DESIGN CRITERIA - STRUCTURES, COMPONENT, EQUIPMENT AND SYSTEMS

CHAPTER 3

TABLE OF CONTENTS

Section	Title	Page
3.0	DESIGN CRITERIA-STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS	3.1-1
3.1	CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA	3.1-1
3.1.1	CRITERION 1 - QUALITY STANDARDS AND RECORDS	3.1-1
3.1.2	CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA	3.1-1
3.1.3	CRITERION 3 - FIRE PROTECTION	3.1-2
3.1.4	CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES	3.1-3
3.1.5	CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS OR COMPONENTS	3.1-4
3.1.10	CRITERION 10 - REACTOR DESIGN	3.1-5
3.1.11	CRITERION 11 - REACTOR INHERENT PROTECTION	3.1-5
3.1.12	CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS	3.1-6
3.1.13	CRITERION 13 - INSTRUMENTATION AND CONTROL	3.1-6
3.1.14	CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY	3.1-7
3.1.15	CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN	3.1-8
3.1.16	CRITERION 16 - CONTAINMENT DESIGN	3.1-9
3.1.17	CRITERION 17 - ELECTRICAL POWER SYSTEMS	3.1-9
3.1.18	CRITERION 18 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS	3.1-10
3.1.19	CRITERION 19 - CONTROL ROOM	3.1-11
3.1.20	CRITERION 20 - PROTECTION SYSTEM FUNCTIONS	3.1-11
3.1.21	CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY	3.1-12
3.1.22	CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE	3.1-12
3.1.23	CRITERION 23 - PROTECTION SYSTEM FAILURE MODES	3.1-13

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.1.24	CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS	3.1-13
3.1.25	CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS	3.1-14
3.1.26	CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY	3.1-14
3.1.27	CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY	3.1-15
3.1.28	CRITERION 28 - REACTIVITY LIMITS	3.1-15
3.1.29	CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES	3.1-16
3.1.30	CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY	3.1-16
3.1.31	CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY	3.1-16
3.1.32	CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY	3.1-17
3.1.33	CRITERION 33 - REACTOR COOLANT MAKEUP	3.1-17
3.1.34	CRITERION 34 - RESIDUAL HEAT REMOVAL	3.1-18
3.1.35	CRITERION 35 - EMERGENCY CORE COOLING	3.1-18
3.1.36	CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM	3.1-19
3.1.37	CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM	3.1-19
3.1.38	CRITERION 38 - CONTAINMENT HEAT REMOVAL	3.1-20
3.1.39	CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM	3.1-20
3.1.40	CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM	3.1-21
3.1.41	CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP	3.1-21

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.1.42	CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEM	3.1-22
3.1.43	CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS	3.1-23
3.1.44	CRITERION 44 - COOLING WATER	3.1-23
3.1.45	CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM	3.1-24
3.1.46	CRITERION 46 - TESTING OF COOLING WATER SYSTEM	3.1-24
3.1.50	CRITERION 50 - CONTAINMENT DESIGN BASIS	3.1-25
3.1.51	CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY	3.1-25
3.1.52	CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING	3.1-26
3.1.53	CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION	3.1-26
3.1.54	CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT	3.1-27
3.1.55	CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT	3.1-27
3.1.56	CRITERION 56 - PRIMARY CONTAINMENT ISOLATION	3.1-28
3.1.57	CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES	3.1-29
3.1.60	CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT	3.1-29
3.1.61	CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL	3.1-30
3.1.62	CRITERION 62 – PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING	3.1-31
3.1.63	CRITERION 63 - MONITORING FUEL AND WASTE STORAGE	3.1-31
3.1.64	CRITERION 64 - MONITORING RADIOACTIVITY RELEASES	3.1-31
	REFERENCES	3.1-32
3.2	CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS	3.2-1
3.2.1	SEISMIC CLASSIFICATION	3.2-1

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.2.2	SYSTEM QUALITY GROUP CLASSIFICATION	3.2-2
3.3	WIND AND TORNADO LOADINGS	3.3-1
3.3.1	WIND LOADINGS	3.3-1
3.3.2	TORNADO LOADINGS	3.3-2
	REFERENCES	3.3-5
3.4	WATER LEVEL (FLOOD) DESIGN	3.4-1
3.4.1	FLOOD PROTECTION	3.4-1
3.4.2	ANALYSIS PROCEDURES	3.4-5
3.4.3	RAB INTERNAL FLOODING DUE TO EQUIPMENT RUPTURE	3.4-5
3.5	MISSILE PROTECTION	3.5-1
3.5.1	MISSILE SELECTION AND DESCRIPTIONS	3.5-1
3.5.2	STRUCTURES, SYSTEMS AND COMPONENTS TO BE PROTECTED FROM EXTERNALLY GENERATED MISSILES	3.5-12
3.5.3	BARRIER DESIGN PROCEDURES	3.5-12
	REFERENCES	3.5-18
3.6	PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE RUPTURE OF PIPING	3.6-1
3.6.1	POSTULATED PIPING FAILURES IN FLUID SYSTEMS	3.6-3
3.6.2	DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING	3.6-8
	REFERENCES	3.6-24
3.6A	HIGH ENERGY PIPE RUPTURE ANALYSIS-INSIDE CONTAINMENT	3.6A-1
3.6A.1	MAIN STEAM AND FEEDWATER INSIDE CONTAINMENT	3.6A-2
3.6A.2	REACTOR COOLANT SYSTEM (INCLUDING PRESSURIZER SURGE, SPRAY AND RELIEF PIPING)	3.6A-5
3.6A.3	SAFETY INJECTION SYSTEM	3.6A-9
3.6A.4	SHUTDOWN COOLING PIPING	3.6A-11

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.6A.5	CHEMICAL AND VOLUME CONTROL SYSTEM INSIDE CONTAINMENT (LETDOWN/AND CHARGING LINE)	3.6A-12
3.6A.6	STEAM GENERATOR BLOWDOWN SYSTEM INSIDE CONTAINMENT	3.6A-13
3.6B	HIGH ENERGY PIPE RUPTURE ANALYSIS - OUTSIDE CONTAINMENT	3.6B-1
3.6B.1	MAIN STEAM AND FEEDWATER OUTSIDE CONTAINMENT	3.6B-2
3.6B.2	CHEMICAL AND VOLUME CONTROL SYSTEM OUTSIDE CONTAINMENT (LETDOWN AND CHARGING LINE)	3.6B-4
3.6B.3	STEAM GENERATOR BLOWDOWN SYSTEM OUTSIDE CONTAINMENT	3.6B-6
3.6B.4	AUXILIARY STEAM SYSTEM	3.6B-6
3.6B.5	AUXILIARY FEEDWATER SYSTEM	3.6B-7
3.6B.6	STEAM SUPPLY TO AUXILIARY FEED PUMP TURBINE	3.6B-8
3.6C	PIPE WHIP RESTRAINTS AND BREAK LOCATIONS	3.6C-1
3.6D	STRUCTURAL DETAILS OF PIPE WHIP RESTRAINTS	3.6D-1
3.6E	MAIN STEAM & FEEDWATER ANALYSIS	3.6E-1
	REFERENCES	3.6E-3
3.6F	MODERATE ENERGY PIPING FAILURE ANALYSIS	3.6F-1
3.6F.1	MODERATE ENERGY PIPING FAILURE - INSIDE CONTAINMENT	3.6F-2
3.6F.2	MODERATE ENERGY PIPING FAILURES - OUTSIDE CONTAINMENT	3.6F-2
3.7	SEISMIC DESIGN	3.7-1
3.7.1	INPUT CRITERIA	3.7-1
3.7.2	SEISMIC SYSTEM ANALYSIS	3.7-4
3.7.3	SEISMIC SUBSYSTEM ANALYSIS	3.7-23
3.7.4	SEISMIC INSTRUMENTATION	3.7-53
	REFERENCES	3.7-55
3.8	DESIGN OF CATEGORY I STRUCTURES	3.8-1

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.8.1	CONCRETE CONTAINMENT	3.8-1
3.8.2	STEEL CONTAINMENT	3.8-1
3.8.3	CONCRETE AND STEEL INTERNAL STRUCTURE OF STEEL CONTAINMENT	3.8-38
3.8.4	OTHER SEISMIC CATEGORY I STRUCTURES	3.8-58
3.8.5	FOUNDATIONS	3.8-72
	REFERENCES	3.8-75
3.8A	EVALUATION OF CONCRETE MASONRY WALLS	3.8A-1
3.8A.1	SUMMARY	3.8A-2
3.8A.2	WALL REINFORCING	3.8A-2
3.8A.3	ANALYTICAL MODEL	3.8A-2
3.8A.4	INTERSTORY DRIFT CONSIDERATIONS	3.8A-3
3.8A.5	LOAD COMBINATIONS AND ALLOWABLE STRESSES	3.8A-4
3.8A.6	ATTACHMENT TO WALLS	3.8A-5
	REFERENCE	3.8A-8
3.9	MECHANICAL SYSTEMS AND COMPONENTS	3.9-1
3.9.1	SPECIAL TOPICS FOR MECHANICAL COMPONENTS	3.9-1
3.9.2	DYNAMIC SYSTEM ANALYSIS AND TESTING	3.9-27
3.9.3	ASME CODE CLASS 1, 2 AND 3 COMPONENTS AND COMPONENT SUPPORTS (INCLUDING ASME CODE CLASS 1, 2 & 3 PIPING AND PIPE SUPPORTS)	3.9-41
3.9.4	CONTROL ELEMENT DRIVE MECHANISMS	3.9-58
3.9.5	REACTOR PRESSURE VESSEL INTERNALS	3.9-62
3.9.6	INSERVICE TESTING OF PUMPS AND VALVES	3.9-68
	REFERENCES	3.9-69
3.9A	OPERABILITY CONSIDERATIONS FOR SEISMIC CATEGORY I ACTIVE PUMPS AND VALVES	3.9A-1
3.9A.1	MATHEMATICAL ANALYSIS METHOD	3.9A-1

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.9B	CONCRETE EXPANSION ANCHOR DESIGN	3.9B-1
3.9B.1	LOADS	3.9B-2
3.9B.2	LOAD COMBINATIONS	3.9B-2
3.9B.3	ALLOWABLE LOADS	3.9B-2
3.9B.4	DETERMINATION OF PRYING FORCES	3.9B-3
3.9B.5	DETERMINATION OF ANCHOR SIZES	3.9B-3
3.9B.6	BASEPLATE FLEXIBILITY ANALYSIS	3.9B-3
3.9B.7	DESIGN OF PIPE RESTRAINTS USING EXPANSION ANCHORS	3.9B-6
3.10	SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT	3.10-1
3.10.1	SEISMIC QUALIFICATION CRITERIA	3.10-1
3.10.2	METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION	3.10-2
3.10.3	METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS FOR ELECTRICAL EQUIPMENT AND INSTRUMENTATION	3.10-3
3.10.4	OPERATING LICENSE REVIEW	3.10-8
	REFERENCES	3.10-9
3.10A	CRITERIA FOR SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT AND THEIR SUPPORTS	3.10A-1
3.10A.1	SEISMIC DESIGN CRITERIA	3.10A-1
3.10A.2	SEISMIC ANALYSES, TESTING PROCEDURES AND RESTRAINT MEASURES	3.10A-1
3.11	ENVIRONMENTAL QUALIFICATION	3.11-1
3.11.1	INTRODUCTION	3.11-1
3.11.2	CRITERIA	3.11-2
3.11.3	IDENTIFICATION OF COMPONENTS	3.11-3
3.11.4	QUALIFICATION OF COMPONENTS	3.11-4
3.11.5	MAINTENANCE	3.11-4

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

Section	Title	Page
3.11.6	RECORDS/QUALITY ASSURANCE	3.11-5
3.11.7	CONCLUSIONS	3.11-5

DESIGN CRITERIA-STRUCTURES, COMPONENTS EQUIPMENT AND SYSTEMS

CHAPTER 3

LIST OF TABLES

Table	Title	Page
3.2-1	DESIGN CLASSIFICATIONS OF STRUCTURES, SYSTEMS AND COMPONENTS	T3.2-1
3.2-2	MINIMUM CODE REQUIREMENTS FOR QUALITY GROUPS	T3.2-12
3.3-1	WIND SPEEDS AND RESULTANT STATIC PRESSURE LOADINGS	T3.3-1
3.4-1	RAB CATASTROPHIC FLOODING ANALYSIS	T3.4-1
3.5-1	MISSILES OUTSIDE CONTAINMENT GENERATED BY HIGH ENERGY SYSTEMS	T3.5-1
3.5-2	MISSILES OUTSIDE CONTAINMENT FROM FAILURE OF OVERSPEED PROTECTION	T3.5-3
3.5-3	ENCLOSURES FOR EQUIPMENT REQUIRED FOR SAFE SHUTDOWN	T3.5-4
3.5-4	INTERNAL MISSILE PARAMETERS	T3.5-11
3.5-5	DELETED	T3.5-13
3.5-6	DELETED	T3.5-14
3.5-7	DELETED	T3.5-15
3.5-8	DELETED	T3.5-16
3.5-9	DELETED	T3.5-17
3.5-9A	DELETED	T3.5-18
3.5-9B	DELETED	T3.5-19
3.5-9C	DELETED	T3.5-20
3.5-10	DESIGN BASE SPECTRUM OF TORNADO MISSILES	T3.5-21
3.5-11	TORNADO MISSILE CONCRETE BARRIER MINIMUM THICKNESS	T3.5-22
3.5-12	ALLOWABLE DUCTILITY FACTORS	T3.5-23
3.5-13	TORNADO MISSILE IMPACTIVE ANALYSIS	T3.5-24
3.6-1	ESSENTIAL SYSTEMS TO MITIGATE CONSEQUENCES OF POSTULATED PIPING FAILURES	T3.6-1
3.6-2	COLD LEG PIPE STOP STIFFNESS	T3.6-2

CHAPTER 3

LIST OF TABLES (Cont'd)

Table	Title	Page
3.6-3	PIPE BREAK AREAS & BREAK OPENING TIMES-PARTIAL AREA GUILLOTINES	T3.6-3
3.6-4	ST. LUCIE NO. 2 RCS DISCHARGE LEG PIPE RESTRAINTS AXIAL GAPS	T3.6-4
3.6C-1	STRESS SUMMARY SAFETY INJECTION SYSTEM (SCI)	T3.6C-1
3.6C-2	STRESS SUMMARY SAFETY INJECTION SYSTEM (SC2)	T3.6C-3
3.6E-1	SUMMARY OF SELECTIVE PIPE WHIP RESTRAINTS AND DYNAMIC LOADS	T3.6E-1
3.6E-2	VOLUME INFORMATION USED FOR RELAP 3/MOD 68 FLUID MODELS OF MAIN STEAM LINES	T3.6E-2
3.6E-3	JUNCTION INFORMATION USED FOR RELAP 3/MOD 68 FLUID MODELS OF MAIN STEAM LINES	T3.6E-3
3.6E-4	RESTRAINT GAPS USED FOR PLAST MODELS OF MAIN STEAM LINE	T3.6E-5
3.6E-5	VOLUME INFORMATION USED FOR RELAP 4/MOD 6 FLUID MODELS OF BOILER FEEDWATER LINE	T3.6E-6
3.6E-6	JUNCTION INFORMATION USED FOR RELAP 4/MOD 6 FLUID MODELS OF BOILER FEEDWATER LINE	T3.6E-8
3.6E-7	RESTRAINT GAPS USED FOR PLAST MODELS OF BOILER FEEDWATER LINE	T3.6E-10
3.7-1	AMPLIFIED ACCELERATIONS AND DISPLACEMENTS FOR THE OPERATING BASIS EARTHQUAKE- HORIZONTAL	T3.7-1
3.7-2	PERCENT CRITICAL DAMPING	T3.7-2
3.7-3	SUPPORTING MEDIA CHARACTERISTICS FOR SEISMIC CATEGORY I STRUCTURES	T3.7-3
3.7-4	REACTOR BUILDING PROPERTIES HORIZONTAL MODEL	T3.7-4
3.7-5	REACTOR BUILDING PROPERTIES VERTICAL MODEL	T3.7-5
3.7-6	REACTOR AUXILIARY BUILDING PROPERTIES HORIZONTAL MODEL	T3.7-6
3.7-7	REACTOR AUXILIARY BUILDING PROPERTIES VERTICAL MODEL	T3.7-7
3.7-8	FUEL HANDLING BUILDING PROPERTIES HORIZONTAL MODEL	T3.7-8

CHAPTER 3

LIST OF TABLES (Cont'd)

Table	Title	Page
3.7-9	INTAKE STRUCTURE PROPERTIES HORIZONTAL MODEL	T3.7-9
3.7-10	DIESEL GENERATOR BUILDING PROPERTIES HORIZONTAL MODEL	T3.7-10
3.7-11	MAIN STEAM TRESTLE PROPERTIES HORIZONTAL MODEL	T3.7-11
3.7-12	MAIN STEAM TRESTLE PROPERTIES VERTICAL MODEL	T3.7-12
3.7-13	COMPONENT COOLING WATER BUILDING PROPERTIES HORIZONTAL MODEL	T3.7-13
3.7-14	COMPONENT COOLING WATER BUILDING PROPERTIES VERTICAL MODEL	T3.7-14
3.7-15	CONDENSATE STORAGE TANK PROPERTIES HORIZONTAL MODEL	T3.7-15
3.7-16	CONDENSATE STORAGE TANK PROPERTIES VERTICAL MODEL	T3.7-17
3.7-17	DIESEL OIL STORAGE BUILDING PROPERTIES	T3.7-18
3.7-18	NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS) REACTOR BUILDING	T3.7-20
3.7-19	NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS) REACTOR AUXILIARY BUILDING	T3.7-21
3.7-20	NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS) FUEL HANDLING BUILDING	T3.7-22
3.7-21	NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS) INTAKE STRUCTURE	T3.7-23
3.7-22	NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS) DIESEL GENERATOR BUILDING	T3.7-24
3.7-23	MS/FW TRESTLE SIGNIFICANT NATURAL FREQUENCIES	T3.7-25
3.7-24	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-26
3.7-25	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-27

CHAPTER 3

LIST OF TABLES (Cont'd)

Table	Title	Page
3.7-26	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-28
3.7-27	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-29
3.7-28	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-30
3.7-29	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-31
3.7-30	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-32
3.7-31	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-33
3.7-32	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-34
3.7-33	COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES USING RESPONSE SPECTRA AND TIME HISTORY METHODS	T3.7-35
3.7-34	COMPARISON OF STRUCTURAL RESPONSES WITH AND WITHOUT THE EFFECT OF CLOSELY SPACED MODES	T3.7-36
3.7-35	SOILS-SUPPORTED SEISMIC CATEGORY I STRUCTURES	T3.7-37
3.7-36	SIDE SPRING CONSTANTS EFFECT ON SOIL STRUCTURE INTERACTION	T3.7-38
3.7-37	COMPOSITE MODAL DAMPING RATIOS FOR SEISMIC CATEGORY I STRUCTURES REACTOR BUILDING, SSE SOIL YOUNG'S MODULUS E_S =40 KS	T3.7-39
3.7-38	MAXIMUM STRESS COMPARISON	T3.7-40
3.7-38a	HIGH STRESS COMPARISON	T3.7-41

CHAPTER 3

LIST OF TABLES (Cont'd)

Table	Title	Page
3.7-38b	HIGH STRESS COMPARISON	T3.7-42
3.7-39	NATURAL FREQUENCIES AND DOMINANT DEGREES OF FREEDOM	T3.7-43
3.7-40	COMPARISON OF CALCULATED MAXIMUM AND SPECIFIED SEISMIC LOAD	T3.7-46
3.7-41	VERTICAL SEISMIC MODEL MASS POINT LOCATIONS AND DESCRIPTION	T3.7-54
3.7-42	LATERAL SEISMIC MODEL MASS POINT LOCATIONS AND DESCRIPTIONS	T3.7-56
3.7-43	CEDM LOADS - PRESSURE HOUSING AND NOZZLE	T3.7-57
3.7-44	CEDM NOZZLE LOADS	T3.7-60
3.7-45	LATERAL DEFLECTIONS OF CEDM - NOZZLE AND PRESSURE HOUSING	T3.7-61
3.7-46	COMPARISON OF COMPUTED CEDM STRESS INTENSITIES WITH STRESS ALLOWABLES	T3.7-64
3.7-47	COMPARISON OF COMPUTED CEDM STRESS INTENSITIES WITH STRESS ALLOWABLES	T3.7-65
3.7-48	COMPARISON OF COMPUTED CEDM STRESS INTENSITIES WITH STRESS ALLOWABLES	T3.7-66
3.7-49	WATERFORD NO. 3 NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS)	T3.7-67
3.7-50	WATERFORD NO. 3 COMPARISON OF ACCELERATION OF DYNAMIC ANALYSIS WITH AND WITHOUT TORSIONAL DEGREE OF FREEDOM	T3.7-68
3.7-51	WATERFORD NO. 3 COMPARISON OF ACCELERATION OF DYNAMIC ANALYSIS WITH AND WITHOUT TORSIONAL DEGREE OF FREEDOM	T3.7-70
3.7-52	COMPARISON OF COMBINATION METHODS FOR RELATIVE SEISMIC DISPLACEMENTS	T3.7-72
3.7-53	COMPARISON OF TIME HISTORY DISPLACEMENTS TO DESIGN VALUES	T3.7-73
3.7-54	EFFECT OF T VARIATION ON COMPARISON OF TIME HISTORY DISPLACEMENTS TO DESIGN VALUES	T3.7-74

CHAPTER 3

LIST OF TABLES (Cont'd)

Table	Title	Page
3.7-55	SEISMIC DISPLACEMENT BETWEEN BUILDINGS	T3.7-75
3.8-1	CONTAINMENT VESSEL LOAD COMBINATIONS	T3.8-1
3.8-2	CONTAINMENT VESSEL PENETRATIONS-LOAD COMBINATIONS AND STRESS LIMITS	T3.8-2
3.8-3	CONTAINMENT SHELL STRESSES AT JUNCTION OF COLUMN AND KNUCKLE	T3.8-5
3.8-4	SUMMARY OF STRESSES IN BOTTOM HEAD KNUCKLE	T3.8-6
3.8-5	SUMMARY OF STRESSES IN BOTTOM HEAD KNUCKLE	T3.8-7
3.8-6	PENETRATION ANALYSIS	T3.8-8
3.8-7	CONTAINMENT VESSEL ALLOWABLE STRESSES	T3.8-9
3.8-8	SUMMARY OF HEMISPHERICAL DOME STRESSES	T3.8-13
3.8-9	SUMMARY OF CYLINDER STRESSES	T3.8-14
3.8-10	CONTAINMENT VESSEL MATERIALS	T3.8-15
3.8-11	CONTAINMENT INTERNAL CONCRETE STRUCTURE LOAD COMBINATIONS	T3.8-16
3.8-12	STEEL INTERNAL STRUCTURES - LOADING COMBINATIONS AND ALLOWABLE STRESSES	T3.8-17
3.8-13	COMPARISON OF MAXIMUM DESIGN AND ALLOWABLE STRESSES FOR UPPER STEAM GENERATOR SUPPORTS	T3.8-18
3.8-14	STEAM GENERATOR SLIDING BASE SUPPORT STRESS COMPARISON	T3.8-19
3.8-15	REACTOR SUPPORT STRUCTURE STRESS SUMMARY TYPICAL LOCA CONDITION	T3.8-20
3.8-16	COMPARISON OF MAXIMUM DESIGN AND ALLOWABLE STRESSES FOR REACTOR COOLANT PUMP SUPPORTS	T3.8-21
3.8-17	PRESSURIZER SUPPORT STRESS COMPARISON	T3.8-22
3.8-18	REACTOR COOLANT PUMP STOPS AND WIRE ROPE RESTRAINT	T3.8-23
3.8-19	CONCRETE INTERNAL STRUCTURES COMPARISON OF REQUIRED DESIGN STRENGTH AND ACTUAL CAPACITY OF STRUCTURAL ELEMENTS	T3.8-24

CHAPTER 3

LIST OF TABLES (Cont'd)

Table	Title	Page
3.8-20	NSSS SUPPORT STEEL MATERIAL SUMMARY	T3.8-25
3.8-21	LIVE LOADS	T3.8-26
3.8-22	MAIN STEAM TRESTLE STRESS COMPARISON	T3.8-28
3.8-23	COMPARISON OF REQUIRED DESIGN STRENGTH AND ACTUAL CAPACITY OF STRUCTURAL ELEMENTS	T3.8-29
3.8-24	PARTIAL LINEUP AGAINST SRP 3.8.4	T3.8-30
3.8-25	RG 1.142 (R0) SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS (OTHER THAN REACTOR VESSELS AND CONTAINMENT)	T3.8-33
3.8-26	COMPLIANCE TO SECTIONS 5.1 AND 10 OF ANSI STANDARD N101.6-1972	T3.8-39
3.8-27	REACTOR BUILDING	T3.8-40
3.8-28	REACTOR AUXILIARY BUILDING	T3.8-41
3.8-29	CONDENSATE STORAGE TANK	T3.8-42
3.8-30	FUEL HANDLING BUILDING	T3.8-43
3.8-31	DIESEL GENERATOR BUILDING	T3.8-44
3.8-32	COMPONENT COOLING	T3.8-45
3.8-33	INTAKE STRUCTURE	T3.8-46
3.8A-1	ALLOWABLE STRESSES IN REINFORCED MASONRY	3.8A-6
3.9-1	QUALITY GROUP A COMPONENTS	T3.9-1
3.9-2	TRANSIENTS USED IN DESIGN AND FATIGUE ANALYSIS	T3.9-4
3.9-3A	A/E SUPPLIED QUALITY GROUP A TRANSIENTS	T3.9-6
3.9-3B	NSSS-SPECIFIED TRANSIENTS	T3.9-8
3.9-3C	COMPONENT CYCLIC OR TRANSIENT LIMITS	T3.9-11 EC289971
3.9-4	COMPARISON OF STRUCTURAL AND HYDRAULIC DESIGN PARAMETERS	T3.9-15
3.9-5	DESIGN LOADING COMBINATIONS FOR A/E QUALITY GROUPS B AND C COMPONENTS (VESSELS, PUMPS, VALVES)	T3.9-17

CHAPTER 3

LIST OF TABLES (Cont'd)

Table	Title	Page
3.9-5A	DESIGN LOADING COMBINATIONS FOR A/E QUALITY GROUPS B AND C PIPING	T3.9-18
3.9-5B	SAMPLE CALCULATIONS	T3.9-19
3.9-6	A/E DESIGN STRESS LIMITS FOR QUALITY GROUP B AND C PUMPS/VALVES	T3.9-20
3.9-7	A/E DESIGN STRESS LIMITS FOR CODE CLASS 2 AND 3 PIPING AND VESSELS	T3.9-21
3.9-8	QUALITY GROUP B AND C ACTIVE PUMPS	T3.9-23
3.9-9	NSSS SUPPLIED ACTIVE VALVES	T3.9-24
3.9-10	A/E SUPPLIED QUALITY GROUP B AND C ACTIVE VALVES	T3.9-27
3.9-11	NSSS-SUPPLIED SEISMIC AND CODE CLASS SAFETY/RELIEF VALVES	T3.9-31
3.9-12	A/E SUPPLIED SEISMIC AND CODE CLASS SAFETY/RELIEF VALVES	T3.9-35
3.9-13	SAFETY RELIEF VALVE LOADING COMBINATION	T3.9-36
3.9-14	LOADING COMBINATIONS ASME CODE CLASS 1 NSSS COMPONENTS EXCEPT VALVES (TABLE 3.9-1)	T3.9-37
3.9-15	STRESS LIMITS FOR ASME CODE CLASS 1 NSSS COMPONENT EXCEPT VALVES	T3.9-38
3.9-16	LOADING COMBINATIONS FOR NSSS VALVES CLASS 1, 2 and 3	T3.9-39
3.9-17	DESIGN STRESS LIMITS FOR NSSS VALVES CLASS 1, 2 AND 3	T3.9-40
3.9-18	LOADING COMBINATIONS FOR NSSS PUMPS CLASS 2 AND 3	T3.9-41
3.9-19	DESIGN STRESS LIMITS FOR CODE CLASS 2 AND 3 NSSS PUMPS	T3.9-42
3.9-20	LOADING COMBINATIONS FOR NSSS ASME CODE CLASS 2 AND 3 COMPONENTS OTHER THAN VALVES AND PUMPS (VESSEL AND SUPPORTS)	T3.9-43
3.9-21	DESIGN STRESS LIMITS FOR CODE CLASS 2 AND 3 NSSS COMPONENTS OTHER THAN VALVES AND PUMPS (VESSELS AND SUPPORTS)	T3.9-44
3.9-22	LIST OF VIBRATION TESTING MODES	T3.9-45
3.9-23	DELETED	T3.9-47

CHAPTER 3

LIST OF TABLES (Cont'd)

Table	Title	Page
3.9-24	REACTOR VESSEL SUPPORT LOADS	T3.9-48
3.9-25	STEAM GENERATOR SUPPORT LOADS	T3.9-49
3.9-26	RCS COMPONENT NOZZLE LOADS	T3.9-50
3.9-27	STRESS LIMITS FOR PIPE SUPPORTS	T3.9-51
3.9-28	LOADING COMBINATIONS AND STRESS LIMITS FOR PIPING SUPPORTS	T3.9-53
3.9-29	LIMITING CORE SUPPORT MARGINS FOR ASYMMETRIC LOADS	T3.9-54
3.9A-1	PERCENT CRITICAL DAMPING	T3.9A-1
3.9A-2	SUMMARY OF RESULTS-CONTAINMENT SPRAY PUMPS	T3.9A-2
3.9A-3	SUMMARY OF RESULTS-CONTAINMENT SPRAY PUMP MOTOR	T3.9A-3
3.9A-4	SUMMARY OF RESULTS-DIESEL OIL TRANSFER PUMPS	T3.9A-4
3.9A-5	SUMMARY OF RESULTS-LPSI PUMPS	T3.9A-5
3.9A-6	SUMMARY OF RESULTS-LPSI MOTORS	T3.9A-6
3.9A-7	SUMMARY OF RESULTS-HPSI PUMPS AND MOTORS	T3.9A-7
3.9A-8	SUMMARY OF RESULTS-CHARGING PUMPS AND MOTORS	T3.9A-8
3.9B-1	BOLT LOAD COMPARISON	T3.9B-1
3.9B-2	BOLT LOAD COMPARISON	T3.9B-2
3.10-1	A/E ELECTRICAL AND INSTRUMENTATION QUALIFICATION DATA	T3.10-1
3.10-2	NSSS ELECTRICAL EQUIPMENT QUALIFICATION DATA	T3.10-9

DESIGN CRITERIA – STRUCTURES, COMPONENT, EQUIPMENT AND SYSTEMS

CHAPTER 3

LIST OF FIGURES

Figure	Title
3.3-1	SHIELD BLDG. TORNADO AND HURRICANE WIND LOAD CONDITION
3.3-2	CONDENSATE STORAGE TANK TORNADO AND HURRICANE WIND LOAD CONDITION
3.4-1	REACTOR AUXILIARY BUILDING EXT. WALLS - MISC DETAILS - M & R
3.4-2	DELETED
3.4-3	SITE GRADING AND DRAINAGE - SH. 1
3.4-4	SITE GRADING AND DRAINAGE - SH. 1
3.4-4a	ISFSI PROJECT - GRADING AND DRAINAGE, DRAINAGE PLAN
3.4-5	ELECTRICAL MANHOLE AND HANDHOLE DRAINAGE SYSTEM
3.4-6	FLOOD CONTROL STOP LOGS
3.4-7	CIRCULATING WATER SYSTEM DISCHARGE CANAL NOSE PROTECTION
3.5-1	TURBINE ORIENTATION RELATIVE TO REACTOR BLDG
3.5-2	TURBINE GENERATOR PLACEMENT AND ORIENTATION
3.6-1a	JET DIVERGENCE OF POSTULATED BREAKS-GUILLOTINE BREAK
3.6-1b	JET DIVERGENCE OF POSTULATED BREAKS-GUILLOTINE BREAK WITH LIMITED SEPARATION
3.6-1c	JET DIVERGENCE OF POSTULATED BREAKS-SLOT BREAK
3.6-2	RESTRAINT SPACING FOR PIPE RUPTURE
3.6-3	DISCHARGE LEG PIPE RESTRAINTS
3.6-4	CUMUL. USAGE FACTOR AND NORMALIZED PRIMARY PLUS SECONDARY STRESS INTENSITY RANGE RESULTS FOR SEISMIC LOADING (LOOP 2A) (SHEET 1 OF 3)
3.6-4	CUMUL. USAGE FACTOR AND NORMALIZED PRIMARY PLUS SECONDARY STRESS INTENSITY RANGE RESULTS FOR SEISMIC LOADING (LOOP 2B) (SHEET 2 OF 3)
3.6-4	DESIGN BASIS PIPE BREAKS TYPES AND LOCATIONS (SHEET 3 OF 3)
3.6-5	LOAD - DEFLECTION CURVE ENCAPSULATED MARINITE

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.6C-1.1	REACTOR BLDG. - BOILER FEEDWATER BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-1.2	REACTOR BLDG. - BOILER FEEDWATER BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-1.3	REACTOR BLDG. - MAIN STEAM BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-1.4	REACTOR BLDG. - MAIN STEAM BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-1.5	TRESTLE STRUCT. - BOILER FEEDWATER BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-1.6	TRESTLE STRUCT. - BOILER FEEDWATER BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-1.7	MAIN STEAM PIPING TRESTLE AREA BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-1.8	MAIN STEAM PIPING TRESTLE AREA BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-2.1	REACTOR COOLANT SYSTEM DESIGN BASIS PIPE BREAKS TYPES & LOCATIONS
3.6C-2.2	REACTOR COOLANT SYSTEM DESIGN BASIS PIPE BREAKS TYPES & LOCATIONS
3.6C-2.3	PRESSURIZER SPRAY PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-2.4	SURGE LINE BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-2.5	REACTOR BLDG. PRESSURIZER RELIEF PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-2.6	REACTOR BLDG. PRESSURIZER RELIEF SYSTEM BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.1	REACTOR BLDG. SAFETY INJECTION PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.2	REACTOR BLDG SAFETY INJECTION PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.3	REACTOR BLDG SAFETY INJECTION PIPING (2A2) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.4	REACTOR BLDG SAFETY INJECTION PIPING (2A2) BREAK POINTS & PIPE WHIP RESTRAINTS

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.6C-3.5	REACTOR BLDG SAFETY INJECTION PIPING (2A1) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.6	REACTOR BLDG SAFETY INJECTION PIPING (2A1) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.7	REACTOR BLDG SAFETY INJECTION PIPING (2A1) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.8	REACTOR BLDG SAFETY INJECTION PIPING (2B1) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.9	REACTOR BLDG SAFETY INJECTION PIPING (2B1) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.10	REACTOR BLDG SAFETY INJECTION PIPING (2B1) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.11	REACTOR BLDG SAEETY INJECTION PIPING (2B1) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.12	REACTOR BLDG SAFETY INJECTION PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.13	REACTOR BLDG SAFETY INJECTION PIPING (2B2) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.14	REACTOR BLDG SAFETY INJECTION PIPING (2B2) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-3.15	REACTOR BLDG SAFETY INJECTION PIPING (2B2) BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-4.1	REACTOR BLDG CVCS LETDOWN PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-4.2	REACTOR BLDG CVCS LETDOWN PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-4.3	REACTOR BLDG CVCS LETDOWN PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-4.4	CONTAINMENT BLDG CVCS LETDOWN LINE
3.6C-4.5	CONTAINMENT BLDG CVCS LETDOWN LINE
3.6C-4.6	PEN. #26 TO LETDOWN HT. EXCHANGER CHEM. & VOL. CONTROL
3.6C-4.7	REACTOR AUX. BLDG. CHEM. & VOLUME CONTROL

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.6C-4.8	REACTOR AUX. BLDG. CHEM. & VOLUME CONTROL
3.6C-4.9	REACTOR AUX. BLDG. CVC PIPING
3.6C-4.10	REACTOR BLDG. CHEM. VOLUME CONTROL
3.6C-4.11	REACTOR BLDG. CVCS PIPING
3.6C-4.12	REACTOR BLDG. CVCS PIPING
3.6C-4.13	REACTOR BLDG. CVCS PIPING
3.6C-4.14	REACTOR BLDG. CVCS PIPING
3.6C-4.15	REACTOR BLDG. CVCS PIPING
3.6C-4.16	REACTOR BLDG. MISCELLANEOUS PIPING
3.6C-4.17	REACTOR BLDG. MISCELLANEOUS PIPING
3.6C-5.1	REACTOR CONTAINMENT BLDG. BLOWDOWN BREAK LOCS. & PIPE WHIP RESTRAINTS
3.6C-5.2	REACTOR AUX BLDG BLOWDOWN BREAK LOCS. & PIPE WHIP RESTRAINTS
3.6C-5.3	REACTOR CONTAINMENT BLDG BLOWDOWN BREAK LOCS. & PIPE WHIP RESTRAINTS
3.6C-5.4	REACTOR CONTAINMENT BLDG BLOWDOWN BREAK LOCS. & PIPE WHIP RESTRAINTS
3.6C-5.5	REACTOR CONTAINMENT BLDG BLOWDOWN BREAK LOCS. & PIPE WHIP RESTRAINTS
3.6C-5.6	REACTOR CONTAINMENT BLDG BLOWDOWN BREAK LOCS. & PIPE WHIP RESTRAINTS
3.6C-5.7	REACTOR CONTAINMENT BLDG BLOWDOWN BREAK LOCS. & PIPE WHIP RESTRAINTS
3.6C-5.8	REACTOR AUX. BLDG. BLOWDOWN BREAK LOCS. & PIPE WHIP RESTRAINTS
3.6C-6.1	AUXILIARY FEEDWATER PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-6.2	AUXILIARY FEEDWATER PIPING BREAK POINTS & PIPE WHIP RESTRAINTS

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.6C-6.3	AUXILIARY FEEDWATER PIPING BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-6.4	AUXILIARY FEEDWATER BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-6.5	AUXILIARY FEEDWATER BREAK POINTS & PIPE WHIP RESTRAINTS
3.6C-7.1	REACTOR BLDG - MS/FW TRESTLE MAIN STEAM PIPING TO AFW PUMP 2C
3.6C-7.2	REACTOR BLDG - MS/FW TRESTLE MAIN STEAM PIPING TO AFW PUMP 2C
3.6C-7.3	REACTOR BLDG - MS/FW TRESTLE MAIN STEAM PIPING TO AFW PUMP 2C
3.6D-1	REACTOR BUILDING PIPE RESTRAINTS SH. 2
3.6D-2	REACTOR BUILDING PIPE RESTRAINTS SH. 4
3.6D-3	REACTOR BUILDING PIPE RESTRAINTS SH. 9
3.6D-4	REACTOR BUILDING PIPE RESTRAINTS SH. 9
3.6E-1	PLAST PIPE RUPTURE MODEL OF THE REACTOR BLDG MAIN STEAM LINE
3.6E-2	PLAST PIPE RUPTURE MODEL OF THE REACTOR BLDG BOILER FEEDWATER LINE
3.6E-3	TYPICAL RELAP3/MOD68 FLUID MODEL MAIN STEAM LINES 2A & 2B
3.6E-4	TYPICAL RELAP4/MOD6 FLUID MODEL BOILER FEEDWATER LINES 2A & 2B
3.6E-5	PIPE THRUST FORCE VS TIME CURVE USED BY PLAST MODEL FL021H2MS
3.6E-6	FORCE HISTORY APPLIED AT NODE 14 OF BOILER FEEDWATER PIPING MODEL FL021PLASTBF4
3.6F-1	YARD SUMP PUMP P&I DIAGRAM
3.7-1	RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE - HORIZONTAL
3.7-2	RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE - VERTICAL
3.7-3	RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE - HORIZONTAL
3.7-4	RESPONSE SPECTRA SAFE SHUTDOWN EARTHQUAKE - VERTICAL
3.7-5	RESPONSE SPECTRA - HORIZONTAL & VERTICAL
3.7-6	ARTIFICIAL SSE HORIZONTAL ACCELEROGRAM MAXIMUM GROUND ACCELERATION 0.10 G DURATION OF TIME 10 SECONDS

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-7	HORIZONTAL RESPONSE SPECTRA 2 PERCENT DAMPING - SSE
3.7-8	HORIZONTAL RESPONSE SPECTRA 4 PERCENT DAMPING - SSE
3.7-9	HORIZONTAL RESPONSE SPECTRA 5 PERCENT DAMPING - SSE
3.7-10	HORIZONTAL RESPONSE SPECTRA 7 PERCENT DAMPING - SSE
3.7-11	HORIZONTAL RESPONSE SPECTRA 10 PERCENT DAMPING - SSE
3.7-12	ARTIFICIAL SSE VERTICAL ACCELEROGRAM MAXIMUM GROUND ACCELERATION 0.10 G DURATION OF TIME 10 SECONDS
3.7-13	VERTICAL RESPONSE SPECTRA 2 PERCENT DAMPING - SSE
3.7-14	VERTICAL RESPONSE SPECTRA 4 PERCENT DAMPING - SSE
3.7-15	VERTICAL RESPONSE SPECTRA 5 PERCENT DAMPING - SSE
3.7-16	VERTICAL RESPONSE SPECTRA 7 PERCENT DAMPING - SSE
3.7-17	VERTICAL RESPONSE SPECTRA 10 PERCENT DAMPING - SSE
3.7-18	ARTIFICIAL OBE HORIZONTAL ACCELEROGRAM MAXIMUM GROUND ACCELERATION 0.05 G DURATION OF TIME 10 SECONDS
3.7-19	ARTIFICIAL OBE VERTICAL ACCELERATION MAXIMUM GROUND ACCELERATION 0.05 G DURATION OF TIME 10 SECONDS
3.7-20	HORIZONTAL RESPONSE SPECTRA 2 PERCENT DAMPING - OBE
3.7-21	HORIZONTAL RESPONSE SPECTRA 4 PERCENT DAMPING - OBE
3.7-22	HORIZONTAL RESPONSE SPECTRA 5 PERCENT DAMPING - OBE
3.7-23	HORIZONTAL RESPONSE SPECTRA 7 PERCENT DAMPING - OBE
3.7-24	HORIZONTAL RESPONSE SPECTRA 10 PERCENT DAMPING - OBE
3.7-25	VERTICAL RESPONSE SPECTRA 2 PERCENT DAMPING - OBE
3.7-26	VERTICAL RESPONSE SPECTRA 4 PERCENT DAMPING - OBE
3.7-27	VERTICAL RESPONSE SPECTRA 5 PERCENT DAMPING - OBE
3.7-28	VERTICAL RESPONSE SPECTRA 7 PERCENT DAMPING - OBE
3.7-29	VERTICAL RESPONSE SPECTRA 10 PERCENT DAMPING - OBE
3.7-30	REACTOR BUILDING HORIZONTAL MATHEMATICAL MODEL
3.7-31	REACTOR BUILDING VERTICAL MODEL

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-32	REACTOR AUXILIARY BUILDING HORIZONTAL MATHEMATICAL MODEL
3.7-33	REACTOR AUXILIARY BUILDING VERTICAL MATHEMATICAL MODEL
3.7-34	FUEL HANDLING BUILDING MATHEMATICAL MODELS
3.7-35	INTAKE STRUCTURE HORIZONTAL (N-S) MODEL
3.7-36	INTAKE STRUCTURE HORIZONTAL (E-W) MODEL
3.7-37	INTAKE STRUCTURE VERTICAL MATHEMATICAL MODEL
3.7-38	DIESEL GENERATOR BUILDING MATHEMATICAL MODELS
3.7-39	MATHEMATICAL MODEL FOR SEISMIC ANALYSIS OF MS/FW TRESTLE (HORIZONTAL EXCITATION)
3.7-40	MATHEMATICAL MODEL FOR SEISMIC ANALYSIS OF MS/FW TRESTLE (VERTICAL EXCITATION)
3.7-41	STEAM GENERATOR BLOWDOWN FACILITY FNDN. MAT - FINITE ELEMENT MODEL
3.7-42	STEAM GENERATOR BLOWDOWN FACILITY HORIZONTAL MATHEMATICAL MODEL
3.7-43	STEAM GENERATOR BLOWDOWN FACILITY VERTICAL MATHEMATICAL MODEL
3.7-44	COMPONENT COOLING WATER BLDG. HORIZONTAL MODEL
3.7-45	COMPONENT COOLING WATER BLDG. VERTICAL MODEL
3.7-46	CONDENSATE STORAGE TANK BLDG. HORIZONTAL MODEL
3.7-47	CONDENSATE STORAGE TANK BLDG. VERTICAL MODEL
3.7-48	DIESEL OIL STORAGE BLDG. HORIZONTAL MODEL
3.7-49	DIESEL OIL STORAGE BLDG. VERTICAL MODEL
3.7-50	TYPICAL HORIZONTAL DYNAMIC MODEL
3.7-51	TYPICAL VERTICAL DYNAMIC MODEL
3.7-52	FLOOR SPECTRA HORIZONTAL N-S & E-W SSE REACTOR BLDG. MASS PT 11
3.7-53	FLOOR SPECTRA HORIZONTAL N-S & E-W SSE REACTOR BLDG. MASS PT 16

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-54	FLOOR SPECTRA HORIZONTAL N-S & E-W SSE REACTOR BLDG MASS PT 22
3.7-55	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 6
3.7-56	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 10
3.7-57	FLOOR SPECTRA HORIZONTAL N-S & E-W OBE REACTOR BLDG. MASS PT 11
3.7-58	FLOOR SPECTRA HORIZONTAL N-S & E-W OBE REACTOR BLDG. MASS PT 16
3.7-59	FLOOR SPECTRA HORIZONTAL N-S & E-W OBE REACTOR BLDG. MASS PT 22
3.7-60	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 6
3.7-61	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 10
3.7-62	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR BLDG. MASS PT 21
3.7-63	FLOOR SPECTRA HORIZONTAL N-S & E-W SSE REACTOR BLDG. MASS PT 22
3.7-64	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR BLDG. MASS PT 23
3.7-65	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR BLDG. MASS PT 24
3.7-66	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR BLDG. MASS PT 27
3.7-67	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 11
3.7-68	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 13
3.7-69	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 14
3.7-70	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 15
3.7-71	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 16
3.7-72	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR BLDG. MASS PT 21
3.7-73	FLOOR SPECTRA HORIZONTAL N-S & E-W OBE REACTOR BLDG. MASS PT 22
3.7-74	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR BLDG. MASS PT 23
3.7-75	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR BLDG. MASS PT 24

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-76	FLOOR SPECTRA HORIZONTAL N-S & E-W OBE REACTOR BLDG. MASS PT 27
3.7-77	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 11
3.7-78	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 13
3.7-79	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 14
3.7-80	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 15
3.7-81	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 16
3.7-82	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR BLDG. MASS PT 1
3.7-83	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR BLDG. MASS PT 6
3.7-84	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR BLDG. MASS PT 10
3.7-85	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 1
3.7-86	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 3
3.7-87	FLOOR SPECTRA VERTICAL SSE REACTOR BLDG. MASS PT 5
3.7-88	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR BLDG. MASS PT 1
3.7-89	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR BLDG. MASS PT 6
3.7-90	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR BLDG. MASS PT 10
3.7-91	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 1
3.7-92	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 3
3.7-93	FLOOR SPECTRA VERTICAL OBE REACTOR BLDG. MASS PT 5
3.7-94	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR AUXILIARY BLDG.
3.7-95	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR AUXILIARY BLDG.
3.7-96	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR AUXILIARY BLDG.
3.7-97	FLOOR SPECTRA HORIZONTAL N-S SSE REACTOR AUXILIARY BLDG.
3.7-98	FLOOR SPECTRA HORIZONTAL E-W SSE REACTOR AUXILIARY BLDG.
3.7-99	FLOOR SPECTRA HORIZONTAL E-W SSE REACTOR AUXILIARY BLDG.
3.7-100	FLOOR SPECTRA HORIZONTAL E-W SSE REACTOR AUXILIARY BLDG.
3.7-101	FLOOR SPECTRA HORIZONTAL E-W SSE REACTOR AUXILIARY BLDG.

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-102	FLOOR SPECTRA VERTICAL SSE REACTOR AUXILIARY BLDG. MASS PT 5
3.7-103	FLOOR SPECTRA VERTICAL SSE REACTOR AUXILIARY BLDG. MASS PT 6
3.7-104	FLOOR SPECTRA VERTICAL SSE REACTOR AUXILIARY BLDG. MASS PT 7
3.7-105	FLOOR SPECTRA VERTICAL SSE REACTOR AUXILIARY BLDG. MASS PT 8
3.7-106	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR AUXILIARY BLDG.
3.7-107	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR AUXILIARY BLDG.
3.7-108	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR AUXILIARY BLDG.
3.7-109	FLOOR SPECTRA HORIZONTAL N-S OBE REACTOR AUXILIARY BLDG.
3.7-110	FLOOR SPECTRA HORIZONTAL E-W OBE REACTOR AUXILIARY BLDG.
3.7-111	FLOOR SPECTRA HORIZONTAL E-W OBE REACTOR AUXILIARY BLDG.
3.7-112	FLOOR SPECTRA HORIZONTAL E-W OBE REACTOR AUXILIARY BLDG.
3.7-113	FLOOR SPECTRA HORIZONTAL E-W OBE REACTOR AUXILIARY BLDG.
3.7-114	FLOOR SPECTRA VERTICAL OBE REACTOR AUXILIARY BLDG. MASS PT 5
3.7-115	FLOOR SPECTRA VERTICAL OBE REACTOR AUXILIARY BLDG. MASS PT 6
3.7-116	FLOOR SPECTRA VERTICAL OBE REACTOR AUXILIARY BLDG. MASS PT 7
3.7-117	FLOOR SPECTRA VERTICAL OBE REACTOR AUXILIARY BLDG. MASS PT 8
3.7-118	FLOOR SPECTRA HORIZONTAL N-S SSE FUEL HANDLING BLDG.
3.7-119	FLOOR SPECTRA HORIZONTAL N-S SSE FUEL HANDLING BLDG.
3.7-120	FLOOR SPECTRA HORIZONTAL N-S SSE FUEL HANDLING BLDG.
3.7-121	FLOOR SPECTRA HORIZONTAL N-S SSE FUEL HANDLING BLDG.
3.7-122	FLOOR SPECTRA HORIZONTAL E-W SSE FUEL HANDLING BLDG.
3.7-123	FLOOR SPECTRA HORIZONTAL E-W SSE FUEL HANDLING BLDG.
3.7-124	FLOOR SPECTRA HORIZONTAL E-W SSE FUEL HANDLING BLDG.
3.7-125	FLOOR SPECTRA HORIZONTAL E-W SSE FUEL HANDLING BLDG.
3.7-126	FLOOR SPECTRA VERTICAL SSE FUEL HANDLING BLDG.
3.7-127	FLOOR SPECTRA VERTICAL SSE FUEL HANDLING BLDG.
3.7-128	FLOOR SPECTRA VERTICAL SSE FUEL HANDLING BLDG.

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-129	FLOOR SPECTRA VERTICAL SSE FUEL HANDLING BLDG.
3.7-130	FLOOR SPECTRA HORIZONTAL N-S OBE FUEL HANDLING BLDG.
3.7-131	FLOOR SPECTRA HORIZONTAL N-S OBE FUEL HANDLING BLDG.
3.7-132	FLOOR SPECTRA HORIZONTAL N-S OBE FUEL HANDLING BLDG.
3.7-133	FLOOR SPECTRA HORIZONTAL N-S OBE FUEL HANDLING BLDG.
3.7-134	FLOOR SPECTRA HORIZONTAL E-W OBE FUEL HANDLING BLDG.
3.7-135	FLOOR SPECTRA HORIZONTAL E-W OBE FUEL HANDLING BLDG.
3.7-136	FLOOR SPECTRA HORIZONTAL E-W OBE FUEL HANDLING BLDG.
3.7-137	FLOOR SPECTRA HORIZONTAL E-W OBE FUEL HANDLING BLDG.
3.7-138	FLOOR SPECTRA VERTICAL OBE FUEL HANDLING BLDG.
3.7-139	FLOOR SPECTRA VERTICAL OBE FUEL HANDLING BLDG.
3.7-140	FLOOR SPECTRA VERTICAL OBE FUEL HANDLING BLDG.
3.7-141	FLOOR SPECTRA VERTICAL OBE FUEL HANDLING BLDG.
3.7-142	FLOOR SPECTRA HORIZONTAL N-S SSE INTAKE STRUCTURE MASS PT 1
3.7-143	FLOOR SPECTRA HORIZONTAL N-S SSE INTAKE STRUCTURE MASS PT 2
3.7-144	FLOOR SPECTRA HORIZONTAL N-S SSE INTAKE STRUCTURE MASS PT 3
3.7-145	FLOOR SPECTRA HORIZONTAL N-S SSE INTAKE STRUCTURE MASS PT 4
3.7-146	FLOOR SPECTRA HORIZONTAL E-W SSE INTAKE STRUCTURE MASS PT 1
3.7-147	FLOOR SPECTRA HORIZONTAL E-W SSE INTAKE STRUCTURE MASS PT 2
3.7-148	FLOOR SPECTRA HORIZONTAL E-W SSE INTAKE STRUCTURE MASS PT 3
3.7-149	FLOOR SPECTRA HORIZONTAL E-W SSE INTAKE STRUCTURE MASS PT 4
3.7-150	FLOOR SPECTRA VERTICAL SSE INTAKE STRUCTURE MASS PT 1

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-151	FLOOR SPECTRA VERTICAL SSE INTAKE STRUCTURE MASS PT 2
3.7-152	FLOOR SPECTRA VERTICAL SSE INTAKE STRUCTURE MASS PT 3
3.7-153	FLOOR SPECTRA VERTICAL SSE INTAKE STRUCTURE MASS PT 4
3.7-154	FLOOR SPECTRA HORIZONTAL N-S OBE INTAKE STRUCTURE MASS PT 1
3.7-155	FLOOR SPECTRA HORIZONTAL N-S OBE INTAKE STRUCTURE MASS PT 2
3.7-156	FLOOR SPECTRA HORIZONTAL N-S OBE INTAKE STRUCTURE MASS PT 3
3.7-157	FLOOR SPECTRA HORIZONTAL N-S OBE INTAKE STRUCTURE MASS PT 4
3.7-158	FLOOR SPECTRA HORIZONTAL E-W OBE INTAKE STRUCTURE MASS PT 1
3.7-159	FLOOR SPECTRA HORIZONTAL E-W OBE INTAKE STRUCTURE MASS PT 2
3.7-160	FLOOR SPECTRA HORIZONTAL E-W OBE INTAKE STRUCTURE MASS PT 3
3.7-161	FLOOR SPECTRA HORIZONTAL E-W OBE INTAKE STRUCTURE MASS PT 4
3.7-162	FLOOR SPECTRA VERTICAL OBE INTAKE STRUCTURE MASS PT 1
3.7-163	FLOOR SPECTRA VERTICAL OBE INTAKE STRUCTURE MASS PT 2
3.7-164	FLOOR SPECTRA VERTICAL OBE INTAKE STRUCTURE MASS PT 3
3.7-165	FLOOR SPECTRA VERTICAL OBE INTAKE STRUCTURE MASS PT 4
3.7-166	FLOOR SPECTRA HORIZONTAL N-S SSE DIESEL GENERATOR BLDG. CANT 1
3.7-167	FLOOR SPECTRA HORIZONTAL N-S SSE DIESEL GENERATOR BLDG. CANT 1
3.7-168	FLOOR SPECTRA HORIZONTAL N-S SSE DIESEL GENERATOR BLDG. CANT 2
3.7-169	FLOOR SPECTRA HORIZONTAL N-S SSE DIESEL GENERATOR BLDG. CANT 3
3.7-170	FLOOR SPECTRA HORIZONTAL N-S SSE DIESEL GENERATOR BLDG.
3.7-171	FLOOR SPECTRA HORIZONTAL E-W SSE DIESEL GENERATOR BLDG. CANT 1
3.7-172	FLOOR SPECTRA HORIZONTAL E-W SSE DIESEL GENERATOR BLDG. CANT 1

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-173	FLOOR SPECTRA HORIZONTAL E-W SSE DIESEL GENERATOR BLDG. CANT 2
3.7-174	FLOOR SPECTRA HORIZONTAL E-W SSE DIESEL GENERATOR BLDG. CANT 3
3.7-175	FLOOR SPECTRA HORIZONTAL E-W SSE DIESEL GENERATOR BLDG.
3.7-176	FLOOR SPECTRA VERTICAL SSE DIESEL GENERATOR BLDG CANT 1
3.7-177	FLOOR SPECTRA VERTICAL SSE DIESEL GENERATOR BLDG CANT 1
3.7-178	FLOOR SPECTRA VERTICAL SSE DIESEL GENERATOR BLDG CANT 2
3.7-179	FLOOR SPECTRA VERTICAL SSE DIESEL GENERATOR BLDG CANT 3
3.7-180	FLOOR SPECTRA VERTICAL SSE DIESEL GENERATOR BLDG
3.7-181	FLOOR SPECTRA HORIZONTAL N-S OBE DIESEL GENERATOR BLDG CANT 1
3.7-182	FLOOR SPECTRA HORIZONTAL N-S OBE DIESEL GENERATOR BLDG CANT 1
3.7-183	FLOOR SPECTRA HORIZONTAL N-S OBE DIESEL GENERATOR BLDG CANT 2
3.7-184	FLOOR SPECTRA HORIZONTAL N-S OBE DIESEL GENERATOR BLDG CANT 3
3.7-185	FLOOR SPECTRA HORIZONTAL N-S OBE DIESEL GENERATOR BLDG
3.7-186	FLOOR SPECTRA HORIZONTAL E-W OBE DIESEL GENERATOR BLDG CANT 1
3.7-187	FLOOR SPECTRA HORIZONTAL E-W OBE DIESEL GENERATOR BLDG CANT 1
3.7-188	FLOOR SPECTRA HORIZONTAL E-W OBE DIESEL GENERATOR BLDG CANT 2
3.7-189	FLOOR SPECTRA HORIZONTAL E-W OBE DIESEL GENERATOR BLDG CANT 3
3.7-190	FLOOR SPECTRA HORIZONTAL E-W OBE DIESEL GENERATOR BLDG
3.7-191	FLOOR SPECTRA VERTICAL OBE DIESEL GENERATOR BLDG CANT 1
3.7-192	FLOOR SPECTRA VERTICAL OBE DIESEL GENERATOR BLDG CANT 1

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-193	FLOOR SPECTRA VERTICAL OBE DIESEL GENERATOR BLDG CANT 2
3.7-194	FLOOR SPECTRA VERTICAL OBE DIESEL GENERATOR BLDG CANT 3
3.7-195	FLOOR SPECTRA VERTICAL OBE DIESEL GENERATOR BLDG
3.7-196	FLOOR SPECTRA HORIZONTAL N-S SSE MS/FW TRESTLE
3.7-197	FLOOR SPECTRA HORIZONTAL N-S SSE MS/FW TRESTLE
3.7-198	FLOOR SPECTRA HORIZONTAL N-S SSE MS/FW TRESTLE
3.7-199	FLOOR SPECTRA HORIZONTAL N-S SSE MS/FW TRESTLE
3.7-200	FLOOR SPECTRA HORIZONTAL N-S SSE MS/FW TRESTLE
3.7-201	FLOOR SPECTRA HORIZONTAL E-W SSE MS/FW TRESTLE
3.7-202	FLOOR SPECTRA HORIZONTAL E-W SSE MS/FW TRESTLE
3.7-203	FLOOR SPECTRA HORIZONTAL E-W SSE MS/FW TRESTLE
3.7-204	FLOOR SPECTRA HORIZONTAL E-W SSE MS/FW TRESTLE
3.7-205	FLOOR SPECTRA HORIZONTAL E-W SSE MS/FW TRESTLE
3.7-206	FLOOR SPECTRA VERTICAL SSE MS/FW TRESTLE
3.7-207	FLOOR SPECTRA VERTICAL SSE MS/FW TRESTLE
3.7-208	FLOOR SPECTRA VERTICAL SSE MS/FW TRESTLE
3.7-209	FLOOR SPECTRA VERTICAL SSE MS/FW TRESTLE
3.7-210	FLOOR SPECTRA VERTICAL SSE MS/FW TRESTLE
3.7-211	FLOOR SPECTRA HORIZONTAL N-S OBE MS/FW TRESTLE
3.7-212	FLOOR SPECTRA HORIZONTAL N-S OBE MS/FW TRESTLE
3.7-213	FLOOR SPECTRA HORIZONTAL N-S OBE MS/FW TRESTLE
3.7-214	FLOOR SPECTRA HORIZONTAL N-S OBE MS/FW TRESTLE
3.7-215	FLOOR SPECTRA HORIZONTAL N-S OBE MS/FW TRESTLE
3.7-216	FLOOR SPECTRA HORIZONTAL E-W OBE MS/FW TRESTLE
3.7-217	FLOOR SPECTRA HORIZONTAL E-W OBE MS/FW TRESTLE
3.7-218	FLOOR SPECTRA HORIZONTAL E-W OBE MS/FW TRESTLE
3.7-219	FLOOR SPECTRA HORIZONTAL E-W OBE MS/FW TRESTLE

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-220	FLOOR SPECTRA HORIZONTAL E-W OBE MS/FW TRESTLE
3.7-221	FLOOR SPECTRA VERTICAL OBE MS/FW TRESTLE
3.7-222	FLOOR SPECTRA VERTICAL OBE MS/FW TRESTLE
3.7-223	FLOOR SPECTRA VERTICAL OBE MS/FW TRESTLE
3.7-224	FLOOR SPECTRA VERTICAL OBE MS/FW TRESTLE
3.7-225	FLOOR SPECTRA VERTICAL OBE MS/FW TRESTLE
3.7-226	FLOOR SPECTRA HORIZONTAL N-S SSE COMP'T COOL WATER BLDG. MASS PT 5
3.7-227	FLOOR SPECTRA HORIZONTAL N-S SSE COMP'T COOL WATER BLDG. MASS PT 6
3.7-228	FLOOR SPECTRA HORIZONTAL N-S SSE COMP'T COOL WATER BLDG.
3.7-229	FLOOR SPECTRA HORIZONTAL E-W SSE COMP'T COOL WATER BLDG. MASS PT 5
3.7-230	FLOOR SPECTRA HORIZONTAL E-W SSE COMP'T COOL WATER BLDG. MASS PT 6
3.7-231	FLOOR SPECTRA HORIZONTAL E-W SSE COMP'T COOL WATER BLDG.
3.7-232	FLOOR SPECTRA VERTICAL SSE COMP'T COOL WATER BLDG. MASS PT 5
3.7-233	FLOOR SPECTRA VERTICAL SSE COMP'T COOL WATER BLDG. MASS PT 6
3.7-234	FLOOR SPECTRA VERTICAL SSE COMP'T COOL WATER BLDG.
3.7-235	FLOOR SPECTRA HORIZONTAL N-S OBE COMP'T COOL WATER BLDG. MASS PT 5
3.7-236	FLOOR SPECTRA HORIZONTAL N-S OBE COMP'T COOL WATER BLDG. MASS PT 6
3.7-237	FLOOR SPECTRA HORIZONTAL N-S OBE COMP'T COOL WATER BLDG.
3.7-238	FLOOR SPECTRA HORIZONTAL E-W OBE COMP'T COOL WATER BLDG. MASS PT 5
3.7-239	FLOOR SPECTRA HORIZONTAL E-W OBE COMP'T COOL WATER BLDG. MASS PT 6
3.7-240	FLOOR SPECTRA HORIZONTAL E-W OBE COMP'T COOL WATER BLDG.
3.7-241	FLOOR SPECTRA VERTICAL OBE COMP'T COOL WATER BLDG. MASS PT 5

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-242	FLOOR SPECTRA VERTICAL OBE COMP'T COOL WATER BLDG. MASS PT 6
3.7-243	FLOOR SPECTRA VERTICAL OBE COMP'T COOL WATER BLDG.
3.7-244	FLOOR SPECTRA HORIZONTAL SSE CONDENSATE STORAGE TANK MASS PT 9
3.7-245	FLOOR SPECTRA VERTICAL SSE CONDENSATE STORAGE TANK MASS PT 9
3.7-246	FLOOR SPECTRA HORIZONTAL OBE CONDENSATE STORAGE TANK MASS PT 9
3.7-247	FLOOR SPECTRA VERTICAL OBE CONDENSATE STORAGE TANK MASS PT 9
3.7-248	FLOOR SPECTRA HORIZONTAL N-S SSE DIESEL OIL STG MASS PT 3 & 8
3.7-249	FLOOR SPECTRA HORIZONTAL E-W SSE DIESEL OIL STG MASS PT 3 & 8
3.7-250	FLOOR SPECTRA VERTICAL SSE DIESEL OIL STG MASS PT 3 & 8
3.7-251	FLOOR SPECTRA HORIZONTAL N-S OBE DIESEL OIL STG MASS PT 3 & 8
3.7-252	FLOOR SPECTRA HORIZONTAL E-W OBE DIESEL OIL STG MASS PT 3 & 8
3.7-253	FLOOR SPECTRA VERTICAL OBE DIESEL OIL STG MASS PT 3 & 8
3.7-254	SEISMIC PROTECTION ANALYSIS SAMPLE PROBLEM NO 1
3.7-255	SEISMIC PROTECTION ANALYSIS SAMPLE PROBLEM NO 2 SHEET 1
3.7-256	SEISMIC PROTECTION ANALYSIS SAMPLE PROBLEM NO 2 SHEET 2
3.7-257	SEISMIC PROTECTION ANALYSIS SAMPLE PROBLEM NO 3
3.7-258	TYPICAL REACTOR COOLANT SYSTEM SEISMIC ANALYSIS MODEL
3.7-259	PRESSURIZER SEISMIC ANALYSIS MODEL
3.7-260	SURGE LINE SEISMIC ANALYSIS MODEL
3.7-261	PRESSURIZER SPRAY LINE SEISMIC ANALYSIS MODEL (Sh 1 of 2)
3.7-261	PRESSURIZER SPRAY LINE SEISMIC ANALYSIS MODEL (Sh 2 of 2)
3.7-262	VERTICAL SEISMIC ANALYSIS MODEL
3.7-263	HORIZONTAL SEISMIC ANALYSIS MODEL
3.7-264	CORE SEISMIC MODEL ONE ROW OF 17 FUEL ASSEMBLIES

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.7-265	CORE-SUPPORT BARREL UPPER FLANGE FINITE-ELEMENT MODEL
3.7-266	SCHEMATIC SHOWING CRITICAL SECTION OF CEDM HOUSING
3.7-267	SCHEMATIC SHOWING CRITICAL SECTION OF CEDM HOUSING
3.7-268	AVERAGE EXTENSION SHAFT POSITION DURING SEISMIC SCRAM
3.7-269	WATERFORD NO. 3 REACTOR BUILDING MATHEMATICAL MODEL (NO TORSIONAL EFFECT)
3.7-270	WATERFORD NO. 3 REACTOR BUILDING MATHEMATICAL TORSION MODEL
3.7-271	RESPONSE SPECTRA USED FOR HIGH STRESS COMPARISON BETWEEN MODIFIED EQUIVALENT STATIC LOAD METHOD & MODE RESPONSE SPECTRA ANALYSIS
3.7-272	RESPONSE SPECTRA USED FOR HIGH STRESS COMPARISON BETWEEN MODIFIED EQUIVALENT STATIC LOAD METHOD & MODE RESPONSE SPECTRA ANALYSIS
3.7-273	RESPONSE SPECTRA USED FOR HIGH STRESS COMPARISON BETWEEN MODIFIED EQUIVALENT STATIC LOAD METHOD & MODE RESPONSE SPECTRA ANALYSIS
3.8-1	CONTAINMENT VESSEL – SHEET 1
3.8-2	CONTAINMENT VESSEL – SHEET 4
3.8-3	GENERAL ARRANGEMENT FOR 2 FT - 6 X 6 FT - 8 PERSONNEL LOCK
3.8-4	GENERAL ARRANGEMENT 5' - 9 I.D. ESCAPE LOCK W/2' - 6 I.D. OPENING
3.8-5	REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS
3.8-6	REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS
3.8-7	REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS
3.8-8	DELETED
3.8-9	REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS SH. 4
3.8-10	CONTAINMENT VESSEL - SHEET 2
3.8-11	CONTAINMENT VESSEL - SHEET 3
3.8-12	ELECTRICAL PENETRATIONS
3.8-13	ROOF TEMPORARY CONSTRUCTION LOADS

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.8-14	CONTAINMENT VESSEL OBE SEISMIC LOADING
3.8-15	CONTAINMENT VESSEL SSE SEISMIC LOADING
3.8-16	COMBINATION OF LATITUDINAL & MERIDIONAL STRESSES FOR THE CONTAINMENT VESSEL
3.8-17	CONTAINMENT VESSEL STRESSES RESULTING FROM HORIZONTAL AXIAL FORCES
3.8-18	CONTAINMENT VESSEL STRESSES RESULTING FROM VERTICAL AXIAL FORCES
3.8-19	COMPUTER MODEL FOR CBI PROGRAM 781
3.8-20	CONTAINMENT VESSEL EMBEDMENT STRESS MODEL
3.8-21	EMBEDMENT LOCA TEMPERATURE GRADIENT
3.8-22	ESCAPE LOCK HORIZONTAL SSE RESPONSE SPECTRUM
3.8-23	PERSONNEL LOCK HORIZONTAL SSE RESPONSE SPECTRUM
3.8-24	PERSONNEL AND ESCAPE LOCKS VERTICAL SSE RESPONSE SPECTRUM
3.8-25	ESCAPE LOCK HORIZONTAL OBE RESPONSE SPECTRUM
3.8-26	PERSONNEL LOCK HORIZONTAL OBE RESPONSE SPECTRUM
3.8-27	PERSONNEL AND ESCAPE LOCKS VERTICAL OBE RESPONSE SPECTRUM
3.8-28	AIRLOCK SEISMIC ANALYSIS MODEL
3.8-29	MODEL FOR CB&I PROGRAM 1027
3.8-30	CONTAINMENT PENETRATION MODEL USED IN CB&I PROGRAM 1036M
3.8-31	CONTAINMENT PENETRATION MODEL USED IN CB&I PROGRAM 1392
3.8-32	CONTAINMENT PENETRATION MODEL USED IN CB&I PROGRAM 1392
3.8-33	CRITICAL SHELL SECTIONS
3.8-34	ALLOWABLE BUCKLING STRESSES FOR UNSTIFFENED HEMISPHERICAL HEAD
3.8-35	ALLOWABLE BUCKLING STRESSES FOR CYLINDRICAL VESSEL
3.8-36	CONTAINMENT VESSEL SUPPORT - PLAN AND SECT. MAS AND REINF
3.8-37	LOCK GASKET CROSS SECTION

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.8-38	INTERNAL CONCRETE - PLANS AND SECTIONS MASONRY SHEET 1
3.8-39	INTERNAL CONCRETE - PLANS AND SECTIONS MASONRY SHEET 2
3.8-40	EQUIPMENT FDNS - M AND R SH. 1
3.8-41	REACTOR BUILDING EQUIPMENT SUPPORTS SHEET 2
3.8-42	REACTOR BUILDING EQUIPMENT SUPPORTS SHEET 1
3.8-43	REACTOR BUILDING EQUIPMENT SUPPORTS SH. 5
3.8-44	REACTOR BUILDING EQUIPMENT SUPPORTS SHEET 11
3.8-45	REACTOR BUILDING EQUIPMENT SUPPORTS SHEET 12
3.8-46	REACTOR BUILDING EQUIPMENT SUPPORTS SHEET 13
3.8-47	REACTOR BUILDING EQUIPMENT SUPPORTS SHEET 4
3.8-48	REACTOR BUILDING EQUIPMENT SUPPORTS SHEET 7
3.8-49	DELETED
3.8-50	DELETED
3.8-51	REACTOR BUILDING EMBEDDED STEEL PRIMARY SHIELD WALL
3.8-52	REACTOR BUILDING EQUIPMENT SUPPORTS SH. 3
3.8-53	REACTOR BUILDING PLATFORMS
3.8-54	CYLINDER WALL PLAN AND SECTIONS MASONRY
3.8-55	CYLINDER DEVELOPMENT - MASONRY
3.8-56	DOME-PLAN AND SECTIONS - MAS.
3.8-57	REACTOR BUILDING - DOME REINFORCEMENT
3.8-58	REACTOR AUXILIARY BUILDING FRAMING PLAN - SLABS & BEAMS SHEET 1
3.8-59	REACTOR AUXILIARY BUILDING FRAMING PLAN - SLABS & BEAMS SHEET 2
3.8-60	DIESEL GENERATOR BUILDING MAS.
3.8-61	MAIN STEAM TRESTLE SHEET 1
3.8-62	MAIN STEAM TRESTLE SHEET 2

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.8-63	MAIN STEAM TRESTLE SHEET 3
3.8-64	MAIN STEAM TRESTLE SHEET 4
3.8-65	MAIN STEAM TRESTLE SHEET 5
3.8-66	MAIN STEAM TRESTLE SHEET 6
3.8-67	STEAM GENERATOR BLOWDOWN
3.8-68	DELETED
3.8-69	STEAM GENERATOR BLOWDOWN
3.8-70	REACTOR BUILDING BASE SLAB - PLAN – MAS.
3.8-71	REACTOR BUILDING BASE SLAB - SECT. REINF. - SH. NO. 2
3.8-72	REACTOR AUXILIARY BUILDING BASE SLAB PLAN - MASONRY
3.8-73	FB-SOIL PRESSURE UNDER EARTHQUAKE
3.8-74	SOIL PRESSURE UNDER EARTHQUAKE RAB DBE
3.8A-1	REACTOR AUXILIARY BUILDING BLOCK WALLS - SHEET 1
3.8A-2	EFFECTIVE AREA FOR FLEXURAL COMPRESSION
3.9-1	CORE SUPPORT BARREL SHELL RESPONSE MODEL (ASHSD)
3.9-2	CORE SUPPORT BARREL DYNAMIC STABILITY (BUCKLING) MODEL
3.9-3	REACTOR INTERNALS LATERAL LOCA MODEL (CESHOCK)
3.9-4	REACTOR INTERNALS VERTICAL LOCA MODEL
3.9-5	REACTOR INTERNALS VERTICAL/HORIZONTAL REDUCED MODEL
3.9-6	REACTOR CORE HORIZONTAL NONLINEAR LOCA MODEL
3.9-7	LOCA ANALYSIS - HOT LEG BREAK MODEL
3.9-8	REACTOR VERTICAL ARRANGEMENT
3.9-9	MAINE YANKEE REACTOR VERTICAL ARRANGEMENT
3.9-10	FORT CALHOUN - UNIT 1 REACTOR VERTICAL ARRANGEMENT
3.9-11	CONTROL EQUIPMENT DRIVE MECHANISM (MAGNETIC JACK)
3.9-12	PRESSURE VESSEL, CORE SUPPORT BARREL SNUBBER ASSEMBLY
3.9-13	CORE SHROUD ASSEMBLY

CHAPTER 3

LIST OF FIGURES (Cont'd)

Figure	Title
3.9-14	UPPER GUIDE STRUCTURE ASSEMBLY
3.9-15	INCORE INSTRUMENT SUPPORT ASSEMBLY
3.9-16	INCORE INSTRUMENT NOZZLE
3.9-18	RCS - FLEXIBILITY ANALYSIS MODEL
3.9-19	RV ASYMMETRIC LOADS ANALYSIS RV SUPPORT LOADS
3.9-20	RV ASYMMETRIC LOADS ANALYSIS RV SUPPORT LOADS
3.9-21	RV ASYMMETRIC LOADS ANALYSIS RV SUPPORT LOADS
3.9-22	MODEL OF REACTOR INTERNALS
3.9-23	GENERAL FLOW CHART FOR PIPING VIBRATION TESTING
3.9-24	GENERAL FLOW DIAGRAM FOR INTERNAL-EXTERNAL ASYMMETRIC LOADS ANALYSIS
3.9-25	RC PIPING DEFORMATION
3.9-25a	REACTOR VESSEL LEVEL DETECTOR HOLDER (UPPER PORTION)
3.9-25b	REACTOR VESSEL LEVEL DETECTOR HOLDER (LOWER PORTION)
3.9B-1	BASE PLATE ANALYSIS

CHAPTER 3

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

The following sections discuss conformance with the NRC "General Design Criteria for Nuclear Power Plants" as specified in Appendix A to 10 CFR 50 effective May 21, 1971 and subsequently amended July 7, 1971 and February 12, 1976. Based on the content herein, the applicant concludes, that St. Lucie Unit 2 fully satisfies and is in compliance with the General Design Criteria.

3.1.1 CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems and components important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection and testing of structures, systems and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

DISCUSSION

All structures, systems and components of the facility are classified according to their relative importance to safety. Those items vital to safety such that their failure might cause or result in an uncontrolled release of an excessive amount of radioactive material are designated seismic Category I. They and items of lesser importance to safety are designed, fabricated, erected and tested according to the provisions of recognized codes and quality standards. Discussions of the applicable codes, standards, records and quality assurance program used to implement and audit the operation processes are presented in Section 17.2. A complete set of facility structural, arrangement and system drawings is maintained under the control of FP&L throughout the life of the plant. Quality assurance written data and comprehensive test and operating procedures are likewise assembled and maintained by FP&L. The classification of seismic Category I structures and safety related systems and components is discussed in Section 3.2.

3.1.2 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 natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

DISCUSSION

The structures, systems and components important to safety are designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. Natural phenomena factored into the design of plant structures, systems and components important to safety are determined from recorded data for the site vicinity with appropriate margin to account for uncertainties in historical data.

The most severe natural phenomena postulated to occur at the site in terms of induced stresses is the safe shutdown earthquake (SSE). Those structures, systems, and components vital for the mitigation and control of accident conditions are designed to withstand the effects of a loss of coolant accident (LOCA) coincident with the effects of the SSE. Structures, systems and components vital to the safe shutdown of the plant are designed to withstand the effects of any one of the most severe natural phenomena, including flooding, hurricanes, tornadoes and the SSE (refer to Chapter 2).

Design criteria for wind and tornado, flood and earthquake are discussed in Sections 3.3, 3.4 and 3.7, respectively.

3.1.3 CRITERION 3 - FIRE PROTECTION

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

DISCUSSION

Noncombustible and fire resistant materials are used wherever practical throughout the facility, particularly in areas containing critical portions of the plant such as containment structure, control room and components of systems important to safety. These systems are designed and located to minimize the effects of fires or explosions on their redundant components. Facilities for the storage of combustible material are designed to minimize both the probability and the effects of a fire.

Equipment and facilities for fire detection, alarm and extinguishment are provided to protect both plant and personnel from fire or explosion and the resultant release of toxic vapors. Both wet and dry type firefighting equipment are provided.

Normal fire protection is provided by deluge systems, hose lines and portable extinguishers and preaction type sprinkler systems.

The Fire Protection System is designed such that a failure of any component of the system:

a) will not cause a significant release of radioactivity to the environment.

b) will not impair the ability of redundant equipment to safely shutdown and isolate the reactor or limit the release of radioactivity to the environment in the event of a LOCA.

All equipment is accessible for periodic inspection. The Fire Protection System is described in Subsection 9.5.1 and the Fire Protection Design Basis Document (Reference 1). EC282743

3.1.4 CRITERION 4 - ENVIRONMENTAL AND MISSILE 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, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analysis reviewed and approved by the Commission demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

DISCUSSION

Structures, systems and components important to safety are designed to accommodate the effects of and to be compatible with the pressure, temperature, humidity, and radiation conditions associated with normal operation, maintenance, testing, and postulated accidents including a LOCA, in the area in which they are located.

Due to the application of leak before break methodology to the RCS hot and cold leg piping, the dynamic effects of a loss of coolant accident do not have to be considered. A technical evaluation was performed to demonstrate that the probability or likelihood of large pipe breaks occurring in the primary coolant loops is sufficiently low that they need not be a design basis (see Reference 13 in Section 3.6).

Protective walls and slabs, local missile shielding, or restraining devices are provided to protect the containment and engineered safety features systems within the containment against damage from missiles generated by equipment failures. The concrete enclosing the Reactor Coolant System serves as radiation shielding and an effective barrier against internally generated missiles. A missile shield is provided for control element drive mechanisms. Penetrations and piping extending outward from the containment, up to and including isolation valves are protected from damage due to pipe whipping, and are protected from damage by external missiles, where such protection is necessary to meet the design bases.

Non-Seismic Category I Piping is arranged or restrained so that failure of any non- seismic Category I piping will neither cause a nuclear accident nor prevent essential seismic Category I structures or equipment from mitigating the consequences of such an accident.

Seismic Category I piping is arranged or restrained such that in the event of rupture of a seismic Category I pipe which causes LOCA, resulting pipe movement will not result in loss of containment integrity or adequate engineered safety features systems operation.

The structures inside the containment vessel are designed to sustain dynamic loads which could result from failure of major equipment and piping, such as jet thrust, jet impingement and local pressure transients, where containment integrity is needed to cope with the conditions.

The external concrete Shield Building protects the steel containment vessel from damage due to external missiles such as tornado propelled missiles.

For those components which are required to operate under extreme conditions such as design seismic loads or containment post- LOCA environmental conditions, the manufacturers submit type test, operational or calculational data which substantiate this capability of the equipment.

Refer to Sections 3.5, 3.6, 3.7, and 3.11 for details on missile protection, pipe rupture and jet impingement protection, seismic design criteria and environmental qualification, respectively.

3.1.5 CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS OR COMPONENTS

Structures, systems and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

DISCUSSION

Safety related components interconnected between the two units include the condensate storage tanks, the diesel generator fuel oil system, and the Class 1E 4.16kV switchgear (1AB and 2AB) station blackout cross-tie. These safety related interconnections are not normally used by both units and employ isolation devices between them. Locked closed isolation valves are provided for the AFW and diesel generator fuel oil inter-ties. The station blackout cross-tie has two breakers in series for isolation between the two units. The failure of equipment on one unit will not impair the ability of the counterpart on the other unit from performing its safety related function. The interconnections provide added redundancy and operational flexibility without compromising unit and system independence.

In accordance with NRC staff requirements, a missile protected inter-tie is provided between the Unit 1 auxiliary feedwater pump suction lines and the Unit 2 condensate storage tank (CST) to be used under administrative control. To add to the system's operational flexibility, the provision to supply the Unit 2 auxiliary feedwater pumps from the Unit 1 condensate storage tank is also provided. To prevent inadvertent draining of the Unit 2 CST to the Unit 1 CST plant procedures for placing the inter-tie in service require that the Unit 1 CST outlet isolation valves be closed prior to placing the inter-tie line in service. This helps to assure that the water level in the Unit 2 CST is maintained at the minimum value required for safe shutdown.

In the unlikely event of loss of offsite power, both St. Lucie Units 1 and 2 have their own 100 percent capacity redundant diesel generator sets which are available for safe shutdown.

In the unlikely event of a station blackout in one unit, i.e., total loss of AC power on- site and off-site, both units can be electrically connected, under administrative control, such that a diesel generator set from the non-blacked out unit is able to provide power to the minimum loads required to maintain both units in a hot standby condition.

The ultimate heat sink (which performs a safety related function) supplies emergency cooling water to both St. Lucie Units 1 and 2. The canal has sufficient cross-sectional water flow area to mitigate the consequences of a LOCA on one unit while safely shutting down the other unit.

3.1.10 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.

DISCUSSION

In ANSI N18.2, plant conditions have been categorized in accordance with their anticipated frequency of occurrence and risk to the public, and design requirements are given for each of the four categories. The categories covered by this criterion are Condition I - Normal Operation and Condition II - Faults of Moderate Frequency.

The design requirement for Condition I is that margin shall be provided between any plant parameter and the value of that parameter which would require either automatic or manual protective action; it is met by providing an adequate control system (refer to Section 7.7). The design requirement for Condition II is that such faults shall be accommodated with, at most, a shutdown of the reactor, with the plant capable of returning to operation after corrective action; it is met by providing a Reactor Protective System (refer to Section 7.2).

Specified acceptable fuel design limits are stated in Section 4.4. Minimum margins to specified acceptable fuel design limits are prescribed in the Technical Specifications (Limiting Conditions for Operations) which support Chapters 4 and 15. The plant is designed such that operation within Limiting Conditions for Operation, with safety system settings not less conservative than Limiting Safety System Settings prescribed in the Technical Specifications, assures that specified acceptable fuel design limits will not be violated as a result of anticipated operational occurrences. During non-accident conditions, operation of the plant within Limiting Conditions for Operation ensures that specified acceptable fuel design limits are not approached within the minimum margins. Operator action, aided by the control systems and monitored by plant instrumentation, maintains the plant within Limiting Conditions for Operation during non-accident conditions.

3.1.11 CRITERION 11 - REACTOR INHERENT PROTECTION

CRITERION:

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.

RESPONSE:

In the power operating range, the combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor power in the power operation range is a decrease in reactivity, i.e., the inherent nuclear feedback characteristics are not positive. The reactivity coefficients are discussed in detail in Section 4.3.

3.1.12 CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

CRITERION:

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.

RESPONSE:

Power level oscillations will not occur. The effect of the negative power coefficient of reactivity (refer to Criterion 11), together with the coolant temperature program maintained by control element assemblies (CEAs) and soluble boron, provide fundamental mode stability. Power level is monitored continuously by neutron flux detectors (refer to Chapter 7) and by reactor coolant temperature difference measuring devices.

Power distribution oscillations are detected by neutron flux detectors. Axial mode oscillations are suppressed by means of CEAs. Radial oscillations are expected to be convergent. It is a design objective that azimuthal xenon oscillations be convergent. Monitoring and protective requirements imposed by Criterion 10 and 20 are discussed in those responses and in Chapter 4.

3.1.13 CRITERION 13 - INSTRUMENTATION AND CONTROL

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

DISCUSSION

Instrumentation is provided, as required, to monitor and maintain significant process variables which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Controls are provided for the purpose of maintaining these variables within the limits prescribed for safe operation.

The principal variables and systems monitored include neutron level (reactor power); reactor coolant temperature, flow, and pressure; pressurizer liquid level; steam generator level and pressure; and containment pressure and temperature.

The following is provided to monitor and maintain control over the fission process during both transient and steady state periods over the lifetime of the core:

a) Twelve independent channels of nuclear instrumentation, which constitute the primary monitor of the fission process. Of these channels, the four wide range logarithmic safety channels are used to monitor the reactor from startup through full power; four will monitor the reactor in the power range and are used to initiate a reactor shutdown in the event of overpower; two linear power range channels are utilized for control purposes and two channels for startup and extended shutdown.

b) Two independent CEA Position Indicating Systems

c) A method of determining soluble poison concentration by sampling and analysis of reactor coolant water

d) Manual and automatic control of reactor power by means of CEAs

e) Manual regulation of coolant boron concentrations.

Incore instrumentation is provided to supplement information on core power distribution and to provide for calibration of out-of-core flux detectors.

Instrumentation measures temperatures, pressures, flows, and levels in the Main Steam System and auxiliary systems and is used to maintain these variables within prescribed limits.

The Reactor Protective System is designed to monitor the reactor operating conditions and to effect reliable and rapid reactor trip if anyone or a combination of conditions deviate from a preselected operating range.

The containment pressure, temperature, and radiation instrumentation is designed to function during normal operation and the postulated accidents.

The instrumentation and control systems are described in detail in Chapter 7.

3.1.14 CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed, fabricated, created and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

DISCUSSION

The RCPB is defined in accordance with 10 CFR 50, Section 50.2(v) and ANSI N18.2, Section 5.4.3.2 (see also response to criterion 55).

Reactor Coolant System (RCS) piping and components are designed to meet the requirements of the ASME Code, Section III. To establish operating pressure and temperature limitations during startup and shutdown of the Reactor Coolant System, the fracture toughness rules defined in Appendix G of the ASME Code, Section III, is followed. Quality control, inspection, and testing as required by this code and allowable reactor pressure-temperature operations ensure the integrity of the RCS.

The RCPB is designed to accommodate the system pressures and temperatures attained under all expected modes of unit operation, including all anticipated transients, and to maintain the stresses within applicable limits.

Piping and equipment pressure parts of the RCPB are assembled and erected by welding unless applicable codes permit flanged, screwed, or compression joints. Welding procedures are employed which produce welds of complete fusion and free of unacceptable defects. All welding procedures, welders, and welding machine operators are qualified in accordance with the requirements of ASME Code, Section IX for the materials to be welded. Qualification records, including the results of procedure and performance qualification tests and identification symbols assigned to each welder are maintained.

The pressure boundary has provisions for inservice inspection in accordance with the requirements of ASME Code, Section XI, to ensure continuance of the structural and leaktight integrity of the boundary (see also response to Criterion 32). For the reactor vessel, a material surveillance program conforming with the requirements of Appendix H of 10 CFR 50 is given in Section 5.3.

3.1.15 CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

CRITERION:

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.

DISCUSSION:

The design criteria and bases for the reactor coolant pressure boundary are described in the response to Criterion 14.

The operating conditions for normal steady state and transient plant operations are established conservatively (see Subsection 4.4.3). Normal operating limits are selected so that an adequate margin exists between operating and design limits. The plant control systems are designed to ensure that plant variables are maintained well within the established operating limits. The plant transient response characteristics and pressure and temperature distributions during normal operations are considered in the design as well as the accuracy and response of the instruments and controls. These design techniques ensure that a satisfactory margin is maintained between the plant's normal operating conditions, including design transients, and the design limits for the reactor coolant pressure boundary.

The plant control systems function to minimize deviations from normal operating limits in the event of most anticipated operational occurrences (ANSI N18.2 Condition II Occurrences). Where control systems response would be inadequate or fail upon demand, the Reactor Protective System (RPS) and the Engineered Safety Features System (ESFS) function to mitigate the consequences of such events.

The RPS and ESFS function to mitigate the consequences of accidents (ANSI N18.2 Condition III or IV occurrences). Analyses show that the design limits for the reactor coolant pressure boundary are not exceeded in the event of any ANSI N18.2 Condition II, III or IV occurrences. See Section 5.2 and Chapter 7 for further information.

3.1.16 CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight 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.

DISCUSSION

The containment system is designed to protect the public from the radiological consequences of a LOCA, based on a postulated break of reactor coolant piping up to and including a double ended break of the largest reactor coolant pipe.

The containment vessel, Shield Building, and the associated engineered safety features systems are designed to safely sustain all 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 of a design basis accident.

Leak tightness of the containment system and short and long term performance are analyzed in Section 6.2.

3.1.17 CRITERION 17 - ELECTRICAL POWER SYSTEMS

An onsite electrical power system and an offsite electrical 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 electrical 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.

Electrical power from the transmission network to the switchyard shall be supplied by physically independent transmission lines (not necessarily on separate rights-of-way) designed and located so as to suitably 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.

DISCUSSION

Offsite power is transmitted to the plant switchyard by four physically independent 230 kV transmission lines. During normal plant operation, the station auxiliary power is normally supplied from the main generator through the plant auxiliary transformers. Upon loss of power from the auxiliary transformers or of a unit generator, there is a "fast dead" automatic transfer to the startup transformers thus providing continuity of power. EC286688

In the event of a loss of the offsite power sources, two emergency onsite diesel generator sets and redundant sets of station batteries provide the necessary ac and dc power for safe shutdown or, in the event of an accident, provide the necessary power to restrict the consequences to within acceptable limits. The onsite emergency ac and dc power systems consist of redundant and independent power sources and distribution systems such that a single failure does not prevent the systems from performing their safety function.

Refer to Sections 8.2 and 8.3 for further discussion of offsite power sources and onsite power sources respectively.

3.1.18 CRITERION 18 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections and switchboards to assess the continuity of the systems and the conditions of their components. The system 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.

DISCUSSION

Electrical power systems important to safety are 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 to detect deterioration, if any, of their components. Capability is provided to periodically test the operability and functional performance of the components of the systems. The diesel generator sets are started and loaded periodically on a routine basis and relays, switches, and buses are inspected and tested for operation and availability on an individual basis.

Transfers from normal to emergency sources of power are made to check the operability of the systems and the full operational sequence that brings the systems into operation.

Refer to Section 8.3 and the Technical Specifications for further discussion.

3.1.19 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 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 total effective dose equivalent for the duration of the accident.

Equipment in appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentations 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.

DISCUSSION

The control stations, switches, controllers and indicators necessary to operate or shut down the unit and maintain safe control of the facility are located in the control room.

The design of the control room permits safe occupancy during design basis accident conditions. The control room is isolated from the outside atmosphere during the initial period following the occurrence of an accident. The control room ventilation system recirculates control room air through HEPA and charcoal filters as discussed in Subsection 9.4.1 and Section 6.4. Radiation detectors and alarms are provided. Emergency lighting is provided as discussed in Subsection 9.5.3.

Alternate local controls and local instruments are available for equipment required to bring the plant to and maintain a hot standby condition. It is also possible to attain a cold shutdown condition from locations outside of the control room through the use of suitable procedures. (Refer to Subsection 7.4.1).

3.1.20 CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

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

DISCUSSION

The Reactor Protective System monitors reactor operating conditions and automatically initiates a reactor trip when the monitored variable or combination of variables exceeds a prescribed operating range. The reactor trip setpoints are selected to ensure that anticipated operational occurrences do not cause specified acceptable fuel design limits to be violated. Specific reactor trips are described in Section 7.2.

Reactor trip is accomplished by deenergizing the control element drive mechanism holding latch coils through the interruption of the CEDM power supply. The CEAs are thus released to drop into the core reducing reactor power.

The Engineered Safety Features Actuation System monitors potential accident conditions and automatically initiates engineered safety features and their supporting systems when the monitored variables reach prescribed setpoints. The parameters which automatically actuate engineered safety features are described in Section 7.3. Manual actuation is provided to the operator.

3.1.21 CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in 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.

DISCUSSION

The protection systems are designed to provide high functional reliability and in-service testability by designing to the requirements of IEEE 279-1971 and IEEE 338-1971. The systems are designed such that the single failure criteria and performance requirements are met with three channels in service.

A coincidence of exceeding any two like sensor trip parameters generates a trip signal. However, four measurement channels with electrical and physical separation are provided for each parameter. To enhance plant availability, a fourth channel is provided as a spare and allows bypassing of one channel while maintaining the requisite two-out-of-three system.

Each channel of the protection system, including the sensors up to the final actuation device is capable of being checked during reactor operation. Those channels that can affect plant operation are tested during scheduled reactor shutdown. Measurement sensors of each channel used in protection systems are checked by observing outputs of similar channels which are presented on indicators and recorders in the control room. Trip units and logic are tested by inserting a signal into the measurement channel ahead of the readout and, upon application of a trip level input, observing that a signal is passed through the trip units and the logic to the logic output relays. The logic output relays are tested individually for initiation of trip action.

Protection system reliability and testability are discussed in Sections 7.2 and 7.3.

3.1.22 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.

DISCUSSION

The protection systems conform to the provisions of IEEE 279-1971, as explained in Subsection 7.2.2.3. Four independent measurement channels complete with sensors, sensor power supplies, signal conditioning units and bistable trip units are provided for each protective parameter monitored by the protection systems. The measurement channels are provided with a high degree of independence by separate connections of the channel sensors to the process systems. Power to the channels is provided by two independent emergency power supply sources.

The protective system is fuctionally tested to ensure satisfactory operation prior to installation in the plant. Environmental and seismic qualifications are also performed utilizing type tests and specific equipment tests (refer to Sections 3.10 and 3.11).

3.1.23 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.

DISCUSSION

Protective system trip channels are designed to fail into a safe state or into a state established as acceptable in the event of loss of power supply or disconnection of the system. A loss of power to the CEDM holding coils results in gravity insertion of the CEAs into the core. Redundancy, channel independence, and separation incorporated in the protective system design minimize the possibility of the loss of a protection function under adverse environmental conditions (refer to Sections 7.2 and 7.3).

3.1.24 CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

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

DISCUSSION

The protection systems are separated from the control instrumentation systems so that failure or removal from service of any control instrumentation system component or channel does not inhibit the function of the protection system. Separation of protection and control systems is discussed in Subsection 7.2.2.3.

3.1.25 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.

DISCUSSION

Reactor shutdown with CEAs is accomplished completely independent of the control functions since the trip breakers interrupt power to the CEA drive mechanisms regardless of existing control signals. The design is such that the system can withstand accidental withdrawal of controlling groups without exceeding acceptable fuel design limits. Analysis of possible reactivity control malfunctions is given in Section 15.4. The Reactor Protection System will prevent specified acceptable fuel design limits from being exceeded for any event of moderate frequency.

3.1.26 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.

DISCUSSION

Two independent reactivity control systems of different design principles are provided. The first system, using CEAS, includes a positive means (gravity) for inserting CEAs and is capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including specified anticipated operational occurrences, specified acceptable fuel design limits are not exceeded. The CEAs can also be mechanically driven into the core. The appropriate margin for stuck rods is provided by assuming in the analyses of anticipated operational occurrences that the highest worth CEA does not fall into the core.

The second system, the Chemical and Volume Control System (CVCS), using neutron absorbing soluble boron, is capable of reliably compensating for the rate of reactivity changes resulting from planned normal power changes (including xenon burnout) such that acceptable fuel design limits are not exceeded. This system is capable of holding the reactor subcritical under cold conditions. For a further description, see Subsection 9.3-4.

Either system is capable of making the core subcritical from a hot operating condition. For further discussion, see Sections 7.4 and 7.7.

3.1.27 CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY CRITERION

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.

DISCUSSION

The reactivity control systems that provide the means for making and holding the core subcritical under postulated accident conditions are discussed in Sections 9.3 and 4.3. Combined use of CEAs and chemical shim control by the CVCS provides the shutdown margin required for plant cooldown and long-term xenon decay, assuming the highest worth CEA is stuck out of the core.

During an accident, the Safety Injection System (SIS) functions to inject concentrated boric acid into the Reactor Coolant System for short term and long term cooling and for reactivity control. Details of the system are given in Section 6.3.

The Safety Injection System, in conjunction with the combined capabilities of the reactivity control systems is available to maintain short and long term cooling of the core even in the event a CEA of highest worth is stuck out of the core. Upon receipt of a safety injection actuation signal, the SIS functions to inject borated water from the refueling water tank into the Reactor Coolant System.

3.1.28 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 RCPB 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.

DISCUSSION

The bases for CEA design include ensuring that the reactivity worth of any one CEA is not greater than a preselected maximum value. The CEAs are divided into two sets: a shutdown set and a regulating set. These sets are further subdivided into groups as necessary. Administrative procedures and control interlocks ensure that the amount and rate of reactivity increase is limited to predetermined values. The regulating groups are withdrawn only after the shutdown groups are fully withdrawn. The regulating groups are programmed to move in sequence and within limits which prevent the rate of reactivity addition and the worth of individual CEAs from exceeding limiting values (see Sections 4.3 and 7.7).

The maximum rate of reactivity addition that may be produced by the CVCS is too low to induce any significant pressure forces that might rupture the RCPB or disturb the reactor vessel internals.

The RCPB (Chapter 5) and the reactor internals (Chapter 4) are designed to appropriate codes (refer for instance to the response to Criterion 14) and accommodate the static and dynamic loads associated with an inadvertent, sudden release of energy, such as that resulting from a CEA ejection or a steam line break, without rupture and with limited deformation which will not impair the capability of cooling the core.

3.1.29 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.

DISCUSSION

Plant conditions designated as Condition I and Condition II in ANSI N18.2 are carefully considered in the design of the Reactor Protective System and the reactivity control systems. Consideration of redundancy, independence and testability in the design, coupled with careful component selection, overall system testing, and adherence to detailed quality assurance, assure an extremely high probability that safety functions are accomplished in the event of anticipated operational occurrences. For additional discussion, see the responses for Criteria 10, 13, 15 and 20.

3.1.30 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.

DISCUSSION

The RCPB components are designed, fabricated, erected, and tested in accordance with ASME Code, Section III. RCPB components are classified as Quality Groups A and B as defined in Subsection 3.2.2. Accordingly, they receive all of the quality measures appropriate to that classification.

Detection and identification of reactor coolant leakage is discussed in Subsection 5.2.5. The system is designed to detect and identify the source of reactor coolant leakage.

3.1.31 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 non-brittle 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.

DISCUSSION

All the RCPB piping and components are designed and constructed in accordance with ASME Code Section III, and comply with the test and inspection requirements of this code. These test and inspection requirements assure that flaw sizes are limited so that the probability of failure by rapid propagation is extremely remote. Particular emphasis is placed on the quality control applied to the reactor vessel, on which tests and inspections exceeding ASME Code requirements are performed. The tests and inspections performed on the reactor vessel are summarized in Sections 5.2 and 5.3.

Carbon and low-alloy steel materials that form part of the pressure boundary are tested for fracture toughness to the acceptance criteria and requirements of 10 CFR 50, Appendix G (see Section 5.2).

3.1.32 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.

DISCUSSION

Provisions are made in the design for inspection, testing and surveillance of the RCS boundary as required by ASME Code Section III and Section XI, as applicable.

The reactor vessel surveillance program conforms with ASTM E-185, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels", as revised in 1973. The details of the reactor surveillance program are given in Subsection 5.2.4.

3.1.33 CRITERION 33 - REACTOR COOLANT MAKEUP

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

DISCUSSION

Reactor coolant makeup during normal operation is provided by the CVCS. The design incorporates a high degree of functional reliability by provision of redundant components and an alternate path for charging. The charging pumps can be powered from either onsite or offsite power sources, including the onsite emergency diesel generators. The CVCS has the capability of replacing the flow loss to the Reactor Coolant System for leaks in the reactor coolant piping up to 0.50 inch equivalent diameter. Additionally, with only one charging pump available, the CVCS has the capability to supplement the HPSI pump injection flow for a certain range of small breaks (see Subsection 6.3.3.4). The system is described in Subsection 9.3.4.

3.1.34 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.

DISCUSSION

The transfer of fission product decay heat and other residual heat from the reactor core is accomplished by the steam generators and the shutdown cooling system at such a rate that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Residual heat removal capacity is provided with sufficient redundancies in design that in the event of a single active failure or a single limited leakage passive failure the system can still perform its function. The steam generator auxiliaries and the Shutdown Cooling System are designed to operate from either offsite or onsite electric power sources. See Subsection 5.4.7 for additional information on Residual Heat Removal.

3.1.35 CRITERION 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming 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.

DISCUSSION

The Safety Injection System provides cooling water at a rate sufficient (1) to assure that the zirconium-water reaction is limited to a negligible rate of less than one percent and (2) to assure that the fuel remains in a coolable geometry. Therefore, compliance with the intent of the acceptance criteria for emergency core cooling systems for light water power reactors of 10 CFR 50, paragraph 50.46(b) is satisfied.

The SIS design includes adequate provisions to ensure that the required safety functions are provided with a single active failure relying on either onsite or offsite electrical power supply (see Section 6.3).

3.1.36 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.

DISCUSSION

The Safety Injection System is designed to facilitate inspection of all critical components. Those components located external to the containment structure are readily accessible for periodic inspection to ensure system leak-tight integrity. Components located inside containment are designed to permit inspection for leak-tightness during maintenance and refueling shutdowns. Reactor vessel internal structures, reactor coolant piping and water injection nozzles are designed to permit visual inspection and/or nondestructive inspection techniques (where these are applicable).

The actual location, arrangement and installation of the system components provide the necessary access for the capability of complying with the periodic inspection requirements of Section XI of the ASME Code (refer to Section 6.3 and the Technical Specifications).

3.1.37 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 on 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.

DISCUSSION

The Safety Injection System is designed to permit appropriate periodic pressure and functional testing. The structural and leaktight integrity, operability, and performance of the SIS components and system are assured through testing conducted during normal plant operation under conditions as close to design as practicable. The operational sequence that bring the SIS into action, including transfer to alternate power sources, is designed to be tested in parts in such a manner as to verify the operability of the actuation system as a whole.

Periodic pressure testing of the high pressure safety injection portion of the SIS to assure system integrity is possible using the cross connection from the charging pumps in the CVCS. Flow path continuity in the high pressure injection lines and suction lines from the refueling water tank (RWT) is assured with the plant at operating pressure by the recirculation of HPSI and LPSI pump discharge back to the RWT. Since LPSI pumps are used as shutdown cooling pumps during normal operation their operability is further demonstrated. Borated water from the safety injection tanks (SITs) may be bled through the recirculation test lines to verify flow path continuity from each tank to its associated main safety injection header. During refueling, blowdown tests will provide additional evidence of SIT operability. Preoperational testing of the SIS provides additional assurance of SIS performance (see Section 6.3 and the Technical Specifications).

3.1.38 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.

DISCUSSION

The containment heat removal systems described in Subsection 6.2.2 consists of the Containment Spray System and the Containment Cooling System. The Containment Spray System consists of two trains, each containing a containment spray pump, shutdown heat exchanger and spray header. The Containment Cooling System consists of four fan coolers. One spray pump and two containment fan coolers have the capacity to reduce containment pressure and temperature following a design basis accident and maintain them at acceptably low levels.

Both the Containment Spray and the Containment Cooling Systems are provided with emergency onsite power necessary for their operation, assuming a loss of offsite power. They are provided with offsite power from the startup transformers if normal onsite power is not available. The Containment Spray and Containment Cooling Systems are provided with redundant equipment so that when assuming a single failure or a failure of an emergency onsite power supply, 100 percent containment cooling capability is available.

3.1.39 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.

DISCUSSION

The Containment Spray System essential equipment except for risers, distribution header piping, spray nozzles and the containment sump are located outside of the containment. The containment sump, the spray piping, and the spray nozzles within the containment can be inspected during refueling shutdowns. Associated equipment outside the containment can be visually inspected at any time.

The Containment Cooling System is entirely within the containment. It can be inspected at appropriate intervals during refueling shutdowns. Cooling water systems external to the containment which service the Containment Cooling System are accessible for inspection at any time during plant operation.

Inservice inspections of the Containment Spray System and the Containment Cooling System are performed as indicated in Section 6.6.

3.1.40 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 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.

DISCUSSION

System piping, valves, pumps, fans, heat exchangers, and other components of the containment heat removal system are designed to permit appropriate periodic testing to assure their structural and leaktight integrity. The components are arranged so that each component can be tested periodically for operability and required functional performance.

The containment cooling units are normally in operation.

The operational sequence that would bring the containment heat removal system into action, including the transfer to alternate power sources, can be tested. With the plant at operating pressure, the containment spray pumps and valves may be operated by recirculation back to the refueling water tank. This permits verification of flow path continuity in the suction lines from the refueling water tank to the first containment spray isolation valve outside the containment. The spray isolation valves can be tested independently of the spray pumps (refer to Section 6.2 and the Technical Specifications).

3.1.41 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 electrical power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

DISCUSSION

As discussed in Subsection 6.5.2, the Containment Spray System, in conjunction with the Iodine Removal System, provides a function of removing fission products from a post-accident containment atmosphere. The Iodine Removal System removes radio-iodines from the containment atmosphere following a LOCA by adding controlled amounts of hydrazine to the containment spray water.

The Shield Building Ventilation System consists of two full capacity redundant fan and filter systems and is designed, consistent with the functioning of other engineered safety features systems, to reduce the concentration and quantity of fission products released to the environment following a LOCA by establishing and maintaining a subatmospheric pressure within the Shield Building annulus to ensure that post-accident activity leakage from the containment vessel is routed through the charcoal filter system (refer to Subsection 6.2.3).

Hydrogen control and sampling systems are provided to prevent the buildup of dangerous concentrations of hydrogen in the containment following a LOCA. The hydrogen control system consists of two full capacity hydrogen recombiners and redundant hydrogen sampling systems. A Continuous Containment/Hydrogen Purge System is also available. The hydrogen recombiners, which are the primary means of control, provide control of hydrogen concentration in the containment without any release to the environment. The hydrogen sampling system can analyze the containment atmosphere either by passing a sample through the automatic hydrogen analyzer or by utilizing a grab sample (refer to Subsections 6.2.3, 6.2.5 and 9.4.8).

The Shield Building Ventilation System, Containment Spray System/Iodine Removal System, and the containment hydrogen control system have suitable redundancy to assure that for normal onsite or for offsite electrical power system failure, their safety functions can be accomplished, assuming a single failure.

3.1.42 CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEM

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.

DISCUSSION

All components of the Shield Building Ventilation System and the containment hydrogen control system are accessible for physical inspection. Ducts, plenums, and casings are provided with access doors for internal inspection.

The only components of the containment atmosphere cleanup systems inside the Shield Building are the duct work of the SBVS, hydrogen recombiners and the containment spray nozzles and piping. These can be inspected during shutdown.

Specific inspection programs are discussed in Subsection 6.2.5.4 for the combustible gas control systems and components, Subsection 6.5.1.4 for the filter systems that are required to perform a safety related function following a design basis accident and Subsection 6.5.2.4 for the Containment Spray System.

3.1.43 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.

DISCUSSION

The Shield Building Ventilation System, Containment Spray System, and hydrogen control and sampling systems are designed and constructed to permit periodic pressure and functional testing. For the purpose of periodically testing the retentive capability of the filter systems, specific plant and vendor procedures are utilized.

High efficiency particulate (HEPA) and charcoal filters are located outside the containment for convenience in testing and inspection. Periodic tests are described in Subsection 6.5.1.4.

Active components of the Shield Building Ventilation System, hydrogen analyzer, hydrogen recombiners and Containment Spray System can be tested periodically for operability and required functional performance.

The full operational sequence that would bring the systems into action, including the transfer to alternate power sources, and the design air flow capability can be tested (refer to Subsections 6.2.5.4, 6.5.1.4 and 6.5.2.4).

3.1.44 CRITERION 44 - COOLING WATER

A system to transfer beat 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 electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

The cooling water systems which function to remove the combined heat load from structures, systems and components important to safety under normal operating and accident conditions, are the Component Cooling Water System and the Intake Cooling Water System.

The Component Cooling Water System is a closed loop system which removes heat from the shutdown heat exchangers, Containment Cooling System and other essential and nonessential components as described in Subsection 9.2.2.

The Intake Cooling Water System is an open loop system which removes heat from the Component Cooling System and transfers it to the ultimate heat sink as described in Subsection 9.2.1.

The primary and secondary sources of water for the ultimate heat sink are as follows. The intake cooling water pumps normally take water from the Atlantic Ocean through the circulating water intake conduits and canal. In the event of interruption of water from this source, water is taken through the emergency cooling water canal from Big Mud Creek. The ultimate heat sink is discussed in Subsection 9.2.5.

Each system is normally pressurized permitting leakage detection by routine surveillance or monitoring instrumentation.

Electrical power for the operation of each system may be supplied from offsite or onsite emergency power sources, with distribution arranged such that a single failure does not prevent the system from performing its safety function.

3.1.45 CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

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

DISCUSSION

The Component Cooling Water System and Intake Cooling Water System are designed to permit periodic inspection, to the extent practical of important components, such as heat exchangers, pumps, valves and accessible piping. Each system is normally pressurized permitting leakage detection by routine surveillance or monitoring instrumentation (refer to Subsections 9.2.1.4 and 9.2.2.4 and Section 6.6).

3.1.46 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.

DISCUSSION

Both the Component Cooling Water and Intake Cooling Water Systems are in operation during normal plant operation or shutdown. The structural and leaktight integrity of the Component Cooling Water and Intake Cooling Water Systems components and the operability and performance of their active components are demonstrated in this way. The operation of pumps and heat exchangers are rotated on a scheduled basis to monitor operational capability of redundant components. Data can be taken periodically during normal plant operation to confirm heat transfer capabilities (refer to Subsections 9.2.1.4 and 9.2.2.4).

The systems are designed to permit testing of system operability encompassing simulation of emergency reactor shutdown or LOCA conditions, including the transfer between normal and emergency power sources.

3.1.50 CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and 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.

DISCUSSION

The containment structure, including access openings and penetrations, is designed to accommodate, without exceeding the design leak rate, the transient peak pressure and temperature associated with a design basis accident.

The containment structure and engineered safety features systems are evaluated for various combinations of energy release. The analysis accounts for system thermal and chemical energy, and for nuclear decay heat. The Safety Injection System is designed such that no single active failure could result in significant metal-water reaction. The combined cooling capacity of two containment cooling units and one containment spray train is adequate, to prevent over pressurization of the structure, and to return the containment to near atmospheric pressure (refer to Subsection 6.2.1).

3.1.51 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.

DISCUSSION

As specified in Subsection 3.8.2, the material selected for the containment vessel is carbon steel normalized to refine the grain which results in improved ductility. In addition, the actual mechanical and chemical properties of the material are documented and are within the limits for minimum ductility.

The containment vessel is built to Subsection NE of Section III of the ASME Code, and in accordance with this code the materials including weld specimens are impact tested.

The design of the vessel reflects consideration of all ranges of temperature and loading conditions which apply to the vessel during operation, maintenance, testing and postulated accident conditions.

All seam welds in the vessel are 100 percent radiographed and the acceptance standards of the radiographs ensure that flaws in welds do not exceed the maximum allowed by the ASME Code.

Since this vessel is post weld heat treated, residual stresses from welding are minimal. Steady state and transient stresses are calculated in accordance with accepted methods (refer to Subsection 3.8.2).

3.1.52 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.

DISCUSSION

The containment vessel is designed so that initial integrated leak rate testing can be performed at design pressure after completion and installation of penetrations and equipment.

Provisions are made in the containment design to permit periodic leakage rate tests, at reduced or peak pressure, to verify the continued leaktight integrity of the containment.

Periodic integrated leakage rate testing are carried out in accordance with the requirements of Appendix J to 10 CFR 50 and/or Appendix J, Option B. A description of the periodic integrated leakage rate testing is provided in Subsections 6.2.1.6 and 6.2.6.

3.1.53 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.

DISCUSSION

The absence of insulation of the containment vessel permits periodic inspection of the exposed interior surfaces of the vessel. The lower portions of the containment vessel are totally encased in concrete and are not accessible for inspection after the acceptance testing. There is no need for any special in-service surveillance program due to the rigorous design, fabrication, inspection and pressure testing the containment vessel receives prior to operation.

Provisions are made to permit periodic testing at containment design pressure of penetrations which have resilient seals or expansion bellows to allow leak tightness to be demonstrated (refer to Subsection 6.2.6).

3.1.54 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.

DISCUSSION

Piping penetrating the containment vessel shell is designed to withstand at least a pressure equal to the containment vessel maximum internal pressure. The isolation system design requires a double barrier on all of the above systems not serving accident consequence limiting systems so that no single active failure can result in loss of isolation or intolerable leakage. These lines are provided with isolation valves as indicated in Subsection 6.2.4.

Valves isolating penetrations serving engineered safety features systems will not automatically close with a containment isolation actuation signal (CIAS), but may be closed by remote manual operation from the control room to isolate any Engineered Safety Feature when required.

Proper valve closing time is achieved by appropriate selection of valve, operator type and operator size. Refer to Subsection 6.2.4 for additional isolation valve information.

To ensure continued integrity of the containment isolation system, periodic closure and leakage tests are performed as stated in Subsection 6.2.4.4.

3.1.55 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:

a) One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

b) One automatic isolation valve inside and one locked closed isolation valve outside containment, or

c) 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

d) 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 in-service 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.

DISCUSSION

Except for Safety Injection System lines, shutdown cooling lines, certain sample lines off the Safety Injection System or Reactor Coolant System, and Chemical and Volume Control System charging and letdown lines, the reactor coolant pressure boundary as defined in 10 CFR 50 is located within the containment. Isolation provisions for these lines are as indicated in Subsection 6.2.4. The safety injection, shutdown cooling and charging lines are closed seismic Category I piping systems outside the containment with isolation valves as indicated in Subsection 6.2.4.

3.1.56 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:

a) One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

b) One automatic isolation valve inside and one locked closed isolation valve outside containment, or

c) 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

d) 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.

DISCUSSION

Lines which connect directly to the containment atmosphere and are not used to mitigate the effects of a LOCA are provided with two valves in series, one inside and one outside the containment. These containment isolation valves are either capable of automatic actuation or normally locked closed.

Lines which connect directly to the containment atmosphere and are used for mitigating the effects of a LOCA are provided with a double containment barrier which consists of the closed piping system pressure boundary outside the containment and one isolation valve capable of remote manual actuation.

Automatic isolation valves, upon loss of power, are selected to fail-close, fail-as-is, or fail-open, whichever position provides the greater safety. Isolation valves are located as close to the containment as practical. Refer to Subsection 6.2.4 for detailed information regarding containment isolation.

3.1.57 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 containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

DISCUSSION

Each line that penetrates the reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere, has at least one containment isolation valve which is either automatic, or locked closed, or capable of remote manual operation, and located outside the containment as close to the containment as practical (refer to Subsection 6.2.4).

3.1.60 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.

DISCUSSION

The Waste Management System is described in Sections 11.2, 11.3 and 11.4, and is designed to provide controlled handling and disposal of liquid, gaseous, and solid wastes. The Waste Management System is designed to ensure that the general public and plant personnel are protected against exposure to radioactive material to meet the intent of 10 CFR 20 and 10 CFR 50, Appendix I.

Liquid and gaseous radioactive releases from the Waste Management System are accomplished on a batch basis. All radioactive effluents are sampled prior to release to ensure compliance with 10 CFR 20 and 10 CFR 50, Appendix I and to determine release rates. Radioactive effluents which do not meet release limits are not discharged to the environment. The Waste Management System is designed with sufficient holdup capacity and flexibility for reprocessing of wastes to ensure that releases are as low as reasonably achievable.

The Waste Management System is designed to preclude the inadvertent release of radioactive material.

All storage tanks in the liquid waste and gaseous waste systems are administratively controlled to prevent the addition of waste to a tank which is being discharged to the environment. Each discharge path is provided with a radiation monitor which alerts plant personnel and initiates automatic closure of redundant isolation valves to prevent further releases in the event of noncompliance with 10 CFR 20 (see Section 11.5 for details).

3.1.61 CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

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

DISCUSSION

Most of the components and systems in this category are in frequent use and no special testing is required. Those systems and components important to safety which are not normally operating are tested periodically, e.g., temperature alarms in the Fuel Pool System (Subsection 9.1.3) and radiation alarms in the fuel pool area, and the fuel handling equipment (prior to each refueling).

The spent fuel storage racks are located to provide sufficient shielding water over stored fuel assemblies to limit radiation at the surface of the water to no more than 2.5 mr/hr during the storage period. The exposure time during refueling is limited so that the integrated dose to operating personnel does not exceed the limits of 10 CFR 20.

The Waste Management System (Chapter 11) is designed to permit controlled handling and disposal of liquid, gaseous, and solid wastes which will be generated during plant operation. The principal design criterion is to ensure that plant personnel and the general public are protected against exposure to radiation from wastes in accordance with limits defined in 10 CFR 20.

The fuel pool is located within the Fuel Handling Building. The liquid waste processing equipment and the gaseous waste storage and disposal equipment are located within a separate area of the Reactor Auxiliary Building. Both of these areas provide confinement capability in the event of an accidental release of radioactive materials, and both are ventilated with filtered discharges to the vent pipe which is monitored.

Analysis (Section 15.7) indicates that the accident release of the maximum activity content of a gas decay tank does not result in doses in excess of the limits set forth in 10 CFR 100.

The Fuel Pool Cooling System is designed to prevent damage to the spent fuel which could result in radioactivity release to the plant operating areas or the public environs (Subsection 9.1.3).

The fuel pool is designed to withstand the postulated tornado driven missiles, seismic event or cask drop without loss of pool water.

3.1.62 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.

DISCUSSION

The new and spent fuel storage and handling facilities are described in Subsections 9.1.1 and 9.1.2. New fuel is stored in air. Spent fuel is stored in borated water. The spacing is sufficient to maintain a subcritical k_{eff} for the new and spent fuel assemblies when in unborated water.

3.1.63 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.

DISCUSSION

There are no residual or decay heat removal systems in the Waste Management System.

The Fuel Pool and Waste Management Systems are provided with appropriate radiation indication and alarms. In addition, alarms are provided in the event of a reduction in fuel pool level.

3.1.64 CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss of coolant accident 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.

DISCUSSION

Means are provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss of coolant accident 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.

Radioactive waste management and monitoring is discussed in Chapter 11. Area and airborne monitoring is discussed in Subsection 12.3.4.

REFERENCES EC282743

1. DBD-FP-1, Fire Protection Design Basis Document

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 SEISMIC CLASSIFICATION

Plant structures, systems and components, including their foundations and supports, that are designed to remain functional in the event of a safe shutdown earthquake (SSE) are designated as seismic Category I, as indicated in Table 3.2-1.

In compliance with General Design Criterion 2, plant structures, systems and components which are important to safety are designed to remain functional in the event of a SSE if they are necessary to assure:

a) the integrity of the reactor coolant pressure boundary,

b) the capability to shut down the reactor and maintain it in a safe shutdown condition, or

c) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guidelines established for design basis accidents.

Non-seismic structures, systems and components are those whose failure would not result in the release of significant radioactivity and would not prevent reactor shutdown or degrade the operation of engineered safety feature systems. Their failure may, however, interrupt power generation.

The Steam Generator Blowdown Treatment Facility which is a Non-seismic category structure, is designed for an Operating Basis Earthquake as discussed in Subsection 3.7.2.1.1(g).

The seismic classifications conform to the recommendations of Regulatory Guide (RG) 1.29, "Seismic Design Classification," September 1978 R(3) with the following two exceptions:

a) The Chemical and Volume Control System letdown line, specifically that portion downstream of the letdown control valves and upstream of the volume control tank outlet isolation valve (V2501), is not seismic Category I. A complete discussion is found in Subsection 9.3.4.3.1. This design was found acceptable by the NRC, as stated in Supplement 1 to the Safety Evaluation Report issued March 3, 1976 (Docket #50-389).

b) The component cooling water (CCW) supply to the reactor coolant pumps (RCPs) is not seismic Category I. A complete discussion on the capability to accommodate loss of CCW to the RCPs is found in Subsections 5.4.1.3 and 9.2.2.3. This design was also found acceptable by the NRC, as stated in Supplement 1 to the Safety Evaluation Report issued March 3, 1976, (Docket #50-389).

   Where only portions of systems are identified as seismic Category I, the boundaries of the seismic Category I portions of the system are shown on the piping and instrument diagrams in appropriate sections of this UFSAR.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

System safety classifications and design and fabrication requirements meet the intent of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," February 1976 (R3).

Water, steam and radioactive containing components (other than turbines and condensers) are designated as Quality Group A, B, C or D in accordance with their importance to safety. This importance as emphasized by quality group assignment is considered in design, material, fabrication, assembly, construction and operation of the component. A single system may have components in more than one quality group.

System components important to safety and the containment boundary are classified in accordance with ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," 1973. The relationship, as stated therein, between Safety Class 1, 2, 3 and non-nuclear safety (NNS) and NRC Quality Group A, B, C and D is as follows:

Safety Class	Quality Group (Regulatory Guide 1.26)
Safety Class 1	A
Safety Class 2	B
Safety Class 3	C
NNS (Non-Nuclear Safety)	D

The quality group designations are given in Table 3.2-1 for applicable components. Corresponding minimum code requirements applied to the various components in each quality group are given in Table 3.2-2. Interfaces between components of different quality groups are designated on the various system P&I diagrams at the end of the appropriate sections in Chapters 5, 6,9,10 and 11.

a) Quality Group A

Quality Group A applies to reactor coolant pressure boundary components whose failure during normal reactor operations would prevent orderly reactor shutdown and cool down assuming makeup is provided by normal makeup systems only.

b) Quality Group B

Quality Group B applies to containment vessel and to those components:

i) of the Reactor Coolant System not in Quality Group A

ii) that are necessary to:

    1) remove directly residual heat from the reactor,

    2) circulate reactor coolant for any safety system* purpose,

    3) control within the reactor containment radioactivity released or control hydrogen in the reactor containment.

c) Quality Group C

Quality Group C applies to those components not in Quality Group A or B:

i) the failure of which would result in significant radioactive release to the environment or that are necessary to:

ii) provide or support any safety system* function,

iii) control of accident airborne radioactivity outside the reactor containment.

d) Quality Group D

Quality Group D applies to those components not related to nuclear safety.

* A safety system (in this context) is any system that functions to shutdown the reactor, cool the core or cool another safety system or the containment, and contains, controls, or reduces radioactivity released in an accident. Only those portions of the secondary systems are included (a) that are designed primarily to accomplish one of the above safety functions or (b) whose failure could prevent accomplishing one of the above functions.

TABLE 3.2-1
DESIGN CLASSIFICATIONS OF STRUCTURES, SYSTEMS AND COMPONENTS

STRUCTURE	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION (*)	FLOOD CRITERION (**)	QUALITY GROUP	NOTES
Shield Building	I	a	a	-	
Containment Vessel	I	b	c	B	
Reactor Building					
Interior structures	I	b	c	-	
Reactor Auxiliary Building	I	a	a	-	
Diesel Generator Building	I	a	a	-	
Intake Structure	I	a	a	-	
Fuel Handling Building	I	a	a	-	
Cask Crane Support Structure	I	-	-	-	(20)
Component Cooling Area Structure	I	a	b	-	
Diesel Oil Storage Tank Building	I	a	a	-	
Condensate Storage Tank Building	I	a	a	-	
Steel Missile Barriers (RAB, Intake Structure, DGB, CCWB, Condensate Storage Tank Building)	I	a	b	-	
Spent Fuel Pool and Liner	I	b	c	-	
Spent and New Fuel Storage Racks	I	b	c	-	
Main Steam Trestle	I	a	-	-	
Biological Shielding within RAB and FHB	I	b	c	-	
Radiation Shielding	I	b	c	-	
Roofs of Safety-Related Structures	I	a	a	-	
Class 1 backfill around safety-related structures	I	-	-	-	(19)
Emergency Cooling Water Canal Slope	-	-	-	-	(15)
SYSTEMS AND C0MPONENTS					
A. Reactor Coolant System					
1. Reactor vessel	I	b	c	A	
2. Reactor vessel internals	I	b	c	-	
3. Control rod drive mechanisms	I	b	c	-	
4 Control element assemblies	I	b	c	-	
5. Pressurizer	I	b	c	A	
6. Steam generator					
a) Primary side	I	b	c	A	
b) Secondary side	I	b	c	B	
7. Reactor coolant pump	I	b	c	A	
are part of RCPB	I	b	c	A,B	(1)
9. Reactor protection instrumentation	I	b	c	-	(2)

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION (*)	FLOOD CRITERION (**)	QUALITY GROUP	NOTES
B. Safety Injection System					
1. Safety injection tanks	I	b	c	B	
2. Refueling water tank	I	-	b	B	(18)
3. Pumps	I	b	c	B	
4. Piping and valves					(3)
a) Part of RCPB	I	b	c	A,B	(1)
b) Required only for initial injection	I	b	c	B	
c) Required for long term post-accident cooling	I	b	c	B	
d) Whose failure would prevent operation of portions of system covered in (a), (b) or (c)	I	b	c	C	
e) Normally isolated or automatically isolated from parts of system covered by (a), (b) or (c)	-	-	-	D	(4)
5. Instrumentation	I	b	c	-	(2)
C. Shutdown Cooling System					
1. Heat exchangers					
a) Reactor coolant side	I	b	c	B	
b) Component cooling water side	I	b	c	C	
2. Piping and valves					(3)
a) Part of RCPB	I	b	c	A	
b) Required for residual heat removal	I	b	c	B	
c) Normally isolated or automatically isolated from parts of system covered by (a) or (b)	-	-	-	D	(4)
3. Instrumentation	I	b	c	-	(2)
D. Chemical Volume and Control System					
1. Charging pumps	I	b	c	B	
2. Boric acid make-up tanks	I	b	c	B	
3. Boric acid pumps	I	b	c	B	
4. Letdown heat exchanger	-	b	c	C	
5. Regenerative heat exchanger	I	b	c	B	
6. Volume control tank	-	b	c	C	

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION (*)	FLOOD CRITERION (**)	QUALITY GROUP	NOTES
D. Chemical Volume and Control System (Cont'd)					
7. Boric acid batching tank	-	b	c	D	
8. Ion exchangers	-	b	c	C	
9. Pulsation dampeners	I	b	c	B	
10. Suction stabilizers	I	b	c	B	
11. Piping and valves					
a) Part of RCPB	I	b	c	A	
b) Required for letdown	-	b	c	C	
c) Required for post-accident injection of boric acid	I	b	c	B	
d) Normally or automatically isolated from parts of system covered by (a), (b) or (c)	-	-	-	D	(4)
12. Instrumentation	I	b	c	-	(2)
E. Containment Spray System					
1. Pumps	I	b	c	B	
2. Spray nozzles	I	b	c	B	
3. Piping and valves					(3)
a) Required for spray and recirculation	I	b	c	B	
b) Normally or automatically isolated from parts of system covered by (a)	-	-	-	D	(4)
4. Instrumentation	I	b	c	-	(2)
F. Waste Management System					
1. Reactor drain tank	-	-	-	D	
2. Flash tank	-	-	-	D	
3. Reactor drain pumps	-	-	-	D	
4. Holdup tanks	-	-	-	D	
5. Spent resin tank	-	-	-	D	
6. Flash tank pumps	-	-	-	D	
7. Gas surge tank	-	-	-	D	
8. Waste gas compressors	-	-	-	D	
9. Gas decay tanks	I	b	-	D	
10. Piping not part of Containment Isolation and Gas Decay Tank Isolation	-	-	-	D	(3)(4)
11. Piping & Valves for Gas Decay Tank Isol	I	b	-	D	
12. Piping & Valves for Containment Isol	I	b	c	B	

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION(*)	FLOOD CRITERION(**)	QUALITY GROUP	NOTES
F. Waste Management System (Cont'd)					
11. Radiation monitoring Instrumentation	-	b	c	-	
12. Flash tank gas vent trap	D	-	-	D	
13. All other WMS components	-	b	c	-	
G. Containment Cooling System					(6)
1. Fan coolers	I	b	c	B	
2. Ductwork	I	b	c	B	
3. Instrumentation	I	b	c	-	(2)
H. Component Cooling Water System					
1. Pumps	I	b	b	C	
2. Surge tank	I	b	c	C	
3. Heat exchangers	I	b	b	C	
4. Piping and valves					(3)
a) Required for performance of safety functions	I	b	b	C	
b) Normally or automatically isolated from parts of System covered by (a)	-	-	-	-	(4)
5. Instrumentation	I	b	b	-	(2)
I. Cooling Water Systems					
1. Intake cooling water pumps	I	b	b	C	
2. Circulating water pumps	-	-	-	D	
3. Piping and valves					(3)
a). Required for performance of safety functions	I	b	a,b	C	
b). Normally or automatically isolated from parts of system covered by (a)	-	-	-	D	(4)
4. Instrumentation	I	b	b	-	(2)

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION(*)	FLOOD CRITERION(**)	QUALITY GROUP	NOTES
J. Containment Isolation System					
1. Piping and valves (of all systems penetrating containment)					(3)
a) From first isolation valve inside containment or from containment penetration weld to outermost isolation valve	I	b	c	B	
2. Instrumentation	I	b	c	B	(2)
K. Main Steam and Feedwater System					
1. Piping and Valves from steam generator to outermost isolation valve	I	b	b	B	(3)
2. Instrumentation	-	b	b,c	-	(2)
L. Auxiliary Feedwater System					
1. Pumps	I	b	b	C	
2. Condensate storage tank	I	b	a	C	
3. Piping and valves					(3)
a) not normally or automatically isolated from Quality Group B components	I	b	b	B	(4)
b) Required for performance of safety functions	I	a	b	C	
4. Instrumentation	I	b	b	-	(2)
M. Emergency Power System					
1. Diesel generator sets	I	b	c	-	(7)
2. Diesel oil storage tanks	I	b	c	C	(7)
3. Diesel oil day tanks	I	b	c	C	(7)
4. Diesel oil transfer pumps	I	b	c	C	(7)
5. Diesel starting systems	I	b	c	C	(7)
6. Diesel generator control boards	I	b	c	-	
7. Safety-related 4.16 kv switchgear	I	b	c	-	
8. Plant emergency batteries and inverters	I	b	c	-	

TABLE 3.2-1 (Cont'd)

	SYSTEMS AND COMPONENTS	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION(*)	FLOOD CRITERION(**)	QUALITY GROUP	NOTES
M.	Emergency Power System (Cont'd)					
	9. Instrumentation	I	b	c	-	(2)
	10. Safety-related 480V switchgear, 125V dc 120V ac panels, transformers, motor control centers	I	b	c	-	
	11. Containment electrical penetrations	I	b	c	-	
	12. Safety system power, control and instrument cables and raceways	I	b	d	-	(12)
	13. Diesel Cooling Water System	I	b	c	C	(7)
	14. Diesel Lube Oil System	I	b	c	C	(7)
	15. Diesel Air Start System	I	b	c	C	(7)
	16. Diesel Air Intake & Exhaust	I	b	c	C	(7)
	17. Electric Manholes for Class 1 components	I	a	a	-	
	18. Cable Splice, connectors	-	b	c	-	(13)
	19. Terminal Blocks	I	b	c	-	(13)
	20. Underground cables	-	b	d	-	
N.	Sampling System					
	1. Piping and valves					(3)
	a) Part of RCPB	I	b	c	B	
	b) Normally or automatically isolated from Quality Group A or B components	-	-	-	D	(4)
0.	Combustible Gas Control System					
	1. Hydrogen recombiners	I	b	c	B	
	2. Containment Purge System	-	-	-	D	(5)
	3. Continuous Containment/Hydrogen Purge System	-	-	-	D	(5)
	4. Hydrogen Sampling System	I	b	c	-	(16)
	5. Instrumentation	-	-	-	-	(13) (14)
P.	Shield Building Ventilation System					(6)
	1. Fans	I	b	c	B	
	2. Filters	I	b	c	B	
	3. Ducting and dampers	I	b	c	B	
	4. Instrumentation	I	b	c	-	(2)

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION(*)	FLOOD CRITERION(**)	QUALITY GROUP	NOTES
Q. Ventilation Systems					(6)
1. Control Room AC and Emergency Cleanup Systems	I	b	c	C	
2. ECCS Area Ventilation System	I	b	c	C	
3. Reactor Auxiliary Building Main Supply System	I	b	c	C	
4. RAB Electrical Equipment and Battery Room Ventilation System	I	b	c	C	
5. Diesel Generator Building Ventilation System	I	b	b	D	
6. Intake Cooling Water Ventilation System	I	b	b	C	
7. Reactor Cavity Cooling System	I	b	c	D	
8. Reactor Support Cooling System	I	b	c	D	
9. CEDM Cooling System	I	b	c	D	
10. FHB Ventilation System (portion connected to SBVS)	I	b	c	C	
R. Fuel Pool Cooling and Purification System					
1. Cooling Loop					
a) Pumps	I	b	c	C	
b) Heat exchanger	I	b	c	C	
c) Piping and Valves	I	b	c	C	
d) Instrumentation	I	b	c	-	(2)
2. Purification Loop					
a) Pump	-	b	c	D	
b) Ion exchanger	-	b	c	D	
c) Piping and Valves	-	b	c	D	
d) Instrumentation	-	b	c	-	(2) (13)
S. Iodine Removal System					
1. Hydrazine pumps	I	b	c	B	
2. Hydrazine storage tank	I	b	c	B	
3. Piping and valves	I	b	c	B	(3)
4. Instrumentation	I	b	c	-	(2)
T. Fuel Handling System					
1 Fuel transfer tube and penetration assembly					
a) Transfer Tube	-	b	a	-	(17)
b) Bellows Assembly	-	b	a	-	(17)
c) Tube Closure Assembly	I	b	a	B	

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION(*)	FLOOD CRITERION(**)	QUALITY GROUP	NOTES
T. Fuel Handling System (Cont'd)					
2. Fuel transfer valve	--	b	c	D	
3. Reactor vessel head lifting rig	I	b	c	D	(17)
4. Containment polar crane	I	b	c	-	(17)
5. Spent fuel handling system	-	b	c	-	(13) (17)
6. Refueling Machine	-	b	c	-	(13) (17)
7. Fuel Transfer System	-	b	c	-	(13)
U. Radiation Monitoring System					
1. Safety area monitors	I	b	b	c	(8)
2. Safety effluent monitors	I	b	b	c	(9)
3. Process Monitors	I	b	b	c	(9)
4 Post-Accident Monitors	-	-	-	-	(11) (13)
V. Containment Vacuum Relief System					
1. Piping	I	b	c	B	
2. Valves	I	b	c	B	
3. Instrumentation	I	-	-	-	(13)
W. Reactor Coolant Gas Vent					
1. Piping & Valves					
a) Part of RCPB	I	b	c	A,B	
b) Required for RCS Post-Accident venting	I	b	c	B	
c) Required for RCS venting during normal operation only	-	b	c	D	
2. Instrumentation	I	b	c	--	(2)
X. Post-Accident Sampling System	-	b	c	D	(13)

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CATEGORY	TORNADO WIND/MISSILE CRITERION(*)	FLOOD CRITERION(**)	QUALITY GROUP	NOTES
Y. Post-Accident Monitoring Instrumentation NUREG-0737					
1) Containment Pressure Monitors (wide range)	I	b	c	-	
2) Containment Water Level Monitors (wide range)	I	b	c	-	
3) In containment High Range Radiation Monitor	I	b	c	-	
4) Plant Vent Accident Range Radiation Monitor	-	b	c	-	(13)
5) ECCS Area Ventilation System Exhaust Monitor	-	b	c	-	(13)
6) Atmospheric Steam Dump Exhaust Monitor	-	b	b	-	(13)

TABLE 3.2-1 (Cont'd)

Footnotes:

*	a	=	structure or component is designed to withstand design wind/tornado loadings and missile impacts.
	b	=	components housed within a structure designed to withstand wind/tornado loadings and missile impacts.
	c	=	separation of redundant components to preclude simultaneous failure by single missile impact.
**	a	=	structures and components designed to withstand flooding effects.
	b	=	positioning structures and components at sufficient elevation to preclude flooding.
	c	=	components housed within waterproof structure.
	d	=	cable is designed to operate in both wet or dry environments.

Notes:

1. Refer to 10 CFR 50 Section 50.2 for definition of reactor coolant pressure boundary (RCPB). Components excluded by footnote 4 to Section 50.55a are Quality Group B. EC290695
2. Instrumentation required to actuate, maintain operation of, or detect failure of equipment needed to safely shutdown, isolate and maintain the reactor in a safe condition and prevent uncontrolled release of radioactivity to the environment is seismic Category I. Instrumentation designated as seismic Category I includes sensing lines, except those whose breakage would cause the instrumentation to assume a "fail safe" position. Nonseismic sensing lines from seismic Category I piping or component are seismic Category I from the piping connection up to and including the root valve. Impulse tubing and associated valves are the same quality group as the piping and root valve from which the tubing commences. When a root valve or orifice is used for a quality group break, the root valve or orifice is classified to the higher quality group classification.
3. Valves are of the same quality group as connected piping. Valves which comprise an interface between piping of different quality groups are of the higher quality group.
4. Components of differing quality group other than Quality Group A are considered to be normally isolated from each other if separated by at least one valve which is always closed during reactor operation or open during testing, sampling or other routine operation of short duration which is under administrative control. Such components are considered to be automatically isolated if separated by at least one valve which closes automatically upon an appropriate Engineered Safety Features actuation signal or by a check valve which prevents flow from the higher to the lower quality group.
5. The only portion of the Containment Purge System and Continuous Containment/Hydrogen Purge System which Quality Group B is that portion performing a containment isolation function. The portion of ductwork located outside of the containment is seismically qualified.
6. Although this is not a steam or water containing system, it functionally corresponds to the Quality Group classification noted as per Regulatory Guide 1.26 (Rev. 3).
7. Those portions of the fuel oil, lube oil, jacket water, air start, combustion air intakes and exhaust systems required for the operation of the diesel generator sets are classified seismic Category I and Quality Group C. The Quality Group C design is considered an owner optional upgrade since RG 1.26 does not require the design of these systems to meet ASME Section III, Quality Group C requirements. See Subsections 9.5.4, 9.5.5, 9.5.6, 9.5.7, and 9.5.8.
8. For further information see Subsection 12.3.4.
9. For further information see Section 11.5
10. Deleted

TABLE 3.2-1 (Cont'd)

Footnotes (Cont'd)

11. For details on Post-Accident monitoring please refer to Chapter 7.

12. The raceway system is seismically analyzed.

13. Although certain systems or components are not designated quality group A, or B, or C nor seismic Category I, these items are subject to the pertinent requirements of the FP&L Quality Assurance program.

14. Instrumentation associated with H_2 Recombiners & H_2 Sampling System are seismic Category I.

15. Failure of canal slopes during seismic event is considered; flow blockage of the Emergency Cooling Water System is precluded by design.

16. See Subsection 6.2.5.2.1 for applicable criteria.

17. These components and associated supporting structures must be designed to retain structural integrity during and after a seismic event, but do not have to retain operability for protection of public safety. The basic requirement is prevention of structural collapse and damage to equipment and structures required for protection of public safety.

18. The RWT is not provided with missile shielding. The safety injection tanks are credited as a backup water source in the event the RWT is unavailable for safe shutdown due to a missile or tornadic wind event (see Section 9.3.4.3.1.3.5).

19. As part of the St. Lucie Unit 2 Component Replacement Projects, Engineering Evaluation No. PSL-ENG-SECS-07-014 was performed to demonstrate the acceptability of using Controlled Low-Strength Material (CLSM) as Class I or lesser classification backfill material in restoring excavated areas. Use of CLSM is limited by PSL-ENG-SECS-07-014 to areas that do not serve as foundation support for any Seismic Category I structures or support for any Seismic Category I buried piping.

20 Cask crane support structure is designed for tornado wind only (not missile impact).

TABLE 3.2-2

MINIMUM CODE REQUIREMENTS FOR QUALITY GROUPS

Component	Quality Group A	Quality Group B	Quality Group C	Quality Group D*
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Class 1	ASME Boiler and Pressure Vessel Code, Section III, Class 2	ASME Boiler and Pressure Vessel Code, Section III, Class 3	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, Manufacturer's Standards
Containment Vessel	-	ASME Boiler and Pressure Vessel Code, Section III, Class MC	-	-
0-15 Psig Storage Tanks	-	ASME Boiler and Pressure Vessel Code, Section III, Class 2	As above	API-620
Atmospheric Storage Tanks	-	As above	As above	API-650, AWWA D 100, ANSI B 96.1 or ASME Section VIII, Division I
Piping	As above	As above	As above	ANSI B 31.1.0
Pumps and Valves	As above	As above	As above	ANSI B 31.1.0 Manufacturer's Standards

Table 3.2-2 reflects minimum code requirements for Quality Groups used in original design. Replacement components may utilize alternate codes and edition/addenda as permitted by the PSL Unit 2 ASME Section XI program.

* As applicable, specific codes used are indicated in the design data tables for specific components.

3.3 WIND AND TORNADO LOADINGS

Structures, systems or components required for safe shutdown of the reactor or to prevent significant uncontrolled releases of radioactivity, are protected from failure due to wind or tornado loading by one of the following methods:

a) the structure or component is designed to withstand design wind and tornado loadings, or

b) the system or components are housed within a structure designed to withstand the wind and tornado loadings.

Table 3.2-1 lists seismic Category I structures and safety related systems and components and the requirement for wind/tornado protection where applicable. The a or b designation in the table refers to items a or b above. Figures 3.3-1 and 3.3-2 show the hurricane and tornado wind loading conditions for the Shield Building and Condensate Storage Tank Building; wind speeds and resultant static pressure loadings for other St. Lucie Unit 2 structures are presented in Table 3.3-1.

3.3.1 WIND LOADINGS

3.3.1.1 Design Hurricane Wind Velocity

The design hurricane wind speed is 194 mph. The design hurricane wind speed is selected by reference to a probable maximum hurricane (PMH) related to the site region. The parameters which are used to arrive at the PMH are described in Section 2.4. Hurricane wind data and history are given in Section 2.3.

Wind loads are determined and applied to all seismic Category I structures in accordance with procedures, incorporated in Reference 1, based on the design hurricane wind of 194 mph.

3.3.1.2 Determination of Applied Forces

The recommendations of ASCE paper No. 3269, "Wind Forces on Structures,[1] are used to derive the equivalent static loads corresponding to the design hurricane wind speed. The following equation is used for the dynamic wind pressure (q) in pounds per square foot:

$$q = 0.002558\ V^2 \qquad (1)$$

where:

V = wind speed, mph.

The local pressure, P_L, at any point on the surface of a building is equal to:

$$PL = C_{pe}q \qquad (2)$$

where:

C_{pe} = local pressure coefficient

C_{pe} depends upon the geometric shape of the building and the relative location of the point in question with respect to the direction of the wind. The values of C_{pe} for structures of different shapes are given in ASCE papers No. 3269[(1)] and No. 4933, "Wind Loads on Dome-Cylinder and Dome Cone Shapes."[(2)] Equation (2) gives the dynamic wind pressure on the surfaces of a building assumed airtight. If there are openings in the building surface, then the internal pressure will be increased or decreased depending on the location of the opening in relation to the wind direction. The corresponding local internal pressure, P_i, on the surface is derived as follows:

$$P_i = C_{pi}\, q \qquad (3)$$

where:

C_{pi} = internal pressure coefficient.

Reference 1 lists values of C_{pi} for different shapes of buildings.

In the design of walls and roofs, the design pressure, P_t, is the summation of both local external and internal pressures as given by equation (4) below:

$$P_t = P_L + P_i = q\,(C_{pe} + C_{pi}) \qquad (4)$$

The total wind force on the building is calculated by multiplying the average pressure, P, given by equation (5) below, with the projected elevation area:

$$P = C_D q \qquad (5)$$

where:

C_D = average drag or shape coefficient for the building, including the effects of both the pressure on the windward wall and the suction on the leeward wall.

3.3.2 TORNADO LOADINGS

3.3.2.1 Applicable Design Parameters

The parameters applicable to the design basis tornado which are in agreement with the requirements of Regulatory Guide 1.76 are:

a) external wind forces resulting from a tornado funnel with a horizontal rotation velocity of 300 mph and a horizontal translational velocity of 60 mph, for a total wind velocity of 360 mph. This is equivalent to RG 1.76 requirements of 290 mph rotational and 70 mph translational for a total wind velocity of 360 mph.

b) a decrease in atmospheric pressure of three psi at a rate of two psi/sec.

c) impact loads from the tornado generated missiles described in Subsection 3.5.1.4.

The design tornado applied to this site is extremely conservative. Florida tornadoes are much less severe as shown by two independent studies. Refer to Section 2.3 for discussions on the design basis tornado.

3.3.2.2 Determination of Forces on Structure

The tornado wind speed is converted into equivalent static pressure loading and the computations for wind pressure, their distribution on surface area of buildings, shape factors, and drag coefficients are based on the procedures outlined in Reference 1. Because of the unique characteristics of tornadoes, gust factor and velocity variation with height are considered uniform. With respect to the pressure distribution around the dome-cylinder Shield Building and Condensate Storage Tank Building wind force data reported in Reference 2 is used in the design.

The Shield Building has a diameter of 154 feet while the overall plan dimensions of the Reactor Auxiliary Building are approximately 115 by 240 feet. References 3 and 4 indicate that the combined maximum velocity of 360 mph is distributed over a narrow band width only. Data extrapolated from Reference 3 indicates an average band width of approximately 50 to 80 feet over which the combined velocity distribution of 360 mph is postulated to act. On this basis, a uniform wind speed of 300 mph for large seismic Category I structures is adopted for the Shield Building and Reactor Auxiliary Building. (The dimensions of the structures are significantly larger than the narrow band width indicated in References 3 and 4 mentioned above.)

Additionally, the tornado design wind speeds specified for the Shield Building and Reactor Auxiliary Building are considered conservative since the tangential velocity versus height above ground is held constant at the maximum average values. Based upon data presented in References 3 and 4, the tangential velocity reduces substantially close to the ground and, in particular in the height range of St. Lucie Unit 2 structures.

The ratio of the natural period of the elements of structures subjected to the pressure drop load and the 1 1/2 second rise time is such that the dynamic load factor does not exceed unity. Thus atmospheric pressure drop (three psi in 1 1/2 seconds) was applied as a static load. Because of the open design of the Diesel Generator Building no pressure differential is expected; nonetheless the structure is designed for a 2.25 psi differential at a rate of 2 psi/second. Based upon the large ventilation and cooling openings in the exterior walls, the reduced differential pressure was considered a conservative assumed design value; no calculations were performed to determine that value. Each of the two equipment housing compartments within the Diesel Generator Building contains approximately 49,200 cubic feet of air volume (before deducting equipment volume) and each compartment has approximately 618 square feet of available ventilation area after deducting wire screen area. Venting is not considered in any other structure.

The total tornado load, W_t, considers the effect of the uniform tornado wind load, W_w, differential pressure, W_p, and tornado missile load, W_m, have been considered using the following combination:

$$W_t = W_w + W_p + W_m \qquad (6)$$

Differential pressure and missile loadings are considered in combination with wind velocity pressure when they act in the same direction so as to ensure the most adverse situation.

Tornado generated missiles considered in the plant design, and analyses to determine the effect of these missiles, is presented in Subsection 3.5.1. Tornado missile analyses demonstrate that structures can absorb sufficient energy to stop missiles without perforation.

3.3.2.3 Effect of Failure of Structures or Components not Designed for Tornado Loads

All nonseismic Category I cranes and supporting structures located outdoors are designed for tornadic wind but not tornado generated missiles. Outdoor cranes are designed to remain on runways under tornado wind. Outdoor cranes are also parked away from safety related equipment as a further safety measure.

The main framing, columns, and bracings of the Turbine Building are designed to prevent the building from collapsing under 360 mph tornado wind loading. The three psi pressure differential is not considered since the Turbine Building is an open structure. Any failures that might occur in the subframing as a result of tornado wind loading do not affect adjacent safety related structures or equipment. There is no safety related equipment in the Turbine Building.

Remaining nonseismic Category I outdoor structures are located away from safety related equipment and are not designed for tornadic winds or tornado generated missiles.

SECTION 3.3: REFERENCES

1) ASCE 3269, "Wind Forces on Structures," American Society of Civil Engineers Transactions, Vol 126, Part II, 1961.

2) Mailer, F.J., "Wind Loads on Dome-Cylinder and Dome-Cone Shapes," Journal of the Structural Division, ASCE Vol 92, No. ST5, Paper 4933, October 1966.

3) Hoecker, W.H. Jr., "Three Dimensional Pressure Pattern of the Dallas Tornado and Some Resultant Implications," Monthly Weather Review, December 1961.

4) Dunlap J.A. and Wiedener K, "Nuclear Power Plant Tornado Design Considerations," ASCE Journal of the Power Division, March 1971.

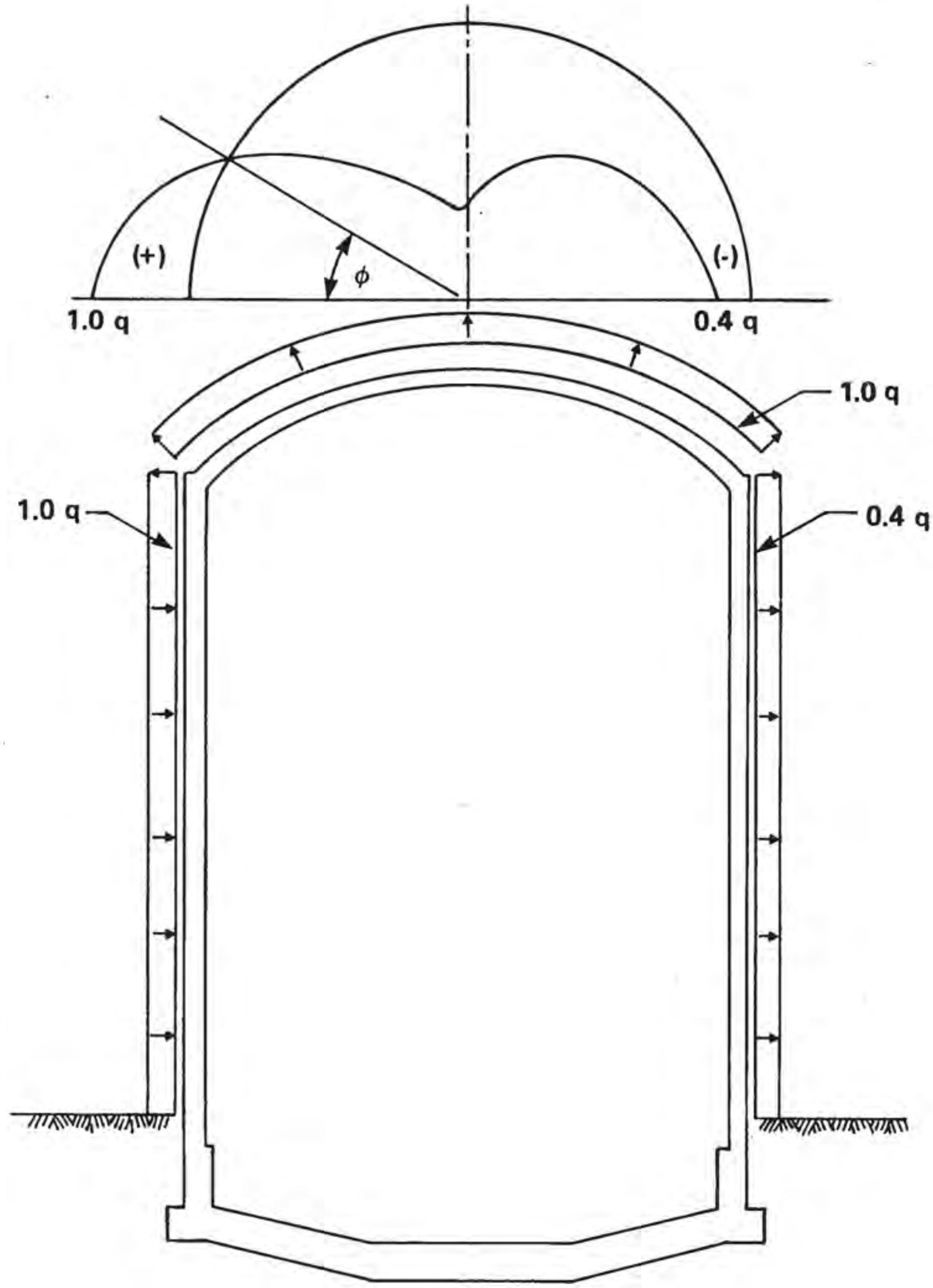
TABLE 3.3-1

WIND SPEEDS AND RESULTANT STATIC PRESSURE LOADINGS

Structure	Wind Speed, mph		Pressure Coefficient	External Pressure, psf	
	Hurricane	Tornado		Hurricane	Tornado
Reactor Auxiliary Building	194	300	.9 (1)	87	207
			-.5 (2)	-48	-115
			-.5 (3)	-48	-115
Fuel Handling Building	194	360	.9 (1)	87	298
			-.5 (2)	-48	-166
			-.8 (3)	-77	-265
Diesel Generator Building	194	360	.9 (1)	87	298
			-.4 (2)	-38	-132
			-.55 (3)	-53	-182
Component Cooling Water Structure	194	360	.9 (1)	87	298
			-.5 (2)	-48	-166
			-.5 (3)	-48	-166
Intake Structure-Pump Missile Protection Structure	194	360	.9 (1)	87	298
			-.5 (2)	-48	-166
			-.8 (3)	-77	-265
Diesel Oil Storage Tank Building	194	360	.9 (1)	87	298
			-.6 (2)	-58	-199
			-.8 (3)	-77	-265
Shield Building	194	300	See Figure 3.3-1		
Condensate Storage Tank Building	194	360	See Figure 3.3-2		

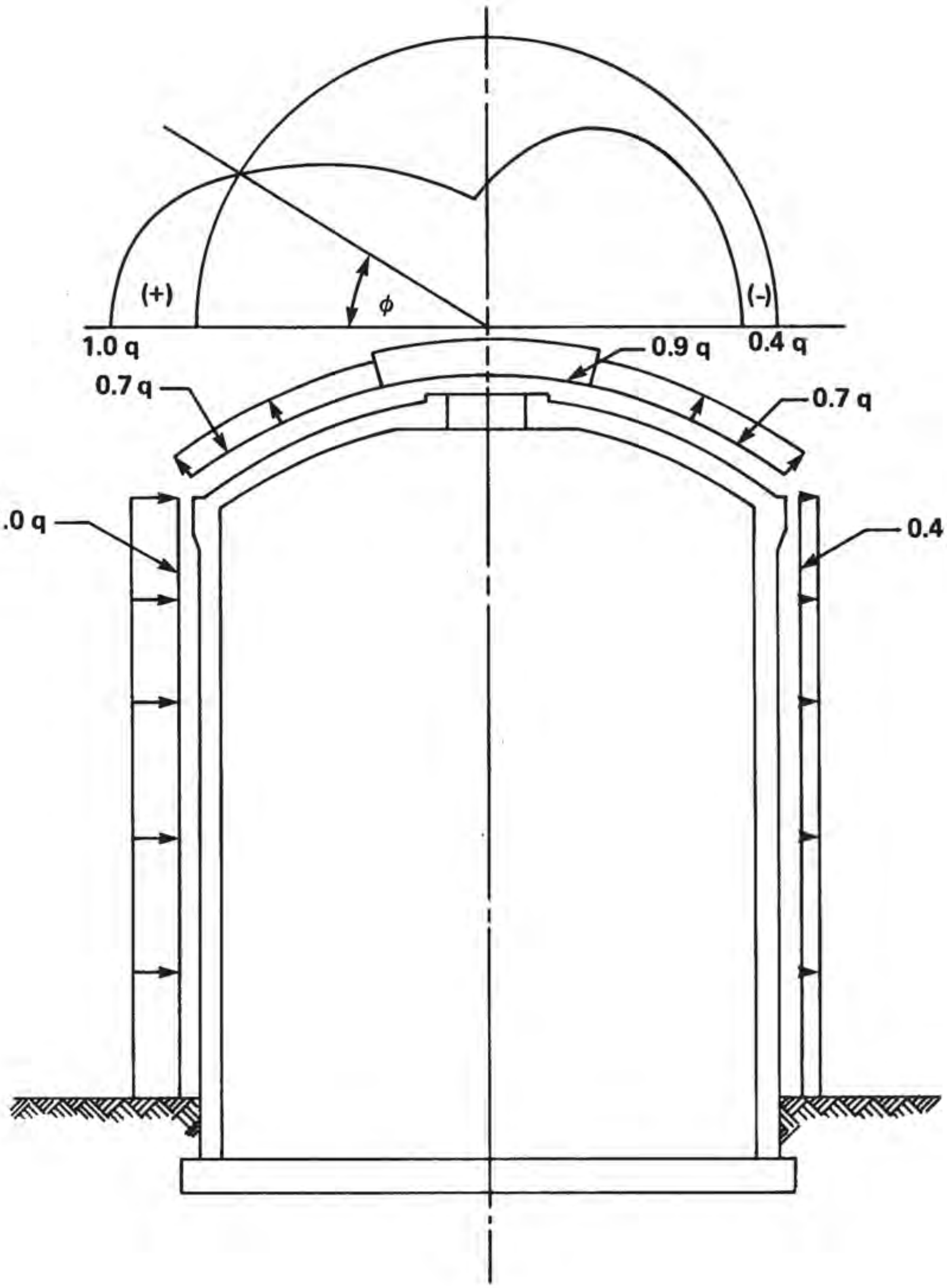
Notes:
(1) Windward
(2) Leeward
(3) Roof

"-" indicates suction



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

SHIELD BLDG. TORNADO &
HURRICANE WIND LOAD CONDITION
FIGURE 3.3–1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONDENSATE STORAGE TANK
TORNADO & HURRICANE WIND
LOAD CONDITION
FIGURE 3.3–2

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD PROTECTION

The plant grade is above the highest possible still-water levels attainable. During the probable maximum flood (PMF), the high water level is 17.2 ft, mean low water (MLW) and wave runups as specified in Table 2.4-9. The recommendations of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," August 1977 (R2), are followed except that the probable maximum hurricane pertinent to the site is the basis for the computation of the probable maximum surge (PMS). The PMS given in the regulatory guide for the St. Lucie plant site open coast is 16.7 ft. MLW. Even so, the plant grade is at elevation 18.5 ft. and minimum entrance elevation to all seismic Category I buildings is 19.5 ft.

Seismic Category I structures and safety related systems, and components are protected from the effects of high water level and wave runup that are associated with probable maximum hurricane (PMH) conditions by one or more of the following:

a) Designing structures and components to withstand such effects where functionally required.

b) Positioning of the structures and components such that they are located at sufficient grade to preclude inoperability due to external flooding.

c) Housing within waterproof structures. The Shield Building and Reactor Auxiliary Building are the only seismic Category I structures with basements. These structures are completely waterproofed to elevation 17.0 ft. Construction joints within seismic Category I structures, except the Component Cooling Water Structure, contain polyvinyl chloride waterstops up to elevation 17.0 ft. The Component Cooling Water Structure is not designed as a waterproof structure since all equipment is located above elevation 23.66 ft. on pedestals.*

Table 3.2-1 lists the flood protection criteria applied to plant structures, systems and components. The a, b, or c designation in the table refers to items a, b, or c above. The recommendations of Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Sept. 1976 (R1), are followed.

Figure 3.4-1 shows details for waterproofing at penetrations and interconnections between seismic Category I structures. Penetrations for pipes or electrical ducts are either encased in concrete where they penetrate the wall, or, where sleeves are used, enclosed in a pipe boot designed to prevent seepage. Boots are not used below the normal groundwater table.

Based upon the PMF high water level, wave runup level and plant island elevation noted above, flood protection stop logs at entrances (whose minimum elevation is at least +19.5 feet) to

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* Intake Cooling Water (ICW) debris discharge valves HCV-21-7A and HCV-21-7B are located below 23.66 feet. However, the actuator and electrical components for these valves are located above 19 feet, which is above the maximum flood level of 17.2 feet and the maximum Component Cooling Water Building backflood level of 18.5 feet.

safety-related buildings are not deemed necessary. Additional wave runup protection is provided to the entrances of the Fuel Handling Building and Reactor Auxiliary Building by the presence of adjacent buildings and structures (see Enlarged Site Plot Plan Figure 1.2-2). Since no permanent structures are located on the south side of the Reactor Auxiliary Building, additional wave runup protection has been provided by installing stop logs in the entrance on the south wall and the southernmost entrance on the east wall.

Figure 3.4-6 details the stop logs provided for the two RAB openings. Rectangular aluminum stop logs would be stacked to Elevation 22.0 feet and secured with bolts. Gaskets provide a seal at both the bottom and sides of the protected openings. Additional security in this design is provided by the ability to bolt the stop logs against the opening frame thus assuring that vertical or horizontal separation cannot take place. The stop logs are stored onsite in a manner that reserves their readiness for use. When a hurricane watch is posted for the plant, the stop logs are removed from storage and prepared for installation; with actual installation occurring when the "hurricane warning" is posted for the plant.

The hurricane protection groins and bulkheads are protected from corrosion by impressed current systems. Florida Power & Light requires periodic checks of the cathodic protection system in accordance with plant procedures. Any equipment found nonfunctional is addressed via the work order/corrective action program.

The sheetpile bulkhead at the nose of the Discharge Canal is covered with a concrete cap from one foot below grade (grade elevation is elevation 18.0) to elevation 22.0 feet. The concrete cap is detailed on Figure 3.4-7.

Since both paving in the vicinity of buildings and the high points of roads are at or below elevation 19.0 ft, no pending could occur above the elevation of seismic Category I building entrances (which are elevation 19.5 ft. as a minimum).

There are only two cases where storm water, resulting from coincident PMH and PMP conditions, could back up into any structure containing safety-related equipment or system; i.e., electrical manholes and the Component Cooling Water Building.

Back flooding of electrical manholes would result only if manhole drain line check valves failed to operate properly. The cables can function in a submerged condition.

Due to the arrangement of the Component Cooling Water Building, back flooding would occur from the site drainage system through the building sump. (The building roof drain piping is not interconnected to the interior drainage system and is serviced by separate branches of the site drainage system.) The elevation of the top of the base mat is at elevation 12.00 feet, elevation of the building sump is at elevation 9.67 feet, and safety-related equipment are supported on concrete piers at elevation 23.66 feet.

Back flooding of the Component Cooling Water Building cannot be greater than the elevation of water within the catch basins. Assuming that the catch basins fill to grade elevation, the water within the catch basins and building will be approximately elevation 18.5 feet. Electrical conduits and mechanical piping and valves located below elevation 18.5 feet can function in a submerged condition.

Section 2.4.2.4 addresses drainage of water from the southern site property and the effect of water pooling caused by the intake canal berm.

For the remaining seismic Category I buildings with roof drains (Shield Building, Reactor Auxiliary Building and Diesel Oil Storage Tank Building), interconnection between roof drain piping and interior floor slab drainage system does not occur.

The Fuel Handling Building has an exterior curb and leader system. DG Building and Condensate Storage Tank Building allow rain water to runoff the edge of the building to grade.

The flood penetration provided on and within exterior walls of seismic Category I structures with basements (Reactor Building and Reactor Auxiliary Building), consists of waterproofing membranes and polyvinyl chloride (PVC) water stops, respectively. Radiation levels at exterior walls of those buildings are well below the levels which PVC can tolerate without appreciable damage (5×10^5 rads) and the level at which corrosive gases could be liberated (approximately 10^6 - 10^7 rads based upon the behavior of similar compounds).

The only area on St. Lucie No. 2 where radiation levels could be high enough to cause damage to PVC water stops was within the Shield Building Steel Containment Structure. Rubber water stops were specified for use in construction joints below elevation 23 within the Steel Containment Structure. Rubber has a threshold to damage above 2×10^6 rads and does not liberate gases until very high radiation levels are reached (above 10^8 rads) which is considerably above the maximum radiation levels predictable for the plant.

At elevation 19.0 feet, roads have the highest contours of plant island grading features.

The list below designates each seismic Category I structure with identification of exterior or access openings.

Structure	Exterior or Access Openings
Shield Building	No openings below elevation 22 feet
Reactor Auxiliary Building	Minimum entrance elevation at 19.5 feet
Fuel Handling Building	Minimum entrance elevation at 19.5 feet
Diesel Generator Building	Floor and equipment above elevation 22 feet and no openings below elevation 22.67 feet
Diesel Oil Storage Tank Building	No openings below elevation 29.5 feet
Condensate Storage Tank Structure	No openings below elevation 22.0 feet
Component Cooling Water Building	Equipment is located above elevation 23.66 feet
Intake Structure	Motors located above elevation 22 feet

The St. Lucie Unit 1 and Unit 2 site drainage plans are shown on Figures 3.4-3 and 3.4-4, respectively. In areas where St. Lucie Unit 1 drain lines carry storm water from both units, the lines are sized to accommodate the additional flow. The connections between St. Lucie Unit 1 and Unit 2 drainage lines and the St. Lucie Unit 1 modifications to handle these connections are shown on Figure 3.4-4.

Drain lines are sized to accommodate runoff in the plant area, estimated by relating the tributary area and the rainfall intensity to an estimated proportion of the rainfall reaching the catch basin as direct runoff.

The ISFSI drainage plan is shown in Figure 3.4-4a.

This procedure is represented by the following formula:

$$Q = ACIp$$

where

Q = design discharge, cfs
A = tributary drainage area, sq ft
C = runoff coefficient based on surface conditions
I = intensity of rainfall, in/hr
p = coefficient based on percent of full pipe flow

The drain line design considered values of C consistent with use in the Rational Method for various ground surface types. The intensity of rainfall, I, used in the calculations was six inches per hour. The tributary drainage area is determined by the location of surrounding catch basins and storm drain lines.

Catch basins are constructed to provide ready access to storm drains for inspection and maintenance as well as to serve as points of concentration for runoff. Runoff computations for catch basins include roof, floor and equipment drains.

The site drainage system and building drainage systems are designed to preclude flooding of safety related structures under PMH conditions, except in the Component Cooling Water Structure where components are located above the wave runup elevation.

Flooding of electrical manholes through backup within the site drainage system is prevented by the installation of check valves on the ends of the electrical manhole drainage system headers within the site drainage system catch basins. When the bottom elevation of an electrical manhole sump is below the catch basin elevation, a manhole sump pump is provided. Figures 3.4-4 and 3.4-5 show the unit grading and drainage, and electrical manhole drainage system. If flooding of an electrical manhole were to occur through inoperability of a check valve or inleakage through manhole construction joints and manhole roof vents, the flood water is prevented from entering connecting structures because the construction openings within those structures are filled with concrete and constructed with waterstops. Operability of submerged cables is addressed in Section 3.11.

The Reactor Building and Reactor Auxiliary Building are constructed with waterproofing to elevation 17.0 feet and therefore protected from inleakage from phenomena such as cracks in exterior walls. The remaining seismic Category I structures are founded above the groundwater table and therefore waterproofing is not required. Potential inleakage within these structures through concrete during PMH flooding is collected by floor drainage systems and routed to sumps for removal.

Permanent doors in the exterior walls at the Reactor Auxiliary, Fuel Handling, and Diesel Generator Buildings provide protection from rain, wind, and other atmospheric effects.

3.4.1.2 Permanent Dewatering System

There is no permanent dewatering system on St. Lucie Unit 2.

3.4.2 ANALYSIS PROCEDURES

Seismic Category I structures can be subjected to the maximum flood level and other environmental effects of the PMH including wind loads. All structures except the Reactor Building are designed for a buoyant soil loading plus a hydrostatic pressure condition up to elevation 17.0 feet (PMF level) and a saturated soil loading condition from elevation 17.0 feet to grade. The Reactor Building is designed for a buoyant soil loading to grade plus a buoyant and hydrostatic pressure condition up to elevation 21.0 feet. This condition, called the buoyant loading condition, accounts for conditions of maximum buoyance and flooding. Subsection 3.8.4 provides the definitions and the equations used.

The gasketed aluminum stop logs are designed to withstand PMH wind loads and hydrostataic pressure resulting from water up to elevation 22.0 feet.

3.4.3 RAB INTERNAL FLOODING DUE TO EQUIPMENT RUPTURE

The following is an analysis of the catastrophic failure of all four 40,000 gallon tanks concurrent with the failure of all other non-seismic Category I components, including all sump pumps, volume control tank, associated piping, and with the worst single active failure, and the resulting flooding of safety-related equipment.

The storage capability of all non-seismic tanks in the RAB has been considered. If it were assumed that every non-seismic tank ruptured during a seismic event, water from such a rupture could eventually drain toward the ECCS pump room sumps located at elevation -10 feet. Each sump is 4 feet x 4 feet x 10 feet deep with a capacity of 1,100 gallons. The pump room is divided into two subcompartments by a flood wall which extends a minimum of 9.5 feet high. Each ECCS compartment houses the minimum complement of the required engineered safety feature pumps.

The fluid from the ruptured tanks would drain into the "A" ECCS pump room via 4 inch and 3 inch lines with a total capacity of no more than 130 gpm, whereas drainage into the "B" ECCS pump room sump is via multiple 4 inch and 3 inch lines with a total capacity of no more than 210 gpm. Any substantial release of water inventory to El -0.5 feet will drain into both ECCS pump rooms (El -10.0 feet) and into the Reactor Drain Pump Room (El -3.5 feet) and flood the -0.5 feet level of the RAB. EC281756

The analysis of the ECCS Flooding Protection and the associated sequence of events are presented in Table 3.4-1. The most limiting components within the ECCS pump room are the HPSI pump conduit boxes located 18 3/4 inches above the floor elevation. The analysis reveals that the fluid level will not reach HPSI Pump B conduit box within 102 minutes and HPSI Pump A within 145 minutes after the accident which is ample time for the operator to isolate the ECCS pump room areas. Upon isolating the ECCS pump room cubicle, the water will accumulate in the -0.5 feet level. However, no safety-related equipment on this level will be affected. The following design modifications were incorporated into the ECCS cubicle design and are available to the operator:

a) Each ECCS pump room contains a seismic Category I, class 1E level switch which provides high sump level alarms in the control room. A high-high sump level signal is also used in each alarm logic circuit for greater reliability.

   The level switches are physically separated and electrically independent from each other. A backup seismic Category I level switch with control room alarms is also provided in each sump in order to provide greater reliability. The analysis shows that within eight minutes the operator receives four signals from four independent sources notifying him of the accident.

b) Floor drain lines entering the ECCS pump room compartments are provided with redundant seismic Category I isolation valves. These isolation valves have the capability of remote-manual operation from the control room. The analysis shows that the flood level will not reach the isolation valves (El -7.5 feet) in ECCS cubicle B until 142 minutes after the accident.

c) Each entrance to the ECCS cubicle is provided with watertight doors thereby assuring that gross water volume will not flood the ECCS pump room.

From the above analysis it is concluded that a potential flooding incident in the Reactor Auxiliary Building cannot impair the ability of redundant equipment to achieve a safe shutdown condition.

TABLE 3.4-1

RAB CATASTROPHIC FLOODING ANALYSIS

Time (Min)	Event
0	Flood is initiated by rupture of all tanks in the RAB
4.1	High level alarm in ECCS Room “B” sump actuated in Control Room
5.0	High-high level alarm in ECCS Room “B” sump actuated in Control Room
7.1	High level alarm in ECCS Room “A” sump actuated in Control Room
8.0	High-high alarm in ECCS Room "A” sump actuated in Control Room
102	Flood level reaches bottom HPSI Pump 2B Conduit box
142	Flood level reaches lowest of ECCS Room B redundant Isolation Valves
145	Flood level reaches bottom of HPSI Pump 2A Conduit Box
197	Flood level reaches bottom of ECCS Room A redundant Isolation Valves

Refer to Drawing
2998-G-584

FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

REACTOR AUXILIARY BUILDING
EXT. WALLS - MISC DETAILS - M & R

FIGURE 3.4-1

FIGURE 3.4-2

HAS BEEN INTENTIONALLY DELETED

Refer to Drawing
8770-G-483

FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

SITE GRADING & DRAINAGE -
SH. 1
FIGURE 3.4-3

Refer to Drawing
2998-G-483

FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

SITE GRADING & DRAINAGE -
SH. 1
FIGURE 3.4-4

Refer to Drawing
8770-G-921 SH 9

FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

ISFSI PROJECT - GRADING & DRAINAGE
DRAINAGE PLAN

FIGURE 3.4-4a

Refer to Dwg.
2998-G-486

FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

ELECTRICAL MANHOLE AND
HANDHOLE DRAINAGE SYSTEM
FIGURE 3.4-5

Refer to Dwg.
2998-G-839 SH 1

FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

FLOOD CONTROL STOP LOGS

FIGURE 3.4-6

Amendment No. 10, (7/96)

Refer to Dwg.
8770-G-719

FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

CIRCULATING WATER SYSTEM
DISCHARGE CANAL NOSE PROTECTION
FIGURE 3.4-7

3.5 MISSILE PROTECTION

Missile protection is provided so that missiles from internal and external sources do not cause or increase the severity of a loss-of-coolant accident (LOCA), damage Engineered Safety Features when their operation is required to mitigate the consequences of an accident, jeopardize primary containment function as a radioactive material barrier during and following accidents that release radioactive material into the containment vessel, damage fuel stored in the spent fuel pool, prevent safe shutdown of the reactor, damage systems or components whose failure could result in a release of radioactivity that could result in potential offsite doses exceeding guidelines established for design basis accidents or jeopardize structural integrity of seismic Category I structures.

3.5.1 MISSILE SELECTION AND DESCRIPTIONS

3.5.1.1 Internally Generated Missiles (Outside Containment)

Two potential sources of missiles outside containment are evaluated:

a) Pressurized component failure, and

b) Rotating component failures.

Internally generated missiles are selected as follows:

a) Pressurized Component Failure Missiles

   1) Pressurized components in systems whose service temperature exceeds 200°F or whose design pressure exceeds 275 psig for more than two percent of the time that it operates below 200°F or 275 psig are evaluated as to their potential for becoming a missile.

   2) Temperature detectors (instrument temperature wells) installed in piping if failure of a single circumferential weld could cause their ejection. As indicated in Table 3.5-1, subcompartment walls are assumed to afford adequate protection to equipment located outside the affected area.

   3) For pressure seal bonnet-type valves, bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke, which would capture the bonnet or reduce bonnet energy. Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable, and hence bonnets are not considered credible missiles.

   4) Bolted bonnets valves are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material by rules set forth in the ASME Code, Section III, and by designing flanges in accordance with applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing a simultaneous complete severance failure is very remote. Therefore, bolted valve bonnets are not considered credible missiles.

5) Valve stems are not considered postulated missiles where at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with back seats, and motor or air operated valves are considered to have sufficient restraints, so that the valve stem does not become a missile.

   Stem ejection from a backseated valve would require the complete severance of all stem threads or the failure of the valve operator and in addition the catastrophic failure of a hardened backseat. This is applicable to valves in an open, closed or any intermediate position. The stems of normally open valves rest against the backseat preventing incipient stem ejection at the onset. In all other cases the backseat is capable of arresting stem ejection after the initiation of movement since the stem cannot attain significant energy in its limited travel before backseating.

6) Nuts, bolts, nut and bolt combinations, and nuts and stud combinations have only a small amount of stored energy and thus are of no concern as potential missiles.

7) The only high pressure gas bottles or accumulators located on the St. Lucie Unit No. 2 site are those provided as part of the hydraulic operators for the main feedwater isolation valves. Each operator has two cylinders containing hydraulic fluid charged with nitrogen gas to a pressure of 1900 psi. These cylinders are designed to withstand pressures greatly in excess of that experienced during normal operation. Additionally, as an integral part of the valve operators the cylinders undergo full environmental, seismic and operability qualification testing. In the unlikely event that a cylinder should become a missile, however; its orientation and location on the steam trestle would prevent it from damaging other safety related equipment.

b) Rotating Component Failure Missiles

1) All rotating components, which are operated during normal operating plant conditions, are evaluated to determine the potential for becoming missiles.

   The conservative design of rotating equipment (i.e. centrifugal pumps, compressors, fans) makes it very unlikely that a rotating component will fail to such an extent that a high energy missile will develop. Highly stressed conditions that would result from overspeed is prevented by the provision of protection devices for turbines (see UFSAR Table 3.5-2) and by the design of induction motors. The low usage factor of most rotating equipment also serves to reduce the probability of missile generation.

   Notwithstanding the above, all centrifugal pumps in St. Lucie Unit 2 have been investigated to determine the possibility that a high energy missile might develop. Using accepted methods for determining missile penetration (specifically, the Ballistic Research Laboratories' formula referenced in ANSI N177, Plant Design Against Missiles) all pump casings have been shown to be either capable of retaining internally generated missiles or have been protected from the effects of missiles by physical barriers. Thus, rotating equipment is not a credible source of missiles. The main turbine is further analyzed in UFSAR Subsection 3.5.1.3.

In addition to the above considerations, all safety related equipment is provided with protection from externally generated missiles (see UFSAR Table 3.5-3). This insures that the failure of a rotating component cannot disable equipment, other than the component itself, necessary for accident mitigation or plant shutdown.

2) The auxiliary feedwater pump turbine and diesel generators, which are designed and manufactured to the Quality Group C standards are designed to prevent overspeed. In addition, the diesel generators are not normally operating and the auxiliary feedwater pumps are normally used only during plant startup and shutdown. Therefore, no missiles are postulated.

3) Motor operated pumps and fans have induction motors which by their design will not allow operation above synchronous speed. All centrifugal fans, located adjacent to the safety related equipment, have a casing thickness exceeding the thickness required to stop the self generated missiles at synchronous speeds. Roof ventilation fans in the electrical equipment and battery rooms (RV-1, 2, 3 and 4) and the Diesel Generator Building (RV-5 and 6) are located above the roof and enclosed in a steel plate design. Therefore missiles generated by rotating blades associated with these fans do not impinge on any safety related equipment. The internal energy of the self generated missiles by the pumps is considered to be insufficient to penetrate the pump casings.

Table 3.5-1 lists the missiles generated by high energy systems considered outside containment. Table 3.5-2 provides a listing of overspeed protection provided for turbines and diesel generators. A tabulation of seismic Category I structures, and safety related systems and components is given in Table 3.2-1. General arrangement and section detail drawings are located in Section 1.2. Enclosures for components required for safe shutdown of the reactor under all conditions of plant operation are provided in Table 3.5-3.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Two potential sources of missiles inside the containment are evaluated:

a) Pressurized component failures, and

b) Rotating component failure.

The bases for selection are identical to those described in Subsection 3.5.1.1.

A tabulation of missiles generated from failures of pressurized components, their source and characteristics, and provided missile protection, is given in Table 3.5-4. A tabulation of seismic

Category 1 structures, and safety- related systems and components is given in Table 3.2-1. General arrangement and section detail drawings are located in Section 1.2.

To preclude internally generated missiles, the following design criteria, procedures and controls have been implemented to avoid damage to safety- related equipment from potential gravity missiles inside the containment.

a) Structural steel inside the containment is designed for the SSE.

b) All Class 1E electrical equipment and associated raceways (cable trays, conduits and boxes) located inside the containment are seismically supported. Maximum use of existing seismic Category I steel is utilized to support non-safety raceway systems. Non-Class 1E electrical equipment and raceways are not seismically supported when an analysis demonstrates that the falling of this equipment will not endanger any Class 1E equipment.

c) All H&V ducts inside containment are seismically supported to prevent gravity missiles.

d) Non-seismically supported piping has been routed away from safety-related equipment.

3.5.1.3 Turbine Missiles

Modern design, manufacturing and testing practices make the possibility of a major turbine structural failure extremely remote. In-service inspection of the turbine ensures that flaws arising during turbine operation are detected and repaired long before they become even a potential challenge to turbine structural integrity. FPL complies with the turbine vendor's NRC approved inspection schedule and refurbishment recommendations.

In the past, evaluation of the likelihood of turbine missiles as related to public health and safety followed Regulatory Guide (RG) 1.115 (Ref. 1), and Standard Review Plan (SRP) (Ref. 2). The probability of unacceptable damage from turbine missiles (P_4) was expressed as the product of the following: the probability of turbine missile generation resulting in the ejection of turbine disc fragments through the casing (P_1), the probability of ejected missiles perforating intervening barriers and striking safety-related SSCs (P_2), and the operability of struck SSCs failing to perform their safety functions (P_3).

The NRC staff has shifted its emphasis in the review of turbine missiles from the strike and damage probability, $P_2 \times P_3$, to the missile generation probability, P_1. The minimum reliability requirement for loading the turbine and bringing the system on line has been established in Appendix U of NUREG-1048 as $P_1 < 10^{-4}$ for favorably oriented turbines and $P_1 < 10^{-5}$ for unfavorably oriented turbines (Ref. 3). A favorable orientation has the turbine generator train perpendicular to the reactor building and an unfavorable orientation has the turbine generator train parallel with the reactor building. St. Lucie has an unfavorable orientation.

Currently, the NRC staff maintains that the maintenance and inspection of turbine rotors and valves are to be based on the P_1 calculation, operating experience of similar equipment and inspection results. The current NRC approved methodology applied by Siemens Energy Inc. for St. Lucie derives the probability of generating an external missile, P_1, as the sum of the probability of an external missile for speeds up to 120% of rated speed and the probability of an external missile for speeds greater than 120% of rated speed due to failure of the overspeed protection system (Ref. 4). This methodology and results of the analyses are described further in Section 3.5.1.3.2.

3.5.1.3.1 Description of Turbine Elements, Placement and Orientation

The placement and orientation of the turbine generator relative to the other structures of the plant is shown in Figures 3.5-1 and 3.5-2.

High Pressure Turbine

The high pressure turbine element is of a double flow design thus it is inherently thrust-balanced. Steam from the four control valves enters at the center of the turbine element through four inlet pipes, two in the base and two in the cover. Steam entering the HP turbine passes through the diagonal stage and flows through four reaction stages, all mounted on the inner casing upstream of the extraction. Downstream of the extraction, steam flows through four reaction stages mounted on the guide blade carriers. The inner casing and the guide blade carriers are mounted on the outer casing.

The outer casing cover and base (upper and lower half) are held together by means of more than 100 studs. Studs have lengths ranging from 17 to 66 inches and diameters ranging from 2.5 inches to 4.5 inches.

Low Pressure Turbine

The LP turbines are of a double flow design. The Siemens 13.9m^2 upgrade for EPU consists of a double flow rotor assembly with blades, inner casing, guide blade carriers and stationary blade rings. The rotor assembly consists of a shaft with six shrunk-on discs of the Advanced Disc Design as described in Ref. 4. The LP outer casing has been retained. Steam enters at the top of each outer cylinder where it flows to the inlet chamber of the inner casing. In the inlet region, the steam is distributed equally to both halves of the rotor and flows through the blading to the condenser. LP turbines are numbered from the high pressure (HP) element to the generator, with the lowest numbered LP element (LP1) located next to the HP element, and the highest numbered LP element (LP2) located next to the generator.

3.5.1.3.2 Turbine Generated Missile Identification and Characteristics

Missiles are generated due to structural failure of turbine discs. Following such a failure, the high rotational energy of the turbine can cause the disc and cylinder fragments to penetrate the turbine casing and become airborne missiles. Due to the large mass and high velocity of these missiles an evaluation is required, to determine the possible damage effects of turbine missiles. The turbine failures are classified into two general types and are referred to as design overspeed failures and destructive overspeed failures.

3.5.1.3.2.1 High Pressure Turbine Generated Missiles

The most significant source of turbine missile is a burst-type failure of one or more bladed shrunk-on disks of the low-pressure (LP) rotors. Failures of the high-pressure (HP) and generator rotors would be contained by relatively massive and strong casings, even if failure occurred at maximum conceivable overspeed of the unit. There is a remote possibility that some minor missiles could result from the failure of couplings or portions of rotors which extend outside the casings. These missiles would be much less hazardous than the LP disk missiles, due to low mass and energy and therefore do not require further consideration.[8]

3.5.1.3.2.2 Low Pressure Turbine Generated Missiles

Previous Missile Analyses:

The initial missile analyses documented in Ref. 6 assumed the probability of missile generation (P_1) to be approximately 10^{-4} per turbine year, based on the historical failure rate. The strike probability (P_2) was estimated on the basis of postulated missile sizes, shapes and energies and on available plant specific information such as turbine placement and orientation, number and type of intervening barriers, target geometry, and potential missile trajectories. The damage probability (P_3) was generally assumed to be 1.0. The overall probability of unacceptable damage to safety-related systems (P_4), which is the sum over all targets of the product of these probabilities, was then evaluated for compliance with the NRC safety objective. This logic places the regulatory emphasis on the strike probability, that is, it necessitates that P_2 be made less than or equal to 10^{-3}, and disregards all of the plant specific factors that determine the actual P_1 and its unique time dependency.

Although the calculation of strike probability is not difficult in principle, for the most part being not more than a straightforward ballistics analysis, it presents a problem in practice. The problem stems from the fact that numerous modeling approximations and simplifying assumptions are required to make tractable the incorporation into acceptable models of available data on the following: (1) properties of missiles, (2) interactions of missiles with barriers and obstacles, (3) trajectories of missiles as they interact with and perforate (or are deflected by) barriers, and (4) identification and location of safety-related targets. The particular approximations and assumptions made tend to have a significant effect on the resulting value of P_2. Similarly, a reasonably accurate specification of the damage probability (P_3) is not a simple matter because of the difficulty in defining the missile impact energy required to render given safety-related systems unavailable to perform their safety functions and the difficulty in postulating sequences of events that would follow a missile-producing turbine failure.

NRC Staff Current Approach:

In view of operating experience and NRC safety objectives, the NRC staff has shifted emphasis in the reviews of the turbine missile issue from the strike and damage probability ($P_2 x P_3$) to the missile generation probability (P_1) and, in the process, has attempted to integrate the various aspects of the issue into a single, coherent evaluation.

Through experience of reviewing various licensing applications, the staff has concluded that $P_2 x P_3$ analyses provide only “ball park” or “order of magnitude” values. Based on simple estimates for a variety of plant layouts, the staff also concludes that the strike and damage probability product ($P_2 x P_3$) can be reasonably taken to fall in a characteristic narrow range which is dependent on the gross features of plant layout with respect to turbine generator orientation; i.e., (a) for favorable oriented turbine generators $P_2 x P_3$ tends to lie in the range of 10^{-3} to 10^{-2}. In addition, detailed analyses such as those discussed in this evaluation show that, depending on the specific combination of material properties, operating environment, and maintenance practices, P_1 can have values from 10^{-9} to 10^{-1} per turbine year depending on the turbine test and inspection intervals. For these reasons, in the evaluation of $P_4 = (P_1 x P_2 x P_3)$, the probability of unacceptable damage to safety-related systems from potential turbine missile, the staff is giving credit for the product of the strike and damage probabilities ($P_2 x P_3$) of 10^{-3} for a favorably oriented turbine and 10^{-2} for an unfavorably oriented turbine (St. Lucie orientation), and is discouraging the elaborate calculation of these values.

By maintaining an initial value of P_1 through turbine testing and inspection provides a reliable means of ensuring that the objectives precluding turbine missiles and unacceptable damage to safety-related structures, systems, and components can be met. It simplifies and improves procedures for evaluation of turbine missile risks and ensures that the public health and safety is maintained.

St. Lucie Current Licensing Basis:

For these reasons, strike and damage calculations were not performed for the current St. Lucie Unit 2 licensing basis for EPU. An alternative methodology to that of Ref. 6 has been developed by Siemens Energy Inc. and approved by NRC (Ref. 4). This methodology determines the probability of an external missile (P_1) to be the sum of the probability of an external missile for turbine speeds up to 120% of rated speed (P_r) and the probability of an external missile for turbine speeds greater than 120% of rated speed (P_o). This methodology determines the external missile probability based on a turbine disc inspection interval of 100,000 hrs and quarterly turbine valve tests provided that no cracks are detected in the discs. These results are then compared to the NRC minimum reliability requirement of $P_1 < 10^{-4}$/yr for favorably oriented turbines and $P_1 < 10^{-5}$/yr for unfavorably oriented turbines (St. Lucie orientation) (Ref. 3). In order to apply the approved methodology, the NRC requires the following:

a. The approximate date for the turbine disc inspection at the end of 100,000 hrs of operation of the rotors,

b. A commitment to inform the NRC about the turbine disc inspection results and plans to reduce the probability of turbine missile generation, P_1, for continued operation should cracks be detected in the inspection, and

c. Justification for any additional turbine missile analyses, or minor deviations that may be plant specific.

Further, as documented in PSL-ENG-SENS-08-077 (Ref. 7), St. Lucie has elected to perform turbine valve test intervals at a frequency of every 6 months instead of quarterly.

A missile probability analysis was then performed for the St. Lucie Unit 2 low pressure turbines which include the upgraded BB281-13.9m2 rotors with Advanced Disc design shrunk-on discs (Ref. 8) by applying the currently approved methodology (Ref. 4) along with the extended 6 month valve test interval (Ref. 7). Based on the conservative assumptions applied, the probability of an external missile for speeds up to 120% of rated speed (P_r) is 2.88×10^{-7}/yr for a disc inspection interval of 100,000 operating hours. Applying the 6 month valve test interval, the probability of an external missile for speeds greater than 120% of rated speed (P_o) is 1.59×10^{-6}/yr for a disc inspection interval of 100,000 operating hours. Therefore, $P_1 = P_r + P_o = 1.88 \times 10^{-6}$/yr which can be compared to the NRC limit of 1.0×10^{-5}/yr (i.e., 11.42×10^{-5}/yr for disc inspection interval of 100,000 operating hours) to demonstrate the probability of an external missile is well below the NRC limit and the Unit can be operated for 100,000 hrs between disc inspections provided no cracking is detected.

As pointed out in Reference 9, the potential for failures of turbines due to brittle fracture and environmentally induced stress corrosion cracking (potential causes of missile generation within design overspeed) has been minimized through current day metallurgical and fabrication processes and turbine operating procedures.

The procedures applied to St. Lucie Unit 2 turbine reflect current metallurgical and inspection practice and are described as follows:

Low Pressure Turbine Rotor

The low pressure turbine rotor body and disc are heat treated nickel-chromium-molybdenum-vanadium alloy steel procured to specifications that define the manufacturing method, heat treating process, and the test and inspection methods. Specific tests and test documentation, in addition to dimensional requirements, are specified for the forging manufacturer.

a) Low Pressure Turbine Rotor Body

   Inspection and tests conducted at the forging manufacturer's plant:

   1) A ladle analysis of each heat of steel for chemical composition is to be within the limits defined by the specification.

   2) Following preliminary machining and heat treatment for mechanical properties but prior to stress relief, all rotor diameters are subjected to an ultrasonic examination by a Siemens Energy Inc. specification which exceeds the requirements of ASTM A-418.

   3) After all heat treatment has been completed, the rotor forging is subjected to a thermal stability test defined by a Siemens Energy Inc. specification which is more restrictive than the requirements of ASTM A-472.

   4) Utilizing specimens removed from the rotor forging at specified locations, tensile, Charpy V Notch impact and FATT properties are determined following the test methods defined by ASTM A-370.

   After the rotor body is finished machining at Siemens Energy Inc., the rotor surface is given a fluorescent magnetic particle examination as defined by a Siemens Energy Inc. specification which is similar to ASTM E-138.

b) Low Pressure Turbine Rotor Discs

   Inspection and tests conducted at the forging manufacturer's plant:

   1) The ladle analysis of each heat of steel is to be within the composition limits defined by the specification.

   2) After all heat treatment, rough machining and stress relief operations, the hub and rim areas for the completed disc forging are subjected to ultrasonic examinations. These ultrasonic tests are defined by a Siemens Energy Inc. specification which exceeds the requirements of ASTM A-418.

3) The tensile, Charpy V Notch impact and FATT properties are determined from specimens removed from the discs at specific locations. The test method used for determining these mechanical properties are defined by ASTM A-370.

c) Design features that provide a higher operational reliability and incorporate measures against stress corrosion cracking are as follows:

1) During the manufacturing process, residual compressive stresses are induced to reduce the effective stresses in the hub area and to prevent stress corrosion cracking. Residual compressive stresses are induced by heat treatment of the forgings (spindle shaft/wheel discs) according to a Siemens Technical Purchasing Specification.

2) Additional residual stresses are induced through shot peening of the bore hole and side surfaces of all discs.

3) Shot peening at the bottom of fir-tree grooves/caulking grooves for L-2R and L-1R.

4) All shaft inner radii greater than or equal to 10mm are rolled. Rolling eliminates the surface tensile stresses due to machining and provides a residual surface compressive stress resulting in a higher cycle fatigue strength and greater resistance to stress corrosion cracking.

5) Circumferential stress-relieving groove at anti-rotation protection holes of disc 1 and rolling of the pin holes.

6) Taper expansion (3 degrees) of disc bore hole at entrance side for lead-off of condensate.

After the discs are finished machined at Siemens Energy Inc., the disc surfaces except blade grooves are given a fluorescent magnetic particle examination as defined by a Siemens Energy Inc. specification which is similar to ASTM E-138.

c) Low Pressure Turbine Rotor Assembly

After the preheated discs are assembled to the rotor body to obtain the specified interference fit, holes are drilled and reamed for axial locking pins at the rotor and disc interface for Disc 1 only. Disc 2 and 3 do not use axial locking pins and rely on the shrink fit only. These holes are given a fluorescent penetrant inspection defined by a Siemens Energy Inc. specification which is similar to ASTM E-165.

d) Prior to shipping, each fully bladed rotor is balanced and tested to 120 percent of rated speed in a shop heater box.

High Pressure Turbine Rotor

The high pressure turbine rotor for low temperature light water reactor applications has the same basic material composition as the low pressure rotors. This nickel-chromium-molybdenum-vanadium alloy steel forging is procured, processed, and subjected to test and inspection requirements the same as the low pressure rotor which includes:

a) Ladle Analysis

b) Ultrasonic tests

c) Magnetic particle inspection

d) Thermal stability test

e) Deleted

f) Tensile and impact mechanical properties

g) Fluorescent magnetic particle inspection

h) Heater box and 120 percent speed test

With respect to destructive overspeed missiles, Reference 9 identifies build-up of foreign materials on valve stems or within piping systems as the most probable cause of turbine overspeed. Reference 9 also observes that the incidence of overspeed occurrences has decreased markedly since 1961 and can be attributed mainly to improvements in valve and overspeed testing procedures.

The testing procedures to be applied to the St. Lucie Unit 2 turbine represents current testing practice.

In addition, the potential for build-up of foreign material on valve stems is virtually eliminated for St. Lucie Unit 2 since the secondary system water chemistry control utilizes volatile chemistry methods and does not employ phosphates, which provide a source of valve material build-up.

Operating chemistry limits for feedwater and secondary steam generator water are given in Subsection 10.3.5.

3.5.1.4 Missile Generated by Natural Phenomena

The postulated missiles generated by natural phenomena are the tornado missiles. The plant is designed for tornado missiles as described in the design bases (opening paragraph) of Section 3.5.

The design bases tornado missiles are listed in Table 3.5-10. Onsite unprotected storage of materials that could become tornado generated missiles are minimized.

Plant structures, system and components required for safe shutdown are protected from the effects of a tornado missile by any of the following means:

a) Design of structures, systems or components to withstand missile impact

b) Protection of systems or components by structures designed to withstand missile impact

c) Separation of redundant components to preclude simultaneous failure by single missile impact

Table 3.2-1 lists the missile protection criteria applied to safety related structures, systems and components. The a, b or c designation refers to items a, b and c above.

Table 3.5-3 provides a listing of components required for safe shutdown and their corresponding protection from missiles. All components required for safe shutdown are located within or protected by structures designed to withstand tornado missile impact, such as the Shield Building, Reactor Auxiliary Building and Diesel Generator Building. In addition, the Fuel Handling Building and the main steam and feedwater trestles are designed to withstand a tornado missile. All reinforced concrete walls and roofs for tornado missile protection are at least two feet thick. Table 3.5-11 summarizes the wall and roof thickness and concrete strength, including the concrete age specified.

Shielding is provided for protection of underground outdoor equipment and piping required for safe shutdown and are separated and designed to preclude damage resulting from tornado missiles. Underground piping has a sufficient separation and a minimum of 6 ft earth or equivalent concrete cover. Under ground cabling is provided with a minimum of two ft of soil cover with a nine inch reinforced concrete protective slab or a minimum of one ft of soil cover with a 15 inch reinforced concrete protective slab. In addition to the buildings listed in Table 3.5-11, the following protection (equivalent to two ft reinforced concrete) is provided:

a) Intake Cooling Water Valve pit area is enclosed in a protective structure.

b) Diesel Generator Building openings are missile protected.

c) The auxiliary feedwater pumps and associated piping, valves and instrumentation required for safe shutdown are housed within a protective structure.

d) The main steam and feedwater lines up to and including the main steam isolation valve are protected.

e) Seismic Category I electric manholes are provided with protective covers.

A Reactor Coolant System make-up water source, with sufficient capacity to maintain coolant system inventory during cooldown to conditions permitting initiation of the Shutdown Cooling System, is located in a protective structure (Reactor Building).

3.5.1.5 Missiles Generated by Events Near the Site

Missiles generated by events near the site are discussed in Subsection 2.2.3.

3.5.1.6 Aircraft Hazards

Aircraft impact is not considered on safety related structures since there are no federal airways or airport approaches passing within two miles of the facility, no airports are located within five miles of the site and no airports have projected operations greater than 500 d^2 movements per year located within 10 miles of the site and greater than 1000 d^2 outside 10 miles, where d is the distance in miles from the site. Section 2.2 contains a description of airports.

3.5.2 STRUCTURES, SYSTEMS AND COMPONENTS TO BE PROTECTED FROM EXTERNALLY GENERATED MISSILES

Plant structures, system and components whose failure could lead to offsite radiological consequences or that are required to shut down the reactor and maintain it in a safe condition assuming a single failure are adequately protected against very low probability missile strikes by barriers when existing structures cannot be used to provide missile protection. These barriers are designed to contain or to deflect the missiles from the safety related component.

Component protection against externally generated missiles (identified in Subsection 3.5.1.4) is provided in Tables 3.5-3 and 3.5-11.

3.5.3 BARRIER DESIGN PROCEDURES

Barriers are designed to withstand both local damage in the impacted area and overall response of missile impact.

3.5.3.1 Local Damage Prediction

Local impact effects include penetration, perforation, scabbing, spalling, and punching shear. The local damage predictions are determined from empirical formulas based upon experimental results.

3.5.3.1.1 Concrete Barriers

Concrete barriers are designed to prevent missile perforation of the barrier. For local damage prediction the following formulae known as the modified Petry formulae[19] were used:

a) Where slab thickness is greater than three times the penetration depth:

$$D = KA_pV'$$

where:

D = penetration of missile, ft

K = material constant

A_p = W/A_c = sectional pressure, lb/ft^2

W = weight of missile, lb

A_c = missile contact area, ft^2

V' = velocity factor = $\log_{10}\left(1 + \frac{V^2}{215,000}\right)$

V = missile impact velocity, ft/sec

b) Where slab thickness is less than three times the penetration depth but greater than two times the penetration depth.

$$D' = D\,(1 + e^{-4(a-2)})$$

where:

D'	=	revised missile penetration, ft
D	=	penetration of missile from equation (a) above, ft
A	=	T/D
T	=	slab thickness, ft.

See Table 3.5-13 for the required wall thicknesses (based on the NDRC formula) compared with the actual wall and roof thicknesses for all Category I structures. Table 3.5-13 summarizes design values obtained by use of the NDRC Formula for missile penetration (x), thickness required to prevent scabbing (s) and the maximum thickness of concrete which a missile will completely penetrate. Please note that the NDRC Formula is only applicable to hard missiles and not the soft missiles; i.e., the automobile and wood missiles.

In order to prevent the generation of secondary missiles by spalling, the minimum thickness of the concrete barrier provided is 2 ft., which is greater than twice the calculated penetration depths. In addition, missile penetration depths were calculated utilizing the Modified Petry formula, to ensure penetration was less than half the wall thickness.

The deepest penetration for a steel missile (1" dia x 3' steel rod) was calculated to be 3.13 in. and the deepest penetration for a wood missile (2" x 4" plank 10' long) was 5.10 inches. Some wood missiles splinter into pieces without causing any local damage for concrete barrier thicknesses of 12 inches or more (see following paragraph for results of conducted tests), the steel rod should penetrate the deepest of all the missiles. The ratio of 2 ft. wall thickness to maximum depth of penetration would be 7.67.

The two foot minimum slab thickness also meets the minimum thickness required by the National Defense Research Committee (NDRC) modified formula for penetration and back of slab scabbing.

The tests conducted by Sandia Laboratories for the Electric Power Research Institute(20) and by Calspan Corporation(11), have indicated that the wood missiles postulated in Subsection 3.5.1.4 splinter into pieces without causing any local damage for concrete barrier thickness of 12 inches or more. Since the postulated steel pipe tornado-generated missiles require greater missile barrier thickness than that necessary for the wood missiles, no investigation of local damage due to wood missiles is required.

Since prevention of perforation satisfies the punching shear requirement, punching shear for non-deformable missiles such as solid steel missiles thus need not be considered as the barrier thickness provided prevents perforation. Moreover, the barrier thickness provided to prevent spalling is in agreement with the results of actual tests.

3.5.3.1.2 Steel Barriers

Steel barriers are analyzed for penetration resistance using the Stanford Research Formula(12) within the range of its applicability, or the Ballistics Research Laboratory Formula(13) where the Stanford Formula is not valid.

a) Stanford Research Formula

$$\frac{E}{D} = \frac{S}{46,500}\left(16,000\,t^2 + 1500\frac{W}{Ws}t\right)$$

where:

E = critical kinetic energy required for penetration (ft-lb)

D = missile diameter (in)

S = ultimate tensile strength of the target steel plate (psi)

t = steel thickness to be just penetrated (in)

W = length of a square side between rigid supports (in)

W_s = length of a standard width (4 in)

The formula may be rewritten as:

$$t = \sqrt{0.045\frac{Wm\,Vm^2}{DS} + 0.0022\left(\frac{W}{Ws}\right)^2} - 0.047\left(\frac{W}{Ws}\right)$$

where:

Wm = missile weight (lb)

Vm = missile velocity (fps)

The formula is valid within the following ranges:

0.1	< t/D	< 0.8
0.002	< t/l	<0.05
10	< L/D	<50
5	< W/D	< 8
8	< w/t	< 100
0.2	< W/L	< 1.0
70	< Vm	< 400

where:

L = missile length (in)

For use of the Stanford formula, rectangular cross-section missiles are converted to equivalent circular cross-section missiles of the same perimeter-to-area ratio.

b) Ballistic Research Laboratory Formula

$$t^{3/2} = \frac{0.5\,MV_m^{\;2}}{17,400\,K^2\,D^{3/2}}$$

where:

M = missile mass ($lb\text{-}sec^2/ft$)

K = constant depending on the grade of steel, usually about 1

t, V_m and D are as defined above

For use of the Ballistics Research Laboratory Formula, rectangular cross-section missiles are converted to equivalent circular cross-section missiles of the same area.

The formulae described above are developed empirically from physical tests where the missile and target are essentially non-deformable. They are thus expected to overpredict local damage for deformable missiles and/or targets.

Most steel protective structures are furnished with barrier plate thicknesses 1.25 times that required to prevent penetration.

3.5.3.1.3 Special Doors

3.5.3.1.3.1 Reactor Building Maintenance Hatch Shielding Door and Fuel Handling Building Cask Area Shielding Door

Both these doors are constructed of steel-lined reinforced concrete. The formulae used to calculate penetration do not treat composite barriers, so the steel liner is conservatively neglected in the calculation.

Available formulae for penetration are not applicable to deformable missiles striking hard, nondeformable targets. The results of tests performed by Sandia Laboratories[20] show no visible damage to the target concrete wall, front or back, when struck by a 4 x 12 plank, utility pole or 3000 lb automobile.

For the cases of nondeformable missiles, the Modified National Defense Research Committee (NDRC) Formula[14] is used to calculate penetration. The value determined by the formula for the case of the 12 in. diameter Sch 40 pipe was compared with the results of a physical test performed by Sandia Laboratories. The actual observed penetration is 5 in versus 5.29 in calculated. Use of the Modified NDRC formula is thus substantiated by the physical test.

The actual concrete thicknesses furnished, 3'-0 for the Reactor Building maintenance hatch shielding door and 2'-0 for the Fuel Handling Building cask area shielding door, are more than twice the maximum calculated penetration depth.

3.5.3.1.3.2 Tornado Resistant Doors for the Diesel Generator Building, Diesel Oil Storage Tank Enclosure and Component Cooling Water Area

These doors are constructed of solid steel, a minimum thickness of 2 1/2 inches for the personnel doors, and 3 1/2 inches for the equipment doors. The thicknesses furnished are more than twice the maximum penetration depth, calculated in accordance with the Ballistics Research Laboratory formula.

3.5.3.2 Overall Damage Prediction - Concrete and Steel Barriers - General Criteria

The overall structural capacity of both concrete and steel barriers is determined to preclude structural collapse of the barrier under missile impactive load.

The dynamic effects of impactive loads are considered by using impulse, momentum and energy balance techniques as presented in the Williamson and Alvy paper[(15)]. The maximum resistance of the barrier is evaluated assuming elasto-plastic response.

The barrier strain energy capacity is limited by the allowable ductility factors listed in Table 3.5-12. A simplified method based on idealization of the actual structure to an equivalent single- degree-of-freedom system, and of the impactive load time history to a simple mathematical form is used in the analysis of seismic Category I structures.

In defining the equivalent single-degree-of-freedom system, References 16 and 17 are used to determine the load-mass factors and the parameters relating the maximum resistance, spring constant, and dynamic reactions of the system.

3.5.3.2.1 Concrete Barriers - Additional Criteria

The ultimate load capacity of concrete barrier members with short spans is based on the yield line theory of reinforced slabs. The collapse mechanism is a circular yield line pattern based on the impact of a concentrated load. Adequate resistance is provided to ensure against an edge type or failure.

In the analysis of the concrete barriers the Williamson and Alvy method is used for both types of impact with and without penetration.

An alternate method is also used in the impactive analysis for soft missiles (characterized by significant local deformation of the missile during impact.) In this method, the peak of the impactive force, (F_{cr}) is equal to the product of the net cross sectional area of the missile and the crushing strength of the missile material. The forcing function is assumed to be rectangular with a duration, t_d:

$$t_d = \frac{mV}{F_{cr}}$$

where:

m = mass of missole, lb-sec^2/ft

V = velocity of missile, ft/sec

3.5.3.2.2 Steel Barriers - Additional Criteria

Steel protective barriers are analyzed using the Williamson and Alvy procedure[(15)] for the case with no penetration.

3.5.3.2.3 Special Doors

3.5.3.2.3.1 Reactor Building, Maintenance Hatch Shielding Door and Fuel Handling Building Cask Area Shielding Door

Equivalent static loads are determined using the method presented in Reference 18, which assumes constant deceleration over the distance of penetration for hard, nondeformable missiles. The method is adapted for use with deformable missiles by using crushing length in lieu of penetration depth in the equations as the distance over which deceleration occurs. Crushing lengths are determined from physical tests performed by Sandia Laboratories[(20)]. This procedure does not consider the dynamic response of the target structure in the calculation of static load, i.e, the structure is treated as rigid. This is expected to result in overprediction of the impact forces.

In the case of the 12 inch diameter Schedule 40 pipe and the utility pole, the calculated impact forces are compared with the actual impact forces measured with load cells in tests by Sandia Laboratories[(20)]. In each case, the calculated force is within five percent of the measured force; demonstrating the validity of the calculational methods used.

3.5.3.2.3.2 Tornado Resistant Doors for the Diesel Generator Building, Diesel Oil Storage Tank Enclosure and Component Cooling Water Area

The maximum impactive load is assumed to occur at the door frame, where there is little opportunity for structural deformation to absorb a portion of the kinetic energy. The equivalent static load is calculated as the crushing or buckling load of the missile. This approach produces results more conservative than that described in Subsection 3.5.3.2.3.1. The door itself is analyzed using the Williamson and Alvy procedure [(15)] for the case with no penetration.

SECTION 3.5: REFERENCES

1. Nuclear Regulatory Commission, Regulatory Guide 1.115, Rev. 1, Protection Against Low Trajectory Turbine Missiles," July 1977.

2. "Standard Review Plan, Section 3.5.1.3- Turbine Missiles," NUREG-75/087, Revision 1-Nuclear Regulatory Commission.

3. NUREG-1048, Appendix U, Supplement No. 6, "Safety Evaluation Report related to the Operation of Hope Creek Generating Station."

4. TP-04124-NP-A, "Missile Probability Analysis for the Siemens 13.9 m^2 Retrofit Design of Low-Pressure Turbine by Siemens AG," June 7, 2004 (Includes NRC Final Safety Evaluation letter dated March 30, 2004).

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7. PSL-ENG-SENS-08-077, Rev. 0, "Engineering Evaluation to Increase Turbine Valve Test Interval to 6 Months."

8. CT-27455, Rev. 1, "Missile Report for FPL St. Lucie Units 1&2, BB281-13.9m^2," August 31, 2009, Siemens Energy, Inc.

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13. R.C. Cwaltney, "Missile Generation and Protection in Light-Water Cooled Power Reactor Plants," ORNL NSIC-22, Oak Ridge National Laboratory (Sept 1968).

14. NDRC, "Effects on Impact and Explosion," Summary of Technical Report of Division 2, National Defense Research Committee, Vol. 1, Washington, D.C. (1946) .

15. R.A. Williamson and R.R. Alvy, "Impact Effects of Fragments Striking Structural Elements," Holmes and Narver, Inc. (Nov, 1973).

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17. Norris, Ch, Hansen, R.J., et al. "Structural Design for Dynamic Loads," McGraw-Hill Book Company, New York, 1969.

18. R.P. Kennedy, "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," Nuclear Engineering and Design 37, 183-203 (1976).

19. Amirikian, A, "Design of Protective Structures," Report NP-3721 Bureau of Yards and Docks, Department of the Navy, August, 1950.

20. "Full-Scale Tornado - Missile Impact Tests," EPRINP-148 Project 399, by Sandia Laboratories, Tonapah Test Range, Tonapah, Nevada for Electric Power Research Institute, Interim Report, April, 1976.

TABLE 3.5-1

MISSILES OUTSIDE CONTAINMENT* GENERATED BY HIGH ENERGY SYSTEMS

	Systems	Postulated Missiles	Line Number	Elevation	Remarks**
1.	CVCS charging)	TE-02-1	I-2-CH-109	+36'-3	Orientated towards RAB interior full height concrete wall and Shield Bldg wall - no equipment impact
2.	CVCS (letdown)	TE-2223	2-CH-304	+28'-0	Orientated toward letdown HX cubicle west wall - no equipment impact
3.	CVCS (letdown)	TE-2224	2-CH-304	+28'-0	Orientated toward letdown HX north wall - no equipment impact
4.	SIS (LPSI pump 2A discharge)	TE-3351Y	I-10-SI-553	+5'-10	Located RAB pipe tunnel - no equipment impact
5.	SIS (SDCRX 2A outlet)	TE-3303W	I-12-SI-406	+2'-6	Located in SDCHX 2A cubicle - oriented straight up - no equipment impact
6.	SIS (SDCRX 2A outlet)	TE-3303X	I-12-SI-406	+2’-6	Located in SDCHX 2A cubicle - orientated due south - no equipment impact
7.	SIS (LPSI pump 2A discharge)	TE-3351X	I-10-SI-475	-5'-0	Orientated straight up - no equipment impact
8.	SIS (LPSI pump 2B discharge)	TE-3352X	I-10-SI-163	-5'-0	Orientated straight up - no equipment impact
9.	SIS (LPSI pump 2B discharge)	TE-3352Y	I-10-SI-556	0'-9	Orientated straight up - no equipment impact
10.	SIS (SDCHX 2B outlet)	TE-3303Z	I-12-SI-407	+2'-6	Located in SDCHX 2B cubicle - orientated due south - no equipment impact
11.	SIS SDCHX 2B outlet)	TE-3303Y	I-12-SI-407	+2'-6	Located in SDCHX 2B cubicle - orientated straight up - no equipment impact
12.	SGBD (SG blowdown line)	TE-23-1	I-3-B-61	+37'-0	Orientated soutwest towards pipe tunnel - no equipment impact
13.	SGBD SG blowdown line)	TE-23-2	I-3-B-62	+34'-0	Orientated directly west towards RAB full height interior concrete wall - no equipment impact

* High energy systems containing valves that may be potential missiles (stems) located in the Turbine Building have those valves either oriented away from safety related equipment or are provided with a barrier

** Walls or ceilings are assigned to contain missiles within cubicle or area

TABLE 3.5-1 (Cont'd)

	Systems	Postulated Missiles	Line Number	Elevation	Remarks*
14.	Sampling System	V5155, VPI5510	3/8-SS-646	+19.51"	These valves are all located in the sample room of the RAB. No equipment impact.
		V5156, VPI5560	3/8-SS-647		
		V5157	3/8-SS-617		
		V5162	3/8-SS-636		
		V5163	3/8-SS-633		
15.	Nitrogen Supply	VPS 6662	1/2-WM-304	-	Nitrogen header as it enters the RAB. No equipment impact.

TABLE 3.5-2

MISSILES OUTSIDE CONTAINMENT

FROM FAILURE OF OVERSPEED PROTECTION

COMPONENT	SAFETY CLASSIFICATION	OVERSPEED PROTECTION	REMARKS
Auxiliary FW Pump Turbine	3	Electrical trip - set at 115% overspeed Mechanical trip - set at 125% overspeed	
Diesel Generators	3	Mechanical trip - set at 115% overspeed	
Main Turbine	Non-safety	Redundant electronic trip - set at 111% overspeed Electrical trip - set at 111.5% overspeed	Analysis of T-G failure is described in Subsection 3.5.1.3. Overspeed protection is described in Subsection 10.2.2.

TABLE 3.5-3

ENCLOSURES FOR EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

Security Related Information
Table Withheld Under 10 CFR 2.390

TABLE 3.5-3 (cont'd)

Security Related Information
Table Withheld Under 10 CFR 2.390

TABLE 3.5-3 (cont'd)

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TABLE 3.5-3 (cont'd)

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TABLE 3.5-3 (cont'd)

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TABLE 3.5-3 (cont'd)

Security Related Information
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TABLE 3.5-3 (cont'd)

Security Related Information
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TABLE 3.5-4

INTERNAL MISSILE PARAMETERS

Item		Kinetic Energy Ft-Lb	Weight (Lb)	Leading Section	Structure/Shield/Barrier
a)	Reactor Vessel				
	1) Closure Head Nut	2,022	116	Annular Ring, OD=10-9/16", ID=6.8"	Missile Shield on Reactor Vessel
	2) Closure Head Nut & Stud	4,932	710	Solid Circle 7" in Diameter	Missile Shield on Reactor Vessel
	3) Incore Detector Instrumentation Assembly	88,556	130	Solid Disk 6-1/2" Diameter and 3" Thick	Missile Shield on Reactor Vessel
b)	Steam Generator				
	1) Primary Manway Stud and Nut	71	4-1/4	Solid Circle 1-1/2" Diameter	Low Energy
	2) Secondary Handhole Stud and Nut	8	1-3/4	Solid Circle 1" Diameter	Low Energy
	3) Secondary Manway Stud	38	4.6	Solid Circle 1-1/4" Diameter	Low Energy
c)	Pressurizer				
	1) Manway Cover Stud and Nut	71	4-1/4	Solid Circle 1-1/2" Diameter	Pressurizer Enclosure
	2) Lower Temperature Element	290	3	Solid Disk 2-3/4" Diameter and 1/2" Thick	Pressurizer Enclosure
d)	Control Element Drive Mechanism	57,600	1141 (wet)	Solid Circle 10" Diameter	Missile Shield on Reactor Vessel
e)	Main Coolant Piping Temperature Nozzle with RTD	1,125	11.1	Solid Disk 2-3/4" Diameter and 1/2" Thick	Secondary Shield Wall

TABLE 3.5-4(cont'd)

	Item	Kinetic Energy (Ft-Lb)	Weight (Lb)	Leading Section	Structure/Shield/Barrier
f)	Surge and Spray Piping Wells with RTD Assembly	277	1-3/4	Solid Disk 2-3/4" Diameter and 1/2" Thick	Secondary Shield Wall
g)	Main Coolant Pump Thermal Well with RTD	1,125	11.1	Solid Disk 2-3/4" Diameter and 1/2" Thick	Secondary Shield Wall
h)	Reactor Coolant Pump Mounting Flange Leakoff Connections: VPI 1150, 1160, 1170, 1180 VPS 1150, 1160, 1170, 1180	Normal service with no gasket leakage will expose these valves to essentially zero pressure.			Valves are oriented to preclude their missile potentiality from being detrimental to the plant.

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TABLE 3.5-10

DESIGN BASE SPECTRUM OF TORNADO MISSILES

	Missile	Density (lb/ft^3)	Impact Area (ft^2)	Impact Velocity (fps)	Weight (lb)	Impact Height
A.	4" x 12" plank, 12 ft long	50	.333	322	200	Grade to top of structure
B.	1" dia x 3' steel rod	490	.00545	163	8	Grade to top of structure
C.	6" dia Sch 40x 15' pipe	490	.196	116	284.5	Grade to top of structure
D.	12" dia Sch 40 x 15' pipe	490	.785	116	743.4	Grade to top of structure
E.	13.5" dia x 35 ft long wooden utility pole	43	.995	153	1497	Grade to max elevation of 25 ft above grade
F.	Automobile	--	20	84	4000	Grade to max elevation of 25 ft above grade
G.	2" x 4" plank, 10 ft long	50	0.0556	403	27.8	Grade to top of structure

TABLE 3.5-11

TORNADO MISSILE CONCRETE BARRIER MINIMUM THICKNESS*

Building	Minimum Thickness (ft)
Reactor Building	
Cylindrical Wall	3
Dome	2 1/2
Reactor Auxiliary Building	
Wall	2
Roof Slab	2
Fuel Handling Building	
Wall	2
Roof Slab	2
Component Cooling Water Area Structure	
Wall	2
Roof Slab	2
Diesel Generator Building	
Wall	2
Roof Slab	2
Intake Cooling Water Structure	
Wall	2
Roof Slab	1 3/4" steel
Condensate Storage Tanks Structure	
Wall	2
Dome	2
Diesel Oil Storage Tanks Structure	
Wall	2
Roof Slab	2

*The required 28 day design strength for the above structures is a minimum 4000 psi (refer to Subsection 3.8.3.6.1).

TABLE 3.5-12

ALLOWABLE DUCTILITY FACTORS (μ)

			μ
I.	Reinforced Concrete		
	a -	Flexure (beams)	$\frac{0.05}{p - p'} \leq 10$
	b -	Flexure (slabs)	$\frac{0.05}{p - p'} \leq 10$
	c -	Compression (walls & columns)	1.3
	d -	Shear (beams & slabs)	
		shear carried by concrete only	1.0
		shear carried by concrete/stirrups	1.3
		shear carried completely by stirrups	3.0
II.	Structural Steel		
	a -	Flexure (beams)	10
	b -	Shear (beams)	1
	c -	Axial compression (columns)	1

$$p = \frac{As}{bd}$$

$$p' = \frac{As'}{bd}$$

where:

As = area of tension reinforcement

As' = area of compression reinforcement

b = width of sections

d = depth of section to centerline of reinforcement

TABLE 3.5-13

TORNADO MISSILE IMPACTIVE ANALYSIS

Penetration of Concrete for Design Base Spectrum of Tornado Missiles
Using Modified National Defense Research Committee Formula (NDRC)

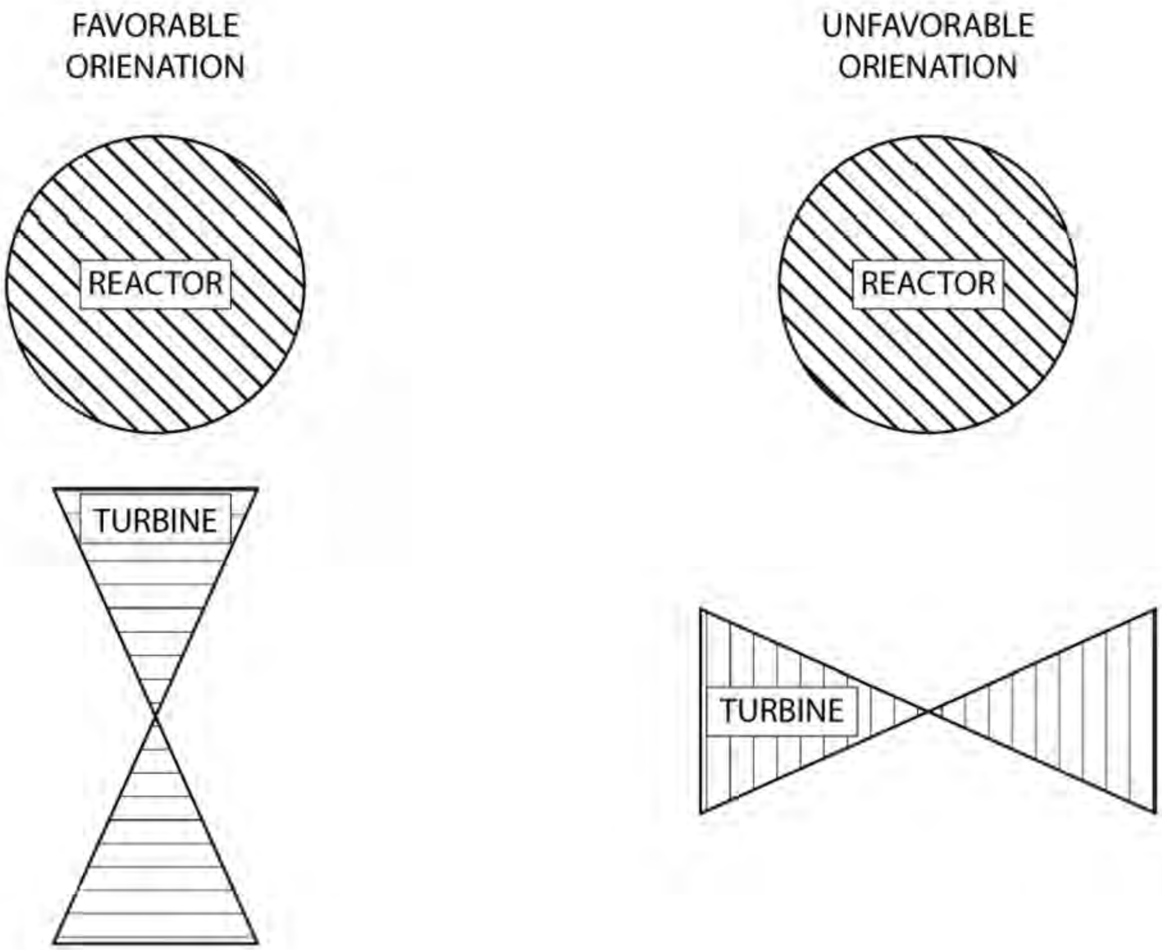
Missile No.	Missile	W (b)	A ($in.^2$)	D (in.)	V_o (ft/sec)	x (in.)	x/d	e/d	s/d	e (in.)	e (in.)	Required Concrete Thickness 1.25 S (in.)	Actual Minimum Concrete Thickness for	Remarks
1	4" x 12" Plank 12' Long	200	48.0	-	322	-	-	-	-	-	-	-	Wall or Roof Slab of All	Soft** Missile
2	1" dia x 3' Steel Rod	8.0	0.785	1.0	163	1.58	1.58	3.28	4.27	3.28	4.27	5.34	Class I Structures (in.)	
3	6" dia Sch 40 x 15' Pipe	284.5	5.58	2.67	116	4.69	1.76	3.50	4.51	9.35	12.05	15.06		
4	12" dia Sch 40 x 15' Pipe	743.4	14.6	4.31	116	6.27	1.45	3.12	4.09	13.44	17.63	22.04	≥24"	Critical Case
5	13.5" dia x 35' long Wooden Utility Pole	1,497	143.0	-	153	-	-	-	-	-	-	-		Soft** Missile
6	Automobile	4,000	2,880	-	84	-	-	-	-	-	-	-		Soft* Missile
7	2" x 4" Plank 10' Long	27.8	8.0	-	403	-	-	-	-	-	-	-		Soft** Missile

22.04"<24" minimum wall or roof slab concrete thickness

* Missile does not penetrate based on ASCE Reference Page 6-28

** NDRC Formula is not applicable based on ASCE Reference Page 6-41

References - Design Based Spectrum of Tornado Missiles Table 3.5-10 SL2-UFSAR
Tornado Missile Concete Barrier Minimum Thickness Table 3.5-11 SL2-UFSAR
ASCE "Manual of Standard Practices for Design of Nuclear Power Plant Facilities"
Chapter 6



FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

TURBINE ORIENTATION RELATIVE TO REACTOR BLDG
FIGURE 3.5-1

Security Related Information
Table Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

TURBINE GENERATOR PLACEMENT
AND ORIENTATION
FIGURE 3.5-2

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE RUPTURE OF PIPING

This section describes the design bases and protective design features that are used in St. Lucie Unit 2 to demonstrate that the essential systems and components, and essential structures are adequately protected against the effects of postulated piping failures. Piping systems are described as high energy (i.e., fluid systems which exceed 200°F and/or 275 psig during normal operating conditions) and moderate energy (i.e., fluid systems which are 200°F or less, and 275 psig or less during normal operating conditions). Postulated piping failures are also divided into those associated with piping located inside the containment and those outside the containment. The failures inside containment are further divided into those which are (or cause) design basis accidents (DBAs) and those which are not. This section describes high energy piping and moderate energy piping located inside and outside the containment, applicable criteria for postulation of pipe break locations, the dynamic effects of pipe rupture; detailed analyses of plant transients are presented in Chapters 6 and 15. Those failures which are not DBAs include all failures outside containment where the plant is maintained in a safe condition and postulated piping failures inside containment associated with systems other than RC Loop, Main Steam and Feedwater. High energy piping failure includes the effects of jet impingement, reactive forces and pipe whip, compartment pressure and environmental conditions. Moderate energy piping failures include the effects of wetting and flooding.

This section describes pipe whip and jet impingement aspects of pipe rupture. The other aspects of pipe rupture such as compartment pressure effects, containment pressure/temperature design impact and environmental parameters identification for equipment qualification are addressed in Subsection 6.2.2.

The following criteria documents are considered for pipe break analysis:

a) Inside Containment:

   Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," May 1973 (R0).

b) Outside Containment:

   The Giambusso Criteria; attachment to the December 1972 letter sent by A. Giambusso Deputy Director for Reactor Projects, Directorate of Licensing to applicants and licensees, entitled "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment" was used in addition to APCSB 3-1. The applicability of this criteria for St. Lucie Unit 2 was accepted during the NRC staff review as delineated in the Safety Evaluation Report (October 1981).

In addition to these original criteria, a moderate energy piping analysis is performed based on criteria given in Subsections 3.6.1.3 and 3.6.2.4.

Subsequent to the requirements of a) and b) above, CEN-367-A, Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems, provides the technical justification to eliminate RCS hot and cold leg piping from the pipe rupture analysis. Further evaluation demonstrates that sufficient LBB margin on crack stability is maintained on the St. Lucie Unit 2 main coolant loop hot and cold leg pipes for the effects on NOP loads from the EPU. This evaluation demonstrates that the primary loop piping meets all of the criteria for application of leak before break presented in NUREG-1061, Volume 3. As a result, the mechanical/structural loads associated with dynamic effects of guillotine and slot breaks in RCS hot and cold leg piping are no longer considered a plant design basis (References 14 and 15). The leak before break methodology concludes that:

a. Small cracks which may go undetected during inspections do not grow significantly during service.

b. Cracks which are assumed to grow through the pipe wall would leak significantly while remaining stable. The amount of leakage is detectable with a safety margin of at least a factor of 10.

c. Cracks of the length that leak at the rate in (b) can withstand normal operation and safe shutdown earthquake loads with a safety factor of at least $(2)^{1/2}$.

d. Cracks twice as long as those addressed in (c) will remain stable when subjected to normal operation and safe shutdown earthquake loads.

Although the dynamic effects associated with a hot or cold leg break have been eliminated from the plant design bases, some of the original design features installed to mitigate the consequences of such a break remain in place and functional (see Section 6.2.1.3.3 for changes to the reactor cavity pressure relief function).

As a result of the installation of the Replacement Steam Generators during the CRP outage, the shim plate attached to the SG sliding base support has been permanently removed by PC/M 05133 (Reference 21), thereby deleting the North-South direction LOCA restraint for the SG sliding base support. The environmental qualification design basis for safety related equipment inside containment remains unchanged (Reference 16).

Also, the lower RCP missile protection system, which includes the RCP restraint cables, was designed to constrain the RCP casing from causing further damage to safety-related equipment in the event of a main loop pipe break (MLPB). With the implementation of leak before break (LBB), MLPBs were eliminated from the design basis faulted loads. An analysis of the RCP and surrounding components to demonstrate the RCP HELB cable restraints for pumps 2A1, 2A2, 2B1, and 2B2 can be permanently removed. The analysis demonstrated the cable restraints around the pump will not be needed for missile protection in the event of such a break. St. Lucie Unit 2 may permanently remove the RCP cable restraints under current and extended power uprate (EPU) conditions. The upper and lower RCP cable restraints have been removed.

Subsequent to the requirements of a) above, Regulatory Guide 1.46 was superceded by Standard Review Plan 3.6.2 in July 1981. NRC Generic Letter 87-11 (Reference 17) revised NRC Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," as contained in the Standard Review Plan (SRP), Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping" (Reference 18) in June 1987. The revision eliminates all dynamic effects (missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, compartment, subcompartment and cavity pressurizations and decompression waves within the ruptured pipe) and all environmental effects (pressure, temperature, humidity and flooding) resulting from arbitrary intermediate pipe ruptures. This action allows the elimination of pipe whip restraints and jet impingement shields installed to mitigate the effects of arbitrary intermediate pipe ruptures provided the requirements of References 17 and 18 are met.

3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS

This subsection presents the design bases, description and safety evaluation for determining the effects of postulated piping failures in fluid systems both inside and outside containment.

3.6.1.1 Design Bases

Systems or components that are needed to shutdown the reactor and/or to mitigate the consequences of the postulated pipe break (defined as essential systems) are listed in Table 3.6-1 for various high energy piping failures inside and outside containment. Depending upon the type and location of the postulated pipe break, certain safety related equipment may not be classified as essential for that particular event.

The essential systems or portions of systems, which may be required to maintain the plant in a safe condition following high energy piping failures, are presented in Table 3.6-1.

The identification of the effect on the essential system by the postulated piping failures is given in Appendices 3.6A and 3.6B. The design approach used to protect essential systems is discussed in Subsection 3.6.1.2. Drawings on which postulated piping failure locations and pipe whip restraint locations have been indicated are provided in Appendix 3.6C.

3.6.1.2 Description

3.6.1.2.1 High Energy Systems

Piping failures are postulated in high energy fluid systems, i.e., fluid systems which exceed 200 F and/or 275 psig during normal operating conditions. The following systems, or portions of those systems, are evaluated as high-energy pipelines for pipe rupture:

a) Inside Containment:

   1) Reactor Coolant System (including pressurizer surge, spray and normally pressurized portion of Pressurizer Safety and relief lines, and piping from RCS nozzle to the first normally closed valve of the Shutdown Cooling System)

   2) Chemical and Volume Control System (letdown and charging)

   3) Safety Injection System (portions included in the RCPB and those lines pressurized by the safety injection tanks)

   4) Main Steam System

   5) Feedwater System

   6) Steam Generator Blowdown System

   Pipe rupture is not postulated in those high energy lines (sampling system, nitrogen supply to SITs) which are one inch nominal pipe size or less.

b) Outside Containment

1) Main Steam System

2) Feedwater System

3) Auxiliary Feedwater System

4) Steam Generator Blowdown System

5) Chemical and Volume Control System (from containment penetration to pressure-reducing valves downstream of letdown heat exchanger, and from discharge of charging pump to containment penetration)

6) Auxiliary Steam System

Additional high energy piping (feedwater, extraction steam, condensate, heater drains) located in the Turbine Building is not analyzed for postulated piping failures since it is physically remote from essential systems, components and structures.

3.6.1.2.2 Moderate Energy Systems

Moderate energy systems include those (fluid) systems where both the following conditions apply during normal operation of the reactor:

a) Normal operating temperature is 200 F or less, and

b) Normal operating pressure is 275 psig or less.

A system is also considered moderate energy if the system exceeds 200 F and/or 275 psig less than two percent of the system normal operating time (not including testing). The following moderate energy systems were considered:

a) Inside Containment

1) Component Cooling Water System

2) Shutdown Cooling System

3) Containment Spray System

4) Waste Management System

5) Fuel Pool System

6) Primary Water System

b) Outside Containment

1) Component Cooling Water System

2) Fire Protection System

3) Shutdown Cooling System

4) Containment Spray System

5) Safety Injection System

6) Sampling System

7) Demineralized Water System

8) Circulating Water System

9) Steam Generator Blowdown System

10) Fuel Pool Cooling System

11) Waste Management System

12) Primary Water System

13) Diesel Oil System

14) Service Water System

15) Chemical and Volume Control System

16) Intake Cooling Water System

Moderate energy line cracks are not considered in the Turbine Building, since it is physically remote from essential systems, components and structures.

3.6.1.2.3 Methods of Protection:

Postulated piping failures are analyzed to determine their possible effect on those essential systems and components required to mitigate the consequences of each postulated pipe break event. Where necessary, protective measures, such as those described below, are incorporated into the plant design to assure the functional capability of these systems.

a) Separation

Wherever plant design separation is shown to assure the functional capability of essential systems, no additional protective measures are required.

b) Barriers, Shields and Enclosures

Where separation cannot be shown to assure the functional capability of essential system, then barriers, shields, or enclosures are provided for protection against the effects and consequences of the pipe break. These structures are evaluated to ensure their capability of accomplishing this function such that any damage to equipment caused by pipe whip, jet impingement, missiles or environmental consequences will not impair the essential systems. Structures providing barrier protection are designed to withstand the pressure, humidity and temperature transients which result from a high-energy piping system break, plus normal operating loads plus safe shutdown earthquake (SSE) loads.

c) Pipe Whip Restraints

Where adequate protection does not exist by separation, barriers, or enclosures, pipe whip restraints are provided as necessary to assure the functional capability of essential systems. Restraints are not provided where it is shown that the broken pipe does not cause unacceptable damage to essential systems. The design criteria for restraints are given in Subsection 3.6.2.7.2.

The pipe whip and jet impingement load is determined by the equation F = KPA (where K is constant, P is system operating pressure and A is break area). This load amplified by a dynamic load factor is used for design of pipe whip and jet impingement protective structures.

3.6.1.2.4 The information contained in this Subsection is Historical

The RCS pressure and break area have not changed due to Stretch Power Operation. Therefore, the existing structure for pipe whip and jet impingement protection is considered adequate without modifications for the following high energy systems inside containment:

(1) Reactor Coolant System (including pressurizer surge, spray and safety & relief systems).

(2) Shutdown Cooling System (from RCS nozzle to normally closed valve).

(3) Safety Injection System (portions included in Reactor Coolant Pressure Boundary (RCPB) and those lines pressurized by the safety injection tanks).

(4) Chemical Volume and Control Systems (letdown and charging).

For Main Steam, Feedwater and Steam Generator Blowdown Systems, the operating pressure and temperature for Stretch Power Operation were determined to be higher than those for 100 percent power operation. The pressure corresponding to Stretch Power Operation has been used in the original pipe rupture analysis of Main Steam and Feedwater Systems inside containment and the entire high energy portion of the Steam Generator Blowdown System. Thus, no revisions or modifications to pipe whip restraints in these systems are required.

To verify the adequacy of the existing pipe whip restraint design in the main steam trestle for Stretch Power Operation, a confirmatory analysis review was initiated. This review consisted of (1) calculating fluid forces on the main steam and feedwater piping due to pipe breaks (RELAP 4 computer code was used for this calculation), and (2) comparing these forces with the existing forcing functions.

For the cases where the original forces enveloped the stretch power forcing functions, it was concluded that pipe whip protective structures are adequate without any modifications. The above review identified only three (3) cases (one feedwater and two main steam) where the original forcing functions did not envelope the stretch power forcing functions. For these cases, structural analyses of the protective structures have been performed to verify the adequacy of the existing design.

Area drawings that show design layout used to protect the essential systems, structures and components are given on Figures 1.2-1 through 1.2-22.

Qualification of Class 1E equipment for the effects of a steam environment is discussed in Section 3.11. An analysis of the potential effects of missiles is discussed in Section 3.5.

3.6.1.3 Safety Evaluation

By means of the design features discussed above such as separation, barriers, and pipe whip restraints, the effects of postulated pipe break are shown not to affect essential systems to an extent that would impair their functional capability.

In conducting the high and moderate energy pipe rupture analyses both inside and outside containment, the following assumptions are used:

a) If the postulated pipe failure results in an automatic separation of the turbine generator from the power grid, then offsite power is assumed to be unavailable.

b) Operator action to mitigate the consequences of the postulated pipe failure, if required, is analyzed for each specific event. The feasibility of initiating operator actions on a timely basis, as well as the accessibility provided to allow the operator actions, is demonstrated.

c) The use of required plant systems, including non-seismic systems, in bringing the plant to a safe shutdown condition, is considered in the analysis of pipe failures.

d) An unrestrained whipping pipe is considered capable of:

    1) rupturing impacted pipes of smaller nominal pipe sizes, and

    2) developing through-wall leakage cracks in larger nominal pipe sizes with thinner wall thicknesses.

e) The energy level in a whipping pipe is considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and heavier wall thickness.

f) Credit is taken for pipe whip restraints to limit the effects of postulated piping failures.

g) The analysis assumes a single active component failure in any of the essential systems required. This single active failure is in addition to the postulated pipe failure and any direct consequences of the pipe break.

h) Where the postulated 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 of components in the other train or trains of that system only are not assumed, provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems.

i) The flow from a moderate energy leakage crack is assumed to result in an environment that wets unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments.

j) A pipe failure is assumed to occur independently of low probability natural phenomena events, e.g., SSE, design basis tornado, etc.

The effects of high energy pipe breaks are not analyzed where it is demonstrated that essential systems, components, or structures are physically remote from a break in that piping run.

High energy lines are described in Subsection 3.6.1.2.1. The results of the analyses of postulated piping failures are presented in Appendices 3.6A and 3.6B.

3.6.2 DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6.2.1 Criteria Used to Define Break Locations for Pipe Whip Analysis

3.6.2.1.1 High Energy Piping Systems

This section provides the criteria used to determine postulated piping failure locations for high energy piping systems both inside and outside containment.

a) Reactor Coolant System Main Loop Piping

Circumferential (guillotine) and longitudinal (slot) breaks were postulated for the RCS hot and cold legs in the original plant design. Since then, however, the NRC revised General Design Criteria (GDC) 4 to eliminate the consideration of dynamic effects of a loss of coolant accident from the plant design bases. The dynamic effects of a LOCA include the effects of missiles, pipe whipping, discharging fluid (i.e., jet impingement), decompression waves within the ruptured pipe and dynamic or nonstatic pressurization in cavities, compartments, and subcompartments. Reference 13 demonstrates that the primary loop piping meets all of the criteria for application of leak before break presented in NUREG-1061, Volume 3. As a result, the mechanical/structural loads associated with dynamic effects of guillotine and slot breaks in RCS hot and cold legs are no longer considered a plant design basis (References 14 and 15).

The discussion in items 2 & 3 below related to RCS piping is historical information. The original pipe whip restraints included in the design of RCS piping, however, remain in place and functional, except for the lower and upper Reactor Coolant Pump (RCP) cables restraints.

1) The design of the RCS pipe whip restraints is based on a stress survey of the St. Lucie Unit 2 Reactor Coolant System Main Loop Piping performed in accordance with the methods described in CENPD 168A[(1)]. St. Lucie

Unit 2 Reactor Coolant System geometries and transients were employed in the analysis. The results of this analysis are presented on Figure 3.6-4. In accordance with the criteria specified in Reference 1, circumferential type pipe breaks are postulated to occur at all terminal ends and pipe breaks are postulated at all intermediate locations throughout the piping system where the range of primary plus secondary stress intensity exceeds 2.4 Sm or the cumulative usage factor exceeds 0.10.

Where all intermediate pipe break locations would be considered unlikely because the stresses and cumulative usage factors calculated for a particular run of piping between terminal ends are everywhere less than the stress and fatigue limits stated above, the two intermediate locations of highest cumulative usage factor are chosen as the most likely break locations for piping runs longer than 10 diameters total length, and for piping runs having more than one change in direction throughout the run.

2) The results presented on Figure 3.6-4 confirm the break location and types of Reference 1 for the main loop pipe.

3) For the partial area guillotine type pipe breaks at the reactor inlet and outlet nozzles and the steam generator inlet nozzles, the methods of Reference 1 were employed to calculate the flow areas and opening times of the break at these locations. The resultant break characteristics are shown in Table 3.6-3.

The pipe whip restraint at the reactor inlet is shown on Figure 3.6-3. The stiffness values of the restraint are provided in Table 3.6-2 and on Figure 3.6-5, and the restraint gaps are provided in Table 3.6-4.

All other guillotine breaks have been assumed to open to full area.

The break locations for RCS are shown on Figures 3.6C-2.1 and 3.6C-2.2.

b) Piping Except RCS Main Loop Piping

A "break anywhere" technique is used to determine requirements for pipe whip restraint locations. This "break anywhere" basis is implemented by postulating full area circumferential piping failures in piping greater than one inch nominal pipe size with failures postulated to occur at any location along the pipe axis. Similarly, longitudinal failures are postulated in piping four inches nominal pipe size and greater. Longitudinal failures are postulated to occur at any location along the pipe axis and at any location about the pipe circumference. Break configurations and areas are those given in Subsection 3.6.2.3. Pipe whip restraints are provided in accordance with the criteria of Subsection 3.6.2.7.2.

NRC Generic Letter 87-11 (Reference 17) revised Branch Technical Position MEB 3-1 as contained in the Standard Review Plan (SRP), Section 3.6.2 (Reference 18). All modifications to Class 2 & 3 piping may invoke this new criteria to eliminate arbitrary intermediate pipe breaks in lieu of the original criteria provided that the requirements stipulated in References 17 and 18 are fully complied with.

The original criteria used for pipe break postulation is described in Section 3.6 items a) and b) in page 3.6-1.

3.6.2.2 Criteria Used to Define Break Locations for Jet Impingement Analysis

3.6.2.2.1 High Energy Piping Systems Inside Containment

a) RCS Main Loop Piping

Due to the application of leak before break methodology to the RCS hot and cold leg piping, the dynamic effects associated with circumferential (guillotine) and longitudinal (slot) breaks do not have to be considered (References 14 and 15). A technical evaluation was performed to demonstrate that the probability or likelihood of such breaks occurring in the primary coolant loops is sufficiently low that they need not be a design basis (see CEN-367-A).

The design features installed to mitigate jet impingement effects, however, remain in place and functional. The jet impingement analysis also remains valid. The break locations assumed in the jet impingement analysis are identical to those for the pipe whip analysis as described in Subsection 3.6.2.1.1a.

b) ASME Section III Code Class 1 Piping other than RCS main loop piping.

Piping failures are postulated to occur at the following locations based on Regulatory Guide 1.46 (R0):

1) Terminal ends.

2) Any intermediate location between terminal ends where the primary plus secondary stress intensities, S_n, derived on an elastically calculated basis under the loading associated with one-half safe shutdown earthquake and operational plant conditions exceeds 2.0 S_m for ferritic steel and 2.4 S_m for austenitic steels.

3) Any intermediate location between terminal ends where the cumulative usage factor (U) derived from the piping fatigue analysis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions exceeds 0.1.

4) Where the stresses calculated for the piping between terminal ends are less than the stress limits stated above intermediate locations are selected on the basis of locations of relative high stress and/or high usage factor, as compared to the remainder of the piping run or branch.

5) Where break locations are selected without the benefit of stress calculations, breaks are postulated at each pipe fitting (e.g., elbow, tee, cross, flange, and non-standard fitting), welded attachment and valve. Where piping contains no fittings, welded attachments, or valves, break locations at one location at each extreme of the piping run adjacent to a protective structure are selected.

6) As a minimum, there are two intermediate break locations in each piping run or branch run as chosen from 2, 3, 4, or 5 above.

c) ASME Code, Section III Code Classes 2 and 3 and Non-Nuclear Piping.

Piping failures are postulated to occur at the following locations in each piping run or branch run. These locations are based on Regulatory Guide 1.46 (R0) for Code Classes 2 and 3. Regulatory Guide 1.46 (R0) does not address non-nuclear piping.

1) Terminal Ends

2) At intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loading associated with one-half safe shutdown earthquake and operational plant conditions exceed 0.8 (S_h + S_A). This stress requirement is based on the sum of Equations (9) and (10) of Paragraphs NC-3652 and ND-3652 of ASME Code, Section III.

3) If no locations exceed the criteria in 2) above, breaks are selected at intermediate locations of relative high stress as compared to the remainder of the piping run.

4) Where break locations are selected without the benefit of stress calculation, breaks are selected at each pipe fitting (e.g., elbow, tee, cross, flange, and non-standard fitting), welded attachment and valve. Where the piping contains no fittings, welded attachments, or valves, break locations at one location at each extreme of the piping run adjacent to a protective structure are selected. These locations are used for non-nuclear safety class piping for which stress and/or usage factor ranking is not available.

5) As a minimum, there are two intermediate locations chosen from 2), 3) or 4) above in each piping run or branch run.

NRC Generic Letter 87-11 (Reference 17) revised Branch Technical Position MEB 3-1 as contained in the Standard Review Plan (SRP), Section 3.6.2 (Reference 18). All modifications to Class 2 & 3 piping may invoke this new criteria to eliminate arbitrary intermediate pipe breaks in lieu of the original criteria provided that the requirements stipulated in References 17 and 18 are fully complied with.

3.6.2.2.2 High Energy Piping Systems Outside Containment

This subsection provides the criteria used to locate postulated piping failures in ASME Code, Section III Code Classes 2 and 3 piping outside containment. The criteria used is based on the letter sent by A. Giambusso which was included as Appendix B to Branch Technical Position APCSB 3-1. There are no Code Class 1 piping runs outside containment.

Piping failures are postulated to occur at the following locations in each piping run or branch run:

1) Terminal Ends

2) At intermediate locations between terminal ends where either the circumferential or longitudinal stresses, derived on an elastically-calculated basis under the loadings associated with seismic events and operational plant conditions, exceed 0.8 (S_h + S_A) or the expansion stresses exceed 0.8 S_A.

3) Where the stresses calculated for the piping between terminal ends are less than the stress limits stated above, intermediate break locations are chosen on the basis of relative high stress as compared to the remainder of the piping run.

4) As a minimum, there are two intermediate locations chosen from 2) or 3) above in each piping run or branch run.

3.6.2.3 Types of Postulated High Energy Pipe Breaks (Inside and Outside Containment)

Pipe breaks are postulated at locations, inside and outside containment, identified in Subsections 3.6.2.1, and 3.6.2.2 with the following types of break configurations:

a) RCS main loop piping

Refer to Subsection 3.6.2.1.1 a).

b) Piping other than RCS main loop

1) Circumferential piping failures are postulated in piping greater than one in. nominal pipe size. Circumferential breaks are perpendicular to the pipe axis and result in complete pipe severance. The break area is equivalent to the pipe flow area.

2) Longitudinal piping failures are postulated in piping four inches nominal pipe size and greater and are assumed to be an axial split of rectangular shape without complete pipe severance. Longitudinal breaks are postulated to occur at any location about the circumference of the pipe to determine the location that results in maximum damage. The area of a longitudinal break is taken to be equal to the cross sectional flow area of the ruptured line. A length twice the inside pipe diameter and a width of $\pi D/8$ is assumed.

Longitudinal breaks for jet impingement analysis are not postulated at:

(a) Terminal ends

(b) At intermediate points where the criterion for a minimum number of break locations must be satisfied.

3 6.2.4 Location and Configuration of Moderate Energy Leakage Cracks

a) For ASME Code Class 2, 3 and non-nuclear moderate energy piping systems routed in areas containing no high energy piping, but which are located near components or structures required for safe shutdown, through wall leakage cracks are postulated to occur at any location that results in the maximum effects from fluid spraying and flooding, with the consequent hazard or environmental conditions developed.

b) Cracks are not postulated in moderate energy fluid system piping located in an area in which a break in high energy fluid system piping is postulated, provided such cracks will not result in more limiting environmental conditions than the high energy piping break.

c) Cracks are postulated in moderate energy fluid system piping and branch runs exceeding a nominal pipe size of one inch.

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

3.6.2.5 Containment Piping Penetrations

The flued head anchor of a containment piping penetration is considered a terminal end.

High energy pipe breaks and moderate energy leakage cracks are postulated based on the criteria of Subsections 3.6.2.2.1(c), 3.6.2.2.2 and 3.6.2.4 for piping located in the penetration area.

Piping failures are postulated at the circumferential welds joining the flued head anchor to: 1) the process line outside containment, and 2) the piping spool which is part of the penetration assembly inside containment. Figures provided in Subsection 3.8.2 give typical arrangements for containment piping penetrations.

Break configurations at the flued head anchors are chosen in accordance with Subsection 3.6.2.3.

3.6.2.6 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.6.1 RCS Main Loop Piping

The methods to define the forcing functions, response models and analysis technique presented in Reference 1 are applicable to St. Lucie Unit 2.

3.6.2.6.2 Forcing Functions Due to High Energy Pipe Breaks for Pipe Whip Analysis (inside and outside containment)

A break opening time equal to or less than one millisecond is assumed in the pipe rupture analysis.

The blowdown reaction force due to high energy fluid escaping from a postulated piping failure is described by the following steady state relationship:

$$\text{Blowdown Reaction Force (lb)} = KP_oA \qquad (1)$$

where:

P_o is the maximum operating pressure in the line prior to failure

A is the break area as defined in Subsection 3.6.2.3b.

K is a fluid phase dependent thrust coefficient defined as follows:

Type of Fluid at Ambient Conditions		K-factors Slot Break	Guillotine Break
1.	Wet Steam or flashing Water	0.68	1.12
2.	Dry Steam	0.63	1.01
3.	Subcooled Water	0.97	1.56

* If the line being considered is a pump discharge header, the K-factor is 1.19.

A dynamic loading factor of 2.0 is used to account for the suddenly applied nature of the blowdown reaction force. Therefore, pipe whip restraints are designed for $2KP_oA$.

To verify that the restraints are adequately designed, since the design is based on a static force applied with a dynamic factor of 2.0, it is necessary to examine the dynamic characteristics of the force as well as the structural response of the restraints.

The dynamic behavior of the reaction force on any pipe in which rupture is postulated depends on several factors among which are:

a) Characteristics of contained fluid (fluid phase: dry steam, wet steam or flashing water, subcooled compressed water, pressure, temperature).

b) Capacity of reservoir behind break

c) Configuration of piping system

d) Location of the break with respect to the reservoir (friction effects)

e) Presence of flow restrictors in the line

f) Type of break (longitudinal or circumferential)

The initial peak value of the reaction force is only slightly affected by friction effects, phase change effects and flow restrictors. However, the subsequent transient phase is very much affected by the above factors and by the actual configuration of the system. Thus, for a given break, it is easier to predict the first peak value of the reaction force than it is to predict the subsequent transient phase. From experiments(2) also published literature(3), the following can be stated with regard to this transient regime:

a) For flashing decompression, such as would occur in high energy lines with fluid temperature above 212 F, the initial force peak decays rapidly as the pressure drops to the saturation pressure for the given temperature then stabilizes at a slowly decaying value.

The duration of this sharp transient is dependent on: 1) the location of the break with respect to the reservoir, 2) the presence of any flow obstacles between the reservoir and the break, and 3) the friction in the line. In general, the transient duration will be less than one or two milliseconds.

b) For steam line breaks, again the peak force value lasts only a few milliseconds. Prior or subsequent values of the force are lower, although the effect is not as pronounced as in a flashing line break.

c) For subcooled decompression, the initial peak force is followed by acoustic pressure wave oscillations which depend on piping system geometry and flow characteristics. Examples of high energy lines in which subcooled decompression would occur following a piping failure are: auxiliary feedwater, charging pump discharge (up to regenerative heat exchanger) and safety injection tank discharge.

Since the broken pipes require times in the order of milliseconds to travel through the gap and strike the restraints, and the peak force will not be acting constantly, the total impulse received will be less than that which would be calculated by using the peak force as constant. Furthermore, the energy imparted will be even less, e.g., half the impulse corresponds to one quarter of the energy. Thus a constant force of lower magnitude is justified in a static analysis.

The actual force on a restraint at the time of impact is equal to the sum of the blowdown force (F_b) acting on the broken pipe at that time (transmitted through a suitable lever arm) and the force due to the energy acquired by the pipe as a result of its acceleration and velocity (F_a) through the existing gap between the pipe and the restraint.

F_b and F_a can only be determined by performing a thermal-hydraulic transient analysis and a structural dynamic analysis. Therefore, the equivalent static force for the restraint design is obtained as the product of the steady state blowdown force and a dynamic load factor (DLF); to account for load increase due to dynamic motion of pipe through the gap and dynamic response of the restraint and the supporting structure.

The peak thrust values used in the static analysis are essentially the peak theoretical values assured in the methodology of Reference 4, corrected to account for friction effects. In case of slot breaks the exit friction losses become very significant, thus even if the initial thrust (no flow) equals P_oA, its duration is such that the impulse may be negligible with respect to the total impulse developed under flow conditions.

In the final analysis, quantitative justification of the coefficient used to predict blowdown thrust(5) and the correctness of the DLF employed can only be provided by performing a dynamic analysis of the piping-restraint system. To verify the adequacy of the static analysis method employed, the Main Steam and Feedwater Systems have been dynamically analyzed(6) and the results are compared to those obtained by the static method. Refer to Appendix 3.6E for the results.

Pipe whip restraints on other high energy systems are selectively analyzed to verify their adequacy.

3.6.2.7 Analysis Methods to Verify Integrity and Operability

3.6.2.7.1 Methods Used to Evaluate Jet Impingement Effects Inside and Outside Containment

The geometry of the jet stream, its pressure distribution and temperature distribution depend on the properties of the discharged fluid, the surrounding medium, the fluid conditions at the exit plane (i.e., critical flow), and the distance from the break plane.

Two types of breaks and three kinds of jet development are considered. For postulated circumferential failures, distances from the failure to a possible target are measured along the axis of the pipe. For slot breaks and circumferential breaks with limited separation, distances are measured perpendicular to the axis of the pipe.

a) Guillotine break jet development: This break is perpendicular to the pipe axis with complete severance. This is postulated to result in the development of two free and clear jets whose shape is independent of the fluid phase (see Figure 3.6-Ia) For this break, the break area equals the inside flow area of the pipe.

b) Guillotine break with limited separation jet development: This special case of jet development is used only where the displacement of the severed ends of the pipe is limited by pipe stops. For this kind of jet development, the dynamic analysis will verify that the pipe ends remain within the following bounds with respect to each other.

   axial separation ≤ 0.5 pipe inside diameter

   lateral separation ≤ 0.5 pipe inside diameter

   The jet which develops from this case is shown on Figure 3.6-1b

c) Slot break jet development: (excluding reactor coolant loop slot breaks): This break is an axial split of rectangular shape whose break area is equivalent to the effective cross-sectional flow area of the pipe at the break location. The length (l) of the slot is assumed to be 2 D_i and the width (w) is assumed to be $\pi D_i/8$. The jet which develops from slot break is shown on Figure 3.6-1c.

d) Slot break jet development: See Figure 3.6-1(b)

   For calculation of fluid jet profiles, three fluid states are considered:

   a) Wet Steam or Flashing water

   b) Dry Steam

c) Subcooled Water

   The operating conditions, such as enthalpy, pressure, and temperature of a fluid inside a pipe, together with ambient pressure and temperature, are used to compute forces imposed by a fluid jet emanating from a ruptured pipe on targets located within the jet profile. Impingement force derivations assume that there is a steady turbulent flow within the pipe at the point of break.

The jet impingement force on a target is calculated from:

$$F = \frac{F_j}{A_j} A_x G \, (DLF)$$

where:

DLF = 2 = dynamic load factor

A_x = projected impact area of the target

G = geometric shape factor

Aj = cross sectional area of jet at the target distance from break plane.

Fj = total jet impingement force at the target plane

= $UmKP_oA$ (if the target is located at a distance greater than five pipe diameters from the break plane of a system containing flashing fluid, P_{sat} is used instead of P_o and for the term "$UmKP_0A$"

A = break area

P_o = maximum operating pressure in the line prior to failure.

K = fluid phase dependent thrust coefficient (refer to Appendices 3.6A and 3.6B for values of K used in the jet impingement analysis).

U_m = ratio of the jet velocity at the target plane to the velocity at the break plane

P_{sat} = saturation pressure corresponding to maximum operating temperature in the pipe to failure

Jet impingement forces for RCS main coolant pipe breaks are calculated from dynamic analyses. The time history blowdown data of Section 6.2.1.2 for Subcompartment Analysis are used. The target jet impingement force is calculated from the following equation.

$$F = \frac{F_j}{A_j} A_x G$$

where:

$$F_j = U_m K P(t) A(t)$$

P(t) is the time-dependent total pressure occurring in the ruptured pipe at the break location. A(t) is the time-dependent break area. Other variables are defined as stated above.

The jet streams [7,8] for all fluid states are divided into two regions as indicated on Figure 3.6-1c. The parameters used in the jet development are provided in a), b) and c) below.

a) Wet Steam or Flashing Water:

1) The initial region divergence angle "a" can be obtained as:

$$\tan a = \frac{c'(1 + 0.5\, X_o)}{(1 + X_o)}$$

Where:

c' = 0.27 for slot case

c' = 0.22 for guillotine case

$$X_o = \frac{Liquid\ mass\ flow\ rate}{Vapor\ mass\ flow\ rate} = \frac{(1.0 - quality)}{quality}$$

2) Initial region length:

$$X_h = \frac{3.7\, R_e\, (1 + X_o)^2}{(1 + 0.5\, X_o)\,(0.416 + 0.307\, X_o)}$$

where:

$$R_e = \frac{(1 + w)}{4}\ for\ slot\ break$$
$$= D_i/2\ for\ guillotine\ break.$$

3) Main region divergence angle:

a' = 12°

b) Dry Steam

1) The divergence of the jet in the initial region is given by

$$\frac{b}{R_e}\ or\ \frac{r}{R_e} = \sqrt{1/(0.13400\theta)} + \frac{0.22}{R_e}(x - x_h)$$

where:

$$\theta = \frac{Absolute\ operating\ temperature\ of\ fluid}{Absolute\ ambient\ temperature}$$

R_e = *defined in a) above.*

2) Initial region length

X_h is determined from Figure 7.35 in Abramovitch[8].

3) Main region divergence

The main region divergence is determined from Abramovitch[8], Figures 7.34 and 7.38 for slot and guillotine cases, respectively.

c) Subcooled Water

The following parameters are kept constant:

Case	Initial Region Divergence	Initial Region Length X_h	Main Region Divergence
Slot	8.33°	8 R_e	12.5°
Guillotine	8.33°	4.5 R_e	12.5°

3.6.2.7.2 Methods Used to Locate Pipe Whip Restraints

3.6.2.7.2.1 RCS Main Loop Piping

Circumferential (guillotine) and longitudinal (slot) breaks were postulated for the RCS hot and cold legs in the original plant design. Since then, however, the NRC revised General Design Criteria (GDC) 4 to eliminate the consideration of dynamic effects of a loss of coolant accident from the plant design bases. The dynamic effects of a LOCA include the effects of missiles, pipe whipping, discharging fluid (i.e., jet impingement), decompression waves within the ruptured pipe and dynamic or nonstatic pressurization in cavities, compartments, and subcompartments. Reference 13 demonstrates that the primary loop piping meets all of the criteria for application of leak before break presented in NUREG-1061, Volume 3. As a result, the mechanical/structural loads associated with dynamic effects of guillotine and slot breaks in the RCS hot and cold legs are no longer considered a plant design basis (References 14 and 15).

The pipe whip restraints installed to mitigate the consequences of such breaks, however, remain in place and functional. Reference 1 described the original criteria used for providing pipe stops in the RCS main loop.

Reference 1 describes the criteria used for providing pipe stops in the RCS main Loop.

In addition the criteria utilized for restraint of the reactor coolant loops include the following:

a) The supports for the steam generators and reactor are designed to accommodate pipe rupture loadings associated with either main steam line break or loss of coolant accident (LOCA).

b) Each reactor coolant pump is restrained from becoming a missile in the event of a LOCA.

c) The containment vessel is protected from possible pipe whip and fluid jet effects of a LOCA pipe rupture by the secondary shield wall.

The locations of the restraints in accordance with criterion b) are shown in Appendix 3.6C. The restraints to meet criterion b) include reactor coolant pump (RCP) suction line stops and restraints around the RCP motor.

The criteria described herein used for pipe rupture analyses, and results presented in Appendices 3.6A, 3.6B, and 3.6F, demonstrate that the protection of safety related systems is equivalent to that afforded by the criteria of Regulatory Guide 1.46 (R0) for inside containment and APCSB 3-1 for outside the containment.

3.6.2.7.2.2 Piping Excluding RCS Main Loop Piping

a) Philosophy

Pipe whip restraints are installed where required to protect essential equipment from possible impact by a whipping high energy line in which a piping failure is postulated.

Whipping of a high energy line following its postulated failure is evaluated per b), and c) of this subsection. This evaluation is based on the piping configuration, the break location and the ultimate moment and torsion carrying capabilities of the failed pipe with respect to its blowdown force. The blowdown force is defined in Subsection 3.6.2.6.2. Typical piping configurations, break locations and restraint requirements to confine pipe whip are shown on Figure 3.6.2. Pipe whip restraint spacing is determined from the evaluation of the load carrying capabilities of the ruptured pipe as per b) of this subsection. To implement the "break anywhere" criteria of Subsection 3.6.2.1.1, the need for pipe whip restraints is considered at all locations on high energy lines. Break anywhere criteria is no longer applicable as described in Subsection 3.6.2.1.1 b). Pipe whip restraints are deleted at those locations where analysis shows that pipe whip would not compromise the function of essential equipment,

The following parameters are considered in determining the minimum gap between the pipe surface and the restraint surface:

1) pipe insulation thickness

2) pipe thermal displacements

3) pipe seismic displacements

4) construction tolerance

For design of the pipe whip restraints, see Subsections 3.8.3 and 3.8.4.

b) Ultimate load Capability

In the evaluation of the ultimate load capability, it is assumed that a structure fails when the applied loads produce maximum true strain value equal to the strain hardening exponent. The object of this approach is to find the maximum unrestrained lengths of pipe that may undergo plastic deformation under specified loading. This idealization is carried out such that the resulting load carrying capacity is less than/equal to that derived from the true stress strain curve.

The method used by Stokey, Peterson and Wunder[9] to evaluate limit loads for tubes under internal pressure, bending moment, axial force and torsion in rigid-plastic material without strain- hardening has been adopted and amplified to account for the effect of strain-hardening. This effect has been evaluated following the method outlined in References 10, 11 and 12. The Tresca, or maximum shear stress theory, has been applied in this development. The effect of axial force is neglected in deriving the following interaction equations.

The interaction equation between ultimate torsion and moment capabilities is given by:

$$\frac{M}{Mo} = 1 + \frac{\pi}{4}\left[\frac{Sm}{Sy} - \sqrt{\frac{Su}{Sy}^{2} - 4\,\frac{T}{To}^{2}} - 1\right] \qquad (1)$$

Since most piping layouts are designed with straight runs and 90 degree bends or elbows, interaction between bending and torsion seldom occurs. Therefore, the ultimate load carrying capacity of pipe based on pure bending and internal pressure is calculated from:

1) Cantilever bending

$$\frac{FLmax}{Mo} = \frac{M}{Mo} = 1 + \frac{\pi}{4}\left[\frac{Su - Sy}{Sy} - \frac{Sm}{Sy}\right] \qquad (2)$$

2) Beam with slot break (see Figure 3.6-2)

$$\frac{\mathrm{FLmax}}{\mathrm{Mo}} = \left[3.2 + \frac{3\pi}{4}\left\{\frac{\mathrm{Su - Sy}}{\mathrm{Sy}} = \frac{\mathrm{Sm}}{\mathrm{Sy}}\right\}\right] \qquad (3)$$

The ultimate load carrying capability of a pipe based on pure torsion and internal pressure is calculated from:

$$\frac{FLmax}{To} = \frac{T}{To} = \frac{1}{2}\sqrt{\left(\frac{Su}{Sy}\right)^{2} - \left(\frac{Sm}{Sy} + 1 - \frac{4}{\pi}\right)^{2}}$$

where:

Mo	$= 4\ tr^2\ Sy$
To	$= 2\ \pi r_t^2\ Sy$
t	= wall thickness of pipe
r	= mean pipe radius
Sy	= minimum yield strength of material
Su	= minimum ultimate material strength
Sm	= circumferential pressure stress $\left(Sm = \frac{Pr}{t}\right)$
P	= operating pressure
M	= applied moment
T	= applied torque
F	= blowdown reaction force
Lmax	= maximum span for pipe whip restraints

c) Maximum Spacing of Restraints for Pipe Rupture Loading

The application of the relationships outlined in b) above for designing the actual spacing requirements for pipe whip restraints is presented below.

Maximum permissible unrestrained spans (Lmax) are computed for three elementary pipe configurations: cantilever bending, cantilever torsion, and beam bending The first two cases involve a guillotine break and the third, a slot break.

The actual restraint span used in the design is given by $L = 0.9\ L_{max}$

where:

L = Restraint spacing used in the design (see Figure 3.6-2)

Figure 3.6-2 provides illustration of how restraint locations are based on the L determined for torsion, bending, and cantilever cases. Appendix 3.6C gives pipe whip restraint locations relative to piping isometric drawing.

3.6.2.8 Guard Pipe Assembly Design Criteria

Guard pipes are supplied for Types I, III, and IV piping penetration assemblies (hot, semi-hot, and containment sump lines, respectively) as discussed in Subsection 3.8.2. The guard pipe on the Type I penetration is inside of and separate from the containment vessel nozzle. The guard pipes for the other types of penetrations are welded to their respective containment vessel nozzles. Penetration design information is provided in Subsection 3.8.2 and typical penetration details are depicted on figures in Section 3.8.

In the event of rupture of a high energy process line within a guard pipe, the guard pipe protects the rest of the penetration assembly and other penetrations in their near proximity from jet impingement. The design directs the release fluid into the containment and maintains containment integrity. Moment limiting restraints are shown on figures in Section 3.8.

3.6.2.9 Material Submitted for the Operating License Review

Design basis piping break locations are provided on piping isometric drawings marked to indicate break location and type of break. These drawings are included in Appendix 3.6C.

Figures provided in Appendix 3.6D give structural detail of pipe whip restraint.

The analysis results, including the jet thrust and impingement functions and the pipe whip dynamic effects are presented in Appendices 3.6A and 3.6B.

The design adequacy of systems, components and component supports, with respect to the effects of postulated piping failures, is presented in Appendices 3.6A and 3.6B.

Pipe whip dynamics analysis for the Main Steam and Feedwater Systems is presented in Appendix 3.6E.

Moderate energy leakage crack analysis is presented in Appendix 3.6F.

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5. Rettig, W.H. et Al, "RELAP-3 - A Computer Program for Reactor Blowdown Analysis," IN-1321, June 1970.

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14. USNRC letter dated March 5, 1993 from J. A. Norris (NRC) to J. H. Goldberg (FP&L).

15. FP&L letter L-92-231 dated August 26, 1992 from D. A. Sager (FP&L) to USNRC.

16. Federal Register, Volume 52, dated October 27, 1987.

17. US NRC Generic Letter 87-11, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements, dated June 19, 1987.

18. NUREG-0800, USNRC Standard Review Plan, Section 3.6.2, Branch Technical Position (BTP) 3-1, “Postulated Rupture Locations in Fluid System Inside and Outside Containment” (Revision 2 - June 1987).

19. PC/M 05215M, Main Steam and Feedwater Modifications For The N-1 Component Replacement Outage.

20. PC/M 05130, Steam Generator 2A & 2B Blowdown Piping Systems for the Unit 2 Component Replacement Projects.

21. PC/M 05133, Steam Generator 2A & 2B Supports Modification for Unit 2 Component Replacement Projects.

TABLE 3.6-1

ESSENTIAL SYSTEMS TO MITIGATE CONSEQUENCES
OF POSTULATED PIPING FAILURES

Essential Systems or Portions of Systems Required to Safely Shutdown the Plant and/or to Mitigate the Consequences of Pipe Break	Loss-of-Coolant Accidents	Main Steam/Feedwater Line Breaks Inside Containment	Main Steam/Feedwater Line Breaks Outside Containment	All other non-DBAs* Pipe Breaks-Outside Containment
1) Reactor Protection System (at onset of accident)	X	X	X	X
2) Engineered Safety Features Actuation System	X	X	X	
3) Containment Isolation System	X	X		
4) Auxiliary Feedwater System	X	X	X	X
5) Main Steam and Feedwater Systems (from SG to MSIV/MFWIV)	X	X	X	X
6) Class 1E Electrical Systems, ac and dc (including switchgear, batteries and distribution systems)	X	X	X	X
7) Diesel Generators (including jacket water cooling and lube oil)	X	X	X	X
8) Diesel Fuel Oil Storage and Transfer System	X	X	X	X
9) Intake Cooling Water System	X	X	X	X
10) Component Cooling Water System	X	X	X	X
11) HVAC Systems (safety related portions)	X	X	X	X
12) Instruments as required	X	X	X	X
13) Reactor Coolant System	X	X	X	X
14) Safety Injection System	X	X	X	
15) Containment Spray/Iodine Removal Systems	X	X		
16) Containment Cooling System	X	X		X
17) Shield Building Ventilation System	X	X		
18) Shutdown Cooling System	X	X	X	X
19) Combustible Gas Control System (recombiners and analyzer)	X			
20) Chemical and Volume Control System (charging portion)	X	X	X	X

* For this Table, DBA breaks are those which result in automatic or manual ESFAS operation. All other breaks are non-DBA breaks.

TABLE 3.6-2
(HISTORICAL)
COLD LEG PIPE STOP STIFFNESS

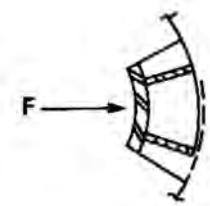
DIRECTION	PIPE STOP \ STOP	STIFFNESS (K/IN) STOP NO. 1	STOP NO. 2	STOP NO. 3	REMARKS
RADIAL	2A1	300.3x10^{3}	295.1x10^{3}	299.3x10^{3}	
	2A2	356.2x10^{3}	252.9x10^{3}	355.9x10^{3}	
	2B1	222.6x10^{3}	222.8x10^{3}	221.0x10^{3}	
	2B2	319.2x10^{3}	366.7x10^{3}	319.3x10^{3}	
LONGITUDINAL	2A1	13.0x10^{3}	13.4x10^{3}	12.7x10^{3}	
	2A2	9.6x10^{3}	13.2x10^{3}	9.6x10^{3}	
	2B1	14.1x10^{3}	13.7x10^{3}	14.2x10^{3}	
	2B2	6.3x10^{3}	15.2x10^{3}	6.3x10^{3}	

TABLE 3.6-3

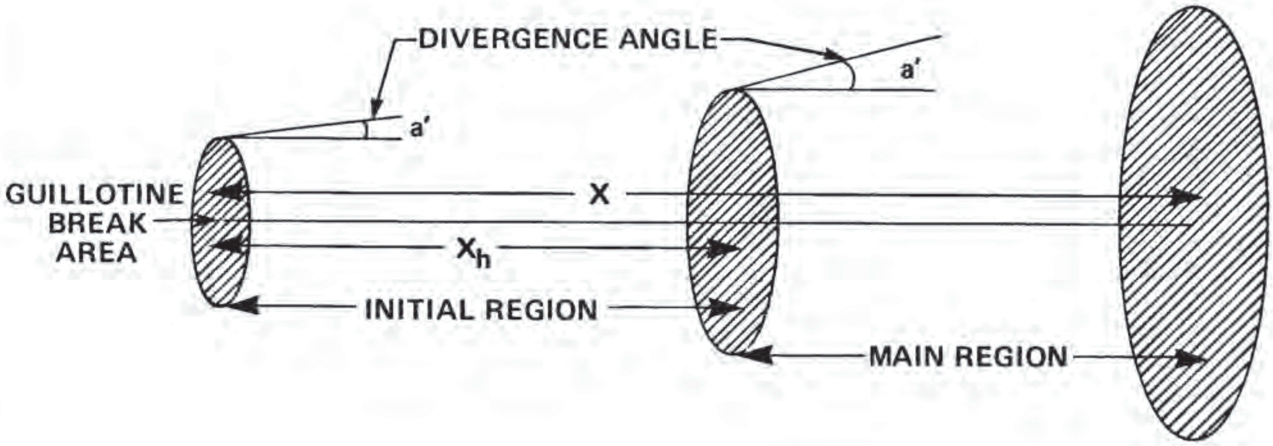
(HISTORICAL)
PIPE BREAK AREAS & BREAK OPENING
TIMES - PARITAL AREA GUILLOTINES

POSTULATED RUPTURE	BREAK FLOW AREA (IN^2)	RISE TIME (MILLISECONDS)
RV INLET GUILLOTINE	200.	6.
RV OUTLET GUILLOTINE	100.	20.
SG INLET GUILLOTINE	1000.	24.

TABLE 3.6-4

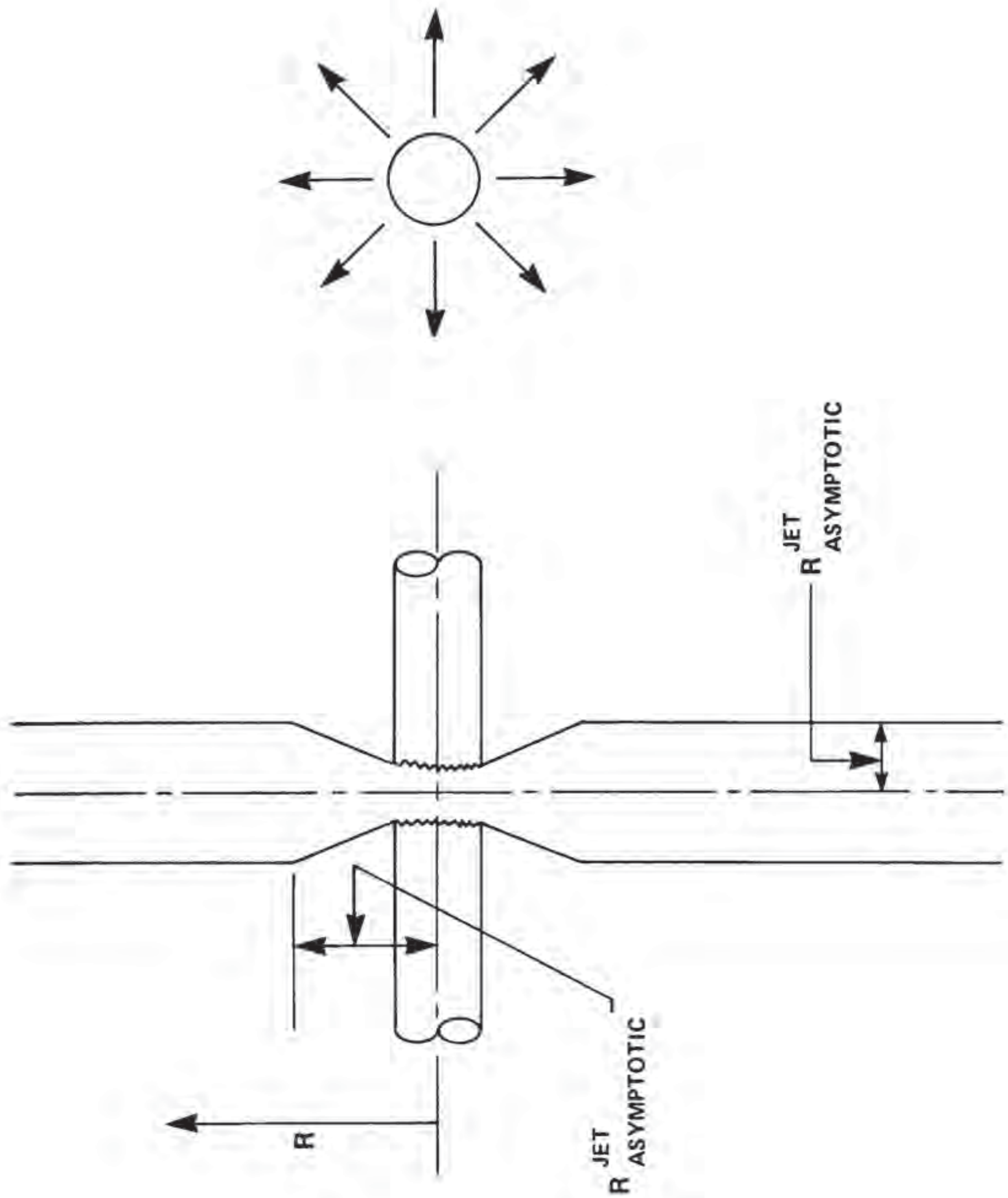
(HISTORICAL)
ST. LUCIE NO. 2 RCS DISCHARGE LEG PIPE RESTRAINTS
AXIAL GAPS

RESTRAINT	LONGITUDINAL STOP	AXIAL GAP (IN.
2A1	1	1/4
	2	1/4
	3	1/4
2A2	1	1
	2	1/2
	3	11/16
2B1	1	1
	2	17/32
	3	1 3/32
2B2	1	5/16
	2	1/4
	3	1/4



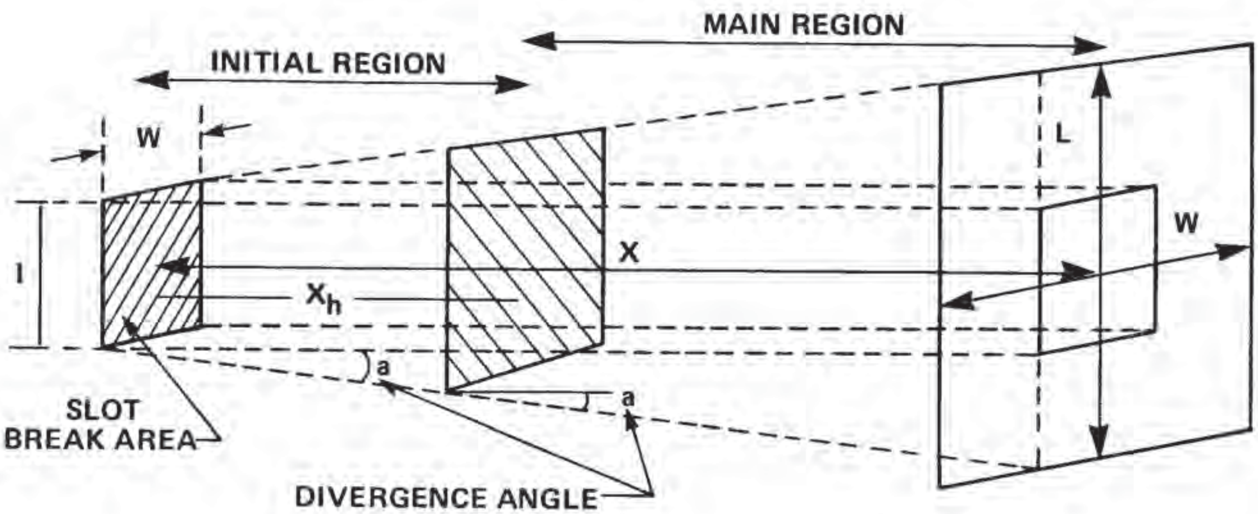
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

JET DIVERGENCE OF POSTULATED BRKS.
GUILLOTINE BREAK
FIGURE 3.6–1a



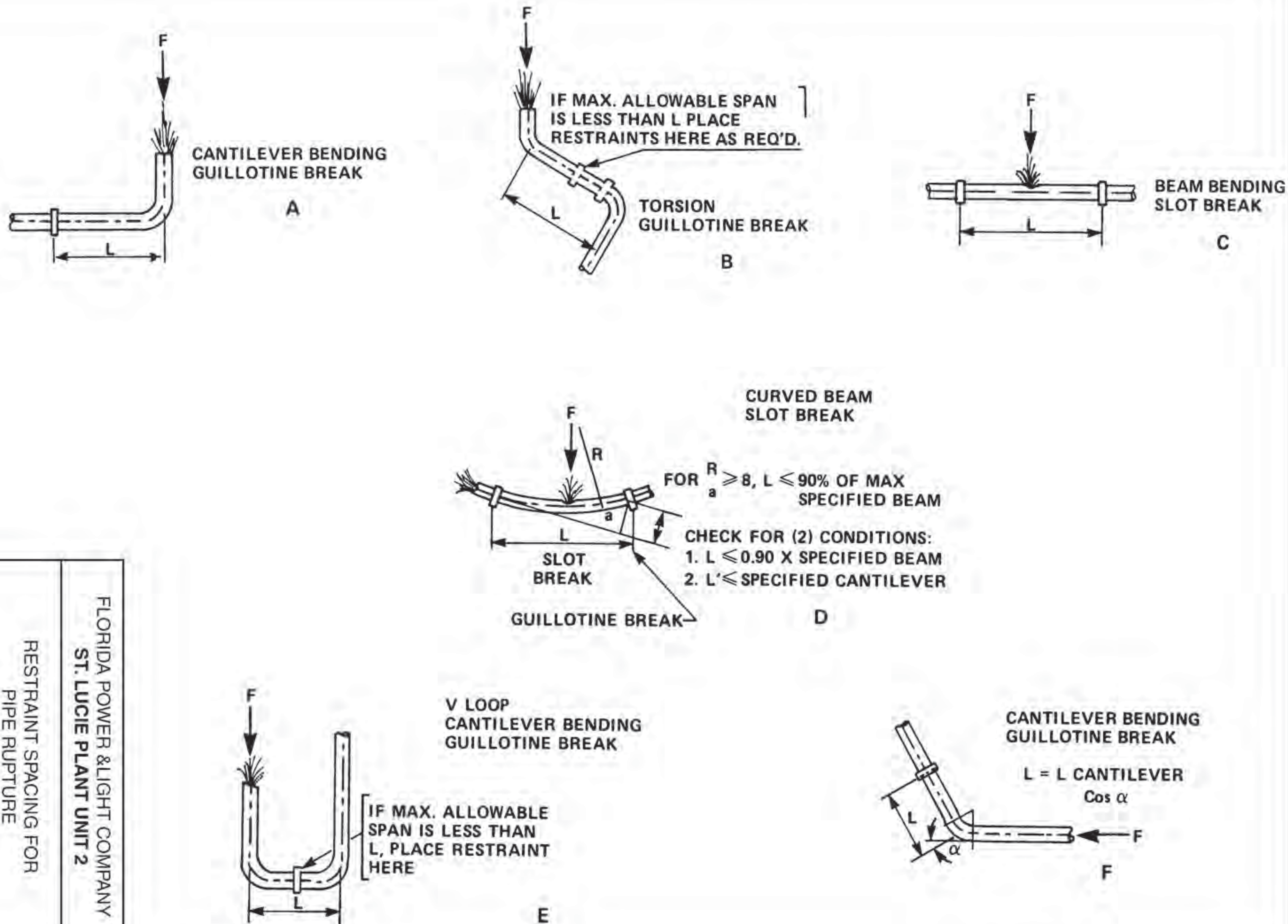
FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

JET DIVERGENCE OF POSTULATED BRKS.
GUILLOTINE BRK. WITH LIMITED SEPAR'TN.
FIGURE 3.6-1b



FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

JET DIVERGENCE OF POSTULATED BRKS
SLOT BREAK
FIGURE 3.6-1c



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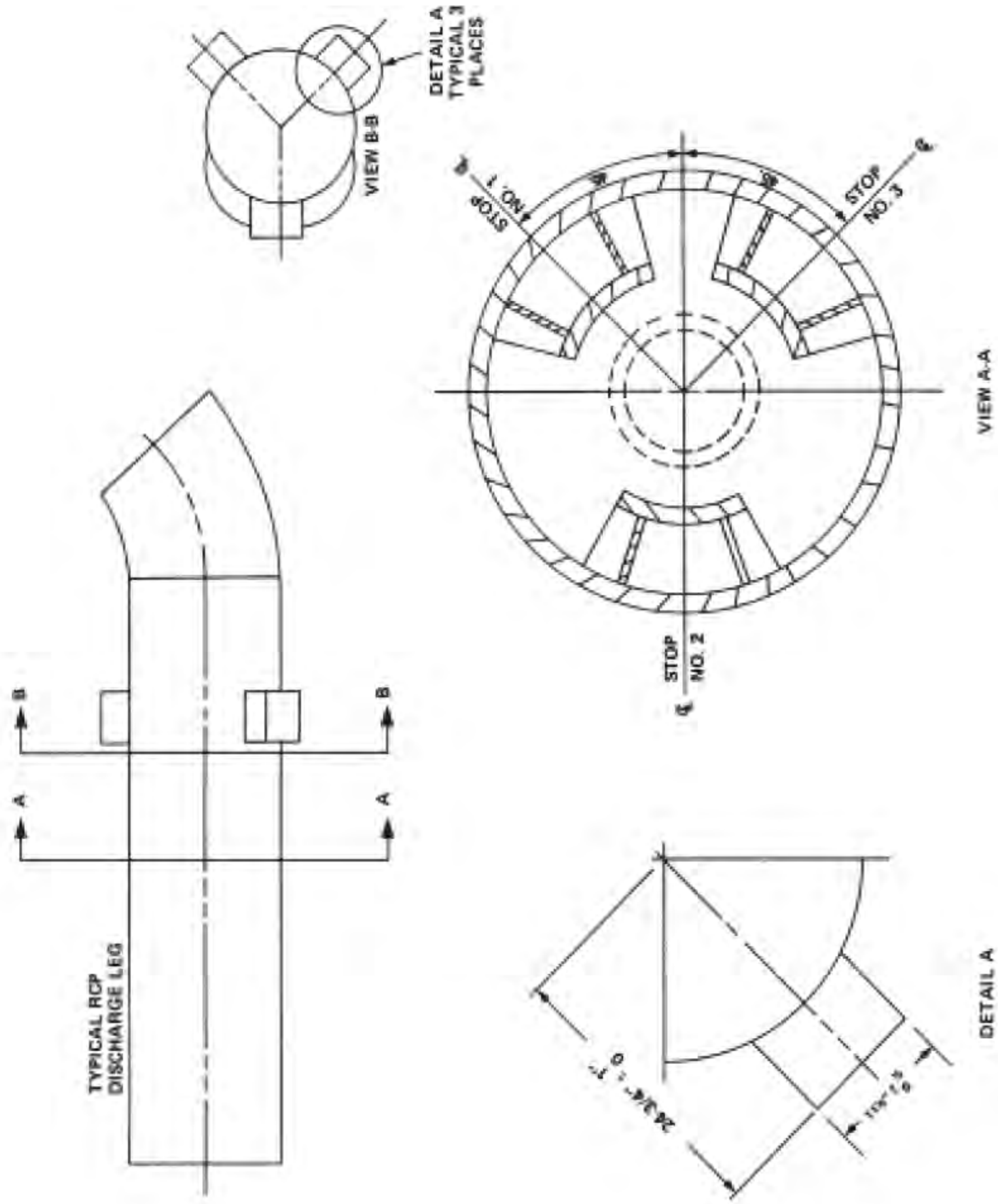
ST. LUCIE PLANT UNIT 2

RESTRAINT SPACING FOR
PIPE RUPTURE
FIGURE 3.6-2



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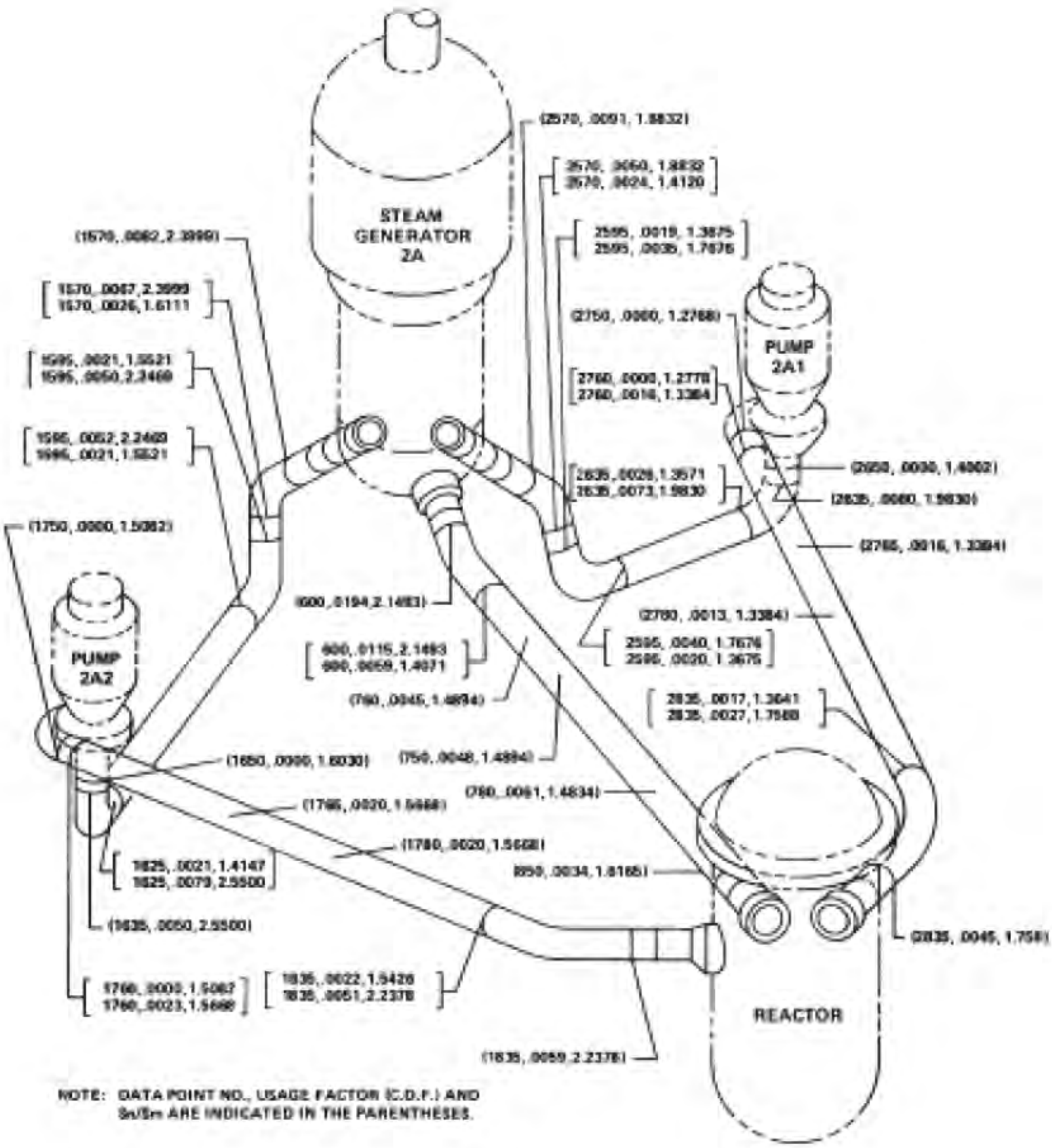
DISCHARGE LEG PIPE RESTRAINTS

FIGURE 3.6-3
(HISTORICAL)



NOTE: DATA POINT NO., USAGE FACTOR (C.D.F.) AND Sn/Sm ARE INDICATED IN THE PARENTHESES.

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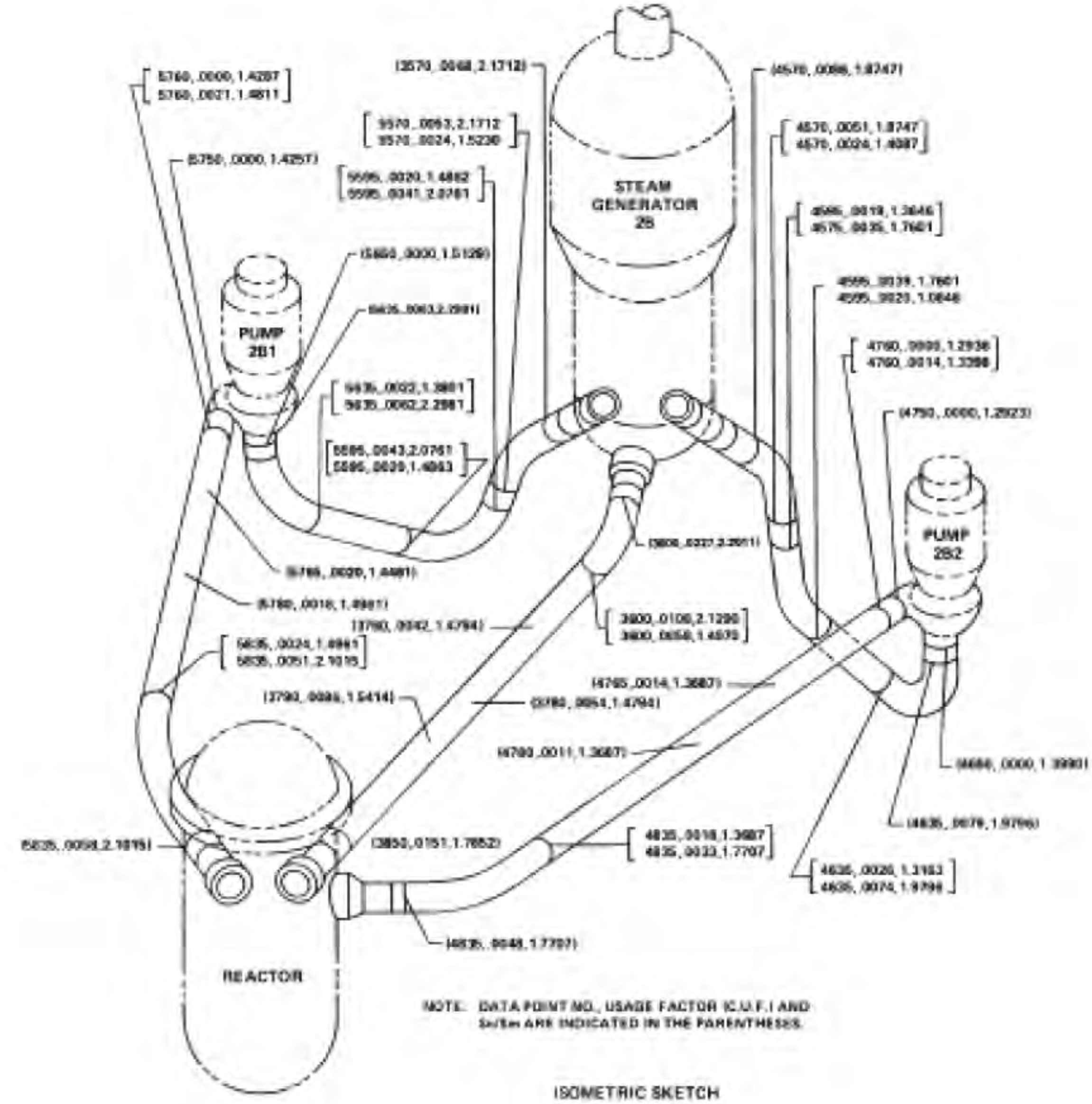
CUMUL. USAGE FACTOR & NORMALIZED PRIMARY PLUS SECONDARY STRESS INTENSITY RANGE RESULTS FOR SEISMIC LOADING (LOOP 2A) SH. 1 OF 3

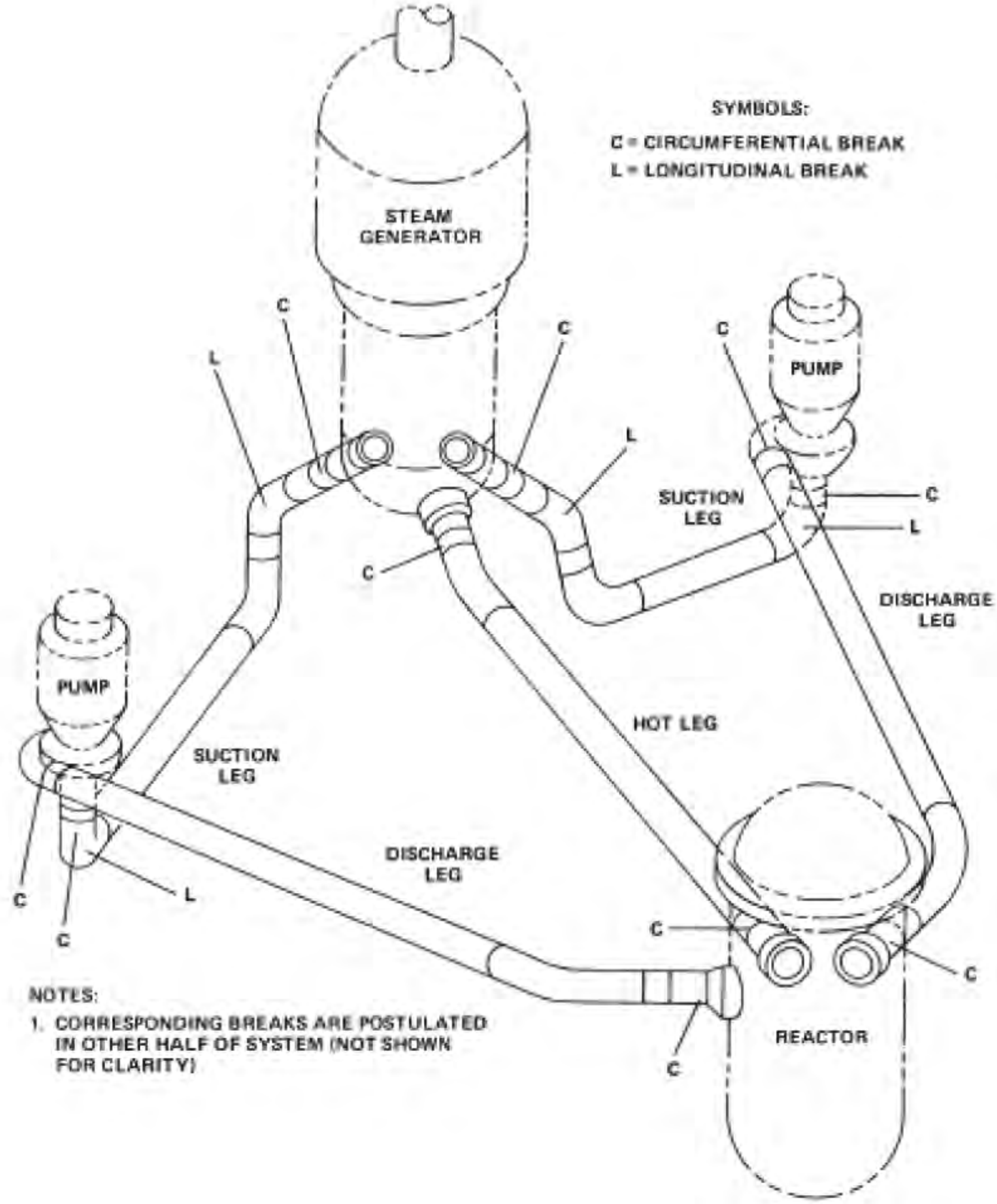
FIGURE 3.6-4
(HISTORICAL)



FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

CUMUL. USAGE FACTOR & NORMALIZED PRIMARY PLUS SECONDARY STRESS INTENSITY RANGE RESULTS FOR SEISMIC LOADING (LOOP 2B) SH. 2 OF 3

FIGURE 3.6-4
(HISTORICAL)



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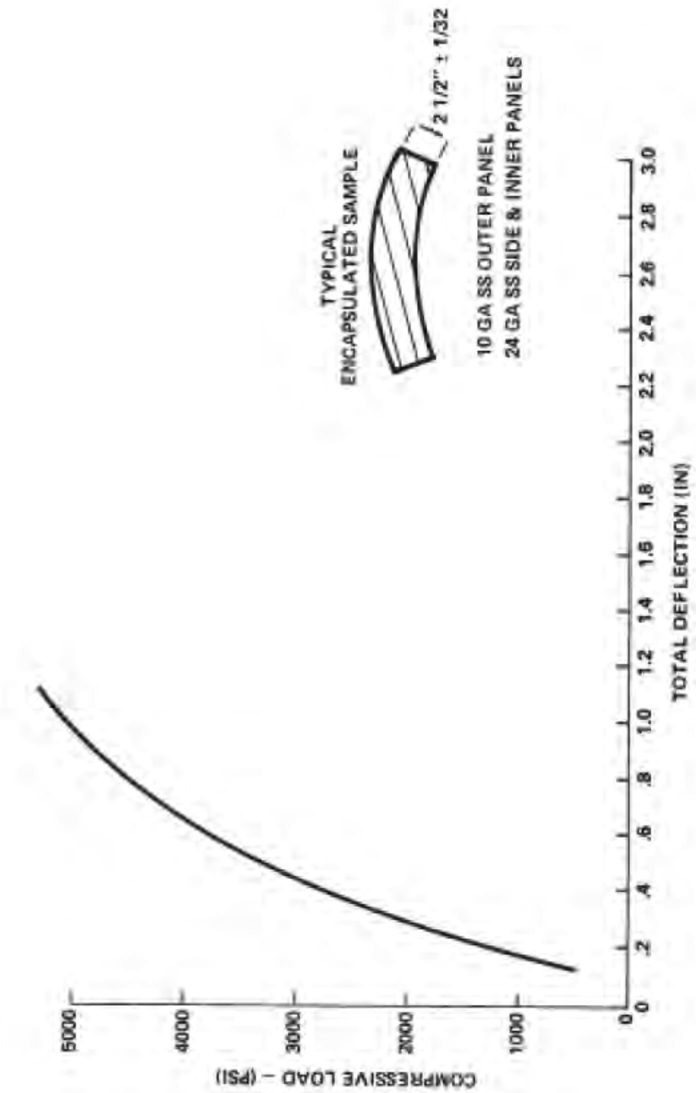
ST. LUCIE PLANT UNIT 2

DESIGN BASIS PIPE BREAKS
TYPES & LOCATIONS
SHEET 3 OF 3
FIGURE 3.6-4
(HISTORICAL)



FLORIDA POWER & LIGHT
ST. LUCIE PLANT UNIT 2

LOAD – DEFLECTION CURVE
ENCAPSULATED MARINITE

FIGURE 3.6-5
(HISTORICAL)

APPENDIX 3.6A

HIGH ENERGY PIPE RUPTURE ANALYSIS – INSIDE CONTAINMENT

APPENDIX 3.6A

3.6A HIGH ENERGY PIPE RUPTURE ANALYSIS - INSIDE CONTAINMENT

This appendix presents the results of high energy piping failure analysis inside containment. High energy piping failure analysis takes into consideration the effects of pipe whip, jet impingement loads and environmental conditions on the essential systems, components and structures. The results of high energy pipe rupture analysis-outside containment and moderate energy analysis are presented in Appendix 3.6B and 3.6F, respectively.

Appendix 3.6C contains isometric drawings for the high energy piping. These figures contain the following:

a) Postulated pipe failure locations for jet impingement analysis

b) Break configurations

c) Pipe whip restraint locations.

The high energy piping systems or portions of a system (i.e., fluid systems which exceed 200°F and/or 275 psig during normal operating conditions) that are considered for pipe failure analysis inside containment are:

a) Main Steam and Feedwater

b) Reactor Coolant (Includes pressurizer surge, spray and safety & relief)

c) Safety Injection (all lines pressurized by the Safety Injection Tanks)

d) Shutdown Cooling (high energy portion only)

e) Chemical and Volume Control (letdown and charging)

f) Steam Generator Blowdown

The results of existing pipe break analyses described in this Section are not impacted by EPU conditions.

3.6A.1 MAIN STEAM AND FEEDWATER INSIDE CONTAINMENT

3.6A.1.1 General Description

Pipe failures are considered in each of the two Main Steam (MS) lines commencing at the steam generators and terminating at the turbine stop valves. The two Feedwater (BF) lines between the feedwater pumps and the steam generators are also considered for pipe rupture analysis. The results of pipe failure analysis for MS and BF piping inside containment are presented in this section. The pipe break analysis for MS and BF piping outside containment are presented in Appendix 3.6B. Criteria used to define break locations and configuration for MS and BF piping inside containment are described in Subsections 3.6.2.1.1 and 3.6.2.2.1.

PC/M 05215M (Reference 19), which was performed in support of the Unit 2 Steam Generator Replacement, permanently deleted the horizontal main steam pipe whip restraints at the steam outlet nozzle and at the second elbow downstream of the steam outlet nozzle (RE-MS-11 and RE-MS-14 for SG2A, and RE-MS-1 and RE-MS-4 for SG2B). These restraint modifications were successfully evaluated by using the criteria of NRC Generic Letter 87-11 (Reference 17) and the Standard Review Plan (SRP), Section 3.6.2 (Reference 18) to eliminate these pipe whip restraints resulting from arbitrary intermediate pipe ruptures.

A discussion of the Main Steam and Feedwater Systems is given in Section 10.3 and Subsection 10.4.7. The MS process line which is part of the containment piping penetration assembly is constructed of ASME SA-106 Gr B seamless material. A guard pipe which is part of the penetration assembly encloses this piping. The remaining portion of Main Steam lines are carbon steel ASTM A-155 Gr KC-65 piping of longitudinally welded construction. The main steam lines inside containment consist of 36.625 inch and 34 inch diameter piping. The maximum operating pressure and temperature for the MS System is 888 psia and 530°F for EPU.

The Feedwater lines between the Steam Generator nozzles and the containment isolation valves are designed in accordance with the ASME Code Section III, Class 2 and seismic Category I criteria. The Feedwater lines consist of 18 inch and 20 inch diameter piping of ASME SA-106 Gr B seamless carbon steel construction. For EPU, the maximum operating pressure and temperature for the BF System is 1108 psig and 440°F. Isometrics of MS and BF piping inside containment are shown in Figures 3.6C-1.1 through 3.6C-1.4.

3.6A.1.2 Pipe Whip Analysis

Pipe whip analysis considers both longitudinal (slot) and circumferential (guillotine) types of failures to occur anywhere (i.e., "break anywhere" criterion) in the MS and BF piping systems. This approach is conservative since break any where criteria is no longer required per Subsection 3.6.2.1.1 b)

Based on the "break anywhere" criterion, the pipe whip restraints and guard pipes are located and designed for the MS and BF systems to accomplish the following in the event of failure of the MS or BF piping:

a) confine the steam generator movement

b) protect safety injection tanks, containment hatch, fan coolers and the containment vessel

c) confine pipe whip and fluid jet envelope (i.e., to prevent traveling jet)

d) prevent missile generation from the biological shield wall due to pipe impact

e) prevent impact of pipe on steam generator cubicle wall

f) protect essential systems, components and structures required to achieve safe plant shutdown.

The locations of pipe whip restraints for MS and BF Systems are shown in Figures 3.6C-1.1 through 3.6C-1.4.

The pipe whip restraints are designed to ensure that pipe whip will be minimized. The adequacy of the restraint systems to prevent pipe whip has been demonstrated by selective dynamic analysis for typical restraints. Appendix 3.6E presents the results of this dynamic analysis.

3.6A.1.3 Jet Impingement Analysis

The essential components, systems and structures required for safe shutdown of the plant in the event of MS or BF line break inside containment have been evaluated for the effects of jet impingement.

Discrete break locations have been chosen by the criteria of Subsection 3.6.2.2.1 for jet impingement analysis. These break locations are shown on Figures 3.6C-1.1 through 3.6C-1.4.

Jet envelopes resulting from MS and BF breaks inside containment have been used to identify jet interactions for each high energy break for electrical, instrumentation and control, HVAC, architectural/structural, civil and mechanical systems and components.

Component jet interactions were judged acceptable or unacceptable according to the plant shutdown requirements. A single active component failure and environmental effects of pipe break have also been considered for this decision. All interactions identified as unacceptable were further analyzed to determine the operability of the essential components under the jet impingement loading. When the analysis indicated loss of function either a barrier has been provided or the target has been reinforced.

In the jet impingement analysis it was observed that the jets from breaks R-BF-1-4, R-BF-2-3, R-BF-2-4, R-MS-1-4 and R-MS-2-4 interact with essential electrical trays. To eliminate these jet- interactions, jet deflectors were incorporated on the end of MS and BF guard pipes. The jet impingement forces on concrete structures such as steam generator cubicles, pressurizer cubicle, secondary and biological shield walls were calculated. The analyses indicate that the structural integrity will be maintained.

Various MS and BF jets impinge on containment spray header piping of 3 inch, 4 inch and 6 inch nominal diameter. The spray piping is located over 90 feet away from the break points. Thus jet impingement forces will not adversely affect the spray piping function.

The jets impinge on component cooling water lines which provide reactor coolant pump seal cooling. Since reactor coolant pumps are non-essential for plant shutdown or mitigating the consequences of a MS and BF pipe failure, the loss of component cooling water to the RC pumps is acceptable.

It was also determined that jet impingement on various non-essential components such as pipe whip restraints, structural frames, platforms, cable tray restraints, and HVAC duct restraints will not compromise plant shutdown capability.

The jets from breaks R-BF-1-4, R-MS-1-4, R-BF-2-4 and R-MS-2-4 impinge on platforms at El. 40.5 feet, structural columns and containment vessel. The loads on these structures are caused by jet expansion in radial directions through the gap between the penetration guard pipe and a second guard pipe extending from the containment vessel to the secondary shield wall. This interaction has been eliminated by the addition of deflectors at the end of penetration guard pipe which redirects jet expansion away from essential systems.

The jets from breaks R-MS-1-2, R-BF-1-2, R-MS-1-1, R-MS-2-2, R-BF-2-2, R-MS-2-1 impinge on the construction hatch cover and containment vessel. These jet impingement loads are included in the design analysis of construction hatch cover and containment vessel.

3.6A.1.4 Environmental Analysis

The safety related systems required to mitigate the consequences of main steam or feedwater line break inside containment are designed to function under environmental conditions resulting from a main steam or feedwater line break accident. Refer to Section 3.11.

3.6A.2 REACTOR COOLANT SYSTEM (INCLUDING PRESSURIZER SURGE, SPRAY AND RELIEF PIPING)

3.6A.2.1 Reactor Coolant System (Historical)

The Reactor Coolant System (RCS) is described in Section 5.1. A system flow diagram is shown in Figure 5.1.3. The reactor coolant loop piping is stainless steel clad, carbon steel piping designed and fabricated in accordance with ASME Code, Section III Class 1 and seismic Category I criteria. The two-loop RCS piping system consists of four 30 inch inside diameter cold legs (reactor coolant pump suction and discharge) and two 42 inch inside diameter hot legs. The layout of RCS piping is shown in Figures 5.1-1 and 5.1-2.

3.6A.2.1.2 Pipe Whip Analysis

Isometrics of the two RCS loops are provided in Figures 3.6C-2.1 and 3.6C-2.2. Criteria used for locations and types of pipe breaks are described in Subsection 3.6.2.1.1.

Pipe stops are provided in various locations of the RC loop to limit the piping motion. The effects of pipe stops are considered in developing the fluid jet expansion profiles used for jet impingement analysis and are described in Subsection 3.6.2.7.1. The location of pipe stops is shown in Figures 3.6C-2.1 and 3.6C-2.2.

The break area and pipe motion due to hot leg guillotine breaks at the reactor vessel nozzle and at the steam generator nozzle are limited by the short length of the hot leg and by the pipe stops on the steam generator base. The pipe stops installed in the primary shield wall limit movement of the reactor coolant pump discharge line due to the postulated break at the reactor vessel nozzle. The pumps, are restrained by cables above and below the discharge line and by vertical and horizontal stops above the motors. However, the postulated pipe failure at the RCP discharge nozzle is conservatively assumed to result in a full area opening.

Four breaks are postulated in each of the four reactor coolant pump suction lines. Guillotine breaks are postulated at the reactor coolant pump nozzle and at the steam generator nozzle. Longitudinal breaks are postulated in each RCP suction leg: at the pump suction nozzle elbow and the elbow closest to the steam generator. Slot breaks are assumed to occur anywhere in the elbow crotch region, 180^0 around the pipe circumference. The guillotine break at the RCP suction nozzle will be of limited opening, since the pipe is restrained from moving downward by a pipe stop and the RCP motion is confined by stops above the motor. The guillotine break of the RCP suction line at the steam generator nozzle will be a full area opening break and will result in pipe whip into the base mat at El. 18.0 feet.

The results of containment and subcompartment pressure analyses are provided in Section 6.2.1. The systems and equipment necessary to mitigate the consequences of a reactor coolant system failure (LOCA) are listed in Table 3.6-1.

3.6A.2.1.3 Jet Impingement Analysis

The essential components and systems located inside containment and required for safe shutdown of the plant in the event of a reactor coolant system pipe failure have been evaluated for the effects of jet impingement. The reactor coolant jets impinge on concrete structures such as the floor at El. 18.0 feet, primary and secondary shield walls, and slab at El. 62.0 feet. The analysis indicates that the integrity of these structures will be maintained. Reinforcement of the biological wall was provided such that integrity of the wall is maintained under the jet loading from breaks R-RC-25-1 and R-RC-30-1.

The jets also impinge on electrical conduits containing cables directed to non-essential equipment. The jets also impinge on non-essential HVAC ducts which are branch lines of the ring header.

3.6A.2.1.4 Environmental Analysis

The safety related systems required to mitigate the consequences of reactor coolant system lines failure inside containment are designed to perform their safety function under environmental conditions resulting from a reactor coolant loop piping failure. Refer to Section 3.11.

3.6A.2.2 Pressurizer Surge Line

3.6A.2.2.1 General Description

The pressurizer operation and design bases are described in Section 5.4.10. The surge line, which runs from the hot leg of the RC loop to the pressurizer, is a 12 inch nominal size schedule 160 stainless steel pipe. It is designed in accordance with ASME Code Section III Class 1 and Seismic Category I requirements.

3.6A.2.2.2 Pipe Whip Analysis

An isometric of the pressurizer surge line indicating postulated break points and pipe whip restraint locations is provided in Figure 3.6C-2.4. Pipe breaks are postulated at terminal ends, each fitting and all spool weld locations. Circumferential as well as longitudinal breaks are postulated at all intermediate pipe rupture locations. Only circumferential breaks are postulated at the terminal ends. The line is restrained to prevent adverse pipe whip effects on essential systems and components. In addition, the pipe whip restraints for the pressurizer surge line confine fluid jet envelope.

Pressurizer cubicle walls can withstand pipe whip and jet impingement effects. If instrumentation lines within the pressurizer cubicle are ruptured by pipe whip or jet impingement, they will produce low pressurizer pressure trip. All transducers required for trip are located outside the cubicle.

Pipe whip restraints for the pressurizer surge line are designed to support pipe rupture thrust loads in any radial direction of the pipe. The systems and equipment necessary to mitigate the consequences of a pressurizer surge line failure (LOCA) are listed in Table 3.6-1.

Surge Line pipe whip restraint RE-RC-30 depicted on Figure 3.6C-2.4 is shown for historical purposes as the pipe whip restraint design basis has been eliminated as addressed in PC/M No. 05131M, Surge Line Rupture Restraints RC-30 & RC-31 Modification for the Unit 2 Component Replacements Projects.

3.6A.2.2.3 Jet Impingement Analysis

The essential components and systems located inside the containment and required for safe shutdown of the plant in the event of pressurizer surge line break have been evaluated for the effects of jet impingement. Break locations for jet impingement analysis are the same as for the pipe whip analysis.

The pressurizer surge line jets impinge on concrete structures such as the floor at El. 18.0 feet, slab at El. 62.0 feet, secondary shield wall, refueling cavity wall, pressurizer foundation and primary shield wall. Analysis indicates that the integrity of these structures will be maintained.

The jets impinge on electrical conduits containing cables directed to non-essential equipment and non-essential HVAC ducts. The jets also impinge on architectural and structural elements such as HVAC duct restraints, non-essential equipment supports, trench covers, platforms, and pipe whip restraints which are not required for safe shutdown of the plant.

The jet from pressurizer surge Line breaks R-RC-31-15 and R-RC-31-5 impinge on pipe whip restraints RE-RC-34 and R-RC-33. The analysis of restraints for simultaneous jet and pipe whip loading indicate that restraints will function after impact.

Jets from the pressurizer surge line impinge on component cooling water lines 4-CC-143 and 4-CC-147, refueling cavity drain line I-3-CS-51 and its valves V07194 and V07195, shutdown cooling, safety injection and hot leg injection lines. These lines are not required to mitigate the consequences of a pressurizer surge line break.

3.6A.2.2.4 Environmental Analysis

The safety related systems required to mitigate the consequences of pressurizer surge line break inside containment are designed to perform their safety function under environmental conditions resulting from pressurizer surge line break. The resulting environmental conditions are bounded by the conditions of a main steam line / feedwater line break or reactor coolant system line break. Refer to sections 3.6A.1.4 and 3.6A.2.1.4.

3.6A.2.3 Pressurizer Spray Lines

3.6A.2.3.1 General Description

The pressurizer spray piping is described in Subsection 5.4.10. The spray lines consists of 3-inch schedule 160 and 4-inch schedule 160 seamless pipe. The stainless steel spray piping is designed in accordance with ASME Code, Section III Class 1 and seismic Category I requirements. Two 3-inch lines from cold legs 2B1 and 2B2 reactor coolant loop are run through spray control valves into a 4-inch line. The 4-inch line discharges to the steam space on the top of the pressurizer.

3.6A.2.3.2 Pipe Whip Analysis

Isometric drawings of the pressurizer spray lines indicating the postulated break points and pipe whip restraint locations are provided in Figure 3.6C-2.3. In addition to pipe breaks at the terminal ends, intermediate breaks are postulated at each fitting and weld location. All breaks in the 3 inch pipes are guillotine (circumferential) type breaks. Circumferential and longitudinal pipe breaks are postulated in the 4 inch portion of the pressurizer spray system per the criteria presented in Subsection 3.6.2.1. Pipe whip is minimized by the restraints such that the function of essential systems and equipment is not affected. The restraints in the 4 inch portion of the spray line are designed to support pipe rupture thrust loads in any radial direction of the pipe. This is to accommodate the assumption that the slot break can occur at any location about the circumference of the pipe. Pipe whip restraints are provided to confine pipe whip and jet envelope.

3.6A.2.3.3 Jet Impingement Analysis

The essential components and systems located inside containment and required for safe shutdown of the plant in the event of pressurizer spray line break have been evaluated for the effects of jet impingement. Break locations are the same as for the pipe whip analysis. Pressurizer spray line jets impinge on concrete structures such as primary and secondary shield walls the and refueling cavity wall. Calculated jet impingement loads indicate that these structures will function after impact. The jets impinge on electrical conduits, HVAC duct, restraints, platforms, pipe whip restraints, cable tray restraints and component cooling lines which are all non-essential for safe shutdown of the plant. Calculations of jet impingement loads on pipe whip restraint RE-RC-42 indicate that it will function after impact.

The jets also impinge on the surge line, shutdown cooling and safety injection lines which are of greater size than the broken line and therefore will not be affected.

3.6A.2.3.4 Environmental Analysis

The safety-related systems required to mitigate the consequences of pressurizer spray line breaks are designed to perform their safety function under environmental conditions resulting from pressurizer spray line break. The resulting environmental conditions are bounded by the conditions of a main steam line / feedwater line break or reactor coolant system line break. Refer to sections 3.6A.1.4 and 3.6A.2.1.4.

3.6A.2.4 Pressurizer Relief Line

3.6A.2.4.1 General Description

The pressurizer relief system is described in Subsection 5.4.13. The pressurizer relief line starts from a 4 inch nozzle in the steam space of the pressurizer and branches to two 3 inch lines. Each line is provided with a solenoid-operated power relief valve and a motor operated isolation valve. The isolation valve is provided for maintenance of each power operated relief valve.

The power operated relief valve discharge lines are joined together and are routed to the Quench tank.

Piping upstream of the relief valves is classified as high energy. Piping downstream of relief valves is classified as moderate energy, since it is not normally pressurized.

Pressurizer relief valve piping is stainless steel and designed in accordance with ASME Code - Section III Class 1 criteria upstream of the power operated valves and designed to ASME Code Section III Class 3 criteria downstream of these valves.

3.6A.2.4.2 Pipe Whip and Jet Impingement Analysis

The high energy portion of pressurizer relief valve piping is within the pressurizer cubicle. The effects of pipe whip caused by postulated failure of this piping will not compromise any essential equipment.

3.6A.3 SAFETY INJECTION SYSTEM

3.6A.3.1 General Description

The Safety Injection System (SIS) is described in Section 6.3. The process and instrument diagram for the Safety Injection System is shown in Figures 6.3-1a, 6.3-1b, and 6.3-1c. Isometrics of cold leg and hot leg injection lines are shown in Figures 3.6C-3.1 through 3.6C-3.15.

The results of pipe rupture analysis for each of the four cold leg injection lines and the two hot leg injection lines inside containment are presented in this section.

The safety injection tanks (SIT) are normally pressurized between 540-570 psig at 120 F. This pressure is seen by SIT discharge piping up to check valves V3217, V3227, V3237 and V3247 which form the Reactor Coolant Pressure Boundary (RCPB) and up to the check valves V3258, V3259, V3260 and V3261 which are located on the ECCS pumps discharge lines inside of the Reactor Containment Building. These lines have been classified as high energy.

Therefore, the terminal ends of the high energy portion of each cold leg injection line inside containment are the safety injection tank outlet nozzle, the reactor coolant system inlet nozzle and the ECCS pump discharge line check valve (valve included). The remainder of the cold leg injection lines located upstream of check valves V3258 through V3261 inside and outside containment are considered as moderate energy, since they are only pressurized during the shutdown cooling mode of normal plant operation. (See Figures 3.6C-3.3 through 3.6C-3.15.)

The portion of hot leg injection piping from the shutdown cooling suction line to the first check valve (V3525 & V3527) is pressurized by the RCS and is classified as high energy. The remainder of the hot leg injection piping inside and outside containment is considered as moderate energy since it is not pressurized during normal operation. For the purpose of pipe rupture analysis, terminal ends are defined at the hot leg injection nozzle on the shutdown cooling suction line and the upstream end of the check valve.

Each of the four cold leg injection lines between the safety injection tanks and the RCS nozzles consists of 12 inch schedules 40S and 160 piping. The branch pipe from this 12 inch line to check valves V3258 through V3261 is 6 inch Schedule 160. All of this piping is stainless steel seismic Category I and has been designed in accordance with ASME Code Section III Class 1 or 2 criteria.

The high energy portion of the hot leg injection lines, described above, is 3 inch nominal size, schedule 160 stainless steel pipe. This portion of the hot leg injection lines is designed in accordance with Seismic Category I and ASME Code Section III Class 1 criteria.

3.6A.3.2 Pipe Whip Analysis

The pipe whip analysis for the cold leg and hot leg injection lines inside containment is based on the break anywhere criterion. For all pipes, double ended guillotine (circumferential) breaks are considered when providing pipe whip protection. Slot breaks are considered to occur in piping 4 inch NPS and greater. The pipe whip restraints provided for SIS piping inside containment are identified in Figures 3.6C-3.1 through 3.6C-3.15.

Based on break anywhere criteria, the pipe whip restraint locations on SIS piping are selected such that the essential systems and components required for plant shutdown and mitigating the accident are protected from the pipe whip. Pipe whip restraints are also provided to accomplish the following:

a) minimize pipe whip and confine jet effects

b) protect component cooling water piping which supplies containment fan coolers.

c) prevent pipe whip on redundant safety injection lines of smaller nominal pipe size and thinner wall

d) protect cold leg injection lines for shutdown cooling function by isolating the safety injection tank discharge line in the event of its failure

The restraints on the SIS piping are designed to support the pipe whip load in the radial direction all around the circumference of the pipe.

3.6A.3.3 Jet Impingement Analysis

The essential components and systems located inside containment and required for safe shutdown of the plant in the event of Safety Injection System failure inside containment are evaluated for the effects of jet impingement. For the jet impingement analysis, discrete breaks are selected on the basis of stress criteria and terminal ends per Subsection 3.6.2.2.1.

The jet impingement forces of SIS breaks on the concrete structures such as primary and secondary shield walls, floor trenches, slab at El. 62.0 feet, and the floor at El. 23.0 feet, were calculated. Analysis indicates that the integrity of these structures will be maintained under jet loading. The SIS jets also impinge on HVAC duct seismic restraints located on non-essential ducts.

The jets impinge on cable trays and cable tray restraints containing cables which serve non-essential equipment. The jets impinge on non-essential elements such as platforms, structural steel, instrument racks and instrument impulse lines.

The jets impinge on Safety Injection, Shutdown Cooling, Hot Leg Injection, Reactor Coolant System, Containment Spray, Letdown, Charging, Reactor Cooling Pump Cooling, Surge and Spray lines. These interactions are acceptable because these lines are either not required for safe shutdown of the plant or are of equal or greater nominal pipe size and equal or heavier wall thickness than the broken line. Jet impingement loads on component cooling water lines which serve the containment fan coolers and are required for safe plant shutdown have been evaluated for breaks R-SI-4-1, R-SI-4-2, R-SI-7-4, R-SI-8-1, R-SI-7-3. The jet loads were found to be insufficient to affect the component cooling water lines.

3.6A.3.4 Environmental Analysis

The safety related systems required to mitigate the consequences of Safety Injection System line break inside containment are designed to function under environmental conditions resulting from safety injection line break. The resulting environmental conditions are bounded by the conditions of a main steam line / feedwater line break or reactor coolant system line break. Refer to sections 3.6A.1.4 and 3.6A.2.1.4.

3.6A.4 SHUTDOWN COOLING PIPING

3.6A.4.1 General Description

The high energy portion of shutdown cooling (SDC) lines are considered for pipe rupture analysis. These lines commence at the hot legs and terminate at valves V3480 and V3652 (see Figures 3.6C-3.1 through 3.6C-3.2).

The high energy portion of shutdown cooling lines is fabricated of austenitic stainless steel and are designed to ASME. Code Section III Class 1 seismic Category I criteria. The lines consists of 10 inch and 12 inch nominal diameter schedule 160 piping.

The Shutdown Cooling System is described in Subsection 6.3. The process and instrument diagram for SDC is shown in Figures 6.3-1b and 5.1-3.

3.6A.4.2 Pipe Whip Analysis

Isometric drawings of the SDC lines inside containment indicating the locations of pipe whip restraints are provided in Figures 3.6C-3.1 and 3.6C-3.2. Failures are postulated at both terminal ends of each SDC line: the reactor coolant loop nozzle and valves V3480 and V3652 (upstream of valve). Intermediate pipe breaks are postulated to occur at all location~ the shutdown cooling lines. For all pipes double ended guillotine (circumferential) breaks and longitudinal breaks are considered for this analysis. The pipe whip restraints are designed to support pipe rupture thrust loads due to guillotine and longitudinal breaks. In addition, the restraints are designed to accomplish the following:

a) confine pipe whip and fluid jet envelope.

b) protect component cooling water piping which serves containment fan coolers.

c) protect containment sump strainers.

3.6A.4.3 Jet Impingement Analysis

The essential components and systems located inside containment and required for safe shutdown of the plant in the event of shutdown cooling system break inside containment are evaluated for the effects of jet impingement. The results of this analysis are presented here. Discrete break locations are based on criteria identified in Subsection 3.6.2.2.1.

The jets impinge on the concrete structures such as the floor at El. 18.0 feet, slab at El. 62.0 feet, reactor cavity wall, refueling cavity wall, biological and secondary shield walls. Calculations indicate that integrity of these structures will be maintained.

The jets are also directed on pipe whip restraints which are non-essential for the break. The jets impinge on non-essential elements such as platforms, HVAC ducts, equipment supports and trench covers. The jets also impinges on instrument racks, instrument impulse lines, electrical conduits and cable trays containing cables directed to non-essential equipment.

The jets impinge on safety injection shutdown cooling hot leg injection, containment spray and pressurizer surge lines which are either not required for safe shutdown of the plant or have equal or greater nominal pipe size or equal or heavier wall thickness than the broken line and therefore these lines will not rupture.

3.6A.4.4 Environmental Analysis

The safety related systems required to mitigate the consequences of a shutdown cooling line break inside containment are designed to perform their safety Function under environmental conditions resulting from SDC line break. The resulting environmental conditions are bounded by the conditions of a main steam line / feedwater line break or reactor coolant system line break. Refer to sections 3.6A.1.4 and 3.6A.2.1.4.

3.6A.5 CHEMICAL AND VOLUME CONTROL SYSTEM INSIDE CONTAINMENT (LETDOWN / AND CHARGING LINE)

3.6A.5.1 General Description

The Chemical and Volume Control System (CVCS) operation and design bases are described in Subsection 9.3.4. The flow diagram for the CVCS is shown in Figure 9.3-3.

The letdown piping inside containment from the Reactor Coolant loop to the containment penetration is 2 inch stainless steel, schedule 160. The portion of the piping from the reactor coolant loop to the second isolation valve (V2516) upstream of the regenerative heat exchanger is designed in accordance with the ASME Code, Class 1 and Seismic Category I criteria. The remainder of the line to the containment penetration is Class 2 piping. The maximum operating conditions are 2235 psig and 600°F.

The charging piping inside containment from the containment penetration to the reactor coolant loop is 2 inch stainless steel, schedule 160. The portions of the two charging lines and the auxiliary spray line up to first isolation valve in each line downstream of the Regenerative Heat Exchanger are designed in accordance with the ASME Code, Section III, Class 2 and seismic Category I. The remainder of this line to the reactor cooling loop is Class 1 piping. The maximum operating conditions are 2377 psig and 650°F.

3.6A.5.2 Pipe Whip Analysis

Isometric drawings 3.6C-4.1 through 3.6C-4.5 and 3.6C-4.10 through 3.6C-4.17 are provided to show break locations and pipe whip restraints. All breaks in the letdown and charging lines are guillotine (circumferential) type. Pipe whip analysis is based on "break anywhere" criteria.

Pipe whip restraints are provided as required:

a) protect the charging control valves.

b) confine pipe whip and fluid jet envelope.

c) maintain Reactor Coolant System integrity.

No pipe whip restraints are provided in the regenerative heat exchanger room, because no essential equipment is located in cubicle.

3.6A.5.3 Jet Impingement Analysis

The essential components and systems located inside containment and required for safe shutdown of the plant in the event of Chemical and Volume Control System failure (Charging and Letdown lines) inside of containment are evaluated for the effects of jet impingement. For jet impingement analysis, breaks are selected on the basis of stress criteria and terminal ends.

The CVCS system jets impinge on concrete structures such as secondary shield wall, floors at El. 18 feet, 41 feet and trench bottom at El. 12 feet. Calculation of jet impingement loads indicate that these structures will maintain their integrity after impact.

The jets also impinge on the electrical conduits containing cables which serve non-essential equipment. The jets also impinge on cable tray restraints, grating at El. 23 feet, and pipe whip restraints which are non-essential. Calculation of jet impingement loads from breaks R-CH-2-2, R-CH-4-1 and R-CH-1-3 on Regenerative Heat Exchanger supports indicated that the supports will function after impact.

The jets are directed on HVAC ducts which are a branch of the ring header and are not required for safe shutdown of the plant.

The jets impinge on Safety Injection, Hot leg Injection, Containment Spray, Auxiliary Spray, Blowdown, Charging and Letdown lines. These lines have equal or greater nominal pipe size or equal or heavier wall thickness than the broken line and therefore will not be affected by jet impingement.

3.6A.5.4 Environmental Analysis

The essential systems required to mitigate the consequences of Chemical and Volume Control System line failures inside containment are designed to perform their safety function under environmental conditions resulting from CVCS line failure. The resulting environmental conditions are bounded by the conditions of a main steam line / feedwater line break or reactor coolant system line break. Refer to sections 3.6A.1.4 and 3.6A.2.1.4.

3.6A.6 STEAM GENERATOR BLOWDOWN SYSTEM INSIDE CONTAINMENT

3.6A.6.1 General Description

The Steam Generator Blowdown System (SGBS) is described in Subsection 10.4.8. The SGBS lines inside containment commencing at the Steam Generators and terminating at the containment penetration assembly are carbon steel ASME SA-106, GR B, 2 inch diameter piping.

The isometrics of the SGBS piping inside containment are shown in Figures 3.6C-5.1 and 3.6C-5.3 through 3.6C-5.7.

3.6A.6.2 Pipe Whip Analysis

Figures 3.6C-5.1 and 3.6C-5.3 through 3.6C-5.7 are provided to show break locations and pipe whip restraints. The breaks in the SGBS are double ended guillotine (circumferential) type. Pipe whip analysis was originally based on break anywhere criteria. PC/M 05130 (Reference 20) which was performed in support of the Unit 2 Steam Generator Replacement, re-evaluated the blowdown piping inside containment using the criteria of NRC Generic Letter 87-11 (Reference 17) and the Standard Review Plan (SRP), Section 3.6.2 (Reference 18) to eliminate pipe whip restraints RE-B-2 & RE-B-3 for SG 2A and RE-B-30 for SG 2B. Terminal end breaks are postulated at the steam generator nozzles and at the containment penetrations. Pipe whip restraints are provided as required:

a) To protect electrical trays in containment area.

b) To protect containment vessel.

3.6A.6.3 Jet Impingement Analysis

The essential components and systems located inside containment and required for safe shutdown of the plant in the event of Steam Generator Blowdown System break inside containment are evaluated for the effects of jet impingement. For the jet impingement analysis, breaks are selected on the basis of stress terminal ends.

The SGBS jets impinge on concrete structures such as primary and secondary shield walls. Calculation of jet impingement loads indicates that these targets will function after impact.

The jets impinge on electrical equipment and elements which are non-essential. The jets also impinge on structural elements such as HVAC duct restraints, cable tray restraints, platforms, equipment supports and pipe whip restraints which are not required for safe shutdown of the plant.

The jets are also directed on HVAC ducts which are a branch of the ring header and they are not required for safe plant shutdown.

3.6A.6.4 Environmental Analysis

The essential systems required to mitigate the consequences of Steam Generator Blowdown System line break inside containment are designed to perform their safety function under environmental conditions resulting from a Steam Generator Blowdown line break. The resulting environmental conditions are bounded by the conditions of a main steam line / feedwater line break or reactor coolant system line break. Refer to sections 3.6A.1.4 and 3.6A.2.1.4.

APPENDIX 3.6B

HIGH ENERGY PIPE RUPTURE ANALYSIS - OUTSIDE CONTAINMENT

3.6B HIGH ENERGY PIPE RUPTURE ANALYSIS - OUTSIDE CONTAINMENT

The results of pipe break analyses contained in this Section are not impacted by EPU conditions. The high energy piping systems (i.e., fluid systems which exceed 200°F and/or 275 psig during normal operating conditions) that are considered for pipe rupture analysis outside containment are:

a) Main Steam (MS) and Feedwater (BF)

b) Chemical and Volume Control System (Charging and Letdown)

c) Steam Generator Blowdown System (SGBS)

d) Auxiliary Steam System (ASS)

e) Auxiliary Feedwater System (AFW)

f) Steam Supply to Auxiliary Feedwater Pump

The criteria used to locate the break points for high energy piping outside containment are described in Subsection 3.6.2.2.2. The various protective methods used to mitigate the consequences of the postulated pipe failure are given in Subsection 3.6.1.2.3. This appendix presents the results of the pipe rupture analysis, which includes pipe break locations, evaluation of the consequences of pipe whip and protection against pipe whip, protection against jet impingement, and environmental effects.

3.6B.1 MAIN STEAM AND FEEDWATER OUTSIDE CONTAINMENT

3.6B.1.1 General Description

Pipe failures are considered in each of the two Main Steam (MS) lines commencing at the flued head anchor and terminating at the turbine stop valves. Similarly, the two Feedwater (BF) lines between the containment penetration flued head anchors and Feedwater Pumps are considered for pipe rupture analysis. The Main Steam and Feedwater Systems are discussed in Section 10.3 and Subsection 10.4.7.

Feedwater lines are carbon steel ASTM A-106 Gr B piping and Main Steam lines are carbon steel ASTM A-155 Gr KC-65 piping. These lines between the penetration anchors and outboard containment isolation valves are designed in accordance with the ASME Code Section III Class 2 and seismic Category I criteria. The lines past the containment isolation valves are classified as ANSI B31.1. The lines past the containment isolation valves are also analyzed to seismic Category I criteria.

The maximum operating pressure and temperature for the Main Steam System is 888 psia and 530F. The maximum operating pressure and temperature for the Feedwater system is 1108 psig and 440 F upstream of Flow Control Valve and 1700 psig and 440°F downstream of Flow Control Valve.

The Main Steam and Feedwater lines are routed from the reactor containment building to the turbine building via two seismic Category I trestles (each trestle supports one Main Steam line and its corresponding Feedwater line).

Steam trestles are partially enclosed with large openings which permit flow to/from ambient surroundings.

The Main Steam lines are separated by approximately 15 feet as they emerge from the reactor containment building and diverge such that at the Main Steam Isolation Valves (MSIV) the lines are approximately 40 feet apart. There are eight safety relief valves and two atmospheric dump valves on each Main Steam line between the penetration flued head and the MSIV.

Isometrics of MS and FW piping outside containment are shown in Figures 3.6C-1.5 through 3.6C-1.8.

3.6B.1.2 Pipe Whip Analysis

The pipe whip analysis for the MS and Feedwater systems outside containment is based on "break anywhere" criteria. The "break anywhere" criteria is not used for the jet impingement analysis. Discrete breaks are selected on the basis of stress criteria and terminal ends for the jet impingement analysis.

Longitudinal (slot) and circumferential (guillotine) types of breaks are considered for pipe whip protection.

The design of the trestle structure provides restraint for the MS and BF piping. The local equipment which requires protection from the dynamic effects of pipe break outside containment area:

a) redundant Main Steam and Feedwater Systems

b) all Auxiliary Feedwater components located under the trestles.

The restraints on the trestle structure provide protection from effects of Main Steam and Feedwater pipe whip for the above essential systems. The Main Steam and Feedwater Systems are also protected by separation from their redundant train. Additional restraints are provided on the non-seismic section of each Main Steam and Feedwater line downstream of the containment isolation valves. These pipe whip restraints assure that rupture in the non-seismic sections of pipe will not affect the safety related sections of Main Steam and Feedwater piping. Since there are no safety related equipment in the turbine building, protective measures against pipe whip are not provided.

The pipe whip restraints are designed to withstand pipe rupture loads due to guillotine and slot breaks. Since it is assumed that the slot break can occur at any location around the circumference of the pipe, the pipe whip restraints are designed to withstand loads in any radial direction of the pipe.

The design of those restraints is based on the static load method. To verify the structural adequacy of pipe whip restraints which are designed using static load method, dynamic analysis was performed for several typical restraints on MS and FW systems outside containment and trestle structure. The computer programs RELAP-3 and PLAST (References 5 and 6 in Section 3.6) were used for this analysis. A detailed description of the dynamic analysis and the results of the analysis are presented in Appendix 3.6E.

3.6B.1.3 Jet Impingement Analysis

The essential components and systems located outside containment (Trestle Area) required for safe plant shutdown in the event of MS or BF pipe break outside containment are evaluated for the effects of jet impingement.

The MS and BF jets impinge on the concrete containment. The analysis indicated that the integrity of the containment will be maintained. The jets from breaks T-BF-9-1, T-BF-8-1, T-MS-1-3, T-MS-1-2, T-MS-1-1, T-MS-2-1, T-MS-2-3 impinge on pipe whip restraints RE-BF-35, RE-BF-39, RE-MS-28, RE-MS-27, RE-MS-33, and RE-MS-34. The functional loss of these restraints does not jeopardize the safe shutdown of the plant, since these pipe whip restraints are not essential for this pipe break.

The jets from breaks T-BF-2-2 and T-BF-1-1 impinge on condensate lines I-6-C-9 and I-8-C-85 which are required for safe shutdown of the plant. Analysis indicated that these lines will function under these jet loads.

3.6B.1.4 Environmental Analysis

Environmental effects on safety related equipment and components due to Main Steam and Feedwater line breaks outside containment are discussed in Section 3.11.

3.6B.2 CHEMICAL AND VOLUME CONTROL SYSTEM OUTSIDE CONTAINMENT (LETDOWN AND CHARGING LINE)

3.6B.2.1 General Description

The Chemical and Volume Control System (CVCS) is described in Subsection 9.3.4. The flow diagram for the Chemical and Volume System is shown in Figures 9.3-5a, b, and c.

The letdown lines outside containment from the Letdown Heat Exchanger to valves V2350 and V2349 are 1, 2 and 3 inch nominal diameter stainless steel, schedule 40S piping. The maximum operating conditions are 460 psig and 120°F for 1 and 2 inch lines and 24 psig and 120°F for 3 inch diameter piping. The remainder of the line located between containment penetration flued heat and Letdown Heat Exchanger is 2 inch stainless steel, schedule 160 piping. The maximum operating fluid conditions of 2188 psig and 450°F are applicable to piping between containment penetration and the letdown heat exchanger.

The Charging line outside containment from charging pumps to the containment penetration Flued Head is 2 inch stainless steel, schedule 160 piping. The maximum operating fluid conditions are 2377 psig and 120 F. The charging line outside containment is designed in accordance with the ASME code, Section III, Class 2 and seismic Category I criteria.

3.6B.2.2 Pipe Whip Analysis

Isometric drawings, Figures 3.6C-4.6 through 3.6C-4.9, are provided to show break locations and pipe whip restraints for the Charging and Letdown lines outside containment. All breaks in the lines are guillotine type. Pipe whip analysis is based on "break anywhere" criteria. Pipe whip restraints are provided as required:

a) To protect integrity of concrete wall forming the boundary of the charging pump rooms.

b) To assure operability of valves which maintain charging pump discharge manifold pressure integrity.

c) To assure operability of valve V2339 in event of rupture of charging line downstream of valve V2338. Thus allowing charging function using Auxiliary HPSI Header.

d) To confine pipe whip and fluid jet envelope.

Pipe break in the charging pump's discharge common header, line I-2-CH-104 along with a single active component failure was analyzed. Pipe break with a single electrical distribution system failure does not compromise cold plant shutdown capability since pump 2C can be powered by interchanging the electrical buses. The pipe break with a single active mechanical failure of pump 2C could result in a loss of charging capability. In this highly unlikely event, the plant will be maintained under hot-standby condition until charging capability is restored.

Pipe whip restraints are not required on manifold I-2-CH-104 and 109 based on the following:

1) Charging can continue in the event of manifold rupture through Auxiliary HPSI Header. Restraints RE-CH-19 and 20 assure this function.

2) Only equipment related to charging function is located in charging pump rooms.

3) Pipe whip of line I-2-CH-104 or 109 into line I-2-SI-141 is not postulated to rupture the safety injection line since these lines are the same size and schedule.

Pipe whip restraints are not provided in the letdown heat exchanger cubicle because no essential equipment is located in the cubicle.

3.6B.2.3 Jet Impingement Analysis

The essential components and systems located outside containment and required for safe plant shutdown in the event of Chemical and Volume Control System failure are evaluated for the effects of jet impingement. Analysis indicate that no essential equipment/component are affected by jet impingement.

3.6B.2.4 Environmental Analysis

Environmental effects on safety related equipment and component due to Chemical and Volume Control System line break outside containment are discussed in UFSAR Section 3.11.

3.6B.3 STEAM GENERATOR BLOWDOWN SYSTEM OUTSIDE CONTAINMENT

3.6B.3.1 General Description

The Steam Generator Blowdown System (SGBS) operation and design bases are described in Subsection 10.4.8.

The SGBS lines outside containment commencing at containment penetration and terminating at check valve V23107 and V23132 are carbon steel ASTM A-106, grade B, 2 and 3 inch diameter piping. This section of piping is designed to seismic Category I criteria. The maximum operating temperature and the design pressure of the system are 532°F and 985 psig. The isometrics of SGBS piping outside containment are show in Figures 3.6C-5.2 through 3.6C-5.8. The line is routed along the roof of the RAB starting from the control room to the SGBTF building.

3.6B.3.2 Pipe Whip Analysis

The pipe whip analysis for SGBS outside containment is based on the "break anywhere" criteria. However, for jet impingement analysis, breaks are chosen based on high stress.

Pipe whip restraints are not required for SGBS lines starting outside the RAB to the SGBTF Building because there are no equipment necessary for safe shutdown of the plant in this area.

3.6B.3.3 Jet Impingement Analysis

The jet impingement analysis for SGBS breaks outside containment indicate that no essential equipment/component are affected. The plant shutdown is not compromised.

3.6B.3.4 Environmental Analysis

The safety-related systems required to mitigate the consequences of Steam Generator Blowdown System line breaks outside containment are designed to perform their safety function under environmental conditions resulting from Steam Generator Blowdown line break.

3.6B.4 AUXILIARY STEAM SYSTEM

The Auxiliary Steam lines are carbon steel ANSI B31.1 non-seismic Category piping.

The flow diagram of the Auxiliary Steam System (ASS) is shown in Figure 10.1-1. ASS starts in the Turbine Building which contains no safety related equipment in the Turbine Building. Portion of ASS (lines 2-AS-2 and 2-1/2-AS-6) are directed to the steam jet air ejectors and to the auxiliary priming ejectors, all located in the Turbine Building. A 12-inch ASS line is routed from the Turbine Building to the Reactor Auxiliary Building (El. 62.00) and to the boric acid concentrator* rooms and the waste concentrator* room (El. 43.00). The line penetrates RAB roof

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* Note: These items are no longer used.

(El. 62.00) in the area adjacent to the concentrator room. Neither of these rooms contain safety related equipment.

The effects of rupture to the high energy ASS lines are not considered because no component or equipment for safe plant shutdown are located in the area. Environmental effects are addressed in Section 3.11.

3.6B.5 AUXILIARY FEEDWATER SYSTEM

3.6B.5.1 General Description

The Auxiliary Feedwater System operation and design bases are described in Section 10.4. The results of pipe rupture analysis for each of the Auxiliary Feedwater System lines are presented below.

The Auxiliary Feedwater System is located under the steam trestle structure. The Auxiliary Feedwater pumps take suction from the condensate storage tank. Pump discharge are routed under the trestle to the main feedwater lines I-20-BF-14 and I-20-BF-19 which feed steam generators 2A and 2B, respectively.

The system consists of three AFW pumps and their associated piping and is designed to provide feedwater for the removal of sensible and decay heat from the Reactor Coolant System and for normal system cool down to below 350°F (design temperature of the LPSI system). Piping from the Auxiliary Feedwater pumps to the branch connections in the main feedwater lines is pressurized to 1115 psig. Each of the pumps is capable of providing sufficient feedwater to bring the plant to below 350°F.

All piping is carbon steel, seismic Category I and has been designed in accordance with the ASME Code Section III Class 3 criteria.

Turbine driven pump 2C supplies auxiliary feedwater to steam generator 2B through lines I-6-BF-32, I-4-BF-35 and I-4-BF-36. Steam generator 2A is supplied by lines I-6-BF-32, I-4-BF-34 and I- 4-BF-33. Valving arrangement allows the entire flow of pump 2C to be directed to either steam generator.

Two motor driven Auxiliary Feedwater pumps, 2A and 2B, are provided. Motor driven pump 2A discharges through lines I-4-BF-28 and I-4-BF-29 into main feedwater line I-20-BF-14 which is directed to steam generator 2A. Motor driven pump 2B discharges to steam generator 2B through Auxiliary Feedwater lines I-4-BF-30 and I-4-BF-31 and main feedwater line I-20-BF-19. A cross tie connection between these lines has been provided to enable the routing of the flow of both pumps to either steam generator.

Isometric drawings of Auxiliary Feedwater lines are shown on Figures 3.6C-6.1 through 3.6C-6.3 for pumps 2A and 2B and on Figures 3.6C-6.4 through 3.6C-6.5 for pump 2C.

3.6B.5.2 Pipe Whip Analysis

The pipe whip analysis for the Auxiliary Feedwater lines outside containment is based on "break anywhere" criteria. Since the lines are 4 inch and 6 inch nominal diameter, a double ended guillotine (circumferential) breaks and slot (longitudinal) breaks are considered for pipe whip protection.

Analysis has shown one pipe whip restraint to be required on the auxiliary feedwater system. It is located to prevent AFW pump 2C discharge line 1-6-BF-32 from possibly impacting AFW pump 2A discharge line I-4-BF-28 due to a slot break in the 6 inch line. In other cases, pipe whip effects are minimal due to separation from other essential piping and equipment or due to impact upon another pipe of equal size and wall thickness.

3.6B.5.3 Jet Impingement Analysis

The essential components and systems required for safe plant shutdown in the event of Auxiliary Feedwater System failure are evaluated for the effects of jet impingement. Postulated piping failure are shown in Figures 3.6C-6.1 through 3.6C-6.5.

The AFW jets impinge on the flood walls. The analysis indicated that the integrity of these walls will be maintained.

The AFW jets impinge on Main Steam, AFW and Condensate System lines. These lines will function after impact because the lines are equal or greater size and thickness than the broken line.

3.6B.5.4 Environmental Analysis

Environmental effects on safety related equipment and components due to Auxiliary Feedwater line break outside containment are discussed in Section 3.11.

3.6B.6 STEAM SUPPLY TO AUXILIARY FEED PUMP TURBINE

3.6B.6.1 General Description

Piping failures are considered in the steam line which supply auxiliary feedwater pump 2C turbine (AFPT). Two steam lines (one from each Main Steam line) commences at the Main Steam lines just upstream of the Main Steam Isolation Valves (MSIVs) and joins a common header which terminates at the inlet to the turbine. This piping is 4 inch Schedule 80 Carbon Steel ASTM A-106 Grade B material. These lines are designed in accordance with ASME Code Section III Class 2 and 3 and seismic Category I criteria. The maximum operating pressure and temperature for this piping for EPU is 866.5 psig and 527.5°F. Isometrics of steam supply piping to the AFPT are shown in Figures 3.6C-7.1 thru 3.6C-7.3.

3.6B.6.2 Pipe Whip Analysis

The pipe whip analysis for the steam supply to AFPT is based on "break anywhere" criteria. Analysis has shown no pipe whip restraints are required on this piping since all piping and components in the area are of equal or greater size and wall thickness compared with these lines. Unrestrained whipping pipes are not considered to rupture lines of equal or greater wall thickness in equal or greater nominal sizes.

3.6B.6.3 Jet Impingement Analysis

The essential components and systems required for safe plant shutdown in the event of failure of steam supply piping to AFPT have been evaluated for the effects of jet impingement. Postulated piping failures are shown in Figures 3.6C-7.1 through 3.6C-7.3. The jet impingement analysis indicated that no essential system or components are affected by these jets.

3.6B.6.4 Environmental Analysis

Environmental conditions resulting from postulated piping failure in steam supply piping to AFPT are bound by the postulated breaks identified in Section 3.6B.1.1.

Environmental effects are addressed in Section 3.6B.1.4.

APPENDIX 3.6C

PIPE WHIP RESTRAINTS AND BREAK LOCATIONS

Note: As described in UFSAR Section 3.6.2.9, the figures contained in Appendix 3.6C were submitted for operating license review. The purpose of these figures is to provide design basis piping break locations. Consequently, these figures provide the design basis piping break locations for the general piping configurations (note: the break locations depicted on Figures 3.6C-2.1 & 3.6C-2.2 are provided for historical purposes since the dynamic effects associated with a hot or cold leg break have been eliminated from the plant design basis based on leak before break methodology). However, ancillary information (i.e., valve numbers, valve fittings, pipe elevations, pipe supports, etc.) shown on these figures may not represent the current as-installed configuration. Details of such ancillary information can be found on plant drawings.

The following break locations and pipe whip restraints depicted on Figures 3.6C-1.3 and 3.6C-1.4 are provided for historical purposes since the dynamic effects associated with the specified pipe breaks and the pipe whip restraints have been eliminated from the plant design basis based upon NRC Generic Letter 87-11 as described in UFSAR Sections 3.6 and 3.6A.1.1:

Figure 3.6C-1.3	Pipe Whip Restraints RE-MS-1 & RE-MS-4 Guillotine Breaks R-MS-2-2 Slot Breaks R-MS-2-1
Figure 3.6C-1.4	Pipe Whip Restraints RE-MS-11 & RE-MS-14 Guillotine Breaks R-MS-1-2 Slot Breaks R-MS-1-1

Surge Line pipe whip restraint RE-RC-30 depicted on Figure 3.6C-2.4 is shown for historical purposes as the pipe whip restraint design basis has been eliminated as addressed in PC/M No. 05131M, Surge Line Rupture Restraints RC-30 & RC-31 Modification for the Unit 2 Component Replacements Projects.

TABLE 3.6C-1

STRESS SUMMARY

SAFETY INJECTION SYSTEM (SCI)
(Historical Information)

Point (1)	Eq 10 Sn/Sm	Eq 9 S/ 1.55	Usage Factor
113	.437	.285	-
114	.202	.267	-
1140	.178	.250	-
115	.533	.314	0
116	.811	.609	-
117	.903	.554	0
118	1.136	.554	.0004
119	.907	.543	0
120	.975	.538	-
121	.882	.511	.03
157	.431	.256	-
158	.353	.243	-
159	-	-	-
160	.370	.215	-
169	.176	.135	-
1210	.586	.271	-
122	1.198	.561	0
123	1.226	.560	-
124	.701	.297	-
125	1.376	.554	0
126	1.395	.540	-
127	.794	.275	-
1270	.823	.287	-
1271	.844	.301	-
128	1.625	.582	-
129	1.498	.560	0
130	.568	.275	-
131	.560	.273	-
132	1.506	.530	-
133	1.638	.521	0
134	1.559	.522	-
135	1.468	.529	0
136	1.043	.522	-
336	.436	.260	-
236	.419	.248	-
237	-	-	-
238	.188	.143	-
239	.176	.135	-
137	.790	.281	-
138	-	-	-
139	1.352	.298	.0024
140 (*)	2.213	.609	-
941	2.307	.674	.0010
8002 (TP)	1.450	.345	-

TABLE 3.6C-1 (Cont'd)

Notes:

(1) Node Points correspond to Figures 3.6C-3.8, 3.9, 3.10

(2) (*) Break location

(3) (TP) Terminal point

TABLE 3.6C-2

STRESS SUMMARY

SAFETY INJECTION SYSTEM (SC2)
(Historical Information)

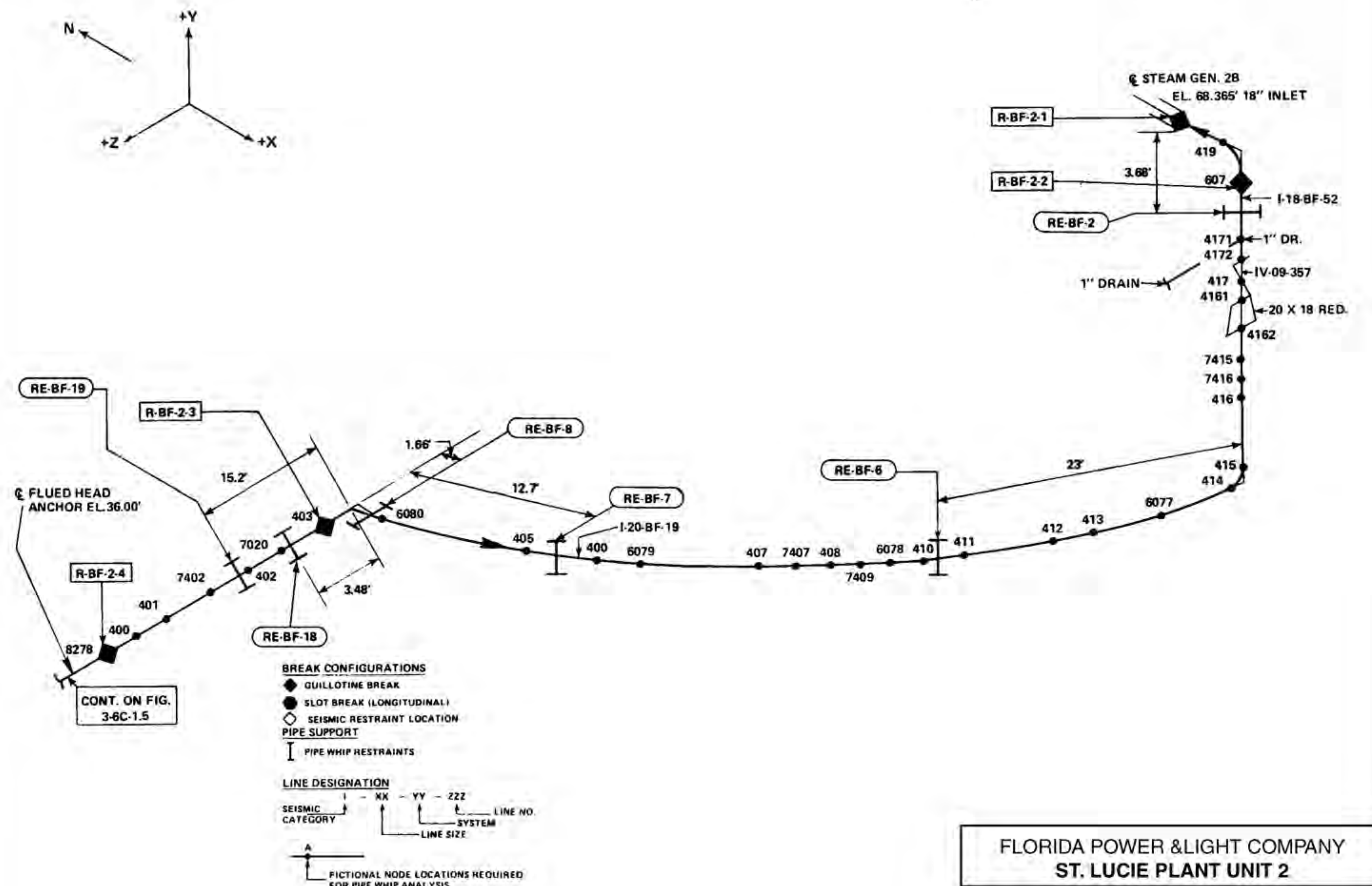
Point Number	Stress Ratio SA	Stress Ratio Sh+SA
8001 (TP)	-	-
100	.424	.266
1000	.335	.147
101	.389	.244
102	.519	.326
103	.355	.233
1030	.417	.262
(104) (*)	.633	.397
105	.402	.252
1050	.099	.062
106	.101	.064
108	.244	.155
109	.278	.177
110	.229	.147
1100	.370	.235
111	.381	.242
112	.336	.214
113	.311	.198
114	.169	.107

Notes:

(1) Node Points correspond to Figures 3.6C-3.8, 3.9, 3.10

(2) (*) Break Location

(3) (TP) Terminal Point



REF DWG: BF-147-4

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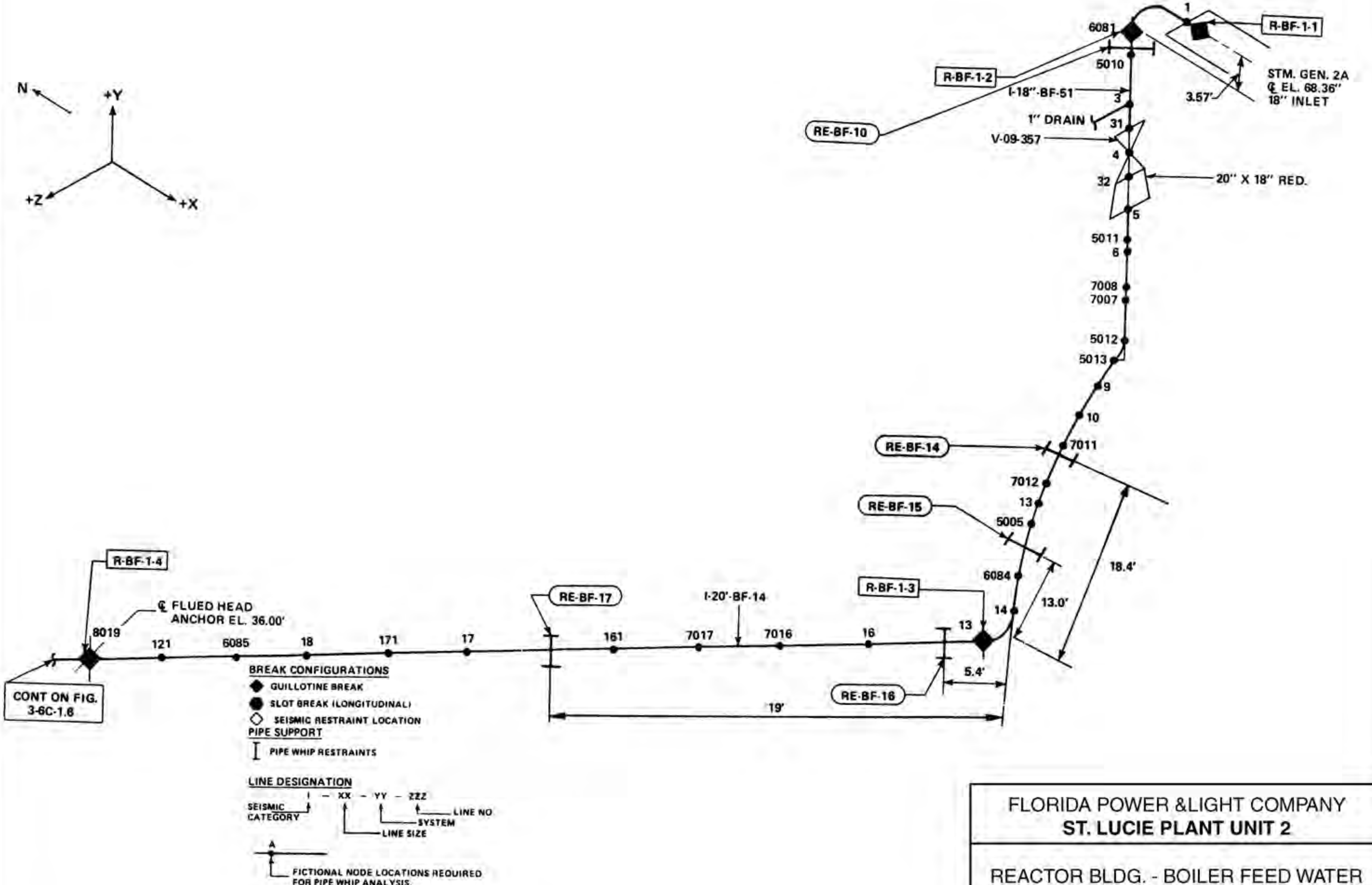
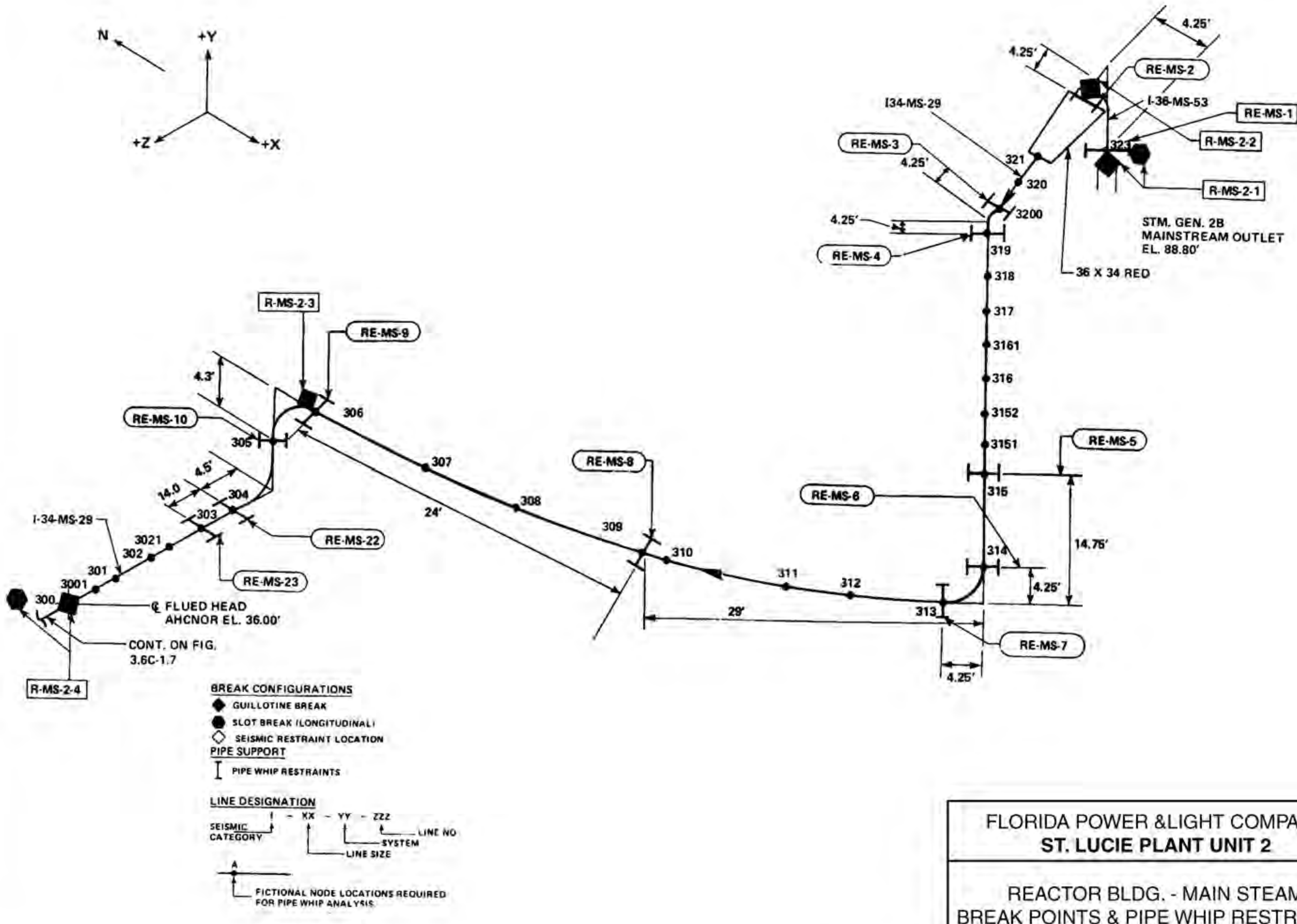
REACTOR BLDG. - BOILER FEED WATER BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-1.1



REF DWG: BF-147-1

FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG. - BOILER FEED WATER
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-1.2



REF DWG: MS-147-2

FLORIDA POWER &LIGHT COMPANY
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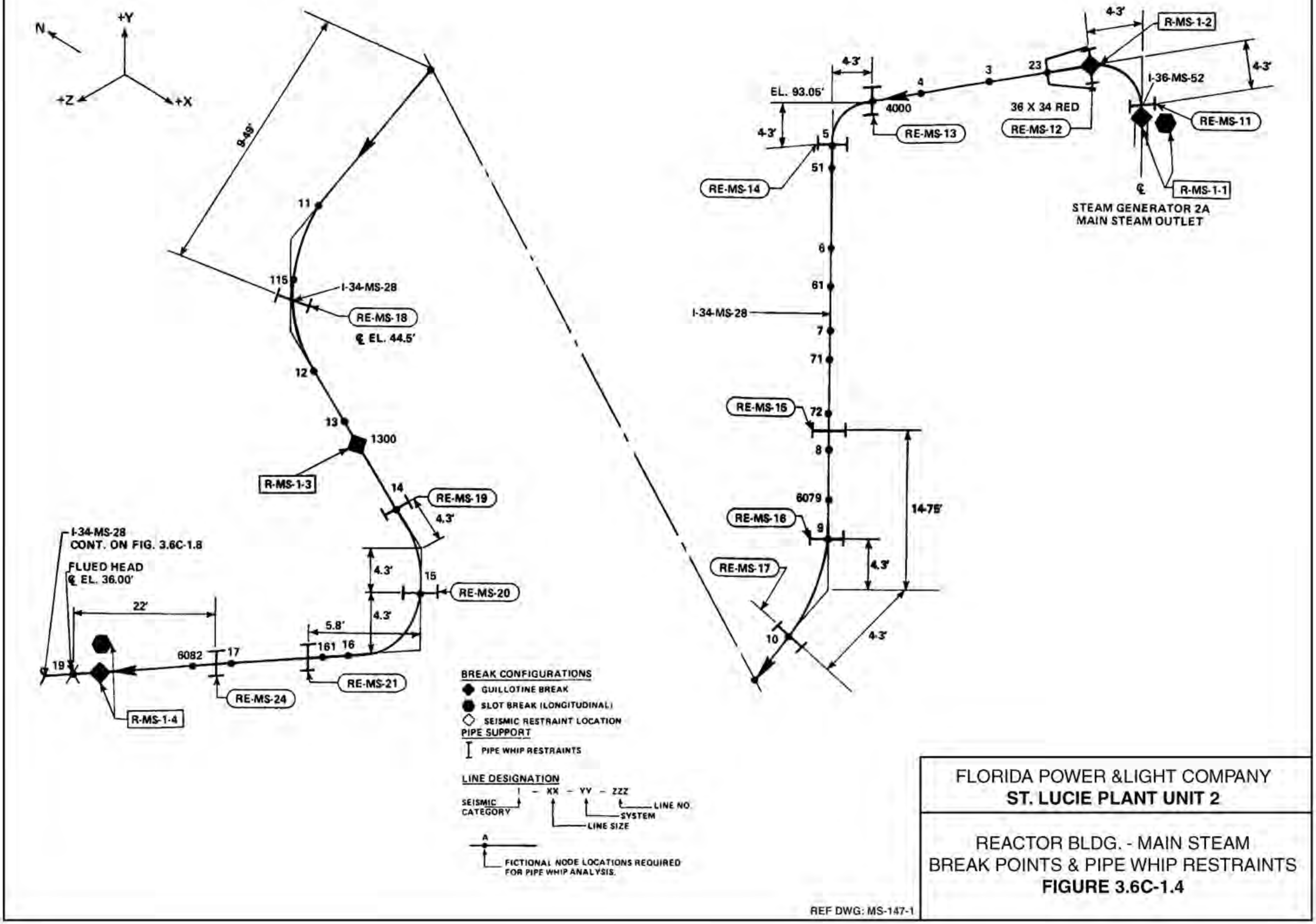
REACTOR BLDG. - MAIN STEAM
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-1.3



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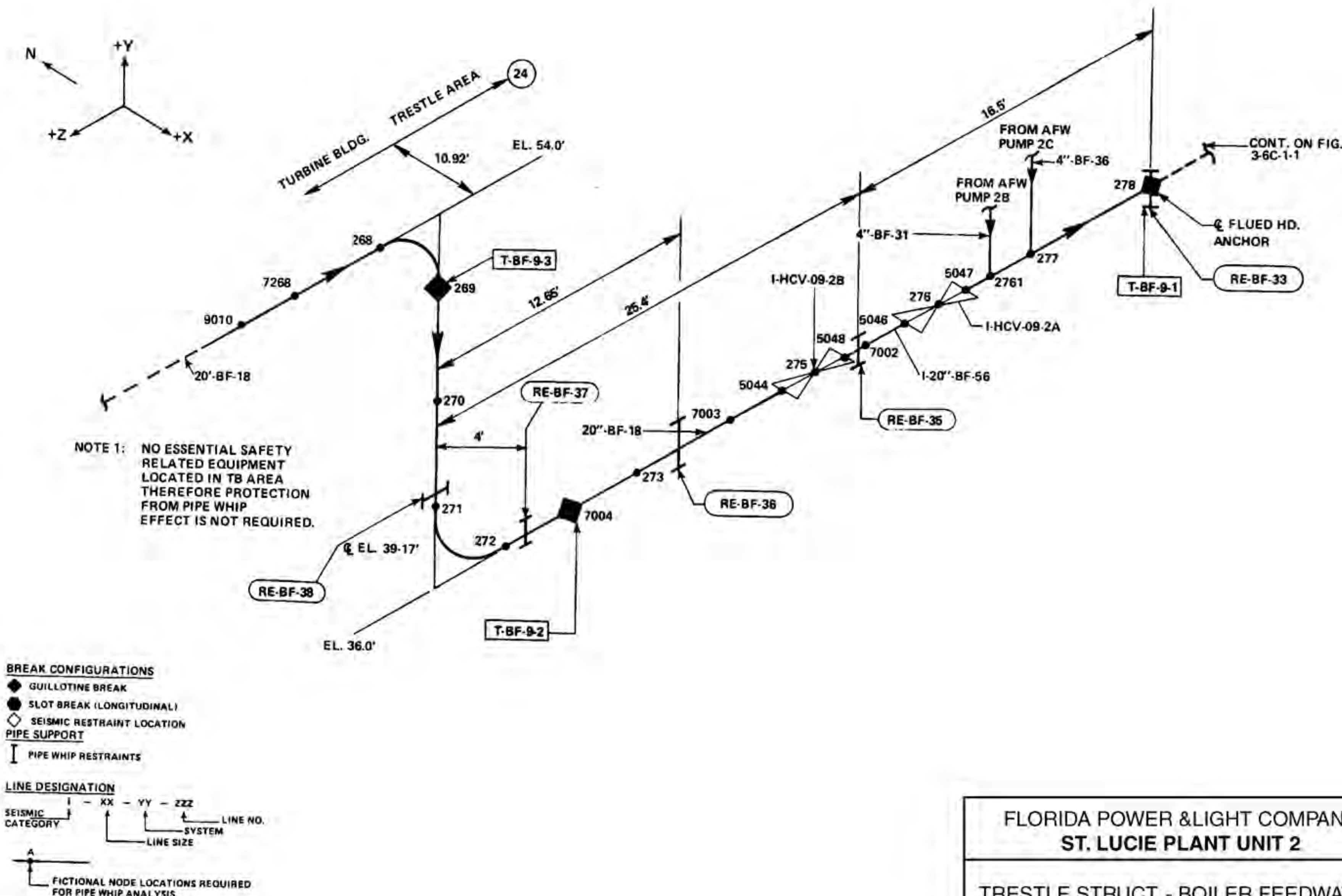
REACTOR BLDG. - MAIN STEAM
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-1.4



REF DWG: BF-149-2

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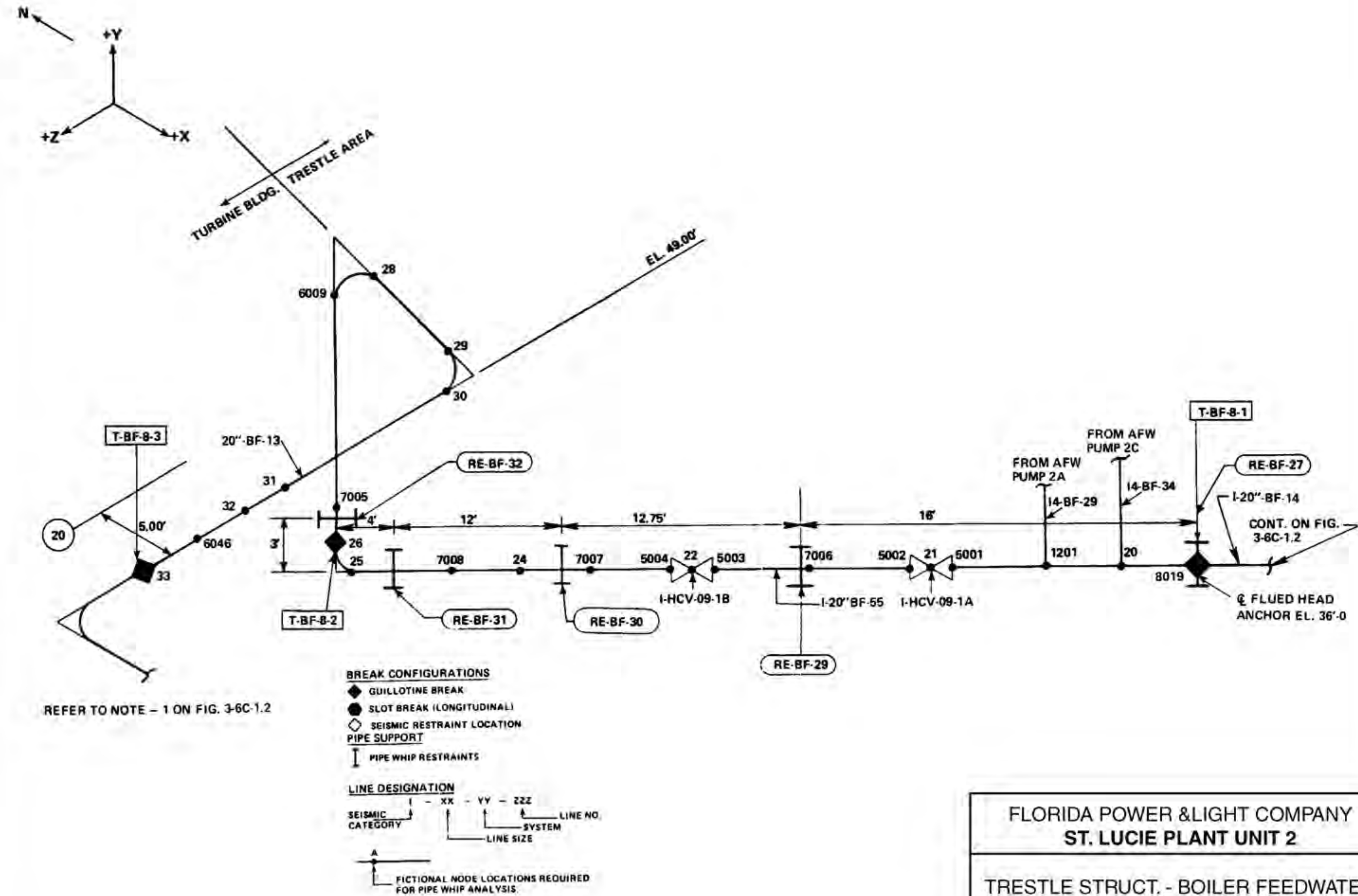
TRESTLE STRUCT. - BOILER FEEDWATER BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-1.5



REF DWG: BF-149-1

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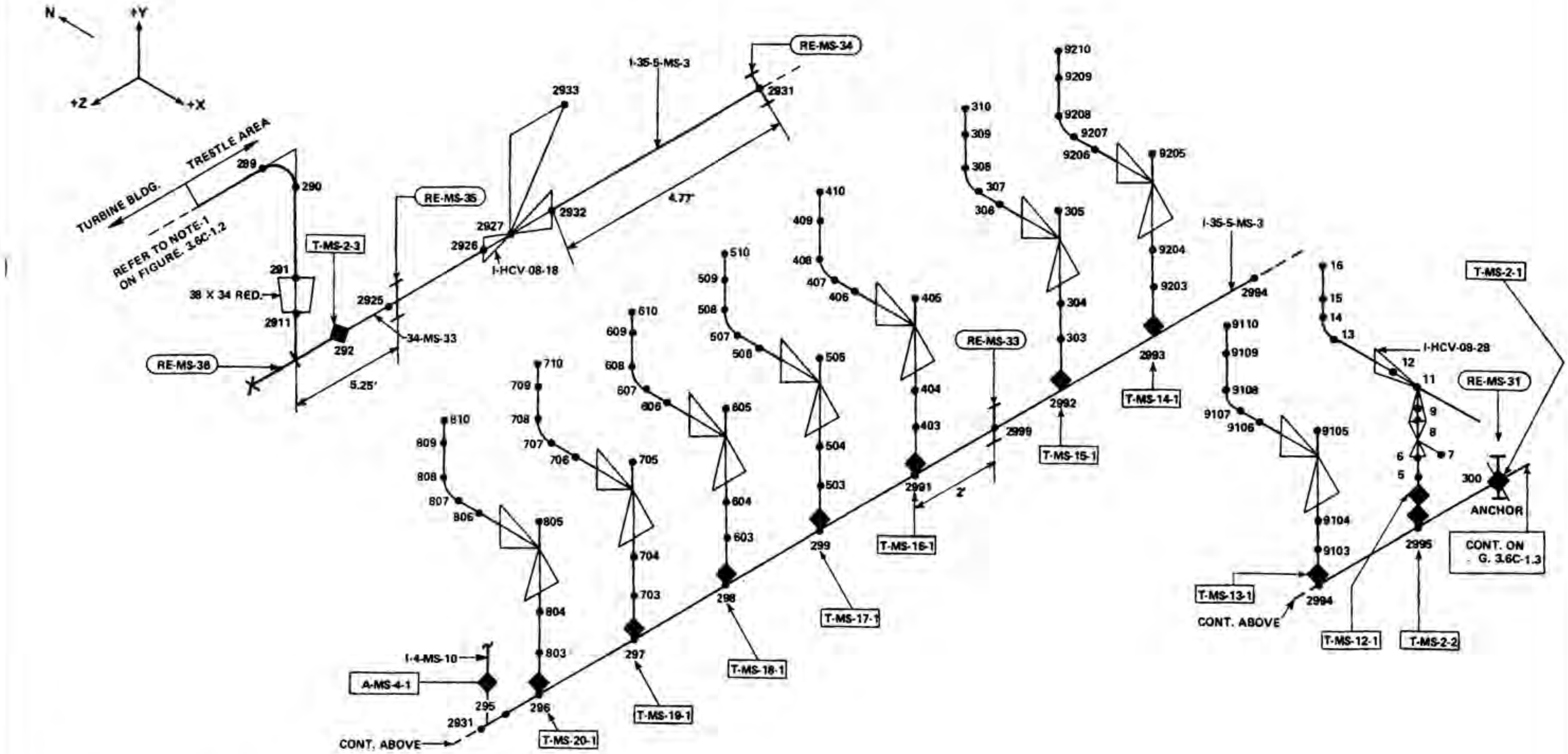
TRESTLE STRUCT. - BOILER FEEDWATER BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-1.6



BREAK CONFIGURATIONS

- GUILLOTINE BREAK
- SLOT BREAK (LONGITUDINAL)
- SEISMIC RESTRAINT LOCATION

PIPE SUPPORT

- PIPE WHIP RESTRAINTS
- PIPE STOP

LINE DESIGNATION

I – XX – YY – ZZZ

SEISMIC CATEGORY — LINE SIZE — SYSTEM — LINE NO.

A — FICTIONAL NODE LOCATIONS REQUIRED FOR PIPE WHIP ANALYSIS.

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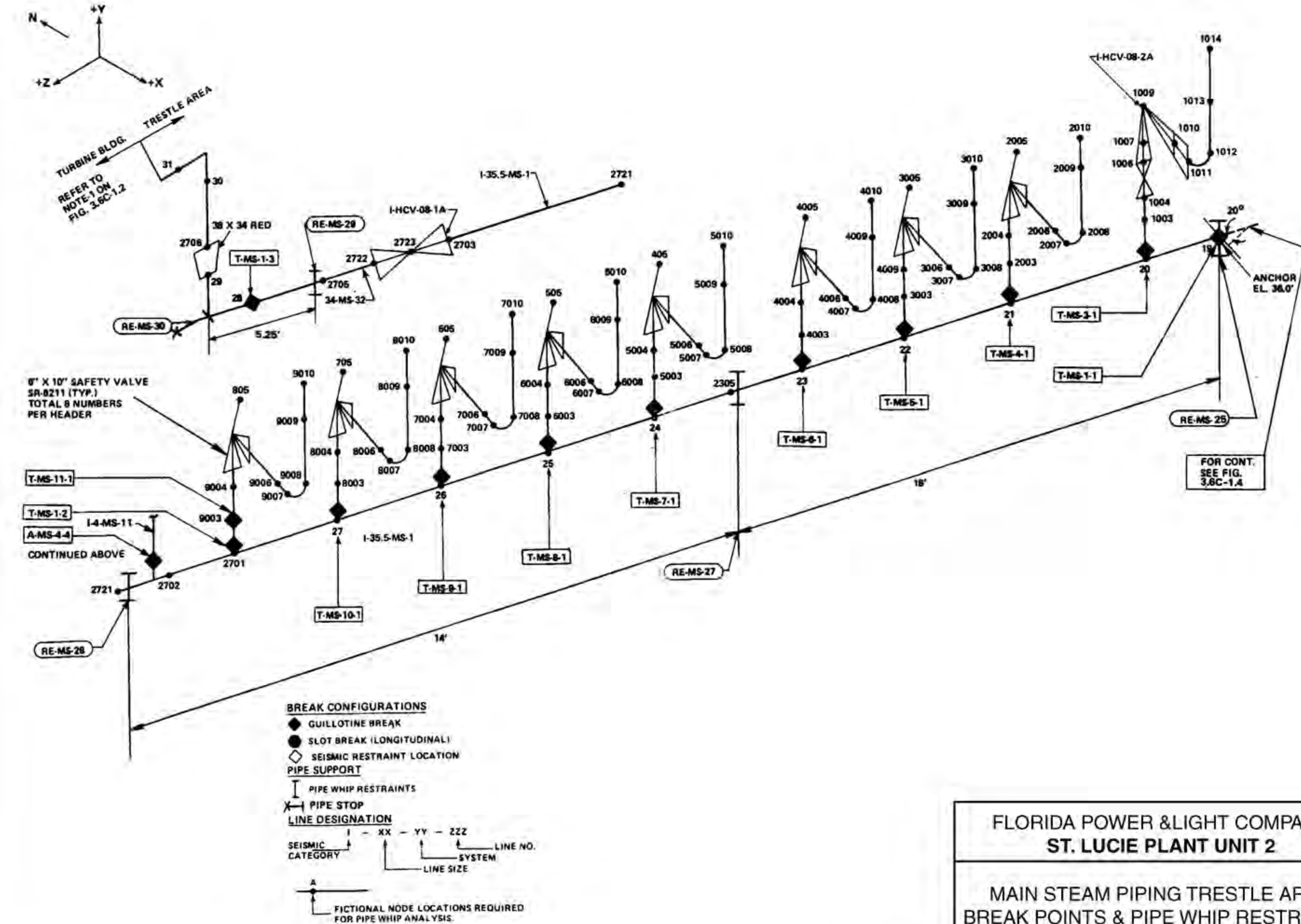
MAIN STEAM PIPING TRESTLE AREA
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-1.7

REF DWG: MS-149-2



REF DWG: BF-149-1

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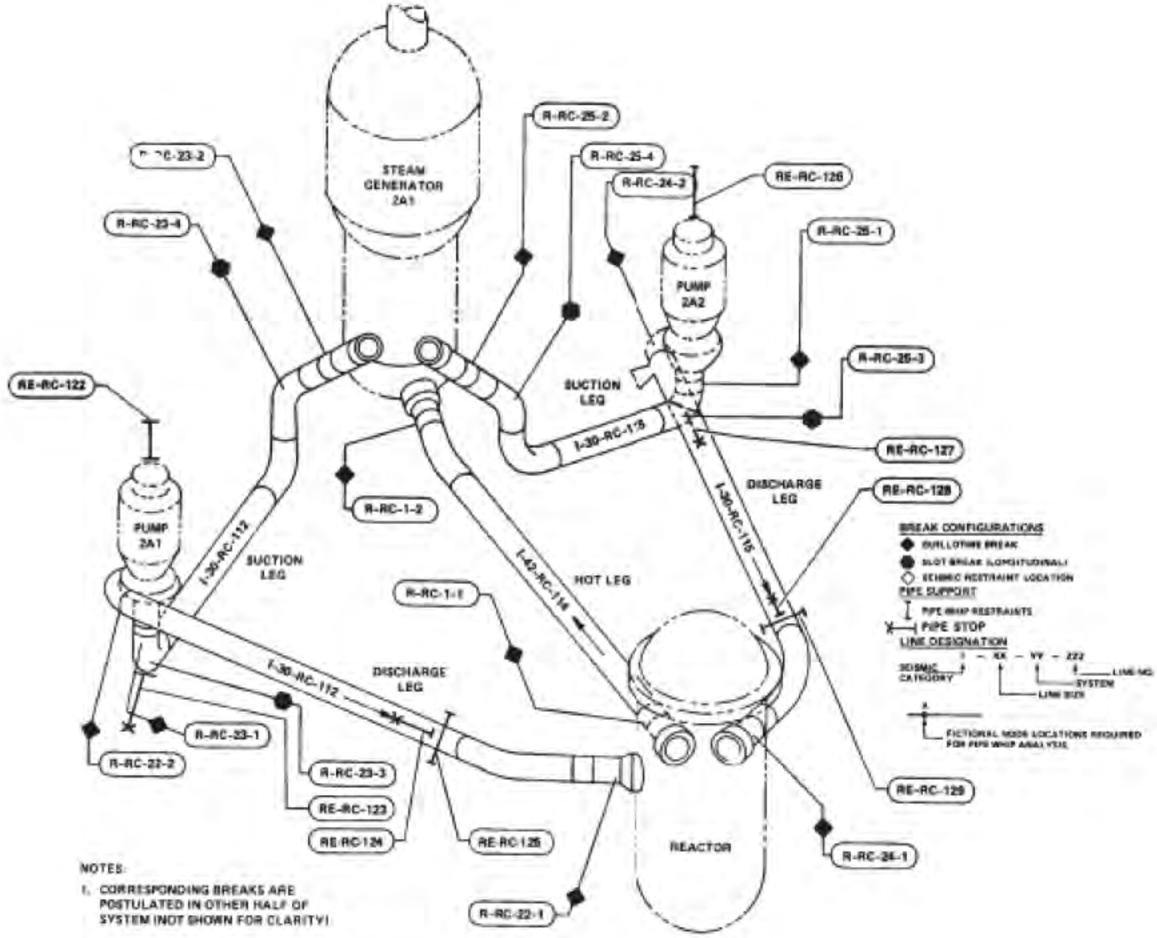
ST. LUCIE PLANT UNIT 2

MAIN STEAM PIPING TRESTLE AREA
BREAK POINTS & PIPE WHIP RESTRAINTS

FIGURE 3.6C-1.8



NOTES:

1. CORRESPONDING BREAKS ARE POSTULATED IN OTHER HALF OF SYSTEM (NOT SHOWN FOR CLARITY)

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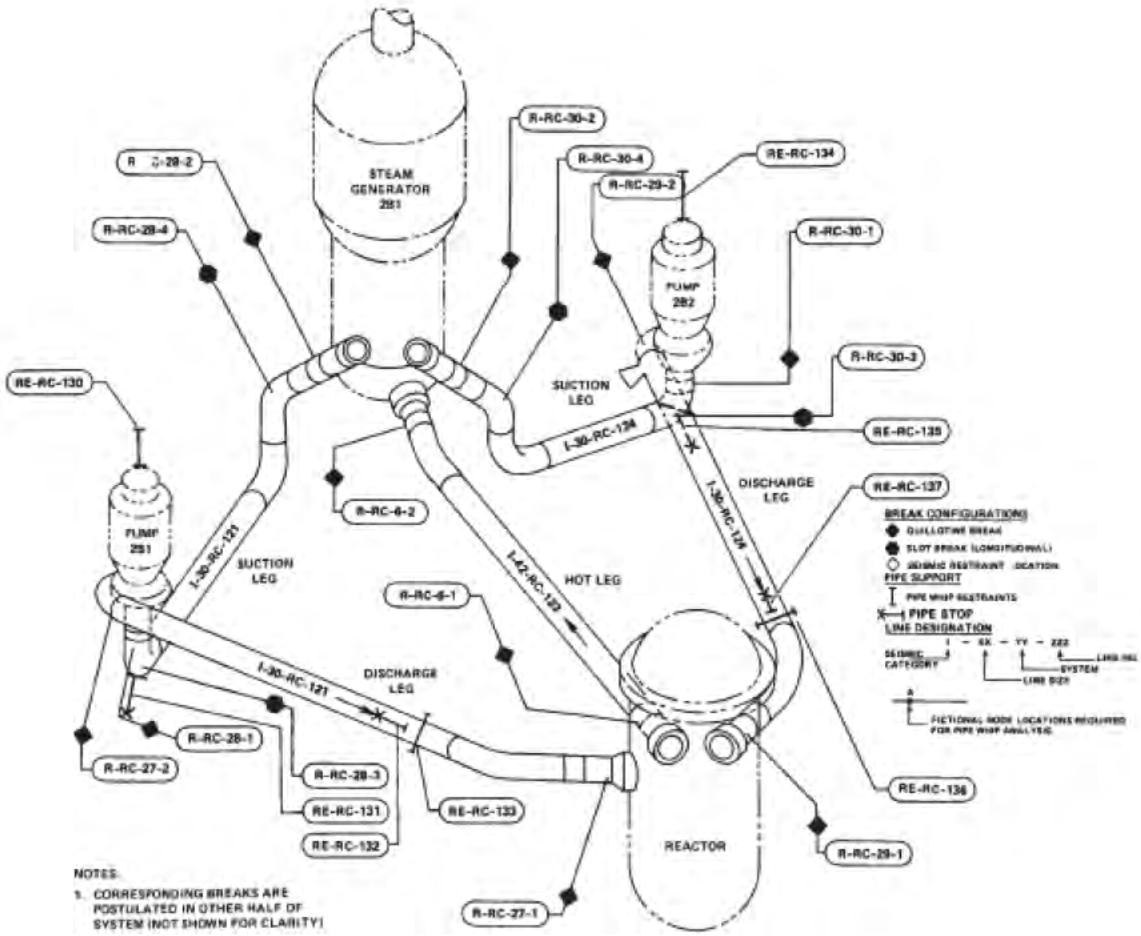
REACTOR COOLANT SYSTEM
DESIGN BASIS PIPE BREAKS
TYPES & LOCATIONS

FIGURE 3.6C-2.1

(HISTORICAL)



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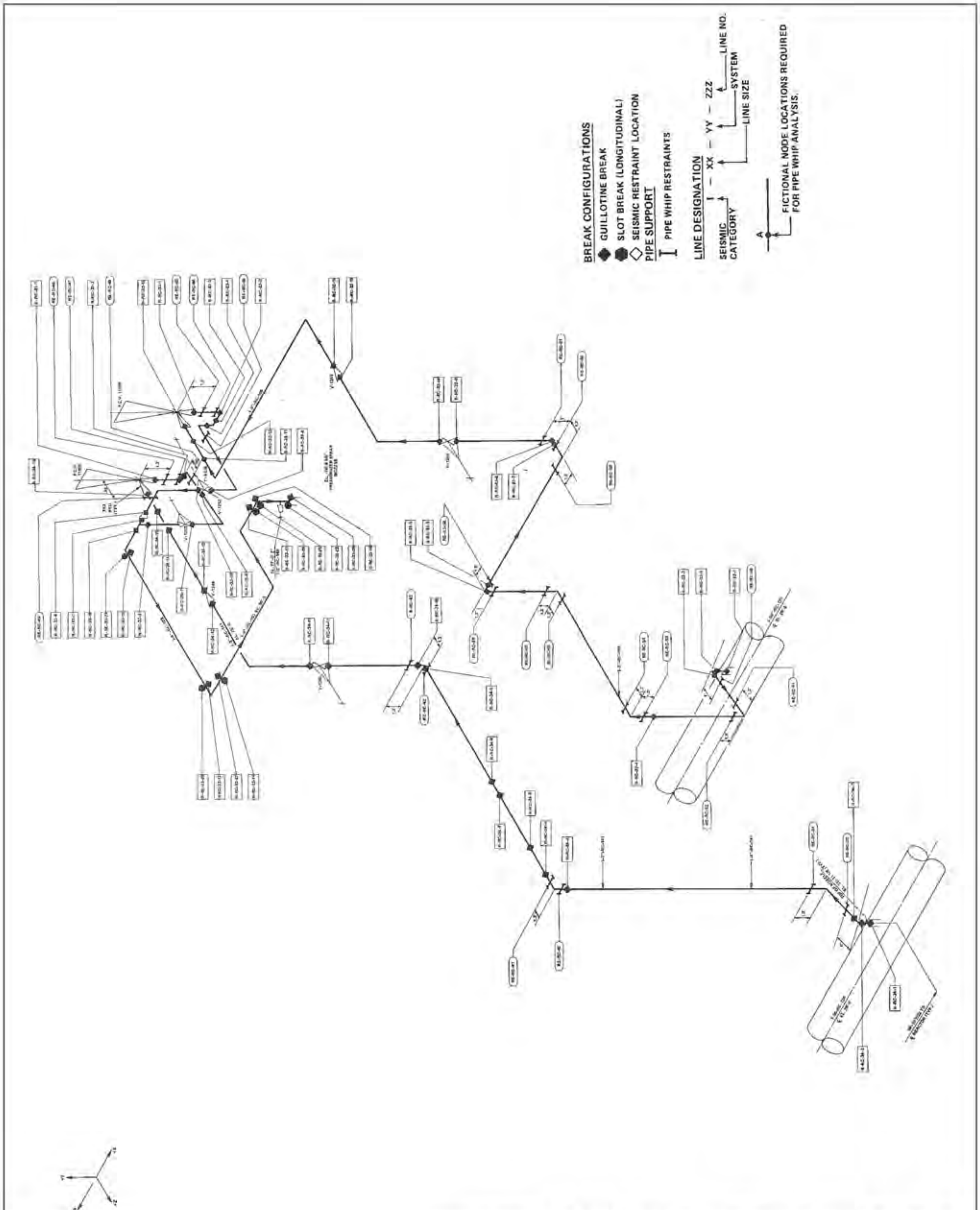
REACTOR COOLANT SYSTEM
DESIGN BASIS PIPE BREAKS
TYPES & LOCATIONS
FIGURE 3.6C-2.2
(HISTORICAL)



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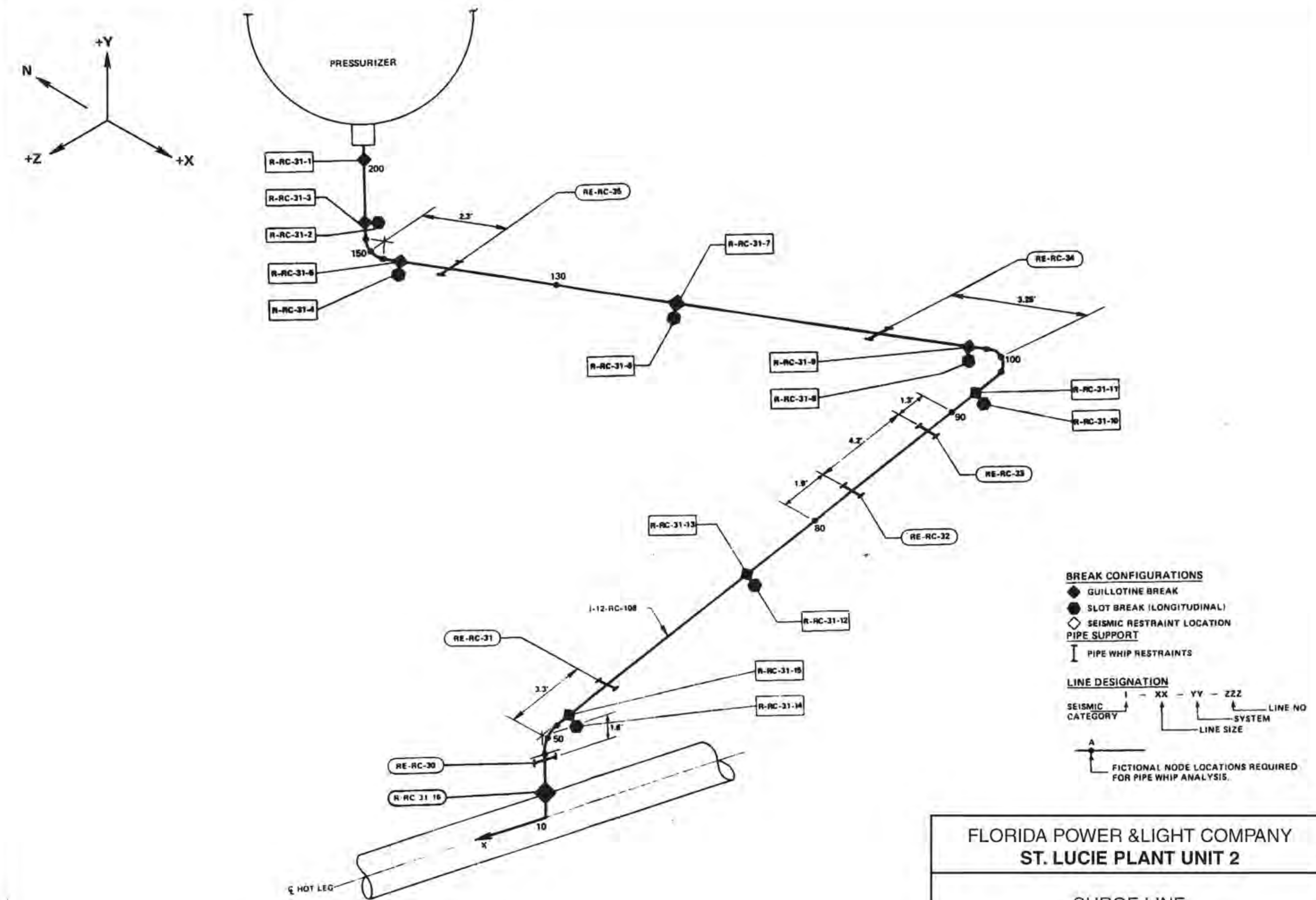
PRESSURIZER SPRAY PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS

FIGURE 3.6C-2.3



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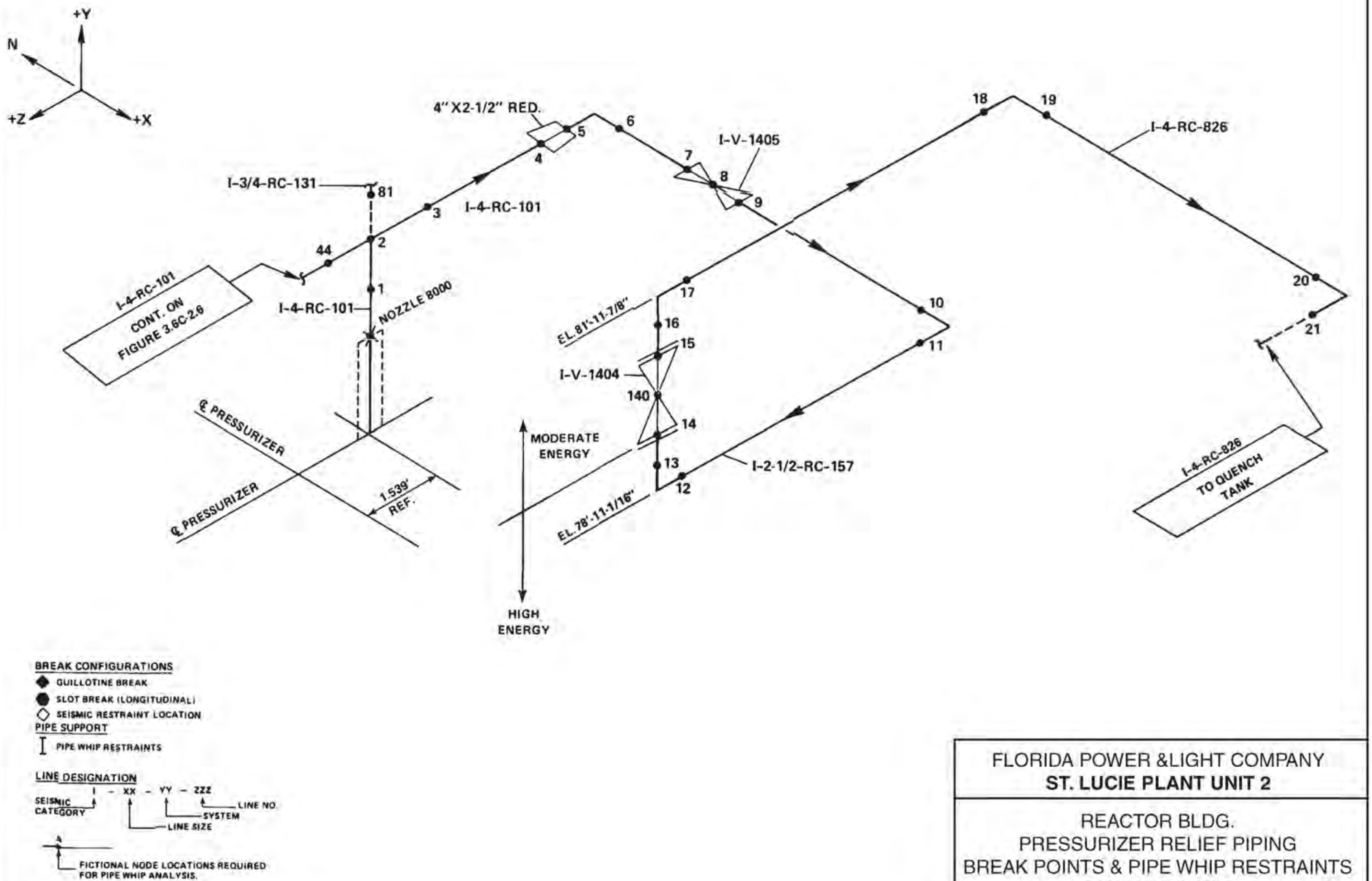
SURGE LINE
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-2.4



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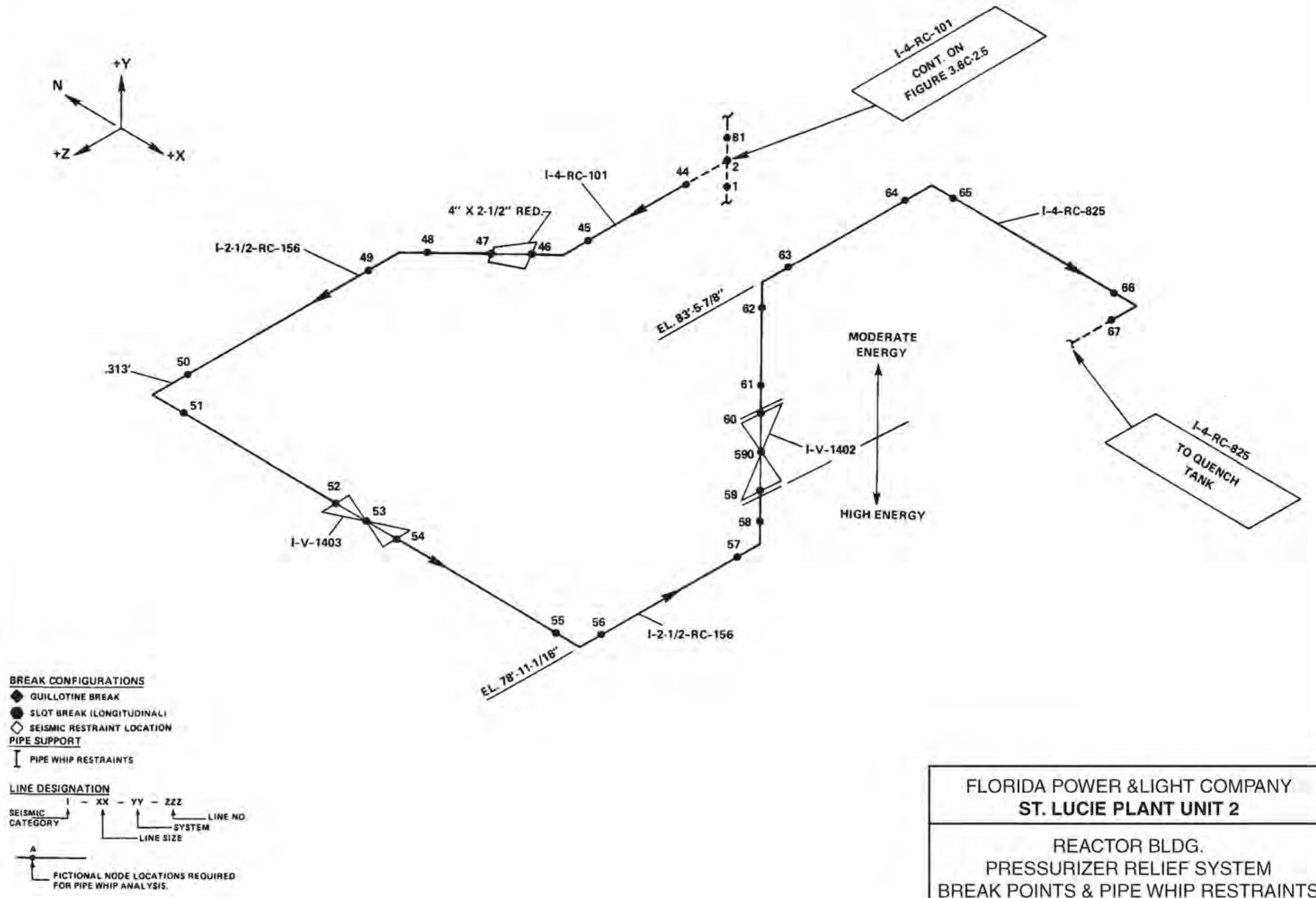
REACTOR BLDG.
PRESSURIZER RELIEF PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-2.5



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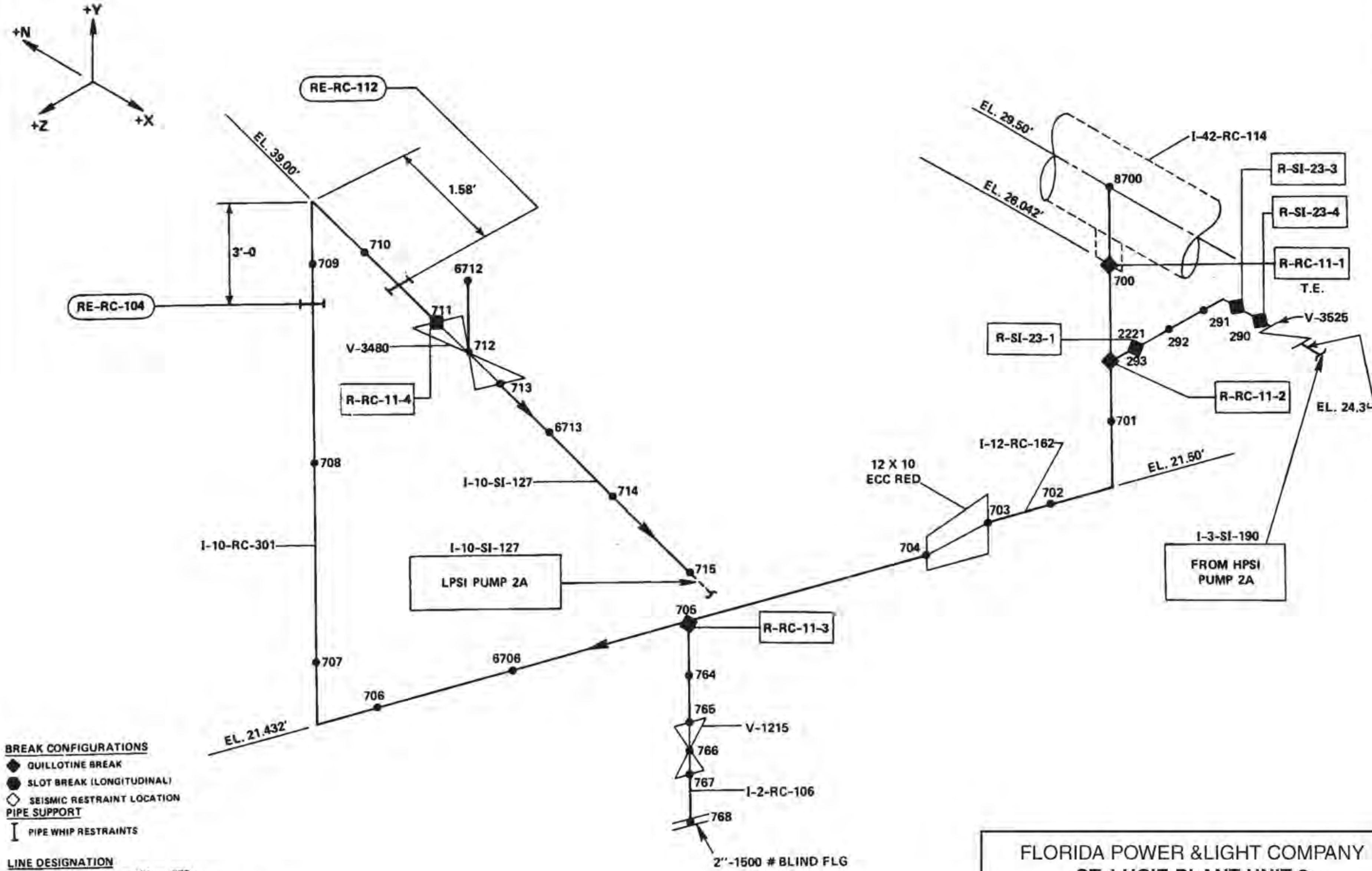
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
PRESSURIZER RELIEF SYSTEM
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-2.6



BREAK CONFIGURATIONS
- GUILLOTINE BREAK
- SLOT BREAK (LONGITUDINAL)
- SEISMIC RESTRAINT LOCATION

PIPE SUPPORT
- PIPE WHIP RESTRAINTS

LINE DESIGNATION

I – XX – YY – ZZZ

SEISMIC CATEGORY — LINE SIZE — SYSTEM — LINE NO.

A — FICTIONAL NODE LOCATIONS REQUIRED FOR PIPE WHIP ANALYSIS.

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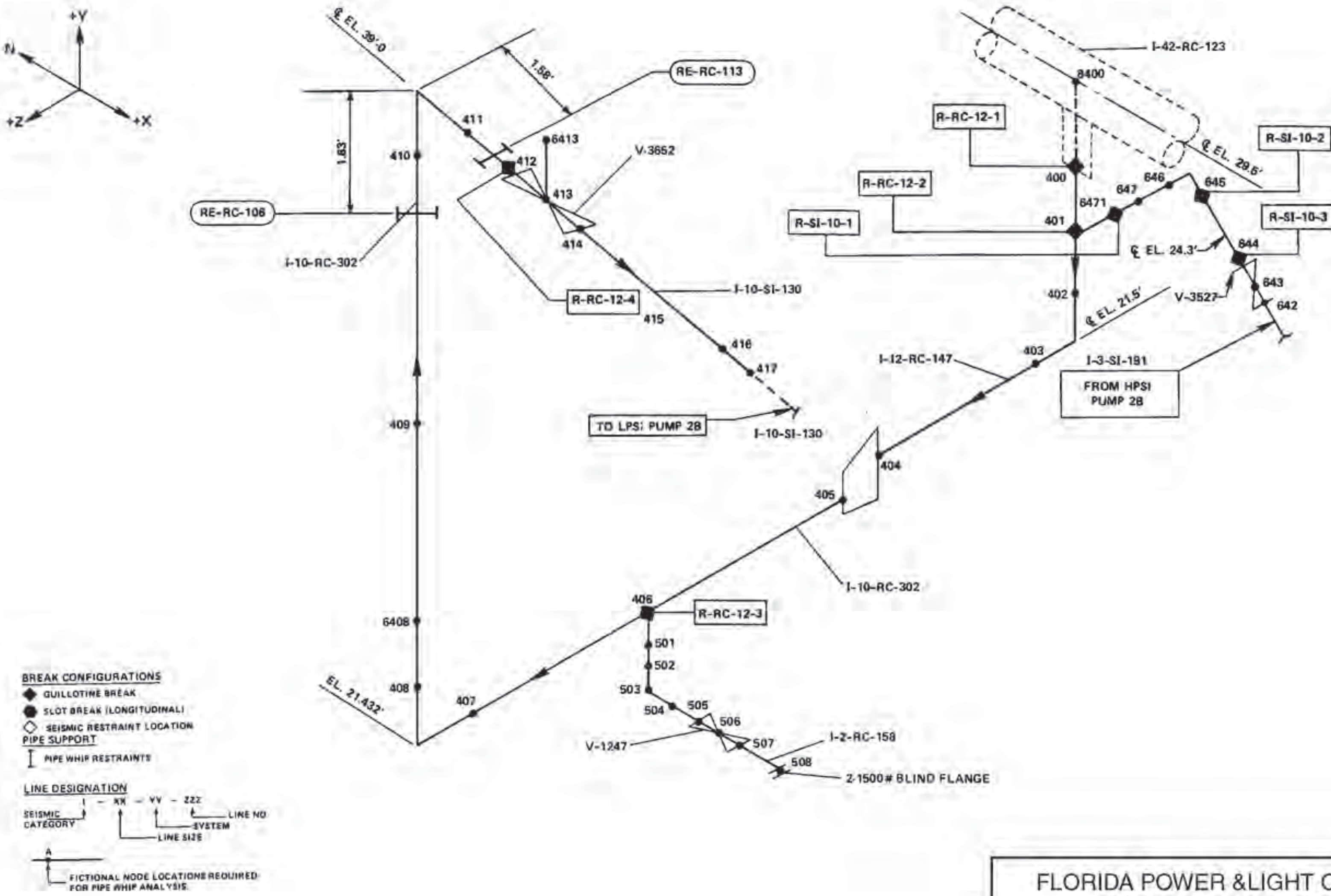
REACTOR BLDG.
SAFETY INJECTION PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.1



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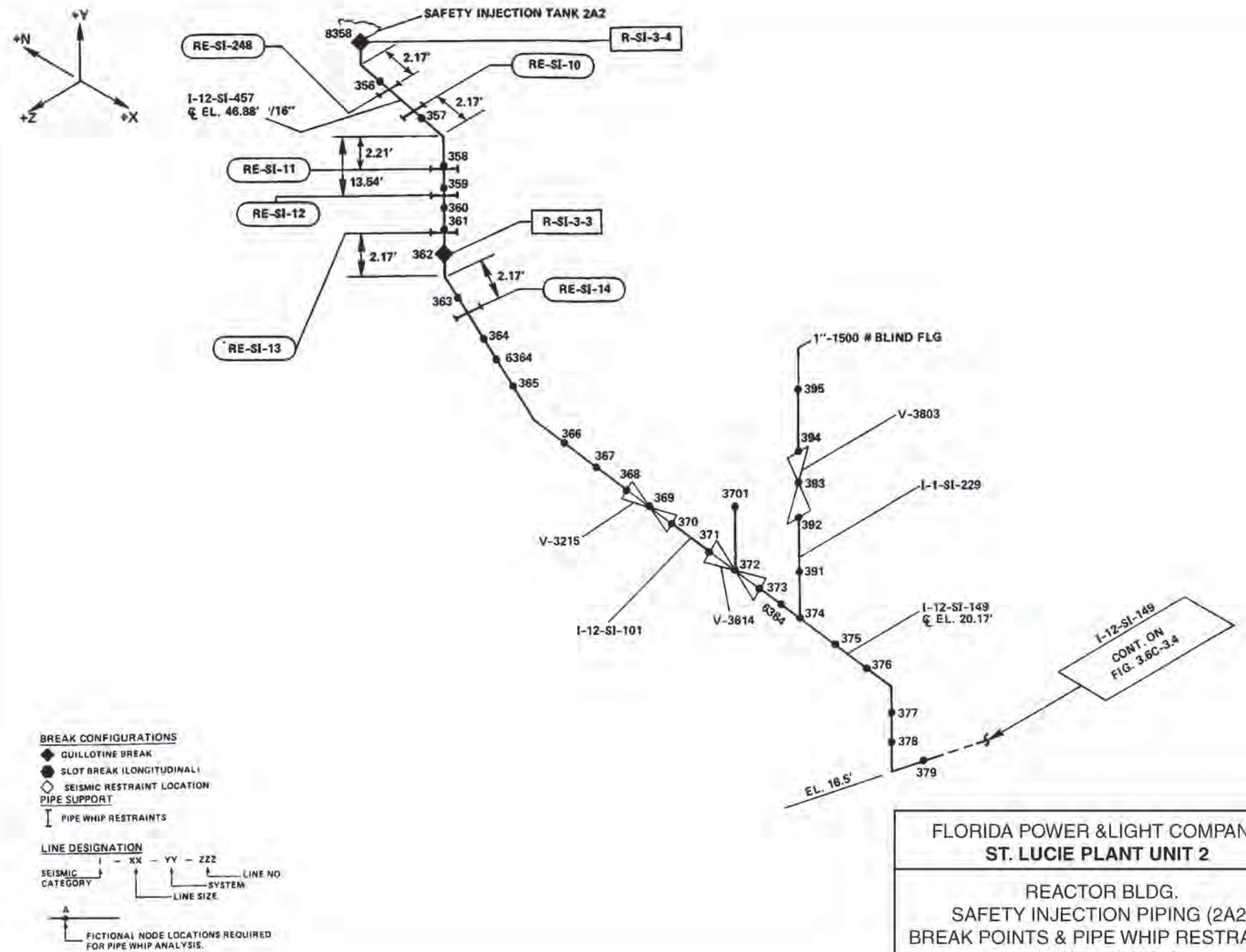
REACTOR BLDG.
SAFETY INJECTION PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.2



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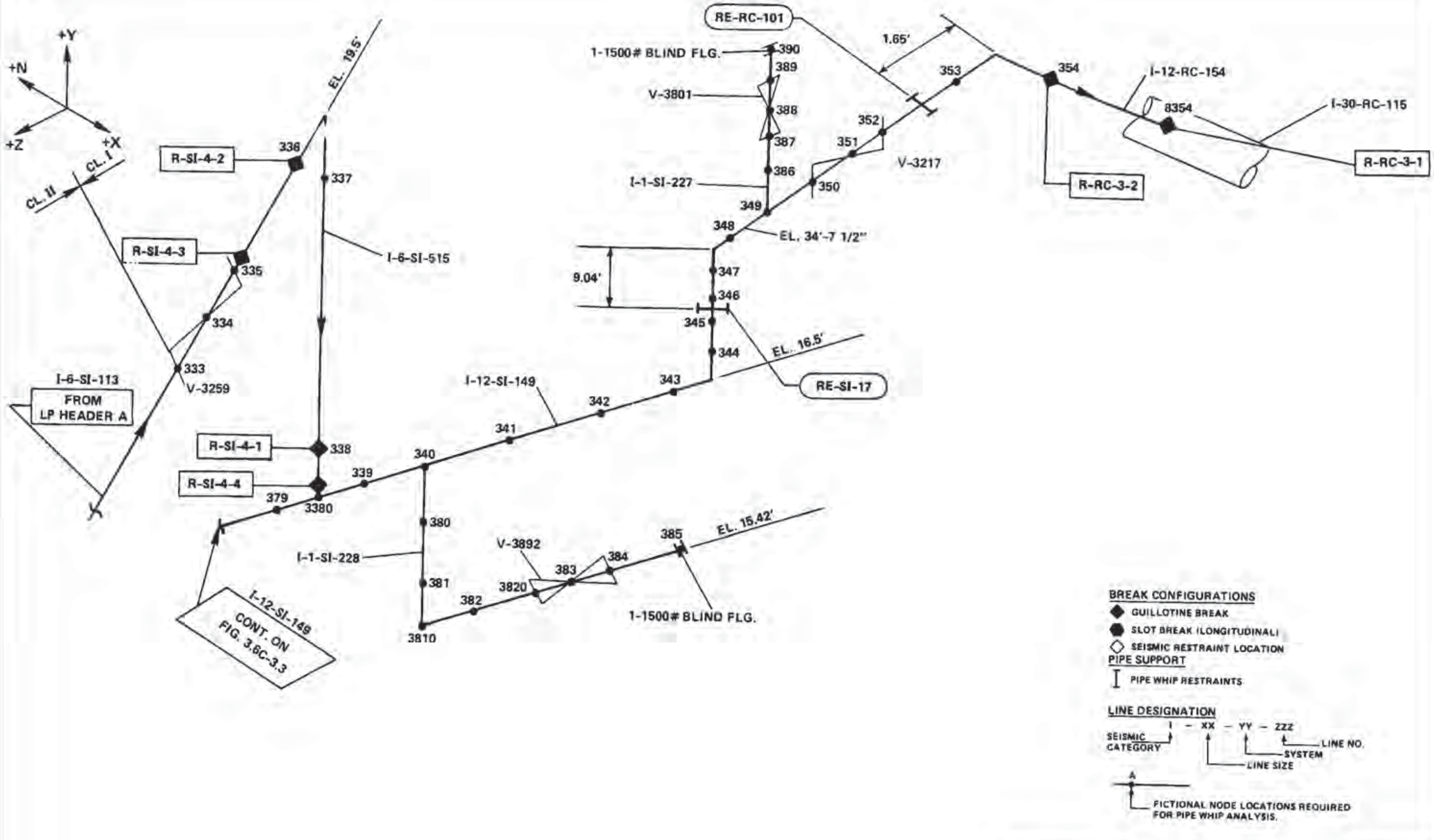
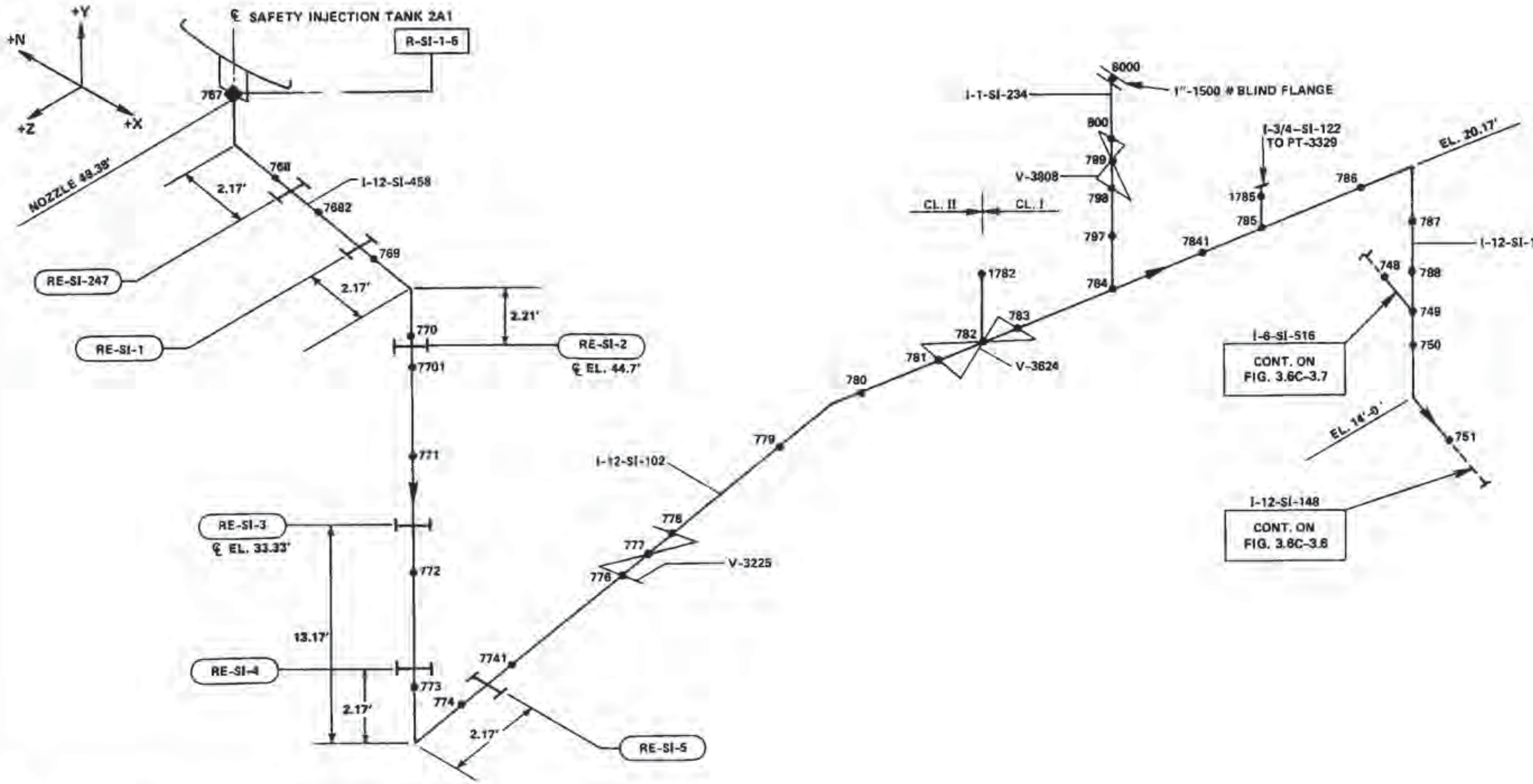
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2A2)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.3



FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2A2)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.4



BREAK CONFIGURATIONS

◆ GUILLOTINE BREAK

⬢ SLOT BREAK (LONGITUDINAL)

◇ SEISMIC RESTRAINT LOCATION

PIPE SUPPORT

I PIPE WHIP RESTRAINTS

LINE DESIGNATION

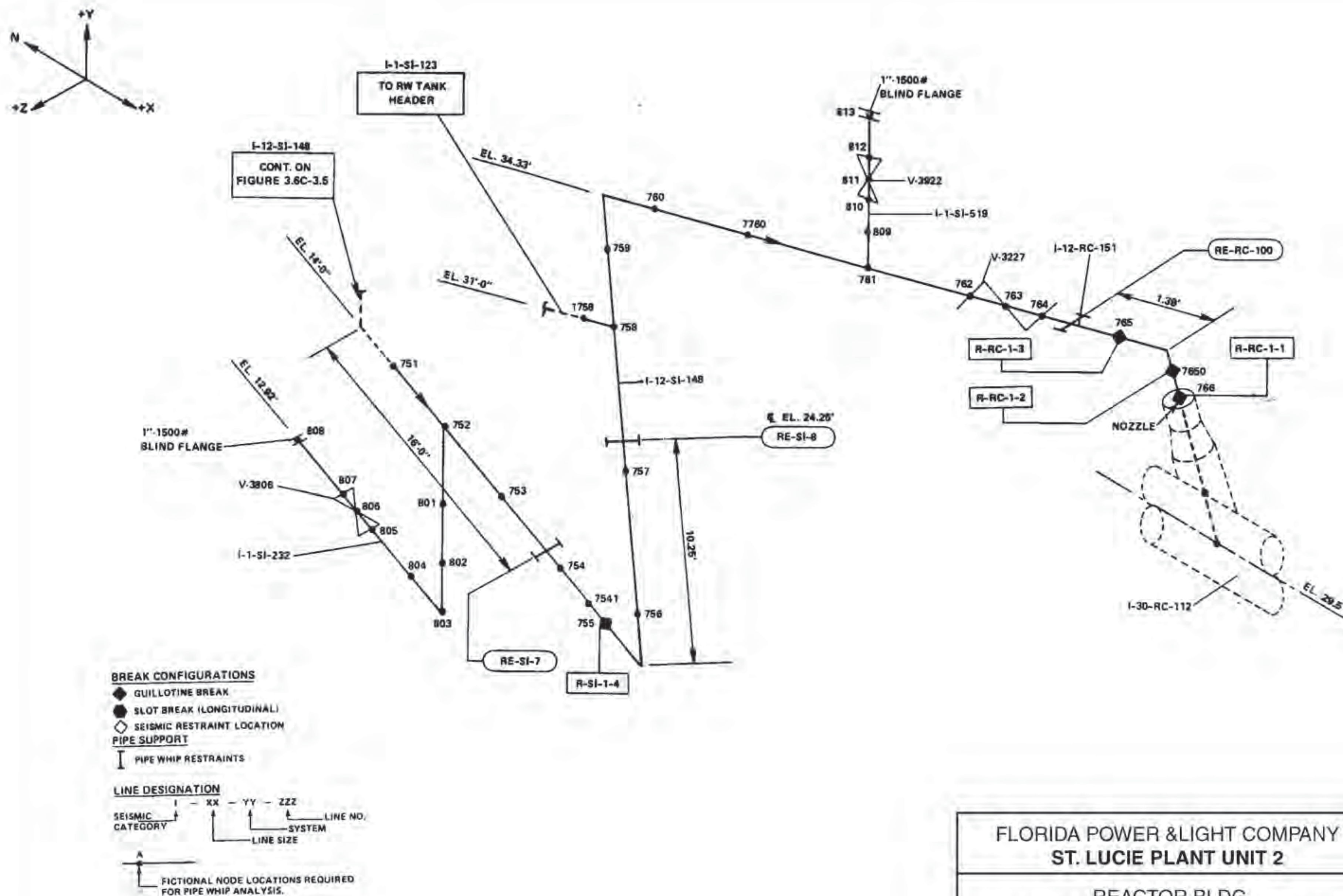
I – XX – YY – ZZZ

SEISMIC CATEGORY — LINE SIZE — SYSTEM — LINE NO.

A — FICTIONAL NODE LOCATIONS REQUIRED FOR PIPE WHIP ANALYSIS.

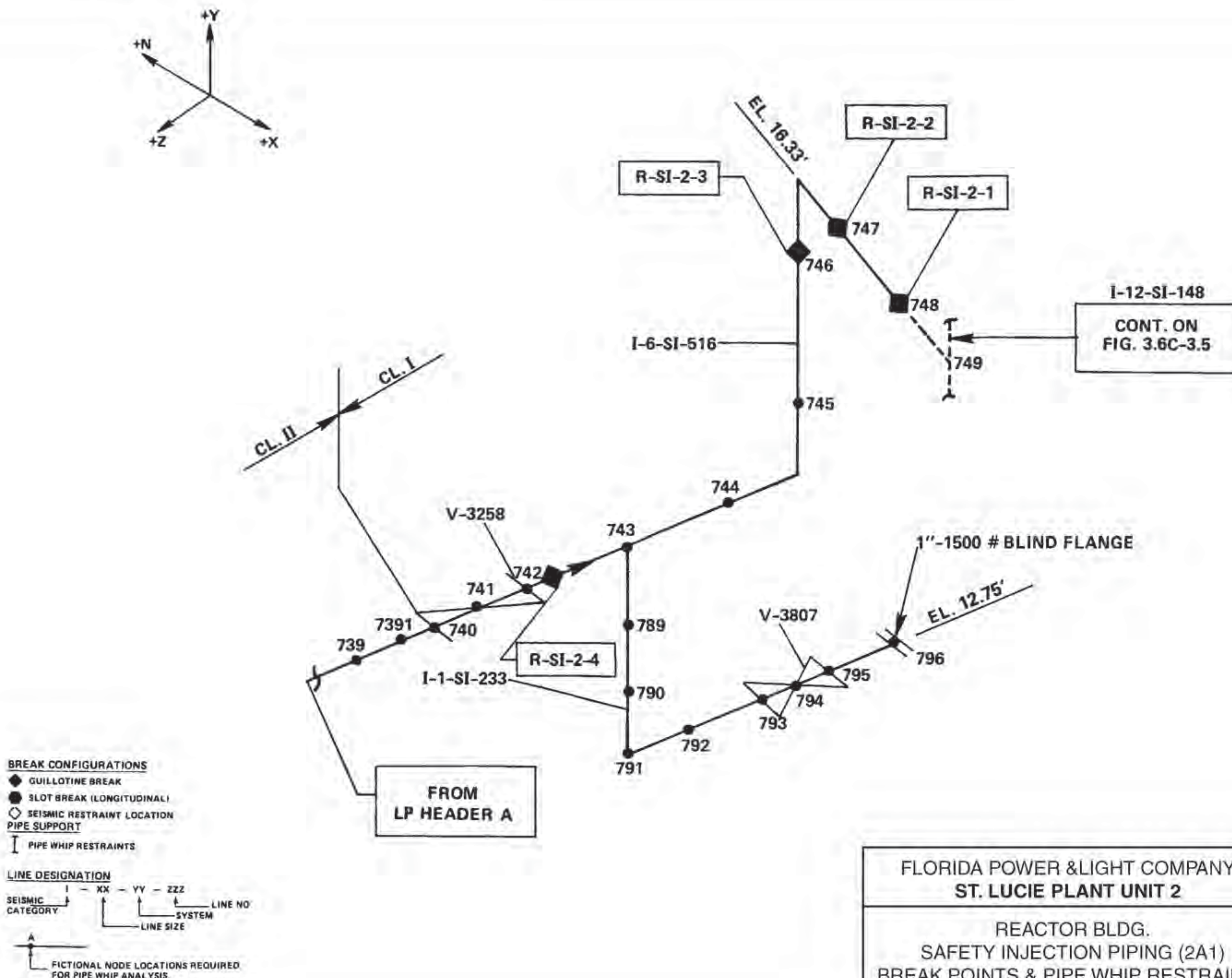
FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2A1)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.5



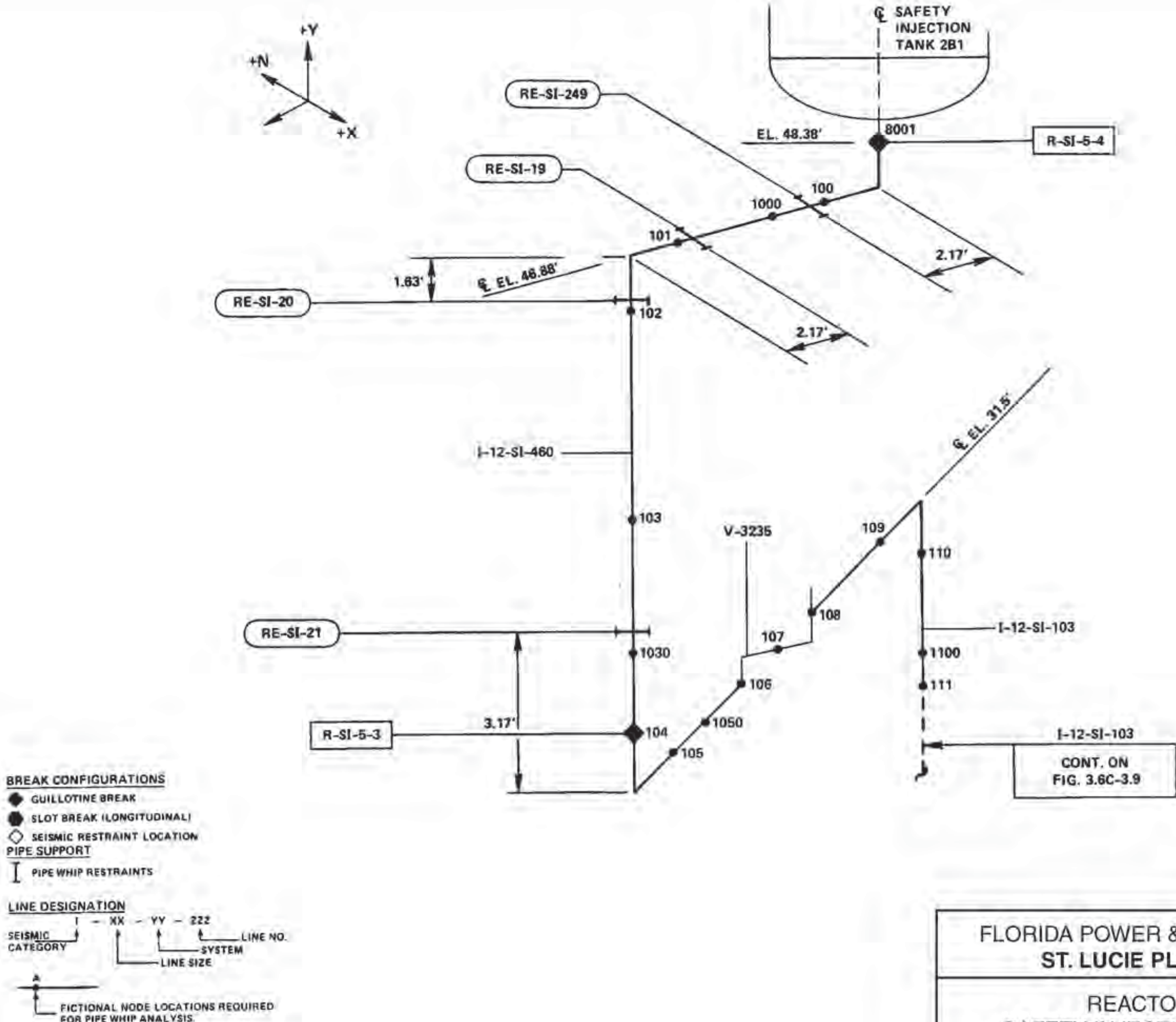
FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2A1)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.6



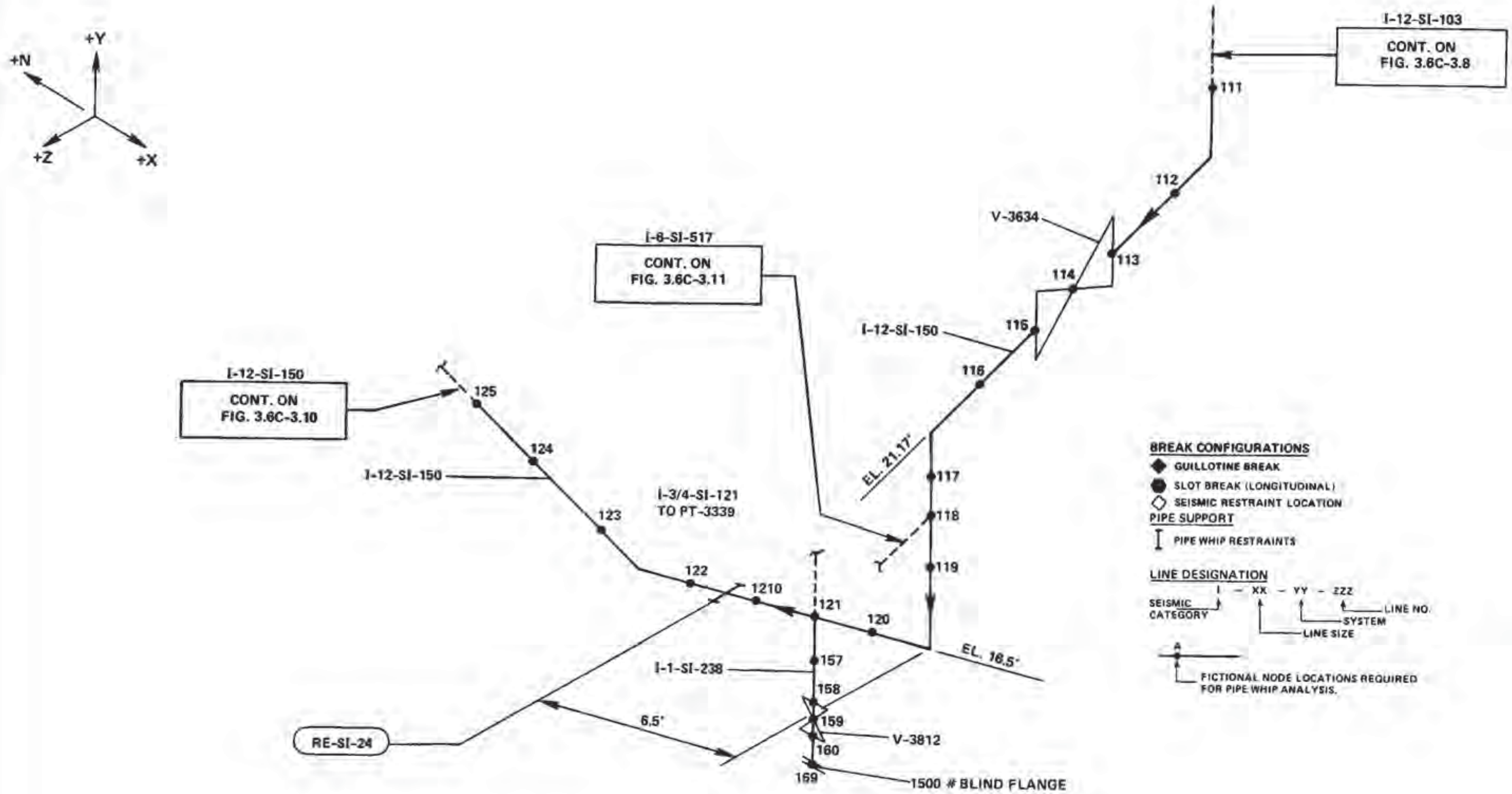
FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2A1)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.7



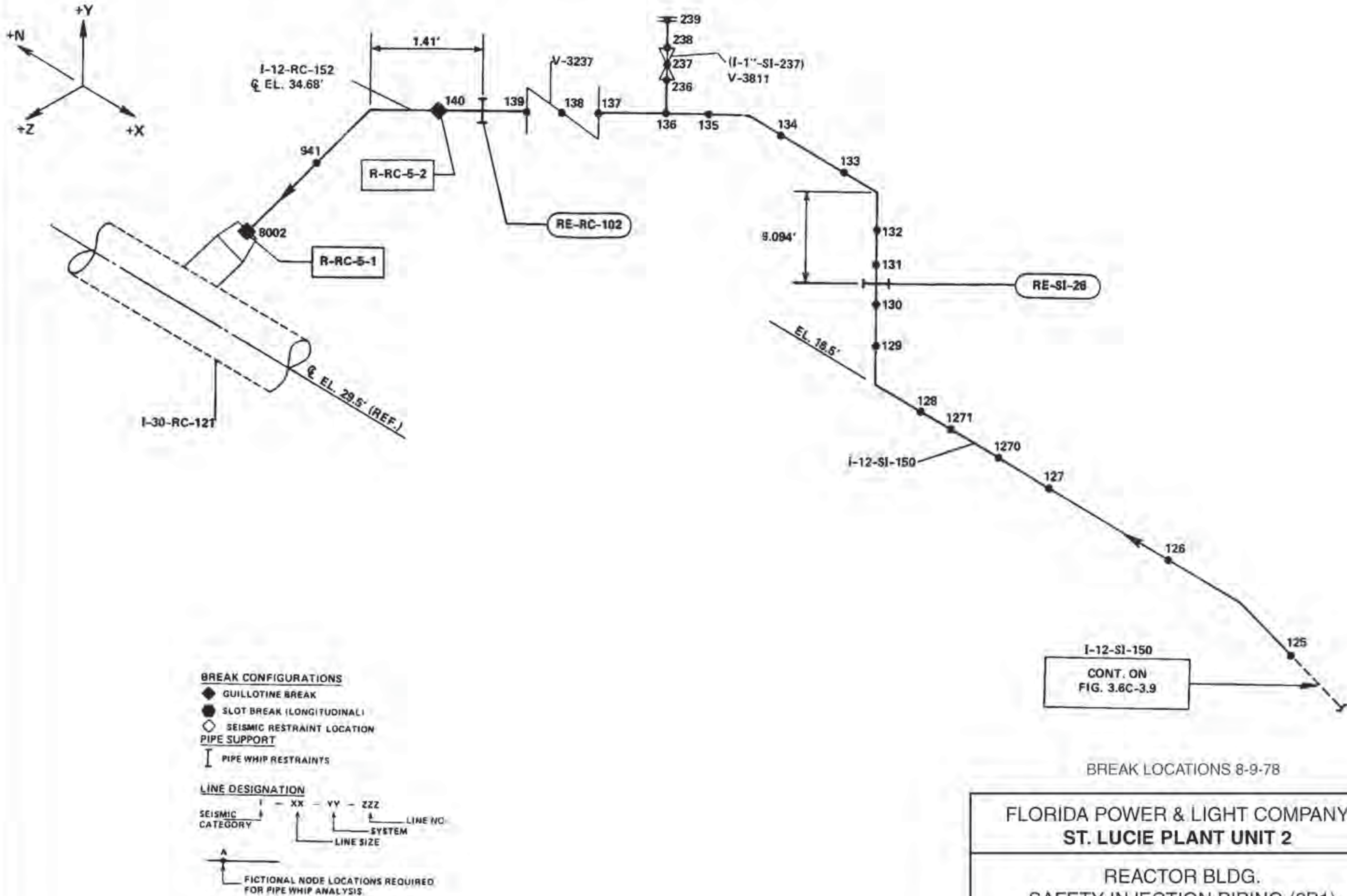
FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2B1)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.8



FLORIDA POWER &LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

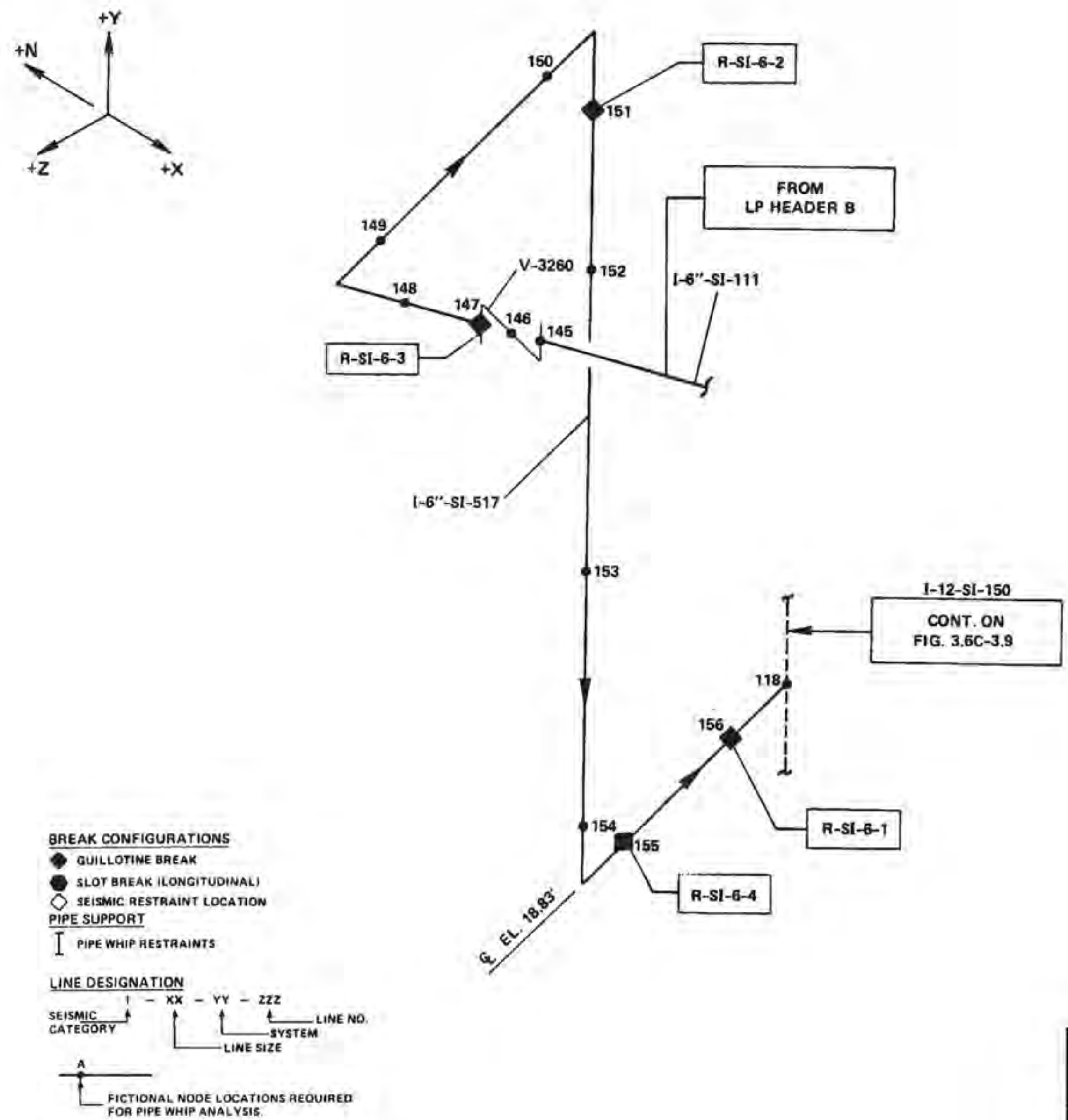
REACTOR BLDG.
SAFETY INJECTION PIPING (2B1)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.9



BREAK LOCATIONS 8-9-78

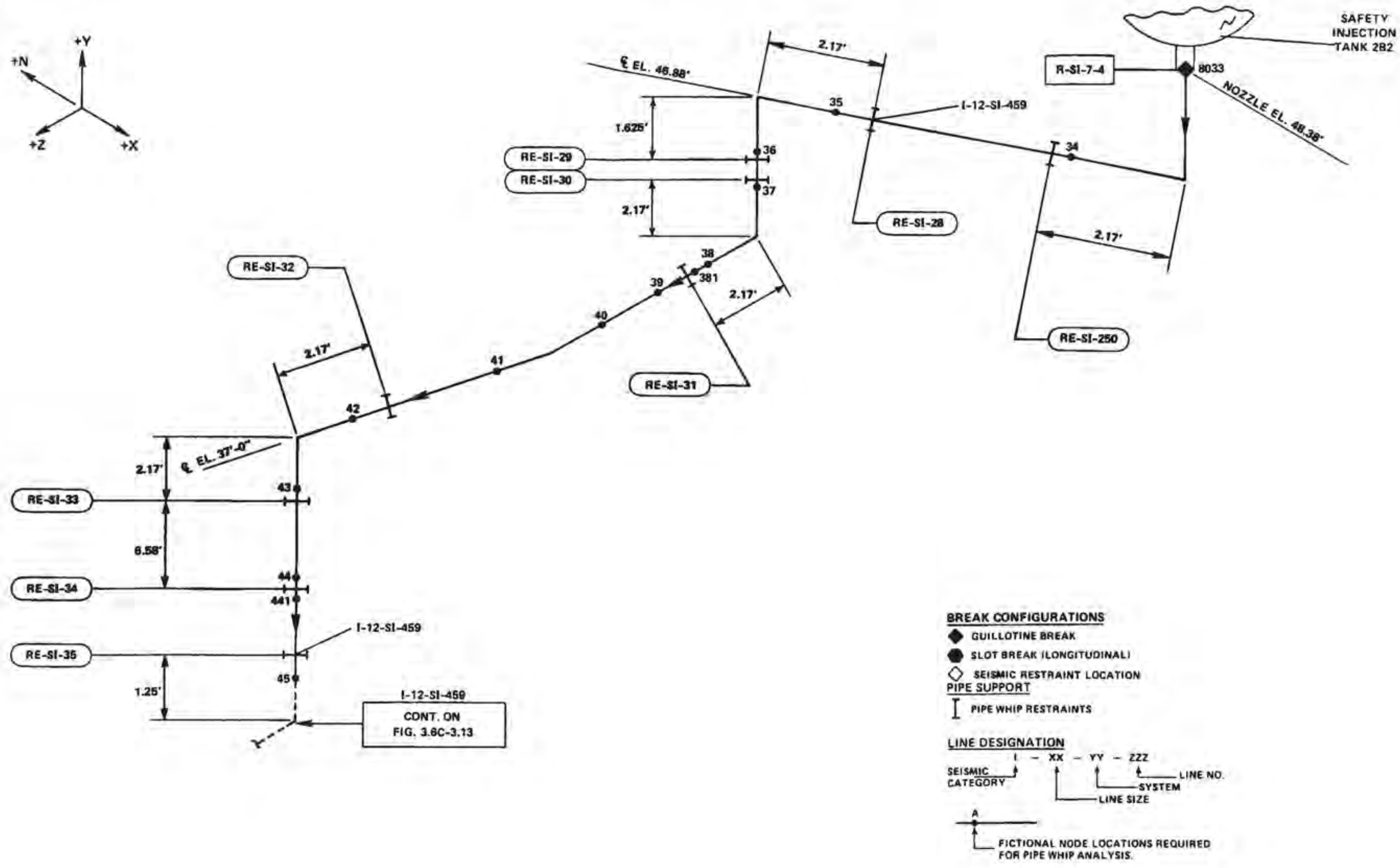
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2B1)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.10



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2B1)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.11

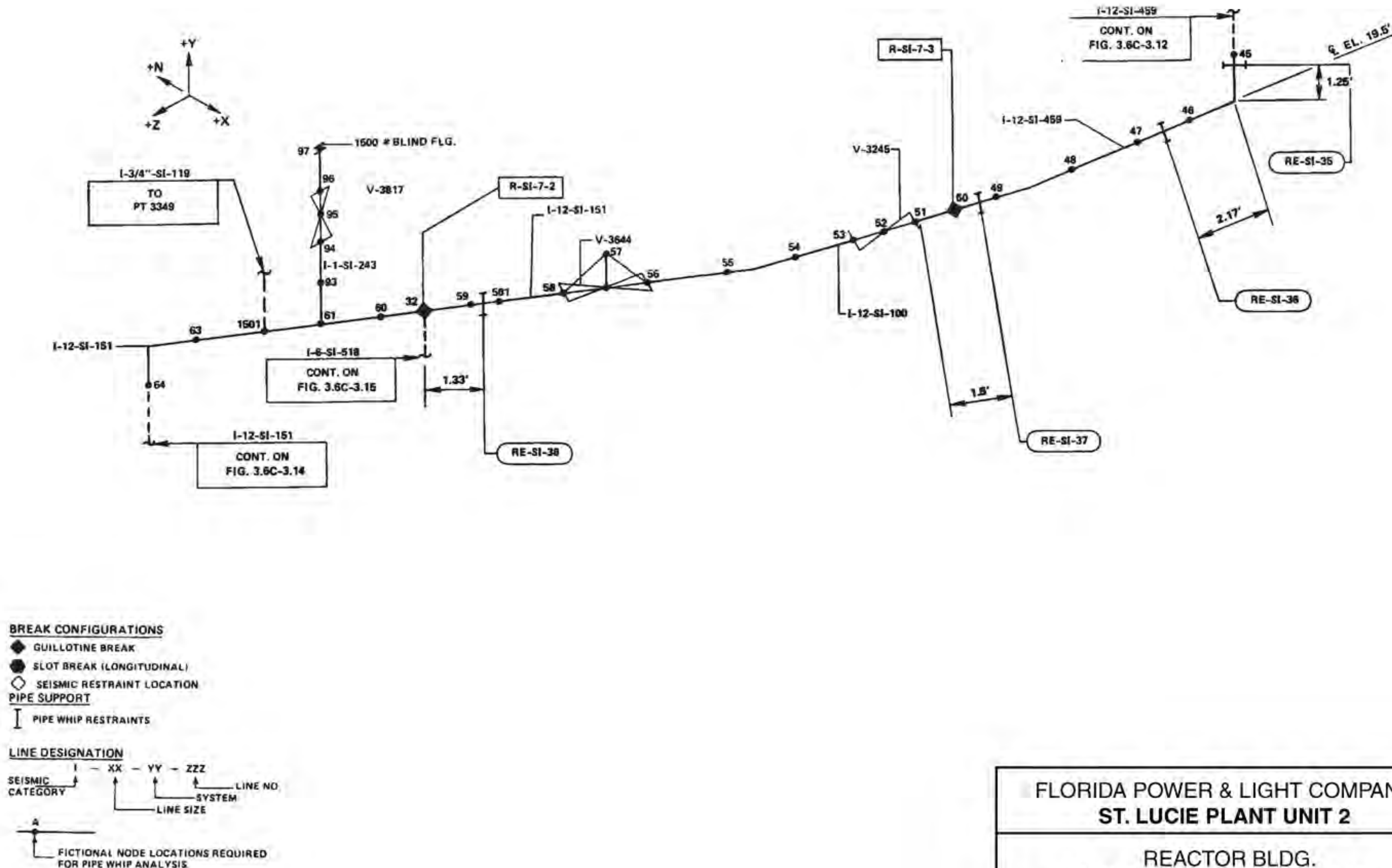


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 2

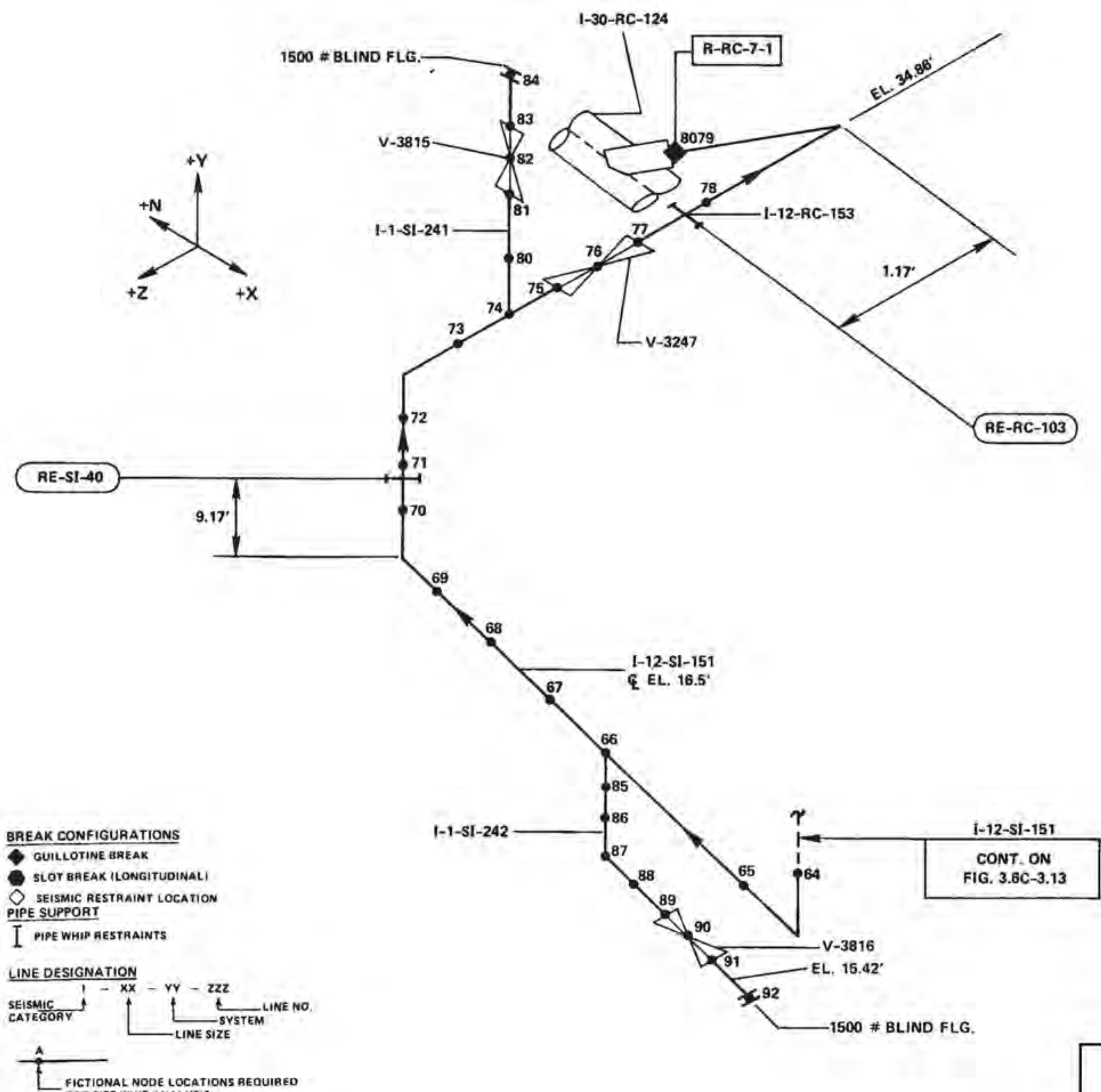
REACTOR BLDG.
SAFETY INJECTION PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS

FIGURE 3.6C-3.12



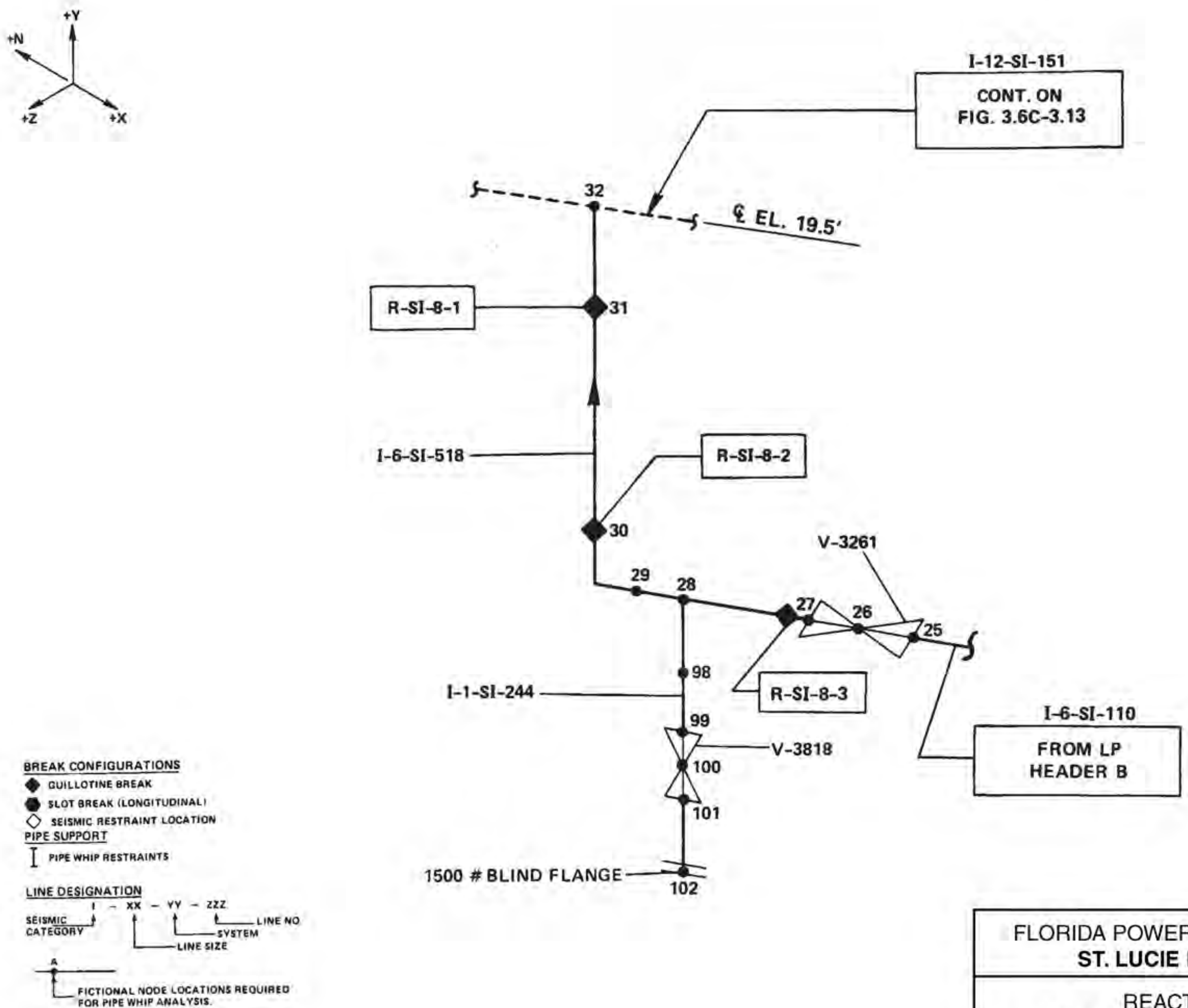
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2B2)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.13



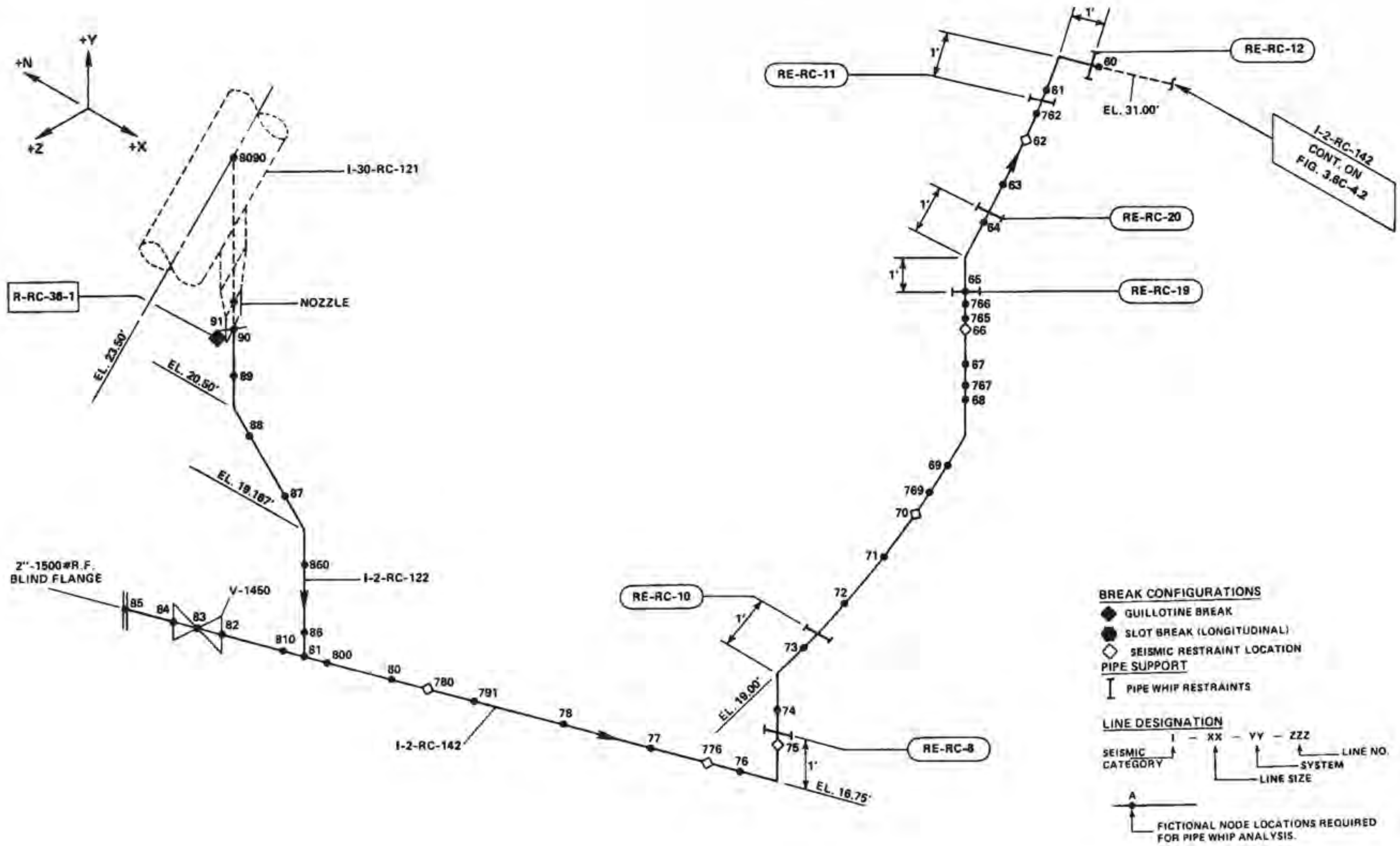
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2B2)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.14



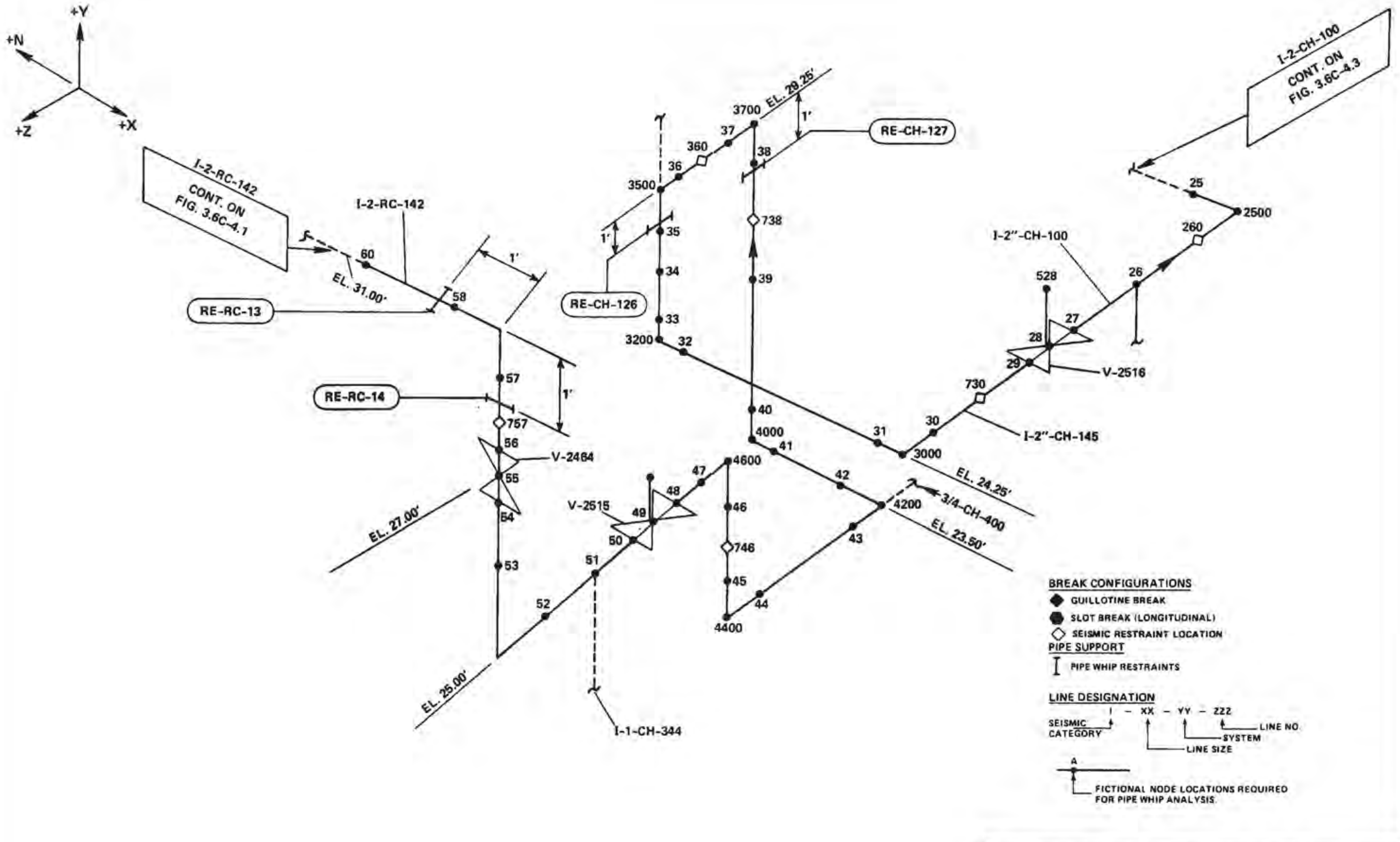
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
SAFETY INJECTION PIPING (2B2)
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-3.15



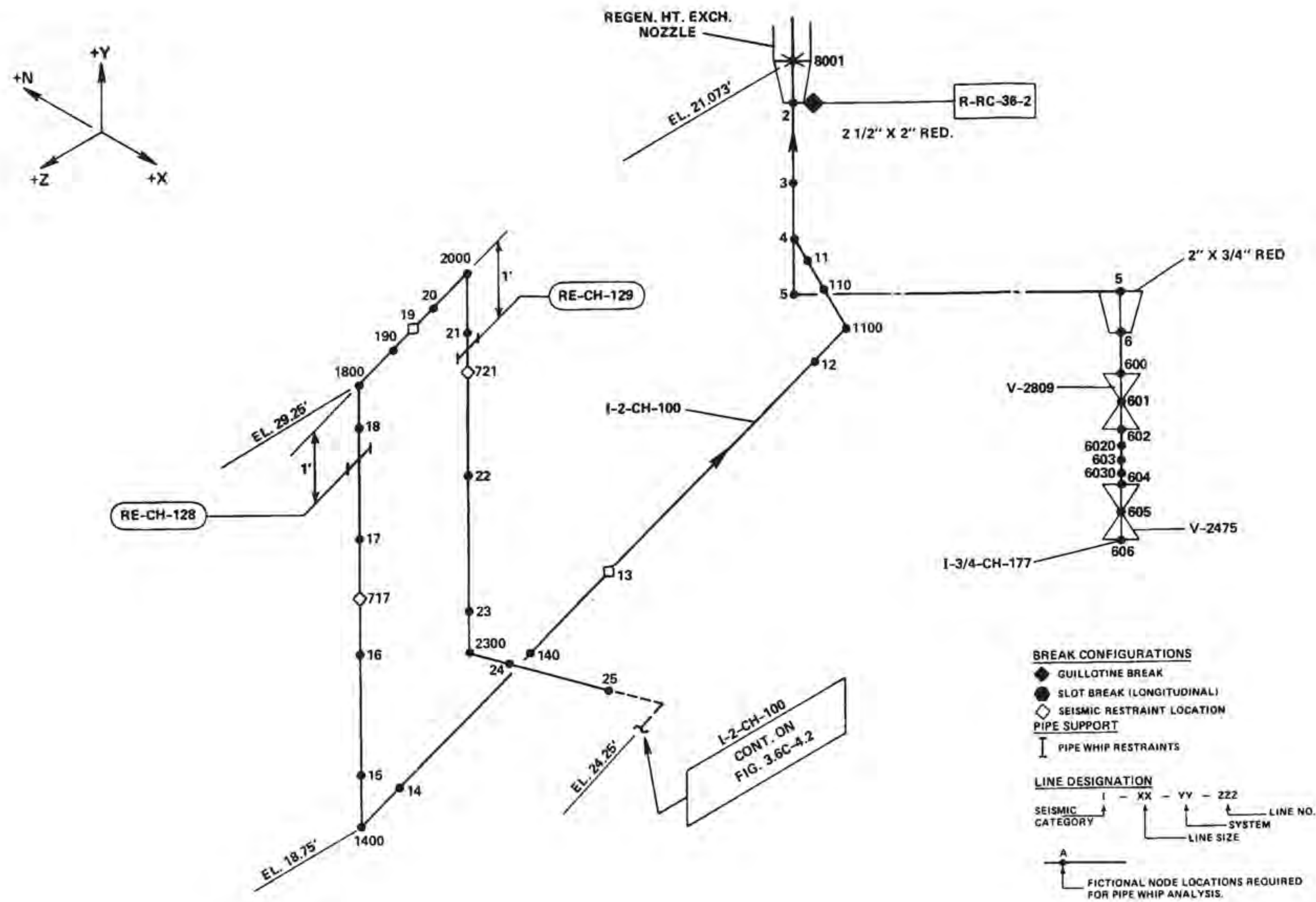
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
CVCS LETDOWN PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-4.1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
CVCS LETDOWN PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-4.2

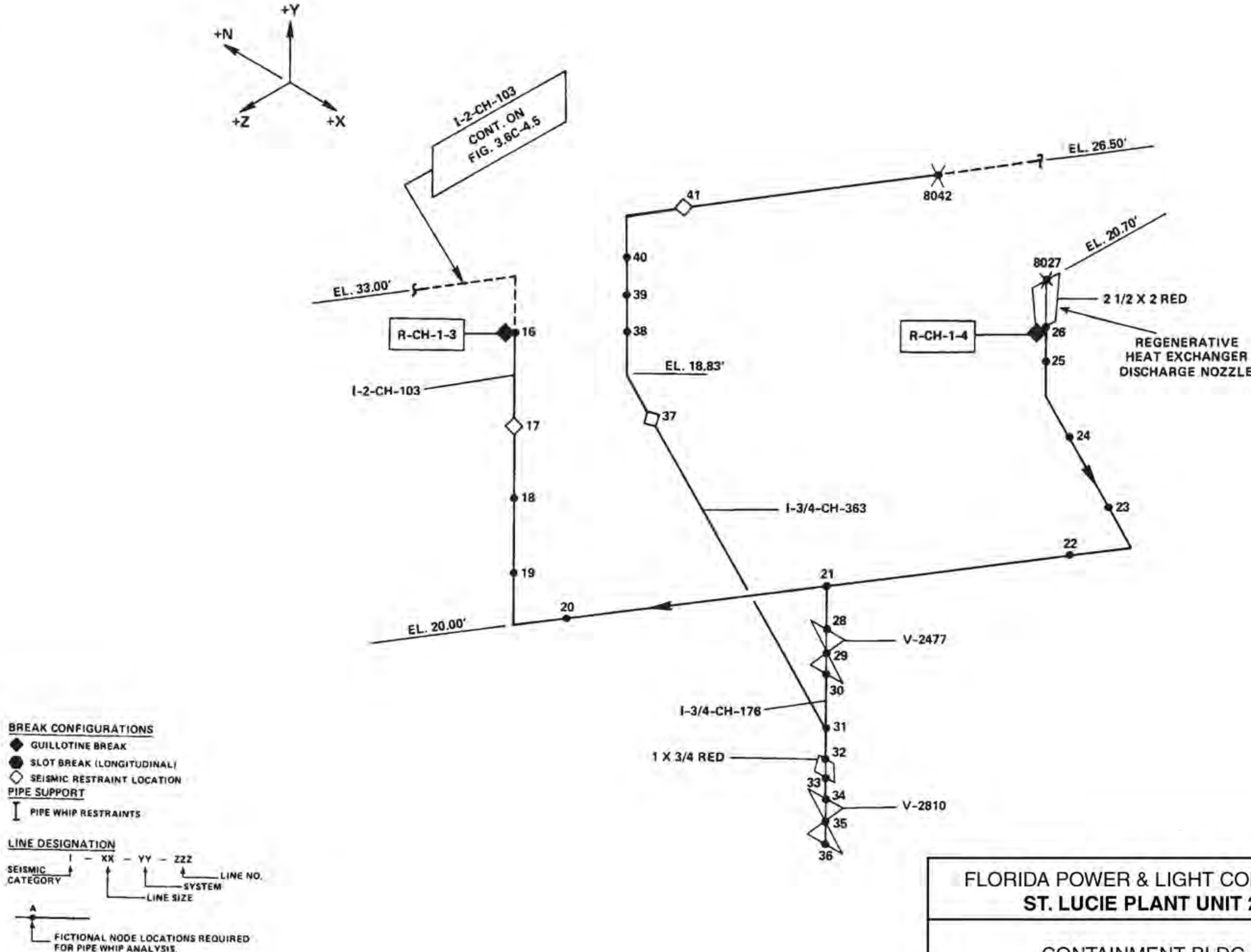


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 2

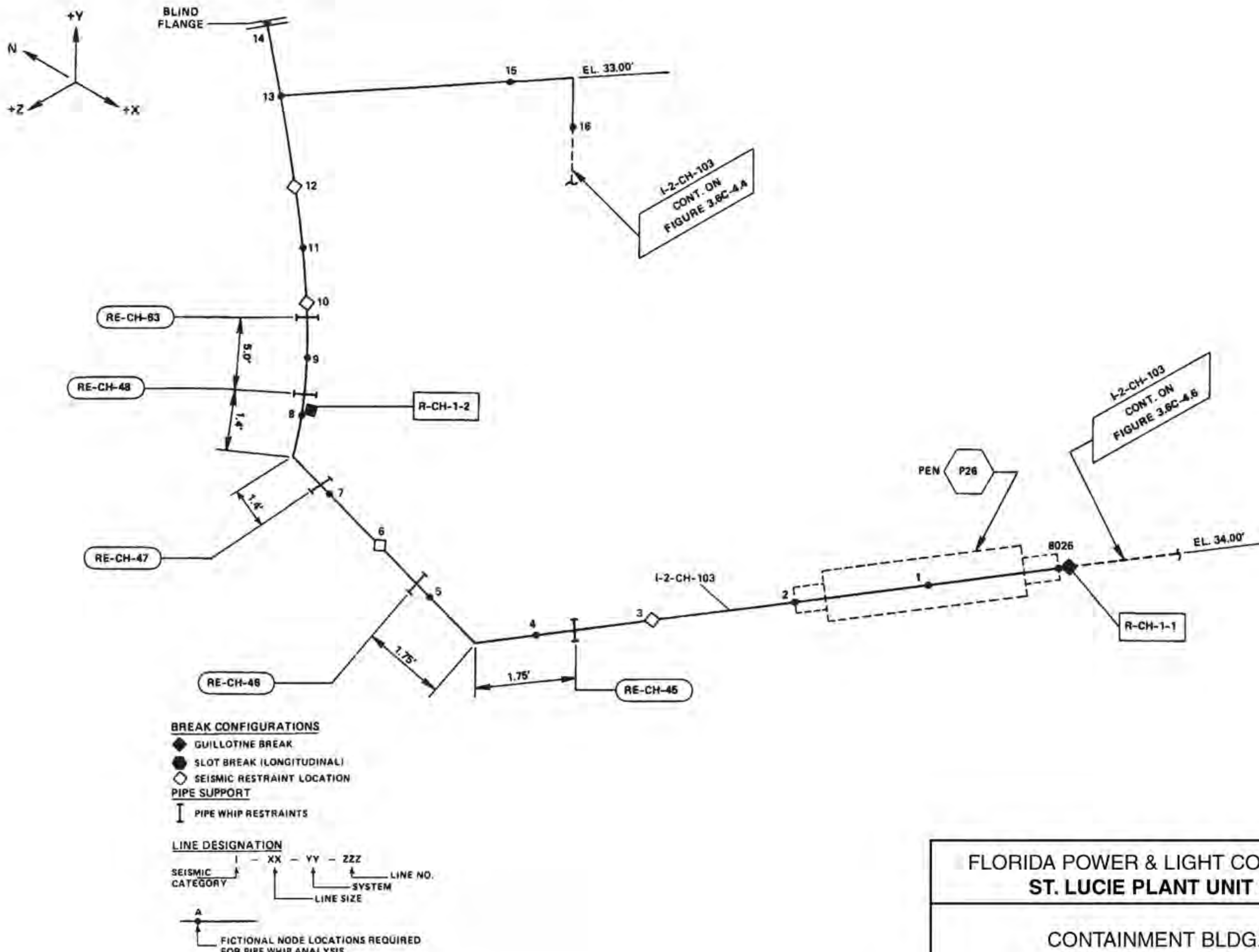
REACTOR BLDG.
CVCS LETDOWN PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS

FIGURE 3.6C-4.3



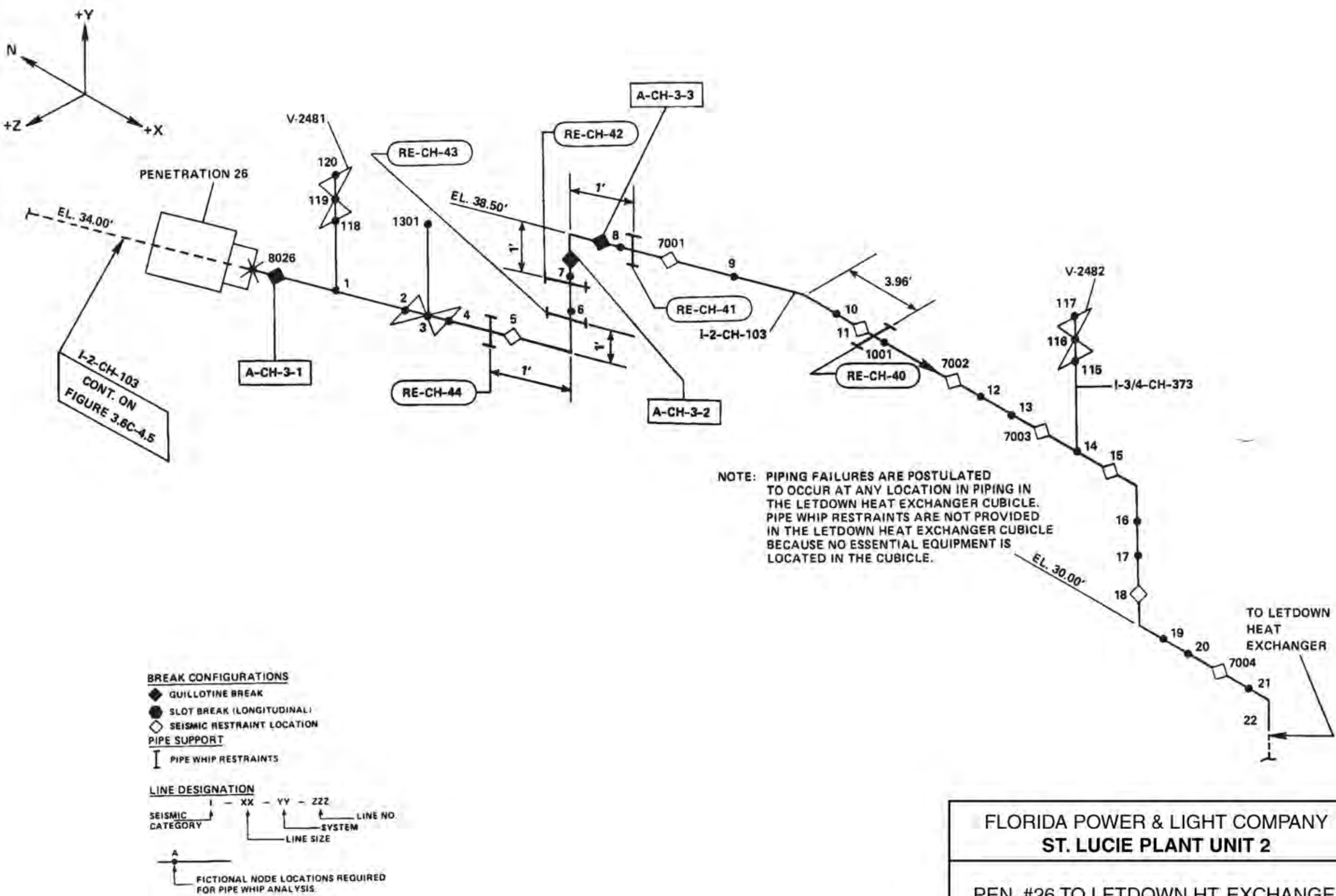
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT BLDG.
CVCS LETDOWN LINE
FIGURE 3.6C-4.4



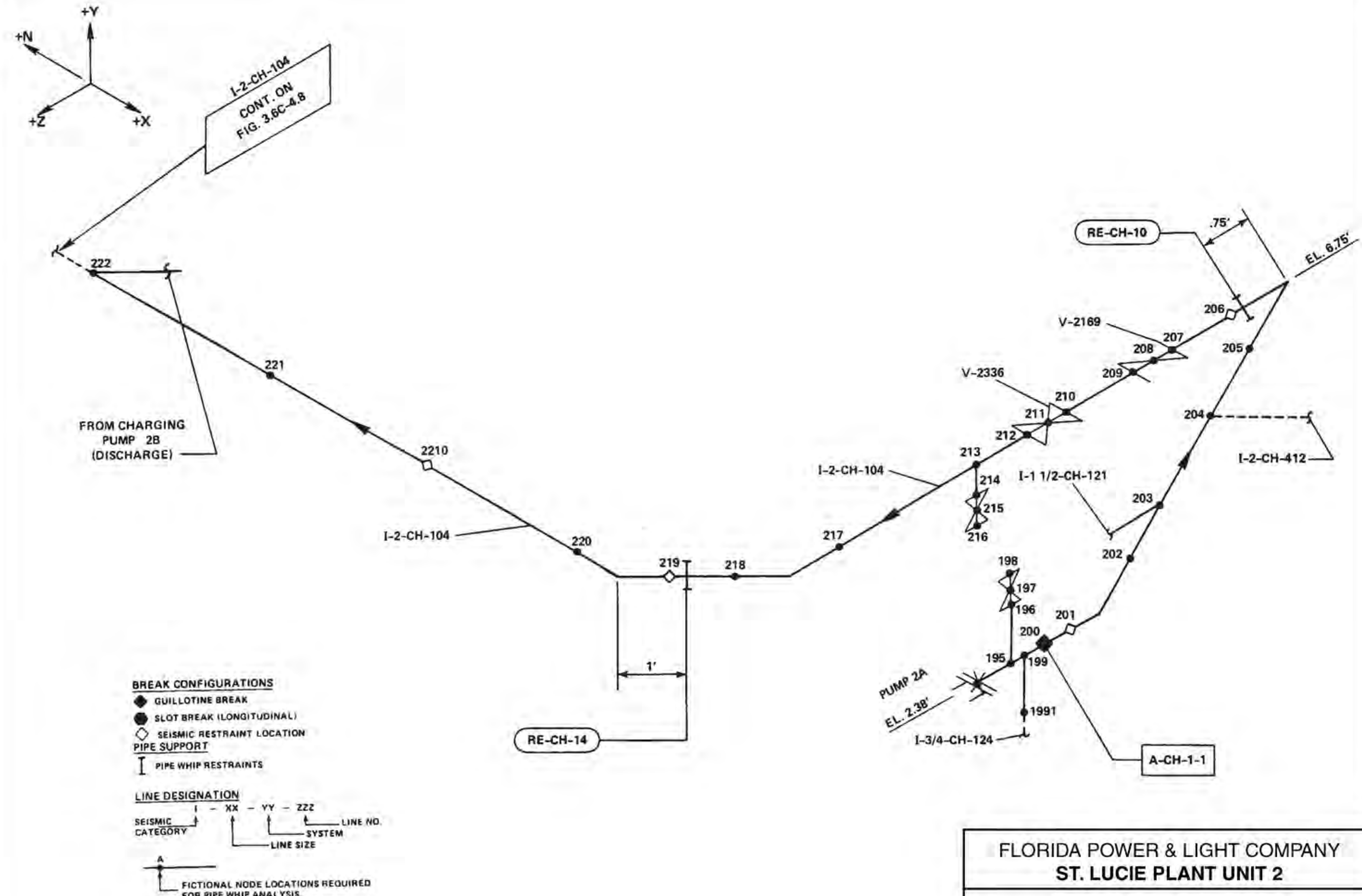
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT BLDG.
CVCS LETDOWN LINE
FIGURE 3.6C-4.5



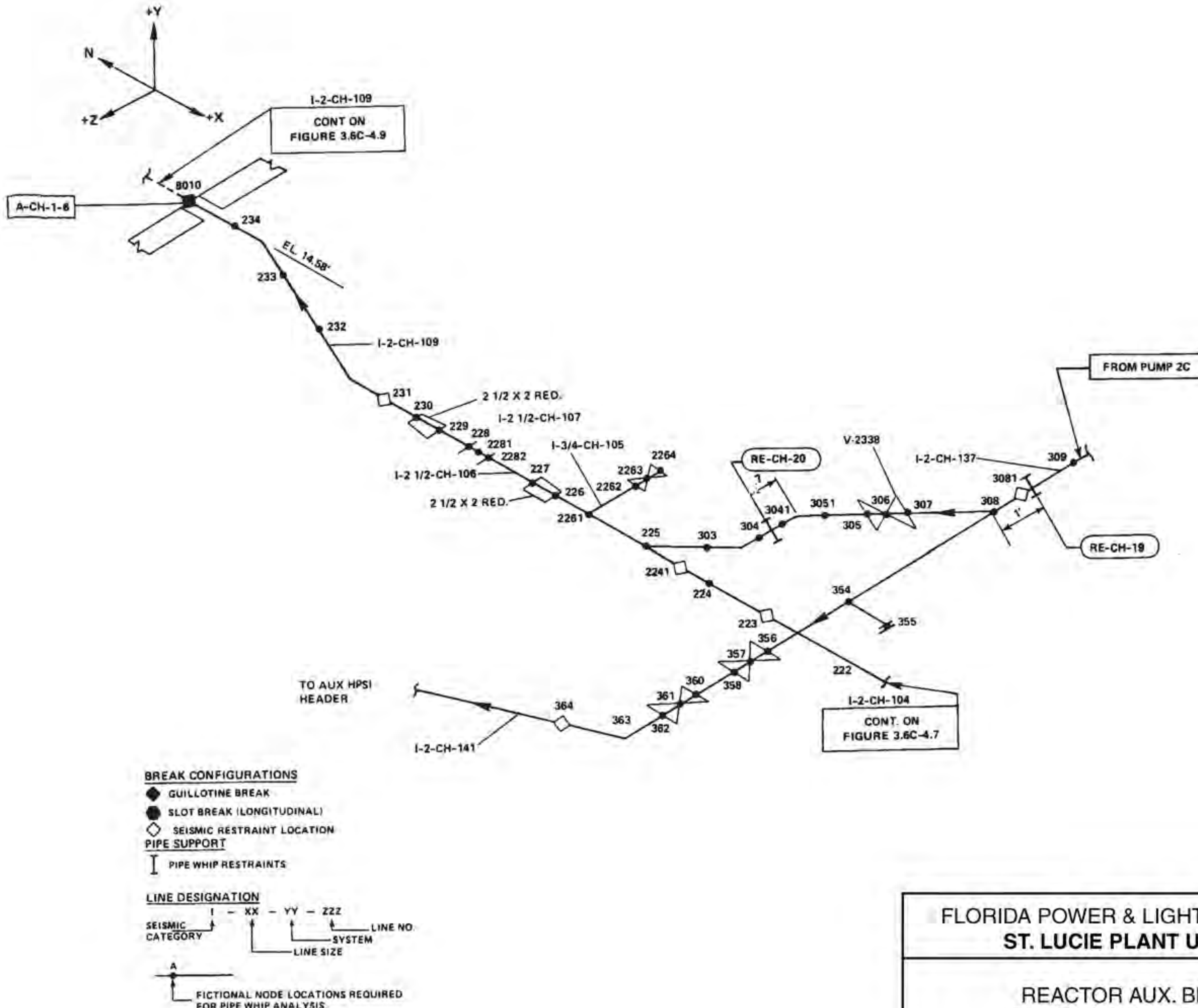
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PEN. #26 TO LETDOWN HT. EXCHANGER
CHEM. & VOL. CONTROL
FIGURE 3.6C-4.6



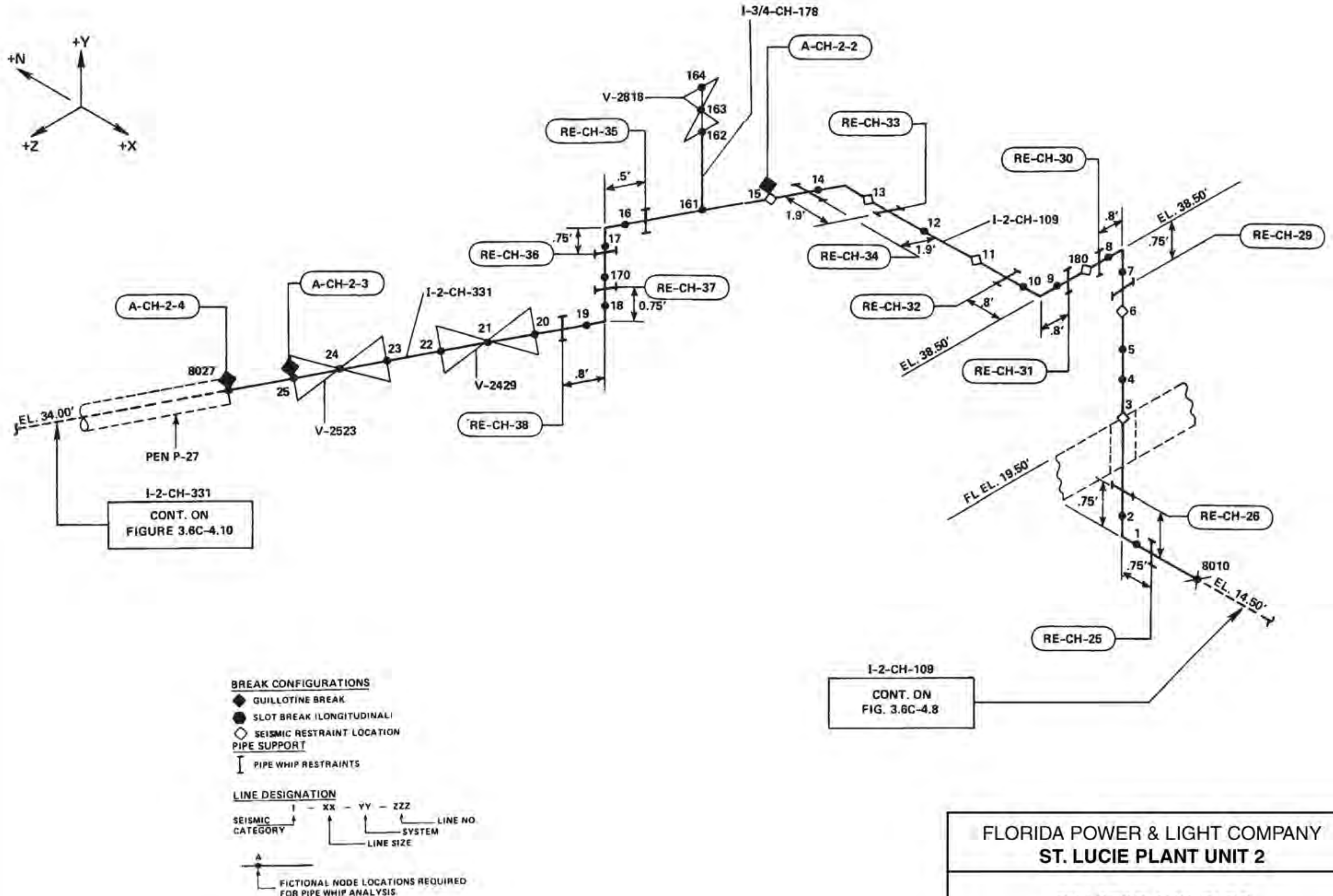
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUX. BLDG.
CHEM. & VOL. CONTROL
FIGURE 3.6C-4.7



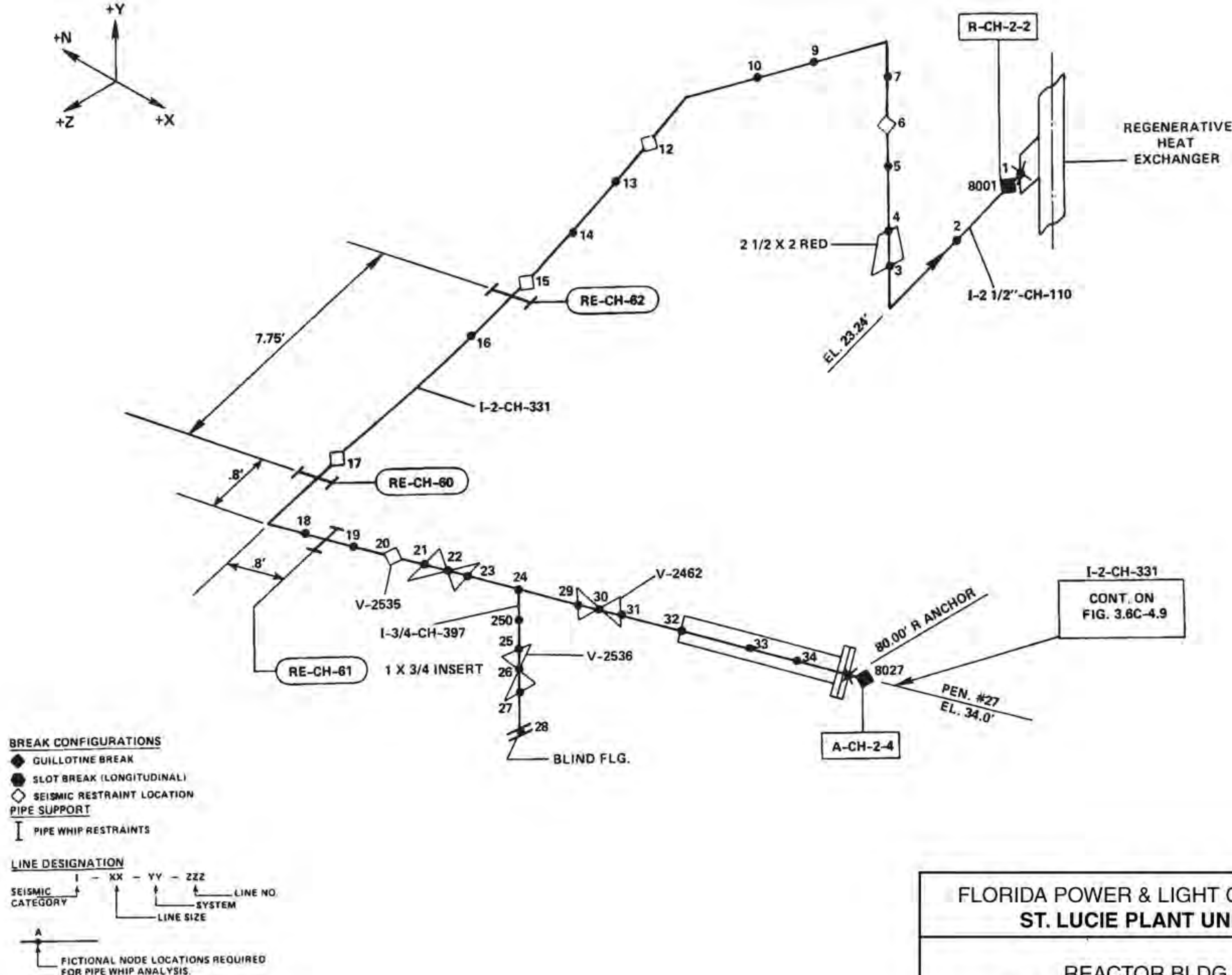
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUX. BLDG.
CHEM. & VOL. CONTROL
FIGURE 3.6C-4.8



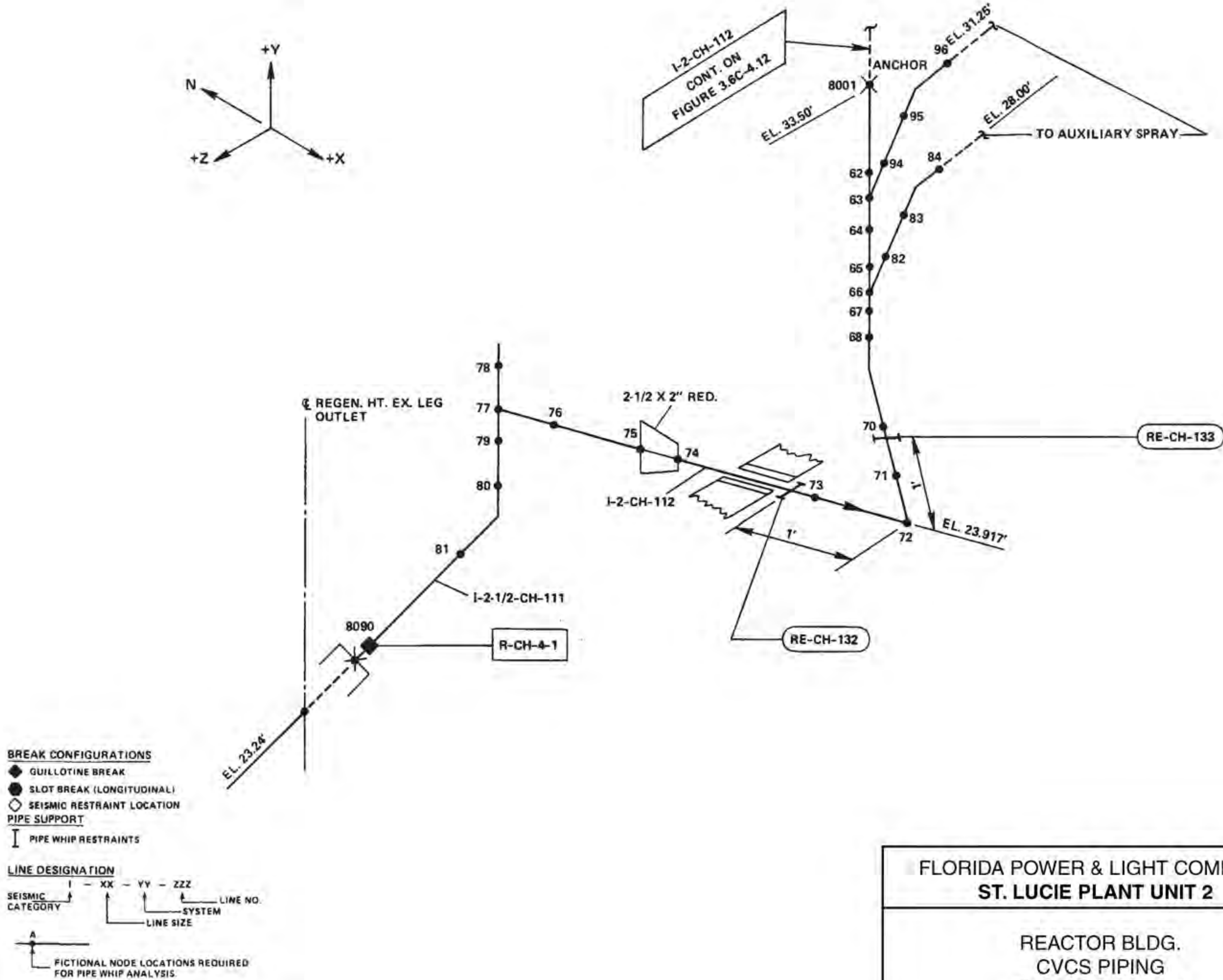
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUX. BLDG.
CVC PIPING
FIGURE 3.6C-4.9



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
CHEMICAL VOLUME CONTROL
FIGURE 3.6C-4.10

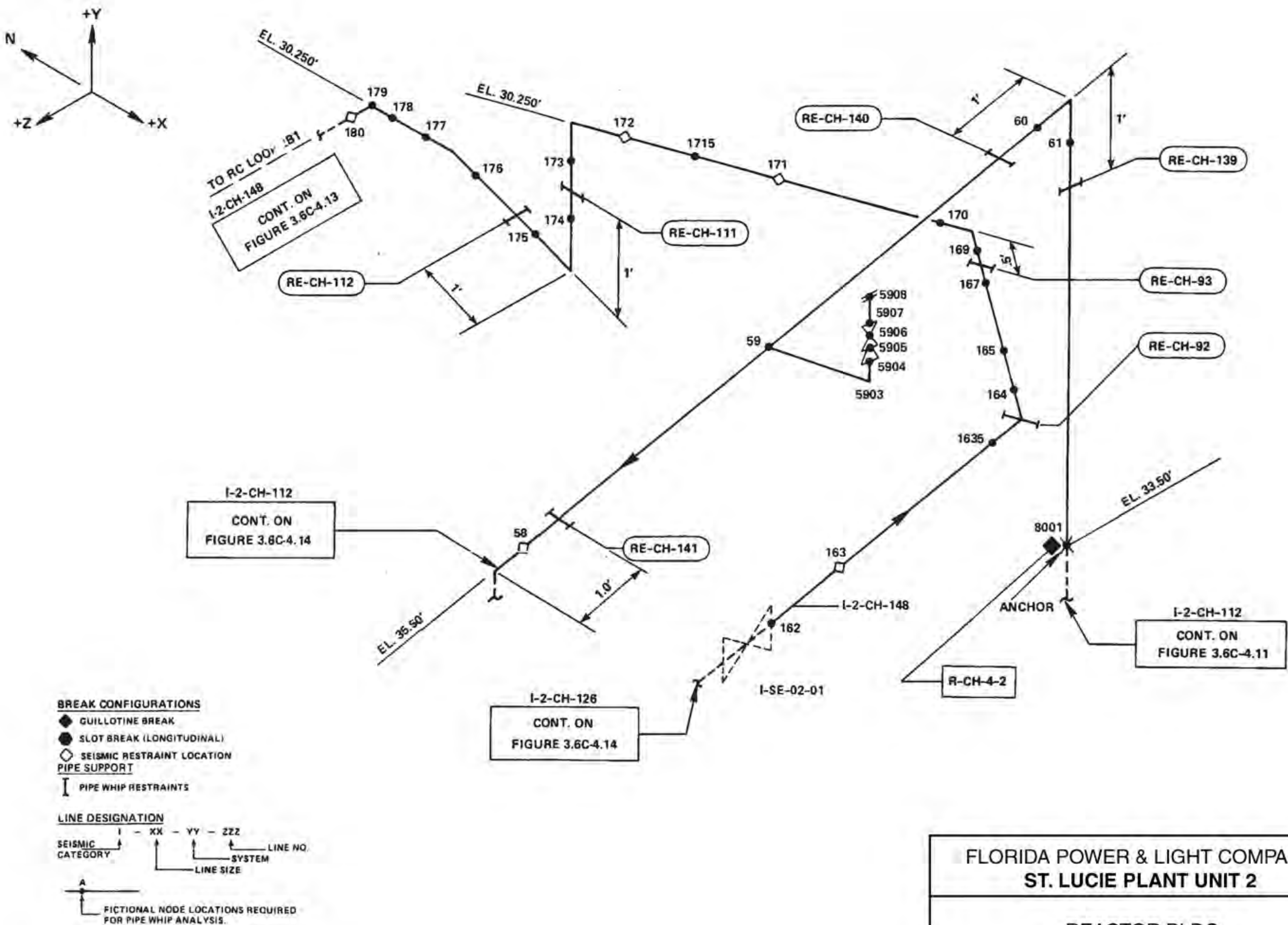


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 2

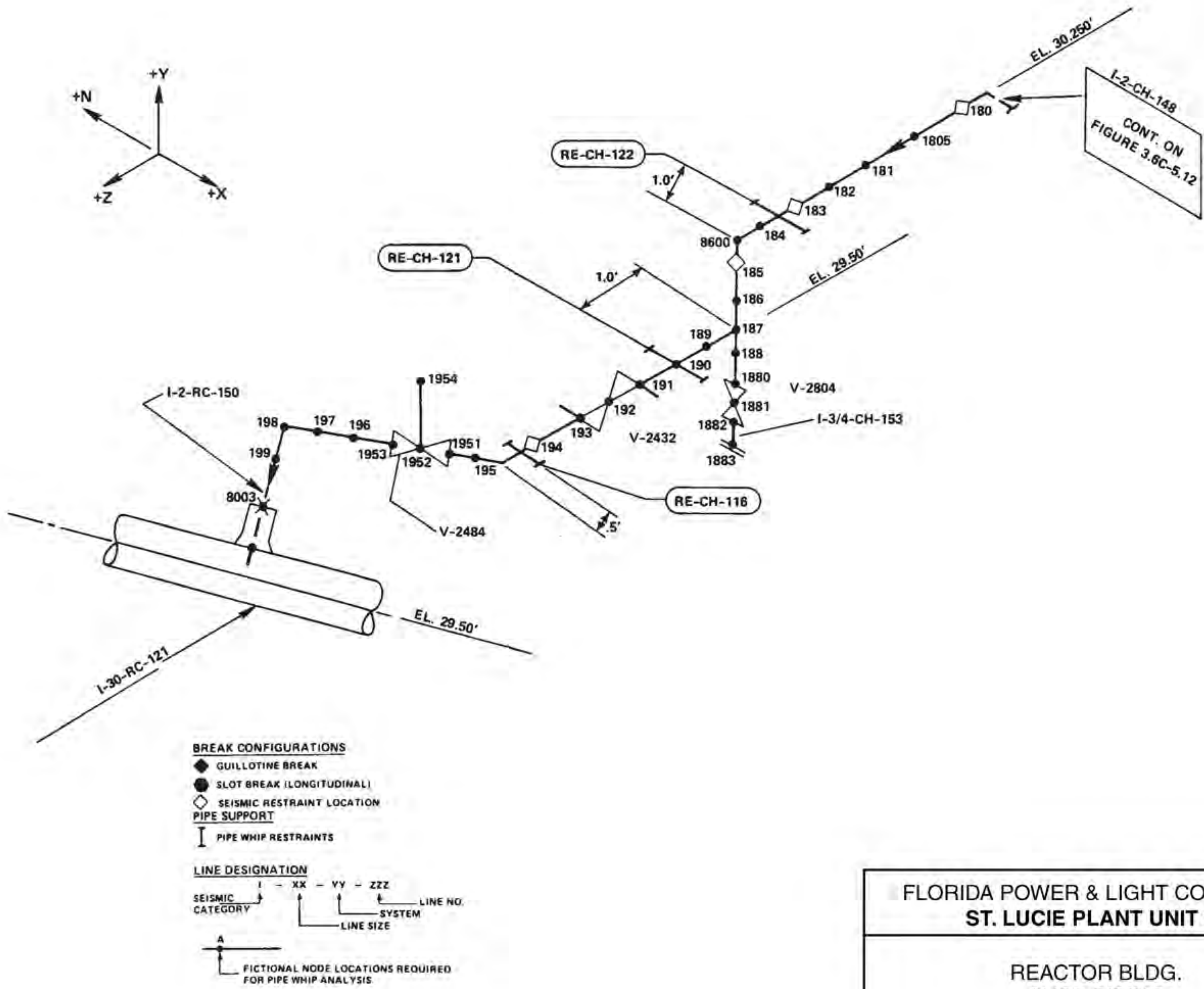
REACTOR BLDG.
CVCS PIPING

FIGURE 3.6C-4.11



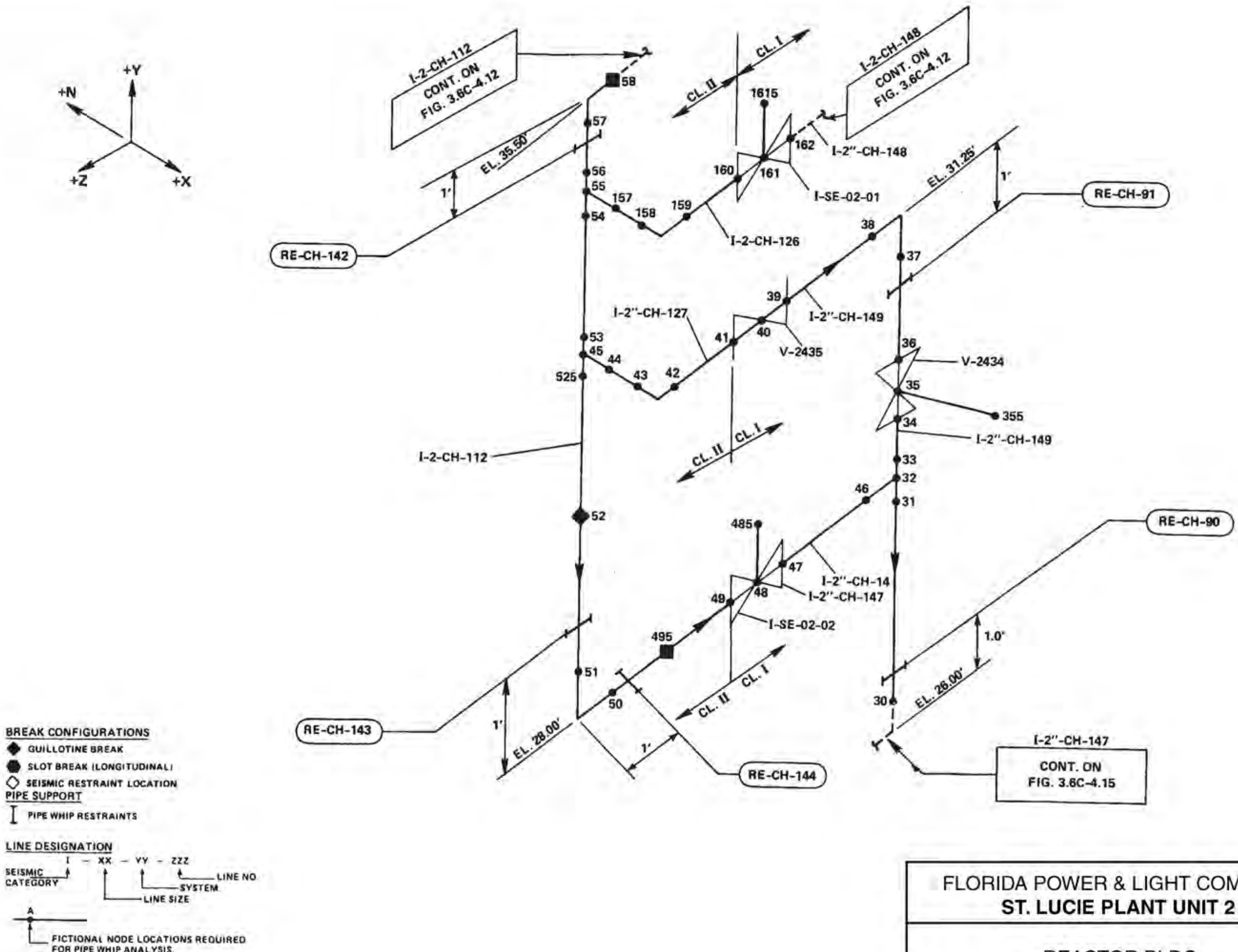
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
CVCS PIPING
FIGURE 3.6C-4.12



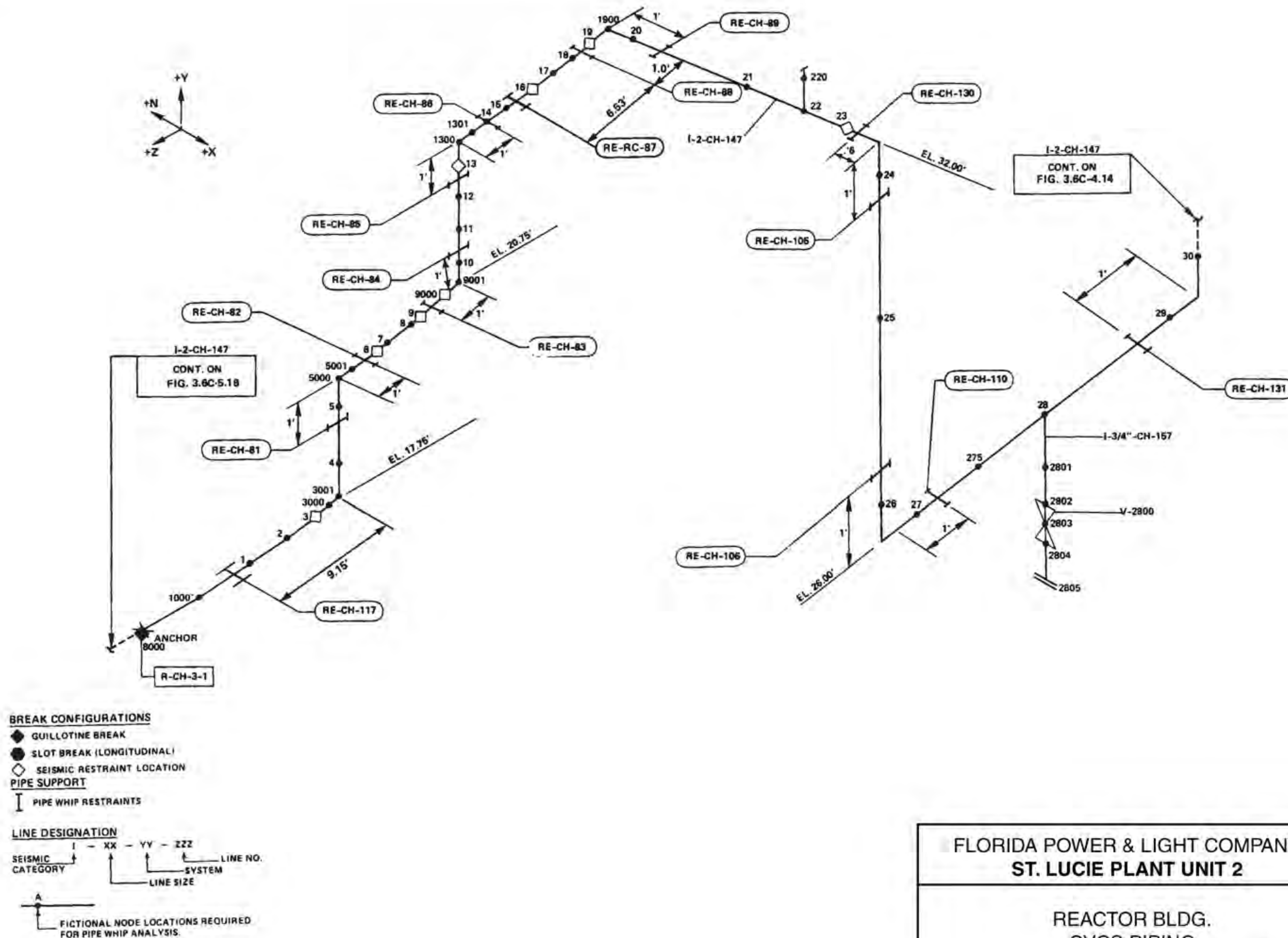
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
CVCS PIPING
FIGURE 3.6C-4.13



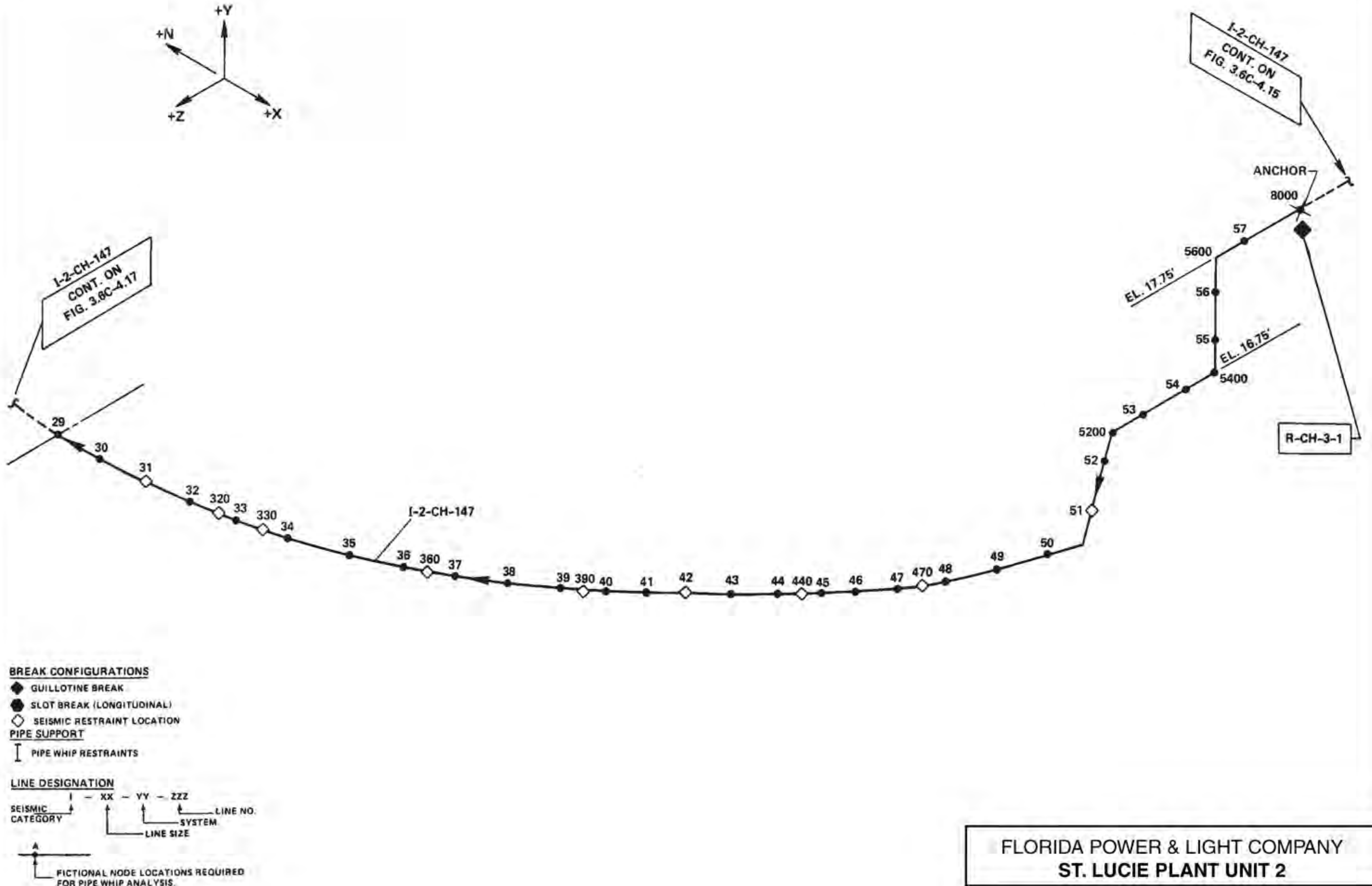
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
CVCS PIPING
FIGURE 3.6C-4.14



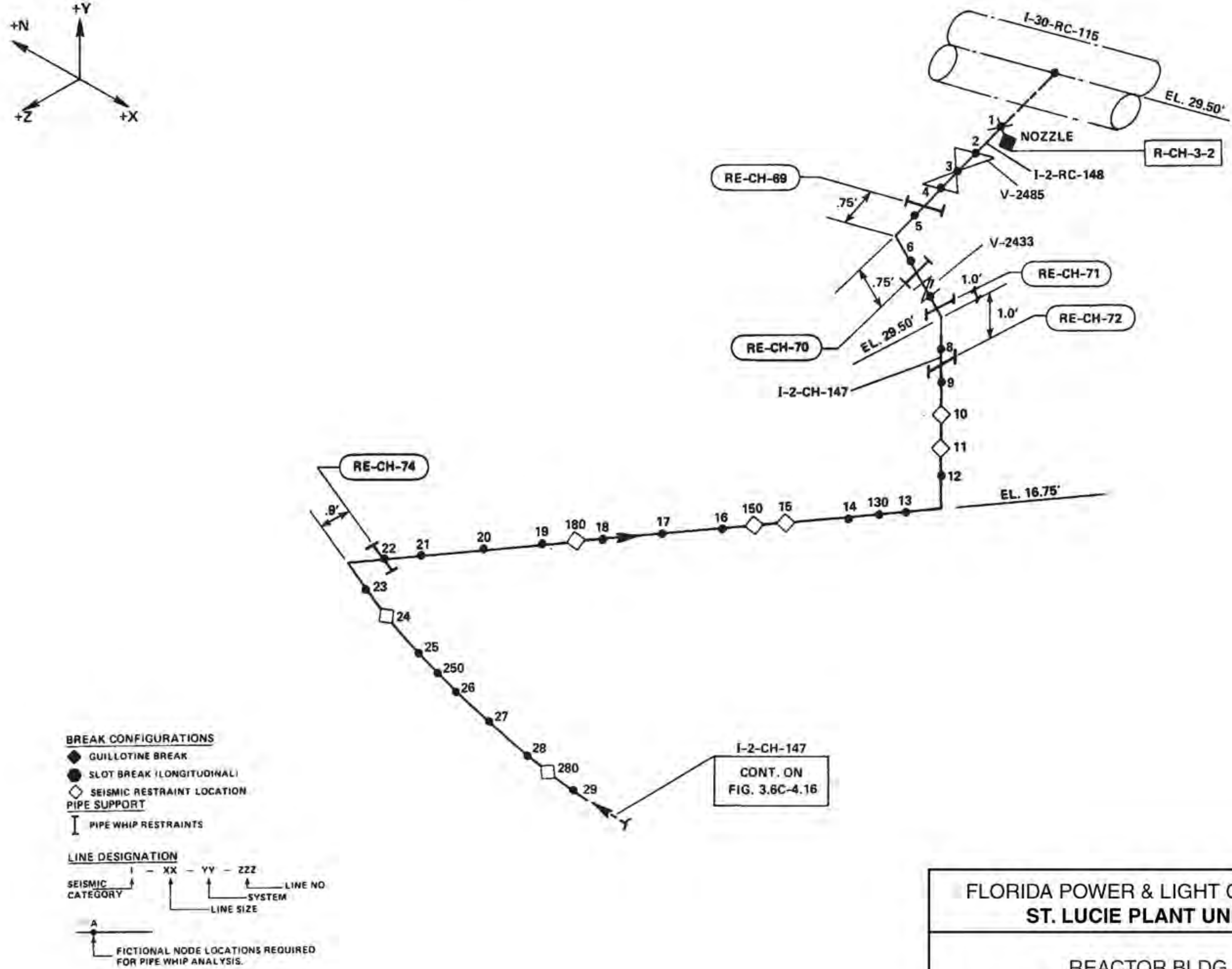
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
CVCS PIPING
FIGURE 3.6C-4.15



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG.
MISCELLANEOUS PIPING
FIGURE 3.6C-4.16



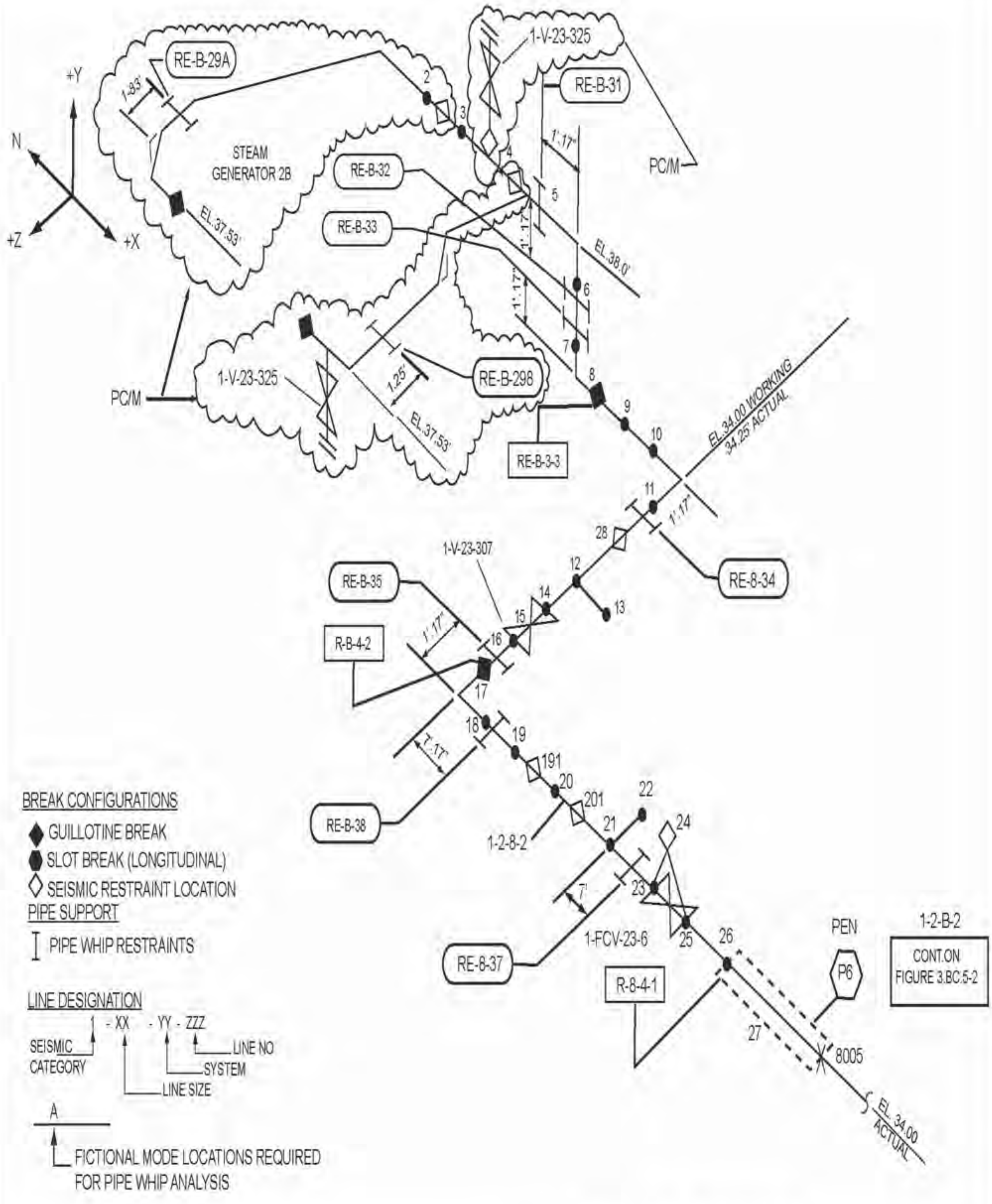
FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 2

REACTOR BLDG.

MISCELLANEOUS PIPING

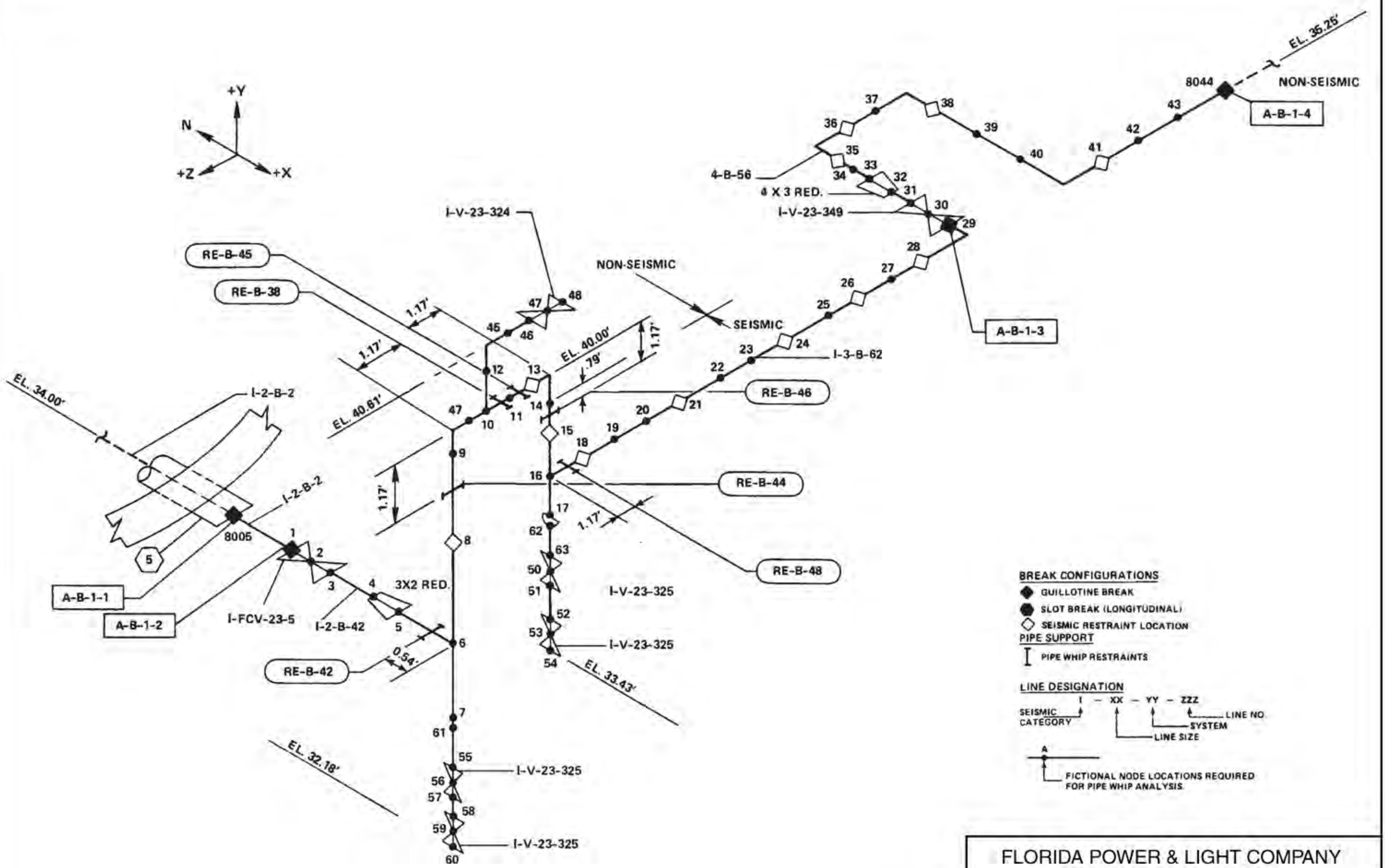
FIGURE 3.6C-4.17



Amendment No. 18 (01/08)

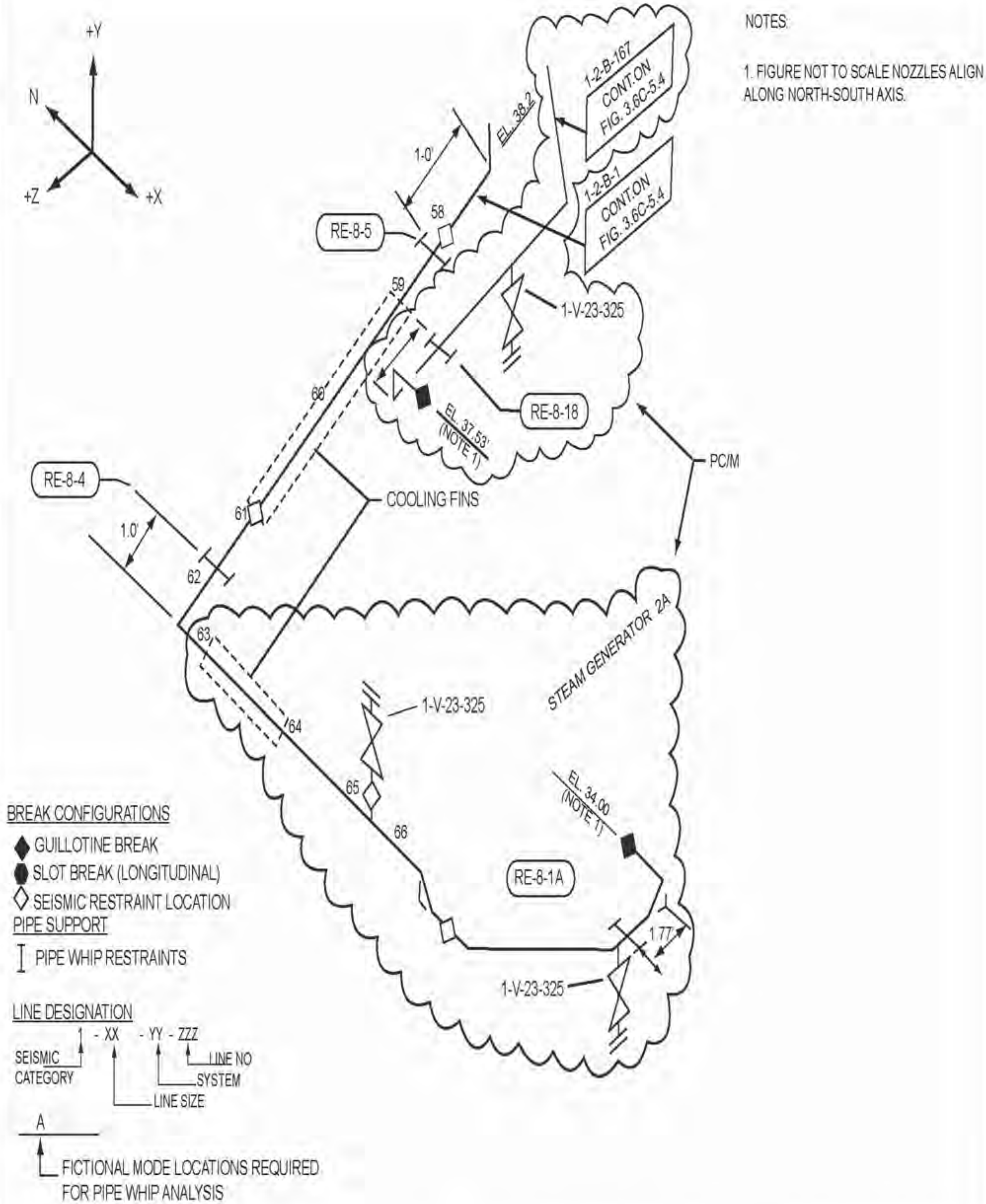
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR CONTAINMENT BLDG.
BLOWDOWN
BREAK LOCS. & PIPE WHIP RESTRAINTS
FIGURE 3.6C-5.1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

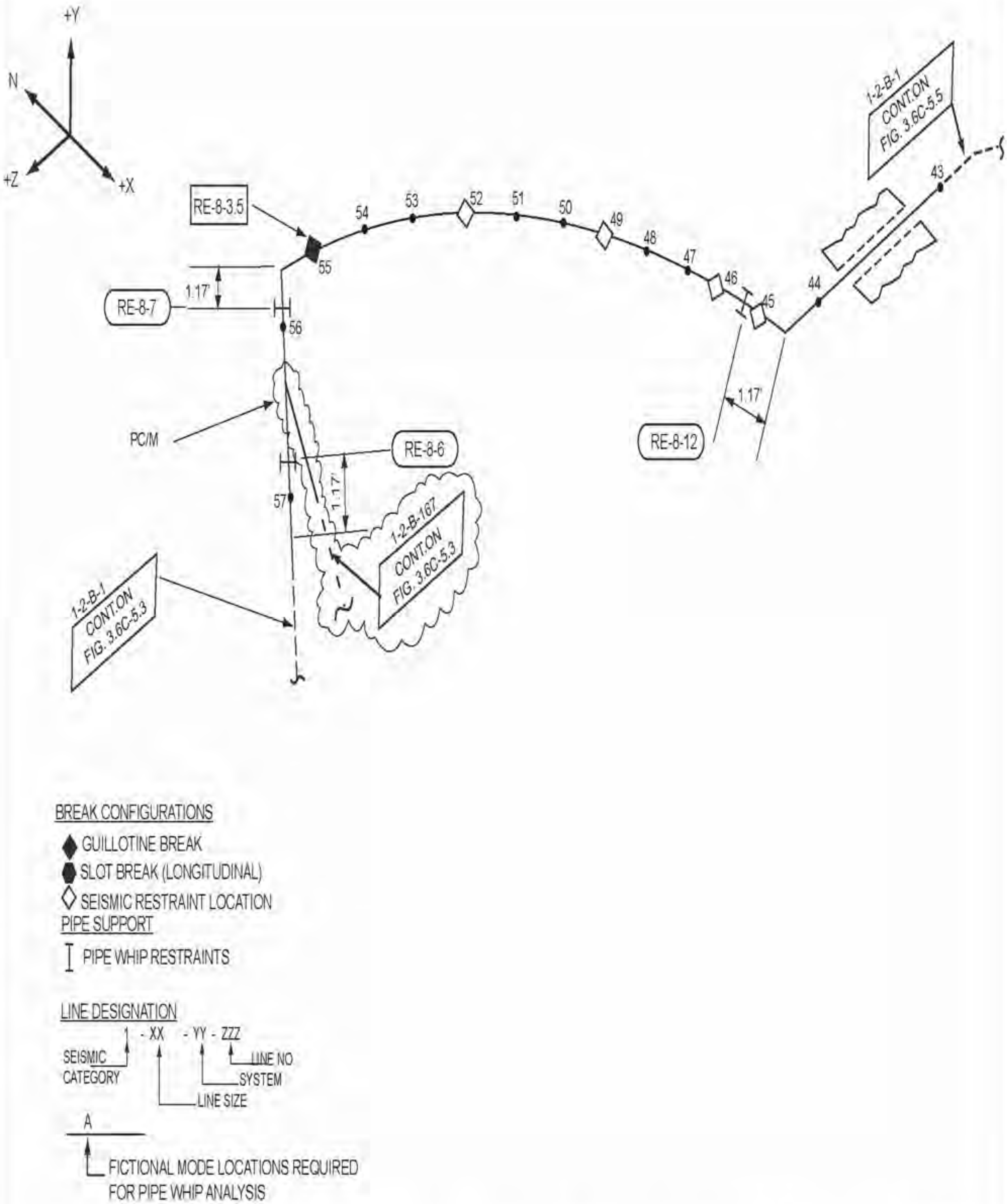
REACTOR AUX. BLDG.
BLOWDOWN
BREAK LOCS. & PIPE WHIP RESTRAINTS
FIGURE 3.6C-5.2



Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

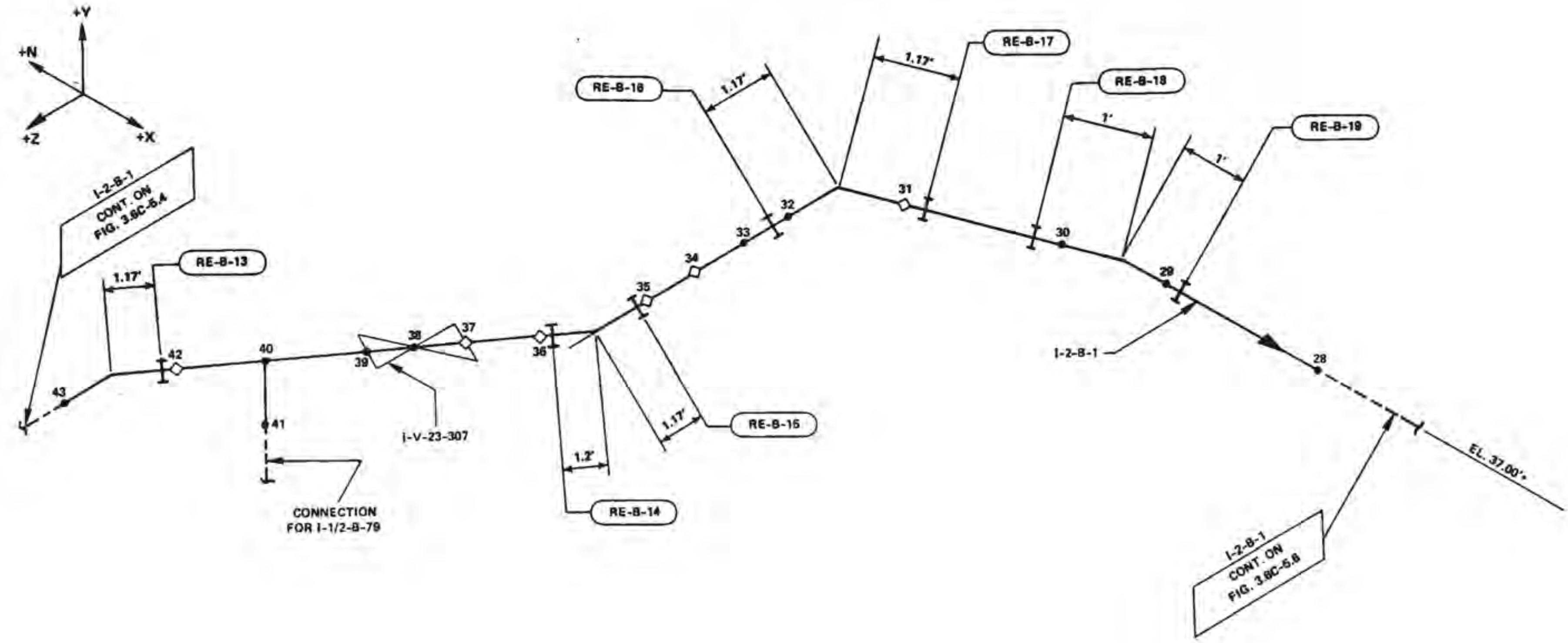
REACTOR CONTAINMENT BLDG.
BLOWDOWN
BREAK LOCS. & PIPE WHIP RESTRAINTS
FIGURE 3.6C-5.3



Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR CONTAINMENT BLDG.
BLOWDOWN
BREAK LOCS. & PIPE WHIP RESTRAINTS
FIGURE 3.6C-5.4



BREAK CONFIGURATIONS

- GUILLOTINE BREAK
- SLOT BREAK (LONGITUDINAL)
- SEISMIC RESTRAINT LOCATION

PIPE SUPPORT

- PIPE WHIP RESTRAINTS

LINE DESIGNATION

I - XX - YY - ZZZ

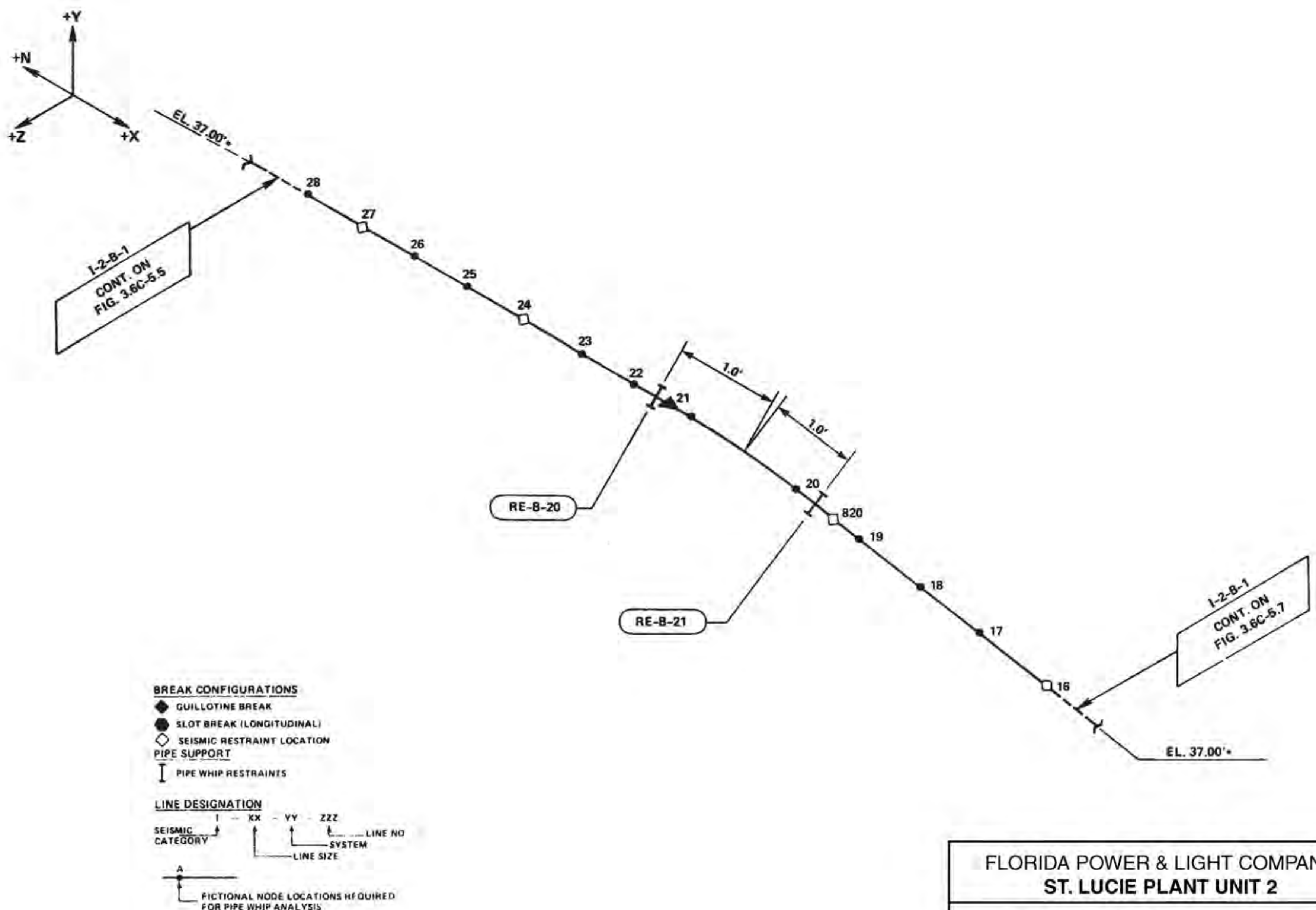
SEISMIC CATEGORY
LINE SIZE
SYSTEM
LINE NO.

A

FICTIONAL NODE LOCATIONS REQUIRED FOR PIPE WHIP ANALYSIS

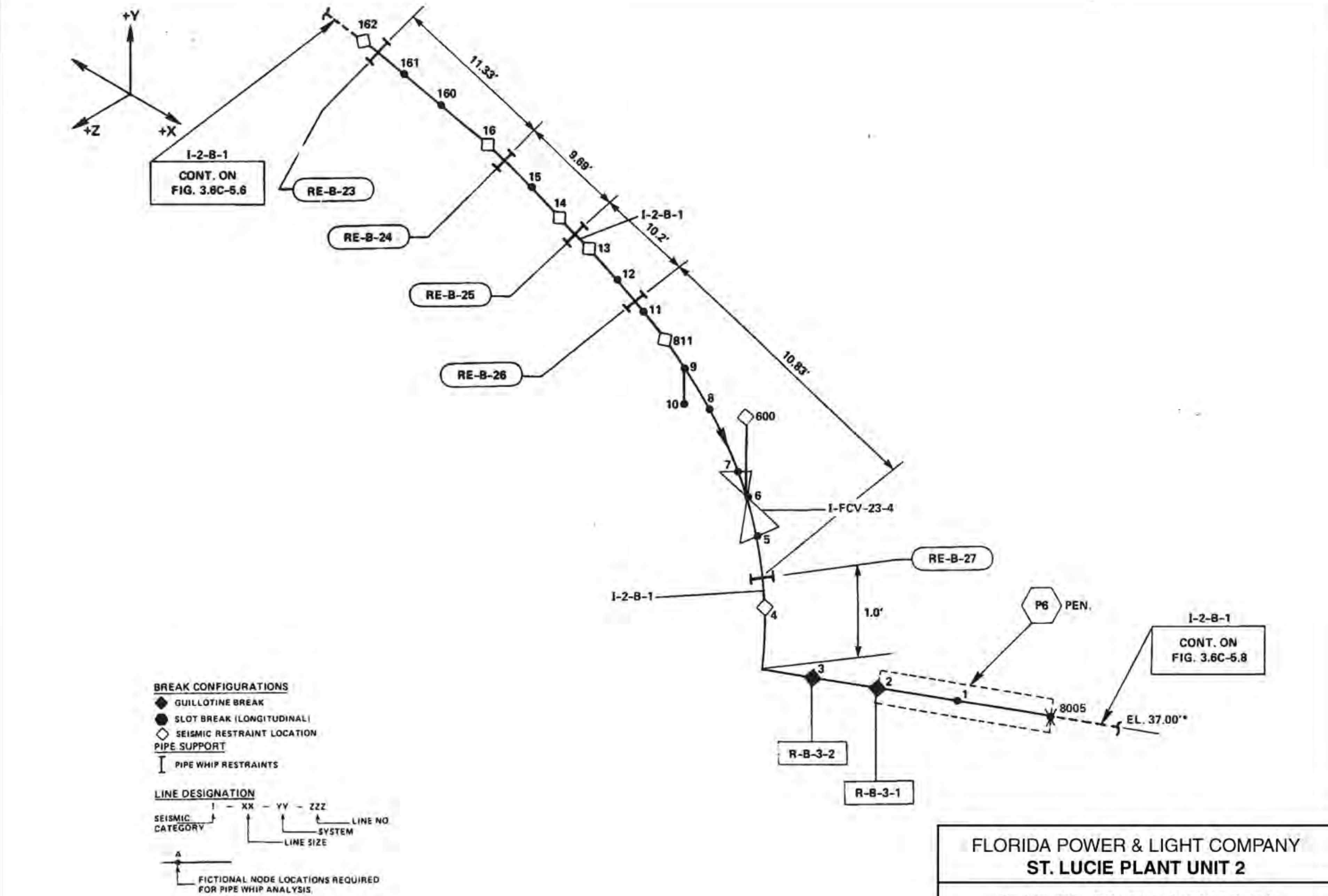
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR CONTAINMENT BLDG.
BLOWDOWN
BREAK LOCS. & PIPE WHIP RESTRAINTS
FIGURE 3.6C-5.5



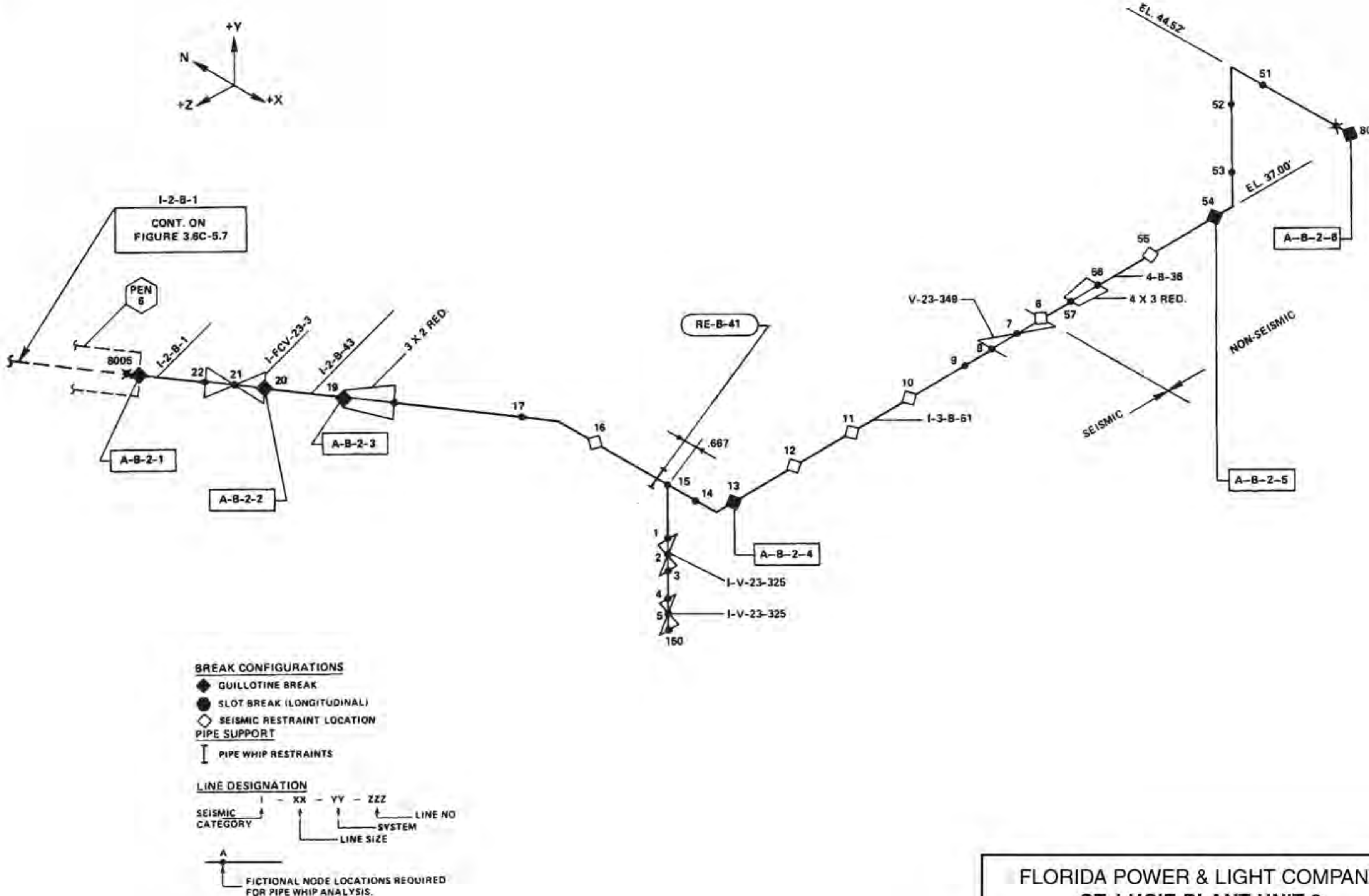
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR CONTAINMENT BLDG.
BLOWDOWN
BREAK LOCS. & PIPE WHIP RESTRAINTS
FIGURE 3.6C-5.6



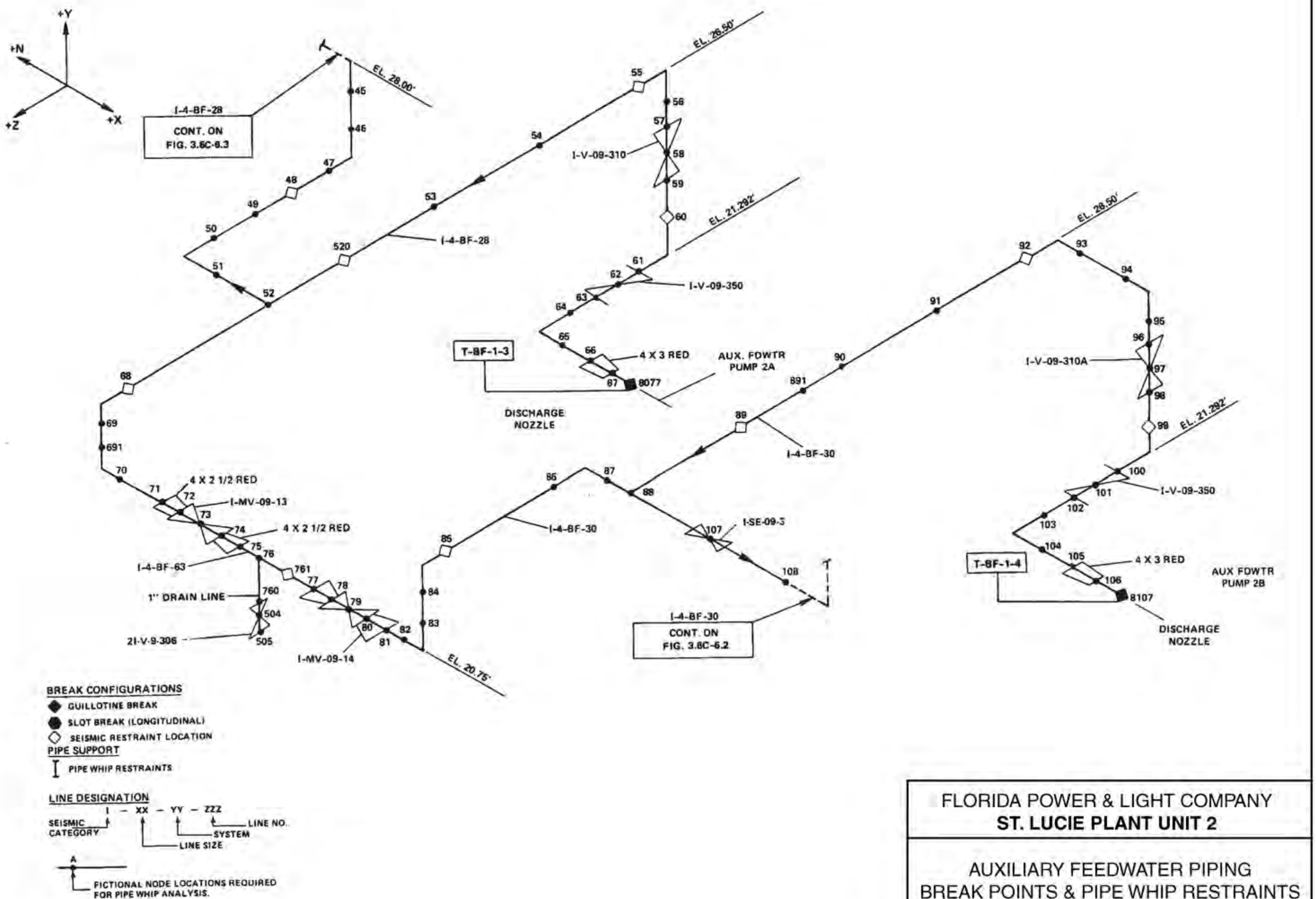
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR CONTAINMENT BLDG,
BLOWDOWN
BREAK LOCS. & PIPE WHIP RESTRAINTS
FIGURE 3.6C-5.7



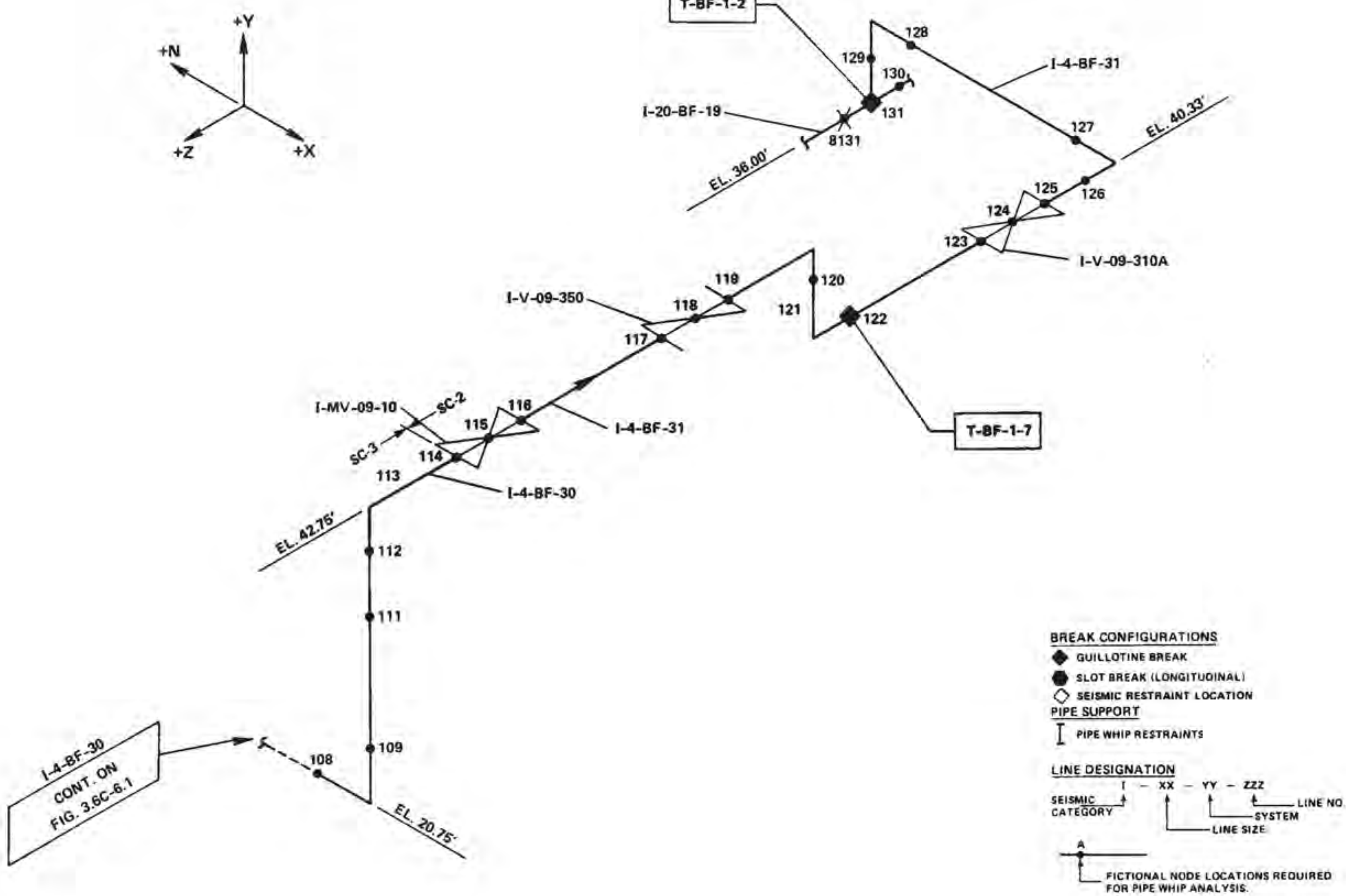
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUX. BLDG.
BLOWDOWN
BREAK LOCS. & PIPE WHIP RESTRAINTS
FIGURE 3.6C-5.8



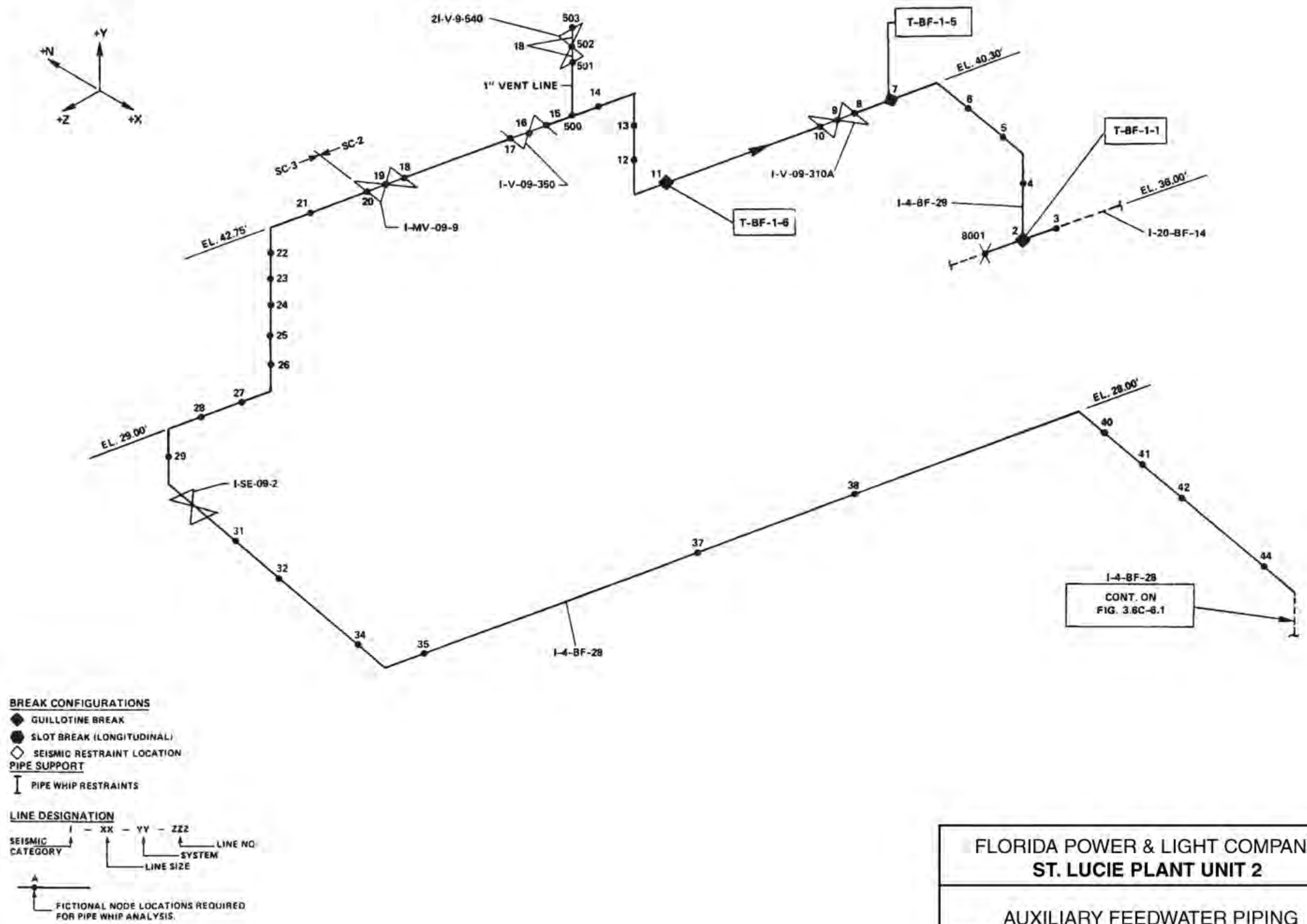
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

AUXILIARY FEEDWATER PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-6.1



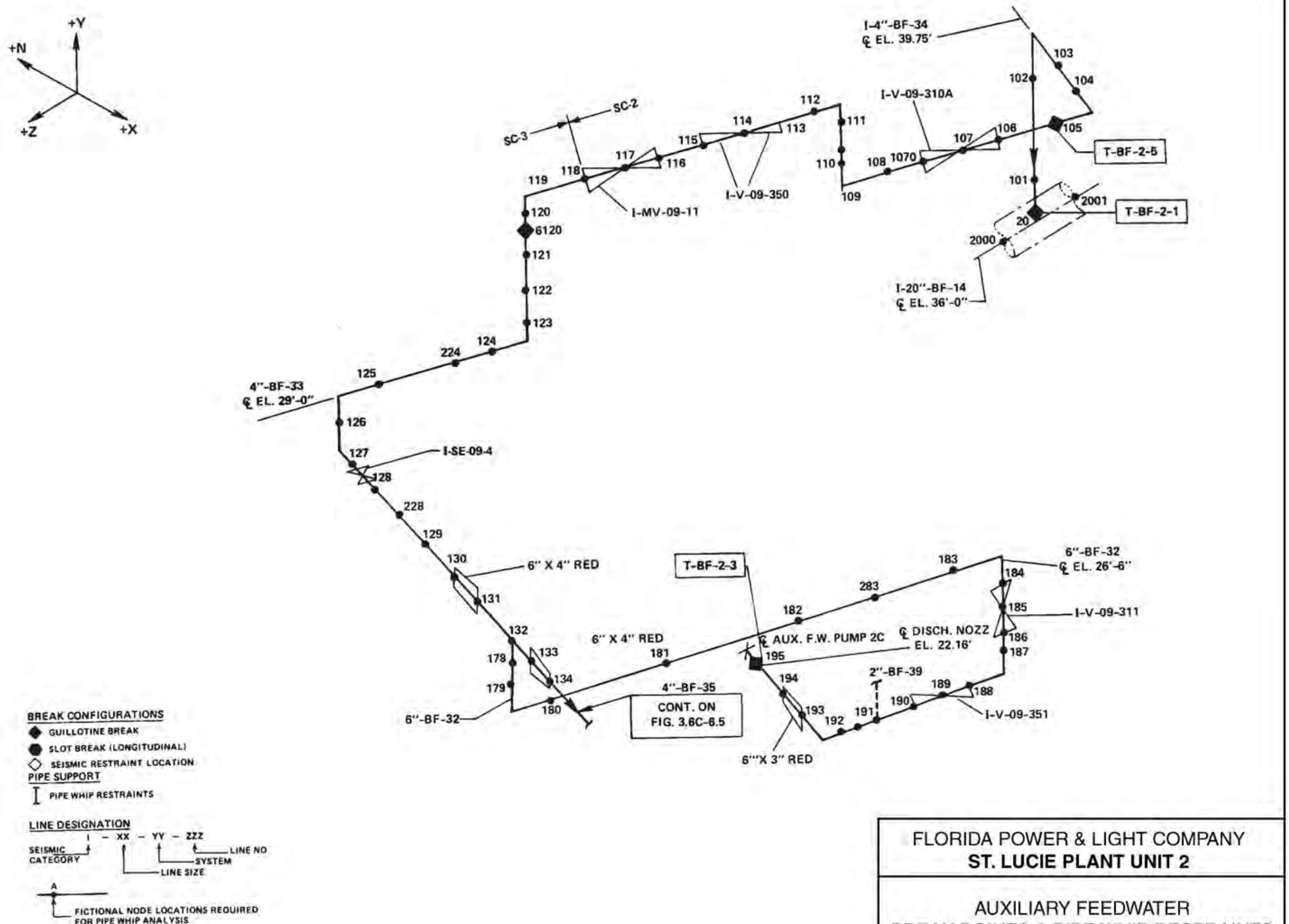
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

AUXILIARY FEEDWATER PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-6.2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

AUXILIARY FEEDWATER PIPING
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-6.3



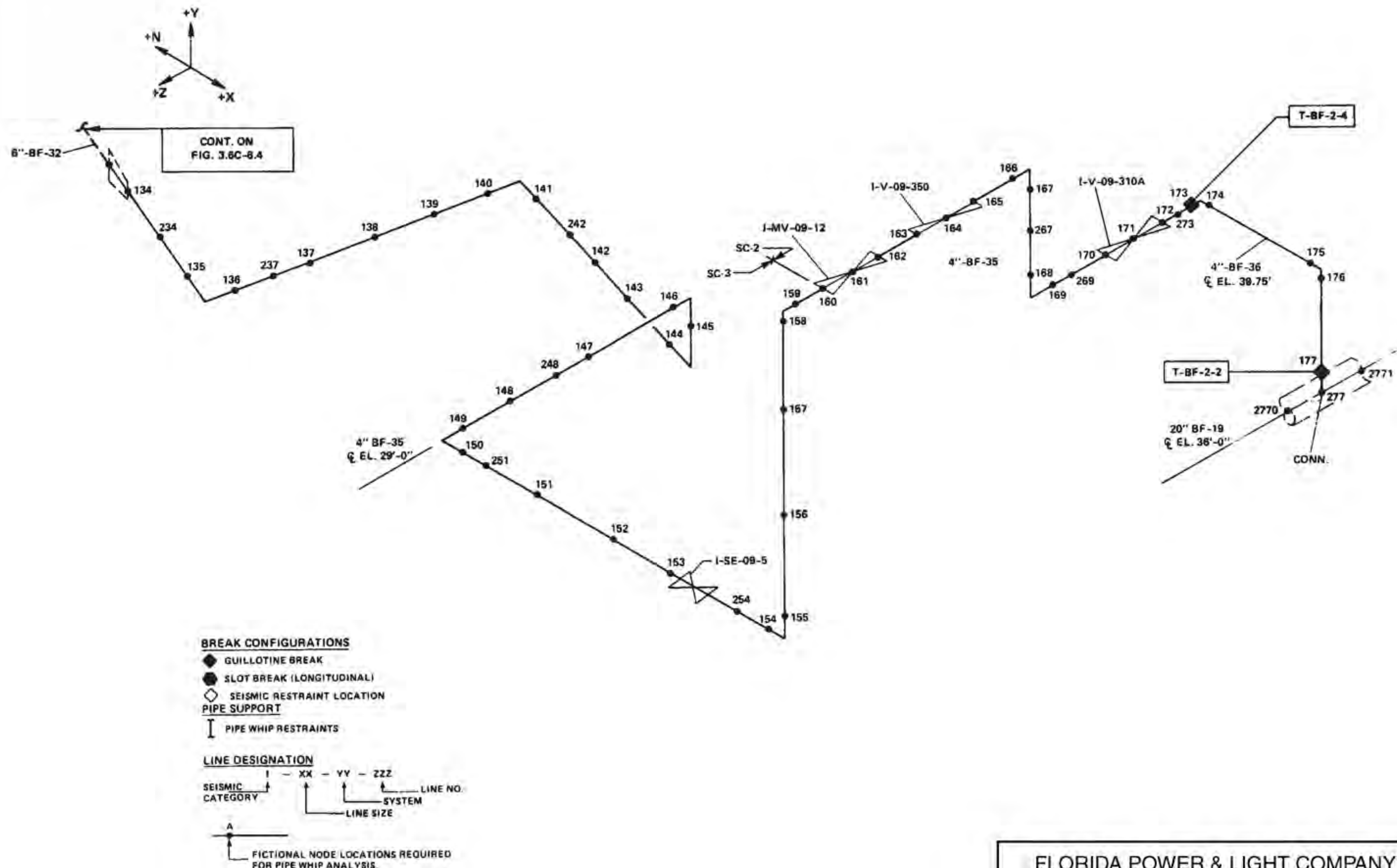
FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 2

AUXILIARY FEEDWATER

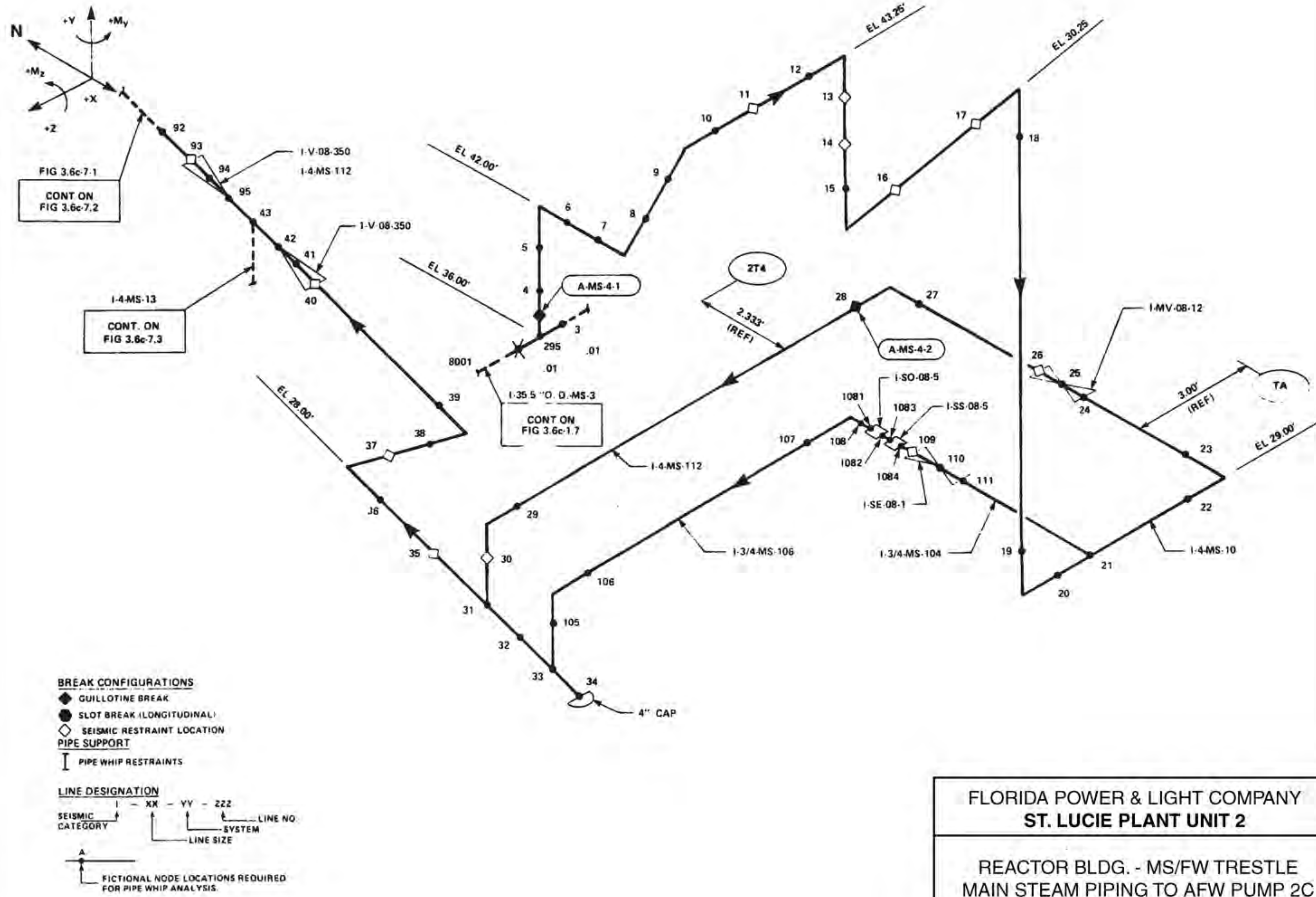
BREAK POINTS & PIPE WHIP RESTRAINTS

FIGURE 3.6C-6.4



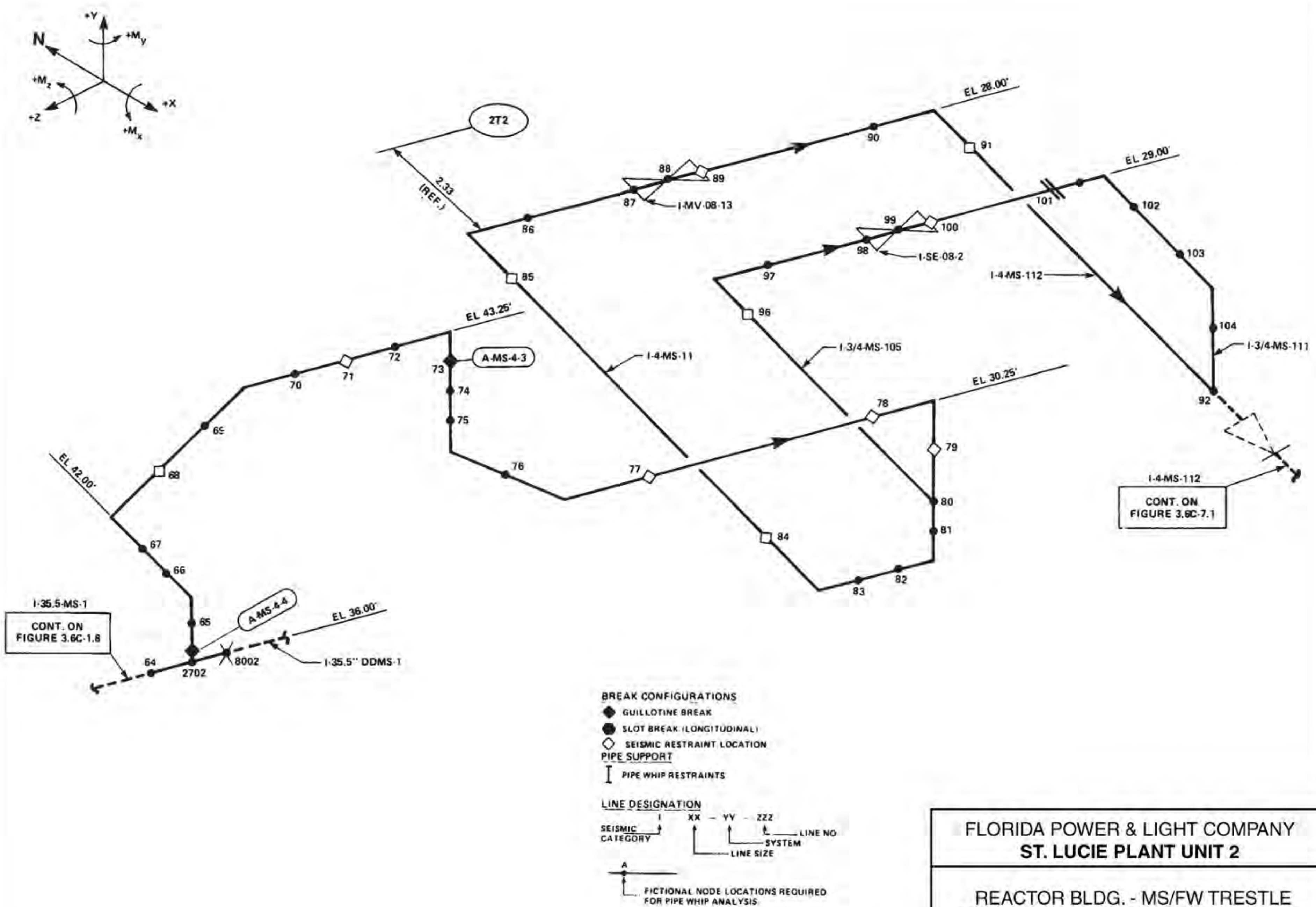
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

AUXILIARY FEEDWATER
BREAK POINTS & PIPE WHIP RESTRAINTS
FIGURE 3.6C-6.5



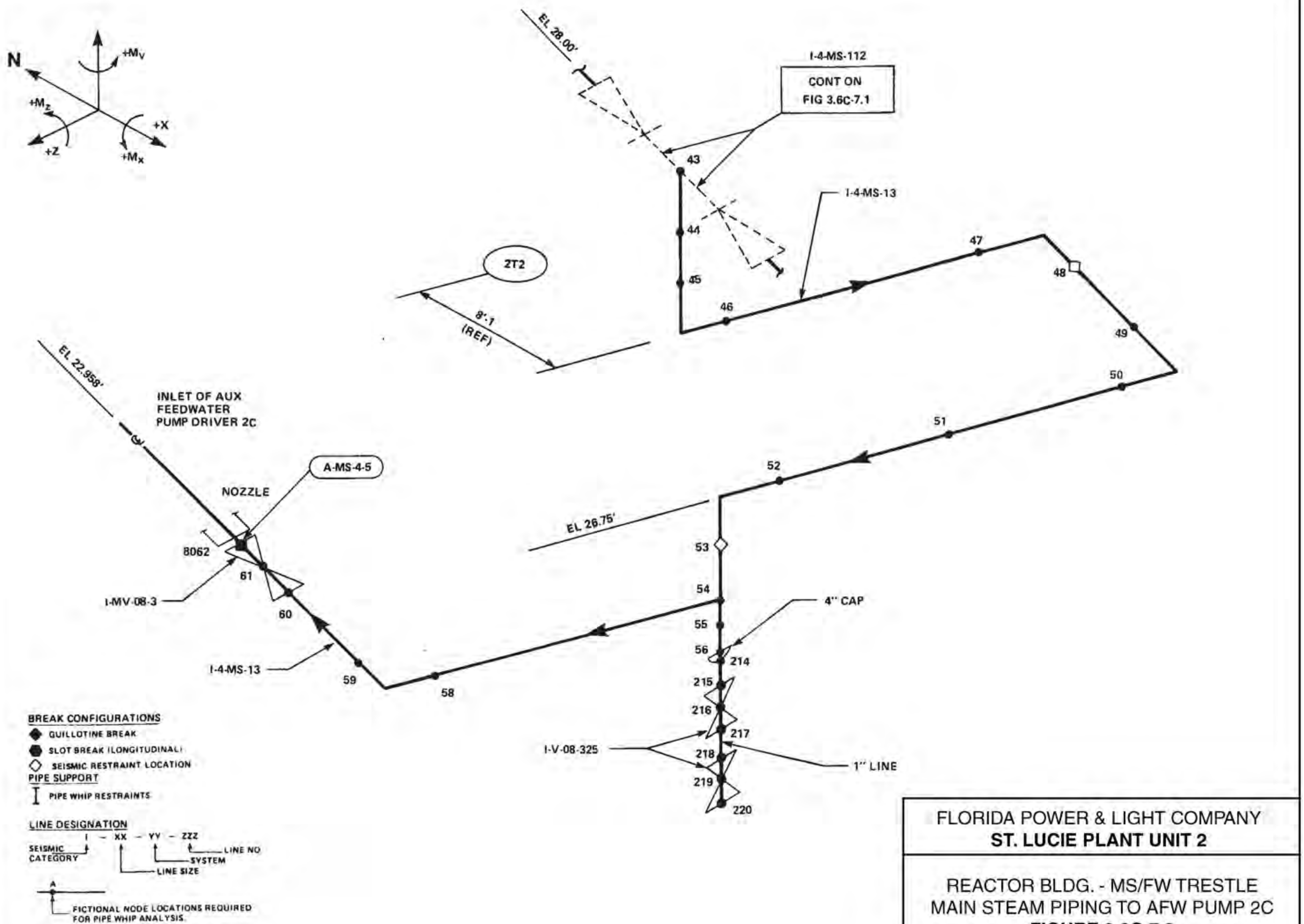
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG. - MS/FW TRESTLE
MAIN STEAM PIPING TO AFW PUMP 2C
FIGURE 3.6C-7.1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG. - MS/FW TRESTLE
MAIN STEAM PIPING TO AFW PUMP 2C
FIGURE 3.6C-7.2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BLDG. - MS/FW TRESTLE
MAIN STEAM PIPING TO AFW PUMP 2C
FIGURE 3.6C-7.3

APPENDIX 3.6D

STRUCTURAL DETAILS OF PIPE WHIP RESTRAINTS

Refer to Drawing
2998-G-799 SH 2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING PIPE
RESTRAINTS SH. 2

FIGURE 3.6D-1

Refer to Drawing
2998-G-799 SH 4

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2
REACTOR BUILDING PIPE
RESTRAINTS SH. 4

FIGURE 3.6D-2

Refer to Drawing
2998-G-799 SH 9

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING PIPE
RESTRAINTS SH. 9

FIGURE 3.6D-3

Refer to Drawing
2998-G-799 SH 9

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING PIPE
RESTRAINTS SH. 9

FIGURE 3.6D-4

APPENDIX 3.6E

MAIN STEAM & FEEDWATER ANALYSIS

3.6E MAIN STEAM & FEEDWATER ANALYSIS

This appendix presents the results of dynamic analyses performed to verify the structural adequacy of the Main Steam and Feedwater pipe whip restraints.

The transient forces resulting from postulated piping failures have been generated using the RELAP computer code. Thrust forces from RELAP are used as input to the PLAST 2267 Code. The PLAST code uses these forces to determine the pipe whip restraint reaction loads by performing a dynamic structural analysis on a lumped mass parameter of the piping system. These computer codes are described in References 3.6E-1 through 3.6E-3.

Typical pipe whip restraint structures used on the Main Steam and Feedwater systems are shown in Figures 3.60-1 and 3.60-2. Elastic stiffness values for these restraints have been determined by two methods:

a) The bolts securing the restraints to the RCB structure were considered infinitely rigid. Credit was taken only for the elasticity of the pipe whip restraint structural steel.

b) The elasticity of the bolts was considered and a combined stiffness was calculated.

Method 2 results in stiffness values approximately 36 percent to 57 percent of the values obtained by Method 1. In addition, pullout loads have been determined for the bolt system.

Reaction loads at the pipe whip restraints have been generated by the PLAST program based on the combined stiffness values. The resulting loads were then compared with the bolt pullout loads. Typical results are tabulated in Table 3.6E-1 for selected Main Steam restraints and show that the restraints perform their design function since the reaction loads are below the bolt pullout loads.

The break locations analyzed are shown in Figure 3.6E-1 for Main Steam piping and Figure 3.6E-2 for Feedwater. These figures also represent the PLAST models for these systems. Figures 3.6E-3 and 3.6E-4 give the RELAP models for the Main Steam and Feedwater piping, respectively. The volume and junction data for Main Steam RELAP are given in Tables 3.6E-2 and 3.6E-3. The pipe whip restraint gaps used in PLAST for Main Steam are given in Table 3.6E-4. This information is tabulated for the Feedwater analyses in Tables 3.6E-5 through 3.6E-7, respectively.

Force versus time history for RELAP is presented for a typical Main Steam break in Figure 3.6E-5 and for a typical Feedwater break in Figure 3.6E-6.

The following break locations and pipe whip restraints depicted on Figure 3.6E-1 and Table 3.6E-4 are provided for historical purposes since the dynamic effects associated with the specified pipe breaks and the pipe whip restraints have been eliminated from the plant design basis based upon NRC Generic Letter 87-11 as described in UFSAR Sections 3.6 and 3.6A.1.1:

Figure 3.6E-1 Pipe Whip Restraints RE-MS-11 (Fig. note) & RE-MS-14
Guillotine Breaks at RE-MS-12 & RE-MS-13

Table 3.6E-4 Pipe Whip Restraint RE-MS-14

REFERENCES: SECTION 3.6E

3.6E-1 "RELAP3 A Computer Program For Reactor Blowdown Analysis" by W H Rettig, G A Jayne, K V Moore, C E Slater, M L Uptmore, Idaho Nuclear Corporation IN-1321 Issued June, 1970, Reactor Technology, TDD-4500

3.6E-2 RELAP4-MOD6, A Computer Code For Transient Thermal Hydraulic Analysis of Nuclear Reactor and Related Systems, User's Manual, EG&G Idaho, Inc., CDAPTR003, January, 1978

3.6E-3 "Design Considerations for the Protection From the Effects of Pipe Rupture", ETR-1002-P (Proprietary Version) and ETR-1002 (non-Proprietary version), by Ebasco Services, Inc. August 1977.

TABLE 3.6E-1

SUMMARY OF SELECTIVE PIPE WHIP RESTRAINTS AND DYNAMIC LOADS

RESTRAINT NUMBER	RESTRAINT STRUCTURE STIFFNESS[1] (KIP/IN) PULLOUT (+x1) (-x1)	RESTRAINT STRUCTURE STIFFNESS[1] (KIP/IN) SHEAR (x2) (x3)	RESTRAINT STRUCTURE STIFFNESS[2] (KIP/IN) PULLOUT (x1)	RESTRAINT STRUCTURE STIFFNESS[2] (KIP/IN) SHEAR (x2)	BOLT PULL-OUT LOAD [3] (KIPS)	RESTRAINT REACTION LOAD (KIPS)	SOURCE RUN	BREAK LOCATION	REMARK
RE-MS-16	+38,749	52,375	+16,202	+35,830	5,187	5,160	FL021J2MS	Slot break betn	
	-181,549	-38,993	-90,728	-25,547				Pts 9 & 6709	
RE-MS-17	+39,304	+37,106	+14,288	+23,103	3,021	542	FL021KLMS	Guill. at S/G	
	-117,548	-40,521	-81,754	-25,538				Nozzle	
RE-MS-20	+28,291	+39,126	+16,143	+32,025	4,795	3,520	FL021HMS	Guill. at Pt. 16	
	-51,075	-28,969	-71,780	-33,924				from Pt. 16	

Notes:

(1) Bolts rigid

(2) Bolts elastic

(3) Applied load at which first both failure occurs.

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TABLE 3.6E-2

VOLUME INFORMATION USED FOR RELAP 3/MOD 68 FLUID MODELS OF MAIN STEAM LINES

VOLUME NUMBER	INITIAL VOLUME PRESSURE (psia)	VOLUME HEIGHT (FT)	ELEVATION AT VOLUME BOTTOM (FT)	VOLUME (FT^3)
1	815.000	.7000E1	.8180E2	.5000E6
2	814.840	.4250E1	.8880E2	.4242E2
3	814.480	.2625E1	.9022E2	.9742E2
4	811.257	.43912E2	.4894E2	.2377E2
5	807.667	.5000E1	.4394E2	.2706E2
6	807.300	.2625E1	.4394E2	.3247E2
7	806.110	.3375E1	.4319E2	.1353E3
8	804.920	.2625E1	.4319E2	.3247E2
9	803.873	.3708E1	.4210E2	.1533E2
10	801.781	.3708E1	.3840E2	.1533E2
11	799.688	.3708E1	.3469E2	.1533E2
12	798.640	.2625E1	.3469E2	.4491E3
13	792.900	.2625E1	.3469E2	.2841E2
14	792.400	.4250E1	.3469E2	.2300E2
15	791.640	.1554E2	.3894E2	.1068E3
16	783.850	.2958E1	.5152E2	.1288E4
17	775.550	.2958E1	.5152E2	.1971E3
18	814.840	.4250E1	.8880E2	.4242E2
19	814.480	.2625E1	.9022E2	.9742E2
20	810.910	.4891E2	.4394E2	.2647E3
21	807.300	.2625E1	.4394E2	.3747E2
22	805.420	.3375E1	.4394E2	.2327E3
23	803.540	.2625E1	.4394E2	.3247E2
24	802.990	.1113E2	.3469E2	.4600E2
25	797.260	.2625E1	.3469E2	.4491E3
26	791.520	.2625E1	.3469E2	.2841E2
27	791.020	.4250E1	.3469E2	.2300E2
28	790.260	.1554E2	.3894E2	.1068E3
29	783.160	.2958E1	.5152E2	.1069E4
30	773.880	.9522E1	.4200E2	.1031E3
31	772.730	.2958E1	.4200E2	.1031E3
32	771.370	.9293E1	.4496E2	.5131E2
33	771.370	.1225E2	.4200E2	.9060E2
34	770.000	.6000E2	.5425E2	.1000E7
35	773.880	.9522E1	.4200E2	.1031E3
36	772.730	.2958E1	.4200E2	.1031E3
37	771.370	.9293E1	.4496E2	.5131E2
38	771.370	.1225E2	.4200E2	.9060E2
39	770.000	.6000E2	.5425E2	.1000E7

NOTE: The mixture level is equal to the volume height for all volume other than volume 39. The mixture level for volume 39 is taken as 20 ft.

TAB1E 3.6E-3

JUNCTION INFORMATION USED FOR RELAP 3/MOD 68 FLUID MODELS
OF MAIN STEAM LINES

FROM VOLUME NUMBER	TO VOLUME NUMBER	JUNCTION HEIGHT (FT.)	MINIMUM FLOW AREA (FT^2)	INITIAL FLOW (LBM./SEC.)	JUNCTION INERTIA (1/FT)
1	2	.8880E2	.6355E1	.15556E4	.5250E0
2	3	.9154E2	.6355E1	.15556E4	.2188E1
3	4	.9154E2	.5412E1	.15556E4	.5719E1
4	5	.4894E2	.5412E1	.15556E4	.4518E1
5	6	.4525E2	.5412E1	.15556E4	.1016E1
6	7	.4525E2	.5412E1	.15556E4	.2864E1
7	8	.4450E2	.5412E1	.15556E4	.2864E1
8	9	.4450E2	.5412E1	.15556E4	.8160E0
9	10	.4210E2	.5412E1	.15556E4	.5230E0
10	11	.3840E2	.5412E1	.15556E4	.5230E0
11	12	.3600E2	.5412E1	.15556E4	.7927E1
12	13	.3600E2	.5412E1	.15556E4	.8150E1
13	14	.3600E2	.5412E1	.15556E4	.8780E0
14	15	.3894E2	.5412E1	.15556E4	.1523E1
15	16	.5300E2	.6874E1	.15556E4	.1476E2
16	17	.5300E2	.6874E1	.15556E4	.1571E2
1	18	.8880E2	.6355E1	.15556E4	.5250E0
18	19	.9154E2	.6355E1	.15556E4	.2188E1
19	20	.9154E2	.5412E1	.15556E4	.6180E1
20	21	.4525E2	.5412E1	.15556E4	.5073E1
21	22	.4525E2	.5412E1	.15556E4	.4527E1
22	23	.4450E2	.5412E1	.15556E4	.4527E1
23	24	.4450E2	.5412E1	.1556E4	.1339E1
24	25	.3600E2	.5412E1	.1556E4	.8451E1
25	26	.3600E2	.5412E1	.1556E4	.8150E1
26	27	.3600E2	.5412E1	.1556E4	.8770E0
27	28	.3893E2	.5412E1	.1556E4	.1523E1
28	29	.5300E2	.6874E1	.1556E4	.1236E2
29	17	.5300E2	.6874E1	.1556E4	.1331E2
17	30	.5152E2	.6874E1	.1556E4	.3176E1
30	31	.4348E2	.6874E1	.1556E4	.2182E1
31	32	.4496E2	.3207E1	.7778E3	.3581E1
31	33	.4348E2	.3207E1	.7778E3	.5496E1
32	34	.5425E2	.3207E1	.7778E3	.2495E1

TABLE 3.6E-3 (Cont'd)

FROM VOLUME NUMBER	TO VOLUME NUMBER	JUNCTION HEIGHT (FT.)	MINIMUM FLOW AREA (FT^2)	INITIAL FLOW (LBM./SEC.)	JUNCTION INERTIA (1/FT.)
33	34	.5425E2	.3207E1	.7778E3	.4405E1
17	35	.5152E2	.6874E1	.1556E4	.3176E1
35	36	.4348E2	.6874E1	.1556E4	.2182E1
36	37	.4496E2	.3207E1	.7778E3	.3586E1
36	38	.4348E2	.3207E1	.7778E3	.5496E1
37	39	.5425E2	.3207E1	.7778E3	.2495E1
38	39	.5425E2	.3207E1	.7778E3	.4405E1
3	0	.9154E2	.5412E1	0.	.1663E1

NOTES:

(1) Moody multipliers and contraction coefficents are taken to be 1 for all junctions.

(2) The friction coefficient for the leak junction, i.e., between volume 3 and volume 0 is taken to be .19E-5. All other junctions have a friction factor taken to be 0.

(3) The leak junction in Table 3.6E-3 is assumed to open instantaneously to a full area equal to the inside area of the pipe at the leak area location.

(4) Table 3.6E-3 describes the RELAP 3/MOD 68 model appropriate for a guillotine break at node 4 of reference isometric MS-147-1. A valve between volumes 3 and 4 is instantaneously closed when the leak junction between volumes 3 and 0 instantaneously opened. The piping between steam generator 2A and the guillotine break is of interest in this particular case.

TABLE 3.6E-4

RESTRAINT GAPS USED FOR PLAST MODELS OF MAIN STEAM LINE

RESTRAINT NAME	LOCAL (1) +Y GAP (IN.)	LOCAL (1) -Y GAP (IN.)	LOCAL (1) +Z GAP (IN.)	LOCAL (1) -Z GAP (IN.)
RE-MS-24	4.00	4.00	4.00	4.00
RE-MS-21	4.25	4.00	4.00	4.25
RE-MS-20	4.50	4.00	4.00	5.50
RE-MS-19	4.00	5.25	4.00	4.50
RE-MS-18	4.00	4.25	5.75	4.00
RE-MS-17	4.00	4.50	100.0(2)	100.00(2)
RE-MS-16	4.00	4.00	4.00	6.00
RE-MS-15	5.75	4.00	4.75	4.00
RE-MS-14	6.25	4.00	99.625(2)	100.375(2)
RE-MS-13	5.75	4.00	4.00	6.500
RE-MS-12	4.00	6.75	4.00	5.750

NOTES:

(1) For local directions see Figure 3.6E-1

(2) Local ± Z is unrestrained for this restraint

TABLE 3.6E-5

VOLUME INFORMATION USED FOR RELAP 4/MOD 6 FLUID MODELS OF BOILER FEEDWATER LINE

VOLUME NUMBER	INITIAL VOLUME PRESSURE (psia)	VOLUME HEIGHT (FT)	ELEVATION AT VOLUME BOTTOM (FT)	VOLUME (FT^3)
1	844.59	1.345	67.692	3.548
2	847.66	9.945	57.749	13.897
3	855.09	22.498	35.253	38.171
4	860.14	1.495	35.253	70.421
5	861.51	1.495	35.253	93.741
6	862.02	1.417	35.292	14.062
7	863.46	1.417	35.292	105.796
8	862.33	1.417	35.292	20.411
9	860.84	1.417	48.235	10.489
10	861.40	1.417	48.235	32.838
11	862.38	1.417	48.235	29.963
12	864.06	1.417	48.235	11.302
13	865.87	1.417	48.235	72.279
14	1011.72	4.488	52.721	81.607
15	1156.93	15.876	55.792	120.598
16	1164.07	8.002	63.666	18.116
17	1169.16	8.017	55.650	83.580
18	1173.12	4.417	52.792	83.371
19	1189.57	29.668	22.875	168.551
20	852.07	1.345	67.692	3.549
21	855.15	9.945	57.749	13.879
22	862.58	22.498	35.253	38.171
23	867.72	1.495	35.253	104.563
24	869.17	1.495	35.253	92.771
25	869.28	1.417	35.292	14.062
26	870.48	1.417	35.292	40.477
27	868.13	19.417	35.292	28.386
28	865.91	1.417	35.292	54.801
29	867.12	1.417	35.292	69.060
30	868.94	1.417	35.292	11.302
31	870.73	1.417	35.292	66.462
32	1013.35	1.985	55.276	68.517
33	1155.83	15.876	55.792	108.635
34	1162.4 2	8.002	63.666	10.666
35	1170.93	8.017	55.650	57.112
36	1174.77	4.417	52.792	57.548
37	1191.13	29.668	22.875	168.551
38	1154.21	1.917	55.542	146.825

TABLE 3.6E-5 (Cont'd)

VOLUME NUMBER	INITIAL VOLUME PRESSURE (psia)	VOLUME HEIGHT (FT)	ELEVATION AT VOLUME BOTTOM (FT)	VOLUME (FT^3)
39	1176.10	1.918	52.541	18.761
40	813.53	58.000	30.797	332.000
41	809.17	58.00	30.797	332.000
42	1010.72	4.488	48.235	460.642
43	1013.35	1.986	53.292	74.374

TABLE 3.6E-6

JUNCTION INFORMATION USED FOR RELAP 4/MOD 6 FLUID MODELS OF BOILER FEEDWATER LINE

JUNCTION NUMBER	FROM VOLUME NUMBER	TO VOLUME NUMBER	JUNCTION HEIGHT (FT)	MINIMUM FLOW AREA (FT^2)	INITIAL FLOW LBM/SEC	FORWARD LOSS CO-EFFICIENT	REVERSE LOSS CO-EFFICIENT
1							
2	2	1	67.693	1.419	1581.6	0.363	0.363
3	3	2	57.750	1.419	1581.6	0.133	0.133
4	4	3	36.000	1.755	1581.6	0.570	0.570
5	5	4	36.000	1.755	1581.6	0.570	0.570
6	6	5	36.000	1.577	1581.6	0.106	0.142
7	7	6	36.000	1.577	1581.6	0.480	0.480
8	8	7	36.000	1.577	1581.6	0.363	0.363
9	9	8	48.943	1.577	1581.6	0.363	0.363
10	10	9	48.943	1.577	1581.6	0.194	0.194
11	11	10	48.943	1.577	1581.6	0.363	0.363
12	12	11	48.943	1.577	1581.6	0.733	0.733
13	13	12	48.943	1.577	1581.6	0.733	0.733
14	42	13	48.943	1.577	1581.6	69.342	69.342
15	38	14	56.500	1.577	1581.6	69.160	69.160
16	15	38	56.500	1.577	1581.6	1.127	1.104
17	16	15	71.667	1.577	1581.6	3.441	3.585
18	17	16	63.667	2.264	1581.6	5.644	5.644
19	18	17	56.500	1.577	1581.6	0.914	1.133
20	39	18	53.500	1.577	1581.6	0.656	0.875
21	19	39	52.542	1.577	1581.6	4.070	3.978
22	37	39	52.542	1.577	1581.6	4.070	3.978
23	20	41	68.365	1.419	1581.6	4.070	3.978
24	21	20	67.693	1.419	1581.6	0.363	0.363
25	22	21	57.750	1.419	1581.6	0.182	0.000
25	23	22	36.000	1.755	1581.6	0.570	0.570
27	24	20	36.000	1.755	1581.6	0.570	0.570
28	25	24	36.000	1.577	1581.6	0.106	0.142
29	26	25	36.000	1.577	1581.6	0.480	0.000
30	27	26	36.000	1.577	1581.6	0.363	0.363
31	28	27	54.000	1.577	1581.6	0.363	0.363
32	29	28	54.000	1.577	1581.6	0.363	0.363
33	30	29	54.000	1.577	1581.6	0.733	0.733
34	31	30	54.000	1.577	1581.6	0.733	0.733
35	43	31	54.000	1.577	1581.6	68.752	68.752

TABLE 3.6.E-6 (Cont'd)

JUNCTION NUMBER	FROM VOLUME NUMBER	TO VOLUME NUMBER	JUNCTION HEIGHT (FT)	MINIMUM FLOW AREA (FT^2)	INITIAL FLOW LBM/SEC	FORWARD LOSS CO-EFFICIENT	REVERSE LOSS CO-EFFICIENT
36	38	33	56.500	1.577	1581.6	68.570	18.134
37	33	38	56.500	1.577	1581.6	1.127	1.104
38	34	33	71.667	1.577	1581.6	3.755	3.585
39	35	34	63.667	2.264	1581.6	5.644	5.644
40	36	35	56.500	1.577	1581.6	0.914	1.113
41	39	36	53.500	1.577	1581.6	0.565	0.875
42	14	42	52.722	1.577	1581.6	0.096	0.096
43	32	43	55.277	1.577	1581.6	0.096	0.096
44	0	19	23.538	1.000	1.0	0.000	0.000
45	0	37	23.538	1.000	1.0	0.000	0.000

TABLE 3.6E-7

RESTRAINT GAPS USED FOR PLAST MODELS OF BOILER FEEDWATER LINE

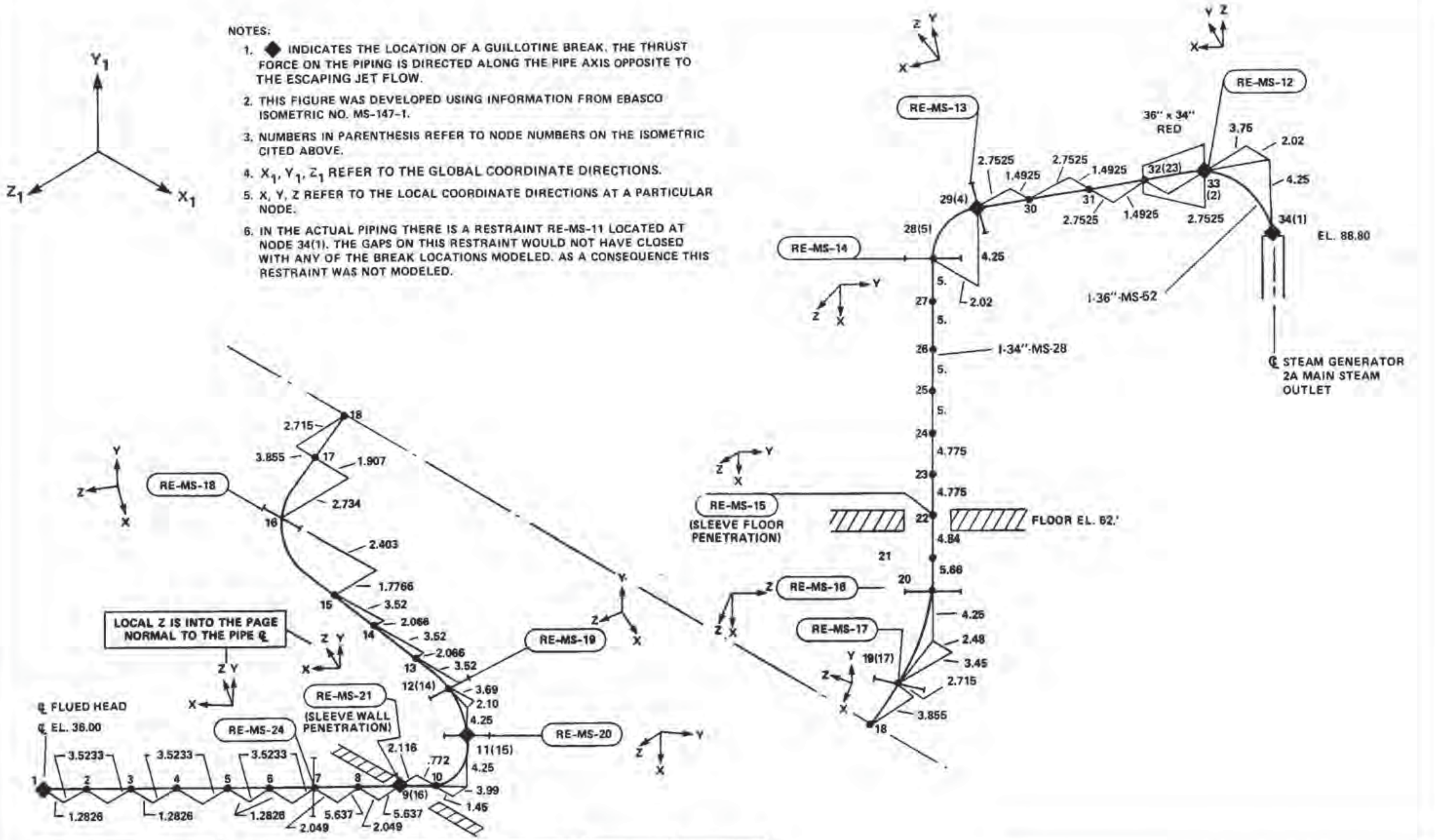
RESTRAINT NAME	LOCAL (1) +Y GAP (IN.)	LOCAL (1) -Y GAP (IN.)	LOCAL (1) +Z GAP (IN.)	LOCAL (1) -Z GAP (IN.)
RE-BF-17	4.00	4.00	4.00	4.50
RE-BF-16	4.00	4.00	4.00	4.50
RE-BF-15	4.00	4.00	4.75	4.00
RE-BF-14	4.00	4.00	4.00	4.25
RE-BF-11	9.00	9.00	9.00	9.00
RE-BF-10	6.50	4.00	4.25	4.00

Note:

1. For local directions (see Figure 3.6E-2)

NOTES:

1. ◆ INDICATES THE LOCATION OF A GUILLOTINE BREAK. THE THRUST FORCE ON THE PIPING IS DIRECTED ALONG THE PIPE AXIS OPPOSITE TO THE ESCAPING JET FLOW.
2. THIS FIGURE WAS DEVELOPED USING INFORMATION FROM EBASCO ISOMETRIC NO. MS-147-1.
3. NUMBERS IN PARENTHESIS REFER TO NODE NUMBERS ON THE ISOMETRIC CITED ABOVE.
4. X_1, Y_1, Z_1 REFER TO THE GLOBAL COORDINATE DIRECTIONS.
5. X, Y, Z REFER TO THE LOCAL COORDINATE DIRECTIONS AT A PARTICULAR NODE.
6. IN THE ACTUAL PIPING THERE IS A RESTRAINT RE-MS-11 LOCATED AT NODE 34(1). THE GAPS ON THIS RESTRAINT WOULD NOT HAVE CLOSED WITH ANY OF THE BREAK LOCATIONS MODELED. AS A CONSEQUENCE THIS RESTRAINT WAS NOT MODELED.



LINE NO.	PIPE O.D.	SCH. NOM. WL. THICKNESS	MATERIAL	OPERATING TEMP. °F	MODULUS OF ELASTICITY @ OP. TEMP. PSI	POISSON'S RATIO	OPERATING PRESSURE PSIG
I-34"-MS-28	34.000"	1.255	C.S.	532	26.26×10^6	.3	885
I-36"-MS-52	36.625"	1.245	C.S.	532	26.26×10^6	.3	885

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PLAST PIPE RUPTURE MODEL OF THE REACTOR BLDG. MAIN STEAM LINE
FIGURE 3.6E-1

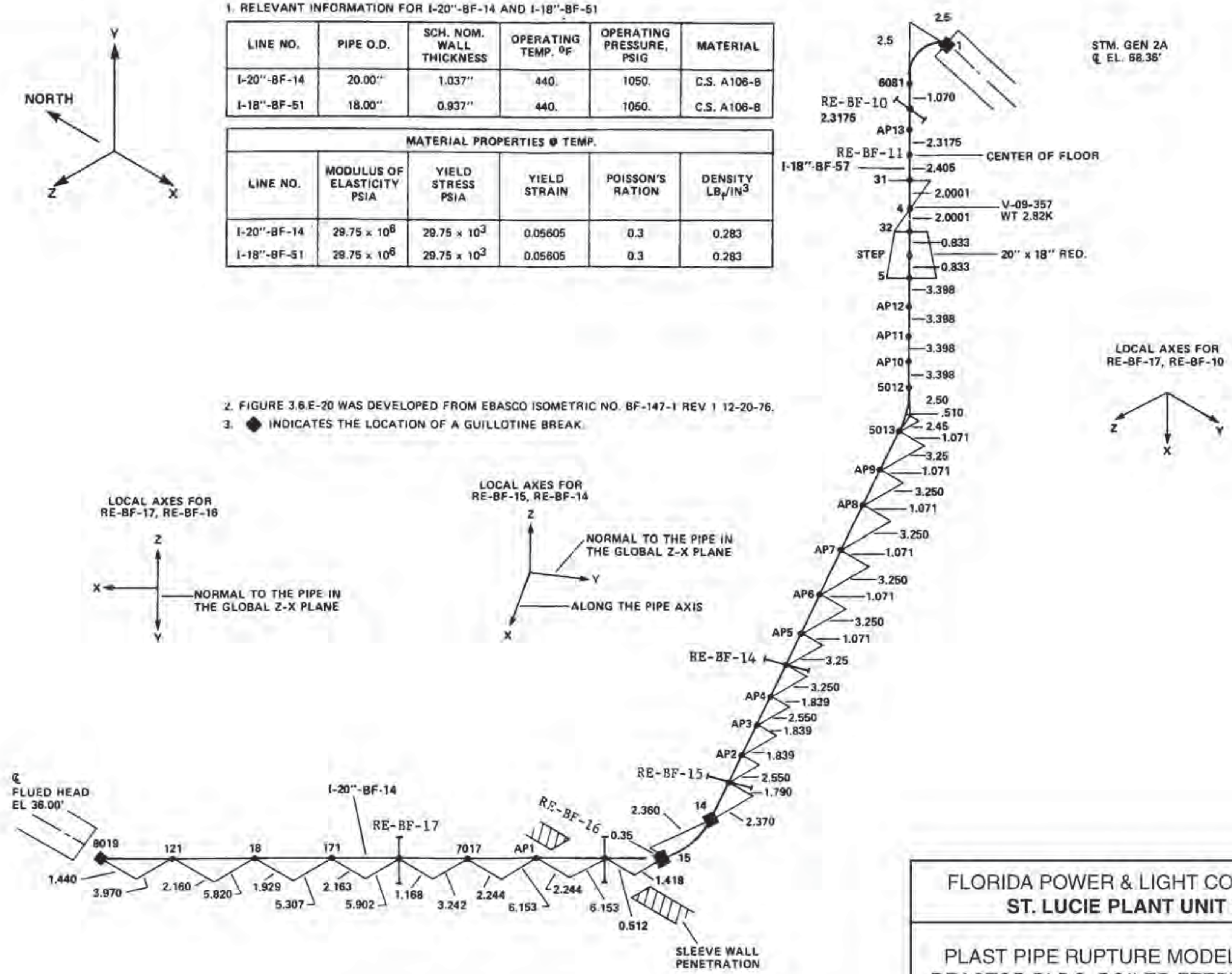
1. RELEVANT INFORMATION FOR I-20"-BF-14 AND I-18"-BF-51

LINE NO.	PIPE O.D.	SCH. NOM. WALL THICKNESS	OPERATING TEMP. °F	OPERATING PRESSURE, PSIG	MATERIAL
I-20"-BF-14	20.00"	1.037"	440.	1050.	C.S. A106-B
I-18"-BF-51	18.00"	0.937"	440.	1050.	C.S. A106-B

MATERIAL PROPERTIES @ TEMP.					
LINE NO.	MODULUS OF ELASTICITY PSIA	YIELD STRESS PSIA	YIELD STRAIN	POISSON'S RATION	DENSITY LB_f/IN^3
I-20"-BF-14	29.75×10^6	29.75×10^3	0.05605	0.3	0.283
I-18"-BF-51	29.75×10^6	29.75×10^3	0.05605	0.3	0.283

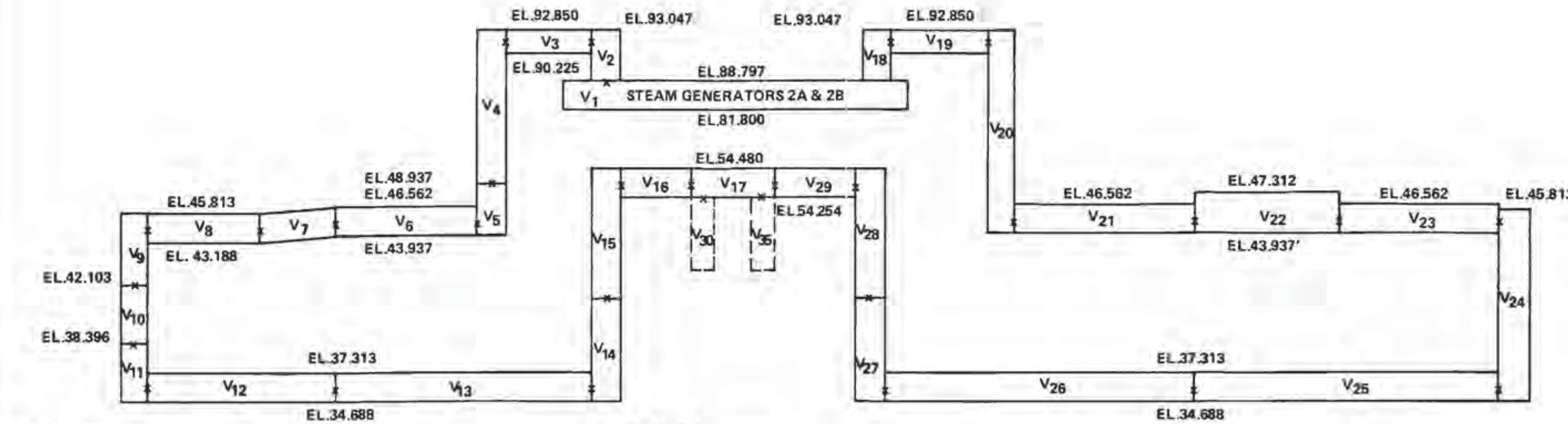
2. FIGURE 3.6.E-20 WAS DEVELOPED FROM EBASCO ISOMETRIC NO. BF-147-1 REV 1 12-20-76.

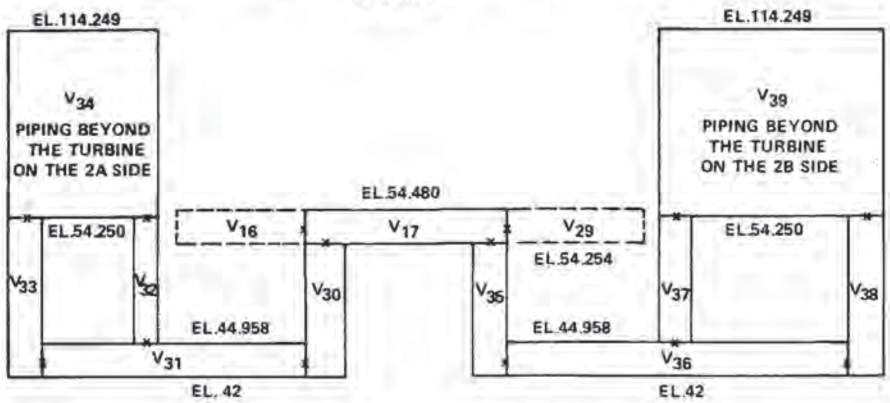
3. ◆ INDICATES THE LOCATION OF A GUILLOTINE BREAK.



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PLAST PIPE RUPTURE MODEL OF THE REACTOR BLDG. BOILER FEEDWTR. LINE
FIGURE 3.6E-2





NOTES:

1. THE STEAM GENERATORS 2A AND 2B ARE MODELED AS VERY LARGE VOLUMES AT A CONSTANT PRESSURE OF 815 PSIA. UNDER THESE CONDITIONS ONE VOLUME CAN BE USED FOR BOTH STEAM GENERATORS.
2. FOR EACH PIPE BREAK THAT WAS MODELED, A PEAK JUNCTION WAS SPECIFIED AT AN APPROPRIATE LOCATION IN A SPECIFIED VOLUME. THE LEAK WAS ASSUMED TO OPEN INSTANTANEOUSLY WITH ZERO DELAY TO AN AREA EQUAL TO THE INSIDE AREA OF THE PIPE.

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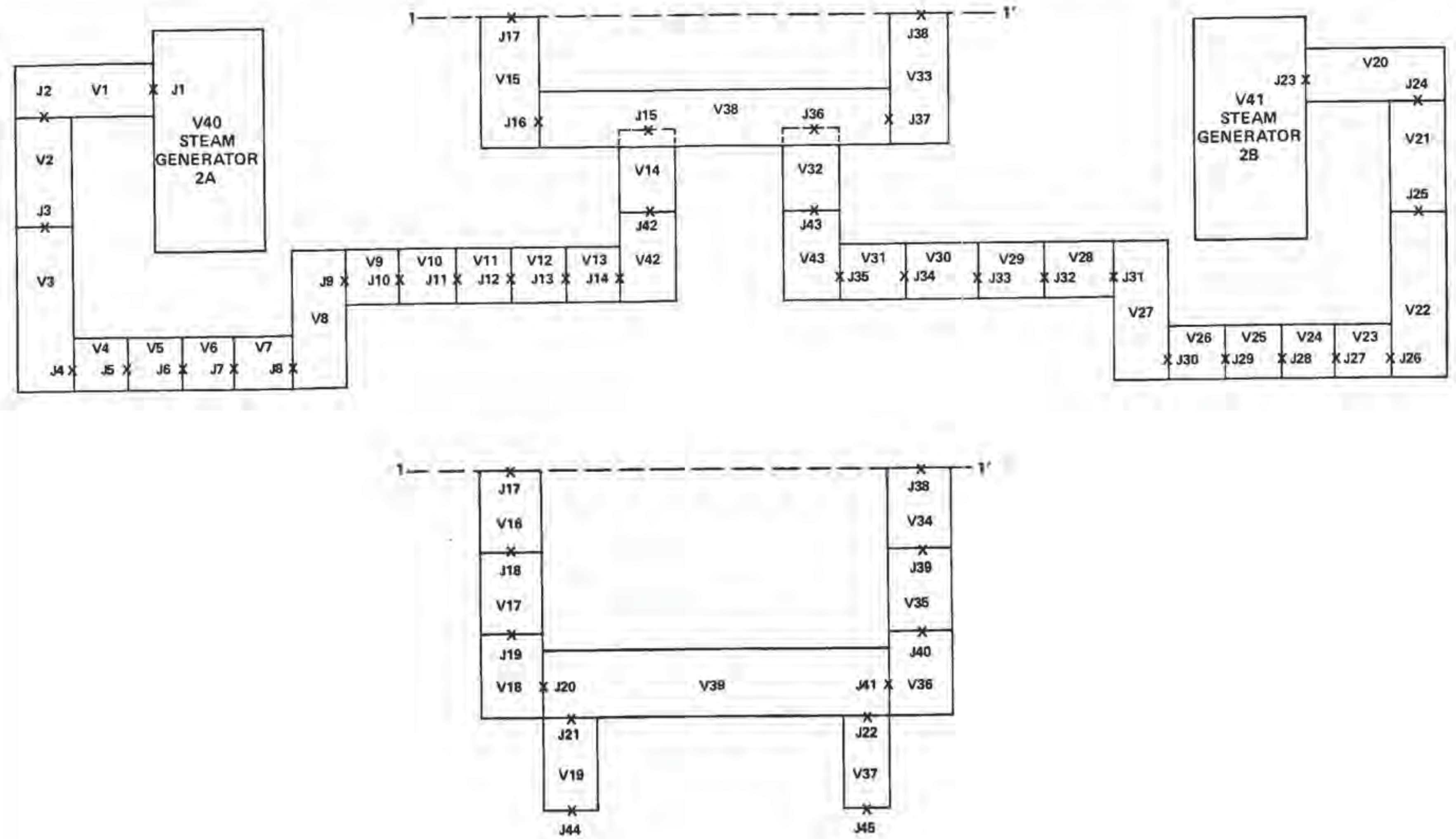
TYPICAL RELAP3/MOD68 FLUID MODEL
MAIN STEAM LINES 2A & 2B
FIGURE 3.6E-3



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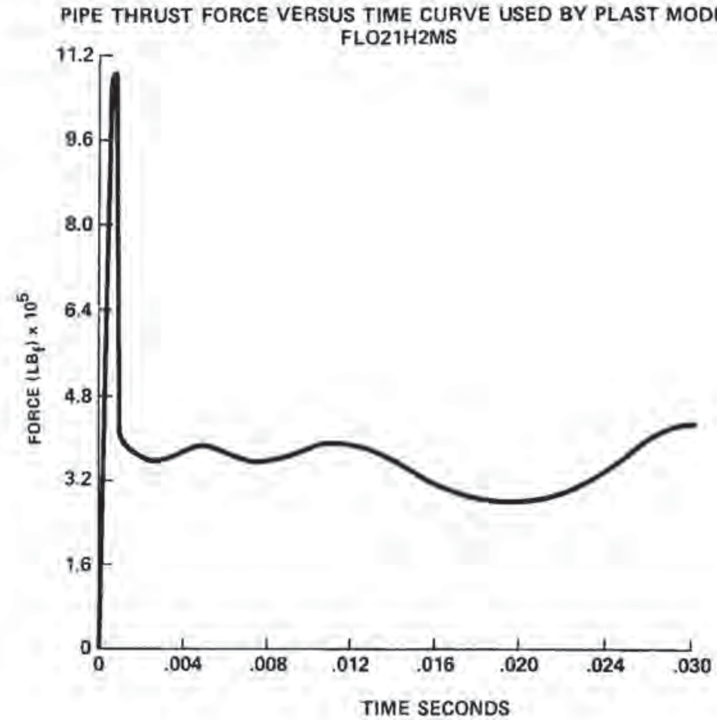
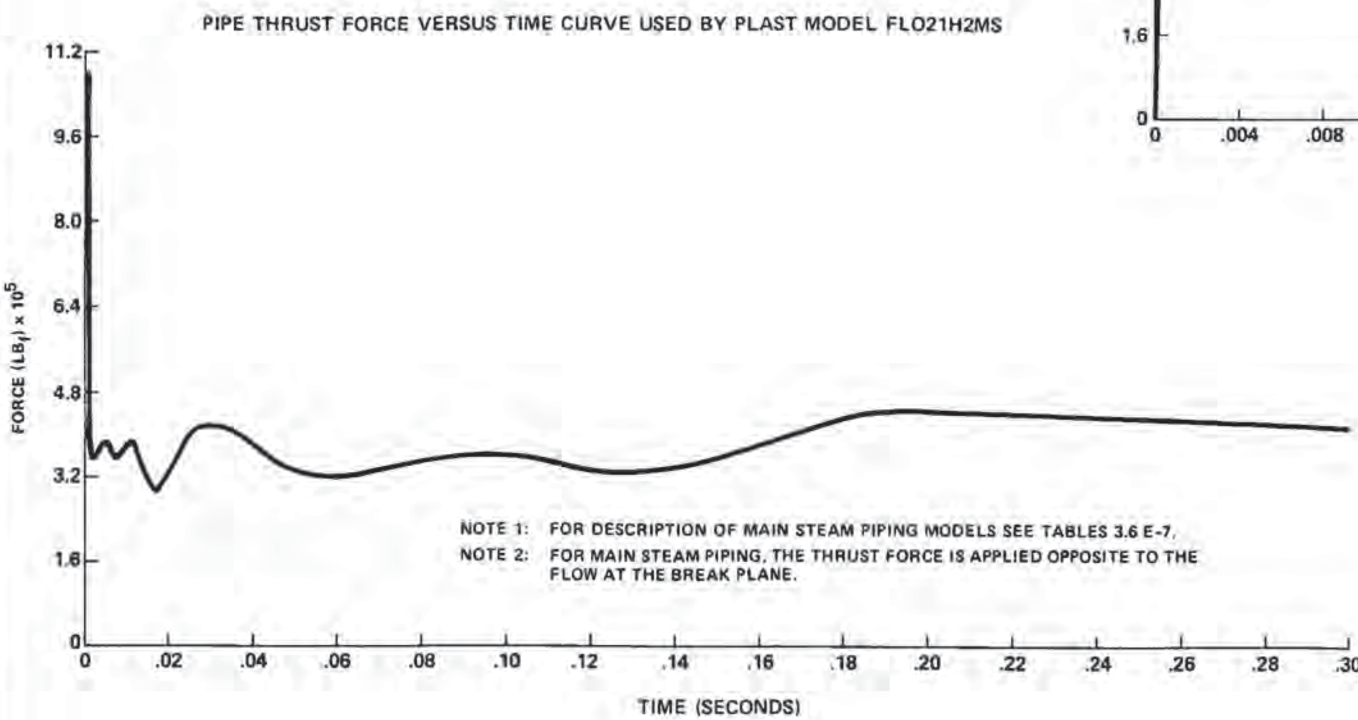
ST. LUCIE PLANT UNIT 2

TYPICAL RELAP4/MOD6 FLUID MODEL
BOILER FEEDWATER LINES 2A & 2B
FIGURE 3.6E-4





NOTE 1: FOR DESCRIPTION OF MAIN STEAM PIPING MODELS SEE TABLES 3.6 E-7.

NOTE 2: FOR MAIN STEAM PIPING, THE THRUST FORCE IS APPLIED OPPOSITE TO THE FLOW AT THE BREAK PLANE.

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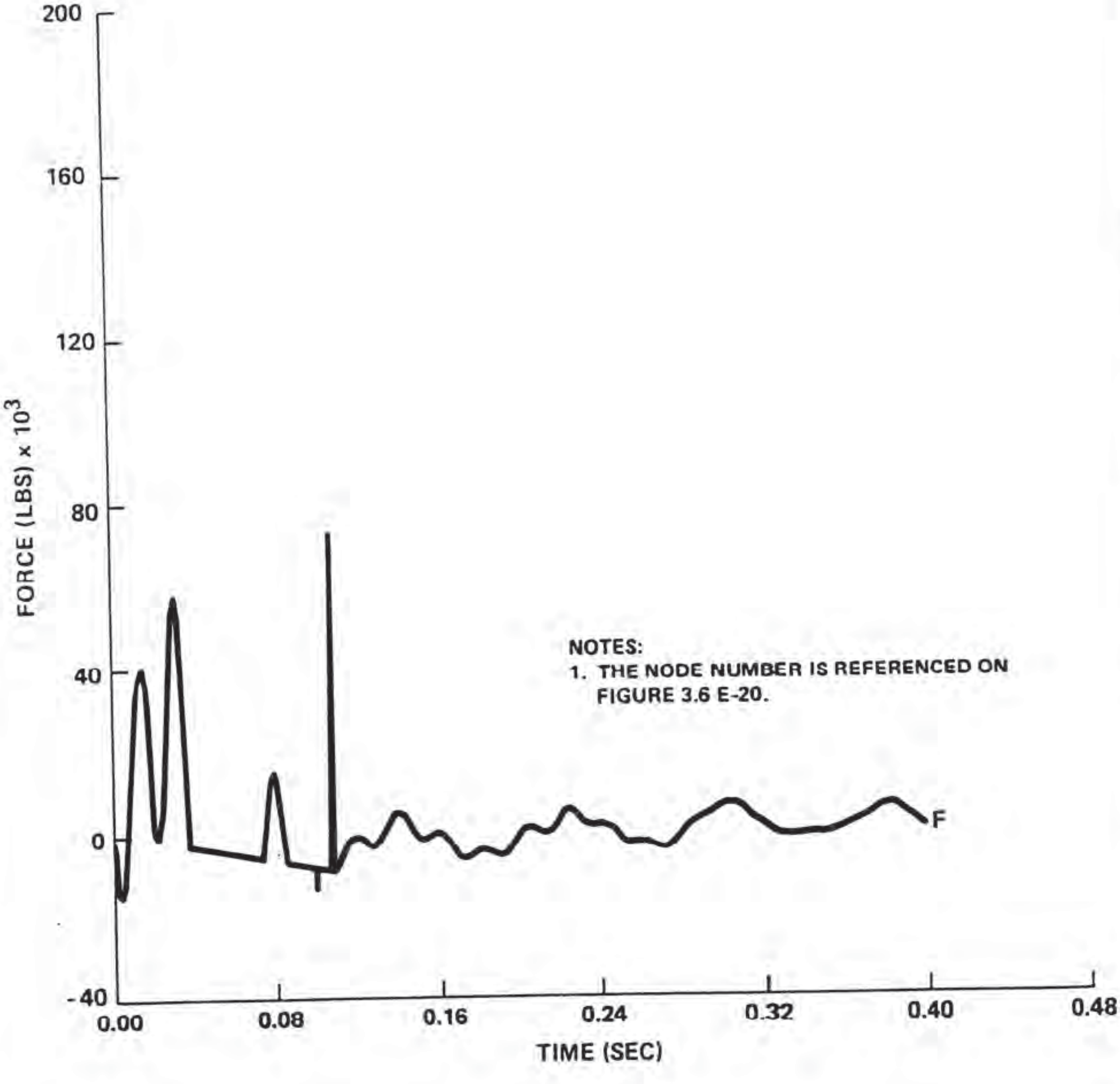
PIPE THRUST FORCE VS. TIME CURVE USED BY PLAST MODEL FL021H2MS
FIGURE 3.6E-5



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FORCE HISTORY APPLIED AT NODE 14 OF BOILER FEEDWATER PIPING MODEL FL021PLASTBF4

FIGURE 3.6E-6

APPENDIX 3.6F

MODERATE ENERGY PIPING FAILURE ANALYSIS

3.6F MODERATE ENERGY PIPING FAILURE ANALYSIS

3.6F.1 MODERATE ENERGY PIPING FAILURE - INSIDE CONTAINMENT

Systems considered for moderate energy analysis inside containment are identified in Subsection 3.6.1.2.2. Design basis environmental conditions inside containment are established by high energy pipe breaks. Therefore, the effects of moderate energy piping failures inside containment are not evaluated.

3.6F.2 MODERATE ENERGY PIPING FAILURES - OUTSIDE CONTAINMENT

This section presents results of the analysis performed for moderate energy piping failures outside containment. The flooding resulting from moderate energy piping failures are considered in evaluating the availability of essential systems and components to mitigate the consequences of the piping failure.

3.6F.2.1 Criteria and Assumptions

In addition to the criteria given in Subsection 3.6.1.3 the following criteria and assumptions are used for moderate energy analysis:

a) Floor drainage system, sump pumps, etc, are considered available to mitigate the flooding consequences of the piping failure.

b) Rate of flow from cracks is assumed to be constant until operator isolates the crack or source volume is depleted.

c) The locations of postulated cracks in the moderate energy piping systems are not based on stress criteria. The crack is assumed to be located anywhere along the run of pipe for the flooding analysis.

d) Moderate energy fluid system pipe failures are considered separately as a single postulated independent event occurring during normal plant operation.

e) No operator action such as closing or opening a valve, stopping or starting a pump is assumed for 30 minutes from the first alarm indication in the control room.

f) Full and part height barriers separating compartments analyzed for flooding are completely watertight. Penetrations connecting compartments are provided with watertight boot seals which have a design temperature of 400F.

3.6F.2.2 Flooding Analysis

3.6F.2.2.1 Evaluation Technique

The Reactor Auxiliary Building (RAB), Fuel Handling Building (FHB), Diesel Generator Building (DGB), Component Cooling Water Building (CCWB), Trestle area and Yard were reviewed to identify all compartments and areas containing safety-related equipment which may be effected by flooding.

Based on this review, the following are considered for the flooding analysis:

a) ECCS compartments A & B in RAB

b) Shutdown Cooling Heat Exchanger Rooms A & B in RAB

c) Boric Acid Make-up Tank Room in RAB

d) Charging Pump Room in RAB

e) Diesel Generator Building

f) Diesel Oil Tank Enclosure

g) Intake Cooling Water Pump Area

h) Component Cooling Water Building

i) Letdown Heat Exchanger Room in RAB

j) Boric Acid Concentrator* Room in RAB

k) Pipe Tunnel in RAB

l) Fuel Pool Heat Exchanger Room in Fuel Handling Building (FHB)

m) Fuel Pool Pumps Room in FHB

n) Fuel Pool Purification Pump Room in FHB

Some of these compartments and areas contain safety related equipment required for safe plant shutdown while others communicate through the floor drain system and corridors with the rooms containing safety related equipment.

The volume of the compartments and areas are taken from the general arrangement drawings, Figures 1.2-12 through 1.2-22.

a) ECCS Room in RAB

The ECCS room is located in the RAB at elevation - 10.0 feet. This room is divided into two compartments, A & B, by a partial height wall. Each compartment contains a high pressure safety injection pump, a low pressure safety injection pump and a containment spray pump. Two reactor drain pumps are located in ECCS Compartment A. There are three watertight doors, with bottom El 0.00 ft, between ECCS compartment B and the main corridor which is at El. -0.5 ft. There is one watertight door, with bottom El. -0.5 ft, between ECCS compartment A and the main corridor. The main corridor is located between the shutdown cooling heat exchanger room and ECCS rooms.

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* Note: The boric acid and waste concentrators are no longer used.

Each ECCS compartment has a sump, bottom El -19.0 feet, and each sump is provided with duplex full capacity sump pumps, 50 gpm each. Level switches and level operated mechanical alternators are provided in each sump for controlling the pump operation. The level control is delineated in the following steps.

1) When water reaches the "high water level", at El -11.25 feet the alternator starts the selected pump and actuates an alarm in the main control room.

2) If the level continues to rise and reaches "high-high water level" at El -10.25 feet, the second pump will start and actuate a second alarm in the main control room. Pumps discharge is routed to the equipment drain tank.

   For flooding analysis, the largest flow crack is assumed to occur, during normal cold shutdown mode, in suction line of the LPSI Pump I-14-SI-424 located in ECCS compartment A. The pipe is 14 inch nominal diameter and operating fluid conditions are 300 psig and 300°F. The shutdown cooling system is categorized as a dual purpose moderate energy system since it is operating in the high energy pressure/temperature region less than two percent of the system normal operating time. Moderate energy cracks are postulated to occur only when the fluid conditions are equal to or less than 275 psig and 200°F. However, for the flooding analysis, the crack was conservatively postulated to occur at 300 psig since this pressure resulted in the largest flow through the piping failure and thereby maximized the flooding of the ECCS cubicles. The resulting flow from the crack is 330 gpm. As a conservative assumption, 2 sump pumps in compartment A are considered to be out of operation. The P&ID for the ECCS room sump pumps is provided on Figure 6.2-41. The analysis assumes that the operator is alerted by the alarm in control room which indicates a "high water level" in the sump and the operator takes corrective actions 30 minutes after the alarm. The flood level in the room at 30 minutes following the first alarm will be 1.6 feet above ECCS room floor level and will reach bottom of HPSI pump conduit box.

   If the leakage crack has occurred in the portion of line between the LPSI pump and valve V3444 in ECCS compartment A, operator action is to isolate the affected shutdown shutdown cooling train by closing valve V3481 (valve V3444 is in the closed position prior to initiation of the shutdown cooling operation). See Figure 6.3-1a.

   The leakage crack in the suction line of LPSI pump reduces Reactor Coolant loop inventory and causes the pressurizer level to drop. Pressurizer low level indicator LI-1103 in the control room along with the ECCS sump high level alarm will alert the operator that the crack has occurred in the LPSI pump suction line. The operator will isolate the affected shutdown cooling loop and continue plant shutdown by means of the redundant shutdown cooling loop.

   A flooding analysis was also performed for ECCS compartment B, assuming a piping failure in LPSI pump 2B suction line during normal plant shutdown mode. The area for ECCS compartment B is larger than compartment A, therefore the flood level will be lower.

As shown above, the flood level in the compartment does not affect any safety-related equipment. In addition, since the flooding is contained within the compartment, the redundant train is not affected. The unaffected train is used to bring the plant to cold shutdown conditions.

b) Shutdown Cooling Heat Exchanger (SDCHX) Room

The two Shutdown Cooling Heat Exchangers are located in the RAB at El -0.5 ft. A seven ft high wall divides the room into two separate compartments. Both compartments communicate with ECCS compartment A through the floor drainage system. A door, with bottom El at 2.0 ft, separates each SDCHX compartment from the main corridor.

For flooding analysis of SDCHX compartment B, the largest flow crack would be in a 12 inch nominal diameter shutdown cooling line (12-SI-164), Figure 9.2-2. The operating conditions of this line are 450 psig and 300F. Moderate energy cracks are to be postulated to occur only when the fluid conditions are equal to or less than 275 psig and 200F. However, for the flooding analysis, the crack was conservatively postulated to occur at 450 psig since this pressure resulted in the largest flow through the piping failure. The flow rate from the crack is 620 gpm of which 38.4 gpm of this fluid drains through a 3 inch diameter floor drain line to sump in ECCS compartment A. The remaining portion of fluid accumulates in the SDCHX compartment B.

When the fluid level in ECCS compartment A sump reaches " high level", the level switch starts a sump pump and actuates an alarm in the control room. The worst flooding condition for ECCS compartment A is postulated to occur when the sump pumps fail to operate. The water level continues to rise and reaches "high water level" in the sump. second alarm is actuated in the control room by the level switch.

The leakage crack in the discharge line of the LPSI pump reduces the flow rate in the system. LPSI pump discharge Flow Indicators FI-3306 or FI-3301 will show a reduction in flow rate. The temperature elements in the LPSI pump discharge lines downstream of shutdown cooling heat exchangers will read temperatures lower than normal Based on this data, the operator can identify the location of the piping failure.

Any significant Shutdown Cooling System leakage would be detected immediately by the Reactor Coolant System parameters displayed in the control room. Pressurizer water level indication and low pressurizer water level alarms are provided in the control room. In addition to the water level instrumentation, both high pressurizer pressure range channels of 1500-2500 psia, and low pressurizer pressure range channels of 0-750 psia are provided. This instrumentation is sufficient to alert the operator of any abnormal system operation.

The time required to reach high water level in the sump, assuming the sump is initially 10 percent full, is 21.2 minutes. It is assumed that 30 minutes after the first alarm in the control room the operator isolates the crack. During this period, water level in the ECCS compartment A will reach 0.1 ft. above floor level. This flood level will not affect any safety- related equipment in ECCS compartment A. The flood level in the shutdown cooling heat exchanger compartment B will reach 7.2 feet above floor level if no fluid leaks through the door. The operator isolates the crack by closing the valves in the suction line of the LPSI pump stopping the pump. If the accumulated fluid in the SDC heat exchanger is allowed to drain to the ECCS room, the water level in the room will reach 3.1 ft.

The operator may close the valves (HCV-25-5 and HCV-25-5A) to prevent draining of the water from the shutdown cooling heat exchanger room. These isolation valves can be closed from the control room. The unaffected SDC train can he used to shutdown the plant.

The above moderate energy line break analysis was performed to demonstrate that the resulting flood would not affect equipment required for safe shutdown. Nonetheless, an additional analysis was performed to show that the plant operator has at least 20 minutes after the first alarm to identify and isolate the damaged train prior to any significant effect on core cooling. This time is available because the water in the RCS above the hot and cold leg piping acts as a reservoir which must be drained prior to any effect on shutdown cooling (SDC) system performance or core cooling (the SDC system takes suction from the RCS hot leg piping).

The RCS volume above the hot and cold leg piping includes the SG tubes, the SG inlet and outlet plenums, the reactor vessel upper head, the pressurizer surge line, and the pressurizer vessel for a total volume of approximately 4700 ft^3. Taking credit for draining of only the SG active tubes and pressurizer volume required to cover the top of the heaters results in a reservoir of 2467 ft^3 (18,454 gallons). With a leak of 620 gpm there is at least 20 minutes between the pressurizer low water level alarm (heater uncovery and uncovery of the SDC suction piping. Therefore, SDC suction performance, coolant circulation through the reactor vessel, and core cooling are maintained.

c) Boric Acid Make-up Tank Room

The Boric Acid Make-up Tank room is located in the RAB at El -0.5 feet. This room contains two Boric Acid Make-up Pumps. The room is open to a corridor which in turns connected to an area containing condensate recovery pumps. The floor drains in these areas are connected to ECCS compartment B Sump.

For flooding analysis, a crack is postulated in a four inch discharge line of Boric Acid Make-up Tank 2B. Tank 2B is considered to be 92.5 percent full. This represents normal operating conditions. The capacity of each tank is 9755 gallons. The operating fluid conditions for the four inch line are 8.0 psig and 170°F. The flow from the crack is 17.0 gpm.

The worst flooding condition for ECCS compartment B will exist when the entire flow from the crack is drained to the ECCS sump. The sump is assumed to be initially 10 percent of full. The "high water level" in the sump is reached after 47.6 minutes from the beginning of pipe failure. The high level switch in the sump actuates an alarm in the control room. Level Indicator LT-2208 in the Boric Acid Make-up Tank will allow the operator to identify the system piping failure. It is conservatively assumed that the sump pumps fail to operate and that the operator will isolate the crack 30 minutes after the first alarm in the control room.

During the 30 minutes the floor level in ECCS compartment B will be 0.4 inch high. This water level will not affect the operation of any safety-related equipment in ECCS compartment B.

If the piping failure is located in the upstream side of Valve V2142, the operator cannot isolate the crack. The entire contents of the tank are considered to be drained to ECCS compartment B. The resulting water level in the ECCS compartment is 0.7 feet. This flood level will not affect the operation of safety-related equipment.

d) Charging Pump Room

The three charging pumps are located in the RAB at El -0.5 feet. The charging pump room is divided into three separate compartments by 6.5 foot high walls. Each charging pump is located in a separate compartment. Doors are provided between the charging pump room and the pipe tunnel at El +0.5 feet and RAB main corridor. Each compartment has a 6 inch high curb at the entrance.

For flooding analysis, the largest flow crack would be in the four inch charging pump 2C suction line 4-CH-967 with operating conditions 27 psig and 120°F (see Figure 9.3-5c). The flow rate from the crack is 31.7 gpm. The entire spillage will drain to ECCS compartment B sump. The operator is alerted by the sump B "high water level" alarm in the control room 25.6 minutes after the pipe failure. The operator will also notice the volume control tank level decreasing via LI-2226. It is assumed that 30 minutes after the alarm, corrective action is taken. The flood level in ECCS compartment B after the 30 minutes will be 0.06 feet, if the sump pumps are assumed to be out of operation. This flood level in ECCS compartment B will not affect operation of any equipment in the room. The operator isolates the piping failure by closing valve V2501 on the discharge line of the volume control tank.

The flooding analysis for charging pump compartments 2A and 2B produces exactly the same result as that of charging pump compartment 2C.

e) Diesel Generator Building

There are two diesel generators installed in separate rooms at El 22.67 feet in the diesel generator building. For flooding analysis, the largest flow crack would be in the Service Water System line 2-SW-108 (see Figure 9.2-4). The pipe is 2 inches nominal diameter and operating conditions are 75 psig and 95°F. The Service Water System serves no safety function since it is not required to achieve safe plant shutdown nor to mitigate the consequences of a design basis accident. The flow rate from the crack is 18 gpm. The entire flow from the crack drains through the drainage system to a pump box located in the yard. Flow would then go to the grade through a 3 inch vent located in top of the pump box at elevation 19 feet. In the case of blockage of the drainage system, the 18 gpm flow rate from the crack is safely handled by out seepage from under the Diesel Generator Building doors. See Subsection 9.5.4 for a discussion of the Diesel Generator building drainage system.

Since there is no accumulation of fluid in the diesel generator room under normal circumstances, and only slight accumulation if blockage of the drains occur, the operation of the diesel generator is not affected by this accident. For flooding analysis, there is no critical time for the operator to isolate the pipe failure.

f) Diesel Oil Tank Enclosure

There are two diesel oil storage tanks with their pumps located in two separate compartments in the diesel oil tank enclosure. Tanks and pumps are installed at El. 19.0 feet. For flooding analysis the highest flow crack would be in the 3 in. nominal diameter diesel oil pump suction line 3-DO-07 (see Figure 9.5-6). The operating fluid conditions are 25 psig and 100F. The drain line in each compartment is normally closed. Each of the compartments is designed to hold the entire capacity of its respective tank should a leak occur.

g) Intake Cooling Water Pump Area

There are three intake cooling water pumps located in the ICWP area at El. 16.5 feet. For flooding analysis the largest flow crack would be in a 30 in. diameter ICWP discharge line I-30-CW-11 (see Figure 9.2-1). The operating conditions are 90 psig and 95F. The resulting flow from the crack is 604 gpm. The entire flow from the crack is drained through the annular area between the discharge pipe and the 42 inch diameter pipe sleeve to the suction well in the intake structure. No safety-related equipment is affected by this flooding.

The intake cooling water pump is designed for 14500 gpm at 130 ft head. Loss of 604 gpm through the crack will not affect the system operation. The normal plant operation and safe plant shutdown are not compromised.

h) Component Cooling Water Building

The component cooling water building contains three CCW pumps and two heat exchangers. The floor elevation of the compartment is 12.0 ft. The pumps and heat exchangers are mounted on pedestals at about El 24.0 ft. There are two sumps in this area. One sump is located inside the compartment with its bottom at El. 19.67 ft. The other sump is located at the pipe tunnel area with bottom elevation at 1.0 ft. The pipe tunnel sump is provided with a sump pump with capacity of 25 gpm. This pump transfers the fluid from the pipe tunnel sump to the sump in the CCW compartment. The P&ID for the yard sump pump is provided in Figure 3.6F-1. The fluid from the CCW compartment sump drains to the existing 36 inch drain pipe via a 3 inch drain line. The 36 inch line discharges to existing grade at El 0.0 ft.

For flooding analysis, the largest flow crack would be in the 30 inch CCW heat exchanger intake cooling water inlet line I-30-CW-78 operating at 60 psig and 95°F (see Figure 9.2-1).

The flow rate from the crack is 490 gpm. The entire flow spills on the floor and fills up the pipe tunnel sump to "high-high water level" in seven minutes. The level switch in the sump starts the sump pump and actuates an alarm in the control room. The sump pump delivers 25 gpm to the CCW compartment sump. All the water will be drained from the CCW compartment because drain capacity is more than 25 gpm.

The alarm in the Control Room due to "high-high water level" in the pipe tunnel sump alerts the operator that there may be piping failure in the pipe tunnel area. The intake cooling water system and component cooling water system piping are located in this area. If the crack occurs in the CCW System, the low level in the surge tank will initiate an alarm in the Control Room. If the alarm is not due to the surge tank low level then the operator could identify the piping failure to be in the ICW System.

It is assumed that 30 minutes after pipe failure, corrective action is taken by the operator. During this 30 minutes the fluid level in the pipe tunnel area will reach El 8.0 feet. No safety-related equipment is affected by this flooding. The operator may close valves SB21185, 21186 and 21192 to isolate the crack. The intake cooling water pump is designed for 14500 gpm at 130 ft head. Loss of 490 gpm through the crack will not affect the operation of the system. Therefore plant shutdown is not compromised by this piping failure. Even if the piping failure is isolated, the unaffected component cooling water loop B is available and capable of supplying the minimum safety feature requirements for plant shutdown.

i) Letdown Heat Exchanger Room

The letdown heat exchanger room is located in the RAB at El 19.5 feet and contains the letdown heat exchanger and associated piping and valves. A door connects this room with the corridor. This room contains no equipment needed for safe plant shutdown but the room is connected with ECCS compartment A, through the floor drainage system.

For flooding analysis, the largest floor crack is in the letdown heat exchanger component cooling water outlet line 8-CC-134 (see Figure 9.2-2). The pipe is eight inch nominal diameter and the operating fluid conditions are 100 psig and 150°F. The resulting flow from the crack is 160 gpm of which 38.4 gpm drains through a three inch diameter floor drain to the equipment drain tank. The remaining 121.6 gpm accumulates in the letdown heat exchanger room.

The temperature indicator TI-04-5 and flow transmitter FT-14-6 in the component cooling water line will allow the operator to identify the piping failure in the system. In addition there will be an alarm in the Control Room due to low level in CCW surge tank about 5.3 minutes after the piping failure.

Within 7.6 minutes after the pipe failure, the water level in the letdown heat exchanger room reaches the curb level and starts spilling into the corridor. Within 11.4 minutes after the pipe failure, the fluid level in the equipment drain tank reaches "high water level." The "high water level" switch in the equipment drain tank actuates an alarm in the control room. It is assumed that 30 minutes after the alarm the operator isolates the line by closing valve SB14241. During this

30 minutes, the equipment drain tank and the chemical drain sump are filled. The operator also closes valves HCV-25-5 and HCV-25-5A from the control room in ECCS compartment A sump drain line, so that the fluid will not reach ECCS compartment A. The fluid will spill into the corridor at El 19.5 feet. Part of the fluid will drain to the equipment drain tank, and part of the fluid spills to the corridor at El -0.5 feet through the stair well. No safety-related equipment is affected by this flooding condition.

j) Boric Acid Concentrator Room

Two boric acid concentrator† rooms and one waste concentrator† room are located in RAB at El 19.5 feet. Each room is connected with the corridor at El 19.5 feet via a door. The boric acid concentrator and waste concentrator are not safety-related. However, these rooms are connected through the drainage system with ECCS compartment A.

The flooding analysis was performed for a postulated crack in the boric acid concentrator component cooling water line. Typical line is 6 inches in diameter with operating conditions of 100 psig and 120F. The flow from the crack is 106 gpm. The result of the analysis indicates that the operator has sufficient time (i.e., 30 minutes from the first alarm in the control room) to isolate the crack before the fluid reaches ECCS compartment A. However, if the operator fails to close the ECCS sump A isolation valves HCV-25-5 and HCV-25-5A, the accumulated fluid in the concentrator room eventually will drain to ECCS compartment A and the water will reach 0.07 feet high. The flood level is insufficient to affect the operation of safety-related equipment.

k) Pipe Tunnels

The upper tunnel is located at El 19.5 feet and the lower tunnel has bilevel floors at El -0.5 feet and at El 0.5 feet. An opening in the floor of the upper tunnel connects the lower tunnel. Two air tight doors separate the upper tunnel from the switchgear room and the H_2 recombiner supply panel 1A ILRT panel room. A door is provided between the lower tunnel and the charging pump room. Another door is provided between the lower tunnel and the corridor which is between the ECCS room and shutdown cooling heat exchanger room. The lower tunnel communicates through the drainage system with ECCS compartments A & B.

The leakage with the greatest crack in the lower pipe tunnel is a postulated crack in containment spray line I-24-CS-41 (see Figure 6.2-41). The pipe line is 24 inch nominal diameter with operating fluid conditions at 30 psig and 120°F.

The flow from the crack is 180 gpm. There are two 3 inch diameter drain lines in the lower tunnel. One drain line delivers fluid to ECCS compartment A sump and the other drain line drains the fluid to ECCS compartment B sump. Flow rate through each drain line is 38.4 gpm. It is conservatively assumed that all 4 sump pumps pumps are not available.

† Note: The boric acid and waste concentrators are no longer used.

The operator is alerted by the sump "high water level" alarm in the control room 21 minutes after pipe failure. The RWT level indicators LIS-07-2A, 2B, 2C, 2D will also alert the operator of the piping failure accident. It is assumed that the operator takes corrective action 30 minutes after pipe failure. During this 30 minute period, the fluid level in ECCS compartment A will reach 0.1 feet. This fluid level will not affect the operation of safety- related equipment in the compartment. The operator can isolate the break by closing valve MV-07-1A. The operator could also close valves HCV-25-5, HCV-25-5A, HCV-25-3 and HCV-25-3A in the drain lines from the control room to prevent further flooding due to draining of fluid accumulated in the pipe tunnel. However, the worst condition is to let all fluid in the pipe tunnel to drain to the ECCS sumps. In this case, the water level in both the ECCS compartments will reach 0.374 feet. This fluid level will not affect operations of any safety-related equipment.

The largest flow crack in the tunnel at El 19.5 feet would be in component cooling water line 20-CC-27 (see Figure 9.2-2). The pipe is 20 inch nominal diameter and the operating conditions are 100 psig and 180°F. The flow from the crack is 433 gpm of which 82 gpm is directly drained to the equipment drain tank. The remainder drains to the lower pipe tunnel. From the lower pipe tunnel, the fluid is drained to the sumps in the ECCS compartments A & B.

The CCW system surge tank low level alarms LS-14-1A & 1B, flow indicators FIS-14-10A and FIS-14-10B indicating higher flow and temperature recorder TR-09-5A registering lower temperature than normal will assist the operator to identify the piping failure location.

The operator is alerted by CCW surge tank "low level" alarm one minute after beginning of leakage from the crack. The CCW makeup system which delivers 100 gpm to surge tank is taken into account to determine the time for level alarm. It is assumed that the operator requires 30 minutes after the alarm in the control room prior to initiating the corrective action. The operator may isolate the crack by closing valves HCV-14-10, MV-14-4, SB14487, SB14133, SB14127, SB14531, MV-14-19 and V14301. The flood level in the ECCS compartments at this time (i.e., 31 minutes) will be 0.1 feet and safety-related equipment is not affected. If the fluid accumulated in the pipe tunnel El -0.5 feet is allowed to drain to the ECCS compartments without operator isolating them, the fluid level will reach 0.49 feet. The flood level does not affect any safety-related equipment. The height from the floor to nearest safety-related item, conduit box for HPSI pump, is 1.6 feet.

l) Fuel Pool Heat Exchanger Room

The Fuel Pool Heat Exchanger room is located in the Fuel Handling Building at El 19.5 feet and contains the fuel pool heat exchanger and associated piping and valves. A door connects this room with the Fuel Pool Purification Filter room. A six inch high curb separates Fuel Pool Heat Exchanger room from Fuel Pool Pump room.

For flooding analysis, the largest flow crack would be in CCW line 12-CC-130 from fuel pool heat exchanger to return header B (see Figure 9.2-2). The pipe is 12 inch nominal diameter and its operating conditions are 100 psig and 150°F.

The resulting flow from the crack is 275 gpm. A 4 inch diameter drain line delivers 82 gpm of the fluid from this room to the equipment drain tank. The equipment drain tank room is connected through the drainage system with the ECCS compartment A.

The operator is alerted by CCW surge tank "low level" alarm two minutes after beginning of leakage. The flow element FIS-14-2 will also aid the operator in defining the piping failure. The CCW make-up system which delivers 100 gpm to surge tank is taken into account to determine the time for alarm level in the surge tank. Operator isolates the crack by closing valves MV-14-17 and MV-14-19.

The flooding analysis indicates that during this 30 minute period, the equipment drain tank, chemical drain sump, chemical drain tank are filled and overflow to the room containing the equipment drain tank and chemical drain tank. From this room the overflow drains to the ECCS compartment A sump. As the accumulated fluid continues to drain from the fuel pool heat exchanger room, the sump will overflow into ECCS compartment A to a depth of 1.29 feet which does not affect safety-related equipment. The fluid drainage to ECCS compartment sump A could also be stopped from the control room by closing valves HCV-25-5, 5A.

m) Fuel Pool Pump Room

The fuel pool pump room is located in Fuel Handling Building at El 19.5 feet and contains two fuel pool pumps. One door with a six inch curb connects this room with fuel pool purification filter room. There is a 6 inch high curb between this room and the corridor. This corridor also has a 6 inch high curb to separate the fuel pool heat exchanger room.

For flooding analysis, the largest flow crack would be in fuel pool pump suction line 12-FS-501 between fuel pool and valves V4203 or V4202 (see Figure 9.1-6). The pipe is 12 inch nominal diameter and its operating conditions are 11 psig and 120°F. The resulting flow is 43.6 gpm. This break cannot be isolated by shutting off the valves in the broken line. Flooding will stop when level of fluid in fuel pool drops below the fuel pool pump suction line at El 56 feet. This will occur 925 minutes after leak initiation. Although normal surveillance would detect the leakage before the fluid level drops below suction level, this analysis assumes the fluid level drops below suction level. Fluid from leak fills fuel pool pumps room, overflows into the corridor and fuel pool heat exchanger room. Thereafter, all flow runs through a four inch diameter drain line to the equipment drain tank. The operator is alerted by equipment drain tank high level alarm 12 minutes after the failure and it is assumed operator closes valves HCV-25-5 or HCV-25-5A after 30 minutes. These valves isolate the drain line from equipment drain tank room which runs to ECCS sump A. With these valves closed, the chemical drain sump overflows on to El -0.5 feet, flooding the floor to a depth of 0.2 feet before the water level in the fuel pool falls below fuel pool pump suction line. Therefore, flooding will not affect any safety-related equipment.

When the water level falls below the fuel pool pump suction line, the accident is similar to loss of all external cooling. The analysis for this accident has been performed and the results are presented in Subsection 9.1.3.

n) Fuel Pool Purification Pump Room

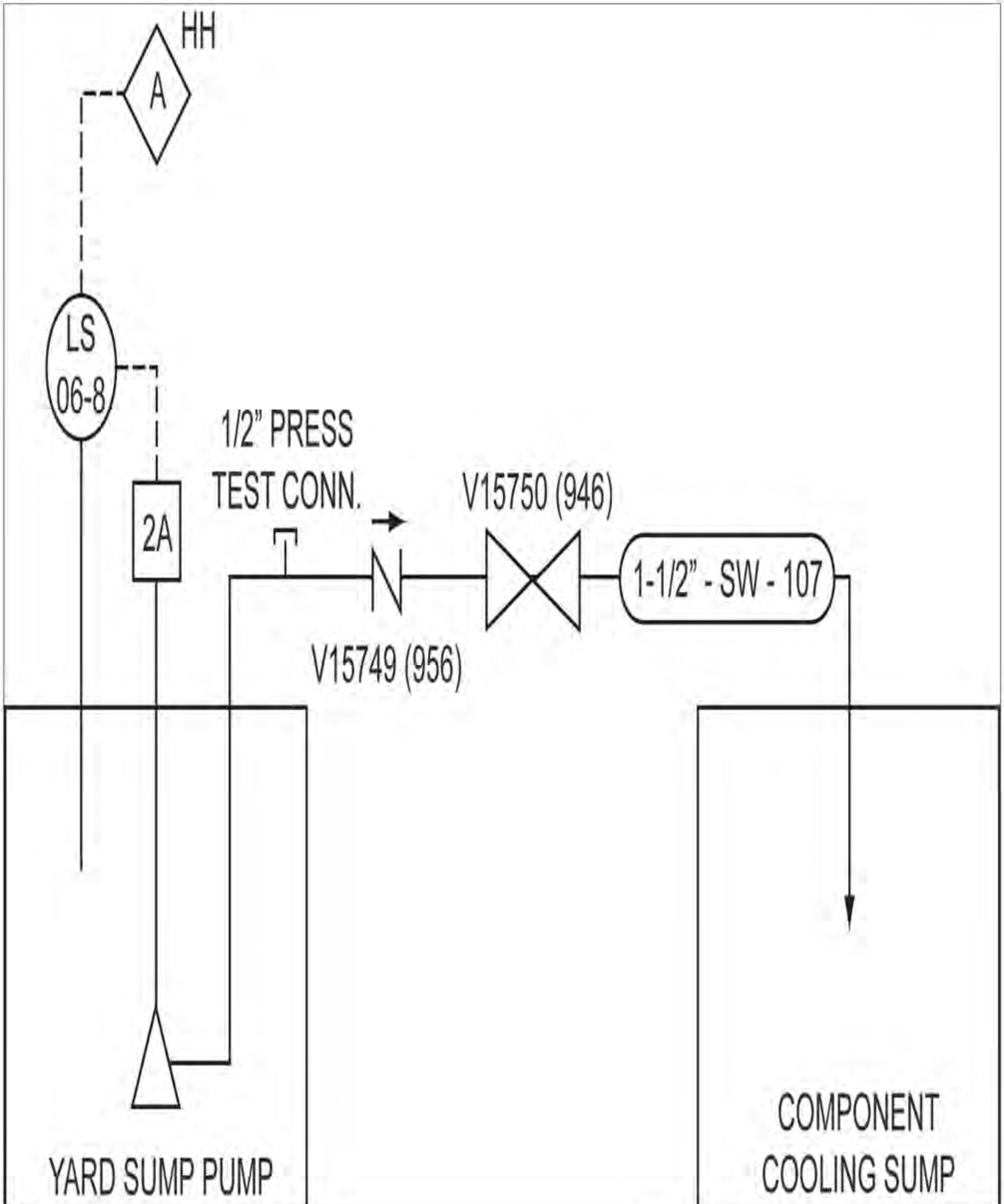
Fuel pool purification pump room is located in the Fuel Handling Building at El 19.5 feet and contains fuel pool purification pump. One door connects this room with fuel pool purification filter room. A wall, top El 26.27 feet, provides separation from the fuel pool pumps room. For flooding analysis, the largest flow crack is the fuel pool ion exchanger outlet line 3-FS-524 (see Figure 9.1-6). The pipe is 3 inches in nominal diameter and its operating conditions are 75 psig and 120°F. The resulting flow is 21 gpm. All flow from the crack runs through 3 inch diameter drain line to the equipment drain tank. The operator is alerted by equipment drain tank high level alarm 23 minutes after beginning of leakage and has 30 minutes to isolate the leakage by closing valve V4220 if failure has occurred downstream of this valve. The operator cannot isolate leakage if the crack is located inside of the fuel pool purification pump room upstream of valve V4220. The crack will be isolated 595 minutes after beginning of leakage when the level of fluid in the fuel pool will drop below fuel pool purification pump suction at El 59.0 feet. Although normal surveillance would detect the leakage before the fluid level drops below suction level, this analysis assumes the fluid level drops below suction level. At this time fluid has filled equipment drain tank and chemical drain sump. The fluid will not reach chemical drain tank or ECCS compartment sump A. Therefore, flooding will not affect safety-related equipment.

3.6F.2.3 Environmental Qualification Effects

This is addressed in UFSAR Section 3.11.

3.6F.2.4 Summary and Conclusion

The consequences of flooding due from the pipe crack were evaluated. The effects of flooding on systems and components required to shutdown the reactor and mitigate the consequences of a postulated piping failure were analyzed. As indicated in the analysis, moderate energy pipe failure does not affect essential equipment and components required for safe plant shutdown.



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

YARD SUMP PUMP
P&I DIAGRAM

FIGURE 3.6F-1

Amendment No. 18 (01/08)

3.7 SEISMIC DESIGN

3.7.1 INPUT CRITERIA

3.7.1.1 Design Response Spectra

The design response spectra for the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) are shown on Figures 3.7-1 through 3.7-4. These spectra are used in the seismic design of safety related structures, systems and components.

Utilizing techniques developed by Newmark[1], the design response spectra are developed for 1, 2, 4, 5, 7 and 10 percent structural damping. These spectra are scaled to maximum ground acceleration of 0.1 g and 0.05 g for the SSE and OBE, respectively.

For the horizontal direction, the amplification factors for the design spectrum frequency control points are computed using the following equations based on Newmark's method (refer to Figure 3.7-5):

Point A, at frequency 33 Hz

A_A = 1.0

Point B, 9 Hz

A_B = 4.25 - 1.02 ln β

Point C, 2.5 Hz

A_C = 1.2 A_B = 5.1 - 1.224 ln β

Point D, 0.25 Hz

D_D = 2.85 - 0.5 ln β

in which

A_A = acceleration amplification at Point A;

A_B = acceleration amplification at Point B;

A_C= acceleration amplification at Point C;

D_D = displacement at Point D;

β = Damping factor, as percentage of critical value

Table 3.7-1 presents the amplification factors and amplified accelerations (or displacements) for the horizontal components of the OBE. Table 3.7-2 gives the percent critical damping values. For the SSE, the values of accelerations and displacements are twice the OBE values.

The design response spectra for horizontal direction as described above are in conformance with Regulatory Guide 1.60, "Design Response Spectra For Seismic Design of Nuclear Power Plants," December 1973 (R1).

For the vertical direction, based on Newmark's procedure, the vertical response spectrum is drawn by taking two-thirds of the horizontal design spectrum from very low frequencies through points D' and C', both of which lie at the same frequencies as points D and C, but at two-thirds of the values of amplification. Line D'C' is extended to Point C", at which the vertical design spectrum becomes equal to the horizontal design spectrum, and then merges into the horizontal ground acceleration value, as presented in Figure 3.7-5.

The procedure described herein for constructing the vertical design response spectrum is in conformance with Regulatory Guide 1.60, (R1) except between the frequency range from 0.25 Hz to 3.5 Hz, in which Regulatory Guide 1.60 (R1) recommends slightly higher amplification factors.

The horizontal and vertical response spectra for both the SSE and the OBE are applied at the foundation levels of the seismic Category I structures.

3.7.1.2 Synthetic Time-History Earthquake Derivation

Artificial accelerograms are developed for the horizontal and vertical components of the SSE using the procedures proposed by Ruiz and Penzien[(2)] and Tsai[(3)]. The Ruiz and Penzien procedure utilizes a linear stochastic model to generate records of filtered nonstationary shot noise to simulate ground motion accelerograms recorded during strong motion earthquakes. The specified principal characteristics of the earthquake, i.e., the expected peak acceleration, duration and variation in intensity with time are reflected in the accelerograms. For the horizontal SSE, the expected peak acceleration is 0.1 g and the duration of the strong shaking is 10 seconds. For the vertical component of the SSE, the peak acceleration is 0.1g.

To ensure that the spectrum of the artificially generated accelerogram envelops the design spectrum discussed in Subsection 3.7.1.1., Tsai's procedures are utilized. Tsai's procedure, an iteration approach applies a deterministic technique to modify the accelerogram by passing the motion successively through a set of frequency filters to suppress or raise any local portion of the response spectrum to match the design spectrum.

The above procedures allow a modification of the simulated accelerograms to ensure that the response spectra generated by the accelerograms are compatible with the design spectra discussed in Subsection 3.7.1.1.

Comparison of the spectral values derived by the horizontal component of the SSE and the design spectra are made at the following frequencies:

Frequency Interval (Cycles/Sec)	Increment (Cycles/Sec)	Number of Frequencies
0.3 – 1.0	0.025	28
1.0 – 2.5	0.05	30
2.5 – 9.0	0.1	65
9.0 – 33	0.5	49
		172

A plot of the horizontal accelerogram is shown on Figure 3.7-6. The spectra are presented on Figures 3.7-7 through 3.7-11 for damping ratios of 0.02, 0.04, 0.05, 0.07 and 0.10.

Comparison of the spectral values derived by the vertical component of the SSE and the design spectra are made at the following frequencies:

Frequency Interval (Cycles/Sec)	Increment (Cycles/Sec)	Number of Frequencies
0.3 – 1.0	0.025	28
1.0 – 3.5	0.05	50
3.5 – 9.0	0.1	55
9.0 – 33	0.5	49
		182

A plot of the vertical accelerogram is shown on Figure 3.7-12. The spectra are shown on Figures 3.7-13 through 3.7-17 for damping ratios of 0.02, 0.04, 0.05, 0.07, and 0.10.

The OBE accelerograms are obtained by reducing the corresponding SSE accelerograms by a scale factor of 0.5. A plot of the horizontal accelerogram for the OBE is shown on Figure 3.7-18 and a plot of the vertical accelerogram for the OBE is shown on Figure 3.7-19.

Comparison of the spectral values derived by the horizontal and vertical components of the OBE and the corresponding design spectra are made using the same methods and frequency intervals described previously for the SSE cases.

Comparisons between the response spectra points computed from the artificial time histories and the design response spectra suggested in RG 1.60 (R1) indicate that some of the response spectra points computed for the artificial time histories fall below the design response spectra. The spectra values are generated at 1/2 percent, two percent, five percent, seven percent and 10 percent damping, as suggested in RG 1.60 (R1). The frequency intervals used are those suggested in SRP Table 3.7.1-1. Results show that only for the 1/2 percent damping curve more than five points (out of 75 points) fall more than 10 percent below the design spectra curve. However, for St. Lucie Unit 2, the lowest damping value specified is one percent (for steel piping) so the case of 1/2 percent damping has no effect on the seismic analysis. Moreover, the lowest frequency value for St. Lucie Unit 2 is 1.22 cycles per second (see Table 3.7-18, Reactor Building), therefore, points falling below design spectra for frequencies less than 1.22 Hz do not affect the results of seismic analysis. For the other points, the design spectra for the time histories show substantially higher values than the RG 1.60 (R1) design spectra. Thus the positive values should compensate for the effects, if any, of the negative values and insure a conservative design.

In summary, the 1/2 percent damping curve is not used for any design purpose on St. Lucie Unit 2 and the remaining response spectra curves meet the criteria of no more than five points falling more than 10 percent below the design spectra.

The horizontal spectra for the OBE are shown on Figures 3.7-20 through 3.7-24 for damping ratios of 0.02, 0.04, 0.05, 0.07, and 0.10.

The vertical spectra for the OBE are shown on Figures 3.7-25 through 3.7-29 for damping ratios of 0.02, 0.04, 0.05, 0.07, and 0.10.

3.7.1.3 Critical Damping Values

Values of critical damping used in the seismic analysis for the various types of structural members are shown in Table 3.7-2. These damping values are equal to the values given in the Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," October 1973 (R0).

3.7.1.4 Supporting Media for Seismic Category I Structures

Major seismic Category I structures for St. Lucie Unit 2, as described in Subsection 3.7.2.1.1, are supported individually by separate foundation mats. The soil layering characteristics and soil properties are discussed in Subsection 2.5.4. A summary of foundation size, structural height and foundation embedment depth for these structures is provided in Table 3.7-3. Soil structure interaction is discussed in Subsection 3.7.2.4.

3.7.2 SEISMIC SYSTEM ANALYSIS

This section includes discussion of seismic analysis of all seismic Category I structures. Seismic analyses of seismic Category I piping systems including the Reactor Coolant System are discussed in Subsection 3.7.3.

3.7.2.1 Seismic Analysis Methods

Seismic analyses of the seismic Category I structures are performed using either the normal mode time history method or the response spectrum method. Seismic inputs used for determining the structural response are those described in Subsections 3.7.1.1 and 3.7.1.2 applied at the various foundation levels of seismic Category I structures.

Seismic Category I structures and associated mathematical models as well as analytical considerations of the methods of analysis, are presented in the following subsections.

3.7.2.1.1 Mathematical Models

For the seismic analysis of the seismic Category I structures, conventional lumped mass mathematical models are selected to represent each structure. In this model, the structure is represented by a cantilever beam with masses lumped at selected elevations simulating floor weights, walls, columns and major equipment. The cantilever beam connecting those lumped masses is assumed weightless and elastic, representing the stiffness of walls or columns between the lumped mass points. The foundation mat supporting the cantilever beam is considered as a rigid body and is supported by rotational and translational springs simulating soil-structure interaction.

For the seismic analysis in the vertical direction, mathematical models are developed using similar lumped mass principles. However, since the major interest In this case is focused at the middle of a floor bay or at column floor junctures, appendages representing floor bay behavior are added to the cantilever beam resulting in a more complex model.

Equivalent soil springs as described in Subsection 3.7.2.4 and critical damping values as described in Subsection 3.7.1.3 are used in the analysis. Significant maximum relative displacements among supports of seismic Category I piping, component and equipment are considered as discussed in Subsection 3.7.3. Differential seismic movements between seismic Category I structures do not cause any structural coupling since sufficient distance are provided between the foundations and super-structures of adjacent buildings.

Hydrodynamic effects are included as follows: the water mass is considered in the lumped masses of the model and the dynamic pressure of the fluid, generated by the seismic event, is used in the analysis of the structural elements.

Details of the mathematical models used for the various seismic Category I structures are discussed below:

a. Reactor Building

For structural responses in the horizontal direction, the mathematical model consists of three independent cantilever beams representing the steel containment vessel, Shield Building and internal structure respectively. Masses are lumped at 10 selected locations for the steel containment and the shield structure and are lumped at four locations for the internal structure. These three cantilever beams are supported by the rigid foundation mat approximately 43.5 ft. in depth. Rotational and translational springs are connected to the mat simulating soil-structure interactions. This model is shown in Figure 3.7-30. Table 3.7-4 describes mass and stiffness characteristics of this model.

For structural responses in the vertical direction, the model consists of three cantilever beams. Five mass points are used to represent both the steel containment vessel and the Shield Building since in the vertical case, less variation of the structural responses is anticipated. For the internal structure, four mass points are used. This model is shown in Figure 3.7-31. Table 3.7-5 describes mass and stiffness characteristics of this model.

b. Reactor Auxiliary Building

For structural responses in horizontal directions, two mathematical models are used corresponding to the N-S and E-W directions of the building. Each model consists of a single cantilever beam with four lumped masses. The cantilever beam is supported on the rigid foundation mat which in turn is supported by rotational and translational springs simulating the soil-structure interaction. The model is shown In Figure 3.7-32. Table 3.7-6 describes the mass and stiffness characteristics of these two models.

For structural responses in the vertical direction the coupled motion between adjacent floor bays, which is anticipated to be small, is neglected in the model, thus allowing establishment of a relatively simple model which is sufficiently detailed to yield reasonable results. As shown in Figure 3.7-33 this model consists of a single cantilever beam which actually represents the total stiffness of all vertical structural elements of the building. At four different floor elevations, appendages representing the behavior of floor bays are attached to the cantilever beam which is supported by the rigid foundation mat. The foundation mat is supported by vertical soil springs simulating soil- structure interaction. Table 3.7-7 describes the mass and stiffness characteristics of this model.

c. Fuel Handling Building

For structural responses in horizontal directions, two models are used corresponding to the N-S (long) and the E-W (short) directions of the building. Each model consists of a cantilever beam with three lumped masses. The cantilever beam is supported by the rigid foundation mat which in turn is supported by rotational and translational springs simulating soil-structure interactions. For responses in the vertical direction, the model consists of a cantilever beam with three lumped mass points. The models are shown in Figure 3.7-34. Table 3.7-8 describes the mass and stiffness characteristics of these models.

d. Intake Structure

For structural responses in the horizontal directions, two mathematical models are used corresponding to each direction of the structure. The model consists of a cantilever beam with five lumped masses. For the lumped masses essentially buried underground, a lateral spring is used at each lumped mass to simulate the interactions. The cantilever beam is supported on the rigid foundation mat which in turn is supported by rotational and translational springs simulating soil-structure interactions. The models are shown in Figures 3.7-35 and 36. Table 3.7-9 describes the mass and stiffness characteristics of this model.

For structural responses in the vertical direction, the mathematical model consists of a cantilever beam with three lumped mass points. One appendage is attached to the top most lumped mass representing the behavior of the top deck of the intake structure. The cantilever beam is supported on the rigid foundation mat which in turn is supported by the vertical soil spring simulating soil- structure interactions. The model is shown in Figure 3.7-37. Table 3.7-9 describes the mass and stiffness characteristics of this model.

e. Diesel Generator Building

For structural responses in the horizontal directions, two mathematical models are used corresponding to each direction of the building. As shown in Figure 3.7-38 the model consists of three cantilever beams: one representing the structure and two representing the diesel generators and their foundations. All three cantilever beams are supported on the foundation mat which in turn is supported by rotational and translational springs simulating soil-structure interaction. Table 3.7-10 describes the mass and stiffness characteristics of the horizontal model.

For responses in the vertical direction, the model consists of three cantilever beams with one appendage attached most mass of the cantilever beam No. 1 to represent the behavior of the roof of the building. This model is shown in Figure 3.7-38. Table 3.7-10 describes the mass and stiffness characteristics of this model.

f. Main Steam Trestle

For structural responses in the horizontal and vertical directions, two mathematical models are used. Each model consists of two cantilever beams representing the structural steel and reinforced concrete portions of the structure. The cantilevers are supported on a rigid mat base which in turn is supported by rotational and translational springs simulating soil-structure interaction. The models are shown in Figures 3.7-39 and 3.7-40. Tables 3.7-11 and 3.7-12 describe the mass and stiffness characteristics of the models.

g. Steam Generator Blowdown Treatment Facility Building

The Steam Generator Blowdown Treatment Facility (SGBTF) Building was designed as a non- seismic category structure. A seismic analysis was performed to determine the capability of the structure to withstand the operating basis earthquake. Figure 3.7-41 shows finite element model for the SGBTF foundation mat.

For structural responses in the horizontal and vertical directions two mathematical models were used (see Figures 3.7-42 and 3.7-43); both models consist of one cantilever representing the structure supported on the foundation mat, which in turn is supported by rotational and translational springs simulating the soil-structure interaction.

The seismic analysis has indicated that the structure is capable of withstanding the OBE loads.

h. Component Cooling Water Building

For structural responses in the horizontal and vertical directions two mathematical models (Figures 3.7-44 and 45) are used. Both models consist of four cantilevers, one representing the structure and the other three the equipment.

The cantilevers are supported on the foundation mat which in turn is supported by rotational and translational springs simulating soil-structure interaction. Tables 3.7-13 and 14 describe the mass and stiffness characteristics of the models.

i. Condensate Storage Tank Building

The mathematical model for structural responses in the horizontal direction consists of three cantilevers, one representing the concrete structure and the other two, the tank and the fluid (see Figure 3.7-46). The three cantilevers are supported on a rigid mat which in turn is supported by rotational and translational springs simulating soil-structure interaction.

For responses in the vertical direction the mathematical model consists of two cantilevers, one for the concrete structure and one for the tank (Figure 3.7-47). Tables 3.7-15 and 16 describe the mass and stiffness characteristics of the models.

j. Diesel Oil Storage Building

For structural responses in the horizontal direction the mathematical model (Figure 3.7-48) consists of five cantilevers, one representing the structure and four representing the two tanks and the fluid. The cantilevers are supported on the foundation mat which in turn is supported by rotational and translational springs simulating soil-structure interaction.

For the responses in vertical direction the mathematical model (Figure 3.7-49) consists of three cantilevers, one for the structure and two for the tanks.

Table 3.7-17 describe the mass and stiffness characteristics of the models.

3.7.2.1.2 Time History Technique

Once the equivalent multidegree of freedom lumped mass-spring mathematical models are established, the design time histories described in Subsection 3.7.1.2 are applied at the foundation levels of seismic Category I structures in the free field, and structural responses such as frequencies, displacement time history and acceleration time history for each mass point are determined. The design time histories described in Subsection 3.7.1.2 were developed utilizing a linear stochastic model to generate records of filtered, nonstationary shot noise to simulate ground motion accelerograms. Deconvolution procedures are not used in the generation of time history data. A brief account of the methods used in the seismic analysis of structures is as follows:

3.7.2.1.2.1 Equations of Motion

The governing equations of motion for lumped-mass multidegree of freedom systems under external excitation may be written in matrix form as:

$$[M]\{\ddot{\Delta}\} + [c]\{\dot{\Delta}\} + [K]\{\Delta\} = \{F\} \qquad (1)$$

where:

$[M]$ = square mass matrix

$\{\ddot{\Delta}\}$ = column matrix of acceleration vectors

$[c]$ = damping matrix

$\{\dot{\Delta}\}$ = column matrix of velocity vectors

$[K]$ = square stiffness matrix

$\{\Delta\}$ = column matrix of displacement vectors

$\{F\}$ = column matrix of eternal load vectors

A typical horizontal model is shown in Figure 3.7-50. Each matrix is described below.

a. Mass Matrix, [M] :

Every mass point including the foundation mat of the two dimensional horizontal model is allowed two degrees of freedom, namely, translation and rotation. For the vertical model, only one translational degree of freedom is considered. Thus for horizontal direction, the mass matrix consists of mass and rotary inertia terms; for vertical direction only mass terms. For a horizontal model with n mass points

$$[M] = \begin{bmatrix} M_1 & & & & & & & & & \\ & M_2 & & & & & & & & \\ & & \ddots & & & & & & & \\ & & & M_n & & & & & & \\ & & & & M_B & & & & & \\ & & & & & I_B & & & & \\ & & & & & & I_1 & & & \\ & & & & & & & I_2 & & \\ & & & & & & & & \ddots & \\ & & & & & & & & & I_n \end{bmatrix} \qquad (2)$$

Where M_i, I_i (i = 1, n), represent the mass and rotary inertia of the i^{th} lumped mass point; M_B, I_B represent the mass and the rotary inertia of the foundation mat, respectively.

b. Column Matrix of Displacements, $\{\Delta\}$:

The transpose matrix for displacement vectors is arranged as:

$$\{\Delta\}^T = \{U_1, U_2, U_{3,}, \ldots \ldots U_n, U_B, \theta_B, \theta_1, \theta_2, \theta_3, \ldots \ldots \theta_n\} \qquad (3)$$

Where, U_l, U_2, U_n are translational displacements of mass points relative to the ground, U_B, θ_B are the translational and rocking displacements of the foundation, and θ_1, θ_2, --- θ_n are the rotational displacements of the mass points.

The column matrices of velocity and acceleration are the first and second time derivatives of Equation (3).

The torsional degree of freedom is not included in the dynamic analysis.

c. Damping Matrix, [c]:

The damping matrix is composed of composite damping factors as discussed in Subsection 3.7.2.15.

d. Stiffness Matrix, [K]:

The stiffness matrix [K] is formulated by computing the stiffness coefficients for each element and assembling them in the proper sequence to form the complete square matrix. The cantilever beam connecting two lumped masses is considered as a beam element and the effects of bending and shear deformation are included in computing the stiffness coefficients. The effects of equivalent soil springs are also included in the formulation of the stiffness matrix [K]. There are three soil springs, two translational and one rocking, considered in the horizontal direction (refer to Figure 3.7-50).

The first translational spring, K_x, represents the shear effect between the mat surface and the soil while the second translational spring, K_{xx}, considers the bearing stress effect between the side surface of the foundation and the soil. The rocking spring K_θ is considered acting at the rotation center of the foundation, K_x at the bottom, and K_{xx} at the mid-point of the foundation thickness. The effects of these soil springs on the stiffness matrix [K] are as follows:

$$[K]\{\Delta\} = \begin{array}{c} \begin{array}{cccc} \mu_1\cdots\mu_n & \mu_B & \theta_B & \theta_1\cdots\theta_n \end{array} \\ \begin{bmatrix} \circ\bullet\circ\bullet & (K_x+K_{xx}) & (K_x t_2 - K_{xx}\ t_{22}) & \bullet\bullet\bullet\bullet \\ \bullet\bullet\circ\bullet & (K_x t_2 - K_{xx} t_{22} & (K_\Theta + K_x t_2^{\ 2} + K_{xx} t_{22}^{\ 2}) & \bullet\bullet\bullet\bullet \end{bmatrix} \end{array} \begin{Bmatrix} U_1 \\ U_2 \\ \vdots \\ U_n \\ U_B \\ \theta_B \\ \theta_1 \\ \vdots \\ \theta_n \end{Bmatrix}$$

The depiction of t_2, t_{22} are given on Figure 3.7-50, the U's and θ's are displacements corresponding to translations and rotations of the mass points.

In addition to the effects of soil springs in the formulation of stiffness [K], the effect due to relative displacements between interconnected mass points are also considered. The connecting members between mass points are modeled as beams or springs and their effects on the structural response are incorporated in the stiffness matrix.

For the vertical model, one translational degree of freedom in the vertical direction is considered. However, two types of stiffness coefficients are computed for structural elements to form the complete stiffness matrix. The walls and columns are modeled as tension or compression elements while the slabs are considered as plate elements in which bending stiffness provides the oscillating restoring force.

For the vertical model with three mass points and three branch mass points shown in Figure 3.7-51, the stiffness matrix is as follows:

	1	2	3	4	5	6	7
1	K_1+K_4	$-K_1$		$-K_4$			
2	$-K_1$	$K_1+K_2+K_5$	$-K_2$		$-K_5$		
3		$-K_2$	$K_2+K_3+K_6$			$-K_6$	$-K_3$
4	$-K_4$			K_4			
5		$-K_5$			K_5		
6			$-K_6$			K_6	
7			$-K_3$				K_3+K_z

K_1, K_2, K_3 are tension or compression stiffness constants;
K_4, K_5, K_6 are bending stiffness constants.

e. Column Matrix of External Load Vectors, {F}

For the horizontal seismic analysis {F} is the column matrix for the inertia forces due to ground accelerations:

$$\{F\} = \begin{Bmatrix} -M_1\,\ddot{X}_g(t) \\ -M_2\,\ddot{X}_g\,(t) \\ \vdots \\ -M_n\,\ddot{X}_g\,(t) \\ -M_B\,\ddot{X}_g\,(t) \\ \circ \\ \circ \\ \vdots \\ \circ \end{Bmatrix} \qquad (5)$$

Where $\ddot{X}_g$ is the ground horizontal acceleration time history, M_1, M_2, ...M_n are the masses of the mathematical model, and M_B the mass of the foundation. Since the input time history is horizontal, the force terms ($-M_i\,\ddot{X}_g$) correspond to displacements only; the rest are zeros indicating no ground rocking acceleration input. There are n + 1 zeros in Eq (5), n being the number of lumped mass points for a mathematical model.

A similar expression for {F} is obtained for the vertical direction seismic analysis.

3.7.2.1.2.2 Natural Frequencies and Mode Shapes

In calculating the natural frequencies and the mode shapes, the damping term [c] $\{\dot{\Delta}\}$ is ignored and the external load vector in equation (1) is set to zero. The displacement vector {Δ} is assumed to take the form of simple harmonic motion:

$$\{\Delta\} = \{ø\}\text{Sin } \omega t \qquad (6)$$

where:

{ø} = Relative amplitude of mode shape vector

ω = Natural frequency of vibration

After substituting into equation (1) and simplifying, the equations of motion are reduced to the following form:

$$[K\,]^{-1}[M]\{\phi\}-\frac{\{\phi\}}{\omega^2}=0 \qquad (7)$$

Solution to this eigenvalue problem exists only for particular values which correspond to the natural frequencies of vibration of the structure. Equation (7) is solved by the Jacobi method to obtain values of natural frequency of vibration (ω) and their corresponding mode shape vector{φ}.

The analyses of natural frequencies and mode shapes are carried out using EBASCO Computer Program DYNAMIC 2037, described in Subsection 3.8.3.4.

3.7.2.1.2.3 Modal Analysis

After all natural frequencies and their mode shapes are determined, the method of modal analysis is employed to calculate the structural responses. This method actually simplifies the analysis of a multidegree of freedom system into an analysis of several equivalent single degree systems, one corresponding to each normal mode. The governing equation of motion is shown in the following:

$$\ddot{A}_n + 2\beta_n \dot{A}_n + \omega_n^2 A_n = \frac{-\ddot{Y}_{so} f_a(t) \sum_{x=1}^{m} M_x \phi_{xn}}{\sum_{x=1}^{N} M_x \phi_{xn}^2} \qquad (8)$$

where:

A_n = displacement of generalized coordinates for the n[th] mode

β_n = damping coefficient = $\lambda_n \omega_n$

λ_n = percentage of critical damping of the nth mode

ω_n = natural frequency of the nth mode

$\ddot{Y}_{so}$ = maximum ground acceleration

$f_a(t)$ = time function of ground motion,

M_x = mass at the xth level

m = number of masses subjected to inertia $M_x \ddot{Y}_{xo} f(t)$

φ_{xn} = normalized displacement of the mass M_x of the nth mode

N = total number of degrees of freedom

If the ratio of the two summations on the right-hand side of the equation (8) are denoted by P_n, which is defined as the model participation factor of the nth mode, then

$$\ddot{A}_n + 2\beta_n \dot{A}_n + \omega_n^2 A_n = -P_n \ddot{Y}_{so} f_a(t) \qquad (9)$$

Since the values of β_n, w_n and P_n are already known for each normal mode, equation (9), which is actually n independent equations, can be solved separately using the method developed by NC Nigam and PC Jennings[(4)].

The total displacement for the ith mass point is the summation of the displacement of each normal mode, that is

$$U_i(t) = \sum_{n=1}^{N} P_n \phi_{in} A_n(t) \qquad (10)$$

Equation (10) gives the displacement time history for mass point i, and similar expressions are obtained for other structural responses such as acceleration, shear and moment.

In equation (10), in order to assure the participation of all significant modes, all modes are included in the actual computation when the value of N is less than 10. When N is greater than 10, all modes with natural frequencies in the range of 33 cps and below are considered significant.

Equation (10) gives the displacement relative to the ground. The maximum values of this time history are sorted out and used as one of the bases for providing clearances among structures or designing of supports.

The analysis of structural responses using time history method are carried out through the use of EBASCO Computer Program DYNAMIC 2037 (see Subsection 3.8.3.4 for a description of the program).

3.7.2.1.3 Response Spectrum Techniques

In spectral analysis, A_n's of Equation (10) are spectral values taken from the design spectral curves, as those described in Subsection 3.7.1.1. Since spectral values are maximum values, the algebraic absolute sum of equation (10) gives the upper limit of the displacement of any mass. However, all the maximum displacements of all normal modes do not necessarily occur at the same time. For the purpose of design, the square root of the sum of the squares (SRSS) method is adopted for combining modal responses:

$$U_i = \left[\sum_{n=1}^{N} (P_n \phi_{in} A_n)^2 \right]^{1/2} \qquad (11)$$

Where U_i is the displacement for the ith mass point. Similar expressions are established for acceleration, shear, moment, etc.

In actual computation, N is taken as the number of total degrees of freedom of the dynamic model when the response spectrum method of analysis is employed.

The analysis of structural responses using response spectrum method are carried out through the use of EBASCO Computer Program DYNAMIC 2037 (see Subsection 3.8.3.4 for a description of the program).

3.7.2.2 Natural Frequencies and Response Loads

A summary of natural frequencies for significant modes is presented in Tables 3.7-18 through 3.7-23. A summary of structural responses determined by the seismic analysis for major seismic Category I structures is presented in Tables 3.7-24 through 3.7-34. In addition, the floor response spectra at major seismic Category I equipment elevations and points of support are presented in Figures 3.7-52 through 3.7-253.

3.7.2.3 Procedure Used for Modeling

Major seismic Category I structures described in Subsection 3.7.2.1.1, such as Reactor Building, Reactor Auxiliary Building, Fuel Handling Building, and others, are modeled as "Seismic Systems". A seismic system is a soil structure interaction model in which major seismic Category I structures are considered in conjunction with foundation media. Other seismic Category I structures, systems and components that are not designated as "Seismic Systems" are considered as "Seismic Subsystems." In general, the frequencies of systems and subsystems alone have negligible effect on the error due to uncoupling. For the reactor building in particular, studies using seismic models with and without subsystems are made to ensure the coupling effect is minimal. Models with major equipment (such as steam generators and reactor vessels) and the supporting structure (i.e., the internal structure), modeled separately and modeled together, are constructed and the Computer Code STARDYNE is employed. Dynamic responses such as frequencies, accelerations, and response spectra are compared. The differences are found negligible. The reactor internal structure response spectra as shown on Figure 3.7-15 illustrates that the peak acceleration occurs approximately at 3 Hz. The RCS loop major components have a fundamental frequency of >10 Hz. Thereby, the coupling effect between the reactor building and the RCS loop is insignificant. The governing factors are the mass ratio and the frequency ratio; these ratios are considered in the analysis of the systems and subsystems. Analyses of seismic systems are described in this Subsection (3.7.2), and analyses of subsystems are presented in Subsection 3.7.3.

Specifically, the following items are considered in analytical modeling:

a. The mathematical models used in all seismic Category I structures include sufficient mass points and corresponding degrees of freedom to provide an adequate representation of the dynamic characteristics of the seismic Category I structures. For structures with uniformly distributed mass, such as the Shield Building and steel containment, enough degrees of freedom of lumped masses are used such that the number of degrees of freedom is greater than twice the number of modes with frequencies less than 33 cps, as can be seen on Figure 3.7-30 and Table 3.7-18. For structures with floors, less points are used since masses are concentrated on the floor. Lumped masses at designated floor levels consist of combining the floor weights, equipment weights and one-half of the wall and column weights from the adjacent upper and lower floors.

b. The selected locations of mass points account for the stiffness, mass, and damping characteristics of the seismic Category I structures as well as floor elevations and locations, elevations and points of support for major seismic Category I equipment.

c. For horizontal models, major seismic Category I equipment and component masses, such as the reactor vessel, are included in the dynamic lumped mass model, and for vertical models, branch masses are used to account for the difference of dynamic behavior, as shown on Figure 3.7-31.

d. Three independent dynamic models, two horizontal and one vertical, are adopted for each seismic Category I structure seismic analysis. Since the seismic Category I structures are supported by separate foundations, and most of these structures are nearly symmetrical in geometric shape, the coupling effects of the torsional degrees of freedom that are omitted from the three-dimensional models are considered not significant. Nevertheless, torsional effects are incorporated in the design of the structures. The procedure is presented in Subsection 3.7.2.11.

e. Two dimensional seismic models are used for analyzing the Category I structures, since the Category I structures are supported independently and the geometrics of the structures are largely symmetrical. In the two dimensional models, torsional degrees of freedom of mass points are considered as fixed conditions. For the soil springs, the torsional component is also fixed but may be visualized to have a spring with very large torsional spring constant. Original analysis of Waterford No. 3 also utilized two dimensional models without torsional degrees of freedom. In response to NRC questions, a new three-dimensional model with torsional degrees of freedom and torsional soil spring was developed. The accelerations obtained from the new model (with torsional degrees of freedom) compared to those of the original two-dimensional model are smaller. See Tables 3.7-49 through 3.7-51 and Figures 3.7-269 and 3.7-270.

3.7.2.4 Soil-Structure Interaction

All structures are soil supported. Due to the rigid and massive behavior of the seismic Category I structures and the relatively soft soil characteristics, it is anticipated that considerable rocking and translating motions of structures may take place during a severe earthquake at the site. To include these motions in the seismic analysis, it is considered appropriate to model the soil into rotational and translational springs to allow for these additional degrees of freedom. The spring constants are calculated using the following formulas from Reference 3:

a. For a circular foundation; horizontal excitation model

$K_{\varnothing} = \dfrac{8Gr^3}{3(1-\mu)}$ Rocking

$K_X = \dfrac{32(1-\mu)Gr}{7-8\mu}$ Shear

$k_{XX} = \dfrac{4G}{1-\mu}\sqrt{\dfrac{2rh}{\pi}}$ Bearing

Where

G = shear modulus of soil

μ = Poisson's ratio of soil

h = depth of foundation

r = radius of foundation

b. For a circular foundation; vertical excitation model

$$k_{xx} = \frac{4Gr}{1-\mu} \quad \text{Bearing}$$

c. For a rectangular foundation; horizontal excitation model

$$K_{\varnothing} = \frac{8G}{3(1-\mu)}\left(\frac{BL^3}{3\pi}\right)^{0.75} \quad \text{Rocking}$$

$$K_x = \frac{32(1-\mu)G}{7-8\mu}\sqrt{\frac{BL}{\pi}} \quad \text{Shear}$$

$$k_{xx} = \frac{4G}{1-\mu}\sqrt{\frac{Bh}{\pi}} \quad \text{Bearing}$$

Where

B = dimension of mat perpendicular to earthquake direction

L = dimension of mat parallel to earthquake direction

d. For a rectangular foundation, vertical excitation model

$$k_{xx} = \frac{4G}{1-\mu}\sqrt{\frac{A}{\pi}} \quad \text{Bearing}$$

Where

A = horizontal bearing area of foundation

From laboratory soil testing and analyses (see Subsection 2.5.4.2) the proper Young's modulus, E, used for the calculation of soil spring constants for all seismic Category I structures is determined as follows:

Reactor Building	E = 40,000 psi
Reactor Auxiliary Building	E = 40,000 psi
Fuel Handling Building	E = 35,000 psi
Diesel Generator Building	E = 30,000 psi

Intake Structure	E = 17,400 psi
Main Steam Trestle	E = 40,000 psi
All Missile Protection Enclosures	E = 30,000 psi

For all above cases the Poisson's ratio for soil is 0.25. As discussed in Subsection 2.5.4.4 Poisson's ratio varies with strain. For the early stages of a first loading of a dense sand, when intermediate strain levels are developed and particle rearrangements are important, "μ" typically has values of about 0.25 as selected above and is consistent with the anticipated soil strains of 10^{-3} to 10^{-4} in/in.

In order to include any uncertainties of the selected soil modulus on the structural responses, a range of soil moduli within ± 20 percent of the selection values are used.

Table 3.7-35 provides a tabulation of all soil-supported seismic Category I structures and the depth of each of the various soil layers to the bottom of the excavation line at elevation -60 ft. The soil layers are shown in Figures 2.5-9 through 2.5-14, 2.5-20 and 2.5-21.

The effects of soil-structure interaction for the seismic Category I structures are considered in the seismic analysis by means of providing equivalent rotational and translational springs based on the theory of rigid plates on elastic half space. To include the effects of any uncertainties of foundation soil engineering properties, a parametric study is made to vary soil properties by ± 20 percent. The maximum responses resulting from the parametric study are used for the actual design of structures.

The analysis demonstrates that by varying the soil properties (Young's modulus and Poisson's ratio) by ± 20 percent, the period at which a floor spectrum peak occurs varies by approximately minus 19 percent to plus 16 percent; a plus or minus 20 percent variation in the period is used in the actual design. An increase in the concrete strength by 50 percent produces a negligible effect on the period at which a floor spectrum peak occurs.

The appropriateness of the methods used in calculating soil-structure interaction are based on the following:

a. Soil properties, such as shear modulus and Poisson's ratio are determined both from laboratory and field tests. The shear modulus versus strain relationship is established as shown in Subsection 2.5.4.4.

b. The shear modulus used for the calculation of equivalent foundation spring constants is consistent with the anticipated soil strain during an SSE. Subsection 2.5.4.4 addresses the procedures used in determining the anticipated soil strain levels during earthquakes.

c. The maximum building embedment depth from plant grade is approximately 43 feet for the Reactor Building. The embedment is approximately 25 percent of the Reactor Building diameter and less than 20 percent of the total building height. In general this type of embedment could be neglected in soil-structure interaction calculations without a significant effect on the structural responses when using the concept of a rigid body resting on an elastic half space.

However, in the actual calculation, the shallow embedment effects are included by providing a side spring at approximately the middle of the embedment.

d. To verify that the model selected and method used for the soil structure interaction effects are appropriate, a set of calculations are made using different values for the side spring constant. The insignificant differences obtained by varying the spring constant indicates that the Reactor Building embedment has very little effect on the overall structural response (refer to Table 3.7-36).

e. In order to demonstrate the adequacy of the lumped mass-spring model approach, a generic study is performed for a nuclear plant island similar to St. Lucie, for which a finite element analysis is available. The floor response spectra developed for this plant from a lumped mass-spring model are compared with the response spectra obtained from the finite element analysis and found to be conservative (see Section 2.5).

3.7.2.5 Development of Floor Response Spectra

A time history method of analysis is used to develop floor response spectra. The procedure adopted is similar to that developed by Nigam and Jennings[(4)].

Consider a viscously damped, simple oscillator subjected to the floor acceleration a(t) obtained by analysis of a seismic system, the equation of motion of the oscillator is given by

$$\ddot{X} + 2\beta\omega\dot{X} + \omega^2 X = -a(t)$$

in which

X = displacement relative to the floor.

β = the fraction of critical damping,

ω = the natural frequency of vibrations of the oscillator and

Assuming that a(t) may be approximated as segmentally linear function for any two consecutive time steps, closed form solution for displacement X and velocity $\dot{X}$ are obtained with appropriate initial conditions. The absolute acceleration, $\ddot{Z}_i$, of the mass at time t_i is given by

$$\ddot{Z}_i = \ddot{X}_i + a_i = -(2\beta\omega\dot{X}_i + \omega^2 X_i)$$

The response spectra for selected values of damping and natural frequency are the maximum values obtained from the following expressions:.

$$S_d(\omega,\beta) = \underset{i=1,n}{\text{Max}} \left[X_i\ (\omega,\beta) \right]$$

$$S_v(\omega,\beta) = \underset{i=1,n}{\text{Max}} \left[\dot{X}_i\ (\omega,\beta) \right]$$

$$S_a(\omega,\beta) = \underset{i=1,n}{\text{Max}} \left[\ddot{Z}\ \ (\omega,\beta) \right]$$

In which S_d, S_v and S_a are the spectral values of displacement, velocity and acceleration respectively; and n is the total number of discrete points at which the response is obtained.

For a floor of interest, acceleration response floor spectra are developed independently for three directions, two horizontal and one vertical. To compute the spectrum ordinates, a total 130 closely spaced frequencies ranging from 0.5 Hz to 50 Hz are selected to produce an accurate spectrum curve. The frequency intervals for calculation of response spectra are:

Frequency Range (Hz)	Increment (Hz)
0.5-2.5	0.050
2.5-8.0	0.157
8.0-25	0.486
25-50	1.250

The frequency intervals shown above have more points than those recommended by Regulatory Guide 1.122, "Development of Floor Response Spectra for Seismic Design of Floor-Supported Equipment or Components," September 1976 (R0).

3.7.2.6 Three Components of Earthquake Motion

The general methodology (square root of the sum of squares method) described in Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," February 1976 (R1) is utilized. The seismic analysis of seismic Category I structures such as the Reactor Building are carried out using two-dimensional lumped-mass spring mathematical models along with response spectrum techniques, as described in Subsection 3.7.2.1. For each seismic Category I structure, there are three mathematical models, two horizontal and one vertical, and the responses such as acceleration and displacements at each mass point for three orthogonal directions are obtained separately.

Then a detailed three-dimensional static model for each seismic Category I structure is employed using the maximum values of the structural response of three directions obtained as input. The resulting codirectional responses caused by each of the three components of earthquake motion at a particular point of structure are combined by SRSS technique at force level (moments, shears, etc).

3.7.2.7 Combination of Modal Responses

When the response spectrum method of analysis is used to determine the dynamic response of seismic systems, the response is obtained as the square root of the sum of squares (SRSS) of the responses from individual modes, without taking into consideration the effect of closely spaced modes, as recommended by Regulatory Guide 1.92 (R1). (Modal frequencies are considered closely spaced when there is less than ± 10 percent difference between them).

To ensure that the SRSS technique yields conservative results, studies have been made using the response spectrum method taking the effect of closely spaced modes into account. Results show that the differences of structural responses due to closely spaced modes are negligible (Table 3.7-34).

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

The Turbine Building and the Cask Handling Facility are the only non-seismic Category I buildings with possible interaction with seismic Category I structures. The structural frames of the Turbine Building are designed to withstand seismic motion such that it will not collapse and impair the integrity of seismic Category I structures or components. There is no safety related equipment located within the Turbine Building. The structural frames of the Cask Handling Facility upper level support steel are designed to withstand seismic motion such that it will not collapse and impair the integrity of the seismic Category I Fuel Handling Building or Cask Crane Supports. There is no safety related equipment located within the Cask Handling Facility.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

The two horizontal and the vertical floor response spectra at the various floor or other equipment-support locations are computed from the time history motions of the floor locations of the supporting seismic system. In general, spectrum peaks normally would be expected to occur at the natural frequencies of the supporting structure, as can be seen on Figures 3.7-52 through 3.7-253.

To account for uncertainties in the structural frequencies due to variations of structural and soil properties, and soil-structure interactions, the computed floor response spectra are smoothed and the peaks associated with each of the structural frequencies are broadened. The amount of peak widening associated with the structural frequency is ±20 percent, as presented in Figures 3.7-52 through 3.7-253.

3.7.2.10 Use of Constant Vertical Static Factors

Constant vertical factors are not used as vertical response loads for seismic design of seismic Category I structures.

A multi-mass dynamic analysis procedure is used for the vertical response loading for the seismic design of buildings and floors. The methods of analysis used for the vertical dynamic analysis are the same as those described in Subsection 3.7.2.1.

3.7.2.11 Method used to Account for Torsional Effects

A static factor is employed for torsional effects in the seismic analysis of seismic Category I structures. The horizontal seismic shear for a structure at a given level is applied at the center of mass of the structure at that level to calculate the moment resulting from the multiplication of this force times the distance between the center of mass and the center of rigidity. The static factor therefore is the distance between the center of mass and the center of rigidity which varies between each level in each structure. If this distance is less than five percent of the length of the side, then the five percent times the length is used.

The resulting torsional moment is distributed along the exterior walls in accordance with methods outlined in Chapter 7 of Reference 5.

3.7.2.12 Comparison of Responses

In order to provide a check on the seismic analysis of seismic Category I structures, an analysis using both the time history method and response spectrum method is conducted. Tables 3.7-24 through 3.7-33 give the response at selected points for major seismic Category I structures using both these methods. These responses illustrate approximate equivalency between these two methods.

3.7.2.13 Methods for Seismic Analysis of Dams

There are no seismic Category I dams associated with St. Lucie Unit 2.

3.7.2.14 Methods to Determine Category I Structure Overturning Moments

The horizontal seismic response loads acting at their corresponding mass point elevations, determined from the dynamic analysis described in Subsection 3.7.2.1, are used in computing the overturning moments about the base of the structure.

Vertical earthquake effects are considered by deducting from the dead load righting moments the vertical response loads determined by the vertical dynamic analysis. Buoyancy, where it is present, is also considered in the summation of moments.

Where structures are embedded in soil strata, resisting soil pressures acting as righting moments are not included in the net overturning moment. However, where dynamic effects of soil strata contribute tn overturning moments these horizontal loads are considered by including this additional overturning in the final summation. The resulting soil reactions, with the appropriate combination of vertical response loads added to give the maximum effects, are compared against the allowable dynamic soil pressures to assure compliance to the criteria.

3.7.2.15 Analysis Procedure for Damping

The procedure used to determine the composite damping matrix is the composite modal damping approach. This approach is based on the use of stiffness as a weighting function in generating the composite modal damping. The formulations lead to:

$$\beta = \sum_{n=i}^{MN} \frac{\{\phi_n\}^T \beta_r [k]_r \{\phi_n\}}{\{\phi_n\}^T [k] \{\phi_n\}} \qquad n = 1,2 \text{ -- } MN$$

Where:

β_n = equivalent model damping ratio of the nth mode,

β_r = damping ratio associated with component, r,

$\{\phi_n\}$ = eigenvector fnr the n^{th} mode,

$[k]_r$ = stiffness matrix for component r,

$[k]$ = assembled stiffness matrix,

MN = number of degrees of freedom of the whole system.

A summary of composite modal damping ratios for the Reactor Building is presented in Table 3.7-37.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

3.7.3.1 Seismic Analysis Methods

3.7.3.1.1 Non-NSSS Seismic Category I Piping

Seismic Category I piping other than the reactor coolant loop piping is seismically analyzed as follows:

ASME Code Class 1 piping 1 1/4 in. and larger for all design temperatures is analyzed by the modal response spectra method as described in this subsection. ASME Code Class 1 piping one in. and under is analyzed either by modal response spectra method or modified equivalent static load method.

ASME Code Class 2 and 3 piping 2 1/2 in. nominal size and larger, with design temperature above 275 F is analyzed by either modal response spectra method or modified equivalent static load method.

ASME Code Class 2 and 3 piping 2 1/2 in. nominal size and larger, with design temperature below 275 F is analyzed by either modified equivalent static load method or simplified seismic analysis method.

ASME Code Class 2 and 3 piping from 1 1/4 in. to two in. nominal size with design temperature above 275 F is analyzed by either modified equivalent static load method or simplified seismic analysis method.

ASME Code Class 2 and 3 piping from 1 1/4 to two in. nominal size with design temperature up to 275 F and piping from 1/2 in. to one in. nominal size for all design temperatures are analyzed by simplified seismic analysis method.

a. Modal Response Spectra Method

   Dynamic analysis by modal response spectra method is described as follows:

   1. Basic Assumptions

      a. The system is linearly elastic.

b. Masses are lumped at discrete points and are connected by weightless elastic members. The maximum spacing between mass points may not exceed one half the distance for which the frequency of a simply supported beam would be 20 Hz.

c. Each mass point has six degrees of freedom, except for points indicated as restrained in a given direction.

d. The system is anchored at one or more locations and these anchor points are fixed for the determination of natural frequencies and mode shapes.

e. Dynamic loadings in the three coordinate directions are determined separately and combined on the basis of excitation occurring in the vertical and two orthogonal directions at the same time.

f. The mass polar moment of inertia, i.e., the mass component involved in rotation, is negligible.

g. Damping is viscous and assumed constant for all modes.

h. Increased flexibility due to pipe bends is included in the analysis.

2. Equations of Motion

The stiffness matrix method of natural mode analysis is employed to determine natural periods of vibration and the associated mode shapes.

The equations of motion for the piping system may be written as

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = \{F\} \qquad (1)$$

[M] = Diagonal matrix of lumped masses, the rows and columns of which are arranged to correspond to the components of the stiffness matrix. The masses effective in the three coordinate directions are taken to be equal to the total mass assumed lumped at the point under study.

$\{\ddot{\Delta}\}$ = Column matrix of acceleration.

[K] = Square symmetric matrix of stiffness coefficients including the effects of axial deformation, bending and torsional shear in the three coordinate directions.

$\{\Delta\}$ = Column matrix of displacement.

{F} = Column matrix of external loads.

Each pipe section has the following properties:

E = Modulus of Elasticity

μ = Poisson's Ratio

I = Moment of Inertia

A = Cross-sectional Area

L = Length

From these properties the characteristics of the section are computed:

$$G = \frac{E}{2(1+\mu)}; GJ = \frac{EI}{1+\mu}; \alpha = \beta = \frac{2(1+\mu)}{AE}$$
$$\varepsilon = \eta = \frac{1}{EI}; \gamma = \frac{1}{AE}; \lambda = \frac{1}{GJ} = \frac{1+\mu}{EI}$$

The end flexibility of the section is contained in the 6 x 6 matrix ϕ:

$$\phi = \begin{Bmatrix} \alpha L + \frac{\varepsilon L^3}{3} & 0 & 0 & 0 & \frac{\varepsilon L^2}{2} & 0 \\ 0 & \beta L + \frac{\eta L^3}{3} & 0 & -\eta\frac{L^2}{2} & 0 & 0 \\ 0 & 0 & \gamma L & 0 & 0 & 0 \\ 0 & -\eta\frac{L^2}{2} & 0 & \eta L & 0 & 0 \\ \frac{\varepsilon L^2}{2} & 0 & 0 & 0 & \varepsilon L & 0 \\ 0 & 0 & 0 & 0 & 0 & \lambda L \end{Bmatrix}$$

A rotation matrix [R] is established to bring the pipe section into the general coordinate system. This matrix is based on the orientation and location of the section in the overall system.

The flexibility in the generalized coordinate system is:

$$[\phi_G] = [R]\,[\phi]\,[R]^T$$

The flexibilities $[\phi_G]$ are accumulated for each mass point and the stiffness coefficients are computed as

$$[K\quad]_A = [\,\phi_G\,] - 1$$

and assembled into the overall stiffness matrix [K].

For the determination of natural frequencies and mode shapes, equation (1) is solved by first setting the external loads F equal to zero and the displacement vector $\{\Delta\} = \{\delta\} \sin \omega t$.

Then:

$$\{\ddot{\Delta}\} = -\{\delta\}\omega^2 \sin \omega t$$

Equation (1) becomes:

$$[K]\{\delta\} = \omega^2 [M]\{\delta\} \qquad (2)$$

This characteristic eigenvalue equation is solved by iterative techniques to determine the natural frequencies and mode shape vectors {δ} of the system. This generalized procedure permits the analysis of multiple fixed branched and looped systems with multiple lumped masses as well as simple single branch systems.

3. Modal Analysis

The response of each mode of vibration is computed as:

$$R_{nd} = \sum_{i=1}^{N} M_i \delta_{ind}$$

where:

n = Mode number

d = Direction X, Y, Z

N = Total number of lumped masses

M_i = Mass for i^{th} component

δ_{ind} = Shape factor (i^{th} component for the n^{th} mode shape for direction d)

M_n = Effective mass = $\dfrac{\sum_{d=1}^{d_3} \left(\sum_{i=1}^{N} M_i \delta_{ind} \right)^2}{\ }$

The disturbance factor for the earthquake in two horizontal coordinate directions and the vertical direction is defined as:

$$D_n = \left\{ [(S_{and_1}) \quad (P_{Fnd_1})]^2 + [(S_{and_2}) \; (P_{Fnd_2})]^2 + [(S_{and_3}) \; (P_{Fnd_3})]^2 \right\}^{1/2}$$

where

d_1, d_2 and d_3 indicate the three directions of motion, and

S_{and} = Floor response spectral acceleration in the d direction for the nth mode with consideration of + 20 percent variation in period.

P_{Fnd} = $\frac{Rnd}{Mn}$ = Participation factor for the n^{th} mode and d^{th} direction.

The modal inertia forces for each mode of vibration are then computed as:

$$F_{idn} = M_i \, \delta_{ind} D_n$$

All significant modes for piping frequencies 33 Hz or less are included in the analysis.

4. Stress and Displacement Analysis

The modal inertia forces Fidn are utilized as response loads in a static analysis to generate modal internal forces F^*_{idn}, moments M^*_{idn} and displacement Δ_{idn}, the final stresses σ_i resulting from the earthquake disturbance in two horizontal coordinate directions and the vertical direction are computed as the maximum resulting from combining the modal stresses by the square root of the sum of squares method. The final inertia of shear forces, moment and displacement to be used for design are determined by combining the results of the modes considered on the same basis, i.e.,

$$F^*_{id} = \left(\sum_n F^{*\,2}_{idn} \right)^{1/2}$$

$$M^*_{id} = \left(\sum_n M^{*\,2}_{idn} \right)^{1/2}$$

$$\Delta_{id} = \left(\sum_n \Delta_{idn}^2 \right)^{1/2}$$

Where

$$\Delta_{idn} = \frac{\delta_{idn}\, D_n}{\omega_n^{2}}$$

and

$$\sigma_i = \left(\frac{\sum_{d=d_1}^{d_3} M^{*}_{id}{}^{2}}{Z} \right)^{1/2}$$

Where "Z" equals the section modulus of pipe cross section.

Dynamic analysis for piping system combines all the modes in flexible and resonant region together with residual terms accounting for higher modes in rigid region.

The responses of the closely-spaced modes are combined by the summation of the absolute values method and, in turn, combined with the response of the remaining significant modes by the square root of the sum of the squares method. Modal frequencies are considered closely spaced when their difference is less than + 10 percent of the lower frequency. The grouping method delineated in Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," February 1976 (R1) has been used to combine the modal responses.

Significant maximum relative seismic displacement among supports of seismic Category I piping and equipment are considered. Interaction from branch connection and supports are considered. All the seismic restraints are considered as rigid constraints.

The computer program (PIPESTRESS 2010) used for the static analysis utilizes the same stiffness matrix method as that described for the modal response spectra method. The program automatically determines forces, moments and deflections in the three coordinate directions and the stresses at selected points in the piping system. The intensification factor is applied to both bending moment and torsional moment. The computer program PIPESTRESS 2010 is discussed in Subsection 3.9.1.2.1.1.

b. Modified Equivalent Static Load Method:

(Simplified dynamic analysis)

For piping systems which would normally be analyzed by the modal response spectra method, if the first mode period of the piping is 70 percent or less of the first mode period of the structure, a modal response spectra method is not performed. The first mode period of the structure is used for analysis. If any floor response spectra curve shows a meaningful peak for other than the first mode period, that mode period is used as the basis for the design of the piping. For instance, Reactor Building internal structure horizontal floor response spectra indicate predominant peak at 3.4 Hz (0.29 sec in period) which represent the second mode of the Reactor Building while the predominant peak on vertical floor spectra are near 2.3 Hz which are correlated to the first mode of the Reactor Building. Therefore a preset period of 0.20 second is selected as the upper bound. The piping system is modeled in the same way as it is described in the modal response spectra method. Detailed dynamic analysis is employed for determining the first mode period of the piping, then a static analysis is performed using the PIPESTRESS 2010 program and an acceleration value of 1.5 times the maximum value of the floor response spectrum in the period range equal to or less than the first mode period of the piping.

To justify this procedure for seismic analysis of piping, three sample problems (Figures 3.7-254 through 3.7-257) are presented using both modal response spectra and frequency based static methods. The frequency based static method uses 1.5g in each of the three orthogonal directions. The modal response spectra method utilizes 18, 14 and 16 modes for sample Problems 1, 2 and 3, respectively. In each case, the analysis includes one mode higher than the number of modes required to reach 33 Hz. For all modes, the acceleration in two horizontal and one vertical directions is taken as 1.0g. The periods for the analyzed modes for all sample problems are between 0.19 seconds and 0.02 seconds.

Because of the difference in the consideration of load distribution between static and dynamic analyses, the maximum computer stress will not always appear at the same point. However, with the consideration in the procedure delineated in Subsection 3.7.3.1.1b, the maximum stress will be higher for the "Modified Equivalent Static Load" method which is a frequency based static analysis.

The original calculations for the three sample problems were performed on the basis of a flat response spectra of 1.0g acceleration. For an actual response spectra generated from ordinary structure, the response values beyond the resonant region usually decay rapidly. This is evidently demonstrated on Figures 3.7-271 through 3.7-273. The three sample problems have been recalculated for both flat response spectra and this more realistic response spectra using combination methods of NRC Regulatory Guide 1.92 Rev 1 and ASME Code Summer 1973 version for Code Class 2 and 3 piping.

Tables 3.7-38a and 38b represent a comparison of pipe stresses computed by both "Modified Equivalent Static Load Method" and the mode response spectra analysis. Table 3.7-38a is the result based on a flat response spectra of 1.0g acceleration. Table 3.7-38b is the result based on the enveloped response on Figures 3.7-271 through 3.7-273. In all cases, the maximum computed stress is higher for the frequency based static analysis. As it is shown in Table 3.7-38b, this is even more evidential in the comparison based on a real response spectra.

While sample problems 2 and 3 were arbitrarily picked from actual piping systems, the sample problem 1 (see Table 3.7-38a) does not reflect any normal restrained piping system. It was purposely modeled and restrained to exemplify the possible dynamic response of the piping. In general practice, the restraints are placed near the valves, corners and offsets as much as possible.

Equipment vendors are requested to provide calculations based on a realistic simple model as an alternative to detailed dynamic analysis.

c. Simplified Seismic Analysis Method

The simplified method of analysis for piping system consists of locating restraints such that the period of the first mode of vibration does not exceed the preset value of 70 percent of the first mode period of the supporting structure. This method involves the use of appropriate and comprehensive charts and tabulations that include correction factors for the concentrated loads, branch connections and other effects. The piping system is studied for loading effects in each of the three coordinate directions to assure that it is adequately restrained in all directions. An additional analysis is performed to evaluate the thermal effects of the restraints on the system. This is done by means of charts that define the minimum distance required for placing restraints adjacent to any expanding leg in order to stay within allowable stress limits.

d. Equipment-Frequency Based Static Method of Analysis

If the frequency analysis of the equipment yields rigid characteristics, i.e., the natural period of vibration of first mode of supported equipment equal to or less than 0.03 seconds, the manufacturer may apply the seismic acceleration coefficients obtained from applicable response spectra curves and perform the static analysis on the equipment and supports (see Appendix 3.9A).

For rigid equipment modeled as one or more degree of freedom system, the equivalent static load factor is the response acceleration on the floor response spectra at the fundamental period of the equipment. For rigid valves, the equivalent static load factors are 3.0g horizontal, 2.0g vertical for safe shutdown earthquake (SSE) and 1.5g horizontal 1.0g vertical for operating basis earthquake (OBE).

Non-rigid valves (less than 33 Hz) are modeled in piping system either with sufficient detail or an amplification factor applied to the center of gravity of the valve assembly.

3.7.3.1.2 Seismic analysis – Reactor Coolant System

3.7.3.1.2.1 General

The seismic analysis of the Reactor Coolant System components is performed using normal mode theory in conjunction with time history and response spectrum techniques, as appropriate.

Time history techniques are employed in the analysis of the reactor vessel, the two steam generators, the four reactor coolant pumps and the interconnecting reactor coolant piping. In the analysis of these components, a single composite mathematical model, which includes integral representations of each of the components and connecting piping, is employed to account for the dynamic interaction effects between components. The analysis of these dynamically coupled multisupported components utilizes different time dependent input excitations applied simultaneously to each support.

The analysis of the pressurizer, spray line and surge line employs separate, uncoupled, mathematical models and utilizes response spectrum techniques.

The input data, time histories and response spectra applied in the analyses are provided by the analysis of the Reactor Building and internal support structure described in Subsection 3.7.2.

For OBE a damping factor of one percent of critical damping is used for each mode. In the SSE analysis a damping factor of two percent of critical damping is used for each mode.

The two original steam generators have been replaced with replacement steam generators that are approximately the same physical size. As stated in reference 26, there are no changes to interfaces with the reactor coolant, main feedwater, or main steam systems, and no significant changes to major component supports or piping systems. Evaluations of the differences between the RSGs and OSGs are presented in reference 26, and the evaluations confirm that the use of the RSGs meets the existing UFSAR design basis acceptance criteria.

As part of the licensing support for replacement of the steam generators, a structural evauation was performed (Section 4.4, reference 26), which demonstrates that the RCS with the RSGs remains in compliance with design basis requirements. This was accomplished through a direct comparison of loads and displacements between structural models of the OSG and RSG.

3.7.3.1.2.2 Mathematical Models

In the description of the mathematical models which follow, the spatial orientations are defined by a set of orthogonal axes where Y is in the vertical direction, and X and Z are in the horizontal plane, in the directions indicated on the appropriate figures. The mathematical representation of the section properties of the structural elements employs a 12 x 12 stiffness matrix for the three dimensional space frame models. Elbows in piping runs include the in-plane/out-of-plane bending flexibility factors as specified in ASME Code, Section III.

a. Reactor Coolant System - Coupled Components

   A schematic diagram of the composite mathematical model used in the analysis of the dynamically coupled components of the Reactor Coolant System is presented on Figure 3.7-258. This model includes 18 mass points with a total of 45 dynamic degrees of freedom. The mass points and corresponding dynamic degrees of freedom are distributed to provide appropriate representations of the dynamic characteristics of the components as follows: the reactor vessel, with internals, is represented by four mass points with a total of 11 dynamic degrees of freedom; each of the two steam generators are represented by three mass points with a total of seven dynamic degrees of freedom; and each of the four reactor coolant pumps are represented by two mass points with a total of five dynamic degrees of freedom. The relatively small mass of the interconnecting reactor coolant piping is addressed in Amendment 20 update.

   This mathematical model, shown on Figure 3.7-258, provides a complete three dimensional representation of the dynamic response of the coupled components to seismic excitations in both the horizontal and vertical directions. The mass is distributed at the selected mass points and corresponding translational degrees of freedom are retained to include rotary inertial effects of the components. The total mass of the entire coupled system is dynamically active in each of the three coordinate directions.

The representation of the reactor vessel internals is formulated in conjunction with the analysis of the reactor vessel internals discussed in Subsection 3.7.3.14 and is designed to simulate the dynamic characteristics of the models used in that analysis. The model is used to generate time histories of absolute accelerations at the reactor vessel flange which are used as forcing functions in the analysis of the reactor vessel internals.

Interaction of the RCS main loop piping and the major components is accounted for directly in the time history analysis (Subsection 3.7.3.1.2.3(c)) of the composite coupled model. Treatment of hydrodynamic effects and non-linear response of the reactor internals and fuel is discussed in Subsection 3.7.3.14.

b. Pressurizer

The mathematical model employed in the analysis of the pressurizer is shown schematically on Figure 3.7-259. This lumped parameter, three dimensional model provides a multi-mass representation of the axially symmetric pressurizer and includes five mass points with a total of 11 dynamic degrees of freedom.

c. Surge Line

The lumped parameter, multi-mass mathematical model employed in the analysis of the surge line is shown schematically on Figure 3.7-260. The surge line is modeled as a three dimensional piping run with end points anchored at the attachments to the pressurizer and the reactor vessel hot leg piping. In the definition of the mathematical model, nine mass points with a total of 27 dynamic degrees of freedom are selected to provide a complete three dimensional representation of the dynamic response of the surge line. All supports and restraints defined for the surge line assembly are included in the mathematical model. The total mass of the surge line is dynamically active in each of the three coordinate directions.

d. Spray Line

The lumped parameter, multi-mass mathematical model employed in the analysis of the spray line is shown schematically on Figure 3.7-261. The spray line is modeled as three dimensional piping runs with end points anchored at the attachments to the pressurizer and the reactor vessel inlet piping. In the definition of the mathematical model, 79 mass points with a total of 194 dynamic degrees of freedom are selected to provide a complete three dimensional representation of the dynamic response of the spray line. All supports and restraints defined for the spray line assembly are included in the mathematical model. The total mass of the spray line is dynamically active in each of the three coordinate directions.

3.7.3.1.2.3 Calculations

a. General

As applied in the analysis, the simultaneous equations of motion for linear structural systems with viscous damping can be written as follows[6]:

$$M\ddot{X} + C\ddot{X} + KX = - M\ddot{Y} - K_{ms}\ X_s$$

where:

M = diagonal matrix of lumped masses

C = square symmetrical damping matrix

K = square symmetrical stiffness matrix which defines the mass point force to displacement relationship

$\ddot{Y}$ = column matrix with elements equal to the absolute accelerations degree of freedom of the structural system

K_{ms} = rectangular matrix of stiffness coefficients which defines the mass point force to non-datum support displacement relationship

X_s = column matrix of displacements relative to the datum at non-datum supports

X = column matrix of mass point displacements relative to the datum

$\dot{X}$ = column matrix of mass point velocities relative to the datum

$\ddot{X}$ = column matrix of mass point acceleration relative to the datum

In this form, the equations define the dynamic response of a multi-mass structural systems subjected to time-dependent support motion. In the analysis of systems with multiple supports, such as the coupled components of the Reactor Coolant System, the equations provide for different time dependent input motions at each of the supports. In this case, one of the supports of the system is designated the reference, or datum, from which the motions of all other points of the structural system are measured. The reactor vessel support is designated as the datum in the analyses of the coupled components of the Reactor Coolant system.

Normal mode theory, described in References 6 and 7, is employed to reduce the equations of motion to a system of independent equations in terms of the normal modes for the time-history and spectrum analyses of the reactor coolant system components. In the analyses, the dynamic response of the components is determined for seismic input excitations in each of the three orthogonal global coordinate directions: X (horizontal), Y (vertical) and Z (horizontal).

b. Frequency Analysis

An eigenvalue analysis is performed utilizing the ICES STRUDL II* computer code[8] (see Subsection 3.9.1.2.2.2.1) to calculate the mode shapes and natural frequencies of the composite mathematical models. Modifications to the standard ICES STRUDL II program have been implemented by Combustion Engineering to include a double precision Jacobi diagonalization procedure in the eigenvalue analysis and to provide appropriate influence coefficients and stiffness matrices for use in the response of reaction calculations. EC283094

The natural frequencies and dominate degrees of freedom calculated are shown in Table 3.7-39 for all modes used in the analysis of the Reactor Coolant System, spray line, surge line, and pressurizer.

c. Mass Point Response Analysis

The time-history mass point responses to seismic excitation are computed using TMCALC,*[9] a CE code. This code performs a closed form integration of the equations of motion for singly or multiply supported dynamic systems utilizing normal mode theory,[7] and Newmark's Beta-Method with beta equal to 1/6[10]. For the multiple supported systems, the separate time-histories of each support are imposed on the system simultaneously. The results are time-history responses of the mass points. The analysis of the Reactor Coolant System utilizes modal data for all frequencies through 50 cps. EC283094

The mass point responses resulting from the spectrum analysis are determined utilizing the ICES/STRUDL II Program. This code performs a normal mode response spectrum analysis resulting in the modal inertial loads at each mass point. The responses of the pressurizer are found using the response spectra for the pressurizer support. The mass point responses of the surge line, and spray line are found using the envelopes of the support spectra of the interconnected components.

d. Seismic Reaction Analysis

The dynamically induced loads at all system design points due to the time history support excitations and mass point responses are calculated utilizing FORCE*, a CE computer code. The code performs a complete load analysis of the deformed structure at each incremental time step by computing internal and external system reactions (forces and moments) by superimposition of the reactions due to the mass point displacements and the non-datum support displacements as follows: EC283094

$$\mathrm{R(t) = C_m X_m(t)\ C_s X_s(t)}$$

* This computer code does not apply to the replacement steam generators (RSGs). The corresponding RSG computer code is described in Section 2.8.8 of the RSG Report, 77-5069878-004 (Reference 26), if applicable. "Methods of Evaluation" within the scope of 10 CFR 50.59(c)(2)(viii) that are used for specific analyses for the RSG computer codes are addressed in the RSG Report. EC283094

where:

R(t)	=	the matrix of all components of the reactions at the system design points
C_m	=	the matrix of mass point displacement influence coefficients
$X_m(t)$	=	the column matrix of time history mass point displacements relative to the datum at each time step
C_s	=	the matrix of support displacement influence coefficients
$X_s(t)$	=	the column matrix of time history support displacements relative to the datum at non-datum supports at each time step.

The support and mass point displacements due to each direction of horizontal and vertical seismic excitation are determined at each time step. The maximum component forces of each reaction for the entire time domain, and its associated time of occurrence, are selected.

The square root of the sum of the squares method is the procedure normally used to combine the modal responses when the modal analysis response spectrum method of analysis is employed. The procedure is modified only in two cases:

a. In the analysis of simple systems where three or less dynamic degrees-of-freedom are involved, the modal responses are combined by the summation of the absolute values method.

b. In the analysis of complex systems where closely spaced modal frequencies are encountered, the responses of the closely spaced modes are combined by the summation of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the square root of the sum of the squares methods. Modal frequencies are considered closely spaced when their difference is less than + 10 percent of the lower frequency.

The maximum reactions for the pressurizer, spray line, and surge line resulting from the response spectrum analysis are found using the ICES/STRUDL II computer code. The surge line analysis includes considerations of the relative end displacements. The reactions found by statically imposing the maximum relative displacements of two ends of the surge line are conservatively included by absolute summation with the inertial response from the spectrum

3.7.3.1.2.4 Results

The demonstration of design adequacy is made by a comparison of the loads specified in the component equipment specifications with those determined through dynamic system analysis. The margins between the specified loads and the loads determined by dynamic analysis demonstrate margin between the stresses that would result from seismic loading and the stresses that have been proven to be acceptable through the design stress reports.

The maximum reactions (force and moments) at all design points for each separate direction of seismic excitation are combined by the square-root-of-the-sum-of-the-squares (SRSS) method. These combined reactions are compared with the seismic loads in each component design specification. The results of this comparison are summarized in Table 3.7-40 for the points of maximum calculated load.

The maximum seismic loads calculated by the time history techniques are the results of a search and comparison over the entire time domain of each individual component of load due to the separate application of each horizontal and vertical excitation. The maximum calculated components of load shown in Table 3.7-40 for each design location, in general do not occur at the same time for the different horizontal and vertical excitations, and therefore result in a conservative analysis.

The maximum seismic loads calculated by the response spectrum techniques are the result of SRSS of two horizontals and the vertical excitation on an absolute sum basis.

The results shown are for both the operating basis earthquake, (OBE) and safe shutdown earthquake (SSE).

It is concluded that the seismic loadings specified for the design of the Reactor Coolant System components and supports are adequate for the OBE and the SSE conditions. All seismic loads calculated by the dynamic seismic analysis are less than the corresponding loads in the component design specification.

3.7.3.2 Determination of Number of Earthquake Cycles

The procedure used to account for the number of earthquake cycles during one seismic event includes consideration of the number of significant motion peaks expected to occur during the event. The number of significant motion peaks during one seismic event would be expected to be equivalent in severity to no more than 40 full load cycles about a mean value of zero and with an amplitude equal to the maximum response produced during the entire event. Based upon this consideration and the assumption that seismic events equivalent to five operating basis earthquakes occurs during the life of the plant, seismic Category I systems, components and equipment are designed for a total of 200 full load OBE cycles. In addition, one safe shutdown earthquake is assumed.

The procedure used to account for the fatigue effect of cyclic motion associated with the OBE recognizes that the actual motion experienced during a seismic event consists of a single maximum or peak motion, and some number of cycles of lesser magnitude. The total or cumulative fatigue effect of all cycles of different magnitude results in an equivalent cumulative usage factor. The equivalent cumulative usage factor can also be specified in terms of a finite number of cycles of the maximum or peak motion. Based on this consideration, seismic Category I systems, components and equipment are designed for a total of 200 full load cycles about a mean value of zero and with an amplitude equal to the maximum response produced during the entire five OBE events.

For seismic analysis of seismic Category I equipment and piping systems, building floor response spectra are developed using synthetic earthquake time history records as described in Subsection 3.7.1. The time history records are developed using a 20 second duration of earthquake motion. The floor response spectra are shown on Figures 3.7-52 through 3.7-253. Generally the floor response spectra exhibit two major peaks corresponding to the first and second modes of vibration. To preclude a resonant condition at these peak accelerations, the design period of vibration of piping has been limited to no more than 70 percent of the second mode period of vibration which is the lowest peak shown on the spectra.

3.7.3.3 Procedure Used for Modeling

The mathematical models used for the seismic Category I piping subsystems include sufficient mass points and corresponding degrees of freedom to provide a three dimensional representation of the dynamic characteristics of the subsystem. The distribution of mass and the selected location of mass points account for torsional effects of valves and other eccentric masses.

For piping systems, adequate mass points and corresponding dynamic degrees of freedom are selected and distributed to provide for appropriate representation of the dynamic characteristics of the subsystem. The maximum spacing between mass points are limited so as to provide fair mode shape for all the significant modes. As indicated in Subsection 3.7.3.1.1.a, the maximum spacing of the mass points does not exceed one half the distance for which the frequency of a simply supported beam would be 20 Hz. Each mass point except for points indicated as restrained in a given direction, have three linear degrees of freedom. Therefore, the degrees of freedom exceed twice the number of modes with frequencies less than 33 Hz.

The criterion for sufficiency in number of dynamic modes is that the inclusion of additional modes does not result in more than 10 percent increase in response. In general, this can be satisfied by including all the dynamic modes below 33 Hz. If the highest mode calculated below 33 Hz has already fallen into the flat rigid response region of the corresponding response spectra, the effect of the remaining high modes are taken care of by adding the dynamic analysis result with an equivalent static solution in SRSS summation.

The analysis used to establish loadings for seismic design of components and equipment depends upon the complexity of the structural model required to define the dynamic response. In each case, the structural (or mathematical) model used provides sufficient detail to reflect the contribution of all significant dynamic modes of response under seismic excitation.

The dynamic models of the cable tray and HVAC duct with their respective support structures were constructed with an adequate number of mass points in order to simulate the dynamic behavior of the subsystem. The number of mass points in the dynamic model is adequate because the number of degrees of freedom exceeds twice the number of modes with frequencies less than 33 Hz.

Dynamic analysis of cable tray/HVAC duct-support systems has combined all modes in the flexible region together with residual terms accounting for higher modes in the rigid region.

NSSS vendor supplied subsystems are modeled with sufficient masses (dynamic degrees of freedom) such that inclusion of additional degrees of freedom results in less than 10 percent increase in responses (Analysis of the pressurizer, surge line, and spray line also meets the alternate criteria that the number of degrees of freedom is greater than twice the number of modes with frequencies less than 33 Hz).

Criterion used to assure that sufficient modes are included in the analysis of NSSS vendor supplied subsystems is that the inclusion of additional modes results in less than a 10 percent increase in response. Analysis of the coupled components of the RCS included all modes less than 50 Hz (Subsection 3.7.3.1.2.3(b).

Modeling of reactor internals, fuel assemblies and control element drive mechanisms is described in Subsection 3.7.3.14.

3.7.3.4 Basis for Selection of Forcing Frequencies

For piping systems which are analyzed by either the modal response spectra method or modified equivalent static load method, the fundamental frequencies are determined by the stiffness matrix method of natural mode analysis, as described in Subsection 3.7.3.1.1.a For piping systems which are analyzed by simplified seismic analysis method, the exact values of fundamental frequencies are not calculated. As described in Subsection 3.7.3.1.1c, the piping is restrained to have fundamental mode periods less than 70 percent of the first mode period of the supporting structure. This was accomplished by comparing and modifying the restraint spacing in design with that of a simply supported beam.

Where feasible, the piping systems are arranged to be in the rigid region (i.e., the fundamental frequencies are more than twice the dominant frequencies of the support structure). If the fundamental frequency of the piping system is less than twice but more than 1.43 of the dominant frequencies of the support structure, the modified Equivalent Static Load Method as delineated in Subsection 3.7.3.1.1b is used. The Modal Response Spectra Method is normally used for piping systems in the flexible or resonant region.

The subsystems in general, are designed or restrained to be in the rigid region to avoid resonance with the supporting system.

The basis for acceptability of the seismic design of equipment and subsystems for the Reactor Coolant System is that the stresses and deformations produced by vibratory motion of the postulated seismic events, in combination with other coincident loadings, be within the established limits.

Within practical limitations, the seismic design of the Reactor Coolant System is accomplished in a manner to maintain the resonant frequencies well above the range which is significantly excited by the forcing frequencies. Based upon the results of analysis of the preliminary design, the stiffness of the restraint and support system is modified as required to maintain the fundamental frequencies of equipment and subsystems sufficiently removed from the resonant range and, thereby, maintain the seismic response within the established limits.

Frequencies for the Reactor Coolant System and reactor internals are calculated in accordance with the procedures described in Subsections 3.7.3.1.2.3 (b) and 3.7.3.14, respectively. The three ranges of equipment/ support behavior (rigid, flexible, resonant) are not delineated for NSSS vendor supplied subsystems. (Subsection 3.7.3.1.2 describes procedures for NSSS vendor supplied systems).

3.7.3.5 Use of Equivalent Static Load Method of Analysis

The piping analysis utilizes the modified equivalent static load method which, as described in Subsection 3.7.3.1.1.b, is a frequency based static analysis. It is applicable when the piping system is proved to be in the relatively rigid side of the dominant frequency of the supporting structure. At first, the fundamental frequency of the piping system is determined by the same stiffness matrix method of natural mode analysis described in Subsection 3.7.3.1.1.a.2, then a static analysis is performed using an acceleration value of 1.5 times the maximum value of the applicable floor response spectrum in the period range equal to or less than the first mode period of the piping system.

The equivalent static load method, as interpreted from SRP 3.7.3 Section II, b does not require demonstration of the fundamental frequency of the piping system, equipment etc., a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum to obtain the equivalent static load.

The piping system stress analysis does not utilize the equivalent static load method.

As indicated in Subsection 3.7.3.1.1, the seismic analysis of non-NSSS piping is done by using one of the three following methods:

a. Modal Response Spectra Method - This method is based on the classical modal analysis which involves the calculation of all the significant natural frequencies and their mode shape vectors and the response combination of these modes of vibration.

b. Modified Equivalent Static Load Method (Simplified dynamic analysis). This method involves the calculation of the first mode period of the piping system to determine the applicable value of accelerations which in turn is used in the equivalent static analysis.

c. Simplified Static Method (chart method) - This method involves the development of reference restraint spacing based on preset values of fundamental piping period to preclude the possible resonance with the support structure. A location of restraint on the piping system is determined by comparing the individual selected restraint spacing with the reference restraint spacing.

Cable tray and HVAC duct seismic supports were designed in the following manner:

A seismic response analysis was performed on a 3-D model, which represented a typical multiple span of cable tray (HVAC duct) and its supports. Each tray (duct) support in the model was assigned a fundamental frequency of 16 to 18 Hz in three directions. The results of this analysis was an amplification factor which was then used to determine the static equivalent "g" values for design of individual cable tray (HVAC duct) supports. Each support is designed to have a fundamental frequency of 16 to 18 Hz.

Regarding considerations of relative motion between supports see Subsection 3.7.3.9.

For NSSS vendor supplied subsystems the equivalent static load method is limited to analysis of components which can be realistically represented as single-degree-of-freedom systems or by simple beam or frame type models. For multiply supported components, the relative motion between supports is applied in the most unfavorable manner using static analysis procedures and responses are added to those due to inertial effects by the absolute sum method.

3.7.3.6 Three Components of Earthquake Motion

For the seismic design of seismic Category I piping and equipment, the horizontal and vertical response loadings are obtained from the floor response spectra that have been generated for the appropriate structures and elevations. The combination of response loadings assumes the occurrence of an earthquake in the vertical and two mutually perpendicular horizontal directions at the same time. The loads corresponding to the three components of the ground motion are computed separately and the maximum co-directional responses are added by the square root of the sum of the squares (SRSS) method, as per the recommendation of Regulatory Guide 1.92 (R1), for obtaining combined response effects.

Combination of the three components of earthquake motion for the reactor internals, fuel assemblies and CEDMs are described in Subsection 3.7.3.14.1.3.

3.7.3.7 Combination of Modal Responses

3.7.3.7.1 A/E Supplied Subsystems

The procedure for combining modal responses for seismic Category I subsystems other than NSSS supplied components is discussed in Subsection 3.7.3.1.1. In the analysis of complex systems where closely spaced modal frequencies are encountered, the responses of the closely spaced modes are combined by the summation of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the SRSS method. Modal frequencies are considered closely spaced when their difference is less than 10 percent of the lower frequency. This procedure is in conformance with Regulatory Guide 1.92 (R1) recommendations.

3.7.3.7.2 NSSS Supplied Subsystems

The SRSS method is the procedure normally used for the NSSS to combine the modal responses, when the modal analysis response spectrum method of analysis is employed. The procedure is modified only in two cases:

a. In the analysis of simple systems, where three or less dynamic degrees-of-freedom are involved, the modal responses are combined by the summation of the absolute values method.

b. In the analysis of complex systems, where closely spaced modal frequencies are encountered, the responses of the closely spaced modes are combined by the summation of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the SRSS method. Modal frequencies are considered closely spaced when their differences are less than 10 percent of the lower frequency.

   The consideration of closely spaced modes for subsystems are stated in Subsections 3.7.3.7.1, 3.7.3.1.1.a)4), and 3.7.3.1.2.3.d)b) following Regulatory Guide 1.92 (R0) for combining modes that are closely spaced.

3.7.3.8 Analytical Procedures for Piping

The analytical procedures applicable to seismic analysis of piping are described in Subsections 3.7.3.1.1 and 3.7.3.5. The method used to consider differential piping support movements at different support points located within a structure and between structures are presented in Subsection 3.7.3.9.

3.7.3.9 Multiple Supported Equipment Components with Distinct Inputs

Piping systems that pass between structures are analyzed by the piping analysis program PIPESTRESS 2010 (See Subsection 3.9.1.2.1.1 for a description of the program) for stresses created by the relative displacement of anchor points that are located on different structures.

Where the location of the subsystem is such that more than one floor response spectrum is applicable, the analysis considers the following:

a. Within the same structure, that spectrum which produces the highest stress is applied to the entire piping or the upper-bound of enveloped response spectra method is used. For piping that is supported by two buildings on a common mat or two structures within the same building, the relative seismic displacements between interface support/restraints are derived by taking the square root of the sum of the squares (SRSS) of each relative seismic displacement towards the common reference. The common reference can either be the common mat or the structure base which is considered as anchorage in the analytical model of the structure.

b. When the support points are on different structures, the upper-bound enveloped response spectrum is used.

   For piping that is supported by two buildings on separate mats, the relative seismic displacements between interface support/restraints, in general, are derived from the combination of co-directional maximum absolute seismic displacement of the two buildings at the supporting elevation by square root of the sum of the square (SRSS) method. If the interface support/restraints at both buildings are located near the ground level or the two adjacent buildings have similar base response, the larger of the two maximum absolute seismic displacements may be considered as the maximum relative displacement between the interface support/restraints.

c. Significant support displacements are considered in the most unfavorable combination using static analysis procedure.

   For NSSS vendor supplied multiple supported components analyzed by the response spectrum method, the support displacements are imposed on the supported item in the most unfavorable combination using static analysis procedures. The responses due to the inertial effect and relative displacement are combined by the absolute sum method. (See Subsection 3.7.3.1.2.3(d) for surge (and spray) line analysis.) The analysis of the multiple supported coupled components of the RCS are analyzed using time history procedures with the relative support displacements applied directly as described in Subsection 3.7.3.1.2.3.

   All the maximum relative seismic displacements are placed at the interface anchorage such as the penetration connections of the steel containment in the seismic displacement analysis. The seismic displacement analyses are at first performed for each of the three orthogonal directions independently. Then the result data are combined by SRSS method.

All the relative displacements between restraint/support points which may contribute an estimated bending stress more than five percent of the code allowable stress limit are considered to be significant.

d. The responses due to the inertia effect and relative displacement are then combined by the absolute sum method.

e. Analytical methods for NSSS supplied multiple supported subsystems with distinct inputs are addressed in Subsections 3.7.3.1.2.3(c) and (d).

The following is a summary of the method used to handle the relative seismic displacement of support in piping systems.

a. The relative seismic displacements between supports/restraints installed on the same building structure are normally negligible in the stress analysis.

b. The relative seismic displacements between supports/restraints located in two buildings on separate mats are to be derived from the combination of co-directional maximum absolute seismic displacement of the two buildings at the supporting elevation by SRSS (square root of the sum of the squares) method.

   Since the adjacent buildings on separate mats are dynamically analyzed as decoupled systems, the real relative seismic displacement between piping restraints on separate buildings are not readily available. As the soil interactions are incorporated in the dynamic analysis of the buildings, the maximum absolute seismic displacements of each building available for piping stress analysis usage normally contain considerable soil movement which is expected to be closely in-phase for locations between the adjacent buildings.

   This is evidently demonstrated in the Maximum Absolute Vertical SSE Seismic Displacement Tables for Reactor Containment Building (RCB) and Reactor Auxiliary Building (RAB) (Table 3.7-52). The values for RCB mat and the top of the steel containment are 0.0304 feet and 0.03067 feet respectively while the value of RAB mat and top mass of RAB are 0.02108 feet and 0.02145 respectively. The displacements for masses in between are approximately in the same magnitude.

   A relative seismic displacement between RCB and RAB mats calculated by Absolute Sum Combination of the two maximum absolute displacements (ABS) results in a value of 0.05148 feet. This essentially represents an assumption of complete out-phase of the adjacent mats each at its peak movement at the same instant. In consideration of similarity in surrounding soil properties, closeness of spacing and the seismic wave motion in ground, this is very unlikely to happen. Furthermore, the fundamental frequencies of the adjacent building which normally dominate the displacement response are also not the same.

   The Application of Absolute Sum Combination of the two maximum absolute building seismic displacements onto the interface piping results in a piping system design overly emphasized in relative displacement condition. This, by the nature of the flexibility requirement, tends to undercut the conservatism usually existing in normal seismic protection design on the basis of seismic response of inertia effect which, being accounted for in the primary pipe stresses, is more probable to cause structural failure.

It is recognized that the pattern of the relative horizontal movement between buildings at high elevation is usually difficult to predict. A less conservative approach is to consider the complete out- phase of the two building structural responses along with the close in-phase of the two adjacent mats. Therefore, the relative seismic displacements between support/restraints located on two buildings on separate mats can be derived by taking the absolute sum of each relative building seismic displacement towards each mat and the difference of the mats displacements. This is assigned as ΔABSR, in Table 3.7-52 for comparison of combination methods for evaluating relative seismic displacements.

As all the piping penetrations are located at the lower part of the building, the strong contribution from the soil movement to the relative displacements between the pipe restraints is anticipated. The Relative Seismic Displacement derived by SRSS method (ΔSRSS) as shown in the comparison table, is considerably higher than ΔABSR.

c. The relative seismic displacements between supports/restraints located in two buildings on a common mat attached to two structures within the same building are derived by taking the square root of the sum of the squares of each relative seismic displacement towards a common reference.

d. For piping connecting to equipment or primary piping systems of which the available maximum seismic displacements are relative to the base support of the major equipment, the base supports are selected as the common reference. The maximum seismic displacement of the subject piping restraint system are to be converted into relative displacements towards the base support of connecting nozzle and the piping restraint system are then to be derived by taking absolute addition of the two relative seismic displacements which in turn are relative to the base support of the equipment.

e. The piping system is analyzed separately with relative seismic displacement input in each of the three orthoganal coordinate directions, The resultant response (such as pipe stress, moment, force, etc.) are obtained by taking SRSS of the response corresponding to each coordinate direction.

Relative displacement among supports located at different floor elevations are not considered in cable tray and HVAC duct seismic analysis. Ducts are provided with flexible joints to accommodate relative displacement of the supports.

For the coupled components of the RCS the relative support displacements are applied directly in the time history analysis methods described in Subsection 3.7.3.1.2.3. For NSSS vendor supplied multiply supported subsystems analyzed by response spectrum methods, relative support displacements are applied statically in the most unfavorable manner.

As indicated in above paragraph (b), the design values for displacement of structures not on a common mat were computed by a SRSS summation of maximum response spectra displacements. The maximum response spectra values are based on enveloping maximum response spectra displacements of each building considering a range of various soil properties. Various scoping studies have been performed to determine appropriate values for maximum expected displacements at St. Lucie Unit 2. These values are compared to those actually used in the pipe stress analysis calculations.

a. The most realistic indication of the true differential seismic displacements can be seen by comparing the time history values between two points on different structures and calculating the maximum difference. Inherent in this comparison are two assumptions for adjacent structures:

    1. Motion of the two structures begins in-phase.

    2. Differential ground motion between the two structures are negligible.

    Table 3.7-53 indicates these maximum difference values for two cases: for piping system going from elevation 28'-0" in the reactor building (RB) to elevation 18'-6" in the reactor auxiliary building (RAB) and for a system from elevation 48'-0" in the RB to 42'-6" in the RAB. These two cases are typical for piping between these two buildings. Note that the time history values are, in all cases, substantially less than the St. Lucie Unit 2 design values.

b. In order to account for any additional displacements resulting from inaccuracy due to the two assumptions stated above, the time history values have also been combined by maximum summations of the values as a function of time. As would be expected, these absolute sum displacements are greater than those calculated previously. However, as shown in Table 3.7-53, these values are in all cases less than the St. Lucie Unit 2 design values.

c. An additional parameter that could be considered is variation in time interval. The time history earthquakes were generated using a time interval of .005 seconds. This could be "spread" by 20 percent analagous to the peak spreading employed for response spectra. Thus time histories with intervals from .004 to .006 seconds would be considered. This spreading would account for uncertainties in design assumptions.

    Table 3.7-54 indicates that even assuming 20 percent time interval spread, the maximum differential displacements are in all cases less than the St. Lucie Unit 2 design values.

d. Finally, the displacement values from the "spread" time history have been combined by the maximum sum method. As shown in Table 3.7-54 the design values are in all cases within 15 percent of the values determined by this technique. The maximum difference is less than 1/8".

Seismic displacement is a secondary stress when applied to piping systems and is combined with other stresses to determine the total effect. For supports and penetrations, seismic displacement is a primary load. However, this is combined with several other primary loads. Variations in the magnitude discussed above would have negligible impact on the total system and would therefore not require any further evaluation.

It should also be noted that the deviation in displacement values exists only for the .006 second time interval. The .006 second time interval reflects a 20 percent increase above the base case of .005 seconds to correspond to the spectra broadening that was used on St. Lucie Unit 2. However, this interval could be reduced to 15 percent and be in compliance with the Standard Review Plan 3.7.2 (11/24/75). The effect would be to reduce or eliminate the deviation between the time history value and the design value.

Based on the comparisons presented in Table 3.7-53 and 3.7-54 and the discussions above, the specific seismic displacement design values developed for St. Lucie Unit 2 are acceptable. This conclusion would apply to all the piping systems which penetrate the Containment and Shield Building listed in UFSAR Table 6.2-52, except as indicated in Table 3.7-55.

3.7.3.10 Use of Constant Vertical Static Load Factors

Vertical floor response spectra instead of constant vertical static factors are used for piping and equipment (refer to Subsection 3.7.3.1.1 item b).

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of motor and air operated valves and other eccentric masses are included in the analysis of Quality Group A systems by taking into account the mass and eccentricity in the mathematical model (refer to Subsection 3.7.3.3).

The mathematical models used in seismic analysis of Reactor Coolant System other than the main loop include sufficient mass points and corresponding dynamic degrees-of-freedom to provide a three dimensional representation of the dynamic characteristics of the system. The distribution of mass and the selected location of mass points accounts for torsional effects of valves and other eccentric masses.

A study performed for the H B Robinson Nuclear Unit No. 2 (Docket No. 50-261) demonstrated that the additional pipe stresses due to the offset weights of valve operators are insignificant except for the case where large operators are installed on two inch and smaller pipes. Therefore, torsional effects of large operators on two inch and smaller pipes are included in the analysis of seismic Category I, Quality Group B and C piping systems, while the weight of valves and operators are lumped at the center line of the pipe in the mathematical models for 2-1/2 inch and larger piping.

If analysis indicates stresses close to allowable code values in the vicinity of the valve, the stress analysis is performed considering the eccentricity of the masses.

Torsional effects of damper operators are considered in the design of HVAC ducts and their seismic restraints.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

The seismic Category I piping between a structure and the ground is provided with sufficient flexibility to absorb the differential motion between the building and ground, or the piping is run underground for some distance in a culvert that allows sufficient displacement of the piping without creating an overstressed condition.

The seismic Category I buried piping between mats and floor slabs are encased in either a sand, cement mixture or a concrete fill. The seismic Category I buried piping outside the buildings are embedded in Class-1 compact backfill soil of sufficient density so that liquefaction shall not take place and backfill will not lose its integrity during a SSE. The effect of the loads imposed by the adjacent non-buried portion piping on the buried piping are considered for a distance into the soil based on Heteny's approach[20] to a semi- infinite beam on elastic foundation. The loads are considered to have died out in the form of a damped wave along the buried pipe further into the soil.

For the portion of buried piping in which the pipe is not fully restrained by the soil and can be influenced by the non-buried piping, the soil resistance is simulated by a system of equivalent lateral restraints. The stiffness of soil restraints and spacing of these restraints are calculated by considering the buried pipe as semi-infinite beam on elastic foundation. The friction and slippage effect which permit relative movement between the soil and the buried pipe are considered in evaluating the pipe stress due to seismic wave motion[17,19,21]. The portion of buried piping is coupled with the nonburied piping and analyzed as a whole by the usual method delineated in Subsection 3.7.3.1.1. The stresses in buried portion of piping due to various load including the loading due to relative movement between the soil and buildings are included in the analysis and the stress limits for various load combinations given in Subsection NC-3650 or ND-3650 of ASME Section III are satisfied.

For the portion of buried piping in which the pipe is fully restrained by the surrounding soil and can not be influenced by the non-buried piping, the "pipe moves with the ground" assumption is used in evaluating the pipe stresses. The maximum axial, bending and shear stresses due to compressional wave and shear wave are calculated separately using the stresses formulas presented by Nemark[22] and Yeh[25]. Since the maximum stresses due to various seismic waves do not occur simultaneously, the maximum combined stresses are calculated by SRSS method using principal stress formula. Finally the combined stresses due to seismic wave motion are combined with the longitudinal pressure stress and compared with the yield stress of the pipe for normal, upset and emergency conditions. For faulted condition, two times the yield stresses of the pipe is considered as the allowable. For buried pipe near the buried elbows, the friction and slippage effect are considered following the technique presented by Goodling[17,18]

In addition to the seismic consideration, the buried piping is also evaluated for plant settlement and maximum thermal expansion stresses with proper consideration of slippage and friction effect[17,19].

The underground duct banks containing Class 1E electrical cables are seismically analyzed.

As part of the St. Lucie Unit 2 Component Replacement Projects, Engineering Evaluation No. PSL-ENG-SECS-07-014 was performed to demonstrate the acceptability of using Controlled Low-Strength Material (CLSM) as Class I or lesser classification backfill material in restoring excavated areas. Use of CLSM is limited by PSL-ENG-SECS-07-014 to areas that do not serve as foundation support for any Seismic Category I structures for any Seismic Category I buried piping.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

In general, all non-seismic Category I piping systems are designed to be isolated from any seismic Category I piping system.

Where seismic Category I piping systems are in close proximity to non-seismic systems, the excessive movement of the non-seismic Category I system due to seismically induced effects is restrained so that no failure of the seismic Category I system occurs.

If not isolated by a barrier, the adjacent non-seismic Category I piping is analyzed and supported according to the same seismic criteria as applicable to the seismic Category I piping system.

Where seismic Category I piping is directly connected to non-seismic piping, the seismic effects of the non- seismic piping are prevented from being transferred to the seismic Category I piping by placing anchors or combinations of restraints beyond the interface. The portion up to the anchor is included in the dynamic modeling of the seismic Category I piping. The attached non-seismic Category I piping, up to the first anchor beyond the interface, is also designed in such a manner that during an earthquake of SSE intensity it does not cause a failure of the seismic Category I piping.

Non-seismic lines are seismically supported where they pass over seismic Category I piping, valves and valve operators.

3.7.3.14 Seismic Analysis of Reactor Internals, Core, and Control Element Drive Mechanisms (CEDM)

Dynamic analyses of the reactor internals, core, and CEDMs are conducted to determine their response to horizontal and vertical seismic excitation, and to verify the adequacy of their seismic design. The seismic analyses of the internals and core utilize response spectra modal analysis methods for the axial direction and step-by-step integration of the equations of motion in the horizontal direction for nonlinear impact conditions, such as exist when the gaps between components close. These analyses are conducted in conjunction with the analyses of the Reactor Coolant System (RCS). The applicable seismic load, stress and deformation criteria are presented in Subsections 3.9.4, 3.9.5, and Section 4.2. In the analysis of reactor internals and the core, closely spaced modes are considered in accordance with Regulatory Guide 1.92 (R0).

3.7.3.14.1 Reactor Internals and Core

The first step in the seismic analysis of the reactor internals and core involves developing the input excitation which consists of the horizontal, vertical and rotational (rocking) time history responses of the reactor vessel. This is obtained from the dynamic analysis of the Reactor Coolant System model which includes the reactor, steam generator, reactor coolant pumps, interconnecting piping and a simplified representative of the reactor internals and core. Details of the Reactor Coolant System seismic analyses are described in Subsection 3.7.3.1. The excitation is then used as input to separate horizontal and vertical models of the internals and core. Separate horizontal and vertical models are formulated to more efficiently account for structural and response differences in these directions.

The horizontal seismic analysis is performed in two stages using time history methods in conjunction with nonlinear lumped-parameter models. In the first state, a coupled internals and core model is analyzed to obtain the internals response and the proper dynamic input for the reactor core model. In the second stage, a more detailed model of the core consisting of an entire row of fuel assemblies is analyzed using the upper and lower core plate and core shroud responses from the first part as excitation.

In the vertical direction, the reactor internals and core are relatively stiff with natural frequencies above the amplified response frequencies. Consequently, the response to vertical seismic acceleration is relatively low level and linear in nature. Therefore, the vertical response of the internals and core is adequately characterized by the modal analysis response spectrum method using the vessel flange vertical spectrum and a linear lumped-parameter model.

The maximum responses are calculated independently for each of the three orthogonal spatial components (two horizontal and one vertical) of the earthquake. The maximum values of the combined response are then determined by taking the square root of the sum of the squares of the maximum values of the response in each direction.

The general methodology used for seismic analysis is presented in Reference 11. The mathematical models and analytical techniques are briefly described below.

3.7.3.14.1.1 Mathematical Models

Equivalent multimass mathematical models are developed to represent the reactor internals and core. The mathematical models of the internals are constructed in terms of lumped masses and elastic-beam elements. At appropriate locations within the internals and core, points (nodes) are chosen to lump the weights of the structure. The criterion for choosing the number and location of mass concentration is to provide for accurate representation of the dynamically significant modes of vibration of each of the internals components. Between the nodes, properties are calculated for moments of inertia, cross-section areas, effective shear areas, and lengths.

The vertical internals and core model is shown on Figure 3.7-262 (node locations are described in Table 3.7-41). A linear analysis has been completed. Because of the low level of excitation, the fuel does not lift off the core support plate. Therefore, a nonlinear analysis is not required.

The model used in the first stage of the horizontal seismic analysis is shown on Figure 3.7-263 (node locations are described in Table 3.7-42). In order to represent the effect of fuel impacting on internals, impacting between the core shroud and peripheral fuel bundles is modeled. The impact stiffness and impair damping (coefficient of restitution) parameters for the gap elements are derived from the impact tests which are described in Subsection 4.2.3. The spacer grid impact representation used for the analysis is capable of representing two types of fuel assembly impact situations. In the first type, only one side of the spacer grid is loaded. This type of impact occurs when the peripheral fuel assembly hits the core shroud or when two fuel assemblies strike one another. The second type of impact loading occurs typically when the fuel assemblies pile up on one side of the core. In this case, the spacer grids are subjected to a through grid compressive loading. Lumped-mass nodes are positioned to coincide with fuel-spacer grid locations. The core is modeled by subdividing it into fuel assembly groupings and choosing stiffness values to adequately characterize its beam response and contacting under dynamic loading.

The horizontal nonlinear reactor core model consisting of one row of 17 individual fuel assemblies is depicted on Figure 3.7-264 (fuel assembly nodes coincide with spacer grid locations). The distribution of mass and stiffness values is based upon experimentally determined fuel assembly vibration characteristics. The 17 fuel assemblies are modeled to properly account for multiple contacting of fuel assemblies. Nonlinear springs are also incorporated between the fuel and core shroud to account for contact with the core shroud.

The damping factors used in the seismic analysis of the reactor internals are in accordance with the values in Table 3.7-2 and are four percent of critical for the DBE and 2 percent of critical for the OBE. Damping values used for fuel assemblies are based on the results of the full-scale structural tests defined in Subsection 4.2.3.

Additional salient details of the internals and core models are discussed in the following paragraphs.

a. Hydrodynamic Effects

   The dynamic analysis of reactor internals presents some special problems due to their immersion in a confined fluid. It has been shown both analytically and experimentally[12] that immersion of a body in a dense fluid medium lowers its natural frequency and significantly alters its vibratory response as compared to that in air. The effect is more pronounced where the confining boundaries of the fluid are in close proximity to the vibrating body as is the case for the reactor internals. The method of accounting for the effects of a surrounding fluid on a vibrating system has been to ascribe to the system additional or "hydrodynamic mass".

   The hydrodynamic mass of an immersed system is a function of the dimensions of the real mass and the space between the real mass and confining boundary. Hydrodynamic mass effects for moving cylinders in a water annulus are discussed in References 12 and 13. The results of these references are applied to the internals structures in the lateral model to obtain the total (structural plus hydrodynamic) mass matrix including hydrodynamic coupling (off-diagonal terms of the mass matrix).

b. Core-Support Barrel

   The core-support barrel is modeled as a beam with shear deformation in the horizontal analysis. It has been shown that the use of the beam theory for cylindrical shells gives sufficiently accurate results when shear deformation is included[14 and 15].

c. Fuel Assemblies

   The fuel assemblies in the horizontal analyses are modeled as uniform beams with rotational springs at each end to represent the proper end conditions. The member properties for the beam elements representing the fuel assemblies are derived from the results of experimental tests of the fuel assembly load deflection characteristics and fundamental natural frequency.

d. Support-Barrel Flanges

   To obtain accurate lateral and vertical stiffnesses of the upper and lower core-support-barrel flanges and the upper guide structure support-barrel upper flange, finite-element analyses of these regions are performed. As shown on Figure 3.7-265 these areas are modeled with quadrilateral and triangular ring elements. Displacements and rotations are applied to the ends of the flange models to obtain member stiffness properties for use in the models.

e. Control Element Assembly (CEA) Shrouds

For the horizontal model, the CEA shrouds are treated as vibrating in unison and are modeled as guided cantilever beams in parallel. In addition, the restraint to relative rotation between the upper guide structure support plate and the fuel alignment plate due to vertical shroud stiffness is modeled. To account for the decreased lateral stiffness of the upper guide structure due to local bending of the fuel alignment plate, a short member with properties approximating the local bending stiffness of the fuel alignment plate is included at the bottom of the CEA shrouds. Since the stiffness of the upper guide structure support plate is large compared to that of the shrouds, the CEA shrouds are assumed to be rigidly connected to the upper guide structure support plate.

f. Upper Guide Structure Support Plate and Lower Support Structure Grid Beams

These grid beam structures are modeled as plane grids. Displacements due to vertical (out-of-plane) loads applied at the beam junctions are calculated through the use of the STRUDL computer program[8]. Stiffness values based on these results yield an equivalent member cross-section area for the vertical model.

3.7.3.14.1.2 Analytical Techniques

a. Vertical Linear Analysis

1. Natural Frequencies and Mode Shapes

The mass- and beam-element properties of the vertical model are utilized in the MODSK computer program (See Subsection 3.9.1.2.2.2.8) to obtain the natural frequencies and mode shapes. The system utilizes the stiffness-matrix method of structural analysis. The natural frequencies and mode shapes are extracted from the system of equations:

$$(\underline{K} - \omega_n^2 \underline{M})\, \phi_n = 0 \qquad (12)$$

where:

K = model stiffness matrix

ω_n = natural circular frequency for the n mode

M = model mass matrix

$ø_n$ = normal mode shape matrix for the nth mode

2. Response Calculation Method

A response spectrum analysis is performed using the modal extraction data and the following relationships for each mode:

(a) Nodal Accelerations

$$\ddot{X}_{in} = \gamma_n A_n \varnothing_{in} \quad (13)$$

where: Install

$\ddot{X}_{in}$ = absolute acceleration at node "i" for mode "n"

γ_n = modal participation factor

A_n = modal acceleration from response spectrum

$\varnothing_{in}$ = mode shape factor at node "i" for mode "n"

(b) Nodal Displacement

$$Y_{in} = \frac{\ddot{X}_{in}}{\omega_n^2} \quad (14)$$

where:

Y_{in} = displacement at node "i" for mode "n" relative to base

ω_n = natural circular frequency for n^{th} mode

(c) Member Forces and Moments

$$F_n = \frac{(\gamma_n A_n)}{\omega_n^2} \overline{F_n} \quad (15)$$

where:

F_n = actual member force for mode "n"

$\overline{F}_n$ = modal member force for mode "n"

Modal responses are combined by using both the square root of the sum of the squares method and the absolute value summation method. While the results are only slightly different, the more conservative absolute value summation results are used.

b. Horizontal Nonlinear Analysis

The nonlinear seismic response and impact forces for the internals are determined using the CESHOCK computer program(16). This computer program provides the numerical solution to transient dynamic problems by step-by-step integration of the differential equations of motion. The input excitation for the model are the horizontal and rotational (rocking) time-history response of the reactor vessel.

Input to the CESHOCK computer program consists of initial conditions, nodal lumped masses, linear- spring coefficients, mass moments of inertia, nonlinear spring curves, and the acceleration time histories. The output from the CESHOCK computer program consists of displacements, velocities, accelerations, impact forces, shears, and moments.

A brief description of the general methods used in the CESHOCK computer program to solve transient dynamic problems can be found in Subsection 3.9.1.2.2.2.5.

3.7.3.14.1.3 Results

Reactor internals and fuel responses due to vertical and horizontal OBE and SSE excitations are calculated using the models and techniques described above. The responses in the three orthogonal directions are combined using the square root of the sum of the squares method. The resulting loads, stresses and deformations for the fuel are compared to the criteria presented in Section 4.2 (see Subsection 4.2.3.1 for a summary of results). The resulting seismic loads for the reactor internals are combined with other loads as described in Subsection 3.9.5 and the resulting stresses and deformations have been compared to the criteria presented in Subsection 3.9.5. Representative results are shown in Table 3.9-29.

The EPU does not impact the reactor internals seismic analysis

Seismic loads on the reactor internals have been adjusted to encompass operation with either Westinghouse fuel or AREVA fuel. EC287528

3.7.3.14.2 Control Element Drive Mechanisms (CEDM)

The pressure-retaining components, see Figure 3.7-266, of the CEDM are designed to the appropriate stress criteria of the ASME code, Section III, for all loadings stipulated in the mechanism design specification. The structural integrity of the CEDM, when subjected to seismic loadings, is verified by dynamic analysis. Methods of modal dynamic analysis employing response spectrum techniques are supported from past analyses and test data correlation.

3.7.3.14.2.1 Mathematical Model

A mathematical model, see Figure 3.7-267, consisting of lumped masses and beam members is used in the dynamic analysis of the CEDM design. The lumped-mass nodal points and member stiffness properties are defined to provide an accurate representation of the dynamically significant modes of vibration within the seismic frequency range.

Due to symmetry of the CEDM structure about its longitudinal axis, a two dimensional representation of the dynamic response suffices.

The effect of different CEDM nozzle lengths on the dynamic response characteristics of the CEDM is accounted for in the analysis.

The dynamic analysis of the CEDM is based on a critical damping ratio of two percent for both OBE and SSE events. This value was confirmed by test for all significant vibration modes of the free standing CEDM type. The model CEDM nozzle length varied from the shortest to longest actual length and results from all computer runs were enveloped.

3.7.3.14.2.2 Analytical Techniques

A dynamic analysis of the mathematical structural model is performed using the SAP IV computer program. Natural frequencies and mode shapes of the composite mathematical model are calculated first. Using the mode shapes and natural frequencies, a modal response analysis using a response spectrum technique is performed for seismic excitations and for deterministic mechanical loadings.

3.7.3.14.2.3 Results

The dynamic responses (forces and moments) at all CEDM design points were combined with operational loads and a stress analysis conforming to ASME Code Section III was performed for all pressure retaining CEDM components. The following is a summary of the most significant results of the dynamic analysis and the stress analysis.

a. The CEDM mechanism was analyzed with 16 different nozzle lengths and all results were enveloped. See Tables 3.7-43 and 44.

b. A stress analysis of the pressure retaining CEDM components demonstrated all stresses to be well within the material allowables, see Tables 3.7-46 to 3.7-48.

c. The ability of the CEDM to allow CEA scram is verified by the fact that the maximum calculated deflection of 3.37 in. at the CEDM top was less than 4.0 in. deflection for which scram was demonstrated by test. See Table 3.7-45 and Figure 3.7-268.

3.7.3.15 Analysis Procedure for Damping

Values of critical damping used in the seismic analysis are tabulated in Appendix 3.9A for the various types of structural members, for the OBE and SSE accelerations. Critical damping for soil is 10 percent for both the OBE and SSE.

3.7.4 SEISMIC INSTRUMENTATION

The St. Lucie Unit 2 structural designs are essentially the same as those of Unit 1, hence the seismic responses expected to be experienced in the St. Lucie Unit 2 plant are similar to those of Unit 1 plant. Using identical seismic inputs, a seismic response analysis of the structures comprising St. Lucie Units 1 and 2 would demonstrate identical effects for both units. As a result, St. Lucie Unit 2 takes credit for the seismic monitoring instrumentation installed in Unit 1. Annunciation is provided in both the Unit 1 and Unit 2 Control Rooms.

The original seismic monitoring system was evaluated against Regulatory Guide 1.12, Revision 01, “Instrumentation for Earthquakes.” The seismic monitoring system was upgraded in 2004, and the new instrumentation was evaluated against Regulatory Guide 1.12, Revision 02 (see Unit 1 UFSAR Section 3.7.4).

Unit 2 Technical Specification Amendment 74 was issued on April 25, 1995 which removed the seismic instrumentation requirements from the Technical Specifications and relocated them to the UFSAR. Unit 2 uses the installed equipment on Unit 1 to meet Technical Specification requirements. See Section 13.7.1.1 for operational and surveillance requirements.

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26. AREVA NP Inc., 77-5069878-004, "Replacement Steam Generator Report for Florida Power and Light St. Lucie Unit 2," December 2015. EC283094

TABLE 3.7-1

AMPLIFIED ACCELERATIONS AND DISPLACEMENTS FOR THE OPERATING BASIS EARTHQUAKE - HORIZONTAL

(See Figure 3.7-5 and Section 3.7.1.1)

	Point A at 33HZ		Point B at 9HZ		Point C at 2.5HZ		Point D at 0.25 Hz	
% Critical Damping	Amplification Factor	Accelera-tion (g)	Amplifi-cation Factor	Acceleration (g)	Amplification Factor	Accelera-tion (g)	Amplification Factor	Displace-ment (in)
1	1.0	0.05	4.25	0.213	5.1	0.255	2.85	5.13
2	1.0	0.05	3.54	0.177	4.25	0.213	2.50	4.50
4	1.0	0.05	2.84	0.142	3.41	0.171	2.16	3.89
5	1.0	0.05	2.61	0.131	3.13	0.157	2.05	3.69
7	1.0	0.05	2.27	0.114	2.72	0.136	1.88	3.38
10	1.0	0.05	1.90	0.095	2.28	0.114	1.70	3.06

TABLE 3.7-2

PERCENT CRITICAL DAMPING

		OBE (0.05g ground surface acceleration)	SSE (0.10g ground surface acceleration)
Welded steel plate assemblies		1	1
Steel containment vessel		2	4
Welded steel framed structures		2	4
Bolted or riveted steel framed structures		4	7
Reinforced concrete equipment supports		4	7
Reinforced concrete frames and buildings		4	7
Steel piping	(pipe dia. > 12 in.)	2	3 (Note 1)
	(pipe dia. $\leq$ 12 in.)	1	2
Soil		10	10

Note 1:

For piping analyses, provisions of ASME Sec. III, Code Case N-411 may be used as described in Section 3.9.3.1.1 h) (page 3.9-37)

TABLE 3.7-3

SUPPORTING MEDIA CHARACTERISTICS FOR SEISMIC CATEGORY I STRUCTURES

	Foundation Size	Structural Height	Embedment Depth
Reactor Building	160 ft. Dia	250.5 ft	44 ft
Reactor Aux Building	237 ft x 113 ft	91 ft	27.5 ft
Fuel Handling Building	130 ft x 51 ft	87.5 ft	9 ft
Diesel Generators Building	96 ft x 76.5 ft	39 ft	9 ft
Component Cooling Water Building	97 ft x 74.5 ft	39.5 ft	9 ft
Condensate Storage Tank Building	55 ft Dia	72 ft	8.5 ft
Diesel Oil Storage Tank Building	62 ft x 31 ft	52.5 ft	2.5 ft
Intake Structure	57.67 ft x 59.5 ft	70.5 ft	Varies
Main Steam Trestle	49.75 ft x 80.08 ft (avg.)	62 ft	24 ft

TABLE 3.7-4

REACTOR BUILDING PROPERTIES

HORIZONTAL MODEL

(See Figure 3.7-30 For Model)

SHIELD BUILDING				STEEL CONTAINMENT			
Mass No.	W (k)	A (Ft^2)	I (1000 Ft^4)	Mass No.	W (k)	A (Ft^2)	I (1000 Ft^4)
1	7,081	573	3,040	11	332	110	293
2	5,251	711	4,057	12	262	139	571
3	4,107	711	4,057	13	308	151	726
4	4,267	711	4,057	14	508	303	1486
5	4,267	711	4,057	15	1,707	303	1807
6	4,267	711	4,057	16	684	303	1486
7	4,267	711	4,057	17	684	303	1486
8	4,267	711	4,057	18	684	303	1486
9	4,267	711	4,057	19	684	303	1486
10	3,200	711	4,057	20	554	359	1746

INTERNAL STRUCTURE				FOUNDATION	
Mass No.	W (k)	A (Ft^2)	I (1000 Ft^4)	W(k)	Weight Moment of Inertia (Ft^2 - k)
21	1,796	593	130	115,104	1.74×10^8
22	10,439	681	2,076		
23	7,246	681	2,076		
24	4,945	968	2,096		

Note: Design concrete strength f'_c = 4000 psi
Young's modulus E_c = 552,130 ksf
Poisson's ratio μ_c = 0.18
W = weight in kips
A = effective area in Ft^2
I = area moment of inertia
Shear Modules G_c = 233,950 ksf

TABLE 3.7-5

REACTOR BUILDING PROPERTIES
VERTICAL MODEL

(See Figure 3.7-31 For Model)

SHIELD BUILDING			STEEL CONTAINMENT		
Mass No.	W (k)	A (Ft^2)	Mass No.	W (k)	A (Ft^2)
1	9824	1422	6	594	276
2	8748	1422	7	2181	605
3	8534	1422	8	1368	605
4	8534	1422	9	1368	605
5	7467	1422	10	886	605

INTERNAL STRUCTURE			FOUNDATION
Mass No.	W (k)	A (Ft^2)	W (k)
11	1796	903	117,242.0
12	9716	2341	
13	7246	2341	
14	4945	2644	
Branch Mass Pt. 15	736	-	

Stiffness of branch mass pt. $= 6.57 \times 10^5$ k/ft.

Note: Design concrete strength f' c = 4,000 psi
Young's modulus E_c = 552,130 ksf
Poisson's ratio μ_c = 0.18
Shear modulus G_c = 233,950 ksf

SOIL-STRUCTURE INTERACTION CONSTANTS

$$K_B = \frac{4GR}{1-\mu}$$

for E_s = 40,000 psi. G = 2300 ksf $\mu = 0.25$

$$K_B = \frac{4\ x G\ x 80'}{0.75} = 426.67\ G$$

$$= 426.67 \times 2300 = 9.813 \times 10^5 \text{ k/ft.}$$

E_s (Psi)	G (ksf)	K_B (k/Ft)
32000	1843	7.86×10^5
36000	2074	8.85×10^5
40000	2300	9.81×10^5
44000	2534	10.81×10^5
48000	2765	11.80×10^5

G(ksf) = .0576 E(Psi)

TABLE 3.7-6

REACTOR AUXILIARY BUILDING PROPERTIES

HORIZONTAL MODEL

(See Figure 3.7-32 For Model)

Mass No.	W (k)	A(N-S) (Ft^2)	I (N-S) (Ft^4)	A(E-W) (Ft^2)	I (E-W) (Ft^4)
1	4,074.	436.	1,380,500	292	765,400
2	11,650.	881.	3,029,400	705	7,687,000
3	21,384.	1,633.	6,072,617	1,765	16,452,164
4	28,670.	1,689.	7,511,170	2,328	26,358,786
Base	40,628.		*32,249,813		*140,247,721

* Weight moment of inertia (k-ft^2)
N-S Short Direction
E-W Long Direction

Design concrete strength f' c = 3000 psi
Young's modulus E_c = 522,130 ksf
Poisson's ratio μ = 0.18
Shear modulus G_c = 233,950 ksf

TABLE 3.7-7

REACTOR AUXILIARY BUILDING PROPERTIES

VERTICAL MODEL

(See Figure 3.7-33 For Model)

Mass No.	W (k)	A (Ft^2)	Floor Stiffness (k/ft)
1	3180.	771.	-
2	11580.	1784.	-
3	21303.	3837.	-
4	28614.	4170.	-
5	894.	-	4.7×10^4
6	70.	-	1.66×10^4
7	81.	-	2.49×10^4
8	56.	-	10.6×10^4
Base	40628.	-	

Design concrete strength f'_c = 4000 psi

Young's modulus E_c = 552,130 ksf

Poisson's ratio μ = 0.18

Shear modulus G_c = 233,950 ksf

TABLE 3.7-8

FUEL HANDLING BUILDING PROPERTIES

HORIZONTAL MODEL

(See Figure 3.7-34 For Model)

Mass No.	W(k)N-S	W(k)E-W	A(N-S) (Ft^2)	I(N-S) (Ft^4)	A(E-W (Ft^2)	I(E-W)(Ft^4)
1	3,813	3,662	406	410,632	229	101,368
2	4,714	4,715	1,130	2,151,618	968	491,800
3	10,315	9,165	1,130	3,052,684	1,025	517,806
Base	16,527	17,497	-	*35,053,836	-	*9,681,330

* Weight Moment of Inertia (k-ft^2)

N-S Long Direction of Structure

E-W Short Direction of Structure

VERTICAL MODEL

(See Figure 3.7-34 For Model)

Mass No.	W(k)	A(Ft^2)
1	3,662	586
2	4,690	1882
3	6,780	1919
Base	19,983	

Design Concrete Strength f ‘c= 4,000 psi

Young's Modulus E_c = 552,130 ksf

Poisson's Ratio μ = 0.18

Shear Modulus G_c = 233,950 ksf

TABLE 3.7-9

INTAKE STRUCTURE PROPERTIES

HORIZONTAL MODEL

(SEE FIGURE 3.7-35 & 36 FOR MODEL)

Mass No.	W(k)	A(N-S) (Ft^2)	I(N-S) (Ft^4)	A(E-W) (Ft^2)	I(E-W)(Ft^4)
1	345	173	44,422	57	6,907
2	554	283	149,650	142	29,645
3	4,093	303	601,601	810	547,400
4	4,878	144	299,570	654	273,414
5	5,182	144	299,570	654	273,414
Base	4,260		*472,434		*502,810

* Weight Moment of Inertia (k-Ft^2)
N-S Long Direction
E-W Short Direction

Design Concrete Strength f'c = 5,000 Psi
Young's Modulus E_C = 617,300 ksf
Poisson's Ratio μ = 0.2
Shear Modulus G_c = 257,210 ksf

VERTICAL MODEL

(See Figure 3.7-37 For Model)

Mass No	W(k)	A (Ft^2)	Floor Stiffness (k)
1	345	214	--
2	554	387	--
3	4,093	1,063	--
4	4,878	771	--
5	5,182	771	--
6	42	--	2,460,000
Base	4,260	--	--

TABLE 3.7-10

DIESEL GENERATOR BUILDING PROPERTIES

HORIZONTAL MODEL

(See Figure 3.7-38 For Model)

Mass No.	W(k)	A(E-W) (Ft^2)	I(E-W) (Ft^4)	A(N-S) (Ft^2)	I(N-S)(Ft^4)
1	3,575	605	656,621	326	287,720
2	2,051	800	878,134	329	338,789
3	3,760	557	924,057	519	399,782
4	1,093	735	330,888	735	6,125
5	1,082	735	330,888	735	6,125
Base	8,943		*7,065,961		*4,580,397

* Weight Moment of Inertia (k-ft^2)
N-S Long Direction
E-W Short Direction

Design Concrete Strength f'c = 4,000 Psi

Young's Modulus E_C = 552,130 ksf

Poisson's Ratio μ = 0.18

Shear Modulus G_c = 233,950 ksf

VERTICAL MODEL

(See Figure 3.7-38 For Model)

Mass No.	W(k)	A(Ft^2)	
1	3,285	871	
2	2,051	1,069	
3	2,929	1,027	
4	290	--	k = 1,825 ksf
5	768	735	
6	767	735	
Base	10,222		

TABLE 3.7-11

MAIN STEAM TRESTLE PROPERTIES

HORIZONTAL MODEL

(See Figure 3.7-39 for Model)

Mass Point	Element No.	Weight (KIPS)	ELEMENT PROPERTIES: Effective Shear Area (FT^2) N-S	Effective Shear Area (FT^2) E-W	Area Moment of Inertia (FT^4) N-S	Area Moment of Inertia (FT^4) E-W	Weight Moment of Inertia ($KIP\text{-}FT^2$) N-S	Weight Moment of Inertia ($KIP\text{-}FT^2$) E-W
1		53.2					3360.	6620
	1		1.28	0.48	15.27	24.39		
2		54.2					3420.	6750
	2		1.28	0.48	15.27	24.39		
3		463.8					29260.	57760
	3		1.39	2.75	16.40	27.84		
4		183.2					11560.	22810
	4		1.39	2.75	16.40	27.84		
5		1356.6					852860.	631440
	5		140.5	94.2	100542.	73079.		
6		4575.0					2532180.	1073490
	6		85.7	80.5	87869.	50652.		
7		239.0					87410.	11170
	7		12.4	15.4	5837.	265.		
MAT		5225.0					2996770.	1273090

MAIN STEAM TRESTLE FOUNDATION PROPERTIES

(See Figure 3.7-39 for Model)

Direction	Spring Constant: Mat			Foundation			
	Kxx (K/Ft) Bearing	Kx (K/Ft) Shear	K_θ (K-Ft/Rad) Rocking	Kxx1 (K/Ft) Bearing	Kxx2 (K/Ft) Bearing	Kxx3 (K/Ft) Bearing	Kxxl3 (K/Ft) Bearing
N-S	144,646.	398,521.	577,247,161.	-	152,688.	102,230.	83,470.
E-W	928,510.	398,521.	272,791,363.	728,880.	980,135.	-	75,944.

E Steel = 4.176×10^6 ksf
E Concrete = 5.5213×10^5 ksf
E Soil = 5760 ksf

μ Steel = 0.30
μ Concrete = 0.18
μ Soil = 0.25

f'_c Concrete = 4000 psi

E = Young's Modulus of Elasticity
μ = Poisson's Ratio
η = E Steel/E Concrete = 7.57

Note: For the concrete foundation structure, the area and area moment of inertia properties shown above are equivalent steel values, calculated by dividing concrete element properties by

TABLE 3.7-12

MAIN STEAM TRESTLE PROPERTIES

VERTICAL MODEL

(See Figure 3.7-40 For Model)

Mass Point	Element No.	Weight (Kips)	Cross-Sectional Areas (Ft^2)
1	1	53.2	3.63
2	2	54.2	4.36
3	3	463.8	7.39
4	4	183.2	7.39
5	5	1356.6	145.3
6	6	4575.0	134.8
7	7	239.0	15.8
MAT		5225.0	

Kxx Bearing = 442800. K/Ft

E Steel = 4.176×10^6 ksf — μ Steel = 0.30

E Concrete = 5.5213×10^5 ksf — μ Concrete = 0.18

Eμ Soil = 5760 ksf — μ Soil = 0.25

f' Concrete = 4000 psi

E^c =Young's Modulus of Elasticity

μ = Poisson's Ratio

η = (E Steel) / (E Concrete) = 7.57

Note: For the concrete foundation structure, the area values shown are equivalent steel values, calculated by dividing concrete area by η

TABLE 3.7-13

COMPONENT COOLING WATER BUILDING PROPERTIES

HORIZONTAL MODEL

(See Figure 3.7-44)

Cantilever	Mass Point	Weight (KIPS)	E-W E.Q A Ft^2	E-W E.Q I Ft^4	N-S E.Q A Ft^2	N-S E.Q I Ft^4
1	1	3011.65	322.	17360.9	412.	33622.2
	2	1333.65	322.	17360.9	412.	33622.2
	3	1257.6	394.5	19978.61	538.5	35989.75
2	4	352.7	57.5	2617.71	126.5	2367.55
3	5	24.13	15.0	2.8	15.	31.25
4	6	75.07	74.75	823.8	74.75	263.18
BASE		3804.4	IW (E-W.E.Q) = 1633709.0 K-Ft^2		IW (N-S E.Q) = 3387163.0 K-Ft^2	

Design concrete strength f'c = 4000 psi, Young's Modulus Ec = 552,130 ksf
Shear modulus G_c = 233, 950 ksf
Poisson's ratio μ_c = 0.18

FOUNDATION PROPERTIES

HORIZONTAL MODEL

(See Figure 3.7-44)

SPRING CONSTANT

Soil E Ksf	E-W DIRECTION Kx K/Ft Shear	E-W DIRECTION Kxx K/Ft Bearing	E-W DIRECTION Kθ K/Ft Rocking	N-S DIRECTION Kx K/Ft Shear	N-S DIRECTION Kxx K/Ft Bearing	N-S DIRECTION Kθ K/Ft Rocking
4320	397,808.	72,422.	575,676,203.	397,808.	63,469.	855,266,507.

TABLE 3.7-14

COMPONENT COOLING WATER BUILDING PROPERTIES

VERTICAL MODEL

(See Figure 3.7-45)

Cantilever	Mass Point	Weight (Kips)	A Sq. Ft
1	1	3011.65	694
	2	1333.65	694
	3	1257.6	694
2	4	352.7	184
3	5	24.13	15
4	6	75.07	74.75
BASE		3804.4	

Foundation Properties

E Soil	= 4320 Ksf
μ Soil	= 0.25
Kxx (bearing)	= 442009 k/ft

TABLE 3.7-15

CONDENSATE STORAGE TANK PROPERTIES

HORIZONTAL MODEL

(See Figure 3.7-46)

No.	Length	Area Ft^2	Shear A Ft^2	Weight	I = Ft^4
1	8.76	500	250	320	83178
2	11.348	314	157	459	98332
3	11.348	314	157	517	98332
4	11.348	314	157	517	98332
5	11.348	314	157	517	98332
6	11.348	314	157	517	98332
7	2.00	314	157	304	98332
8	8.00	13	6.6	9	1395
9	9.15	24	12.1	12.1	4812
10	10.10	24	12.1	525.3	4812
11	10.10	36	18	813	7218
12	12.65	48	24	921	9624
13	7.50	48	24	814	9624
14	2.00	252	126	173	50391
15	42.35		1.	500	100.

BASE Weight = 2615 kips IW = 196168.5 K-Ft

Spring Constant for Mass 15 = 136. K/Ft

Foundation Properties:

E_s = 4320 Ksf; μ =0.25
G_s = 1728 Ksf
K Rocking = 127,776,000 K-Ft/Radian
K Shear = 228,096 K/Ft
K Bearing = 66,790 K/Ft
K Bearing = 74,989 K/Ft - For Mass 7

Young's Modulus
E_i = 552, 130 KSF

Poisson's Patio
μ_c = 0.18

Shear Modulus
G_c = 233,950 KSF

TABLE 3.7-15 (Cont'd)

Foundation Properties: (Cont'd)

Design Concrete Strength	$f'c$	= 4000 psi
Young's Modulus -	E_c	= 552,130 Ksf
Poisson's Ratio -	μ_c	= 0.18
Shear Modulus -	G_c	= 233,950 Ksf

TABLE 3.7-16

CONDENSATE STORAGE TANK PROPERTIES
VERTICAL MODEL

(See Figure 3.7-47)

Mass Point	Area Ft^2	Weight (Kips)
1	500.4	320
2	314	459
3	314	517
4	314	517
5	314	517
6	314	517
7	314	304
8	13.1	9
9	24.1	370.8
10	24.1	770.3
11	36.	813.0
12	48.2	921
13	48.2	814
14	251.4	173
BASE		2615

Spring Constant
K = 253,440k/ft

TABLE 3.7-17

DIESEL OIL STORAGE BUILDING PROPERTIES

(See Figure 3.7-48 & 49)

		Horizontal Model				Vertical
		North-South		East-West		
Mass Point	Weight (Kips)	A Shear (Ft^2)	I (Ft^4)	A Shear (Ft^2)	I (Ft^4)	Area (Ft^2)
1	1,165.00	211	162,288	126	44,410	375.0
2	1,208.00	211	162,288	126	44,410	375.0
3	5.44	5	344	5	344	10.36
4	89.29	5	344	5	344	10.36
5	97.00	5	344	5	344	10.36
6	63.00	5	344	5	344	10.36
7	65.00	5	344	5	344	10.36
8	5.44	5	344	5	344	10.36
9	89.29	5	344	5	344	10.36
10	97.00	5	344	5	344	10.36
11	63.00	5	344	5	344	10.36
12	65.00	5	344	5	344	10.36
13	29.72	0.5	5	0.5	5	0.5
14	29.72	0.5	5	0.5	5	0.5
BASE	1820.4	I_W = 565,589 K-Ft^2		I_W = 172,151 K-Ft^2		

Spring Constant K_W (Between Mass Points 4&13, 9&14) = 6.83 K/Ft

For Vertical Model:

Weight of Mass Point 3&8 are $5.44 + \frac{29.72}{2}$ = 20.30 Kips

Weight of Mass Point 4&9 are $89.29 + \frac{29.72}{2}$ = 104.15 Kips

Design concrete strength f'_c = 4000 psi
Young's modulus E_c = 552,130 ksf
Poisson's ratio μ_c = 0.18
Shear modulus G_c = 233,950 ksf

TABLE 3.7-17 (Cont'd)

Foundation Properties:

Soil (30 Ksi)	E-Q Motion	Spring Constants ^{K}Bearing (K/ft)	^{K}Shear (K/ft)	^{K}Rocking (K-ft/radian)
E = 4320 Ksf	N-S	45,774	205,157	161,864,061
G = 1728 Ksf				
μ = 0.25	E-W	64,734	205,157	57,227,588
	Vert	227,952	-	-

TABLE 3.7-18

NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS)

REACTOR BUILDING

	E-W Direction			N-S Direction			Vertical Direction		
Mode No.	Es = 32 Ksi	Es = 40 Ksi	Es = 48 Ksi	Es = 32 Ksi	Es = 40 Ksi	Es = 48 Ksi	Es = 32 Ksi	Es = 40 Ksi	Es = 48 Ksi
1	1.22	1.36	1.48	1.22	1.36	1.48	1.83	2.04	2.23
2	3.08	3.41	3.71	3.07	3.41	3.71	13.05	13.06	13.07
3	7.17	7.21	7.26	7.14	7.19	7. 23	22.25	22.25	22.25
4	11.39	11.47	11.54	10.69	10.74	10.79	26.66	26.66	26.66
5	14.29	14.29	14.29	14.02	14.03	14.03	34.82	34.82	34.82
6	23.42	23.43	23.45	18.31	18.31	18.41	42.73	42.73	42.73
7	24.28	24.28	24.28	24.15	24.15	24.15	54.86	54.86	54.86
8	26.50	26.50	26.51	26.06	26.07	26.07	58.25	58.25	58.25
9	35.60	35.60	35.60	35.58	35.58	35.58	70.36	70.36	70.36
10	40.49	40.49	40.49	40.47	40.47	40.47	77.47	77.47	77.47

TABLE 3.7-19

NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS)

REACTOR AUXILIARY BUILDING

	E-W Direction			N-S Direction			Vertical Direction		
Mode No.	Es = 32 Ksi	Es = 40 Ksi	Es = 48 Ksi	Es = 32 Ksi	Es = 40 Ksi	Es = 48 Ksi	Es = 32 Ksi	Es = 40 Ksi	Es = 48 Ksi
1	2.51	2.80	3.05	2.37	2.63	2.87	2.60	2.90	3.18
2	6.05	6.72	7.32	5.43	6.04	6.58	6.57	13.91	10.91
3	17.79	17.92	18.06	23.08	23.14	23.20	13.91	15.84	15.84
4	24.28	24.32	24.36	27.29	27.50	27.72	15.84	19.62	19.62
5	31.40	31.41	31.42	32.99	33.01	33.04	19.62	21.60	21.60
6	37.93	37.94	37.94	39.00	39.00	39.00	21.60	22.01	22.01
7							22.01	22.74	22.74
8							22.74	33.61	33.63
9							34.61	39.29	39.29
10							39.29	54.18	54.18

TABLE 3.7-20

NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS)

FUEL HANDLING BUILDING

	E-W Direction			N-S Direction			Vertical Direction		
Mode No.	Es = 28 Ksi	Es = 35 Ksi	Es = 42 Ksi	Es = 28 Ksi	Es = 35 Ksi	Es = 42 Ksi	Es = 28 Ksi	Es = 35 Ksi	Es = 42 Ksi
1	1.75	1.95	2.14	2.58	2.88	3.75	3.11	3.47	3.80
2	4.10	4.58	5.01	4.98	5.56	6.08	41.01	41.02	41.03
3	18.83	18.87	18.90	24.70	24.74	24.79	69.43	69.44	69.44
4	40.96	41.02	41.07	36.89	36.94	36.99	154.12	154.12	154.12
5	64.98	64.99	64.99	72.28	72.28	72.28			

TABLE 3.7-21

NATURAL FREQUENCIES IN CYCLES ER SECOND (CPS)

INTAKE STRUCTURE

Mode No.	E-W Direction Es = 17.4 Ksi	N-S Direction Es = 17.4 Ksi	Vertical Direction Es = 17.4 Ksi
1	3.82	3.89	4.07
2	6.84	6.75	51.91
3	37.69	28.73	95.22
4	53.41	32.48	116.58
5	67.61	55.66	132.60
6	86.36	72.15	219.50
7	92.50	131.72	237-14

TABLE 3.7-22

NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS)

DIESEL GENERATOR BUILDING

	E-W Direction			N-S Direction			Vertical Direction		
Mode No.	Es = 24 Ksi	Es = 30 Ksi	Es = 36 Ksi	Es = 24 Ksi	Es = 30 Ksi	Es = 36 Ksi	Es = 24 Ksi	Es = 30 Ksi	Es = 36 Ksi
1	3.29	3.67	4.02	3.23	3.60	3.94	3.74	4.18	4.58
2	5.66	6.32	6.92	5.89	6.57	7.18	59.20	59.21	59.23
3	35.29	35.35	35.40	27.96	28.05	28.14	135.15	135.15	135.15
4	62.10	62.11	62.12	51.56	51.57	51.59	194.02	194.02	194.02
5	82.36	82.38	82.40	55.53	55.53	55.53	201.91	201.91	201.91
6	105.07	105.07	105.07	57.37	57.38	57.38	212.16	212.17	212.17
7	107.89	107.89	107.89	70.31	70.35	70.39			
8	121.62	121.63	121.64	76.41	76.41	76.42			
9	154.52	154.54	154.54	122.43	122.43	122.44			
10	197.73	197.73	197.73	134.54	134.54	134.54			

TABLE 3.7-23

MS/FW TRESTLE SIGNIFICANT NATURAL FREQUENCIES

Mode	NATURAL	FREQUENCY	HZ
No.	N-S	E-W	Vertical
1	4.06	4.79	5.44
2	7.16	10.50	48.21
3	8.79	13.68	98.56
4	12.64	15.48	
5	21.54	21.30	
6	38.78	28.80	
7	43.13	34.26	
8	49.24	41.28	
9	59.61	49.50	

TABLE 3.7-24

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

E - W EARTHQUAKE MOTION, SSE, REACTOR BUILDING

SOIL YOUNG'S MODULUS E_s = 40 Ksi

			Response Spectrum Method				Time History Method			
REACTOR BUILDING	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)
	1	210.0	0.387	0.137	64,500	2,750	0.425	0.170	70,700	3,010
CONCRETE	4	148.0	0.253	0.101	397,500	5,780	0.304	0.126	445,300	7,700
CONTAINMENT	7	88.0	0.155	0.068	893,300	8,870	0.190	0.083	1,027,700	10,560
	10	28.0	0.163	0.036	1,357,500	9,580	0.173	0.041	1,601,700	11,870
	11	206.5	0.312	0.123	- 2,070	103	0.357	0.152	- 2,370	- 118
STEEL	14	142.0	0.221	0.094	-17,700	364	0.268	0.115	20,500	426
CONTAINMENT	17	88.0	0.159	0.064	65,200	902	0.191	0.078	79,100	1,120
	20	28.0	0.166	0.035	116,300	1,110	0.178	0.041	143,902	1,383
	21	68.5	0.152	0.054	2,320	273	0.160	0.069	3,350	292
INTERNAL	22	60.0	0.153	0.050	32,200	1,870	0.152	0.060	32,862	1,694
STRUCTURES	23	44.0	0.157	0.042	92,000	3,000	0.153	0.050	90,900	2,903
	24	24.0	0.167	0.033	114,700	3,800	0.184	0.038	112,500	3,745
FDN MAT	25	- 0.54	0.194	0.024	2,000,600	28,360	0.225	0.024	2,558,100	16,636

TABLE 3.7-25

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

N-S EARTHQUAKE MOTION, SSE, REACTOR BUILDING

SOIL YOUNG'S MODULUS E_s = 40 Ksi

			Response Spectrum Method				Time History Method			
REACTOR BUILDING	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)
	1	210.0	0.391	0.139	65,100	2,770	0.430	0.171	71,100	3,020
CONCRETE	4	148.0	0.255	0.103	401,000	6,840	0.305	0.126	447,200	7,730
CONTAINMENT	7	88.0	0.157	0.069	901,900	8,970	0.181	0.083	1,031,900	10,590
	10	28.0	0.163	0.036	1,371,500	9,570	0.173	0.041	1,607,500	11,900
	11	206.5	0.316	0.125	2,100	105	0.351	0.152	2,330	- 116
STEEL	14	148.0	0.234	0.095	17,900	369	0.268	0.115	20,200	423
CONTAINMENT	17	88.0	0.160	0.065	66,200	914	0.191	0.078	78,800	1,115
	20	28.0	0.166	0.036	118,000	1,130	0.177	0.041	143,600	1,383
	21	68.5	0.157	0.055	2,400	281	0.163	0.065	3,400	298
INTERNAL	22	60.0	0.157	0.051	33,100	1,920	0.155	0.060	19,000	1,730
STRUCTURE	23	44.0	0.161	0.043	94,600	3,070	0.158	0.050	92,600	2,953
	24	24.0	0.170	0.034	117,800	3,880	0.185	0.038	114,500	3,877
FDN MAT	25	- 0.54	0.193	0.024	2,020,100	28,480	0.224	0.024	2,560,000	16,625

TABLE 3.7-26

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

VERT EARTHQUAKE MOTION, SSE, REACTOR BUILDING

SOIL YOUNG'S MODULUS E_s = 40 Ksi

			Response Spectrum Method				Time History Method			
REACTOR BUILDING	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment	Max Force (K)	Max Acce (G)	Max Disp (Ft)	Max Moment	Max Force (K)
	1	210.0	0.144	0.0283		1,219	0.144	0.0275		1,413
CONCRETE	2	168.0	0.144	0.0232		1,259	0.143	0.0274		1,254
CONTAINMENT	3	128.0	0.143	0.0281		1,221	0.142	0.0273		1,215
	4	88.0	0.142	0.0278		1,211	0.141	0.0271		1,203
	5	48.0	0.140	0.0275		1,098	0.139	0.0268		1,039
	6	206.5	0.141	0.0276		84	0.139	0.0268		83
STEEL	7	148.0	0.140	0.0275		306	0.139	0.0268		303
CONTAINMENT	8	108.0	0.140	0.0275		192	0.128	0.0267		189
	9	68.0	0.140	0.0274		191	0.138	0.0267		189
	10	28.0	0.139	0.0273		123	0.138	0.0266		122
	11	68.5	0.140	0.0274		250	0.138	0.0266		248
INTERNAL	12	60.0	0.140	0.0274		1,355	0.138	0.0266		1,340
STRUCTURES	13	44.0	0.189	0.0273		1,010	0.138	0.0266		998
	14	24.0	0.139	0.0273		688	0.138	0.0265		680
FDN MAT	16	- 0.54	0.139	0.0273		16,308	0.137	0.0265		15,775

TABLE 3.7-27

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

N-S EARTHQUAKE MOTION, SSE, REACTOR BUILDING

SOIL YOUNG'S MODULUS E_s = 48 Ksi

			Response Spectrum Method				Time History Method			
REACTOR BUILDING	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)
	1	210.0	0.410	0.124	68,200	2,900	0.375	0.147	62,470	2,660
CONCRETE	4	148.0	0.270	0.092	420,900	7,180	0.287	0.110	400,000	6,980
CONTAINMENT	7	88.0	0.165	0.061	948,000	9,440	0.200	0.074	940,000	9,900
	10	28.0	0.167	0.032	1,443,500	10,230	0.160	0.038	1,491,500	11,570
	11	206.5	0.325	0.110	2,160	110	0.377	0.132	2,605	125
STEEL	14	148.0	0.235	0.084	18,550	380	0.297	0.100	22,240	470
CONTAINMENT	17	88.0	0.171	0.057	69,490	970	0.213	0.069	87,460	1,240
	20	28.0	0.170	0.031	124,707	1,200	0.163	0.037	159,600	1,540
	21	68.5	0.166	0.048	2,540	300	0.182	0.058	2,784	330
INTERNAL	22	60.0	0.166	0.045	35,100	2,040	0.172	0.054	36,810	2,130
STRUCTURES	23	44.0	0.168	0.038	99,810	3,238	0.152	0.046	101,270	3,220
	24	24.0	0.172	0.079	124,140	4,070	0.168	0.035	124,300	3,840
FDN MAT	25	-0.54	0.194	0.021	2,103,900	29,200	0.201	0.023	2,521,630	23,200

TABLE 3.7-28

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

E-W EARTHQUAKE MOTION, SSE, REACTOR AUXILIARY BUILDING

SOIL YOUNG'S MODULUS 40 Ksi

			Response Spectrum Method				Time History Method			
	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)
	1	81.0	0.314	0.0322	25,500	1,280	0.305	0.0324	24,800	1,240
	2	61.0	0.291	0.0301	111,800	4,660	0.286	0.0303	109,400	4,570
	3	42.5	0.268	0.0279	361,100	10,390	0.268	0.0281	356,600	10,300
	4	18.5	0.242	0.0261	767,100	17,300	0.248	0.0254	764,600	17,390
FDN MAT	5	-5.0	0.213	0.0220	971,000	25,870	0.227	0.0223	915,840	20,280

TABLE 3.7-29

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

N-S EARTHQUAKE MOTION, SSE, REACTOR AUXILIARY BUILDING

SOIL YOUNG'S MODULUS 40 Ksi

			Response Spectrum Method				Time History Method			
	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)
	1	81.0	0.392	0.0441	31,900	1,600	0.371	0.0452	30,200	1,510
	2	61.0	0.335	0.0386	133,500	5,490	0.330	0.0397	129,200	5,350
	3	42.5	0.283	0.0332	409,700	11,530	0.290	0.0344	406,300	11,550
	4	18.5	0.224	0.0262	826,100	17,790	0.237	0.0274	838,200	18,400
FDN MAT	5	-5.0	0.175	0.0180	984,400	24,090	0.198	0.0191	977,800	19,690

TABLE 3.7-30

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

VERT EARTHQUAKE MOTION, SSE, REACTOR AUXILIARY BUILDING

SOIL YOUNG'S MODULUS 40 Ksi

	Response Spectrum Method				Time History Method	
	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Acce (G)	Max Disp (Ft)
	1	81.0	0.193	0.0186	0.232	0.0222
	2	61.0	0.192	0.0186	0.231	0.00221
	3	42.5	0.191	0.0185	0.230	0.0220
	4	18.5	0.170	0.0187	0.229	0.0219
FDN MAT	5	-5.0	0.188	0.0182	0.227	0.0217

TABLE 3.7-31

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

E-W EARTHQUAKE MOTION, SSE FUEL HANDLING BUILDING

SOIL YOUNG'S MODULUS 35 Ksi

			Response Spectrum Method				Time History Method			
	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)
	1	96.83	0.351	0.0704	44,500	1,285	0.452	0.0842	57,300	1,653
	2	62.0	0.213	0.0454	76,500	2,287	0.245	0.0527	96,600	2,808
	3	48.0	0.177	0.0358	180,300	3,915	0.172	0.0404	211,100	4,323
FDN MAT	4	15.75	0.195	0.0122	267,400	7,330	0.228	0.0127	310,700	6,021

TABLE 3.7-32

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

N-S EARTHQUAKE MOTION, SSE, FUEL HANDLING BUILDING

SOIL YOUNG'S MODULUS 35 Ksi

			Response Spectrum Method				Time History Method			
	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)
	1	96.83	0.345	0.0331	45,600	1,316	0.322	0.0319	42,500	1,228
	2	62.0	0.270	0.0264	81,800	2,586	0.262	0.0261	74,800	2,399
	3	48.0	0.243	0.0239	216,700	5,092	0.248	0.0239	203,300	4,931
FDN MAT	4	15.75	0.185	0.0169	347,400	8,157	0.210	0.0177	322,800	8,401

TABLE 3.7-33

COMPARISON OF STRUCTURAL RESPONSES FOR SEISMIC CATEGORY I STRUCTURES

USING RESPONSE SPECTRA AND TIME HISTORY METHODS

VERT EARTHQUAKE MOTION, SSE, FUEL HANDLING BUILDING

SOIL YOUNG'S MODULUS 35 Ksi

	Response Spectrum Method				Time History Method	
	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Acce (G)	Max Disp (Ft)
	1	96.83	0.222	0.0150	0.225	0.0150
	2	62.0	0.221	0.0149	0.224	0.0147
	3	48.0	0.221	0.0149	0.223	0.0149
FDN MAT	4	15.75	0.219	0.0148	0.322	0.0148

TABLE 3.7-34

COMPARISON OF STRUCTURAL RESPONSES FOR WITH AND WITHOUT

THE EFFECT OF CLOSELY SPACED MODES

E-W OBE RESPONSE SPECTRUM METHOD

SOIL YOUNG'S MODULUS E_s = 48 Ksi

			SRSS				Effect of Closely Spaced Modes Included			
REACTOR BUILDING	Mass No.	Elevation (Ft)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)	Max Acce (G)	Max Disp (Ft)	Max Moment (K-Ft)	Max Shear (K)
	1	210.0	0.258760	0.079322	43,058.51	1,832.28	0.258760	0.079323	43,058.51	1,832.28
CONCRETE	4	148.0	0.170922	0.058998	266,108.02	4,545.28	0.170922	0.058998	266,108.02	4,545.28
CONTAINMENT	7	88.0	0.105656	0.039045	600,287.13	5,992.96	0.010565 6	0.039045	600,287.15	5,772.96
	10	28.0	0.104577	0.020145	915,605.76	6,513.18	0.104577	0.020145	915,605.82	6,513,18
	11	206.5	0.206962	0.070158	1,374.23	68.71	0.206962	0.070158	1,374.23	68.71
STEEL	14	148.0	0.150324	0.053451	11,822.74	244.87	0.150324	0.053451	11,822.75	244.71
CONTAINMENT	17	88.0	0.109099	0.036307	44,360.69	618.09	0.107099	0.036387	44,360.70	618.09
	20	28.0	0.106395	0.019852	79,641.62	764.81	0.106395	0.019852	79,641.63	764.81
	21	65.0	0.012379	0.000482	1,561.63	183.72	0.102294	0.030483	1,561.63	183.72
INTERNAL	22	60.0	0.101836	0.028207	21,516.33	1,247.45	0.101836	0.028207	21,516.33	1,247.40
STRUCTURE	23	44.0	0.103060	0.023764	61,202.43	1,986.26	0.103060	0.023964	61,202.43	1,986.28
	24	24.0	0.108145	0.018799	76,151.82	2,500.18	0.108145	0.018799	76,151.82	2,500.18
FDN MAT	25	-0.54	0.121003	0.013136	1,337,298.24	18,375.53	0.121003	0.013136	1,337,295.34	18,335.53

TABLE 3.7-35

SOILS-SUPPORTED

SEISMIC CATEGORY I STRUCTURES

Structure	Foundation Elevation in Feet	Depth of Engineered Back Fill
Reactor Building	-25	35
Reactor Auxiliary Building	-18	42
Fuel Handling Building	+12	72
Intake Structure	-35	25
Diesel Generator Building	+10	70
Steam Trestle	-5	55
Component Cooling Water Heat Exchange Structure	+10	70
Condensate Storage Tank	+10	70
Refueling Water Tank	+10	70
Diesel Fuel Oil Storage Tanks	+10	70

TABLE 3.7-36

SIDE SPRING CONSTANTS EFFECT ON SOIL STRUCTURE INTERACTION

Magnitude of Side Spring, %	Maximum Acceleration, g Top Shield Bldg	Top Steel Containment Vessel	Foundation Mat
120	0.3294	0.2613	0.1737
110	0.3294	0.2620	0.1715
100	0.3293	0.2626	0.1692
90	0.3290	0.2631	0.1667
80	0.3285	0.2635	0.1642

TABLE 3.7-37

COMPOSITE MODAL DAMPING RATIOS FOR SEISMIC CATEGORY I STRUCTURES

REACTOR BUILDING, SSE

SOIL YOUNG'S MODULUS E_s = 40 KS

Mode No.	Composite Modal Damping Factors % E-W	N-S	Vertical
1	6.49	6.47	8.71
2	6.69	6.68	7.37
3	6.04	6.02	4.22
4	6.43	6.39	7.03
5	6.98	6.93	7.03
6	6.25	6.52	7.39
7	4.70	4.04	7.03
8	6.81	6.93	4.03
9	6.99	7.00	7.02
10	4.02	4.02	4.03
11	7.00	6.99	7.00
12	7.00	7.00	4.00
13	7.00	7.00	7.03
14	6.96	6.97	7.00
15	4.04	4.03	4.03
16	7.00	7.90	7.07
17	4.00	4.00	
18	7.00	7.00	
19	7.00	7.00	
20	7.00	6.99	
21	4.00	7.00	
22	4.00	4.00	
23	6.98	4.00	
24	4.02	4.00	
25	4.00	4.00	
26	4.00	4.00	

TABLE 3.7-38

MAXIMUM STRESS COMPARISON

Sample Problem	Static		Dynamic	
	Point	Stress (psi)	Point	Stress psi
No. 1 Fig 3.7-254	50	16047	50	8200
No. 2 Fig 3.7-255, 256	28	9235	4	7583
No. 3 Fig 3.7-257	2	8528	2	6039

TABLE 3.7-38a

HIGH STRESS COMPARISON
based on Flat 1.0g response

Sample Problem 1
(Fig. 3.7-254)

Point No.	Seismic Stress (PSI) Static	Dynamic	Difference
16	13083	7304	5779*
14	8371	5559	2812
18	7355	4789	2566
1	6260	4921	1339
4	5525	4386	1139
15	4422	3743	679
30	1969	5572	-3603
29	1964	5566	-3602

Sample Problem 2
(Fig. 3.7-255)
(Fig. 3.7-256)

Point No.	Seismic Stress (PSI) Static	Dynamic	Difference
25	8755	7031	1724
31	8291	5112	3179*
19	8002	5026	2976
28	7733	6521	1212
21	6732	5293	1439
17	6708	4143	2565
5	6337	5385	952
13	6008	4699	1309
36	2663	3505	-842
32	1544	2389	-845

Sample Problem 3
(Fig. 3.7-257)

Point No.	Seismic Stress (PSI) Static	Dynamic	Difference
2	8179	5717	2462*
27	6577	4708	1869
8	6553	4459	2094
4	6317	5180	1137
3	6174	4734	1440
1	5428	4008	1420
2501	5198	4091	1107
5	5169	3723	1446
100	1357	2054	-697

*greatest difference

TABLE 3.7-38b

HIGH STRESS COMPARISON
based on floor response spectra

Sample Problem 1
(Fig. 3.7-254)

Point No.	Seismic Stress (PSI) Static	Dynamic	Difference
16	6853	1634	5219*
18	3645	1087	2558
1	3226	1936	1290
14	3152	1502	1650
15	2647	644	2003
4	2384	1524	860
30	711	1322	-611
29	709	1320	-611

Sample Problem 2
(Fig. 3.7-255)
(Fig. 3.7-256)

Point No.	Seismic Stress (PSI) Static	Dynamic	Difference
31	3894	1315	2597
19	3673	1315	2358
17	3649	1010	2639*
21	3165	1236	1929
5	3042	1330	1712
13	2983	1101	1882
12	2784	1055	1729
30	2745	1005	1740

Sample Problem 3
(Fig. 3.7-257)

Point No.	Seismic Stress (PSI) Static	Dynamic	Difference
27	3289	1134	2155*
1	3029	1324	1705
13	2804	1100	1704
2	4212	2518	1694
4	3162	1510	1652
20	2615	991	1624
8	3286	2055	1231
3	3139	2088	1051

- greatest difference

TABLE 3.7-39

NATURAL FREQUENCIES AND DOMINANT DEGREES OF FREEDOM

Mode Number	Frequency (CPS)	Dominant Degree of Freedom Joint No.	Direction	Location
1	3.32	9932	Z	Reactor Internal
2	3.32	9932	X	Reactor Internal
3	4.60	1103	X	R.C.P Top Mass
4	4.70	5103	X	R.C.P Top Mass
5	5.11	2103	X	R.C.P Top Mass
6	5.13	4103	X	R.C.P Top Mass
7	6.25	2103	Z	R.C.P Top Mass
8	6.27	4103	Z	R.C.P Top Mass
9	6.91	1103	Z	R.C.P Top Mass
10	8.27	5103	Z	R.C.P Top Mass
11	10.40	1103	X	R.C.P Top Mass
12	10.51	404	X	S.G. Top Mass
13	10.52	3404	X	S.G. Top Mass
14	10.63	5103	X	R.C.P Top Mass
15	10.82	2103	X	R.C.P Top Mass
16	10.94	4103	X	R.C.P Top Mass
17	12.07	9901	Z	Reactor Internal
13	12.09	9901	X	Reactor Internal
19	19.14	409	Z	S.G. Lower Mass
20	19.18	3409	Z	S.G. Lower Mass
21	22.68	2101	Z	R.C.P Lower Mass
22	22.74	4101	Z	R.C.P Lower Mass
23	24.11	1101	Z	R.C.P Lower Mass
24	24.57	5101	Z	R.C.P Lower Mass
25	24.82	9932	Y	Reactor Internal
26	27.40	3409	X	S.G. Lower Mass
27	27.68	3404	Y	S.G. Top Mass
28	27.80	404	Y	S.G. Top Mass
29	30.26	409	Z	S.G. Lower Mass
30	31.94	404	Z	S.G. Top Mass
31	31.96	3404	Z	S.G. Top Mass
32	35.84	9998	Z	R.V. Lower Mass
33	38.91	1101	X	R.C.P Lower Mass
34	40.67	2301	X	R.C.P Lower Mass
35	41.42	5301	X	R.C.P Lower Mass
36	42.82	2101	X	R.C.P Lower Mass
37	43.24	4101	X	R.C.P Lower Mass
38	53.72	9996	Z	R.V.Mass at Flange
39	57.28	9996	X	R.V.Mass at Flange
40	62.85	9901	Y	Reactor Internal
41	105.73	9996	Y	R.V. Mass at Flange
42	250.48	3500	Z	S.G. Bottom Tube Sheet
43	250.53	500	Z	S.G. Bottom Tube Sheet
44	253.53	3500	X	S.G. Bottom Tube Sheet
45	258.19	500	X	S.G. Bottom Tube Sheet

TABLE 3.7-39 (Cont'd)

Mode Number	Frequency (CPS)	Dominant Degree of Freedom		
		Joint No.	Direction	Location
1,2	14.34	106	X, Z	Pressurizer
3	61.14	106	Y	Pressurizer
4,5	62.60	106	X, Z	Pressurizer
6,7	167.82	106	X, Z	Pressurizer
8,9	209.94	101	X, Z	Pressurizer
10,11	363.32	103	X, Z	Pressurizer
1	5.65	5	Y	Surge Line
2	13.82	11	X, Z	Surge Line
3	16.42	6,3	X	Surge Line
4	20.94	11	Y	Surge Line
5	24.48	13,7	Z	Surge Line
6	30.65	6,4	Y	Surge Line
7	38.23	7,8	X	Surge Line
8	49.21	4,5	X	Surge Line
9	69.70	3	X	Surge Line
10	74.15	13	Y	Surge Line
11	84.56	13,8	X	Surge Line
12	157.63	5,4	X, Z	Surge Line
13	162.21	5,4	Y	Surge Line
14	179.83	3	Y	Surge Line
15	206.49	7	Y	Surge Line
16	230.90	7	X	Surge Line
17	254.34	7	Y	Surge Line
18	293.40	7	Y	Surge Line
19	348.27	8,7	Y	Surge Line
20	403.80	4,8,5	X, Y, Z	Surge Line
21	515.11	6,4	Z	Surge Line
22	623.61	8,7	Y	Surge Line
23	648.56	2	X, Z	Surge Line
24	685.94	3,4	Z	Surge Line
25	785.87	2	X, Z	Surge Line
26	883.67	7	Z	Surge Line
27	1831.5	2	Y	Surge Line

TABLE 3.7-39 (Cont'd)

Mode	Frequency	Location
I	5.18	Spray Line Region 3
2	5.22	Spray Line Region 2
3	5.41	Spray Line Region 1
4	6.15	Spray Line Region 2
5	6.33	Spray Line Region 2
6	7.17	Spray Line Region 1
7	8.15	Spray Line Region 3
8	8.35	Spray Line Region 2
9	8.41	Spray Line Region 1
10	9.50	Spray Line Region 2
11	10.49	Spray Line Region 1
12	11.30	Spray Line Region 2
13	12.06	Spray Line Region 1
14	13.48	Spray Line Region 1
15	14.08	Spray Line Region 2
16	15.01	Spray Line Region 1
17	16.11	Spray Line Region 1
18	16.28	Spray Line Region 2
19	16.88	Spray Line Region 3
20	17.31	Spray Line Region 1
21	18.66	Spray Line Region 3
22	18.93	Spray Line Region 2
23	19.23	Spray Line Region I
24	19.90	Spray Line Region 1
25	20.64	Spray Line Region 2
26	20.91	Spray Line Region 2
27	21.00	Spray Line Region 3
28	22.03	Spray Line Region 1
29	23.13	Spray Line Region 1
30	25.55	Spray Line Region 2
31	23.62	Spray Line Region 3

TABLE 3.7-40

(HISTORICAL)
COMPARISON OF CALCULATED MAXIMUM AND SPECIFIED
SEISMIC LOAD

Seismic Excitation	System Location	Component of Reactions	Seismic Load: Calculated Maximum	Seismic Load: Specified for Design
$\sqrt{X^2+Y^2+Z^2}$ OBE	Reactor Inlet Nozzle	Fx	108.0	109.
		Fy	15.8	16.
		Fz	104.3	105.
		Mx	5611.8	5632.
		My	3281.7	3282.
		Mz	4224.3	4250.
	Reactor Outlet Nozzle	Fx	123.2	124.
		Fy	137.7	138.
		Fz	136.0	136.
		Mx	4059.5	4060.
		My	15582.5	15583.
		Mz	10991.1	10992.
$\sqrt{X^2+Y^2+Z^2}$ OBE	Steam Generator Inlet Nozzle	Fx	177.6	178.
		Fy	147.3	148.
		Fz	136.0	136.
		Mx	5753.6	5769.
		My	8032.9	8044.
		Mz	11453.5	11454.
	Steam Generator Outlet Nozzle	Fx	79.3	82.
		Fy	74.3	75.
		Fz	79.3	82.
		Mx	7149.3	7157.
		My	6037.8	6041.
		Mz	7149.3	7157.

X,Y,Z in Global Coordinates

Forces = Kips

Moments = Inch-Kips

TABLE 3.7-40 (Cont'd)

(HISTORICAL)

Seismic Excitation	System Location	Component of Reactions	Seismic Load Calculated Maximum	Seismic Load Specified for Design
$\sqrt{X^2+Y^2+Z^2}$ OBE	Pump Inlet Nozzle	Fx	36.1	74.
		Fy	74.0	100.
		Fz	30.8	114.
		Mx	2790.3	9428
		My	2754.9	10685.
		Mz	8125.5	11701.
	Pump Outlet Nozzle	Fx	108.5	251.
		Fy	15.7	98.
		Fz	79.4	99.
		Mx	3785.4	17781.
		My	4511.0	3875.
		Mz	5100.4	12637.
$\sqrt{X^2+Y^2+Z^2}$ OBE	Reactor Vessel Inlet Piping	M	6410.1	12000.
	Reactor Vessel Outlet Piping	M	19469.2	19503.
	Steam Generator Inlet Piping	M	14932.4	15138.
	Steam Generator Outlet Piping	M	7558.2	12000.
	Pump Inlet Piping	M	8622.6	12000.
	Pump Outlet Piping	M	6743.1	12000.
	Cold Leg Elbow	M	6474.4	12000.

Forces = KIPS
Moments = INCH-KIPS

$$M = \sqrt{M_x^2 + M_y^2 + M_z^2}$$

TABLE 3.7-40 (Cont'd)

(HISTORICAL)

Seismic Excitation	System Location	Component of Reactions	Seismic Load Calculated Maximum	Seismic Load Specified for Design
$\sqrt{X^2+Y^2+Z^2}$ OBE	Steam Generator Upper Key	Fz	109.0	240.
	Steam Generator Snubbers	Fx	135.2	160.
	Steam Generator Support Skirt	Fy	142.0	142.
		Fz	651.4	652.
		Mx	44205.9	44206.
		My	2211.8	2219.
		Mz	32842.3	32843.
	Reactor Vessel	H	261.1	262.
	Outlet Support	V	400.9	401.
	Reactor Vessel	H	310.8	311.
	Inlet Support	V	236.7	237.
	Pump Hanger	Fy	2.7	4.
	Pump Snubber	F_a	31.7	32.
	Pressurizer Support	Fx	32.3	82.
		Fy	22.0	81.
		Fz	28.0	83.
		Mx	5863.3	16833.
		Mz	6765.3	16844.

X, Y, Z in Global Coordinates

V - Vertical

H - Horizontal

a - Axial

TABLE 3.7-40 (Cont'd)

(HISTORICAL)

Seismic Excitation	System Location	Component of Reactions	Seismic Load: Calculated Maximum	Seismic Load: Specified for Design
$\sqrt{X^2+Y^2+Z^2}$ OBE	Surge Line Piping (Press Side)	M	131.5	150.0
	Surge Line Piping (Hot Leg Side)	M	77.9	100.0
	Surge Line Hanger VI	Fy	0.14	0.3
	Surge Line Hanger V2	Fy	0.20	0.5
	Surge Line Hanger V3	Fy	0.06	0.15

Forces = Kips

Moments = Inch-Kips

Y = Vertical

TABLE 3.7-40 (Cont'd)

(HISTORICAL)

Seismic Excitation	System Location	Component of Reactions	Seismic Load: Calculated Maximum	Seismic Load: Specified for Design
$\sqrt{X^2+Y^2+Z^2}$ SSE	Reactor Inlet Nozzle	Fx	185.6	218.
		Fy	27.0	32.
		Fz	175.5	210.
		Mx	10481.0	11264.
		My	5597.0	6564.
		Mz	7939.9	8500.
	Reactor Outlet Nozzle	Fz	221.9	248.
		Fy	240.5	276.
		Fz	231.0	272.
		Mx	6963.7	8120.
		My	26457.6	31166.
		Mz	19173.0	21984.
	Steam Generator Inlet Nozzle	Fx	314.2	356.
		Fy	258.3	296.
		Fz	231.0	272.
		Mx	9817.4	11592.
		My	13703.1	16088.
		Mz	19837.5	22908.
	Steam Generator Outlet Nozzle	Fx	141.9	164.
		Fy	136.5	150.
		Fz	141.9	164.
		Mx	13308.6	14314.
		My	11224.1	12082.
		Mz	13308.5	14314.

X,Y,Z in Global Coordinates

Forces = Kips

Moments = Inch-Kips

TABLE 3.7-40 (Cont'd)

(HISTORICAL)

Seismic Excitation	System Location	Component of Reactions	Seismic Load: Calculated Maximum	Seismic Load: Specified for Design
$\sqrt{X^2+Y^2+Z^2}$ SSE	Pump Inlet Nozzle	Fx	61.5	148.
		Fy	137.6	200.
		Fz	48.8	228
		Mx	5092.5	18856.
		My	4514.3	21370.
		Mz	14838.8	23402.
	Pump Outlet Nozzle	Fx	180.8	502.
		Fy	26.7	196.
		Fz	133.7	198.
		Mx	6710.4	35562.
		My	7810.0	17750.
		Mz	8707.2	25274.
$\sqrt{X^2+Y^2+Z^2}$ SSE	Reactor Vessel Inlet Piping	M	14290.5	24000.
	Reactor Vessel Outlet Piping	M	33408.1	39006.
	Steam Generator Inlet Piping	M	26032.3	30276.
	Steam Generator Outlet Piping	M	21913.8	24000.
	Pump Inlet Piping	M	15970.3	24000.
	Pump 0utlet Piping	M	13484.8	24000.
	Cold Leg Elbow	M	11692.4	24000.

Forces = KIPS
Moments = Inch-Kips
X,Y,Z in Global Coordinates
M = $\sqrt{Mx^2+My^2+Mz^2}$

TABLE 3.7-40 (Cont'd)

(HISTORICAL)

Seismic Excitation	System Location	Component of Reactions	Seismic Load: Calculated Maximum	Seismic Load: Specified for Design
$\sqrt{X^2+Y^2+Z^2}$ SSE	Steam Generator Upper Key	Fz	181.9	480.
	Steam Generator Snubbers	Fx	194.3	320.
	Steam Generator Support Skirt	Fy	251.2	284.
		Fz	1048.6	1304.
		Mx	72418.3	88412.
		My	3607.6	4438.
		Mz	56314.4	65686.
	Reactor Vessel Outlet Support	H	425.4	524
		V	657.5	802.
	Reactor Vessel Inlet Support	H	520.5	622.
		V	379.3	474.
	Pump Hanger	Fy	4.9 (5.9)	8.
	Pump Snubber	F_a	53.7 (64.5)	64.*
	Pressurizer Support	Fx	43.1	164.
		Fy	44.0	162.
		Fz	34.5	166.
		Mx	7216.3	33666.
		Mz	9020.4	33688.

( ) denotes 2B1 RCP only.

* 64 k specified for original design. Snubber capacity is 150 k.

X,Y,Z in Global Coordinates

V - Vertical

H - Horizontal

a - Axial

TABLE 3.7-40 (Cont'd)

(HISTORICAL)

Seismic Excitation	System Location	Component of Reactions	Seismic Load Calculated Maximum	Seismic Load Specified for Design
$\sqrt{X^2+Y^2+Z^2}$ SSE	Surge Line Piping (Pressurizer Side)	M	196.5	300.0
	Surge Line Piping (Hot Leg Side)	M	125.8	200.0
	Surge Line Hanger VI	Fy	0.25	0.6
	Surge Line Hanger V2	Fy	0.38	1.0
	Surge Line Hanger V3	Fy	0.11	0.3

Force = Kips

Moment = Inch-Kips

Y - Vertical

TABLE 3.7-41

VERTICAL SEISMIC MODEL MASS POINT LOCATIONS AND DESCRIPTION

Node	Description of Node Coordinate
1	Top of lower Core Support Barrel flange
2	Bottom of Core Support Barrel
3	Core Support Barrel at section change
4	Core Support Barrel at section change (lwr.-center sect.)
5	Core Support Barrel (center section)
S	Core Support Barrel (center section)
7	Core Support Barrel at section change (cntr.-upper sect.)
8	Core Support Barrel Bottom of outlet nozzle
9	Core Support Barrel top of outlet nozzle
10	Top of Core Support Barrel
11	Top of Core Support Barrel upper flange
12	Middle of Core Support Plate
13	Core shroud
14	Core shroud
15	Core shroud
16	Top of core shroud
17	Top of Lower Support Structure grid beams
13	Top of Core Support Plate
19	Top of the lower end fitting
20	Guide tube at inconnel spacer grid
21	Guide tube at spacer grid
22	Guide tube at spacer grid
23	Guide tube at spacer grid
24	Guide tube at spacer grid
25	Guide tube at spacer grid
26	Guide tube at spacer grid
27	Guide tube at spacer grid
28	Guide tube at spacer grid
29	Guide tube at spacer grid
30	Guide tube at spacer grid
31	Bottom of the upper end fitting
32	Fuel at a spacer grid
33	Fuel at a spacer grid
34	Fuel at a spacer grid
35	Fuel at a spacer grid
36	Fuel at a spacer grid
37	Fuel at a spacer grid
33	Fuel at a spacer grid
39	Fuel at a spacer grid
40	Fuel at a spacer grid
41	Fuel at a spacer grid
42	Fuel at a spacer grid
43	Middle of the fuel alignment plate
44	CEA shrouds at the scupper

TABLE 3.7-41 (Cont'd)

Node	Description of Node Coordinate
45	CEA shrouds near center of outlet nozzle
46	CEA shrouds
47	Middle of Upper Guide Structure Support Plate
48	Upper Guide Structure cylinder at the Upper Guide Structure Support Plate
49	Top of the Upper Guide Structure cylinder
50	Top of the Upper Guide Structure flange
51	Reactor vessel--internals interface
52	Reactor vessel--internals interface

TABLE 3.7-42

LATERAL SEISMIC MODEL MASS POINT LOCATIONS AND DESCRIPTIONS

Node	Description of Node
1	Reactor vessel--Internals interface
2	Reactor vessel
3	Reactor vessel
4	Reactor vessel
5	Reactor vessel
6	Reactor vessel
7	Reactor vessel
8	Reactor vessel
9	Reactor vessel
10	Reactor vessel @ snubber elevation
11	Reactor vessel
12	Core support barrel
13	Core support barrel
14	Core support barrel
15	Core support barrel
16	Core support barrel
17	Core support barrel
18	Core support barrel
19	Core support barrel
20	Core support barrel @ snubber elevation
21	Top of core support barrel lower flange
22	Core support plate
23	Top of lower support structure columns
24	CEA shroud extensions (91)
25	Upper guide structure support plate
26	CEA shrouds (91)
27	Fuel alignment plate (centerline)
28	Core shroud
29	Core shroud
30	Core shroud
31	Core shroud
32	Core shroud
33	18 outer fuel bundles
34	18 outer fuel bundles
35	18 outer fuel bundles
36	18 outer fuel bundles
37	18 outer fuel bundles
38	181 inner fuel bundles
39	181 inner fuel bundles
40	181 inner fuel bundles
41	181 inner fuel bundles
42	181 inner fuel bundles
43	18 outer fuel bundles
44	18 outer fuel bundles
45	18 outer fuel bundles
46	18 outer fuel bundles
47	18 outer fuel bundles

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TABLE 3.7-43

CEDM LOADS - PRESSURE HOUSING AND NOZZLE

Height a long CEDM (inches)	Axial Forces(lbs) Mech.	Axial Seismic OBE	Axial Seismic SSE	Shear Forces(lbs) Mech.	Shear Seismic OBE	Shear Seismic SSE	Bending Moments (inch Kips) Mech.	Bending Seismic OBE	Bending Seismic SSE
-26.768	1201	57	107	302	537	740	18.89	69.31	94.28
-20.70	1201	57	107	302	537	740	17.88	67.25	91.67
-14.619	1201	57	107	302	537	740	16.87	65.20	89.10
-8.538	1201	57	107	302	537	740	15.85	63.13	86.60
-2.456	1201	56	107	302	536	740	14.85	61.08	84.14
3.625	1200	56	106	302	536	740	13.84	59.03	81.68
5.813	1150	56	105	302	535	740	13.56	57.86	80.13
10.063	1148	54	102	302	534	734	13.01	55.61	77.02
15.75	1061	51	96	277	528	731	12.45	52.62	72.91
16.812	1060	50	95	277	526	729	12.20	52.06	72.14
18.352	955	46	88	228	516	718	11.94	51.27	71.04
19.522	955	46	87	228	516	718	11.92	50.67	70.20
21.062	955	46	87	228	515	718	11.66	49.88	69.12
22.562	955	46	87	228	515	717	11.70	49.11	68.05
24.102	955	46	87	228	515	717	11.47	49.32	66.96
24.772	955	46	86	228	515	717	11.47	47.98	66.49
26.312	937	46	86	228	514	716	11.25	47.19	65.39
27.812	937	45	85	228	513	715	11.25	46.43	64.33
29.352	937	45	85	227	513	715	11.03	45.64	63.24
30.522	937	45	85	227	513	715	11.04	45.05	62.42

TABLE 3.7-43 (Cont'd)

Height a long CEDM (inches)	Axial Forces(lbs) Mech.	Axial Forces(lbs) Seismic OBE	Axial Forces(lbs) Seismic SSE	Shear Forces(lbs) Mech.	Shear Forces(lbs) Seismic OBE	Shear Forces(lbs) Seismic SSE	Bending Moments (inch Kips) Mech.	Bending Moments (inch Kips) Seismic OBE	Bending Moments (inch Kips) Seismic SSE
32.062	832	41	77	200	489	700	10.82	44.31	61.38
36.062	832	41	77	176	371	508	10.61	42.82	59.40
37.602	832	40	76	175	369	505	10.39	42.27	58.64
38.897	832	40	76	175	369	504	10.40	41.79	58.00
40.437	831	40	75	175	368	504	10.41	41.22	57.25
42.062	831	40	75	175	368	500	10.19	40.63	56.45
43.602	797	39	74	175	366	498	10.20	40.06	55.70
46.772	796	38	73	175	362	496	10.00	38.92	54.16
48.312	796	38	72	173	361	490	10.00	38.37	53.43
51.25	672	32	61	123	338	458	9.63	37.38	52.09
54.312	195	10	18	25	58	80	1.08	3.76	5.19
64.496	194	9	17	26	54	74	1.05	3.22	4.45
74.679	194	9	16	26	52	72	.78	2.69	3.73
84.863	177	8	16	3	50	69	.76	2.18	3.04
95.046	176	8	15	3	48	66	.50	1.71	2.4
105.23	176	8	14	3	44	62	.50	1.27	1.8
115.414	158	7	14	2	41	56	.52	.88	1.26
125.597	158	7	13	2	37	50	.53	.54	.79
135.781	157	6	13	2	31	43	.55	.33	.48
145.964	124	6	11	2	27	36	.31	.31	.5
156.148	123	5	10	2	20	28	.32	.42	.64
166.332	123	5	9	2	14	20	.31	.54	.76
176.515	71	4	7	2	8	11	.31	.57	.82

TABLE 3.7-43 (Cont'd)

Height a long CEDM (inches)	Axial Forces(lbs) Mech.	Axial Seismic OBE	Axial Seismic SSE	Shear Forces(lbs) Mech.	Shear Seismic OBE	Shear Seismic SSE	Bending Moments (inch Kips) Mech.	Bending Seismic OBE	Bending Seismic SSE
186.699	70	3	6	2	10	15	.29	.55	.78
196.882	70	3	5	2	18	26	.28	.45	.64
207.066	36	2	3	2	27	37	.27	.26	.37
217.25	35	1	2	2	27	37	.05	.02	.02
219.25	35	0	1	0	7	9	0.00	.00	.00

TABLE 3.7-44

CEDM NOZZLE LOADS

(a) Computed Nozzle Loads

Load Type	Axial Forces (Kips)	Shear Force(Kips)	Bending Moment (in. Kips)
Mechanical			
Deterministic	0.059	0.017	.59
Random	1.142	0.285	18.29
TOTAL	1.201	0.302	18.88
Seismic			
OBE	0.057	0.537	69.31
SSE	0.10	0.740	94.28
Mech. + OBE	1.258	0.839	88.19
Mech. + SSE	1.308	1.042	113.16

(b) Allowable vs Calculated Nozzle Stresses in Ksi

	Load Category	
	Mech. + OBE	Mech. + SSE
Allowable Loads	23.30	55.93
Calculated Load	19.34	23.64

CALCULATED LOAD:

Mech.+ OBE CASE $3.984 + .151V + .172\ M$

Mech. + SSE CASE $3.978 + .151V + .172\ M$

Where V = Axial Force (Kips)
M = Bending Moment (in-Kips)

TABLE 3.7-45

LATERAL DEFLECTIONS OF CEDM - NOZZLE AND PRESSURE HOUSING

Height along CEDM (inches)	Seismic Loading OBE (inches)	Seismic Loading SSE (inches)	Mechanical Loading (inches)
-26.768	.0	.0	.0
-20.700	.004	.004	.0
-14.619	.014	.018	.005
- 8.538	.031	.040	.010
- 2.456	.055	.070	.017
3.625	.085	.108	.024
5.813	.096	.124	.029
10.063	.120	.154	.034
I5.750	.151	.194	.043
16.812	.156	.202	.046
18.352	.165	.213	.048
19.522	.172	.221	.051
21.062	.180	.233	.053
22.562	.189	.244	.055
24.102	.I99	.256	.058
24.772	.202	.261	.058
26.312	.211	.273	.062
77.8I2	.220	.284	.065
29.352	.230	.296	.067
30.522	.237	.305	.070

TABLE 3.7-45 (Cont'd)

Height along CEDM (inches)	Seismic Loading OBE (inches)	Seismic Loading SSE (inches)	Mechanical Loading (inches)
32.062	.247	.319	.072
36.062	.272	.351	.079
37.602	.282	.363	.082
38.897	.290	.374	.084
40.437	.300	.387	.087
42.062	.311	.400	.089
43.602	.321	.414	.094
46.772	.342	.442	.099
48.312	.354	.455	.101
51.25	.373	.482	.108
54.312	.395	.509	.113
64.496	.472	.608	.134
74.679	.558	.719	.158
84.863	.651	.841	.184
95.04O	.752	.972	.201
105.23	.901	1.114	.240
115.414	.967	1.261	.264
125.597	1.078	1.412	.313
135.781	1.191	1.565	.337
145.964	1.304	1.719	.361

TABLE 3.7-45 (Cont'd)

Height along CEDM (inches)	Seismic Loading OBE (inches)	SSE (inches)	Mechanical Loading (inches)
156.148	1.416	1.872	.408
166.332	1.528	2.023	.432
176.515	1.638	2.172	.455
186.699	1.746	2.320	.502
196.882	1.853	2.464	.525
207.066	1.958	2.608	.566
217.25	2.063	2.750	.572
219.25	2.084	2.778	.596

TABLE 3-7-46

COMPARISON OF COMPUTED CEDM STRESS INTENSITIES
WITH STRESS ALLOWABLES

(CRITICAL WALL SECTION STRESSES IN KSI)

Location	Conditions	Design (Normal plus Upset)	Emergency (a)	Test
Section I	Stress Intensity	9.9	8.9	12.4
	Pm Allowable	Sm = 16.7	1.2Sm = 20.3	0.9Sy = 19.3
Section II	Stress Intensity	11.0	13.2	13.8
	Pm Allowable	Sm = 16.7	1.2Sm = 20.3	0.9Sy = 19.3
Section III	Stress Intensity	14.1	12.7	17.7
	Pm Allowable	Sm = 18.2	1.2Sm = 21.9	0.9SY = 21.5
Section IV	Stress Intensity	25.6	27.9	30.4
	Pm Allowable	Sm = 29.9	1.2Sm = 36.5	0.9Sy = 64.2
Section V	Stress Intensity	11.3	10.2	14.2
	Pm Allowable	Sm = 23.3	1.2Sm = 28.0	0.9Sy = 26.8

a. Stress intensity also applies to the faulted condition, however, faulted allowables exceed emergency allowables and thus are not shown.

TABLE 3.7-47

COMPARISON OF COMPUTED CEDM STRESS INTENSITIES
WITH STRESS ALLOWABLES

(OMEGA SEAL STRESSES IN KSI)

Location	Conditions	Design (Normal plus Upset)	Emergency (a)	Test
Seal A	Stress Intensity	10.9	9.8	13.6
	Pm Allowable	SM = 16.7	1.2Sm = 20.3	0.9Sy = 19.3
Seal B	Stress Intensity	11.7	10.6	14.7
	Pm Allowable	Sm = 16.7	1.2Sm = 20.3	0.9Sy = 19.3
Seal C	Stress Intensity	11.7	10.6	14.7
	Pm Allowable	Sm = 16.7	1.2Sm = 20.3	0.9Sy = 19.3
Seal D	Stress Intensity	11.7	10.6	14.7
	Pm Allowable	Sm = 23.3	1.2Sm = 28.0	0.9Sy = 26.8

a. Stress intensity also applies to the faulted condition, however, faulted allowables exceed emergency allowables and thus are not shown.

TABLE 3.7-48

COMPARISON OF COMPUTED CEDM STRESS INTENSITIES

WITH STRESS ALLOWABLES

(SCREW THREAD STRESSES IN KSI)

Location	Conditions	Design (Normal plus Upset)	Emergency (a)	Test
Thread A	Stress Intensity	1.6	1.4	1.9
	Allowable	0.6Sm = 10.0	0.6Sm = 10.1	0.6Sm = 11.6
Thread B	Stress Intensity	1.8	1.6	2.2
	Allowable	0.6Sm = 10.0	0.6Sm = 10.1	0.6Sm = 11.6
Thread C	Stress Intensity	6.3	5.7	7.0
	Allowable	0.6Sm = 10.0	0.6Sm = 10.1	0.6Sm = 11.6
Thread D	Stress Intensity	2.5	2.2	2.7
	Allowable	0.6Sm = 14.0	0.6Sm = 14.0	0.6Sm = 14.0

a. Stress also applies to the faulted condition, however, faulted allowables exceed emergency allowables and thus are not shown.

TABLE 3.7-49

WATERFORD NO. 3, NATURAL FREQUENCIES IN CYCLES PER SECOND (CPS)

Node No.	E-W EARTHQUAKE G = 6,400 PSI Without Torsion	E-W EARTHQUAKE G = 6,400 PSI With Torsion	E-W EARTHQUAKE G = 16,050 Psi Without Torsion	E-W EARTHQUAKE G = 16,050 Psi With Torsion	N-S EARTHQUAKE G = 6,400 PSI Without Torsion	N-S EARTHQUAKE G = 6,400 PSI With Torsion	N-S EARTHQUAKE G = 16,050 PSI Without Torsion	N-S EARTHQUAKE G = 16,050 PSI With Torsion
1	1.091	1.086	1.706	1.700	1.087	1.086	1.702	1.700
2	2.445	1.684	3.334	2.620	2.468	1.815	3.410	2.833
3	4.562	2.450	5.248	3.363	4.275	2.468	4.883	3.410
4	7.535	4.545	7.571	4.684	7.475	4.265	7.491	4.701
5	10.936	4.678	10.965	5.184	10.254	4.680	10.284	4.860
6	11.975	6.529	11.982	6.587	10.807	6.741	10.863	6.767
7	12.154	7.626	12.155	7.696	12.125	7.511	12.129	7.539
8	14.874	11.464	15.046	11.471	14.914	10.054	14.940	10.083
9	20.438	12.004	20.464	12.009	19.270	10.826	19.303	10.877
10	21.640	13.113	21.640	13.176	21.637	12.105	21.638	12.108

Notes:

1. G denotes the soil shear modules
2. Results without torsion and with torsion shown above were analyzed using STARDYNE. Figures 3.7-269 and 3.7-270 present accordingly these two 3-D mathematical models.

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TABLE 3.7-50

WATERFORD NO. 3 COMPARISON OF ACCELERATION OF DYNAMIC ANALYSIS

WITH AND WITHOUT TORSIONAL DEGREE OF FREEDOM

SOIL SHEAR MODULUS G = 6400 PSI, SSE, SPECTRUM METHOD, 5% DAMPING

(Unit of Acceleration g or 32.2 ft/sec^2)

	E - W DIRECTION			N - S DIRECTION		
	STARDYNE - Version 3			STARDYNE Version 3		
MASS NO.	CASE -I*	CASE - II**	DIFF %	CASE I*	CASE II**	DIFF %
1	0.278	0.272	-2.2	0.231	0.231	0
2	0.257	0.251	-2.3	0.218	0.217	-0.5
3	2.240	0.234	-2.5	0.208	0.207	-0.5
4	0.226	0.220	-2.7	0.199	0.198	-0.5
5	0.211	0.205	-2.8	0.190	0.189	-0.5
6	0.194	0.188	-3.1	0.179	0.179	0
7	0.178	0.172	-3.4	0.169	0.169	0
8	0.167	0.161	-3.6	0.162	0.161	-0.6
9	0.156	0.150	-3.9	0.154	0.154	0
10	0.148	0.143	-3.4	0.149	0.148	-0.7
11	0.141	0.137	-2.8	0.144	0.144	0
12	0.234	0.228	-2.6	0.200	0.200	0
13	0.223	0.217	-2.7	0.194	0.193	-0.5
14	0.211	0.205	-2.8	0.187	0.186	-0.5
15	0.200	0.195	-2.5	0.180	0.180	0
16	0.190	0.184	-3.2	0.174	0.174	0
17	0.180	0.174	-3.3	0.168	0.168	0
18	0.170	0.165	-2.9	0.162	0.162	0
19	0.161	0.156	-3.1	0.157	0.156	0.6
20	0.153	0.148	-3.3	0.152	0.151	0.7
21	0.146	0.141	-3.4	0.147	0.146	0.7

* Without torsional degree of freedom

** With torsional degree of freedom

TABLE 3.7-50 (Cont'd)

	E - W DIRECTION STARDYNE - Version 3			N - S DIRECTION STARDYNE Version 3		
MASS NO.	CASE -I*	CASE - II**	DIFF %	CASE I*	CASE II**	DIFF %
22	0.167	0.161	-3.6	0.160	0.159	-0.6
23	0.164	0.159	-3.1	0.158	0.158	0
24	0.162	0.156	-3.7	0.156	0.156	0
25	0.157	0.152	-3.2	0.154	0.154	0
26	0.153	0.148	-3.3	0.152	0.152	-0.7
27	0.148	0.143	-3.4	0.148	0.148	0
28	0.145	0.140	-3.5	0.146	0.146	0
29	0.179	0.164	-8.4	0.169	0.169	0
30	0.161	0.147	-8.7	0.158	0.157	-0.6
31	0.153	0.154	+0.7	0.152	0.151	-0.7
32	0.145	0.149	+2.8	0.147	0.148	+0.7
35	0.171	0.180	+5.3	0.164	0.164	0
36	0.163	0.170	+4.3	0.158	0.159	+0.6
39	0.137	0.135	-1.5	0.141	0.140	-0.7

* Without torsional degree of freedom
** With torsional degree of freedom

TABLE 3.7-51

WATERFORD NO.3 COMPARISON OF ACCELERATION OF DYNAMIC ANALYSIS

WITH AND WITHOUT TORSIONAL DEGREE OF FREEDOM

SOIL SHEAR MODULUS G = 16050 PSI, SSE, SPECTRUM METHOD, 5% DAMPING

	E - W DIRECTION			N - S DIRECTION		
	STARDYNE - Version 3			STARDYNE Version 3		
MASS NO.	CASE -I*	CASE - II**	DIFF %	CASE I*	CASE II**	DIFF %
1	0.492	0.479	-2.6	0.430	0.429	-0.2
2	0.453	0.440	-2.9	0.401	0.399	-0.5
3	0.423	0.411	-2.8	0.379	0.377	-0.5
4	0.395	0.384	-2.8	0.358	0.357	-0.3
5	0.367	0.356	-3.0	0.337	0.336	-0.3
6	0.333	0.322	-3.3	0.311	0.310	-0.3
7	0.299	0.290	-3.0	0.286	0.285	-0.3
8	0.275	0.266	-3.3	0.268	0.266	-0.8
9	0.250	0.242	-3.2	0.249	0.248	-0.4
10	0.232	0.224	-3.5	0.234	0.233	-0.4
11	0.216	0.209	-3.2	0.222	0.221	-0.5
12	0.356	0.344	-3.4	0.312	0.311	-0.3
13	0.341	0.330	-3.2	0.303	0.301	-0.7
14	0.325	0.315	-3.1	0.293	0.292	-0.3
15	0.310	0.300	-3.2	0.284	0.282	-0.7
16	0.295	0.286	-3.1	0.274	0.273	-0.4
17	0.280	0.271	-3.2	0.265	0.264	-0.4
18	0.265	0.257	-3.0	0.255	0.254	-0.4
19	0.251	0.243	-3.2	0.245	0.244	-0.4
20	0.237	0.229	-3.4	0.236	0.235	-0.4
21	0.223	0.217	-2.7	0.226	0.226	0

* Without torsional degree of freedom

** With torsional degree of freedom

TABLE 3.7-51 (Cont'd)

	E - W DIRECTION STARDYNE - Version 3			N - S DIRECTION STARDYNE Version 3		
MASS NO.	CASE -I*	CASE - II**	DIFF %	CASE I*	CASE II**	DIFF %
22	0.259	0.250	-3.5	0.248	0.247	-0.4
23	0.255	0.246	-3.5	0.246	0.245	-0.4
24	0.251	0.243	-3.2	0.243	0.242	-0.4
25	0.244	0.236	-3.3	0.239	0.238	-0.4
26	0.236	0.229	-3.0	0.235	0.234	-0.4
27	0.228	0.220	-3.5	0.229	0.228	-0.4
28	0.221	0.214	-3.2	0.225	0.224	-0.4
29	0.277	0.254	-8.3	0.268	0.267	-0.4
30	0.251	0.229	-8.8	0.248	0.247	-0.4
31	0.238	0.239	+0.4	0.238	0.236	-0.8
32	0.224	0.228	+1.8	0.228	0.229	+0.4
35	0.268	0.282	+5.2	0.259	0.262	+1.2
36	0.254	0.267	+5.1	0.250	0.252	+0.8
39	0.206	0.203	-1.5	0.215	0.213	-0.9

* Without torsional degree of freedom
** With torsional degree of freedom

TABLE 3.7-52

COMPARISON OF COMBINATION METHODS FOR RELATIVE SEISMIC DISPLACEMENTS

VERTICAL SSE

MAXIMUM ABSOLUTE SEISMIC DISPLACEMENT (ft)				RELATIVE SEISMIC DISPLACEMENT (ft)		
Reactor Containment Building		Reactor Auxiliary				
MASS #	Δ_{XA}	MASS #	Δ_{XB}	Δ_{ABS}	Δ_{SRSS}	Δ_{ABSR}
6 (top of PCV)	0.03067	1 (top)	0.02145	0.05212	0.03743	0.00996
16 (Mat)	0.03040	23 (Mat)	0.02108	0.05148	0.03699	0.00932

N-S SSE MAXIMUM ABSOLUTE SEISMIC DISPLACEMENT (ft)				RELATIVE SEISMIC DISPLACEMENT (ft)		
Reactor Containment Building						
MASS #	Δ_{XA}	MASS #	Δ_{XB}	Δ_{ABS}	Δ_{SRSS}	Δ_{ABSR}
I9 (EL. 44'-0")	0.04066	3 (EL. 42'-5")	0.0402	0.0808	0.05717	0.03646
25 (Mat)	0.02783	5 (Mat)	0.0222	0.05003	0.03560	0.00563

ABS = $|\Delta_{XA}| + |\Delta_{XB}|$ (1)

SRSS = $(\Delta_{XA}^2 + \Delta_{XB}^2)1/2$ (2)

ABSR = $|(\Delta_{XA}) - (\Delta_{Amat})| + |(\Delta_{XB} - \Delta_{Bmat})| + |(\Delta_{Amat} - \Delta_{Bmat})|$ (3)

Where:

Δ_{XA}: Maximum Absolute Seismic displacement of building A at restraint location.

Δ_{XB}: Maximum Absolute Seismic displacement of building B at restraint location.

Δ_{mat}: Maximum Absolute Seismic displacement of the mat.

TABLE 3.7-53

COMPARISON OF TIME HISTORY DISPLACEMENTS TO DESIGN VALUES

Direction	Max ABS Sum	Maximum Difference	Design Value
	Displacement between RB El 28'-0" & RAB El 18'-6"		
N-S	0.0484'	0.0462'	0.0515'
E-W	0.0462'	0.0426"	0.0522'
Vert	0.0371'	0.0286'	0.0372'
	Displacement between RB El 48'-0' & RAB El 42'-6"		
N-S	0.0615'	0.0593'	0.0649'
E-W	0.0591'	0.0531'	0.0625'

NOTE:

Maximum absolute summation is the maximum value found by adding the absolute value of the displacement of one building to the absolute value of the displacement of the other building both taken at the same time over the duration of the seismic event.

Maximum difference is the maximum value found by algebraic subtraction of the displacement of one building from the displacement of the other building both taken at the same time over the duration of the seismic event.

TABLE 3.7-54

EFFECT OF T VARIATION ON COMPARISON OF TIME HISTORY DISPLACEMENTS TO DESIGN VALUES

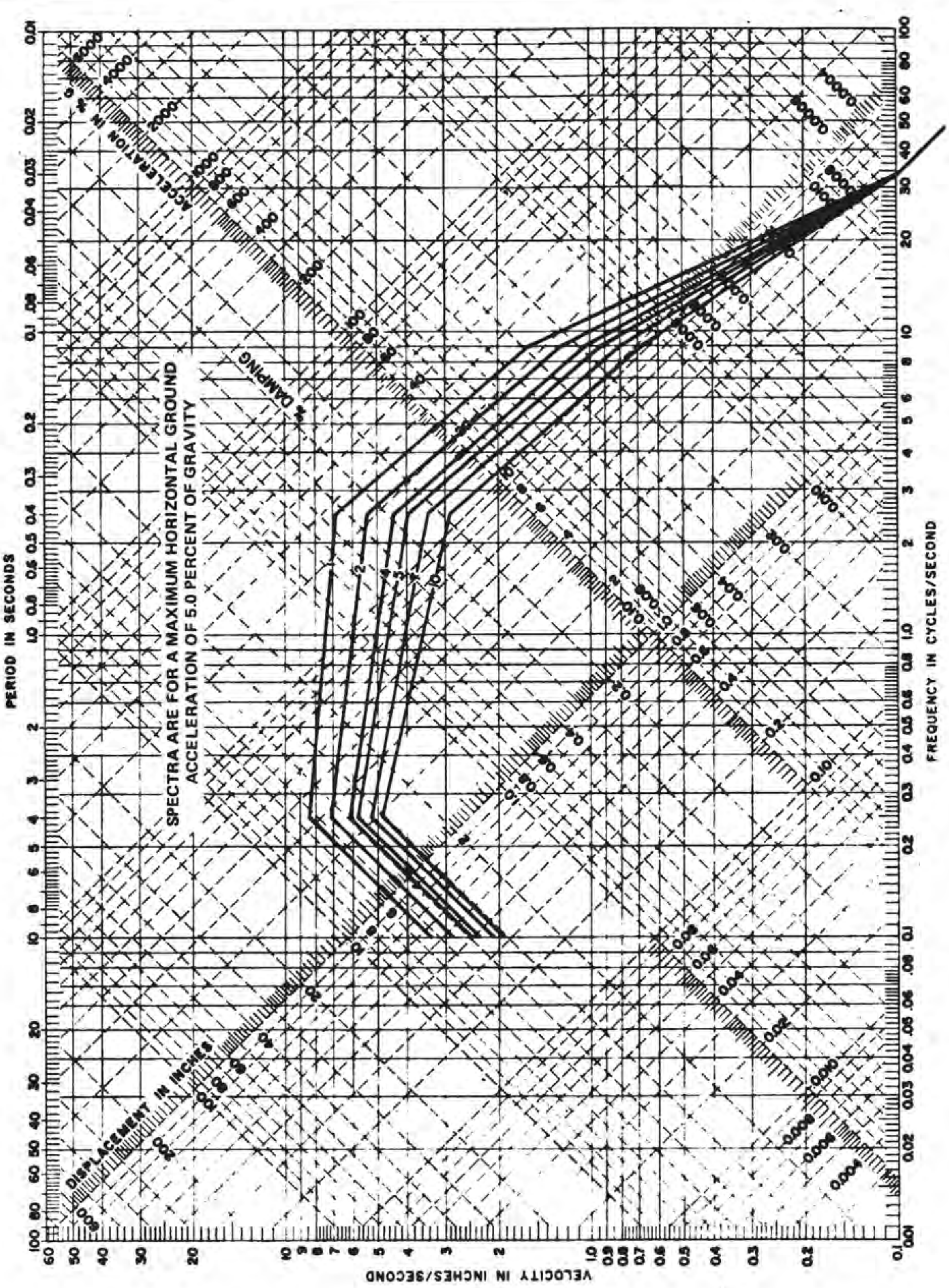
	ΔT	ABS Sum (ft)	Max. Diff. (ft)	Design Values
N-S	.004	.0457	.0391	.0515'
Displacement betw RB El 28'-0"	.005	.0484	.0462	.0515'
& RAB El 18'-0"	.006	.0567	.0414	.0515'
Vert	.004	.0357	.0309	.0372'
Disp betw RB El 28'-0" &	.005	.0371	.0286	.0372'
RAB El 18'-6"	.006	.0431	.0364	.0372'
E-W	.004	.0410	.0392	.0522'
Disp betw RB El 28'-0" &	.005	.0462	.0426	.0522'
RAB El 18'-6"	.006	.0590	.0394	.0522'
N-S	.004	.0551	.0496	.0649'
Disp betw RB El 48'-0" &	.005	.0615	.0593	.0649'
RAB El 42'-6"	.006	.0713	.0528	.0649'
E-W	.004	.04806	.04806	.0625'
Disp betw RB El 48'-0" &	.005	.0591	.0531	.0625'
RAB El 42'-6"	.006	.0728	.0502	.0625'

TABLE 3.7-55

SEISMIC DISPLACEMENT BETWEEN BUILDINGS

*Penetration#	Elevation	Pipe Size	System	Reason For Not Analyzing Penetration
10	54'-0	48"	Main Containment Purge	Flexible Connection in ductwork in RAB immediately downstream containment isolation valve
11	33'-0	48"	Main Containment Purge	HVAC duct ends approx 5 ft. into RAB HVAC duct ends approx 2 ft. into RB
12B	34'-0	-	Spare	Penetrates concrete shield wall only
25	27'-6	36"	Fuel Transfer Tube	No piping connections on either FHB side or containment side
45,53	25'-0/28'-0	3/8"	Instrument tubing	Is analyzed for ABS SUM displacement
55	25'-0	-	Spare	No piping
56	26'-0	8"	HVAC Mini Purge Inlet	HVAC duct ends approx. 5 ft. into RAB HVAC duct ends approx 2 ft. into RB
57	51'-0	8"	HVAC Mini Purge Outlet	Flexible connection on ductwork in RAB immediately downstream containment isolation valve
58	51'-0	-	Spare	Blind Flanged
59	56'-0	31"	Shield Bldg Vent System	Flexible connection on duct work in RAB immediately downstream containment isolation valve
60	56'-0	31"	Shield Bldg Vent System	Flexible connection on ductwork in RAB immediately downstream containment isolation valve
61	51'-0	-	Spare	Blind flanged
62	51'-0	3/4"	Instrument Air to Construction Hatch	Is analyzed for ABS SUM displacement
65	58'-0	-	Spare	Blind flanged
66	58'-0	-	Spare	Blind flanged
67		24"	Containment Vacuum Relief	Penetrates Steel Containment only
68		24"	Containment Vacuum Relief	Penetrates Steel Containment only

* Shown on drawing 2998-G-213 Sheets 3 and 4



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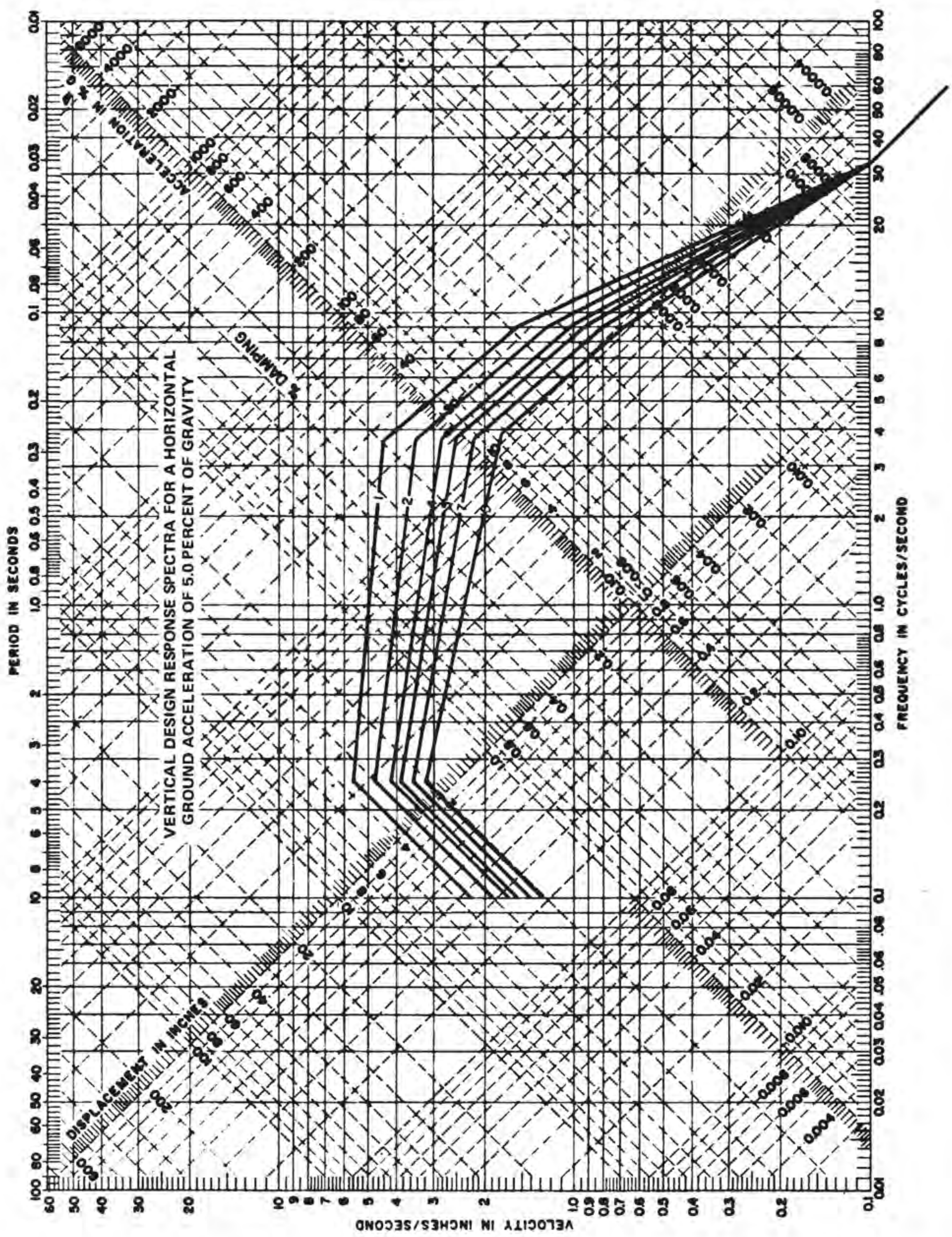
ST. LUCIE PLANT UNIT 2

RESPONSE SPECTRA OPERATING
BASIS EARTHQUAKE – HORIZONTAL

FIGURE 3.7-1



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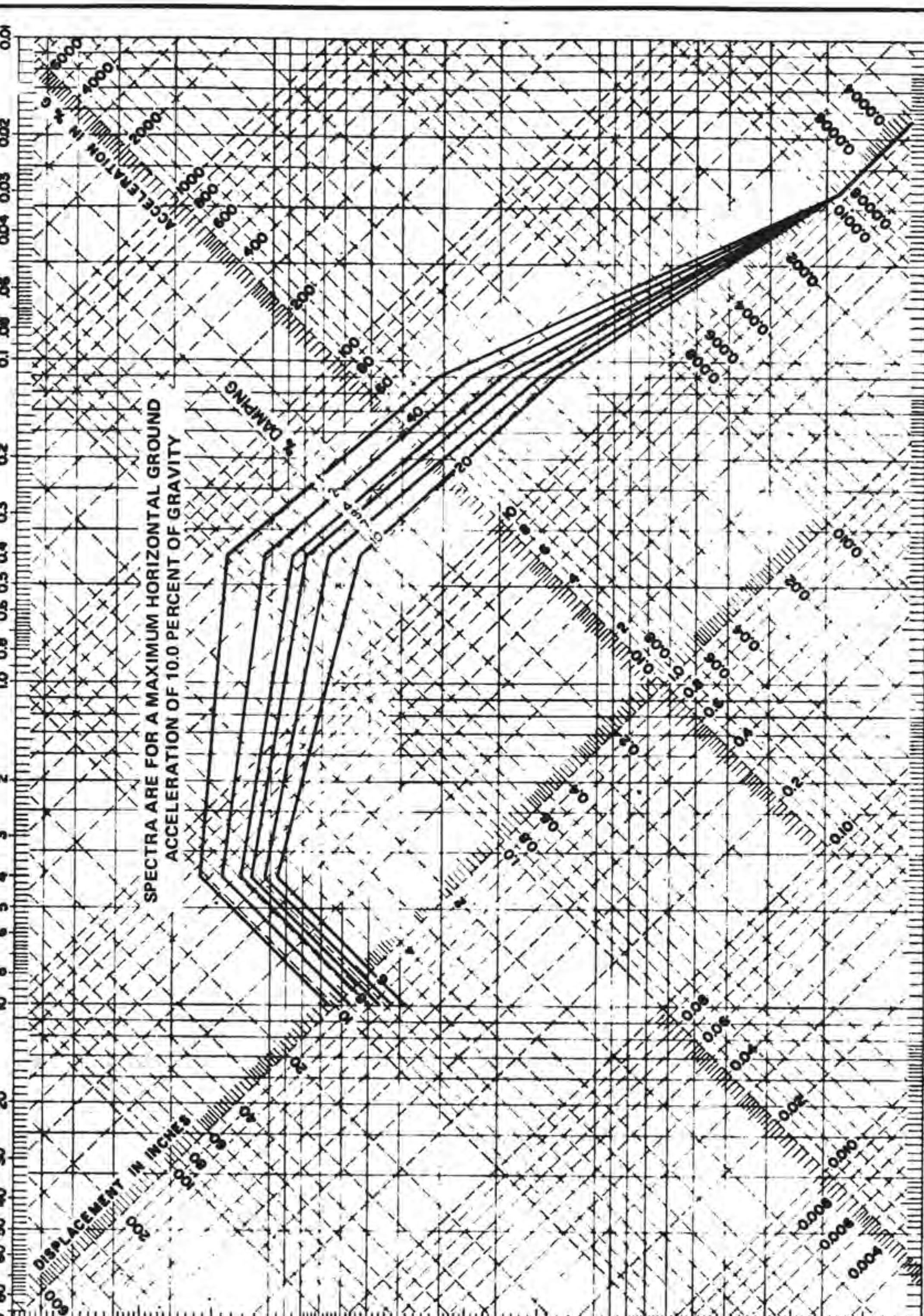
ST. LUCIE PLANT UNIT 2

RESPONSE SPECTRA OPERATING BASIS EARTHQUAKE – VERTICAL

FIGURE 3.7-2



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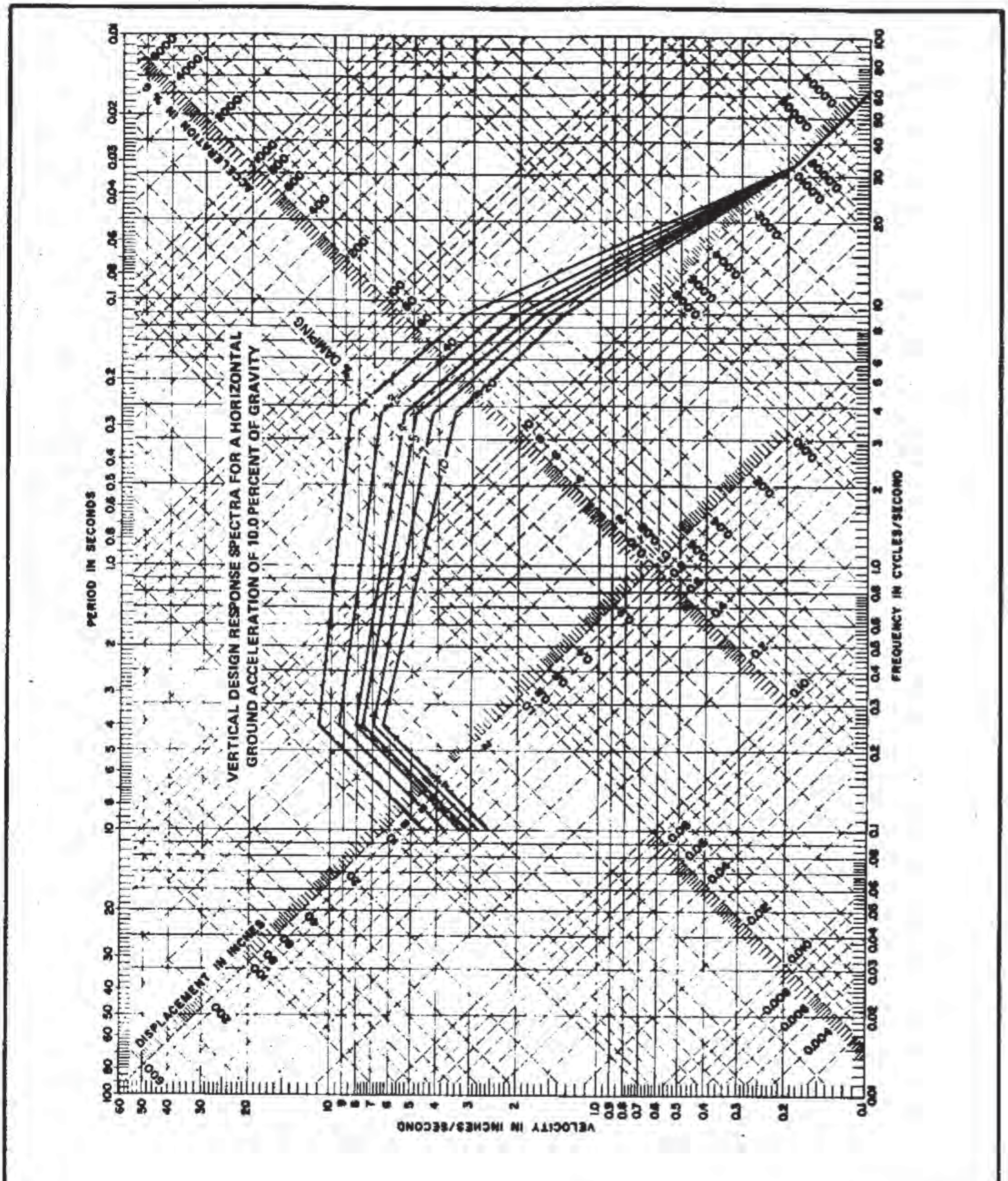
ST. LUCIE PLANT UNIT 2

RESPONSE SPECTRA SAFE
SHUTDOWN EARTHQUAKE – HORIZONTAL

FIGURE 3.7-3



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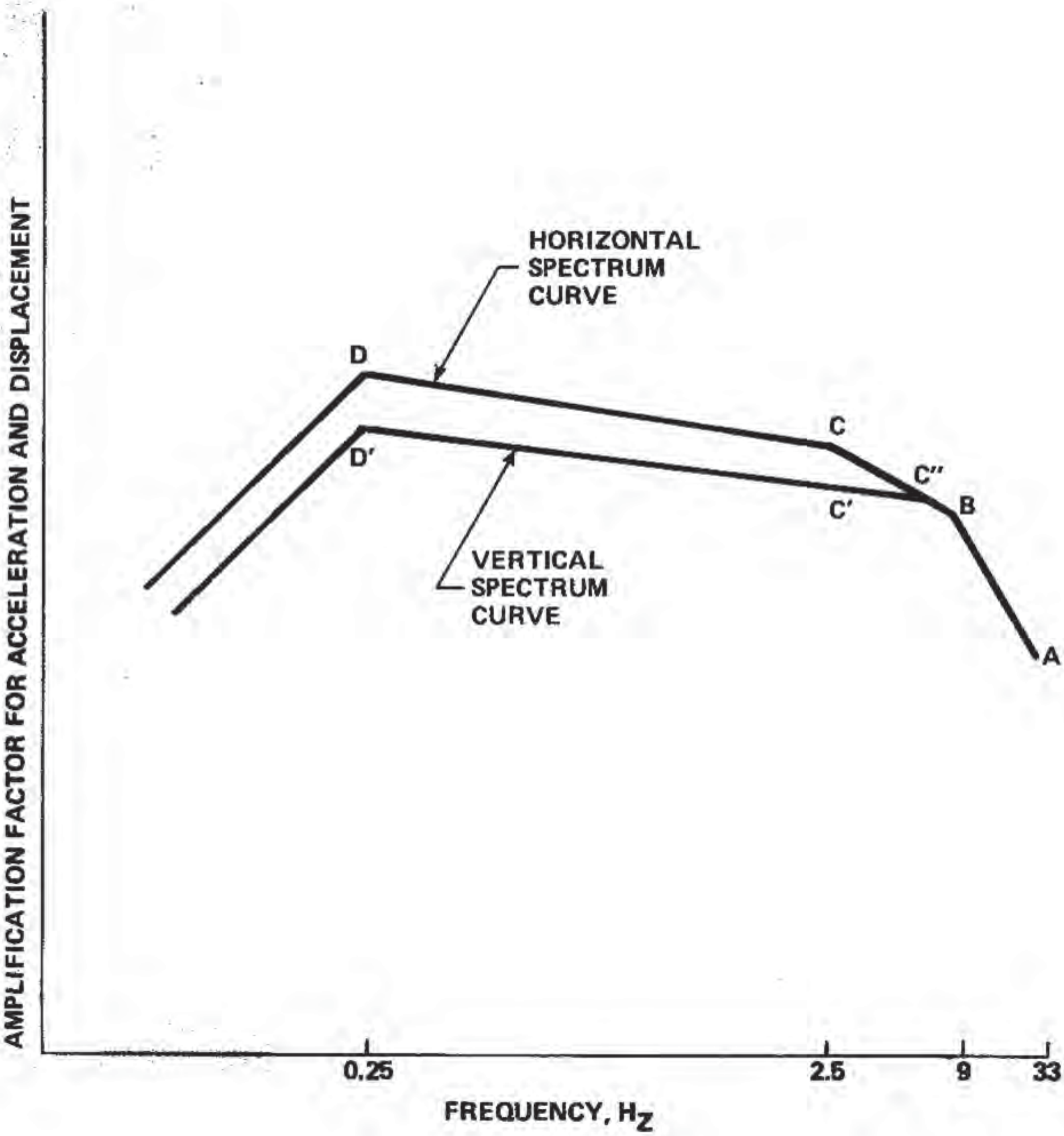
RESPONSE SPECTRA SAFE
SHUTDOWN EARTHQUAKE – VERTICAL

FIGURE 3.7-4

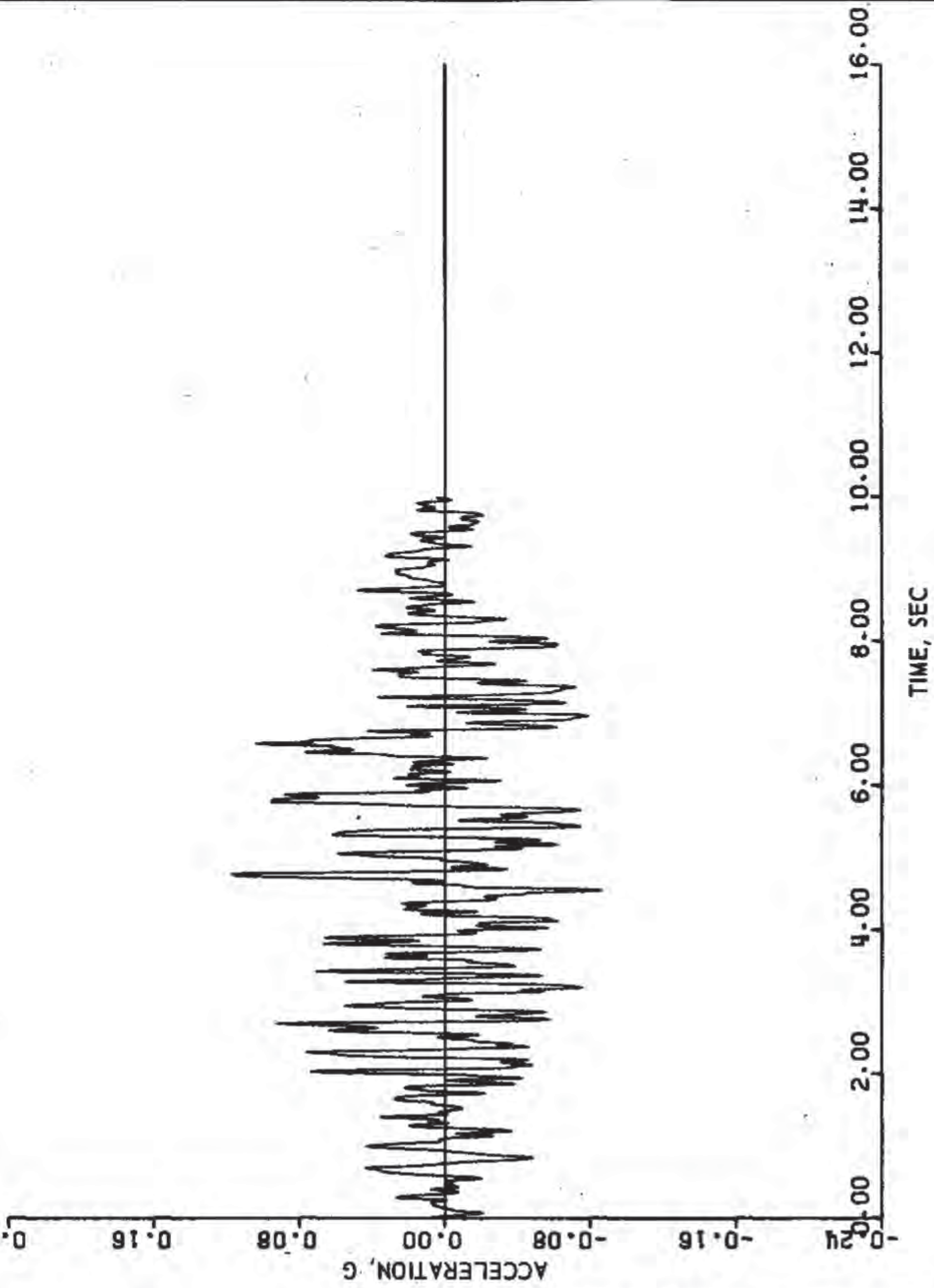


A) FOR HORIZONTAL SPECTRUM CURVE SEE TABLE 3.7-1

B) FOR VERTICAL SPECTRUM CURVE SEE SECTION 3.7.1.1 FOR DIRECTIONS TO OBTAIN VALUES.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

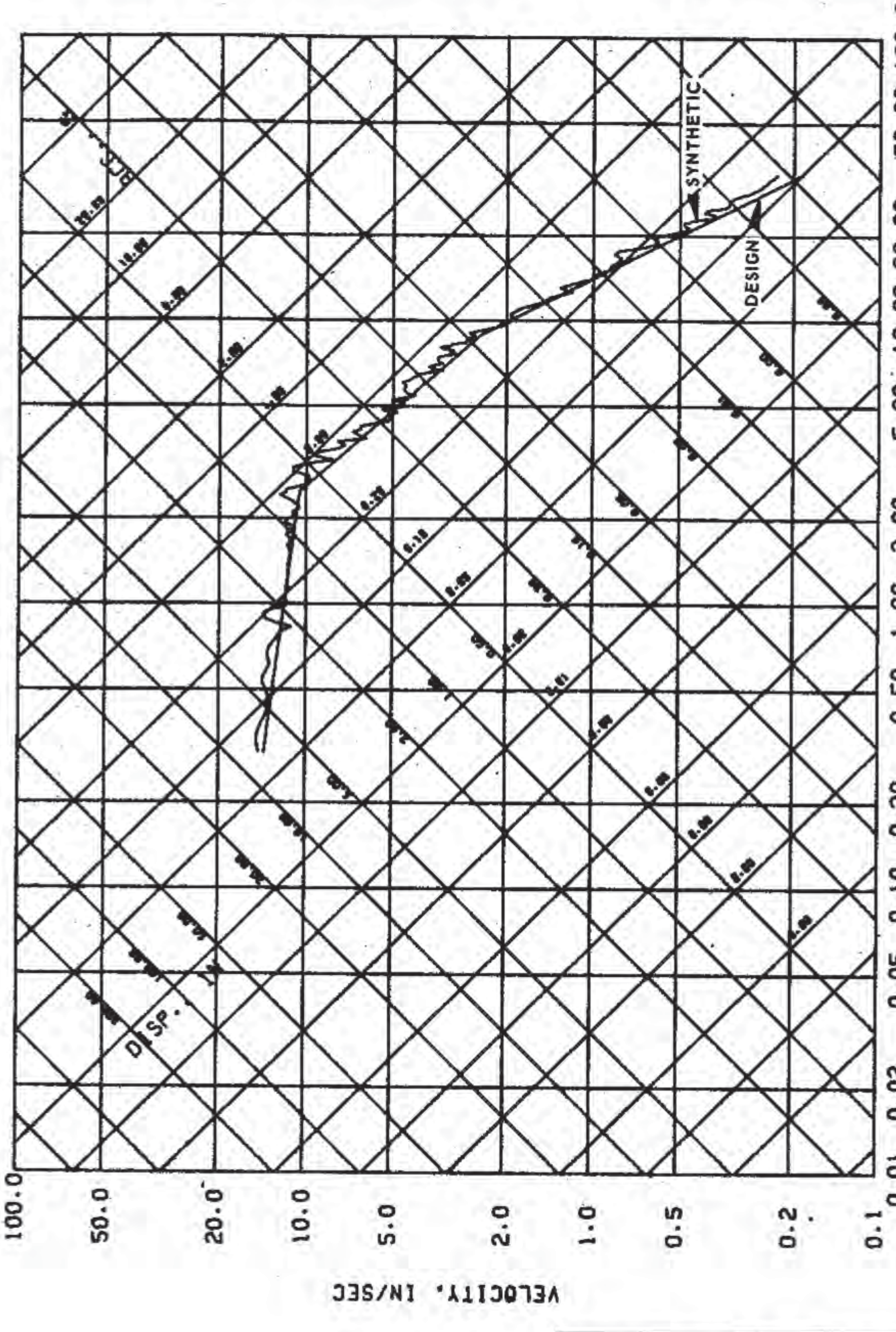
RESPONSE SPECTRA – HORIZONTAL & VERTICAL
FIGURE 3.7-5



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ARTIFICIAL SSE HORIZONTAL ACCELEROGRAM MAXIMUM GROUND ACCELERATION 0.10 G DURATION OF TIME 10 SECONDS

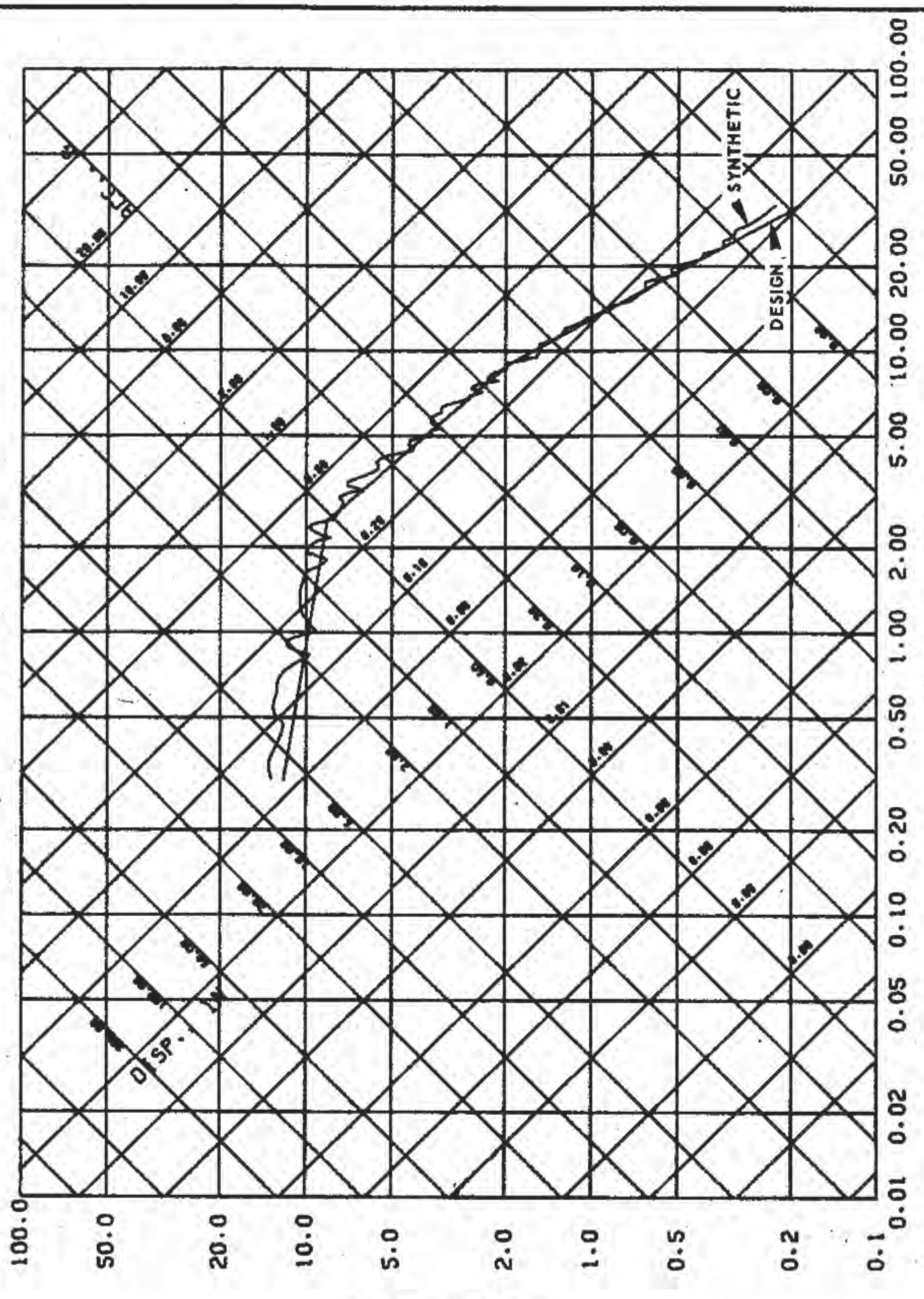
FIGURE 3.7-6



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
2 PERCENT DAMPING – SSE

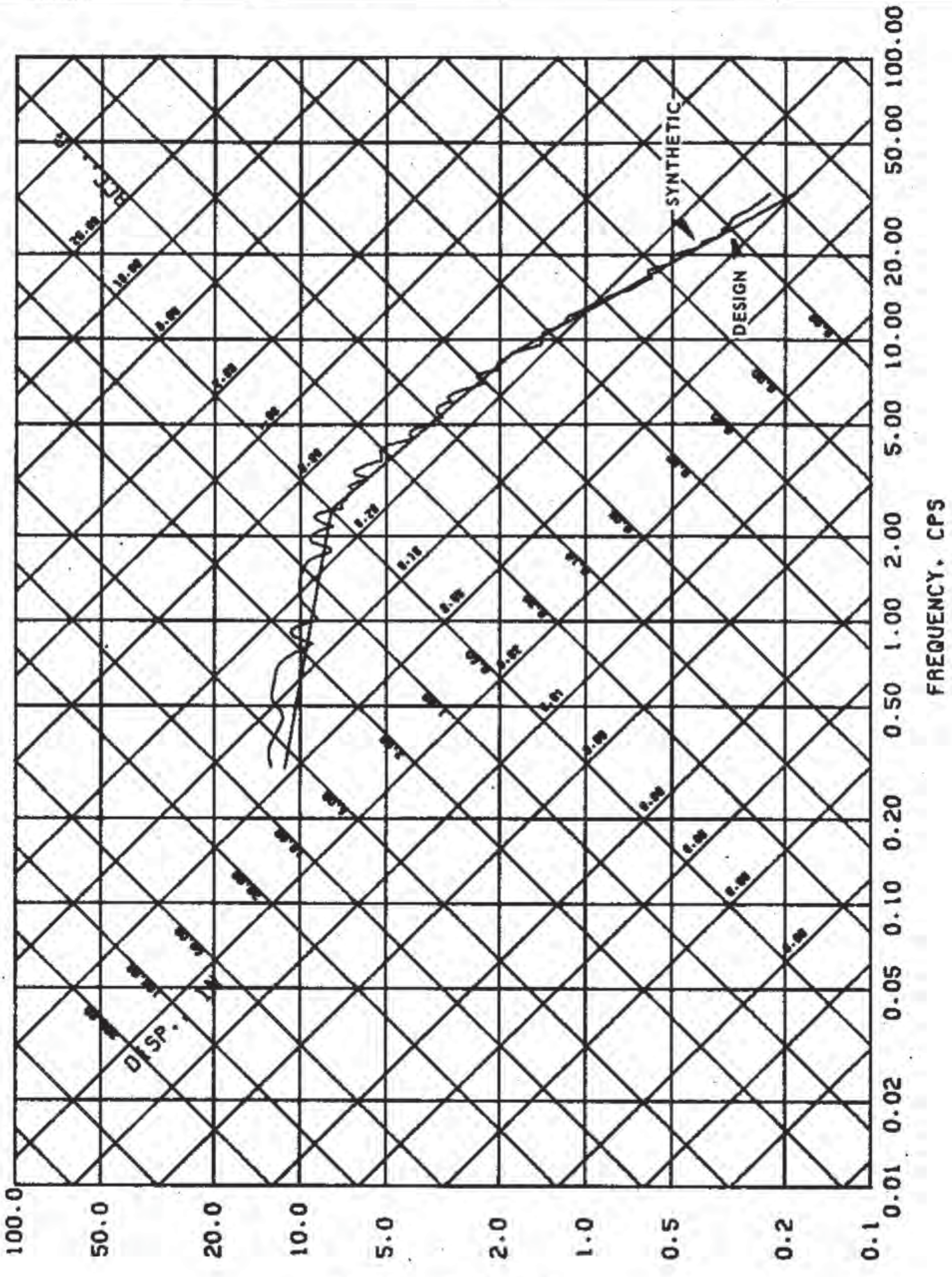
FIGURE 3.7-7



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
4 PERCENT DAMPING – SSE

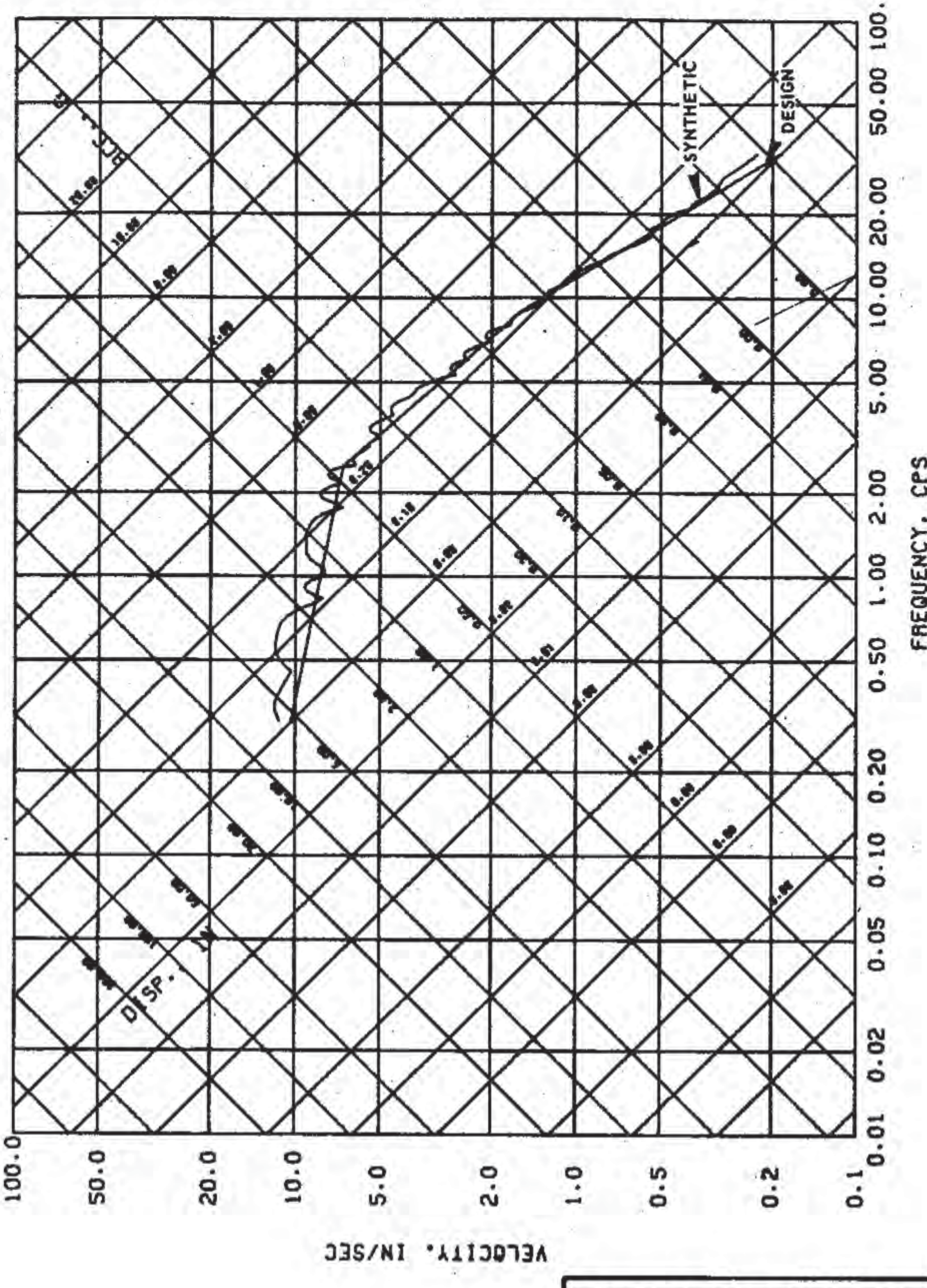
FIGURE 3.7-8



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
5 PERCENT DAMPING – SSE

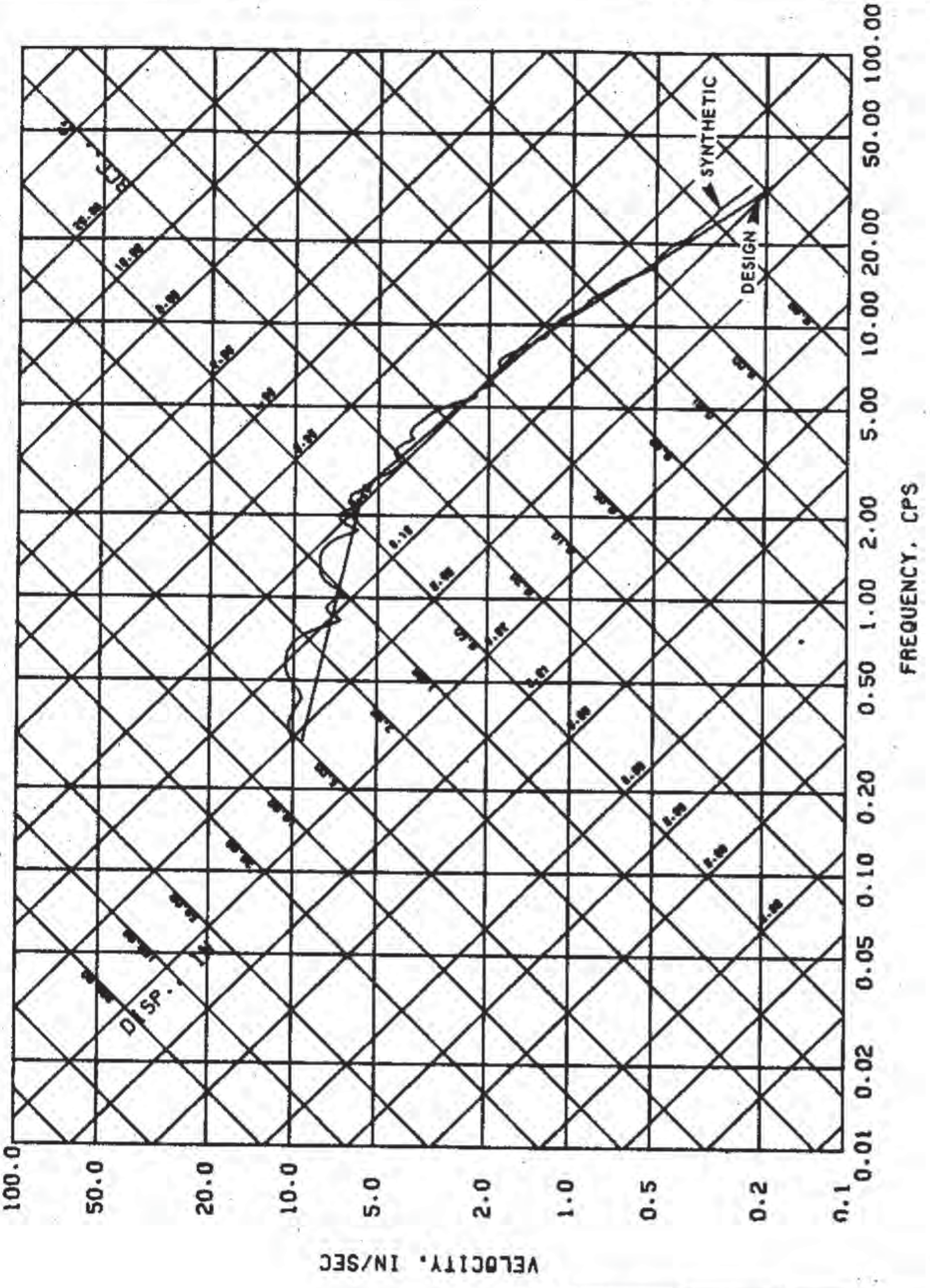
FIGURE 3.7-9



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
7 PERCENT DAMPING – SSE

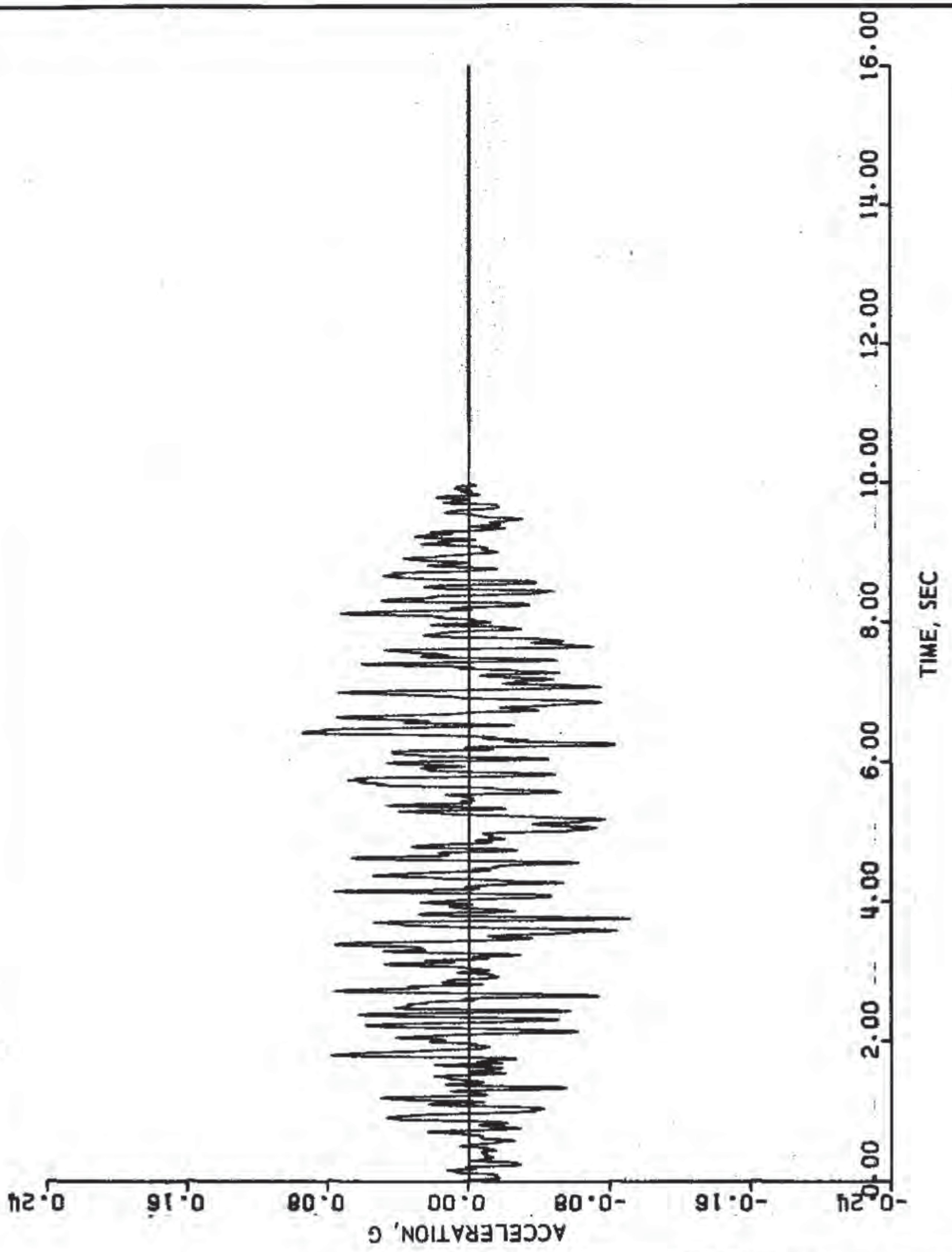
FIGURE 3.7-10



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
10 PERCENT DAMPING – SSE

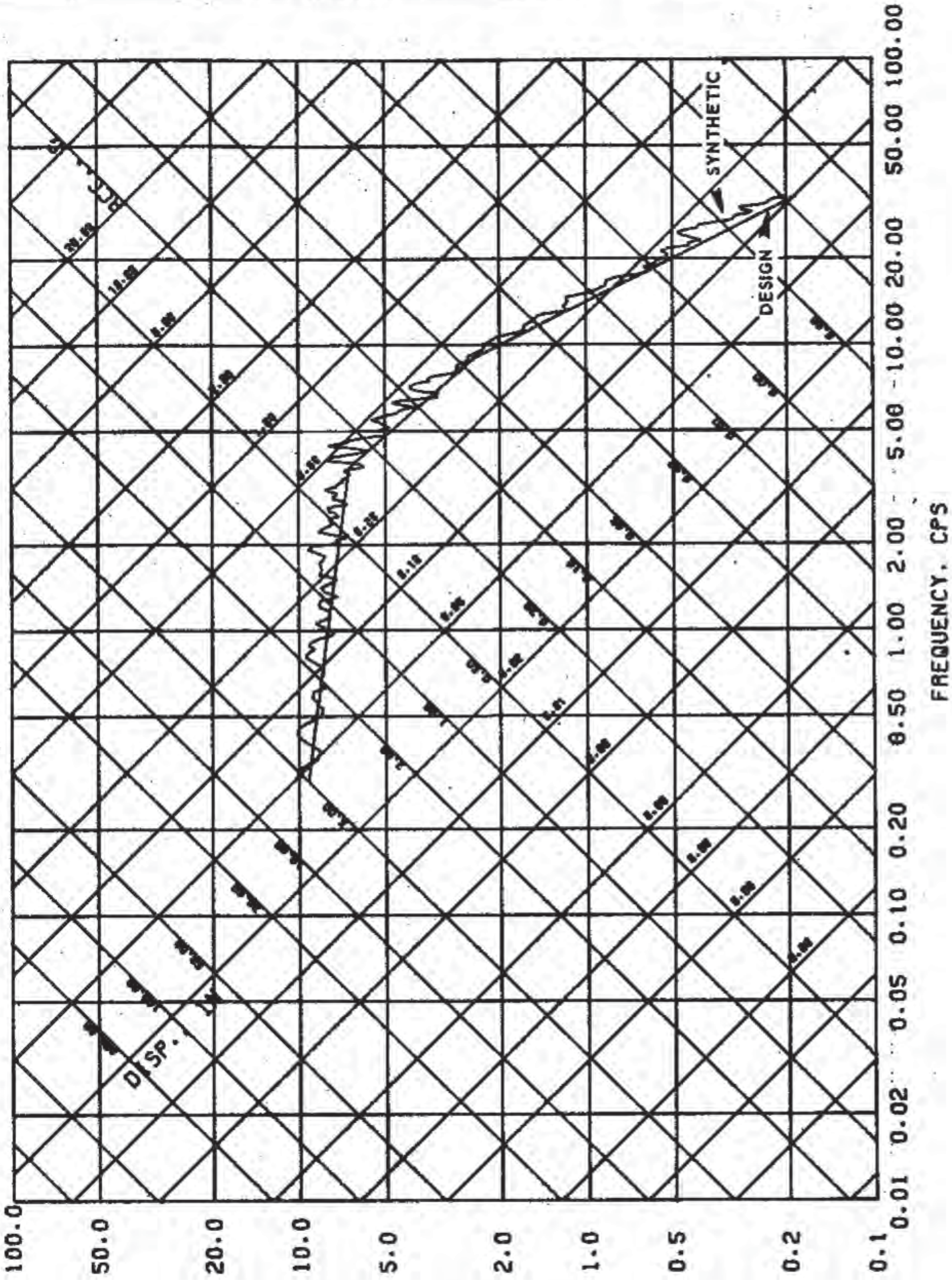
FIGURE 3.7-11



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ARTIFICAL SSE VERTICAL ACCELEROGRAM MAXIMUM GROUND ACCELERATION 0.10 G DURATION OF TIME 10 SECONDS

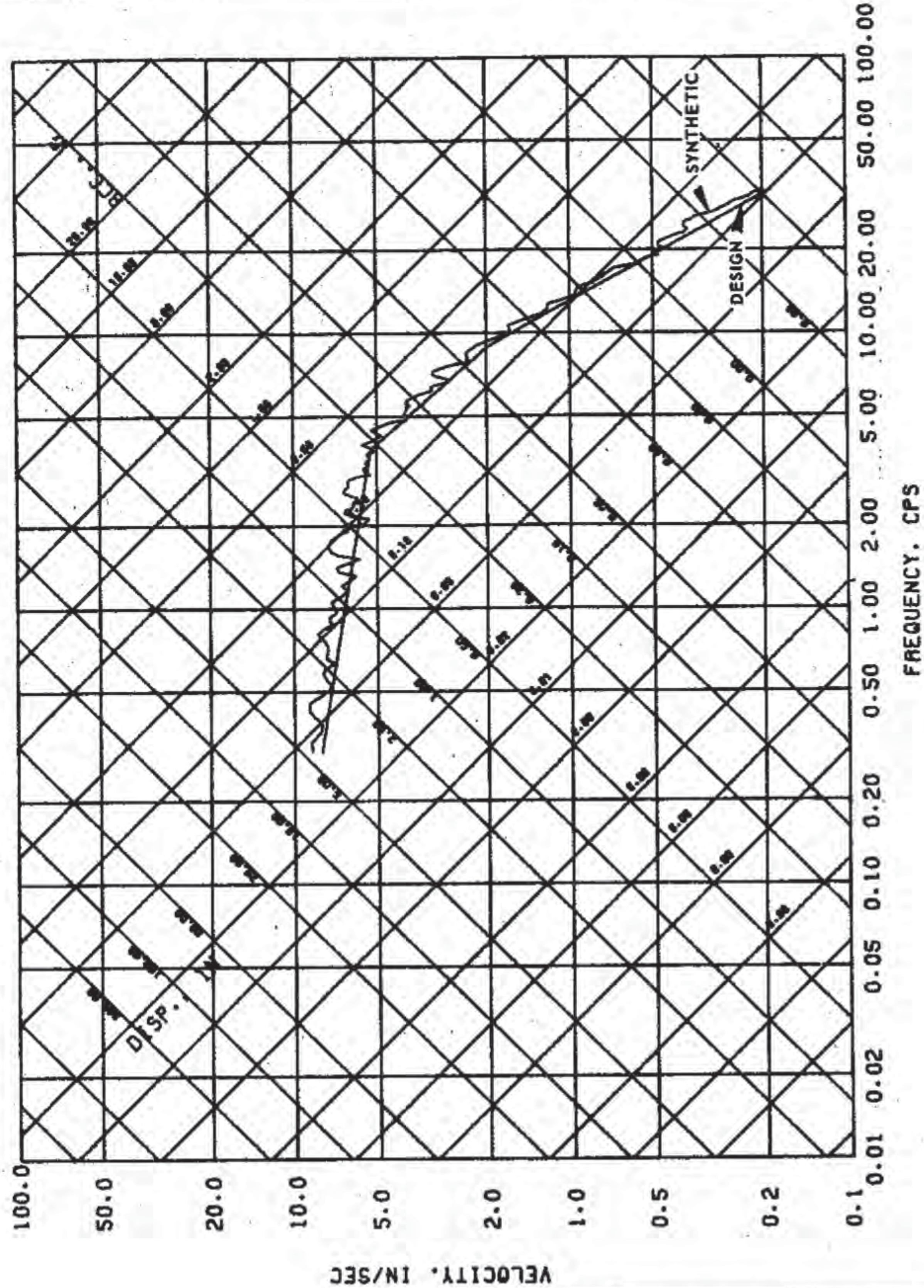
FIGURE 3.7-12



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
2 PERCENT DAMPING – SSE

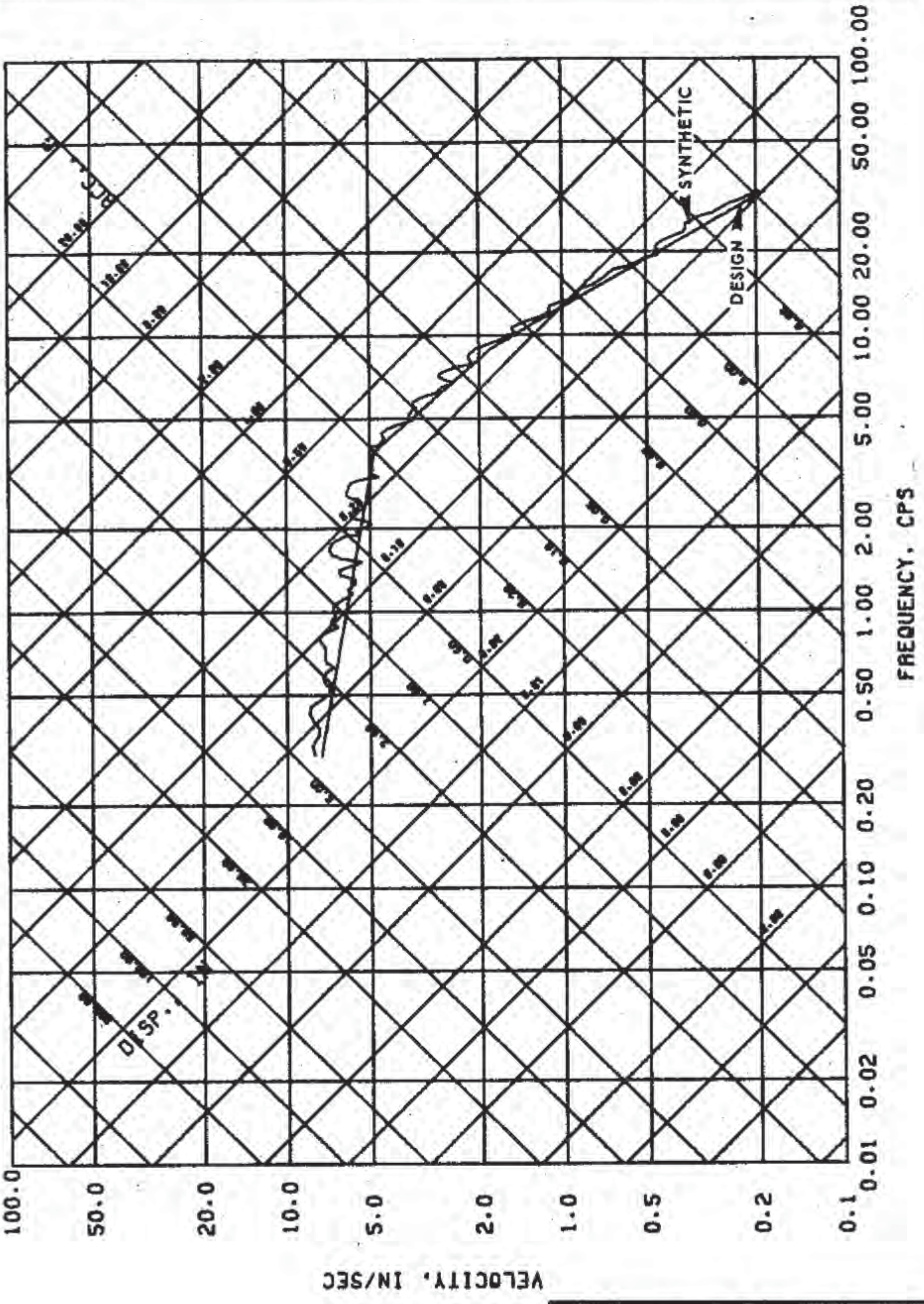
FIGURE 3.7-13



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
4 PERCENT DAMPING – SSE

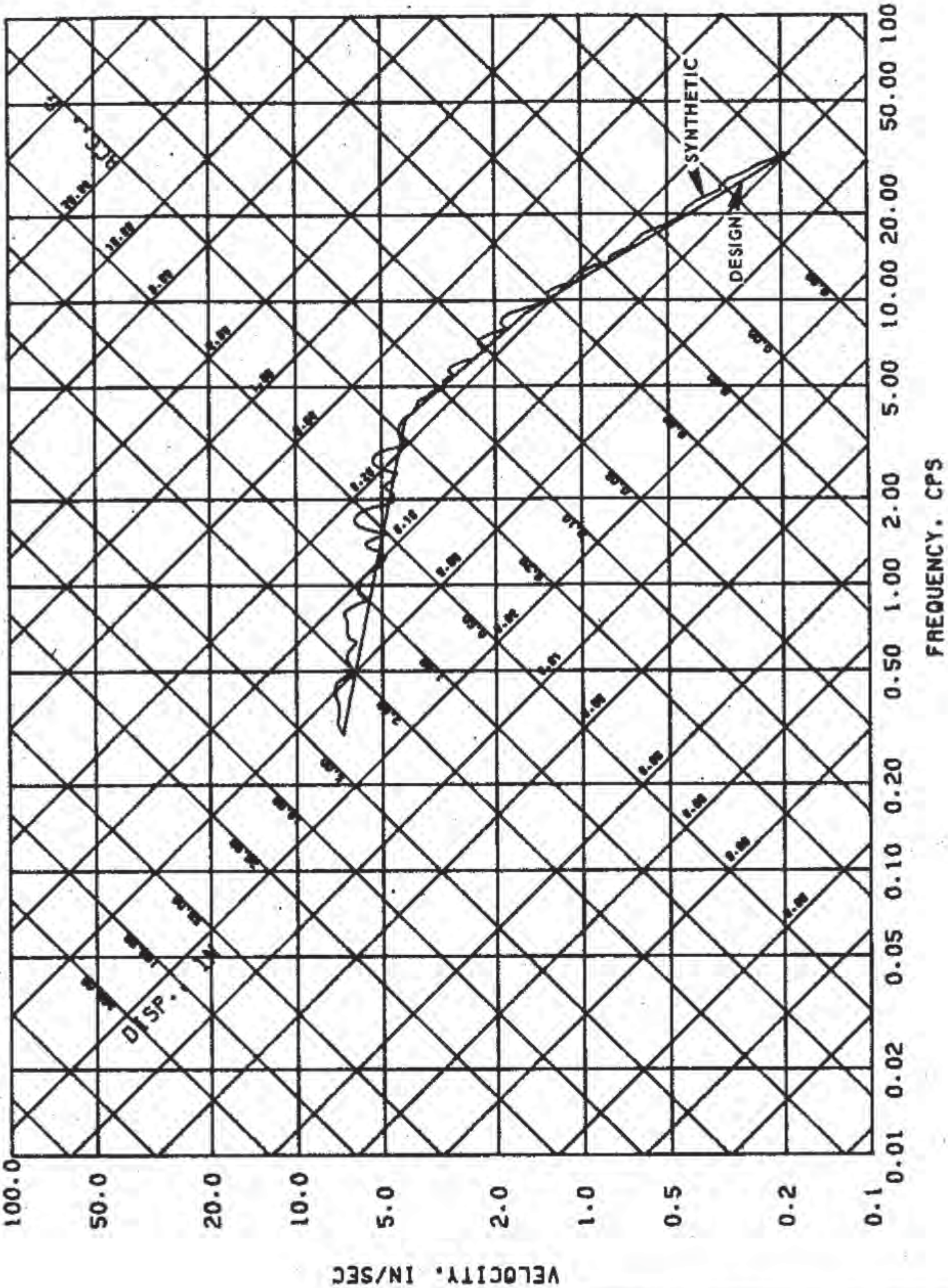
FIGURE 3.7-14



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
5 PERCENT DAMPING – SSE

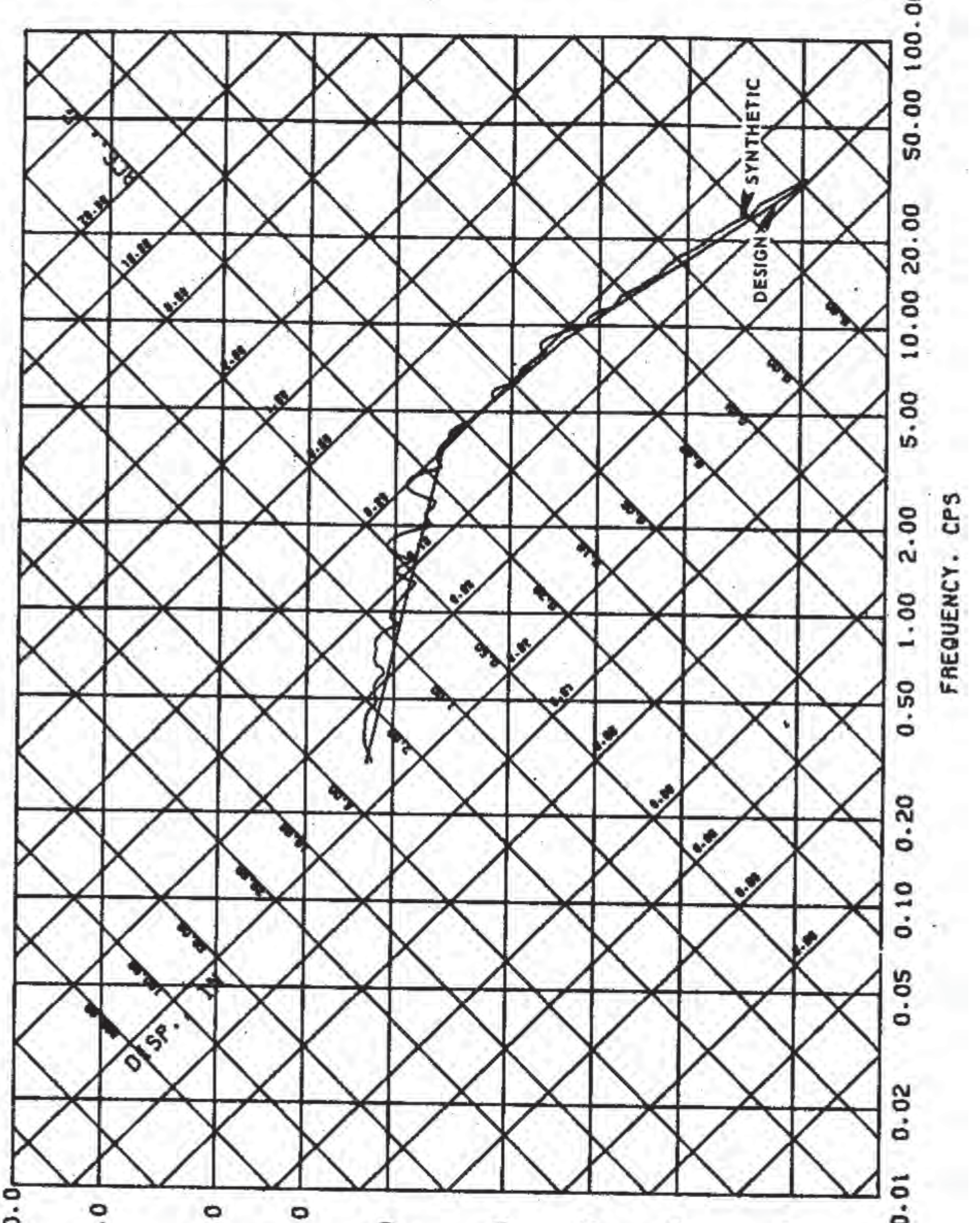
FIGURE 3.7-15



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
7 PERCENT DAMPING – SSE

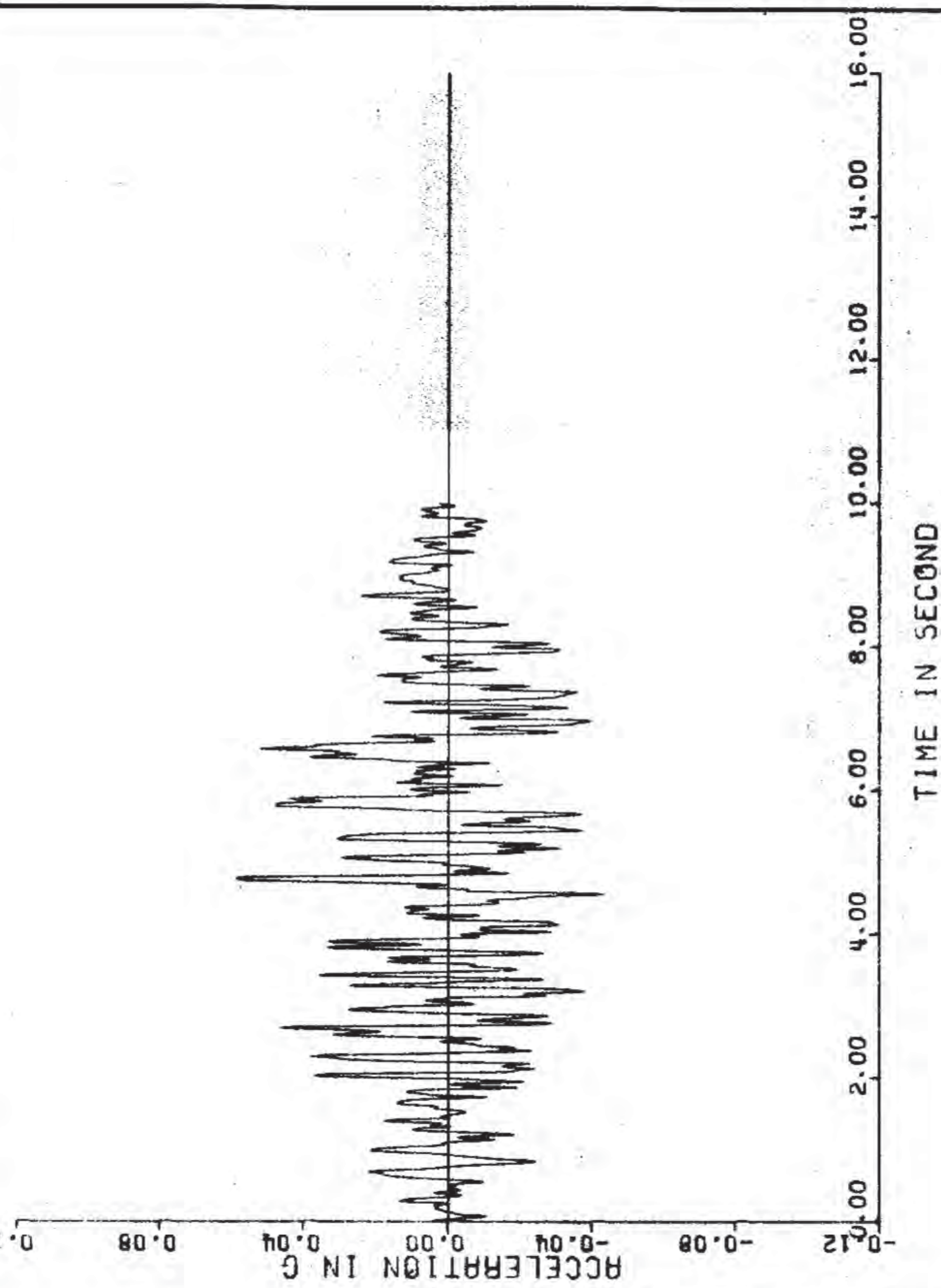
FIGURE 3.7-16



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
10 PERCENT DAMPING – SSE

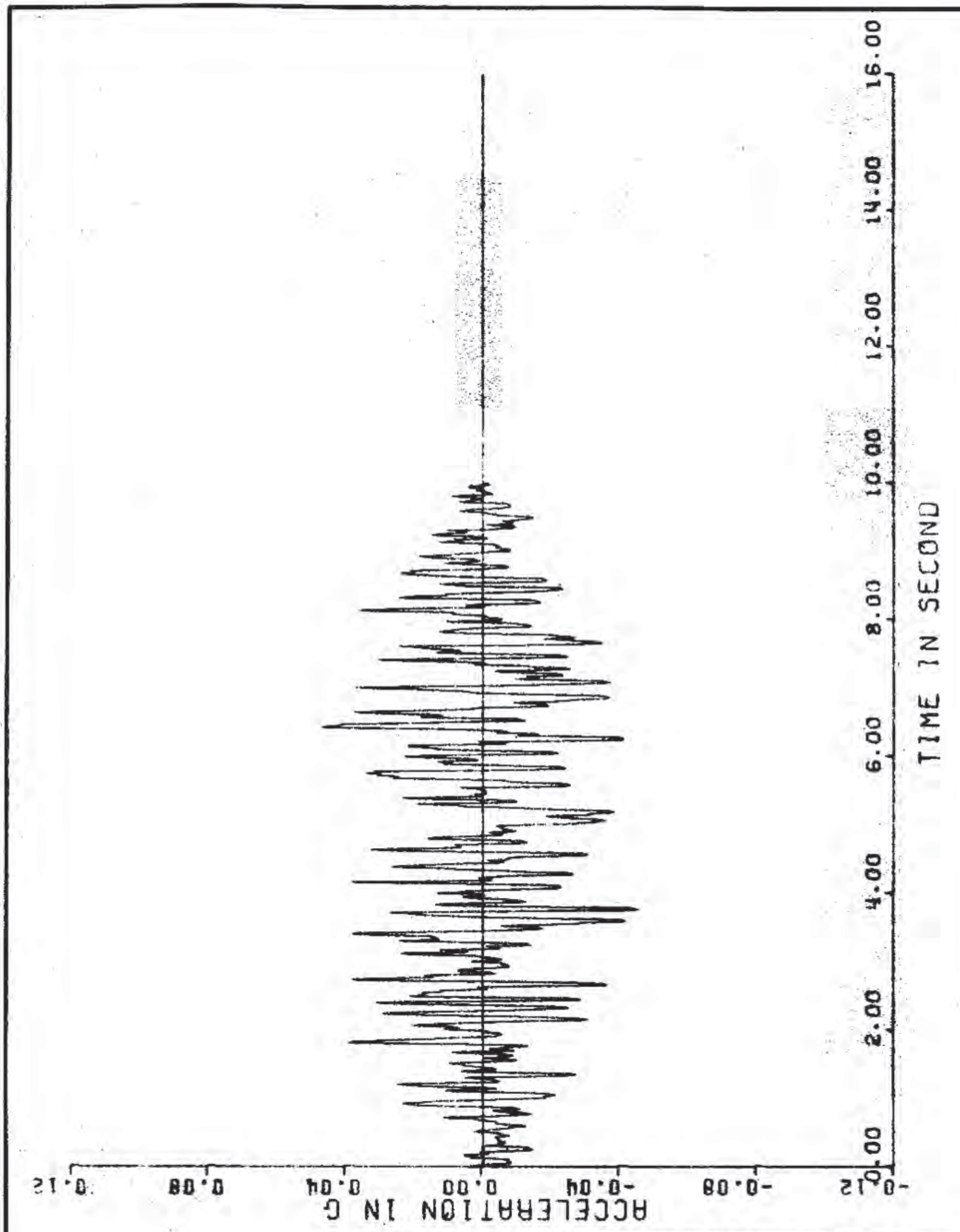
FIGURE 3.7-17



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ARTIFICAL OBE HORIZONTAL ACCELEROGRAM MAXIMUM GROUND ACCELERATION 0.05 G DURATION OF TIME 10 SECONDS

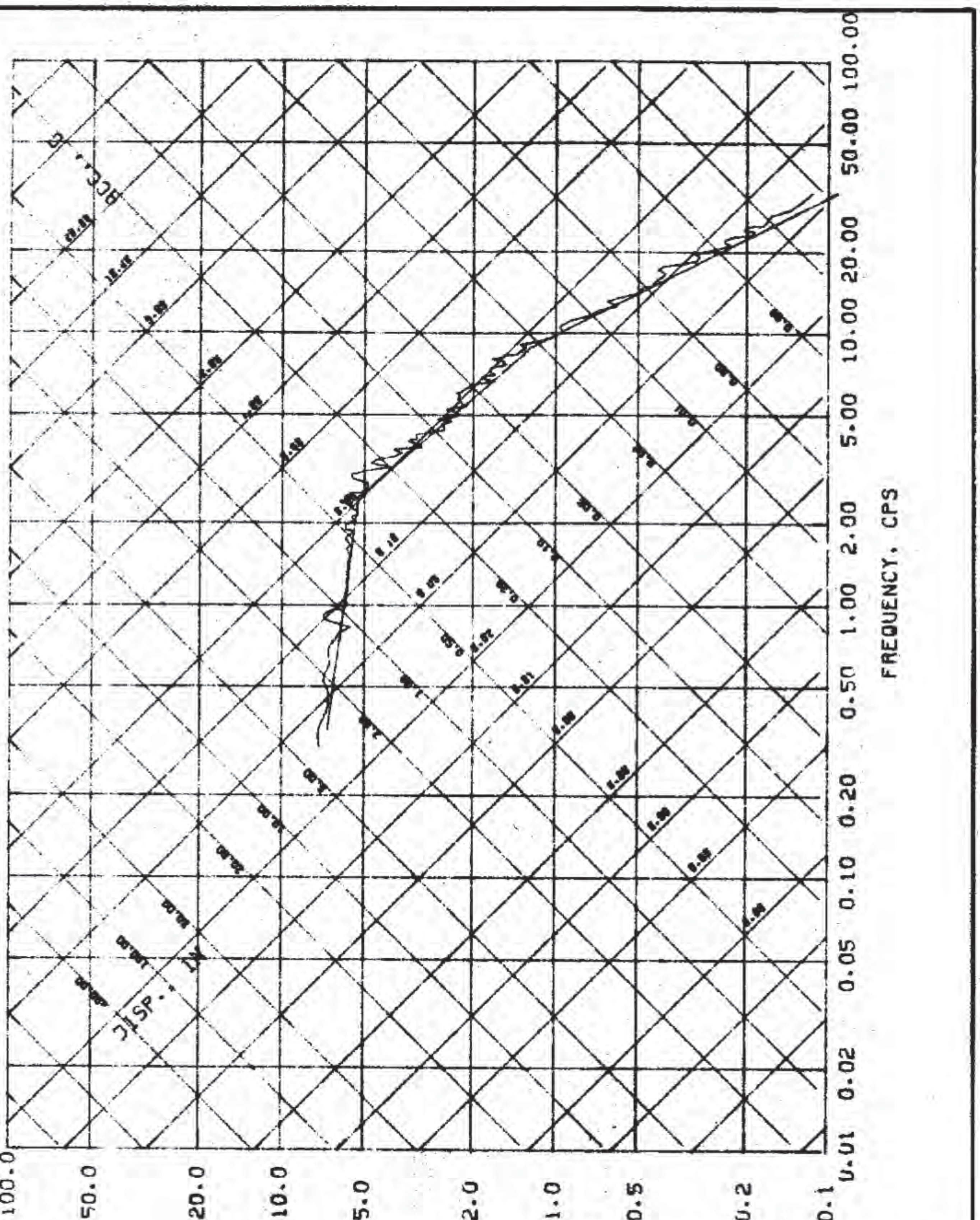
FIGURE 3.7-18



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ARTIFICAL OBE VERTICAL ACCELEROGRAM MAXIMUM GROUND ACCELERATION 0.05 G DURATION OF TIME 10 SECONDS

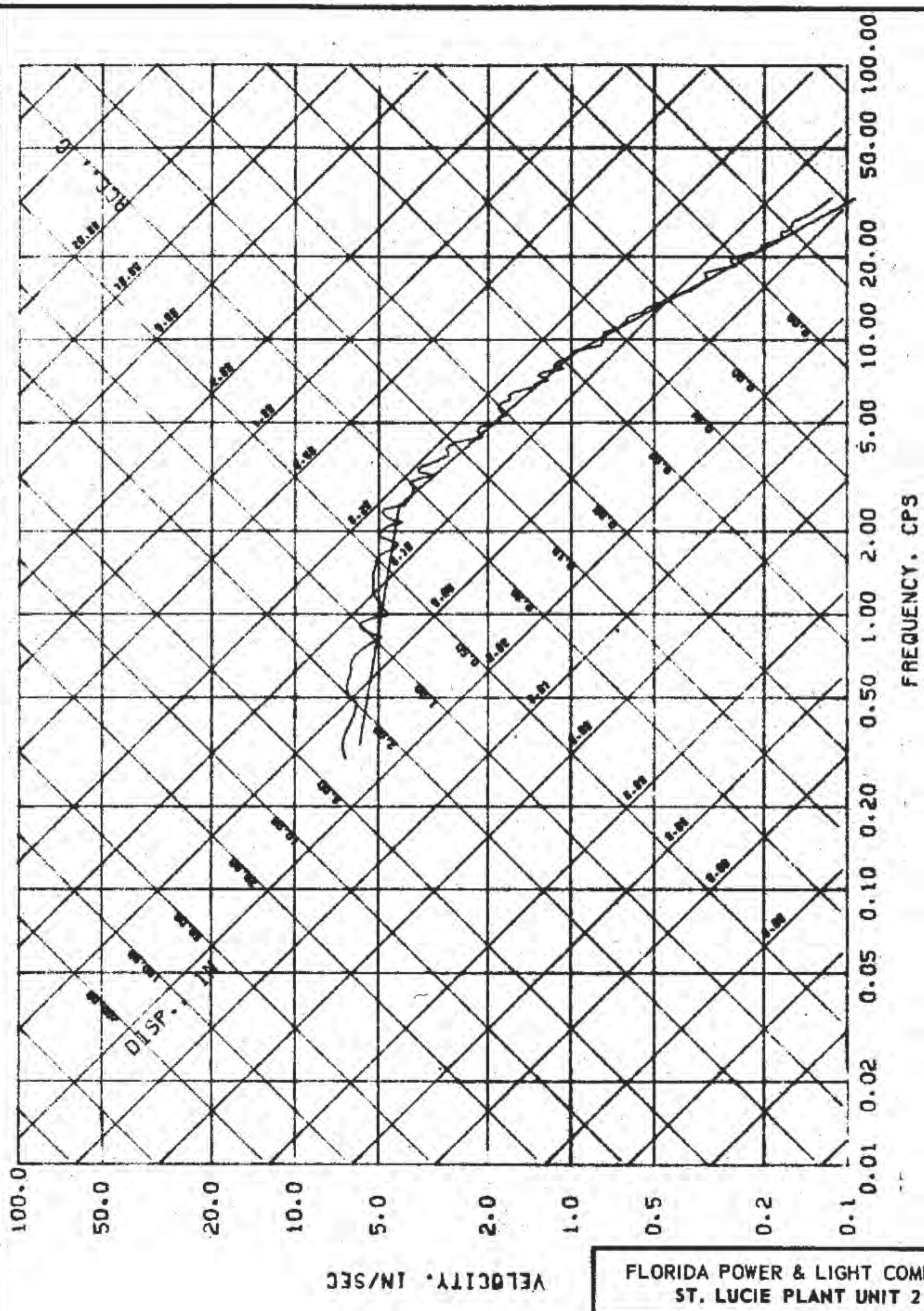
FIGURE 3.7-19



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
2 PERCENT DAMPING – OBE

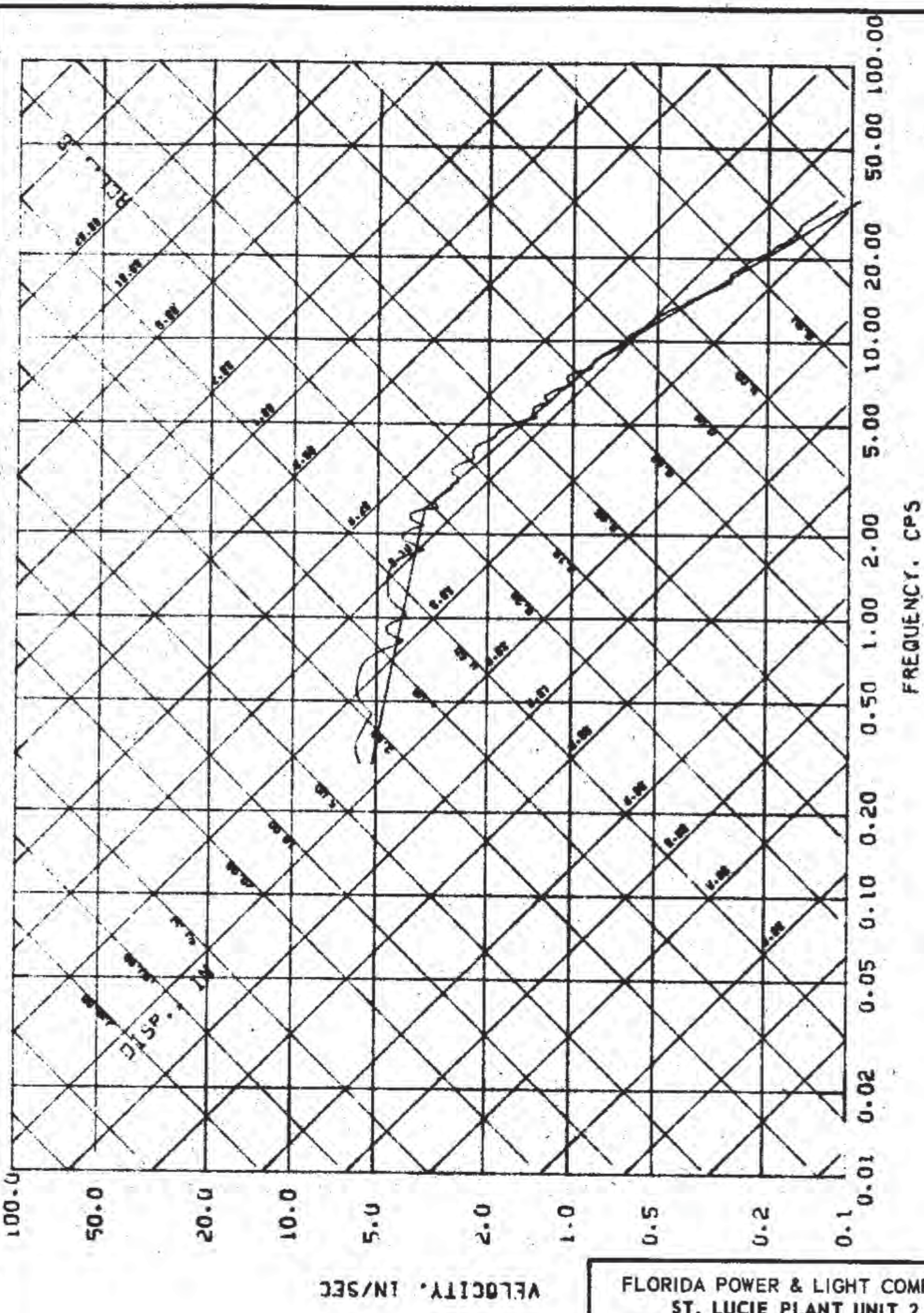
FIGURE 3.7-20



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
4 PERCENT DAMPING – OBE

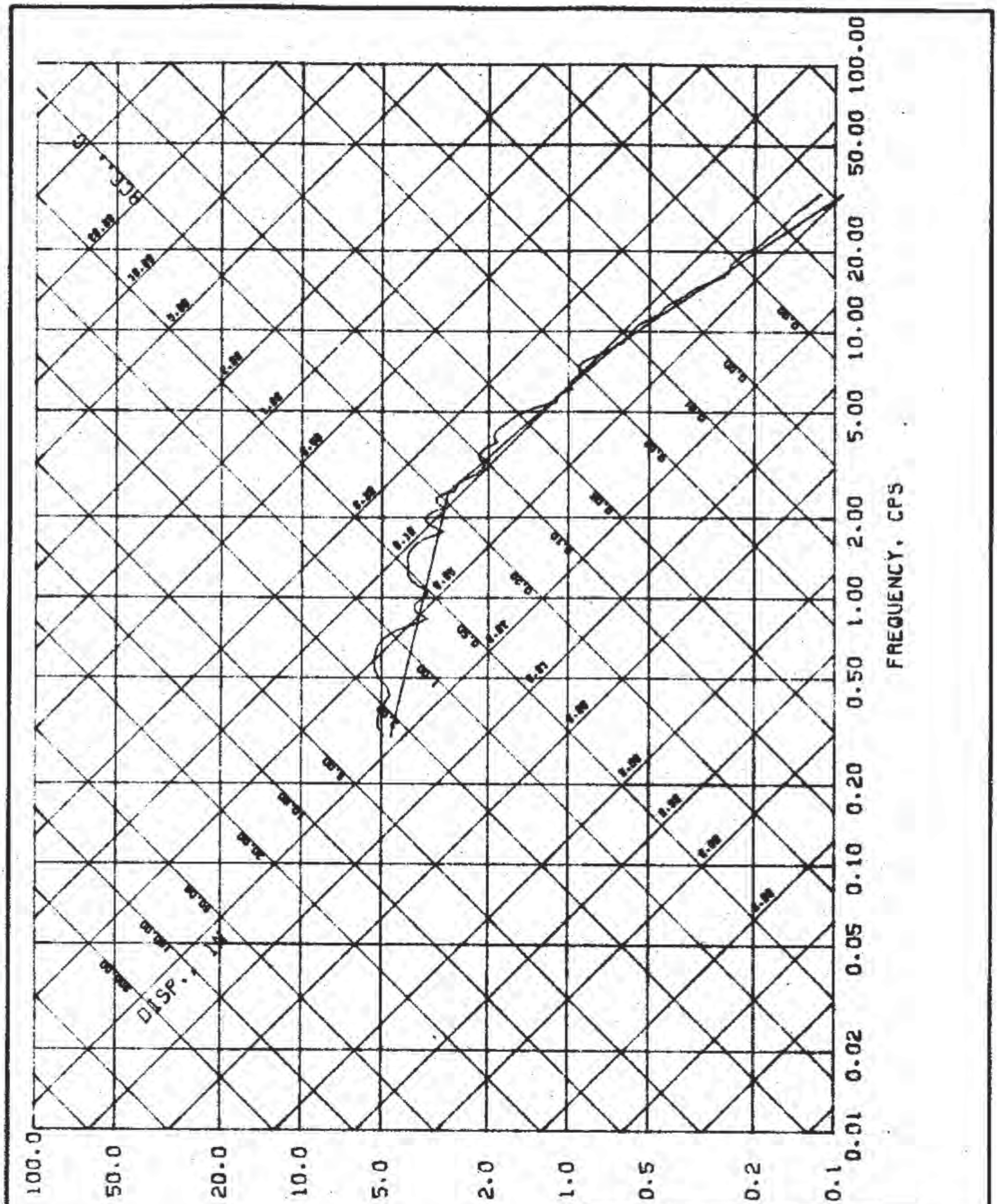
FIGURE 3.7-21



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
5 PERCENT DAMPING – OBE

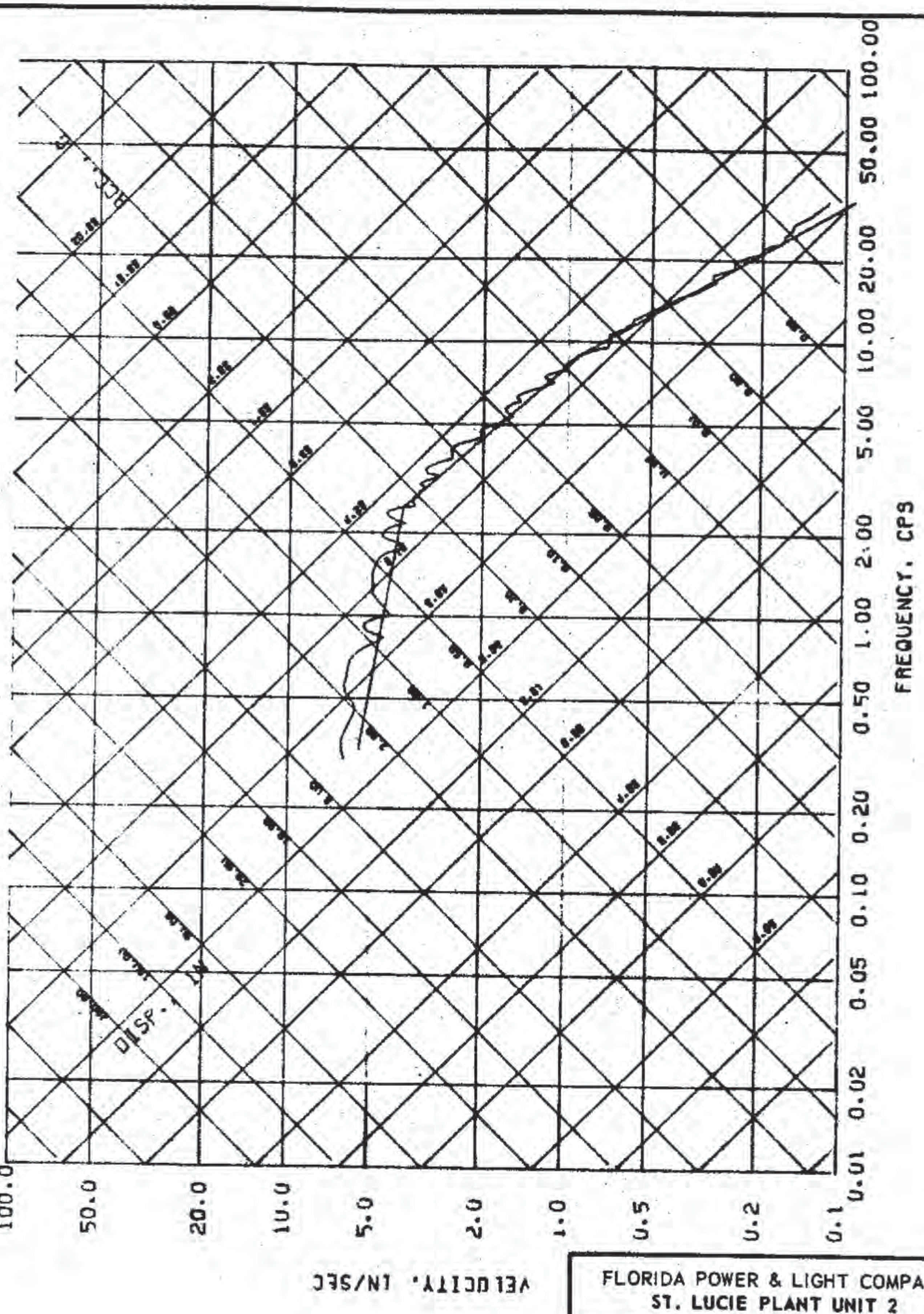
FIGURE 3.7-22



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
7 PERCENT DAMPING – OBE

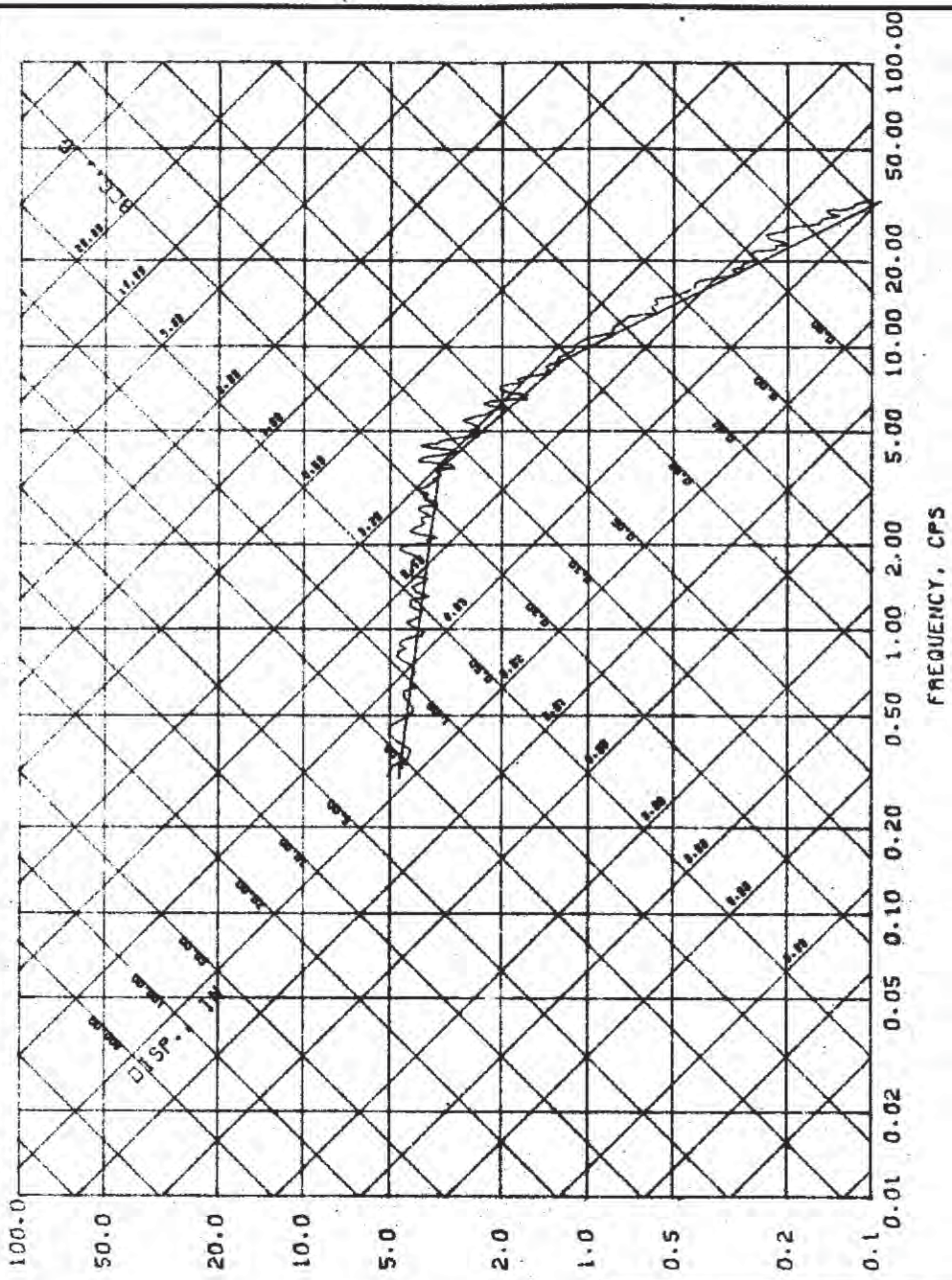
FIGURE 3.7-23



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL RESPONSE SPECTRA
10 PERCENT DAMPING – OBE

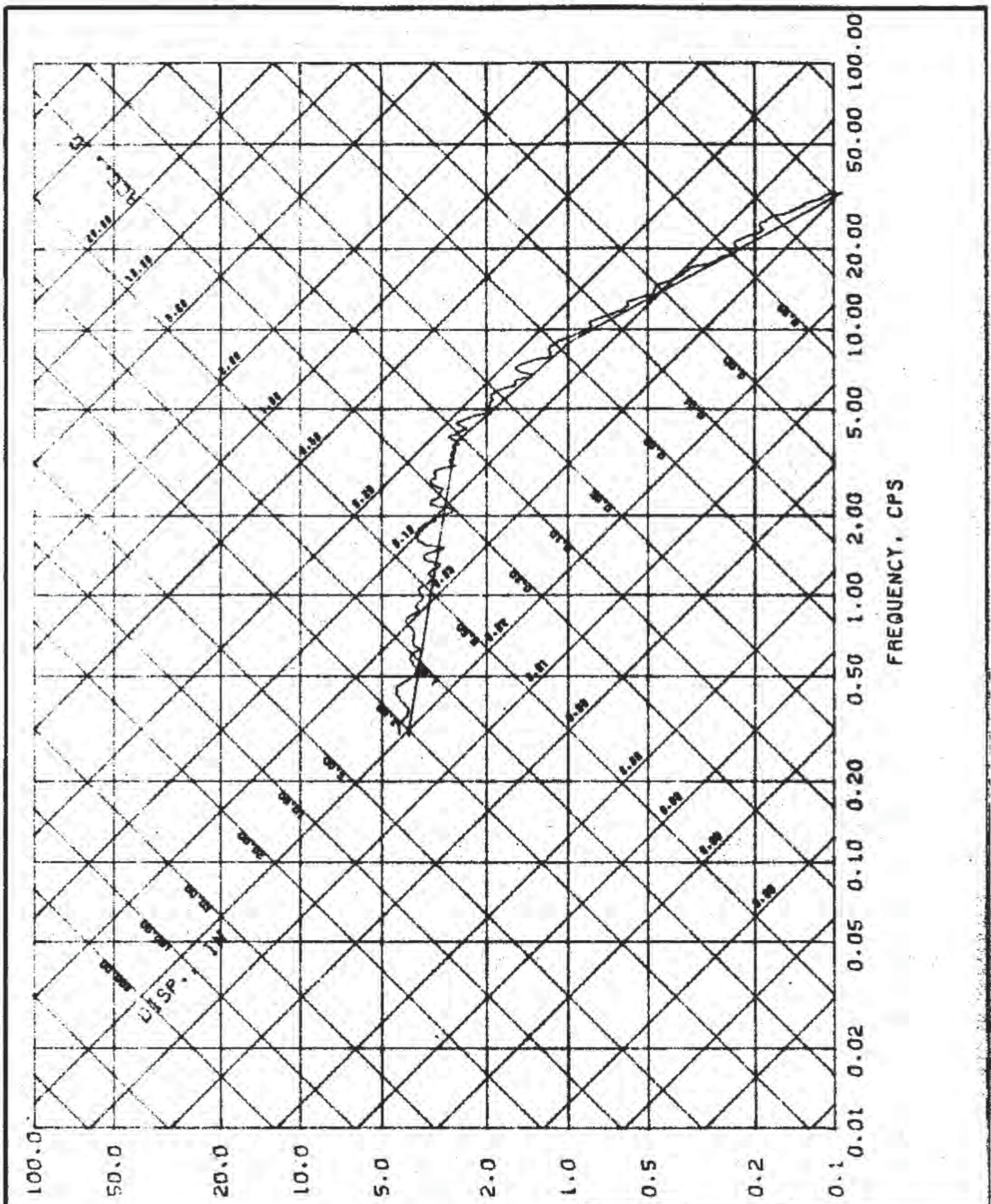
FIGURE 3.7-24



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
2 PERCENT DAMPING – OBE

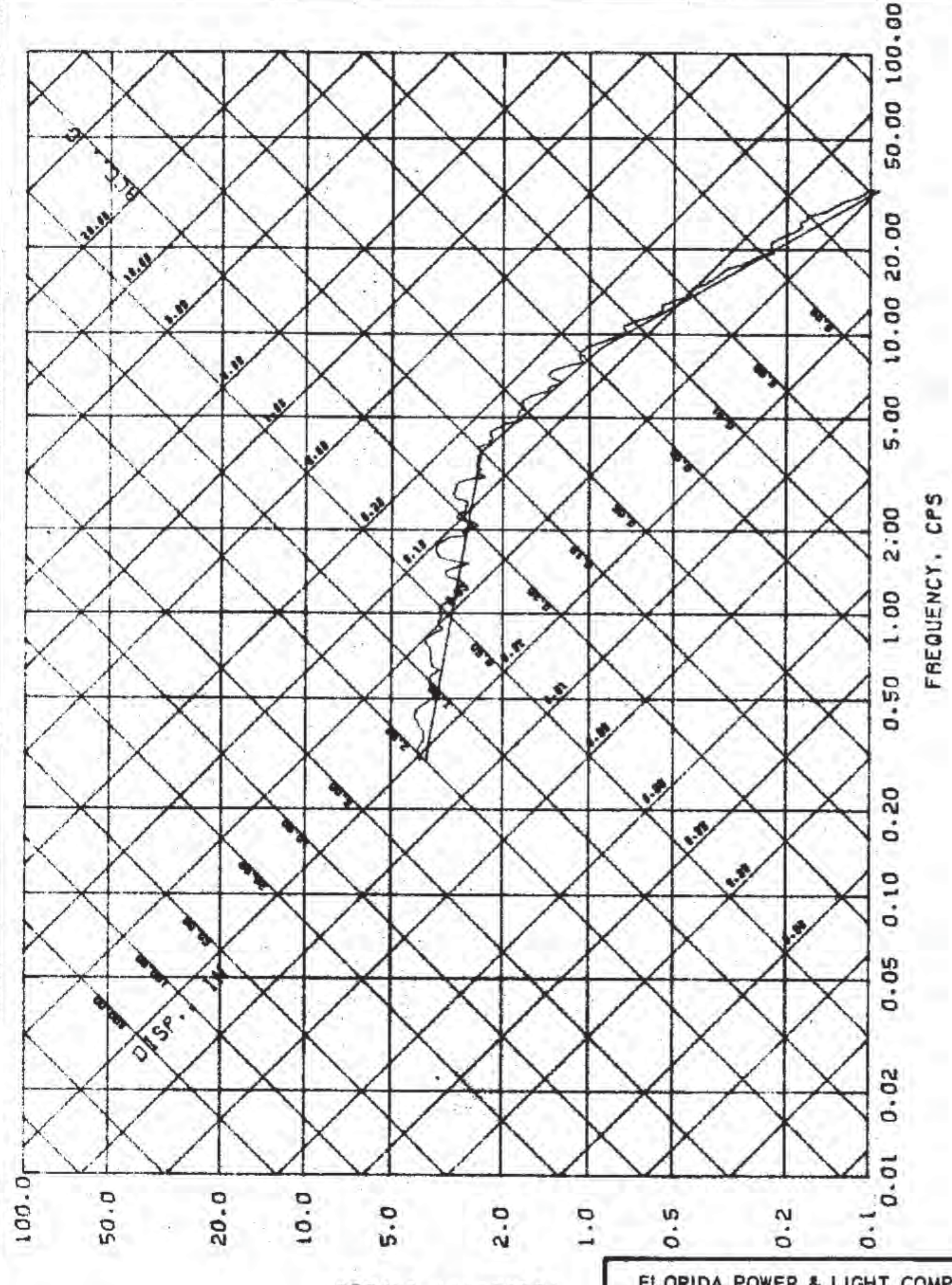
FIGURE 3.7-25



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
4 PERCENT DAMPING – OBE

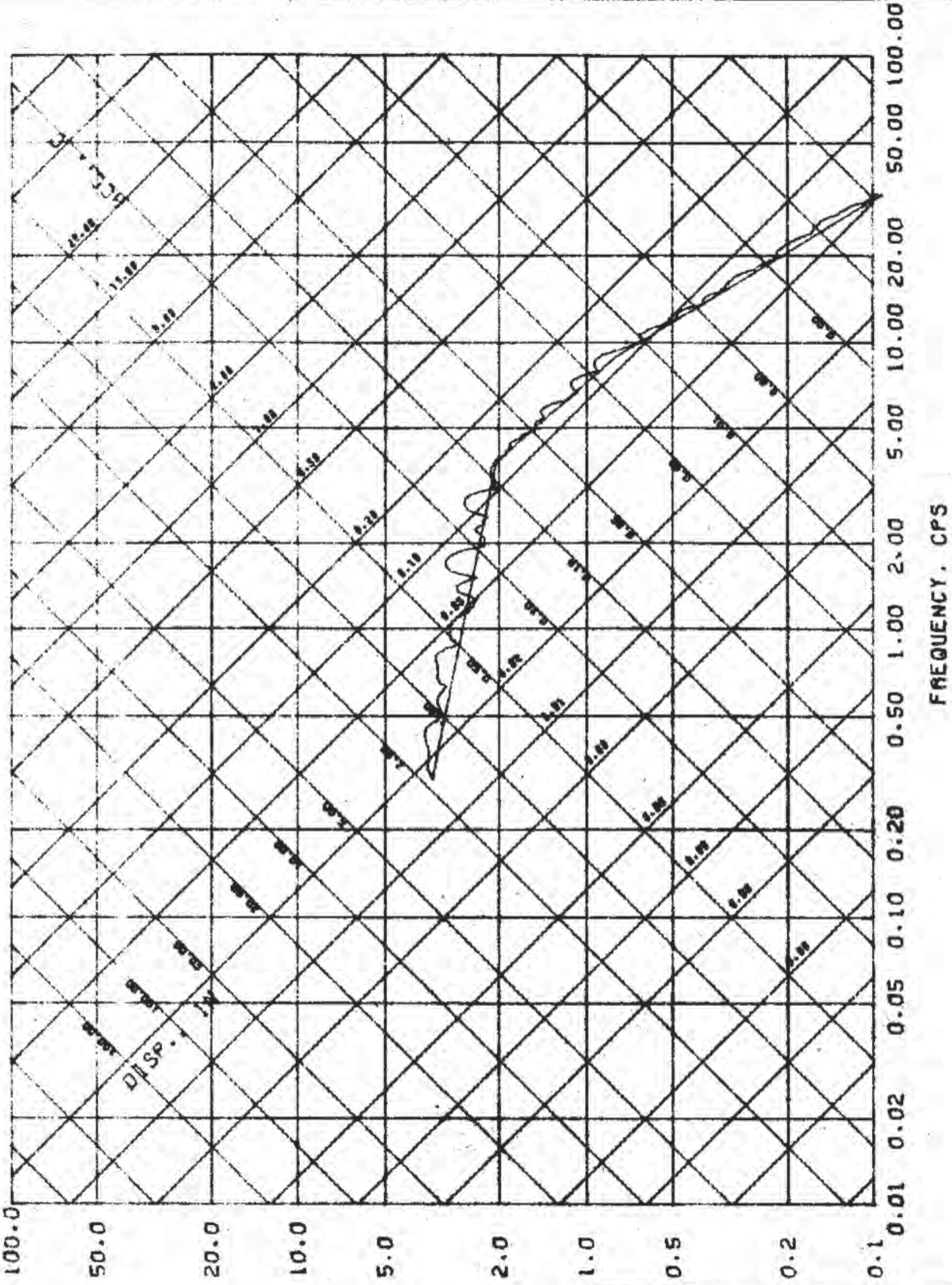
FIGURE 3.7-26



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
5 PERCENT DAMPING – OBE

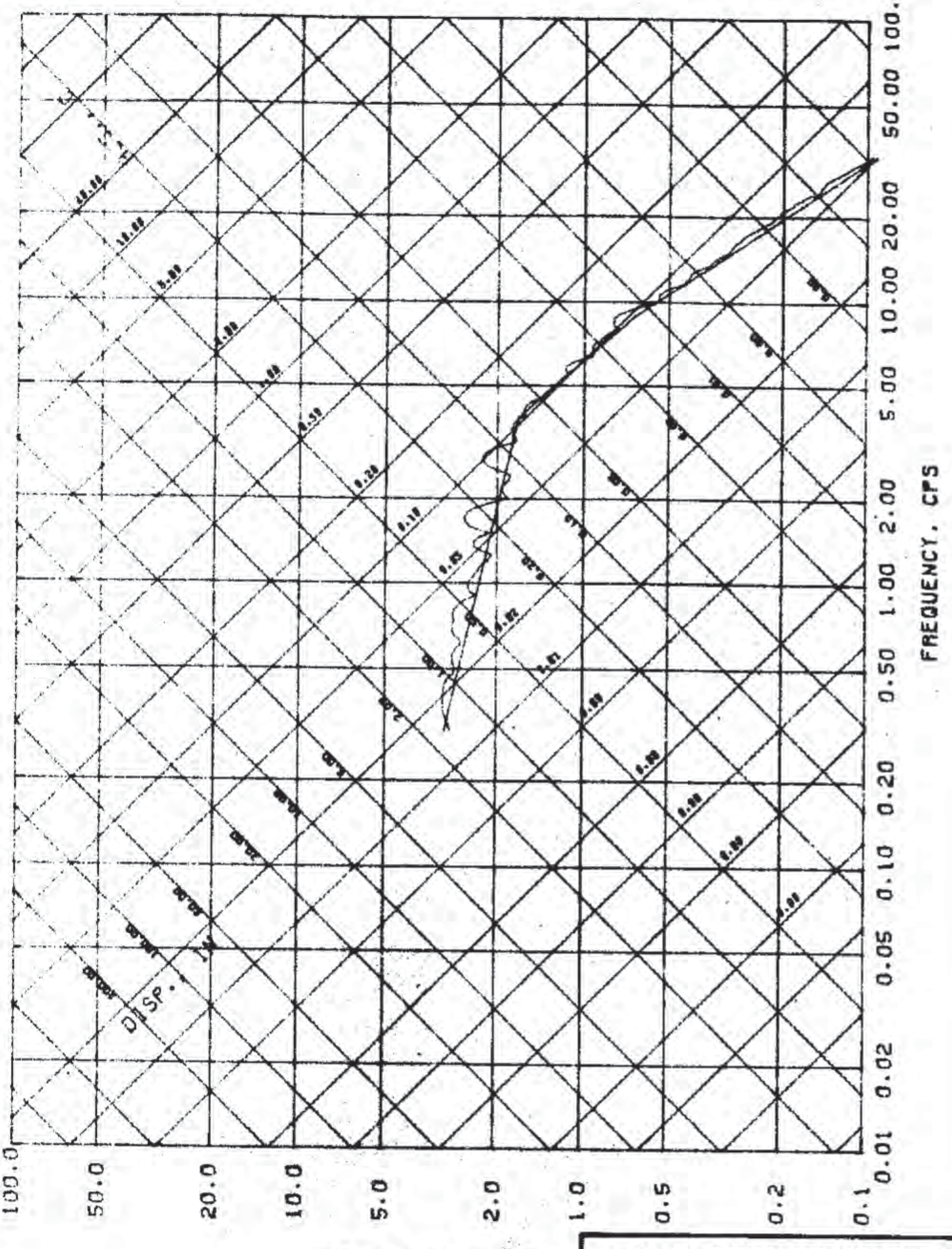
FIGURE 3.7-27



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
7 PERCENT DAMPING – OBE

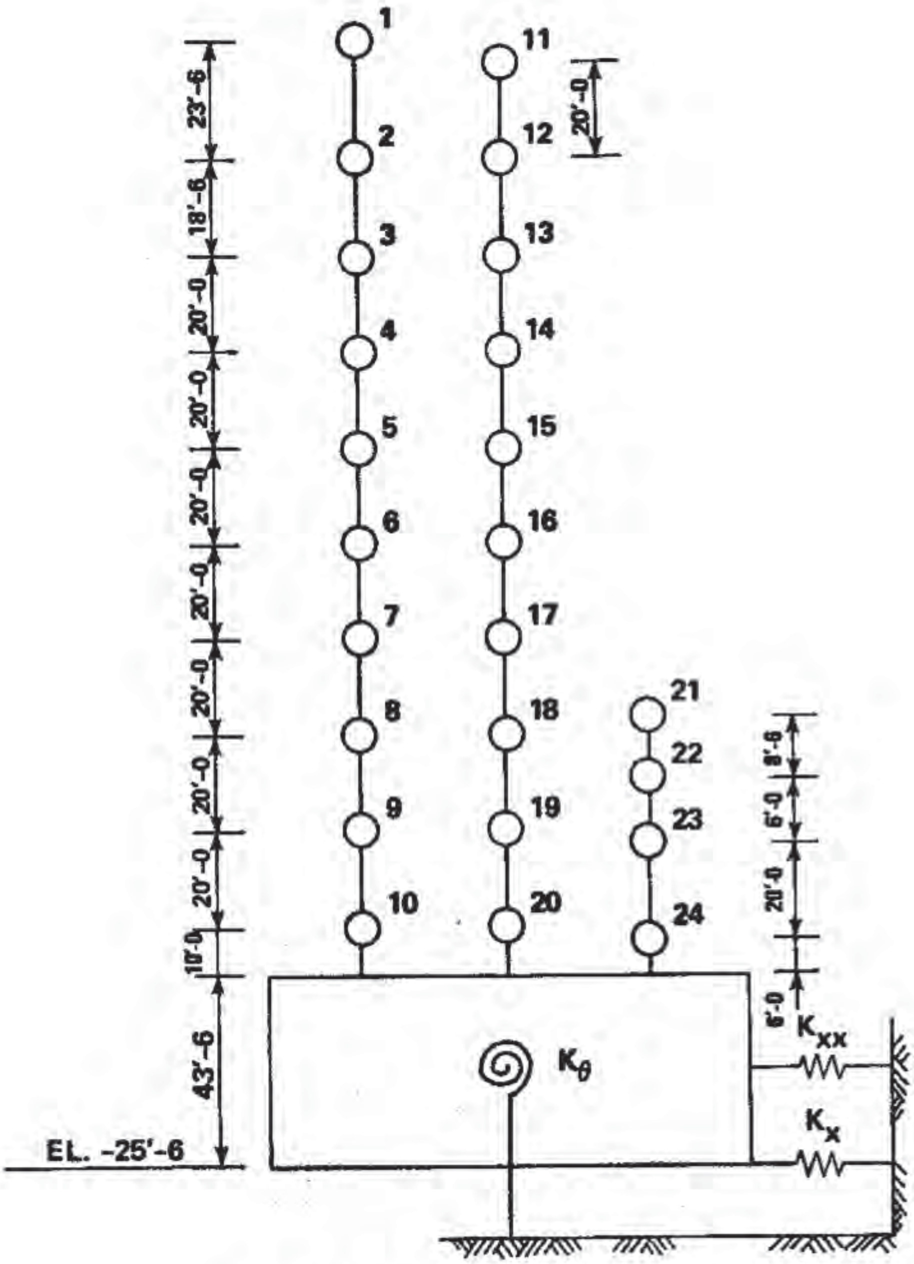
FIGURE 3.7-28



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL RESPONSE SPECTRA
10 PERCENT DAMPING – OBE

FIGURE 3.7-29



MASS 1 ~ 10 SHIELD STRUCTURE
11 ~ 20 STEEL CONTAINMENT
21 ~ 24 INTERNAL STRUCTURE

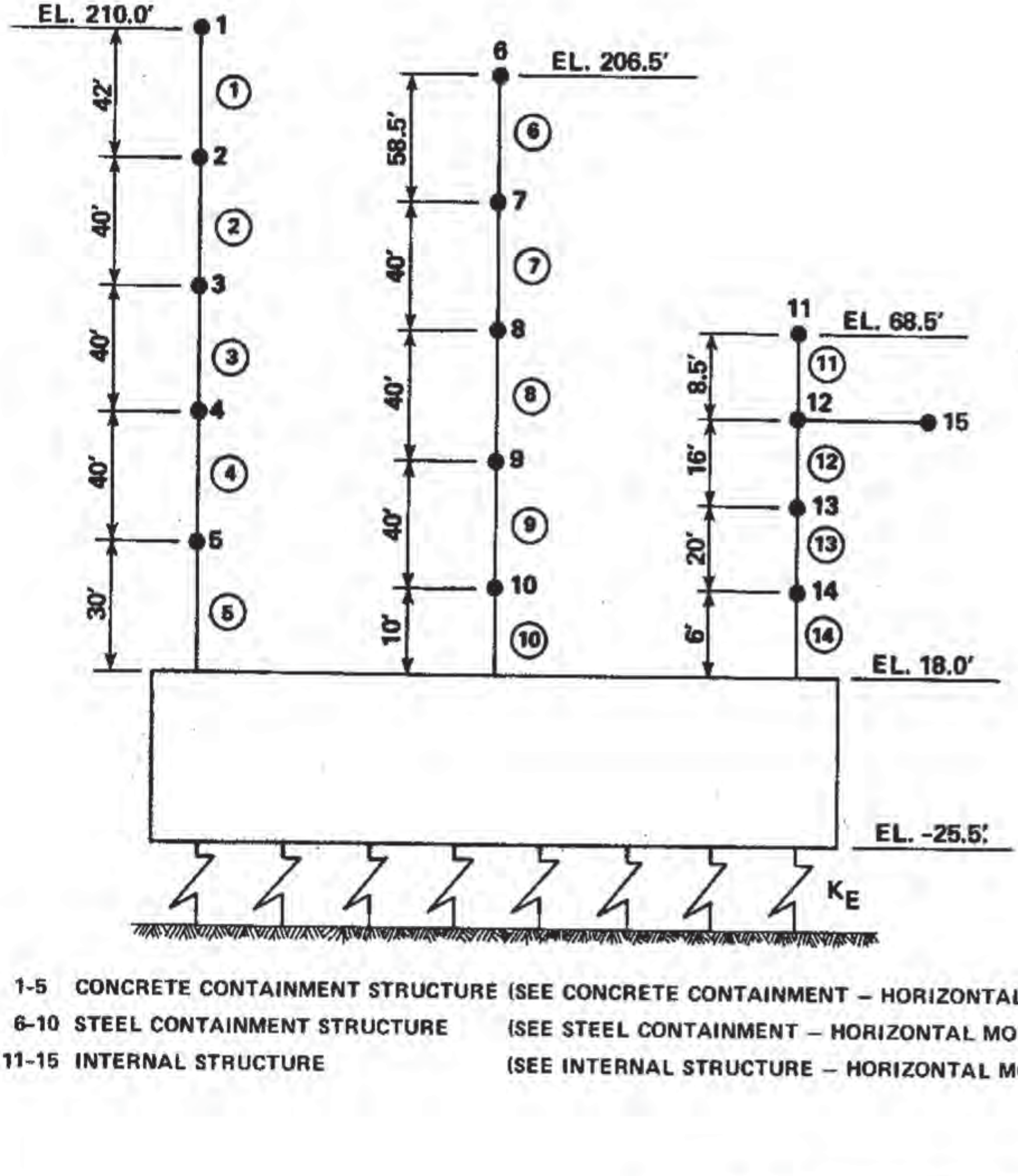
DIAMETER OF FDN = 160'-0

STRUCTURAL PROPERTIES = TABLE 3.7-4

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING HORIZONTAL
MATHEMATICAL MODEL

FIGURE 3.7-30



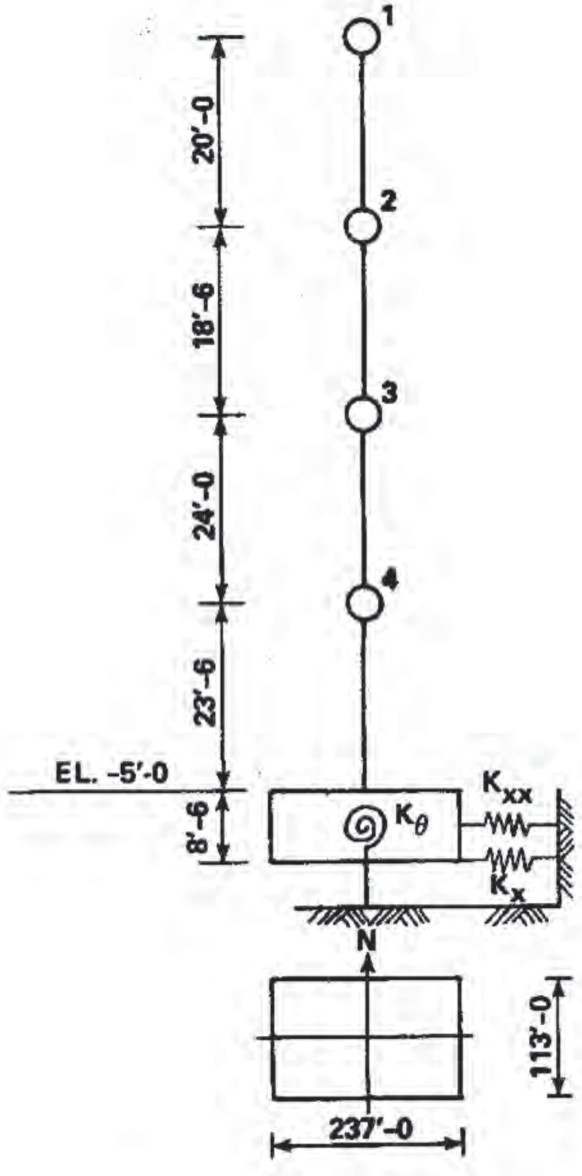
1-5 CONCRETE CONTAINMENT STRUCTURE (SEE CONCRETE CONTAINMENT – HORIZONTAL MODEL)

6-10 STEEL CONTAINMENT STRUCTURE (SEE STEEL CONTAINMENT – HORIZONTAL MODEL)

11-15 INTERNAL STRUCTURE (SEE INTERNAL STRUCTURE – HORIZONTAL MODEL)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING
VERTICAL MODEL
FIGURE 3.7-31

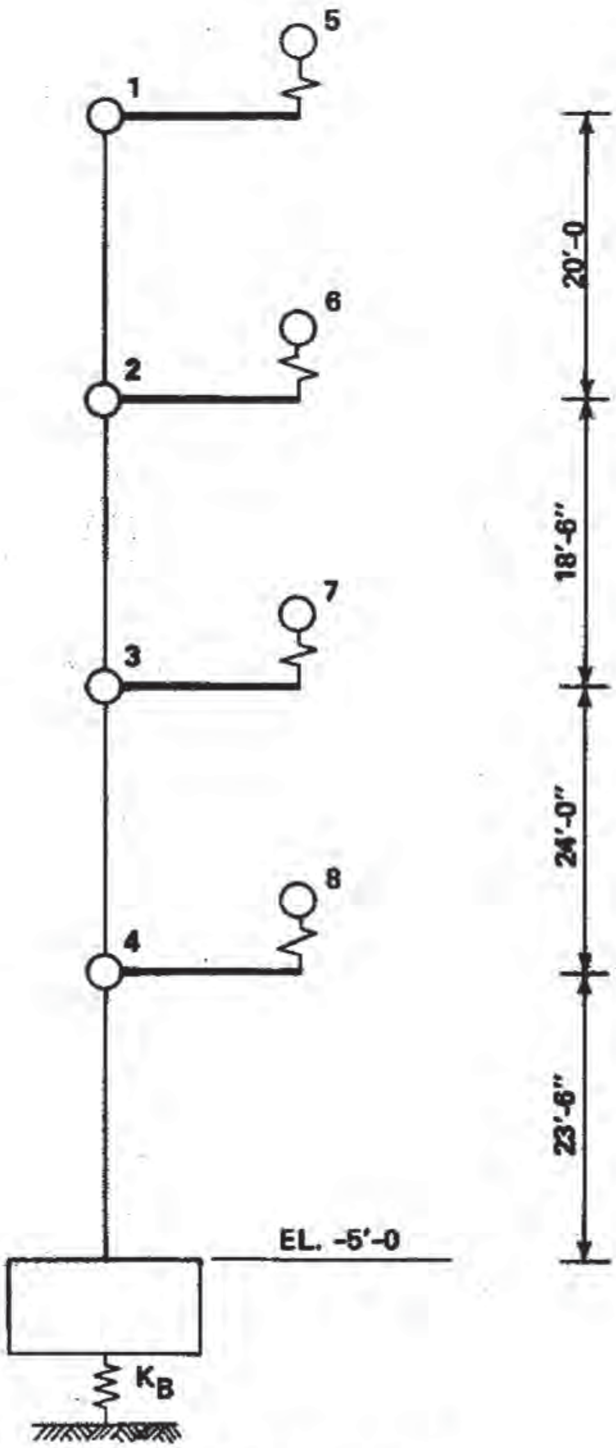


STRUCTURAL PROPERTIES = TABLE 3.7-6

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUXILIARY BUILDING
HORIZONTAL MATHEMATICAL MODEL

FIGURE 3.7-32



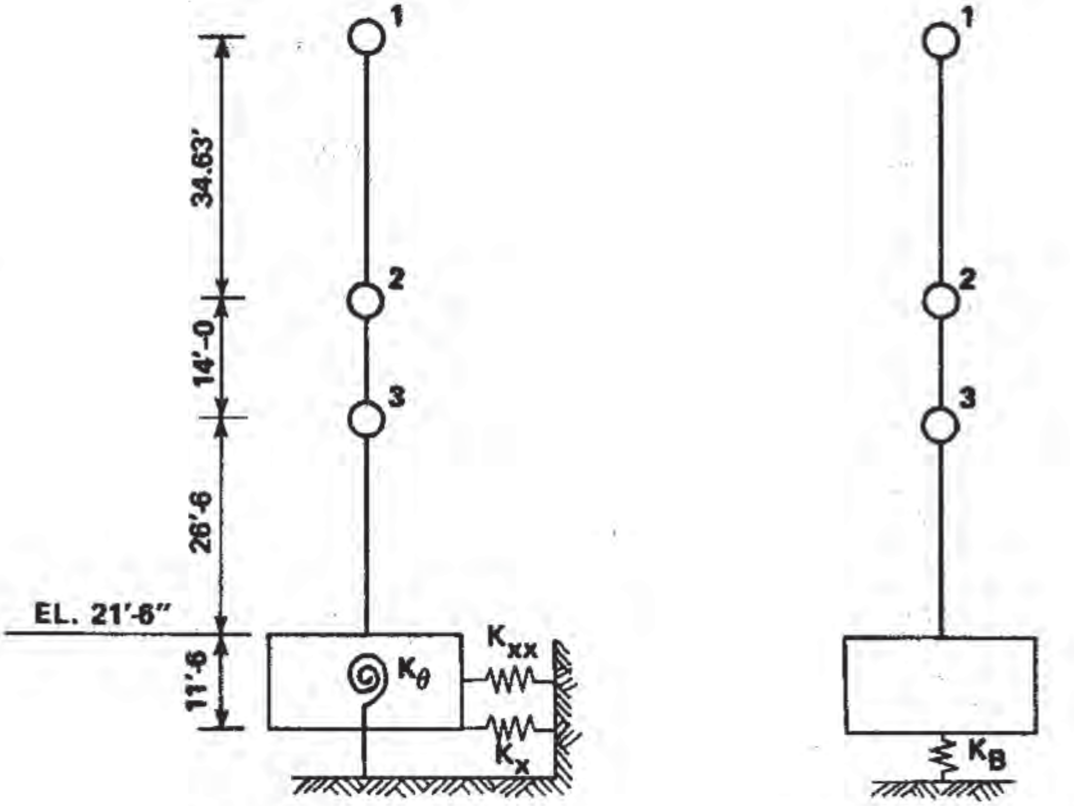
NOTE: HEAVY LINE INDICATES RIGID MEMBER

STRUCTURAL PROPERTIES = TABLE 3.7–7

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUXILIARY BUILDING
VERTICAL MATHEMATICAL MODEL

FIGURE 3.7-33



HORIZONTAL MODEL

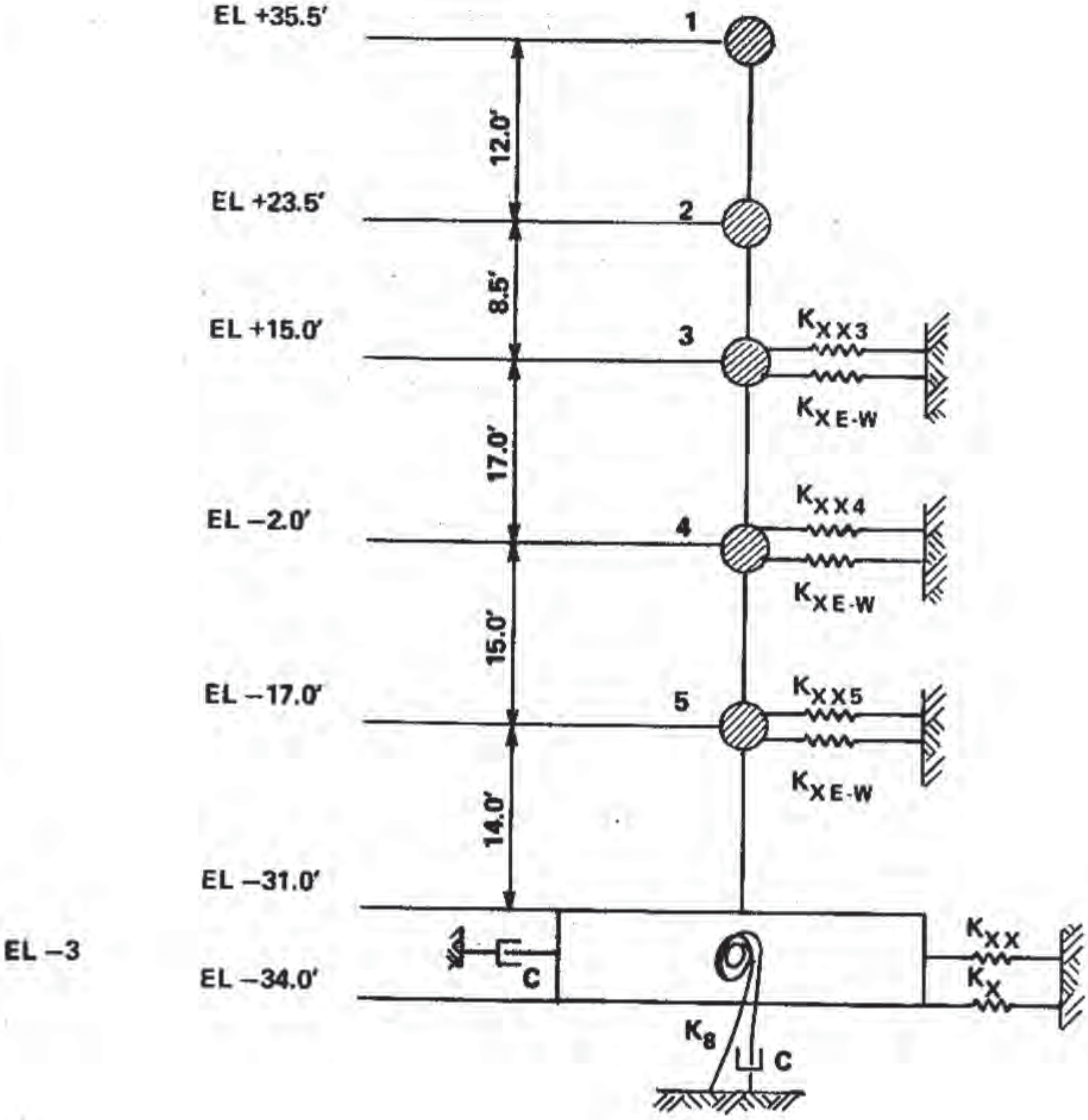
VERTICAL MODEL

STRUCTURAL PROPERTIES = TABLE 3.7–8

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FUEL HANDLING BUILDING
MATHEMATICAL MODELS

FIGURE 3.7-34



K_{XX}: N-S BEARING SPRING

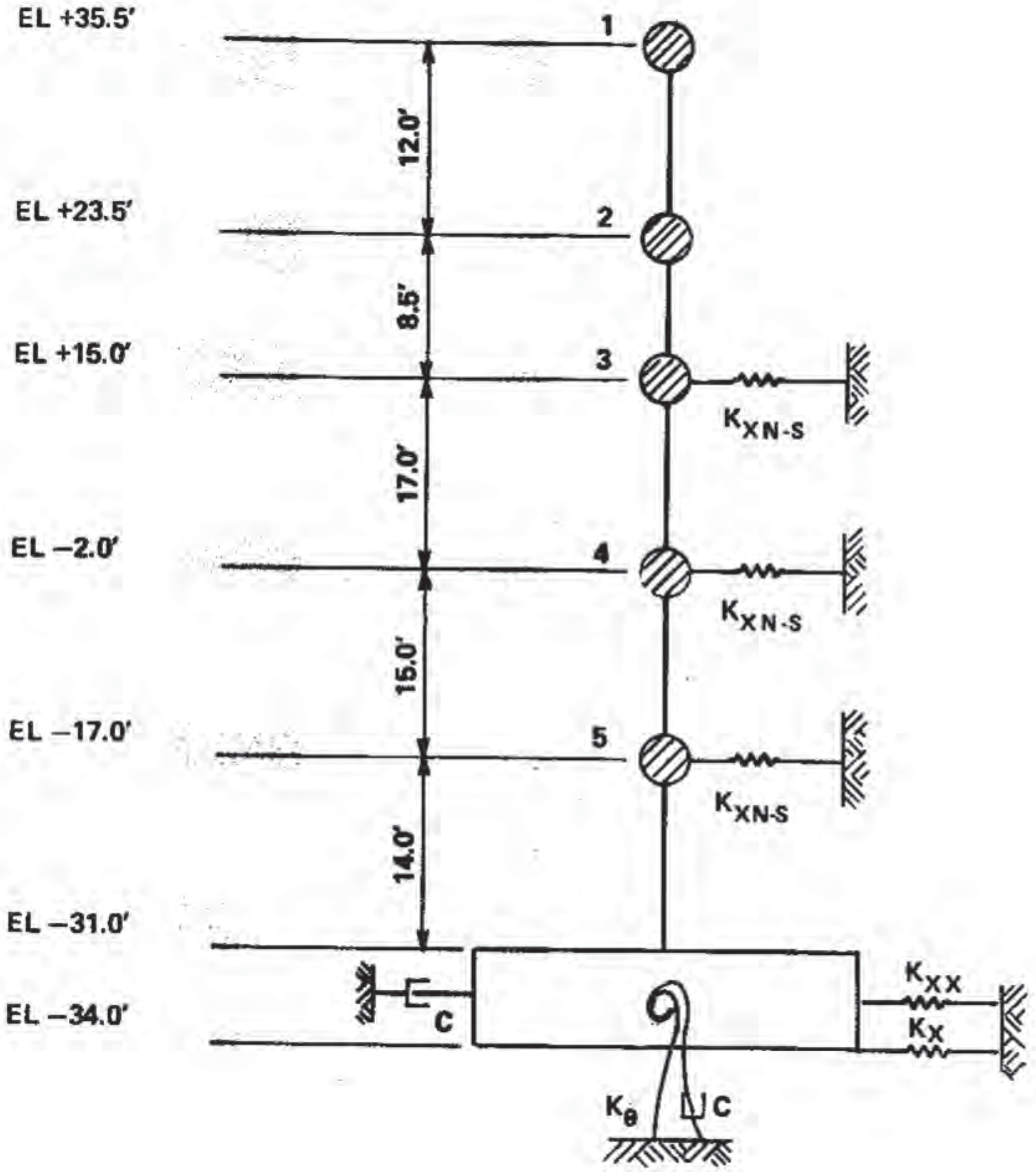
$K_{XE\text{-}W}$: E-W SHEAR SPRING

STRUCTURAL PROPERTIES = TABLE 3.7-9

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

INTAKE STRUCTURE
HORIZONTAL (N-S) MODEL

FIGURE 3.7-35

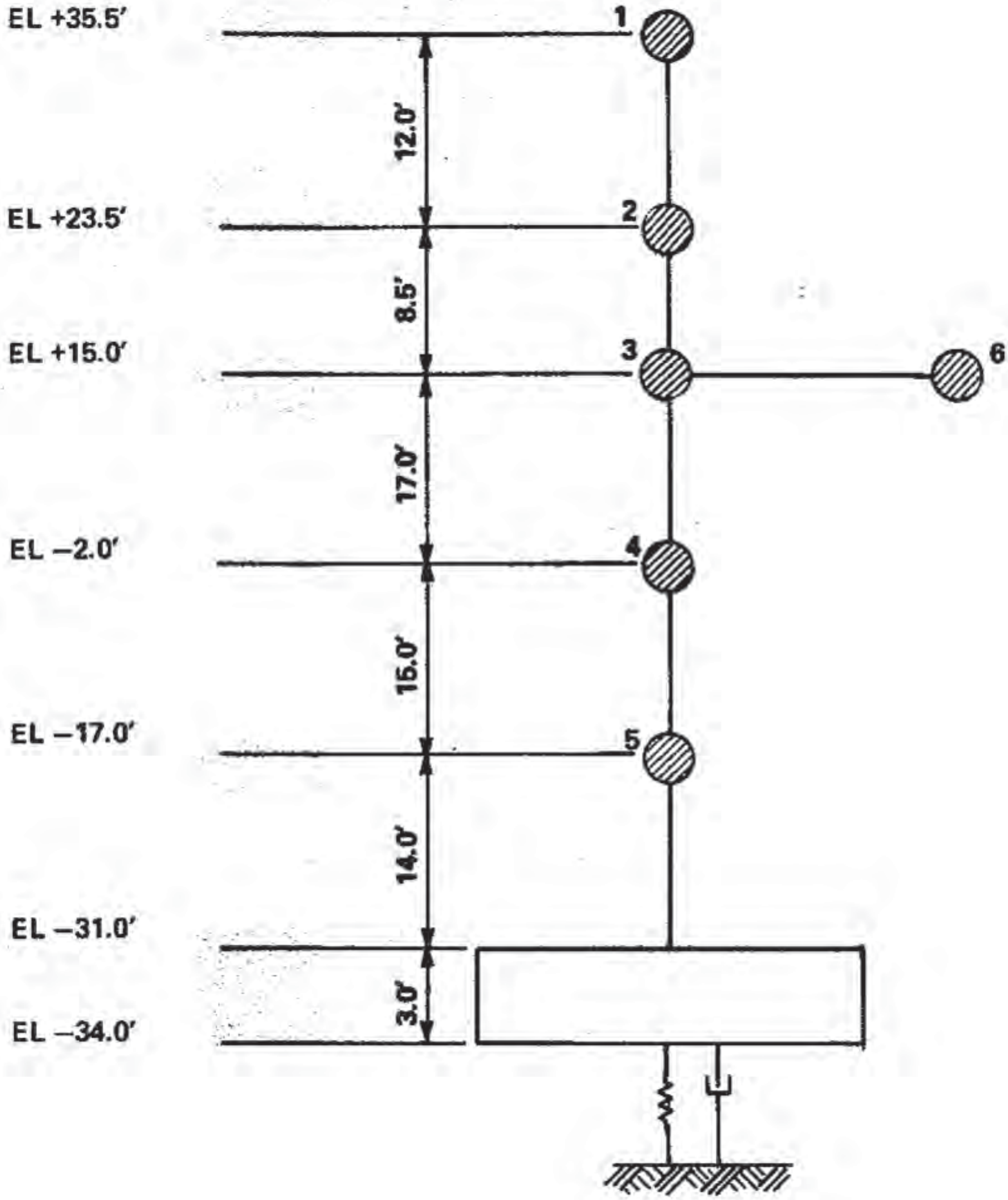


K_{XN-S}: N-S – SHEAR SPRING
STRUCTURAL PROPERTIES = TABLE 3.7-9

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

INTAKE STRUCTURE
HORIZONTAL (E-W) MODEL

FIGURE 3.7-36

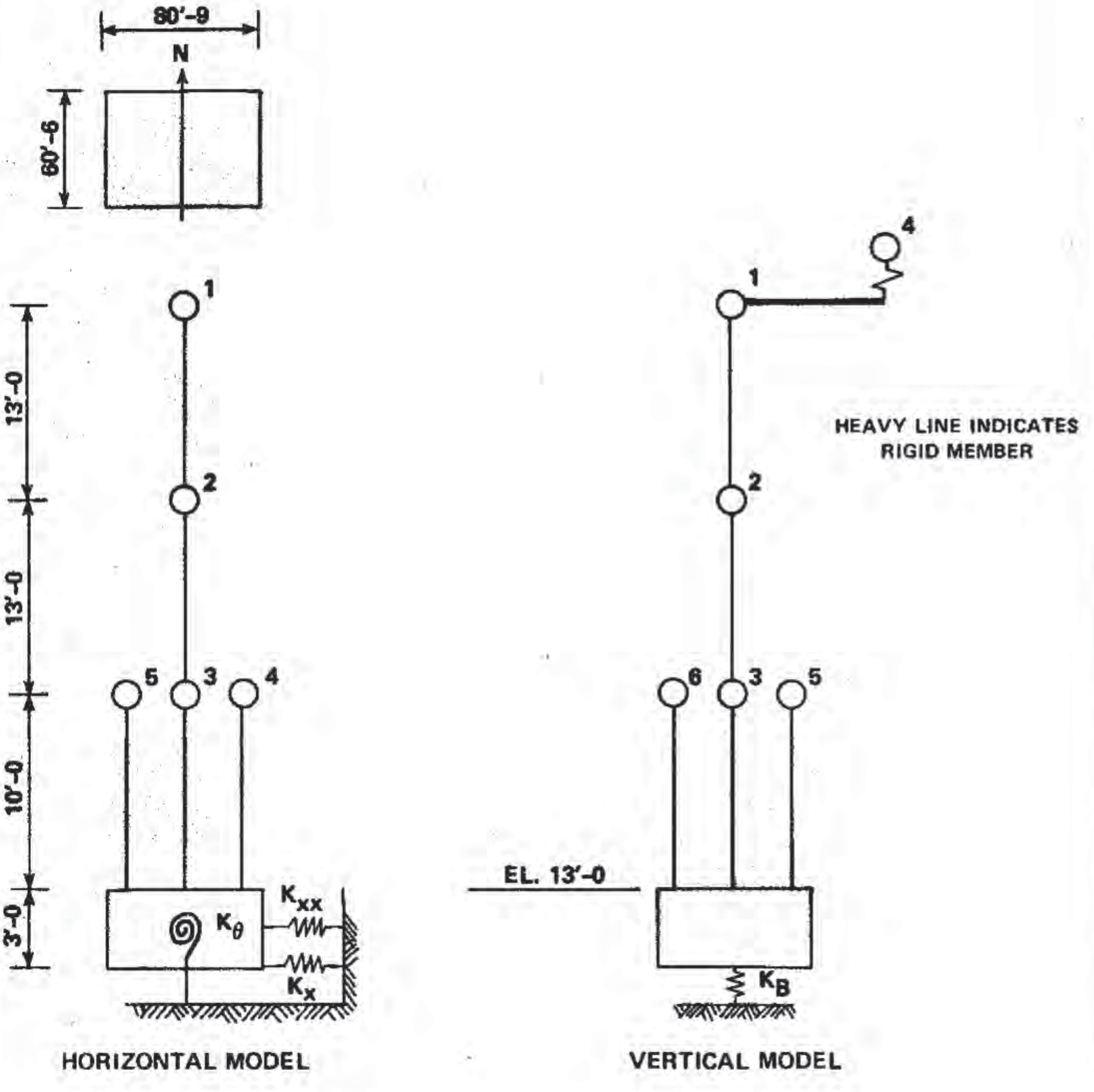


STRUCTURAL PROPERTIES = TABLE 3.7-9

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

INTAKE STRUCTURE VERTICAL MATHEMATICAL MODEL

FIGURE 3.7-37

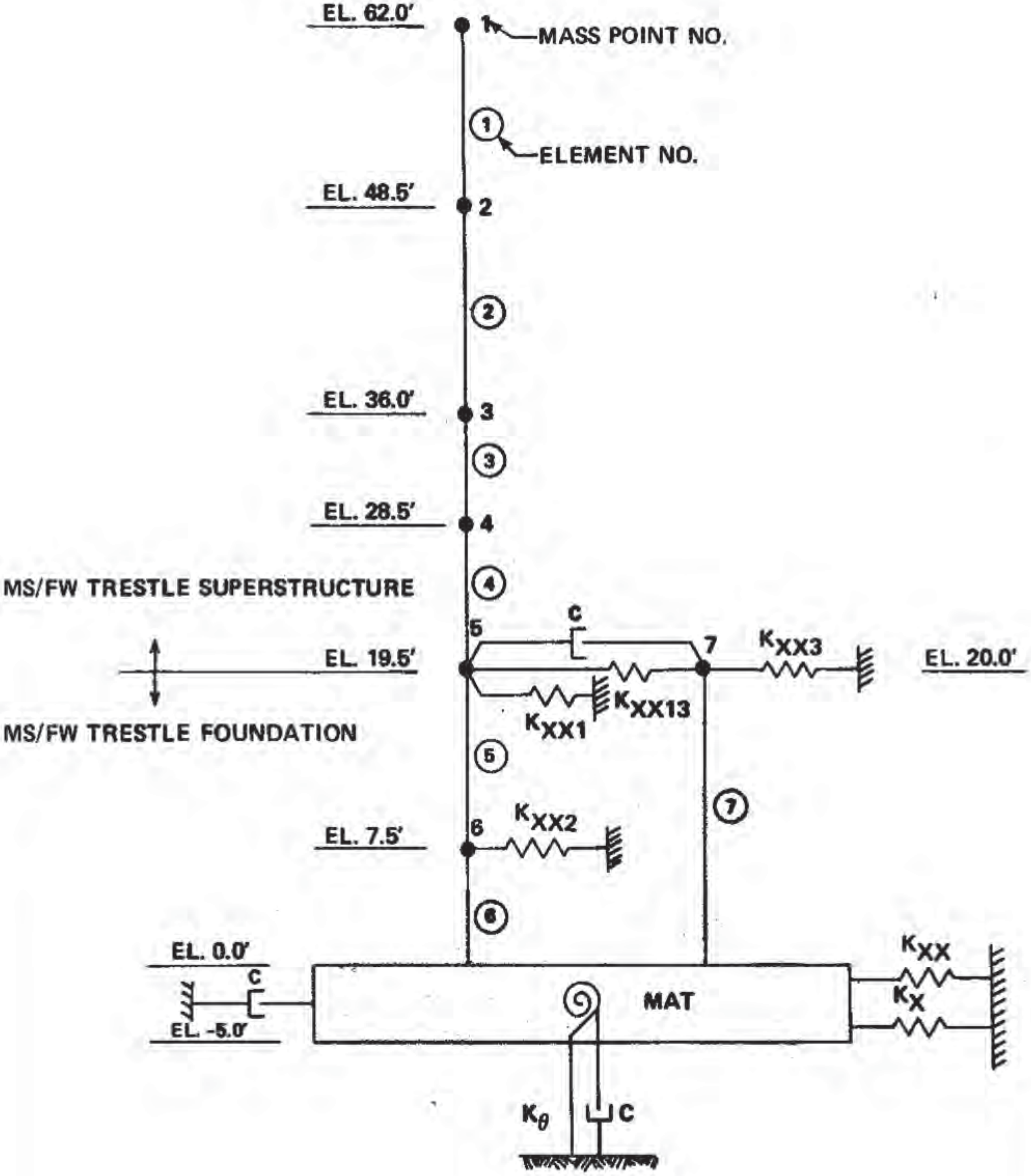


STRUCTURAL PROPERTIES = TABLE 3.7-10

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

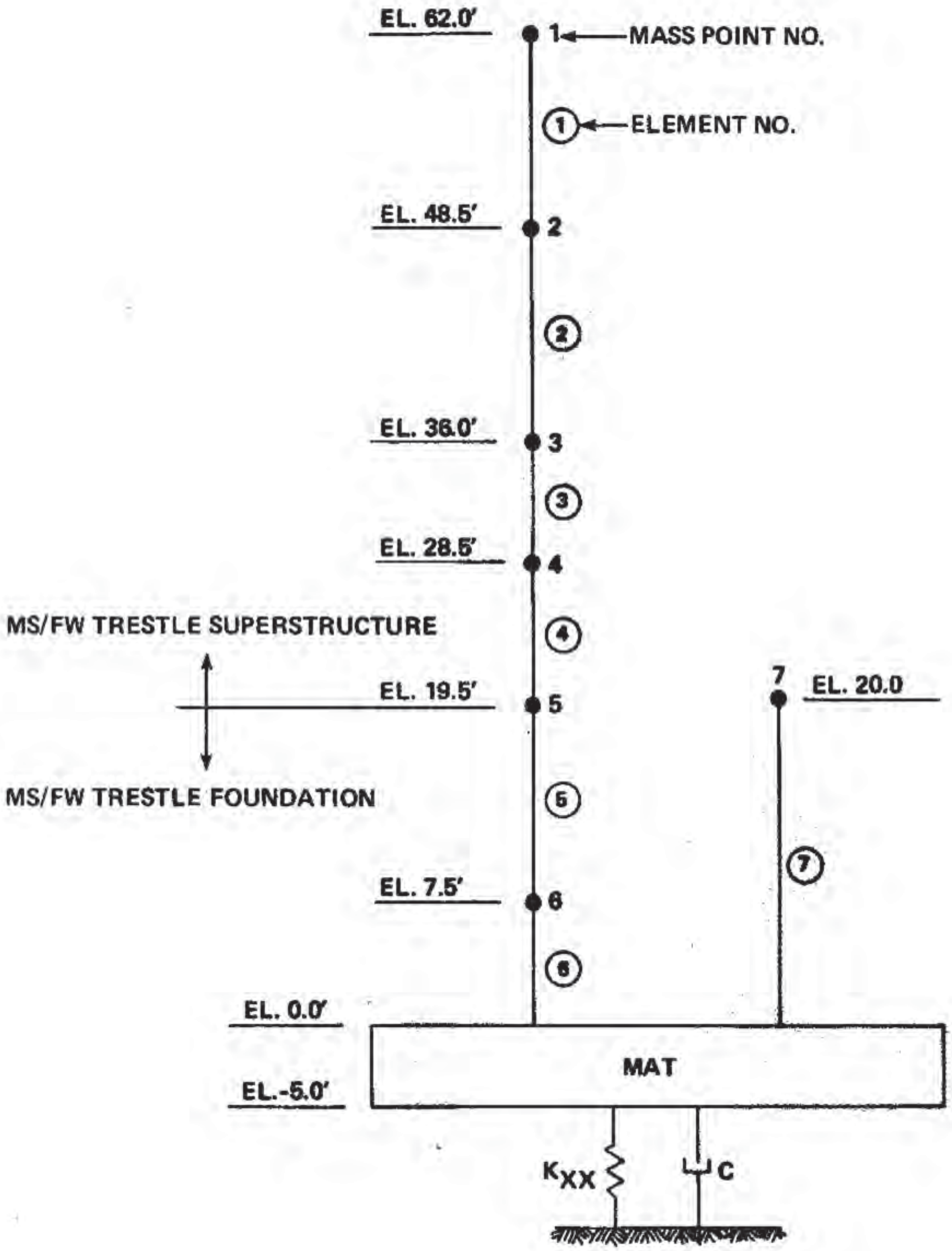
DIESEL GENERATOR BUILDING
MATHEMATICAL MODELS

FIGURE 3.7-38



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

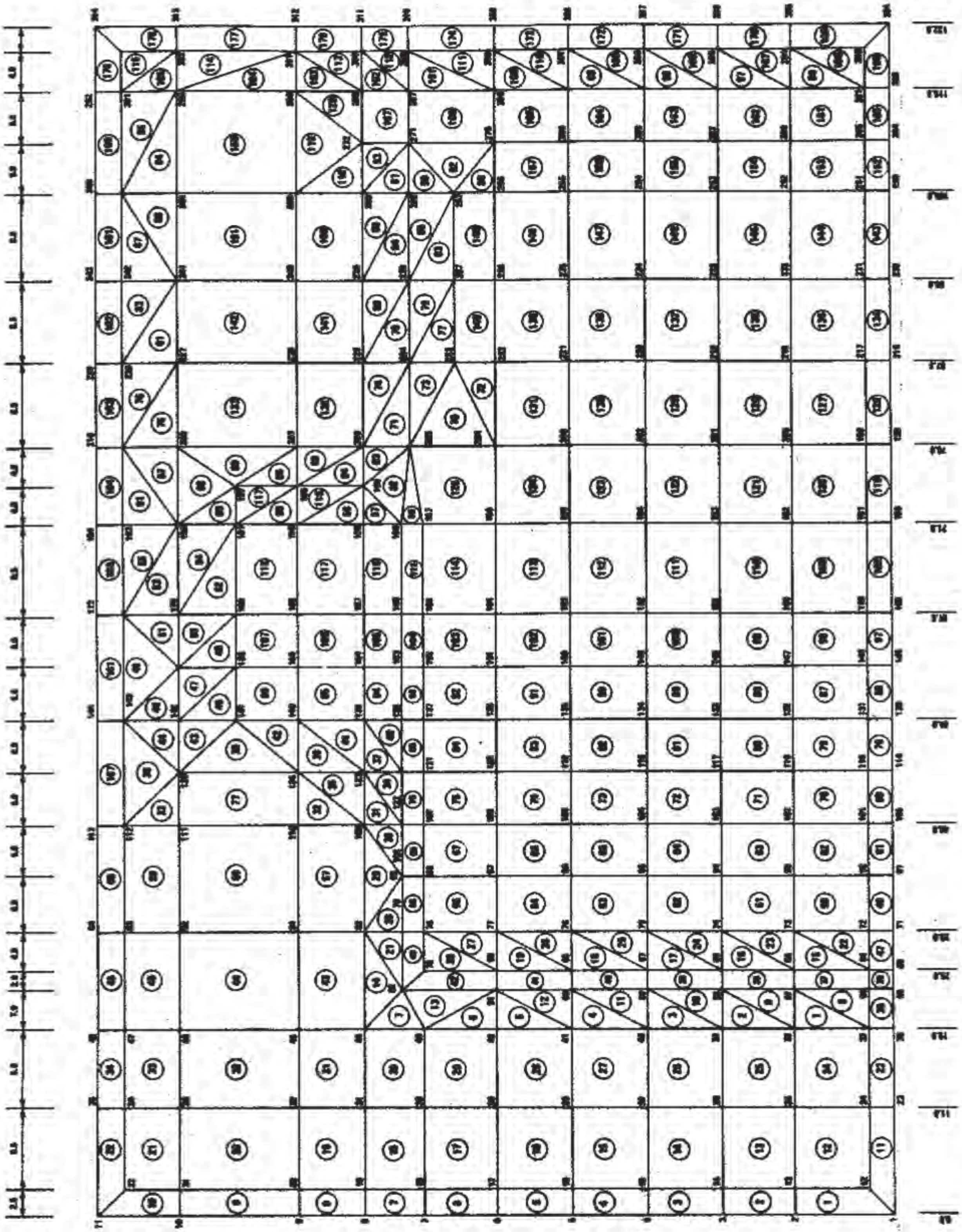
MATHEMATICAL MODEL FOR SEISMIC ANALYSIS OF MS/FW TRESTLE (HORIZONTAL EXCITATION)
FIGURE 3.7-39



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MATHEMATICAL MODEL FOR SEISMIC ANALYSIS OF MS/FW TRESTLE (VERTICAL EXCITATION)

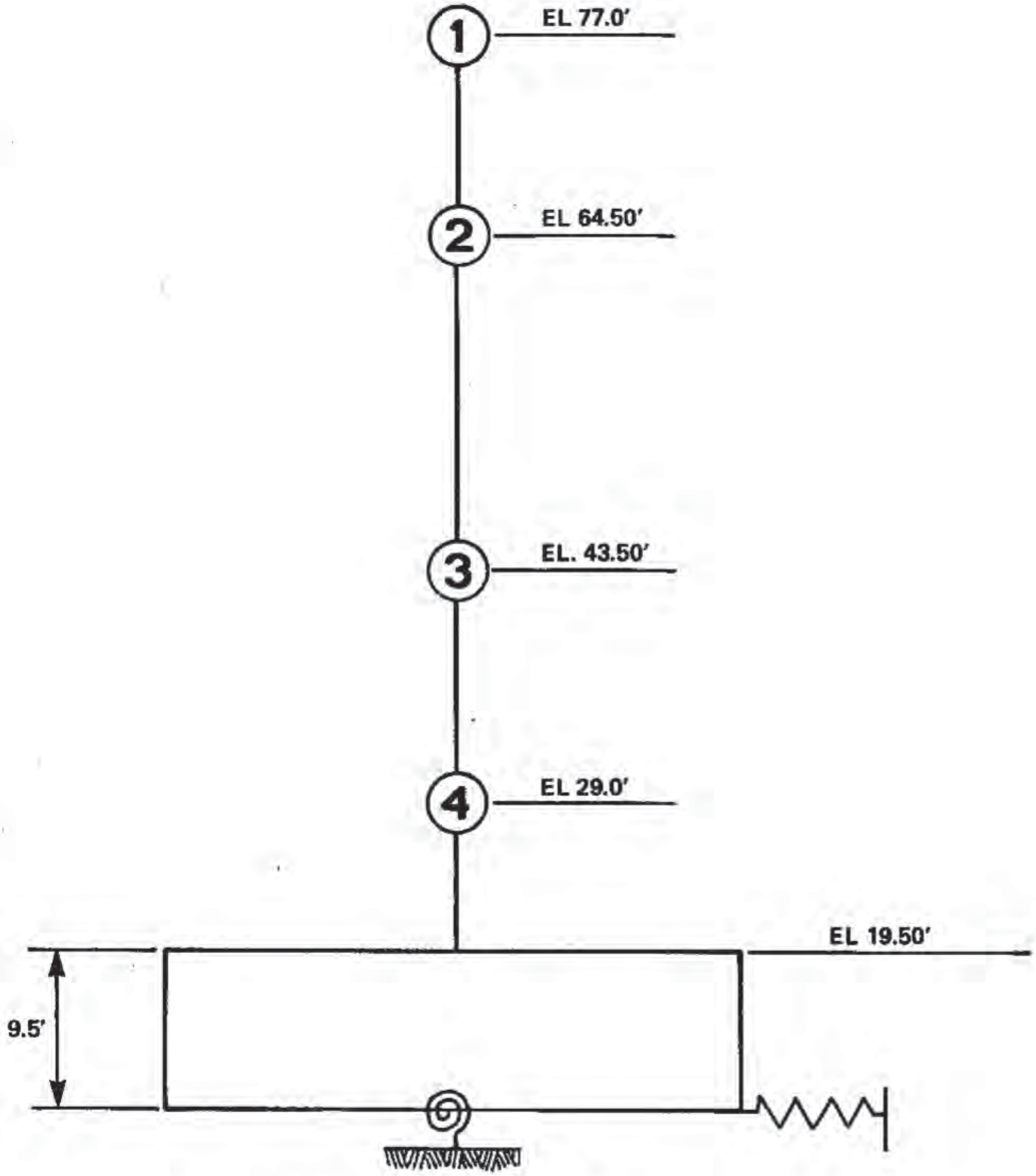
FIGURE 3.7-40



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

STEAM GENERATOR BLOWDOWN FACILITY
FNDN. MAT - FINITE ELEMENT MODEL

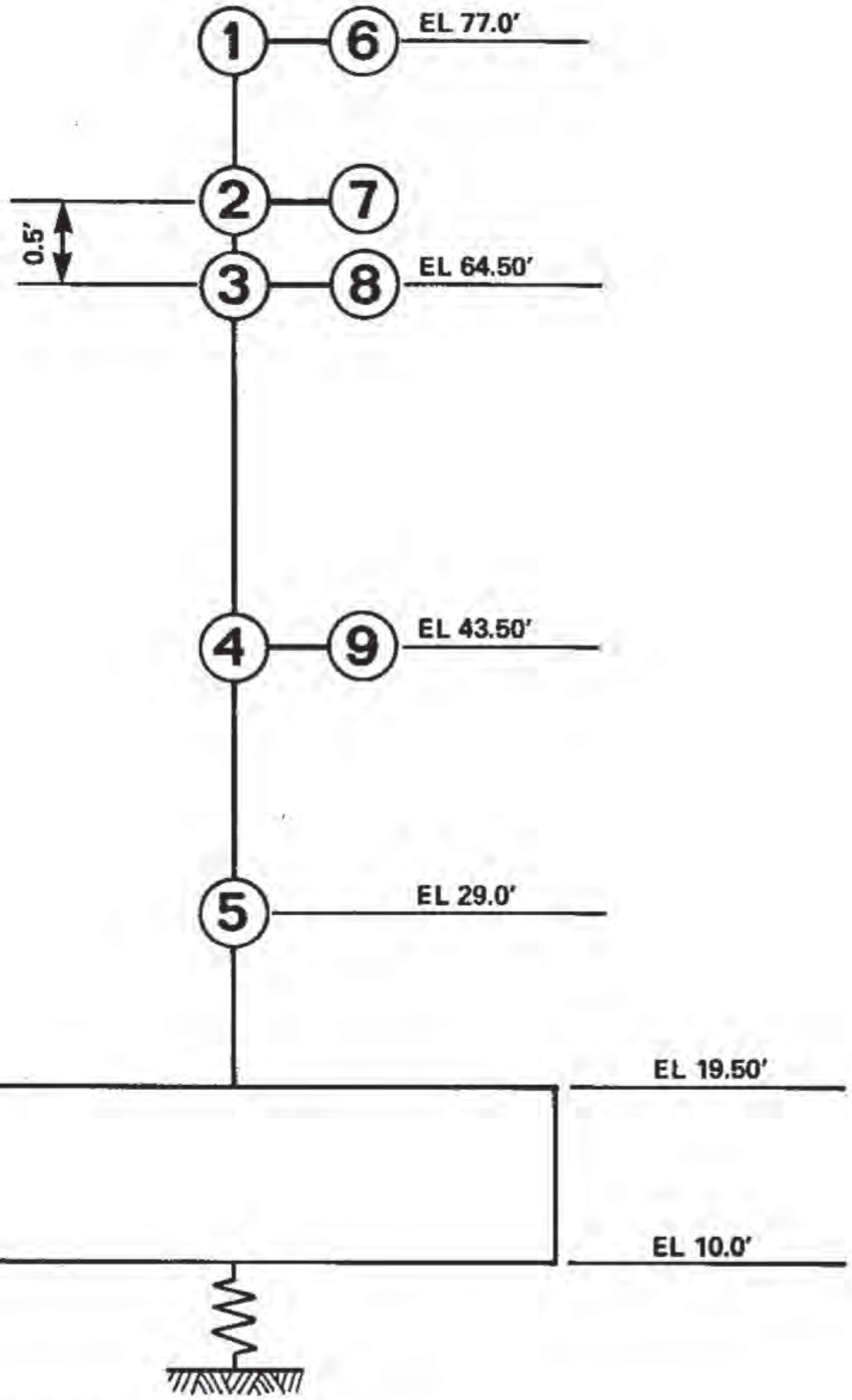
FIGURE 3.7-41



HORIZONTAL MATHEMATICAL MODEL

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

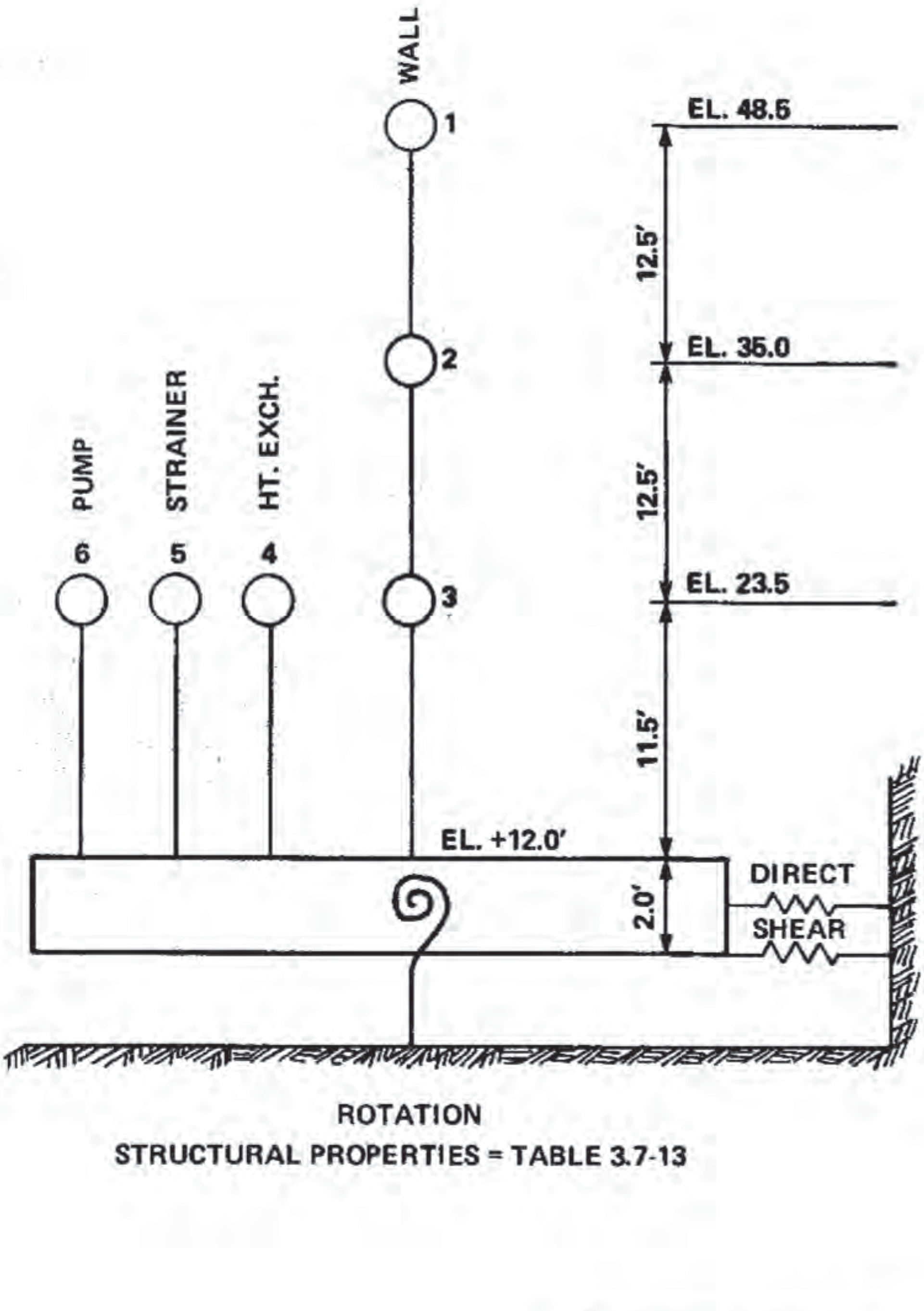
STEAM GENERATOR BLOWDOWN FACILITY
HORIZONTAL MATHEMATICAL MODEL
FIGURE 3.7-42



VERTICAL MATHEMATICAL MODEL

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

STEAM GENERATOR BLOWDOWN FACILITY
VERTICAL MATHEMATICAL MODEL
FIGURE 3.7-43



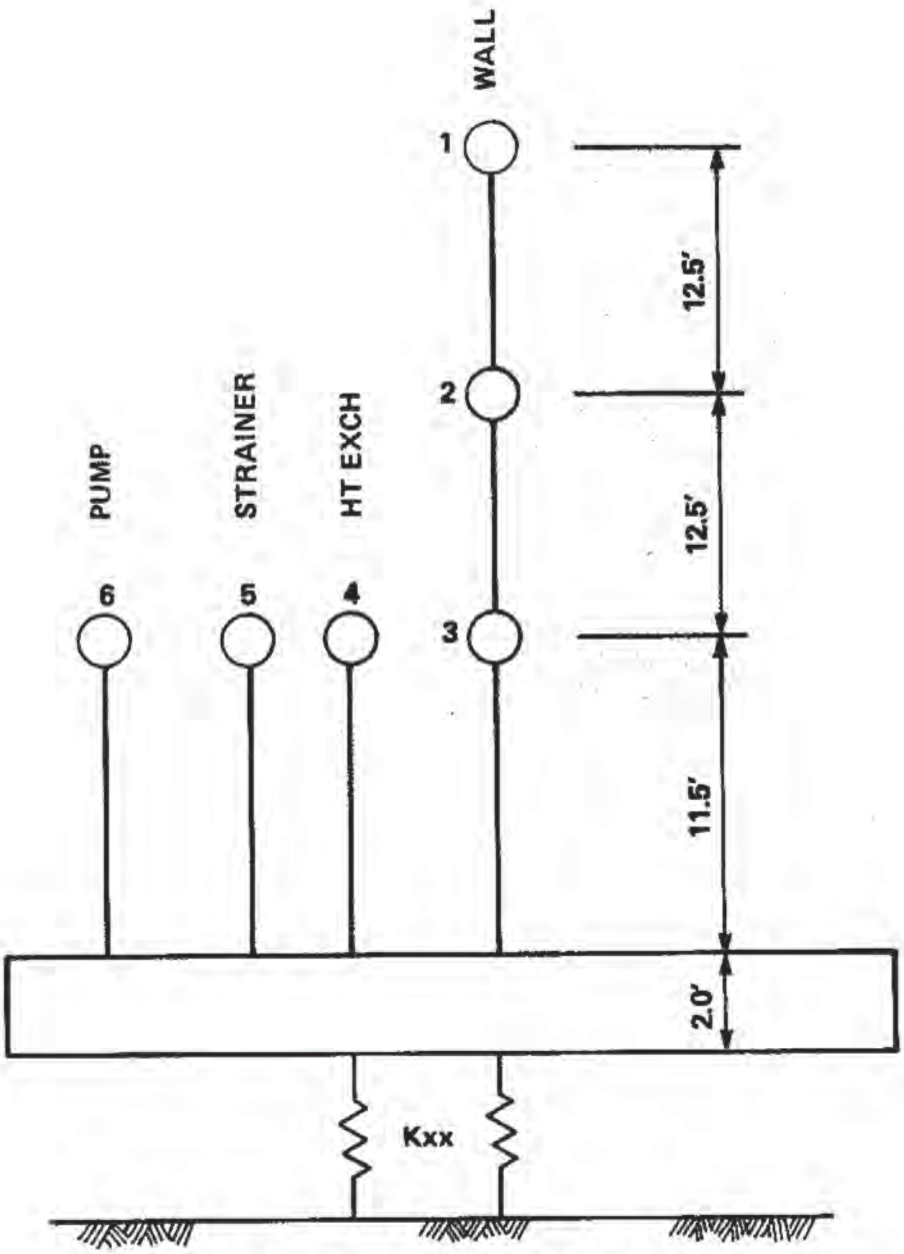
ROTATION

STRUCTURAL PROPERTIES = TABLE 3.7-13

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

COMPONENT COOLING WATER BLDG.
HORIZONTAL MODEL

FIGURE 3.7–44



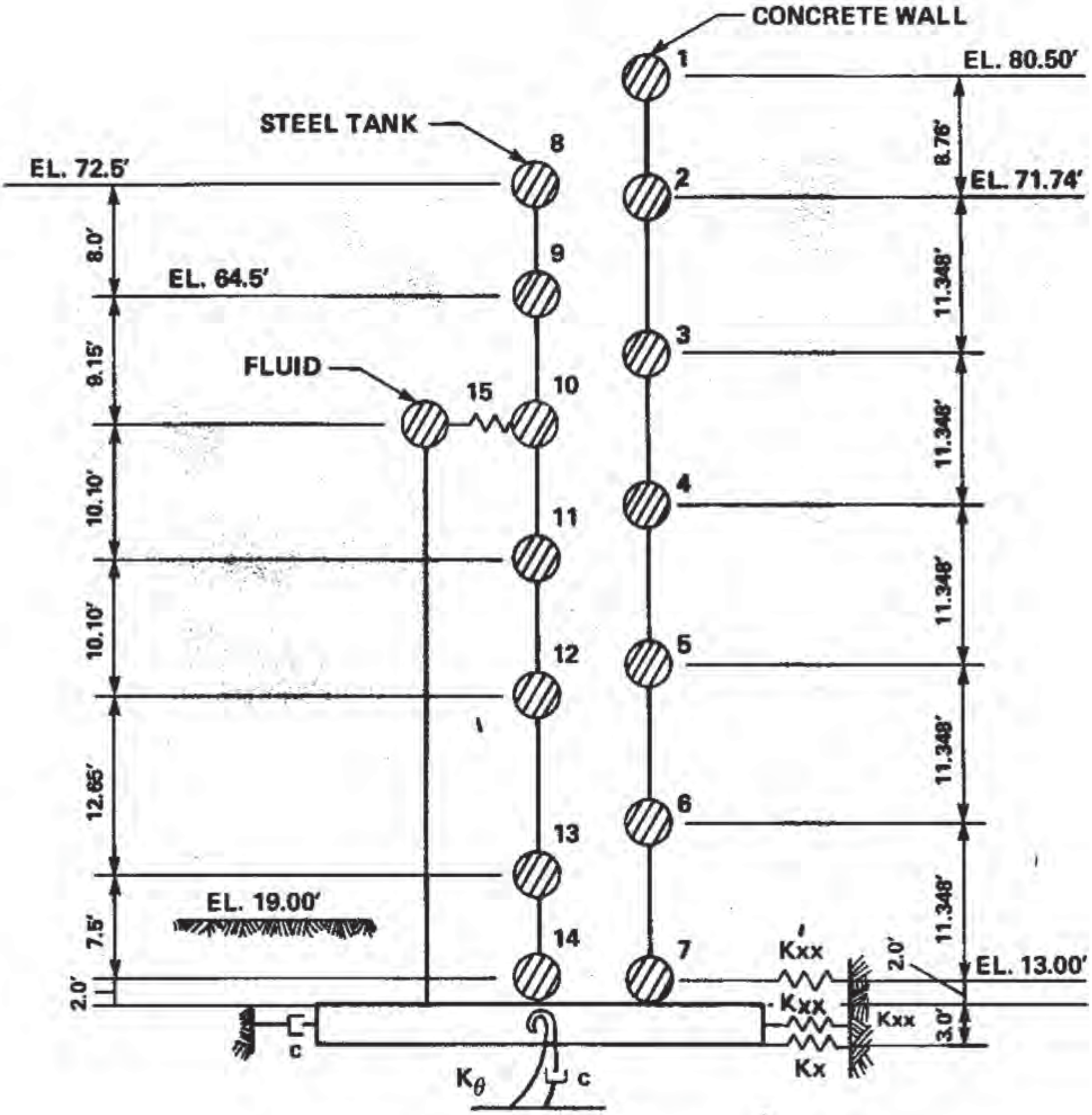
STRUCTURAL PROPERTIES = TABLE 3.7-14

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

COMPONENT COOLING WATER BLDG.
VERTICAL MODEL

FIGURE 3.7–45

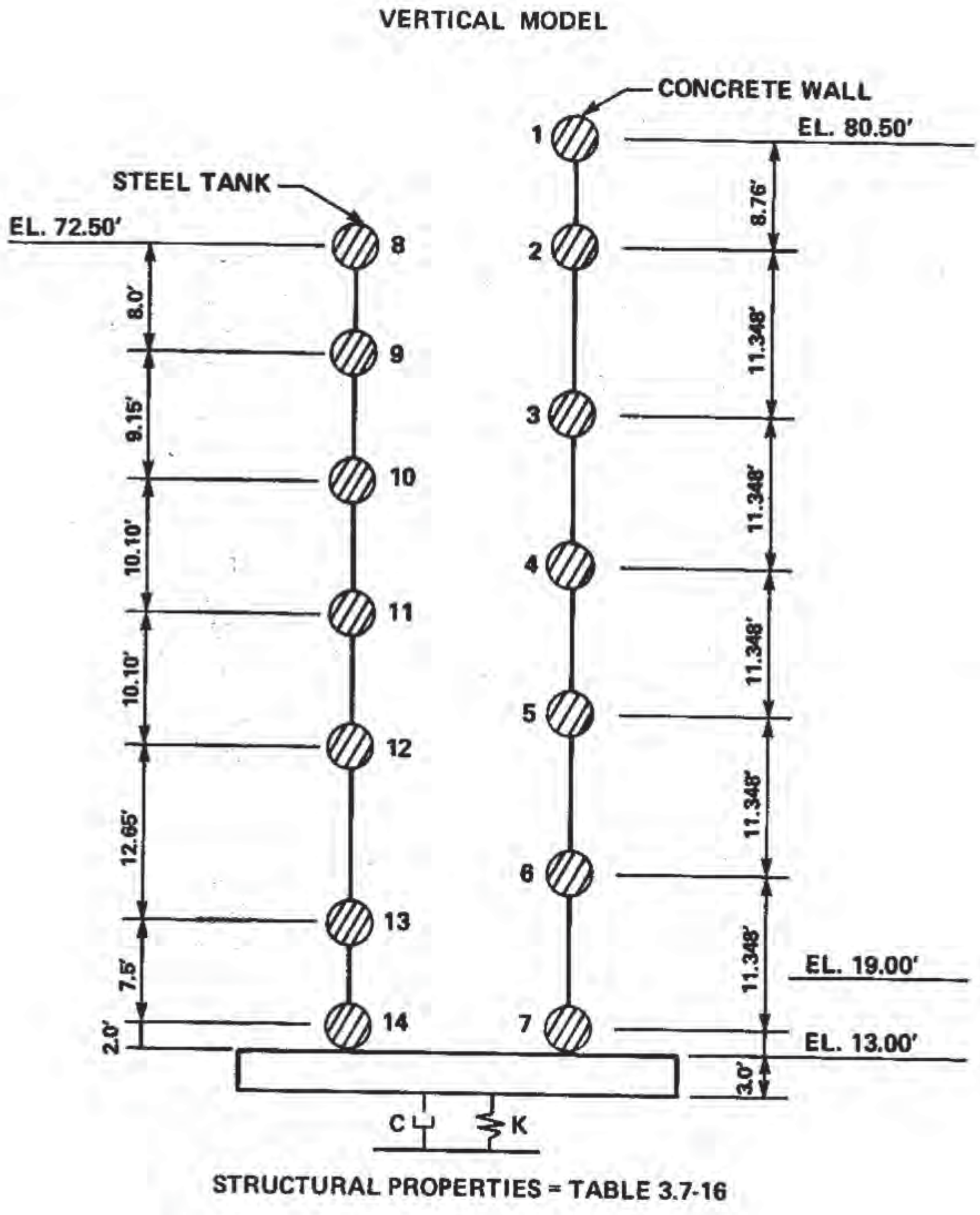
HORIZONTAL MODEL



STRUCTURAL PROPERTIES = TABLE 3.7-15

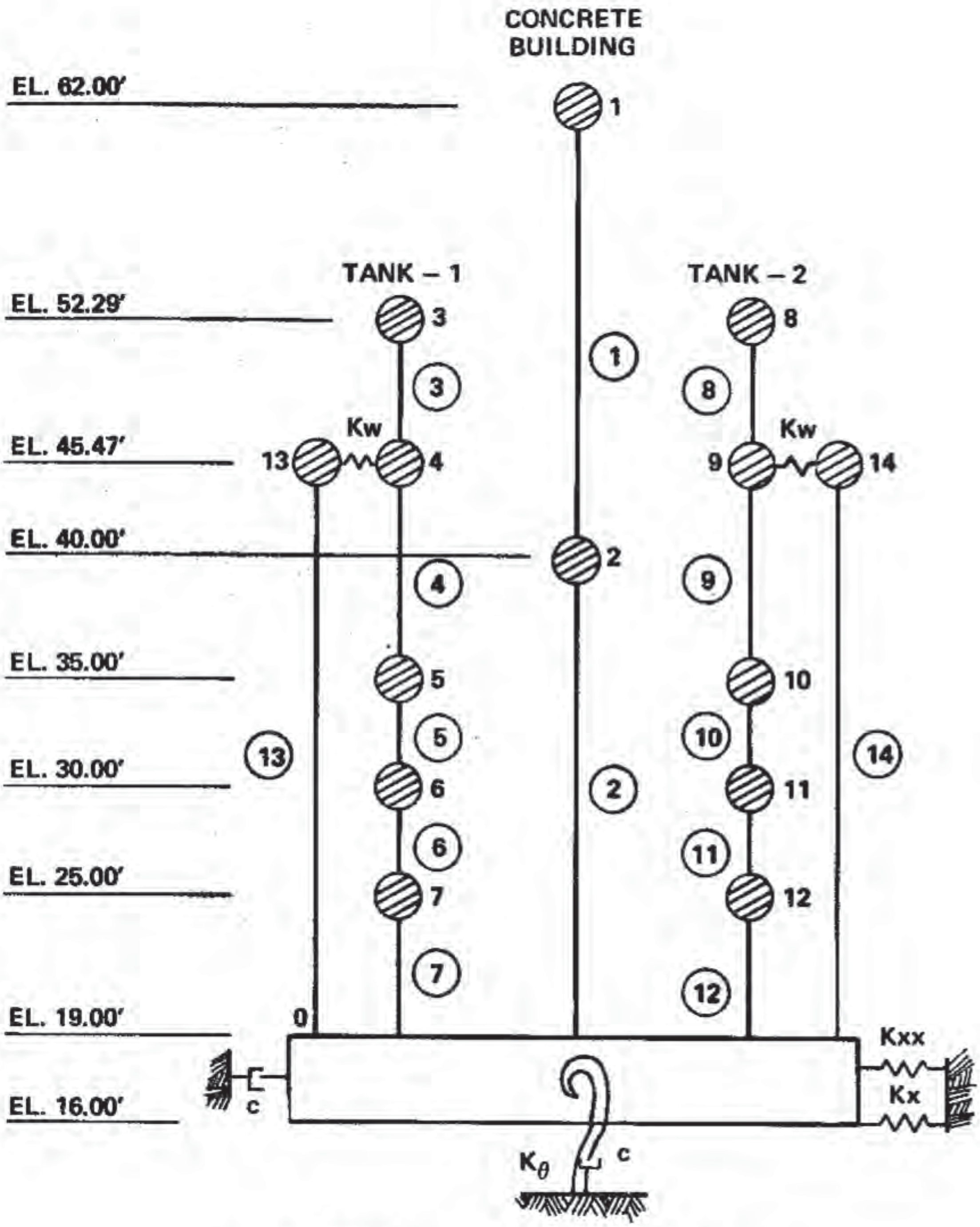
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONDENSATE STORAGE TANK BLDG.
HORIZONTAL MODEL
FIGURE 3.7–46



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONDENSATE STORAGE TANK BLDG.
VERTICAL MODEL
FIGURE 3.7-47

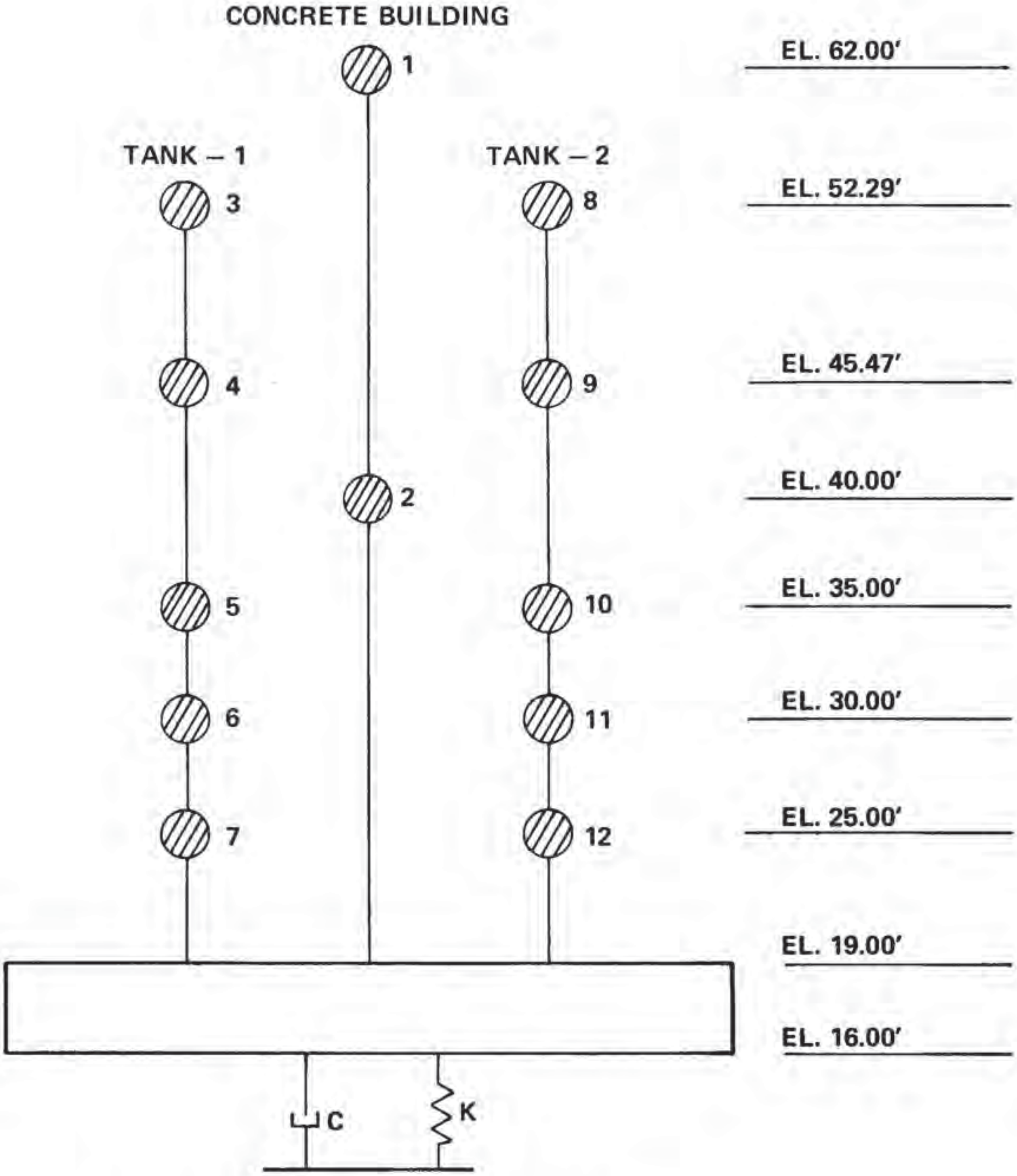


STRUCTURAL PROPERTIES – TABLE 3.7-17

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

DIESEL OIL STORAGE BLDG.
HORIZONTAL MODEL

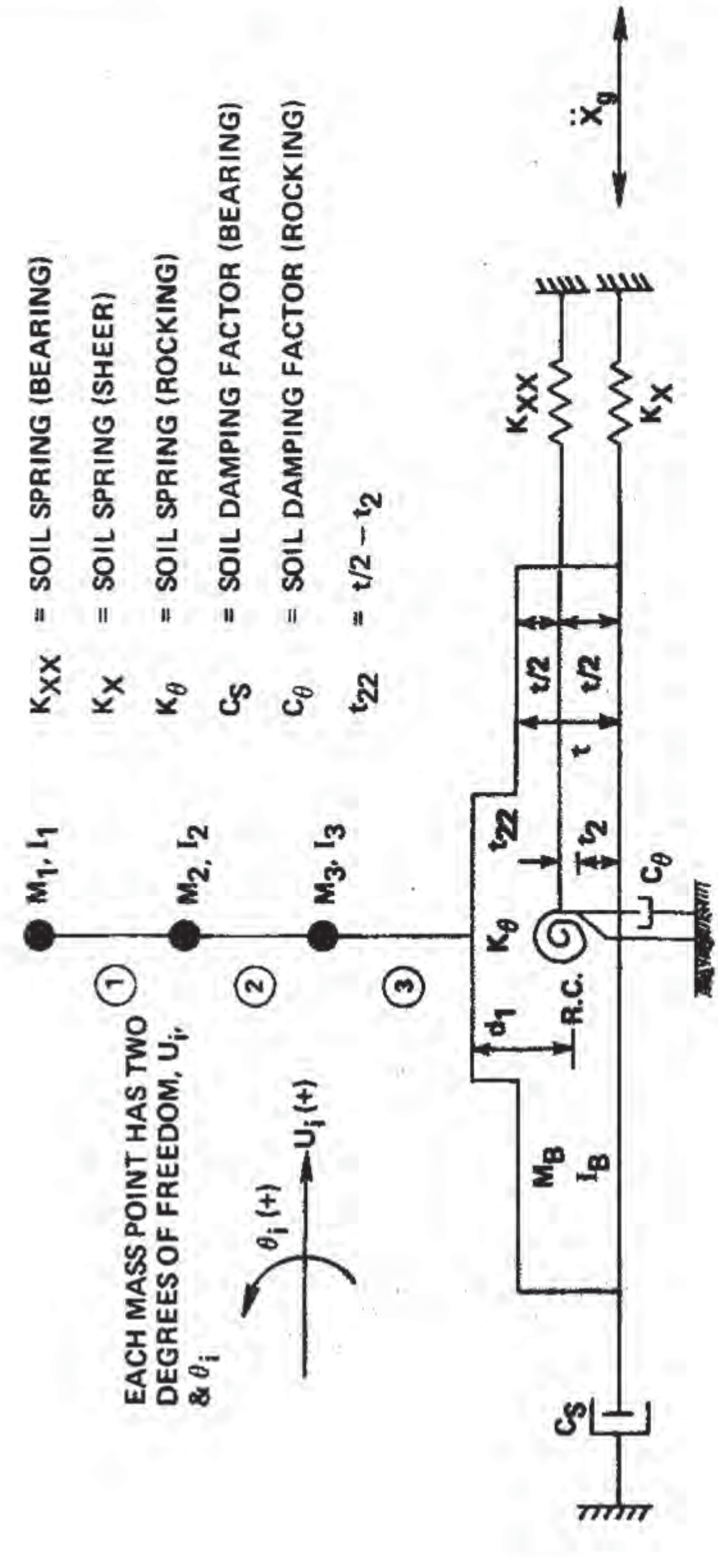
FIGURE 3.7–48



STRUCTURAL PROPERTIES – TABLE 3.7-17

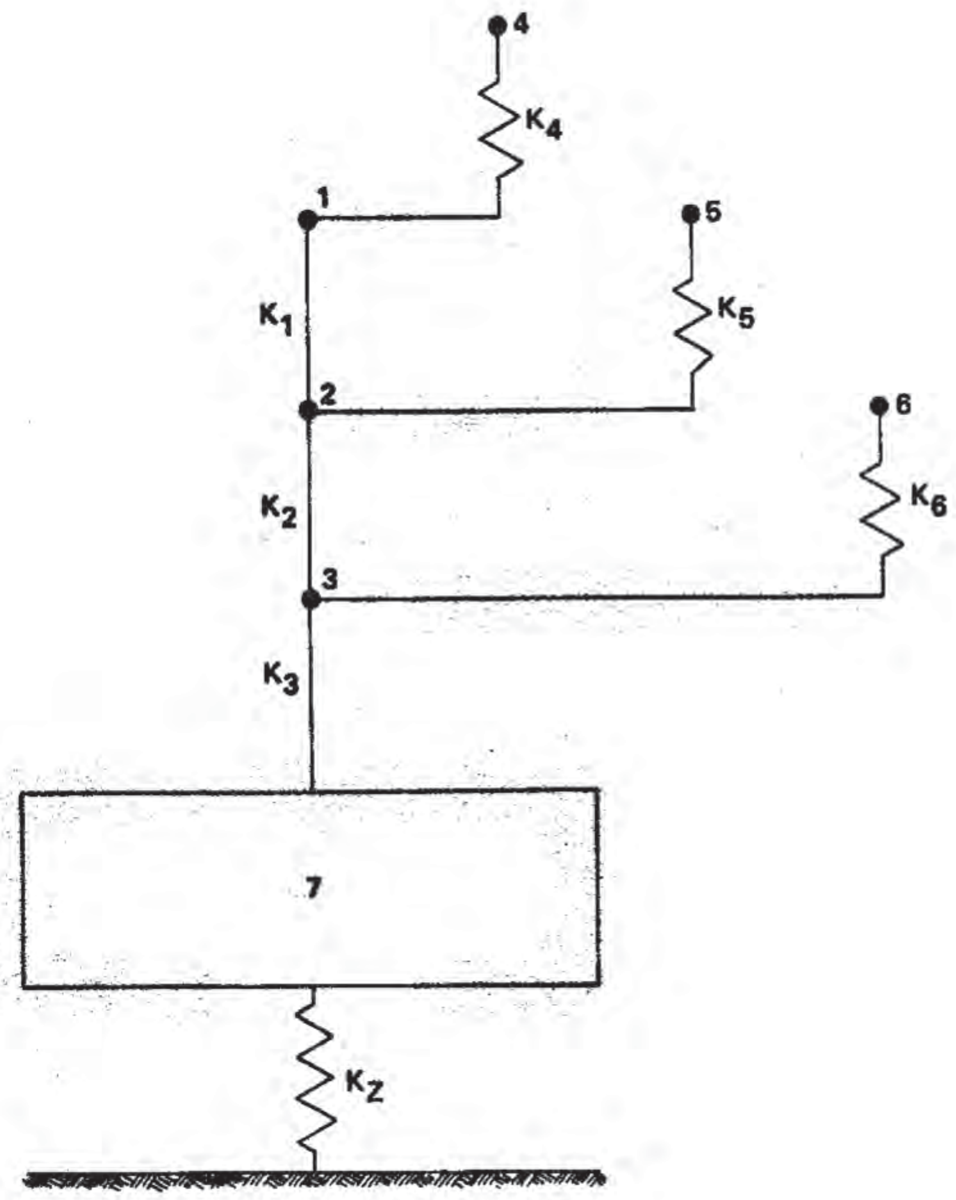
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

DIESEL OIL STORAGE BLDG.
VERTICAL MODEL
FIGURE 3.7–49



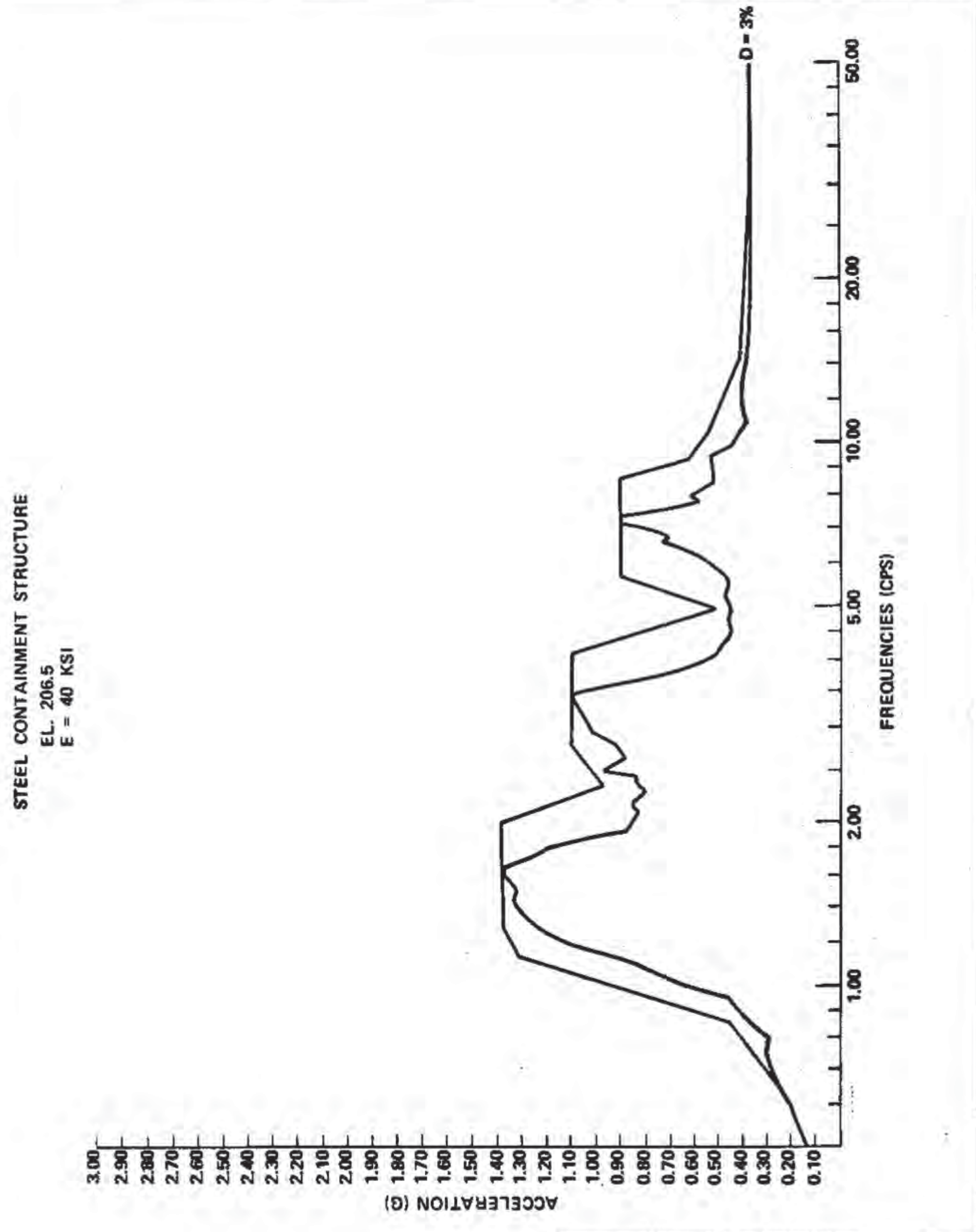
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TYPICAL HORIZONTAL
DYNAMIC MODEL
FIGURE 3.7-50



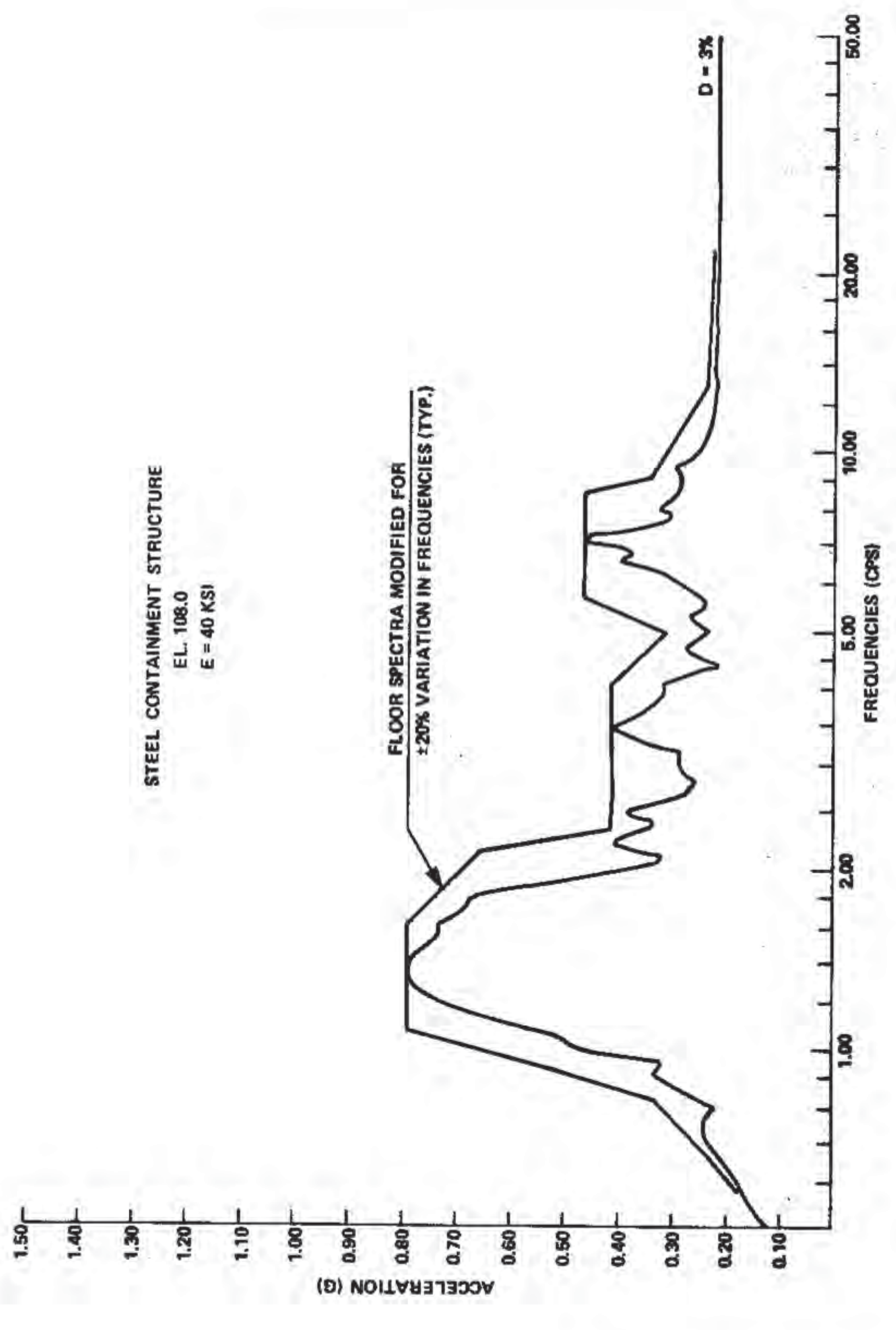
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TYPICAL VERTICAL
DYNAMIC MODEL
FIGURE 3.7-51



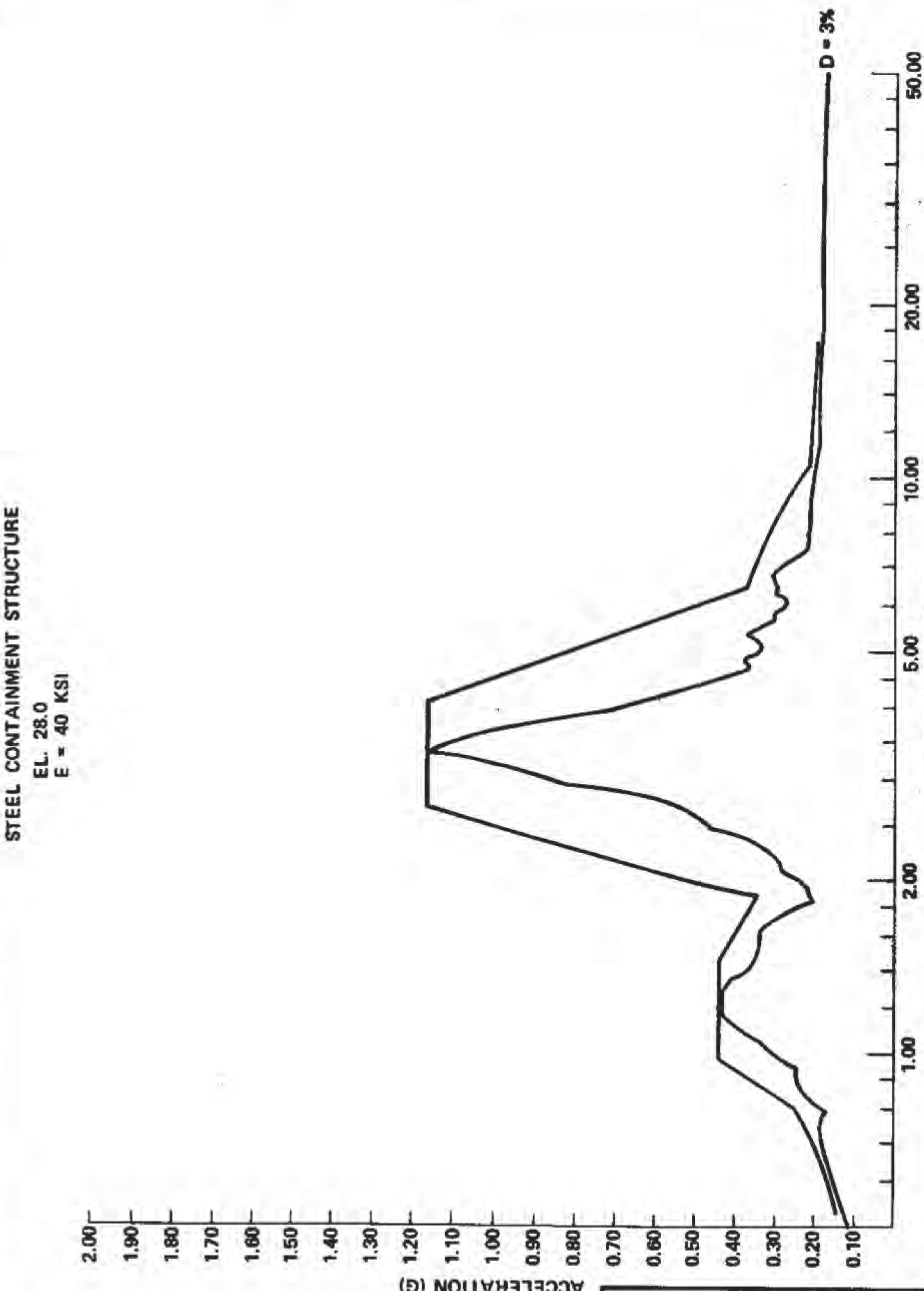
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S & E-W SSE
REACTOR BLDG MASS PT 11
FIGURE 3.7-52



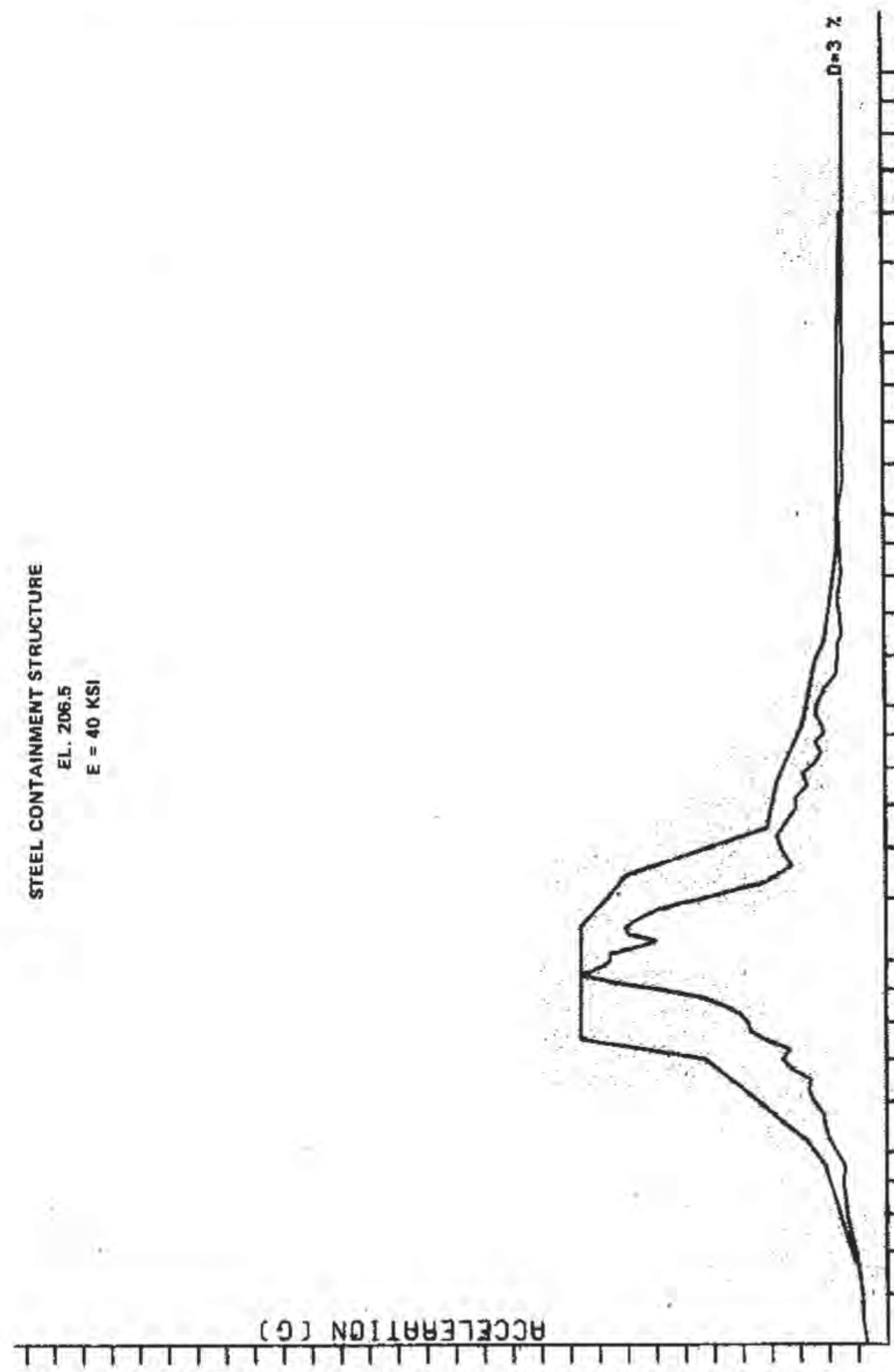
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S & E-W SSE
REACTOR BLDGMASS PT.16
FIGURE 3.7–53



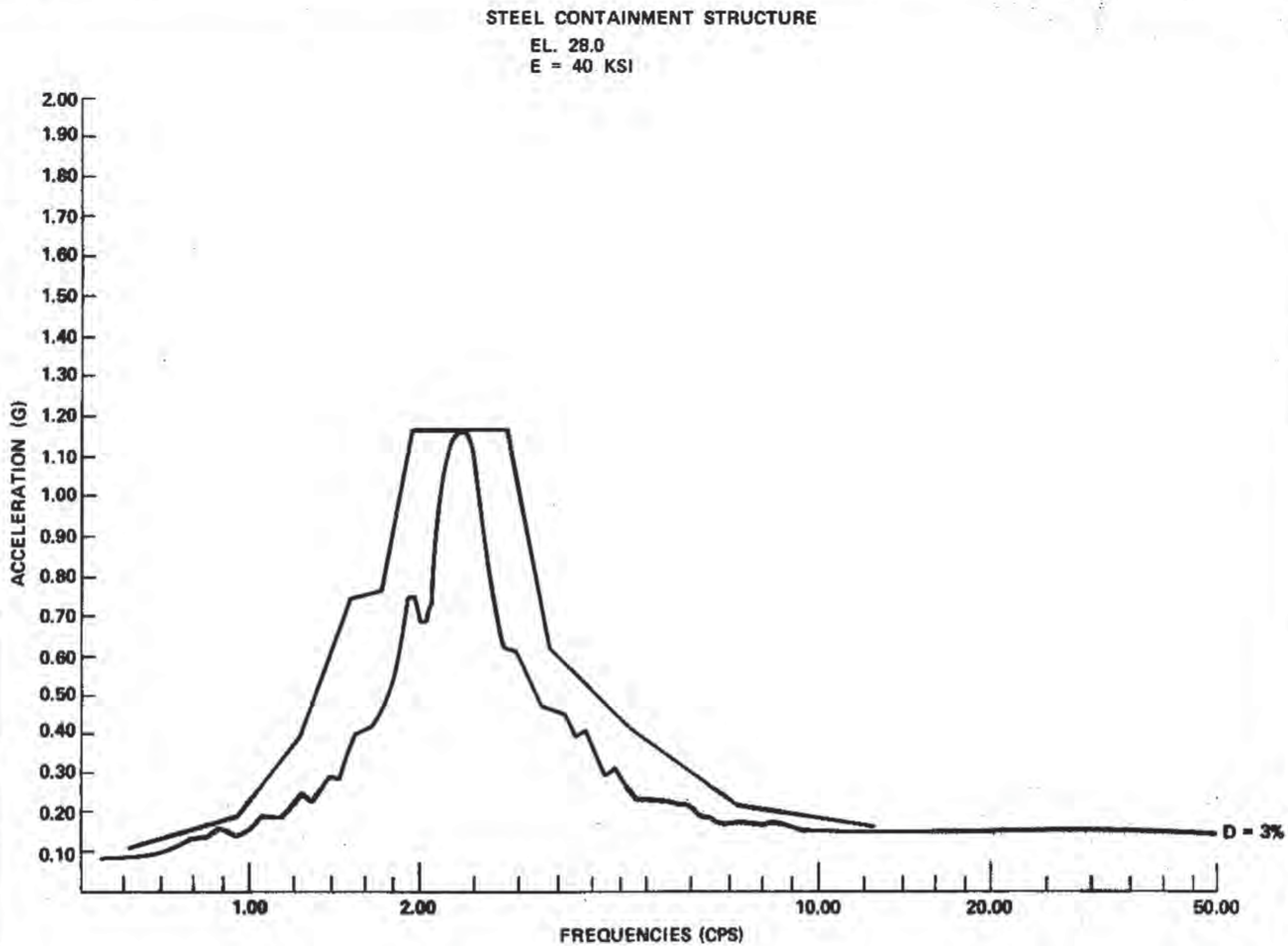
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S & E-W SSE
REACTOR BLDG MASS PT 22
FIGURE 3.7-54



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

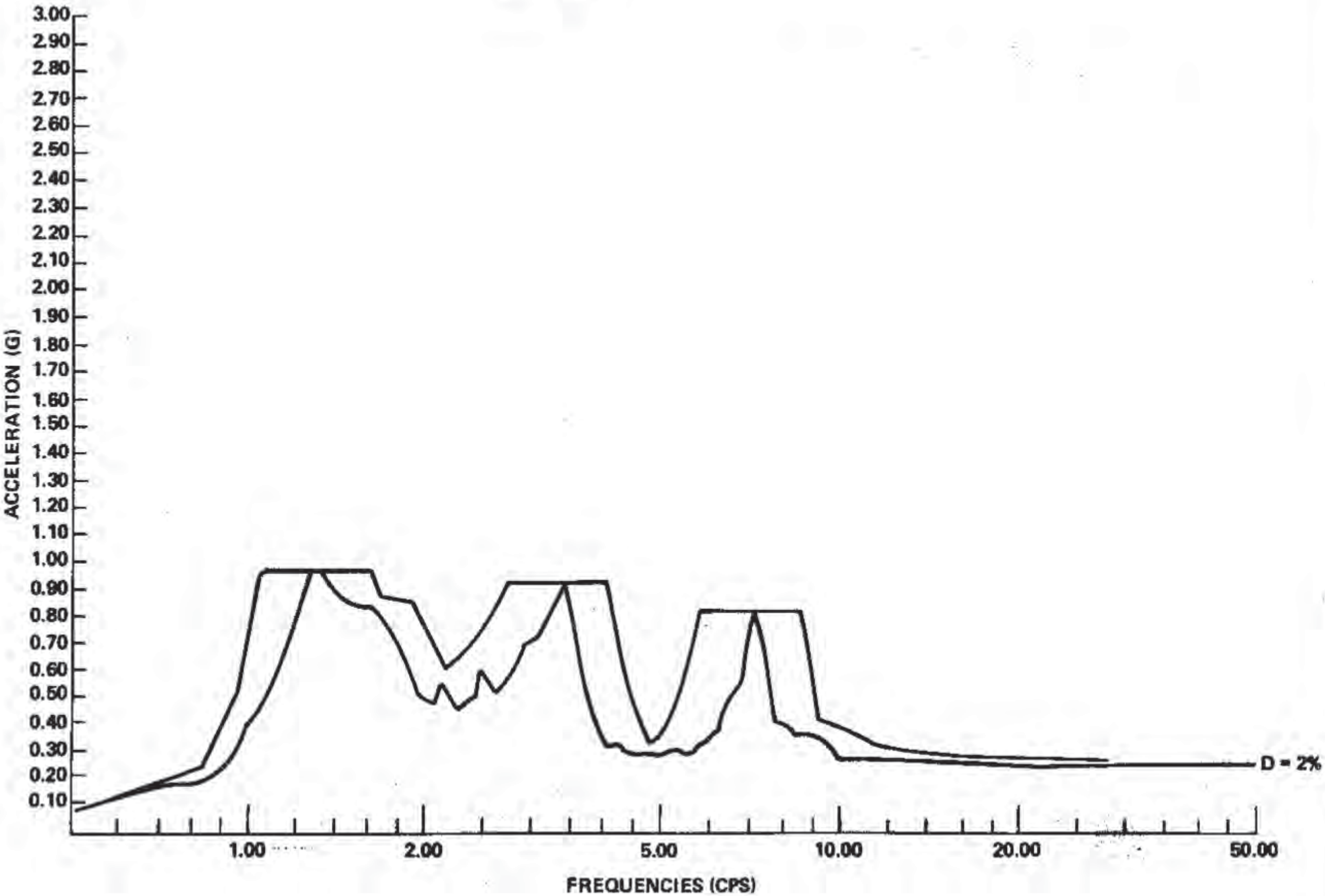
FLOOR SPECTRA VERTICAL SSE
REACTOR BLDG. MASS PT. 6
FIGURE 3.7–55



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

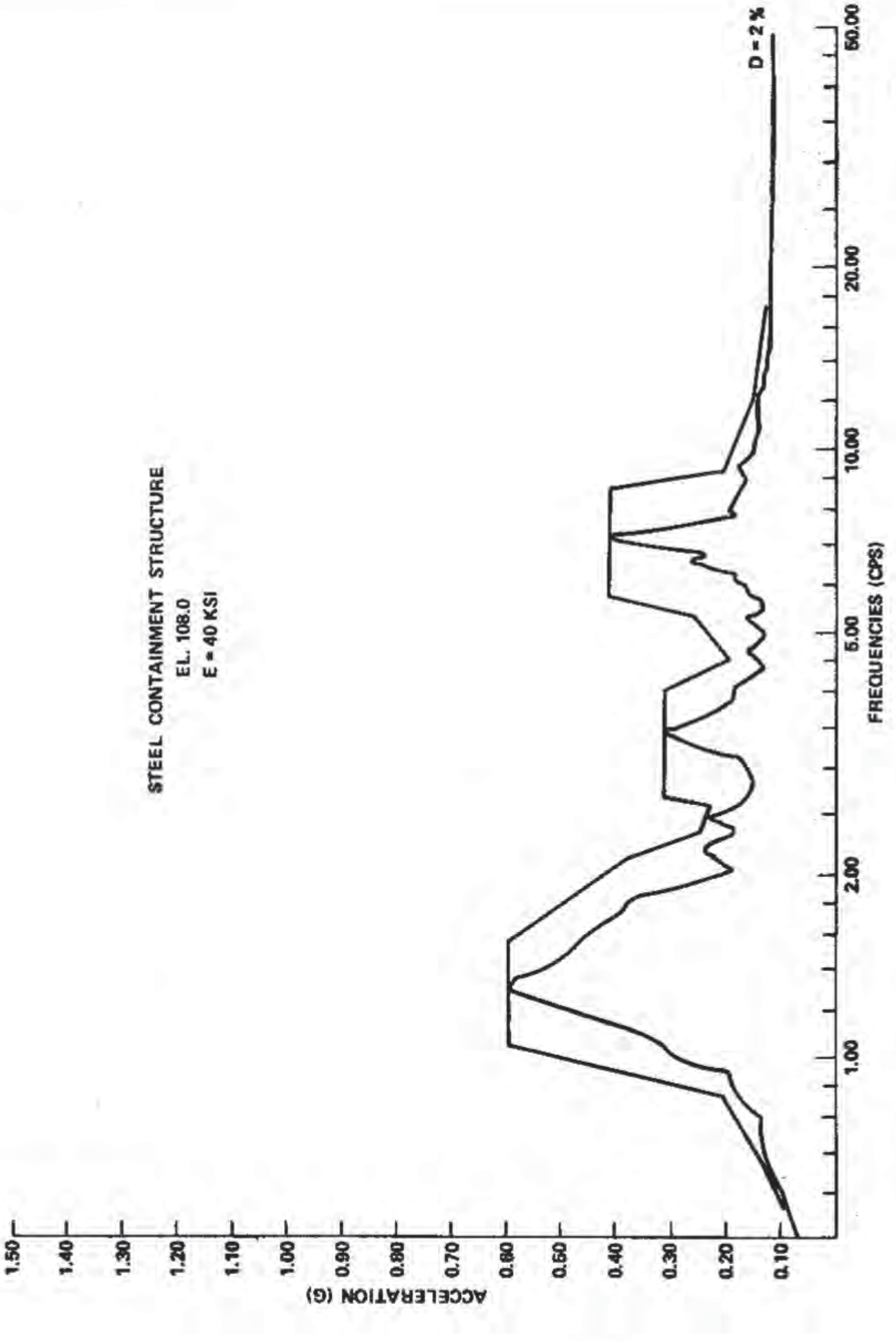
FLOOR SPECTRA VERTICAL SSE
REACTOR BLDG MASS PT 10
FIGURE 3.7-56

STEEL CONTAINMENT STRUCTURE
EL. 206.5
E = 40 KSI



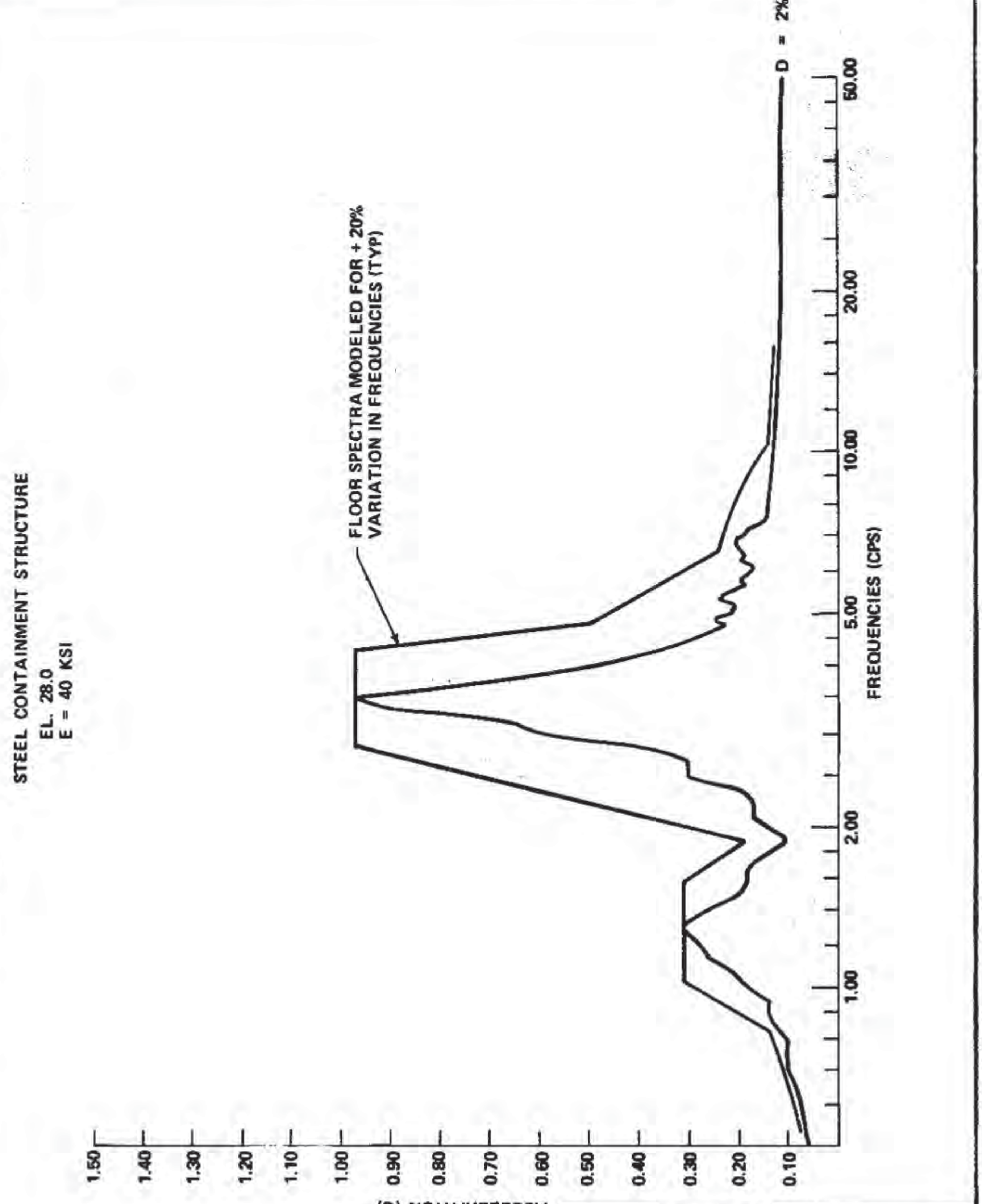
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S & E-W OBE
REACTOR BUILDING MASS PT 11
FIGURE 3.7-57



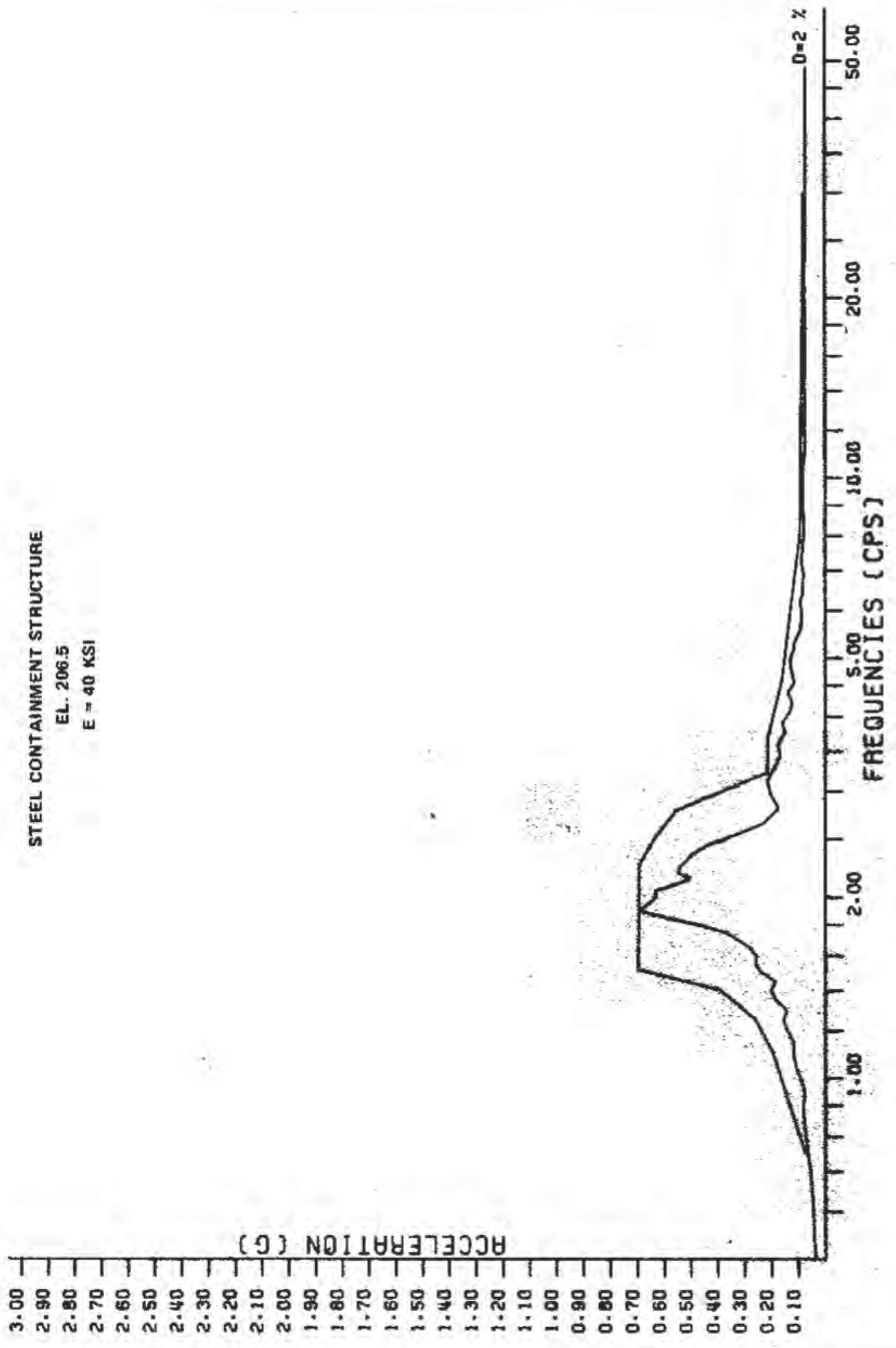
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S & E-W OBE
REACTOR BLDG. MASS PT. 16
FIGURE 3.7–58



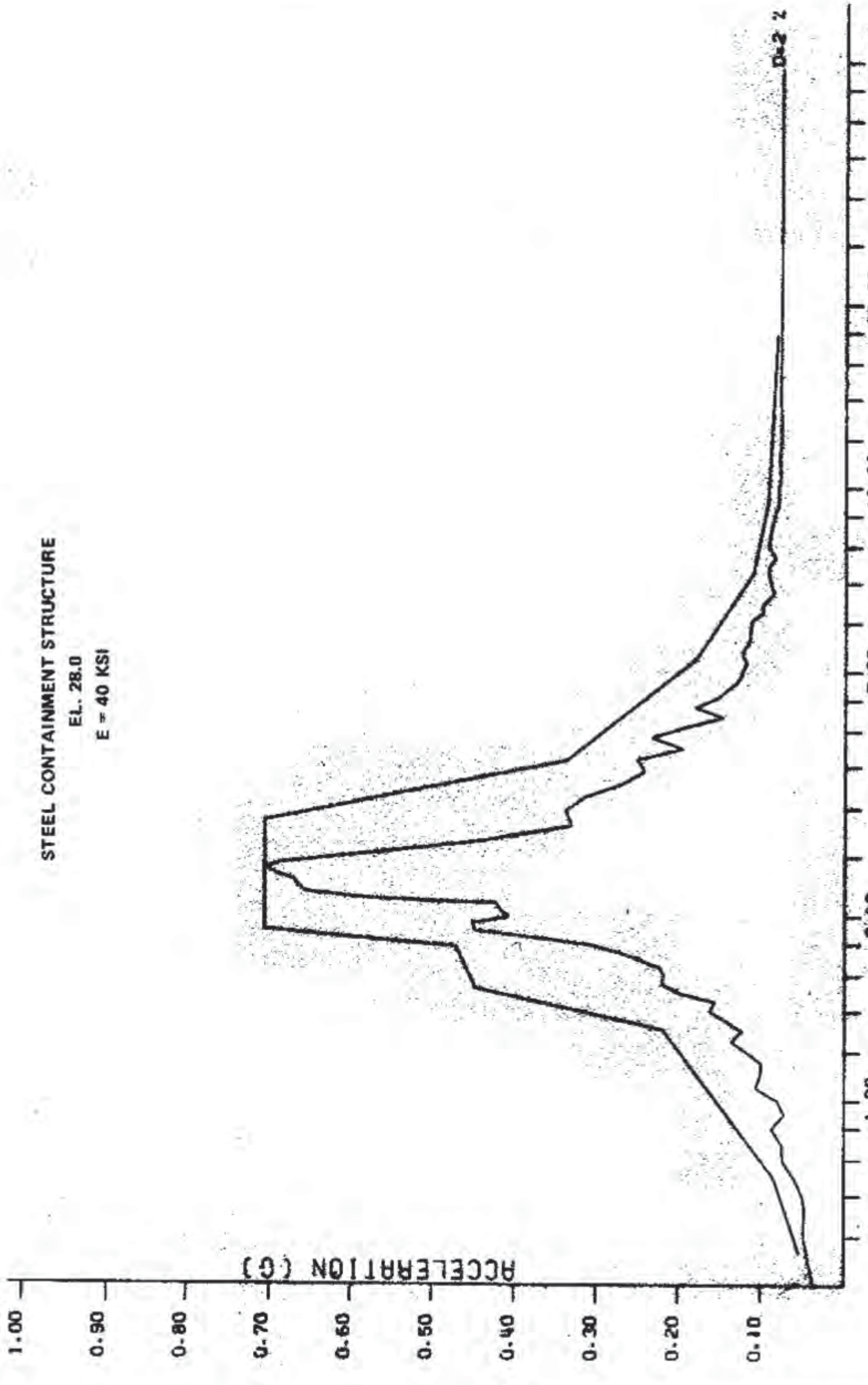
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S & E-W OBE
REACTOR BLDG MASS PT 22
FIGURE 3.7-59



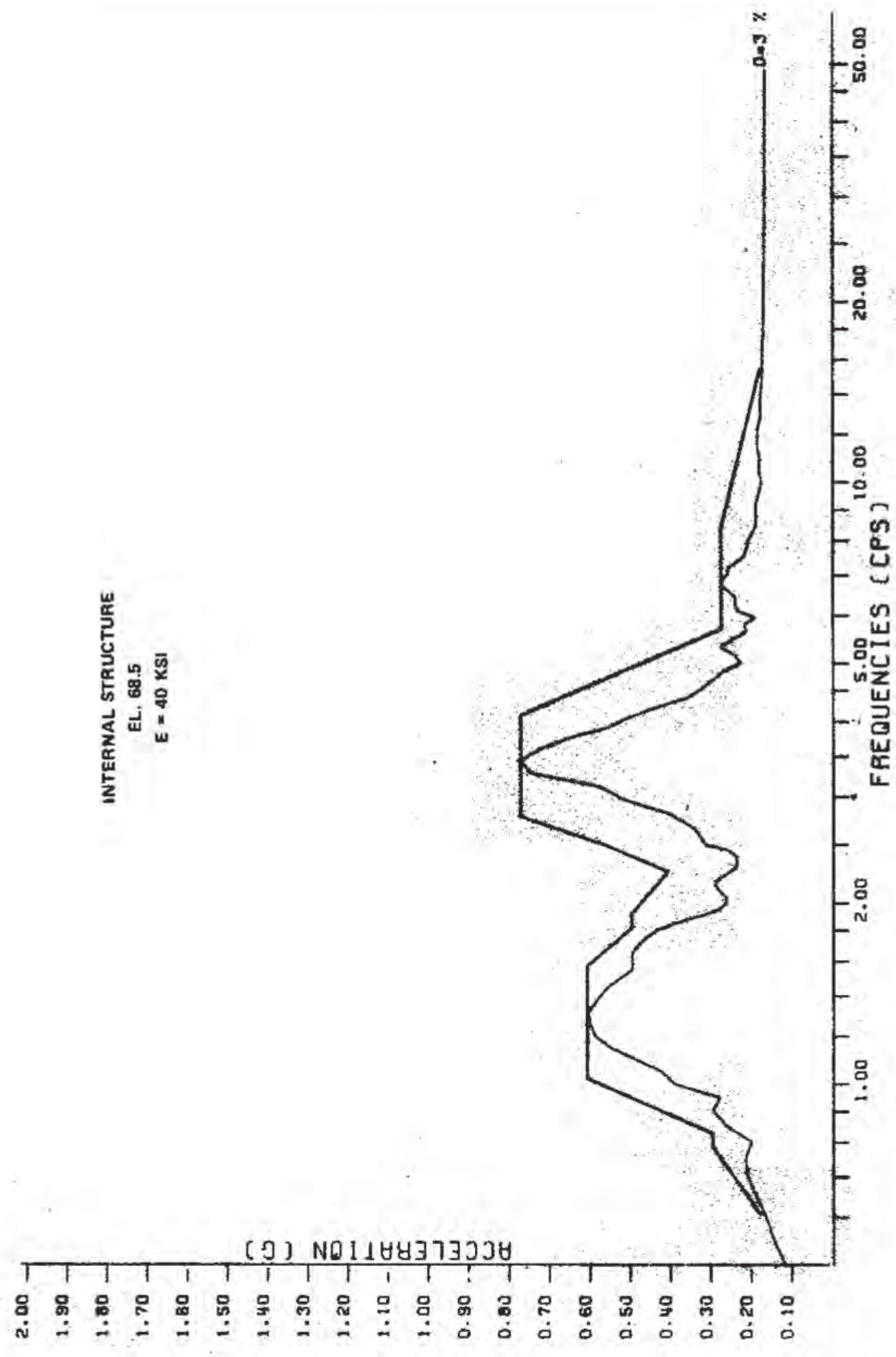
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG. MASS PT. 6
FIGURE 3.7–60



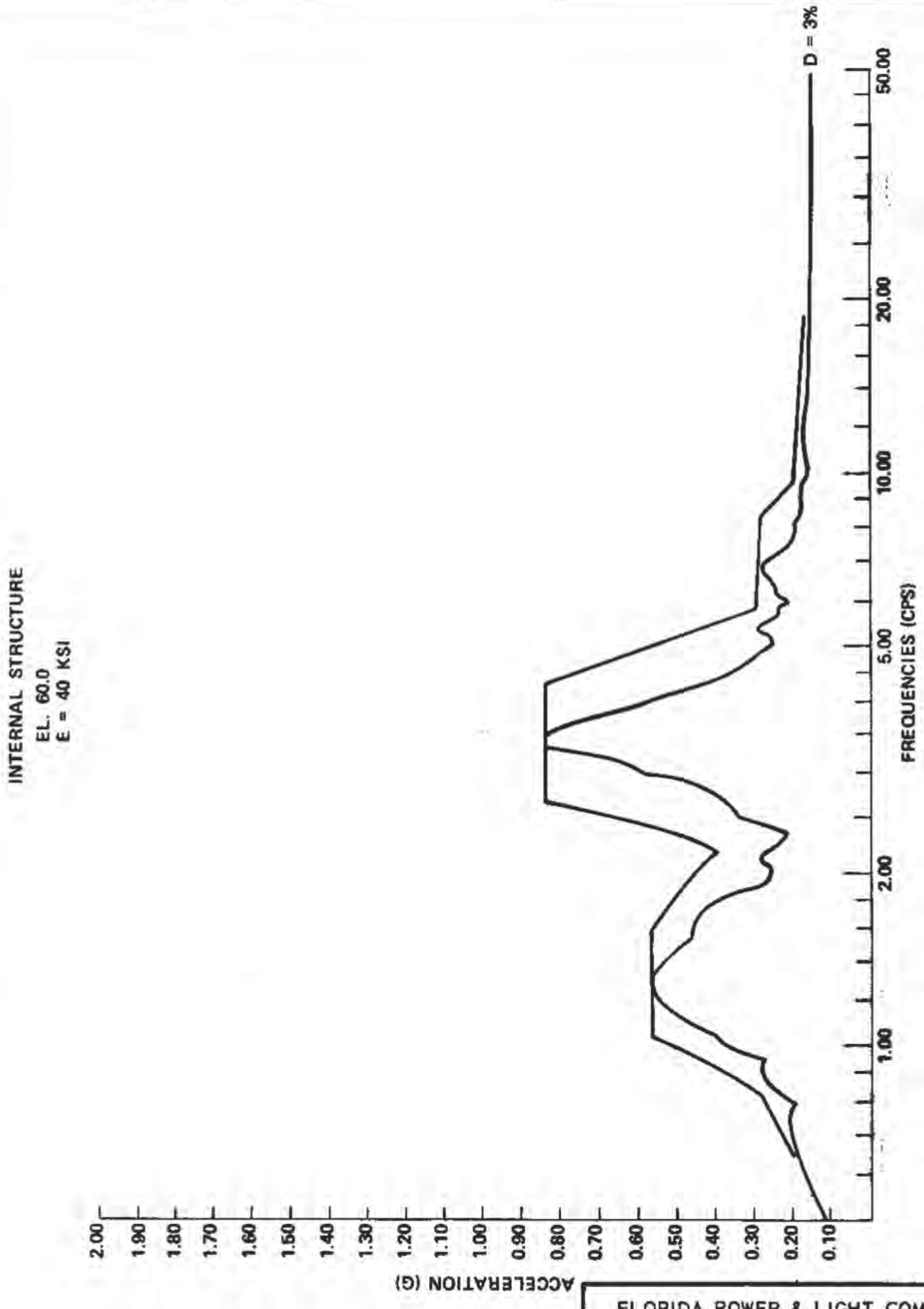
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG. MASS PT. 10
FIGURE 3.7-61



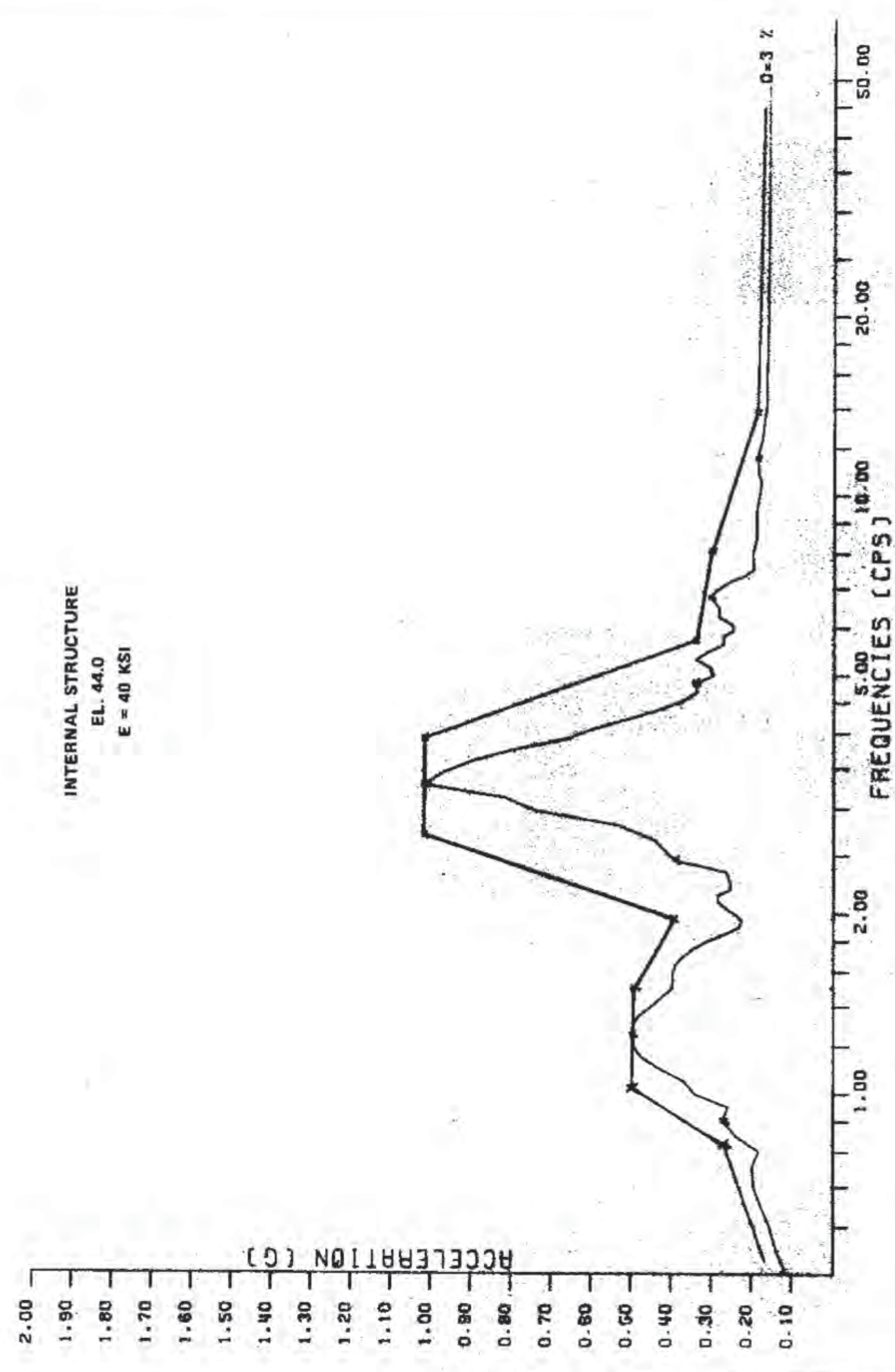
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S SSE
REACTOR BLDG. MASS PT.21
FIGURE 3.7-62



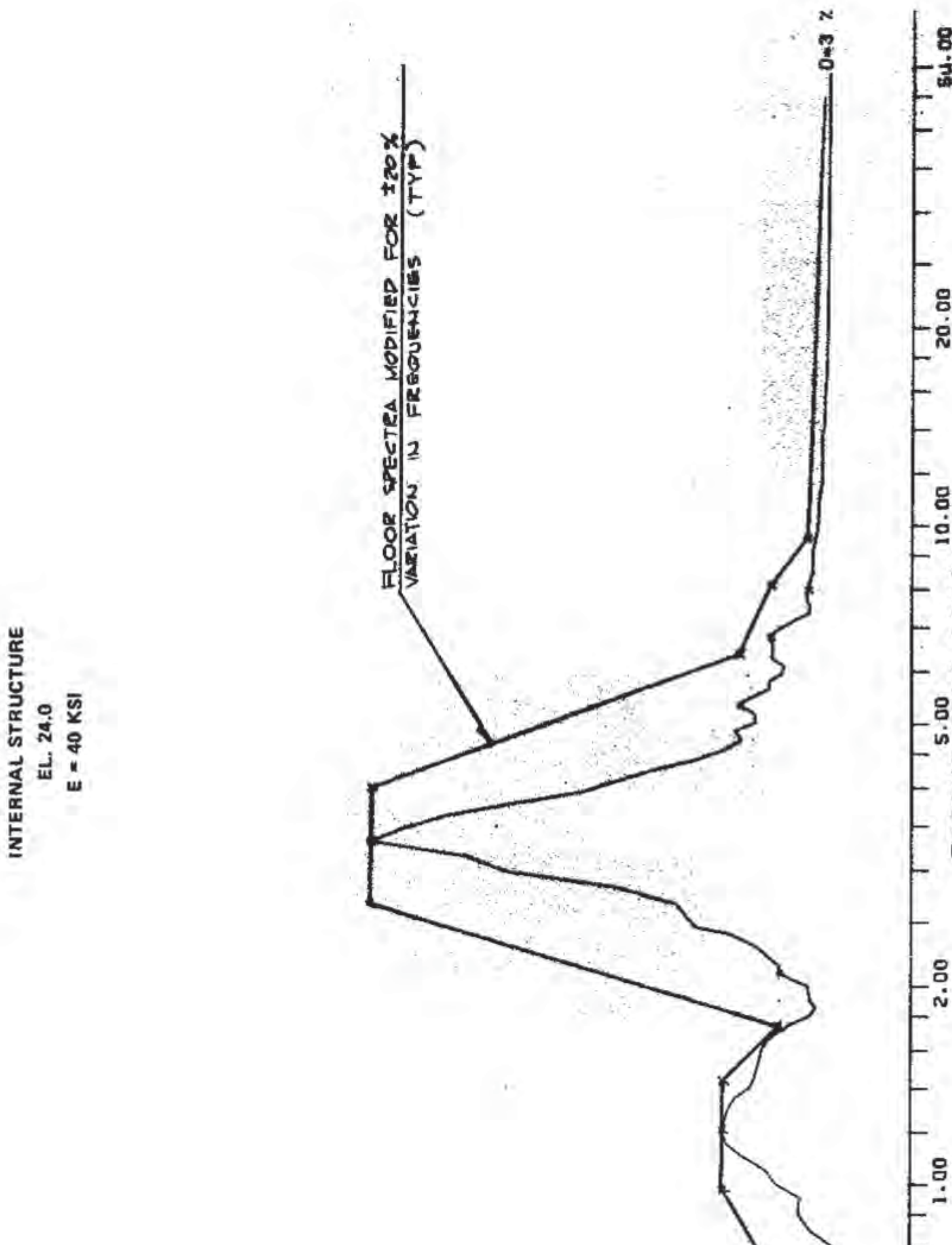
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S AND E-W SSE
REACTOR BUILDING MASS PT 22
FIGURE 3.7-63



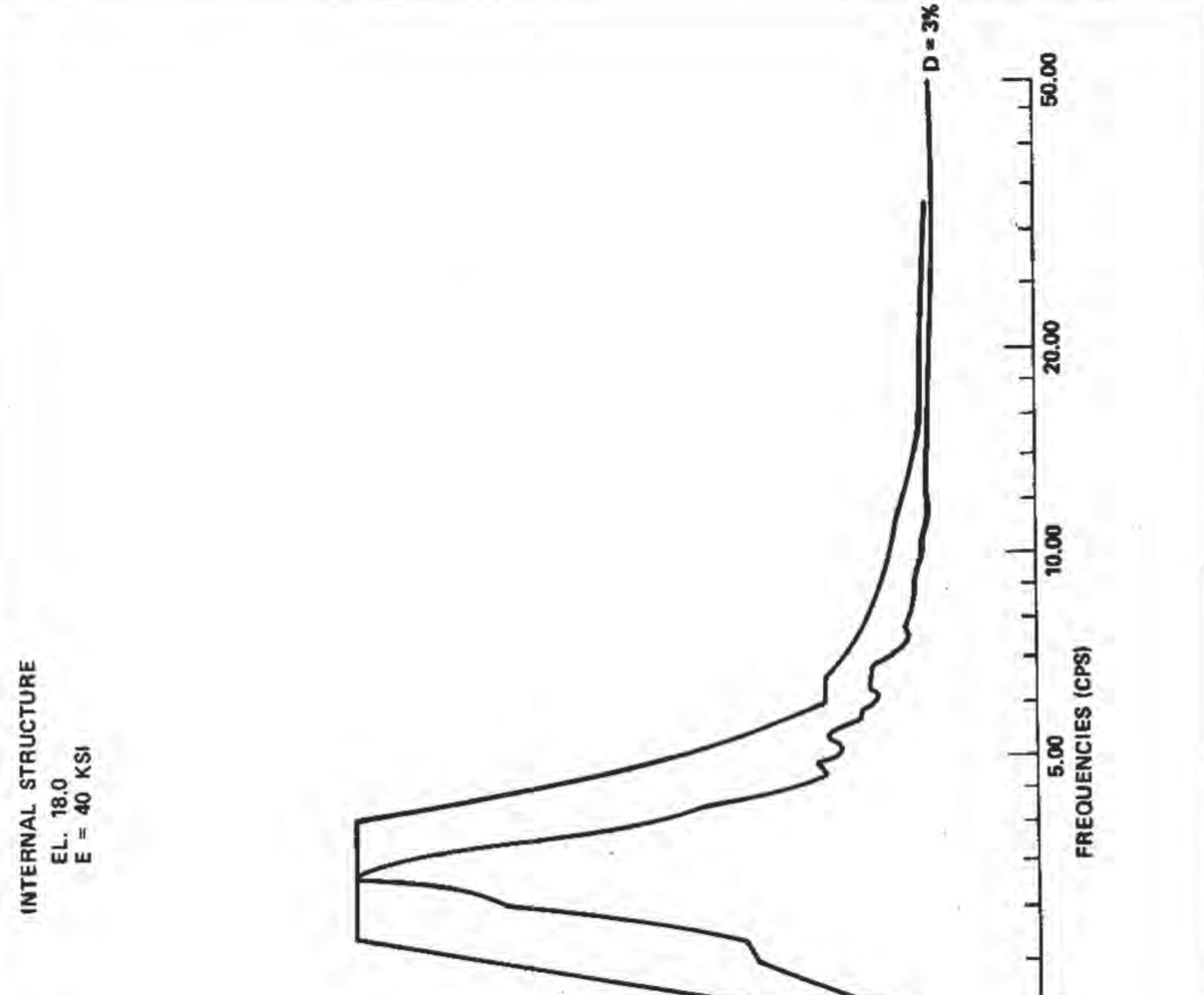
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S SSE
REACTOR BLDG. MASS PT.23
FIGURE 3.7–64



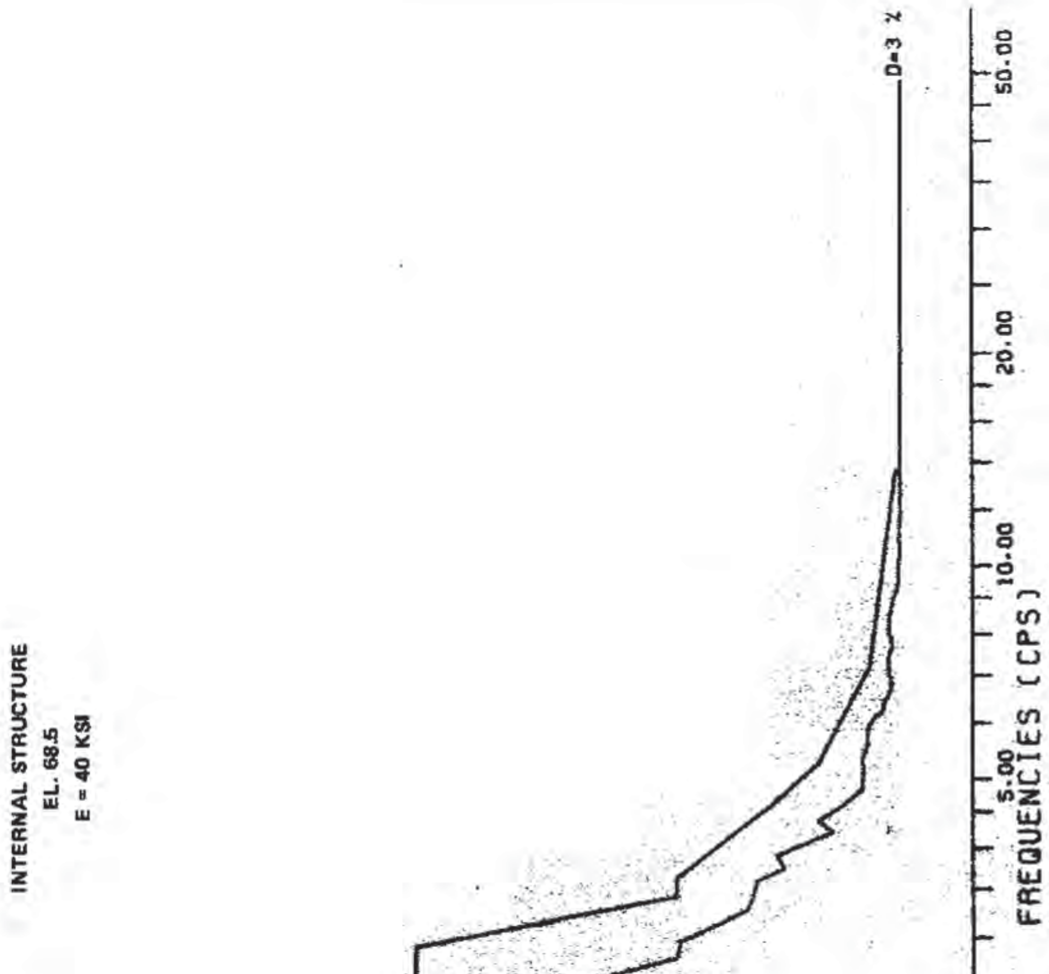
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
REACTOR BLDG. MASS PT. 24
FIGURE 3.7–65



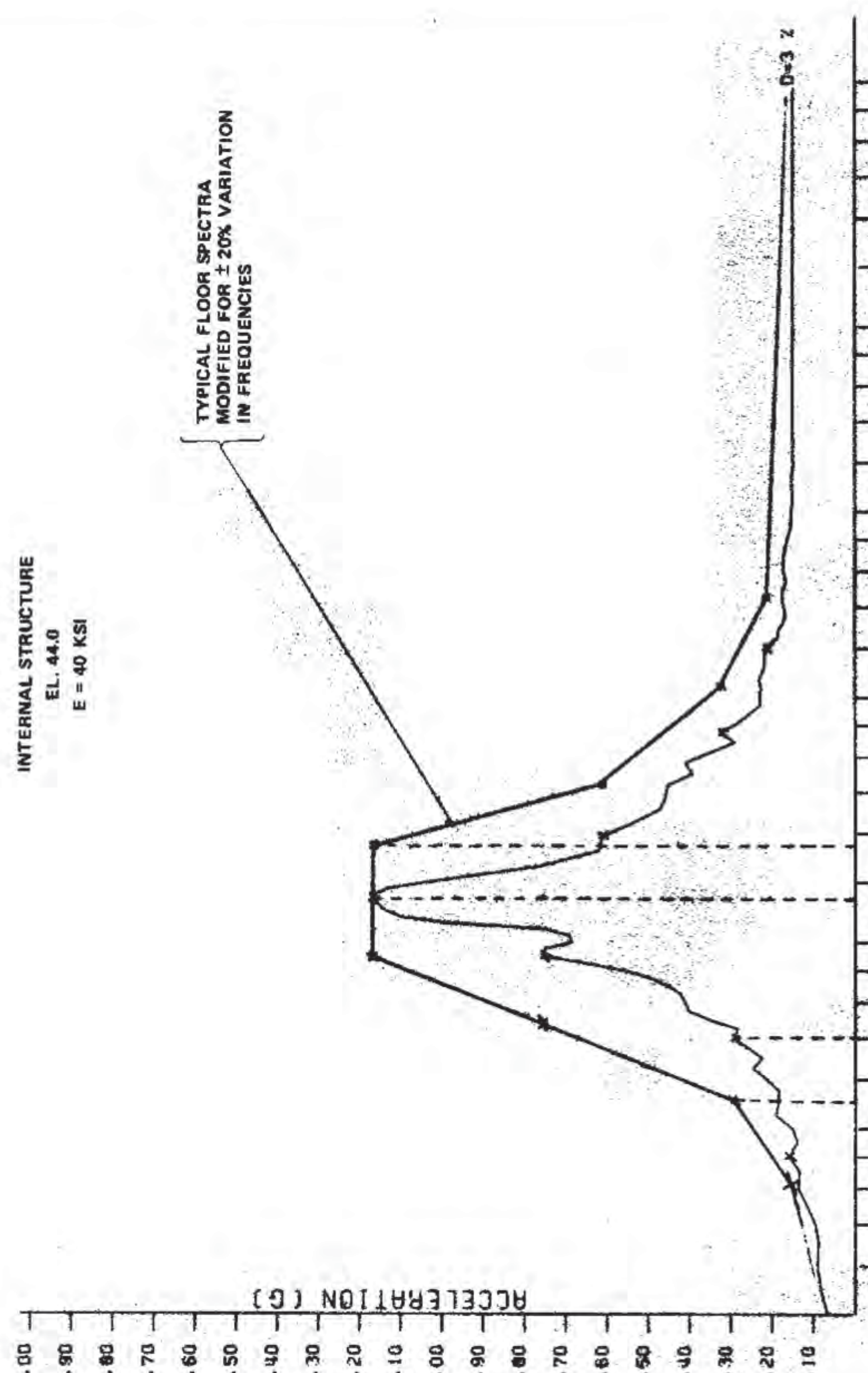
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S & E-W SSE
REACTOR BLDG MASS PT 27
FIGURE 3.7-66



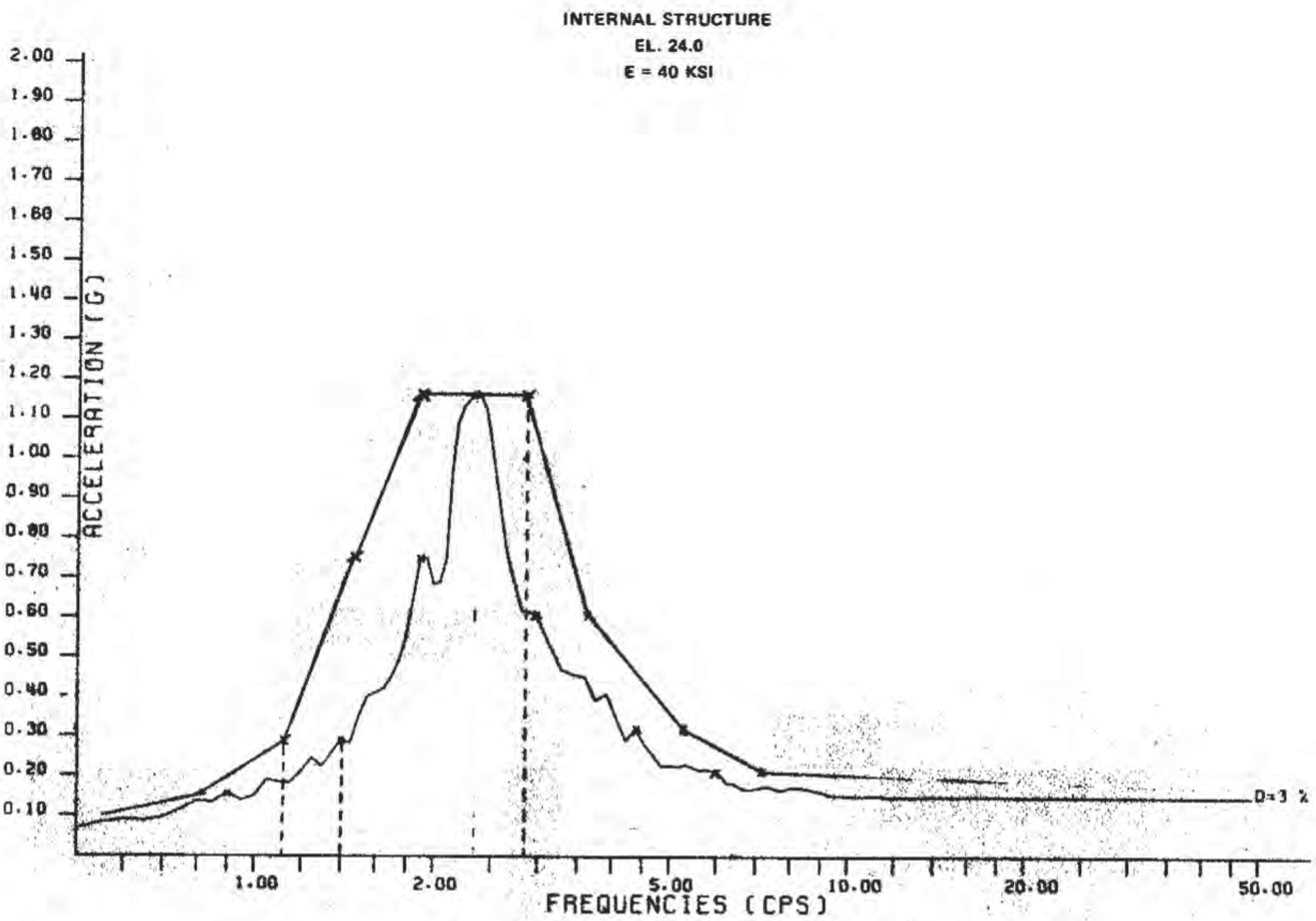
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR BLDG. MASS PT.11
FIGURE 3.7–67



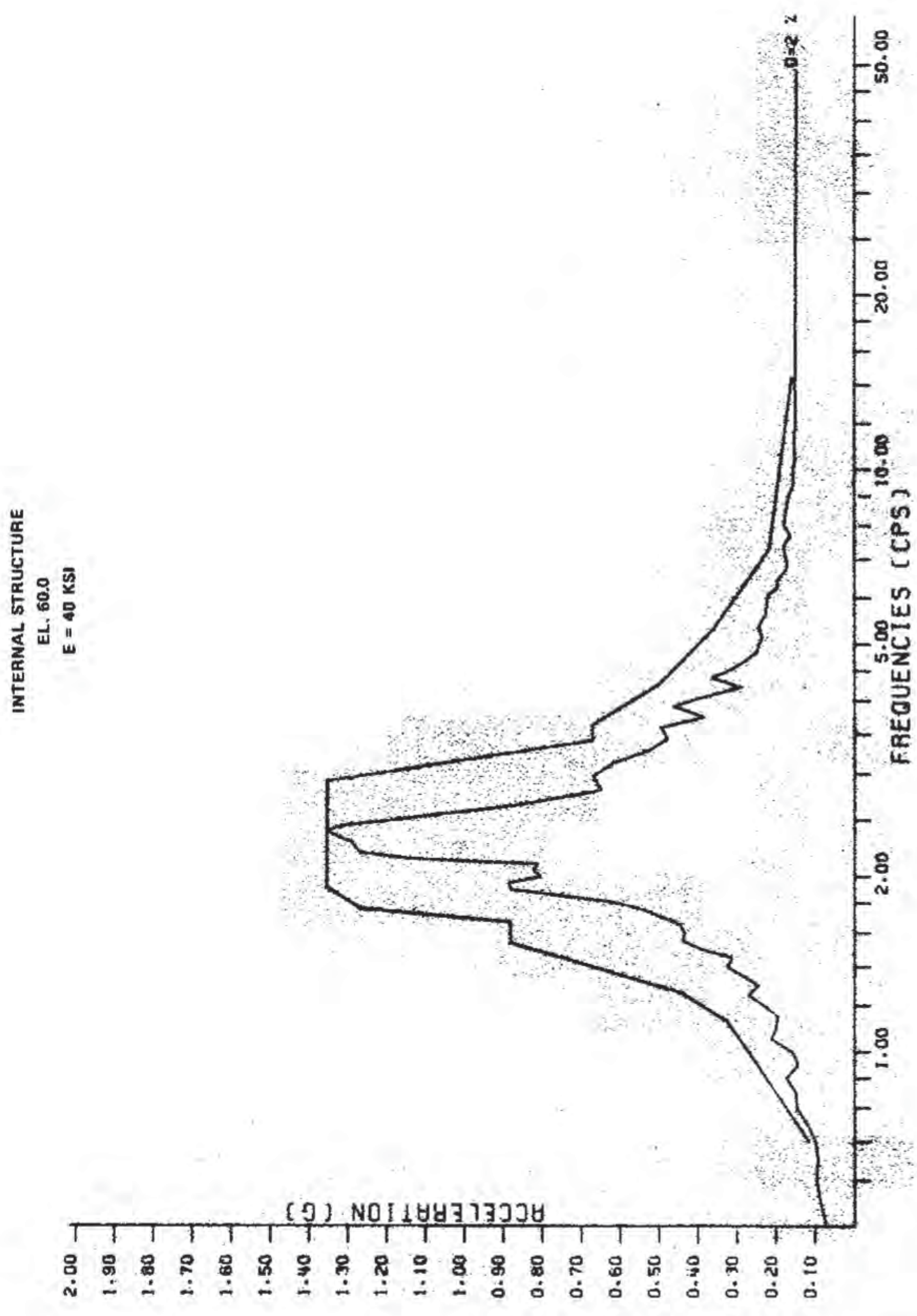
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR BLDG. MASS PT. 13
FIGURE 3.7–68



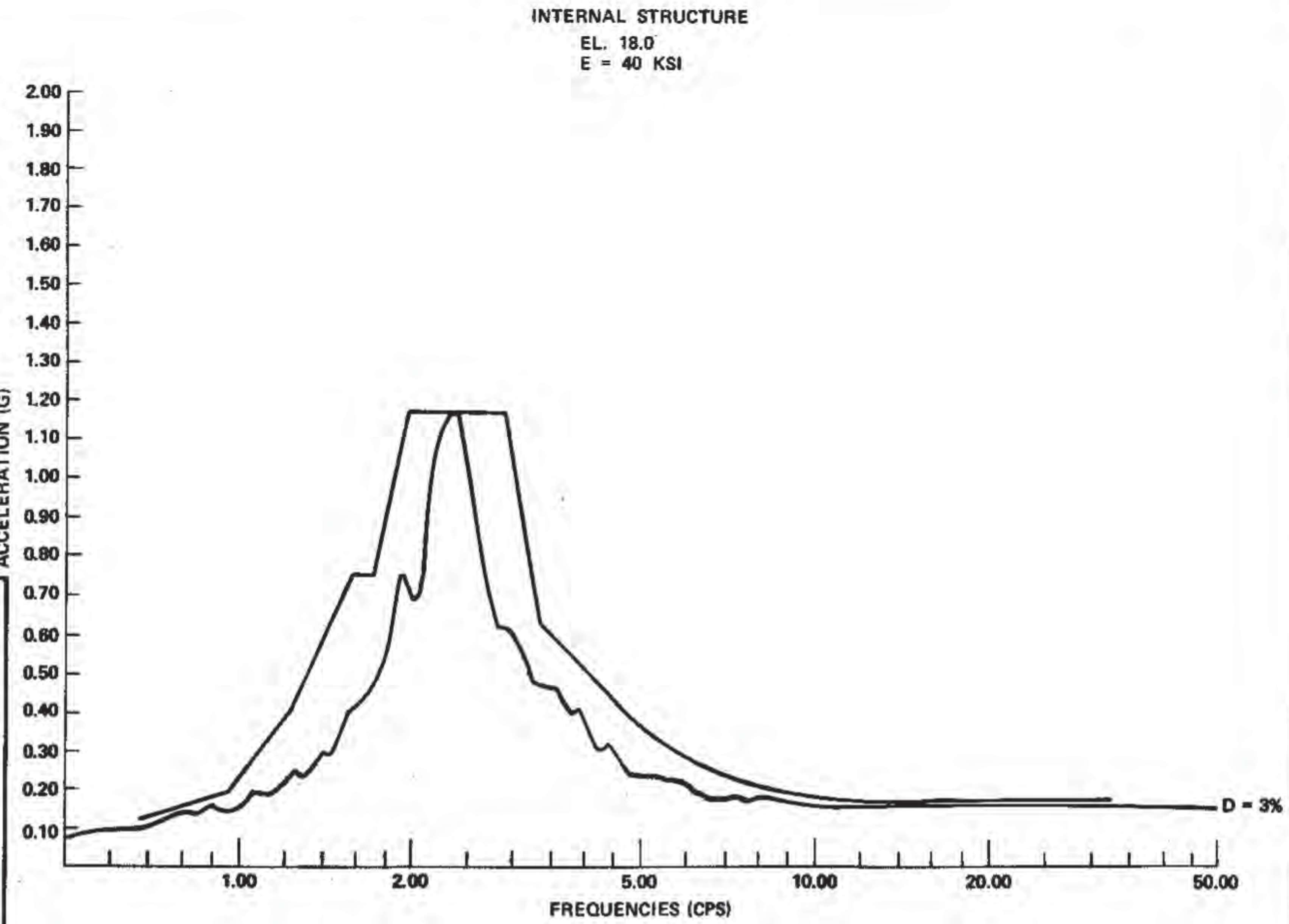
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL – SSE
REACTOR BUILDING MASS PT 14
FIGURE 3.7-69



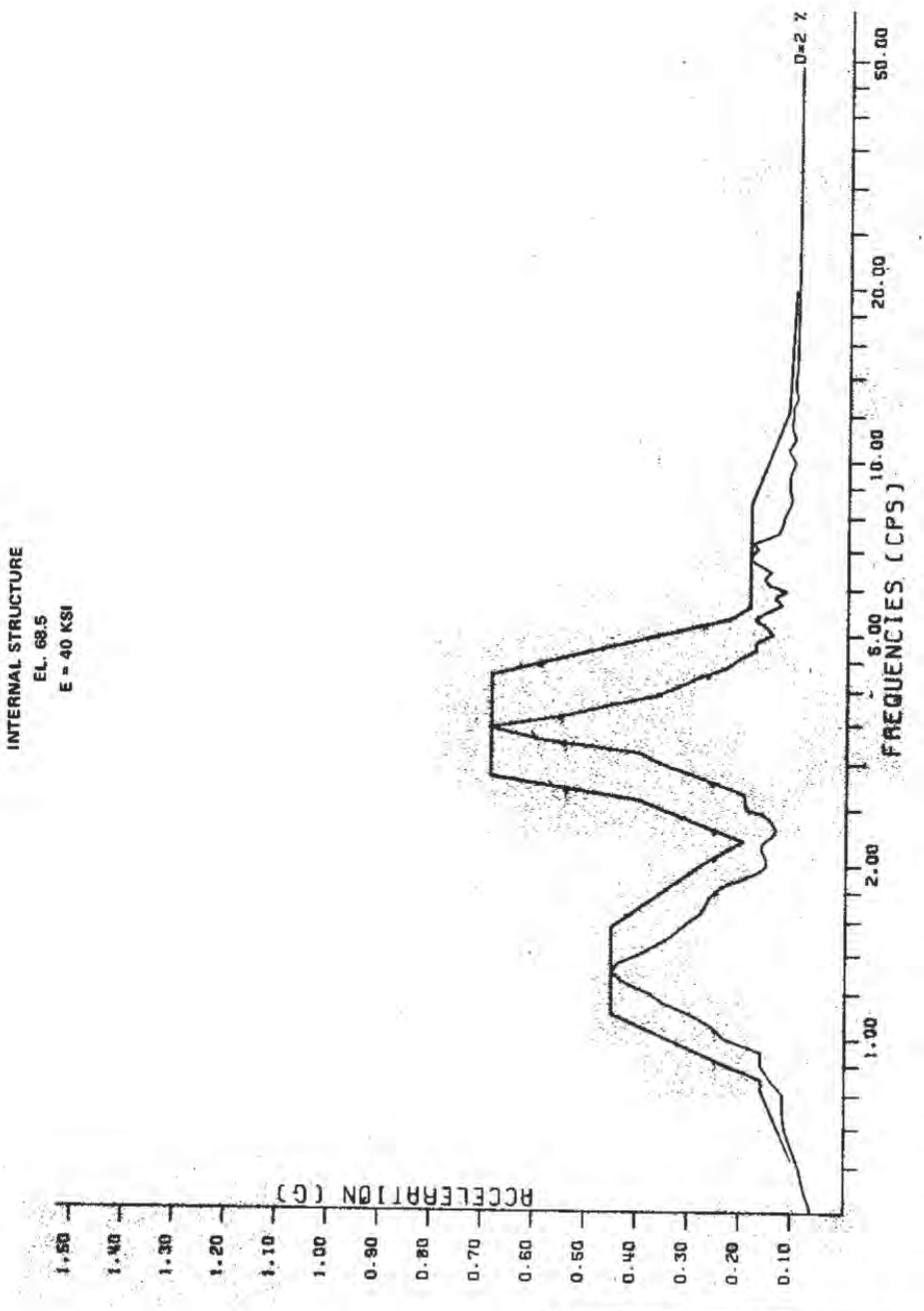
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL – SSE
REACTOR BUILDING MASS PT 15
FIGURE 3.7-70



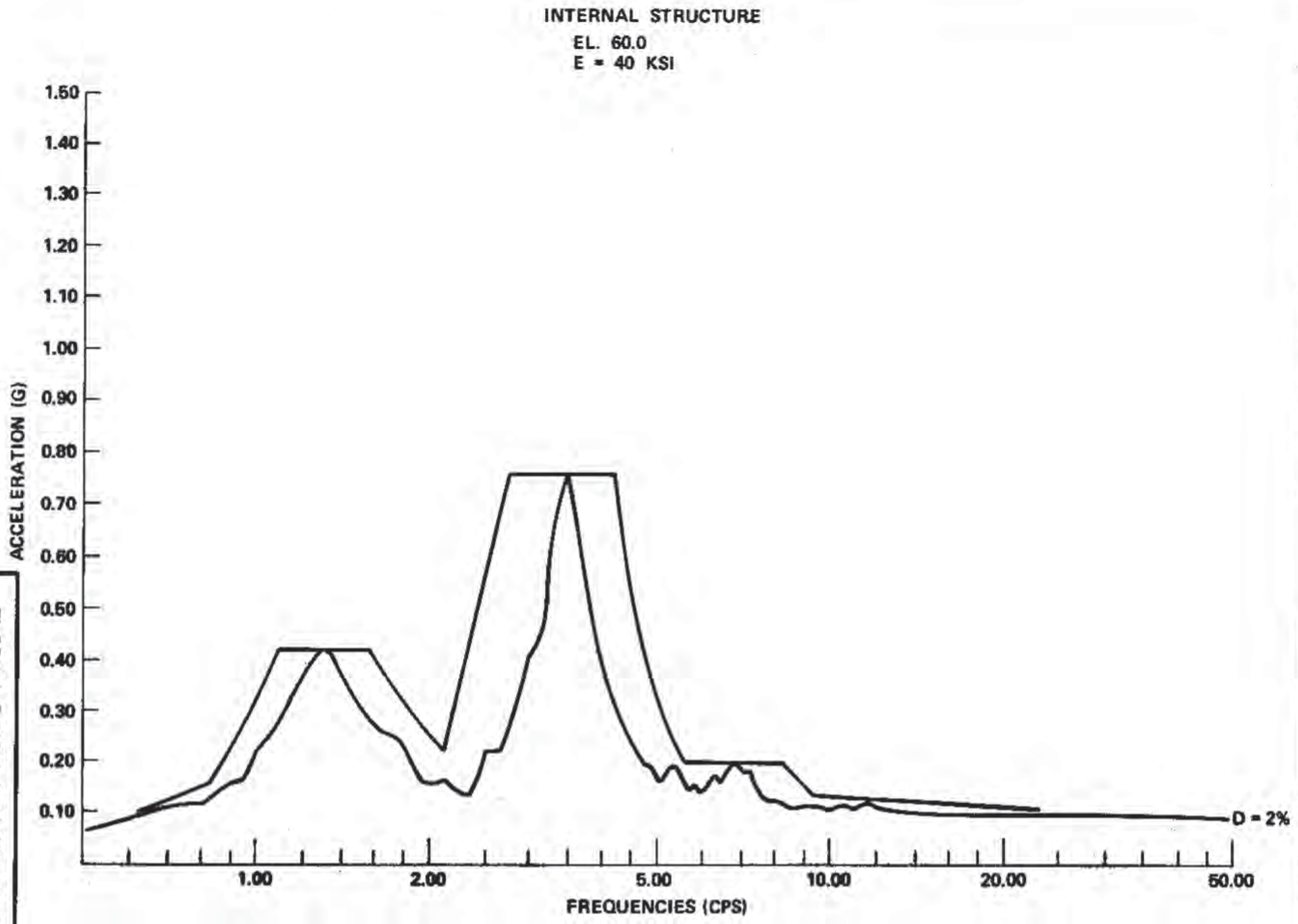
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL
SSE
REACTOR BLDG MASS PT 16
FIGURE 3.7-71



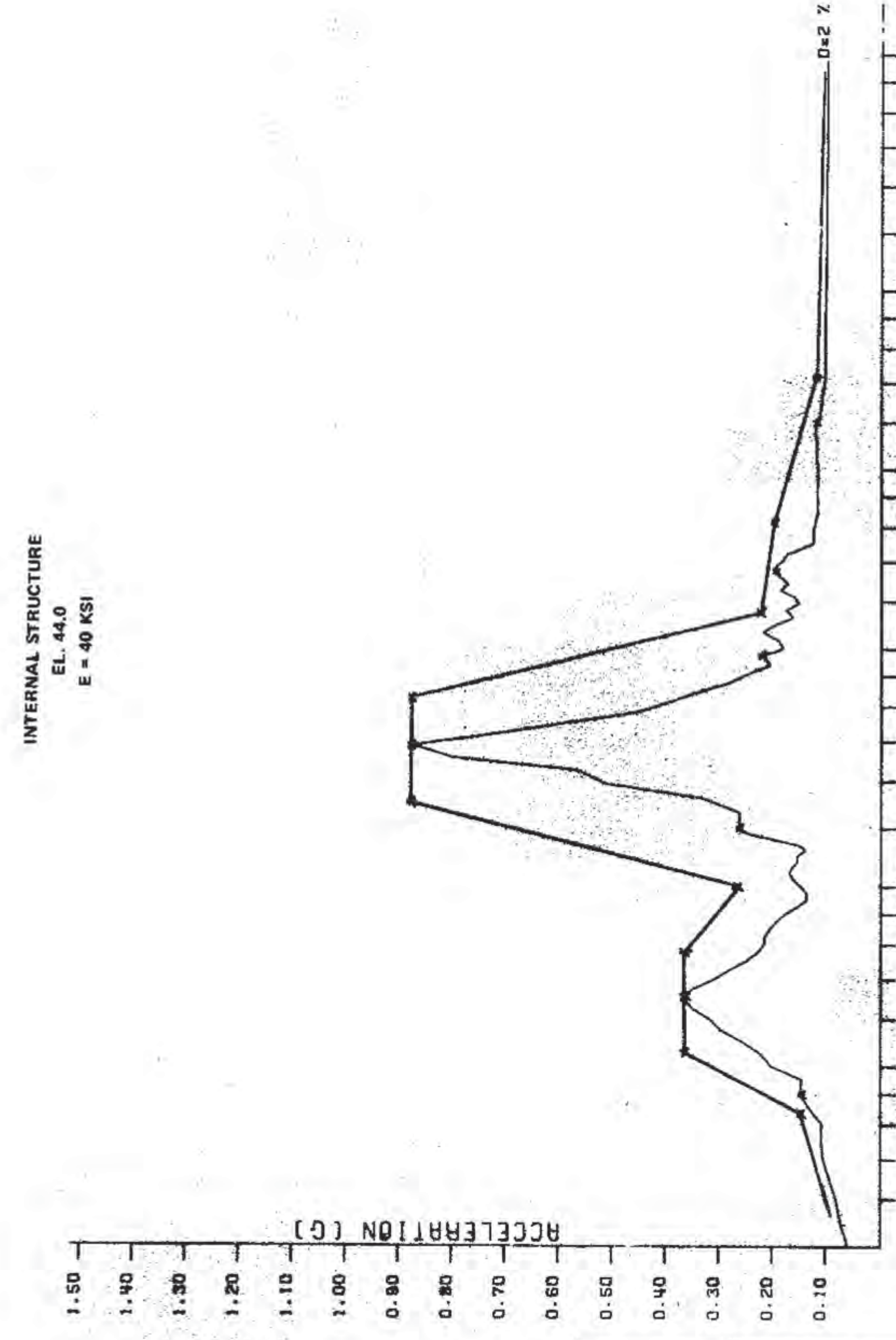
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S OBE
REACTOR BUILDING MASS PT 21
FIGURE 3.7-72



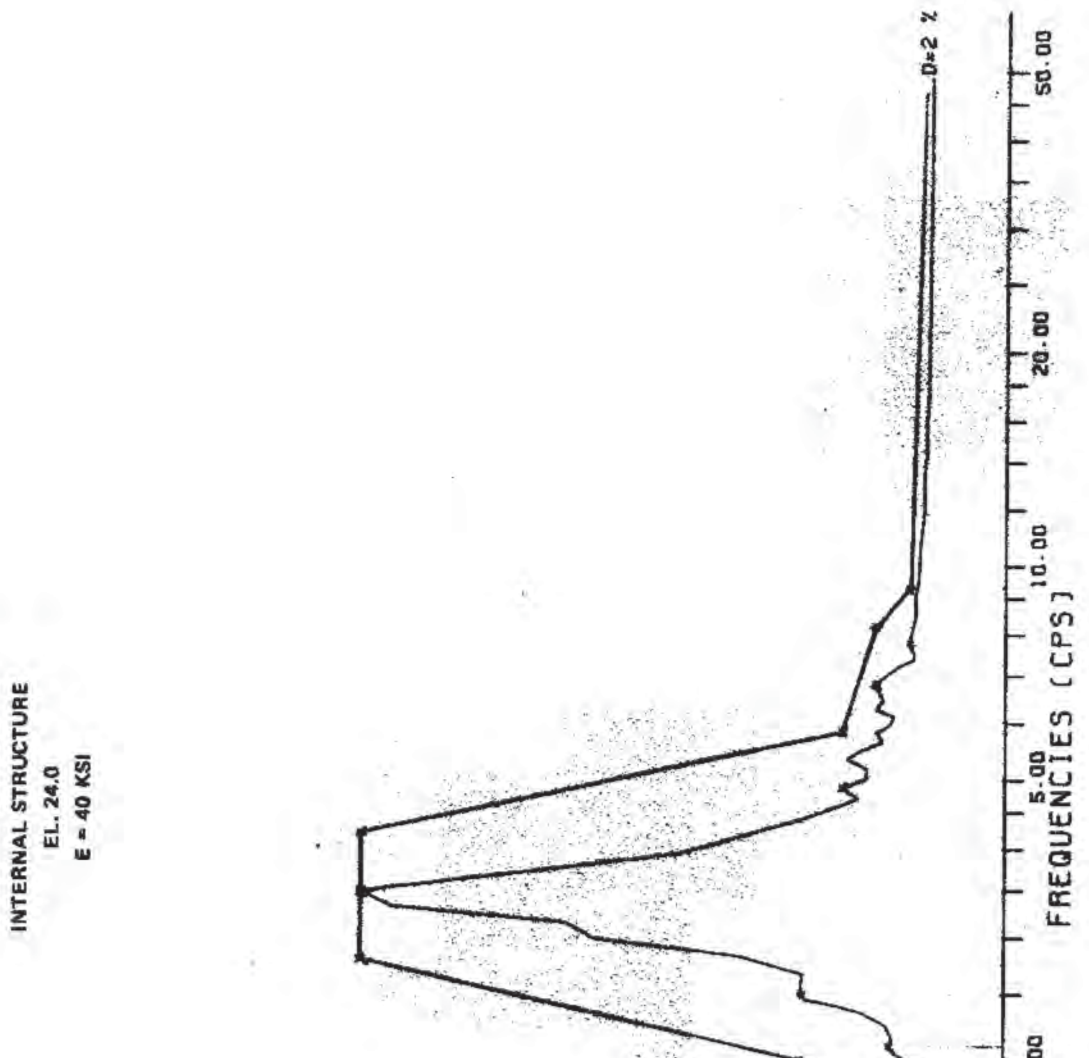
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S AND E-W OBE
REACTOR BLDG MASS PT 22
FIGURE 3.7-73



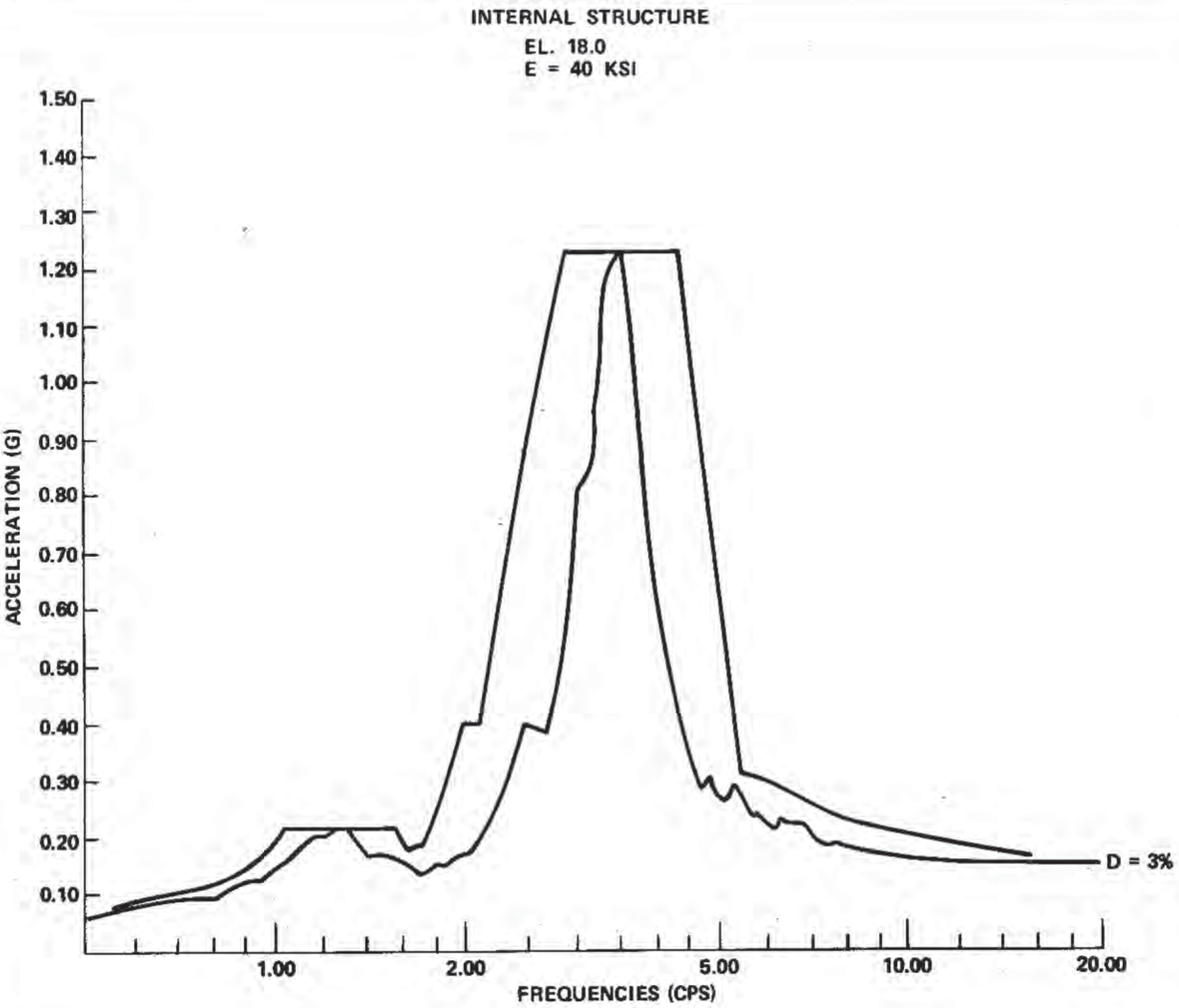
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR BLDG. MASS PT. 23
FIGURE 3.7–74



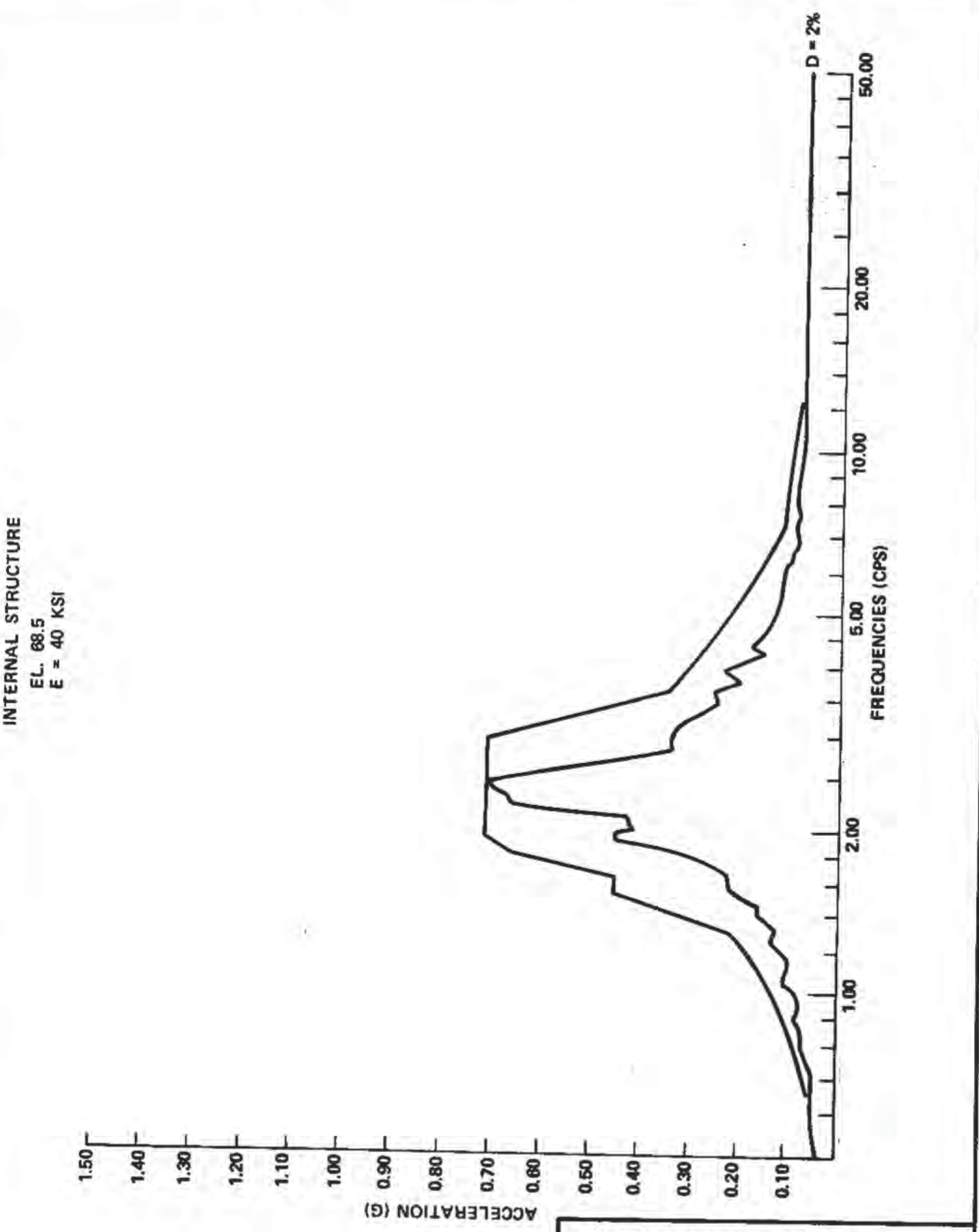
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR BLDG. MASS PT. 24
FIGURE 3.7–75



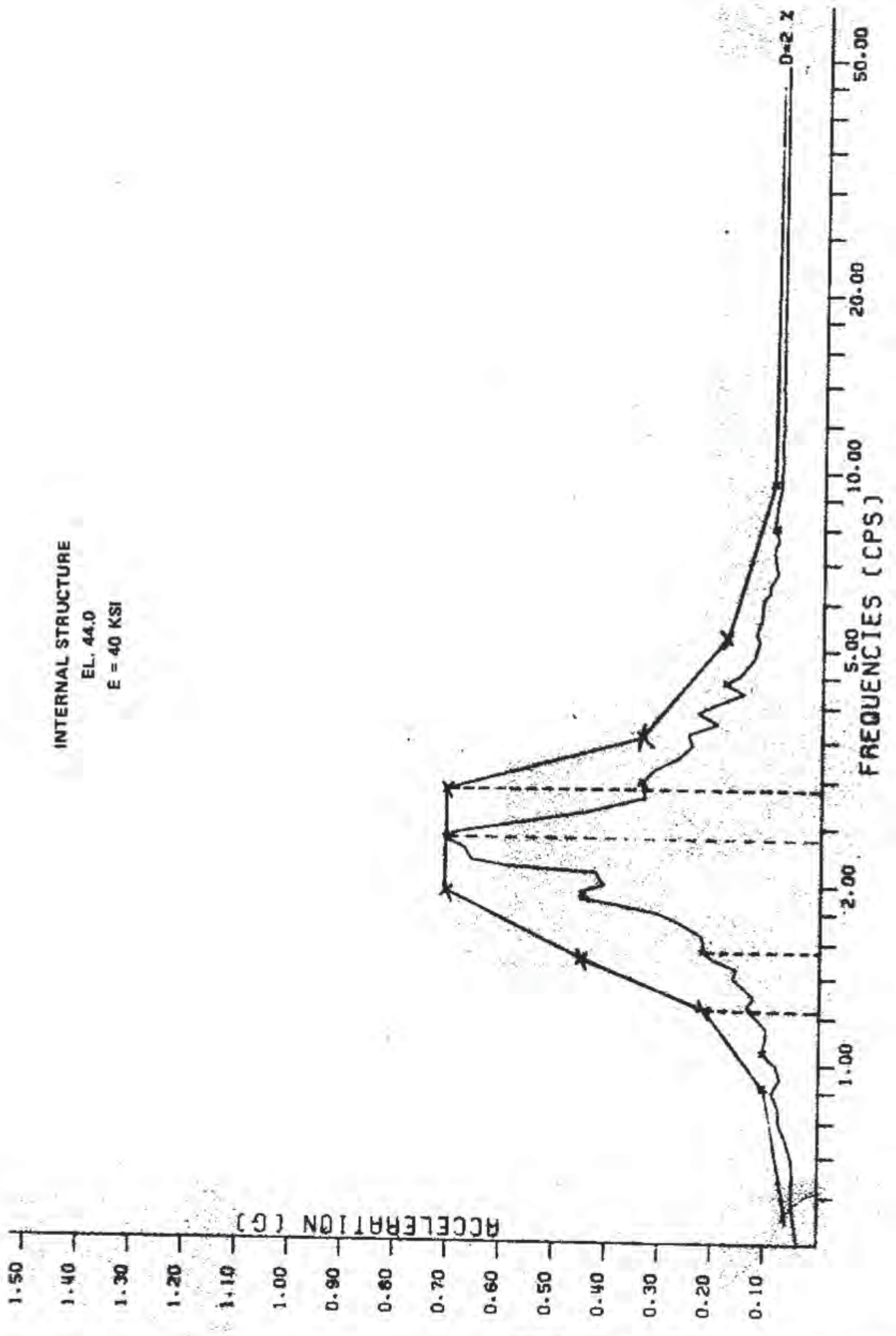
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL
N-S & E-W OBE
REACTOR BLDG MASS PT 27
FIGURE 3.7-76



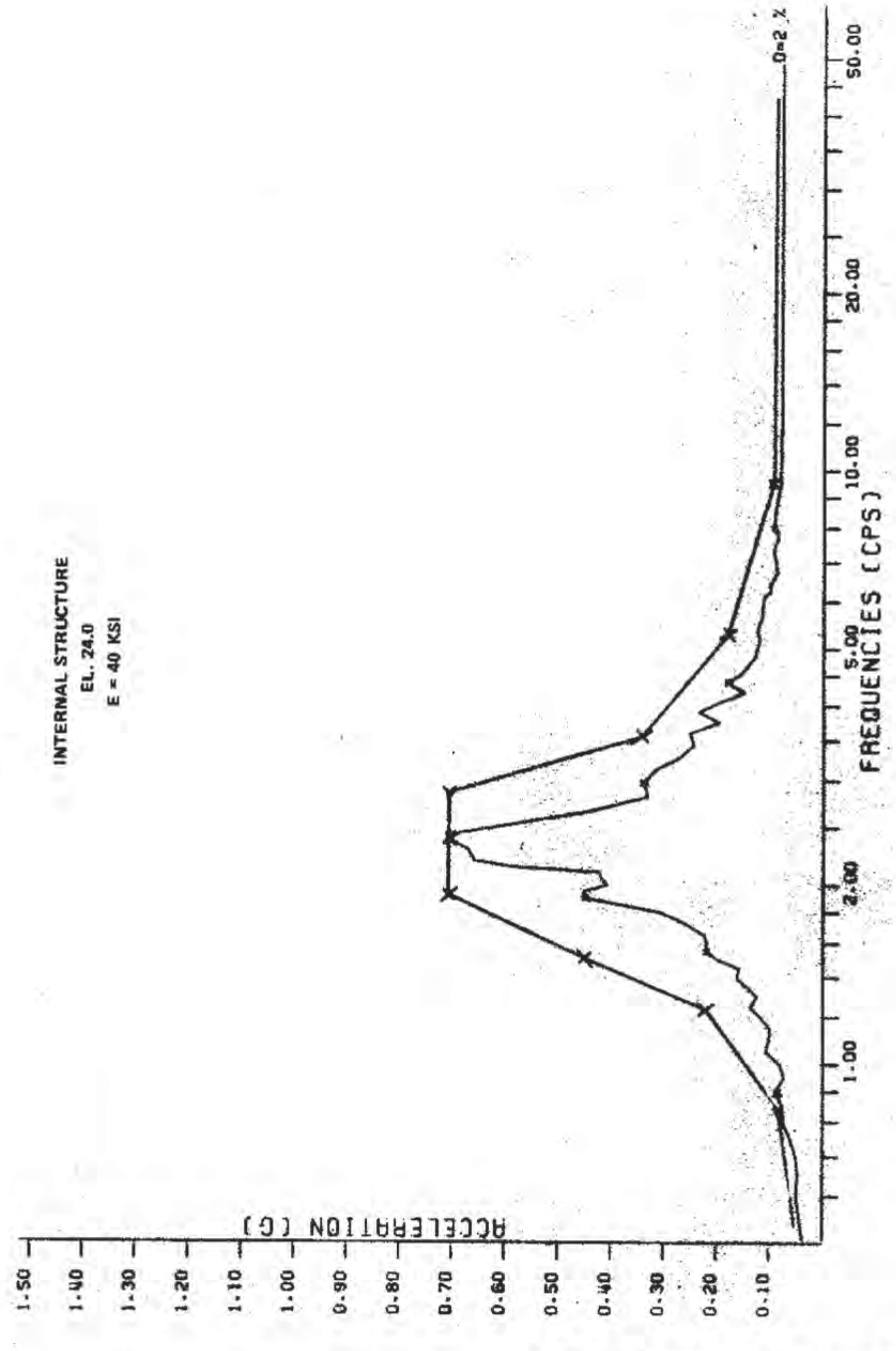
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG MASS PT 11
FIGURE 3.7-77



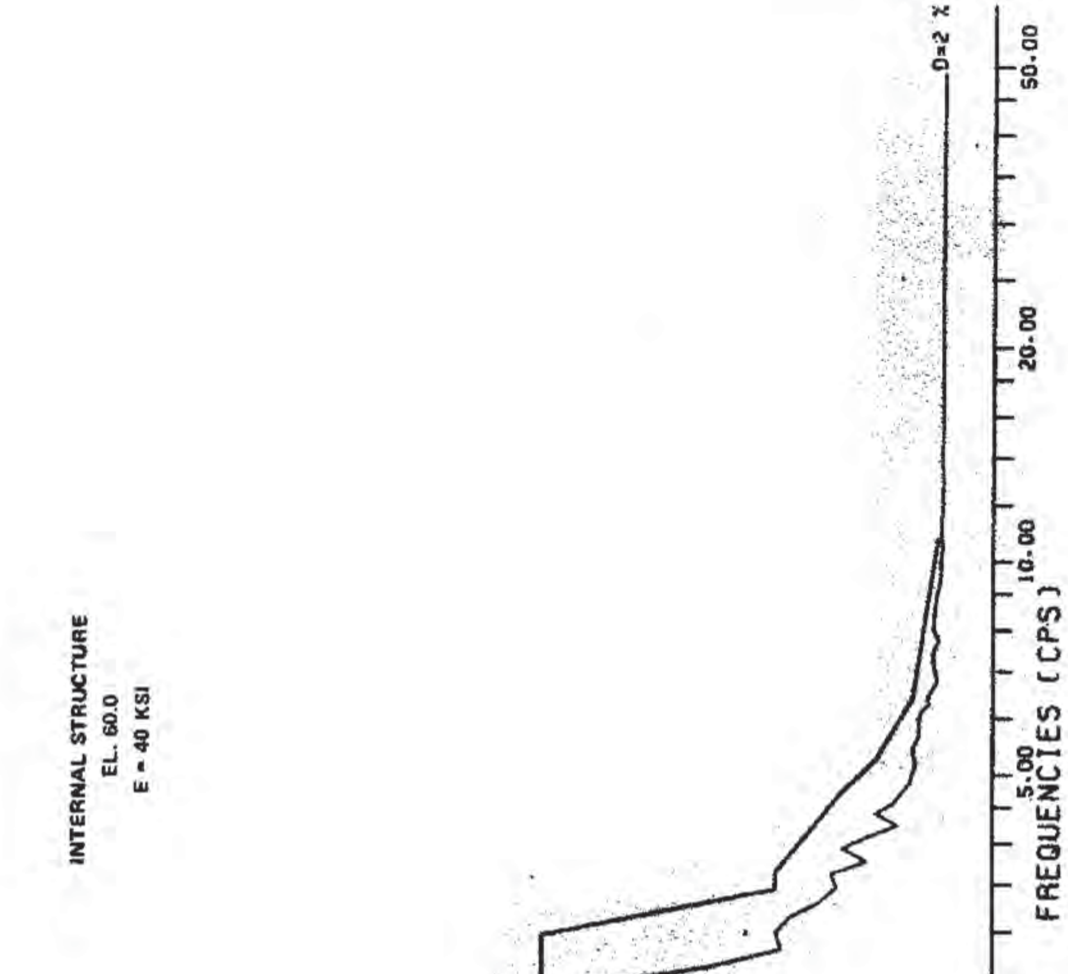
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG. MASS PT. 13
FIGURE 3.7–78



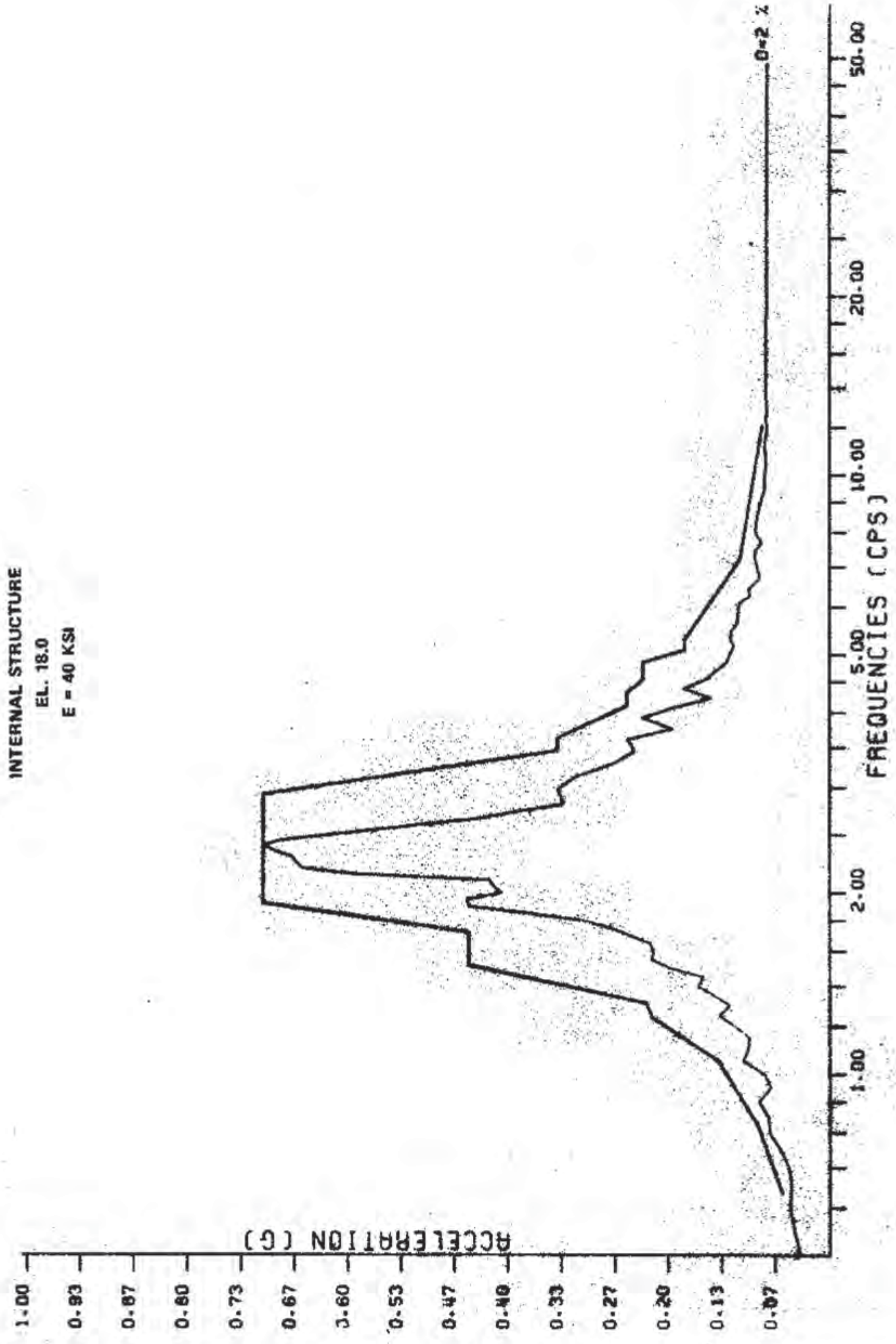
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG. MASS PT. 14
FIGURE 3.7–79



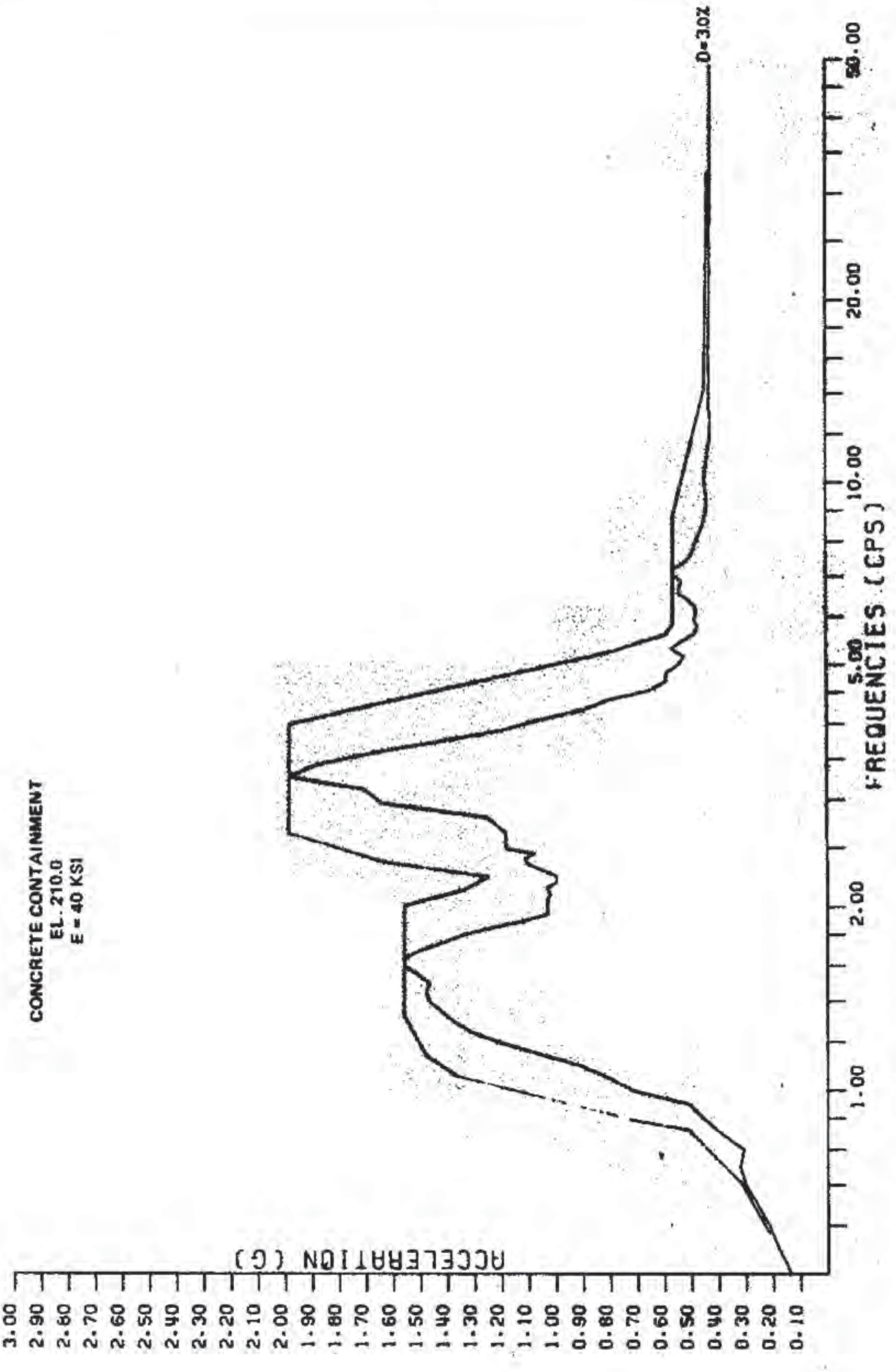
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG. MASS PT.15
FIGURE 3.7-80



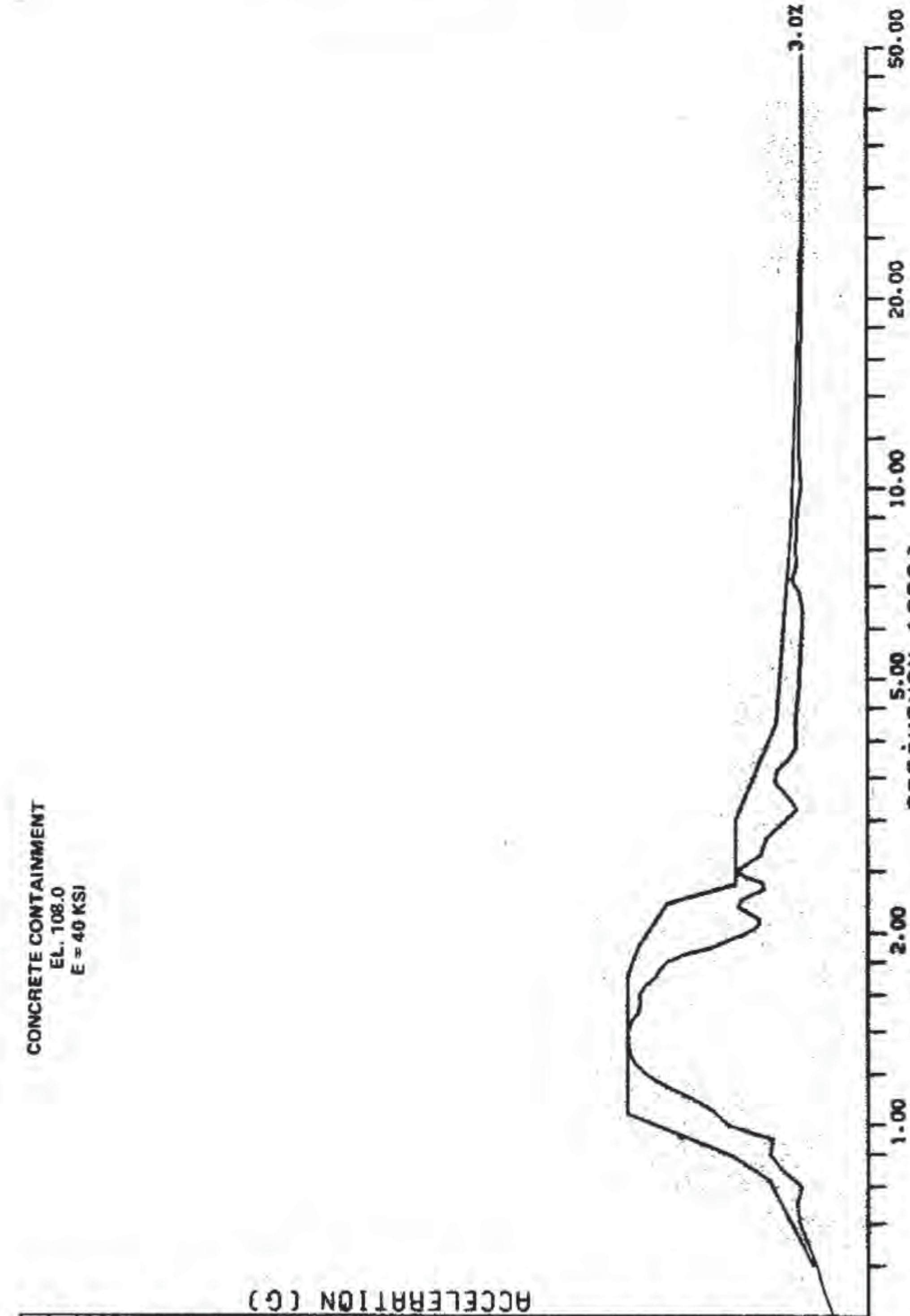
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG. MASS PT. 16
FIGURE 3.7–81



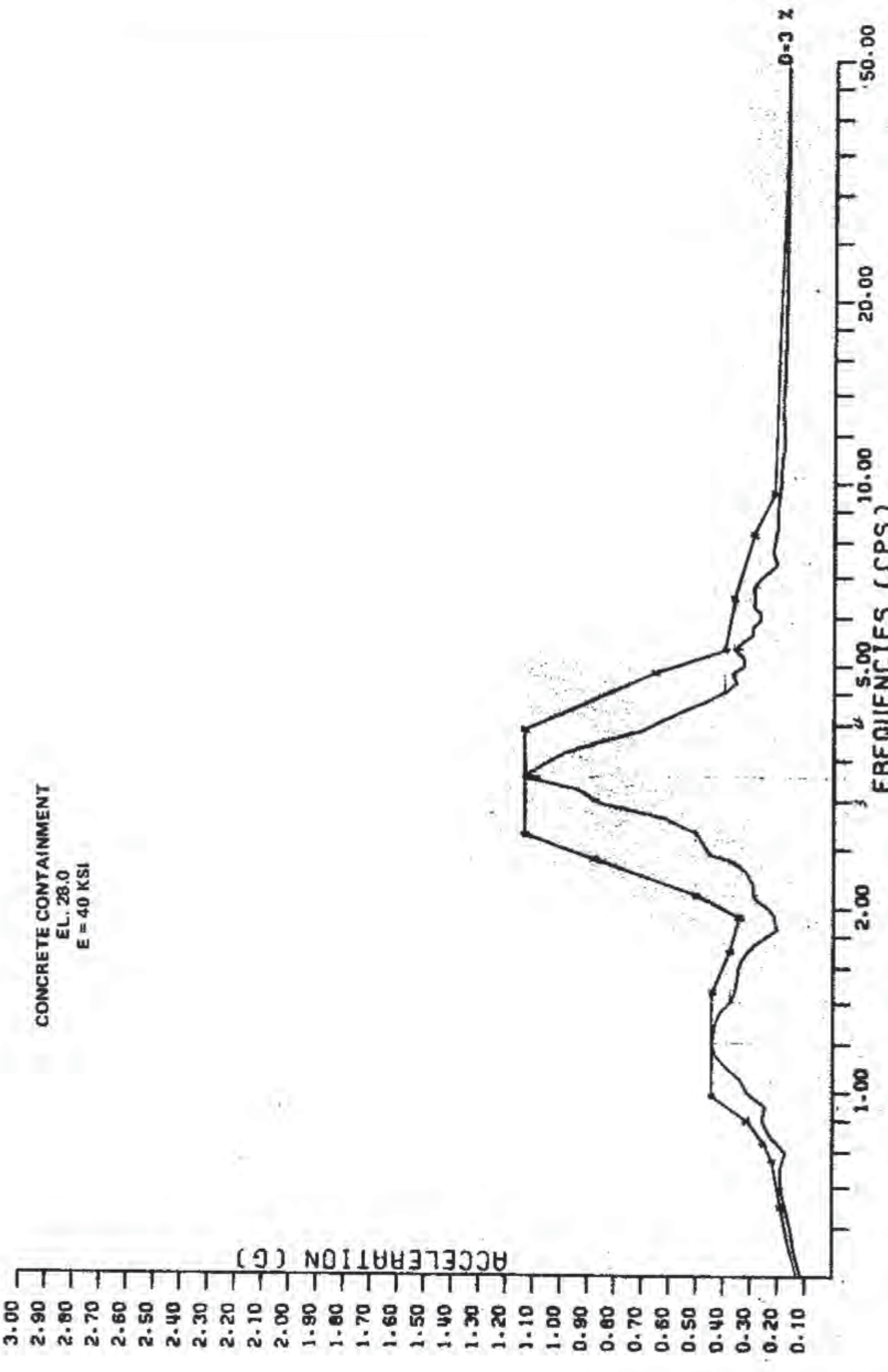
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
REACTOR BLDG. MASS PT. 1
FIGURE 3.7-82



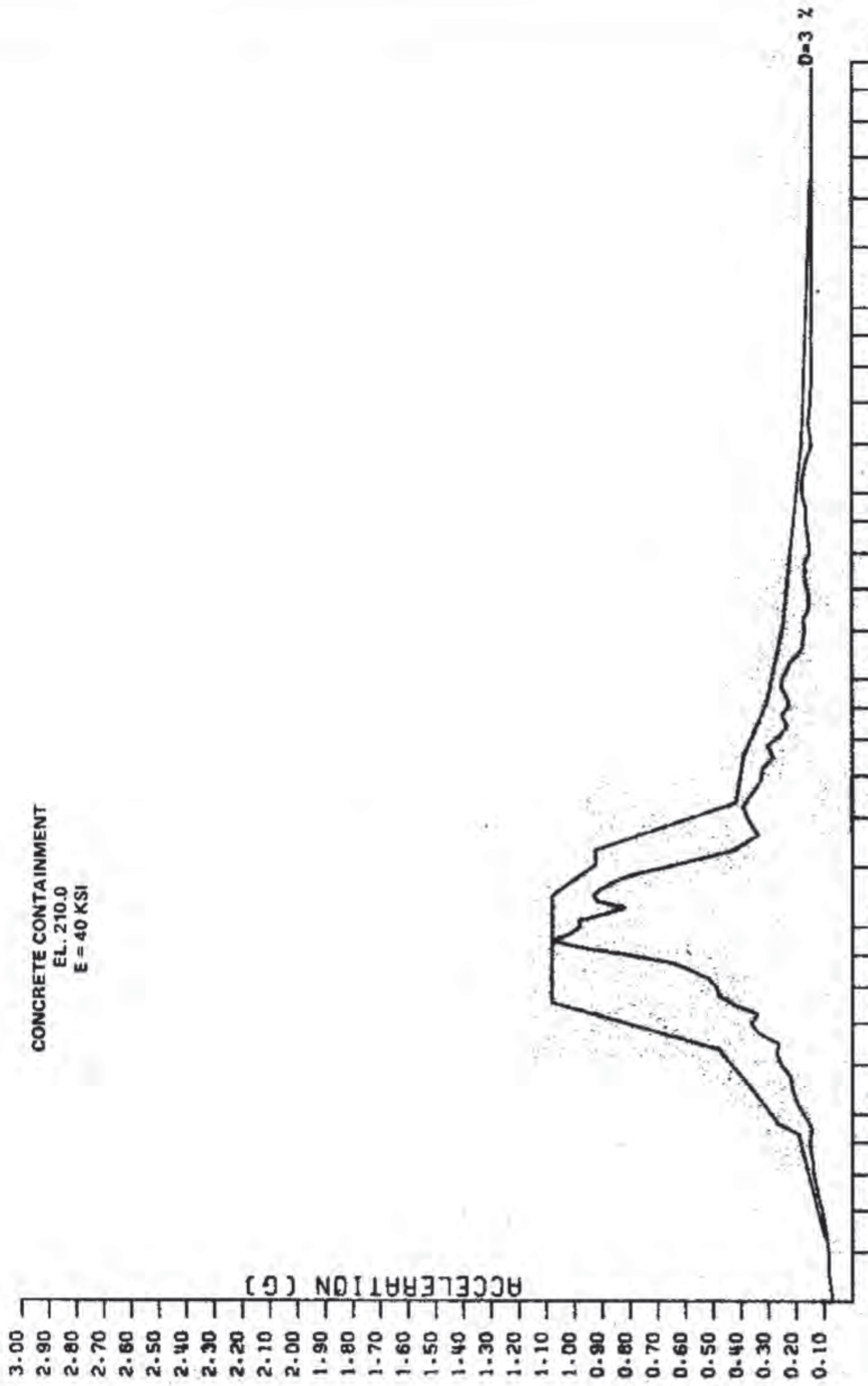
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
REACTOR BLDG. MASS PT. 6
FIGURE 3.7-83



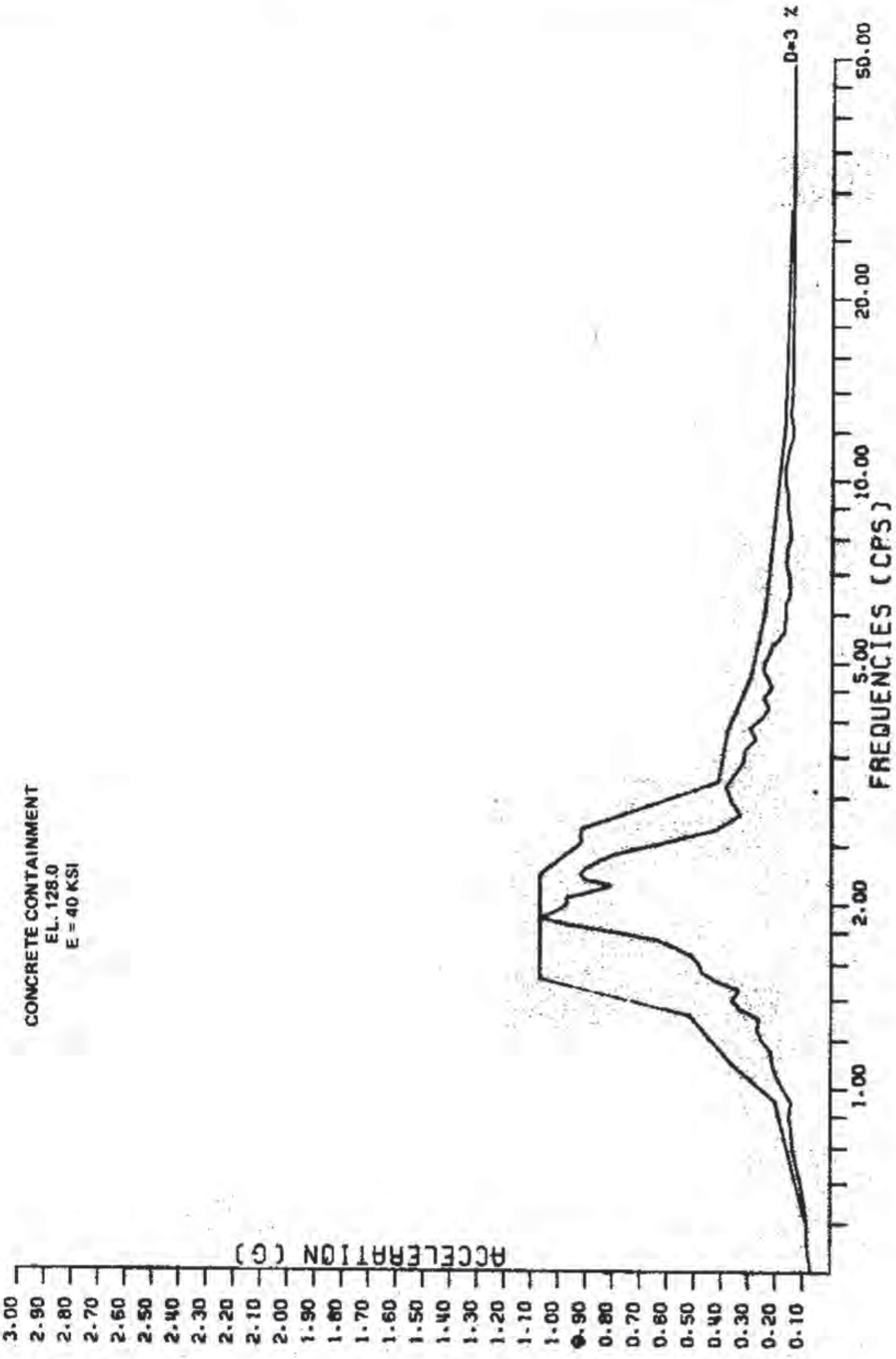
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
REACTOR BLDG. MASS PT. 10
FIGURE 3.7-84



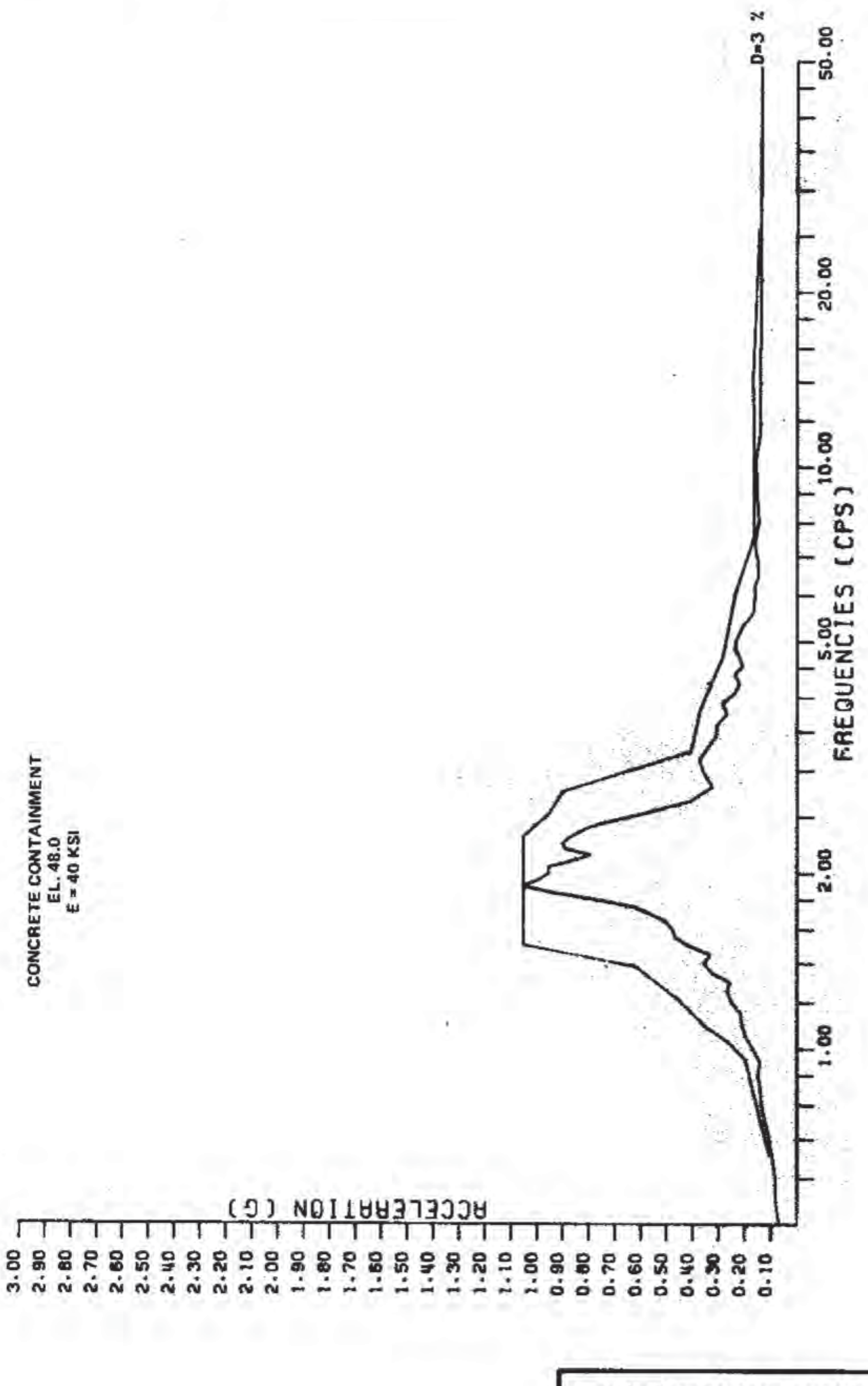
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR BLDG. MASS PT. 1
FIGURE 3.7–85



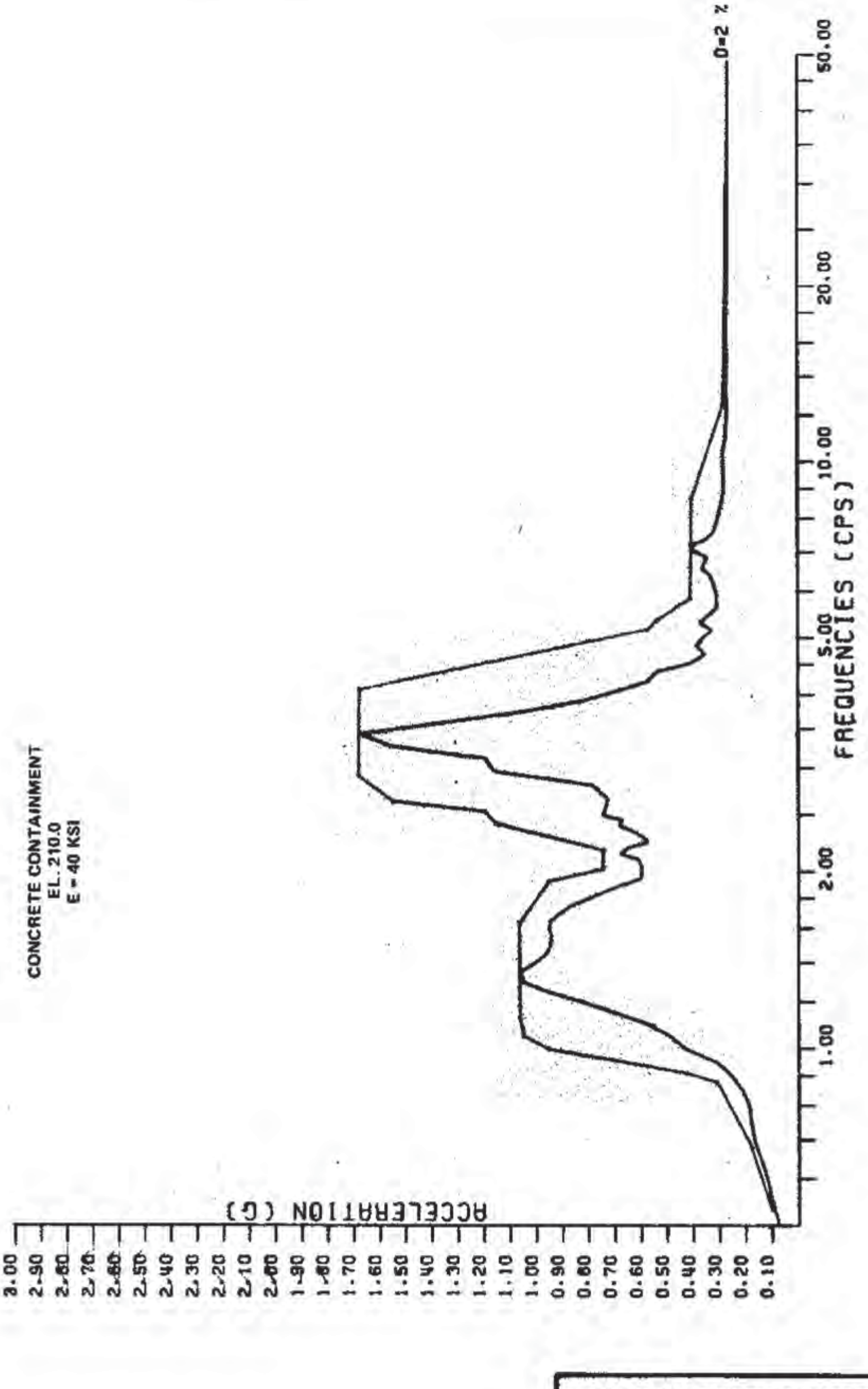
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR BLDG. MASS PT. 3
FIGURE 3.7–86



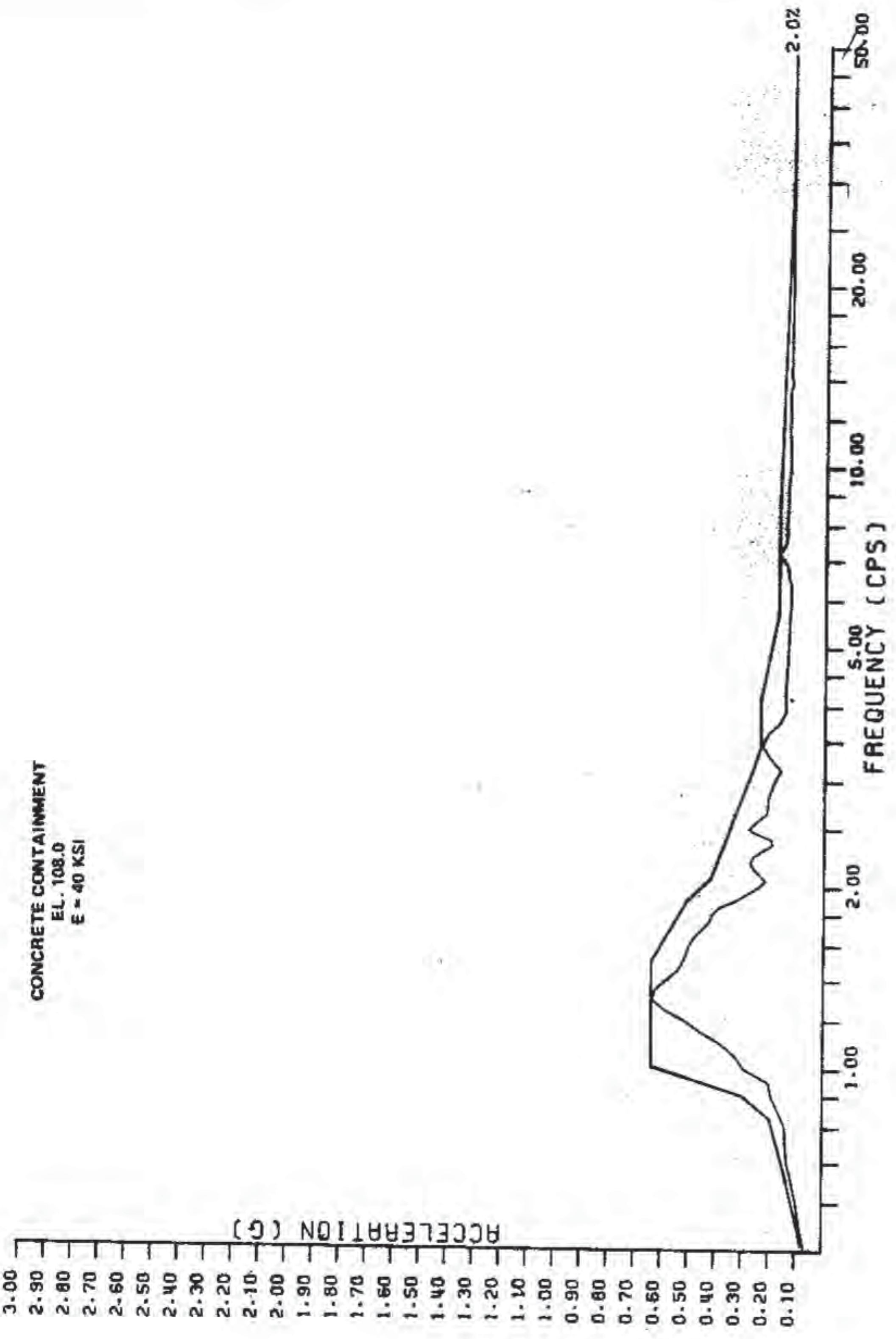
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR BLDG. MASS PT. 5
FIGURE 3.7–87



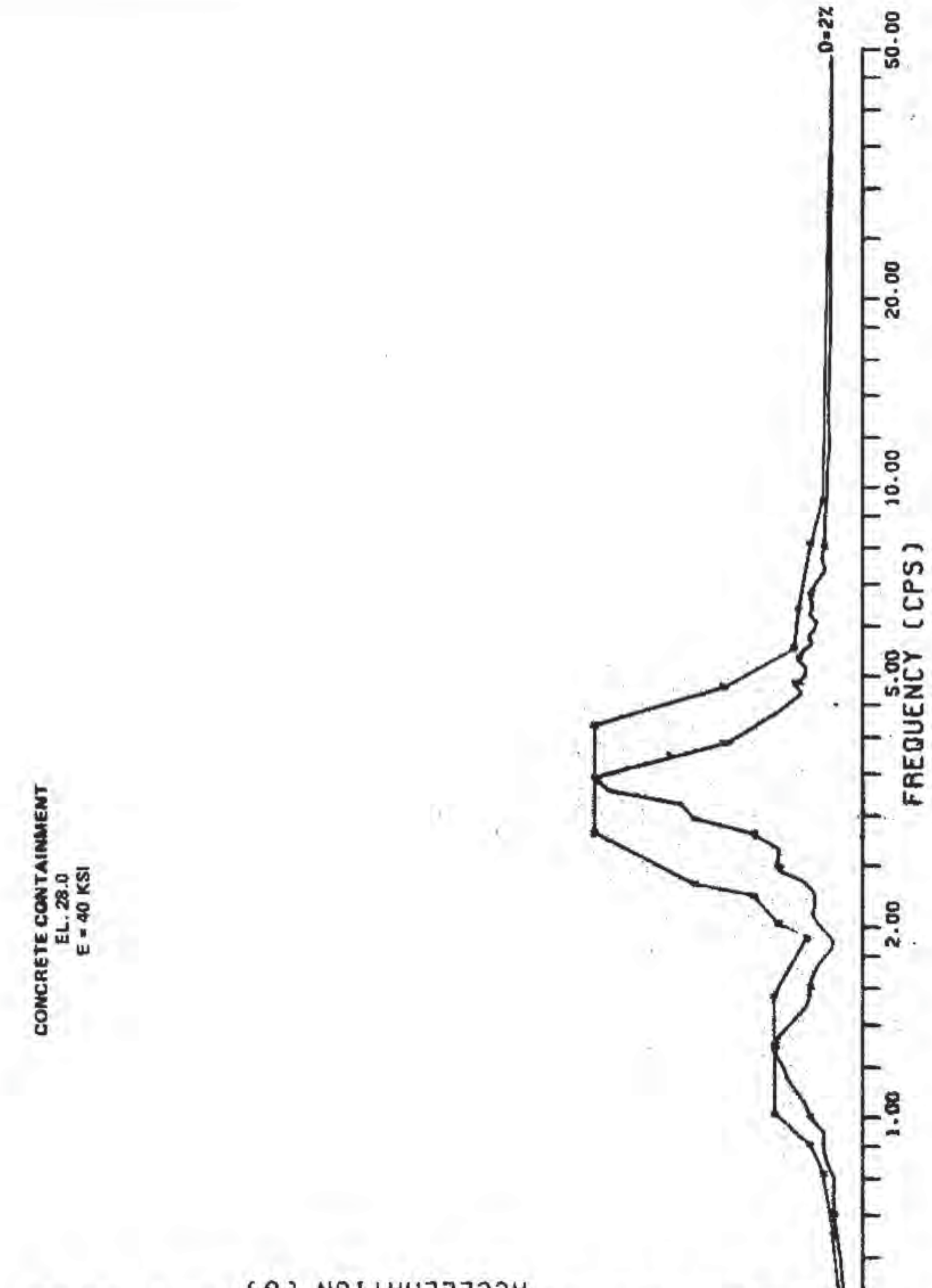
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR BLDG. MASS PT. 1
FIGURE 3.7-88



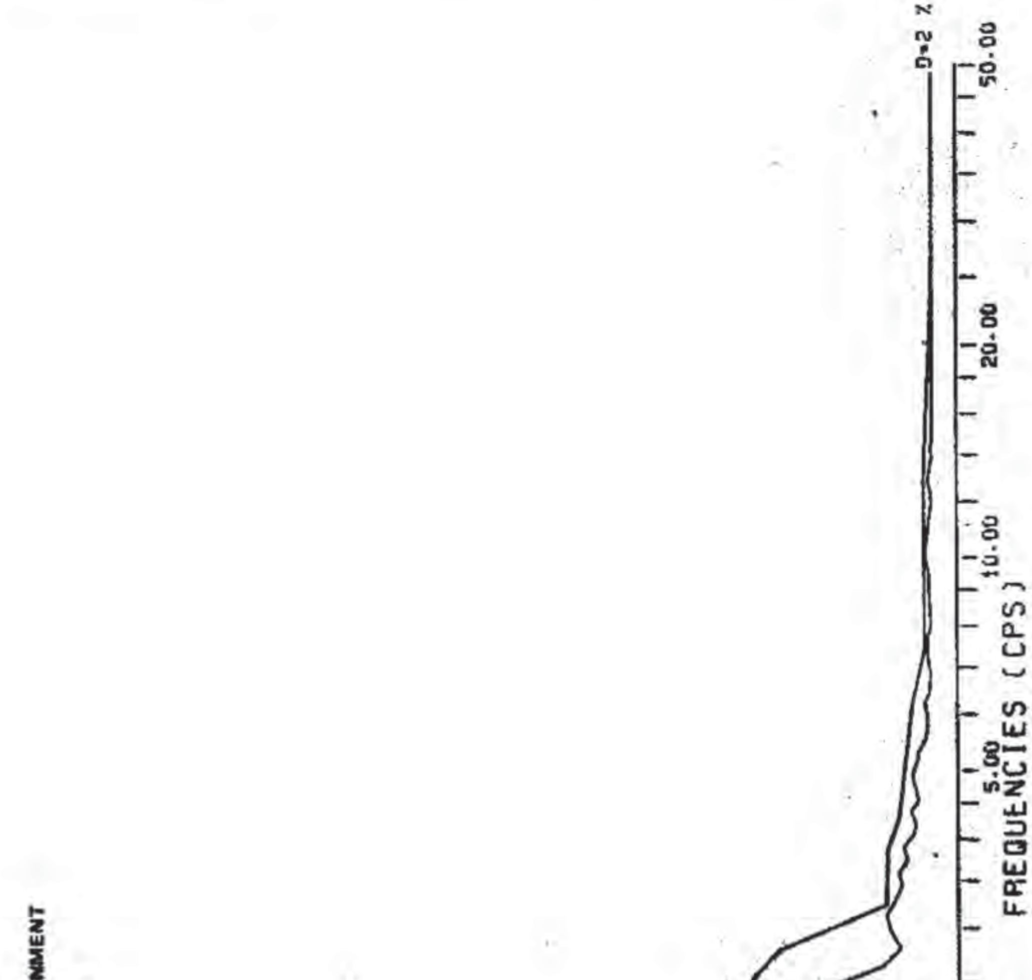
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR BLDG. MASS PT. 6
FIGURE 3.7–89



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

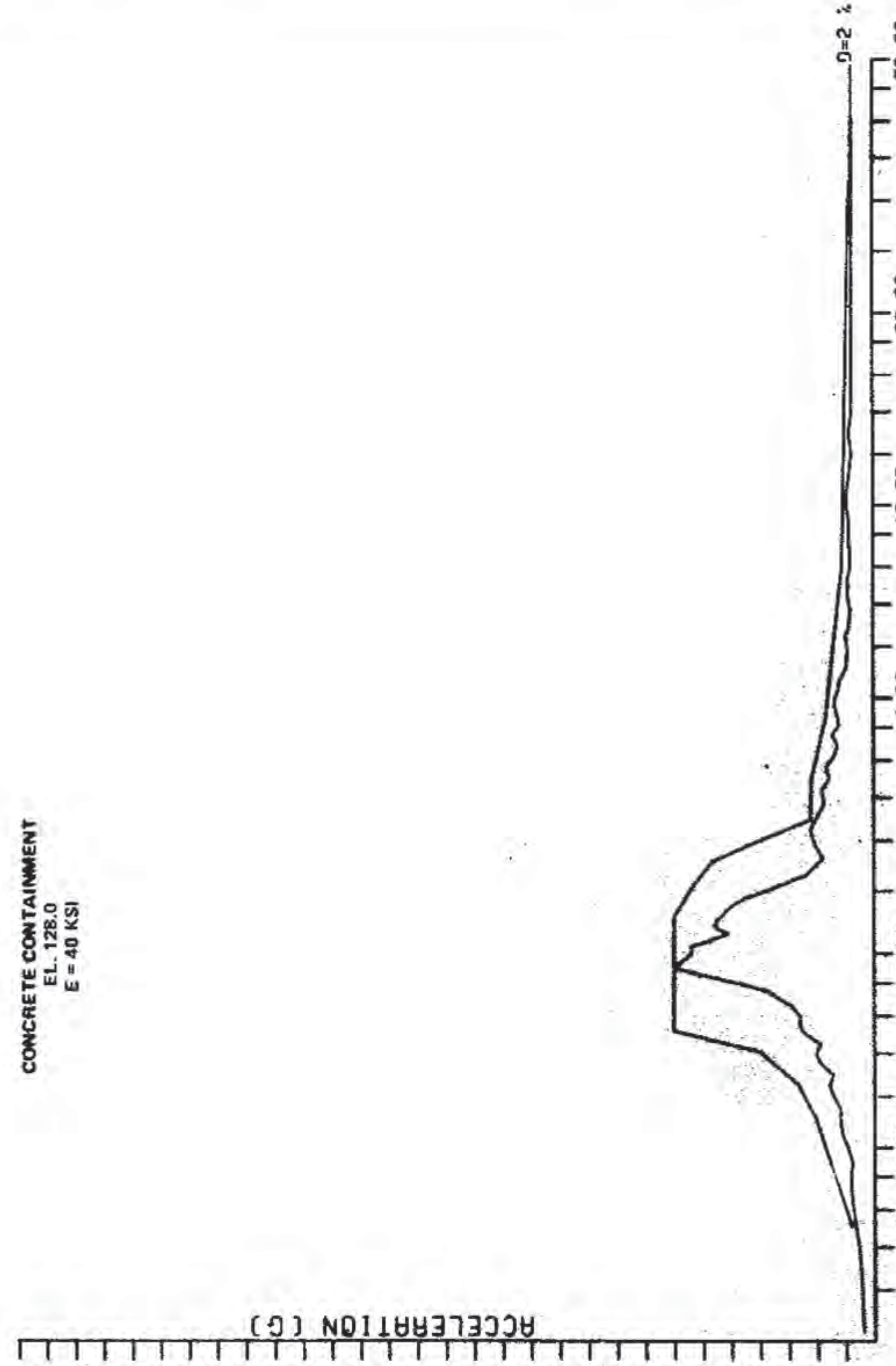
FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR BLDG. MASS PT. 10
FIGURE 3.7–90



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

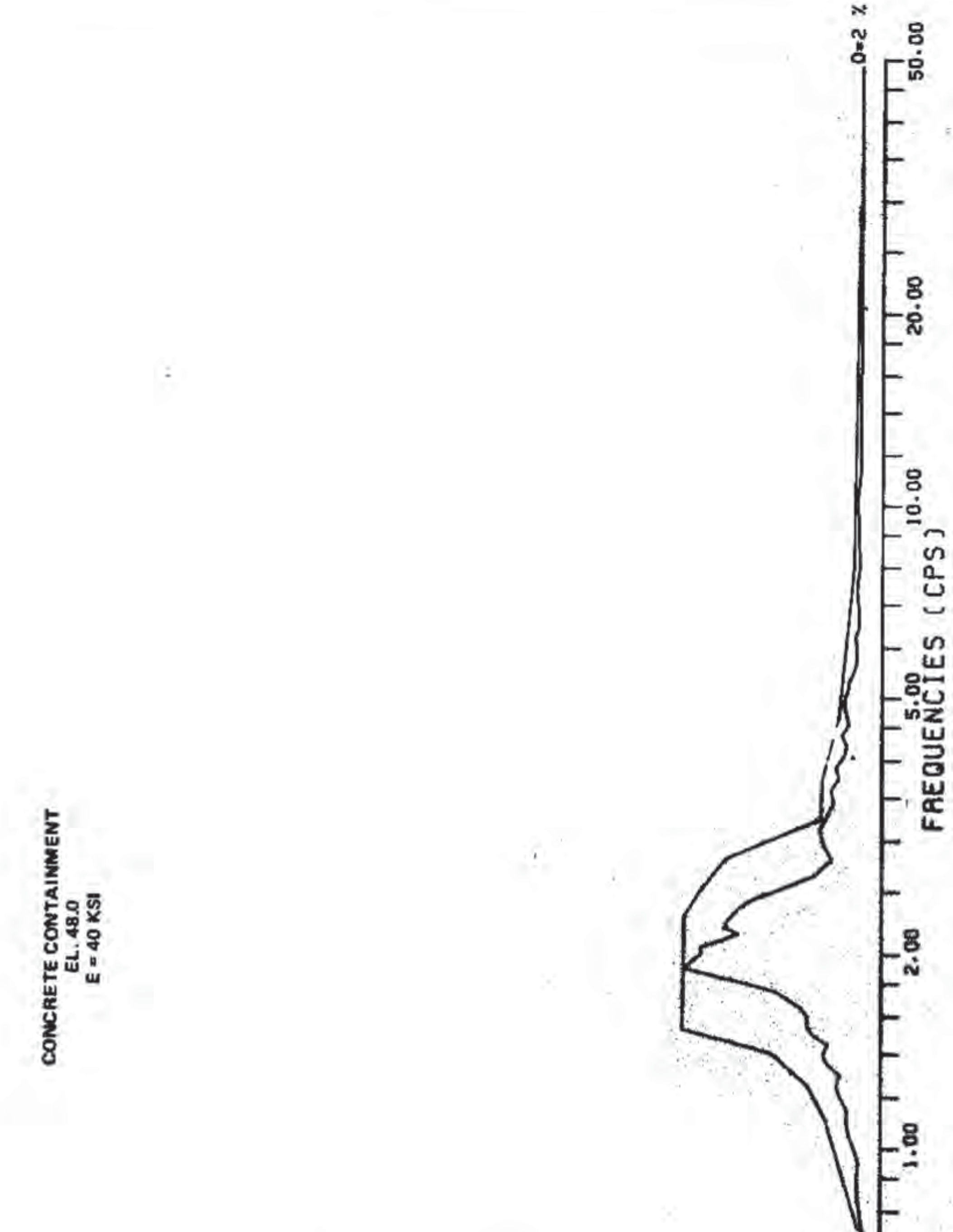
FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG MASS PT 1

FIGURE 3.7-91



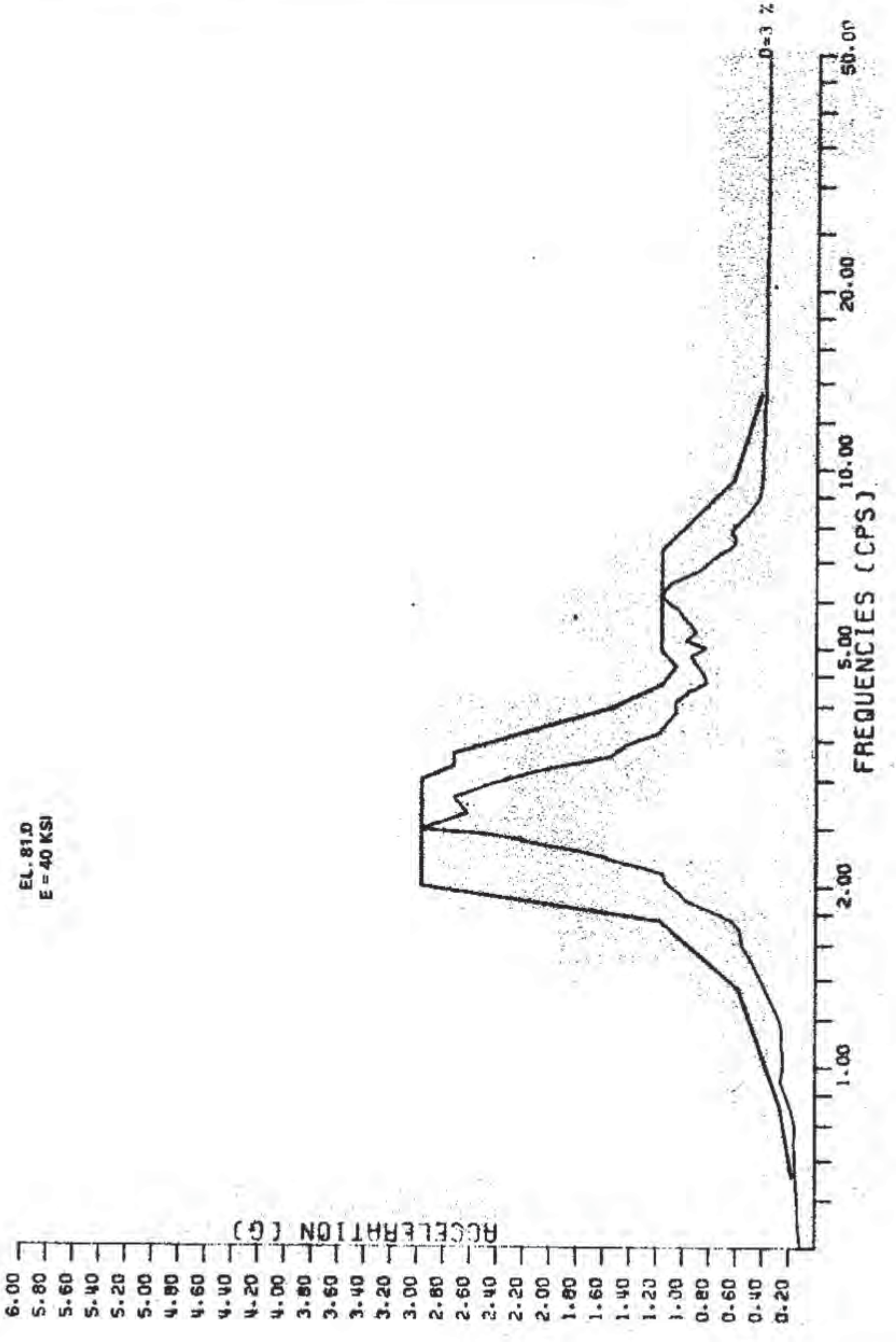
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG. MASS PT. 3
FIGURE 3.7–92



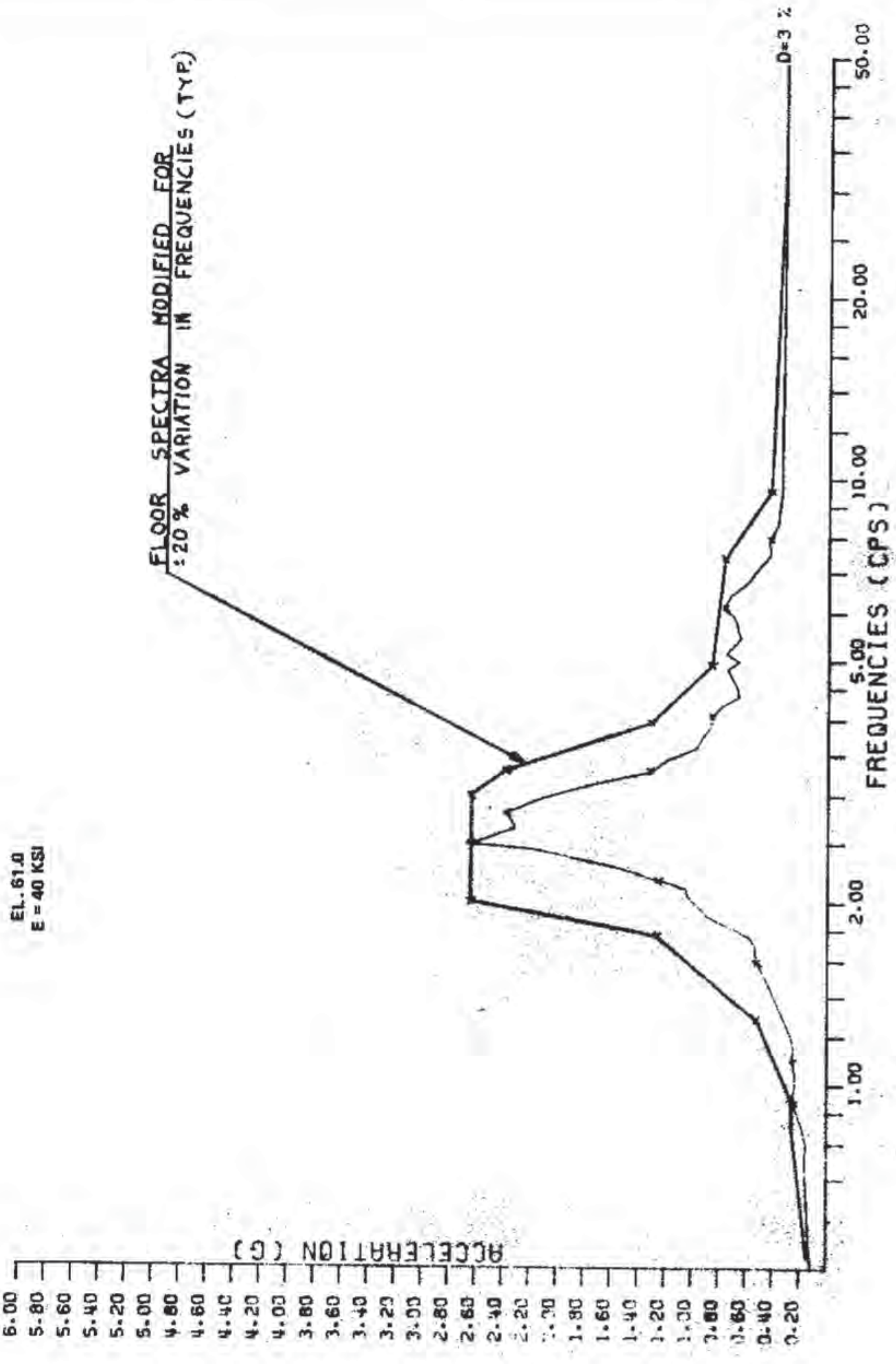
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR BLDG. MASS PT. 5
FIGURE 3.7-93



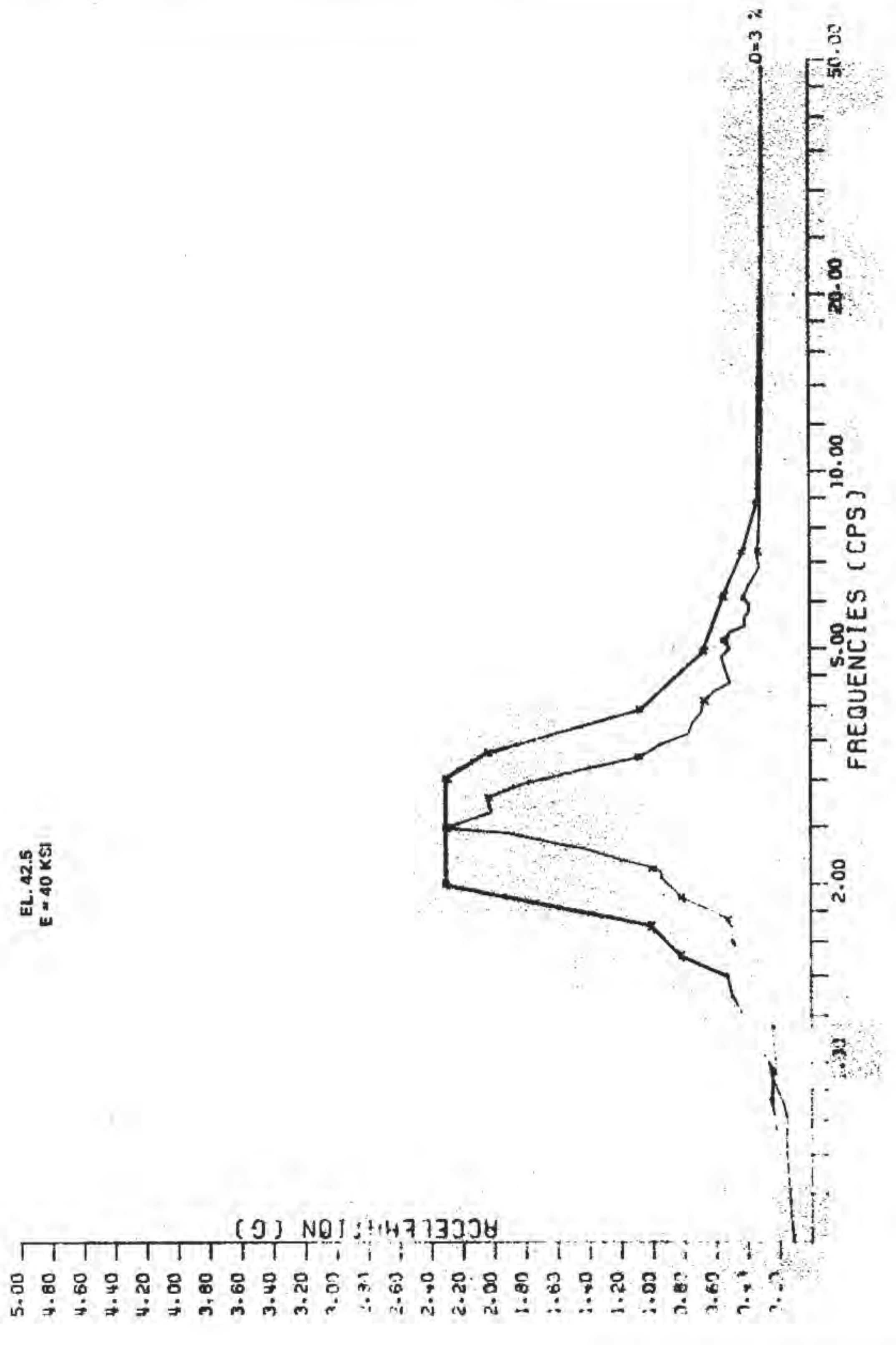
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–94



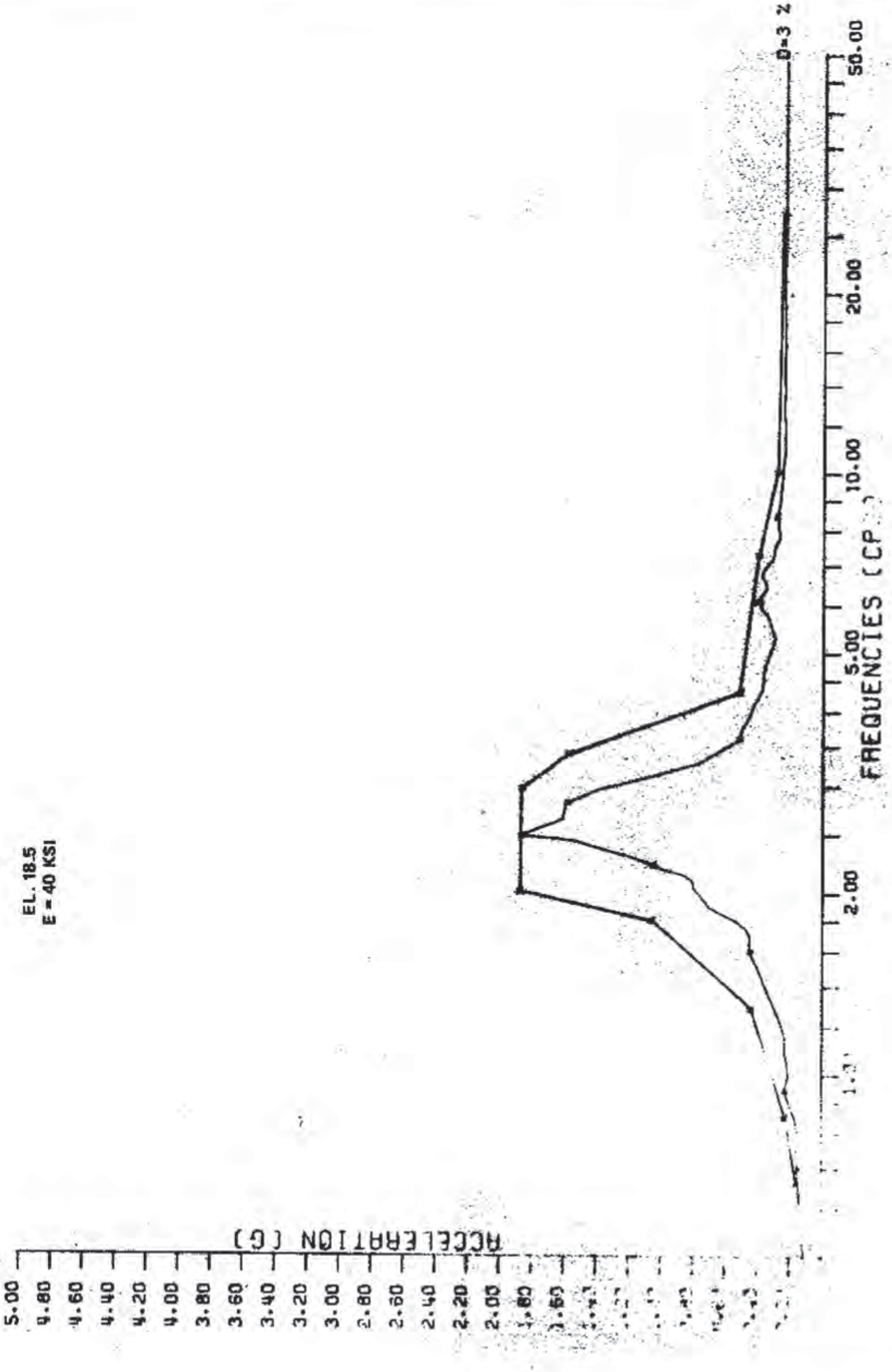
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–95



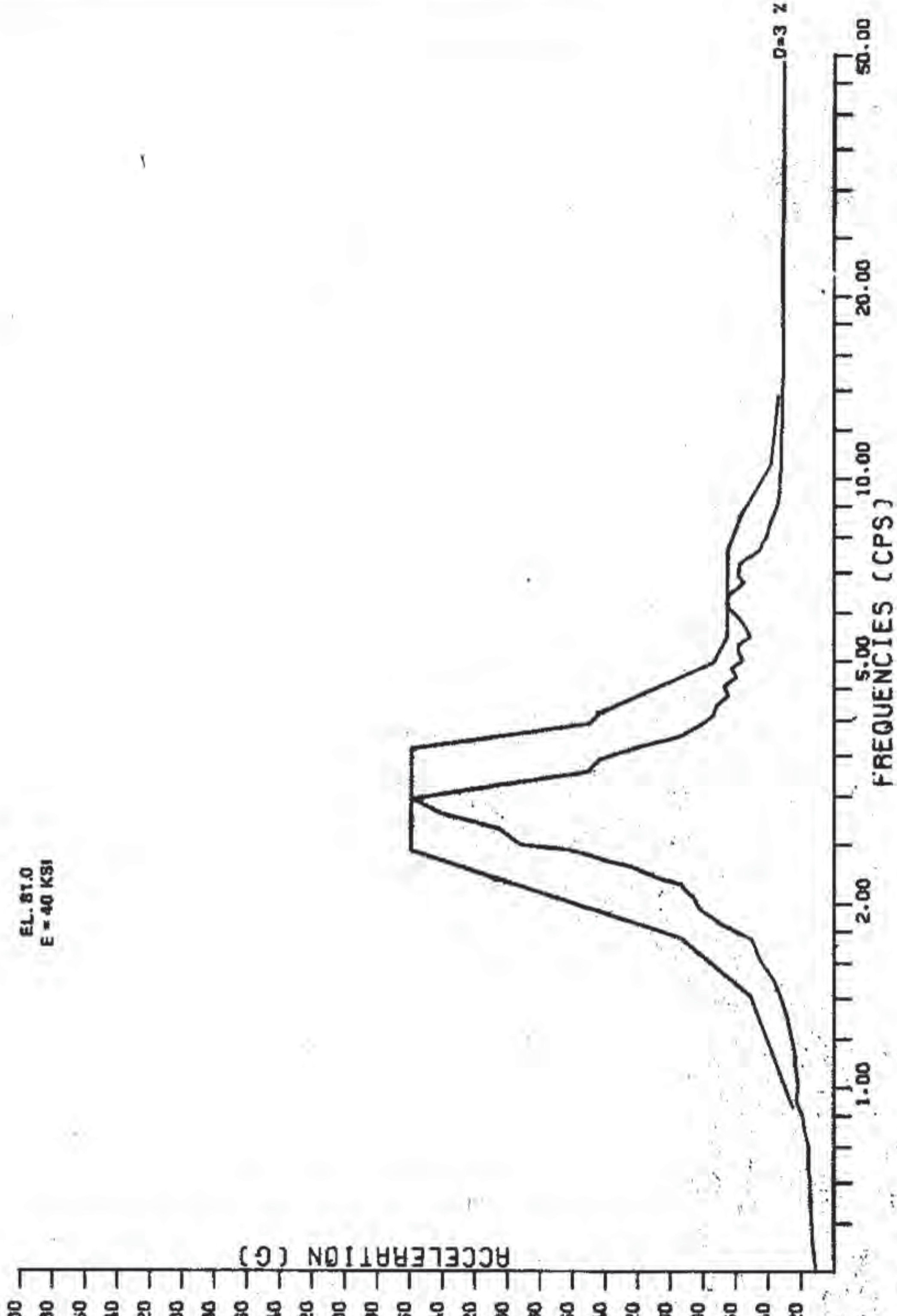
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
REACTOR AUXILIARY BUILDING
FIGURE 3.7-96



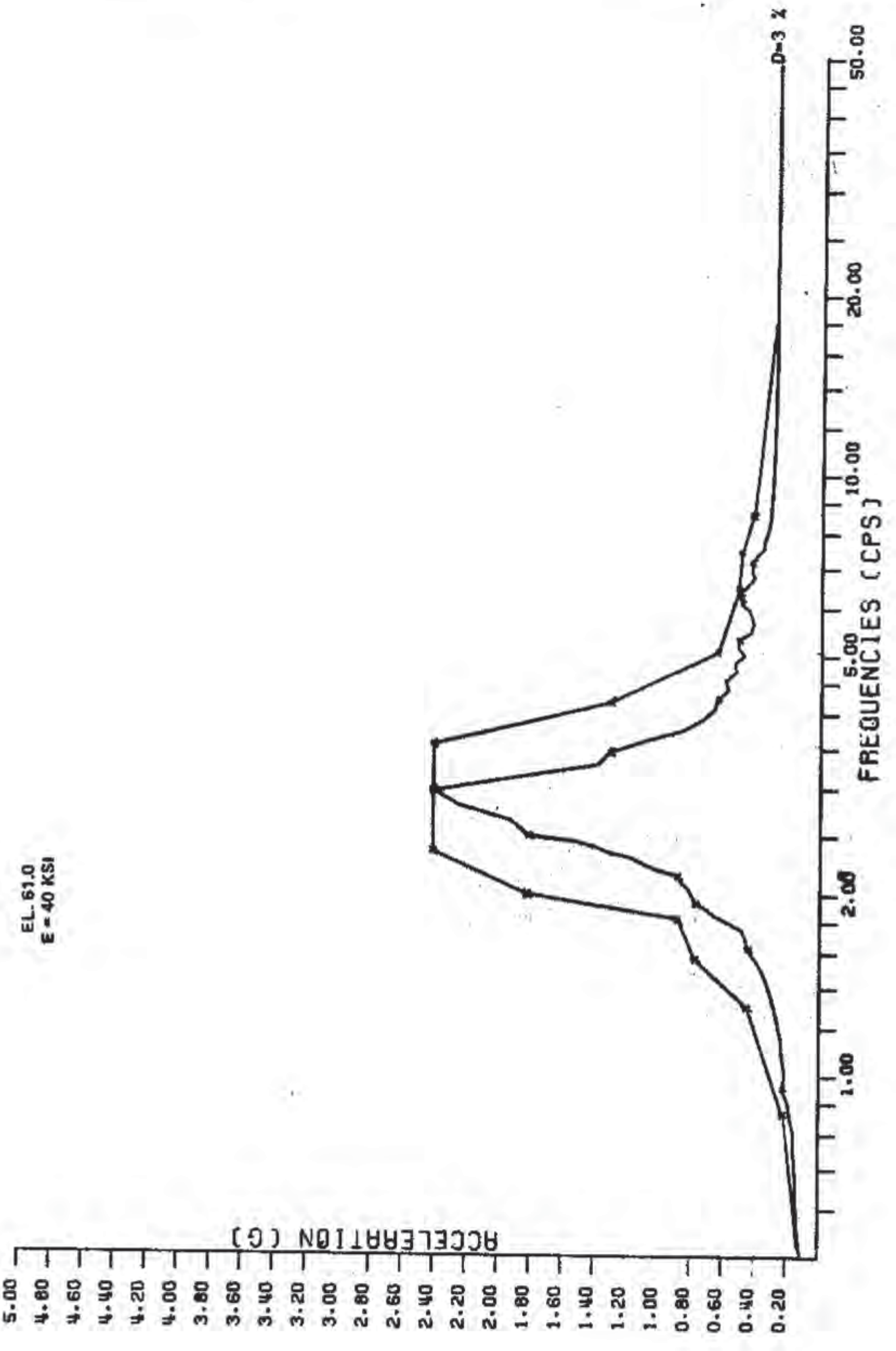
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–97



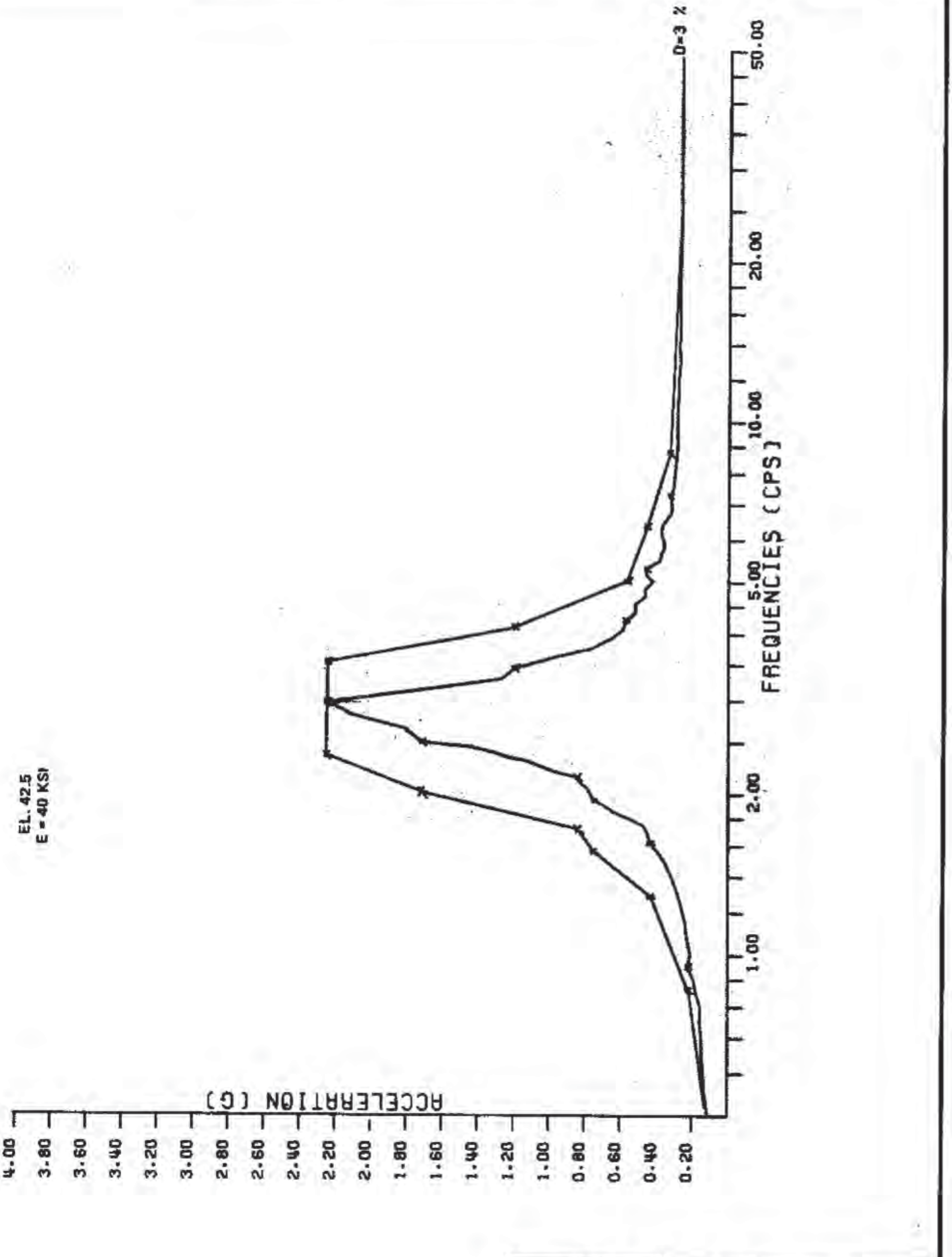
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
REACTOR AUXILIARY BUILDING
FIGURE 3.7-98



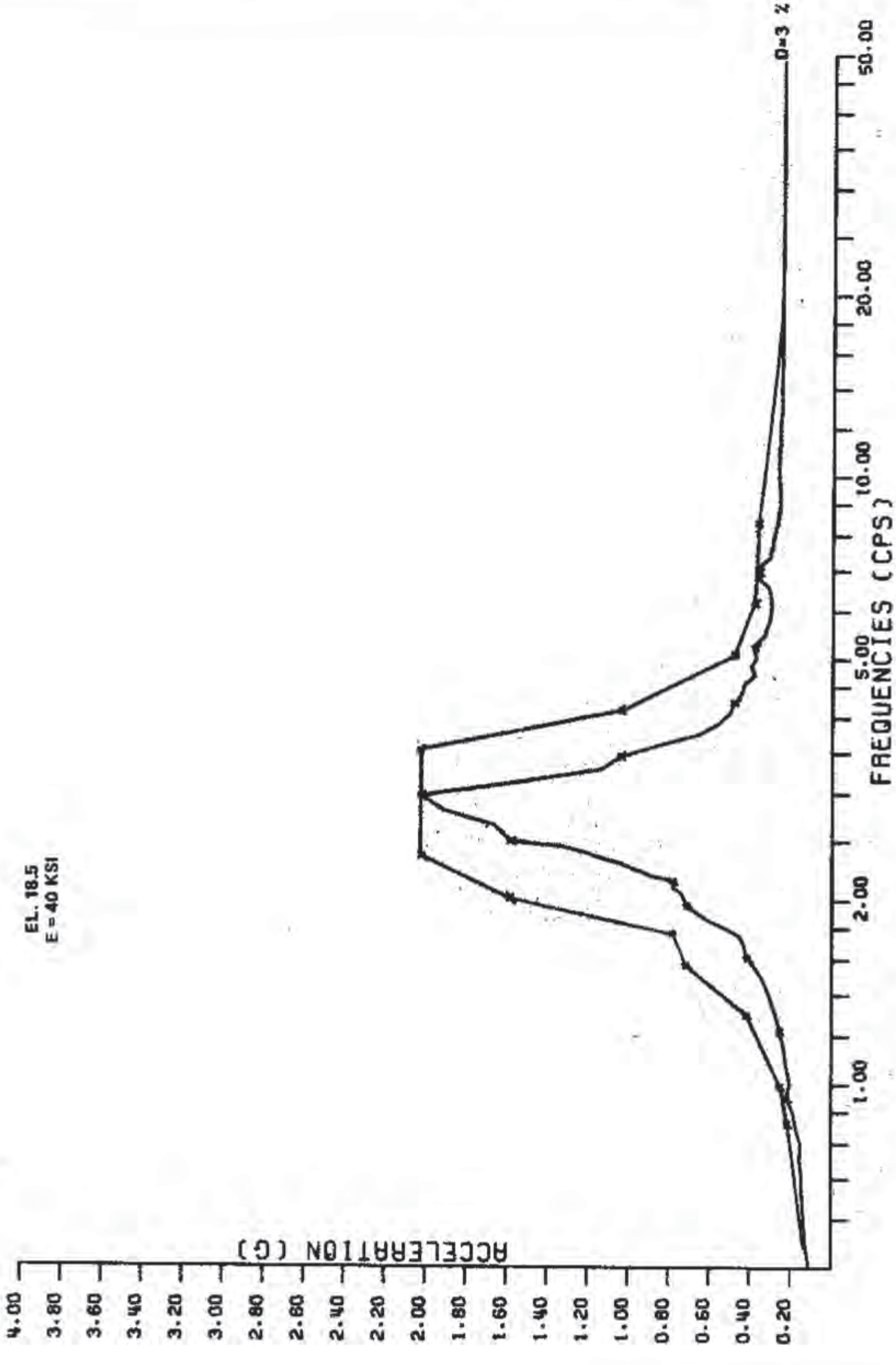
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
REACTOR AUXILIARY BUILDING
FIGURE 3.7-99



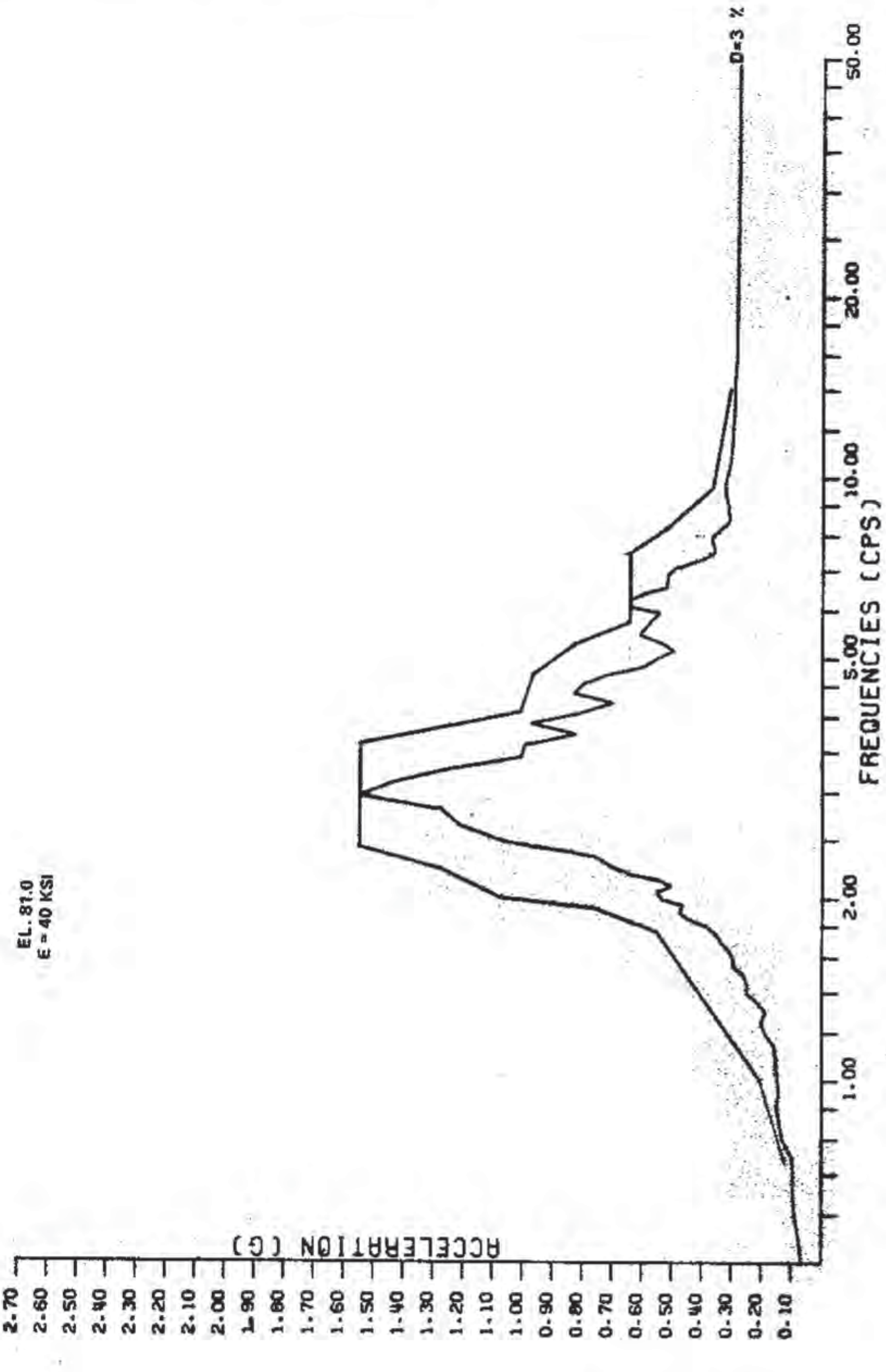
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–100



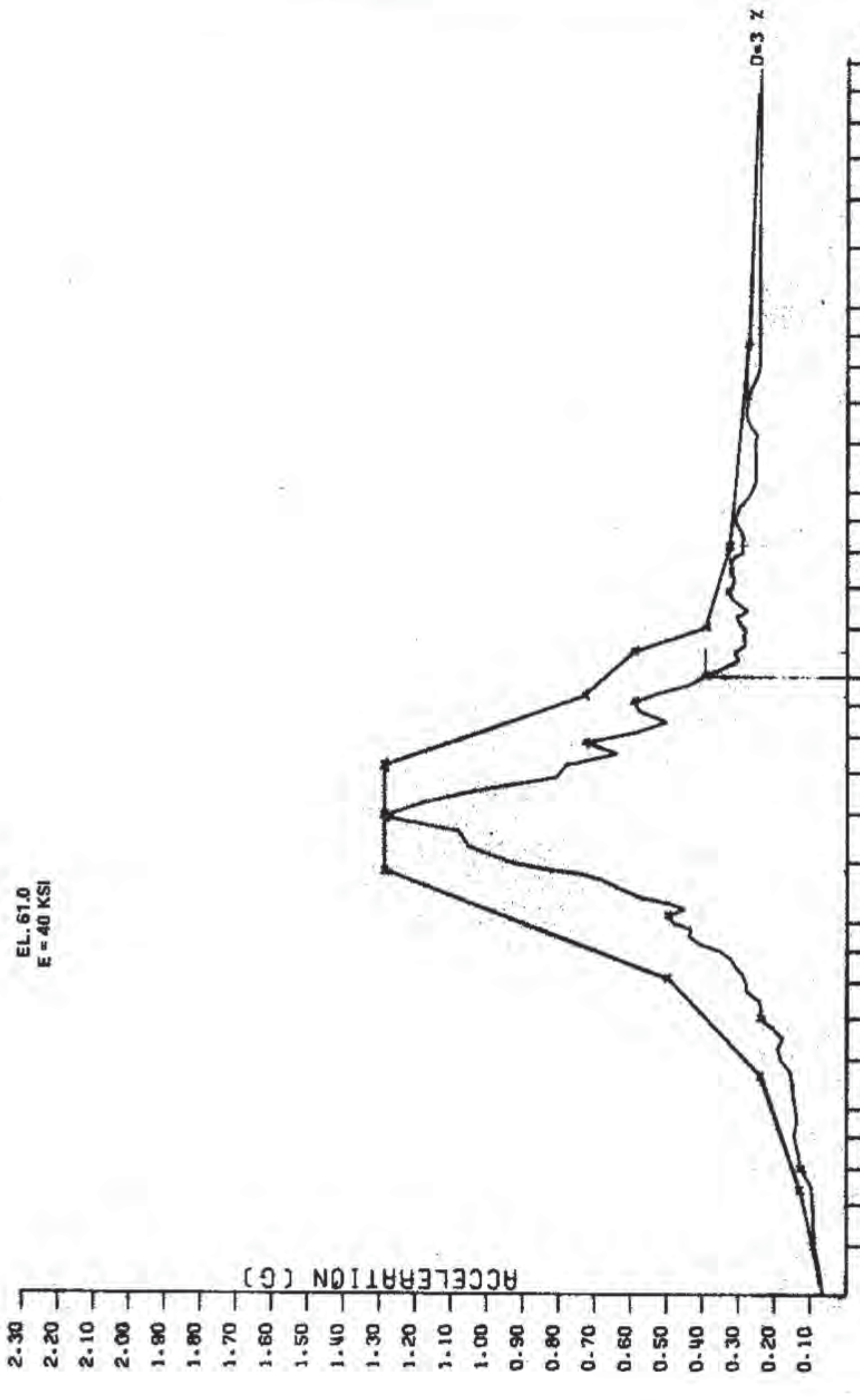
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–101



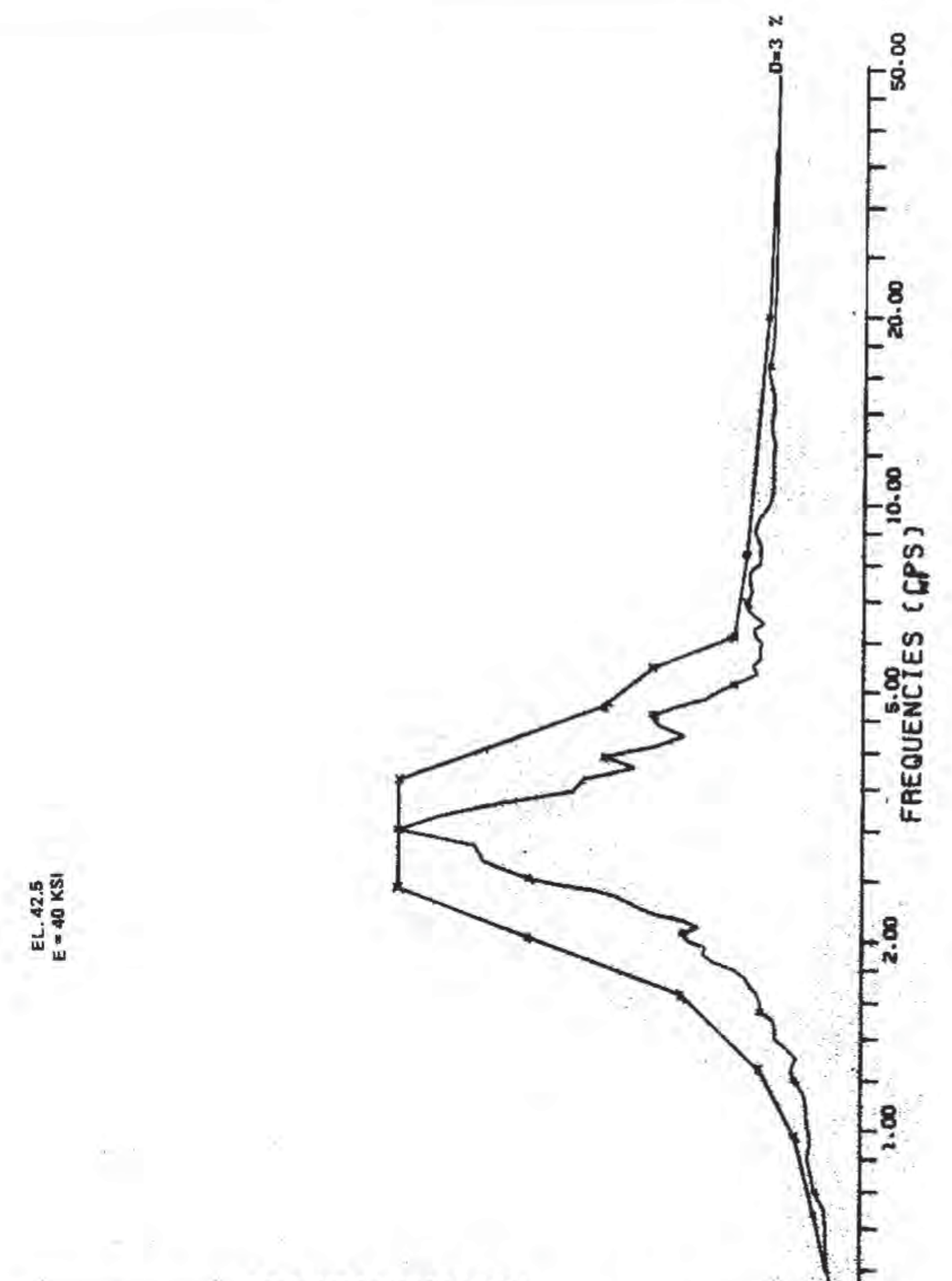
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR AUXILIARY BLDG MASS PT. 5
FIGURE 3.7–102



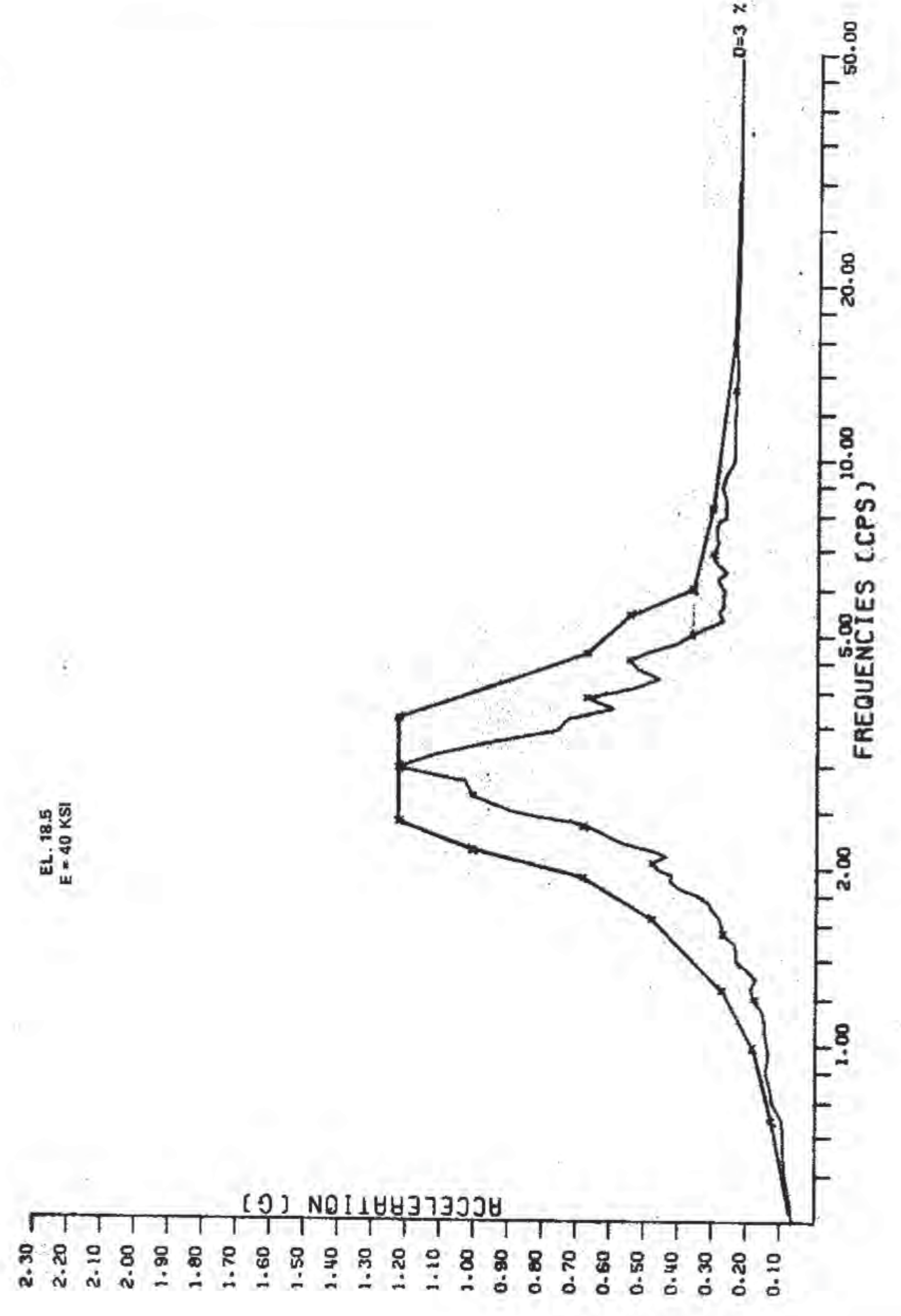
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR AUXILIARY BLDG. MASS PT. 6
FIGURE 3.7–103



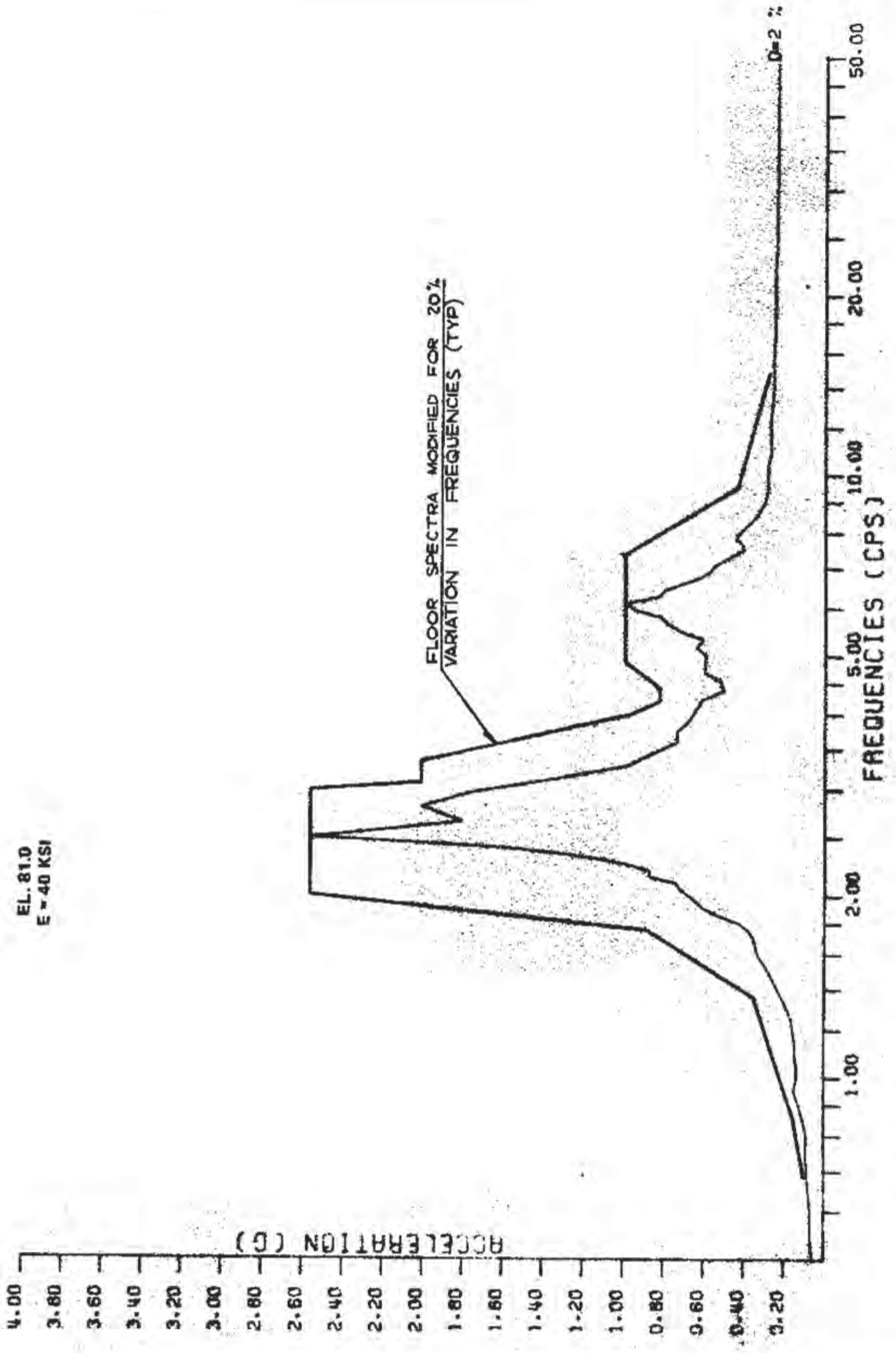
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR AUXILIARY BLDG. MASS PT. 7
FIGURE 3.7–104



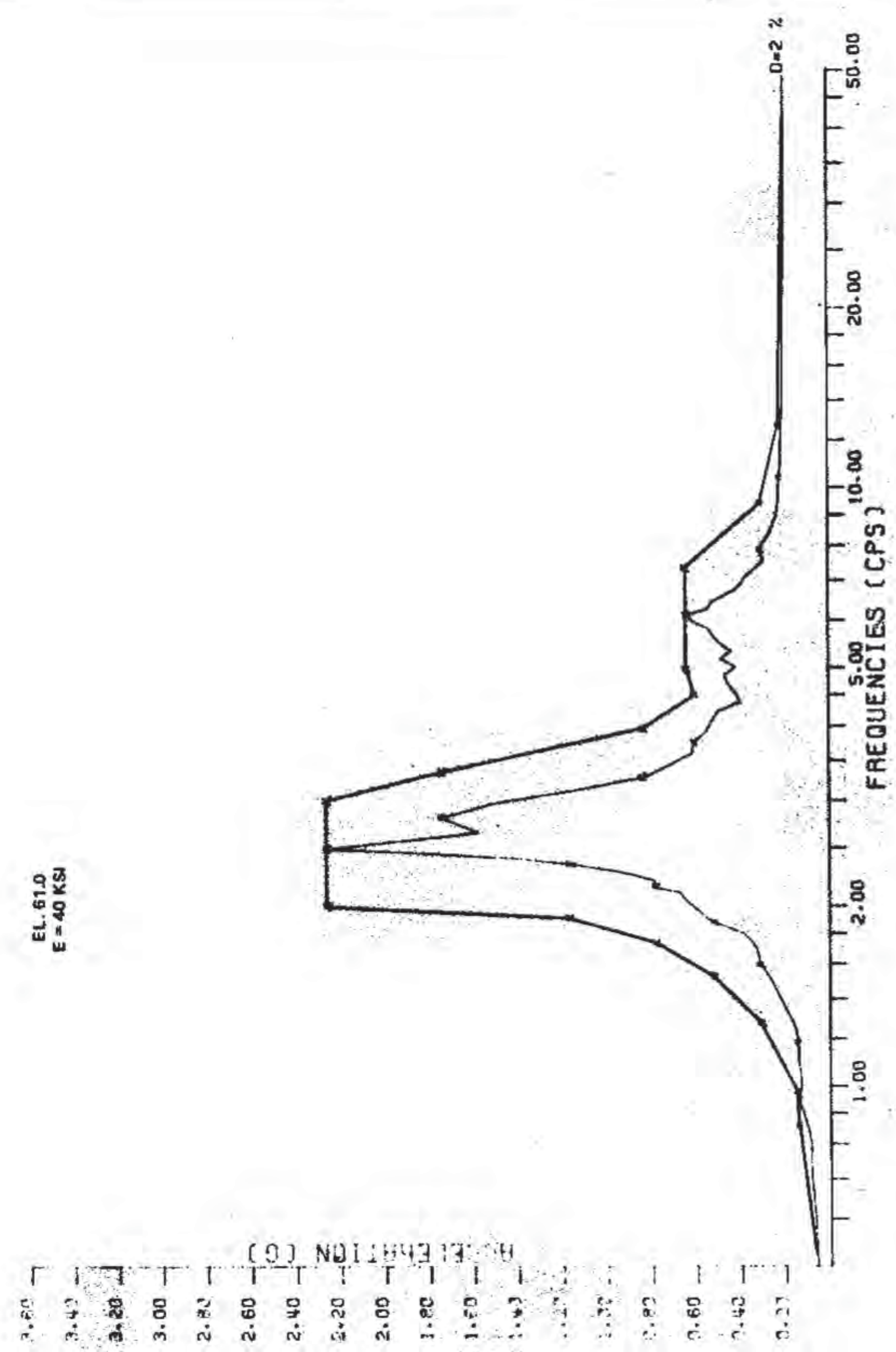
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
REACTOR AUXILIARY BLDG MASS PT. 8
FIGURE 3.7–105



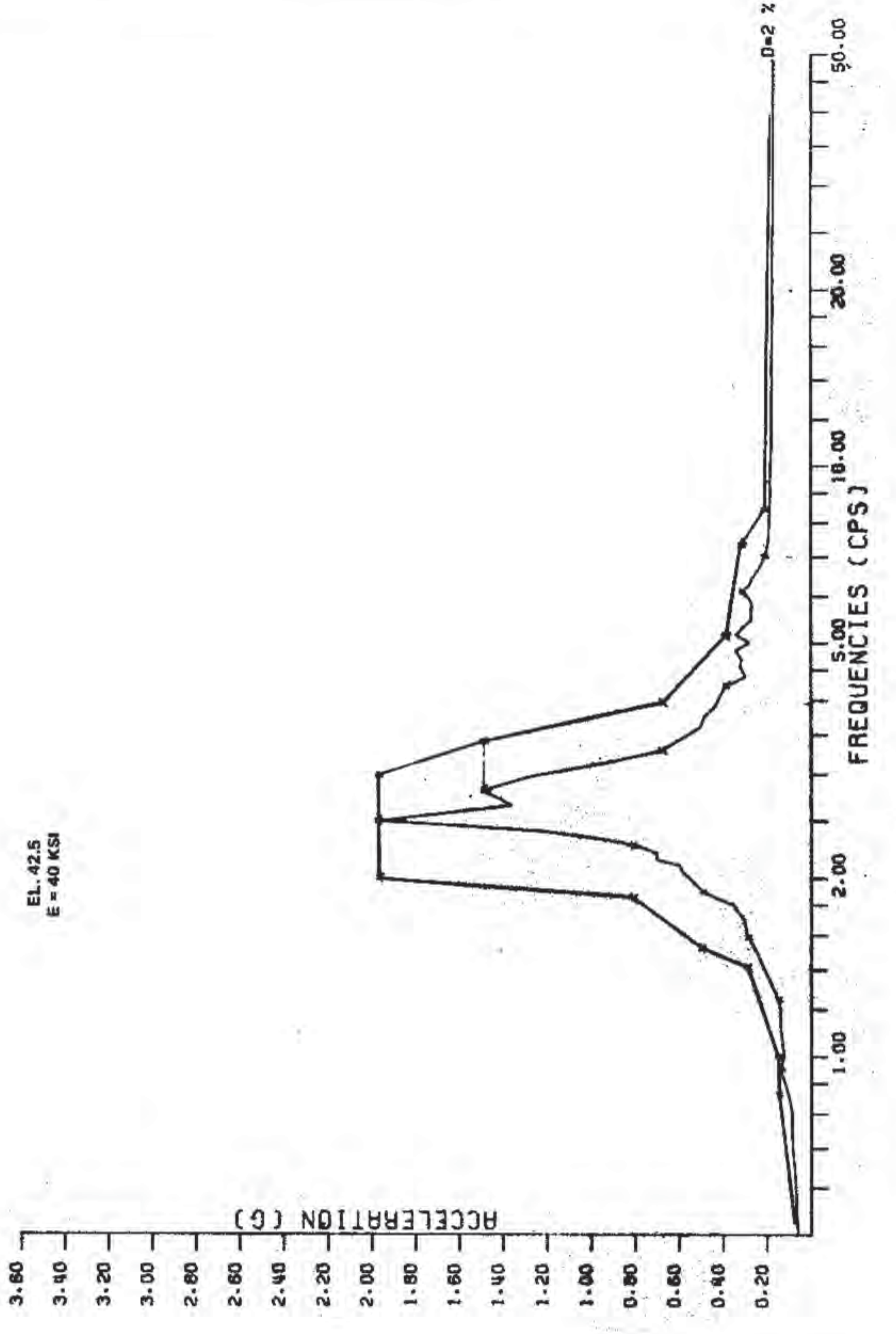
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–106



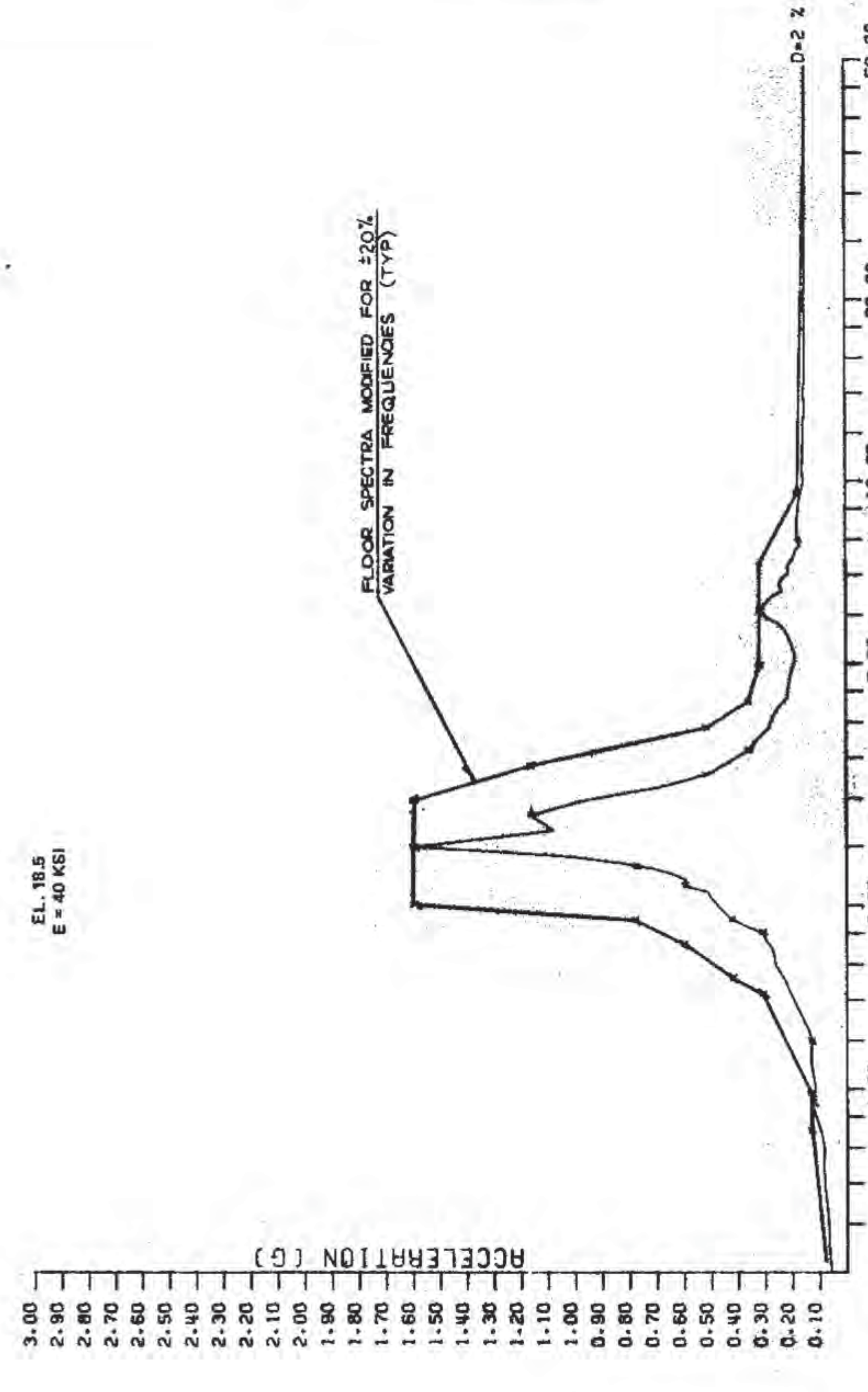
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–107



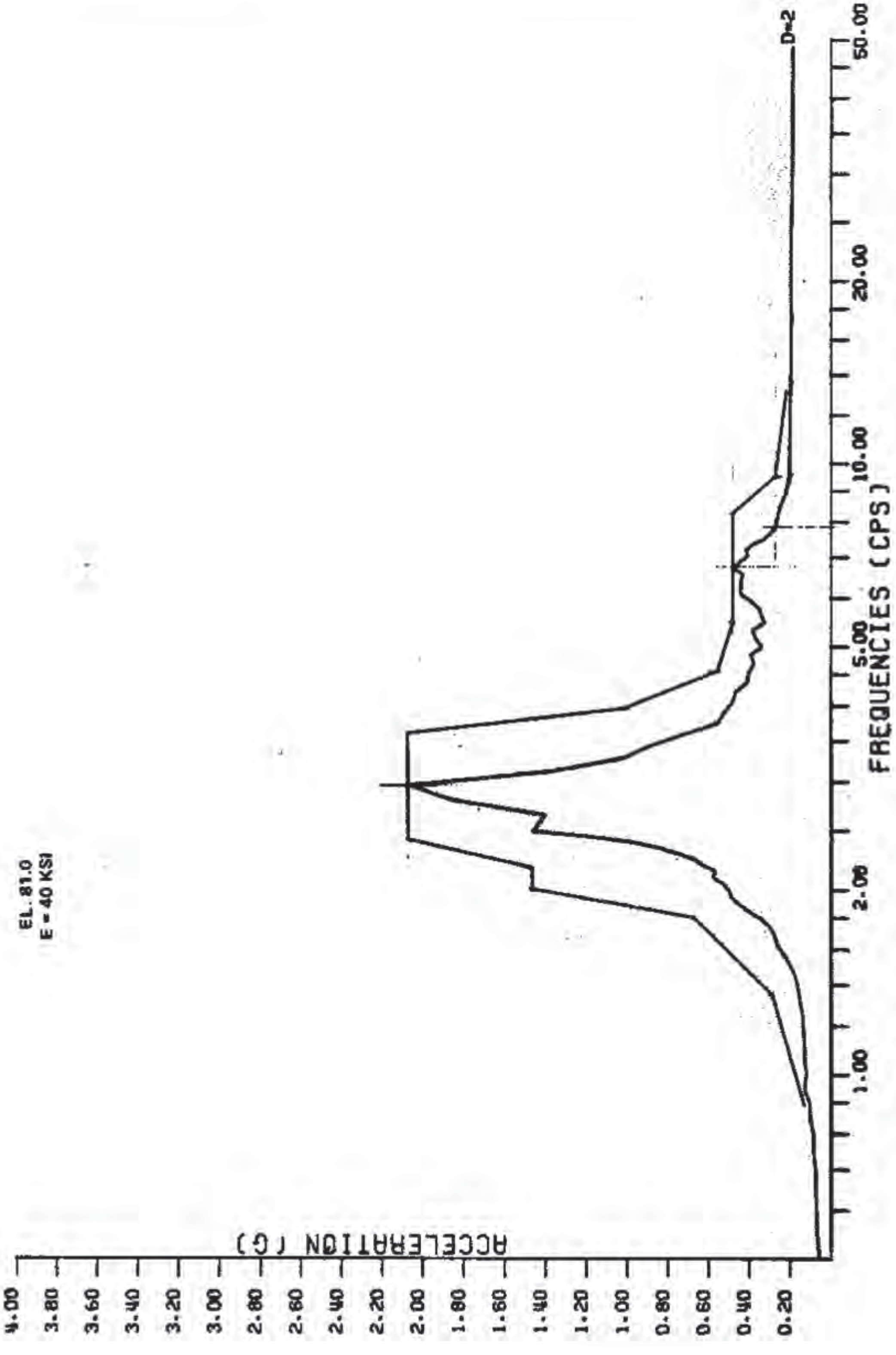
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–108



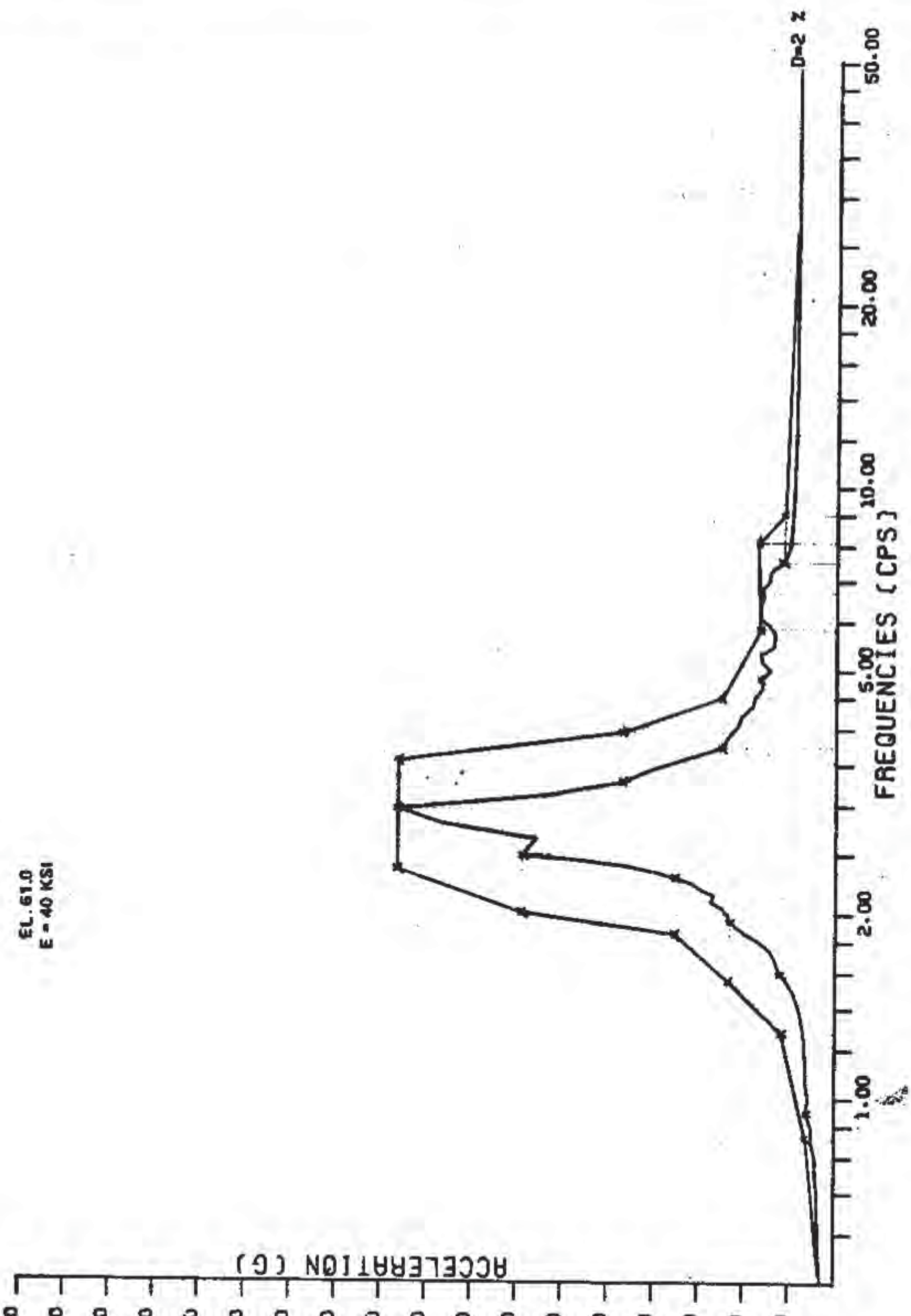
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
REACTOR AUXILIARY BUILDING
FIGURE 3.7-109



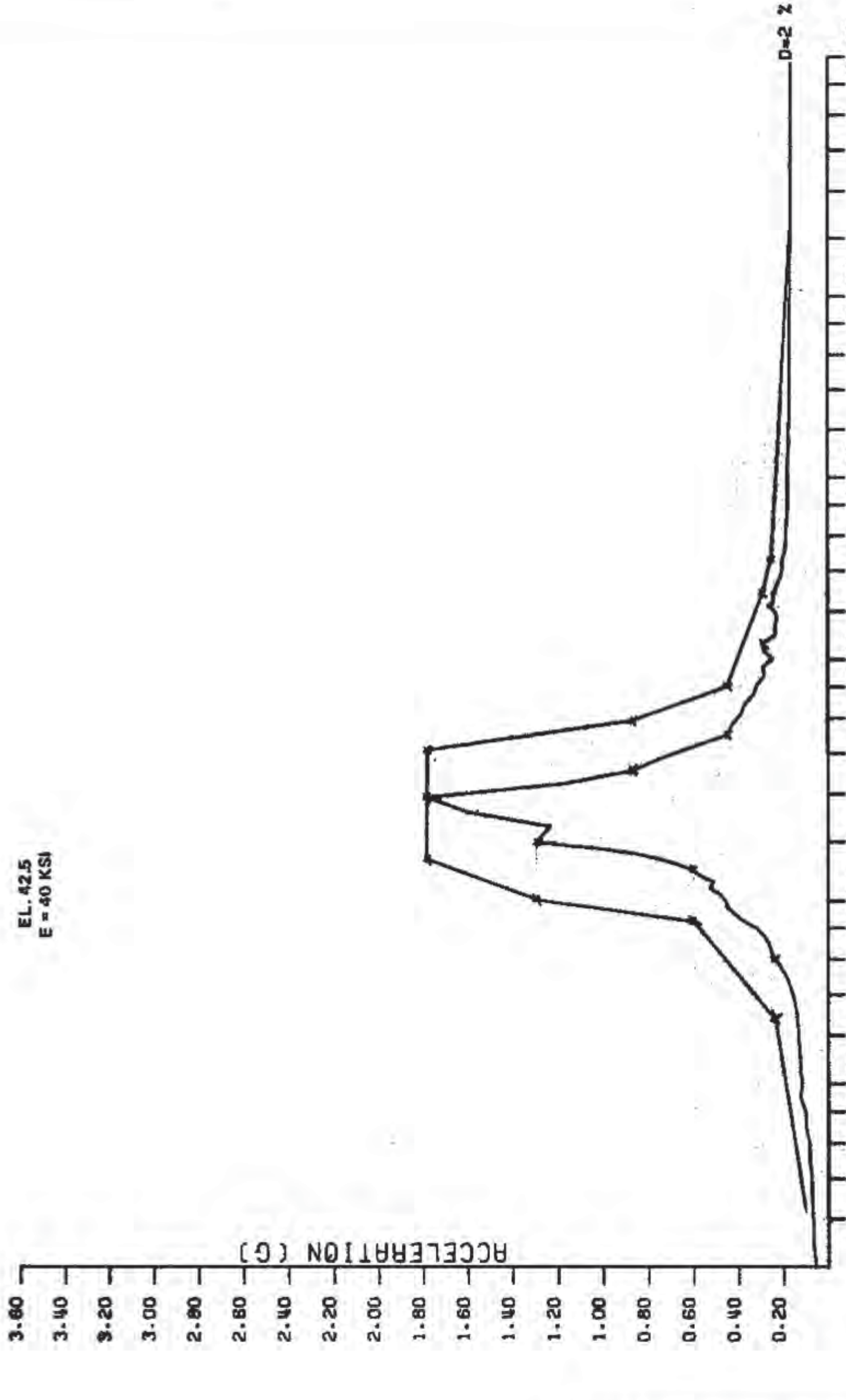
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–110



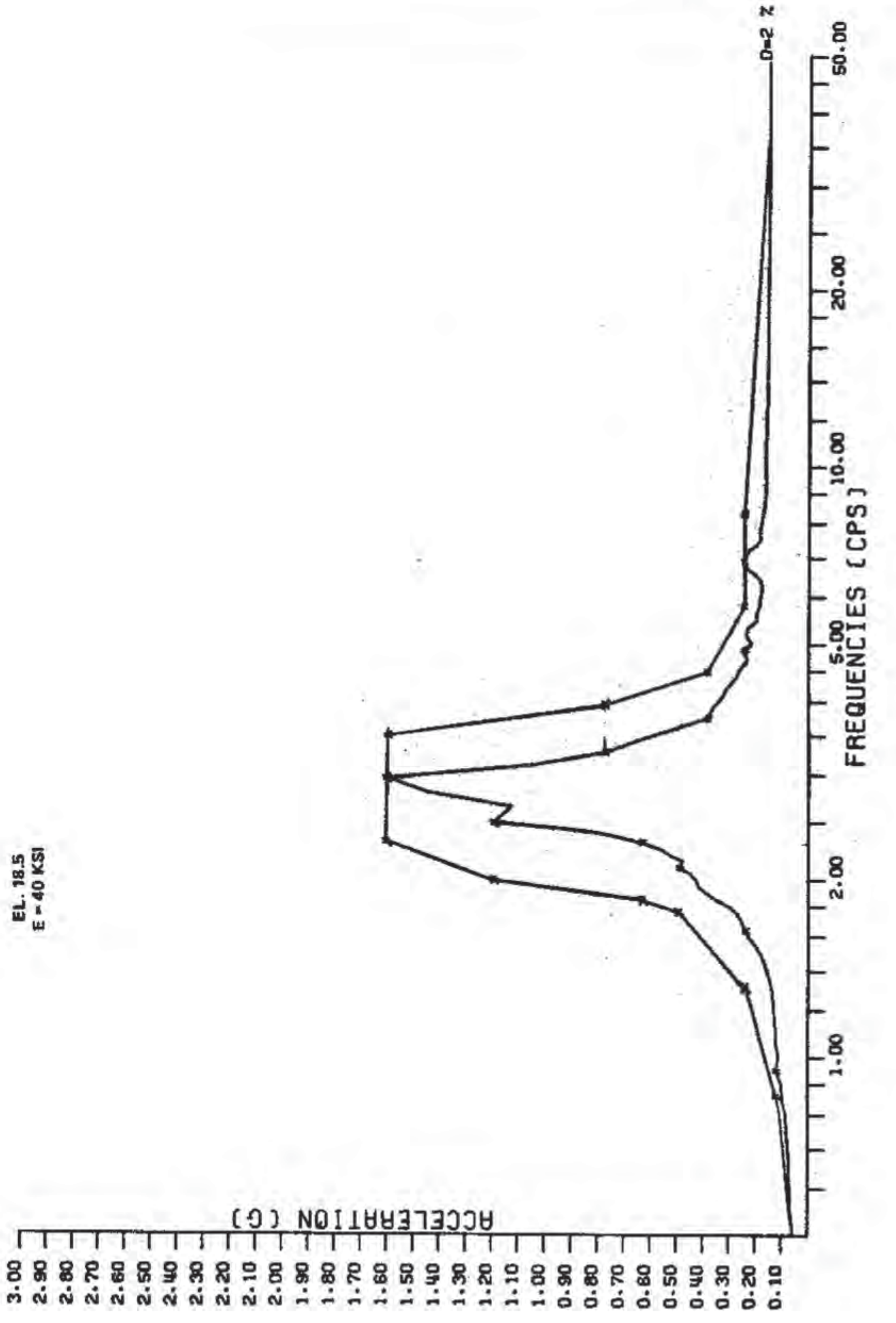
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–111



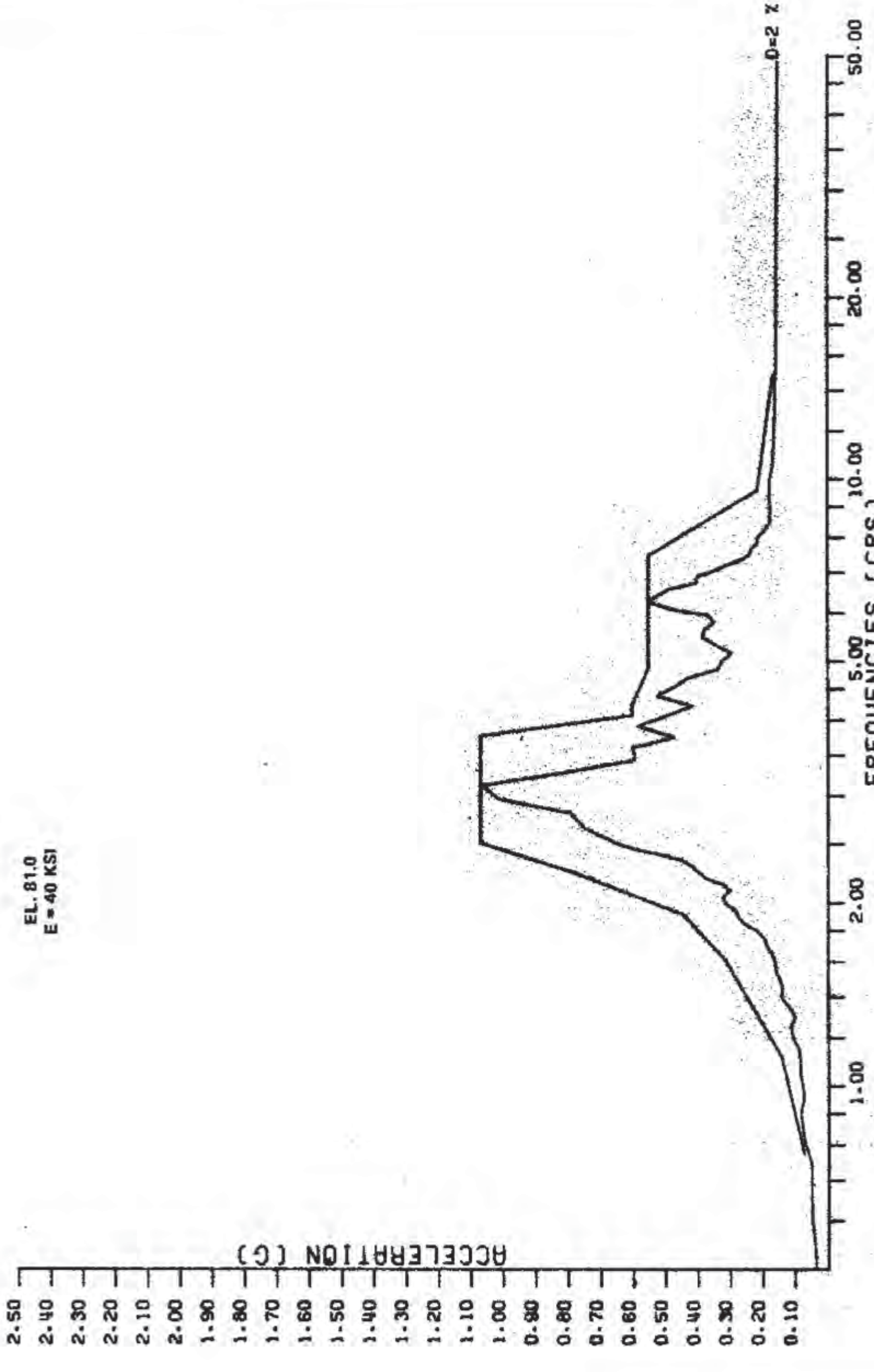
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–112



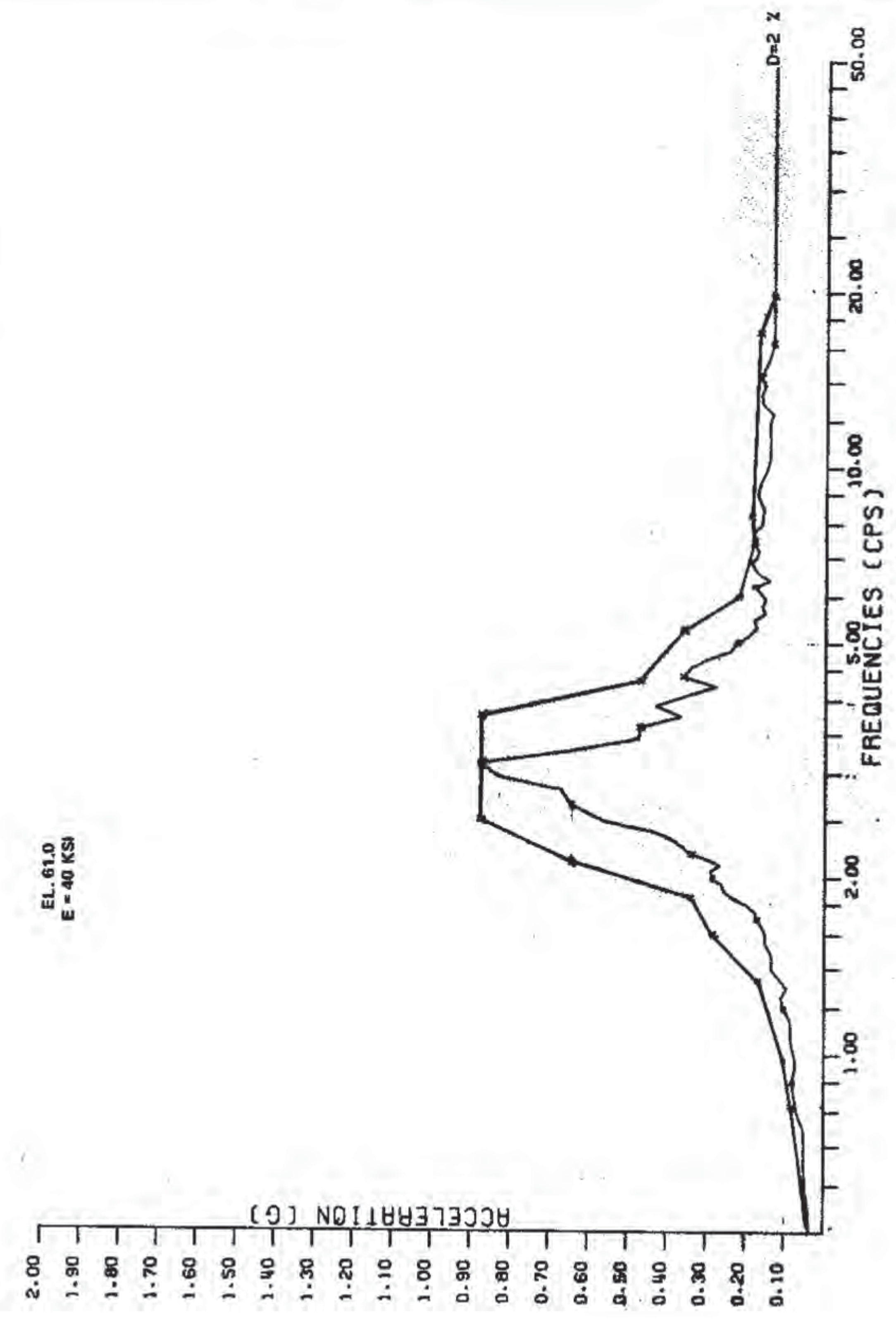
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
REACTOR AUXILIARY BUILDING
FIGURE 3.7–113



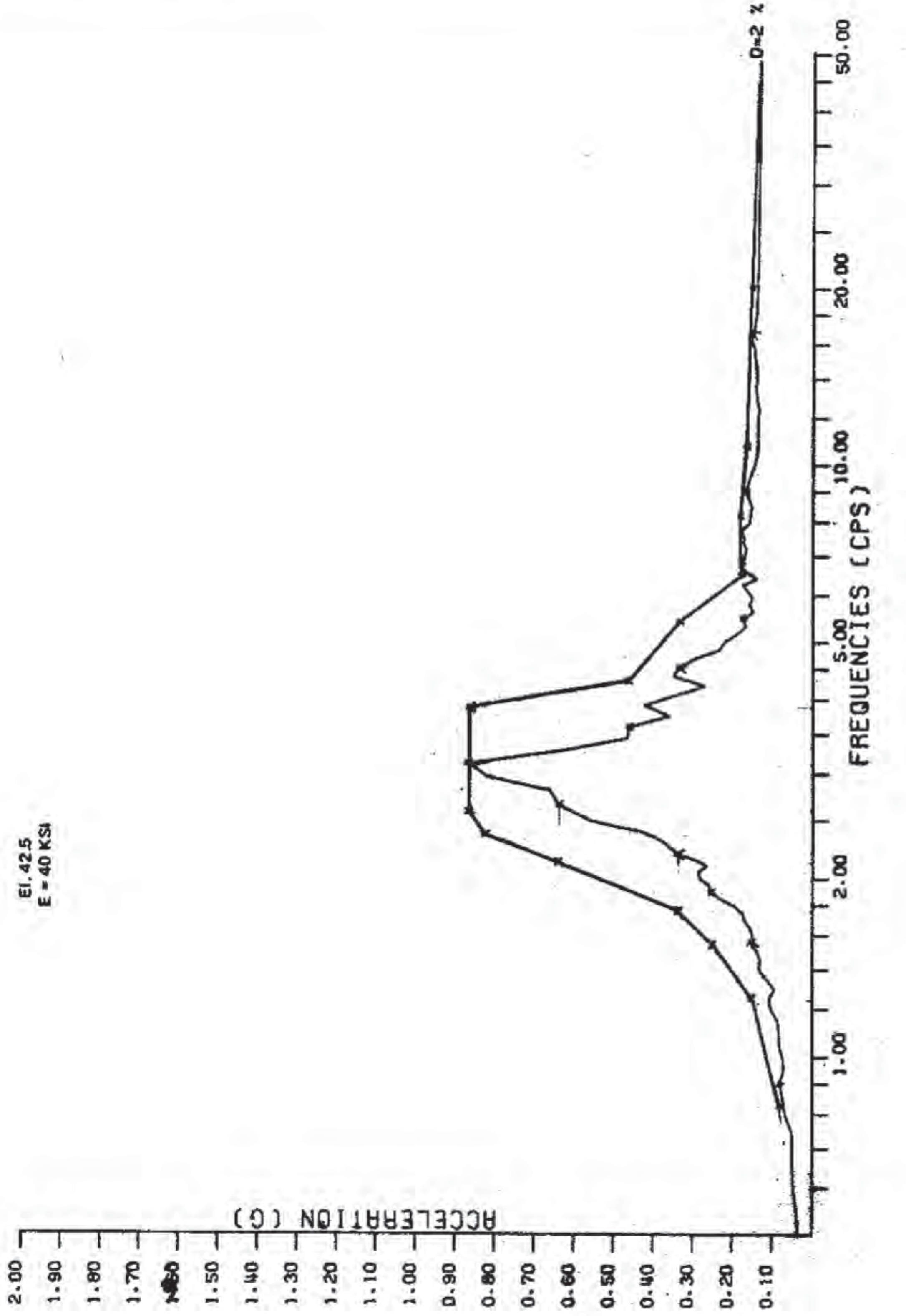
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR AUXILIARY BLDG. MASS PT. 5
FIGURE 3.7-114



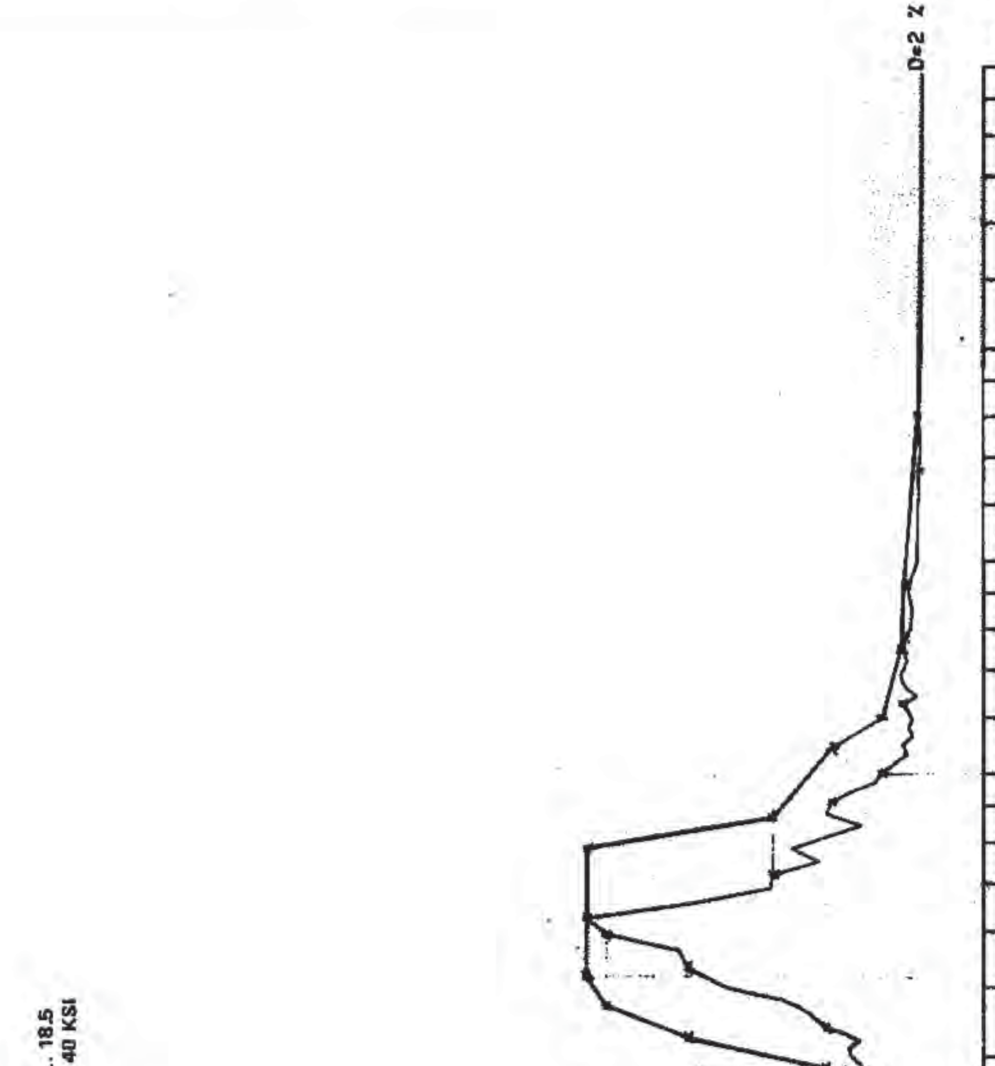
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR AUXILIARY BLDG. MASS PT. 6
FIGURE 3.7-115



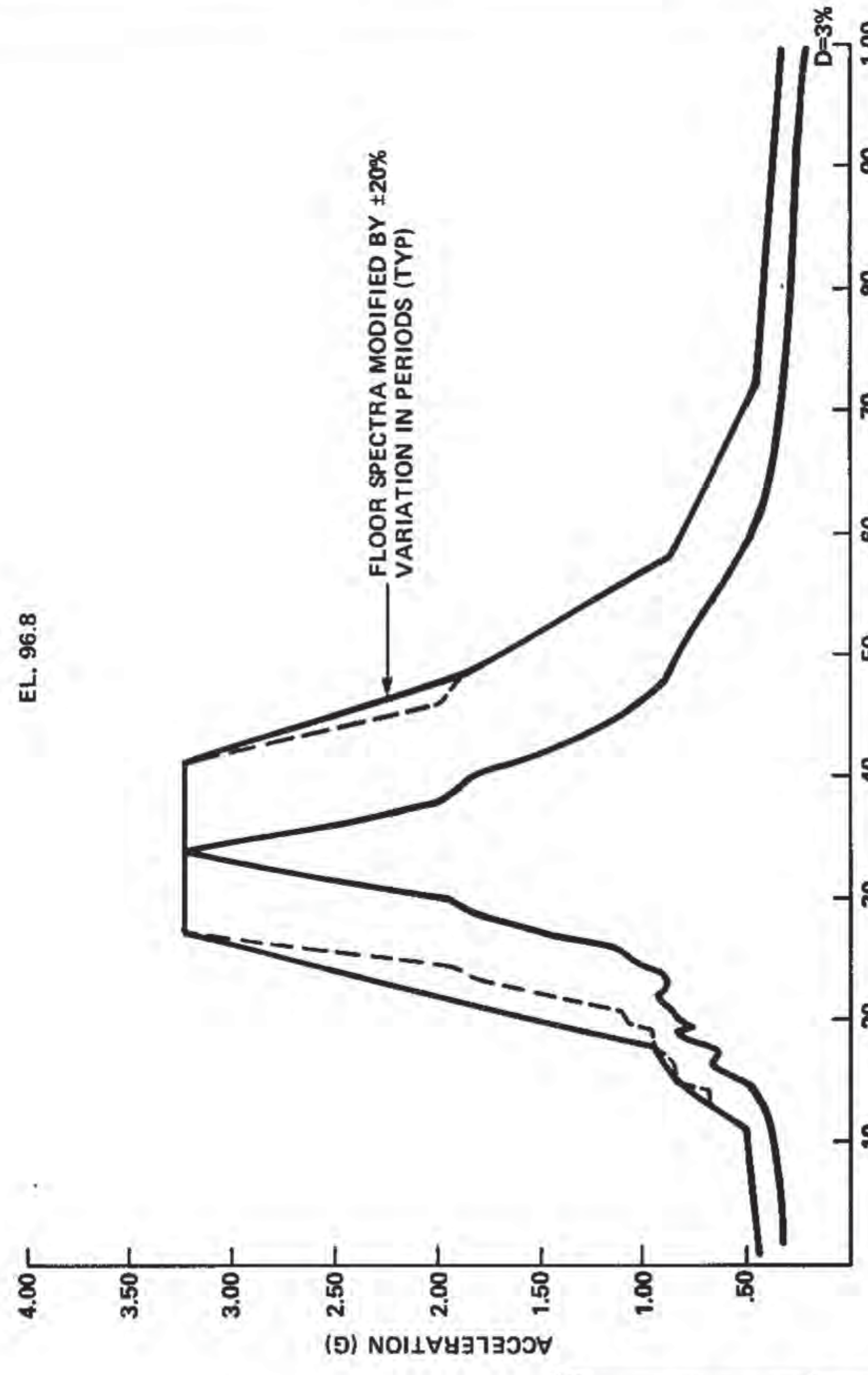
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR AUXILIARY BLDG. MASS PT. 7
FIGURE 3.7–116



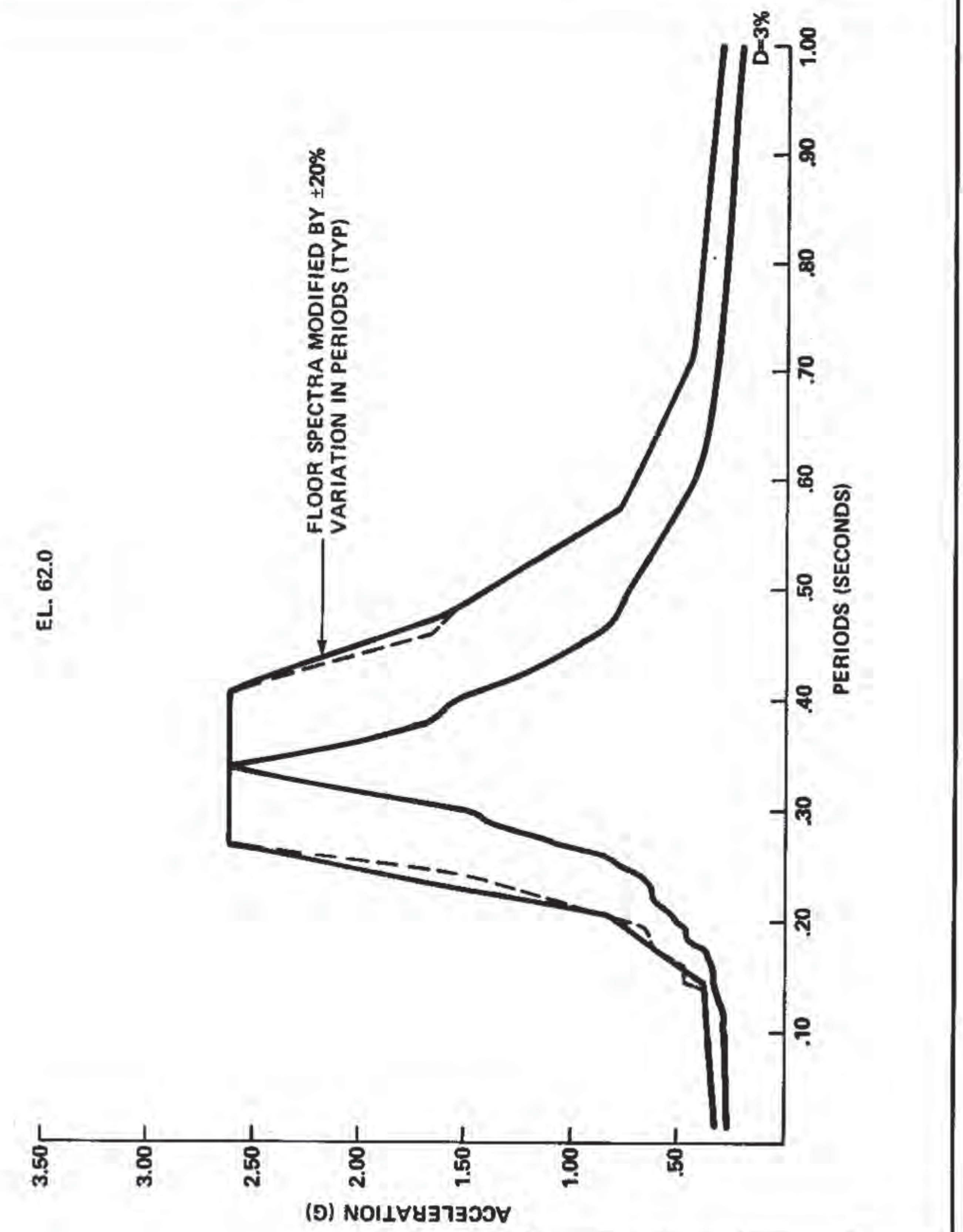
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
REACTOR AUXILIARY BLDG. MASS PT. 8
FIGURE 3.7–117



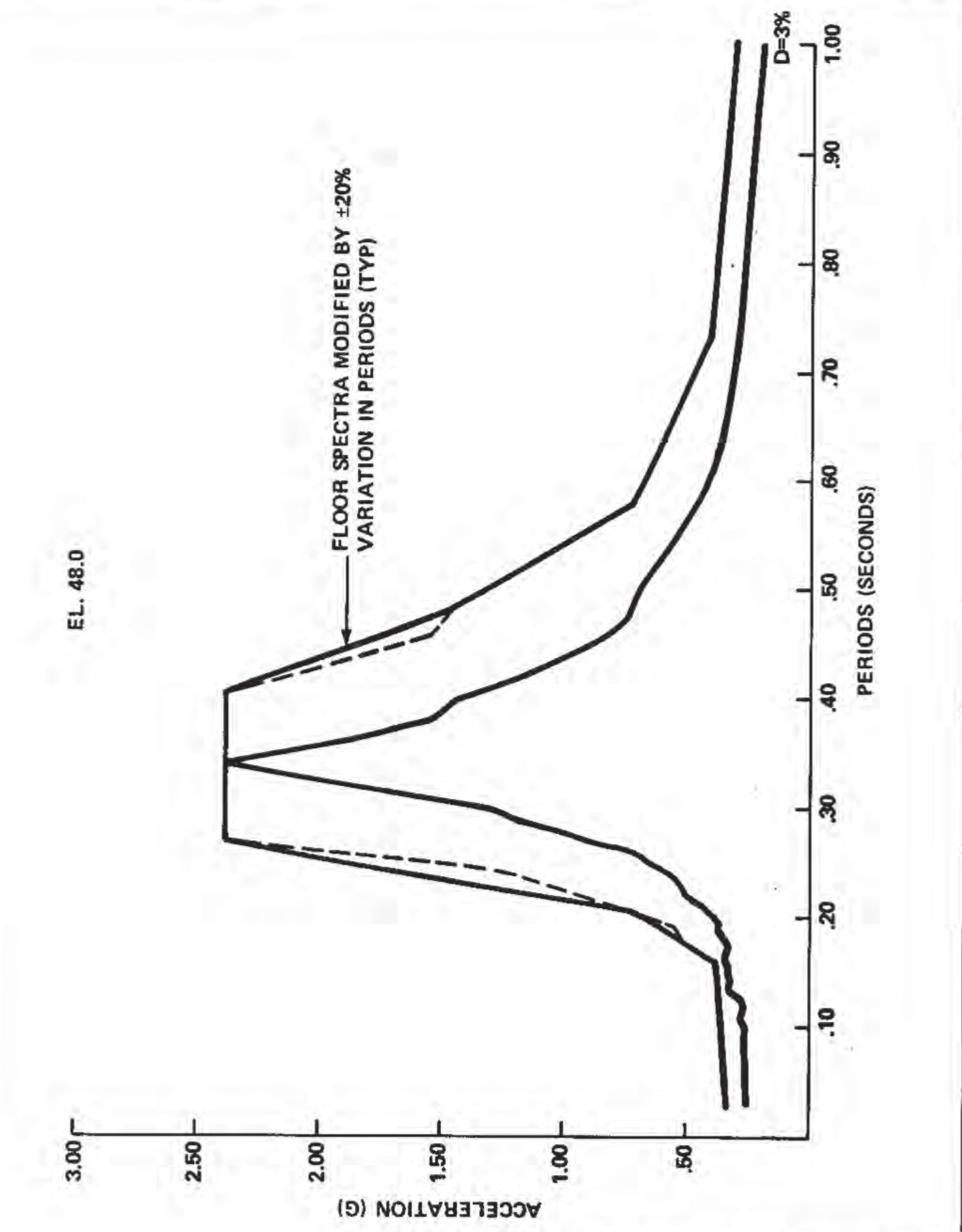
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
FUEL HANDLING BLDG.
FIGURE 3.7-118



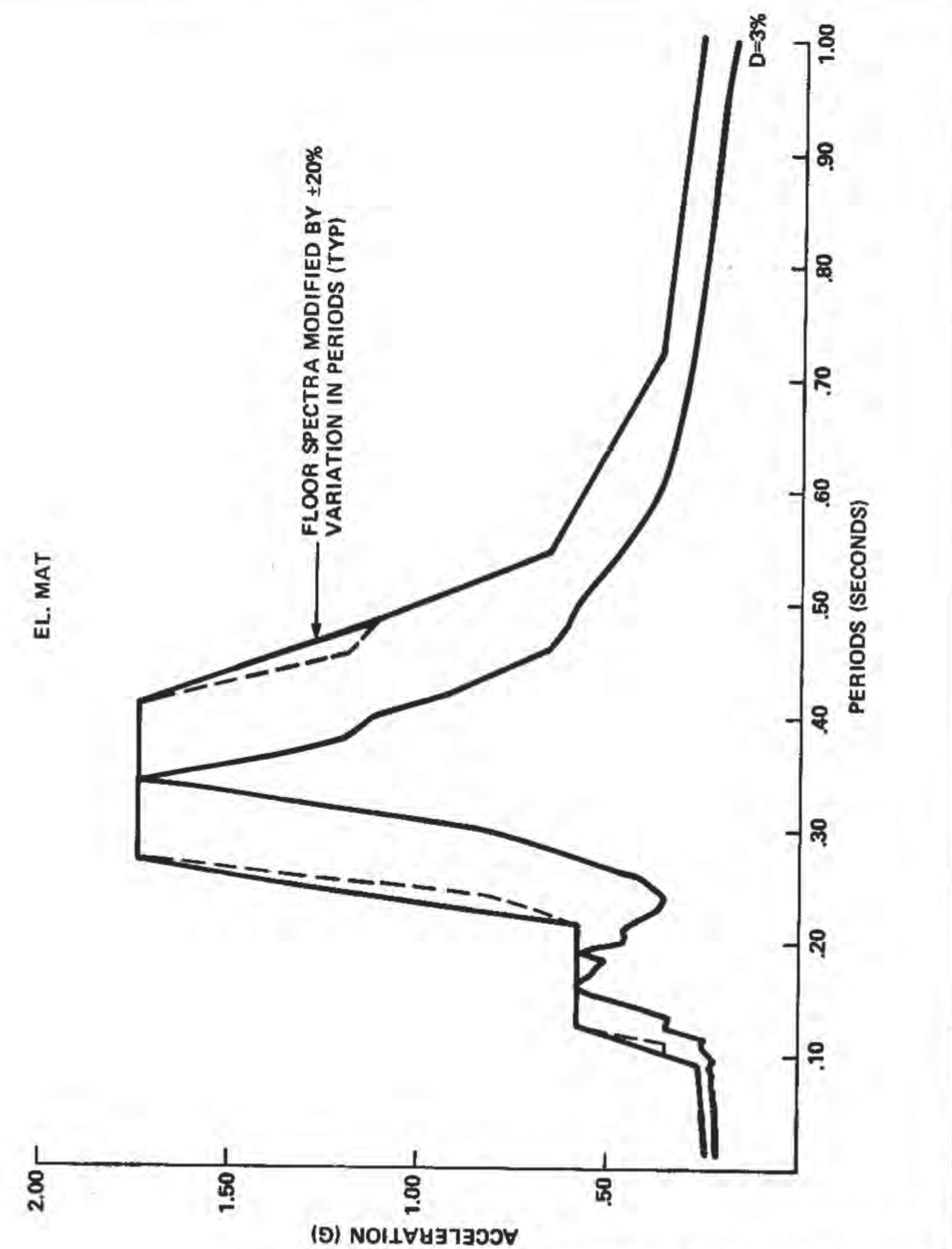
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
FUEL HANDLING BLDG.
FIGURE 3.7-119



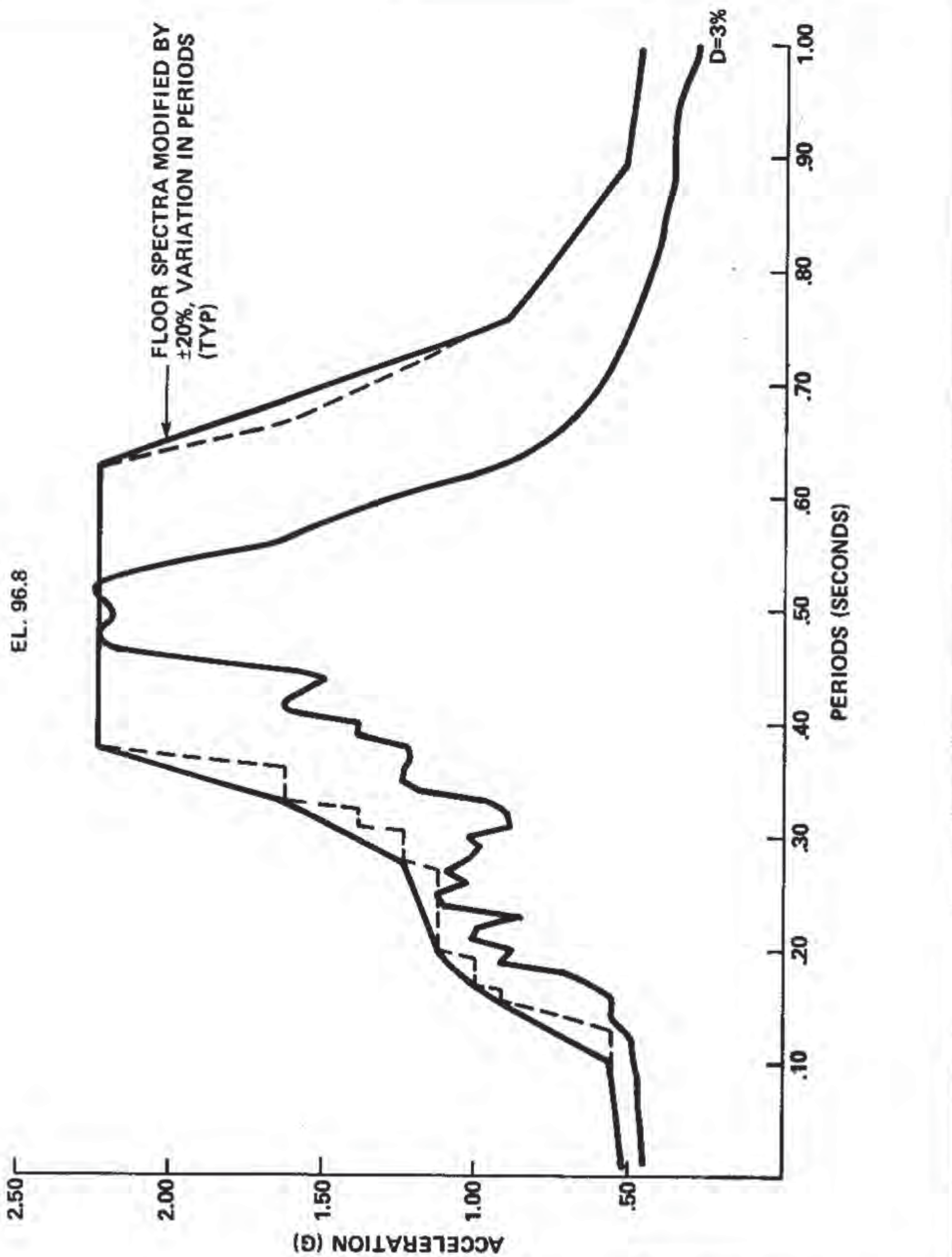
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
FUEL HANDLING BLDG.
FIGURE 3.7-120



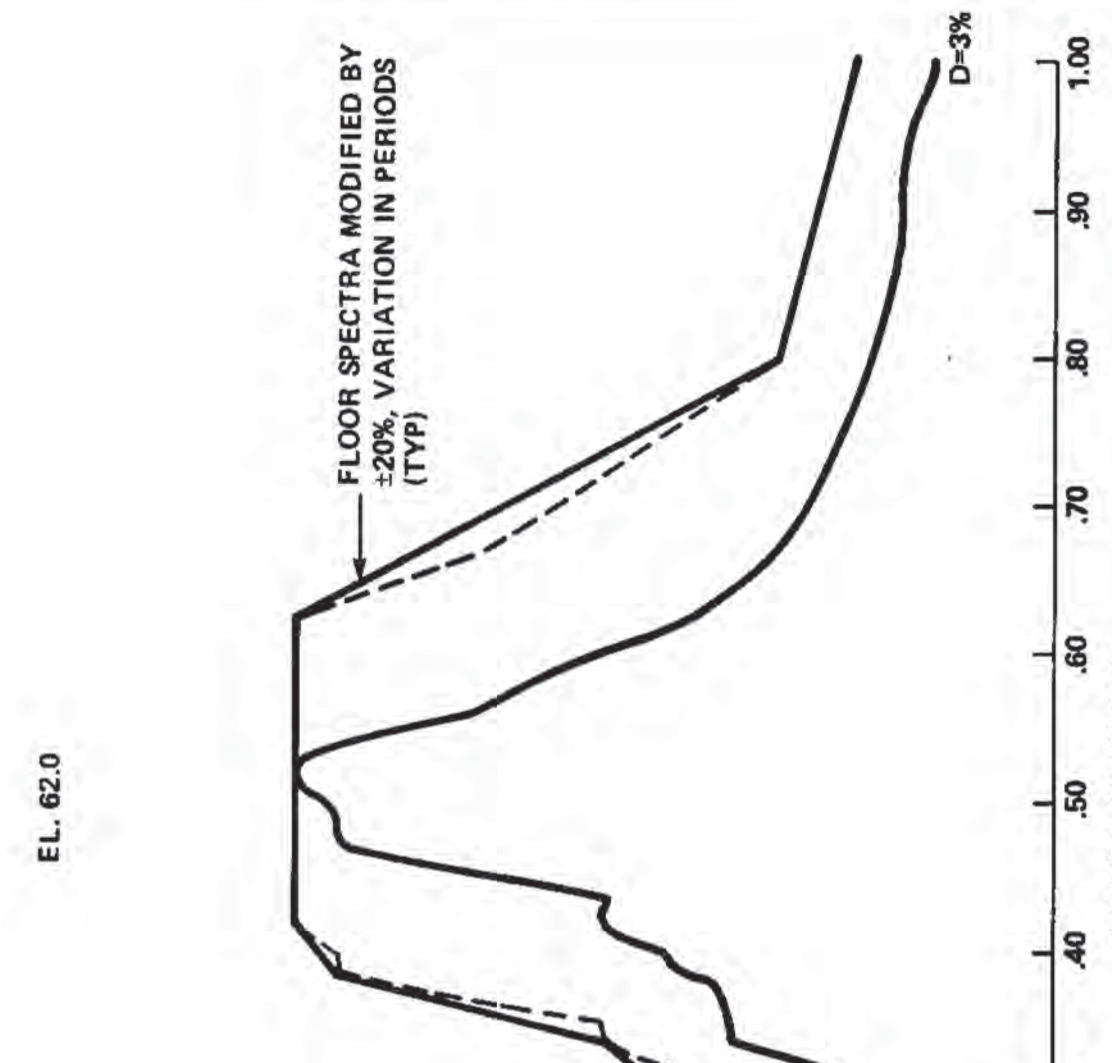
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
FUEL HANDLING BLDG.
FIGURE 3.7–121



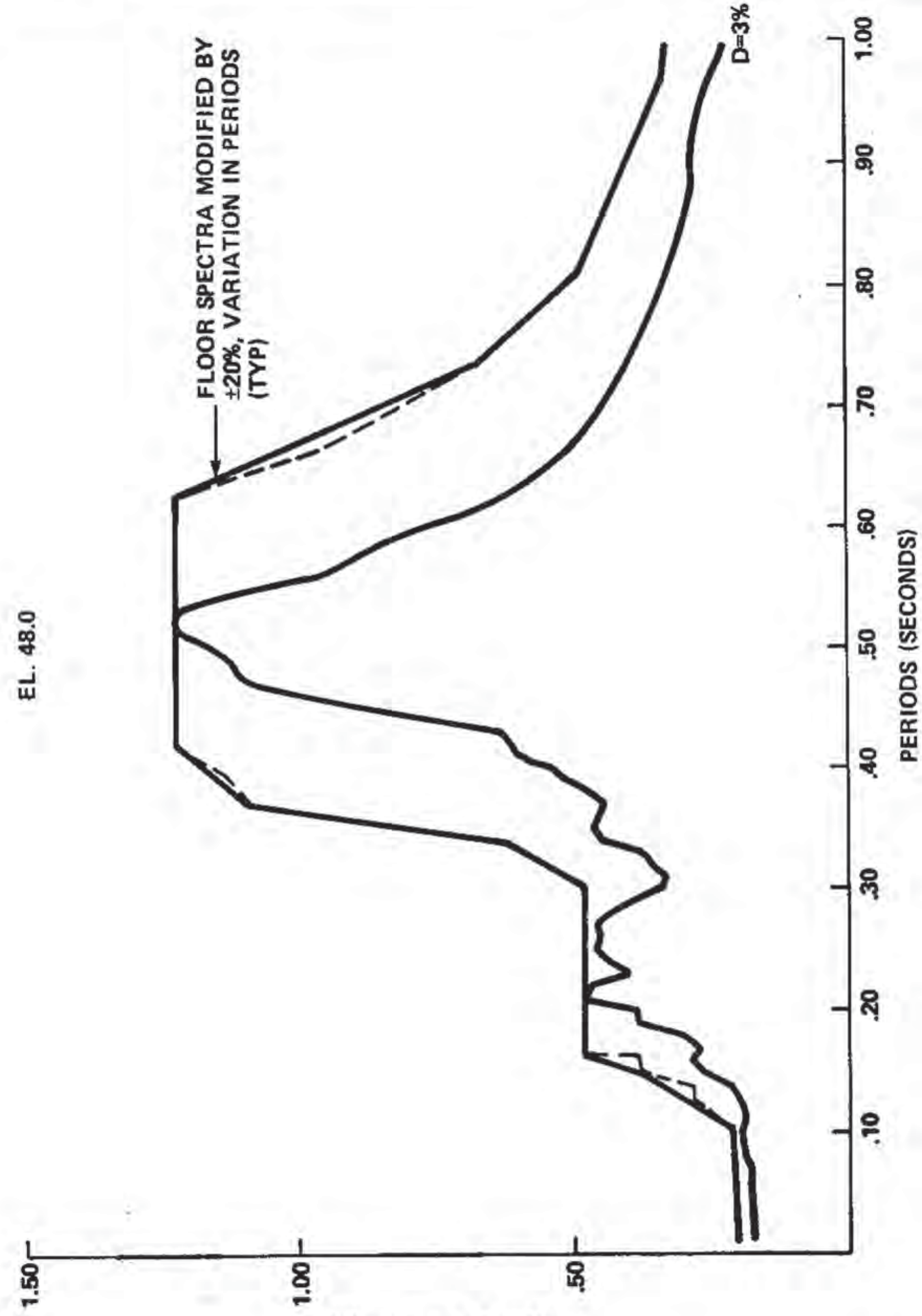
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
FUEL HANDLING BLDG.
FIGURE 3.7–122



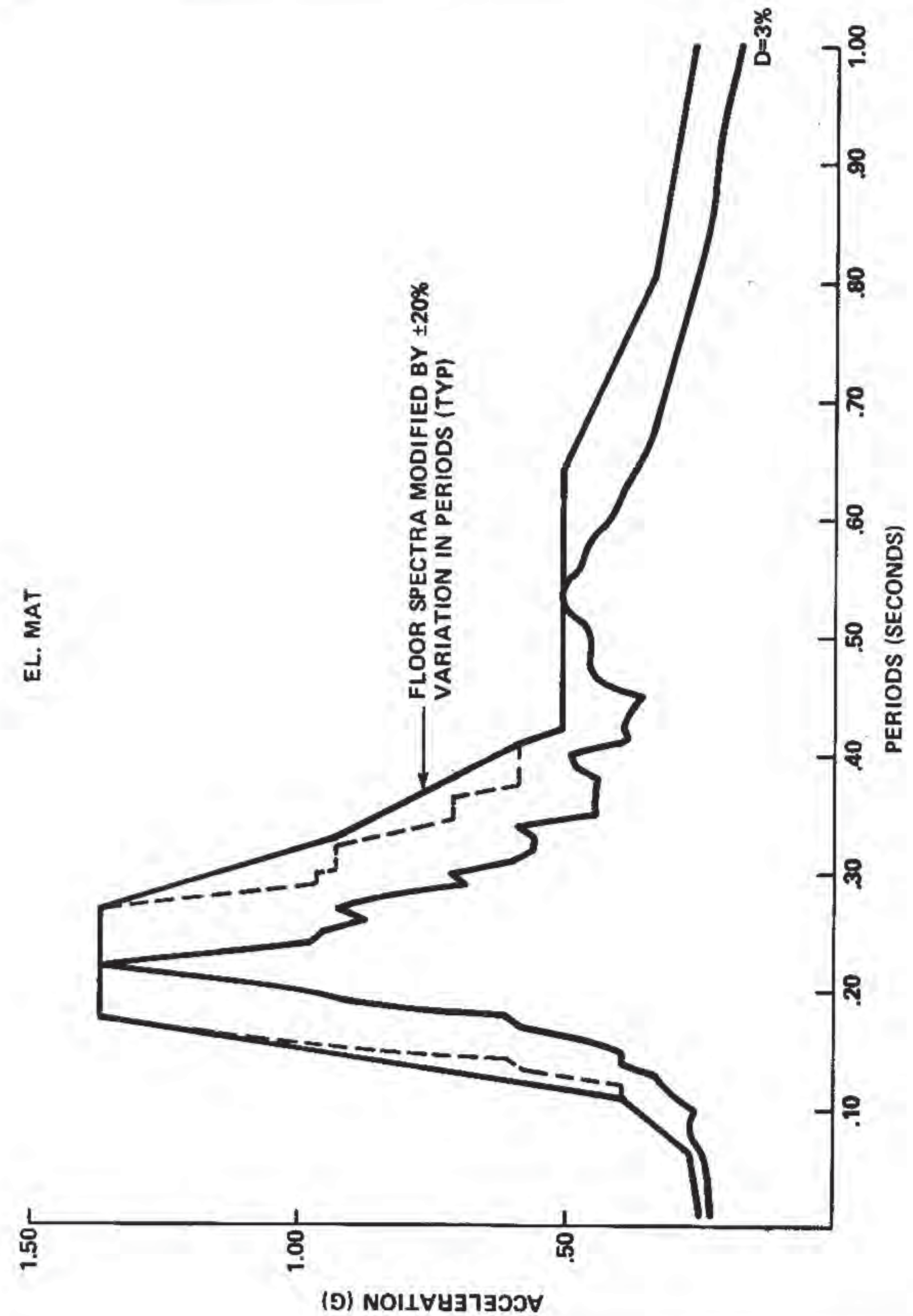
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
FUEL HANDLING BLDG.
FIGURE 3.7–123



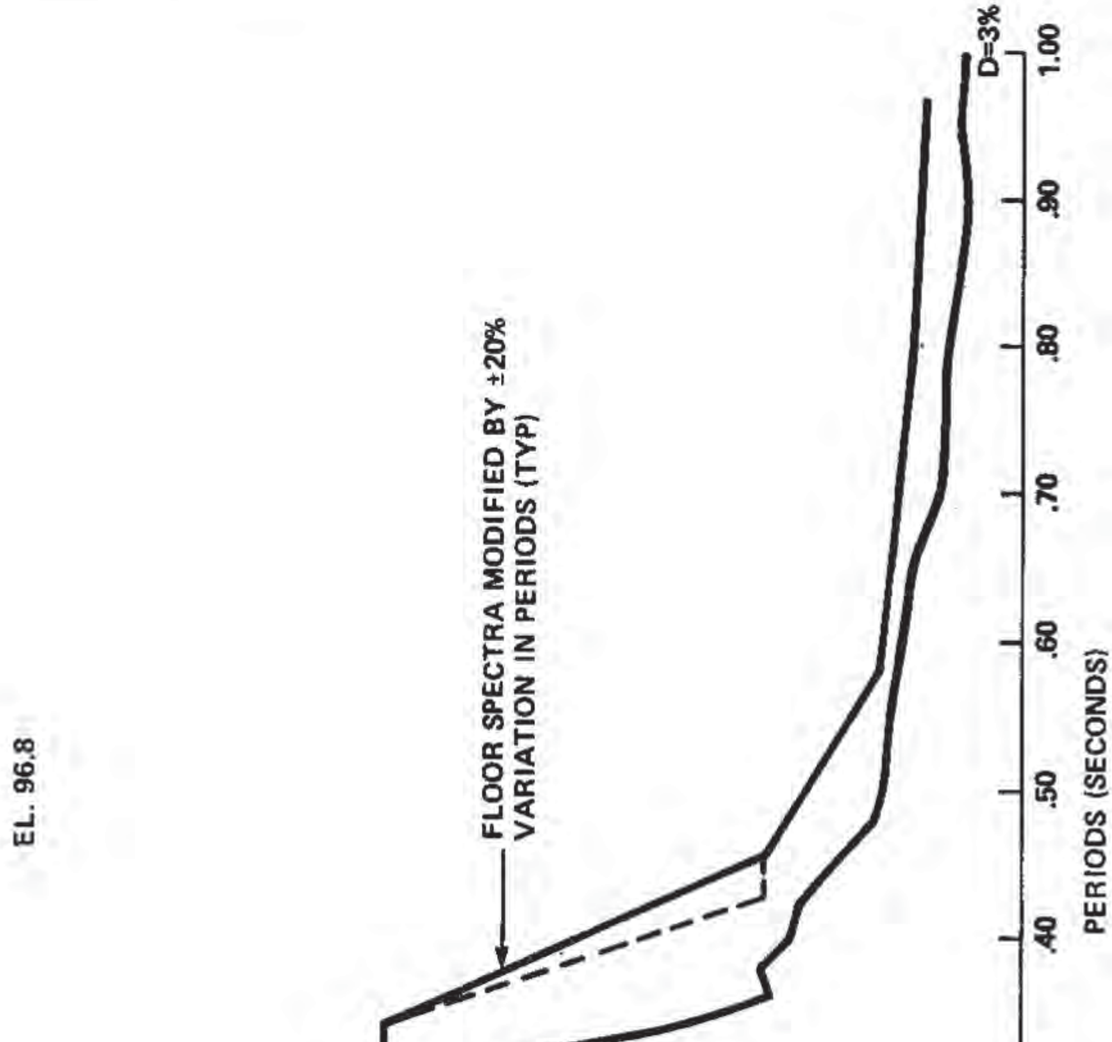
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
FUEL HANDLING BLDG.
FIGURE 3.7–124



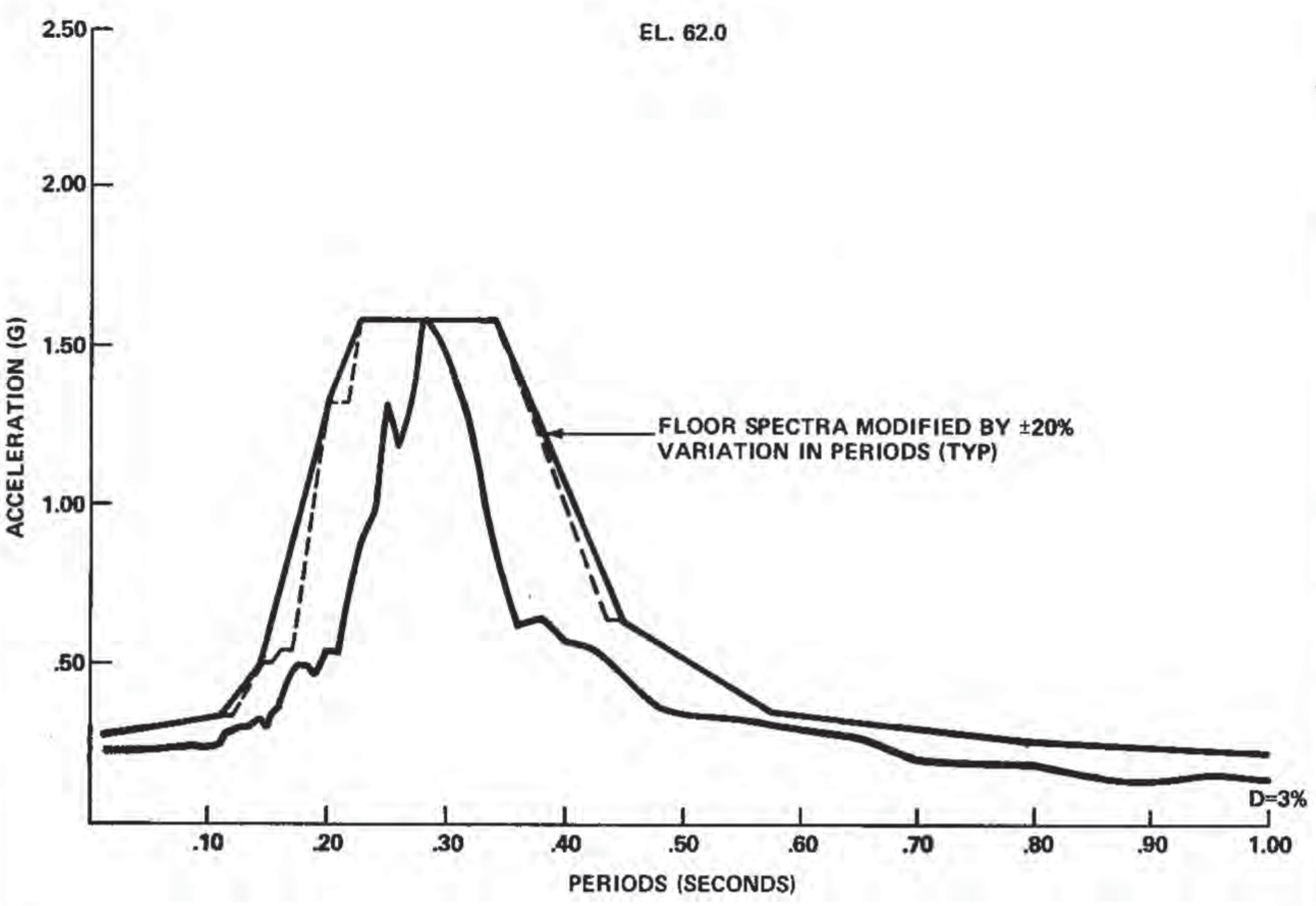
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
FUEL HANDLING BLDG.
FIGURE 3.7–125



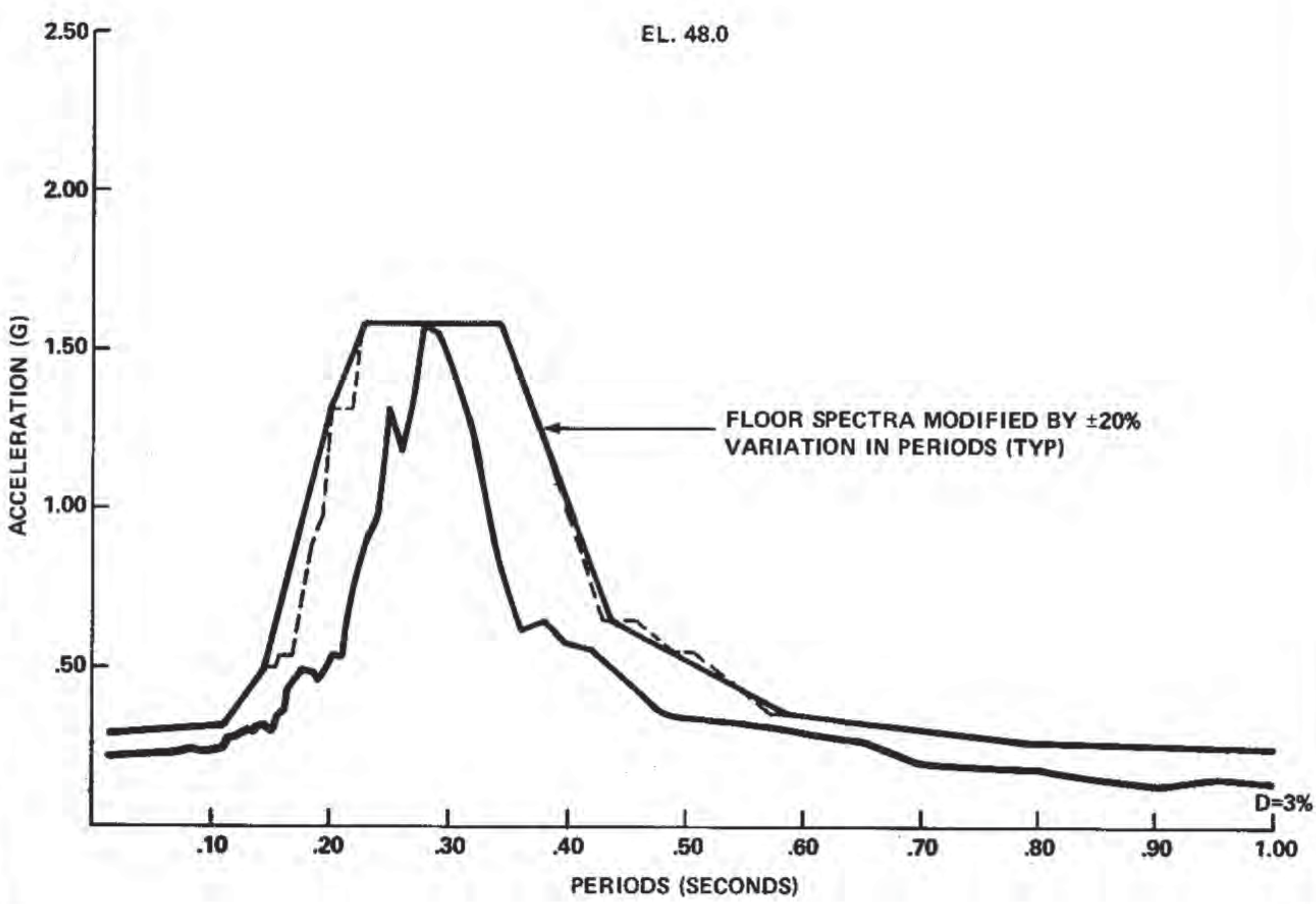
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
FUEL HANDLING BLDG.
FIGURE 3.7–126



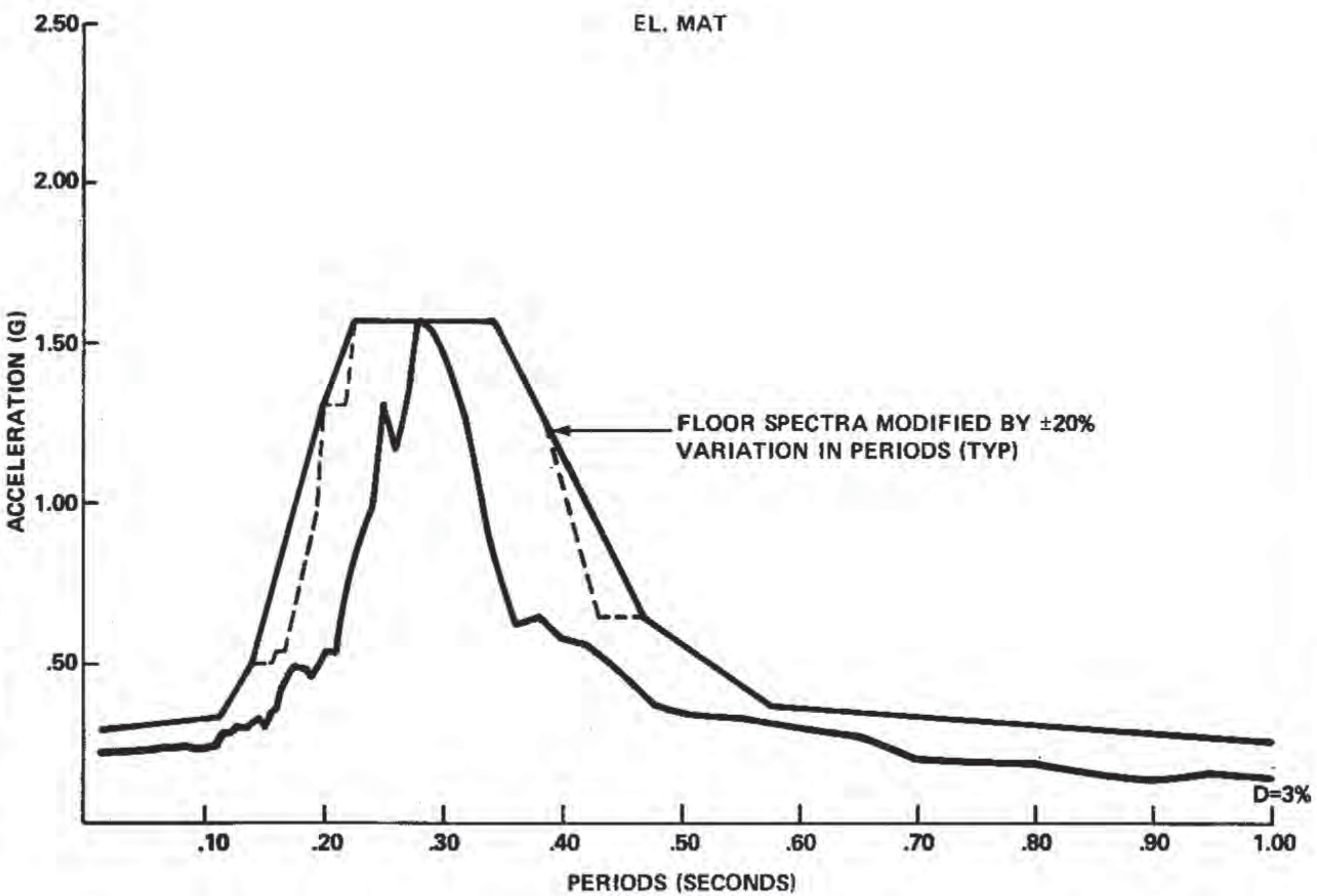
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
FUEL HANDLING BLDG.
FIGURE 3.7–127



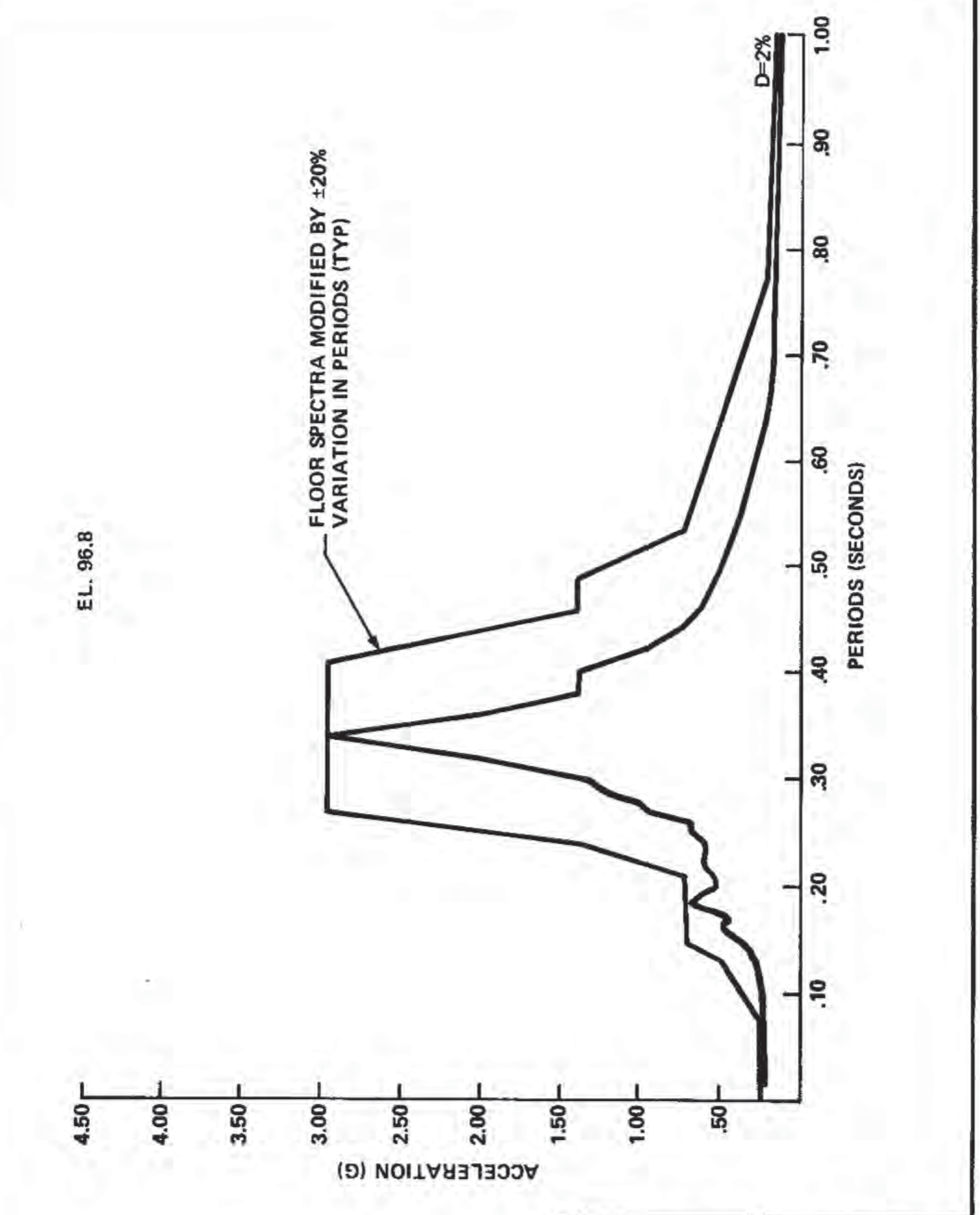
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
FUEL HANDLING BLDG.
FIGURE 3.7–128



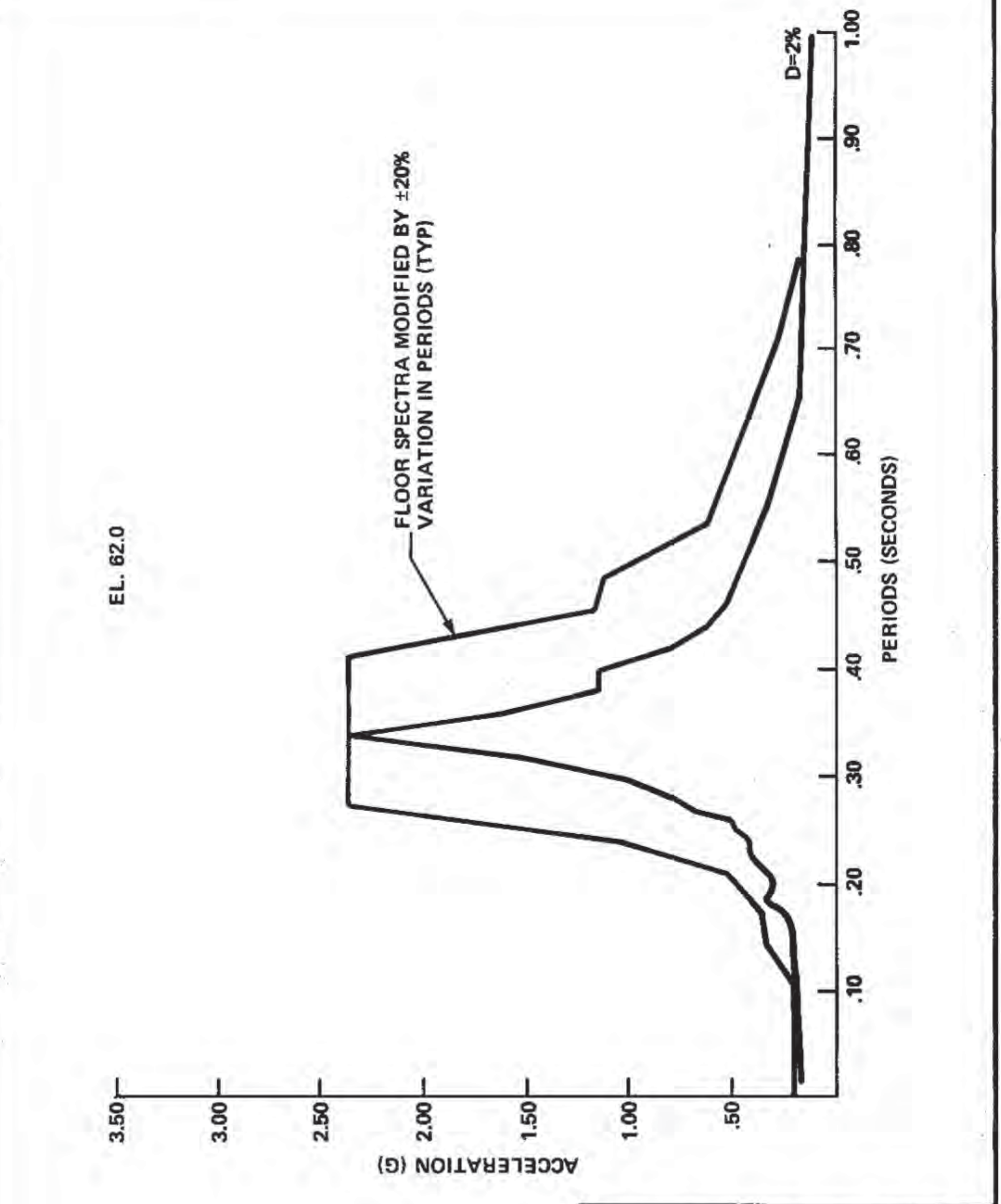
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
FUEL HANDLING BLDG.
FIGURE 3.7-129



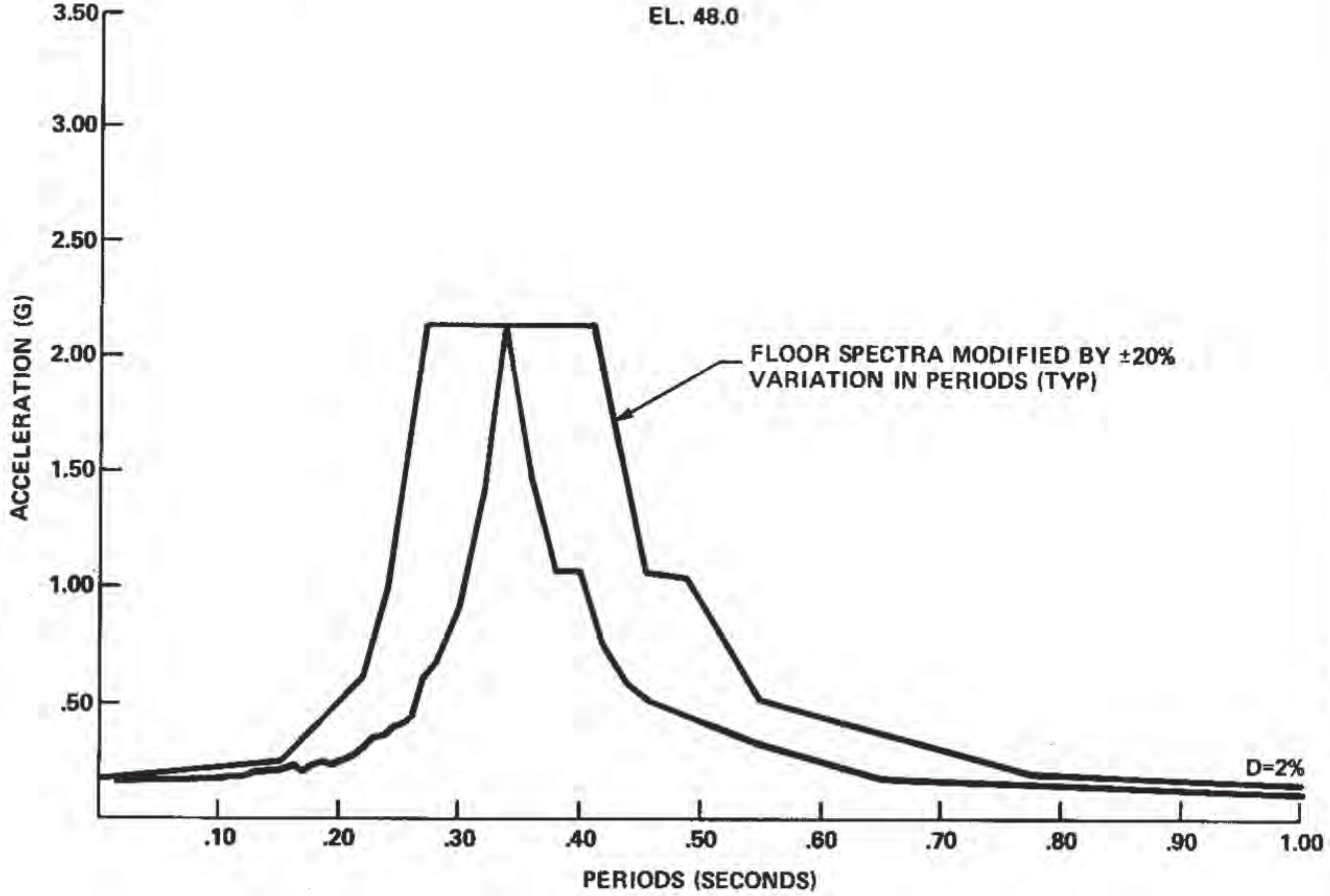
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
FUEL HANDLING BLDG.
FIGURE 3.7-130



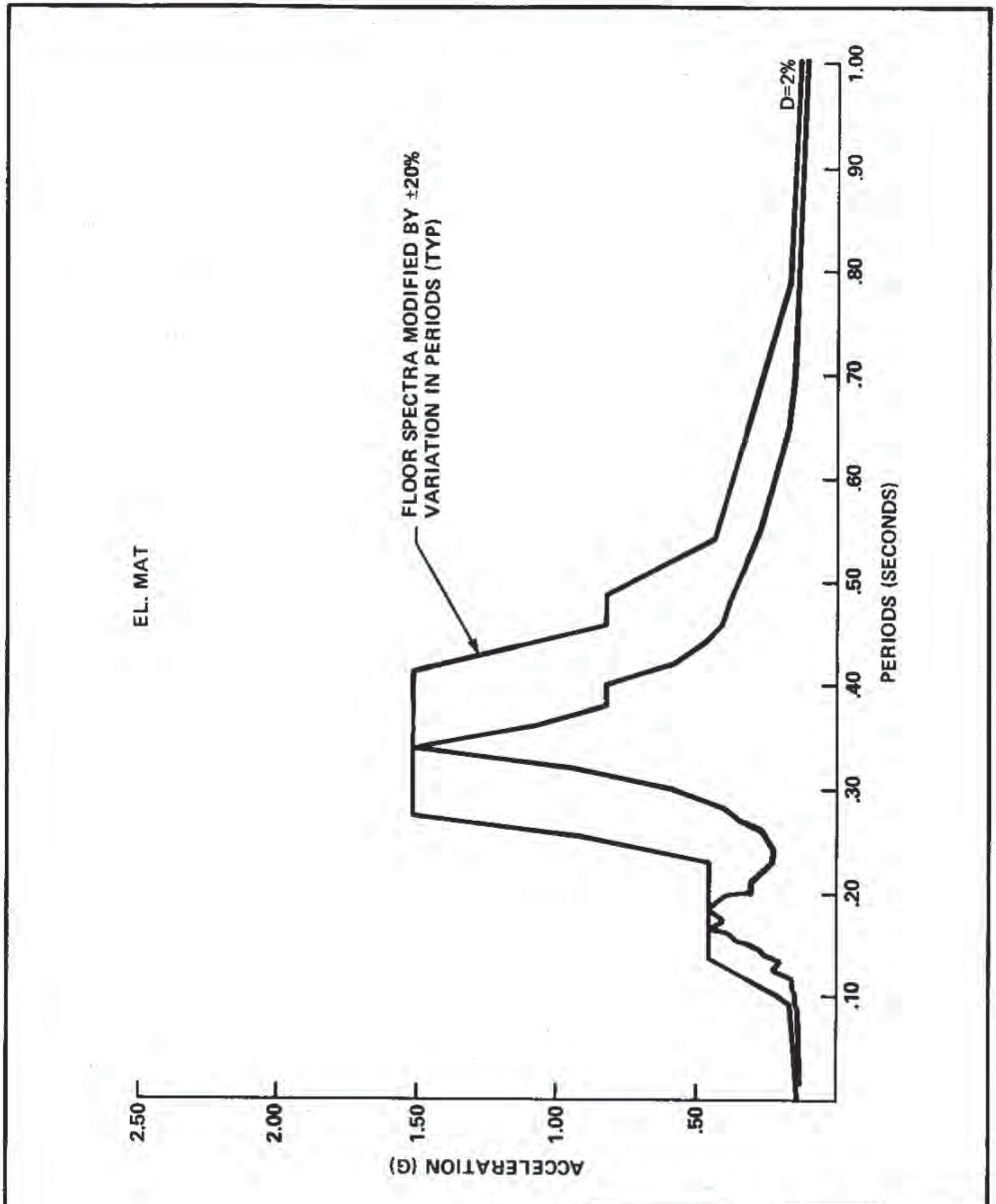
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
FUEL HANDLING BLDG.
FIGURE 3.7–131



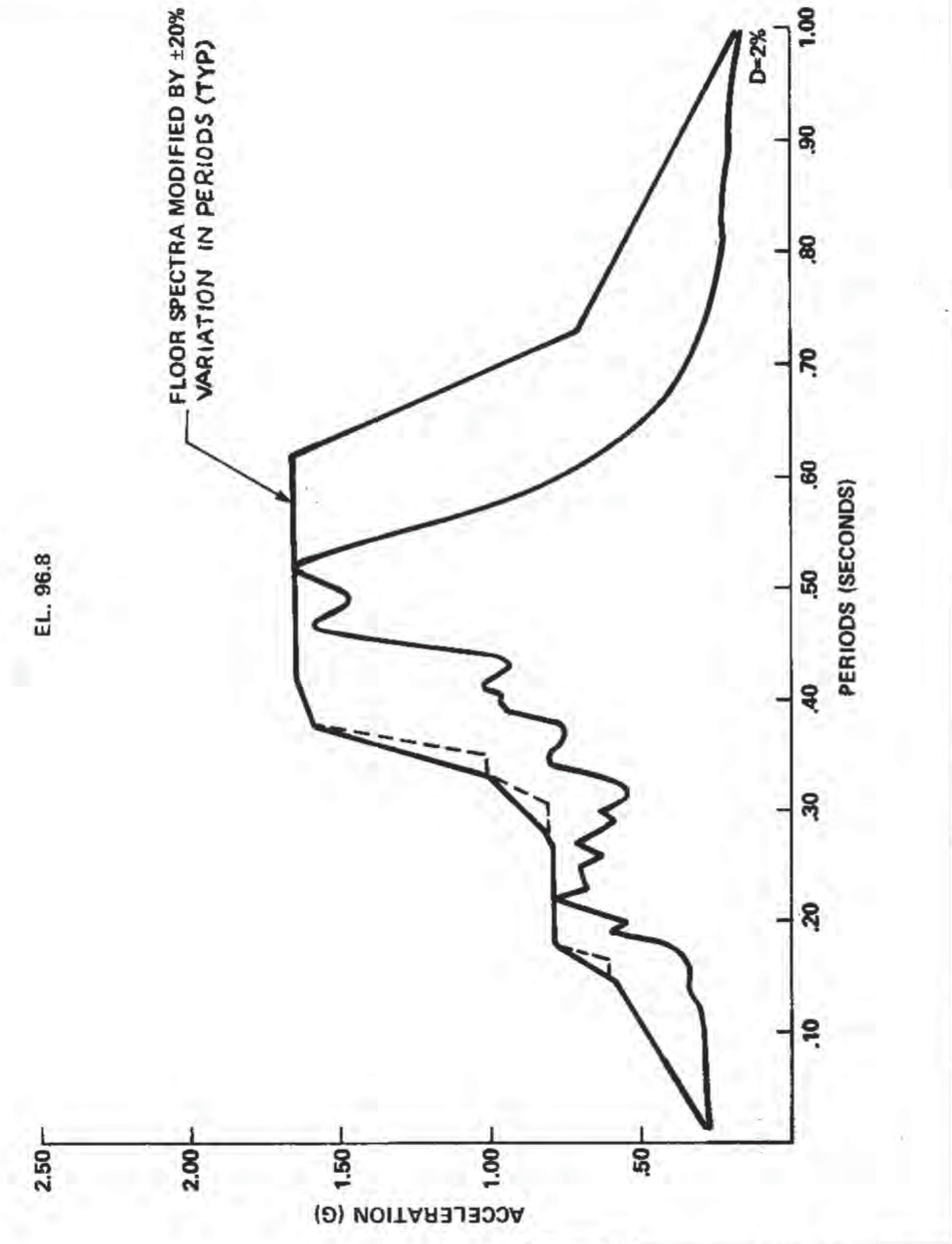
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
FUEL HANDLING BLDG.
FIGURE 3.7–132



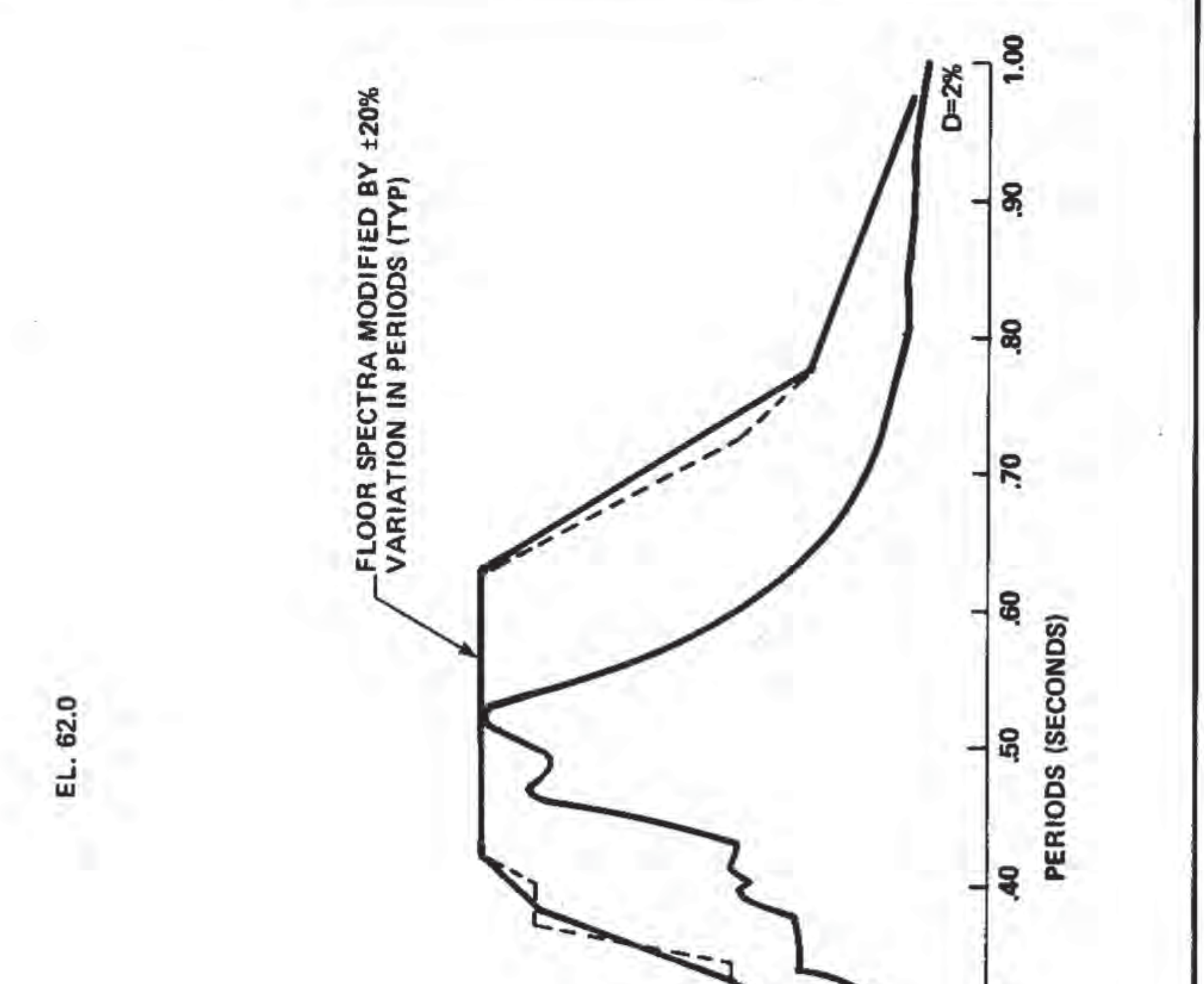
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
FUEL HANDLING BLDG.
FIGURE 3.7–133



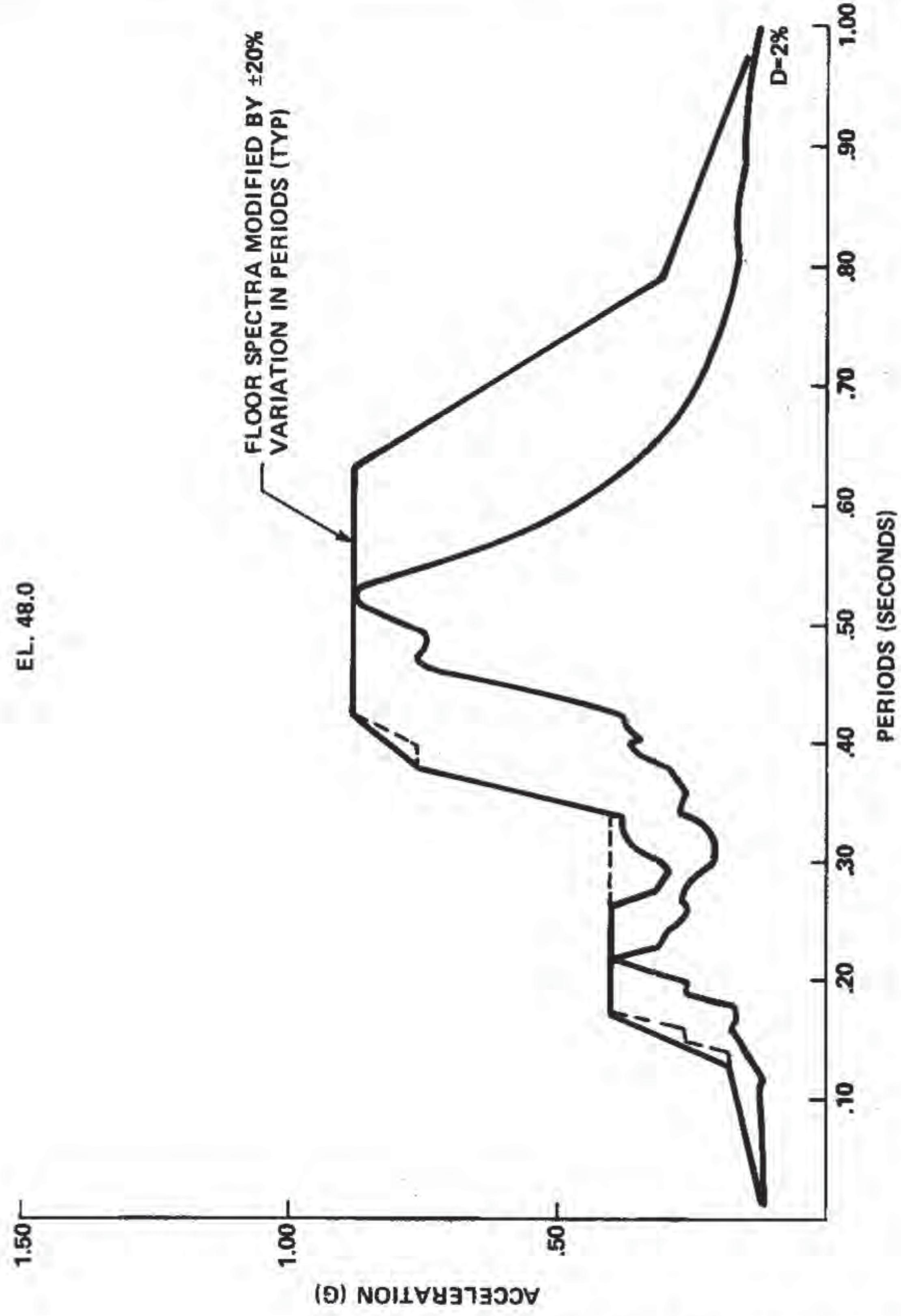
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
FUEL HANDLING BLDG.
FIGURE 3.7–134



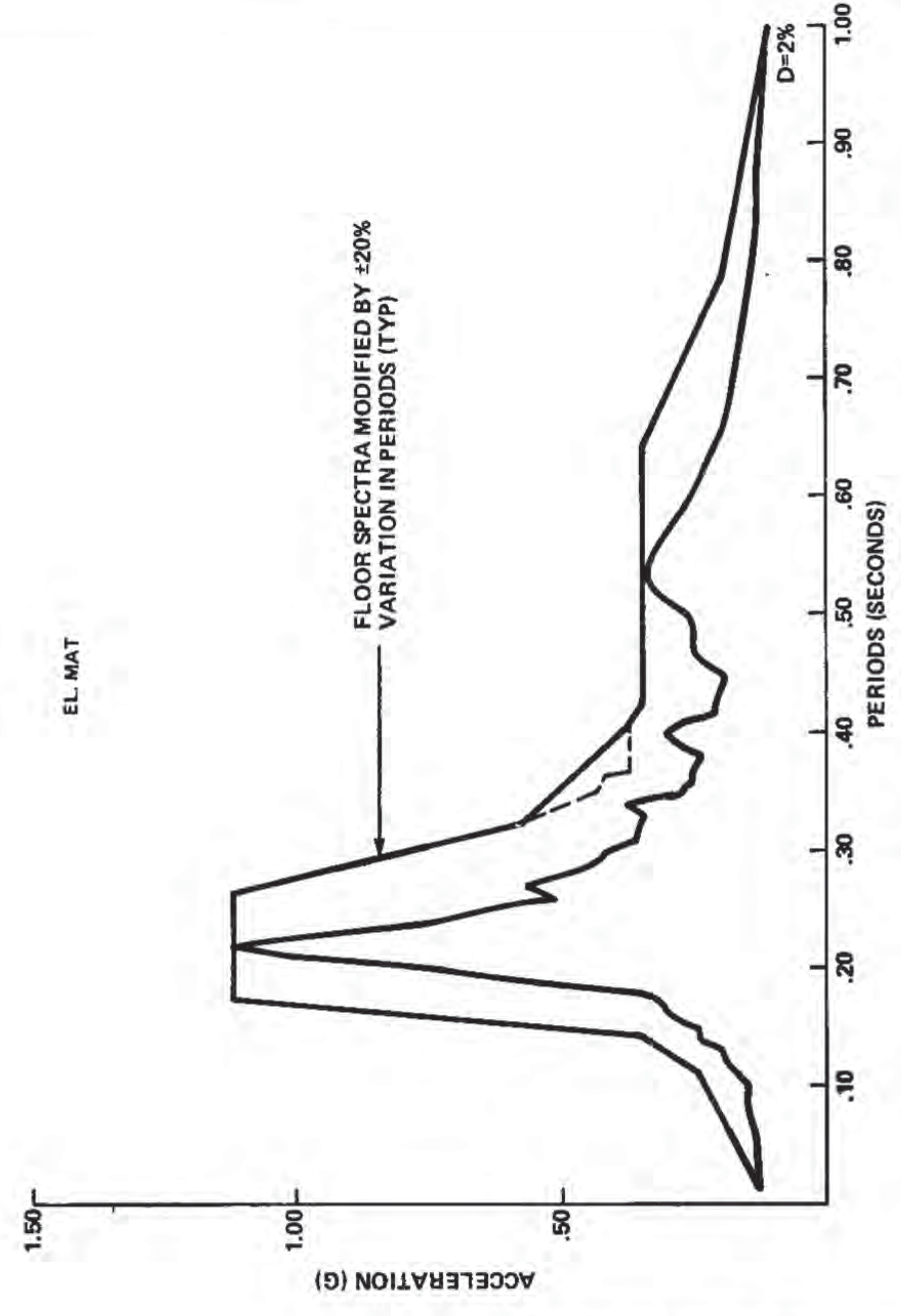
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
FUEL HANDLING BLDG.
FIGURE 3.7–135



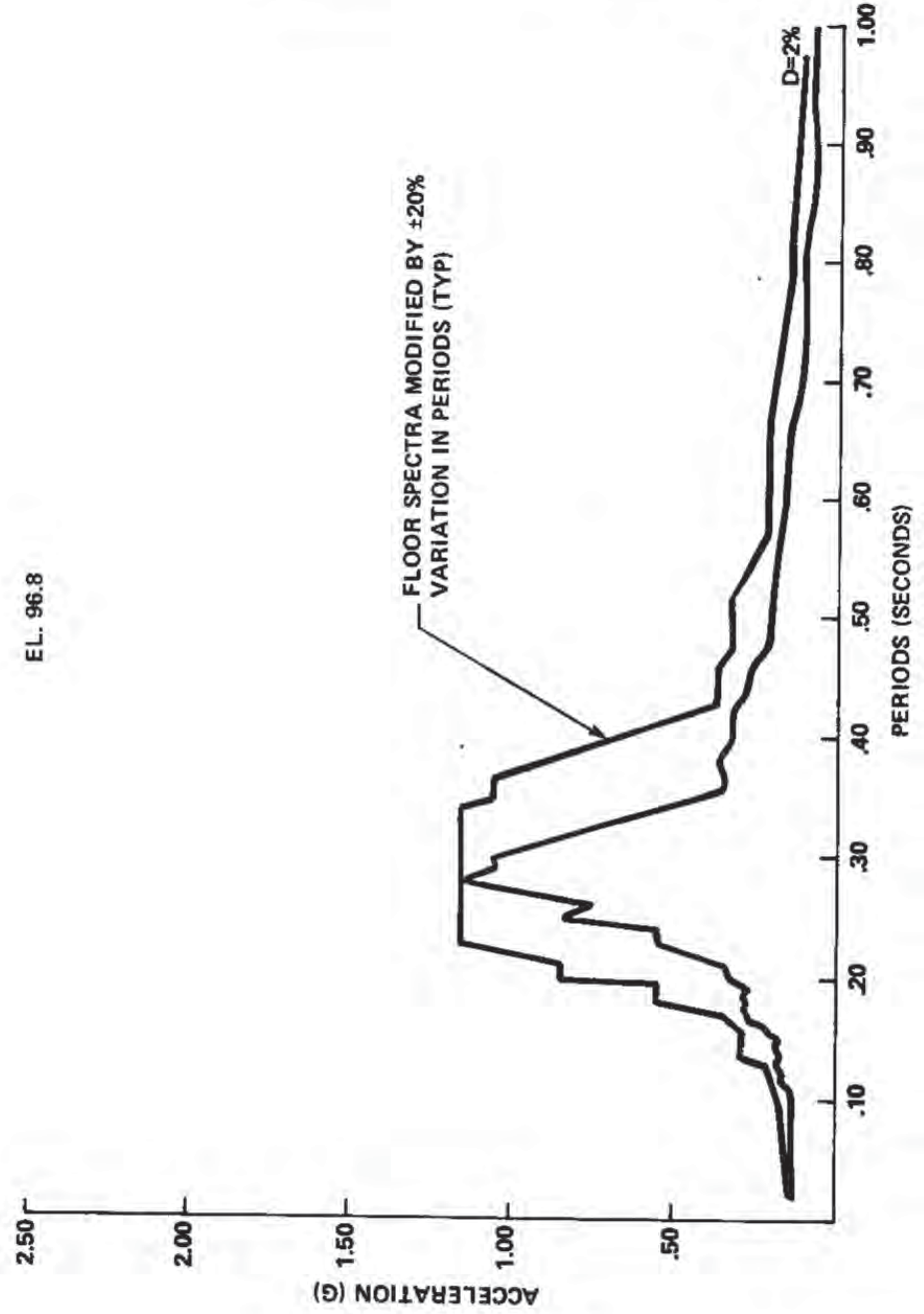
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
FUEL HANDLING BLDG.
FIGURE 3.7-136



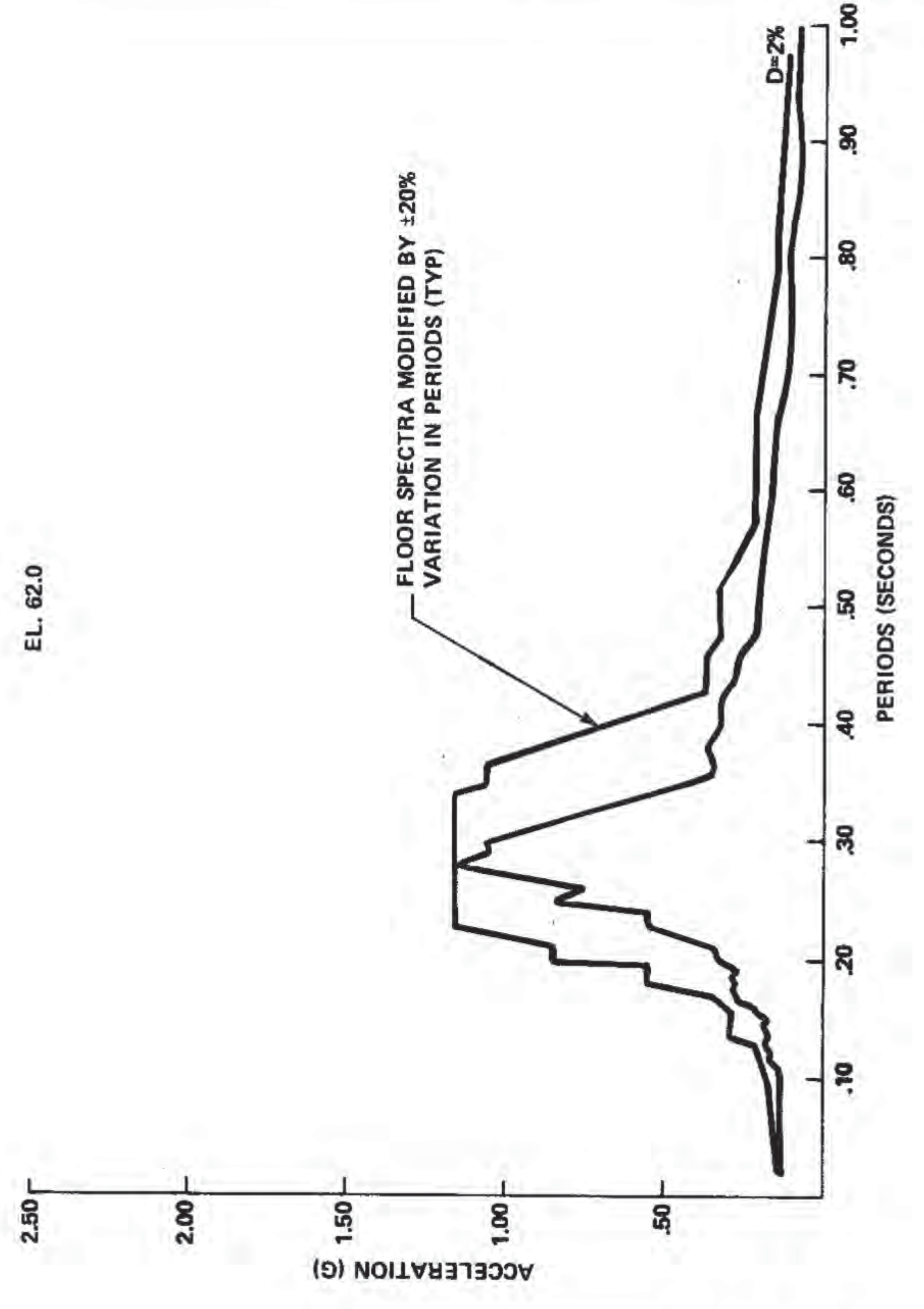
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
FUEL HANDLING BLDG.
FIGURE 3.7–137



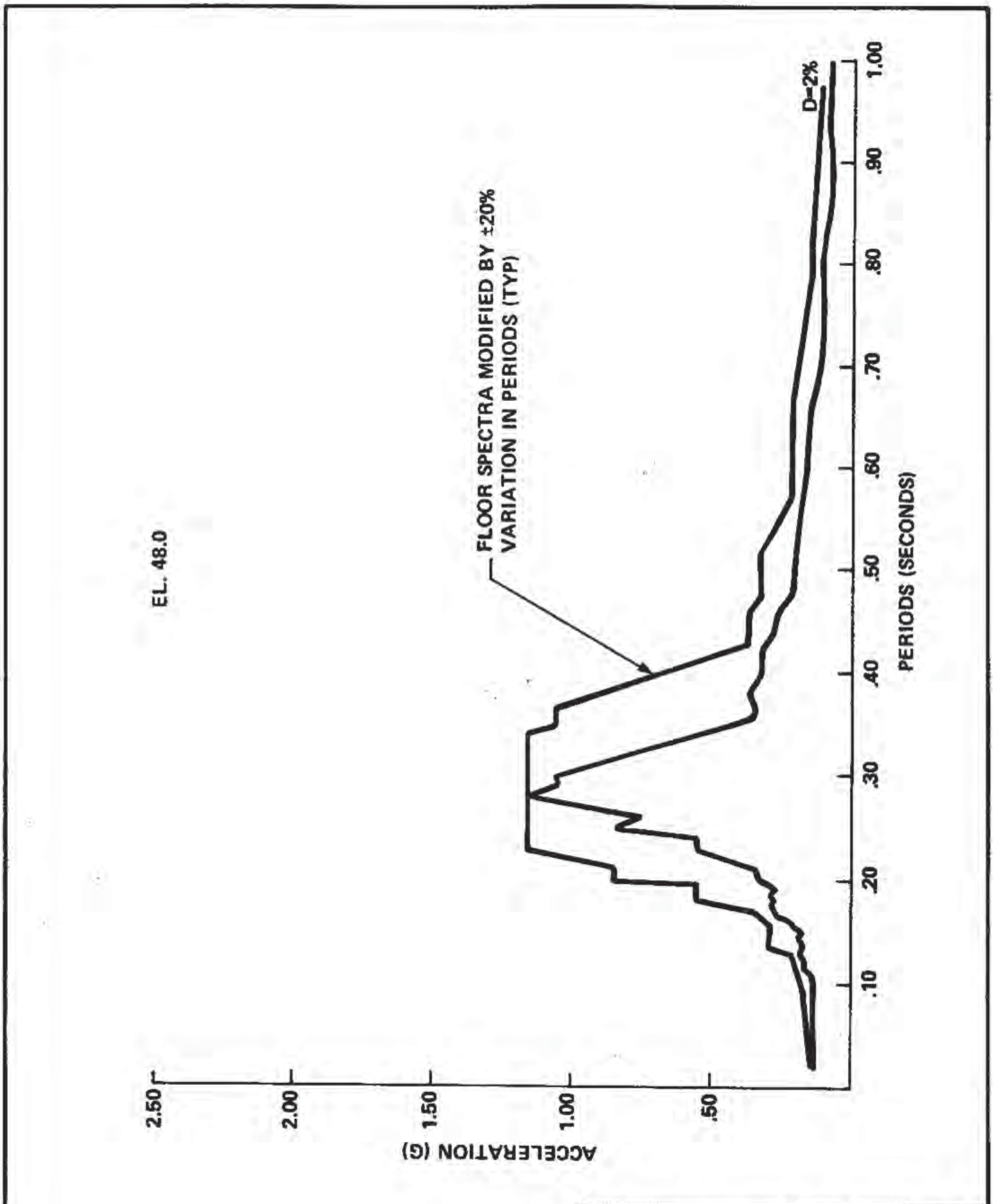
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
FUEL HANDLING BLDG.
FIGURE 3.7-138



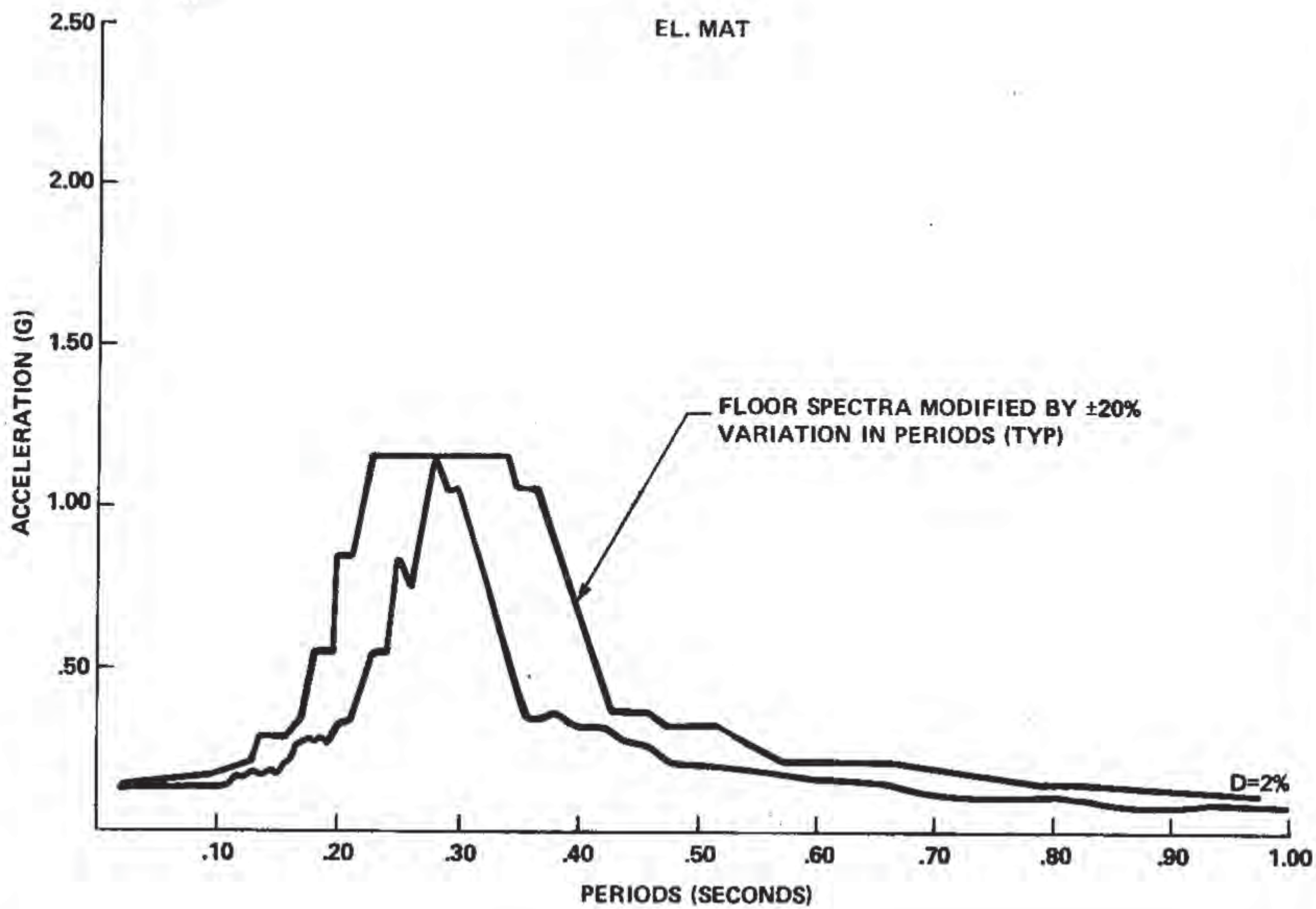
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
FUEL HANDLING BLDG.
FIGURE 3.7-139



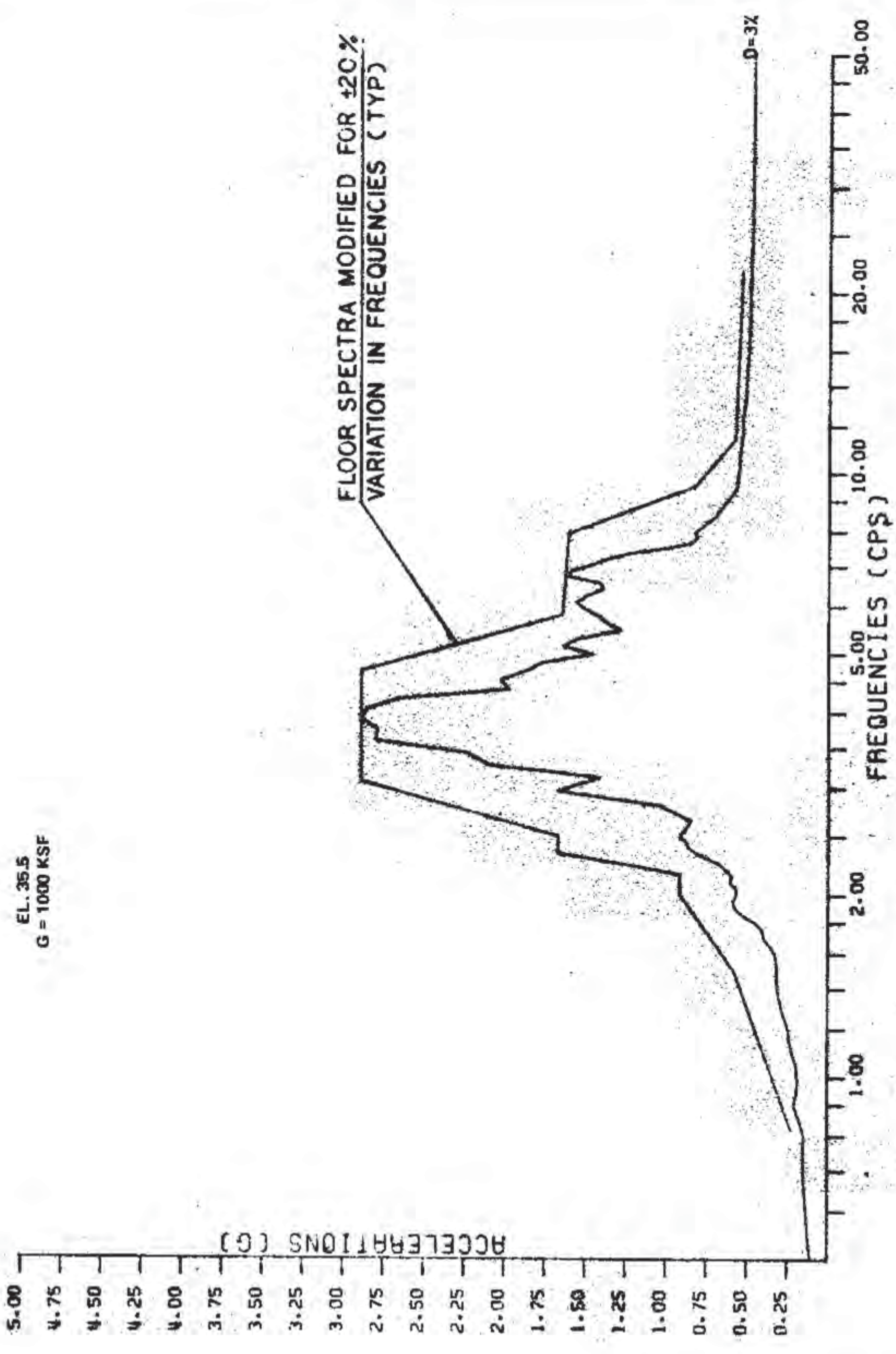
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
FUEL HANDLING BLDG.
FIGURE 3.7–140



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

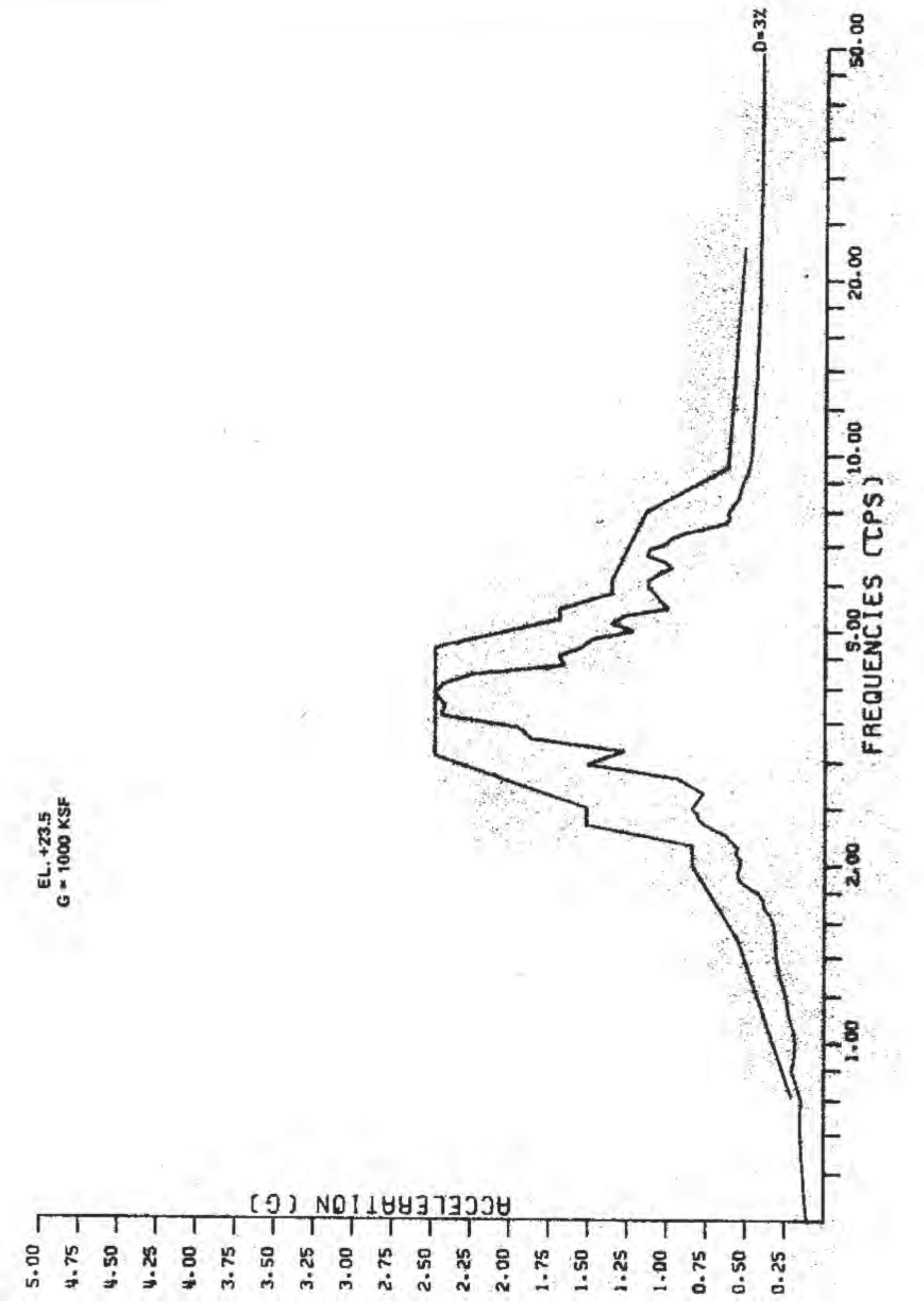
FLOOR SPECTRA VERTICAL OBE
FUEL HANDLING BLDG.
FIGURE 3.7-141



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

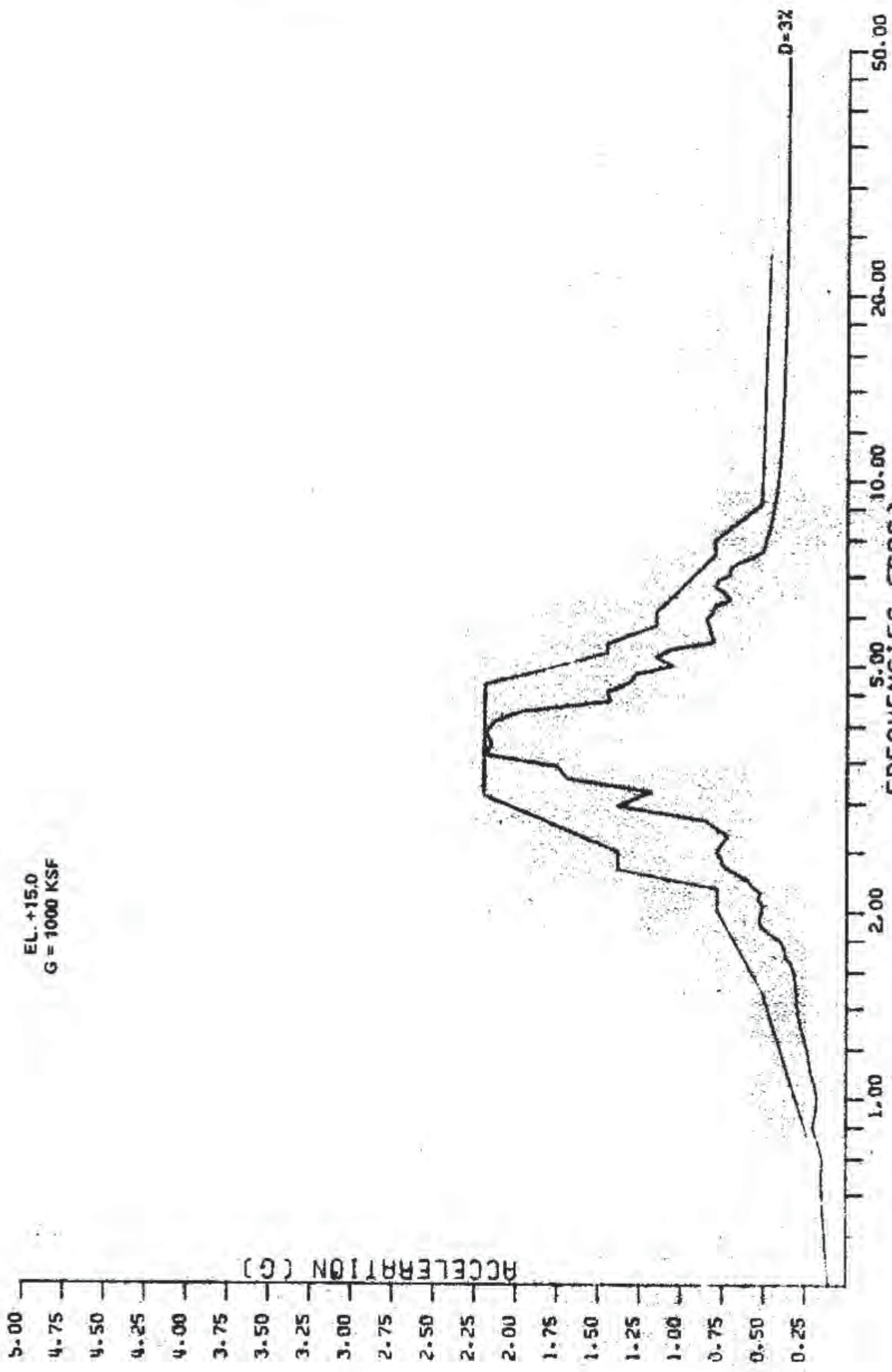
FLOOR SPECTRA HORIZONTAL N-S SSE
INTAKE STRUCTURE MASS PT. 1

FIGURE 3.7–142



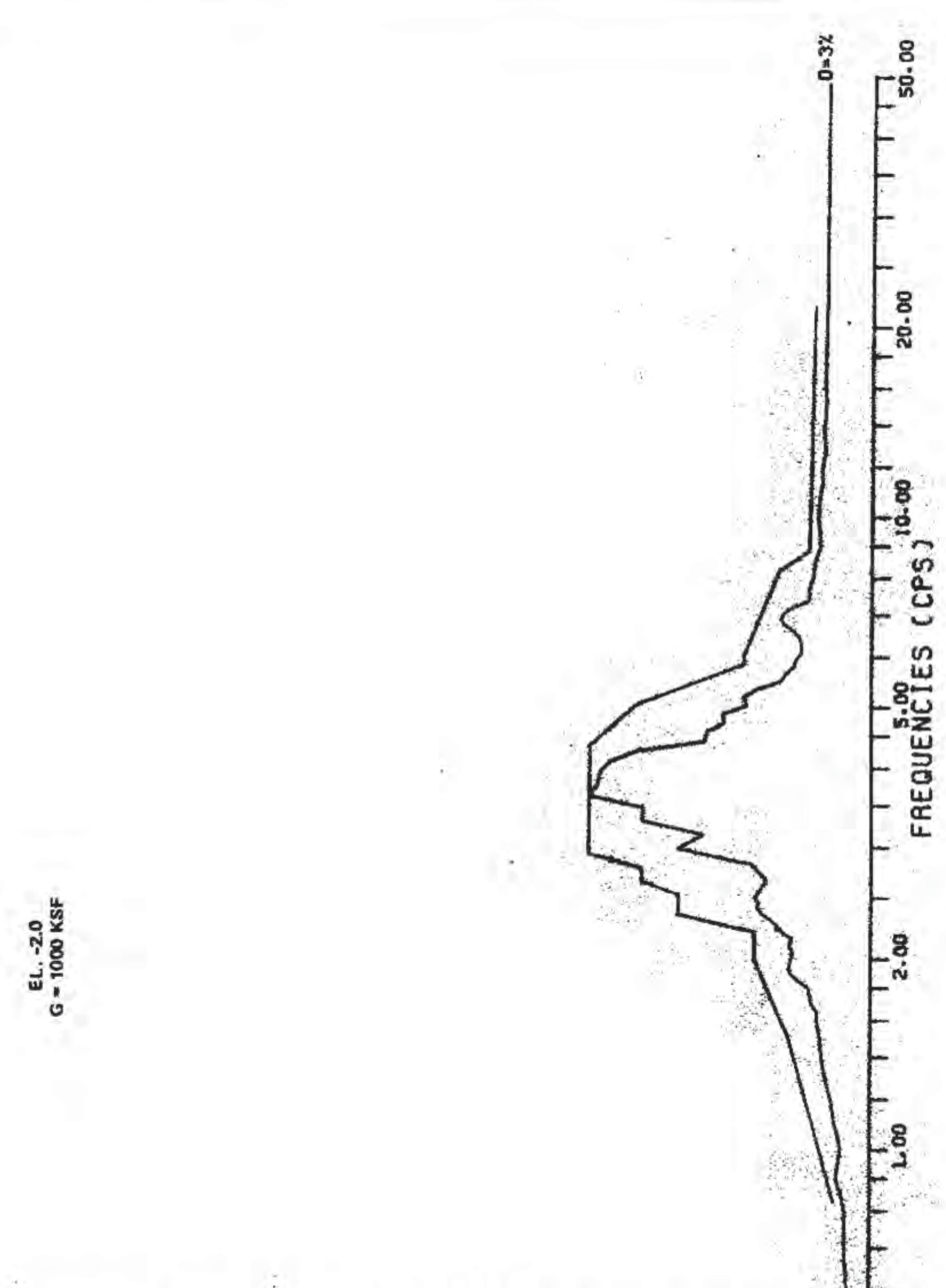
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
INTAKE STRUCTURE MASS PT. 2
FIGURE 3.7–143



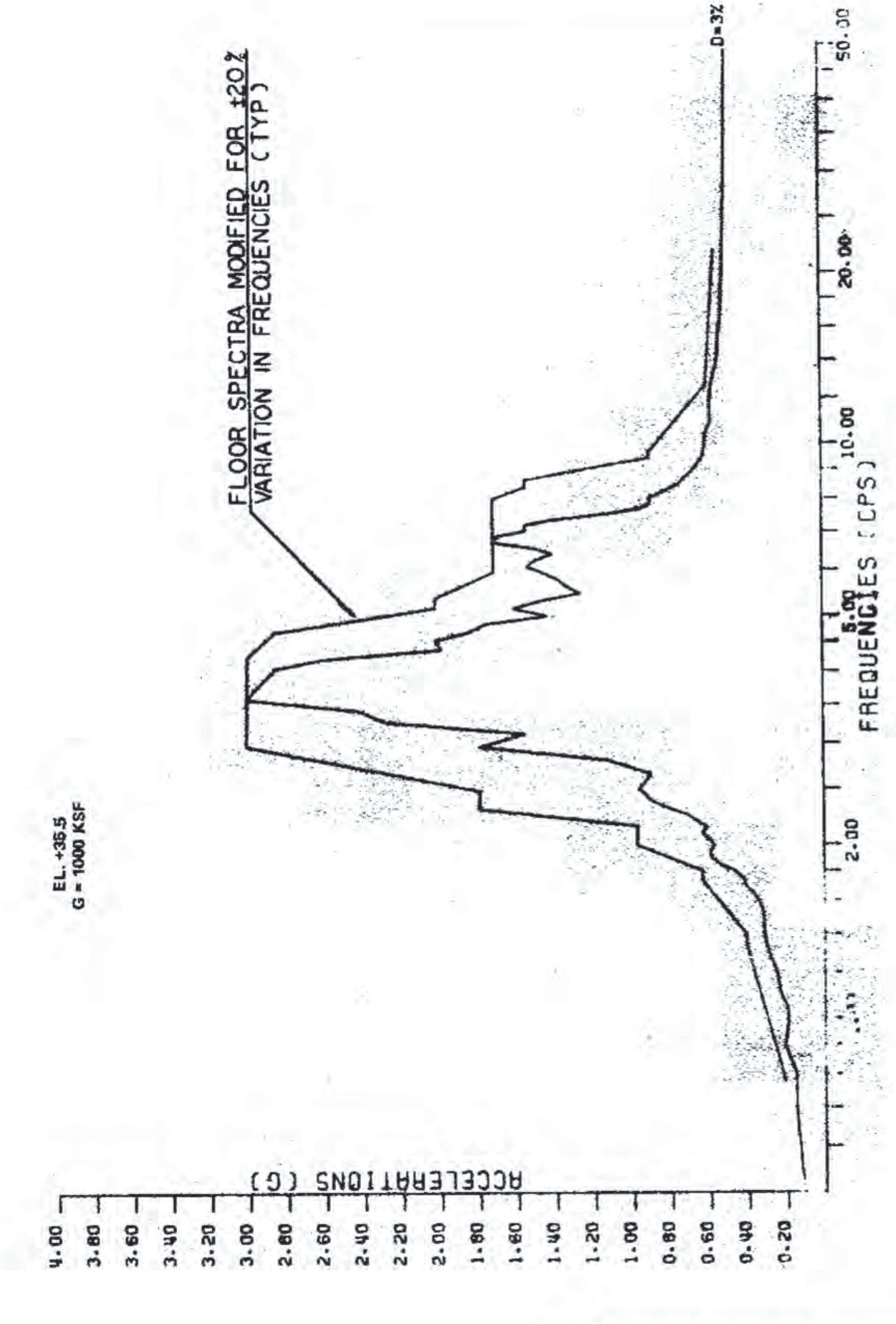
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
INTAKE STRUCTURE MASS PT. 3
FIGURE 3.7–144



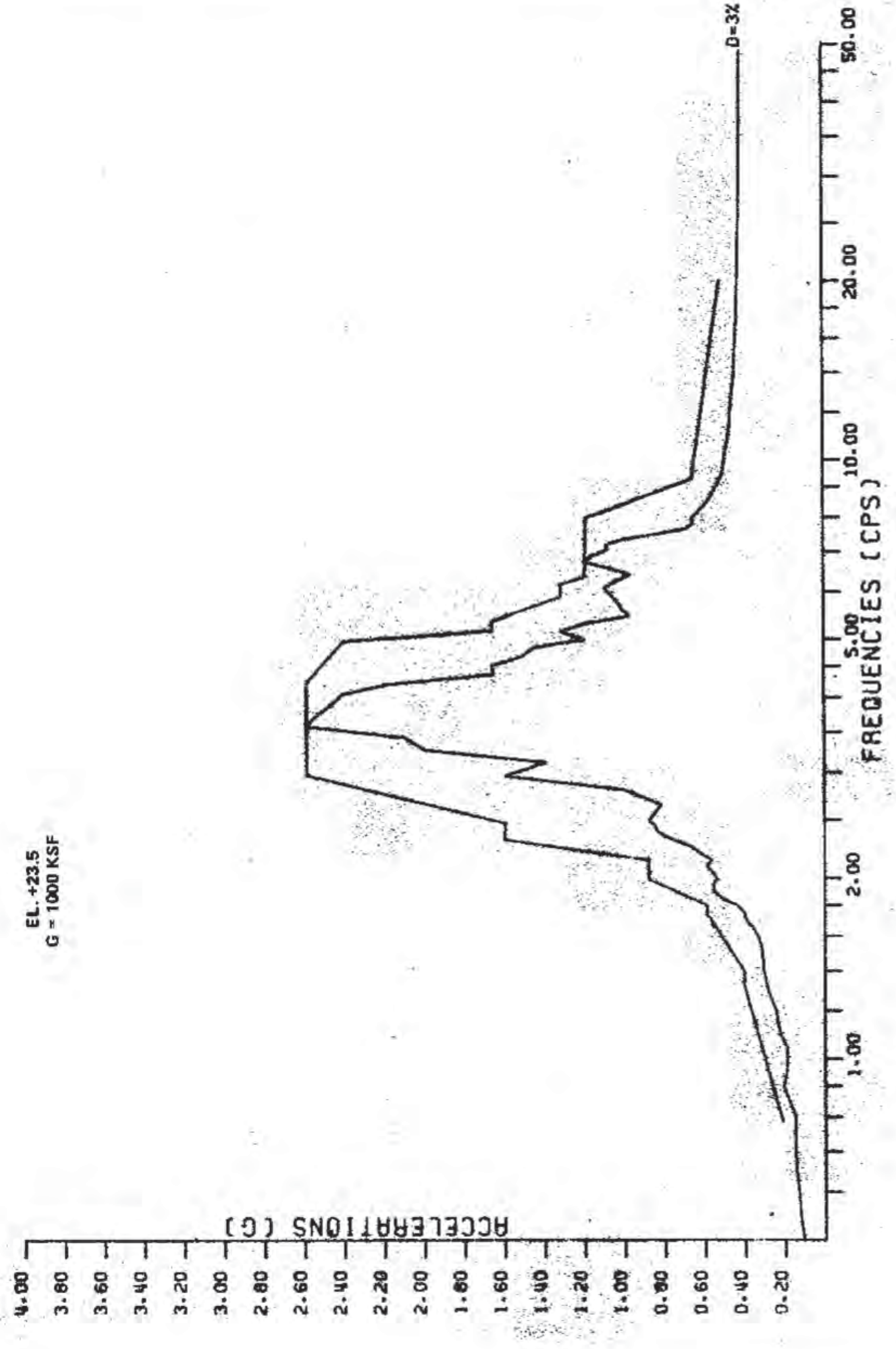
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
INTAKE STRUCTURE MASS PT. 4
FIGURE 3.7-145



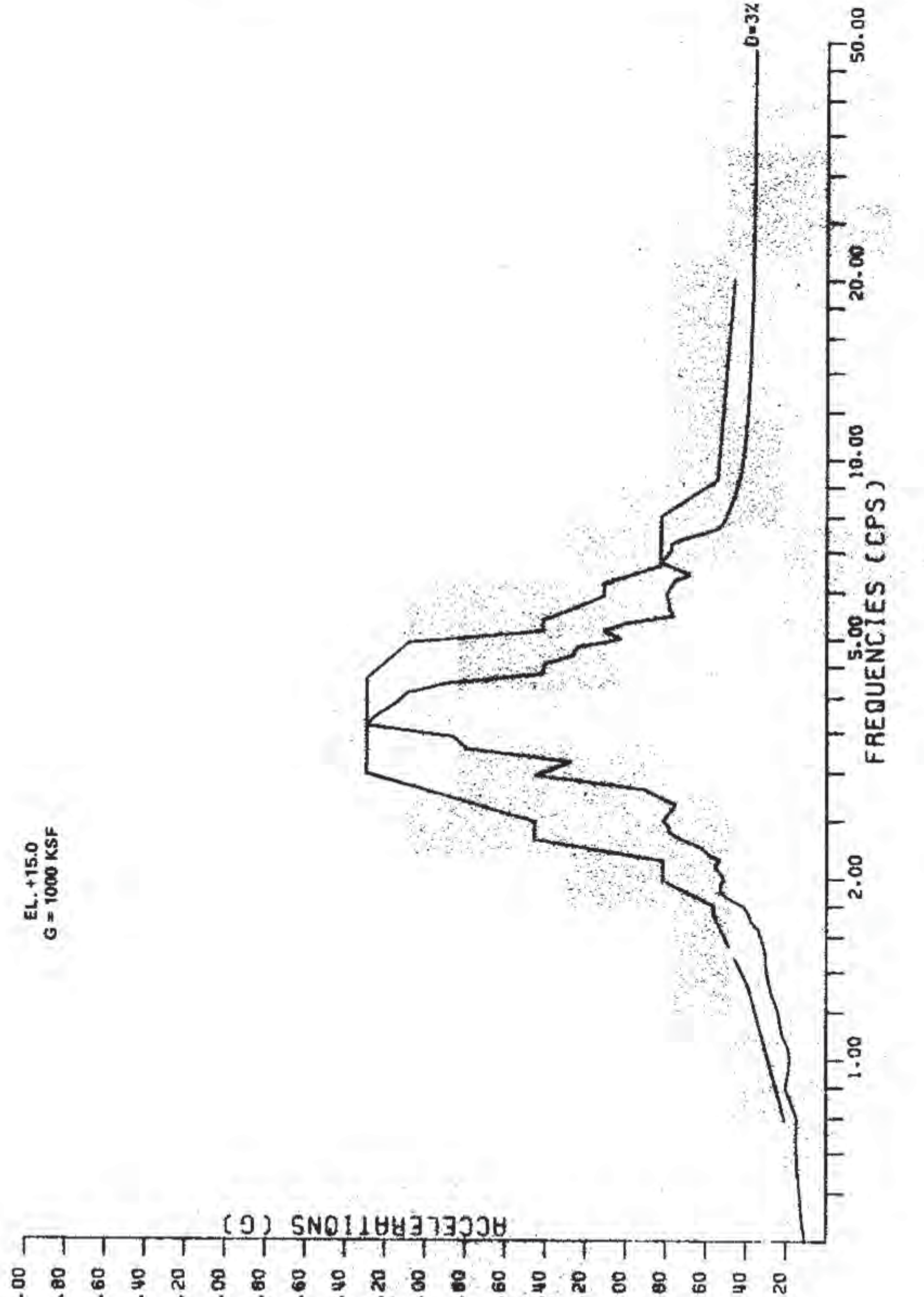
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
INTAKE STRUCTURE MASS PT. 1
FIGURE 3.7-146



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

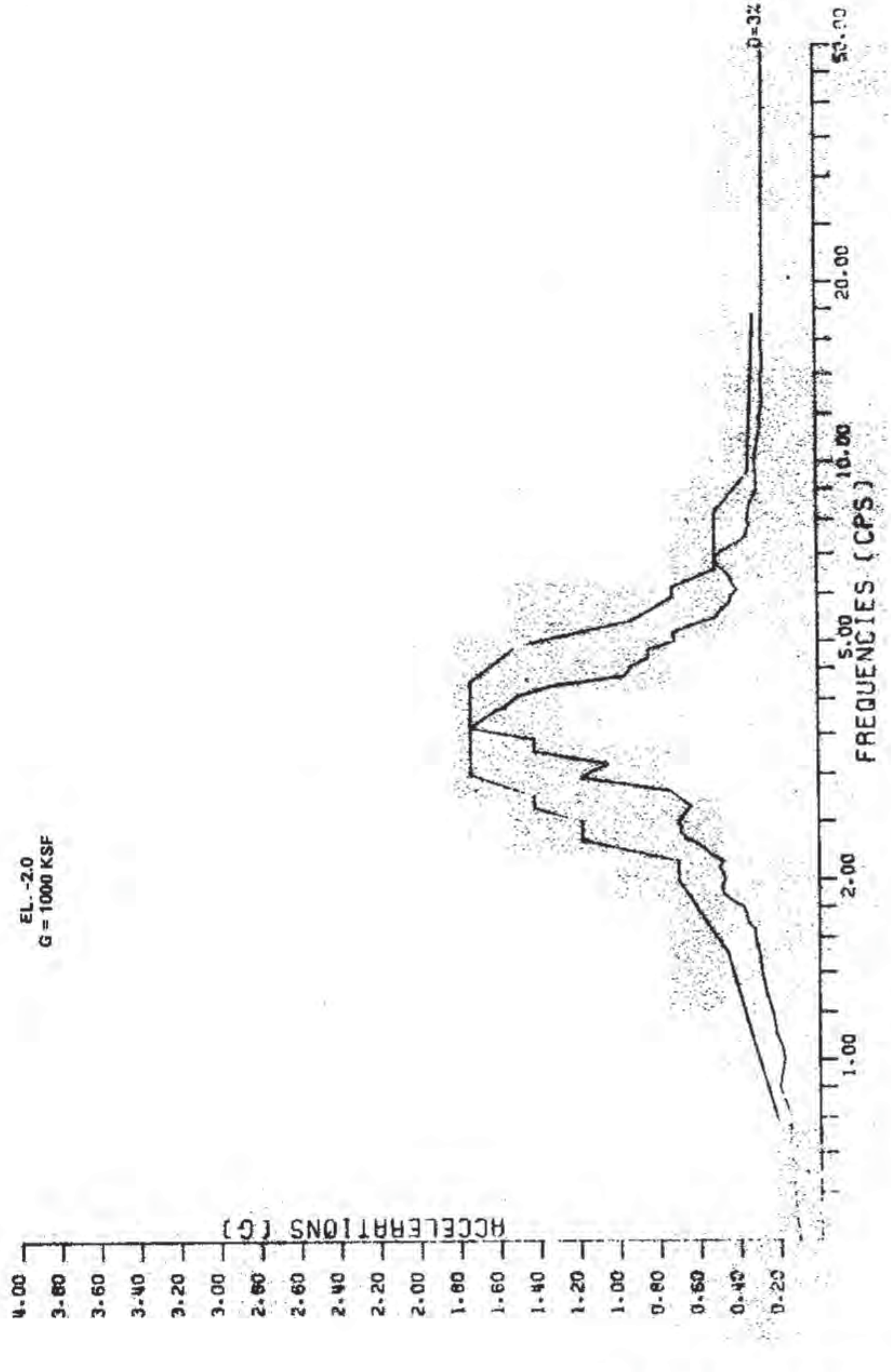
FLOOR SPECTRA HORIZONTAL E-W SSE
INTAKE STRUCTURE MASS PT. 2
FIGURE 3.7-147



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

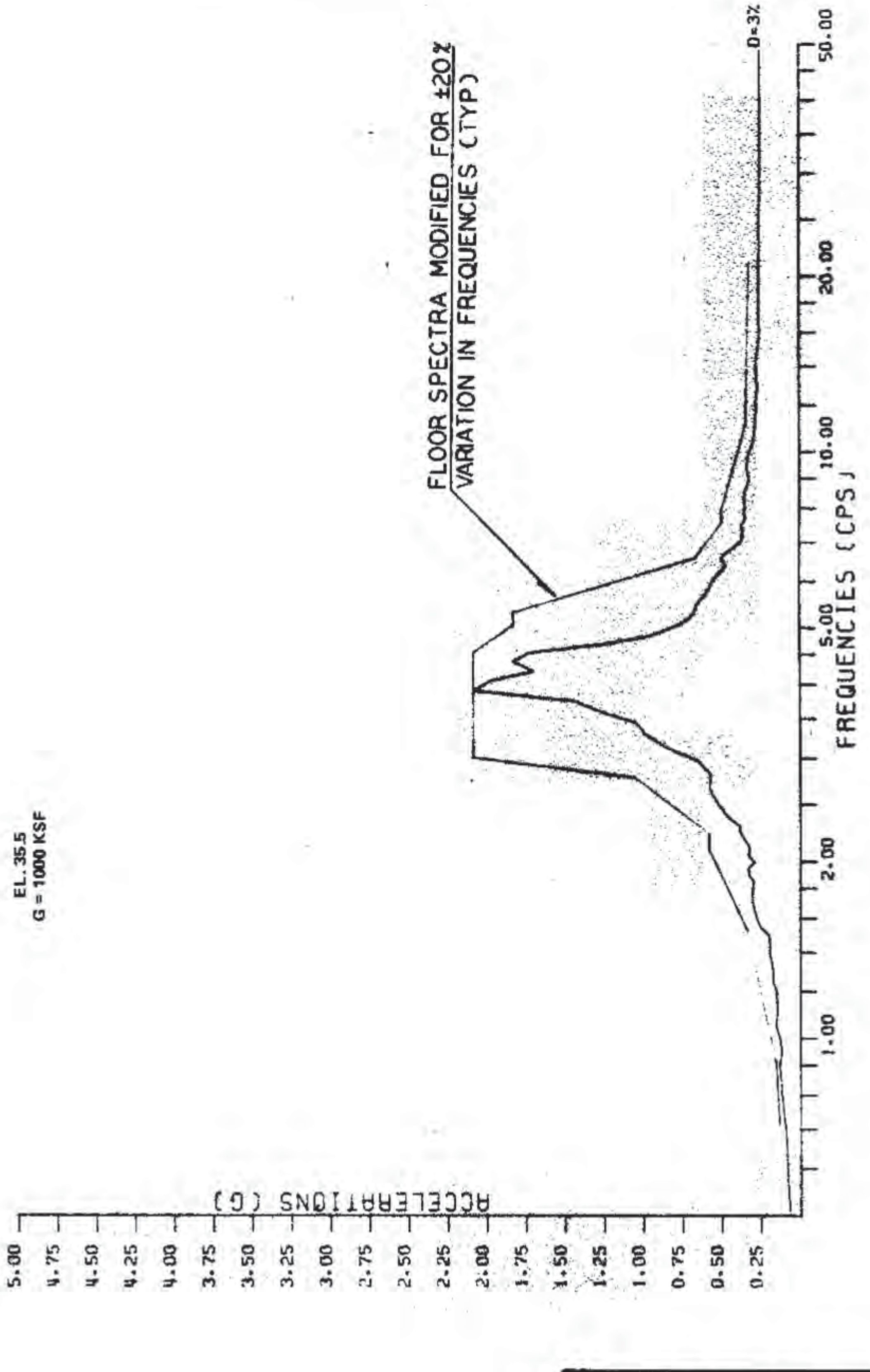
FLOOR SPECTRA HORIZONTAL E-W SSE
INTAKE STRUCTURE MASS PT. 3

FIGURE 3.7-148



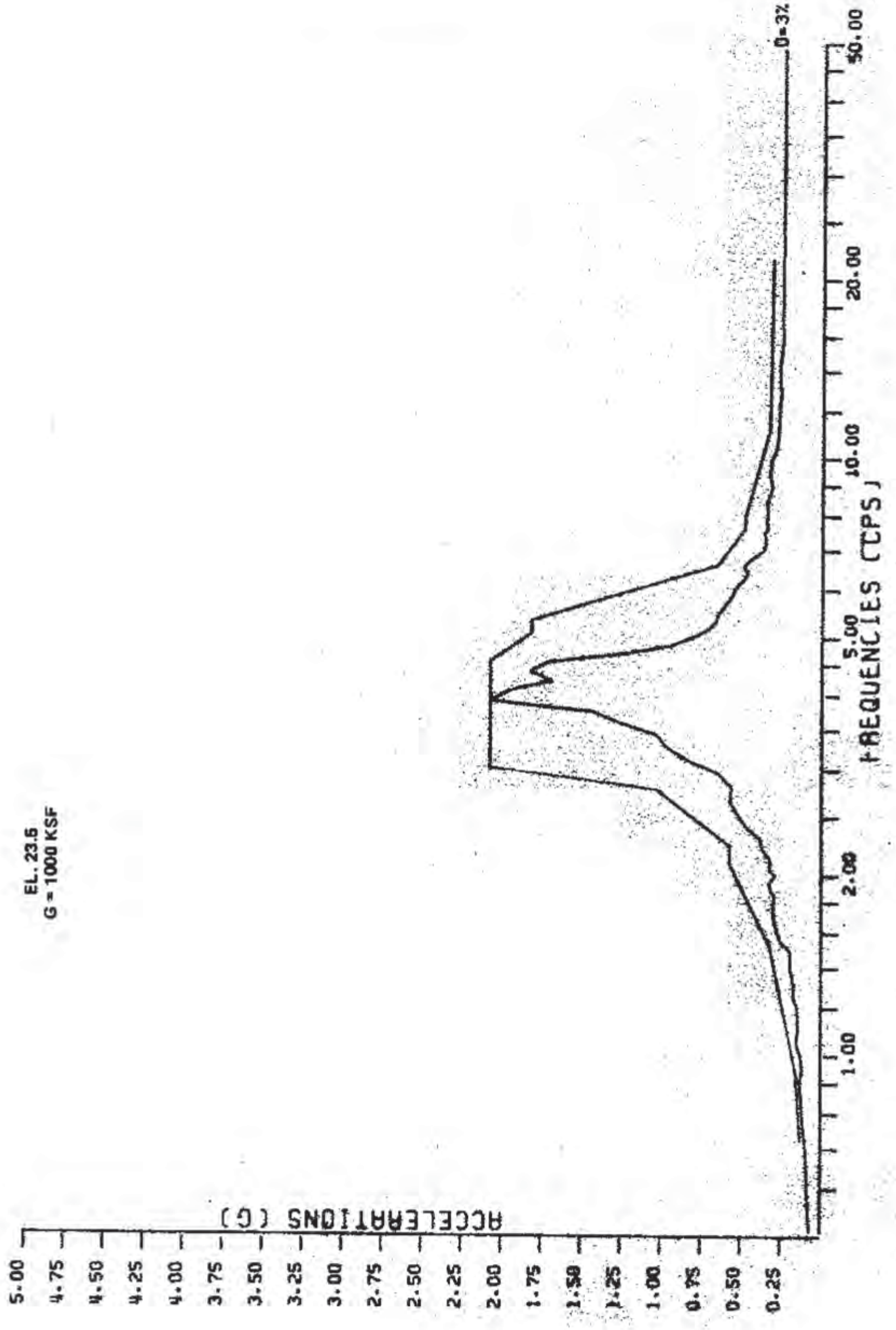
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
INTAKE STRUCTURE MASS PT. 4
FIGURE 3.7–149



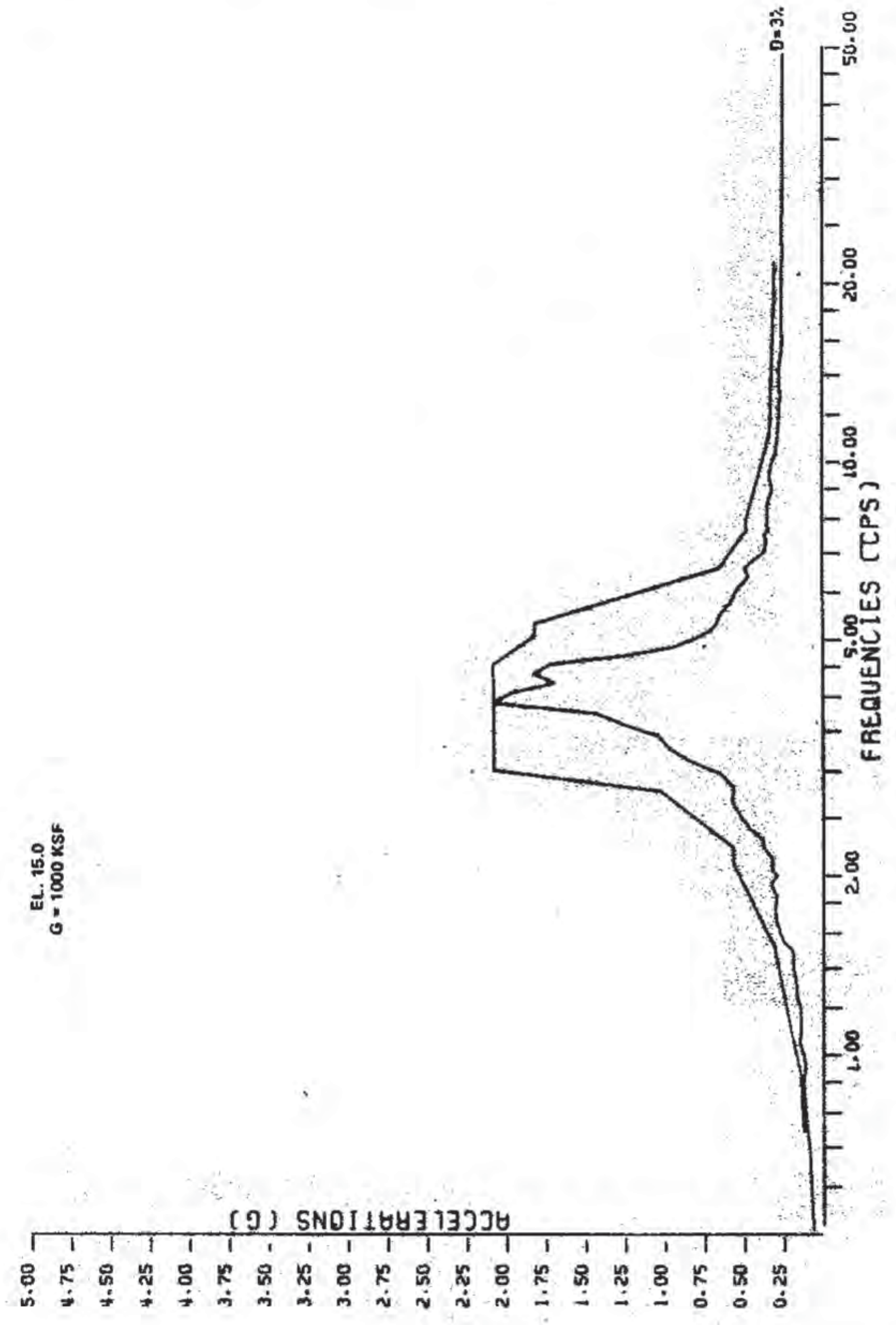
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
INTAKE STRUCTURE MASS PT. 1
FIGURE 3.7-150



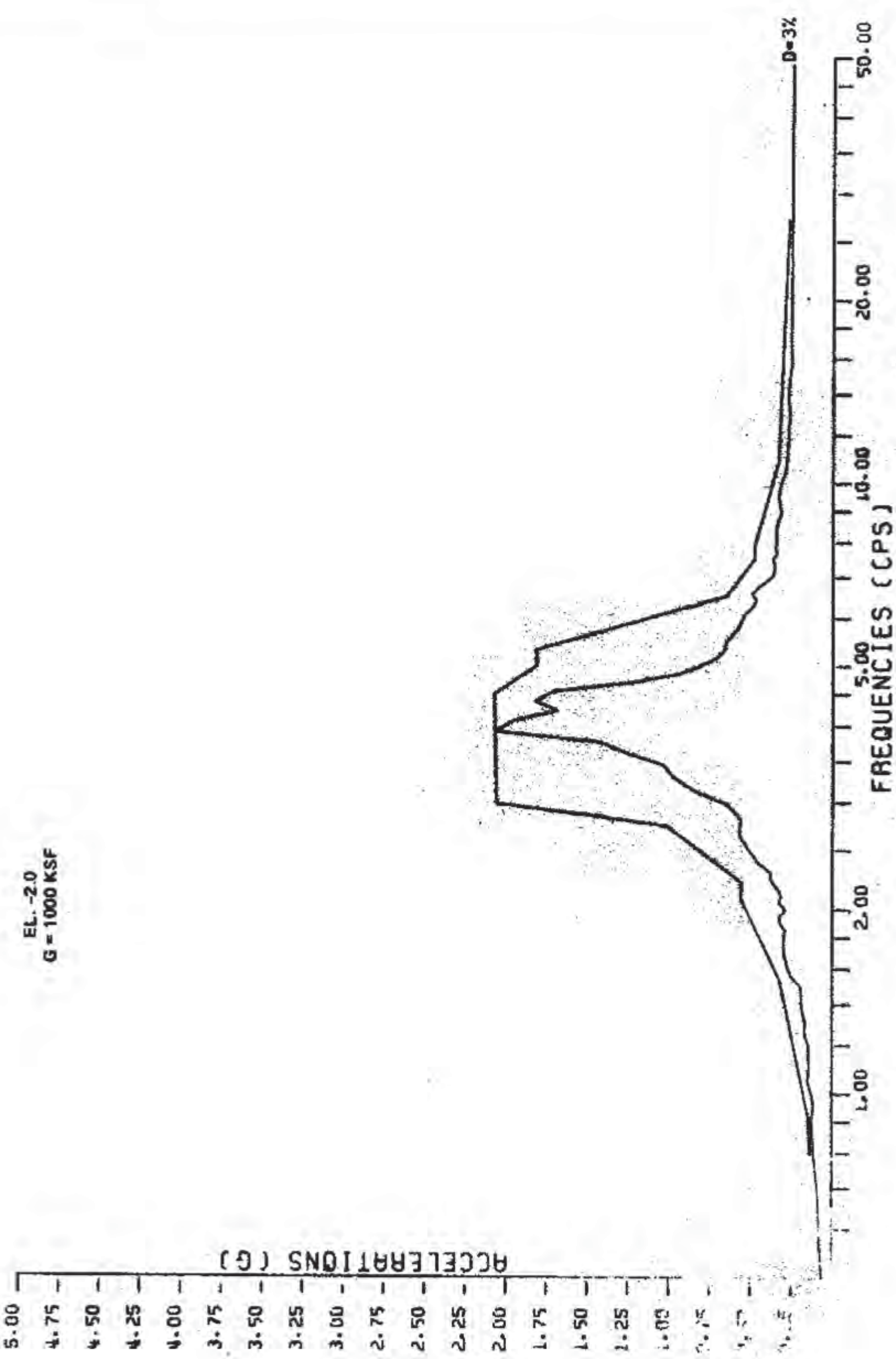
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
INTAKE STRUCTURE MASS PT. 2
FIGURE 3.7–151



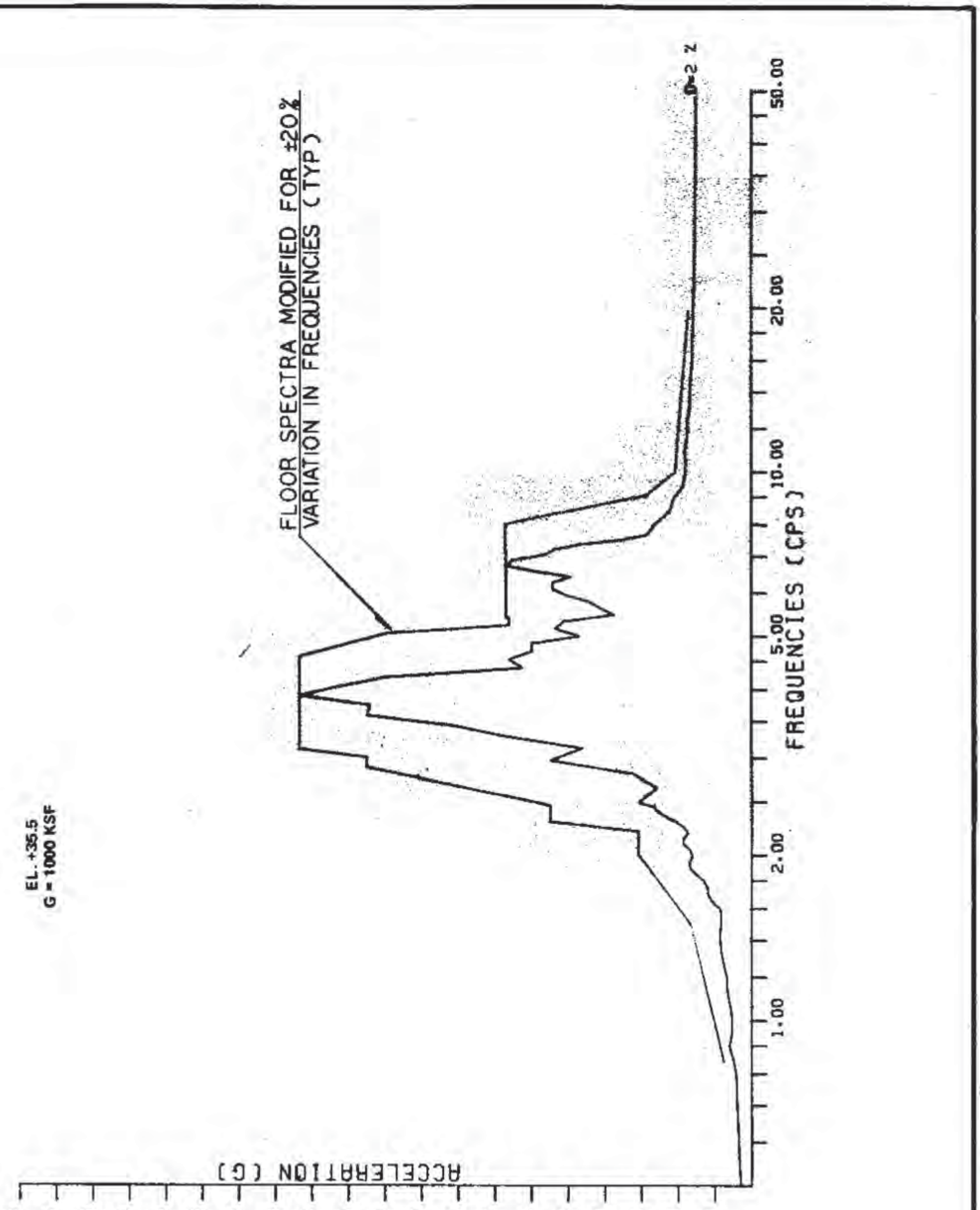
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
INTAKE STRUCTURE MASS PT. 3
FIGURE 3.7–152



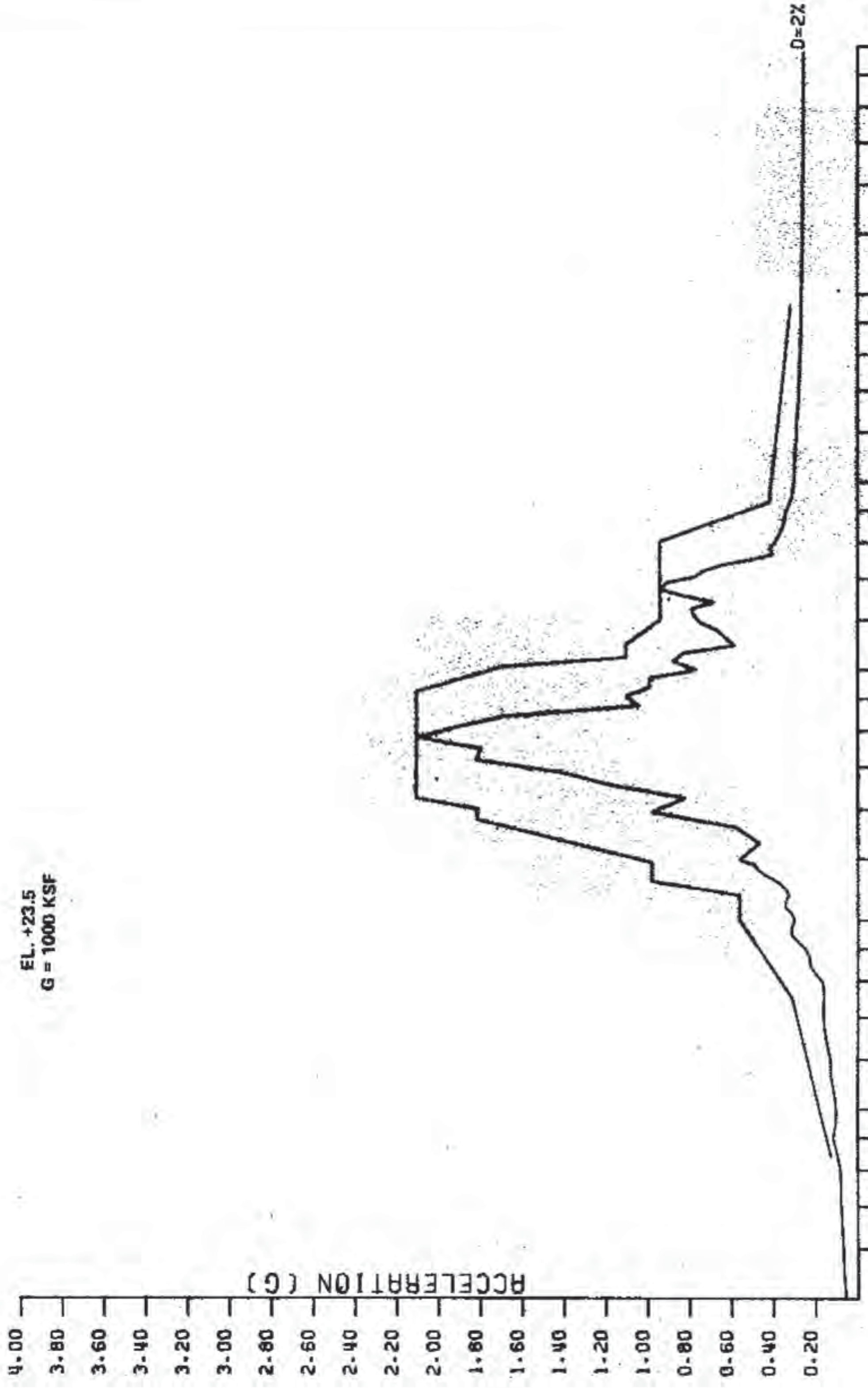
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
INTAKE STRUCTURE MASS PT. 4
FIGURE 3.7–153



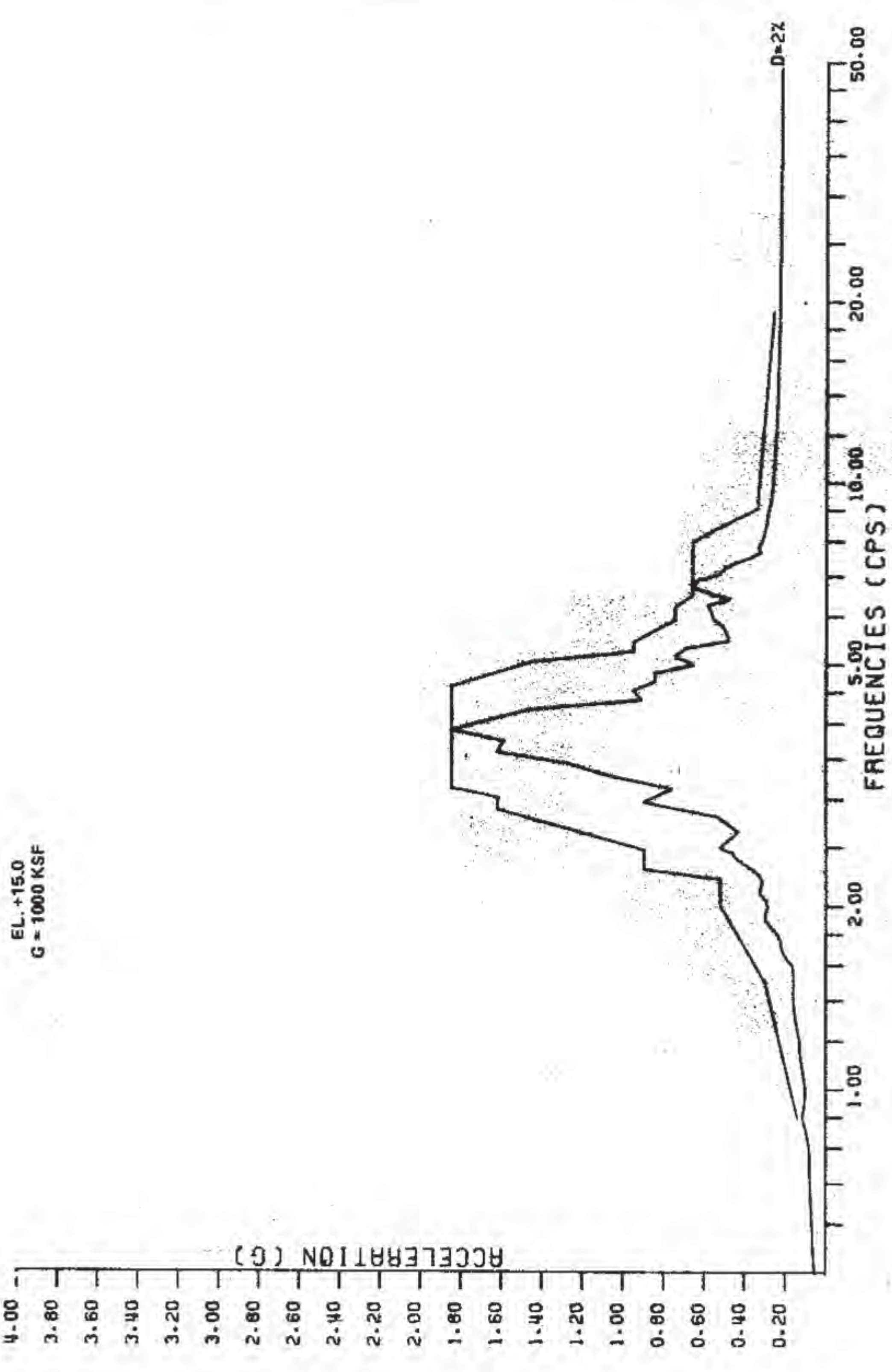
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
INTAKE STRUCTURE MASS PT. 1
FIGURE 3.7–154



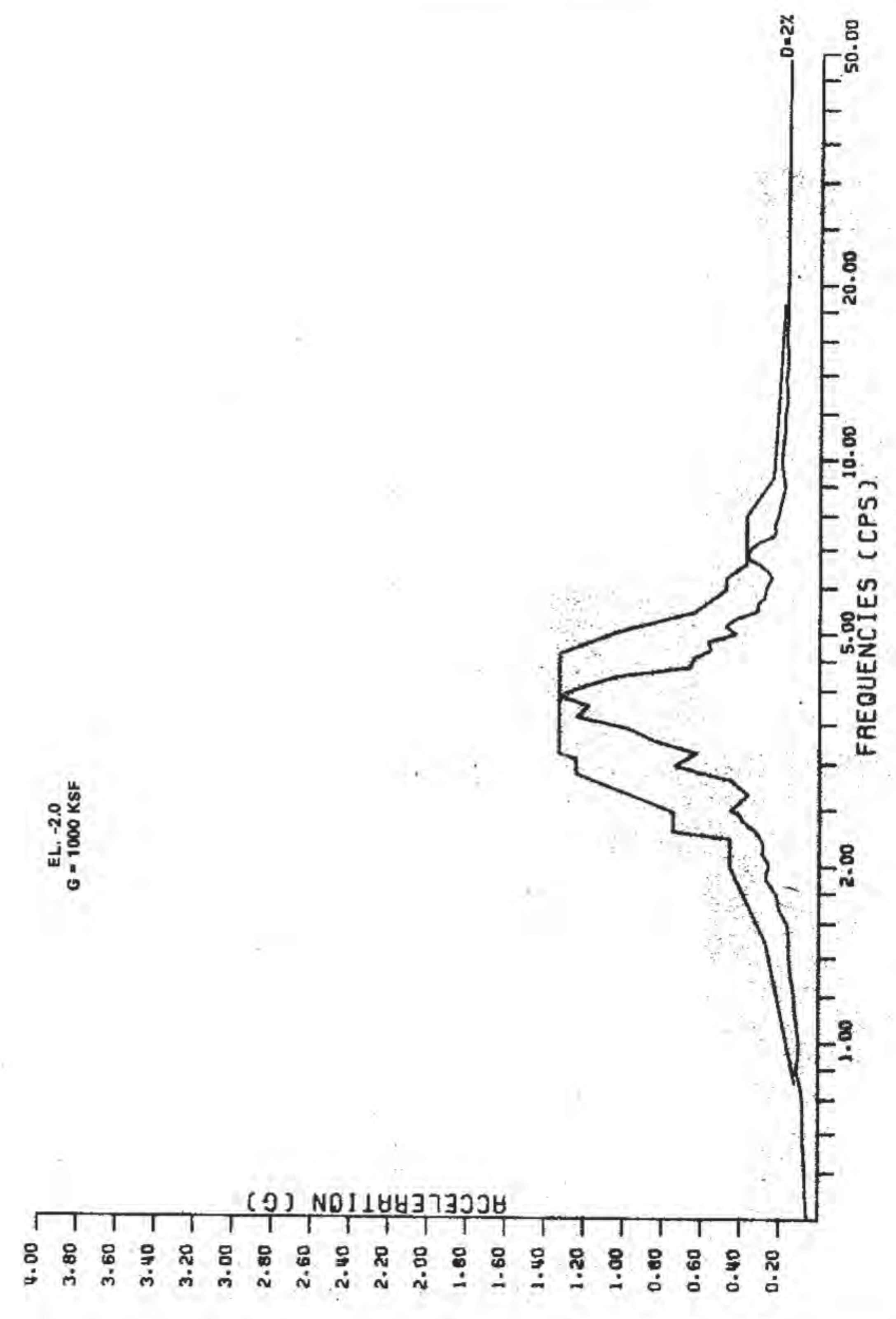
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
INTAKE STRUCTURE MASS PT. 2
FIGURE 3.7–155



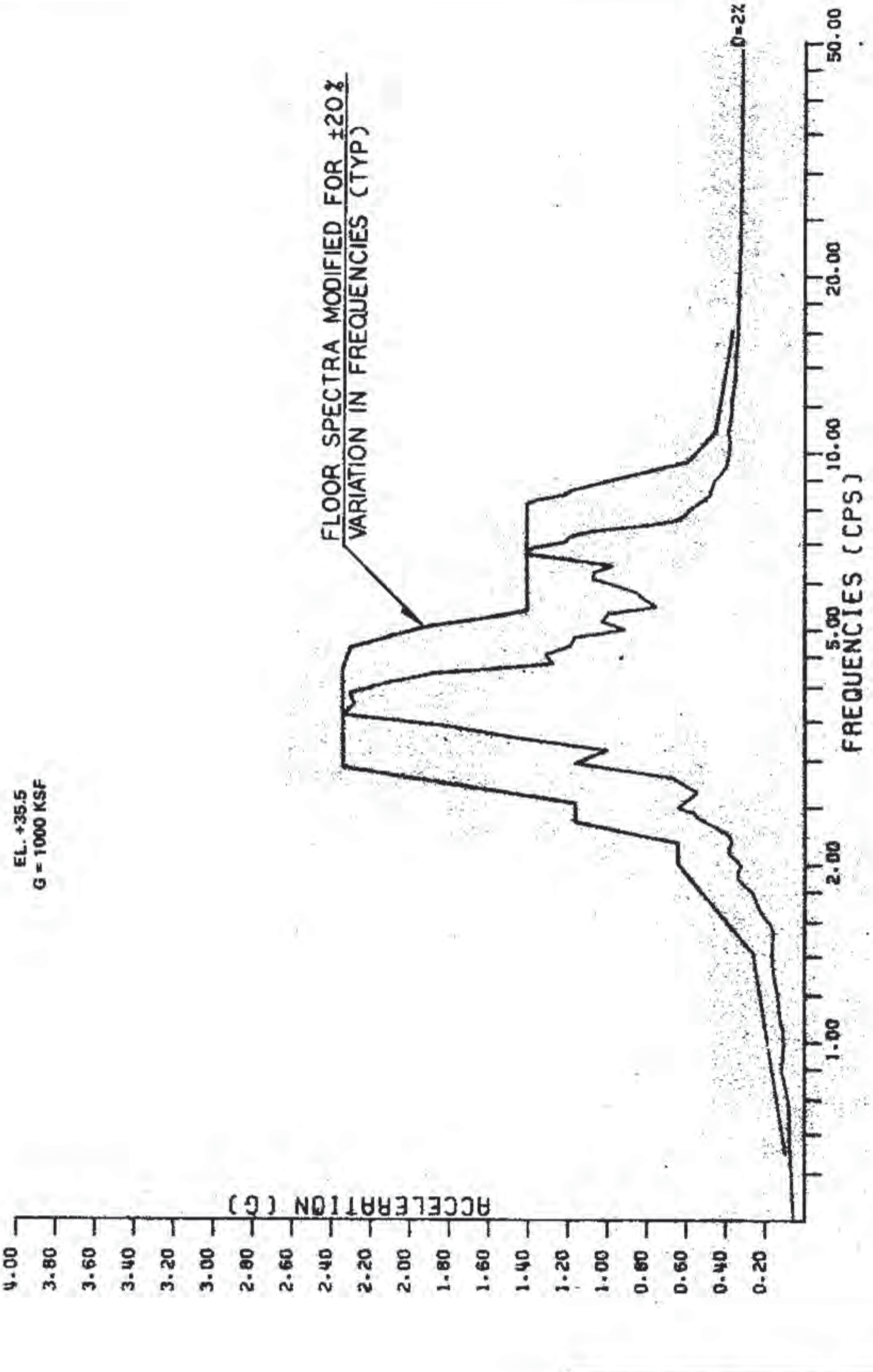
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
INTAKE STRUCTURE MASS PT.3
FIGURE 3.7–156



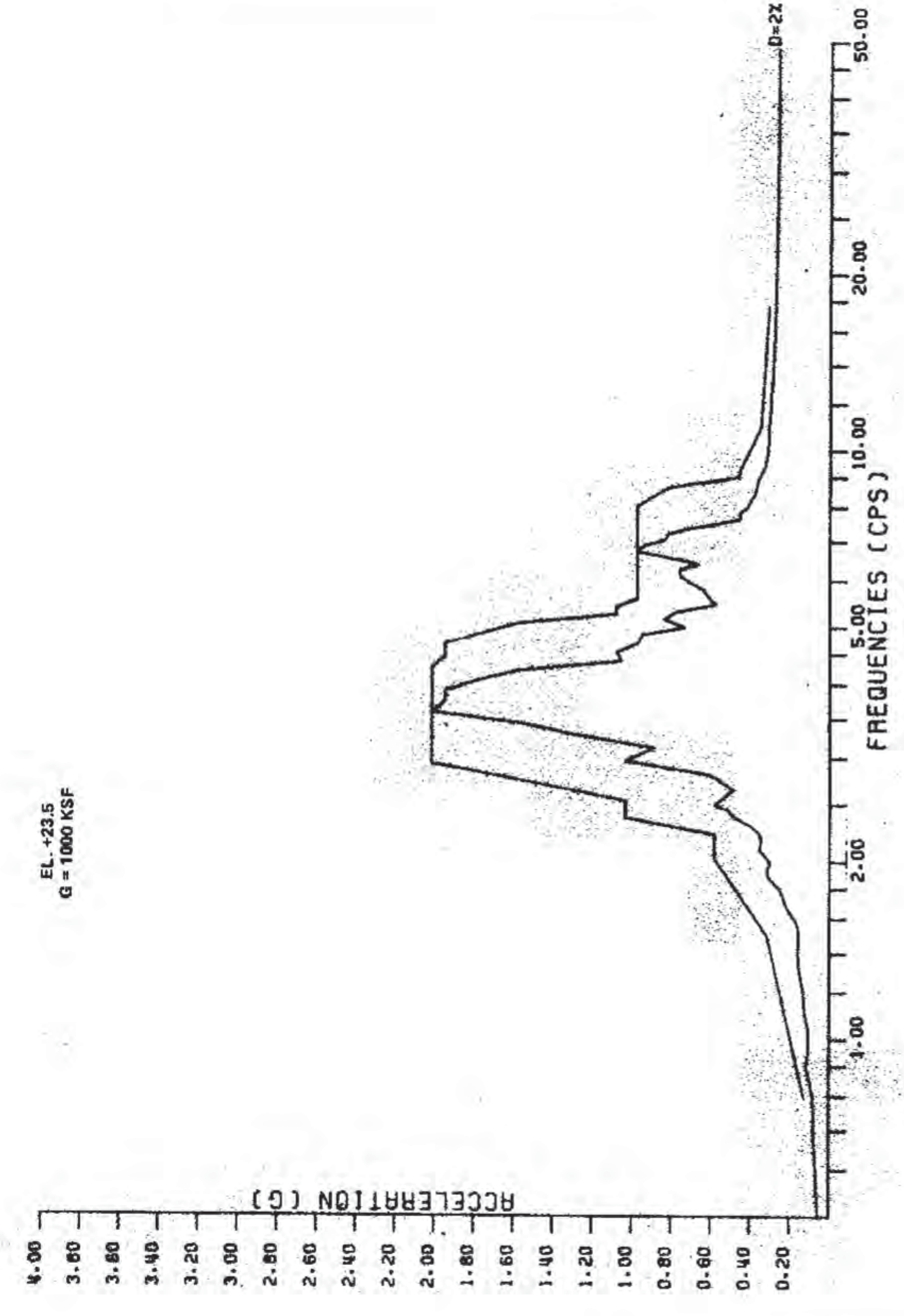
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
INTAKE STRUCTURE MASS PT.4
FIGURE 3.7-157



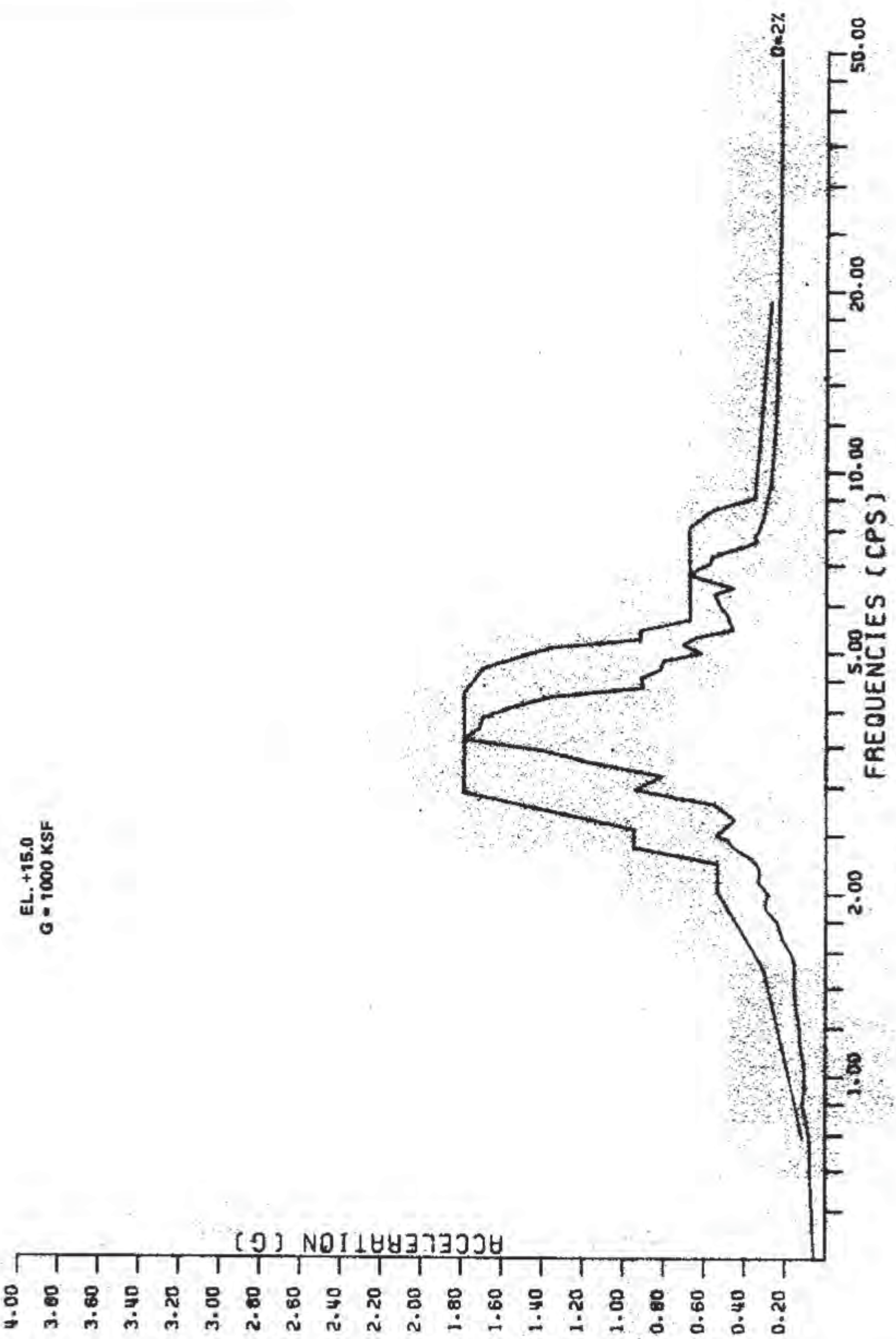
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
INTAKE STRUCTURE MASS PT.1
FIGURE 3.7-158



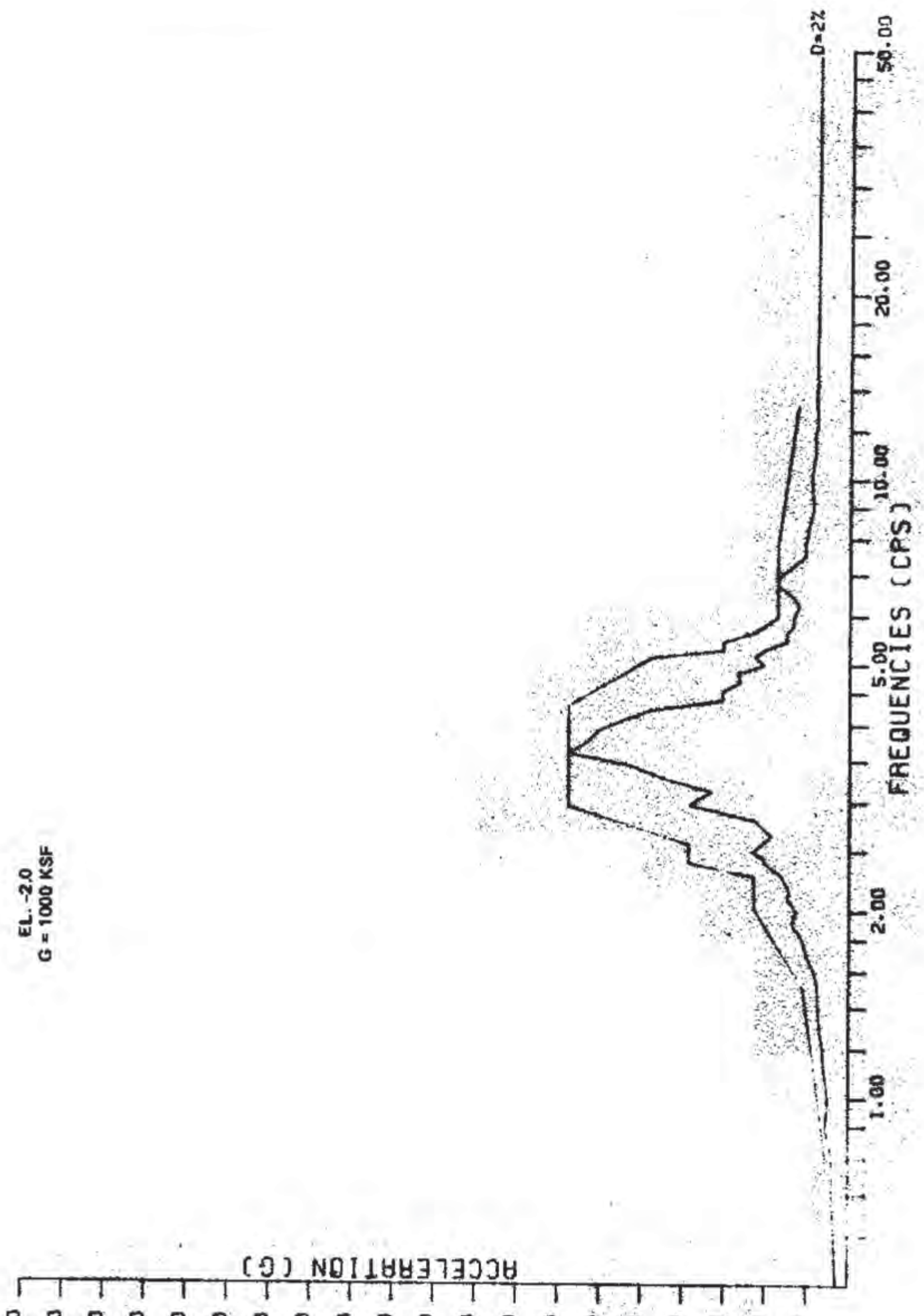
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
INTAKE STRUCTURE MASS PT.2
FIGURE 3.7-159



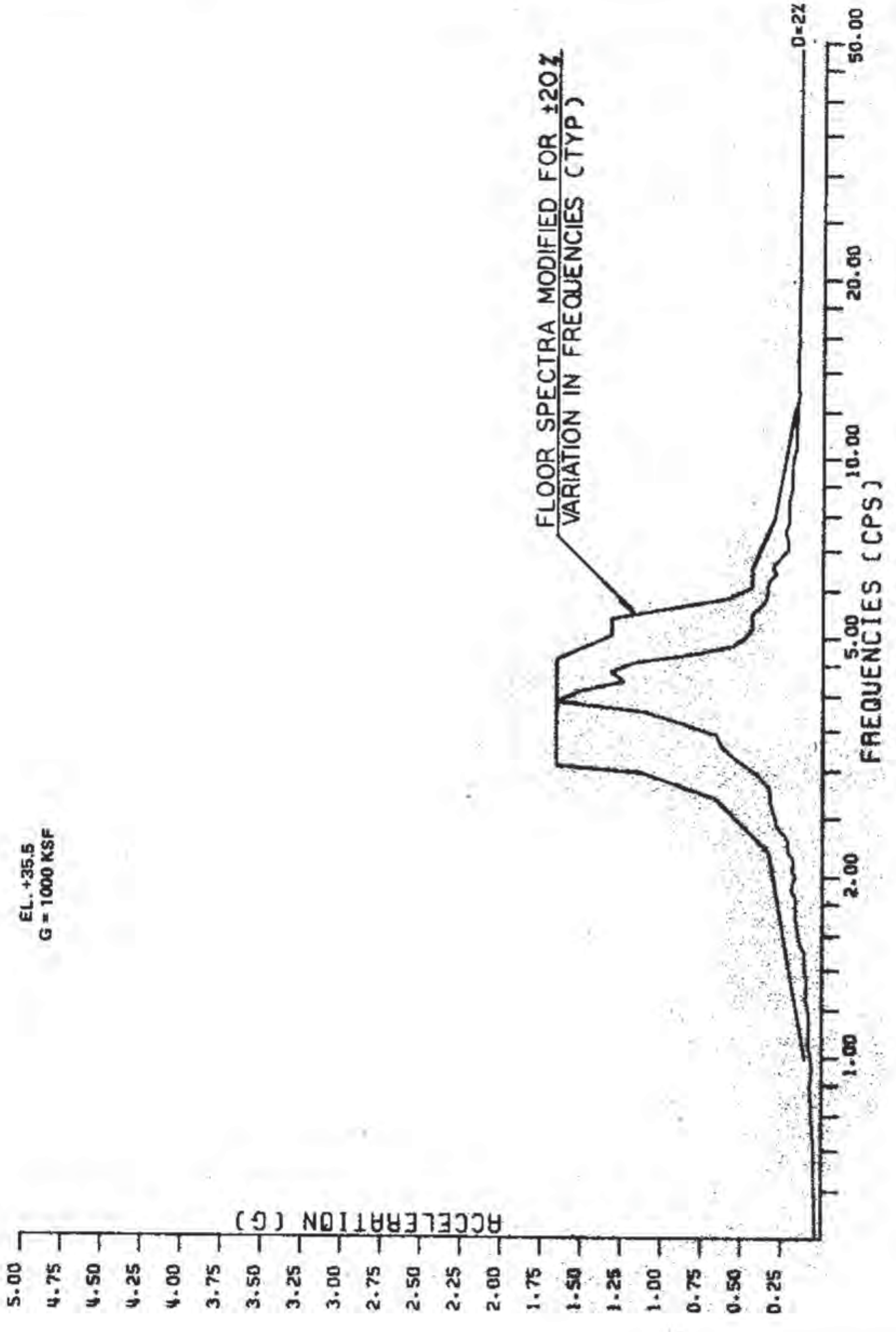
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
INTAKE STRUCTURE MASS PT.3
FIGURE 3.7–160



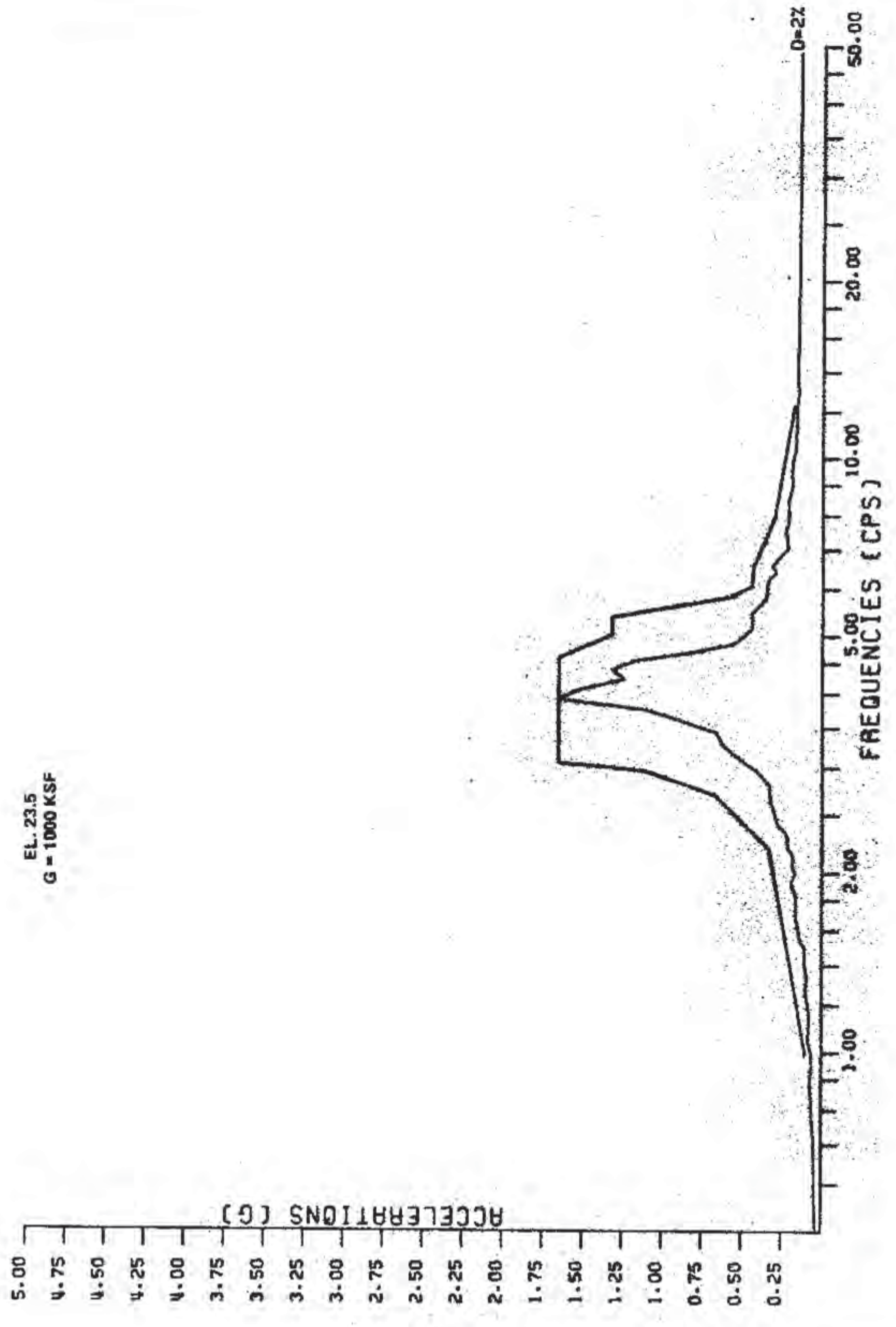
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
INTAKE STRUCTURE MASS PT.4
FIGURE 3.7–161



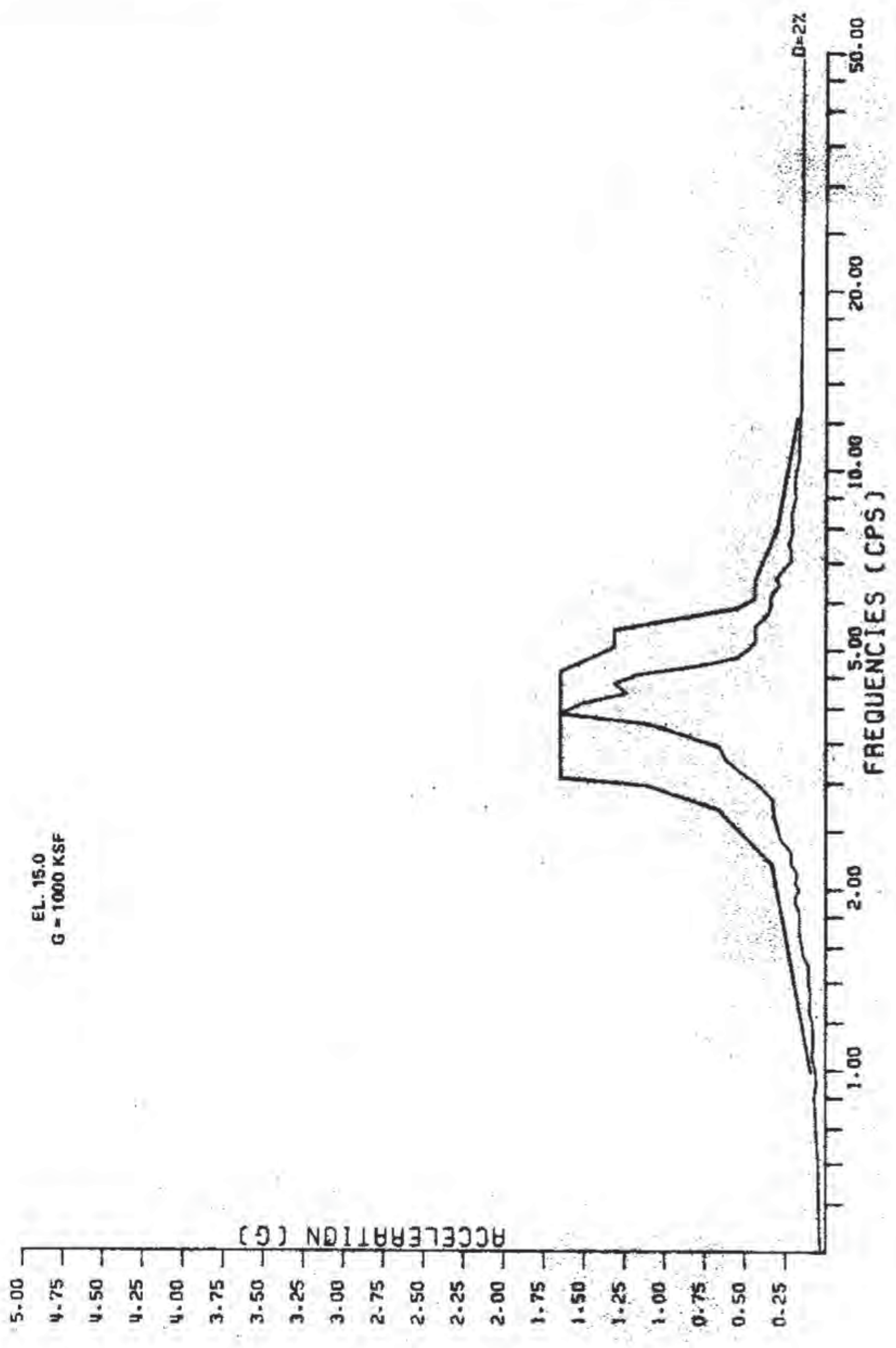
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
INTAKE STRUCTURE MASS PT.1
FIGURE 3.7–162



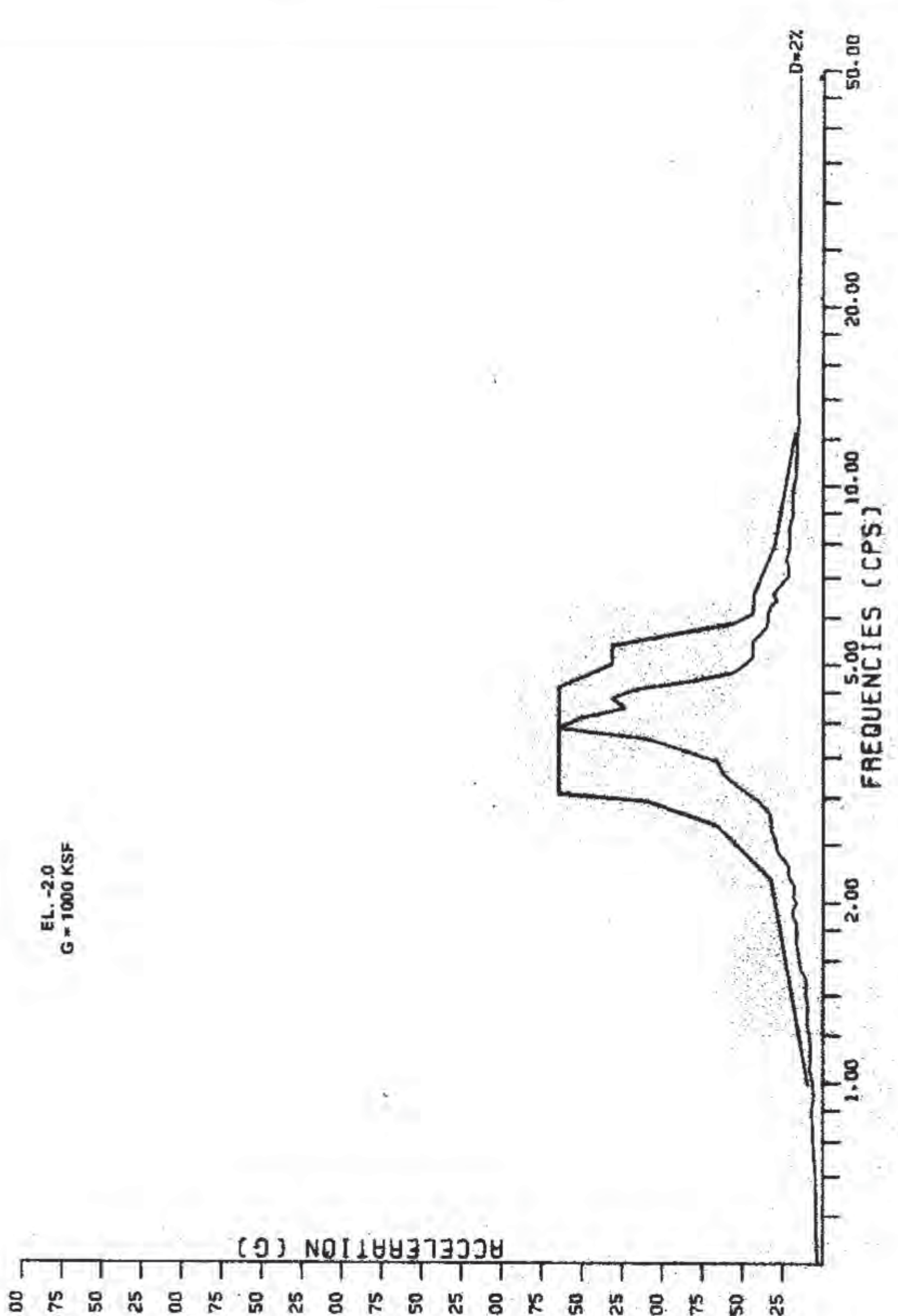
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
INTAKE STRUCTURE MASS PT.2
FIGURE 3.7-163



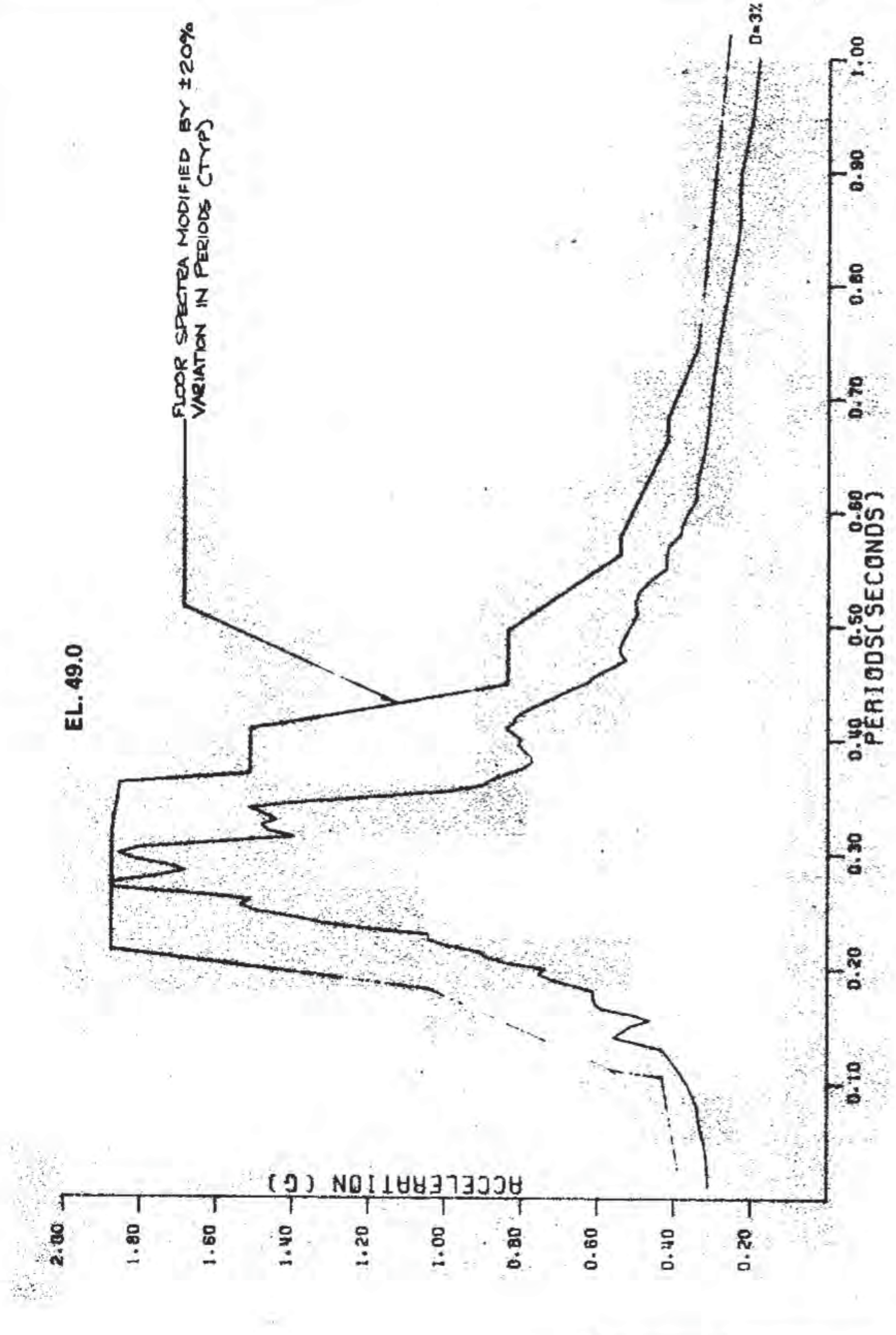
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
INTAKE STRUCTURE MASS PT.3
FIGURE 3.7–164



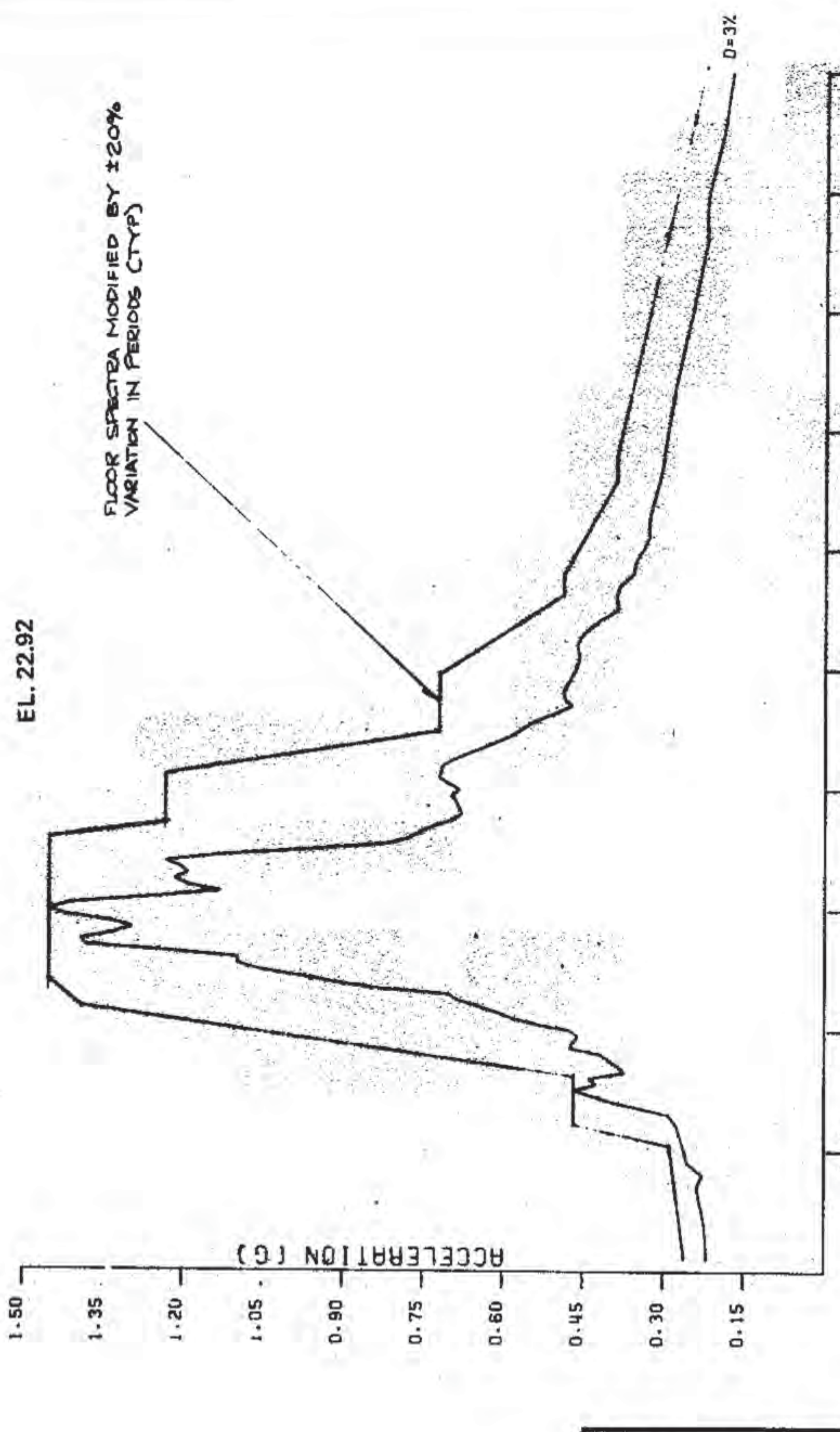
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
INTAKE STRUCTURE MASS PT.4
FIGURE 3.7-165



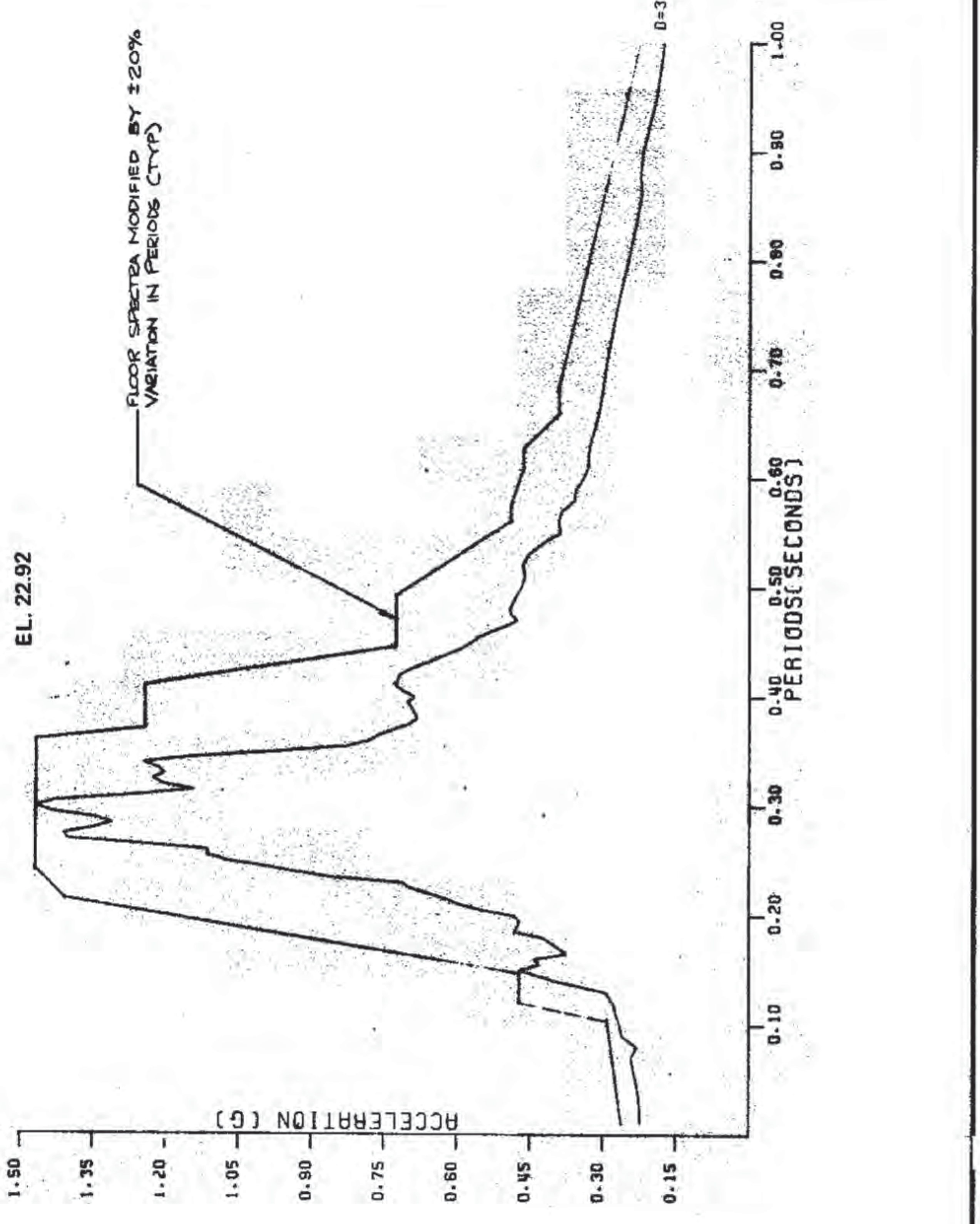
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7-166



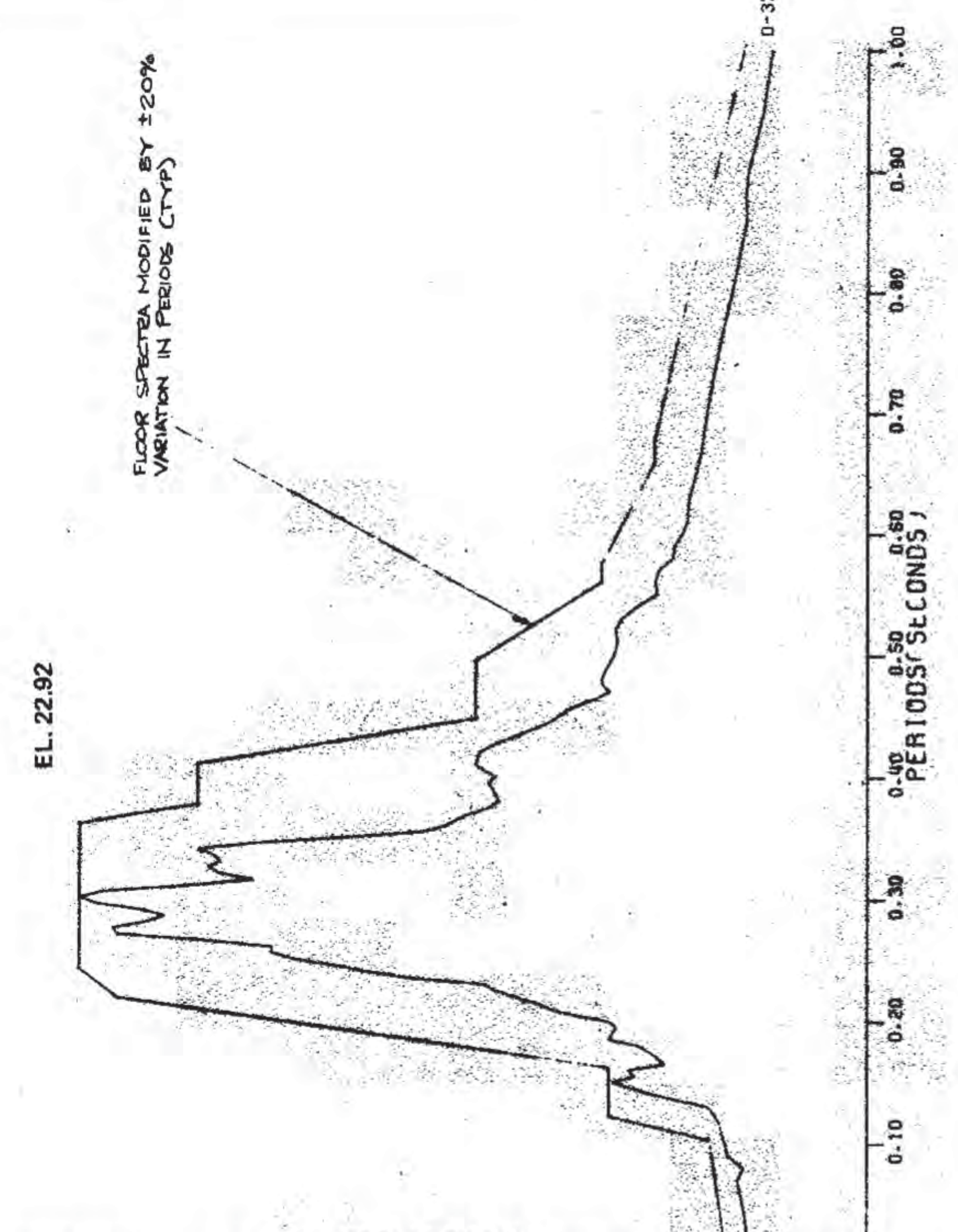
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7–167



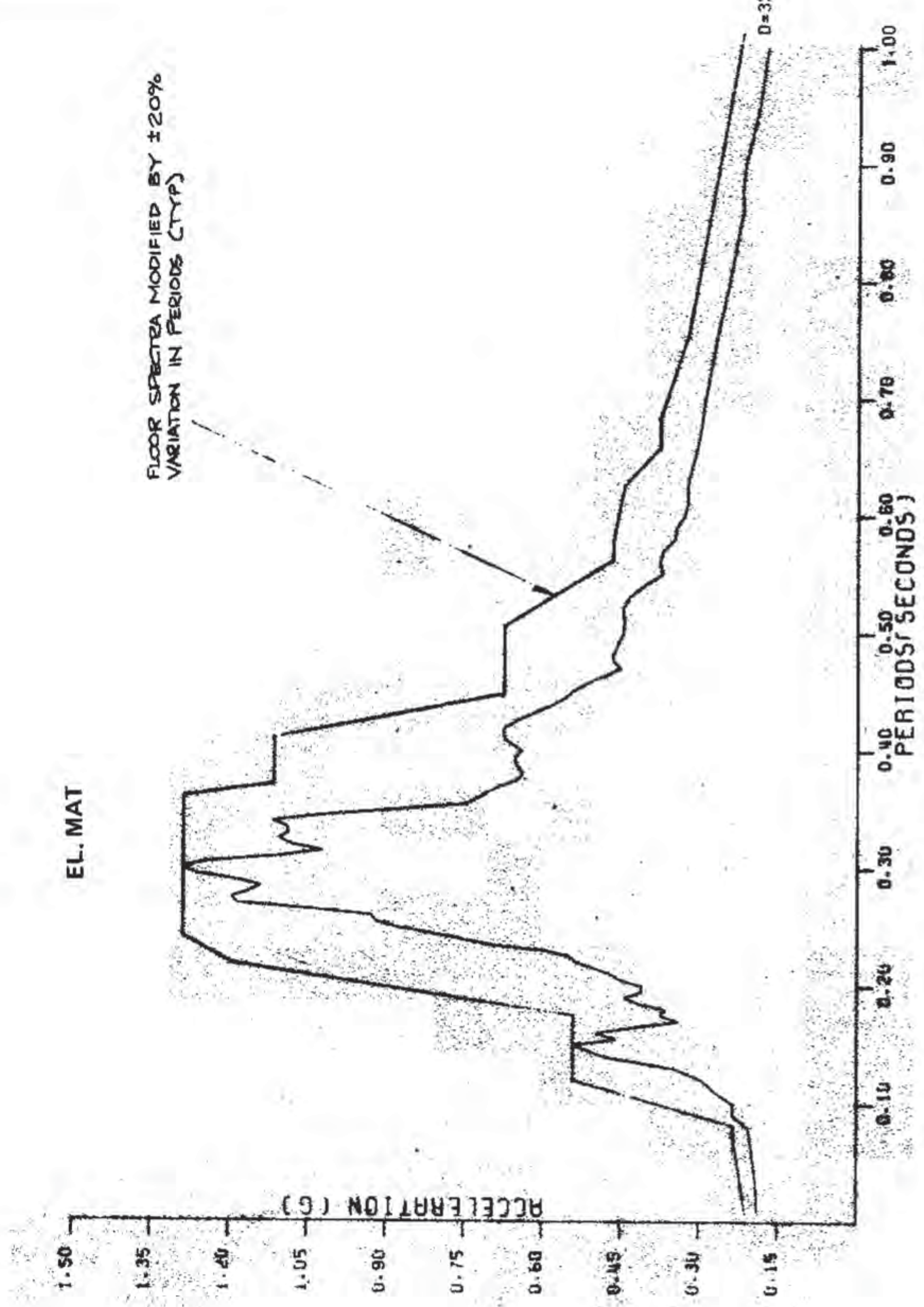
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
DIESEL GENERATOR BLDG. CANT. 2
FIGURE 3.7-168



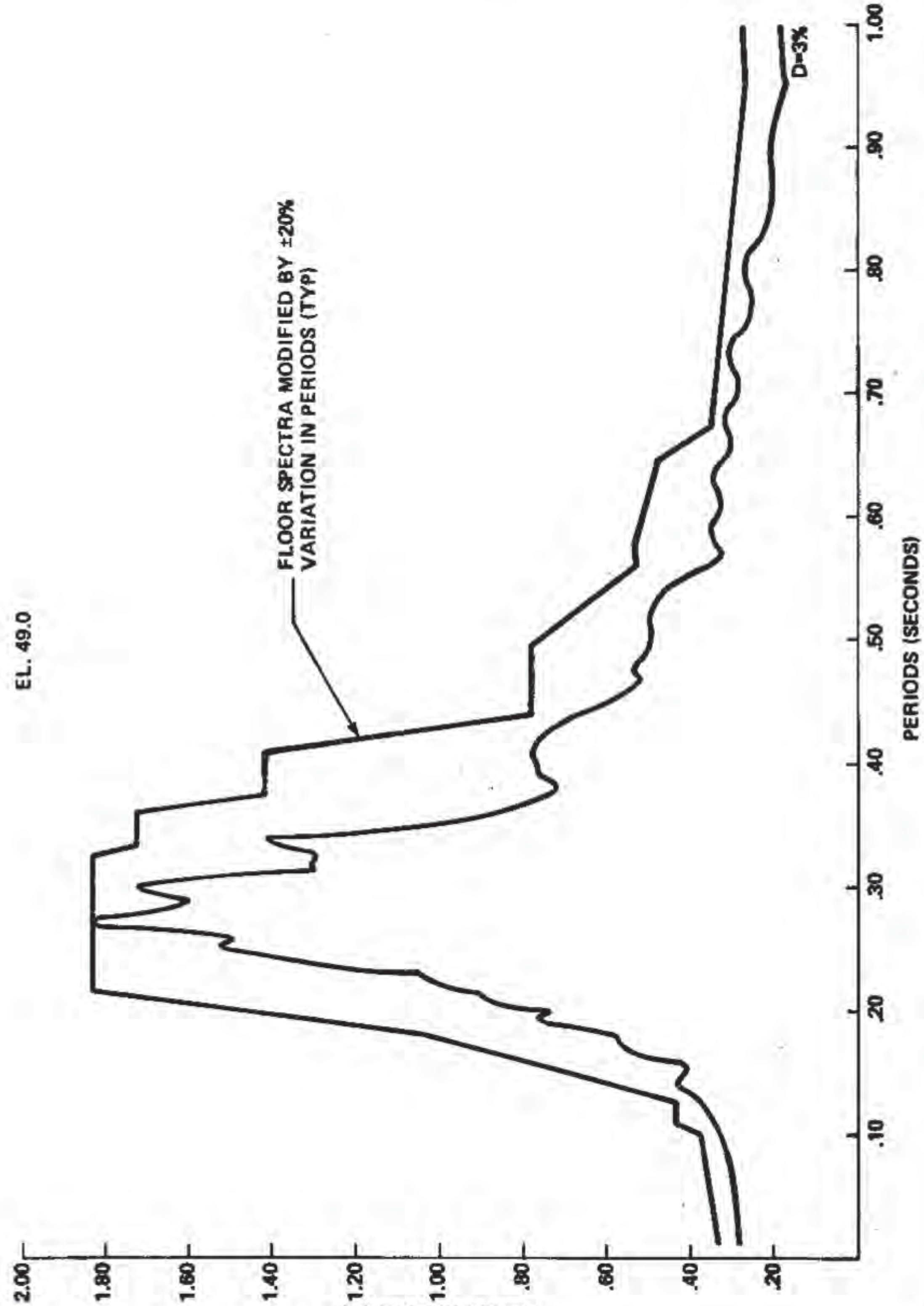
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
DIESEL GENERATOR BLDG. CANT.3
FIGURE 3.7–169



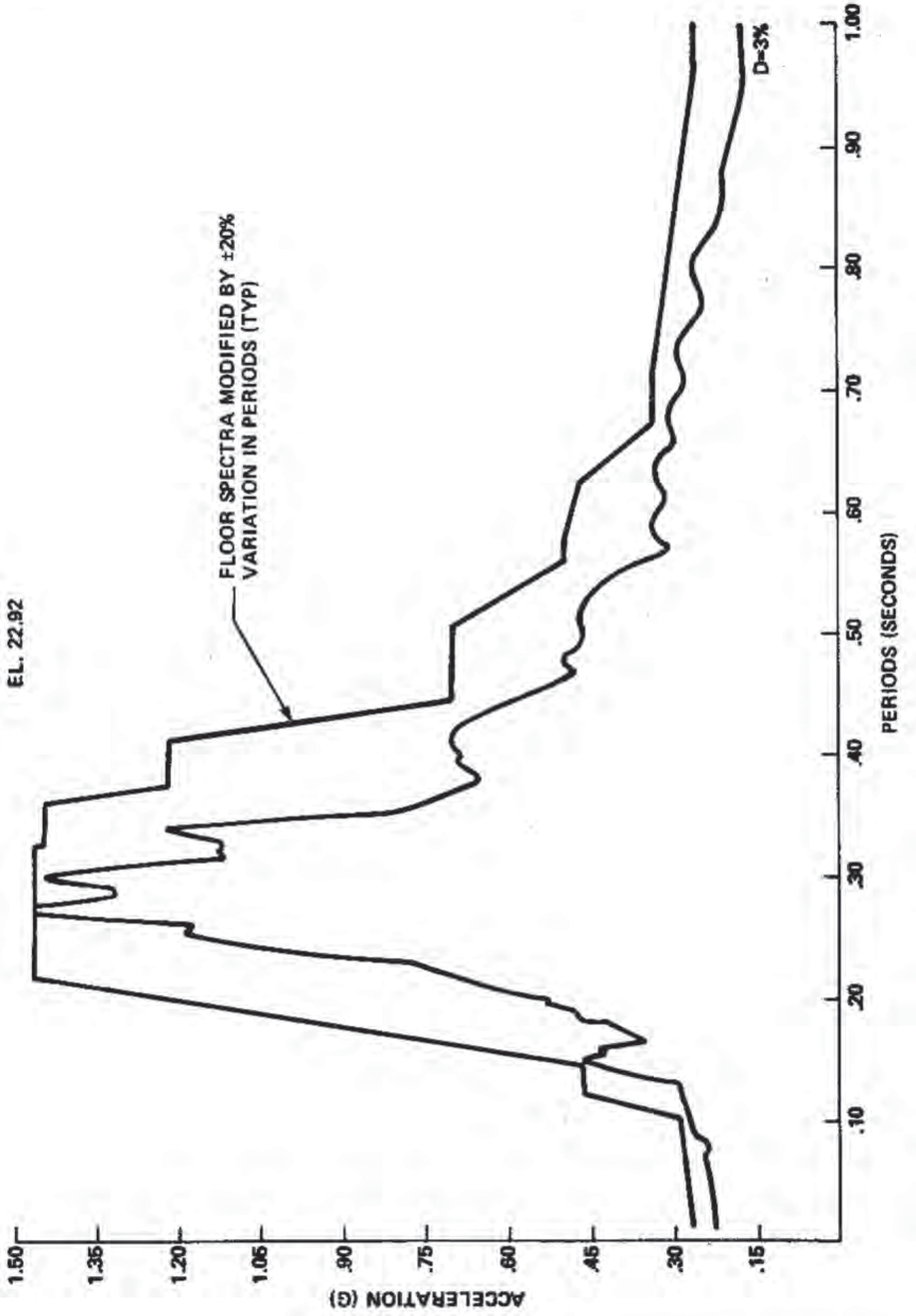
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
DIESEL GENERATOR BLDG.
FIGURE 3.7–170



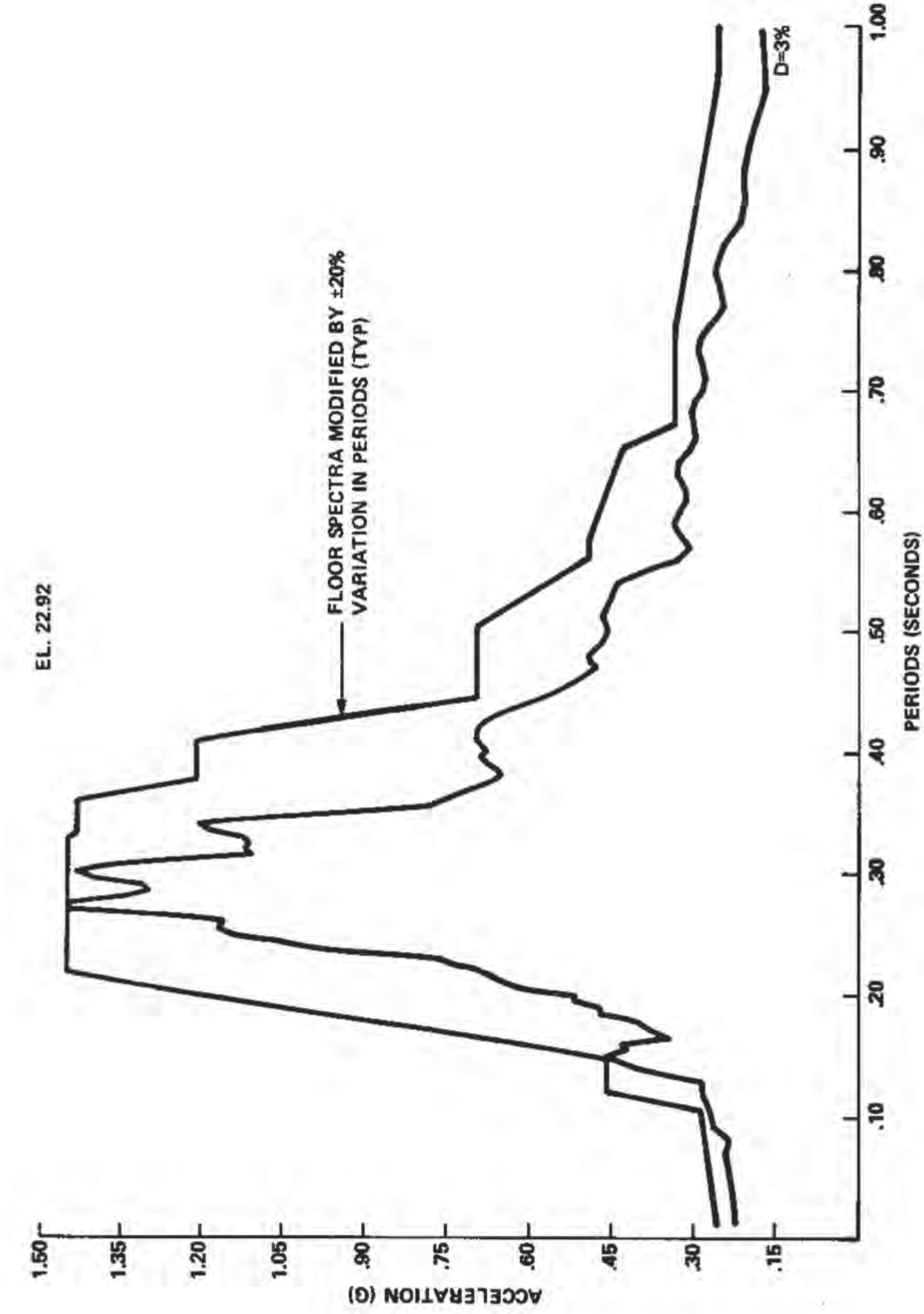
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7-171



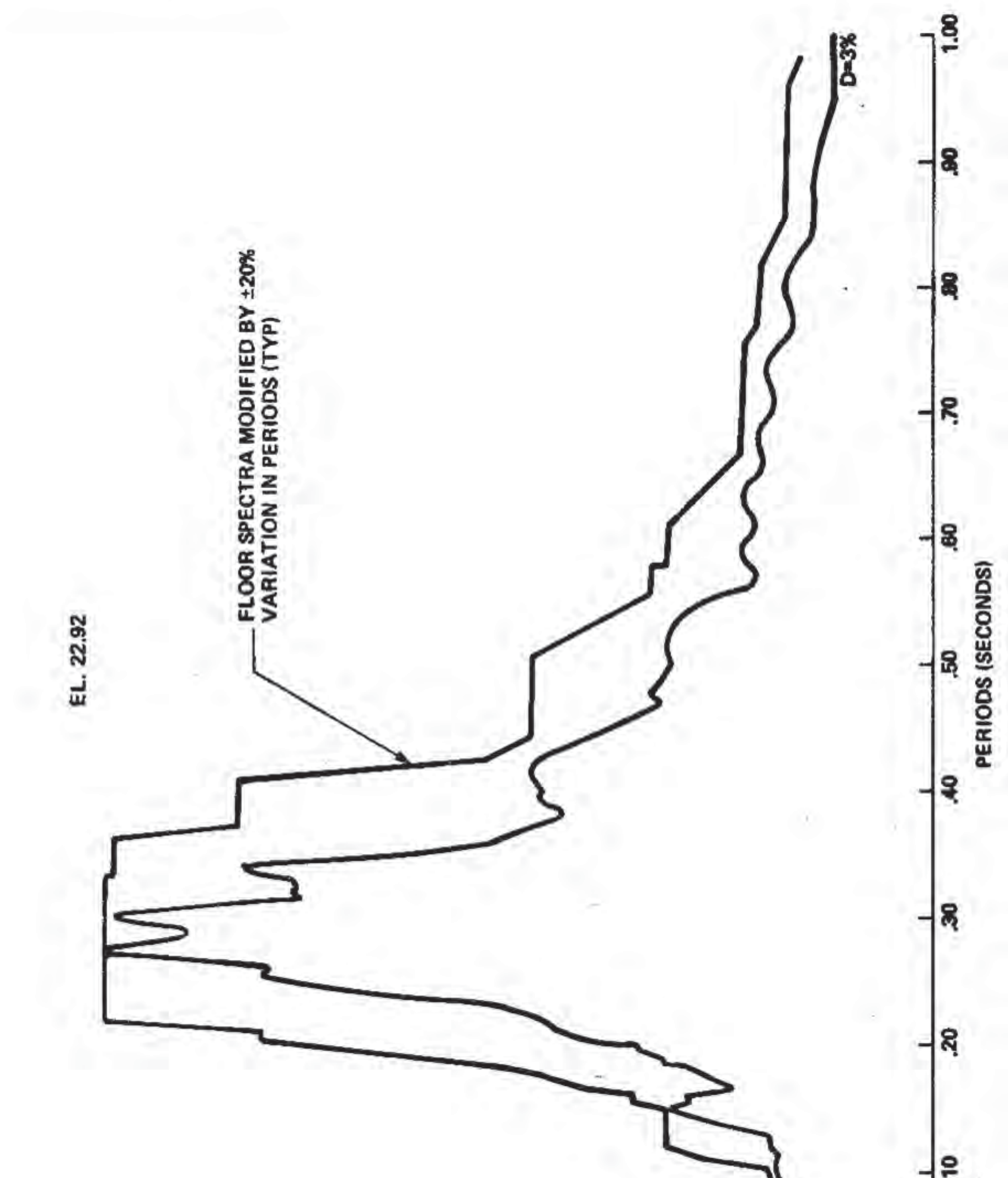
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7-172



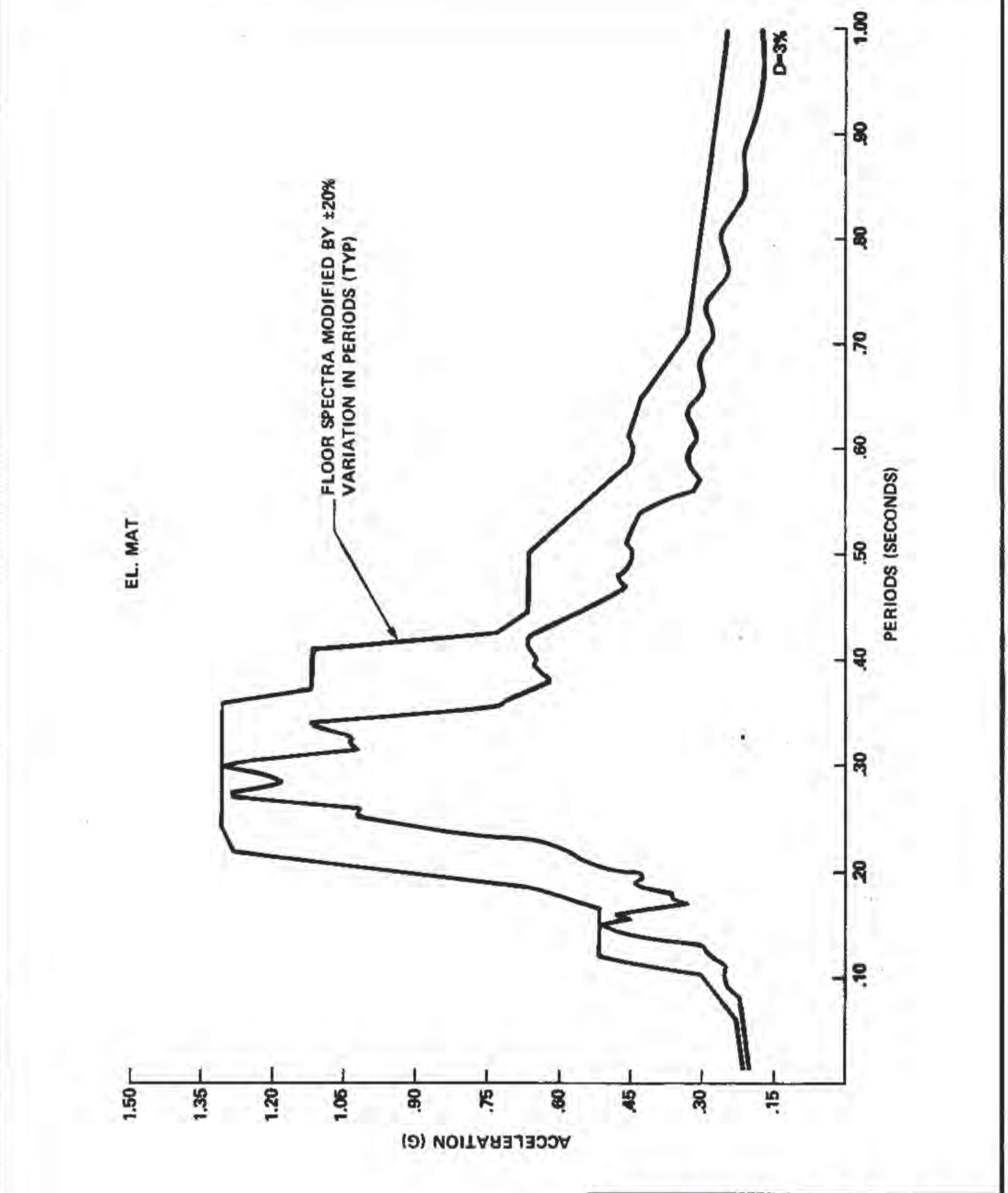
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
DIESEL GENERATOR BLDG. CANT.2
FIGURE 3.7–173



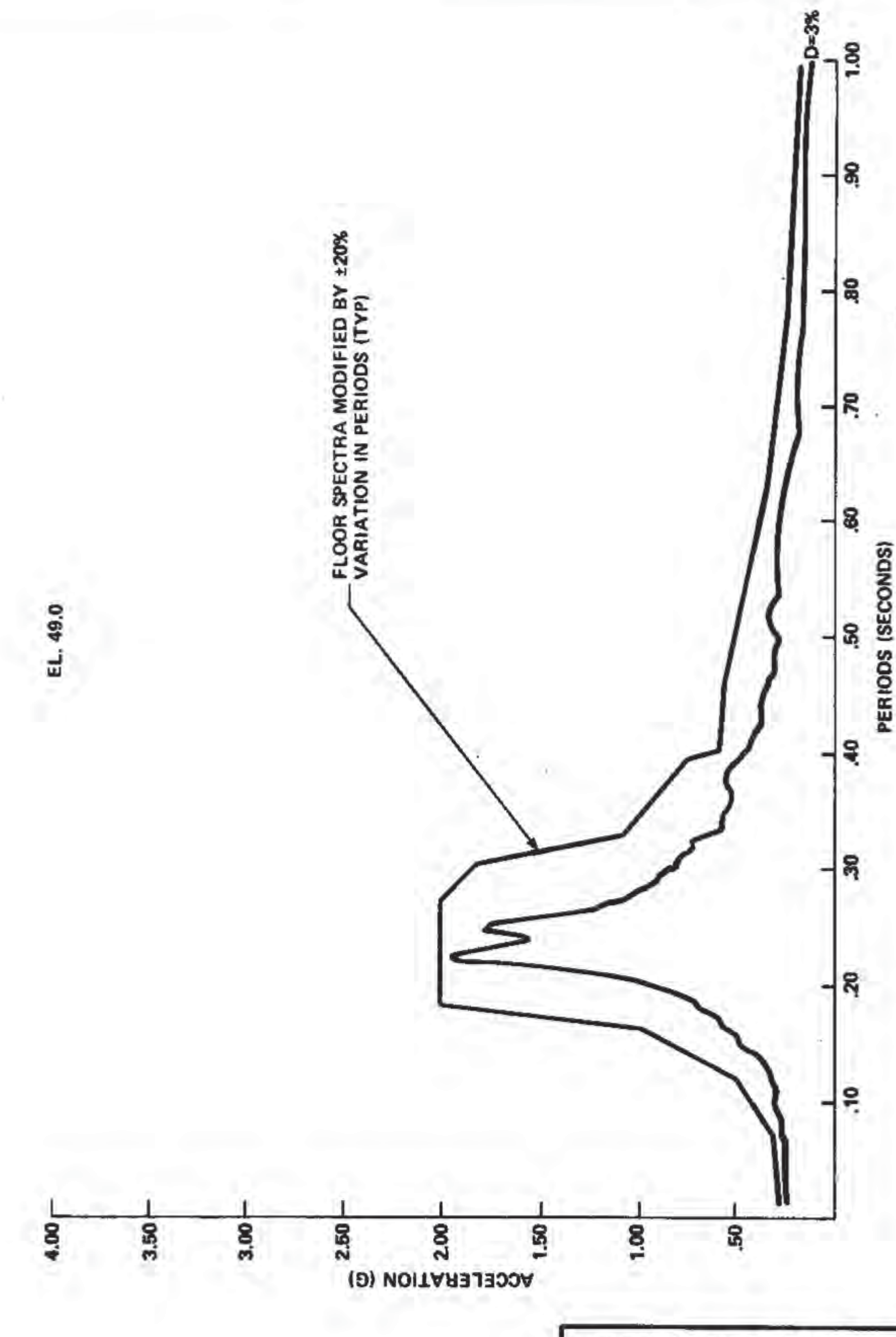
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
DIESEL GENERATOR BLDG. CANT. 3
FIGURE 3.7-174



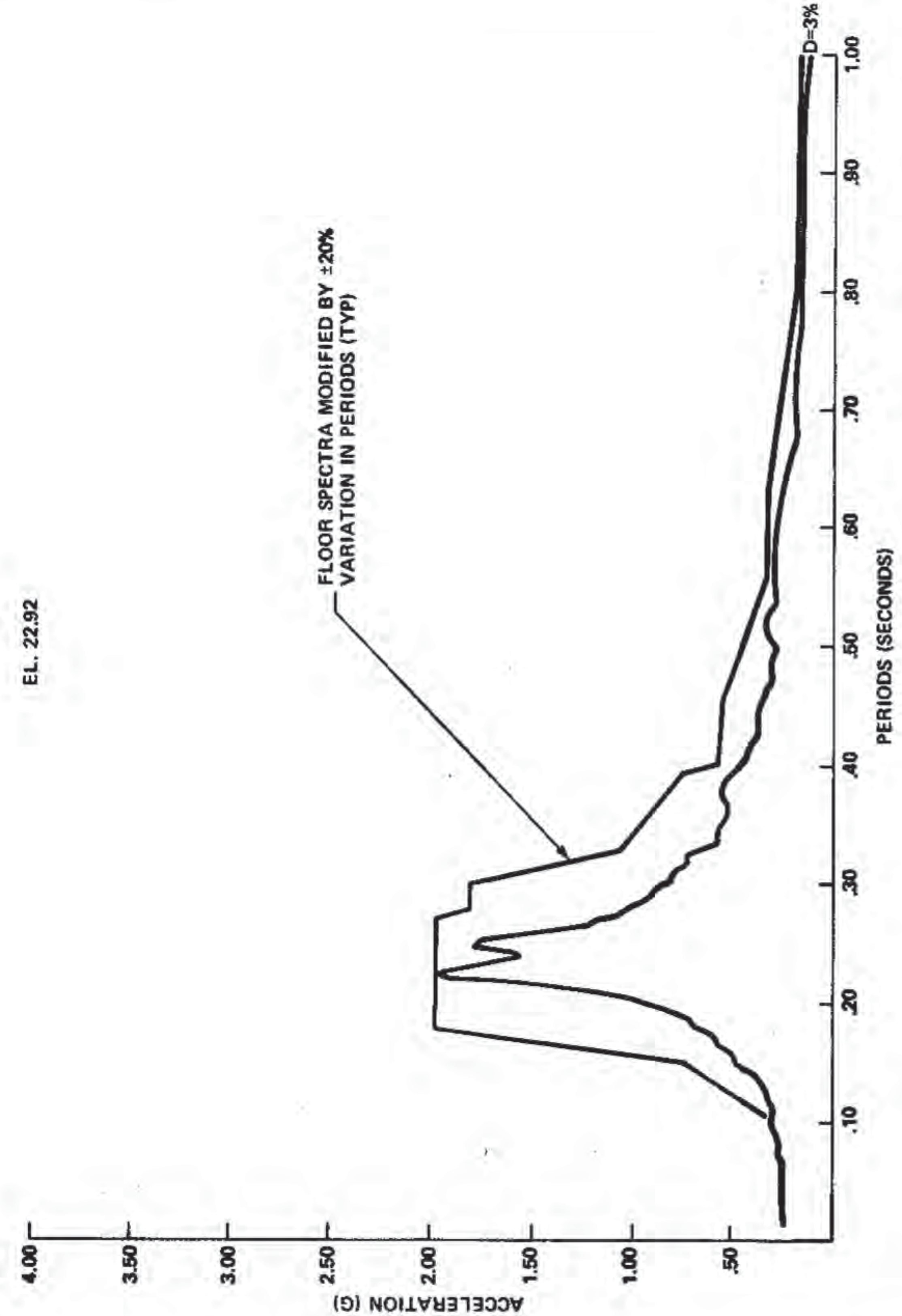
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
DIESEL GENERATOR BLDG.
FIGURE 3.7–175



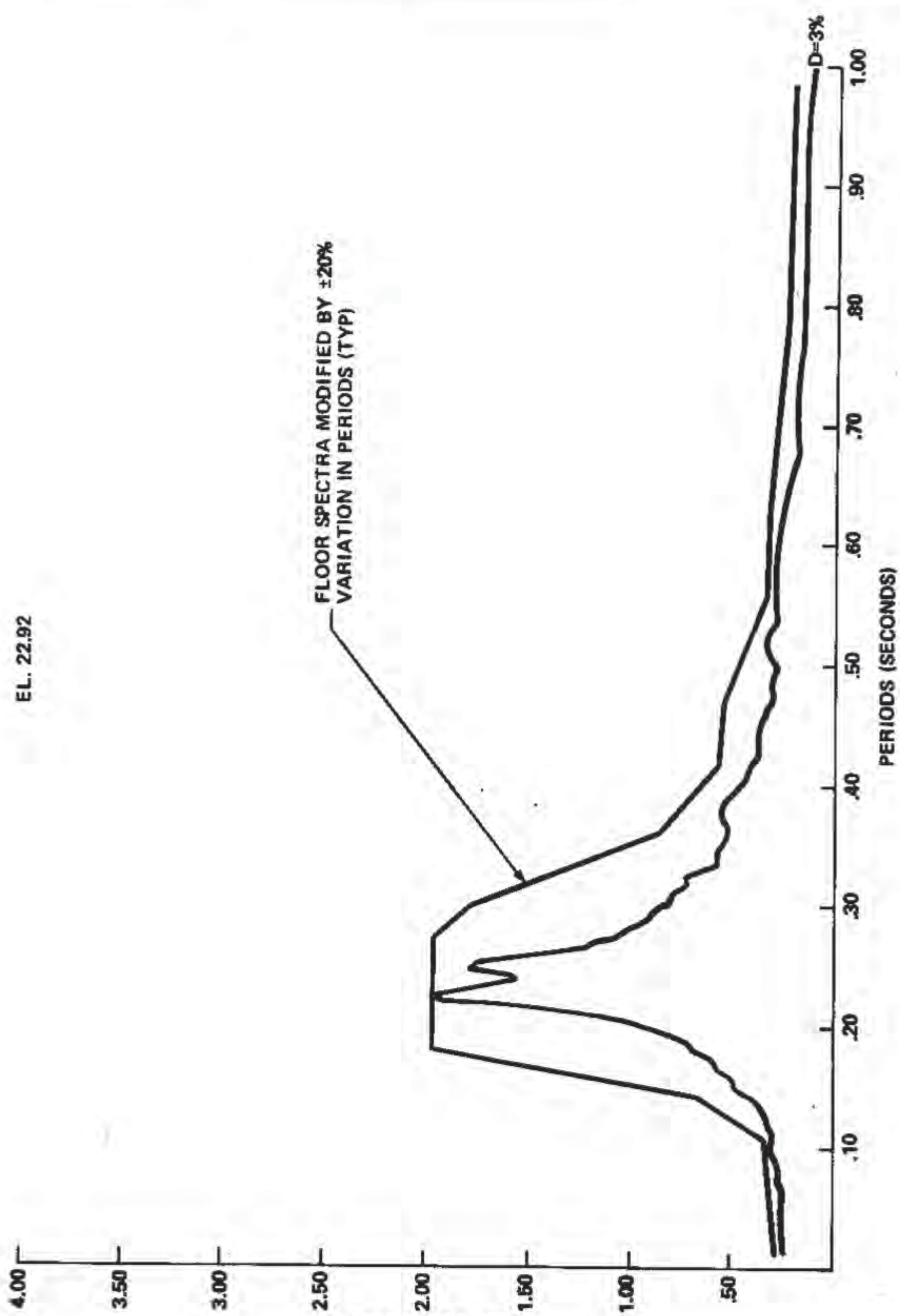
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7-176



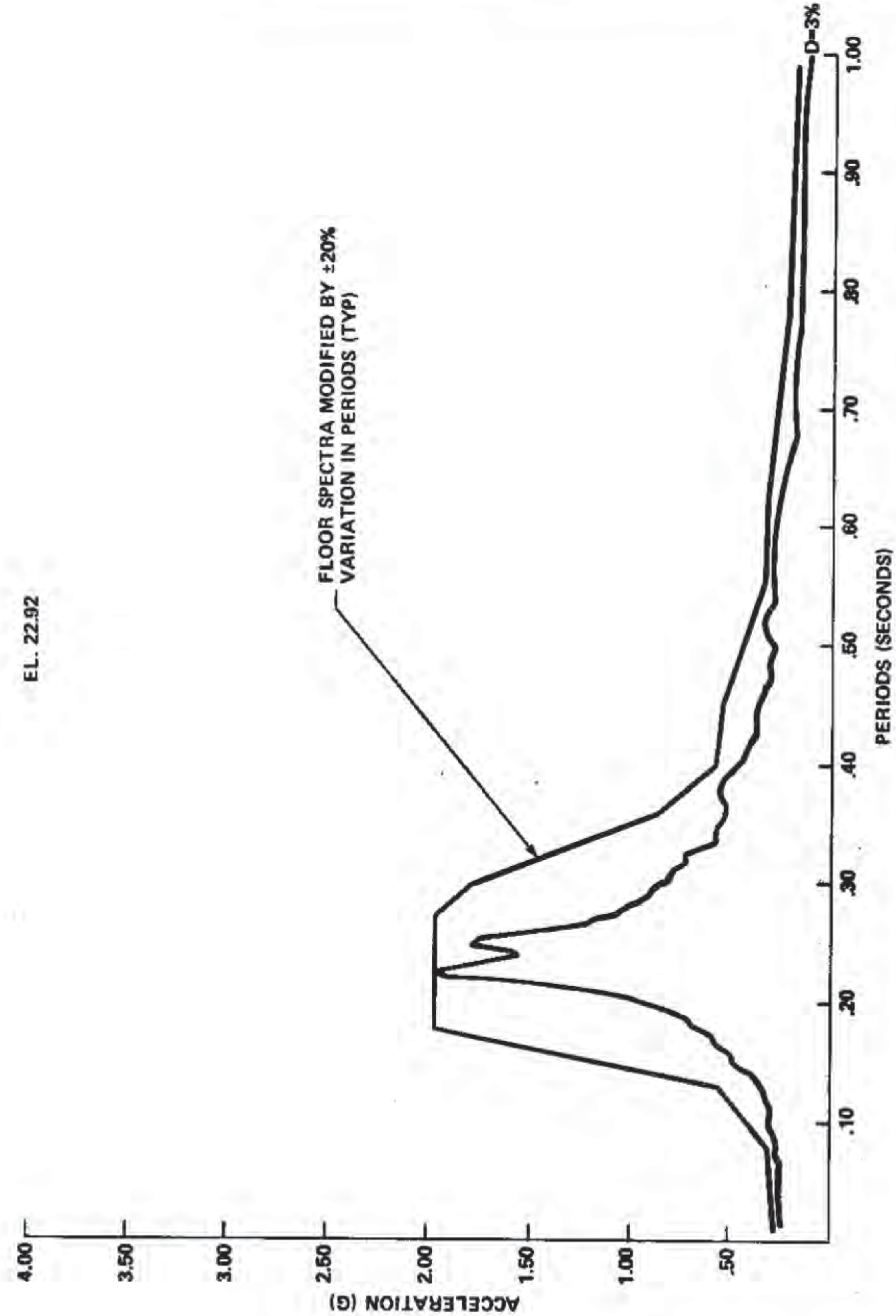
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7–177



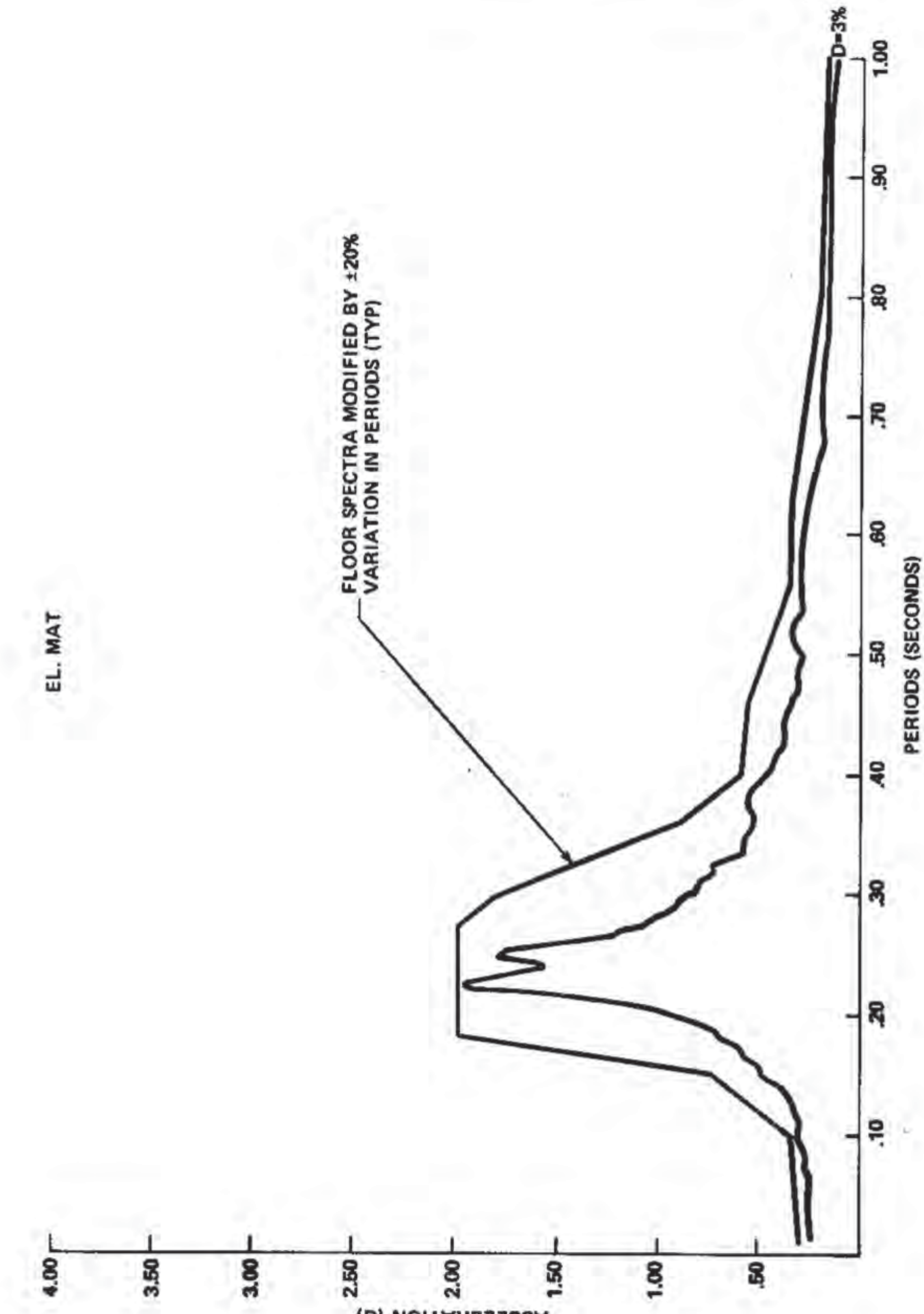
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
DIESEL GENERATOR BLDG. CANT. 2
FIGURE 3.7–178



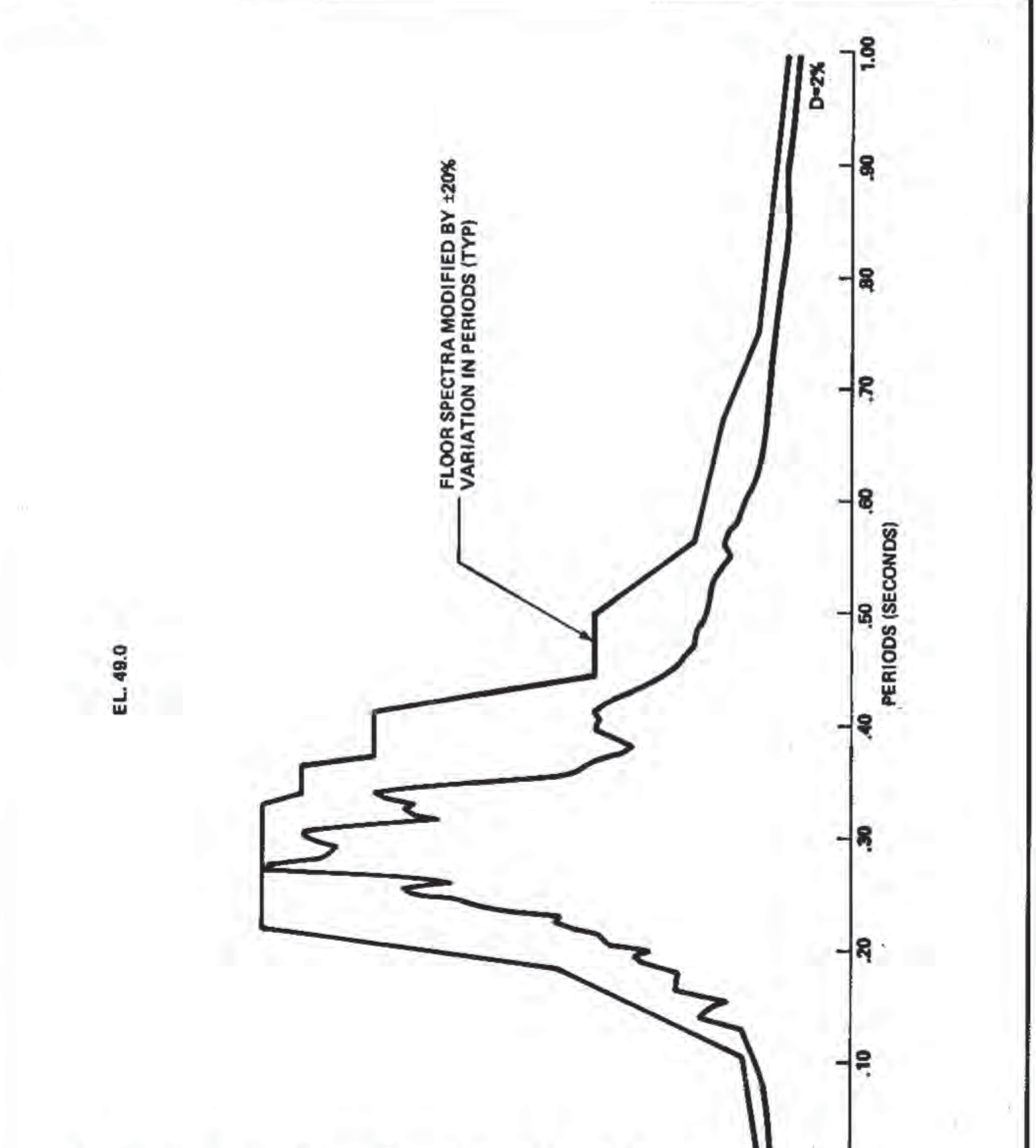
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
DIESEL GENERATOR BLDG. CANT. 3
FIGURE 3.7–179



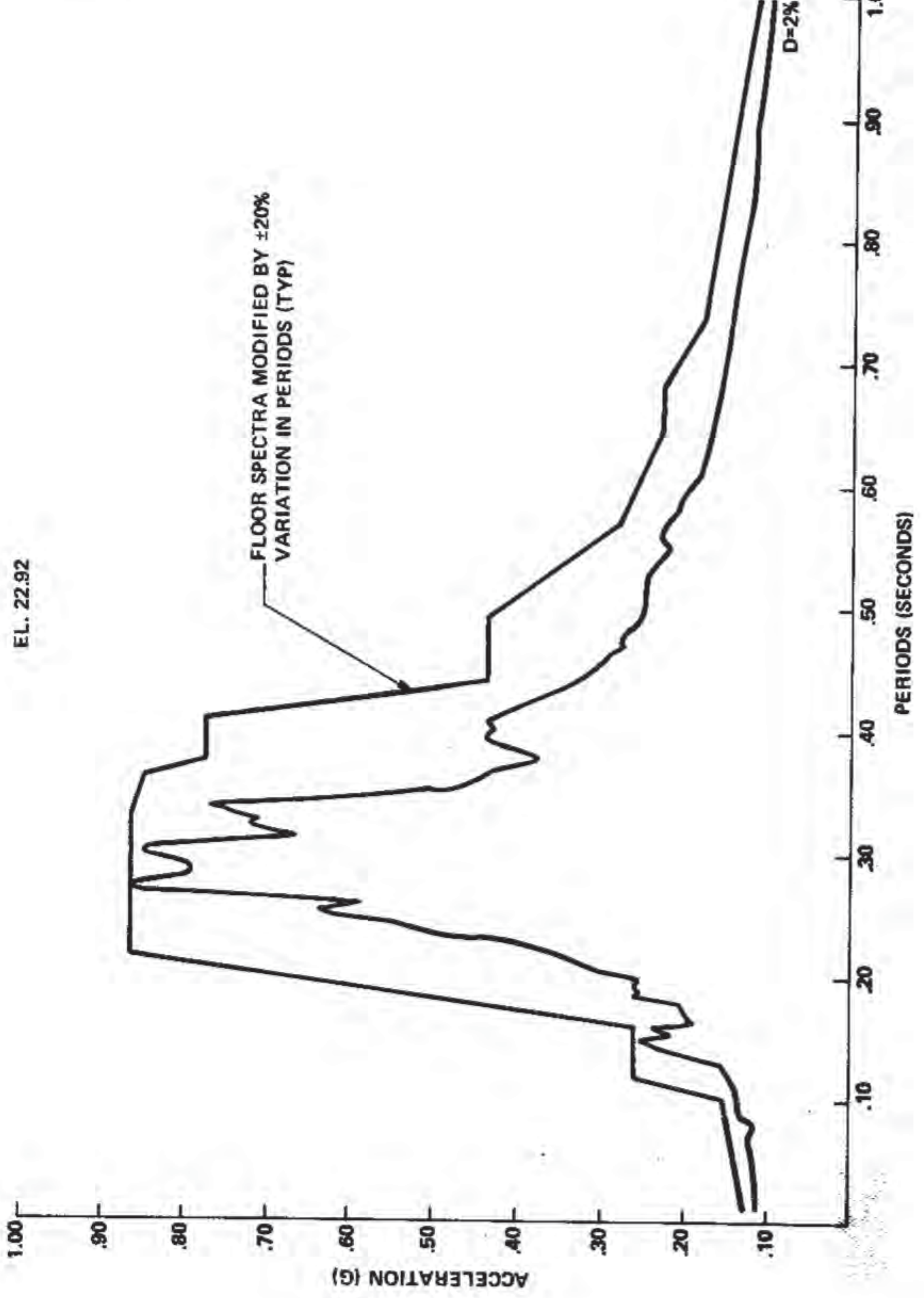
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
DIESEL GENERATOR BLDG.
FIGURE 3.7–180



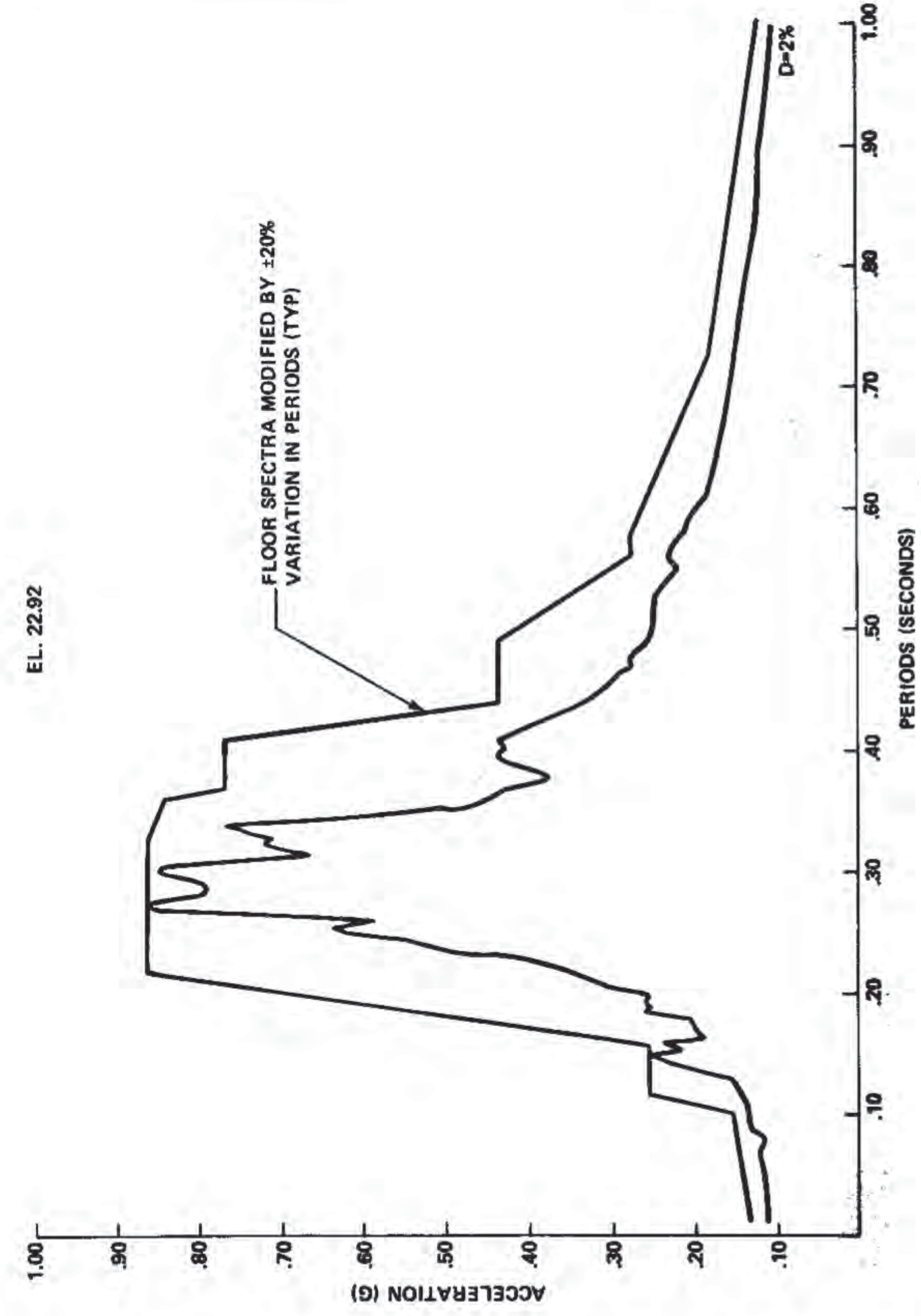
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7–181



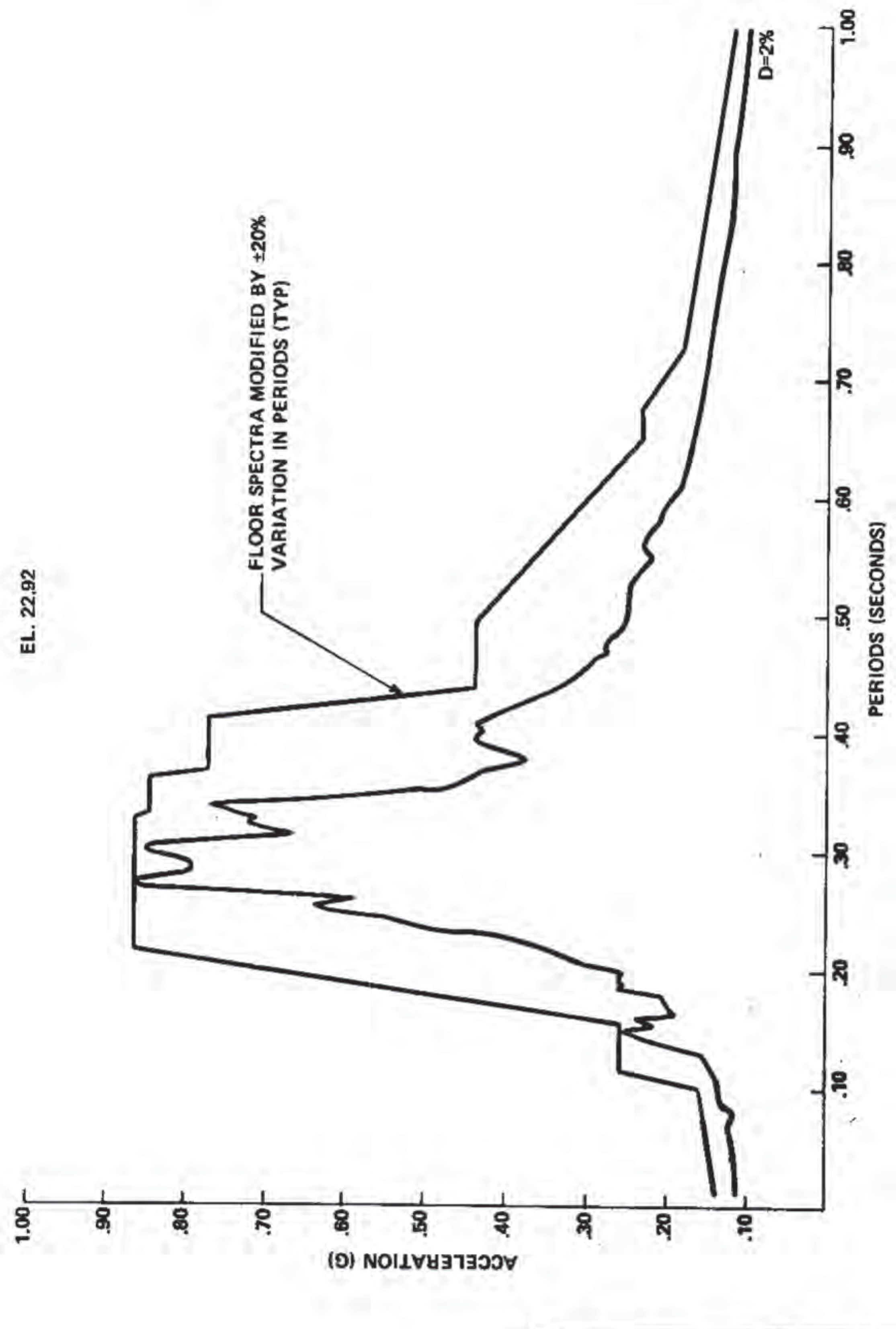
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7–182



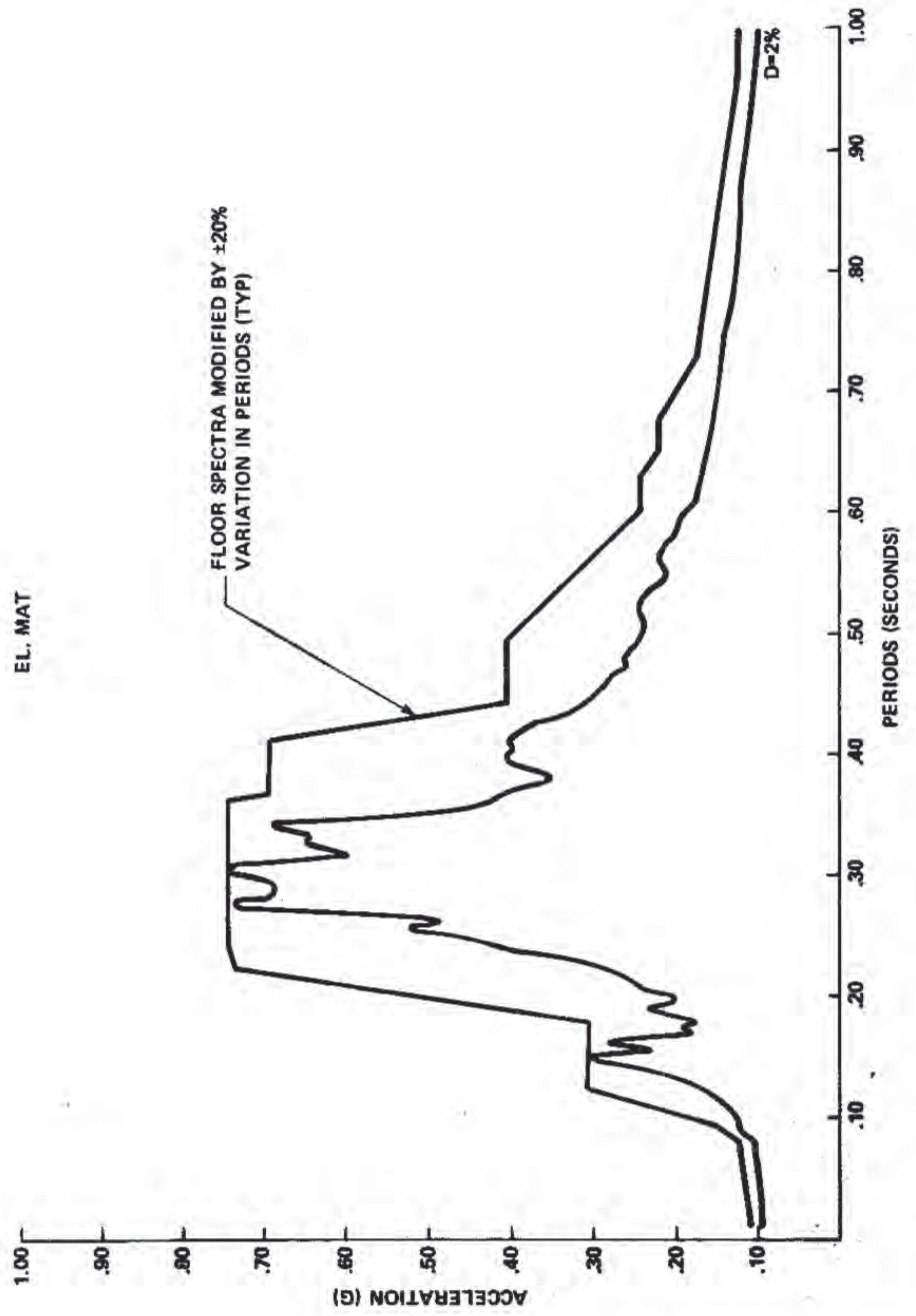
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
DIESEL GENERATOR BLDG. CANT. 2
FIGURE 3.7–183



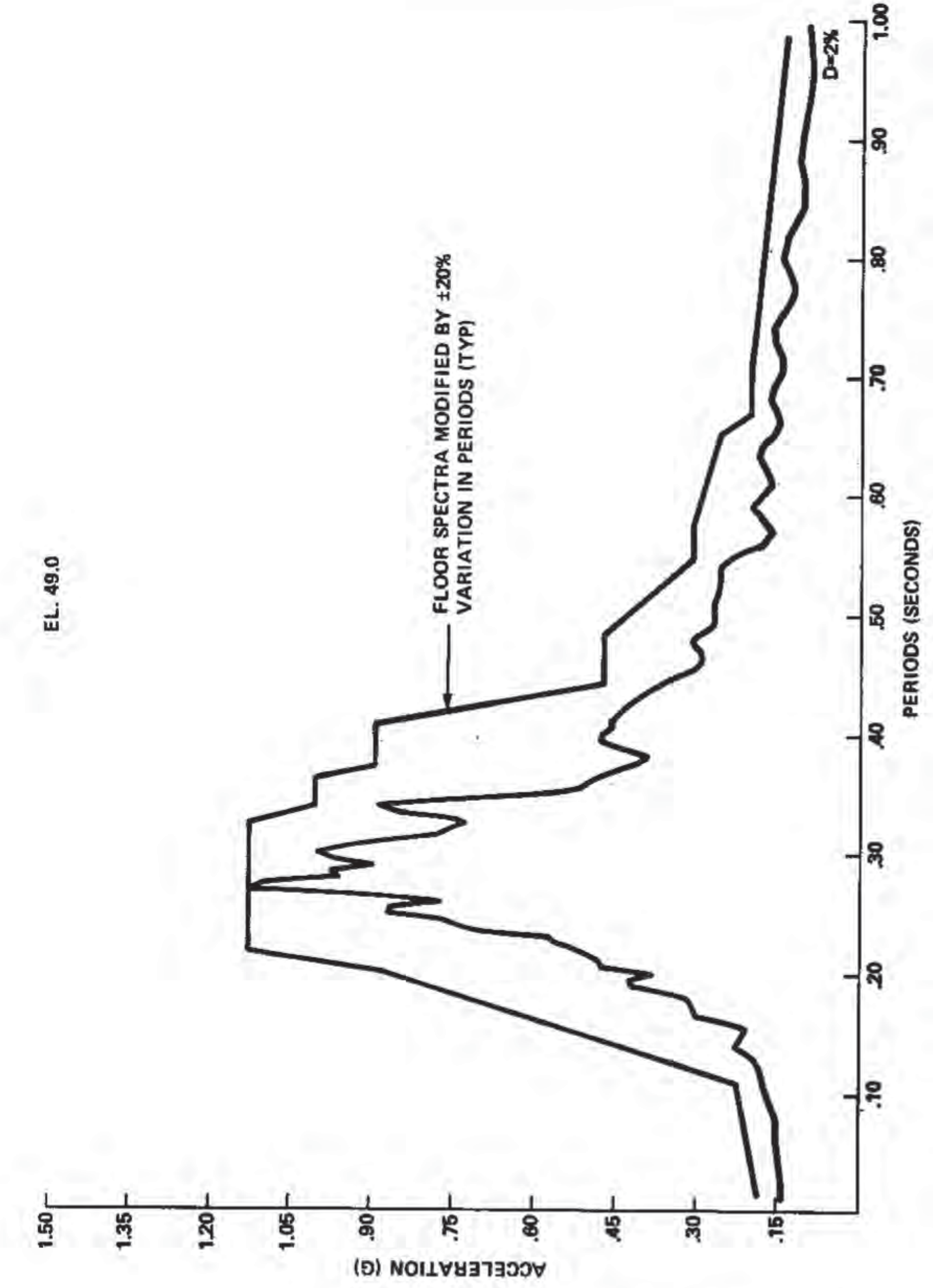
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
DIESEL GENERATOR BLDG. CANT. 3
FIGURE 3.7-184



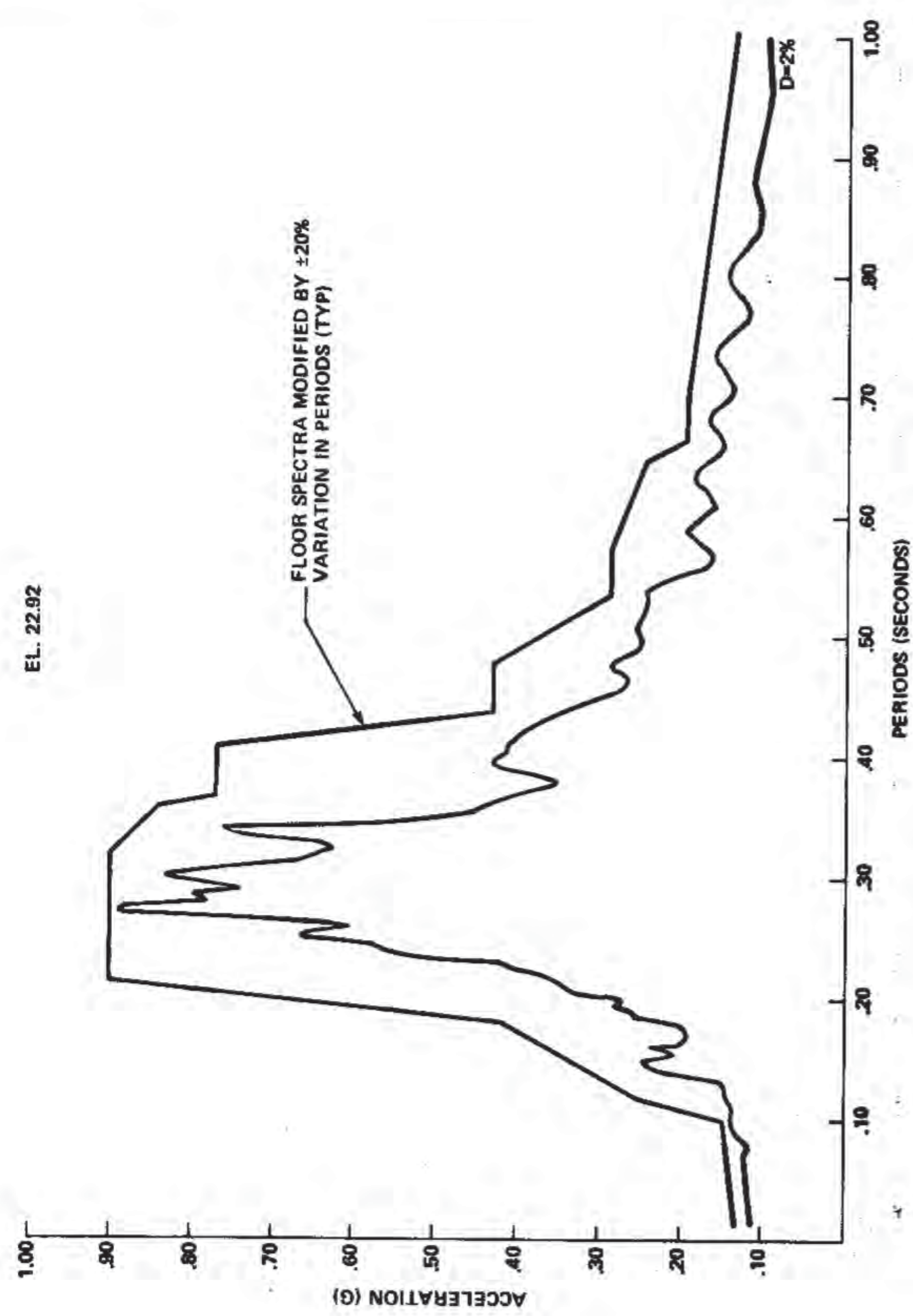
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
DIESEL GENERATOR BLDG.
FIGURE 3.7-185



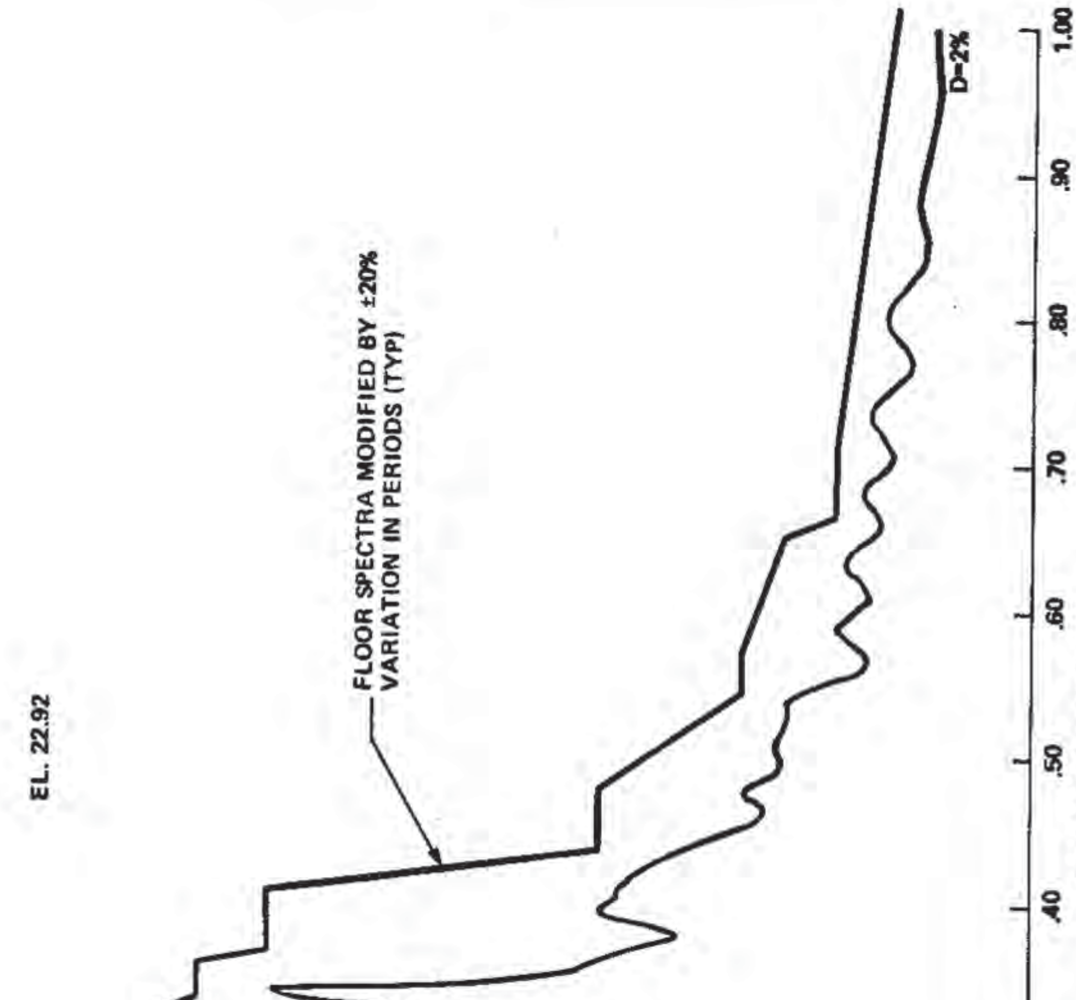
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7–186



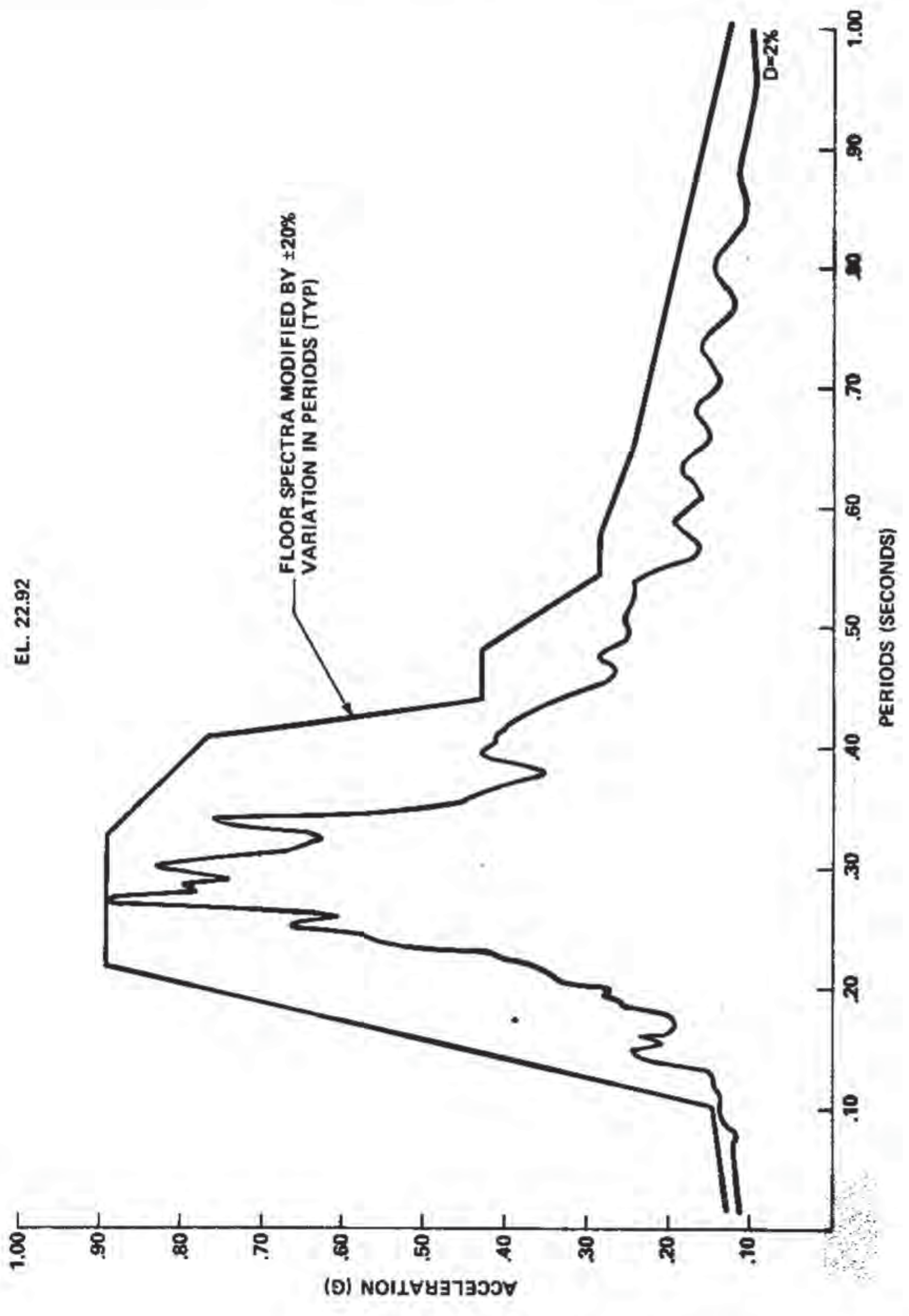
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7–187



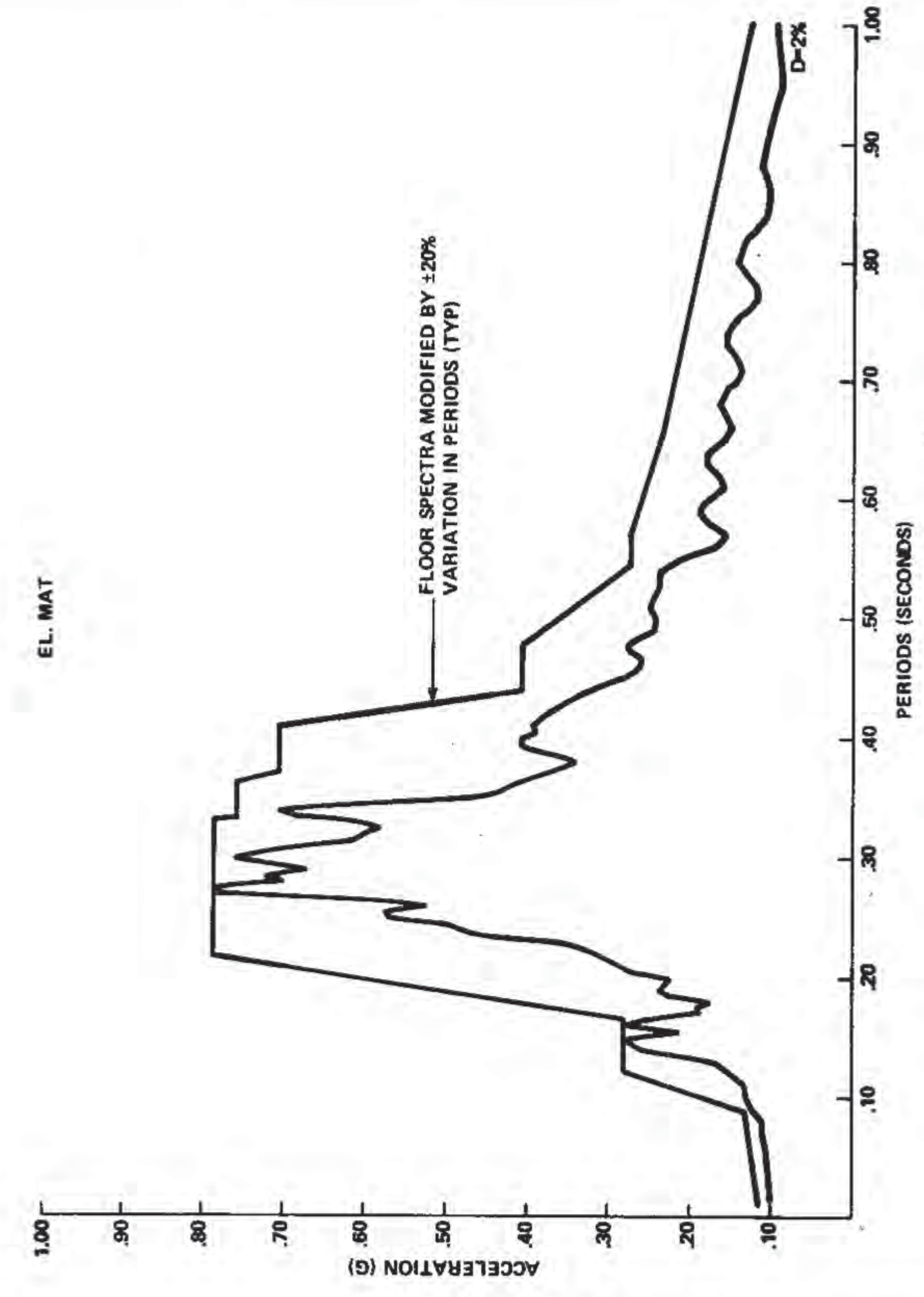
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
DIESEL GENERATOR BLDG. CANT. 2
FIGURE 3.7–188



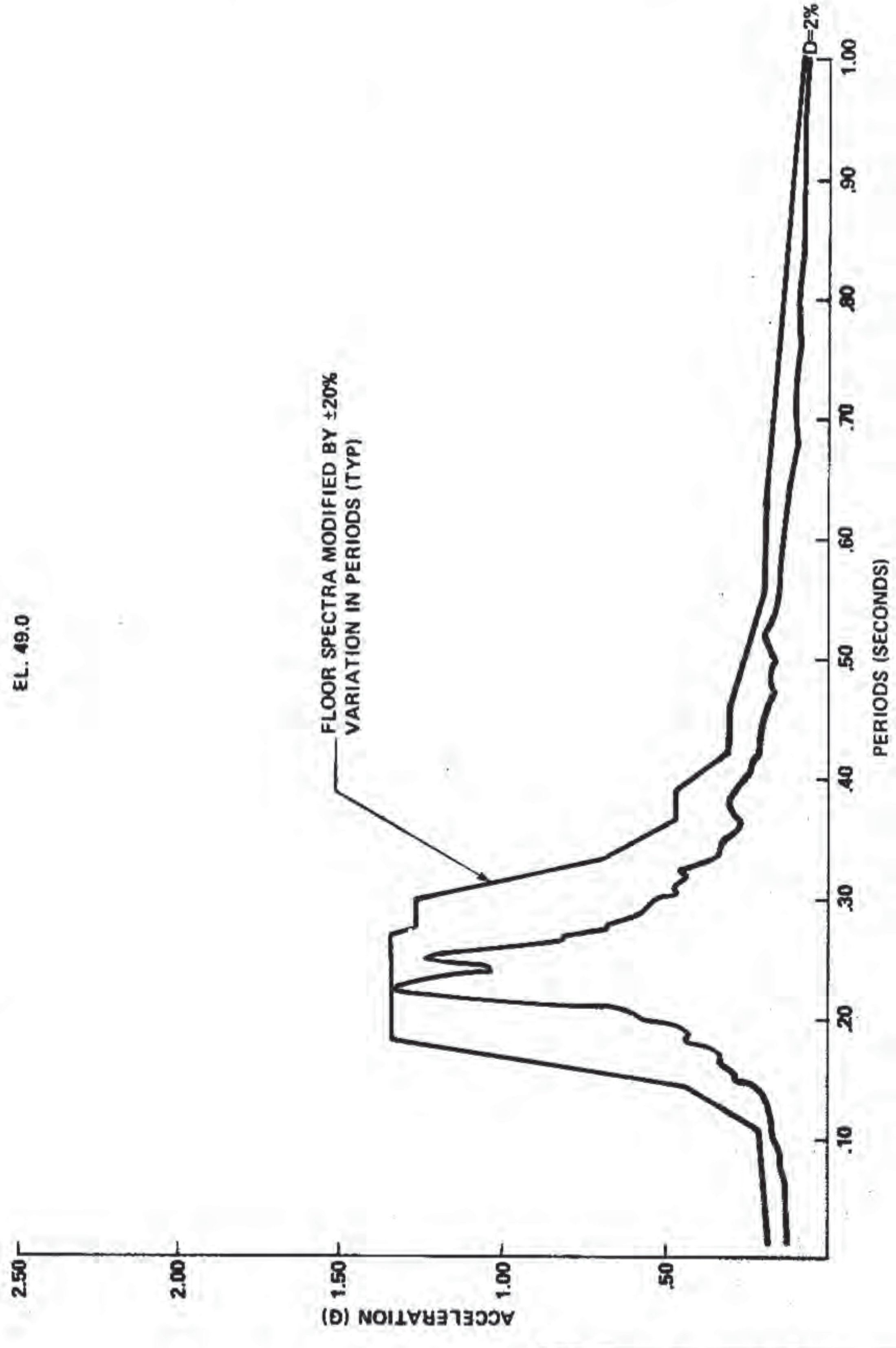
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
DIESEL GENERATOR BLDG. CANT. 3
FIGURE 3.7–189



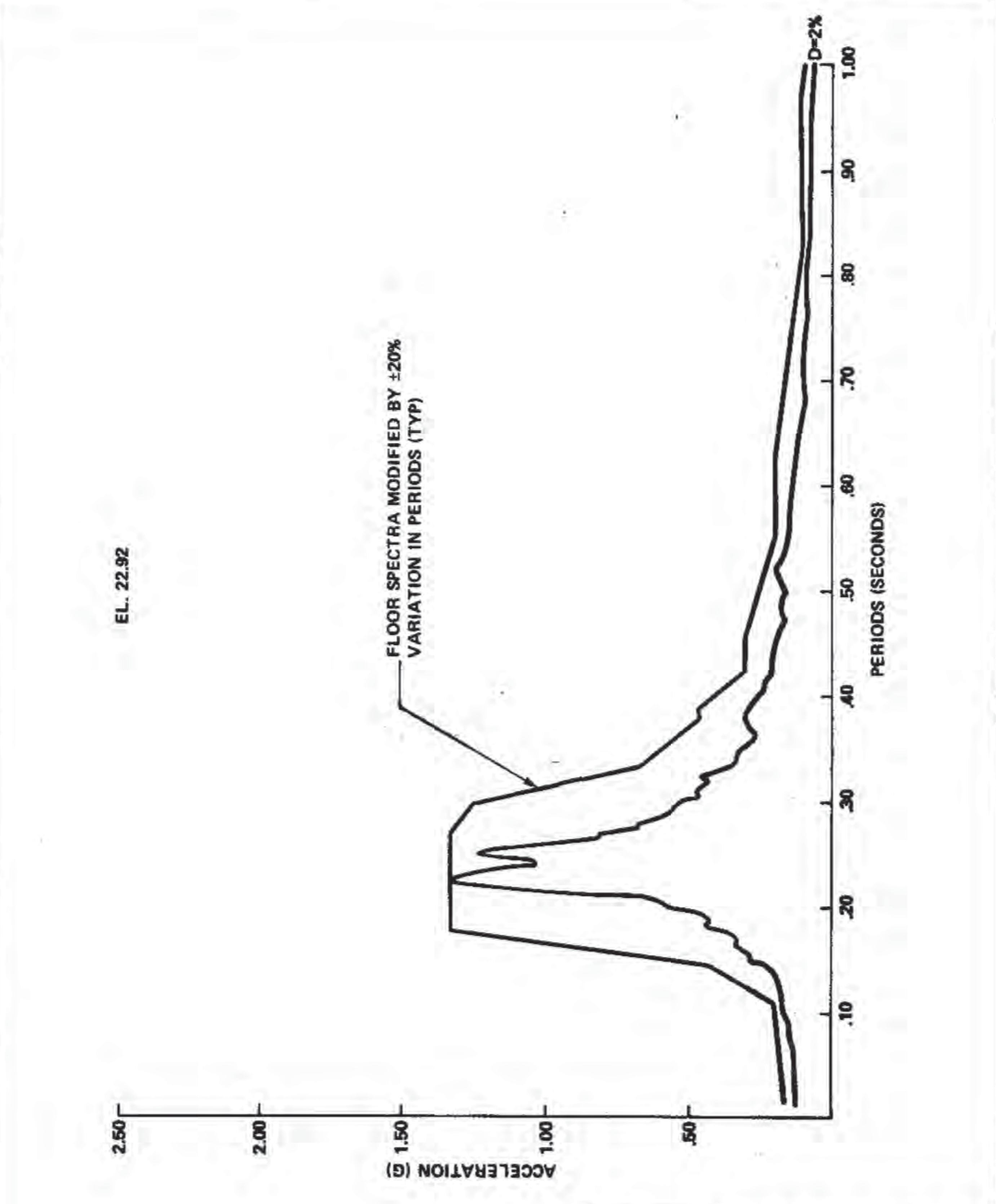
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
DIESEL GENERATOR BLDG.
FIGURE 3.7–190



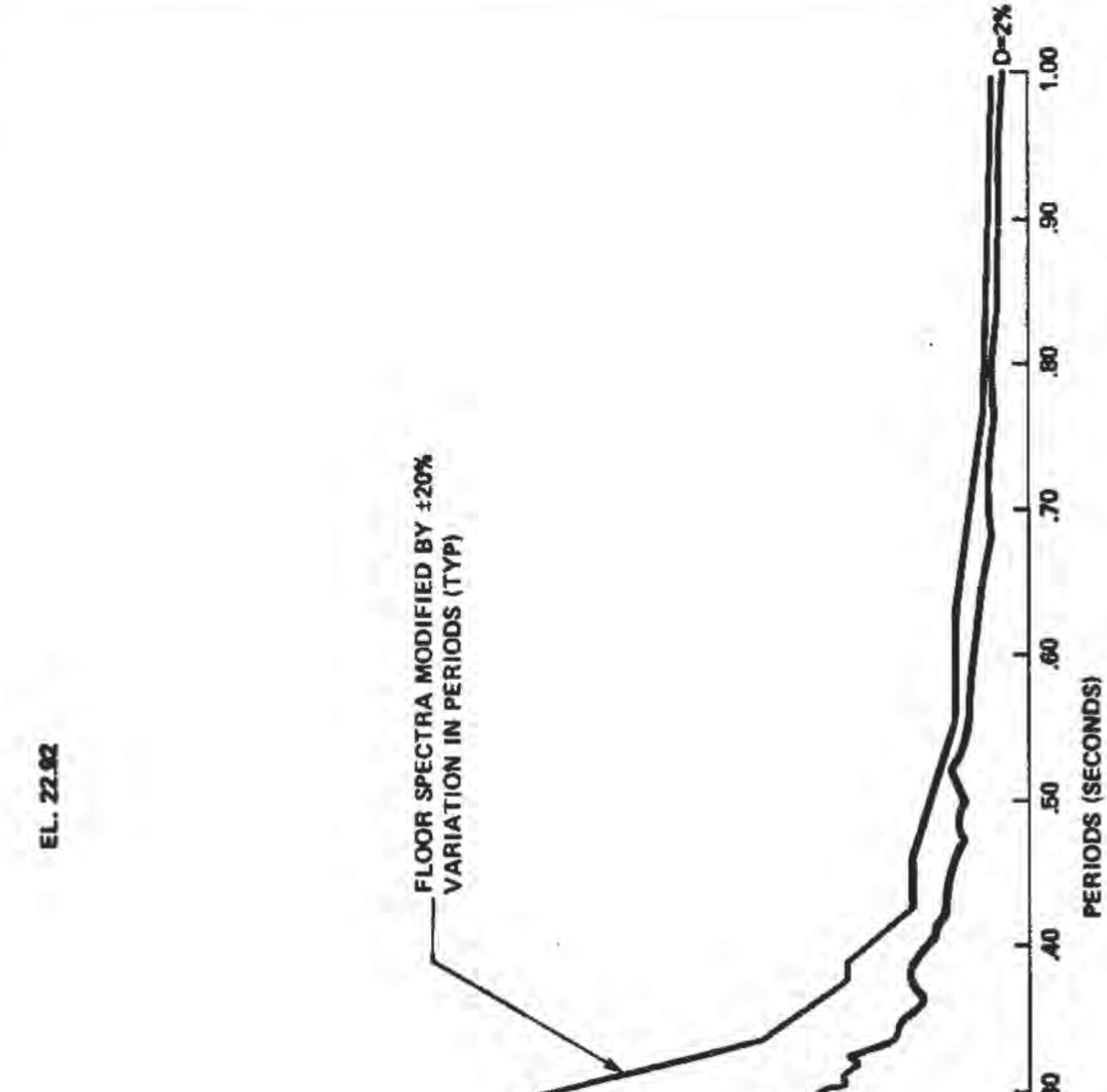
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7–191



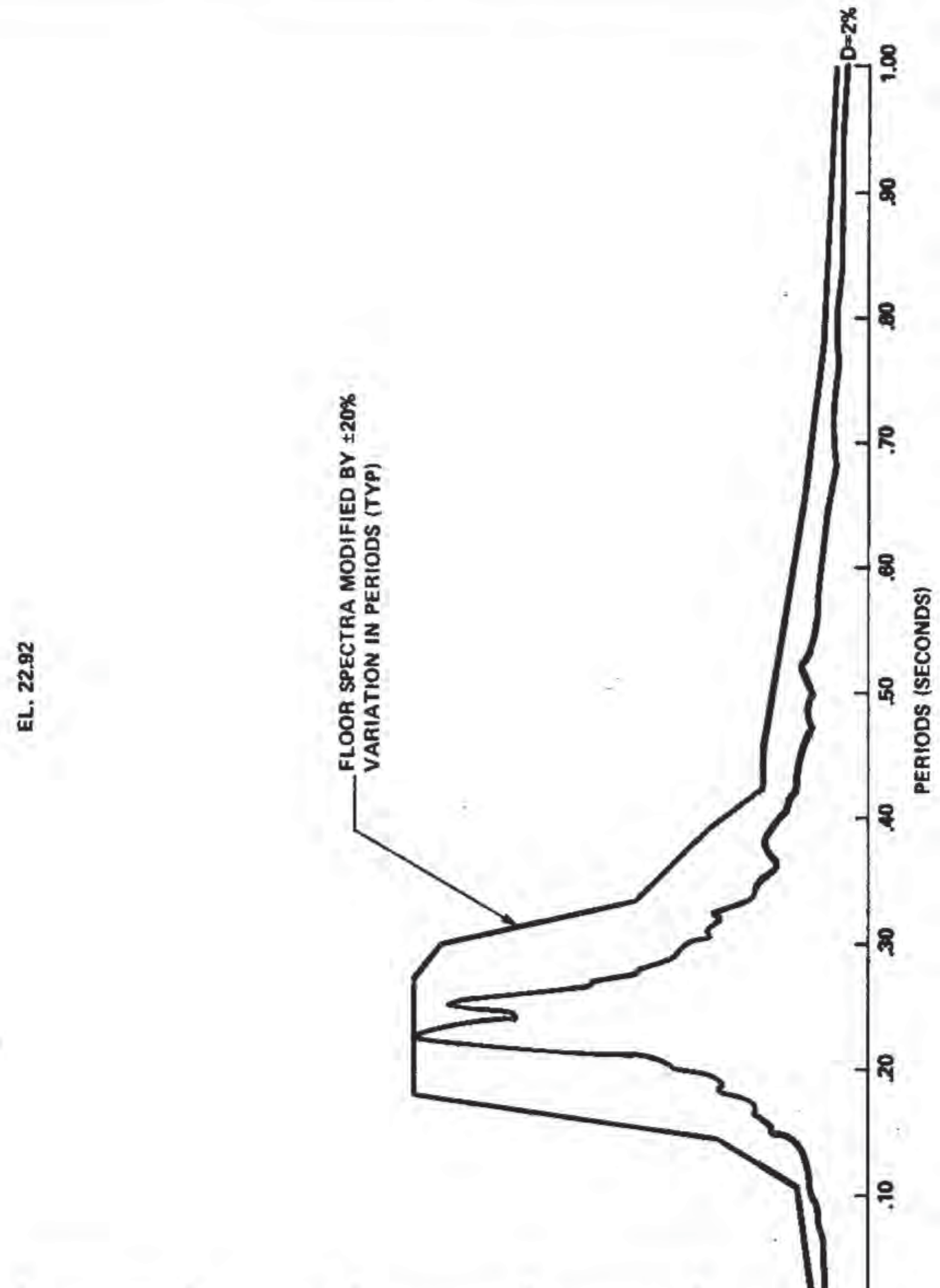
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
DIESEL GENERATOR BLDG. CANT. 1
FIGURE 3.7–192



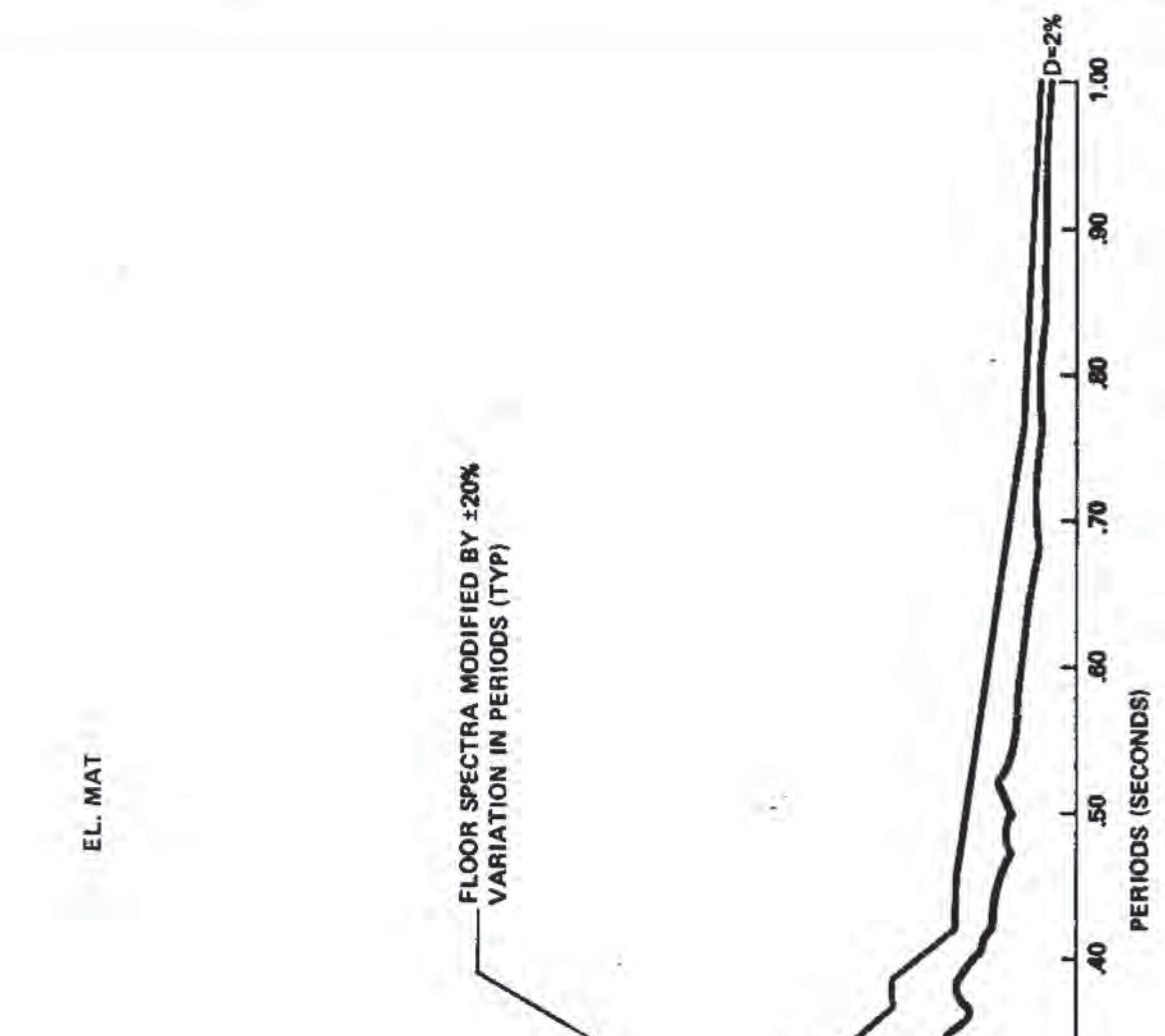
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
DIESEL GENERATOR BLDG. CANT. 2
FIGURE 3.7-193



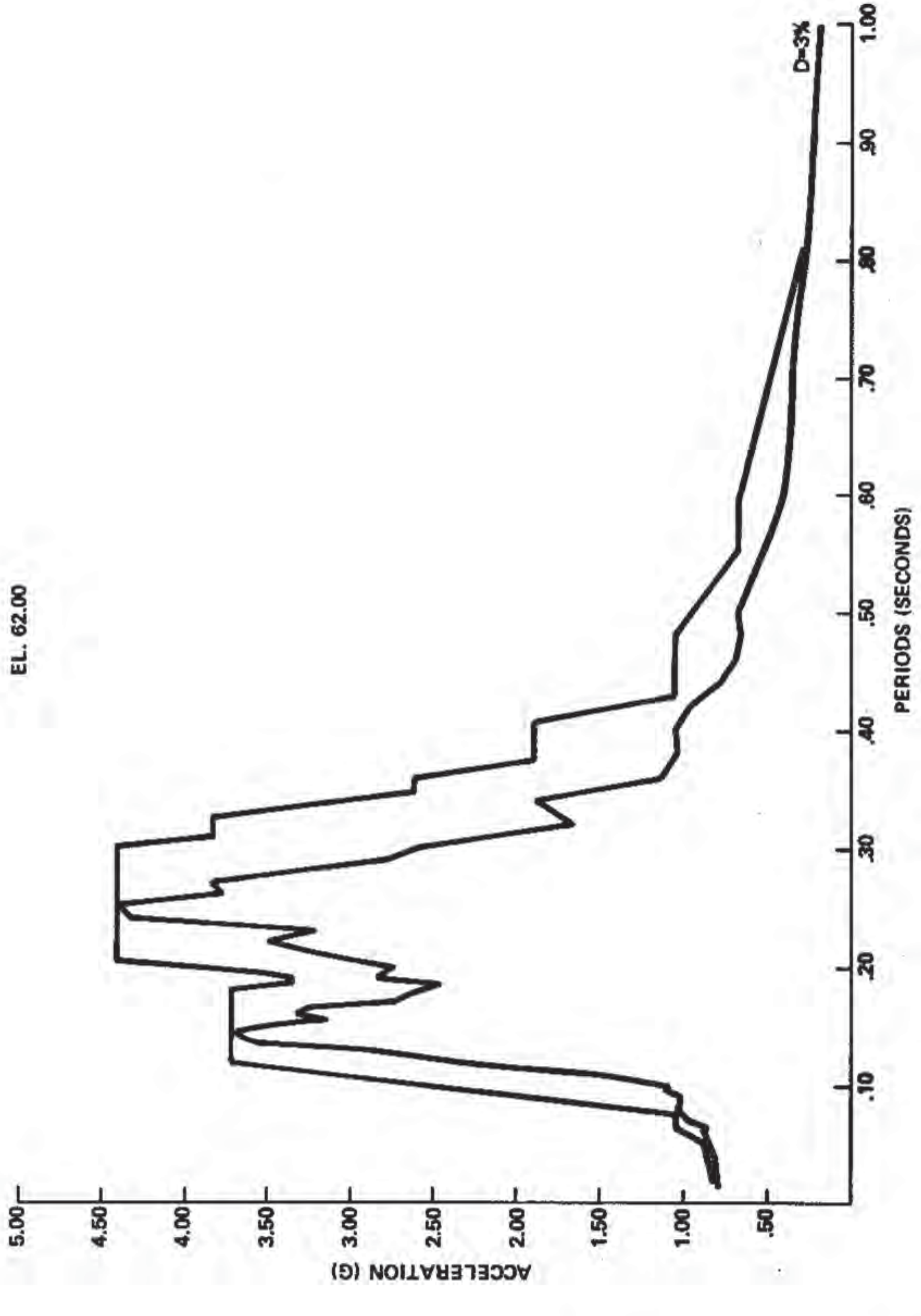
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
DIESEL GENERATOR BLDG. CANT. 3
FIGURE 3.7-194



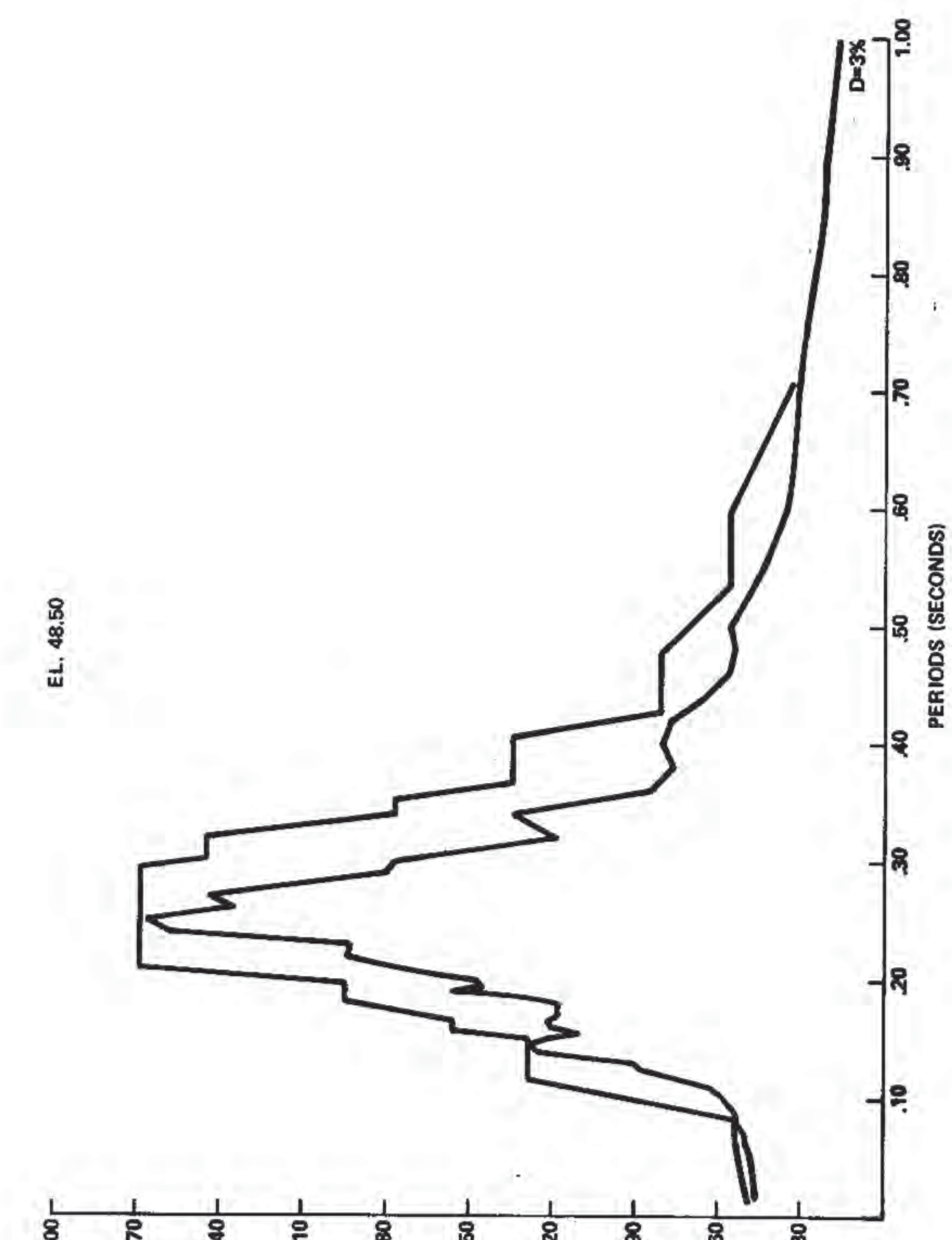
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
DIESEL GENERATOR BLDG.
FIGURE 3.7–195



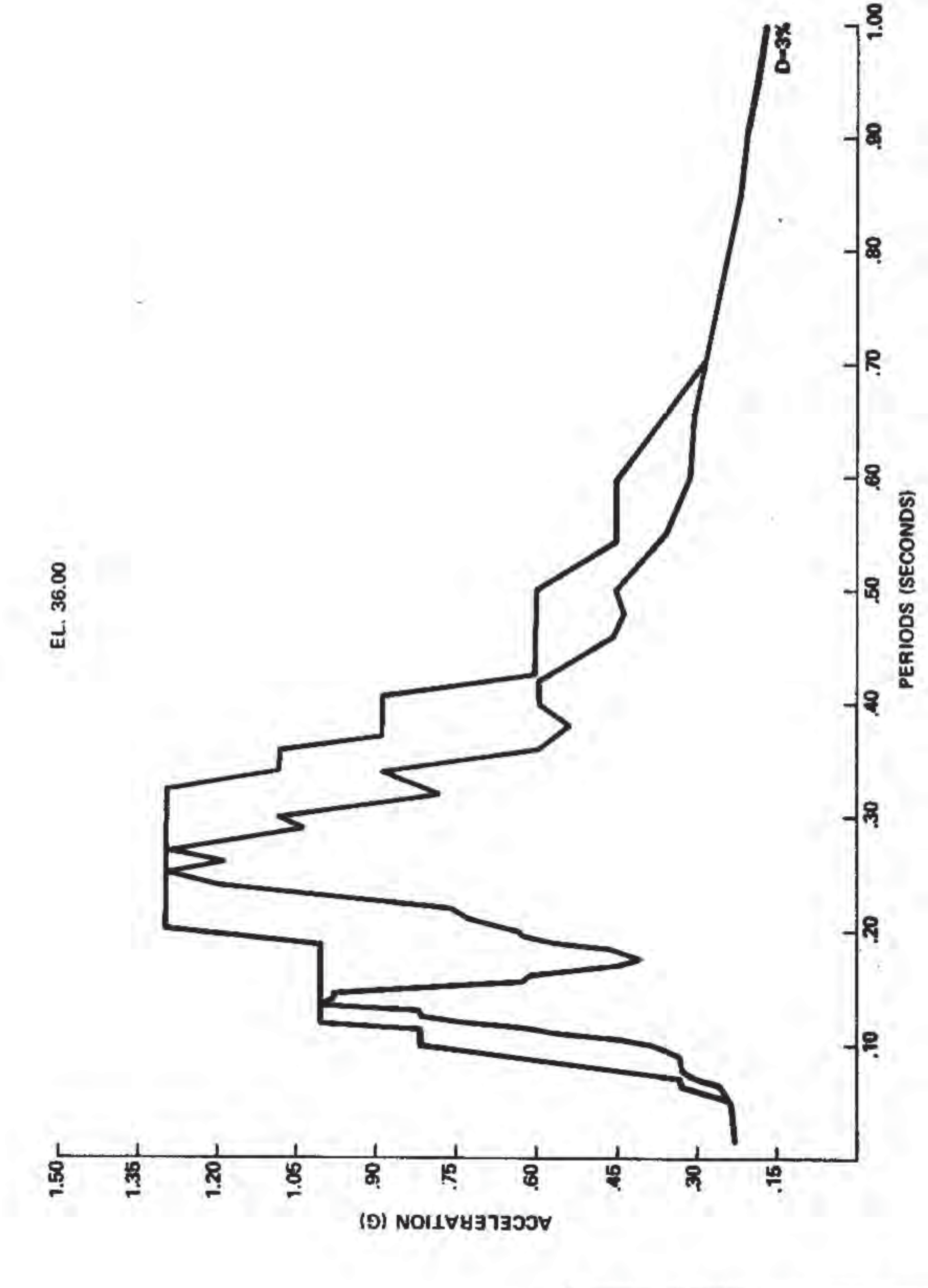
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
MS/FW TRESTLE
FIGURE 3.7-196



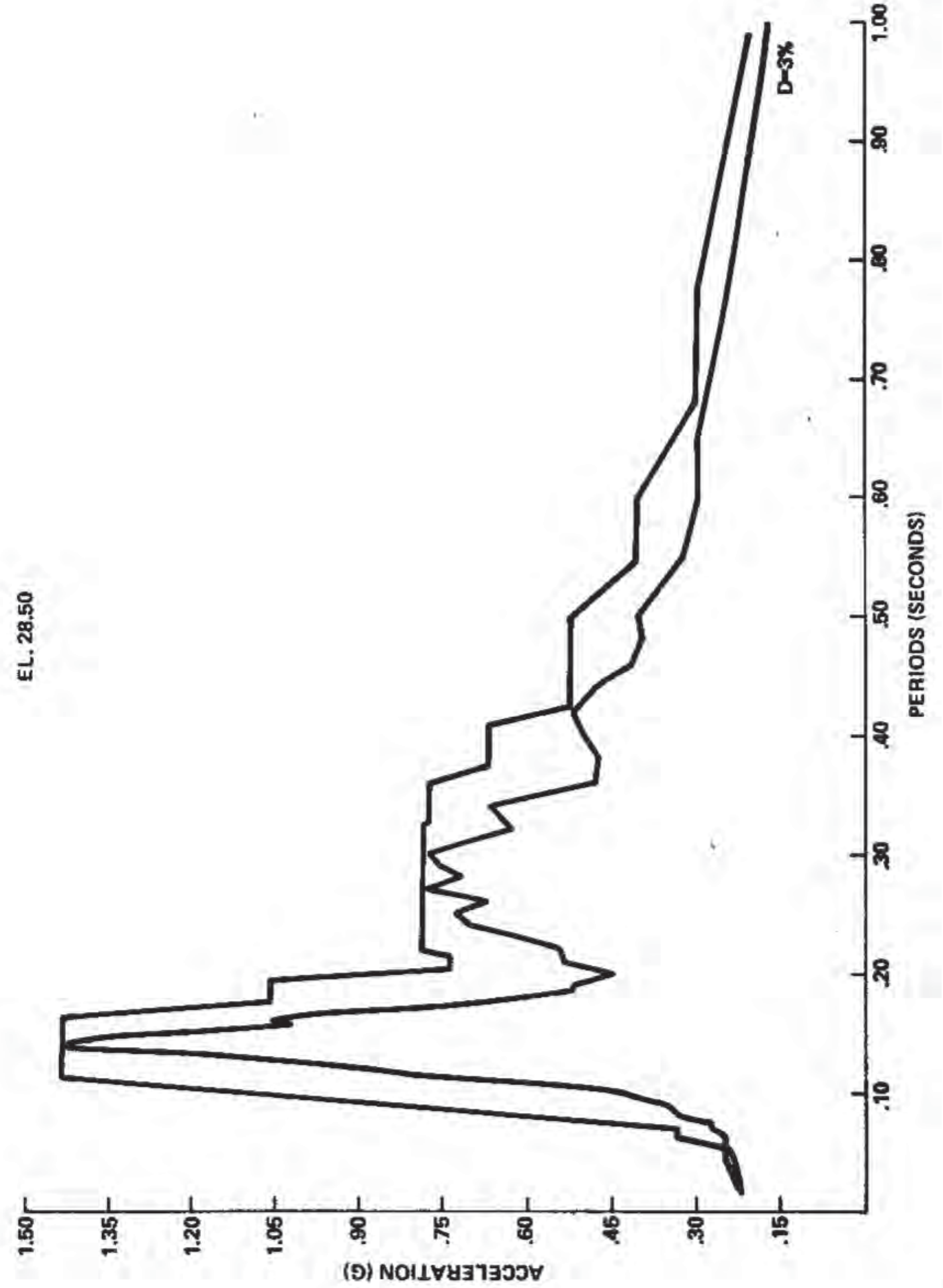
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
MS/FW TRESTLE
FIGURE 3.7-197



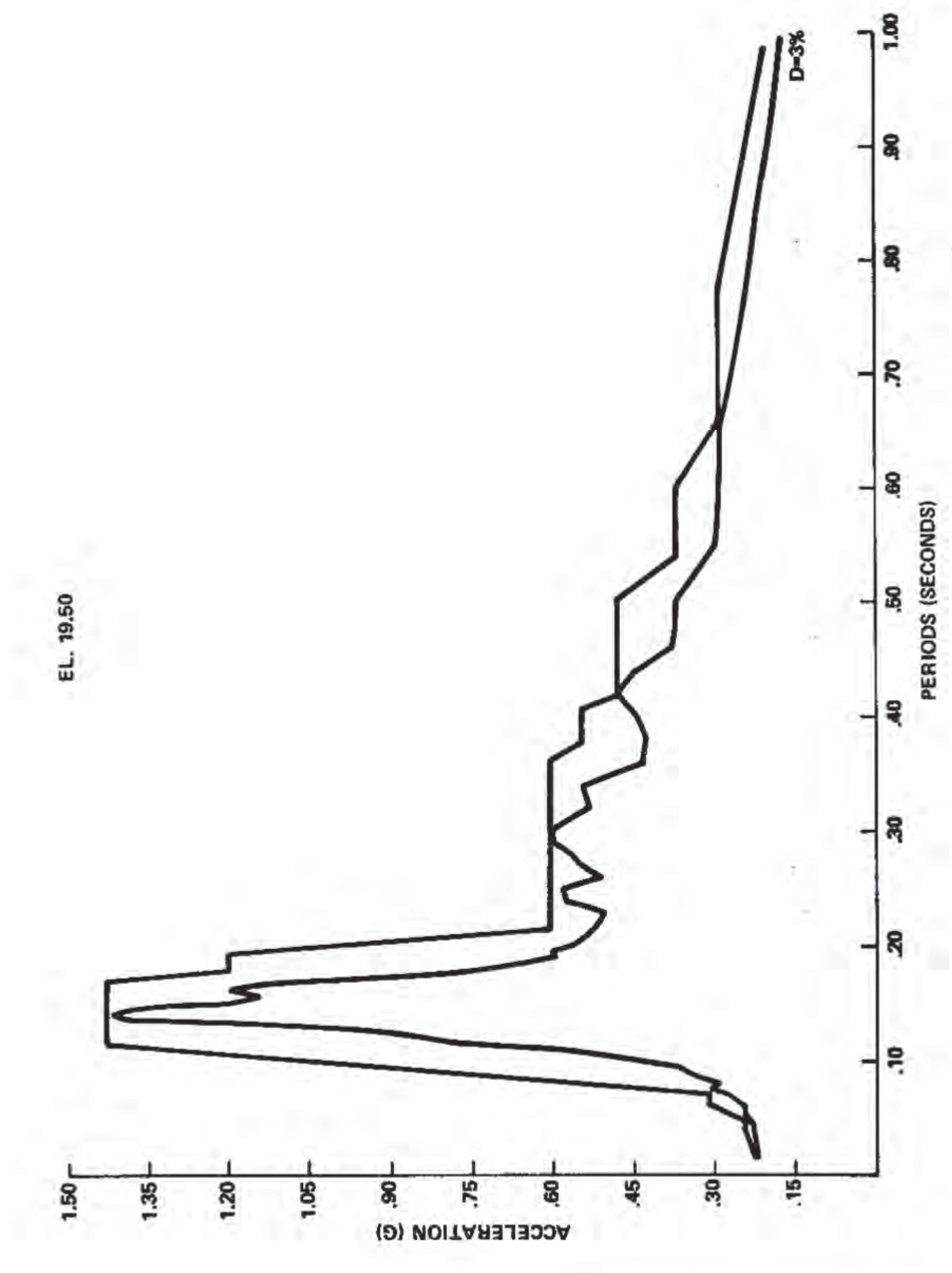
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
MS/FW TRESTLE
FIGURE 3.7-198



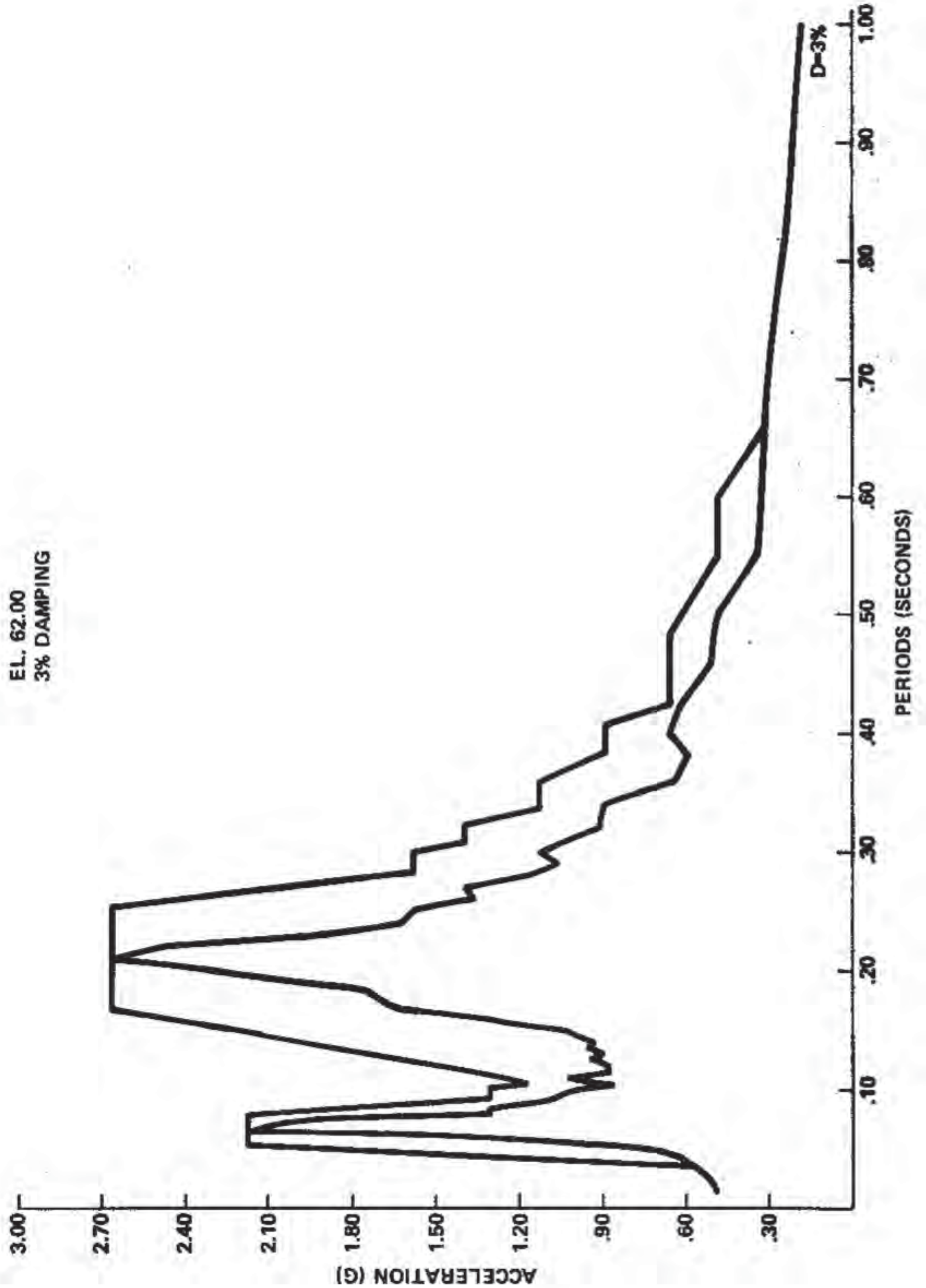
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
MS/FW TRESTLE
FIGURE 3.7-199



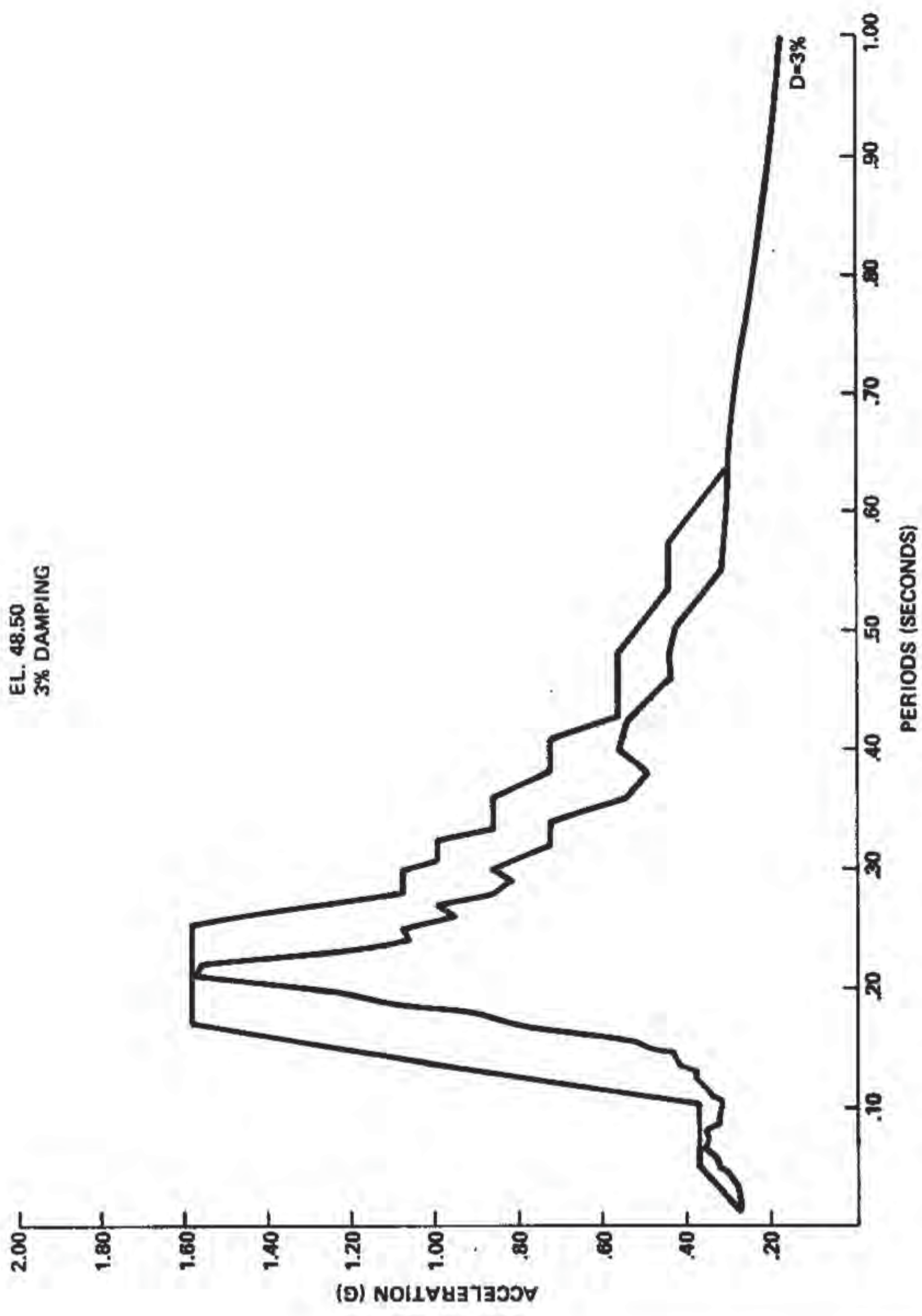
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
MS/FW TRESTLE
FIGURE 3.7-200



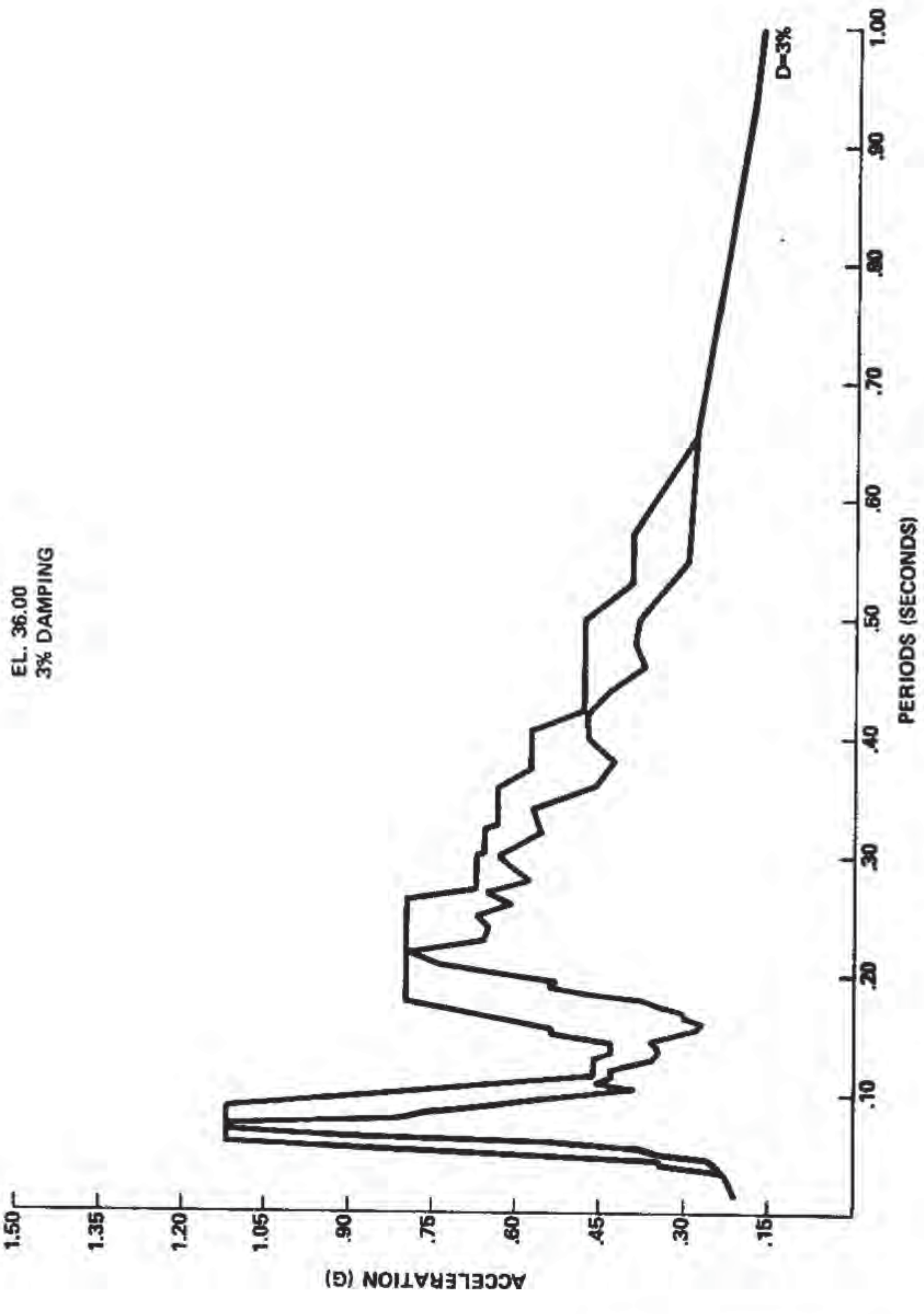
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
MS/FW TRESTLE
FIGURE 3.7-201



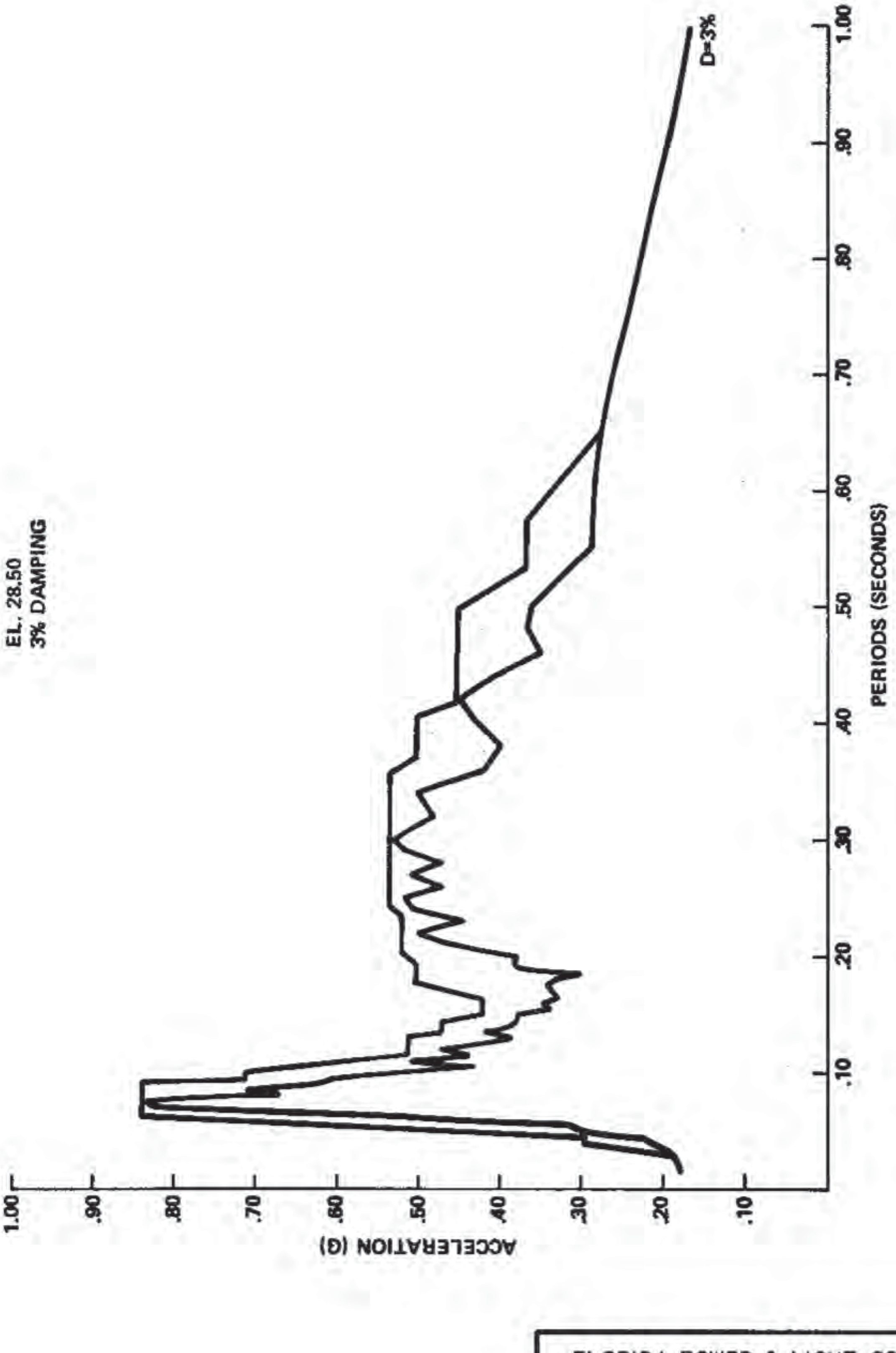
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
MS/FW TRESTLE
FIGURE 3.7-202



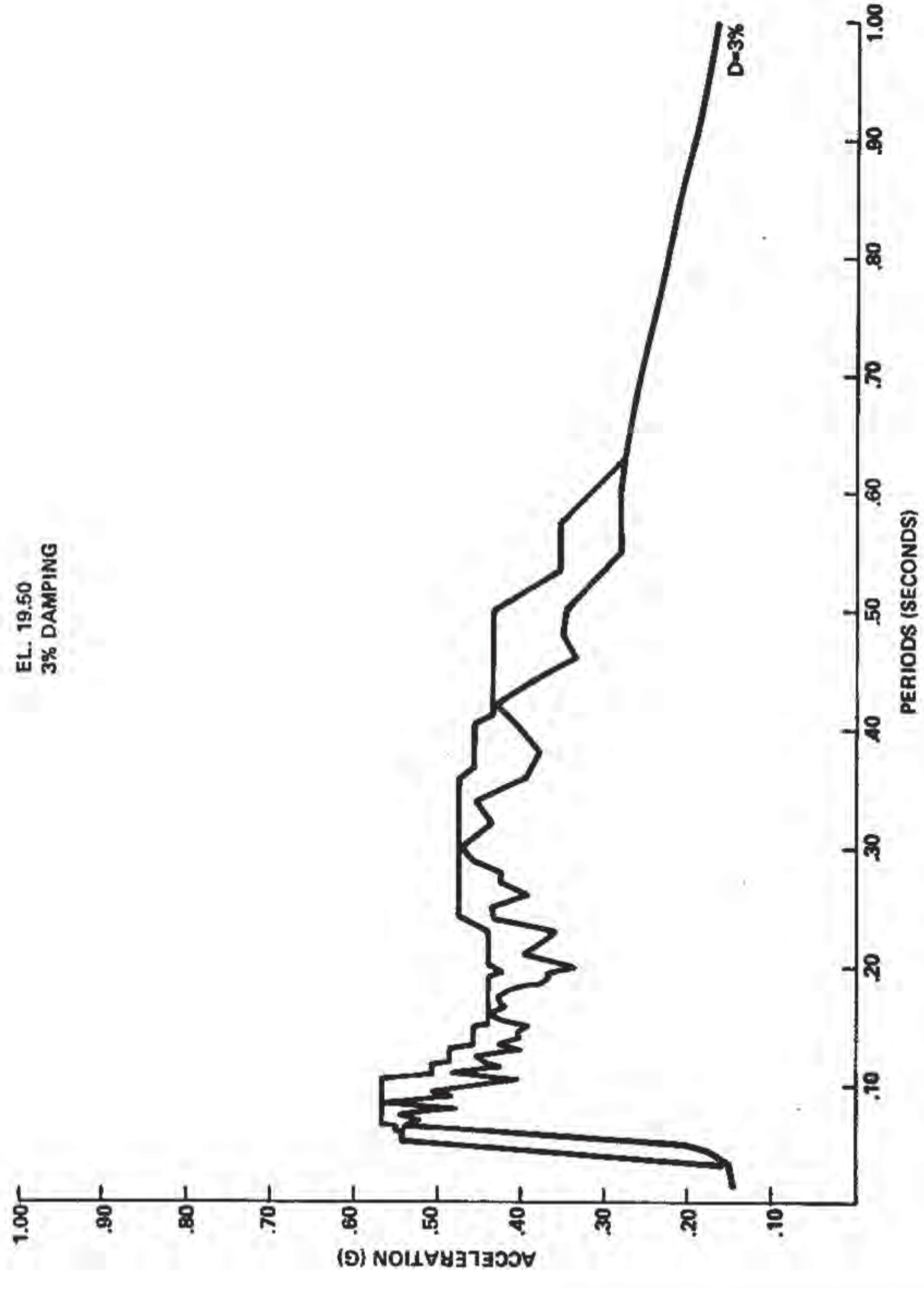
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
MS/FW TRESTLE
FIGURE 3.7-203



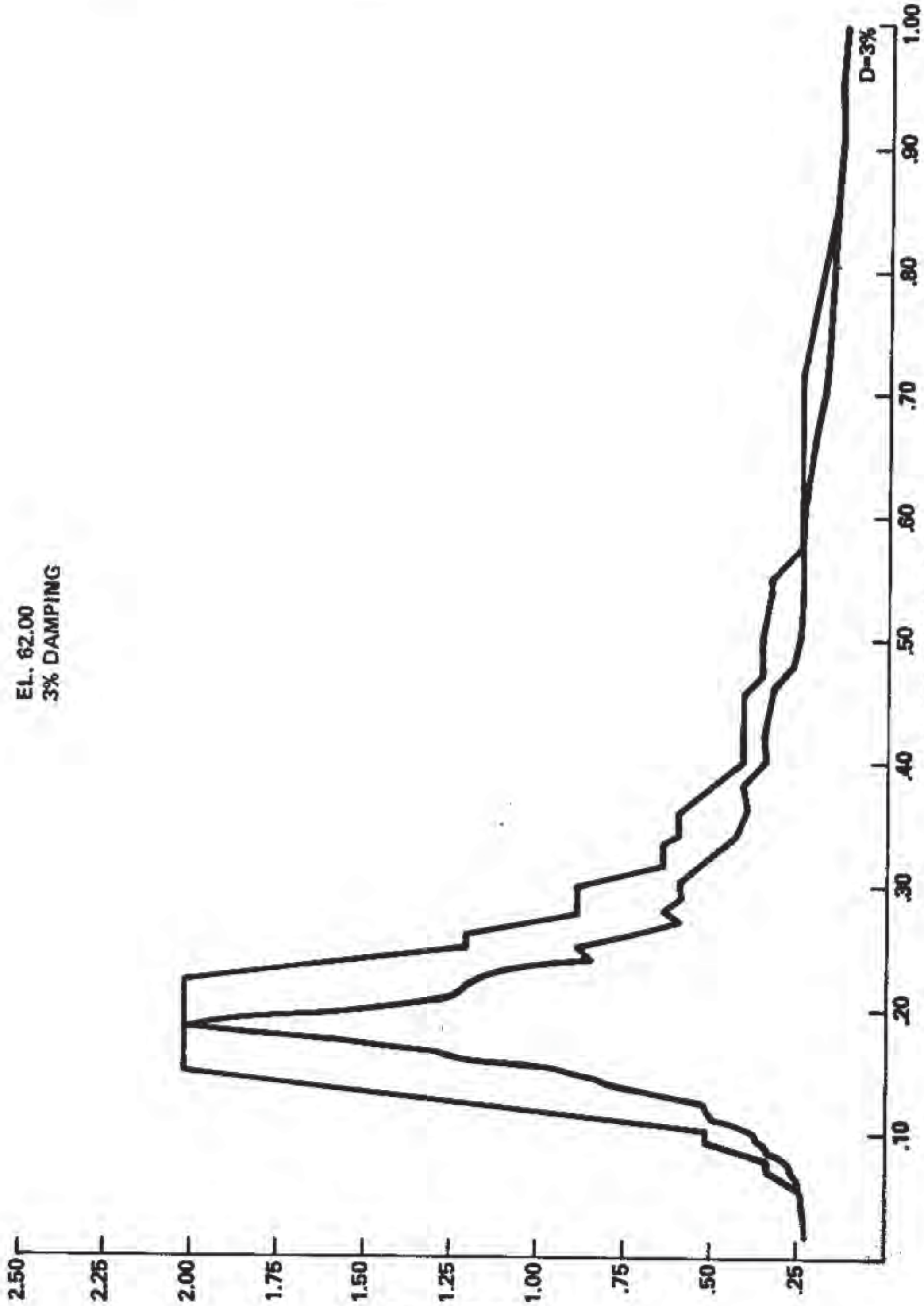
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
MS/FW TRESTLE
FIGURE 3.7-204



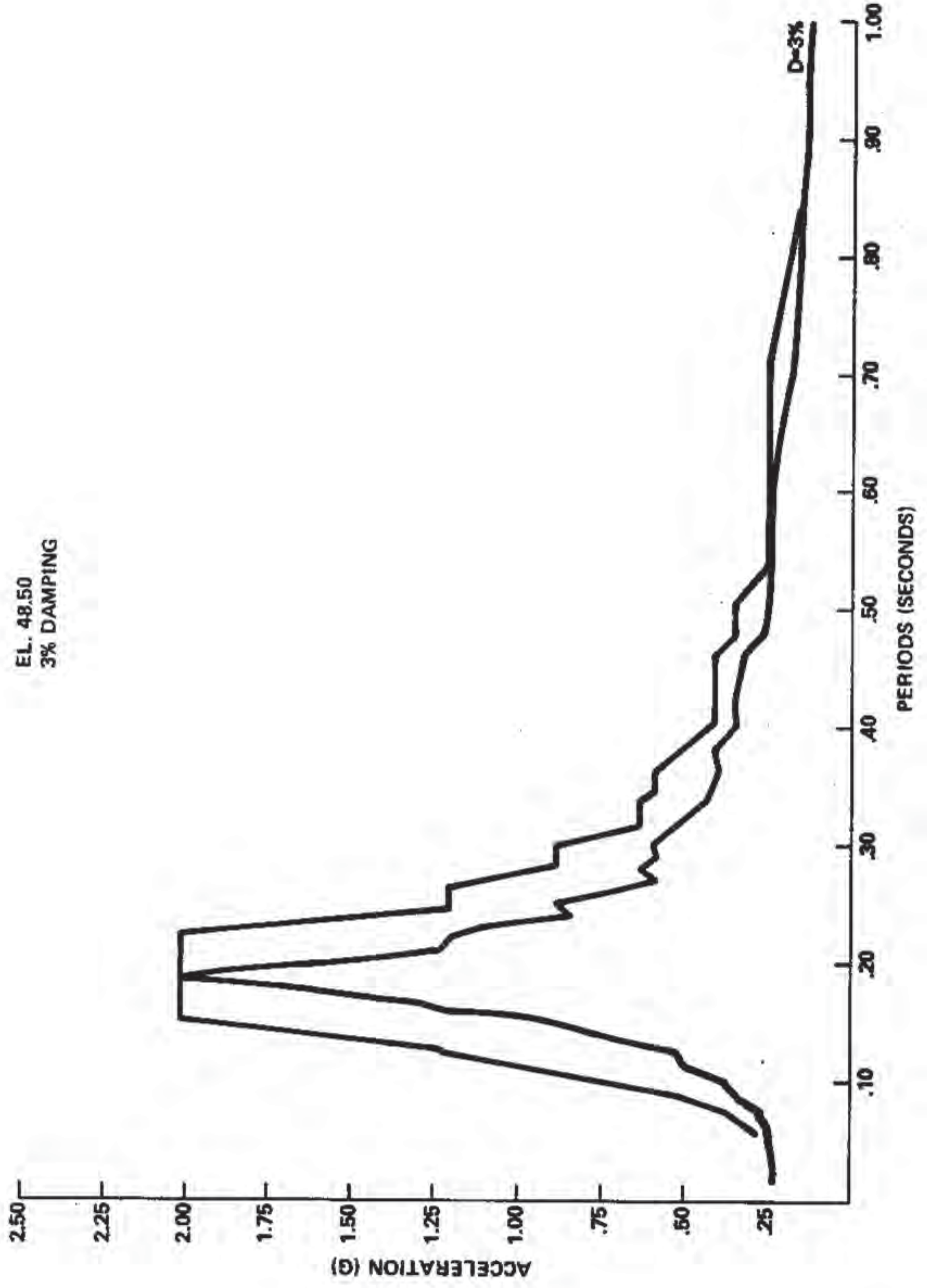
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
MS/FW TRESTLE
FIGURE 3.7-205



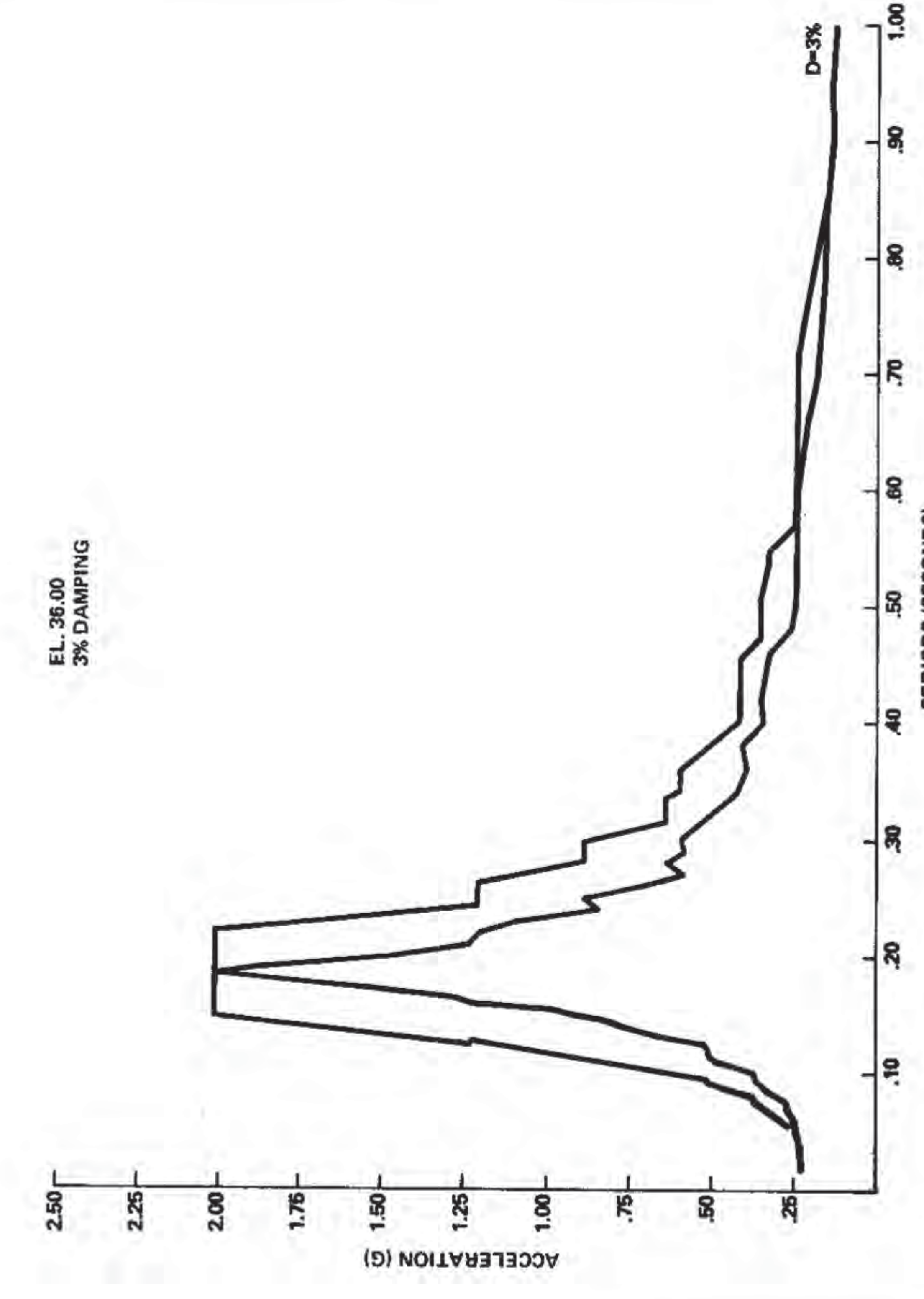
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
MS/FW TRESTLE
FIGURE 3.7-206



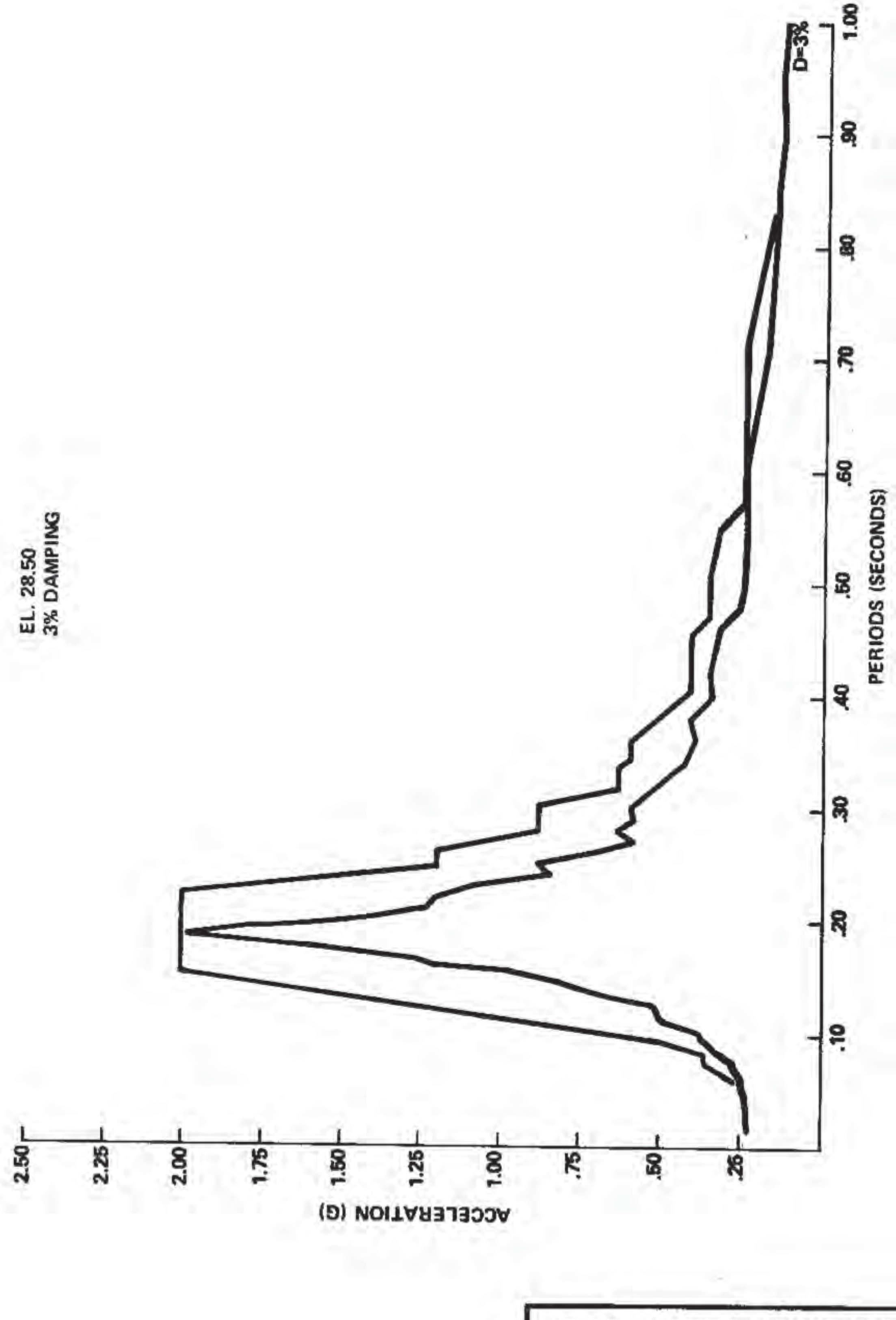
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
MS/FW TRESTLE
FIGURE 3.7-207



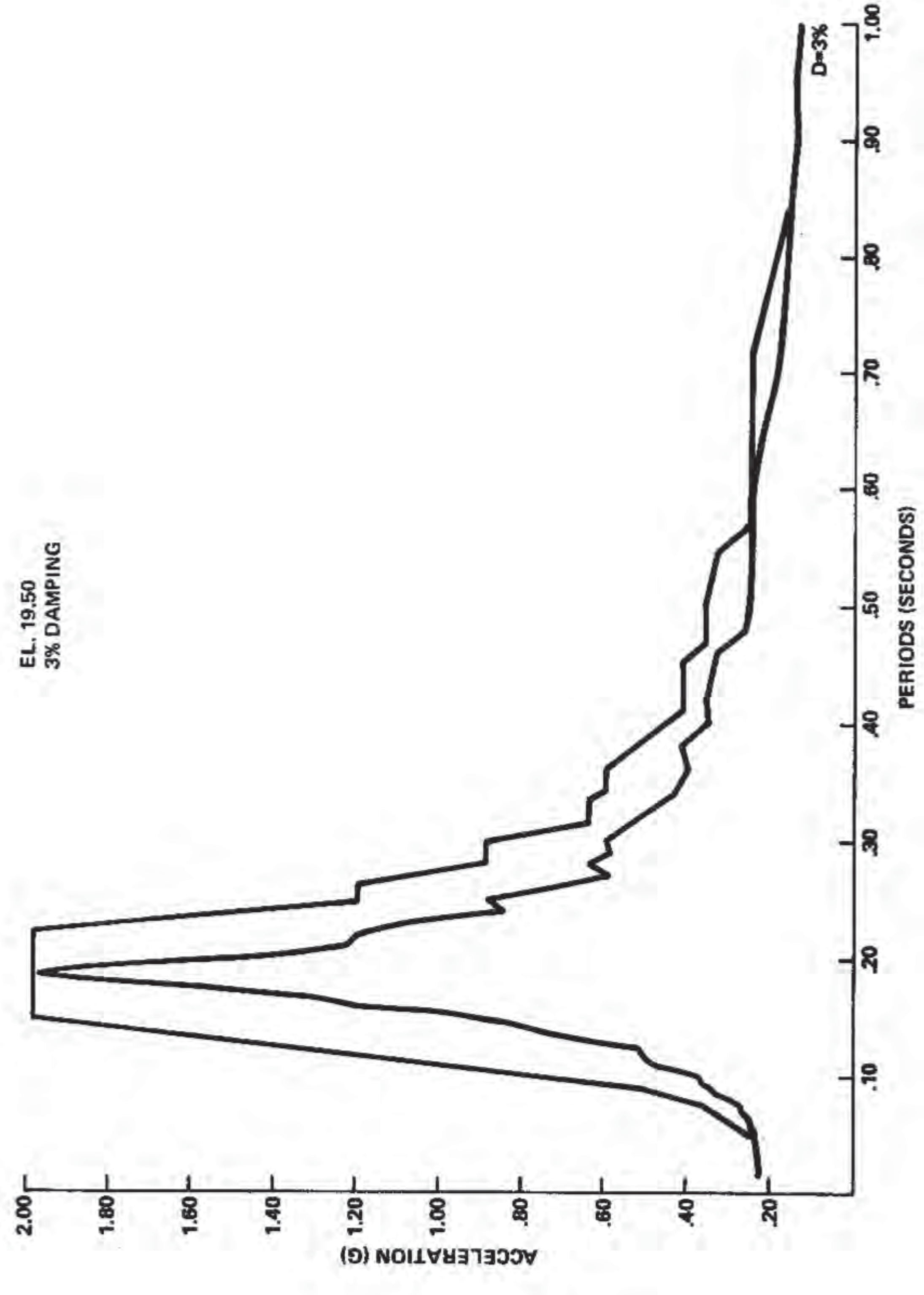
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
MS/FW TRESTLE
FIGURE 3.7-208



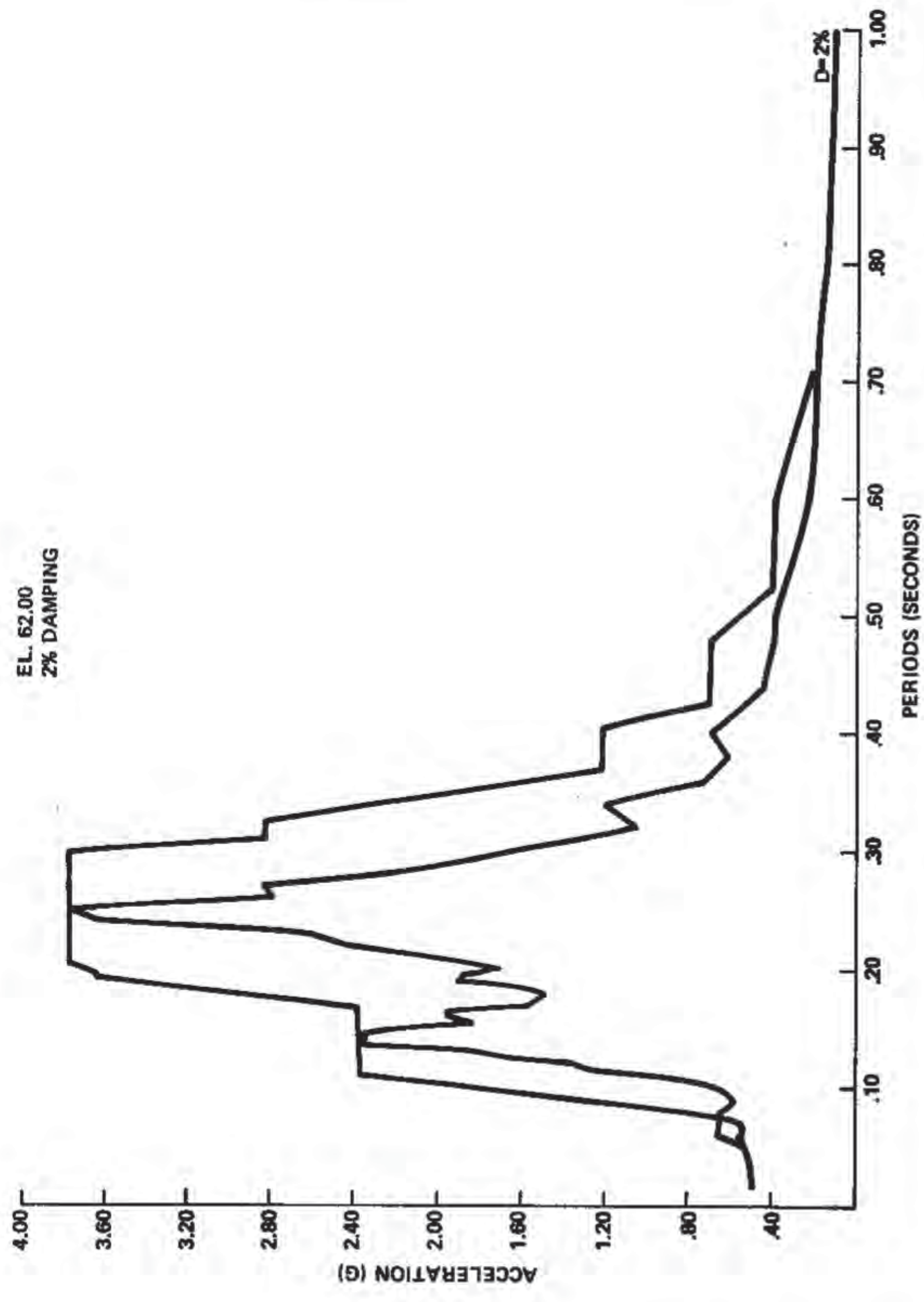
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
MS/FW TRESTLE
FIGURE 3.7-209



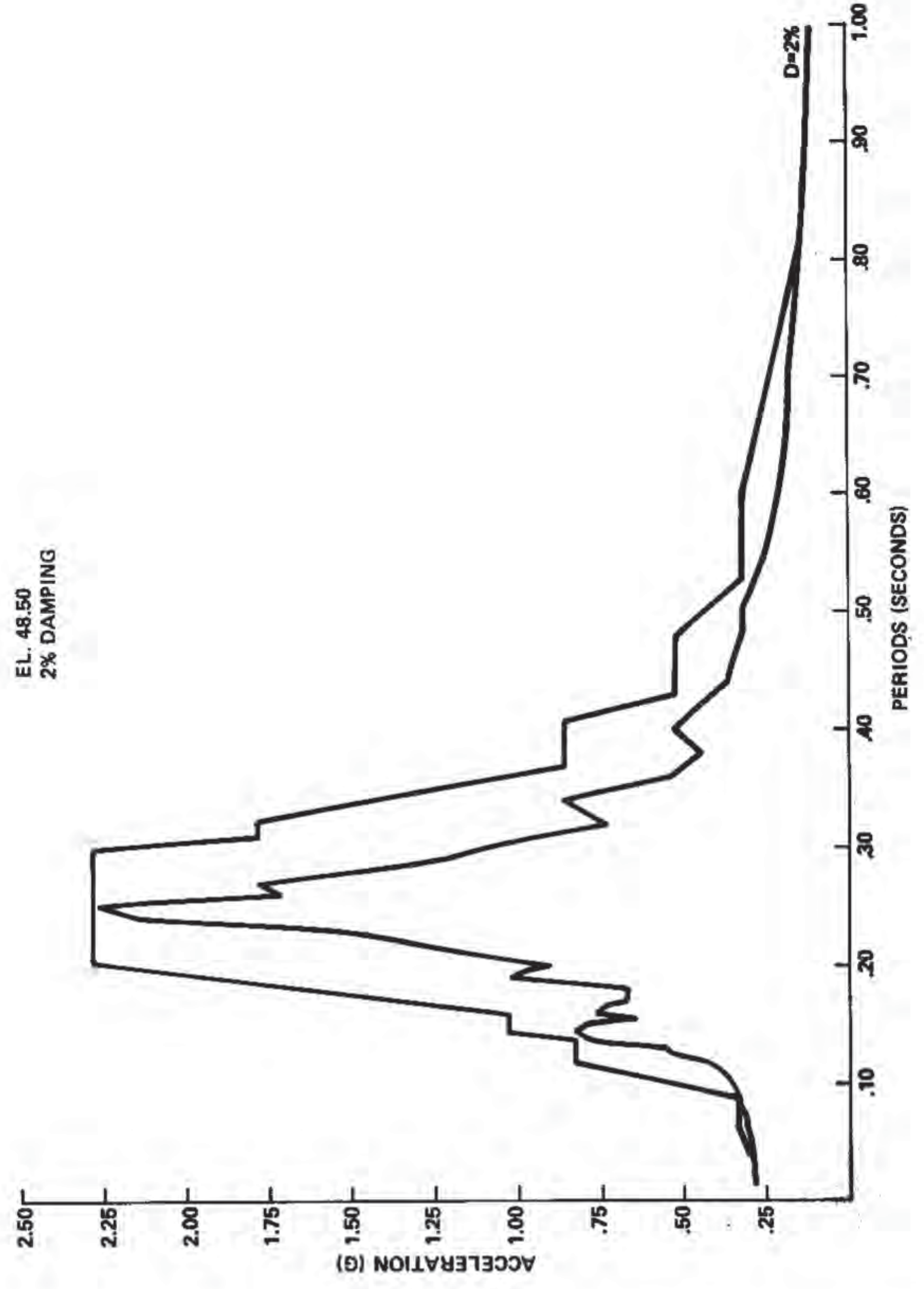
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
MS/FW TRESTLE
FIGURE 3.7-210



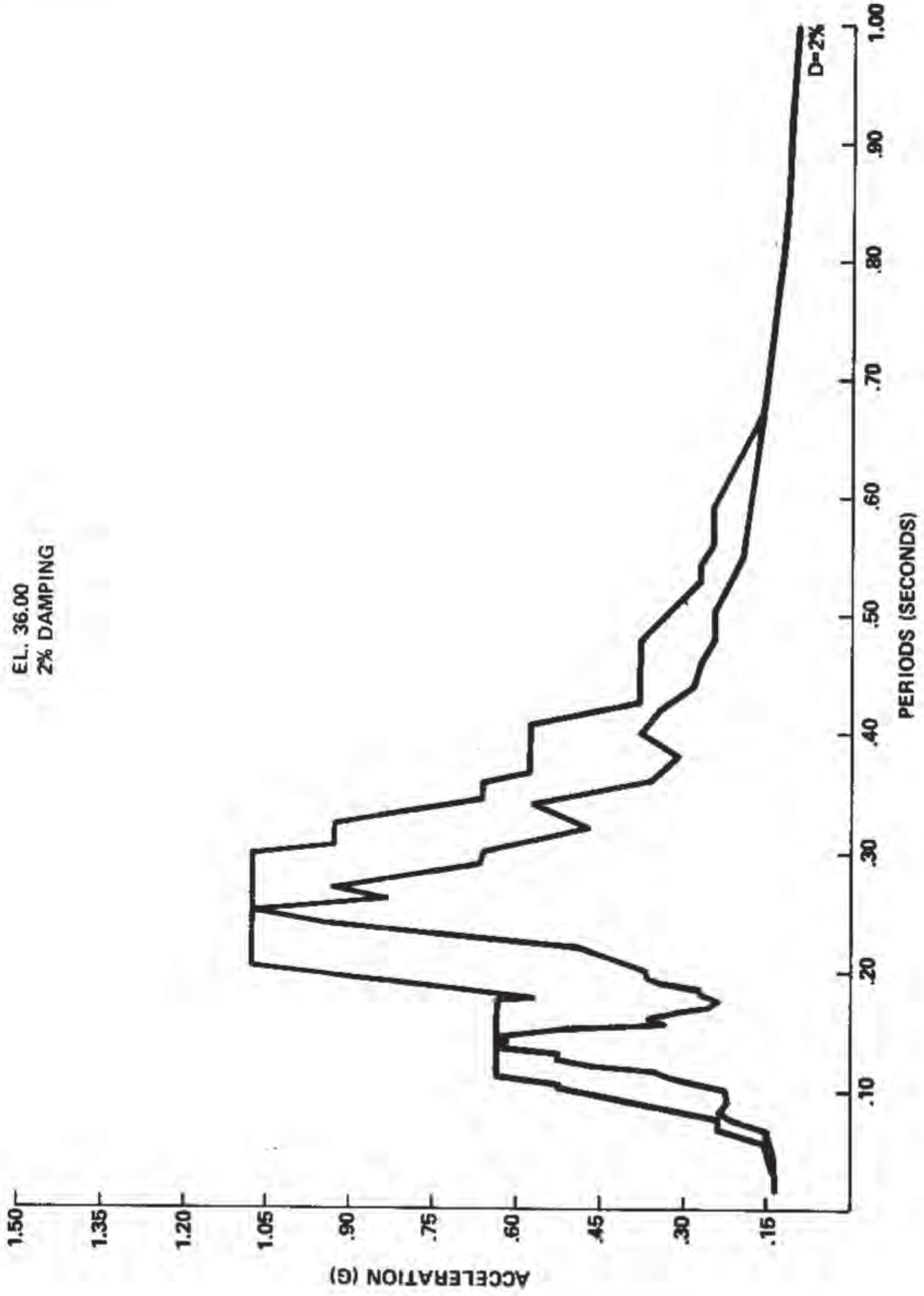
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
MS/FW TRESTLE
FIGURE 3.7-211



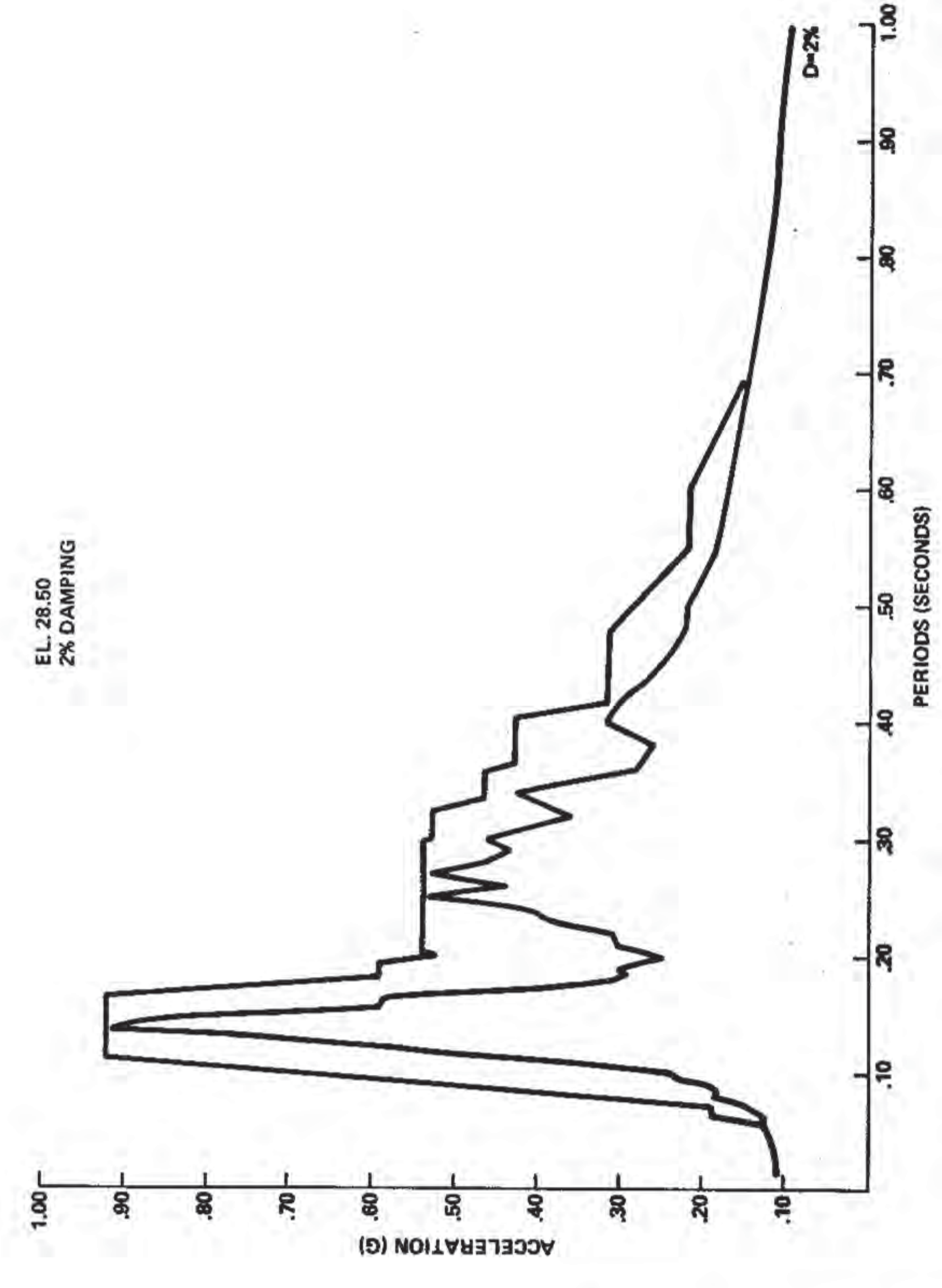
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
MS/FW TRESTLE
FIGURE 3.7-212



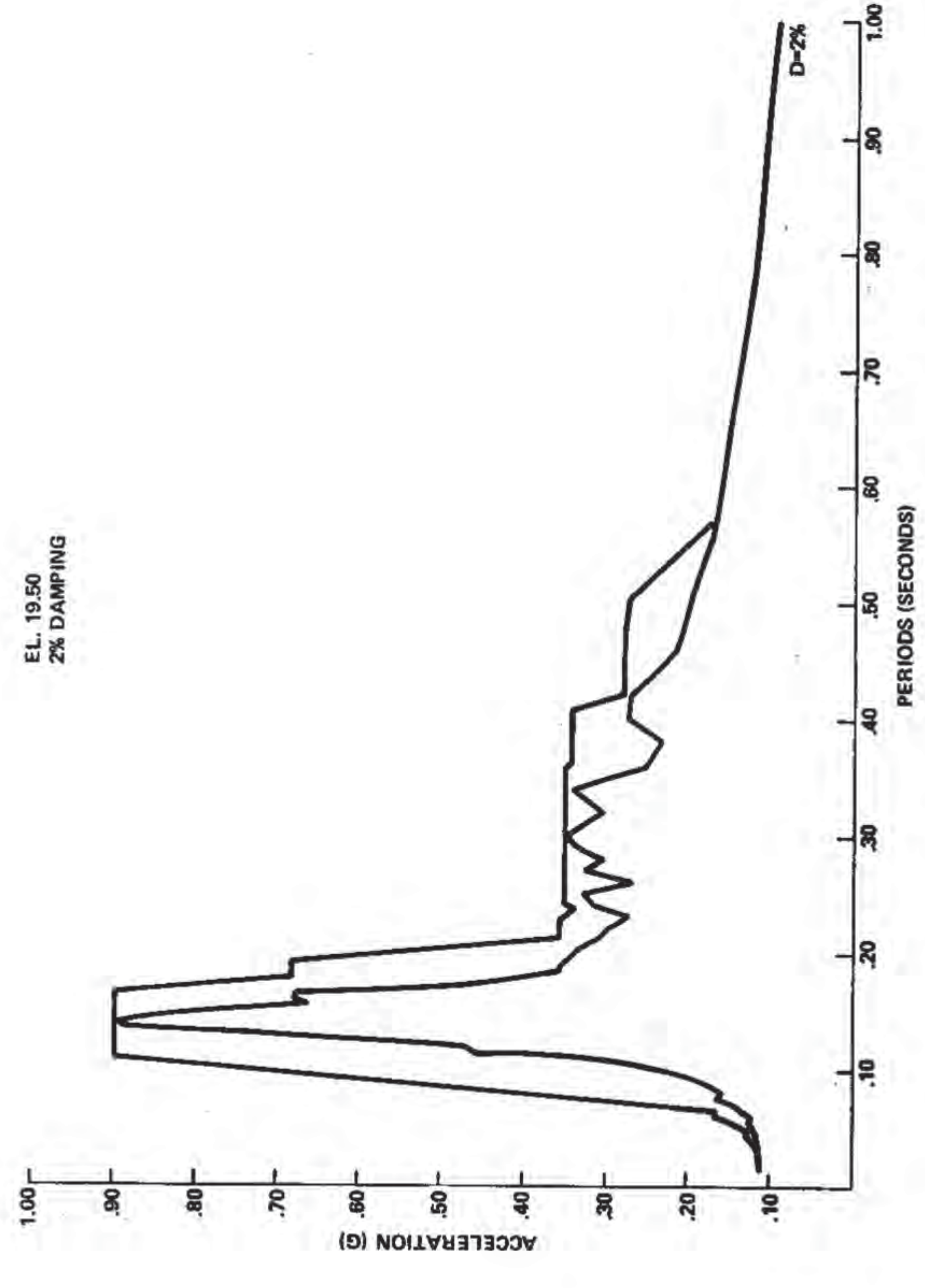
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
MS/FW TRESTLE
FIGURE 3.7-213



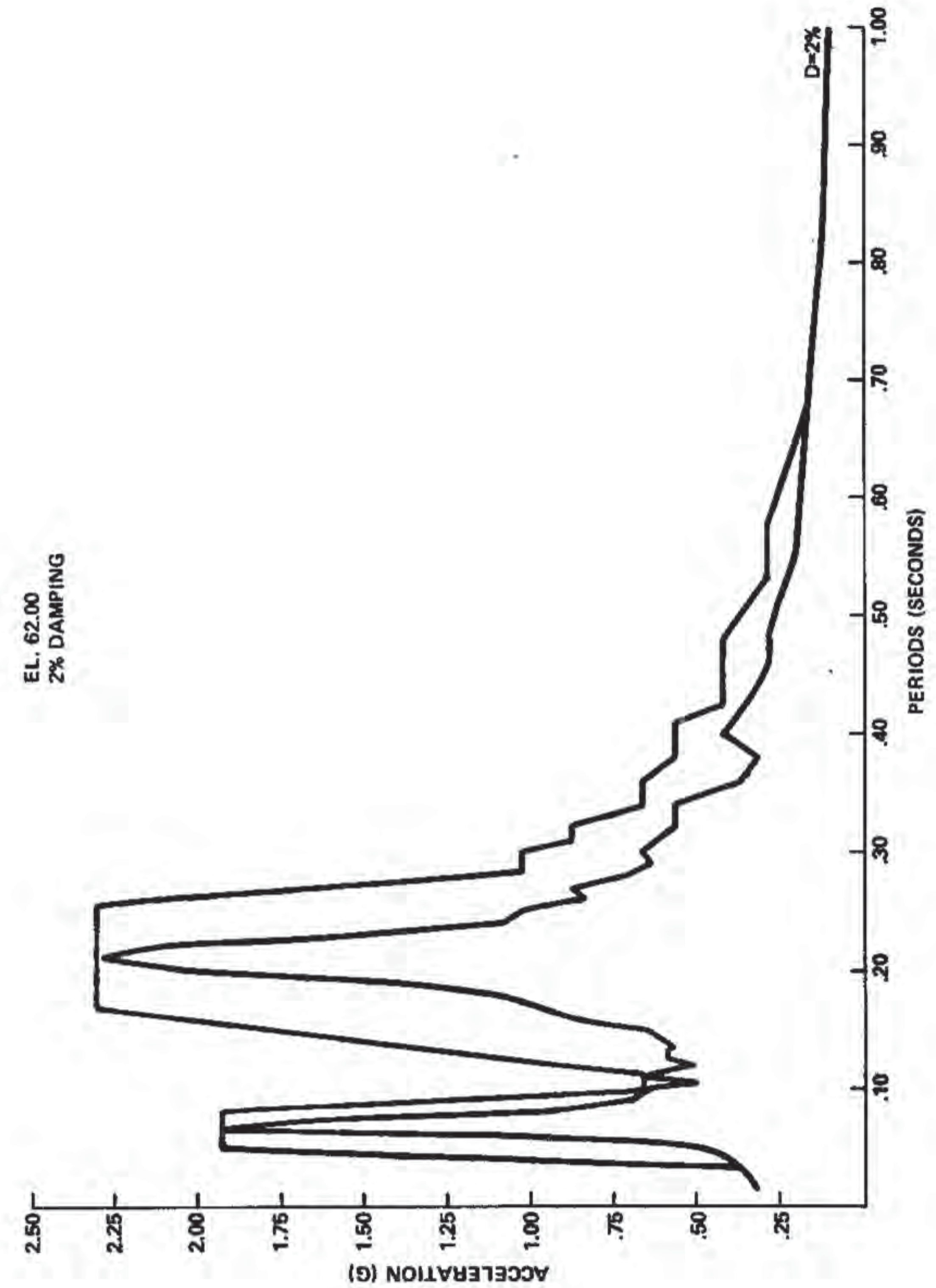
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
MS/FW TRESTLE
FIGURE 3.7-214



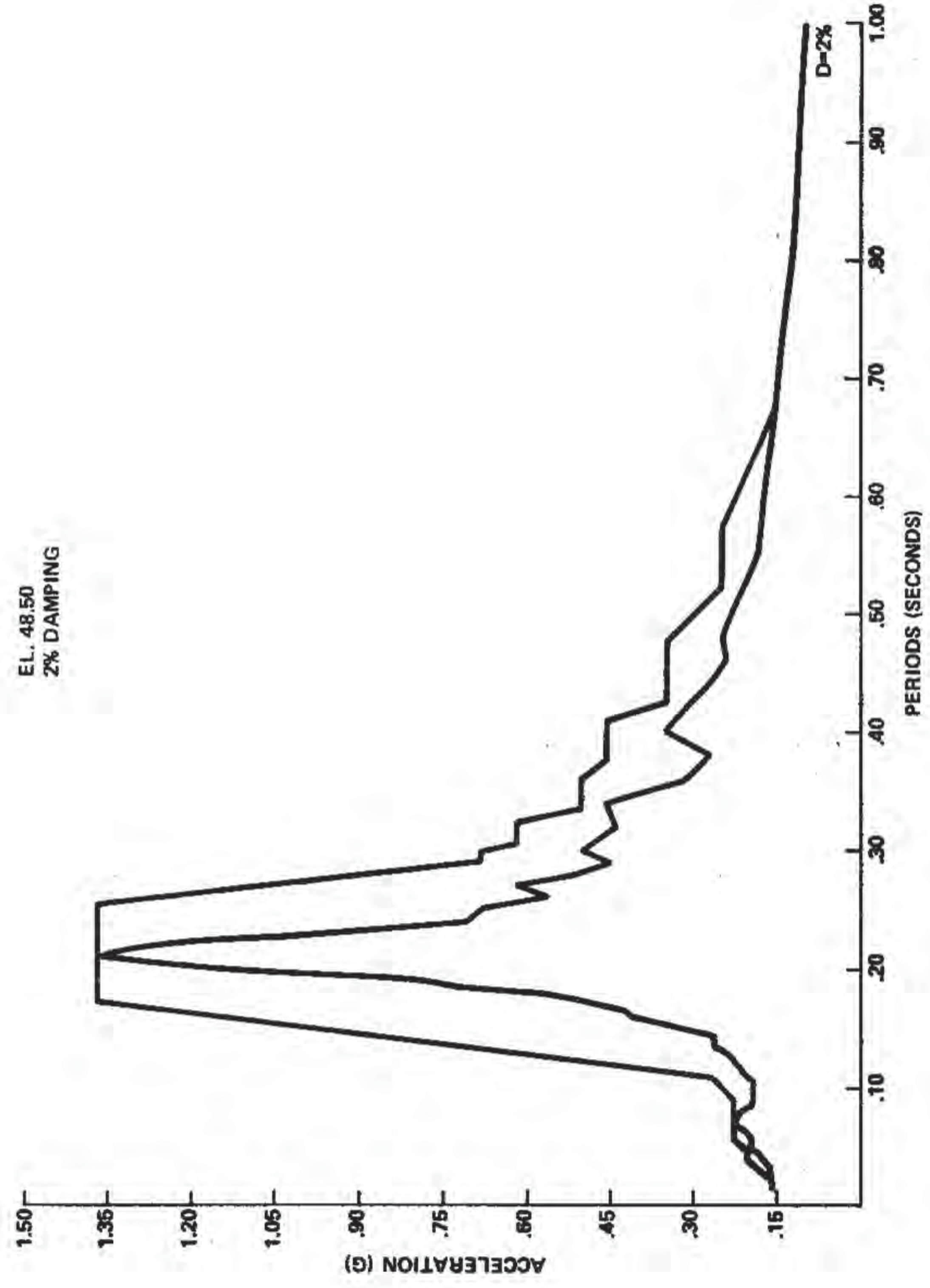
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
MS/FW TRESTLE
FIGURE 3.7-215



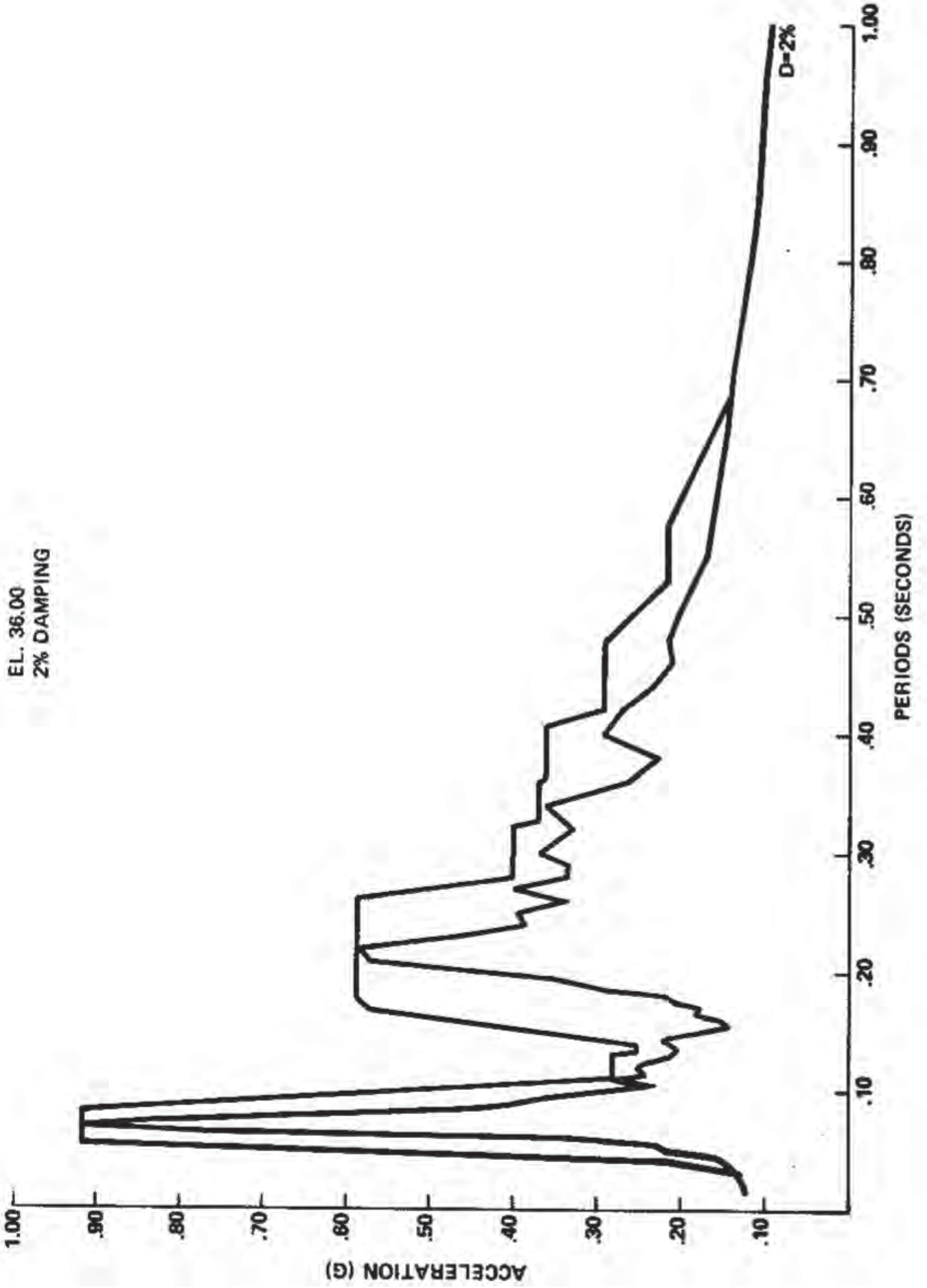
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
MS/FW TRESTLE
FIGURE 3.7-216



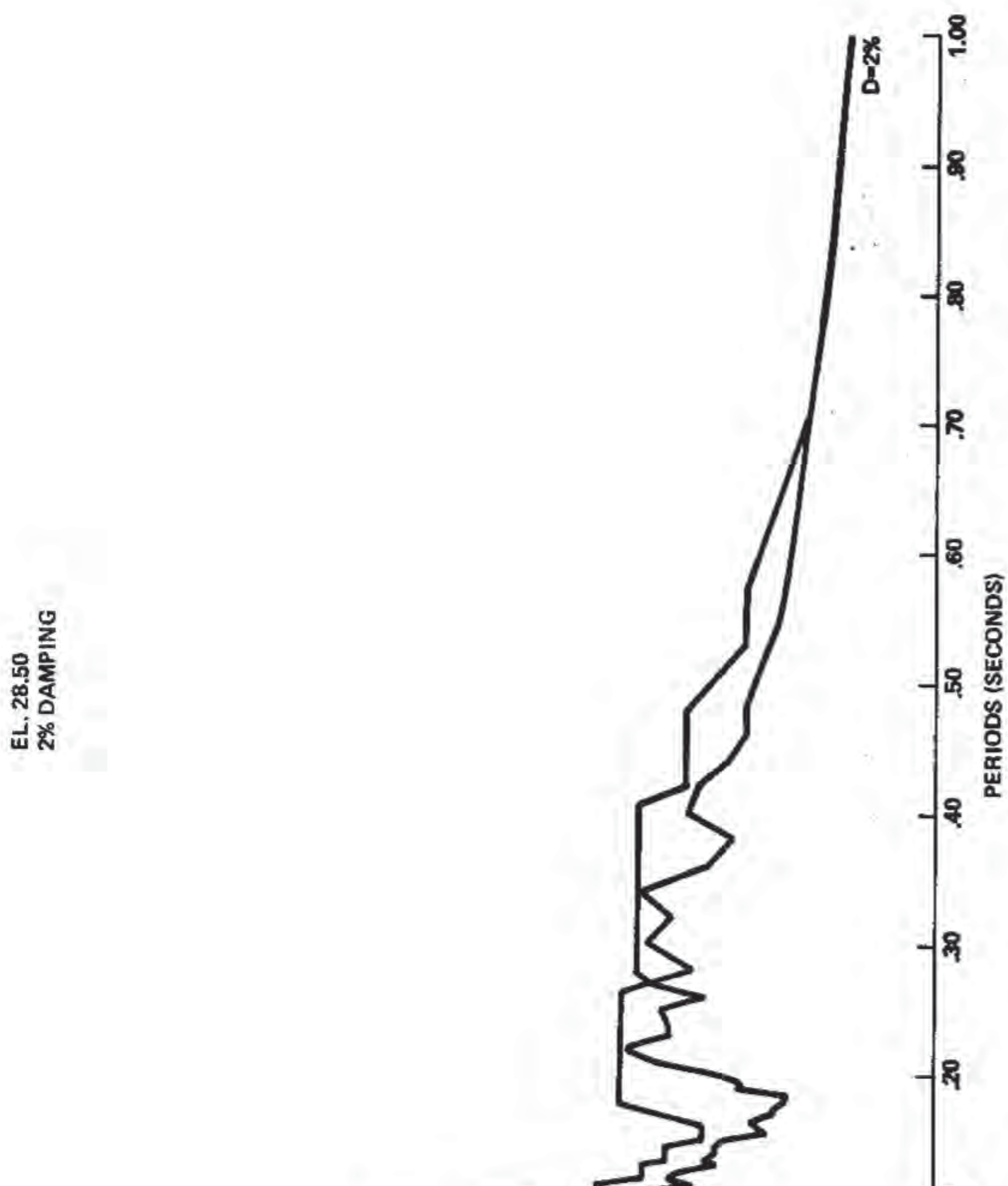
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
MS/FW TRESTLE
FIGURE 3.7-217



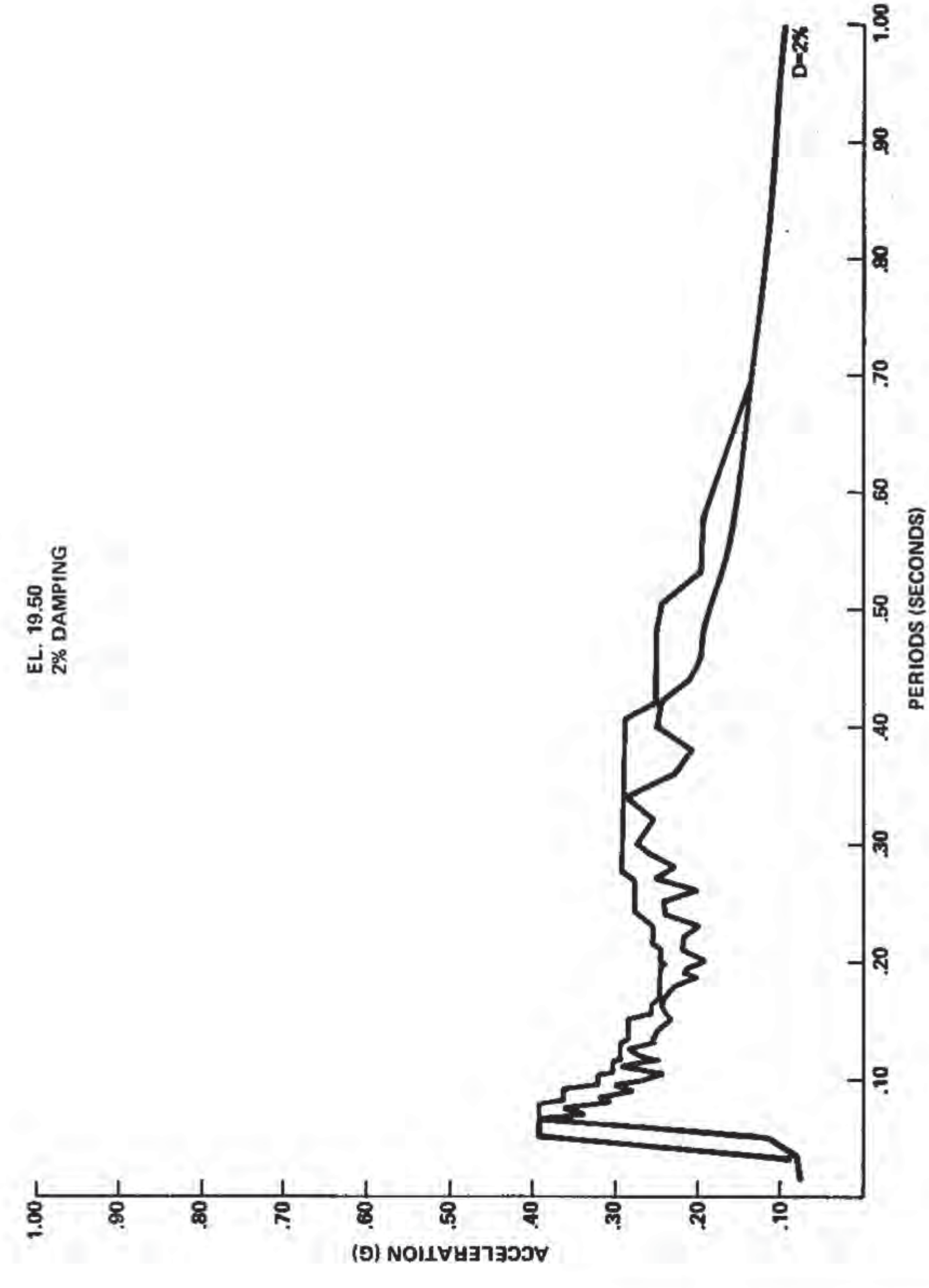
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
MS/FW TRESTLE
FIGURE 3.7-218



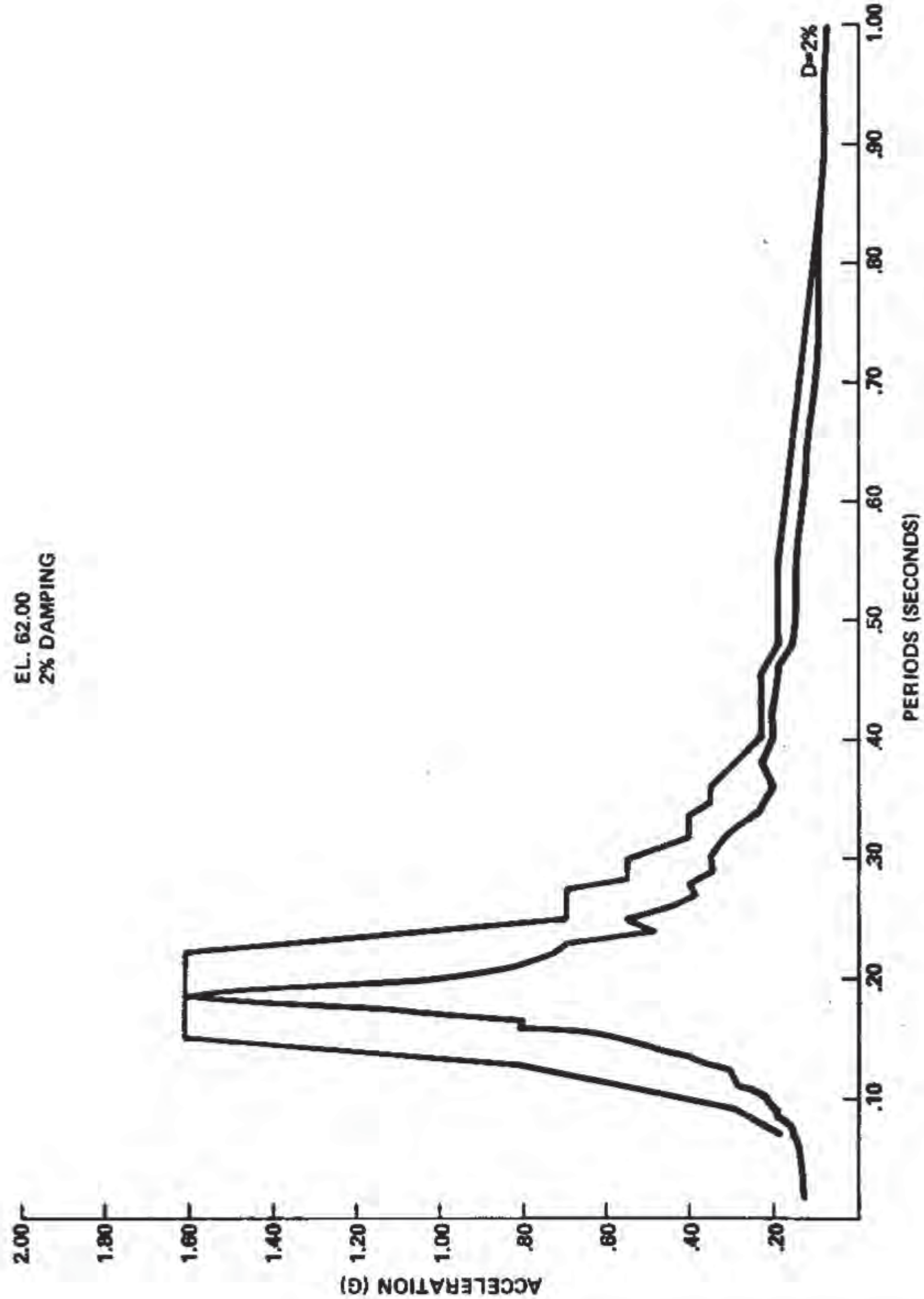
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
MS/FW TRESTLE
FIGURE 3.7-219



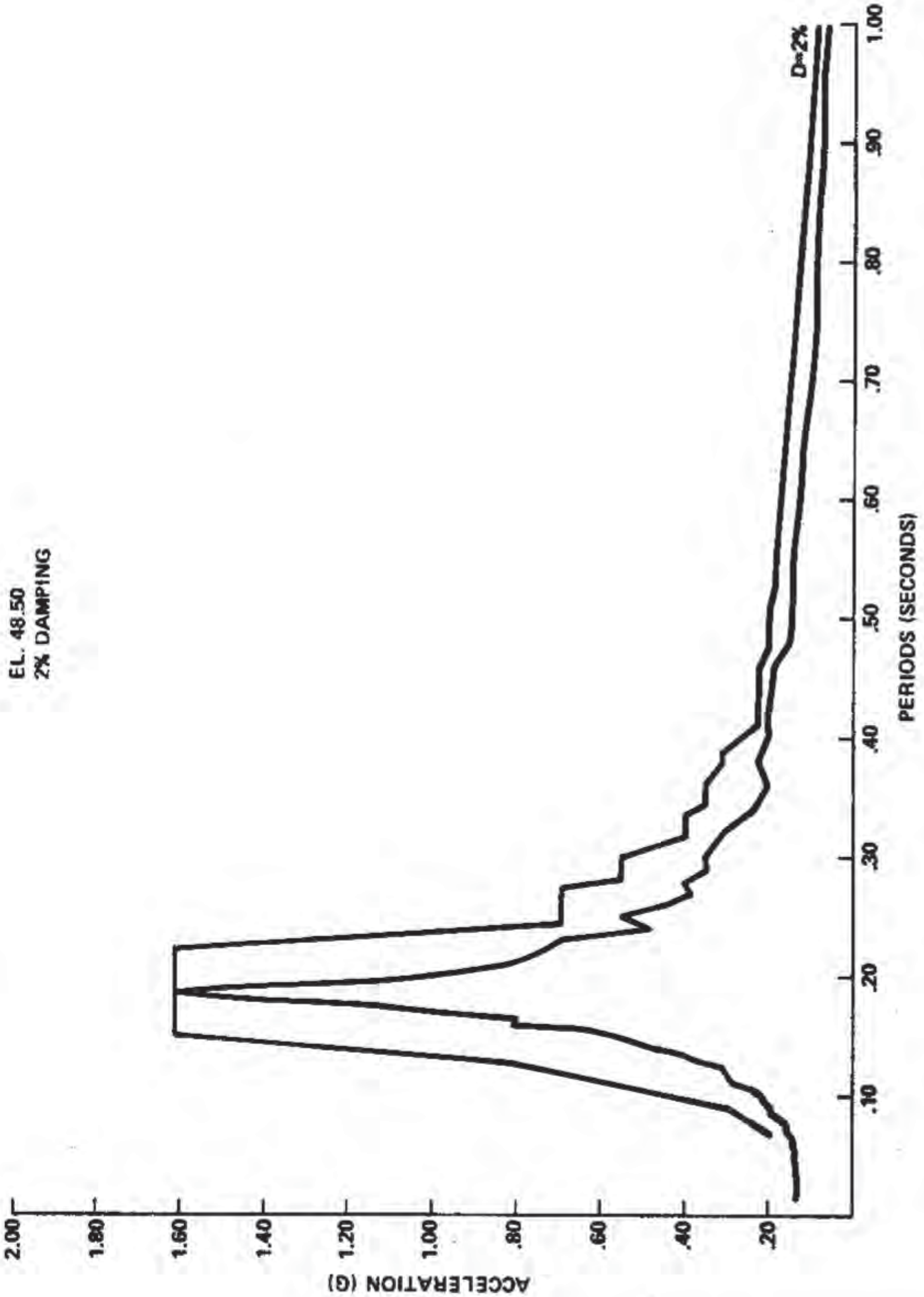
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
MS/FW TRESTLE
FIGURE 3.7-220



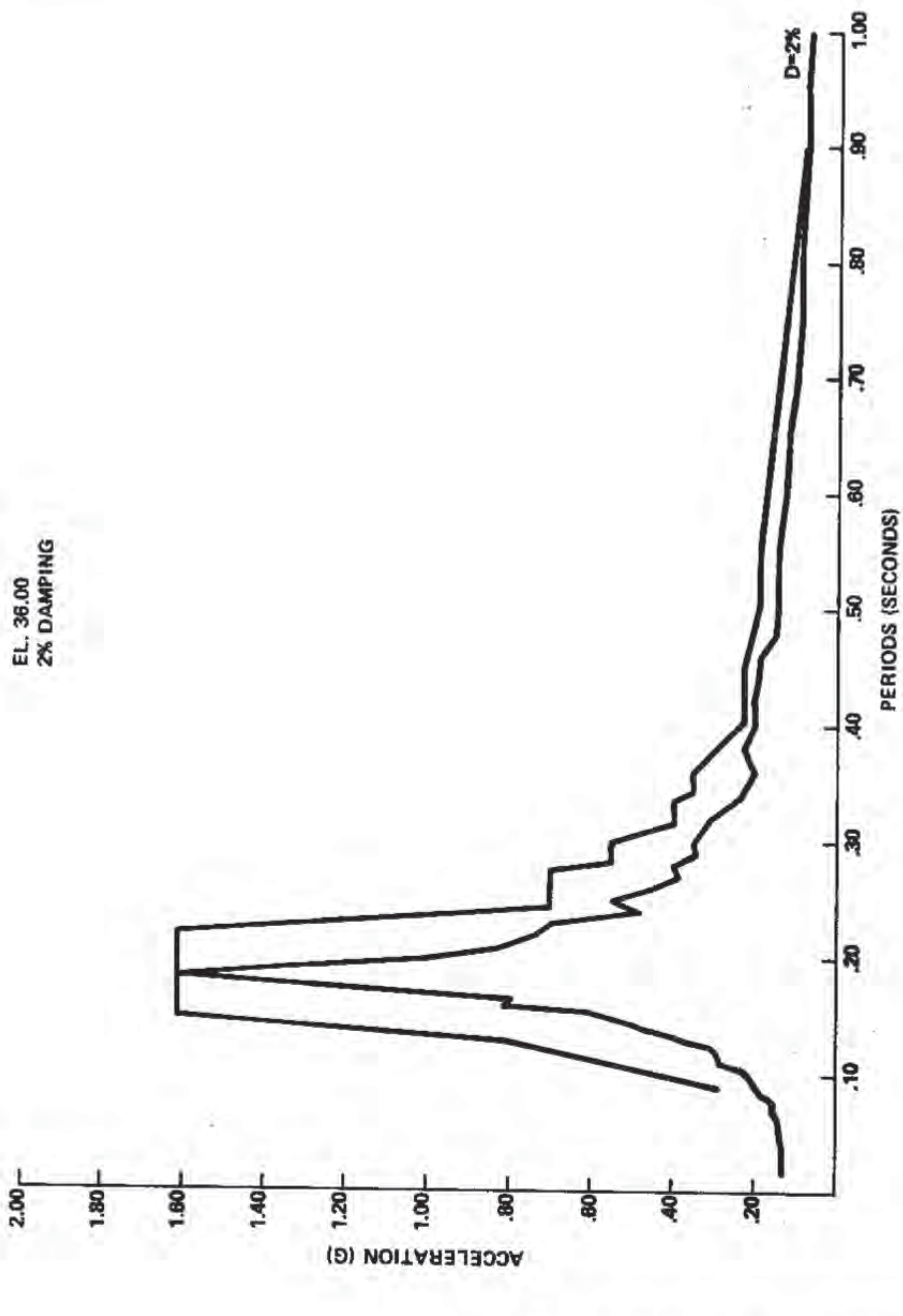
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
MS/FW TRESTLE
FIGURE 3.7-221



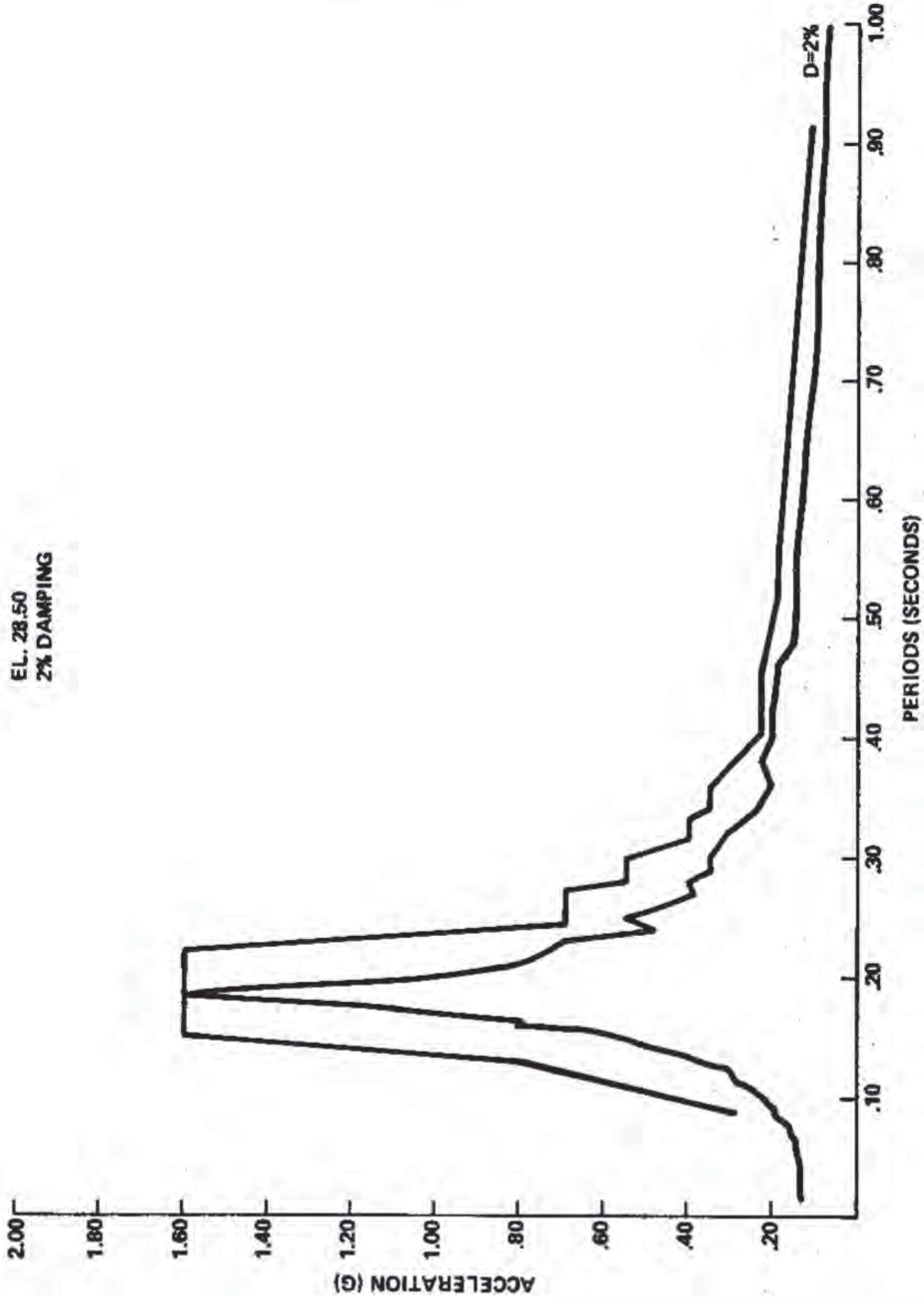
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
MS/FW TRESTLE
FIGURE 3.7-222



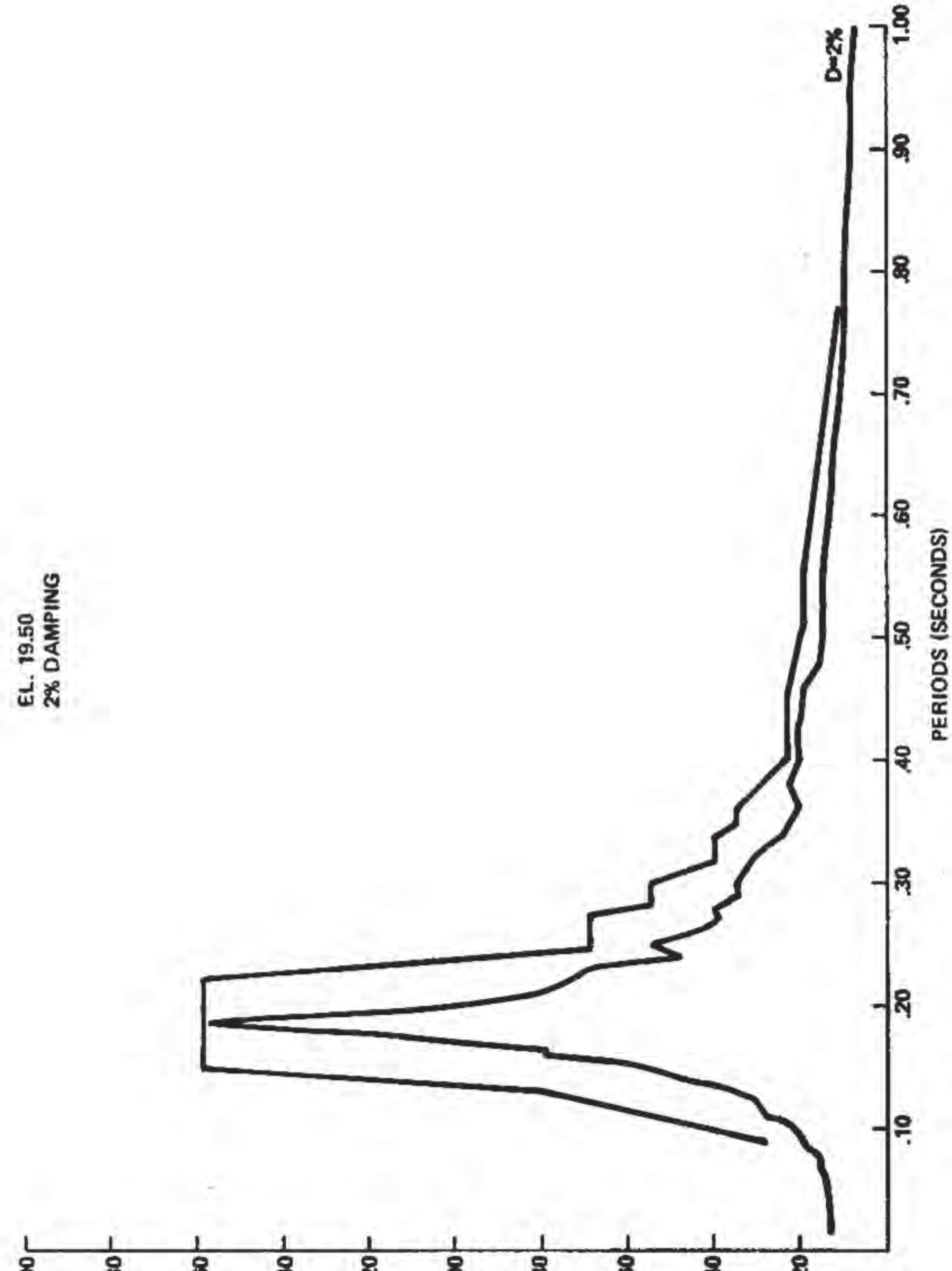
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
MS/FW TRESTLE
FIGURE 3.7-223



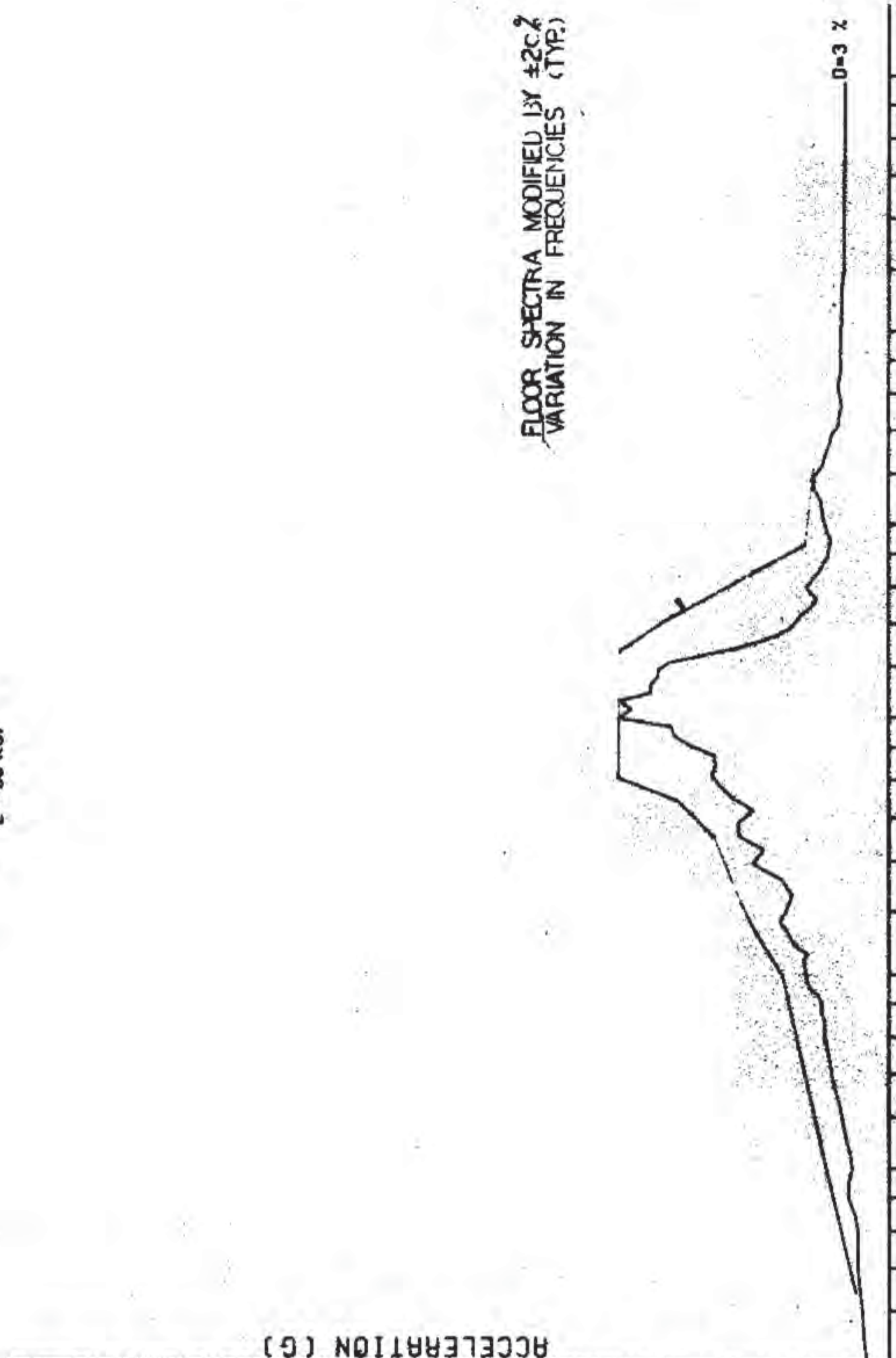
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
MS/FW TRESTLE
FIGURE 3.7-224



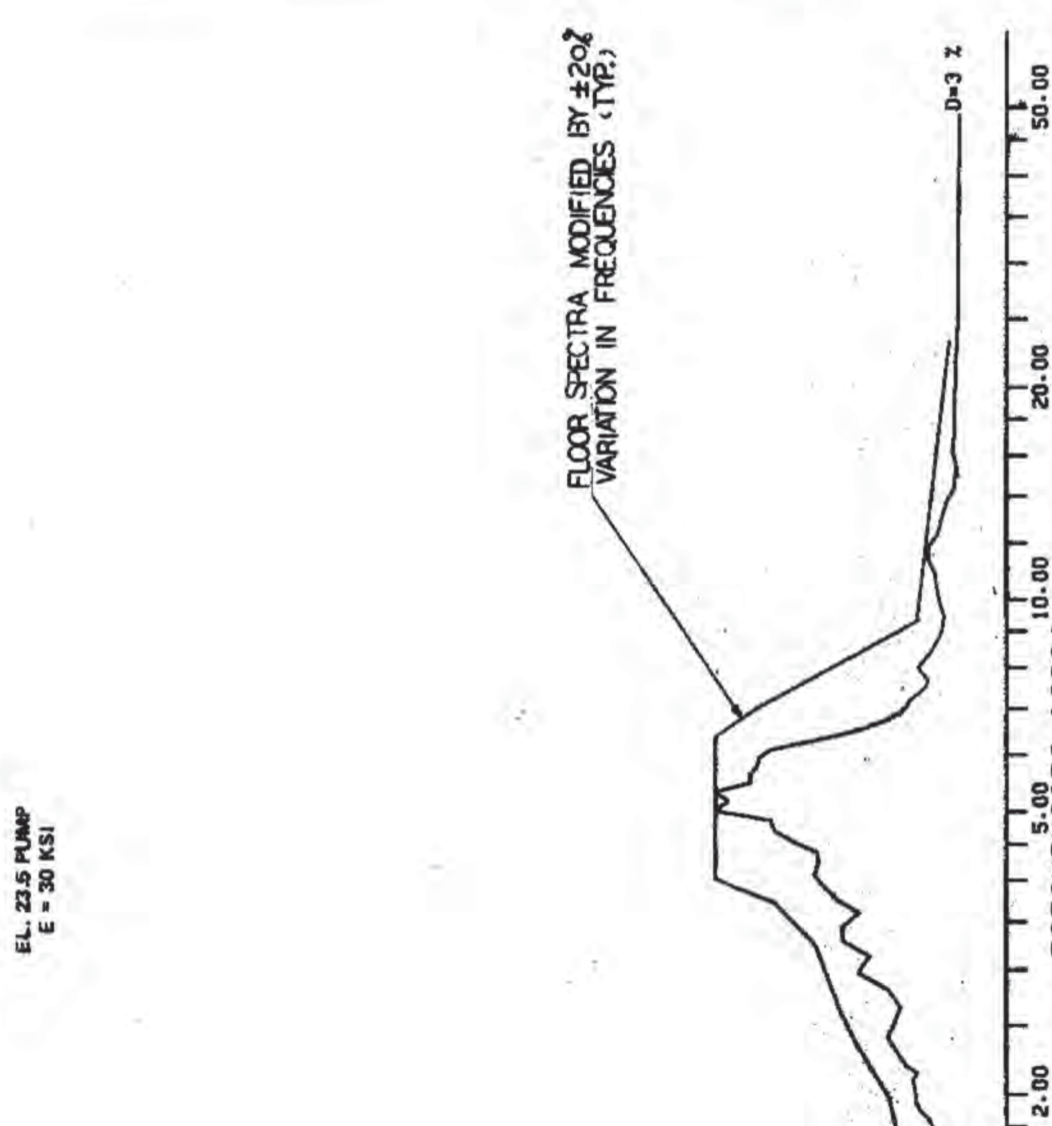
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
MS/FW TRESTLE
FIGURE 3.7-225



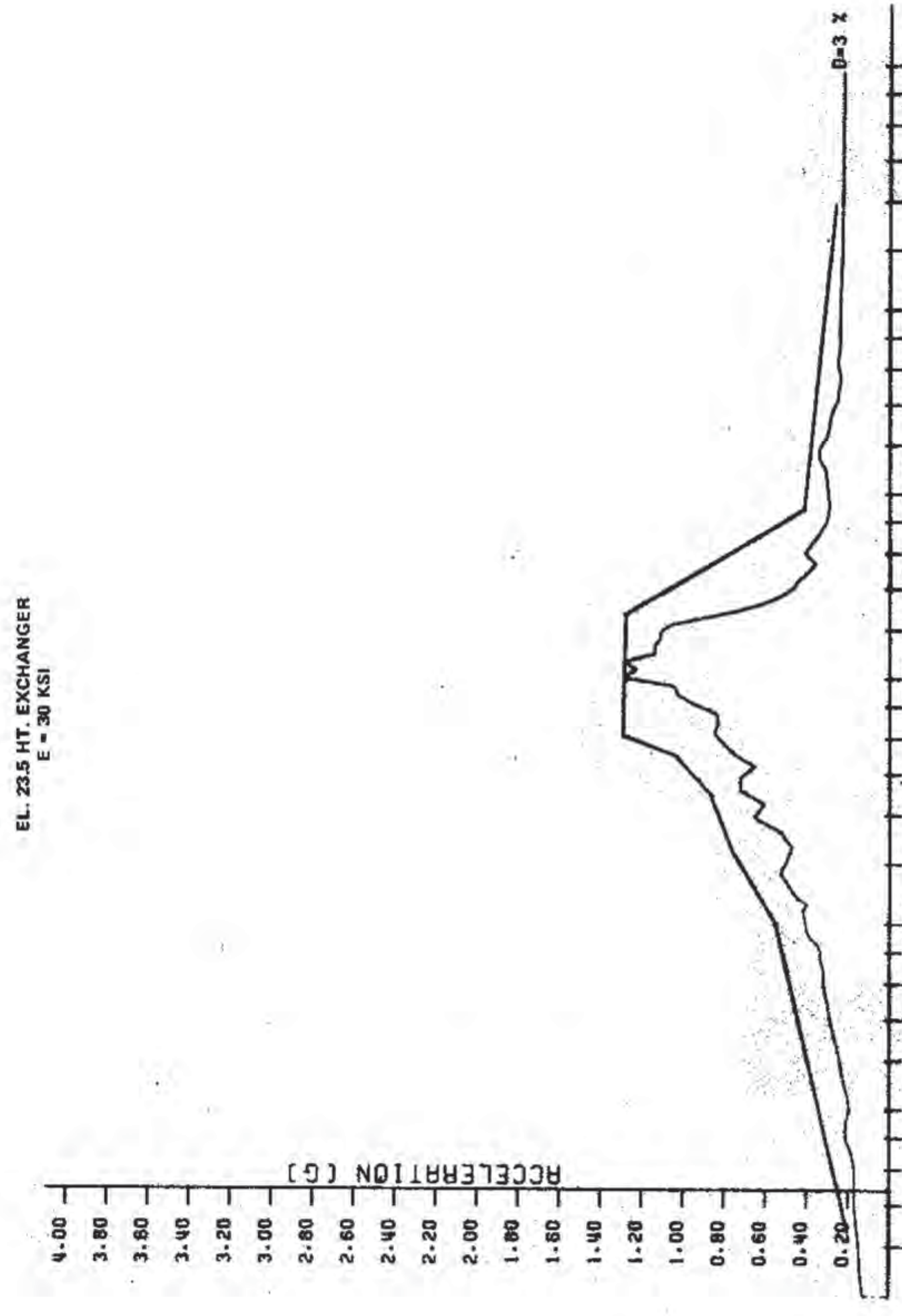
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
COMP'T COOL. WATER BLDG. MASS PT 5
FIGURE 3.7-226



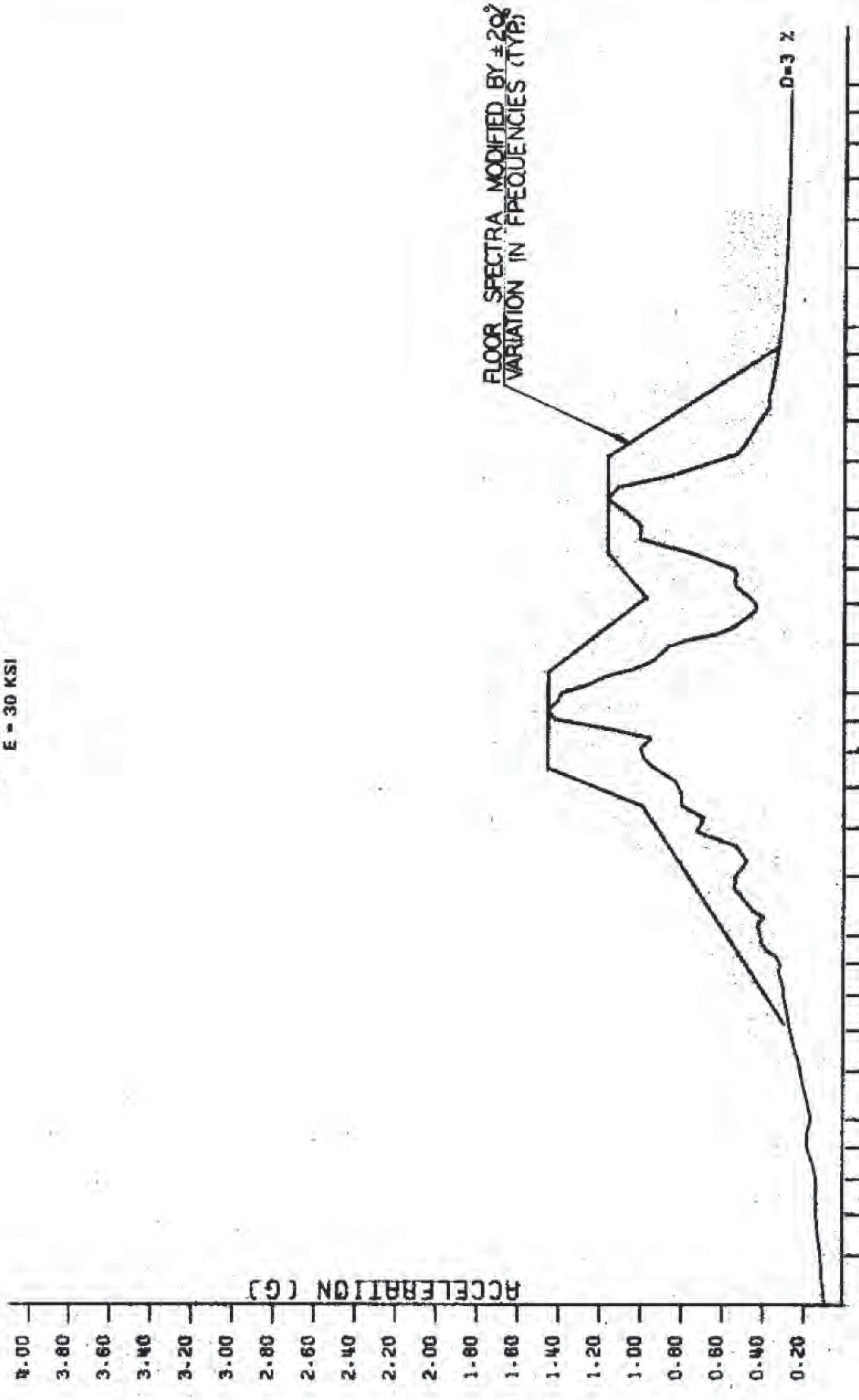
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
COMP'T COOL. WATER BLDG. MASS PT 6
FIGURE 3.7-227



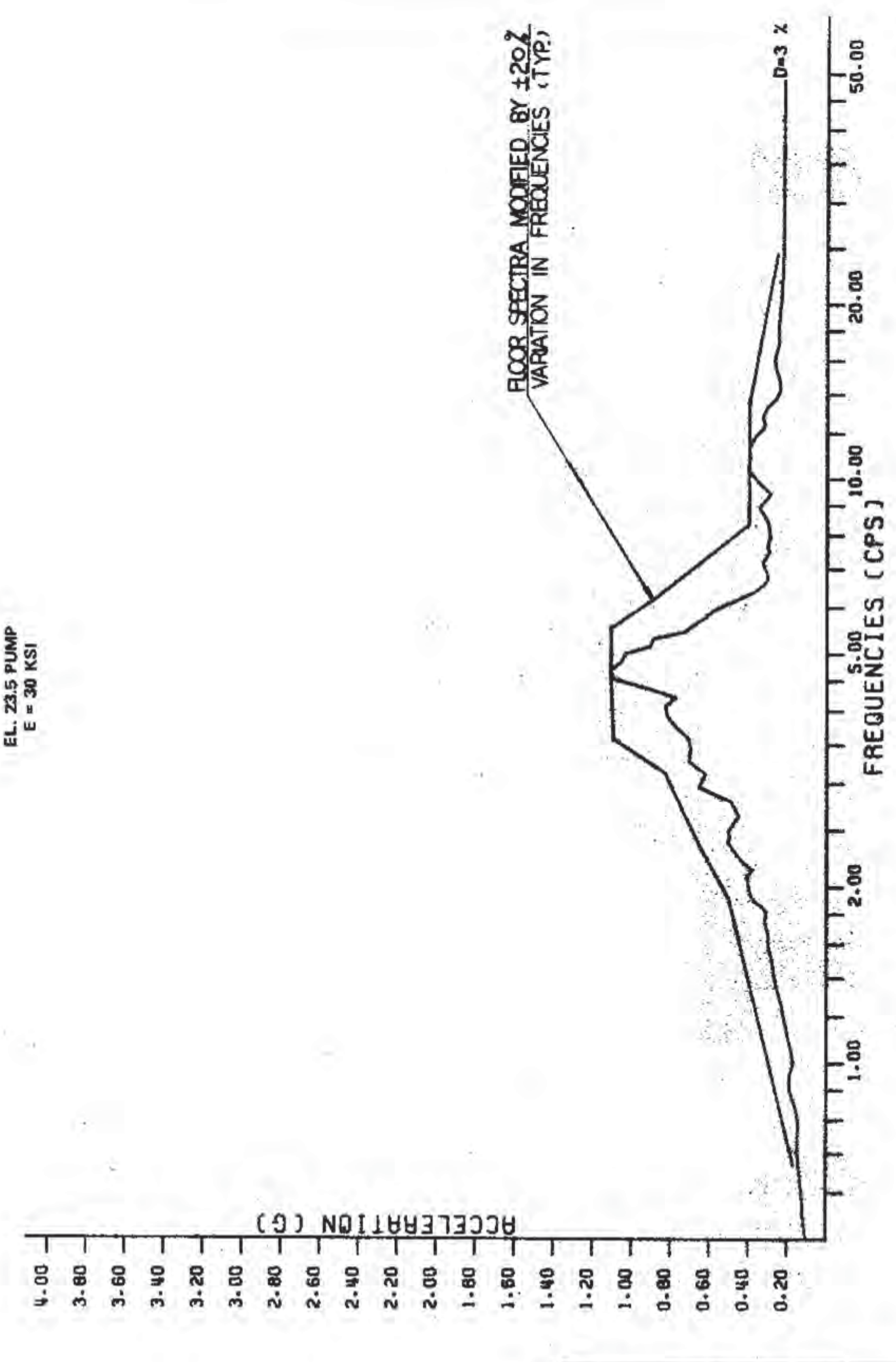
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S SSE
COMPONENT COOLING WATER BLDG.
FIGURE 3.7-228



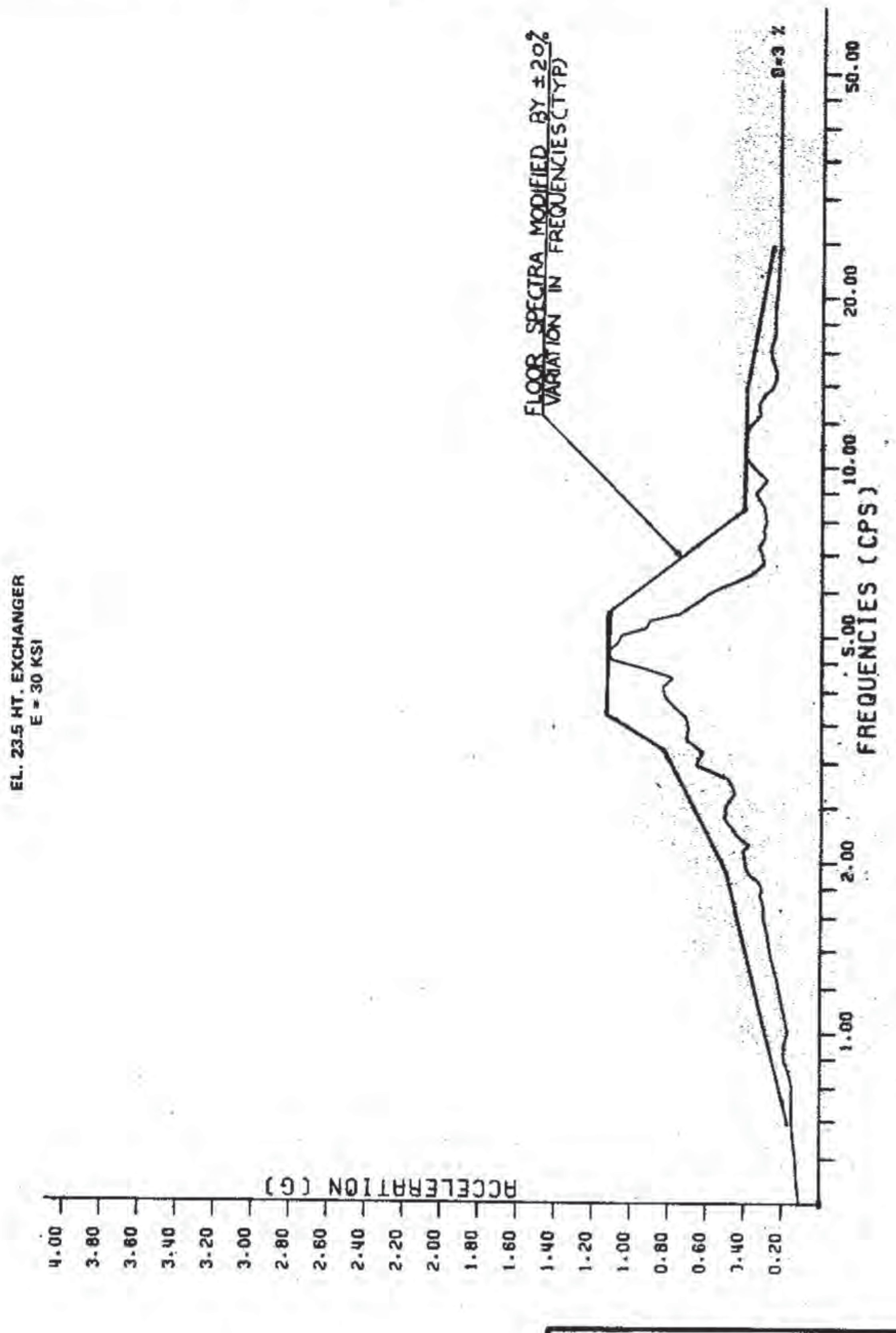
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
COMP'T COOL. WATER BLDG. MASS PT 5
FIGURE 3.7-229



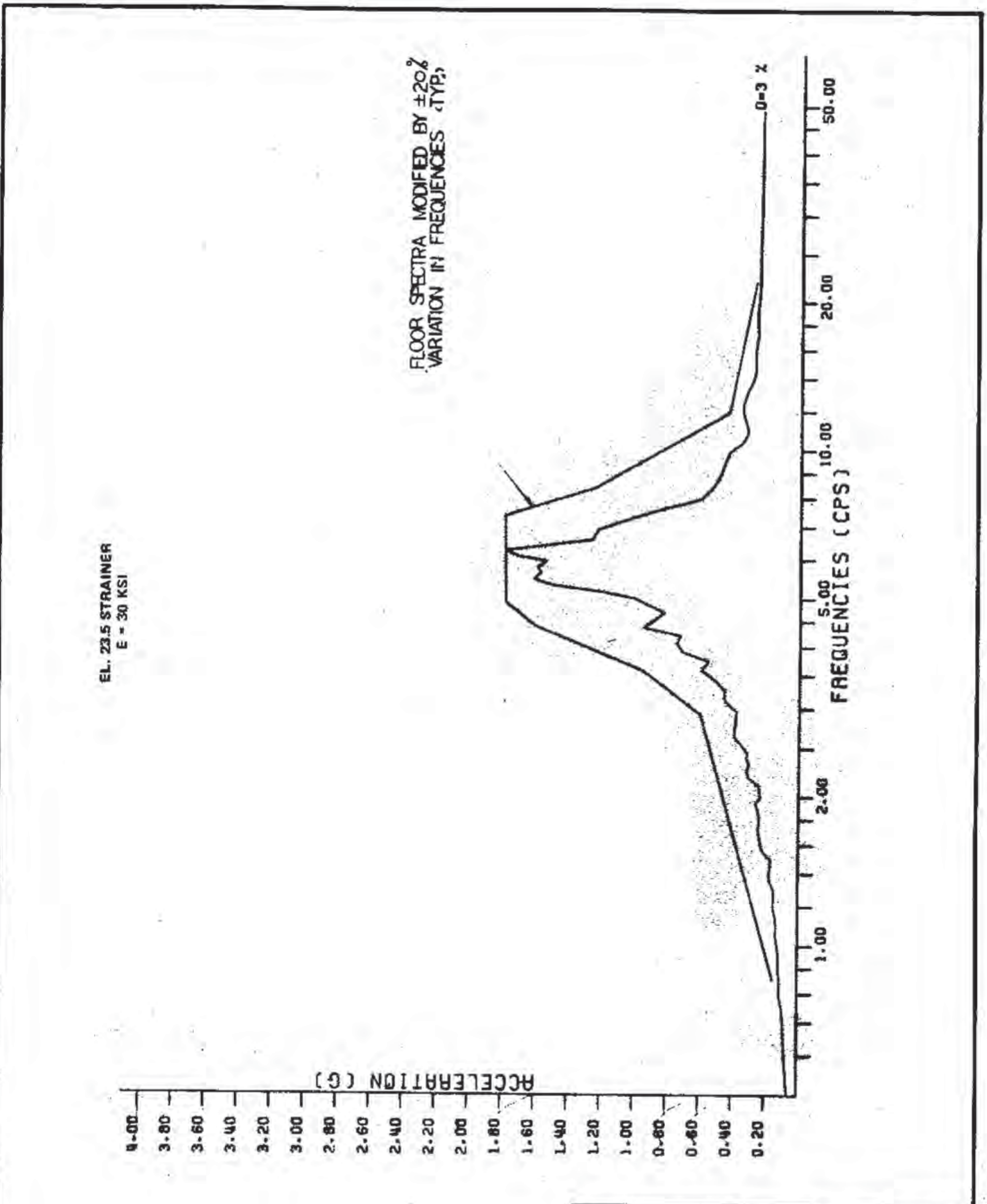
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
COMP'T COOL. WATER BLDG. MASS PT 6
FIGURE 3.7-230



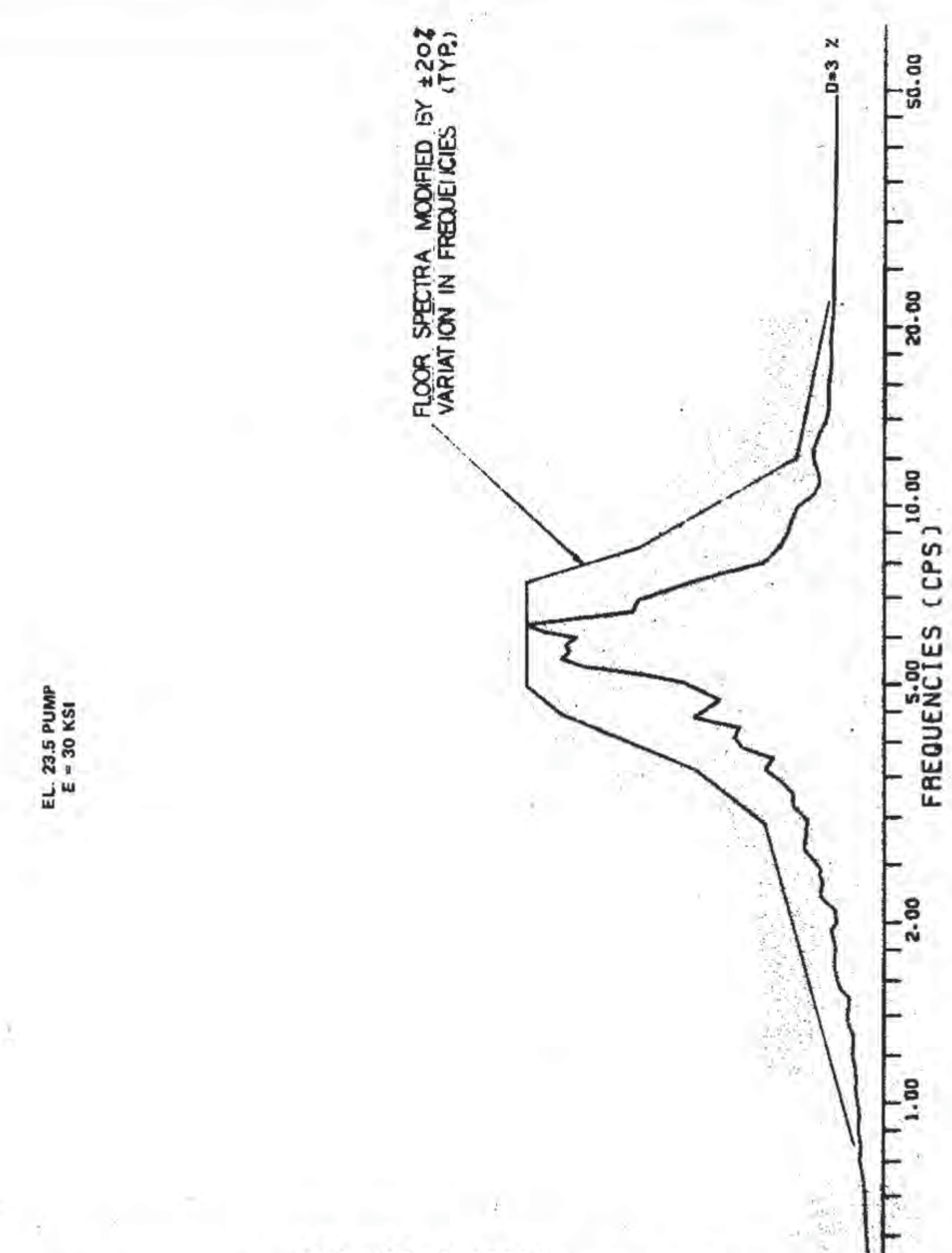
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
COMPONENT COOLING WATER BLDG.
FIGURE 3.7-231



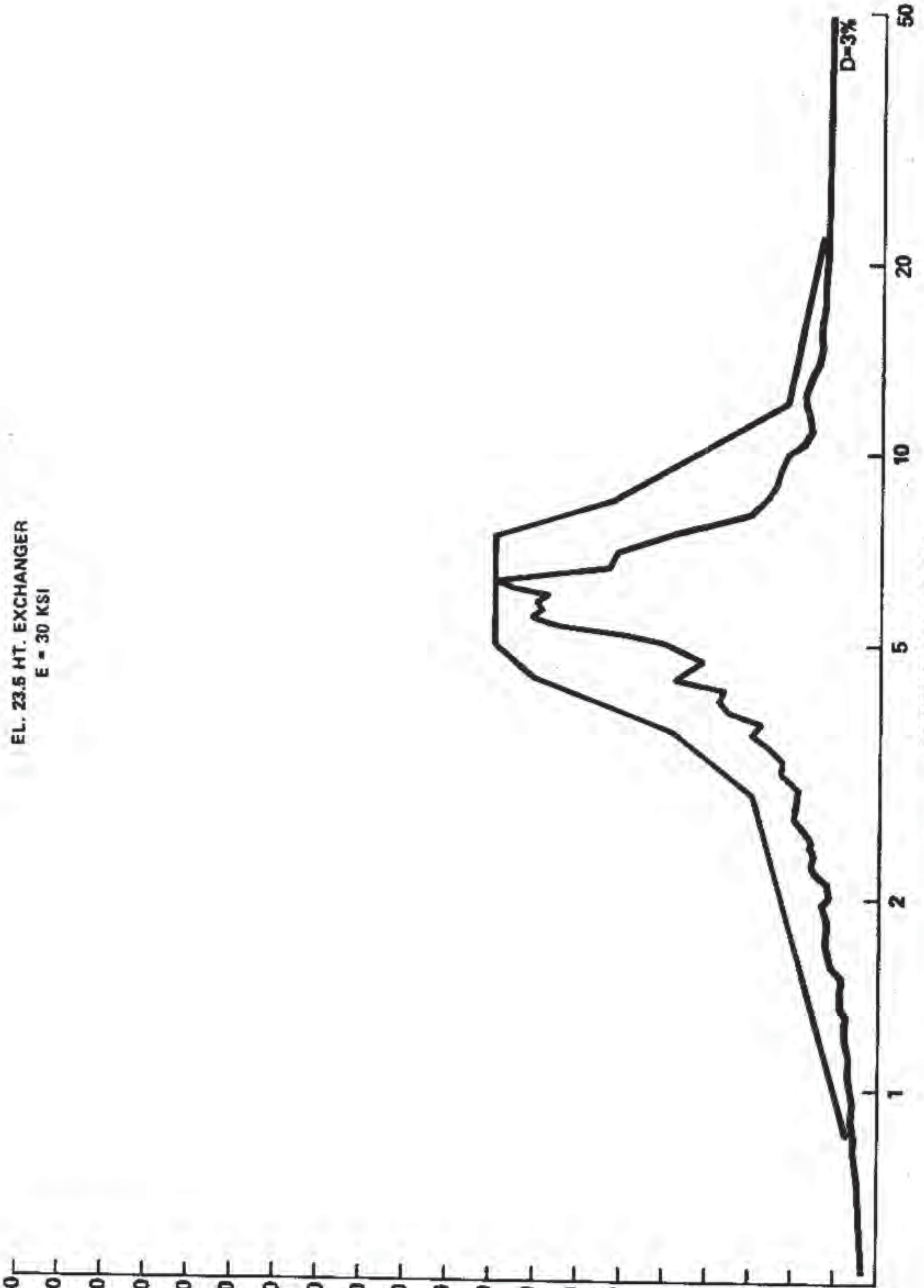
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
COMP'T COOL. WATER BLDG. MASS PT 5
FIGURE 3.7-232



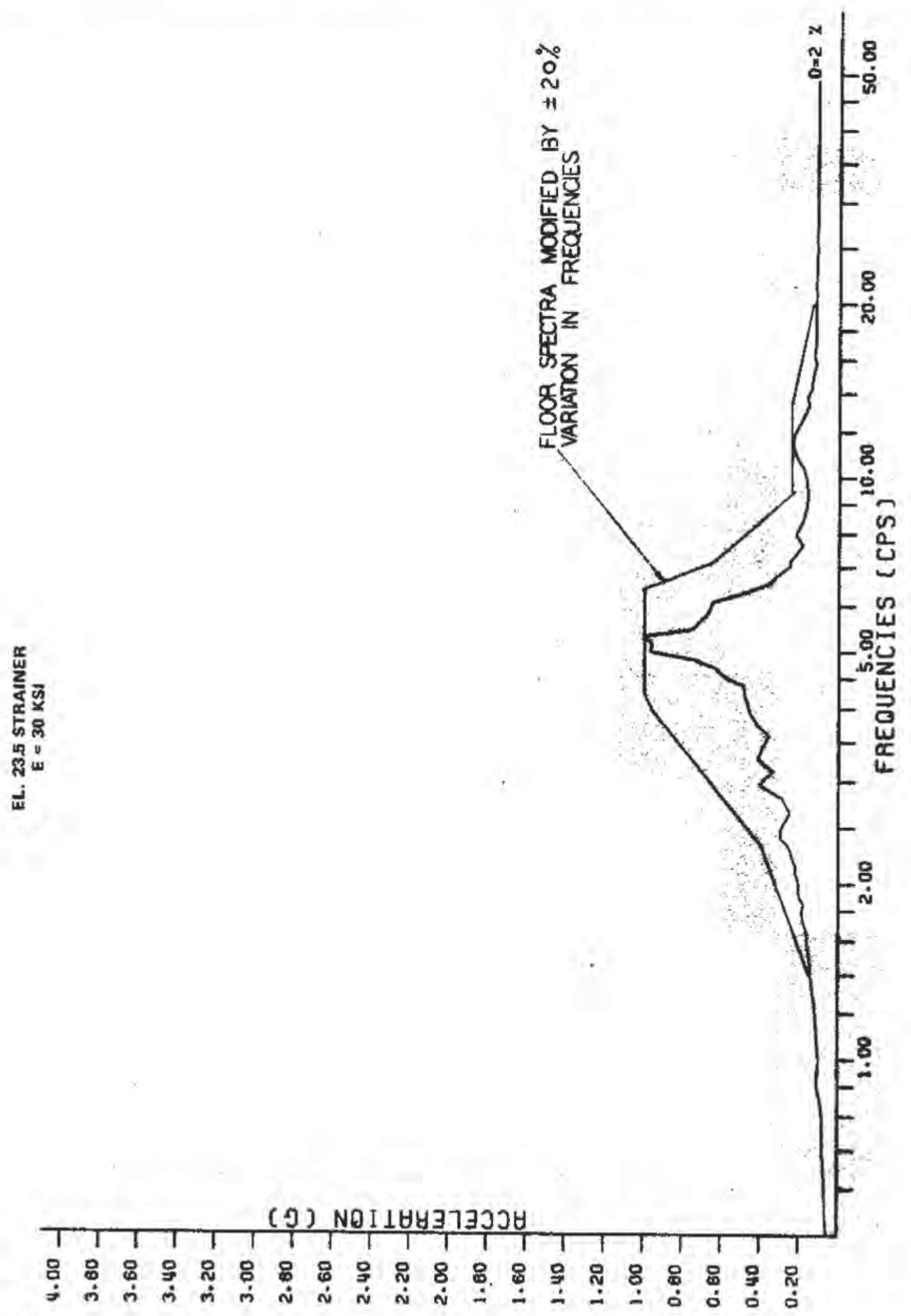
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
COMP'T COOL. WATER BLDG. MASS PT 6
FIGURE 3.7-233



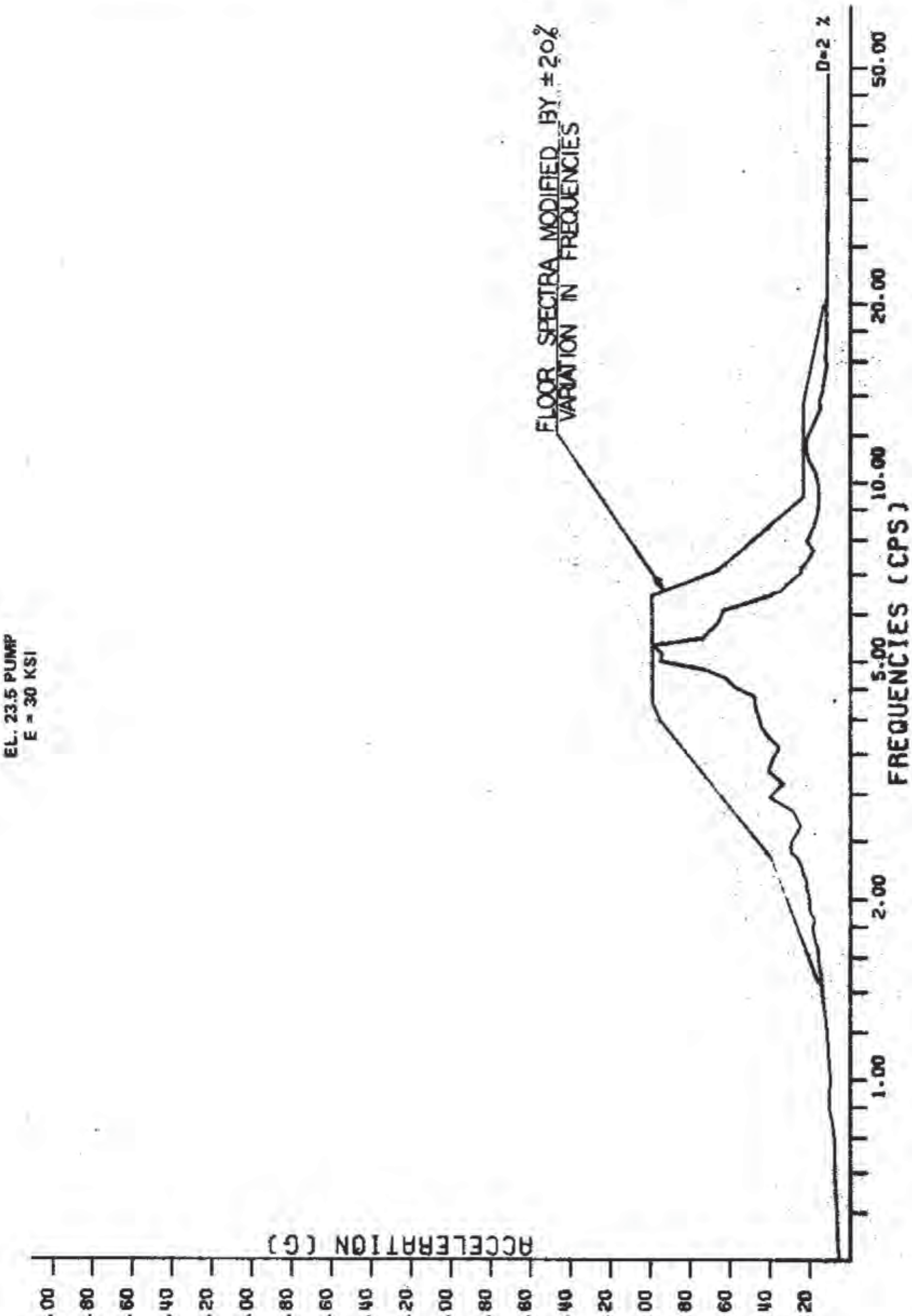
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
COMPONENT COOLING WATER BLDG.
FIGURE 3.7-234



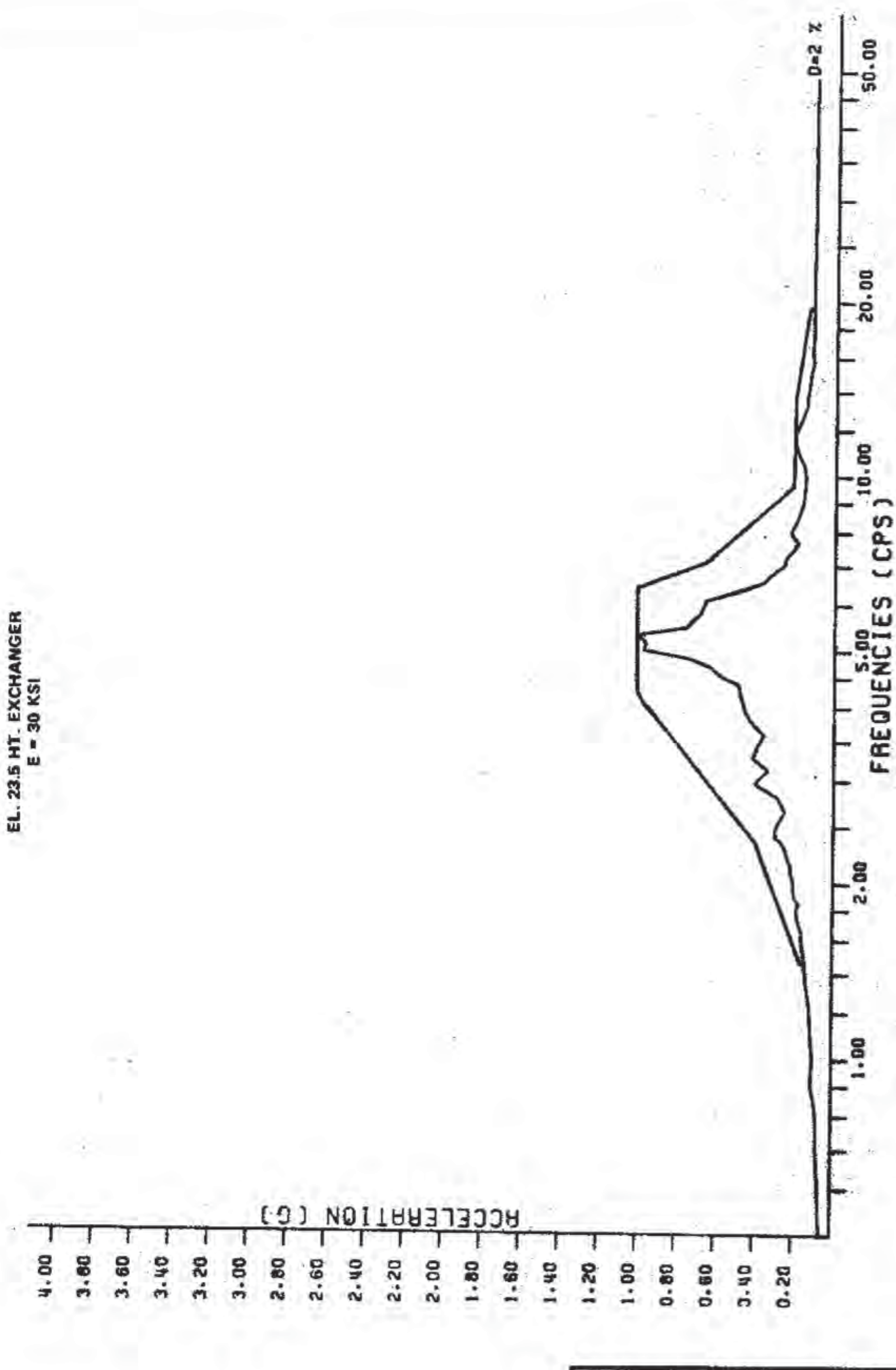
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
COMP'T COOL. WATER BLDG. MASS PT 5
FIGURE 3.7-235



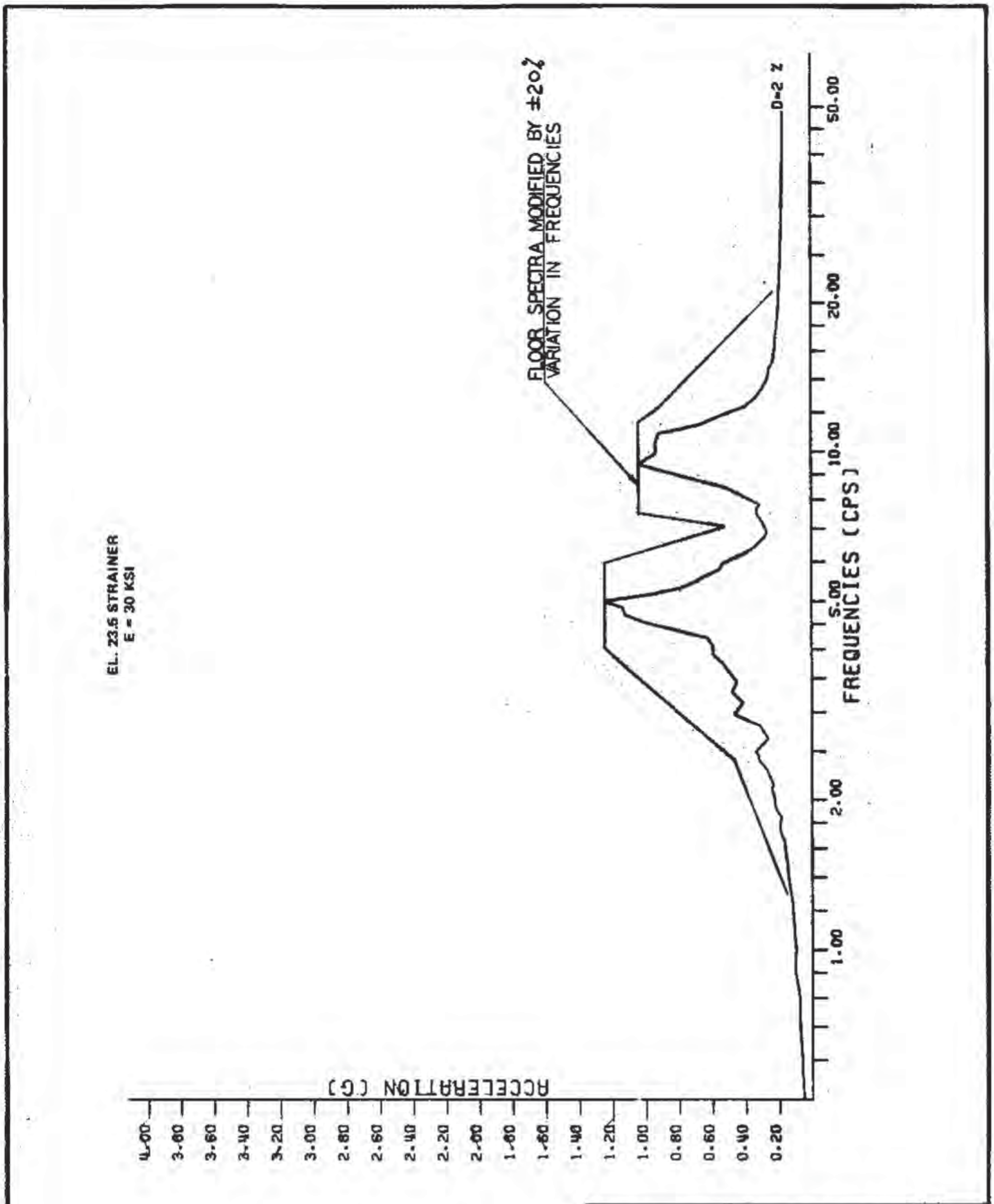
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
COMP'T COOL. WATER BLDG. MASS PT 6
FIGURE 3.7-236



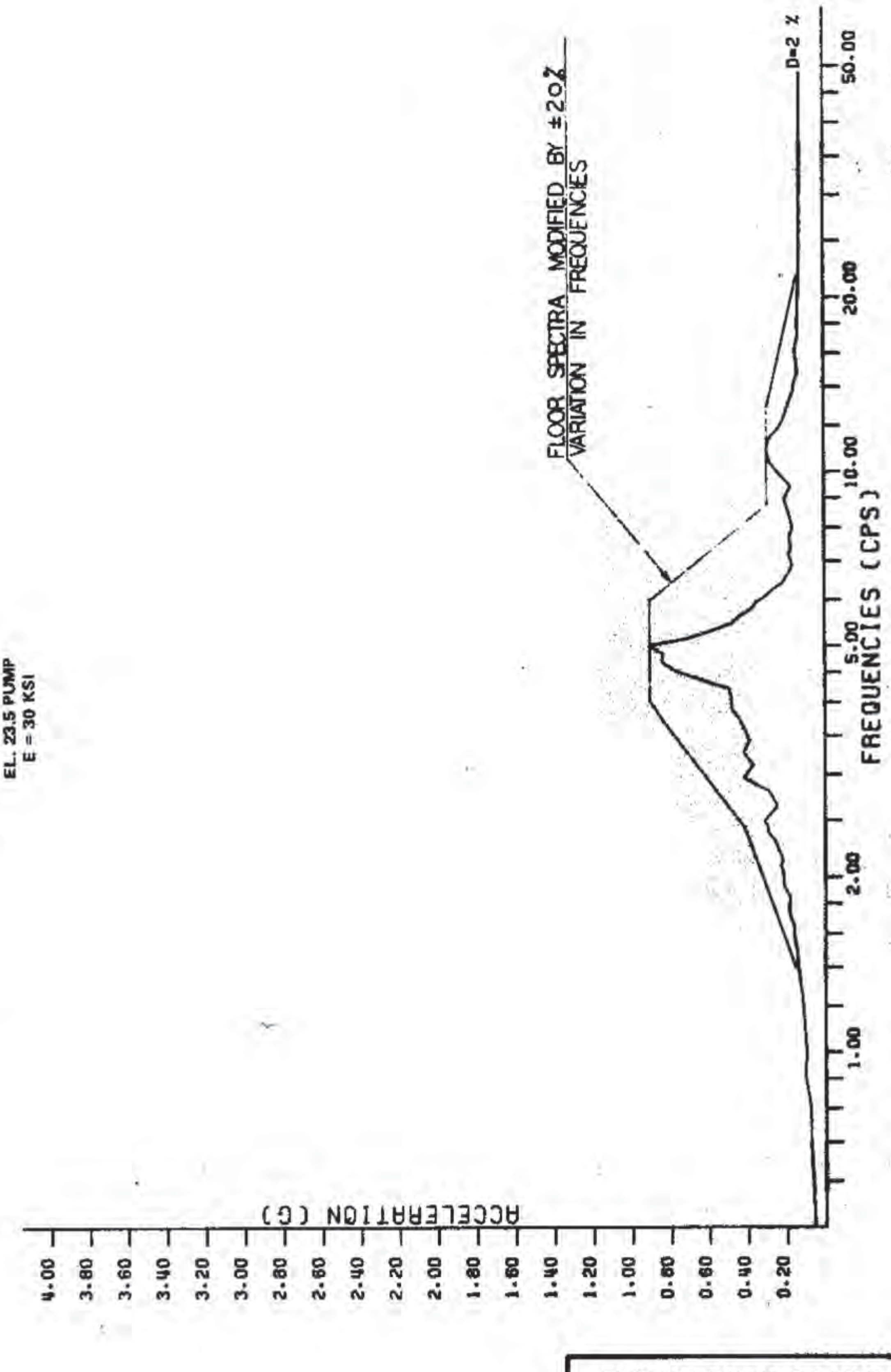
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
COMPONENT COOLING WATER BLDG.
FIGURE 3.7-237



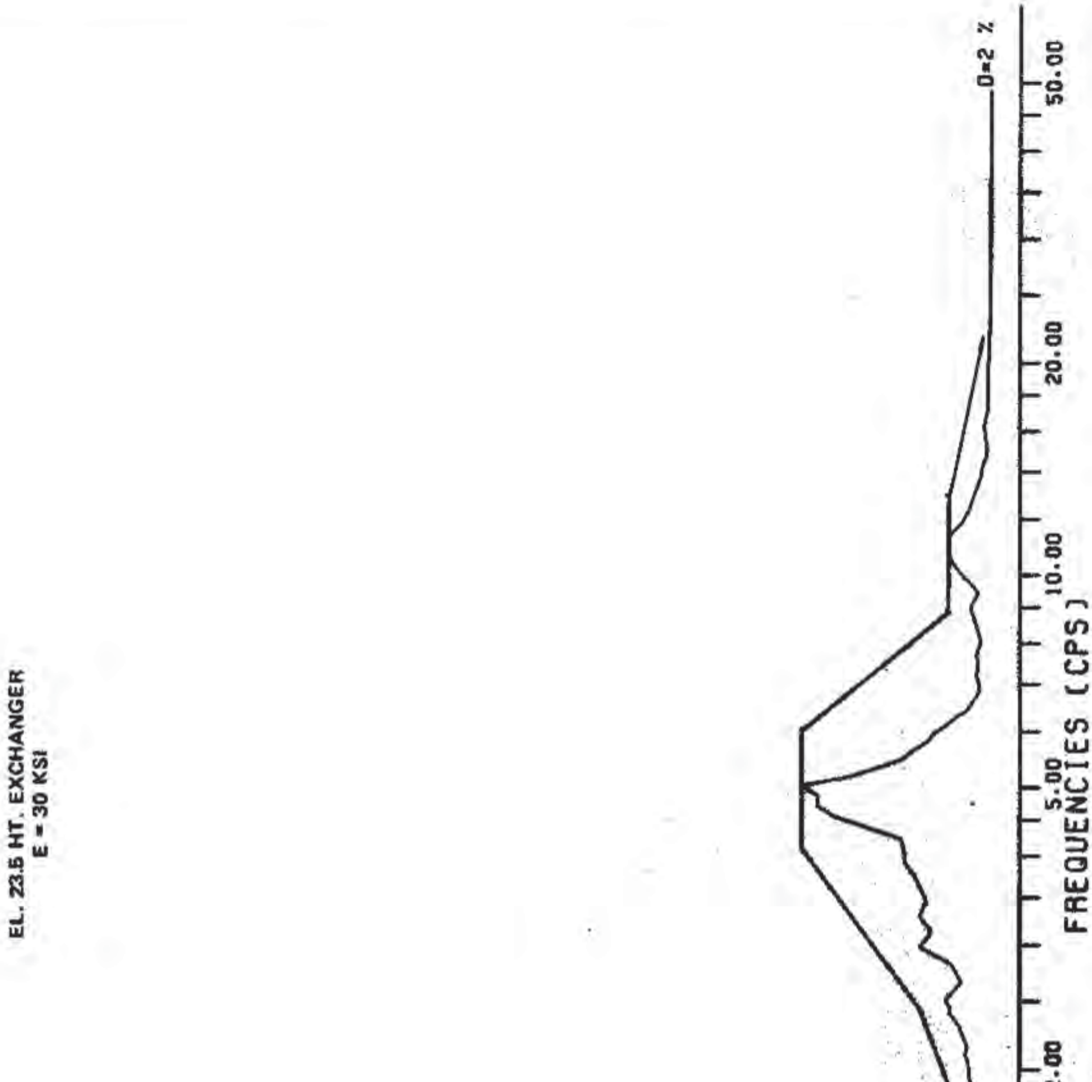
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
COMP'T COOL. WATER BLDG. MASS PT 5
FIGURE 3.7-238



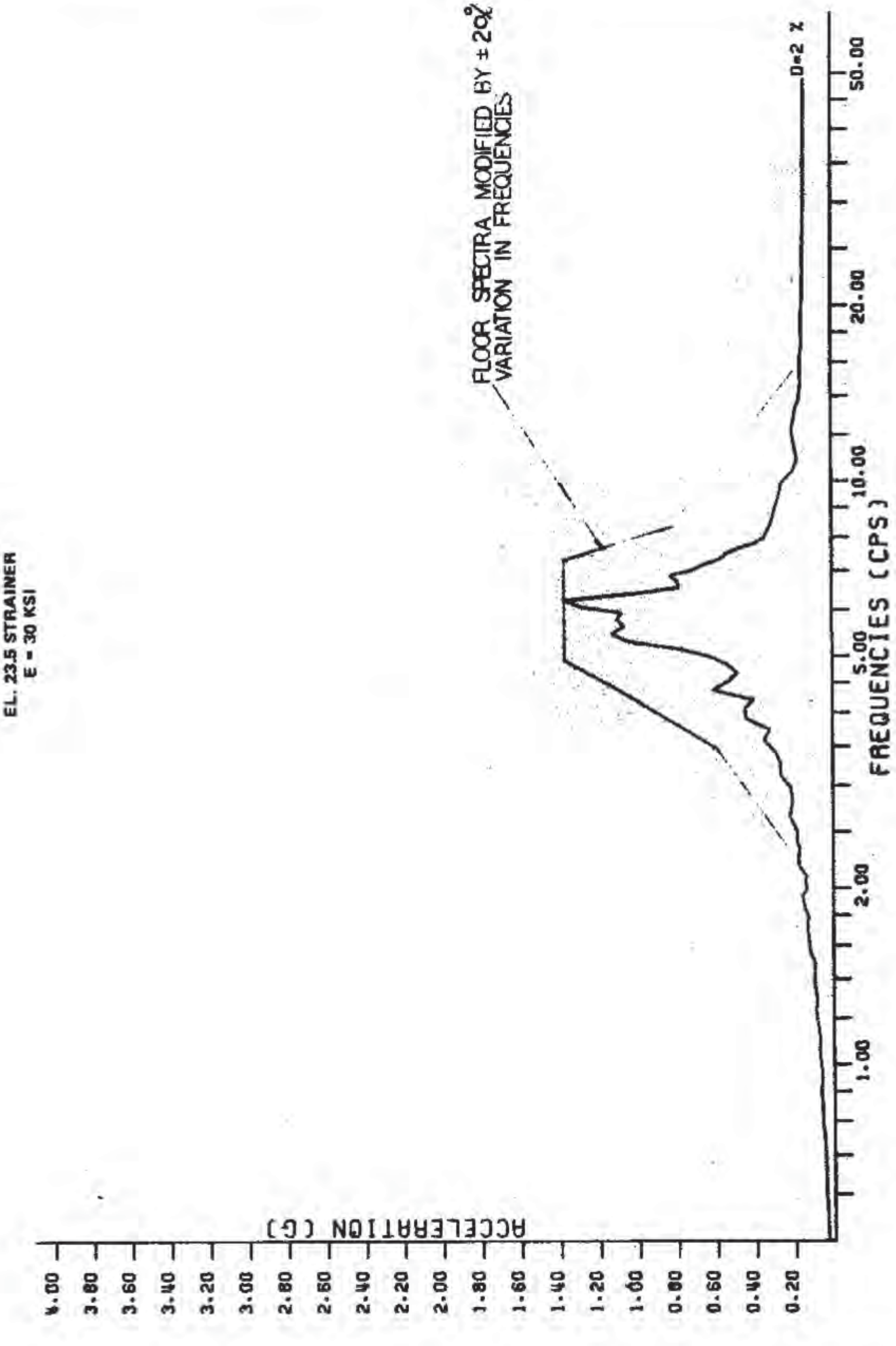
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
COMP'T COOL. WATER BLDG. MASS PT 6
FIGURE 3.7-239



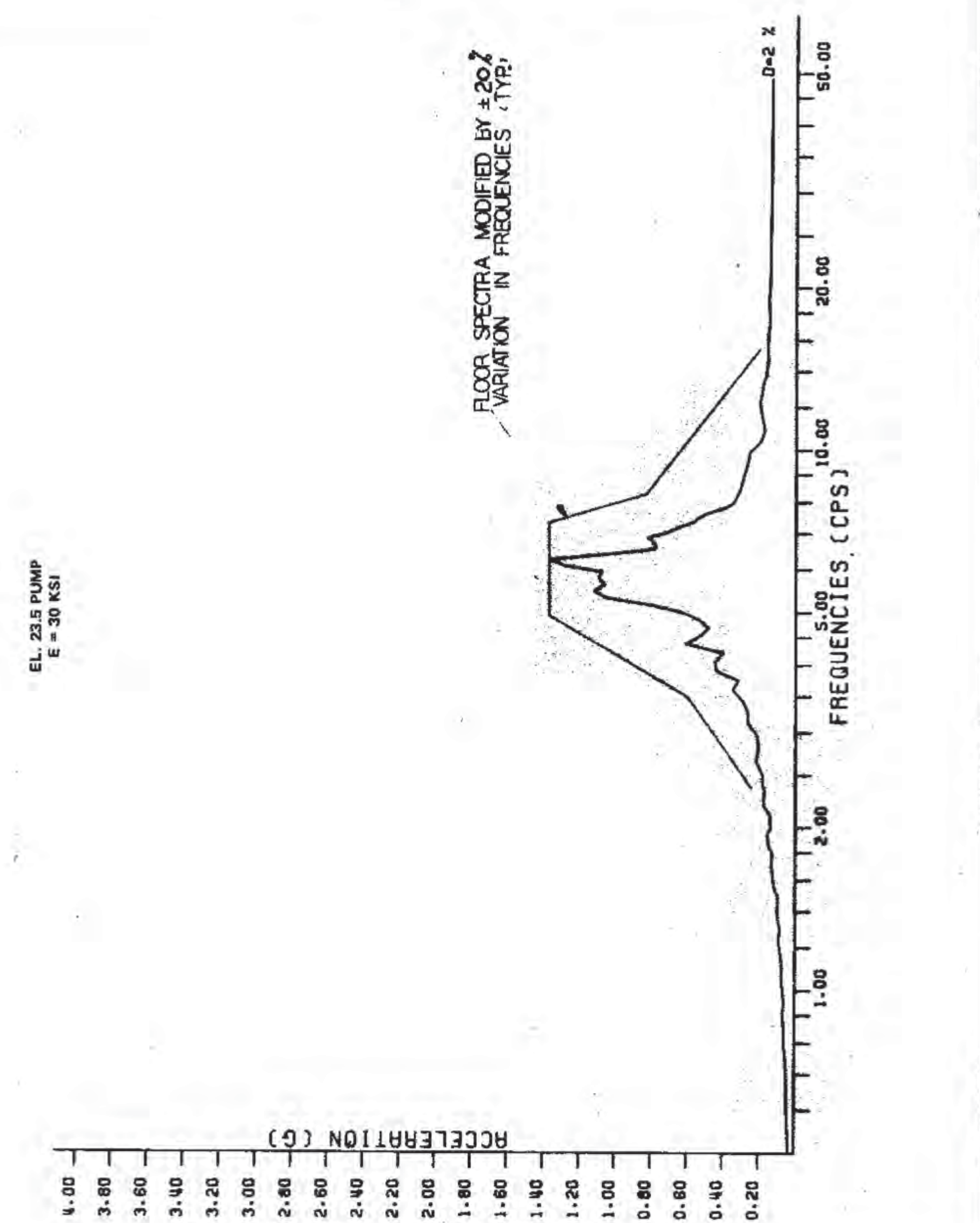
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
COMPONENT COOLING WATER BLDG.
FIGURE 3.7-240



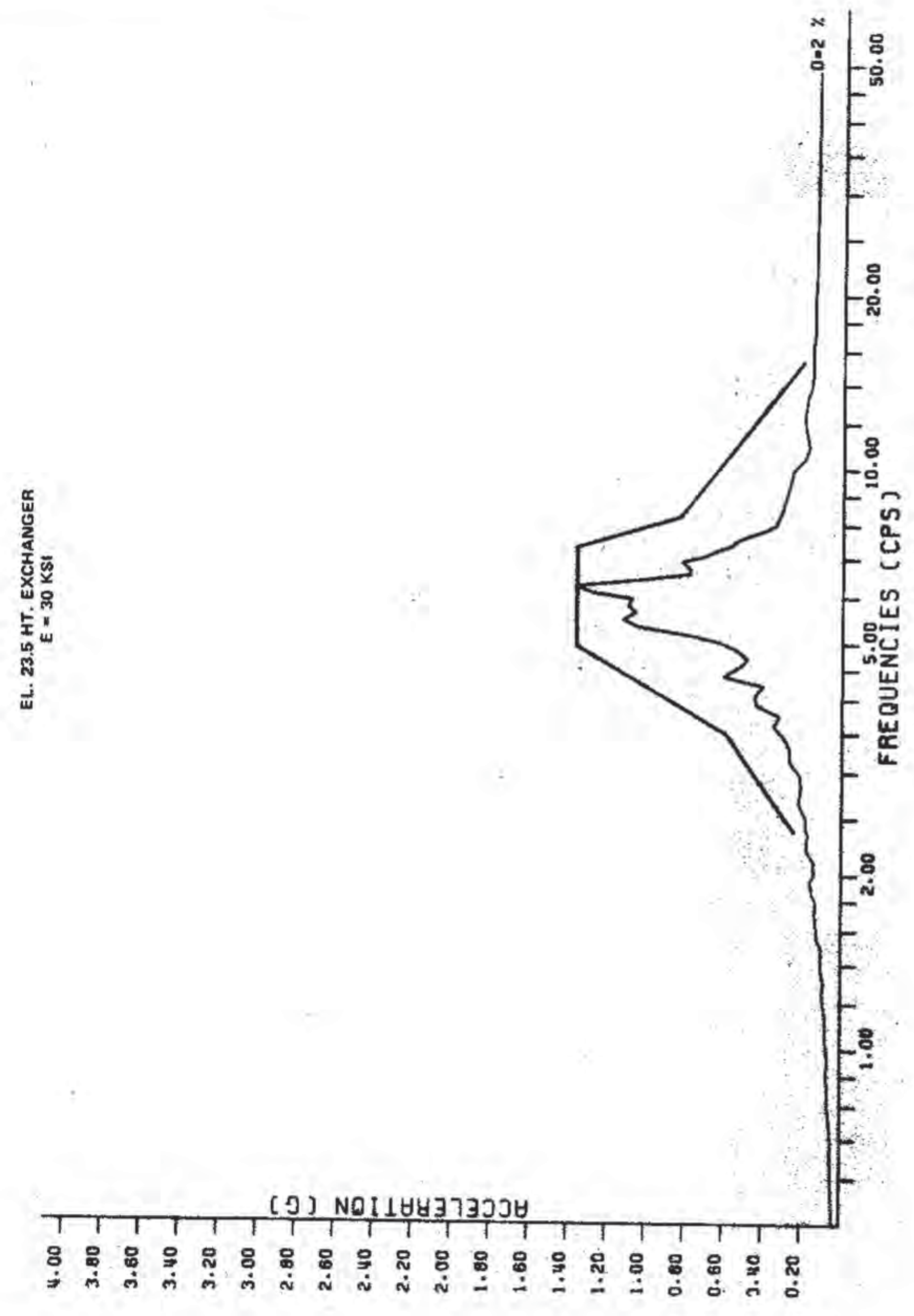
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
COMP'T COOL. WATER BLDG. MASS PT 5
FIGURE 3.7-241



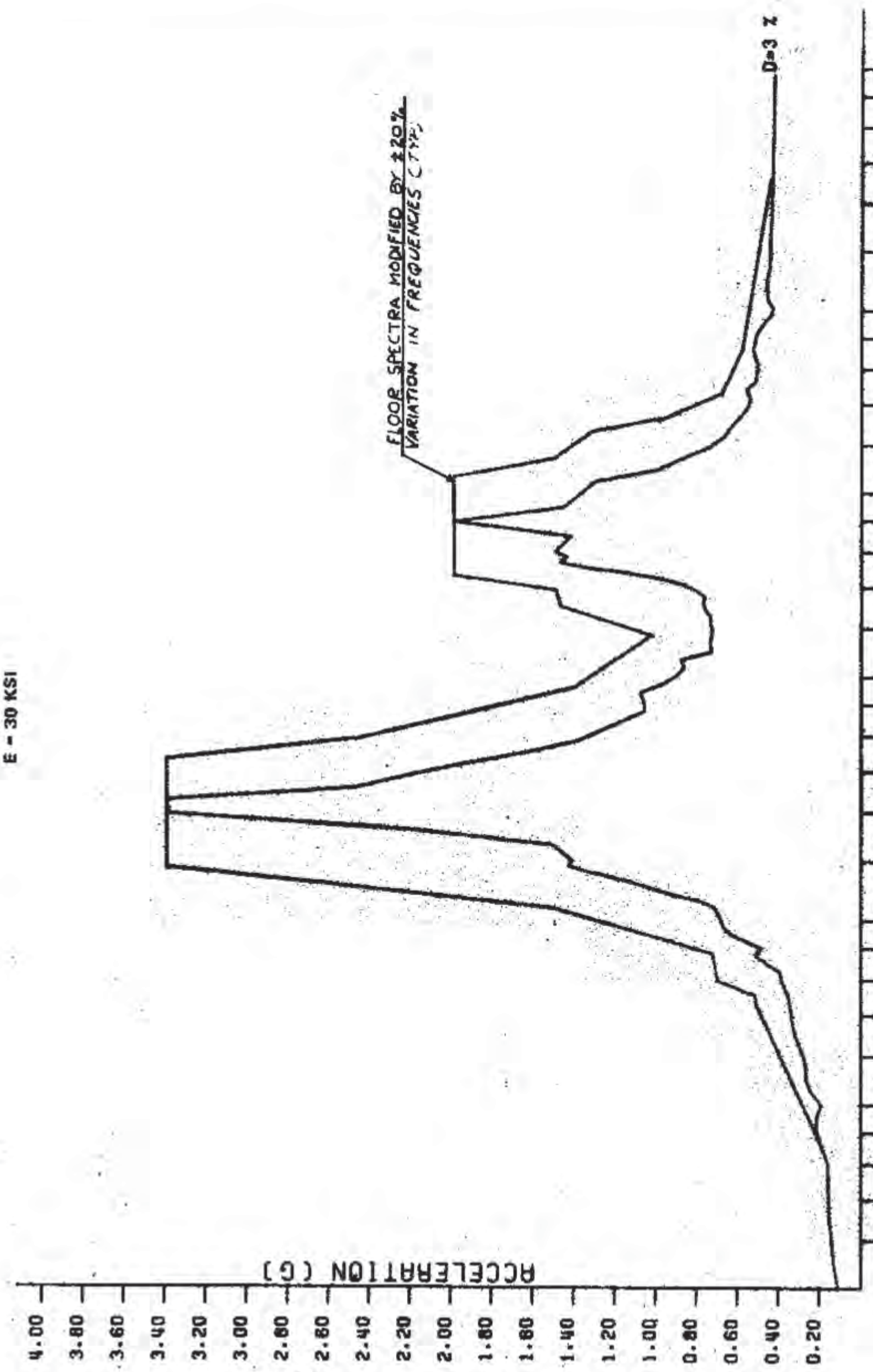
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
COMP'T COOL. WATER BLDG. MASS PT 6
FIGURE 3.7-242



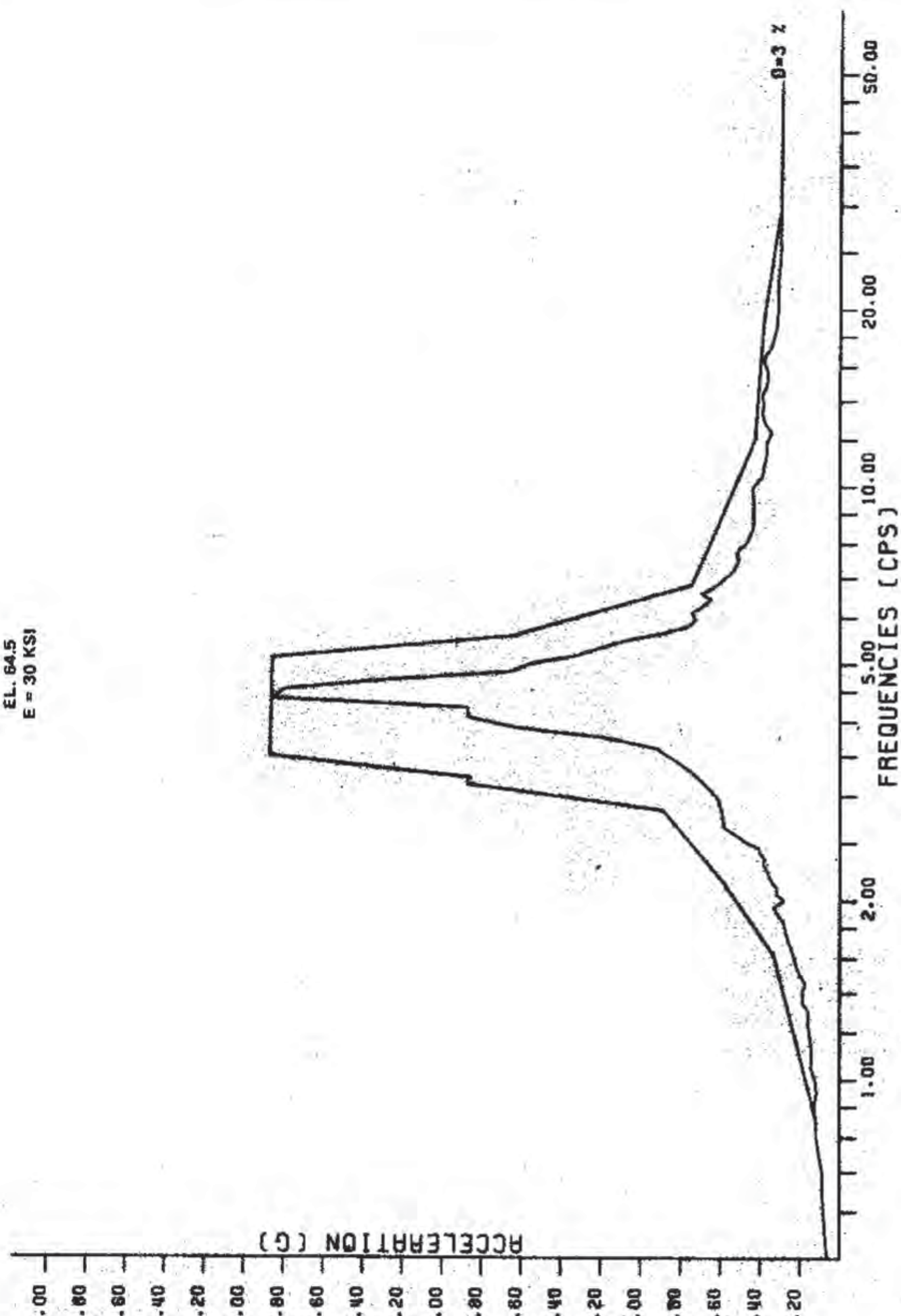
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
COMPONENT COOLING WATER BLDG.
FIGURE 3.7-243



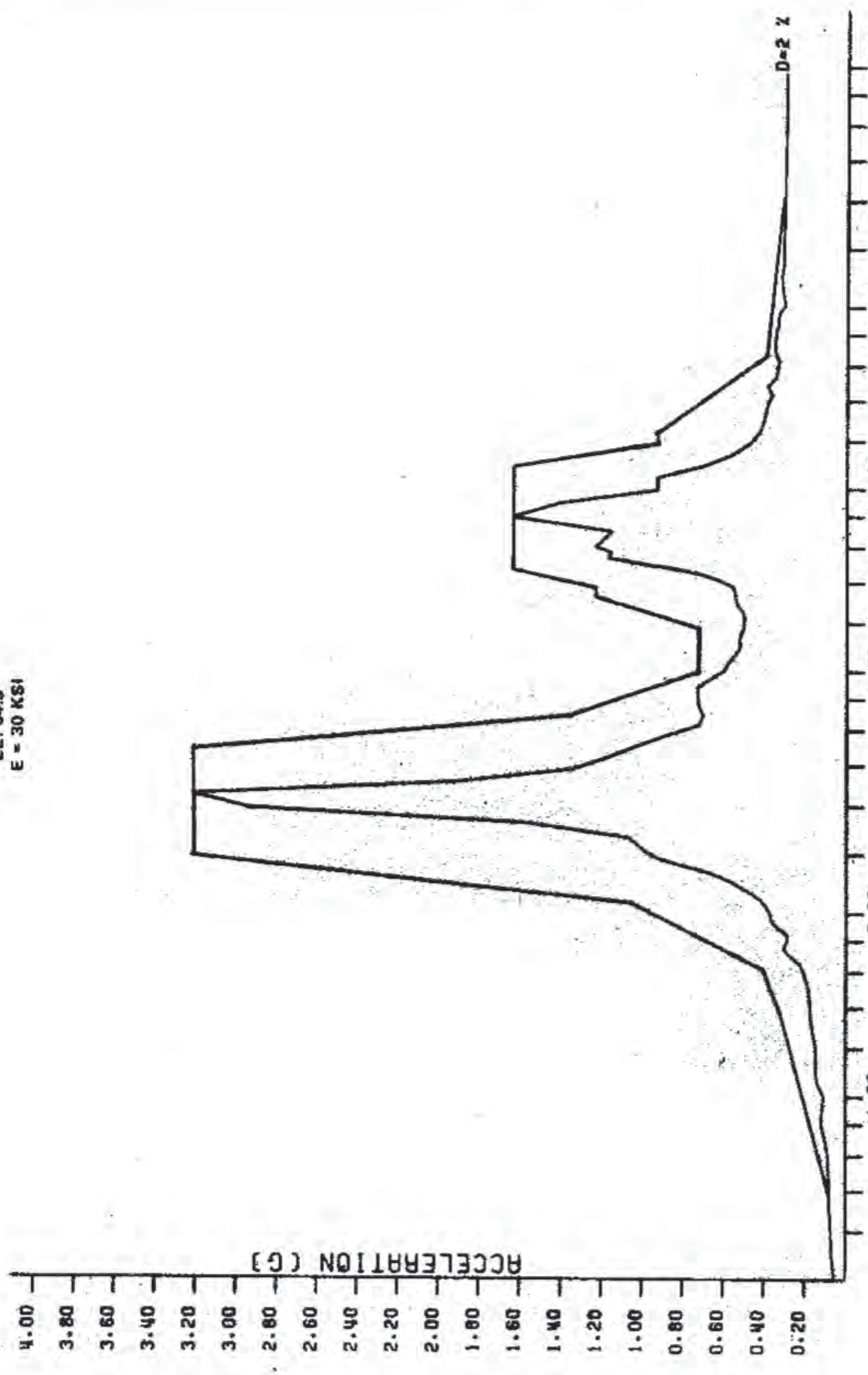
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL SSE
CONDENSATE STORAGE TANK MASS PT 9
FIGURE 3.7-244



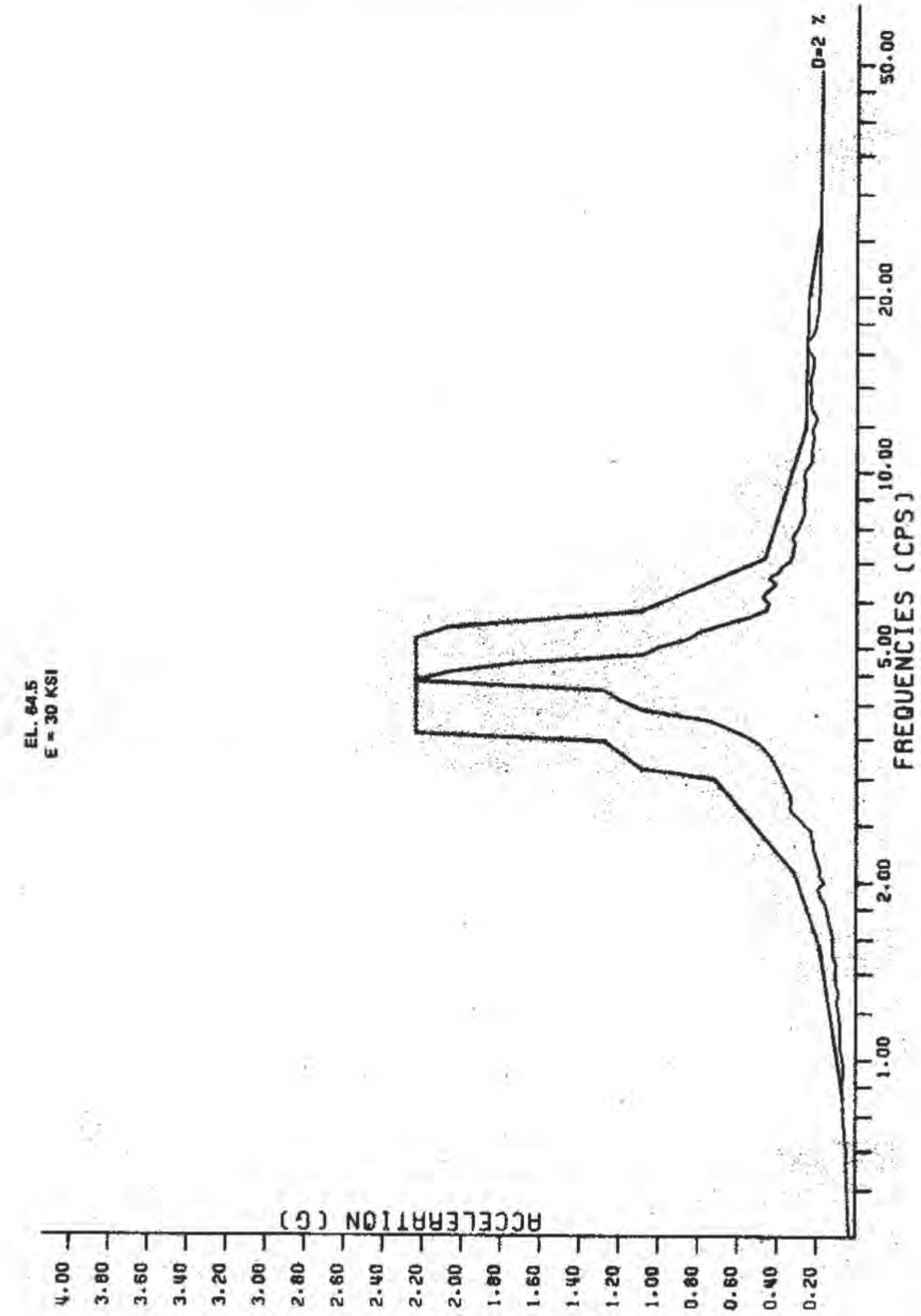
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL SSE
CONDENSATE STORAGE TANK MASS PT 9
FIGURE 3.7-245



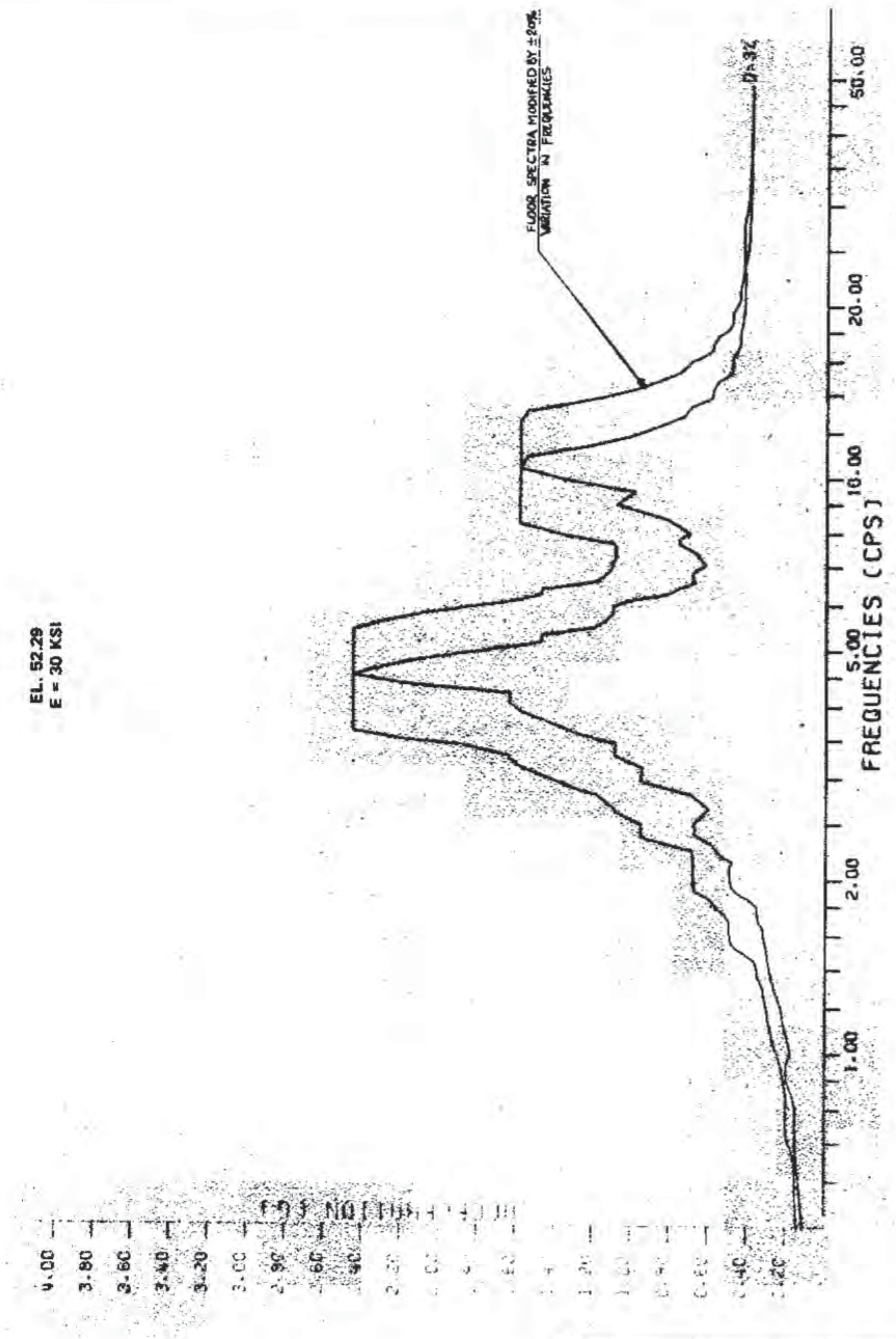
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL OBE
CONDENSATE STORAGE TANK MASS PT 9
FIGURE 3.7-246



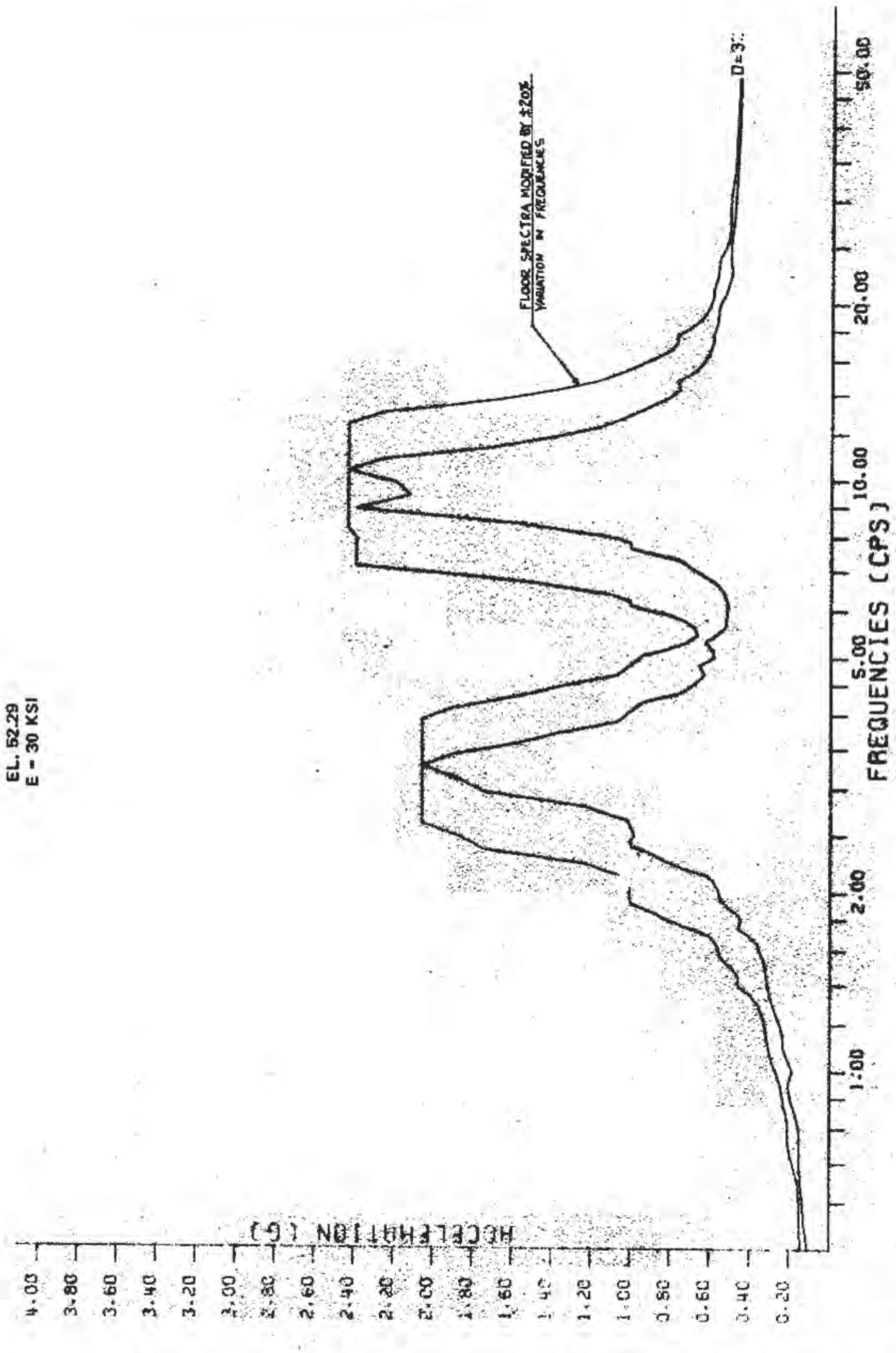
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA VERTICAL OBE
CONDENSATE STORAGE TANK MASS PT 9
FIGURE 3.7-247



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

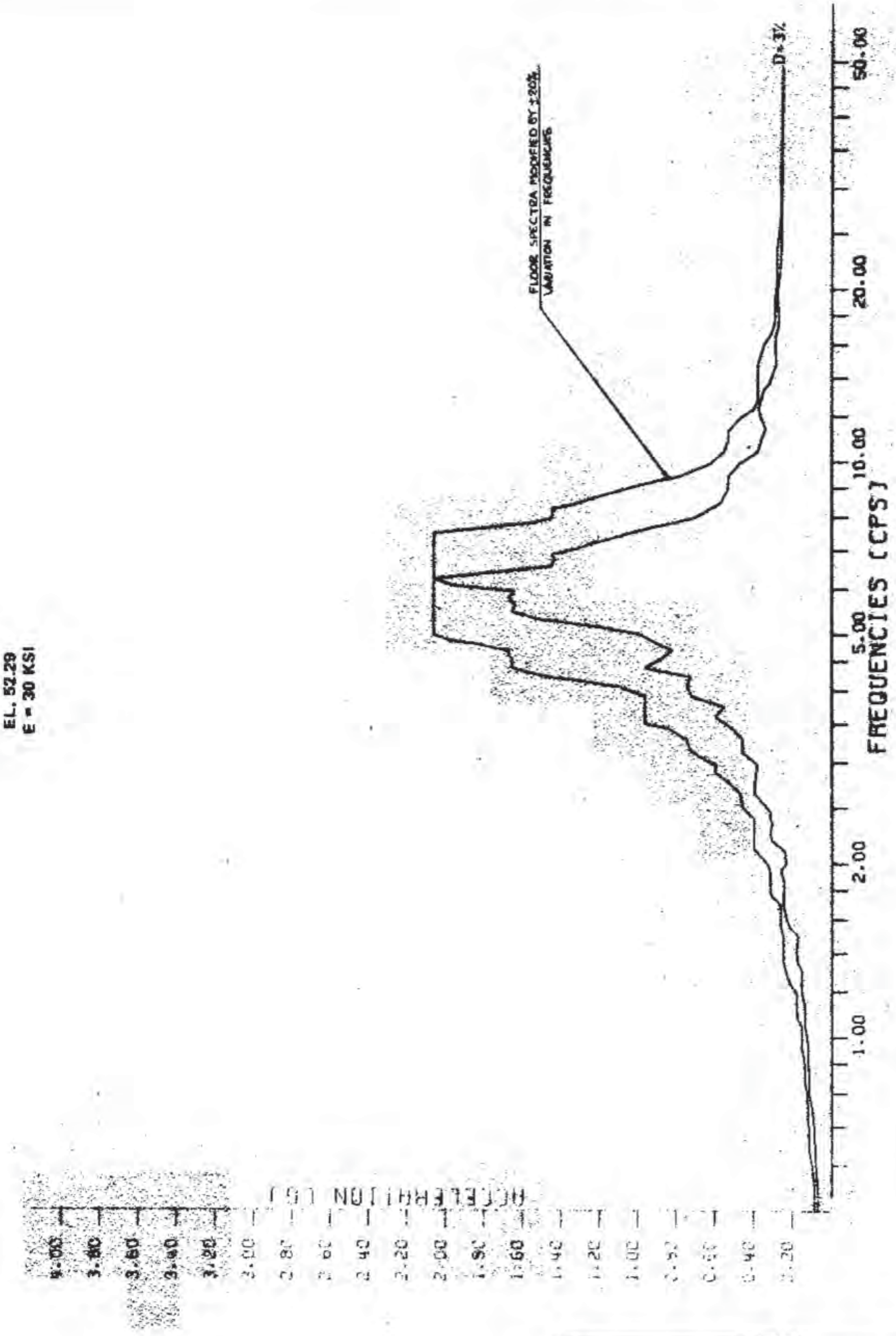
FLOOR SPECTRA HORIZONTAL N-S SSE
DIESEL OIL STG MASS PT 3 & 8
FIGURE 3.7-248



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W SSE
DIESEL OIL STG MASS PT 3 & 8

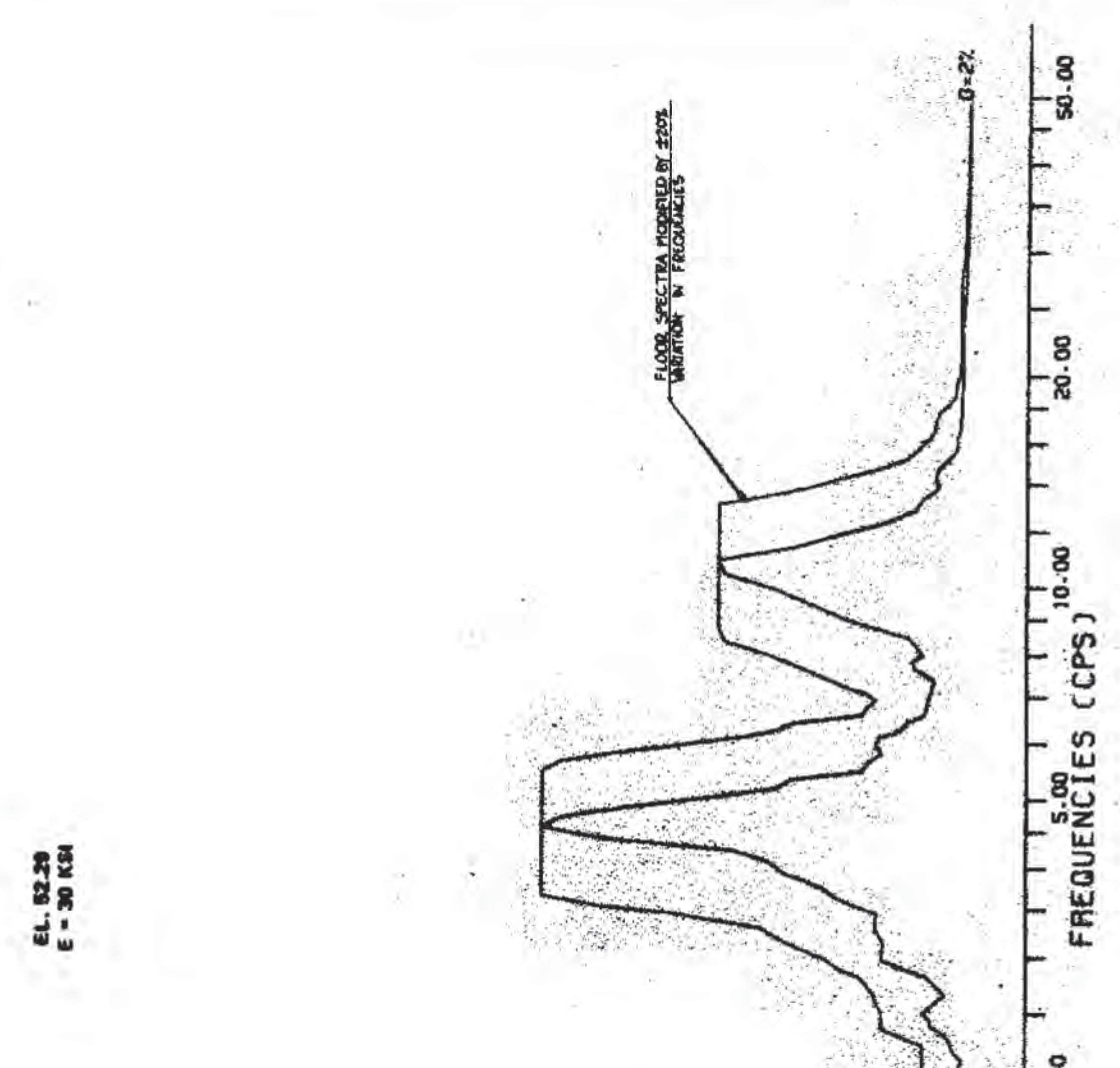
FIGURE 3.7-249



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

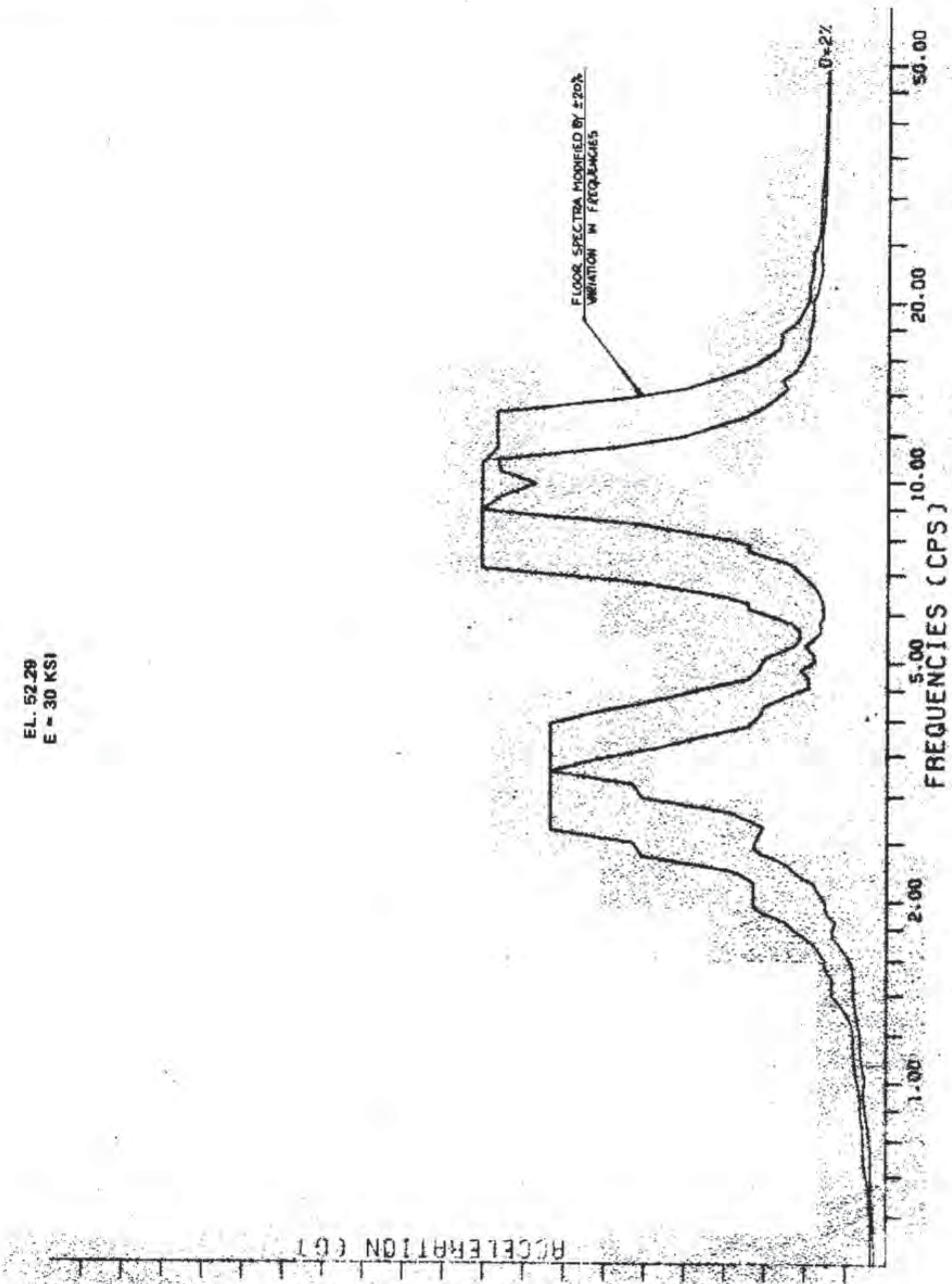
FLOOR SPECTRA VERTICAL SSE
DIESEL OIL STG MASS PT 3 & 8

FIGURE 3.7-250



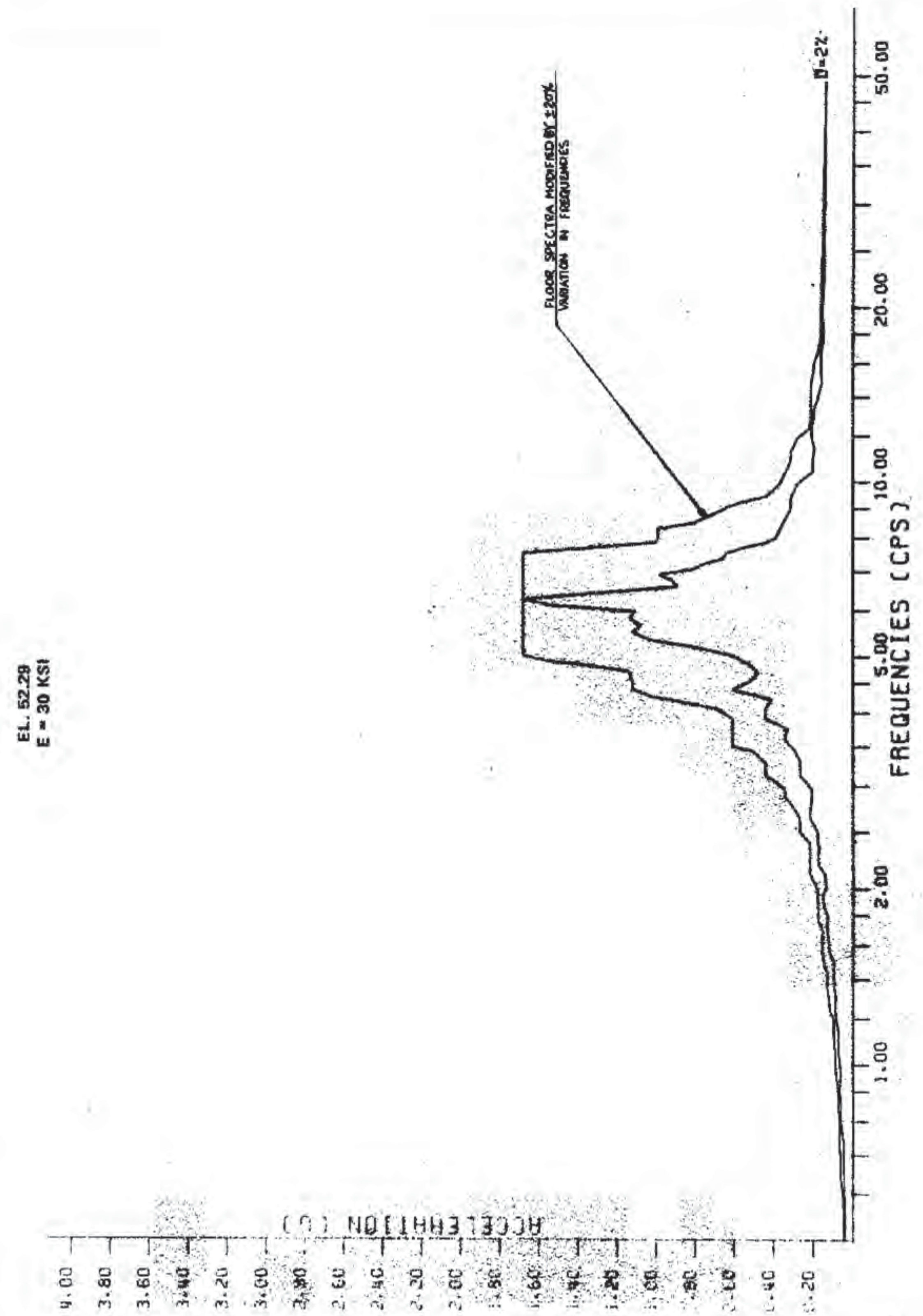
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL N-S OBE
DIESEL OIL STG MASS PT 3 & 8
FIGURE 3.7-251



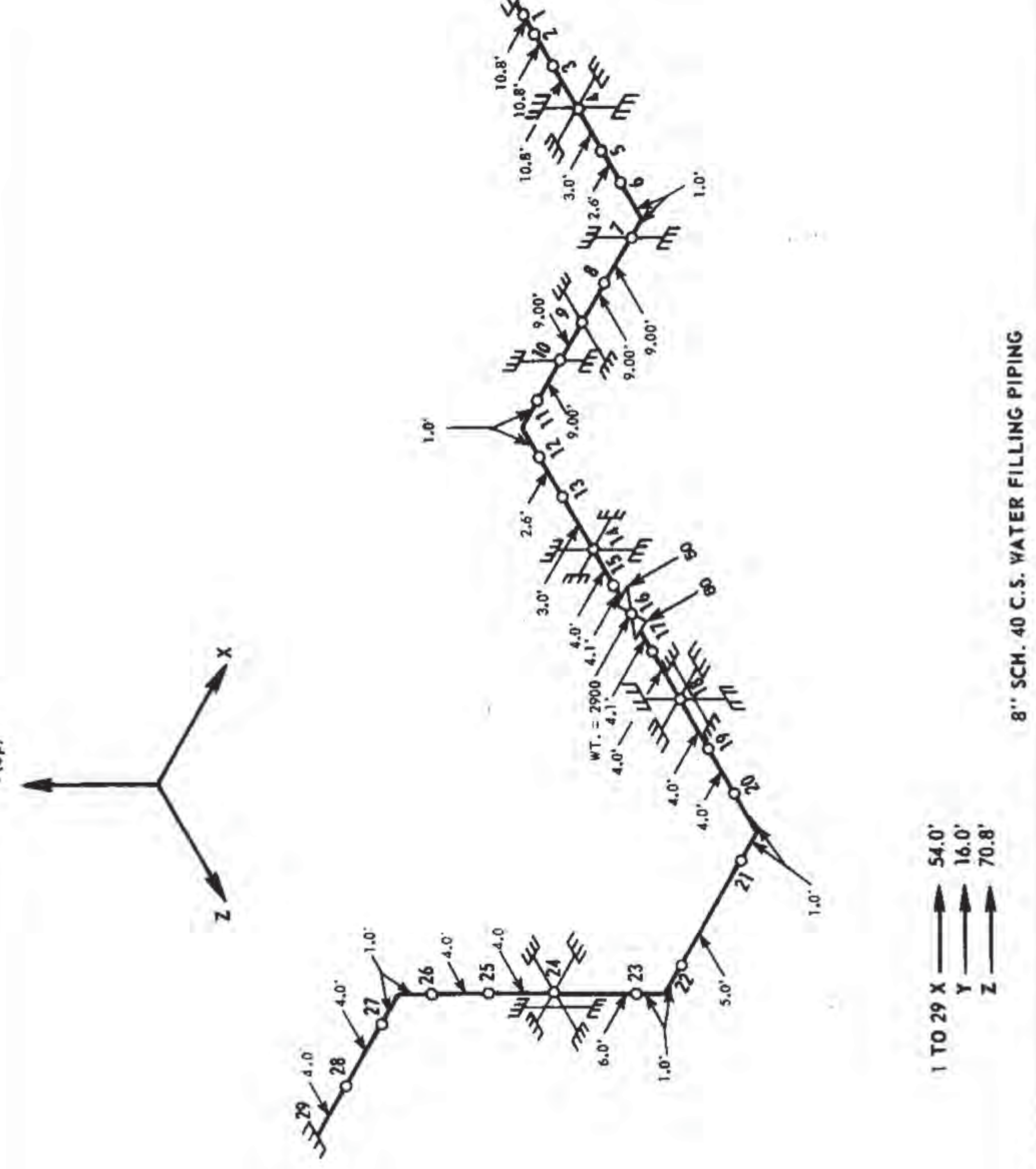
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOOR SPECTRA HORIZONTAL E-W OBE
DIESEL OIL STG MASS PT 3 & 8
FIGURE 3.7-252



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

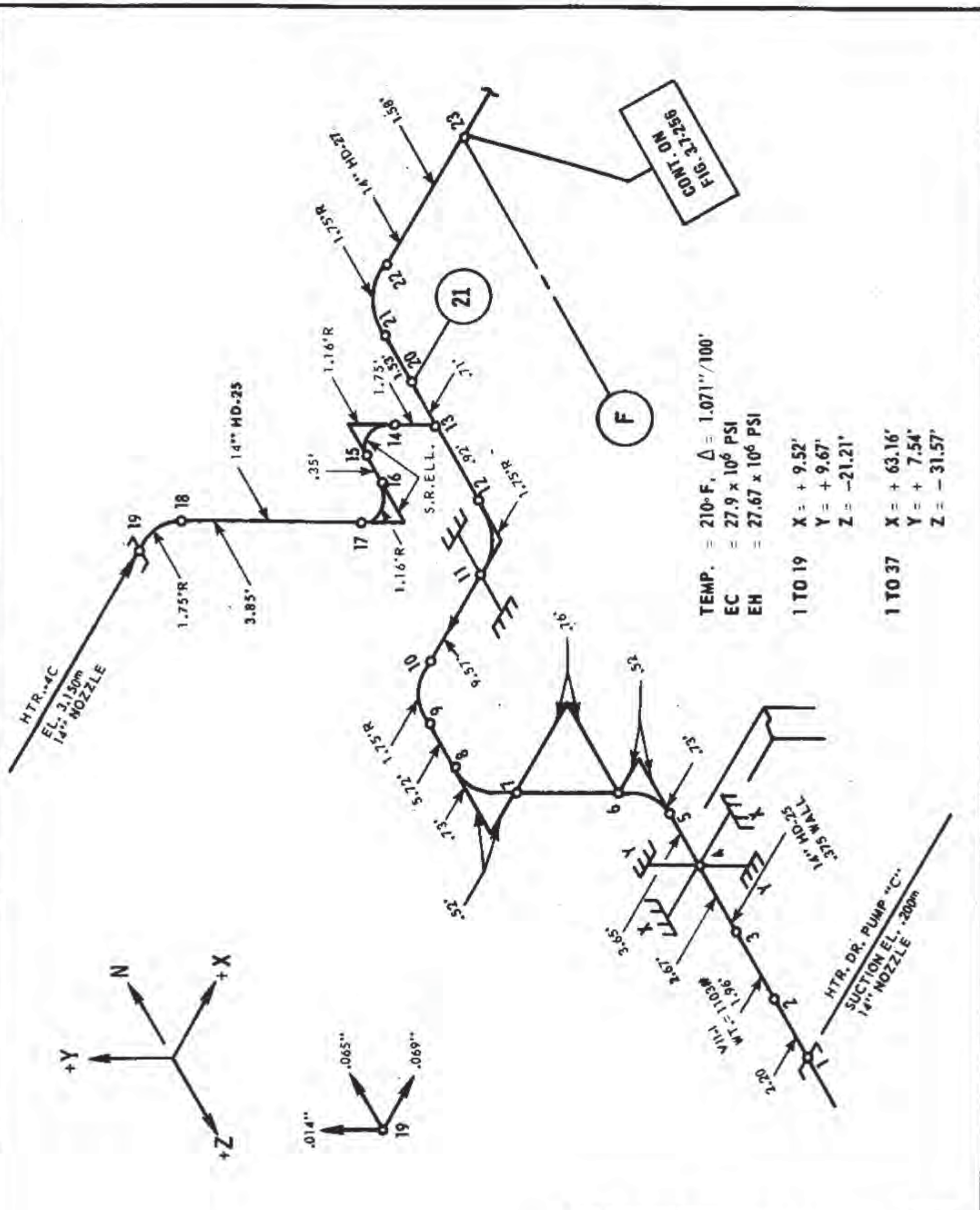
FLOOR SPECTRA VERTICAL OBE
DIESEL OIL STG MASS PT 3 & 8
FIGURE 3.7-253



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

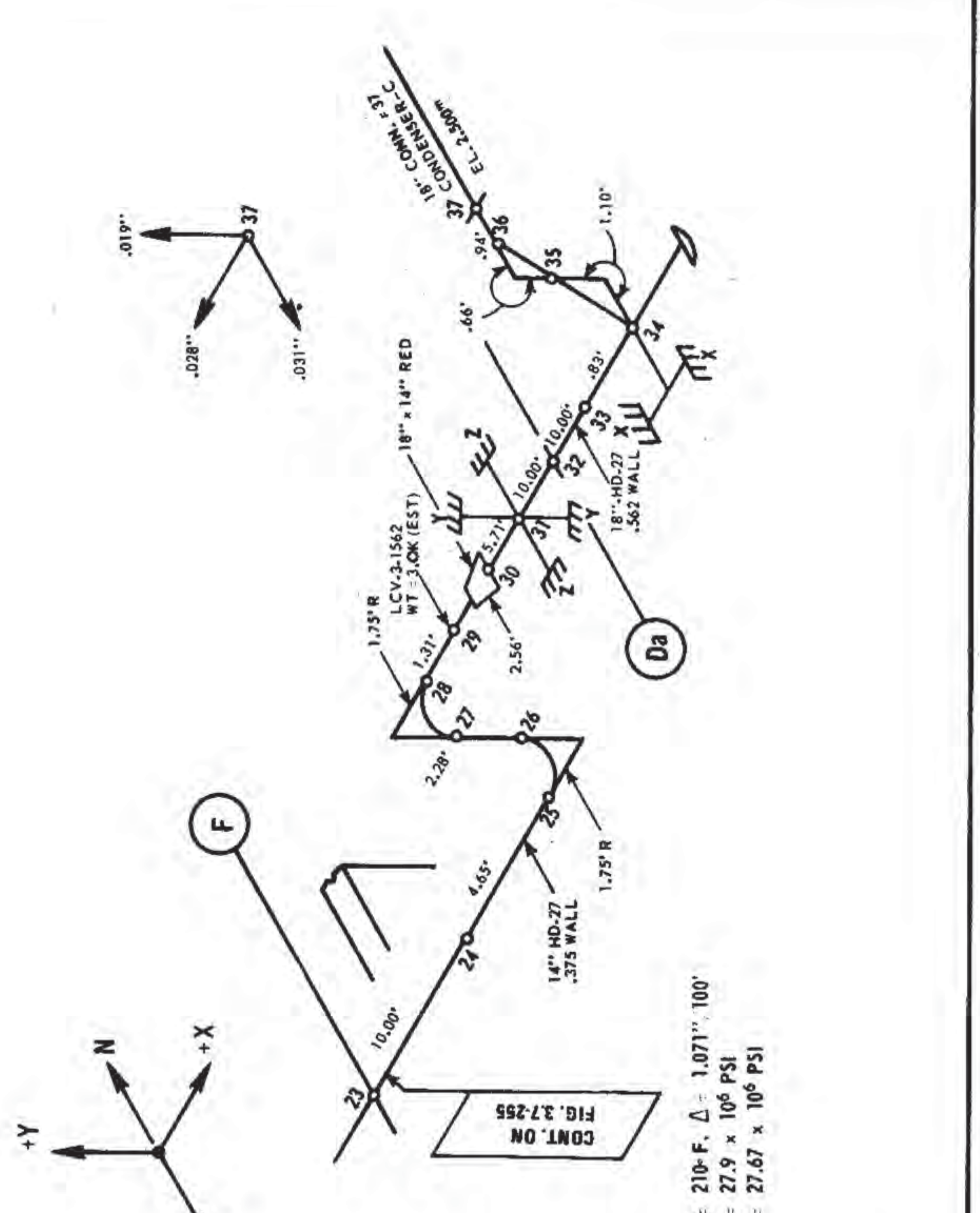
SEISMIC PROTECTION ANALYSIS
SAMPLE PROBLEM NO. 1

FIGURE 3.7-254



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

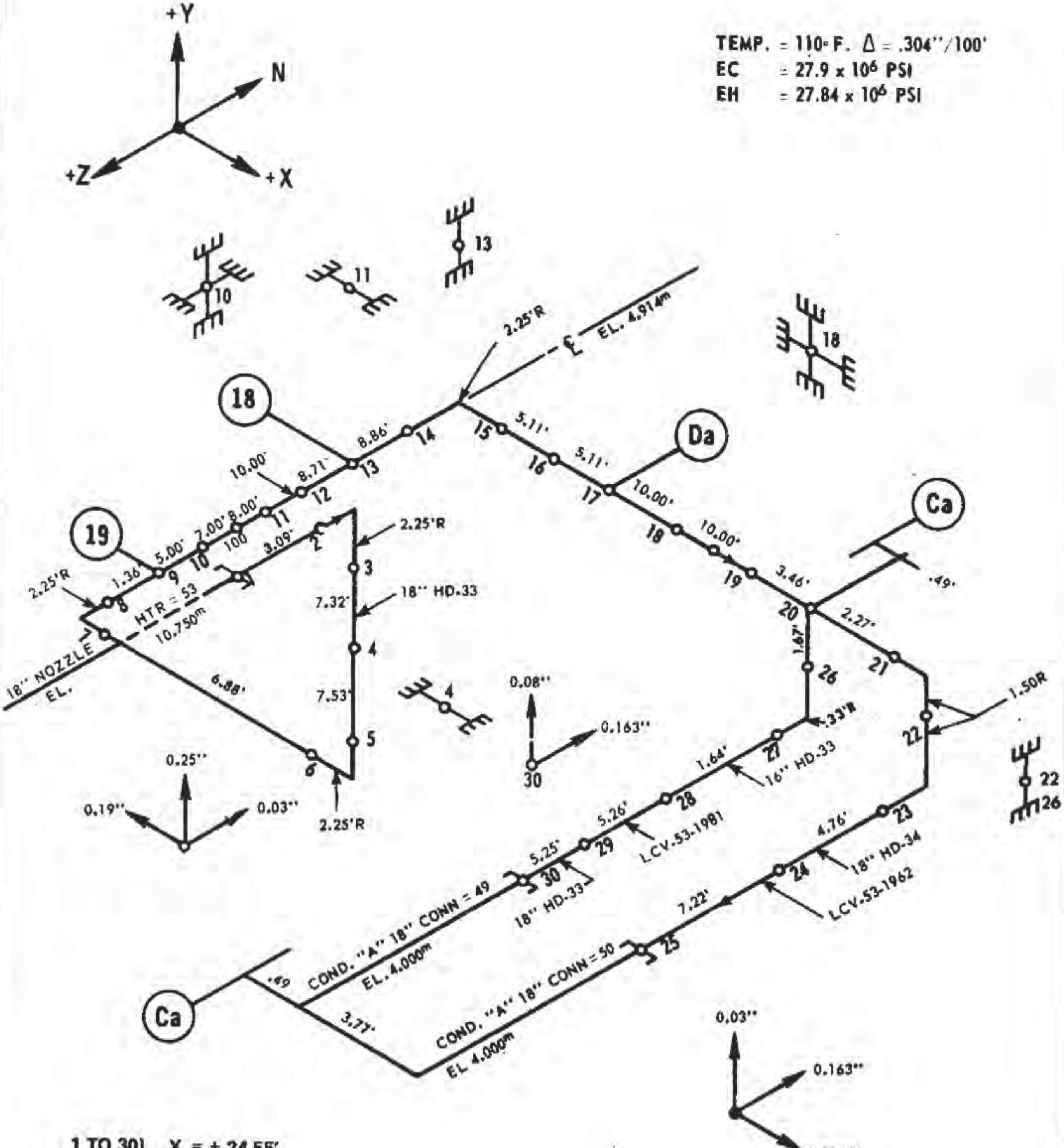
SEISMIC PROTECTION ANALYSIS
SAMPLE PROBLEM NO. 2
SHEET 1
FIGURE 3.7-255



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

SEISMIC PROTECTION ANALYSIS
SAMPLE PROBLEM NO. 2
SHEET 2

FIGURE 3.7-256

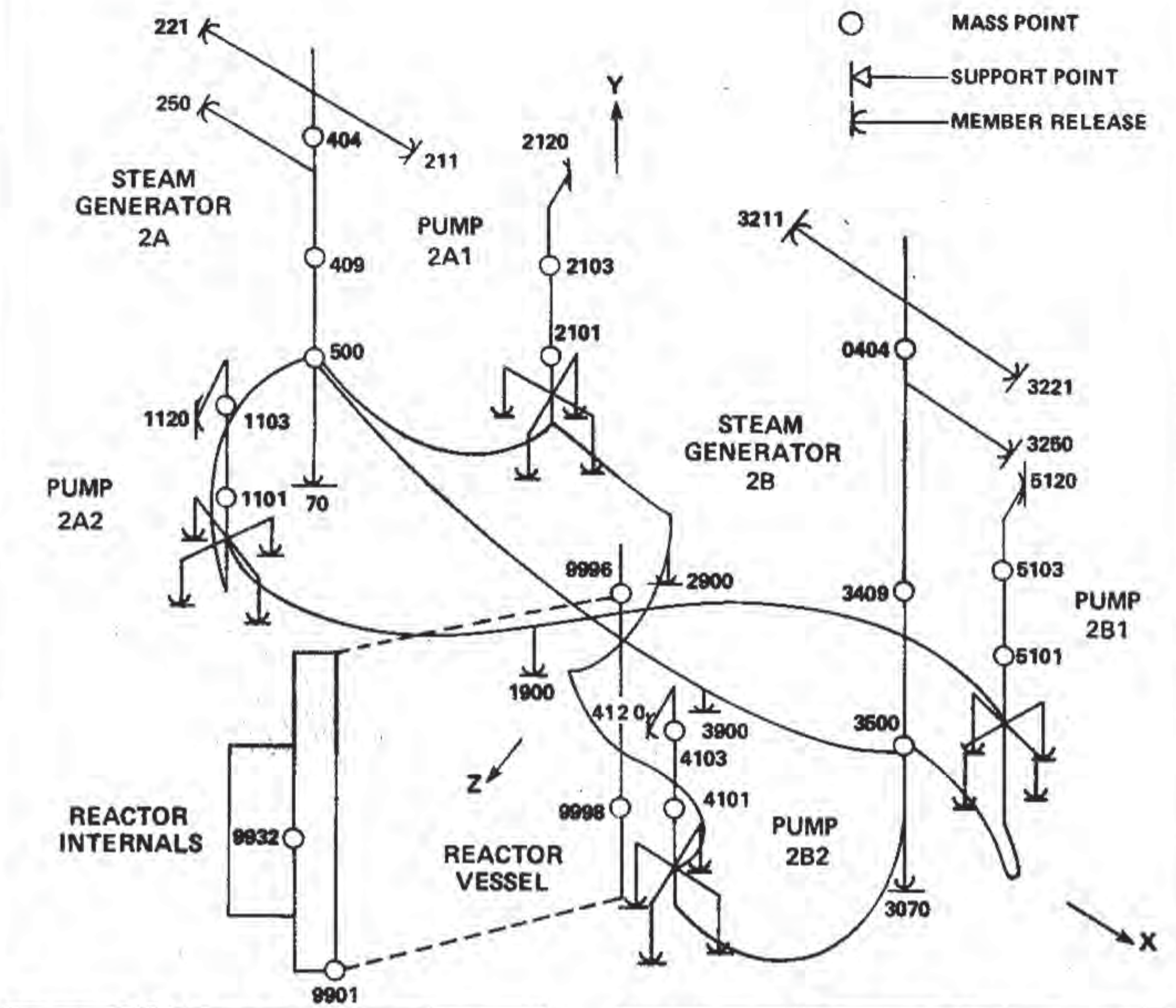


1 TO 30) X = + 24.55'
Y = − 22.35'
Z = − 45.28

1 TO 25) X = + 28.32'
Y = − 22.35'
Z = − 45.12'

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

SEISMIC PROTECTION ANALYSIS
SAMPLE PROBLEM NO. 3
FIGURE 3.7-257



COMPONENT NAME	MASS POINT NAME	DEGREES OF FREEDOM
REACTOR	9996,9901,9932	X, Y, Z
	9998	X, Z
STEAM GENERATOR	404, 3404	X, Y, Z
	409, 3409 500, 3500	X, Z
REACTOR COOLANT PUMPS	1103, 2103	X, Y, Z
	4103, 5103	
	1101, 2101	X, Y
	4101, 5101	

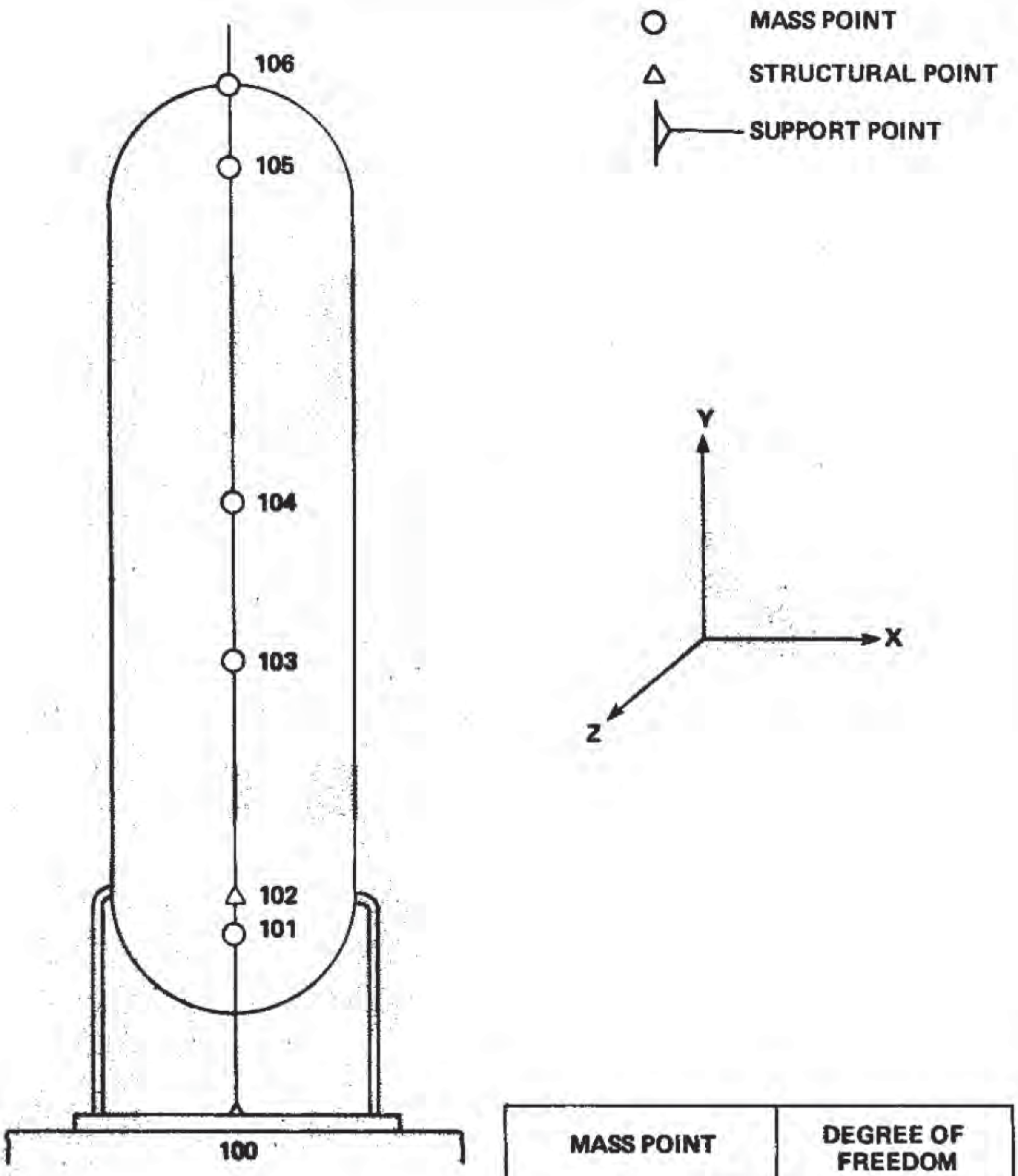
* FH RESTRAINTS ARE PERPENDICULAR TO THE REACTOR VESSEL RADIAL DIRECTION IN THE HORIZONTAL PLANE

COMPONENT NAME	SUPPORT NAME	RESTRAINT DIRECTIONS
REACTOR	3900	Fy, Fz
	1900, 2900	Fy, FH*
STEAM GENERATOR	70, 3070	Fy, Fz, My
	250, 3250	Fx
	211, 221	Fz
	3211, 3221	
REACTOR COOLANT PUMPS	VERTICAL SUPPORTS (4 PLACES ON EACH PUMP)	Fy
	SNUBBERS	Fx, Fz

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TYPICAL REACTOR COOLANT SYSTEM SEISMIC ANALYSIS MODEL

FIGURE 3.7-258

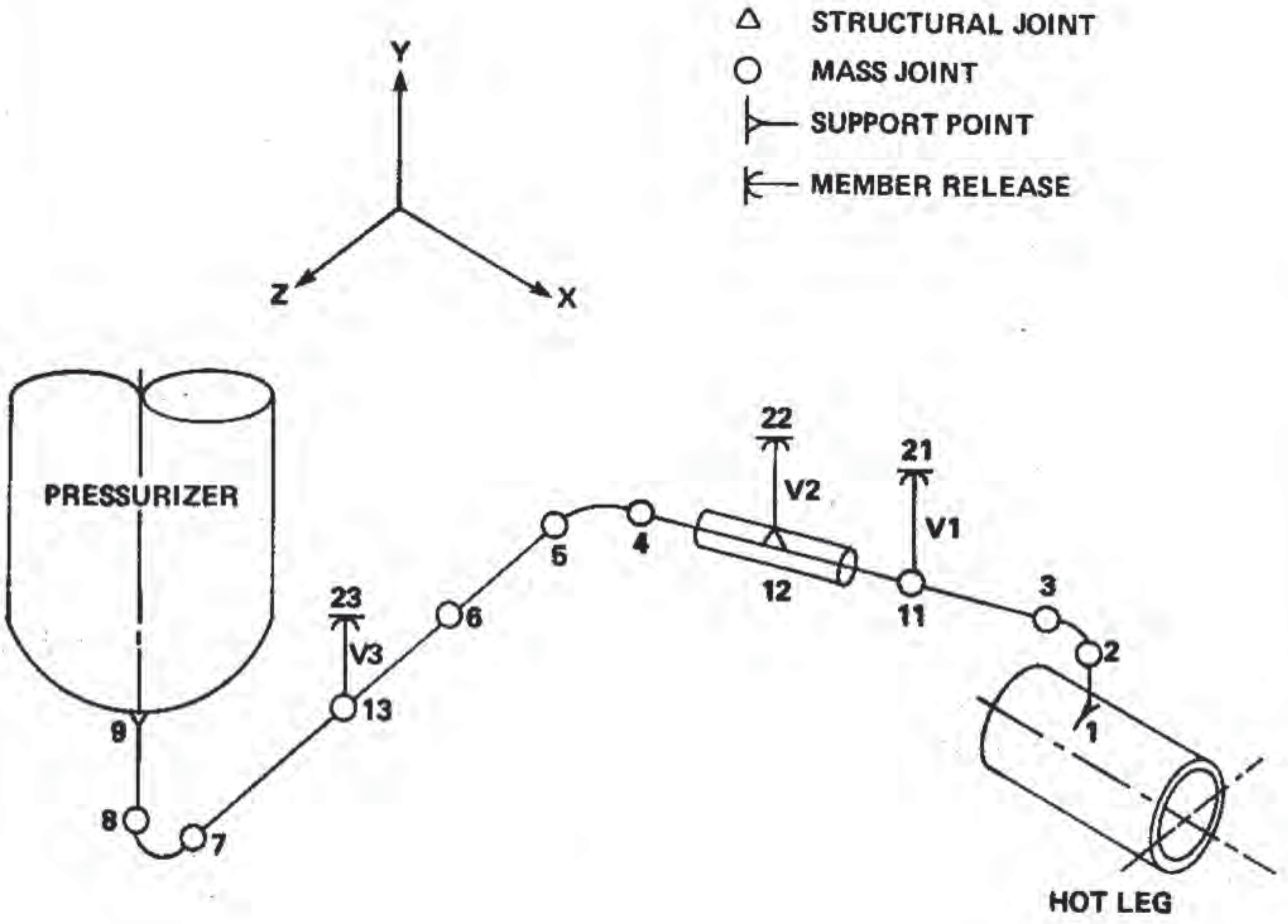


MASS POINT	DEGREE OF FREEDOM
101, 103 TO 105	X, Z,
106	X, Y, Z

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PRESSURIZER SEISMIC ANALYSIS MODEL

FIGURE 3.7-259



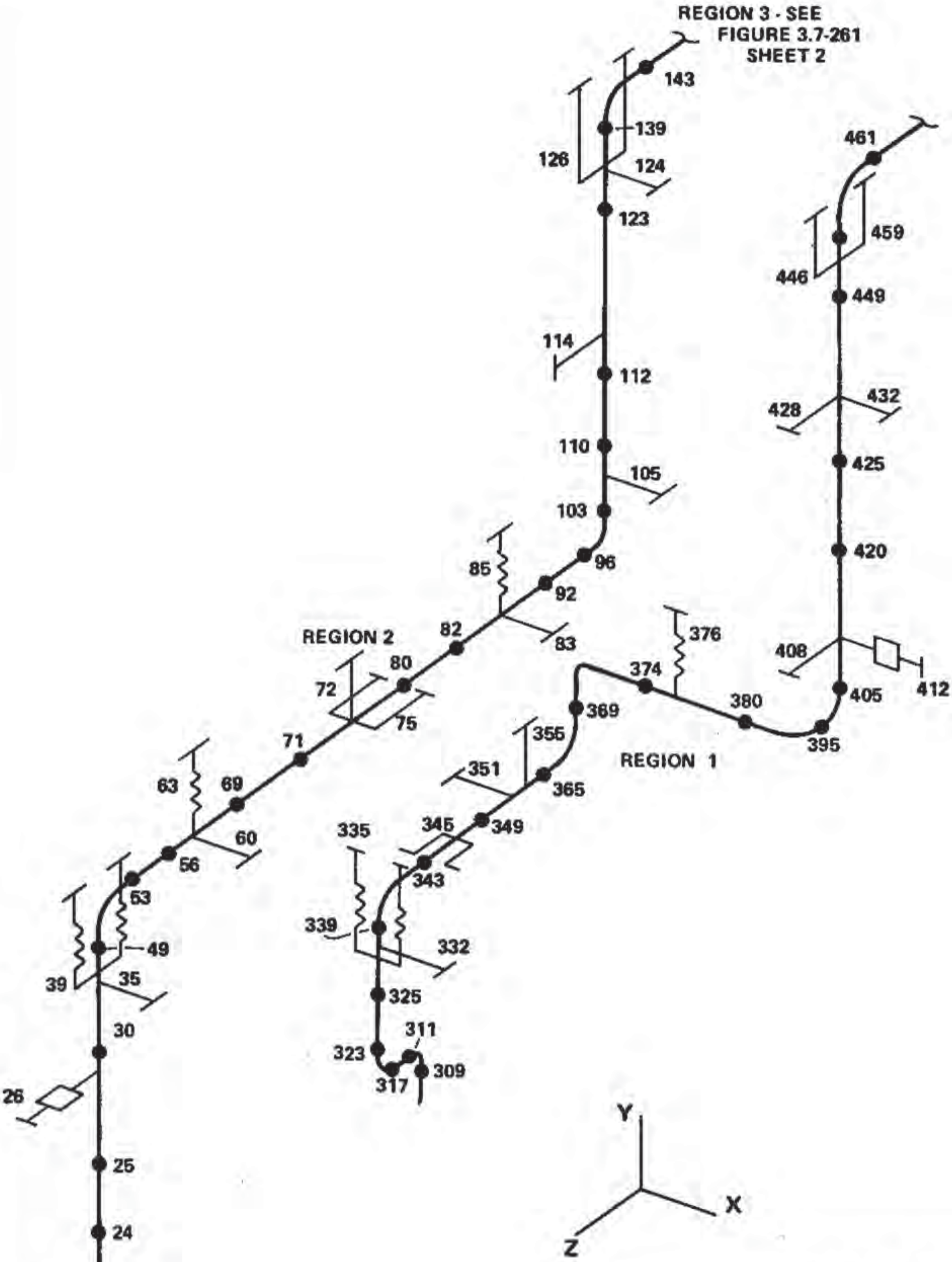
MASS POINT	DEGREE OF FREEDOM
2-8, 11, 13	X, Y, Z

SUPPORT NAME	RESTRAINT DIRECTION
21, 22, 23	Y

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

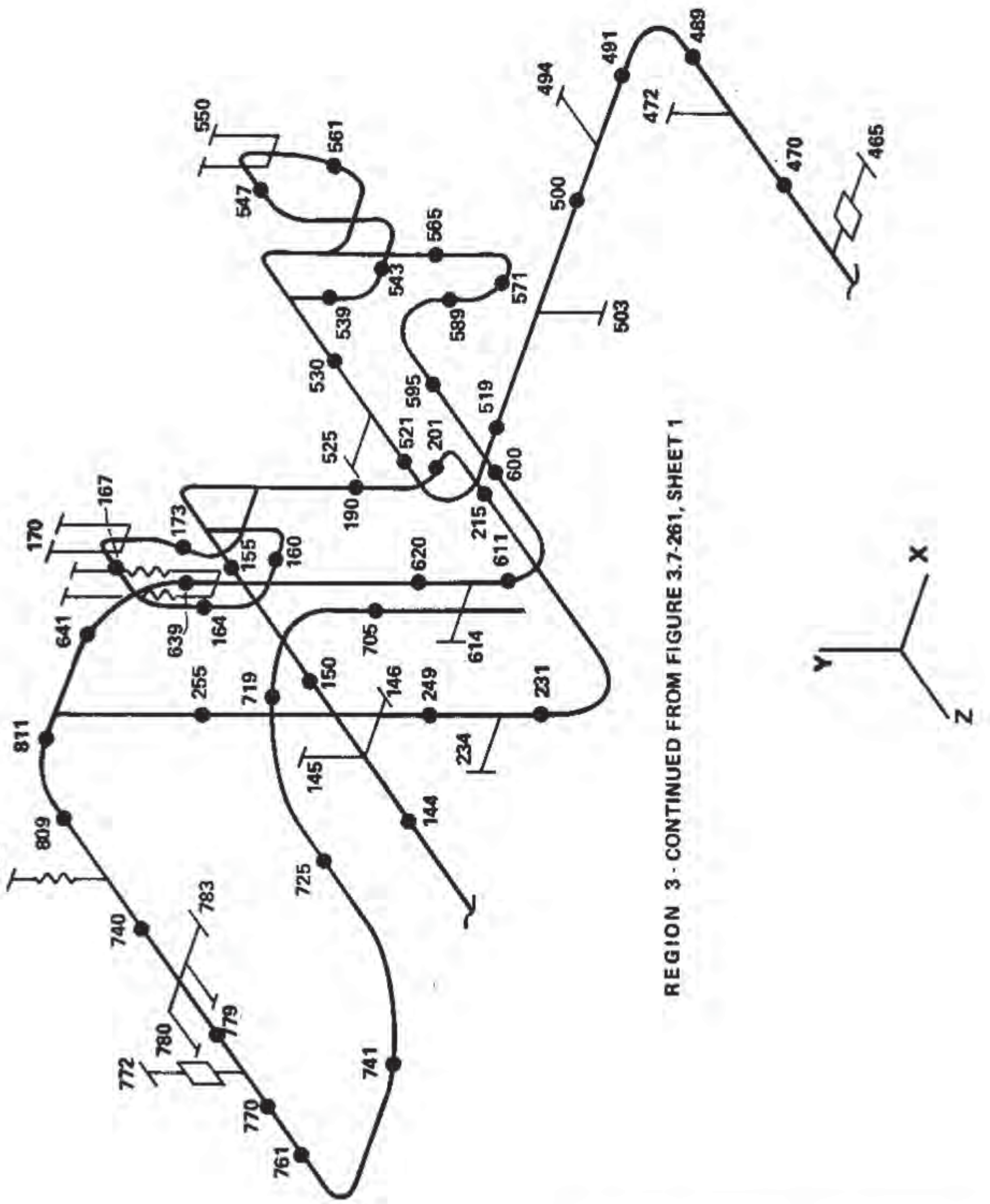
SURGE LINE SEISMIC
ANALYSIS MODEL

FIGURE 3.7-260



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

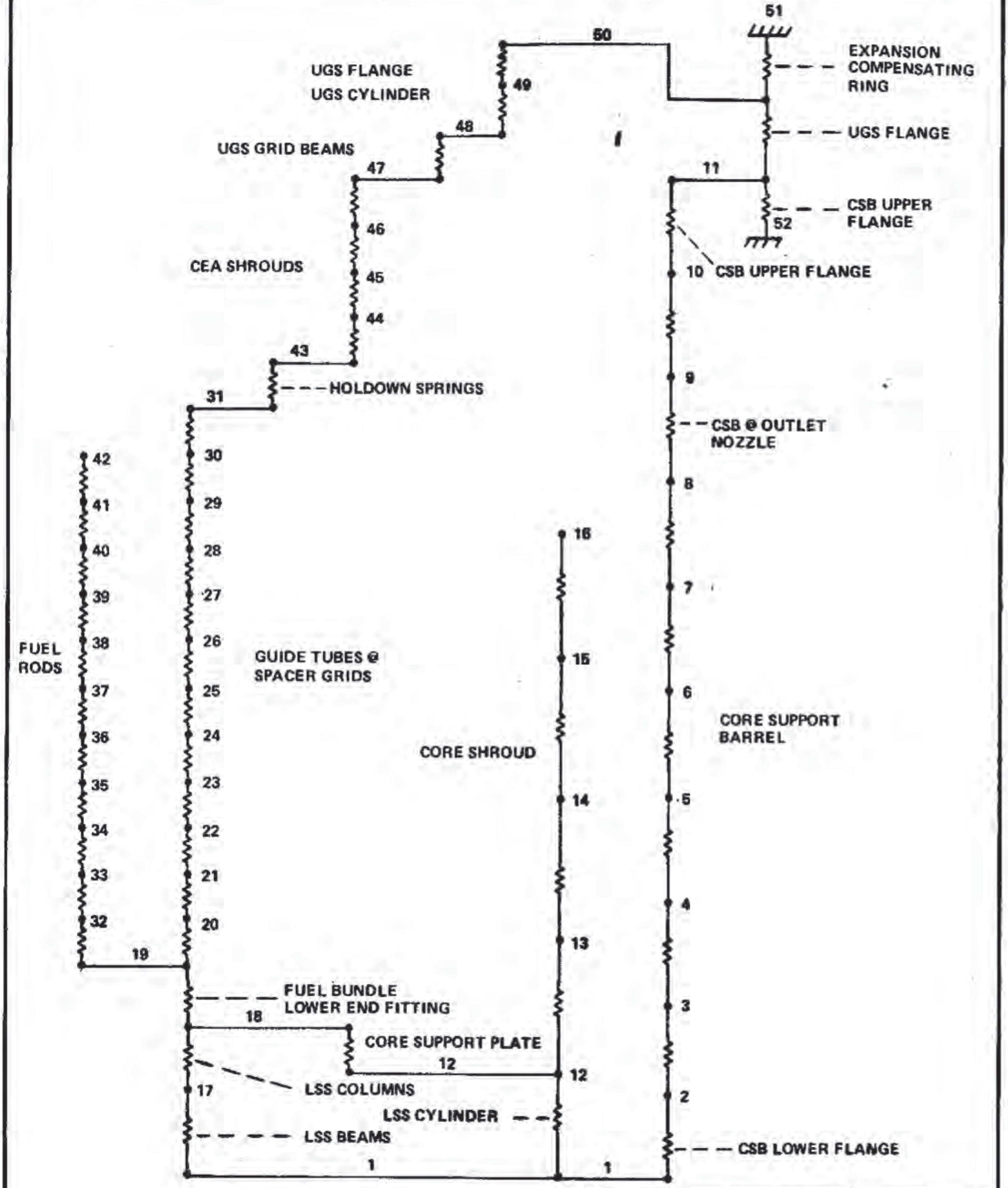
PRESSURIZER SPRAY LINE
SEISMIC ANALYSIS MODEL
SHEET 1 OF 2
FIGURE 3.7–261



REGION 3 - CONTINUED FROM FIGURE 3.7-261, SHEET 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

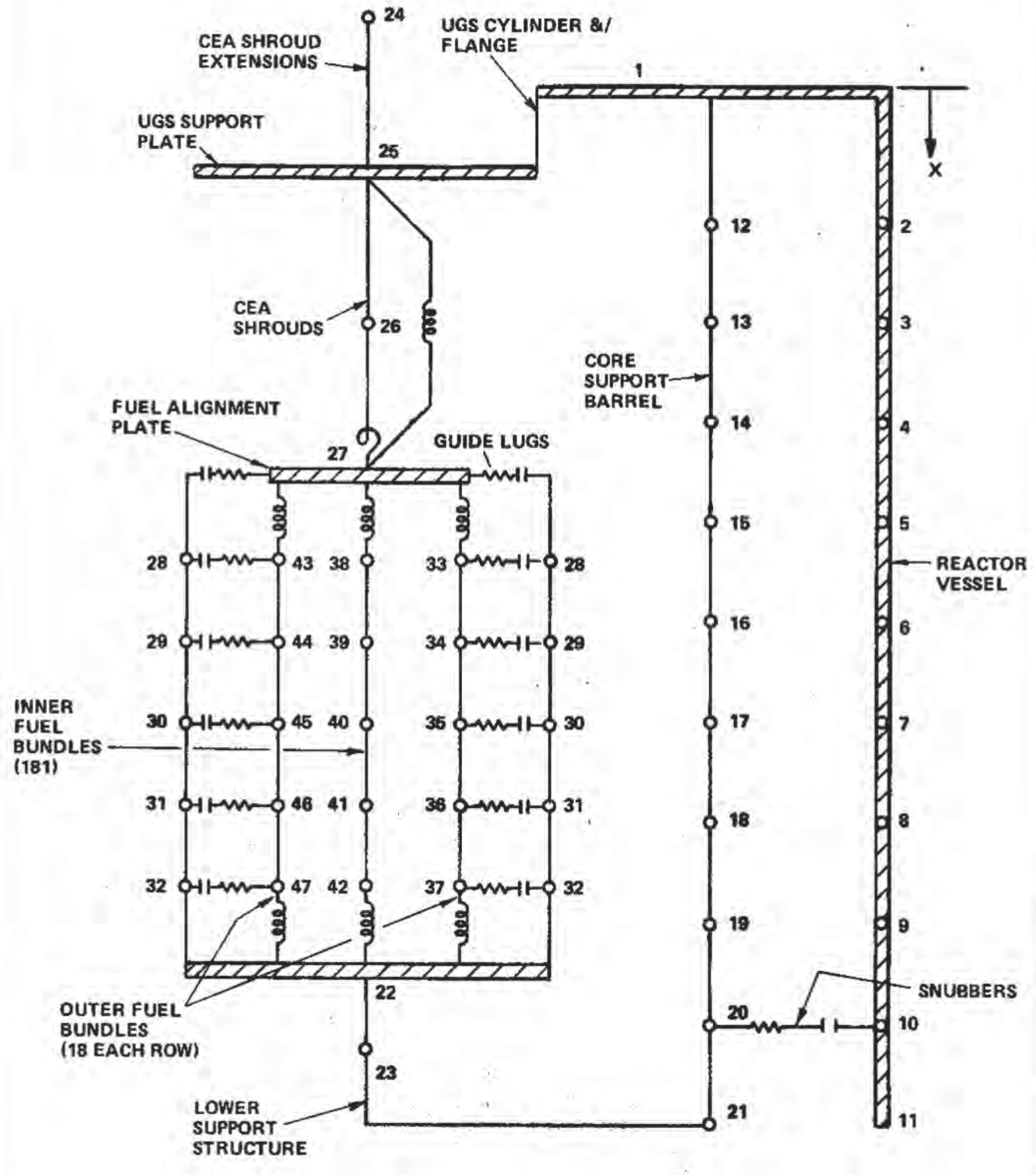
PRESSURIZER SPRAY LINE
SEISMIC ANALYSIS MODEL
SHEET 2 OF 2
FIGURE 3.7–261



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VERTICAL SEISMIC ANALYSIS MODEL

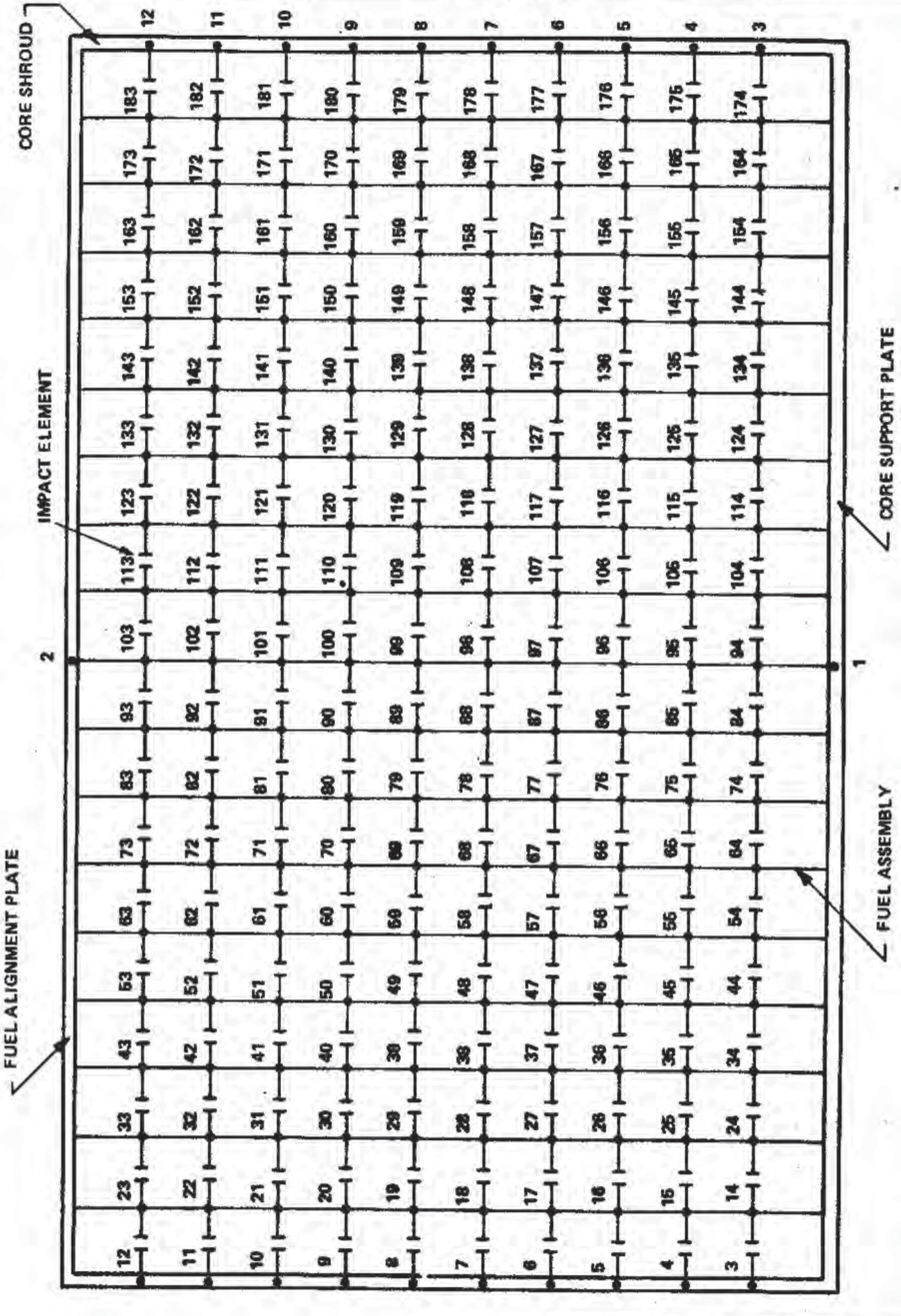
FIGURE 3.7-262



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

HORIZONTAL SEISMIC ANALYSIS MODEL

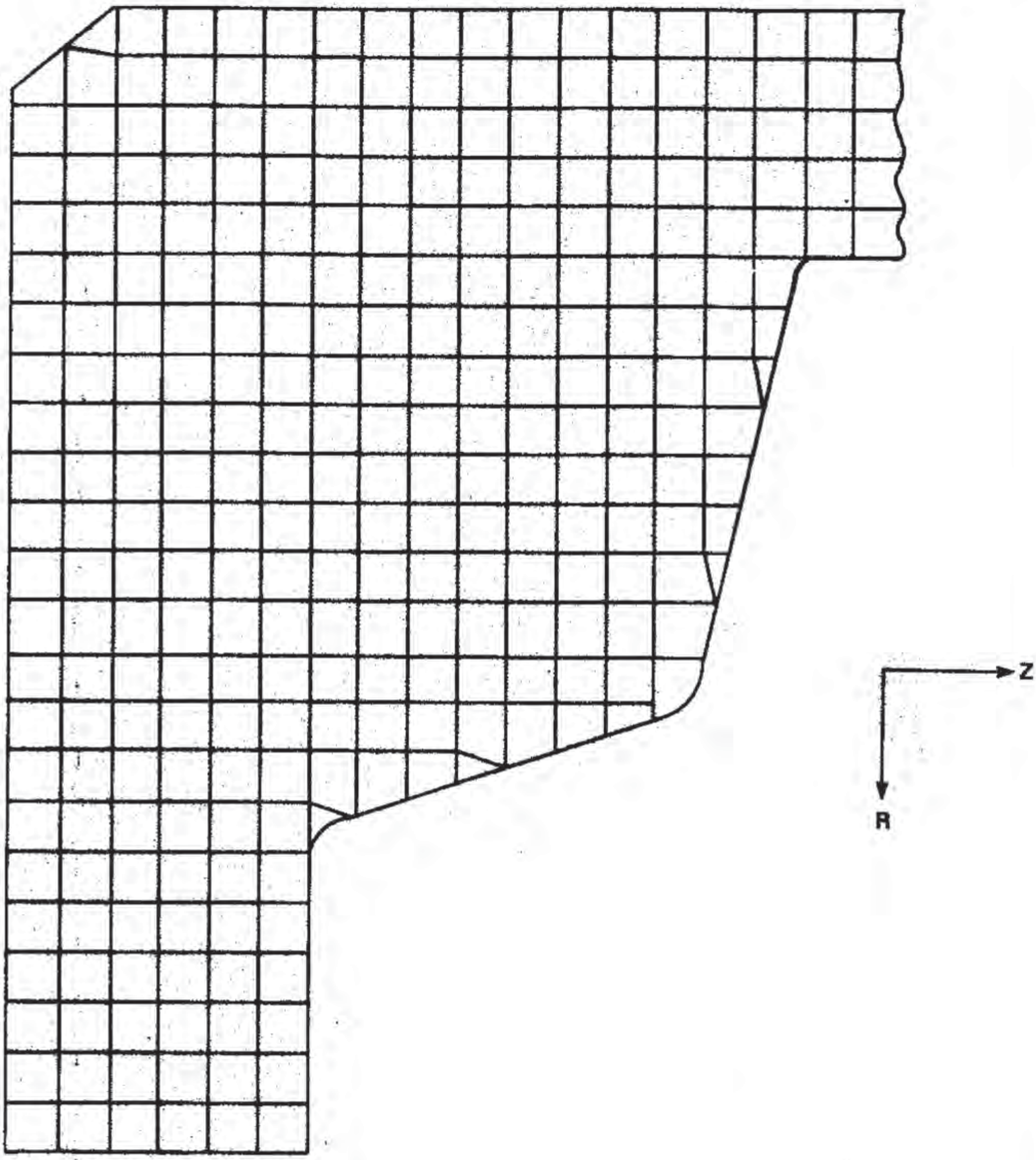
FIGURE 3.7-263



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CORE SEISMIC MODEL
ONE ROW OF 17 FUEL ASSEMBLIES

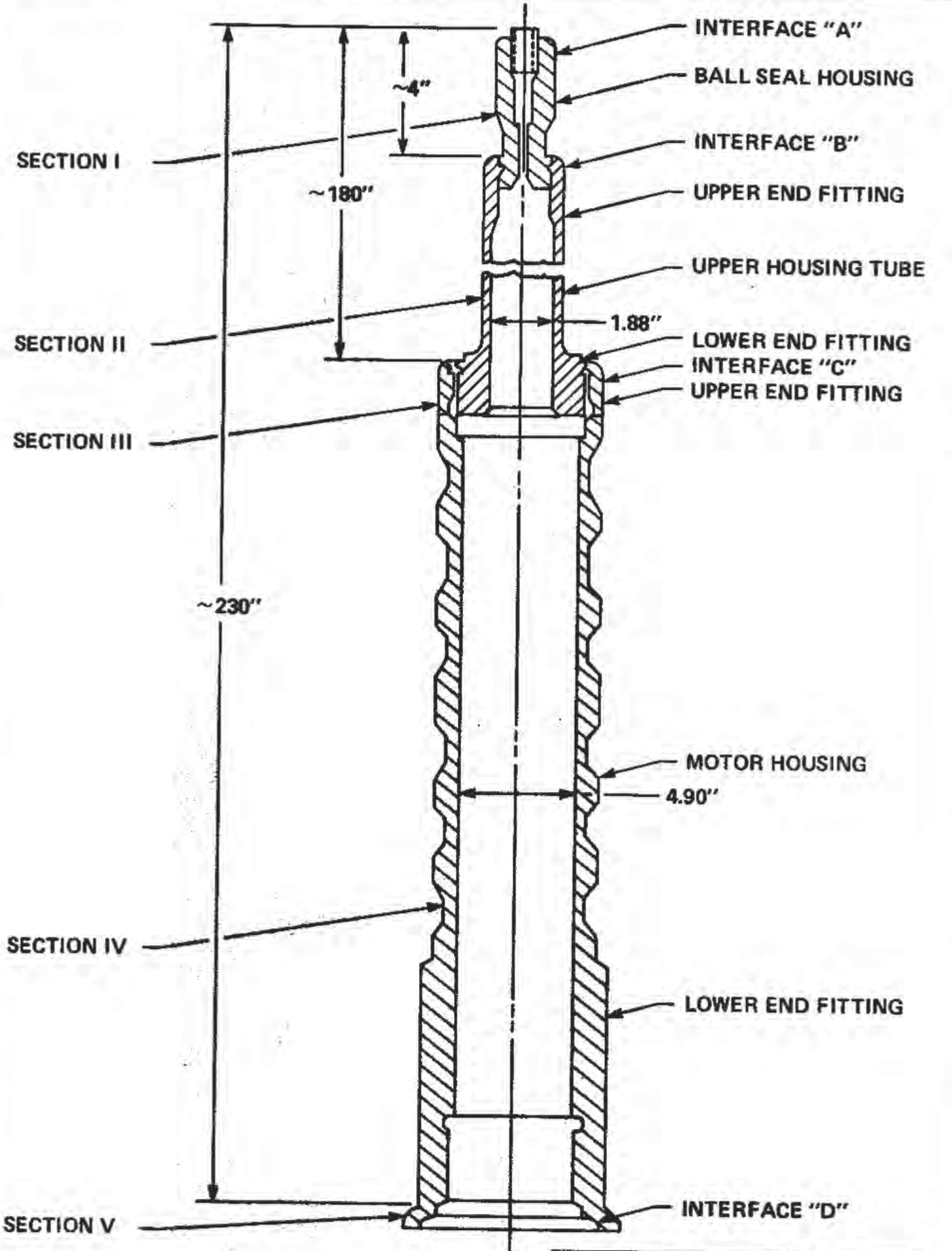
FIGURE 3.7-264



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CORE-SUPPORT BARREL UPPER FLANGE FINITE-ELEMENT MODEL

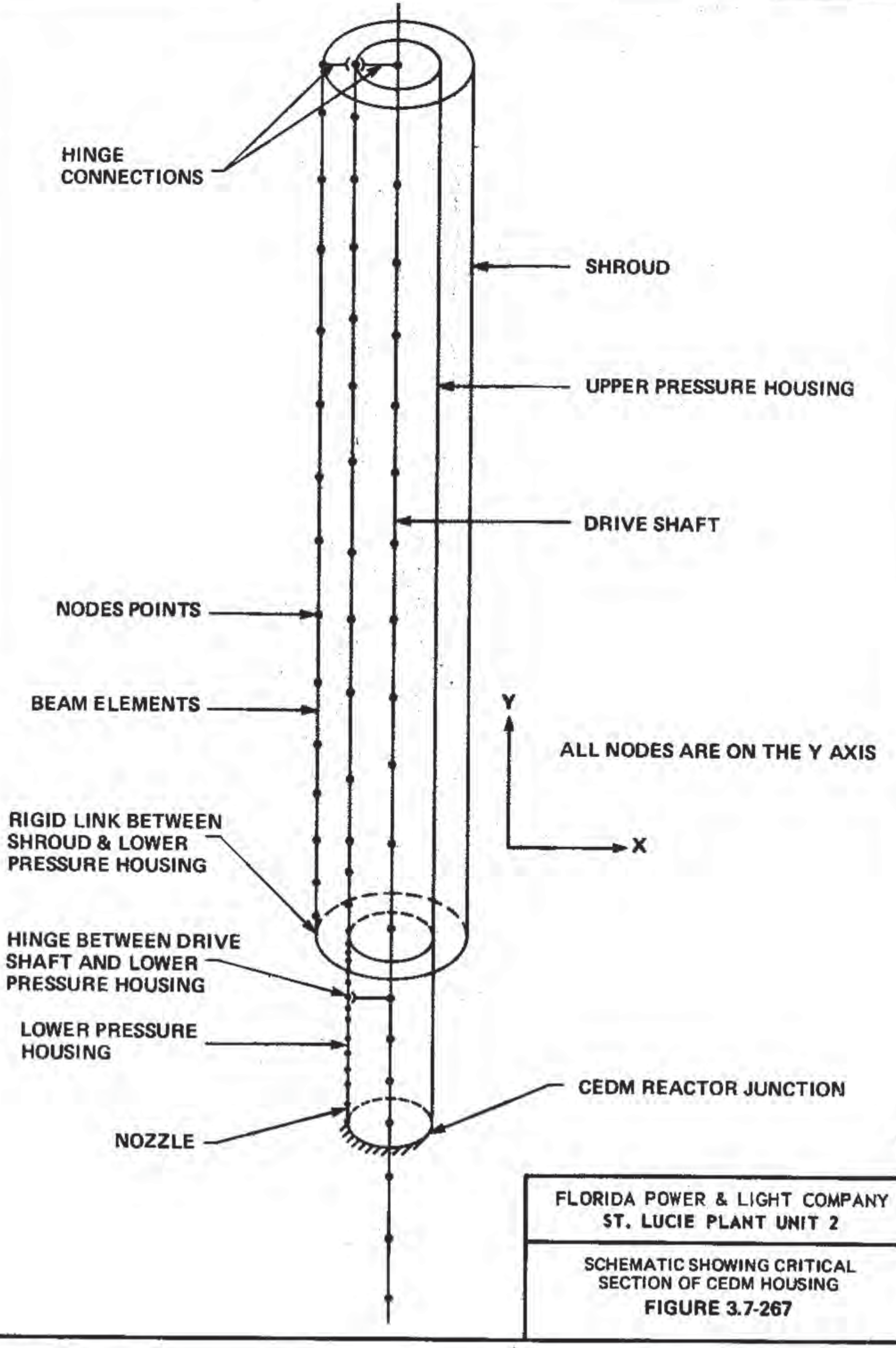
FIGURE 3.7-265



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

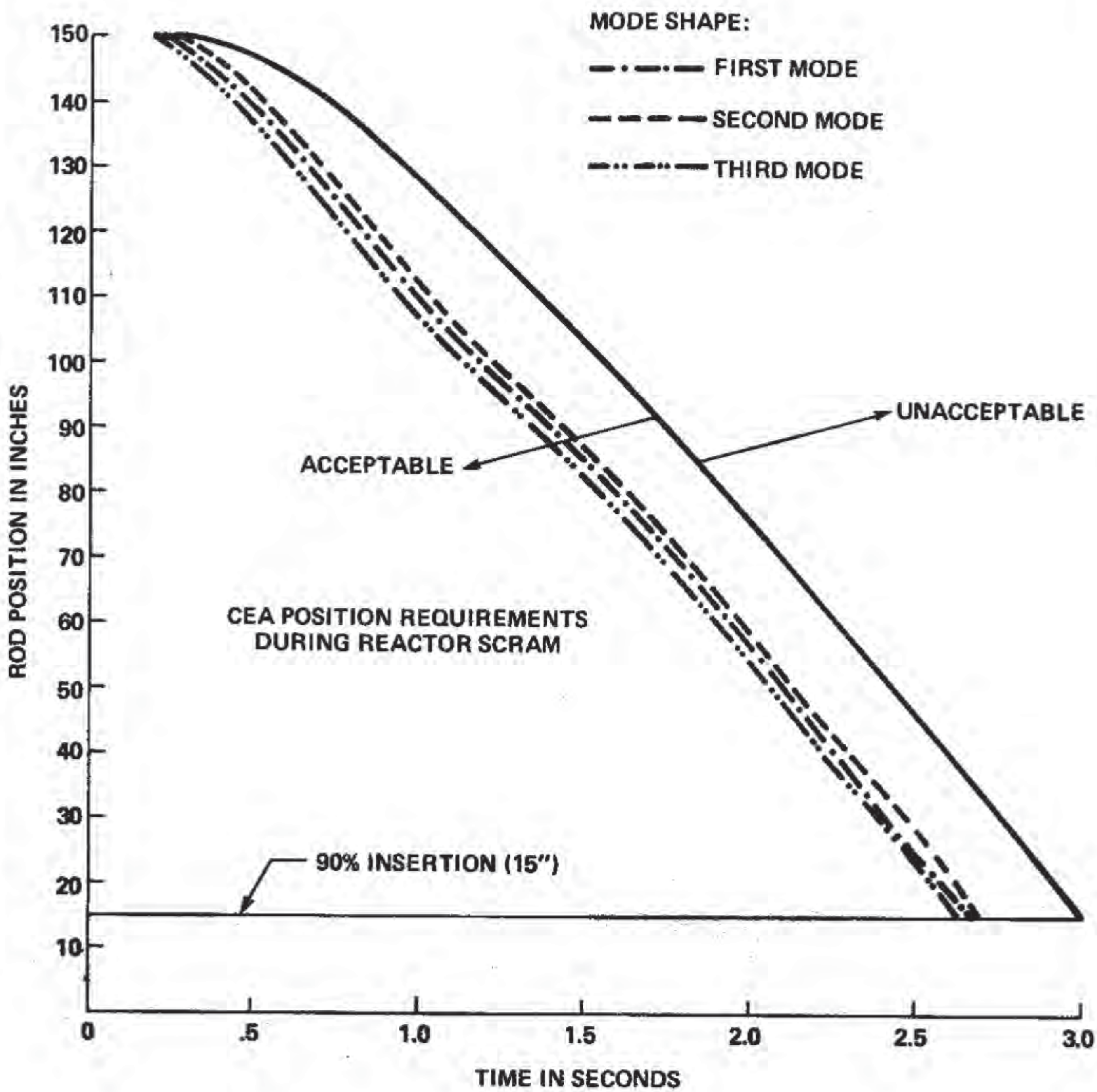
SCHEMATIC SHOWING CRITICAL SECTION OF CEDM HOUSING

FIGURE 3.7-266



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

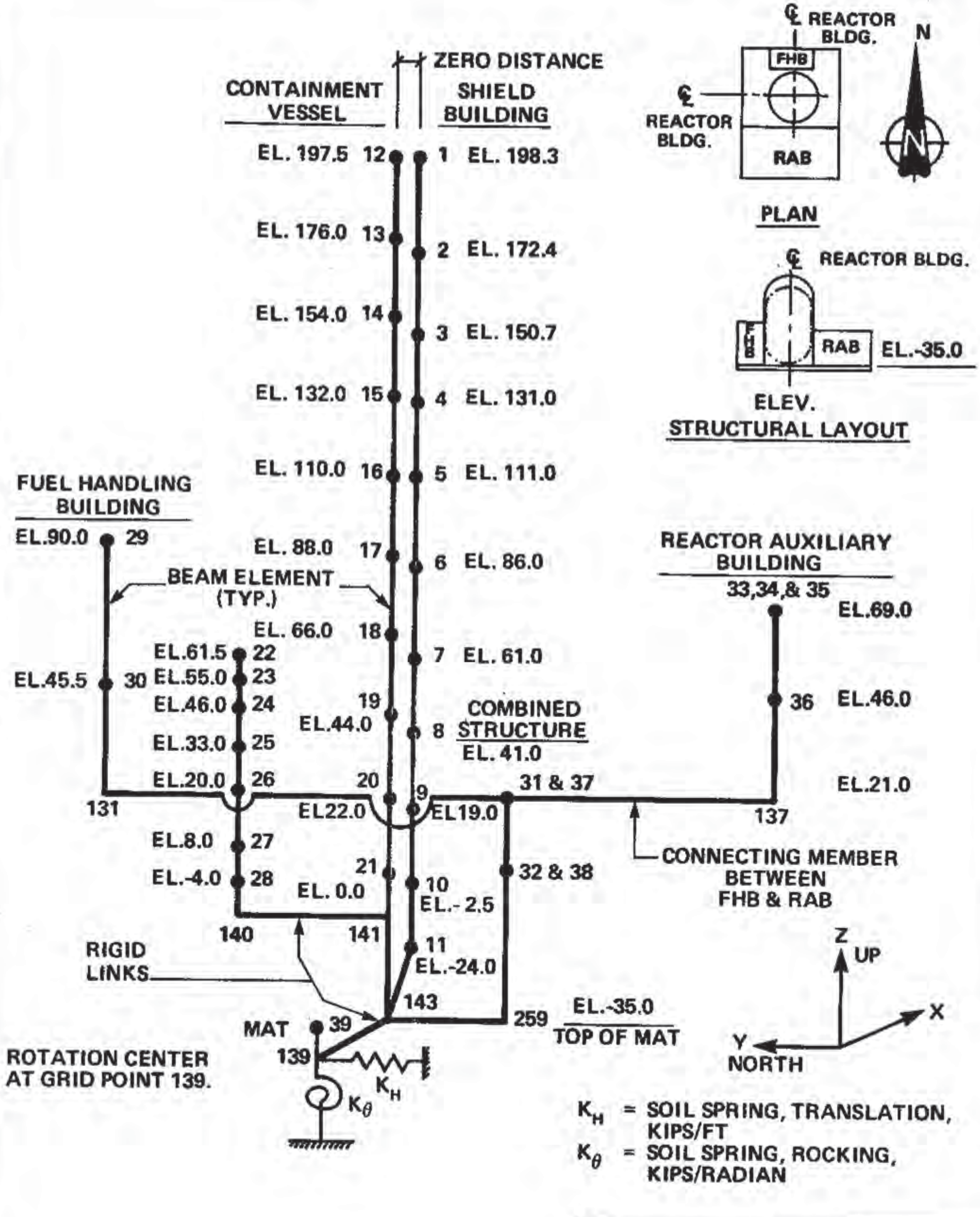
SCHEMATIC SHOWING CRITICAL
SECTION OF CEDM HOUSING
FIGURE 3.7-267



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

AVERAGE EXTENSION SHAFT POSITION DURING SEISMIC SCRAM

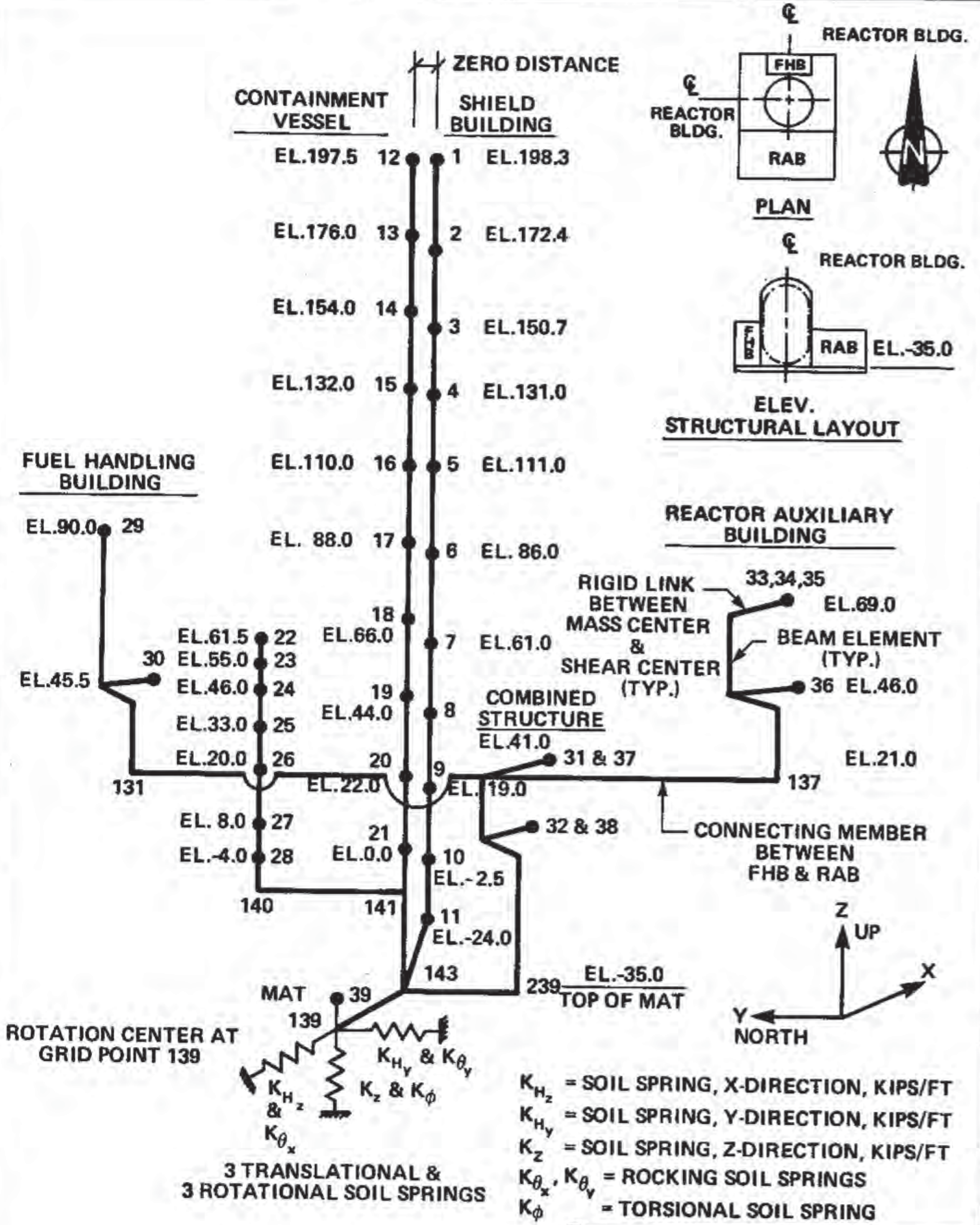
FIGURE 3.7-268



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

WATERFORD NO. 3 REACTOR BLDG.
MATHEMATICAL MODEL (NO TORSIONAL EFFECT)

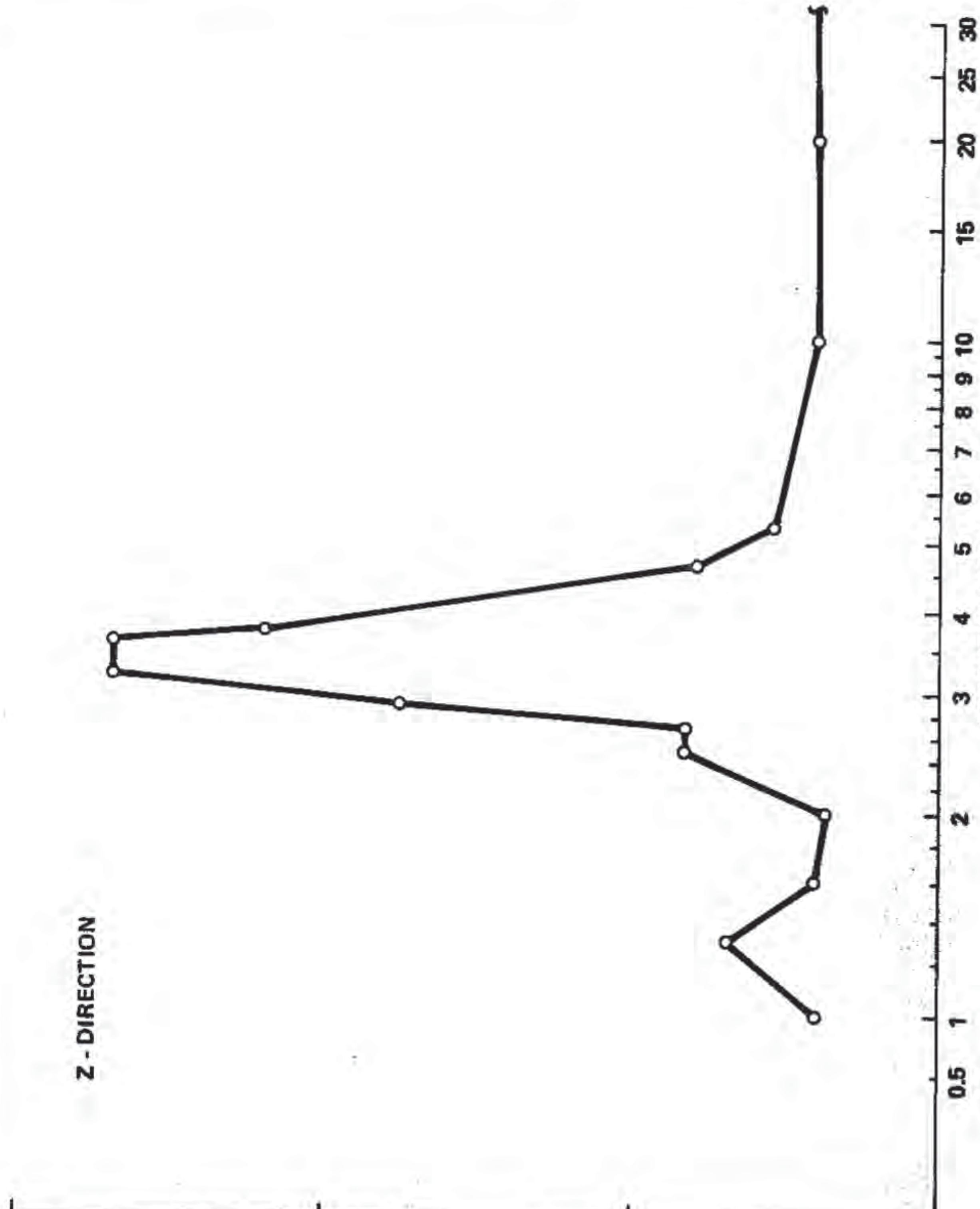
FIGURE 3.7-269



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

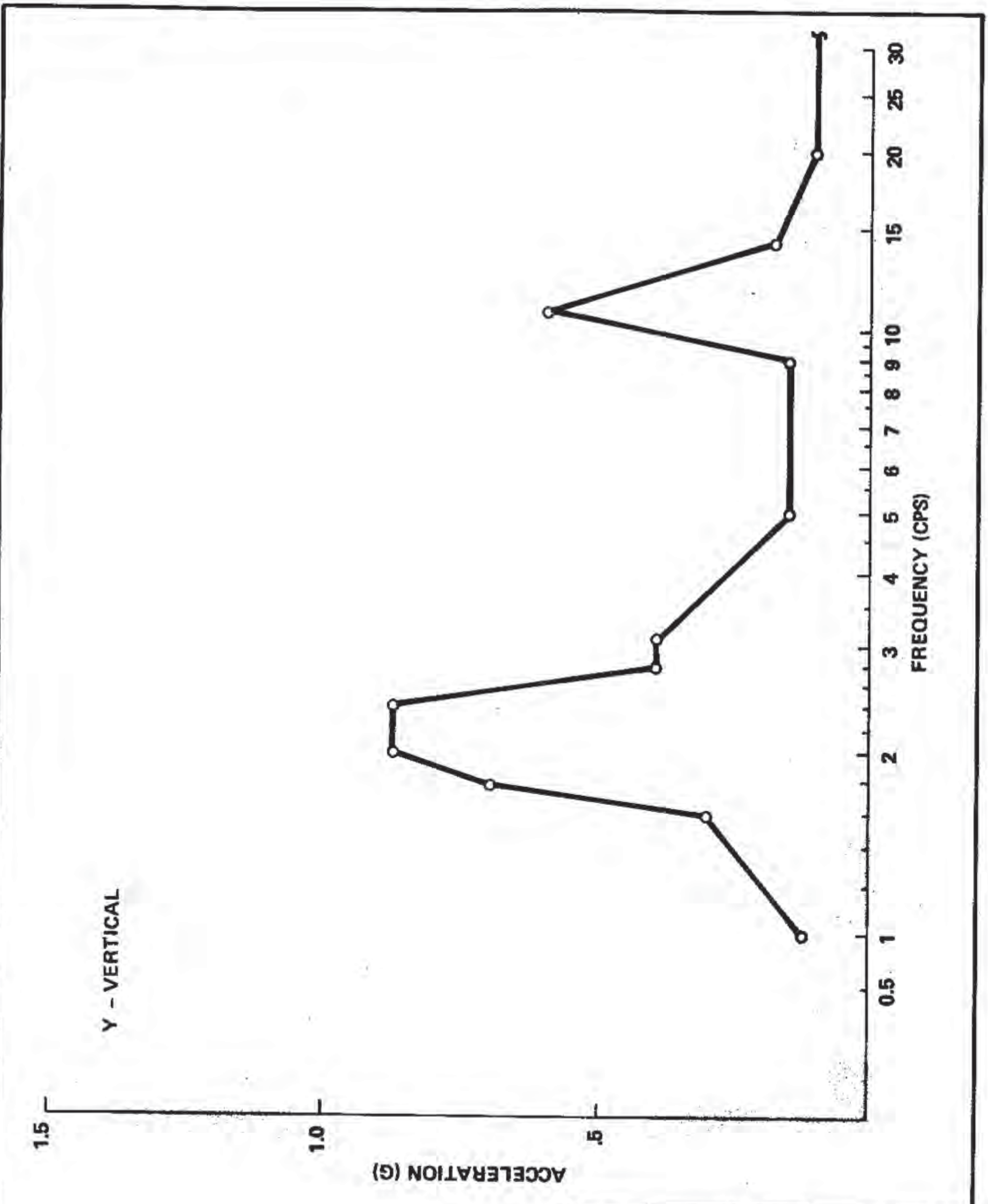
WATERFORD NO. 3 REACTOR BUILDING MATHEMATICAL TORSION MODEL

FIGURE 3.7-270



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

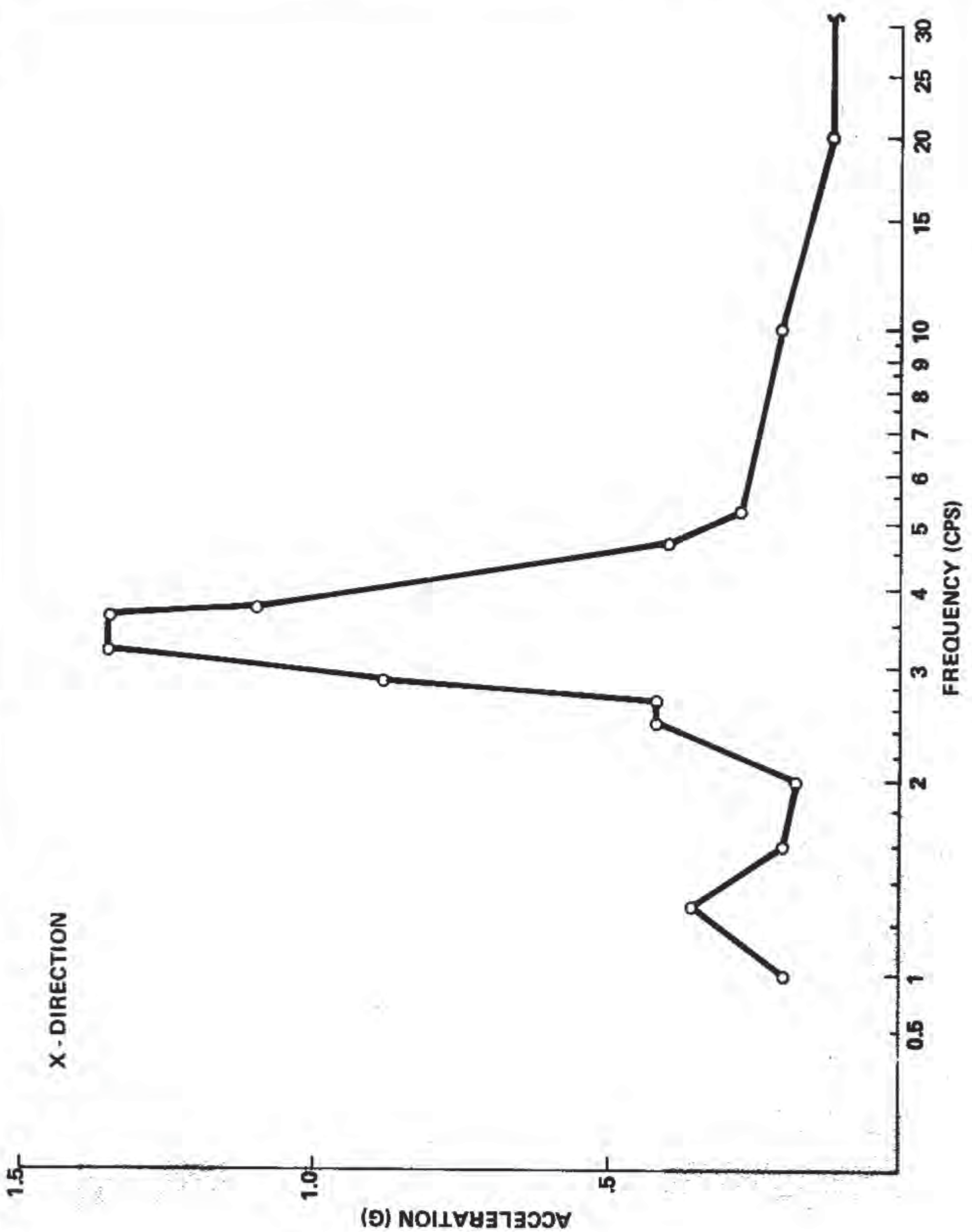
RESPONSE SPECTRA USED FOR HIGH STRESS COMPARISON BETWEEN MODIFIED EQUIVALENT STATIC LOAD METHOD & MODE RESPONSE SPECTRA ANALYSIS
FIGURE 3.7-271



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

RESPONSE SPECTRA USED FOR HIGH STRESS COMPARISON BETWEEN MODIFIED EQUIVALENT STATIC LOAD METHOD & MODE RESPONSE SPECTRA ANALYSIS

FIGURE 3.7-272



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

RESPONSE SPECTRA USED FOR HIGH STRESS COMPARISON BETWEEN MODIFIED EQUIVALENT STATIC LOAD METHOD & MODE RESPONSE SPECTRA ANALYSIS

FIGURE 3.7-273

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

This containment design is not applicable to St. Lucie Unit 2.

3.8.2 STEEL CONTAINMENT

3.8.2.1 Description

The containment vessel, including all its penetrations, is a low leakage steel shell designed to withstand a postulated design basis accident (DBA) and to confine the radioactive materials that could be released by accidental loss of integrity of the reactor coolant pressure boundary. Plans and sectional views are shown on Figures 1.2-8, 9, 10 and 11. The containment vessel is a right circular cylinder (approximately 2 inch thick), as shown on Figure 3.8-1, with hemispherical dome (approximately 1 inch thick) and ellipsoidal bottom (approximately 2 inch thick) which houses the Reactor Vessel, the Reactor Coolant System piping and pumps, the steam generators, the pressurizer and the pressurizer quench tank, and other branch connections of the Reactor Coolant System including the safety injection tanks. The containment vessel penetrations include a construction hatch (Figure 3.8-2), a maintenance hatch, a personnel air lock (Figure 3.8-3), and escape lock (Figure 3.8-4) and various sized penetration nozzles which are described further below. The containment vessel is also equipped with a dome inspection walkway, access ladder and a circular crane girder with a crane rail attached to the shell of the vessel. The containment vessel is enclosed by the reinforced concrete Shield Building described in Subsection 3.8.4.

An annular space is provided between the walls and domes of the containment vessel and the Shield Building in order to permit construction operations and in-service inspection, and to filter any leakage from containment during a loss of coolant accident (LOCA) to minimize site doses.

The containment vessel is an independent free standing structure with a net free volume of approximately 2.5×10^6 ft^3. The containment vessel is rigidly supported at its base near the elevation of its bottom spring line. The concrete base is placed after the cylindrical shell and the ellipsoidal bottom have been constructed and post weld heat treated. Both the Shield Building and the containment vessel are supported on a common Foundation mat. With the exception of the concrete placed underneath and near the knuckles at the sides of the vessel, there are no structural ties between the containment vessel and the Shield Building above the Foundation slab. Therefore there is virtually unlimited freedom for differential movement between the containment vessel and the Shield Building above the top of the concrete base at elevation 23 feet MSL. Concrete floor fill as stated in Subsection 3.8.2.6.3 is placed above the ellipsoidal shell bottom, after the vessel has been post weld heat treated, to anchor the vessel.

The cylindrical portion of the steel containment shell has a minimum thickness of 1.92 inches on an inside radius of 70 feet. The polar crane girder support plates are welded to the shell at approximately six feet on center. Except for some miscellaneous platform framing and some minor seismic restraints, no major floor framing or seismic restraint supports are attached to the shell. Immediately below the crane girder a heating and ventilating duct for the containment ring header, approximately five ft wide x five ft deep and running the entire containment circumference, is structurally supported at 30 places and attached to the shell by means of welded clips. The containment shell is also used to support temporary construction loads from the pedestal cranes.

The 1.92 in. minimum shell plate thickness increases to a minimum of four in. adjacent to all penetrations and openings. The inside radius of the hemispherical dome is 70 ft. with a dome plate 0.96 in. thick connected to the cylindrical portion of the shell at the tangent line by means of a full penetration weld. The containment spray piping is attached to the dome by means of welded clips as are the dome inspection walkway and platforms. The containment vessel is protected from external missiles by the Shield Building. Protection from internal missiles is provided by the primary and secondary shield walls and other containment internal structures (see Section 3.5).

3.8.2.1.1 Penetrations

The function of the containment penetration assemblies is to provide for passage of process, service, sampling and instrumentation pipe lines and electrical cabling (or in the case of the fuel transfer assembly, new or spent fuel) into the containment vessel, while maintaining the desired containment integrity and providing a leak-tight seal with adequate provisions for movement between the pipe lines (or fuel transfer tube) and the containment structure during operation (start up, shutdown, power testing), emergency and accident conditions.

The following design characteristics of the containment penetrations maintain the desired containment integrity:

a) penetrations are capable of withstanding the maximum internal pressure which would occur due to the postulated rupture of any pipe inside the containment vessel.

b) penetrations are capable of withstanding the jet forces associated with the flow from a postulated rupture of a pipe in the penetration or adjacent to it, while still maintaining the integrity of the containment, and

c) penetrations are capable of safely accommodating the thermal and mechanical stresses which may be encountered during all modes of operation and test.

The materials used for penetrations, including the personnel access air locks, the equipment access hatch, the piping and duct penetration sleeves and the electrical penetration sleeves, conform to the requirements set forth by ASME Code, Section III. In accordance with this code, the penetration materials meet the necessary nil ductility transition temperature impact values as specified in Subsection 3.8.2.6.

3.8.2.1.1.1 Mechanical Penetrations

The penetrations listed in Tables 6.2-52 and 53, except the vacuum breakers, penetrate the Shield Building as well as the containment vessel. Penetrations 59, 60, 62, 65 and 66 are for the Shield Building Ventilation System and HVAC instrumentation and do not penetrate the containment; as a minimum, these penetrations are Quality Group C/ASME Class 3 design (reference Subsection 3.2.2, System Quality Group Classification). Both the containment vessel and Shield Building are provided with capped spare penetrations for possible future use.

The process lines traverse the boundary between the inside of the containment vessel and the outside of the Shield Building by means of piping penetration assemblies made up of several elements. Six general types of piping penetration assemblies are provided:

- Type I - Those which must accommodate considerable thermal movements (hot penetrations).
- Type II - Those which are not required to accommodate thermal movements (cold penetrations).
- Type III - Those which must accommodate moderate thermal movements (semi-hot penetrations)
- Type IV - Containment sump recirculation suction lines (designed to reduce the possibility of leakage of sump water).
- Type V - Fuel transfer tube penetration.
- Type VI - Containment vacuum breaker penetration.

The penetration assemblies consist of a containment vessel penetration nozzle, a process pipe, a Shield Building penetration sleeve and a shield building bellows seal. In the case of cold penetrations the containment vessel penetration nozzle is an integral part of the process pipe. In the case of hot and semi-hot penetrations, a multiple fluid head is provided as an integral part of the process pipe. A guard pipe, which encloses the process pipe and directs any fluid released back into the containment, is welded to the flued head. For hot penetrations an expansion joint bellows is welded to the flued head and the containment vessel penetration nozzle to accommodate thermal movements.

At the terminal of a piping penetration assembly near the Shield Building a low pressure leakage barrier is provided in the form of a Shield Building bellows seal. The bellows provides a flexible membrane type closure between the Shield Building penetration sleeve, which is embedded in the Shield Building, and the process pipe.

The Shield Building bellows is designed to withstand a design differential pressure of five psig and provide an adequate leak-tight seal consistent with overall allowable Shield Building leakage.

The containment vessel penetration nozzles are designed to meet the requirements for Class MC vessels under ASME Code, Section III. In compliance with the code, the operating stresses in a containment vessel penetration nozzle caused by the attached penetration assembly are limited to the allowable values given in the code.

The multi-ply bellows expansion joint in the hot pipe penetration assemblies and the Shield Building bellows seals for pipes are designed to accommodate maximum combination of vertical, radial and horizontal differential movements between the containment vessel, the Shield Building and the piping. This design considers the calculated displacements resulting from earthquake, pressure and temperature and relative building settlement.

Type I and III process pipe penetrations are provided with guard pipes to preclude an energy release to the Shield Building annulus due to a rupture in a process line. These guard pipes enclose the process pipe and direct any fluid releases back into containment. Because of this design certain welds on the enclosed piping are not accessible for inservice inspection hence the design approach appropriately stressed the integrity of the guard pipe assembly. By design, the need to perform inservice inspection of enclosed process pipe welds is obviated. The bases are provided below.

Piping for the penetration assemblies meets the requirements for ASME Code, Section III, Code Class 2 piping. The multiple flued head fittings are one piece forgings. They are designed in accordance with ASME Code, Section III, Subsection NE. The multiple flued head fittings are designed to serve as an anchor that accommodates forces and moments imposed on the piping. Insofar as the flued heads are attached directly to the process line and form a portion of process fluid pressure boundary, analyses done for the process line assure that ASME Code, Section III Code Class 2 design rules and stress limits are met for that portion of the flued head required for process line integrity. The guard pipe is designed to be within the allowable stress limits for the maximum design pressure at the design temperature of the process line and is designed so that the maximum stresses in the guard pipe are not more than the code allowable stress values of the material for all load combinations including combined pressure, thermal, seismic, and hydraulic jet forces or pipe whip forces for the assumed rupture.

The process pipes within the guard pipe assemblies are seamless.

Shield Building secondary bellows were procured per the requirements for ASME Code, Section III, Class 3 components. Replacement secondary bellows for Penetrations 15 – 24 are specified to be procured per Expansion Joint Manufacturing Association Standards with material procured from an ASME Section III supplier.

The design of the penetration assemblies is as follows:

a) Type I Penetrations

A Type I (hot) piping penetration assembly is used where large thermal movements of the process pipe have to be accommodated and where the differential between the normal operating temperature of the fluid carried by a process line and the containment vessel wall temperature would create unacceptable thermal or cyclic stress at the attachment of the vessel penetration nozzle. A Type I penetration is shown on Figure 3.8-5.

Type I penetrations are used on the main steam lines and the main feedwater lines. A hot penetration assembly has a multiple flued head machined from a solid forging to which are welded in sequence a length of process pipe, a guard pipe, and a bellows expansion joint. The multiple flued head is welded into, and becomes an integral part of, the process line. The inner flue provides support for the guard pipe and the outer flue provides support for the expansion joint bellows. The length of guard pipe is set so that it extends past the containment vessel penetration nozzle into the vessel. Near the open end of the guard pipe lugs are provided on the process pipe to serve as limit stops for lateral movement to facilitate distribution of pipe rupture loads, in the unlikely event of a slot rupture or pipe whip of the process pipe line within the guard pipe. The guard pipe protects the bellows element against a direct steam impingement in case of a process line rupture. In order to prevent deflection of the guard pipe from overstressing the flued head, at the junction of the flued head and guard pipe a bellows hinge is provided, protected by a liner, to prevent it from being impinged upon.

The expansion joint bellows is attached at one end to the outer flue on the flued head and at the other end to the containment vessel penetration sleeve. The expansion joint is provided with a two-ply bellows that has a connection between the plies for integrity testing.

A bellows located between the Shield Building wall and the flued head seals the penetration where it passes through the concrete shield wall. This bellows is a two-ply bellows element constructed to permit a pressure test of the annulus between the plies.

b) Type II Penetrations

A Type II penetration assembly is shown on Figure 3.8-6. This type of penetration is provided for pipe lines carrying low temperature (below 200 F) and low pressure fluids and gases. The principal consideration in this design is the provision of a leak-tight seal between the pipe and the containment vessel.

This is accomplished by use of sleeves welded into the steel containment vessel by the vessel fabricator. The process line is welded directly to a sleeve penetrating the containment vessel. The sleeve and containment shell are designed to carry the forces and moments due to all normal and upset conditions.

c) Type III Penetrations

A Type III penetration assembly is shown on Figure 3.8-5.

For moderate thermal movement (temperatures over 200 F) where there is a possibility of a pipe rupture overpressurizing the annulus, a multiple flued head is used to provide a leak-tight seal between the penetration nozzle and the process pipe.

In case of a rupture of the process pipe in the annulus area, the guard pipe acts to direct the fluid back into the containment vessel, thus preventing overpressurization of the annulus. The penetration is designed to accommodate all forces and moments due to thermal expansion, seismic, fluid transient and pipe rupture.

d) Containment Sump Recirculation Suction Lines

A special type of penetration assembly (Type IV) is provided for the suction lines from the containment sump. These lines are used following a LOCA to allow recirculation of containment sump water by the containment spray and high pressure safety injection pumps.

As shown on Figure 3.8-6, each line consists of a double barrier concentric pipe from the sump up to the suction line isolation valve outside the containment. The penetration assembly is designed for the differential motion associated with the SSE.

e) Fuel Transfer Tube Penetration

A fuel transfer tube penetration (Type V) is provided to transport fuel rods between the refueling transfer canal and the spent fuel pool during refueling operations of the reactor. The penetration is shown on Figure 3.8-7 and consists of a 36 inch diameter stainless steel pipe installed inside a 48 inch pipe. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pool. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment vessel and provision is made for testing welds essential to the integrity of containment. Bellows expansion joints are provided on the pipe to compensate for building settlement and differential seismic motion between the Reactor Building and the Fuel Handling Building.

The bellows expansion joints which form a part of the containment boundary meet the requirements of ASME Code, Section III. The fuel transfer tube bellows are designed for a 35 foot head of water.

Bellows design and construction is such that the bellows does not deflect more than its designed amount: The bellows is designed to withstand a 60-year lifetime total of 7,000 cycles of expansion and compression due to operating thermal expansion and 200 cycles of differential settlement and seismic motion.

f) Containment Vacuum Breaker Penetration

The penetration consists of a nozzle welded on the containment vessel with a check valve inside the containment and a butterfly valve outside the containment (see Figure 9.4-9).

The containment vessel penetration details are shown on Figures 3.8-9, 3.8-10 and 3.8-11. Shield Building penetration details are shown on Figure 3.8-9.

3.8.2.1.1.2 Electrical Penetrations

Electrical penetrations are divided into the following three basic types with each performing a specific function:

a) Medium voltage penetrations (over 600 volts)

b) Low voltage power and control (600 volts and lower)

c) Low voltage level instrumentation which includes coaxial cables, thermocouple extension wires and other types of low level signal cables.

Electrical penetration assemblies are used to provide means for carrying electric circuits through the containment, the annulus and the Shield Building. Cable protection sleeves are provided to give support and protection to cables in the annular space. Typical electrical penetration assemblies are shown on Figure 3.8-12.

Medium voltage penetrations consist of a canister (double seal), a header plate extension tube (single seal) and flexible connecting conductors. The primary canister sub-assembly consists of two stainless steel header plates welded to a 14 inch diameter carbon steel pipe and is approximately 15 inches long. The secondary header plate sub-assembly consists of a 2 1/2 inch thick stainless steel header plate with a 14 inch schedule 40 by 8 inch long carbon steel pipe welded to it. The secondary header plate sub-assembly (single seal configuration) is welded to the outboard end of the nozzle, which is embedded in the concrete shell of the Shield Building aligned with the containment vessel nozzle.

In the medium voltage penetrations, three solid bus conductors pass through the canister and are terminated with ceramic bushings which are secured and sealed with stainless steel compression fittings. The terminal bushings are internally sealed to the header plate with Vitron O-rings. The canister and its seals form a small pressure vessel which is used for leakage monitoring of the seals.

Each medium voltage penetration has a dual thermocouple (Chromel-Constantan) to measure the operating temperature in the primary canister. Electrical leads extend through the secondary header plate to the outboard containment end.

Low voltage power, control and instrumentation penetrations consist of stainless steel header plate on the primary (containment) side and secondary (Shield Building) side, connected with flexible conductors in the annular space.

The feed through modules containing various insulated electrical conductors pass through the header plates and are secured and sealed to the plate with stainless steel compression fittings. On the containment side these feed through modules contain solid conductors which are sealed in resilient thermoplastic sealants at both ends of a stainless steel tube. A double seal is provided for leakage monitoring of the module.

In the Shield Building the solid conductor feedthrough module has a single seal.

The low voltage power penetration assemblies are furnished with a dual thermocouple (Chromel-Constantan) to measure the operating temperature of the primary header plate. Electrical leads extend through the secondary header plate to the outboard containment end.

3.8.2.1.1.3 Equipment and Personnel Access

Two equipment hatches are provided. These are welded steel assemblies of 28 feet diameter and 12 feet diameter openings respectively. The 28 feet diameter hatch cover is welded into position on completion of construction. The design is such that post-weld heat treatment is not required.

The 12 feet diameter hatch has a double gasketed flanged and bolted cover. Provisions are made to pressurize the space between the gaskets to 44 psig for leak rate testing.

Two personnel air locks are provided. These are welded steel assemblies with two double gasketed doors in series. Provision is made to pressurize the space between the gaskets for leak testing. The doors are mechanically interlocked to ensure that one door cannot be opened until the second door is sealed. Provisions are made for deliberately violating the interlock by the use of special tools and procedures under strict administrative control. Each door is equipped with quick acting valves for equalizing the pressure across the doors. The doors are not operable unless the pressure is equalized. Pressure equalization is possible from every point at which the associated door can be operated. The valves for the two doors are properly interlocked so that only one valve can be opened at one time, and only when the opposite door is closed and sealed. Each door is designed so that with the other door open, it withstands and seals against the design and testing pressures of the containment vessel. There is visual indication outside each door showing whether the opposite door is open or closed and whether its valve is open or closed. In addition, limit switches are provided to indicate remotely whether doors are open or closed. Control room annunciation is provided. Provision is made outside each door for remotely closing and latching the opposite door so that in the event that one door is accidentally left open it can be closed by remote control. The air-locks have nozzles installed which permit pressure testing of the lock at any time. An interior lighting system and a communication system are installed. The lighting system is capable of operating from an emergency power supply. The Communication System consists of PAX sets which are powered by the St. Lucie Unit 1 Communication System.

3.8.2.2 Applicable Codes, Standards and Specifications

The following codes, standards and specifications are used in the design, fabrication, erection and testing of the containment vessel:

a) American Society of Mechanical Engineers (ASME)

1) ASME Section II, "Material Specifications," 1971 Edition, Winter 1972 Addenda

2) ASME Section III, "Nuclear Power Plant Components," 1971 Edition, Winter 1972 Addenda

3) ASME Section VIII, "Unfired Pressure Vessels," 1971 Edition, Winter 1972 Addenda

4) ASME Section IX, "Welding Qualifications," 1971 Edition, Winter 1972 Addenda

b) American Society of Testing and Material (ASTM) - 1971

1) ASTI A36 - Structural Steel

2) ASTM E376 - Recommended Practice for Measuring Coating Thickness by Magnetic Field or Eddy - current "Electromagnetic" Test Methods.

c) American Institute of Steel Construction (AISC)

   1) Specification for the Design, Fabrication and Erection of Structural Steel for Buildings - seventh edition, February 1969

d) American Welding Society (AWS)

   1) Structural Welding Code, AWS D1.1-72

e) Steel Structures Painting Council

   1) SSPC-SP-3, Power Tool Cleaning

   2) SSPC-SP-7, Brush-off Blast Cleaning

   3) SSPC-SP-10, Near White Blast Cleaning

   4) SSPC-PA-1, Shop, Field and Maintenance Painting

f) Ebasco Services Incorporated

   1) Ebasco Specification FL0-2998-757, "Steel Containment Vessel," November, 1973

   2) Ebasco Specification 873 "Nondestructive Testing Procedures," 1973

   3) Ebasco Specification 860, "Quality Control Requirements," 1971

   4) Ebasco Coating Guide CP-43, Instructions for Application of Inorganic Zinc Coatings to Metal Surfaces, March 1970

The containment vessel is code stamped for a pressure of 44 psig in accordance with Article NE-8000 of Section III of the ASME Code.

The design internal pressure, referred to as "maximum calculated peak internal pressure" in Article NE-3112, for the containment vessel is specified in accordance with the provisions of Section III of the ASME Code. The design requirements for Class MC vessels are contained in Article NE-3000 of Section III.

The containment vessel is pressure tested in accordance with the rules of ASME Code, Section VIII, Paragraph UG-100 and Section III NE-6300. The maximum test pressure is 50 psig.

The design of supports and bracing and similar structures not within the scope of the ASME Code conform to the requirements of American Institute of Steel Construction (AISC) Specification, seventh edition.

The containment vessel design and construction meet the requirements of applicable Florida state and local building codes.

Refer to Subsections 3.1.50 and 3.1.51 for a discussion of General Design Criteria 50 and 51.

3.8.2.3 Loads and Load Combinations

The vessel is designed to exhibit a general elastic behavior under accident and earthquake conditions of loading. No permanent deformations due to primary stresses are permitted in the design under any condition of loading. The design of the containment vessel is based on permissible stresses as set forth in the applicable codes. The structure safely functions within the normal design limits as specified in Section III of the ASME Code, Article NE-3000 "Design". The purchase specification of the containment predates the formal issuance date of the Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components", June 1973 (RO), hence the Regulatory Guide is not used in the design.

The areas of the vessel adjacent to penetrations that are not subject to externally applied loads are designed by the area replacement method in accordance with NE-3332 of Section III of the ASME Code.

Where external loads and moments are applied to penetrations the secondary and local stresses are evaluated in accordance with the Welding Research Council Bulletin 107[(1)] and the basic stress intensity limits are in accordance with ASME Code, Section III. Where substantial thermal or mechanical loads other than pressure loads exist, analyses are performed using the Yale computer program developed by Professor A Kalnins[(25)]. The program is used to analyze the following:

a) discontinuity stresses at embedment,

b) head to shell discontinuties,

c) discontinuities around the crane girder, and

d) areas where the membrane stress spectrum is disturbed due to change in geometry or loading conditions.

A containment vessel of thickness suitable to meet the specified internal, pressure requirements is capable of withstanding an external pressure differential of 1.05 psig in accordance with NE-3133 of Section III of the ASME Code. Since the ASME Code charts have a safety factor of three, the collapsing pressure for the containment vessel is about three times greater than the design external pressure differential. The containment vessel is continuously purged which eliminates pressure fluctuations caused by air temperature changes during various operating modes. The containment purge system is described in Subsection 9.4.8.8. EC280084

Protection of the containment vessel against excessive external pressure is provided by two independent vacuum breaker lines. The arrangement of instrumentation and valving is shown on Figure 9.4-9.

Each vacuum breaker assembly consists of a check valve inside and an automatic air operated butterfly valve outside the containment vessel. Actuation of the butterfly valve is controlled by differential pressure between the containment vessel and the shield building annulus. A transmitter senses the differential pressure and provides a signal to the pilot solenoid on the air operated butterfly valve to open the valve at a differential pressure of -9.85 $\pm$ 0.35 in. wg. and to close the valve at a differential pressure of -7.75 in. wg. An alarm signal is actuated at a pressure differential of -9 in. wg. The check valve is counter weight balanced to open at a differential pressure of 1.1 in. wg.

The design criterion used in sizing the vacuum breaker system is to prevent the occurrence of a differential pressure between the inside of the containment and the annulus of less than -1.05 psi, and between the Shield Building annulus and the environment of less than -3 psi due to inadvertent actuation of both containment spray pumps at runout conditions plus four fan coolers. Refer to Subsection 6.2.1.1 for a discussion of the containment vacuum breaker analysis. EC280084

3.8.2.3.1 Design Conditions

Pressures, temperatures, design loads and the corresponding nomenclature for the design of the containment vessel and penetrations are listed below. The calculated containment environment is shown in Section 6.2 where it is shown that these design specified containment environment values are appropriate. The design conditions (pressure/temperatures) noted below are not impacted by EPU conditions per Reference 13.

a)	Test overpressure	50 psig	
b)	Maximum Calculated Peak Internal Pressure (P_{g1}) and Temperature (T_{g1}) (as defined by 1971 Edition of ASME III)	44 psig @ 264 F	
c)	Leakage Rate Test Pressure	43.48 psig (Table 6.2-3, -4)	
d)	Maximum containment pressure (P_{c1}) and temperature (T_{c1}) after loss of coolant accident (LOCA)	44 psig @ 264 F	
e)	Maximum external to internal pressure differential and temperature after cooling of the containment by the containment spray system and actuation of the vacuum breaker system.	1.05 psi @ 120 F	EC280084
f)	Process line maximum operating pressure (P_o) and temperature (T_o)	See Figures 3.8-5 and -9	
g)	Penetration guard pipe, internal pressure P_g and temperature (T_g), ambient	14.7 psia @ 120 F	
h)	Penetration guard pipe internal pressure (P_{g1}) and temperature (T_{g1}) after LOCA	44 psig @ 264 F	
i)	Process line design pressure (P_d) and temperature (T_d)	See Figures 3.8-5 and -9	

j)	Penetration guard pipe internal pressure (P_{gr}) and temperature (T_{gr}) after rupture of the process line inside the penetration. (P_{gr}), (T_{gr}) equal (P_o), (T_o) respectively.	See Figures 3.8-5 and -9
k)	Containment ambient pressure (P_c) and ambient temperature (T_c)	14.7 psia @ 120 F
l)	Annulus Pressure, Normal (P_a)	14.54 to 14.62 psia
m)	Annulus Temperature, Normal (T_a)	103 F to 115 F
n)	Annulus Pressure After LOCA (P_{a1})	14.7 to 14.54 psia
o)	Annulus Temperature After LOCA (T_{a1})	218 F to 230 F
p)	Pipe Assembly Load (W)	See Figure 3.8-10

q) Penetration Seismic Load (EP). Occasional loads such as loads due to fluid transient events are included where applicable along with seismic loads which consists of the following components:

		<u>OBE</u>	<u>SSE</u>
Axial Load	-	EPA	EPA´
Transverse Load	-	EPT	EPT´
Bending Moment	-	EPM	EPM´
Torsional Moment	-	EPT_o	EPT'_o

r) Penetration Pipe Rupture Local (PR) which consists of the following components:

Axial Load	-	PRA
Transverse Load	-	PRT
Bending Moment	-	PRM
Torsional Moment	-	PRT_o

s) Penetration Jet Impingement Load (J) which consists of the following components:

Impinging Force on Guard Pipe	-	JGT
Bending Moment on Process Pipe	-	JPM
Transverse Load on Process Pipe	-	JPT

t) Penetration thermal load (T) which consists of the following components:

Axial Load	-	TA
Transverse Load	-	TT
Bending Moment	-	TM
Torsional Moment	-	TT_O

u) Gravity Loads (include but are not limited to):

Item		Estimated Weight	
Vessel shell & appurtenances	=	7,304,000	lb
Penetrations	=	380,000	lb.
Equipment hatch	=	266,000	lb.
Maintenance hatch	=	66,000	lb.
Personnel lock	=	62,200	lb.
Escape lock	=	20,900	lb.
Ventilation duct	=	35,200	lb.
Crane girder & rail & girder	=	482,500	lb.
Trolley & crane	=	456,000	lb.
Platforms, ladders, stairs, etc.	=	70,000	lb.

v) Live Loads (include but are not limited to):

Load Item			
Weight of Contained Test Air	=	100,000	lb.
Crane Operating Live Load	=	200	tons
Impact from crane	=	30	tons
Air locks	=	150	psf
Platforms on Dome	=	50	psf
Access Ladder	=	500	lb.
Maintenance Hatch	=	50	tons
Insulation (for PWHT)	=	233,000	lb.
Concrete dome temporary construction loads	See Figure 3.8-13		

w) Seismic Loads:

1) The operating basis Earthquake seismic loads are lateral and vertical forces equal to the seismic coefficients shown on Figure 3.8-14 multiplied by the permanent gravity loads applied to the vessel, assumed to be acting concurrently.

2) The safe shutdown earthquake seismic loads are lateral and vertical forces equal to the seismic coefficients shown on Figure 3.8-15 multiplied by the permanent gravity loads applied to the vessel, assumed to be acting concurrently.

3) Stresses due to seismic loads are combined with stresses caused by the maximum hypothetical accident, dead loads and appropriate indicated loads to obtain the total stresses.

4) The seismic loads include the seismic effects due to the inertia of the mass of the air locks and equipment hatches and the effects of the air locks vibrating as independent systems.

x) Wind Loads

The portion of the containment vessel which is exposed above grade prior to the completion of the shield structure is designed for the wind loads on the projected area of the circular shape in accordance with the height zones below:

Weight Above Grade Feet	Wind Load Psf
0-30	18
30-50	24
50-100	30
Above 100	36

Tabulated wind pressures include the reduction for the circular shape of the vessel.

The maximum water elevation post LOCA is 21 ft. 3 in. which is below the exposed internal surface of the containment vessel. Hence, the containment does not experience flooding loads.

3.8.2.3.2 Load Combinations

Various combinations of loads are considered in the design of the containment vessel corresponding to loading conditions during construction, test, normal operation, earthquake and accident conditions. Thirteen load combinations are considered as follows:

a) Case 1 - Construction loads at Post Weld Heat Treatment (PWHT)

b) Case 2 - Acceptance Test loads at ambient temperature

c) Case 3 - Pre-Operation Test loads at ambient temperature

d) Case 4 - Normal Operating Condition with OBE and a temperature range of 30F to 150F

e) Case 5 - Cold shutdown with OBE and a temperature range of 30F to 120F

f) Case 6 - LOCA loads plus OBE loads

g) Case 7 - LOCA loads plus SSE loads

h) Case 8 - Pipe rupture loads plus OBE loads, pipe thermal and seismic loads at penetrations

i) Case 9 - Pipe rupture loads plus SSE loads, pipe thermal and seismic loads at penetrations

j) Case 10 - Condition with OBE loads plus jet forces, and pipe thermal plus thermal and seismic loads at penetrations

k) Case 11 - LOCA loads plus SSE loads, jet forces, and pipe thermal plus thermal and seismic loads at penetrations

l) Case 12 - Normal Operating Condition with SSE loads at temperature range of 30F to 150F.

m) Case 13 - Cold shutdown with SSE loads at temperature range of 30F to 120F.

The load combinations for each of the cases are summarized in Table 3.8-1. Table 3.8-2 shows the load combinations considered at penetrations. Table 3.8-3 shows the load combinations used to determine stresses at the junction of the containment vessel knuckle region and column.

3.8.2.4 Design and Analysis Procedures

The design and analysis of the steel containment is in accordance with ASME Code, Section III Subsection NE. The analysis of the steel containment consists of two parts: the overall analysis of the containment and the local analyses. The local analyses include such items as the air lock and penetrations.

The steel containment ellipsoidal bottom head is completely embedded in concrete. The containment dead weight and any overturning moments due to a seismic event are assumed to be transferred to the concrete in bearing. Shear stresses are assumed to be zero.

3.8.2.4.1 Shell Analysis

Stresses in the vessel shell remote from penetrations or other appurtenances are analyzed as described below. Shell stresses adjacent to appurtenances are analyzed along with the appurtenance design.

Stresses resulting from each specified load condition are calculated separately at critical locations and combined to obtain total meridional and circumferential stresses at each point. Stress intensities are then determined and compared to specified allowable stresses.

In accordance with the maximum shear stress failure criterion and thin shell theory, stress intensities are found as follows:

a) since shear stress is much less than circumferential or meridional stress, shear stresses are neglected in calculating stress intensities, and

b) since radial stress is much less than circumferential or meridional stress, for calculating stress intensities, radial stress $\sigma_r = 0$.

For latitudinal stress σ_θ and meridional stress σ_Φ of like sign (refer to Figure 3.8-16), the stress intensity is the larger of $|(\sigma_\theta) - \sigma_r| = |(\sigma_\theta)|$ or $|(\sigma_\Phi - \sigma_r| = |(\sigma_\Phi)|$. For σ_θ and σ_Φ of unlike signs, the stress intensity is equal to $|\sigma_\theta - \sigma_\Phi|$.

In addition to the stress intensity evaluation, compressive buckling loads are investigated in the construction, normal operating, and accident conditions. An axial load in the longitudinal direction induces a meridional membrane stress σ_Φ (local buckling) in the cylindrical portion of the containment vessel shell. In the cylindrical portion of the vessel:

$$2\pi \ r \ N_{\phi} + F = 0 \qquad \text{(from equilibrium)}$$

where:

r = shell radius

N_ϕ = meridional force per unit length = $\frac{-F}{2\pi r}$

F = axial load

σ_ϕ = meridional stress (longitudinal) = $\frac{N_\phi}{t} = \frac{-F}{2\pi rt}$

t = shell thickness

From Timoshenko's Theory of Plates and Shells(3)

$$\frac{N_\phi}{r_1} + \frac{N_\theta}{r_2} = -Z \qquad (1)$$

where:

N_θ = latitudinal (hoop force) per unit length

r_1 = radius in longitudinal direction

r_2 = radius in latitudinal direction

Z = radial load = 0 for a cylinder with $r_1 = \infty$ and $r_2 = r$

therefore: $N_\theta = 0$.

In the spherical portion (dome), an axial load in the longitudinal direction induces a meridional buckling stress (σ_Φ) and a latitudinal buckling stress (σ_θ). In the spherical portion:

$$2 - r_O \; N_\phi \, Sin\phi + F = 0 \qquad \text{(for equilibrium(3))}$$

where:

r_O = radius at point of interest

ϕ = refer to Figure 3.8-17

$r_o = r \sin \phi$

$$N_\phi = \frac{-F}{2 r_o \pi \sin \phi}$$

$$\sigma_\phi = \frac{N_\phi}{t} \qquad \sigma_\phi = \frac{-F}{2\pi rt \; Sin^2 \phi}$$

$Z = 0$ and for a sphere $r_1 = r_2 = r$, hence equation (1) gives

$$\frac{N_\theta}{r} = \frac{N_\phi}{r} \text{ and } N_\theta = -N_\phi$$

Figure 3.8-18 shows the equation used for the total vertical axial load to be considered at any location.

An axial load in the latitudinal (circumferential) direction induces a meridional (load buckling) stress in the cylindrical portion of the containment vessel:

$$\sigma_\phi = \frac{Mr}{I} = \frac{Mr}{\pi r^3 t} = \frac{M}{\pi r^2 t}$$

$$N_\phi = \frac{M}{\pi r^2}$$

where:

M = moment at point of interest

I = moment of inertia of cylindrical shell at point of interest

r = shell radius

t = shell thickness

Since $N\phi / r_1 + N\theta / r_2 = -Z$ and Z is equal to zero when $r_1 = \infty$
and $r_2 = r$, then $N_\theta = 0$.

In the spherical portion (dome), an axial latitudinal load induces a longitudinal buckling stress (σ_ϕ) as well as a latitudinal buckling stress (σ_θ):

$$N\boldsymbol{\phi}_{\max} \operatorname{Sin} \boldsymbol{\phi}\, t = \frac{Mr_o}{I} = \frac{Mr_o}{\boldsymbol{\pi} r_o^3}$$

(assuming a stress distribution in accordance with Beam Flexure Theory)

$$\sigma_\phi = \frac{N_\phi}{t} = \frac{Mr_o}{\pi r_o^3 t \sin\phi} = \frac{M}{\pi r^2 t \sin^3\phi}$$

where:

r_o = radius at point of interest

r = radius of dome

ϕ = see Figure 3.8-17

Since $N\phi / r_1 + N / r_2 = -Z$ and Z is equal to zero when $r_1 = r_2$, then $(\sigma_\theta) = -\sigma_\phi$.

Figure 3.8-17 shows the equations used for the total horizontal axial load to be considered at any location.

The design of the containment to guard against buckling is in accordance with ASME Code Section III, NE-3133 design rules with assumptions and boundary conditions inherent in the design rules. The loading combinations considered critical for buckling are:

a) Case 5 - Cold shutdown at ambient temperature. This case includes OBE seismic with external pressure.

b) Case 9 - Condition with safe shutdown earthquake. This case includes SSE seismic with no internal pressure.

The regions of the shell most likely affected by axial compressive loadings are the top head near the cylinder junction and the bottom tangent line on the cylinder. External pressure for cylinder and head are checked using design rules in NE-3133.3 and NE-3133.4. The cylinder is checked for axial compression using the design rules in NE-3133.6. Seismic and dead loads are considered to cause axial compression. For those areas with unequal biaxial compressive stresses, the ASME rules as modified by WRC 69 have been used. (Basically, the ASME Code allowable stress for compression determined by NE-3133 has remained the same for all "design conditions" except for SSE where the ASME Section III, Winter 1972 Addenda NE-3131 C(2) allows a 20 percent increase. The St. Lucie Unit 2 design did not use this 20 percent increase.)

3.8.2.4.2 Bottom Head Analysis

The bottom head knuckle is analyzed for the loading conditions specified in Table 3.8-3. Stresses produced by pressure are calculated using Chicago Bridge & Iron computer program 781 (see Subsection 3.8.2.4.6), and added directly to the stresses produced by external loads. Figure 3.8-19 shows the model used for program 781.

Results of the analysis are shown in Tables 3.8-4 and 3.8-5.

CBI's program 781 is also employed to analyze the portion of the bottom ellipsoidal head in the region of the ethafoam zone around the outside of the containment. The program is based on the Kalnins shells of revolution program[2]. The Kalnins program[4] is used widely by industry, and its results have been found to be in good agreement with other analytical methods.

The analysis is based on the fact that a rotationally symmetric shell may be divided into a number of short segments in the meridional direction and that the stiffness properties of each of these segments can be determined in relation to eight fundamental variables. By enforcing equilibrium and compatibility between each segment and applying boundary conditions, the value of the fundamental variables can be determined for each segment. Values between each segment can then be determined by integration.

The model used for the analysis is as shown on Figure 3.8-20. The vessel is taken to be rigidly fixed at elevation 19 feet, and the model is taken to extend upward to a point remote from any effect of the local discontinuity due to the fixity at the lower boundary or point of embedment. There has been no consideration of the concrete or shell below elevation 19 feet.

The pressure is included all along the model as an internal pressure equal to the maximum internal pressure of 44 psi.

The temperature gradients which are assumed to exist along the shell are as shown on Figure 3.8-21. The cases analyzed are for a steady state condition only, with the shell assumed to be at some ambient temperature before the gradient is applied along the shell. The maximum temperature differential along the shell is taken to be the difference between the ambient temperature and the temperature of the shell above the embedment region at a specified time after the accident.

It is also assumed that the temperature at the inside surface of the shell plate is the same as that at the outside surface, or that there is no temperature gradient across the thickness of the plate.

Thermal stresses are added to those due to dead loads and seismic loads. In accordance with maximum shear stress failure criterion and thin shell theory, stress intensities are then found as described previously in Subsection 3.8.2.4.1.
Buckling of the ellipsoidal head is not considered since it is embedded in concrete.

3.8.2.4.3 Air Lock Seismic Analysis

The containment vessel earthquake design includes the seismic effects of the air locks vibrating as an independent system. The seismic effect of this independent vibration is then added vectorially to all other seismic effects.

In the analysis, the vibration driving force on the air locks is determined by accelerations derived from the response spectra curves, shown on Figures 3.8-22 through 3.8-27. The vibrating driving force is considered to be independent of the vibration modes of the composite containment vessel shield building and foundation system.

For analytical purposes the locks are assumed to vibrate in three independent directions simultaneously as shown on Figure 3.8-28.

Mode I results in forces and moments being applied to the containment in the meridional plane. Mode II results in forces and moments being applied to the containment in the circumferential plane. Case III results in a radial thrust being applied to the containment shell.

Once the natural frequency of the lock is calculated for the longitudinal, circumferential, and radial direction of the containment vessel, it is possible to determine the fundamental period and thus the response acceleration to be applied to the lock. The response acceleration is calculated for the insert to shell junction and then applied to the lock dead loads to find stresses in the shell, using analytical methods developed from Reference 1.

For longitudinal and circumferential direction: (Mode I & II)

$$K = \frac{M}{\theta}$$

$$w = \left[\frac{K}{I_o}\right]^{0.5}$$

$$w = \left[\frac{M}{\theta I_o}\right]^{0.5}$$

$$T = \frac{2\pi}{\omega}(\text{sec})$$

where:

K = Spring constant of shell

M = Moment at shell

θ = Unit rotation at shell to insert junction

ω = Angular frequency of lock

I_O = Mass moment of inertia of lock about point of support on shell

T = Fundamental period of lock

The spring constant for the longitudinal and circumferential direction is determined by applying a unit deflection (1 radian) at the shell and determining M using Reference 5.

For the radial direction (Mode III):

$$K = \frac{P}{w}$$

$$T = 2\pi\left[\frac{W}{Kg}\right]^{0.5}$$

$$T = 0.32\left[\frac{W}{K}\right]^{0.5}$$

where:

W = Weight of lock plus insert

g = 3.86.4 in/sec^2

P = load

w = unit deflection at shell to insert junction

Stresses are checked at three locations; at the neck to insert junction, at the insert to shell junction, and at 1/2 $[Rt]^{0.5}$ from any local stress area.

Stresses are calculated in the insert and in the shell. An equivalent stress intensity iscalculated (per maximum shear theory) and compared to the ASME Code allowables.

3.8.2.4.4 Penetration Analysis

The penetrations are analyzed for compliance with the ASME Code. Area replacement is calculated using code rules. Welds for nozzles employing partial penetration attachment are analyzed using code rules. Nozzles with specified loads are investigated for pipe wall stresses and for stresses in the vessel shell.

Inserts for penetrations larger than 2 1/2 inch pipe size are checked for area replacement in accordance with paragraph NE 3332 of ASME Code, Section III. The pipe walls of nozzles with loads specified as thermal plus seismic are analyzed for primary stresses.

Stresses in the vessel shell resulting from loads applied to penetrations are calculated using CBI programs 1027 and 1036M (see Subsections 3.8.2.4.7 and 3.8.2.4.8).

Loads are applied in specified combinations on a penetration of interest and on adjacent penetrations that are on cardinal lines of the central penetration within a distance of $2\sqrt{RT}$ (80.0 in), where R is the radius of the shell and T is the thickness of the shell. This limit is chosen since the results of this type of analysis are questionable for greater distances. Load combinations are shown in Table 3.8-2. Each load is considered reversible for purposes of determining maximum stress intensity.

Pressure produces a complex state of stress in the shell and penetration at their intersection. As a rational means of estimating these stresses paragraph NE-3221 of Section III, has been used as a guide. This paragraph assumes that in the vicinity of a penetration reinforced in accordance with ASME rules, maximum membrane pressure stress does not exceed 1.0 S_m and the maximum surface stress does not exceed 1.5 S_m.

The loading combinations of thermal plus seismic and occasional loads due to fluid transient events are evaluated using ASME Code allowable stresses. ASME Code, Section III states that requirements of paragraphs NE 3221.1 through NE-3221.3 must be met for allowable stress intensities. Paragraph NB 3222.2 limits the primary plus secondary stress intensity to 3.0 S_m. An additional requirement of the code states that the local membrane stress intensity due to pressure and mechanical loads be limited to 1.5 S_m. In defining a local stress region, paragraph NB-3213 of Section III states that the distance over which the total membrane stress intensity exceeds 1.1 S_m may not extend more than $.5\sqrt{RT}$ and may not be closer than 2.5 $\sqrt{RT}$ to another region where the total membrane stress intensity exceeds 1.1 S_m. R is the mean radius of the vessel and t is the vessel wall thickness.

The loading combinations of thermal plus seismic and occasional load cases for penetration nozzles are evaluated using allowables of 90 percent of the allowables for an emergency condition defined in Figure NB-3224-1. Load combinations which include pipe rupture, occasional, and seismic loads are evaluated using the following allowable stress limits:

a) For loadings which include OBE, the primary membrane stresses do not exceed 0.9 S_y and local stresses at discontinuities do not exceed 1.5 S_y.

b) For loadings which include SSE, the primary membrane stresses do not exceed 1.0 S_y and local stresses at discontinuities do not exceed 1.5 S_y.

3.8.2.4.5 Program 405

This is a CBI program used for the analysis of a ring with a constant moment of inertia and modulus of elasticity. The loads are in the plane of the ring. The mathematics are based upon the Hardy-Cross Column Analogy for rings[(6)]. The loads can be moments, tangential, or radial to the ring. The printout is coefficients at incremental distances around the ring. The printout titles for the output are as follows:

X	=	angle and degrees as measured from a reference axes
V	=	a radial shear with force units acting in a radial direction through the ring
T	=	an axial thrust in the ring with units of force
M/R	=	a coefficient with units of force which when multiplied by the radius to the centroid equals a moment
EI/RR	=	a coefficient which when multiplied by the radius squared equals the rotation of the ring at the point
REI/RRR	=	a coefficient which when multiplied by the radius cubed equals the radial deflection of the point
CEI/RRR	=	a coefficient which when multiplied by the radius cubed equals the tangential deflection of the point

The following assumptions and limitations are considered:

a) ring has a uniform cross section and is made of homogeneous material,

b) depth of ring relative to its radius is too small to significantly influence the elementary flexural theory for straight beams,

c) unit stresses do not exceed proportional limit,

d) deflections from shear and axial stresses are negligible,

e) deflections are so small, the basic geometry remains essentially unchanged,

f) when the ring is attached to a cylindrical shell, distortion of the ring with respect to the shell is so small the only significant reactions in the shell are membrane-type shears, and

g) when the ring is attached to a cylindrical shell, the shell is held and loaded in such a way that membrane shear patterns are limited to the usual beam shear S = VcQ/I or torsional shear S_1 = Torsional moment/2 πR^2 or any combination of the two (Vc = total shear perpendicular to axis of a thin walled cylinder, Q = moment of area, I = moment of inertia and R = radius to centroid of ring).

Program 405 has been validated by comparison with solutions obtained from the formulae of Reference 7.

3.8.2.4.6 Program 781

The Shells of Revolution Program is the CBI Program 781. The program calculates the stresses and displacements in thin walled elastic shells of revolution when subjected to static edge, surface and/or temperature loads with arbitrary distribution over the surface of the shell. The geometry of the shell must be symmetric, but the shape of the median is arbitrary. It is possible to include up to three branch shells with the main shell in a single model. In addition the shell wall may consist of four layers of different orthotropic materials, and the thickness of each layer and the elastic properties of each layer may vary along the median.

The 781 program numerically integrates the eight ordinary first order differential equations of thin shell theory derived by E. Reissner[8]. The equations are derived such that the eight variables are chosen which appear on the boundaries of the axially symmetric shell so that the entire problem can be expressed in these fundamental variables.

CBI's program is an extensively revised Kalnin's Program[2]. The program has been altered such that a four x four force-displacement relation can be used as a boundary condition as an alternative to the usual procedure of specifying forces or displacements. This force-displacement relation can be used to describe the forces at the boundary in terms of displacements at the boundary, or the displacements at the boundary in terms of forces or some compatible combination of the two. In this manner, it is possible to study the behavior of a large complex structure.

It is also possible to introduce a "spring matrix" at the end of any part of the stress model. This matrix must be expressed in the form: force = spring matrix x displacement. In this manner it is possible to model the restraint of the sand cushion in the transition zone at the point of embedment. In addition to the above changes, the Kalnin's Program has been modified to increase the size of the problem that can be considered and to improve the accuracy of the solution.

3.8.2.4.7 Program 1027

This CBI program determines the stress intensities in a sphere or cylinder at a maximum of 12 points around an externally loaded round or square attachment. Stresses resulting from external loads are superimposed on an initial pressure stress situation. The program computes stresses at three levels of plate thicknesses: outside, inside, and centerline of plate. The 12 points investigated are shown on Figure 3.8-29: four points at the edge of attachment, at $1/2\sqrt{RT}$ (where R is the radius of the shell and T is the thickness of the shell) from the edge of attachment and at the edge of reinforcement.

The program determines three components for each stress intensity:

σ_x = normal stress parallel to the vessels longitudinal axis

σ_ϕ = normal stress in a circumferential direction

τ = shear stress

The program has an option whereby the penetration load can be considered reversible or nonreversible in direction. Under the reversible option, only the data associated with the most severe loading situation is printed.

Most of the analysis and notation used in the program is taken directly from Reference 1. Use of the program requires complete familiarity with this publication.

The program contains extrapolations of the curves for cylinders in Reference 1 for γ up to 570.

3.8.2.4.8 Program 1036M

This CBI program determines the stress intensities in a "Jumbo" insert plate (a reinforcing plate with multiple penetrations) in a cylindrical vessel at eight points around one of these penetrations due to the loading on that penetration plus the loadings on the four adjacent penetrations all as superimposed on an initial stress situation. It does this at three levels of plate thickness: outside, inside, and centerline of plate. The eight points investigated are shown on Figure 3.8-30. The four points on radius R are at the junction of the penetration and the insert plate. The other four points are other points of interest, normally, they are at the midpoints in the clear space between penetrations or at the edge of reinforcing. Although five penetrations are considered, each point is analyzed as though it were only influenced by two (the central penetration plus the penetration on the same axis as the point concerned).

The program also determines three components for each stress intensity:

σ_x = normal stress parallel to the vessel's longitudinal axis

$\sigma_\emptyset$ = normal stress in a circumferential direction

τ = shear stress

Each of these is composed of three subcomponents, one due to the central penetration's loading, one due the loading on the next adjacent penetration and an initial stress component (input).

The program has an option whereby the penetration loads are considered reversible or nonreversible in direction. Under the reversible option only the data associated with the most severe loading situations is printed out.

Most of the analysis and notation used in the program is taken directly from the Welding Research Council (WRC) Bulletin #107(1).

The analysis in WRC 107 is for a single penetration. This program analyzes the several penetrations individually, using WRC 107 techniques verbatim, and then through superposition obtains the composite results. The adjacent penetrations must be on a cardinal line of the central penetration in order to use WRC 107 methods. This has required a very conservative extension of the WRC 107 analysis. WRC 107 analysis applies only to the points on the penetration to shell juncture. This program makes stress determinations at points removed from the junction by fictitiously extending the radius of any penetration to any point at which a stress determination is desired. This disregards the statement in WRC 107 that "these stresses attenuate very rapidly at points removed from the penetration to shell juncture". Furthermore, in some cases, the moment induced stresses at both the juncture and at points removed from the juncture are increased by 20 percent per discussion in WRC 107. Table 3.8-6 shows the cases for the calculation of the parameters (per WRC 107) and stresses.

The program contains extrapolations of the curves in WRC 107 for T up to 600. The program is limited to the domains and range of Figures 1A through 4C in WRC 107 ($0 < \beta \leq 0.5$ and $5 \leq T \leq 600$).

3.8.2.4.9 Program 1392

Stresses in pipe are computed using CBI's Program 1392, which utilizes classical beam theory for computing stresses due to external loads. Pressure stresses are added in by superposition in the appropriate directions. Within the limits of reinforcement or next to the shell, the circumferential pressure stress is the membrane stress in the circumferential direction due to pressure in the vessel (PMSϕ). Outside the limits of reinforcement or away from the shell, the circumferential pressure stress is the membrane stress in the circumferential direction due to pressure in the pipe (PMϕ). The stresses are computed for locations A,B,C and D (refer to Figure 3.8-31) and stress intensities are then computed as per WRC 107.

Referring to Figures 3.8-31 and 32:

$$S_x \text{ at A } = \frac{-P}{A} - \frac{M_L}{S} + \frac{pr_m}{2t_n} \quad ; \quad S_X \text{ at C } = \frac{-P}{A} - \frac{M_c}{S} + \frac{pr_m}{2t_n}$$

$$S_x \text{ at B } = \frac{-P}{A} + \frac{M_L}{S} + \frac{pr_m}{2t_n} \quad ; \quad S_x \text{ at D } = \frac{-P}{A} - \frac{M_c}{S} + \frac{pr_m}{2t_n}$$

$$S_\phi \text{ at A, B, C} = \frac{pr_m}{t_n} \quad \text{if element is away from shell}$$

= PMSϕ if element is within limits of reinforcing

$$\tau \; at \; A = \frac{V_c}{\pi \, r_m \, t_n} + \frac{M_T \, (r)}{J} \; ; \quad \tau \; at \; C = \frac{-V_L}{\pi \, r_m \, t_n} + \frac{M_T \, (r)}{J}$$

$$\tau \; at \; B = \frac{V_c}{\pi \, r_m \, t_n} + \frac{M_T \, (r)}{J} \; ; \quad \tau \; at \; D = \frac{-V_L}{\pi \, r_m \, t_n} + \frac{M_T \, (r)}{J}$$

Stress intensities are determined as follows:

$$S1 = \frac{1}{2} \left\{ S_x + S_\phi + \sqrt{(S_x - S_\phi)^2 + 4\tau^2} \right\}$$

$$S2 = \frac{1}{2} \left\{ S_x + S_\phi + \sqrt{(S_x - S_\phi)^2 + 4\tau^2} \right\}$$

S3 = 0

The stress intensity SI = the maximum of | S1-S2 | , | S2-S3 | ,or | S1-S3 |.

3.8.2.5 Structural Acceptance Criteria

The load combinations for the containment vessel are specified in Subsection 3.8.2.3 and summarized in Tables 3.8-1 and 3.8-3. The allowable stresses for each of the load cases are summarized in Table 3.8-7. Tables 3.8-8 and -9 compare the calculated stresses with the allowable stresses for those critical sections of the containment shown on Figure 3.8-33.

Pipe loads at penetrations are investigated as local effects separately using the load combinations and stress limits for each type of penetration as specified in Table 3.8-2.

The allowable stresses are determined by the methods discussed below.

3.8.2.5.1 Allowable Buckling Stresses for Unstiffened Hemispherical Head

Compressive stress resultants in the top head are compared to the allowable stresses obtained from the paragraphs entitled "Biaxial Compression-Equal Unit Forces" and "Biaxial Compression-Unequal Unit Forces" of the Welding Research Council Bulletin #69[9]. Use of these allowables for the spherical dome is based on the assumption that the dome acts as a cylinder with the radius equal to the radius of the dome.

Three cases are considered (refer to Figure 3.8-34):

a) For a uniaxial compressive stress resultant and for biaxial unequal tensile and compressive stress resultants:

N_φ allowable = $1.8 \times 10^6\ t^2/R$

where:

t = shell thickness

R = dome radius

b) For biaxial equal compressive stress resultants:

N_φ allowable = $0.9 \times 10^6\ t^2/R$

c) For biaxial unequal stress resultants: This case is treated as the summation of the uniaxial condition with equal stress.

$$\frac{N_\theta - N_\phi}{1.8 \times 10^6\ t^2/R} + \frac{N_\phi}{0.9 \times 10^6\ t^2/R} \leq 1$$

3.8.2.5.2 Allowable Buckling Stress for Cylindrical Vessel

Allowable buckling stresses for cylindrical vessel are shown on Figure 3.8-35.

a) Meridional or Axial Stress

The maximum allowable compressive stress used in the design of cylindrical shell subjected to loadings that produce longitudinal compressive stress is in accordance with ASME Code, Section III, Article NE-3133.

b) Circumferential Stress

Generally speaking, circumferential compression results from external pressure loading. The criteria of ASME Code, Section III, Article NE-3133 is used to analyze circumferential buckling. These rules provide a safety factor of 3.0 against shell buckling.

3.8.2.5.3 Allowable Buckling Stresses for Cylindrical Vessel During Post Weld Heat Treatment

Using E. O. Bergman's formula[10] for the allowable buckling stress in a cylinder:

σ_ϕ = longitudinal compressive stress = $\frac{PD}{t}$

where:

$P = \frac{E}{16} \frac{t^2}{L_e}$

E = modulus of elasticity

L_e = radius

D = diameter

t = shell thickness

To calculate the allowable circumferential compressive stress:

σ_θ = S/2

where:

$S = \frac{1.3\, Et^{1.5}}{L\sqrt{D}}$

t = thickness of cylindrical shell

L = vertical direction length (tangent line to tangent line)

D = diameter of cylindrical shell

3.3.2.5.4 Allowable Weld Stresses

Weld metal joining or attaching pressure parts meets specified Charpy V notch impact test requirements.

a) ASME Code Allowable Weld Stresses

For full fusion welds, the allowable weld stresses are in accordance with Subsection NE of the ASME Code, Section III. The allowable is the same as the parent metal.

For partial depth groove welds, the allowable weld stress on the effective depth is calculated by multiplying an inspection factor by a load factor and then by S_m of weaker material. The inspection factor used is 0.8. The load factor used is 1.0 for loads perpendicular to the axis of the weld, 0.875 for any combination of perpendicular and parallel loads, and 0.75 for a load parallel to the axis of the weld. For simplicity, an allowable stress of $0.8 \times 0.75 \times S_m = 0.6\ S_m$ is used for all partial groove welds except where a higher allowable is required and is permissible as above.

For fillet welds, the allowable stress is per NE 3356(b) of ASME Code, Section III. The allowable stress is equal to 0.55 S_m (of material) on a minimum leg equal to 0.55/0.707 S_m on throat.

b) AISC Allowable Weld Stresses

The allowable stress for fillet welds and groove welds are in accordance with paragraph 1.5.3 of AISC.

3.8.2.6 Materials, Quality Control and Special Construction Techniques

The materials used in the construction of the steel containment unless otherwise identified in Subsection 3.8.2.6.1 comply to Article NE-2000 of Subsection NE, Section III of ASME Code. The quality control program proposed for fabrication and construction is in accordance with the applicable provisions of Articles NE 4000 and NE-5000 of the Code.

3.8.2.6.1 Materials

The materials used in the containment vessel are listed in Table 3.8-10. The containment vessel and the equipment hatches and personnel air locks are fabricated of ASME-SA 516 Grade 70 fully killed pressure vessel quality steel plate except that impact test requirements are as specified in the ASME Code, Section III, NE-2300 for a minimum service temperature of 30 F.

Charpy V-Notch specimens (ASME SA 370 Type A) used for impact testing of all product forms are in accordance with the requirements of the ASME Code, Section III, NE-2353.

The ferritic material in the fabrication of the containment vessel has a nil ductility transition temperature of zero degrees maximum when tested in accordance with the appropriate specification of the material.

During reactor operation or leak rate testing the containment vessel metal temperature is above 30 F.

The following materials are used in the mechanical penetration assemblies:

a) Flued head fittings are in accordance with the requirements of:

(1) The ASME Code, Section III, Paragraphs NB 2100, NB 2200 and NB 2300;

(2) ASTM Material Specifications A182 Grade F304 alloy steel for stainless steel flued heads and A105 Grade II carbon steel for carbon steel flued heads.

b) Carbon steel pipe is per ASTM A106, Grade B. For pipe greater than 24 in. NPS ASTM A155, Grade KC-65, Class I, Fire-box Quality may be used.

c) Stainless steel pipe is per ASTM A376 or A312 (seamless only) grade TP-304 or TP-316. For pipe greater than 12 in. NPS ASTM A358, Class I, Grade TP-304 or TP-316 may be used.

d) The expansion joint material is ASTM A240 Type 316L.

The foregoing materials in all cases are compatible with the materials of the process line.

After fabrication surfaces are cleaned to remove oil, grease, dirt, loose rust, loose mill scale, and other foreign substances. The removal of oil and grease is accomplished before the mechanical cleaning, using mineral spirits or other paraffin-free solvents. Clean cloths and clean fluids are used to avoid leaving a film of greasy residue. The use of chipping tools which produce cuts, burrs, and other forms of excessive roughness are not permitted.

The carbon steel surfaces are blast cleaned prior to painting in accordance with Steel Surfaces Painting Council Specification SSPC-SP-10 "Near White Blast Cleaning". See Subsection 6.1.2 for a discussion on coating applied to the containment surface.

3.8.2.6.2 Construction Quality Control
(historical Information, Ref.: specification FLO-2998-757)

a) General Requirements

Test, code and cleanliness requirements accompany each specification or purchase order for materials and equipment. Tests to be performed by the manufacturers are enumerated in the specifications together with any requirements for test witnessing by inspectors. Fabrication and cleanliness standard including final cleaning and sealing are also described, together with shipping procedures. Standards and tests are specified in accordance with applicable regulations, recognized technical society codes and current industrial practices.

b) Welding Procedures

Welders and welding procedures are qualified in strict accordance with, and meet the requirements of, Section IX of the ASME Code. Prior to the start of welding operations, the vessel manufacturer provides the applicant and his engineer with copies of the qualified welding procedure specifications and reports of the results of the qualification tests for each welder or welding operator.

Longitudinal and circumferential welds in the shell of the containment vessel are double-bevel full penetrations butt welds. Joints in any accessories subject to the ASME Code are full penetration welded.

Welds subject to the ASME Code are 100 percent radiographed or otherwise examined in accordance with the ASME Code. Welds which cannot be radiographed, or where the interpretation of radiographs would be open to doubt are examined by the magnetic particle, liquid penetrant or ultrasonic method.

In manual arc-welding, the electrodes are of the low hydrogen type. Welding filler metal has mechanical properties which are similar to the base metal. Automatic welding is by the submerged-arc process.

Preheat at 200 F minimum in accordance with the ASME Code is applied to seams whose thickness exceeds 1 inch regardless of the surrounding air temperature. Preheat at 100 F is applied to thinner seams if the surrounding air temperature falls below 50 F and/or the surfaces to be welded are damp. Preheat is applied to ferritic material less than one in. thick and more than 3/4 in. thick, regardless of surrounding air temperature, unless the vessel manufacturer's qualified procedures have been performed without preheat under these conditions.

Post-well heat treatment is performed as required by and in accordance with the ASME Code (See Subsection 3.8.2.6.4)

c) Nondestructive Examination

Nondestructive testing, in addition to the containment pressure and leak rate tests, is performed after the post weld heat treatment operation by means of magnetic particle or liquid penetrant examinations on welds that join the containment penetrations and access openings to the containment wall.

The radiographic examinations on the welds are made prior to the post weld heat treatment operation in accordance with Section III, Subsection NE of the ASME Code, with controlled heating and cooling rates during postweld heat treatment which provides a safeguard against propagation of defects.

Repairs, if required to meet acceptance standards, are examined by the original nondestructive testing examination acceptance method.

d) Materials Testing

Charpy V-Notch impact tests are made on material, weld deposit, and the base metal weld heat affected zone employing a test temperature of not higher than 0 F. The requirements of the ASME Code, Subsection NE, are met for all materials under jurisdiction of the code.

An impact test of the weld deposit and base metal weld heat affected zone is made for each welding procedure requiring ASME Code, Section IX qualifications.

Specimen removal from the test weld conforms to the requirements of ASME Code Section IX and removal of the impact specimens is in accordance with Paragraph NE-4300 of ASME Code Section III.

e) Penetration Assemblies

Since certain welds are not accessible for inspection subsequent to completion of a guard pipe assembly, special attention is placed upon fabrication and the tests and inspections performed during manufacture. For example:

1) The flued head fitting, and that section of the process pipe and guard pipe between the flued head fitting and the nearest weld joint outside the containment vessel are hydrostatically tested in accordance with the requirements of ASME Code, Section III.

2) Water used for the hydrostatic testing of stainless steel penetration assemblies does not have a halide content exceeding 20 ppm.

3) The expansion bellows assembly is tested in accordance with ASME Code, Section III, Articles NC-3649 and ND-3600 (as applicable.)

4) Each bellows is flexed 10 times through the axial equivalent of its specified axial and lateral deflections (without damage or deformation). This test is performed prior to hydrostatic tests.

5) The carbon and low alloy steel pipe and fittings that make up the process pipe or the guard pipe are subject to impact testing. The impact testing conforms to the procedures and configuration for Charpy V-notch specimens, Type A, Figure II, as specified in ASME SA-370, and is in accordance with the requirements of ASME Code, Section III, Articles NB-2300 and NC-2300. The specimen temperature during impact testing is 20 F. Impact values for pipe and wrought fittings meet the ft lb values indicated in Section III. Materials failing to meet this requirement are rejected.

6) Welding procedure qualifications and welder performance qualifications are in accordance with Section III and IX of the ASME Code and Ebasco specifications as applicable. Repair welding of base materials follows a procedure that is qualified to join base materials. Repaired base materials are stress relieved as per Section III of the ASME Code and Ebasco specifications. The finished repaired areas are nondestructively tested to the same level originally specified for the base material. Prior to weld repair, repair cavities are examined by magnetic particle or liquid

penetrant tests to assure complete defect removal. Electrodes used for repair welding deposit a composition within the limits of the chemical composition of the base material. Welding materials are identified and controlled so as to be traceable to each welded section. Major weld repairs (exceeding 20 percent of wall thickness, one in. or encroaching on minimum wall thickness) are reported to the Applicant prior to repair.

7) Process piping materials, welds or pressure retaining components undergo nondestructive testing examination. These tests include;

(a) Longitudinal Seam Welds of Primary Bellows and Pipes other than Process Pipe:

Full 100 percent radiography is performed.

(b) Secondary Bellows:

Longitudinal seam welds are examined by liquid penetrant methods.

(c) Joint of Primary Bellows to Flued Head:

These joints are fully radiographed in accordance with Article NE-5120 of Section III. Closure welds for which radiography is impractical are examined by a magnetic particle method for carbon steel or a liquid penetrant method for stainless steel on the root and final weld surface in lieu of radiography. Permanent backings rings may be used provided the requirements of Article NC-3649 of ASME Code, Section III are satisfied.

(d) Guard Pipe Joint Welds:

Examination by either a magnetic particle or liquid penetrant method is performed.

Permanent backing rings may be used.

(e) Fillet Welds:

Examination by liquid penetrant methods is performed.

(f) Process Pipe:

Girth and longitudinal seam welds are 100 percent radiographed and examined by either magnetic particle or liquid penetrant method. In addition, all other examinations required by applicable ASME or ASTM specifications for the associated process line are performed.

(g) Forging:

The pressure retaining forging of penetrations assemblies are given 100 percent ultra-sonic examination. Whenever practicable, the stage at which ultra-sonic testing is performed for acceptance is in the finished condition after final heat treat unless otherwise specified in writing by the Project Quality Assurance Engineer. In addition, the external and accessible internal surfaces are examined by either magnetic particle or liquid penetrant methods including all final machined surfaces and weld end preparations.

(h) Castings:

The pressure retaining castings of penetration assemblies are 100 percent radiographed after final heat treatment. The radiography is performed at a stage of machining that is in accordance with ASME Code, Section III, Paragraph NB-2577.2. In addition, the external and accessible internal surfaces are examined by either magnetic particle or liquid penetrant methods including the final machined surfaces and weld end preparations.

8) The vendor is required to document that the wall thickness for each pressure retaining component is within acceptable limits as specified by specification and/or engineering drawing. The vendor submits for approval the proposed method and documentation procedure as well as marked up engineering drawings or sketches depicting those sections to be measured in order to confirm manufacturer's compliance.

The tests and inspections described above were applied to complete penetration assemblies procured as part of original plant construction. Replacement component parts for the secondary penetrations (e.g., secondary bellows) have been designed, fabricated, and tested in accordance with standard manufacturing, design and Code requirements to be functionally equivalent to the original design.

3.8.2.6.3 Construction

During the erection of the containment vessel, it is supported by 24 temporary steel pipe column assemblies welded directly to the vessel shell. The temporary supports are removed after the containment vessel is completely constructed and post weld heat treated and after a portion of permanent base foundation is placed. The supports are cut not closer than 1/4 in. from the surface of the shell plate and the remaining support material and welds are removed by chipping and grinding smooth with the shell face.

A placing and grouting procedure is used to fill void areas beneath the containment vessel. The placing and grouting procedure results in a continuous support of the vessel.

Concrete placements are made according to the plan and sequence shown on Figure 3.8-36.

Vessel fabrication and erection tolerances are in accordance with ASME Code, Section III, Paragraph NE-4220 with the following exceptions:

a) the difference between the maximum and minimum diameter at any cross section does not exceed 0.5 percent of the nominal diameter of the vessel,

b) the diameter at any cross section does not deviate more than 0.25 percent of the nominal diameter,

c) out of plumb does not exceed 0.25 percent measured between tangent lines after making allowances for out of roundness as specified above, and

d) the surfaces of the top and bottom heads do not deviate outside the specified shape by more than 1.25 percent of the nominal diameter or inside by more than 0.625 percent of the nominal diameter.

3.8.2.6.4 Post-Weld Heat Treatment

Post weld heat treatment is performed as required by and in accordance with the ASME Code. For field post weld heat treatment, after the vessel shell and ellipsoidal bottom has been completely erected and welded, and the top temporarily closed with a diaphragm, it is externally insulated with a temporary blanket type insulation suitable for the post weld heat treatment operation, and is attached mainly by banding. Temporary supports, covers and insulation required for effecting the post weld heat treatment operation are attached with a minimum of welding to the vessel.

Thermocouples of the iron-constantan type are used to monitor temperatures during the post weld heat treatment operation, and are so located as to indicate representative temperatures of areas of the vessel. Thermocouples are used to monitor temperatures of the vessel shell and bottom head and to serve as control points on the installed portions of the top head (which do not require heat treatment because of lesser thickness) during the heat treatment cycle. The thermocouples are attached by welding to the outside surfaces of the vessel. The hot or measuring junctions of the thermocouples are protected by special sleeves which are welded to the part being monitored. The heating of the vessel is done with luminous flame oil burners firing through openings in the bottom and sides of the vessel and arranged in such a way that the heat will be evenly distributed throughout the vessel during the heatup and holding periods without flame impingement on any part of the vessel. Combustion products are exhausted through an opening in the top of the vessel.

Temperatures of the thermocouple locations are simultaneously recorded against time on a direct reading strip chart or charts using multiple point potentiometer type instruments.

Heatup rates, holding temperatures and times, cooldown rates and temperature gradient restrictions are in accordance with Section III, Subsection NE of the ASME Code.

During heatup from ambient to holding temperature, the vessel becomes approximately 14 in. larger in diameter. Special attention is given to the temporary peripheral supporting columns during the post weld heat treatment cycle. In order to prevent the development of excessive stresses at the columns-to-vessel connections and in the columns themselves, provisions are made for the bases of the columns to move radially outward during the heatup period and inward during a cooldown.

Upon completion of the post weld heat treatment operation, the insulation and other temporary items are removed and temporary attachment weldments ground smooth.

3.8.2.7 Testing and Inservice Inspection Requirements (Historical Information, Ref. Specification no. FLO-2998-757)

Pneumatic testing is performed in accordance with the applicable requirements of Article NE-6000 of the ASME Code to demonstrate the structural integrity and leaktightness of the completed vessel. Testing is performed after concrete is placed under and within the containment vessel. The vessel is not pressurized until the ambient temperature is 30 F or above. A halogen sniffer leach test is performed for all bottom head welds that are later embedded in concrete, which are inaccessible during the leak rate and soap bubble tests described below. The test procedure utilizes a combination pressure-vacuum box to pressurize one side of the weld joint with a halogen-air mixture. The evacuated outer compartment of the box seals the inner compartment over the weld joint. Penetrations that are eventually embedded in concrete are provided with blockouts so that the concrete may be placed after the testing is performed.

The temporary blind flanges, blanking off plates and gasketing required to seal the containment vessel for testing purposes are removed following the successful performance of tests. Ends of pipe and cable sealing details are properly prepared for connections. The testing connections are properly sealed and are permanently left in place for future testing.

Tests and retests on the containment vessel as listed below are made upon completion of vessel erection and after placement of concrete.

a) Soap Bubble Tests

Upon completion of the containment vessel, a soap bubble test at five psig is performed for all welds and seals. The tests are also performed on each door of the personnel air locks. A second soap bubble test of the welds and seals is conducted at a pressure of 44 psig, upon completion of the over pressure test described below. Soap bubble tests are performed by applying a thick soap solution to the welds and seals after pressurizing, checking for bubbles or dry flaking as indications of leaks. Any leak detected by the soap bubble tests are repaired prior to proceeding with further tests.

b) Leak Testing of Personnel Locks

With the containment vessel at atmospheric pressure, the air locks are pressurized with air to a pressure of 50 psig. The welds and seals are observed for visual signs of distress or noticeable leakage. The air lock pressure is then reduced to a pressure of 44 psig and a soap bubble test performed. All leaks and questionable areas are clearly marked for identification and subsequent repair. During the overpressure test, the inner door is blocked with hold down devices to prevent unseating of the seals. If leaks are detected, the internal pressure of the air locks is reduced to atmospheric pressure and all leaks are repaired, after which the air locks are pressurized to a pressure of 44 psig with air and all areas suspected or known to have leaked during the previous test are retested by the soap bubble method. This procedure is repeated until no leaks are discernible by this means of testing.

If a personnel lock has undergone initial tests before it is attached to the containment vessel, a soap bubble test of the seals and welds at the pressure of 44 psig is performed after the lock has been made an integral part of the vessel.

c) Overload Pressure Test

After successful completion of the initial soap bubble test, a pneumatic pressure test is made on the containment vessel and on each of the personnel locks at a pressure of 50 psig, held for a minimum of one hour. The inner, as well as outer doors, of the personnel locks are tested at this pressure. The test pressure is maintained on each individual airlock door for at least one half hour.

d) Leak Rate Test

After successful completion of the soap bubble and overload pressure tests, a leak rate test at a pressure of 44 psig is performed on the containment vessel with the personnel air lock inner doors closed and atmospheric pressure in the locks. Pressure is maintained for the length of time required to demonstrate full compliance with the air-tightness requirements. Continuous readings are taken at least once an hour and continued over a minimum period of 24 hours until it is satisfactorily shown that the leakage in any 24 hour period does not exceed 0.2 percent of the total contained weight of air at test pressure and ambient temperature. Leakage is determined by the "Reference System Method" which consists of measuring the pressure differential between the contained air and that of a hermetically sealed references system within the containment vessel.

e) Integrated Leak Rate Testing

Per 10 CFR 50 Appendix J (see Subsection 6.2.6) and ANSI N45.4, the equipment used for the integrated leak rate tests (ILRT) consists of but is not limited to the following:

1) Absolute pressure is measured by utilizing precision pressure indicators.

2) A barometer and thermometer for measuring outside pressure and temperature are provided.

3) In order to account for temperature effects, resistance temperature detectors (RTDs) are located within the containment. These detectors are placed spatially within a calculated fractional volume. Therefore, the average temperature is a weighted average.

4) Relative humidity or dewpoint detectors are located in the same manner as the RTDs discussed above.

5) Leakproof stuffing boxes are provided for all lines passing through the vessel shell.

The accuracy of the detectors are as follows:

(a) Pressure $\pm$ 0.02 percent of reading (0-100 psia scale)

(b) Temperature $\pm$ 0.1 °F

(c) Humidity $\pm$ 2.5 percent RH or $\pm$ 1.0 to 2.0 °F dewpoint

6) A supplemental test is performed whereby the leak rate measurement are validated independently. This validation is performed for a sufficient duration to accurately establish validation following the integrated leak rate test measurements.

f) Operational Testing

After installation, each personnel lock including all latching mechanisms and interlocks is given an operational test consisting of repeated operation of each door and mechanism to determine that all parts are operating smoothly without binding or other defects. Defects encountered, if any, are corrected and retested. The process of testing and correcting is continued until no defects are detected. Provisions are also made whereby the doors are leak tested after each opening through pressurization of the gasket interspace (see Figure 3.8-37).

g) Penetration Testing

Pipe penetrations which must accommodate thermal movement are provided with expansion bellows. The bellows expansion joints are designed to withstand containment vessel maximum internal pressure and are checked for leak-tightness when the containment vessel is pressurized. In addition, these joints are provided with a second seal and test tap so that the space between the seals can be pressurized to the maximum internal pressure to permit testing the individual penetrations for leakage at any time.

Penetrations which are welded directly to the containment vessel are leak tested by pressurizing the entire containment vessel.

Electrical penetrations are also provided with seals that are testable. The test taps and seals are so located that the leakage test of the electrical penetrations can be conducted without entering or pressurizing the containment vessel. Electrical penetration assemblies are tested in accordance with IEEE 317-76, "Electrical Penetration Assembly in Containment Structure of Nuclear Power Generating Stations."

The containment closures which are fitted with resilient seals or gaskets are separately tested to verify leak tightness. The covers on flanged closures are provided with double seals and with a test tap which allow pressurizing the space between the seals without pressurizing the entire containment system. In addition, provision is made so that the space between the airlock doors can be pressurized to full containment vessel maximum internal pressure.

h) In-service Testing

In-service periodic leakage rate tests of the containment vessel and leak tests of the testable penetrations are conducted to verify their continued leak-tight integrity. See Subsection 6.2.6.

i) Surveillance of Structural Integrity

The steel shell pressure containment vessel is designed, fabricated, inspected and pressure tested in accordance with the ASME Code and is protected by the concrete Shield Building to assure continued structural integrity over the life of the unit. The vessel receive a Code stamp from an authoritative body (ASME) and represents the most recent developments in the techniques of pressure vessel design and fabrication that are backed up by years of research, testing and successful in-service experience. Therefore, there is no need for any special in-service surveillance program other than visual inspection of the exposed interior and exterior surfaces of the containment vessel.

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURE OF STEEL CONTAINMENT

3.8.3.1 Description of the Internal Structures

The internal structures of the steel containment consist of concrete and steel components. The major concrete components are the primary and secondary shield walls, the refueling cavity, the operating floor and the enclosures around the pressurizer and steam generators. The major internal steel components are the Reactor Coolant System supports, the refueling cavity liner, main steel framing, miscellaneous platforms, and restraints for cable trays, HVAC ducting and piping. The concrete and steel internal structures which are necessary to ensure the integrity of the reactor coolant pressure boundary, to maintain the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate accident consequences that could result in potential offsite doses which are significant fractions of the guidelines established for design basis accidents are designed to seismic Category I requirements.

The internal structures are supported on the concrete floor fill (Subsection 3.8.5) placed in the bottom of the steel containment. The entire Reactor Coolant System is located within the compartments formed by the concrete fill floor, the primary and secondary shield walls and the concrete enclosures around the steam generators and pressurizer (see Figures 3.8-38 and 39).

3.8.3.1.1 Primary Shield Wall

The primary shield wall around the reactor vessel (RV) extends from the concrete fill at the base (elevation 18.0 feet) up to the RV flange section and is connected to the refueling cavity walls (see Figures 3.8-38 and 39). It is a cylindrical structure with an internal diameter of 22 feet and a minimum wall thickness of 7.25 feet. The steel girders supporting the reactor vessel have their ends embedded in the primary shield wall. Piping penetrations are provided through the shield wall for the Reactor Coolant System.

The primary shield wall is designed to provide the following functions:

a) biological shield during normal operation

b) missile shield to prevent missiles from impacting upon the reactor vessel

c) support structure for the reactor vessel, pipe and HVAC duct restraints and intermediate platforms.

3.8.3.1.2 Secondary Shield Wall

The secondary shield wall is a cylindrical structure which encloses the pressurizer, steam generators, reactor coolant pumps and reactor coolant piping. It extends from the top of the concrete fill at elevation 18 feet to elevation 58 feet. The secondary shield wall is 4 feet thick and has an inside radius of 49 feet. Vent openings are provided at the top and bottom of the secondary shield wall to minimize the internal pressure from a postulated pipe break accident. The upper vent openings are constructed so that the biological and missile shielding functions are not compromised. The lower openings, which also facilitate the drainage of containment spray water into and out of the area within the secondary shield wall, are provided with carbon steel trash racks (see Figures 3.8-38 and 39).

The secondary shield wall provides biological shielding to permit access inside the containment vessel during plant operation, provides a missile shield to protect the steel containment vessel, and provides support for the operating floor, intermediate platforms and for the pipe, HVAC duct and cable tray restraints.

3.8.3.1.3 Refueling Cavity

The refueling cavity provides storage for the reactor vessel internals and enables fuel transfer to the Fuel Handling Building via the fuel transfer tube. The refueling cavity walls extend from elevation 18 ft to elevation 58 ft. The N-S concrete walls are six ft thick and the E-W concrete wall is three ft thick. The interior face of the refueling cavity walls and floor is lined with 3/16 in. or 1/4 in. thick stainless steel plate to make it watertight (see Figures 3.8-38 and 39). A leak detection system is provided to monitor liner leak during refueling operations.

The walls of the refueling cavity provide a biological shield and a watertight compartment, and serve as a support structure for the operating floor, intermediate platform and the missile shield above the reactor vessel.

3.8.3.1.4 Enclosures Around Steam Generators, Pressurizer and Regenerative Heat Exchanger

The rectangular enclosure for the steam generators consists of a wall with a minimum thickness of 2.16 ft, extending from the operating floor (elevation 62 ft) to elevation 76 ft. This enclosure provides biological shielding, contains potential missiles and is also the support structure for the steam generator upper supports.

The pressurizer enclosure consists of a two ft thick wall with a removable two ft thick roof. The wall extends from the operating floor to elevation 85.67 ft. The concrete roof has a steel liner on the interior face and is attached to the wall with anchor bolts. The enclosure was designed to provide biological and missile shielding. Evaluation JPN-PSL-SENP-95-046 justified removal of this shield during power operations.

The regenerative heat exchanger enclosure consists of a two ft thick wall extending from the fill concrete (at elevations 18 and 23 ft) to elevation 38 ft, where a steel shield plate is attached to the wall with anchor bolts. This enclosure provides biological shielding.

Concrete support foundations for the pressurizer and steam generator are shown in Figure 3.8-40.

3.8.3.1.5 Steel Internal Structures

The reactor vessel (RV) is supported at three points on a steel girder column assembly within the reactor vessel cavity (see Figure 3.8-41). The columns are bolted to the underside of the steel girders and to the RV cavity floor. The horizontal ends of the steel girders are embedded in the primary shield wall. Loads from the RV are transferred to the steel girder - column assemblies by means of the support shoes welded to the nozzle of the RV and the bearing plates on top of the steel girders. The support shoes are free to slide longitudinally along the axis of the nozzles and only a frictional load is transmitted to the support structures in this direction. Transverse and downward loads are transmitted into the girder in direct bearing. The horizontal and vertical loads are then transferred to the supporting concrete by means of the embedded ends of the steel girders and the anchor bolt/anchor plate anchorages.

The steam generator is supported on a sliding base and anchored to a concrete pier (see Figure 3.8-42). The sliding base rests on bearings which permit free movement of the steam generator during thermal expansion of the primary coolant loop. The sliding base also restrains the steam generator in the unlikely event of a pipe break. Load transfer into the concrete is by means of bearing and anchor bolt/anchor plate assemblies. The steam generator also has an upper support composed of hydraulic snubbers and builtup steel assemblies. The primary purpose of the upper support is to take the seismic and LOCA loads which are then transferred to the concrete of the SG enclosure by means of the anchor bolt/anchor plate anchorages.

The reactor coolant pumps are supported by vertical compression spring supports integral with the pump. These spring supports are supported on pipe columns, braced steel structures or base plates which are anchored to the supporting concrete by anchor bolt/anchor plate anchorages (Figure 3.8-43). Horizontal upper supports (snubbers and structural steel brackets) are also provided for pump motors. In a seismic event, the loads are taken by the snubber and bracket and then transferred to the supporting concrete through the embedded plate with shear lugs and anchor bolt/anchor plate anchorages (see Figures 3.8-44, 45 and 46). Pump stops

(Figures 3.8-47 and 48) are also provided to restrain the reactor coolant pumps in the unlikely event of a break of the Reactor Coolant System piping. The loads are transferred to the surrounding concrete through embedded plates and anchor bolt/anchor plate anchorages. The upper and lower RCP cable restraints have been permanently removed.

Pipe stops are provided in the primary shield wall penetration sleeves to restrain the reactor coolant piping cold leg and limit break opening size in the unlikely event of a pipe break (see Figure 3.8-51). The loads are transferred to the surrounding concrete by means of bumper plate assemblies and anchor studs.

The pressurizer is supported by a built-up steel ring girder - four columns assembly (Figure 3.8-52). The pressurizer skirt is bolted to the top flange of the ring girder. The four columns are bolted to the bottom flange of the ring girder and anchored into the concrete pedestal. The ring girder is also tied to the concrete walls by means of a steel diaphragm welded to embedded plates in the walls. Vertical loads are transferred to the base concrete through the columns and anchor bolt/anchor plate anchorages. Horizontal loads are transferred to the top flange of the ring girder and then to the surrounding walls through the diaphragm plate, embedded plate and anchor bolt/anchor plate anchorages.

The safety injection tanks are supported by the main platform framing at the operating floor, elevation 62 feet (Figure 3.8-53). This platform as well as one at elevation 45 feet is supported by 20 columns located on the circumference of a 138 feet diameter circle and by embedded plates in the internal concrete. Vertical loads are taken by the columns and embedded plates and transferred to the concrete. Horizontal loads are taken by the horizontal bracing system and then transferred to the supporting concrete walls by the embedded plates.

Restraint framing is provided for all pipes, equipment, electrical trays and heating and ventilating ducts where failure of any of these items could affect the safe shutdown of the reactor. All other equipment supports, piping, cable tray and HVAC restraint loads are transferred from the structural steel to the concrete internal structures by means of attaching these supports and restraints to embedded plates.

The polar crane built up ring girder support and the crane loads are carried by the steel containment vessel cylindrical wall. A detailed description of the polar crane is given in Subsection 9.1.4.2.2.

3.8.3.2 Applicable Codes, Standards and Specifications

3.8.3.2.1 General Codes and Standards

a) Concrete Internal Structures

The concrete internal structures are designed in accordance with ACI-318-71 "Building Requirements for Reinforced Concrete".* A listing of other codes and standards is as follows:

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* ACI-349 "Proposed Code Requirements for Nuclear Safety Related Concrete Structures" is not used in St. Lucie Unit 2 design, since the concrete design predates the formal issuance date of the code. The major difference between the ACI 318-71 and ACI 349 codes is in the area of design loading. The design loads specified in ACI 349 are supplemented by RG 1.142 (R0)

1) ACI-214-65 Recommended Practice for Evaluation of Compression Test Results of Field Concrete

2) ACI-301-72 Specification for Structural Concrete for Buildings (Exceptions noted in Subsection 3.8.3.6.1)

3) ACI-315-65 Manual of Standard Practice for Detailing Reinforced Concrete.

4) ACI-347-68 Recommended Practice for Concrete Formwork

5) ACI-211-74 Recommended Practice for Selecting Proportions for Normal Weight Concrete

6) ANSI N45.2.5-1974 Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants. All listed tests were performed for the duration of construction except for ASTM C-235 which was discontinued July 1980. ASME NQA-1-1994, Subpart 2.5 was substituted for ANSI N45.2.5 as described in the FPL Quality Assurance Topical Report discussed in Section 17.2.

b) Steel Internal Structures

Design, fabrication, and erection of steel structures is in accordance with the applicable requirements of the following general codes and standards:

1) American Institute of Steel Construction (AISC) "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings - 7th Edition, February, 1969".

2) AISC “Code of Standard Practice for Steel Buildings and Bridges,” July, 1970

3) American Welding Society (AWS) Structural Welding Code D1.1-74

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which makes the design loads consistent with those presented in SRP 3.8.4 (11/75). Refer to Table 3.8-24 for partial comparison of SRP 3.8.4 (11/75) for comparison of load combinations specified therein and those used in the design of seismic Category I structures. Refer to Table 3.8-25 for the delineation of RG 1.142 (R0) which gives supplemented requirements on design procedures to the ACI 349 Code, with statements of compliance, alternate compliance and remarks on impact of deviations.

RG 1.142 (R0) also requires, that for concrete structures used to provide radiation shielding, the provisions of Section 5.1 and 10 of ANSI Standard N101.6-1972, "Concrete Radiation Shields" be followed. The provisions of those sections are followed with the clarifications shown in Table 3.8-26.

4) ASME Section III, Division 1, Subsection ND 1974 edition and Winter 1976 addenda. Subsection NF is not invoked for the steel structural supports of the Reactor Coolant System (RCS) in the St. Lucie Unit 2 design. Also the purchase specification of the steel supports of the RCS predates the formal issuance date of NF.

5) ASME Section IX, "Welding Qualification," 1974 edition and winter 1977 addenda - Used for pool liner plate

3.8.3.2.2 NRC Regulatory Guides

Recommendations continued in the regulatory guides listed below are followed:

a) Regulatory Guide 1.10 "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures," Jan. 1973 (RI)

b) Regulatory Guide 1.15 "Testing of Reinforcing Bars for Category I Concrete Structures," Dec. 1972 (RI)

c) Regulatory Guide 1.55 "Concrete Placement in Category I Structures," June 1973 (R0) with exception of the sampling requirement which is done per ACI 318-1971.

3 8.3.2.3 Specifications

a) Material Specifications

ASTM, ASME and ACI Specifications as listed in Subsection 3.8.3.6

b) Cleaning Steel

Steel Structures Painting Council SSPC - SP-6 Commercial Blast Cleaning

c) Coatings

1) Steel Structure Painting Council (SSPC) - PA-I-Shop, Field and Maintenance Painting

2) ANSI - N5.12 - Protective Coatings (Paints) for Nuclear Industry

3) ANSI - N101.2 - Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities

4) ANSI - N101.4 - Quality Assurance for Protective Coatings Applied to Nuclear Facilities

d) Purchase Specifications

The purchase specifications include the requirements for materials, design criteria, fabrication, erection, inspection and quality assurance.

3.8.3.3 Loads and Loading Combinations

3.8.3.3.1 Loads

The major loads encountered or which are postulated are listed below.

a) Normal Loads

Normal loads are those loads encountered during normal plant operation, startup and shutdown; and include the following:

1) D = Dead load consisting of the weight of the concrete and steel internal structures, any permanent equipment loads and hydrostatic loads. For equipment supports, dead load also includes static and dynamic head and fluid flow effects.

The specific weights used to establish the dead loads are as follows:

(a) Concrete: plain 138 pcf

(b) Concrete: reinforced 150 pcf

(c) Steel reinforcing: 490 pcf

(d) Structural steel: 490 pcf

2) L = Live Load is applied to the various floors and slabs to assure a structure sufficiently strong to support a random temporary load condition during reactor shutdown and to assure structural adequacy for normal loading. For equipment supports, it also includes loads due to vibration and any support movement effects. These loads are as follows:

(a) Operating floor: 1000 psf or equipment load in a designated laydown area, whichever is greater

(b) Platforms, stairs and walkways: 100 psf

(c) Pressurizer platform: 50 psf

3) F = Water load exerted by the water in the refueling cavity which is filled only during refueling operations. The specific weight of water is assumed to be 62.4 pcf.

4) T_o = Thermal Load during normal operating or shutdown conditions based on the most critical transient or steady state condition. The temperature of components of the internal structures is assumed to stabilize uniformly at the same temperature as the environment (120 F). The as-constructed temperature is assumed as 70 F.

5) R_o = Pipe or Equipment Anchor Loads (R_o) exerted upon the various structural elements in the containment internal structure by the pipe or equipment restraints for normal thermal expansion of

the various piping systems, based on the most critical transient or steady state condition.

b) Severe Environmental Loads

Severe environmental loads are those loads that could infrequently be encountered during the plant life. Postulated for this category is:

E = Operating basis earthquake (OBE) load with a horizontal ground acceleration of 0.05g (refer to Section 3.7)

c) Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but are highly improbable. The extreme environmental load postulated is:

E' = Safe shutdown earthquake (SSE) load with a horizontal ground acceleration of 0.10g (refer to Section 3.7)

d) Abnormal Loads

Abnormal loads are those loads generated by a postulated high energy pipe break accident within the containment and/or a compartment thereof. Included in this category are the following:

P_a = Pressure equivalent static load within or across a compartment. This load is generated by the postulated pipe break and includes an appropriate dynamic load factor to account for dynamic nature of the load. The dynamic load factor is determined by analysis of pressure transients inside the primary and secondary shield walls (refer to Subsection 6.2.1).

R_a = Pipe reaction caused by thermal conditions generated by the postulated break and including R_o.

T_a = Thermal loads under thermal condition generated by the postulated break and including T_o.

Y_r = Equivalent static load on the structure generated by the reaction of 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.

Y_j = Jet impingement equivalent static load on the structure generated by the postulated break, and including a dynamic load factor.

Y_m = Missile impact equivalent static load on the structure generated by or during the postulated break (e.g., pipe whipping) and including an appropriate dynamic load factor.

3.8.3.3.2 Load Combinations

The design of the internal concrete structures is based upon limiting load factors which are the ratio by which loads are multiplied for design purposes to assure that the load/deformation behavior of the structure is one of elastic, small strain behavior at the design load condition. The load factor approach is a means of making a rational evaluation of the individual factors which must be considered in assuring an adequate safety margin for the structure. This approach permits placing the greatest conservatism on those loads which most directly control the overall safety of the structure. The factored loads utilized to determine the required strength of the concrete structural element are computed using the load combination operations shown in Table 3.8-11.

Elastic working stress design methods are used in the design of the steel internal structures. The design of these structures is based on the loading combinations shown in Table 3.8-12.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 General Considerations

The primary and secondary shield walls, operating floor, refueling cavity and steam generators shield walls are the structural components of an integral monolithic unit which transfers the design loads to the containment base slab. The design loads and loading combinations are described in Subsection 3.8.3.3. These load combinations are developed per the NRC guidance provided in reference 11.

Conventional methods involving simplifying assumptions such as beam theory as well as plate and shell theories with various degrees of approximation, are utilized in the analysis of the internal structures. In general, the structures are proportioned to maintain elastic behavior under the various load combinations. The upper limit of elastic behavior is considered to be the yield strength of the effective load-carrying structural material. The yield strength for steel is considered to be the guaranteed minimum in appropriate ASTM specifications. The yield strength for reinforced concrete structures is considered to be the strength capacity as calculated from the equations given in the ACI 318-71 Code. The reinforced concrete structures are designed for ductile behavior, that is with reinforcing steel stresses controlling. Under impactive and impulsive loadings elasto-plastic behavior (or localized plastic deformation) of the structural components is considered acceptable provided that the overall integrity of the structure is not impaired.

Structural steel is designed in accordance with the elastic working stress methods in Part 1 of AISC, 7th Edition, February 12, 1969. The design, loads and loading combinations are described in Subsection 3.8.3.3. Increased allowable stresses are used in loading combinations 3 to 8 (see Table 3.8-12). The steel structures are designed to withstand the design loads without loss of function and with the corresponding deflections and deformations within acceptable limits so that the functions of the steel containment and the engineered safety feature systems are not impaired.

All safety-related component supporting structures are designated seismic Category I. Load combinations and allowable stresses are shown in Table 3.8-12. The margin of safety for those structures is inherent in the design equations in the AISC Specifications.

For linear and plate and shell type component supports subjected to the accident (faulted) load condition, the design stresses are limited to ninety percent of the critical buckling stress as applicable. (Axial compressive stresses in supports of ASME Section III, Class 1, 2 and 3 components do not exceed 67 percent of the critical buckling stress.) For design of support bolts and bolted connections, refer to the above paragraph.

For linear and plate and shell type component supports subjected to normal operation, loads are as specified in Section 1.5.1.3 of the AISC Code. This section identifies a minimum factor of safety of 1.67 which is in agreement with Appendix XVII of the ASME Code, (Article XVII-2110b).

3.8.3.4.2 Computer Programs Utilized for Structural and Seismic Analyses

The following computer programs have been used for the static and dynamic analyses of the seismic Category I structures:

EASE

EASE is a public domain computer program developed by Engineering/Analysis Corporation, which applies the finite element method for the static structural analyses of linear three-dimensional systems. The program uses beam, membrane and plate elements.

NASTRAN

NASTRAN is a public domain computer program developed by Computer Science Corporation and Mac Neal Schwendler Company for NASA. The program is used for the static and dynamic analysis of large complex structures.

ANSYS* EC283094

ANSYS is a public domain computer program developed by Swanson Analysis Systems Inc (SASI); it is a large scale general purpose computer program with a wide range of analytical capabilities: static and dynamic structural analyses (elastic and plastic, small and large deflections), steady state and transient heat transfer and fluid flow.

SHELLS

This program developed by the Service Bureau Corporation, Inglewood, California uses techniques of the finite difference method to determine stresses and deformations of shell structures in the form of surfaces of revolution about an axis. This computer code is in the public domain.

SOLIDS II

* This computer code does not apply to the replacement steam generators (RSGs). The corresponding RSG computer code is described in Section 2.8.8 of the RSG Report, 77-5069878-004 (Reference 13), if applicable. "Methods of Evaluation" within the scope of 10 CFR 50.59(c)(2)(viii) that are used for specific analyses for the RSG computer codes are addressed in the RSG Report. EC283094

This is a computer program in public domain developed by the Service Bureau Corporation of Englewood, California and utilizes a finite element method for the analysis of solids with an axis of symmetry.

SAMIS

This computer program developed by Philco Corporation - Western Development Laboratories employs the finite element method for the analysis of general structural problems.

DYNAMIC 2037

DYNAMIC 2037 developed by Ebasco, is very similar to another Ebasco program called FIXMAT 2037. The program uses iteration technique to determine natural periods and associated vibrational modal shapes of the structural system. The dynamic responses of the structures are obtained through multidegree-of-freedom and modal analysis methods together with either a time history analysis or response spectrum method.

Since this program is not a recognized program in public domain, a comparison of its predecessor FIXMAT 2037 with STARDYNE (April, 1972 version) and NASTRAN, both proven programs in public domain is made to demonstrate its validity and applicability.
(See Subsection 3.8.3.4.1.1, Waterford - Unit 3-FSAR, Docket No. 50-382 for a description of FIXMAT 2037).

MRI/STARDYNE

This is a computer program in public domain developed by Mechanics Research Incorporated and designed to analyze linear elastic structural models using finite element techniques. This program computes the structural deformations and stresses caused by any arbitrary thermal, static or dynamic loading.

3.8.3.4.3 Analysis and Design Procedures

The analytical techniques for the seismic analysis are described in Section 3.7. The analysis for the protection against dynamic effects associated with a postulated pipe break is presented in Section 3.6. The dynamic analysis for missile impact is described in Section 3.5.

The elements of the internal structures are analyzed statically for the loading combinations described in Subsection 3.8.3.3. The equivalent static loads resulting from the application of the seismic accelerations at various levels, obtained from the dynamic analysis, are included.

a) Primary Shield Wall

The MRI/STARDYNE computer program is used for the analysis of the primary shield wall. Both the concrete wall and the reactor vessel steel support structure are included in the model to determine their interaction under the different loading conditions. The steel girders supporting the reactor vessel transfer the horizontal loads through their embedded ends into the cavity wall. The cavity pressure resulting from LOCA is a load directly applied to the primary shield wall.

The results of the analyses indicate that the controlling loading condition is load combination (6) of Table 3.8-11 (abnormal/extreme environmental).

b) Secondary Shield Wall and Refueling Cavity Wall

The analysis of the secondary shield wall, refueling cavity and operating floor is done using NASTRAN. Since the structure is symmetrical only half of it is modeled. A fixed boundary condition is assumed at the bottom of the model, where the secondary shield wall is connected to the massive concrete fill.

In the analysis of the secondary shield wall an equivalent static load is utilized for the differential compartment pressure caused by LOCA.

c) Steam Generator Support

For the analysis of the steam generator enclosure and upper support the ANSYS program is used. Solid elements are utilized in the model. A fixed boundary is assumed at the bottom of the model.

For the lower concrete support of the steam generator conventional analysis method is used. The horizontal loads are transferred from the steel plate to the concrete through shear lugs. The vertical compression loads are transferred in bearing and the uplift loads are transferred through anchor bolts.

For the structural steel upper supports, the finite element method is used. The model consists of steel plate elements. The built-up steel assemblies at the key locations are modified in support of the Replacement Steam Generator per PC/M 05133 (Reference 12) based upon a reduction in design loads due to the application of Leak-Before-Break (LBB) to the RCS piping. LBB for the RCS piping is described in Section 3.6. The analysis of the modified structural steel upper supports is carried out using the ANSYS program. Forces that are postulated to occur during a DBA are transferred to the snubbers and to the builtup steel assemblies at the key locations. These loads in turn are transferred to the concrete walls by means of the anchor bolts. Table 3.8-13 shows a comparison of calculated and allowable stresses for selected elements of the Steam Generator upper supports.

For the lower sliding base support, the finite element method is also used. The steel plate assembly and concrete pier are modelled together as one unit; the entire structure is fixed at elevation 18 feet. The top plate of the steel assembly is allowed to expand freely in the lateral direction by providing a gap between the top steel plate and the top six or 10 inches of the concrete all around.

The analysis is carried out using the STARDYNE program. The horizontal forces are transferred through bearing between the key and stiffener plate to the bottom of the steel plate assembly and in turn to the concrete pier.

Table 3.8-14 shows a comparison of calculated and allowable stresses for selected elements of the steam generator sliding base support.

d) Reactor Vessel Support Structure

Design and analysis is based on elastic theory. The steel girder-column assembly is analyzed using the MRI/STARDYNE finite element program. Horizontal loads are transferred to the girder through the bearing plates on top of the steel girder and then transferred to the primary shield wall through the embedded ends of the steel girder. Vertical compressive loads are transferred (by bearing) to the primary shield wall and the base concrete, by the steel girder and column. Tension or uplift loads are resisted by the anchor bolt anchorages. Table 3.8-15 shows a comparison of calculated and allowable stresses for selected elements of the reactor vessel support structure.

e) Reactor Coolant Pump Supports

The analysis of the reactor coolant pump supports is done using conventional analytical techniques. The lower structures supporting the vertical compression spring supports are designed for both tension and compression. This is also true for the upper horizontal snubber and snubber steel support assembly. The appropriate design loads are transferred to the concrete walls or slabs by means of embedded plates with shear lugs and anchor bolts where required or by anchor bolt/anchor plate anchorages. Table 3.8-16 shows a comparison of calculated and allowable stresses for selected elements of the reactor coolant pump supports.

f) Reactor Coolant Cold Leg Stops

The reactor coolant piping cold leg stops are analyzed using the MRI/STARDYNE finite element program. The stops are composed of steel plates or forgings supported by a stiffened plate assembly. The plate or forging is shimmed in the field to maintain a prescribed gap. The compressive and shear loads are transferred into the surrounding concrete by bearing and anchor studs.

g) Pressurizer Support

The pressurizer support is analyzed using the MRI/STARDYNE finite element program. For a description of the pressurizer support structure, see Subsection 3.8.3.1.5. Table 3.8-17 shows a comparison of calculated and allowable stresses for selected elements of the pressurizer support.

h) Reactor Coolant Pump Stops and Wire Ropes

The reactor coolant pump stops are analyzed using the MRI/STARDYNE finite element program. The pump stop is composed of a steel forging supported by a stiffened plate assembly or a steel frame. The steel forging is machined and positioned in the field to maintain a prescribed gap. The appropriate design loads are taken by the steel forging through the supporting steel structure and into the internal concrete walls or slabs. Compressive loads are transferred by bearing while tension or uplift loads by embedded plates with anchor bolts where required or by anchor bolt/anchor plate anchorages. Table 3.8-18 shows a comparison of calculated and allowable stresses for selected elements of a typical reactor coolant pump stop.

One 1 7/8 inch diameter wire rope is provided at the top of the pump to restrain the pump motor casing in the unlikely event of a pipe break, see Figure 3.8-45. The appropriate tension design loads are taken by the wire ropes, through the supporting steel structure and into the surrounding concrete walls by means of the embedded plate/anchor bolt with anchor plate anchorages. Conventional analytical techniques are used to analyze these wire rope restraint

i) Main Steam and Feedwater Pipe Restraints

The main steam and feedwater pipe restraints are supported by plane frames or space frames. These frames are analyzed using the MRI/STARDYNE finite element program. The design loads are taken through the frames and transferred to the supporting concrete by means of embedded plates or anchor bolt/anchor plate anchorages.

j) Other Internal Structures

Cable tray restraints are designed as nonrigid structures with a minimum natural frequency within 16 Hz. HVAC restraints are also designed as nonrigid structures with a minimum natural frequency of 15 Hz. These systems are dynamically analyzed using the response spectra modal analysis technique to determine seismic accelerations.

The design and analysis of the platform framing at elevation 62 ft, and elevation 45 ft. is based on elastic theory. For a description of the design load transfers to the supporting concrete walls and slabs, see Subsection 3.8.3.1.5.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Concrete Structures

The calculated stresses for various components of the containment internal structures under the different loading combinations (as described in Subsection 3.8.3.3) do not exceed the yield strength of the materials used. Table 3.8-19 compares the required design strength and the ultimate capacity of the structural elements for the governing loading condition.

For the calculation of the strength capacity of the concrete structural components, capacity reduction factors in accordance with Section 9.2 of ACI-318-71 are used. These capacity reduction factors are provided to allow for possible understrength of the structural element due to variations in material strength, dimensions or workmanship. These factors also reflect the degree of certainty with which the various types of strength (bending, shear, torsional) can be predicted.

3.8.3.5.2 Steel Structures

The allowable stress limits for each loading combination are represented in Table 3.8-12. The calculated results for representative structural elements of the Nuclear Steam Supply System equipment support structures are shown in Tables 3.8-13 to 3.8-18 and indicate the margins of safety in their design.

3.8.3.6 Materials, Quality Control and Special Construction Techniques

Materials used for the construction of internal structures are concrete, reinforcing steel and structural steel. The material specifications testing requirements, and quality control measures delineated in this section form a part of the overall quality assurance program described in Chapter 17.

3.8.3.6.1 Concrete Construction

The requirements for concrete construction conform to the purchase specifications and to the appropriate ASTM, ACI and ANSI Standards or portions thereof as required by the purchase specifications.

Concrete Materials

a) Cement

Type II low alkali cement conforming to ASTM C150 is used. The cement contains no more than 0.60 percent by weight of the alkalies calculated as $Na_20 + 0.658K_20$.

Cement with a temperature in excess of 150° F is not used.

b) Aggregate

Both the fine and coarse aggregate conform to ASTM C33 with the exceptions that:

1) Use of blast-furnace slag for coarse aggregate is not permitted.

2) Grading requirements are developed for the local aggregate.

c) Admixtures

Air entraining admixture and chemical admixtures conform to ASTM C260 and ASTM C494 respectively.

1) Air Entraining Admixture

The air entraining admixture is used in concrete within the dosage limits recommended by the manufacturer. Total air content, including that due to use of chemical admixture generally ranges between 3.5 and 6.5 percent by volume.

2) Chemical Admixtures

Two types of chemical admixtures, i.e., water reducing (type A), and water reducing set retarding (type D) are used in accordance with the concrete specification. Type D admixture is used whenever the dry bulb temperature equals or exceeds 85 F or the temperature of the fresh concrete equals or exceeds 75 F.

d) Water

Water used for mixing or curing of concrete is free from any injurious amounts of acids, alkalines, salts, oil, sediment or organic matter. The chlorides and dissolved solids are within the following limits:

Chloride ion content (mixing water)	$\leq$ 125 ppm
Total soluble chloride ion content	$\leq$ 250 ppm
(concrete mixture)	
Total dissolved solids	$\leq$ 1000 ppm

e) Concrete

The concrete ingredients consist of fine and coarse aggregates, Type II Portland cement, water and admixtures. The concrete mixes are designed and tested to produce concrete of required consistency, workability and a minimum strength in excess of the required design strength by the values specified in ACI 318-71.

Concrete with a design strength of 4000 psi and 5000 psi (as detailed on the design drawings) is used in the internal concrete structures. Slumps for various types of placements are four in. or less. The concrete is produced from an automatic batch plant of 150 cu ft per hour capacity installed in the vicinity of the work.

Quality Control for Concrete Construction

The quality control requirements for concrete construction are in accordance with ANSI N45.2.5 (1974), Regulatory Guide 1.55 (RO), and ACI-318-1971. ASME NQA-1-1994, Subpart 2.5 was substituted for ANSI N45.2.5 as described in the FPL Quality Assurance Topical Report discussed in Section 17.2. Quality control procedures are established and implemented at the site for installation, inspection and testing of concrete construction to verify conformance to specified requirements. These are discussed in the following paragraphs.

a) Cement

Sampling and testing of cement is performed on each 1200 tons delivered to the site in accordance with ASTM C183-1971 and ASTM C150-1972. A certified mill test report, attesting to the conformance of cement to the specification by the manufacturer's chemist is furnished with each shipment and reviewed. The testing program is supplemented by inspection at the batch plant and measurement of cement temperature.

b) Aggregate

Sampling and testing of aggregates are performed during production in accordance with ASTM C33-1971a and the test frequency specified below:

Gradation (ASTM C136-1971)	-	daily
Organic impurities (ASTM C40-1966)	-	weekly
Material finer than No. 200 sieve (ASTM C117-169)	-	daily
Clay lumps and friable particles (ASTM C142-1971)	-	monthly
Specific gravity and absorption (ASTM C127-1968 or each C128-1968) before	-	once for stockpile
production		
Percentage voids (ASTM C30-1970) each	-	once for
stockpile before		
production		
Lightweight pieces (ASTM C123-1969)	-	monthly
*Soft fragments (ASTM C235-1968)	-	monthly
Los Angeles abrasion (ASTM C131-1969 or C535-1969)	-	6 months
Flat and elongated particles (CRD C119-1953)	-	6 months
Potential reactivity (ASTM C289-1971)	-	6 months
Soundness (ASTM C88-1971a)	-	6 months
Moisture content (ASTM C566-1967)	-	daily

The above program is supplemented by visual inspection of aggregate stockpiles, daily during concrete production or weekly during off-production periods, to verify conformance to the specifications.

c) Admixtures

Inspection is performed for each batch of admixture delivered to the site to assure that certified infrared spectrophotometry analyses reports are

* Test discontinued July 1980

supplied by the manufacturer along with certification by a qualified chemist, attesting to the conformance of the admixture supplied to admixture used in trial mixes. Inspection is also performed to verify the type of admixture, Type A or Type D, used in concrete in accordance with the ambient temperature limitations.

d) Water

Sampling and testing of water is performed every month for chloride content and every six months for dissolved solids. In addition, tests for effect on compressive strength setting time and soundness are also performed to verify compliance with American Association of State Highway Officials (AASHO) T-26 every month.

e) Concrete

Inspections are performed to verify that the concrete batch plant and the transporting equipment comply in all respects, including storage provisions and precision of measurements, with ASTM C94-1972 and National Ready Mixed Concrete Association (NRMCA) Certification of Ready Mixed Concrete Production Facilities and Measuring the Uniformity of Concrete Produced in Truck Mixer (1972).

Daily inspections are performed of all truck mixers and agitator units to verify: satisfactory interior condition, no appreciable accumulation of hardened concrete, blades free of excessive wear, charging and discharging chute in good condition, equipped with counter in working condition to indicate the number of total revolutions including mixing revolutions of drum.

Six month inspections are performed on truck mixers for the following:

1) to check blade wear in the drum

2) to check that each truck is equipped with a plate showing manufacturer, mixer capacity and mixing speed, and

3) to verify uniformity of concrete produced.

During concrete production, periodic inspections were also made for the following:

1) to check scale remote readout for zero reading prior to the weighing of a batch

2) to check batch selected to assure that proper mix is being batched

3) to check moisture meter reading to verify moisture and aggregate weight compensation,

4) to witness the weighing of cement, sand, coarse aggregate, water and measurement of admixtures to assure that weights are within the specified limits,

5) to check recording ticket for confirmation of batch weights, and

6) to verify that the truck ticket has batch time stamped on it.

Inspections are also made prior to, during and after placement of concrete to ensure conformance to concrete specification and ANSI N45.2.5-74. ASME NQA-1-1994, Subpart 2.5 was substituted for ANSI N45.2.5 as described in the FPL Quality Assurance Topical Report discussed in Section 17.2.

In-process tests are performed on concrete samples at the truck mixer discharge, in accordance with ASTM C172-1971, to control the consistency and other structural properties of concrete. For pumped concrete, sampling is initially performed at both the truck discharge and pump line discharge to first establish a correlation in concrete properties (slump, air content, temperature and strength); thereafter sampling and testing is permitted at the truck discharge. No concrete is placed into the forms prior to testing and acceptance of concrete. The test requirements, test method and test frequency for concrete conform to the following:

Requirement	Test Method	Total Frequency
Slump	ASTM C143-1971	First batch produced and every 50 cu yd
Air Content	ASTM C173-1971 or C231-1971T	First batch produced and every 50 cu yd
Temperature		First batch produced and every 50 cu yd
Unit Weight	ASTM C138-1971T	First batch produced and every 10 cu yd
Compressive Strength	ASTM C39-1971	Each 100 cu yd (1 cylinder @ 7 day, 2 @ 28 days and 1 spare)

Seven day strength results are used to establish a correlation between seven day and 28 day strengths. After a correlation is established, the seven-day tests are used as an indicator of the compressive strength which are expected at 28 days. If seven day tests indicate low strengths, corrective measures are taken immediately without waiting for the results of the 28 day tests.

The strength test is statistically evaluated in accordance with ACI-301 with a coefficient of variation for the tests not exceeding 10 percent (ACI-214-1965). The actual coefficient of variation for concrete is within 10 percent.

The fabrication and placement of concrete in the containment internal structure does not utilize any special construction techniques.

3.8.3.6.2 Reinforcing Steel

ASTM A 615-68, “Standard Specification for Deformed and Plain Billet Steel Bars for Concrete Reinforcement", is used to purchase reinforcing steel.

All reinforcing bars used are new billet steel in accordance with ASTM A615-68 Grade 60 (60,000 psi minimum yield strength).

The quality control requirements for reinforcing steel are in conformance with ANSI N45.2.5-74. ASME NQA-1-1994, Subpart 2.5 was substituted for ANSI N45.2.5 as described in the FPL Quality Assurance Topical Report discussed in Section 17.2.

Certified mill test reports are furnished by the reinforcing steel supplier for each heat of steel, showing that the reinforcing steel meets the specified composition, strength and ductility requirements. The testing frequency is in accordance with ANSI N45.2.5-74 and Regulatory Guide 1.15 (R1). In addition, tests are performed on representative specimens to supplement the standard mill test.

Reinforcing steel is shipped to the site in bundles bearing a tag identifying size, grade and code number keyed to the delivery ticket and a second tag identifying heat number. This information is verified by certified mill test report, which accompanies each shipment of reinforcing steel.

Visual inspection of fabricated reinforcement is performed to ascertain dimensional conformance with specifications and drawings. Visual inspections are made of in-place reinforcement to assure dimensional and locational conformance with drawings and specifications.

Placing and splicing of bars meet the requirements of ACI 318-71 Code. Welding of reinforcing bars is not performed.

The fabrication and placement of the reinforcement do not utilize any special construction techniques.

3.8.3.6.3 Structural and Miscellaneous Steel

Structural and miscellaneous steel in general conform to the following ASTM and ASME specifications wherein the chemical and mechanical properties are specified. Certified mill test reports or certificates of compliance are supplied by the structural steel vendors to verify these properties. Erection tolerances of the structural steel are specified in the AISC code.

a) ASTM

Carbon Steel	
Rolled Shapes, Bars and Plates	A36, A441, A533 Class 2 Grade B, A572 Gr 50, A992
High Strength Bolts	A325, A490
Other Bolts	A307

b) ASME

Stainless Steel	
Sheet and Plate	SA240 Type 304
Rolled Shapes	SA276 Type 304, SA479 Type 304
Carbon Steel Rolled Shapes,	SA36, SA 572 Gr 50
Bars and Plates	

Other types of steel are used in small quantities in the internal structures as required. A summary of the materials used for the NSSS supporting steel is given in Table 3.8-20. Mill test reports are obtained for the material used in the supporting structures.

3.8.3.7 Testing and Inservice Inspection Requirements

Following construction the complete refueling cavity liner is hydrostatically tested. Leaks, if any are repaired, and rewelded and retested. There are no testing and inservice inspection requirements for the other internal structures.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

3.8.4.1 Description of the Structures

The general description of the seismic Category I structures other than the containment is provided herein, with plan and section views of general arrangement drawings provided in Section 1.2.

3.8.4.1.1 Shield Building

The Shield Building (also termed the Reactor Building) is a reinforced concrete right cylinder structure with a shallow dome roof surrounding the containment vessel. It has an outside diameter of 154 ft and a height of 230.5 ft measured from the top of the base slab to the top of the dome. The thickness of the wall is three ft except at the base (below elevation 1.0 ft) where it is nine ft. The thickness of the dome is 2.5 ft. A nominal four ft annular space is provided between the interior face of the concrete shield building and the containment vessel. The volume contained within the annulus is 543,000 cu ft. This space provides the means of access for periodic visual inspection of the containment vessel as well as collecting and diluting any leakage from the containment vessel. The Shield Building is a free standing structure, with concrete fill placed in the bottom of the structure to support the steel containment. Shield Building plan - masonry and reinforcement drawings are shown on Figures 3.8-54 through 57. See Subsection 3.8.5 for the Shield Building foundation masonry and reinforcements.

The Shield Building functions:

a) as a biological shield during normal operation and after any accident within the steel containment, including LOCA,

b) as a low leakage structure,

c) as a shield for the primary steel containment for adverse atmospheric conditions due to tornados and hurricane winds, external missiles and flooding.

The Shield Building is designed to seismic Category I requirements considering the loading combinations specified in Subsection 3.8.4.3.

3.8.4.1.2 Reactor Auxiliary Building

The Reactor Auxiliary Building (RAB) is a three-story reinforced concrete structure located immediately south of the Reactor Building. The interior floors are a beam and girder construction supported by columns. The building occupies an area approximately 237 ft by 113 ft and extends from the top of mat at elevation -14.5 ft and -5 ft to roof levels varying from elevation 43 to 82 ft (see Figures 3.8-58 and 59). The RAB houses the waste treatment facilities, Engineered Safety Features, switchgear, laboratories, offices and control room. It further provides protection to the cable and piping penetration areas of the Reactor Building. The building exterior walls, floors and interior partitions are designed to provide plant personnel with the necessary biological radiation shielding and protect the equipment inside from the effects of adverse atmospheric conditions, including tornado and hurricane winds, external missiles, temperature, external flooding, corrosive environment and pipe whip and jet impingement design conditions.

The RAB is designed to seismic Category I requirements and the load combinations considered are specified in Subsection 3.8.4.3.

3.8.4.1.3 Fuel Handling Building and Cask Handling Crane Support Structure

The Fuel Handling Building (FHB) is a reinforced concrete structure located immediately east of the Reactor Building. It occupies an area approximately 45 ft by 126 ft and it extends from the top of the mat at elevations 15 ft and 17 ft to roof level at elevation 96.5 ft. The FHB houses a spent fuel pool, a spent fuel cask pit, a refueling canal, spent fuel pool pumps, purification pumps, heat exchanger, filter and heating and ventilating equipment. In addition the building provides space for the storage of new fuel.

An outdoor spent fuel cask handling crane is capable of hoisting a multi-element spent fuel cask through an opening in the northeast corner of the FHB roof that is normally covered by an L-shaped door. The crane's external runways and steel frame structure are partially supported by the FHB roof and east exterior wall, as well as by columns on concrete foundations at the grade elevation.

The FHB exterior walls, the floors and the interior partitions are designed to provide plant personnel with the necessary biological radiation shielding and to protect the equipment inside from the effects of adverse atmospheric conditions such as tornado and hurricane winds, external missiles and flooding. The FHB and the spent fuel cask handling crane support structure are designed to seismic Category I requirements and the loading conditions are specified in Subsection 3.8.4.3

The spent fuel pool, the spent fuel cask pit, and the refueling canal are cast-in-place, stainless steel lined, reinforced concrete structures that provide space for storage of spent fuel, the spent fuel cask, and miscellaneous equipment.

3.8.4.1.4 Diesel Generator Building

The Diesel Generator Building is a reinforced concrete structure housing duplicate diesel generator units separated by a reinforced concrete wall. The building consists of a base mat, exterior walls, one interior wall separating the units and a concrete roof. The diesel generator sets are supported on pedestals on the concrete mat. The exterior walls and the roof are designed to protect the equipment from the effects of adverse atmospheric conditions such as tornado and hurricane winds, external missiles and flooding. The Diesel Generator Building is designed to seismic Category 1 requirements and the loading conditions are specified in Subsection 3.8.4.3 (see Figure 3.8-60).

3.8.4.1.5 Intake Structure

The intake structure is a reinforced concrete structure housing the circulating water pumps and the intake cooling water pumps. The structure consists of four bays on a common base mat. The top of the mat is at elevation -31.0 ft and the intake deck is at elevation 16.5 ft. A valve pit is located to the north of the intake, cantilevered from the Intake Structure wall at elevation 5 ft. The intake cooling water pumps are protected from tornado missiles by an enclosure consisting of reinforced concrete walls and structural steel roof extending above the deck to elevation 36.5 ft. A similar structure for missile protection is provided over a portion of the valve pit. The intake structure is serviced by a bridge crane.

To the north and south of the intake structure, two retaining walls provide support for the fill and serve as foundations for the bridge crane. The intake structure and the north and south retaining walls are designed to seismic Category I requirements, and the loading conditions are specified in Subsection 3.8.4.3.

3.8.4.1.6 Component Cooling Water Building

The component cooling water pumps and heat exchangers are housed in a rectangular missile protection structure. The structure consists of a base mat, exterior walls and a concrete roof slab, supported on the exterior walls and on reinforced concrete columns.

The pumps and the heat exchangers are supported on concrete pedestals. The structure is designed to seismic Category I requirements and the loading conditions are specified in Subsection 3.8.4.3.

3.8.4.1.7 Condensate Storage Tank Building

The condensate storage tank is protected from tornado missiles by a cylindrical reinforced concrete structure with a shallow dome roof. The thickness of the wall and of the roof is two feet. The structure is supported on a three foot thick reinforced concrete mat. The structure is designed to seismic Category I requirements. The loading conditions are specified in Subsection 3.8.4.3.

3.8.4.1.8 Diesel Oil Storage Tanks

The missile protection structure housing the diesel oil storage tanks is a rectangular reinforced concrete structure supported on a three feet thick mat, with two foot thick walls and a two foot thick roof slab. An interior wall separates the two tanks. The structure is designed to seismic Category I requirements. The loading conditions are specified in Subsection 3.8.4.3.

3.8.4.1.9 Main Steam Trestles

The main steam trestles are two braced steel tower structures. One tower is located immediately west of the Reactor Building and the other immediately northwest (see Figures 3.8-61 through 66). Each tower occupies an area approximately 31 feet by 51 feet and extends from the top of pier elevation 19.5 ft to elevation 62 ft. Each tower is structurally independent from the other except for a small connecting walkway at elevation 42.83 feet and the Auxiliary Feedwater System crossover piping. The crossover piping is enclosed by a one inch steel plate which provides protection against all postulated missiles. The two trestle compartments are two totally enclosed structures which are physically separated from each other. Each trestle compartment houses the following equipment:

a) One main steam line

b) One main steam isolation valve (MSIV)

c) Eight main steam safety valves

d) One main feedwater line

e) Two main feedwater isolation valves (MFIVs)

f) Two atmospheric dump valves (ADVs)

g Two motor driven auxiliary feedwater pumps or one steam driven auxiliary feedwater pump (with associated piping and valves).

The main steam trestles provide support and missile protection for the main steam and feedwater piping and the required restraints between the Reactor and Turbine Buildings. The main steam trestles also provide support for the missile protection enclosing the auxiliary feedwater pumps located beneath both tower structures. The loading conditions are specified in Subsection 3.8.4.3. Three sides of the main steam trestle are completely enclosed with a one inch steel plate along the entire vertical run with a nine inch opening left on the base perimeter to provide for natural ventilation. The forth side utilizes the containment structure as a missile barrier and is recessed several feet from the containment in order to provide adequate ventilation. The roof of the trestle structure utilizes a steel grating (several inches thick) for missile protection purposes. The openings in this grating have been designed to inhibit the smallest missile provided in Section 3.5 and to provide sufficient main steam mass and energy blowdown area to accommodate a main steam line break outside the containment.

3.8.4.1.10 Steam Generator Blowdown Treatment Facility

The Steam Generator Blowdown Treatment Facility (SGBTF) is located east of the St. Lucie Units 1 and 2. The building houses equipment for the blowdown treatment. The SGBTF building is capable of withstanding the OBE loads or 120 mph wind.

The SGBTF is a common facility which was licensed on the St. Lucie Unit 1 Docket (Docket No. 50-335) and is shared between St. Lucie Units 1 and 2. See Subsection 10.4.8 for a description of the SGBTF. The seismic analysis for the SGBS is outlined in Subsection 3.7.2.1.1.

SGBTF Building is a reinforced concrete building with structural steel supported roofs and structural steel bays on the north and west sides. The building occupies an area 122 feet x 80 feet and has a height of 60 feet above grade except for the bay at the west end of the building which has a height of 18 feet above grade.

The building is constructed with cast-in-place reinforced concrete exterior bearing walls on the south and east sides and structural steel columns and siding on the north and west sides. The building is founded on a reinforced concrete mat. The top elevation of the mat is 13 feet MLW, except under the equipment drain tank and spent resin storage tank where the top of mat is elevation 7.50 ft MLW.

Three steel monitor storage tanks are located outdoors to the south of the building. Each tank is supported by a concrete ring beam. The monitor storage tanks occupy an area approximately 120 ft x 60 ft. The monitor storage tanks have not been analyzed for OBE conditions (refer to Figures 3.8-67 and 69).

3.8.4.1.11 Concrete Masonry Walls

Concrete masonry walls as used in seismic Category I structures have been classified as either safety-related or non-safety-related.

At completion of construction a program of field inspections and re-evaluation of design adequacy of concrete masonry walls was implemented as follows:

a) Field surveys of all masonry walls were performed to identify those masonry walls which are in proximity to or have attachments from safety-related piping or equipment such that wall failure could affect a safety-related system.

b) Masonry walls identified by the field inspection program as safety-related were re-evaluated to demonstrate their capacity to withstand the postulated design loads. All data used in the analysis was verified by a field inspection program.

The masonry wall evaluation program is described in Appendix 3.8A.

3.8.4.1.12 Turbine Building

Although the Turbine Building is a non-seismic building, the equivalent static "g" loads have been taken into account in the stress analysis of the Turbine Building framing.

The seismic required design "g" values for the Turbine Building structure evaluation were obtained from the dynamic seismic response analysis using a simplified model to represent the dynamic behavior of the Turbine Building structure.

3.8.4.2 Applicable Codes, Standards and Specifications

The applicable codes, standards and specifications are given in Subsection 3.8.3.2.

3.8.4.3 Loads and Loading Combinations

3.8.4.3.1 Loads

3.8.4.3.1.1 Loads (other than for Cask Crane Support Structure)

The major loads to be encountered or to be postulated are listed below;

a) Normal Loads

Normal loads are those loads to be encountered during normal plant operation and shutdown. They include the following:

1) D = Dead load consisting of dead weight of the concrete structure, structural steel, permanent equipment loads, hydrostatic loads.

The specific weights used to establish the dead loads are given in Subsection 3.8.3.3.1. For the Fuel Handling Building the water level in the spent fuel pool is assumed to be at operating floor level, elevation 60 ft.

2) L = Live load, set on the various floors and slabs and includes any movable equipment loads to assure a structure sufficiently strong under normal operation to support a random temporary load condition. See Table 3.8-21 for a list of live loads.

3) B = Buoyancy, the uplift load exerted by groundwater under normal environmental condition, and B' = Uplift load under severe environmental condition (hurricane).

4) T_o = Thermal load, induced by thermal gradient existing across the walls between the building interior and the ambient external environment during normal operating or shutdown conditions (including the most critical transient and steady state condition). Both winter and summer operating conditions are considered.

5) H = lateral earth loads due to soil pressure under normal conditions. These loads are based on the following soil properties:

(a) Dry unit weight, a = 105 pcf

(b) Moist unit weight, m = 115 pcf

(c) Saturated unit weight, s = 125 pcf

(d) Submerged unit weight, sn = 60 pcf

Angle of internal friction = 40°

The lateral soil pressure during both normal conditions and postulated earthquake conditions is calculated as described in Subsection 2.5.4.10.

6) R_o = Pipe load, the load exerted upon the various structural elements by the pipe reaction during normal operating or shutdown conditions, including the most critical transient and steady state condition.

b) Severe Environmental Loads

Severe environmental loads are those loads that could infrequently be encountered during the plant life. Included in this category are:

E = Loads generated by the operating basis earthquake (OBE) having a horizontal acceleration of 0.05g, (refer to Section 3.7).

W = Loads generated by the design wind specified for the plant. (Refer to Subsection 3.3.2).

H′ = Lateral earth loads due to soil pressure under normal and earthquake conditions. Soil properties can be found in Subsection 3.8.4.3.1.1.a)5).

c) Extreme Environmental Loads

Extreme environmental loads are those loads which are credible but are highly improbable. They include:

E′ = Loads generated by the safe shutdown earthquake (SSE) having a horizontal ground acceleration of 0.1 g (refer to Section 3.7).

W_t = Loads generated by the plant specific design basis tornado. This includes loads resulting from a tornado funnel with a horizontal rotational velocity of 300 mph and a horizontal translational velocity of 60 mph, from tornado-created differential pressures, and from missiles, combined in a manner to produce most severe loading condition for each case.

H′ = Lateral earth loads due to soil pressure under normal and earthquake conditions. Soil properties can be found in Subsection 3.8.4.3.1.1.a)5).

The tornado design velocity is 300 mph for the Shield Building and Reactor Auxiliary Building, and 360 mph for other buildings and structures needed for missile protection. A lower design velocity is selected for the Shield and Reactor Auxiliary Buildings, because their width is significantly larger than the narrow band width over which a combined maximum velocity of 360 mph is distributed. Hence a uniform wind of 300 mph is selected (see Subsection 3.3.2).

The tornado created differential pressure is three psi in three seconds for all these buildings with the exception of the Diesel Generator Building, for which a reduced value of 2.25 psi, is used based on the large vent area for this building. The spectrum of tornado missiles used in design is given in Section 3.5.

d) Abnormal Loads

Abnormal loads are those generated by a postulated high-energy pipe break accident within a building and/or compartment. Only equipment within the Reactor Building or the Reactor Auxiliary Building experience abnormal compartment loads.

Following is a list of abnormal loads considered:

P_a = Pressure load, the pressure equivalent static load within or across a compartment and/or building, generated by the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.

T_a = Thermal load under thermal conditions generated by the postulated break and including the normal thermal load T_o.

R_a = Pipe reactions under thermal conditions generated by the postulated break and including the normal pipe anchor load, R_o.

Y_r = Pipe loads, the equivalent static loads on the structure generated by the reaction on the broken high energy pipe during the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.

Y_j = Jet impingement equivalent static loads on the structure generated by the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.

Y_m = Missile load, the missile impact equivalent static load on the structures generated by or during the postulated break, like pipe whipping, and including an appropriate dynamic factor to account for the dynamic nature of the load.

3.8.4.3.1.2 Cask Crane Support Structure

The cask handling crane steel support structure, including the runway girders, is designed and qualified for the following loads:

D = Dead load of the crane and of the support structure

L = Crane lifted load of 150 tons

I = Impact load resulting from the operation of the crane

W_O = Operating wind load resulting from a wind speed of 50 mph

W_H = Hurricane wind load resulting from a wind speed of 120 mph

W_T = Tornado wind load resulting from a wind speed of 360 mph (300 mph rotational speed plus 60 mph translational speed)

E = OBE seismic load

E' = DBE seismic load

The impact loads resulting from operation of the crane that are used in the design are enveloped by the impact load calculation methods given in the AISC Manual of Steel, Ninth Edition; CMAA Specification 70-2000, "Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes"; and ASME NOG-1-1998, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)", including May 3, 2000 addenda.

The wind loads are determined using the methods given is ASCE Paper No. 3269, "Wind Forces on Structures", 1961.

For seismic analysis, a conventional lumped mass three dimensional model is used. The mathematical model represents the coupled FHB, the crane steel support structure, and the crane, including the bridge girders, trolley, end trucks, and end ties. The boundary conditions at the crane wheels are in accordance with ASME NOG-1-998, Table NOG-4154.3-1. The OBE and DBE loads are determined in accordance with the response spectrum method of analysis given in Section 3.7.2. The horizontal and vertical seismic input motions used in the analysis are based on the response spectra described in Section 3.7.1.1. The damping values used in the analysis are consistent with Section 3.7.1.3. Composite damping is conservatively calculated for structural elements having different damping based on the method described in Section 3.7.2.15. The soil spring constants for the foundations are based on the formulae given in Section 3.7.2.4. The modal responses are combined using the root-mean-square method specified in Section 3.7.2.1.3. The co-directional responses (axial force, shear and moment) caused by each of the three components of earthquake motion are combined using the square root sum of the squares (SRSS) method as described in Section 3.7.2.6.

3.8.4.3.2 Load Combinations

3.8.4.3.2.1 Concrete Structures

The design of the Shield Building and other seismic Category I structures outside the containment is based upon limiting load factors (see Subsection 3.8.3.3.2) to assure the structural integrity and adequate margin of safety of these structures. The load combinations for the required strength (U) are as follows (see definition of loads in Subsection 3.8.4.3.1):

a) Normal Operating

$U = 1.4 (B + D) + 1.3 (R_o + T_o) + 1.7 (L + H)$

$U = 1.4 (B + D) + 1.3 (R_o + T_o)$

b) Severe Environmental (OBE)

$U = 1.4 (B + D) + 1.3 (R_o + T_o) + 1.7 (L + H) + 1.9E$

$U = 1.4 (B + D) + 1.3 (R_o + T_o) + 1.9E$

$U = 1.2 (B + D) + 1.9E$

c) Severe Environmental (Hurricane)

$U = 1.4 (B' + D) + 1.3 (R_o + T_o) + 1.7 (L + H + W)$

$U = 1.4 (B' + D) + 1.3 (R_o + T_o) + 1.7W$

$U = 1.2 (B' + D) + 1.7W$

d) Extreme Environmental (SSE)

$U = 1.0 (R + D + R_o + T_o) + 1.0 (H' + L + E')$

$U = 1.0 (B + D + R_o + T_o) + 1.0E'$

e) Extreme Environmental (Tornado)

$U = 1.0\ (B + D + R_o + T_o) + 1.0\ (H + L + W_t)$

$U = 1.0\ (B + D + R_o + T_o) + 1.0W_t$

f) Abnormal (Pipe break accident)

$U = 1.0\ (B + D) + 1.0\ (H + L + R_a + T_a) + 1.5\ P_a$

$U = 1.0\ (B + D) + 1.0\ (R_a + T_a) + 1.5\ P$

g) Abnormal/Severe Environmental (OBE)

$U = 1.0\ (B + D) + 1.0\ (H' + L + R_a + T_a) + 1.0\ (Y_r + Y_j + Y_m\ ) + 1.25\ (P_a + E)$

$U = 1.0\ (B + D) + 1.0\ (R_a + T_a) + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ (P_a + E)$

h) Abnormal/Extreme Environmental (SSE)

$U = 1.0\ (B + D) + 1.0\ (H' + L + R_a + T_a) + 1.0\ (Y_r + Y_j + Y_m) + 1.0(P_a + E')$

$U = 1.0\ (B + D) + 1.0\ (R_a + T_a) + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ (P_{a\ +} E')$

3.8.4.3.2.2 Steel Structures

Elastic working stress design methods are used in the design of the steel structures. The design of these structures is based on the following loading combinations:

a) Service Load Conditions

1) $D + L$

2) $D + L + E$

3) $D + L + W$

4) $D + L + T_o + R_o$

5) $D + L + T_o + R_o + E$

6) $D + L + T_o + R_o + W$

b) Factored Load Conditions

1) $D + L + T_o + R_o + E'$

2) $D + L + T_o + R_o + W_t$

3) $D + L + T_a + R_a + P_a$

4) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E$

5) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E$

In loading combinations b3, b4 and b5, R_a and T_a are applied statically without consideration of a dynamic load factor.

3.8.4.3.2.3 Cask Handling Crane Support Structure

The following load combinations and acceptance criteria are used in the design of the FHB cask handling crane steel support structure:

		Load Combination	Acceptance Criteria
a)	Normal Operation	$D + L + I + W_O$	S
b)	Design Hurricane	$D + W_H$	S
	(The crane is assumed to be in its parked position with the storage locks set.)		
c)	Operating Basis Earthquake (OBE)	$D + L + E + W_O$	S
	(Seismic-induced pendulum effects are considered in the design.)		
d)	Design Basis Earthquake (DBE)	$D + L + E' + W_O$	The lesser of 1.6S, or 0.90 times yield stress,
	(Seismic-induced pendulum effects are considered in the design.)		or 0.90 times critical buckling stress
e)	Design Tornado	$D + W_T$	The lesser of 1.6S, or 0.9 times yield stress,
	(The crane is assumed to be in its parked position with the storage locks set.)		or 0.90 times critical buckling stress

Acceptance criterion "S" above is the required section strength based on the elastic design methods and the normal allowable stresses defined in the AISC Manual of Steel Construction, Seventh Editon.

The FHB cask handling crane column foundations at the grade elevation are designed for the loads and load combinations given in Section 3.8.4.3, subject to the following clarifications:

D = Dead load of the concrete structure, the crane, and the steel support structure

L = Crane lifted load of 150 tons

W_H = Hurricane wind load resulting from a wind speed fo 120 mph

W_T = Tornado wind load resulting from a wind speed of 360 mph (300 mph rotational speed plus 60 mph translational speed)

I = Impact load resulting from the operation of the crane

Thermal loads (T_o and T_a), pipe loads (R_o, R_a, and Y_r), jet impingement loads (Y_j), and missile loads (Y_m) are not applicable to the design of the cask handling crane superstructure and column foundations. The impact load (I) is added to the normal operation combination with a factor of 1.7, as follows:

$$U = 1.4 (B + D) + 1.7 (L + H + I)$$

The column foundations are designed in accordance with the ultimate strength design (USD) methods of ACI-318-71 as described in Section 3.8.5.

3.8.4.4 Design and Analysis Procedure

3.8.4.4.1 General Considerations

The seismic Category I buildings are designed for the load combinations indicated in Subsection 3.8.4.3 in accordance with the strength design provisions of ACI 318-71. Fixed boundary conditions are assumed for the connection of the Shield Building cylinder wall and for the columns and structural walls of the other buildings to the foundation mats. The seismic and impactive analyses are described in Sections 3.7 and 3.5. The equivalent static loads derived from these dynamic analyses are utilized in the static analysis of the buildings.

a) Shield Building

The cylinder wall and the dome are designed to resist the membrane stresses imposed by the different load combinations (Subsection 3.8.4.3) as well as the local stresses caused by discontinuities.

The SHELLS computer program (see Subsection 3.8.3.4) is used for the analysis of the Shield Building. The structural model used in the analysis is divided in four regions to adequately represent the geometrical configuration of the building. Each region is divided into equal finite increments referred to as stations.

A separate model is used for the areas of the shell with major penetrations (larger than 7.5 ft in diameter). For opening less than 7.5 ft in diameter the stress concentration is considered in design by using the equations available in the technical literature for circular openings in plates subjected to tension and bending.

b) The Reactor Auxiliary Building, Fuel Handling Building, Intake Structure and missile protection structures are analyzed by conventional methods.

The structures are designed to maintain elastic behavior under the various load combinations. The upper limit of elastic behavior is considered to be yield strength of the structural material, for steel, the guaranteed minimum in the ASTM specification. For reinforced concrete the strength capacity is calculated from the equations in the ACI 318-71 Code.

The reinforced concrete structures are designed for ductile behavior, that is with reinforcing steel stresses controlling. Under impactive and impulsive loading, elasto-plastic behavior (or localized plastic deformation) of the structural components is considered acceptable provided that the overall integrity of the structure is not impaired.

Structural steel is designed in accordance with the elastic working stress methods in Part 1 of AISC, 7th Edition, February 1969. The design loads and loading combinations are described in Subsection 3 8.4.3. Increased allowable stresses are used in loading combinations a4 to a6 and b1 to b5. The steel structures are designed to withstand the design loads without loss of function and with the corresponding deflections and deformations within acceptable limits.

The computer programs used in the static and dynamic analyses of seismic Category I steel structures are MRI/STARDYNE and ANSYS. See Subsection 3.8.3.4 for a description of the computer programs.

3.8.4.4.2 Procedures Used in Design and Analysis

Analytical techniques for the seismic dynamic analysis are described in Section 3.7. Analytical techniques for the missile protection are described in Section 3.5. Analytical techniques for the protection against dynamic effects associated with a postulated pipe break are described in Section 3.6.

The steel structures are analyzed statically, based on loading combinations described in Subsection 3.8.4.3. The equivalent static loads resulting from the application of the seismic accelerations or displacements at various levels obtained from the above mentioned dynamic analysis are utilized. The method of transforming the dynamic loads with equivalent static loads is as follows:

a) The dynamic piping loads are transformed into static loads in the piping stress analysis by either utilizing the model response spectra method or by applying an amplification factor to the excitation load. These pipe loads on structures are considered in the design for both the positive and negative directions with the same magnitude.

b) The dynamic effects of pipe whip and jet impingement are discussed in Section 3.6. The pipe whip loads on the structural components (i.e., pipe whip restraints) were calculated utilizing static methods while various confirmatory dynamic analysis were utilized to confirm these piping loads (refer to Appendix 3.6E). The jet impingement analysis utilized a Dynamic Load Factor (DLF) of two applied to the KPA load to determine the loads on structural components (refer to Subsection 3.6.2).

c) The dynamic effects associated with the containment subcompartment pressure analysis were considered in the structural design of the subcompartment walls. The equivalent static loads were obtained by applying a Dynamic Load Factor to the peak of each dynamic loads. The calculated peak pressure for each subcompartment and the corresponding design values (with calculated margin) is provided in Subsection 6.2.3.

The two tower structures of the main steam trestle are analyzed using finite element techniques. The analysis is done using the MRI/STARDYNE computer program. Horizontal and vertical design loads are transmitted through the two braced towers and into the supporting concrete by means of the column base plate with shear lugs and the anchor bolt/anchor plate anchorages.

See Section 3.7 for a description of the dynamic analysis of the main steam trestle. Table 3.8-22 shows a comparison of actual and allowable stresses for selected elements of the main steam trestle.

3.8.4.5 Structural Acceptance Criteria

The structural acceptance criteria for the seismic Category I, concrete structures are given in Subsection 3.8.3.5.1. For all loading conditions the stresses in the structure do not exceed the yield strength of the materials used. Table 3.8-23 compares the required design strength and the ultimate capacity of various structural elements for the governing loading conditions.

For each of the loading combinations described in Subsection 3.8.4.3.2.2, the following defines the allowable limits which constitute the structural acceptance criteria for steel structures,

Load Combination	Limit
a1, a2, a3	S
a4, a5, a6	1.5S
b1, b2, b3, b4, b5	1.6S

S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC Specification.

3.8.4.6 Materials, Quality Control and Special Construction Techniques

The basic materials used for the construction of the seismic Category I structures described in Subsection 3.8.4.1 are concrete, reinforcing steel and structural steel. The material specifications, testing requirements and quality control measures specified in this section form part of the overall Quality Assurance Program described in Chapter 17. See Subsection 3.8.3.6.1 for codes and standards, concrete materials and quality control applied to concrete structures.

3.8.4.6.1 Shield Building Tolerances

The finished concrete tolerances for the Shield Buildings cylinder wall are such that variation from plumb is not more than ± four in. for the total structure height taken at the vertical axis of the building, or more than one in. in any 20 ft of wall height; variation from true circular section is not more than ± three in. in radius; and variation of wall thickness is not less than -1/4 in. or more than + one in.

3.8.4.6.2 Special Construction Techniques

The three ft thick portion of the Shield Building cylinder wall is built by using the slip forming construction technique. Since in a slip forming operation the penetration sleeves cannot be set within the specification tolerances, blockouts are provided in which the sleeves are set in a second stage concreting operation.

For two major size adjoining blockouts in the electrical and mechanical penetrations area, the reinforcing steel is interrupted at the periphery of the blockouts (after being fully developed), and replaced with structural steel skin plates within the blockouts. The skin plates are continuously welded to plates embedded in concrete all around the blockouts. To ensure a positive connection between the skin plates and the fill concrete, Nelson studs are welded to the plates at a spacing close enough to limit the stress intensity to well below the buckling stress level.

The dome of the Shield Building is constructed in two stages. First a thin dome is formed, supported by shores resting on the steel containment dome; in the second stage the dome is completed to its final thickness with the weight of the additional concrete and construction loads supported by the thin dome placed first.

3.8.4.6.3 Reinforcing Steel

See Subsection 3.8.3.6.2

3.8.4.6.4 Structural Steel

See Subsection 3.8.3.6.3

3.8.4.7 Testing and In-Service Surveillance Requirements

There are no planned systematic testing or surveillance programs for the seismic Category I structures after the plant has been placed in operation. The structural steel framing and connections are generally accessible to visual inspection.

3.8.5 FOUNDATIONS

3.8.5.1 Description of the Foundations

Seismic Category I structures are supported on separate reinforced concrete mats.

The Reactor Building base slab as shown on Figures 3.8-70 and 71 supports the containment vessel and the Shield Building. The base slab is a dish shaped slab, 160 ft in diameter, 10 ft. thick consisting of a central lower section 57.5 ft in diameter (top elevation-15.5 ft) and an inclined section with a slope of four horizontal to one vertical. The outer portion of the mat is 12 ft thick and levels with the top at elevation - 3 ft. The Shield Building cylinder wall is directly connected to the base slab. The steel containment vessel is supported on the fill concrete which transfers the loads by bearing to the base slab. To assure proper contact between the containment and the concrete the interface is grouted with epoxy.

The internal structures described in Subsection 3.8.3 are supported on fill concrete which transfers the loads to the foundation mat.

The foundation mats supporting the Reactor Auxiliary Building (Figure 3.8-72), Fuel Handling Building and Diesel Generator Building, are all rectangular in shape and of different thicknesses: three ft for the Diesel Building, four ft for the Reactor Auxiliary Building and five ft/eight feet six inches for the Fuel Handling Building. The walls and columns of these structures transfer the loads directly to the foundation mats.

Waterproofing membranes are provided under and around the edges of the mats used for the Shield Building and Reactor Auxiliary Building.

3.8.5.2 Applicable Codes, Standards and Specifications

The applicable codes, standards and specifications are as given in Subsection 3.8.3.2.

Foundations for St. Lucie Unit 2 are designed in accordance with ACI-318-71, "Building Requirements for Reinforced Concrete." A review of the 1977 edition of ACI 318 Code has determined that the changes have insignificant effect on foundation design requirements. See Subsection 3.8.3.2.1 for discussion of the comparison of SRP 3.8.4 load combination requirements and those used in the St. Lucie Unit 2 design.

3.8.5.3 Loads and Loading Combinations

Each foundation mat is subjected to loads transferred from the structure supported on it. Applicable loads and loading combinations considered in the design of the foundation mats are as described in Subsection 3.8.4.3. In addition, the foundations are checked against sliding and overturning due to earthquake, tornado and hurricane winds and against flotation due to maximum water level.

For tables indicating the factors of safety against sliding, overturning and floatation for the major and typical seismic Category I Buildings, see Tables 3.8-27 through 3.8-33.

3.8.5.4 Design and Analysis Procedures

Conventional methods involving simplifying assumptions such as rigid mat procedure and flat slab design, as well as slab on elastic foundation theory are utilized in the analysis of the foundation mats. The Shield Building mat is analyzed as a circular slab resting on an elastic foundation, using the finite element program SOLIDS II (See Subsection 3.8.3.4.2). The Reactor Auxiliary Building mat is analyzed as a flat slab in accordance with the provisions of ACI 318-71.

The MRI STARDYNE static analyses program is used for the analysis of the Fuel Handling Building mat. (See Subsection 3.8.3.4.2 for a description of the code).

The dynamic analysis of the seismic Category I structures including their foundation mats is described in Section 3.7. The maximum structure loads resulting from this analysis are used in the mat design.

The foundation mats have been designed such that the resulting soil pressures are within the allowable limits as discussed in Section 2.5.

Load transfer from the seismic Category I structures to the foundation mat is achieved through conventional wall-to-mat and column-to-mat connections. The load transfer to the foundation mat from seismic Category I equipment is achieved through the supporting structural members such as concrete pedestals, floor slabs and concrete walls.

The load transfer to the soil materials is discussed in Section 2.5.

3.8.5.5 Structural Acceptance Criteria

The structural acceptance criteria for the seismic Category I foundation mats are as discussed in Subsection 3.8.3.5. In addition adequate margins of safety are provided against overturning and sliding as well as shear failure of the foundation materials, as presented in Subsection 2.5.4.

Waterproofing membrane utilized on the Shield Building and Reactor Auxiliary Building, does not have significant effect on the capability of the foundations to transfer shears. The design calculations for resistance to sliding are based upon the combination of resisting earth pressure and the waterproofing membrane shear strength value documented by the manufacturer, as follows:

Shield Building - A substantial portion of the Shield Building has structural concrete and fill concrete throughout the building cross-section below plant island grade elevation. Therefore the soil passive pressure loads on the building are of no design significance. However, the design analysis of the building determined a seismic movement and resisting earth pressure less than the full passive earth pressure. This requires the waterproofing membrane to be able to transfer shear forces. Therefore, the design calculations for resistance to sliding are based upon the combination of resisting earth pressure and the membrane shear strength value documented by the manufacturer. See Figure 3.8-73 for the coefficient of passive pressure and design resisting pressure.

Reactor Auxiliary Building - The building general arrangement and resulting mat layout allows sliding to be resisted through internal shear resistance of the soil and resisting earth pressures in the east, west and south directions. The ability of the waterproofing membrane to transfer shear forces is not required in those directions. In the north direction, the earth pressure required to resist sliding is greater than the design capacity of the building walls. Therefore, the design calculations for resistance to sliding in that direction are based upon the combination of resisting earth pressure and the membrane shear strength value. See Figure 3.8-74 for resisting pressures.

3.8.5.6 Materials, Quality Control and Special Construction Techniques

For details of applicable material specifications, quality control provisions and any special construction techniques for the seismic Category I concrete foundations refer to Subsections 3.8.3.6 and 3.8.4.6.

3.8.5.7 Testing and Inservice Surveillance Requirements

The only instrumentation required for the surveillance of the foundations for seismic Category I structures are settlement monuments. The results and analysis of settlement data is presented in Subsection 2.5.4.10.

SECTION 3.8: REFERENCES

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12) PC/M 05133, Steam Generator 2A & 2B Supports Modification for Unit 2 Component Replacement Projects.

13) AREVA NP Inc., 77-5069878-004, "Replacement Steam Generator Report for Florida Power and Light St. Lucie Unit 2," December 2015. EC283094

TABLE 3.8-1
CONTAINMENT VESSEL LOAD COMBINATIONS

Load	Load Combinations												
	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9	Case 10	Case 11	Case 12	Case 13
Internal Pressure (psi)		50	44			44	44				44		
External Pressure (psi)				1.05	1.05							1.05	1.05
Dead Load of Vessel Appurtenances	X	X	X	X	X	X	X	X	X	X	X	X	X
Containment Air @ Test		X	X										
Dead Load of Ventilation Duct		X	X	X	X	X	X	X	X	X	X	X	X
Dead Load of Penetration Internals		X	X	X	X	X	X	X	X	X	X	X	X
Crane Live Load					X								X
Crane Dead Load		X	X	X	X	X	X	X	X	X	X	X	X
Lateral Load Due to Wind	X	X											
OBE Horizontal Load			X	X	X	X		X		X			
SSE Horizontal Load							X		X		X	X	X
OBE Vertical Load			X	X	X	X		X		X			
SSE Vertical Load							X		X		X	X	X
Live Load on Air Locks		X	X	X	X	X	X	X	X	X	X	X	X
Live Load on Equip. Hatches					X								X
Live Load on Platforms	X												
Attachment Load (Thermal & Seismic)*			X	X	X	X	X	X	X	X	X	X	X
Thermal Loads at Embedment										X	X		
Pipe Rupture Loads at Penetration				X				X	X			X	
Thermal Loads at Penetration				X		X	X	X	X	X	X	X	
Jet Forces										X	X		
Seismic Loads at Penetration			X	X	X	X	X	X	X	X	X	X	X

EC280084

*For pipes, HVAC ducts and electrical trays and conduits.

TABLE 3.8-2

CONTAINMENT VESSEL PENETRATIONS - LOAD COMBINATIONS AND STRESS LIMITS

Load Combination	Temperature Condition	Stress Limits as Per ASME Code (Section III)
Type I Penetration		
$P_c + P_a + EPT + W$	$T_{c1}\ T_a$	Fig. NE-3221-1
$P_{c1} + P_{a1} + EPT + W$	$T_{c1}\ T_{a1}$	Fig. NE-3221-1
$P_c + P_a + TT + 2EPT + W$	$T_{c1} T_a$	NB-3222.2, NE (for Sm)
$P_{c1} + P_{a1} + TT + 2EPT + W$	T_{c1}, T_{a1}	NB-3222.2, NE (for Sm)
$P_{c1} + P_{a1} + EPT' + W$	T_{c1}, T_{a1}	NE-3131(c)
$P_c + P_a + PRT + EPT' + W$	$T_{c1}\ T_{a1}$	NE-3131.2
Type II Penetration		
$P_g + P_d + P_c + P_a + EPA + EPM + W$	T_g, T_d, T_c, T_a	Fig. NE-3221-1
$P_g + P_d + P_c + P_a + EPT + EPT_o + W$	T_g, T_d, T_c, T_a	Fig. NE-3221-1
$P_{g1} + P_o + P_{c1l} + P_{a1} + EPA + EPM + W$	$T_{g1}, T_o, T_{c1}, T_{a1}$	Fig. NE-3221-1
$P_{g1} + P_o + P_{c1} + P_{a1} + EPT + EPT_o + W$	$T_{g1}\ T_o, T_{c1}, T_{a1}$	Fig. NE-3221-1
$P_g + P_o + P_c + P_a + TA + TM + 2\ EPA + 2\ EPM + W$	$T_g, T_o, T_{c1}\ T_a$	NB-3222.2, NE (for Sm)
$P_g + P_o + P_c + P_a + TT + TT_o + 2\ EPT + 2EPT_o + W$	$T_{g1} T_o, T_c, T_a$	NB-3222.2, NE (for Sm)
$P_{g1} + P_o + P_{c1} + P_{a1} + TA + TM + 2EPA + 2EPM + W$	$T_{g1}\ T_o, T_{c1}, T_{a1}$	NB-3222.2, NE (for Sm)
$P_{g1} + P_o + P_{c1} + P_{a1} + TT + TT_o + 2EPT + 2EPT_o + W$	$T_{g1}, T_o, T_{c1}, T_{a1}$	NB-3222.2, NE (for Sm)
$P_{g1} + P_o + P_{c1} + P_{a1} + EPA + EPM + W$	$T_{g1}, T_o, T_{c1}, T_{a1}$	NE-3131 (c)
$P_{g1} + P_o + P_{c1} + P_{a1} + EPT + EPT_o + W$	$T_{g1}, T_o, T_{c1}, T_{a1}$	NE-3131 (c)

Historical Information. For loading combinations refer to specification FLO-2998-757.

TABLE 3.8-2 (Cont'd)

Load Combination	Temperature Condition	Stress Limits as Per ASME Code (Section III)
	Type II Penetration	
$P_g + P_o + P_c + P_a$ + PRA + PRM + EPA' + EPM' + W	T_g, T_o, T_c, T_a	NE-3131.2
$P_g + P_o + P_c + P_a$ + PRT + PRT_o + EPT' + EPT_o' + W	T_g, T_o, T_c, T_a	NE-3131.2
$P_g + P_o + P_c + P_a$ + PRT + PRM + EPT' + EPM' + W	T_g, T_o, T_c, T_a	NE-3131.2
	Type III Penetration	
$P_d + P_g + P_c + P_a$ + EPA + EPM + W	T_d, T_g, T_c, T_a	Fig. NE-3221-1
$P_d + P_g + P_c + P_a$ + EPT + EPT_o + (120 in.)x(EPT) + W	T_d, T_g, T_c, T_a	Fig. NE-3221-1
$P_o + P_{g1} + P_{c1} + P_{a1}$ + EPA + EPM + W	T_o, T_{g1}, T_{c1} T_{a1}	Fig. NE-3221-1
$P_o + P_{g1} + P_{c1} + P_{a1}$ + EPT + EPT_o + (120 in.)x(EPT) + W	T_o, T_{g1}, T_{c1} T_{a1}	Fig. NE-3221-1
$P_o + P_g + P_c + P_a$ + TA + TM + 2EPA + 2EPM + W	T_o, T_g, T_c, T_a	NB-3222.2, NE (for Sm)
$P_o + P_g + P_c + P_a$ + TT + TT_o + 2EPT + $2EPT_o$ + (120 in.)x(TT + 2EPT) + W	T_o, T_g, T_c, T_a	NB-3222.2, NE (for Sm)
$P_o + P_{g1} + P_{c1} + P_{a1}$ + TA + TM + 2EPA + 2EPM + W	T_o, T_{g1}, T_{c1}, T_{a1}	NB-3222.2,
$P_o + P_{g1} + P_{c1} + P_{a1}$ + TT + TT_o + 2EPT + $2EPT_o$ + (120 in.)x(TT + 2EPT) + W	T_o, T_{g1} T_{c1}, T_{a1}	NB-3222.2, NE (for Sm)
$P_o + P_{g1} + P_{c1} + P_{a1}$ + EPA' + EPM' + W	T_o, T_{g1} T_{c1} T_{a1}	NE-3131 (c)

Historical Information. For loading combinations refer to specification FLO-2998-757.

TABLE 3.8-2 (Cont'd)

Load Combination	Temperature Condition	Stress Limits as Per ASME Code (Section III)
Type III Penetration		
$P_o + P_{g1} + P_{c1} + P_{a1} + EPT' + EPT_o' + (120\ in.)x(EPT') + W$	T_o, T_{g1} T_{c1} T_{a1}	NE-3131 (C)
$P_o + P_g + P_c + P_a + PRA + PRM + EPA' + EPM' + W$	T_o, T_g, T_c, T_a	NE-3131.2
$P_o + P_g + P_c + P_a + PRT + PRT_o + (120\ in.) x(PRT + EPT') + EPT' + EPT ' + W$	T_o, T_g, T_c, T_a	NE-3131.2
$P_o + P_{gr} + P_c + P_a + PRT + EPT' + EPM' + W$	T_o, T_{gr}, T_c, T_a	NE-3131.2
$P_o + P_{gr} + P_c + P_a + PRT + EPT' + EPT_o' + (120\ in.)x(EPT') + W$	T_o, T_{gr}, T_c, T_a	NE-3131-2
$P_o + P_{gr} + P_c + P_a + PRT + EPT' + EPM' + W$	T_o, T_{gr}, T_c, T_a	NE-3131.2
$P_o + P_{gr} + P_c + P_a + PRT + PRM + EPT' + EPM' + W$	T_o, T_g, T_c, T_a	NE-3131.2

Historical Information. For loading combinations refer to specification FLO-2998-757.

TABLE 3.8-3

CONTAINMENT SHELL STRESSES AT JUNCTION OF COLUMN AND KNUCKLE

Conditions	Load Combinations
I. P.W.H.T	1) Vertical Bending Due to Column Loads
	2) Horizontal Bending Due to Wind Shear and Column Eccentricity
	3) Torsion a) Due to A-Frame Eccentricity b) Due to A-Frame Pin Connections
II. Construction State (with Concrete in Bottom Head & Wind Load	1) Vertical Bending Due to Column Loads
	2) Horizontal Bending Due to Wind Shear and Column Eccentricity
	3) Torsion Due to A-Frame Eccentricity
III. Construction State (with Concrete in Bottom Head & Hard) Point Loads	1) Vertical Bending Due to Column Loads
	2) Horizontal Bending Due to Column Eccentricity
	3) Torsion Due to A-Frame Eccentricity
IV. During Final Test	1) Vertical Bending Due to Column Loads
	2) Horizontal Bending Due to Wind Shear and Column Eccentricity
	3) Torsion Due to A-Frame Eccentricity
	4) Internal Pressure

Historical Information. For loading combinations refer to design bases calculation by CB&I, PSL-73-7302.

TABLE 3.8-4

SUMMARY OF STRESSES
IN BOTTOM HEAD KNUCKLE

Membrane Stresses

	At Column			At Midspan		
	Φ	θ	SI	σΦ	σθ	SI
Location	(psi)	(psi)	(psi)	(psi)	(psi)	(psi)
90	7330	1095	7330	9650	530	9650
92.1	7250	-3730	10980	9675	-4175	13850
94.2	7145	-10235	17380	9725	-3456	18180
96.3	7040	-12120	19160	9780	-12070	21850
98.4	6960	-15390	22350	9855	-14915	24770
100.5	6925	-17955	24880	9950	-16990	26940
102.6	6995	-19800	26795	10060	-18350	28410
104.7	7195	-20930	28125	10180	-19090	29270
106.8	7595	-21330	28925	10210	-19320	29630
108.9	8290	-21000	29240	10400	-19130	29530
111.0	9150	-19960	29110	10600	-18615	29215
113.6	9475	-18300	27775	10730	-17685	28415
115.3	9785	-17085	26870	10815	-16930	27745
117.0	10110	-15915	26025	10905	-16100	27005
118.7	10410	-14820	25240	10995	-15220	26215
120.4	10705	-13810	24515	11090	-14305	25395
122.1	10965	-12875	23840	11200	-13380	24580
123.8	11190	-12000	23190	11320	-12440	23760
125.5	11400	-11155	22555	11450	-11505	22955
127.2	11600	-10315	21915	11600	-10565	22165
128.5	11745	-9650	21395	11730	-9825	21555
129.5	11855	-9155	21010	11830	-9280	21110
130.4	11960	-8645	20605	11935	-8730	20665

Allowable S.I. <.9 Sy = 34200 psi NE-6322

Historical Information. For stress summary refer to design bases calculation by CB&I, PSL-73-7302.

TABLE 3.8-5

SUMMARY OF STRESSES
IN BOTTOM HEAD KNUCKLE

Max Surface Stresses

	At Column			At Midspan		
Location	σΦ (psi)	σθ (psi)	SI (psi)	σΦ (psi)	σθ (psi)	SI (psi)
90°	6400	1990	6400	9585	460	9585
92.1	12320	-945	13765	16545	-7105	18650
94.2	15870	-4090	19960	20255	-5255	25510
96.3	17665	-7090	24750	21670	-8460	30130
98.4	18325	-9675	27990	21565	-11380	32945
100.5	18385	-11715	30100	20550	-13830	34380
102.6	18230	-13135	31365	19100	-16705	34805
104.7	18095	-13970	32065	17510	-16995	34505
106.8	17935	-14405	32340	16005	-17745	33750
108.9	17240	-14795	32035	14660	-18015	32675
111.0	14525	-15760	30280	13510	-17880	31390
113.6	10260	-16880	27140	12355	-17310	29665
115.3	9205	-16725	25930	11780	-16725	28515
117.0	8920	-16145	25065	11365	-16020	27385
118.7	9080	-15335	24415	11095	-15215	26310
120.4	9475	-14405	23880	10960	-14360	25320
122.1	9975	-13430	23405	10940	-13465	24405
123.8	10500	-12445	22945	11020	-12535	23555
125.5	10985	-11480	22465	11185	-11605	22790
127.2	11420	-10530	21950	11400	-10660	22060
128.5	11715	-9795	21510	11605	-9910	21515
129.5	11905	-9260	21165	11760	-9360	21120
130.4	12070	-8720	20790	11915	-8810	20725

Allowable S.I. < 1.25 Sy = 47500 psi NE 6322

Historical Information. For stress summary refer to design bases calculation by CB&I, PSL-73-7302.

TABLE 3.8-6

PENETRATION ANALYSIS

Style Or Reinforcement	Location Of Analysis	Reference For Curves	Radial M Parameter (1) Stress	Radial M % Increment	Tangential M Parameter Stress	Tangential M % Increment	Radial N Parameter Stress	Radial N % Increment	Tangential N Parameter Stress	Tangential N % Increment	Comments
Insert (width < 1.65 $\sqrt{RTins}$)	Insert to Neck	WRC 107	Tins Tins	20	Tins Tins	20	Ts Tins	-	Ts Tins	-	
	1/2 $\sqrt{RTins}$ from local stress	WRC 107	Tins Tins	20	TinsTins	20	Ts Tins	-	Ts Tins	-	See Note 3
	Insert to Shell	WRC 107	Ts Ts	-	Ts Ts	-	Ts Ts	-	Ts Ts	-	See Note 4
Pad (width < 1.65 $\sqrt{RTeq}$)	Pad to Neck	WRC 107	Tp+s Tp+s	20	Tp+s Tp+s	20	Ts Tp+s	-	Ts Tp+s	-	
	1/2 TЯ√ from local stress	WRC 107	Tp+s Tp+s	20	TP+s Tp+s	20	Ts p+s	-	Ts Tp+s	-	See Note 3
	Edge of Pad	WRC 107	Ts Ts	-	Ts Ts	-	Ts Ts	-	Ts Ts	-	See Note 4
Shell (No reinforce- ment width≥ 1.65 $\sqrt{RT\ reinf.}$)	Neck to Shell	WRC 107	Ts Ts	-	Ts Ts	-	Ts Ts	-	Ts Ts	-	
	1/2 $\sqrt{RTs}$ from Neck	WRC 107	Ts Ts	-	Ts Ts	-	Ts Ts	-	Ts Ts	-	See Note 3

Notes:
(1) Indicates thickness for calculation of parameter and stresses
(2) any alternate method may be used for a location of analysis in combination with the above method for other locations
(3) check only if stresses at other location > 1.1 Sm. Use Ts and no increase in stresses if outside reinforcing
(4) stresses due to loads may be reduced in accordance with WRC 95

Nomenclature:

Tins = thickness of insert
Teq = equivalent thickness
Tp+s = thickness of pad plus shell
Ts = thickness of shell

Historical Information. For penetration analysis refer to design bases calculation by CB&I, PSL-73-7302.

TABLE 3.8-7
CONTAINMENT VESSEL
ALLOWABLE STRESSES

ASME Code is used in the design of the steel shell and its penetrations. AISC refers to all other steel structures, interacting with the containment vessel, such as crane girders, platforms, and temporary supports.

CASE 1 - Construction at Post Weld Heat Treatment (PWHT)

No ASME Design (Shell is analyzed using methods consistent with the ASME Code)

AISC Design

AISC Allowables

CASE 2 - Acceptance Test at Ambient Temperature

ASME Design

$PM \leq 0.9\ Sy$

$PL + PB \leq 1.25\ Sy$

AISC Design

AISC Allowables

CASE 3 - Preoperation Test at Ambient Temperature

ASME Design with OBE earthquake

$PM \leq 1.0\ Sm$ (includes seismic stress)

$PL + PB \leq 1.5\ Sm$

$PL + PB + Q \leq 3.0\ Sm$

AISC Design with OBE earthquake

AISC Allowables with normal increase

CASE 4 - Normal Operating Condition with OBE Earthquake and a Temperature Range of 30 F to 150 F

ASME Design with OBE earthquake

$PM \leq 1.0\ Sm$ (includes seismic stress)

$PL + PB \leq 1.5\ Sm$

$PL + PB + Q \leq 3.0\ Sm$

AISC Design with OBE earthquake

AISC Allowables with normal increase

TABLE 3.8-7 (Cont'd)

CASE 5 - Cold Shutdown with OBE Earthquake and a Temperature Range of 30 F to 120 F

ASME Design with OBE earthquake
PM ≤ 1.0 Sm (Includes Seismic Stress)
PL + PB ≤ 1.5 Sm
PL + PB + Q ≤ 3.0 Sm

AISC Design with OBE earthquake

AISC Allowables with normal increase

CASE 6 - Accident Condition with OBE

ASME Design with OBE earthquake
PM ≤ 1.0 Sm (Includes Seismic Stress)
PL + PB ≤ 1.5 Sm
PL + PB + Q ≤ 3.0 Sm

AISC Design with OBE earthquake

AISC Allowables with normal increase

CASE 7 - Accident Condition with SSE

ASME Design with SSE earthquake
PM ≤ 0.9 Sy (includes Seismic Stress)
PL + PB ≤ 1.5 Sy

AISC Design with SSE earthquake

AISC Allowables with normal increase

CASE 8- Condition with OBE, Pipe Rupture, Pipe Thermal, and Seismic Load at Penetration

ASME Design with OBE earthquake
PM ≤ 1.0 Sm (Includes Seismic Stress)
PL + PB ≤ 1.5 Sm
PL + PB + Q ≤ 3.0 Sm

Note:
Pipe loads are investigated as
a local effect separately
Pm ≤ 0.9 Sy
PL + PB ≤ 1.5 Sy

AISC Design with OBE earthquake

AISC Allowables with normal increase

TABLE 3.8-7 (Cont'd)

CASE 9 - Condition with SSE, Earthquake, Pipe Rupture, Pipe Thermal, and Seismic Loads at Penetrations

ASME Design with SSE earthquake

$PM \leq 0.9$ Sy (Includes Seismic Stress)

$PL + PB \leq 1.5$ Sy

Note:
Pipe loads are investigated as a local effect separately

$PM \leq 0.9$ Sy

$PL + PB \leq 1.5$ Sy

AISC Design with SSE earthquake

AISC Allowables with normal increase

CASE 10 - Condition with OBE, Earthquake, Jet Forces, Thermal Loads at Penetrations, Pipe Thermal and Seismic Loads at Penetrations

ASME Design with OBE earthquake

$PM \leq 1.0$ Sm (Includes Seismic Stress)

$PL + PB \leq 1.5$ Sm

$PL + PB + Q \leq 3.0$ Sm

Note:
Pipe loads are investigated as a local effect separately using ASME Code allowables

AISC Design with OBE earthquake

AISC Allowables with normal increase

CASE 11 - Accident Condition with SSE Earthquake, Jet Forces, Thermal Plus Seismic Loads at Penetrations and Pipe Thermal

ASME Design with SSE earthquake

$PM \leq 0.9$ Sy (includes seismic stress)

$PL + PB \leq 1.5$ Sy

Note:
Pipe loads are investigated as a local effect separately using ASME Code allowables

AISC Design with SSE earthquake

AISC Allowables with normal increase

TABLE 3.8-7 (Cont'd)

CASE 12 - Normal Operating Condition with SSE loads at temperature range of 30F to 150F

ASME Design

$PM \leq 0.9$ Sy (includes seismic stress)

$PL + PB \leq 1.5$ Sy

AISC Design

AISC Allowables (no increase due to seismic loads)

CASE 13 - Cold shutdown with SSE loads at temperature range of 30F to 150 F

ASME Design

$PM \leq 0.9$ Sy (includes seismic stress)

$PL + PB \leq 1.5$ Sy

AISC Design

AISC Allowables (no increase due to seismic loads)

TABLE 3.8-8

SUMMARY OF HEMISPERICAL DOME STRESSES

	Meridional (Longitudinal) Stresses				Circumferential Stresses			
	Compressive Stress (psi)		Tensile Stress (psi)		Compressive Stress (psi)		Tensile Stress (psi)	
Load Case	Allowable	Max. Calc	Allowable	Max. Calc	Allowable	Max. Calc	Allowable	Max. Calc
1	-2055*	-304	19300	0	-2055*	-34	19300	304
2	-2569	0	34200	21722	-2569	0	34200	22211
3	-2055	0	19300	19098	-2055	0	19300**	19585
4	-2055*	-599	19300	0	-2055*	-127	19300	0
5	-2055*	-599	19300	0	-2055*	-127	19300	0
6	-2055	0	19300	19098	-2055	0	19300**	19585
7	-2466	0	34200	19132	-2466	0	34200	19619
8	-2055	-293	19300	0	-2055	-55	19300	293
9	-2466	-327	34200	0	-2466	-89	34200	327
10	-2055	-293	19300	0	-2055	-55	19300	293
11	-2466	0	34200	19132	-2466	0	34200	19619

* For Biaxial Buckling $\dfrac{N\theta - N\phi}{1.8x\,10^{6}\left(\dfrac{t}{R}\right)^{2}} + \dfrac{N\phi}{.9x\,10^{6}\left(\dfrac{t}{R}\right)^{2}} \leq 1$

** The Code allows 10% over allowable in the absence of substantial mechanical or thermal loads other than pressure.

Historical Information. For stress summary refer to design bases calculation by CB&I, PSL-73-7302.

TABLE 3.8-9

SUMMARY OF CYLINDER STRESSES

	Meridional (Longitudinal) Stresses				Circumferential Stresses			
	Compressive Stress (psi)		Tensile Stress (psi)		Compressive Stress (psi)		Tensile Stress (psi)	
Load Case	Allowable	Max. Calc	Allowable	Max. Calc	Allowable	Max. Calc	Allowable	Max. Calc
1	-4105	-930	19300	0	-534	0	19300	0
2	-5131	0	34200	10455	-668	0	34200	21923
3	-4105	0	19300	9182	-534	0	19300	19292
4	-4105	-1150	19300	0	-534	-307	19300	0
5	-4105	-1200	19300	0	-534	-307	19300	0
6	-4105	0	19300	9182	-534	0	19300	19292
7	-4926	0	34200	9343	-641	0	34200	19292
8	-4105	-997	19300	0	-534	0	19300	0
9	-4926	-1158	34200	0	-641	0	34200	0
10	-4105	-997	19300	0	-534	0	19300	0
11	-4926	0	34200	9343	-641	0	34200	19292

Historical Information. For stress summary refer to design bases calculation by CB&I, PSL-73-7302.

TABLE 3.8-10

CONTAINMENT VESSEL MATERIALS

Material	Specification	Design Stress Intensity (psi)	Code	Remarks
Plate	SA 516, Gr 70	19300	ASME	
	SA 240, Type 304			
Forgings	SA 350, Gr LF 1*	16500	ASME	
	LF 2*	19300		
	SA 182, F304	18300	ASME	
Pipe	SA 333, Gr 6 seamless*	16500	ASME	Thru 14"
	SB 168	20700	ASME	10"
	SB 167	20700	ASME	Thru 5"
	SB 166	20700	ASME	6" & 8"
	SA 516, Gr 70	19300	ASME	16" & Greater
Castings	SA 352, Gr LC1*	16200	ASME	
	SA 351, Gr CF8	16000	ASME	
Bolting	SA 193 B7**	25000	ASME	
	SA 193 B8	15000	ASME	
Structural	A 36	% Fy	AISC	Not used for pressure parts nor within 4" of pressure parts, except painter's angle.

* All of the above designated carbon steel materials comply with the requirements of the applicable ASME Code Material Specification for low temperature service (as specified in Article NE-2000 of Subsection NE of Section III of the ASME Code) except that the impact testing, as a minimum requirement, is performed as specified in Section III of the ASME Code, Paragraph NE 2300. Charpy V-Notch specimens (SA370-Type A) are used for all impact testing at a maximum temperature of 0 F.

** All carbon steel bolting material is impact tested in accordance with the requirements of ASME SA 320.

Historical Information. For loading combinations refer to specification FLO-2998-757.

TABLE 3.8-11

CONTAINMENT INTERNAL CONCRETE STRUCTURE LOAD COMBINATIONS

1. Normal operating

$$U = 1.4D + 1.3(R_O + T_O) + 1.7L$$
$$U = 1.4D + 1.3(R_O + T_O)$$

2. Severe Environmental (OBE)

$$U = 1.4D + 1.3(R_O + T_O) + 1.7L + 1.9E$$
$$U = 1.2D + 1.9E$$

3. Extreme Environmental (SSE)

$$U = 1.0(D + R_O + T_O) + 1.0(L + E')$$
$$U = 1.0(D + R_O + T_O) + 1.0E'$$

4. Abnormal (Pipe break accident)

$$U = 1.0D + 1.0(L + R_a + T_a) + 1.5P_a$$
$$U = 1.0D + 1.0(R_a + T_a) + 1.5P_a$$

5. Abnormal/Severe Environmental (OBE)

$$U = 1.0D + 1.0(L + R_a + T_a) + 1.0(Y_r + Y_j + Y_m) + 1.25(P_a + E)$$
$$U = 1.0D + 1.0(R_a + T_a) + 1.0(Y_r + Y_j + Y_m) + 1.25(P_a + E)$$

6. Abnormal/Extreme Environmental (SSE)

$$U = 1.0D + 1.0(L + R_a + T_a) + 1.0(Y_r + Y_j + Y_m) + 1.0(P_a + E')$$
$$U = 1.0D + 1.0(R_a + T_a) + 1.0(Y_r + Y_j + Y_m) + 1.0(P_a + E')$$

Notes:

U = Ultimate Design Load
Loads (D, Ro etc) are defined in Subsection 3.8.3.3.1

TABLE 3.8-12

STEEL INTERNAL STRUCTURES - LOADING COMBINATIONS & ALLOWABLE STRESSES

Service load conditions

1) $D + L \leq S$

2) $D + L + E \leq S$

3) $D + L + T_o + R_o \leq 1.5S$

4) $D + L + T_o + R_o + E \leq 1.5S$

Factored load conditions

5) $D + L + T_o + R_o + E' \leq 1.6S$

6) $D + L + T_a + R_a + P_a \leq 1.6S$

7) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E \leq 1.6S$

8) $D + L + T_a + R_a + P_a + 1.0\ (Y_j + Y_r + Y_m) + E' \leq 1.6S$

Notes:

(1) Loads (D,L etc) are defined in Subsection 3.8.3.3.1.

(2) For load combinations 6, 7 and 8, R_a and T_a are applied statically.

(3) S is the required section strength based on the elastic design method and the allowable stresses defined in Part I of AISC Code.

TABLE 3.8-13

COMPARISON OF MAXIMUM DESIGN AND ALLOWABLE STRESSES FOR UPPER STEAM GENERATOR SUPPORTS
See Figure 3.8-42

MEMBER	MAX. STRESSES (KSI) CALCULATED STRESS	ALLOWABLE	REMARKS
1-1/2" Φ Anchor Bolt for Snubber Bracket	42.44[(a)]	64.0	
Anchor Plate for Snubber Bracket	59.53[(b)]	67.2	
1-3/4" Plate of Support Beam	55.59[(c)(d)]	67.2	Point D
2" Φ Anchor Bolt for Support Beam	33.04[(a)]	64.0	
Anchor Plate for Support Beam	47.84[(b)]	67.2	
W 14 x 342	41.65[(b)(d)]	48.0	Point E

Notes:

(a) axial stress
(b) bending stress
(c) principal stress
(d) historical

TABLE 3.8-14

STEAM GENERATOR SLIDING BASE SUPPORT STRESS COMPARISON
See Figure 3.8-42

MEMBER	MAX. STRESSES (KSI) CALCULATED STRESS	ALLOWABLE STRESS	REMARKS
Key	53.6[d]	67.2	
10" Top Plate	18.3[c]	67.2	
3" Stiff Plate	58.48[b]	67.2	Point A
2-1/4" ϕ A354 Gr BD Anchor Bolt	81.84[a]	86.4	
5" Anchor Plate	50.06[b]	67.2	Point B
2-1/2" Anchor Plate	49.44[b]	67.2	Point C

Notes:

(a) axial stress
(b) bending stress
(c) principal stress
(d) bearing stress

TABLE 3.8-15

REACTOR SUPPORT STRUCTURE STRESS SUMMARY
TYPICAL LOCA CONDITION(1)

LOCATION OF MEMBER	ELEMENT ID No.	MAXIMUM STRESS(2) fa (CALCULATED)(3)	(KSI) Fa (ALLOWABLE)(3)	REMARKS
Upper Flange of Horiz Girder:				
Near One Third Point	96	-39.5	-40.32	4" Plate
Near End Point	104	-29.2	-40.32	4" Plate
Lower Flange Of Horiz Girder:				
Near One Third Point	110	-15.8	-40.32	4" Plate
Near End Point	120	-20.7	-40.32	4" Plate
Horiz Girder Web Plate (Near End)	19	-30.1	-40.32	3" Plate
Horiz Girder Stiff Plate (Near End)	84	-14.1	-40.32	2-1/2" Plate
Vertical Column Web Plate (Near Top)	122	-4.6	-40.32	3" Plate

Notes:

(1) Loading = Steady state normal operating thermal load + 6370 kip horizontal mechanical load

(2) Steady state maximum stresses for the LOCA condition defined in (1) for elements shown in tables from computer runs NNCG 097 and NNRFOJR.

(3) fa = Computed horizontal axial stress, (-) indicates compression
Fa = Allowable axial stress

TABLE 3.8-16

COMPARISON OF MAXIMUM DESIGN AND ALLOWABLE STRESSES FOR REACTOR COOLANT PUMP SUPPORTS

		MAX. STRESSES (KSI)		
	MEMBER	CALCULATED STRESS	ALLOWABLE	REMARKS
Lower Support	10" ϕ Pipe Post	17.17[(b)]	44.12	Fig. 3.8-43 Fig. 3.8-47 Point A
Lower Support	W12 x 40 Post	10.3[(b)]	48.0	Fig. 3.8-47 Point B
Lower Support	W12 x 53 Beam	34.01[(b)]	48.0	Fig. 3.8-47 Point C
Lower Support	1-1/4" ϕ A354 Gr BD Anchor Bolt	31.53[(a)]	86.4	Fig. 3.8-47 Point D
Lower Support	2" Base Plate	32.3[(b)]	40.32	Fig. 3.8-47 Point E
Lower Support	1" Shear Lug	31.32[(b)]	44.16	Fig. 3.8-47 Point F
Snubber Support	W8 x 31	17.2[(a)]	40.9	Fig. 3.8-44 Point A
Snubber Support	1½" Anchor Plate	8.7[(b)]	43.2	Fig. 3.8-44 Point B
Snubber Support	1” ϕ A325 Anchor Bolt	49.7[(a)]	64.0	Fig. 3.8-44 Point C

Notes:

(a) Axial Stress
(b) Bending Stress

TABLE 3.8-17

PRESSURIZER SUPPORT STRESS COMPARISON

See Figure 3.8-52

MEMBER	MAX. STRESSES (KSI) CALCULATED STRESS	ALLOWABLE	REMARKS
Column	22.14[(a)]	41.4	
Base Plate of Column	23.03[(b)]	40.3	
2" ϕ A354 Gr BD Anchor Bolt for Column	23.87[(a)]	64.0	
Anchor Plate for 2"ϕ Anchor Bolt	29.96[(b)]	44.16	
Embedded Plate	28.56[(b)]	44.16	
Shear Lug	17.63[(b)]	40.32	Point A
Plate of Ring Girder	28.56[(b)]	44.16	
Stiffener Plate	32.4[(C)]	48.0	Point B

Note:
(a) axial stress
(b) bending stress
(c) principal stress

TABLE 3.8-18

REACTOR COOLANT PUMP STOPS & WIRE ROPE RESTRAINT

MEMBER		MAX. STRESSES (KSI) CALCULATED	ALLOWABLE	REMARKS
Lower Stop	3-1/2" Top Plate	15.31[d]	23.28	Fig. 3.8-47 Point G
Lower Stop	1-1/2" Support Plate	29.22[C]	40.32	Fig. 3.8-47 Point H
Lower Stop	W14 x 730 Support Member	29.53[a]	30.56	Fig. 3.8-48 Point A
Lower Stop	4" Base Plate	33.6[b]	40.32	Fig. 3.8-48 Point B
Lower Stop	2-1/2" ϕ A490 Anchor Bolt	79.74[a]	86.4	Fig. 3.8-48 Point
Upper Stop	Stop Beam	39.6[b]	40.32	Fig. 3.8-45 Point A
Upper Stop	Bracing WT12 x 58.5	12.73[a]	41.5	Fig. 3.8-46 Point B
Upper Stop	Beam Support	9.3[a]	36.9	Fig. 3.8-46 Point C
Upper Stop	2" Base Plate	35.7[b]	40.32	Fig. 3.8-46 Point D

Notes:

(a) Axial Stress
(b) Bending Stress
(c) Principal Stress
(d) Shear Stress

1) Newly calculated stresses are bounded by the stresses shown in Table 3.8-18

TABLE 3.8-19

CONCRETE INTERNAL STRUCTURES
COMPARISON OF REQUIRED DESIGN STRENGTH AND ACTUAL CAPACITY
OF STRUCTURAL ELEMENTS

STRUCTURAL ELEMENT		GOVERNING LOADING CONDITION (1)	CALCULATED DESIGN VALUE (REQUIRED STRENGTH)(2)		ULTIMATE CAPACITY(2)		
Secondary Shield Wall	Vert	5	Axial Force	= 142 K/Ft	Axial Force,	N_u	= 163 K/Ft
			Moment	= 433 Ft-K/Ft	Moment,	M_u	= 498 Ft-K/F
			In-Plane Shear	= 65 K/Ft	In-Plane Shear,	V_{xy}	= 138 K/Ft
			Shear	= 147 K/Ft	Shear	V_u	= 209 K/Ft
Secondary Shield Wall	Horiz	5	Axial Force	= 300 K/Ft	Axial Force,	N_u	= 330 K/Ft
			Moment	= 354 Ft-K/Ft	Moment,	M_u	= 390 Ft-K/F
			In-Plane Shear	= 65 K/Ft	In-Plane Shear,	V_{xy}	= 174 K/Ft
			Shear	= 174 K/Ft	Shear	V_u	= 209 K/Ft
Refueling Cavity Wall	Vert	4	Axial Force	= 22 K/Ft	Axial Force,	N_u	= 27 K/Ft
			Moment	= 82 Ft-K/Ft	Moment,	M_u	= 98 Ft-K/Ft
			In-Plane Shear	= 18 K/Ft	In-Plane Shear,	V_{xy}	= 150 K/Ft
			Shear	= 61 K/Ft	Shear	V_u	= 170 K/Ft
	Horiz	4	Axial Force	= -0.9 K/Ft	Axial Force,	N_u	= -20 K/Ft
			Moment	= 155 Ft-K/Ft	Moment	M_u	= 250 Ft-K/F
			In-Plane Shear	= 62 K/Ft	In-Plane Shear,	V_{xy}	= 228 K/Ft
			Shear	= 84 K/Ft	Shear	V_u	= 170 K/Ft
Primary Shield Wall @	Vert	4	Moment	= 17038 Ft-K	Moment		= 17610 Ft-K
Reactor Steel Supports	Horiz	4	Axial Force	= 921 K	Axial Force		= 1095 K

Notes:

(1) See Table 3.8-11 for Load Combination
(2) For Axial Force + = Tension, - = Compression

TABLE 3.8-20

NSSS SUPPORT STEEL MATERIAL SUMMARY

Reactor Vessel	
Support structure	A441, A533 Class 2 - Grade B
Bolts	A325
Shims	A533 Class 2 - Grade B
Support Shim plate	A240 Type 410
Sliding support	See Table 5.2-3
Steam Generator	
Sliding base support	See Table 5.2-3
Shims (bet. sliding base & skirt of steam generator)	SA-240 Type 304
Upper support: structural	A572, A533 Class 2 - Grade B
bolts	A325, A320 Gr L43, A490
Reactor Coolant Pump Snubber Supports	
Snubber support	A441
Anchor bolts	A325
Reactor Coolant Pump Stops, Lower and Upper Supports	
Forgings	A350-Gr IF-1
Steel support assembly	A441
Anchor bolts	A354 Grade BD
Reactor Coolant Piping	
Cold leg stops	A441 & A508 Class 2
Reactor Coolant Pump Wire Rope	
Wire rope	A603
Pins and nuts	A193 Gr B7 for pins A194 Gr 7 for nuts

TABLE 3.8-21

LIVE LOADS

Building	Specific Area	Load in Psf
Shield Building	Top Surface of dome	30 (for the horizontal plan projection)
Reactor [1] Auxiliary Building	Roof	30
	Roof at el. 43 ft	200
	Floor at el. 43 ft	200
	Floor at el. 19.5 ft	
	Equipment area	200
	Drumming area	1000
	Electrical and Piping penetration area	200
	All other areas	100
	Floor at el. 0.5 ft	200
	Floor at el. 10 ft	200
	Piping and Cable trays hung from all floors and roof	50
	Floor in pipe tunnel and cable ways	200
Fuel Handling Building[1,2]	Roof at el. 96.5 ft	30
	Roof at el. 62 ft	30
	Spent fuel pool floor at el. 62 ft	100
	Floor at el. 48 ft	150
	New fuel storage area	1200 (in lieu of actual weight of fuel elements)
	Floor at el. 19.5 ft	100
	Floor for truck access to new fuel storage	H20 or 100
	Spent fuel pool floor at el. 21.5 ft	1200 (In lieu of actual weight of fuel elements)
	Piping and Cable trays from all floors and roof	50
Diesel Generator Building	Roof at el. 49.5 ft	30
	Floor at el. 22.67 ft	200
Missile Protection Structures-Diesel Oil Storage Tank	Roof at el. 63 ft	30
	floor at el 19 ft	100

TABLE 3.8-21 (Cont'd)

Building	Specific Area	Load in Psf
Missile Protection Structure-component Cooling Area	Roof at el. 49.5 ft Platform at el. 23.5 ft	30 100
Missile Protection Structure-Condensate Storage Tank	Roof	30
All Areas[(3)]	Miscellaneous Platform stairs and walkways	100

Notes:

(1) All floors and roof are designed for 8000 lb concentrated load at any one point

(2) Roof loads from crane runway column are as applicable

(3) All areas of the Main Steam Trestles and the Intake Structure are designed for a uniform load of 100 psf.

TABLE 3.8-22

MAIN STEAM TRESTLE STRESS COMPARISON

MEMBER	MAX STRESSES (KSI) CALCULATE[1]	ALLOWABLE[2]	LOAD COMBINATION	REMARKS
Flange of Column TB	32.07[b]	44.16	DL+LL+Piping Load +SSE+Pipe Rupture	Top of Column TB-2T6
Web of Column TB	8.06[d]	27.71	DL+LL+Piping Load +SSE+Pipe Rupture	Bottom Column TB-2T6
Frame "A" Flange of Girder	12.59[b]	44.16	DL+LL+Piping Load +SSE+Pipe Rupture	End of Girder
Frame "A" Web of Girder	2.76[d]	27.71	DL+LL+Piping Load +SSE+Pipe Rupture	End of Girder TA-2T4
Flange of Column TA	31.51[b]	40.32	DL+LL+Piping Load +SSE+Pipe Rupture	Top of Column TA-2T4
Web of Column TA	9.07[d]	25.50	DL+LL+Piping Load +SSE+Pipe Rupture	Bottom Column
Frame "B" Flange of Girder	12.71[b]	44.16	DL+LL+Piping Load +SSE+Pipe Rupture	End of Girder
Frame "B" Web of Girder	5.51[d]	25.50	DL+LL+Piping Load +SSE+Pipe Rupture	End of Girder
G5 Girder (W 24 x 145)	27.8[b]	48.00	DL+LL+Piping Load +SSE+Pipe Rupture	Center of Girder
Silencer Support Beam at El 62 ft. W 12 x 27 2L 3 x 2 x 1/4	36.48[b]	48.00	DL+LL+Piping Load +Missile Load+ Tornado Load	Near Center of Beam
Horizontal Frame @ El 36 ft. Member W 14 x 84	32.4[a]	48.00	DL+LL+Piping Load +SSE+Pipe Rupture	End & Center of Beam
Post (W 14 x 119) Above El 41 ft.	35.32[c]	40.42	DL+LL+Piping Load +Missile Load+ Tornado Load	Middle of Column
Flued Head (FW) Bottom Flange of G2	35.24[c]	44.16	DL+LL+Piping Load +SSE+Pipe Rupture	End of Girder Element 401

Notes:
1. (a) axial stress
   (b) bending stress
   (c) principal stress
   (d) shear stress
2. ASTM A-441 Material

TABLE 3.8-23

COMPARISON OF REQUIRED DESIGN STRENGTH AND ACTUAL CAPACITY OF STRUCTURAL ELEMENTS

STRUCTURAL ELEMENT			GOVERNING LOADING[a] CONDITION	CALCULATED DESIGN VALUE[b][c] (REQUIRED STRENGTH)		ULTIMATE CAPACITY[b][c]	
Reactor Building Mat (R - 45'-8" to R = 55'-8")		Radial	2	Shear Axial load Moment	= 59 = -258 2484	Shear Axial Load Moment	= 180 =-312 3006
		Tangential	2	Shear Axial load Moment	= 45 = 381 =1917	Shear Axial Load Moment	= 77 = 384 = 1931
Reactor Building Cylinder Wall	BELOW EL 80.0 (REGION III STA 99)	Vertical (Meridional)	2	Shear Axial load Moment	=20.6 =-61 =399	Shear Axial Load Moment	= 42.5 =-73 = 479
		Horizontal (Hoop)	2	Shear Axial load Moment	=0.2 =76 =253	Shear Axial Load Moment	= 21.8 = 97 = 323
	ABOVE EL 80.0 (REGION II STA 1)	Vertical (Meridional)	2	Shear Axial load Moment	=11.4 =-19.6 =263.1	Shear Axial Load Moment	= 40.4 = -23.5 = 315.7
		Horizontal (Hoop)	2	Shear Axial load Moment	=0.2 =93.6 =230.1	Shear Axial Load Moment	= 21.7 = 97.6 = 240
Reactor Building Dome (Station 42)		Meridional	5	Axial load Moment	=24.3 =119.2	Axial Load Moment	= 27.7 = 136.1
		Circumferential	5	Axial load Moment	=43.0 =116.5	Axial Load Moment	= 51.6 = 139.8
Reactor Auxiliary Building El. 19.5, Girder 1G1			2	Shear Moment	=290 K =1683 Ft-K	Shear Moment	= 438 K = 2824 Ft-K
Fuel Handling Building Floor @ El. 32.0, Typ. Girder			2	Shear Moment	= 72 K = 299 Ft-K	Shear Moment	= 80 K = 465 Ft-K

Note:
a) See Subsection 3.8.4.3.2 for Load Combination.
b) Shear and Axial Load is in K/Ft except where indicated otherwise. Moment is in the Ft-K/Ft- except where indicated otherwise.
c) Positive Force = Tension.

TABLE 3.8-24

PARTIAL LINEUP AGAINST SRP 3.8.4 (11/75)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
Load Combinations for Concrete Structures			
For concrete structures, the load combinations are acceptable if found in accordance with the following:			
a- for service load conditions, either the working stress design (WSD) method or the strength design method may be used.	a- The strength design method was used		
i- If the WSD method is used, the following load combinations should be considered: (1) D + L (2) D + L + E (3) D + L + W If thermal stresses due to T_o and R_o, are present, the following combinations should be considered: (1a) $D + L + T_o + R_o$ (2a) $D + L + T_o + R_o + E$ (3a) $D + L + T_o + R_o + W$ Both cases of L having its full value or being completely absent should be checked.	i- Not applicable		

TABLE 3.8-24 (Cont'd)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
ii- If the strength design method is used, the following load combinations should be considered: (1) 1.4 D + 1.7 L (2) 1.4 D + 1.7 L + 1.9 E (3) 1.4 D + 1.7 L + 1.7 W If thermal stresses due to T_o and R_o are present, the following combinations should also be considered: (1b) (0.75) (1.4 D + 1.7 L + 1.7 T_o + 1.7 R_o) (2b) (0.75) (1.4 D + 1.7 L + 1.9 E + 1.7 T_o + 1.7 R_o) (3b) (0.75) (1.4 D + 1.7 L +1.7 W + 1.7 T_o + 1.7 R_o) Both cases of L having its full value or being completely absent should be checked. In addition the following combination should be considered: (2b') 1.2 D + 1.9 E (3b') 1.2 D + 1.7 W	ii - St. Lucie Unit 2 design complies with the load combinations listed, with the exception of load combinations (1b), (2b) and (3b).	ii - The alternate load combinations used are: (1b) 1.4 (B + D) + 1.3 (R_o + T_o) + 1.7 (L + H) (2b) 1.4 (B + D) + 1.3 (R_o + T_o) + 1.7 (L + H') + 1.9 E (3b) 1.4 (B' + D) + 1.3 (R_o + T_o) + 1.7 (L + H) + W) where B = Buoyancy at normal groundwater level B' = Buoyancy at maximum groundwater level resulting from a PMH H = Lateral earth loads under normal conditions H' = Lateral earth loads under normal and earthquake conditions	Since the acceptance criteria load combinations have a multiplication factor of 0.75, the combined loads used as identified in alternate compliance for the St. Lucie Unit 2 design would be greater for all design cases. The load combinations used on St. Lucie Unit 2 were based upon guidance provided in AEC letter to FP&L Co., August 30, 1973, "Enclosure 2 - Structural Design Criteria for Category I Structures Outside the Containment, " in addition to those given in the ACI 318-71 Code.
Where soil and hydrostatic pressures are present, in addition to all the above combinations where they have been included in L and D respectively, the requirements of Sections 9.3.4 and 9.3.5 of ACI-318-71 (Ref 1) should also be satisfied.	Soil and hydrostatic pressures are included in the design and the requirements of ACI-318-71 Sections 9.3.4 and 9.3.5 were considered.		

TABLE 3.8-24 (Cont'd)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
b - For factored load conditions, which represent extreme environmental abnormal, abnormal/severe environmental and abnormal/ extreme environmental conditions, the strength design method should be used and the following load combinations should be considered. (4) $D + L + T_o + R_o + E$ (5) $D + L + T_o + R_o + W_t$ (6) $D + L + T_a + R_a + 1.5\ P_a$ (7) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$ (8) $D + L + T_a + R_a + 1.0\ P_a = 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$ In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_j, Y_r and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8) and the corresponding structural acceptance criteria of Section II.5 of this plan should be satisfied first without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). When considering these concentrated loads, local section strength capacities may be exceeded provided there will be no loss of function of any safety-related system. Both cases of L having its full value or being completely absent should be checked.	b - St. Lucie Unit 2 design complies with the load combinations listed.		

TABLE 3.8-25

RG 1.142 (R0) SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS

(OTHER THAN REACTOR VESSELS AND CONTAINMENT)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
The procedures and requirements described in ACI Standard 349-76, "Code Requirements for Nuclear Safety Related Concrete Structures," are generally acceptable to the NRC staff and provide an adequate basis for complying with the Commission's regulations with regard to the design of safety-related concrete structures other than reactor vessels and containments, subject to the following:		The design and analysis procedures utilized for safety-related concrete structures are in accordance with the ACI 318-71 Code.	Design and analysis of St. Lucie Unit 2 started before ACI 349-76 was issued.
1. The applicability of strength design methods to structures whose principal function is to provide a barrier to contain or retain pressure such as the divider barrier of the ice-condenser of the PWR containment is questionable. Therefore, for those structures, mere conformance with the requirements of ACI 349-76 is unacceptable to the staff, who will continue to review the design of these structures on a case-by- case basis.			1. Not applicable to St. Lucie- Unit 2.
2. When concrete structures are used to provide radiation shielding, the provisions of Sections 5.1 and 10 of ANSI Standard N101.6-1972,[2] "Concrete Radiation Shields," and those of ANSI Standard N101.4-1972,[3] as endorsed by Regulatory Guide 1.54,"Quality Assurance Requirements for Protective Coatings Applied to Water- Cooled Nuclear Power Plants," are applicable.			2. Refer to Table 3.8-26 for compliance to Sections 5.1 and 10 of ANSI Standard N101.6-1972

TABLE 3.8-25 (Cont'd)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
3. ACI Standard 349-76 lacks specific requirements to ensure ductility of framed structures. Adherence to the requirements of Appendix A to ACI Standard 318-71 is acceptable			3. Appendix A of ACI 318 "Special Provisions for Seismic Design" is applicable when seismic loads are based on empirical formulate such as those of the Unified Building Code. For the category I structures, seismic loads are obtained from dynamic analysis of the structures based on SSE and OBE design response spectra. Shear walls and bracing systems are designed to take the seismic forces calculated from each analysis. For these reasons FP&L feels that the requirements of Appendix A of ACI 318 are not applicable to the nuclear plant structures whose design is based on conservative criteria and detailed seismic analysis
4. Section 5.1.2 permits depositing concrete without the prior removal of water from the place of deposit at the discretion of the owner. Since the presence of water in the place of deposit may seriously affect the strength properties of concrete, it is important that water be removed before concrete is deposited unless a tremie is used	4. Water is removed before concrete is deposited		
5. Section 5.4.1 allows concrete that has partially hardened or has been contaminated with foreign materials or remixed after initial set to be reused at the discretion of the engineer. Such a material would be defective and therefore should not be used.	5. The placement of partially hardened, Contaminated or Retempered concrete Is not permitted.		

TABLE 3.8-25 (Cont'd)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
6. In addition to the requirements of Section 1.3.1 of ACI Standard 349-76, the inspectors should have sufficient experience in reinforced and prestressed concrete practice to interpret plans and specifications. The inspectors should be thoroughly familiar with the applicable ACI and ASTM Standards. ACI Standard 311-74,[1] "Recommended Practice for Concrete Inspection," should be followed except where the requirements of Section 1.5 of ACI Standard 349-76 control.			6. Compliance based on site inspection practices.
7. The frequency of cylinder testing required by Section 4.3.1 of ACI Standard 349-76 is not consistent with generally accepted practice. A test frequency in conformance with ANSI Standard N45.2.5-1974,[7] as endorsed by Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," is acceptable.	ANSI N45.2.5-1974 concrete cylinder testing frequency was followed on St. Lucie Unit 2.		7. ASME NQA-1-1994, Subpart 2.5 was substituted for ANSI N45.2.5 as described in the FPL Quality Assurance Topical Report discussed in Section 17.2.
8. The minimum pressure-testing requirements for embedded piping of ACI Standard 318-71 have been deleted from ACI Standard 349-76. In order to ensure that minimum pressure-testing requirements are met, the pressure tests of embedded pipes in Section 6.3.2.4 of ACI 349-76 should also satisfy the requirements of Subsection 6.3.2.4 of ACI 318-71.			8. Not applicable. ACI Standard 318-71, Section 6.3.2.4, indicates that piping, with the exception of Section 6.3.2.5, is to be tested prior to concreting. Section 6.3.2.5 is as follows: "Drain pipes and other piping designed for pressures of not more than 1 psi above atmospheric pressure need not be tested as required in Section 6.3.2.4."

TABLE 3.8-25 (Cont'd)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
9. More conservative load factors are appropriate in accounting for the effects of normal or shutdown thermal loads, postulated pipe break accidents, and an operating basis earthquake (OBE) in combination with a postulated pipe break. The load factors used in Section 9.3.1 of ACI Standard 349-76 are acceptable to the staff except for the following: a. In load combinations (9), (10), and (11), 1.7 T_o should be used in place of 1.4 T_o. b. In load combination (6), 1.5 P_a should be used in place of 1.25 P_a. c. In load combination (7), 1.25 P_a and 1.25 E_o should be used in place of 1.15 P_a and 1.15 E_o, respectively. d. In load combinations (2) and (10), 1.9 E_o should be used in place of 1.7 E_o.			9. Load factors utilized are presented in SRP 3.8.4 line-up. The noted load factor changes make the load combination consistent with those presented in SRP 3.8.4.
10. Structures must be able to withstand the effects of differential settlement under environmental loads as well as under abnormal loads. Thus, in Section 9.3.2 of ACI 349-76, consideration of the effects of differential settlement should be included in load combinations (1) through (11).			10. Not applicable since each safety-related concrete structure is supported on an individual mat. Differential settlement within a building was not expected to occur and was not included as a design consideration.
11. The provisions of Section 9.3.3 of ACI Standard 349-76 to account for the effects of transitory loads are not sufficiently general. Thus, in Section 9.3.3 of ACI Standard 349-76, when any load reduces the effects of other loads, the corresponding coefficient for that load should be taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the coefficient for that load should be taken as zero.		11. Load combinations used in design either useful dead and live loads or full dead and zero live loads.	11. The load combinations used on St. Lucie Unit 2 were based upon guidance provided in AEC letter to FP&L Co., 8-30-73, "Enclosure 2-Structural Design Criteria for Category I Structures Outside the Containment," in addition to those given in the ACI 318-71 Code.

TABLE 3.8-25 (Cont'd)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
			(Continued) 11. Exception is taken to the regulatory position which requires that a coefficient of 0.9 or zero be applied to any load which reduces the effects of other loads. Considering live load as having its full value or being completely absent satisfies the requirement for setting a transitory load to zero. However, applying the 0.9 coefficient to all other such mitigating loads, which are always present or occur simultaneously, would increase the number of load combinations to an impractical level with no demonstrated or meaningful increase in the overall conservatism of the governing load combinations. Our position is consistent with ACI 318-77 and ACI 349-76.
12 The provision in Section 9.3.6 of ACI Standard 349-76 permitting local exceedance of section strength under concentrated dynamic loads does not ensure that the section can withstand associated distributed loadings. Thus, if the provision of Section 9.3.6 of ACI 349-76 permitting exceedance of local section strengths is invoked, it should be demonstrated that section strengths are adequate to accommodate load combinations (7) and (8) without the dynamic loads Y_j, Y_m, and Y_r.			12. Design criteria used on St. Lucie Unit 2 does not permit local exceedance of section strength

TABLE 3.8-25 (Cont'd)

ACCEPTANCE CRITERIA	COMPLIANCE	ALTERNATE COMPLIANCE	REMARKS
13. The NRC staff would accept the local exceedance of section strength for concentrated tornado-generated-missile loading under load combination (5). However, an analysis should be performed to demonstrate that section strengths are adequate to accommodate load combination (5) without the dynamic load effect of tornado-generated missiles.			13. Same as 12 above
14. ACI Standard 349-76 does not address the subject of openings in slabs and footings. Provisions of Section 11.12 of ACI 318-71are acceptable for this purpose.	14. Provisions of Section 11-12 of ACI 318-71 are followed		

TABLE 3.8-26

COMPLIANCE TO SECTIONS 5.1 AND 10 OF ANSI STANDARD N101.6-1972

ANSI N101.6-72 Section	Clarifications
5.1.2	No high density concrete is used.
5.1.3	No hydrous aggregate is used.
5.1.4	No boron containing aggregates are used.
5.1.6	Coatings of clay, silt, gypsum, calcite or caliche on coarse aggregate total no more than three and one half percent of the total weight of the aggregate. Radiation attenuation calculations take this into account.
10.1.2	Dimensional tolerances for hatches and openings as specified in ACI-347 are used rather than those given in Table 1 of ANSI N101.6-72. Minimum practicable joint clearances are specified.
10.1.3	Service trenches are not used.
10.2.2	The weight of each block is indicated on the design drawing, not marked on the block.
10.2.3	Blocks are cured according to good construction practice, e.g., use of wet burlap or curing compound, but not necessarily in the absence of direct sunlight or heat. This sunlight or heat, however, does not result in the loss of shielding efficiency.
10.3.1	There are no present plans for penetrations through shielding plugs. However, if they are required, streaming is prevented by proper design of the penetration.
10.4	No movable or removable poured walls are used.
10.6	Precast shielding components are fabricated at the site.

TABLE 3.8-27

REACTOR BUILDING

LOAD COMBINATION	FACTOR OF SAFETY AGAINST SLIDING	OVERTURNING	FLOATATION
D + H + E	>1.5	> 2.69	-
D + H + W	>8.7	> 21.7	-
D + H + E^1	1.23	2.69	-
D + H + W_t	8.7	21.7	-
D + F^1	-	-	3.12

D =Dead Loads

E = OBE

E^1 = DBE

W = Hurricane Wind @ 194 mph

W_t = Tornado Wind @ 300 mph

F^1 =Buoyancy, Max GWT EL + 21.00

H = Soil Pressure

TABLE 3.8-28

REACTOR AUXILIARY BUILDING

LOAD COMBINATION	FACTOR OF SAFETY AGAINST SLIDING	OVERTURNING	FLOATATION
D + H + E	1.35	2.64	-
D + H + W	3.88	4.14	-
$D + H + E^1$	1.16	2.15	-
$D + H + W_t$	2.84	3.15	-
$D + F^1$	-	-	2.35

D = Dead Loads

E = OBE

E^1 = DBE

W = Hurricane Wind @ 194 mph

W_t = Tornado Wind @ 300 mph

F^1 = Buoyancy, Max GWT EL + 17.00

H = Soil Pressure

TABLE 3.8-29

CONDENSATE STORAGE TANK

LOAD COMBINATION	FACTOR OF SAFETY AGAINST SLIDING	OVERTURNING	FLOATATION
D + H + E	>2.24	> 2.54	-
D + H + W	>4.71	> 3.93	-
D + H + E^1	2.24	2.54	-
D + H + W_t	4.71	3.93	-
D + F^1	-	-	5.90

D = Dead Loads

E = OBE

E^1 = DBE

W = Hurricane Wind @ 194 mph

W_t = Tornado Wind @ 360 mph

F^1 = Buoyancy, Max GWT EL + 17.00

H = Soil Pressure

Note:

Factors of safety for load combinations D+H+E and D+H+W will be higher than for D+H+E^1 and D+H+W_t respectively

TABLE 3.8-30

FUEL HANDLING BUILDING

LOAD COMBINATION	FACTOR OF SAFETY AGAINST SLIDING	OVERTURNING	FLOATATION
D + H + E	2.11	2.33	-
D + H + W	>1.50	>1.50	-
D + H + E^1	1.25	1.39	-
D + H + W_t	4.09	4.45	-
D + F^1	-	-	9.1

D = Dead Loads

E = OBE

E^1 = DBE

W = Hurricane Wind @ 194 mph

W_t = Tornado Wind @ 360 mph

F^1 = Buoyancy, Max GWT EL + 17.00

H = Soil Pressure

Note:

Dyn soil pressure > active soil pressure in calculations H is neglected and this is conservative.

TABLE 3.8-31

DIESEL GENERATOR BUILDING

LOAD COMBINATION	FACTOR OF SAFETY AGAINST SLIDING	OVERTURNING	FLOATATION
D + H + E	2.98	9.68	-
D + H + W	>7.45	>23.92	-
D + H + E^1	1.55	4.71	-
D + H + W_t	7.45	23.92	-
D + F^1	-	-	6.44

D = Dead Loads

E = OBE

E^1 = DBE

W = Hurricane Wind @ 194 mph

W_t = Tornado Wind @ 360 mph

F^1 = Buoyancy, Max GWT EL + 17.00

H = Soil Pressure

TABLE 3.8-32

COMPONENT COOLING

LOAD COMBINATION	FACTOR OF SAFETY AGAINST SLIDING	OVERTURNING	FLOATATION
D + H + E	>1.52	>4.16	-
D + H + W	>3.07	>12.68	-
D + H + E^1	1.52	4.16	-
D + H + W_t	3.07	12.68	-
D + F^1	-	-	3.11

D Loads= Dead

E = OBE

E^1 = DBE

W = Hurricane Wind @ 194 mph

W_t = Tornado Wind @ 360 mph

F^1 = Buoyancy, Max GWT EL + 17.00

H = Soil Pressure

TABLE 3.8-33

INTAKE STRUCTURE

LOAD COMBINATION	FACTOR OF SAFETY AGAINST SLIDING	OVERTURNING	FLOATATION
D + H + E	1.64	1.52	-
D + H + W	4.38	1.65	-
D + H + E^1	1.13	1.23	-
D + H + W_t	3.83	1.61	-
D + F^1	-	-	2.31

D = Dead Loads

E = OBE

E^1 = DBE

W = Hurricane Wind @ 194 mph

W_t = Tornado Wind @ 360 mph

F^1 = Buoyancy, Max GWT EL + 16.00

H = Soil Pressure

Refer to Dwg.
2998-G-793 SH 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL
SHEET 1
FIGURE 3.8-1

Refer to Dwg.
2998-G-793 SH 4

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL
SHEET 4
FIGURE 3.8-2

Refer to Drawing
2998-2788

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT FOR
2 FT. 6 X 6 FT - 8
PERSONNEL LOCK
FIGURE 3.8-3

Refer to Drawing
2998-2843

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL ARRANGEMENT 5' - 9 I.D. ESCAPE LOCK W/2' - 6 I.D. OPENING
FIGURE 3.8-4

Refer to Dwg.
2998-G-213 SH. 3

FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 2**
REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS **FIGURE 3.8-5**

Refer to Dwg.
2998-G-213 SH. 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS
FIGURE 3.8-6

Refer to Dwg.
2998-G-213 SH. 2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS
FIGURE 3.8-7

DELETED

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FIGURE 3.8-8

Refer to Dwg.
2998-G-213 SH. 4

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR CONTAINMENT BUILDING
PIPING PENETRATIONS SH. 4
FIGURE 3.8-9

Refer to Dwg.
2998-G-793 SH 2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL
SHEET 2
FIGURE 3.8-10

Refer to Dwg.
2998-G-793 SH 3

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL
SHEET 3

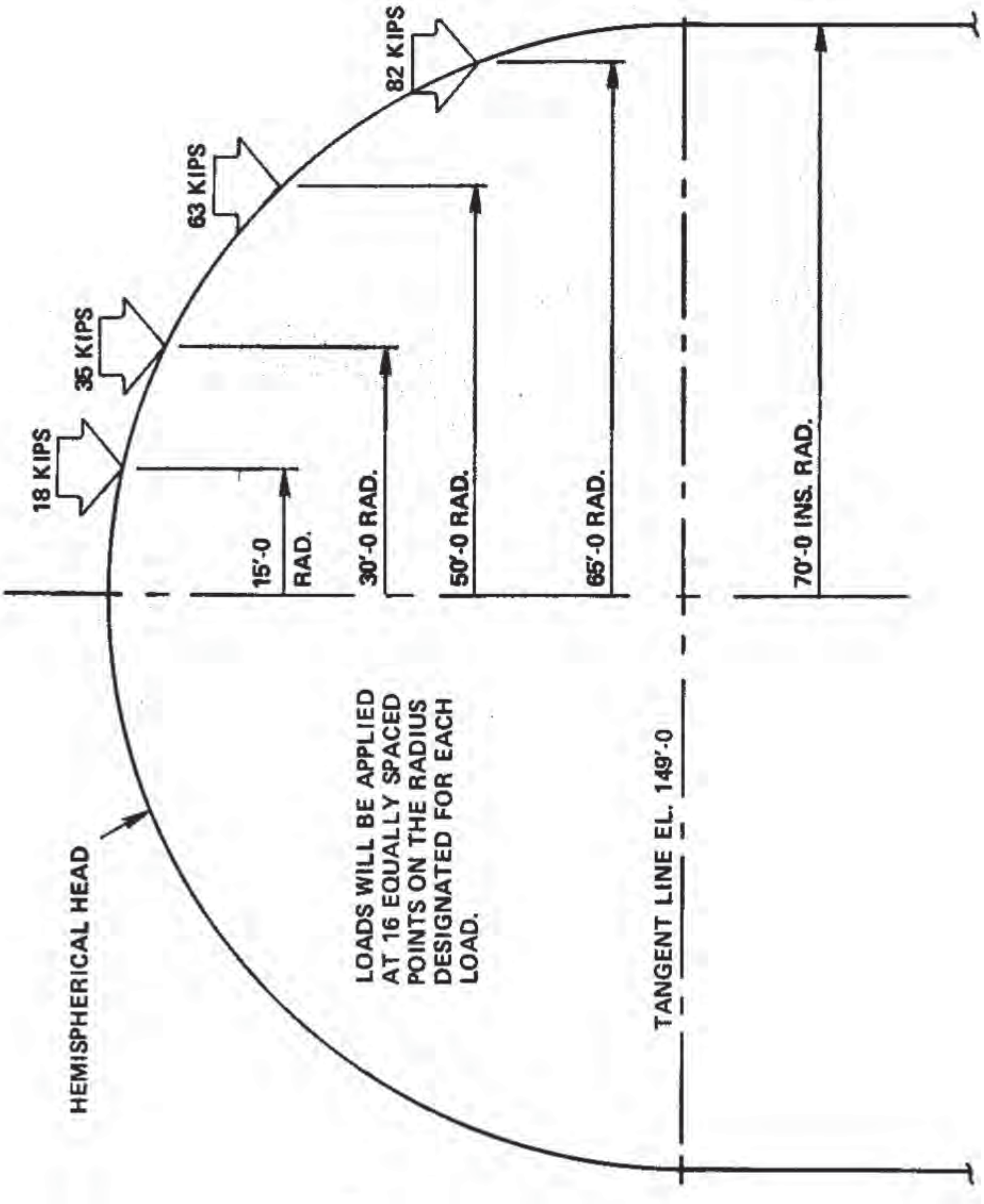
FIGURE 3.8-11

Refer to Dwg.
2998-B-337 SH 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ELECTRICAL PENETRATIONS

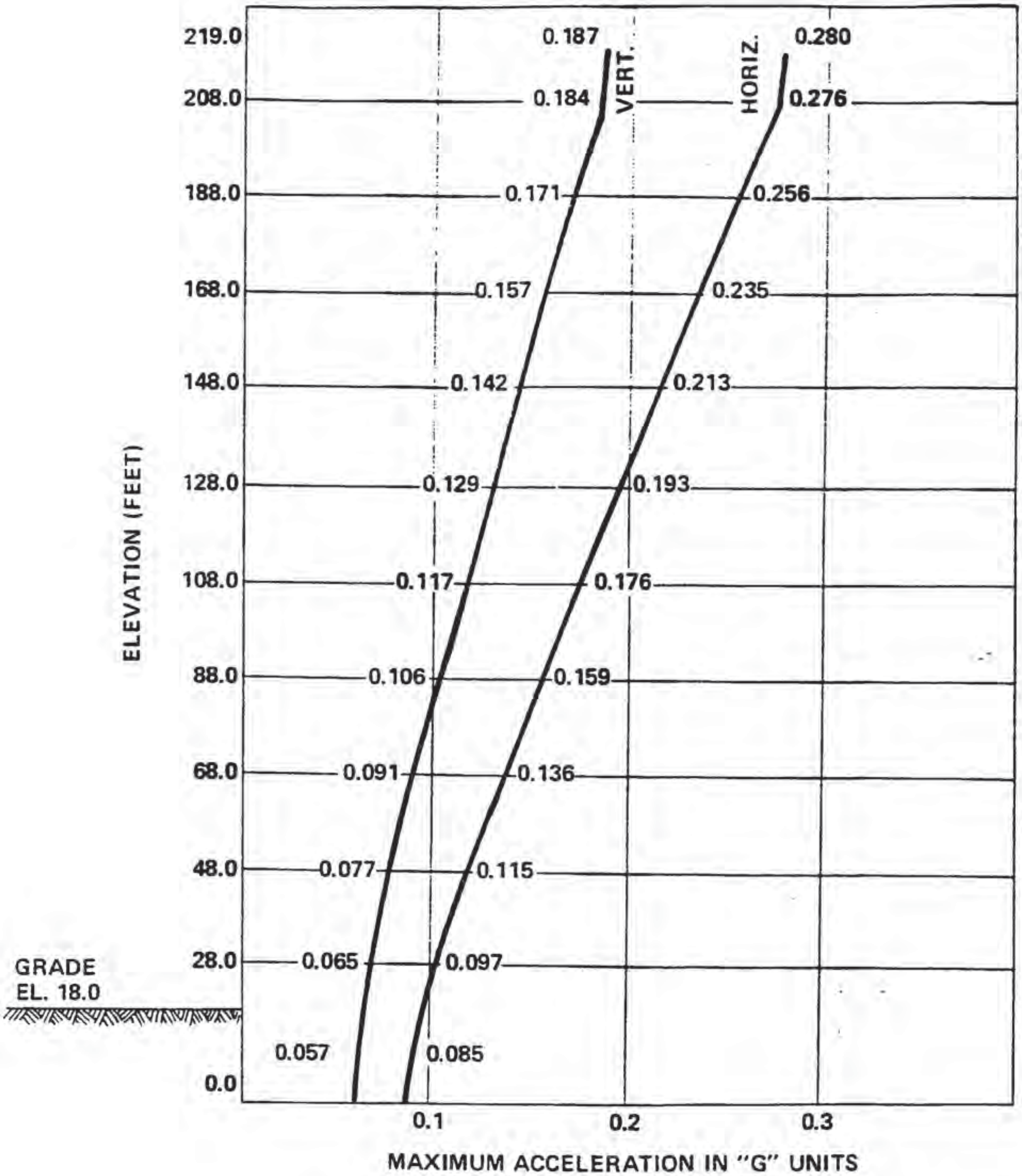
FIGURE 3.8-12



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ROOF TEMPORARY CONSTRUCTION LOADS

FIGURE 3.8-13



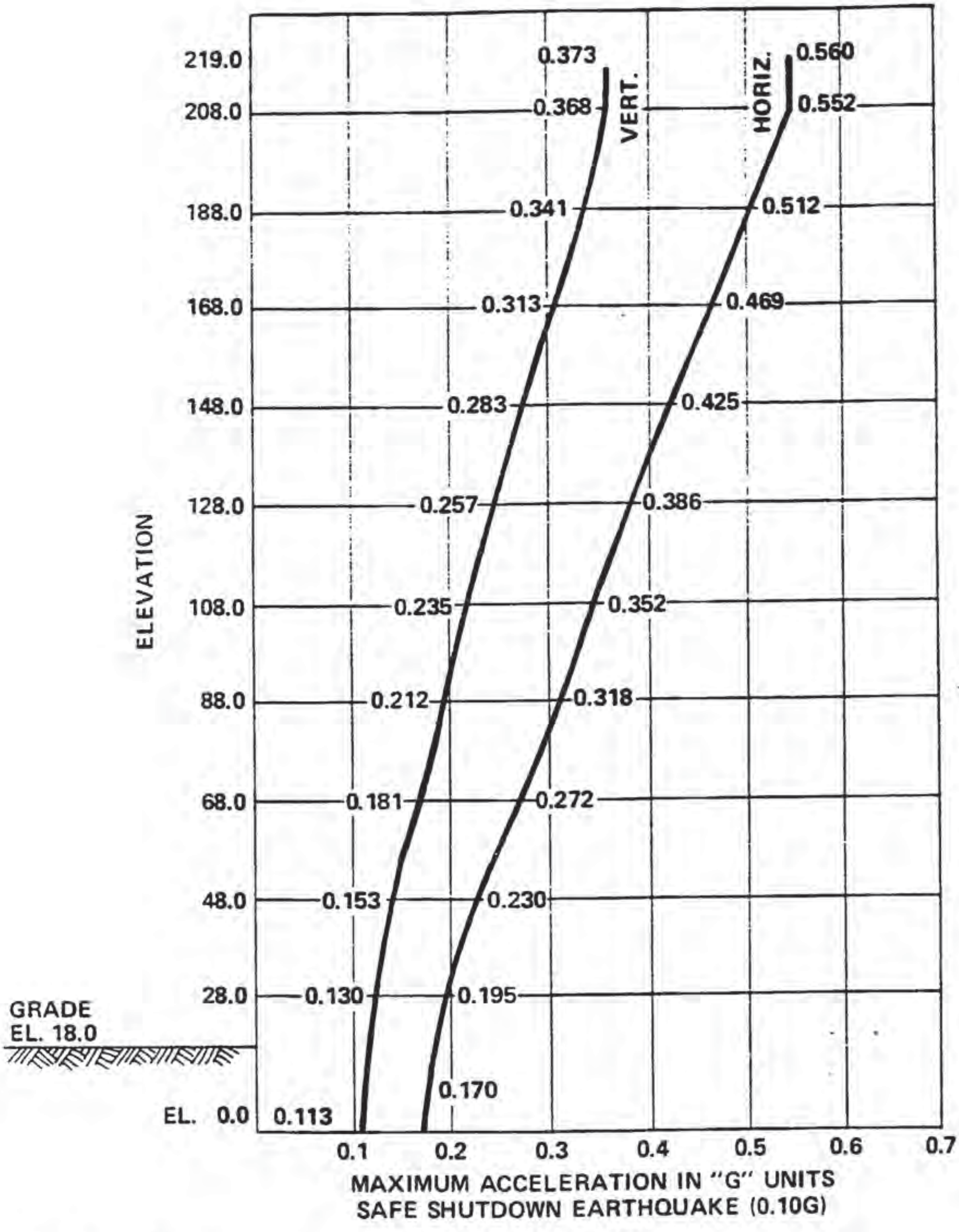
Historical Information, for Acceleration curves refer to specification FLO-2998-757.

Amendment No. 13 (05/00)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL OBE
SEISMIC LOADING

FIGURE 3.8-14



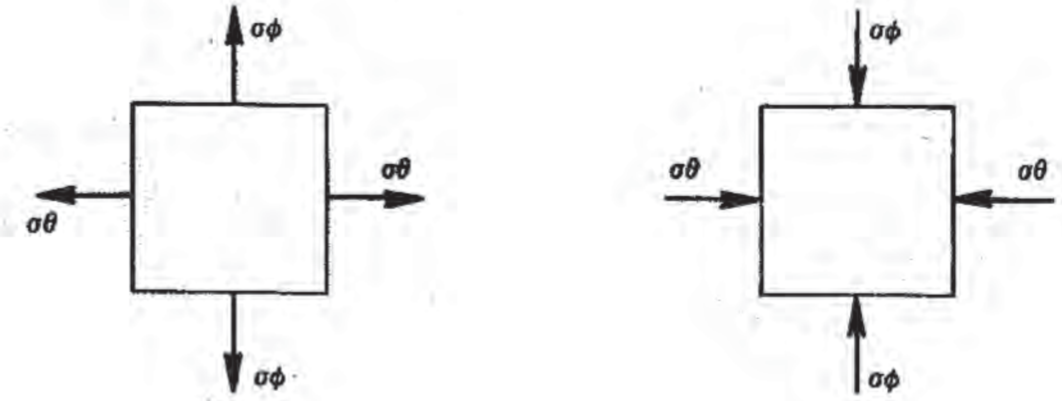
Historical Information, for Acceleration curves refer to specification FLO-2998-757.

Amendment No. 13 (05/00)

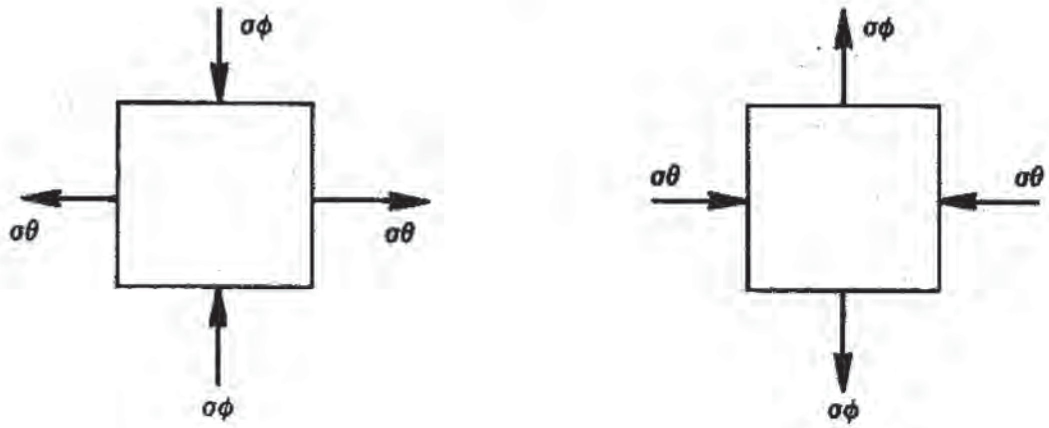
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL SSE SEISMIC LOADING
FIGURE 3.8-15

FOR $\sigma\theta$ AND $\sigma\phi$ OF LIKE SIGN:



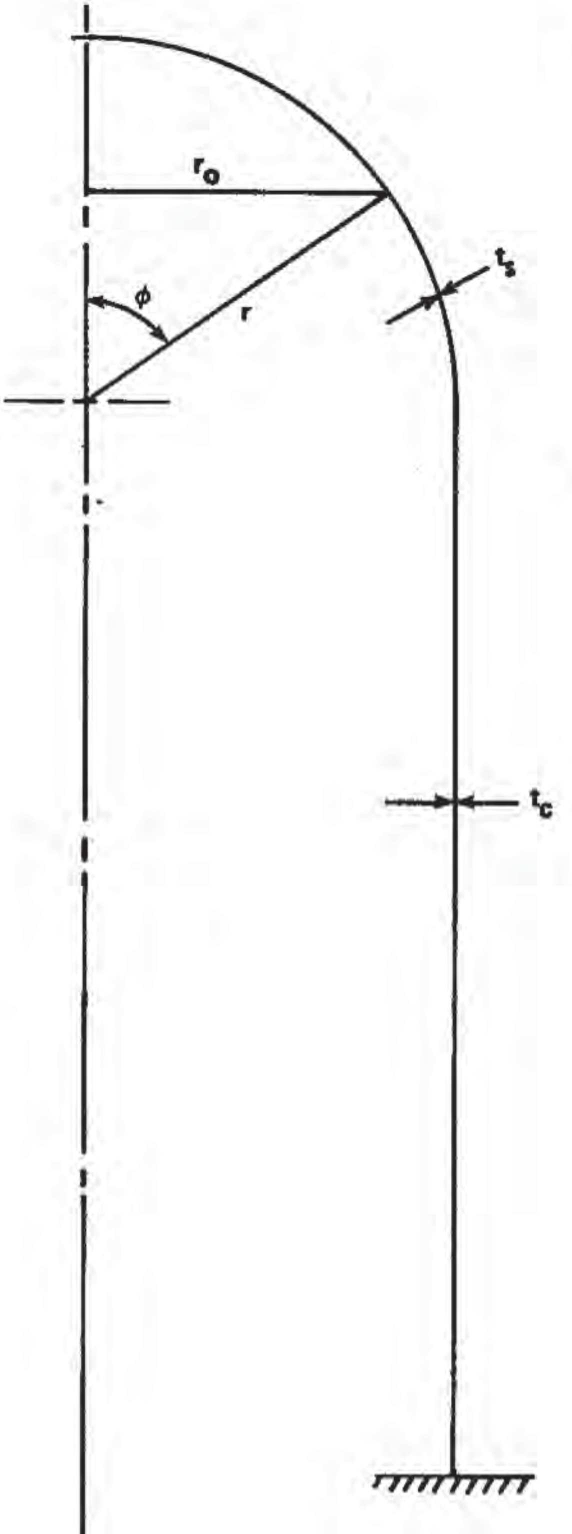
FOR $\sigma\theta$ AND $\sigma\phi$ OF UNLIKE SIGN:



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

COMBINATION OF LATITUDINAL & MERIDIONAL STRESSES FOR THE CONTAINMENT VESSEL

FIGURE 3.8-16



FOR SPHERE

$$\sigma_\phi = \frac{M}{\pi r^2 t_s \sin^3\phi}$$

$$\sigma_\theta = \sigma_\phi$$

t_s = THICKNESS OF SPHERE

M = MOMENT AT POINT OF INTEREST

FOR CYLINDER

$$\sigma_\phi = \frac{M}{\pi r^2 t_c}$$

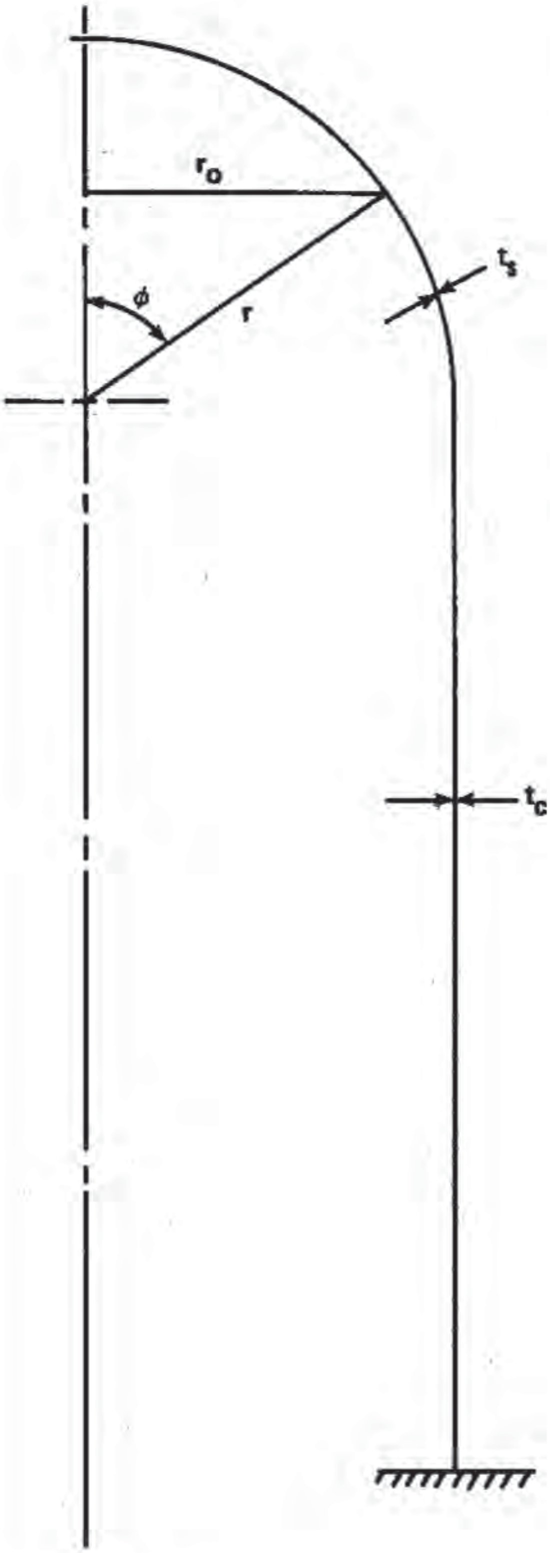
$$\sigma_\theta = 0$$

t_c = THICKNESS OF CYLINDRICAL SHELL

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL STRESSES RESULTING FROM HORIZONTAL AXIAL FORCES

FIGURE 3.8-17



FOR SPHERE

$$\sigma_\phi = \frac{-F}{2\pi r t_s \sin^2\phi}$$

$$\sigma_\theta = -\sigma_\phi$$

t_s = THICKNESS OF SPHERE

F = SUM OF VERTICAL FORCES ABOVE POINT OF INTEREST

FOR CYLINDER

$$\sigma_\phi = \frac{-F}{2\pi R t_c}$$

$$\sigma_\theta = 0$$

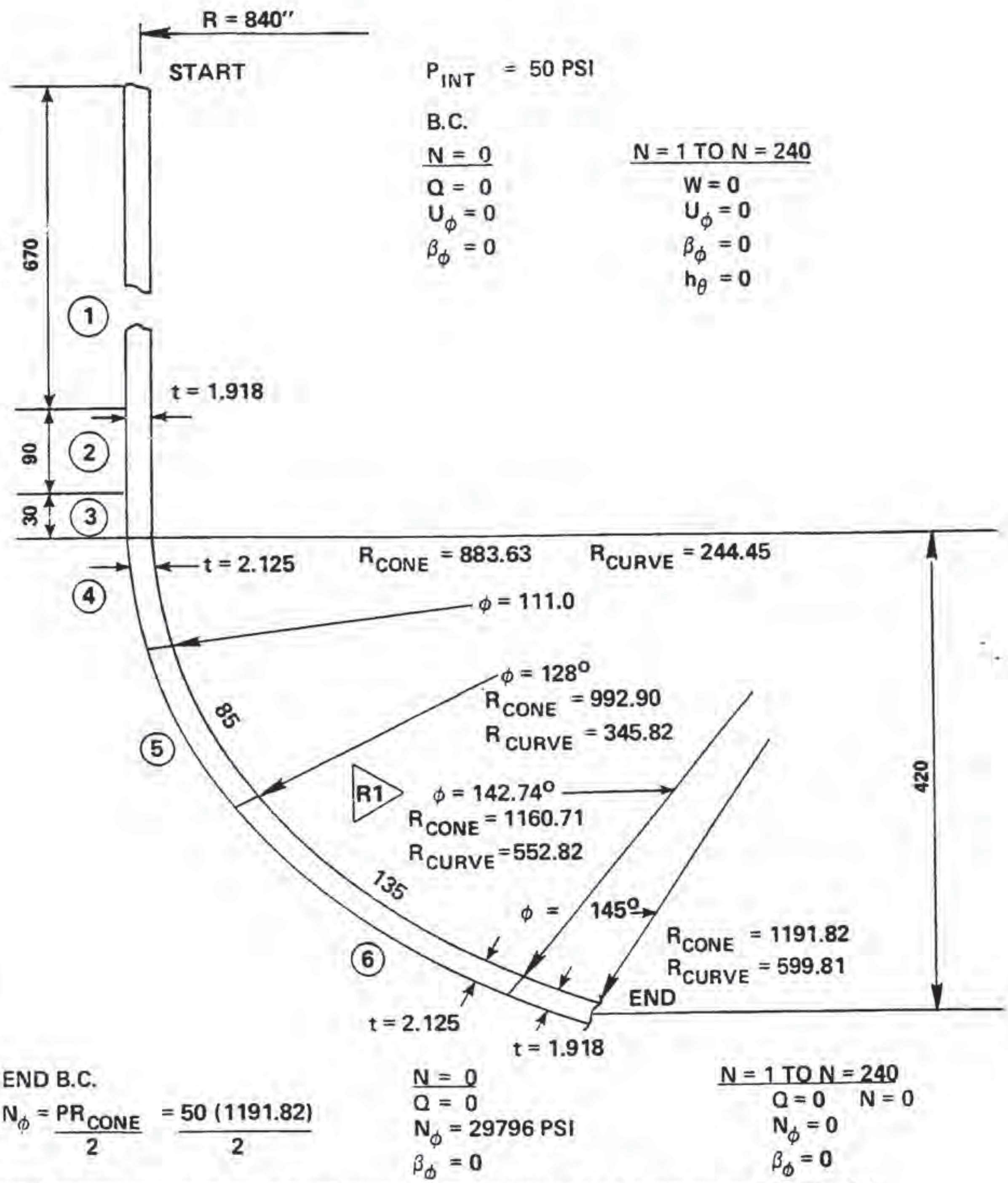
t_c = THICKNESS OF CYLINDER

F = SUM OF VERTICAL FORCES ABOVE POINT OF INTEREST

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL STRESSES RESULTING FROM VERTICAL AXIAL FORCES

FIGURE 3.8-18



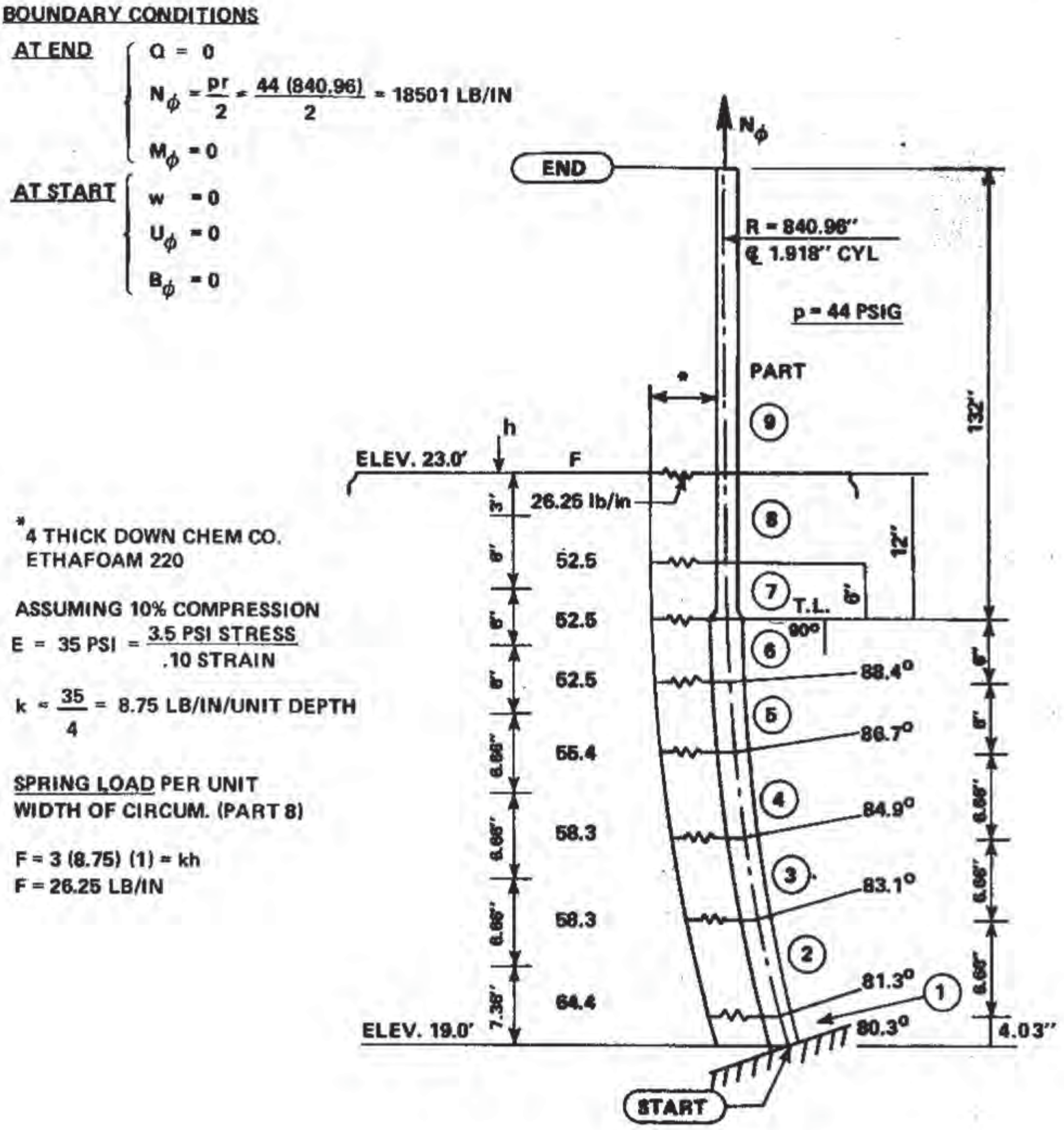
Historical Information, for more updated computer model refer to CB& I calculation PSL-73-7302.

Amendment No. 13 (05/00)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

COMPUTER MODEL FOR CBI PROGRAM
781

FIGURE 3.8-19



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

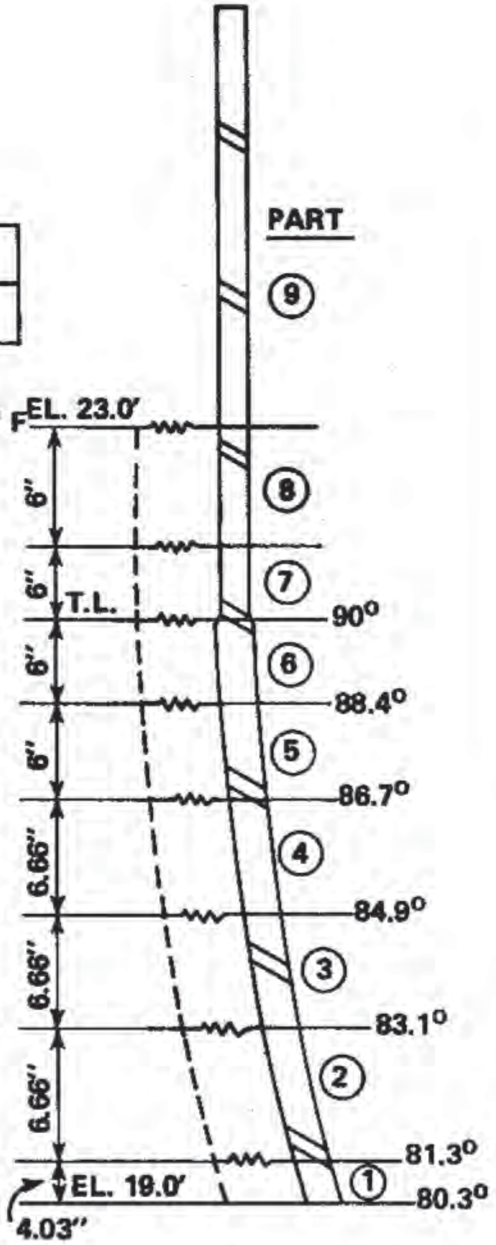
CONTAINMENT VESSEL EMBEDMENT
STRESS MODEL

FIGURE 3.8-20

SHELL TEMPERATURES UNDER ACCIDENT CONDITIONS

(120°F AMBIENT BEFORE ACCIDENT)

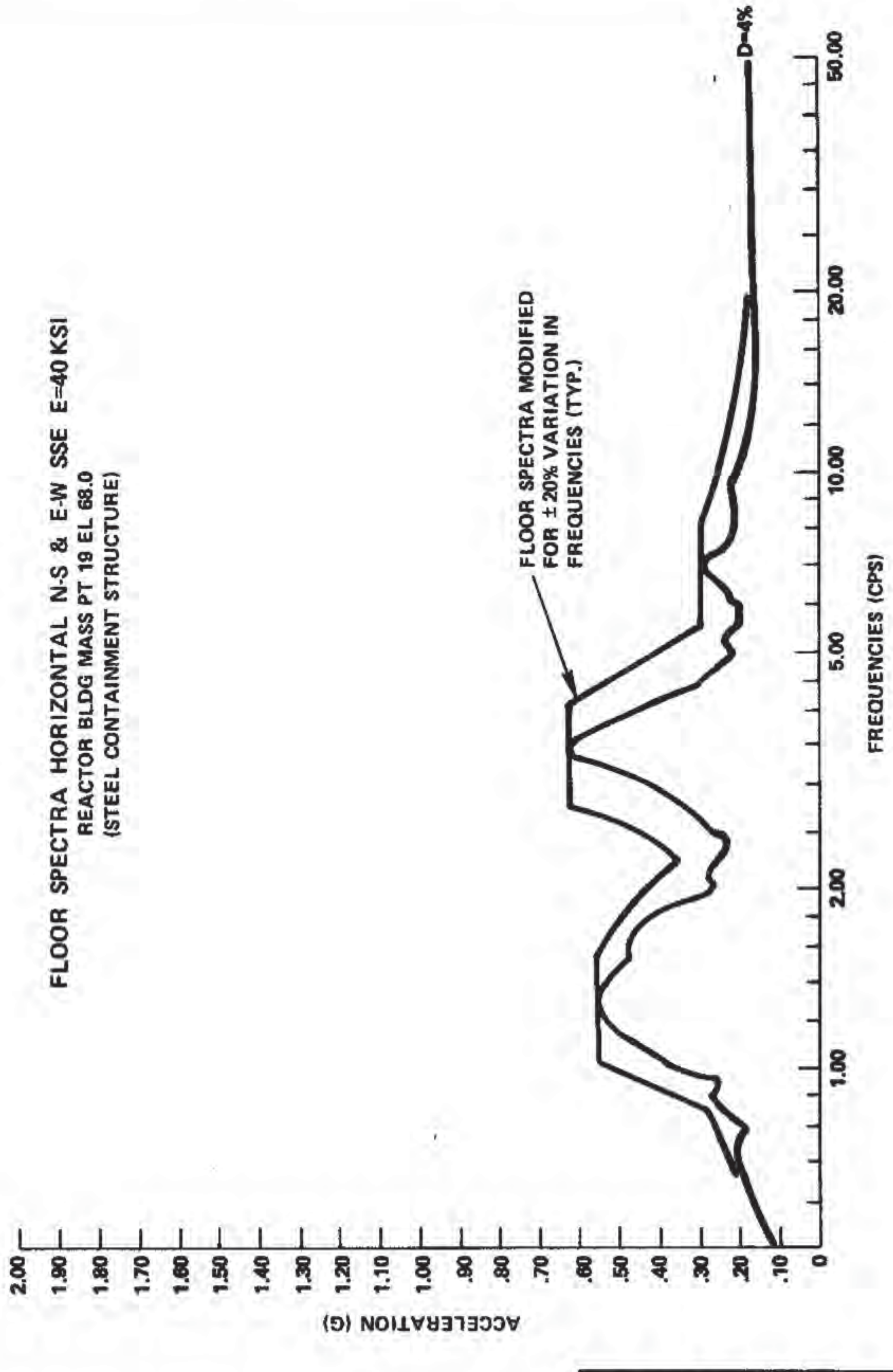
CASE	1	2	3	4
TIME	5 MIN	25 MIN	1 HR	3 HRS
	169°F	227° F	236° F	235° F
	169	227	236	235
	167	225	234	235
	121	155	182	207
	120	124	144	183
	120	122	134	163
	120	120	123	143
	120	120	121	135
	120	120	120	127



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

EMBEDMENT LOCA TEMPERATURE GRADIENT

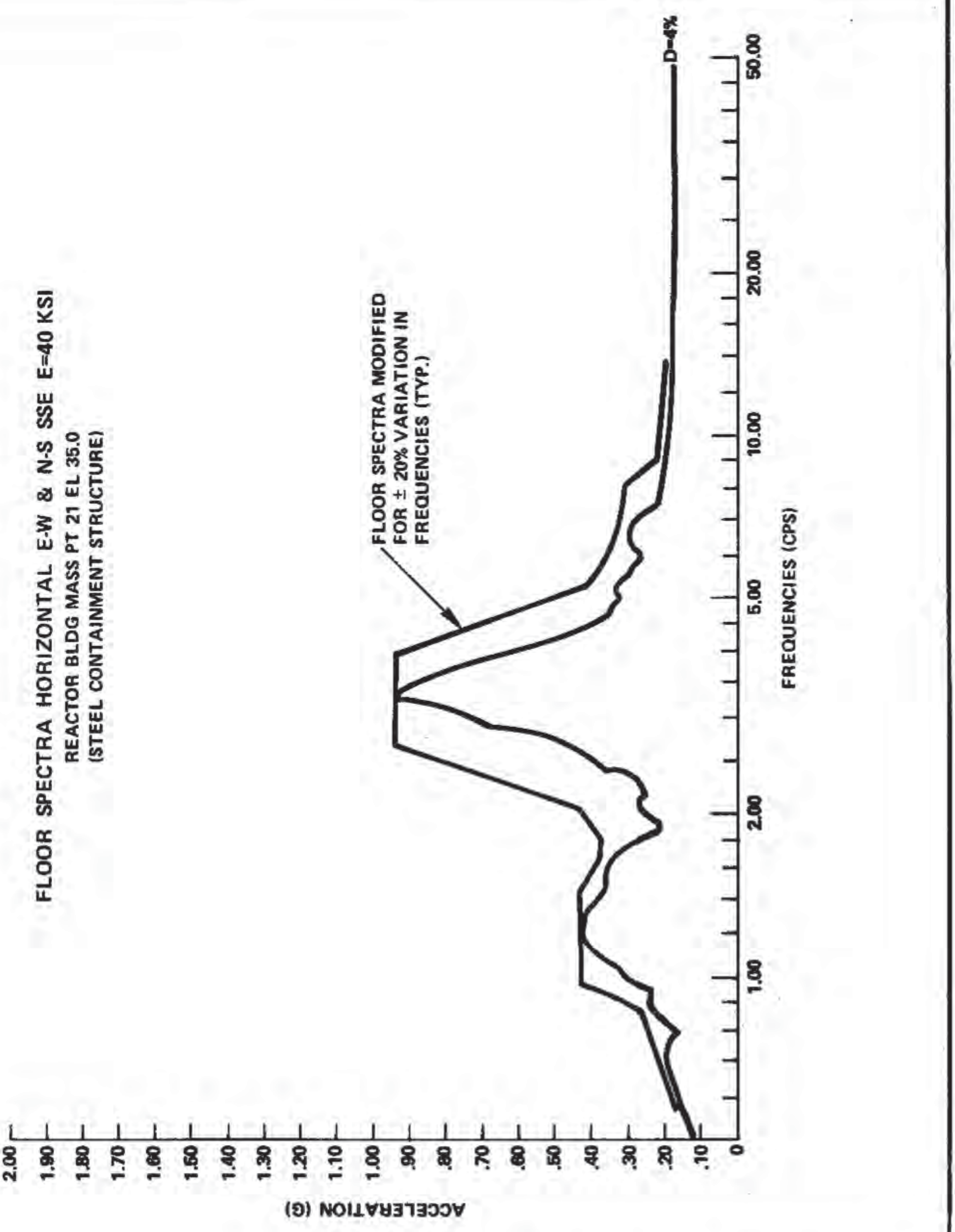
FIGURE 3.8-21



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

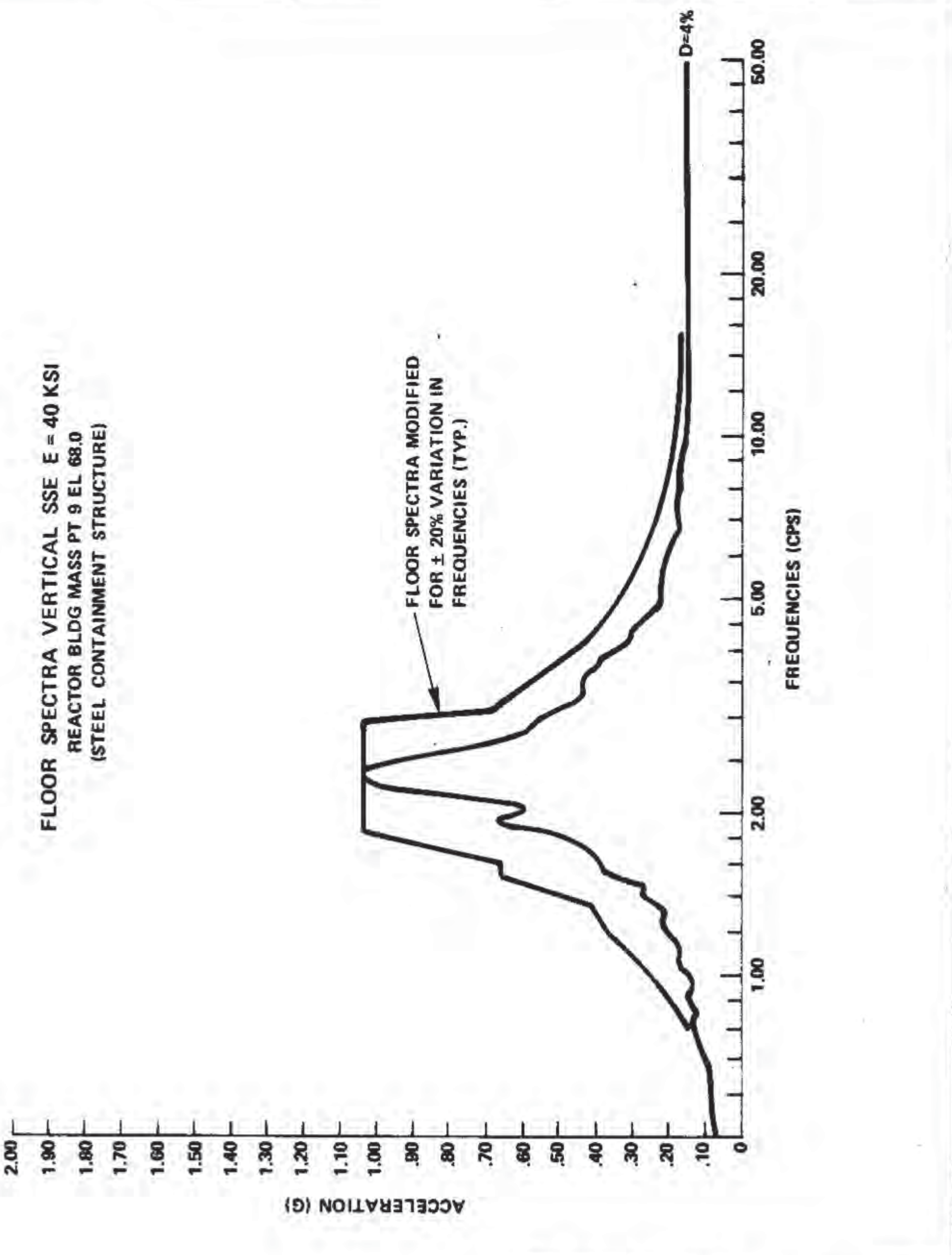
ESCAPE LOCK HORIZONTAL SSE
RESPONSE SPECTRUM

FIGURE 3.8-22



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

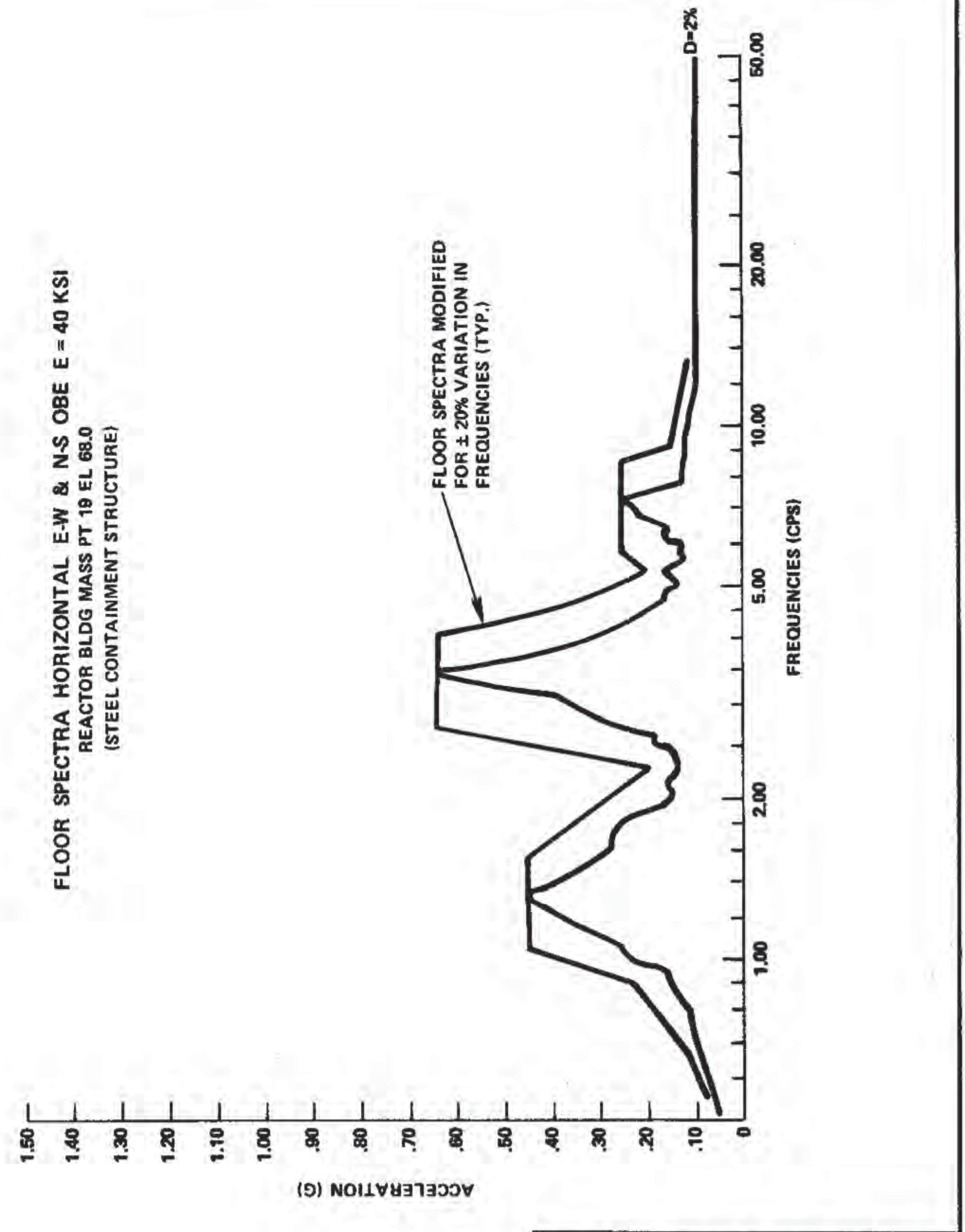
PERSONNEL LOCK HORIZONTAL SSE
RESPONSE SPECTRUM
FIGURE 3.8-23



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PERSONNEL AND ESCAPE LOCKS
VERTICAL SSE RESPONSE SPECTRUM

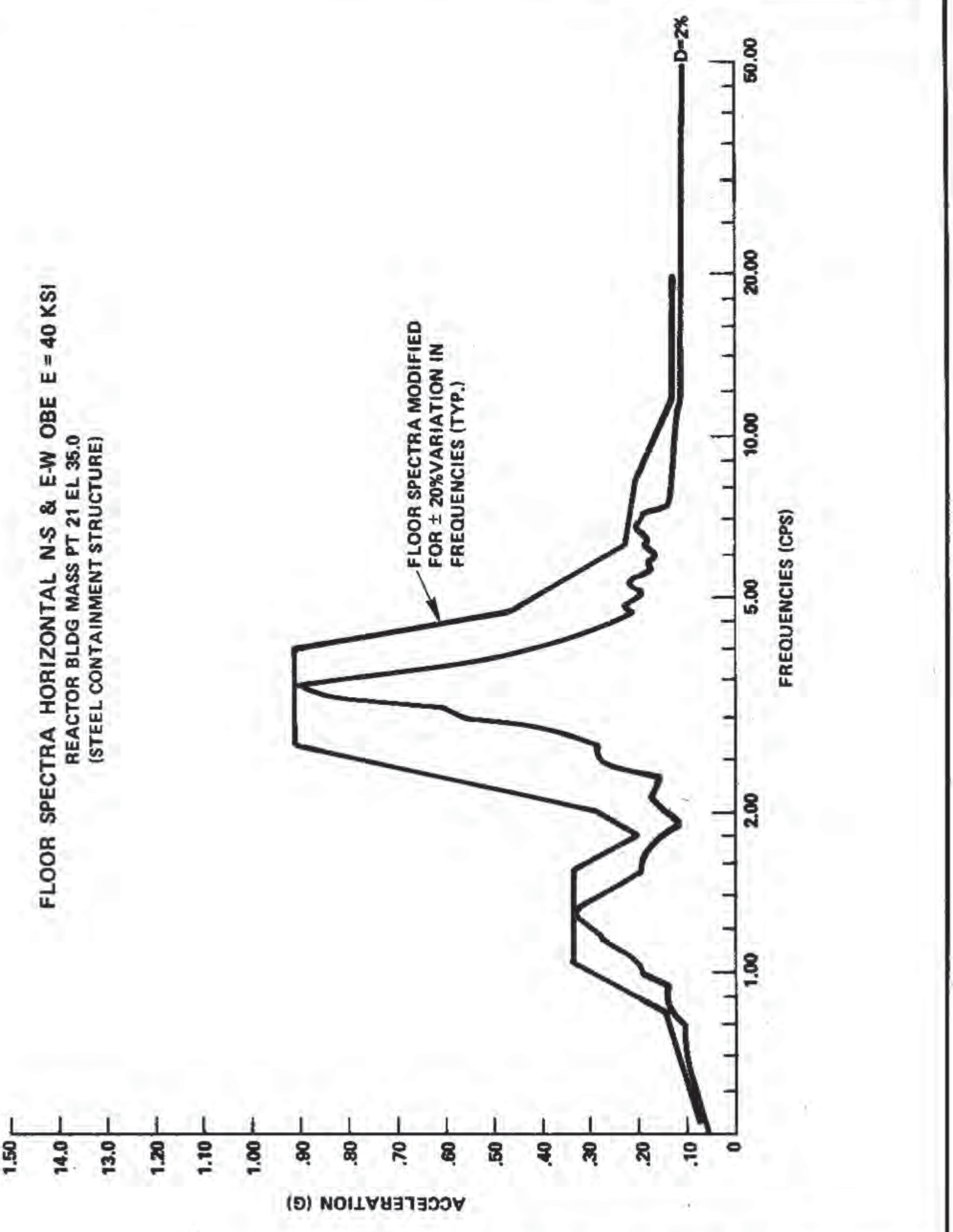
FIGURE 3.8-24



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

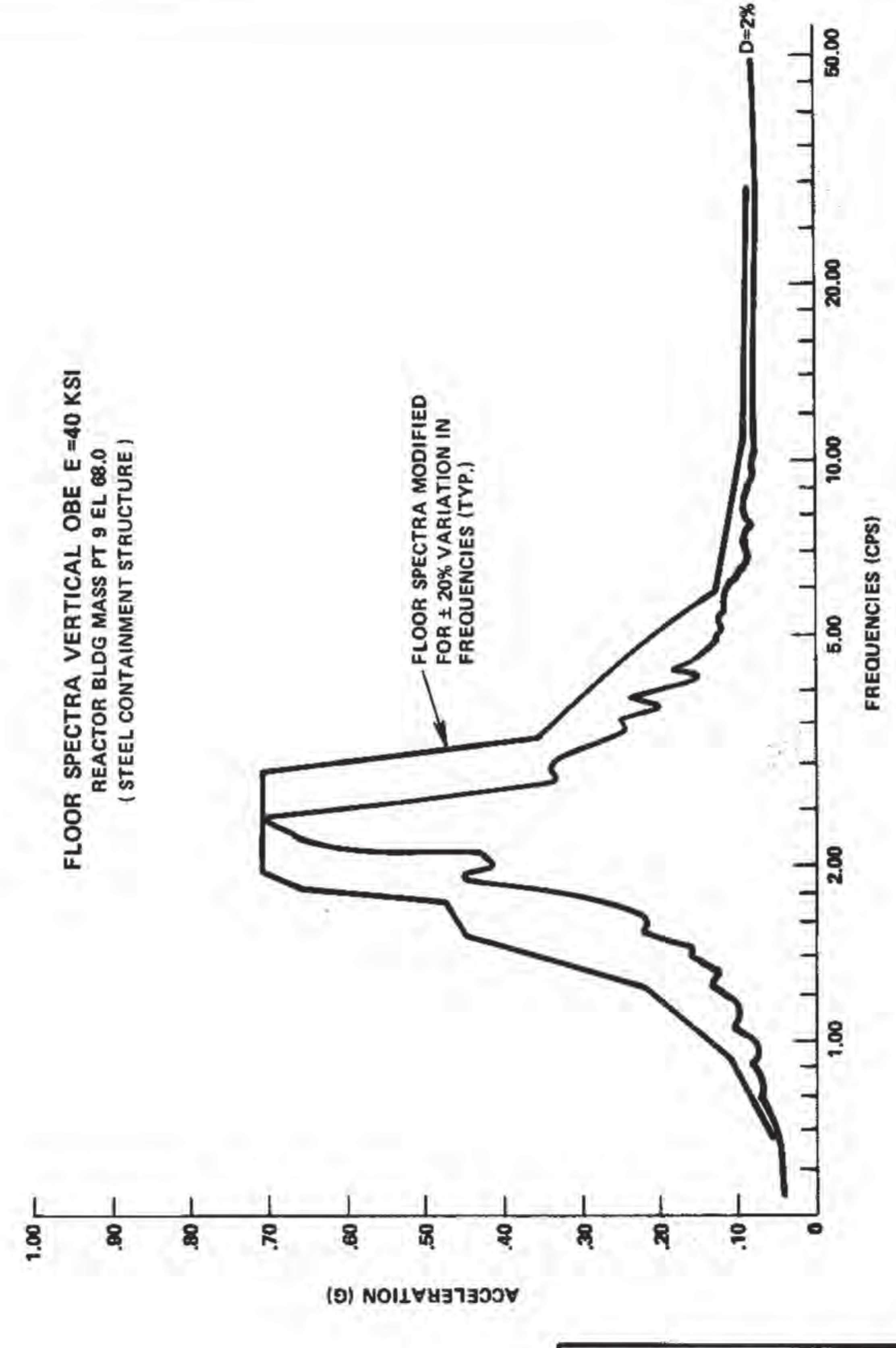
ESCAPE LOCK HORIZONTAL OBE RESPONSE SPECTRUM

FIGURE 3.8-25



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

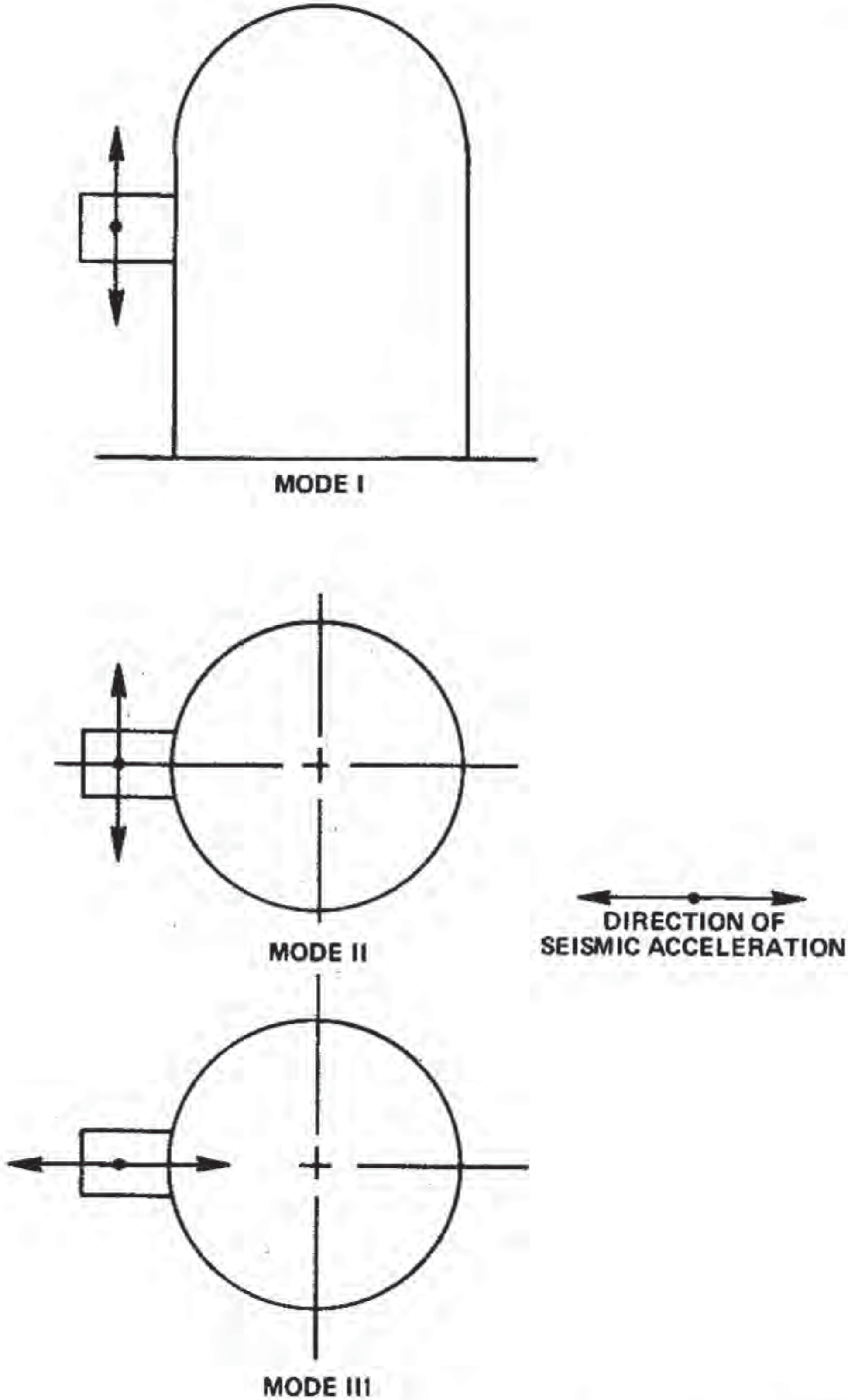
PERSONNEL LOCK HORIZONTAL OBE
RESPONSE SPECTRUM
FIGURE 3.8-26



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

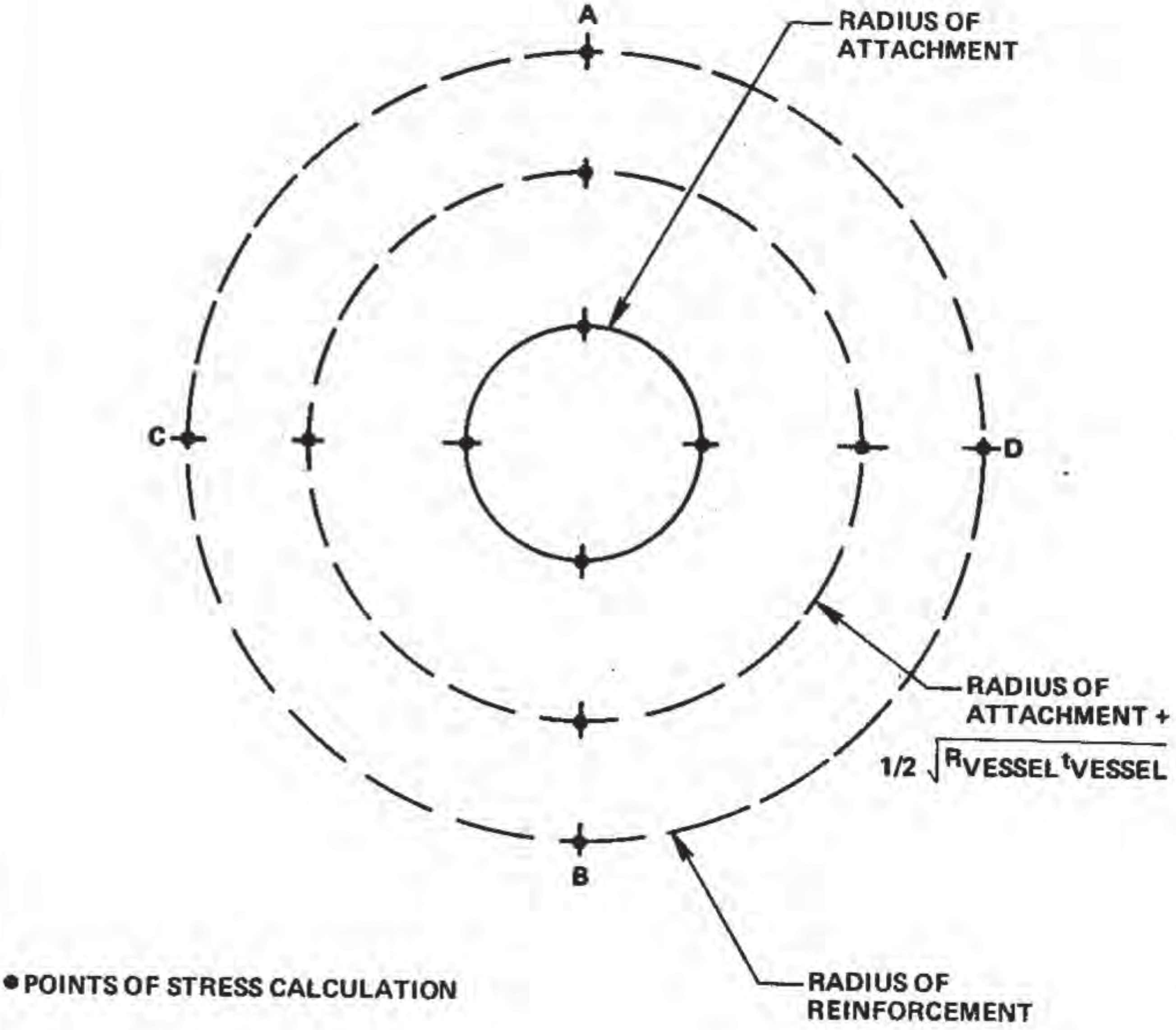
PERSONNEL AND ESCAPE LOCKS
VERTICAL OBE RESPONSE SPECTRUM

FIGURE 3.8-27



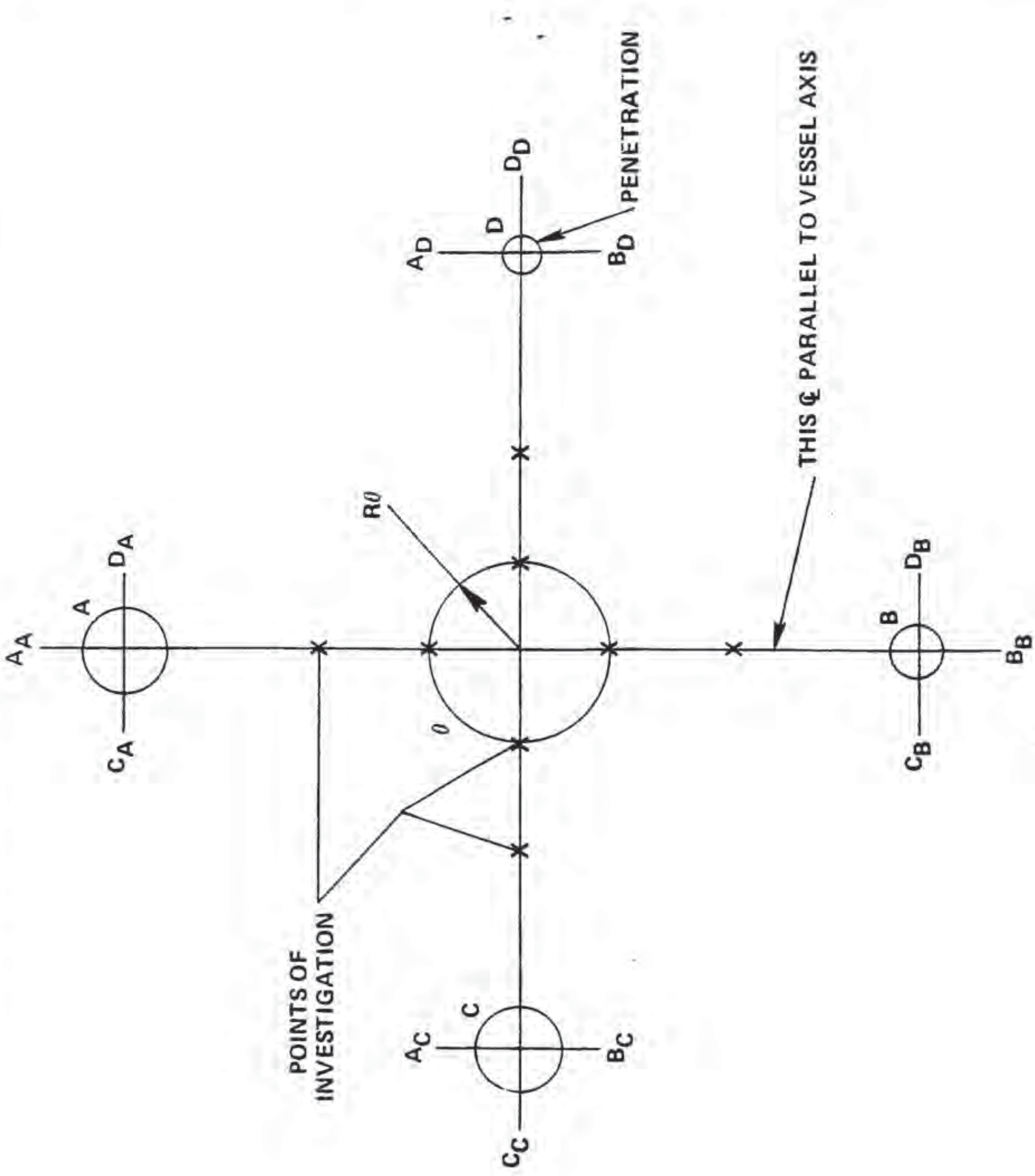
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

AIRLOCK SEISMIC ANALYSIS MODEL
FIGURE 3.8-28



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MODEL FOR CB&I PROGRAM 1027
FIGURE 3.8-29



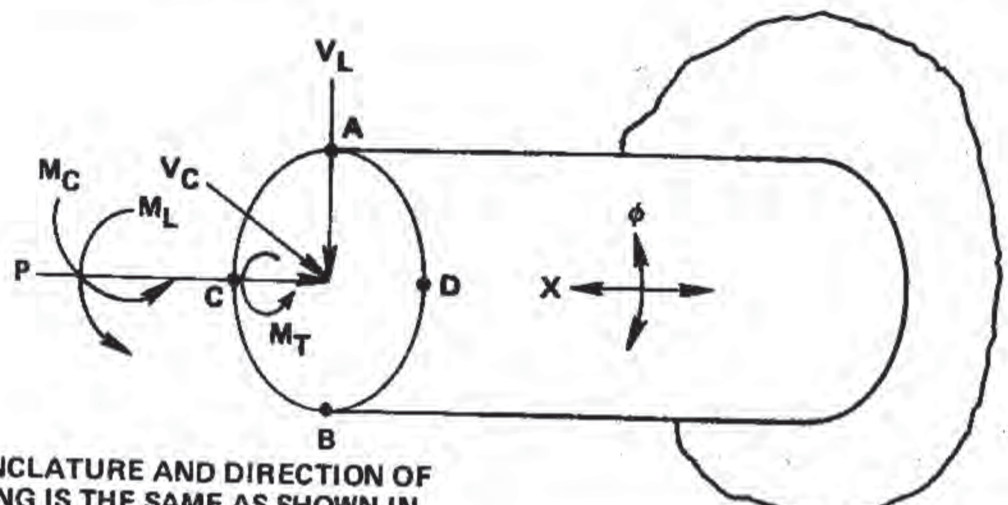
Historical Information, for Containment penetration computer model refer to CB& I calculation PSL-73-7302.

Amendment No. 13 (05/00)

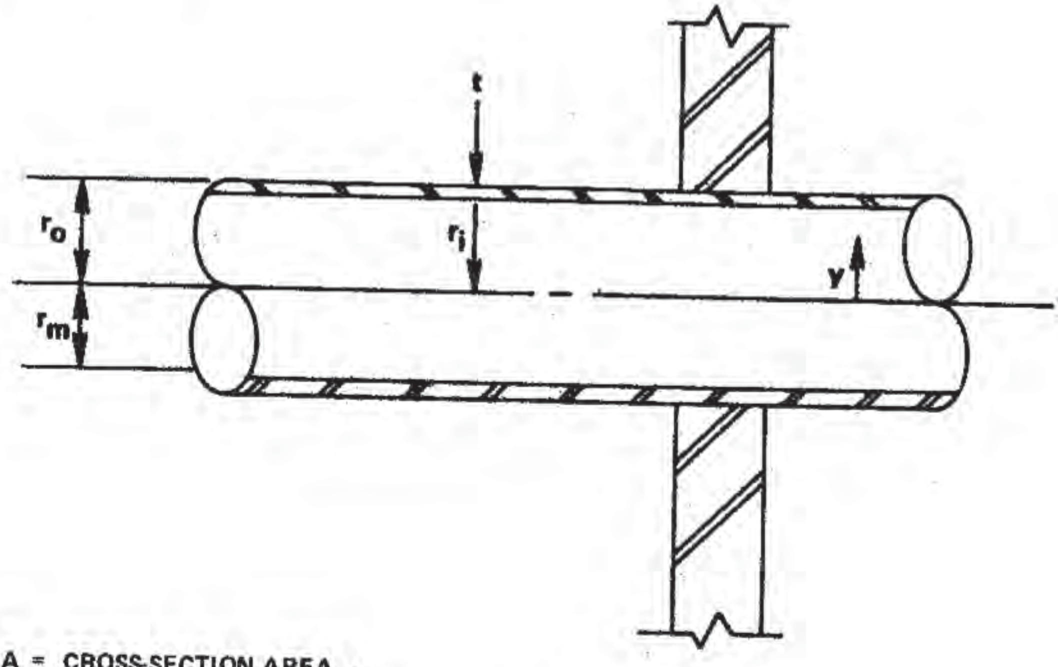
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT PENETRATION MODEL
USED IN CB&I PROGRAM 1036M

FIGURE 3.8-30



NOMENCLATURE AND DIRECTION OF LOADING IS THE SAME AS SHOWN IN WELDING RESEARCH COUNCIL BULLETIN 107



A = CROSS-SECTION AREA

S = SECTION MODULUS $= \frac{I}{y}$

J = POLAR MOMENT OF INERTIA $= \frac{\pi (r_o^4 - r_i^4)}{2}$

P = PRESSURE IN PIPE

r = DISTANCE FROM CENTERLINE OF PIPE TO POINT OF INVESTIGATION = r_m FOR CENTERLINE STRESS

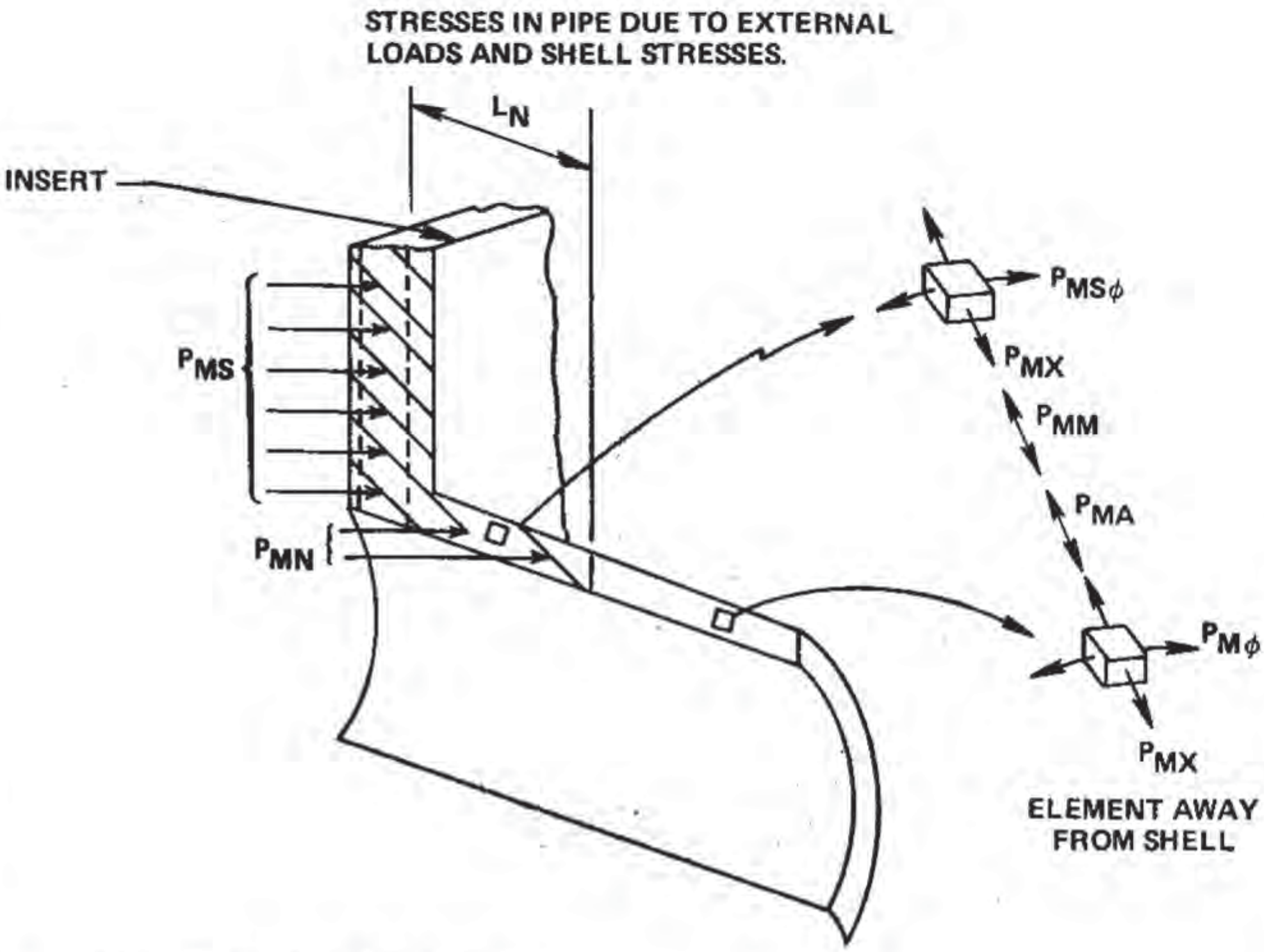
= r_o FOR SURFACE STRESS

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 2

CONTAINMENT PENETRATION MODEL USED IN CB&I PROGRAM 1392

FIGURE 3.8-31



L_N = NORMAL LIMIT FOR AREA REPLACEMENT

P_{MS} = MEMBRANE STRESS DUE TO PRESSURE IN VESSEL SHELL

P_{MN} = MEMBRANE STRESS DUE TO PRESSURE IN NECK

$P_{MS\phi}$ = MEMBRANE STRESS IN CIRCUMFERENTIAL DIRECTION DUE TO PRESSURE IN THE VESSEL

P_{MX} = MEMBRANE STRESS IN THE MERIDIONAL DIRECTION DUE TO PRESSURE IN PIPE

$P_{M\phi}$ = MEMBRANE STRESS IN THE CIRCUMFERENTIAL DIRECTION DUE TO PRESSURE IN PIPE

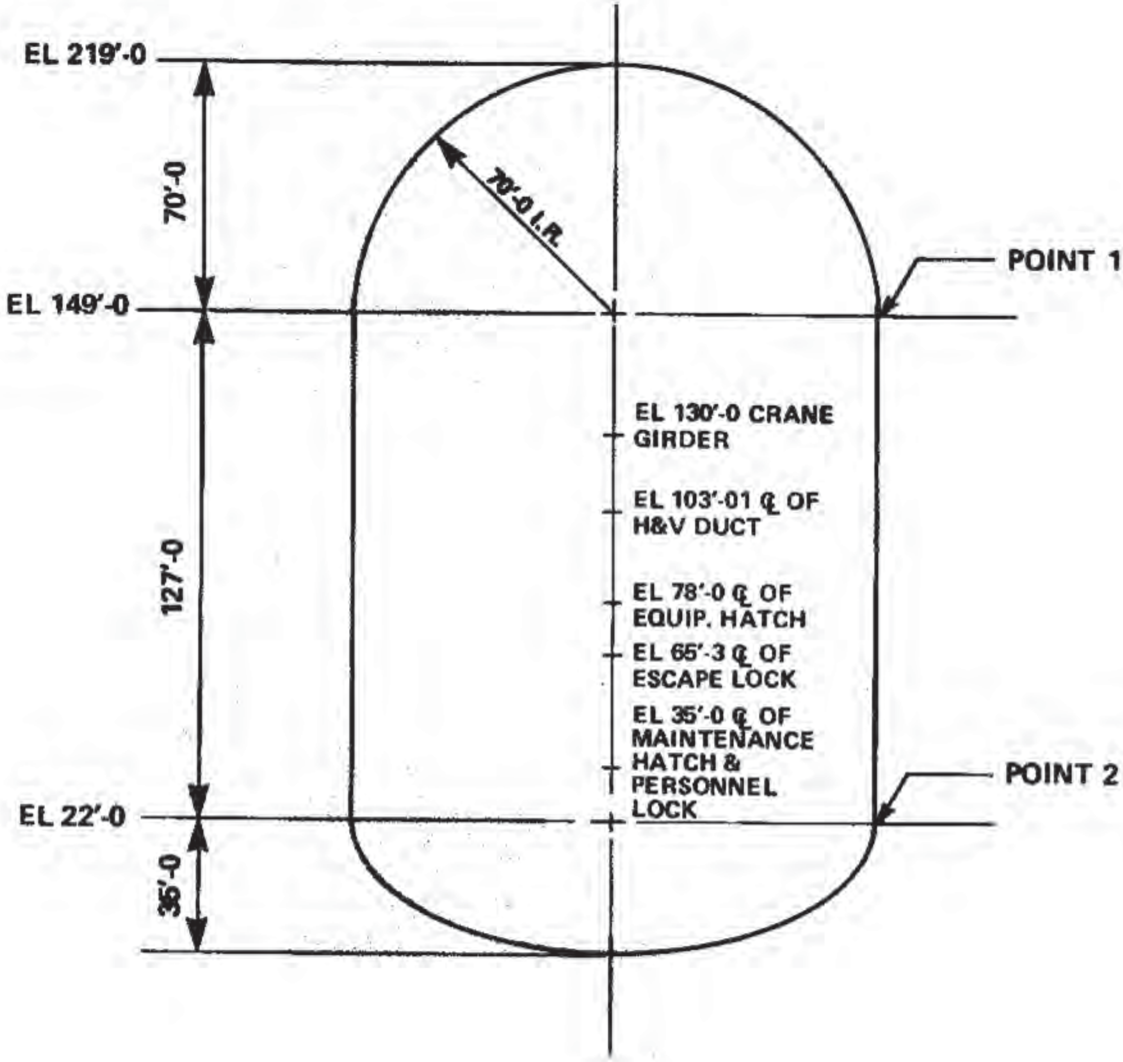
P_{MM} = MEMBRANE STRESS DUE TO EXTERNAL MOMENT

P_{MA} = MEMBRANE STRESS DUE TO AXIAL LOAD

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT PENETRATION MODEL USED IN CB&I PROGRAM 1392

FIGURE 3.8-32



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CRITICAL SHELL SECTIONS

FIGURE 3.8-33

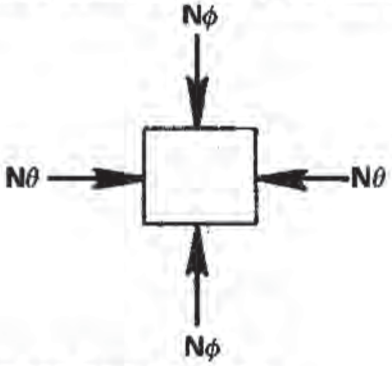
1. UNIAXIAL COMPRESSIVE STRESS RESULTANT AND BIAXIAL UNEQUAL TENSILE AND COMPRESSIVE STRESS RESULTANTS.

$N\phi$ (–) COMPRESSIVE
$N\theta$ 0

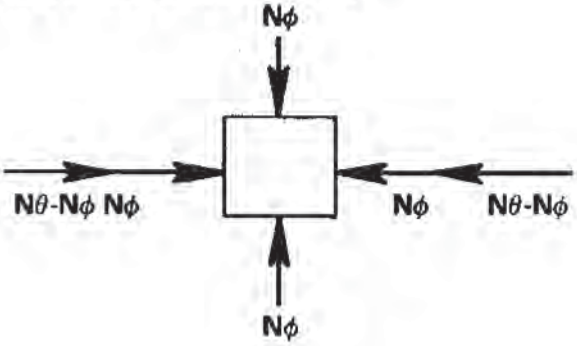
$N\phi$ (–) COMPRESSIVE
$N\theta$ (+) TENSILE

2. BIAXIAL EQUAL COMPRESSIVE STRESS RESULTANTS.

$N\phi$ (–) COMPRESSIVE = $N\theta$ (–) COMPRESSIVE



3. BIAXIAL UNEQUAL STRESS RESULTANTS.

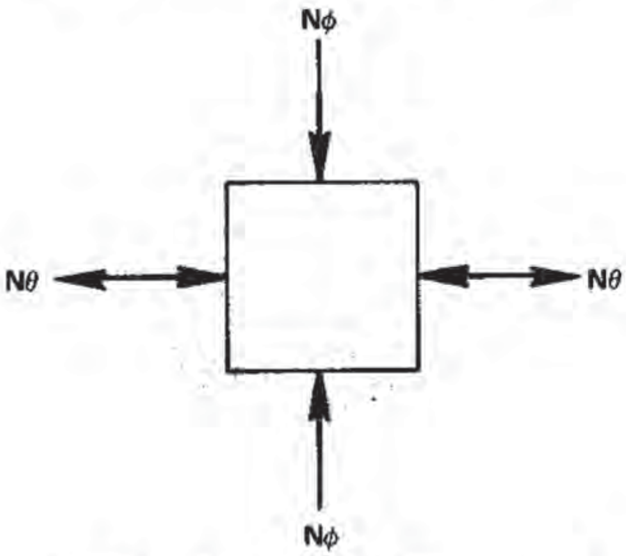


FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ALLOWABLE BUCKLING STRESSES FOR UNSTIFFENED HEMISPHERICAL HEAD

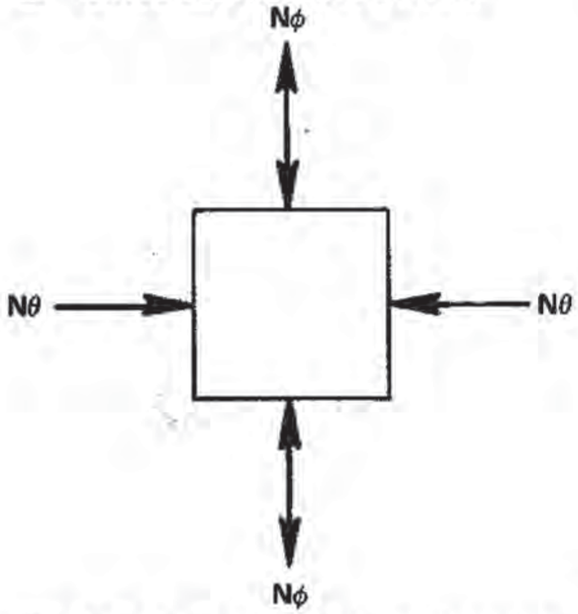
FIGURE 3.8-34

1. MERIDIONAL OR AXIAL STRESS



$N\theta$ CAN BE TENSILE OR COMPRESSIVE

2. CIRCUMFERENTIAL STRESS



$N\phi$ CAN BE TENSILE OR COMPRESSIVE

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ALLOWABLE BUCKLING STRESSES
FOR CYLINDRICAL VESSEL

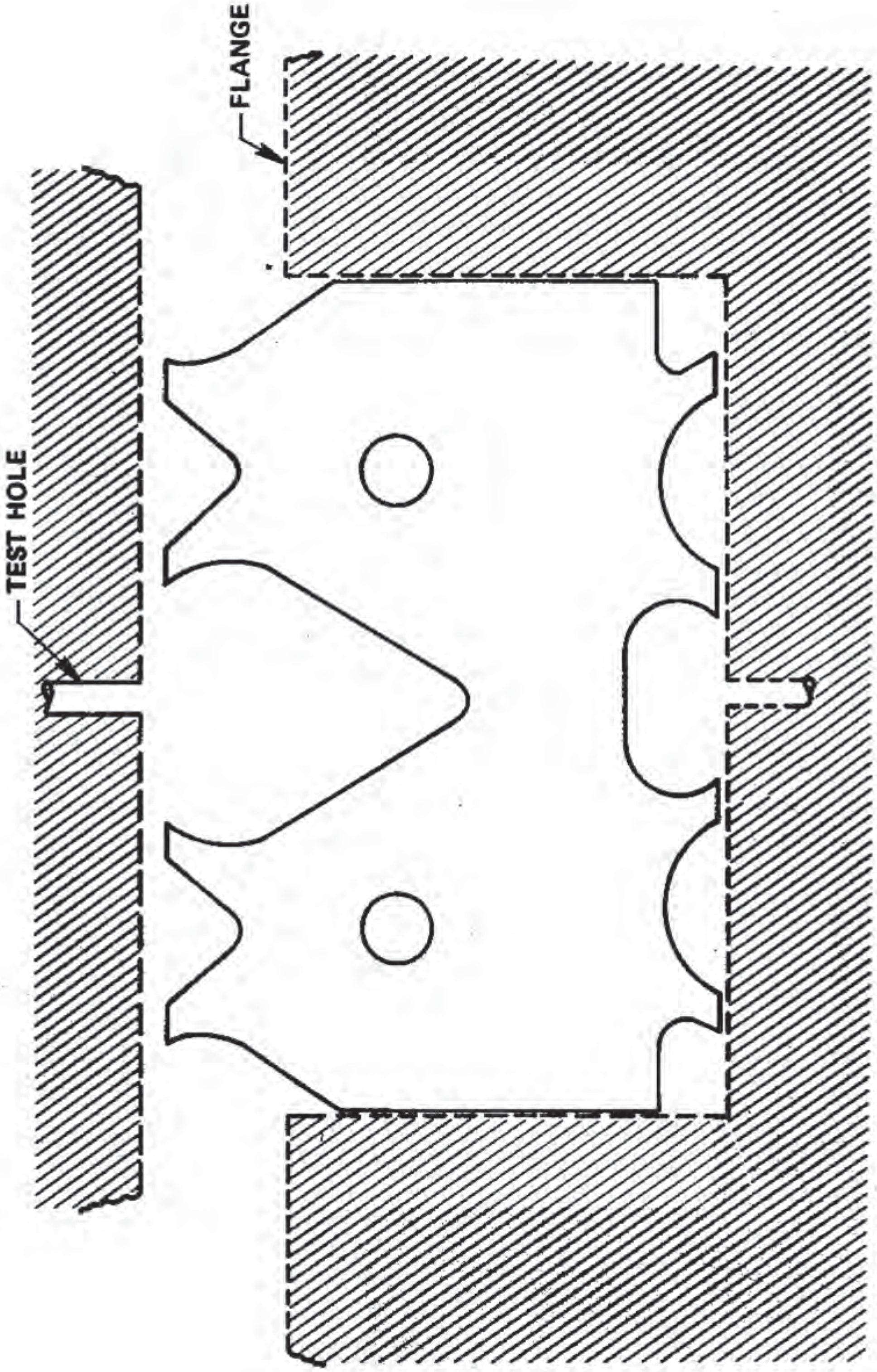
FIGURE 3.8-35

Refer to Drawing
2998-G-493

Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CONTAINMENT VESSEL SUPPORT -
PLAN AND SECT. MAS AND REINF
FIGURE 3.8-36



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

LOCK GASKET CROSS SECTION

FIGURE 3.8-37

Refer to Drawing
2998-G-518

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

INTERNAL CONCRETE - PLANS
AND SECTIONS MASONRY
SHEET 1
FIGURE 3.8-38

Refer to Drawing
2998-G-519

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

INTERNAL CONCRETE - PLANS AND
SECTIONS MASONRY-
SHEET 2
FIGURE 3.8-39

Refer to Drawing
2998-G-539

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

EQUIPMENT FDNS - M AND R
SH. 1
FIGURE 3.8-40

Refer to Dwg.
2998-G-794 SH 2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING
EQUIPMENT SUPPORTS
SHEET 2
FIGURE 3.8-41

Refer to Dwg.
2998-G-794 SH 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING
EQUIPMENT SUPPORTS
SHEET 1
FIGURE 3.8-42

Refer to Drawing
2998-G-794 SH 5

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EQUIPMENT
SUPPORTS SH. 5
FIGURE 3.8-43

Refer to Drawing
2998-G-794 SH 11

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EQUIPMENT
SUPPORTS
SHEET 11
FIGURE 3.8-44

Refer to Drawing
2998-G-794 SH 12

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EQUIPMENT SUPPORTS
SHEET 12
FIGURE 3.8-45

Refer to Drawing
2998-G-794 SH 13

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EQUIPMENT SUPPORTS
SHEET 13
FIGURE 3.8-46

Refer to Drawing
2998-G-794 SH 4

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EQUIPMENT
SUPPORTS
SHEET 4
FIGURE 3.8-47

Refer to Drawing
2998-G-794 SH 7

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EQUIPMENT SUPPORTS
SHEET 7
FIGURE 3.8-48

DELETED

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EQUIPMENT SUPPORTS
SHEET 8
FIGURE 3.8-49

DELETED

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ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EQUIPMENT
SUPPORTS SH. 9
FIGURE 3.8-50

Refer to Drawing
2998-G-803

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING EMBEDDED
STEEL PRIMARY SHIELD WALL
FIGURE 3.8-51

Refer to Drawing
2998-G-794 SH 3

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING
EQUIPMENT SUPPORTS
SHEET 3
FIGURE 3.8-52

Refer to Dwg.
2998-G-795 SH 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING PLATFORMS
FIGURE 3.8-53

Amendment No. 10, (7/96)

Refer to Drawing
2998-G-501

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

CYLINDER WALL -
PLAN AND SECTIONS MASONRY
FIGURE 3.8-54

Refer to Drawing
2998-G-503 SH 1

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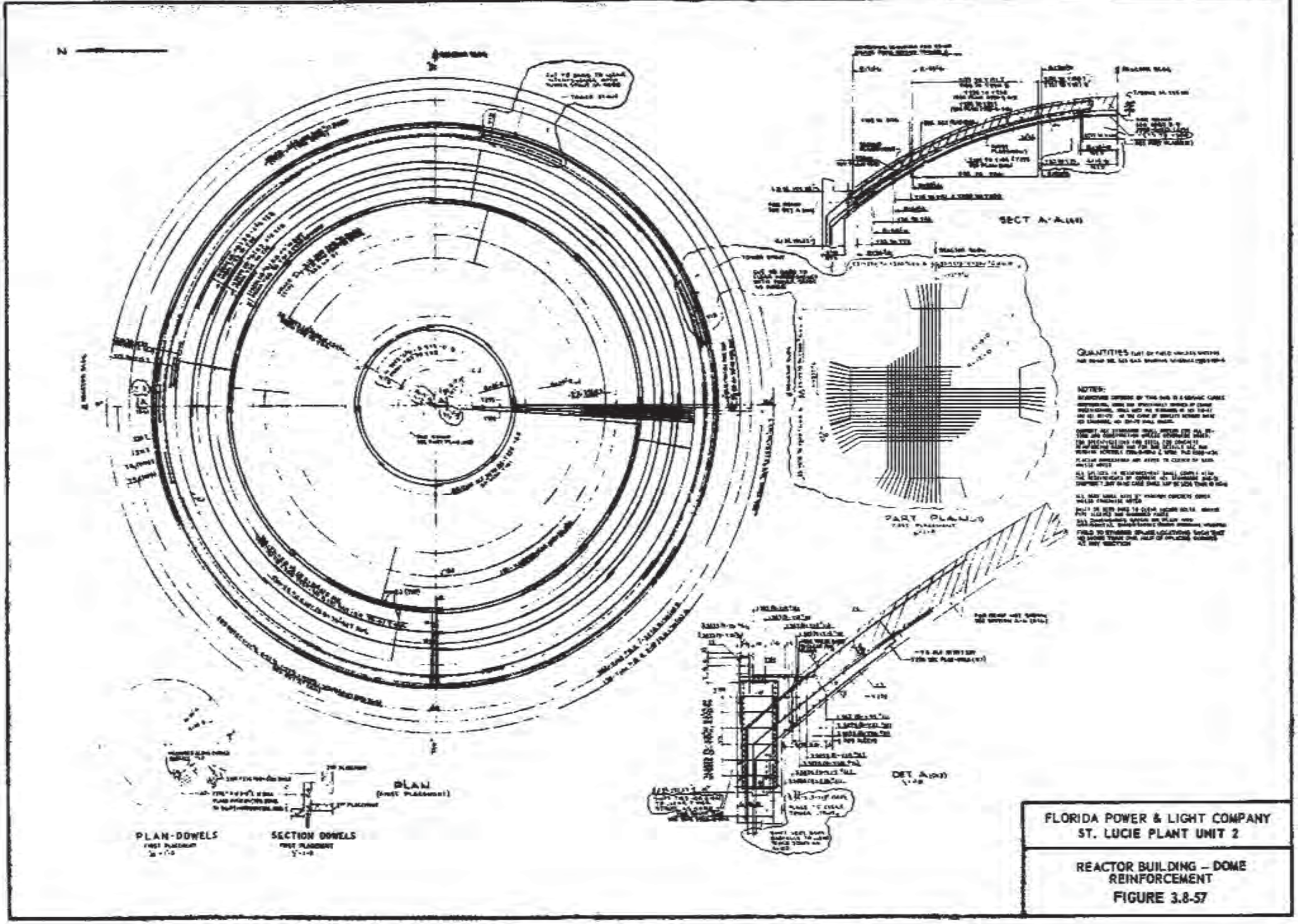
ST. LUCIE PLANT UNIT 2

CYLINDER DEVELOPMENT - MASONRY
FIGURE 3.8-55

Refer to Drawing
2998-G-511

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

DOME - PLAN AND SECTIONS -
MAS.
FIGURE 3.8-56



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING – DOME REINFORCEMENT

FIGURE 3.8-57

Refer to Drawing
2998-G-568

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUXILIARY
BUILDING FRAMING PLAN -
SLABS & BEAMS
SHEET 1
FIGURE 3.8-58

Refer to Drawing
2998-G-569

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUXILIARY BUILDING FRAMING PLAN - SLABS & BEAMS
SHEET 2
FIGURE 3.8-59

Refer to Drawing
2998-G-667 SH 1

FLORIDA POWER & LIGHT COMPANY
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DIESEL GENERATOR BUILDING MAS
FIGURE 3.8-60

Refer to Dwg.
2998-G-838 SH 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MAIN STEAM TRESTLE
SHEET 1
FIGURE 3.8-61

Amendment No. 10, (7/96)

Refer to Dwg.
2998-G-838 SH 2

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

MAIN STEAM TRESTLE
SHEET 2
FIGURE 3.8-62

Refer to Dwg.
2998-G-838 SH 3

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

MAIN STEAM TRESTLE
SHEET 3
FIGURE 3.8-63

Refer to Dwg.
2998-G-838 SH 4

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MAIN STEAM TRESTLE
SHEET 4
FIGURE 3.8-64

Refer to Dwg.
2998-G-838 SH 5

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MAIN STEAM TRESTLE
SHEET 5
FIGURE 3.8-65

Refer to Dwg.
2998-G-838 SH 6

FLORIDA POWER & LIGHT
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MAIN STEAM TRESTLE
SHEET 6
FIGURE 3.8-66

Refer to Drawing
3509-G-101

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

STEAM GENERATOR
BLOWDOWN
FIGURE 3.8-67

Amendment No. 18 (01/08)

Deleted

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FIGURE 3.8-68

Refer to Drawing
3509-G-103

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ST. LUCIE PLANT UNIT 2

STEAM GENERATOR BLOWDOWN

FIGURE 3.8-69

Refer to Dwg.
2998-G-488

FLORIDA POWER & LIGHT
COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING
BASE SLAB - PLAN -
MAS.
FIGURE 3.8-70

Refer to Dwg.
2998-G-491

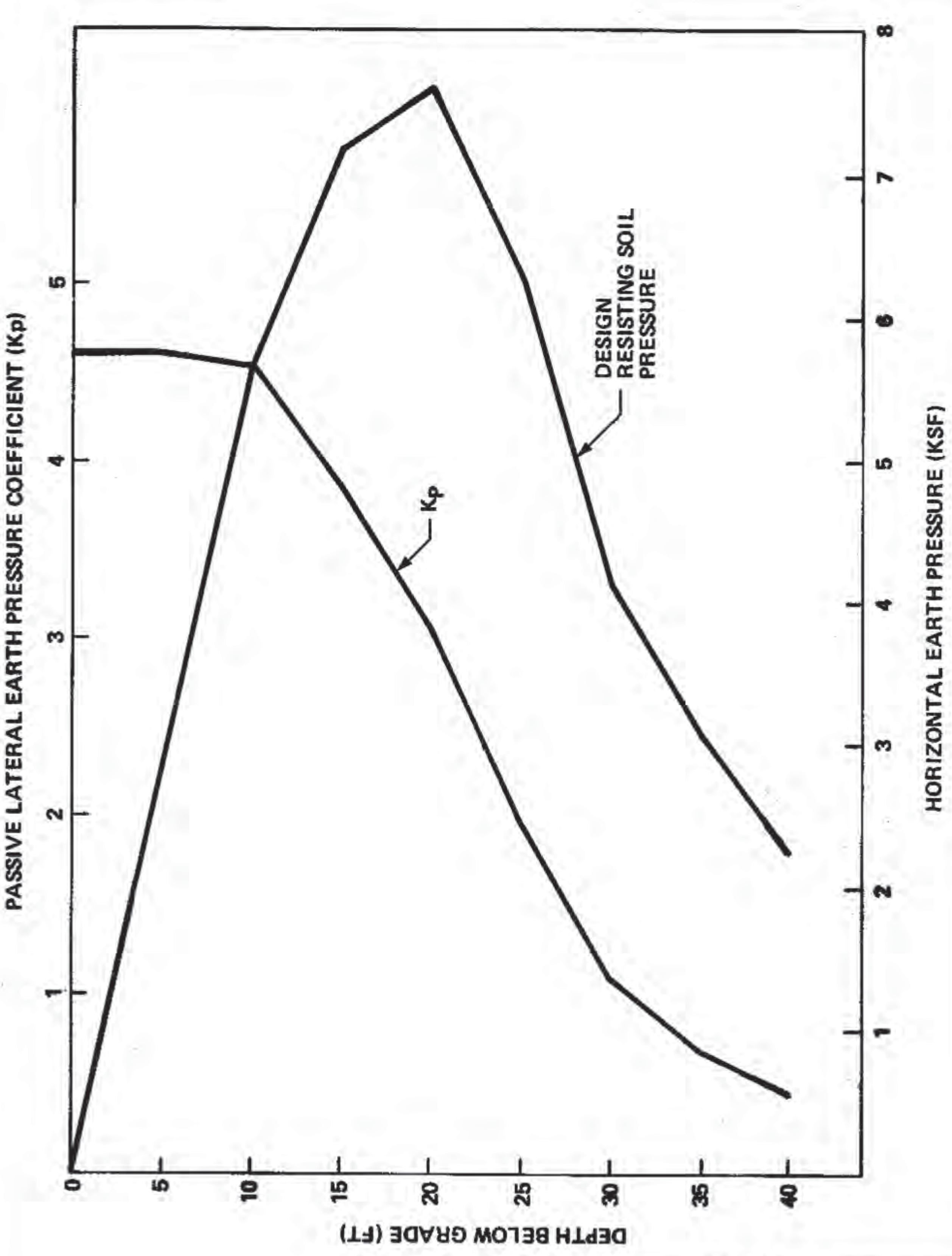
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR BUILDING
BASE SLAB - SECT. REINF. -
SH. NO. 2
FIGURE 3.8-71

Refer to Drawing
2998-G-554

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

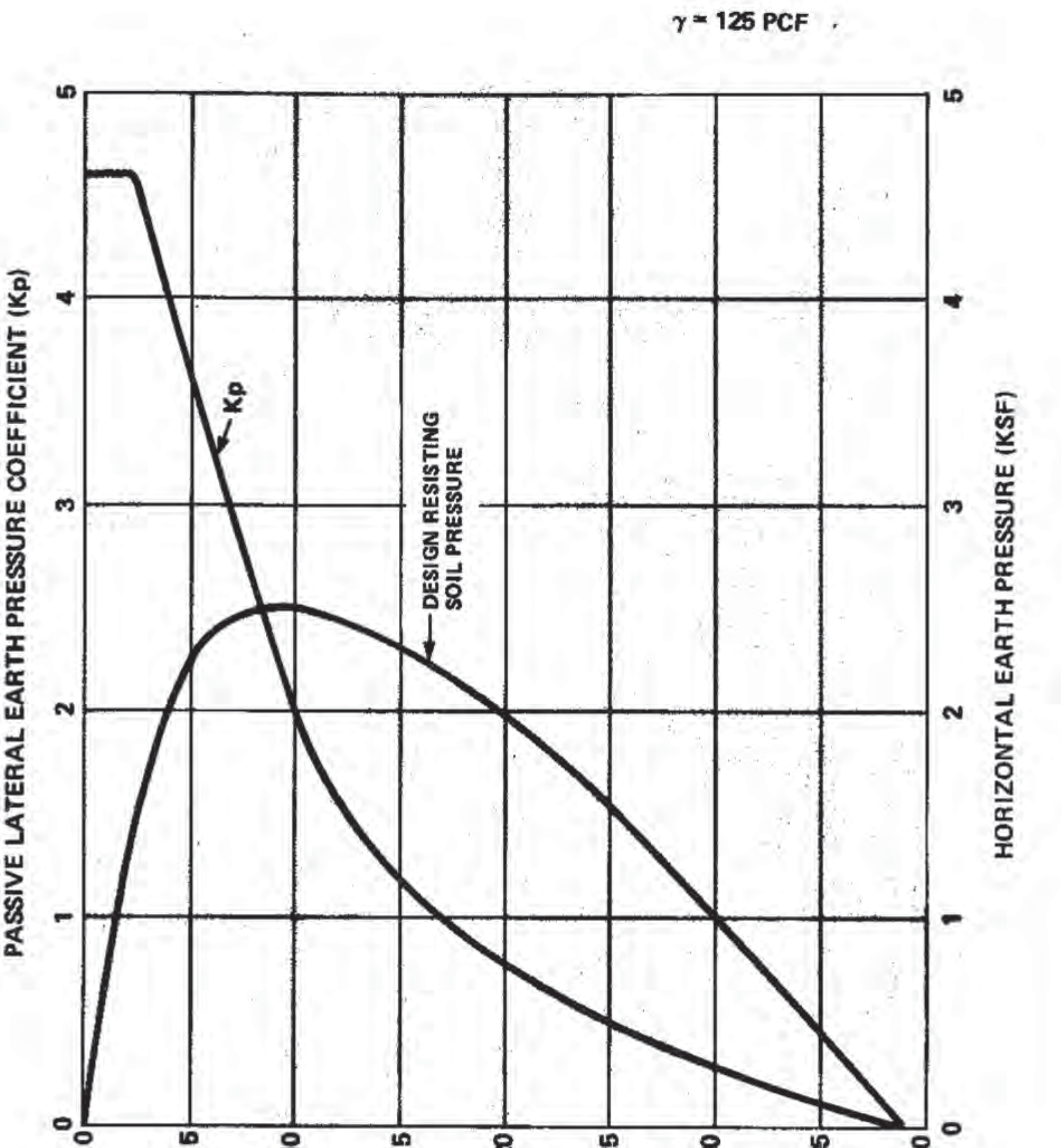
REACTOR AUXILIARY BUILDING
BASE SLAB PLAN - MASONRY
FIGURE 3.8-72



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FB–SOIL PRESSURE UNDER EARTHQUAKE

FIGURE 3.8-73



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

SOIL PRESSURE UNDER EARTHQUAKE
RAB DBE

FIGURE 3.8-74

APPENDIX 3.8A

EVALUATION OF CONCRETE MASONRY WALLS

3.8A EVALUATION OF CONCRETE MASONRY WALLS

3.8A.1 SUMMARY

Safety-related masonry walls that are located in St. Lucie Unit 2 are evaluated in accordance with the "SEB Criteria for Safety-Related Masonry Wall Evaluation" and supplementary criteria as described below.

All safety-related walls are reinforced. Reinforced units are spaced 4'-0 on centers. Due to similarity among walls, some of them are grouped together and the typical one is analyzed to represent the group.

The masonry walls are not shear resistant elements in the building structural system. They primarily function as shielding and partition walls. Therefore, the primary effort of the masonry wall evaluation was focused on the out-of-plane bending analysis due to seismic loads and pressure loads.

3.8A.2 WALL REINFORCING

Some walls require external reinforcement to meet the criteria stated above. There are two types of external wall reinforcing. The first type uses through-bolting to achieve composite action for multi-wythe walls. Figure 3.8A-1 shows a typical through-bolting detail. The second type of wall reinforcing is for walls subjected to accident pressure load. Structural steel is used to support the wall.

Sample calculations illustrating the two types of reinforcing have been submitted to the NRC. [(1)]

3.8A.3 ANALYTICAL MODEL

The masonry walls are transformed into equivalent homogenous plate elements spanning vertically to resist out-of-plane bending loads. For full height walls, the top of the wall is assumed to be simply supported, since the walls are restrained by two clip angles on both sides. The bottom of the wall is assumed as simply supported at the top of the starter wall, which is a three foot high reinforced concrete pier doweled into the floor slab. For cantilever walls, the partial fixity at the bottom of the wall is taken into account. For walls that are restrained laterally at the top by the Houdaille precast slab, the simple support condition is also assumed at the top.

For stack bond walls, the effective width of each reinforced unit according to ACI 531-79, Paragraph 9.4.6.1 is a little less than the actual spacing of the reinforced units. However, DUR-O-WALL reinforcement is provided for every course so that the entire width of the wall between reinforced units is considered effective for the model.

For multi-wythe walls, non-composite action was initially assumed for evaluation of the walls since no shear connectors were originally provided. The distribution of bending moment among the wythes is thus proportional to their stiffness or moment of inertia. However, for walls composed of two six-inch wythes, the bending capacity of the walls is so small that the design fix of through-bolting was introduced directly for analysis.

If large openings exist, finite element models are simulated to represent the masonry walls. The weights of attachments are considered as mass inputs for frequency analysis. All attachments are rigidly connected to the walls. Also, the maximum weight of the attachment is less than one percent of the total weight of the masonry wall itself. Therefore, the dynamic amplification of the attachments is not considered likely to have a significant effect. The ANSYS computer program is used for all analysis.

3.8A.4 INTERSTORY DRIFT CONSIDERATIONS

A full-height wall is laterally restrained at the top by a pair of angles as shown in the typical details of Figure 3.8A-1. The rotation of the wall is not restrained. Therefore a simple support is assumed at the top. The bottom of the wall is afforded partial fixity by means of dowels projecting from a poured concrete starter wall (see Figure 3.8A-1). However, it is more conservative to assume a simply supported condition for both frequency and static analyses. The frequency of walls are all larger than the peak of the floor response spectra curve. Lower frequency is obtained when the wall is assumed simply supported. Consequently, conservatively higher earthquake coefficients are used for analysis. The maximum positive bending moment of a simply supported one way plate is larger than the moment of the same plate assumed fixed or partially fixed at one end and simply supported at the other end when the plate is subjected to uniform seismic inertial loads (i.e., $1/8 \times WL^2 > 9/128 \times WL^2$). The maximum negative bending moment for a bottom-fixed wall is equal to $1/8\ WL^2$. Therefore, for a bottom partially fixed wall, the negative bending moment is less than $1/8\ WL^2$ and also is less than the positive bending moment of the bottom-hinged wall as assumed.

The influence of out-of-plane drift effects on the flexural bending moment calculation is not significant due to the small magnitude of the relative displacement between floors. As indicated in the calculation (1), the maximum relative displacement between floors is equal to 0.0065 in. For a typical bottom fully fixed, top simply supported wall (20 feet high, 4 feet wide, 1 foot thick) the fixed end moment due to the drift (Δ) is equal to $3\ EI\ \Delta/L^2 = (3 \times (0.97 \times 10^3) \times (4915) \times 0.017)/ (20 \times 12)^2 = 4.22\ ''^K = 0.35'^K$. The moment capacity of the wall is equal to $f_s\ A_s\ jd = 24 \times 0.88 \times 8.8 = 167''^K = 14'^K$. The ratio of the fixed end moment to the moment capacity of the wall is only 2.5 percent.

The in-plane interstory drift for the walls is evaluated by comparing the shear strains derived from the dynamic analysis with the following allowable strains:

$\gamma u = 0.0001$ for unconfined walls

$\gamma c = 0.001$ for confined walls.

The above values are used for normal and severe environmental load combinations. For other load combinations, the allowable strains are multiplied by a factor of 1.67.

An unconfined wall is attached on one vertical boundary and its base. A confined wall is attached in one of the following ways: (a) on all four sides; (b) on the top and bottom of the wall (c) on the top, bottom and one vertical side of the wall (d) on the bottom and two vertical sides of the wall.

The in - plane strains of masonry walls are so low that the increase of allowable strains by a factor of 1.67 has not been considered for any load combination. The acceptance criteria for strains are established sufficiently conservative for in plane effects alone that a reasonable margin still remains for out-of-plane loading. Since the walls do not carry a significant part of the associated story shear, and their stiffness is extremely difficult to define and since the experimental evidence to date demonstrates that the apparent in-plane strength of a masonry wall depends heavily upon the in-place stress boundary conditions, load or stress on the wall is not a reasonable basis for acceptance criteria for in-plane effects.

3.8A.5 LOAD COMBINATIONS AND ALLOWABLE STRESSES

The loads that are imposed on the masonry walls are:

a) Dead Load (D) - This includes the weight of the wall and of structures or equipment supported by the wall. The attachment loads as shown on the field inspection sheets are due to conduits, pipes, junction boxes, switches and transformers.

b) Seismic Loads

   1) Feqo - This is the load generated by the operating basis earthquake (OBE) specified for the site of the plant and developed for the wall by the dynamic analyses performed for the building. In-plane and out-of-plane loadings and the effects of lateral displacements of wall ends relative to each other are considered.

   2) Feqs - This is the load generated by the safe shutdown earthquake (SSE) specified for the site of the plant, and developed as described for OBE.

c) Pressure Load (Pa) - This is the pressure equivalent static load within the masonry wall compartment caused by a postulated pipe rupture.

There are five possible load combinations when combining the above four (4) different individual loads:

1) Severe Environmental Condition = D + OBE

2) Extreme Environmental Condition = D+ SSE

3) Abnormal Condition = D + 1.5 Pa

4) Abnormal/Severe Environmental Condition = D +1.25 Pa + 1.25 OBE

5) Abnormal/Extreme Environmental Condition = D + Pa +SSE

It has been found that combination (1) is generally more critical than combination (2) and that combination (3) is more critical than combination (4) or (5).

The allowable stresses for combination (1) are "S" as defined in Table 3.8A-1. The allowable stresses for combinations (2) through (5) are "U" as defined in Table 3.8A-1.

Colinear stresses due to in-plane and out-of-plane seismic loads are combined by square root of the sum of the squares (SRSS).

Stresses due to attachment loads and pressure loads are combined with wall inertial loads by absolute sum.

3.8A.6 ATTACHMENTS TO WALLS

The attachments to masonry walls are typically by means of two plates and through bolts. Equipment (piping, conduits, etc) is supported by rigid structural framing which is then welded to the plates. Two typical types of support framing and their calculations and all types of wall attachment plates are contained in Reference 1.

The attachment loads are input as masses in the model for frequency analysis. Their seismic loads are obtained by the wall masses multiplied by their corresponding earthquake coefficients in both horizontal and vertical directions when performing the static analysis. Two critical support framings were selected for frequency analysis. They were found to be in the rigid region and therefore no amplification was made in the static analysis. Furthermore, the heaviest attachment is approximately 500 lbs, which is very small compared to the wall weight. The local stresses of the masonry wall due to the attachments are determined locally and added to the global stresses as illustrated in the sample calculation.[(1)]

TABLE 3.8A-1

ALLOWABLE STRESSES IN REINFORCED MASONRY

	S		U	
Description	Allowable (psi)	Maximum (psi)	Allowable (psi)	Maximum (psi)
Compressive				
Axial [(1)]	0.22 f'_m	1000	0.55 f'_m	2500
Flexural [(2)]	0 33 f'_m	1200	0.825 f'_m	3000
Bearing				
On Full area	0.25 f'_m	900	0.625 f'_m	2250
On One-third area	0.375 f'_m	1200	0.94 f'_m	3000
Shear				
Flexural members [(2)]	1.1 $\sqrt{(f'_m)}$	50	1.43 $\sqrt{(f'_m)}$	65
Shear Walls [(3, 4)]				
Masonry Takes Shear				
M/Vd > 1	0.9 $\sqrt{(f'_m)}$	34	1.17 $\sqrt{(f'_m)}$	44
M/Vd = 0	2.0 $\sqrt{(f'_m)}$	74	2.6 $\sqrt{(f'_m)}$	96
Reinforcement Takes Shear				
M/Vd > 1	1.5 $\sqrt{(f'_m)}$	75	2.25 $\sqrt{(f'_m)}$	112
M/Vd = 0	2.0 $\sqrt{(f'_m)}$	120	3.0 $\sqrt{(f'_m)}$	180
Reinforcement				
Bond (6)				
Deformed Bars		160		208
Tension				
Grade 60		24,000		48,000
Joint Wire		5F_y		0.75F_y
Compression	Not to exceed 30,000			
Grade 60		24,000		48,000
Structural Steel	Per AISC Code		1.6 x AISC allowables	0.96 Fy. or 0.90 times buckling stress

Notes to Table 3.8A-1.

(1) These values should be multiplied by $(1-(h/40t)^3)$ if the wall has a significant vertical load.

(2) This stress should be evaluated using the effective area shown on Figure 3.8A-2 except as provided in Paragraph 7.2.1(a).

(3) Net bedded area shall be used with these stresses.

(4) M is the maximum bending occurring simultaneously with the shear load V at the section under consideration. d is the depth from compression face of wall to centroid of tensile reinforcement. For M/Vd values between 0 and 1 interpolate between the values given for 0 and 1.

(5) f'_m = specified compressive strength of masonry per ASTM C90 Grade N = 800 psi (on net cross-sectional area).

f_y = specified yield strength of joint wire material.

(6) Alternately, use development length 1d = 0.0015 $d_b f_s$, where d_b = nominal diameter of bar or wire in inches, f_s = calculated stress in reinforcement in psi. 1_d shall not be less than 12".

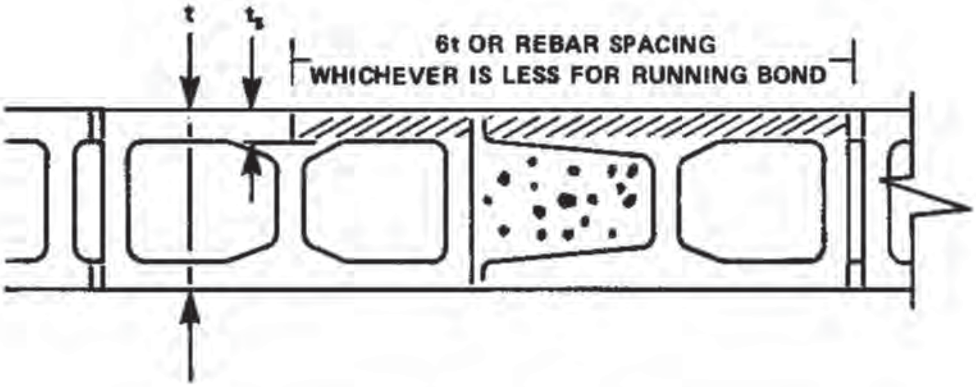
SECTION 3.8A: REFERENCE

1) Dr R E Uhrig (FP&L) to Mr D Eisenhut (NRC), "Masonry Walls," L-82-459, dated October 27, 1983.

Refer to Dwg.
2998-G-820 SH 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR AUXILIARY BUILDING
BLOCK WALLS
SHEET 1
FIGURE 3.8A-1



AREA ASSUMED EFFECTIVE IN FLEXURAL COMPRESSION
FORCE NORMAL TO FACE

FLORIDA POWER & LIGHT COMPANY
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EFFECTIVE AREA FOR
FLEXURAL COMPRESSION

FIGURE 3.8A-2

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

Quality Group A components as listed in Table 3.9-1 are designed, fabricated and installed in accordance with ASME Code, Section III requirements. A list of operating transient conditions utilized in the design and fatigue analysis is presented in Tables 3.9-2, 3.9-3A, 3.9-3B, and 3.9-3C for the Quality Group A components. Loading conditions for the design of ASME Code Class 2 and 3 components, as applicable, are discussed in Subsection 3.9.3. The applicable design transients that are contained in the ASME Code-required "Design Specification" for the reactor coolant pressure boundary components are also addressed. The transients specified represent conservative estimates for design purposes only and do not purport to be accurate representations of actual transients, or necessarily reflect actual operating procedures. However, the envisioned actual transients are accounted for, and the number and severity of the design transients, exceeds those which may be anticipated during the life of the plant. The transients include, as a minimum, plant start-up/shutdown conditions, power level changes, hydrostatic testing, system and component malfunction and reactor trip. A conservative number of events is specified for each transient. (Note: differences exist between the cycles and transients assumed in the design of Unit 1 and those assumed in the design of Unit 2. Further, there may also be unit differences with respect to those cycles and transients required by plant procedure to be tracked). Table 3.9-3 provides the above information for the Quality Group A equipment along with the appropriate classification as either Normal (N), Upset (U), Emergency (E), Faulted (F) or Testing (T), conditions. The following definitions are utilized in determining the appropriate classifications: EC289971

<u>Normal</u> (N) - Normal conditions are those conditions in the course of system start-up, operation in the design power range, hot standby and system shut-down, other than Upset, Emergency, Faulted or Testing conditions.

<u>Upset</u> (U) - Upset conditions are 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 a 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 abnormal incidents not resulting in a forced outage, and also forced outages for which the corrective action does not include repair of mechanical damage.

<u>Emergency</u> (E) - Emergency conditions are those deviations from Normal conditions which require shutdown for correction of the conditions or repair of damage in the system. These conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system.

<u>Faulted</u> (F) - Faulted conditions are those combinations of conditions associated with extremely low probability; ie, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

Test (T) - Test conditions are those tests in addition to the 10 hydrostatic or pneumatic tests[*][†] permitted by ASME Code, Section III, including leak tests or subsequent hydrostatic tests. EC284513

The appropriate loading combination and stress limits for each of the above conditions are discussed in Subsection 3.9.3.1.

In support of the design of each Quality Group A component, a fatigue analysis of the combined effects of mechanical and thermal loads is performed in accordance with the requirements of ASME Code, Section III. The purpose of the analysis is to demonstrate that fatigue failure does not occur when the components are subjected to typical dynamic events which may occur in the power plant.

The fatigue analysis is based upon a series of dynamic events depicted in the respective component specifications. Associated with each dynamic event is a mechanical, thermal-hydraulic transient presentation along with an assumed number of occurrences for the event. The presentation is generally simple and straightforward, since it is meant to envelop the actual plant response. The intent is to present material for purposes of design. A best-estimate representation of the expected plant dynamic response is neither intended nor appropriate. The fundamental concept is to ensure that the consequences of the normal and upset conditions which are expected to occur in the power plant are enveloped by one or more of the dynamic event portrayals in the component specifications. The number of occurrences selected for each dynamic event is considered to be conservative, so that in the aggregate a 60-year useful life is provided by this design process.

A stress analysis is performed on Quality Group A piping in accordance with the ASME Code, Section III, 1971 edition and all addenda up to and including Summer 1973 addenda. A stress report is developed in accordance with Section NB of ASME Code, Section III. The Quality Group A piping is listed in Table 3.9-1 along with Group A Components.

The Quality Group A components listed in Table 3.9-1 are analyzed with the appropriate loading combinations of pressure, temperature and flow transients for the normal, upset, emergency, faulted and test conditions. Design load combinations and stress limits for the above components are given in Subsection 3.9.3.

Quality Group A piping is classified as seismic Category I and is analyzed as such. The operating basis earthquake (OBE) loading is considered to occur five times over the plant life with 40 cycles for each event. One safe shutdown earthquake (SSE) event is assumed to occur for Quality Group A piping for the life of the plant.

The ASME Quality Group A valves are designed in accordance with Article NB-3000 of ASME Code, Section III. The Quality Group A valves are as listed in Table 3.9-1. When required by ASME Code, Section III the Quality Group A valves are designed for the cyclic loading conditions shown in Table 3.9-2. The system transients for valves are supplied to the manufacturers. The manufacturers perform cyclic and transients analyses in accordance with the ASME Code. A stress report is submitted by each manufacturer to demonstrate that the requirements of Subarticle NB-3500 are satisfied.

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[*] Per EC 284419, no additional secondary-side hydrostatic tests are permitted on the 2B RSG.
[†] Per EC 284513, no additional primary-side hydrostatic tests at the pressure/temperature conditions in Table 3.9-2 are permitted on the 2B RSG. EC284513

3.9.1.2 Computer Programs Used in Analyses

3.9.1.2.1 A/E Supplied Systems Components and Piping

3.9.1.2.1.1 Pipe Stress Analysis Programs

a. Description

Several computer programs are utilized for piping stress analysis. Examples of the programs used, but not necessarily limited to, are PIPESTRESS2010, PS&CAEPIPE and SUPERPIPE. These programs are capable of performing linear elastic analysis of three- dimensional piping systems including multiple branches and closed loops. Pipes are modeled using the load-deflection relationships based on the displacement method.

b. Application

The programs have the following capabilities:

1. Stress calculation in conformance with either

   (a) American National Standards Institute B31.1 Piping Code, 1967 Edition, or

   (b) ASME Code, Section III 1974 Edition for Code Class 1, up to and including Summer 1975 Addenda, (For piping stress analysis purposes, the 1971 Edition including Summer 1973 Addenda is identical to the 1974 Edition), or

   (c) ASME Code, Section III, 1971 Edition for Quality Groups B and C up to and including Winter 1972 Addenda (For piping stress analysis purposes, the Winter 1972 Addenda is identical to the Summer 1973 Addenda).

2. Static analysis for loading conditions due to pressure, applied loads, thermal expansion, dead weight, support movement, differential settlement, and seismic acceleration.

3. Frequency analysis of lumped mass model to compute natural frequencies and mode shapes.

4. Response analysis uses floor response spectra data to compute inertial forces at mass points for each vibrational mode. These forces are applied to the structure and solved as a static loading. The resulting forces and moments for individual modes are then combined.

5. Thermal transient analysis using the finite difference approximation to find thermal gradients in the pipe walls, due to step or ramp temperature changes. The programs determine the time during each transient when the various stress terms are maximized, by an iterative technique.

6. Combination cases to combine components of forces, moments and deflections from independent loading conditions, using a choice of methods:

   (a) algebraic addition,

   (b) addition to absolute values,

   (c) square root of sum of squares, or

   (d) addition in the direction of a specified loading case.

7. Fatigue analysis as prescribed in ASME Code, Section III, for Quality Group A piping. The forces and stresses due to cyclic load sets are found, in order to determine the cumulative usage factor, which must not exceed 1.0 for an acceptable design.

c. Verification

The programs solutions to ASME sample problems have been compared with the solutions to the same sample problems generated by similar, independently written programs in the public domain, namely, ANSYS[(1)], PIPESD[(2)] and ADLPIPE[(3)]. The comparison shows the programs results to be substantially identical to results generated by the above programs and by hand calculations.

3.9.1.2.1.2 EQLOADFACT 2423

a. Description

The purpose of this program is to develop response spectra using the time history response of the structure to specified earthquake excitations, to calculate dynamic load factors at safety and relief valves, and to calculate the dynamic response of a piping system to combinations of time dependent forces or to combinations of time dependent accelerations. Solutions are limited to small elastic deformations resulting from a transient force or acceleration input.

b. Application

This program is used to find the dynamic response of a lumped parameter piping system to a variety of transient forces or accelerations and to furnish dynamic solutions when modal - response techniques are inadequate. Specifically, it is used to find dynamic load factors for relief valves and for response spectra generation for main steam and feedwater lines.

c. Verification

EQLOADFACT 2423 is a time - history dynamic analysis program which has the same pedigree as the Ebasco program code PLAST 2267. While PLAST2267 performs a non-linear dynamic analysis, the EQLOADFACT 2423 code is limited to elastic analysis solutions. The details of the analytic description of PLAST 2267 are found in the Ebasco Topical Report ETR-1002[(4)] , "Design

Considerations for the Protection from Effects of Pipe Rupture." The validation cases described in ETR-1002 for PLAST 2267 for the elastic solution are applicable to EQLOADFACT 2423.

3.9.1.2.1.3 MRI/STARDYNE

a. Description

The MRI/ STARDYNE[5] Analysis System consists of a series of compatible digital computer programs designed to analyze linear elastic structural models using the "Finite Element" method. The basic concept of the finite element method is that every structure may be considered as a mathematical assemblage of individual structural components or elements. There must be a finite number of such elements, interconnected at a finite number of nodal points. The behavior of this finite element structural model closely approximates the behavioral characteristics of the real structure.

The finite element method is essentially a generalization of standard structural analysis procedures which permits the calculation of stresses and deflections in two-dimensional and three-dimensional structures, such as plates and shells, by the same techniques which are applied in the analyses of ordinary frame structures.

The STARDYNE system can be used for the static and dynamic analysis of structures subjected to any arbitrary thermal, static, or dynamic loading. The dynamic loading may include transient, steady-state harmonic, random and shock spectra excitation types. The STARDYNE system computes the structural deformations and member loads and stresses caused by the applied loads.

The finite elements used are the beam and plate elements. The primary assumptions of the beam element are: linear elastic behavior, plane sections remain plane, no coupling of axial, torque and bending, and the inclusion of shear distortions. The assumptions of the plate element are: linear elastic behavior, no coupling of in-plane and bending loads, neglecting normal stress and displacement defined by three translations and two rotations at each node.

For further information refer to CDC Publication No. 76069900, "MRI/ STARDYNE User Information Manual." This program is also described in Subsection 3.8.3.4.1.1.

b. Application

The MRI/STARDYNE program is used in the analysis of the main steam trestle, cable tray restraints, HVAC duct restraints, equipment supports and pipe whip restraints.

For the main steam trestle, the program is used to obtain the response of the structure subjected to dynamic loads. For the cable tray and HVAC duct support structures, the program is used for the eigenvalue/eigenvector analyses to obtain the structural natural frequencies and the responses of the tray-restraint system due to the seismic loads. For the equipment supports and pipe whip restraints, the program is used to analyze the response of the structure subjected to thermal and static mechanical loads.

c. Verification

MRI/STARDYNE is in the public domain and further verification is not required.

3.9.1.2.1.4

The following computer programs are specialty programs used by an A/E for supplied systems and components:

PITRUST-PC

PITRUST-PC (ME-094.01) is a computer program, developed by Shaw-Stone & Webster. This program calculates local stress intensity at the junction of two cylindrical vessels. The calculated stresses, including those due to pressure, are determined for the run cylinder. The program has application where a trunnion is welded to a run-pipe or where a branch pipe exits from a vessel or run-pipe.

The method and theory of calculating stresses follows that promulgated by the Welding Research Council Bulletin No. 107 (Wichman et al 1965, 1979.) The program is capable of complying with the requirements of ASME Boiler and Pressure Vessel Code - Section III - Nuclear Power Plant Components and ANSI-B31.1 Power Piping.

PITRUST-PC input consists basically of program control options, run-pipe dimensions, internal operating pressure, trunnion outside diameter, and loading specification.

Program output tabulates the applied loadings and local stresses at the junction of the trunnion and run-pipe.

PSPECTRA-PC

The PSPECTRA-PC computer program (ME-164.01), (Version 1, Level 00), is a computer program developed by Shaw-Stone & Webster. It is used to envelope the OBE (Operating Basis Earthquake) and DBE (Design Basis Earthquake) seismic response spectra curves from St. Lucie Unit 1 and Unit 2 Pipe Stress Analysis Criteria Document (STD-C-004.) The PSPECTRA-PC program is run on a stand-alone personal computer. The program was bench marked and verified for that computer at the time of installation.

RELAP5

The RELAP5 computer code was used to determine forcing functions for the Pressurizer Power Operated Relief Valve (PORV) opening event.

The RELAP5 computer code is a PC based QA Category 1 light water reactor (LWR) transient analysis code developed at the Idaho National Engineering Laboratory (INEL) for the Nuclear Regulatory Commission.

The RELAP 5 program is a highly generic code that, in addition to calculating the behavior of a reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and non-nuclear systems involving mixtures of steam, water, non-condensable, and solute.

STEHAM-PC

The STEHAM-PC program, ME-167.01 is a computer program, developed by Shaw-Stone & Webster. It was used to determine forcing functions for a main steam isolation valve (MSIV) closure event and a turbine stop valve (TSV) closure / steam bypass valve closure fluid transient event.

The STEHAM-PC computer program is a generalized fluid transient analysis code that is used to perform steady state and transient analyses of a steam filled flow network. The program has the capability to model any compressible fluid flow network containing valves, safety / relief valves, reservoirs, branch piping, and steam chests. The steam is modeled as an ideal gas with homogenous and adiabatic fluid properties.

WATHAM-PC

The WATHAM-PC program, ME-168.01 is a computer program, developed by Shaw-Stone & Webster. It was used to determine forcing functions for the feedwater regulatory valve and isolation valve closure and feedwater pump trip events.

The WATHAM-PC computer program is a generalized fluid transient code that is used to perform transient analysis of a water filled flow network due to pump start-up, pump trip and valve opening and closing. The program has the capability to model any incompressible fluid flow network containing in-line and discharge pumps, reservoirs, branch piping, check valves, air inlet valves, in-line and discharge valves, trapped air pockets and voids.

PC-PREPS

PC-PREPS (ME-323) is a PC-based, integrated pipe support analysis software package developed by Shaw-Stone & Webster. It is interactive, menu-driven, with built-in structural analysis and graphics capabilities. This package is totally self-contained, except for a word processor used for the final Calculation document production. All operations, including the finite element analyses, are performed on the Personal Computer.

The package allows a pipe support analyst to prepare data, view associated graphics, and execute frame and baseplate analyses. It can automatically perform load combinations and convert loads computed with Pipe Stress software to the pipe support frame, and the frame to any of the defined baseplates. The post-processing capabilities of PC-PREPS include AISC and NF17 Code Checks and maximum displacement checks, Weld Stress Check, and Local Stress check. In addition, PC-PREPS can perform auxiliary pipe support calculations to relieve the analyst from carrying them out manually.

3.9.1.2.2 NSSS Systems and Components

3.9.1.2.2.1 Reactor Coolant System

The following subsections provide a summary of the applicable computer programs used in the structural analyses for Quality Group A systems, components, and supports. The summaries include individual descriptions and applicability data. The computer programs employed in these analyses have been verified in conformance with design control methods, consistent with 10 CFR 50, Appendix B.

3.9.1.2.2.1.1 TMCALC*

See Subsection 3.7.3.1.2.3

3.9.1.2.2.1.2 FORCE*

See Subsection 3.7.3.1.2.3

3.9.1.2.2.1.3 Closure Head or Bottom Head Penetration Reinforcement Program

This program calculates reinforcement available and reinforcement required for penetration in hemispherical heads. The technique described in Paragraph NB-3332 of the ASME Code, Section III is used.

This program is used to perform preliminary sizing and reinforcement calculations for hemispherical heads in the reactor vessel.

3.9.1.2.2.1.4 Flange Fatigue Program BCH10102

This program computes the redundant reactions, forces, moments, stresses and fatigue usage factors in a reactor vessel head, head flange, closure studs, vessel flange, and upper vessel wall for pressure and thermal loadings. Classical shell equations are used in the interaction analysis.

This program is used to perform the fatigue analysis of the reactor vessel closure head and vessel flange assembly.

3.9.1.2.2.1.5 Nozzle Fatigue Program BCH10105*

This program computes the redundant reactions, forces, moments and fatigue usage factors for nozzles in cylindrical shells.

This program is used to perform the fatigue analysis of reactor vessel nozzles and steam generator feedwater nozzle.

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* This computer code does not apply to the replacement steam generators (RSGs). The corresponding RSG computer code is described in Section 2.8.8 of the RSG Report, 77-5069878-004 (Reference 45), if applicable. "Methods of Evaluation" within the scope of 10 CFR 50.59(c)(2)(viii) that are used for specific analyses for the RSG computer codes are addressed in the RSG Report.

3.9.1.2.2.1.6 Edge Coefficients BCH10026

This program calculates the coefficients for edge deformations of conical cylinder and tapered cylinders when subjected to axisymmetric unit shears and moments applied at the edges.

This program is used to perform the fatigue analysis of reactor vessel wall transition.

3.9.1.2.2.1.7 Generalized 4 x 4 BCH10124

This program computes the redundant reactions, forces, moments, stresses and fatigue usage factors for the reactor vessel wall at the transition from a thick to thinner section and at the bottom head juncture.

This program is used to perform fatigue analysis of reactor vessel bottom head juncture.

3.9.1.2.2.1.8 Load Transfer Program BCHEP007

This program transfers input loads to the nozzle cross-section being evaluated. For cross-sections outside the limit of reinforcement, the loads are applied and stresses calculated in one degree increments around the perimeter of the nozzle. This determines the worst load combination. For the nozzle-to-vessel juncture, the stresses are calculated in 90 degree increments, i.e., longitudinal and circumferential planes with respect to the reactor vessel.

This program is used to perform structural and faulted analysis of the reactor vessel.

3.9.1.2.2.1.9 Secondary Nozzle Stresses, BCHEP006

This program calculates stress intensities in the control element drive mechanism (CEDM) and instrument nozzles for evaluation of range-of-stress and fatigue.

This program is used to perform the fatigue analysis of the reactor vessel CEDM and instrument nozzles.

3.9.1.2.2.1.10 ANSYS*

This is a large-scale, general-purpose, finite element program for linear and nonlinear structural and thermal analysis. This program is in the public domain. Additional descriptive information on this program is provided in Subsection 3.9.1.2.2.2.2.

This program is used for numerous applications for all components in the areas of structural, fatigue, thermal and eigenvalue analysis.

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* This computer code does not apply to the replacement steam generators (RSGs). The corresponding RSG computer code is described in Section 2.8.8 of the RSG Report, 77-5069878-004 (Reference 45), if applicable. "Methods of Evaluation" within the scope of 10 CFR 50.59(c)(2)(viii) that are used for specific analyses for the RSG computer codes are addressed in the RSG Report.

3.9.1.2.2.1.11 Mare Island Computer Program, MEC-21/MECOL

This program is used for piping flexibility checks and to do a vibration analysis on the pressurizer heaters. This program is in the public domain and further verification is not required.

This program is used in numerous piping applications.

3.9.1.2.2.1.12 Reinforcement Analysis of Skewed Penetrations and Non-Radial Nozzles, BC101047

This program is designed to compute the limits of compensation for penetrant openings that are non-radial or skewed to a spherical head. The program is used as an aid in satisfying the requirements of ASME Section III.

This program is used in the preliminary sizing and reinforcement calculations for the pressurizer hemispherical heads.

3.9.1.2.2.1.13 Primary Plus Secondary and Peak Stresses for the Pressurizer Manway, BC10324

The program is designed to compute and tabulate the primary plus secondary stresses and the peak stresses in the manway assembly. The program is used as an aid to satisfy the requirements of ASME Code, Section III.

This program is used in the fatigue analysis of the pressurizer manway.

3.9.1.2.2.1.14 The Structural Analysis for Partial Penetration Nozzles, Heater Tube Plug Welds, and the Water Level Boundary of the Pressurizer Shell, BC10301

This program computes various analytical parameters, primary plus secondary stresses and stress intensities, peak stresses and stress intensities and the cyclic fatigue analysis with usage factors at cuts of interest. This program is utilized to satisfy the requirements of ASME Code, Section III.

This program is used in the fatigue analysis of partial penetration nozzles in the pressurizer and piping.

3.9.1.2.2.1.15 Nozzle Primary Plus Secondary Stress Range Check, Peak Stress Calculation and Maximum Usage Factor Location, BC10192

This program is designed to combine the required stress components to develop the primary, plus secondary stress intensities and determine their maximum ranges. The program also calculates peak stresses and stress intensities and develops a usage factor guide.

This program is used in the fatigue analysis of nozzles in the pressurizer and piping.

3.9.1.2.2.1.16 A Three Variable Summation Program for Computing Thermal Stresses BC10126

The program is a three variable computer program which evaluates for each time (transient) and location along nozzle, the summation of the constant thermal stress term (K) and the product of H (thermal force) and M (thermal moment) by their respective coefficients, $C_1$and C_2.

This program is used in the fatigue analysis of nozzles in the pressurizer and piping.

3.9.1.2.2.1.17 Seal, Shell II Code*

This program computes stresses and deformations of axisymmetric shells for pressure and thermal loads.

This program is used in the fatigue analysis of various nozzles in the pressurizer, piping and steam generator.

3.9.1.2.2.1.18 ICES/STRUDL II*

General purpose, finite element program for framed structures and continuous mechanics problems. Additional descriptive information on this code is provided in Subsection 3.9.1.2.2.2.1.

This program is used in the eigenvalue analysis of piping and component internals.

3.9.1.2.2.1.19 Primary Structure Interaction, BC10223*

This program calculates redundant loads, stresses, and fatigue usage factors in the primary head, tubesheet, secondary shell, and stay cylinder for pressure and thermal loadings.

This program is used in the fatigue analysis of the steam generator primary structure.

3.9.1.2.2.1.20 Tube-to-Tubesheet Weld, BC10362*

This program performs a three body interaction analysis of the tube-to-tubesheet weld juncture. The program calculates primary, secondary, and peak stresses and computes range of stress and fatigue usage factors.

This program is used in the fatigue analysis of steam generator tube-to-tubesheet welds.

3.9.1.2.2.1.21 Support Skirt Loading, BC10286*

This program calculates the stresses in the conical support skirt of the steam generator for external loads.

This program is used in the structural analysis of steam generator support skirt.

3.9.1.2.2.1.22 Principal Stress Program, BC10210*

This program sums stresses for three load conditions and computes principal stress intensity, stress intensity range, and fatigue usage factor.

This program is used in the fatigue analysis of steam generator components.

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* This computer code does not apply to the replacement steam generators (RSGs). The corresponding RSG computer code is described in Section 2.8.8 of the RSG Report, 77-5069878-004 (Reference 45), if applicable. "Methods of Evaluation" within the scope of 10 CFR 50.59(c)(2)(viii) that are used for specific analyses for the RSG computer codes are addressed in the RSG Report.

3.9.1.2.2.1.23 OUTRND Program*

This program calculates the bending stresses in an out-of-round cylinder subjected to internal pressure. The application of this program is limited to evaluation of secondary shell out-of-round deviation exceeding the ASME Code allowables.

This program is used for fabrication deviations on steam generator shells.

3.9.1.2.2.1.24 Nozzle Load Resolution, BC10211*

A special purpose program, used to calculate stresses in nozzles produced by piping loads in combination with internal pressure.

This program is used in the fatigue analysis of steam generator nozzles.

3.9.1.2.2.1.25 Analysis of Axisymmetric Solids, BCH10311*

A finite element program used to determine stresses and deformations of axisymmetric structures.

This program is used in the fatigue analysis of the steam generator secondary shell.

3.9.1.2.2.1.26 Zipper, CDC Timesharing Zipper, Siddon*

This program is used to determine the neutral axis in bending for the bolted flange of the steam generator support skirt.

This program is used in the structural analysis of the steam generator support skirt.

3.9.1.2.2.1.27 CHAT 12100*

A general purpose finite difference heat transfer program is used for steady state and transient thermal analysis.

This program is used in numerous thermal relaxation analysis for all components.

3.9.1.2.2.1.28 CEFLASH-4A*

This program is used to calculate transient conditions resulting from flow line rupture in a water/steam flow system. The program is used to calculate steam generator internal loadings following a postulated main steam line break.

This program is used in a steam line break accident structural analysis.

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* This computer code does not apply to the replacement steam generators (RSGs). The corresponding RSG computer code is described in Section 2.8.8 of the RSG Report, 77-5069878-004 (Reference 45), if applicable. "Methods of Evaluation" within the scope of 10 CFR 50.59(c)(2)(viii) that are used for specific analyses for the RSG computer codes are addressed in the RSG Report.

3.9.1.2.2.1.29 CRIB*

This program is one dimensional, two phase thermal hydraulic code, utilizing a momentum integral model of the secondary flow. This program is used to establish the recirculation ratio and fluid mass inventories as a function of power level. The program is in the public domain and further verification is not required.

This program is used for determining steam generator performance.

3.9.1.2.2.1.30 PWR Code BCH10107*

This program is used for preliminary sizing of the steam generator heat transfer area. The required number of tubes and the average tube length is calculated to satisfy the specification performance and pressure drop requirements.

This program is used for determining steam generator performance.

3.9.1.2.2.1.31 HEAT05

The HEAT05 computer program applies the finite element analysis techniques to the transient and steady state heat conduction analysis of axisymmetric solids with temperature and heat flux boundary conditions. The finite element idealization of the structure may be represented by ring elements of either triangular or quadrilateral cross-section which are interconnected along modal circles.

The solution for the temperature distribution within the structure is determined by the standard Rayleigh-Rinz procedure in which the generalized coordinates are selected as the temperatures at the nodal points of the finite element idealization. The form of the assumed temperature field within an element depends on the specific element type.

This program is used to analyze thick walled pump components subjected to internal pressure and thermal loads, and to external piping loads.

3.9.1.2.2.1.32 SOLIDS II

The computer program, SOLIDS II, applies the finite element analysis to axisymmetric solids subjected to either axisymmetric or non-axisymmetric distributed, concentrated, temperature loading. The finite element idealization represents the continuous structure by a system of ring elements which are interconnected at circumferential points or nodal circles.

Equilibrium equations are developed at each nodal circle. A solution of this net of equations for the unknown nodal circle displacements constitutes a solution for the system, since stresses within each element can be calculated from the appropriate nodal circle displacements.

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* This computer code does not apply to the replacement steam generators (RSGs). The corresponding RSG computer code is described in Section 2.8.8 of the RSG Report, 77-5069878-004 (Reference 45), if applicable. "Methods of Evaluation" within the scope of 10 CFR 50.59(c)(2)(viii) that are used for specific analyses for the RSG computer codes are addressed in the RSG Report.

The formulation of the element stiffness matrices assumes a linear displacement field to assure continuity between adjacent elements. For nonaxisymmetric loads, which are symmetric about a plane containing the axis of revolution, the formulation expands the nodal circle displacements, the temperature distribution, and the nodal circle forces in Fourier series.

This program is used to analyze the thick walled pump components subjected to internal pressure and thermal loads, and to external piping loads.

3.9.1.2.2.1.33 BJS-BJT

The BJS-BJT analysis program is a generalized computer program developed to perform complex thermal gradient and stress analysis problems. The program logic is divided into two sections, one of which performs the thermal analysis, and one which calculates stresses due to thermal distributed and concentrated loadings. Both portions of the program use a finite element method of analysis.

The thermal analysis portion determines the steady state or transient temperature distribution throughout the structure being analyzed. These temperature distributions are presented as specific temperatures specified for each nodal point in the finite element model as a function of time for each thermal condition analyzed.

The stress analysis portion of the computer program is a general purpose three dimensional linear, elastic analysis program based on the finite element displacement method.

Loads imposed on the structure to be analyzed may consist of temperature, concentrated mechanical loads, or distributed loads such as gravity and pressure. Load cases may also consist of a linear combination of node and element type loads.

This program is used to analyze thick walled pump components subjected to internal pressure and thermal loads, and to external piping loads.

3.9.1.2.2.1.34 DAGS

The computer program DAGS (Dynamic Analysis of Gapped Structure) performs a piecewise linear direct integration solution of the coupled equations of motion of a three dimensional structure which may have clearances or gaps between the structure and any of its supports or restraints (boundary gaps) or between points within the structure (internal gaps). The contacted boundary points may be oriented in any selected direction and may respond rigidly, elastically, or plastically. The structure may be subjected to applied dynamic loads or boundary motions.

The DAGS program is used to calculate the dynamic response of piecewise linear structural systems subjected to time varying load forcing functions resulting from postulated LOCA conditions.

3.9.1.2.2.2 Reactor Internals, Fuel and CEDMS

The following computer programs are used in the static and dynamic analyses of reactor internals, fuel, and CEDMs.

3.9.1.2.2.2.1 ICES/STRUDL-II*

a. Description:

The ICES/STRUDL-II computer program provides the ability to solve static and dynamic analysis of framed or solid two or three dimensional structures.

Analytic procedures in the pertinent portions of ICES/STRUDL-II apply to framed structures. Framed structures are two or three dimensional structures composed of slender, linear members which can be represented by properties along a centroidal axis. Such a structure is modeled with joints (including support joints) and members connecting the joints. A variety of force conditions on members or joints can be specified. The member stiffness matrix is computed from beam theory. The total stiffness matrix of the modeled structure is obtained by appropriately combining the individual member stiffnesses.

The stiffness analysis method of solution treats the joint displacements as unknowns. The solution procedure provides results for joints and members. Joint results include displacements and reactions and joint loads as calculated from member end forces. Member results are member end forces and distortions. The assumptions governing the beam element representation of the structure are as follows: linear, elastic, homogenous, and isotropic behavior, small deformations, plane sections remain plane, and no coupling of axial, torque, and bending. Further description is provided in Reference 6.

b. Application

The ICES/STRUDL-II code is used in the analysis of reactor internals. For reactor internals, the program is used to obtain stiffness properties of lower support structure and upper guide structure grid beams due to transverse loads. The results of the analyses are incorporated into overall reactor vessel internals models, which calculate the dynamic response due to seismic and LOCA conditions.

c. Verification

ICES/STRUDL-II is in the public domain and further verification is not required.

* This computer code does not apply to the replacement steam generators (RSGs). The corresponding RSG computer code is described in Section 2.8.8 of the RSG Report, 77-5069878-004 (Reference 45), if applicable. "Methods of Evaluation" within the scope of 10 CFR 50.59(c)(2)(viii) that are used for specific analyses for the RSG computer codes are addressed in the RSG Report.

3.9.1.2.2.2.2 ANSYS

a. Description

ANSYS is a general purpose nonlinear finite element program with structural and heat transfer capabilities. It is described in Reference 1.

b. Application

ANSYS is used to perform detailed stress analyses of the fuel assembly due to combined lateral and vertical dynamic loads resulting from postulated seismic and loss-of-coolant-accident conditions.

Static finite element analyses of reactor internal structures such as flanges, and the expansion compensating ring are performed with ANSYS to determine vertical and lateral stiffnesses.

Thermal stress analyses of the core shroud and the core support barrel cylinders for EPU conditions are performed with a three-dimensional ANYSYS finite element model.

c. Verification

ANSYS is a proprietary code in the public domain. The developers, Swanson Analysis Systems, Incorporated have published an ANSYS verification manual with numerous examples of its usage. See Reference 1.

3.9.1.2.2.2.3 SAPIV

a. Description

The SAPIV computer code is a structural analysis program capable of analyzing two and three dimensional linear complex structures subjected to any arbitrary static and dynamic loading or base acceleration. The analysis technique is based on the finite element displacement method. The structure to be analyzed can be represented using bars, beams, plates, membranes and three dimensional finite elements.

Structural stiffness and load vectors are assembled from the element matrices which are derived assuming various displacement functions within each element whereas lumped mass matrices are used to represent inertia characteristics of the structure. In the static analysis, the assembled equations of equilibrium are solved by using a linear equation solver. Dynamic analysis capabilities include modal analysis, modal superposition and direct integration methods of computing dynamic response and response spectrum techniques.

b. Application

The SAPIV code is used in the computation of dynamic response of control element drive mechanisms under mechanical and seismic loads. Both modal analysis and response spectrum capabilities of the code are used to find the natural frequencies and mode shapes and the dynamic loads in CEDM components.

c. Verification

SAPIV is in the public domain. A complete description with sample problems is given in Reference 7. Verification of the CE version has been performed to supplement the public domain documentation.

3.9.1.2.2.2.4 ASHSD

a. Description

The ASHSD program uses a finite-element technique for the dynamic analysis of complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base acceleration. The three-dimensional axisymmetric continuum is represented as an axisymmetric thin shell or as a solid of revolution, or a combination of both. The axisymmetric shell is discretized as a series of frustums of cones and the solid of revolution as triangular or quadrilateral "toroids" connected at their nodal circles.

Hamilton's variational principle is used to derive the equations of motion for these discrete structures. This leads to a mass matrix, stiffness matrix, and load vectors which are all consistent with the assumed displacement field. To minimize computer storage and execution time, the non-diagonal "consistent" mass matrix is diagonalized by adding off- diagonal terms to the appropriate diagonal terms. These equations of motion are solved numerically in the time domain by a direct step-by-step integration procedure.

The assumptions governing the axisymmetric thin shell finite element representation of the structure are those consistent with linear orthotropic thin elastic shell theory. Further description is provided in Reference 8.

b. Application

ASHSD is used to obtain the dynamic response of the core support barrel due to a LOCA. An axisymmetric thin shell model of the structure is developed. The spatial Fourier series components of the time varying LOCA loads are applied to the modeled structure. The program yields the dynamic shell and beam mode response of the structural system.

c. Verification

ASHSD has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted experimental tests or analytical results. The details of these comparisons may be found in References 8 and 9.

3.9.1.2.2.2.5 CESHOCK

a. Description

The computer program CESHOCK solves for the response of structures which can be represented by lumped mass and spring systems and are subjected to a variety of arbitrary type loadings. This is done by numerically solving the differential equations of motion for an n^{th} degree of freedom system using the Runge-Kutta-Gill technique. The equations of motion can represent an axially responding system or a laterally responding system; i.e., an axial motion, or a coupled lateral and rotational motion. The program is designed to handle a large number of options for describing load environments and includes such transient conditions as time-dependent forces and moments, initial displacements and rotations, and initial velocities. Options are also available for describing steady-state loads, preloads, accelerations, gaps, nonlinear elements, hydrodynamic mass, friction, hysteresis, and coefficient of restitution.

The output from the code consists of minimum and maximum values of translational and angular accelerations, forces, shears, and moments for the problem time range. In addition, the above quantities are presented for all printout times requested. Plots can also be obtained for displacements, relative displacements and member forces as a function of time if desired. Further description is provided in Reference 10.

b. Application

The CESHOCK program is used to obtain the transient response of the reactor vessel internals and fuel assemblies due to LOCA and seismic loads.

Lateral and vertical lumped-mass and spring models of the internals are formulated. Various types of springs; linear, compression only, tension only, or nonlinear springs are used to represent the structural components. Thus, judicious use of load-deflection characteristics enables effects of components impacting to be predicted. Transient loading appropriate to the horizontal and vertical directions is applied at mass points and a dynamic response (displacements and internals forces) is obtained.

c. Verification

CESHOCK has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted analytical results via an independent computer code. The details of these comparisons may be found in References 9 and 10.

3.9.1.2.2.2.6 SAMMSOR/DYNASOR

a. Description

SAMMSOR-DYNASOR provides the ability to perform non-linear dynamic analyses of shell structures represented by axisymmetric finite-elements and subjected to arbitrarily varying load configurations.

The program employs the matrix displacement method of structural analysis, utilizing a curved shell element. Geometrically nonlinear dynamic analyses can be conducted using this code.

Stiffness and mass matrices for shells of revolution are generated utilizing the SAMMSOR part of this code. This program accepts a description of the structure in terms of the coordinates and slopes of the nodes, and the properties of the elements joining the nodes. Utilizing the element properties, the structural stiffness and mass matrices are generated for as many as twenty harmonics and stored on magnetic tape. The DYNASOR portion of the program utilizes the output tape generated by SAMMSOR as input data for the respective analyses.

The equations of motion of the shell are solved in DYNASOR using Houbolt's numerical procedure with the non-linear terms being moved to the right-hand side of the equilibrium equations and treated as generalized pseudo-loads. The displacements and stress resultants can be determined for both symmetrical and asymmetrical loading conditions. Asymmetrical dynamic buckling can be investigated using this program. Solutions can be obtained for highly non-linear problems utilizing as many as five circumferential Fourier harmonics. Further description is provided in References 11 and 12.

b. Application

This program is used to analyze the dynamic buckling characteristic of the core support barrel (CSB) during a LOCA hot-leg break. The program's non-linear characteristics provide this capability.

A finite element model of the CSB is formulated which is consistent with the computer program. Taking into account the initial deviation of the structure and the shell mode which is most likely to give the minimum critical pressure, the time-dependent pressure load is applied to the barrel. The maximum displacement occurring in the barrel is obtained.

c. Verification

SAMMSOR/DYNASOR has been verified by demonstration that its solutions are substantially identical to those calculations obtained by hand calculations, accepted experimental test or analytical results, and results obtained with a similar independently written program in the public domain. The details of these comparisons are found in Reference 9.

3.9.1.2.2.2.7 MARC

a. Description

The MARC program is a general purpose nonlinear finite element program with structural and heat transfer capabilities. It is described in Reference 13.

b. Application

MARC is used to perform detailed stress analyses of the fuel assembly due to combined lateral and vertical dynamic loads resulting from postulated seismic and loss-of-coolant- accident conditions.

c. Verification

MARC is in the public domain. The developers, MARC-CDC, have published a set of demonstration problems, see Reference 13. Extensive verification of the CE version has been performed to supplement the public documentation.

3.9.1.2.2.2.8 MODSK

a. Description

MODSK is a CE computer program which solves for the natural frequencies and mode shapes of a structural system. The natural frequencies and mode shapes are extracted from the system of equations:

$$(K - W_n^2 M)\, \Phi_n = 0$$

Where

K = Model stiffness matrix

W_n = natural circular frequency for the n^{th} mode

M = model mass matrix

Φ_n = Normal mode shape matrix for the n^{th} mode

The solution to the general eigenvalue problem is obtained using Jacobi's method(14).

b. Application

The MODSK code is used in the analyses of reactor internals to obtain frequencies and mode shapes. The results of these analyses are incorporated into overall reactor vessel internals models, which are used to calculate dynamic response due to seismic and LOCA conditions.

c. Verification

The MODSK program was developed by CE and is used on the CDC 7600 computer. To demonstrate the validity of the MODSK program, results from lateral and vertical test problems were obtained and shown to be substantially identical to those obtained from an equivalent analysis using the public domain program ANSYS(1).

3.9.1.2.2.2.9 CEFLASH-4B

The CEFLASH-4B computer code (Reference 36) predicts the reactor pressure vessel pressure and flow distribution during the subcooled and saturated portion of the blowdown period of a Loss-of-Coolant-Accident (LOCA). The equations for conservation of mass, energy and momentum along with a representation of the equation of state are solved simultaneously in a node and flow path network representation of the primary Reactor Coolant System.

CEFLASH-4B provides transient pressures, flow rates and densities throughout the primary system following a postulated pipe break in the Reactor Coolant System.

The CEFLASH-4B computer code is a modified version of the CEFLASH-4A code (References 37-39). The CEFLASH-4A and CEFLASH-4B computer codes have been approved by the NRC (References 36, 40 and 41). The capability of CEFLASH-4B to predict experimental blowdown data is presented in Reference 36.

3.9.1.2.2.2.10 LOAD

LOAD calculates the applied forces of the axial internals model which is contained within water control volumes using results from the CEFLASH-4B blowdown loads analysis as input. The fluid momentum equation is applied to each volume and a resultant force is calculated. Each force is then apportioned to the various structural nodes contained within the volume. Use of the fluid momentum equation takes into account pressure forces, fluid friction, water weight, and momentum changes within each volume. The resultant forces are combined with the reactor vessel motions obtained from the Reactor Coolant System analysis before the structural responses are determined. The LOAD code has been verified by demonstrating that its solutions are substantially identical to those obtained from hand calculations.

3.9.1.3 Experimental Stress Analysis

Experimental stress analysis methods are not utilized.

3.9.1.4 Consideration for the Evaluation of the Faulted Condition

3.9.1.4.1 Seismic Category 1 NSSS Items

The major components of the Reactor Coolant System (RCS) are designed to withstand the forces associated with the design basis pipe breaks discussed in Subsection 3.6.2 in combination with the forces associated with the Safe Shutdown Earthquake and normal operating conditions. See Subsections 3.9.1.1 and 3.9.3 for discussions of loading combinations. Figure 3.9-24 contains a general flow diagram for the major components evaluated under the faulted condition. The forces associated with the postulated pipe breaks include pipe thrust forces at the break location resultant subcompartment differential pressurization forces, and internal asymmetric hydraulic forces acting on the reactor internals. The pipe break thrust forces are determined by the methods discussed in Subsection 3.6.2.6.1. The time and spatially dependent asymmetric hydraulic loads acting on the reactor internals are determined by the methods discussed in Subsection 3.9.2.5.

A dynamic non-linear time history analysis was performed to generate reactor vessel loads and motions due to the forces associated with the partial area pipe breaks at the reactor inlet and outlet nozzles and the steam generator inlet nozzles (see Subsection 3.6.2.1.1.3). The analysis used the DAGS code to perform a direct integration of the coupled equations of motion, in which the system characteristics are updated at each integration step to account for local non-linearities. These non-linearities include initial gaps and preloads at system restraints or local plastic response which may occur following a pipe break. The FORCE code post-processes DAGS response output in order to provide the loads and motions at pre-specified locations.

The analysis used a lumped parameter model including details of the reactor vessel and supports major connected piping and components and the reactor internals (Figures 3.9-19 through 3.9-22). This mathematical model provides a three-dimensional representation of the dynamic response of the RCS major components subjected to the simultaneous time varying pipe break forcing functions. This model is defined mathematically in terms of the ICES STRUDL II computer code to develop appropriate matrices for the elements of the three-dimensional space frame model.

The results generate reactor vessel and support loads and time history motions of RCS piping at ECCS piping juncture points, and RV shell motions at internals and CEDM support points. These motions provide input excitations for the pipe break analyses of the reactor internals fuel, CEAS, CEDMS and ECCS piping.

The components and support loads for the Steam Generator, Reactor Coolant Pump and Pressurizer were determined by equivalent static analyses. A load factor equal to 2.0 on the calculated thrust, jet impingement and subcompartment pressure loads is employed to account for the dynamic response of the structure. The model employed for static analysis is shown on Figure 3.9-18.

The system or subsystem analysis used to establish or confirm, loads which are specified for the design of components and supports is performed on an elastic basis.

When an elastic system analysis is employed to establish the loads which act on components and supports, elastic stress analysis methods are also used in the design calculations to evaluate the effects of the loads on the components and supports. In particular, inelastic methods such as plastic instability and limit analysis methods, as defined in Section III of the ASME Code, are not used in conjunction with an elastic system analysis.

Analysis of the reactor coolant system components (reactor vessel, steam generator, reactor coolant pump, pressurizer, and reactor coolant piping) and their supports have been performed in accordance with the methods described above. For each component and support member, the calculated loads, in combination with the seismic loads, are below the loads specified for design, and the stresses (piping rupture in combination with SSE) are below those allowed by Section III of the ASME B&PV code for Service Level D. Results are summarized in Tables 3.9-24, 3.9-25, 3.9-26 and on Figure 3.9-25.

3.9.1.4.2 Reactor Internals

See Subsections 3.7.3.14 and 3.9.2.5.

3.9.1.4.3 Control Element Drive Mechanisms (CEDMs)

The capability of the control element drive mechanisms (CEDMs) to withstand the effects of design basis pipe breaks in combination with safe shutdown seismic (SSE) loadings was evaluated by analysis. This dynamic loading is experienced by the CEDMs via the motion of the reactor vessel head. The reactor vessel head/CEDM motions due to pipe rupture and seismic loadings are calculated using the models described in Subsections 3.9.1.4.1 and 3.7.3.1.2.

3.9.1.4.3.1 Method of Analysis

An elastic plastic dynamic analysis was performed to determine if the St. Lucie 2 CEDMs maintain their integrity when subject to pipe breaks and SSE loadings. The motions of the RV head were input to the finite element model of the CEDM, and time history analyses were performed to determine moments, displacements, strains and stresses during the postulated events. The moments were then compared to the plastic instability moment for the most severely loaded section, which was calculated by elastic plastic static analysis on a separate detailed model, and the strains were compared to the strain capability of the CEDM material.

3.9.1.4.3.2 Models

Models for dynamic analysis considered detailed CEDM beam models - one of a CEDM with the shortest nozzle and another for a CEDM with the longest nozzle. All models were made up of beam type elements from the library available in the finite element computer program MARC.

The plastic instability moment was determined from static analysis using a model made up of shell elements, which modeled a short section of the CEDM nozzle. The nozzle at the RV head was the most severely loaded section of CEDM.

3.9.1.4.3.3 Material Properties

Recently the material properties necessary for elastic plastic analysis have been developed by the CE Metallurgical and Materials Laboratory. These properties are available for all of the materials at all of the temperatures that the CEDM normally experiences.

3.9.1.4.3.4 Loading

The effects of pipe break and SSE are transmitted to the CEDM by the motion of the reactor vessel head resulting from the analysis of Subsections 3.9.1.4.1 and 3.7.3.1.2.

A response spectrum is calculated for the motion of the reactor vessel head resulting from the primary system dynamic analysis for pipe break loads. This response spectrum is combined with the SSE response spectrum by taking the square root of the sum of the squares (SRSS) of the ordinates of the two spectra. An artificial time history of motion is then developed from the combined acceleration spectrum and used as the input to the dynamic CEDM analysis.

The loading cases performed considered acceleration spectra resulting from pipe rupture at the RV inlet, the RV outlet, and the steam generator inlet.

These load cases were applied to a CEDM with a short nozzle and to a CEDM with a long nozzle, resulting in six elastic plastic CEDM analyses.

3.9.1.4.3.5 Response

The results of the MARC elastic plastic dynamic analyses contain bending moments, deflections, stresses and strains as functions of time.

The experimentally based stress-strain material properties were used both in the static instability and dynamic analysis.

The plastic instability moment for the CEDM nozzle at the RV head determined from the static analysis is greater than 510 kip-inches.

The maximum bending moment during all cases analyzed reached 427 kip-inches in a CEDM with the shortest nozzle for the steam generator inlet guillotine. The corresponding maximum plastic strains were 6.7 percent and the maximum stresses were 48 ksi.

3.9.1.4.3.6 Evaluation

3.9.1.4.3.6.1 Acceptance Criteria

The CEDMs are not required to operate for safe shutdown after a loss of coolant event resulting from the design basis pipe breaks. In order to comply with existing ECCS analysis methods, however, the integrity of the CEDMs must be maintained and leakage must be prevented. The ASME Boiler and Pressure Vessel Code Section III Division 1 Appendix F lists a number of criteria which assure that the pressure boundary will not be violated. These criteria include strain limits for comparison to elastic plastic analysis results. The NRC standard review plan NUREG-0800 Subsection 3.6.2 recommends a strain of 50 percent of the strain at ultimate stress as a maximum acceptable limit.

3.9.1.4.3.6.2 Evaluation of Integrity

The results of each of the dynamic analyses were compared to the allowable strain limit. The plastic strain at ultimate stress for the CEDM nozzle material is 20 percent strain. The maximum strain computed, 6.7 percent, is significantly lower than half the strain at ultimate stress thereby assuring the integrity of the CEDMs for all examined load cases.

3.9.1.4.4 Other NSSS Components

The components not covered by the ASME Code but which are related to plant safety include: (1) fuel, (2) non pressure boundary portions of control element drive mechanisms (CEDMs) and (3) control element assemblies (CEAs). Each of these components is designed in accordance with specific criteria to insure their operability as it relates to safety.

3.9.1.4.5 Emergency Core Cooling System (ECCS) Piping and Supports

The capability of the Emergency Core Cooling System (ECCS) piping and supports to withstand the effects of design basis pipe breaks are evaluated by analysis. The capability of the ECCS piping and supports to withstand the combined effects of pipe break and safe shutdown seismic (SSE) loadings are also evaluated. Pipe rupture loadings are experienced by the ECCS piping via the motion of the primary system piping, and the SSE loadings are experienced by the ECCS piping via the motion of the primary system piping and the ECCS piping supports.

The primary piping motions due to pipe rupture loadings are calculated using the models described in Subsection 3.9.1.4.1. The seismic loadings are provided from the code stress analysis of the ECCS lines.

3.9.1.4.5.1 Method of Analysis

Previous studies on other CE plants(34) have indicated that the motion of the primary system piping at the ECCS injection nozzle due to pipe rupture loads contains frequencies which are in the range of the natural frequencies of the ECCS piping. A review of all pipeline geometries was therefore performed to evaluate which hot leg injection and cold leg injection lines are loaded most severely. The most severely loaded pipelines were analyzed for the effects of vibratory motion due to design basis pipe breaks.

Each ECCS pipeline evaluated was analyzed by non-linear time history dynamic elastic analysis and was evaluated according to appropriate elastic stress limits for ASME Level B and Level D conditions.

3.9.1.4.5.2 Models

The elastic dynamic analysis was performed by using distributed mass models and the appropriate ECCS nozzle motion history. The DAGS computer program was used to determine the motion history of the ECCS pipeline and the loads in the supports by performing the time history analysis.

3.9.1.4.5.3 Materials

The material used for the ECCS piping is ASME SA376 TP304H stainless steel. The elastic properties required for analysis have been taken directly from the ASME Code.

3.9.1.4.5.4 Loading

The effects of primary system pipe breaks are transmitted to the ECCS piping by the motion of the primary piping. For the evaluation of pipe break loads only, the displacement time history of the primary piping (at the ECCS injection nozzle) has been applied directly to each dynamic ECCS pipeline analysis. The displacement time history is obtained from a dynamic analysis of the reactor coolant system for postulated pipe breaks at the vessel inlet, outlet nozzles and steam generator inlet nozzle.

3.9.1.4.5.5 Response

Those pipelines evaluated to be the most severely loaded were analyzed for the effects of vibratory motion due to RCS pipe breaks. Hot leg injection and intact cold leg injection were analyzed for cold leg pipe rupture loads. Cold leg injection and intact hot leg injection were analyzed for hot leg pipe rupture loads.

The analysis resulted in motions and stresses in the piping and pipe support loads.

3.9.1.4.5.6 Evaluation

3.9.1.4.5.6.1 Acceptance Criteria

The integrity and functionability of the ECCS piping must be demonstrated. Integrity and functionability are assured if the Level B (upset condition) limits of the ASME Boiler and Pressure Vessel Code Section III, are not exceeded. If the Level B limits are exceeded, then

Level D or faulted limits may be used to demonstrate that integrity is maintained. Functionability may be assured by demonstrating that the deformations of the piping are acceptable.

3.9.1.4.5.6.2 Evaluation of Integrity and Functionability

The evaluation of the effects of pipe break loads and SSE loads is the comparison of the square root of the sum of the squares (RSS) of the stresses caused by the two loadings plus the dead weight loading with elastic stress allowable. The evaluation of piping supports is the comparison of the RSS of pipe break and SSE loads, plus normal operation loads with the allowable support loads.

A review of piping results showed that not all piping stresses satisfied Level B limits, but that all piping stresses did satisfy Level D limits, thereby demonstrating integrity of the piping. For piping that did not satisfy Level B limits, the maximum ratio of calculated stress to allowable Level B stress was 1.77. For this piping, maximum deformation was calculated. Maximum deformation, in terms of percentage reduction in flow area, was found to be only 1 percent, thereby assuring functionability. Integrity and functionability has therefore been demonstrated for all ECCS piping.

Allowable loads on piping supports were not exceeded by the combined pipe rupture, SSE and normal operation support loads, thereby confirming the integrity of the supports.

3.9.1.5 Program for Monitoring of Thinning of Pipe Walls of High Energy Carbon Steel Piping

In response to the feedwater pipe rupture event at the Surry Plant and the issuance of I&E Notice 86-106 and I&E Bulletin 87-01, a program for monitoring pipe wall thinning in carbon steel piping due to erosion/corrosion has been developed. Generally, piping wall thicknesses are monitored to ensure that code requirements for all thickness are satisfied. The program includes all moderate and high energy piping systems, both nuclear safety related and non-nuclear safety related.

Inspection locations are established in accordance with accepted industry methods such as those provided by EPRI for single and two phase systems. Within specific piping systems, locations for inspections are selected based upon such factors as fluid velocity, temperature, material composition, piping geometry, moisture content (for steam) and chemistry. Areas which are subjected to flow disturbances such as elbows, branch connections and piping and fittings downstream of control valves or flow orifices are preferred locations for inspections.

The program is designed to first inspect the most likely points for erosion/corrosion and to collect "baseline" data on other locations, with program expansion required if wall thinning in any location was more severe than anticipated. Frequency of inspection is based upon the rate of erosion/corrosion. Each operating cycle, inspection data is reviewed to determine which locations, based upon measured maximum erosion/corrosion rates, may be approaching code minimum wall thickness values.

The method of examination is selected based upon the ability to accurately provide a profile of wall thickness readings over the entire area of the piping or fitting expected to experience significant erosion/corrosion. In general, ultrasonic devices have been used for this purpose.

Decisions to take corrective action for piping and fittings which have suffered erosion/corrosion damage are based upon the ability of the piping or fitting to satisfy code minimum wall thickness requirements during the subsequent operating cycle. If the lowest wall thickness reading in a piping section less the erosion/corrosion expected during the subsequent operating cycle is less than the minimum value required by the applicable code, the piping section must be repaired or replaced.

The NRC, in Reference 44 confirms that the implemented erosion/corrosion program meets the requirements of Generic Letter 89-08.

3.9.2 DYNAMIC SYSTEM ANALYSIS AND TESTING

3.9.2.1 Preoperational Vibration, Thermal Expansion and Dynamic Testing on Piping (HISTORICAL)

Piping vibration, thermal expansion and dynamic effect testing will be conducted during preoperational and startup testing. The purpose of these tests is to confirm, by observation or measurement, as appropriate, that the piping systems, restraints, components and supports are capable of withstanding the flow-induced dynamic loadings under steady state and anticipated transient operating conditions. In addition, thermal motions are observed or monitored as appropriate to verify movements predicted by analysis and ensure that adequate clearances exist to allow the required normal thermal movement of systems, components and supports.

This testing program is designated to fulfill the requirements of Regulatory Guide 1.68, Revision 2. The following piping is included in the Test Program:

- ASME Code 1, 2 and 3 Systems
- Other high energy systems within seismic Category I structures
- High energy portions of non-safety systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable level
- Seismic Category I portion of moderate energy piping systems located both inside and outside containment.

Certain lines which fall in the categories above are exempted from testing for the following reasons:

- Line is rarely used, or when used, is not related to plant shutdown
- Line is both isolated from source of vibration and has a low momentum flow
- Line is continuously supported (e.g., buried lines)
- Line cannot be tested under the operational conditions for which it is designed during preoperational or startup testing (e.g., containment spraying headers).

Test boundaries of each system subject to test are marked on isometrics as well as corresponding allowable vibratory and thermal motion for points which are to be observed.

3.9.2.1.1 Vibrational Testing

EPU implemented a Piping Vibration Monitoring Plan to ensure that any steady state flow-induced piping vibration and thermal expansion displacements on secondary systems piping were not detrimental to the plant, piping, pipe supports or connected equipment at pre-EPU / post modification, EPU power ascension and post-EPU conditions. The piping system vibration test plan excluded valve internal vibration, unstable check valve operation, active component operation, heat exchanger vibration and primary system piping vibration.

The vibration tests are performed during those system operating modes where significant vibratory response is anticipated, based on operating experience with similar systems in nuclear power plants. Prior to implementation of the test program, a test procedure is written which contains a description of the tests, a complete listing of the systems to be tested and of the various modes of operations under which they are to he tested and the acceptance criteria for each test. For example, Table 3.9-22 gives a summary listing of possible testing modes for selected systems.

They are divided into two categories:

> Steady State - Repetitive vibrations, such as when pumps are operating, which occur for relatively long periods of time during the normal plant operations;
>
> Transient - Vibrations which occur during relatively short periods of time. Examples are single and multiple pump start, rapid valve opening or closing and safety relief valve operation.

To simplify the testing efforts, four levels of test (based on their sophistication) are identified:

3.9.2.1.1.1 Level 1 - Visual Observation Test

The purpose of this test is to visually determine the acceptability of the vibration for the piping subject to test.

Testing at Level 1 is judged sufficient to determine the acceptability of steady state and transient vibration for many cases, based on industrial experience with similar systems. This flexibility results in high allowable peak-to-peak displacements which might be easily observed visually. Locations having allowable peak-to-peak displacements in excess of 20 mils are clearly observable visually and require no specific definition of their location. All locations with allowable peak-to-peak displacements less than 20 mils are marked up on the isometrics as well as the respective distances from which these vibrations must be imperceivable to be acceptable. The distances, marked on the isometrics, are derived by determination of a visually observable maximum amplitude which results in a dynamic stress less than or equal to 50 percent of the alternating stress amplitude at 106 cycles as shown in the ASME Code. In addition to marked points, special attention is paid to observing:

a. Elbow spans and spans adjacent to elbows;

b. Spans with lumped masses such as valves and flanges;

c. Vents, drains and instrumentation lines.

Simple charts, which quickly and conservatively determine allowable peak-to-peak displacement for any piping span configuration, is provided for this purpose. If the Level 1 test procedure led to inconclusive results, a Level 2 test was performed.

3.9.2.1.2 Level 2 - Hand Held Amplitude Test

The purpose of this test is to determine the vibratory displacement of those piping segments for which Level 1 visual observations are inconclusive.

This Test Procedure is applicable for both steady state and transient conditions. A Level 2 Test, utilizing a handheld vibration indicator to measure peak-to-peak displacement, is performed at prescribed locations. The frequency range of the instruments is 5-1000 Hz and the amplitude range is appropriate with the anticipated vibration. The locations will be chosen on the basis of dividing the piping systems into a series of representative spans. A span is defined as any part of a piping system between two consecutive restraints which function in the same direction, of a cantilever. Instruction on how to break down each piping system into different span configurations is provided as part of the test procedure. The measurement locations and acceptable criteria for the different span configurations are given in the Test Procedure.

Stress amplitudes due to vibration are considered acceptable if they do not exceed 50 percent of S_a and 10^6 cycles as shown on Figure I-9 of the ASME B&PV Code, Section III 1971 edition up to and including the Summer 1973 addenda.

For low cycle (< 10^6 cycles) transient vibrations, the acceptance criteria is predicted on the following:

a. If observed displacements are such that the maximum dynamic amplitude stress does not exceed 50 percent of S_a at 10^6 cycles as shown on Figure I-9 of the ASME B&PV Code, Section III 1971 edition up to and including the Summer 1973 addenda, then the vibration is acceptable.

b. If measured displacements are larger than a) above, then:

   1. A cumulative usage factor U_v is computed from

$$U_V = \sum^{i} \frac{N_i}{N^i_{AL}}$$

where:

N_i = the effective number of cycles for each type i transient, and

N^i_{AL} = allowable number of cycles for type i transient corresponding to the alternating stress, S_i, where

S_i = where the maximum alternating stress produced by the type i transient.

i = The number of type i transients
The vibration is acceptable if $U_v \leq 0.1$.

If the test results do not meet the Level 2 acceptance criteria, then a Level 3, Hand Held Amplitude/Frequency Test will be performed for cases of steady state vibration and Level 4, Instrumentation-Stress Test for cases of transient vibrations.

3.9.2.1.1.3 Level 3 - Hand Held Amplitude/Frequency Test

The purpose of this test is to determine the vibratory and respective peak-to-peak displacements of piping segments for which the results of the Level 2 testing are inconclusive. Portable instruments are used for this test. Acceptance criteria incorporated in the same charts used for Level 2 tests and/or computer analysis used to determine dynamic stresses, based on the measurement results, enable a final conclusion regarding the acceptability of steady state vibration.

3.9.2.1.1.4 Level 4 - Instrumentation - Stress

This test is performed for those transient events for which the results of Level 3 testing are inconclusive. A time history analysis of the piping system response to the transients is performed utilizing the computer program PLAST. The location of maximum stress points, maximum displacement points and maximum restraint loads is calculated. The results given all necessary information to establish acceptance criteria with proper testing sensors. Fluid parameters are measured. A data acquisition system is used to record information during testing.

In addition, Level 4 Testing may be used for shock or pulse type transients. A computer time history analysis of the piping and support system response to the pulse is generated to optimize transducer locations. During the test, real time data is recorded for later analysis.

3.9.2.1.1.5 Corrective Action

In the unlikely event that the piping vibration exceeds the acceptance criteria for Level 3 or 4 tests, then corrective actions are initiated. Possible corrective action includes: (1) identification and reduction or elimination of the offending force, (2) detuning of resonant piping spans by appropriate modifications to the restraint system, (3) addition of bracing to stiffen the system, and (4) changes in operating procedures to eliminate troublesome operating conditions.

Following corrective action, additional testing is performed to determine if the vibrations have been sufficiently reduced to satisfy the acceptance criteria and the piping stress analysis shall be revised to include the corrective measures. Corrective action is documented in preoperational test procedures as required and is available for NRC review.

The methodology described above is summarized in the General Flow Chart in Figure 3.9-23.

3.9.2.1.2 Thermal Expansion Testing

Thermal expansion testing is performed to verify that the measured movements at particular locations are approximately equal to those predicted by analysis and to ensure that the piping is not restrained due to interferences with other components.

Prior to the implementation of the testing program, a test procedure is written identifying systems to be tested and expected movements at those chosen points. A rationale is provided for the choice of measurement points. Information concerning inspection and testing of snubbers is contained in the Technical Specifications 3/4.7.9.

3.9.2.2 Seismic Qualification Testing of Safety Related Mechanical Equipment (HISTORICAL)

Equipment specifications for seismic Category I mechanical components contain requirements for seismic testing or analysis. Seismic forces in the horizontal and vertical direction are determined by estimating the amplification of each floor acceleration due to the operating and design basis earthquakes. The seismic loads are forwarded to the equipment manufacturer in the form of OBE (horizontal and vertical) and SSE (horizontal and vertical) floor response spectra at various floor elevations. The manufacturer is required to demonstrate that the equipment and equipment supports do not suffer loss of function under the maximum seismic loads.

Seismic Category I equipment is qualified to ensure its structural and functional integrity (when required) when subjected to either the OBE or the SSE as specified by the loading combinations of Subsection 3.9.3. In order to demonstrate operability there is no differentiation made between equipment that is required to function during or after an earthquake. The methods and procedures used and the results of tests and analyses that confirm the implementation of the seismic qualification program for safety related mechanical equipment are provided in:

Appendix 3.9A - Operability Considerations for Seismic Category I Active Pumps and Valves.

Appendix 3.10A - Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under

Operational Flow Transients and Steady-State Conditions

3.9.2.3.1 Introduction

The flow-induced vibration of the core support barrel system during normal operation can be characterized as a forced response to both deterministic (period and transient) and random pressure fluctuations in the coolant. Methods are developed to predict the components of the hydraulic forcing function and the response of the reactor internals to such excitation. An analytical method based on a theoretical solution of the appropriate hydrodynamic differential equations for a mathematically tractable geometry is used to develop the periodic hydraulic forcing function. The random component of the load is developed by analytical and experimental methods.

Normal operating hydraulic loads are not affected by EPU.

Normal operating hydraulic loads on the Upper Guide Structure have been adjusted to encompass operation with either Westinghouse fuel or AREVA fuel. All other normal operating hydraulic loads on reactor internals components are not affected by the AREVA fuel transition.

The response of the core support barrel system to the normal operating hydraulic loads is calculated by finite-element techniques. The results are combined with seismic and LOCA analyses for comparison with the criteria specified in Subsection 3.9.5.

3.9.2.3.2 Periodic Forcing Function

An analysis based on an idealized hydrodynamic model is employed to obtain a basic understanding of the relationship between reactor coolant pump pulsations in the inlet ducts and the periodic pressure fluctuations on the core support barrel. The idealized model represents the annulus of coolant between the core support barrel and the reactor vessel. In deriving the governing hydrodynamic differential equation for the above model, the fluid is taken to be compressible but inviscid. Linearized versions of the equations of motion and continuity are used. The excitation of the hydraulic model is assumed harmonic with the frequencies of excitation corresponding to the pump rotational speed, 15 Hz; its first harmonic, 30 Hz; the blade passing frequency, 75 Hz; and its first harmonic, 150 Hz. The result of the hydraulic analysis is a system of equations that defined the forced response, natural frequencies, and natural modes of the hydrodynamic model. The forced-response equations define the distribution of pressure on the core support barrel system as a function of time and space. The details of this analytical procedure are given in References 18 through 20

Reference 18 shows that the magnitude of the deterministic hydraulic load is keyed to the amplitude of the inlet pressure fluctuations, and that when best estimate inlet pressure values are used to calculate the hydraulic load, the induced responses compare well with the measured data. It is also shown that the predicted and measured responses are well below the allowable stress criteria. Predictions for St. Lucie Unit 2 are made on a best estimate basis, utilizing inlet pressure values measured during the previous Maine Yankee and Fort Calhoun precritical vibration monitoring programs (PVMPs) (see Subsection 3.9.2.6.3).

3.9.2.3.3 Random Forcing Function

The random hydraulic forcing function is developed by analytical and experimental methods. An analytical expression is developed to define the turbulent pressure fluctuation for fully developed flow. This expression is modified, based upon the result of scale model testing, to account for the fact that flow in the downcomer is not fully developed. In addition, experimentally adjusted analytical expressions are developed to define the peak value of the pressure spectral density associated with the turbulence and the maximum area of coherence, in terms of the boundary layer displacement, across which the random pressure fluctuations are in phase.

3.9.2.3.4 Response Analysis

3.9.2.3.4.1 Deterministic Response

The natural frequencies and mode shapes of the core support barrel, which form the basis for all forced response analyses, are obtained through the use of the axisymmetric shell finite-element computer program, ASHSD[8] This computer program is capable of obtaining natural frequencies and mode shapes of complex axisymmetric shells; e.g., arbitrary meridional shape, varying thickness, branches, multi-materials, orthotropic material properties, etc. An inverse iteration technique is used in the program to obtain solutions of the characteristic equation, which is based on a diagonalized form or consistent mass and stiffness matrices developed using the finite- element method. Four degrees of freedom - radial displacement, circumferential displacement, vertical displacement and meridional rotation - are taken into account in the analysis giving rise to coupled mode shapes and frequencies.

A finite-element model of the core support barrel system is developed as shown on Figure 3.9-1. Evaluation of the reduction of these frequencies for the system immersed in the coolant is made by means of the “virtual mass" method. The normal mode method is used to obtain the structural response of the core support barrel to the deterministic forcing functions. Generalized masses based on mode shapes and the mass matrix from the shell finite-element computer program are calculated for each core support barrel mode of vibration. Modal force participation factors, based on the mode shapes and the predicted periodic forcing functions, are calculated for each mode and forcing function. The generalized coordinate response for each mode is then obtained through solution of the corresponding set of independent second order, single-degree-of-freedom equations. Utilizing displacement and stress mode shapes from the shell finite-element computer program, the structural response of the core support barrel for each mode is obtained by means of the appropriate coordinate transformation. Response to any specific forcing function is obtained through summation of the component modes for that forcing function.

3.9.2.3.4.2 Random Response

The random response analysis considers the response of the core support barrel system to the turbulent downcomer flow during steady-state operation. The random forcing function is assumed to be wideband stationary random process with a pressure spectral density equal to the peak value associated with the turbulence. The root-mean-square (rms) vibration level of the core support barrel system in terms of a beam mode is obtained based upon a damped, single degree-of-freedom analysis assuming the rms random pressure fluctuations to be spatially invariant. The maximum rms response calculated is considerably less than the design operating clearances available at the snubbers.

3.9.2.3.5 Transient Forcing Conditions

The transients that occur during loop startup or shutdown represent gradual transitions from one steady-state mode of operation to another taking place over many seconds. It is recognized, therefore, that no dynamic magnification of structural response will occur and no dynamic transient response calculations are required.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

The Maine Yankee and Fort Calhoun precritical vibration monitoring programs together constitute a valid prototype design for St. Lucie Unit 2. The St. Lucie Unit 2 reactor is designated as non- prototype seismic Category I designs.

In accordance with NRC Regulatory Guide 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” May 1976 (R2), prototype prediction, measurement, and inspection programs were developed and performed for the Maine Yankee and Fort Calhoun reactor internals. Theoretical prediction analyses were performed for Maine Yankee[(21)] and Fort Calhoun[(22)] to estimate the amplitude, time, and spatial dependency of the steady-state and transient hydraulic and structural responses to be encountered during precritical testing. The precritical vibration monitoring programs for Maine Yankee [(23)] and Fort Calhoun[(24)]were completed successfully. Comparisons of the measured and predicted responses for Maine Yankee and Fort Calhoun demonstrate that the theoretical prediction methods used provide accurate estimates of the steady-state response of the core support barrel system, when reasonable best estimate values for the magnitude of the inlet pressure fluctuations are used. It was concluded from these programs that flow induced vibrations of the Maine Yankee and Fort Calhoun reactor internals are well within design allowables and are acceptable for all normal, steady-state, and transient flow modes or reactor coolant pump operation.

Presented in Table 3.9-4 is a summary of the significant hydraulic and structural design parameters for each of the three reactor designs. The effects of these structural and hydraulic parameters on the flow-induced vibratory response of the reactor internals are presented in Subsection 3.9.2.6, where it is shown that the nominal differences have no significant effects on the stress levels. In general, the analysis of St. Lucie Unit 2 demonstrates that:

a. The predicted structural response of the St. Lucie Unit 2 reactor internals are well within design allowables and are acceptable for all normal, steady-state, and transient flow modes of primary coolant pump operation

b. The prototype precritical vibration monitoring programs for Maine Yankee and Fort Calhoun adequately account for the specific design features of the St. Lucie Unit 2 which are shared by the valid prototype designs.

The applicant is proceeding to implement a PVMP for St. Lucie Unit 2 consistent with the recommendations of Regulatory Guide 1.20 (R2) as it relates to non-prototype seismic Category I units. The reactor vessel internals are subjected, during the preoperational and functional testing program, to the significant flow modes of normal reactor operation for a sufficient period of time to determine whether the reactor vessel internals exhibit any unexpected vibration problems. Prior to and during the PVMP, the reactor vessel internals are subjected to full examination to detect any evidence of unanticipated or excessive vibrations. The internals are removed from the vessel for these visual and nondestructive inspections. The following points are investigated:

a. All major load bearing elements

b. Lateral, vertical and torsional restraints within the vessel

c. Locking and bolting devices

d. All other locations examined on the prototype designs and

e. The reactor vessel interior for loose parts and/or foreign material.

The results of the full examination program are separately reported in a summary report submitted to the NRC to confirm that the observed vibrational characteristics are similar to those of the prototype design.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

Dynamic analyses are performed to determine blowdown loads and structural responses of the reactor internals and fuel to postulated LOCA (Loss-of Coolant Accident) loadings and to verify the adequacy of their design. A brief description of these methods is provided below.

The LOCA maximum stress intensities in the reactor internals are determined using the combinations of lateral and vertical LOCA time-dependent loadings which result in maximum stress intensities. The maximum LOCA stresses and the maximum stresses resulting from the SSE are then combined using the SRSS method to obtain the total stress intensities.

3.9.2.5.1 Dynamic Analysis Forcing Functions

The hydrodynamic forcing functions during a postulated LOCA result from transient pressure, flow rate, and density distributions throughout the primary reactor coolant system.

3.9.2.5.1.1 Hydraulic Pressure Loads

The transient pressure, flow rate and density distributions are computed for the subcooled and saturated portions of the blowdown period during a LOCA. The computer code utilized is based on a node-flow path concept in which control volumes (nodes) are connected in any desired manner by flow areas (flow paths). A complex node-flow path network is used to model the Reactor Coolant System (RCS). The modeling procedure has been compared to a large scale experimental blowdown test with excellent agreement.

The laws of conservation of mass, energy and momentum along with a representation of the equation of state are solved simultaneously. The hydraulic transient of the reactor is coupled to the thermal response of the core by analytically solving the one-dimensional radial heat conduction equation in each core node.

Pre-blowdown steady state conditions in the RCS are established through the use of specified input quantities.

The blowdown loads model uses a nonequilibrium critical flow correlation for computing the subcooled and saturated critical fluid discharge through the break.

3.9.2.5.1.2 Drag Loads

A break in the primary coolant system will result in large local pressure differences across various reactor vessel internal components and an acceleration of the local fluid velocity in various regions. The acceleration of the local fluid velocity can result in higher component drag loads than occur during steady state reactor operation.

3.9.2.5.1.3 Core Loads

The total instantaneous load across the core is given by the summation of the pressure and drag forces acting parallel to the flow. The loads are obtained using a control volume approach utilizing an integrated fluid momentum equation. The drag forces are represented by the fluid shear term in this equation and consist of both frictional and form drag.

3.9.2.5.1.4 CEA Shroud Loads

During normal operation, the reactor coolant flows axially through the core into the upper guide structure. Within the upper guide structure, the coolant flow changes direction so that it exits radially through the hot leg nozzles. During a LOCA, the transverse flow of the coolant across the CEA shroud gives rise to loads which induce deflections in these shrouds.

The transverse drag forces were determined from flow model experiments which were geometrically and dynamically similar to the full-scale upper guide structure design. The measured experimental model forces were scaled up to represent the actual forces on the upper guide structure using the computed transient flow rate and density information.

3.9.2.5.1.5 Results of Blowdown Loads Analysis

Analysis was performed of a postulated pipe break at the reactor vessel inlet nozzle. The transient pressure differences throughout the vessel are evaluated and used in the structural response calculation described below. The pressure difference across the core is also evaluated for the break.

A postulated pipe break occurring at the reactor vessel outlet nozzle was also analyzed. The pressure difference throughout the vessel is calculated. The decompression in the annulus is symmetric early in the transient because the pressure wave must travel through the core barrel internals to reach the lower plenum from where the wave propagates uniformly up through the downcomer. The axial pressure difference across the core was also calculated.

A postulated pipe break occurring at the steam generator inlet nozzle was also analyzed. The pressure difference throughout the reactor vessel was calculated. The axial pressure difference across the core was also calculated.

3.9.2.5.2 Structural Response Analyses

The dynamic LOCA anlyses of the reactor internals and core determine the shell, beam and rigid body motions of the internals, using established computerized structural response techniques. The analyses consist basically of three parts. In the first part, the time-dependent shell response of the core support barrel to the transient loading is calculated using the finite-element computer code, ASHSD[8]. The second part of the analysis evaluates the buckling potential of the core support barrel for hot leg break conditions using the finite-element computer code, SAMMSOR- DYNASOR[11,12]. In the third part, the nonlinear dynamic time history responses of the reactor internals and core to vertical and horizontal loads resulting from hot and cold leg breaks are determined with the CESHOCK code, which is further described in Reference 10.

3.9.2.5.2.1 Shell Response of the Core Support Barrel

A cold leg break causes a pressure transient on the core support barrel that varies circumferentially as well as longitudinally. The ASHSD finite-element computer code is used to analyze the shell response of the CSB to the pressure transient from a cold leg break.

The CSB is modeled as a series of shell elements joined at their nodal point circles as shown in Figure 3.9-1. The length of the elements in each model is selected to be a fraction of the shell attenuation length.

A damped equation of motion is formulated for each degree of freedom of the system. Four degrees of freedom, radial displacement, circumferential displacement, vertical displacement, and meridional rotation are considered in the analysis. The differential equations of motion are solved numerically using a step-by-step integration procedure.

The circumferential variation of the pressure time-history is considered by representing the pressure as a Fourier expansion. The pressure at each elevation in the model is determined by linear interpolation. Thus, a complete spatial time load distribution compatible with the ASHSD computer program is obtained. Each load harmonic is considered separately by ASHSD. The results for each harmonic are then added to obtain the nodal displacements, resultant shell forces and shell stresses as a function of time.

3.9.2.5.2.2 Dynamic Stability Analysis of CSB

A hot leg break causes net external radial pressure on the core support barrel. A stability analysis of the CSB is performed using the finiteelement computer code, SAMMSOR-DYNASOR. The effects of an initially imperfect shape based on actual out-of-roundness measurements are included in the analysis.

The CSB is modeled as a series of shell elements, as shown in Figure 3.9-2. Stiffness and mass matrices for the barrel are generated utilizing the SAMMSOR part of the code. The equations of motion of the shell are solved in DYNASOR using the Houbolt numerical procedure.

An initial imperfection is applied to the core support barrel by means of a pseudoload for each circumferential harmonic considered. The actual pressure transient loading generated by the outlet break is uniform circumferentially but varies longitudinally. The response is obtained for each of the imperfection harmonics.

Appendix F, Section III of the ASME Boiler and Pressure Vessel Code requires that permissible dynamic external pressure loads be limited to 75 percent of the dynamic instability pressure loads, or alternately, the dynamic instability loads must be greater than 1.33 times the actual loads. Consequently, this analysis is repeated with the imperfection applied in the critical harmonic and the pressure loading is increased beyond 1.33 times the actual loads in order to demonstrate the stability of the core support barrel.

3.9.2.5.2.3 Dynamic System Analysis of the Reactor Internals

Dynamic analyses are performed to determine the structural response of the reactor internals to postulated asymmetric LOCA loading (including reactor vessel motion effects) and to verify the adequacy of their structural design. The postulated pipe breaks result in horizontal and vertical forcing functions which cause the internals to respond to both beam and shell modes.

Detailed structural mathematical models of the reactor internals are developed based on the geometrical design. These models are constructed in terms of lumped masses connected by beam or bar elements, and include nonlinear effects such as impacting and friction. The models are developed for input to the CESHOCK code which solves the differential equations of motion for lumped parameter models by a direct step-by-step numerical integration procedure. The model was developed according to the procedures established in Reference 9, and, in addition, include hydrodynamic coupling effects and a detailed representation of the core support barrel to upper guide structure to reactor vessel interfaces. Separate models are formulated for the horizontal (Fig. 3.9-3) and vertical (Fig. 3.9-4) directions to more efficiently account for structural and response differences in those directions.

The horizontal and vertical models which were used to determine the LOCA structural responses Of the reactor internals were nonlinear. The CESHOCK code, References 9 and 10, is used to calculate the maximum component loads that resulted from the postulated hot and cold leg breaks. These analyses considered nonlinearities such as gaps, damping, friction, hysteresis coefficient of restitution.

The models for the horizontal directions are developed in terms of lumped masses connected by beam elements. The stiffness values for the beam elements are generally evaluated using beam characteristic equations. The lumped-mass weights are based upon the mass distribution of the internals structures. Local masses such as plates and snubber blocks are included at appropriate nodes. The effect of the surrounding water on the dynamics of the internals for horizontal motion is accounted for by hydrodynamically coupling the components separated by a narrow annulus - the vessel, core barrel, core shroud, lower support structure cylinder, and upper guide structure cylinder. (The effects of system flow on the structural properties of reactor internals are secondary.) The hydrodynamic effect of these components is dominated by hydrodynamic coupling and hydrodynamic added mass. Both of these effects are considered in the dynamic response analyses of these components. (A description of CE methodology for hydrodynamic mass is presented in Reference 9. Additional references[32,33] describe this hydrodynamic mass methodology.) The clearances between the core support barrel and the reactor vessel snubbers and between the core shroud guide lugs and the fuel alignment plate is simulated by non-linear spring which account for the loads generated should impacting occur. A representation of the core is included in the internals models to provide appropriate inertial and impact feedback effects on the internals response.

The vertical model stiffness values are generally calculated using bar characteristic equations. Nonlinear couplings are included between components to account for structural interactions such as those between the fuel and core support plate, and between the core support barrel and upper guide structure upper flanges. Pre-loads, which are caused by the combined action of applied external forces, dead weights, and holddowns are also included. Friction elements are used to simulate the coupling between the fuel rods and spacer grids.

The axial and lateral models are uncoupled to provide more spatial detail to account for important structural characteristics in each model. Separate responses can be calculated because the axial and lateral modes of vibration are independent of each other since the resulting displacements are small. This technique is standard practice in the dynamic analysis of elongated symmetrical structures. The validity of the method has been further verified by analytical studies of fuel assemblies subjected to simultaneous application of axial loads and lateral displacements at magnitudes typical of LOCA responses (Reference 9). These studies concluded that no significant axial lateral interaction (beam-column behavior) results from this combined loading.

A reduced model of the reactor vessel internals (Fig. 3.9-5) is developed for incorporation into the reactor coolant system model. The detailed nonlinear horizontal and vertical internals (plus core) models are condensed and combined into a three--dimensional model compatible with the reactor coolant system model and the computer programs through which the latter model is analyzed. The purpose of this reduced internals model is to account for the effects of the internal LOCA loads on the reactor vessel support motion and the structural loading interaction between the internals and the vessel. A reduced internals model is developed to produce reactor vessel support motions and loadings equivalent to those produced by the detailed internals model.

The dynamic responses of the reactor internals to the postulated pipe breaks are determined with the CESHOCK code utilizing the detailed models. Horizontal and vertical analyses are performed for both hot and cold leg breaks to determine the lateral and axial responses of the internals to the simultaneous internal fluid forces and vessel motion excitation.

The vertical excitation of the internals is calculated by the LOAD2 computer code (Subsection 3.9.1.2.2) using the control volume method. In this method, the reactor internals are divided into volumes containing both structure and fluid or structure alone. The momentum equation is then applied to each volume, and resultant force is calculated and assigned to the structural node within the volume. This method takes into consideration pressure, fluid friction, momentum changes, and gravitational forces acting on each volume. The resulting load time histories are in a form consistent with the CESHOCK code input.

In order to achieve an initial (prior to the pipe break) equilibrium, the initial static deflections and gaps are calculated. The resulting initial conditions and load time histories are input to the CESHOCK code and the dynamic response of the model is calculated.

The horizontal input excitations resulting from a cold leg break are the core support barrel force time history and the vessel motion time history determined from the reactor coolant system analysis. The core support barrel forces are obtained by representing the asymmetric pressure distribution time history as a Fourier expansion. The two terms (sine and cosine) which excite the beam mode of vibration are then integrated over the core support barrel and transformed into nodal force time histories.

The horizontal input excitations resulting from a hot leg break are the CEA shroud crossflow load time histories and the vessel motion time history determined from the reactor coolant system analysis. The forces applied to the shroud mass points are determined directly from the blowdown pressure time history and include the drag force and forces due to the pressure differential on the shrouds.

The input to the reactor internals pipe break analysis is the reactor coolant system LOCA analysis and blow-down loads, which are not impacted by EPU conditions. Hence, the pre-EPU analysis remains applicable for EPU.

Pipe break loads on the reactor internals have been adjusted to encompass operation with either Westinghouse fuel or AREVA fuel.

Following the lateral analysis of the reactor internals, a series of detailed lateral analyses of the core are performed also using the CESHOCK code. The model used is shown in Figure 3.9-6 and the analysis was performed for 4-, 9-, 11-, and 17-bundle rows of fuel assemblies to obtain the highest bundle loads. The applied forcing functions consisted of displacement time-histories of the fuel alignment plate, core support plate, and core shroud which are calculated in the internals analyses described above. The detailed core analyses provided the maximum spacer grid impact loads and the most severe fuel assembly displaced shapes. These are used for the fuel assembly stress analysis which is described in Section 4.2.

The results from these analyses consist of time-dependent member forces, and nodal displacements, velocities and accelerations. The load and displacement responses are used in the detailed stress analyses of the internals.

Detailed stress analysis of the reactor internals has confirmed the adequacy of the structural design (see Subsections 3.9.3 and 3.9.5).

3.9.2.6 Correlation of Reactor Internals Vibration Test With Analytical Results

3.9.2.6.1 Introduction

Since the St. Lucie Unit 2 reactor has been categorized as nonprototype seismic Category I design, an analysis and full inspection program has been performed for the plant in lieu of a measurement program. The results of the analysis are presented in this section. The results of the inspection are reported separately.[43]

The analysis procedures utilized are presented in detail in References 18 through 20. Only the pertinent results are presented in the following sections.

3.9.2.6.2 Comparison of Structural and Hydraulic Parameters

Elevation views of the St. Lucie Unit 2, Maine Yankee and Fort Calhoun reactor internals are presented on Figures 3.9-8, 3.9-9 and 3.9-10, respectively. Presented in Table 3.9-4 is a summary of the significant hydraulic and structural design parameters for each of the three reactor designs. In general, the designs are similar but some variations do exist. For example, the St. Lucie Unit 2 reactor is simpler in design than either prototype in that it does not include a thermal shield.

The most significant hydraulic region is the downcomer annulus, where the coolant flow is undeveloped and highly turbulent. These reactor designs vary in three aspects with regard to the evaluation of hydraulic pressure fluctuations in the downcomer. These are the number of coolant loops, the presence or absence of a thermal shield, and the magnitude of the coolant velocity. A brief discussion of each is presented below.

The number of loops and their azimuthal relationship affects the spatial distribution of the fluctuating pressure field within the downcomer. That there is a nonuniform distribution has been shown in model tests and the Maine Yankee PVMP [27][28][29]. Using the principle of superposition it is a simple matter, having determined the pressure field in the annulus for the case of one functioning pump, to develop the fields corresponding to any azimuthal pattern of operating loops with varying phase relationships. The validity of superimposing pump effects was initially investigated during the hydraulic forcing function development. Subsequently, with actual PVMP measurements of the fluctuating pressure at various locations in the downcomer annulus for various operating conditions, the applicability of the principle was checked. Specifically, data obtained from single pump operation were combined to predict multiple pump pressures at various transducer locations. These values were compared with the actual measurements. The results[23][24] indicated an average variation from perfect correlation of less than 25 percent. The majority of the predicted values exceeded the measured values indicating conservatism in the estimates. From these results, it was concluded that the superposition principle is an acceptable procedure for developing hydraulic forcing functions in the downcomer annulus.

The random component of the hydraulic loading on the CSB could be affected by the presence of a thermal shield in the annulus. The St. Lucie Unit 2 plant has no thermal shield. However, the random hydraulic load was developed from PVMP pressure data with the shield present. The results in a conservative estimate of the loading.

A usual assumption in prediction of hydraulic fluctuations, whether of a periodic or random character, is that the magnitude is dependent on the fluid density multiplied by the square of a characteristic velocity. Data obtained in the Maine Yankee and Fort Calhoun PVMPs indicate this assumption to be valid. Comparison of the Fort Calhoun hot post-core estimate and measurement[24] indicates the validity of the assumption over a sizeable range of the postulated variables (e.g., a change in density compounded with a change in velocity). The agreement for other prototype is nearly as good despite Maine Yankee being a three-loop system. This would indicate that the effects of flow velocity in St. Lucie Unit 2 would be to reduce the hydraulic load as they have a slightly lower downcomer coolant velocity (see Table 3.9-4).

3.9.2.6.3 Deterministic Structural Response Results

Predictions of the periodic forcing functions were based on the steady-state hot core coolant conditions. All predictions were made using the best estimate of the inlet duct pressure pulsations available as derived from the Maine Yankee and Fort Calhoun PVMPs. The fundamental forcing frequencies were the pump rotational speed (15 Hz) and the blade passing speed (75 Hz). A higher harmonic of each of the fundamental frequencies was included (30 and 150 Hz respectively).

From the results of the response analysis (described in Subsection 3.9.2.3.4.1) for St. Lucie Unit 2, the maximum stress intensity is experienced in the CSB lower flange region and is below the allowable stress criteria, viz:

$$(\sigma_{max} = 3909 \text{ psi}) << (\sigma_{allowable} = 26000 \text{ psi})$$

The analytical results provide a high degree of assurance that the structural integrity of the reactor internals will be maintained during all normal operating steady-state and transient conditions of coolant pump operation.

3.9.2.6.4 Random Structural Response Results

The random response analysis considers the response of the CSB system to the turbulent component of the flow during steady-state operation. The random forcing function is assumed to be a wide-band stationary random process representing the random pressure fluctuations that result from the flow turbulence. The power spectral density (PSD) of the pressure fluctuations was estimated from a representative analytical expression modified by the results of flow model testing. The PSD used were for full design flow conditions (all pumps operating). The response of the CSB system in the beam mode at the snubber elevation was considerably less than the nominal design gap at the core barrel support - reactor vessel snubbers.

3.9.3 ASME CODE CLASS 1, 2 and 3 COMPONENTS and COMPONENT SUPPORTS (including ASME Code Class 1, 2 & 3 Piping and Pipe Supports)

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

ASME Code Class 1, 2 and 3 system components are designed in accordance with the rules and methods specified in the ASME code. The design stress limits of the ASME Code (including code cases) are selected to insure the pressure retaining integrity of safety class equipment. Code cases utilized by the A/E have been approved by Regulatory Guide 1.84 "Code Case Acceptability ASME Section III Design and Fabrication," (R9) and 1.85, "Code Case Acceptability ASME Section III Materials," (R9) is discussed in UFSAR Section 5.2.

Design transients for ASME Code Class 1 components are provided in Table 3.9-3A, 3.9-3B, and 3.9-3C. Stress limits for A/E Supplied Class 1 components are described in EC289971

Subsection 3.9.3.1.1. The stress limits and loading combinations for NSSS Supplied Class 1, 2, and 3 components are described in Subsection 3.9.3.1.3.

ASME Code Class 2 and 3 components are designed for the concurrent loadings produced by pressure, deadweight, temperature distributions, the vibratory motion of the safe shutdown earthquake (SSE), and the dynamic system loadings associated with the appropriate plant faulted condition. The design loading combinations for specific plant operating conditions are listed in Table 3.9-5 and Table 3.9-5A for Group B & C Components and Piping respectively. Additionally, an investigation was performed for all Safety Class 2 and 3 piping systems (irrespective of operating temperature) to demonstrate that the number of equivalent thermal cycles, as defined in ASME Subsection NC 3611.2, was sufficiently low to confirm the conservatism of the existing stress analyses.

In accordance with the agreement reached at a meeting with the NRC and Florida Power & Light Company on October 14, 1982 an acceptance criteria of 1000 "Realistic" cycles was employed. In conducting this analysis, the following Safety Class 2 and 3 systems were reviewed:

Reactor Coolant	Component Cooling Water
Charging	Letdown
Safety Injection	Auxiliary Feedwater
Main Steam	Containment Spray
Main Feedwater	Intake Cooling Water

A sample calculation specifying methodology and a summary of the results is provided in Table 3.9-5b.

Using realistic values of cycle frequencies, all systems were shown to exhibit approximately 700 equivalent cycles. Using all the thermal transients that appear in the Safety Class 1 specification (Refer to Table 3.9-5b), which is conservative both in frequency and temperature variation, all systems were shown to have less than 1000 equivalent thermal cycles. Therefore, the above results confirm the conservatism of the existing stress analyses for Class 2 and 3 systems and was approved by the NRC (NUREG-0843 Supplement 3, April 1983).

Class 2 and 3 piping systems were reviewed for thermal fatigue and confirmed to be acceptable for 60 years of operation. See Section 18.3.2.2.

The specific criteria that provide the bases for design of a particular component are given in the specific sections that describe the corresponding fluid systems. The design pressure, temperature and other design transients that are considered in the design of each mechanical component are also listed.

The design rules and associated design stress limits applied to the design of ASME Code Class 2 and 3 components are in accordance with the ASME Code, Section III, Subsections NC and ND, respectively. In those areas of design where the applicable rules of Subsections NC and ND are not explicit, the rules are supplemented as described herein, and in Tables 3.9-6 and 3.9-7.

The plant conditions governing design are normal, upset, emergency, and faulted. The loads are combined for each component to insure that the severest combination is specified for non-faulted systems. The loads specified are those that occur within these systems, Dynamic loads that may be imposed by a faulted system on a non-faulted system, i.e., fluid jet impingement and pipe whip impingement are considered separately (see Section 3.6). These effects are accounted for on a case by case basis and are accommodated by:

a. Routing systems such that impingement is not a problem; or

b. If systems are routed in proximity of each other, pipe whip restraints and barriers, or some combination thereof are utilized to protect the essential systems,

It should be noted that for the emergency and faulted conditions the fundamental design criterion is that the functional integrity of the pressure boundary be maintained for non-faulted system piping, vessels and inactive components, and that non-faulted system active components maintain minimum required performance capability.

The stress limit and operability criteria for Code Class 2 and Class 3 components are specified in Tables 3.9-6 and 3.9-7. The criteria provide guidance that results in component designs that comply with the fundamental design criterion for all plant operating conditions.

All ASME III Class 2 and 3 austenitic stainless steel pipe bends and elbows were reviewed and evaluated for functional capability based on General Electric Topical Report NEDO-21985. All systems have been verified to be acceptable except those noted below.

The restraint/support system for the Containment Spray System and the Fuel Pool Cooling System piping systems comply with the stress limit of 1.5 Sy.

Two different elbows on the Intake Cooling Water (ICW) system (CW-76 and CW-77) have a Do/t ratio exceeding the limit listed in the GE report (120 vs. 100). However, the following information is provided to verify the adequacy of this existing piping design:

1. The coefficient of 0.75i used in the original stress calculations is 6.445. The modified coefficient value based on NEDO-21985 for Do/t >50 is 19.98. The primary stresses under faulted conditions calculated with the modified coefficient of 19.98 are 21,816 psi and 18,142 psi for the 30 inch lines (CW-76 and CW-77, respectively). These are well below the allowable stress of 1.5 Sy (i.e., 41,850 psi). Based on the low stresses in the elbows, no deformation is expected.

2. Each of the ICW lines (CW-76 and CW-77) contains a restriction orifice which limits the flow for pump protection. The restriction orifice inside diameter of 12.25 inches represents a reduction in area of approximately 84 percent which is expected to be far greater than the area reduction induced by the piping stress at any bend or elbow. Thus, the restriction orifice is considered to be the limiting element to flow in the piping system.

Therefore, since the calculated stresses are well below the allowable stress (i.e., 23,420 vs 41,850) and since the system orifice is a more critical component for flow restriction than the potential deformation of the pipe bends or elbows, the functional capability of the system is assured.

Testing and/or analysis is representative of the combinations of seismic and operating stresses in order to assure component operability. Vendors are required to provide qualification by testing and/or analysis of all equipment and components prior to installation. Additional data on vendor qualification was submitted to the NRC under separate cover.[(42)] Those components, once installed, are tested preoperationally and are also subject to periodic inspections and tests during the course of normal plant maintenance.

Further consideration for loading combinations under plant faulted conditions are described in Subsection 3.9.1.4.

3.9.3.1.1 Design Stress Limits for A/E Supplied Components

The design stress limits for the mechanical components utilize the guidance of Regulatory Guide 1.48[*], "Design Limits and Loading Combinations for Seismic Category I Fluid System Components", May 1973 (RO). The design stress limits for the various plant components are discussed below:

a. ASME Code Class 1 vessels and piping:

   The design limits specified in NB-3223, NB-3224, and NB-3225 for vessels and NB-3654, NB-3655 and NB-3656 for piping are not exceeded for the design loading combinations specified in Regulatory Position C.1.a,b,c respectively.

b. Non-Active ASME Code Class 1 pumps and valves that are designed by analysis: The design limits specified in NB-3223, NB-3224 and NB-3225 are not exceeded for design loading combinations specified in Regulatory Position C.2.a,b,c, respectively.

c. Non-Active ASME Code Class 1 valves that are designed by standard or alternative design rules:

   1. The primary-pressure rating P_r is not exceeded by more than 10 percent for the design loading combination specified in Regulatory Position C.3 (a).

   2. P_r is not exceeded by more than 20 percent for the design loading combination specified in Regulatory Position C.3 (b).

   3. P_r is not exceeded by more than 50 percent for the design loading combination specified in Regulatory Position C.3 (c).

d. Active ASME Code Class 1 pumps and valves that are designed by analysis:

   1. The design limits specified in NB-3222 of the ASME Code are not exceeded for any of the design loading combinations specified in Regulatory Position C.4.

   2. As alternates, the design limits for non-active pumps and valves are applied for the applicable loading combinations and assurance is

[*] This Reg. Guide was withdrawn in March 1985.

provided by detailed stress and deformation analysis that operability is not impaired when designed to these limits.

e. Active ASME Code Class 1 valves that are designed by standard or alternate rules:

1. The primary-pressure rating P_r is not exceeded for any of the design loading combinations specified in Regulatory Position C.5.

2. As alternates, the pressure limitations for non-active valves designed by standard or alternate design rules are used for the applicable loading combinations and appropriate testing demonstrated that operability is not impaired when the valve is so rated.

f. Pressure Vessels - ASME Code Class 2 and 3 vessels designed to Division 1 of ASME Code, Section VIII utilize the stress limits provided in the ASME Code Case 1607-1, which was approved by the ASME Council on November 4, 1974, was approved per Regulatory Guide 1.84 (R9), and is now incorporated in the ASME Code. Loading conditions are specified with the requirement that all applicable loading combinations be considered and only limiting conditions are required to be analyzed. The design stress levels are provided in tabular form in Table 3.9-7.

g. Pressure Vessels - ASME Code Class 2 and 3 vessels designed to ASME Code, Section VIII, Division 2 are in general agreement with Class 1 requirements. ASME Section III subsections NB and NC apply rather than Section VIII. Refer to Table 3.9-7 for the design stress limits.

h. Piping - ASME Code Class 2 and 3 piping designed to ASME Code, Section III, Division 1 utilizes the stress limits specified in the ASME Code Case 1606-1. (Approved by the Council on December 16, 1974 and accepted by the NRC Regulatory Guide 1.84 (R9), and is now incorporated in the ASME Code.) The design stress levels are provided in Table 3.9-7.

   Code Case N-411, “Alternative Damping Values for Response Spectra of Class 1, 2 and 3 Piping, Section III, Division 1,” may be applied to new systems analyzed by response spectrum methods. Code Case N-411 may also be utilized to qualify proposed modifications to existing systems. Based on the use of Code Case N-411, all piping qualification analyses shall include verification that:

   - All piping supports are properly designed and capable of withstanding design loads.

   - Excessive pipe deflections are not introduced by use of the code case (i.e., displacements shall be checked to verify that proper clearances exist with respect to adjacent structures, components and equipment). Pipe mounted equipment shall also be checked to assure that the equipment is able to withstand the pipe motion.

   - Postulated pipe break locations have been properly considered.

- Affected equipment nozzle loads are not adversely affected.

Each new analysis or reanalysis performed utilizing the PVRC damping values shall include specific reference to Code Case N-411 in the Quality Assurance Records associated with the calculation. For each anchor group (analysis package) where the code case is applied, the code case shall be applied to the entire analysis (i.e., PVRC damping would not be mixed in a given analysis with Regulatory Guide 1.61 criteria).

Code Case N-411 may be applied to systems analyzed by response spectrum methods.

i. Piping Weld Attachments, ASME Code Class 1, 2 and 3 - Stress analysis of piping performed for various loading conditions is presented in Table 3.9-5A. This analysis also determines the loads on welded attachments which are used as part of pipe supports. Local stresses on piping due to welded attachments are calculated using WRC Bulletin 107 and provisions of Code Cases N-318, N-392. The local stresses are combined with other stresses determined by pipe stress analysis in the welded attachment location and ASME Code Section III allowable Stress Criteria are satisfied.

The trunnion analysis utilizes the CYLNOZ computer program to determine the adequacy of the piping system by including the local stresses into the appropriate loading combination. The computer program uses the piping load in global coordinates and transforms these loads into the following components:

1. Radial/shear loads (P/V_L, V_c)

2. Circum./longit. bending (M_C/M_L)

3. Torsional moment (M_T)

Membrane and bending stresses in hoop direction due to P, M_C and M_L are calculated at four locations around the trunnion in outer and inner surfaces of the pipe. Similar calculation for membrane and bending stresses in the longitudinal direction due to P, M_c, and M_L are calculated at the same eight locations of the pipe as described above. Shear stresses due to V_L, V_c and M_T are calculated at these locations. Using the above stresses, stress intensity is calculated at each of these eight locations. The largest stress intensity value is added with pressure stress (if applicable) and normal stress in the pipe, due to corresponding load case and compared with the allowable stress. The results of the analysis conclude that the calculated stresses are within the ASME allowable for all loading combinations.

j. Non-Active Pumps

The design stress limits for the ASME Class 2 and 3 pumps are based upon the approved recommendations of the ASME Task Group. The recommendations were issued as Code Case 1636-1 and have been accepted by the NRC per Regulatory Guide 1.84 (R9), and are now incorporated in the ASME Code. The design stress limits for the active and non- active pumps are summarized in Table 3.9-6.

k. A/E Supplied Active Pumps

Stress limits employed in the design of active pumps, which are defined in Subsection 3.9.3.2.1 as provided in Table 3.9-6, are less than those utilized for the non-active pumps. The use of these conservative stress limits, in conjunction with the operability assurance programs discussed in Subsection 3.9.3.2, provide an acceptable basis for the design of the active pumps. Current requirements, including pre- and post- installation testing, seismic qualification and supplemental analyses provide adequate assurance of operability.

l. Active and Non-active Valves

The design stress limits for the active and non-active ASME Code Class 2 and 3 valves utilize the recommendation of the ASME Task Group. The ASME approved criteria have been issued as Code Case 1635-1 and have been found acceptable to the NRC via Regulatory Guide 1.84 (R9), and are now incorporated in the ASME Code. The design stress limits for the active and non-active valves are provided in Table 3.9-6.

The design stress limits and design conditions presented in Table 3.9-6 are intended to ensure that no gross deformation of the component occurs. These limits are applicable for an elastic system and component analysis. Inelastic methods are not used in conjunction with an elastic system analysis.

3.9.3.1.2 Analysis of Code Class 1 Components and Supports

The major components, supports and main loop piping of the Reactor Coolant System were analyzed using the MEC-21 computer program described in Subsection 3.9.1.2.2.1.11 to determine the loads and displacements at locations throughout the system due to all plant normal operating conditions.

The model of the RCS used in this analysis includes equivalent beam element representations of the reactor vessel, both steam generators, the four reactor coolant pumps, the interconnecting piping and all component supports down to and including the foundation flexibilities. Figure 3.9-18 illustrates the RCS flexibility model. The results of these analyses include forces and moments at all interface locations for inclusion in the component design specifications. Also obtained from these analyses were thermal displacements of interface locations such as tributary nozzles for use in subsequent subsystem design/analysis. See Subsection 3.7.3.1.2 for discussion of seismic analyses of the RCS. See Subsection 3.9.1 for discussion of faulted condition analyses of the RCS.

The analysis and design procedures for A/E supplied supports for the reactor vessel supports (Figure 3.8-41), steam generator sliding base support (Figure 3.8-42), reactor coolant pump pipe column supports (Figure 3.8-43), and pressurizer frame structure (Figure 3.8-52), are discussed in Subsection 3.8.3.4.3.

The method of seismic analysis for A/E supplied Code Class 1 piping system is described in Subsection 3.7.3.1.1.a. As indicated, only elastic analyses are used for analysis and design of Code Class 1 piping systems under faulted conditions. Actual mathematical models are delineated in individual piping stress reports. A/E supplied supports for Code Class 1 piping is discussed in Subsection 3.9.3.4.

3.9.3.1.3 NSSS Components Loading Combinations and Stress Limits

ASME Code Class 1, 2 and 3 fluid system components are designed in accordance with the rules and methods specified in the ASME Code, Section III, Subsections NB, NC and ND respectively. The design stress limits of the ASME Code are selected to insure the pressure retaining integrity of safety class equipment.

The ASME Code is recognized by industry and in the Code of Federal Regulations, as a standard whose rules and procedures provide a reliable, conservative basis for the design of nuclear safety-related equipment of very high integrity. The design limits specified by the ASME Code inherently contain safety factors themselves, so that if certain maximum stresses, deformations, or fatigue usage factors are less than the code allowable limits by any amount, the design is conservative.

3.9.3.1.3.1 NSSS Supplied Valves, Class 1, 2 and 3

NSSS Class 1, 2 and 3 valves are designed and manufactured to the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, NC and ND respectively. Applicable addenda range from winter 1972 and later depending on the purchase order date. Regulatory Guide 1.48* was issued in 1973 and therefore was not imposed on the valves. As an alternative, Appendix 3.9A (Seismic Analysis and Summary of Results for Active Valves) were implemented.

Line valves with extended operators were analyzed for the combined loadings of maximum seismic loads (3.0 g in any direction), maximum operator thrust load, dead weight, and design pressure. This information as well as the stress limits used for the design of these valves is summarized in Tables 3.9-16 and 3.9-17.

Class 1 safety valves were analyzed for the combined loadings of seismic, operating pressure, reaction force, pipe loads and dead weight. This information is summarized in Tables 3.9-16 and 3.9-17. Design transient loads for Class 1 valves were considered but were not required to be analyzed based on the notes to Table 3.9-3B. For active valves, operability is further discussed in Subsection 3.9.3.2 and Appendix 3.9A.

3.9.3.1.3.2 NSSS Supplied Pumps, Class 1, 2 and 3

NSSS supplied pumps are designed for the concurrent loadings provided by pressure, dead weight, nozzle loads and seismic.

Non-active ASME Code Class 1 pumps are designed by analysis and utilize the guidance of Regulatory Guide 1.48*, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components", May 1973. The design limits specified in NB-3223, NB-3224 and NB-3225 are not exceeded for design loading combinations as specified in Paragraphs C.2.a, b and c respectively of the Regulatory Guide. Loading combinations and allowable stress limits are defined in Tables 3.9-14 and 3.9-15 respectively. There are no active ASME Code Class 1 pumps supplied by NSSS.

ASME Code Class 2 and 3 assemblies are also designed by analysis. The stress limits used for each pump are defined in Table 3.9-19 and the loading combinations are defined in

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* This Reg. Guide was withdrawn in March 1985.

Table 3.9-18. For active pumps, operability assurance is further discussed in Subsection 3.9.3.2 and Appendix 3.9A.

3.9.3.1.3.3 Class 1 NSSS Components Except for Pumps and Valves

The NSSS components which are ASME Code Class 1 (Quality Group A) are included in Table 3.9-1. The specific components covered by this subsection are the reactor vessel, steam generators, pressurizer and NSSS supplied Reactor Coolant System piping. ASME code requirements are supplemented by additional requirements of Regulatory Guide 1.48*, "Design Limits and Loading Combinations Seismic Category I Fluid System Components", May 1973. The design limits specified in NB-3223, NB-3224 and NB-3225 for vessels and NB-3654, NB-3655 and NB-3656 for piping are not exceeded for the design loading combinations as specified in Paragraphs C.1a, b and c respectively of the Regulatory position. The loading combinations for ASME Code Class 1 components are shown in Table 3.9-14. The allowable stress limits are defined in Table 3.9-15.

3.9.3.1.3.4 Class 2 and 3 Components, Other Than Valves and Pumps

ASME Code Class 2 and 3 vessels utilize the stress limits provided in the ASME Code Case 1607-1, which was approved by the ASME Council on November 4, 1974, was approved per Regulatory Guide 1.84 (R9), and is now incorporated in the ASME Code. The loading combinations for these components and supports are defined in Table 3.9-20. Allowable stress limits are shown in Table 3.9-21.

3.9.3.2 Pumps and Valve Operability Assurance

ASME Code Class 2 and 3 components are designed in accordance with the ASME Code, Subsections NC and ND respectively. The design standards employed are as specified in those subsections. Additionally, the rules of Subsections NC and ND are supplemented as described previously.

Procurement specifications for safety-related active components stipulate that Seller shall submit either detailed calculations and/or test data to demonstrate operability when subjected to the specification loading and stress criteria (normal through faulted conditions). The decision to accept actual or prototype test data, or analysis for operability assurance is made during the normal design/procurement process. The determination to test is based on (1) whether the component is amenable to analysis, (2) whether proven analytical methods are available, and (3) whether applicable prototype test is available. If analysis or prototype test data is not sufficient, testing is conducted to qualify the component or to verify the analytical technique.

Where appropriate, the static shaft deflection calculations for pumps and static valve stem deflection calculations for valves are performed to determine deflections due to short term seismic and other applicable loadings. Deflections so determined are compared to allowable clearances. It must be noted that seismic events are of short duration; thus, contact (if it occurs) does not demonstrate that operability is adversely affected. Cases where contact occurs are reviewed on a case by case basis to determine acceptability.

The operability of active Code Class 1, 2 and 3 components is assured because of an extensive program of design verification, qualification testing and thorough surveillance of the

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* This Reg. Guide was withdrawn in March 1985.

manufacturing, assembly and testing of each active component. Each aspect of the design related to pressure boundary integrity and operability is either tested or verified by calculations. Procedures for testing are developed by component manufacturers and reviewed and approved before the tests are conducted. The design analyses of the component take into consideration environmental conditions including loadings developed from the thermal, seismic and operational effects. Where necessary and feasible, the conclusions of these analysis are confirmed by test in accordance with the quality assurance program.

3.9.3.2.1 Active Pumps

Active pumps are defined as those pumps required to undergo a mechanical motion to either mitigate the effects of an accident or safely shutdown the plant. (Table 3.9-1 provides a tabulation of Code Class 1 components). Code Class 2 and 3 active pumps utilized in St. Lucie Unit 2 are identified in Table 3.9-8. It must be noted that the list does not intend to imply that all of the components listed undergo simultaneous loadings equivalent to those specified in Table 3.9-5 for the faulted conditions. The list merely represents those components which are investigated for such loading combinations to determine the applicable combination for design.

Active pumps are qualified for operability by subjecting them to rigid tests both prior to and after installation in the plant. Each active pump is given factory tests, as necessary, to determine that the work and materials are free from defects and to establish that the design and construction are satisfactory. The in-shop tests include a sufficient combination of the following tests:

a. Hydrostatic tests of the pump casings.

b. Complete performance test to measure capacity, total developed head, power input and efficiency.

c. Net positive suction head (NPSH) requirements verified by suction pressure suppression tests.

d. Temperature transient tests performed at pump design conditions.

e. Mechanical seal leakage tests.

f. Vibration tests.

g. Measurements to determine pump casing minimum wall thickness.

h. Operability qualification of pump motors as outlined in Section 3.11.

After the pump is installed in the plant, it undergoes the cold hydro tests, hot functional tests, and the required periodic inservice inspection and inservice testing. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to the above, the manufacturer is required to demonstrate the equipment's ability to perform its required safety function during and after the time it is subjected to the forces resulting from seismic conditions. In order to avoid damage during the faulted condition, the stresses caused by the combination of normal operating loads, SSE and dynamic system loads are limited to allowable values as indicated in Table 3.9-6. An analysis is further required to prove that adequate clearance has been provided to prevent binding of the shaft and other rotating parts within the pump casing when the pump is subjected to these design loading conditions.

Performance of these analyses and tests with the loads stated and with the restrictive allowable stress limits of Table 3.9-6 assures that the critical parts of the active pumps are not damaged during the short duration of the faulted condition and that the reliability of the active pumps for post-faulted condition operation is not impaired by the seismic event. Appendix 3.9A "Operability Considerations for Seismic Category I Active Pumps and Valves" provides the criteria for seismic qualification provided to the manufacturers of seismic Category I equipment and supports.

Appendix 3.9A also contains a description of the analyses and the results of these analyses for the active pumps and their supports.

3.9.3.2.2 Active Valves

Active valves are those valves in which mechanical movement is required in order to perform its intended safety function. The active valves are identified in Tables 3.9-9 and 10.

The active valves, like active pumps, are similarly subjected to extensive analysis and testing in order to assure that specification requirements are met. Prior to installation the following tests are performed:

a. Hydrostatic tests to ASME Code, Section III requirements

b. Seat leakage tests

c. Vibration tests or analyses

d. Functional tests to verify valve opening/closing capability.

e. Operability qualification of operators (i.e. environmental qualification) as outlined in Section 3.11.

f. Measurements to assure minimum wall thickness for valves greater than 1 inch nominal pipe size.

Various post-installations tests are performed to assure valve performance. The active valves are subjected to cold hydro qualification tests, hot functional testing, periodic inservice testing and inservice inspection. These tests are performed on site in order to verify and assure the functional ability of the active valves and guarantee reliability of the valves for the design life of the plant.

In addition to the above tests, each valve that is designated as an active component must demonstrate operational capabilities during and after a seismic event. The manufacturer is provided with criteria for seismic qualification, as outlined in Appendix 3.9A. The results of the valve operability analysis or testing is provided in the Appendix.

NRC Generic Letter 89-10 and 96-05 Programs

NRC Generic Letter 89-10 requires that operating nuclear plants develop and implement a program to ensure that switch settings on all safety-related motor-operated valves (MOVs) are correctly selected, set and maintained to accommodate the maximum differential pressures expected on these valves during all postulated events within the design basis. Item a) of the Letter requires that the design basis for the MOVs be reviewed to determine the maximum differential pressure expected during both opening and closing strokes (as applicable) for all postulated events.

Item b) of Generic Letter 89-10 requires that the licensee establish the correct MOV switch settings based on the previously determined maximum differential pressure. All switches, including torque switches, torque bypass switches, position limit, position indication, overloads, etc., shall be considered. This requires that the actuator and valve capabilities at degraded voltage be evaluated. Modifications to the valves and actuators have been performed where appropriate to allow incorporation of the proper switch settings.

Once the correct switch settings have been incorporated into the respective motor-operated valves, Item c) of Generic Letter 89-10 establishes requirements for stroke testing motor-operated valves against the maximum differential pressure established in Item a) to verify operability. The results of these tests shall be used to trend the condition of the valves and operators and to help determine appropriate maintenance frequencies.

The requirements of NRC Generic Letter 89-10 have been completed for the applicable Unit 2 valves, which are listed in different sections of the UFSAR.

NRC Generic Letter 96-05 requires that operating nuclear plants establish or ensure the effectiveness of their current program to identify and account for age-related changes which may affect design basis capability of motor-operated valves. Specifically, potential degradation which may result in an increase in thrust or torque requirements or a decrease in motor actuator capability must be addressed.

The periodic verification program established as a result of GL 96-05 includes all motor-operated valves within the scope of GL 89-10. This program addresses all elements of GL 96-05 and provides for a blend of maintenance, static and dynamic testing, and trending of specific parameters to assure overall MOV capability. The program is focused, utilizing risk significance and margin, to determine the appropriate activities and frequency.

3.9.3.2.3 Deep Draft Pumps

St. Lucie Unit 2 has vertical deep draft pumps which are used in the Intake Cooling Water System. These pumps are manufactured by Byron Jackson (Model 37 KXL) and are single stage, 130 ft head, 14,500 gpm capacity, vertical circulator pumps and are the only pumps of this type in safety-related systems at this facility. The pumps are used for long term cooling.

The Intake Cooling Water Pumps at St. Lucie Unit 2 are required for normal plant operations as well as for long term accident conditions. There are three such pumps installed, two of which will run continuously during Mode 1 through 5, and a third which is an installed spare able to provide flow should one of the other pumps fail. Continuous operation of two pumps provides positive indication of pump operability and availability for emergency situations, as well as longer run times for more accurate identification of potential pump problems. All three pumps are operated extensively during preoperational testing such that their operability is assured prior to core load. These pumps were placed in service prior to the RCS Cold Hydro and accumulated at least 300 hours run time each prior to core load. Based on the requirement for continuous pump operation and confidence in preoperational testing, additional testing of Intake Cooling Water Pumps per the NRC guidelines is not deemed necessary.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

Safety and relief valves for overpressure protection of ASME Code Class 1 and 2 system components are designed and installed in accordance with ASME Code, Section III, 1971 edition up to and including the 1973 Summer Addenda. Code Class 1 and 2 safety and relief valves are listed in Tables 3.9-11 and 12 including sizes, operating and design conditions and locations.

Analysis of the safety and relief valves includes the effects of local stresses at the junction of the valve branch and header and stresses at the flange connections.

In the dynamic analysis of the Safety/Relief valve discharge piping system, the same stiffness matrix method, as described in Subsection 3.7.3.1.1 a) 2), is used for the representation of the structural response of the piping Supports.

The Design Stress limits, as delineated in Subsection 3.9.3.1.1 and Tables 3.9-6, 3.9-7 and Design Loading as defined in Table 3.9-5, are applicable to the mounting of pressure relief valves.

The safety relief valve stations are listed in Tables 3.9-11 and 12. The dynamic effects of discharge piping are evaluated and incorporated into the applicable piping analysis. The simultaneous discharge of all the relief valves on any single run pipe in a multiple-valve installation is considered as a dynamic load factor (DLF), in the header and the support/restraint system design.

As recommended in Regulatory Guide 1.67, "Installation of Overpressure Protection Devices," October 1973 (R0) and ASME Code Case N-40 (1569), "Design of Piping for Pressure Relief Valve Station Section III," the magnitude of the reaction force and anticipated transient behavior as supplied by the valve manufacturer is used in the safety relief valve station design.

Safety and relief valve stations can be categorized as either open discharge or closed discharge.

a. Open discharge installations have the fluid discharging directly to atmosphere or vent pipe that is uncoupled from the safety valve. The following information is included as part of total design consideration:

    1. Thrust forces include pressure and momentum effects

    2. The minimum moment used in the stress analyses are those specified in ASME Code Case-1569.

    3. The valve thrust loads are considered on the valve inlet piping from the header.

    4. The reaction forces and moments used in the stress calculations are modified by a dynamic load factor (DLF) or by the maximum instantaneous value obtained from a dynamic time-history analysis. A DLF of 2.0 is used in lieu of a dynamic analysis to determine the specific DLF. See Subsection 3.6.2.

5. Stresses due to thermal, internal pressure, dead weight. Seismic effects and thrust loads are compared to the allowable for the header, local stresses and valve inlet piping. These stresses are calculated according to ASME Code Section III and combined as shown in Table 3.9-13.

6. Where multiple safety and/or safety and relief valves are installed on the same headers. The worst sequence of the valve opening is considered.

7. The nozzle spacing for multiple valve installation meets the recommendations of ASME Code, Section III.

b. Closed Discharge Installation:

An installation where the effluent is carried to distant spot by a discharge pipe which is connected directly to the safety valve. The stresses developed after the initial valve thrust are calculated by either a conservative static method or a time-history computer solution. Conservative calculations described below are performed on each valve and resulting stresses compared to the ASME allowable.

The initial analysis of the valves conservatively assumes that they are open discharge valves supported only by the valve inlet pipe as a cantilever beam from the header. The valve discharge forces conservatively include a DLF of 2.0.

The closed discharge system of the safety and relief valves from the pressurizer are analyzed by a time-history dynamic analysis. As a more conservative, less complex approach, the closed discharge system of the safety and relief valves on a safety-related auxiliary system such as the Safety Injection System and Chemical and Volume Control System are analyzed by a static analysis. A transient hydraulic force equal to the freely blowing reaction force acting in both directions with a dynamic load factor of 2 is applied to each long straight leg of the piping system for flashing service. For short intermediate straight leg (L) the unbalanced hydraulic transient force will be modified by a factor (1.5L/c.t) to account for the valve opening time (t), piping length (L) and the acoustical velocity of the propagating wave (c).

For non-flashing liquid discharge system, the same transient hydraulic forces are applied to the valve outlet and the first elbow and the transient hydraulic forces in the downstream of the discharge are considered in the analysis for the maximum momentum of the fluid.

The seismic loads are based on the maximum accelerations given in the appropriate floor response spectra. The valves meet the ASME Code allowables under these very conservative assumptions.

Dynamic analyses for the pressurizer safety/relief valves (V1200, V1201, V1202, V1474, V1475) were performed. The computer program PIPESHK calculates the forcing functions acting on the pressurizer relief valve discharge piping system generated by the initial shock wave. PIPESHK was developed on the basis of the technical paper, "Analysis of Safety Valve Discharging Into Closed Piping

System," by C H. Luk[30]. The forcing functions generated by the fluid flow itself were calculated with computer programs RELAP3 and CALPLOTF. The results are used in computer program PIPESTRESS 2010 and then compared to ASME Code allowable stresses.

3.9.3.4 Component Supports

For NSSS supplied ASME Code Class 1 vessel supports, piping supports and supports for the reactor coolant pumps (including the attachment welds to the vessel or piping assemblies) procured prior to July 1, 1974, the design stress limits are defined in the applicable design / procurement specification. For the faulted condition the stress limits are defined as: the limits of Section III, NB-3220 using an Sm value equal to the greater of 1.5 times the tabulated Sm value and 1.2 times the tabulated Sy value, but not exceeding 0.7 times the material tensile strength, with the values taken at the appropriate temperature. Design stress limits for other loading conditions are those identified in applicable subsections of the ASME Code.

A/E supplied supports for Code Class 1, 2 and 3 components are designed in accordance with codes in effect at the time of the purchase order. The supports for piping, instrument tubing and field installed instrumentation are designed in accordance with the code in effect at the time of the original piping purchase order, the summer of 1973 Addenda to Section III of the ASME Code. The supports for components procured prior to July 1, 1974 are designed per AISC guidelines. For normal and upset conditions, normal AISC stress limits apply.

The allowable stresses in the AISC Code and the ASME Code Appendix XVII have been reviewed and are similar in most respects. The following differences are identified:

a. At the contact surface of a weld producing a tension load in the through thickness direction of plates and elements of rolled shapes, the allowable tension stress is limited to 60 percent of yield by AISC and 30 percent of yield by ASME Appendix XVII.

   Ultrasonic testing of both shop and field full penetration tee welds has been specified for materials used in A/E supplied NSSS component supports where lamellar tearing may be a problem. In addition, ultrasonic examinations are required to be performed on a flange prior to welding the flange to the web on built up girders and columns.

   The materials used in piping supports are less than one inch thick in the majority of cases, therefore lamellar tearing is not considered a problem. (Refer to "Significance and Control of Lamellar Tearing of steel plate in the shipbuilding industry" 1979). In addition, lamellar tearing generally initiates during or shortly after the welding process. This has not been identified as a problem with piping supports.

b. ASME Code Appendix XVII Load Increase Factor is 1.2 Sy/Ft not to exceed 0.7 Su/Ft. An increase factor of 1.6 is used. This is applied to minimize redesign and refabrication.

   The St. Lucie 2 design criteria for NSSS supports in the factored (faulted) condition allows the AISC allowables to be increased by 1.6 but less than

0.96 x F_y or 0.90 x critical buckling where applicable. MEB requested that the NSSS supports in the faulted condition be reviewed for compliance to ASME III Subsection NF F-1370 (a) wherein the allowable stresses in Appendix XVII may be increased by a factor ranging from 1.4 to 2.0 depending upon the material tensile and yield stresses. Critical elements in the NSSS supports have been reviewed and the actual design stresses are below those allowed by ASME III Subsection NF.

The only material used for seismic Category I pipe supports for which S_y/Su exceeds 0.73 is SA-500 Grade B structural tubing. Support designs have been reviewed and in no cases are normal AISC allowable of 0.6 Sy exceeded. (Note Ft=0.6 Sy).

No reduction in yield strength is taken for application below 700 F, in accordance with the AISC Manual of Steel Construction (Reference: Effect of Heat on Structural Steel). Austenitic material used in seismic Category I component supports for hypochlorination piping supports for the intake cooling water pumps. The austenitic piping supports are located in the intake structure. Austenitic material is also used for welded attachments on austenitic pipe and is designed to the same allowables as the pipe,

Material documentation required for seismic Category I support components are Certificates of Compliance and CMTRs, as appropriate.

Integrally attached supports supplied with components procured after July 1, 1974 are in accordance with ASME Code, Section III, Subsection NF, with its applicable stress limits. The only supports designed to Section NF are for the following components:

Containment Spray Pumps

Intake Cooling Water Pump.

Auxiliary Feedwater Pumps

Diesel Oil Transfer Pumps

Intake Cooling Water Basket Strainers

Safety Injection Tanks

SG Sliding Base and Bearings

Fuel Pool Heat Exchanger

Shutdown Cooling Heat Exchangers

Vendor Supplied Instrument Racks

Regenerative Heat Exchanger

Boric Acid Makeup Tank

Hydrazine Pumps

The extent of deformation of the supports is limited by the allowable stresses discussed above.

The waste gas compressor supports and anchor bolts are designed to the criteria of the AISC Manual of Steel Construction, 1970, except that the increase in the allowable stresses due to seismic and wind loads per Paragraph 1.5.6 Part 5, is not permitted.

Thermal stresses are considered as primary stresses for supports and as secondary stresses for components.

The operability assurance program for active components and their supports is discussed in Subsection 3.9.3.2. Preoperational tests for piping systems and their supports are discussed in Subsection 3.9.2.1.

A/E Supplied Equipment and Pipe Supports

All safety-related component supporting structures are designated "seismic Category I." Load combinations and allowable stresses are in accordance with Standard Review Plan 3.8.3 and Standard Review Plan 3.8.4. Refer to Subsections 3.8.3 and 3.8.4. The margin of safety for these structures is inherent in the design equations in the AISC Specifications.

For linear and plate and shell type component supports subjected to the accident (faulted) load condition, the design stresses are limited to 90* percent of the critical buckling stress as applicable. For design of support bolts and bolted connections, refer to the above paragraph.

Piping supports and restraints are designed to accommodate the loading combinations shown in Table 3.9-28. The normal allowable stress limits of AISC and MSS-SP-58, as summarized in Table 3.9-27, are used in original support and restraint design for all loading combinations, including faulted. The allowable stresses of MSS-SP-58 are used without the addition of a shape factor to account for bending stresses. To minimize redesign and refabrication, which might result from revised stress analyses, the following criteria apply as necessary when evaluating existing restraint designs against revised faulted loading: stresses in hangers and restraints shall be less than 1.6 times AISC limits, not to exceed 0.96 times material yield stress, where shear yield stress is assumed to be 0.577 times tensile yield stress. Also, stresses shall not exceed 0.90* times critical buckling stress, when that is a controlling factor.

Piping supports and restraints are normally attached to embedded plates. Where embedded plates are not available, bridging between embedded plates or thru bolting is implemented. If these methods proved unreliable, concrete expansion anchors are then used.

When using concrete expansion anchors a factor of safety 15 is used for seismic applications, except where the pressure of large loads results in a significant prying effect (Appendix 3.9B). Prying effects have been studied and are accounted for in the design. Expansion anchors' ultimate capacities have been established by a field testing program.

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*Cases where buckling stresses in the supports of the ASME Class 1, 2 or 3 components exceed 67 percent of critical buckling stress will be justified on an individual basis to determine that the margin against buckling is sufficient.

The design procedure for using expansion anchors is discussed in Appendix 3.9B.

NSSS Vendor Supplied Equipment

a. Buckling failure of the RCS supports is not credible due to the design characteristics of the supports.

b. The bolts in CE scope of supply (Steam Generator Skirt to Sliding Base) are designed to be below 70 percent of ultimate which, for the material, is less than 75 percent of yield.

c. Required Preload of interface Anchor Bolts (S.G. Snubber, Pressurizer Skirt) were specified to Ebasco.

3.9.4 CONTROL ELEMENT DRIVE MECHANISMS

3.9.4.1 Descriptive Information of CEDM

The control element drive mechanism (CEDMs) are magnetic jack type drives used to vertically position and indicate the position of the control element assemblies (CEAs) in the core. Each CEDM is capable of withdrawing, inserting, holding, or tripping the CEA from any point within its 137 inch stroke in response to operation signals.

The CEDM is designed to function during and after all normal plant transients. The design life of the CEDM is defined as 40 years of operation or 100,000 feet of rod travel without loss of function. The CEDM is designed to operate without maintenance for a minimum of 1-1/2 years and without replacing components for a minimum of three years. The CEDM is designed to function normally during and after being subjected to the operating basis earthquake loads. The CEDM allows for tripping and drive-in of the CEA during and after a safe shutdown earthquake.

The design and construction of the CEDM pressure housings fulfill the requirements of the ASME Code, Section III, Class 1. The CEDM pressure housings are part of the reactor coolant pressure boundary, and they are designed to meet stress requirements consistent with those of the vessel. The pressure housings are capable of withstanding, throughout the design life, all normal operating loads, which include the steady-state and transient operating conditions specified for the vessel. Mechanical excitations are also defined and included as a normal operating load. The CEDM pressure housings are service rated at 2500 psia and 650°F. The loading combinations and stress limit categories are presented in Subsection 3.9.4.3 and are consistent with those defined in the ASME Code.

The test programs performed in support of the CEDM design are described in Subsection 3.9.4.4.

3.9.4.1.1 Control Element Drive Mechanism Design Description

The CEDMs are mounted and seal welded on nozzles on top of the reactor vessel closure head. The CEDMs consist of the upper and lower CEDM pressure housings, motor assembly, coil stack assembly, reed switch assemblies, and extension shaft assembly. The CEDM is shown on Figure 3.9-11. The drive power is supplied by the coil stack assembly, which is positioned around the CEDM housing. A position indicating reed switch assembly is supported by the upper pressure housing shroud, which encloses the upper pressure housing assembly.

The components outside the pressure boundaries are the coil stack, the pressure housing shroud, and the cooling shroud. All are designed to be a slip fit over the motor housing and are capable of being removed at temperature. A test was performed to verify this requirement. Dimensions and materials used for the St. Lucie Unit 2 CEDMs are identical to those on operating reactors.

All failure modes of non-pressurized active components will not effect the safety function of the CEDM. The coil stack is designed and has been tested to verify its capability to withstand loss of air coolant flow for up to four hours without loss of function.

Parts within the pressure boundary, such as the motor assembly, have been sized for thermal deflections caused by dissimilar material so that clearances are available above the maximum design temperature of 650 F.

The lifting operation consists of a series of magnetically operated step movements. Two sets of mechanical latches are utilized engaging a notched extension shaft. To prevent excessive latch wear, a means has been provided to unload the latches during the engaging operations. The magnetic force is obtained from large dc magnet coils mounted on the outside of the lower pressure housing.

Power for the electromagnets is obtaned from two separate supplies. A control programmer actuates the stepping cycle and moves the CEA by a forward or reverse stepping sequence. Control element drive mechanism hold is obtained by energizing one coil at a reduced current, while all other coils are deenergized. The CEDMs are tripped upon interruption of electrical power to all coils. Each CEDM is connected to the CEAs by an extension shaft. The weight of the CEDMs and the CEAs is carried by the pressure vessel head. Installation, removal, and maintenance of the CEDM is possible with the reactor vessel head in place; however, the CEDM is inaccessible during operation of the plant.

The axial position of a CEA in the core is indicated by two independent readout systems. One counts the CEDM steps electronically, and the other consists of magnetically actuated reed switches located at regular intervals along the CEDM upper pressure housing. These systems are designed to indicate CEA position to within ± 2-1/2 inches of the true location. This accuracy requirement is based on ensuring that the axial alignment between CEAs is maintained within acceptable limits.

The materials in contact with the reactor coolant used in the CEDM are listed in Subsection 4.5.1.

3.9.4.1.1.1 CEDM Pressure Housing

The CEDM pressure housing consists of the motor housing assembly and the upper pressure housing assembly. The motor housing assembly is attached to the reactor vessel head nozzle by means of a threaded joint and seal welded. Once the motor housing assembly is seal welded to the head nozzle, it need not be removed since all servicing of the CEDM is performed from the top of the housing. The upper pressure housing is threaded into the top of the motor housing assembly and seal welded. The upper pressure housing encloses the CEDM extension shaft and contains a vent. The top of the upper pressure housing is closed by means of a threaded cap with a welded Omega seal.

3.9.4.1.1.2 Motor Assembly

The motor assembly is an integral unit that fits into the motor housing and provides the linear motion to the CEA. The motor assembly consists of a latch guide tube, driving latches, and holding latches.

The driving latches are used to perform the major stepping of the CEA. The holding latches hold the CEA during repositioning of the driving latches and perform a load transfer function to minimize latch and extension shaft wear. Engagement of the extension shaft occurs when the appropriate set of magnetic coils is energized. This moves sliding magnets which cam a two-bar linkage moving the latches inward. The driving latches move vertically a maximum of 3/4 inch. The holding latches move vertically 1/16 inch to perform the load transfer.

3.9.4.1.1.3 Coil Stack Assembly

The coil stack assembly for the CEDM consists of five large dc magnet coils mounted on the outside of the motor housing assembly. The coils supply magnetic force to actuate mechanical latches for engaging and driving the extension shaft. Power for the magnet coils is supplied from two separate supplies. A magnetic coil power programmer actuates the stepping cycle and obtains the correct CEA position by a forward or reverse stepping sequence. CEDM hold is obtained by energizing one coil at a reduced current while all other coils are deenergized. The CEDMs are tripped upon interruption of electrical power to all coils. Electrical pulses from the magnetic coil power programmer provide one of the means of transmitting CEA position indication.

A conduit assembly containing the lead wires for the coil stack assembly is located at the side of the upper pressure housing.

3.9.4.1.1.4 Reed Switch Assembly

The reed switch assembly provides the means for transmitting CEA position indication. Reed switches and voltage divider networks are used to provide output voltages proportional to the CEA position. The reed switch assembly is positioned so as to utilize the permanent magnet in the top of the extension shaft. The permanent magnet actuates the reed switches as it passes them. The reed switch assembly is provided with an accessible electrical connector at the top of the upper pressure housing. Three additional pairs of reed switches on each CEDM provide upper electrical limit, lower electrical limit, and dropped rod indications.

3.9.4.1.1.5 Extension Shaft Assembly

The extension shaft assemblies are used to link the CEDMs to the CEAs. The extension shaft assembly is a 304 stainless steel rod with a permanent magnet assembly at the top for actuating reed switches in the reed switch assembly, a center section called the drive shaft, and a lower end with a coupling device for connection to the CEA.

The drive shaft is a long tube made of 304 stainless steel. It is threaded and pinned to the extension shaft. The drive shaft has circumferential notches along the shaft to provide the means of engagement to the CEDM.

The magnetic assembly consists of a housing, magnet, and plug. The magnet is made of two cylindrical Alnico-5 magnets. This magnet assembly is used to actuate the reed switch position indicators. The magnets are contained in a housing, which is plugged at the bottom. The housing also provides a means of attaching the lifting tool for disengaging the CEA from the extension shaft.

3.9.4.2 Applicable CEDM Design Specifications

The pressure boundary components consist of a lower and upper pressure housing and vent assembly at the top of the upper housing. The vent assembly is composed of a ball seal housing, vent stem, housing nut and steel ball. With the exception of the steel ball (see Subsection 4.5.1.2), all items are designed and fabricated in accordance with the requirements of the ASME B&PV Code, Section III, 1998 Edition through 2000 Addenda. One Code Case is also used in the design and fabrication of the pressure boundary components. Specifically, the section of the motor housing surrounding the motor electromagnets employs material in conformance with Code Case N-2 (1334-4). The pressure boundary material selection complies with the requirements of the ASME B&PV Code, Section II, 1998 Edition through 2000 Addenda and ASME B&PV Code, Section XI, 1998 Edition through 2000 Addenda.

3.9.4.3 Design Loads, Stress Limits and Allowable Deformations

Loads are combined into the following loading conditions for the CEDM pressure boundary stress analysis;

a. Normal and Upset Conditions (Service Levels A and B)

   1. Reactor coolant pressure and temperature
   2. Normal and Upset reactor operating transients
   3. Dynamic stresses produced by operating basis earthquake forces.
   4. Dynamic stresses produced by mechanical excitations
   5. Loads produced by operation and tripping of the mechanism

The stress limits employed are given in Figures NB-3221-1 and 3222-1 including notes from Section III of the ASME Code 1998 Edition through the 2000 Addenda.

The CEDM is designed to function normally during and after normal and upset deflection conditions.

b. Faulted Conditions (Service Level D)

   1. Reactor coolant pressure and temperature
   2. Loss of coolant reactor transients
   3. Dynamic stresses produced by safe shutdown earthquake forces
   4. Dynamic stresses produced by mechanical excitations
   5. Loads produced by operation and tripping of the mechanism

The Stress limits employed are given in Paragraph F-1330 or F-1340, Appendix F of Section III, of the ASME Code 1998 Edition through 2000 Addenda, Rules for Evaluation of Service Loadings With Level D Service Limits.

CEDM deflections are limited such that CEAs can be inserted after exposure to faulted conditions.

c. Testing Conditions

1. Testing plant transients

The stress limits employed are given in Paragraph NB-3226 of Section III of the ASME Code 1998 Edition through 2000 Addenda.

3.9.4.4 CEDM Performance Assurance Program

The CEDM prototype performance assurance program is described in Waterford Steam Electric Station Unit No. 3 UFSAR, Docket number 50-382.

ALL CEDM production units are tested for a minimum of 400 feet of total travel at combinations of pressure and temperature from ambient up to reactor operating conditions. The CEDMS are also tested for six full-height gravity drop tests at simulated reactor operating conditions.

After installation of the CEDMS and prior to power operation, the CEDMs were field tested in accordance with plant procedures.

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

3.9.5.1 Design Arrangements

The components of the reactor internals are divided into two major parts consisting of the core support barrel assembly and the upper guide structure assembly. The flow skirt, although functioning as an integral part of the coolant flow path, is separate from the internals and is affixed to the bottom head of the pressure vessel. The incore instrumentation support system is also considered as part of the reactor internals structures and assemblies. The arrangement of these components is shown on Figure 3.9-8.

3.9.5.1.1 Core Support Barrel Assembly

The major structural member of the reactor internals is the core support barrel assembly. This assembly consists of the core support barrel and the lower support structure. The material for the assembly is Type 304 stainless steel.

The core support barrel assembly is supported at its upper end by the upper flange of the core support barrel, which rests on a ledge in the reactor vessel. Alignment is accomplished by means of four equally spaced keys in the flange, which fit into the keyways in the vessel ledge and closure head. The lower flange of the core support barrel supports, secures, and positions the lower support structure and is attached to the lower support structure by means of a welded flexural connection. The lower support structure provides support for the core by means of a core support plate supported by columns mounted on support beams that transmit the load to the core support barrel lower flange. The core support plate provides support and orientation for the lower ends of the fuel assemblies. The core shroud, which provides a flow path for the coolant and lateral support for the fuel assemblies, is also supported and positioned by the core support plate. The lower end of the core support barrel is restricted from excessive radial and torsional movement by six snubbers that interface with the reactor vessel wall.

3.9.5.1.1.1 Core Support Barrel

The core support barrel is a right circular cylinder including a heavy external ring flange at the top end and an internal ring flange at the lower end. The core support barrel is supported from a ledge on the pressure vessel. The core support barrel, in turn, supports the lower support structure upon which the fuel assemblies rest. Press-fitted into the upper flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head, and upper guide structure assembly flange are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

The upper section of the barrel contains two outlet nozzles that interface with internal projections on the vessel nozzles to minimize leakage of coolant from inlet to outlet.

Amplitude limiting devices, or snubbers, are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally-spaced lugs around the circumference of the barrel and act as a tongue-and-groove assembly with the mating lugs on the pressure vessel. Minimizing the clearance between the two mating pieces limits the amplitude of vibration. During assembly, as the internals are lowered into the pressure vessel, the pressure vessel lugs engage the core support barrel lugs in an axial direction. Radial and axial expansion of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted. The reactor vessel lugs have bolted, captured Inconel X shims, and the core support barrel lug mating surfaces are hardfaced with Stellite to minimize wear. The shims are machined during initial installation to provide minimum clearance. The snubber assembly is shown on Figure 3.9-12.

3.9.5.1.1.2 Core Support Plate and Lower Support Structure

The core support plate is a Type 304 stainless steel plate into which the necessary flow distribution holes for the fuel assemblies have been machined. Fuel assembly locating pins (four for each assembly) are shrunk-fit into this plate.

The fuel assemblies and core shroud are positioned on the core support plate, which forms the top support member of a welded assembly consisting of a cylinder, a bottom plate, support columns, and support beams. The core support plate is supported by an arrangement of columns welded at the base to support beams. The bottoms of the beams are welded to the bottom plate, which contains flow holes to provide proper flow distribution. The ends of the beams and the top periphery of the bottom plate are welded to a cylinder that supports the outer edge of the core support plate. The cylinder guides the reactor coolant flow and limits the core shroud bypass flow by means of holes located near the base of the cylinder.

3.9.5.1.1.3 Core Shroud

The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. The shroud consists of two Type 304 stainless steel ring sections welded to each other and to the core support plate.

A small gap is provided between the core shroud outer perimeter, and the core support barrel and holes are provided in the girth rings in order to provide upward coolant flow between the core shroud and the core support barrel, thereby minimizing thermal stresses in the core shroud and eliminating stagnant pockets. The core shroud is shown on Figure 3.9-13.

3.9.5.1.2 Upper Guide Structure Assembly

The upper guide structure assembly consists of the upper guide structure support plate assembly, control element assembly shrouds, and a fuel assembly alignment plate (Figure 3.9-14). The upper guide structure assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, holds down the fuel assemblies during operation, prevents fuel assemblies from being lifted out of position during a severe accident condition, protects the control element assemblies (CEAs) from the effect of coolant cross flow in the upper plenum, and supports the incore instrumentation plate assembly. The upper guide structure assembly is handled as one unit during installation and refueling.

The upper end of the assembly is a structure consisting of a support flange welded to the top of a cylinder. A support plate is welded to the inside of the cylinder approximately in the middle. The support plate is welded to a grid array of deep beams, the ends of which are welded to the cylinder. The support flange contains four accurately machined and located alignment keyways, equally spaced at 90 degree intervals, which engage the core barrel alignment keys. This system of keys and slots provides an accurate means of aligning the core with the closure head and thereby with the CEA drive mechanisms. The support plate aligns and supports the upper end of the CEA shrouds. The shrouds extend from the fuel assembly alignment plate to an elevation above the upper guide structure support plate. The CEA shroud consists of a cylindrical upper section welded to a base, and a flow channel structure shaped to provide flow passage for the coolant through the alignment plate, while shrouding the CEAs from cross flow. The shrouds are bolted and lockwelded to the fuel assembly alignment plate. At the upper guide structure support plate, the shrouds are connected to the plate by spanner nuts. The spanner nuts are tightened with proper torque to assure a rigid connection and lockwelded.

The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds. Precision machined and located holes in the fuel assembly alignment plate engage machined posts on the fuel assembly upper end fittings to provide accurate alignment. The fuel assembly alignment plate also has four equally spaced slots on its outer edge that engage with Stellite hardfaced pins protruding from the core shroud to limit lateral motion of the upper guide structure assembly during operation. The fuel alignment plate bears the upward force of the fuel assembly holddown devices. This force is transmitted from the alignment plate through the CEA shrouds to the upper guide structure support plate. A holddown ring is located between the upper surface of the flange of the upper guide structure and the lower surface of the reactor vessel head. This holddown ring provides the axial force on the core support structures necessary to overcome the hydraulic forces and prevent movement of the structures during operation. The holddown ring is designed to accommodate the differential thermal expansion between the reactor vessel and the core support structures in the vessel ledge region.

3.9.5.1.3 Flow Skirt

The Inconel flow skirt is a right circular cylinder, perforated with flow holes, and reinforced at the top and bottom with stiffening rings. The flow skirt functions to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is supported by nine equally spaced machined sections that are welded to the bottom head of the reactor vessel.

3.9.5.1.4 Incore Instrumentation Support System

The incore neutron flux monitoring system includes self-powered incore detector assemblies, and supporting structures.

The self-powered incore detector assemblies and the computer system are described in Section 7.7.

The fixed incore detector support system consists of the instrument plate structure guide tubes, and the thimbles that extend downward into selected fuel bundles. The incore instrumentation guide tubes route the instruments so that the detectors are located and spaced throughout the core. The guide tubes and the incore thimbles are attached to and supported by the instrument plate assembly shown on Figure 3.9-15.

The instrumentation plate assembly fits within the confines of the reactor vessel head and rests in the recessed section of the upper guide structure assembly. Its weight is supported by four bearing pins. The upper guide structure CEA shrouds extend through the instrumentation plate clearance holes. Above the instrumentation plate, the guide tubes bend and are gathered to form stalks that extend into the reactor vessel head instrumentation nozzles. The instrumentation plate assembly is raised and lowered during refueling to insert or withdraw all instruments and their thimbles simultaneously. The pressure boundaries for the individual instruments are at the instrumentation nozzle flange, where the external electrical connections to the incore instruments are also made.

The incore instrument assemblies have a Swagelock fitting, which forms a seal at the instrument flange, and through which the signal cables pass. Carbon packing rings fitted in a recess in the instrument flange are used to seal against operating pressure. The incore instrument nozzle sealing arrangement is shown on Figure 3.9-16.

3.9.5.1.5 Reactor Vessel Level Detector Support System

Provisions for two reactor vessel level detector assemblies are incorporated into the reactor vessel internals. These level detector holders provide a guide path for the Heated Junction Thermocouple (HJTC) probe assemblies which enable monitoring of reactor vessel water level from the reactor vessel closure head down to the fuel alignment plate. The level detector support system is similar to the fixed incore instrumentation support system described above (See Figures 3.9-25a and 3.9-25b). The HJTC support tubes and guide paths are specially ported with holes and slots to provide good communication between the outside of the guide path and the inside where the probe is located. The HJTC probes penetrate the reactor vessel head via existing instrument flanges in a manner identical to that for incore instrumentation.

3.9.5.2 Design Loading Conditions

The following loading conditions are considered in the design of the reactor internals:

a. Normal operating temperature differentials

b. Normal operating pressure differentials

c. Flow impingement loads

d. Weights, reactions and superimposed loads (fuel assembly weights and spring loads have been adjusted to encompass operation with either Westinghouse fuel or AREVA fuel).

e. Vibration loads

f. Shock loads (operating basis and safe shutdown earthquakes)

g. Anticipated transient loadings not requiring forced shutdown

h. Handling loads (not combined with other loads above)

i. Loads resulting from postulated loss-of-coolant accidents

3.9.5.3 Design Loading Categories

The design loading conditions are categorized below:

3.9.5.3.1 Normal Operating and Upset

The normal and upset category includes the combinations of design loadings consisting of normal operating temperature and pressure differentials loads due to flow, weights, reactions, superimposed loads, vibration, shock loads including operating basis earthquake, and transient loads not requiring shutdown.

3.9.5.3.2 Faulted

The faulted category consists of the mechanical loading combinations of Subsection 3.9.5.3.1 with the exception that the safe shutdown earthquake (SSE) (in place of the operating basis earthquake) and the loads resulting from the loss-of-coolant accident (LOCA) are included.

3.9.5.4 Design Bases

3.9.5.4.1 Reactor Internals

Reactor internals are designed according to Subsection NG of the ASME Code Section III with the exception of stamping and a code stress report.

No emergency condition has been identified for the applicable components, therefore, no appropriate stress criteria are provided.

Dynamic systems analyses including the effects of asymmetric loads have been performed to verify the adequacy of the structural design of the reactor internals during the postulated LOCA. The LOCA maximum stresses in the reactor internal components have been determined using the combinations of the lateral and vertical LOCA time-dependent loadings which resulted in the maximum stress intensities in the structural analysis. These LOCA maximum stresses and the maximum stresses resulting from the SSE were combined using the SRSS method to obtain the total stress intensities. The total stress intensities were compared to the allowable stress intensities for faulted conditions, and the reactor internals stress intensities are within acceptable limits. The comparison is shown on Table 3.9-29.

To properly perform their functions, the reactor internal structures are designed to meet the deformation limits listed below:

a. Under design loadings plus operating basis earthquake forces, deflection is limited so that the control element assemblies (CEAs) can function and adequate core cooling is provided.

b. Under normal operating loadings, plus SSE forces, plus pipe rupture loadings resulting from a break equivalent in size to the largest line connected to the Reactor Coolant System piping, deflections are limited so that the core is held in place, adequate core cooling is preserved, and all CEAs can be inserted. Those deflections which would influence CEA movement are limited to less than 80 percent of the deflections required to prevent CEA insertion.

c. Under normal operating loads, plus SSE forces, plus the maximum pipe breaks loadings resulting from the full spectrum of pipe breaks, deflections are limited so that the core is held in place and adequate core cooling is preserved. Although CEA insertion is not required for a safe and orderly shutdown for break sizes greater than the largest line connected to the Reactor Coolant System piping, calculations show that the CEAs are insertable for larger breaks, except for a few CEAs located near the vessel outlet nozzle which is feeding the postulated break.

The allowable deformation limits are listed in the following tabulation. Allowable limits are established as 80 percent of the loss-of-function deflection limits.

Location	Allowable Deflection (in.)
Fuel lower end fitting lower support structure	1.842 (Disengagement)
Fuel upper end fitting, upper guide structure	1.243 (Disengagement)
CEA shroud (lateral)	1.178 (CEA Insertion

In the design of critical reactor vessel internals components which are subject to fatigue, the stress analysis is performed utilizing the design fatigue curve of Figure 1-9-2 of Section III of the ASME Code. A cumulative usage factor of less than 1.0 is used as the limiting criterion. The highest usage factor for the reactor internals is found to occur in the core support barrel flange region and is less than .15.

As indicated in the preceding subsections, the stress and fatigue limits for reactor internals components are obtained from the ASME Code. Allowable deformation limits are established as 80 percent of the loss-of-function deflection limits. These limits provide adequate safety factors assuring that so long as calculated stresses, usage factors, or deformations do not exceed these limits, the design is conservative.

3.9.5.4.2 Incore Instrumentation Plate Assembly

The incore instrumentation plate assembly is an internal structure as defined in Paragraph NG-1122 of Subsection NG of Section III of the ASME Code. The rules of this subsection were applied to the design of the incore instrumentation plate assembly. The construction of the incore instrumentation plate assembly does not adversely affect the integrity of the core support structure. The incore instrumentation plate assembly by itself is not a safety-related component since the satisfactory performance of the incore instrumentation plate assembly does not prevent accidents nor mitigate the consequences of accidents that could cause undue risk to the health and safety of the public.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

The preservice and inservice testing programs for Code Class 1, 2 and 3 safety-related pumps and valves are developed employing the R.G. 1.26, Revision 3, criteria for quality group classifications and standards (Quality Group A is the same as ASME Class 1, etc.). The programs are provided under separate cover and are implemented to assess operational readiness. The inservice testing program is based on the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, and the ASME OM Code. The program is updated periodically to meet 10 CFR 50.55a(f) requirements. Specific code commitments are documented in the plant's inservice testing program procedure. EC290695

3.9.6.1 (deleted)

3.9.6.2 (deleted)

3.9.6.3 Relief Requests

The inservice testing program is updated periodically to meet the 10 CFR 50.55a(f) requirements. Where it become impractical to meet this criteria, relief from these requirements, on a case-by-case basis, shall be requested.

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44. J. A. Norris (NRC) to C. O. Woody (FPL), dated August 17, 1989. "Closure of Generic Letter 89-08, Erosion/Corrosion Induced Pipe Wall Thinning."

45. AREVA NP Inc., 77-5069878-004, “Replacement Steam Generator Report for Florida Power and Light St. Lucie Unit 2,” December 2015.

TABLE 3.9-1

QUALITY GROUP A COMPONENTS

1. NSSS SUPPLIED COMPONENTS

(a) Equipment
Reactor Vessel
Steam Generators*
Reactor Coolant Pumps
Pressurizer

(b) Valves

V1474, V1475	Power Operated Reliefs
V1476, V1477	Power Operated Reliefs Isolation
V1239	Power operated Reliefs
V1200, V1201, V1202	Pressurizer Safeties
V1436, V1439, V1435 V1438, V1437, V1440	Pressurizer Instrumentation
PCV-1100E, PCV-1100F	Pressurizer Spray
V1453, V1454	Pressurizer Spray Bypass
V1441, V1442, V1443, V1444	Pressurizer Spray Isolation
V1248, V1249	Pressurizer Spray Check
V1455, V1456	Pressurizer Spray Vent
V1215, V1247	Shutdown Cooling Outlet Drain
V1234, V1235, V1449, V1450	RC Cold Leg Drain
V1212	Reactor Vessel Head Vent
V1384, V1385, V1386, V1387	RC Pump Lower Seal Cavity Vent
V1280, V1281, V1282, V1283	RC Pump Middle Seal Cavity Vent
V1288, V1289, V1290, V1291	RC Pump Upper Seal Cavity Vent
V2593, V2515, V2516	Letdown Line
V2539, V2463	Letdown Line Drain
V2540	Etdown Line Vent
V2483, V2431	Auxiliary Spray Line
V2807	Auxiliary Spray Line Drain
V2489	Auxiliary Spray Line Vent
V2484, V2432, V2485, V2433	Charging Line
V2804, V2803, V2801, V2800	Charging Line Drain
V2802, V2805	Charging Line Vent
V2434, V2435	Charging Line Bypass
V3217, V3227, V3237, V3247 V3614, V3624, V3634, V3644 V3258, V3259, V3260, V3261	Safety Injection
V3118, V3128, V3138, V3148	SIT Recirculation
V3803, V3801, V3885, V3765 V3922, V3862, V3811, V3912 V3815, V3817, V3906	Safety Injection Vents

* The steam generators were replaced (reference 45) and the replacement steam generators are not part of the original NSSS equipment.

TABLE 3.9-1 (Cont'd)

V3802, V3806, V3807 V3863, V3768, V3818, V3816 V3911, V3905	Safety Injection Drain
V3525, V3574, V3572, V3524 V3527, V3573, V3571, V3526	Hot Leg Injection
V3542, V3709, V3544, V3711	Hot Leg Injection Drain
V3708, V3710	Hot Leg Injection Vent
V3480, V3481, V3545, V3652 V3651	Shutdown Cooling
V3482, V3469	Shutdown Cooling Relief
V3841, V3800	Shutdown Cooling Drain
V2505, V2507	Reactor Coolant Pump Leakoff
I-SE-02-1,2	Charging Line
I-SE-02-3,4	Aux. Pressurizer Spray

(c) Piping

Reactor Coolant System

Line Number	Nearest Equipment/Header
42-RC-114	Hot Leg 2A
30-RC-112	Cold Leg 2A1
30-RC-115	Cold Leg 2A2
42-RC-123	Hot Leg 2B
30-RC-121	Cold Leg 2B1
30-RC-124	Cold Leg 2B2
12-RC-10-3	Surge Line
3-RC-141	Spray Line
3-RC-109	Spray Line
3/4-RC-303	Spray Line
3/4-RC-304	Spray Line
3/4-RC-305	Spray Line Vent
3/4-RC-306	Spray Line
3/4-RC-307	Spray Line
3/4-RC-308	Spray Line Vent
4-RC-103	Spray Line

2. Non - NSSS Components

Piping

Reactor Coolant System

Line Number	Nearest Equipment/Header
I-4-RC-101	Pressurizer
I-2-RC-106	RC Loop 2A
I-2-RC-113	RC Loop 2A1

TABLE 3.9-1 (Cont'd)

Line Number	Nearest Equipment/Header
I-2-RC-116	RC Loop 2A2
I-2-RC-122	RC Loop 2B1
I-2-RC-125	RC Loop 2B2
I-2-RC-142	Letdown (2B1)
I-12-RC-147	RC Loop 2B
I-2-RC-148	RC Loop 2A2
I-2-RC-149	Press Aux Spray
I-2-RC-150	RC Loop 2B1
I-12-RC-151	RC Loop 2A1
I-12-RC-152	RC Loop 2B1
I-12-RC-153	RC Loop 2B2
I-12-RC-154	RC Loop 2A2
I-2-RC-158	RC Loop 2B
I-12-RC-162	RC Loop 2A
I-10-RC-301	SDC Line 2A
I-10-RC-302	SDC Line 2B
I-3-RC-309	Pressurizer
I-3-RC-310	Pressurizer

Safety Injection System

Line Number	Nearest Equipment/Header
I-6-SI-515	SI Tank 2A2
I-6-SI-516	SI Tank 2A1
I-6-SI-517	SI Tank 2B1
I-6-SI-518	SI Tank 2B2
I-10-SI-127	RC Loop 2A (Pen 64)
I-10-SI-130	RC Loop 2B (Pen 40)
I-12-SI-148	SI Tank 2A1
I-12-SI-149	SI Tank 2A2
I-12-SI-150	SI Tank 2B1
I-12-SI-151	SI Tank 2B2
I-1-SI-125	SIT 2A2
I-1-SI-123	SIT 2A1
I-1-SI-120	SIT 2B1
I-1-SI-118	SIT 2B2
I-2-SI-152	RC Loop 2A
I-3-SI-190	Hot Leg Inj
I-3-SI-191	Hot Leg Inj
I-10-SI-378	SDC Header
I-2-SI-153	Safety/Relief Valves
I-1-SI-533	SIT Fill Header (HPSIP 2B)
I-1-SI-535	SIT Fill Header (HPSIP 2A)

Chemical and Volume Control System

I-2-CH-145	Regenerative HX
I-2-CH-146	Regenerative HX
I-2-CH-147	Regenerative HX
I-2-CH-148	Regenerative HX
I-2-CH-149	Regenerative HX
I-2-CH-347	Regenerative HX

TABLE 3.9-2

TRANSIENTS USED IN DESIGN AND FATIGUE ANALYSIS

NOTE: Class 1 piping and components were reviewed for thermal fatigue and were confirmed to be acceptable for a 60 year design file, utilizing the original 40-year design cycles. See Section 18.3.2.1.

1. Normal Conditions

   (a) 500 heatup and cooldown cycles during the design life of the components with heating and cooling at a rate of 100°F/hr between 70°F and 532°F (653°F for the pressurizer). The heatup and cooldown rate of the system is administratively limited to 75°F/hr and 85°F/hr, respectively, to assure that these limits will not be exceeded. This is based on the original 40-year design life cycle and a normal plant cycle of one heatup and cooldown per month rounded to the next highest hundred. The heatup and cooldown cycles permitted on the 2B RSG were reduced from 500 to 120 per EC 284513. EC290592 EC284513

   (b) 15,000 power change cycles over the range of 15 percent to 100 percent of full load at 5 percent of full load per minute increasing and decreasing. This is based on a normal plant operation involving one cycle per day for 40 years rounded to the next highest 1000. (CEDM repairs implemented via PCM 03021 reduces the power change cycles from 15,000 to 2,000 cycles for the affected penetrations.) The power change cycles permitted on the 2B RSG were reduced from 15000 to 2000 per EC 284419.

   (c) 2,000 cycles of step power changes of 10 percent of full load, increasing in the 15 percent to 90 percent of full load range and 2000 cycles decreasing in the 100 percent to 25 percent of full load range. This is based on the original 40-year design life cycle and a normal plant operation involving one cycle per week for 50 weeks of the year.

   (d) The Reactor Vessel, Replacement Steam Generator and Reactor Coolant Pump are designed for 1 x 10^6 cycles of normal variations of ± 100 psi and ± 6°F when at operating temperature and pressure. The pressurizer normal design transient is 1 x 10^6 cycles of normal variations of ± 50 psi and ± 6°F. The 1 x 10^6 cycles is based on such a large value being equivalent to infinite cycles and thus the limiting stress is the endurance limit. The pressure and temperature variations are selected to be well within the actual fluctuations which are limited by control systems.

2. Upset Conditions

   (a) 40 cycles of complete loss of reactor coolant flow when at 100 percent power. This is based on the original 40-year design life cycle and one reactor trip per year for the life of the plant resulting from failure of electrical supply to the reactor coolant pumps.

TABLE 3.9-2 (Cont'd)

(b) 400 reactor trips from full load. This is based on the original 40-year design life cycle and one reactor trip per month for the life of the plant and includes trips due to operator error and equipment failure. Allowing for down time and refueling, the design number of cycles is reduced to 400.

(c) 40 cycles of turbine trip from 100 percent power with delayed reactor trip. This is based on the original 40-year design life cycle and one reactor trip per year for the life of the plant considering failure of the turbine trip/ reactor trip circuit as credible.

3. Emergency Conditions

5 cycles of complete loss of secondary pressure. This transient would follow a steam line break. A steam line break is not considered credible in forming the basis for design of the Reactor Coolant System. However, system components will not fail structurally in the unlikely event that it does happen.

4. Faulted Conditions

The loading combination resulting from the combined effects of the design basis earthquake and normal operation at full power are categorized as faulted condition.

The loading combinations resulting from the design basis earthquake, normal operation at full power and pipe rupture conditions are categorized as faulted condition. Design basis earthquake and pipe rupture loadings are combined by the SRSS method.

5. Test Conditions

10 cycles of system hydrostatic testing at 3110 psig and at a temperature not less than 60 F above the highest component reference temperature (RT_{NDT}) or 100 F above the highest component section (RT_{NDT}) value. This is based on the original 40-year design life cycle and one initial hydrostatic test plus a major repair every four years for 36 years which includes equipment failure and normal plant cycles. No additional system hydrostatic testing at the above pressure/temperature conditions is permitted on the 2B RSG per EC 284513. EC284513

200 cycles of leak testing at 2235 psig and at a temperature not less than 60 F above the highest component reference temperature (RT_{NDT}) or 100 F above the highest pipe section RT_{NDT}. This is based on the original 40-year design life cycle and normal plant operation involving five shutdowns for head removal or valve repair per year for 40 years. The number of cycles of leak testing permitted on the 2B RSG at the above pressure/temperature conditions was reduced from 200 to 30 per EC 284513. EC284513

TABLE 3.9-3A

A/E SUPPLIED QUALITY GROUP A TRANSIENTS

PLANT EVENT	LIFETIME OCCURRENCES	COMPONENT* CONDITION	
Plant Cooldown	500†	N	EC284513
Plant Heatup	500†	N	EC284513
Power Operation	-	N	
Loading/Unloading Ramp 5% per Min Step 10%	 15,000 each‡ 2,000 each		
Reactor Trip	400	U	
Hydrostatic Tests, (3125 psia)	10†	T	EC284513
Leak Test, (2250 psia)	200†	T	EC284513
Normal Pressure and Temperature variations ±100 psi (RCS is ±50 psi) ±6°F (Pressurizer is ±7°F	10^6	N	
Loss of Primary Flow	40	U	
Loss of Secondary Pressure	5	E	
Loss of Turbine-Gen. Load	40	U	
Purification, & Boron Dilution (CVCS)	24,000	N	
Loss of Charging Flow (CVCS)	100	U	

* Definitions of the events (Component Condition) Normal (N), Upset (U), Emergency (E), Faulted (F) and Test (T) are given in ASME III, Para. NB-3113.

† For the 2B RSG, the lifetime occurrences of these plant events were reduced per EC 284513 as follows: heatup/cooldown reduced from 500 to 120; hydrostatic tests reduced from 10 to 1; leak tests reduced from 200 to 30. EC284513

‡ The lifetime occurrences of plant Loading/Unloading for the 2B RSG were reduced from 15000 to 2000 per EC 284419.

TABLE 3.9-3A (Cont'd)

PLANT EVENT	LIFETIME OCCURRENCES	COMPONENT* CONDITION
Regenerate Heat Exchanger Isolation and Loss of Letdown (CVCS)	270	U
Isolation Check Valve Leaks	40	U
LOCA (Safety Injection)	1	F
LOCA (Hot Leg Injection)	1	F

* Definitions of the events (Component Condition) Normal (N), Upset (U), Emergency (E), Faulted (F) and Test (T) are given in ASME III, Para. NB-3113.

TABLE 3.9-3B

NSSS - SPECIFIED TRANSIENTS

TRANSIENT	Transient Caterory	Reactor Vessel	Stm.Gen	Press	RC Pipe	Surge Line	Spray Line	R C Pump	
1. Heatups									
a) 70° to 532° @100°F/HR	NORMAL	500	500*		500		500	500	EC284513
b) 70° to 653° @100°F/HR	NORMAL			500		500			
2. Cooldowns									
a) 532° to 70° @100°F/HR	NORMAL	500	500*		500		500	500	EC284513
b) 653° to 70° @100°F/HR	NORMAL			500		500			
3. Plant Loading									
a) 5% F.P/Min-15% to 100% F.P	NORMAL	15,000†	15,000‡	15,000					
b) 5% F.P/Min-10% to 100% F.P.					15,000	15,000	15,000	15,000	
4. Plant Unloading									
a) 5% F.P./Min-100% F.P. to 15%	NORMAL	15,000†	15,000‡	15,000					
b) 5% F.P./Min-100% F.P. to 10%	NORMAL				15,000	15,000	15,000	15,000	
5. 10% Step Load Increase									
b) 90% to 100% F.P	NORMAL	2,000	2,000	2,000	2,000	2,000	2,000	2,000	
6. 10% Step Load Decrease									
b) 100% to 90%	NORMAL	2,000	2,000	2,000	2,000	2,000	2,000	2,000	
7. Plant Variations									
b) ± 100 psi, ± 6°F	NORMAL	10^6	10^6					10^6	
c) ± 100 psi, ± 7°F	NORMAL			10^6					
d) ± 50 psi, ± 6°F	NORMAL				10^6	10^6	10^6		

* The transient limits for Heatups and Cooldowns for the 2B RSG were reduced from 500 to 120 per EC 284513. EC284513
† PCM 03021 reduced the loading/unloading cycles to 2,000 cycles for Rx head penetrations repaired via the PCM.
‡ The transient limits for Plant Loading/Unloading for the 2B RSG were reduced from 15000 to 2000 per EC 284419.

TABLE 3.9-3B (Cont'd)

TRANSIENT		Transient Caterory	Reactor Vessel	Stm.Gen	Press	RC Pipe	Surge Line	Spray Line	R C Pump	
8.	Chg. Noz.-Purification & Boron Dilution	NORMAL				24,000				
9.	Loss of Flow	UPSET	40	40	40	40	40	40	40	
10.	Loss of Full Load	UPSET	40	40	40	40	40	40	40	
11.	Reactor Trip/Loss of Load	UPSET	400	400	400	400	400	400	400	
12.	Oper. Basis Earthquake	UPSET	200	200	200	200	200	200	200	
13.	Loss of Charging	UPSET	N/A	N/A	N/A	20	N/A	N/A	N/A	
14.	Loss of Letdown	UPSET	N/A	N/A	N/A	50	N/A	N/A	N/A	
15.	Loss of Secondary Pressure	EMERGENCY	5	5	5	5	5	5	5	
16.	Safe Shutdown Earthquake + Normal Operation	FAULTED	1	1	1	1	1	1	1	
17.	Safe Shutdown Earthquake + Normal Operation + Pipe Rupture	FAULTED	1	1	1	1	1	1	1	
18.	Hydro Test (3110 psig)	TEST	10	10*	10	10	10	10	10	EC284513
19.	Leak Test (2235 psig)	TEST	200	200*	200	200	200	200	200	EC284513

* For the 2B RSG, the limits for these transients were reduced per EC 284513 as follows: Hydro Test reduced from 10 to 1; Leak Test reduced from 200 to 30. EC284513

TABLE 3.9-3B (Cont'd)

4" and under Class 1

These valves are covered under NB-3513 of ASME Code Section III. As such no transients were specified in the components design specification.

Over 4" Class 1

A. V3217, V3227, V3237, V3247 (12" Check Valves)

These valves are in the safety injection inlet lines and have no flow through them except during safety injection or shutdown cooling. In addition their physical location is such that the volume of fluid entrapped between these valves and the reactor coolant loop ameliorates the effect of reactor coolant loop transients to a point which produces negligible transients at the valve. As such no transients were specified in the component design specification.

B. V3258, V3259, V3260, V3261 (6" check valves)

These valves are in the safety injection inlet lines, upstream of the valves listed in A above, (further removed from the RV loop). These valves are insulated from the R.C. loop by valves in A above, the leg of fluid discussed in A above and the leg of fluid between these valves and the valves discussed in A above. As such, due to their location, these valves experience no transients. Therefore, no transients were specified in component design specification.

C. V3614, V3624, V3634, V3444 (12" M.O. Gate Valves)

These valves are the safety injection tank isolation valves are connected to the safety injection lines between the valves discussed in A and B above. For the reasons stated in B above, these valves also experience no transients and none were specified in component design specification.

D. V3480, V3481, V3545, V3651, V3652 (10" M.O. gate valves)

The valves are in the shutdown cooling suction lines, V3480 and V3652 are the two valves closest to the R C Loop. The other three valves are insulated from the R C Loop transients by V3480 and V3652. V3480 and V3652 are locked closed valves and have no flow through them. Their location is such that the volume of fluid entrapped between the valves and the R C Loop ameliorates the effect of reactor coolant loop transients to a point which produces negligible transients at the valve. When the valve is opened to permit flow, the reactor coolant loop (the flowing medium) temperature is below the design temperature of the valve. The valve stroking time (60 seconds) produces a ramped flow increase. Both of these combined result in negligible transients. Based on all the above, no transients were specified in the component design specifications.

EC289971

TABLE 3.9-3C

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT	
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤100°F/hr.*	Heatup cycle - T_{avg} from ≤200°F to ≥532°F; cooldown cycle - T_{avg} from ≥532°F to ≤200°F.	EC284513
	500 pressurizer heatup and cooldown cycles at rates ≤200°F/hr.	Heatup cycle - Pressurizer temperature from ≤200°F to ≥653°F; cooldown ≥653°F to ≤200°F.	
	10 hydrostatic testing cycles.*	RCS pressurized to 3110 psig with RCS temperature ≥60°F above the most limiting components' NDTT value.	EC284513
	200 leak testing cycles.*	RCS pressured to 2250 psia with RCS temperature greater than minimum for hydrostatic testing, but less than minimum RCS temperature for criticality.	EC284513
	400 reactor trip cycles.	Trip from 100% of RATED THERMAL POWER.	
	40 turbine trip cycles with delayed reactor trip.	Turbine trip (total load rejection) from 100% of RATED THERMAL POWER followed by resulting reactor trip.	

* For the 2B RSG, these cyclic limits were reduced per EC 284513 as follows: system heatup and cooldown reduced from 500 to 120; hydrostatic testing reduced from 10 to 1; leak testing reduced from 200 to 30. EC284513

EC289971

TABLE 3.9-3C (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	40 complete loss of reactor coolant flow cycles.	Simultaneous loss of all Reactor Coolant Pumps at 100% of RATED THERMAL POWER.
	5 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODE 1, 2, or 3.
	100 pressurizer spray cycles per year with pressurizer/spray water $\Delta T >$ 200°F or as otherwise calculated by the following method:	Spray operation consisting of opening and closing either the main or auxiliary spray valve(s) spray water/pressurizer $\Delta T >$ 200°F.

EC289971

TABLE 3.9-3C (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Method for Calculating Pressurizer Spray Nozzle Cumulative Usage Factor

ΔT	N_A	N	N/N_A
201 - 300	13,000		
301 - 400	5,000		
401 - 500	3,000		
501 - 600	1,500		
		$\Sigma\ N/N_A$	

Where:

ΔT = Temperature difference between pressurizer water and spray in °F.

N_A = Allowable number of spray cycles.

N = Number of cycles of ΔT range indicated.

EC289971

TABLE 3.9-3C (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Calculation Method:

1. At 12-month intervals the cumulative spray cycles shall be totaled.
   If the total is equal to or less than 1000, no further action is required.
2. If the cumulative total exceeds 1000, the spray nozzle usage factor shall be calculated as follows:

   A. Fill in Column “N” above.

   B. Calculate “N/N_A” (Divide N and N_A).

   C. Add Column “N/N_A” to find $\Sigma\ N/N_A$.

   $\Sigma\ N/N_A$ is the cumulative spray nozzle usage factor. If the calculated usage factor is equal to or less than 0.75, no further action is required.
3. If the calculated usage factor exceeds 0.75, subsequent pressurizer spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 200°F when spray is operated. An engineering evaluation of nozzle fatigue shall be performed and shall determine that the nozzle remains acceptable for additional service prior to removing this restriction.

TABLE 3.9-4

COMPARISON OF STRUCTURAL AND HYDRAULIC DESIGN PARAMETERS

Parameters			Fort Calhoun	Maine Yankee	St. Lucie Unit 2
Structural(1)					
Upper CSB	R_{mean},	in.	61-5/16	75-1/4	75-1/2
	t,	in.	2	2-1/2	3
	L,	in.	101-3/8	135-5/8	153-1/2
Middle CSB	R_{mean},	in.	61-1/16	74-7/8	75-1/4
	t,	in.	1-1/2	1-3/4	2-1/2
	L,	in.	166-1/8	144-3/4	113-3/4
	R_{mean},	in.	60-11/16	75-5/8	75-1/4
	t,	in.	2-1/4	2-1/4	3
	L,	in.	35-5/8	38	65-1/2
Lower cylinder ID,		in.	Integral	141	141
Core cylinder OD,		in.	Integral	145	145
Support cylinder L.		in.	Integral	42	37-5/8
Structure supported			Integral	CSB flange	CSB flange
Core shroud support			Bolted to CSB	Core support plate	Core support plate
UGS Cylinder	R_{mean},	in.	59-1/16	72-5/8	72-1/4
	t,	in.	1-1/2	2	2-1/2
	L	in.	24	24	44-1/2
Beams,		in.	24x1-1/2	24x1-1/2	24x1-1/2
Plate t		in.	3-1/4	4	3-1/2
Thermal Shield			Yes	Yes	No
Hydraulic(2)					
No. of loops			2	3	2
Design min flow, 10^6 1bm/h /h			71.7	122	139.5
Inlet design temperature, °F			547	546	548
Inlet ID, in.			28-3/4	39	35-3/16
Outlet ID, in.			37	39-5/8	48-1/8

(1) CSB = Core support barrel
UGS = Upper guide structure

(2) Velocity = Design minimum velocity

TABLE 3.9-4 (Cont'd)

Parameters	Fort Calhoun	Maine Yankee	St. Lucie Unit 2
Hydraulic (Cont'd)			
Inlet pipe velocity, ft/s	33.7	39.0	42.0
Downcomer velocity, ft/s	25.2	24.9	22.3
Core inlet velocity, ft/s	12.8	12.9	14.5
Outlet pipe velocity, ft/s	41.5	39.0	46.5
Pump rotational speed, r/min	1200.	1200.	900.0

TABLE 3.9-5

DESIGN LOADING COMBINATIONS FOR A/E QUALITY GROUPS B AND C COMPONENTS (VESSELS, PUMPS, VALVES)

Plant Operating Condition	Design Loading Combination
Normal	a) PO + DW
Upset	a) PO + DW + OBE
	b) PO + DW + OBE + RVO
	c) PO + DW + OBE + FVC
	d) PO + DW + DU
Emergency	a) PO + DW + OBE + RVO + FVC
Faulted	a) PO + DW + DBE
	b) PO + DW + DBE + RVO
	c) PO + DW + DBE + FVC
	d) PO + DW + FC
	e) PO + DW + DBE(*) + FC(*)

Notation

PO - operating pressure and temperature
DW - live and dead weight (including nozzle load)
OBE - operating basis earthquake (inertia portion)
RVO - relief valve operation (including open or closed, as applicable)
FVC - fast valve operation (as applicable)
DU - other dynamic system loading associated with plant upset conditions
DBE - design basis earthquake (inertia portion)
FC - dynamic system loadings associated with plant faulted conditions

(*) These loads are combined in accordance with NUREG-0484, Rev. 1.

TABLE 3.9-5A
DESIGN LOADING COMBINATIONS FOR A/E QUALITY
GROUPS B AND C PIPING

Plant Operating Condition	Design Loading Combinations
Normal	a) PO + DW
	b) TO
Upset	a) PO + DW + OBE
	b) PO + DW + OBE + RVO
	c) PO + DW + OBE +FVC
	d) PO + DW + DU
	e) TI
Emergency	a) PO + DW + OBE + RVO + FVC
Faulted	a) PO + DW + DBE
	b) PO + DW + DBE + RVO
	c) PO + DW + DBE + FVC
	d) PO + DW + FC
	e) PO + DW + DBE (*) + FC (*)

Notation

PO - operating pressure and temperature
DW - live and dead weight (including nozzle load)
OBE - operating basis earthquake (inertia portion)
RVO - relief valve operation (open or closed, as applicable)
FVC - fast valve operation (as applicable)
DU - other dynamic system loading associated with plant upset conditions
DBE - design basis earthquake (inertia portion)
FC - dynamic system loadings associated with plant faulted conditions
TO - thermal loads
TI - restrained thermal expansion and the relative movement of anchor points produced by the OBE

(*) These loads are combined on the basis of SRSS in accordance with NUREG-0484, Rev. 1.

TABLE 3.9-5B

SAMPLE CALCULATIONS

Thermal Transient	ΔTE	ΔTI	NI	N
5% Ramp Up	455	70	15000	2
5% Ramp Down	455	208	15000	299
10% Step Up	455	47	2000	0
10% Step Down	455	145	2000	7
Loss of RCP Flow	455	204	40	1
Reactor Trip	455	187	400	5
Loss of Load	455	204	40	1
Loss of Sec ss	455	330	5	1
Normal Var.	455	6	10^6	1
Hydro	455	335	10	2
Leak Test	455	335	200	43
Loss of Charging	455	335	20	6
Loss of Letdown	455	400	50	26
Regen H-X Iso	455	349	120	32
Max Purification	455	128	1000	16
Max Dilution	455	126	8000	14
Low VCT	455	45	2000	0
Norm Start	455	455	500	500
Aux FW Inj	-	-	-	-
TOTALS				955

1. Charging T max = 520°F
2. Ambient Temp (Tamb) 65°F
3. ΔTE = T max - Tamb = 455°F
4. Number of cycles = NI
5. Temp change of transient = ΔTI
6. Equiv # of cycles at ΔTI

$$N = NI\,(\Delta TI/\Delta TE)^5$$

TABLE 3.9-6

A/E DESIGN STRESS LIMITS FOR QUALITY GROUP
B AND C PUMPS/VALVES

COMPONENTS	CONDITION	σ_m	STRESS LIMITS[1][2] $(\sigma_m$ or $\sigma_L) + \sigma_b$		P_{MAX}
A/E supplied Inactive Pumps	Normal	ASME III,	NC-3400	or	ND-3400
	Upset	1.1 S	1.65 S		
	Emergency	1.5 S	1.8 S		
	Faulted	2.0 S	2.4 S		
A/E supplied Active Pumps	Normal	ASME III,	NC-3400	or	ND-3400
	Upset	1.0 S	1.5 S		
	Emergency	1.1 S	1.65 S		
	Faulted	1.5 S	1.8 S		
A/E supplied Valves (Active/Inactive)	Normal	ASME III,	NC-3500	or	ND-3500
	Upset	1.1 S	1.65 S		1.1P
	Emergency	1.5 S	1.8 S		1.2P
	Faulted	2.0 S	2.4 S		1.5P

TABLE 3.9-7

A/E DESIGN STRESS LIMITS FOR
CODE CLASS 2 AND 3 PIPING AND VESSELS

COMPONENT	CONDITION	STRESS LIMITS[1,2]	^{S}ALLOWABLE	^{P}MAX
Piping	Normal	ASME III, NC-3600		
	Upset	NC-3611.1 (b)(4)(c)(b)(1) - 1.2 sh.		
	Emergency	NC-3611.1 (b)(4)(c)(b)(2) - 1.8 Sh.		1.5P
	Faulted	NC-3611.1 (b)(4)(c)(b)(2) - 2.4 Sh.		2P
		ASME III, 1971 Edition through Summer 1973 and the ASME Code Case 1606 Addenda		
Pressure Vessels (ASME VIII, Div. 2) (App. F)	Normal	ASME III, NC-3300 or ND-3300		
	Upset	ASME III, NB-3223		
	Emergency	ASME III, NB-3224		
	Faulted	ASME III, NB-3225		
Pressure Vessels (ASME VIII, Div. 1)	Normal	ASME III, NC-3300 or ND 3300		
		σ_m $(\sigma_m$ or $\sigma_L) + \sigma_b$		
	Upset	1.1 S 1.65 S		
	Emergency	1.5 S 1.8 S		
	Faulted	2.0 S T 2.4 S		

Note:

Allowable values for pipe stress analysis noted in the Table are based on ASME Code, Section III, 1971 Edition including Addenda through Summer 1973 and the ASME Code Case 1606.

[ASME Code Class 2 and 3 piping designed to ASME Code, Section III, Division 1 utilizes the stress limits specified in ASME Code Case 1601. (Approved by the council on December 16, 1974 and accepted by the NRC Regulatory Guide 1.84 (R9)). This Code Case was annulled by 1606-1 (Case N-53) and the provisions were incorporated into ASME Section III, Division 1, Winter 1976 Addenda.]

NOTES TO TABLES 3.9-6 AND 3.9-7

1. Components are designed to ASME Section III Class 2 and Class 3 requirements in effect at time of purchase order. These Design Criteria provide guidance for treatment of the loading combinations specified in Table 3.9-5 and Table 3.9-5A for the normal, upset, emergency, and faulted plant conditions.

2. Terms relating to stress analysis are as defined in ASME Section III, NB-3213 & NC-3611.1(a).

   σ_m = general membrane stress. This stress is equal to the average stress across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by mechanical loads.

   σ_L = local membrane stress. This stress is the same as σ_m except that it includes the effect of discontinuities.

   σ_b = bending stress. This stress is equal to the linear varying portion of the stress across the solid section under consideration. Excludes discontinuities and concentrations. Produced only by mechanical loads.

   P = design pressure.

   S = allowable stress value given in Tables I-7.1, I-7.2 and I-7.3 of Appendix I of ASME Section III. The allowable stress shall correspond to the highest metal temperature at the section under consideration during the condition under consideration.

   The term "stress" in the above definitions means the maximum normal stress.

3. The maximum pressure (Pmax) shall not exceed the design pressure (P) by the multiple specified in the stress limit column. If the pressure rating limits are met at the operating conditions, the stress limits are considered to be satisfied.

TABLE 3.9-8
QUALITY GROUP B AND C ACTIVE PUMPS

	Pumps	Safety Function
1.	Auxiliary Feedwater Pump (Motor Driven)	Operate (AFAS)
	Auxiliary Feedwater Pump (Turbine Driven)	Operate (AFAS)
2.	Component Cooling Water Pumps	Operate (SIAS)
3.	Intake Cooling Water Pumps	Operate (SIAS)
4.	Containment Spray Pumps	Operate (CSAS)
5.	Diesel Oil Transfer Pumps	Operate (SIAS)
6.	High Pressure Safety Injection Pumps	Operate (SIAS)
7.	Low Pressure Safety Injection Pumps	Operate (SIAS)
8.	Charging Pumps	Operate (SIAS)
9.	Boric Acid Makeup	Operate (SIAS)
10.	Hydrazine Pump	Operate (CSAS)

TABLE 3.9-9
NSSS SUPPLIED ACTIVE VALVES

Component	Service	Temp °F	Press Psig	RTG Lbs	Function	Size (In.)	Type Valve	Type Operator
1. Safety Injection								
HCV-3618	SIT Recirc/Drain	650	2485	1500	Close (SIAS)	1	Globe	Pneumatic
HCV-3628	SIT Recirc/Drain	650	2485	1500	Close (SIAS)	1	Globe	Pneumatic
HCV-3638	SIT Recirc/Drain	650	2485	1500	Close (SIAS)	1	Globe	Pneumatic
HCV-3648	SIT Recirc/Drain	650	2485	1500	Close (SIAS)	1	Globe	Pneumatic
V3572	SIT Fill Header	350	2485	1500	Close (SIAS)	1	Globe	Pneumatic
V3571	SIT Fill Header	350	2485	1500	Close (SIAS)	1	Globe	Pneumatic
V3444	RWT Isolation	350	350	300	Open/Close (R-M)	14	Gate	Motor
V3432	RWT Isolation	350	350	300	Open/Close (R-M)	14	Gate	Motor
V3495	S I Pump Recirc.	350	1750	900	Close (RAS)	6	Globe	Solenoid
V3496	S I Pump Recirc.	350	1750	900	Close (RAS)	6	Globe	Solenoid
V3550	Hot Leg Inj.	650	2485	1500	Open (R-M)	3	Globe	Motor
V3551	Hot Leg Inj.	650	2485	1500	Open (R-M)	3	Globe	Motor
V3540	Hot Leg Inj.	650	2485	1500	Open (R-M)	3	Globe	Motor
V3523	Hot Leg Inj.	650	2485	1500	Open (R-M)	3	Globe	Motor
HCV-3615	L.P. Header	650	2485	1500	Open (SIAS)	6	Globe	Motor
HCV-3625	L.P. Header	650	2485	1500	Open (SIAS)	6	Globe	Motor
HCV-3635	L.P. Header	650	2485	1500	Open (SIAS)	6	Globe	Motor
HCV-3645	L.P. Header	650	2485	1500	Open (SIAS)	6	Globe	Motor
HCV-3616	HPSI Header	650	2485	1500	Open (SIAS)	2	Globe	Motor
HCV-3626	HPSI Header	650	2485	1500	Open (SIAS)	2	Globe	Motor
HCV-3636	HPSI Header	650	2485	1500	Open (SIAS)	2	Globe	Motor
HCV-3646	HPSI Header	650	2485	1500	Open (SIAS)	2	Globe	Motor
HCV-3617	HPSI Reader	650	2485	1500	Open (SIAS)	2	Globe	Motor
HCV-3627	HPSI Header	650	2485	1500	Open (SIAS)	2	Globe	Motor
HCV-3637	HPSI Header	650	2485	1500	Open (SIAS)	2	Globe	Motor
HCV-3647	HPSI Header	650	2485	1500	Open (SIAS)	2	Globe	Motor
V3517	SDC HX Isol.	350	500	300	Open (R-M)	12	Gate	Motor
V3658	SDC HX Isol.	350	500	300	Open (R-M)	12	Gate	Motor
FCV-3301	LPSI Discharge	350	500	300	Modulate SDC Flow	10	B'fly	Motor
FCV-3306	LPSI Discharge	350	500	300	Modulate SDC Flow	10	B'fly	Motor
HCV-3512	SDC-HX Outlet	350	500	300	Open (R-M)	10	B'fly	Motor
HCV-3657	SDC-HX Outlet	350	500	300	Open (R-M)	10	B'fly	Motor
V3456	SDC-HX Outlet	350	500	300	Open (R-M)	10	Gate	Motor
V3457	SDC-HX Outlet	350	500	300	Open (R-M)	10	Gate	Motor
V3611	SIT Recirc/Drain	350	700	900	Close (SIAS)	1	Globe	Pneumatic
V3621	SIT Recirc/Drain	350	700	900	Close (SIAS)	1	Globe	Pneumatic
V3631	SIT Recirc/Drain	350	700	900	Close (SIAS)	1	Globe	Pneumatic
V3641	SIT Recirc/Drain	350	700	900	Close (SIAS)	1	Globe	Pneumatic
V3659	SI Pump Recirc.	350	1750	900	Close (RAS)	3	Gate	Motor
V3660	SI Pump Recirc.	350	1750	900	Close (RAS)	3	Gate	Motor
V3654	HPSI Discharge	350	1750	900	Close (R-M)	6	Gate	Motor
V3656	HPSI Discharge	350	2485	1500	Close (R-M)	6	Gate	Motor
V3536	SDC Warmup	350	350	300	Open (R-M)	4	Globe	Motor

TABLE 3.9-9 (Cont'd)

Component	Service	Temp °F	Press Psig	RTG Lbs	Function	Size (In.)	Type Valve	Type Operator
V3539	SDC Warmup	350	350	300	Open (R-M)	4	Globe	Motor
V3480(1)	SDC Isol.	650	2485	1500	Open (R-M)	10	Gate	Motor
V3481(1)	SDC Isol.	650	2485	1500	Open (R-M)	10	Gate	Motor
V3664(1)	SDC Isol.	350	350	300	Open (R-M)	10	Gate	Motor
V3652(1)	SDC Isol.	650	2485	1500	Open (R-M)	10	Gate	Motor
V3651(1)	SDC Isol.	650	2485	1500	Open (R-M)	10	Gate	Motor
V3665(1)	SDC Isol.	350	350	300	Open (R-M)	10	Gate	Motor
V3545(1)	SDC Isol.	650	2485	1500	Open (R-M)	10	Gate	Motor
V3614	SIT Isol.	650	2485	1500	Open (BIAS)	12	Gate	Motor
V3624	SIT Isol.	650	2485	1500	Open (SIAS)	12	Gate	Motor
V3634	SIT Isol.	650	2485	1500	Open (SIAS)	12	Gate	Motor
V3644	SIT Isol.	650	2485	1500	Open (SIAS)	12	Gate	Motor
V3733,3734 3735,3736 3737,3738 3739,3740	SIT Vent Valves	350	700	600	Open & Close	1	Globe	Solenoid
Chemical & Volume Control System								
V2650	BAMT Recir.	200	200	150	Close (SIAS)	1	Globe	Pneumatic
V2651	BAMT Recir.	200	200	150	Close (SIAS)	1	Globe	Pneumatic
V2515	Letdown Isol.	650	2485	1500	Close (SIAS)	2	Globe	Pneumatic
V2516	Letdown Isol.	650	2485	1500	Close(SIAS/CIAS)	2	Globe	Pneumatic
V2522	Letdown Isol.	550	2485	1500	Close (CIS)	2	Globe	Pneumatic
V2501	VCT Isol.	250	200	150	Close (SIAS)	4	Gate	Motor
V2504	RWT Isol.	200	200	150	Close (R-M)	3	Gate	Motor
V2508	BAMT Suction	200	200	150	Open (SIAS)	3	Gate	Motor
V2509	BAMT Suction	200	200	150	Open (SIAS)	3	Gate	Motor
V2525(3)	VCT Isol.	250	200	150	Closes (SIAS)	4	Gate	Motor
FCV-2210Y	BAMP Discharge	200	200	150	Closes (SIAS)	1	Globe	Pneumatic
V2524(2)	RCP Leakoff	550	2485	1500	Close (CIS)	3/4	Globe	Pneumatic
V2505(2)	RCP Leakoff	250	2485	1500	Close (CIS)	3/4	Globe	Pneumatic
V2514	BAMP Discharge	200	200	150	Open (SIAS)	3	Gate	Motor
V2553	Charging Pump Bypass	250	2735	1500	Close (R-M)	2	Globe	Motor
V2554	Charging Pump Bypass	250	2735	1500	Close (R-M)	2	Globe	Motor
V2555	Charging Pump Bypass	250	2735	1500	Close (R-M)	2	Globe	Motor

TABLE 3.9-9 (Cont'd)

Component	Service	Temp °F	Press Psig	RTG Lbs	Function	Size (In.)	Type Valve	Type Operator
Sampling System								
V5201	Press. Surge Sample	675	2485	1500	Close (CIS)	3/8	Globe	Solenoid
V5202	Press. Steam Sample	675	2485	1500	Close (CIS)	3/8	Globe	Solenoid
V5200	Hot Leg Sample	675	2485	1500	Close (CIS)	3/8	Globe	Solenoid
V5203	Hot Leg Sample	675	2485	1500	Close (CIS)	3/8	Globe	Pneumatic
V5204	Press. Surge Sample	675	2485	1500	Close (CIS)	3/8	Globe	Pneumatic
V5205	Press. Steam Sample	675	2485	1500	Close (CIS)	3/8	Globe	Pneumatic
Waste Management								
V6341	RDT Discharge	250	80	150	Close (CIS)	3	Diaph.	Pneumatic
V6342	RDT Discharge	250	80	150	Close (CIS)	3	Diaph.	Pneumatic
V6718	Cont. Vert Header	200	80	150	Close (CIS)	1	Diaph.	Pneumatic
V6750	Cont. Vent Header	200	80	150	Close (CIS)	1	Diaph.	Pneumatic
V6741	Nitrogen Supply	150	900	400	Close (CIS)	1	Diaph.	Pneumatic
Reactor Coolant System								
V1460, 1461 1463, 1464, 1466,1462,1465	Head Vent System	700	2485	1500	Open & Close	1	Globe	Solenoid
V1474, 1475	Pressurizer Relief	675	2485	2500	Open & Close	3 x 8	Relief	Solenoid
V1476, 1477	PORV Isolation	675	2485	2500	Open & Close	3	Gate	Motor

Note: Operability of the above valves have been verified as per FP&L letter L-83-428, dated July 26, 1983.

For Reasons Listed Below the Below Identified Valves were Stroked Dry or Not Checked.

(1) Operation would result in loss of suction to an ESF pump - MV-07-1A, MV-07-1B, V3480, V3481, V3664, V3652, V3651, V3665 and V3545.

(2) Operation would isolate RCP bleed off flow - V2524 and V2505.

(3) Operation would result in excessive boration or dilution of plant during fuel loading - V2525.

TABLE 3.9-10

A/E SUPPLIED QUALITY GROUP B AND C ACTIVE VALVES

	Component	Service	Temp °F	Press Psig	RTG Lbs	Function	Size (In)	Type
1.	Main Steam System							
	MV-08-12,13	Aux Turbine Supply	550	985	600	Open(AFAS)	4	Gate
	MV-08-3	Aux Turbine Stop			600	Open(RM)	4	Globe
	MV-08-18A,18B, 19A, 19B	ADV	550	985	600	Open(RM)	10	Globe
	HCV-08-1A, 1B	MSIV	550	985	600	Close(MSIS)	34	Angle Globe
	MV-08-14 to 17	ADV Block	550	985	600	Close(M)	8	Gate
	MV-08-1A, 1B	MFIV Bypass	530	995	1500	NC/Close (MSIS)	3	Globe
2.	Main Feedwater							
	HCV-09-1A,1B (8) 2A,2B (8)	MFIV	500	1875	900	Close(MSIS/AFAS)*	20	Gate
	MV-09-9,10,11,12	Aux Feed Isol.	120	1420	600	Open(AFAS)**	4	Globe
	MV-09-13,14	Aux Feed Intertie	120	1420	600	Open(M)	2-1/2	Gate
	SE-09-2,3,4,5	Aux Feed Isol.	120	1420	900	Open(AFAS)**	4	Globe
3.	Component Cooling							
	MV-14-1, 3 (6)	CCW Pump Isol	200	150	150	NO (9)	24	B'tfly
	MV-14-2, 4 (6)	CCW Pump Isol	200	150	150	NC (10)	24	B'tfly
	HCV-14-6,7	RCP Isol	200	150	150	Close(SIAS)	8	B'tfly
	HCV-14-3A,3B	SDC HX Isol	200	150	150	Open(SIAS)	14	B'tfly
	HCV-14-8A,8B	CCW Train Isol	200	150	150	Close(SIAS)	16	B'tfly
	MV-14-9 to 16	Fan Cooler Isol	200	150	150	Opens (NO)	8	B'tfly
	MV-14-17,18	Fuel Pool Iso.	200	150	150	Close(SIAS)	12	B'tfly
	HCV-14-9,10	CCW Train Isol	200	150	150	Close(SIAS)	16	B'tfly
	HCV-14-1,2	RCP Isol	200	150	150	Close(SIAS)	8	B'tfly
	MV-14-19,20	Fuel Pool Isol.	200	150	150	Close (RM)	12	B'tfly
4.	Intake Cooling							
	MV-21-2,3	Turbine Cool HX Isol	125	100	150	Close(SIAS)	24	B'tfly
	HCV-21-7A, 7B	Debris Discharge	125	150	150	Close(SIAS)	6	Ball
5.	Cont Spray							
	MV-07-1A,B (3)	RWT Isol	300	65	150	Close(RAS)	24	B'tfly
	FCV-07-1A,B (5)	Cont Isol	250	500	300	Open(CSAS)	12	B'tfly
	MV-07-2A,B (4)	Sump Isol	300	65	150	Open(RAS)	24	B'tfly
	LCV-07-11A,B	Cont Isol	240	100	150	Close(CIS,SIAS)	2	Globe
	SE-07-3A,3B	Hydrazine Isol	120	100	600	Open(CSAS)	1/2	Globe
	MV-07-3,4	Containment Spray Isol	300	500	300	Close(RM)	12	Gate
	SE-07-5A,5B, 5C,5D,5E,5F	Containment Sampling	420	175	600	Close(RM)	3/8	Globe

*The AFAS maybe overridden and the valve re-opened by the control room operator only during 2-EOP-06, total loss of feedwater.
** These valves have a close function in the event of a faulted S/G.

TABLE 3.9-10 (Cont'd)

	Component	Service	Temp °F	Press Psig	Rtg Lbs	Function(1)	Size (in)	Type
6.	Emer Diesel							
	SE-59-1A1, 1A2	Day Tank 2A1, 2A2	-	-	600	Open(NC)	1.5	Globe
	SE-59-1B1, 1B2	Day Tank 2B1, 2B2	-	-	600	Open(NC)	1.5	Globe
7.	HVAC/Drainage							
	HCV-25-5 to 7 (7)	ECCS Sump Isol	125	150	150	Close(RM)	4	Globe
	HCV-25-5A to 7A(7)	ECCS Sump Isol	125	150	150	Close(RM)	4	Globe
	HCV-25-1 to 4 (7)	ECCS Sump Isol	125	150	150	Close (RM)	3	Gate
	HCV-25-1A to 4A(7)	ECCS Sump Isol	125	150	150	Close (RM)	3	Gate
	FCV-25-1,6	Containment Purge	265	-1/2 to 15	75	Close(CIAS)	48	B'tfly
	FCV-25-2,3,4,5	Containment Purge	265	-1/2 to 15	75	Close(CIAS)	48	B'tfly
	FCV-25-20,21,26,36	Containment Isolation	350	-3 to 65	150	Close(CIAS)	8	B'tfly
	FCV-25-29	H_2 Cont Purge	100	44	150	Open(RM)	4	B'tfly
	FCV-25-34	H_2 Cont Purge	100	44	150	Open(RM)	4	B'tfly
	FCV-25-30,31	SFP Exhaust	200	-3 to 5	150	Close(CIAS)	20	B'tfly
	FCV-25-32,33	SBVS Inlet	200	-3 to 5	150	Open(CIAS)	30	B'tfly
	FCV-25-7,8	Cont. Vacuum relief	300	-1 to 65	150	Close/Open	24	B'tfly
	FCV-25-11,12	SBVS Isol	150	-3 to Atm	150	Close	16	B'tfly
	FCV-25-13	SBVS Bypass	200	-3 to 5	150	Open	12	B'tfly
8.	Control Room A/C							
	FCV-25-14 to 17	Air Intake	150	-15 WG	150	Close (CIAS)	12	B'tfly
	FCV-25-18,19	Toilet Exhaust	150	2 in	150	Close(CIAS)	6	B'tfly
	FCV-25-24,25	Kitchen Exhaust	150	2 in	150	Close(CIAS)	10	B'tfly
9.	Cont Air Monit.							
	FCV-26-1,3,5	Rad Monit Isol	300	60	150	Closes(CIAS)	1	Globe
	FCV-26-2,4,6	Rad Monit Isol	300	60	150	Closes(CIAS)	1	Globe
10.	Hydrogen Sampling							
	FSE-27-8 to14	Cont Isol	340	44	150	Closes(RM)	3/8	Globe
	FSE-27-15 to18	Cont Isol	340	44	150	Closes(RM)	3/8	Globe
11.	Stm Gen Blowdown							
	FCV-23-3,5	SG Blowdown	550	985	600	Closes(CIAS)	3	Globe
	FCV-23-7,9	SG Sample	550	985	600	Closes(CIAS)	1/2	Globe
12.	Inst Air							
	HCV-18-1	Inst Air to Cont	115	100	150	Closes(CIAS)	1	Globe

TABLE 3.9-10 (Cont'd)

Component	Service	Temp °F	Press Psig	RTG Lbs	Function(1)	Size (in)	Type
13. Make Up Water							
HCV-15-1	Primary Makeup	120	125	150	Close(CIAS)	2	Globe
14. Sampling System & Safety Injection							
SE-03-1A,1B, 1C,1D	SIT Isolation	350	700	600	Close(SIAS)	1	Globe
SE-05-1A to 1E	Containment Isolation	300	700	600	Close (CIAS)	3/8	Globe
SE-03-2A,2B	Containment Isolation	350	700	600	Close (CIAS,SIAS)	2	Globe
15. CVCS							
SE-02-1	Charging Line Isol	650	2735	1500	Close(RM)	2	Globe
SE-02-2	Charging Line Isol	650	2735	1500	Close(RM)	2	Globe
SE-02-3	Aux Spray Isol	650	2735	1500	Close(RM)	2	Globe
SE-02-4	Aux Spray Isol	650	2735	1500	Close(RM)	2	Globe
16. STATION AIR							
HCV-18-2	Service Air to Cont	125	150	600	Close(CIAS)	2	Globe

NOTES: (1)
M = Manual
NC = Normally closed
NO = Normally open
RM = Remote manual
SIA = Safety Injection Action Signal
MSI = Main Steam Isolation Signal
CS = Containment Spray Actuation Signal
CIA = Containment Isolation Actuation Signal
RA = Recirculation Actuation Signal
AFA = Auxiliary Feedwater Actuation Signals

(2) Operability of the above valves has been verified as per FP&L letter L-83-428, dated July 26, 1983.

For reasons listed below the below identified valves were stroked dry or not checked.

(3) Operation would result in loss of suction to an ESF pump - MV-07-1A, MV-07-1B, V3480, V3481, V3664, V3652, V3651, V3665 and V3645.

(4) Operation under flow would require flooding of containment sump - MV-07-2A, MV-07-2B, V07119 and V07120.

(5) Operation under flow would spray down the containment - FCV-07-1A and FCV-07-1B.

TABLE 3.9-10(Cont'd)

NOTES: (Cont'd)

(6) The potential for loss of suction to CCW pump would result from operation deeming unnecessary since valve positions are not altered under flow for any design based accident - MV-14-1, MV-14-2, MV-14-3, MV-14-4.

(7) Flow through valves is due to gravity and would not amount to a significant DP - HCV-25-1 to HCV-25-7 and HCV-25-1A to HCV-25-7A.

(8) Operation would result in undue transient on secondary plant-HCV-09-1A, HCV-09-1B, HCV-09-2A and HCV-09-2B.

(9) These valves will be closed if the “C” CCW pump is supplying the “B” CCW header

(10) These valves will be open if the “C” CCW pump is supplying the “B” CCW header

TABLE 3.9-11

NSSS - SUPPLIED SEISMIC AND CODE CLASS SAFETY/RELIEF VALVES

System	Seismic Category	Code Class	Valve Number	Line No.	Service	Size(In.)	Set Pressure (psig)	Design Pressure (psig)	Design Temp (°F)
Reactor Coolant	I	1	V1200	6-RC-827	Pressurizer	3 x 6	2485	2485	700
	I	1	V1201	6-RC-828	Pressurizer	3 x 6	2485	2485	700
	I	1	V1202	6-RC-829	Pressurizer	3 x 6	2485	2485	700
	-	3	V1242	1-1/2-RC-502	Quench Tank	1-1/2 x 2	70	100	350
	I	1	V1474	3-RC-309	Pressurizer	3 x 8	2400	2485	700
	I	1	V1475	3-RC-310	Pressurizer	3 x 8	2400	2485	700
Chemical and Volume Control	-	2	V2115	4-CH-554	Volume Control Tank Discharge	4 x 6	75	75	250
	I	2	V2199	1-1/2-CH-140	RC Pumps Bleedoff	1-1/2 x 2	150	2485	550
	I	2	V2311	1/2-CH-957	Charging Pumps Suction	1/2 x 1	150	200	250
	I	2	V2588	1/2-CH-534	Charging Bypass	1/2 x 1	150	200	250

TABLE 3.9-11 (Cont'd)

System	Seismic Category	Code Class	Valve Number	Line No.	Service	Size(In.)	Set Pressure (psig)	Design Pressure (psig)	Design Temp (°F)
Chemical and Volume Control (Cont'd)	I	2	V2318	1/2-CH-948	Charging Bypass	1/2 x 1	150	200	250
	I	2	V2321	1/2-CH-542	Charging Bypass	1/2 x 1	150	200	250
	I	2	V2324	1-1/2-CH-121	Charging Pump 2A	1-1/2 x 2	2735	2735	250
	I	2	V2325	1-1/2-CH-117	Charging Pump 2B	1-1/2 x 2	2735	2735	250
	I	2	V2326	1-1/2-CH-113	Charging Pump 2C	1-1/2 x 2	2735	2735	250
	-	2	V2345	2-CH-312	Letdown	2 x 3	600	650	550
	-	2	V2531	3-CH-516	Letdown	2 x 3	200	200	250
	-	2	V2446	1/2-CH-630	RWT to VCT	3/4 x 1	200	200	250
	-	2	V2447	1/2-CH-911	RWT to VCT	3/4 x 1	200	200	250
Safety Injection	I	2	V3211	1-1/2-SI-529	SIT 2A2	1-1/2 x 2-1/2	669	700	200
	I	2	V3221	1-1/2-SI-528	SIT 2A1	1-1/2 x 2-1/2	669	700	200
	I	2	V3231	1-1/2-SI-530	SIT 2B1	1-1/2 x 2-1/2	669	700	200
	I	2	V3241	1-1/2-SI-531	SIT 2B2	1-1/2 x 2-1/2	669	700	200
	I	3	V3407	1/2-SI-490	SIT Recirc	1/2 x 1	650	700	350
	I	2	V3412	1/2-SI-225	High-Press. Header B	1 x 2	1585	1600	350
	I	2	V3417	1-SI-136	High-Press. Header A	1 x 2	2485	2485	650

TABLE 3.9-11 (Cont'd)

System	Seismic Category	Code Class	Valve Number	Line No.	Service	Size(In.)	Set Pressure (psig)	Design Pressure (psig)	Design Temp (°F)
Safety Injection (Cont'd)	I	2	V3430	1-SI-806	Shutdown Hx 2B Outlet	1 x 2	500	500	350
	I	2	V3431	1-SI-807	Shutdown Hx 2A Outlet	1 x 2	500	500	350
	I	2	V3439	1/2-SI-465	Low-Press. Header A	1 x 2	535	500	350
	I	3	V3466	1-1/2-SI-489	RWT Return	1-1/2x2-1/2	700	700	350
	I	2	V3468	2-SI-574	Loop 2B Shutdown	2 x 3	335	350	350
	I	1	V3469	3/4-SI-135	Loop 2B Shutdown	3/4 x 1	2485	2485	650
	I	1	V3482	3/4-SI-201	Loop 2A Shutdown	3/4 x 1	2485	2485	650
	I	2	V3483	2-SI-575	Loop 2A Shutdown	2 x 3	335	350	350
	I	2	V3507	1/2-SI-168	Low-Press. Header B	1 x 2	535	500	350
	I	2	V3513	2-SI-174	RWT Return	2 x 3	500	500	350
	I	2	V3570	1-SI-532	High-Press. Header	1 x 2	2400	2485	650
	I	2	V3666	6-SI-365	Loop 2B Shutdown	6 x 8	335	350	350
	I	2	V3667	6-SI-364	Loop 2A Shutdown	6 x 8	335	350	350
	I	2	V3688	2-SI-483	RWT Return	2 x 3	500	500	350
Main Steam	I	2	V8201	6-MS-63	Main Steam	6 x 10	985	1025	550
	I	2	V8202	6-MS-64	Main Steam	6 x 10	985	1025	550
	I	2	V8203	6-MS-65	Main Steam	6 x 10	985	1025	550
	I	2	V8204	6-MS-66	Main Steam	6 x 10	9B5	1025	550

TABLE 3.9-11 (Cont'd)

System	Seismic Category	Code Class	Valve Number	Line No.	Service	Size(In.)	Set Pressure (psig)	Design Pressure (psig)	Design Temp (°F)
Main Steam (Cont'd)	I	2	V8205	6-MS-67	Main Steam	6 x 10	985	1025	550
	I	2	V8206	6-MS-68	Main Steam	6 x 10	985	1025	550
	I	2	V8207	6-MS-69	Main Steam	6 x 10	985	1025	550
	I	2	V8208	6-MS-70	Main Steam	6 x 10	985	1025	550
	I	2	V8209	6-MS-71	Main Steam	6 x 10	1025	1025	550
	I	2	V8210	6-MS-72	Main Steam	6 x 10	1025	1025	550
	I	2	V8211	6-MS-73	Main Steam	6 x 10	1025	1025	550
	I	2	V8212	6-MS-74	Main Steam	6 x 10	1025	1025	550
	I	2	V8213	6-MS-75	Main Steam	6 x 10	1025	1025	550
	I	2	V8214	6-MS-76	Main Steam	6 x 10	1025	1025	550
	I	2	V8215	6-MS-77	Main Steam	6 x 10	1025	1025	550
	I	2	V8216	6-MS-78	Main Steam	6 x 10	1025	1025	550

TABLE 3 9-12
A/E SUPPLIED SEISMIC AND CODE CLASS SAFETY/RELIEF VALVES

System	Seismic Category	Code Class	Valve Number	Line No.	Service	Size(In.)	Set Pressure (psig)	Design Pressure (psig)	Design Temp (psig)
Containment Spray	I	2	SR-07-1A	14-SI-511	LPSI Pump 2A Suction	3/4 x 1	60	60	300
	I	2	SR-07-1B	14-SI-512	LPSI Pump 2B Suction	3/4 x 1	60	60	300
	I	2	SR-07-1C		IRS Tank	1 x 1	20	20	120
	I	2	SR-07-2A	1/2-CS-89	Hydrazine Pump 2A Disch	1/2 x 1/2	100	100	120
	I	2	SR-07-2B	1/2-CS-90	Hydrazine Pump 2B Disch	1/2 x 1/2	100	100	120
Component Cooling	I	2	SR14307 SR14318 SR14329 SR14342	8-CC-41 8-CC-42 8-CC-43 8-CC-44	Containment Cig Unit Disch	1 x 1 1/2	150	150	200
	I	3	SR14350 SR14359	14-CC-20 14-CC-21	Sht Dn HX Inlets	1 x 1 1/2	150	150	200
Diesel Oil	I	3	SR17221	3/4-DO-19	DO Pump 2A Disch	3/4/ x I	100	100	120
	I	3	SR17222	3/4-DO-20	DO Pump 2B Disch	3/4 x 1	100	100	120
Instrument Air	I	3	SR-18-6A	1/2-IA-62	Main Hatch Door Seal A	1/2 x 1	35	150	125
	I	2	SR-18-6B	1/2-IA-64	Main Hatch Door Seal B	1/2 x 1	35	150	125
Main Steam	I	2	MV-08-18A	10-MS-124	Atmospheric Dump	10 x 12	variable	985	550
	I	2	MV-08-18B	10-MS-126	Atmospheric Dump	10 x 12	variable	985	550
	I	2	MV-08-19A	10-MS-125	Atmospheric Dump	10 x 12	variable	985	550
	I	2	MV-08-19B	10-MB-127	Atmospheric Dump	10 x 12	variable	985	550

TABLE 3.9-13

CODE CLASS I

SAFETY RELIEF VALVE LOADING COMBINATION

Component Operating Conditions	Equation	Stress Level in Connecting Pipe
1) Design	NB-3652 Eq. 9	1.5 S_m
2) Normal	NB-3653.1 Eq. 10	3.0 S_m
	NB-3653.2 Eq 11	Cumulative usage factor less than 1.0
3) Upset	NB-3653.1 Eq 10	3.0 S_m
	NB-3653.2 Eq. 11	Cumulative usage factor less than 1.0
4) Emergency	NB-3652 Eq. 9	2.25 S_m
5) Faulted	NB-3652 Eq. 9	3.0 S_m
	Code Class 2 & 3	
1) Normal	NC-3652.1 Eq. 8	S_h
2) Upset	NC-3652.2 Eq. 9	1.2 S_h
3) Emergency	NC-3652.2 Eq. 9	1.8 S_h
4) Faulted	NC-3652.2 Eq. 9	2.4 S_h

TABLE 3.9-14

LOADING COMBINATIONS ASME CODE CLASS 1 NSSS COMPONENTS
EXCEPT VALVES (TABLE 3.9-1)

Condition	Design Loading Combination[(a)]
Design (c)	PD
Normal (b)	PO + DW
Upset (b)	PO + DW + OBE
Emergency	PO + DW + DE
Faulted	PO + DW + DBE + DF

(a) Legend:

PD	=	design pressure
PO	=	operating pressure
DW	=	dead weight
OBE	=	operating basis earthquake
DBE	=	design basis earthquake
DE	=	dynamic system loadings associated with the emergency condition (5 cycles of complete loss of secondary pressure)
DF	=	dynamic system loadings associated with a postulated pipe rupture (LOCA) or steam line break

(b) As required by ASME Code Section III, Division I, other loads such as thermal transient, thermal gradient, and anchor point displacement portions of the OBE require consideration in addition to the primary stress producing loads listed.

(c) For the Design Condition for the replacement CEDMs, Design Pressure, Design Mechanical Loads (including Deadweight), plus the OBE Loads were considered.

Method of combination: NUREG-0484

TABLE 3.9-15

STRESS LIMITS FOR ASME CODE CLASS 1
NSSS COMPONENT EXCEPT VALVES

(TABLE 3.9-1)

Condition	Stress Limits[a]
Normal and upset	NB 3223 and NB 3654
Emergency	NB 3224 and NB 3655
Faulted	NB 3225 and NB 3656

(a) As specified in ASME Section III, 1971 and applicable addenda.

STRESS LIMITS FOR ASME CODE CLASS I
REPLACEMENT CEDMS

Service Level (Condition)	Stress Categories & Limits of Stress Intensities[b]
Design	Figure NB-3221-1 including notes
A + B (Normal and Upset)	Figure NB-3222-1 including notes
C (Emergency)	Figure NB-3224-1 including notes
D (Faulted)	Paragraph F-1330 or F-1340, Appendix F, Rules for Evaluation of Service Loadings with Level D Service Limits

(b) As specified in ASME Section III, 1998 Edition through 2000 Addenda.

TABLE 3.9-16

LOADING COMBINATIONS FOR
NSSS VALVES CLASS 1, 2 and 3

Conditions	Loading
Design	PD
Normal	PO + DW + SSE + T
Upset[(1)]	PO + DW + SSE + T
Emergency[(2)]	PO + DW + SSE + T

[(1)]Jamesbury supplied valves only

[(2)]Fischer supplied valves only

PO - Operating Pressure

PD - Design Pressure

DW - Dead Weight

SSE - Safe Shutdown Earthquake

T - Transients

Note: All loads are absolutely summed

TABLE 3.9-17

DESIGN STRESS LIMITS FOR
NSSS VALVES CLASS 1, 2 and 3

Condition	Stress Limits[2]		
	σm	$(\sigma m \text{ or } \sigma_l) + \sigma_b$	P max
Design	Sm	1.5 Sm	P
Normal	Sm	1.5 Sm	P
Upset	1.1 Sm	1.65 Sm	1.1 P
Emergency	1.2 Sm or Sy [1]	1.8 Sm[1]	1.2 P

Notes:

(1) Greater of 1.2 Sm or Sy

(2) For Class 2 and 3 valves Sm is replaced by S

TABLE 3.9-18

LOADING COMBINATIONS FOR
NSSS PUMPS CLASS 2 AND 3

Design = DP+DW+NL+SSE

DP	=	design pressure
DW	=	dead weight
NL	=	nozzle loads (include piping imposed thermal expansion, dead weight and seismic)
SSE	=	safe shutdown earthquake

Note:
All loads are absolutely summed.

TABLE 3.9-19

DESIGN STRESS LIMITS
FOR
CODE CLASS 2 AND 3 NSSS PUMPS

Condition	Stress Limit	
Design	$\frac{\sigma_m}{S}$	$\frac{(\sigma_m \text{ or } \sigma_1) + \sigma_b}{1.5S}$

Notes:

(1) For the LPSI Pump, S is replaced by Sm

(2) Supports satisfy the stress limits of ASME III, NF. For those pumps ordered prior to the issuance of NF, supports satisfy the AISC stress limits.

TABLE 3.9-20

LOADING COMBINATIONS FOR NSSS ASME CODE
CLASS 2 AND 3 COMPONENTS OTHER THAN VALVES AND PUMPS
(VESSEL AND SUPPORTS)

Condition	Design Loading Combinations [(a)]
Design	PD + NL
Normal	PO + DW + NL
Upset	PO + DW + OBE + NL
Faulted	PO + DW + SSE + NL[(1)]

(a) Legend:

PD = design pressure

PO = operating pressure

DW = dead weight

OBE = operating basis earthquake

SSE = safe shutdown earthquake

NL = nozzle loads (includes piping imposed thermal expansion, dead weight and seismic)

(1) Nozzle loads are increased by 50 percent for the faulted loading condition. The 50 percent increase for faulted nozzle loads is confirmed to be acceptable by the piping analysis.

Note:
All loads are absolutely summed.

TABLE 3.9-21

DESIGN STRESS LIMITS FOR
CODE CLASS 2 & 3 NSSS COMPONENTS
OTHER THAN VALVES AND PUMPS
(VESSEL & SUPPORTS)

Components	Condition	Stress Limits	
		σ_m	$(\sigma_m$ or $\sigma_1) + \sigma_b$
Safety Injection Tank	Normal	Sm	1.5 Sm
	Upset	Sm	1.5 Sm
	Faulted	2.0 Sm	2.4 Sm
Pressure Vessels	Normal	ASME III NC 3300 or ND 3300	
		σ_m	$(\sigma_m$ or $\sigma_1) + \sigma_b$
	Upset	1.1 S	I.65 S
	Emergency	1.5 S	1.8 S
	Faulted	2.0 S	2.4 S
Supports	All	ASME III, NF	

0947W - 1

TABLE 3.9-22

LIST OF VIBRATION TESTING MODES

	Flow Modes for Preoperational Vibration Testing				
Piping Systems	Steady State	Test Level	Transient	Test Level	Instrumentation Required
Main Steam from Steam Generators to MSIVs	100% Power	1	Transient trip at 100% power	2 -	(to be added) -
	Full flow through atmospheric dump Valves, all valves open	1	None	-	-
Main Steam to Auxiliary Feedwater Pump Turbine	Run at full pump flow	1	AFW turbine trip at full pump flow	1	None
Feedwater and Auxiliary Feedwater	Single AFW Pump Operation for Pumps 2A, 2B, 2C; recirculation mode	1	Pump start, recirculation mode FW reg valv	1	None
Intake Cooling Water Pumps Discharge Piping	Pump(s) Operating	1	None	-	None
Component Cooling Water	Pump(s) Operating	1	None	-	None
Diesel Oil Transfer Pump Discharge Piping	Pump(s) Operating	1	None	-	None
Steam Generator Blowdown	Flow at normal rate	1	Initiate flow, system cold	1	None
	Flow at maximum rate	1	None		
Reactor Coolant Main Loop	Single and Multiple Pump Operation	1	Pump(s) starts and stops	1	None
			Pressurizer Spray Valve Cycling	2	Hand-held Vibration Amplitude Meter

TABLE 3.9-22 (Cont'd)

Flow Modes for Preoperational Vibration Testing

Piping Systems	Steady State	Test Level	Transient	Test Level	Instrumentation Required
Relief Valve Discharge Piping			PORV operation	4	(to be added)
Chemical & Volume Control System	Letdown flow modes	1	None	-	-
	Boric acid makeup pumps 2A and 2B	1	None	-	-
	Charging Pumps 2A, 2B and 2C Single and Multiple pump operation	2	Single and Multiple Pump starts and stops stops	2	Hand-held Vibration Amplitude Meter
Low Pressure Safety Injection	LPSI Pumps 2A and 2B operating in minimum recircula-tion mode	1	None	-	-
	Shutdown Cooling mode	1	None	-	-
High Pressure Safety Injection	HPSI Pumps 2A and 2B operating in minimum recircula-tion mode	1	None	-	-
	Safety injection mode	1	None	-	-
Fuel Pool Cooling	Pump(s) 2A and 2B operating	1	None	-	-
Containment Spray	Pumps 2A and 2B in minimum recirculation mode	1	None	-	-
Hydrazine Injection	Pumps 2A and 2B operating	1	Pump(s) start and stop	1	None

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TABLE 3.9-24

REACTOR VESSEL SUPPORT LOADS

LOCATION	LOCA ONLY	COMBINED LOCA + N.Op. + SSE	SPECIFICATION
H_1	4.291	4.74	8.00
V_1	4.697	6.47	8.50
H_2	4.100	4.71	7.00
V_2	2.642	3.75	7.00
H_3	3.904	4.44	7.00
V_3	3.216	4.29	7.00

Units - - millions of pounds

TABLE 3.9-25

STEAM GENERATOR SUPPORT LOADS

LOCATION		COMBINED LOCA + N.Op. + SSE	SPECIFICATION
Upper keys (ea.)	Z_1	1.51	2.172
	Z_2	2.00	2.172
Snubbers (ea.)	5	0.22	0.55
SLIDING BASE			
Vertical pads	Y_1	1.71	5.974
	Y_2	2.33	3.588
	Y_3	2.23	2.458
	Y_4	1.72	2.586
Anchor bolts	Y_1	1.85	2.716
(per pair of bolts)	Y_2	1.72	2.856
	Y_3	0.58	2.086
	Y_4	1.73	2.948
Lower stop	X_3[(a)]	5.648	7.085
Lower keys	Z_{11}	3.28	3.755
	Z_{12}	1.06	2.772

Units -.- millions of pounds

Notes:
(a) Historical - Lower stop X_3 is deleted per LBB (See Section 3.6).

TABLE 3.9-26

RCS COMPONENT NOZZLE LOADS

NOZZLE LOCATION	COMBINED LOCA + N.Op. + SSE	RSS MOMENTS SPECIFICATION
R V Inlet	3.47	9.93
R V Outlet	14.01	42.43
S G Inlet	6.73	21.75
S G Outlet	6.20	7.79
RCP Suction	3.90	4.45
RCP Discharge	3.98	5.42

Units - - millions of pounds

TABLE 3.9-27

STRESS LIMITS FOR PIPE SUPPORTS

Reference: MSS SP-58 & AISC Manual - 7th Edition

Shape & Use	Fb Bending	Ft Tension	Fv Shear	Fp Bearing	Tension at Pin Hole
Supplementary Steel	21,600[1] 23,760[2]	N/A	14,400	14,400	N/A
Standard Hanger Components	14,500	14,500	11,600	23,200	10,850
Plates and Bars	14,500	14,500	11,600	21,600	10,850
Rods at Threads	N/A	9,000	N/A	N/A	N/A
Rods - Plain	N/A	14,500	N/A	N/A	N/A
Pins	14,500	N/A	11,600	23,200	N/A
Pipe	15,000	15,000	12,000	See Note 3	N/A
Bars & Plates 304 Steel	11,200	11,200	8,950	17,900	8,400
Pipe 304 Steel	11,200	11,200	8,950	See Note 4	N/A
Bolts	N/A	15,000	12,000	21,600	N/A

For Notes, see next page.

TABLE 3.9-27 (Cont'd)

NOTES:

1. For compact sections as defined in AISC 1.5.1.4.1.

2. For other sections as defined in AISC 1.5.1.4.4 and 1.5.1.4.5.

3. Tables for compression values in AISC 1.5.1.3.1.

4. Calculated per AISC 1.5.1.3.1.

5. These values are used in original design for all loading combinations including faulted. When checking an existing support or restraint design against revised faulted loads stresses are limited to the following:

   stresses in hangers and restraints shall be less than 1.6 times AISC limits, not to exceed 0.96 times material yield stress, where shear yield stress is assumed to be 0.577 times tensile yield stress. Also, stresses shall not exceed 0.90 times critical buckling stress, when that is a controlling factor. Cases where buckling stresses in supports of ASME Class 1, 2 or 3 components exceed 67% of critical buckling stress will be justified on an individual basis that the margin against buckling is sufficient.

TABLE 3.9-28

LOADING COMBINATIONS AND STRESS LIMITS FOR PIPING SUPPORTS

	Plant Operating Condition	Design Load Combination	Piping Support Stress Limit
ASME CODE CLASS 1	Normal	SW	Refer to Table 3.9-27
	Upset	a) SW+OBE	Refer to Table 3.9-27
		b) SW+OBE+FVC	Refer to Tab]e 3.9-27
		c) SW+OBE+RVO	Refer to Table 3.9-27
		d) SW+OBE+T	Refer to Table 3.9-27
	Emergency	SW+OBE+RVO+FVC	Refer to Table 3.9-27
	Faulted(1)	$SW+(SSE^2+FC^2)^{1/2}$	Refer to Table 3.9-27
ASME CODE CLASS 2 & 3	Normal	SW	Refer to Table 3.9-27
	Upset	a) SW+OBE	Refer to Table 3.9-27
		b) SW+OBE+FVC	Refer to Table 3.9-27
		c) SW+OBE+RVO	Refer to Table 3.9-27
	Emergency	SW+OBE+FVC+RVO	Refer to Table 3.9-27
	Faulted(1)	$SW+(SSE^2+FC^2)^{1/2}$	Refer to Table 3.9-27

Notations

SW=	=	Largest of:	a) DW+Max(+)TH
			b) DW+Max(-)TH
			c) DW
DW	=	Deadweight (includes sustained mechanical loads)	
OBE	=	Operating Basis Earthquake	
SSE	=	Safe Shutdown Earthquake	
RVO	=	Relief Valve - includes both open and closed systems	
FVC	=	Fast Valve Closure	
TH	=	Thermal expansion	
SSEI	=	Inertia Portion of SSE	
SSED	=	Displacement Portion of SSE	
T	=	Transient	
FC	=	Dynamic loads associated with plant faulted condition	

(1) a) Where the fundamental frequency of the piping systems is beyond the resonant region of the supporting structure the SSE will be combined in the following manner:

$$SSE = \sqrt{SSEI^2 + SSED^2}$$

b) Where the piping fundamental frequency is not beyond the structural resonant region the SSE will be combined in the following manner:

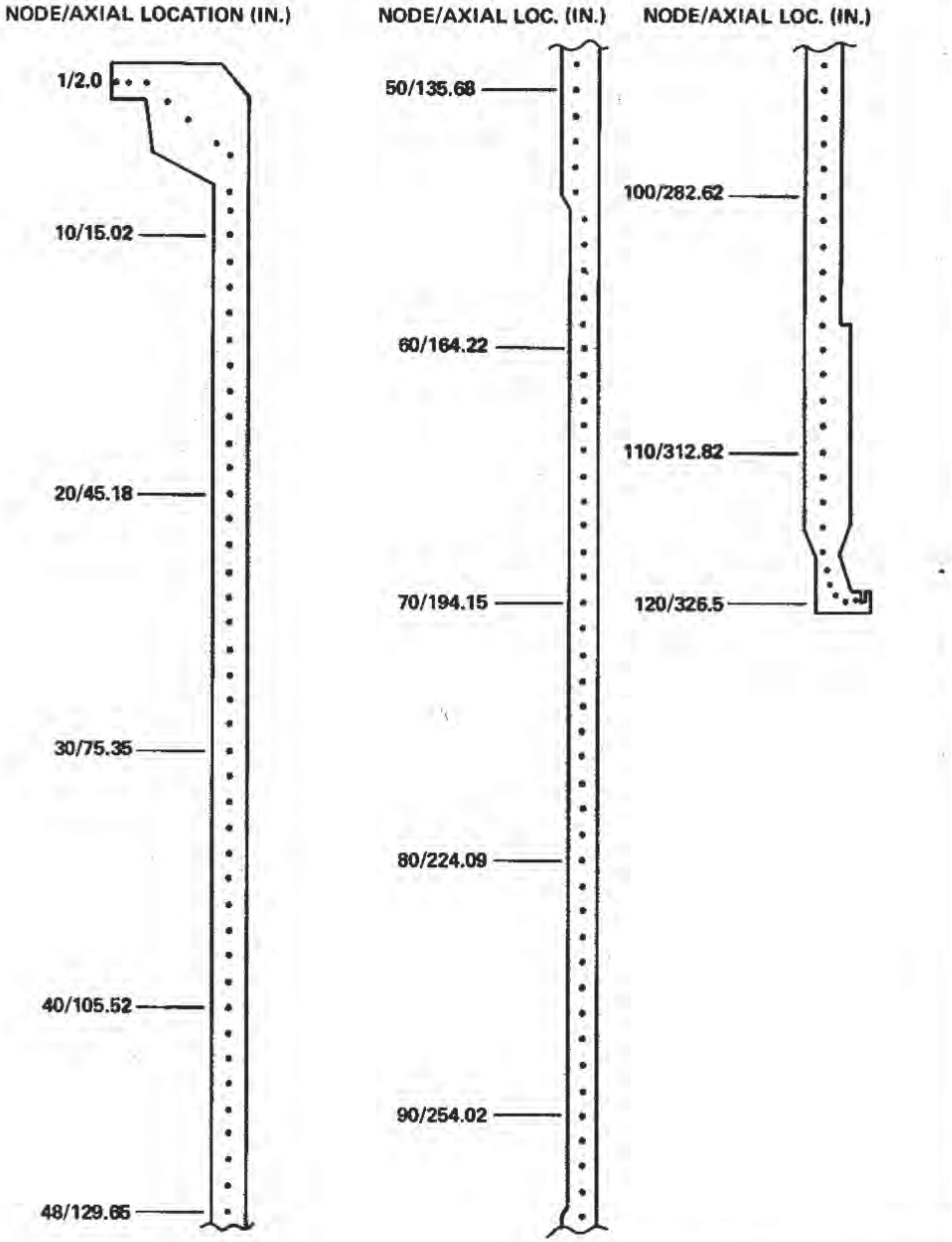
$$SSE = |SSEI| + |SSED|$$

TABLE 3.9-29

LIMITING CORE SUPPORT MARGINS FOR ASYMMETRIC LOADS

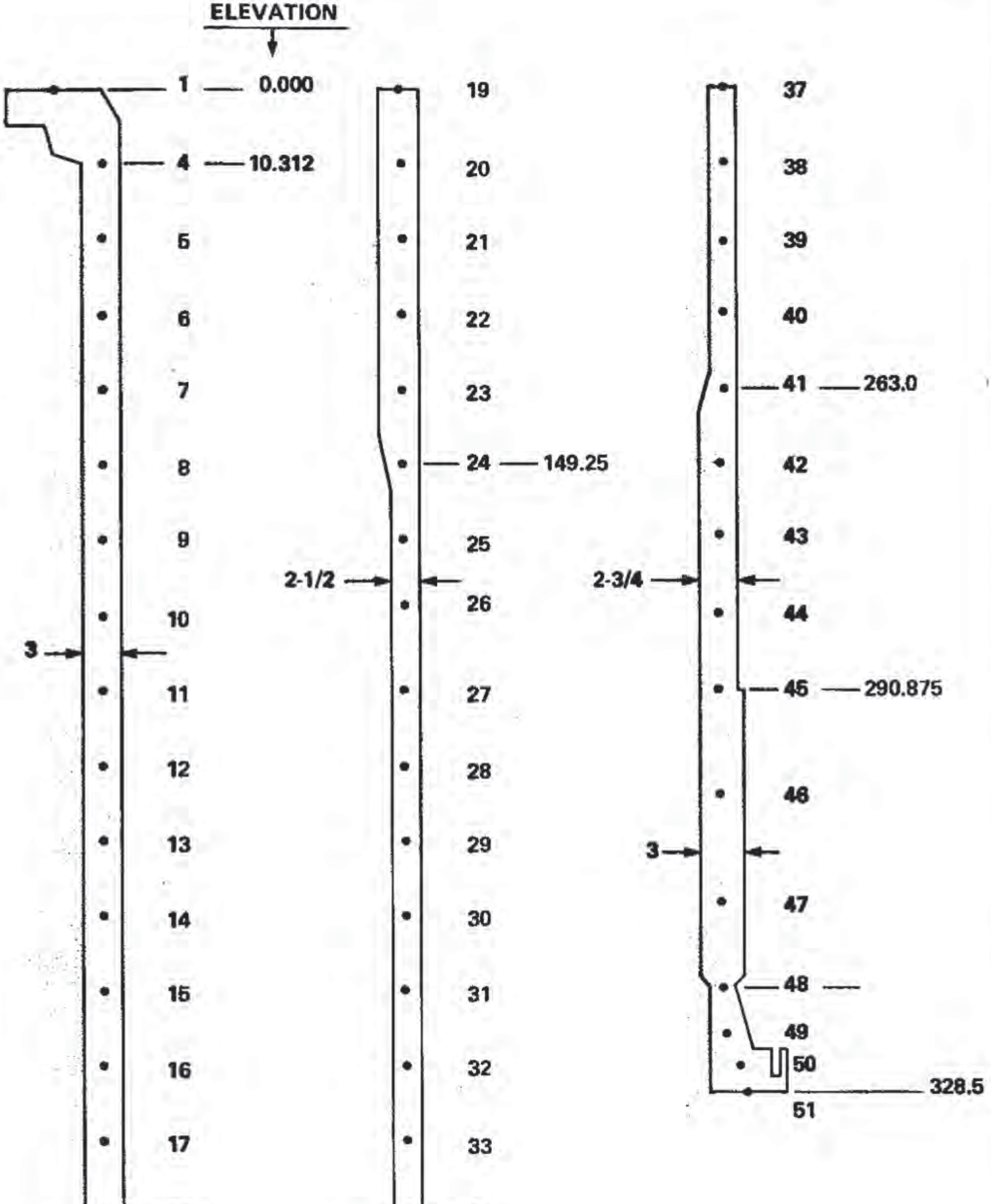
COMPONENT	CONDITION*	LIMITING STRESS, PSI	ALLOWABLE STRESS, PSI
UGS Grid Beams	Pm + Pb Faulted	27,696	57,960
CEA Shroud Bolts	Pm Faulted	81,032	86,600
UGS Flange	Pm Faulted	33,696	38,640
LSS	Pm + Pb Faulted	45,726	57,960
CSB Lower Flange Flexure	Pm Faulted	29,862	34,776
CSB Upper Flange	Pm + Pb Faulted	54,871	57,960
CSB Barrel Sections	Pm + Pb Faulted	48,884	52,164

* Pm = Primary Membrane Stress
  Pm + Pb = Primary Membrane Plus Primary Bending Stresses



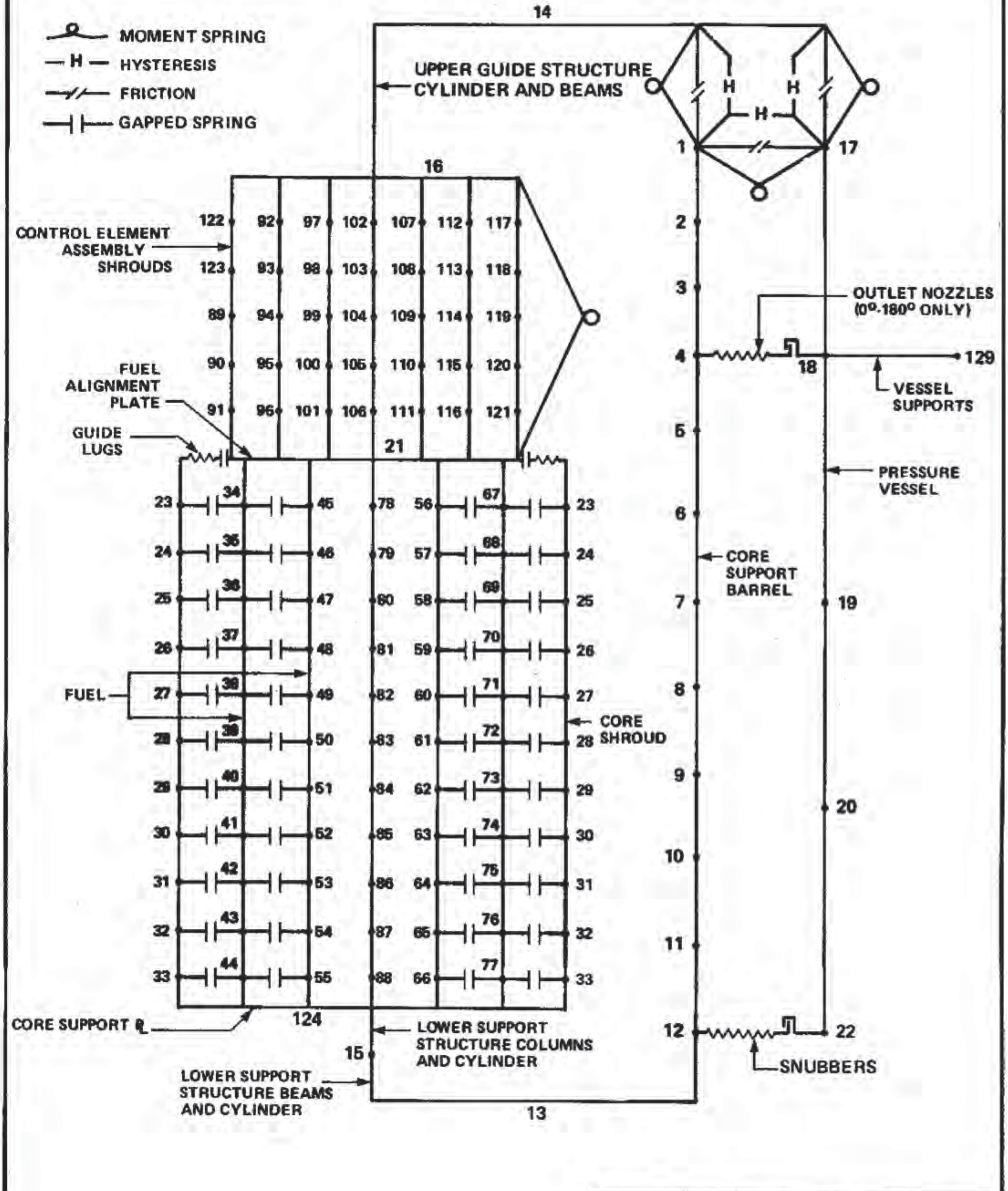
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CORE SUPPORT BARREL
SHELL RESPONSE MODEL (ASHSD)
FIGURE 3.9-1



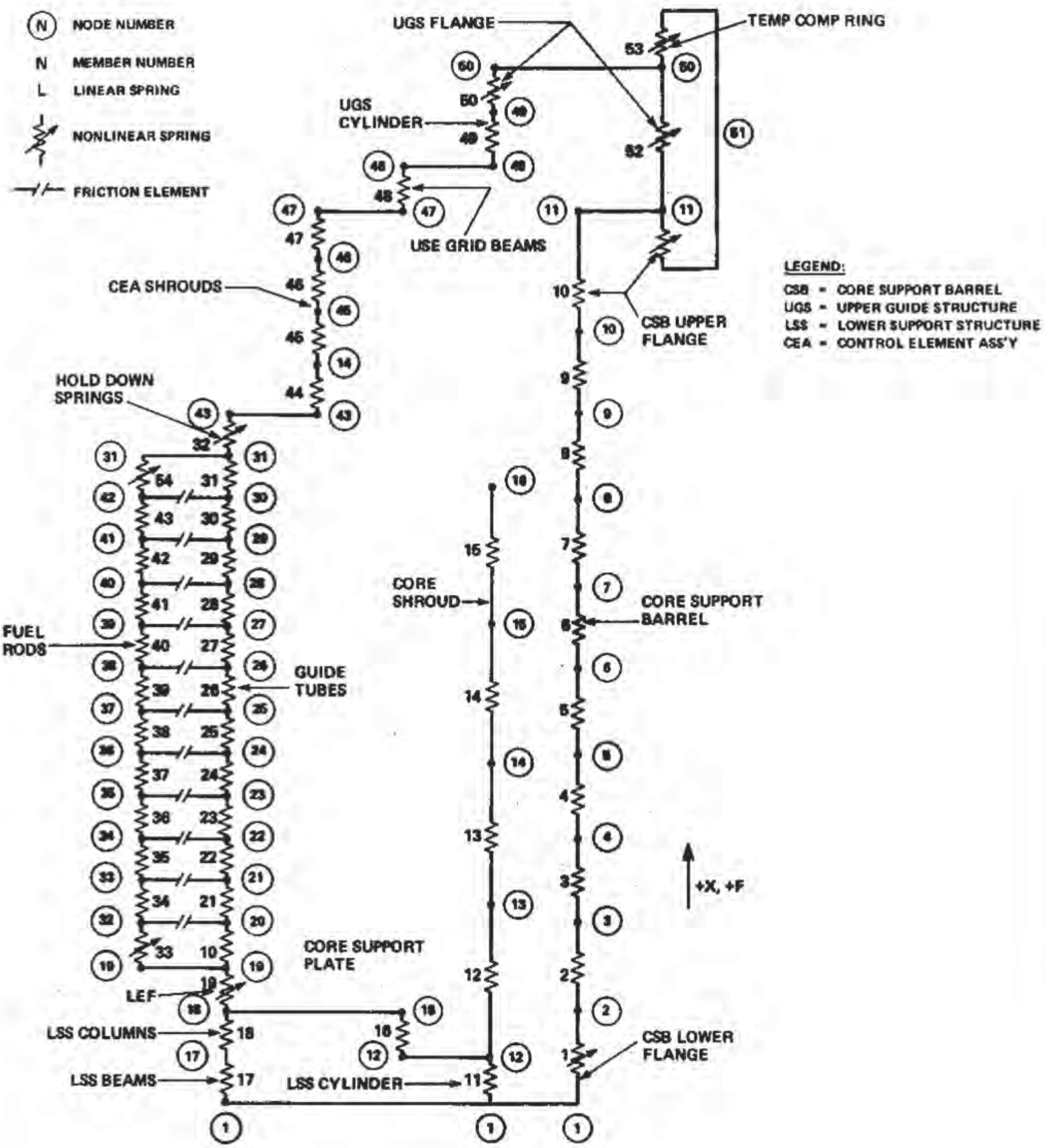
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CORE SUPPORT BARREL DYNAMIC STABILITY (BUCKLING) MODEL
FIGURE 3.9–2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

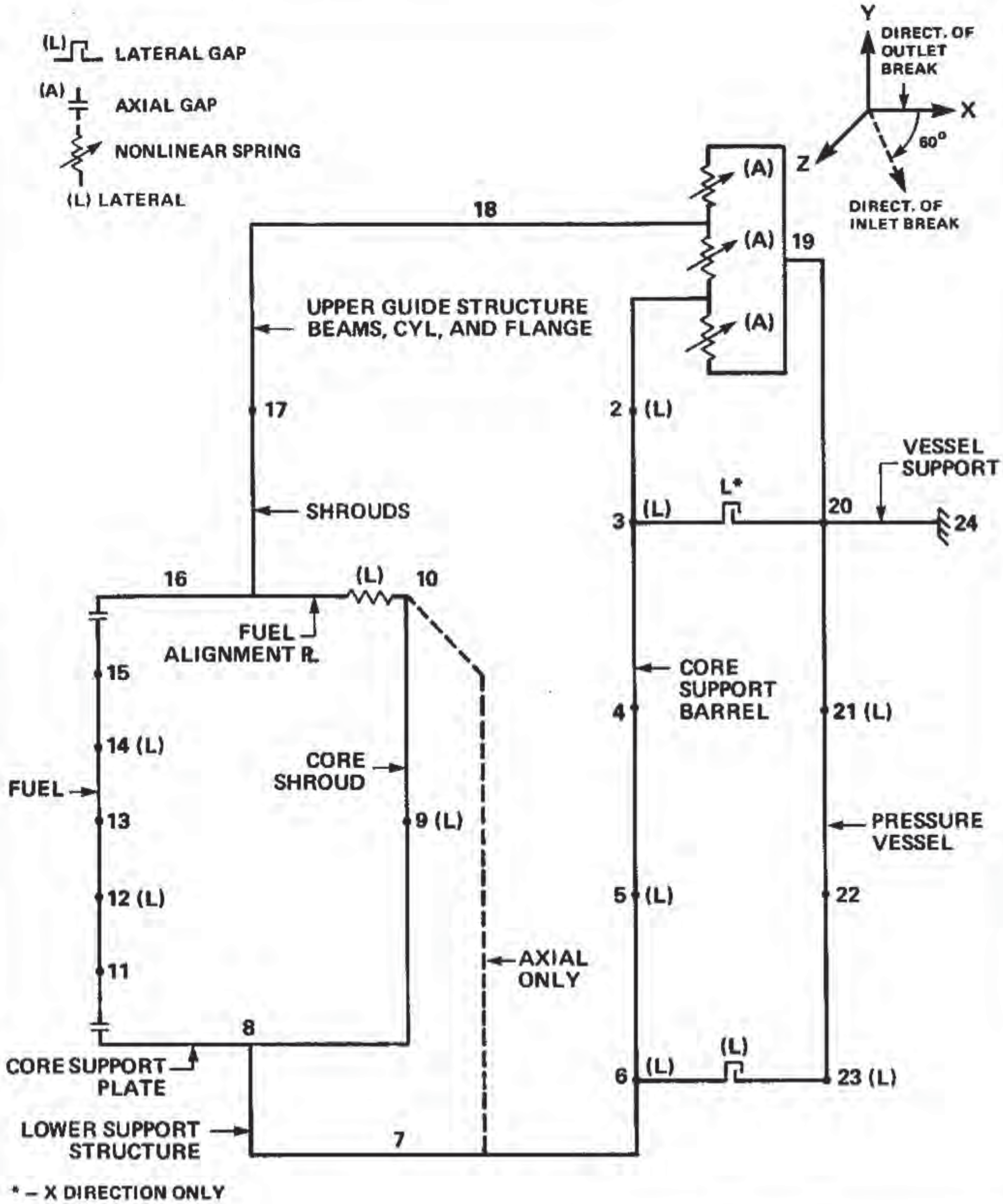
REACTOR INTERNALS LATERAL
LOCA MODEL (CESHOCK)
FIGURE 3.9–3



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

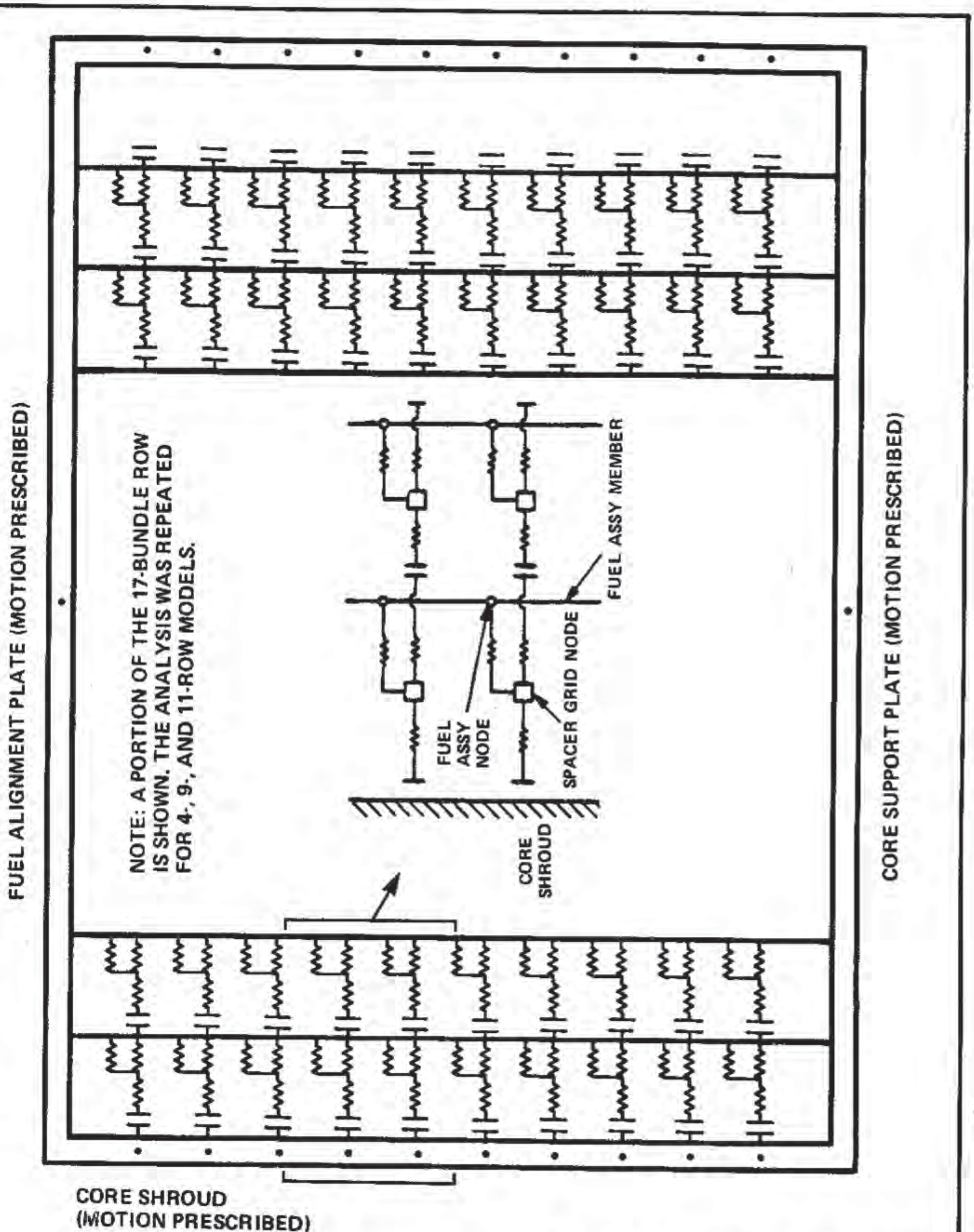
REACTOR INTERNALS VERTICAL LOCA MODEL

FIGURE 3.9–4



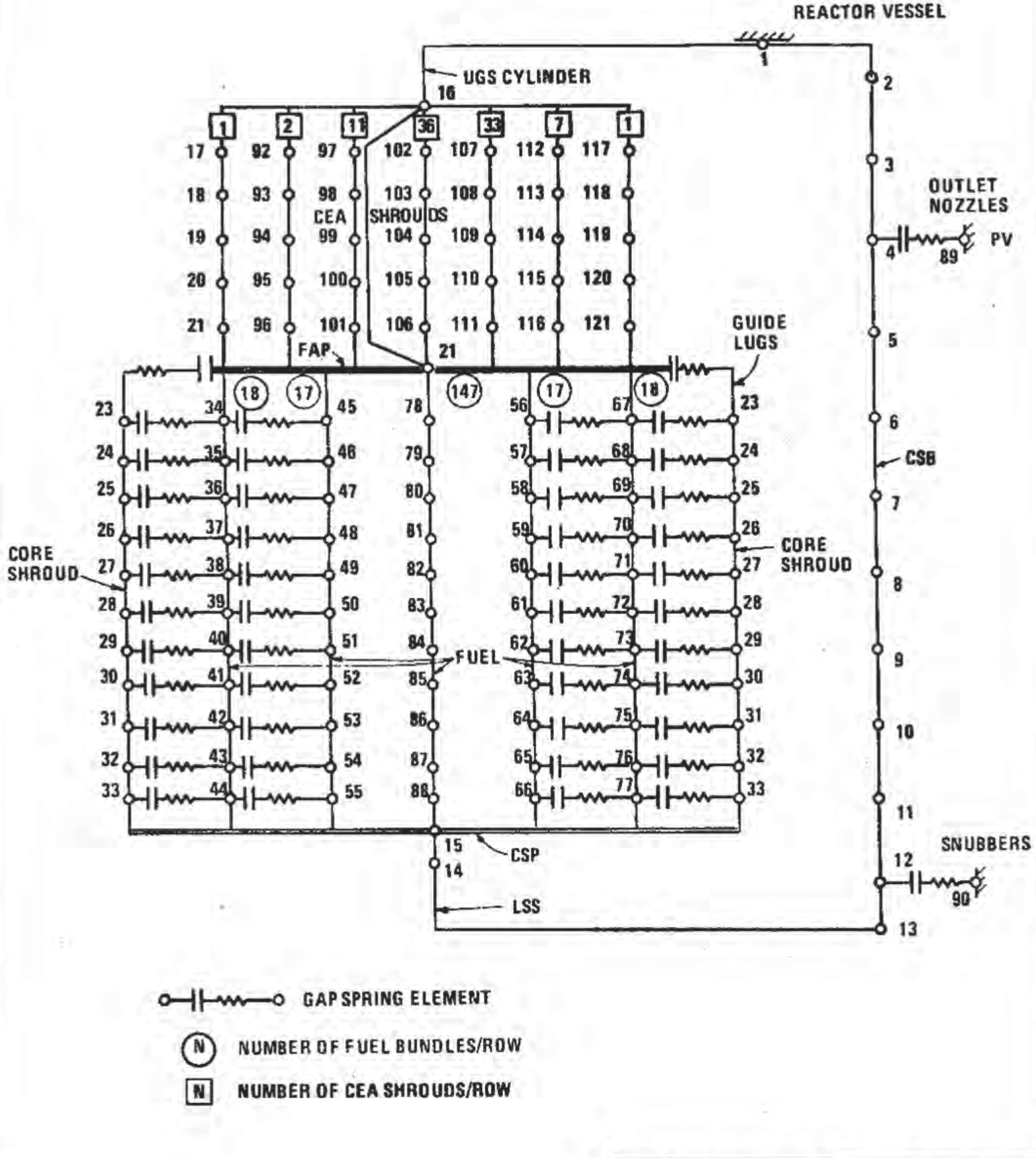
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR INTERNALS VERTICAL/HORIZONTAL REDUCED MODEL
FIGURE 3.9-5



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

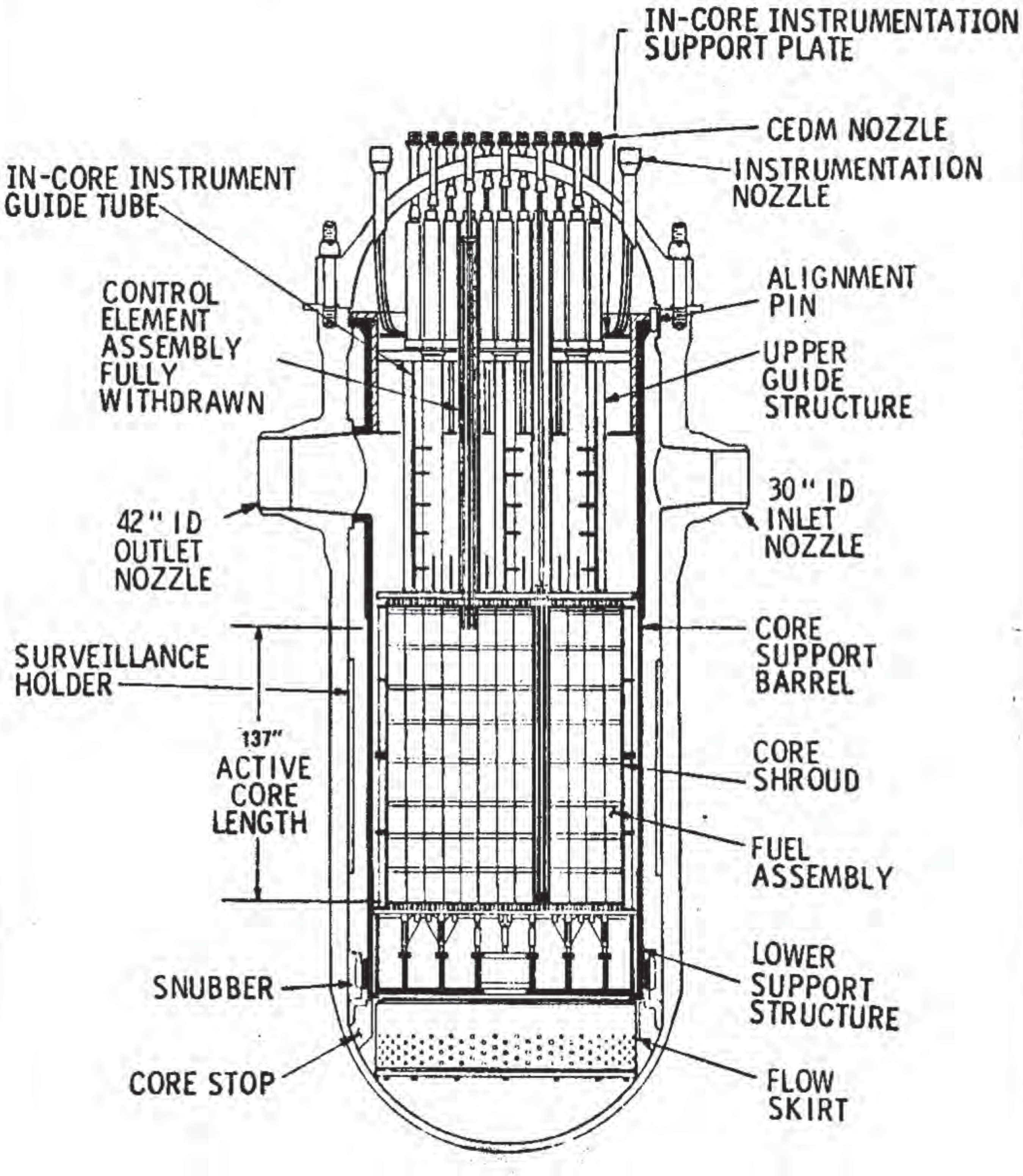
REACTOR CORE HORIZONTAL
NONLINEAR LOCA MODEL
FIGURE 3.9-6



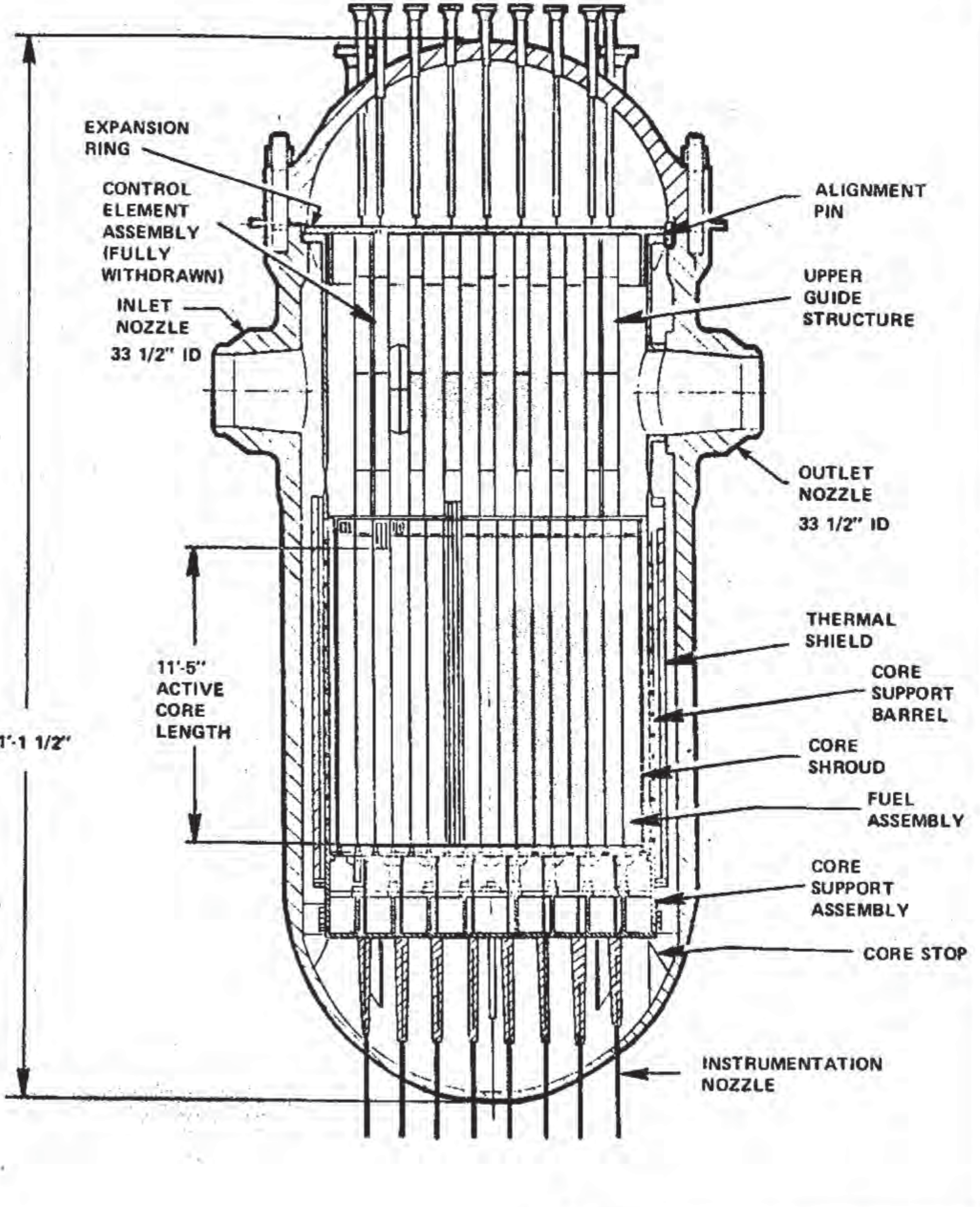
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

LOCA ANALYSIS – HOT LEG BREAK MODEL

FIGURE 3.9-7



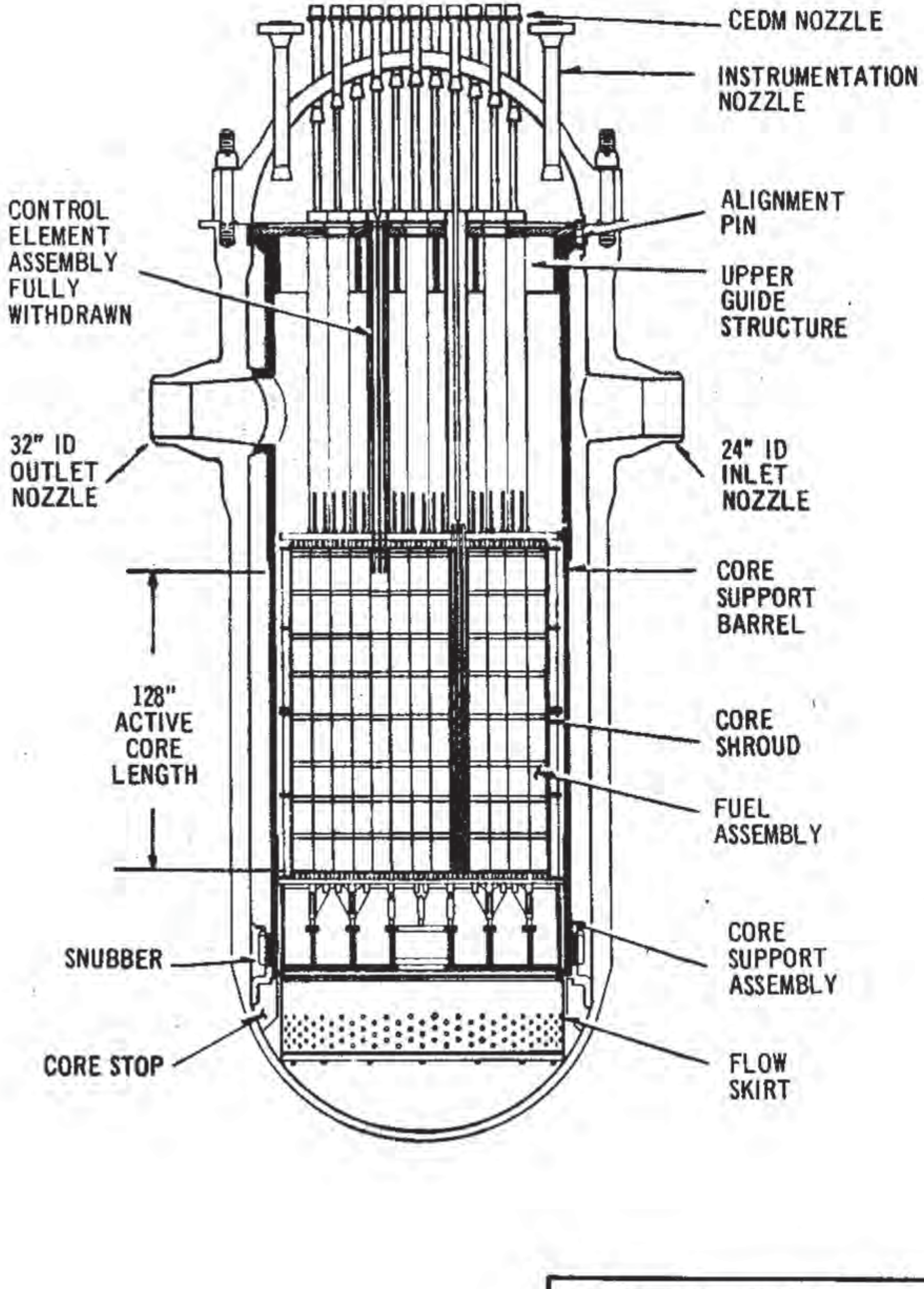
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 2
ST LUCIE – UNIT 2 REACTOR VERTICAL ARRANGEMENT FIGURE 3.9-8



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MAINE YANKEE
REACTOR VERTICAL ARRANGEMENT

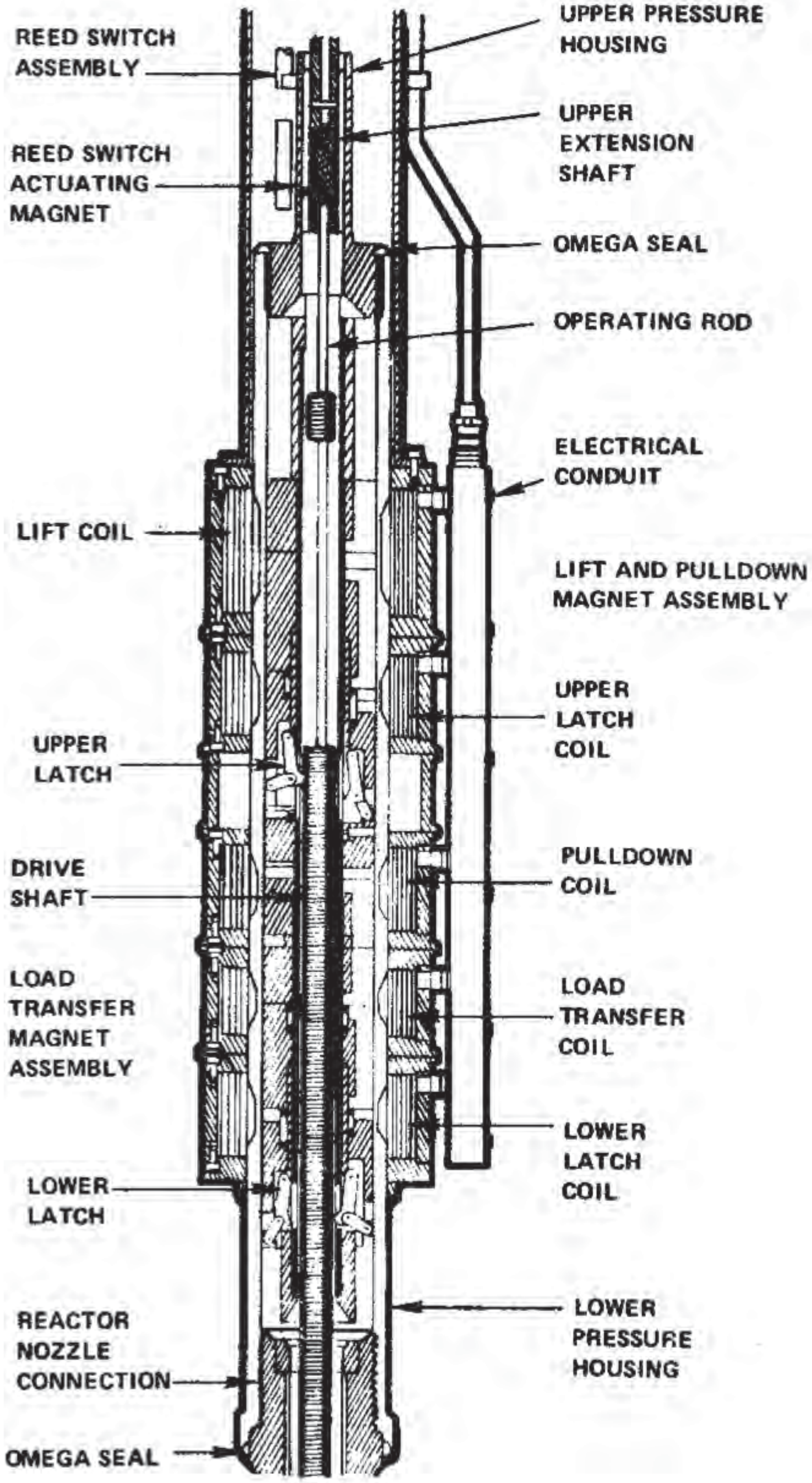
FIGURE 3.9-9



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FORT CALHOUN – UNIT 1
REACTOR VERTICAL ARRANGEMENT

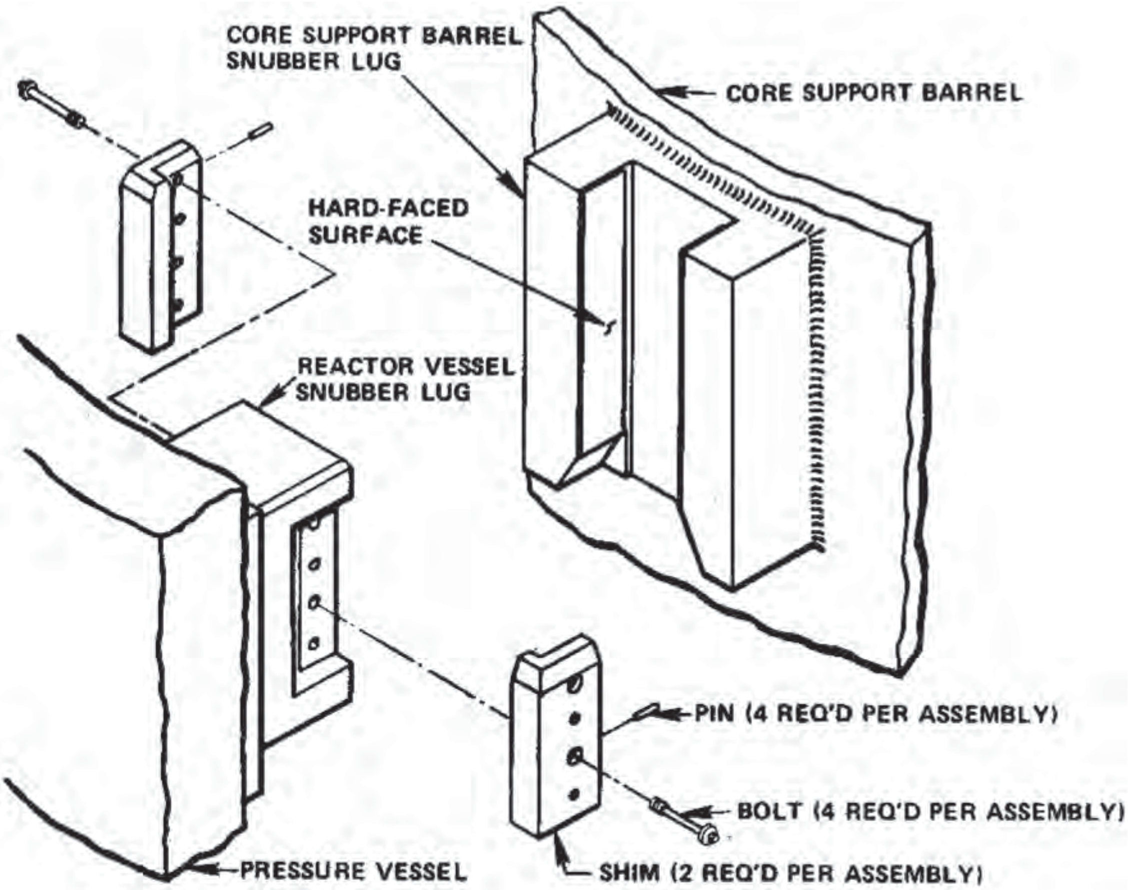
FIGURE 3.9-10



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

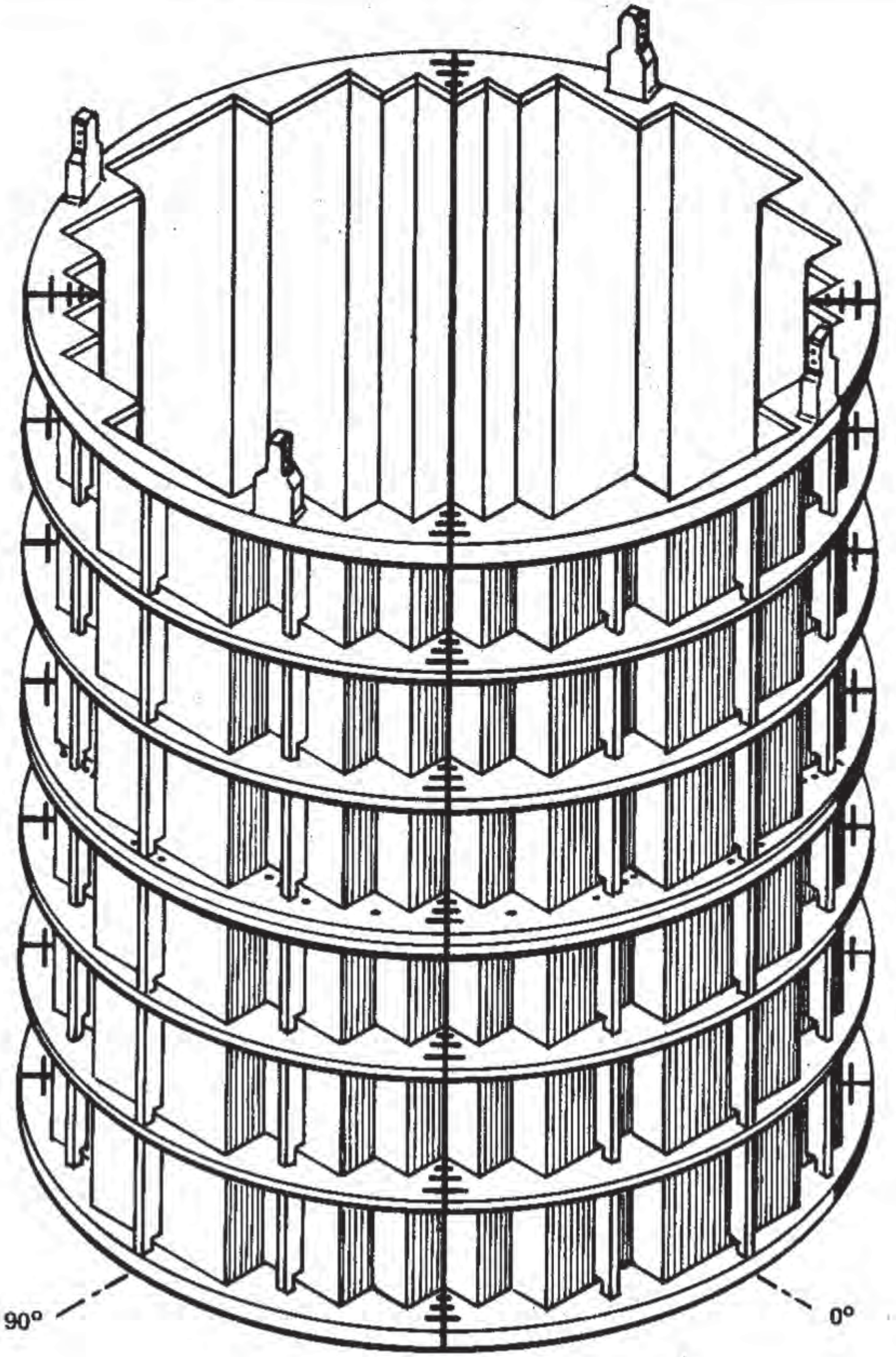
CONTROL ELEMENT DRIVE MECHANISM (MAGNETIC JACK)

FIGURE 3.9-11



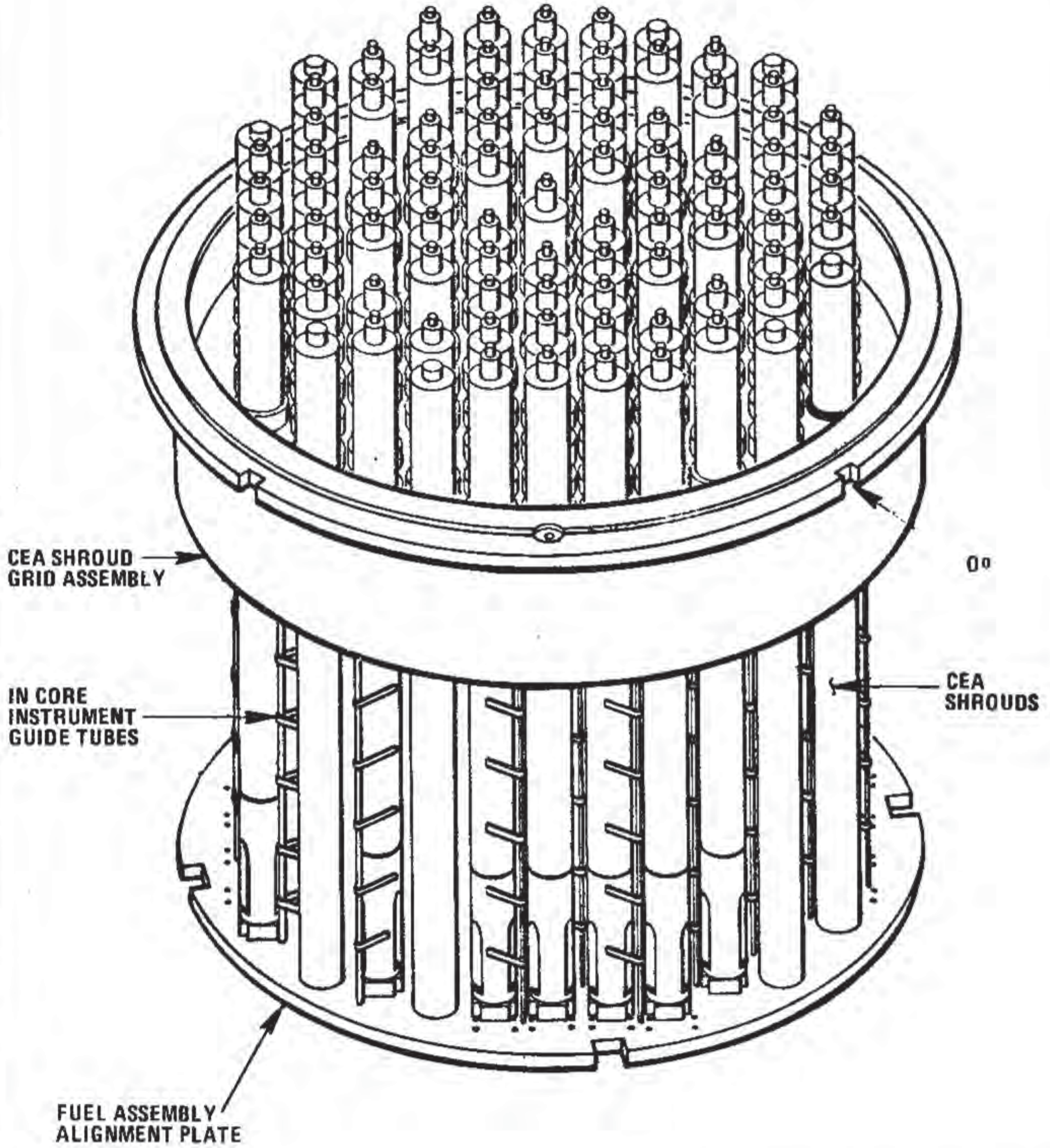
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PRESSURE VESSEL, CORE SUPPORT BARREL SNUBBER ASSEMBLY
FIGURE 3.9-12



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

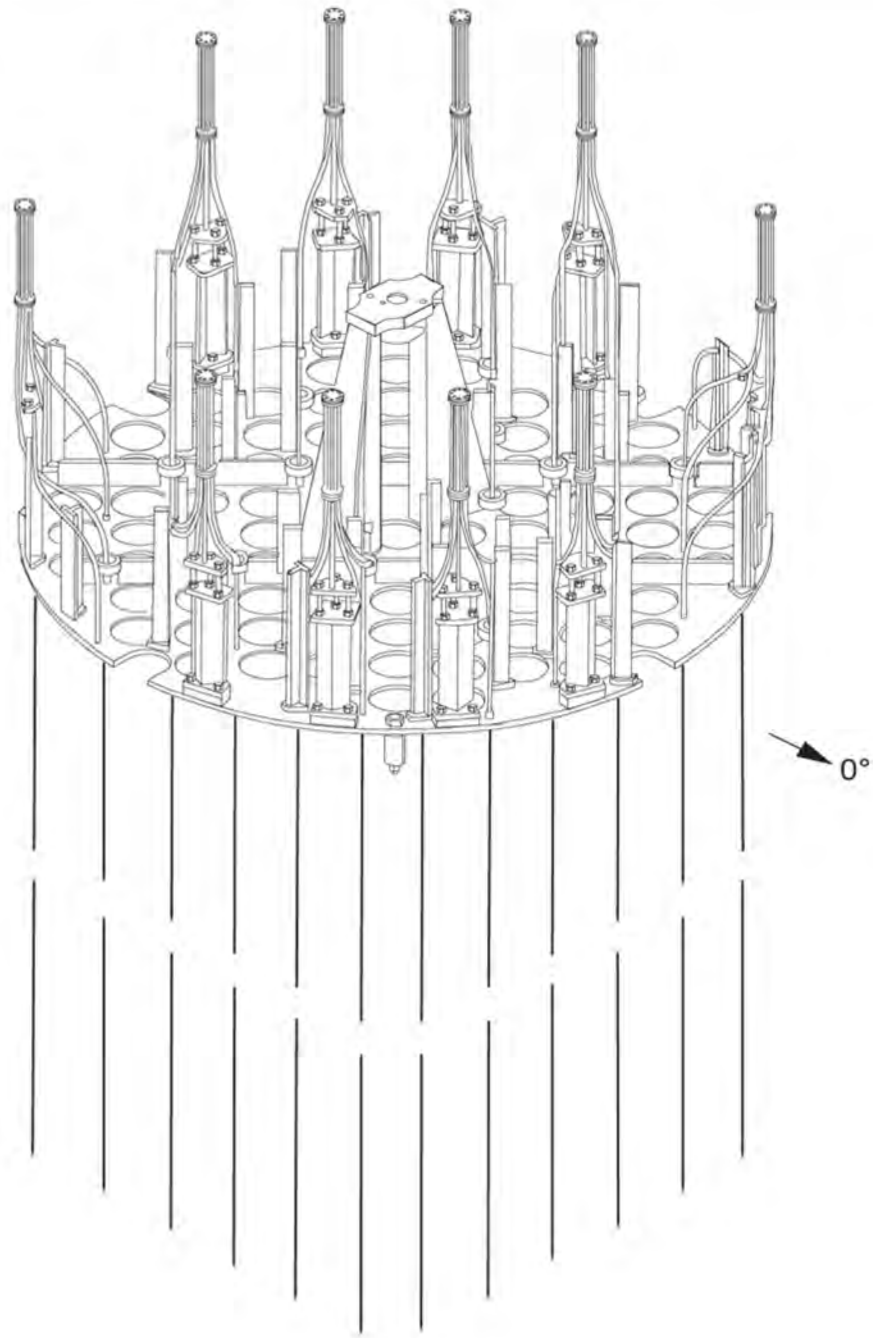
CORE SHROUD ASSEMBLY
FIGURE 3.9–13



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

UPPER GUIDE STRUCTURE
ASSEMBLY

FIGURE 3.9-14



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

INCORE INSTRUMENT SUPPORT ASSEMBLY

FIGURE 3.9-15

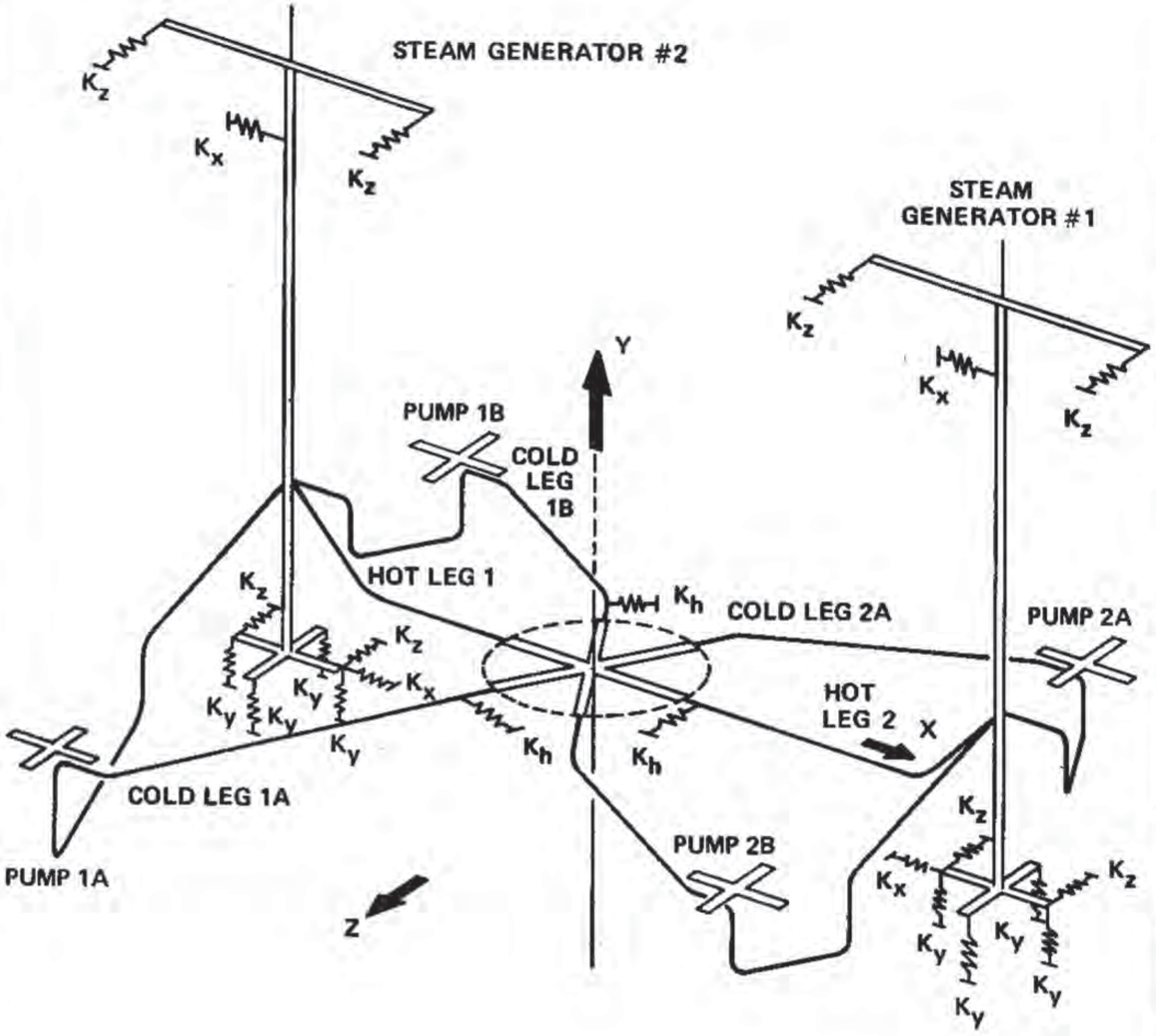
Refer to Dwg.
2998-19709 & 2998-19710

Amendment No. 11, (5/97)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

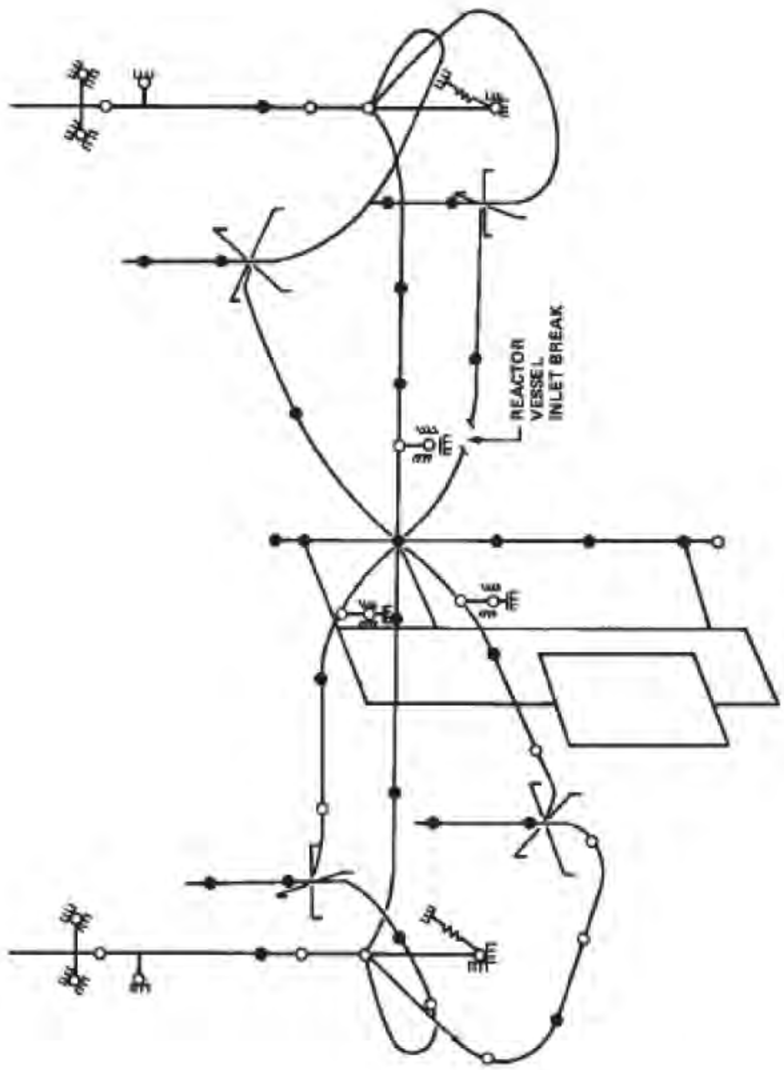
INCORE INSTRUMENT NOZZEL

FIGURE 3.9-16



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

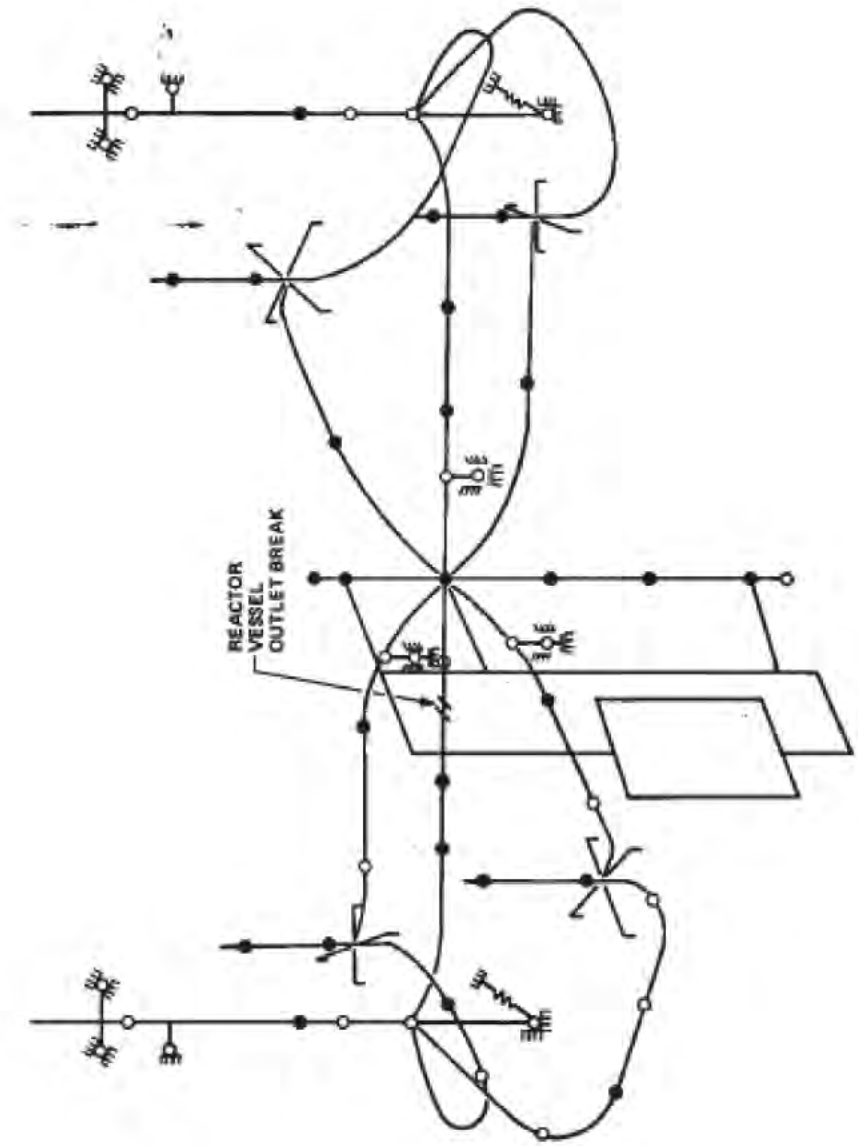
RCS - FLEXIBILITY ANALYSIS MODEL
FIGURE 3.9–18



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

RV ASYMMETRIC LOADS ANALYSIS
RV SUPPORT LOADS

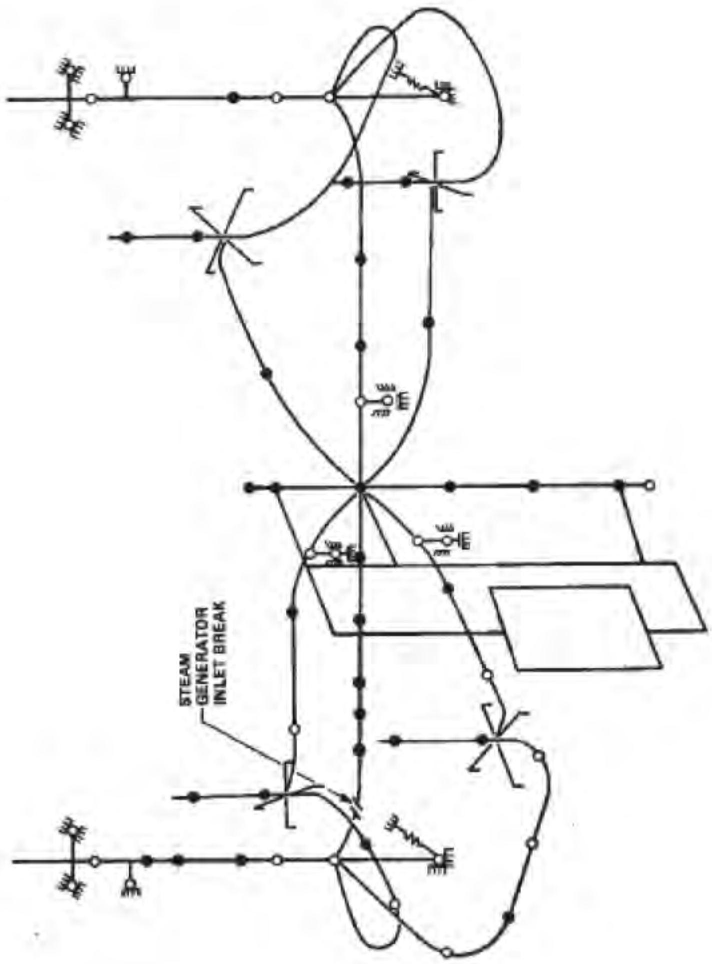
FIGURE 3.9-19
(HISTORICAL)



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

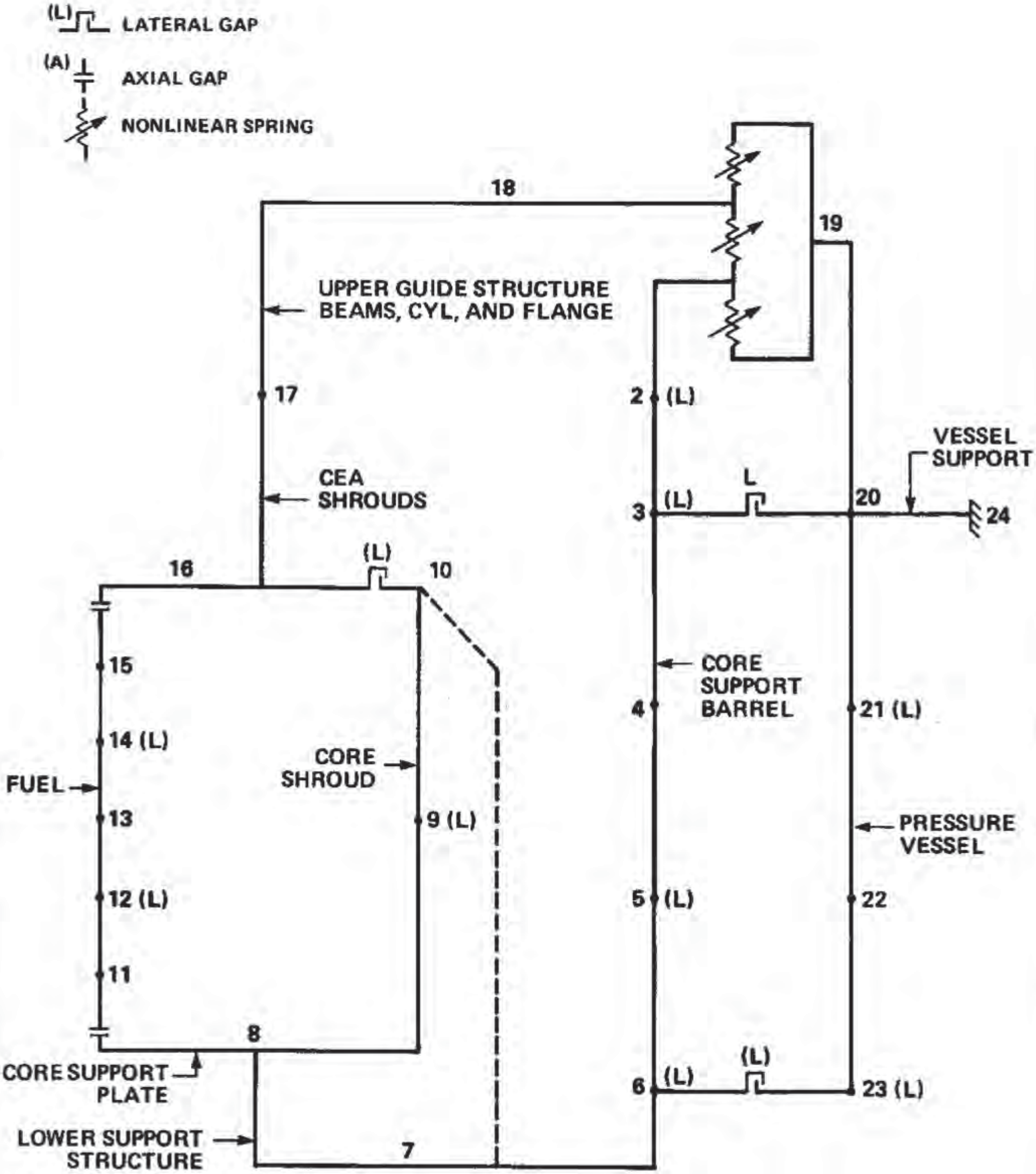
RV ASYMMETRIC LOADS ANALYSIS
RV SUPPORT LOADS

FIGURE 3.9-20
(HISTORICAL)



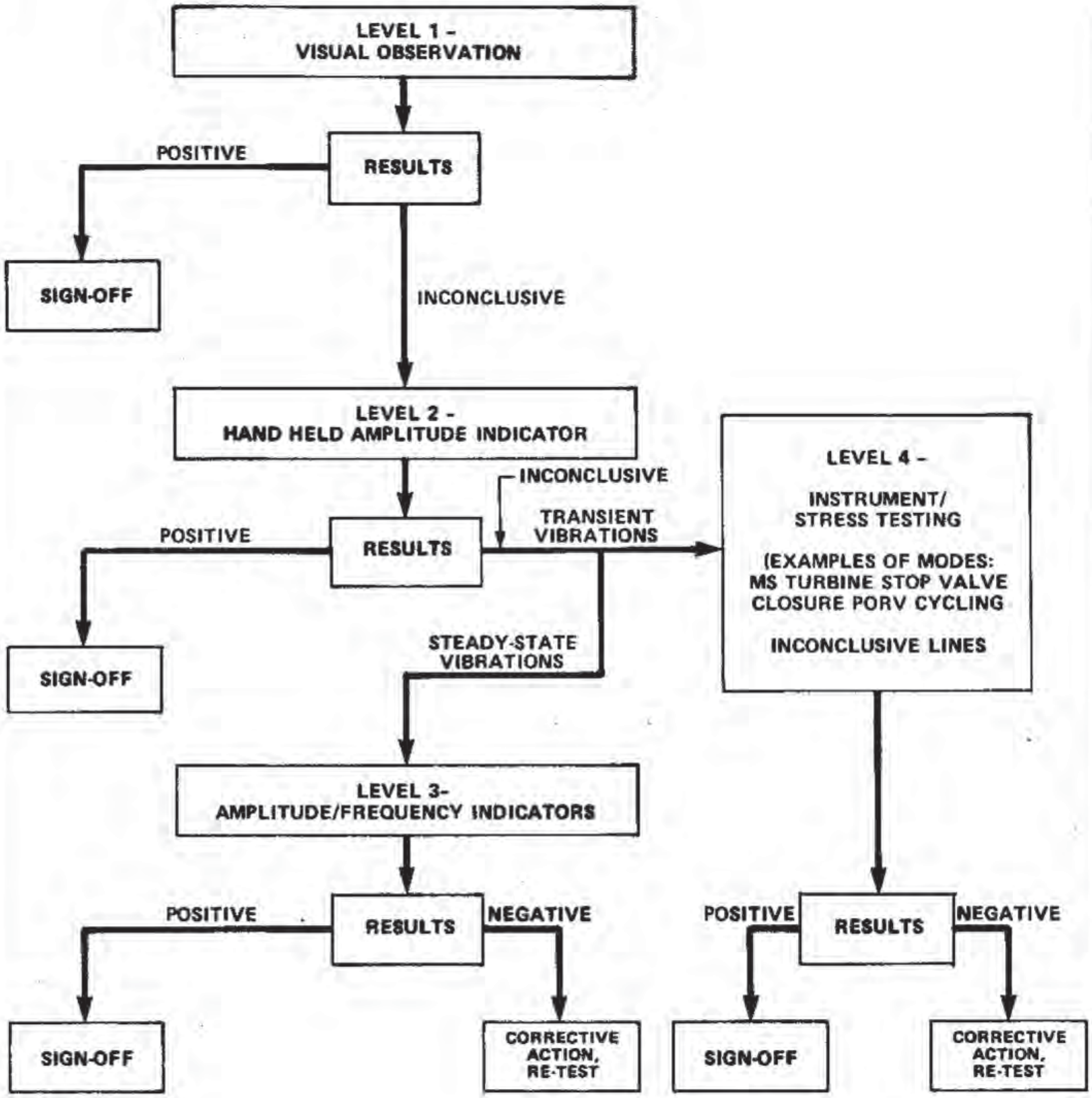
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

RV ASYMMETRIC LOADS ANALYSIS
RV SUPPORT LOADS
FIGURE 3.9-21
(HISTORICAL)



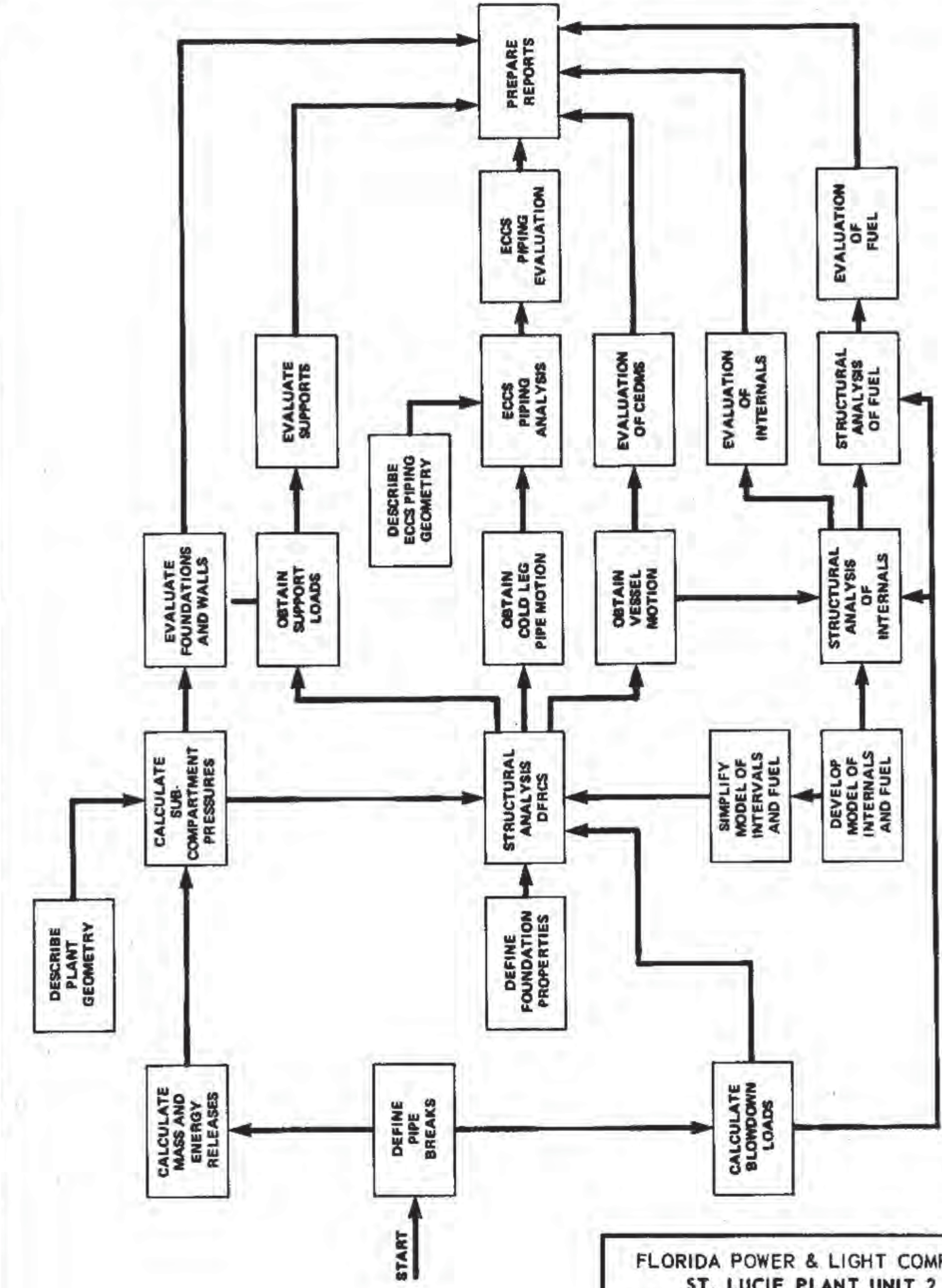
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MODEL OF REACTOR INTERNALS
FIGURE 3.9–22



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

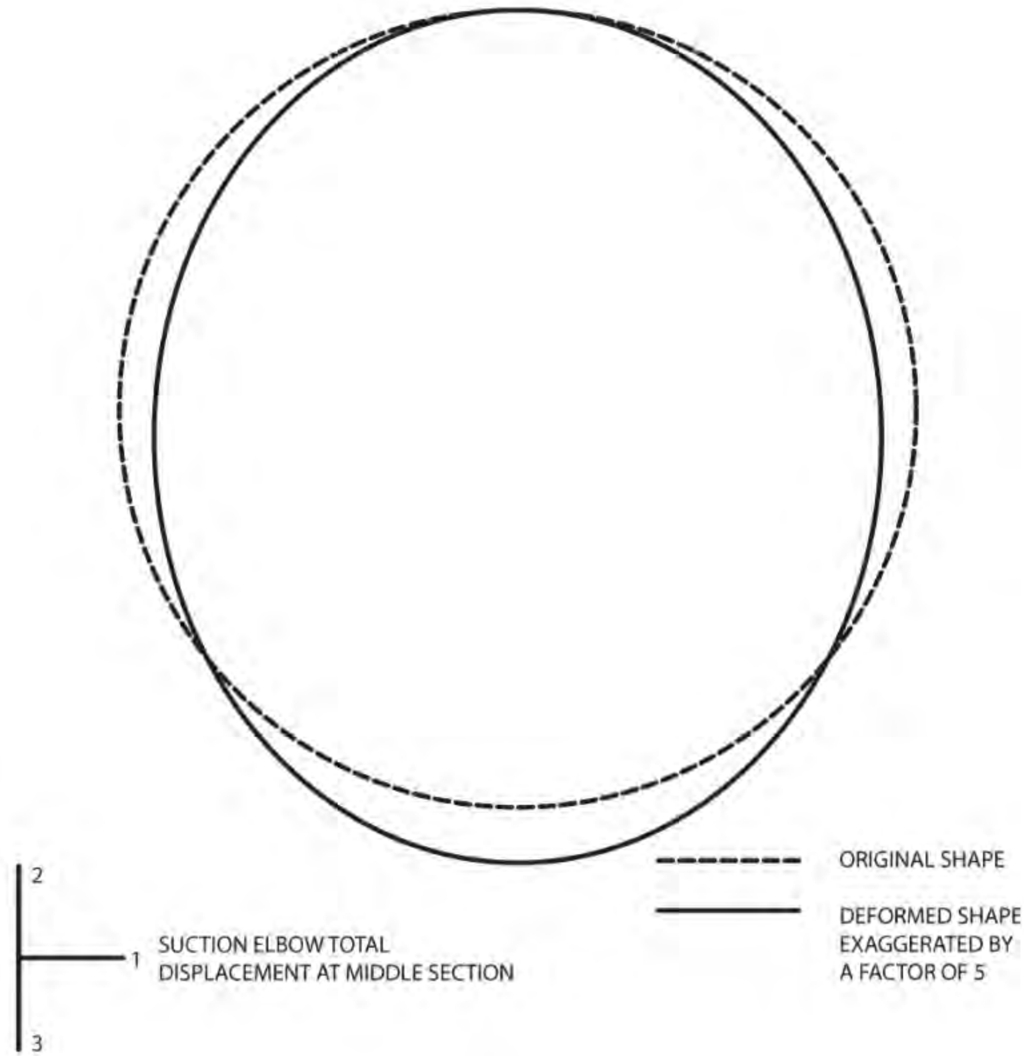
GENERAL FLOW CHART FOR
PIPING VIBRATION TESTING
FIGURE 3.9-23



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

GENERAL FLOW DIAGRAM FOR INTERNAL-EXTERNAL ASYMMETRIC LOADS ANALYSIS

FIGURE 3.9-24

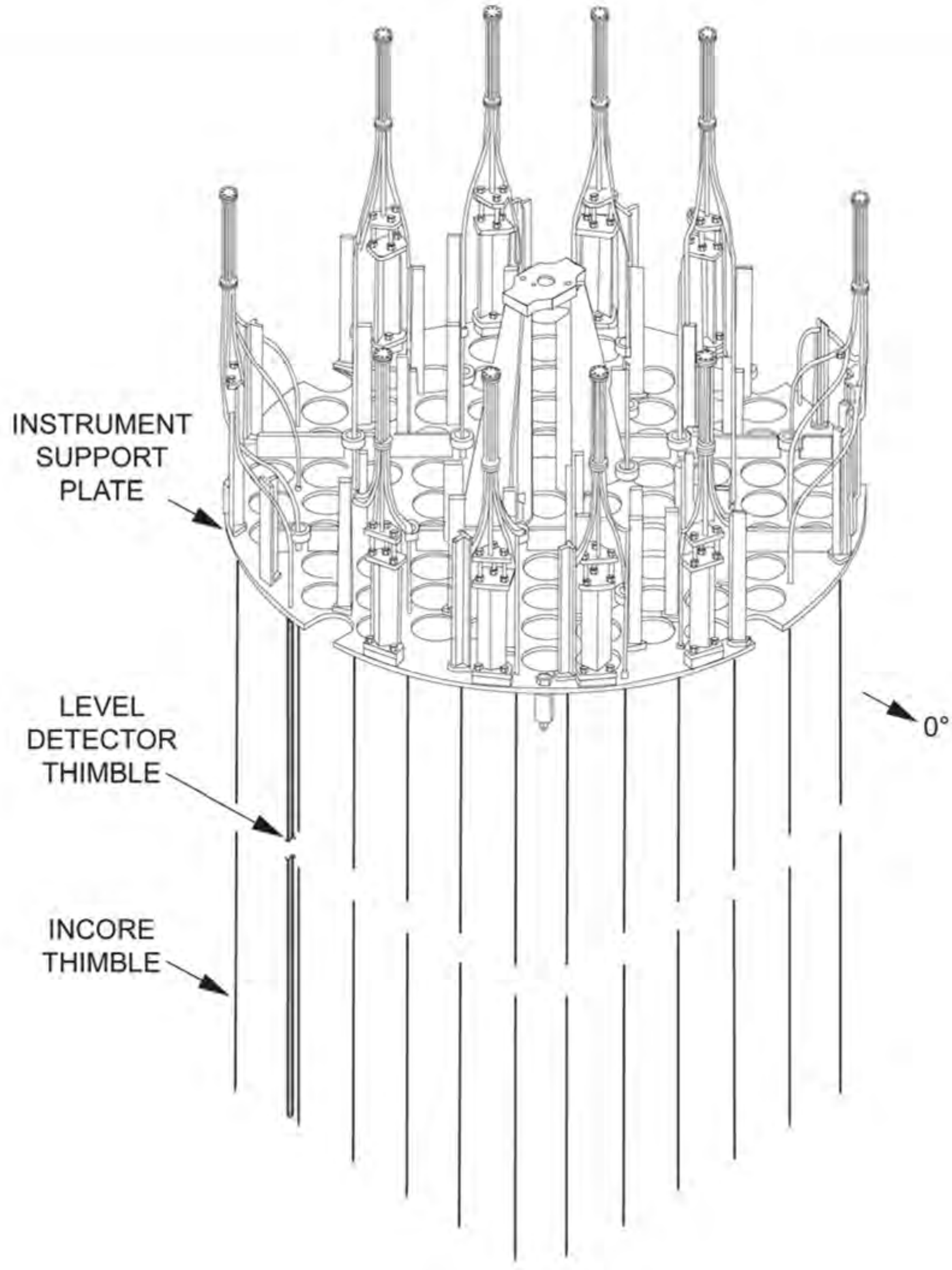


EFFECT OF MOMENT OF $60.2X10^6$ IN-LB

NOTE:
A CALCULATION OF THE DEFORMATION OF THE ST. LUCIE UNIT 2 RC PIPING WHEN SUBJECTED TO THE MAXIMUM MOMENT ALLOWED BY SECTION III, NB 3652 WAS PERFORMED.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

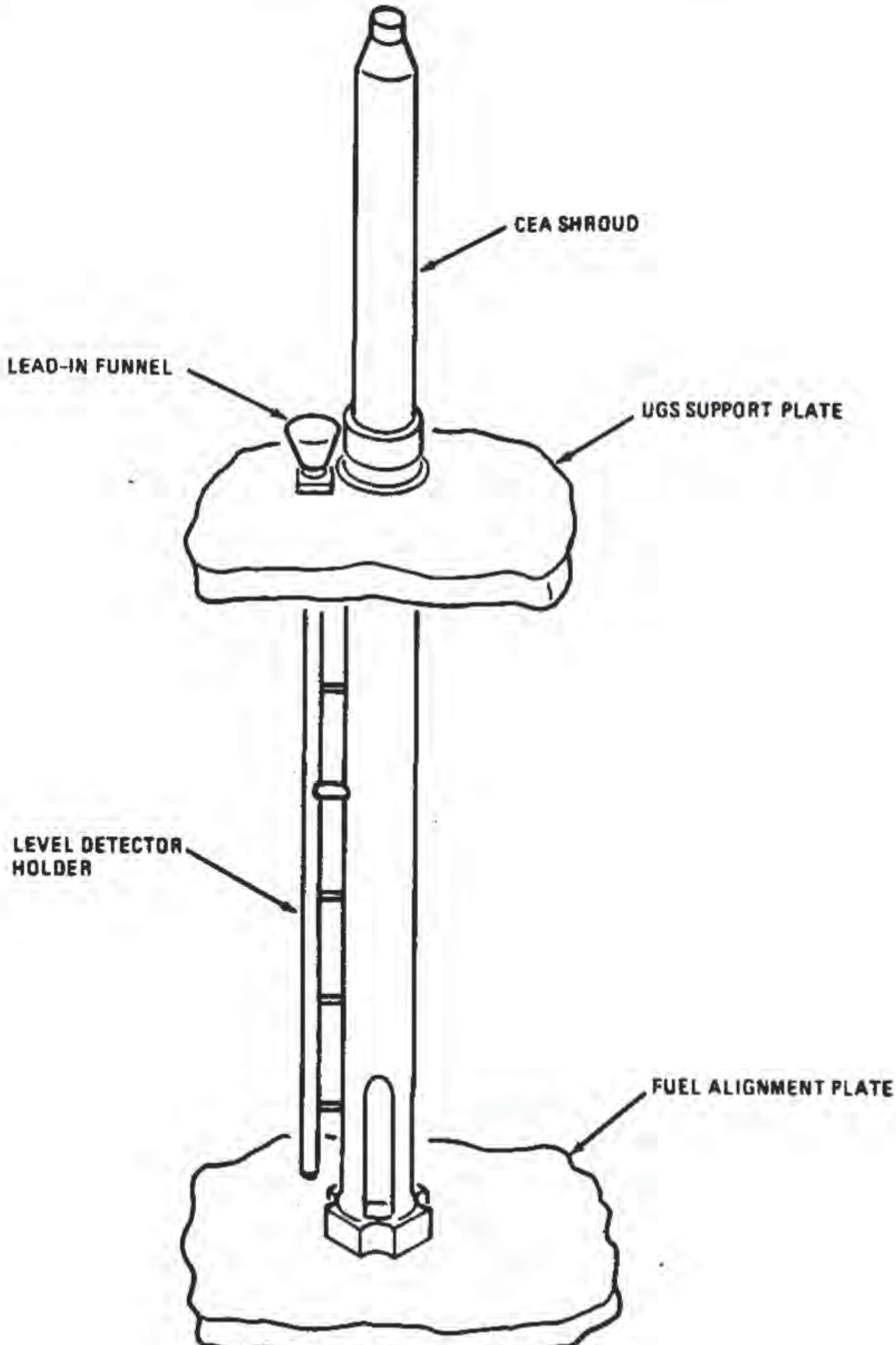
RC PIPING DEFORMATION
FIGURE 3.9-25



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR VESSEL LEVEL DETECTOR HOLDER (UPPER PORTION)

FIGURE 3.9-25a



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR VESSEL LEVEL DETECTOR
HOLDER (LOWER PORTION)
FIGURE 3.9-25b

APPENDIX 3.9A

OPERABILITY CONSIDERATIONS FOR SEISMIC CATEGORY I ACTIVE PUMPS AND VALVES

Seismic criteria are provided to the manufacturer wherein the manufacturer is required to demonstrate the operability of the active pumps and valves by adhering to the following criterion:

For seismic Category I equipment and supports the vendor must demonstrate the capability of the equipment to perform its required function during and after the time that it is subject to the forces resulting from the seismic conditions specified. This can be accomplished in various ways. Two methods commonly used are to predict the equipment performance by mathematical analysis (detailed stress and displacement analyses), or to test the equipment under simulated seismic conditions (in-situ testing).

3.9A.1 MATHEMATICAL ANALYSIS METHOD

This method is used for equipment which can be modeled to predict its responses. Detailed stress analyses are performed to verify that the equipment suffers no loss of function when subjected to the design loading combinations specified in Table 3.9-5. The equipment is considered acceptable to perform its intended safety function if the actual stresses are lower than the allowable stress limits given in Tables 3.9-6. The analyses also verify that adequate clearances have been provided to prevent binding of rotating parts within the active components when the equipment is subjected to the design loading combinations specified.

The nature of the calculation depends on the inherent design features of the equipment. The mathematical analysis method used takes into consideration the following:

a. Model the equipment and supports with sufficient degrees of freedom to ensure adequate representation.

b. Determine the natural frequencies and mode shapes of the equipment and supports as it is mounted in service.

c. The damping factors specified to be used in the seismic analysis are given in Table 3.9A-1.

3.9A1.1 Rigid Components

If the nature of the equipment is such that the calculations indicate that the natural period of the equipment (including its supports) is less than 0.03 seconds, then the subject equipment is considered rigid and it is analyzed statically. In this static analysis, the seismic forces of each component of the equipment are obtained by concentrating its mass at its center of gravity.

Seismic Category I active pumps are specified to withstand the following loading conditions:

a. The operating basis earthquake (OBE) and safe shutdown earthquake (SSE) which are considered in the design of the pumps are obtained by utilizing the seismic acceleration(g) values from the applicable floor response spectra curves.

The minimum seismic acceleration(g) values utilized in the static analysis correspond to the equipment's lowest natural frequency.

The seismic acceleration curves are established based on an estimated amplification of the acceleration of the floor on which the equipment is supported. Floor response spectra curves showing floor spectral accelerations as a function of natural frequencies of vibration are available for all major floor levels where equipment is located. (See Section 3.7).

b. The vertical force and horizontal forces in both horizontal orthogonal directions are assumed to be acting simultaneously for both the OBE and the SSE conditions.

c. The natural frequency of vibration of the equipment and supports is above the frequency limit required to define these components as rigid.

The seismic Category I active valves are specified to withstand the following loading conditions:

The operating basis earthquake (OBE) load consisting of the most severe combination of two horizontal seismic load coefficient of 1.5g, which can act in either of the two major horizontal directions, acting simultaneously with a vertical seismic load coefficient of 1.0g, which can act upward or downward.

The safe shutdown earthquake (SSE) load consisting of the most severe combination of a horizontal seismic load assuming a coefficient of 3g, and a vertical seismic load assuming a coefficient of 2g acting as above. The specified requirement for NSSS valves is that they be capable of withstanding a resultant seismic load of 3g in any direction applied at the pipe connections on the valve.

Seismic loads are assumed to act at the center of gravity of the equipment.

The stresses induced from the earthquake loads are combined in accordance with the applicable equipment codes. If the codes are not specific, the seismic loads are added directly to the stresses from other applicable loading. The allowable stresses for the upset and emergency conditions are not allowed to be increased due to the addition of the OBE seismic load; but the allowable stresses may be increased due to the addition of the SSE seismic load to a limiting value that does not cause loss of function. The allowable stress limits are provided in Table 3.9-6.

Where valves are provided with operators, the valve yoke structure with operator is analyzed to exhibit a fundamental frequency of greater than or equal to 33 Hz (20 Hz for NSSS supplied pneumatically operated valves) in its most limiting configuration. The valve system capability to sustain the SSE seismic event is established by satisfying the stress limit with an analysis based on static forces resulting from equivalent earthquake accelerations of equal to or greater than 3.0g horizontal and 2.0g vertical acting at the center of gravity of the operator. If the fundamental frequency of the yoke structure is less than 33 Hz, a detailed analysis and/or test is performed to demonstrate that the valve system will not experience a loss of function during and following a SSE seismic event. NSSS supplied pneumatically operated valves are required to have a natural frequency greater than 20 Hz, and be analyzed to withstand a resultant seismic load of 3g applied in any direction at the pipe connection.

3.9A.1.2 Non Rigid Components

If the natural frequency of the equipment and supports is less than 33 Hz (20 Hz for NSSS supplied pneumatically operated valves), a dynamic analysis is performed using the response spectral modal analysis technique using floor response spectra or a time history analysis. Other methods of analysis may be used if properly justified. The stress analysis is performed using the inertia forces or the equivalent static loads obtained from the dynamic analysis for each mode.

The square root of the sum of the squares (SRSS) method normally is used to combine the modal responses when the response spectrum modal analysis method is employed. In those cases however, where modal frequencies are closely spaced, the responses of the closely spaced modes are combined by the sum of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the square root of the sum of the squares method.

If the time-history analysis is used, the maximum responses are determined by obtaining the greatest sum of the response of all significant modes at a particular time.

In each of the preceeding analyses the response and seismic stress in the three perpendicular directions (one vertical and two horizontal) are considered acting simultaneously. The maximum value of a particular response (seismic stress) of a component subjected to a single independent spatial component of a three component earthquake is obtained by taking the square root of the sum of the squares of the corresponding maximum values of the response of the element attributed to the individual significant modes.

Combining by SRSS applies whether the response spectrum method or the time - history modal analysis is employed. The analysis includes evaluation of the effects of the calculated stresses on mechanical strength, alignment, electrical performance, and noninterruption of function as related to the functional requirements of the equipment during an SSE.

<u>TESTING METHOD</u>

Test data or operating experience history of similar equipment designed and operating under similar conditions is acceptable. If it is not feasible to test the entire unit, testing of components which are crucial to the operation of the equipment is acceptable.

Seismic tests are performed by subjecting the equipment to vibratory motion which conservatively simulates that to be seen at the equipment mounting during an SSE. The equipment to be tested is mounted on the vibration generator in a manner that simulates the intended service mounting. The vibration motion is applied to each of the three major perpendicular axes simultaneously unless symmetry justifies otherwise. The equipment being tested must demonstrate its ability to perform its intended function and sufficient monitoring equipment is used to evaluate performance before, during and following the test. Detailed testing procedures are submitted for prior approval.

Actual testing generally involves the following procedures:

a. Performing a low amplitude frequency search to determine potential resonance regions (1-33 Hz).

b. Testing the equipment at these resonance frequencies with amplitude and test duration equivalent to that produced by the floor time history motions. The Sine Beat Test is the preferred method of testing, however, other methods of testing are permitted if properly justified.

The equipment is subjected to the seismic response indicated above and the test data are submitted which substantiate that the equipment and accessories do not suffer loss of function due to these seismic considerations.

SEISMIC ANALYSIS AND SUMMARY OF RESULTS FOR ACTIVE PUMPS

<u>CONTAINMENT SPRAY PUMPS</u>
(REPORT NO. EAS-TR-7714 RP dated November 7, 1977)

This Containment Spray Pumps are Ingersoll-Rand Pumps type 8 x 23 WDF. These pumps are vertical, single stage, diffuser type pumps with an operating speed of 1780 RPM.

The Containment Spray Pumps have been qualified for their intended service by analysis. The analysis includes the development of a mathematical model to determine the natural frequency and mode shapes by using, the ANSYS computer program. The results of the computer analysis indicates the lowest natural frequency to be greater than 33 Hz, hence the equipment is considered rigid and qualified by static analysis. The first five natural frequencies calculated were:

44.5, 44.6, 120.0, 120.9 and 218.1 Hz.

The analysis considered the following seismic loads applied to the pump:

Horizontal	- 0.25g (In each of two orthogonal directions)
Vertical	- 0.25g

A summary of calculated stresses, deflections and loads is provided in Table 3.9A-2. The actual values are compared with the ASME code allowables.

<u>CONTAINMENT SPRAY PUMP MOTOR</u>
(REPORT NO. EL-8-5117-90301-01)

The Containment Spray Pump Motors are Allis Chalmers Type FODVS 500 H.P. Motors.

The CSP motors have been qualified for their intended service by mathematical analysis. The results of the analysis indicates the lowest natural frequency (52.5 Hz) to be greater than 33 Hz, hence the equipment was considered rigid and qualified by static analysis.

The following seismic acceleration values were used in the analysis:

Horizontal	- 0.22g (In each of two directions)
Vertical	- 0.22g

A summary of the calculated stresses, deflections and loads is provided on Table 3.9A-3. The actual values are compared with the ASME codes allowables.

<u>DIESEL OIL TRANSFER PUMPS AND MOTORS</u>
(REPORT NO. ME -414 dated May 6, 1977)

The Diesel Oil Transfer Pumps are Gould Pumps, Inc Model 3196

MT Size 1 x 2 - 10.

The Diesel Oil Transfer Pumps and Motor Combination have been qualified for their intended safety function by analysis. A dynamic model was developed and the natural frequencies were calculated utilizing the ICES - STRUDL computer program. The lowest natural frequency was calculated to be 40 Hz and therefore the pumps and motor assembly are considered rigid and qualified by static analysis.

The analysis considered the following loadings applied to the center of mass of each individual pump component:

Horizontal	- 0.5g (In each of two directions)
Vertical	- 0.5g

A summary of calculated stresses, deflections and loads is shown in Table 3.9A-4. The actual values are compared with the ASME code allowables to verify structural integrity and operability.

LOW PRESSURE SAFETY INJECTION PUMPS

The LPSI Pumps are Ingersoll-Rand pumps type 8 x 20 WDF.

The original close coupled LPSI Pumps have been qualified for their intended safety function by analysis. The results of the calculation indicates that the lowest natural frequency to be greater than 33 Hz, hence the equipment is considered rigid and qualified by static analysis. The first five natural frequencies calculated were: EC275737

36.7, 41.8, 70.2, 70.3 and 119.2 Hz.

The analysis considered the following DBE seismic loads applied to the pump:

Horizontal	- 1.1g (In each of two orthogonal directions)
Vertical	- 0.6g

For the modified coupled design, the first four natural frequencies calculated were: EC275737

24.7, 24.8, 71.4, 115.3 Hz (Modified coupled design with original motor)

25.6, 26.0, 79.6, 124.6 Hz (Modified coupled design with spare motor)

The minimum natural frequency of the pump-motor assembly is 24.7 Hz, which corresponds to the zero period acceleration (ZPA) of the applicable seismic response spectra curves. The pump support structure is a rigid steel frame; thus it is appropriate to apply the floor ZPAs at the pump base multiplied by a conservative amplification factor of 1.5. The resulting amplified accelerations were used for the qualification of the pump's pressure boundary components and hold-down bolting.

A summary of calculated stresses, deflections and loads is provided in Table 3.9A-5. The actual values are compared with the ASME code allowables.

LOW PRESSURE SAFETY INJECTION - PUMP MOTOR

The LPSI Pump Motors are Westinghouse Type 5010P39, VSWF.

The LPSI Pump Motors have been qualified for their intended safety function by analysis and testing. The results of the analysis indicates that the lowest natural frequency is greater than 33 Hz and therefore the equipment is considered rigid and qualified by static analysis.

The seismic analysis considered the following DBE seismic loads applied to the motor:

Horizontal	- 1.0g
Vertical	- 0.66g

The summary of the calculated stresses, deflections and loads is provided on Table 3.9A-6. The actual values are compared with the ASME code allowables.

HIGH PRESSURE SAFETY INJECTION PUMPS AND MOTORS

The HPSI Pumps are Bingham Willamette Pumps, Type 3 x 4 x 9 CP7.

The HPSI Pumps have been qualified for their intended safety function by analysis. The results of the analysis indicates that the lowest natural frequency is greater than 33 Hz and therefore the equipment is considered rigid and qualified by static analysis. The HPSI Pump and motor combination experienced the following natural frequencies:

	Pump	Motor
X - Direction	67.01	100.13
Y - Direction	238.61	300.08
Z - Direction	72.65	67.92

The analysis considered the following seismic accelerations applied to the pump/motor combinations:

Horizontal	- 0.36g
Vertical	- 0.18g

A summary of the calculated stresses, deflections and loads is provided on Table 3.9A-7. The actual values are compared with the ASME allowables.

CHARGING PUMPS AND MOTORS

The Charging Pumps are reciprocating pumps, Type TX-125, manufactured by the Union Pump Company.

The Charging Pumps have been qualified for their intended safety function by analysis. The results of the calculations reveal that the lowest natural frequency was greater than 33 Hz and therefore, the pumps were considered rigid and qualified for static analysis. The natural frequencies calculated are as follows:

Pump Fluid End	- 947.40 Hz
Pump Frame	- 594.60
Pump Tank Platform	-58.50
Motor	-86.68

The analysis considered the following seismic loads applied to the pump and motor:

Horizontal	-0.6 g
Vertical	-0.4g

A summary of the calculated stresses, deflections and loads is provided on Table 3.9A-8. The actual values are compared with the ASME code allowables to verify the pump/motor combination operability.

SEISMIC ANALYSIS AND SUMMARY OF RESULTS FOR ACTIVE VALVES

1. MAIN STEAM SYSTEM

a. MAIN STEAM TO AUX TURBINE: TAG NO. MV-08-12, 13
Report No. R93.230 dated Dec. 16, 1993

The main steam to auxiliary turbine valves are products of Anchor/Darling Valve

Company and are 4" 600# double disc gate valves. It has been shown by a static seismic analysis that the valves are structurally adequate to withstand specified operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	2g

A summary of calculated actual stresses are compared with ASME allowables. Note that many resulting stress levels equal the allowable; this is because the stresses are due to the component's maximum allowed actuator thrust and torque in addition to seismic loading. Also the calculated natural frequency is 62.81 Hz. The lowest natural frequency for a static seismic analysis is 33 Hz, as specified in Appendix 3.9A.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
Body Neck	17500	17500
Retainer Groove	12340	17500
Yoke Clamp	29400	29400
Clamp Bolting	12990	43450
Yoke Legs	27850	27850
Yoke Top Bolting	67500	67500
Yoke Top Flange	27850	27850
CRITICAL DEFLECTION		
Calculated	5.103×10^{-3} in.	
Allowed	3.1×10^{-2} in.	

b. MAIN STEAM ISOLATION VALVES – TAG No. HCV-08-1A, 1B
Report No. RAL 3048, dated August 1, 1975

The main steam isolation valves are products of Rockwell International Company and are type: 32 inch Figure 612 GJMMTY. It has been shown by a static seismic analysis that the valves are structurally adequate to withstand specified operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	2g

The Summary of calculated actual stresses are compared with the allowable. Also the calculated natural frequency is 37 Hz. The lowest natural frequency for a static seismic analysis is 33 Hz.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
Yoke Cylinder	5,818	36,000
Operator/Mounting Place Bolts	17,622	45,000
Yoke Base	856	36,000
Yoke-Leg Mounting Base	716	45,000
Deflection		
Upper Structure, Inch	7.2×10^{-3}	

2. FEEDWATER SYSTEM

a. STEAM GENERATOR MAIN FEED: TAG Nos. MV-09-1, 2, 7, 8
(Report No. E-6047-3, dated August 20, 1975)

The steam generator main feed valves are products of Anchor Darling Valve Company and are 20 inch No. 900 Gate Valves. A static seismic analysis was performed and shows by calculation that the valves are structurally adequate to withstand the specified operating and seismic conditions.

THE SEISMIC LOADS SPECIFIED ARE:

	DBE
Horizontal	3g
Vertical	2g

A summary of calculated actual stresses are compared with the allowable. Also the calculated natural frequency is 130 Hz. The lowest allowable natural frequency for static seismic analogy is 33 Hz.

The valve components most highly stressed were the yoke clamp area, yoke clamp bolts and the yoke body.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
Yoke Clamp Area	3334	17500
Yoke Clamp	3770	17500
Yoke Clamp Bolts	21066	25000
Yoke Body	4376	17500

3. COMPONENT COOLING

a. FUEL POOL ISOLATION VALVES - TAG NO. MV-14-17,18

Fuel Pool isolation valves are products of Henry Pratt Company and are 12 inch NMKII W/SMB0002-HIBC, Class 3. It has been shown through calculations that the valves are structurally adequate to withstand specified operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	3g

A Summary of calculated actual stresses are compared with the ASME allowables and are tabulated below. Also the natural frequencies for various components is greater than 783 Hz.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
1. Body:		
a) Primary Membrane (PM)	1,365	17,500
b) Primary + Secondary (SN)	7,080	52,500
2. Operator Mounting		
a) Bolts, Max.	5,382	30,000
b) Bonnet Body	2,088	12,600
3. Banjo Assembly		
a) Disc	9,307	17,500
b) Shaft	17,459	33,700

b. CCW TO SHUTDOWN H.X. TAG NO. HCV-14-3A,B
Report No. D0066-7 dated June 3, 1977

The CCW to shutdown H.X. valves are products of Henry Pratt Company and are 14 inch NMK II W/T-312-SR 4 Class 3. It has been shown by calculations that these valves are structurally adequate to withstand operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	3g

A summary of calculated actual stresses are compared with the ASME allowables and are tabulated below. Also calculated natural frequencies are greater than 1,579 Hz.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
1. Body:		
a) Primary Membrane (PM)	1,414	17,500
b) Primary + Secondary (SN)	7,235	52,500
2. Operator Mounting		
a) Bolts, Max.	25,819	33,700
b) Trunnion Body	2,774	17,500
3. Banjo Assembly		
a) Disc	12,468	16,000
b) Shaft	11,219	33,700

c. CCW TO NORMAL HEADER TAG NO. HCV-14-8 A, B, 9, 10
Report No. D0066-8 dated June 3,1977

The CCW to normal header valves are products of Henry Pratt Company and are 16 inch NMK II W/T-316-SR2 Class 3. It has been shown through calculation that the valves are structurally adequate to withstand specified operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	2g

A summary of calculated actual stresses are compared with the ASME allowables and are tabulated below. Also the natural frequencies for various components are greater than 1309 Hz.

	COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
1.	Body:		
	a) Primary Membrane (PM)	1,377	17,500
	b) Primary + Secondary (SN)	6,196	52,500
2.	Banjo Assembly		
	a) Disc	13,268	16,500
	b) Shaft	11,021	33,700
3.	Thrust Bearing		
	a) Clamp Ring	398	1,500
	b) Thrust Bearing Bolts	375	14,500

4. INTAKE COOLING WATER

a. TURBINE COOLING H.X. ISOLATION: TAG NO. MV-21-2, 3
Report No. D0064-5, dated 5/4/77

The Turbine Cooling H.X. isolation valves are products of Henry Pratt Company and are 24 inch NMK II with SMB 0005/H3/BC Class 3. It has been found through calculation that these valves are structurally adequate to withstand specified operating and design basis earthquake conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	3g

A summary of actual stresses are compared with allowables and are tabulated below. The lowest natural frequency calculated for any valve component was 577 Hz.

	COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
1.	Body:		
	a) Primary Membrane (PM)	1,977	17,500
	b) Primary + Secondary (SN)	8,099	52,500
2.	Operator Mounting		
	a) Bonnet Body	2,775	12,600
	b) Bolting, Max.	20,403	23,000
	c) Welding, Max	1,248	8,400
3.	Banjo Assembly		
	a) Disc	15,598	16,000
	b) Shaft	11,152	17,200
	c) Pins, Max.	6,329	9,300

b. ICW TO CIRCULATING WATER PUMPS: TAG NO. MV-21-4A&B
Report No. D0064-6, dated May 4, 1977

The Intake Cooling Water to Circulating Water Pump valves are products of Henry Pratt Company and are 3 inch NMK II with SMB 0002/ HOBC, Class 3. It has been shown through calculations that the valves are structurally adequate to withstand operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	3g

A summary of calculated actual stresses are compared with the ASME allowables and are tabulated below. The lowest natural frequency calculated was 6099 Hz. The lowest natural frequency for a static analysis is 33 Hz as outlined in appendix 3.9.

	COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
1.	Body		
	Primary Membrane (PM)	557	17,500
	Primary + Secondary (SN)	4,200	52,500
2.	Operator Mounting		
	Bolts, Max.	24,699	37,500
	Trunnion Body	679	17,500
3.	Banjo Assembly		
	Disc	3,585	16,000
	Shaft	10,598	17,200
	Pins	5,182	9,300

5. CONTAINMENT SPRAY

a. PUMP SUCTION FROM RWT: TAG NO. MV-07-1A, B
Report No. D-0064-1, dated May 4, 1977

The pump suction from Refueling Water Tank valves are products of Henry Pratt Company and are 24 inch NMK II with SMB 0005/H3BC, Class 2. It has been shown through calculations that these valves are structurally adequate to withstand specified operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	3g

A summary of calculated actual stresses are compared with ASME allowables and are tabulated below. The lowest natural frequency was calculated to be 557 Hz. Since this is higher than the 33 Hz specified, a static analysis was performed.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
1. Body:		
Primary Membrane (PM)	1,977	16,000
Primary + Secondary (SN)	8,099	48,000
2. Operator Mounting		
Bolts, Max.	17,249	23,000
Trunnion Body	675	16,000
3. Banjo Assembly		
Disc	15,593	16,000
Shaft	11,152	33,700
Pins	6,329	9,300

b. PUMP SUCTION FROM SUMP: TAG NO. MV-07-2A
Report No. D0064-2, dated May 4, 1977

This pump suction from sump valve is a product of Henry Pratt Company and is a 24 inch NMK II W/SMB0010/H3BC Class 2. It has been shown through calculations that this valve is structurally adequate to withstand specified operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	3g

A summary of calculated actual stresses are compared with the ASME allowables and are tabulated below. The lowest natural frequency calculated was 557 Hz. Since this exceeds a natural frequency of 33 Hz, a static analysis was performed.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
1. Body:		
Primary Membrane (PM)	1,977	16,000
Primary + Secondary (SN)	3,099	48,000
2. Operator Mounting		
Bolts, Max.	19,496	23,000
Trunnion Body	745	16,000
3. Banjo Assembly		
Disc	15,593	16,000
Shaft	11,152	33,700
Pins	6,329	9,300

c. PUMP SUCTION FROM SUMP - TAG NO. MV-07-2B
Report No. D-0088-1, dated June 3, 1977

The pump suction from sump valves are products of Henry Pratt Company and are 24 inch NMK II W/SMB0010/H3BC Class 2. It has been shown by calculation that these valves are structurally adequate to withstand operating and seismic conditions,

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	3g
Vertical	3g

A summary of calculated actual stresses are compared with ASME allowables and are tabulated below. The lowest natural frequency calculated was 557 Hz. Since this exceeds a natural frequency of 33 Hz, a static analysis was performed.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
1. Body:		
Primary Membrane	1,977	16,000
Primary + Secondary	8,099	48,000
2. Operator Mounting		
Bolting	19,496	23,000
Trunnion Body	745	16,000
3. Banjo Assembly		
Disc	15,598	16,000
Shaft	11,152	33,700
Pins	6,329	9,300

6. SHIELD BUILDING EXHAUST

SHIELD BUILDING VENTILATION SYSTEM INLET-TAG NO. FCV-25-32,33

The air purge valves are products of Henry Pratt Company and are 30 inch NRIA W/SMB0010-H3BC, Class 2. It has been shown through calculation that the valves are structurally adequate to withstand specified operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	5.0g
Vertical	5.0g

A summary of calculated actual stresses are compared with the ASME allowables and are tabulated below. The lowest natural frequency was calculated to be 650 Hz. Therefore, a static analysis was performed.

COMPONENTS	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
Primary Membrane (PM)	431	17,500
Primary + Secondary (SN)	2,579	52,500
Disc	5,041	26,250
Shaft	21,262	33,700
Trunnion Bolts	6,862	30,000
Bonnet Bolts	6,687	30,000
Operator Bolts	4,246	30,000
Bonnet Body	1,311	12,600
Trunnion Body	243	17,500
Bonnet Welds	854	7,200

7. CONTROL ROOM A/C

AIR INTAKE VALVES - TAG NO. FCV-25-14, 15, 16 & 17

The air intake valves are products of Henry Pratt Company and are 12 inch NRS W/SMB0002-HIBC, Class 3. It has been shown through calculations that the valves are structurally adequate to withstand specified operating and seismic conditions.

THE SPECIFIED SEISMIC LOADS ARE:

	DBE
Horizontal	4g
Vertical	4g

A summary of calculated actual stresses are compared with the ASME allowables and are tabulated below. The lowest natural frequency was calculated to be 357 Hz. Therefore, a static analysis was performed.

COMPONENT	ACTUAL STRESS (PSI)	ALLOWABLE STRESS (PSI)
Primary Membrane (PM)	787	17,500
Primary + Secondary (SN)	5,386	52,500
Disc	3,931	26,250
Shaft	18,945	33,700
Disc Pin	3,140	13,600
Shaft Bearing	3,061	4,000
Trunnion Bolts	24,197	30,000
Bonnet Bolts	15,839	30,000
Operator Bolts	3,646	30,000
Bonnet Body	2,879	12,600
Trunnion Body	1,128	17,500
Bonnet Welds	1,291	7,200

8. CONTAINMENT AIR MONITOR

Radiation Monitor Isolation Valves - Tag No. FCV-26-1, 2, 3, 4, 5, 6

The radiation monitor isolation valves are products of WKM Valves Company and are type 1/2" 6C 70-14-2 S35 DRT. The "ANSYS" finite element program was used to determine the natural frequencies:

X - Direction 45.14 Hz
Y - Direction 46.07 Hz

Static seismic calculations show the following results:

The Specified Seismic loads Are:

	Upset	Faulted
Horizontal	1.5g	3.0g
Vertical	1.0g	2.0g

Components	Actual Stress (PSI)	Allowable Stress (PSI)	Actual Stress (PSI)	Allowable Stress (PSI)
Actuator Yoke Legs	2793	32500	4028	25000
Body/Bonnet	4281	27500	6168.6	50000

9. CHEMICAL & VOLUME CONTROL SYSTEM

a. Letdown Line Isolation Valves Nos. V2516, V2515
Report No. ES-107, dated October 18, 1978

The letdown line isolation valves are products of the Fisher Valve Company and are 2 inch pneumatic operated valves. The lowest calculated natural frequency was 23 Hz and thus the valves were considered rigid and qualified for static analysis.

A summary of actual stresses are compared with the ASME Code allowables and are tabulated below:

Component	Actual Stress (PSI)	Allowable (PSI)
Yoke Leg	24,837	36,000
Lock Nut	14,330	40,000
Bonnet	3,458	18,500
Bonnet Thread	4,766	18,500
Body to Bonnet Bolt	22,351	53,800

b. Letdown Line Isolation Valve No. V2522
Report from Fisher dated April 19, 1977

The letdown line isolation valves are products of the Fisher Valve Company and are 2 inch pneumatic operated valves. The lowest calculated natural frequency was 22.5 Hz and therefore a static analysis was performed.

A summary of the actual stresses are compared with the ASME Code allowables and are tabulated as follows:

Component	Actual Stress	Allowables
Yoke Leg	28,986	36,000
Lock Nut	43,786	54,375
Bonnet	6,876	30,000
Bonnet Thread	10,817	18,000
Body to Bonnet Bolt	29,084	54,100

10. SAFETY INJECTION SYSTEM

a. SIT Drain Isolation Valve - No. V-3472 (This valve is not installed in the field)
Report from Fisher, dated April 19, 1979

The SIT Drain Isolation Valve is a 2 inch pneumatic operated valve supplied by the Fisher Valve Company. The lowest natural frequency calculated was 21.8 Hz and therefore a static analysis was performed.

A summary of the actual stresses are compared with the ASME code allowables and are tabulated as follows:

Component	Actual Stress	Allowables
Yoke Leg	28,617	36,000
Lock Nut	43,464	54,375
Bonnet	6,846	30,000
Bonnet Thread	10,741	18,000
Body to Bonnet Bolt	52,103	54,300

b. Shutdown Cooling Line Valves FCV-3301, 3306, HCV-3512, 3657
Action Test Report No. 13865 and John Henry Associates Report JHA-77-94A

The shutdown cooling line valves are 10 in motor operated butterfly valves supplied by the Jamesbury Valve Company. These valves are supplied with Limitorque SMB000/2-HIBC operators. The lowest natural frequency calculated was 31.46 Hz and therefore, a static analysis was performed. This analysis was verified by tests.

A summary of the actual stresses are compared with the ASME code allowables and are tabulated as follows:

Component	Actual Stress	Allowables
Neck	13,183	26,480
Wafer Plate	15,106	26,480
Bolts	11,191	18,000
Shaft	23,685	60,750
Wafer Support	25,336	26,480
Bracket	18,880	34,200

A Sine Beat Test and Resonance Search was performed to verify the validity of the analysis and to assure the operability of the valve and operator combination. The test was performed in accordance with Appendix 3.9A and verified the validity of the static analysis.

11. WASTE MANAGEMENT SYSTEM

a. Containment Isolation Valves Nos. V6718, V6750
Report No. W105 dated March, 1976

The Waste Management Containment Isolation Valves are products of the Grinnell Valve Company. It has been shown by calculations that the lowest natural frequency was 28 Hz and, therefore, a static analysis was justified.

A summary of calculated stresses are compared with the ASME Code allowables and are tabulated below:

Components	Actual Stress	Percent Yield
Bolting	19,656	24%
Bonnet	12,576	53%

b. Reactor Drain Tank Isolation Valves Nos. V6341, V6342
Report No. W105, dated March, 1976

The Reactor Drain Tank isolation valves are products of the Grinnell Valve Company. It has been shown by calculation that the lowest natural frequency was 28 Hz, and, therefore, a static analysis was justified.

The summary of the calculated stresses are compared with the ASME code allowables and are tabulated as follows:

Component	Actual Stress	Percent Yield
Bolting	17,266	21%
Bonnet	2,838	12%
Adapter Bushing	42,688	71%
Yoke	7,868	20%

TABLE 3.9A-1

PERCENT CRITICAL DAMPING

	OBE (0.05g Ground Surface Acceleration)	SSE (0.10g Ground Surface Acceleration)
Welded Steel Plate Assemblies	1	1
Steel Containment Vessel	2	4
Welded Steel Framed Structures	2	4
Bolted or Riveted Steel Framed Structures	4	7
Reinforced Concrete Equipment Supports	4	7
Reinforced Concrete Frames and Buildings	4	7
Steel Piping		
(Pipe dia > 12 in.)	2	3 (Note 1)
(Pipe dia ≤ 12 in.)	1	2 (Note 1)

Note 1:

For piping, damping values specified by ASME Code Case N-411 may be used as described in sub section 3.9.3.1.1 h) (page 3.9.37.)

TABLE 3.9A-2

SUMMARY OF RESULTS - CONTAINMENT SPRAY PUMPS

Structural Integrity (Faulted Condition)

Component		Calculated, psi	Allowable, psi
Casing Foot Attachment		15,431	36,000
Casing Disch. Noz. Attachment		29,213	36,000
Casing Suct. Noz. Attachment		11,953	36,000
Suction Flange		28,798	44,820
Discharge Flange		29,661	44,820
Main Flange Bolting		16,111	37,500
Foot		22,090	29,880
Foot Weld		21,100	29,880
Anchor Bolting	Tension	18,801	40,000
	Shear	5,806	15,390
Support Head		207	22,680
Motor Attachment Bolting		1,837	45,000

Operability

Description	Calculated	Allowable
Rotor/Stator Deflection (Motor Air Gap)	.0001	.050 in.
Impeller/Ring Deflection	.003	.0115
Shaft/Cover Deflection at Mechanical Seal	.0015	.010

TABLE 3.9A-3

SUMMARY OF RESULTS - CONTAINMENT SPRAY PUMP MOTOR

		Actual	Allowable
1.	Anchorage System		
1.1	Normal Operational Loading		
	Tensile Stress	0	20,000
	Shear Stress/Bolt	423	10,000
1.2	Externally Applied Thrust Loading		
	Tensile Stress/Bolt	1,308	20,000
	Shear Stress	0	10,000
1.3	Seismic Loading		
	Horizontal Acceleration Induced		
	Tensile Stress Max.	1,042	20,000
	Shear Stress/Bolt	616	10,000
	Vertical acceleration Induced		
	Tensile Stress/Bolt	436	20,000
	Shear Stress	0	10,000
1.4	Combined Loading		
	Tensile Stress Max.	1,517	20,000
	Shear Stress Max.	1,114	10,000
2.	Rotor Response		
	Deflection @ Core	0.0015 in.	
	Shaft End Bearing Reaction	1,093 lbs.	
	Front End Bearing Reaction	662 lbs.	
	Deflection @ Shaft Extension	0.0011 in.	
	Lateral Critical Speed	2,217 cpm	
3.	Bearings		
	Top		
	Equivalent Load	5,778 lbs.	
	Life	84,504 Hours	
	Bottom		
	Equivalent Load	1,093 lbs	
	Life	330,533 Hours	
4.	Conduit Box		
	Tensile Stress Max.	18,526	20,000
	Shear Stress/Bolt	2,863	10,000

TABLE 3.9A-4

SUMMARY OF RESULTS - DIESEL OIL TRANSFER PUMPS

A summary of the stresses, deflections, and loads are given here. Faulted values are given and compared to the Normal allowable values.

Components		Actual	Allowable
Motor Hold Down Bolts Stress, PSI	- Shear	2,651	10,000
	- Tensile	4,670	20,000
Pump Hold Down Bolt Stress, PSI	- Shear	9,056	12,320
	- Tensile	22,522	35,510
Anchor Bolt Stress, PSI - Shear		3,953	10,000
- Tensile		9,615	20,000
Shaft Stress, PSI		4,333	17,500
Frame Stress, PSI		10,879	21,750
Thrust Retainer Bolt Stress, PSI		2,414	20,000
Pump Frame Bolt Stress, PSI - Shear		7,818	10,000
- Tensile		7,906	20,000
Frame Adapter Bolt Stress, PSI - Tensile		13,629	25 000
Frame Adapter Flange Stress, PSI		11,960	21,000
Maximum Nozzle Stress, PSI - Discharge		14,053	20,760
- Suction		9,152	20,760
Nozzle Flange Stress, PSI - Discharge		19,256	20,760
- Suction		18,438	20,760
Pump Bearing Loads, lbs. - Inboard		320	9,104
- Outboard		1,849	15,038
Flexible Coupling Misalignment, Radians		.004	.004
Impeller Connection Stress, PSI - Shear		711	8,750
- Tensile		1,541	17,500
Impeller Relative Deflection, Inches		.002	.025

TABLE 3.9A-5

SUMMARY OF RESULTS - LPSI PUMPS

EC275737

Original Close Coupled Design

Component	Calculated, psi	Allowable, psi
Casing Foot Attachment	15,913	28,050
Casing Disch. Noz. Attach.	7,849	28,050
Casing Suction Noz. Attach.	17,800	28,050
Main Flange Bolting	22,225	37,500
Foot	20,720	24,300
Foot Weld	20,054	24,300
Anchor Bolting		
Tension	17,143	40,000
Shear	6,202	15,390
Support Head	l,029	18,900
Motor Attachment Bolt	9,845	37,500
Rotor/Stator Deflection	.002 in	.05 in
Impeller Ring Deflection	.0055	.0115
Shaft/Cover Deflection	.0023	.0100

EC275737

Modified Coupled Design

Component or Part	Actual Value	Allowable Value
Seal Gland Bolting Stress	7,029.2 psi	11,450 psi
Support Head Stress	1,810 psi	20,000 psi
Hold Down Bolting	14,307 psi	25,000 psi
Pump Case Foot Stress	11,805 psi	24,600 psi
Pump Foot Weld Stress	9,735 psi	24,000 psi
Shaft Deflection at Impeller	0.0026 in	0.012 in (Max)
Rotor Deflection	0.0007 in	0.022 in

TABLE 3.9A-6

SUMMARY OF RESULTS - LPSI MOTORS

Component	Calculated, psi	Allowable, psi
Shaft - Normal	15,835	40,000
- Shear	14,363	24,000
Bearing Loads -Upper	1,304	15,600
- Lower	6,192	93,600
Bearing Casing:		
Tapped Holes in Housing	1,202	12,000
Tapped Holes in Cap	4,762	20,000
Hold Down Bolts	8,452	61,640
Stator Core Welds	5,804	21,000
Stator Core Sppt. Welds	1,398	21,000
Motor Frame End Flange:		
Tapped Holes in Flange	770	12,000
Flange Welds	1,564	21,000
Rotor Deflection	.0014 inches	.022 inches
Shaft Deflection (at impeller)	.0036 inches	.0115 inches
Shaft Slope, (at impeller)	.018 degrees	*

* The allowable value is not applicable for shaft slope.

EC275737

The values above are for the original close coupled design. See Table 3.9A-5 for deflections that apply to the modified coupling design.

TABLE 3.9A-7
SUMMARY OF RESULTS - HPSI PUMPS AND MOTORS

Component	Calculated, psi	Allowable, psi
Pump Hold Down Bolts:		
Tension	3,653	40,000
Motor Hold Down Bolts:		
Compression	448	40,000
Motor Dowel Pins:		
Shear	3,280	21,000
Pump Foot Taper Pins:		
Shear	6,882	21,000
Base Hold Down Bolts:		
Shear	7,835	20,000
Tension	9,132	54,000

TABLE 3.9A-8
SUMMARY OF RESULTS -CHARGING PUMPS AND MOTORS

Component	Calculated, psi	Allowable, psi
Pump Hold Down Bolts:		
Shear	5,128	24,000
Tension	4,684	30,000
Motor Hold Down Bolts:		
Shear	I,972	24,000
Tension	3,694	30,000
Foundation Bolts:		
Shear	5,938	36,000
Tension	6,604	36,000
Dowel Pins - Shear	12,667	23,400
Tie Studs - Tension	1,756	28,000
Motor Bearing Loads:		
Rear	410 lbs	20,100 lbs
Forward	201 lbs	10,900 lbs
Motor Feet	273	26,000
Motor Mounting Bolts	1,892	44,600
Shaft	1,938	5,714
Shaft Deflection	.00176 in.	.0597

APPENDIX 3.9B

CONCRETE EXPANSION ANCHOR DESIGN

3.9B CONCRETE EXPANSION ANCHOR DESIGN

3.9B.1 LOADS

The following loads are considered applicable to the design of concrete expansion anchors:

D	=	Dead loads, including the weight of stationary structures, piping and equipment
L	=	Live loads, including any movable equipment loads and loads due to the operation of equipment.
W	=	Hurricane Load.
F_{eqs}	=	Design Basis Earthquake (DBE), including effects of differential movement of supports.
R_o	=	Pipe reaction loads during normal operating or shutdown conditions.
R_a	=	Pipe reaction loads under thermal conditions generated by a postulated pipe break, including R_o.
T_o	=	Thermal effects and loads during normal operating or shutdown conditions.
T_a	=	Thermal effects under conditions generated by a postulated pipe break, including T_o
Y_r	=	Reactions loads generated by a postulated pipe break, including an appropriate factor to account for the dynamic nature of the load.

3.9B.2 LOAD COMBINATIONS

The following load combinations are used to compute the tensions, moments, and shears for use in the design of concrete expansion anchors:

$$D+L+W+R_o+T_o$$

$$D+L+F_{eqs}+R_o+T_o$$

$$D+L+F_{eqs}+R_a+T_a+Y_r$$

3.9B.3 ALLOWABLE LOADS

The ultimate loads carried by concrete expansion anchors are obtained from applicable design specification(s), modified by a capacity reduction factor of safety determined in accordance with paragraph 3.9B.4. In addition, criteria for capacity reduction due to bolt spacing and edge distance shall also be obtained from the applicable design specification(s).

3.9B.4 DETERMINATION OF PRYING FORCES

The method of evaluation is the use of the safety factors provided in NRC IEB 79-02, Revision 2 (and draft Revision 2 of NRC Standard Review Plan Section 3.8.4) and manual calculation methodology, or finite element computer analysis for the determination of prying forces.

3.9B.5 DETERMINATION OF ANCHOR SIZES

Concrete expansion anchors are selected such that:

$$\frac{B}{Ba} + \frac{V}{Va} \leq 1$$

where:

B	=	total factored tensile load per anchor, including prying forces, if any
B_a	=	allowable anchor tensile capacity, including capacity reduction factor and effects of close spacing
V	=	total factored shear load per anchor
V_a	=	allowable anchor shear capacity, including capacity reduction factor and effects of close spacing.

3.9B.6 BASEPLATE FLEXIBILITY ANALYSIS (Historical)

To account for the flexibilities of both the concrete expansion anchor and the baseplate, the "ANSYS" computer program is employed. This program utilizes the finite element method of analysis. To facilitate the use of the "ANSYS" program Ebasco developed a preprocessor "EMBEDP". A brief description of the "EMBEDP" computer program is given below. The test problem presented in Section 5.0 of "Summary Report of Generic Response to USNRC IE Bulletin No. 79-02 Baseplate/Concrete Expansion Anchor Bolts by Teledyne Engineering Services, August 30, 1979," is used to verify the "EMBEDP" program. Subsection 5.5.3 of that report shows the plate geometry and Subsection 5.6.1 gives the bolt load. The EMBEDP output (ANSYS input data) results from these sample problems are introduced into the ANSYS program, the output of which is then compared with that of the Teledyne report. Table 3.9B-1 compares the EMBEDP and Teledyne results for the bolt loads.

3.9B.6.I Description of Model

The plate is divided into a finite number of elements (STIF63). While dividing the plate into elements it is desirable to increase the number of elements in the region of expected maximum stresses. In other areas fewer elements may be used. To increase the convergence of the results it is common to have two rows of elements between the edge of the plate and the bolt line.

The concrete is replaced by compression-only springs (STIF10) derived from the half-space theory as given by Barkan. The total stiffness Kc of concrete subgrade is given as:

$$K_c = \frac{G_c}{1 - V_c}(2.2) \quad \sqrt{WL}$$

G_c = shear modulus of concrete

V_c = Poisson's ratio of concrete

W = width of baseplate

L = length of baseplate

Compression springs representing the concrete subgrade are attached to each node of the model.

Bolts are represented by tension springs (STIF10) in the longitudinal direction. The longitudinal stiffnesses of the bolts are obtained from tests performed at the job site by the bolt manufacturer.

Shear stiffnesses of the bolts (STIF14) are also derived from test results. In this analysis these values were taken from, "Anchor Bolt Shear and Tension Stiffness," Teledyne Engineering Services, May 25, 1979. Since the stiffness of the plate in the horizontal direction (in-plane stiffness) is relatively large compared to the shear stiffness of the bolt, the shear force distribution among the bolts (all of the same type and size) is not affected by the shear stiffness of the bolt. For this reason, it is possible to distribute the total shear force among the bolts without resorting to the ANSYS analysis. However, in the analyses performed for St. Lucie Unit 2, all loads, pullout and moment as well as shear, are applied in the same run in the knowledge that the shear force taken by the bolts would affect neither the tension in the bolt nor the plate stress.

A part of the attachment is included in the model as plate elements. The load is applied to this attachment.

3.9B.6.2 Study of Model Mesh Size

The baseplate for restraint CH-71-R1 is selected to study the effect of element sizes on the stresses and the bolt tension. This restraint baseplate is typical of the majority of expansion anchored restraint applications. Pullout load is applied to the plate. A part of the attachment is modeled as plate elements. Two computer runs are performed with the 3/4" plate divided into 5 x 5 elements and 8 x 8 elements as shown on Figure 3.9B-1. Table 3.9B-2 presents the values of maximum bolt load and maximum plate stress obtained from the two computer runs for two different mesh sizes. From these two cases it can be seen that the difference in bolt tension is small (0.06%) while the maximum stresses differ by 2.20%.

3.9B.6.3 Study of Superposition of Loads

An analysis of a sample baseplate subjected to pullout load and moment is performed, considering these effects applied separately, and an analysis considering these effects applied simultaneously.

Four individual load cases are considered separately. The four load cases are (1) pullout load F_z', (2) moment M_x, (3) moment M_y and (4) combined loads F_z, M_x', and M_y' applied simultaneously. The plate selected was CH-71-R1 (see Figure 3.9B-1). Results from these cases are summarized below. It may be pointed out that the location of maximum stress is different for each load case. However, in the actual analysis using the "ANSYS" program for St Lucie 2 critical combinations of individual loads are used. Single load applications are never considered in the analysis.

Applied Load	Maximum Bolt Tension Load	Maximum Plate Stress
Fz' = 1.398	0.350	2.19
Mx' = 0.4.96	0.026	0.20
My' = 0.496	0.026	0.20
Fz' = 1.398 Mx' = 0.496 My' = 2.622	0.512	2.72

Load in KIP
Stress in KSI

Moment in IN-KIP

3.9B.6.4 Ebasco Computer Program "EMBEDP"

The "EMBEDP" computer program was developed by Ebasco as a preprocessor for the "ANSYS" finite element program for baseplate and anchorage nonlinear analysis. This program automatically generates the finite element model including the load data using a minimum number of input cards. The preprocessor minimizes engineering time and allows solution of a large number of baseplate problems economically. The program has been completed and verified.

The program structure is sufficiently flexible to allow the user to exercise options in considering special features of different problems. The following special features are included and can be handled by the program:

a. Selection of the type of element (bending only or membrane plus bending) for the baseplate. For the case with uplift force only, the bending type element can be used to reduce the computer cost.

b. Generation of the spring constants of the concrete subgrade using the half-space formula developed by Barkan.

c. Consideration of the pretorque in the anchor bolts.

d. Consideration of the friction between the baseplate and concrete surfaces. If it is required to take into account the friction between the baseplate and the concrete, the friction element (STF52) may be included in the analysis. When this element is selected, the baseplate is automatically represented by a membrane plus bending element. (In the analyses performed for St. Lucie Unit 2, the friction element was not used to carry shear loads.)

e. Location and Number of bolts - Any random distribution, up to 20 bolts can be input.

f. The attachments - any attachment having components parallel to the sides of the baseplate can be input.

"EMBEDP" together with "ANSYS" provides stresses in baseplates and forces in bolts on plate assemblies subject to various loadings.

3.9B.7 DESIGN OF PIPE RESTRAINTS USING EXPANSION ANCHORS

A safety factor of 15 is used across the board (i.e., applied to dead, thermal and seismic loads) in the design of the small-bore safety-related pipe supports which require the use of expansion anchors. All the small-bore safety-related pipe supports were designed by Ebasco Engineering.

The large-bore safety-related pipe supports at St. Lucie Unit 2 were designed by Bergen-Paterson. Bergen-Paterson does not use a factor of safety of 15 in their design of these supports. In order to verify the design adequacy of expansion anchor applications for large-bore safety-related pipe supports, 19 of the Bergen-Paterson designs which represent the worst cases are analyzed using the ANSYS computer program. The results were compared to the ultimate concrete expansion anchor capacities to determine the actual factors of safety.

All but one case resulted in a factor of safety of at least five for bolt tension and shear. The one exception is a main steam restraint whose analysis resulted in a factor of safety of 2.4 for bolt tension (bolt shear is negligible). This restraint is unique in that the loads are very large. The pullout force is an order of magnitude greater than that of any other expansion-anchored restraint. The design of this restraint has been modified to achieve a factor of safety of at least four.

TABLE 3.9B-1

BOLT LOAD COMPARISON

Load Case	Bolt Number	Bolt Load (lb) EMBEDP	Bolt Load (lb) Teledvne
Case 1	1	2324	2350
	2	2324	2350
(Axial Load)	3	2324	2350
	4	2324	2350
Case 2	1	2272	2316
(45° Shear/	2	972	1024
Moment	3	0.0	0
	4	972	1024
Case 3	1	1860	1942
	2	1860	1942
(0° Shear/ Moment	3	0	0
	4	0	0

Note: Primary difference is due to different formulas used for concrete spring.

TABLE 3.9B-2

BOLT LOAD COMPARISON

1/4 Pull Out Load Kip	1/4 Plate 5 x 5 Elements Maximum Plate Stress in KSI	5 x 5 Elements Maximum Bolt Load in KIP	8 x 8 Elements Maximum Plate Stress in KSI	8 x 8 Elements Maximum Bolt Load in KIP
0.3495	2.19	0.3498	2.24	0.3496

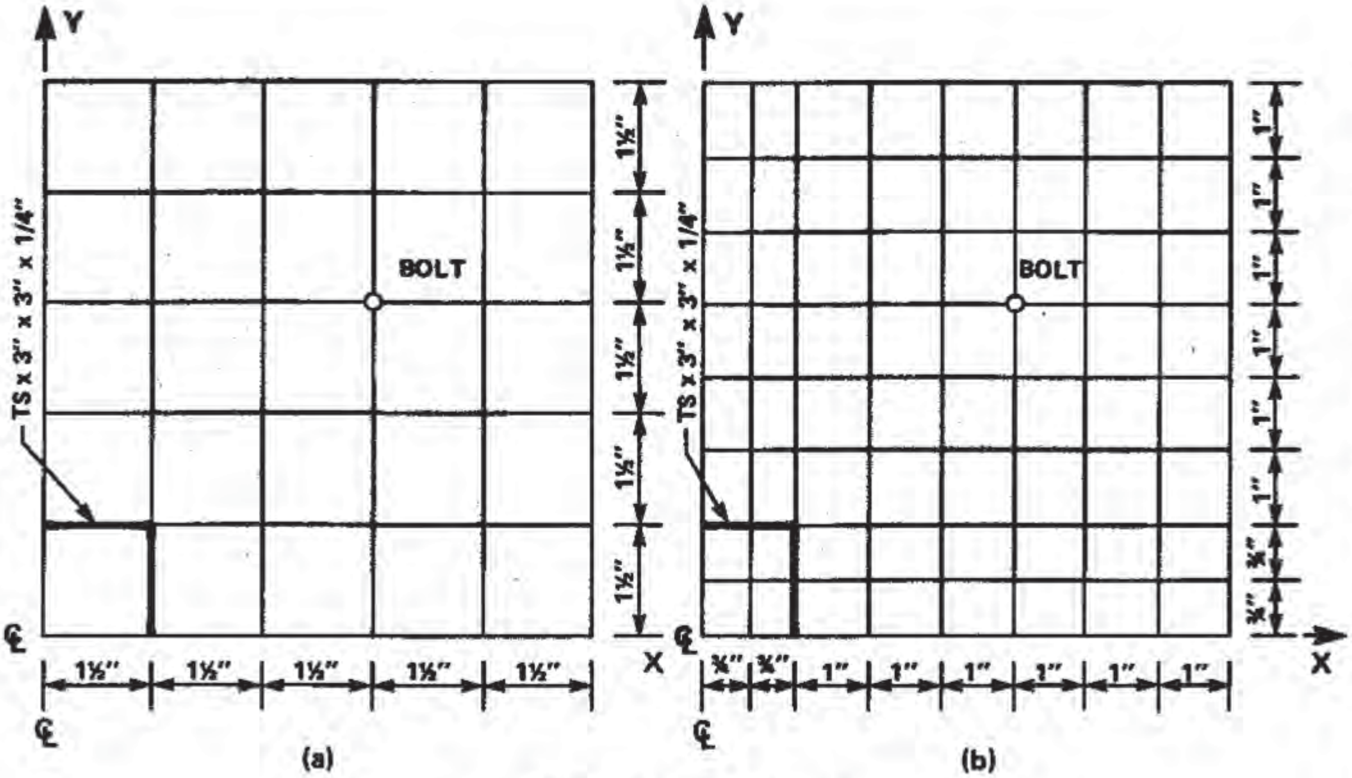


PLATE THICKNESS = 3/4 INCH

MODEL OF 1/4 PLATE
5 x 5 ELEMENTS

MODEL OF 1/4 PLATE
8 x 8 ELEMENTS

3/4" ϕ PHILLIPS WEDGE ANCHOR
1/4 PULL OUT LOAD = 0.3495 K

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

BASE PLATE ANALYSIS

FIGURE 3.9B-1

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 SEISMIC QUALIFICATION CRITERIA

Class 1E instrumentation and electrical equipment associated with the Reactor Protective System, engineered safety features equipment, emergency power system and auxiliary safety related systems are designed as seismic Category I to ensure their ability to perform their required function during, and/or following a postulated safe shutdown earthquake (SSE) and to supply standby electrical power following a SSE to safety related components. This capability is accomplished by one of the following three methods, where the choice of method is based on the practicability of the method for the type, size, shape, and complexity of the equipment and the reliability of the conclusions:

a. Mathematical Analysis Method -

   Equipment performance is predicted by mathematical analysis techniques using mathematical predictions of the natural frequency and the determination of equipment maximum responses (stress and displacement) at a critical section and/or point of interest under the effects of the postulated seismic loading. Analysis without testing is deemed acceptable when structural integrity alone will assure the intended functions.

b. Testing Method -

   The equipment performance is verified by testing under simulated seismic conditions as specified in the equipment specifications. Test data demonstrates that the equipment remains functional during and after the postulated seismic event.

c. Combined Analysis and Testing Method -

   Equipment is qualified using analysis to verify or to extrapolate test results. This is particularly apt when there are many different combinations of equipment which are basically of the same type, where changes are due to different models or sizes and it is impractical to test every variation. The tests are designed to gather sufficient data to enable valid mathematical models to be established.

A list of seismic Category I instrumentation, electrical and auxiliary equipment is found in Tables 3.10-1 and 2. The data presented therein provide the response to an earlier NRC request[1,2] for additional seismic qualification information for Class 1E equipment. (Environmental qualification is discussed in Section 3.11.)

The operating basis and safe shutdown earthquake horizontal and vertical floor response spectra, worst case acceleration loadings or the static equivalent loading are provided to the vendor for each given instrumentation or electrical equipment location. The vendors are required to submit qualification data which demonstrates that the component is capable of functioning under the specified seismic loading. The qualification data may consist of test results, or mathematical analysis, or a combination of these.

In designing the equipment, the vendor combines the effects of gravity loads, normal operating loads, operating temperature loads, other loads that may be included, as appropriate, in the specification, and the appropriate SSE and OBE loads.

The adequacy of the seismic qualification program is demonstrated by documentation requirements the vendor fulfills for each equipment type. This documentation demonstrates that the equipment meets its performance requirements when subjected to the loads for which it was qualified. Additional data on vendor qualification was submitted to the NRC under separate cover.(5)

Appendix 3.10A contains the information provided to manufacturers which presents the criteria for seismic qualification of seismic Category I equipment for St. Lucie Unit 2 except for NSSS instrumentation equipment which are contained in CENPD-182.(3)

3.10.2 METHODS AND PROCEDURES FOR QUALIFYING ELECTRICAL EQUIPMENT AND INSTRUMENTATION

CENPD-182 provides a description of the seismic qualification program for NSSS supplied Class 1E instrumentation. Seismic Category I A/E supplied instrumentation, and A/E and NSSS supplied electrical equipment and supports are qualified by either analysis or testing, or by a combination of analysis and testing as indicated in Tables 3.10-1 and 2.

As required by the Safety Evaluation Report commitment, seismic qualification of seismic Category I instrumentation and electrical equipment is in accordance with IEEE 344-1971 and multi-axis and multi-frequency testing unless specific requirements are met that demonstrate single-frequency or single-axis testing is sufficient. However, as indicated in Tables 3.10-1 and 2, the applicant has purchased Class 1E equipment with qualifications exceeding the SER commitments as far as the state-of-the-art knowledge was available at the time of the purchase order.

As stated in Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," March 1976 (R0) in effect at the time of the Construction Permit, this guide is to be used "in the evaluation of submittals for construction permit applications docketed after November 15, 1976," and thus is not applicable to St. Lucie Unit 2.

Certification is obtained from each manufacturer of Class 1E equipment that his equipment will perform without a loss of function resulting from the stipulated qualifying seismic loading conditions of the SSE in combination with all other applicable loadings as specified in equipment specifications, and that the equipment will remain operable during and after the occurrence of an OBE. Non-Class 1E equipment that interfaces with Class 1E equipment is analyzed or tested to demonstrate that it does not impair the functional capability of the Class 1E equipment under the effects of a safe shutdown earthquake.

For seismic qualification by analysis, the supporting data and design calculations show that the equipment satisfies the specifications. Data and calculations are required to be submitted with a certificate of compliance to substantiate that the equipment will not lose its structural integrity during or after an SSE as a result of the encountered seismic loadings.

3.10.3 METHODS AND PROCEDURES OF ANALYSIS OR TESTING OF SUPPORTS FOR ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Cable Trays Conduits and Field Fabricated Instrument Racks

Supports for electrical cable trays, conduits and field fabricated instrument racks are analyzed to ensure their structural capability to withstand a postulated seismic event. With the exception of certain justified cases, a three dimensional model of a given support is used to perform a seismic response analysis. The simultaneous effects of three orthogonal components of the earthquake on a support are considered and the results (stress and displacement at the location of interest) are evaluated by the square-root-of-the-sum-of-the-squares method.

The first natural frequency of the cable tray (HVAC) restraint is determined after member selection to ensure that the minimum natural frequency of 16 hz (15 hz) is satisfied. Amplification factors are used to account for the participation of higher modes. The cable tray (HVAC) restraints are welded to steel embedments with a fillet weld all around.

Vendor Designed Instrument Racks

The Instrument Racks provided by Mercury Company of Norwood, Mass, were seismically qualified by Acton Environmental Testing Corporation. The method Acton used was a finite element digital computer analysis of three representative instrument racks. The racks used in the test were the worst case examples of all the racks which were qualified. The racks were modeled with beam elements for general structural members and plate elements to simulate gusset stiffeners. The mounted equipment is modeled as a mass element at the equipment's center of gravity, connected to the appropriate support structure with stiff beams. The response spectra imposed was provided for the applicable floor locations and elevations. The spectra was imposed in three axes simultaneously and the modes summed by a square root sum of the square summation, except where closely spaced modes (within 10 percent) are combined absolutely. The maximum principal stresses were calculated and compared to 90 percent of material minimum yield strength from ASME or AISC code values.

Battery Racks

The battery racks were tested on a Wyle multi-axis seismic simulator table, with batteries installed. The racks were initially mounted on the test table in a side-to-side/vertical orientation. Upon completion of the specified sequence of tests, the specimens were rotated 90 degrees in the horizontal plane to the front-to-back/vertical orientation. A low level (approximately 0.2g horizontally and 0.1g vertically) biaxial sine sweep was performed to determine major resonances in both orientations. The sweep rate was one-half octave per minute over the frequency range of one Hz to 40 Hz. Five OBE, followed by one SSE test were performed in both vertical orientations.

RTG Boards

The seismic analysis of the Reactor Turbine Generator (RTG) Control Board was implemented utilizing a multi-degree of freedom mathematical model to ensure adequate representation in two major horizontal and the vertical directions. A Stardyne computer program was used to calculate natural frequencies and mode shapes, a gravity load case, and seismic load cases. Stardyne is a static and dynamic system of computer programs which analyze linear elastic structural models and is designed to run on the Control Data Cyber Computer System. Using the mathematical model, the first five lowest natural frequencies and mode shapes were calculated. The first load case was used to determine the gravitational forces on the boards and the seismic load cases were calculated by the program with the NRC 10 percent summation technique using the natural frequencies and mode shapes values. An analysis of the anchorage of the structure to its base was performed and the margin of safety for the welds which were used as attachments (M.S. (shear) = 1.08) was determined. These margins indicate that the structure can maintain its structural integrity.

Engineered Safeguards Logic Panels

The seismic qualification test for the Engineered Safeguards Panels were performed on a test specimen by the American Environments Company. The cabinets were bolted to U-channels in front and rear, and the U-channels were secured to a steel plate by a continuous fillet weld, and the cabinets were bolted together. Upon completion of this set up, the fixture and specimen were secured to the seismic vibration table to simulate normal in-service mounting and orientation. During the Seismic Random Tests 24 trips and resets were inserted (one per test for the two principal horizontal axis during each SSE and for each of the five OBE, totaling 24) to insure that the equipment will actuate as required. ALL sensor channels were tripped simultaneously in both measurement channels to produce a 2/2 trip in the safety channel. A resonant frequency search was performed in the frequency range of 1.0 to 35 Hz at an input excitation level of approximately 0.2g peak. The frequency range was searched by sweeping the input frequency at a rate approximately one-half octave per minute and remaining at each discrete frequency for a period of 15 seconds. The resonant frequency survey was performed in the major horizontal, vertical, and minor horizontal axis sequence.

The test specimen was subjected to biaxial multi-frequency random input motion at one-third octave intervals from 1.0 to 40 Hz. The duration of the seismic event was a minimum of 30 seconds.

There were a total of six seismic events; five OBEs followed by one SSE for each of the in-phase and out of phase conditions. Electrical tests were performed on the panels both before and after the seismic tests.

Isolation Cabinets

The Isolation Cabinets were tested on a Wyle multi-axis seismic simulator table mounted initially in the side-to-side and vertical orientation. After the completion of these tests, the cabinets are rotated 90 degrees in the horizontal plane to the front-to-back and vertical test orientation. To establish major resonances, a low level (approximately 0.1g horizontally and vertically) biaxial sine sweep for both orientations is performed over the frequency range of one Hz to 40 Hz and at a sweep rate of one octave per minute. The cabinets are subjected to 30 second duration simultaneous horizontal and vertical inputs of phase-incoherent random motion consisting of frequency band widths spaced one-third octave apart over the frequency range of one Hz to 40 Hz. Before the application of the SSE test in each orientation is performed, five OBE tests are to be applied.

Hot Shutdown Panel and HVAC-Board

The Hot Shutdown Panel and the Heating Ventilating and Plant Auxiliary Control Board were tested on a Wyle multi-axis seismic simulator table. The Panel and the Control Board were oriented such that their horizontal axis were colinear with the longitudinal axis of the test table for the first orientation, and rotated 90 degrees in the horizontal plane for the second orientation. A low-level (approximately 0.2g horizontally and vertically) single-axis sine sweep in each test was performed from one Hz to 35 Hz to determine major resonances. The sweep rate was set at one octave per minute.

Both the Panel and the Control Board are then subjected to 30-second duration simultaneous horizontal and vertical phase-incoherent inputs of random motion with frequency bandwidths spaced one-third octave apart over the frequency range of one Hz to 40 Hz. Five OBE tests, followed by a SSE test, are performed in both the side-to-side/vertical and the front-to-back/vertical orientation.

To ensure qualification for the required forces for other equipment and instrumentation, acceleration requirements are included in equipment specifications as design parameters. Vendors will use this information as the basis for analysis or testing depending on the type, size, shape, or complexity of equipment to be qualified.

The equipment specifications include, as a minimum, the following Seismic requirements:

a. The appropriate seismic excitation for which the equipment must qualify will be determined based on location in the plant;

b. The equipment is required to perform its intended function during and after a Safe Shutdown Earthquake;

c. The vendor is required to substantiate the adequacy of the design by analysis, testing, or a combination of these depending on the type of equipment and its intended safety function; and

d. The quality assurance program used in assuring the implementation of the requirements of CENPD-182[3] are discussed in CENPD-210A[4]. The seismic qualification program, as described in CENPD-182 meets the specified requirements for seismic Category I equipment.

e. Analyses or tests shall be performed for all supports of electrical and associated mechanical equipment and instrumentation to ensure their structural capability to withstand seismic excitation.

f. The analytical results will include the following:

    1. The required input motions to the mounted equipment

       The characteristics of the required input motion shall be specified by one of the following:

       (a) response spectrum

       (b) power spectral density function

       (c) time history

Such characteristics, as derived from the structures or systems seismic analysis, shall be representative of the input motion at the equipment mounting locations.

2. The combined stresses of the support structures shall be within the allowable limits found in the applicable codes.

3. Mathematical Model

4. Summary of the results which include the stresses of all major components, and displacements at the points of interest.

5. Conclusions

g. Supports shall be tested with either equipment or dynamically equivalent models installed. If the equipment is not operating or not installed during the support test, the response at the equipment mounting locations shall be monitored and characterized in the manner as stated in item f.1. In such a case, equipment shall be tested separately and the actual input to the equipment shall be more conservative in amplitude and frequency content than the monitored response.

h. The characteristics of the required input motion shall be specified by one of the following:

1. response spectrum

2. power spectral density function

3. time history

Such characteristics, as derived for the structures or systems seismic analysis, shall be representative of the input motion at the equipment mounting locations.

i. The actual input motion shall be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content shall be demonstrated. In applying this item to the electrical equipment, the frequency spectrum used shall cover the range from one through 33 Hz.

j. Seismic excitations generally have a broad frequency content. Random vibration input motion shall be used. However, single frequency input, such as sine beats, may be utilized provided one of the following conditions are 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, such that the testing response spectra will envelop the corresponding response spectra of the individual modes.

k. The input motion shall be applied to one vertical and one principal (or two orthogonal) horizontal axes simultaneously unless it can be demonstrated that the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions will be such that a purely rectilinear resultant input is avoided. The acceptable alternative is to have vertical and horizontal inputs in-phase, and then repeated with inputs 180 degrees out-of-phase. In addition, the test will be repeated with the equipment rotated 90 degrees horizontally.

l. The fixture design shall meet the following requirements:

   1. Simulate the actual service mounting

   2. Cause no dynamic coupling to the test item

   Specifically, cabinet and support test requirements will be conducted as follows:

   The design seismic environment of equipment located within support structures (cabinets) will be determined by either test or analysis.

m. Testing will consist of one of the following procedures:

   1. <u>Fully Operational Cabinet Test</u>

      The cabinet, fully loaded with equipment, will be tested in its operating state. During testing, a sample of safety-related functions will be monitored. This test will demonstrate both structural integrity and functional operability.

   2. <u>Weighed Cabinet Test With Subsequent Equipment Tests</u>

      (a) The cabinet will be tested with simulated equipment in place of the actual equipment. The simulated equipment will be equal in mass, mass distribution, and mounting to the actual equipment such that the dynamic response of the weighted cabinet is equal to that of the fully loaded cabinet. During testing the motions present at the equipment mounting points will be recorded. This test will demonstrate the cabinet structural integrity and determine the local seismic environment of the actual equipment.

      (b) The actual equipment will be independently tested or analyzed to those motions determined by the weighted cabinet test. The equipment will be operational and all safety related functions will be monitored during the test. This test will demonstrate functional operability of the equipment.

   3. <u>Equipment Test</u>

      Equipment which is not mounted in a cabinet, will be tested or analyzed in its operating state in a configuration which simulates its intended mounting.

n. For structures which can be modeled, a dynamic analysis may be substituted for the weighted cabinet test to determine the motions at the enclosed equipment mounting points.

For both testing and analysis, the input motions to the cabinet shall be derived from the building motions at the cabinet's intended location.

3.10.4 OPERATING LICENSE REVIEW

Seismic Category I equipment is designed and qualified to perform its intended function during and after the SSE. The reports of test results and analyses are referenced in Tables 3.10-1 and 2.

REFERENCES: SECTION 3.10

1. D.B. Vassalo (NRC) letter to Dr. R. E. Uhrig (FP&L), "Environmental and Seismic Qualification of Class 1E Equipment," dated July 28, 1978.

2. Dr. R. E. Uhrig (FP&L) letter L-78-334 to D.B. Vassalo (NRC) dated October 16, 1978.

3. CENPD-182, "Seismic Qualification of CE Instrumentation Equipment," Parts 1 and 2, Combustion Engineering Inc., May 1977. (Rev 01)

4. CENPD-210A, "Quality Assurance Program - A Description of the Nuclear Steam Supply System Quality Assurance Program," Combustion Engineering Inc., July 7, 1977.

5. Dr. R. E. Uhrig (FP&L) Letter L-81-507, December 2, 1981, List of Safety-Related Equipment & Systems; Qualification Summary of Equipment Sheets.

TABLE 3.10-1

A/E ELECTRICAL AND INSTRUMENTATION QUALIFICATION DATA

Equipment	Manufacturer/ID	Location*	Function	Conformance to EEE 344 (Year)	Qualification By Test Or Analysis or Both	Frequency Range Tested HZ	Seismic Excitation Waveform Input	Simultaneous Biaxial or Single Axis Input	Acceptance Criteria	Qualification Report #	
4.16 kV Switchgear	Westinghouse Co.	E	Furnish all emergency 4.16 kV power	1975	Test/Analysis	5-25 HZ	Sine Beats	Single	Without loss of function	76-7E7-SHAKE-R1 NLI Report R-037088-2, Rev. 1	
480 Volt Switchgear	ITE Corp.	E	Furnish all emergency 480 Volt power	1971 (and Appendix 3-10 A)	Test	0 5 - 50 HZ	Random Wave Forms (multi- Frequency)	Biaxial	Without loss of function	R09399 (4-27-1974)	
1500 KVA, 4160/480 V Station Power Transformer	Westinghouse (Purchased through ITE)	E	Furnish emergency power to the 480 V Switchgear	1975 (and Appendix 3.10 A)	Type Testing (Done by Westinghouse)	10 Cycles of Strong Motion Acceleration	30 Seconds Multi-Frequency Damped Sine Wave	Biaxial	Without loss of function	Seismic Cert. Report for Class 1E/Seismic Report Westinghouse 2000 KVA EL. 53B, Sept. 1975	
480 Volt Motor Control Centers	General Electric	E, F, G	Supply power to essential motors 100 HP and to panelboards	1975 (and Appendix 3.10 A)	Test	1 - 33 HZ (Test #1) 5.- 500 HZ (Test #2) 0.3 - 33 HZ (Test #3)	Sine Wave	Single	Unmodified Equipment suit for 0.5 g (5.-500HZ) Modified Equipment suit for 1 0 g(l-33 HZ)	75ICS001A	
Diesel Generator Controls	Morrison/Knudsen	G	Provide emergency 4.16 kV switchgear power	1975 (and Appendix 3.10 A)	Testing of individual components/analysis	1 0 40 HZ(max for some components					
125 VDC Station Batteries and Chargers	C and D Batteries Co.	E	Provide 125 VDC emergency power	1975	Test	1 - 40 HZ	Random Wave Forms (multi- frequency	Biaxial	Without loss of function	43517-1 (6-9-77) for control panel, FDI Report No. A-4-77 (9-8-1977) for diesel engine, report dated 7/26/78 for synchronous generator	
Static Inverters	Solid-state Controls	E	Provides 120 VAC, 1 phase regulated instrument power	1975	Test	1- 40 HZ	Random Wave Forms (multi-frequency)with Shine beat testing	Biaxial	Without loss of function	C&D Test Report #1948 and Wyle Lab Test Reports 43291-1 (6/24/76), 43450-1 (12/7/76) and 53116-1 (3/14/06)	
AC and DC Panels	System Controls Corp	Throughout the plant		1975	Test		Random Wave Forms (multi-frequency)				
						a) 1-35	a) Sine Sweep	Biaxial	Without loss of function	44001-1	
						b) 1-40	b) Random freq. Wave form	Biaxial	Structural integrity of panels will be maintained	44938-1	
Cable Trays	Husky-Burndy Co	Throughout the plant	Carries cables throughout the plant	1975	Analysis	NA	NA	NA	6" and 9" Tray safety factor 1.1 was achieved	Input data supplied via vendor's report S0 #201500 Rev. 1 5/22/78	
AC Motors	General Electric or HES	Intake structure steam trestle area, CCW area	Driven equip. opera- tors for ICW, Aux FW and CCW pumps	1975	Analysis or Test	NA	NA	NA	Meets or exceeds design specifications	Reports No. ME-384 ME-413 and ME-597 HES-NK0-2H08-WW Rev. 0 HES-GLS-11HPC-N17462 Rev. 0	EC 284 667
Heat Tracing	Nelson Electrical	E,D	Temp, control of boric acid lines	1975	Test	1.25 to 35 Hz	Random Multi-Frequency	Simultaneous Biaxial	Without visible evidence of physical damage	Wyle 58622 (6-31-81)	

*For symbols location, see notes following this table.

TABLE 3.10-1 (Cont'd)

Equipment	Manufacturer/ID	Location*	Function	Conformance to IEEE 344 (Year)	Qualification By Test Or Analysis or Both	Frequency Range Tested HZ	Seismic Excitation Waveform Input	Simultaneous Biaxial or Single Axis Input	Acceptance Criteria	Qualification Report #
Containment Penetrations	Conax Corp	Containnent wall of reactor bldg	Provides connections to electrical equipment on both sides of containment	1975	Test	1.-90 hz (TRS)	Random 30 second	Biaxial	No seal failure or loss of function	Conax Report PS-596, IPS-453, IPS-620, IPS-603, IPS-602, IPS-567
Lighting Transformers	Square D Co. Sorgel Transformer	Throughout the plant	Provides 120/208 Volt power for area ltg.	1975	Test	a) 1-35 b) 1-40	a) Sine Sweep b) Random wave form	Biaxial	Maintain structural and operational integrity	44509-1
Local Control Stations	Gould Inc.	Throughout the plant	"Start-stop" capability near motors and motor operated valves	1975	Test	1-40	a) Random	Biaxial	Without loss of function	Wyle 45316-1
Level Switches	Magnetrol	H, outside, D, E, I	Control and Alarm	1975	Test	1.0 to 40 0	Random Multi-frequencies	Simultaneous Biaxial	Without loss of function	Wyle Laboratories Report No. 43235-1
Diff Pressure Switches	ITT Barton Model 580 & 581	D, H, I, P Outside	Control and Alarm	1975	Test	1.0-33Hz	Random Multi-frequency	Biaxial	Without loss of function	Barton ITT Report No R3-580-6
Electronic Transmitters	Rosemount 153 Series B	B, E, outside, F, I, H	Control and Alarm	1975	Test	1-60Hz	Random Multi-frequency	Biaxial	Without loss of function	Rosemont Report No. 108025, Wyle Report No.45353-1
RTG Boards	Reliance	C	Control and Plant Protection	1975	Both					
					Test Components	1 0-35.0	Random Multi-frequency	Simultaneous Biaxial	Without loss of function or physical damage	Action Labs Report No. 17414-1
					Analysis-Structure	NA	NA	NA	Meets or exceeds design specification	Analysis Report, Action Labs Report No. 17387-82N
Engineered Safeguard Panels	Consolidated Controls	C	ESFAS Initiation	1975	Test	NA	Random Multi-frequency	Simultaneous Biaxial	Without evidence or physical damage or loss of function	American Environments Co. Inc. STR-136279-1 (Consolidated Controls Co. No. ER 1039)
Isolation Cabinets	Rochester Instrument	E	Provide separation between safety and non-safety channels	1975	Test	1.0 to 40 0	Random Multi-frequency	Simultaneous Biaxial	Without loss of function	Wyle Lab Report #44685-1
Heating and Ventilating Control Boards	Systems Control	C	Environment, Plant and Personnel Protection	1975	Test	1.0 to 40 0	Random Multi-frequency	Simultaneous Biaxial	Without loss of function	Wyle Lab Report #44685-1
Hot Shutdown Control Panels	Systems Control	E	Enable remote shutdown of plant	1975	Test	1.0 to 40 0	Random Multi-frequency	Simultaneous Biaxial	Without loss of function	Wyle Lab Report #44686-1

*For symbols locations, see notes following this table.

TABLE 3.10-1 (Cont'd)

Equipment	Manufacturer/ID	Location*	Function	Conformance to IEEE 344 (Year)	Qualification By Test Or Analysis or Both	Frequency Range Tested HZ	Seismic Excitation Waveform Input	Simultaneous Biaxial or Single Axis Input	Acceptance Criteria	Qualification Report #
Containment Fan Coolers (HVS-1A, 1B, 1C and 1D)										
i) Fans	Joy Mfg. Co. Model 45-26-1770/1170	B	Remove containment Heat (Supplement to Containment Spray System)	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	AAF Seismic Analysis NESE 392 (9-18-80)
ii) Motors	Reliance Electric Co 125/83 Hp: 1770/1170 rpm Part No. 600287-49	B	Remove containment Heat (Supplement to Containment Spray System)	1975	Test	NA	Single Frequency	Single	Meets or exceeds design specifications	Joy Mfg Report No. X-604 (4-6-77)
Radiation Monitoring System	General Atomics	B, C, D, E, F, H	Plant Radiation Monitoring	1975	Test	1 25-33Hz	Random Multi-frequency	Biaxial	Without loss of function	Model RD-23 Wyle 58567 (10-30-80) Model CD-15, 2HRD-2OPI Wyle 55651 (6-2-1981) Wyle 58602 (12-30-1980) Wyle 58645 (6-2-1981) Part No. GA0357-2170 Wyle 58695 (11-10-1981)
Plant Aux. Control Board 2 (PACB-2)	Systems Control	C	Misc. Elec. Controls & Instrumentation	1975	Analysis	NA	NA	NA	Meets design specifications	Acton Lab Report #16845
Isol. Relay Boxes	Systems Control	E	Misc. Elec. Controls & Instrumentation	1975	Analysis	NA	NA	NA	Meets design specifications	Acton Lab Report #16322
Main Feedwater Isol. Valve Relay Boxes	Systems Control		Elec. Signal Isol.	1975	Analysis	NA	NA	NA	Meets design specifications	Acton Lab Report #16549
Charging Pump Relay Box	Systems Control		Elec. Signal Isol.	1975	Analysis	NA	NA	NA	Meets design specifications	Acton Lab Report #16549
Shield Building Ventilation System (HVE-6A and 6B)										
i) Fans	Westinghouse Model H-413-M	D	Maintain sub-atmospheric pressure in the annulus and control the release of radioactive materials.	1975	Test	1 0-33 Hz	30 seconds Biaxial, multi-frequency input wave form.	Biaxial	Without loss of function	Westinghouse Test Report El:1075 dated 2-27-80
ii) Motors	Westinghouse 60 Hp/3545 rpm	E	Maintain sub-atmospheric pressure in the annulus and control the release of radioactive materials.	1975	Test	1 0-33 Hz	30 seconds Biaxial, multi-frequency input wave form.	Biaxial	Without loss of function	Westinghouse Test Report EL: 1075 dated 2-27-80
RAB Supply Air System										
i) Fans	Buffalo Forge Co. Model 980 BLAD	E	Supply Ventilation air and control equipment environment.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26280-81 dated 5-13-80
ii) Motors										
4A	Westinghouse 150 Hp/1775 rpm	E	Supply Ventilation air and control equipment environment	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26280-81 dated 5-13-80
4B	Westinghouse 150 Hp/1775 rpm	E	Supply Ventilation air and control equipment environment.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26280-81 dated 5-13-80

*For symbols location, see notes following this table.

TABLE 3.10-1 (Cont'd)

Equipment	Manufacturer/ID	Location*	Function	Conformance to IEEE 344 (Year)	Qualification By Test Or Analysis or Both	Frequency Range Tested HZ	Seismic Excitation Waveform Input	Simultaneous Biaxial or Single Axis Input	Acceptance Criteria	Qualification Report #
ECCS Area Ventilation System (HVE-9A and 9B)										
i) Fans	Buffalo Forge Co. Model 805 BL	D	Maintain slightly sub-atmospheric pressure in ECCS area and control the release of radioactive material to atmosphere.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26284-85 dated March, 1979
ii) Motors	Westinghouse 60 Hp/1775 rpm	E	Maintain slightly sub-atmospheric pressure in ECCS area and control the release of radioactive material to atmosphere	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26284-85 dated 5-13-80
Electrical Equipment Room Supply Fans (HVS-5A and 5B)										
i) Fans	Buffalo Forge Co Model 1085 BLD	E	Maintain a controlled environment in the electrical room	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26282-83 dated March, 1979
ii) Motors	Westinghouse 100 Hp/1775 rpm.	E	Maintain a controlled environment in the electrical room	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26282-83 dated 5-13-80
Electrical Equipment Room Exhaust Fans (HVE-11 and 12)										
i) Fans	Westinghouse Model 3066	E	Maintain a controlled environment in the electrical room	1975	Test	1-33Hz	Random Multi- frequency input wave	Biaxial	Without loss of function	Westinghouse Test Report EL: 1061
ii) Motors	Westinghouse 50 Hp/1780 rpm.	E	Maintain a controlled environment in the electrical room.	1975	Test	1-33Hz	Random Multi- frequency input wave	Biaxial	Without loss of function	Westinghouse Test Report EL: 1061
Electrical Equipment Room Roof Ventilators (RV-3 and 4)										
i) Fans	Buffalo Forge Model 36MB	E	Maintain a controlled environment in the electrical room	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26463-64 dated April, 1979
ii) Motors	Westinghouse 5 Hp/1750 rpm	E	Maintain a controlled environment in the electrical room.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26463-64 dated March 26, 1980
Battery Room Roof Ventilators (RV-1 and 2)										
i) Fans	Buffalo Forge 16A Breezo	E	Maintain a controlled environment in the Battery room and exhaust hydrogen from the Battery room.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26461-62 dated April, 1979
ii) Motors	Westinghouse 3/4 Hp/1750 rpm	E	Maintain a controlled environment in the Battery room and exhaust hydrogen from the Battery room.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26461-62 dated April, 1979

*For symbols location, see notes following this table.

TABLE 3.10-1 (Cont'd)

Equipment	Manufacturer/ D	Location*	Function	Conformance to IEEE 344 (Year)	Qualification By Test Or Analysis or Both	Frequency Range Tested HZ	Seismic Excitation Waveform Input	Simultaneous Biaxial or Single Axis Input	Acceptance Criteria	Qualification Report #
Intake Structure Ventilation Fan (HVE-41A and 41B)										
i) Fans	Buffalo Forge Co. Model 36MB	I	Maintain a controlled environment in the intake structure.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26467-68 dated April, 1979
ii) Motor	Westinghouse 7-1/2 Hp/1745 rpm	I	Maintain a controlled environment in the intake structure.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26467-68 dated March 26, 1980
SBVS Electric Heating Coils (EHC-HVE-6A1, 6B1, 6A2 and 6B2)										
i) Heating Coils	CVI Corporation	D	Maintain relative humidity of air entering charcoal absorber at or below 70%.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	CVI Corp. Calculations & Reports B772-9910, September 28, 1979 B771-9911, March 2, 1979
ii) Panel	CVI Corporation	D	Maintain relative humidity of air entering charcoal absorber at or below 70%.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Action Environmental Testing Corporation Report #17870-83N January 12, 1983 for Panels & Report #17414-1, May 2, 1983 for Components
Control Room Emergency Cleanup System (HVE-13A and 13B)										
i) Fans	Buffalo Forge Co. Model 21 MW	C	Maintain habitability of control room.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26286-87 dated March, 1979
ii) Motors	Westinghouse 10 Hp/3515 rpm	C	Maintain habitability of control room.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Buffalo Seismic Calculation No. 78L-26286-87 dated May 13, 1980
Control Room Air-conditioning Units HVA/ACC-3A, 3B and 3C)										
i) Fans	Twin City Fan & Blower Co. 200 BAF	C	Maintain habitability of control room.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Twin City Seismic Calculations for CVI R-120 (3-26-79)
ii) Motors for Fans	Westinghouse 15 Hp	C	Maintain habitability of control room.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Westinghouse Seismic Analysis S.O. No. CB62667 (3-15-79)
iii)Compressor Motor	Westinghouse, 40 Hp	C	Maintain habitability of control room.	1975	Analysis	NA	NA	NA	Meets or exceeds design specifications	Westinghouse Seismic Analysis S.O. No. CB62667 dated March 15,1979

* For symbols location, see notes following this table.

TABLE 3.10-1 (Cont'd)

Equipment	Manufacturer/ID	Location*	Function	Conformance to EEE 344 (Year)	Qualification By Test Or Analysis or Both	Frequency Range Tested HZ	Seismic Excitation Waveform Input	Simultaneous Biaxial or Single Axis Input	Acceptance Criteria	Qualification Report #
Dampers for CRECS D-17A, 17B, D-18 and 19										
i) Operators	ITT NH-95	C	Allow air to pass through CRECS filter	1975	Test	1 25 to 35	½ SSE Test Spectrum	Biaxial random excitation	No damage shown after test	Wyle Test Report No 58072 dated 6/8/76
ii) Limit switches	NAMCO controls EA 170	C	Indicate Damper position	1975	Test	1 to 35	Maximum 9.52 G's Acceleration	Single	No loss of contact during the test	NAMCO Seismic Qualification Test of limit controls switches dated June, 1977
Control Room Air- Conditioning Dampers (D-20, 21 and 22)										
i) Operators	ITT NH - 95	C	Allow return air to pass through the cooling coils	1975	Test	1 25 to 35	1/2 SSE Test Spectrum	Biaxial Random Exitation	No damage shown after the test	Wyle Test Report No. 58072 dated 6/8/76
Control Room Return Air Damper D-39 and D-40										
i) Operators	ITT NH - 91	C	Modulate Return Air to maintain positive pressure inside the control room.	1975	Test	1 25 to 35	1/2 SSE Test Spectrum	Biaxial Random Exitation	No damage shown after the test	Wyle Test Report No. 58072 dated 6/8/76
SBVS Fan Discharge Damper (D-23 and D-24)										
i) Operators	ITT - NH - 91	E	Modulate and control air flow	1975	Test	1 25 to 35	1/2 SSE Test Spectrum	Biaxial Random Exitation	No damage shown after the test	Wyle Test Report No. 58072 dated 6/8/76
RAB Supply Air Dampers (D-1, D-2, D-3 and D-4)										
i) Operators	ITT NH - 95	D	Open on SIAS to supply air to ECCS area.	1975	Test	1 25 to 35	1/2 SSE Test Spectrum	Biaxial Random Exitation	No damage shown after the test	Wyle Test Report No. 58072 dated 6/8/76
ii) Limit Switches	NAMCO Controls EA 180	D	Indicate damper position	1975	Test	1 to 35	Maximum 9.52 G's Acceleration	Single	No loss of contact during the test	NAMCO Seismic Qualification Test of Limit Control Switches dated June, 1977
RAB Supply Air Dampers (D-7A, 7B, 8A and 8B)										
i) Operators	ITT NH-95	E	Close on SIAS to isolate supply air to non-essential areas	1975	Test	1 25 to 35	1/2 SSE Test Spectrum	Biaxial Random Excitation	No damage shown after the test	Wyle Test Report No. 58072 dated 6-8-76
ii) Limit Switches	NAMCO Controls EA-180	E	Indicate damper position	1975	Test	1 to 35	Maximum 9.52 G's Acceleration	Single	No loss of contact during the test	NAMCO Seismic Qualification Test of Limit Control Switches dated June,1977

*For symbols location, see notes following this table.

TABLE 3.10-1 (Cont'd)

Equipment	Manufacturer/ID	Location*	Function	Conformance to IEEE 344 (Year)	Qualification By Test Or Analysis or Both	Frequency Range Tested HZ	Seismic Excitation Waveform Input	Simultaneous Biaxial or Single Axis Input	Acceptance Criteria	Qualification Report #
RAP Exhaust Air Dampers. (D-5A, 5B, 6A, 6B, 9A, 9B, 12A and 12B.)										
i) Operators	ITT NH-95	D	Close on SIAS to isolate normal exhaust air from ECCS area.	1975	Test	1.25 to 35	1/2 SSE Test Spectrum	Biaxial Random Excitation	No damage shown after the test	Wyle Test Report No. 58072 dated 6-8-76
ii) Limit Switches	NAMCO Controls EA 180	D	Indicate damper position	1975	Test	1 to 35	Maximum 9.52 G's Acceleration	Single	No loss of contact during the test	NAMCO Seismic Qualification of Limit Control Switches dated June, 1977
Dampers on ECCS Filter System (D-13, 14, 15, 16, 2L-7A and 2L-7B)										
i) Operators	ITT NH-95	D	Open on SIAS to allow air to pass through ECCS filter train.	1975	Test	1.25 to 35	1/2 SSE Test Spectrum	Biaxial Random Excitation	No damage shown after the test	Wyle Test Report No. 58072 dated 6/8/76
ii) Limit Switches	NAMCO Controls EA-170 & EA-180 No Switches for 2L-7A, 2L-7B	D	Indicate damper position.	1975	Test	1 to 35	Maximum 9.52 G's Acceleration	Single	No loss of contact during the test	NAMCO Seismic Qualification of Limit Control Switches dated June, 1977
Dampers on FHB Vent System (D-29,30,31,32, 33,34,35 & 36										
i) Operators	ITT NH-95	E	Close on High-High radiation signal to allow air to pass through SBVS filter trains.	1975	Test	1.25 to 35	1/2 SSE Test Spectrum	Biaxial Random Excitation	No damage shown after the test	Wyle Test Report No. 58072 dated 6/8/76
ii) Limit Switches	NAMCO Controls EA-170 & EA-180	F	Indicate damper position.				Maximum 9.52 G's Acceleration	Single	No loss of contact during the test	NAMCO Seismic Qualification of Limit Control Switches dated June, 1977

*For symbols location, see notes following this table.

TABLE 3.10-1 (Cont'd)

Equipment	Manufacturer/ D	Location*	Function	Conformance to IEEE 344 (Year)	Qualification By Test Or Analysis or Both	Frequency Range Tested HZ	Seismic Excitation Waveform Input	Simultaneous Biaxial or Single Axis Input	Acceptance Criteria	Qualification Report #
Hydrogen Recombiner	Westinghouse Electric Corp Model B	A (Recombiner) E (Power Supply) C (Control Panel)	Recombine liberated hydrogen with free oxygen to form water.	1975	Test	1.25 to 35	Random and Sine beat input	Biaxial	Without loss of function	WCAP-9346 WCAP-7709L Supplement 1 to 7
Hydrogen Analyzer	Comsip Delphi K-111	D (local panel) C (remote panel)	Monitor containment hydrogen levels.	1975	Resonance test & analysis	1-40	Random Multi-Frequency	Single	Without loss of function	EA & T Project No. 1035-1 (Sept. 1981) 1035-2 (July 1980)
Rotameters	Brooks Instrument Division		Indicator	1975	Analysis structure by computer	NA	N/A	N/A	Meets design specification without loss of function	Acton Environmental Testing Corporation Report No. 1520
Low Differential Electronic Transmitters	Air Monitor Corporation		Indication and alarm.	1975	Test	1.25 to 33	Random Multi-frequency	Biaxial	Ebasco Specification Requirements	Wyle Report #58539
Acoustical Valve Flow Monitors	Technology for Energy Corp.	A	Indication and Alarm.	1975	Test	1-35 Hz	Random Multi-frequency	Biaxial	Ebasco Specification Requirements	TEC Report No. 517-TR-03
Containment Level Transmitters	Fluid Components Incorp.	A	Indication and Alarm	1975	Test	1.25-32 Hz	Random Multi-frequency	Biaxial	Ebasco Specification Requirements	FCI Report No. 708111-R1
Ultrasonic Level	National Sonic		Alarm	1975	Test	1-35 Hz	Random Multi-frequency	Biaxial	Ebasco Specifica- tion Requirements	Acton Lab Report No. 17376-82N

NOTE: *For symbols locations, see notes following this table

A - Inside containment, inside the secondary shield wall
B - Inside containment, outside the secondary shield wall
C - Control room
D - RAB high radiation zone
E - RAB low radiation zone
F - FHB
G - DGB
H - CCW area
I - Intake structure

TABLE 3.10-2

NSSS ELECTRICAL EQUIPMENT QUALIFICATION DATA

Equipment Mechanical	Equipment Electrical	Manufacturer	Location	Function	Conformance To IEEE 344 (Year)	Test or Analysis	Frequency Tested	Wave Form	Biaxial or Single Axis	Acceptance Criteria	Report Number
HPSI Pump	Motor	GE or HES	RAB	Continuous Operation	1971	Analysis or Test	-	-	-	*	34A842271 Rev.A HES-NKO-2H08-WW Rev. 0 HES-GLS-11HPC-N17462 Rev. 0
LPSI Pump	Motor	W	RAB	Continuous Operation	1975	Analysis & Test	2-120 Hz	Sine	**	*	WCAP-8754(1)
Charging Pump	Motor	W	RAB	Continuous Operation	1971	Analysis	-	-	-	*	74C22954
B.A. Makeup Pump	Motor	Reliance	RAB	Continuous	1975	Analysis	-	-	-	*	77-A-27
Valves	Motor	Limitorque	Inside, Outside Containment	Open, Close Modulate	1971 (and IEEE 332-1972)	Test	5-35 Hz	Sine	**	*	P-600456
Instrumentation	-	-	-	-	-	-	Refer To CENPD-182	-	-	-	

* Natural frequency above 33 Hz.
** Single axis input in three separate directions

(1) Ref. W Seismic Analysis Report for S.O. 77F14404, C.E. Letter L-CE-4047

3.10A CRITERIA FOR SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT AND THEIR SUPPORTS

3.10A.1 SEISMIC DESIGN CRITERIA

Equipment and equipment supports are designed and evaluated by analytical and/or testing methods described herein to insure that they will remain functional during and following the postulated safe shutdown earthquake, and remain operable during and after the occurrence of an operating basis earthquake.

Equipment and equipment supports subject to seismic Category I considerations are designed to safely withstand simultaneously-acting two horizontal and one vertical earthquake effects (vertical earthquake effects acting in either upward or downward direction to give the most severe combinations in accordance with values given in the corresponding floor response spectra.

For the equipment and its supports, vendors are required to demonstrate the ability of the supported equipment to remain fully functional during and after the seismic disturbances. The preferred acceptable method of demonstrating this ability is by testing of the supported equipment under simulated earthquake conditions.

The test program is required to confirm the functional operability of electrical equipment and instrumentation during and after specified earthquake events. The analysis method is used for equipment and its supports which can be properly modeled and mathematically analyzed to obtain their response during the seismic event. This method may not provide sufficient assurance as to maintaining the proper internal operation of certain equipment (such as switches, level indicators, etc.); for these cases actual performance testing procedures are applied.

3.10A.2 SEISMIC ANALYSES, TESTING PROCEDURES AND RESTRAINT MEASURES

The following briefly describes the procedure followed in using the methods for seismic qualification:

.01 Testing Method

a. The characteristics of the required input motion are specified by response spectra.

   One of the following input motion representation characteristics is used in performing the tests:

   1. Response spectrum

   2. Power spectral density function

   3. Time history

Such characteristics, represented by spectra curves, are derived from the structures or systems seismic analysis test motion and are representative of the input motion at the equipment mounting locations, i.e., the motion is such that the resulting response spectra envelopes the design response spectra provided.

b. Electrical and instrumentation and control equipment as identified in the base specification are tested in the operational condition. Operability is verified during and after the testing.

c. The actual input motion is characterized in the same manner as the required input motion. The test input motion is conservative and can be achieved by varying the input parameters such as amplitude and the range of test frequencies.

d. As seismic excitations generally have a broad frequency content, random vibration input motion should be used. However, single frequency input, such as sine beats or other waveforms, may 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, such that the testing response spectra will envelope the corresponding response spectra of the individual modes.

e. The input motion is applied to one vertical and one principal (or two orthogonal) horizontal axes simultaneously unless it can be demonstrated that the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction and vice versa.

f. In the case of a single frequency input, the time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. If the test inputs in the two axes are identical and in phase, then the test is repeated with the inputs 180 degrees out of phase. Therefore, for multiaxis testing, independent random inputs are recommended.

g. Unless it can be shown that the equipment has symmetry about the vertical axis, the equipment is rotated 90 degrees and re-tested.

h. Where practical, the fixture design meets the following requirements:

   1. Simulate the actual service mounting

   2. Cause no dynamic coupling to the test item

   3. In designing the actual simulated service mounting and support, there shall be no dynamic amplification due to flexibility of supports.

i. The equipment being tested must demonstrate its ability to perform its intended function, and sufficient monitoring equipment should be used to evaluate performance and operability before, during and following the test.

j. Where testing is practicable, the 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 as stated in (a). In such a case, equipment is required to be tested separately and the actual input to the equipment is more conservative in amplitude and frequency content than the monitored response.

.02 Mathematical Analysis Method

The mathematical analysis method consists of the following:

a. Model the equipment and supports with sufficient degrees of freedom to ensure adequate representation in two major horizontal and the vertical directions.

b. Determine the natural frequencies and mode shapes of the equipment and supports.

c. If the analysis of the model yields "rigid" characteristics, i.e., natural period of vibration of predominant mode of supported equipment equal to or less than 0.03 seconds, seismic acceleration coefficients obtained from applicable response spectra curves may be applied statically to perform the analysis on the equipment and supports. The vertical and two horizontal seismic effects are applied simultaneously to the weights of components at their gravitational centers for the seismic load calculation and design.

d. For supported equipment with “flexible" characteristics, i.e. having a natural period of vibration more than 0.03 seconds, the equipment responses are obtained by use of Response Spectra Modal Analysis Techniques. The input response acceleration values are obtained from the applicable response spectra curves. The Time History Analysis method may also be used if the proper input time history data is available.

   The square root of the sum of the squares method is normally used to combine the modal responses when the Response Spectrum Modal Analysis method is employed. In those cases, however, where modal frequencies are closely spaced, the responses of the closely spaced modes are combined by the sum of the absolute values method and in turn combined with the responses of the remaining significant modes by the square root of the sum of the square method.

e. The response effects resulting from individual earthquake directions are combined by the square root of the sum of the squares method. This is applicable to the aforementioned paragraphs "c" and "d".

f. The equipment stress induced from the earthquake loads as obtained from above, is combined with stresses from other loads in accordance with the load combinations for seismic Category I structures as specified in Section 3.8.

g. The analysis includes evaluation of the effects of the calculated stresses and deflection under earthquake load effects on mechanical strength, alignment and electrical characteristics.

.03 Combined Analysis and Testing

Impedance testing is used when the equipment to be qualified is too large to be mounted on a shake table. Portable vibration generators are used to excite the modes of the equipment. This data is used to establish mathematical models in qualification by analysis, or to determine transfer functions for component testing. An alternative is to simulate the effects of a SSE by single or multiple mode excitation as applicable.

Equipment is qualified using analysis to extrapolate test results. This is particularly apt when there are many different combinations of equipment which is basically of the same type and changes are due to different models or sizes. Here it is impractical to test every variation. Tests must be designed to gather sufficient data to enable valid mathematical models to be established.

Other combined methods include shipping vibration and shock testing. These methods are intended primarily to produce valuable back-up information. However, it is not recommended procedure to use them solely for seismic qualification.

The following documentation is required to be provided in the seismic qualification report:

a. Where testing is used for equipment qualification, the following documentation is provided:

1. Equipment identification

2. Test facility (location and description)

3. Test equipment used and calibration records

4. Test method

5. Seismic input used for testing

6. Variables to be measured including accuracy

7. Number, type and location of test monitoring sensors for each variable

8. Test data and accuracy

9. Equipment responses, at support interface (shears, axial loads, moments, etc) resulting from two horizontal and the vertical input excitation, for Purchaser's support design.

10. Results and conclusions

11. Attestation

b. Where computational techniques are utilized for equipment qualification, the following documentation is provided. Analytical procedures are presented in a step-by-step form which is readily auditable by persons skilled in such analysis.

1. Analytical method

2. Complete mathematical model

3. Properties of mathematical model

4. Method of combining modes

5. Result of analysis

6. Response loads

7. Governing codes

8. Conclusion

9. Certification of compliance with specified seismic requirements.

3.11 ENVIRONMENTAL QUALIFICATION

3.11.1 INTRODUCTION

Safety-related equipment must be capable of maintaining functional operability under conditions postulated to occur during its installed life. This requirement is embodied in 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criteria 1, 2, 4, 23 and 50.

The NRC has used a variety of methods to ensure that these general requirements are met for electric equipment important to safety. For nuclear plants after 1971, qualification was judged on the basis of IEEE Std 323-1971, "IEEE Trial-Use Standard: General Guide for Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations." For later day plants the commission has used Regulatory Guide 1.89 which endorses IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," subject to supplementary provisions. The St. Lucie Unit 2 construction permit was issued on May 2, 1977. Originally St. Lucie Unit 2 was committed to meet IEEE 323-1971, but revised that commitment to comply with IEEE 323-1974 guidelines wherever practical for equipment qualification.

The NRC has subsequently issued more definitive criteria in NUREG-0588, "Interim Staff Position on Environmental Qualification (EQ) of Safety-Related Electrical Equipment," which contains two sets of criteria:

- Category I, for plants whose CP was issued after July 1, 1974, incorporates and supplements IEEE Std 323-1974

- Category II, for plants whose CP was issued before July 1, 1974, incorporates and supplements IEEE Std 323-1971.

Because the St. Lucie Unit 2 construction permit was issued on May 2, 1977, NUREG-0588 Category I Criteria are applicable.

Environmental qualification (EQ) of safety-related equipment located in a mild environment is ensured by conformance to the general quality and surveillance requirements identified in 10 CFR Part 50, Appendix B, and Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," as implemented in the FPL Quality Assurance Topical Report. A separate controlled drawing, "Environmental Qualification Report and Guidebook," has been developed which provides the detailed information required to certify qualification of electrical components located in a potential harsh environment. The report describes the scope of the St. Lucie EQ Program. The EQ program development, methodology and the equipment qualification status is also discussed in the reports.

The Mechanical Equipment Qualification status is as follows. Although there are no detailed requirements for mechanical equipment, 10 CFR 50 Appendix A General Design Criteria 1, "Quality Standards and Records," and 4, "Environmental and Missile Design Basis;" Appendix B to 10 CFR 50," Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Sections III, "Design Control," and XVII, "Quality Assurance Records;" and Standard

Review Plant Section 3.11, Revision 1 contain the following requirements and guidelines related to equipment qualification of mechanical components:

- Components shall be designed to be compatible with the postulated environmental conditions, including those associated with loss of coolant accidents (LOCA).
- Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety related functions.
- Design Control measures shall be established and shall include the results of tests and material analysis.

FPL complies with the above requirements.

3.11.2 CRITERIA

3.11.2.1 Code Of Federal Regulations

Environmental Qualification requires commitment to various sections of 10 CFR. These sections include 10 CFR 21, "Reporting of Defects and Noncompliance," and 10 CFR 50 "Domestic Licensing of Production and New Utilization Facilities." Discussed below is the Environmental Qualification Rule. For more information refer to Drawing 2998-A-451-1000, "Environment Qualification Report and Guidebook."

3.11.2.1.1 10 CFR 50.49 Environmental Qualification Of Electric Equipment Important To Safety For Nuclear Power Plants

The US Code of Federal Regulations, Part 10, Section 50.49 (10 CFR 50.49) requires that each holder of a license to operate a nuclear power plant establish a program for qualifying: (1) Safety Related electric equipment which is relied upon to remain functional during and following a Design Basis Accident; (2) Non-Safety related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions; (3) certain post-accident monitoring equipment defined in Regulatory Guide 1.97, Revision 3, as discussed in 10 CFR 50.49 (b)(1), (b)(2) and (b)(3).

3.11.2.2 NUREG-0588, "Interim Staff Position Of Environmental Qualification Of Safety Related Electric Equipment"

As part of the NRC's review of operating license applications, a number of positions have been developed on the methods and procedures used to environmentally qualify safety related equipment. These positions which are described in more details in NUREG-0588, supplement the requirements found in the 1971 and 1974 version of IEEE standard 323. While alternatives to these positions may be proposed, these positions will be used together with the standards, as the basis for environmental qualification. As stated previously, St. Lucie Unit 2 is a NUREG-0588 category I plant.

3.11.2.3 NRC Regulatory Guides

The following regulatory guides are followed at St. Lucie Unit 2. A more inclusive listing of Regulatory Guides committed to can be found in Section 1.8.

- 1.30 Rev. 3 - "Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment."
- 1.40 Rev. 0 - “Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants."
- 1.63 Rev. 0 - "Electrical Penetration Assemblies in Containment Structures for Light Water Cooled Nuclear Power Plants.”
- 1.73 Rev. 0 - “Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants."
- 1.89 Rev. 0 - “Qualification of Class IE equipment for Nuclear Power Plants.”
- 1.97 Rev. 3 - “Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During the Following an Accident.”
- 1.131 Rev. 0 - "Qualification Tests of Electric Cables Field Splices and Connections for Light Water Cooled Nuclear Power Plants."

3.11.2.4 Other Applicable Documents

For identification of other documents St. Lucie Unit 2 is committed to, and for a more detailed description of the St. Lucie Unit 2 commitments on EQ refer to Drawing 2998-A-451-1000, "Environmental Qualification Report and Guidebook."

3.11.3 IDENTIFICATION OF COMPONENTS

Electric equipment covered in 10 CFR 50.49 is characterized as follows:

a. Safety Related electric equipment that is relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR 100 guidelines. Design Basis Events are defined as conditions of normal operation, including anticipated operational occurrances, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) of this paragraph.

b. Non safety electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified previously.

c. Certain post-accident monitoring equipment (Refer to Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.")

These components are identified and controlled on plant drawing 2998-A-450.

3.11.4 QUALIFICATION OF COMPONENTS

If the equipment in question meets the requirement found in Subsection 3.11.3, it must be qualified to 10 CFR 50.49. The "Environmental Qualification Report and Guidebook," Drawing 2998-A-451-1000 provides the information required to properly identify the environment to which the specific equipment must be qualified. Operability requirements associated with the component are discussed along with the required temperature, pressure, humidity, radiation, aging and submergence.

Each parameter is defined in a specific subsection. Most parameters are identified on Zone Maps as a convenient reference. Zone Maps indicate the normal and abnormal values associated with specific areas of the plant at a given period of time.

Harsh environments are characterized by abnormally high temperatures and pressures, high radiation doses, corrosive chemical spray, and/or high relative humidity. Also, in some cases, submergence may have to be considered based on equipment location with respect to the maximum flood level.

A mild environment, as defined in 10 CFR 50.49, is an environment that would at no time be significantly more severe than the environment which would occur during normal operation, including operational occurrences. Equipment located in a mild environment is not covered under 10 CFR 50.49. Mild environments operability is assured by either: (a) periodic maintenance, inspection and/or a replacement program based on sound engineering judgement or manufacturer's recommendations; (b) a periodic testing program; (c) an equipment surveillance program.

Environments in which radiation is the only parameter of concern are considered to be mild if the total radiation dose (includes 60-year normal dose plus the post accident dose) is 1.0E5 rads or less. This value is the threshold for evaluation and consideration. Excluded from this consideration, however, are most solid state electronic components and components that utilize teflon. Class 1E equipment located in environments between 1.0E3 and 1.0E5 are evaluated on a case by case basis.

For additional detail on the identification of environmental conditions refer to Drawing 2998-A-451-1000," Environmental Qualification Report and Guidebook."

3.11.5 MAINTENANCE

The purpose of the St. Lucie Unit 2 Equipment Qualification Program is the preservation of the qualification of safety related systems, structures and components. In order to accomplish the task, the plant has developed approved Design Control, Procurement and Maintenance Procedures. Each procedure has incorporated the requirements of environmental qualification according to the functional requirements of the program/system/component. The plants procedures are prepared to maintain proper design control, for plant modifications, procurement of new equipment and spare parts. The plants maintenance program is designed to provide preventative as well as corrective maintenance which is identified by field operational experience and industry correspondence. In addition, the component specific documentation package contains, in Section 5, the equipments qualified life. This qualification interval is developed based upon the vendors test report reviewed in conjunction with the environmental parameter associated with the area. After this review is completed a qualified life is established and operation with this piece of equipment up to the equipments end point is acceptable.

3.11.6 RECORDS/QUALITY ASSURANCE

A documentation package is prepared for the qualification of each manufacturers piece of equipment under the auspices of 10 CFR 50.49. This package contains the information, analysis and justifications necessary to demonstrate that the equipment is properly and validly qualified for the environmental effects of 60 years of service plus a design basis accident.

This documentation package is developed from the criteria stipulated in the Environment Qualification Report and Guidebook.

A complete listing of equipment under the auspices of 10 CFR 50.49 is maintained.

All three of the above documents are drawings and are developed and controlled under the procedures involving drawing preparation, updating and storage as specified in the FPL Quality Assurance Program.

The generic elements of the FPL Quality Assurance Program are described in the Florida Power and Light Quality Assurance Topical Report (QATR) discussed in Section 17.2. The QATR defines departmental responsibilities by which FPL implements the corporate Quality Assurance program.

3.11.7 CONCLUSIONS

The Equipment Qualifications Report and Guidebook, together with the manufacturers' specific Documentation Packages and the 10 CFR 50.49 list of equipment have been developed for the purpose of documenting the environmental qualification of safety related equipment. This program has insured the systems selected for qualification are complete, the environmental conditions resulting from the design basis accident are indentified and that the methods used for qualification are appropriate.

Based on these checks and the ongoing environmental qualification program, St. Lucie Unit 2 is in compliance with 10 CFR 50.49.