the maximum reactivity addition rate for normal operation and for accidental withdrawal of two banks.

4.3.1.5 Shutdown Margins

Basis

Minimum shutdown margin as specified in the Core Operating Limits Report is required at any power operating condition, in the hot standby shutdown condition and in the cold shutdown condition.

In all analyses involving reactor trip, the single, highest worth rod cluster control assembly is postulated to remain untripped in its full-out position (stuck rod criterion). This satisfies GDC-26.

Discussion

Two independent reactivity control systems are provided, namely control rods and soluble boron in the coolant. The control rod system can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full-load to no-load. In addition, the control rod system provides the minimum shutdown margin under Condition I events and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits assuming that the highest worth control rod is stuck out upon trip.

The boron system can compensate for all xenon burnout reactivity changes and will maintain the reactor in the cold shutdown. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical shim control system which satisfies GDC-26.

Basis

When fuel assemblies are in the pressure vessel and the vessel head is not in place, k_{eff} will be maintained at or below 0.95 with control rods and soluble boron. Further, the fuel will be maintained sufficiently subcritical so that removal of all rod cluster control assemblies will not result in criticality.

Discussion

ANSI Standard NI8.2 specifies a k_{eff} not to exceed 0.95 in spent fuel storage racks and transfer equipment flooded with pure water and a k_{eff} not to exceed 0.98 in normally dry new fuel storage racks assuming optimum moderation. No criterion is given for the refueling operation; however, a 5 percent margin, which is consistent with spent fuel storage and transfer and the new fuel storage, is adequate for the controlled and continuously monitored operations involved.

The boron concentration required to meet the refueling shutdown criteria is specified in the Technical Specifications. Verification that this shutdown criteria is met, including uncertainties, is achieved using standard design methods. The subcriticality of the core is continuously monitored as described in the Technical Specifications.

4.3.1.6 Stability

Basis

The core will be inherently stable to power oscillations at the fundamental mode. This satisfies GDC-12. Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

Discussion

Oscillations of the total power output of the core, from whatever cause, are readily detected by the loop N-16 power monitors and by the nuclear instrumentation. The core is protected by these systems and a reactor trip would occur if power increased unacceptably, preserving the design margins to fuel design limits. The stability of the turbine/steam generator/core systems and the reactor control system is such that total core power oscillations are not normally possible. The redundancy of the protection circuits ensures an extremely low probability of exceeding design power levels.

The core is designed so that diametral and azimuthal oscillations due to spatial xenon effects are self-damping and no operator action or control action is required to suppress them. The stability to diametral oscillations is so great that this excitation is highly improbable. Convergent azimuthal oscillations can be excited by prohibited motion of individual control rods. Such oscillations are readily observable and alarmed, using the excore long ion chambers. Indications are also continuously available from incore thermocouples and loop power measurements. Moveable incore detectors can be activated to provide more detailed information. In all proposed cores these horizontal plane oscillations are self-damping by virtue of reactivity feedback effects designed into the core.

However, axial xenon spatial power oscillations may occur late in core life. The control banks and excore detectors are provided for control and monitoring of axial power distributions. Assurance that fuel design limits are not exceeded is provided by reactor Overpower N-16 and Overtemperature N-16 trip functions which use the measured axial power imbalance as an input.

4.3.1.7 Anticipated Transients Without Trip

The effects of anticipated transients with failure to trip are not considered in the design bases of the plant. Analysis has shown that the likelihood of such a hypothetical event is negligibly small. Furthermore, analysis of the consequences of a hypothetical failure to trip following anticipated transients has shown that no significant core damage would result, system peak pressures would be limited to acceptable values and no failure of the Reactor Coolant System would result [1]. However, Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) has been incorporated into the CPNPP design in accordance with the requirement of 10CFR50.62. For detailed discussion see FSAR Section 7.8.1.

4.3.2 DESCRIPTION

4.3.2.1 Nuclear Design Description

The reactor core consists of a specified number of fuel rods which are held in bundles by spacer grids and top and bottom fittings. The fuel rods are constructed of Zircaloy or ZIRLO cylindrical tubes containing UO_2 fuel pellets. The bundles, known as fuel assemblies, are arranged in a pattern which approximates a right circular cylinder.

Each fuel assembly normally contains a 17 x 17 rod array composed of 264 fuel rods, 24 rod cluster control thimbles and an incore instrumentation thimble. Further details of the fuel assembly are given in Section 4.2.

The core will normally operate approximately one to two years between refuelings, accumulating approximately 14,000 MWD/MTU per year. The exact reloading pattern, initial and final positions of assemblies, number of fresh assemblies and their placement are dependent on the energy requirement for the next cycle and burnup and power histories of the previous cycles.

The core average enrichment is determined by the amount of fissionable material required to provide the desired core lifetime and energy requirements. The physics of the burnout process is such that operation of the reactor depletes the amount of fuel available due to the absorption of neutrons by the U-235 atoms and their subsequent fission. The rate of U-235 depletion is directly proportional to the power level at which the reactor is operated. In addition, the fission process results in the formation of fission products, some of which readily absorb neutrons. These effects, depletion and the buildup of fission products, are partially offset by the buildup of plutonium which occurs due to the non-fission absorption of neutrons in U-238. Typical plutonium buildup is shown in Figure 4.3-2. Therefore, at the beginning of any cycle a reactivity reserve equal to the depletion of the fissionable fuel and the buildup of fission product poisons over the specified cycle life must be “built” into the reactor. This excess reactivity is controlled by removable neutron absorbing material in the form of boron dissolved in the primary coolant, burnable absorber rods, and/or ZrB_2 coated fuel pellets (IFBA).

The concentration of boric acid in the primary coolant is varied to provide control and to compensate for long term reactivity requirements. The concentration of the soluble neutron absorber is varied to compensate for reactivity changes due to fuel burnup, fission product poisoning including xenon and samarium, burnable absorber depletion, and the cold-to-operating moderator temperature change. Using its normal makeup path, the Chemical and Volume Control System (CVCS) is capable of inserting negative reactivity at a rate of approximately

30 pcm/min when the reactor coolant boron concentration is 1000 ppm and approximately 35 pcm/min when the reactor coolant boron concentration is 100 ppm. If the emergency boration path is used, the CVCS is capable of inserting negative reactivity at a rate of approximately 65 pcm/min when the reactor coolant concentration is 1000 ppm and approximately 75 pcm/min when the reactor coolant boron concentration is 100 ppm. The peak burnout rate for xenon is approximately 25 pcm/min (Section 9.3.4 discusses the capability of the CVCS to counteract xenon decay). Rapid transient reactivity requirements and safety shutdown requirements are met with control rods.

As the boron concentration is increased, the moderator temperature coefficient becomes less negative. The use of a soluble poison alone would result in a positive moderator coefficient at beginning-of-life. Therefore, burnable absorber rods are used to reduce the soluble boron concentration sufficiently to ensure that the moderator temperature coefficient is negative for full power operating conditions. During operation the poison content in these rods is depleted, thus adding positive reactivity to offset some of the negative reactivity from fuel depletion and fission product buildup. The depletion rate of the burnable absorber rods is not critical since chemical shim is always available and flexible enough to cover any possible deviations in the expected burnable absorber depletion rate. Figure 4.3-3 is a graph of a typical core depletion with and without burnable absorber rods. Note that even at end-of-life conditions some residual absorber remains in the burnable absorber rods resulting in a net decrease in the first cycle lifetime. In addition to reactivity control, the burnable absorber rods and/or ZrB_2 coated fuel pellets (IFBA) are strategically located to provide a favorable radial power distribution.

Control rods are located for use in the core to provide control for rapid changes in reactivity. The total reactivity worth of the control rods is dependent on the particular absorber material used and the power distribution is affected by the number, worth, and location of the inserted control rods.

4.3.2.2 Power Distributions

Two methods can be employed for performing core power distribution calculations. The Power Distribution Monitoring System (PDMS) generates a continuous measurement of the core power distribution using the methodology documented in Reference 12. The measured core power distribution is used to determine the most limiting core peaking factors, which are used to verify that the reactor is operating within the design limits.

The PDMS requires information on current plant and core conditions in order to determine the core power distribution using the core peaking factor measurement and measurement uncertainty methodology described in Reference 12. The core and plant condition information is used as input to the continuous core power distribution measurement software that continuously and automatically determines the current core peaking factor values. The core power distribution calculation software provides the measured peaking factor values at nominal one-minute intervals to allow confirmation that the core peaking factors are within design limits. In order for the PDMS to accurately determine the peaking factor values, the core power distribution measurement software requires accurate information about the current reactor power level average reactor vessel inlet temperature, control bank positions, the power range detector currents, and the core exit thermocouples.

Data obtained from the movable neutron flux detectors, described in Section 7.7.1, are used to calibrate the PDMS, and may also be used independent of the PDMS to generate a flux map of

the core power distribution. The accuracy of power distribution calculations has been confirmed through approximately one thousand flux maps during some twenty years of operation under conditions similar to those expected. Details of this confirmation are given in Reference 2 and in Section 4.3.2.2.7.

4.3.2.2.1 Definitions

Power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core and the total energy produced in a coolant channel and are expressed in terms of quantities related to the nuclear or thermal design namely:

Power density is the thermal power produced per unit volume of the core (kW/liter).

Linear power density is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, this is the unit of power density most commonly used. For all practical purposes, it differs from kW/liter by a constant factor which includes geometry and the fraction of the total thermal power which is generated in the fuel rod.

Average linear power density is the total thermal power produced in the fuel rods divided by the total active fuel length of all rods in the core.

Local heat flux is the heat flux at the surface of the cladding ($Btu\text{-}ft^{-2}\text{-}hr^{-1}$). For nominal rod parameters this differs from linear power density by a constant factor.

Rod power or rod integral power is the length integrated linear power density in one rod (kW).

Average rod power is the total thermal power produced in the fuel rods divided by the number of fuel rods (assuming all rods have equal length).

The hot channel factors used in the discussion of power distributions in this section are defined as follows:

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F^N_Q, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod parameters.

F^E_Q, Engineering Heat Flux Hot Channel Factor, is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta}{}^{N}{}_{H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Manufacturing tolerances, hot channel power distribution and surrounding channel power distributions are treated explicitly in the calculation of the DNBR described in Section 4.4.

It is convenient for the purposes of discussion to define subfactors of F_Q, however, design limits are set in terms of the total peaking factor.

F_Q = Total peaking factor of heat flux hot-channel factor

= Maximum KW/ft / Average KW/ft

without densification effects

$$F_Q = F^N_Q \times F^E_Q$$

$$= F^N_{XY} \times F^N_Z \times F^N_U \times F^E_Q$$

where

F^N_Q and F^E_Q are defined above.

F^N_U = factor for conservatism, assumed to be 1.05

F^N_{XY} = ratio of peak power density to average power density in the horizontal plane of peak local power

F^N_Z = ratio of the power per unit core height in the horizontal plane of peak local power to the average value of power per unit core height. If the plane of peak local power coincides with the plane of maximum power per unit core height the F^N_Z is the core average axial peaking factor.

To include the allowances made for densification effects, which are height dependent, the following quantities are defined.

S(Z) = the allowance made for densification effects at height Z in the core (see Section 4.3.2.2.5).

P(Z) = ratio of the power per unit core height in the horizontal plane at height Z to the average value of power per unit core height.

Then

F_Q = Total peaking factor

= Maximum KW/ft / Average KW/ft

Including densification allowance

$$F_Q = \max (F^N_{XY}(Z) \times P(Z) \times S(Z)) \times F^N_{U\,X}\, F^E_Q$$

4.3.2.2.2 Radial Power Distributions

The power shape in horizontal sections of the core at full power is a function of the fuel and burnable absorber loading patterns and the presence or absence of a single bank of full length control rods.

Thus, at any time in the cycle a horizontal section of the core can be characterized as unrodded or with group D control rods. These two situations combined with burnup effects determine the radial power shapes which can exist in the core at full power. The effect on radial power shapes of power level, xenon, samarium and moderator density effects are considered also but these are quite small. The effect of non-uniform flow distribution is negligible. While radial power distributions in various planes of the core are often illustrated, the core radial enthalpy rise distribution as determined by the integral of power up each channel is of greater interest. Figures 4.3-6 through 4.3-11 show typical radial power distributions for one eighth of the core for representative operating conditions. These conditions are: 1) hot full power (HFP) at beginning-of-life (BOL) - unrodded - no xenon, 2) HFP at BOL - unrodded - equilibrium xenon, 3) HFP at BOL - bank D in - equilibrium xenon, 4) HFP near middle-of-life (MOL) - unrodded - equilibrium xenon, 5) HFP at end-of-life (EOL) - unrodded - equilibrium xenon, and 6) HFP at EOL - bank D in - equilibrium xenon.

Since hot channel location varies from time to time, a single reference radial design power distribution is selected for DNB calculations. This reference power distribution, normalized to core average power, is chosen conservatively to concentrate power in one area of the core, minimizing the benefits of flow redistribution.

4.3.2.2.3 Assembly Power Distributions

For the purpose of illustration, typical assembly power distributions from BOL and EOL conditions corresponding to Figures 4.3-7 and 4.3-10 respectively, are given in Figures 4.3-12 and 4.3-13, respectively. The power distributions provided are consistent with the burnable poison rod patterns shown in Figure 4.3-5.

Since the detailed power distribution surrounding the hot channel varies from time to time, a conservatively flat assembly power distribution is assumed in the DNB analysis, described in Section 4.4, with the rod of maximum integrated power artificially raised to the design value of

$F^N\Delta_H$. The nuclear design considers all fuel cycles and all operating conditions to ensure that a flatter assembly power distribution does not occur with limiting values of $F^N\Delta_H$.

4.3.2.2.4 Axial Power Distributions

The shape of the power profile in the axial or vertical direction is largely under the control of the operator either through the manual operation of the full length control rods or automatic motion of full length rods responding to manual operation of the CVCS. Nuclear effects which cause variations in the axial power shape include moderator density, Doppler effect on resonance absorption, spatial xenon, and burnup. Automatically controlled variations in total power output and full length rod motion are also important in determining the axial power shape at any time. Signals are available to the operator from the excore ion chambers which are long ion chambers outside the reactor vessel which contain four active sections running parallel to the axis of the core. Signals from the top two and bottom two sections are averaged to obtain top and bottom signals. The difference between top and bottom signals from each of four multi-section of detectors is displayed on the control panel and called the flux difference, ΔI. Calculations of core average peaking factor for many plants and measurements from operating plants under many operating situations are associated with either ΔI or axial offset in such a way that an upper bound can be placed on the peaking factor. For these correlations axial offset is defined as:

$$\text{axial offset} = \frac{\varphi t - \varphi b}{\varphi t + \varphi b}$$

and φt and φb are the top and bottom detector readings.

Representative axial power shapes for BOL, MOL, and EOL conditions are shown in Figures 4.3-14 through 4.3-16. These figures cover a wide range of axial offset including values not permitted at full power.

The typical radial power distributions shown in Figures 4.3-8 and 4.3-11 involving the insertion of control rods represent a synthesis of power shapes from the rodded and unrodded planes. The applicability of the separability assumption upon which this procedure is based is assured through extensive three-dimensional calculations of possible rodded conditions. As an example, Figure 4.3-17 compares the axial power distribution for several assemblies at different distances from inserted control rods with the core average distribution. The only significant difference from the average occurs in the low power peripheral assemblies, thus confirming the validity of the separability assumption.

4.3.2.2.5 Local Power Peaking

Fuel densification, which has been observed to occur under irradiation in several operating reactors, causes the fuel pellets to shrink both axially and radially. The pellet shrinkage combined with random hangup of fuel pellets results in gaps in the fuel column when the pellets below the hung-up pellet settle in the fuel rod. The gaps vary in length and location in the fuel rod. Because of decreased neutron absorption in the vicinity of the gap, power peaking occurs in the adjacent fuel rods resulting in an increased power peaking factor. A quantitative measure of this local peaking is given by the power spike factor S(Z) where Z is the axial location in the core.

In previous analyses of power peaking factors, it was necessary to apply a penalty on calculated overpower transient F_Q values to allow for interpellet gaps caused by pellet hang-ups and pellet shrinkage due to densification. This penalty is known as the densification spike factor. However, studies have shown (Reference 3) that this penalty can be eliminated for the fuel type present in the Comanche Peak Units 1 and 2 cores.

4.3.2.2.6 Limiting Power Distributions

According to the ANSI classification of plant conditions (see Chapter 15), Condition I occurrences are those which are expected frequently or regularly in the course of power operation, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Condition I operation.

The list of steady state and shutdown conditions, permissible deviations and operational transients is given is Section 15.0. Implicit in the definition of normal operation is proper and timely action by the reactor operator. That is, the operator follows recommended operating procedures for maintaining appropriate power distributions and takes any necessary remedial actions when alerted to do so by the plant instrumentation. Thus, as stated above, the worst or limiting power distribution which can occur during normal operation is to be considered as the starting point for analysis of ANSI Conditions II, III, and IV events.

Improper procedural actions or errors by the operator are assumed in the design as occurrences of moderate frequency (ANSI Condition II). Some of the consequences which might result are discussed in Section 15.0. Therefore, the limiting power shapes which result from such Condition II events are those power shapes which deviate from the normal operating condition at the recommended axial offset band, e.g. due to lack of proper action by the operator during a xenon transient following a change in power level brought about by control rod motion. Power shapes which fall in this category are used for determination of the reactor protection system setpoints so as to maintain margin to overpower or DNB limits.

The means for maintaining power distributions within the required hot channel factor limits are described in the Technical Specifications. A complete discussion of power distribution control in Westinghouse PWRs is included in Reference 6. Detailed background information on the following design constraints on local power density in a Westinghouse PWR, the defined operating procedures, and the measures taken to preclude exceeding design limits are presented in the topical report on power distribution control and load following procedures (Reference 13). The following paragraphs summarize these reports and describe the calculations used to establish the upper bound on peaking factors.

The calculations used to establish the upper bound on peaking factors, F_Q and $F_{\Delta}{}^{N}{}_{H}$, include all of the nuclear effects which influence the radial and/or axial power distributions throughout core life for various modes of operation including load follow, reduced power operation, and axial xenon transients.

The core average axial profile, however, can experience significant changes which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the parameters which are readily observed on the plant. Specifically, the nuclear design parameters which are significant to the axial power distribution analysis are:

1. Core power level.

2. Core height.

3. Coolant temperature and flow.

4. Coolant temperature program as a function of reactor power.

5. Fuel cycle lifetimes.

6. Rod bank worths.

7. Rod bank overlaps.

Normal operation of the plant assumes compliance with the following conditions:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 12 steps (indicated) from the bank demand position;

2. Control banks are sequenced with overlapping banks;

3. The control full length bank insertion limits are not violated;

4. Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures which are followed in normal operation. During normal operation with relaxed axial offset control (RAOC), they require control of the axial flux difference at power levels above 50% RTP within a permissible operating space. The limits on axial flux difference (AFD) are given in the COLR. This minimizes xenon transient effects on the axial power distribution, since the procedures essentially keep the xenon distribution in phase with the power distribution.

Calculations are performed for normal operation of the reactor at various points in the operating cycles. Different operation maneuvers are assumed prior to calculating the effect on the axial power distributions. For a given plant and fuel cycle, a finite number of maneuvers are studied to determine the general behavior of the local power density as a function of core elevation.

These cases represent many possible reactor states in the life of one fuel cycle, and they have been chosen as sufficiently definitive of the cycle by comparison with much more exhaustive studies performed on some 20 or 30 different, but typical, plant and fuel cycle combinations. The cases are described in detail in Reference 7, and they are considered to be necessary and sufficient to generate a local power density limit which, when increased by 5 percent for

conservatism, will not be exceeded with a 95-percent confidence level. Many of the points do not approach the limiting envelope. However, they are part of the time histories which lead to the hundreds of shapes which do define the envelope. They also serve as a check that the reactor studied is typical of those studied more exhaustively.

Thus it is not possible to single out any transient or steady state condition which defines the most limiting case. It is not even possible to separate out a small number which form an adequate analysis. The process of generating a myriad of shapes is essential to the philosophy that leads to the required level of confidence. A maneuver which provides a limiting case for one reactor fuel cycle is not necessarily a limiting case for another reactor or fuel cycle with different control bank worths, enrichments, burnup, coefficients, etc. Each shape depends on the detailed history of operation up to that time and on the manner in which the operator conditioned xenon in the days immediately prior to the time at which the power distribution is calculated.

The calculated points are synthesized from axial calculations combined with radial factors appropriate for rodded and unrodded planes. In these calculations, the effects on the unrodded radial peak of xenon redistribution that occurs following the withdrawal of a control bank (or banks) from a rodded region is obtained from two-dimensional X-Y calculations. A conservative factor is applied on the unrodded radial peak that was obtained from calculations in which xenon distribution was preconditioned by the presence of control rods and then allowed to redistribute for several hours. A detailed discussion of this effect may be found in Reference 7. The calculated values have been increased by a factor of 1.05 for conservatism and a factor of 1.03 for the engineering factor for the overpower (kW/ft) evaluations.

The envelope drawn over the calculated [max (F_Q • Power)] points represents an upper bound envelope on local power density versus elevation in the core. It should be emphasized that this envelope is a conservative representation of the bounding values of local power density. Expected values are considerably smaller and, in fact, less conservative bounding values may be justified with additional analysis or surveillance requirements.

This upper bound envelope is verified for operation within an allowed AFD operating space, per the RAOC methodology outlined in Reference 13. Axial offset control is detailed in the Technical Specifications, with AFD limits specified in the COLR, and followed by relying only upon excore surveillance supplemented by the normal monthly power distribution measurement requirement and by computer-based alarms on deviation from the allowed AFD operating space.

Allowing for fuel densification effects the average linear power is 5.77 kW/ft at 3612 MWt. The conservative upper bound value of normalized local power density, including uncertainty allowances, is 2.50.

To determine reactor protection system setpoints, with respect to power distributions, three categories of events are considered, namely rod control equipment malfunctions, operator errors of commission, and operator errors of omission. In evaluating these three categories of events, the core is assumed to be operating within the four constraints described above.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence) for full length banks (automatic rod withdrawal is no longer available). Also included are motions of the full-length banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions were

calculated throughout these occurrences, assuming short term corrective action. That is, no transient xenon effects were considered to result from the malfunction. The event was assumed to occur from typical normal operating situations, which include normal xenon transients. It was further assumed in determining the power distributions that total core power level would be limited by reactor trip to below 118.5 percent. Since the study is to determine protection limits with respect to power and axial offset, no credit was taken for trip setpoint reduction due to flux difference. The peak power density which can occur in such events, assuming reactor trip at or below 118.5 percent, is less than that required for center-line melt, including uncertainties and densification effects.

The second category assumes that the operator mispositions the full-length rod bank in violation of the insertion limits and creates short-term conditions not included in normal operating conditions. The third category assumes that the operator fails to take action to correct a flux difference violation. The results for FQ are multiplied by a factor to include an allowance for calorimetric error. The peak linear power does not exceed that required for centerline melt. Since the peak kW/ft is below the limit, no flux difference penalties are required for overpower protection. It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error. Additional detailed discussion of these analyses is presented in Reference 7.

Analyses of possible operating power shapes show that the appropriate hot channel factors F_Q and $F^N\Delta_H$ for peak local power density and for DNB analysis at full power are the values addressed in the Technical Specifications and the COLR.

The maximum allowable F_Q can be increased with decreasing power, as shown in the Technical Specifications. Increasing $F^N\Delta_H$ with decreasing power is permitted by the DNB protection set points and allows radial power shape changes with rod insertion to the insertion limits, as described in Section 4.4.4.3. The allowance for increased permitted becomes a design basis criterion which is used for establishing acceptable control rod patterns and control bank sequencing. Like-wise, fuel loading patterns for each cycle are selected with consideration of this design criterion. The worst values of for possible rod configurations occurring in normal operation are used in verifying that this criterion is met. The worst values generally occur when the rods are assumed to be at their insertion limits. Maintenance of relaxed axial offset control establishes rod positions which are at or above the allowed rod insertion limits. As discussed in Section 3.2 of Reference 8, it has been determined that the Technical Specification limits are met, provided the above conditions a and b are observed. These limits are taken as input to the thermal-hydraulic design basis, as described in Section 4.4.4.3.1.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the precondition for a subsequent hypothetical accident, but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition.

4.3.2.2.7 Experimental Verification of Power Distribution Analysis

This subject is discussed in depth in Reference 2. A summary of this report is given below. The measured and calculational comparison is normally followed periodically throughout the cycle lifetime of the reactor as required by Technical Specifications.

In a measurement of the heat flux hot channel factor, F_Q, with the movable detector system described in Sections 7.7.1 and 4.4.6, the following uncertainties have to be considered:

1. Reproducibility of the measured signal.

2. Errors in the calculated relationship between detector current and local flux.

3. Errors in the calculated relationship between detector flux and peak rod power some distance from the measurement thimble.

The appropriate allowance for category 1 above has been quantified by repetitive measurements made with several inter-calibrated detectors by using the common thimble features of the incore detector system. This system allows more than one detector to access any thimble. Errors in category 2 above are quantified to the extent possible, by using the fluxes measured at one thimble location to predict fluxes at another location which is also measured. Local power distribution predictions are verified in critical experiments on arrays of rods with simulated guide thimbles, control rods, burnable poisons, etc. These critical experiments provide quantification of errors of types 2 and 3 above.

Reference 2 describes critical experiments performed at the Westinghouse Reactor Evaluation Center and measurements taken on two Westinghouse plants with incore systems of the same type as used in this plant. The report concludes that the uncertainty associated with F_Q (heat flux) is 4.58 percent at the 95 percent confidence level with only 5 percent of the measurements greater than the inferred value. This is the equivalent of a 1.645 limit on a normal distribution and is the uncertainty to be associated with a full core flux map with movable detectors reduced with a reasonable set of input data incorporating the influence of burnup on the radial power distribution. The uncertainty is usually rounded up to 5 percent.

In comparing measured power distributions (or detector currents) against the calculations for the same situation it is not possible to subtract out the detector reproducibility. Thus a comparison between measured and predicted power distributions has to include some measurement error. Such a comparison is given in Figures 4.3-24 for one of the maps used in Reference 2. Since the first publication of the report, hundreds of maps have been taken on these and other reactors. The results confirm the adequacy of the 5 percent uncertainty allowance on the calculated F_Q.

A similar analysis for the uncertainty in $F^N_{\Delta H}$ (rod integral power) measurements results in an allowance of 3.65 percent at the equivalent of a 1.645 confidence level. For historical reasons an 8 percent uncertainty factor is allowed in the nuclear design calculational basis; that is, the predicted rod integrals at full power must not exceed the design $F^N_{\Delta H}$ less 8 percent. This 8 percent may be reduced in final design to 4 percent to allow a wider range of acceptable axial power distributions in the DNB analysis and still meet the design bases of Section 4.3.1.3.

A recent measurement in the second cycle of a 121 assembly, 12 foot, core is compared with a simplified one dimensional core average axial calculation in Figure 4.3-25. This calculation does not give explicit representation to the fuel grids.

The accumulated data on power distributions in actual operation was basically of three types:

1. Much of the data was obtained in steady state operation at constant power in the normal operating configuration;

2. Data with unusual values of axial offset were obtained as part of the excore detector calibration exercise which was performed every 92 EFPD;

3. Special tests have been performed in load follow and other transient xenon conditions which have yielded useful information on power distributions.

These data are presented in detail in Reference [8]. Figure 4.3-26 contains a summary of measured values of F_Q as a function of axial offset for five plants from that report.

4.3.2.2.8 Testing and Operations Support

A very extensive series of physics tests is performed on the first cores. These tests and the criteria for satisfactory results are described in Chapter 14. Since not all limiting situations can be created at BOL, the main purpose of the tests is to provide a check on the calculational methods used in the predictions for the conditions of the test. Limited tests are performed at the beginning of each reload cycle to verify that the core is loaded as designed and can be safely operated.

4.3.2.2.9 Monitoring Instrumentation

The adequacy of instrument numbers, spatial deployment, required correlations between readings and peaking factors, calibration and errors are described in References 2, 6, and 8. The relevant conclusions are summarized here in Sections 4.3.2.2.7 and 4.4.6.

Reference 12 describes the instrumentation requirements and calibration of the PDMS and the uncertainties applied to the calculated peaking factors.

Provided the limitations given in Section 4.3.2.2.6 on rod insertion and flux difference are observed, the excore detector system provides adequate online monitoring of power distributions. Further details of specific limits on the observed rod positions and flux difference are given in the Technical Specifications together with a discussion of their bases.

Limits for alarms, reactor trip, etc. are given in the Technical Specifications. Descriptions of the systems provided are given in Section 7.7.

4.3.2.3 Reactivity Coefficients

The kinetic characteristics of the reactor core determine the response of the core to changing plant conditions or to operator adjustments made during normal operation, as well as the core response during abnormal or accidental transients. These kinetic characteristics are quantified in reactivity coefficients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions such as power, moderator or fuel temperatures, or less significantly due to a change in pressure or void conditions. Since reactivity coefficients change during the life of the core, ranges of coefficients are employed in transient analysis to determine the response of the plant throughout life. The results of such simulations and the reactivity coefficients used are presented in Chapter 15. The reactivity coefficients are calculated

on a corewise basis using diffusion theory methods. The effect of radial and axial power distribution on core average reactivity coefficients is implicit in those calculations and is not significant under normal operating conditions. For example, a skewed xenon distribution which results in changing axial offset by 5 percent changes the moderator and Doppler temperature coefficients by less than 0.01 pcm/°F and 0.03 pcm/°F respectively. An artificially skewed xenon distribution which results in changing the radial $F^N_{\Delta H}$ by 3 percent changes the moderator and Doppler temperature coefficients by less than 0.03 pcm/°F and 0.001 pcm/°F respectively. The spatial effects are accentuated in some transient conditions (for example, in postulated rupture of the main steam line break and rupture of rod cluster control assembly mechanism housing described in Sections 15.1.5 and 15.4.8) and are included in these analyses.

The analytical methods and calculational models used in calculating the reactivity coefficients are given in Section 4.3.3.

Quantitative information for calculated reactivity coefficients, including fuel-Doppler coefficient, moderator coefficients (density, temperature, pressure, void) and power coefficient is given in the following sections.

4.3.2.3.1 Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature and is primarily a measure of the Doppler broadening of U-238 and Pu-240 resonance absorption peaks. Doppler broadening of other isotopes such as U-236, Np-237, etc. are also considered but their contributions to the Doppler effect is small. Fission products have a negligible effect on the Doppler coefficient. An increase in fuel temperature increases the effective resonance absorption cross sections of the fuel and produces a corresponding reduction in reactivity.

The fuel temperature coefficient is calculated with the standard nuclear design methods described in Section 4.3.3.

A typical Doppler temperature coefficient is shown in Figure 4.3-27 as a function of the effective fuel temperature (at BOL and EOL conditions). The Doppler-only contribution to the power coefficient, defined later, is shown in Figure 4.3-28 as a function of relative core power. The integral of the differential curve on Figure 4.3-28 is the Doppler contribution to the power defect and is shown in Figure 4.3-29 as a function of relative power. The Doppler coefficient becomes more negative as a function of life as the Pu-240 content increases, thus increasing the Pu-240 resonance absorption, but overall becomes less negative since the fuel temperature changes with burnup as described in Section 4.3.3.1. The upper and lower limits of Doppler coefficient used in accident analyses are given in Chapter 15.

4.3.2.3.2 Moderator Coefficients

The moderator coefficients is a measure of the change in reactivity due to a change in specific coolant parameters such as density, temperature, pressure or void. The coefficients so obtained are moderator density, temperature, pressure and void coefficients.

Moderator Density and Temperature Coefficients

The moderator temperature (density) coefficient is defined as the change in reactivity per degree change in the moderator temperature. Generally, the effect of the changes in moderator density as well as the temperature are considered together. A decrease in moderator density means less moderation which results in a negative moderator coefficient. An increase in coolant temperature, keeping the density constant, leads to a hardened neutron spectrum and results in an increase in resonance absorption in U-238, Pu-240 and other isotopes. The hardened spectrum also causes a decrease in the fission to capture ratio in U-235 and Pu-239. Both of these effects make the moderator coefficients more negative. Since water density changes more rapidly with temperature as temperature increases, the moderator temperature coefficient become more negative with increasing temperature.

The soluble boron used in the reactor as a means of reactivity control also has an effect on moderator density coefficient since the soluble boron poison density as well as the water density is decreased when the coolant temperature rises. A decrease in the soluble poison concentration introduces a positive component in the moderator density coefficient.

Thus, if the concentration of soluble poison is large enough, the net value of the coefficient may be positive. With the burnable absorber rods present, however, the initial hot full power boron concentration is sufficiently low that the moderator temperature coefficient is negative at operating temperatures. The effect of control rods is to make the moderator coefficient more negative by reducing the required soluble boron concentration and by increasing the “leakage” of the core.

With burnup, the moderator coefficient becomes more negative primarily as a result of boric acid dilution but also to a significant extent from the effects of the buildup of plutonium and fission products.

The moderator coefficient is calculated for the various plant conditions discussed above with the standard nuclear design methods described in Section 4.3.3. A typical moderator coefficient is shown as a function of core temperature and boron concentration for the unrodded and rodded core in Figures 4.3-30 through 4.3-32. The temperature range covered is from cold (68°F) to about 600°F. The contribution due to Doppler coefficient (because of change in moderator temperature) has been subtracted from these results. Figure 4.3-33 shows a representative hot full power moderator temperature coefficient plotted as a function of cycle lifetime for the just critical boron concentration condition based on the design boron letdown condition.

The moderator coefficients presented here are calculated on a corewide bases, since they are used to describe the core behavior in normal and accident situations when the moderator temperature changes can be considered to affect the entire core.

Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is of much less significance in comparison with the moderator temperature coefficient. A change of 50 psi in pressure has approximately the same effect on reactivity as a half-degree change in moderator temperature. This coefficient can be determined from the moderator temperature coefficient by relating change in pressure to the corresponding change in density.

The moderator pressure coefficient is negative over a portion of the moderator temperature range at BOL but is always positive at operating conditions and becomes more positive during life.

Moderator Void Coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. In a PWR this coefficient is not very significant because of the low void content in the coolant. The core void content is less than 1/2 of 1 percent and is due to local or statistical boiling. The void coefficient varies from approximately 50 pcm/percent void at BOL and at low temperatures to approximately -250 pcm/ percent void at EOL and at operating temperatures. The negative void coefficient at operating temperature becomes more negative with fuel burnup.

4.3.2.3.3 Power Coefficient

The combined effect of moderator temperature and fuel temperature change as the core power level changes is called the total power coefficient and is expressed in terms of reactivity change per percent power change. A typical power coefficient at BOL and EOL conditions is given in Figure 4.3-34.

It becomes more negative with burnup reflecting the combined effect of moderator and fuel temperature coefficients with burnup. A typical defect (integral reactivity effect) at BOL and EOL is given in Figure 4.3-35.

4.3.2.3.4 Comparison of Calculated and Experimental Reactivity Coefficients

Section 4.3.3 discusses the comparison of calculated and experimental reactivity coefficients.

Experimental evaluation of the calculated coefficients will be completed during the physics startup tests described in Chapter 14.

4.3.2.3.5 Reactivity Coefficients Used in Transient Analysis

The limiting values of reactivity coefficients are used as design limits in the transient analysis. The exact values of the coefficient used in the analysis depend on whether the transient of interest is examined at the BOL or EOL, whether the most negative or the most positive (least negative) coefficients are appropriate, and whether spatial nonuniformity must be considered in the analysis. Conservative values of coefficients, considering various aspects of analysis are used in the transient analysis. This is described in Chapter 15.

The reactivity coefficients shown in Figures 4.3-27 through 4.3-35 are typical best estimate values. The limiting values are chosen to encompass the best estimate reactivity coefficients calculated for the current cycle. The most positive as well as the most negative values are selected to form the design basis range used in the transient analysis. A direct comparison of the best estimate and design limit values can be misleading since in many instances, the most conservative combination of reactivity coefficients is used in the transient analysis even though the extreme coefficients assumed may not simultaneously occur at the conditions of lifetime, power level, temperature and boron concentration assumed in the analysis. The need for a re-evaluation of any accident in a subsequent cycle is contingent upon whether or not the

coefficients for that cycle fall within the identified range used in the analysis presented in Chapter 15 with due allowance for the calculational uncertainties given in Section 4.3.3. Reactivity components for the shutdown margin calculation are given in Table 4.3-3 for the first cycle and for a hypothetical equilibrium cycle. These latter numbers are provided for information only and their validity in a particular cycle would be an unexpected coincidence.

4.3.2.4 Control Requirements

To ensure the shutdown margin stated in the Core Operating Limits Report under conditions where a cooldown to ambient temperature is required, concentrated soluble boron is added to the coolant.

For all core conditions including refueling, the boron concentration is well below the solubility limit. The rod cluster control assemblies are employed to bring the reactor to the hot shutdown condition. The minimum required shutdown margin is given in the Core Operating Limits Report.

The ability to accomplish the shutdown for hot conditions is demonstrated in Table 4.3-3 by comparing the typical difference between the rod cluster control assembly reactivity available with an allowance for the worst struck rod with that required for control and protection purposes. The shutdown margin includes an allowance of 10 percent for analytic uncertainties (see Section 4.3.2.4.9). The largest reactivity control requirement appears at the EOL when the moderator temperature coefficient reaches its peak negative value as reflected in the larger power defect.

The control rods are required to provide sufficient reactivity to account for the power defect from full power to zero power and to provide the required shutdown margin. Typical Ag-In-Cd absorber design control rod reactivity worths listed in Table 4.3-3 are the most limiting values for ensuring shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler, variable average moderator temperature, flux redistribution, and reduction in void content as discussed below.

4.3.2.4.1 Doppler

The Doppler effect arises from the broadening of U-238 and Pu-240 resonance peaks with an increase in effective pellet temperature. This effect is most noticeable over the range of zero power to full power due to the large pellet temperature increase with power generation.

4.3.2.4.2 Variable Average Moderator Temperature

When the core is shutdown to the hot, zero power condition, the average moderator temperature changes from the equilibrium full load value determined by the steam generator and turbine characteristics (steam pressure, heat transfer, tube fouling, etc.) to the equilibrium no load value, which is based on the steam generator shell side design pressure.

Since the moderator coefficient is negative at most operating conditions, there is typically a reactivity addition with power reduction. The moderator coefficient becomes more negative as the fuel depletes because the boron concentration is reduced. This effect is the major contributor to the increased requirement at EOL.

4.3.2.4.3 Redistribution

During full power operation the coolant density decreases with core height; and this, together with partial insertion of control rods, results in less fuel depletion near the top of the core. Under steady state conditions, the relative power distribution will be slightly asymmetric towards the bottom of the core. On the other hand, at hot zero power conditions, the coolant density is uniform up the core, and there is no flattening due to Doppler. The result will be a flux distribution which at zero power can be skewed toward the top of the core. The reactivity insertion due to the skewed distribution is calculated with an allowance for effects of xenon distribution.

4.3.2.4.4 Void Content

A small void content in the core is due to nucleate boiling at full power. The void collapse coincident with power reduction makes a small reactivity contribution.

4.3.2.4.5 Rod Insertion Allowance

At full power, the control bank is operated within a prescribed band of travel to compensate for additional reactivity changes. Since the insertion limit is set by a rod travel limit, a conservatively high calculation of the inserted worth is made which exceeds the normally inserted reactivity.

4.3.2.4.6 Burnup

Excess reactivity is installed at the beginning of each cycle to provide sufficient reactivity to compensate for fuel depletion and fission products throughout the cycle. This reactivity is controlled by the addition of soluble boron to the coolant and by burnable absorber. Since the excess reactivity for burnup is controlled by soluble boron and/or burnable poison, it is not included in control rod requirements.

4.3.2.4.7 Xenon and Samarium Poisoning

Changes in xenon and samarium concentrations in the core occur at a sufficiently slow rate, even following rapid power level changes, that the resulting reactivity change is controlled by changing the soluble boron concentration.

4.3.2.4.8 pH Effects

Changes in reactivity due to a change in coolant pH, if any, are sufficiently small in magnitude and occur slowly enough to be controlled by the boron system. Further details are provided in Reference 11.

4.3.2.4.9 Experimental Confirmation

Following a normal shutdown, the total core reactivity change during cooldown with a stuck rod has been measured on a 121 assembly, 10 foot high core and 121 assembly, 12 foot high core. In each case, the core was allowed to cooldown until it reached criticality simulating the steamline break accident. For the ten foot core, the total reactivity change associated with the cooldown is overpredicted by about 0.3 percent $\Delta\rho$ with respect to the measured result. This represents an error of about 5 percent in the total reactivity change and is about half the uncertainty allowance for this quantity. For the 12 foot core, the difference between the

measured and predicted reactivity change was an even smaller 0.2 percent $\Delta\rho$. These measurements and others demonstrate the capability of the methods described in Section 4.3.3.

4.3.2.4.10 Control

Core reactivity is controlled by means of a chemical poison dissolved in the coolant, rod cluster control assemblies, and burnable poison rods as described below.

4.3.2.4.11 Chemical Poison

Boron in solution as boric acid is used to control relatively slow reactivity changes associated with:

1. The moderator temperature defect in going from cold shutdown at ambient temperature to the hot operating temperature at zero power,

2. The transient xenon and samarium poisoning, such as that following power changes or changes in rod cluster control position,

3. The excess reactivity required to compensate for the effects of fissile inventory depletion and buildup of long-life fission products.

4. The burnable poison depletion.

4.3.2.4.12 Rod Cluster Control Assemblies

Full length rod cluster control assemblies exclusively are employed in this reactor. The full length rod cluster control assemblies are used for shutdown and control purposes to offset fast reactivity changes associated with:

1. The required shutdown margin in the hot zero power, stuck rod condition,

2. The reactivity compensation as a result of an increase in power above hot zero power (power defect including Doppler, and moderator reactivity changes),

3. Unprogrammed fluctuations in boron concentration, coolant temperature, or xenon concentration (with rods not exceeding the allowable rod insertion limits),

4. Reactivity ramp rates resulting from load changes.

The allowed full length control bank reactivity insertion is limited at full power to maintain shutdown capability. As the power level is reduced, control rod reactivity requirements are also reduced and more rod insertion is allowed. The control bank position is monitored and the operator is notified by an alarm if the limit is approached. The determination of the insertion limit uses conservative xenon distributions and axial power shapes. In addition, the rod cluster control assembly withdrawal pattern determined from these analyses is used in determining power distribution factors and in determining the maximum worth of an inserted rod cluster control assembly ejection accident. For further discussion, refer to the Technical Specifications on rod insertion limits.

Power distribution, rod ejection and rod misalignment analyses are based on the arrangement of the shutdown and control groups of the rod cluster control assemblies shown in Figure 4.3-36. All shutdown rod cluster control assemblies are withdrawn before withdrawal of the control banks is initiated. In going from zero to 100 percent power, control banks A, B, C and D are sequentially withdrawn. The limits of rod positions and further discussion on the basis for rod insertion limits are provided in the Technical Specifications.

4.3.2.4.13 Reactor Coolant Temperature

Reactor coolant (or moderator) temperature control has added flexibility in reactivity control of the Westinghouse PWR. This feature takes advantage of the negative moderator temperature coefficient inherent in a PWR to:

1. Maximize return to power capabilities,

2. Provide ±5 percent power load regulation capabilities without requiring control rod compensation, and

3. Extend the time in cycle life for which daily load follow operations can be accomplished.

Reactor coolant temperature control supplements the dilution capability of the plant by lowering the reactor coolant temperature to supply positive reactivity through the negative moderator coefficient of the reactor. After the transient is over, the system automatically recovers the reactor coolant temperature to the programmed value.

Moderator temperature control of reactivity, like soluble boron control, has the advantage of not significantly affecting the core power distribution. However, unlike boron control, temperature control can be rapid enough to achieve reactor power change rates of 5 percent/minute.

4.3.2.4.14 Burnable Absorber Rods

The burnable absorber rods (discrete and/or integral) provide partial control of the excess reactivity available during the fuel cycle. In doing so, these rods prevent the moderator temperature coefficient from being positive at normal full power operating conditions. They perform this function by reducing the requirement for soluble poison in the moderator at the beginning of the fuel cycle as described previously. For purposes of illustration, a typical burnable absorber rod pattern in the core together with the number of rods per assembly is shown in Figure 4.3-5, while the arrangements within an assembly are displayed in Figure 4.3-4. The boron in the rods is depleted with burnup but at a sufficiently slow rate so that the resulting critical concentration of soluble boron is such that the moderator temperature coefficient remains non-positive for full power operating conditions.

4.3.2.4.15 Peak Xenon Startup

Compensation for the peak xenon buildup is accomplished using the boron control system. Startup from the peak xenon condition is accomplished with a combination of rod motion and boron dilution. The boron dilution may be made at any time, including during the shutdown period, provided the shutdown margin is maintained.

4.3.2.4.16 Load Follow Control and Xenon Control

During load follow maneuvers, power changes are accomplished using control rod motion and dilution or boration by the boron system as required. Control rod motion is limited by the control rod insertion limits on full length rods as provided in the Technical Specifications and discussed in Sections 4.3.2.4.12 and 4.3.2.4.13. The power distribution is maintained within acceptable limits through the location of the full length rod bank. Reactivity changes due to the changing xenon concentration can be controlled by rod motion and/or changes in the soluble boron concentration. Late in cycle life, extended load follow capability is obtained by augmenting the limited boron dilution capability at low soluble boron concentrations by temporary moderator temperature reductions.

Rapid power increases (5 percent/min) from part power load follow operation are accomplished with a combination of rod motion, moderator temperature reduction, and boron dilution. Compensation for the rapid power increase is accomplished initially by a combination of rod withdrawal and moderator temperature reduction. As the slower boron dilution takes affect after the initial rapid power increase, the moderator temperature returns to the programmed value.

4.3.2.4.17 Burnup

Control of the excess reactivity for burnup is accomplished using soluble boron and/or burnable absorber. The boron concentration must be limited during operating conditions to ensure the moderator temperature coefficient is within the Technical Specifications. Sufficient burnable absorber is installed at the beginning of a cycle to give the desired cycle lifetime without exceeding the boron concentration limit.

4.3.2.5 Control Rod Patterns and Reactivity Worth

The full length rod cluster control assemblies are designated by function as the control groups and the shutdown groups. The terms “group” and “bank” are used synonymously throughout this report to describe a particular grouping of control assemblies. The rod cluster assembly pattern is displayed in Figure 4.3-36 which is not expected to change during the life of the plant. The control banks are labeled A, B, C, and D and the shutdown banks are labeled SA, SB, etc., as applicable. Each bank, although operated and controlled as a unit, is comprised of two subgroups. The axial position of the full length rod cluster control assemblies may be controlled manually or automatically. The rod cluster control assemblies are all dropped into the core following actuation of reactor trip signals.

Two criteria have been employed for selection of the control groups. First the total reactivity worth must be adequate to meet the control requirements (i.e., fuel temperature, moderator temperature, redistribution and rod insertion allowance). Typical values for these parameters are listed in Table 4.3-3. Second, in view of the fact that these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that the power capability requirements are met. Analyses indicate that the first requirement can be met either by a single group or by two or more banks whose total worth equals at least the required amount. The axial power shape would be more peaked following movement of a single group of rods worth three to four percent $\Delta\rho$; therefore, four banks (described as A, B, C and D in Figure 4.3-36) each worth approximately one percent $\Delta\rho$ have been selected.

The position of control banks for criticality under any reactor condition is determined by the concentration of boron in the coolant. On an approach to criticality, boron is adjusted to ensure that criticality will be achieved with control rods above the insertion limit set by shutdown and other considerations (see the Technical Specifications). Early in the cycle there may also be a withdrawal limit at low power to maintain the moderator temperature coefficient within limits assumed in the safety analysis.

Allowable deviations due to misaligned control rods are discussed in the Technical Specifications.

A representative calculation for two banks of control rods withdrawn simultaneously (rod withdrawal accident) is given in Figure 4.3-37.

Calculation of control rod reactivity worth versus time following reactor trip involves both control rod velocity and differential reactivity worth. A representative rod position versus time of travel after rod release assumed is given in Figure 4.3-38. For nuclear design purposes, the reactivity worth versus rod position is calculated by a series of steady state calculations at various control rod positions assuming all rods out of the core as the initial position in order to minimize the initial reactivity insertion rate. Also to be conservative, the rod of highest worth is assumed stuck out of the core and the flux distribution (and thus reactivity importance) is assumed to be skewed to the bottom of the core. The result of a typical calculation is shown on Figure 4.3-39.

The shutdown groups provide additional negative reactivity to assure an adequate shutdown margin. Shutdown margin is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped, but assuming that the highest worth control assembly remains fully withdrawn and no changes in xenon or boron take place. The loss of control rod worth due to the material irradiaiton is negligible since only bank D may be in the core under normal operating conditions.

The values given in Table 4.3-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design bases minimum shutdown margin allowing for the highest worth cluster to be at its fully withdrawn position. An allowance for the uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.6 Criticality of the Reactor During Refueling and Criticality of Fuel Assemblies

The basis for maintaining the reactor subcritical during refueling is presented in Section 4.3.1.5 and a discussion of how control requirements are met is given in Sections 4.3.2.4 and 4.3.2.5.

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and fuel storage facilities and by administrative control procedures. This section identifies those criteria important to criticality safety analyses.

New fuel is stored in 21 inch center to center racks in the new fuel storage facility with no water present but which are designed so as to prevent accidental criticality even if unborated water is present. For the flooded condition assuming new fuel of the highest anticipated enrichment in place, the effective multiplication factor does not exceed 0.95. For the normally dry condition the effective multiplication factor does not exceed 0.98 with fuel of the highest anticipated enrichment in place assuming possible sources of moderation such as those that could arise during fire

fighting operations. Credit is taken for the inherent neutron-absorbing effect of materials of construction of the racks.

In the analysis for the new fuel storage facilities, the fuel assemblies are assumed to be in their most reactive condition, namely fresh or undepleted and with no control rods or removable neutron absorbers present. Assemblies can not be closer together than the design separation provided by the storage facility except in special cases such as in fuel shipping containers where analyses are carried out to establish the acceptability of the design. The mechanical integrity of the fuel assembly is assumed.

A mechanical description of the fuel storage racks can be found in Section 9.1. A discussion of the SFP Region I and Region II oversized inspection cells is contained in the Technical Specification Bases 3.7.17.

High Density Region I Racks in Spent Fuel Pool 1 (SFP1) (any storage configuration) and in Spent Fuel Pool 2 (SFP2) (any storage configuration)

For High Density Region I Racks in SFP1/SFP2 (any storage configuration), fuel assembly spacing is such that subcriticality is ensured, $k_{eff} \leq 0.95$ taking credit for soluble boron and $k_{eff} < 1.0$ assuming unborated water.

Verification that the design criteria for fuel storage are met is achieved through the use of the three dimensional KENO5a Monte Carlo code as described in Reference 42.

High Density Region II Racks in SFP1/SFP2

For High Density Region II Racks in SFP1/SFP2, fuel assembly spacing is such that subcriticality is ensured, $k_{eff} \leq 0.95$ taking credit for soluble boron and $k_{eff} < 1.0$ assuming unborated water.

Verification that the design criteria for fuel storage are met is achieved through the use of the taking credit for soluble boron and $k_{eff} < 1.0$ assuming unborated water as described in Reference 42.

Low Density Racks in Containment Refueling Cavity (5 x 5 module, any storage configuration)

For the low density racks in the containment refueling cavity (any storage configuration), fuel assembly spacing is such that subcriticality is ensured ($k_{eff} \leq 0.95$) even if assemblies are immersed in unborated water.

Verification that the design criteria for fuel storage are met is achieved through the use of the three dimensional Monte Carlo code package NITAWL-KENO5a, as described in References 39 and 45.

A boron concentration equal to or greater than the Technical Specification limit assures that a dilution event which will result in a k_{eff} greater than 0.95 is not credible. This is demonstrated by a boron dilution analysis performed for the CPNPP Spent Fuel pools. This conclusion is based on the following: (1) a substantial amount of water is needed in order to dilute the SFP to the design k_{eff} of 0.95, (2) since such a large water volume turnover is required, a SFP dilution event would be readily detected by plant personnel via alarms, flooding in the fuel and auxiliary

buildings or by normal operator rounds through the SFP area, and (3) evaluations indicate that, based on the flow rates of non-borated water normally available to the SFP, taken in conjunction with significant operator errors, and equipment failures, sufficient time is available to detect and respond to a dilution event. In addition, there is significant conservatism built into this evaluation; for example, the cooling of the spent fuel pools can be performed by one train supplying common water to both pools. This cooling configuration would allow credit of the volume of both pools and substantially increase the dilution time estimates presented. However, because the flexibility exists for the cooling system to be totally dedicated to one pool, only one pool volume is considered in this evaluation. As discussed above, calculations have been performed on a 95/95 basis to show that the spent fuel rack k_{eff} remains less than 1.0 with non-borated water in the pool. Thus, even if the SFP were diluted to concentrations approaching zero ppm, the fuel in the racks would remain subcritical.

4.3.2.7 Stability

4.3.2.7.1 Introduction

The stability of the PWR cores against xenon-induced spatial oscillations and the control of such transients are discussed extensively in References 6, 14, 15, and 16. A summary of these reports is given in the following discussion and the design bases are given in Section 4.3.1.6.

In a large reactor core, xenon-induced oscillations can take place with no corresponding change in the total power of the core. The oscillation may be caused by a power shift in the core which occurs rapidly by comparison with the xenon-iodine time constants. Such a power shift occurs in the axial direction when a plant load change is made by control rod motion and results in a change in the moderator density and fuel temperature distributions. Such a power shift could occur in the diametral plane of the core as a result of abnormal control action.

Due to the negative power coefficient of reactivity, PWR cores are inherently stable to oscillations in total power. Protection against total power instabilities is provided by the Control and Protection System as described in Section 7.7. Hence, the discussion on the core stability will be limited here to xenon-induced spatial oscillations.

4.3.2.7.2 Stability Index

Power distributions, either in the axial direction or in the X-Y plane, can undergo oscillations due to perturbations introduced in the equilibrium distributions without changing the total core power. The overtones in the current PWRs, and the stability of the core against xenon-induced oscillations can be determined in terms of the eigenvalues of the first flux overtones. Writing, either in the axial direction or in the X-Y plane, the eigenvalue ς of the first flux harmonic as:

$$\varsigma = b + ic \qquad (4.3\text{-}1)$$

then b is defined as the stability index and T = 2 /c as the oscillation period of the first harmonic. The time-dependence of the first harmonic in the power distribution can now be represented as:

$$(t) = Ae^{t} = ae^{bt}\cos ct \qquad (4.3\text{-}2)$$

where A and a are constants. The stability index can also be obtained approximately by:

$$b = \frac{1}{T} \ln \frac{An + 1}{An} \qquad (4.3\text{-}3)$$

where A_n, A_{n+1} are the successive peak amplitudes of the oscillation and T is the time period between the successive peaks.

4.3.2.7.3 Prediction of the Core Stability

The stability of the core described herein (i.e., with 17 x 17 fuel assemblies) against xenon-induced spatial oscillations is expected to be equal to or better than that of earlier designs. The prediction is based on a comparison of the parameters which are significant in determining the stability of the core against the xenon-induced oscillations, namely: 1) the overall core size is unchanged and spatial power distributions will be similar, 2) the moderator temperature coefficient is expected to be similar to or slightly more negative, and 3) the Doppler coefficient of reactivity is expected to be equal to or slightly more negative at full power.

Analysis of both the axial and X-Y xenon transient tests, discussed in Section 4.3.2.7.5, shows that the calculational model is adequate for the prediction of core stability.

4.3.2.7.4 Stability Measurements

1. Axial measurements

Two axial xenon transient tests conducted in a PWR with a core height of 12 feet and 121 fuel assemblies is reported in Reference 17, and will be briefly discussed here. The tests were performed at approximately 10 percent and 50 percent of cycle life.

Both a free-running oscillation test and a controlled test were performed early in cycle life. The second test at mid-cycle consisted of a free-running oscillation test only. In each of the free-running oscillation tests, a perturbation was introduced to the equilibrium power distribution through an impulse motion of the control Bank D and the subsequent oscillation period. In the controlled test conducted early in the cycle, the part length rods were used to follow the oscillations to maintain an axial offset within the prescribed limits. The axial offset of power was obtained from the excore ion chamber readings (which had been calibrated against the incore flux maps) as a function of time for both free-running tests as shown in Figure 4.3-40.

The total core power was maintained constant during these spatial xenon tests, and the stability index and the oscillation period were obtained from a least-square fit of the axial offset data in the form of Equation (4.3-2). The axial offset of power is the quantity that properly represents the axial stability in the sense that it essentially eliminates any contribution from even order harmonics including the fundamental mode. The conclusions of the tests are:

a. The core was stable against induced axial xenon transients both at the core average burnups of 1550 MWD/MTU and 7700 MWD/MTU. The measured

stability indices are -0.041 hr^{-1} for the first test (Curve 1 of Figure 4.3-40) and -0.014 hr^{-1} for the second test (Curve 2 of Figure 4.3-40). The corresponding oscillation periods are 32.4 hrs. and 27.2 hrs., respectively.

b. The reactor core becomes less stable as fuel burnup progresses and the axial stability index was essentially zero at 12,000 MWD/MTU.

2. Measurements in the X-Y plane

Two X-Y xenon oscillation tests were performed at a PWR plant with a core height of 12 feet and 157 fuel assemblies. The first test was conducted at a core average burnup of 1540 MWD/MTU and the second at a core average burnup of 12900 MWD/MTU. Both of the X-Y xenon tests show that the core was stable in the X-Y plane at both burnups. The second test shows that the core became more stable as the fuel burnup increased and all Westinghouse PWRs with 121 and 157 assemblies are expected to be stable throughout their burnup cycles.

In each of the two X-Y tests, a perturbation was introduced to the equilibrium power distribution through an impulse motion of one rod cluster control unit located a long the diagonal axis. Following the perturbation, the uncontrolled oscillation was monitored using the moveable detector and thermocouple system and the excore power range detectors. The quadrant tilt difference (QTD) is the quantity that properly represents the diametral oscillation in the X-Y plane of the reactor core in that the differences of the quadrant average powers over two symmetrically opposite quadrants essentially eliminates the contribution to the oscillation from the azimuthal mode. The QTD data were fitted in the form of Equation (4.3-2) through a least-square method. A stability index of -0.076 hr^{-1} with a period of 29.6 hours was obtained from the thermocouple data shown in Figure 4.3-41.

It was observed in the second X-Y xenon test that the PWR core with 157 fuel assemblies had become more stable due to an increased fuel depletion and the stability index was not determined.

4.3.2.7.5 Comparison of Calculations with Measurements

The analysis of the axial xenon transient tests was performed in an axial slab geometry using a flux synthesis technique. The direct simulation of the axial offset data was carried out using the PANDA Code [18]. The analysis of the X-Y xenon transient tests was performed in an X-Y geometry using TORTISE which is an advanced version of TURTLE Code [10]. Both the PANDA and TORTISE codes solve the two-group time-dependent neutron diffusion equation with time-dependent xenon and iodine concentrations. The fuel temperature and moderator density feedback is limited to a steady-state model. All the X-Y calculations were performed in an average enthalpy plane.

The basic nuclear cross-sections used in this study were generated from the unit cell depletion program ARK, which has evolved from the codes LEOPARD [19] and CINDER [20]. The detailed experimental data during the tests including the reactor power level, enthalpy rise and the impulse motion of the control rod assembly, as well as the plant follow burnup data were closely simulated in the study.

The results of the stability calculation for the axial tests are compared with the experimental data in Table 4.3-5. The calculations show conservative results for both of the axial tests with a margin of approximately -0.01 hr^{-1} in the stability index.

An analytical simulation of the first X-Y xenon oscillation test shows a calculated stability index of -0.081 hr^{-1}, in good agreement with the measured value of -0.076 hr^{-1}. As indicated earlier, the second X-Y xenon test showed that the core had become more stable compared to the first test and no evaluation of the stability index was attempted. This increase in the core stability in the X-Y plane due to increased fuel burnup is due mainly to the increased magnitude of the negative moderator temperature coefficient.

Previous studies of the physics of xenon oscillations, including three dimensional analysis, are reported in the series of topical reports, References 14, 15 and 16. A more detailed description of the experimental results and analysis of the axial and X-Y xenon transient tests is presented in Reference 17 and Section 1 of Reference 21.

4.3.2.7.6 Stability Control and Protection

The excore detector system is utilized to provide indications of xenon-induced spatial oscillations. The readings from the excore detectors are available to the operator and also form part of the protection system.

1. Axial power distribution

   For maintenance of proper axial power distributions, the operator is instructed to maintain an axial offset within a prescribed operating band, based on the excore detector readings. Should the axial offset be permitted to move far enough outside this band, the protection limit will be reached and power will be automatically reduced.

   Twelve foot PWR cores become less stable to axial xenon oscillations as fuel burnup progresses. However, free xenon oscillations are not allowed to occur except for special tests. The fuel length control rod banks are sufficient to dampen and control any axial xenon oscillations present. Should the axial offset be inadvertently permitted to move far enough outside the control band due to an axial xenon oscillation, or any other reason, the protection limit on axial offset will be reached and power will be automatically reduced.

2. Radial power distribution

   The core described herein is calculated to be stable against X-Y xenon induced oscillations at all times in life.

   The X-Y stability of large PWRs has been further verified as part of the startup physics test program for cores with 193 fuel assemblies. The measured X-Y stability of the cores with 157 and 193 assemblies was in good agreement with the calculated stability as discussed in Sections 4.3.2.7.4 and 4.3.2.7.5. In the unlikely event that X-Y oscillations occur, back-up actions are possible and would be implemented, if necessary, to increase the natural stability of the core. This is based on the fact that several actions could be

taken to make the moderator temperature coefficient more negative (less positive), which will increase the stability of the core in the X-Y plane.

Provisions for protection against non-symmetric perturbations in the X-Y power distribution that could result from equipment malfunctions are made in the protection system design. This includes control rod drop, rod misalignment and asymmetric loss of coolant flow.

A more detailed discussion of the power distribution control in PWR cores is presented in References 6 and 7.

4.3.2.8 Vessel Irradiation

A brief review of the methods and analyses used in the determination of neutron and gamma-ray flux attenuation between the core and the pressure vessel is given below. A more complete discussion on the pressure vessel irradiation and surveillance program is given in Section 5.3.

The materials that serve to attenuate neutrons originating in the core and gamma rays from both the core and structural components consist of the core baffle, core barrel, neutron pads and associated water annuli all of which are within the region between the core and the pressure vessel.

In general, few group neutron diffusion theory codes are used to determine fission power density distributions within the active core, and the accuracy of these analyses is verified by incore measurements on operating reactors. Region and rodwise power sharing information from the core calculations is then used as source information in two-dimensional S_n transport calculations which compute the flux distributions throughout the reactor.

The neutron flux distribution and spectrum in the various structural components varies significantly from the core to the pressure vessel. Representative values of the neutron flux distribution and spectrum are presented in Table 4.3-6. The values listed are based on time averaged equilibrium cycle reactor core parameters and power distributions; and, thus, are suitable for long term nvt projections and for correlation with radiation damage estimates.

As discussed in Section 5.3, the irradiation surveillance program utilizes actual test samples to verify the accuracy of the calculated fluxes at the vessel.

4.3.3 ANALYTICAL METHODS

Calculations required in nuclear design consist of the following three distinct types which are performed in sequence:

1. Determination of effective fuel temperatures

2. Generation of macroscopic few-group parameters

3. Space-dependent, few-group diffusion calculations

4.3.3.1 Fuel Temperature (Doppler) Calculations

Temperatures vary radially within the fuel rod, depending on heat generation rate in the pellet, the conductivity of the materials in the pellet, gap and cladding, and coolant temperature.

Fuel temperatures for use in most nuclear design Doppler calculations are obtained from a simplified version of the Westinghouse fuel rod design model described in Section 4.2.1.3.1, which considers the effect of radial variation of pellet conductivity, expansion-coefficient and heat generation rate, elastic deflection of the cladding, and a gap conductance which depends on the initial fill gas, the hot open gap dimension, and the fraction of the pellet over which the gap is closed. The fraction of the gap assumed closed represents an empirical adjustment to produce good agreement with observed reactivity data at BOL. Further gap closure occurs with burnup and accounts for the decrease in Doppler defect with burnup which has been observed in operating plants. For detailed calculations of the Doppler coefficient, such as for use in xenon stability calculations, a more sophisticated temperature model is used which accounts for the effects of fuel swelling, fission gas release, and plastic cladding deformation.

Radial power distributions in the pellet as a function of burnup are obtained from LASER (Reference 4) calculations.

The effective U-238 temperature for resonance absorption is obtained from the radial temperature distribution by applying a radially dependent weighting function. The weighting function was determined from REPAD (Reference 5) Monte Carlo calculations of resonance escape probabilities in several steady state and transient temperature distributions. In each case, a flat pellet temperature was determined which produced the same resonance escape probability as the actual distribution. The weighting function was empirically determined from these results. The effective Pu-240 temperature for resonance absorption is determined by a convolution of the radial distribution of Pu-240 number densities from LASER burnup calculations and the radial weighting function. The resulting temperature is burnup dependent, but the difference between U-238 and Pu-240 temperatures, in terms of reactivity effects, is small.

The effective pellet temperature for pellet dimensional change is that value which produces the same outer pellet radius in a virgin pellet as that obtained from the temperature model. The effective cladding temperature for dimensional change is its average value.

The temperature calculational model has been validated by plant Doppler defect data and Doppler coefficient data. Stability index measurements also provide a sensitive measure of the Doppler coefficient near full power (see Section 4.3.2.7). It can be seen that Doppler defect data are typically within 0.2 percent Δp of prediction.

4.3.3.2 Macroscopic Group Constants

The PHOENIX-P computer code is a two-dimensional, multi-group, transport based lattice code and capable of providing all necessary data for PWR analysis. Being a dimensional lattice code, PHOENIX-P does not rely on pre-determined spatial/spectral interaction assumptions for a heterogeneous fuel lattice, hence, provides a more accurate multi-group flux solution than versions of LEOPARD/CINDER. The PHOENIX-P computer code is approved by the USNRC as the lattice code for generating macroscopic and microscopic few group cross sections for PWR analysis (Reference 22).

The solution for the detailed spatial flux and energy distribution is divided into two major steps in PHOENIX-P. In the first step, a two-dimensional fine energy group nodal solution is obtained which couples individual subcell regions (pellet, cladding and moderator) as well as surrounding pins. PHOENIX-P uses a method based on the Carlvik's collision probability approach and heterogeneous response fluxes which preserves the heterogeneity of the pin cells and their surroundings. The nodal solution provides accurate and detailed local flux distribution, which is then used to spatially homogenize the pin cells to fewer groups.

The second step in the solution process solves for the angular flux distribution using a standard S4 discrete ordinates calculation. This step is based on the group-collapsed and homogenized cross sections obtained from the first step of the solution. The S4 fluxes are then used to normalize the detailed spatial and energy nodal fluxes. The normalized nodal fluxes are used to compute reaction rates, power distribution and to deplete the fuel and burnable absorbers. A standard B1 calculation is employed to evaluate the fundamental mode critical spectrum and to provide an improved fast diffusion coefficient for the core spatial codes.

The PHOENIX-P code employs a 42 energy group library, which has been derived mainly from ENDF/B-V files. The PHOENIX-P cross sections library was designed to properly capture integral properties of the multi-group data during group collapse, and enabling proper modeling of important resonance parameters. The library contains all neutronic data necessary for modeling fuel, fission products, cladding and structural data, coolant, and control/burnable absorber materials present in Light Water Reactor cores.

Group constants for burnable absorber cells, guide thimbles, instrument thimbles, control rod cells and other non-fuel cells can be obtained directly from PHOENIX-P without any adjustments such as those required in the cell or 1D lattice codes.

4.3.3.3 Spatial Few-Group Diffusion Calculations

ANC (Advanced Nodal Code) is a multidimensional nodal analysis program used to predict nuclear reactor core reactivity and assembly and rod distributions for normal and off-normal conditions (Reference 9).

Nodal three-dimensional calculations are carried out to determine the critical boron concentrations and power distributions. The moderator coefficient is evaluated by varying the inlet temperature in the same calculations used for power distribution and reactivity predictions. Validation of reactivity calculations is associated with the validation of the group constants themselves, as discussed in Section 4.3.3.2. Validation of the Doppler calculations is associated with the fuel temperature validation, as discussed in Section 4.3.3.1. Validation of the moderator coefficient calculation is obtained by comparison with plant measurements at HZP conditions.

Axial calculations are used to determine differential control rod worth curves (reactivity versus rod insertion) and axial power shapes during steady state and transient xenon conditions. Group constants are obtained from three-dimensional nodal calculations homogenized by flux volume weighting.

Validation of the spatial codes for calculating power distributions involves the use of incore and excore detectors, and is discussed in Section 4.3.2.2.7.

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33. Deleted.

34. Deleted.

35. Deleted.

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TABLE 4.3-1
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TABLE 4.3-2A
NUCLEAR DESIGN PARAMETERS
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TABLE 4.3-2B
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TABLE 4.3-3
TYPICAL REACTIVITY PARAMETERS FOR ROD CLUSTER CONTROL ASSEMBLIES

Reactivity Effects, percent $\Delta\rho$		Beginning-of-Life (First Cycle)	End-of-Life (First Cycle)	End-of-Life (Equilibrium Cycle)
1.	Control requirements			
	Fuel temperature, Doppler (%$\Delta\rho$)	1.36	1.12	1.10
	Moderator temperature[(a)] (%$\Delta\rho$)	0.10	1.17	1.34
	Redistribution (%$\Delta\rho$)	0.50	0.85	0.85
	Rod insertion allowance (%$\Delta\rho$)	0.50	0.50	0.50
2.	Total control (%$\Delta\rho$)	2.46	3.64	3.79
3.	Estimated rod cluster control assembly worth (53 rods, Ag-In-Cd or Hf)			
	a. All full length assemblies inserted (%$\Delta\rho$)	7.59	7.68	7.30
	b. All but one (highest worth) assemblies inserted (%$\Delta\rho$)	6.62	6.52	6.20
4.	Estimated rod cluster control assembly credit with 10 percent adjustment to accomodate uncertainties, 3b - 10 percent (%$\Delta\rho$)	5.96	5.87	5.58
5.	Shutdown margin available, 4-2 (%$\Delta\rho$)	3.50	2.23	1.79

a) Includes void effects.

TABLE 4.3-4
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TABLE 4.3-5
AXIAL STABILITY INDEX PRESSURIZED WATER REACTOR CORE WITH A 12 FOOT HEIGHT

Burnup (MWD/MTU)	F_Z	C_B (ppm)	Stability Index (hr^{-1}) Exp	Calc
1550	1.34	1065	-0.041	-0.032
7700	1.27	700	-0.014	-0.006
		Difference:	+0.027	+0.026

TABLE 4.3-6
TYPICAL NEUTRON FLUX LEVELS (n/cm^2-sec) AT FULL POWER

	E>1.0MeV	0.111MeV < E <1.0MeV	0.3eV < E <0.111MeV	<E 0.3eV
Core Center	9.98×10^{13}	1.11×10^{14}	2.17×10^{14}	5.36×10^{13}
Core Outer Radius At Mid-Height	4.24×10^{13}	4.85×10^{13}	9.52×10^{13}	2.21×10^{13}
Core Top, on Axis	2.62×10^{13}	2.13×10^{13}	1.31×10^{14}	4.35×10^{13}
Core Bottom, on Axis	2.70×10^{13}	2.25×10^{13}	1.33×10^{14}	4.74×10^{13}
Pressure Vessel Inner Diameter Azimuthal Peak, Core Mid-Height	2.08×10^{10}	2.83×10^{10}	6.18×10^{10}	1.20×10^{11}

TABLE 4.3-7
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TABLE 4.3-8
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TABLE 4.3-9
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TABLE 4.3-10
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TABLE 4.3-11
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4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 DESIGN BASES

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core such that heat removal by the Reactor Coolant System (RCS) or the Emergency Core Cooling System (ECCS) (when applicable) assures that the following performance and safety criteria requirements are met:

1. Fuel damage (defined as penetration of the fission product barrier, i.e, the fuel rod clad) is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These will be within the capability of the plant cleanup system and are consistent with the plant design bases.

2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged (see above definition) although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

To satisfy the above requirements, the following design bases have been established for the thermal and hydraulic design of the reactor core.

4.4.1.1 Departure from Nucleate Boiling Design Basis

Basis

There will be at least a 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events) at a 95 percent confidence level.

Discussion

The design method employed to meet the DNB design basis for both OFA and VANTAGE 5 fuel assemblies is the Revised Thermal Design Procedure (RTDP), Reference 3. Note, the VANTAGE+ fuel design is often referred to as VANTAGE 5 or OFA to distinguish assemblies containing IFMs from assemblies without IFMs, respectively. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined such that there is at least a 95 percent probability at a 95 percent confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events as defined in ANSI N18.2).

Uncertainties in the plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, reactor coolant system flow, etc.) have been evaluated. Only the random portion of the plant operating parameter uncertainties is included in the statistical combination. Instrumentation bias is treated as a direct DNBR penalty. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

The RTDP design limit DNBR for OFA fuel and the WRB-1 correlation is 1.23 for both typical and thimble cells. The RTDP design limit DNBR values are 1.23 and 1.22 for the typical and thimble cells, respectively, for VANTAGE 5 fuel and the WRB-2 correlation. The design limit DNBR values are used as a basis for the technical specifications and for consideration of the applicability of unreviewed safety questions as defined in 10CFR 50.59.

To maintain DNBR margin to offset DNB penalties such as that due to fuel rod bow (see paragraph 4.4.2.1.5), the safety analyses were performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operating and design flexibility.

The Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety analyses input values to give the lowest minimum DNBR. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

By preventing DNB, adequate heat transfer is assured between the fuel clad and the reactor coolant, thereby preventing clad damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis as it will be within a few degrees of coolant temperature during operation in the nucleate boiling region. Limits provided by the nuclear control and protection systems are such that this design basis will be met for transients associated with Condition II events including overpower transients. There is an additional large DNBR margin at rated power operation and during normal operating transients.

4.4.1.2 Fuel Temperature Design Basis

Basis

During modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature at the 95 percent confidence level. The melting temperature of UO_2 is taken as 5080°F, Reference [1], unirradiated and decreasing 58°F per 10,000 MWD/MTU. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated. To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations as described in Section 4.4.2.6.1.

Discussion

Fuel rod thermal evaluations are performed at rated power, maximum overpower and during transients at various burnups. These analyses assure that this design basis as well as the fuel integrity design bases given in Section 4.2 are met. They also provide input for the evaluation of Condition III and IV events given in Chapter 15.

4.4.1.3 Core Flow Design Basis

Basis

Most of the thermal flow rate will pass through the fuel rod region of the core and be effective for fuel rod cooling. Coolant flow through the thimble tubes as well as the leakage from the core barrel-baffle region into the core are not considered effective for heat removal.

Discussion

Core cooling evaluations are based on the thermal flow rate (minimum flow) entering the reactor vessel. A small fraction of this value is allotted as bypass flow. This includes Rod Cluster Control (RCC) guide thimble cooling flow, head cooling flow, baffle leakage, and leakage to the vessel outlet nozzle. The amount of the bypass flow is a function of the fuel design and is confirmed for each reload.

4.4.1.4 Hydrodynamic Stability Design Basis

Basis

Modes of operation associated with Condition I and II events shall not lead to hydrodynamic instability.

4.4.1.5 Other Considerations

The above design bases together with the fuel clad and fuel assembly design bases given in Section 4.2.1 are sufficiently comprehensive so additional limits are not required.

Fuel rod diametral gap characteristics, moderator-coolant flow velocity and distribution, and moderator void are not inherently limiting. Each of these parameters is incorporated into the thermal and hydraulic models used to ensure the above mentioned design criteria are met. For instance, the fuel rod diametral gap characteristics change with time and the fuel rod integrity is evaluated on that basis. The effect of the moderator flow velocity and distribution and moderator void distribution are included in the core thermal-hydraulic evaluation and thus affect the design bases.

Meeting the fuel clad integrity criteria covers possible effects of clad temperature limitations. As noted in Section 4.2, the fuel rod conditions changed with time. A single clad temperature limit for Condition I or Condition II events is not appropriate since of necessity it would be overly conservative. A clad temperature limit is applied to the loss of coolant accident (Section 15.6.5), control rod ejection accident (Section 15.4.8), and locked rotor accident (Section 15.3.3).

4.4.2 DESCRIPTION OF THERMAL AND HYDRAULIC DESIGN OF THE REACTOR CORE

4.4.2.1 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology

DNBRs are calculated using approved correlations and the definitions described in the following sections. An approved core thermal-hydraulic computer code is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is discussed in Section 4.4.4.3.1 (nuclear hot channel factors) and in Section 4.4.2.1.4 (engineering hot channel factors).

4.4.2.1.1 DNB Technology

The primary DNB correlation used for the analysis of 17 x 17 OFA fuel is the WRB-1 correlation (Reference 83). The WRB-1 correlation was developed based exclusively on the large bank of mixing vane grid rod bundle CHF data (over 1100 points) that Westinghouse has collected. The WRB-1 correlation, based on local fluid conditions, represents the rod bundle data with better accuracy over a wide range of variables than the previous correlation used in design. This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and nonuniform heat flux profiles, and variations in rod heated length and in grid spacing. A correlation limit DNBR of 1.17 is applicable for the WRB-1 correlation.

The applicable range of parameters for the WRB-1 correlation is:

Pressure	$1440 \leq P \leq 2490$ psia
Local mass velocity	$0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft^2-hr
Local quality	$-0.2 \leq X_{loc} \leq 0.3$
Heated length, inlet to CHF location	$L_h \leq 14$ ft
Grid spacing	$13 \leq g_{sp} \leq 32$ in.
Equivalent hydraulic diameter	$0.37 \leq d_e \leq 0.60$ in.
Equivalent heated hydraulic diameter	$0.46 \leq d_h \leq 0.59$ in.
Distance from last grid to CHF site	$0 \leq d_g \leq g_{sp}$

The primary DNB correlation used for the analysis of VANTAGE 5 fuel is the WRB-2 correlation (Reference 97). The WRB-2 DNB correlation was developed to take credit for the VANTAGE 5 intermediate flow mixer (IFM) grid design. A limit of 1.17 is applicable for the WRB-2 correlation.

The applicable range of parameters for the WRB-2 correlation is:

Pressure	$1440 \leq P \leq 2490$ psia
Local mass velocity	$0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft^2-hr
Local quality	$-0.1 \leq X_{loc} \leq 0.3$
Heated length, inlet to CHF location	$L_h \leq 14$ ft
Grid spacing	$10 \leq g_{sp} \leq 26$ in.
Equivalent hydraulic diameter	$0.33 \leq d_e \leq 0.5101$ in.
Equivalent heated hydraulic diameter	$0.45 \leq d_h \leq 0.66$ in.

The W-3 DNB correlation, References 5 and 6, is used for both fuel types where the primary DNBR correlations are not applicable. The WRB-1 and WRB-2 correlations were developed based on mixing vane data and therefore are only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 to 1000 psia, the W-3 correlations limit is 1.45, Reference 19. For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30. A cold wall factor, Reference 8, is applied to the W-3 DNB correlation to account for the presence of the unheated thimble surfaces.

4.4.2.1.2 Definition of DNBR

The DNB heat flux ratio (DNBR) as applied to typical cells (flow cells with all walls heated) and thimble cells (flow cells with heated and unheated walls) is defined as:

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}} \quad (1)$$

where:

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F} \quad (2)$$

$q''_{DNB,EU}$ = the uniform DNB heat flux as predicted by the WRB-1, WRB-2, or W-3 DNB correlation (typical cell only).

F = the flux shape factor to account for nonuniform axial heat flux distributions [10] with the term “C” modified as in Reference 5.

q''_{loc} = the actual local heat flux.

The DNBR as applied to the W-3 DNB correlation when a cold wall (CW) is present is:

$$DNBR = \frac{q''_{DNB,N,CW}}{q''_{loc}} \quad (3)$$

where:

$$q''_{DNB,N,CW} = \frac{q''_{DNB,EU,Dh} \times CWF}{F} \quad (4)$$

where:

$q''_{DNB,EU,Dh}$ = the uniform DNB heat flux as predicted by the W-3 cold wall DNB correlation (5) when not all flow cell walls are heated (i.e. - thimble CW cell).

$$CWF = \frac{q''_{coldwall}}{q''_{W-3,D_h}} = 1.0 - R_u\left[13.76 - 1.372e^{1.78\chi} - 4.732\left(\frac{G}{10^6}\right)^{-0.0535} - 0.0619\left(\frac{P}{1000}\right)^{0.14} - 8.509D_h^{0.107}\right]$$

R_u = 1.0-De/Dh

4.4.2.1.3 Mixing Technology

The rate of heat exchange by mixing between flow channels is proportional to the difference in the local mean fluid enthalpy of the respective channels, the local fluid density, and the flow velocity. The proportionality is expressed by the dimensionless thermal diffusion coefficient (TDC) which is defined as:

$$TDC = \frac{w'}{\rho V a}$$

where:

w' = flow exchange rate per unit length (lbm/ft-s).

ρ = fluid density (lbm/ft^3).

V = fluid velocity (ft/s).

a = lateral flow area between channels per unit length (ft^2/ft).

The application of the TDC in the thermal hydraulic analysis for determining the overall mixing effect or heat exchange rate is presented in Reference 7.

Various mixing tests have been performed at Columbia University.(12) These series of tests, using the "R" mixing vane grid design on 13-, 26-, and 32-in. grid spacing, were conducted in pressurized water loops at Reynolds numbers similar to that of a pressurized water reactor (PWR) core under the following single- and two-phase (subcooled boiling) flow conditions:

- Pressure 1500 to 2400 psia
- Inlet temperature 332 to 642°F
- Mass velocity 1.0 to 3.5 x 10^6 lbm/hr-ft^2
- Reynolds number 1.34 to 7.45 x 10^5
- Bulk outlet quality -52.1 to -13.5 percent

DC was determined by comparing the THINC code predictions with the measured subchannel exit temperatures. TDC is found to be independent of the Reynolds number, mass velocity, pressure, and quality over the ranges tested. The two-phase data (local, subcooled boiling) fell within the scatter of the single-phase data. The effect of two-phase flow on the value of TDC has been demonstrated by Cadek,(12) Rowe and Angle,(13)(14) and Gonzalez-Santalo and Griffith.(15) In the subcooled boiling region, the values of TDC were indistinguishable from the single-phase values. In the quality region, Rowe and Angle show that in the case with rod spacing similar to that in PWR core geometry, the value of TDC increased with quality to a point and then decreased, but never below the single-phase value. Gonzalez-Santalo and Griffith show that the mixing coefficient increased as the void fraction increased. The data from these tests on the R-grid showed that a design TDC value of 0.038 (for 26-in. grid spacing) can be used in determining the effect of coolant mixing in the THINC analysis. A mixing test program similar to the one described above was conducted at Columbia University for the current 17 x 17 geometry and mixing vane grids on 26-in. spacing.(16) The mean value of TDC obtained from these tests was 0.059, and all data were well above the current design value of 0.038.

The inclusion of three intermediate flow mixer grids in the upper span of the VANTAGE 5 fuel assembly results in a grid spacing of approximately 10 inches. Per Reference 97, a TDC value of 0.038 was chosen as a conservatively low value for use in VANTAGE 5 to determine the effect of coolant mixing in the core thermal performance analysis.

4.4.2.1.4 Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratios of these quantities.

The heat flux hot channel factor considers the local maximum linear heat generation rate at a point (the hot spot), and the enthalpy rise hot channel factor involves the maximum integrated value along a channel (the hot channel).

Each of the total hot channel factors considers a nuclear hot channel factor (see Section 4.4.4.3) describing the neutron power distribution and an engineering hot channel factor, which allows for variations in flow conditions and fabrication tolerances. The engineering hot channel factors are made up of subfactors which account for the influence of the variations of fuel pellet diameter, density, enrichment and eccentricity; fuel rod diameter pitch and bowing inlet flow distribution; flow redistribution; and flow mixing.

Heat Flux Engineering Hot Channel Factor, F^{E}_{Q}

The heat flux engineering hot channel factor is used to evaluate the maximum heat flux. This subfactor is determined by statistically combining the tolerances for the fuel pellet diameter and has a value of 1.03 at the 95-percent probability level with 95-percent probability level with 95-percent confidence, density, enrichment, eccentricity and the fuel rod diameter. As shown using approved methodologies [90] and [92], no DNB penalty need be taken for the relatively low intensity heat flux spikes caused by variations in the above parameters, as well as fuel pellet eccentricity and fuel rod diameter variation.

Enthalpy Rise Engineering Hot Channel Factor,$F^{E}_{\Delta H}$

The effect of variations in flow conditions and fabrication tolerances on the hot channel enthalpy rise is directly considered in the core thermal-hydraulic analysis under any reactor operating condition. The items considered contributing to the enthalpy rise engineering hot channel factor are discussed below:

1. Pellet diameter, density and enrichment:

   Variations i pellet diameter, density, and enrichment are considered statistically in establishing the limit DNBRs (see Section 4.4.1.1) for the Revised Thermal Design Procedure (Reference 101) employed in this application. Variances in these design values are met for 95 percent of the limiting channels at a 95 percent confidence level.

2. Inlet flow maldistribution:

   The consideration of inlet flow maldistribution in core thermal performances is discussed in Section 4.4.4.2.2. A design basis of 5 percent reduction in coolant flow to the hot assembly is used in the core thermal-hydraulic analysis.

3. Flow redistribution:

   The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the non-uniform power distribution is inherently considered in the core thermal-hydraulic analysis for every operating condition which is evaluated.

4. Flow mixing:

   The spacer grid design induces additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances.

4.4.2.1.5 Effects of Rod Bow on DNBR

The phenomenon of fuel rod bowing, as described in Reference 81, must be accounted for in DNBR safety analysis of Condition I and Condition II vents for each plant application. The methodology described in Reference 81 has been accepted by the NRC for use in design applications (Reference 99) and used in the analysis of fuel rod bowing.

The rod bow penalties accounted for in the design safety analysis are based on an assembly average of 24,000 MWD/MTU. Credit is taken for the effect of $F^N_{\Delta H}$ burndown due to the decrease in fissionable isotopes and the buildup of fission product inventory (Reference 100).

In the upper spans of the VANTAGE+ fuel assembly, additional restraint is provided with the intermediate flow mixer (IFM) grids such that the grid-to-grid spacing in those spans with IFM grids is approximately 10 inches compared to approximately 20 inches in the other spans. Therefore, no rod bow DNBR penalty is required in the 10 inch spans in the VANTAGE+ safety analyses.

For the safety analysis for the CPNPP units, sufficient DNBR margin is maintained to accommodate DNBR penalties due to rod bow.

4.4.2.2 Linear Heat Generation Rate

The core average and maximum linear heat generation rates are given in Table 4.4-1. The method of determining the maximum linear heat generation rate is given in Section 4.3.

4.4.2.3 Void Fraction Distribution

The void models used in the VIPRE-01 computer code are described in Subsection 4.4.2.6.3.

4.4.2.4 Normalized Core Flow and Enthalpy Rise Distributions

Normalized core flow and enthalpy rise distributions are shown in Figures 4.4-2, 4.4-3, and 4.4-4 which would be typical of the Cycle 1 core design. These general distributions would also be typical of later operating cycles.

4.4.2.5 Core Pressure Drops and Hydraulic Loads

4.4.2.5.1 Core Pressure Drops

The core pressure drop includes the fuel assembly, lower core plate, and upper core plate pressure drops. These pressure drops are based on the best estimate flow for expected plant operating conditions as described in Section 5.1.1. Section 5.1.1 also defines and describes the thermal design flow (minimum flow) which is the basis for reactor core thermal performance and the mechanical design flow (maximum flow) which is used in the mechanical design of the reactor vessel internals and fuel assemblies. Since the best estimate flow is that flow which is most likely to exist in an operating plant, the calculated core pressure drop values are based on this best estimate flow rather than the thermal design flow.

Uncertainties associated with the core and vessel pressure drop values are discussed in Section 4.4.2.8.2.

4.4.2.5.2 Hydraulic Loads

The fuel assembly hold down springs are designed to keep the fuel assemblies in contact with the lower core plate under all Condition I and II events with the exception of the turbine overspeed transient associated with a loss of external load. The hold down springs are designed to tolerate the possibility of an over deflection associated with fuel assembly liftoff for this case and to provide contact between the fuel assembly and the lower core plate following this transient. More adverse flow conditions occur during a loss of coolant accident. These conditions are presented in Section 15.6.5.

Hydraulic loads at normal operating conditions are calculated considering the best estimate flow which is described in Section 5.1, accounting for the best estimate core bypass flow and adding margin. Core hydraulic loads at cold plant startup conditions are based on the cold best estimate flow, but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which could possibly create flow rates 18 to 20 percent

greater than the best estimate flow, are evaluated to be approximately twice the fuel assembly weight.

Core hydraulic loads were measured during the prototype assembly tests of the optimized fuel assembly. Reference [86] contains a detailed discussion of the results.

4.4.2.6 Correlation and Physical Data

4.4.2.6.1 Surface Heat Transfer Coefficients

Forced convection heat transfer coefficients are obtained from the familiar Dittus-Boelter correlation (Reference 20), with the properties evaluated at bulk fluid conditions:

$$\frac{hDe}{K} = 0.023\left(\frac{D_e G}{\mu}\right)^{0.8}\left(\frac{C_p \mu}{K}\right)^{0.4} \qquad (6)$$

where:

h = heat transfer coefficient, (Btu/hr-ft^2°F)

De = equivalent diameter, (ft)

K = thermal conductivity, (Btu/hr-ft-°F)

G = mass velocity, (lb_m/hr-ft^2)

μ = dynamic viscosity, (lb_m/ft-hr)

C_P = heat capacity, (Btu/lb_m -°F)

This correlation has been shown to be conservative (Reference 21) for rod bundle geometries with pitch-to-diameter ratios in the range used by PWRs. The onset of nucleate boiling occurs when the clad wall temperature reaches the amount of superheat predicted by Thom's correlation, Reference 22. After this occurrence the outer clad wall temperature is determined by:

$$\Delta T_{sat} = (0.072 \exp(-P/1260))\,(q")^{0.5} \qquad (7)$$

where:

ΔT_{sat} = wall superheat, $T_w - T_{sat}$, (°F)

q" = wall heat flux, (Btu/hr-ft^2)

P = pressure, (psia)

T_w = outer clad wall temperature, (°F)

T_{sat} = saturation temperature of coolant at P, (°F)

4.4.2.6.2 Total Core and Vessel Pressure Drop

Unrecoverable pressure losses occur as a result of viscous drag (friction) and/or geometry changes (form) in the fluid flow path. The flow field is assumed to be incompressible, turbulent, single- phase water. These assumptions apply to the core and vessel pressure drop calculations for the purpose of establishing the primary loop flow rate. Two-phase considerations are neglected in the vessel pressure drop evaluation because the core average void is negligible. Two-phase flow considerations in the core thermal subchannel analyses are considered and the models are discussed in Subsection 4.4.4.2.3. Core and vessel pressure losses are calculated by equations of the form:

$$P_L = \left(K + \frac{FL}{D_e}\right)\frac{\rho V^2}{2g_c(144)} \qquad (8)$$

Where:

P_L = unrecoverable pressure drop, (lbf/in^2)

ρ = fluid density, (lbm/ft^3)

L = length, (ft)

D^e = equivalent diameter, (ft)

V = fluid velocity, (ft/sec)

g_c = $32.174 \frac{lb_m - ft}{lb_f - sec^2}$

K = form loss coefficient, dimensionless

F = friction loss coefficient, dimensionless

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

Values are quoted in Table 4.4-1 for unrecoverable pressure loss across the reactor vessel. The results of full-scale tests of core components and fuel assemblies were utilized in developing the core pressure loss characteristic. The pressure drop for the vessel was obtained by combining the core loss with correlation of 1/7th scale model hydraulic test data on a number of vessels

(References 23 and 24) and form loss relationships (Reference 25). Moody curves (Reference 26) were used to obtain the single-phase friction factors.

4.4.2.6.3 Void Fraction Correlation

Empirical correlations are used in VIPRE to model the void fraction in two-phase flow. The subcooled void correlation used to model the non-equilibrium transition from single phase to nucleate boiling is given in Reference 27. The bulk (saturated) void model relates flow quality with void fraction which can account for phase slip.

4.4.2.7 Thermal Effects of Operation Transients

DNB core safety limits are generated as a function of coolant temperature, pressure, core power and axial power imbalance. Steady- state operation within these safety limits insures that the minimum DNBR is not less than the DNBR limit.

The overtemperature N-16 system provides adequate protection against anticipated operational transients that are slow with respect to fluid transport delays in the primary system. For fast transients, e.g., uncontrolled rod bank withdrawal at power incident (Section 15.4.2) specific protection functions are provided as described in Section 7.2 and the use of these protection functions is described in Chapter 15.

4.4.2.8 Uncertainties in Estimates

4.4.2.8.1 Uncertainties in Fuel and Clad Temperatures

As discussed in Section 4.4.2.10, the fuel temperature is a function of crud, oxide, clad, pellet-clad gap, and pellet conductances. Uncertainties in the fuel temperature calculation are essentially of two types: fabrication uncertainties such as variations in the pellet and clad dimensions and the pellet density; and model uncertainties such as variations in the pellet conductivity and the gap conductance. These uncertainties have been quantified by comparison of the thermal model to inpile thermocouple measurements, by out-of pile measurements of the fuel and clad properties, and by measurements of the fuel and clad dimensions during fabrication. The resulting uncertainties are then used in all evaluations involving the fuel temperature. The effect of densification on fuel temperature uncertainties is also considered.

In addition to the temperature uncertainty described above, the measurement uncertainty in determining the local power, and the effect of density and enrichment variations on the local power are considered in establishing the heat flux hot channel factor. These uncertainties are described in Section 4.3.2.2.1.

Reactor trip setpoints, as specified in the Technical Specifications, include allowance for instrument and measurement uncertainties such as calorimetric error, instrument drift and channel reproducibility, temperature measurement uncertainties, noise, and heat capacity variations. Uncertainty in determining the cladding temperature results from uncertainties in the crud and oxide thicknesses. Because of the excellent heat transfer between the surface of the rod and the coolant, the film temperature drop does not appreciably contribute to the uncertainty.

4.4.2.8.2 Uncertainties in Pressure Drops

Core and vessel pressure drop values are based on the best estimate flow, as described in Section 5.1. The uncertainties are based on the uncertainties in both the test results and the analytical extension of these values to the reactor application.

A major use of the core and vessel pressure drop value is to determine the primary system coolant flow rates as discussed in Section 5.1. In addition, as discussed in Section 4.4.5.1, tests on the primary system prior to initial criticality will be made to verify that a conservative primary system coolant flow rate has been used in the design and analyses of the plant.

4.4.2.8.3 Uncertainties Due to Inlet Flow Maldistribution

The effects of uncertainties in the inlet flow maldistribution criteria used in the core thermal analyses are discussed in Section 4.4.4.2.2.

4.4.2.8.4 Uncertainty in DNB Correlations

The uncertainty in the DNB correlations can be written as a statement on the probability of not being in DNB based on the statistics of the DNB data. This is discussed in Section 4.4.4.2.2.

4.4.2.8.5 Uncertainties in DNBR Calculations

The uncertainties in the DNBRs calculated by the core thermal-hydraulic analysis due to uncertainties in the nuclear peaking factors are accounted for by applying conservatively high values of the nuclear peaking factors and including measurement error allowances in the statistical evaluation of the limit DNBR using the Revised Thermal Design Procedure (Reference 101). In addition, conservative values for the engineering hot channel factors and flow mixing coefficients are used. The results of sensitivity studies [18] and [92] show that the minimum DNBR in the hot channel is relatively insensitive to variations in the core-wide radial power distribution (for the same value of $F^N_{\Delta H}$).

Studies have been performed [18] and [92] to determine the sensitivity of the minimum DNBR in the hot channel to the void fraction correlation; the inlet velocity and exit pressure distributions assumed as boundary conditions for the analysis; and the grid pressure loss coefficients. The results of these studies show that the minimum DNBR in the hot channel is relatively insensitive to variations in these parameters. The range of variations considered in these studies covered the range of possible variations in these parameters.

4.4.2.8.6 Uncertainties in Flow Rates

The uncertainties associated with loop flow rates are discussed in Section 5.1. For core thermal performance evaluations, a thermal design loop flow is used which accounts for both prediction and measurement uncertainties. In addition, a small fraction of the thermal design flow is assumed to be ineffective for core heat removal capability because it bypasses the core through the various available vessel flow paths described in Section 4.4.4.2.1.

4.4.2.8.7 Uncertainties in Hydraulic Loads

As discussed in Section 4.4.2.5.2, hydraulic loads on the fuel assembly are evaluated for a pump overspeed transient which could possibly create flow rates 18 to 20 percent greater than the mechanical design flow. The mechanical design flow as stated in Section 5.1 is greater than the best estimate or most likely flow rate value for the actual plant operating condition.

4.4.2.9 Flux Tilt Considerations

Significant quadrant power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation. A dropped or misaligned rod cluster control assembly could cause changes in hot channel factors; however, these events are analyzed separately in Chapter 15. This discussion will be confined to flux tilts caused by conditions such as x-y xenon transients, inlet temperature mismatches, and enrichment variations within tolerances.

In addition to unanticipated quadrant power tilts as described above, other readily explainable asymmetries may be observed during calibration of the excore detector quadrant power tilt alarm. During operation, incore maps are taken at least once per month and, periodically, additional maps are obtained for calibration purposes. Each of these maps is reviewed for deviations from the expected power distributions. Asymmetry in the core, from quadrant to quadrant, is frequently a consequence of the design when assembly and/or components shuffling and rotation requirements do not allow exact symmetry preservation. In each case, the acceptability of an observed asymmetry, planned or otherwise depends solely on meeting the required accident analyses assumptions.

In practice, once acceptability has been established by review of the incore maps, the quadrant power tilt alarms and related instrumentation are adjusted to indicate zero Quadrant Power Tilt Ratio as the final step in the calibration process. This action ensures that the instrumentation is correctly calibrated to alarm in the event an unexplained or unanticipated change occurs in the quadrant to quadrant relationships between calibration intervals. Proper functioning of the quadrant power tilt alarm is significant because no allowances are made in the design for increased hot channel factors due to unexpected developing flux tilts since all likely causes are prevented by design or procedures or specifically analyzed. Finally in the event that unexplained flux tilts do occur, the Technical Specifications provide appropriate corrective actions to ensure continued safe operation of the reactor.

4.4.2.10 Fuel and Cladding Temperatures

Consistent with the thermal-hydraulic design bases in Section 4.4.1, the following discussion pertains mainly to the fuel pellet temperature evaluation. A discussion of fuel clad integrity is presented in Section 4.2.3.1.

The thermal-hydraulic design assures that the maximum fuel temperature is below the melting point of UO_2. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. However, the computation of radial fuel temperature distributions combines crud, oxide, clad gap and pellet conductances. The factors which influence these conductances, such as gap size (or contact pressure), internal gas pressure, gas composition, pellet density, time dependent fuel densification and radial power

distribution within the pellet, etc., have been combined into a semi-empirical thermal model. This thermal model enables the determination of these factors and their net effects on temperature profiles. The temperature predictions have been compared to inpile fuel temperature measurements with good results.

Fuel rod thermal evaluations (fuel centerline, average and surface temperatures) are determined throughout the fuel rod lifetime with consideration of time dependent densification. To determine the maximum fuel temperature, various burnup rods, including the highest burnup rod, are analyzed over the rod linear power range of interest. The principal factors which are employed in the determination of the fuel temperature are discussed below.

4.4.2.10.1 UO_2 Thermal Conductivity

The thermal conductivity of uranium dioxide was evaluated from data reported from a number of measurements.

At the higher temperatures, thermal conductivity is best obtained by utilizing the integral conductivity to melt which can be determined with more certainty. From an examination of the data, it has been concluded that the best estimate for the value of 2800°C Kdt is 93 watts/cm.

The design curve is in excellent agreement with the recommendation of the IAEA panel (Reference 28).

4.4.2.10.2 Radial Power Distribution in UO_2 Fuel Rods

An accurate description of the radial power distribution as a function of burnup is needed for determining the power level for incipient fuel melting and other important performance parameters such as pellet thermal expansion, fuel swelling and fission gas release rates. Radial power distribution in UO2 fuel rods is determined with the neutron transport code LASER. The LASER code has been validated by comparing the code predictions on radial burnup and isotopic distributions with measured radial microdrill data (References 29 and 30). A “radial power depression factor,” f, is determined using radial power distributions predicted by LASER. The factor, f, enters into the determination of the pellet centerline temperature, TC, relative to the pellet surface temperature, T_S, through the expression:

$$\int_{T_s}^{T_c} K(T)dT = \frac{q'f}{4\pi}$$

where:

K(T) = the thermal conductivity for UO_2 with a uniform density distribution

q' = the linear power generation rate

4.4.2.10.3 Gap Conductance

The temperature drop across the pellet-clad gap is a function of the gap size and the thermal conductivity of the gas in the gap. The gap conductance model is selected such that when

combined with the UO_2 thermal conductivity model, the calculated fuel centerline temperatures reflect the inpile temperature measurements. A more detailed discussion of the gap conductance model is presented in Reference 31.

4.4.2.10.4 Surface Heat Transfer Coefficients

The fuel rod surface heat transfer coefficients during subcooled forced convection and nucleate boiling are presented in Subsection 4.4.2.7.1.

4.4.2.10.5 Fuel Clad Temperatures

The outer surface of the fuel rod at the hot spot operates at a temperature of slightly higher than the bulk fluid temperature at the hot spot for steady state operation at rated power throughout core life due to nucleate boiling.

Initially (beginning-of-life), this temperature is that of the clad metal outer surface.

During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the clad surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. Since the thermal-hydraulic design basis limits DNB, adequate heat transfer is provided between the fuel clad and the reactor coolant so that the core thermal output is not limited by considerations of clad temperature.

4.4.2.10.6 Treatment of Peaking Factors

The total heat flux hot channel factor, F_Q, is defined by the ratio of the maximum to core average heat flux. The design value of F_Q for normal operation is 2.42 for the 3458 MWt design and 2.5 for the uprated design providing 3612 MWt. This results in peak linear powers at full-power conditions of 13.36 and 14.41 kW/ft respectively.

The centerline temperature at the peak linear power resulting from overpower transients/ overpower errors must be below the UO_2 melt temperature over the lifetime of the rod, including allowances for uncertainties. The fuel temperature design basis is discussed in Section 4.4.1.2. The centerline temperature at the peak linear power resulting from overpower transients/ overpower errors is below that required to produce melting.

4.4.3 DESCRIPTION OF THE THERMAL AND HYDRAULIC DESIGN OF THE REACTOR COOLANT SYSTEM

4.4.3.1 Plant Configuration Data

Plant configuration data for the thermal-hydraulic and fluid systems external to the core are provided in the appropriate Chapters 5, 6, and 9. Implementation of the Emergency Core Cooling System (ECCS) is discussed in Chapter 15.

4.4.3.2 Operating Restrictions on Pumps

The minimum net positive suction head (NPSH) and minimum seal injection flow rate must be established before operating the reactor coolant pumps. With the minimum labyrinth seal

injection flow rate established, the operator will have to verify that the system pressure satisfies NPSH requirements.

4.4.3.3 Power-Flow Operating Map (BWR)

Not applicable to the CPNPP.

4.4.3.4 Temperature-Power Operating Map

A representative relationship between Reactor Coolant System temperature and power is shown in Figure 4.4-21.

The effects of reduced core flow due to inoperative pumps is discussed in Sections 15.3.1 and 15.3.2. Natural circulation capability of the system is discussed in Section 15.2.6.

4.4.3.5 Load Following Characteristics

The Reactor Coolant System is designed on the basis of steady state operation at full power heat load. The reactor coolant pumps utilize constant speed drives as described in Section 5.4 and the reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load, as described in Section 7.7.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal and hydraulic characteristics are given in Tables 4.1-1 and 4.4-1.

4.4.4 EVALUATION

4.4.4.1 Critical Heat Flux

The critical heat flux correlations utilized in the core thermal analyses is explained in detail in Section 4.4.2.

4.4.4.2 Core Hydraulics

4.4.4.2.1 Flow Paths Considered in Core Pressure Drop and Thermal Design

The following flow paths for core bypass flow are considered:

1. Flow through the spray nozzles into the upper head for head cooling purposes.

2. Flow entering into the RCC guide thimbles.

3. Leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel.

4. Flow introduced between the baffle and the barrel for the purpose of cooling these components and which is not considered available for core cooling.

5. Flow in the gaps between the fuel assemblies on the core periphery and the adjacent baffle wall.

The above contributions are evaluated for each cycle to confirm that the design value of the core bypass flow is met.

4.4.4.2.2 Inlet Flow Distributions

Data have been considered from several 1/7 scale hydraulic reactor model tests, References [23], [24], and [62], in arriving at the core inlet flow maldistribution criteria used in the core thermal-hydraulic analyses.

The effect of the total flow rate on the inlet velocity distribution was studied in the experiments of Reference [23]. As was expected, on the basis of the theoretical analysis, no significant variation could be found in inlet velocity distribution with reduced flow rate.

4.4.4.2.3 Empirical Friction Factor Correlations

Two empirical friction factor correlations are used in the VIPRE-01 computer code (described in Subsection 4.4.4.5.1).

The friction factor in the axial direction, parallel to the fuel rod axis, is evaluated using the correlations described in Reference 27).

The flow in the lateral directions, normal to the fuel rod axis, views the reactor core as a large tube bank. Thus, the lateral friction factor proposed by Idel'chik (Reference 25) is applicable. This correlation is of the form:

$$F_L = A\ Re_L^{-0.2}$$

where:

A is a function of the rod pitch and diameter as given in Reference 25.

Re_L is the lateral Reynolds number based on the rod diameter.

4.4.4.3 Influence of Power Distribution

The core power distribution which is largely established at beginning- of-life by fuel enrichment, loading pattern, and core power level is also a function of variables such as control rod worth and position, and fuel depletion throughout lifetime. Radial power distributions in various planes of the core are often illustrated for general interest; however, the core radial enthalpy rise distribution as determined by the integral of power up each channel is of greater importance for DNB analyses. These radial power distributions, characterized by $F^N_{\Delta H}$ defined in Section 4.3.2.2.1) as well as axial heat flux profiles are discussed in the following two sections.

4.4.4.3.1 Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$

Given the local power density q' (kW/ft) at a point x, y, z in a core with N fuel rods and height H,

$$F^N_{\Delta H} = \frac{\text{hot rod power}}{\text{average rod power}} = \frac{\text{Max}\int_o^h q'(xo, yo, z)dz}{\frac{1\Sigma}{N_{\text{all rods}}}\int_o^h q'(x, y, z)dz}$$

The way in which $F^N_{\Delta H}$ is used in the DNB calculation is important. The location of minimum DNBR depends on the axial profile and the value of DNBR depends on the enthalpy rise to that point. Basically, the maximum value of the rod integral is used to identify the most likely rod for minimum DNBR. An axial power profile is obtained, which when normalized to the design value of $F^N_{\Delta H}$, recreates the axial heat flux along the limiting rod. The surrounding rods are assumed to have the same axial profile with rod average powers which are typical distributions found in hot assemblies. In this manner, worst case axial profiles can be combined with worst case radial distributions for reference DNB calculations.

It should be noted again that $F^N_{\Delta H}$ is an integral and is used as such in DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal power shapes throughout the core. The sensitivity of the core thermal-hydraulic analysis to radial power shapes is discussed in References [18] and [92]. For operation at a fraction of full power, the design $F^N_{\Delta H}$ used is given by:

$$F^N_{\Delta H} = F^{RTD}_{\Delta H}[1 + PF_{\Delta H}(1-P]$$

$F^{RTP}_{\Delta H}$ is the limit at rated thermal power (RTP) specified in the Core Operating Limits Report (COLR).

$PF_{\Delta H}$ is the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR.

P is the fraction of rated thermal power.

The permitted relaxation of $F^N_{\Delta H}$ is included in the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits, (Reference 32) thus allowing greater flexibility in the nuclear design.

4.4.4.3.2 Axial Heat Flux Distributions

As discussed in Section 4.3.2.2, the axial heat flux distribution can vary as a result of rod motion, power change, or due to spatial xenon transients which may occur in the axial direction. Consequently, it is necessary to measure the axial power imbalance by means of the excore nuclear detectors (as discussed in Section 4.3.2.2.7) and protect the core from excessive axial power imbalance.

The Reactor Trip System provides automatic reduction of the trip setpoint in the overtemperature N-16 channels on excessive axial power imbalance; that is, when an extremely large axial offset corresponds to an axial shape which could lead to a DNBR which is less than that calculated for the reference DNB design axial shape.

To determine the penalty to be taken in protection setpoints for extreme values of flux difference, this reference shape is supplemented by other axial shapes skewed to the bottom and top of the core. The initial conditions for the accidents for which DNB protection is required are assumed to be those permissible within the constant axial offset control strategy for the load maneuvers described in Reference [67]. The course of those accidents in which DNB is a concern is analyzed in Chapter 15, assuming that the protection setpoints have been set on the basis of these shapes. In many cases, the axial power distribution in the hot channel changes throughout the course of the accident due to rod motion, coolant temperature and power level changes.

4.4.4.4 Core Thermal Response

A general summary of the steady-state thermal-hydraulic design parameters including thermal output, flow rates, etc., is provided in Table 4.4-1. As stated in Section 4.4.1, the design bases for the thermal hydraulic design of the reactor core are to prevent DNB and to prevent fuel melting for Condition I and II events. The protective systems described in Chapter 7 are designed to meet these bases. The response of the core to Condition II transients is given in Chapter 15.

4.4.4.5 Analytical Techniques

4.4.4.5.1 Core Analysis

The objective of reactor core thermal design is to determine the maximum heat-removal capability in all flow subchannels and to show that the core safety limits are not exceeded using the most conservative power distribution. The thermal design takes into account local variations in dimensions, power generation, flow redistribution, and mixing. VIPRE-01 is a realistic three-dimensional matrix model which has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core (Reference 27). The behavior of the hot assembly is determined by superimposing the power distribution among the assemblies upon the inlet flow distribution while allowing for flow mixing and flow distribution between assemblies. The local variations in power, fuel rod and pellet fabrication, and mixing within the hottest assembly are superimposed on the average conditions of the hottest assembly in order to determine the conditions in the hot channel.

4.4.4.5.2 Steady State Analysis

The VIPRE-01 computer program and subchannel analysis methodology, as approved by the NRC (Reference 27) is used to determine coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distributions within the reactor core hot subchannel under all expected operating conditions. The VIPRE-01 code is described in detail in Reference 27, including models and correlations used.

4.4.4.5.3 Experimental Verification

Experimental verification of VIPRE-01 is presented in References 33 and 27. The VIPRE-01 analysis methodology is based on a knowledge and understanding of the heat transfer and hydrodynamic behavior of the coolant flow and the mechanical characteristics of the fuel elements. VIPRE-01 analysis provides a realistic evaluation of the core performance and is used in the thermal analyses as described above.

4.4.4.5.4 Transient Analysis

The approved VIPRE-01 methodology (Reference 27) was shown to be conservative for transient thermal-hydraulic analysis.

4.4.4.6 Hydrodynamic and Flow Power Coupled Instability

Boiling flows may be susceptible to thermohydrodynamic instabilities [72]. These instabilities are undesirable in reactors since they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components. Therefore, a thermohydraulic design was developed which states that modes of operation under Condition I and II events shall not lead to thermohydrodynamic instabilities.

Two specific types of flow instabilities are considered for Westinghouse PWR operation. These are the Ledinegg or flow excursion type of static instability and the density wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flow rate from one steady state to another. This instability occurs[72] when the slope of the reactor coolant system pressure drop-flow rate curve $(\partial \Delta P / \partial G_{internal})$ becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve $(\partial \Delta P / \partial G_{external})$. The criterion for stability is thus $(\partial \Delta P / \partial G_{internal} > \partial \Delta P / \partial G_{external})$. The Westinghouse pump head curve has a negative slope $(\partial \Delta P / \partial G_{external} < 0)$, whereas the reactor coolant system pressure drop-flow curve has a positive slope $(\partial \Delta P / \partial G_{internal} > 0)$ over the Condition I and Condition II operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody[73]. Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single-phase region and causes quality or void perturbations in the two-phase regions which travel up the channel with the flow. The quality and length perturbations in the two-phase region create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by characteristics of the fluid system external to the core, the two-phase pressure drop perturbation feeds back to the single-phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by Ishii[74] for parallel closed channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability. This method had been used to assess the stability of typical Westinghouse reactor

designs[70,71,75,76,77] under Condition I and II operation. The results indicate that a large margin to density wave instability exists, e.g., increases on the order of 200% of rated reactor power would be required for the predicted inception of this type of instability.

The application of the method of Ishii[74] to Westinghouse reactor designs is conservative due to the parallel open channel feature of Westinghouse PWR cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high power density to low power density channels. There is also energy transfer from channels of high power density to lower power density channels. This coupling with cooler channels has led to the opinion that an open channel configuration is more stable than the above closed channel analysis under the same boundary conditions. Flow stability tests[77] have been conducted where the closed channel systems were shown to be less stable than when the same channels were cross connected at several locations. The cross connections were such that the resistance to channel-to-channel crossflow and enthalpy perturbations would be greater than that which would exist in a PWR core which has a relatively low resistance to crossflow.

Flow instabilities which have been observed have occurred almost exclusively in closed channel systems operating at low pressures. Kao, Morgan and Parker[77b] analyzed parallel closed channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Additional evidence that flow instabilities do not adversely affect thermal margin is provided by the data from the rod bundle DNB tests. Many Westinghouse rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundle.

In summary, it is concluded that thermohydrodynamic instabilities will not occur under Condition I and II modes of operation for W PWR reactor designs. A large power margin, greater than doubling rated power, exists to predicted inception of such instabilities. An analysis has been performed which shows that minor plant to plant differences in Westinghouse reactor designs such as fuel assembly arrays, core power to flow ratios, fuel assembly length, etc., will not result in gross deterioration of the above power margins.

4.4.4.7 Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior are more pronounced than external blockages of the same magnitude. In both cases, the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where it occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools. Inspection of the DNB correlations (Subsection 4.4.2 and Reference 8) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

Thermal-hydraulic codes are capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs.

In Reference 49, it is shown that for a fuel assembly similar to the Westinghouse design, the flow distribution within the fuel assembly when the inlet nozzle is completely blocked can be accurately predicted. Full recovery of the flow was found to occur about 30 inches downstream of the blockage. With the reference reactor operating at the nominal full power conditions specified in Table 4.4-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching a minimum DNBR below the safety analysis limit. From a review of the open literature, it is concluded that flow blockage in "open lattice cores" similar to the CPNPP cores causes flow perturbations which are local to the blockage. For instance, Ohtsubo [78], et. al., show that the mean bundle velocity is approached asymptotically about four inches downstream from a flow blockage in a single flow cell. Similar results were also found for two and three cells completely blocked. Basmer [79], et. al., tested an open lattice fuel assembly in which 41 percent of the subchannels were completely blocked in the center of the test bundles between spacer grids. Their results show the stagnant zone behind the flow blockage essentially disappears after 1.65 L/De or about 5 inches for their test bundle. They also found that leakage flow through the blockage tended to shorten the stagnant zone or, in essence, the complete recovery length. Thus, local flow blockages within a fuel assembly have little effect on subchannel enthalpy rise. The reduction in local mass velocity is then the main parameter which affects the DNBR. If the plant was operating at full power and nominal steady state conditions as specified in Table 4.4-1, a substantial reduction in local mass velocity would be required to reduce the DNBR close to the DNBR Safety Analysis Limits. The above mass velocity effect on the DNB correlations was based on the assumption of fully developed flow along the full channel length. In reality, a local flow blockage is expected to promote turbulence and thus would likely not affect DNBR at all.

Coolant flow blockages induce local crossflows as well as promote turbulence. Fuel rod behavior is changed under the influence of a sufficiently high crossflow component. Fuel rod vibration could occur, caused by this crossflow component, through vortex shedding or turbulent mechanisms. If the crossflow velocity exceeds the limit established for fluid elastic stability, large amplitude whirling results. The limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. The crossflow velocity required to exceed fluid elastic stability limits is dependent on the axial location of the blockage and the characterization of the crossflow (jet flow or not). These limits are greater than those for vibratory fuel rod wear. Crossflow velocity above the established limits can lead to mechanical wear of the fuel rods at the grid support locations. Fuel rod wear due to flow induced vibration is considered in the fuel rod fretting evaluation (Section 4.2).

4.4.5 TESTING AND VERIFICATION

4.4.5.1 Tests Prior to Initial Criticality

A reactor coolant flow test is performed following fuel loading but prior to initial criticality. Reactor coolant flow is measured using data obtained from installed elbow tap differential pressure instrumentation. This data allows determination of the coolant flow rates at reactor operating conditions. This test verifies that proper coolant flow rates have been used in the core thermal and hydraulic analysis.

4.4.5.2 Initial Power and Plant Operation

Core power distribution measurements are made at several core power levels (see Chapter 14). These tests are used to insure that conservative peaking factors are used in the core thermal and hydraulic analysis.

4.4.5.3 Component and Fuel Inspections

Inspections performed on the manufactured fuel are delineated in Section 4.2.4. Fabrication measurements critical to thermal and hydraulic analysis are obtained to verify that the engineering hot channel factors employed in the design analyses (Section 4.4.2.1.4) are met.

4.4.6 INSTRUMENTATION REQUIREMENTS

4.4.6.1 Incore Instrumentation

Instrumentation is located in the core so that by correlating movable neutron detector information with fixed thermocouple information, radial, axial, and azimuthal core characteristics may be obtained for all core quadrants.

The incore instrumentation system is comprised of thermocouples, positioned to measure fuel assembly coolant outlet temperatures at preselected positions, and fission chamber detectors, positioned in guide thimbles which run the length of selected fuel assemblies to measure the neutron flux distribution. Figure 4.4-20 shows the number and location of instrumented assemblies in the core.

The core-exit thermocouples provide a backup to the flux monitoring instrumentation for monitoring power distribution. The routine, systematic collection of thermocouple readings by the operator provides a data base. From this data base, abnormally high or abnormally low readings, quadrant temperature tilts, or systematic departures from a prior reference map can be deduced.

The movable incore neutron detector system would be used for more detailed mapping if the thermocouple system indicated an abnormality. These two complementary systems are more useful when taken together than either system alone would be. The incore instrumentation system is described in more detail in Section 7.7.1.9.

The incore instrumentation is provided to obtain data from which fission power density distribution in the core, coolant enthalpy distribution in the core, and fuel burnup distribution may be determined.

4.4.6.2 Overtemperature and Overpower N-16 Instrumentation

The overtemperature N-16 trip protects the core against low DNBR. The overpower N-16 trip protects against excessive power (fuel rod rating protection).

As discussed in Section 7.2.1.1.2, factors included in establishing the overtemperature N-16 and overpower N-16 trip setpoint include the reactor coolant temperature in each loop and the axial distribution of core power through the use of the four section excore neutron detectors.

4.4.6.3 Instrumentation to Limit Maximum Power Output

The output of the three ranges (source, intermediate, and power) of the detectors, with the electronics of the nuclear instruments, is used to limit the maximum power output of the reactor within their respective ranges.

There are six radial locations containing a total of eight neutron flux detectors installed around the reactor in the primary shield, and two proportional counters for the source range installed on opposite “flat” portions of the core containing the primary startup sources at an elevation approximately one quarter of the core height. Two compensated ionization chambers for the intermediate range, located in the same instrument wells and detector assemblies as the source range detectors, are positioned at an elevation corresponding to one half of the core height; also positioned are four, four-section uncompensated ionization chamber assemblies for the power range installed vertically at the four corners of the core and located equidistant from the reactor vessel at all points and, to minimize neutron flux pattern distortions, within one foot of the reactor vessel. Each power range detector provides two signals corresponding to the neutron flux in the upper and in the lower sections of the core. The three ranges of detectors are used as inputs to monitor neutron flux from a completely shutdown condition to 120 percent of full power with the capability of recording overpower excursions up to 200 percent of full power.

The output of the power range channels is used for:

1. The rod speed control function,

2. To alert the operator to an excessive power imbalance between the quadrants,

3. Protect the core against rod ejection accidents, and

4. Protect the core against adverse power distributions.

Details of the neutron detectors and nuclear instrumentation design and the control and trip logic are given in Chapter 7. The trip setpoints are given in the Technical Specifications.

4.4.6.4 Loose Parts Monitoring System

The Loose Parts Monitoring System uses an array of active and passive accelerometers to detect metal to metal impacts indicative of loose parts in the RCS. The accelerometers are permanently installed at selected locations where a loose part would tend to collect or impact. The locations of the accelerometers provide a high degree of reliability in the detection of metal to metal impact in the reactor coolant system. Two accelerometers are located at each of the following locations:

1. Reactor Pressure Vessel - Upper Head Region

2. Reactor Pressure Vessel - Lower Region

3. All Steam Generators - Primary Coolant inlet Region

One accelerometer is located at each of the following locations:

1. All Reactor Coolant Pumps

2. All Steam Generators - Upper Region

The major source of noise is expected to be due to primary flow turbulence, reactor coolant pump vibrations, feedwater and steam flow turbulences. A spectral comparison of the measured local metal-to-metal acoustical resonances and the normal background is performed to minimize the noise-effect on the LPMS. From this comparison, narrow band filters are tuned to eliminate/ minimize the effects of background noise on the LPMS. In addition automatic gain control is used to maintain background noise at fairly constant level to allow easier detection of metal-to-metal impact. (The accelerometers operate on the piezoelectric crystal effect with deformation of the crystal producing a measurable proportional charge. This signal is amplified, filtered, and conditioned to accentuate the frequency band known by measurement to correspond to metal to metal impacts.) The active channels are designed to alarm the presence of unusual noises above normal background noise present in the plant. Baseline data will be taken during the initial startup of the plant to allow determination of the appropriate alarm settings for each channel and to use it later as a comparison for loose part detection and analysis. The system comprises eight active and twelve passive accelerometer channels. The active channels are monitored continuously for detection of loose parts. The passive channels are used for diagnostics (location determination) of a loose part that has been detected. Detection of a loose part by the active channels will activate an audible alarm in the main control room and the data is digitally recorded on a hard disk. The LPMS microprocessor is capable of printing out noise location information. The control room operator responds to the alarm upon referring to his annunciator response manual and no additional operator training program is necessary.

The LPMS vendor has seismically analyzed the LPMS and found it to be functional following an OBE. CPNPP has installed the system as a non safety related system.

The retention of plant records including those of the LPMS is addressed in plant administrative procedures.

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TABLE 4.4-1
THERMAL AND HYDRAULIC COMPARISON TABLE

(Sheet 1 of 2)

Design Parameters[a]	Pre-Uprate Design	Uprate Design
Reactor core heat output (MW_t)[b]	3458	3612
Reactor core heat output (10^6) BTU/h)[b]	11,799	12,325
Heat generated in fuel (%)	97.4	97.4
System pressure, nominal (psia)	2250	2250
Minimum DNBR at nominal conditions		
Typical flow channel	>2.46	>2.46
Thimble (cold wall) flow channel	>2.34	>2.34
Minimum DNBR for design transients		
Typical flow channel		
WRB-1	1.23	1.23
WRB-2	1.23	1.23
Thimble (cold wall) flow channel		
WRB-1	1.23	1.23
WRB-2	1.22	1.22
DNB correlations[c]	WRB-2, W-3 & WRB-1	WRB-2, W-3 & WRB-1
Coolant conditions		
Vessel minimum measured flow rate (MMF		
gpm	396,400	396,400
Vessel thermal design flow rate (TDF)[c]		
10^6 lbm/h	142.04	142.3
gpm	382,800	382,800
Effective flow-rate for heat transfer		
(based on TDF, excluding bypass)[d]		
10^6 lbm/h	133.80	134.04
gpm	360,598	360,598
Effective flow area for heat transfer (ft^2)	54.13	54.13
Average mass velocity along fuel rods		
(based on TDF, excluding bypass)[d]		
ft/s	14.84	14.84
Average mass velocity		
(based on TDF, excluding bypass)[d]		
10^6 lbm/h-ft^2	2.47	2.48

TABLE 4.4-1
THERMAL AND HYDRAULIC COMPARISON TABLE

(Sheet 2 of 2)

Design Parameters[a]	Pre-Uprate Design	Uprate Design
Coolant temperature		
Nominal inlet (°F)	559.2	558.0
Average rise in vessel (°F)	60.0	62.4
Average rise in core (°F)	63.3	65.8
Average in core (°F)	592.6	592.8
Average in vessel (°F)	589.2	589.2
Heat transfer		
Active heat transfer surface area (ft^2)	57,505	57,505
Average heat flux (BTU/h-ft^2)	199,900	208,802
Maximum heat flux for normal[e][f]operation (BTU/h-ft^2)	483,758	522,005
Average linear power (kW/ft)[f]	5.52	5.77
Peak linear power for normal operation (kW/ft)[e][f]	13.36	14.41
Peak linear power resulting from overpower transients/operator errors, assuming a maximum overpower of 118.5% (kW/ft)	$\leq$22.4	$\leq$22.4
Peak linear power for prevention of centerline melt (kW/ft)	22.4	22.4
Fuel central temperature Peak at peak linear power for prevention of centerline melt (°F)	4,700	4,700
Pressure drop across core[g]	29.1	28.9

a) Unit 1 flow rates are used for comparison. The lower Unit 1 flow rates are more limiting with respect to DNB. Also, these flows are to be used for both units for uprate conditions.

b) The proposed power level of 3612 MW_t has been used for all thermal-hydraulic design analyses.

c) See paragraph 4.4.2.1.1 for the use of the W-3 correlation.

d) Based on thimble plugs inserted.

e) Based on maximum F_Q of 2.42 for current design and 2.50 for uprate design.

f) Based on densified active fuel length.

g) Based on best estimate flow and design bypass flow rates.

TABLE 4.4-1B
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TABLE 4.4-2
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TABLE 4.4-3
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TABLE 4.4-4
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4.5 REACTOR MATERIALS

4.5.1 CONTROL ROD SYSTEM STRUCTURAL MATERIALS

4.5.1.1 Materials Specifications

All parts exposed to reactor coolant are made of metals which resist the corrosive action of the water. Three types of metals are used exclusively: stainless steels, nickel-chrome-iron and cobalt based alloys. In the case of stainless steels, only austenitic and martensitic stainless steels are used, the martensitic stainless steels are not used in the heat treated conditions which cause susceptibility to stress corrosion cracking or accelerated corrosion in the Westinghouse pressurized water reactor (PWR) water chemistry.

1. Pressure vessel

   All pressure containing materials comply with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, and are fabricated from austenitic (Type F304LN for Unit 1, Type 304 for Unit 2) stainless steel.

2. Coil stack assembly

   The coil housings require a magnetic material. Both low carbon cast steel and ductile iron have been successfully tested for this application. The choice, made on the basis of cost, indicates that ductile iron will be specified on the control rod drive mechanism (CRDM). The finished housings are zinc plated or flame sprayed to provide corrosion resistance.

   Coils are wound on bobbins of molded Dow Corning 302 material, with double glass insulated copper wire. Coils are then vacuum impregnated with silicon varnish. A wrapping of mica sheet is secured to the coil outside diameter. The result is a well insulated coil capable of sustained operation at 200 degrees centigrade.

3. Latch assembly

   Magnetic pole pieces are fabricated from Type 410 stainless steel. All non-magnetic parts, except pins and springs, are fabricated from Type 304 stainless steel. Haynes 25 is used to fabricate link pins. Springs are made from nickel-chrome-iron alloy (Inconel-X). Latch arm tips are clad with Stellite-6 to provide improved wearability. Hard chrome plate and Stellite-6 are used selectively for bearing and wear surfaces.

4. Drive rod assembly

   The drive rod assembly utilizes a Type 410 stainless steel drive rod. The coupling is machined from Type 403 stainless steel. Other parts are Type 304 stainless steel with the exception of the springs which are nickel-chrome-iron alloy and the locking button which is Haynes 25.

4.5.1.2 Fabrication and Processing of Austenitic Stainless Steel Components

The discussions provided in Section 5.2.3 concerning the processes, inspections, and tests on austenitic stainless steel components to assure freedom from increased susceptibility to intergranular corrosion caused by sensitization, and the discussions provided in Section 5.2.3 on the control of welding of austenitic stainless steels, especially control of delta ferrite, are applicable to the austenitic stainless steel pressure housing components of the CRDM.

4.5.1.3 Contamination Protection and Cleaning of Austenitic Stainless Steel

The CRDM's are cleaned prior to delivery in accordance with the guidance of ANSI 45.2.1. Process specifications in packaging and shipment are discussed in Section 5.2.3. Although the procedure at the construction site is not in the Westinghouse Nuclear Steam Supply System scope of supply, Westinghouse personnel do conduct surveillance of these operations to assure that manufacturers and installers adhere to appropriate requirements as discussed in Section 5.2.3.

4.5.2 REACTOR INTERNALS MATERIALS

4.5.2.1 Materials Specifications

All the major material for the reactor internals is Type 304 stainless steel. Parts not fabricated from Type 304 stainless steel include bolts and dowel pins which are fabricated from Type 316 stainless steel and radial support key bolts which are fabricated of Inconel- 750. These materials are listed in Table 5.2-4. There are no other materials used in the reactor internals or core support structures which are not otherwise included in the ASME Code, Section III, Appendix I.

4.5.2.2 Controls on Welding

The discussions provided in Section 5.2.3 are applicable to the welding of reactor internals and core support components.

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products and Fittings

The discussion provided in Appendix 1A(N) verifies conformance of reactor internals and core support structures with Regulatory Guide 1.66.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel Components

The discussions provided in Section 5.2.3 and Appendix 1A(N) verify conformance of reactor internals and core support structures with Regulatory Guide 1.44.

Regulatory Guide 1.36 is not applicable to the reactor vessel internals since no insulation material of any kind is used on these structures.

The discussions provided in Section 5.2.3 and Appendix 1A(N) verify conformance of reactor internals and core support structures with Regulatory Guide 1.31.

The discussion provided in Appendix 1A(N) verifies conformance of reactor internals with Regulatory Guide 1.34.

The discussion provided in Appendix 1A(N) verifies conformance of reactor internals and core support structures with Regulatory Guide 1.71.

4.5.2.5 Contamination Protection and Cleaning of Austenitic Stainless Steel

The discussions provided in Section 5.2.3 and Appendix 1A(N) are applicable to the reactor internals and core support structures and verify conformance with ANSI 45 specifications and Regulatory Guide 1.37.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

4.6.1 INFORMATION FOR CONTROL ROD DRIVE SYSTEM (CRDS)

The CRDS is described in Section 3.9N.4.1. Figures 3.9N-5 and 3.9N-6 provide the details of the control rod drive mechanisms, and Figure 4.2-8 provides the layout of the CRDS. No hydraulic system is associated with its functioning. The instrumentation and controls for the Reactor Trip System are described in Section 7.2 and the Reactor Control System is described in Section 7.7.

4.6.2 EVALUATION OF THE CRDS

The CRDS has been analyzed in detail in a failure mode and effects analysis (FMEA) [1]. This study, and the analyses presented in Chapter 15, demonstrates that the CRDS performs its intended safety function, a reactor trip, by putting the reactor in a subcritical condition when a safety system setting is approached, with any assumed credible failure of a single active component. The essential elements of the CRDS (those required to ensure reactor trip) are isolated from non-essential portions of the CRDS (the Rod Control System) as described in Section 7.2.

Despite the extremely low probability of a common mode failure impairing the ability of the Reactor Trip System to perform its safety function, analyses have been performed in accordance with the requirements of WASH-1270. These analyses documented in References [2] and [3] have demonstrated that acceptable safety criterion would not be exceeded even if the CRDS were rendered incapable of functioning during a reactor transient for which their function would normally be expected.

The design of the control rod drive mechanism is such that failure of the control rod drive mechanism cooling system will, in the worst case, result in an individual control rod trip or a full reactor trip (see Section 9.2).

4.6.3 TESTING AND VERIFICATION OF THE CRDS

The CRDS is extensively tested prior to its operation. These tests may be subdivided into five categories, 1) prototype tests of components, 2) prototype CRDS tests, 3) production tests of components following manufacture and prior to installation, 4) onsite preoperational and initial startup tests, and 5) periodic inservice tests. These tests which are described in Sections 3.9N.4.4, 4.2, 14.2, and the Technical Specifications are conducted to verify the operability of the CRDS when called upon to function.

4.6.4 INFORMATION FOR COMBINED PERFORMANCE OF REACTIVITY SYSTEMS

As is indicated in Chapter 15, the only postulated events which assume credit for reactivity control systems other than a reactor trip to render the plant subcritical are the steam line break, feedwater line break, and loss of coolant accident. The reactivity control systems for which credit is taken in these accidents are the reactor trip and the Safety Injection System (SIS). Additional information on the CRDS is presented in Section 3.9N.4 and on the SIS in Section 6.3. Note that no credit is taken for the boration capabilities of the Chemical and Volume Control System (CVCS) as a system in the analysis of transients presented in Chapter 15. Information on the capabilities of the CVCS is provided in Section 9.3.4. The adverse boron dilution possibilities due to the operation of the CVCS are investigated in Section 15.4.6. Prior proper operation of

the CVCS has been presumed as an initial condition to evaluate transients and appropriate Technical Specifications have been prepared to ensure the correct operation or remedial action.

4.6.5 EVALUATION OF COMBINED PERFORMANCE

The evaluation of the steam line break, feedwater line break and loss of coolant accident which presume the combined actuation of the Reactor Trip System to the CRDS and the SIS are presented in Sections 15.1.5, 15.2.8, and 15.6.5. Reactor trip signals and safety injection signals for these events are generated from functionally diverse sensors and actuate diverse means of reactivity control, i.e., control rod insertion and injection of soluble poison.

Non-diverse but redundant types of equipment are only utilized in the processing of the incoming sensor signals into appropriate logic which initiates the protective action. This equipment is described in detail in Section 7.2 and 7.3. In particular, note that protection from equipment failures is provided by redundant equipment and periodic testing. Effects of failures of this equipment have been extensively investigated as reported in Reference [4]. This FMEA verifies that any single failure will not have a deleterious effect upon the Engineering Safety Features Actuation System. Adequacy of the Emergency Core Cooling System and SIS performance under faulted conditions is verified in Section 6.3.

4.6.6 STARTUP REPORTS

A summary report of unit startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level unless the power increase is based on measurement uncertainty recapture within the existing safety analyses, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the unit.

The initial Startup Report shall address each of the startup tests identified in Chapter 14 of the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications. Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e, initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

REFERENCES

1. Shopsky, W. E., “Failure Mode and Effects Analysis (FMEA) of the Solid State Full Length Rod Control System,” WCAP-8976, September 1977.

2. “Westinghouse Anticipated Transients Without Trip Analysis,” WCAP-8330, August 1974.

3. Gangloff, W. C. and Loftus, W. D., “An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients,” WCAP-7706-L (Proprietary) and WCAP-7706 (Non-Proprietary), February 1971.

4. Eggleston, F. T., Rawlins, D. H., Petrow, J. R., “Failure Mode and Effects Analysis (FMEA) of the Engineering Safeguard Features Actuation System,” WCAP-8584 (Proprietary) and WCAP-8760 (Non- Proprietary), April 1976.